PMLevyCOLPEm Resource

From: Habib, Donald
Sent: Friday, May 29, 2015 12:57 PM
To: PMLevyCOLPEm Resource
Subject: FW: Reviewer Aid - DCD markup for Levy Condensate Return
Attachments: Integrated DCD_COLA CondensateReturnChanges_Updated051415_External (3).pdf

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From: Kitchen, Robert [mailto:Robert.Kitchen@duke-energy.com]
Sent: Friday, May 29, 2015 11:51 AM
To: Habib, Donald
Cc: Waters, David; Cross-Dial, Andrea
Subject: Reviewer Aid - DCD markup for Levy Condensate Return

Don – Attached is a DCD markup to be used as a reviewer's aid that reflects an integration of changes to the AP1000 DCD Rev 19 information due to Condensate Return design change that will be incorporated into the Levy COLA. These changes were provided to NRC in the following Duke correspondence:

- NPD-NRC-2014-034 "LNP Submittal of COL Application, Revision 7"
- NPD-NRC-2014-038 "LNP Condensate Return Submittal Supplement 5"
- NPD-NRC-2015-015 "RESPONSE TO NRC RAI LETTER 124- SRP SECTION 6.3, AND SUPPLEMENT 6 TO SUBMITTAL OF EXEMPTION REQUEST AND DESIGN CHANGE DESCRIPTION FOR DEPARTURE FROM AP1000 DCD REVISION 19 TO ADDRESS CONTAINMENT CONDENSATE RETURN COOLING DESIGN"

Let me know if you have questions.

Hearing Identifier: Levy_County_COL_Public
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Priority: Standard
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CONDENSATE RETURN

TIER 1 CHANGES

NOTE: These excerpts from DCD R19 are provided for context. The Tier 1 changes are reflected in the COLA Tables contained in pages 15-17 of this document.

2.2.3 Passive Core Cooling System

Design Description

The passive core cooling system (PXS) provides emergency core cooling during design basis events.

The PXS is as shown in Figure 2.2.3-1 and the component locations of the PXS are as shown in Table 2.2.3-5.

1. The functional arrangement of the PXS is as described in the Design Description of this Section 2.2.3.

2. a) The components identified in Table 2.2.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.3-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.3-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.2.3-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.3-1 are powered from their respective Class 1E division.

c) Separation is provided between PXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The PXS provides the following safety-related functions:

   a) The PXS provides containment isolation of the PXS lines penetrating the containment.

   b) The PRHR HX provides core decay heat removal during design basis events.

   c) The CMTs, accumulators, in-containment refueling water storage tank (IRWST) and containment recirculation provide reactor coolant system (RCS) makeup, boration, and safety injection during design basis events.

   d) The PXS provides pH adjustment of water flooding the containment following design basis accidents.

9. The PXS has the following features:

   a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.

   b) The accumulator discharge check valves (PXS-PL-V028A/B and V029A/B) are of a different check valve type than the CMT discharge check valves (PXS-PL-V016A/B and V017A/B).

   c) The equipment listed in Table 2.2.3-6 has sufficient thermal lag to withstand the effects of identified hydrogen burns associated with severe accidents.

10. Safety-related displays of the parameters identified in Table 2.2.3-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).

    b) The valves identified in Table 2.2.3-1 as having protection and safety monitoring system (PMS) control perform their active function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.3-1 as having diverse actuation system (DAS) control perform their active function after receiving a signal from the DAS.

12. a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.3-1 assume the indicated loss of motive power position.

13. Displays of the parameters identified in Table 2.2.3-3 can be retrieved in the MCR.

Inspection, Tests, Analyses, and Acceptance Criteria

Table 2.2.3-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the PXS.

Table 2.2.3-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V016B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT A Discharge Check Valve	PXS-PL-V017A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V017B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Accumulator A Pressure Relief Valve	PXS-PL-V022A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator B Pressure Relief Valve	PXS-PL-V022B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator A Discharge Isolation Valve	PXS-PL-V027A	Yes	Yes	Yes	- / -	Yes	- /No	None	As Is
Accumulator B Discharge Isolation Valve	PXS-PL-V027B	Yes	Yes	Yes	- / -	Yes	- /No	None	As Is
Accumulator A Discharge Check Valve	PXS-PL-V028A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V028B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator A Discharge Check Valve	PXS-PL-V029A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V029B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Nitrogen Supply Containment Isolation Valve	PXS-PL-V042	Yes	Yes	Yes	Yes/No	Yes (position)	Yes/No	Transfer Closed	Close
Nitrogen Supply Containment Isolation Check Valve	PXS-PL-V043	Yes	Yes	No	- / -	No	- / -	Transfer Closed	-
PRHR HX Inlet Isolation Motor-operated Valve	PXS-PL-V101	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
PRHR HX Control Valve	PXS-PL-V108A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
PRHR HX Control Valve	PXS-PL-V108B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
Containment Recirculation A Isolation Motor-operated Valve	PXS-PL-V117A	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/Yes	None	As Is
Containment Recirculation B Isolation Motor-operated Valve	PXS-PL-V117B	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/Yes	None	As Is
Containment Recirculation A Squib Valve	PXS-PL-V118A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V118B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Containment Recirculation A Check Valve	PXS-PL-V119A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation B Check Valve	PXS-PL-V119B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation A Squib Valve	PXS-PL-V120A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V120B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection A Check Valve	PXS-PL-V122A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V122B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V123A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V123B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
IRWST Injection A Check Valve	PXS-PL-V124A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V124B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V125A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V125B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Gutter Isolation Valve	PXS-PL-V130A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed
IRWST Gutter Isolation Valve	PXS-PL-V130B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed
CMT A Level Sensor	PXS-011A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011D	-	Yes	-	Yes/Yes	Yes	- / -	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
CMT B Level Sensor	PXS-012A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-014A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014D	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-045	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-046	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-047	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-048	-	Yes	-	Yes/Yes	Yes	- / -	-	-
PRHR HX Flow Sensor	PXS-049A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
PRHR HX Flow Sensor	PXS-049B	-	Yes	-	Yes/Yes	Yes	- / -	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Containment Flood-up Level Sensor	PXS-050	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-051	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-052	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RNS Suction Leak Test Valve	PXS-PL-V208A	Yes	Yes	No	- / -	No	-/-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
PRHR HX inlet line from hot leg and outlet line to steam generator channel head	RCS-L134, PXS-L102, PXS-L103, PXS-L104A, PXS-L104B, PXS-L105, RCS-L113	Yes	Yes	Yes
	PXS-L107	Yes	Yes	No
CMT A inlet line from cold leg C and outlet line to reactor vessel direct vessel injection (DVI) nozzle A	RCS-L118A, PXS-L007A, PXS-L015A, PXS-L016A, PXS-L017A, PXS-L018A, PXS-L020A, PXS-L021A	Yes	Yes	Yes
	PXS-L019A, PXS-L070A	Yes	Yes	No
CMT B inlet line from cold leg D and outlet line to reactor vessel DVI nozzle B	RCS-L118B, PXS-L007B, PXS-L015B, PXS-L016B, PXS-L017B, PXS-L018B, PXS-L020B, PXS-L021B	Yes	Yes	Yes
	PXS-L019B, PXS-L070B	Yes	Yes	No
Accumulator A discharge line to DVI line A	PXS-L025A, PXS-L027A, PXS-L029A	Yes	Yes	Yes
Accumulator B discharge line to DVI line B	PXS-L025B, PXS-L027B, PXS-L029B	Yes	Yes	Yes
IRWST injection line A to DVI line A	PXS-L125A, PXS-L127A	Yes	Yes	Yes
	PXS-L123A, PXS-L124A, PXS-L118A, PXS-L117A, PXS-L116A, PXS-L112A	Yes	No	Yes
IRWST injection line B to DVI line B	PXS-L125B, PXS-L127B	Yes	Yes	Yes
	PXS-L123B, PXS-L124B, PXS-L118B, PXS-L117B, PXS-L116B, PXS-L114B, PXS-L112B, PXS-L120	Yes	No	Yes

Table 2.2.3-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No

Table 2.2.3-1									
Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-

LNP DEP 3.2-1

Note: Dash (-) indicates not applicable.

Table 2.2.3-1 (cont.)

LNP DEP 3.2-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Downspout Screen 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Ye s	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Ye s	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Ye s	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Ye s	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
LNP DEP 3.2-1 Downspout drain lines from polarcrane girder and internal stiffener to collection box A	PXS-L301A, PXS-L302A, PXS-L303A,PXS-L304A, PXS-L305A, PXS-L306A,PXS-L307A, PXS-L308A, PXS-L309A,PXS-L310A	Yes	No	Yes
Downspout drain lines from polarcrane girder and internal stiffener to collection box B	PXS-L301B, PXS-L302B, PXS-L303B,PXS-L304B, PXS-L305B, PXS-L306B,PXS-L307B, PXS-L308B, PXS-L309B,PXS-L310B	Yes	No	Yes

CONDENSATE RETURN

TIER 2 CHANGES

- Pumps (Class A, B, C) – ASME OM Code, Subsection ISTB
- Valves (Class A, B, C) – ASME OM Code, Subsection ISTC
- Equipment supports (Class A, B, C) – ASME Code, Section XI, Subsection IWF
- Metal containments and vessels – ASME Code, Section XI, Subsection IWE
- Other Class A components such as pipes and tanks – ASME Code, Section XI, Subsection IWB
- Other Class B components such as pipes and tanks – ASME Code, Section XI, Subsection IWC
- Other Class C components such as pipes and tanks – ASME Code, Section XI, Subsection IWD.

The inspection requirements, if applicable, for Class D structures, systems, and components are established by the designer for each structure, system, and component. These inspection requirements are developed so that the reliability of the structures, systems, and components is not degraded. The inspection req [illegible] or maintenance plans.

NOTE: The DCD R19 excerpts are provided for context - the COLA Table contained on page 22 of this document identifies the changes

3.2.4 Application of AP1000 Safety-Related Equipment and Seismic Classification System

The application of the AP1000 equipment and seismic classification system to AP1000 systems and components is shown in Table 3.2-3. Table 3.2-3 lists safety-related and seismic Category I mechanical and fluid system component and associated equipment class and seismic category as well as other related information. The table also provides information on the systems that contain Class D components. Additional information on the Class D functions of the various systems can be found in the description in the Design Certification Document (DCD) for the systems. Mechanical and fluid systems that contain no safety-related or Class D systems are included in the table and general information provided on the system. Supports for piping and components have the same classification as the component or piping supported. Supports for AP1000 equipment Class A, B, and C mechanical components and piping are constructed to ASME Code, Section III, Subsection NF requirements. The principal construction code for supports for nonsafety-related components and piping is the same as that for the supported component or piping.

Following the name of each system is the building location of the system components. Some of the systems supply all or most of the buildings. This is indicated by identifying the location as various. Where a system includes piping or ducts that only passed through a building without including any components that building is generally not included in the list.

The following list includes the systems in Table 3.2-3. The three letters in the beginning of each line is the acronym for the system. The systems included in Table 3.2-3 are listed alphabetically by three letter acronym. Those systems marked with an asterisk * are electrical or instrumentation systems and are not included in Table 3.2-3. The components in the incore instrumentation system

Table 3.2-3 (Sheet 15 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Primary Sampling System (Continued)					
PSS-PY-C03	Containment Atmosphere Sample Line Penetration	B	I	ASME III, 2	
Balance of system components are Class E					
Potable Water System (PWS)					Location: Various
PWS-PL-V418	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V420	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V498	PWS MCR Vacuum Relief	C	I	ASME III-3	
Balance of system components are Class E					
Passive Core Cooling System (PXS)					Location: Containment
PXS-ME-01	Passive Residual Heat Removal Heat Exchanger	A	I	ASME III-1	
PXS-MT-01A	Accumulator Tank A	C	I	ASME III-3	
PXS-MT-01B	Accumulator Tank B	C	I	ASME III-3	
PXS-MT-02A	Core Makeup Tank A	A	I	ASME III-1	
PXS-MT-02B	Core Makeup Tank B	A	I	ASME III-1	
PXS-MT-03	In-Containment Refueling Water Storage Tank	C	I	ACI 349/AISC N690	ACI 349 is used for Evaluation of Structural Boundary
PXS-MT-04	IRWST Gutter	C	I	Manufacturer Std.	
PXS-MW-01A	Reactor Coolant Depressurization Sparger A	C	I	ASME III-3	
PXS-MW-01B	Reactor Coolant Depressurization Sparger B	C	I	ASME III-3	
PXS-MY-Y01A	IRWST Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y01B	IRWST Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.

Table 3.2-3 (Sheet 16 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y01C	IRWST Screen C	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02A	Containment Recirculation Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02B	Containment Recirculation Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y03A	pH Adjustment Basket A	C	I	Manufacturer Std.	
PXS-MY-Y03B	pH Adjustment Basket B	C	I	Manufacturer Std.	
PXS-MY-Y03C	pH Adjustment Basket C	C	I	Manufacturer Std.	
PXS-MY-Y03D	pH Adjustment Basket D	C	I	Manufacturer Std.	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V002B	CMT B CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V010A	CMT A Upper Sample	B	I	ASME III-2	
PXS-PL-V010B	CMT B Upper Sample	B	I	ASME III-2	
PXS-PL-V011A	CMT A Lower Sample	B	I	ASME III-2	
PXS-PL-V011B	CMT B Lower Sample	B	I	ASME III-2	
PXS-PL-V012A	CMT A Drain	A	I	ASME III-1	
PXS-PL-V012B	CMT B Drain	A	I	ASME III-1	
PXS-PL-V013A	CMT A Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V013B	CMT B Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V014A	CMT A Discharge Isolation	A	I	ASME III-1	

Table 3.2-202
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

LNP DEP 3.2-1

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y01C	IRWST Screen C	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02A	Containment Recirculation Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02B	Containment Recirculation Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y03A	pH Adjustment Basket A	C	I	Manufacturer Std.	
PXS-MY-Y03B	pH Adjustment Basket B	C	I	Manufacturer Std.	
PXS-MY-Y03C	pH Adjustment Basket C	C	I	Manufacturer Std.	
PXS-MY-Y03D	pH Adjustment Basket D	C	I	Manufacturer Std.	
PXS-MY-Y81	Downspout Screen 1A	C	I	Manufacturer Std.	
PXS-MY-Y82	Downspout Screen 1B	C	I	Manufacturer Std.	
PXS-MY-Y83	Downspout Screen 1C	C	I	Manufacturer Std.	
PXS-MY-Y84	Downspout Screen 1D	C	I	Manufacturer Std.	
PXS-MY-Y85	Downspout Screen 2A	C	I	Manufacturer Std.	
PXS-MY-Y86	Downspout Screen 2B	C	I	Manufacturer Std.	
PXS-MY-Y87	Downspout Screen 2C	C	I	Manufacturer Std.	
PXS-MY-Y88	Downspout Screen 2D	C	I	Manufacturer Std.	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V002B	CMT B CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V010A	CMT A Upper Sample	B	I	ASME III-2	
PXS-PL-V010B	CMT B Upper Sample	B	I	ASME III-2	
PXS-PL-V011A	CMT A Lower Sample	B	I	ASME III-2	
PXS-PL-V011B	CMT B Lower Sample	B	I	ASME III-2	

Table 3.2-3 (Sheet 17 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V014B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015A	CMT A Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V016A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V016B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V017A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V017B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V019A	RNS to CMT Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V019B	RNS to CMT Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V020A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V020B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V021A	Accumulator A Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V021B	Accumulator B Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V022A	Accumulator A Pressure Relief	C	I	ASME III-3	
PXS-PL-V022B	Accumulator B Pressure Relief	C	I	ASME III-3	
PXS-PL-V023A	Accumulator A Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V023B	Accumulator B Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V024A	Accumulator A Pressure Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V024B	Accumulator B Pressure Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V025A	Accumulator A Sample	C	I	ASME III-3	
PXS-PL-V025B	Accumulator B Sample	C	I	ASME III-3	
PXS-PL-V026A	Accumulator A Drain	C	I	ASME III-3	
PXS-PL-V026B	Accumulator B Drain	C	I	ASME III-3	
PXS-PL-V027A	Accumulator A Discharge Isolation	C	I	ASME III-3	
PXS-PL-V027B	Accumulator B Discharge Isolation	C	I	ASME III-3	
PXS-PL-V028A	Accumulator A Discharge Check	A	I	ASME III-1	

Table 3.2-3 (Sheet 18 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V028B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V029A	Accumulator A Discharge Check	A	I	ASME III-1	
PXS-PL-V029B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V030A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V030B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V033A	Accumulator A Check Valve Drain	B	I	ASME III-2	
PXS-PL-V033B	Accumulator B Check Valve Drain	B	I	ASME III-2	
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC	B	I	ASME III-2	
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC	B	I	ASME III-2	
PXS-PL-V052	Accumulator Nitrogen Containment Penetration TC	B	I	ASME III-2	
PXS-PL-V080A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V080B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V082A	CMT A Upper Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V082B	CMT B Upper Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V083A	CMT A Upper Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V083B	CMT B Upper Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V084A	CMT A Upper Level A Vent	B	I	ASME III-2	
PXS-PL-V084B	CMT B Upper Level A Vent	B	I	ASME III-2	
PXS-PL-V085A	CMT A Upper Level A Drain	B	I	ASME III-2	
PXS-PL-V085B	CMT B Upper Level A Drain	B	I	ASME III-2	

Table 3.2-3 (Sheet 19 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V086A	CMT A Upper Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V086B	CMT B Upper Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V087A	CMT A Upper Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V087B	CMT B Upper Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V088A	CMT A Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V088B	CMT B Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V089A	CMT A Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V089B	CMT B Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V092A	CMT A Lower Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V092B	CMT B Lower Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V093A	CMT A Lower Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V093B	CMT B Lower Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V094A	CMT A Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V094B	CMT B Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V095A	CMT A Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V095B	CMT B Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V096A	CMT A Lower Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V096B	CMT B Lower Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V097A	CMT A Lower Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V097B	CMT B Lower Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V098A	CMT A Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V098B	CMT B Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V099A	CMT A Lower Level B Drain	B	I	ASME III-2	
PXS-PL-V099B	CMT B Lower Level B Drain	B	I	ASME III-2	

Table 3.2-3 (Sheet 20 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V101	PRHR HX Inlet Isolation	A	I	ASME III-1	
PXS-PL-V102A	PRHR HX Inlet Head Vent	B	I	ASME III-2	
PXS-PL-V102B	PRHR HX Inlet Head Drain	B	I	ASME III-2	
PXS-PL-V103A	PRHR HX Outlet Head Vent	B	I	ASME III-2	
PXS-PL-V103B	PRHR HX Outlet Head Drain	B	I	ASME III-2	
PXS-PL-V104A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	
PXS-PL-V104B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	
PXS-PL-V105A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	
PXS-PL-V105B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	
PXS-PL-V106	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V107	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V108A	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V108B	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V109	PRHR HX/RCS Return Isolation	A	I	ASME III-1	
PXS-PL-V111A	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V111B	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V113	PRHR HX Pressure Transmitter Isolation	B	I	ASME III-2	
PXS-PL-V115A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V115B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V116A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V116B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V117A	Containment Recirculation A Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 21 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V117B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V118A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V118B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V119A	Containment Recirculation A Check	C	I	ASME III-3	
PXS-PL-V119B	Containment Recirculation B Check	C	I	ASME III-3	
PXS-PL-V120A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V120B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V121A	IRWST Line A Isolation	C	I	ASME III-3	
PXS-PL-V121B	IRWST Line B Isolation	C	I	ASME III-3	
PXS-PL-V122A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V122B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V123A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V123B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V124A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V124B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V125A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V125B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V126A	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V126B	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V127	IRWST Injection Line A Drain	C	I	ASME III-3	
PXS-PL-V128A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V128B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V130A	IRWST Gutter Bypass A Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 22 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V130B	IRWST Gutter Bypass B Isolation	C	I	ASME III-3	
PXS-PL-V131A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V131B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V132A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V132B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V133A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V133B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V135A	IRWST Injection Line A Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V135B	IRWST Injection Line B Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V149	RNS Suction Pump Line Drain	C	I	ASME III-3	
PXS-PL-V150A	IRWST Level Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V150B	IRWST Level Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V150C	IRWST Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V150D	IRWST Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V151A	IRWST Level Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V151B	IRWST Level Transmitter B Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 23 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V151C	IRWST Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V151D	IRWST Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V201A	Accumulator A Leak Test	B	I	ASME III-2	
PXS-PL-V201B	Accumulator B Leak Test	B	I	ASME III-2	
PXS-PL-V202A	Accumulator A Leak Test	C	I	ASME III-3	
PXS-PL-V202B	Accumulator B Leak Test	C	I	ASME III-3	
PXS-PL-V205A	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V205B	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V206	RNS Discharge Leak Test	C	I	ASME III-3	
PXS-PL-V207A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V207B	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V208A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V217	PXS Leak Test Line Isolation	D	NS	ANSI B31.1	
PXS-PL-V221	Test Header to IRWST	D	NS	ANSI B31.1	
PXS-PL-V230A	CMT A Fill Isolation	B	I	ASME III-2	
PXS-PL-V230B	CMT B Fill Isolation	B	I	ASME III-2	
PXS-PL-V231A	CMT A Fill Check	B	I	ASME III-2	
PXS-PL-V231B	CMT B Fill Check	B	I	ASME III-2	
PXS-PL-V232A	Accumulator A Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V232B	Accumulator B Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V250A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V250B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252A	CMT A Check Valve Test Valve	A	I	ASME III-1	

Table 3.2-3 (Sheet 24 of 75)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V252B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PY-C01	Nitrogen Makeup Containment Penetration	B	I	ASME III, 2	
Balance of system components are Class E					
Reactor Coolant System (RCS)					Location: Containment
RCS-MB-01	Steam Generator 1	A	I	ASME III-1	
RCS-MB-02	Steam Generator 2	A	I	ASME III-1	
RCS-MP-01A/B	SG 1A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	
n/a	Pump Motor Cooling Water from HX Outlet Connector	A	I	ASME III-1	
RCS-MP-02A/B	SG 2A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This subsection is not applicable to the AP1000.

3.8.2 Steel Containment

NOTE: This section and figures excerpted from DCD R19 are provided for context - the change to figure 3.8.2-1 is reflected on page 51 of this document

3.8.2.1 Description of the Containment

3.8.2.1.1 General

This subsection describes the structural design of the steel containment vessel and its parts and appurtenances. The steel containment vessel is an integral part of the containment system whose function is described in Section 6.2. It serves both to limit releases in the event of an accident and to provide the safety-related ultimate heat sink.

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. Final detailed analyses will be documented in the ASME Design Report.

The containment arrangement is indicated in the general arrangement figures in Section 1.2. The portion of the vessel above elevation 132′-3″ is surrounded by the shield building but is exposed to ambient conditions as part of the passive cooling flow path. A flexible watertight and airtight seal is provided at elevation 132′-3″ between the containment vessel and the shield building. The portion of the vessel below elevation 132′-3″ is fully enclosed within the shield building.

Figure 3.8.2-1 shows the containment vessel outline, including the plate configuration and crane girder. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. [*The containment vessel has the following design characteristics:*

Diameter: 130 feet
Height: 215 feet 4 inches
Design Code: ASME III, Div. 1
Material: SA738, Grade B
Design Pressure: 59 psig
Design Temperature: 300°F
Design External Pressure: 1.7 psid
Lower Personnel Airlock: Elevation 110′-6″ and 107 degrees azimuth
Lower Equipment Hatch: Elevation 112′-6″ and 126 degrees azimuth
Upper Personnel Airlock Elevation 138′-7″ and 107 degrees azimuth
Upper Equipment Hatch Elevation 141′-6″ and 67 degrees azimuth
External Stiffener: Elevation 131′-9″
Internal Stiffener: Elevation 170′-0″
Bottom Head Tangent Line Elevation 104′-1 1/2″
Upper Head Tangent Line Elevation 244′-2 1/2″

The tangent line is the elevation at which the vessel transitions from the cylinder to the head.

The wall thickness in most of the cylinder is 1.75 inches. The wall thickness of the lowest course of the cylindrical shell is increased to 1.875 inches to provide margin in the event of corrosion in the embedment transition region. The thickness of the heads is 1.625 inches.]* The heads are [*ellipsoidal*]* with a major diameter of 130 feet and a height of 37 feet, 7.5 inches.

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments. The design for external pressure is dependent on the spacing of the hoop stiffeners and crane girder, which are shown on Figure 3.8.2-1. [*The spacing between each pair of ring supports (the bottom flange of the crane girder, the hoop stiffeners, and the concrete floor at elevation 100′-0″) is less than 50 feet, 6 inches. The design of the stiffeners and polar crane girder provides equal or greater radial and rotational stiffness than the design evaluated for the design certification.*]*

The polar crane is designed for handling the reactor vessel head during normal refueling. The crane girder and wheel assemblies are designed to support a special trolley to be installed in the event of steam generator replacement.

The containment vessel supports most of the containment air baffle as described in subsection 3.8.4. The air baffle is arranged to permit inspection of the exterior surface of the containment vessel. Steel plates are welded to the dome as part of the water distribution system, described in subsection 6.2.2. The polar crane system is described in subsection 9.1.5.

3.8.2.1.2 Containment Vessel Support

The bottom head is embedded in concrete, with concrete up to elevation 100′ on the outside and to the maintenance floor at elevation 107′-2″ on the inside. The containment vessel is assumed as an independent, free-standing structure above elevation 100′. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by shear studs, friction, and bearing. The shear studs are not required for design basis loads. They provide additional margin for earthquakes beyond the safe shutdown earthquake.

Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete. A typical cross section design of the seal is presented in Figure 3.8.2-8, sheets 1 and 2.

3.8.2.1.3 Equipment Hatches

Two equipment hatches are provided. One is at the operating floor (elevation 135′-3″) with an inside diameter of 16 feet. The other is at floor elevation 107′-2″ to permit grade-level access into the containment, with an inside diameter of 16 feet. The hatches, shown in Figure 3.8.2-2, consist

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

of a cylindrical sleeve with a pressure seated dished head bolted on the inside of the vessel. The containment internal pressure acts on the convex face of the dished head and the head is in compression. The flanged joint has double O-ring or gum-drop seals with an annular space that may be pressurized for leak testing the seals. Each of the two equipment hatches is provided with an electrically powered hoist and with a set of hardware, tools, equipment and a self-contained power source for moving the hatch from its storage location and installing it in the opening. [*The information in Figure 3.8.2-2 that is considered to be Tier 2* information is the minimum thickness of the hatch cover, the inside diameter of the sleeve, the diameter of the insert plate, the minimum thickness of the insert plate, and the nominal spherical radius of the hatch cover.*]*

3.8.2.1.4 Personnel Airlocks

Two personnel airlocks are provided, one located adjacent to each of the equipment hatches. Figure 3.8.2-3 shows the typical arrangement. Each personnel airlock has about a 10-foot external diameter to accommodate a door opening of width 3 feet 6 inches and height 6 feet 8 inches. The airlocks are long enough to provide a clear distance of 8 feet, which is not impaired by the swing of the doors within the lock. The airlocks extend radially out from the containment vessel through the shield building. They are supported by the containment vessel. [*Area reinforcement for the personnel airlocks is provided by a minimum of 3-3/4-inch-thick insert plates. The surface area of the personnel airlock insert plate, not including the sleeve, is a minimum of 51.9 ft*^{2}.]*

Each airlock has two double-gasketed, pressure-seated doors in series. The doors are mechanically interlocked to prevent simultaneous opening of both doors and to allow one door to be completely closed before the second door can be opened. The interlock can be bypassed by using special tools and procedures.

3.8.2.1.5 Mechanical Penetrations

The mechanical penetrations consist of the fuel transfer penetration and mechanical piping penetrations and are listed in Table 6.2.3-1. Area is added to the shell by the addition of an insert plate that is thicker than the shell or by increasing the thickness of the nozzle neck or a combination of both. This piping penetration design is then evaluated for external loads on the penetration imposed by the piping system.

Figure 3.8.2-4, sheet 1, shows typical details for the main steam penetration. This includes bellows to minimize piping loads applied to the containment vessel and a guardpipe to protect the bellows and to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the feedwater penetration. [*The main steam and feedwater penetrations are combined into a common 3-3/4-inch-thick insert plate. This thickness is a minimum value. The main steam penetration has an inside sleeve diameter of 57 inches. The feedwater penetration has an inside sleeve diameter of 38 inches.*]* The insert plates for the main steam and feedwater penetrations are shown in Figure 3.8.2-4, Sheet 7. The insert plate also includes the penetration for the 6-inch-diameter startup feedwater pipe. The insert plate is designed in accordance with NE-3330, “Openings and Reinforcement,” of the ASME Code.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Figure 3.8.2-4, sheet 2, shows typical details for the startup feedwater penetration. This includes a guardpipe to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the steam generator blowdown penetration.

Figure 3.8.2-4, sheet 3, shows typical details for the normal residual heat removal penetration. Similar details are used for other penetrations below elevation 107′-2″ where there is concrete inside the containment vessel. The flued head is integral with the process piping and is welded to the containment sleeve. The welds are accessible for in-service inspection. The containment sleeve is separated from the concrete by compressible material.

Figure 3.8.2-4, sheet 4 shows typical details for the other mechanical penetrations. These consist of a sleeve welded to containment with either a flued head welded to the sleeve (detail A), or with the process piping welded directly to the sleeve (detail B). Flued heads are used for stainless piping greater than 2 inches in nominal diameter and for piping with high operating temperatures.

Design requirements for the mechanical penetrations are as follows:

- Design and construction of the process piping follow ASME, Section III, Subsection NC. Design and construction of the remaining portions follow ASME Code, Section III, Subsection NE. The boundary of jurisdiction is according to ASME Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

- Guard pipes are designed for pipe ruptures as described in subsection 3.6.2.1.1.4.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the containment vessel. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. These cover plates are removable to permit in-service inspection.

The fuel transfer penetration, shown in Figure 3.8.2-4, sheet 5, is provided to transfer fuel between the containment and the fuel handling area of the auxiliary building. The fuel transfer tube is welded to the penetration sleeve. The containment boundary is a double-gasketed blind flange at the refueling canal end. The expansion bellows are not a part of the containment boundary. Rather, they are water seals during refueling operations and accommodate differential movement between the containment vessel, containment internal structures, and the auxiliary building.

3.8.2.1.6 Electrical Penetrations

Figure 3.8.2-4, sheet 6, shows a typical 18-inch-diameter electrical penetration. The penetration assemblies consist of conductor modules (or medium voltage cable modules in a similar 18-inch-diameter penetration) passing through a bulkhead attached to the containment nozzle. Electrical design of these penetrations is described in subsection 8.3.1.1.6.

Electrical penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly to verify that containment integrity is maintained. Design and testing is according to IEEE Standard 317-83 and IEEE Standard 323-74.

3.8.2.1.7 Instrument Line Penetrations

Instrument line penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

Figure 3.8.2-4, sheet 4, detail B, shows typical details for the containment pressure instrumentation penetrations. The penetrations consist of sleeves welded to the containment vessel. Pressure transmitters outside containment are connected to pressure sensors inside containment by sealed, fluid-filled tubing (capillary), which passes through the sleeves. The capillary tubing is welded directly to the sleeve at a tubing coupling, which has a thicker wall and larger diameter than the capillary tubing.

Design and construction of the penetrations are in accordance with ASME Section III. The penetration sleeves, including the welds to the tubing couplings, follow ASME Section III, Subsection NE. Because ASME Section III, Subsection NCA excludes the sealed-tubing instrument configuration from the scope of Section III, the capillary tubing is designed and fabricated in accordance with ASME B31.1.

3.8.2.2 Applicable Codes, Standards, and Specifications

[*The containment vessel is designed*]* and constructed [*according to the 2001 edition of the ASME Code, Section III, Subsection NE, Metal Containment, including the 2002 Addenda. Stability of the containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001.*]*

Structural steel nonpressure parts, such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in subsection 3.8.4.

Section 1.9 discusses compliance with the Regulatory Guides and the Standard Review Plans.

3.8.2.3 Loads and Load Combinations

Table 3.8.2-1 summarizes the design loads, load combinations and ASME Service Levels. They meet the requirements of the ASME Code, Section III, Subsection NE. The loads and load combinations used in the analysis are considered to be part of the method of evaluation. The containment vessel is designed for the following loads specified during construction, test, normal plant operation and shutdown, and during accident conditions:

- D Dead loads or their related internal moments and forces, including any permanent piping and equipment loads

- L Live loads or their related internal moments and forces, including crane loads

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

P_o Operating pressure loads during normal operating conditions resulting from pressure variations either inside or outside containment

T_o Thermal effects and loads during normal operating conditions, based on the most critical transient or steady-state condition

R_o Piping and equipment reactions during normal operating conditions, based on the most critical transient or steady-state condition

W Loads generated by the design wind on the portion of the containment vessel above elevation 132′, as described in subsection 3.3.1.1

E_s Loads generated by the safe shutdown earthquake (SSE) as described in Section 3.7

W_t Loads generated by the design tornado on the portion of the containment vessel above elevation 132′, as described in subsection 3.3.2

P_t Test pressure

P_d Containment vessel design pressure that exceeds the pressure load generated by the postulated pipebreak accidents and passive cooling function

P_e Containment vessel external pressure

T_a Thermal loads under thermal conditions generated by the postulated break or passive cooling function and including T_o. This includes variations around the shell due to the surrounding buildings and maldistribution of the passive containment cooling system water.

R_a Piping and equipment reactions under thermal conditions generated by the postulated break, as described in Section 3.6, and including R_o

Y_r Loads generated by the reaction on the broken high-energy pipe during the postulated break, as described in Section 3.6

Y_j Jet impingement load on a structure generated by the postulated break, as described in Section 3.6

Y_m Missile impact load on a structure generated by or during the postulated break, as from pipe whipping, as described in Section 3.6

Post-accident flooding load combination is not applicable in the design of the AP1000 containment vessel. The post-loss-of-coolant accident (LOCA) flooding event is enveloped by the other design cases.

The AP1000 addresses the production of large quantities of hydrogen from the oxidation of zirconium and other metals as a result of a postulated severe accident. The AP1000 includes hydrogen igniters inside containment to ensure that hydrogen generated in a severe accident is

burned prior to reaching an explosive mixture. The discussion of the generation and burning of hydrogen as a result of a severe accident is included in Section 19.41.

The containment vessel is protected from the direct effects of wind/tornado loads (and associated potential missiles) by virtue of its location inside the shield building. The differential pressure effects of a tornado are also reduced because of the location and are bounded by other pressure loadings for which the containment vessel is designed.

The containment is evaluated for the deterministic severe accident pressure capacity. This evaluation is discussed in subsection 3.8.2.4.2, "Evaluation of Ultimate Capacity." According to 10 CFR 50.44, the hydrogen generated pressure loads from 100 percent fuel clad-coolant reaction plus the peak pressure from a hydrogen burn must be less than ASME Service Level C (not including buckling). The Service Level C maximum capacity is 117 psig at 300°F as presented in subsection 3.8.2.4.2.8. The peak pressure from the 100 percent fuel clad-coolant reaction plus the hydrogen burn (Pg1 + Pg2) is 90.3 psig as reported in Section 41.11 and Table 41-4 of the AP1000 Probabilistic Risk Assessment report. The severe accident conditions are beyond design basis accidents, and the load combinations for these severe accident evaluations are not included in the load combinations and service limits for the containment vessel.

The AP1000 does not have a post-accident inerting system. Therefore, there is no load combination that includes inerting of the containment.

Note that loads associated with flooding of the containment below elevation 107′ are resisted by the concrete structures and not by the containment vessel.

3.8.2.4 Design and Analysis Procedures

The design and analysis procedures for the containment vessel are according to the requirements of the ASME Code, Section III, Subsection NE.

The analyses are summarized in Table 3.8.2-4. The detailed analyses will use a series of general-purpose finite element, axisymmetric shell and special purpose computer codes to conduct such analyses. Code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

3.8.2.4.1 Analyses for Design Conditions

3.8.2.4.1.1 Axisymmetric Shell Analyses

The containment vessel is modelled as an axisymmetric shell and analyzed using the ANSYS computer program. A model used for static analyses is shown in Figure 3.8.2-6.

Dynamic analyses of the axisymmetric model, which is similar to that shown in Figure 3.8.2-6, are performed to obtain frequencies and mode shapes. These are used to confirm the adequacy of the containment vessel stick model as described in subsection 3.7.2.3.2. Stress analyses are performed for each of the following loads:

- Dead load

- Internal pressure
- Seismic
- Polar crane wheel loads
- Wind loads
- Thermal loads

The seismic analysis performed envelopes all soil conditions. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. These accelerations are the maximum accelerations from the nuclear island stick model on hard rock. The global member forces from the equivalent static case exceed those from the soil cases for soil conditions described in Appendix 3G. Based on these comparisons, the design acceleration values used for the global analyses are appropriate for both the hard rock and the soil sites. The seismic analysis of the nuclear island is discussed in Section 3.7 and Appendix 3G. The torsional moments, which include the effects of the eccentric masses, are increased to account for accidental torsion and are evaluated in a separate calculation.

The results of these load cases are factored and combined in accordance with the load combinations identified in Table 3.8.2-1. These results are used to evaluate the general shell away from local penetrations and attachments, that is, for areas of the shell represented by the axisymmetric geometry. The results for the polar crane wheel loads are also used to establish local shell stiffnesses for inclusion in the containment vessel stick model described in subsection 3.7.2.3. The results of the analyses and evaluations are included in the containment vessel design report.

Design of the containment shell is primarily controlled by the internal pressure of 59 psig. The meridional and circumferential stresses for the internal pressure case are shown in Figure 3.8.2-5. The most highly stressed regions for this load case are the portions of the shell away from the hoop stiffeners and the knuckle region of the top head. In these regions the stress intensity is close to the allowable for the design condition.

Table 3.8.2-1 includes a design load combination to address external pressure. For the design external pressure, a conservatively large magnitude of 1.7 psi differential pressure is used. Design external pressure is defined as a value greater than the external pressure at which the vacuum relief system will open and mitigate the external pressure. This is a part of the containment air filtration system (see subsection 9.4.7). Upon actuation, the external pressure transient is immediately controlled and the external pressure is relieved. This design external pressure is combined with a coincident -40°F outside air temperature, which corresponds to a -18.5°F metal temperature for the portions of the containment vessel shell not insulated from ambient conditions. The portions of the containment vessel shell that are below the external stiffener are insulated from the cold outside air conditions and result in a metal temperature of 70°F.

A bounding case was analyzed to provide an indication of the margin to acceptance criteria associated with the minimum allowable service metal temperature for the AP1000 containment vessel. Various types of transients were considered to evaluate the minimum service metal temperature, including inadvertent fan cooler cases, inadvertent passive containment cooling system (PCS) actuation, and loss of ac power. The evaluation considered variations in initial

conditions for parameters, including humidity, internal temperature, external temperature, and wind speed. These evaluations demonstrate that the -18.5°F service metal temperature is adequate.

Design external pressure is used in load combinations that include thermal loads and are used to evaluate Service Level A and D stress limits. These external pressure conditions are included in the loading combinations in Table 3.8.2-1.

Operating pressures range from -0.2 psig to 1.0 psig, which are then combined with an ambient temperature for the containment vessel. Design internal pressure is 59 psig combined with a containment vessel metal temperature of 300°F to be evaluated in the ASME service limits as well as the design conditions.

A load combination that combines design wind plus internal design pressure is not included in Table 3.8.2-1 because the wind loads are small (within the normal operating range for containment pressure) and because the combination of the design wind and accident pressure is a lower probability than either the design wind or the accident pressure acting alone.

Major loads that induce compressive stresses in the containment vessel are internal and external pressure and crane and seismic loads. Each of these loads and the evaluation of the compressive stresses are discussed below.

- Internal pressure causes compressive stresses in the knuckle region of the top head and in the equipment hatch covers. The evaluation methods are similar to those discussed in subsection 3.8.2.4.2 for the ultimate capacity.

- Evaluation of external pressure loads is performed in accordance with ASME Code, Section III, Subsection NE, Paragraph NE-3133.

- Crane wheel loads due to crane dead load, live load, and seismic loads result in local compressive stresses in the vicinity of the crane girder. These are evaluated in accordance with ASME Code, Case N-284.

- Overall seismic loads result in axial compression and tangential shear stresses at the base of the cylindrical portion. These are evaluated in accordance with ASME Code, Case N-284.

The bottom head is embedded in the concrete base at elevation 100 feet. This leads to circumferential compressive stresses at the discontinuity under thermal loading associated with the design basis accident. The containment vessel design includes a Service Level A combination in which the vessel above elevation 107′-2″ is specified at the design temperature of 300°F and the portion of the embedded vessel (and concrete) below elevation 100 feet is specified at a temperature of 70°F. The temperature profile for the vessel is linear between these elevations. Containment shell buckling close to the base is evaluated against the criteria of ASME Code, Case N-284.

Revision 1 of Code Case N-284 is used for the evaluation of the containment shell and equipment hatches.

3.8.2.4.1.2 Local Analyses

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response (Reference 53).

The personnel airlocks and equipment hatches are modeled in a 3-D shell finite element model of the containment. A 3D shell, finite element model of the containment vessel was developed in ANSYS to consider the effect of the penetrations and their quasi-static response due to a seismic event. The large masses and local stiffness of the personnel locks and equipment hatches are discretely modeled. The polar crane wheel loads are incorporated by appropriate loadings (dead load and seismic loadings). The bottom of the model is fixed at elevation 100′ where the containment vessel is embedded in concrete. This means that rotations and displacements are conservatively fixed at this location.

Static analyses are performed using the finite element model shown in Figure 3.8.2-7 for internal pressure, dead load (including the polar crane in the parked position), thermal loads and seismic loads. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. The amplified local responses are included separately for each of the four penetrations. Local seismic axial and rotational accelerations about both horizontal and vertical axes are applied based on the maximum amplified response determined from a time history analysis on a less refined dynamic model with seismic time histories at elevation 100′.

Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE for the load combinations described in Table 3.8.2-1. Stability is evaluated against ASME Code Case N-284. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1700 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

3.8.2.4.2 Evaluation of Ultimate Capacity

The capacity of the containment vessel has been calculated for internal pressure loads for use in the probabilistic risk assessment analyses and severe accident evaluations. These analyses include the evaluation of the peak pressure from the hydrogen-generated pressure loads from 100-percent fuel cladding metal-water reaction plus the hydrogen burn. Each element of the containment vessel boundary was evaluated to estimate the maximum pressure at an ambient temperature of 100°F corresponding to the following stress and buckling criteria:

- Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME paragraph NE-3222, and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY 93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outline in the Staff Requirements

Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, Dated July 21, 1993.

- Best estimate capacity corresponding to gross membrane yield at the ASME-specified minimum yield stress (SA738, Grade B, yield stress = 60 ksi, ultimate stress = 85 ksi), and critical buckling for the equipment hatch covers and top head.

The results are shown in Table 3.8.2-2. The analyses at a temperature of 100°F are described in the following paragraphs for each element. The critical regions identified in this table are then examined further for their response at higher temperatures. This results in the best-estimate capacity based on the ASME-specified minimum yield properties. The evaluation considered the containment boundary elements including:

- Cylindrical shell
- Top and bottom heads
- Equipment hatches and covers
- Personnel airlocks
- Mechanical and electrical penetrations

The evaluation identified the most likely failure mode to be that associated with gross yield of the cylindrical shell. Loss of containment function would be expected to occur because the large post-yield deflections would lead to local failures at penetrations, bellows, or other local discontinuities.

3.8.2.4.2.1 Tensile Stress Evaluation of Shell

Results of the axisymmetric analyses of the cylinder and top head described in subsection 3.8.2.4.1 for dead load and internal pressure were evaluated to determine the pressure at which stresses reach yield at an ambient temperature of 100°F. The analyses assume the shell is fixed at elevation 100′, where the bottom head is embedded in concrete. The steel bottom head is identical to the top head and has a pressure capability greater than the top head due to the additional strength of the embedment concrete.

The allowable stress intensity under Service Level C loads is equal to yield. This corresponds to an internal pressure of 135 psig. The critical section is the cylinder, where the general primary membrane stress intensity is greatest.

The best-estimate yield analysis uses the von Mises criterion to establish yield rather than the more conservative ASME stress intensity approach. This increases the yield stress by about 15 percent for the cylinder, where the longitudinal stress is equal to one-half of the hoop stress resulting in first yield at an internal pressure of 155 psig. At this pressure, hoop stresses in the cylinder reach yield. The radial deflection is about 1.6 inches. As pressure increases further, large deflections occur. For a material such as SA738, where the yield plateau extends from a strain of 0.2 percent to 0.6 percent, deflections would increase to 4.8 inches at yield without a substantial increase in pressure. Strain hardening would then permit a further increase in pressure with large radial deflections, as described in subsection 3.8.2.4.2.6.

3.8.2.4.2.2 Buckling Evaluation of Top Head

The top head has a radius-to-height ratio of 1.728. This is not as shallow as most ellipsoidal or torispherical heads, which typically have a radius-to-height ratio of 2. The ratio was specifically selected to minimize the local stresses and buckling in the knuckle region due to internal pressure. As the ratio decreases, the magnitude of compressive stresses in the knuckle region decreases; for a radius-to-height ratio of 1.4 or smaller, there are no compressive stresses and therefore there is no potential for buckling.

Theoretical Buckling Capacity

The top head was analyzed using the BOSOR-5 computer code (Reference 1). This code permits consideration of both large displacements and nonlinear material properties. It calculates shell stresses and checks stability at each load step. The analysis included a portion of the cylinder with a thickness of 1.625 inches. In this analysis, yield of the cylinder started at a pressure of 144 psig using elastic – perfectly plastic material properties, a yield stress of 60 ksi, and the von Mises yield criterion. Yield of the top of the crown started at an internal pressure of 146 psig. Yield of the knuckle region started at 152 psig. A theoretical plastic buckling pressure of 174 psig was determined. At this pressure, the maximum effective prebuckling strain was 0.23 percent in the knuckle region where buckling occurred and 2.5 percent at the crown. The maximum deflection at the crown was 15.9 inches. A similar analysis was performed using nonlinear material properties considering the effects of residual stresses; buckling did not occur in this analysis, and failure would occur once strains at the crown reach ultimate. The failure mode was found to be an axisymmetric plastic collapse resulting from excessive vertical displacements at the crown. The maximum displacement was 43 inches at 195 psig.

Predicted Pressure Capacity

The actual buckling capacity may be lower than the theoretical buckling capacity because of effects not included in the analysis such as imperfections and residual stresses. This is considered by the use of capacity reduction factors that are based upon a correlation of theory and experiment. The capacity reduction factor for the top head was evaluated based on comparisons of BOSOR-5 analyses against test results of ellipsoidal and torispherical heads. This evaluation is described below and concludes that no reduction in capacity need be considered; that is, a capacity reduction factor of 1.0 is appropriate.

The knuckle region of ellipsoidal and torispherical heads is subjected to meridional tension and circumferential compression. The meridional tension tends to stabilize the knuckle region and reduces its sensitivity to imperfection. The radius-to-height ratio of 1.728 of the AP1000 head results in a larger ratio of meridional tension to circumferential compression than on shallower heads, further reducing the sensitivity to imperfection.

Welding Research Council Bulletin 267 (Reference 22) shows a comparison of BOSOR-5 predictions of buckling against the results of 20 tests of small head models. These results are summarized in Table 4 of the reference and show ratios (capacity reduction factors) of actual buckling to the BOSOR-5 prediction with an average of 1.2. Only one of the 20 cases shows a capacity reduction factor less than 1.0.

Table 3.8.2-3 shows the key parameters, test results, and BOSOR-5 predictions for two large, fabricated 2:1 torispherical heads tested and reported in NUREG/CR-4926 (Reference 23). The theoretical plastic buckling pressure predicted by BOSOR-5 represents initial buckling based on actual material properties. The initial buckling did not cause failure for either of the tests, and test pressure continued to increase until rupture occurred in the spherical cap. The collapse pressures were three to four times the initial buckling pressures.

- **Test Head 1** – The test result of 58 psig is 79 percent of the predicted theoretical plastic buckling pressure of 74 psig. Many of the buckles occurred directly on the meridional weld seams of the knuckle. The knuckle welds were noticeably flatter than the corresponding welds of the Test 2 head. The as-built configuration extended inside the theoretical shape at some of the meridional weld seams and was most pronounced at the location of the first observed buckle. Model 1 exceeded the tolerances for formed heads specified for containment vessels in NE-4222.2 of ASME, Section III, Subsection NE.

- **Test Head 2** – The test result of 106 psi is 100 percent of the BOSOR-5 predicted theoretical plastic buckling pressure. For test head 2, the welds had no noticeable flat spots and there was a smooth transition between the sphere and knuckle sections. Test head 2 was well within the Code allowable deviations.

The low-capacity reduction factor of 0.79 for test head 1 is attributed to excessive imperfections associated with the fabrication of relatively thin plate (0.196 inch). These imperfections were visible and were outside the tolerances permitted by the ASME Code. The results of test head 1 are therefore not considered applicable to the AP1000. The results of test head 2 and of the small-scale models described in the Welding Research Council Bulletin support the application of a capacity reduction factor of 1.0.

The capacity of the AP1000 head was also investigated using an approach similar to that permitted in ASME Code, Case N284. This code case provides alternate rules for certain containment vessel geometries such as cylindrical shells. The theoretical elastic buckling pressure was calculated to be 536 psi using the linear elastic computer code, BOSOR-4 (Reference 24). A reduction factor (defined as the product of the capacity reduction factor and the plastic reduction factor) was established as 0.385 based on the lower bound curve of test results of 20 ellipsoidal and 28 torispherical test specimens, which also include the two large fabricated heads previously discussed. This resulted in a predicted buckling capacity of 206 psig.

The preceding paragraphs addressed incipient buckling. It is concluded that buckling would not occur prior to reaching the pressure of 174 psig predicted in the BOSOR-5 analyses. Tests indicate that pressure can be significantly increased prior to rupture after the formation of the initial buckles. Failure would occur when local strains reach ultimate either close to a local buckle in the knuckle or at the center of the crown. The best estimate capacity of the head is taken as the theoretical plastic buckling pressure of 174 psig predicted in the BOSOR-5 analyses.

The deterministic severe accident pressure capacity is taken as 60 percent of critical buckling. This is consistent with the safety factor for Service Level C in ASME Code, Case N-284 and results in a containment head capacity of 104 psig.

3.8.2.4.2.3 Equipment Hatches

SECY 93-087 permits evaluation of certain severe accident scenarios against ASME Service Level C limits. The equipment hatch covers were evaluated for buckling against ASME paragraph NE-3222 and according to ASME Code, Case N-284. Use of ASME Code, Case N-284 for this application was confirmed to be appropriate by ASME. The containment internal pressure acts on the convex face of the dished head and the hatch covers are in compression under containment internal pressure loads. The critical buckling capacity is based on classical buckling capacities reduced by capacity reduction factors to account for the effects of imperfections and plasticity. These capacity reduction factors are based on test data and are generally lower-bound values for the tolerances specified in the ASME Code.

The critical buckling pressure is 211 psig for the 16-foot-diameter hatch at an ambient temperature of 100°F. For the Service Level C limits in accordance with paragraph NE 3222, a safety factor of 2.50 is specified, resulting in capabilities of 84 psig (16-foot-diameter). For the Service Level C limits in accordance with Code Case N284, a safety factor of 1.67 is specified, resulting in capabilities of 126 psig (16-foot-diameter).

Typical gaskets have been tested for severe accident conditions as described in NUREG/CR-5096 (Reference 25). The gaskets for the AP1000 will be similar to those tested with material such as Presray EPDM E 603. For such gaskets the onset of leakage occurred at a temperature of about 600°F.

3.8.2.4.2.4 Personnel Airlocks

The capacity of the personnel airlocks was determined by comparing the airlock design to that tested and reported in NUREG/CR-5118 (Reference 3). Critical parameters are the same, so the results of the test apply directly. In the tests the inner door and end bulkhead of the airlock withstood a maximum pressure of 300 psig at 400°F. The capacity of the airlock is therefore at least 300 psig at ambient temperature. The maximum pressure corresponding to Service Level C is conservatively estimated by reducing this capacity in the ratio of the minimum specified material yield to ultimate.

3.8.2.4.2.5 Mechanical and Electrical Penetrations

Subsections 3.8.2.1.3 through 3.8.2.1.6 describe the containment penetrations. Penetration reinforcement is designed following the area replacement method of the ASME Code. The insert plates and sleeves permit development of the hoop tensile yield stresses predicted as the limiting capacity in subsection 3.8.2.4.1. Capacities of the equipment hatch covers are discussed in subsection 3.8.2.4.2.3 and of the personnel airlocks in subsection 3.8.2.4.2.4.

Mechanical penetrations welded directly to the containment vessel are generally piping systems with design pressures greater than that of the containment vessel. Thicknesses of the flued head or end plate are established based on piping support loads or stiffness requirements. The capacities of these penetrations are greater than the capacity of the containment vessel cylinder.

Mechanical penetrations for the large-diameter high-energy lines, such as the main steam and feedwater piping, include expansion bellows. The piping and flued head have large pressure

capability. The response of expansion bellows to severe pressure and deformations is described in NUREG/CR-5561 (Reference 4). The bellows can withstand large pressure loading but may tear once the containment vessel deflection becomes large. Testing reported in NUREG/CR-6154 (Reference 26) has shown that the bellows remain leaktight even when subjected to large deflections sufficient to fully compress the bellows. Such large deflections do not occur as long as the containment vessel remains elastic. As described in subsection 3.8.2.4.2.6, the radial deflection of the shell increases substantially once the containment cylinder yields. The resulting deflections are assumed to cause loss of containment function. The containment penetration bellows are designed for a pressure of 90 psig at design temperature within Service Level C limits, concurrent with the relative displacements imposed on the bellows when the containment vessel is pressurized to these magnitudes.

Electrical penetrations have a pressure boundary consisting of the sleeve and an end plate containing a series of modules. The electrical pressure boundary is designed and built to the requirements of the ASME Code, Section III, Class MC, Subsection NE. The pressure capacity of these elements is large and is greater than the capacity of the containment vessel cylinder at temperatures up to the containment design temperature. Electrical penetration assemblies are also designed to satisfy ASME Service Level C stress limits under a pressure of 90 psig at design temperature. Tests at pressures and temperatures representative of severe accident conditions are described in NUREG/CR-5334 (Reference 5), where typical nuclear industry penetrations were irradiated, aged, then tested. One design was tested to 135 psia at 700°F. Other electrical penetration assemblies were tested to 75 psia at 400°F and 155 psia at 361°F. These tests showed that the electrical penetration assemblies withstand severe accident conditions. The electrical penetration assemblies are qualified for the containment design basis event conditions as described in Appendix 3D. The assemblies are similar to one of those tested by Sandia as reported in NUREG/CR-5334 (Reference 5). The ultimate pressure capacity of the electrical penetration assemblies is primarily determined by the temperature. The maximum temperature of the containment vessel below the operating deck during a severe accident is below the temperature at which the assemblies from the three suppliers in the Sandia tests were tested.

3.8.2.4.2.6 Material Properties

The containment vessel is designed using SA738, Grade B material. This has a specified minimum yield of 60 ksi and ultimate of 85 ksi. Test data for materials having similar chemical properties were reviewed. In a sample of 122 tests for thicknesses equaling or exceeding 1.50 inches and less than 1.75 inches, the actual yield had a mean value of 69.1 ksi with a standard deviation of 3.3 ksi. Thus, the actual yield is expected to be about 15 percent higher than the minimum yield. Membrane yield of the cylinder is predicted to occur at an internal pressure of 178 psig.

A stress-strain curve for material with chemistry similar to SA738, Grade B, indicated constant yield stress of 81.3 ksi from a strain of 0.002 to 0.006 followed by strain-hardening up to a maximum stress of 94.5 ksi at a strain of 0.079. The first portion of the strain-hardening is nearly linear, with a stress of 90 ksi at a strain of 4 percent. This strain occurs at a stress 10 percent above yield. Thus, a pressure load 10 percent higher than that corresponding to yield of the shell would result in 4 percent strain and a 31-inch radial deflection of the containment cylinder. Such a deflection is expected to cause major distress for penetrations, the air flow path, and local areas

where other structures are close to the containment vessel. Loss of function is therefore assumed for the containment once gross yield of the containment cylinder occurs.

3.8.2.4.2.7 Effect of Temperature

The evaluations described in the preceding subsections are based on an ambient temperature of 100°F. Nonmetallic items, such as gaskets, are qualified to function at the design temperature. The capacity of steel elements is reduced in proportion to the reduction due to temperature in yield stress, ultimate stress, or elastic modulus. The cylinder is governed by yield stress, and elastic buckling of the hatch covers is governed by the elastic modulus. The reduction in capacity is estimated using the tables given for material properties in the ASME Code. At 400°F, the yield stress is reduced by 17 percent and the pressure capacity corresponding to gross yield is reduced from 155 to 129 psig.

3.8.2.4.2.8 Summary of Containment Pressure Capacity

The ultimate pressure capacity for containment function is expected to be associated with leakage caused by excessive radial deflection of the containment cylindrical shell. This radial deflection causes distress to the mechanical penetrations, and leakage would be expected at the expansion bellows for the main steam and feedwater piping. There is high confidence that this failure would not occur before stresses in the shell reach the minimum specified material yield. This is calculated to occur at a pressure of 155 psig at ambient temperature and 129 psig at 400°F. Failure would be more likely to occur at a pressure about 15 percent higher based on expected actual material properties.

The deterministic severe accident pressure that can be accommodated according to the ASME Service Level C stress intensity limits and using a factor of safety of 1.67 for buckling of the top head is determined by the capacity of the 16-foot-diameter equipment hatch cover and the ellipsoidal head. The maximum capacity of the hatch cover, calculated according to ASME paragraph NE-3222, Service Level C, is 84 psig at an ambient temperature of 100°F and 81 psig at 300°F. When calculated in accordance with ASME Code, Case N-284, Service Level C, the maximum capacity is 126 psig at an ambient temperature of 100°F and 121 psig at 300°F. The maximum capacity of the ellipsoidal head is 104 psig at 100°F and 91 psig at 300°F.

The maximum pressure that can be accommodated according to the ASME Service Level C stress intensity limits, excluding evaluation of instability, is determined by yield of the cylinder and is 135 psig at an ambient temperature of 100°F and 117 psig at 300°F. This limit is used in the evaluations required by 10 CFR 50.44.

3.8.2.5 Structural Criteria

The containment vessel is designed, fabricated, installed, and tested according to the ASME Code, Section III, Subsection NE, and will receive a code stamp.

Stress intensity limits are according to ASME Code, Section III, Paragraph NE-3221 and Table NE-3221-1. [*Critical buckling stresses are checked according to the provisions of ASME Code, Section III, Paragraph NE-3222, or ASME Code Case N-284.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Materials for the containment vessel, including the equipment hatches, personnel locks, penetrations, attachments, and appurtenances meet the requirements of NE-2000 of the ASME Code. [*The basic containment material is SA-738, Grade B, plate. The procurement specification for the SA-738, Grade B, plate includes SA-20 supplemental requirements S1, Vacuum Treatment and S20, Maximum Carbon Equivalent for Weldability*.]* This material has been selected to satisfy the lowest service metal temperature requirement of -18.5°F. This temperature is established by analysis for the portion of the vessel exposed to the environment when the minimum ambient air temperature is -40°F. Impact test requirements are as specified in NE-2000.

[*The material of construction for the insert plates and fabricated nozzle necks of penetrations is SA-738 Grade B. The material of construction for forged nozzle neck forgings is SA-350, LF2, Class 1 for penetrations greater than 2 inches nominal diameter and less than 24 inches inside diameter, and for the containment air filtration system penetration nozzle necks. The maximum carbon equivalent for the SA-350, LF2, Class 1 used for penetrations is 0.52 percent. A vacuum refining process is required for SA-350, LF2, Class 1. The SA-350, LF2, Class 1 material is used in portions of the containment vessel that are below the external stiffener.*]* These portions of the containment vessel are insulated from the cold outside air temperature. Insulation is provided around the upper equipment hatch and personnel airlock, including the insert plates, in order to insulate the penetrations from the ambient air temperature in the upper annulus.

The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor and up to 8 feet above the operating floor also has a phenolic top coat. Below elevation 100′ the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations (see Figure 3.8.2-4, sheet 3 of 6, for typical details). Embedding the steel vessel in concrete protects the steel from corrosion.

The AP1000 configuration is shown in the general arrangement figures in Section 1.2 and in Figure 3.8.2-1. The exterior of the vessel is embedded at elevation 100′ and concrete is placed against the inside of the vessel up to the maintenance floor at elevation 107′-2″. Above this elevation the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete.

Seals are provided at the surface of the concrete inside and outside the vessel so that moisture is not trapped next to the steel vessel just below the top of the concrete. The seal on the inside accommodates radial growth of the vessel due to pressurization and heatup.

The plate thickness for the first course (elevation 104′-1.5″ to 116′-10″) of the cylinder is 1.875 inches, which is 1/8-inch thicker than the rest of the vessel. This provides margin in the event there would be any corrosion in the transition region despite the coatings and seals described previously. Equivalent margin is available for the 1.625-inch-thick bottom head in the transition region (elevation 100′ to 104′-1.5″). The plate thickness for the head is a constant thickness and is established by the stresses in the knuckle. As a result, the pressure stresses in the transition zone are well below the allowable stress, providing margin in the event of corrosion in this region.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The quality control program involving welding procedures, erection tolerances, and nondestructive examination of shop- and field-fabricated welds conforms with Subsections NE-4000 and NE-5000 of the ASME Code. The containment vessel is designed to permit its construction using large subassemblies. These subassemblies consist of the two heads and three ring sections. Each ring section comprises three or four courses of plates and is approximately 38 to 51 feet high. These are assembled in an area near the final location, using plates fabricated in a shop facility.

3.8.2.7 Testing and In-Service Inspection Requirements

Testing of the containment vessel and the pipe assemblies forming the pressure boundary within the containment vessel will be according to the provisions of NE-6000 and NC-6000, respectively.

Subsection 6.2.5 describes leak-rate testing of the containment system including the containment vessel.

In-service inspection of the containment vessel will be performed. See Section 6.6 for information on inservice inspection for the containment vessel and penetrations.

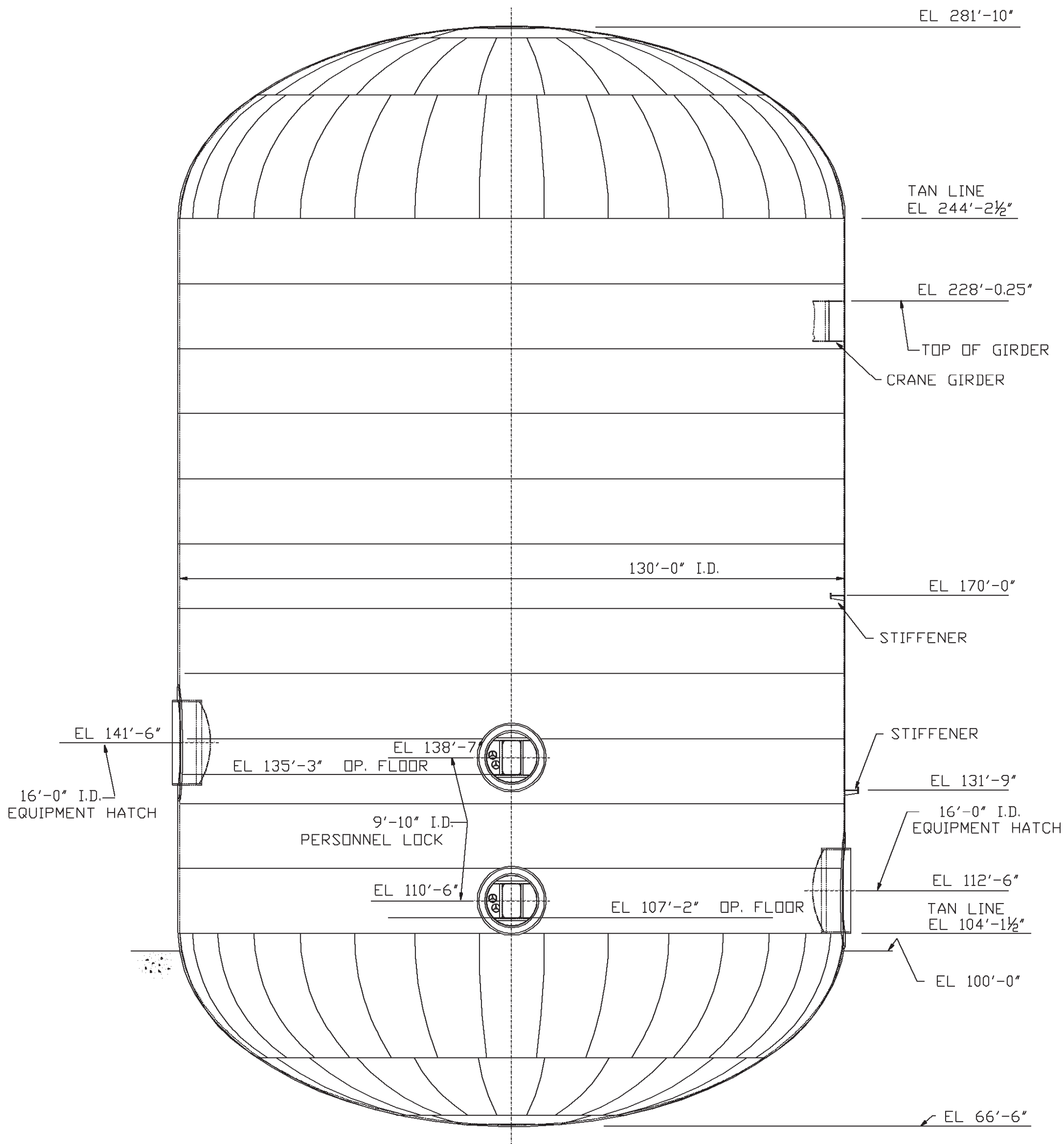
3.8.3 Concrete and Steel Internal Structures of Steel Containment

3.8.3.1 Description of the Containment Internal Structures

The containment internal structures are those concrete and steel structures inside (not part of) the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. The concrete and steel structures also provide radiation shielding. The containment internal structures are shown on the general arrangement drawings in Section 1.2. The containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms. The polar crane girders are considered part of the containment vessel. They are described in subsection 3.8.2.

Component supports are those steel members designed to transmit loads from the reactor coolant system to the load-carrying building structures. The component configuration is described in this subsection including the local building structure backing up the component support. The design and construction of the component supports are described in subsection 5.4.10.

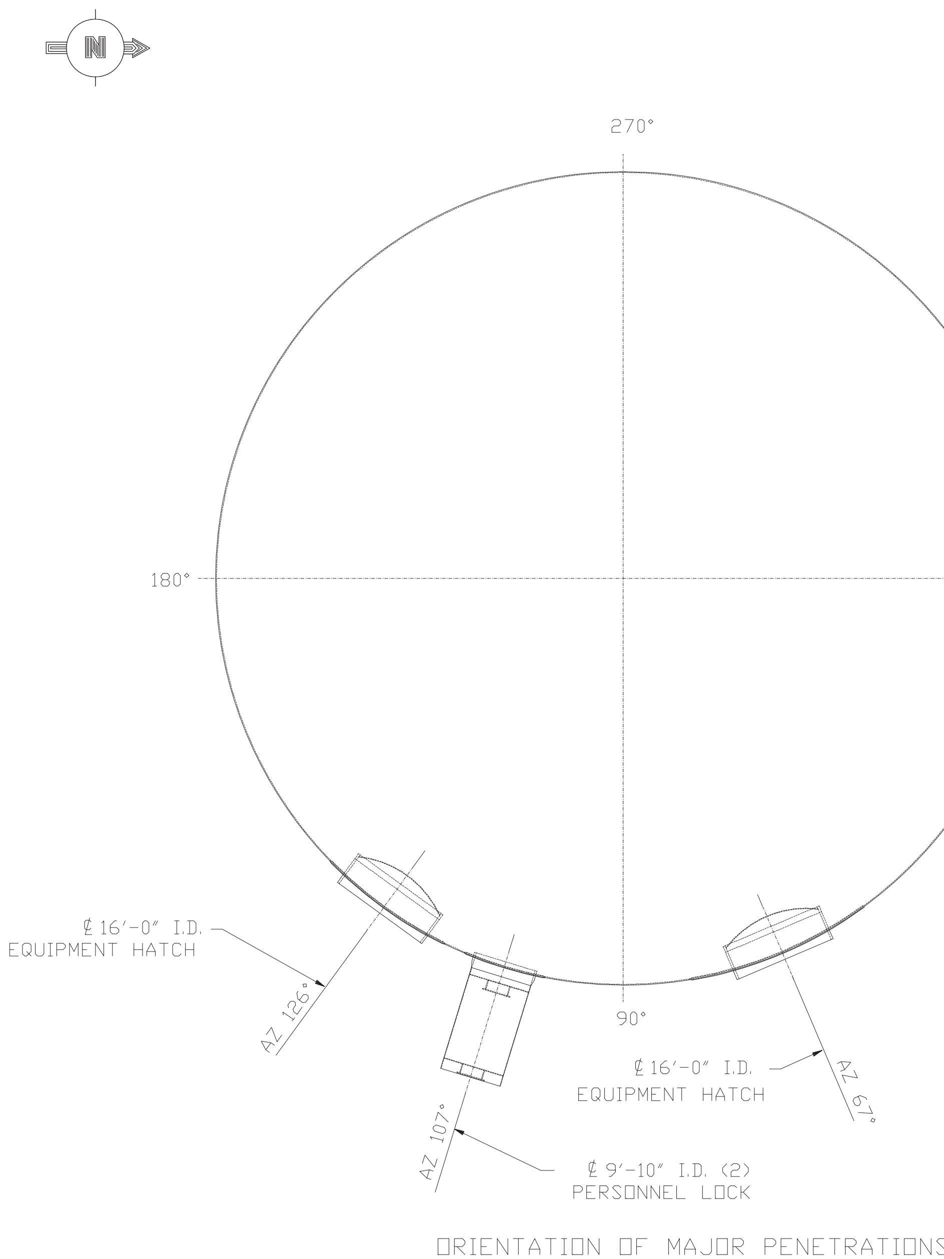
The containment internal structures are designed using reinforced concrete and structural steel. At the lower elevations conventional concrete and reinforcing steel are used, except that permanent steel forms are used in some areas in lieu of removable forms based on constructibility considerations. These steel form modules (liners) consist of plate reinforced with angle stiffeners and tee sections, as shown in Figure 3.8.3-16. The angles and the tee sections are on the concrete side of the plate. Welded studs, or similar embedded steel elements, are attached on the concrete face of the permanent steel form where surface attachments transfer loads into the concrete. Where these surface attachments are seismic Category I, the portion of the steel form module transferring the load into the concrete is classified as seismic Category I.



Equipment hatches rotated out of position to show configuration: see Sheet 2 for azimuth orientation.

Figure 3.8.2-1 (Sheet 1 of 3)

Containment Vessel General Outline

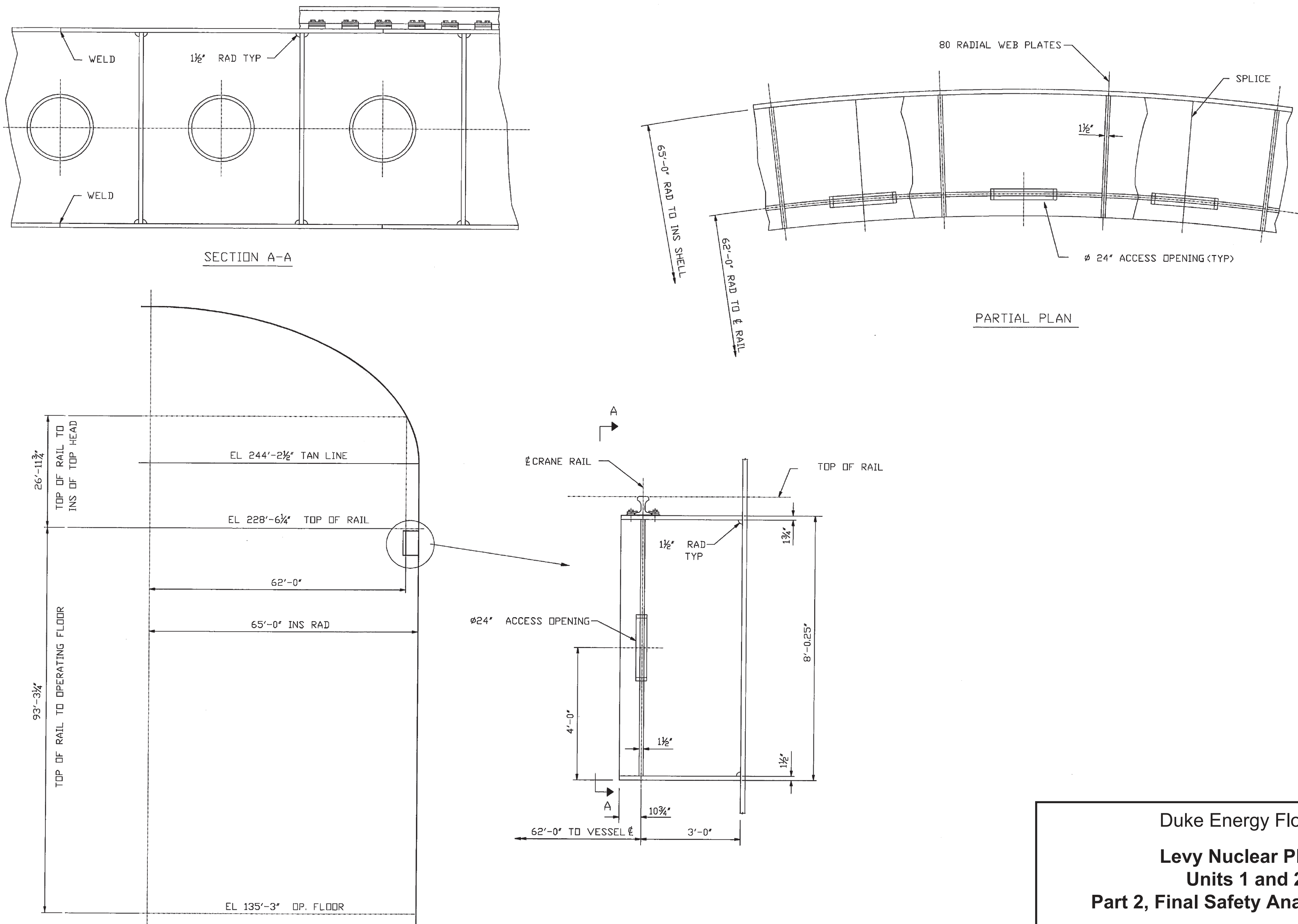


Equipment hatch at azimuth of 126° is the lower hatch.

Equipment hatch at azimuth of 67° is the upper hatch.

Figure 3.8.2-1 (Sheet 2 of 3)

Containment Vessel General Outline



Duke Energy Florida

Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

Containment Vessel General Outline

FIGURE 3.8-201

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5.4.11 Pressurizer Relief Discharge

The AP1000 does not have a pressurizer relief discharge system. The AP1000 has neither power operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP1000.

The safety valves connected to the top of the pressurizer provide for overpressure protection of the reactor coolant system. First-, second-, and third-stage automatic depressurization system valves provide for depressurization of the reactor coolant system and venting of noncondensable gases in the pressurizer following an accident. These functions are discussed in subsections 5.2.2, 5.4.12, and in Section 6.3. The AP1000 does not have power operated relief valves connected to the pressurizer.

The discharge of the safety valves is directed through a rupture disk to containment atmosphere.

The discharge of the first-, second-, and third-stage automatic depressurization system valves is directed to the in-containment refueling water storage tank. For the automatic depressurization system valves, the following discussion considers only the gas venting function. Only the first stage automatic depressurization valves are used to vent non-condensible gases following an accident. The sizing considerations and design basis for the in-containment refueling water storage tank for the depressurization function are discussed throughout Section 6.3. The provisions to minimize the differential pressure between the containment atmosphere and the interior of the in-containment refueling water storage tank are also discussed in subsection 6.3.2.

The safety valve on the normal residual heat removal system, which provides low temperature overpressure protection, discharges into the containment atmosphere. See subsection 5.4.7 for a discussion of the connections to and location of the safety valve in the normal residual heat removal system.

5.4.11.1 Design Bases

The containment has the capability to absorb the pressure increase and heat load resulting from the discharge of the safety valves to containment atmosphere. The in-containment refueling water storage tank has the capability to absorb the pressure increase and heat load from the discharge, including the water seal, steam and gases, from a first-stage automatic depressurization system valve when used to vent noncondensable gases from the pressurizer following an accident. The venting of noncondensable gases from the pressurizer following an accident is not a safety-related function.

5.4.11.2 System Description

Each safety valve discharge is directed to a rupture disk at the end of the discharge piping. A small pipe is connected to the discharge piping to drain away condensed steam leaking past the safety valve. The discharge is directed away from any safety related equipment, structures, or supports that could be damaged to the extent that emergency plant shutdown is prevented by such a discharge.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-~~2~~1. The in-containment refueling water storage tank is a stainless steel lined compartment integrated into the containment interior structure. The discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank design pressure. Additionally, vents on the top of the tank protect the tank from overpressure, as described in subsection 6.3.2.

Overflow provisions prevent overfilling of the tank. The overflow is directed into the refueling cavity. The in-containment refueling water storage tank does not have a cover gas and does not require a connection to the waste gas processing system. The normal residual heat removal system provides nonsafety-related cooling of the in-containment refueling water storage tank.

5.4.14 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal heat exchanger (PRHR HX) is the component of the passive core cooling system that removes core decay heat for any postulated non-loss of coolant accident event where a loss of cooling capability via the steam generators occurs. Section 6.3 discusses the operation of the passive residual heat removal heat exchanger in the passive core cooling system.

5.4.14.1 Design Bases

The passive residual heat removal heat exchanger~~s~~ automatically ~~removes~~ actuates to remove core decay heat for an ~~unlimited~~ extended period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15.~~keep the reactor coolant subcooled and prevent water relief from the pressurizer.The passive residual heat removal heat exchanger in conjunction with the passive containment cooling system can remove heat for an indefinite time in a closed-loop (that is, no pipe break) mode of operation.~~ In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system ~~can be depressurized~~pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

The passive residual heat removal heat exchanger is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Those portions of the passive residual heat exchanger that support the primary-side pressure boundary and falls under the jurisdiction of ASME Code, Section III, Subsection NF are AP1000 equipment Class A (ANS Safety Class 1, Quality Group A). Stresses for ASME Code, Section III equipment and supports are

maintained within the limits of Section III of the Code. Section 5.2 provides ASME Code, Section III and material requirements. Subsection 5.2.4 discusses inservice inspection.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. Subsection 5.2.3 discusses the welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel

6.3 Passive Core Cooling System

The primary function of the passive core cooling system is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the passive core cooling system is designed to perform the following functions:

- Emergency core decay heat removal

  Provide core decay heat removal during transients, accidents or whenever the normal heat removal paths are lost. This heat removal function is available at reactor coolant system conditions including shutdowns. During refueling operations, when the IRWST is drained into the refueling cavity, other passive means of core decay heat removal are utilized. Subsection 6.3.3.4.4 provides a description of how this is accomplished.

- Reactor coolant system emergency makeup and boration

  Provide reactor coolant system makeup and boration during transients or accidents when the normal reactor coolant system makeup supply from the chemical and volume control system is unavailable or is insufficient.

- Safety injection

  Provide safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accidents, up to and including the double-ended rupture of the largest primary loop reactor coolant system piping.

- Containment pH control

  Provide for chemical addition to the containment during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment and to prevent corrosion of containment equipment during long-term floodup conditions.

The passive core cooling system is designed to operate without the use of active equipment such as pumps and ac power sources. The passive core cooling system depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The passive core cooling system does require a one-time alignment of valves upon actuation of the specific components.

6.3.1 Design Basis

The passive core cooling system is designed to perform its safety-related functions based on the following considerations:

- It has component redundancy to provide confidence that its safety-related functions are performed, even in the unlikely event of the most limiting single failure occurring coincident with postulated design basis events.

- Components are designed and fabricated according to industry standard quality groups commensurate with its intended safety-related functions.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- It is tested and inspected at appropriate intervals, as defined by the ASME Code, Section XI, and by technical specifications.

- It performs its intended safety-related functions following events such as fire, internal missiles or pipe breaks.

- It is protected from the effects of external events such as earthquakes, tornadoes, and floods.

- It is designed to be sufficiently reliable, considering redundancy and diversity, to support the plant core melt frequency and significant release frequency goals.

6.3.1.1 Safety Design Basis

The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory. Subsection 6.3.3 provides a description of the design basis events. The performance criteria are provided in subsection 6.3.1 and also described in Chapter 15, under the respective event sections.

6.3.1.1.1 Emergency Core Decay Heat Removal

For postulated non-LOCA events, where a loss of capability to remove core decay heat via the steam generators occurs, the passive core cooling system is designed to perform the following functions:

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling~~ and to prevent water relief through the pressurizer safety valves~~.

- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).

- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in Subsection 6.3.1.1.4.

- The passive residual heat removal heat exchanger is capable of ~~automatically removing core decay heat following such an event~~performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- ~~The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed loop mode of operation. The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.~~

- During a steam generator tube rupture event, the passive residual heat removal heat exchanger removes core decay heat and reduces reactor coolant system temperature and pressure, equalizing with steam generator pressure and terminating break flow, without overfilling the steam generator.

6.3.1.1.2 Reactor Coolant System Emergency Makeup and Boration

For postulated non-LOCA events, sufficient core makeup water inventory is automatically provided to keep the core covered and to allow for decay heat removal. In addition, this makeup prevents actuation of the automatic depressurization system for a significant time.

For postulated events resulting in an inadvertent cooldown of the reactor coolant system, such as a steam line break, sufficient borated water is automatically provided to makeup for reactor coolant system shrinkage. The borated water also counteracts the reactivity increase caused by the resulting system cooldown.

For a Condition II steam line break described in Chapter 15, return to power is acceptable if there is no core damage. For this event, the automatic depressurization system is not actuated.

For a large steam line break, the peak return to power is limited so that the offsite dose limits are satisfied. Following either of these events, the reactor is automatically brought to a subcritical condition.

For safe shutdown, the passive core cooling system is designed to supply sufficient boron to the reactor coolant system to maintain the technical specification shutdown margin for cold, post-depressurization conditions, with the most reactive rod fully withdrawn from the core. The automatic depressurization system is not expected to actuate for these events.

6.3.1.1.3 Safety Injection

The passive core cooling system provides sufficient water to the reactor coolant system to mitigate the effects of a loss of coolant accident. In the event of a large loss of coolant accident, up to and including the rupture of a hot or cold leg pipe, where essentially all of the reactor coolant volume is initially displaced, the passive core cooling system rapidly refills the reactor vessel, refloods the core, and continuously removes the core decay heat. A large break is a rupture with a total cross-sectional area equal to or greater than one square foot. Although the criteria for mechanistic pipe break are used to limit the size of pipe rupture considered in the design and evaluation of piping systems, as described in subsection 3.6.3, such criteria are not used in the design of the passive core cooling system.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Sufficient water is provided to the reactor vessel following a postulated loss of coolant accident so that the performance criteria for emergency core cooling systems, described in Chapter 15, are satisfied.

The automatic depressurization system valves, provided as part of the reactor coolant system, are designed so that together with the passive core cooling system they:

- Satisfy the small loss of coolant accident performance requirements
- Provide effective core cooling for loss of coolant accidents from when the passive core cooling system is actuated through the long-term cooling mode.

6.3.1.1.4 Safe Shutdown

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, ~~For these events,~~ the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions~~, cooling the reactor coolant system to about 420°F in 36 hours, with or without the reactor coolant pumps operating~~ in the reactor coolant system.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level~~as the temperature decreases and pressurizer level decreases, actuating the core makeup tanks~~. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available. ~~However in scenarios when ac power sources are unavailable for as long as 24 hours, the automatic depressurization system will automatically actuate.~~

In most sequences the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where ac power sources are lost but the passive residual heat removal heat exchanger operation is not extended or exhausted~~, or when the core makeup tank levels reach the automatic depressurization system actuation setpoint,~~ the automatic depressurization system will be initiated~~ initiates~~. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

The basis used to define the passive core cooling system functional requirements ~~are~~ is derived from Section 7.4 of the Standard Review Plan. The functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in subsection 5.4.7 and Section 7.4. The performance of the passive residual heat removal heat exchanger to bring the plant to 420^{0}F in 36 hours is summarized in Subsection 19E.4.10.2.

6.3.1.1.5 Containment pH Control

The passive core cooling system is capable of maintaining the desired post-accident pH conditions in the recirculation water after containment floodup. The pH adjustment is capable of maintaining containment pH within a range of 7.0 to 9.5, to enhance radionuclide retention in the containment and to prevent stress corrosion cracking of containment components during long-term containment floodup.

6.3.1.1.6 Reliability Requirements

The passive core cooling system satisfies a variety of reliability requirements, including redundancy (such as for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the system provides protection in a number of areas including:

- Single active and passive component failures
- Spurious failures
- Physical damage from fires, flooding, missiles, pipe whip, and accident loads
- Environmental conditions such as high-temperature steam and containment floodup

Subsection 6.3.1.~~2~~ 3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Long-Term Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in- containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for 14 days in a closed-loop mode of operation.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in Subsection 7.4.1.1.

6.3.1.~~2~~3 Power Generation Design Basis

The passive core cooling system is designed to be sufficiently reliable to support the probabilistic risk analysis goals for core damage frequency and severe release frequency. In assessing the reliability for probabilistic risk analysis purposes, more realistic analysis is used for both the passive core cooling system performance and for plant response.

In the event of a small loss of coolant accident, the passive core cooling system limits the increase in peak clad temperature and core uncovery with design basis assumptions. For pipe ruptures of less than eight-inch nominal diameter size, the passive core cooling system is designed to prevent core uncovery with best estimate assumptions.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour.

The frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outages. Equipment is located so that it is not flooded or it is designed so that it is not damaged by the flooding. Major plant equipment is designed for multiple occurrences without damage.

The pH control equipment is designed to minimize the potential for and the impact of inadvertent actuation.

The passive core cooling system is capable of supporting the required testing and maintenance, including capabilities to isolate and drain equipment.

6.3.2 System Design

The passive core cooling system is a seismic Category I, safety-related system. It consists of two core makeup tanks, two accumulators, the in-containment refueling water storage tank, the passive residual heat removal heat exchanger, pH adjustment baskets, and associated piping, valves, instrumentation, and other related equipment. The automatic depressurization system valves and spargers, which are part of the reactor coolant system, also provide important passive core cooling functions.

The passive core cooling system is designed to provide adequate core cooling in the event of design basis events. The redundant onsite safety-related class 1E dc and UPS system provides power such that protection is provided for a loss of ac power sources, coincident with an event, assuming a single failure has occurred.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Figure~~s~~ 6.3-1 ~~and 6.3-2~~ shows the piping and instrumentation drawings of the passive core cooling system. Simplified flow diagrams are shown in Figures 6.3-3 and 6.3-4. The accident analysis results of events analyzed in Chapter 15 provide a summary of the expected fluid conditions in the passive core cooling system for the various locations shown on the simplified flow diagrams, for the specific plant conditions identified -- safety injection and decay heat removal.

The passive core cooling system is designed to supply the core cooling flow rates to the reactor coolant system specified in Chapter 15 for the accident analyses. The accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values.

The passive core cooling system design is based on the six major components, listed in subsection 6.3.2.2, that function together in various combinations to support the four passive core cooling system functions:

- Emergency decay heat removal
- Emergency reactor makeup/boration
- Safety injection
- Containment pH control

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

For events not involving a loss of coolant, the emergency core decay heat removal is provided by the passive core cooling system via the passive residual heat removal heat exchanger. The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel head arrangement at the top (inlet) and bottom (outlet). The passive residual heat removal heat exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg (through a tee from one of the fourth stage automatic depressurization lines) and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).

The inlet line is normally open and connects to the upper passive residual heat removal heat exchanger channel head. The inlet line is connected to the top of the hot leg and is routed continuously upward to the high point near the heat exchanger inlet. The normal water temperature in the inlet line will be hotter than the discharge line.

The outlet line contains normally closed air-operated valves that open on loss of air pressure or on control signal actuation. The alignment of the passive residual heat removal heat exchanger (with a normally open inlet motor-operated valve and normally closed outlet air-operated valves) maintains the heat exchanger full of reactor coolant at reactor coolant system pressure. The water temperature in the heat exchanger is about the same as the water in the in-containment refueling water storage tank, so that a thermal driving head is established and maintained during plant operation.

The heat exchanger is elevated above the reactor coolant system loops to induce natural circulation flow through the heat exchanger when the reactor coolant pumps are not available. The passive residual heat removal heat exchanger piping arrangement also allows actuation of the heat exchanger with reactor coolant pumps operating. When the reactor coolant pumps

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

are operating, they provide forced flow in the same direction as natural circulation flow through the heat exchanger. If the pumps are operating and subsequently trip, then natural circulation continues to provide the driving head for heat exchanger flow.

The heat exchanger is located in the in-containment refueling water storage tank, which provides the heat sink for the heat exchanger.

Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, can provide core cooling for at least 72 hours~~in conjunction with the passive containment cooling system, can provide core cooling for an indefinite period of time~~. After the in-containment refueling water storage tank water reaches its saturation temperature (in ~~about 2~~several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of t~~T~~he condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located ~~at~~ near the operating deck ~~level~~ elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it~~which returns the condensate~~ to the in-containment refueling water storage tank. The gutter and downspouts normally drain~~s~~ to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an ~~indefinite~~ extended period of time.

The passive residual heat removal heat exchanger is used to maintain ~~a safe shutdown~~an acceptable, stable reactor coolant system condition. It ~~removes~~ transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power

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sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

The core makeup tanks provide reactor coolant system makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two core makeup tanks located inside the containment at an elevation slightly above the reactor coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves discussed above. They use different globe valve body styles and different air operator types.

The pressure balance line from the cold leg is normally open to maintain the core makeup tanks at reactor coolant system pressure, which prevents water hammer upon initiation of core makeup tank injection.

The cold leg pressure balance line is connected to the top of the cold leg and is routed continuously upward to the high point near the core makeup tank inlet. The normal water temperature in this line will be hotter than the discharge line.

The outlet line from the bottom of each core makeup tank provides an injection path to one of the two direct vessel injection lines, which are connected to the reactor vessel downcomer annulus. Upon receipt of a safeguards actuation signal, the two parallel valves in each discharge line open to align the associated core makeup tank to the reactor coolant system.

There are two operating processes for the core makeup tanks, steam-compensated injection and water recirculation. During steam-compensated injection, steam is supplied to the core makeup tanks to displace the water that is injected into the reactor coolant system. This steam is provided to the core makeup tanks through the cold leg pressure balance line. The cold leg line only has steam flow if the cold legs are voided.

During water recirculation, hot water from the cold leg enters the core makeup tanks, and the cold water in the tank is discharged to the reactor coolant system. This results in reactor coolant system boration and a net increase in reactor coolant system mass.

The operating process for the core makeup tanks depends on conditions in the reactor coolant system, primarily voiding in the cold leg. When the cold leg is full of water, the cold leg pressure balance line remains full of water and the injection occurs via water recirculation. If reactor coolant system inventory decreases sufficiently to cause cold leg voiding, then steam flows through the cold leg balance lines to the core makeup tanks.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Following an event such as steam-line break, the reactor coolant system experiences a decrease in temperature and pressure due to an increase of energy removed by the secondary system as a consequence of the break. The cooldown results in a reduction of the core shutdown margin due to the negative moderator temperature coefficient. There is a potential return to power, assuming the most reactive rod cluster control assembly is stuck in its fully withdrawn position. The actuation of the core makeup tanks following this event provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.

In case of a steam generator tube rupture, core makeup tank injection together with the steam generator overfill prevention logic terminates the reactor coolant system leak into the steam generator. This occurs without actuation of the automatic depressurization system and without operator action. In a steam generator tube rupture, the core makeup tanks operate in the water recirculation mode to provide borated water to compensate for reactor coolant system inventory losses and to borate the reactor coolant system. In case of a leak rate of 10 gallons per minute, the passive core cooling system can delay the automatic depressurization system actuation for at least 10 hours while providing makeup water to the reactor coolant system. After the actuation of the automatic depressurization system, the passive core cooling system provides sufficient borated water to compensate for reactor coolant system shrinkage and to provide the reactor coolant system boration.

6.3.2.1.3 Safety Injection During Loss of Coolant Accidents

The passive core cooling system uses four different sources of passive injection during loss of coolant accidents.

- Accumulators provide a very high flow for a limited duration of several minutes.
- The core makeup tanks provide a relatively high flow for a longer duration.
- The in-containment refueling water storage tank provides a lower flow, but for a much longer time.
- The containment is the final long-term source of water. It becomes available following the injection of the other three sources and floodup of containment.

The operation of the core makeup tanks is described in the subsection 6.3.2.1.2. During a loss of coolant accident, they provide injection rates commensurate with the severity of the loss of coolant accident. For a larger loss of coolant accident, and after the automatic depressurization system has been actuated, the cold legs are expected to be voided. In this situation, the core makeup tanks operate at their maximum injection rate with steam entering the core makeup tanks through the cold leg pressure balance lines.

Downstream of the parallel discharge isolation valves, the core makeup tank discharge line contains two check valves, in series, that normally remain open with or without flow in the line. These valves prevent reverse flow through this line, from the accumulator, that would bypass the reactor vessel in the event of a larger loss of coolant accident in the cold leg or the cold leg pressure balance line.

For smaller loss of coolant accidents the core makeup tanks initially operate in the water recirculation mode since the cold legs are water filled. During this water recirculation, the

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

core makeup tanks remain full, but the cold, borated water is purged with hot, less borated cold leg water. The water recirculation provides reactor coolant system makeup and also effectively borates the reactor coolant system. As the accident progresses, when the cold legs void, the core makeup tanks switch to the steam displacement mode which provides higher flow rates.

The two accumulators contain borated water and a compressed nitrogen cover gas to provide rapid injection. They are located inside the reactor containment and the discharge from each tank is connected to one of the direct vessel injection lines. These lines connect to the reactor vessel downcomer. A deflector in the annulus directs the water flow downward to minimize core bypass flow. The water and gas volumes and the discharge line resistance provide several minutes of injection in a large loss of coolant accident.

The in-containment refueling water storage tank is located in the containment at an elevation slightly above the reactor coolant system loop piping. Reactor coolant system injection is possible only after the reactor coolant system has been depressurized by the automatic depressurization system or by a loss of coolant accident. Squib valves in the in-containment refueling water storage tank injection lines open automatically on a 4th stage automatic depressurization signal. Check valves, arranged in series with the squib valves, open when the reactor pressure decreases to below the in-containment refueling water storage tank injection head.

After the accumulators, core makeup tanks, and the in-containment refueling water storage tank inject, the containment is flooded up to a level sufficient to provide recirculation flow through the gravity injection lines back into the reactor coolant system.

The time that it takes until the initiation of containment recirculation flow varies greatly, depending on the specific event. With a break in a direct vessel injection line, the in-containment refueling water storage tank spills out through the break and floods the containment, along with reactor coolant system leakage, and recirculation can occur in several hours. In the event of automatic depressurization without a reactor coolant system break and with condensate return, the in-containment refueling water storage tank level decreases very slowly. Recirculation may not initiate for several days.

Containment recirculation initiates when the recirculation line valves are open and the containment floodup level is sufficiently high. When the in-containment refueling water storage tank level decreases to a low level, the containment recirculation squib valves automatically open to provide redundant flow paths from the containment to the reactor.

These recirculation flow paths can also provide a suction flow path from the containment to the normal residual heat removal pumps, when they are operating after containment flood up. In addition, the squib valves in the recirculation paths containing normally open motor-operated valves can be manually opened to intentionally drain the in-containment refueling water storage tank to the reactor cavity during severe accidents. This action is modeled in the AP1000 probabilistic risk assessment.

A range of break sizes and locations are analyzed to verify the adequacy of passive core cooling system injection. These events include a no-break case, a complete severance of one (eight-inch) direct vessel injection line case, and other smaller break cases. Successful reactor coolant system depressurization to in-containment refueling water storage tank injection is achieved, as shown in Chapter 15.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

In larger loss of coolant accidents, including double ended ruptures in reactor coolant system piping, the passive core cooling system can provide a large flow rate, from the accumulators, to quickly refill the reactor vessel lower plenum and downcomer. The accumulators provide the required injection flow during the first part of the event including refilling the downcomer and lower plenum and partially reflooding the core. After the accumulators empty, the core makeup tanks complete the reflooding of the core. The subsequent in-containment refueling water storage tank injection and recirculation provide long-term cooling. Both injection lines are available since the injection lines are not the source of a large pipe break.

6.3.2.1.4 Containment pH Control

Control of the pH in the containment sump water post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP). The baskets are located below the minimum post-accident floodup level, and chemical addition is initiated passively when the water reaches the baskets. The baskets are placed at least a foot above the floor to reduce the chance that water spills in containment will dissolve the TSP.

The TSP is designed to maintain the pH of the containment sump water in a range from 7.0 to 9.5. This chemistry reduces radiolytic formation of elemental iodine in the containment sump, consequently reducing the aqueous production of organic iodine, and ultimately reducing the airborne iodine in containment and offsite doses.

The chemical addition also helps to reduce the potential for stress corrosion cracking of stainless steel components in a post floodup condition, where chlorides can leach out of the containment concrete and potentially affect these components during a long-term floodup event.

6.3.2.1.5 Passive Core Cooling System Actuation

Table 6.3-1 lists the remotely actuated valves used by the various passive core cooling system components. The engineered safeguards features actuation signals used for these valves are described in Section 7.3. Table 6.3-1 shows the normal valve position, the valve position to actuate the associated component, and the failure position of the valve. The failed position represents the position that the valve fails upon loss of electrical power or other motive sources, such as instrument air.

Table 6.3-3 contains the failure mode and effects analysis of the passive core cooling system.

6.3.2.2 Equipment and Component Descriptions

Table 6.3-2 contains a summary of equipment parameters for major components of the passive core cooling system.

6.3.2.2.1 Core Makeup Tanks

The two core makeup tanks are vertical, cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel, clad on the internal surfaces with stainless steel. The core makeup tanks are AP1000 Equipment Class A and are designed to meet seismic Category I requirements. They are located inside containment on the 107-foot floor elevation. The core makeup tanks are located above the direct vessel injection line connections to the reactor vessel, which are located at an elevation near the bottom of the hot leg.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

During normal operation the core makeup tanks are completely filled with borated water and are maintained at reactor coolant system pressure by the cold leg pressure balance line. The temperature of the borated water in the core makeup tanks is about the same as the containment ambient temperature since the tanks are not insulated or heated.

The inlet line from the cold leg is sized for loss of coolant accidents, where the cold legs become voided and higher core makeup tank injection flows are required. The discharge line from each core makeup tank contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates assumed in the core makeup tank design. The core makeup tanks provide injection for an extended time after core makeup tank actuation. The duration of injection will be much longer when the core makeup tanks operate in the water recirculation mode as compared to the steam condensation mode.

Connections are provided for remotely adjusting the boron concentration of the borated water in each core makeup tank during normal plant operation, as required. Makeup water for the core makeup tank is provided by the chemical and volume control system. Samples from the core makeup tanks are taken periodically to check boron concentration.

Each core makeup tank has an inlet diffuser which is designed to reduce steam velocities entering the core makeup tank; thereby minimizing potential water hammer and reducing the amount of mixing that occurs during initial core makeup tank operation. The inlet diffuser flow area is $\geq 165\ in^2$.

The core makeup tanks are located inside the containment but outside the secondary shield wall. This facilitates maintenance and inspection.

Core makeup tank level and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action as required to meet the technical specification requirements for core makeup tank operability.

6.3.2.2.2 Accumulators

The two accumulators are spherical tanks made of carbon steel and clad on the internal surfaces with stainless steel. The accumulators are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. They are located inside the containment on the floor just below the core makeup tanks.

The accumulators are mostly filled with borated water and pressurized with nitrogen gas. The temperature of the borated water in the accumulators is about the same as the containment ambient temperature since the tanks are not insulated or heated. Each accumulator is connected to one of the direct vessel injection lines. During normal operation, the accumulator is isolated from the reactor coolant system by two check valves in series. When the reactor coolant system pressure falls below the accumulator pressure, the check valves open and borated water is forced into the reactor coolant system by the gas pressure. Mechanical operation of the check valves is the only action required to open the injection path from the accumulators to the core.

The accumulators are designed to deliver a high flow of borated water to the reactor vessel in the event of a large loss of coolant accident. This large flow rate is used to quickly establish core cooling following the large loss of reactor coolant system inventory.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The injection line from each accumulator contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates assumed in the accumulator design. The accumulator provides injection for several minutes after reactor coolant system pressure drops below the static accumulator pressure.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal plant operation, as required. Accumulator water level may be adjusted either by draining or by pumping borated water from the chemical and volume control system to the accumulator. Samples from the accumulators are taken periodically to check the boron concentration.

Accumulator pressure is provided by a supply of nitrogen gas and can be adjusted as required during normal plant operation. However, the accumulators are normally isolated from the nitrogen supply. Gas relief valves on the accumulators protect them from overpressurization. The system also includes the capability to remotely vent gas from the accumulator, if required.

The accumulators are located inside the containment and outside the secondary shield wall. This facilitates maintenance and inspection.

Accumulator level and pressure are monitored by indication and alarms. The operator can take action, as required, to meet the technical specification requirements for accumulator operability.

6.3.2.2.3 In-Containment Refueling Water Storage Tank

The in-containment refueling water storage tank is a large, stainless-steel lined tank located underneath the operating deck inside the containment. The in-containment refueling water storage tank is AP1000 Equipment Class C and is designed to meet seismic Category I requirements. The tank is constructed as an integral part of the containment internal structures, and is isolated from the steel containment vessel. See subsection 3.8.3 for additional information.

The bottom of the in-containment refueling water storage tank is above the reactor coolant system loop elevation so that the borated refueling water can drain by gravity into the reactor coolant system after it is sufficiently depressurized. The in-containment refueling water storage tank is connected to the reactor coolant system through both direct vessel injection lines. The in-containment refueling water storage tank contains borated water, at the existing temperature and pressure in containment.

Vents are installed in the roof of the in-containment refueling water storage tank. These vents are normally closed in order to contain water vapor and radioactive gases within the tank during normal operation and to prevent debris from entering the tank from the containment operating deck. The vents open with a slight pressurization of the in-containment refueling water storage tank. These vents provide a path to vent steam released by the spargers or generated by the passive residual heat removal heat exchanger, into the containment atmosphere. Other vents also open on small pressure differentials to allow air/steam to enter the in-containment refueling water storage tank from containment, such as during a loss of coolant accident, to prevent damage to the tank. Overflows are provided from the in-containment refueling water storage tank to the refueling cavity to accommodate volume and

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

mass increases during passive residual heat removal heat exchanger or automatic depressurization system operation, while minimizing the floodup of the containment.

The IRWST is stainless steel lined and does not contain material either in the tank or the recirculation path that could plug the outlet screens.

The in-containment refueling water storage tank contains one passive residual heat removal heat exchanger and two depressurization spargers. The top of the passive residual heat removal heat exchanger tubes are located underwater and extend down into the in-containment refueling water storage tank. The spargers are also submerged in the in-containment refueling water storage tank, with the spargers midarms located below the normal water level.

The in-containment refueling water storage tank is sized to provide the flooding of the refueling cavity for normal refueling, the post-loss of coolant accident flooding of the containment for reactor coolant system long-term cooling mode, and to support the passive residual heat removal heat exchanger operation. Flow out of the in-containment refueling water storage tank during the injection mode includes conservative allowances for spill flow during a direct vessel injection line break.

The in-containment refueling water storage tank can provide sufficient injection until the containment sump floods up high enough to initiate recirculation flow. The injection duration varies greatly, depending upon the specific event. A direct vessel injection line break more rapidly drains the in-containment refueling water storage tank and speeds containment floodup.

The containment floodup volume for a LOCA in PXS room B is less than 73,500 ft^3 (excluding the in-containment refueling water storage tank) below a containment elevation of 108 feet.

Connections to the in-containment refueling water storage tank provide for transfer to and from the reactor coolant system/refueling cavity via the normal residual heat removal system, purification and sampling via the spent fuel pit cooling system, and remotely adjusting boron concentration to the chemical and volume control system. Also, the normal residual heat removal system can provide cooling of the in-containment refueling water storage.

In-containment refueling water storage tank level and temperature are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements for in-containment refueling water storage tank operability.

6.3.2.2.4 pH Adjustment Baskets

The passive core cooling system utilizes pH adjustment baskets for control of the pH level in the containment sump. The baskets are made of stainless steel with a mesh front that readily permits contact with water. The baskets are designated AP1000 Equipment Class C, and are designed to meet seismic Category I requirements.

The total weight of TSP contained in the baskets is at least 26,460 pounds. The TSP, in granular form, is provided to raise the pH of the borated water in the containment following an accident to at least 7.0. After extended plant operation, the granular TSP may cake into a solid form as it absorbs moisture. Assuming that the TSP has caked, the dissolution time of

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

the TSP is approximately 3 hours. Good mixing with the sump water is expected due to both basket construction and because the baskets are placed in locations conducive to recirculation flows post-accident. The baskets are designed for ease of replacement of the TSP.

6.3.2.2.5 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal exchanger consists of inlet and outlet channel heads connected together by vertical C-shaped tubes. The tubes are supported inside the in-containment refueling water storage tank. The top of the tubes is several feet below the in-containment refueling water storage tank water surface. The component data for the passive residual heat removal heat exchanger is shown in Table 6.3-2. The passive residual heat removal heat exchanger is AP1000 Equipment Class A and is designed to meet seismic Category I requirements.

The heat exchanger inlet piping connects to an inlet channel head located near the outside top of the tank. The inlet channel head and tubesheet are attached to the tank wall via an extension flange. The heat exchanger is supported by a frame which is attached to the IRWST floor and ceiling. The heat exchanger supports are designed to ASME Code, Section III, subsection NF. The extended flange is designed to accommodate thermal expansion. Figure 6.3-5 illustrates the relationship between these parts and the boundaries of design code jurisdiction. The heat exchanger outlet piping is connected to the outlet channel head, which is vertically below the inlet channel head, near the tank bottom. The outlet channel head has an identical structural configuration to the inlet channel head. Both channel head tubesheets are similar to the steam generator tubesheets and they have manways for inspection and maintenance access.

The passive residual heat removal heat exchanger is designed to remove sufficient heat so that its operation, in conjunction with available inventory in the steam generators, provide reactor coolant system cooling and prevents water relief through the pressurizer safety valves during loss of main feedwater or main feedline break events.

Passive residual heat removal heat exchanger flow and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements or follow emergency operating procedures for control of the passive residual heat removal heat exchanger operation.

6.3.2.2.6 Depressurization Spargers

Two reactor coolant depressurization spargers are provided. Each one is connected to an automatic depressurization system discharge header (shared by three automatic depressurization system stages) and submerged in the in-containment refueling water storage tank. Each sparger has four branch arms inclined downward. The connection of the sparger branch arms to the sparger hub are submerged below the in-containment refueling water storage tank overflow level by ≤11.5 feet. The component data for the spargers is shown in Table 6.3-2. The spargers are AP1000 Equipment Class C and are designed to meet seismic Category I requirements.

The spargers perform a nonsafety-related function -- minimizing plant cleanup and recovery actions following automatic depressurization. They are designed to distribute steam into the in-containment refueling water storage tank, thereby promoting more effective steam condensation.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The first three stages of automatic depressurization system valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads on the in-containment refueling water storage tank and other structures.

Each sparger is sized to discharge at a flow rate that supports automatic depressurization system performance, which in turn, allows adequate passive core cooling system injection.

6.3.2.2.7 IRWST and Containment Recirculation Screens

The passive core cooling system~~s~~ has two different sets of screens that are used ~~following~~ to prevent debris from entering the reactor and blocking core cooling passages during a LOCA; IRWST screens and containment recirculation screens. ~~These screens prevent debris from entering the reactor and blocking core cooling passages during a LOCA.~~ The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames, attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The~~se~~ IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:

- GDC 35 of 10 CFR 50 Appendix A
- Regulatory Guide 1.82
- NUREG-0897

The operation of the passive core cooling system following a LOCA is described in subsection 6.3.2.1.3. Proper screen design, plant layout, and other factors prevent clogging of these screens by debris during accident operations.

6.3.2.2.7.1 General Screen Design Criteria

The IRWST screens and containment recirculation screens are designed to comply with the following criteria.

1. Screens are designed to Regulatory Guide 1.82, including:

    - Separate, large screens are provided for each function.

    - Screens are located well below containment floodup level. Each screen provides the function of a trash rack and a fine screen. A debris curb is provided to prevent high density debris from being swept along the floor to the screen face.

    - Floors slope away from screens (not required for AP1000).

    - Drains do not impinge on screens.

    - Screens can withstand accident loads and credible missiles.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- Screens have conservative flow areas to account for plugging. Operation of the non-safety-related normal residual heat removal pumps with suction from the IRWST and the containment recirculation lines is considered in sizing screens.

- System and screen performance are evaluated.

- Screens have solid top cover. Containment recirculation screens have protective plates that are located no more than 1 foot above the top of the screens and extend at least 10 feet in front and 7 feet to the side of the screens. The plate dimensions are relative to the portion of the screens where water flow enters the screen openings. Coating debris, from coatings located outside of the ZOI, is not transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation considering the use of high density coatings discussed in subsection 6.1.2.1.5.

- Screens are seismically qualified.

- Screen openings are sized to prevent blockage of core cooling.

- Screens are designed for adequate pump performance. AP1000 has no safety-related pumps.

- Corrosion resistant materials are used for screens.

- Access openings in screens are provided for screen inspection.

- Screens are inspected each refueling.

2. Low screen approach velocities limit the transport of heavy debris even with operation of normal residual heat removal pumps.
3. [*Metal reflective insulation is used on ASME class 1 lines because they are subject to loss-of-coolant accidents. Metal reflective insulation is also used on the reactor vessel, the reactor coolant pumps, the steam generators, and on the pressurizer because they have relatively large insulation surface areas and they are located close to large ASME class 1 lines. As a result, they are subject to jet impingement during loss-of-coolant accidents.*]* A suitable equivalent insulation to metal reflective may be used. A suitable equivalent insulation is one that is encapsulated in stainless steel that is seam welded so that LOCA jet impingement does not damage the insulation and generate debris. Another suitable insulation is one that may be damaged by LOCA jet impingement as long as the resulting insulation debris is not transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

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[*In order to provide additional margin, metal reflective insulation is used inside containment where it would be subject to jet impingement during loss-of-coolant accidents that are not otherwise shielded from the blowdown jet.*]* As a result, fibrous debris is not generated by loss-of-coolant accidents. Insulation located within the zone of influence (ZOI), which is a spherical region within a distance equal to 29 inside diameters (for Min-K, Koolphen-K, or rigid cellular glass insulation) or 20 inside diameters (for other types of insulation) of the LOCA pipe break is assumed to be affected by the LOCA when there are intervening components, supports, structures, or other objects.

[*The ZOI in the absence of intervening components, supports, structures, or other objects includes insulation in a cylindrical area extending out a distance equal to 45 inside diameters from the break along an axis that is a continuation of the pipe axis and up to 5 inside diameters in the radial direction from the axis.*]* A suitable equivalent insulation to metal reflective may be used as discussed in the previous paragraph.

[*Insulation used inside the containment, outside the ZOI, but below the maximum post-DBA LOCA floodup water level (plant elevation 110.2 feet), is metal reflective insulation, jacketed fiberglass, or a suitable equivalent.*]* A suitable equivalent insulation is one that would be restrained so that it would not be transported by the flow velocities present during recirculation and would not add to the chemical precipitates. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

[*Insulation used inside the containment, outside the ZOI, but above the maximum post-design basis accident (DBA) LOCA floodup water level, is jacketed fiberglass, rigid cellular glass, or a suitable equivalent.*]* A suitable equivalent insulation is one that when subjected to dripping of water from the containment dome would not add to the chemical precipitates; suitable equivalents include metal reflective insulation.

4. Coatings are not used on surfaces located close to the containment recirculation screens. The surfaces considered close to the screens are defined in subsection 6.3.2.2.7.3. Refer to subsection 6.1.2.1.6. These surfaces are constructed of materials that do not require coatings.

5. The IRWST is enclosed which limits debris egress to the IRWST screens.

6. Containment recirculation screens are located above lowest levels of containment.

7. Long settling times are provided before initiation of containment recirculation.

8. Air ingestion by safety-related pumps is not an issue in the AP1000 because there are no safety-related pumps. The normal residual heat removal system pumps are evaluated to show that they can operate with minimum water levels in the IRWST and in the containment.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

9. A commitment for cleanliness program to limit debris in containment is provided in subsection 6.3.8.1.

10. [*Other potential sources of fibrous material, such as ventilation filters or fiber-producing fire barriers, are not located in jet impingement damage zones or below the maximum post-DBA LOCA floodup water level.*]*

11. Other potential sources of transportable material, such as caulking, signs, and equipment tags installed inside the containment are located:

    - Below the maximum flood level, or
    - Above the maximum flood level and not inside a cabinet or enclosure.

    Tags and signs in these locations are made of stainless steel or another metal that has a density ≥ 100 lbm/ft^3. Caulking in these locations is a high density (≥ 100 lbm/ft^3).

    The use of high-density metal prevents the production of debris that could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break location that is submerged during recirculation. If a high-density material is not used for these components, then the components must be located inside a cabinet or other enclosure, or otherwise shown not to transport; the enclosures do not have to be watertight, but need to prevent water dripping on them from creating a flow path that would transport the debris outside the enclosure. For light-weight (< 100 lb_m/ft^3) caulking, signs or tags that are located outside enclosures, testing must be performed that subjects the caulking, signs, or tags to conditions that bound the AP1000 conditions and demonstrates that debris would not be transported to an AP1000 screen or into the core through a flooded break. Note that in determining if there is sufficient water flow to transport these materials, consideration needs to be given as to whether they are within the ZOI (for the material used) because that determines whether they are in their original geometry or have been reduced to smaller pieces. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing must be approved by the NRC.

12. An evaluation consistent with Regulatory Guide 1.82, Revision 3, and subsequently approved NRC guidance, has been performed (Reference 3) to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation considered resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris was based on sample measurements from operating plants. The evaluation also considered the potential for the generation of chemical debris (precipitants). The potential to generate such debris was determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The evaluation considered the following conservative considerations:

- [*The COL cleanliness program will limit the total amount of resident debris inside the containment to ≤ 130 pounds and the amount of the total that might be fiber to ≤ 6.6 pounds.*]*

- In addition to the resident debris, the LOCA blowdown jet may impinge on coatings and generate coating debris fines, which because of their small size, might not settle. The amount of coating debris fines that can be generated in the AP1000 by a LOCA jet will be limited to less than 70 pounds for double-ended cold leg and double-ended direct vessel injection LOCAs. In evaluating this limit, a ZOI of 4 IDs for epoxy and 10 IDs for inorganic zinc will be used. A DEHL LOCA could generate more coating debris; however, with the small amount of fiber available in the AP1000 following a LOCA, the additional coating debris fines that may be generated in a DEHL LOCA are not limiting.

- The total resident and ZOI coating debris available for transport following a LOCA is ≤ 193.4 pounds of particulate and ≤ 6.6 pounds of fiber. The percentage of this debris that could be transported to the screens or to the core is as follows:

  – Containment recirculation screens is ≤ 100 percent fiber and particles

  – IRWST screens is ≤ 50 percent fiber and 100 percent particles

  – Core (via a direct vessel injection or a cold leg LOCA break that becomes submerged) is ≤ 90 percent fiber and 100 percent particles

- Fibrous insulation debris is not generated and transported to the screens or into the core as discussed in item 3.

- Metal reflective insulation, including accident generated debris, is not transported to the screens or into the core.

- Coating debris is not transported to the screens or into the core as discussed in item 1.

- Debris from other sources, including caulking, signs, and tags, is not generated and transported to the screens or into the core as discussed in item 11.

- The total amount of chemical precipitates that could form in 30 days is ≤ 57 pounds.

- The percentage of the chemical precipitates that could be transported to the:

  – Containment recirculation screens is ≤ 100 percent.
  – IRWST screens is ≤ 100 percent.
  – Core is ≤ 100 percent.

- The range of flow rates during post-LOCA injection and recirculation is as follows:

  – CR screens: 2320 to 539 gpm

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- IRWST screens: 2320 to 464 gpm
- Core: 2012 to 484 gpm

These flows bound operation of the PXS and the RNS. Note that if the RNS operates during post-LOCA injection or recirculation, the RNS flow is limited to 2320 gpm. This limit ensures that the operation of the plant is consistent with screen head loss testing. In addition, the screens will be designed structurally to withstand much higher flow rates and pressure losses to provide appropriate margin during PXS and RNS operation.

No chemical precipitates are expected to enter the IRWST because the primary water input to the IRWST is steam condensed on the containment vessel. However, during a direct vessel injection LOCA, recirculation can transport chemical debris through the containment recirculation screens and to the IRWST screens. As a result, 100 percent of the chemical debris is conservatively assumed to be transported to the IRWST screens.

The AP1000 containment recirculation screens and IRWST screens have been shown to have acceptable head losses. The head losses for these screens were determined in testing performed using the above conservative considerations. It has been shown that a head loss of 0.25 psi at the maximum screen flows is acceptable based on long-term core cooling sensitivity analysis.

Considering downstream effects as well as potential bypass through a cold leg LOCA, the core was shown to have acceptable head losses. The head losses for the core were determined in testing performed using the above conservative considerations. It has been shown that a head loss of 4.1 psi at these flows is acceptable based on long-term core cooling sensitivity analysis.

6.3.2.2.7.2 IRWST Screens

The IRWST screens are located inside the IRWST at the bottom of the tank. Figure 6.3-6 shows a plan view and Figure 6.3-7 shows a section view of these screens. Three separate screens are provided in the IRWST, one at either end of the tank and one in the center. A cross-connect pipe connects all three IRWST screens to distribute flow. The IRWST is closed off from the containment; its vents and overflows are normally closed by louvers. The potential for introducing debris inadvertently during plant operations is limited. A cleanliness program (refer to subsection 6.3.8.1) controls foreign debris from being introduced into the tank during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The IRWST design eliminates sources of debris from inside the tank. Insulation is not used in the tank. Air filters are not used in the IRWST vents or overflows. Wetted surfaces in the IRWST are corrosion resistant such as stainless steel or nickel alloys; the use of these materials prevents the formation of significant amounts of corrosion products. In addition, the water is required to be clean because it is used to fill the refueling cavity for refueling; filtering and demineralizing by the spent fuel pit cooling system is provided during and after refueling.

During a LOCA, steam vented from the reactor coolant system condenses on the containment shell~~,~~ and drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell ~~to the operating deck elevation~~ and is collected in a gutter near the

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of ~~its~~ their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

The design of the IRWST screens reduces the chance of debris reaching the screens. The screens are oriented vertically such that debris that settles out of the water does not fall on the screens. The lowest screening surface of the IRWST screens is located 6 inches above the IRWST floor to prevent high density debris from being swept along the floor by water flow to the IRWST screens. The screen design provides the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inch from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has sufficient surface area to accommodate debris that could be trapped on the screen. The design of the IRWST screens is described further in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed considering the operation of the nonsafety-related normal residual heat removal system pumps which produce a higher flow than the safety-related gravity driven IRWST injection/recirculation flows. As a result, when the normal residual heat removal system pumps are not operating, there is a large margin to screen clogging.

6.3.2.2.7.3 Containment Recirculation Screens

The containment recirculation screens are oriented vertically along walls above the loop compartment floor (elevation 83 feet). Figure 6.3-8 shows a plan view and Figure 6.3-9 shows a section view of these screens. Two separate screens are provided as shown in Figure 6.3-3. The loop compartment floor elevation is significantly above (11.5 feet) the lowest level in the containment, the reactor vessel cavity. A two-foot-high debris curb is provided in front of the screens.

During a LOCA, the reactor coolant system blowdown will tend to carry debris created by the accident (pipe whip/jets) into the cavity under the reactor vessel which is located away from and below the containment recirculation screens. As the accumulators, core makeup tanks and IRWST inject, the containment water level will slowly rise above the 108 foot elevation. The containment recirculation line opens when the water level in the IRWST drops to a low level setpoint a few feet above the final containment floodup level. When the recirculation lines initially open, the water level in the IRWST is higher than the containment water level and water flows from the IRWST backwards through the containment recirculation screen. This back flow tends to flush debris located close to the recirculation screens away from the screens. A flow connection between Screen A and Screen B is provided so that both recirculation screens will operate. This connection increases the reliability of the PXS in a PRA sequence where there are multiple failures of valves in one of the PXS subsystems.

The water level in the containment when recirculation begins is well above (~ 10 feet) the top of the recirculation screens. During the long containment floodup time, floating debris does

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

not move toward the screens and heavy materials settle to the floors of the loop compartments or the reactor vessel cavity. During recirculation operation, the containment water level will not change significantly nor will it drop below the top of the screens.

The amount of debris that may exist following an accident is limited. Reflective insulation is used to preclude fibrous debris that can be generated by a loss of coolant accident and be postulated to reach the screens during recirculation. The nonsafety-related coatings used in the containment are designed to withstand the post accident environment. The containment recirculation screens are protected by plates located above them. These plates prevent debris from the failure of nonsafety-related coatings from getting into the water close to the screens such that the recirculation flow can cause the debris to be swept to the screens before it settles to the floor. Stainless steel is used on the underside of these plates and on surfaces located below the plates, above the bottom of the screens, 10 feet in front and 7 feet to the side of the screens to prevent coating debris from reaching the screens.

A cleanliness program (refer to subsection 6.3.8.1) controls foreign debris introduced into the containment during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The design of the containment recirculation screens reduces the chance of debris reaching the screens. The screens are orientated vertically such that debris settling out of the water will not fall on the screens. The protective plates described above provide additional protection to the screens from debris. A 2-foot-high debris curb is provided to prevent high density debris from being swept along the floor by water flow to the containment recirculation screens. The screen design provides the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inch from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has more surface area to accommodate debris that could be trapped on the screen. The design of the containment recirculation screens is further described in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed, considering the operation of the normal residual heat removal system pumps, which produce a higher flow than the gravity driven IRWST injection/recirculation flows. As a result, when the normal residual heat removal system pumps are not operating there is even more margin in screen clogging.

6.3.2.2.8 Valves

Design features used to minimize leakage for valves in the passive core cooling system include:

- Packless valves are used for manual isolation valves that are 2 inches or smaller.

- Valves which are normally open, except check valves and those which perform control function, are provided with back seats to limit stem leakage.

6.3.2.2.8.1 Manual Globe, Gate, and Check Valves

Gate valves have backseats and external screw and yoke assemblies.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Globe valves, both “T” and “Y” styles, are full-ported with external screw and yoke construction.

Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check valve hinge is serviced through the bonnet.

The gasket of the stainless steel manual globe and gate valves is similar to those described in subsection 6.3.2.2.8.3 for motor-operated valves.

6.3.2.2.8.2 Manual Valves

Manual valves are generally used as maintenance isolation valves. When used for this function they are under administrative control. They are located so that no single valve can isolate redundant passive core cooling system equipment or they are provided with alarms in the main control room to indicate mispositioning.

To help preclude the possibility of passive core cooling system degradation due to valve mispositioning, line connections such as vent and drain lines, test connections, pressure points, flow element test points, flush connections, local sample points, and bypass lines are provided
with double isolation or sealed barriers. The isolation is provided by one of the following methods:

- Two valves in series
- A single valve with a screwed cap or blind flange
- A single locked-closed valve
- A blind flange

6.3.2.2.8.3 Motor-Operated Valves

The motor operators for gate valves are conservatively sized, considering the frictional component of the hydraulic unbalance on the valve disc, the disc face friction, and the packing box friction. For motor-operated valves, the valve disc is guided throughout the full disc travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard-faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound asbestos (or a qualified asbestos substitute) gasket with provisions for seal welding or it is of the pressure seal design with provisions for seal welding.

The motor operator incorporates a hammer-blow feature that allows the motor to impact the disc away from the back seat upon closing. This hammer-blow feature impacts the discs and allows the motor to attain its operational speed prior to impact.

6.3.2.2.8.4 Motor-Operated Valve Controls

Remotely operated valves which do not receive a safeguards actuation signal, have their positions indicated on the main control board. When one of these valves is not in the ready position for injection during plant operation, this condition is indicated and alarmed in the main control room.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Spurious movement of a motor-operated valve due to an electrical fault in the motor actuation circuitry, coincident with loss of coolant accident, has been analyzed (Reference 1) and found to be an acceptably low probability event. In addition, power lockout in accordance with Branch Technical Position ICSB-18 is provided for those valves whose spurious movement could result in degraded passive core cooling system performance.

Table 6.3-1 provides a list of the remotely operated isolation valves in the passive core cooling system. These valves have various interlocks, automatic features, and position indication. Some valves have their control power locked out during normal plant operation. Periodic visual inspection and operability testing of the motor-operated valves in the passive core cooling system confirm valve operability. In addition, the location of the motor-operated valves within the containment, which are identified in Table 6.3-1, has been examined to identify remotely operated valves which may be submerged following a postulated loss of coolant accident.

See Section 3.4 for additional information on containment flooding effects.

6.3.2.2.8.5 Automatic Depressurization Valves

The automatic depressurization system consists of four different stages of valves. The first three stages each have two lines and each line has two valves in series; both normally closed. The fourth stage has four lines with each line having two valves in series; one normally open and one normally closed. The four stages, therefore, include a total of 20 valves. The four valve stages open sequentially.

The first stage, second-stage and third-stage valves have dc motor operators. The stage 1/2/3 control valves are normally closed globe valves; the isolation valves are normally closed gate valves. The fourth-stage valves are interlocked so that they can not open until reactor coolant system pressure has been substantially reduced. The fourth stage control valves are squib valves. There is a normally open motor-operated gate valve in series with each squib valve.

The first three stages have a common inlet header connected to the top of the pressurizer. The outlet of the first to third stages then combine to a common discharge line to one of the spargers in the in-containment refueling water storage tank. There is a second identical group of first- to third-stage valves with its own inlet and outlet line and sparger.

The fourth-stage valves connect directly to the top of the reactor coolant hot leg and vent directly to the steam generator compartment. There are also two groups of fourth stage valves, with one group in each steam generator compartment.

The automatic depressurization valves are designed to automatically open when actuated and to remain open for the duration of an automatic depressurization event. Valve stages 1 and 4 actuate at discrete core makeup tank levels, as either tank’s level decreases during injection or from spilling out a broken injection line. Valve stages 2 and 3 actuate based upon a timed delay after actuation of the preceding stage. This opening sequence provides a controlled depressurization of the reactor coolant system. The valve opening sequence prevents simultaneous opening of more than one stage, to allow the valves to sequentially open. The valve actuation logic is based on two-of-four level detectors, in either core makeup tank for automatic depressurization system stages 1 and 4.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The stage 1/2/3 automatic depressurization control valves are designed to open relatively slowly. During the actuation of each stage, the isolation valve is sequenced open before the control valve. Therefore, there is some time delay between stage actuation and control valve actuation.

The operators can manually open the first-stage valves to a partially open position to perform a controlled depressurization of the reactor coolant system. Additional information on the automatic depressurization valves is provided in subsection 5.4.6.

6.3.2.2.8.6 Low Differential Pressure Opening Check Valves

Several applications in the passive core cooling system gravity injection piping use check valves that open with low differential pressures. These check valves are installed in the following locations:

- The gravity injection line flow paths from the in-containment refueling water storage tank
- The containment recirculation lines that connect to the gravity injection lines

The check valves selected for these applications incorporate a simple swing-check design with a stainless steel body and hardened valve seats. The passive core cooling system check valves are safety-related, designed with their operating parts contained within the body, and with a low pressure drop across each valve. The valve internals are exposed to low temperature reactor coolant or borated refueling water.

During normal plant operation, these check valves are closed, with essentially no differential pressure across them. Confidence in the check valve operability is provided by operation at no differential pressure clean/cold fluid environment, the simple valve design, and the specified seat materials.

The check valves normally remain closed, except for testing or when called upon to open following an event to initiate passive core cooling system operation. The valves are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating, and they do not experience significant wear of the moving parts.

These check valves are periodically tested during shutdown conditions to demonstrate valve operation. These check valves are equipped with nonintrusive position sensors to indicate when the valves are open or closed.

In current plants, there are many applications of simple swing-check valves that have similar operating conditions to those in the passive core cooling system. The extensive operational history and experience derived from similar check valves used in the safety injection systems of current pressurized water reactors indicate that the design is reliable. Check valve failure to open and common mode failures have not been significant problems.

6.3.2.2.8.7 Accumulator Check Valves

The accumulator check valve design is similar to the accumulator check valves in current pressurized water reactor applications. It is also similar to the low differential pressure opening check valve design described in subsection 6.3.2.2.8.6. The accumulator check

valves are diverse from the core makeup tank valves because they use different check valve types.

During normal operation, the check valves are in the closed position with a nominal differential pressure across the disc of about 1550 psid. The valves remain in this position, except for testing or when called upon to open following an event. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. They do not experience significant wear of the moving parts and they are expected to function with minimal backleakage.

The accumulators can accept some inleakage from the reactor coolant system without affecting availability. Continuous inleakage requires that the accumulator water volume and boron concentration be adjusted periodically to meet technical specification requirements.

The AP1000 accumulator check valves are periodically tested during shutdown conditions to demonstrate their operation.

6.3.2.2.8.8 Relief Valves

Relief valves are installed for passive core cooling system accumulators to protect the tanks from overpressure.

The passive core cooling system piping is reviewed to identify those lengths of piping that are isolated by normally closed valves and that do not have pressure relief protection in the piping section between the valves.

These piping sections include:

- Portions of in-containment passive core cooling system test lines that are not passive core cooling system accident mitigation flow paths and are not needed to achieve safe shutdown

- Piping vents, drains, and test connections that typically have two closed valves or one closed valve and a blind flange

- Check valve test lines with sections isolated by two normally closed valves.

The piping vents, drains, test connections, and check valve lines have design pressure/ temperature conditions compatible with the process piping to which they connect. Valve leakage does not overpressurize the isolated piping sections and pressure relief provisions are not required.

6.3.2.2.8.9 Explosively Opening (Squib) Valves

Squib valves are used in several passive core cooling system lines in order to provide the following:

- Zero leakage during normal operation
- Reliable opening during an accident
- Reduced maintenance and associated personnel radiation exposure

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Squib valves are used to isolate the incontainment refueling water storage tank injection lines and the containment recirculation lines. In these applications, the squib valves are not expected to be opened during normal operation and anticipated transients. In addition, after they are opened it is not necessary that they re-close.

In the incontainment refueling water storage tank injection lines, the squib valves are in series with normally closed check valves. In the containment recirculation lines, the squib valves are in series with normally closed check valves in two lines and with normally open motor operated valves in the other two lines. As a result, inadvertent opening of these squib valves will not result in loss of reactor coolant or in draining of the incontainment refueling water storage tank.

The type of squib valve used in these applications provides zero leakage in both directions. It also allows flow in both directions. A valve open position sensor is provided for these valves. The IRWST injection squib valves and the containment recirculation squib valves in series with check valves are diverse from the other containment recirculation squib valves. They are designed to different design pressures. The IRWST injection and the containment recirculation squib valves are qualified to operate after being submerged; this capability adds margin to the performance of the PXS in handling debris during long-term core cooling following a LOCA.

Squib valves are also used to isolate the fourth stage automatic depressurization system lines. These squib valves are in series with normally open motor operated gate valves. Actuation of these squib valves requires signals from two separate protection logic cabinets. This helps to prevent spurious opening of these squib valves. The type of squib valve used in this application provides zero leakage of reactor coolant out of the reactor coolant system. The reactor coolant pressure acts to open the valve. A valve open position sensor is provided for these valves.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP1000 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Equipment Class C components and piping, that provide an emergency core cooling function, have augmented weld inspection requirements (see subsection 3.2.2.5). Some system piping and components that do not perform safety-related functions are nonsafety-related.

The requirements for the control, actuation, and Class 1E devices are presented in Chapters 7 and 8.

6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to the meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.

- Valve seating surfaces are hard-faced to prevent failure and to reduce wear.

- Valve stem materials are selected for their corrosion resistance, high-tensile properties, and their resistance to surface scoring by the packing.

Section 6.1 summarizes the materials used for passive core cooling system components.

6.3.2.5 System Reliability

The reliability of the passive core cooling system is considered including periodic testing of the components during plant operation. The passive core cooling system is a redundant, safety-related system. The system is designed to withstand credible single active or passive failures.

The initiating signals for the passive core cooling system are derived from independent sources as measured from process parameters (pressurizer low pressure) or environmental (containment high pressure) variables. Redundant, as well as functionally independent variables, are measured to initiate passive core cooling system operation.

Redundant passive core cooling system components are physically separated and protected so that a single event cannot initiate a common failure.

Power sources for the passive core cooling system are divided into four independent divisions that are supplied from the Class 1E dc and UPS system. Sufficient battery capacity is maintained to provide required power to the emergency loads when onsite and offsite ac power sources are not available. Section 8.3 provides additional information.

The preoperational testing program confirms that the systems, as designed and constructed meet the functional design requirements. Section 14.2 provides additional information. The passive core cooling system is designed with the capability for on-line testing of its active components so the availability and operation status can be readily determined. Testing of passive components such as check valves, tanks, heat exchanger, and flow paths can be conducted during shutdown conditions. In addition, the integrity of the passive core cooling system is verified through examination of critical components during the routine in-service inspection. Section 3.9.6 provides additional information.

The reliability assurance program described in Section 16.2, extends to the procurement of passive core cooling system components. The procurement quality assurance program is described in Chapter 17.

The passive core cooling system is a redundant, safety-related system. During the long-term cooling period following a loss of coolant accident, once the passive core cooling system equipment has actuated, there is no long-term maintenance required. Components actuate to the safeguards actuation alignment and do not need subsequent position changes for long-term operation.

For long-term cooling, the reactor coolant system is depressurized to containment ambient pressure following a loss of coolant accident. During this period, the heat generated in the reactor core is the residual decay heat and the passive core cooling system provides the required decay heat removal.

Proper initial filling and venting of the passive core cooling system prevents water hammer from occurring in the passive core cooling system lines. In addition, the head of water provided by the various tanks keeps system lines full. The arrangement of the core makeup tank pressure equalization line design also reduces the potential for water hammer. High-point vents in the passive core cooling system lines are provided as a means for venting of lines. Fill and venting procedures for the passive core cooling system provide for the removal of air from the system.

The existence of high-point vents and the positive head of water provide means by which the operator can confirm water-solid passive core cooling system lines, where required.

6.3.2.5.1 Response to Active Failure

Treatment of active failures is described in Section 15.0.12.

An active failure is the failure of a powered component, a component of the electrical supply system, or instrumentation and control equipment to act on command to perform its function. One example is the failure of a motor-operated valve to move to its intended safeguards actuation position.

One change in the definition of active failures has been incorporated into the passive core cooling system design. The system has been specifically designed to treat check valve failures to reposition as active failures. More specifically, it is assumed that normally closed check valves may fail to open and normally open check valves may fail to close. Check valves that remain in the same position before and after an event are not considered active failures.

There are two exceptions to this treatment of check valve failures in the passive core cooling system. One exception is made for the accumulator check valves, which is consistent with the treatment of these specific check valves in currently licensed plant designs. The other exception is made for the core makeup tank check valves failure to re-open after they have closed during an accident. The valves are normally open, biased-opened check valves. This exception is based on the low probability of these check valves not re-opening within a few minutes after they have cycled closed during accumulator operation.

The failure mode and effects analysis provided in Table 6.3-3 provides a summary of the passive core cooling system response to single failure of the various components.

The following passive core cooling system motor-operated valves are not included in this analysis:

- Both accumulator discharge line motor-operated valves
- Both in-containment refueling water storage tank gravity injection line motor-operated valves
- Both containment recirculation line motor-operated valves
- Both core makeup tank inlet line motor-operated valves

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- The passive residual heat removal heat exchanger inlet line motor-operated valve

These valves are normally in the required position for actuation of the associated component, they have redundant position indications and alarms, and they also receive confirmatory open actuation signals. The accumulator, incontainment refueling water storage tanks and passive residual heat removal heat exchanger valves have their power removed and locked out. The core makeup tank and the containment recirculation line have redundant series controllers. Therefore, these valves are not considered in the failure modes and effects analysis.

The analysis illustrates that the passive core cooling system can sustain an active failure in either the short-term or long-term and meet the required level of performance for core cooling. The short-term operation of the active components of the passive core cooling system following a steam line rupture or a steam generator tube rupture is similar to that following a loss of coolant accident. The same analysis is applicable and the passive core cooling system can sustain the failure of a single active component and meet the level of performance for the addition of shutdown reactivity.

Portions of the passive core cooling system are also relied upon to provide boration and makeup during a safety-related shutdown. The passive core cooling system can sustain an active failure and perform the required functions necessary to establish safe shutdown conditions. Safe shutdown operation of the passive core cooling system is described in Section 7.4.

6.3.2.5.2 Response to Passive Failure

Treatment of passive failures is described in subsection 15.0.12.

A passive failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. Examples include cracking of pipes, sprung flanges, or valve packing leaks. The passive core cooling system can sustain a single passive failure during the long-term phase and still retain an intact flow path to the core to supply sufficient flow to keep the core covered and to remove decay heat.

Since the passive core cooling system equipment is inside the containment, offsite dose caused by passive failures is not a concern. Also, with actuation of the automatic depressurization system, the reactor coolant system pressure is very close to containment pressure. Therefore, it is not necessary to isolate or realign the passive core cooling system following a passive failure.

The passive core cooling system flow paths are separated into redundant lines, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the reactor coolant system. For the long-term passive core cooling system function, adequate core cooling capacity exists with one of the two redundant flow paths.

6.3.2.5.3 Lag Times

Lag times for initiation and operation of the passive core cooling system are controlled by repositioning of valves. Some valves are normally in the position required for safety-related system function and therefore, their valve operation times are not considered. For those valves that reposition to initiate safety-related system functions, the valve repositioning times

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

are less than the times assumed in the accident analyses. These lag times refer to the time after initiation of the safeguards actuation signal.

It is acceptable for the core makeup tank injection to be delayed several minutes following actuation due to high initial steam condensation rates in the tank.

6.3.2.5.4 Potential Boron Precipitation

Boron precipitation in the reactor vessel is prevented by sufficient flow of passive core cooling system water through the core to limit the increase in boron concentration of the water remaining in the reactor vessel. Water along with steam leaves the core and exits the RCS through the fourth stage ADS lines. These valves connect to the hot leg and open in about 20 minutes after a loss of coolant accident or an automatic depressurization system actuation.

6.3.2.5.5 Safe Shutdown

During a safe shutdown, the passive core cooling system provides redundancy for boration, makeup, and heat removal functions. Section 7.4 provides additional information about safe shutdown.

6.3.2.6 Protection Provisions

The measures taken to protect the system from damage that might result from various events are described in other sections, as listed below.

- Protection from dynamic effects is presented in Section 3.6.
- Protection from missiles is presented in Section 3.5.
- Protection from seismic damage is presented in Sections 3.7, 3.8, 3.9, and 3.10.
- Protection from fire is presented subsection 9.5.1.
- Environmental qualification of equipment is presented in Section 3.11.
- Thermal stresses on the reactor coolant system are presented in Section 5.2.

6.3.2.7 Provisions for Performance Testing

The passive core cooling system includes the capability for determination of the integrity of the pressure boundary formed by series passive core cooling system check valves. Additional information on testing can be found in subsection 6.3.6.

6.3.2.8 Manual Actions

The passive core cooling system is automatically actuated for those events as presented in subsection 6.3.3. Following actuation, the passive core cooling system continues to operate in the injection mode until the transition to recirculation initiates automatically following containment floodup.

Although the passive core cooling system operates automatically, operator actions would be beneficial, in some cases, in reducing the consequences of an event. For example, in a steam generator tube rupture with no operator action, the protection and safety monitoring system automatically terminates the leak, prevents steam generator overfill, and limits the offsite doses. However, the operator can initiate actions, similar to those taken in current plants, to

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

identify and isolate the faulted steam generator, cool down and depressurize the reactor coolant system to terminate the break flow to the steam generator, and stabilize plant conditions.

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

6.3.3 Performance Evaluation

The events described in subsection 6.3.1 result in passive core cooling system actuation and are mitigated within the performance criteria. For the purpose of evaluation in Chapters 15 and 19, the events that result in passive core cooling system actuation are categorized as follows:

A. Increase in heat removal by the secondary system

   1. Inadvertent opening of a steam generator power-operated atmospheric steam relief or safety valve

   2. Steam system piping failure

B. Decrease in heat removal by the secondary system

   1. Loss of Main Feedwater Flow
   2. Feedwater system piping failure

C. Decrease in reactor coolant system inventory

   1. Steam generator tube rupture

   2. Loss of coolant accident from a spectrum of postulated reactor coolant system piping failures

   3. Loss of coolant due to a rod cluster control assembly ejection accident

      (This event is enveloped by the reactor coolant system piping failures.)

D. Shutdown Events (Chapter 19)

   1. Loss of Startup Feedwater

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

2. Loss of normal residual heat removal system with reactor coolant system pressure boundary intact

3. Loss of normal residual heat removal system during mid-loop operation

4. Loss of normal residual heat removal system with refueling cavity flooded

The events listed in groups A and B are non-LOCA events where the primary protection is provided by the passive core cooling system passive residual heat removal heat exchanger. For these events, the passive residual heat removal heat exchanger is actuated by the protection and monitoring system for the following conditions:

- Steam generator low narrow range level, coincident with startup feedwater low flow
- Steam generator low wide range level
- Core makeup tank actuation
- Automatic depressurization actuation
- Pressurizer water level - High 3
- Manual actuation

The events listed in group C above are events involving the loss of reactor coolant where the primary protection is by the core makeup tanks and accumulators. For these events the core makeup tanks are actuated by the protection and monitoring system for the following conditions:

- Pressurizer low pressure

- Pressurizer low level

- Steam line low pressure

- Containment high pressure

- Cold leg low temperature

- Steam generator low wide range level, coincident with reactor coolant system high hot leg temperature

- Manual actuation

In addition to initiating passive core cooling system operation, these signals initiate other safeguards automatic actions including reactor trip, reactor coolant pump trip, feedwater isolation, and containment isolation. The passive core cooling system actuation signals are described in Section 7.3.

The core makeup tanks and passive residual heat removal heat exchangers are also actuated by the Diverse Actuation System as described in subsection 7.7.1.11.

Upon receipt of an actuation signal, the actions described in subsection 6.3.2.1 are automatically initiated to align the appropriate features of the passive core cooling system.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in Subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition of 420°F within 36 hours as described in Subsection 19E.4.10.2. A non-bounding, conservative analysis of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of Subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

For loss of coolant accidents, the core makeup tanks deliver borated water to the reactor coolant system via the direct vessel injection nozzles. The accumulators deliver flow to the direct vessel injection line whenever reactor coolant system pressure drops below the tank static pressure. The in-containment refueling water storage tank provides gravity injection once the reactor coolant system pressure is reduced to below the injection head from the in-containment refueling water storage tank. The passive core cooling system flow rates vary depending upon the type of event and its characteristic pressure transient.

As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available. If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in Subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe shutdown condition and maintaining that condition.

The events listed in group D occur during shutdown conditions that are characterized by slow plant responses and mild thermal-hydraulic transients. In addition, some of the passive core cooling system features need to be isolated to allow the plant to be in these conditions or to perform maintenance on the system. The protection and monitoring system automatically actuates gravity injection from the IRWST to provide core cooling during shutdown conditions prior to refueling cavity floodup. In addition, the operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, to provide core cooling during shutdown conditions when the equipment does not automatically actuate.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.3.1 Increase in Heat Removal by the Secondary System

A number of events that could result in an increase in heat removal from the reactor coolant system by the secondary system have been postulated. For each event, consideration has been given to operation of nonsafety-related systems that could affect the event results. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.1. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.1.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Subsection 15.1.4 provides a description of an inadvertent opening of a steam generator relief or safety valve, including criteria and analytical results.

For this event, upon generation of a safeguards actuation signal the reactor is tripped, the core makeup tanks are actuated, and the reactor coolant pumps are tripped. Since the core makeup tanks are actuated, the passive residual heat removal heat exchanger is also actuated. The main steam lines are also isolated to prevent blowdown of more than one steam generator. The core makeup tanks operate with water recirculation injection to provide borated water to the reactor vessel downcomer plenum for reactor coolant system inventory and reactivity control. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met, thereby preventing fuel damage.

During this event, the startup feedwater system is assumed to malfunction so that it injects water at the maximum flow rate. This injection continues until feedwater isolation occurs on low reactor coolant system temperature. The feedwater isolation signal terminates the feedwater addition from the startup feedwater system. The passive residual heat removal heat exchanger is also assumed to function in this event. This heat removal mechanism continues throughout the duration of the event.

For this event, the core makeup tanks operate in the water recirculation mode, providing boration and injection flow without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level.

Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.1.2 Steam System Pipe Failure

The most severe core conditions resulting from a steam system piping failure are associated with a double-ended rupture of a main steam line, occurring at zero power. Effects of smaller piping failures at higher power levels are bounded by the double-ended rupture at zero power.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Subsection 15.1.5 provides a description of this event, including criteria and analytical results.

For this event, the passive core cooling system functions as described in subsection 6.3.3.1.1 for the inadvertent opening of a steam generator relief or safety valve. However, this piping failure constitutes a more severe cooldown transient. The malfunctioning of the startup feedwater system is considered as it was in the inadvertent steam generator depressurization. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met.

For this event, the reactor coolant system may depressurize sufficiently to permit the accumulators to deliver makeup water to the reactor coolant system. The core makeup tanks inject via water recirculation without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2 Decrease in Heat Removal by the Secondary System

A number of events have been postulated that could result in a decrease in heat removal from the reactor coolant system by the secondary system. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of an event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.2. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.2.1 Loss of Main Feedwater

The most severe core conditions resulting from a loss of main feedwater system flow are associated with a loss of flow at full power. The heat-up transient effects of loss of flow at reduced power levels are bounded by the loss of flow at full power. Subsection 15.2.7 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level. The passive residual heat removal heat exchanger serves to remove core decay heat and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer annulus. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation, without draining, to maintain reactor coolant system inventory. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Since the event is characterized by a heat-up transient, the injection of

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in Subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.2.2 Feedwater System Pipe Failure

The most severe core conditions resulting from a feedwater system piping failure are associated with a double-ended rupture of a feed line at full power. Depending on break size and power level, a feedwater system pipe failure could cause either a reactor coolant system cooldown transient or a reactor coolant system heat-up transient. Only the reactor coolant system heat-up transient is evaluated as a feedwater system pipe failure, since the spectrum of cooldown transients is bounded by the steam system pipe failure analyses. The heat-up transient effects of smaller piping failures at reduced power levels are bounded by the

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double-ended feed line rupture at full power. Subsection 15.2.8 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated. The passive residual heat removal heat exchanger serves to remove core decay heat, and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation to maintain reactor coolant system inventory. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.3 Decrease in Reactor Coolant System Inventory

A number of events have been postulated that could result in a decrease in reactor coolant system inventory. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of the event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.6. For those events which result in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.3.1 Steam Generator Tube Rupture

Although a steam generator tube rupture is an event that results in a decrease in reactor coolant system inventory, severe core conditions do not result from a steam generator tube rupture. The event analyzed is a complete severance of a single steam generator tube that occurs at power with the reactor coolant contaminated with fission products, corresponding to continuous operation with a limited amount of defective fuel rods. Effects of smaller breaks are bounded by the complete severance. Subsection 15.6.3 provides a description of this event, including criteria and analytical results.

For this event, the nonsafety-related makeup pumps are automatically actuated when reactor coolant system inventory decreases and a reactor trip occurs, followed by actuation of the startup feedwater pumps. The startup feedwater flow initiates on low steam generator level following the reactor trip and automatically throttles feedwater flow to maintain programmed steam generator level, limiting overfill of the faulted steam generator. The makeup pumps automatically function to maintain the programmed pressurizer level. The operators are expected to take actions similar to those in current plants to identify and isolate the faulted steam generator, cooldown and depressurize the reactor coolant system to terminate the break flow into the steam generator, and stabilize plant conditions.

If the operator fails to take timely or correct actions in response to the leak, or if the makeup pumps and/or the startup feedwater pumps malfunction with excessive flow, then the water level in the faulted steam generator continues to increase. This actuates safety-related overfill

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protection and automatically isolates the startup feedwater pumps and the chemical and volume control system makeup pumps. The core makeup tanks subsequently actuate on low pressurizer level, if they are not already actuated. Actuation of the core makeup tanks automatically actuates the passive residual heat removal system heat exchanger.

The core makeup tanks operate via water recirculation to provide borated water directly into the reactor vessel downcomer to maintain reactor coolant system inventory. The passive residual heat removal heat exchanger serves to remove core decay heat. Since the reactor coolant pumps are automatically tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation flow conditions. The passive residual heat removal heat exchanger, in conjunction with the core makeup tanks, remove core decay heat and reduce reactor coolant system temperature. As the reactor coolant system cools and the inventory contracts, pressurizer level and pressure decrease, equalizing with steam generator pressure and terminating break flow.

If the nonsafety-related systems fail to start, the core makeup tanks and the passive residual heat removal heat exchangers automatically actuate. Their response is similar to that previously described, except that the faulted steam generator level is lower.

In these events, the plant conditions are stabilized without actuating the automatic depressurization system. Once plant conditions are stable, the operator completes a normal plant shutdown.

6.3.3.3.2 Loss of Coolant Accident

A loss of coolant accident is a rupture of the reactor coolant system piping or branch piping that results in a decrease in reactor coolant system inventory that exceeds the flow capability of the normal makeup system. Ruptures resulting in break flow within the capability of the normal makeup system do not result in decreasing reactor coolant system pressure and actuation of the passive core cooling system. The maximum break size for which the normal makeup system can maintain reactor coolant system pressure is obtained by comparing the calculated flow from the reactor coolant system through the postulated break with the charging pump makeup flow at a reactor coolant system pressure that is above the low pressure safeguards actuation setpoint. The makeup flow rate from one makeup pump is adequate to maintain pressurizer pressure for a break through a 0.375-inch diameter hole. Therefore, the normal makeup system can maintain reactor coolant system pressure and permit the operator to execute an orderly shutdown.

For the purpose of evaluation, the spectrum of postulated piping breaks in the reactor coolant system is divided into major pipe breaks (large break) and minor pipe breaks (small breaks). The large break is a rupture with a total cross-sectional area equal to or greater than one square foot. The small break is defined as a rupture with a total cross-sectional area less than one square foot. Section 15.6 provides a description of this event, including criteria and analytical results.

For either event, the core makeup tanks are actuated upon receipt of a safeguards actuation signal. These tanks provide high-pressure injection. For large breaks, or after the automatic depressurization system is actuated, the accumulators also provide injection. After automatic depressurization system actuation, the in-containment refueling water storage tank, and the containment recirculation sump, provide low pressure injection.

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The core makeup tanks can operate via water recirculation or steam-compensated injection during LOCAs. For smaller loss of coolant accidents, the reactor coolant system inventory is sufficient to establish water recirculation. For larger break sizes, when the pressurizer empties and voiding occurs in the cold legs steam-compensated injection initiates. When the cold legs void, the core makeup tank flow increases.

As the core makeup tanks drain, their level sequences the automatic depressurization system valve stages. As the level drops in the core makeup tank, the first-stage actuates. The first-stage valves are connected to the top of the pressurizer and discharge to the in-containment refueling water storage tank via the automatic depressurization system spargers. After a time delay, the second-stage is actuated. The second stage valves are connected with the same flow path as the first-stage valves. After an additional time delay, the third-stage is actuated. The third stage valves are identical to the second-stage valves. As the core makeup tank drops to a low level the fourth-stage is actuated. The fourth stage valves are connected to both hot legs and they discharge directly to the reactor coolant system loop compartments at an elevation just above the maximum containment floodup level.

The in-containment refueling water storage tank line squib valves are opened on the fourth stage actuation signal. Check valves arranged in series with the squib valves remain closed until the reactor depressurizes. After depressurization, the in-containment refueling water storage tank provides injection flow. The flow continues until containment floodup initiates containment recirculation.

For large breaks or following automatic depressurization system initiation, the accumulators provide rapid injection to the reactor vessel through the same connections used by the core makeup tanks and the in-containment refueling water storage tank injection. The accumulators begin to inject when the reactor coolant system depressurizes to about 700 psig. During the loss of coolant accident transient, flow to the reactor coolant system is dependent on the reactor coolant system pressure transient. The passive core cooling system water injected into the reactor coolant system provides for heat transfer from the core, prevents excessive core clad temperatures, and refloods the core (for large loss of coolant accidents) or keeps the core covered (for small loss of coolant accidents).

For small loss of coolant accidents, the control rods provide the initial core shutdown and the boron in the passive core cooling system tanks add negative reactivity to provide adequate shutdown at low temperatures.

Following the initial thermal-hydraulic transient for a loss of coolant accident event, the passive core cooling system continues to supply water to the reactor coolant system for long-term cooling. When the water level in the in-containment refueling water storage tank drops to a low-low level, the water level in the containment has increased to a sufficient level to provide recirculation flow. The in-containment refueling water storage tank low-low level signal opens the squib valves in the lines between the containment and the gravity injection line. Initially, some of the water remaining in the tank drains to the containment until the water levels equalize. During this drain, injection to the core continues. The redundant flow paths provide continued cooling of the core by recirculation of the water in the containment. Figure 6.3-3 provides process flow information illustrating passive core cooling system performance for the various modes of system operation.

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6.3.3.3.3 Passive Residual Heat Removal Heat Exchanger Tube Rupture

Although a passive residual heat removal heat exchanger tube rupture is an event that results in a decrease in reactor coolant system inventory, severe core conditions do not result from this event. There is a spectrum of heat exchanger tube leak sizes that are possible. For a small initiating leak, the passive core cooling system temperature instrumentation for the heat exchanger is used to identify that this is a heat exchanger leak. If the leak rate is less than the Technical Specification limits, plant operation can continue indefinitely. If the leak rate exceeds the Technical Specification limits the plant would be shut down to repair the heat exchanger.

If a severe tube leak occurs, the operators can use available instrumentation to identify the leak source. Action can then be taken to remotely isolate the heat exchanger by closing the motor-operated inlet isolation valve, which is normally open. The plant would be shut down to repair the heat exchanger.

This event is addressed in Section 15.6.

6.3.3.4 Shutdown Events

The passive core cooling system components are available whenever the reactor is critical and when reactor coolant energy is sufficiently high to require passive safety injection. During low-temperature physics testing, the core decay heat levels are low and there is a negligible amount of stored energy in the reactor coolant. Therefore, an event comparable in severity to events occurring at operating conditions is not possible and passive core cooling system equipment is not required. The possibility of a loss of coolant accident during plant startup and shutdown has been considered.

During shutdown conditions, some of the passive core cooling system equipment is isolated. In addition, since the normal residual heat removal system is not a safety-related system, its loss is considered.

As a result, gravity injection is automatically actuated when required during shutdown conditions prior to refueling cavity floodup, as discussed in subsection 6.3.3.3.2. The operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, if required for accident mitigation during shutdown conditions when the equipment does not automatically actuate.

6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

During normal cooldowns, the steam generators are supplied by the startup feedwater pumps and steam from the steam generator is directed to either the main condenser or to the atmosphere. There are two nonsafety-related startup feedwater pumps, each of which is capable of providing sufficient feedwater flow to both steam generators to remove decay heat. These pumps are also automatically loaded on the nonsafety-related diesel-generators in the event offsite power is lost. Since these pumps are nonsafety-related, their failure is considered.

In the event of a loss of startup feedwater, the passive residual heat removal heat exchanger is automatically actuated on low steam generator water level and provides safety-related heat removal. The passive residual heat removal heat exchanger can maintain the reactor coolant

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system temperature, as well as provide for reactor coolant system cooldown to conditions where the normal residual heat removal system can be operated.

Since the chemical and volume control system makeup pumps are nonsafety-related, they may not be available. In this case, the core makeup tanks automatically actuate as the cooldown continues and the pressurizer level decreases. The core makeup tanks operate in a water recirculation mode to maintain reactor coolant system inventory while the passive residual heat removal heat exchanger is operating.

The in-containment refueling water storage tank provides the heat sink for the passive residual heat removal heat exchanger. Initially, the heat addition increases the water temperature. Within one to two hours, the water reaches saturation temperature and begins to boil. The steam generated in the in-containment refueling water storage tank discharges to containment. Because the containment integrity is maintained during cooldown Modes 3 and 4, the passive containment cooling system provides the safety-related ultimate heat sink. Therefore, most of the steam generated in the in-containment refueling water storage tank is condensed on the inside of the containment vessel and drains back into the in-containment refueling water storage tank via the condensate return gutter arrangement. This allows it to ~~indefinitely~~ function as a heat sink.

6.3.3.4.2 Loss of Normal Residual Heat Removal Cooling With The Reactor Coolant System Pressure Boundary Intact

During normal shutdown conditions, the normal residual heat removal system is placed into service at about 350°F to accomplish reactor coolant system cooldown to refueling temperatures. The normal residual heat removal system piping is safety-related and meets seismic Category I requirements to prevent pipe breaks that could result in a significant loss of reactor coolant during system operation. The pump motors and the electrical power supplies are nonsafety-related.

The system is designed so that with single failure of an active system component, it can maintain the plant in a hot shutdown condition (<350°F). It is also possible to perform a reactor coolant system cooldown, but at a slower rate than with full system capability. Heat removed by the normal residual heat removal system is transferred to the component cooling water system and then to the service water system. The heat removal path is powered by the nonsafety-related diesel-generators in the event that offsite power is lost.

Since the normal residual heat removal pumps are nonsafety-related, they may not be available. In this case, the reactor coolant system pressure boundary remains intact and the passive residual heat removal heat exchanger provides the safety-related heat removal flow path.

The normal residual heat removal system is operated once the reactor coolant system temperature is too low to support sufficient steam production for decay heat removal. With a loss of shutdown cooling, the reactor coolant system temperature does not increase sufficiently to initiate steam generator steaming and to reduce steam generator level. This is because the steam generators are normally filled, with a nitrogen purge established, during shutdown conditions. The loss of cooling would result in the heat up of the reactor coolant system and a pressure increase resulting in the normal residual heat removal system relief valve opening. This loss of fluid would result in a decrease in the pressurizer level; which a low pressurizer level signal automatically actuates the core make tanks and the passive

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residual heat removal heat exchanger. The passive residual heat removal heat exchanger could also be manually actuated.

The passive residual heat removal heat exchanger is capable of functioning at low reactor coolant system temperatures and pressures, but it may not be able to maintain the initial reactor coolant system temperature. It can remove sufficient heat to maintain the reactor coolant system within the normal residual heat removal system design limits (400°F). This permits the normal residual heat removal system to be placed back in operation when it becomes available.

For this event, the reactor coolant system temperature is expected to increase and expand into the pressurizer. Reactor coolant system injection should not be required. The makeup pumps are aligned for automatic operation in the event that pressurizer level decreases, due to leakage. However, since they are nonsafety-related, they are considered unavailable for reactor coolant system makeup. Therefore should safety-related makeup be required, the core makeup tanks would automatically actuate and operate via water recirculation injection. For some scenarios, the core makeup tanks could drain down and actuate the automatic depressurization system valves. This would lead to injection via the in-containment refueling water storage tank and containment recirculation paths.

6.3.3.4.3 Loss of Normal Residual Heat Removal Cooling During Reduced Inventory

During reactor coolant system maintenance, the most limiting shutdown condition anticipated is with the reactor coolant level reduced and the reactor coolant system pressure boundary opened. It is normal practice to open the steam generator channel head manway covers to install the hot leg and cold leg nozzle dams during a refueling outage. In this situation, the normal residual heat removal system is used to cool the reactor coolant system. The AP1000 incorporates many features to reduce the probability of losing the normal residual heat removal system. However, since the normal residual heat removal system is nonsafety-related, its failure has been considered. The normal residual heat removal system is described subsection 5.4.7.

In reduced inventory operation with the reactor coolant system depressurized and the pressure boundary opened, the passive residual heat removal heat exchanger is unable to remove the decay heat because the reactor coolant system cannot heat sufficiently above the in-containment refueling water storage tank temperature.

In this situation, core cooling is provided by the safety-related passive core cooling system, using gravity injection from the in-containment refueling water storage tank, while venting through the automatic depressurization system valves (and possibly through other openings in the reactor coolant system).

Prior to draining the reactor coolant system inventory below the no-load pressurizer level, the core makeup tanks are isolated to preclude inadvertent draining into the reactor coolant system while preparing for midloop operation. During plant shutdown, at 1000 psig, the accumulators are isolated to prevent inadvertent injection. In this configuration, the core makeup tanks and accumulators are isolated from the reactor coolant system, however these valves can be remotely opened with operator action to provide additional makeup water injection, if required.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Before the core makeup tanks are isolated, the automatic depressurization first-, second-, and third-stages valves are opened manually be the operators. The automatic depressurization system first-, second- and third-stage valves are required to remain open whenever the reactor coolant inventory is reduced or the upper core internals are in place. During an extended loss of normal residual heat removal system operation the stage one, two and three vent paths may not provide sufficient vent capability to allow gravity injection of water from the in-containment refueling water storage tank because of pressurizer surge line flooding. As a result, two of the automatic depressurization stage four paths are required to be operable in these conditions. The stage four valves are automatically opened by a signal from the protection and monitoring system on a low hot leg level signal following a time delay.

The in-containment refueling water storage tank injection squib valves automatically open via the same low hot leg level signal that opens the automatic depressurization stage four valves. The operators can also open these injection and depressurization valves via the diverse actuation system. Once these valves open, injection from the in-containment refueling water storage tank provides gravity injection for core cooling. When the in-containment refueling water storage tank level drops to a low level, the squib valves in the containment recirculation line automatically open. This action initiates containment recirculation flow, with flow passing through the in-containment refueling water storage tank gravity injection lines, which provides long-term core cooling.

This arrangement provides automatic core cooling protection, while in reduced inventory operation while also providing protection (an evacuation alarm and sufficient time to evacuate) for maintenance personnel in containment during midloop operation. The time delay also provides the operators with time to take actions to restore nonsafety-related decay heat removal prior to actuating the passive core cooling system.

During reduced inventory conditions the capability of closing the containment is required. After the containment is closed, containment recirculation can continue indefinitely, with the decay heat generating steam which condenses on the containment vessel and drains back into the in-containment refueling water storage tank.

6.3.3.4.4 Loss of Normal Residual Heat Removal Cooling During Refueling

The normal residual heat removal system is normally used for decay heat removal during refueling operation. Its failure is considered because it is not a safety-related system. In this case, it is assumed that the reactor vessel head is removed and the water from the in-containment refueling water storage tank has been transferred to the refueling cavity, which is flooded to its high level condition. The passive residual heat removal heat exchanger is not available and containment integrity is expected to be relaxed with air locks and/or equipment hatches open.

Assuming that the refueling cavity was just flooded when the normal residual heat removal system fails, the refueling cavity water heats up to saturation temperature in about nine hours. With the slow heat-up of the refueling cavity water, there is ample time to close containment before significant steaming to the containment begins. The Technical Specifications require that containment closure capability be maintained during refueling MODES such that closure of the containment can be assumed. With the containment closed, water will not be lost from containment and long-term cooling can be maintained without subsequent need for cooling water makeup. Without closing the containment, boiling would reduce the water level to the top of the fuel assemblies in about five days.

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6.3.4 Post-72 Hour Actions

The AP1000 passive core cooling system design includes safety-related equipment that is sufficient to automatically establish and maintain safe shutdown conditions for the plant following design basis events. The passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power.

There is only one action that may be required to provide long-term core cooling. There is a potential need for containment inventory makeup. The need for makeup to containment is directly related to the leakrate from the containment. With the maximum allowable containment leakrate, makeup to containment is not needed for about one month. A safety-related connection is available in the normal residual heat removal system to align a temporary makeup source to containment.

6.3.5 Limits on System Parameters

The analyses show that the design basis performance of the passive core cooling system is sufficient to meet the core cooling requirements following an event, with the minimum engineered safety features equipment operating. To provide this capability in the event of the single failure of components, technical specifications are established for reactor operation. The technical specifications are provided in Chapter 16.

The passive core cooling system equipment is not required to operate to support either normal power operation or shutdown operation of the plant. This reduces the probability that the passive core cooling system equipment is unavailable due to maintenance. Planned maintenance on the passive core cooling system equipment is accomplished during shutdown operations when the core temperatures are low, decay heat levels are low, and the Technical Specifications do not require availability of the equipment.

The principal system parameters and the number of components that may be out of operation during testing, quantities and concentrations of coolant available, and allowable time for operation in a degraded status are provided in the technical specifications.

If efforts to restore the operable status of the passive core cooling system equipment are not accomplished within technical specification requirements, the plant is required to be placed in a lower operational mode.

6.3.6 Inspection and Testing Requirements

6.3.6.1 Preoperational Inspection and Testing

Preoperational inspections and tests of the passive core cooling system are performed to verify the operability of the system prior to loading fuel. This testing includes valve inspection and testing, flow testing, and verification of heat removal capability.

Preoperational testing of the passive core cooling system is completed in conjunction with testing of the reactor coolant system following flushing and hydrostatic testing, with the system cold and the reactor vessel head removed. The passive core cooling system is aligned for normal power operation. This testing provides the following information:

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- Satisfactory safeguards actuation signal generation and transmission
- Valve operating times
- Injection starting times
- Injection delivery rates

The preoperational testing program includes testing of the following passive core cooling system components:

- Core makeup tanks
- Accumulators
- In-containment refueling water storage tank
- Containment recirculation
- Passive residual heat removal heat exchanger

Conformance with the recommendations of Regulatory Guide 1.79 is described in subsection 1.9.1. Preoperational testing of the passive core cooling system is conducted in accordance with the requirements presented in subsection 14.2.9.1.3.

6.3.6.1.1 Flow Testing

Initial verification of the resistance of the passive core cooling injection lines is performed by conducting a series of flow tests for the core makeup tanks, accumulators, in-containment refueling water storage tank, and containment recirculation piping. The calculated flow resistances are bounded by the resistances used in the Chapter 15 safety analyses.

6.3.6.1.2 Heat Transfer Testing

Initial verification of the heat transfer capability of the passive residual heat removal heat exchanger is performed by conducting a natural circulation test. This test is conducted during hot functional testing of the reactor coolant system. Measurements of heat exchanger flow rate and inlet and outlet temperatures are recorded, and calculations are performed to verify that the heat transfer performance of the heat exchanger is greater than that provided in Table 6.3-2.

6.3.6.1.3 Preoperational Inspections

Preoperational inspections are performed to verify that important elevations associated with the passive core cooling system components are consistent with the accident analyses presented in Chapter 15. The following elevations are verified:

- The bottom inside surface of each core makeup tank is at least 7.5 feet above the direct vessel injection nozzle centerline.

- The bottom inside surface of the in-containment refueling water storage tank is at least 3.4 feet above the direct vessel injection nozzle centerline.

- The centerline of the upper passive residual heat removal heat exchanger channel head is at least 26.3 feet above the hot leg centerline.

- The pH baskets are located below plant elevation 107 feet, 2 inches.

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Inspections of the passive core cooling system tanks and pH adjustment baskets are conducted to verify that the actual tank volumes are greater than or equal to volume assumed in the Chapter 15 accident analyses. Inspections to determine dimensions of the core makeup tanks, accumulators, in-containment refueling water storage tank, and pH adjustment baskets are conducted, and calculations are performed to verify that actual volume is not less than the corresponding minimum required volume listed in Table 6.3-2.

6.3.6.2 In-Service Testing and Inspection

In-service testing and inspection of the passive core cooling system components and the associated support systems are planned. The passive core cooling system components and systems are designed to meet the intent of the ASME Operations and Maintenance (OM) Code, for in-service testing. A description of the in-service testing program is provided in subsection 3.9.6.

Two basic types of in-service testing are performed on the passive core cooling system components:

- Periodic exercise testing of active components during power operation (for example, cycling of specific valves)
- Operability testing of specific passive core cooling system features during plant shutdown (for example, accumulator injection flow to the reactor vessel or leak testing of containment isolation valves during selected plant shutdown.

The passive core cooling system includes specific features to support in-service test performance:

- Remotely operated valves can be exercised during routine plant maintenance
- Level, pressure, flow, and valve position instrumentation is provided for monitoring required passive core cooling system equipment during plant operation and testing
- Permanently installed test lines and connections are provided for operability testing

6.3.6.3 Mitigation of Gas Accumulation

Periodic system surveillance and venting procedures, in addition to specific design features, are implemented that aim to prevent gas accumulation and minimize or eliminate gas whenever found. Locations identified by the gas accumulation assessment have been equipped with manual vent valves or continuously monitored and alarmed pipe stubs with manual vent valves. These locations are specified within the periodic system surveillance and venting procedures as locations of high importance. Locations outfitted with pipe stub collection and alarm features have Technical Specifications including Surveillance Requirements to continuously monitor for gas accumulation and Required Actions subsequent to identifying gas accumulation in those locations to vent the identified gas accumulation. Plant startup and operational procedures include venting and surveillance steps that provide a means to track and trend accumulated gas such that problem areas can be systematically identified, monitored, and corrected.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.6.3.1 System Gas Accumulation Assessment

Reviews of pipe layout and routing drawings to identify high-point vent and low-point drain locations are included as part of system design finalization activities. This existing design activity was expanded for the AP1000 passive safety systems, to integrate the draft Interim Staff Guidance (ISG) document ISG-019 regarding gas intrusion assessment guidance into the design process, helping to confirm that the potential issues identified in Generic Letter 2008-01 have been addressed within the design. Westinghouse also performed a comprehensive assessment for gas intrusion within the passive safety systems, consistent with the methodology in NEI 09-10 as applied in current operating plants, and consistent with the additional guidance in the ISG.

6.3.6.3.2 System Design Features to Mitigate Gas Intrusion

The gas intrusion assessment described in subsection 6.3.6.3.1 helped to identify:

- Potential gas accumulation locations in the passive core cooling system piping.
- Potential gas intrusion mechanisms during various plant conditions (including plant startup, shutdown, post-maintenance system restoration and filling, power operation, and accident conditions).
- Passive core cooling system design features to provide the capability to perform system high-point venting and to continuously monitor several high-point locations.

These passive core cooling system design features help to eliminate the potential for significant gas accumulation in specific passive safety system injection lines that could adversely impact passive safety system operation. System venting capabilities are provided for the following passive safety system locations:

- IRWST injection line squib valve inlet lines
  - Vents located at the piping high points upstream of the parallel paths in both IRWST safety injection lines
  - Vents located between the check and squib valves in each line of the parallel paths in both IRWST safety injection lines
- Core makeup tank outlet lines
  - Vents located upstream of the first check valves in both core makeup tank outlet lines
  - Vents located between the series check valves in the core makeup tank A outlet line
  - Vents located between the second check and manual isolation valves in both core makeup tank outlet lines
- Containment recirculation Line A

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– Vents located at the high points in the common containment recirculation Line A path between the recirculation squib valves and the IRWST injection line (before and after the spent fuel system connection tee)

The potential for gas accumulation in passive safety system IRWST injection lines following accumulator injection is precluded by connecting the accumulator injection line in the direct vessel injection line riser section vertically above the IRWST injection line connection to the direct vessel injection line riser.

In addition, passive safety system design features are provided to monitor for gas accumulation at several specific locations. These design features include pipe stub gas collection chambers with redundant instrumentation at each high point, are continuously monitored and alarmed, have hard piped vent lines, are accessible during power operation, and include Technical Specifications and Surveillance Requirements specifically intended to identify unintended gas accumulations that could potentially challenge passive safety system operability for the following locations:

- Core makeup tank inlet highpoints
- Passive residual heat removal heat exchanger inlet high point
- IRWST injection line squib valve outlet high points

To ensure that all of the vent locations identified above function properly, notes are included on the system piping and instrumentation diagrams that specify layout sloping requirements. The intent of the layout requirements is to help ensure that the installed vents can effectively vent accumulated gases from the associated line segments. These notes also appear on the isometric drawings to make certain that the layout sloping requirements are observed during fabrication, construction, and installation.

The continuously monitored and alarmed pipe stub gas collection chamber, including Technical Specifications and Surveillance Requirements, was not utilized for the high points in the containment recirculation Line A because there are no credible postulated gas intrusion mechanisms by which gas is expected to migrate into these lines (except in the event of improper venting during line filling operations such as after maintenance). This isolated piping section is maintained in a standby condition prior to passive safety system actuation. This local high point is located between the containment recirculation squib valves and the IRWST injection line squib valves, and it remains connected to the IRWST so that the tank water elevation head maintains the pressure in this line prior to actuation.

Passive safety system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures to eliminate any identified gas accumulation. Locations equipped with pipe stub gas collection and alarm features will be continuously monitored via alarm indications in the main control room. Because the locations with pipe stub collection and alarm features are continuously monitored and have Surveillance Requirements and Required Actions, the potential exists that these locations will be vented at RCS pressure. Consequently, these locations have manual vent valves and are hard-piped to either the IRWST or the reactor coolant drain tank for potential venting at RCS pressure.

For the AP1000, the structures, systems, and components (SSCs) of the passive safety systems that are used to establish and maintain safe shutdown conditions for the plant are identified and discussed in subsections 7.4.1.1 and 7.4.2, and listed in Table 7.4-1. These

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

same SSCs that provide the AP1000 safe shutdown capability also provide the passive, safety-related accident mitigation functions, including those that are equivalent to the emergency core cooling system, decay heat removal, and containment spray system functions for active plants specified in the generic letter.

6.3.7 Instrumentation Requirements

Instrumentation channels employed for actuation of passive core cooling system operation are described in Section 7.3. This subsection describes the instrumentation provided for monitoring passive core cooling system components during normal plant operation and also during passive core cooling system post-accident operation. Alarms are annunciated in the main control room.

6.3.7.1 Pressure Indication

6.3.7.1.1 Accumulator Pressure

Two pressure channels are installed on each accumulator. The pressure indications are used to confirm that accumulator pressure is within bounds of the assumptions used in the safety analysis. Each channel provides pressure indication in the main control room and also provides high-pressure and low-pressure alarms.

6.3.7.1.2 Passive Residual Heat Removal Heat Exchanger Pressure

One pressure indicator is installed on the passive residual heat removal heat exchanger inlet line. The pressure indication is used to assist the operators in determining if there is a leak in the passive residual heat removal heat exchanger. The instrument provides pressure indication in the main control room.

6.3.7.2 Temperature Indication

6.3.7.2.1 Core Makeup Tank Inlet Line Temperature

Individual temperature channels are installed on the inlet line for each core makeup tank. The temperature indication is used to determine if there is a sufficient thermal gradient for system operation. Each channel provides temperature indication in the main control room and also provides a low-temperature alarm.

6.3.7.2.2 Passive Residual Heat Removal Heat Exchanger Inlet Temperature

One temperature channel is installed on the inlet line to the passive residual heat removal heat exchanger. The temperature indication is used to detect reactor coolant system leakage into the passive residual heat removal heat exchanger, either through the discharge valves or from tube leakage into the in-containment refueling water storage tank, and to identify the leakage path. The channel provides temperature indication in the main control room and also provides a high-temperature alarm.

6.3.7.2.3 In-Containment Refueling Water Storage Tank Temperature

Four temperature channels are installed on the in-containment refueling water storage tank. The temperature indications are used to confirm that in-containment refueling water storage

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

tank temperature is within the bounds of the assumptions used in the safety analysis. The temperature indications are also used to monitor in-containment refueling water storage tank temperature during passive core cooling system operation. Each channel provides temperature indication and high-temperature alarms in the main control room.

6.3.7.2.4 Core Makeup Tank Outlet Line Temperature

Two temperature channels are installed, one on each core makeup tank outlet line. The temperature indication is used to detect reactor coolant system leakage into the core makeup tanks. Each channel provides temperature indication in the main control room and also provides a high-temperature alarm.

6.3.7.2.5 Direct Vessel Injection Line Temperature

Two temperature channels are installed, one on each direct vessel injection line. The temperature indication is used to detect reactor coolant system leakage back through the direct vessel injection lines to the core makeup tanks, accumulator, or in-containment refueling water storage tank. Each channel provides temperature indication in the main control room.

6.3.7.2.6 Passive Residual Heat Removal Heat Exchanger Inlet High Point Temperature

One temperature channel is installed on the passive residual heat removal heat exchanger inlet line. The temperature indication is used to determine that the temperature in the inlet is within the bounds of the assumptions used in the safety analysis. The channel provides temperature indication and a low temperature alarm in the main control room.

6.3.7.3 Passive Residual Heat Removal Heat Exchanger Outlet Flow Indication

Two flow channels are installed on the passive residual heat removal outlet line. The flow indications are used to monitor and control passive residual heat removal heat exchanger operation. Each channel provides flow indication in the main control room.

6.3.7.4 Level Indication

6.3.7.4.1 Core Makeup Tank Level

Ten level channels are installed on each core makeup tank. There are 2 wide range level channels which are used to confirm that the core makeup tanks are maintained at full water level during normal operation. There are four narrow range level channels which are used to control the actuation of the automatic depressurization system stage 1 valves. There are four narrow range level channels which are used to control the actuation of the automatic depressurization system stage 4 valves. Each wide range channel provides level indication and alarms in the main control room. Each narrow range channel provides level indication and alarms in the main control room and actuation of the automatic depressurization system. Each set of two narrow range channels share upper and lower level tap connections with the core makeup tanks; a failure modes and effects analysis confirms the ability of this arrangement to tolerate single failures (Reference 2).

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.7.4.2 Accumulator Level

Two level channels are installed on each accumulator. The level indications are used to confirm that accumulator level is within bounds of the assumptions used in the safety analysis. Each channel provides level indication and both high and low level alarms in the main control room.

6.3.7.4.3 In-Containment Refueling Water Storage Tank Level

Six level channels are installed on the in-containment refueling water storage tank. There are two narrow range channels. These level indications are used to confirm that in-containment refueling water storage tank level is within the bounds of the assumptions used in the safety analysis. There are four wide range level channels. These level indications are used to provide containment recirculation valve repositioning. Each channel provides level indication in the main control room and provides level alarms.

The in-containment refueling water storage tank is sized and the level alarm setpoints selected to provide adequate in-containment refueling water storage tank injection (and spill flow to containment for a direct vessel injection line break) until containment floodup is sufficient to provide recirculation flow.

6.3.7.4.4 Containment Level

Three level channels are installed on the containment. The level indications are used to monitor containment level from the reactor vessel cavity up to the maximum containment floodup elevation. Each channel provides level indication and alarms in the main control room.

6.3.7.5 Containment Radiation Level

Four channels are installed for the containment radiation. The radiation indications are used to monitor containment conditions. Each channel provides radiation indication and high radiation alarms in the main control room. Section 11.5 provides additional information.

6.3.7.6 Valve Position Indication and Control

6.3.7.6.1 Valve Position Indication

Individual valve position is provided for the safety-related, remotely actuated valves listed in Table 6.3-1. In addition, valve position is provided for certain manually operated valves, as described in subsection 6.3.2.2.8.2, that can isolate redundant passive core cooling equipment, if mispositioned. The incontainment refueling water injection check valves and containment recirculation check valves have nonintrusive position indication.

For certain passive core cooling system valves with position indication, alarms in the main control room are provided to alert the operators to valve mispositioning. For the passive residual heat removal heat exchanger discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.7.6.2 Valve Position Control

Valve controls are provided for remotely operated passive core cooling system valves. Table 6.3-1 provides a list of the passive core cooling system remotely operated valves. These remotely operated valves have controls in the main control room.

6.3.7.6.2.1 Accumulator Motor-Operated Valve Controls

As part of the plant shutdown procedures, the operator is required to close the accumulator motor-operated valves. This prevents a loss of accumulator water inventory to the reactor coolant system when the reactor coolant system is depressurized. The valves are closed after the reactor coolant system has been depressurized to below the setpoint to block the safeguards actuation signal. The redundant pressure and level alarms on each accumulator function to alert the operator to close these valves, if any are inadvertently left open. Power is locked out after the valves are closed. During plant startup, the operator is directed by plant procedures to energize and open these valves prior to reaching the reactor coolant system pressure setpoint that unblocks the safeguards actuation signal. Redundant indication and alarms are available to alert the operator if a valve is inadvertently left closed once the reactor coolant system pressure increases beyond the setpoint. Power is also locked out after these valves are opened.

The accumulator isolation valves are not required to move during power operation. For a description of limiting conditions for operation and surveillance requirements of these valves, refer to the technical specifications. The accumulator isolation valves receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, technical specifications, and the redundant position indication and alarms, the valve controls are nonsafety-related.

6.3.7.6.2.2 In-Containment Refueling Water Storage Tank Injection Motor-Operated Valve Controls

The motor-operated valves in each in-containment refueling water storage tank injection line are normally open during all modes of normal plant operation. Power to these valves is locked out. Redundant valve position indication and alarms are provided to alert the operator if a valve is inadvertently closed. The technical specifications specify surveillances to show that these valves are open. These valves also receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, the redundant position indication and alarms and the technical specifications the valve controls are nonsafety-related.

6.3.7.6.2.3 Passive Residual Heat Removal Heat Exchanger Inlet Motor-Operated Valve Control

The motor-operated valve in the passive residual heat removal heat exchanger inlet line is normally open during normal plant operation. Power to this valve is locked out. Redundant valve position indications and alarms are provided to alert the operator if the valve is open. This valve also receives an actuation signal to confirm that it is open in the event of an accident.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.7.7 Automatic Depressurization System Actuation at 24 Hours

A timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This prevents discharging the Class 1E dc power sources such that they are no longer able to operate the automatic depressurization system valves. If power becomes available to the dc batteries and they are no longer discharging prior to activation of the timer, then the automatic depressurization system actuation would be delayed. If the plant does not need actuation of the automatic depressurization system based on having stable pressurizer level, full core makeup tanks, and high and stable in-containment refueling water storage tank levels, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the automatic depressurization system and allow for its actuation later should the plant conditions unexpectedly degrade.

6.3.8 Combined License Information

6.3.8.1 Containment Cleanliness Program

The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation to items that do not produce debris (physical or chemical), which could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. The cleanliness program shall limit the amount of latent debris and fibrous material located within the containment, as identified in subsection 6.3.2.2.7.1, item 12.

6.3.8.2 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA

The Combined License information requested in this subsection has been fully addressed in APP-GW-GLR-079 (Reference 3), and the applicable changes are incorporated into the DCD. The design of the recirculation screens is complete. Testing to assess the screen performance and downstream effects is complete. A study of the effects of screen design and performance on long-term cooling is complete. No additional work is required by the Combined License applicant to address the aspects of the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

> The Combined License applicants referencing the AP1000 will perform an evaluation consistent with Regulatory Guide 1.82, revision 3, and subsequently approved NRC guidance, to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in DCD subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation will consider resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris will be based on sample measurements from operating plants. The evaluation will also consider the potential for

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

the generation of chemical debris (precipitants). The potential to generate such debris will be determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

6.3.9 References

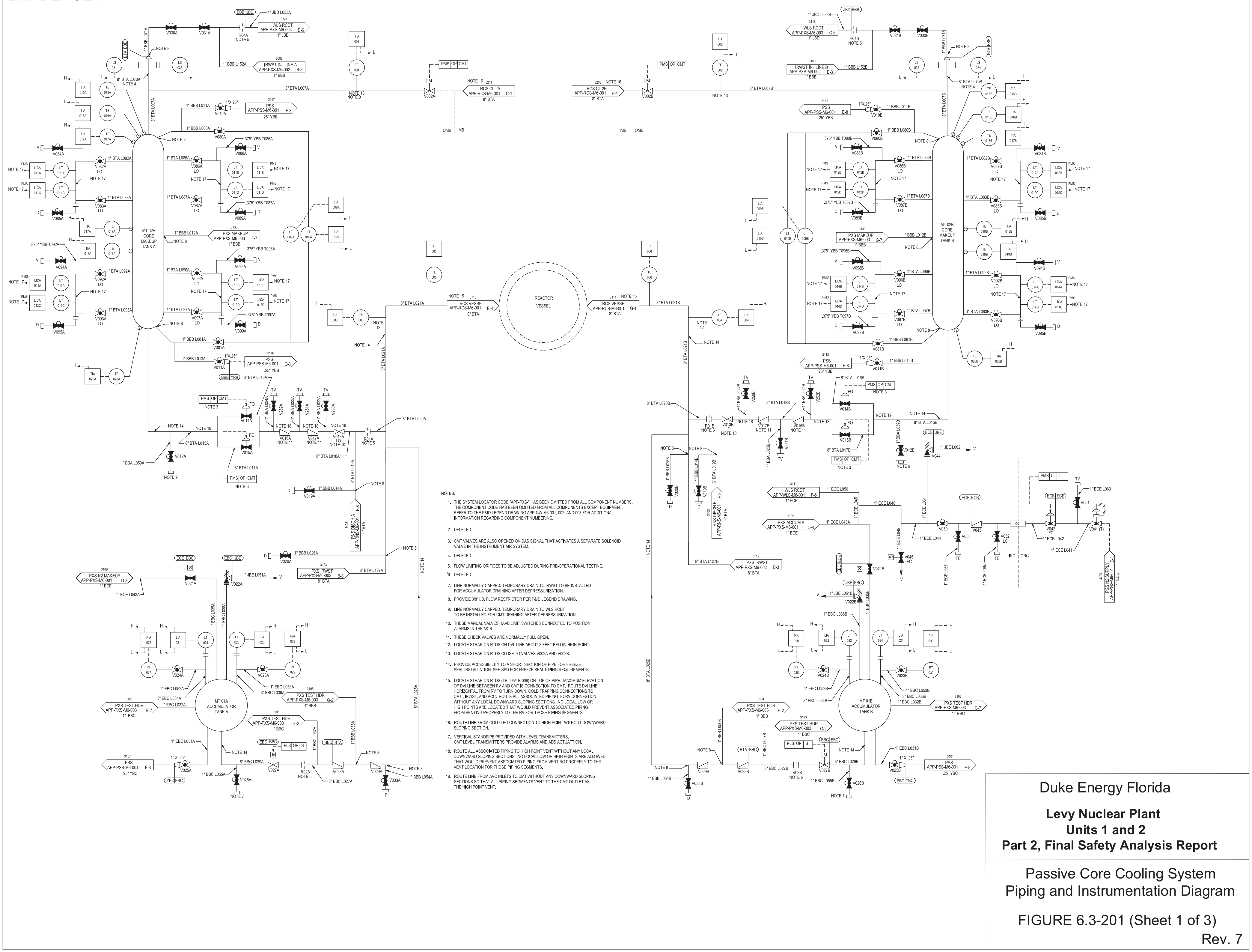
1. WCAP-8966, "Evaluation of Mispositioned ECCS Valves," September 1977.

2. WCAP-13594 (P), WCAP-13662 (NP), "FMEA of Advanced Passive Plant Protection System," Revision 1, June 1998.

3. APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Westinghouse Electric Company LLC.

4. APP-GW-GLN-147, "AP1000 Containment Recirculation and IRWST Screen Design," Westinghouse Electric Company LLC.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.



NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002, AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.
2. DELETED
3. CMT VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTIVATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
4. DELETED
5. FLOW LIMITING ORIFICES TO BE ADJUSTED DURING PRE-OPERATIONAL TESTING.
6. DELETED
7. LINE NORMALLY CAPPED. TEMPORARY DRAIN TO IRWST TO BE INSTALLED FOR ACCUMULATOR DRAINING AFTER DEPRESSURIZATION.
8. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER P&ID LEGEND DRAWING.
9. LINE NORMALLY CAPPED. TEMPORARY DRAIN TO WLS RCDT TO BE INSTALLED FOR CMT DRAINING AFTER DEPRESSURIZATION.
10. THESE MANUAL VALVES HAVE LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN THE MCR.
11. THESE CHECK VALVES ARE NORMALLY FULL OPEN.
12. LOCATE STRAP-ON RTDS ON DVI LINE ABOUT 3 FEET BELOW HIGH POINT.
13. LOCATE STRAP-ON RTDS CLOSE TO VALVES V002A AND V002B.
14. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE SSD FOR FREEZE SEAL PIPING REQUIREMENTS.
15. LOCATE STRAP-ON RTDS (TE-005/TE-006) ON TOP OF PIPE. MAXIMUM ELEVATION OF DVI LINE BETWEEN RV AND CMT IS CONNECTION TO CMT. ROUTE DVI LINE HORIZONTAL FROM RV TO TURN DOWN, COLD TRAPPING CONNECTIONS TO CMT , IRWST, AND ACC. ROUTE ALL ASSOCIATED PIPING TO RV CONNESTION WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE LOCATED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE RV FOR THOSE PIPING SEGMENTS.
16. ROUTE LINE FROM COLD LEG CONNECTION TO HIGH POINT WITHOUT DOWNWARD SLOPING SECTION.
17. VERTICAL STANDPIPE PROVIDED WITH LEVEL TRANSMITTERS. CMT LEVEL TRANSMITTERS PROVIDE ALARMS AND ADS ACTUATTION.
18. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATION FOR THOSE PIPING SEGMENTS.
19. ROUTE LINE FROM AVO INLETS TO CMT WITHOUT ANY DOWNWARD SLOPING SECTIONS SO THAT ALL PIPING SEGMENTS VENT TO THE CMT OUTLET AS THE HIGH POINT VENT.

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FIGURE 6.3-201 (Sheet 1 of 3)
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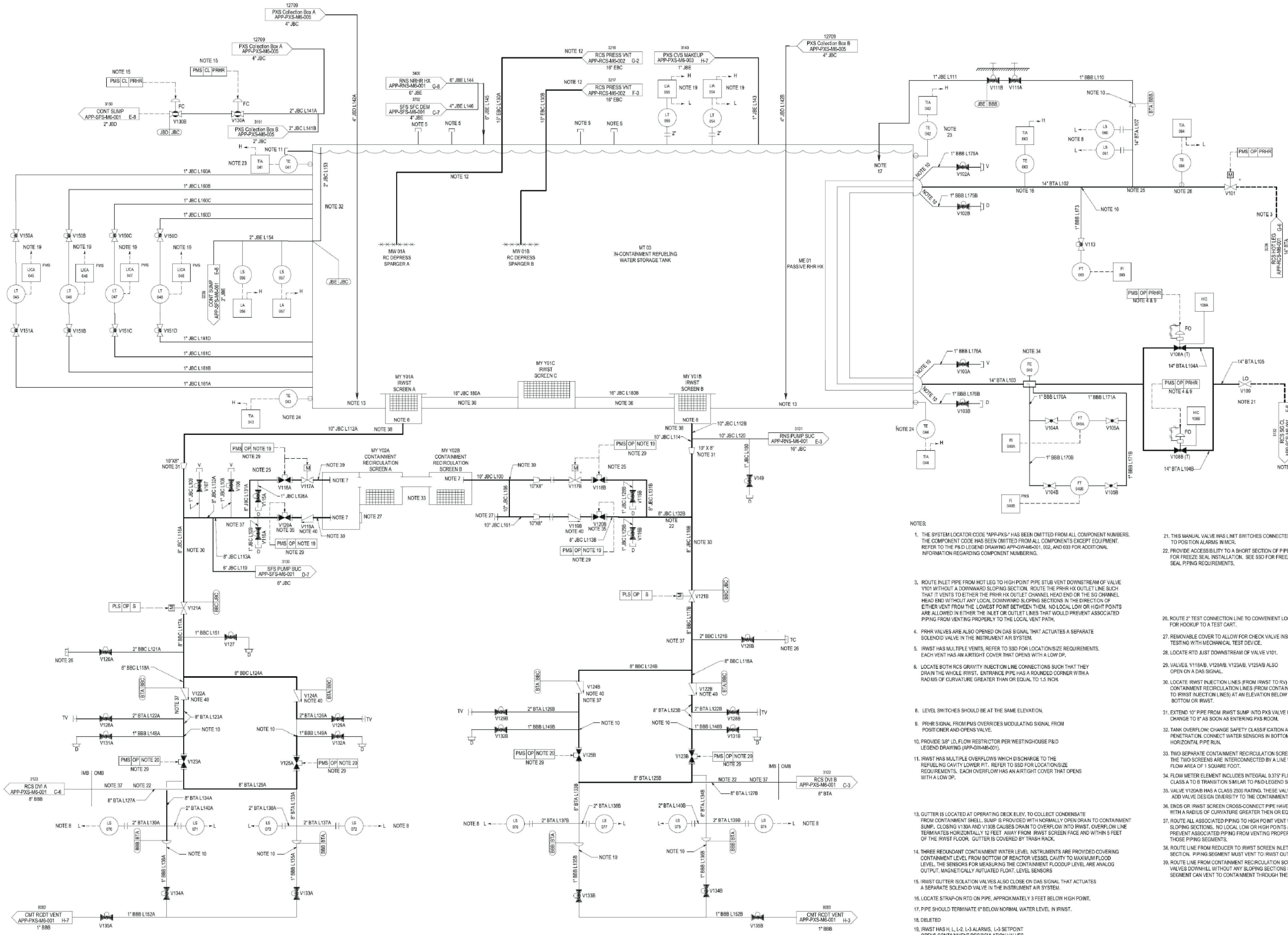

NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002, AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.

3. ROUTE INLET PIPE FROM HOT LEG TO HIGH POINT PIPE STUB VENT DOWNSTREAM OF VALVE V101 WITHOUT A DOWNWARD SLOPING SECTION. ROUTE THE PRHR HX OUTLET LINE SUCH THAT IT VENTS TO EITHER THE PRHR HX OUTLET CHANNEL HEAD END OR THE SG CHANNEL HEAD END WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS IN THE DIRECTION OF EITHER VENT FROM THE LOWEST POINT BETWEEN THEM. NO LOCAL LOW OR HIGHT POINTS ARE ALLOWED IN EITHER THE INLET OR OUTLET LINES THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE LOCAL VENT PATH.

4. PRHR VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

5. IRWST HAS MULTIPLE VENTS, REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH VENT HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

6. LOCATE BOTH RCS GRAVITY INJECTION LINE CONNECTIONS SUCH THAT THEY DRAIN THE WHOLE IRWST. ENTRANCE PIPE HAS A ROUNDED CORNER WITH A RADIUS OF CURVATURE GREATER THAN OR EQUAL TO 1.5 INCH.

8. LEVEL SWITCHES SHOULD BE AT THE SAME ELEVATION.

9. PRHR SIGNAL FROM PMS OVERRIDES MODULATING SIGNAL FROM POSITIONER AND OPENS VALVE.

10. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER WESTINGHOUSE P&ID LEGEND DRAWING (APP-GW-M6-001).

11. IRWST HAS MULTIPLE OVERFLOWS WHICH DISCHARGE TO THE REFUELING CAVITY LOWER PIT. REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH OVERFLOW HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

13. GUTTER IS LOCATED AT OPERATING DECK ELEV. TO COLLECT CONDENSATE FROM CONTAINMENT SHELL. SUMP IS PROVIDED WITH NORMALLY OPEN DRAIN TO CONTAINMENT SUMP. CLOSING V130A AND V130B CAUSES DRAIN TO OVERFLOW INTO IRWST. OVERFLOW LINE TERMINATES HORIZONTALLY 12 FEET AWAY FROM IRWST SCREEN FACE AND WITHIN 5 FEET OF THE IRWST FLOOR. GUTTER IS COVERED BY TRASH RACK.

14. THREE REDUNDANT CONTAINMENT WATER LEVEL INSTRUMENTS ARE PROVIDED COVERING CONTAINMENT LEVEL FROM BOTTOM OF REACTOR VESSEL CAVITY TO MAXIMUM FLOOD LEVEL. THE SENSORS FOR MEASURING THE CONTAINMENT FLOODUP LEVEL ARE ANALOG OUTPUT, MAGNETICALLY AUTUATED FLOAT, LEVEL SENSORS

15. IRWST GUTTER ISOLATION VALVES ALSO CLOSE ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

16. LOCATE STRAP-ON RTD ON PIPE, APPROXIMATELY 3 FEET BELOW HIGH POINT.

17. PIPE SHOULD TERMINATE 6" BELOW NORMAL WATER LEVEL IN IRWST.

18. DELETED

19. IRWST HAS H, L, L-2, L-3 ALARMS. L-3 SETPOINT OPENS CONTAINMENT RECIRCULATION VALVES.

20. ADS STAGE 4 ACTUATION SIGNAL OR A LOW HL LEVEL SIGNAL OPENS IRWST INJECTION VALVES.

21. THIS MANUAL VALVE HAS LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN MCR.

22. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE SSD FOR FREEZE SEAL PIPING REQUIREMENTS.

26. ROUTE 2" TEST CONNECTION LINE TO CONVENIENT LOCATION FOR HOOKUP TO A TEST CART.

27. REMOVABLE COVER TO ALLOW FOR CHECK VALVE INSERVICE TESTING WITH MECHANICAL TEST DEVICE.

28. LOCATE RTD JUST DOWNSTREAM OF VALVE V101.

29. VALVES, V118A/B, V120A/B, V123A/B, V125A/B ALSO OPEN ON A DAS SIGNAL.

30. LOCATE IRWST INJECTION LINES (FROM IRWST TO RV) AND CONTAINMENT RECIRCULATION LINES (FROM CONTAINMENT TO IRWST INJECTION LINES) AT AN ELEVATION BELOW THE BOTTOM OR IRWST.

31. EXTEND 10" PIPE FROM IRWST SUMP INTO PXS VALVE ROOM. CHANGE TO 8" AS SOON AS ENTERING PXS ROOM.

32. TANK OVERFLOW; CHANGE SAFETY CLASSIFICATION AT TANK PENETRATION. CONNECT WATER SENSORS IN BOTTOM OF HORIZONTAL PIPE RUN.

33. TWO SEPARATE CONTAINMENT RECIRCULATION SCREENS PROVIDED. THE TWO SCREENS ARE INTERCONNECTED BY A LINE WITH A MINIMUM FLOW AREA OF 1 SQUARE FOOT.

34. FLOW METER ELEMENT INCLUDES INTEGRAL 0.375" FLOW RESTRICTOR FOR CLASS A TO B TRANSITION SIMILAR TO P&ID LEGEND SHEET 2 NOTE 7.

35. VALVE V120A/B HAS A CLASS 2500 RATING. THESE VALVES ARE EMPLOYED TO ADD VALVE DESIGN DIVERSITY TO THE CONTAINMENT RECIRCULATION LINE.

36. ENDS OR IRWST SCREEN CROSS-CONNECT PIPE HAVE ROUNDED CORNERS WITH A RADIUS OF CURVATURE GREATER THEN OR EQUAL TO 2.25".

37. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATIONS FOR THOSE PIPING SEGMENTS.

38. ROUTE LINE FROM REDUCER TO IRWST SCREEN INLET WITHOUT A DOWNWARD SLOPING SECTION. PIPING SEGMENT MUST VENT TO IRWST OUTLET AS THE HIGH POINT.

39. ROUTE LINE FROM CONTAINMENT RECIRCULATION SCREEN TO PARALLEL SQUIB VALVES DOWNHILL WITHOUT ANY SLOPING SECTIONS SO THAT ALL PIPING SEGMENT CAN VENT TO CONTAINMENT THROUGH THE RECIRCULATION SCREENS.

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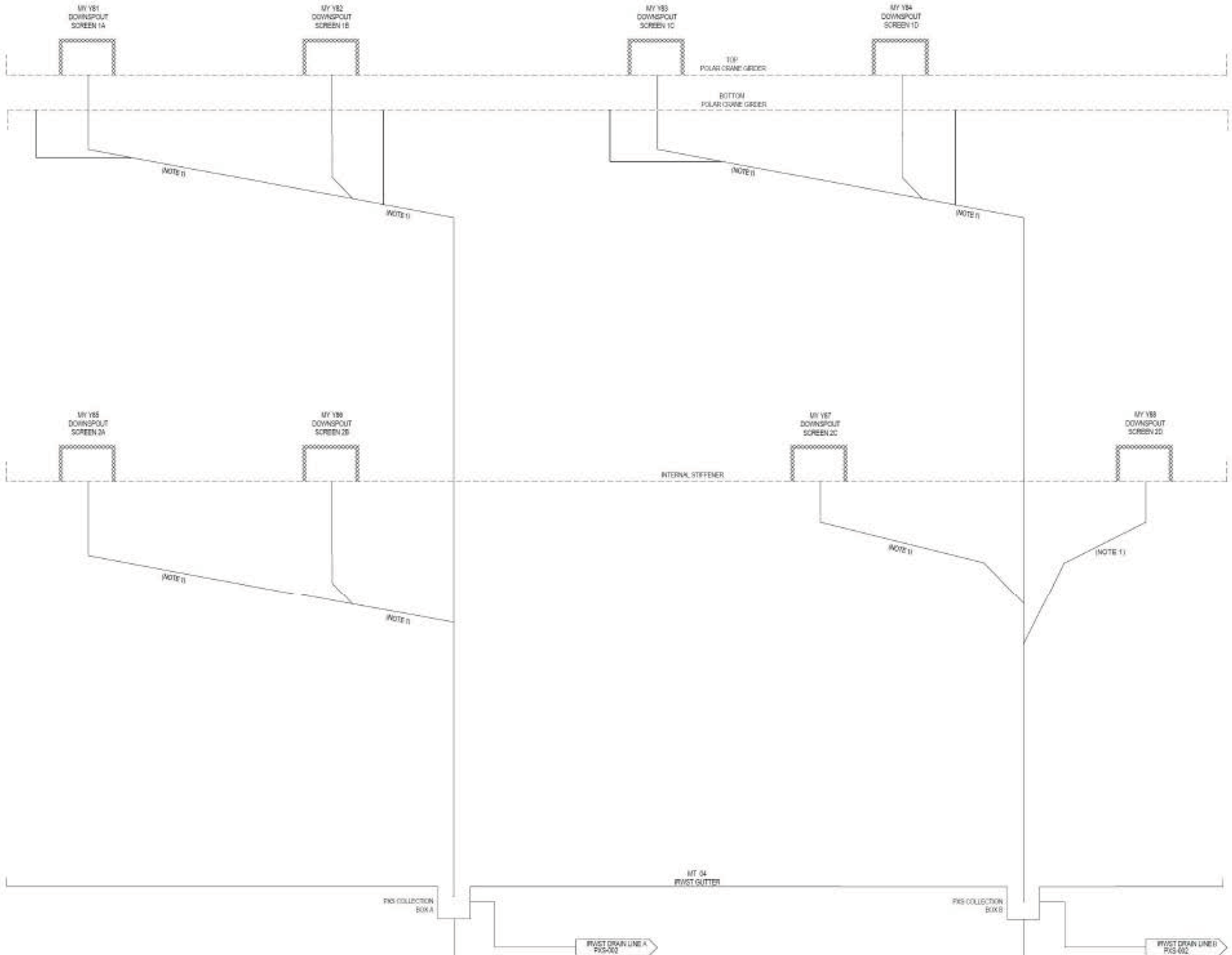

NOTES:
1. HORIZONTAL DOWNSPOUT SECTIONS MUST BE SLOPED DOWNWARD AT A MINIMUM OF 1/8" PER FOOT OR GREATER TO THE COLLECTION BOXES WITH NO LOCAL HIGHPOINTS.
2. PIPING AND COMPONENTS SHOWN ON THIS FIGURE ARE SAFETY CLASS C.

VENTS, DRAINS AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

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FIGURE 6.3-201 (Sheet 3 of 3)

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7.4 Systems Required for Safe Shutdown

Systems to establish safe shutdown conditions perform two basic functions. First, they provide the necessary reactivity control to maintain the core in a subcritical condition. Boration capability is provided to compensate for xenon decay and to maintain the required core shutdown margin. Second, these systems must provide residual heat removal capability to maintain adequate core cooling.

The designation of systems required for safe shutdown depends on identifying those systems that provide the following capabilities for maintaining a safe shutdown:

- Decay heat removal
- Reactor coolant system inventory control
- Reactor coolant system pressure control
- Reactivity control

There are two different safe shutdown conditions that are expected following a transient or accident condition. Short-term safe shutdown refers to the plant conditions from the start of an event until about 36 hours later. Long-term safe shutdown refers to the plant conditions after this 36-hour period.

The short-term safe shutdown conditions include maintaining the reactor subcritical, the reactor coolant average temperature less than or equal to no load temperature, and adequate coolant inventory and core cooling. These shutdown conditions shall be achieved following any of the design basis events using safety-related equipment. The specific safe shutdown condition achieved is a function of the particular accident sequence.

The long-term safe shutdown conditions are the same as the short-term conditions except that the coolant temperature shall be less than 420°F. This long-term condition must be achieved within 36 hours and ~~maintained indefinitely using safety-related equipment~~following a non-LOCA event using the PRHR HX as shown in Chapter 19E. These safe shutdown conditions can be maintained by the PRHR HX for at least 14 days based on a realistic analysis that only credits using safety-related equipment. In addition, these safe shutdown conditions can be maintained indefinitely using the ADS and passive injection / recirculation as discussed in 7.4.1.1. Also refer to Chapter 6.3.1.1 for additional discussion on safe shutdown requirements.

There are no systems specifically and solely dedicated as safe shutdown systems. However, there are a number of plant systems that are available to establish and maintain safe shutdown conditions. Normally, in the event of a turbine or reactor trip, nonsafety-related plant systems automatically function to place the plant in short-term safe shutdown, as described in subsection 7.4.1.2. During the short-term safe shutdown condition, an adequate heat sink is provided to remove reactor core residual heat and boration control is available. Redundancy of systems and components is provided to enable continued maintenance of the short-term safe shutdown condition. Additional redundant nonsafety-related systems are normally available to manually perform a plant depressurization and cooldown.

The engineered safety systems are designed to establish and maintain safe shutdown conditions for the plant. Nonsafety-related systems are not required for safe shutdown of the plant.

This section focuses on safety-related systems used to establish and maintain safe shutdown conditions. The discussion of safe shutdown does not include accident response and/or mitigation since the standard review plan for this section addresses safe shutdown not related to accident mitigation. However, safe shutdown conditions are also established and maintained by these safety-related systems following accident conditions. For example, the control rods are released to initially place the plant in a shutdown condition to mitigate the consequences of various accidents. The passive core cooling system, on the other hand, is used to provide core cooling in an accident, but it is also one of the principal systems used for safe shutdown. Only those specific engineered safety systems listed in Table 7.4-1 are used to establish and maintain safe shutdown of the plant. These engineered safety systems automatically function to place the plant in a safe shutdown condition without operator action.

The instrumentation functions necessary for safe shutdown are available through instrumentation channels associated with the safety-related systems in the primary plant. These channels automatically actuate the protective functions provided by the safety-related systems. Manual actuation of the associated safety-related systems is also provided.

The instrumentation systems discussed in this section are those which are required during nonaccident conditions to align the safety-related systems and perform the specified safe shutdown functions.

The specific systems available for safe shutdown are discussed in subsection 7.4.2 and are listed in Table 7.4-1.

Maintenance of safe shutdown conditions with these systems, and the associated instrumentation and controls, includes consideration of the accident consequences that might challenge safe shutdown conditions. The accident consequences that are germane are those that tend to degrade the capabilities for coolant circulation, boration, heat removal, and depressurization. Safe shutdown is achieved following any of the accidents analyzed in Chapter 15. The specific safe shutdown condition reached is a function of the particular accident sequence.

The instrumentation and controls discussed in subsection 7.4.1 are used to control and/or monitor shutdown. These safety-related systems allow the maintenance of safe shutdown, even under accident conditions that tend toward a return to criticality or a loss of heat sink.

In addition to the operation of safety-related systems used for safe shutdown, as described in subsection 7.4.1, the following are part of the safe shutdown provisions:

- The turbine is tripped. (This can be accomplished at the turbine as well as from the main control room.)

- The reactor is tripped. (This can be accomplished at the reactor trip switchgear as well as from the main control room.)

- Support of engineered safety systems actuation is provided by safety-related onsite dc power.

7.4.1 Safe Shutdown

7.4.1.1 Safe Shutdown Using Safety-Related Systems

The following describes the process that establishes safe shutdown conditions for the plant, using the safety-related systems, and no operator action. The reactor coolant system is assumed to be intact for this discussion of safe shutdown.

Since this discussion only considers the use of safety-related systems, offsite electrical power sources are assumed to be lost at the start of the event. This results in a loss of the reactor coolant pumps. Even though the reactor coolant pumps are tripped during the initiation of certain engineered safety system actuation, it is assumed that no engineered safety system actuation signal is generated for this initiating event. With loss of the reactor coolant pumps, reactor coolant system natural circulation flow initiates and transfers core heat to the steam generators. Since feedwater flow is lost, the existing steam generator water inventory provides initial decay heat removal capability.

The initial loss of main ac power results in the Class 1E dc batteries automatically supplying power to the Class 1E dc power distribution network and the four Class 1E 120 Vac instrumentation divisions via the inverters.

The initial response of the passive safety systems is to actuate the passive residual heat removal heat exchanger due to low steam generator water level. The passive residual heat removal heat exchanger removes decay heat from the core by transferring this heat to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger removes core decay heat, cooling the reactor coolant system. As reactor coolant system cooldown continues, the reactor coolant system pressure decreases due to contraction of the reactor coolant system inventory since the pressurizer heaters are de-energized. An engineered safety system actuation signal occurs when reactor coolant system pressure decreases below a setpoint. This actuates the core makeup tanks, if they had not been previously actuated due to low pressurizer level. The core makeup tanks provide borated water injection to the reactor coolant system.

The engineered safety system actuation signal generated on low pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits ~~indefinite~~ extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

The in-containment refueling water storage tank starts to boil about one to two hours after passive residual heat removal operation is initiated. Once boiling occurs, the in-containment refueling water storage tank begins steaming to containment, transferring heat to the air flowing on the outside of the containment shell. As steaming to containment continues, containment pressure slowly increases. As containment pressure slowly increases, an engineered safety system actuation signal is generated on containment high pressure, resulting in the initiation of passive containment cooling. This provides water flow on the outside of the containment shell to improve the heat removal performance from containment through evaporative cooling to the outside air.

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste

sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system ~~indefinitely~~ provides core decay heat removal in this configuration with~~out a significant~~ a limited increase in the containment water level.

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

Operation in this configuration may be limited in time duration by reactor coolant system leakage. The core makeup tanks can only supply a limited amount of makeup in the event there is reactor coolant system leakage. Eventually the volume of the water in the core makeup tanks will decrease to the first stage automatic depressurization setpoint. The time to reach this setpoint depends upon the reactor coolant system leak rate and the reactor coolant cooldown.

The Class 1E dc batteries that power the automatic depressurization system valves provide power for at least 24 hours. There is a timer that measures the time that ac power sources are unavailable. This timer provides for automatic actuation of the automatic depressurization system before the Class 1E dc batteries are discharged. The emergency response guidelines direct the operator to assess the need for automatic depressurization before the timer completes its count (approximately 22 hours). The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, a high and stable pressurizer level, and decreasing or stable reactor coolant system temperature.~~and a high and stable in-containment refueling water storage tank level.~~ If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

The automatic depressurization system can be manually initiated by the operator at any time, but no operator action is needed to provide safe shutdown conditions. Once the automatic depressurization system sequence initiates, the plant automatically transitions to lower pressure and temperature conditions that establish and maintain long-term safe shutdown of the plant.

When the automatic depressurization system is actuated, the first stage depressurization valves open and the reactor coolant system depressurization starts. The second and third stage depressurization valves open in sequence, based on automatic timers that are started upon the actuation of the first stage depressurization valves. As reactor coolant inventory continues to be lost, the core makeup tanks continue to inject. If the volume of the water in the core makeup tanks decrease to the fourth stage automatic depressurization setpoint, the fourth stage depressurization valves open. The water and steam vented from the reactor coolant system initially flows into the in-containment refueling water storage tank and overflows into the refueling canal. Eventually this overflows into the reactor vessel cavity, where any moisture from the fourth stage automatic depressurization system valves also collects from discharge in the loop compartments. This overflow initiates the floodup of containment, along with condensate from the containment shell and other cool surfaces in containment.

As the reactor coolant system pressure decreases, the accumulators inject borated water into the reactor coolant system. After the fourth stage automatic depressurization system valves open, the reactor coolant system pressure is reduced sufficiently so that in-containment refueling water storage tank injection can begin as the core makeup tanks empty.

The drain down of the in-containment refueling water storage tank is relatively slow, depending on the injection rates and the reactor coolant system pressure. As the in-containment refueling water storage tank continues to inject, the containment floodup also continues and eventually the floodup volume is sufficient to initiate flow from the recirculation sump.

As the reactor coolant system voids during the cooldown and depressurization process, water flow through the passive residual heat removal heat exchanger is replaced by steam flow, which also provides core cooling. As the in-containment refueling water storage tank empties and uncovers the passive residual heat removal heat exchanger, heat transfer via this path decreases. Eventually, the passive residual heat removal heat exchanger is uncovered, heat removal by the passive residual heat removal heat exchanger stops, and decay heat is removed by automatic depressurization system venting.

The final long-term safe shutdown plant conditions are maintained with the reactor coolant system depressurized to about 10 psig at saturated conditions, venting steam through the automatic depressurization system valves to containment, with heat transferred to the outside atmosphere via the passive containment cooling system. With containment isolation established, the water inventory inside containment provides an indefinite cooling water supply for core decay heat removal.

7.4.1.2 Safe Shutdown Using Safety-Related and Nonsafety-Related Systems

This subsection describes situations where nonsafety-related features of the plant are used together with safety-related systems to establish safe shutdown conditions. As discussed in subsection 7.4.1.1, the AP1000 can be placed in a safe shutdown condition and maintained there using safety-related systems and no operator actions. Section 6.3 provides additional discussion of these situations.

Following passive residual heat removal heat exchanger actuation, the in-containment refueling water storage tank heats up and starts to boil after several hours of operation. If normal steam generator heat removal is not re-established, the operators align the normal residual heat removal system to cool the in-containment refueling water storage tank. This operation prevents significant steaming to the containment.

In case the automatic depresurization system is actuated, the operators align the normal residual heat removal system to provide injection to the reactor coolant system. This action causes the core makeup tank level to remain above the fourth stage valve actuation setpoint and prevents significant steaming to and flooding of the containment.

7.4.1.3 Safe Shutdown Using Nonsafety-Related Systems

This subsection describes the process to establish and maintain safe shutdown conditions using the nonsafety-related systems. As discussed in Section 7.4, the review of the plant safe shutdown capability, including the capabilities provided by the nonsafety-related systems, does not include accident response or mitigation. The nonsafety-related systems normally used to support plant shutdown operations are expected to be available. Offsite power is also expected to be available to support safe shutdown operations, although the nonsafety-related systems can establish and maintain safe shutdown conditions using only onsite electrical power.

For the purposes of this discussion, the nonsafety-related system operation following a reactor trip is described. As assumed in the discussion in subsection 7.4.1.1 on safe shutdown using safety-related systems, the reactor coolant system is assumed to be intact during plant safe shutdown operations.

The nonsafety-related systems and equipment used to establish and maintain safe shutdown conditions are the same systems and equipment that are operated during normal plant startup and shutdown evolutions. The safe shutdown capability using the safety-related systems, described in subsection 7.4.1.1, is only expected to be used in the event that the nonsafety-related systems are not available.

The nonsafety-related systems operate to establish and maintain safe shutdown conditions by providing the safe shutdown functions described in Section 7.4, except that reactivity control is only needed for long-term safe shutdown. If offsite power is available, the operation of these nonsafety-related systems is automatic.

The nonsafety-related systems actuate to establish and maintain the short-term safe shutdown conditions. The systems can also establish and maintain long-term safe shutdown conditions within the time limits discussed in Section 7.4. The operational philosophy following any event is to maintain appropriate safe shutdown conditions based on the duration of the shutdown, until the plant is able to re-start.

Cold shutdown conditions would only be established if it becomes necessary for equipment repair or due to limitations of the nonsafety-related systems in maintaining safe shutdown conditions (such as feedwater system water inventory). This philosophy reduces unnecessary challenges to plant safety due to the transition from operating systems to infrequently-operated standby systems.

Normally, offsite electrical power is available and the nonsafety-related systems automatically maintain short-term safe shutdown conditions as follows:

- Reactor coolant system forced flow to the steam generators by the reactor coolant pumps
- Feedwater from the main or startup feedwater systems
- Heat removal by the steam generators to the main condenser using turbine bypass valves
- Condenser heat removal provided by the main circulating water system

- Reactor coolant system inventory and boration control by the chemical and volume control system

- Reactor coolant system pressure control using pressurizer heaters and normal spray

If offsite power is not available, the reactor coolant pumps, main feedwater pumps, and main circulating water pumps will not be operating. However, the nonsafety-related systems maintain short-term safe shutdown conditions without offsite electrical power as follows:

- Electrical power provided to the required nonsafety-related systems by the diesel-generators of the onsite standby power system

- Heat removal by the steam generators directly to the atmosphere through the power-operated relief valves

- Feedwater from the startup feedwater system

- Reactor coolant system flow to the steam generators via natural circulation

- Reactor coolant system inventory and boration control by the chemical and volume control system

- Reactor coolant system pressure control using pressurizer heaters and auxiliary spray

In case the main feedwater is unavailable, the initial response of the nonsafety-related systems following a reactor trip is to automatically actuate the startup feedwater system, on low steam generator water level, to provide decay heat removal. The steam generators can remove decay heat from the core by either forced or natural circulation in the reactor coolant system. If offsite electrical power is available, the reactor coolant pumps continue to provide forced circulation in the reactor coolant system and the circulating water system continues to operate to provide a heat sink for the steam discharged from the steam generators to the main condenser.

With offsite power and the main condenser available, the turbine bypass valves automatically actuate after the reactor trip to control reactor coolant system temperature, based on the pre-set steam generator pressure control set point that is normally established for standby turbine bypass valve operation. The main feedwater system or the startup feedwater system automatically maintains steam generator water level as the turbine bypass valves continue to throttle steam flow to match the decreasing core decay heat levels. The pressurizer heaters and spray automatically maintain reactor coolant system subcooling with pressure at normal reactor coolant system conditions.

The chemical and volume control system makeup pumps automatically actuate as required to provide borated makeup water to maintain pressurizer level in the programmed band for no-load conditions. The makeup source is the boric acid tank which provides long-term reactivity control. The makeup pumps are expected to operate infrequently during these conditions to compensate for normal reactor coolant system inventory losses such as valve leakage.

Operation of the nonsafety-related systems in this mode maintains short-term safe shutdown conditions and reactor coolant system temperature and pressure remain near no-load

conditions. If it becomes necessary to perform a plant cooldown and depressurization to establish long-term safe shutdown conditions, the nonsafety-related systems are used, following the normal plant cooldown procedures. Manual boration to the cold shutdown boron concentration is provided by the chemical and volume control system by initiating reactor coolant system letdown in combination with makeup pump operation. After the boration is completed and letdown is secured, the makeup pumps automatically maintain reactor coolant system inventory throughout the remainder of the cooldown process.

After the required boration is completed the turbine bypass valves are used to initiate the cooldown, with manual control of pressurizer heaters and spray to maintain the reactor coolant system pressure, temperature, and cooldown rate within the limits specified in the technical specifications. The main feedwater system automatically provides feedwater and maintains steam generator level throughout the cooldown process.

When the reactor coolant system temperature and pressure are reduced to within the capabilities of the normal residual heat removal system, at approximately 350°F and 400 psig, the system is manually aligned to the reactor coolant system and started to continue the cooldown process. The final long-term safe shutdown conditions established would be dependent upon the specific maintenance required.

The use of the nonsafety-related systems and equipment for both short-term and long-term safe shutdown also requires the operation of associated support systems. These normally operating support systems include component cooling water, chilled water, compressed air, area ventilation, and nonsafety-related instrumentation and control power. These systems are started as required following a loss of offsite power, once the nonsafety-related diesel-generators are started.

If offsite electrical power is unavailable, the nonsafety-related systems actuate to establish and maintain safe shutdown conditions. There are some differences in the decay heat discharge flow path and the reactor coolant system remains at a slightly higher temperature resulting from the natural circulation flow conditions. With the loss of offsite electrical power, the nonsafety-related diesel-generators provide electrical power for the required nonsafety-related equipment. However, the reactor coolant pumps, main feedwater pumps, and main circulating water pumps are not available. Therefore, core decay heat is transferred to the steam generators using natural circulation in the reactor coolant system, the startup feedwater pumps supply the steam generators, and the steam generators discharge directly to the atmosphere to remove decay heat.

When offsite electrical power is unavailable, reactor coolant temperature is automatically maintained by the steam generator atmospheric power-operated relief valves instead of the turbine bypass valves. The steam generator power-operated relief valves maintain a pre-set steam generator pressure by throttling the steam discharged directly from the steam generators to the atmosphere. The relief valve operation maintains a slightly higher steam generator pressure than the pressure maintained with turbine bypass valve standby operation, resulting in a slight increase in the reactor coolant system temperature. The automatic operation of the startup feedwater subsystem maintains steam generator inventory with the pumps powered from the diesel-generators. In addition, the direct discharge of steam to the atmosphere prevents condensate recovery, which limits the water inventory for the startup feedwater system.

Following a loss of offsite power, the reactor coolant system temperature is slightly higher than for a reactor trip when offsite electrical power is available, resulting from natural circulation flow and steam generator power-operated relief valve operation. Since the transition to natural circulation flow is relatively slow, the reactor coolant system pressure remains stable without operator action. Operator action is not required to maintain reactor coolant system pressure.

Without offsite electrical power, the pressurizer heaters are manually re-energized after the diesel-generators start. Without reactor coolant pump operation, normal pressurizer spray is unavailable to counteract system pressure increases. Therefore, auxiliary spray provided by the chemical and volume control system makeup pumps is manually initiated to decrease reactor coolant system pressure, if necessary. The operation of the chemical and volume control system makeup pumps to maintain reactor coolant system inventory is similar to their operation when offsite power is available, except that the pumps are manually controlled and powered from the diesel-generators.

The nonsafety-related systems are normally expected to maintain short-term safe shutdown conditions when offsite power is not available. If it is required to establish long-term safe shutdown conditions for equipment maintenance, the cooldown would normally be delayed until offsite power is recovered.

However, the nonsafety-related systems can be used to perform a natural circulation cooldown, if necessary. When performing a natural circulation plant cooldown and depressurization, the operation of the nonsafety-related systems is similar to the normal cooldown operation except that they are powered from the diesel-generators. The primary difference in operation is the use of the steam generator power-operated relief valves to control the cooldown process.

7.4.2 Safe Shutdown Systems

To effect a safe shutdown, with safety-related systems, the plant is initially brought to a stable condition with heat removal provided by the passive residual heat removal heat exchanger. For safe shutdown conditions, control is possible from either the main control room or the remote shutdown workstation. To accomplish a safe shutdown, the functions required are: coolant circulation, boration, heat removal, and depressurization. The portions of the protection and safety monitoring system required to achieve the safe shutdown condition are described in Sections 7.2 and 7.3. The minimum systems required to maintain safe shutdown conditions under a nonaccident condition are listed and discussed in the following paragraphs.

7.4.2.1 Passive Core Cooling System

A description of the passive core cooling system and its operation is provided in Section 6.3. The passive residual heat removal heat exchanger, the core makeup tanks, the in-containment refueling water storage tank, the containment recirculation, and the automatic depressurization system actuate automatically. They can also be manually initiated. Actuation controls are located at the remote shutdown workstation as well as in the main control room.

The safety injection flow from the accumulators, initiates automatically by the reactor coolant system depressurization process. The operation of the accumulator is integrated with the automatic actuation of the other passive core cooling subsystems.

7.4.2.2 Passive Containment Cooling System

A description of the passive containment cooling system and its operation is provided in subsection 6.2.2. The passive containment cooling system actuates automatically. It also can be manually initiated. Actuation controls are located at the remote shutdown workstation as well as in the main control room.

7.4.2.3 Containment Isolation

A description of containment isolation valves and their operation is provided in various subsections. Each system that has piping that penetrates the containment vessel and therefore, requires containment isolation valves is discussed in its own subsection. Most of these systems are nonsafety-related; however, the containment isolation valves and the associated piping are safety-related and automatically close on a safeguards actuation (S) signal. The containment isolation system is discussed in subsection 6.2.3.

7.4.2.4 Reactor Coolant System Circulation

The preferred method of coolant circulation is forced circulation with the reactor coolant pumps supplying the driving head. Upon the loss of main ac power, or when the reactor coolant pumps are tripped during engineered safety system actuation, the reactor coolant pumps are not available. However, the reactor coolant system is designed to provide sufficient natural circulation to achieve safe shutdown conditions with the steam generators and passive residual heat removal heat exchanger removing decay heat. Natural circulation flow is verified by monitoring the reactor coolant system temperatures.

7.4.2.5 Other Systems Required for Safe Shutdown

The other safety-related equipment and systems used to maintain the plant in safe shutdown are identified in Table 7.4-1. They are also listed below, with a reference to the respective section or subsection which discusses their operation in more detail:

- Protection and safety monitoring system — Sections 7.2, 7.3, and 7.5
- Class 1E dc and UPS system — Subsection 8.3.2

These systems are either normally operating or they start automatically when required. The instrumentation for these systems is described in the particular section containing the system description.

The monitoring instrumentation available in the main control room for safe shutdown is safety-related and is part of the protection and safety monitoring system. The instrumentation available for safe shutdown monitoring is listed in Section 7.5.

LNP DEP 6.3-1

Table 9.5.1-201
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp(1)	Remarks
Safe Shutdown Capability			
72. Fire damage should be limited so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the main control room or emergency control station is free of fire damage.	C.5.b(1)	C	
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for at least 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.
74. Separation requirements for verifying that one train of systems necessary to achieve and maintain hot shutdown is free of fire damage.	C.5.b (2)	C	

Table 14.3-202 (Sheet 1 of 2)
Design Basis Accident Analysis

LNP DEP 3.2-1

Reference	Design Feature	Value
DCD Subsection 6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4
DCD Subsection 6.3.6.1.3	The pH baskets are located below plant elevation 107' 2".	
DCD Figure 6.3-1	The passive core cooling system has two direct vessel injection lines.	
DCD Table 6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2500
DCD Table 6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2000
DCD Table 6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,900
DCD Subsection 6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 108 feet.	73,500
DCD Table 6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274
DCD Table 6.3-2	The passive core cooling system has two pH adjustment baskets each with a minimum required volume (ft^3).	280
DCD Subsection 14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) - With 520°F hot leg and 80°F IRWST - With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08
DCD Subsection 6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3
DCD Figure 6.3-1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap centerlines are located below the centerline of the upper level tap connection to the CMTs (in).	1" ± 1"
DCD Figure 6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
DCD Figure 6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.	
DCD Figure 6.3-1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

Table 14.3-202 (Sheet 2 of 2)
Design Basis Accident Analysis

	Reference	Design Feature	Value
LNP DEP 3.2-1	DCD Figure 6.3-1	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
	DCD Figure 6.3-1	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
	DCD Subsection 7.1.2.10	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	
	DCD Subsection 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
	DCD Subsection 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
	DCD Subsection 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
	DCD Subsection 7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
	DCD Subsection 7.3.1.2.2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
	DCD Subsection 7.3.1.2.3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
	DCD Subsection 7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a low-2 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

15.0.13 Operator Actions

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to ~~the~~ a safe, stable ~~shutdown~~ condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in subsection 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- Upon reaching one of the trip setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal.
- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below occurs:

- Plant vital instruments are supplied from the Class 1E and uninterruptable power supply.
- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The onsite standby power system, if available, supplies ac power to the selected plant non-safety loads.
- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.
- If startup feedwater is not available, the PRHR heat exchanger is actuated.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, ~~keeps the reactor coolant subcooled indefinitely~~provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria. After the IRWST water reaches saturation (in about two and half hours), steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine trip initiated decrease in secondary heat removal without loss of ac power, which is discussed in subsection 15.2.3. A loss of offsite power to the plant auxiliaries will also result in a loss of normal feedwater.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in subsection 15.0.8 and listed in Table 15.0-6.

LNP DEP 6.3-1

Table 19.59-202
AP1000 PRA-Based Insights

Insight	Disposition
1e. (cont.)	
Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1

19E.4.10.2 Shutdown Temperature Evaluation

~~In SECY-94-084, Item C, Safe Shutdown (Reference 14), the NRC staff recommended the Commission's approval of 420°F or below, rather than cold shutdown condition as a safe stable condition, which the PRHR HX must be capable of achieving and maintaining following non-LOCA events, predicated on acceptable passive safety system performance and an acceptable resolution of the regulatory treatment of nonsafety systems (RTNSS) issue. The NRC requested a safety analysis to~~ As discussed in Subsection 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a ~~stable~~ safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

As discussed in subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the specified safe shutdown condition following a~~n~~ non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to ~~a stable safe~~this condition following postulated transients. A non-bounding, conservative analysis is represented in ~~The results of this analysis are presented in~~ Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in Subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to ensure the results were conservative for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time- dependent condensate return rate was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip. The PRHR ~~heat exchanger~~HX is actuated on the low steam generator narrow range level coincident with low startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on ~~Low~~ low cold leg temperature and the CMTs are actuated.

Once actuated, at about ~~600~~ 2,700 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold ~~flow rate~~water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about ~~3,500~~6000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about ~~31,000~~46,700 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1 the cold leg temperature in the loop with the PRHR is reduced to 420°F ~~at 82,600~~within 52,900 seconds, while the core average temperature reaches 420°F ~~in 123,600~~within 120,900 seconds (approximately 34 hours).

As discussed in subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de- energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time.~~this mode of operation can last for up to 72 hours. However, in about 22 hours after the event, if no ac power is available, or if condensate return is not available, then the operator is instructed to actuate the ADS.~~ Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

19E.9 References

1. Letter, Westinghouse to NRC, DCP/NRC1385, AP600 Emergency Response Guidelines.

2. NUREG-1449, "Shutdown and Low Power Operations at Commercial Nuclear Power Plants in the United States," September 1993.

3. NRC Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," July 24, 1992.

4. Letter, Westinghouse to NRC, DCP/NRC0124, APWR-0452, "AP600 Vortex Mitigator Development Test for RCS Mid-loop Operation," July 6, 1994.

5. NUREG-0897, Rev. 1, "Containment Emergency Sump Performance," October 1985.

6. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants."

7. NRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3, February 1976.

8. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1988 with 1989 Addenda.

9. Lewis, R. N., Huang, P., Behnke, D. H., Fittante, R. L., and Gelman, A., WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Non-Proprietary), "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987.

10. WCAP-14171, Revision 2 (Proprietary) and WCAP-14172, Revision 2 (Non-Proprietary), "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," March 1998.

11. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Model."

12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 1, July 1981.

13. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

14. ~~SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.~~Not used.

15. Title 10, Code of Federal Regulations, Part 50, (10 CFR 50.46).

16. NRC letter, SECY-93-190, "Regulatory Approach to Shutdown and Low-Power Operations," July 12, 1993.

17. Title 10, Code of Federal Regulations, Part 50.36, "Technical Specifications."

18. NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," April 1995.

LNP DEP 3.2-1

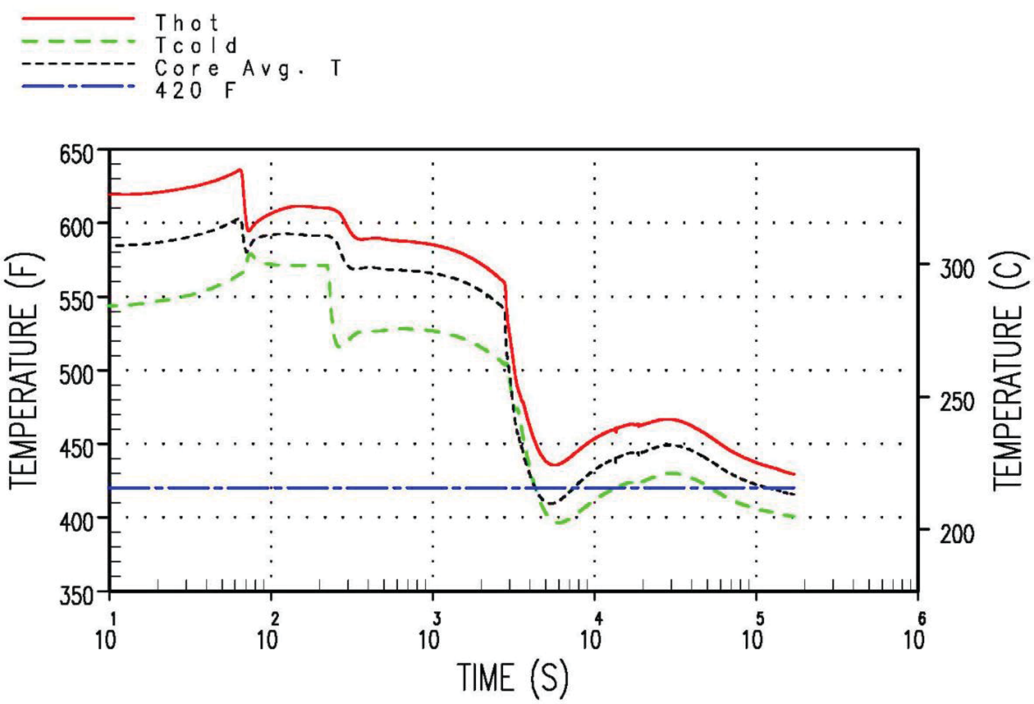
Table 19E.4.10-201

SEQUENCE OF EVENTS FOLLOWING A LOSS OF AC POWER FLOW WITH CONDENSATE FROM THE CONTAINMENT SHELL BEING RETURNED TO THE IRWST

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	60.6
Rods Begin to Drop	62.6
Low Steam Generator Water Level (Wide-Range) Reached	209.5
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	221.5
Low T_{cold} Setpoint Reached	2,752
Steam Line Isolation on Low T_{cold} Signal	2,764
CMTs Actuated on Low T_{cold} Signal	2,764
IRWST Reaches Saturation Temperature	15,900
Heat Extracted by PRHR HX Matches Core Decay Heat	46,700
Cold Leg Temperature Reaches 420°F (loop with PRHR)	52,900
Core Average Temperature Reaches 420°F	120,900

Event CMTs Stop Circulating has been deleted

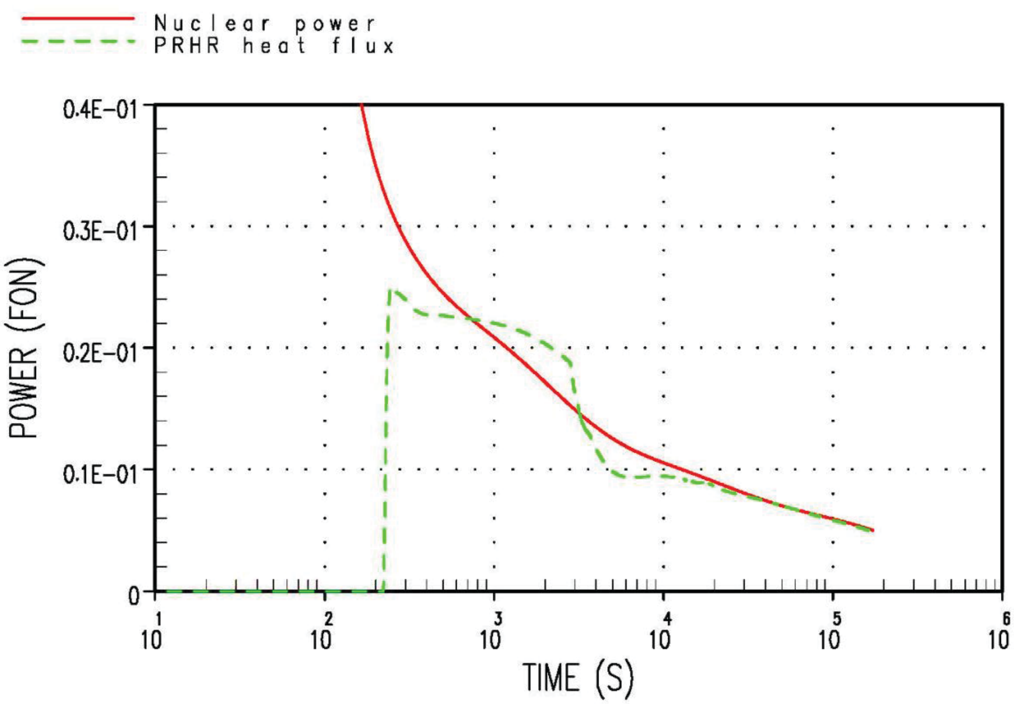
deleted (loop with PRHR)



Duke Energy Florida
Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

Shutdown Temperature Evaluation,
RCS Temperature
Figure 19E.4.10-201

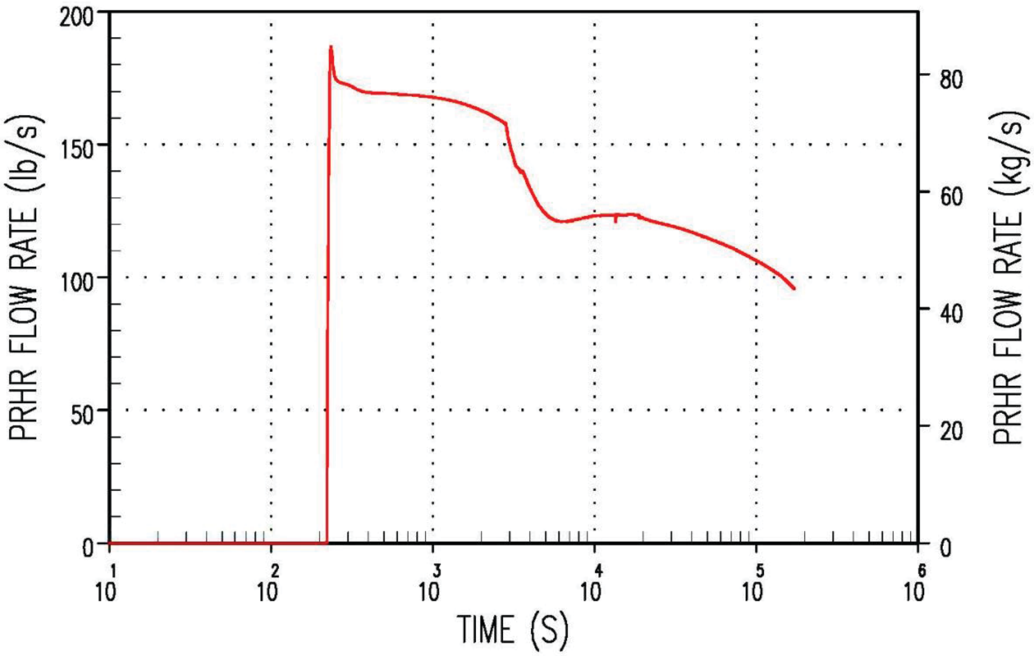
Rev. X



Duke Energy Florida
Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

Shutdown Temperature Evaluation,
PRHR Heat Transfer
Figure 19E.4.10-202

Rev. X



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Units 1 and 2
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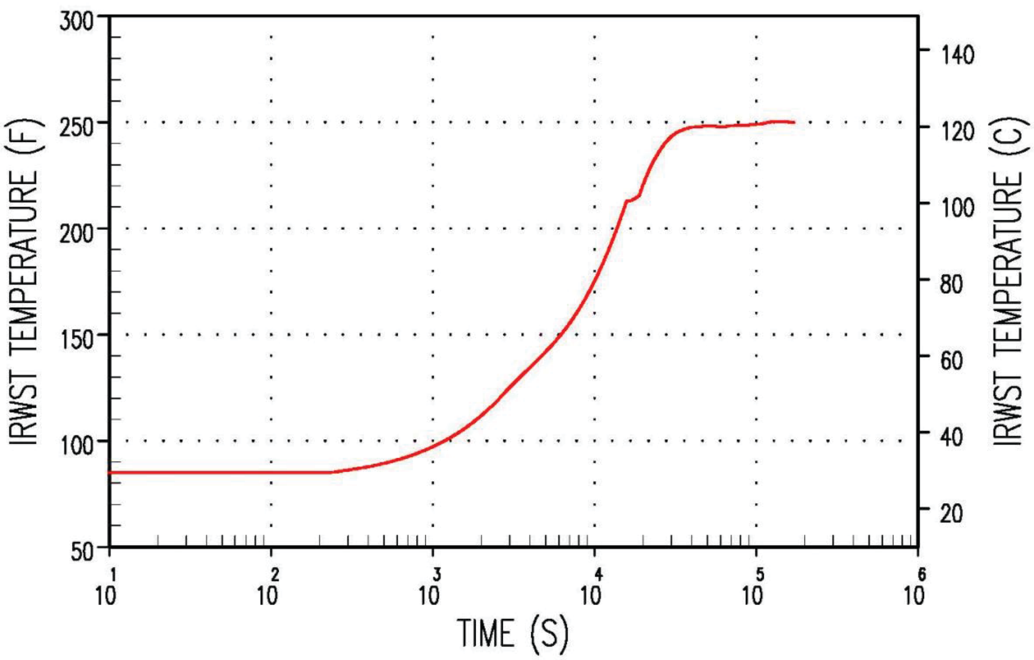
Shutdown Temperature Evaluation,
PRHR Flow Rate
Figure 19E.4.10-203

Rev. X



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Units 1 and 2
Part 2, Final Safety Analysis Report

Shutdown Temperature Evaluation,
IRWST Heatup
Figure 19E.4.10-204

Rev. X

CONDENSATE RETURN

TECHNICAL SPECIFICATION AND BASES CHANGES

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	Verify the outlet manual isolation valve is fully open.	12 hours
SR 3.5.4.2	Verify the inlet motor operated isolation valve is open.	12 hours
SR 3.5.4.3	Verify the volume of noncondensible gases in the PRHR HX inlet line has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.4.4	Verify that power is removed from the inlet motor operated isolation valve.	31 days
SR 3.5.4.5	Verify both PRHR air operated outlet isolation valves and both IRWST gutter isolation valves are OPERABLE by stroking open the valves.	In accordance with the Inservice Testing Program
SR 3.5.4.6	Verify PRHR HX heat transfer performance in accordance with the System Level OPERABILITY Testing Program.	10 years
SR 3.5.4.7	Verify by visual inspection that the IRWST ~~gutters~~ gutter and downspout screens are not restricted by debris.	24 months

B 3.3.3 LCO

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level

The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter and downspouts.

During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

B 3.5.4 BACKGROUND

In order to preserve the IRWST water for long term PRHR HX operation, downspouts and a gutter ~~is~~ are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

B 3.5.4 SURVEILLANCE REQUIREMENTS

SR 3.5.4.7

This surveillance requires visual inspection of the IRWST ~~gutters~~ gutter and downspout screens to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the ~~gutters~~ gutter or downspout screens could become restricted.