ArevaEPRDCPEm Resource

From:	BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent:	Monday, December 13, 2010 10:25 AM
To:	Tesfaye, Getachew
Cc:	DELANO Karen (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); HALLINGER Pat (EXTERNAL AREVA); WILLIFORD Dennis (AREVA); Miernicki, Michael; PANNELL George (AREVA); BROWNSON Doug (AREVA)
Subject:	DRAFT Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7, 9 Questions
Attachments:	RAI 413 Supplement 2 Response US EPR DC - DRAFT.pdf

Getachew,

To support the final response dates for nine questions in RAI 413, a draft response to RAI 413 questions 07.08-15, 07.08-18, 07.08-20, 07.08-22, 07.08-23, 07.08-25, 07.08-33, 07.08-34, and 07.08-38 is attached.
Let me know if the staff has questions or if the response can be sent as final.

Thanks,

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

From: BRYAN Martin (External RS/NB)
Sent: Friday, November 19, 2010 4:51 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); PANNELL George (CORP/QP)
Subject: Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7

Getachew,

AREVA NP provided a schedule for technically complete and correct responses to the questions in RAI 413 on September 08, 2010. To provide additional time to interact with the NRC a revised schedule is provided below for questions 07.08-36, 07.08-39, and 07.08-41.

Question #	Response Date
RAI 413 07.08-10	March 15, 2011
RAI 413 07.08-11	March 15, 2011
RAI 413 07.08-12	March 15, 2011
RAI 413 07.08-13	March 15, 2011
RAI 413 07.08-14	March 15, 2011
RAI 413 07.08-15	December 17, 2010
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	December 17, 2010
RAI 413 07.08-19	January 28, 2011

RAI 413 07.08-20	December 17, 2010
RAI 413 07.08-21	January 28, 2011
RAI 413 07.08-22	December 17, 2010
RAI 413 07.08-23	December 17, 2010
RAI 413 07.08-24	January 28, 2011
RAI 413 07.08-25	December 17, 2010
RAI 413 07.08-26	December 17, 2010
RAI 413 07.08-27	December 17, 2010
RAI 413 07.08-28	December 17, 2010
RAI 413 07.08-29	January 28, 2011
RAI 413 07.08-30	January 28, 2011
RAI 413 07.08-31	January 28, 2011
RAI 413 07.08-32	January 28, 2011
RAI 413 07.08-33	December 17, 2010
RAI 413 07.08-34	December 17, 2010
RAI 413 07.08-35	January 28, 2011
RAI 413 07.08-36	**December 15, 2010**
RAI 413 07.08-37	January 28, 2011
RAI 413 07.08-38	December 17, 2010
RAI 413 07.08-39	**December 15, 2010**
RAI 413 07.08-40	January 28, 2011
RAI 413 07.08-41	**December 15, 2010**
RAI 413 07.08-42	March 15, 2011

Sincerely,

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

From: BRYAN Martin (External RS/NB)
Sent: Wednesday, September 08, 2010 4:33 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); PANNELL George (CORP/QP)
Subject: Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information RAI 413.

The following table indicates the respective pages in the response document, "RAI 413 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 413 07.08-10	2	2
RAI 413 07.08-11	3	3
RAI 413 07.08-12	4	4
RAI 413 07.08-13	5	5

RAI 413 07.08-14	6	6
RAI 413 07.08-15	7	7
RAI 413 07.08-16	8	8
RAI 413 07.08-17	9	9
RAI 413 07.08-18	10	10
RAI 413 07.08-19	11	11
RAI 413 07.08-20	12	12
RAI 413 07.08-21	13	13
RAI 413 07.08-22	14	14
RAI 413 07.08-23	15	15
RAI 413 07.08-24	16	16
RAI 413 07.08-25	17	18
RAI 413 07.08-26	19	19
RAI 413 07.08-27	20	20
RAI 413 07.08-28	21	21
RAI 413 07.08-29	22	22
RAI 413 07.08-30	23	23
RAI 413 07.08-31	24	24
RAI 413 07.08-32	25	25
RAI 413 07.08-33	26	26
RAI 413 07.08-34	27	27
RAI 413 07.08-35	28	28
RAI 413 07.08-36	29	29
RAI 413 07.08-37	30	30
RAI 413 07.08-38	31	31
RAI 413 07.08-39	32	32
RAI 413 07.08-40	33	33
RAI 413 07.08-41	34	34
RAI 413 07.08-42	35	35

A complete answer is not provided for 33 of the 33 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	**Response Date**
RAI 413 07.08-10	March 15, 2011
RAI 413 07.08-11	March 15, 2011
RAI 413 07.08-12	March 15, 2011
RAI 413 07.08-13	March 15, 2011
RAI 413 07.08-14	March 15, 2011
RAI 413 07.08-15	December 17, 2010
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	December 17, 2010
RAI 413 07.08-19	January 28, 2011
RAI 413 07.08-20	December 17, 2010
RAI 413 07.08-21	January 28, 2011
RAI 413 07.08-22	December 17, 2010
RAI 413 07.08-23	December 17, 2010
RAI 413 07.08-24	January 28, 2011
RAI 413 07.08-25	December 17, 2010
RAI 413 07.08-26	December 17, 2010

RAI 413 07.08-27	December 17, 2010
RAI 413 07.08-28	December 17, 2010
RAI 413 07.08-29	January 28, 2011
RAI 413 07.08-30	January 28, 2011
RAI 413 07.08-31	January 28, 2011
RAI 413 07.08-32	January 28, 2011
RAI 413 07.08-33	December 17, 2010
RAI 413 07.08-34	December 17, 2010
RAI 413 07.08-35	January 28, 2011
RAI 413 07.08-36	November 19, 2010
RAI 413 07.08-37	January 28, 2011
RAI 413 07.08-38	December 17, 2010
RAI 413 07.08-39	November 19, 2010
RAI 413 07.08-40	January 28, 2011
RAI 413 07.08-41	November 19, 2010
RAI 413 07.08-42	March 15, 2011

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: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, August 09, 2010 3:46 PM
To: ZZ-DL-A-USEPR-DL
Cc: Mott, Kenneth; Spaulding, Deirdre; Jackson, Terry; Canova, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 413(4772), FSAR Ch. 7

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on June 4, 2010, and discussed with your staff on July 22, 2010. Draft RAI Questions 07.08-19, 07.08-21, 07.08-23, and 07.08-41,were modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period 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: 2457

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB71086B9FF2)

Subject: DRAFT Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7, 9 Questions
Sent Date: 12/13/2010 10:24:57 AM
Received Date: 12/13/2010 10:25:07 AM
From: BRYAN Martin (EXTERNAL AREVA)

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

Recipients:
"DELANO Karen (AREVA)" <Karen.Delano@areva.com>
Tracking Status: None
"ROMINE Judy (AREVA)" <Judy.Romine@areva.com>
Tracking Status: None
"RYAN Tom (AREVA)" <Tom.Ryan@areva.com>
Tracking Status: None
"HALLINGER Pat (EXTERNAL AREVA)" <Pat.Hallinger.ext@areva.com>
Tracking Status: None
"WILLIFORD Dennis (AREVA)" <Dennis.Williford@areva.com>
Tracking Status: None
"Miernicki, Michael" <Michael.Miernicki@nrc.gov>
Tracking Status: None
"PANNELL George (AREVA)" <George.Pannell@areva.com>
Tracking Status: None
"BROWNSON Doug (AREVA)" <Douglas.Brownson@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	7473	12/13/2010 10:25:07 AM
RAI 413 Supplement 2 Response US EPR DC - DRAFT.pdf	884652	

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

Request for Additional Information No. 413(4772), Revision 1, Supplement 2

8/9/2010

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 07.08 - Diverse Instrumentation and Control Systems
Application Section: ANP-10304

QUESTIONS for Instrumentation, Controls and Electrical Engineering 1 (AP1000/EPR Projects) (ICE1)

Question 07.08-15

Provide the design basis and the detailed description of why the RCSL is credited for certain events and not credited for others.

10CFR52.47(a)(2) states that an application must contain a “description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations….”

The RCSL design descriptions do not permit the staff to sufficiently understand when to credit the RCSL for event mitigation within the “U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report” (D3-TR), ANP-10304, Revision 1 (ML093420199), event analysis. It states in Section 4.1 of the D3-TR that a “…decision [was] made not to credit the RCSL in the D3 assessment.” This position is also stated in Section A.2.2 of Appendix A. However, Section A.3.5.5, CVCS Malfunction, takes credit for RCSL control rod (RCCA) insertion to alert the control room operation of a dilution event. Therefore, the staff could not identify clear design descriptions that will permit sufficient understanding of credited accident mitigation actions for the RCSL within the D3-TR analysis.

Response to 07.08-15:

As stated in U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report (D3-TR), ANP-10304, Revision 1, Section 4.1 and Section A.2.2, the reactor control, surveillance and limitation (RCSL) is not credited in the D3 assessment for event mitigation. Although not part of the protection system (PS), the RCSL has multiple similarities with the PS; therefore, a software common cause failure (SWCCF) in the PS could potentially impact the availability of the RCSL to perform its function. For events where the RCSL causes the response to be more severe, RCSL is assumed to function. In the CVCS malfunction event cases of the RCSL available and not available are evaluated to establish the limiting condition. When RCSL functions during the CVCS malfunction event, the plant response results in the insertion of rod cluster control assemblies (RCCAs). Once the RCCAs are inserted to the power dependent insertion limit, an alarm alerts the operator that a dilution is in progress. When the RCSL does not function, the RCCAs do not move. In both cases, the BTP 7-19 acceptance criteria are met.

The RCSL is also assumed to function in the RCCA drop event. In this case, the RCSL automatically withdraws the RCCAs causing the reactor power to increase.

FSAR Impact:

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

Question 07.08-18

Provide the following additional information regarding the use of nominal DAS reactor trip setpoints in the D3 analyses:

a) Clarification of whether the DAS setpoint values listed in Table A.2-3 of ANP-10304 Rev 1 are used directly in the S-RELAP5 simulation of the D3 transient events,

b) Justification for the use of the Table A.2-3 DAS nominal trip setpoints without uncertainty in the D3 analyses, specifically addressing instrumentation channel uncertainty components such as sensor calibration tolerance and drift, basic sensor accuracy, and random process measurement uncertainty,

c) Identification of the setpoint methodology used to derive the DAS reactor trip setpoints, and

d) Identification of the design reference document that contains the DAS setpoint values.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Table A.2-3 lists the PS and DAS reactor trip setpoints. The table implies that the DAS setpoints are used directly in the D3 safety analysis with no allowance for uncertainty, and several of the evaluation results sections (e.g., Section A.3.4.2) of ANP-10304 Rev 1 state that nominal reactor trip setpoints are used in their respective D3 analysis. Protection system setpoints are normally derived from the Analytical Limits that are utilized in the safety analysis. From the Analytical Limits the Limiting Trip Setpoints (corresponding to the Limiting Safety System Settings defined in 10CFR50.36) are derived taking into account total instrumentation channel uncertainty (e.g., calibration tolerance, drift, basic sensor accuracy). Additional margin is usually added to the Limiting Trip Setpoints, resulting in Nominal Trip Setpoints. The PS setpoints listed in ANP-10304 Rev 1 Table A.2-3 are consistent with the Limiting Trip Setpoints provided in the plant Technical Specifications, FSAR Tier 2, Chapter 16 Table 3.3.1-1.

Response to 07.08-18

U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report (D3-TR), ANP-10304, Revision 1, Table A.2-3 includes the diverse actuation system (DAS) setpoint values used in the diversity and defense-in-depth (D3) transient analysis. The DAS setpoints represent nominal values and were used directly in the S-RELAP5 simulations for those events where specific analysis was performed. This approach differs from that used in the safety analysis supporting the design basis. For the design basis, protection system (PS) setpoints are derived from the analytical limits used in the safety analysis. From the analytical limits, the limiting trip setpoints, which correspond to the limiting safety system settings defined in 10CFR50.36, take into account total instrumentation channel uncertainty, such as calibration tolerance, drift, and basic sensor accuracy. The D3 analysis uses best-estimate assumptions for the DAS setpoints. These represent expected setpoints dialed-in the plant instrumentation. Because the dialed-in setting meets the Technical Specification limit, it is typically set well below the analytical limit used in the safety analysis and including uncertainties as well as administrative margin. In the D3 analysis, the DAS setpoints used represent conditions that are closer to actual plant conditions. The specific setpoints for the DAS were chosen to provide reasonable assurance they are reached only after the corresponding PS setpoint is reached.

FSAR Impact:

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

Question 07.08-20

Provide the basis or a reference for the containment structure ultimate pressure capacity stated in Section A.2.4 of Appendix A to ANP-10304 Rev 1.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.2.4 of Appendix A to ANP-10304 states that the acceptance criterion for meeting containment structure integrity is the ultimate pressure capacity of the containment. The ultimate pressure capacity is cited in Section A.2.4 to be 156 psig, or 2.52 times the containment design pressure of 62 psig. US EPR FSAR Tier 2, Table 3.8-6 lists the containment ultimate capacity as 119 psig, which is also the value used in the severe accident evaluations in FSAR Tier 2, Section 19.2.4.4.2.4.

Response to 07.08-20

The containment structure ultimate pressure capacity stated in U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report (D3-TR), ANP-10304, Revision 1, Section A.2.4 is based on a containment ultimate capacity deterministic analysis for the various containment structural elements. The containment ultimate capacity deterministic analysis was performed in accordance with the guidance provided in U.S. NRC Standard Review Plan (SRP) Section 3.8.1.II.4.K (Revision 2 – March 2007). The ultimate pressure capacity of 156 psig corresponds to the loss of structural integrity of the equipment hatch, the limiting containment structural component.

The containment structure ultimate capacity was the subject of RAI 155, Question 03.08.01-10. As a result of this RAI, U.S. EPR FSAR Tier 2, Table 3.8-6 was updated in U.S. EPR FSAR, Revision 2 as follows.

Table 3.8-6 – Containment Ultimate Pressure Capacity (P_u) at Accident Temperature of 309°F

Section	P_u (psig)	Ratio P_u/P_d	Failure Mode/ Location
Cylinder	267	4.31	Failure due to maximum allowable membrane strains away from structural discontinuities.
Dome	249	4.02	Failure due to maximum allowable membrane strains away from structural discontinuities.
Dome belt	173	2.79	Failure due to maximum allowable flexural strains at structural discontinuities.
Gusset base	315	5.08	Failure due to maximum allowable flexural strains at structural discontinuities.
Equipment hatch	156	2.52	Loss of structural integrity in protruding sleeve area due to principal strain, which approaches ultimate.
Equipment hatch	125	2.02	Loss of leak tightness in protruding sleeve area due to principal strain, which approaches ultimate.

The 119 psig value presented in U.S. EPR FSAR Tier 2, Section 19.2.4.4.2.4 will also be replaced with a reference to U.S. EPR FSAR Tier 2, Table 3.8-6.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 19.2.4.4.2.4 will be revised as described in the response and indicated on the enclosed markup.

Question 07.08-22

Provide a plot of DNBR normalized to SAFDL for the D3 analysis of the Increase in Steam Flow event, along with an explanation of the basis for the initial DNBR margin. In addition, address margin to the LPD SAFDL for the D3 analysis of the Increase in Steam Flow event.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.2.3 of Appendix A to ANP-10304 Rev 1 states that no fuel failure is predicted for the Increase in Steam Flow event. In comparing the D3 analysis of ANP-10304 Rev 1 to the corresponding FSAR Tier 2, Chapter 15 analysis, the maximum predicted power level is substantially higher in the D3 analysis (130% versus approximately 108%). Although initial DNBR margin is expected to be higher given the best-estimate assumptions of the D3 analysis, the decrease in DNBR due to the much higher power transient condition could be more than offsetting.

Response to 07.08-22

A plot of departure from nucleate boiling ratio (DNBR) and linear power density (LPD) normalized to the specified acceptable fuel design limit (SAFDL) for the beginning of cycle (BOC) increase in steam flow event is included in Figure 07.08-22-1—D3 Increase in Steam Flow – Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL below.

The Response to RAI 413, Question 07.08-19 will provide an overview of the differences between the best-estimate assumptions used in the diversity and defense-in-depth (D3) analysis as compared to the parameters within the U.S. EPR FSAR Tier 2, Chapter 15. The key items affecting the initial margin in DNBR and the LPD for the increase in steam flow event, as compared to the U.S. EPR FSAR Tier 2, Chapter 15 analysis, include the following:

a) Power Distributions

   The power distribution applied for the D3 increase in steam flow DNB and LPD analysis is a representative nominal power distribution for BOC conditions. These best-estimate power distributions are described in RAI 07.08-19 part b.

   For the FSAR safety analysis, numerous power distributions are evaluated for actuation of the low DNBR reactor trip, as described in the Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report, ANP-10287P-000.

   To initiate the increase in steam flow event for the U.S. EPR FSAR Tier 2 Chapter 15 analysis, these power distributions are scaled to the Low DNBR limiting conditions for operation (LCO), the High LPD LCO, or the FΔH LCO, whichever occurs first. For the representative case presented in U.S. EPR FSAR Tier 2, Figure 15.1-59, the power distribution was scaled from a nominal FDH of 1.385 to 1.684 (22 percent increase) to initiate the event. This event is terminated by a Low DNBR reactor trip at 7.1 seconds, which limits the reactor power to 108.7 percent of nominal.

b) Application of Uncertainties

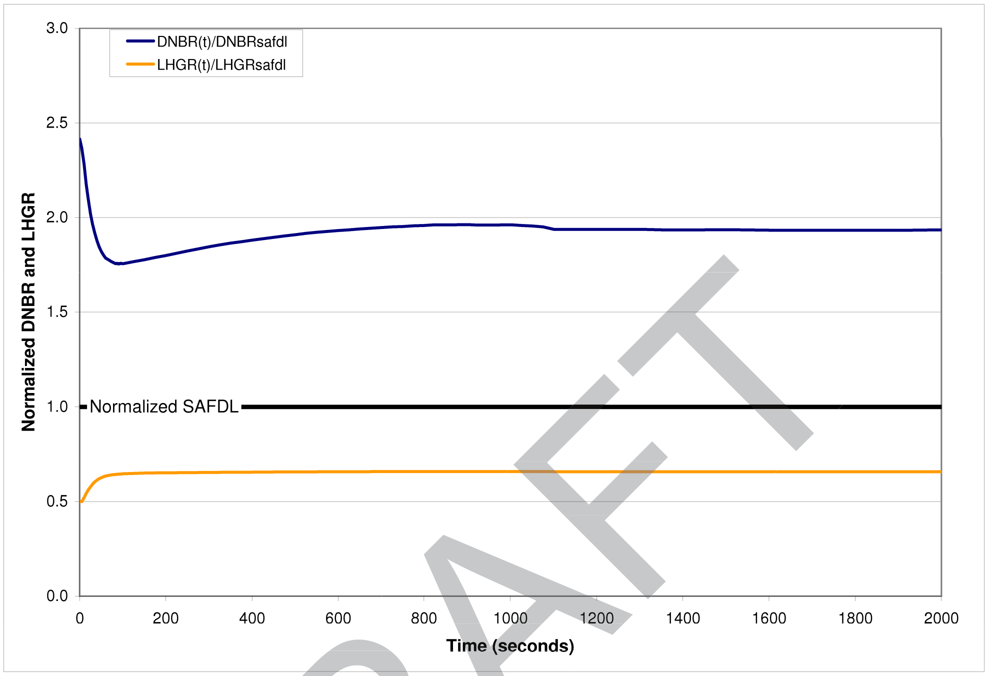
D3 DNBR analyses use best-estimate S-RELAP5 thermal-hydraulic boundary conditions within the subchannel code LYNXT to confirm the DNBR is above the technical specification safety limit of 1.0 with a 95 percent probability and 95 percent confidence (95/95) level. Since best-estimate conditions are applied for D3 DNBR analyses, the 95/95 level is the critical heat flux correlation limit with no additional uncertainties.

D3 LPD analyses use the best-estimate power distribution and the maximum reactor power, without additional uncertainties, to determine the maximum linear heat rate during the transient event. This linear heat rate is compared to the limiting linear heat rate corresponding to fuel centerline melt or 1 percent clad strain.

For the increase in steam flow analysis in the U.S. EPR FSAR Tier 2 Chapter 15, the uncertainties applicable to the Low DNBR and High LPD reactor trips are applied as described in the the Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report, ANP-10287P-000. To make sure the DNBR is above the safety limit of 1.0 with a 95 percent probability and 95 percent confidence level, uncertainties described in Section 5 of ANP-10287P-000 are applied. For LPD, the uncertainties described in Section 6 of ANP-10287P-000 are applied to make sure the LPD is less than the LPD corresponding to fuel centerline melt or 1 percent clad strain with 95 percent probability and a 95 percent confidence level.

The differences between the power distributions and the application of uncertainties applied for the D3 analysis as compared to the U.S. EPR FSAR Tier 2 Chapter 15, analysis account for the additional margin realized in the increase in steam flow analysis for D3. Both analyses demonstrate adequate margin to the safety limit.

Figure 07.08-22–1 — D3 Increase in Steam Flow – Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



FSAR Impact:

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

Question 07.08-23

For the D3 analysis provided in "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report" (D3-TR), ANP-10304, Revision 1 (ML093420199), Appendix A, the following additional information is needed for the MSLB case with PS SWCCF at full power conditions:

a) Demonstrate that a return to criticality does not occur following reactor trip,

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The FSAR Tier 2, Chapter 15 analysis identifies the main steam line break (MSLB) initiated from HZP as the limiting Steam System Piping Failure event, resulting in a small number of fuel failures due to the SAFDL on cladding strain (Local Power Density) being exceeded. [NOTE: HZP is typically more limiting than HFP because the absence of decay heat will allow greater cooldown of the RCS and therefore an increase in moderator temperature reactivity.] Therefore, additional information is needed for the case with PS SWCCF at full power conditions for the staff to complete its review.

Response to 07.08-23

The main steam line break (MSLB) event was not specifically analyzed with S-RELAP5 for the diversity and defense-in-depth (D3) assessment but was evaluated by a quantitative comparison to the FSAR Tier 2, Chapter 15 analysis using best estimate assumptions. Hot full power (HFP) was assumed as the initial condition in all the D3 assessments. HFP represents the normal plant operating condition and is consistent with best estimate conditions. To establish whether the core would return to critical following an MSLB, the temperature at which the core would be critical under best-estimate assumptions was determined. For this calculation, the PRISM reactor analysis tool was used to determine the reactor state (k_{eff}) as a function of temperature with all rods in (ARI), HFP xenon, and at end of cycle (EOC). The cases employ both thermal and Doppler feedback mechanisms to determine the reactivity response as a function of inlet temperature at isothermal conditions. This limiting set of conditions coincides with an equilibrium cycle, and the results are summarized in Table 07.08-23-1.

Table 07.08-23-1 K_{eff} Summary for MSLB Event

Moderator Temperature (°F)	Effective Multiplication Factor (k_{eff})	Reactivity	
		(Δk/k)	(pcm)
596.15	0.924745	-.081380	-8138
600	0.923059	-0.083354	-8335
500	0.953973	-0.048248	-4825
400	0.971725	-0.029097	-2910
300	0.983901	-0.016362	-1636
200	0.993078	-0.006971	-697
100	1.00018	0.000178	18

These data illustrate that the temperature at which a return to critical is expected to occur post-MSLB is ~ 105°F under best-estimate core conditions at EOC, ARI with HFP xenon conditions. This is the basis for the conclusion that a return to power would not occur following a MSLB considering a software common cause failure (SWCCF).

The difference between this analysis and that documented in the U.S. EPR FSAR Tier 2, Chapter 15 involves the use of best estimate moderator temperature coefficient (MTC), Doppler fuel temperature coefficient (DTC), and scram reactivity. The difference in these parameters between the best estimate and FSAR Tier 2, Chapter 15 values are given in Table 07.08-23-2 and Figures 07.08-23-1 through 07.08-23-3.

The U.S. EPR FSAR Tier 2, Chapter 15 values include biases and account for calculational uncertainties. The best estimate values are determined from the core analysis models for projected Cycle 1 and the equilibrium cycle. The scram reactivity use in the D3 analysis does not assume a stuck rod.

Table 07.08-23-2 Reactivity Parameters Comparison

Parameter	Best Estimate (Equilibrium Cycle)	U.S. EPR™ FSAR Tier 2, Chapter 15
MTC (pcm/°F) EOC HFP	-39.4	-50
DTC (pcm/°F) EOC HFP	-1.63	-1.85
Scram (pcm) EOC HFP	10349	7353

Figure 07.08-23-1

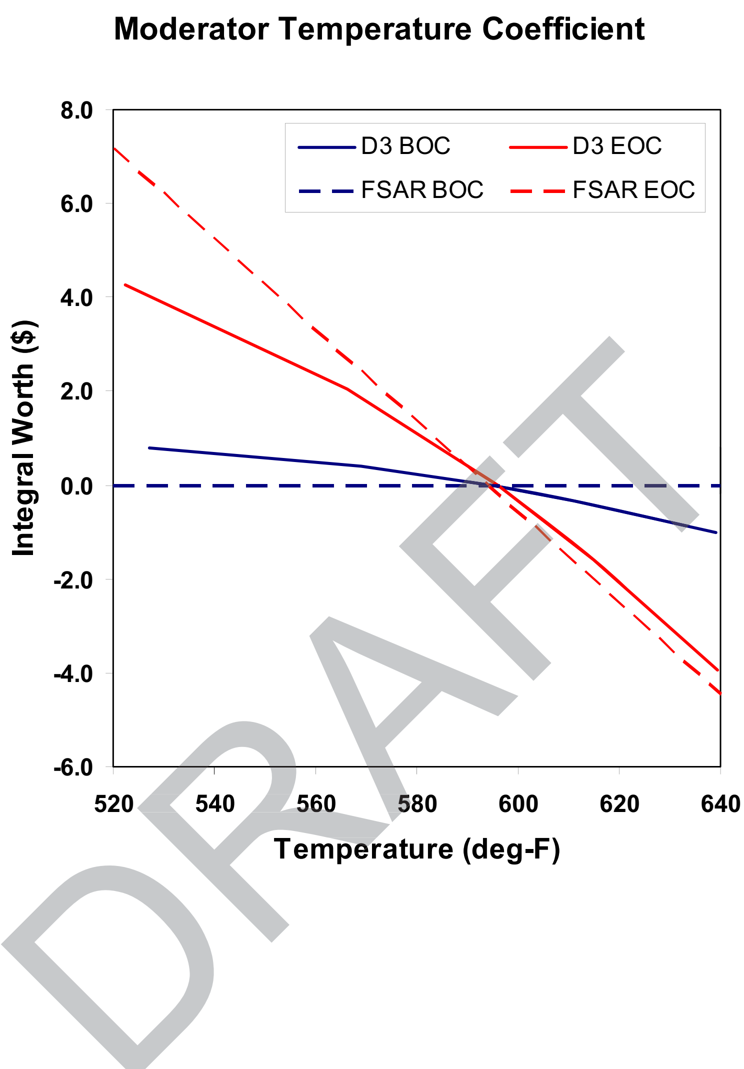


Figure 07.08-23-2

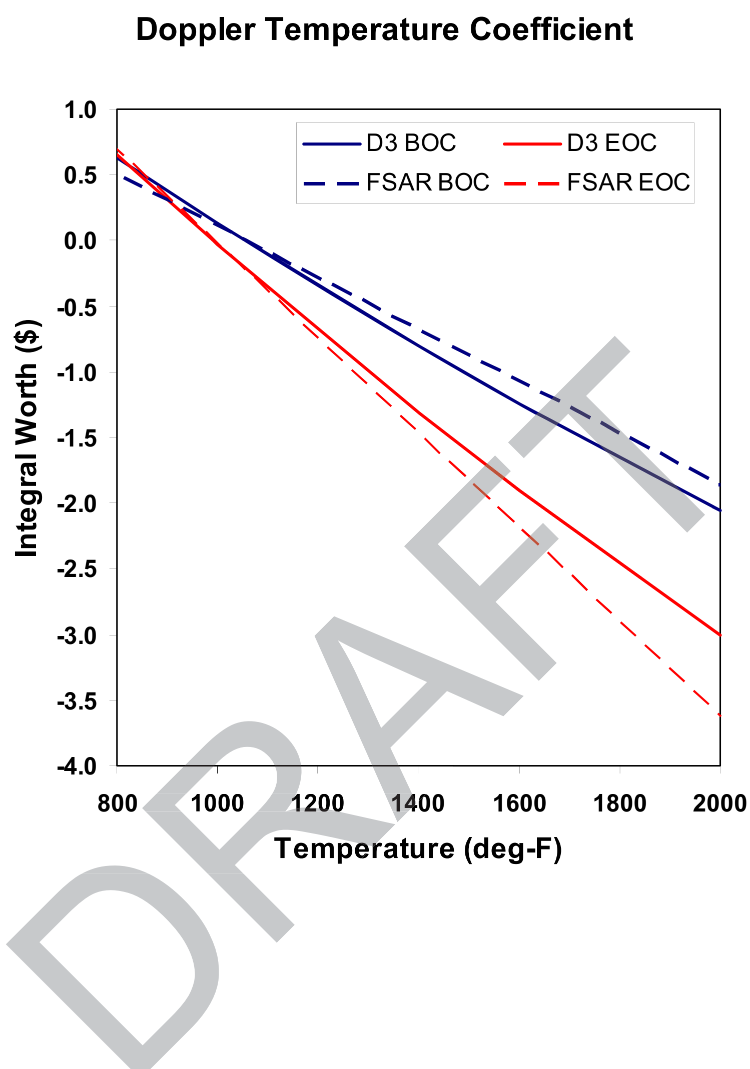
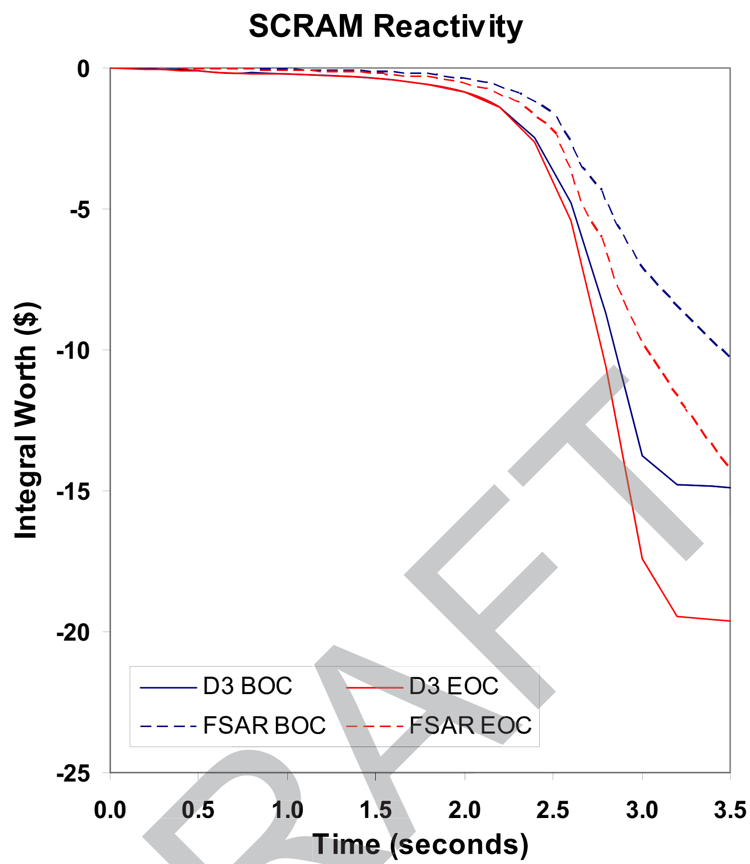


Figure 07.08-23-3



FSAR Impact:

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

Question 07.08-25

Identify the credited diverse means to address the loss of the LOOP emergency diesel generator (EDG) initiation start signal from the PS in order to operate the EFWS, in accordance with the D3 policy stated in SRM to SECY-93-087, Point 3. If the credited diverse means is a manual actuation, provide the detailed design descriptions that would address the guidance of Standard Review Plan (SRP) Appendix 18-A, "Crediting Manual Operator Actions in Diversity and Defense-In-Depth (D3) Analyses." Applying the credited diverse means, provide the following additional information:

a) Clarification of whether the EDGs or the SBO DGs are assumed to be utilized to power the EFW pumps in the D3 analysis,

b) A detailed description of how the EDGs are started following the occurrence of Loss of Non-Emergency AC Power to Station Auxiliaries event, including the following:

   I. Identification of the control systems (e.g., SICS, PICS) utilized to start the diesel generators and/or load the EFW pumps to the diesel generator buses,

   II. Identification of the location where the operator action to start diesel generators and/or load the EFW pumps to the diesel generator buses must take place, e.g., main control room or field local control panel,

   III. An assessment of the time required for the operator to achieve the necessary start of EFW, relative to the SG boil off time (reported as 1 ½ hours from the time of event initiation), and

   IV. An assessment of whether existing operating procedures (e.g., EPGs/EOPs) provide adequate symptom-based instructions to ensure the operator response assumed in the D3 analysis, or whether a special D3 coping procedure is required.

c) An evaluation of the results and consequences of the Loss of Non-Emergency AC Power to Station Auxiliaries event with SWCCF in the PS assuming that the steam generators boil dry (in approximately 1 ½ hours, according to the Section A.3.3.3 analysis) before EFW can be initiated.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events. Point 3 of the Commission's SRM to SECY 93-087, states, in part:

If a postulated common-[cause] failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-*[cause] failure, shall be required to perform either the same function as or a different function….*

The Loss of Non-Emergency AC Power to Station Auxiliaries event addressed in Section A.3.3.3 of ANP-10304 Rev 1 explains that the EDGs must be manually started for powering the EFW pumps, consistent with the unavailability of a LOOP signal from the PS. FSAR Tier 2, Section 7.3.1.2.12 states that the EDGs are manually started through SICS in the main control room, but according to ANP-10304 Rev 1 Section 4.1, SICS is not credited in the D3 analysis. Section A.3.3.3 also states that the Station Blackout diesel generators (SBO DGs) start automatically upon loss of AC power, but the EFW pumps need to be manually loaded to the SBO DGs. According to the D3 analysis, the steam generators will boil dry within approximately 1 ½ hours from the start of the event. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-25

The loss of non-emergency alternating current (AC) power to station auxiliaries event with a software common cause failure (SWCCF) in the protection system (PS) relies on operator action to start emergency feedwater (EFW) pumps. Before the EFW pumps can be started, emergency power must be available to power the pumps. Under this scenario the station blackout diesel generators (SBODG) will auto-start on loss of offsite power, but the emergency diesel generators (EDGs) need a PS signal to start. With an SWCCF in the PS, the EDGs would not receive an auto-start signal. To remove decay heat through the steam generators (SGs), two emergency feedwater (EFW) pumps are required. The two SBODGs can power two EFW pumps, while the EDGs can power four EFW pumps. In this scenario, therefore, either the SBODGs or the EDGs can be used to power the EFW pumps.

For diversity and defense-in-depth (D3), the EDGs would not start with an SWCCF in the PS. Upon failure of the EDGs to auto-start, the operator would, as a first option, manually start the EDGs. If the operator cannot manually start the EDGs during the loss of offsite power (LOOP) event, a station blackout (SBO) condition is declared and the SBODGs are connected to the associated emergency buses; the required loads, including the EFW pumps in Divisions 1 and 4, are powered. All these actions can be taken from the control room using the process information and control system (PICS) and the process automation system (PAS). The time required to start and load the EFW pumps is within the one and a half hours available before the steam generators boil dry.

At this time the U. S. EPR Emergency Procedure Guidelines/Emergency Operating Procedures are still under development. Symptom-based recovery instructions for all secondary inventory loss scenarios are envisioned so that a special D3 coping procedure will not be required.

If EFW pumps cannot be started within the one and a half hours and the SGs boil dry, alternative actions are available. Once the SGs boil dry, the primary system will initiate a heat-up. If feedwater sources cannot be recovered, the operator initiates a primary system feed and bleed. The pressurizer safety relief valves (PSRVs) opens to depressurize the primary system, while at the same time activating the medium head safety injection (MHSI) and the low head safety injection (LHSI). Decay heat is removed by the vented steam and water through the PSRVs, and the safety injection (SI) pumps would provide make-up to keep the core covered. This process could continue indefinitely with recirculation from the in-containment refueling water storage tank (IRWST) or until secondary feedwater sources are recovered.

FSAR Impact:

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

Question 07.08-33

For the large break loss of coolant accident (LBLOCA) event described in ANP-10304 Rev 1 Section A.3.7.3, provide the following additional information:

a) A summary of the sensitivity calculation results referred to in Section A.3.7.3.1 that demonstrate the conclusion that continued operation of RCPs does not significantly affect the LBLOCA results, and

b) An assessment of the effects of continued operation of the RCPs during two phase and vapor LBLOCA conditions on the RCPs themselves, i.e., cavitation or overspeed damage resulting in subsequent unavailability of the RCPs.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.7.3.1 of ANP-10304 Rev 1 states that an automatic RCP trip does not occur during a LBLOCA with SWCCF in the PS because the RCP trip is provided by the PS. It is further stated in Section A.3.7.3.1 that the RCPs are expected to continue in operation, and that such continued operation does not significantly affect the LBLOCA results, with reference to sensitivity calculation results. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-33:

Item (a):

A sensitivity calculation was performed to assess the effect of not actuating this automatic reactor coolant pump (RCP) trip on peak cladding temperature (PCT). This -study remodels the automatic RCP trip for selected cases from the U.S. EPR uncertainty analysis. These realistic large-break loss-of-coolant accident (RLBLOCA) cases are representative of an initial fuel cycle and were performed in accordance with ANP-10278, Revision 0. Since performing this sensitivity study, the RLBLOCA analysis was updated in accordance with ANP-10278, Revision 1.

The results of the sensitivity study show that not actuating the automatic RCP trip has a minor impact of <30 °F on the PCT and causes no discernible differences in break flow between original calculation cases with automatic RCP trip and new sensitivity cases without automatic RCP trip. In the RLBLOCA analysis, with the automatic RCP trip case, the pump trips at approximately 10 seconds.

Figure 07.08-33-1 is a comparison of PCT between new and original cases and shows that the deactivation of the automatic RCP trip has a minor impact (<30 °F) on the PCT calculation.

Figure 07.08-33-2 is a comparison of vessel-side break mass flow rate reactor between the new and original cases and shows that the deactivation of the automatic RCP trip has no discernible impact on vessel-side break mass flow rate.

Figure 07.08-33-3 is a comparison of RCP-side break mass flow rate (between new and original cases and shows that the deactivation of the automatic RCP trip has no discernible impact on RCP-side break mass flow rate.

Item (b):

This question was addressed by the Response to RAI 30, Question 15.06.05-17, which states that the maximum reactor coolant pump (RCP) overspeed due to a loss-of-coolant accident (LOCA) is based on the largest break size after application of the leak-before-break (LBB) analysis and not a large-break LOCA (LBLOCA). As discussed in U.S. EPR FSAR Tier 2, Section 5.4.1.4, the largest break size remaining after the application of the LBB analysis does not result in a significant overspeed. The overspeed associated with the largest break size remaining after application of LBB is less than the design RCP overspeed of 125 percent of normal operating speed even if the pumps continue to run after automatic RCP trip initiation.

The response to this question also references the Response to RAI 15 concerning ANP-10278, which provides an evaluation of the broken-loop RCP failing and locking during the realistic large break loss-of-coolant accident (RLBLOCA) transient. The referenced evaluation found a peak cladding temperature (PCT) impact less than 50ºF. The response concluded that the additional resistance associated with seizure of the pump rotor is not significant to the determination that the criteria of 10 CFR 50.46 are met with high probability.

From Item (a), subsequent unavailability of RCPs has no significant effects on the realistic large-break LOCA (RLBLOCA) event. No credit is taken for subsequent restart of the RCPs.

Figure 07.08-33-1—Comparison of PCT: RCP NO Trip (New) vs. RCP TRIP (Original)

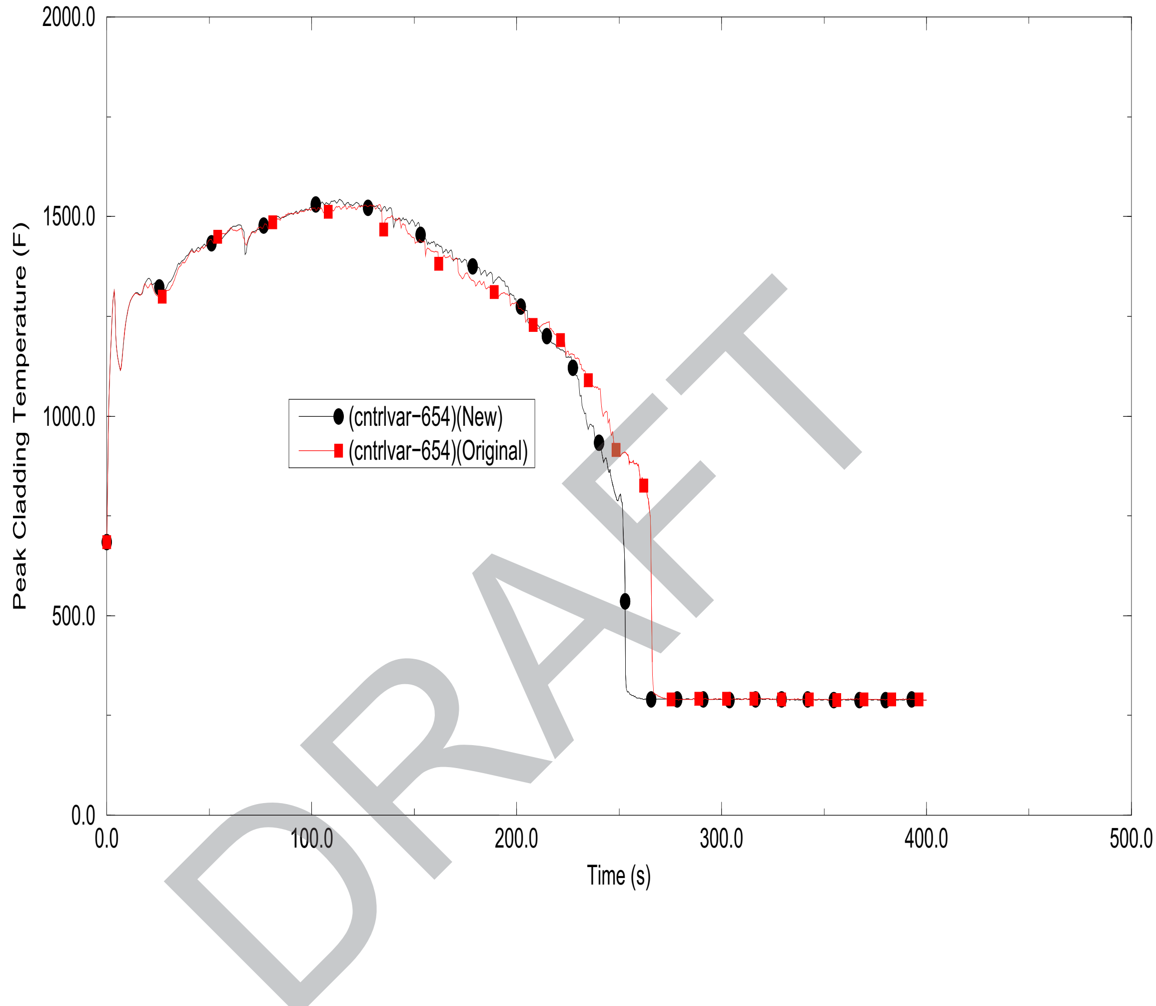


Figure 07.08-33-2—Comparison of Break Mass Flow rate (Reactor Vessel-side): RCP NO Trip (New) vs. RCP TRIP (Original)

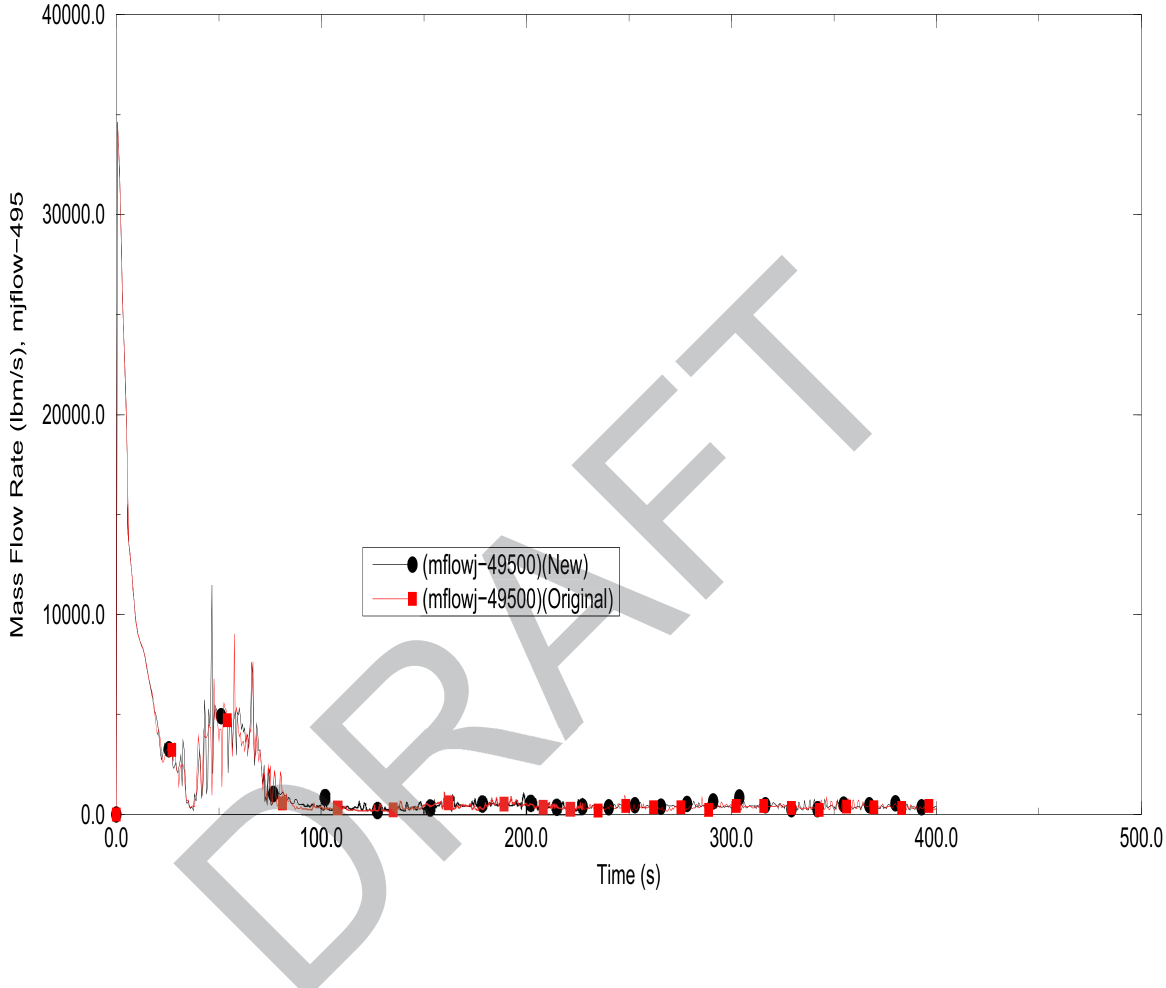
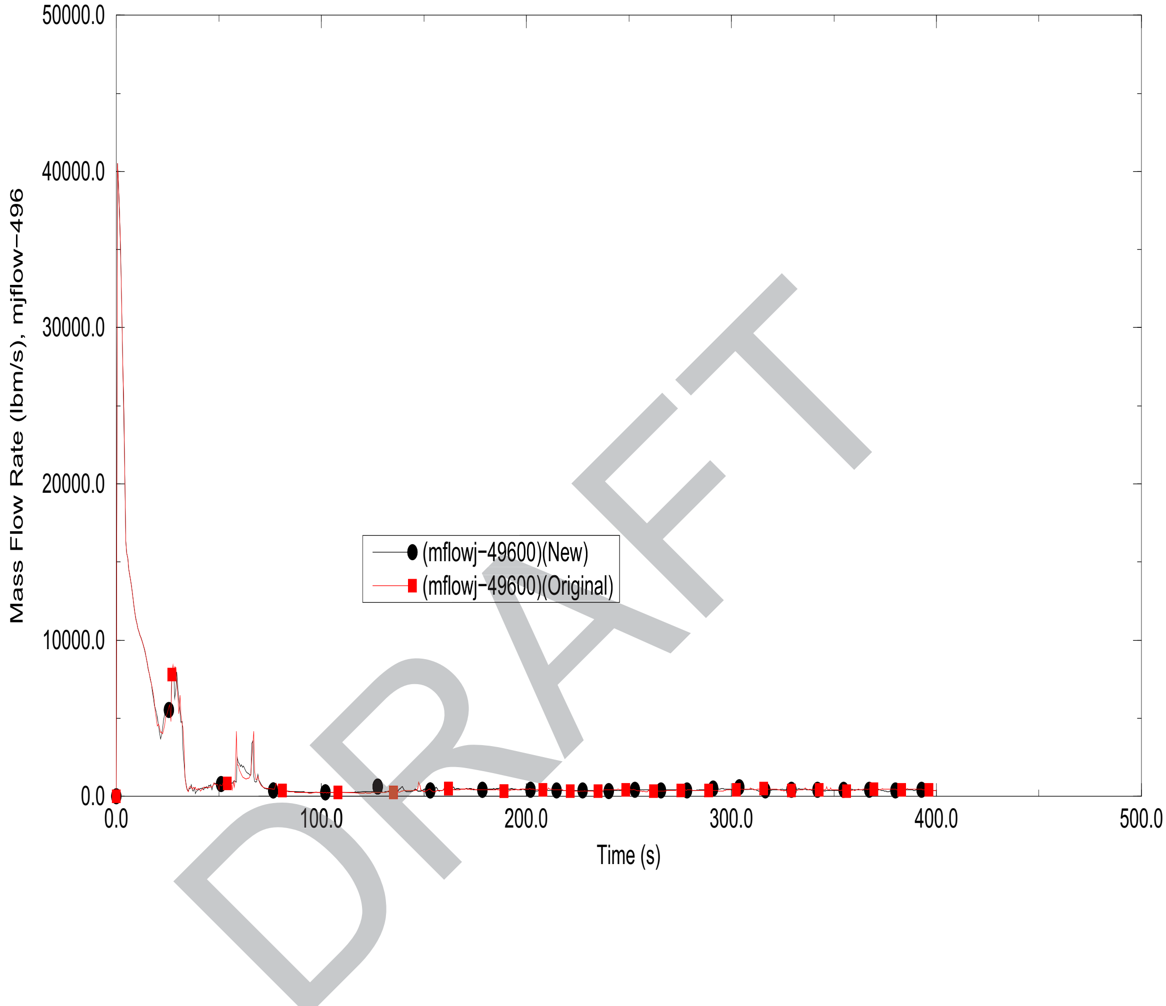


Figure 07.08-33-3—Comparison of Break Mass Flow rate (RCP-side): RCP NO Trip (New) vs. RCP TRIP (Original)



FSAR Impact:

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

Question 07.08-34

Provide a resolution of whether the turbine bypass system may be relied upon for partial cooldown during a SBLOCA with SWCCF in the PS. If manual operator actions are required to be performed, provide the timing requirements and a description of the operator actions, including whether such operator actions are covered by existing plant operating procedures or symptom-based EPGs.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

ANP-10304 Rev 1 Section A.3.7.3.2 describes a condition where the MSRT partial cooldown function is unavailable during a SBLOCA with SWCCF in the PS and the turbine bypass system provides the partial cooldown function, as long as the MSIVs remain open. The applicant notes that an on-going review could affect the use of the turbine bypass system during SBLOCA. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-34:

The U.S. EPR FSAR Tier 2, Chapter 15 analysis of the small-break loss-of-coolant accident (SBLOCA) event relies on the protection pystem (PS) to initiate the main steam relief train (MSRT) partial cooldown function. The partial cooldown facilitates reactor coolant system (RCS) depressurization and enables the injection of medium head safety injection (MHSI) for smaller breaks. The SBLOCA analysis in support of the diversity and defense-in-depth (D3) assumes a software common cause failure (SWCCF) in the PS system with the diverse actuation system (DAS) available. The partial cooldown function of the MSRT is not available in DAS.

For the scenario of an SWCCF with a complete failure of the PS, the turbine bypass system (TBS) can implement the partial cooldown function and reduce secondary system pressure similarly to the cooldown by the MSRT. The TBS-programmed cooldown function is available during the SBLOCA as long as the main steam isolation valve (MSIV) remains open. With an SWCCF that results in a partial failure of the PS such that the containment isolation on Stage 1 pressure signal isolates the MSIVs, the TBS is not available for the partial cooldown.

An SBLOCA sensitivity analysis was performed to address the SWCCF in the PS with partial cooldown unavailable by MSRT or TBS. A spectrum of break sizes ranging from 1.0 inch to 9.7 inch diameter was analyzed. The results of the analysis show the following:

Intermediate and Larger SBLOCA Cases:

SBLOCA breaks at the upper end of the spectrum rely less on the steam generators (SGs) to remove primary energy. Figure 07.08-34-1 shows that decay heat is removed through the break; the loop seals clear early in the transient, opening the path for the RCS depressurization. Figure 07.08-34-2 shows that the RCS depressurizes to the MHSI actuation signal independent of the SG ability to remove heat. Figures 07.08-34-3 and 07.08-34-4 show how as the RCS depressurizes further, the MHSI flow overcomes the break flow and replenishes the core . Subsequent injection from the LHSI and accumulators assures extended core cooling.

For this break category, the highest calculated peak cladding temperature (PCT) is 1152°F for the 9.7-inch diameter break with reactor coolant pumps (RCPs) tripped at 60 seconds.

Smaller SBLOCA Cases:

As the break size decreases toward the lower end of the spectrum, heat transfer to the SG plays a higher role. Initially, the SGs remove heat through the two banks of main steam safety valves (MSSV) (Figure 07.08-34-5). Figure 07.08-34-6 shows how primary pressure remains constant and coupled with the secondary pressure, while a small amount of heat is continually transferred from the primary to the secondary and through the banks of MSSV. This causes the secondary inventory to boil off to the atmosphere and the SGs levels to decrease to the emergency feedwater (EFW) injection setpoint, which is shown in Figures 07.08-34-7 and 07.08-34-8. The injection of the EFW colder water at 86°F causes secondary condensation and reduction in pressure, increasing the energy removal from the primary system. This effect, combined with the clearing of the loop seals, which is shown in Figure 07.08-34-9, depressurize the RCS to the MHSI injection setpoint.

Once the four MHSI trains start injecting, which is shown in Figure 07.08-34-10, their makeup capacity exceeds the inventory loss out the break, core uncovery is minimal or precluded, and the RCS inventory begins to recover, which is shown is Figure 07.08-34-11.

For breaks smaller than 2.0-inch diameter, it may take several hours for the loop seal to clear and allow primary depressurization to the MHSI injection setpoint. For these cases, the operator must take manual control of the EFW system and initiate cooldown through the MSRTs to reduce the RCS pressure and actuate the MHSI. Injection from the chemical and volume control system (CVCS), shown in Figure 07.08-34-12, is sufficient to keep the core covered prior to MHSI injection. There is sufficient time to manually initiate the cooldown for these scenarios so the partial cooldown function does not need to be automated in DAS.

For this break category, the highest PCT calculated is 1260°F for the 2.5-inch diameter break with the RCPs tripped at 60 seconds.

The results of the sensitivity analysis demonstrated that the RCS depressurization can be accomplished when the TBS is unavailable. The RCS depressurization leads to actuation of the four SI trains, which recover the RCS inventory and maintain the maximum PCT below the 10.CFR 50.46 criteria for the entire break spectrum analyzed.

Therefore, it is not necessary to rely on the TBS for partial cooldown during an SBLOCA with SWCCF in the PS, and TBS manual operator actions do not need to be performed.

Results for Larger SBLOCA Breaks: 6.0 inch DIAMETER Break

Figure 07.08-34-1—SBLOCA 6.5 inch diameter Break: Loop Seal Clearing Time

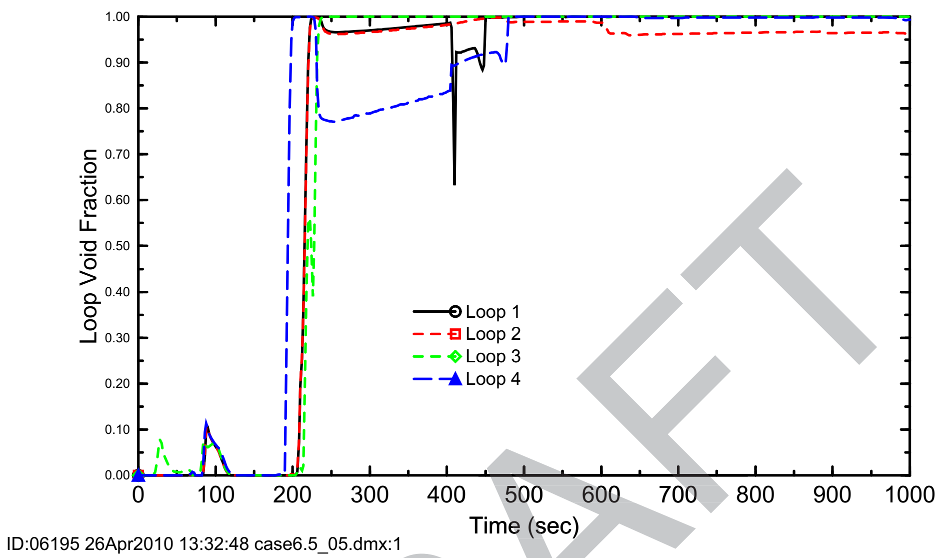


Figure 07.08-34-2: SBLOCA 6.5 inch diameter Break: Primary/ Secondary System Pressure

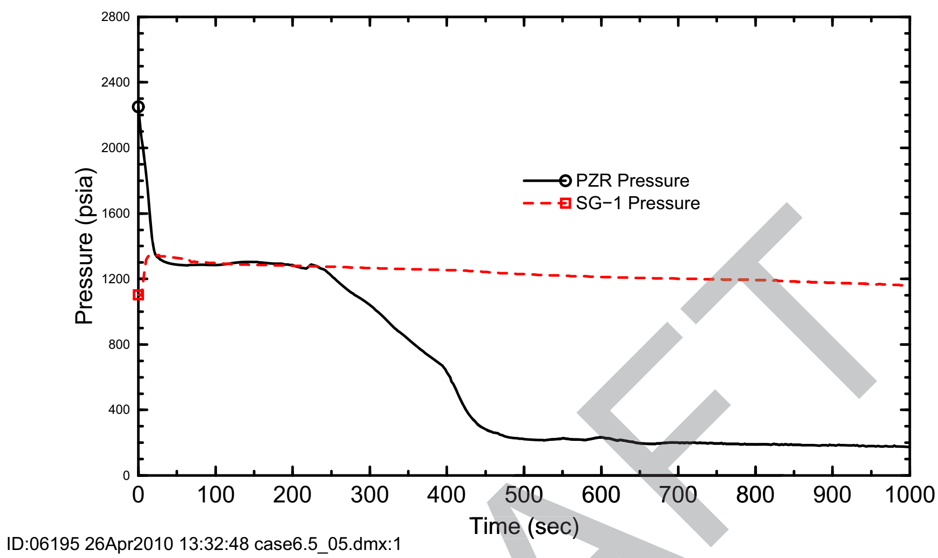


Figure 07.08-34-3—SBLOCA 6.5 inch diameter Break: MHSI Flow Rate

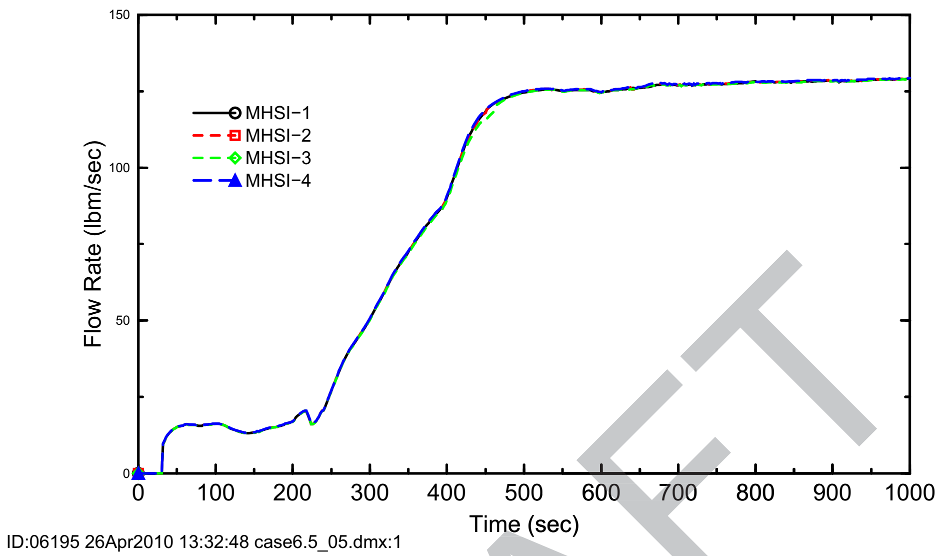
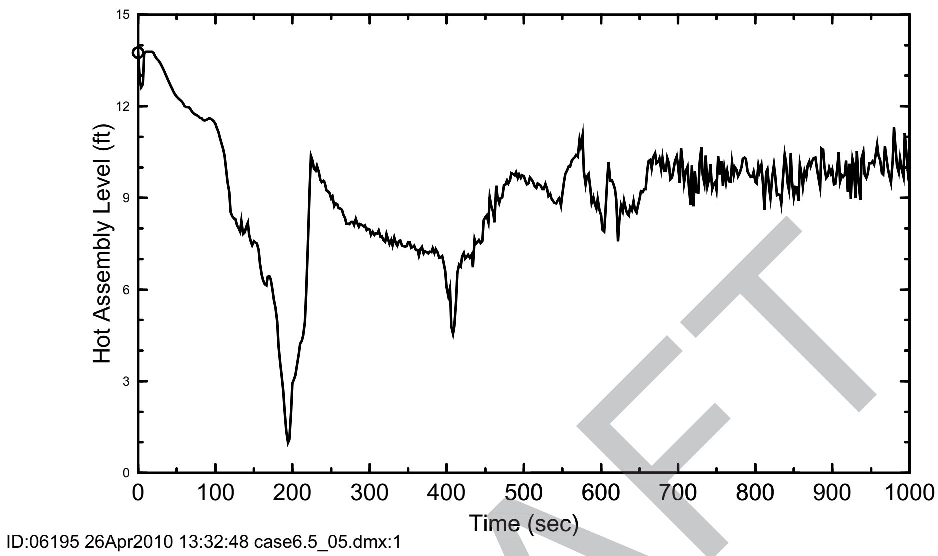


Figure 07.08-34-4—SBLOCA 6.5 inch diameter Break: Hot Assembly Collapsed Liquid Level



Results for Smaller SBLOCA Breaks: 2.5 inch DIAMETER Break

Figure 07.08-34-5—SBLOCA 2.5 inch diameter Break: MSSV-01 Mass Flow Rate

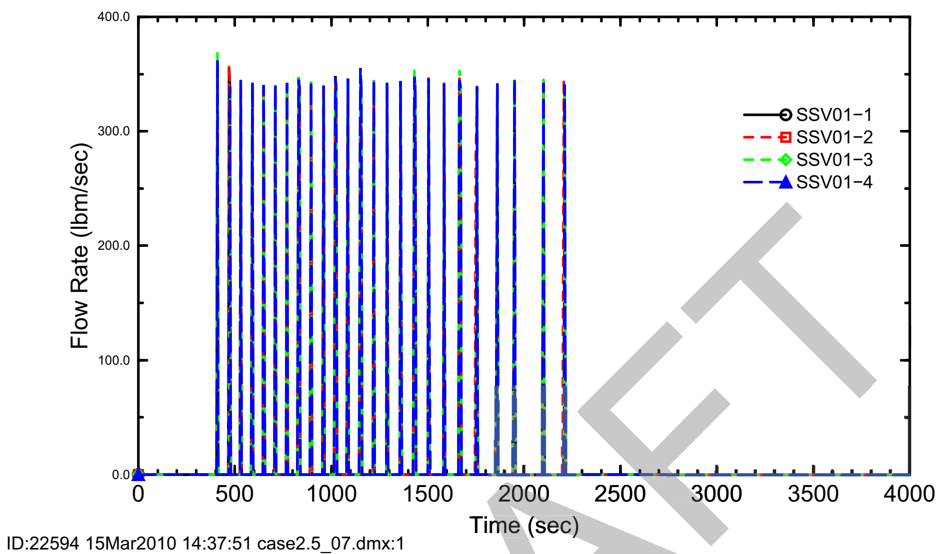


Figure 07.08-34-6—SBLOCA 2.5 inch diameter Break: Primary/ Secondary System Pressure

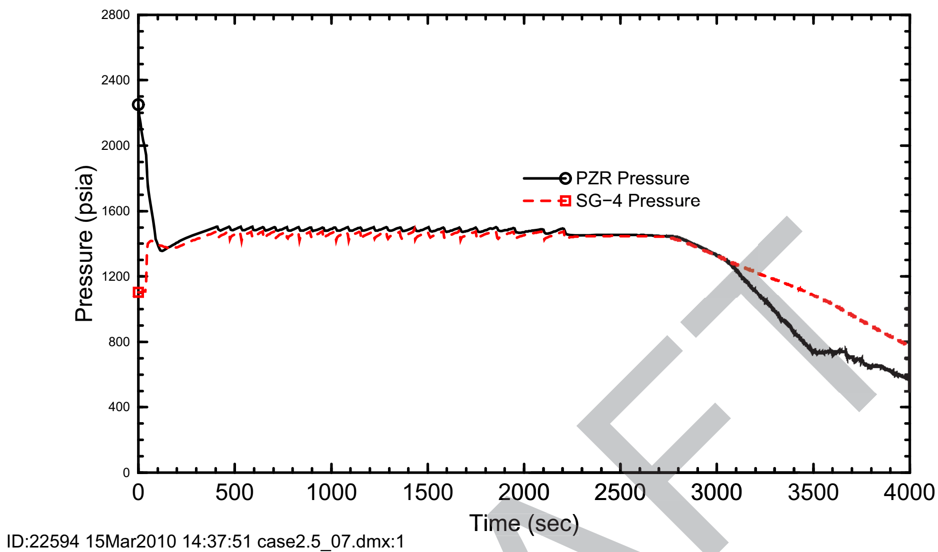


Figure 07.08-34-7—SBLOCA 2.5 inch diameter Break: Steam Generators Water Level

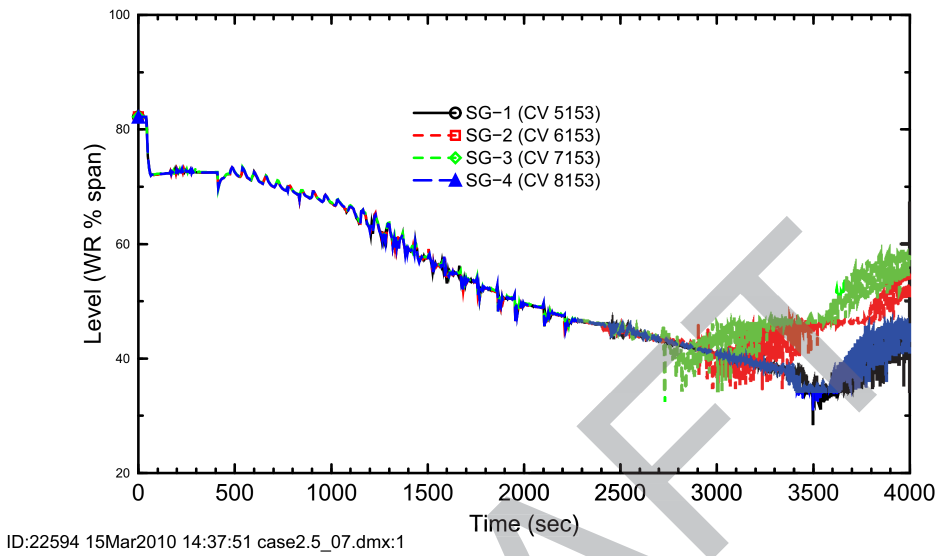


Figure 07.08-34-8—SBLOCA 2.5 inch diameter Break: EFW System Mass Flow Rate

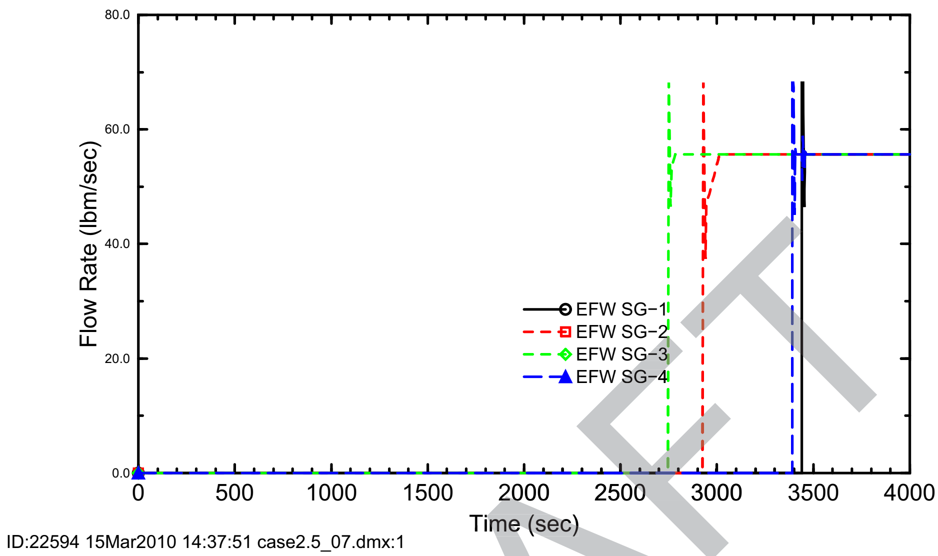


Figure 07.08-34-9—SBLOCA 2.5 inch diameter Break: Loop Seal Void Fraction

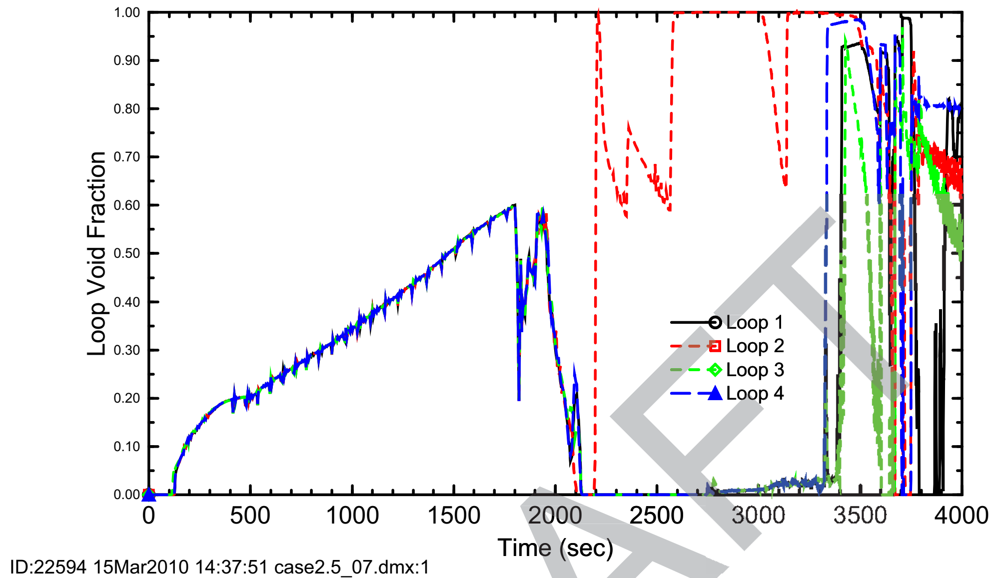


Figure 07.08-34-10—SBLOCA 2.5 inch diameter Break: MHSI System Mass Flow Rate

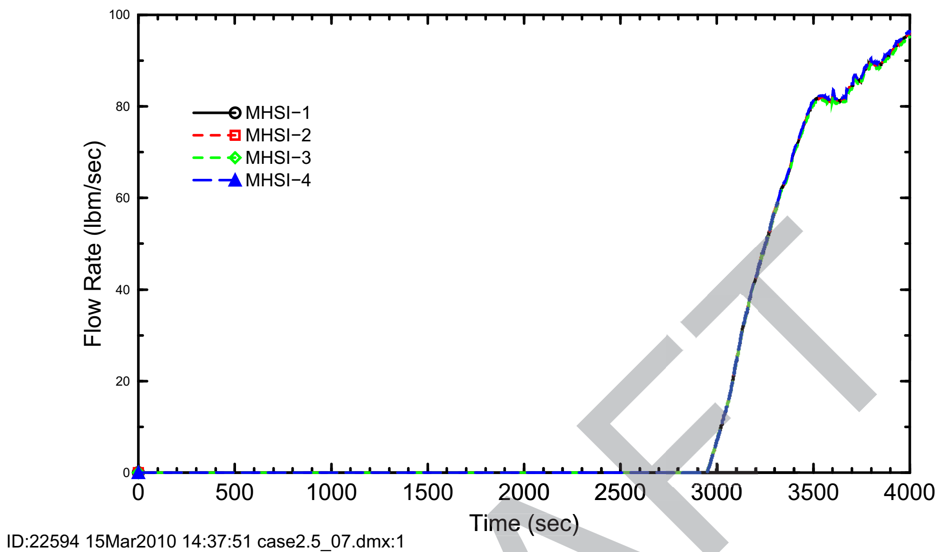


Figure 07.08-34-11—SBLOCA 2.5 inch diameter Break: RCS/ RV Mass Inventory

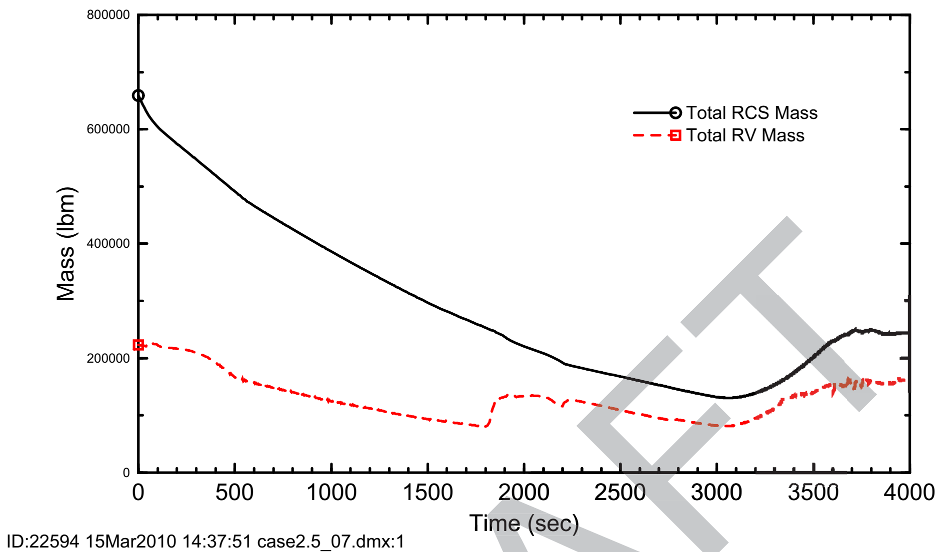
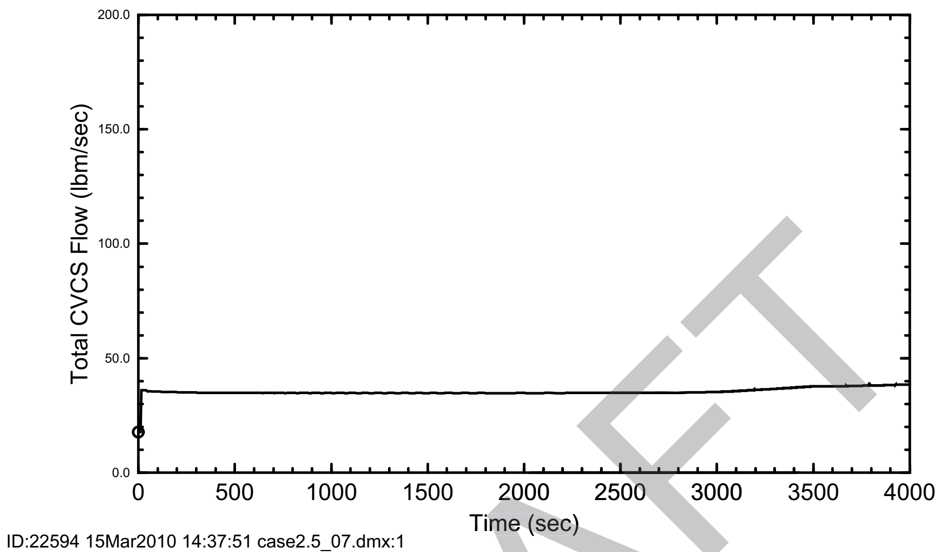


Figure 07.08-34-12—SBLOCA 2.5 inch diameter Break: Chemical and Volume Control System Flow Rate



Question 07.08-38

For Section A.3.2.3 of ANP-10304 Rev 1, describe the neutronics calculations that are done to determine the decalibration. In particular, address the issue of the temperature decrease in neutron reflector coolant. Also describe how the decalibration is implemented in the S-RELAP5 model.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.2.3 of ANP-10304 Rev 1 states that the reduction in downcomer water temperature that results from the Increase in Steam Flow event causes the excore neutron detectors to become decalibrated, thereby delaying reactor trip. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-38:

The temperature decalibration factors are determined by an independent adjoint calculation. This adjoint calculation uses the Oak Ridge National Laboratory (ORNL) programs Group-Organized Cross-Section Input Program (GIP) and Discrete Ordinates Transport (DORT). The GIP program generates 47 group neutron cross sections for the materials internal and adjacent to the U.S. EPR pressure vessel. The DORT program calculates the adjoint fluxes necessary to obtain the desired excore detector response factors for a 25ºF temperature variation around the nominal inlet coolant temperature. In addition, due to the uncertainty in the exact location of the excore detector, the factor is calculated for three different excore detector locations. The temperature decalibration factor (DF) for all the locations is calculated to be 0.51%/ºF.

In the S-RELAP5 best estimate model used for diversity and defense-in-depth (D3) analysis, the decalibration factor is applied as follows:

$$currentIndicatedPower(\%) = reactorPower(\%) + \Delta T(^\circ F) \times DF\left(\frac{\%}{^\circ F}\right) \quad (1)$$

$$where \quad \Delta T(^\circ F) = T_{calibration}^{Downcomer} - T_{current}^{Downcomer}$$

When the temperature decreases, as shown in the Increased Steam Flow event (ANP-10304 Rev 1, Figure A.3.2-2), the correction $T(^\circ F) \times DF\left(\frac{\%}{^\circ F}\right)$ is negative; the indicated reactor power is therefore lower than the current reactor power, and the reactor trip on high neutron flux is delayed as stated ANP-10304, Section A.3.2.3.

AREVA NP identified an error in formula (1) and this error was entered into the AREVA NP corrective action program. The correct formula for use in the best estimate D3 analyses is:

$$revisedIndicatedPower(\%) = reactorPower(\%) \times \left\{ 1 + \frac{\left[\Delta T(°F) \times DF\left(\frac{\%}{°F}\right) \right]}{100(\%)} \right\}$$

The power correction due to the change in downcomer temperature depends on the current reactor power. This term was omitted from the previous equation.

In the AREVA NP corrective action process, it has been established that a revision of the decalibration formula in the S-RELAP5 deck would not change the D3 analyses conclusions.

For the increased steam flow event presented in ANP-10304 (Appendix A, Section A.3.2.3), the indicated reactor power increases and stabilizes below the diverse actuation system (DAS) high neutron flux (power range) reactor trip setpoint (Figure A.3.2-1). No other reactor trip (RT) is actuated.

According to the numerical simulation below, the revised indicated power would be lower due to the negative ΔT, and the conditions would be farther from the reactor trip conditions. Thus, the transient analysis would not be affected by this correction.

If the decalibration coefficient calculated for a 25ºF variation is not applicable to the predicted -34.5ºF variation during the increased steam flow event, the extreme case is that the revised indicated power would be higher than previously calculated, and the DAS high neutron flux reactor trip setpoint would be reached. In this case, the transient would be interrupted and the consequences would be less severe than in the case presented in the U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report, ANP-10304, Section A.3.2.3.

Numerical simulation :

The calculation with the approximate final conditions of the increased steam flow event — i.e., reactor power = 131 percent and ΔT = -34.5ºF, shown in ANP-10304, Figures A.3.2-1 and A.3.2-2 — gives 113.41 percent for the current indicated power and 107.95 percent for the revised indicated power, which are less than the reactor trip setpoint of 115 percent.

FSAR Impact:

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

U.S. EPR Final Safety Analysis Report Markups

Hydrogen combustion can have two damaging effects on the containment and equipment, those resulting from either pressure or temperature. The primary function of the CGCS is to minimize the threat of combustion by maintaining the global concentration of hydrogen below 10 percent by volume, as required by 10 CFR 50.44. This is accomplished through global convection and the distribution of the PARs (which itself aids in global convection). Figure 19.2-6—Tolerance Limit Plot of Hydrogen Concentration shows that the global hydrogen concentration did not reach or exceed 10 percent by volume for any of the scenarios.

Containment structural integrity must be maintained per 10 CFR 50.44. Thus, the containment response was monitored to ensure that the pressure loads resulting from the accumulation and combustion of hydrogen did not exceed the containment ultimate capacity pressure limit. To provide reasonable assurance that structural integrity was not compromised, the containment was qualified with regard to two phenomena: (1) global hydrogen deflagration and (2) flame acceleration.

With regard to global deflagration, the AICC pressure was used as a bounding value for the pressure that would result should a single large deflagration occur. From Figure 19.2-7—Tolerance Limit Plot of Containment AICC Pressure the global maximum AICC pressure is 105 psia, for all the uncertainty cases. This does not exceed the containment ultimate capacity pressure presented in Table 3.8-6 in Section 3.8~~of approximately 119 psig (see Section 3.8.1.4.11)~~.

RAI 413 Question 07.08-20

To further address Part 50.44(c)(5), the MAAP4 computer code was used to calculate the containment thermal and pressure loads of a severe accident coincident with a combustion event. That calculation assumed 100 percent metal-water reaction of the clad surrounding the active fuel and a combustion event occurring at the moment of maximum AICC pressure.

An ANSYS thermal time history analysis of the reactor containment building subject to this internal temperature was conducted to determine tensile strength and modulus of elasticity reduction factors. Using these factors to determine concrete properties, a structural analysis of the RCB subject to the above mentioned pressure load as well as dead loads and pre-stressing loads, was also conducted using the ANSYS computer code. In addition, a separate analysis was performed to determine the effects of the pressure load on containment penetrations.

From these analyses, it is determined that the liner strains will not exceed the allowable values listed in ASME Boiler and Pressure Vessel Code Section III, Division 2, Subsubarticle CC-3720. Specifically, the Service Level C allowable membrane strain in the liner of 0.003 in./in. in tension and 0.005 in./in. in compression bounds the analytical results, demonstrating that the provisions of Part 50.44(c)(5) are met.

formation reduces the rate of steam generation; and, subsequently, the pressure rise. Eventually, heat transfer drops below that necessary to vaporize all the water entering the spreading compartment, and a water pool forms. As the liquid levels in the spreading compartment and the IRWST equalize, compartment flooding ends. The water pool temperature will rise to saturation; however, steam generation will occur at a much slower rate driven by the decay level, a level adequately mitigated by the U.S. EPR SAHRS.

Figure 19.2-15—Containment Pressure following Gate Failure shows the response of containment pressure following gate failure. This plot shows a peak appearing shortly after gate failure that corresponds to the moment when passive flooding coolant contacts the melt. A second peak follows corresponding to steady-state steaming that occurs following the water fill-up phase. The worst case in the uncertainty analysis has a maximum pressure in the containment of 74 psia. This is well below the containment ultimate pressure presented in Table 3.8-6 in Section 3.8~~of approximately 119 psig (refer to Section 3.8.1.4.11)~~.

RAI 413 Question 07.08-20

To address the SECY-93-087 containment deterministic structural performance expectation, the containment pressure and thermal loads were calculated from the five “Relevant Scenarios” listed in Section 19.2.4.2.2 plus a large break LOCA and a steam line break outside of containment. These scenarios are defined as those having a Core Damage Frequency (CDF) greater than 1.0E-8/yr., which captures categories of events covering over 95% of the CDF.

For the initial 24 hours after initiation of the accident scenario, bounding pressure and temperature time histories associated with these accident scenarios were identified as the accident scenarios with the highest temperature and largest pressure effects. The elastic model of containment described in U.S. EPR FSAR Tier 2, Section 3.8.1.4.1 was used to study these two accident scenarios. The elements associated with the liner plate, containment wall, ring girder, dome, foundation, and reactor building internal structures foundation were isolated from the overall static model. The material models used in the static model were modified to incorporate the effects of temperature on the concrete materials. Additionally, a nonlinear model was created from a 6 degree slice of the RCB liner, wall, ring girder, and dome, which implements axisymmetric boundary conditions. This nonlinear model allows concrete cracking and the tensile capability of the reinforcement bars. The nonlinear model was analyzed for dead and pre-stressing loads as well as the entire pressure time histories for the controlling accident scenarios. A separate analysis was performed to determine the effects of the pressure load on containment penetrations. The analyzed impact of the more likely accident scenarios showed that the containment remains an effective barrier against the uncontrolled release of fission products by demonstrating that the stresses or strains for containment structural elements remain within the ASME Service Level C (or Factored Load) limits.