ArevaEPRDCPEm Resource

From: BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent: Tuesday, November 02, 2010 12:20 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (AREVA); ROMINE Judy (AREVA); BENNETT Kathy (AREVA); LENTZ Tony (EXTERNAL AREVA); RYAN Tom (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No.311, FSAR Ch. 16 OPEN ITEM, Supplement 4
Attachments: RAI 311 Supplement 4 Response US EPR DC.pdf

Getachew,

AREVA NP provided a schedule for technically correct and complete responses to the 2 questions in RAI No. 311 on April 5, 2010. AREVA provided an updated schedule for the remaining 2 responses on May 20, 2010. AREVA provided an updated schedule for the remaining 2 responses on August 27, 2010 to allow for additional interaction with the NRC. Supplement 3 response to RAI No. 311 was sent on September 2, 2010 to provide a response to one of the remaining two questions, 16-316. AREVA NP originally submitted Supplement 4 on the due date October 21, 2010, but is now resubmitting the response to remove the duplicate information in that response.

The attached file, “RAI 311 Supplement 4 Response US EPR DC.pdf” provides technically correct and complete FINAL response to the remaining question, 16-317. Appended to this file are the affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 311 Question 16-317.

The following table indicates the respective pages in the response document, “RAI 311 Supplement 4 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject question.

Question #	Start Page	End Page
RAI 311 — 16-317	2	5

This concludes the formal AREVA NP response to RAI 311, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (External RS/NB)
Sent: Thursday, September 02, 2010 6:05 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); Miernicki, Michael
Subject: Response to U.S. EPR Design Certification Application RAI No. 311, FSAR Ch. 16 OPEN ITEM, Supplement 3

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the 2 questions in RAI No. 311 on April 5, 2010. AREVA provided an updated schedule for the remaining 2 responses on May 20, 2010. AREVA provided an updated schedule for the remaining 2 responses on August 27, 2010 to allow for additional interaction with the NRC.

The attached file, “RAI 311 Supplement 3 Response US EPR DC.pdf” provides technically correct and complete FINAL response to 1 of the remaining 2 questions.

The following table indicates the respective pages in the response document, “RAI 311 Supplement 3 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject question.

Question #	Start Page	End Page
RAI 311 — 16-316	2	6

The schedule for a technically correct and complete response to the remaining question is unchanged and is provided below.

Question #	Response Date
RAI 311 — 16-317	October 21, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (External RS/NB)
Sent: Friday, August 27, 2010 12:09 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 311, FSAR Ch. 16 OPEN ITEM, Supplement 2

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the 2 questions in RAI No. 311 on April 5, 2010. AREVA provided an updated schedule for the remaining 2 responses on May 20, 2010 to allow for additional interaction with the NRC.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the two remaining questions.

A complete answer is not provided for the remaining 2 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date

RAI 311 — 16-316	October 21, 2010
RAI 311 — 16-317	October 21, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (EXT)
Sent: Thursday, May 20, 2010 12:26 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); ROMINE Judy (AREVA NP INC); RYAN Tom (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 311, FSAR Ch. 16 OPEN ITEM, Supplement 1

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the 2 questions in RAI No. 311 on April 5, 2010. As agreed with the NRC, additional time is needed for the NRC to review and discuss the draft responses to these questions with AREVA.

The schedule for technically correct and complete responses to these questions has been revised as provided below.

Question #	Response Date
RAI 311 — 16-316	August 31, 2010
RAI 311 — 16-317	August 31, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (EXT)
Sent: Monday, April 05, 2010 5:06 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); ROMINE Judy (AREVA NP INC); PANNELL George L (AREVA NP INC); LENTZ Tony F (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 311,FSAR Ch. 16

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 311 Response US EPR DC.pdf" provides a schedule since a technically correct and complete response to the 2 questions is not provided.

The following table indicates the respective pages in the response document, "RAI 311 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 311 — 16-316	2	2
RAI 311 — 16-317	3	3

A complete answer is not provided for the 2 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 311 — 16-316	May 20, 2010
RAI 311 — 16-317	May 20, 2010

Sincerely,

Martin (Marty) C. Bryan
Licensing Advisory Engineer
AREVA NP Inc.
Tel: (434) 832-3016
Martin.Bryan@areva.com

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Tuesday, October 13, 2009 8:01 AM
To: ZZ-DL-A-USEPR-DL
Cc: DeMarshall, Joseph; Le, Hien; Kowal, Mark; Hearn, Peter; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 311 (3839, 3846),FSAR Ch. 16

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 9, 2009, and on the October 12, 2009, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The questions in this RAI are considered potential open items for Phases 2 and 3 reviews. As such, the schedule we have established for your application assumes technically correct and complete responses prior to the start of Phase 4 review. For any RAI that cannot be answered prior to the start of Phase 4 review, it is expected that a date for receipt of this information will be provided so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 2238

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB71081F2C59)

Subject: Response to U.S. EPR Design Certification Application RAI No.311, FSAR Ch. 16 OPEN ITEM, Supplement 4
Sent Date: 11/2/2010 12:20:13 PM
Received Date: 11/2/2010 12:21:42 PM
From: BRYAN Martin (EXTERNAL AREVA)

Created By: Martin.Bryan.ext@areva.com

Recipients:
"DELANO Karen (AREVA)" <Karen.Delano@areva.com>
Tracking Status: None
"ROMINE Judy (AREVA)" <Judy.Romine@areva.com>
Tracking Status: None
"BENNETT Kathy (AREVA)" <Kathy.Bennett@areva.com>
Tracking Status: None
"LENTZ Tony (EXTERNAL AREVA)" <Tony.Lentz.ext@areva.com>
Tracking Status: None
"RYAN Tom (AREVA)" <Tom.Ryan@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	8318	11/2/2010 12:21:42 PM
RAI 311 Supplement 4 Response US EPR DC.pdf	237696	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 311 (3839, 3846), Supplement 4

10/13/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: FSAR Ch. 16

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-317:

POTENTIAL OPEN ITEM

Follow-up to RAI Question No. 16-293

In RAI 16-293, the applicant was asked to provide an assessment to confirm that all the LCO values in the proposed TS are consistent with the initial conditions assumed in the safety analyses. The staff, in particular, cited the proposed pressurizer minimum water level of 75% specified in LCO 3.4.9.a as an example where a TS requirement is not consistent with the initial condition assumed in the safety analyses (e.g., a pressurizer water level of 59% is assumed in heat-up transients in FSAR Section 15.2). In its response letter dated September 30, 2009, the applicant proposed to revise LCO 3.4.9.a and the associated discussion in the TS Bases B 3.4.9 to reflect the assumed value of 59% in a feedwater line break event which is identified as the limiting event for pressurizer water level concerns. The staff finds the revised LCO 3.4.9.a and its associated TS Bases B 3.4.9 acceptable, however, the requested assessment of all the LCO values proposed in the EPR GTS is not provided. The applicant is requested to provide an overall assessment as stated in the original RAI 16-293. This is identified as an open item in the SER w/OI for Chapter 16 of the EPR FSAR.

Response to Question 16-317:

An assessment was made of the US EPR GTS including LCOs, Actions, Surveillances and Bases against the accident analyses in U.S. EPR FSAR Tier 2, Chapter 15. The following changes to the US EPR GTS were identified:

1. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Sections 3.1.8, 3.4.6, 3.4.7, 3.4.8, 3.9.4, 3.9.5, and Bases Sections B3.1.1, B3.1.8, B3.4.6, B3.4.7, B3.4.8, B3.9.1, B3.9.4, and B3.9.5 were revised in Revision 2 of the U.S. EPR FSAR to reflect the Anti-Dilution Mitigation system described in U.S. EPR FSAR Tier 2, Section 9.3.4 and Chapter 15 boron dilution analysis. The safety analysis credits this system in all modes. In Mode 1 it protects against the loss of shutdown margin. One isolation valve is allowed to be inoperable for 72 hours consistent with single failure assumptions in the analysis.

   U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.1.8, Bases B3.1.1, B3.1.8, B3.9.4, and B3.9.5 will be revised for clarification and consistency with U.S. EPR FSAR Tier 2, Chapter 15 analyses.

2. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.2.2 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the U.S. EPR FSAR Tier 2, Chapter 15 analyses.

3. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.2.5 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to reflect AREVA usage of Azimuthal Power Imbalance in core and accident analyses. Conforming changes were also made to Sections 1.1, 3.1.4, 5.6.3, B3.1.4, B3.1.6, B3.1.9, B3.2.2, and B3.4.1 in Revision 2 of the U.S. EPR FSAR.

4. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.4.5 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 analyses of uncontrolled control rod assembly withdrawal. In Mode 3, the safety analysis requires 4 loops in operation when rods are capable of withdrawal. Therefore, a short time

(2 hours) is given to place the rods in a condition where they are not capable of being withdrawn.

5. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.4.9 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 analysis assumptions.

   U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.4.9 establishes limitations on the maximum pressurizer level based on assumptions in the safety analysis. Pressurizer level is one of the initial conditions selected in the safety analysis to describe the plant state from which the postulated event occurs. The limiting case conditions for each event are summarized in U.S. EPR FSAR Tier 2, Table 15.0-63. For each event, parameters are selected within the allowed operating band including uncertainties. The parameter is selected to produce the greatest challenge to acceptance criteria. When the variation of a parameter has a negligible impact on results, the nominal value is selected. For example, for the chemical and volume control system (CVCS) malfunction event, the pressurizer level listed in U.S. EPR FSAR Tier 2, Table 15.0-63 that was selected for the limiting case is 54.3 percent, which is the nominal pressurizer level. In this event the limiting results would produce the maximum pressurizer level. Because design features of the U.S. EPR include a reactor trip and CVCS isolation on high pressurizer level setpoints, the final pressurizer level at the end of the event is independent of initial level. In this case the selection of initial pressurizer level has a negligible impact on the final result and is, therefore, established as nominal.

   For the CVCS malfunction event the end result is the pressurizer does not overfill. AREVA applies this criteria as a conservative target realizing that if the pressurizer results in overfill the PSRVs are designed to operate in steam, two phase mixture, or sub-cooled liquid conditions. This design feature is described in U.S. EPR FSAR Tier 2, Section 5.4.13.2. In U.S. EPR FSAR Tier 2, Chapter 15, the small feedwater line break results show that the pressurizer level exceeds 100 percent. In this case, because the pressurizer safety relief valves (PSRVs) are designed to operate with steam, two phase mixture, or sub-cooled liquid discharge, the valves are expected to complete their function and reseat. This is consistent with NUREG-0737 item II.D.1 – Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2).

   U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.4.9 and associated Bases will be revised for clarification and consistency with U.S. EPR FSAR Tier 2, Chapter 15 analyses.

6. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.5.3 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 LOCA analyses. Mode 4 loss of coolant accident (LOCA) analysis assumes three medium head safety injection (MHSI) pumps are available. One MHSI injecting to the vessel is success. One MHSI feeds the break and one MHSI is the single failure.

   U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Bases Sections B3.5.3 will be revised for clarification and consistency with Chapter 15 analyses.

7. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.5.5 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 pH

analyses. Tank volume was modified to be consistent with the design and what was used in SGTR analysis.

8. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.1 and associated Bases were revised to agree in Revision 2 of the U.S. EPR FSAR to reduce power to match the available relief capacity. This approach is conservative because the Chapter 15 safety analysis supports full power operation with one main steam safety valve (MSSV) inoperable as long as the MSRT in same loop is operable.

   U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.1 and associated Bases will be revised for clarification and consistency with Chapter 15 analyses.

9. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.2 was revised in Revision 2 of the U.S. EPR FSAR to agree clarify testing requirements.

10. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.3 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 analyses assumptions. Main feedwater (MFW) valves are relied upon to isolate each corresponding flow path. The changes reflect when they are required to perform their isolation function. One valve is allowed to be inoperable for 72 hours consistent with single failure assumptions.

11. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.4 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 analyses assumptions. The safety analysis assumes that the main steam relief trains (MSRTs) function during overpressure events. Because the safety analysis assumes one MSRT fails to open as a single failure a MSRT may be inoperable for 72 hours.

    U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Section 3.7.4 and associated Bases will be revised for clarification and consistency with Chapter 15 analyses.

12. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Sections 3.7.5 and 3.7.6 and associated Bases were revised in Revision 2 of the U.S. EPR FSAR to agree with the U.S. EPR FSAR Tier 2, Chapter 15 analyses assumptions. The safety analysis requires two emergency feedwater (EFW) pumps for success. That means that if one is the single failure, then three pumps are required to be operable at all times in Modes 1, 2, and 3. In Mode 4 only one pump is required for success. Two pumps are therefore required to account for single failure. Because there are four EFW pumps, one pump can be out for maintenance.

    U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Sections 3.7.5 and 3.7.6 and associated Bases will be revised for clarification and consistency with Chapter 15 analyses.

13. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.4.4 was revised in Revision 2 of the U.S. EPR FSAR to agree with the assumed design overpower condition.

    U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.3.1 will be revised for clarification and consistency with Chapter 15 analyses.

14. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section 3.4.10 was revised in Revision 2 of the U.S. EPR FSAR to delete reference to an immediate shutdown condition that is not part of the U.S. EPR plant design.

15. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.4.16 was revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 steam generator tube rupture analysis. Offsite dose is mostly released to atmosphere via a stuck open MSRT not through the condenser.
16. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.5.4 was revised in Revision 2 of the U.S. EPR FSAR to agree with the Chapter 15 LOCA analyses.

    U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.5.4 will be revised for clarification and consistency with Chapter 15 analyses.
17. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Sections B3.5.2 and B3.5.8 will be revised for clarification and consistency with Chapter 15 analyses.
18. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Section 3.7.19 and associated Bases will be revised to replace a Surveillance Requirement and with a COLA requirement because the UHS makeup pump is not in DC scope.
19. U.S. EPR FSAR Tier 2, Chapter 16 Technical Specifications, Bases Section B3.9.2 will be revised for clarification.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications and Bases will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

16-317

Table 15.0-63—Transient Analysis Limiting Case Conditions[9]
(Sheet 1 of 3)

Event	Limiting Acceptance Criteria	Power (MW_t)	T_{avg} (°F)	RCS Flow per loop (gpm)	Pressurizer level (%)	SG Level (%)	SG Tube Plugging (%)
15.1 Increase in Heat Removal by Secondary System							
Decrease in feedwater temperature	SAFDLs	4612	594[2]	119,692[3]	54.3	49	0
Increase in feedwater flow	SAFDLs	4612	594[2]	119,692[3]	54.3	49	0
Increase in steam flow	SAFDLs	4612	594[2]	119,692[3]	54.3	49	0
Inadvertent opening of a SG relief or safety valve	SAFDLs	4612	594[2]	119,692[3]	54.3	49	0
Steam system piping failure	SAFDLs	1.0E-06	578[2]	119,692[3]	34	49	0
15.2 Decrease in Heat Removal by Secondary System							
Turbine Trip	RCS pressure	4612	594	119,692	59.3	49	5
Closure of a MSIV	SG pressure	4612	598	119,692	59.3	49	0
Loss of non-emergency AC power[1]		—	—	—	—	—	—
Loss of normal feedwater flow	RCS and SG pressure[8]						
	Decay heat removal	4612	594	119,692	59.3	49	0
	Pressurizer overfill	4612	579	119,692	59.3	49	0
Feedwater system pipe break	RCS Pressure	4612	594	119,692	59.3	49	0
	SG pressure	4612	594	119,692	59.3	49	0
	Decay heat removal	4612	594	119,692	59.3	49	0
	Pressurizer overfill	4612	584	119,692	59.3	49	5

16-317

Table 15.0-63—Transient Analysis Limiting Case Conditions[9]
(Sheet 2 of 3)

Event	Limiting Acceptance Criteria	Power (MW_t)	T_{avg} (°F)	RCS Flow per loop (gpm)	Pressurizer level (%)	SG Level (%)	SG Tube Plugging (%)
15.3 Decrease in Reactor Coolant System Flow Rate							
Partial loss of forced reactor coolant flow	SAFDLs	4612	594[2]	119,692	54.3	49	5
Complete loss of forced reactor coolant flow	SAFDLs	4612	594[2]	119,692	54.3	49	5
RCP rotor seizure	SAFDLs	4612	594[2]	119,692	54.3	49	5
15.4 Reactivity and Power Distribution Anomaly							
Uncontrolled RCCA withdrawal from a subcritical or low power startup condition	SAFDLs	4.59E-06	578[2]	119,692	34	49	5
Uncontrolled RCCA bank withdrawal at power	SAFDLs	0-4612[4]	578-594[2]	119,692	34-54.3	49	5
Single RCCA withdrawal	SAFDLs	0-4612[4]	578-594[2]	119,692	34-54.3	49	5
RCCA drop	SAFDLs	4612[4]	594[2]	119,692	54.3	49	5
Startup of a RCP in an inactive loop[7]		2754	594[2]	119,692	54.3	49	5
Decrease in the boron concentration in the RCS[5]	SAFDLs	0-4612[4]	578-594[2]	119,692	34-54.3	49	5
RCCA ejection	RCS pressure	0	578	119,692	34	49	—
	Fuel deposition limits	4612	594	119,692	54.3	49	—
15.5 Increase in RCS Inventory							
Inadvertent operation of the ECCS or EBS	Pressurizer overfill	4612	594	119,692	54.3	49	5

Table 15.0-63—Transient Analysis Limiting Case Conditions[9] (Sheet 3 of 3)

Event	Limiting Acceptance Criteria	Power (MW_t)	T_{avg} (°F)	RCS Flow per loop (gpm)	Pressurizer level (%)	SG Level (%)	SG Tube Plugging (%)
CVCS malfunction that increases reactor coolant inventory	Pressurizer overfill	4612	594	119,692	54.3	49	5
15.6 Decrease in RCS Inventory							
Inadvertent opening of a pressurizer relief valve	SAFDL	4612	594	119,692	54.3	49	5
SGTR	Radiological dose	4612	584	119,692	59.3	49	5
	SG Overfill	4612	584	119,692	59.3	49	5
Loss-of-coolant accident	10CFR50.46 LBLOCA	See Note 6	See Note 6	See Note 6	See Note 6	—	5
	10CFR50.46 SBLOCA	4612	594	119,692	54.3	49	5

Notes:

1. Not analyzed. Event is bounded by complete loss of flow for DNB and loss of normal feedwater for RCS and SG Pressure.
2. Nominal T_{avg} at full power. Operating band and measurement uncertainties on temperature and pressure are applied in the DNB analysis.
3. Thermal design flow is assumed in the system analysis for DNB limiting events.
4. No specific limiting case. Low DNBR trip provides protection.
5. This event is analyzed at power as part of the spectrum of uncontrolled rod withdrawal events. In the shutdown modes, this event establishes the setpoints for the anti-dilution mitigation system.
6. These parameters are statistically sampled. See Section 15.6.5.1.

7. Acceptance criteria are not challenged.

16-317

8. Bounded by turbine trip and MSIV closure events, respectively.

9. Parameters are selected within the allowed operating band that would produce the greatest challenge to the acceptance criteria. When the variation of a parameter has a negligible impact on the results, the nominal value is selected.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Anti-Dilution Mitigation (ADM)

LCO 3.1.8 The volume control tank (VCT) and letdown isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

16-317

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~or more~~ VCT or letdown isolation valves inoperable.	A.1	Restore VCT or letdown isolation valve~~s~~ to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Isolate the affected boron dilution flow path by use of at least one closed and de-activated automatic valve or closed manual valve.	8 hours
	AND		
	B.2	-------------NOTE--------------- Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. -------------------------------------	
		Verify the affected boron dilution flow path is isolated.	31 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The Pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 59%;

b. Three groups of emergency supply pressurizer heaters OPERABLE with the capacity of each group ≥ 144 kW; and

c. The Chemical Volume and Control System (CVCS) charging and auxiliary spray isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 > P17.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Fully insert all rods.	6 hours
	AND	
	A.3 Place the Control Rod Drive Control System in a condition incapable of rod withdrawal.	6 hours
	AND	16-317
	A.4 Be in MODE 4 at ≤ P17.	~~12~~ 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
-------------NOTE--------------- Separate Condition entry is allowed for each valve. ------------------------------------ C. CVCS charging isolation valve or auxiliary spray isolation valve inoperable.	C.1 Isolate the associated flow path. 16-317	6 hours
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4 at ≤ P17.	6 hours 16-317 ~~12~~ 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 59%.	12 hours
SR 3.4.9.2	Verify capacity of each required group of emergency supply pressurizer heaters is ≥ 144 kW. 16-317	92 days
SR 3.4.9.3	Verify the CVCS charging and auxiliary spray isolation valves actuate to the correct position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

16-317

LCO 3.7.1 Two MSSVs per steam generator shall be OPERABLE.

--NOTE--
Only one MSSV per steam generator is required to be OPERABLE in MODE 4 when steam generator is relied upon for heat removal.

APPLICABILITY: MODES 1, 2, and 3,~~.~~
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MSSV inoperable.	A.1 Verify associated Main Steam Relief Train is OPERABLE. AND A.2 Reduce power to less than 75% RTP.	Immediately 4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two or more required MSSVs inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours ~~12~~ 24 hours

3.7 PLANT SYSTEMS

3.7.4 Main Steam Relief Trains (MSRTs)

16-317

LCO 3.7.4 Four MSRTs shall be OPERABLE.

---NOTE---
Only two MSRTs are required to be OPERABLE in MODE 4 when steam generator is relied upon for heat removal.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSRIVs inoperable due to one or both pilot valves in one control line inoperable for opening. ~~OR~~	A.1 Restore pilot valve(s) to OPERABLE status.	72 hours~~30 days~~
B. One or more required MSRIVs inoperable due to one pilot valve open in one or both control lines.	B.1 Restore pilot valve(s) to OPERABLE status.	30 days

ACTIONS (continued)

16-317

CONDITION	REQUIRED ACTION	COMPLETION TIME
C~~B~~. One required MSRT inoperable for opening.	C~~B~~.1 Verify associated main steam safety valve(s) are OPERABLE. AND C.2 Reduce power to less than 50% RTP. AND C.3~~B.2~~ Restore MSRT to OPERABLE status.	Immediately 4 hours 72 hours
D~~C~~. Required Action and associated Completion Times of Condition A, ~~or~~ B, or C not met. OR Two or more required MSRTs inoperable. ~~OR~~ ~~Two or more required MSRCVs inoperable.~~	D~~C~~.1 Be in MODE 3. AND D~~C~~.2 Be in MODE 4 without reliance upon steam generator~~s~~ for heat removal.	6 hours ~~12~~ 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each MSRIV. 16-317	In accordance with the Inservice Testing Program ~~24 months on STAGGERED TEST BASIS for each control line~~
SR 3.7.4.2	Verify one complete cycle of each MSRCV.	24 months
SR 3.7.4.3	Verify each MSRIV automatically actuates on an actual or simulated steam pressure setpoints.	24 months
SR 3.7.4.4	Verify each MSRCV is automatically positioned on an actual or simulated actuation signal.	24 months
SR 3.7.4.5	Verify each MSRCV is automatically switched into steam generator pressure control mode on an actual or simulated MSRIV opening signal.	24 months

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Four EFW pump trains along with the downstream injection pathways and ~~common~~ supply and discharge headers shall be OPERABLE.

---NOTE---
Only two EFW pump trains are required to be OPERABLE in MODE 4 when steam generator is relied upon for heat removal.

16-317

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable for two or more EFW pump trains inoperable when entering MODE 1.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EFW pump train inoperable in MODE 1, 2, or 3.	A.1	Restore EFW pump train to OPERABLE status.	120 days
B. Two EFW pump trains inoperable in MODE 1, 2, or 3.	B.1	Restore one EFW pump train to OPERABLE status.	72 hours
C. Downstream injection pathway or ~~common~~ supply or discharge header inoperable.	C.1	Restore the injection pathway and ~~common~~ supply and discharge headers to OPERABLE status.	~~72~~ 8 hours

16-317

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B , or C not met. OR Three EFW pump trains inoperable in MODE 1, 2, or 3.	D.1 Be in MODE 3. AND D.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours ~~12~~ 24 hours 16-317
E. Four EFW pump trains inoperable in MODE 1, 2, or 3.	E.1 ------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW pump train is restored to OPERABLE status. ------------------------------------- 16-317 Initiate action to restore one EFW pump train to OPERABLE status.	 Immediately
F. One of the two required EFW pump trains inoperable in MODE 4 when ~~a~~ steam generator is relied upon for heat removal.	F.1 Restore required EFW pump train to OPERABLE status. OR F.2 Be in MODE 4 without reliance upon steam generator for heat removal.	72 hours 96 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	Cycle each EFW discharge header cross-connect valve.	In accordance with the Inservice Testing Program
SR 3.7.5.3	Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.4	16-317 Verify, on an actual or simulated actuation signal, each EFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position and each EFW pump starts automatically.	24 months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying flow from the EFW storage pool to its respective steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

3.7 PLANT SYSTEMS

3.7.6 Emergency Feedwater (EFW) Storage Pools

LCO 3.7.6 Four EFW Storage Pools shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more EFW Storage Pools inoperable.	A.1 Declare associated EFW pump train inoperable.	Immediately
	<u>AND</u>	
	A.2 Verify by administrative means, the availability of the back-up water supply~~ies~~. 16-317	4 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>	
	A.3 Restore EFW Storage Pools to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4 without reliance upon steam generator for heat removal.	24 hours

16-317

SURVEILLANCE REQUIREMENTS

16-317

--REVIEWER'S NOTE--
A surveillance to verify the ability to supply makeup water to each UHS cooling tower basin at ≥ 300 gpm will be provided by the COL applicant.

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	Verify water level of each UHS cooling tower basin is ≥ 23.75 feet.	24 hours
SR 3.7.19.2	Verify water temperature of each UHS cooling tower basin is ≤ 90°F.	24 hours
SR 3.7.19.3	Operate each UHS cooling tower fan for ≥ 15 minutes in each speed setting and direction, including reverse.	31 days
SR 3.7.19.4	Verify each UHS cooling tower fan starts automatically on an actual or simulated actuation signal.	24 months
~~SR 3.7.19.5~~	~~Verify the ability to supply makeup water to each UHS cooling tower basin at ≥ 300 gpm.~~	~~24 months~~
SR 3.7.19.~~6~~5	Verify each UHS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
[SR 3.7.19.6	Verify the ability to supply makeup water to each UHS cooling tower basin at ≥ 300 gpm.]	[In accordance with the Inservice Testing Program]

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loop~~s~~ in operation. 16-317	B.1	Initiate action to restore required RHR loop to operation.	Immediately
	<u>AND</u>		
	B.2	Close and secure equipment hatch.	4 hours
	<u>AND</u>		
	B.3	Close one door in each air lock.	4 hours
	<u>AND</u>		
	B.4	Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Ventilation System.	4 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

16-317

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not result in unacceptable consequences.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or trip of all shutdown and control RCCAs, assuming that the single RCCA of highest reactivity worth is fully withdrawn. For postulated accidents, some fuel damage is acceptable as long as offsite dose consequences remain within 10 CFR 100 limits.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod Drive Control System (CRDCS) can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the CRDCS, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the RCCA of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Anti-Dilution Mitigation (ADM)

BASES

BACKGROUND

The primary purpose of ADM is to mitigate the consequences of the inadvertent addition of unborated primary grade water into the Reactor Coolant System (RCS) when the reactor is in MODES 1, 2, 3, 4, 5, and 6 with no RCPs in operation. The RCS boron concentration that is measured by the Boron Concentration – CVCS Charging Line sensors is continuously compared to pre-established setpoints that ~~is~~ depends on the following conditions:

16-317

- Reactor critical;
- Shutdown, with RCPs in operation; and
- Shutdown, without RCPs in operation.

The ADM setpoint ensures:

- The dilution is terminated when the Protection System setpoint is actuated;
- The available SDM is sufficient to shutdown the core with the RCCA with highest worth unable to insert, and
- The core remains sub-critical if already shutdown.

The ADM setpoint is periodically adjusted to compensate for core burnup and the indicated boron concentration is periodically compared against boron titration samples and boron isotopic analyses to confirm that the measured boron concentration is within analysis assumptions. The instrumentation requirements are specified in Technical Specification 3.3.1 "Protection System."

The volume control tank (VCT) and letdown isolation valves actuate to the isolation position on a signal from the Protection System. The OPERABILITY requirements for these isolation valves help ensure that a dilution path is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the isolation function assumed in the safety analyses will be maintained.

BASES

ACTIONS

A.1

If one of the VCT and letdown isolation valves are inoperable, the automatic capability for mitigation of dilution events is no longer available. In this case, the isolation valve is required to be restored to OPERABLE status within 72 hours. In this Condition, ability to maintain subcriticality may be reduced. Thus~~,~~ 72 hours are allowed to return the VCT and letdown isolation valve~~s~~ to OPERABLE status.

B.1 and B.2

If the VCT and letdown isolation valve cannot be returned to OPERABLE status within the associated Completion Time, ~~the plant must be brought to a configuration in which the LCO does not apply. As an alternative (Required Actions B.1 and B.2),~~ the VCT and letdown return line must be closed and secured within 8 hours to isolate the unborated water sources. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to isolate the boron dilution flow path. ~~return the isolation valve to an OPERABLE condition in an orderly manner.~~ The boron dilution flow path will be verified to be isolated every 31 days. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that make valve opening an unlikely possibility.

16-317

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Periodic surveillance testing of VCT and letdown isolation valves is required by the ASME OM Code (Ref. 3). This verifies that the measured performance, receipt of isolation signal to full closure, is within an acceptable tolerance of the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME OM Code.

The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

16-317

BASES

ACTIONS

A.1

If one ~~required~~ RCS loop is inoperable or not in operation, and the CRDCS is capable of rod withdrawal, the Required Action is to place the CRDCS in a condition incapable of rod withdrawal (e.g., removing power from CRDMs). When the CRDCS is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of four RCS loops in operation. If three RCS loops are in operation, the CRDCS must be rendered incapable of rod withdrawal. The Completion Time of 2 hours to make the CRDCS incapable of rod withdrawal is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

B.1

If one or more required RCS loops is inoperable with CRDS incapable of rod withdrawal, redundancy for decay heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without a redundant loop because of the low probability of a failure in the remaining loops occurring during this period.

C.1

If no required RCS loop is in operation, except as allowed by the Note in the LCO section, and the CRDCS is incapable of rod withdrawal, the Required Action is to initiate action to place one RCS loop in operation. The immediate Completion Time reflects the importance of maintaining operation of an RCS loop for heat removal. The action to restore must be continued until one loop is restored to operation.

16-317

D.1

~~If restoration for~~ When Required Action A.1, B.1, or C.1 ~~is not possible within 72 hours~~cannot be completed within the required Completion Times, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

BASES

LCO (continued)

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE LHSI pump capable of providing forced flow to an OPERABLE LHSI heat exchanger. RCPs and LHSI pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RCS or two loops of RHR provides sufficient circulation for decay heat removal. One RCS or RHR loop provides boron mixing. However, additional loops consisting of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1

If one required RCS or one or two required RHR loop(s) is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or required number of RHR loops to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until two paths are restored to OPERABLE status.

16-317

BASES

LCO (continued)

Note 3 requires that the secondary side water temperature of each SG be ≤ 50 °F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

LHSI pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 20%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1, A.2, B.1, and B.2

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels < 20%, or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until two paths are restored to OPERABLE status.

16-317

BASES

ACTIONS (continued)

16-317

C.1

If a required RHR loop is not in operation, except during conditions permitted by Note 1 in the LCO section, or if no required loop is OPERABLE, action to restore one RHR loop to OPERABLE status and operation must be initiated. The immediate Completion Time~~s~~ reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 20% ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that each required LHSI pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required LHSI pump. If secondary side water level is ≥ 20% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

16-317

ACTIONS

A.1

If one required RHR loop is inoperable, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until two paths are restored to OPERABLE status and operation

B.1

If no required loop is OPERABLE or the required loop is not in operation, action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters and their controls, and Chemical Volume and Control System (CVCS) valves that limit the increase in water level. Pressurizer Safety Relief Valves are addressed by LCO 3.4.10, "Pressurizer Safety Relief Valves (PSRVs)."

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for anticipated design basis transients. The water level limit thus serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and this is in the preferred state for heat transport; and

16-317

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus both sprays and heaters can operate to maintain the design operating pressure. The level limit also ~~prevents~~ minimizes the potential for overfilling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PSRVs) can control pressure by steam relief rather than water relief. A small feedwater line break (Ref. 1) results in the maximum transient pressurizer level. The PSRVs are designed for water relief. The maximum RCS pressure does not exceed the 110% ASME Code limit. ~~If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2803 psia or damage may occur to the PSRVs.~~

BASES

BACKGROUND (continued)

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of emergency supply pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSRV with water overflow. The CVCS charging and auxiliary spray isolation valves isolate on an increasing pressurizer level to perform this function. The CVCS charging isolation valve is normally open during operation for pressurizer level control while the CVCS auxiliary spray isolation valve is opened during plant cooldown to reduce pressurizer temperature and provide pressure control when the RCS normal sprays are not functional (RCPs removed from service).

16-317

16-317

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, ~~and~~ 3, and MODE 4 > P17, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety for lower MODES are not limiting. All analyses performed from a critical analyses performed reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in FSAR Chapter 15 (Ref. 1) do not take credit for emergency supply pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The limiting event for pressurizer level is a feedwater line break.

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSRV with water overflow. The closure of the CVCS charging and auxiliary spray isolation valves perform this function. This function is required in MODES 1, 2, 3, and MODE 4 > P17.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the emergency supply heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

BASES

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ≤ 59% ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

Each required emergency powered heater group contains 6 heaters with an electrical requirement of 144 kW with each heater requiring 24 kW. The three groups required to be OPERABLE contain 18 emergency powered heaters for a total of 432 kW (3x6x24 kW). If a single failure occurs, one group of heaters would be inoperable and the required 288 kW needed for maintaining a bubble in the pressurizer during the design basis event would be maintained.

16-317

The LCO requires the CVCS charging and auxiliary spray isolation valves to be OPERABLE to minimize ~~prevent~~ overfilling the pressurizer which would lead to pressurization and opening of the PSRV with water overflow. For the valves to be OPERABLE they must be capable of automatically closing on the CVCS Charging Line Isolation on High-High Pressurizer Level signal generated from the Protection System (LCO 3.3.1, "Protection System (PS)").

APPLICABILITY

16-317

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3 and MODE 4 > P17. The purpose is to prevent solid water RCS operation during heatup and cooldown in MODE 4 > P17 to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, 3, and MODE 4 > P17, there is a need to maintain the availability of the emergency supply pressurizer heaters. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODES 4 ≤ ~~<~~ P17, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in operation.

16-317

ACTIONS

A.1, A.2, A.3, and A4~~, and A.5~~

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.~~ Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Level > Max 1p Trip Setpoint.~~

If the pressurizer water level is not within the limit, action must be taken to restore the level within limit within 6 hours or by placing the unit in a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 at ≤ P17 within ~~12~~ 24 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of emergency supply pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using the remaining heaters.

C.1

Condition C is modified by a Note indicating that separate Condition entry is allowed for each valve. If the CVCS charging isolation valve or the CVCS auxiliary spray isolation valve is inoperable and cannot be restored in the allowed Completion Time, the associated flow path must be isolated. This ensures the function of the valves has been performed. The flow path can be isolated using additional valves which are in the flow path, but do not get the automatic closure signal on increasing pressurizer level. The Completion Time of 6 hours is reasonable considering the controls for the additional valves are located in the control room.

BASES

ACTIONS (continued)

D.1 and D.2

If one required group of emergency supply pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, or the CVCS charging isolation valve or auxiliary spray isolation valve and the associated flow path is not isolated in the allowed Completion Time of Required Action C.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 at ≤ P17 within ~~12~~ 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

16-317

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated emergency supply pressurizer heaters are verified to be at their design rating. This SR may be verified by energizing the heaters and measuring circuit current. The Frequency of 92 days is considered adequate to detect heater degradation and has shown by operating experience to be acceptable.

BASES

BACKGROUND (continued)

If it is necessary to remove one LHSI train from service, an isolatable ECCS cross-connect ensures LHSI delivery in the event of a cold leg break. Whenever the cross-connects are opened, the isolation valve electrical breakers are racked-out to avoid single failure. Otherwise, both ECCS cross-connects are isolated to maintain LHSI train separation. 16-317

The ECCS accumulators and the IRWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

16-317

Cross-connect capability is provided between Trains 1 and 2 and between Trains 3 and 4 for the LHSI pumps. Cross-connect capability allows one LHSI ~~SI~~ train to be removed from service for maintenance and provides alternate injection paths for the LHSI trains that remain in service. This configuration mitigates the effect of potential degraded injection flow due to steam entrainment during a loss of coolant event. At higher pressures when the MHSI pumps provide the injection flow, steam entrainment is not a concern; therefore no cross-connects are required between the MHSI trains.

Each ECCS flow path consists of piping, valves, heat exchangers, and pumps such that water from the IRWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the MHSI pumps, the LHSI pumps, and LHSI heat exchangers. Each of the two subsystems (MHSI and LHSI) consists of four 100% capacity trains that are independent and redundant such that each train is capable of supplying 100% of the flow required to mitigate the accident consequences.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the MHSI pumps, the secondary side is cooled down to approximately 870 psia at a rate of approximately 180°F/hr by means of the relief valves to ensure adequate injection from the MHSI System.

16-317

BASES

BACKGROUND (continued)

Due to the large miniflow lines, it is not necessary to limit the number of MHSI or LHSI pumps in service during low temperature conditions in the RCS. Refer to the Bases for LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for the basis of low RCS temperature operation.

The ECCS subsystems are actuated upon receipt of a Protection System (PS) signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the IRWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "In-Containment Refueling Water Storage Tank (IRWST) - Operating," provide the cooling water necessary to meet GDC 35 (Ref. 1). 16-317

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO also limits the post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The MHSI and LHSI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the MHSI pumps. The SGTR and MSLB events also credit the MHSI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power; and

16-317

b. A small break LOCA event, with a loss of offsite power.

ECCS cross-connects are provided for the LHSI trains to allow the removal of one train from service. Cross-connects are normally isolated by two motor-operated valves (MOVs) arranged in a series configuration. The MOVs provide separation of LHSI trains when the valves are closed. The cross-connects provide the ability to cross-tie between Trains 1 and 2 or between Trains 3 and 4. The cross-connects do not provide the ability to cross-tie between any other train combinations. When the cross-connects are opened and the associated LHSI trains are cross-tied, the electrical power is required to be removed from the MOVs to ensure the valves remain in the required position.

16-317

If LHSI Train 1 is removed from service, the MOVs between Train 1 and 2 are opened, and additionally the MOVs between Trains 3 and 4 are also opened. With this configuration, if a DBA were to occur Train 2 would provide 50 percent of the required design flow to RCS loop 1, and 50 percent to RCS loop 2. For Train 3 and Train 4, 100 percent of the required design flow would be provided to RCS loops 3 and 4. The most limiting event is an RCP cold leg break. If Train 2 was lost because of a single failure, and Train 4 assumed to have the break, flow would be lost and Train 3 would provide sufficient flow to the core.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The MHSI pump performance requirements are based on a small break LOCA. MODE 2 and 3 requirements are bounded by the MODE 1 analysis.

16-317

This LCO is only applicable in MODE 3 and above. Below 350°F, the PS signal setpoint for the LHSI System is manually bypassed by operator control (in order to allow the alignment of the LHSI System in RHR mode), and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown, MODE 4."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "~~LHSI / RHR and Coolant Circulation~~ RHR Loops - High Water Level," and LCO 3.9.5, "~~LHSI / RHR and Coolant Circulation~~RHR Loops - Low Water Level."

ACTIONS

A.1, B.1 and B.2

With one MHSI train inoperable, the inoperable components must be returned to OPERABLE status within 120 days. The 120 day Completion Time is based on the assumption in the FSAR Chapter 15 analysis that one ECCS train is assumed out of service for maintenance at the time of the accident.

With one LHSI train inoperable, an acceptable ECCS configuration can be achieved by opening both ECCS cross connections. In the event of a cold leg break, one train is assumed lost due to steam entrainment to the broken loop, one train is assumed to mitigate the event, and one train is assumed to have a single failure.

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown, MODE 4

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

16-317

In MODE 4, a single ECCS train consisting of a Medium Head Safety Injection (MHSI) train is capable of providing the core cooling function. A second MHSI train is assumed to spill out of the break. Low Head Safety Injection (LHSI) is not automatically actuated.

The ECCS MHSI flow paths consist of piping, valves, ~~heat exchangers,~~ and pumps such that water from the In-Containment Refueling Water Storage Tank (IRWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Below P14 and RHR connected, LHSI is not automatically actuated by the Protection System (PS). However, MHSI is automatically actuated by the PS.

Three MHSI trains of ECCS are required for MODE 4.

The ECCS MHSI trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, three of the four independent (and redundant) ECCS MHSI trains are required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA. One MHSI train is required to accomplish the safety function, one MHSI train is lost due to single failure, and one MHSI train is assumed to feed the break. ~~The ECCS cross-connects are not needed for events postulated in MODE 4.~~

In MODE 4, an ECCS train consists of an MHSI train~~System~~. Each MHSI train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the IRWST.

BASES

LCO (continued)

16-317

During an event requiring ECCS MHSI actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the MHSI pumps and to its associated cold leg injection nozzles. ~~In the long term, this flow path may be switched to deliver its flow to the RCS hot and cold legs.~~

APPLICABILITY

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

In MODE 4, three OPERABLE ECCS MHSI trains are acceptable and provide for single failure consideration ~~on the basis of the stable reactivity of the reactor and the limited core cooling requirements~~.

In MODES 5 and 6, the OPERABILITY requirements for ECCS are covered by LCO 3.5.8 “ECCS - Shutdown, MODES 5 and 6”. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "~~LHSI/RHR and Coolant Circulation~~RHR Loops - High Water Level," and LCO 3.9.5, "~~LHSI/RHR and Coolant Circulation~~RHR Loops - Low Water Level."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS MHSI train. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS MHSI train and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

16-317

A.1

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

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

BASES

LCO

The IRWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS pump operation.

To be considered OPERABLE, the IRWST must meet the valve position, temperature, water volume, and boron concentration and enrichment limits established in the SRs.

APPLICABILITY

16-317

In MODES 1, 2, 3, and 4, IRWST OPERABILITY requirements are dictated by ECCS OPERABILITY requirements. Since the ECCS must be OPERABLE in MODES 1, 2, 3, and 4, the IRWST must also be OPERABLE to support its operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." The requirements for the IRWST in MODE 5 are specified in LCO 3.5.6, “In-Containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 5.” MODE 6 core cooling requirements are addressed by LCO 3.9.4, "~~LHSI/RHR and Coolant Circulation~~RHR Loops - High Water Level," and LCO 3.9.5, "~~LHSI/RHR and Coolant Circulation~~RHR Loops - Low Water Level." The requirements for the IRWST in MODE 6 are specified in LCO 3.5.7, “In-Containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 6.”

ACTIONS

A.1

With IRWST temperature, water volume, boron concentration, or enrichment not within limits, it must be returned to within limits within 8 hours. Under these conditions the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit is acceptable considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.8 ECCS - Shutdown, MODES 5 and 6

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

16-317

In MODES 5 and 6, a single ECCS train consisting of a Medium Head Safety Injection (MHSI) train is capable of providing the core cooling function. Low head Safety Injection is not automatically actuated.

The ECCS MHSI flow paths consist of piping, valves, ~~heat exchangers,~~ and pumps such that water from the In-Containment Refueling Water Storage Tank (IRWST) can be injected into the Reactor Coolant System (RCS) following loss of shutdown cooling event.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

16-317

Due to the stable conditions and low RCS pressure associated with operation in MODES 5 and 6 and the reduced probability of occurrence of a shutdown event, the ECCS operational requirements are reduced. Below P14 and RHR connected, LHSI is not automatically actuated by the Protection System (PS). However, MHSI is automatically actuated by the PS.

Two MHSI trains of ECCS are required for MODES 5 and 6. Protection against single failures is provided for this MODE of operation.

The ECCS MHSI trains in MODES 5 and 6 satisfy~~ies~~ Criteria 4 of 10 CFR 50.36(c)(2)(ii).

LCO

16-317

In MODES 5 and 6, two of the four independent (and redundant) ECCS MHSI trains are required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following shutdown events. One MHSI train is required to accomplish the safety function and one MHSI train is assumed lost to a single failure. ~~The ECCS cross-connects are not needed for events postulated in MODES 5 and 6.~~

In MODES 5 and 6, an ECCS train consists of an MHSI train~~System~~. Each MHSI train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the IRWST.

BASES

LCO (continued)

16-317

During an event requiring ECCS MHSI actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the ECCS pumps and to its associated four cold leg injection nozzles. ~~In the long term, this flow path may be switched to deliver its flow to the RCS hot and cold legs.~~

APPLICABILITY

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

In MODES 5 and 6, two OPERABLE ECCS MHSI trains are acceptable and provide for single failure consideration ~~on the basis of the stable reactivity of the reactor and the limited core cooling requirements~~.

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

ACTIONS

A.1

16-317

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

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

B.1

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

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The MSSVs and Main Steam Relief Trains (MSRTs) provide overpressure protection for the secondary system. The MSSVs and MSRTs also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the condenser, is not available. This is done in conjunction with the Emergency Feedwater System (EFW) providing cooling water from the EFW Storage Pools.

16-317

The MSSVs are spring-loaded safety valves. Each~~The~~ MSSV rated capacity passes 25% of the full steam flow per steam generator at 110% RTP with the valves full open. Two MSSVs are located on each Main Steam Line, outside containment, upstream of the main steam isolation valves and downstream of the ~~Main Steam Relief Train (~~MSRT~~)~~ branch line, as described in FSAR Section 10.3 (Ref. 1).

The MSSVs along with the MSRTs provide overpressure protection of the main steam piping and steam generators. Together, the MSSVs and MSRTs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure during an anticipated operational occurrence (AOO) or an accident considered in the design basis accident (DBA) and transient analysis.

16-317

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in FSAR Section 15.2 (Ref. 3). Of these, the closure of a single main steam isolation valve without main steam bypass or partial trip function is the limiting AOO. Closure of a single Main Steam Isolation Valve (MSIV) results in a smaller isolated volume on the secondary side, therefore this event is more limiting than a turbine trip event for secondary system overpressure. The safety analysis demonstrates that the transient response for a single MSIV closure occurring from full power ~~without a direct reactor trip~~ presents no hazard to the integrity of the RCS or the Main Steam System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSRTs and MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. These events are bounded by the single MSIV closure event.

The safety analyses discussed above assume that the low setpoint MSSV of the affected steam generator is out of service and that the MSRT is the single failure.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires that the two MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 100.5% RTP. The LCO requires that the two MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the postulated accident analysis.

The OPERABILITY of the MSSVs are defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and be closed or reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

16-317

The LCO is modified by a Note. In MODE 4 when a steam generator is relied upon for heat removal, only one MSSV is required to be OPERABLE. Because of the reduced heat removal requirements and the short period of time in MODE 4, one MSSV is sufficient to relieve steam generator overpressure.

BASES

APPLICABILITY

In MODES 1, 2, and 3, two MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

16-317

In MODE 4 when a steam generator is relied upon for heat removal, one MSSV per steam generator is required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 when a steam generator is not relied upon for heat removal, ~~and~~ 5, and 6, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, and thus cannot be overpressurized; there is no requirement for the MSRTs or MSSVs to be OPERABLE in these MODES.

ACTIONS

A.1 and A.2

With one required MSSV inoperable, the associated MSRT is verified OPERABLE. Verification of MSRT OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSRT is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSRT. If the OPERABILITY of the associated MSRT cannot be verified, however, Condition B must be immediately entered.

An alternative to restoring the inoperable MSSV to OPERABLE status is to reduce power so that the available MSSV and MSRT relieving capacity meets ASME Code, Section III requirements for the power level. Operations may continue, provided the RATED THERMAL POWER is reduced by the application of the following formula:

RTP = Y / Z x 100%

where:

RTP = Reduced power requirement (not to exceed RTP);

Y = Total OPERABLE MSSV and MSRT design relieving capacity per steam generator of 4,266,219 lb/hr with one MSSV inoperable;

Z = Required relieving capacity per steam generator of 5,688,292 lb/hr.

BASES

ACTIONS (continued)

16-317

The Completion Time of 4 hours for Required Action A.2 is a reasonable time period to reduce reactor power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs.

B.1 and B.2

If the Required Action and associated Completion Time cannot be met or if two or more MSSVs are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 without reliance upon a steam generator for heat removal within ~~12~~ 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code (Ref. 4) requires that safety valve tests shall be performed as required by Appendix I of the ASME OM Code.

The ASME OM Code requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME OM Code specifies the activities and frequencies necessary to satisfy the requirements. The SR allows a ±[3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Steam Relief Trains (MSRTs)

BASES

BACKGROUND

16-317

The MSRTs provide overpressure protection for the secondary system. They also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB). The MSRTs also provide a method for cooling the unit to Residual Heat Removal (RHR) System entry conditions should the preferred heat sink via the condenser not be available. This is done in conjunction with the Feedwater or Emergency Feedwater System.

One MSRT is provided for each steam generator, outside containment, on a branch line that is upstream of the Main Steam Safety Valves (MSSVs) and the Main Steam Isolation Valves (MSIVs). The MSRT rated capacity passes 50% of the full steam flow per steam generator at 110% RTP with the valves full open. Each MSRT consists of one Main Steam Relief Control Valve (MSRCV) located downstream of one Main Steam Relief Isolation Valve (MSRIV).

The MSRCVs are motorized control valves, normally open, which allow control of the steam generator steam pressure. The MSRCVs provide a means of controlling MSRT steam flow to prevent overcooling the RCS. The MSRCVs allow mitigation of the effects of a stuck open MSRIV. The MSRCVs are automatically positioned based on THERMAL POWER.

The MSRIVs are angle globe valves with a motive steam-operated piston actuator, operated by two parallel sets of two pilot valves in series. The arrangement of pilot valves prevents a failure in any pilot valve from causing either a spurious opening (two pilot valves in series) or a failure to open (two sets of pilot valves in parallel). The MSRIVs are normally closed, with the pilot valves kept closed (de-energized). The MSRIVs are designed to open quickly and automatically on demand from the Protection System.

A description of the MSRCVs and the MSRIVs is found in FSAR Section 10.3 (Ref. 1)

BASES

BACKGROUND (continued)

Each MSRT minimum required capacity is 50% of the full steam generation of the assigned steam generator, at a design pressure of 1435 psig, thus limiting the system pressure to ≤ 110% of the steam generator design pressure, in order to meet the requirements of the ASME Code, Section III (Ref. 2). The minimum required capacity, combined with the MSSV capacity, provides 100% flow relief at steam generator design pressure per steam generator (SG) at ≤ 110% of the SG design pressure.

Each MSRT maximum capacity is limited to 61% of the full load steam generation of its assigned steam generator, at design pressure of 1435 psig, thus limiting the consequences of MSRIV spurious opening with regards to Reactor Coolant System overcooling and reactivity control.

The MSRTs are actuated automatically by the Protection System, but can be controlled manually by the operator.

APPLICABLE SAFETY ANALYSES

The MSRTs, along with the MSSVs (LCO 3.7.1), are credited in the mitigation of anticipated operational occurrences (AOOs) and postulated accidents in Reference 3. The MSRTs control secondary system pressure to less than the 110% design limit without challenging the MSSVs in the following events:

a. Loss of normal feedwater,

b. Loss of non-emergency AC power,

c. Inadvertent Extra Borating System actuation,

d. Uncontrolled control bank withdrawal at power,

e. Small break LOCA (SBLOCA).

16-317

For other analyzed events, an MSRT would normally actuate to control secondary system pressure but is assumed to be the limiting single failure:

a. Inadvertent opening of an ~~SG relief or safety valve~~MSSV,

b. Main Steam System piping failure,

BASES

LCO (continued)

This LCO provides assurance that the MSRTs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

16-317

The LCO is modified by a Note. In MODE 4 when a steam generator is relied upon for heat removal, only two MSRTs are required to be OPERABLE. Because of the reduced heat removal requirements and the short period of time in MODE 4, two MSRTs are sufficient to relieve steam generator overpressure.

APPLICABILITY

In MODES 1, 2, 3, and 4 when ~~the~~ a steam generator is relied upon for heat removal, the MSRTs are required to be OPERABLE to prevent Main Steam System overpressurization and to provide a decay heat removal path in conjunction with the Emergency Feedwater System.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, there are no credible transients requiring the MSRTs. In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, decay heat removal is provided by the Residual Heat Removal System.

ACTIONS

A.1

With one control line inoperable for opening in one or more MSRIVs (i.e., one pilot valve is inoperable for opening), the affected MSRIVs are still OPERABLE, however the control line(s) must be restored to OPERABLE status in ~~30 days~~72 hours. This Completion Time is based on the following:

a. Redundancy for MSRIV opening is provided by the second control line.

b. In case of an event with loss of the condenser and assuming a single failure on the second control line of one MSRIV to open, the residual heat removal can still be ensured by the other MSRTs.

c. In case of an overpressure event and assuming a single failure of the second control line of one MSRIV to open which leads to failure to open of the associated MSRIV, the redundancy provided by the two associated OPERABLE MSSVs ensure the pressure limitation in the affected SG.

BASES

ACTIONS (continued)

16-317

B.1

In case one pilot valve is open in one or both control lines of one or more MSRIVs, the isolation function of the MSRIV is not assured. The control line(s) must be restored to OPERABLE status in 30 days. This Completion Time is based on redundancy for MSRIV closure provided by the second pilot valve in series and by the MSRCV.

C.1, C.2, and C.3~~B.1 and B.2~~

With one required MSRT inoperable, the associated MSSVs are verified OPERABLE and action must be taken to reduce power to 50% RTP so that the available MSSV relieving capacity meets ASME Code, Section III requirements for the power level. The MSRT must be restored ~~the MSRT~~ to OPERABLE status within 72 hours. Verification of MSSVs OPERABILITY is performed as an administrative check by examining logs or other information to determine if MSSVs are out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSSVs. If the OPERABILITY of the associated MSSVs cannot be verified, however, Condition D~~C~~ must be immediately entered.

D.1 and D.2~~C.1 and C.2~~

If Required Action A.1, ~~or~~ B.1, or C.1 and C~~B~~.2 cannot be met within the required Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within ~~12~~ 24 hours.

With two or more required MSRTs inoperable~~.~~, the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. The unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 without reliance upon a steam generator~~s~~ for heat removal within ~~12~~ 24 hours.

BASES

SURVEILLANCE REQUIREMENTS

16-317

SR 3.7.4.1

This SR verifies each MSRIV OPERABILITY by opening the valve and then by closing the MSRIV. This SR verifies the valve OPERABILITY in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. ~~is performed once every refueling outage on a STAGGERED TEST BASIS for each control line (i.e., twice per MSRIV) in hot shutdown conditions.~~ The Frequency for this SR is in accordance with the Inservice Testing Program. ~~is reasonable based on the fact that complete opening of an MSRIV is not possible during power operation and on the operating experience of similar MSRIVs on existing plants.~~

SR 3.7.4.2

This SR verifies each MSRCV OPERABILITY by stroking the valve through a full cycle. The test can be performed during power operation under stable conditions without impairing power operation because the MSRIV stays closed during the test. The test can also be performed in hot shutdown conditions before plant shutdown. The Frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Protection System during power operation, which ensures that the valve is not blocked in a specific position.

SR 3.7.4.3

This SR demonstrates that each MSRIV actuates on an actual or simulated steam pressure setpoint signal. The 24 month Frequency is based on the need to perform the test during either hot or cold shutdown conditions. The Frequency is reasonable based on the fact that opening a MSRIV is not possible during power operation and on operating experience of similar MSRIVs on existing plants.

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System. The EFW pumps take suction from their respective EFW storage pool (SP) (LCO 3.7.6) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1) or Main Steam Relief Trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

16-317

The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the ~~common~~ supply header. The supply header isolation valves and the discharge header isolation valves are maintained closed during normal plant operation and can be opened, as necessary, to change component alignments. The supply header isolation valves can be operated locally. The discharge header isolation valves are motor operated and also have manual hand wheels so that they can be operated from the MCR or locally.

The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each EFW pump is powered from an independent Class 1E power source.

The non-safety Startup and Shutdown System (SSS) is used for supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The EFW System is designed to supply sufficient water to the steam generator(s) (SG) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the EFW System supplies sufficient water to cool the unit to Low Head Safety Injection (LHSI) entry conditions, with steam released through the MSRTs.

The EFW System actuates automatically on low steam generator water level signal generated by the Protection System (LCO 3.3.1). The EFW System also actuates on loss of offsite power signal with safety injection.

The EFW System is discussed in FSAR Section 10.4.9 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with loss of normal feedwater supplies.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest ~~steam generator~~ MSSV set pressure plus 3%.

16-317

In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

16-317

There are four EFW trains. Each EFW train has a separate SP. All four EFW SPs, the ~~common~~ supply and discharge headers, and the four injection paths are required to be OPERABLE. One EFW pump train is assumed to be unavailable due to maintenance and a second EFW pump train or its normal injection pathway is assumed to be lost to a single failure. Note, an EFW pump train includes the pump, discharge check valve, flow control valve, and piping to the manual isolation valves on the suction and discharge of the pump.

The two remaining EFW trains provide sufficient flow for decay heat removal as required by the accident analysis. For certain sized feedwater line breaks, one of the remaining EFW pumps feeds a faulted steam generator. This pump is re-aligned from the MCR at 30 minutes to feed through the injection pathway associated with the EFW train whose pump is unavailable due to maintenance. The reactor coolant pumps in the two unfed loops are tripped.

The limiting accident for the EFW System is a Main Feedwater Line Break (MFWLB).

In addition, the minimum available EFW flow and system characteristics are considerations in the analysis of a small break loss of coolant accident (SBLOCA).

The Protection System automatically actuates the EFW pumps and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power with safety injection.

The EFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) for operation in MODES 1, 2, and 3, and Criterion 4 of 10 CFR 50.36(c)(2)(ii) for operation in MODE 4.

BASES

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary.

16-317

Four EFW pumps, the ~~common~~ supply and discharge headers, and the four injection paths are required to be OPERABLE to ensure decay heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering each of the pumps from independent emergency buses.

The EFW System is configured into four trains, which share ~~common~~ supply and discharge headers. The EFW System is considered OPERABLE when the components and ~~common~~ flow paths required to provide redundant EFW flow to the steam generators are OPERABLE. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

In MODE 4 when a steam generator is relied upon for heat removal with two EFW pumps OPERABLE, operation is allowed to continue because only two EFW pumps are required in accordance with the Note that modifies the LCO. Because of the reduced heat removal requirements and the short period of time in MODE 4 when a steam generator is relied upon for heat removal, one EFW pump is sufficient to remove decay heat. The second required pump provides single failure protection.

APPLICABILITY

In MODES 1, 2, ~~and~~ 3, and 4 when a steam generator is relied upon for heat removal, the EFW System is required to be OPERABLE in the event that it is called upon to function when MFW or offsite power are lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary inventory, lost as the unit cools to MODE 4 conditions where a steam generator is not relied upon for heat removal.

~~In MODE 4, the EFW System may be used for heat removal via the steam generators.~~

In MODES 4 when a steam generator is not relied upon for heat removal, 5 and 6, ~~the SGs are not normally used for heat removal, and~~ the EFW System is not required.

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b for two or more EFW pump trains inoperable when entering MODE 1. There is an increased risk associated with entering MODE 1 with two or more EFW pump trains inoperable and the provisions of LCO 3.0.4.b, which allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With one EFW pump train inoperable in MODE 1, 2, or 3, action must be taken to restore OPERABLE status within 120 days. The 120 day Completion Time is reasonable, based on the FSAR Chapter 15 analysis assumption that one EFW pump train is not available due to maintenance.

B.1

With two EFW pump trains inoperable in MODES 1, 2, or 3, action must be taken to restore at least one inoperable EFW pump train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period.

16-317

C.1

With a downstream injection pathway or ~~common~~ supply or discharge header inoperable in MODE 1, 2, or 3, action must be taken to restore the inoperable pathway or header to OPERABLE status in ~~72~~ 8 hours. The ~~72~~ 8 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period. ~~It is noted that the containment isolation valves in the downstream injection pathway are covered by LCO 3.6.3, "Containment Isolation Valves."~~

BASES

ACTIONS (continued)

D.1 and D.2

16-317

When Required Action A, B, or C and associated Completion Time cannot be met; or if three EFW trains are inoperable in MODE 1, 2, or 3; the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within ~~12~~ 24 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one EFW pump train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

With four EFW pump trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown and only non-safety means for conducting a cooldown with the SSS. In such a condition, the unit should not be perturbed by any action, including a power change that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one EFW pump train to OPERABLE status.

F.1 and F.2

In MODE 4, either the reactor coolant pumps or the LHSI loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one of the two required EFW pump trains inoperable when a steam generator is relied upon for heat removal, action must be taken to restore an inoperable EFW pump train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable based on redundant capabilities of the EFW System.

An alternative to Required Action F.1 is to be in MODE 4 without reliance upon a steam generator for heat removal within an additional 24 hours. The 96 hour Completion Time is reasonable based on redundant capabilities of the EFW System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-317

SR 3.7.5.4

This SR verifies that EFW can be delivered to the appropriate steam generators in the event of any accident or transient that generates a Protection System actuation, by demonstrating that each automatic valve in the flow path actuates to its correct position and each EFW pump starts automatically on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.5

This SR verifies that the EFW is properly aligned by verifying the flow path from each storage pool to its respective steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the SP to the steam generators is properly aligned.

REFERENCES

1. FSAR Section 10.4.9.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater (EFW) Storage Pools

BASES

BACKGROUND

The EFW storage pools (SP) provide a safety-related source of water to respective steam generator (SG) for removing decay and sensible heat from the Reactor Coolant System (RCS). The EFW SPs provide a passive flow of water, by gravity, to the EFW System (LCO 3.7.5). The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) (LCO 3.7.1) or the Main Steam Relief Trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

16-317

The inventory of the four EFW SPs is available to all EFW pumps through the ~~common~~ supply header. The supply header isolation valves and the discharge header isolation valves are maintained closed during normal plant operation and can be opened, as necessary, to change component alignments. The supply header isolation valves can be operated locally. The discharge header isolation valves are motor operated and also have manual hand wheels so that they can be operated from the MCR or locally. Because the EFW SPs are principal components in removing residual heat from the RCS, they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The EFW SPs are designed to Seismic Category I to ensure availability of the feedwater supply.

A description of the EFW SPs is found in FSAR Section 10.4.9 (Ref. 1).

The Demineralized Water Distribution System (DWDS), with more than 260,000 gallons available, provides the normal make-up supply to the EFW SPs. The DWDS can be aligned to the EFW SPs from the MCR. ~~If needed, the Fire Water Distribution System can be used to provide approximately 280,000 gallons of additional make-up water to the EFW SPs from standpipes located in each Safeguard Building.~~

APPLICABLE SAFETY ANALYSES

The EFW SPs provide cooling water to remove decay heat and to cool down the unit following the loss of normal feedwater supplies due to anticipated operational occurrences and accidents addressed in FSAR Chapter 15 (Ref. 3).

The limiting case for sizing of the EFW SPs is a natural circulation cooldown following a LOOP in accordance with BTP 5-4 requirements. A failed closed EFW level control valve is the assumed single failure that results in an unfed SG and stagnant RCS loop.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EFW SPs satisfy the requirements of Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The EFW SPs required usable volume of 365,000 gallons is based on a cooldown to RHR entry conditions for the bounding BTP 5-4 natural circulation cooldown described in FSAR Section 5.4.7.3.3 (Ref. 2). This basis is established in Reference 1 and exceeds the volume required by the accident analysis.

The OPERABILITY of the EFW SPs is determined by summing the available tank volumes. The volume in an SP is considered usable when it is aligned to its respective EFW pump.

16-317

APPLICABILITY

In MODES 1, 2, ~~and~~ 3, and ~~in MODE~~ 4 when a steam generator is ~~being~~ relied upon for heat removal, the EFW SPs are required to be OPERABLE to support EFW System OPERABILITY.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, ~~or~~ and 6, the EFW SPs are not required because the EFW System is not required.

ACTIONS

A.1, A.2, and A.3

16-317

With one or more of the EFW SPs inoperable in MODE 1, 2, or 3, or MODE 4 when a steam generator is ~~being~~ relied upon for heat removal, action must be taken to declare the associated EFW pump train(s) inoperable and verify the availability of the ~~back-up water supplies~~DWDS by administrative means within 4 hours and once every 12 hours thereafter. ~~The minimum volume of water available to the EFW trains is to ensure the accident analyses assumptions for cooldown rates may be achieved to bring the plant to less than 350°F within a calculated time assuming a loss of main feedwater event. The EFW SPs must be restored to OPERABLE status within 7 days.~~ The 4 hour Completion Time is reasonable, based on operating experience, to verify the required volume of water. Additionally, verifying the volume of water every 12 hours is adequate to ensure an adequate supply continues to be available. The EFW SPs must be restored to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the volume of ~~water available~~availability of the DWDS, and the low probability of an event occurring during this time period requiring the inoperable EFW SP.

B 3.7 PLANT SYSTEMS

B 3.7.19 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the UHS also provides this function for the associated safety related and non safety related systems. The safety related function is covered by this LCO.

The UHS consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, piping, valving, and instrumentation. Each safety related 2-cell Seismic Category I mechanical draft cooling tower rejects energy from the essential service water (ESW) fluid to ambient and returns the cooled fluid to the UHS cooling tower basin, from which the ESW pumps take suction. Each UHS cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related Seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA.

The mechanical draft cooling towers and basins are safety related, Seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

16-317

[The Seismic Category ~~1~~ I makeup necessary to support 30 days of post accident mitigation is site specific and details are to be provided by the COL applicant.]

Additional information about the design and operation of the UHS is presented in FSAR Section 9.2.5 (Ref. 1).

BASES

ACTIONS (continued)

C.1 and C.2

If a UHS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

16-317

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

SURVEILLANCE REQUIREMENTS

[COL applicant to provide a surveillance for makeup water to UHS cooling tower.]

SR 3.7.19.1

This SR verifies that adequate short term (3 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESW pumps during the first 3 days post LOCA. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS basin water level is ≥ 23.75 feet from the bottom of the basin.

SR 3.7.19.2

This SR verifies that the UHS is available to cool the CCW System and EDG to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a postulated accident. With water temperature of the UHS basin ≤ 90°F, the design basis assumptions associated with initial UHS temperature are bounded. With the water temperature of the UHS basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Emergency Diesel Generators (EDG) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.19.3

Operating each cooling tower fan for ≥ 15 minutes in each speed setting and direction, including reverse verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.

SR 3.7.19.4

This SR verifies proper automatic operation of the UHS cooling tower fans on an actual or simulated actuation signal. The UHS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

16-317

~~SR 3.7.19.5~~

~~This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. This SR verifies that the UHS makeup flowrate is ≥ 300 gpm.~~

BASES

16-317

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.19.5~~6~~

This SR verifies proper automatic operation of the UHS makeup valves on an actual or simulated actuation signal. The UHS is part of the ESW System, a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

16-317

[SR 3.7.19.6

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The Frequency is in accordance with the Inservice Testing Program and is in accordance with the ASME OM Code (Ref. 5). This SR verifies that the UHS makeup flowrate is ≥ 300 gpm.]

REFERENCES

1. FSAR Section 9.2.5.

2. FSAR Section 6.2.

3. FSAR Section 5.4.7.

4. Regulatory Guide 1.27, Rev. 2, January 1976.

16-317

[5. ASME Code for Operation and Maintenance of Nuclear Power Plants.]

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND	Three source range neutron flux monitors are used during refueling operations and prior to criticality to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the nuclear instrumentation system. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. 16-317 The installed source range neutron flux monitors are ~~BF3~~ boron-lined detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers the lower six decades of neutron flux (1E+6 cps). The detectors also provide visual indication in the control room and can provide an audible count rate to alert operators to a possible dilution accident. The nuclear instrumentation is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	The source range neutron flux monitors have no safety function and are not assumed to function during any design basis accident or transient analysis. However, the source range neutron flux monitors provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in the Technical Specifications. The source range neutron flux monitors satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.
APPLICABILITY	In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) Loops - High Water Level

BASES

BACKGROUND

16-317

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34~~, and to prevent boron stratification~~ (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the LHSI heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the LHSI heat exchanger(s) and the bypass. ~~Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.~~

~~An OPERABLE RHR loop includes an LHSI pump, an LHSI heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.~~

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the LHSI pump to be removed from operation for short durations, under the condition that the boron concentration is not diluted. This conditional stopping of the LHSI pump does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat; and

b. Indication of reactor coolant temperature.

16-317

An OPERABLE RHR loop includes an LHSI pump, an LHSI heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration." Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) Loops - Low Water Level

BASES

16-317

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34~~, and to prevent boron stratification~~ (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the LHSI heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the LHSI heat exchanger(s) and the bypass lines.~~ Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.~~

~~An OPERABLE RHR loop consists of an LHSI pump, an LHSI heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.~~

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, two RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat; and

b. Indication of reactor coolant temperature.

BASES

LCO (continued)

16-317

An OPERABLE RHR loop consists of an LHSI pump, an LHSI heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.4, "Residual Heat Removal (RHR) Loops – High Water Level," and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.2, B.3, and B.4

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured;

b. One door in each air lock must be closed; and