ArevaEPRDCPEm Resource

From: BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent: Wednesday, July 14, 2010 11:42 AM
To: Tesfaye, Getachew
Cc: DELANO Karen (AREVA); ROMINE Judy (AREVA); BENNETT Kathy (AREVA); RYAN Tom (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No. 293, FSAR Ch. 16 OPEN ITEM, Supplement 2
Attachments: RAI 293 Supplement 2 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the15 questions in RAI No. 293 on April 5, 2010. Supplement 1 responses to RAI No. 293 were sent on May 20, 2010, to provide a revised schedule to allow additional time for the NRC to review and discuss the draft responses to these questions with AREVA, as agreed with the NRC.

The attached file, "RAI 293 Supplement 2 Response US EPR DC.pdf" provides technically correct and complete responses to all 15 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which supports the responses.

The following table indicates the respective pages in the response document, "RAI 293 Supplement 2 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 293 — 16-294	2	2
RAI 293 — 16-295	3	3
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RAI 293 — 16-300	7	7
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RAI 293 — 16-308	19	19
RAI 293 — 16-309	20	20

This concludes the formal AREVA NP response to RAI 293, 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 (EXT)
Sent: Thursday, May 20, 2010 11:43 AM
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. 293, FSAR Ch. 16 OPEN ITEM, Supplement 1

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the15 questions in RAI No. 293 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 293 — 16-294	August 31, 2010
RAI 293 — 16-295	August 31, 2010
RAI 293 — 16-296	August 31, 2010
RAI 293 — 16-297	August 31, 2010
RAI 293 — 16-299	August 31, 2010
RAI 293 — 16-300	August 31, 2010
RAI 293 — 16-301	August 31, 2010
RAI 293 — 16-302	August 31, 2010
RAI 293 — 16-303	August 31, 2010
RAI 293 — 16-304	August 31, 2010
RAI 293 — 16-305	August 31, 2010
RAI 293 — 16-306	August 31, 2010
RAI 293 — 16-307	August 31, 2010
RAI 293 — 16-308	August 31, 2010
RAI 293 — 16-309	August 31, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
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From: BRYAN Martin (EXT)
Sent: Monday, April 05, 2010 5:10 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); PANNELL George L (AREVA NP INC); LENTZ Tony F (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 293, FSARCh. 16

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 293 Response US EPR DC.pdf" provides a schedule since a technically correct and complete response to the 15 questions is not provided.

The following table indicates the respective pages in the response document, "RAI 293 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 293 — 16-294	2	2
RAI 293 — 16-295	3	3
RAI 293 — 16-296	4	4
RAI 293 — 16-297	5	5
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RAI 293 — 16-300	7	7
RAI 293 — 16-301	8	8
RAI 293 — 16-302	9	9
RAI 293 — 16-303	10	10
RAI 293 — 16-304	11	11
RAI 293 — 16-305	12	12
RAI 293 — 16-306	13	13
RAI 293 — 16-307	14	14
RAI 293 — 16-308	15	15
RAI 293 — 16-309	16	16

A complete answer is not provided for the 15 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 293 — 16-294	May 20, 2010
RAI 293 — 16-295	May 20, 2010
RAI 293 — 16-296	May 20, 2010
RAI 293 — 16-297	May 20, 2010
RAI 293 — 16-299	May 20, 2010
RAI 293 — 16-300	May 20, 2010
RAI 293 — 16-301	May 20, 2010
RAI 293 — 16-302	May 20, 2010
RAI 293 — 16-303	May 20, 2010
RAI 293 — 16-304	May 20, 2010
RAI 293 — 16-305	May 20, 2010
RAI 293 — 16-306	May 20, 2010
RAI 293 — 16-307	May 20, 2010
RAI 293 — 16-308	May 20, 2010
RAI 293 — 16-309	May 20, 2010

Sincerely,

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Licensing Advisory Engineer
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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, October 26, 2009 10:31 AM
To: ZZ-DL-A-USEPR-DL
Cc: Le, Hien; DeMarshall, Joseph; Kowal, Mark; Hearn, Peter; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 293 (3713), FSARCh. 16

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 11, 2009, and discussed with your staff on October 22, 2009. No changes were made to the draft RAI as a result of that discussion. 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: 1689

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB7106DCF5C0)

Subject: Response to U.S. EPR Design Certification Application RAI No. 293, FSAR Ch. 16 OPEN ITEM, Supplement 2
Sent Date: 7/14/2010 11:42:25 AM
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MESSAGE	6807	7/14/2010 11:42:30 AM
RAI 293 Supplement 2 Response US EPR DC.pdf		808815

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

Response to

Request for Additional Information No. 293 (3713), Revision 0, Supplement 2

10/26/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: TS 3.4

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-294:

POTENTIAL OPEN ITEM

Follow-up to Question 16-54.

The EPR GTS, Section 3.4.1, contains limits for RCS departure from nucleate boiling (DNB). Section 3.4.1 of EPR GTS models the DNB limits in the Westinghouse STS. During its review, however, the staff noted that the applicant did not include an RCS flow limit as part of LCO 3.4.1. In the Westinghouse STS, an numerical flow limit which is based on the maximum allowable steam generator (SG) tube plugging is listed in addition to the flow limit specified in the core operating limits report (COLR). Justify not including this flow limit in the EPR GTS. In RAI 16-54, the applicant was asked to justify not specifying the minimum thermal design flow of 119692 gpm in LCO 3.4.1. This minimum flow is listed as an initial condition for design basis accident analyses in EPR FSAR subsection 15.0.0.3.1. In its response letter dated March 12, 2009, the applicant stated that since the SG tube plugging limit is provided in the STS as a bracketed value, it selects not to include the flow of 119692 gpm in the EPR GTS. The staff finds this answer inadequate. In the GTS bases, the applicant needs to state that the RCS flow limit specified in the COLR is at least equal or more restrictive than the SG tube plugging limit before the staff can determine the adequacy of LCO 3.4.1 requirements.

Response to Question 16-294:

As discussed in U.S. EPR FSAR Tier 2, Chapter 15, Section 15.0.0.3.1, and Table 15.0-63, the thermal design flow of 119,692 gpm per loop is used in the accident analysis for the RCS response. This thermal design flow is the minimum allowed by plant TSs. The analyses bound up to five percent SG tube plugging. This same value for thermal design flow will be specified in the COLR.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.1 Bases will be revised to reflect that the RCS flow specified in the COLR is greater than or equal to the RCS flow at the SG plugging limit.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.1 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-295:

POTENTIAL OPEN ITEM

Follow-up to Question 16-77

The EPR GTS, Section 3.4.6 and Section 3.4.7 contain operability requirements for RCS Loops - MODE 4 and MODE 5 with loops filled, respectively. Sections 3.4.6 and 3.4.7 of the EPR GTS model the RCS Loops, in MODE 4 and MODE 5 with loops filled, operability requirements in the Westinghouse STS. As part of its review, however, the staff noted that the applicant omitted a precautionary “Note” regarding low temperature overpressure protection (LTOP) before starting an idle reactor coolant pump (RCP). In RAI 16-77, the applicant was asked to clarify on this omission. In its response letter dated March 12, 2009, the applicant proposed to revise GTS 3.4.6 and its associated bases to include this "Note" in LCO 3.4.6. In reviewing this response, the staff believes the same Note should also be placed in LCO 3.4.7, since during planned heatup to Mode 4, at least one RCS loop can be placed in operation.

Response to Question 16-295:

A cautionary note will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.4.7, RCS Loops – MODE 5, Loops Filled.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.7 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-296:

POTENTIAL OPEN ITEM

Follow-up to Question 16-64.

In RAI 16-64, the applicant was asked to provide a discussion in the Bases 3.4.6 on Required Action A.2 and its associated completion time of 24 hours. In its response letter dated December 12, 2008, the applicant proposed to revise the EPR GTS and its associated bases to add the requested information. In reviewing this response, the staff noted that the proposed changes to the description of Condition A and Required Action A.1 are in conflict with the discussion provided for Required Actions A.1 and A.2 in the bases.

Response to Question 16-296:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.6 Bases will be revised to clarify the discussion of Actions A.1 and A.2.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.6 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-297:

POTENTIAL OPEN ITEM

Follow-up to Question 16-79

The EPR GTS, Section 3.4.9, contains operability requirements for the pressurizer. The EPR GTS 3.4.9 models similar operability requirements in the Westinghouse STS. During its review, the staff noted that the discussion on capacities of emergency powered heaters in the bases is not clear. In RAI 16-79, the applicant was asked to clarify on the designed capacity of these heaters. In its response letter dated March 12, 2009, the applicant provided the needed information including revising the bases to incorporate clarifying details. The staff finds the response acceptable, however noted an editorial error that needed to be corrected.

Response to Question 16-297:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.9 Bases will be revised to reflect that the three heater groups required to be OPERABLE contain 18 emergency powered heaters.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.9 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-299:

POTENTIAL OPEN ITEM

Follow-up to Question 16-68

The EPR GTS, Section 3.4.11, contains operability requirements for the low temperature overpressure protection (LTOP) system. The EPR GTS 3.4.11 models similar operability requirements in the Westinghouse STS. During its review, the staff noted incomplete information provided in the TS bases and relevant information in the FSAR Subsection 5.2.2. In RAI 16-68, the applicant was asked to address the missing details. In its response letter dated April 9, 2009, the applicant provided the requested details including revising affected areas in both EPR FSAR and the GTS Bases to complete the missing details. In reviewing this response, the staff, however, noted that the applicant also proposed to remove a discussion of single failure criteria applicable to the required number of PSRVs which appears to be relevant and consistent with approach discussed in the Westinghouse STS. The applicant is requested to provide further explanation on this information removal.

Response to Question 16-299:

The discussion of the need for redundant PSRVs that was removed by Question 16-68 will be restored.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.11 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-300:

POTENTIAL OPEN ITEM

Follow-up to Question 16-69

In RAI 16-69, the staff questioned the completeness of actions to resolve the MHSI miniflow line problem which causes entry into Condition A of LCO 3.4.11. In its response letter dated March 12, 2009, the applicant proposed to revise Condition C to close the action loop. The staff reviewed the proposed change and noted that the alternate Required Action C.2 is effective only for the accumulators, not for the MHSI pumps. The applicant should consider a change to Condition F to resolve this issue.

Response to Question 16-300:

Condition F for LCO 3.4.11 and the Bases discussion will be revised to provide entry from Conditions A and C.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.4.11 and the Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-301:

POTENTIAL OPEN ITEM

Follow-up to Question 5.4.2.2-3

The EPR GTS, Section 3.4.12, contains RCS operational leakage limits. The EPR GTS 3.4.12 models similar operability requirements in the Westinghouse STS. During its review, the staff noted that the applicant has adopted the NRC approved TSTF-449, Steam Generator Tube Integrity, Revision 4, in EPR GTS 3.4.12, 3.4.16 and 5.5.8, however, TSTF-449 requirements were not properly incorporated in LCO 3.4.12.d and TS 5.5.8. In RAIs 5.4.2.2-3 and 5.4.2.2-9, the applicant was asked to address these discrepancies. In its response letter dated November 14, 2008, the applicant proposed to revise LCO 3.4.12.d, TS 5.5.8 and the affected supporting information in the GTS Bases to resolve these discrepancies. The staff reviewed Revision 1 of the EPR FSAR and found that required changes were incorporated in LCO 3.4.12.d and TS 5.5.8, but conforming changes (8 places) were not accordingly made in the GTS bases (B 3.4.12 and B 3.4.16).

Response to Question 16-301:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.12 and 3.4.16 Bases will be revised to conform to TSTF-449, Revision 4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.12 and 3.4.16 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-302:

POTENTIAL OPEN ITEM

Follow-up to Question 16-81

The EPR GTS, Section 3.4.15, contains RCS Dose Equivalent I-131 and Dose Equivalent XE 133 limits. Section 3.4.15 of the EPR GTS models similar limits in the Westinghouse STS. As part of its review, however, the staff noted that the GTS Bases contain information that was different from the information contained in the NRC approved TSTF-490, Revision 0. In RAI 16-81, the applicant was asked to address these differences. In its response letter dated December 18, 2008, the applicant adequately provided the basis for the specified requirements including revising the EPR GTS Bases to include clarifying details. The staff finds the stated response acceptable, however, editorial errors still exist in the GTS bases that need to be corrected.

Response to Question 16-302:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.4.15 Bases will be revised to correct noted editorial errors.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.4.15 Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-303:

POTENTIAL OPEN ITEM

Follow-up to Question 16-62

In RAI 16-62, the applicant was asked to provide justification for the as-found tolerance of 3% for the setpoint setting of MSSVs. This 3% value is not consistent with requirements of ASME Code, Section III, NC 7000 which is listed as a reference in the EPR GTS Bases B 3.7.1. ASME Code Subsection NC 7512 requires a tolerance of 1% unless a greater tolerance is established as permissible in the Overpressure Protection Report per NC 7200.

In its response letter dated December 12, 2008, the applicant reiterated a discussion of SR 3.7.1.1 in the GTS Bases 3.7.1 which states "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," and concluded that "The ASME Code requirement is met provided the +/- 1% tolerance is met during pre-service and in-service testing." The staff disagreed with this conclusion.

In the Westinghouse STS, as discussed in the bases of LCO 3.4.6 for the pressurizer safety valves and of LCO 3.7.1 for the main steam safety valves, ASME Code requirements apply to the setpoint tolerance for OPERABILITY. Further, in the Westinghouse STS, the 3% value is bracketed as preliminary and subject to additional conformance of referenced ASME Code requirements (e.g. If the 3% value is selected for use it should be addressed in the applicable Overpressure Protection Report).

Response to Question 16-303:

The +/- 3% value in SR 3.7.1.1 and Bases will be changed to a bracketed value.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Chapter 16, Technical Specifications 3.7.1 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-304:

POTENTIAL OPEN ITEM

Follow-up to Question 16-90

In RAI 16-90, the applicant was asked to address an inconsistency between requirements of ASME Code, Section III, Article NC-7512.1 and relevant information provided in FSAR Table 15.2-1 regarding the MSSV rated lift capacity.

In its response letter dated December 12, 2008, the applicant stated "the flow area of the main steam safety valves is calculated based on rated flow rate being achieved when system pressure is equal to nominal setpoint pressure, plus 3% accumulation. The MSSV setpoint pressure used in the safety evaluation is calculated as the nominal setpoint pressure, plus 3% setpoint uncertainty (to account for drift). Since accumulation and uncertainty have the same value of 3%, rated flow will be achieved at the initial valve opening." The applicant also indicated that no change to the EPR FSAR is needed.

The staff finds the above response unacceptable in that combining an one-sided + 3% accumulation with a two-sided +/- 3% setpoint uncertainty can result in a total of + 6% difference from the nominal setpoint value. The rated lift capacity assumed in the accident analysis should be at the upper limit of the combined tolerance (e.g., at the RCS pressure of 1608.7 psia instead 1518.5 psia for the first MSSV).

Response to Question 16-304:

ASME Code, Section III, Division 1

NC-7512.1 Antichattering and Lift Requirements

(a) Safety valves shall be constructed to operate without chattering and to attain rated lift at a pressure which does not exceed the set pressure by more than 3% or 2 psi (15 kPa), whichever is greater.

NC-7512.2 Set Pressure Tolerance

(a) The set pressure tolerance plus or minus shall not exceed the following: …. 1% for pressures over 1000 psi (7000 kPa). The set pressure tolerance shall apply unless a greater tolerance is established as permissible in the Overpressure Protection Report.

The starting pressure for both the rated lift (accumulation) and tolerance (uncertainty) is based on the set pressure per the ASME code. The ASME code does not require that these requirements be stacked.

The lower and upper main steam safety valves (MSSV) have set pressures of 1460 psig and 1490 psig, respectively. Both MSSVs are attached directly to the main steam line. The U.S. EPR MSSVs use an accumulation of 3% above the set pressure before rated relieving capacity is achieved. This is the accumulation (rated lift) value specified by ASME Code, Section III, Division 1, NC-7512.1(a). A tolerance of ±3% is utilized in the overpressure analyses to account for potential set pressure drift during plant operation rather than utilizing the less limiting

ASME specified ±1% tolerance. A ±1% tolerance is utilized as the “as left” set pressure condition after testing and inspection. Per AREVA response to RAI 293, Question16-303, justification for the use of a ±3% tolerance will be included in the U.S. EPR Overpressure Protection Report as required by ASME Code, Section III, Division 1, NC-7512.2(a).

The rated relieving capacity is reached at the set pressure plus accumulation (rated lift). For the U.S. EPR, the rated relieving capacity of the MSSVs corresponds to 25% of 110% of the full power steam flow in each steam line, or 1,422,073 lb_m/hr.

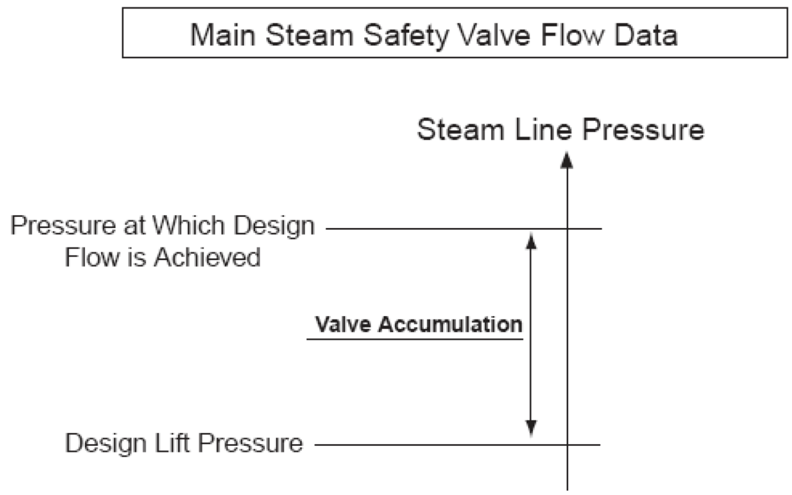
Accumulation (rated lift) for any safety valve is independent of drift. The valve accumulation (rated lift) is the design lift at which a safety valve attains its rated relieving capacity. If drift occurs above the set pressure, the steam flow will be equal to or greater than the rated relieving capacity.

At the operating nuclear plant, MSSV set pressure tolerance is established by inservice testing of the set pressure which accounts for drift or any other uncertainty.

For overpressure events, the valve is not credited to open until the inlet pressure equals the set pressure plus drift. If the valve drift is equal to the accumulation (as is the case for the U.S. EPR MSSVs), the valve is capable of relieving rated flow once the inlet pressure equals the set pressure plus drift. Since for this case the valve is opening at the same inlet pressure as the set pressure plus accumulation, only the physical time required for the valve to fully open needs to be accounted for. Safety valves typically require 20 to 30 milliseconds to fully open. A 40 millisecond delay is assumed for the opening of the U.S. EPR MSSVs.

Figures 16-304-1 and 16-304-2 provide a graphical explanation as to the application of accumulation and tolerance for the main steam safety valves. It should be noted that for the accumulation description, the design lift pressure is equal to the set pressure of the U.S. EPR MSSVs (i.e., 1460 and 1490 psig).

Figure 16-304-1. Treatment of Accumulation for Main Steam Safety Valves



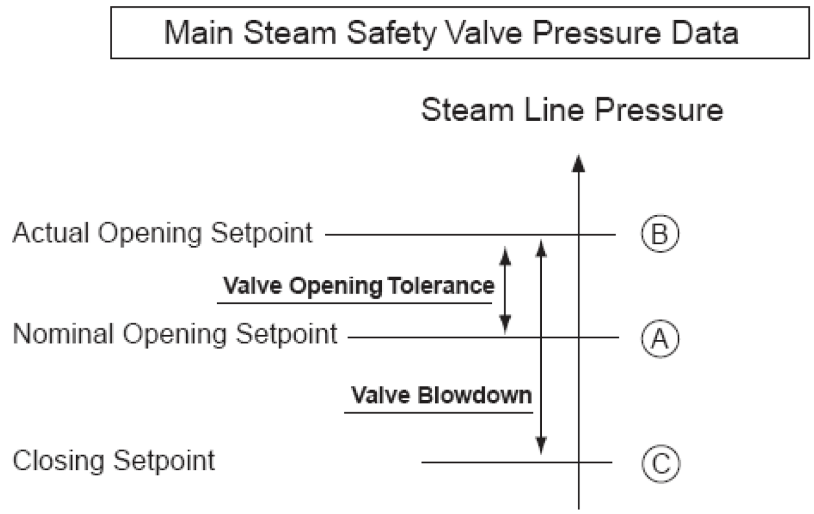
Modeling Assumptions

The MSSV valves are designed to be operable over a range of pressures. The "Design Lift Pressure" represents a nominal pressure supplied by the valve manufacturer for defining rated flow. The valve rated flow is achieved at a pressure, $P_{Accumulation}$, equal to the "Design Lift Pressure" + "Accumulation", as

$$P_{Accumulation} = P_{Design.Lift}(psig) * (1 + (\% \ Accumulation/100)) + 14.7 \quad .$$

Valve sizing calculations for the RELAP5 model define the valve flow area (mssv_area - see Database Definitions below) such that the rated flow will occur at the design lift pressure plus accumulation.

Figure 16-304-2. Treatment of Tolerance for Main Steam Safety Valves



Modeling Assumptions

1. The RELAP model and the setpoint codes (CEBASE and WBASE) assume the valve to open at point B given by the rated "Nominal Opening Setpoint, A" + "Valve Opening Tolerance" as

   $P_{open} = P_{open.nominal}(psig) * (1 + (\% tolerance/100)) + 14.7$

2. The valve closing position, C, is based on the "Actual Opening Setpoint, B, "decreased by the rated "Valve Blowdown" , as

   $P_{close} = P_{open.nominal}(psig) * (1 + (\% tolerance/100))(1- (\%blowdown/100)) + 14.7$

3. Valve sizing calculations for the RELAP5 model define the valve flow area such that the design rated flow will occur at the design lift pressure plus accumulation. The design flow and lift pressure are supplied by the valve manufacturer. The design lift pressure does not necessarily correspond to the nominal opening setpoint (above) used at the plant.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-305:

POTENTIAL OPEN ITEM

Follow-up to Question 16-291

The EPR GTS, Sections 3.7.8, combines operability requirements for both the ESW System (ESWS) and the Ultimate Heat Sink (UHS). The staff noted that, in the Westinghouse STS, separate sections are provided for the ESWS and the UHS since their respective design and operating characteristics are completely different (e.g., the UHS contains a common system parameter that affects all four ESW redundant trains). In RAI 16-291, the applicant was asked to justify this change from the STS format and content. In its response letter dated March 26, 2009, the applicant proposed to revise the EPR GTS to reflect separate sections for these two systems (TS 3.7.8 for the ESWS and TS 3.7.19 for the UHS). The staff finds this response acceptable, however editorial errors still exist that need to be corrected (e.g., a discussion of the two Notes in the action statements was deleted in error, a SR to verify automatic operation of UHS valves is missing, etc.).

Response to Question 16-305:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.7.8 Bases will be revised to add the discussion of the Notes that was deleted in error in response to RAI 16-291 and to delete Reference 4 in the Bases.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications 3.7.19 and Bases will be revised to add SR 3.7.19.6 similar to SR 3.7.8.2 for applicable valves in the UHS.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.8 and Section 3.7.19 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-306:

POTENTIAL OPEN ITEM

Follow-up to Question 16-121

The EPR GTS Section 3.7.9 includes operability requirements for the safety chilled water (SCW) system. The Westinghouse STS does not have a separate TS LCO for a safety chilled water system. In RAI 16-121, the applicant was asked to provide further explanation on a Note that requires entering only LCO 3.4.6 action statements for RHR loops made inoperable by SCW system. In the EPR FSAR Subsection 9.2.8, the SCW system is shown to provide cooling loads not only to the Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) pumps bearing coolers, but the Control Room Air Conditioning units and the Safeguard Building Ventilation Units for Electrical Divisions. In its response letter dated December 12, 2008, the applicant stated the reason for that Note is the longer completion time allowed for an inoperable SCWS train in LCO 3.7.9 when compared to that allowed for the affected RHR loop in LCO 3.4.6. The staff finds this response acceptable, however, a discussion of this Note and the reason for not to enter other LCOs 3.5.2, 3.7.11 and 3.7.13 should have been provided in the TS bases 3.7.9. Also, considering a longer completion time allowed for an inoperable CCW system train in LCO 3.7.7 when compared to that allowed for the affected SCW system train in LCO 3.7.9, the staff believes a similar Note in GTS Section 3.7.7 for the CCW system should be revised to include LCO 3.7.9 in addition to LCO 3.4.6.

Response to Question 16-306:

The response to RAI 174, Question 09.02.02-53 described changes to the safety chilled water (SCW) system that have been made to improve the reliability of the system. Cross-ties have been added to the SCW system which interconnect the supply and return piping of Train 1 and Train 2, and the supply and return piping of Train 3 and Train 4. Revised pages for the U.S. EPR FSAR Tier 2, Section 9.2.8 and Chapter 16, Technical Specification Sections 3.7.9 and Bases which described the revised SCW configuration, were included with the response to Question 09.02.02-53. As a result of this design change, the Note in TS Section 3.7.9 is no longer needed since the loss of a single SCW train will not result in the inoperability of a Low Head Safety Injection/Residual Heat Removal pump or other single train load. The note will be deleted.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.9 will be revised as described in the response and indicated on the enclosed markup.

Question 16-307:

POTENTIAL OPEN ITEM

Follow-up to Question 16-93

Section 3.5.2 of the EPR GTS contains operability requirements for the ECCS. Section 3.5.2 of the EPR GTS is modeled after the ECCS operability requirements in the Westinghouse STS. Due to the increased level of train redundancy (e.g., four trains in the EPR design versus two trains in the Westinghouse design, it is not clear to the staff that the LCO conditions as specified are equivalent to those provided in the Westinghouse STS. In RAI 16-93, the applicant was asked to provide further clarifications on LCO 3.5.2 conditions. In its response letter December 12, 2008, the applicant provided the requested information including revising Condition C and various supporting information in the bases to include the clarifying details. In reviewing this response, the staff noted that the revised discussion in the bases on Actions B.1 , B.2 and C.1 is incomplete.

Response to Question 16-307:

While the U.S. EPR GTS generally used NUREG 1431, Rev.3.1, Standard Technical Specifications (STS) for Westinghouse Plants, as a model, there are ECCS configuration differences that need to be accounted for in the U.S. EPR GTS Section 3.5.2, ECCS-Operating specification and Bases.

The typical Westinghouse plant configuration is two ECCS trains injecting into a flow header that distributes ECCS flow to the individual loops. One ECCS train is needed to accomplish the safety function. With one train inoperable, the single failure criterion is not met and 72 hours is the accepted time frame for restoring the inoperable train.

For the U.S. EPR, four independent trains powered by four emergency diesel generators are provided. Each train injects into a separate RCS loop. The flow from one train is needed to accomplish the safety function of core cooling.

Because of these system configuration differences, there is not a one-for-one correspondence with the STS. One MHSI or one LHSI train (with the cross connects open) are assumed not available by the FSAR Chapter 15 analysis and are conservatively limited to a 120 day Completion Time by Actions A and B. These Conditions have no parallel in the Westinghouse STS.

With two ECCS trains inoperable, the U.S. EPR has two trains available for injection. Since one of these two trains is conservatively assumed lost to a break, the single failure criteria is no longer met and a 72 hour Action Statement is entered. From the standpoint of ECCS flow, having two ECCS trains inoperable is equivalent to one train inoperable with an additional train assumed lost to a single failure. Since a 72 hour Action Statement is entered for two ECCS trains being inoperable, an additional single failure is not assumed.

The Bases for Conditions A.1, B.1 and B.2, and C.1 and the Applicable Safety Analyses discussion will be revised to add further clarification with one or two ECCS trains inoperable.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.5.2 will be revised as described in the response and indicated on the enclosed markup.

Question 16-308:

POTENTIAL OPEN ITEM

Follow-up to Question 16-292

In responding to RAI 142(1623)/19-269 regarding the use of Criteria 4 of 10 CFR 50.36 (c)(ii) for establishing LCO requirements in the EPR GTS, the applicant proposed three new LCOs 3.5.6, 3.5.7 and 3.5.8. These new LCOs establish controls for operation of MHSI pumps which are automatically actuated in the event of a loss of RCS inventory during Mid-Loop operation. The staff reviewed these new LCOs and, in RAI 207(2453)/16-292, requested further clarifications from the applicant regarding effectiveness of various specified actions for the described conditions in these new TSs and completeness of supporting information provided in the respective bases. In its responses letters, dated May 8, 2009 and July 5, 2009, the applicant provided the requested information including revising the EPR GTS and Bases to capture the new details. The staff finds the response acceptable, however, the proposed changes still contain editorial errors that need to be corrected.

Response to Question 16-308:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.5.7 and 3.5.8 and Bases will be revised to conform to correct noted editorial errors.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.5.7 and 3.5.8 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-309:

POTENTIAL OPEN ITEM

Follow-up to Question 16-116

Section 3.6.3 of the EPR GTS contains operability requirements for Containment isolation valves. Section 3.6.6 of the EPR GTS is modeled after similar operability requirements in the Westinghouse STS. During its review, the staff noted differences between the EPR GTS and the Westinghouse STS regarding requirements for Containment purge valves. In RAI 16-116, the applicant was asked to provide clarification on these differences. In its response letter dated December 12, 2009, the applicant provided the requested information including revising TS 3.6.3 and its associated bases to resolve these differences. The staff finds the response acceptable, however, a conforming change to the Condition D description should also have been made.

Response to Question 16-309:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification 3.6.3 and Bases will be revised to conform to correct noted editorial errors.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.6.3 and Bases will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be ≥ 20%.

--NOTES--

1. The LHSI pump may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

16-295

3. No RCP shall be started with any RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR, unless the secondary side water temperature of each steam generator (SG) is ≤ 50 °F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 5 with RCS Loops Filled.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One required PSRV inoperable in MODE 4.	D.1	Restore required PSRV to OPERABLE status.	72 hours
E. One required PSRV inoperable in MODE 5 or 6.	E.1	Restore required PSRV to OPERABLE status.	12 hours
F. Required Action and associated Completion Time of Condition A, C, D, or E not met. 16-300 OR Two required PSRVs inoperable. OR LTOP inoperable for any reasons other than Condition A, B, C, D, or E.	F.1	Depressurize RCS and establish RCS vent of ≥ 10.1 square inches.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify each accumulator is isolated.	12 hours
SR 3.4.11.2	Verify each MHSI pump capable of injecting into the RCS has its associated miniflow line open.	12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.7 In-Containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 6

LCO 3.5.7 The IRWST shall be OPERABLE.

APPLICABILITY: MODE 6 with the refueling cavity not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. IRWST~~, refueling canal, and refueling cavity~~ ~~water volume not within~~ limits.	A.1 Restore IRWST~~, refueling canal, and refueling cavity~~ ~~water volume to within limits~~.	8 hours
B. One motor-operated passive flooding line valve not closed. OR Power is not removed from one or more motor-operated passive flooding line valves.	B.1 Restore motor operated passive flooding line valves to the closed position with power removed.	1 hour
~~B. IRWST, refueling canal, and refueling cavity inoperable for reasons other than Condition A.~~	~~B.1 Restore IRWST, refueling canal, and refueling cavity to OPERABLE status.~~	~~1 hour~~

16-308

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.8 ECCS - Shutdown, MODES 5 and 6

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

16-308

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MHSI train inoperable.	A.1 Restore required MHSI train to OPERABLE status.	72 hours
B. Two required MHSI trains inoperable. ~~Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ ~~Two required MHSI trains inoperable.~~	B.1 Initiate action to restore at least one MHSI train to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met. 16-308	C.1.1 Initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥25% pressurizer level.	Immediately
	OR	
	C.1.2 Initiate action to achieve refueling cavity water level ≥ 23 feet above the reactor vessel flange.	Immediately
	AND	
	C.2 Suspend positive reactivity additions.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	-------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days
16-309 D. One or more penetration flow paths with one or more ~~full~~ purge valves not within purge valve leakage limits.	D.1 AND	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- 16-303 Verify, for each steam generator, one MSSV lift setpoint of ≥ [1416~~.2~~] psig and ≤ [1503~~.8~~] psig and other MSSV lift setpoint of ≥ [1445~~.3~~] psig and ≤ [1534~~.7~~] psig in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.9 Safety Chilled Water (SCW) System

LCO 3.7.9 Four SCW trains shall be OPERABLE and in operation.

16-306

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

ACTIONS

~~NOTE~~
~~Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SCW System.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCW train inoperable ~~or not in operation~~.	A.1 Restore SCW train to OPERABLE status ~~and in operation~~.	~~72 hours~~30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~SR 3.7.9.1 Verify each SCW train is in operation.~~	~~24 hours~~

SURVEILLANCE REQUIREMENTS

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 ~~all~~each speed setting~~s~~ 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 basin at ≥ 300 gpm.	24 months
SR 3.7.19.6	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

16-305

BASES

APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure limit and RCS average coolant temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS Pressure, Temperature, and Flow DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables – RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

16-294

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. The RCS flow specified in the COLR is greater than or equal to the RCS flow at the SG tube plugging limit (Ref. 1).

The numerical values for RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on RCS pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

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

16-296

BASES

ACTIONS (continued)

16-296

A.2

If one required RCS or one required RHR loop is inoperable and restoration is not accomplished and two RHR loops are OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only two RHR loops OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of one of the two remaining RHR loops, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

16-296

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if two RHR loops are OPERABLE. With only one RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of an RHR loop, rather than a cooldown of extended duration.

B.1 ~~and B.2~~

If two RCS or three RHR required loops are inoperable or the required loop(s) are not in operation, except during conditions permitted by the Note in the LCO section, ~~all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and~~ action to restore one RCS or two RHR loops to OPERABLE status and operation must be initiated. ~~The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.~~ The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until the required loop(s) are restored to OPERABLE status or operation.

BASES

LCO (continued)

Note 2 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

16-295

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.

BASES

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water ~~volume~~ level ≤ 59% ~~1858 cubic feet, which is equivalent to 75%,~~ 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.

16-297

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 ~~24~~ 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.

The LCO requires the CVCS charging and auxiliary spray isolation valves to be OPERABLE to 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

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. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, ~~and~~ 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 MODE 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 ~~service~~operation.~~, and therefore, the LCO is not applicable.~~

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. By having the MHSI miniflow lines open, the MHSI pumps can be made available in the event of loss of inventory.

The RCPs are administratively restricted from being started unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

16-299

The LTOP System for pressure relief consists of two PSRVs with reduced lift settings, or a depressurized RCS and an RCS vent of sufficient size. One PSRV has adequate relieving capability to keep from overpressurization for the assumed coolant input capability. Two PSRVs are required for redundancy.

PSRV Requirements for LTOP Capability

As designed for the LTOP System, each PSRV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic in Technical Specification Section 3.3. The LTOP actuation logic monitors the RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. If the indicated pressure meets or exceeds the calculated value, a PSRV is signaled to open.

The PTLR presents the PSRV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of the valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PSRV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PSRV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

In addition to the LTOP actuation logic the PSRVs are considered LTOP capable if they have the appropriate power supplies to perform their overpressure protection function.

BASES

ACTIONS (continued)

F.1

The RCS must be depressurized and a vent must be established within 6 hours when:

16-300

a. Two or more required PSRVs are inoperable;

b. Any Required Action and associated Completion Time of Condition A, C, D or E is not met; or

c. The LTOP System is inoperable for reasons other than Condition A, B, C, D, or E.

The vent must be sized ≥ 10.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability the accumulator discharge isolation valves are verified closed and the motor will be de-energized.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.11.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability the miniflow lines for each MHSI pump capable of injecting into the RCS is verified open. If the miniflow lines are

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

16-301

d. Primary to Secondary LEAKAGE Through ~~Each~~ Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.2

16-301

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one~~each~~ SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.16, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one~~each~~ SG. If it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG, in which case the LEAKAGE rate limit of 150 gallons per day would still apply.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, Rev. 1, May 2008~~ May 1973~~.

3. FSAR Chapter 15.

4. NEI 97-06, Steam Generator Program Guidelines.

5. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

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The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10CFR100.21 (Ref. 1). Doses to the control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a main steam line break (MSLB) or a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following an MSLB or an SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit. The MSLB safety analysis (Ref. 4) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit and an existing ~~reactor coolant system~~ steam generator (SG) tube leakage rate of 0.125 gpm in the affected steam generator and 0.375 gpm combined in the unaffected steam generators. The safety analysis for both accidents assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

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The ~~analysis~~ analyses for the MSLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each of the above analyses must consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases, by a factor of 500 (MSLB) and 335 (SGTR), the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after an MSLB or SGTR, respectively. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas specific activity is assumed to be 210 μCi/gm DOSE EQUIVALENT XE-133.

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These analyses also assume a loss of offsite power at the same time as the SGTR or the MSLB event. The SGTR requires operator action to initiate a manual reactor trip after 30 minutes and the loss of offsite power is assumed at this time. The MSLB causes a reduction in reactor coolant temperature and pressure. The temperature decrease causes an increase in reactor power. A reactor trip is initiated on either a low SG pressure or high SG pressure decrease.

For the SGTR and the MSLB radiological analysis, the coincident loss of offsite power causes the turbine bypass valves to close to protect the condenser. For the SGTR, a rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the main steam relief valves. A failure to close of the main steam relief valve on the affected SG is also assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until RHR System is placed in service. For the MSLB, an uncontrolled (i.e., released to atmosphere) blowdown of only one steam generator is assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until RHR System is placed in service. Radioactively contaminated steam is released to the atmosphere through the faulted SG as well as the intact SGs assuming the primary to secondary leak rates shown above.

The applicable safety analysis shows the radiological consequences of either an SGTR or MSLB accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 1~~60~~ μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The iodine specific activity in the reactor coolant is limited to 0.45 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 210 μCi/gm DOSE EQUIVALENT XE-133, as contained in SR 3.4.15.2 and SR 3.4.15.1 respectively. The limits on specific activity ensure that offsite and Control Room doses will meet the appropriate Standard Review Plan acceptance criteria (Ref. 2).

The SGTR accident analysis (Ref. 3) and the MSLB accident analysis (Ref. 4) show that the calculated dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an MSLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY

In MODES 1, 2, 3 and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENTXE-133 is necessary to limit the potential consequences of an SGTR and an MSLB to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

16-302

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that the specific activity is ≤ ~~60.0~~1 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours~~done to continue~~ to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were no iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of an MSLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-302

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.15~~16~~.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.15.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR 100.21.

2. NUREG-0800 Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

3. FSAR Section 15.0.3.7.

4. FSAR Section 15.0.3.6.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.12, "RCS Operational LEAKAGE." The dose analysis for an SGTR assumes that the ensuing radioactive release to the atmosphere is initially via the condenser and the vent stack until manual reactor trip and concurrent LOOP, and via the main steam relief trains and safety valves thereafter for the second duration.~~The accident analysis for an SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via main steam relief trains and safety valves and the majority is discharged to the main condenser.~~

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.5 gallon per minute or is assumed to increase to 0.5 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.15, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits.)

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

16-301

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair~~plugging~~ criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair~~plugging~~ criteria is removed from service by plugging. If a tube was determined to satisfy the repair~~plugging~~ criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

BASES

LCO (continued)

16-301

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.12, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one~~each~~ SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The Actions are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.16.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-301

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair~~plugging~~ criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.16.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.16.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

16-307

There are two phases of ECCS operation: injection and hot leg recirculation. In the injection phase, water is taken from the in-containment refueling water storage tank (IRWST) and injected into the Reactor Coolant System (RCS) through the cold legs. After approximately one ~~20~~ hour~~s~~, a portion of the LHSI flow is shifted to hot leg ~~recirculation~~ injection to backflush the loops, reduce the boiling in the top of the core, and recapture any boron precipitation.

Each ECCS train consists of two separate subsystems: Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI). Each ECCS flow path consists of piping, valves, one LHSI heat exchanger~~s~~, and pumps that provide a means to transfer the water from IRWST to one RCS loop cold leg. Four separate suction supply lines supply water from the IRWST to the ECCS pumps. Each ECCS train is capable of providing 100 percent of the required flow to mitigate the design basis accident (DBA). With the ECCS cross-connects closed, each of the four trains is independent and injects into a single RCS cold leg.

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

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 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 trains are cross-tied, the electrical power is required to be removed from the MOVs to ensure the valves remain in the required position.

16-307

If 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 ~~t~~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~~3~~ was lost because of a single failure, and Train 4 assumed to have the break, ~~would split its~~ flow would be lost and Train 3 would provide sufficient flow to the core.~~provide 50 percent to RCS loop 3 and 50 percent to RCS loop 4. In this configuration, no matter in which RCS loop the LOCA occurred a minimum of 150 percent of the required design flow would be delivered to the reactor core by the remaining two trains.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boil off rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the MHSI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the MHSI pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling. The cooling capability of the steam generators is enhanced by the operation of the secondary side main steam relief trains.

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

LCO

16-307

In Modes 1, 2 and 3, four~~three~~ 100% capacity independent (cross-connect closed) ECCS trains are required to ensure that sufficient ECCS flow is available. One of the four ECCS trains is assumed unavailable due to maintenance. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

An ECCS train consists of an MHSI ~~subs~~System, and an LHSI ~~subs~~System. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of injecting upon an PS signal.

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

The IRWST is sized to ensure that an adequate volume of borated water is available in the event of a DBA. The IRWST is addressed in LCO 3.5.4.

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.

This LCO is only applicable in MODE 3 and above. Below 350~~356~~°F, the PS signal setpoint for the LHSI ~~subs~~System is manually bypassed by operator control (in order to allow the alignment of the LHSI ~~subs~~System in RHR mode), and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

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 - High Water Level," and LCO 3.9.5, " LHSI / RHR and Coolant Circulation - Low Water Level."

ACTIONS

~~A.1~~

~~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.~~

16-307

~~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.1 and B.2~~

~~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. A reliability analysis (Ref. 5) has shown that the impact of having only one full ECCS train operable is sufficiently small to justify continued operation for 72 hours. 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 during the DBA.~~

BASES

ACTIONS (continued)

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.

16-307

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored.

C.1

With two ECCS trains inoperable, at least one train must be restored to OPERABLE status in 72 hours. A reliability analysis (Ref. 5) has shown that the impact of having only one full ECCS train ~~operable~~OPERABLE is sufficiently small to justify continued operation for 72 hours. This allowed ~~c~~Completion ~~t~~Time is reasonable since two trains are available and only one train is required to accomplish the safety function. Even with the most adverse configuration of two ECCS trains in the same pair (i.e. Trains 1 and 2 or Trains 3 and 4) inoperable and a cold leg break between the RCP and the vessel with the ECCS cross connects open, adequate flow is delivered to the core. With only two trains OPERABLE, the single failure criterion is not met.

BASES

ACTIONS (continued)

16-307

~~C.1~~

~~With two ECCS trains inoperable, at least one train must be restored to OPERABLE status in 72 hours. This allowed completion time is reasonable since two trains are available and only one train is required to accomplish the safety function. With only two trains OPERABLE, the single failure criterion is not met.~~

D.1 and D.2

If the inoperable train(s) cannot be returned to OPERABLE status within the associated Completion Time, 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 MODE 4 within 12 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.

E.1

Condition E is applicable with three or more trains inoperable. With less than 100% of the ECCS flow equivalent to two OPERABLE ECCS trains available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. The ECCS flow path verification includes the LHSI pump cold leg cross-connect valves are in their required position. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

BASES

ACTIONS

A.1

16-308

With IRWST, ~~refueling canal, and refueling cavity~~ water volume not within limit, it must be returned to within limit 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.1~~

~~With the IRWST inoperable for reasons other than Condition A, it must be restored to OPERABLE status within 1 hour.~~

~~In this Condition, the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a condition in which the probability and consequences of an event are minimized to the extent possible. The short time limit of 1 hour to restore the IRWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.~~

B.1

With one motor-operated passive flooding line valve not closed or with valve power not removed, a portion of the IRWST inventory could become unavailable to the ECCS pumps. In this situation, the valves must be restored to the closed position with power removed in one hour. This Completion Time is acceptable based on risk considerations.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.7.1

16-308

The IRWST, ~~refueling canal, and refueling cavity~~ should be verified every 24 hours to be within limits. The required minimum volume is verified in order to ensure that a sufficient NPSH is available for injection and to support continued ECCS pump operation. Since the IRWST volume is normally stable and is protected by an alarm, a 24 hour Frequency is appropriate and has been shown to be acceptable through operating experience

SR 3.5.7.2

The LCO 3.5.4 Surveillance Requirements and Frequencies (SR 3.5.4.1 and 3.5.4.5) are applicable to the IRWST in MODE 6. Refer to the corresponding Bases for LCO 3.5.4 for a discussion of each SR.

REFERENCES

None

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the probability and consequences of an event are minimized to the extent possible. This is done by immediately initiating action to place the plant in MODE 5 with the RCS intact with ≥ 25 % pressurizer level or, if in MODE 6, to achieve refueling cavity water level ≥ 23 feet above the reactor vessel flange. This provides adequate RCS inventory in support of RHR cooling. Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of rod cluster control assemblies, and excessive cooling of the RCS.

16-308

SURVEILLANCE REQUIREMENTS

SR 3.5.8.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

BASES

LCO (continued)

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

16-309

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. ~~The requirements for containment isolation valves during movement of recently irradiated fuel assemblies within containment are addressed in LCO 3.9.3, "Containment Penetrations."~~

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 39 inch full flow purge flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the full flow purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single full flow purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The ~~ANSI/~~ASME OM Code ~~Standard~~ 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.

16-303

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.

REFERENCES

1. FSAR Section 10.3.
2. ASME~~,~~ Boiler and Pressure Vessel Code, Section III, Article NC-7000~~, Class 2 Components~~.
3. FSAR Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ~~ANSI/ASME OM-1-1987.~~

BASES

ACTIONS

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The actions have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources Operating," should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops MODE 4," should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

If an ESW 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.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-305

SR 3.7.8.3

This SR verifies proper automatic operation of the ESW pumps ~~and cooling tower fans~~ on an actual or simulated actuation signal. The ESW System 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.

REFERENCES

1. FSAR Section 9.2.1.

2. FSAR Section 6.2.

3. FSAR Section 5.4.7.

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~~4. Regulatory Guide 1.27.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

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.

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SR 3.7.19.6

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.