AREVA

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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response to U.S. EPR Design Certification Application RAI No. 413, Supplement 3, Questions 07.08-24, 07.08-30, 07.08-31, 07.08-36, 07.08-37, 07.08-40, and 07.08-41

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Martin Bryan, et al (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7," August 9, 2010.

Ref. 2: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7," September 8, 2010.

Ref. 3: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7," November 19, 2010.

Ref. 4: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7, Supplement 2" December 13, 2010.

In Reference 1, the NRC provided a request for additional information (RAI) regarding the U.S. EPR design certification application (i.e., RAI No. 413). In Reference 2, AREVA NP Inc. (AREVA NP) provided a schedule indicating when the responses to RAI No. 413 would be provided. In Reference 3, and Reference 4, AREVA NP Inc. provided a revised response schedule for some of the questions in RAI No. 413. Enclosed with this letter are the final responses for questions 07.08-24, 07.08-30, 07.08-31, 07.08-36, 07.08-37, 07.08-40, and 07.08-41.

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NRO

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The following table indicates the respective pages in the response document, "RAI 413 Supplement 3 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 413 07.08-24	2	11
RAI 413 07.08-30	12	12
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In addition, a revised schedule is included in the enclosure as indicated in the table below for questions 07.08-15, and 07.08-18 thru 07.08-39.

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	March 2, 2011
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	March 2, 2011
RAI 413 07.08-19	March 31, 2011
RAI 413 07.08-20	March 2, 2011
RAI 413 07.08-21	March 2, 2011
RAI 413 07.08-22	March 2, 2011
RAI 413 07.08-23	March 2, 2011
RAI 413 07.08-25	March 2, 2011
RAI 413 07.08-26	March 31, 2011
RAI 413 07.08-27	March 2, 2011
RAI 413 07.08-28	March 2, 2011
RAI 413 07.08-29	March 31, 2011
RAI 413 07.08-32	March 31, 2011
RAI 413 07.08-33	March 2, 2011
RAI 413 07.08-34	March 2, 2011
RAI 413 07.08-35	March 2, 2011
RAI 413 07.08-38	March 2, 2011
RAI 413 07.08-39	March 2, 2011
RAI 413 07.08-42	March 15, 2011

AREVA NP considers portions of the material contained in the RAI responses listed above to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the enclosures to this letter are provided.

Sincerely,

Sandra M. Sloan

Sandra M. Sloan
Regulatory Affairs Manager, New Plants
AREVA NP Inc.

Enclosure
cc: G. Tesfaye
Docket No. 52-020

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Sandra M. Sloan. I am Manager, New Builds Regulatory Affairs for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the enclosed "Response to U.S. EPR Design Certification Application RAI No. 413" and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) “Trade secrets and commercial or financial information”.

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP’s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraph 6(b,d) above.

7. In accordance with AREVA NP’s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Sandra M. Sloan

SUBSCRIBED before me this 28th

day of January, 2011.

Kathleen A. Bennett

Kathy Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/2010
REGISTRATION #110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2011

Request for Additional Information No. 413(4772), Revision 1, Supplement 3

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 07.08-24

Provide additional information to support the reliability of the predicted DAS trip actuation on Low SG Level, including the following:

a. A description of the S-RELAP5 SG level model and the SG narrow range level instrumentation model, including validation basis, and
b. An evaluation of the decalibration effects of the MSIV closure on the narrow range instrumentation and how decalibration is treated in the simulation 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.

The engineering analysis of the Inadvertent MSIV Closure event presented in Section A.3.3.2 of ANP-10304 Rev 1 shows that a DAS reactor trip actuation occurs on Low SG Level, terminating the secondary system pressure excursion at about 1640 psia, or 100 psia below the D3 analysis criterion of 120 percent of secondary system design pressure (1.20 x 1450 psia = 1740 psia). The Inadvertent Closure of MSIV event analyzed in the FSAR results in PS reactor trip actuation (at approximately 6 seconds) on High SG Pressure, which is not available through DAS. Therefore, an explicit D3 engineering analysis is provided in ANP-10304 Rev 1 Section A.3.3.2 for the Inadvertent Closure of MSIV event. The D3 analysis shows that at approximately 130 seconds DAS provides a reactor trip actuation on Low SG Level. The pressure in the affected SG reaches approximately 1640 psia, or 113% of the secondary system design pressure (design pressure is 1450 psia, per FSAR Table 10.3-1). Considering the rate of change of the SG pressure excursion and its calculated peak value relative to the D3 analysis criterion (1640 psia peak SG pressure versus 1740 psia criterion), additional information on the SG level model and the DAS Low SG Level trip function is required in order for staff to complete its review of the Inadvertent Closure of MSIV D3 analysis.

Response to 07.08-24

Item 07.08-24(a):

Description of S-RELAP5 Narrow Range (NR) and Wide Range (WR) steam generator (SG) Level Model

The S-RELAP5 model for the best estimate non loss-of-coolant accident (non-LOCA) analyses implements a mechanistic tap-to-tap differential pressure measurement of the SG narrow range and wide range level signals. This scheme is used only in the best estimate non-LOCA model, where steam generator (SG) pressure can rise rapidly following main steam isolation valve (MSIV) closure or turbine trip. This can potentially cause SG level shrink that may not be tracked with complete accuracy by the more conventional collapsed liquid level scheme used in the U.S. EPR FSAR Tier 2, Chapter 15 analyses. Since nominal instrument setpoints are used in the best estimate analyses, capturing the SG level shrink/swell phenomena, if it occurs, is an important consideration. The best estimate small-break LOCA (SBLOCA) S-RELAP5 model retains the collapsed liquid level scheme used in the U.S. EPR FSAR Tier 2, Chapter 15 design basis analyses because rapid SG secondary pressure and level changes are not expected for SBLOCA scenarios.

The tap-to-tap differential pressure is comprised of the constituent liquid and vapor static heads, as follows:

$$P_{lower\ \mathrm{tap}} - P_{upper\ \mathrm{tap}} = \rho_f g \Delta z_f + \rho_g g \Delta z_g \,, \tag{1}$$

where the effective heights of the liquid and vapor constituents sum to the overall tap-to-tap elevation difference:

$$\Delta z_f + \Delta z_g = z_{upper\ \mathrm{tap}} - z_{lower\ \mathrm{tap}} \tag{2}$$

The steam generator liquid level is indicated as the effective height of the liquid, which is expressed as a percentage of the tap-to-tap span, is as follows:

$$\text{Liquid level} = \frac{\Delta z_f}{z_{\text{upper tap}} - z_{\text{lower tap}}} \times 100\% \tag{3}$$

The effective height of the liquid, in terms of the tap-to-tap differential pressure, for substitution in this relation may be obtained from equation (1), by first solving equation (2) for the effective height of the vapor and substituting it into equation (1),as follows:

$$\begin{aligned} P_{lower\ \mathrm{tap}} - P_{upper\ \mathrm{tap}} &= \rho_f g \Delta z_f + \rho_g g \left(z_{upper\ \mathrm{tap}} - z_{lower\ \mathrm{tap}} - \Delta z_f \right) \\ &= \left(\rho_f - \rho_g \right) g \Delta z_f + \rho_g g \left(z_{upper\ \mathrm{tap}} - z_{lower\ \mathrm{tap}} \right), \end{aligned}$$

which may be rearranged to obtain:

$$\Delta z_f = \frac{P_{lower\ \mathrm{tap}} - P_{upper\ \mathrm{tap}} - \rho_g g \left(z_{upper\ \mathrm{tap}} - z_{lower\ \mathrm{tap}} \right)}{\left(\rho_f - \rho_g \right) g}$$

By incorporating this result into equation (3) and simplifying, the liquid level may be expressed as:

$$\text{Liquid level} = \left(\frac{P_{\text{lower tap}} - P_{\text{upper tap}}}{z_{\text{upper tap}} - z_{\text{lower tap}}} - \rho_g g \right) \times \frac{100\%}{\left(\rho_f - \rho_g \right) g} \tag{4}$$

When implementing this into the S-RELAP5 model, it was assumed based on preliminary I&C design information that the plant's narrow range and wide range liquid level indication algorithms would both be calibrated for saturated conditions at 1200 psia, and would not be corrected for transient-related deviations from those conditions. The densities in equation (4) are therefore treated as constant values and are evaluated at the 1200 psia saturated state regardless of the dynamic conditions in the SG. The constants are defined in the non-LOCA model supplied in the Response to RAI 413, Question 07.08-41, as control variables (CVs) 40 (for ρ_f) and 41 (for ρ_g).

It is also assumed that the S-RELAP5-calculated pressures for the model volumes containing the taps as listed in Table 07.08-24-1, with appropriate adjustments to account for the dynamic heads and the static head differences between the volume-centered elevations and the tap elevations, are suitable to use as the tap pressures in equation (4).

For example, using the S-RELAP5 model volume numbers shown in Figure 07.08-24-1 for SG 1:

[ ]

Table 07.08-24-1 —SG Level Tap and S-RELAP5 Volume Elevations

Location	Elevation Above Top of Tubesheet	Comments

Applying the elevations in Table 07.08-24-1 and equations (4), (5), and (7) above, the narrow range liquid level for SG 1 becomes:

[ ]

Similarly, applying the elevations in Table 07.08-24-1 and equations (4), (5), and (7) above, the wide range liquid level for SG 1 becomes

[ ]

[

]. The original collapsed liquid level control functions are retained in the non-LOCA model for comparison purposes, but all trips and controls that reference SG level use the new differential pressure signals. [ ]

Item 07.08-24(b):

<u>Steam Generator Level Transient Response</u>

The models of SG level are compared for the main steam isolation valve (MSIV) event because it presents the more severe drop of SG level.

Figure 07.08-24-2 shows the NR level response for the MSIV closure event for end of cycle (EOC) conditions. The reactor trip on diverse actuation system (DAS) low SG level at 15 percent NR level occurs at 133 seconds after MSIV closure. The plots show that if the collapsed liquid level scheme had been used for DAS input instead of the differential pressure algorithm the reactor trip (see Figure 07.08-24-3) on low level would have occurred 5 to 10 seconds earlier.

Figure 07.08-24-4 shows the WR level response for the MSIV closure event. In this case the collapsed liquid level signal leads the differential pressure WR signal, demonstrating that the shrink/swell phenomenon has a larger influence on the full-range level transmitters than for the narrow range transmitters.

Figure 07.08-24-5 shows the outlet pressure for the affected SG. The timing of the pressure peak at ~140 seconds coincides with the approximate time of minimum SG WR level (differential pressure signal) shown in Figure 07.08-24-4.

Figure 07.08-24-1 — Diversity and defense-in-depth (D3) Secondary System Nodalization

Figure 07.08-24-2 — SG-4 NR Level - D3 MSIV Closure

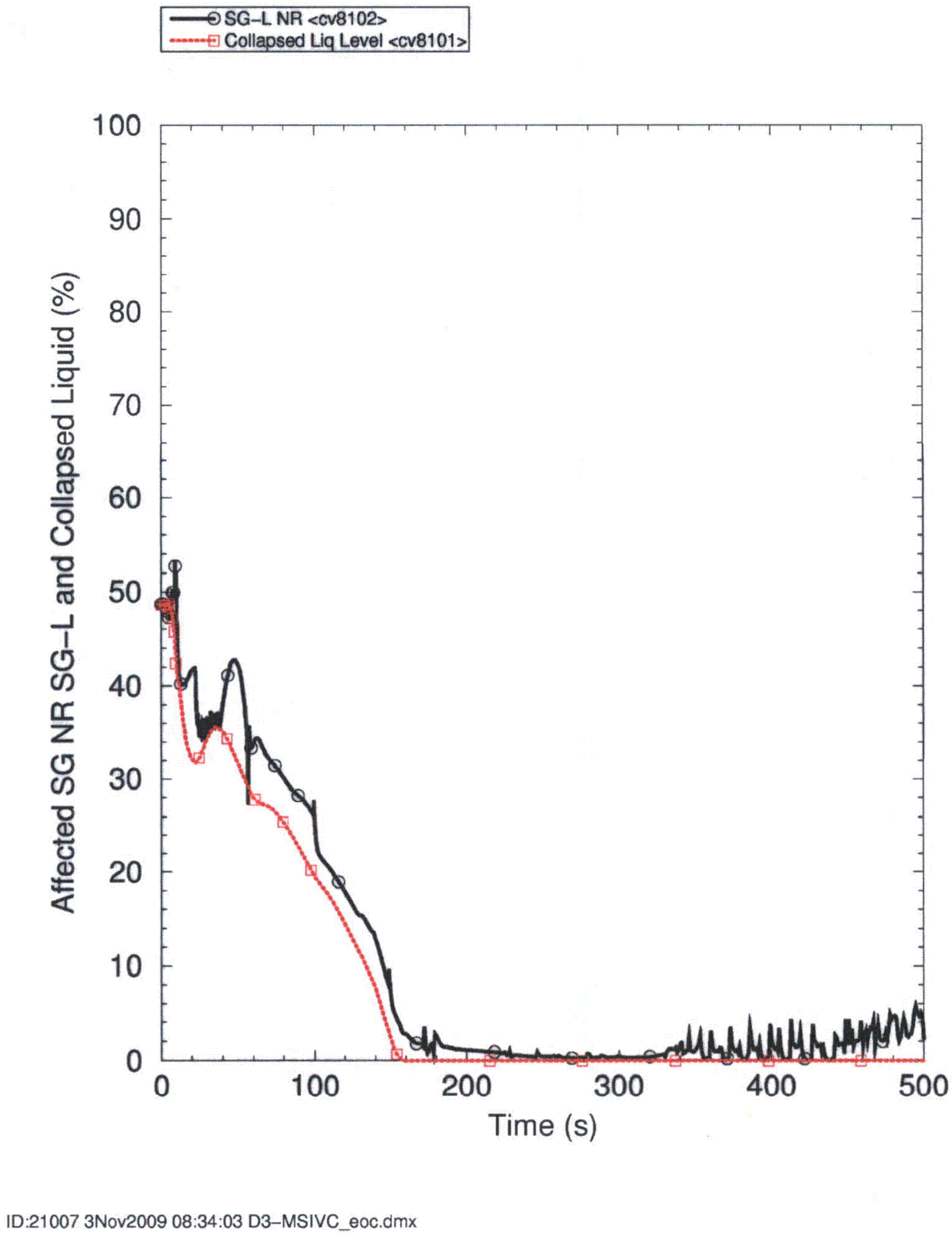
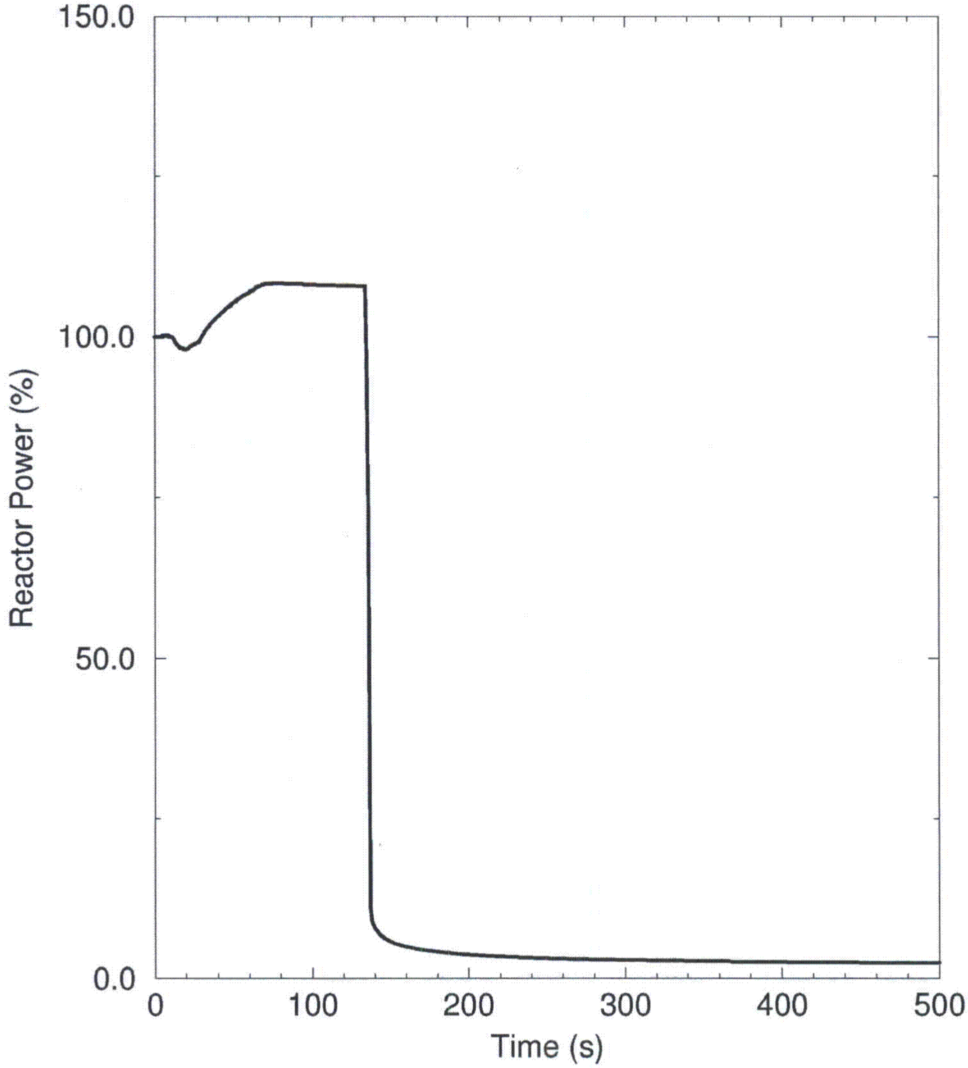


Figure 07.08-24-3 — Reactor Power - D3 MSIV Closure



ID:21007 3Nov2009 08:34:03 D3–MSIVC_eoc.dmx

Figure 07.08-24-4 — SG-4 WR Level - D3 MSIV Closure

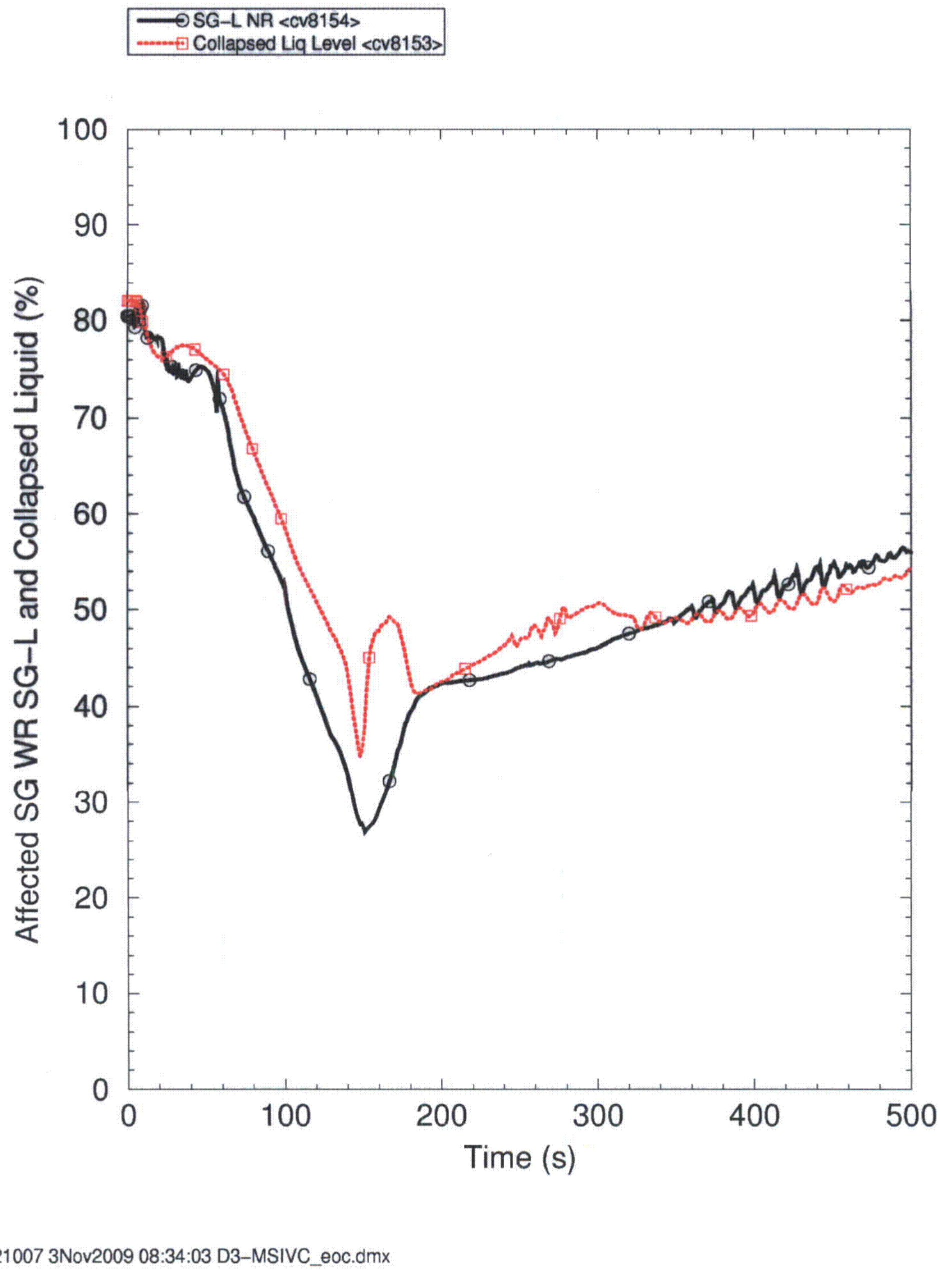
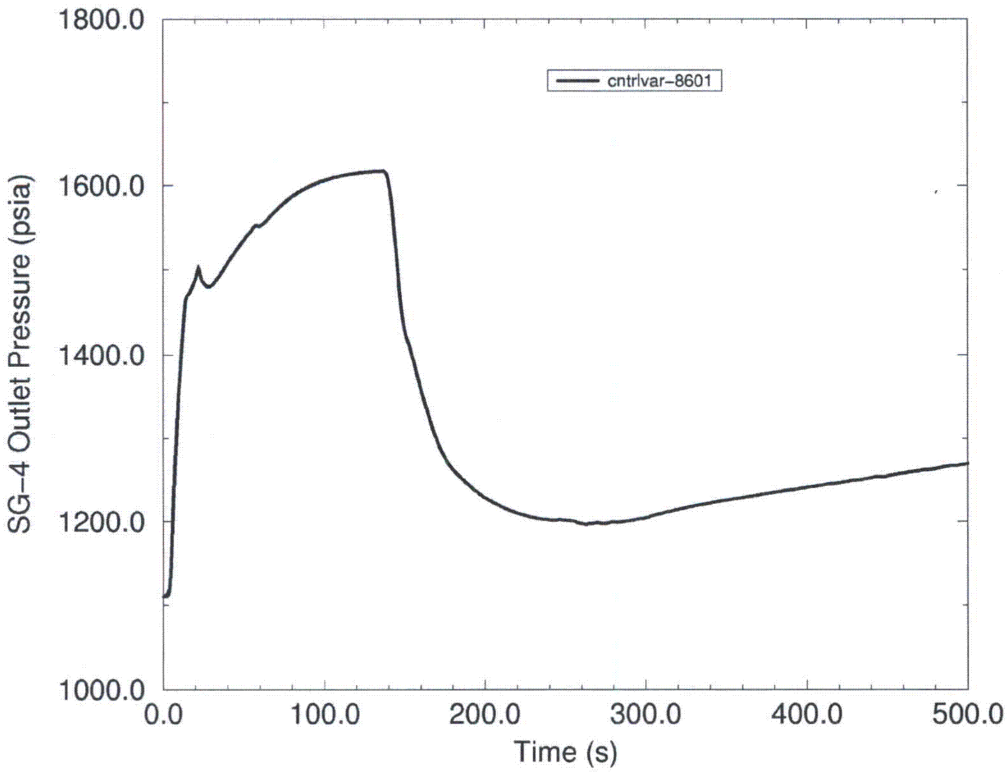


Figure 07.08-24-5 — SG-4 Outlet Pressure - D3 MSIV Closure



FSAR Impact:

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

Question 07.08-30

Explain the assumption that RCSL will respond to the Boron Dilution event as described in Section A.3.5.5 of 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.3.5.5 of ANP-10304 Rev 1 describes a Boron Dilution event where RCSL responds to the reactivity transient by automatic insertion of RCCAs, thereby alerting the main control room operator of the dilution event. Sections 4.1, 4.12, and A.2.2 state that RCSL is assumed not to be available in the D3 analysis. The U.S. EPR, Tier 2, Section 7.7.2.3.10, state that "...RCS boron concentration is calculated in the PS..." and "Four redundant limitation signals from the PS are transferred to RCSL." Therefore, upon postulated failure of the PS, the "transfer" of PS limitation signals to the RCSL would not occur. The staff could not identify sufficient design descriptions that would clearly describe how the RCSL would respond to the Boron Dilution event as described in Section A.3.5.5 of ANP-10304 Rev 1.

Response to 07.08-30

There are two possible plant responses for a boron dilution event at power. One possible response is that the reactor control, surveillance and limitation system (RCSL) is functioning properly and the other is that the RCSL is either in manual control or is not functioning. Section A.3.5.5 of ANP-10304 Revision 1 discussed both situations. Both cases were evaluated to cover the possible responses. If a software common cause failure (SWCCF) is present in the protection system (PS) resulting in a complete failure of the PS, the transfer of limitation signals to the RCSL will most likely not occur and the RCSL will not function. This is the same as when the RCSL is in manual control. Conversely, when the SWCCF in the PS resulted in a partial failure, it is possible for the RCSL to function normally. The diversity and defense-in-depth evaluation discussed in Section A.3.5.5 of ANP-10304 Revision 1 presents both possibilities to demonstrate that acceptance criteria are met in the event of either scenario.

For the case where RCSL is functioning properly (partial failure of PS), RCSL inserts rod cluster control assemblies (RCCAs) to maintain the reactor coolant system (RCS) average temperature (T_{avg}) and core power. As the position of the RCCAs approaches the power-dependent insertion limit (PDIL), an alarm will alert the operator that a possible dilution is in progress. Rod movement is blocked so that the PDIL is not exceeded. As in the case for when the RCSL is not functioning or is in manual operation, sufficient time (several hours) is available for the operator to detect and terminate a dilution of the RCS before the shutdown margin is lost.

FSAR Impact:

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

Question 07.08-31

Provide an explanation of difference in the DNBR transient between the no-rupture and with-rupture RCCA Ejection cases described in Section A.3.5.6, including:

a. Identification of any differences in analysis assumptions (e.g., initial conditions, reactivity parameters) that affect the transient,

b. A comparison of the key reactor parameters affecting DNBR, e.g., power level, peaking factors, reactor pressure, coolant temperatures, core flow.

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 applicant reports that the RCCA Ejection event, assuming no vessel rupture, does not exceed the DNBR SAFDL, whereas for the rupture cases the DNBR decreases below the SAFDL. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-31

Item 07.08-31(a):

The rod ejection event for diversity and defense-in-depth (D3) considered three different scenarios for coolant leakage from "break" sizes identified in Section A.3.5.6 of ANP-10304, ranging from no break area to maximum break area, sized from the control drive flange housing inner diameter. The three sizes provide a spectrum of possible coolant leakage path sizes if the control drive were to be ejected from the reactor by the pressure driving head from a flange break. This also allows consideration of the impact of the depressurization of the reactor coolant system (RCS) on the departure from nucleate boiling ratio (DNBR) performance.

Each of the events is initialized to the same operating conditions and each "ejects" a control rod with a worth of 65 pcm by adding the equivalent point kinetic worth in dollars (1$ = 1 beta, or delayed neutron fraction) over a timeframe of 0.05 second to simulate the withdrawal from the hot full power (HFP) dependent insertion limit (~50 percent inserted)

The three-dimensional (3D) transient power shapes were determined for the fuel assembly of interest from a rod ejection calculation with the three-dimensional nodal kinetics code NEMO-K using constant inlet thermal hydraulic conditions. This captured the initial power shape redistribution in the assembly of interest, following the methods of U.S. EPR Control Rod Ejection Accident Methodology Topical Report, ANP-10286P. The total core power histories were determined from the point kinetics S-RELAP5 model. These accounted for the reactivity feedback effects from the depressurization and heatup of the RCS occurring after the addition of reactivity from the ejected control rod. The inputs to the LYNXT DNBR calculation were a combination of the transient 3D power shapes in the form of peaking factors and the total core power, mass flux, RCS pressure, and inlet temperature in the form of histories normalized to the initial conditions.

Item 07.08-31(b):

The sequence of events for each break size case is provided in Tables 07.08-31-1, 07.08-31-2, and 07.08-31-3. The axial power shapes along with their associated radial peaking factors for the peak fuel rod are provided in Figure 07.08-31-1. The transient shapes are shown for key time points of the transient and are a subset of the full data set from the 3D kinetics calculation. A comparison of the responses for the total core power, the core exit pressure, the core inlet temperature, the minimum departure from nucleate boiling ratio (MDNBR)/specified acceptable fuel design limit (SAFDL), the peak fuel and cladding temperatures, and the peak average enthalpy rise in the fuel are provided in Figures 07.08-31-2 through 07.08-31-6. The information is shown for each case out beyond the time of the MDNBR point. The information shown for each case is beyond the time of the minimum DNBR point. The time ranges presented are from 200 seconds for the no break case to a few seconds after reactor trip for the break cases. The information in the figures allows the following observations to be made:

- The depressurization rate is proportional to the break size (see Figure 07.08-31-3).
- The moderator density reduction is proportional to the magnitude of the depressurization and, therefore, so is the magnitude of negative reactivity addition.
- Core power reduction is proportional to the magnitude of negative reactivity addition (see Figure 07.08-31-2). The initial heat up of the core inlet temperature is about the same for each of the cases due to the initial energy deposition into the coolant during the initial 6 seconds of the event; beyond this point, lower core powers lead to lower increases in the inlet temperature (see Figure 07.08-31-3).
- MDNBR degradation increases as the rate of depressurization increases. For each case with break, the SAFDL is violated (see Figure 07.08-31-4). The results of fuel failure census, performed in accordance with the methods presented in ANP-10286P, indicate less than 0.3 percent of the rods enter DNB (one criterion for fuel failure) which is far below the limit of 30 percent fuel failures.
- Once the MDNBR has exceeded the SAFDL the peak cladding temperatures rapidly increase but are limited by the reduction in the fuel rod heat fluxes due to the decreasing core power. At no time do the clad temperatures exceed the limit established in Section 2.2 of ANP-10286P (see Figure 07.08-31-5).
- The peak fuel temperatures are influenced primarily by the core power history because the peak linear heat generation rate is occurring at elevations below the location at which MDNBR exceeds the SAFDL limit. This is reflected in the peak average enthalpy rise response provided in Figure 07.08-31-6. At no time do the fuel temperatures exceed the melting point limit established in Section 7.3 of ANP-10286P (see Figure 07.08-31-5).
- The peak fuel enthalpy rise is impacted by the local heat transfer capability as well as the internal heat generation rate. Violation of the MDNBR SAFDL is followed by an enthalpy rise excursion as a result of the degradation of heat removal capability. The reactor trip eventually terminates the local temperature excursions. This occurs for these cases before the MDNBR SAFDL is violated at the elevation where the peak LHGR is located. At no time

does the enthalpy rise exceed the limit of 150 cal/gm established in Section 2.2 of ANP-10286P (see Figure 07.08-31-6).

Table 07.08-31-1 — Sequence of Events for Rod Ejection for Case with No Break

Event	Parameter	Time (sec)
Peak core power reached	110.7%	0.066
High core power level delay (protection system (PS) reactor trip (RT) not active)	Trip 455	6.8
Minimum MDNBR/SAFDL reached	1.198	159
Transient terminated (without diverse actuation system (DAS) RT)		1800.0

Table 07.08-31-2 — Sequence of Events for Rod Ejection for Case with 0.025 ft^2 Break

Event	Parameter	Time (sec)
Peak core power reached	110.7%	0.066
High core power level delay (PS RT not active)	Trip 455	7.2
MDNBR/SAFDL limit reached	1.000	29.0
Low hot leg saturation margin delay (PS RT not active)	Trip 460	29.5
Low PZR pressure delay (PS RT not active)	Trip 15	55.8
Low hot leg pressure delay (PS RT not active)	Trip 5	57.8
Minimum MDNBR/SAFDL reached	0.862	69.0
DAS low hot leg pressure delay	Trip 88	69.1
DAS RT with delay	Trip 900	69.5
DAS turbine trip (TT) with delay	Trip 899	70.1
Transient terminated		1281.6

Table 07.08-31-3 — Sequence of Events for Rod Ejection for Case with 0.048 ft^2 Break

Event	Parameter	Time (sec)
Peak core power reached	110.7%	0.072
High core power level delay (PS RT not active)	Trip 455	8.8
MDNBR/SAFDL limit reached	1.000	14.5
Low hot leg saturation margin delay (PS RT not active)	Trip 460	15.1
Low PZR pressure delay (PS RT not active)	Trip 15	28.2
Low hot leg pressure delay (PS RT not active)	Trip 59	28.8
DAS low hot leg pressure delay	Trip 88	34.8
Minimum MDNBR/SAFDL reached	0.864	35.0
DAS RT with delay	Trip 900	35.2
DAS TT with delay	Trip 899	35.8
Safety injection system (SIS) by PS or by DAS	Trip 1058	58.1
Transient terminated		195.1

Figure 07.08-31-1 — Peak Fuel Rod Axial Shapes from NEMO-K and Best Estimate Models

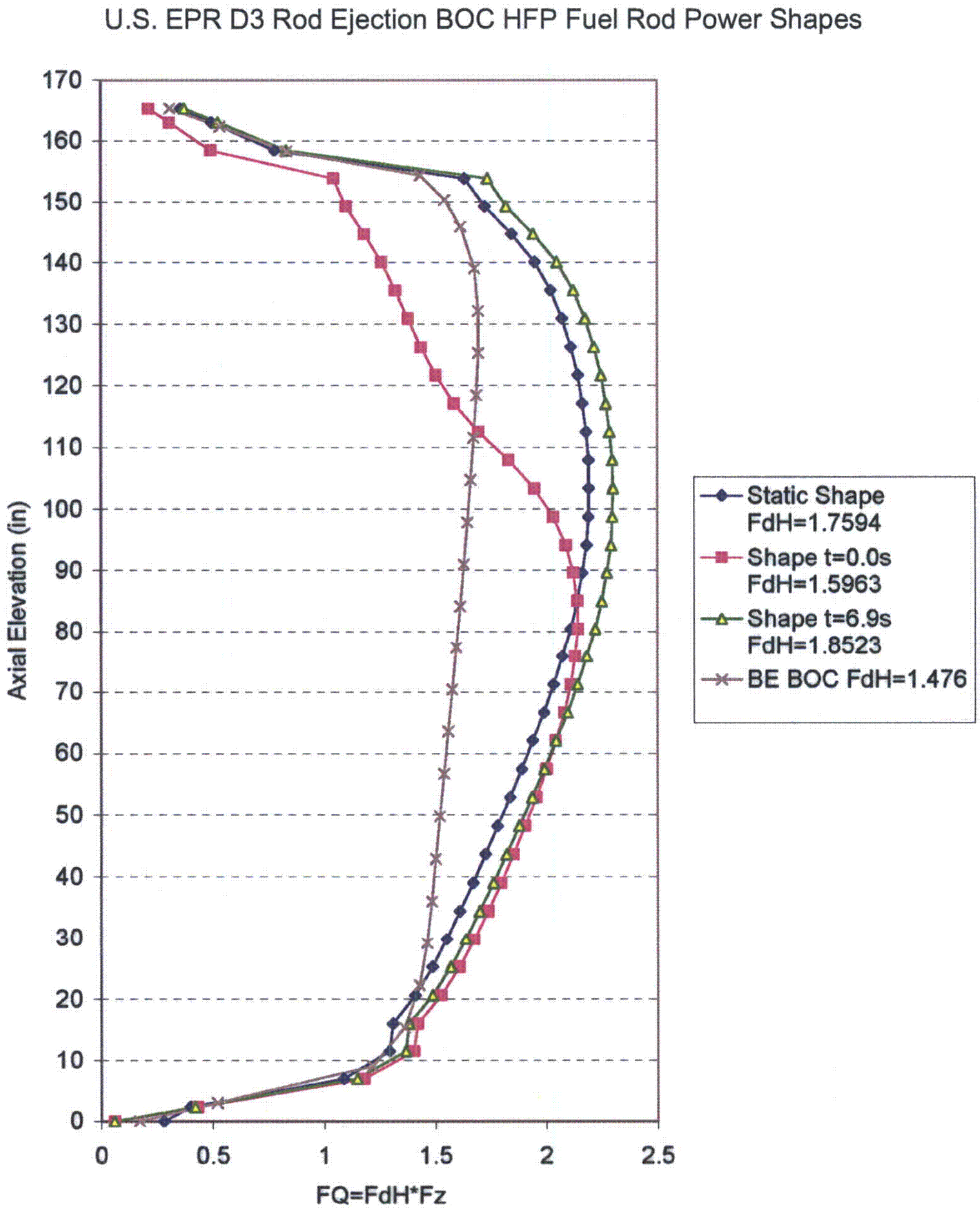


Figure 07.08-31-2 — Fraction of Core Power Response to Rod Ejection from BOC HFP

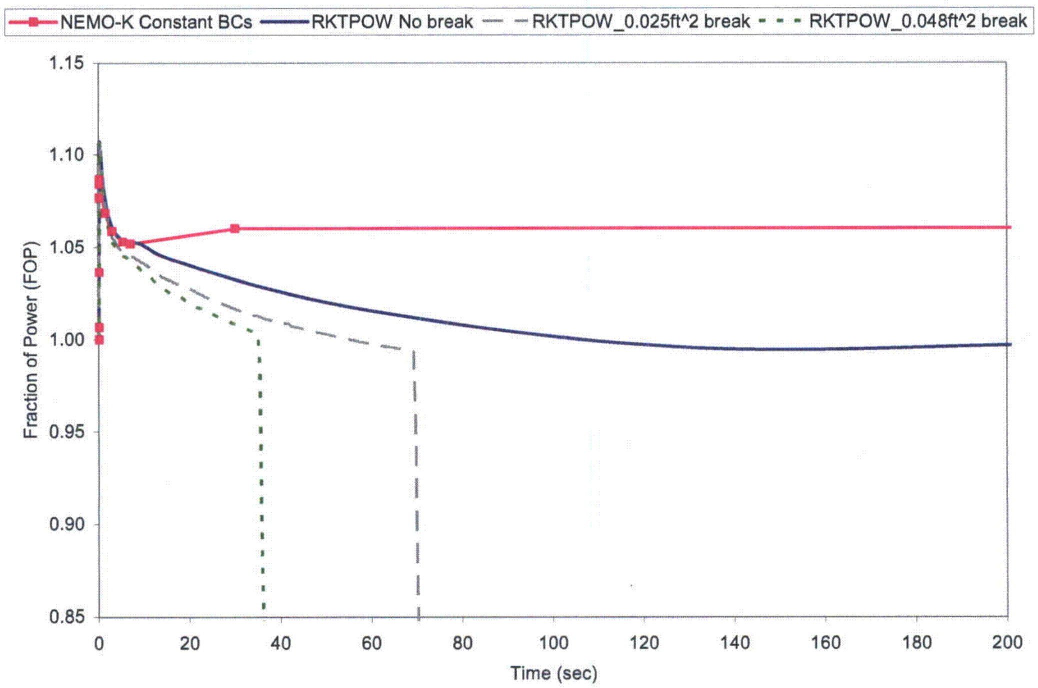


Figure 07.08-31-3— Core Exit Pressure and Inlet Temperature Responses to Rod Ejection from BOC HFP

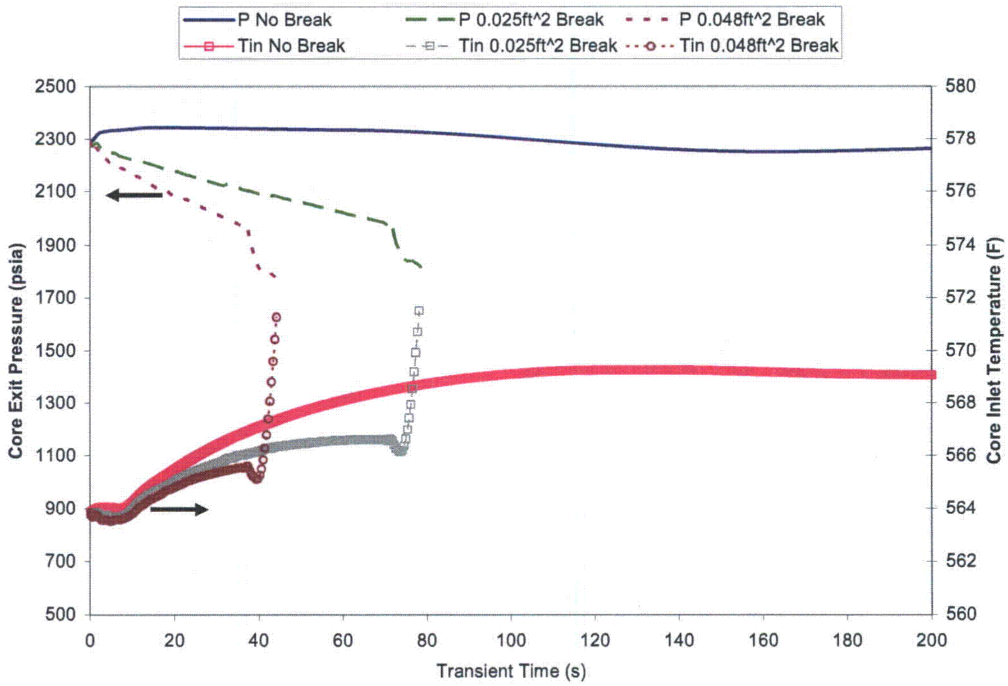


Figure 07.08-31-4 — MDNBR/SAFDL Response to Rod Ejection from BOC HFP

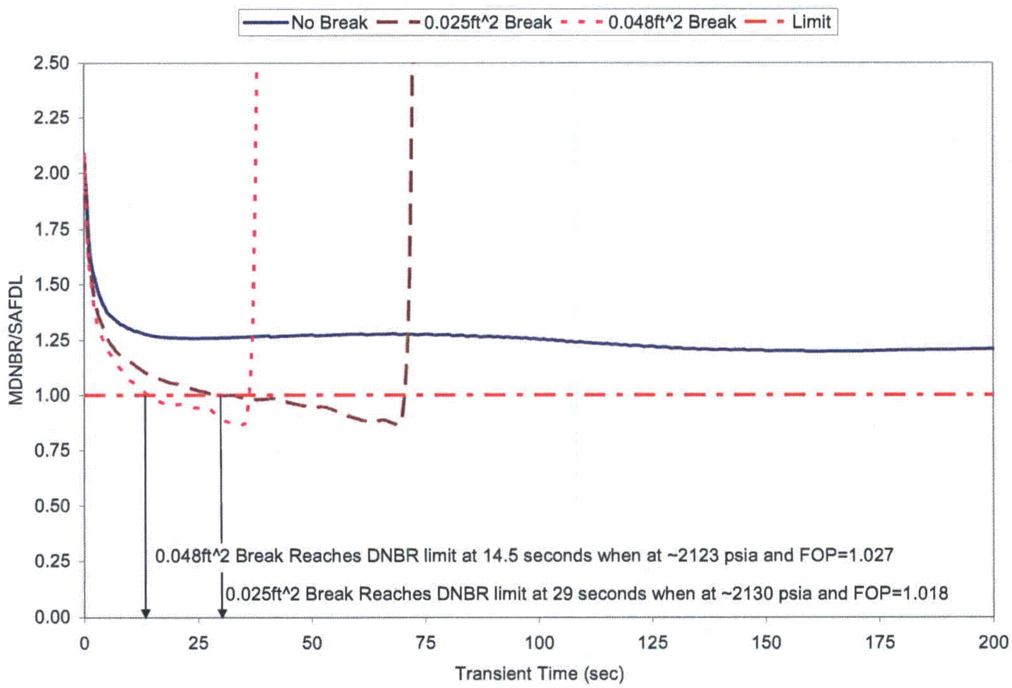


Figure 07.08-31-5 — Peak Fuel and Cladding Temperature Responses to Rod Ejection from BOC HFP

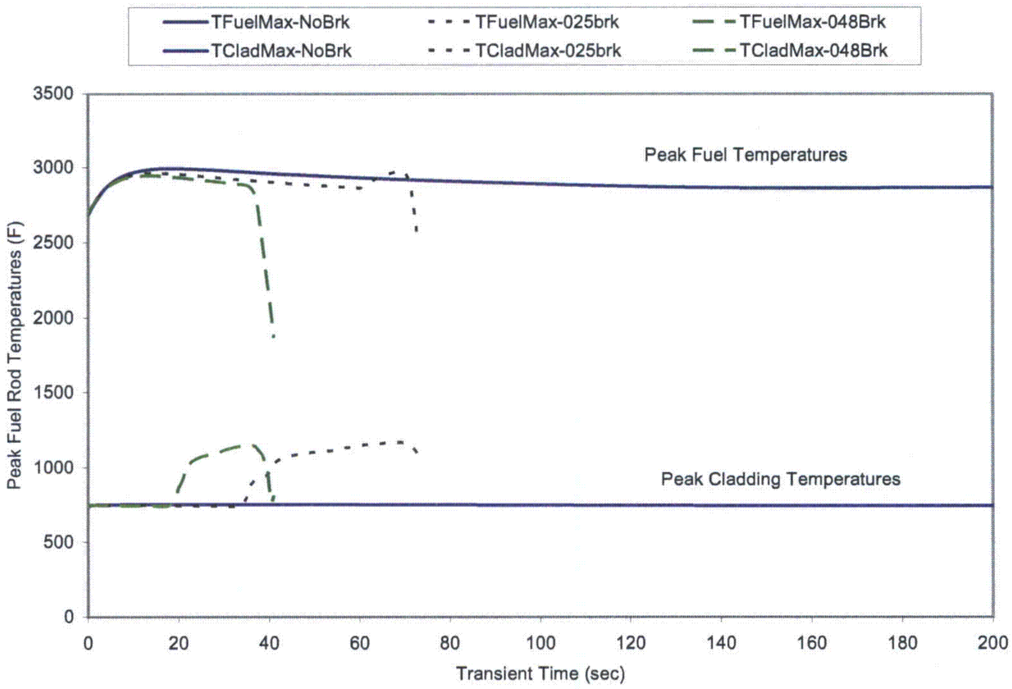
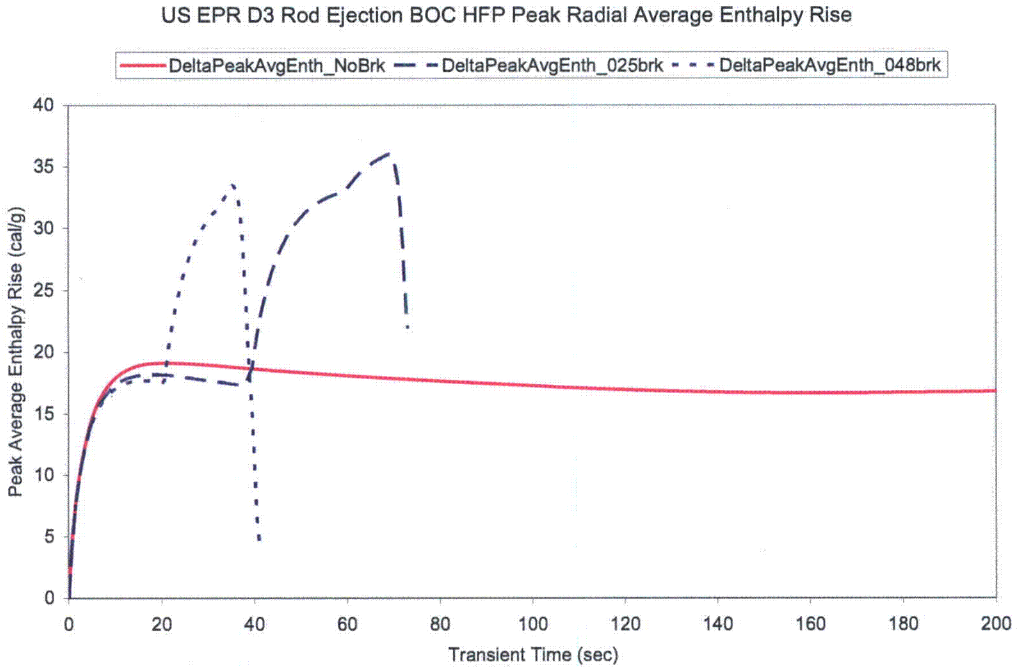


Figure 07.08-31-6 — Peak Average Enthalpy Rise Responses to Rod Ejection from BOC HFP



FSAR Impact:

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

Question 07.08-36

For the D3 analysis of SBLOCA, described ANP-10304 Rev 1 Section A.3.7.3.2, provide the following additional information relative to RCP trip during SBLOCA with SWCCF in the PS:

a. The criteria for operator determination of the need for RCP trip, i.e., LOCA as confirmed by two-phase RCS flow conditions,

b. The displays available to the main control room operators to determine the need for RCP trip, and

c. Identification of the procedure or procedure type (e.g., EPGs) that will prescribe the steps to accomplish the required operator action and whether a special D3 coping procedure is required.

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 D3 analysis of SBLOCA, described ANP-10304 Rev 1 Section A.3.7.3.2, states that an RCP trip does not automatically occur. Although SBLOCA sensitivity analyses referred to in Section A.3.7.3.2 reportedly demonstrate that 10 CFR 50.46 criteria are met even without RCP trip, the staff's position is that the RCPs should be tripped during the occurrence of a SBLOCA either automatically or manually by operator action. The lack of automatic RCP trip for SBLOCA is not in conformance with TMI Action Plan requirement II.K.3.5 (NUREG-0737) as stated in FSAR Sections 15.6.5.2.2 and 15.6.5.2.6. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-36:

The protection system (PS) design for the U.S. EPR includes an automatic reactor coolant pump (RCP) trip on low differential pressure. The U.S. EPR design, therefore, conforms to the NUREG-0737 TMI Action Plan requirement II.K.3.5.

In support of the diversity and defense-in-depth (D3) assessment, the small break loss of coolant accident (SBLOCA) analysis assumes a software common cause failure (SWCCF) in the PS rendering it unavailable. Under these conditions the diverse actuation system (DAS) provides protection. D3 is beyond the U.S. EPR design basis and the analysis supporting D3 is performed in conformance with BTP 7-19 (Ref. [1]). BTP 7-19 includes the use of best-estimate assumptions for the assessment of the adequacy of the DAS protection features. Under best-estimate conditions each of the four safety injection (SI) trains would be available.

The DAS does not include an RCP trip function; therefore, during the SBLOCA event the RCPs will continue to run until the operator manually trips them in accordance with the RCP trip verification steps implemented in the design Emergency Procedure Guidelines (EPG) governing a loss of coolant accident (LOCA) event.

Continued operation of the RCPs can result in earlier loop seal clearing, which in the longer term could result in more overall inventory loss out the break. If the RCPs are tripped when the minimum inventory occurs, it could cause a collapse of voids in the core, a deeper core uncovery, and a potentially higher peak cladding temperature.

An SBLOCA RCP trip time sensitivity analysis was performed with an SWCCF failure in the PS. A spectrum of break sizes ranging from 1.0 inch diameter to 9.71 inch diameter, representing 10 percent pipe cross-sectional area, were analyzed with various RCP trip times, including the time of core minimum inventory. The analysis demonstrated that the four trains of medium head safety injection (MHSI)/ low head safety injection (LHSI) enable ample core inventory and tripping of the RCPs at various transient times, or at minimum vessel inventory, does not adversely impact the transient response.

The results of the break-spectrum analysis show that the maximum PCT (1220°F for the 9.71 inch ID with the RCP tripped at 60 seconds) remains below the 10 CFR 50.46 criterion for the entire break spectrum while the RCPs are operating or not.

The analysis also demonstrates that an RCP trip during an SBLOCA event with an SWCCF in the PS is not needed to mitigate the event. SWCCF in the PCs during an SBLOCA event is beyond design basis and the analysis of this event is subject to best estimate conditions and no single failure assumptions, as outlined in BTP 7-19 (Ref. [1]). Therefore, operator criteria or a D3 coping procedure for tripping the RCPs during this event are not necessary. Technical Report ANP-10304 Rev 1 Section A.3.7.3.2 will be revised to include this information.

References

1. U.S. NRC, Standard Review Plan, Branch Technical Position 7-19.

FSAR Impact:

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

Technical Report ANP-10284 Impact:

Technical Report ANP-10284, Section A.3.7.3.2, "Small Break LOCA" will be revised in the next revision as described in the response and indicated on the enclosed markup.

Question 07.08-37

Provide information to justify that manual isolation of the main control room as described in Section A.3.9 of ANP-10304 Rev 1, will occur in a timely manner, including:

a. A description of how the need for manual isolation of the main control room is recognized,

b. The time line for manual control room isolation, including recognition of need for isolation and achievement of isolation,

c. Identification of the procedure or procedure type (e.g., EPGs) that will prescribe the steps to accomplish the required operator action and whether a special D3 coping procedure is required, and

d. A discussion of whether the operator actions required to manually isolate the main control room represent a diverse means of protective action to ensure the D3 radiological analysis acceptance criteria are met.

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.

For the Radiological Consequences of accidents with SWCCF in the PS, Section A.3.9 of ANP-10304 Rev 1 states that DAS does not provide automatic control room isolation. Analyses performed by the applicant, however, indicate that manual isolation of the main control room should take place within 30 minutes of an event initiation. The staff is concerned that manual isolation of the main control room may represent a required D3 coping action and is a considered a vulnerability which should address the guidance contained in the BTP-7-19 acceptance criteria, which states, in part:

The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.

Therefore, the staff request additional information to address this vulnerability.

Response to 07.08-37

Isolation of the main control room (MCR) is required following events with radiological consequences. The events of interest include loss-of-coolant accident (LOCA), steam generator tube rupture, main steam line break, reactor coolant pump locked rotor, rod ejection, and fuel handling accident. Under normal conditions, the protection system (PS) would automatically actuate the MCR emergency filtration system upon receipt of a high radiation signal in the MCR air intakes or a primary containment isolation signal. In the event of a software common cause failure (SWCCF), automatic isolation of the MCR is assumed not available, but the radiation signal and alarms are still present. In each of these scenarios with a

SWCCF in the PS, the MCR high radiation air intake alarms would alert the operator that MCR isolation is required. Since the air intakes to the MCR are close to the release point in each scenario, the alarm is expected to occur relatively early in the event (within approximately 5 minutes). The diversity and defense-in-depth (D3) analysis described in Section A.3.9 of ANP-10304, Revision 1 relies on this alarm function to alert the operator so that the MCR is isolated within 30 minutes.

Emergency response procedures for the U.S. EPR are not developed as a part of Design Certification. It is, however, anticipated that either abnormal operating procedures or emergency operating procedures will include instructions in response to high radiation at the MCR intakes. These instructions will either direct the operator to confirm MCR filtration system actuation or provide for manual isolation. A special D3 coping procedure is not anticipated.

FSAR Impact:

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

Question 07.08-40

For Section A.2.5 of ANP-10304 Rev 1, describe how the models for pressurizer pressure and level control are validated to assure that they accurately describe the plant response.

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.2.5 of ANP-10304 Rev 1 states that the pressurizer pressure and level control systems are included in the S-RELAP5 best estimate non-LOCA model. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-40

The S-RELAP5 models for the best estimate analyses feature a 10-node pressurizer component that utilizes a standard collapsed liquid level scheme for level measurement identical to that used in the U.S. EPR FSAR Tier 2, Chapter 15 design basis models. The reactor coolant system (RCS) primary nodalization for non loss-of-coolant accident (non-LOCA) analyses is shown in Figure 07.08-40-1, and the RCS nodalization for small-break LOCA (SBLOCA) analyses is shown in Figure 07.08-40-2. The pressurizer (PZR) components are almost the same as the non-LOCA model except that the normal pressurizer spray and the pressurizer safety relief valves (PSRVs) are not included in the SBLOCA model because those subsystems are not usually challenged or required for SBLOCA events. The PSRVs in the best estimate non-LOCA model are identical to those from the U.S. EPR FSAR Tier 2, Chapter 15 design basis models except that nominal opening and closing setpoints are used instead of nominal plus instrument uncertainty.

Pressurizer Level Indication and Control

Table 07.08-24-1 in the Response to RAI 413, Question 07.08-24-1 shows the PZR level tap locations relative to the bottom of the PZR vessel.

PZR level control is accomplished by the chemical and volume control system (CVCS). The S-RELAP5 CVCS model is described in the Response to RAI 413, Question 07.08-39. The nominal control setpoint at 100 percent rated power is 54.3 percent. Table 07.08-40-3 shows the level and pressure control setpoints that are implemented.

Figure 07.08-40-3 and Figure 07.08-40-4 show the CVCS logic that is implemented in the S-RELAP5 models to control PZR level during transients. Figure 07.08-40-5 shows the pressurizer level response to three representative transients:

1. Loss of all four reactor coolant pumps (RCPs).
2. 1006 pcm rod drop at end of cycle (EOC).
3. Rod ejection accident with pressure boundary break.

Figure 07.08-40-6 shows the corresponding PZR pressure response for the same events. The rate of change of pressure for these typical events is not extreme enough for concern about the decalibration effects associated with the differential pressure level taps in the as-built pressurizer. There is no transient primary pressure profile for a diversity and defense-in-depth (D3) event that would invalidate the collapsed liquid level scheme used in the S-RELAP5 model.

Pressurizer Pressure Indication and Control

The parameters presented in Table 07.08-40-3 apply to the operation of the PID controller used in the RCS pressure control function.

Table 07.08-40-4 and Table 07.08-40-5 correspond to the function blocks in Figure 07.08-40-7 that are used to determine spray valve position. A linear interpolation is used to find values intermediate to those provided.

The desired flow (lb/s) is the flow through the normal spray valves, in addition to the constant bypass spray flow. Flow is based on constant volumetric flowrate at cold leg temperature = 562.5°F and pressurizer pressure = 2250 psia.

Table 07.08-40-1 — Pressurizer Level Tap Elevations

Location	Elevation	Comments
Lower level tap position	27.7559 in.	Measured relative to bottom of PZR
Upper level tap position	461.024 in.	Measured relative to bottom of PZR

Table 07.08-40-2 — Pressurizer Control Trips

Signal	Action(s)	Setpoint	S-RELAP5 Trip(s) Specifying Setpoint
High-high PZR level	Terminate PZR auxiliary spray and RCS charging	70%	Trip 379
High PZR level	Maximize RCS letdown	7% above control setpoint	Trip 474
High PZR pressure	If pressurizer normal spray is demanded but is unavailable or insufficient, actuate PZR auxiliary spray	2349.7 psia	Trip 383
Low-low-low PZR level	Deenergize all PZR heater rods	12%	Trip 373
Low-low PZR level	Terminate letdown	15%	Trip 476
Low PZR level	Minimize letdown and start standby charging pump	7% below control setpoint	Trip 472

Table 07.08-40-3 — Pressurizer Pressure PID Controller Constants

Parameter	Value
Gain, K_p	[ ]
Integral action time, T_r	[ ]
Derivative action time, T_d	[ ]
Lag time of derivative action, T_1	[ ]
C-OUTP upper limit, UL	[ ]
C-OUTP lower limit, LL	[ ]

Table 07.08-40-4 — 102F or 103F = f (C – OUTP)

C-OUTP Value [%]	Desired Spray Flow [lb/s]

Table 07.08-40-5 — 102P or 103P = f(F)

Desired Spray Flow [lb/s]	Desired Valve Position [% open]
-10	0
0	0
3.3	30
6.6	40
11.9	50
21.4	60
66.1	100
88.2	100

Figure 07.08-40-1 — Non-LOCA RCS and Pressurizer Nodalization

Figure 07.08-40-2 — SBLOCA RCS and Pressurizer Nodalization

Figure 07.08-40-3 — CVCS Control Logic

Figure 07.08-40-4 — CVCS Flow Control

Figure 07.08-40-5 — Pressurizer Level Response for Various Events

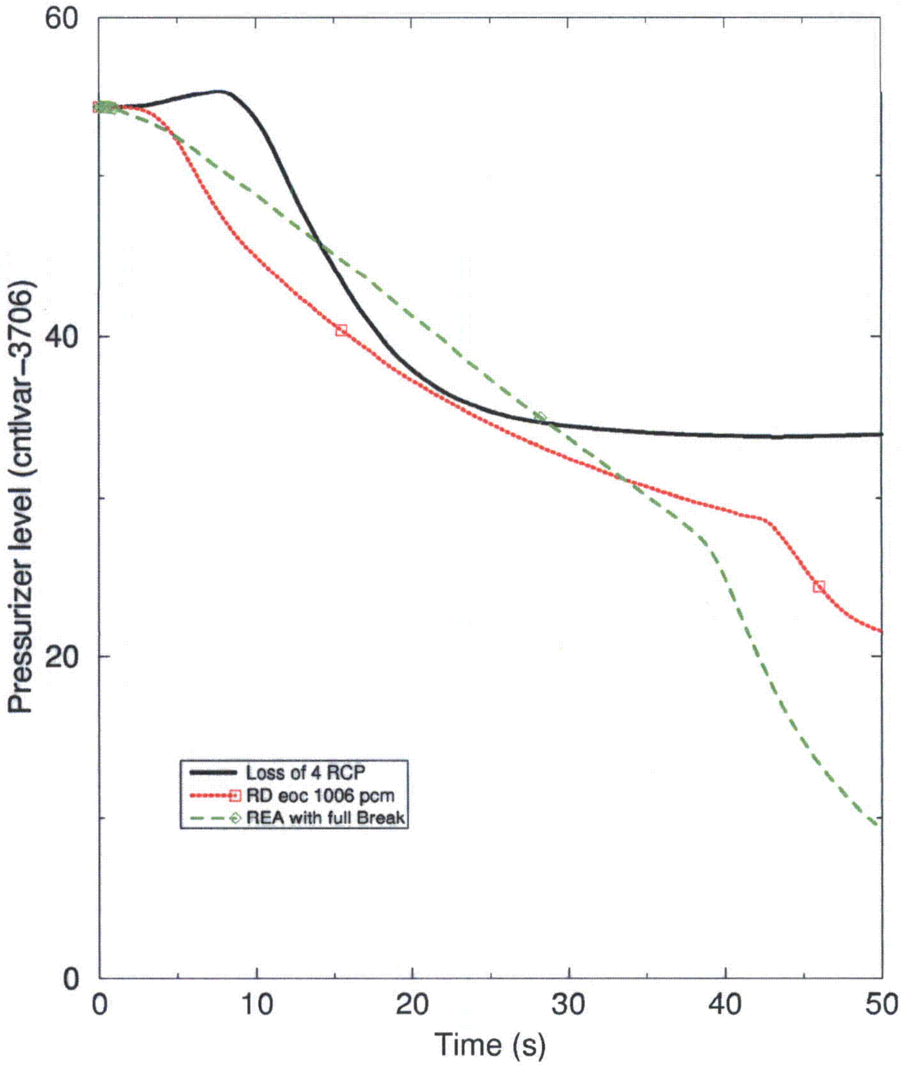


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Figure 07.08-40-6 — Pressurizer Pressure Response for Various Events

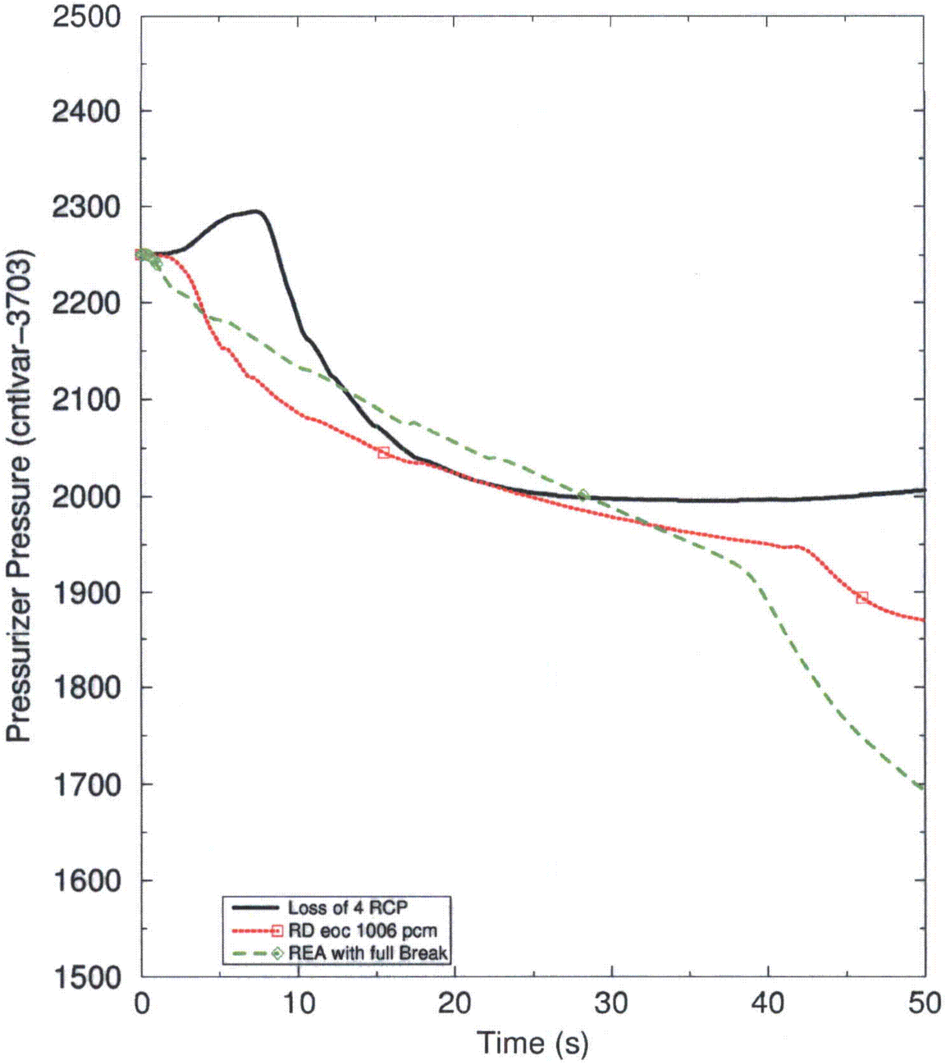

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Figure 07.08-40-7 — Mode 1 RCS Pressure Control

FSAR Impact:

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

Question 07.08-41

Provide the S-RELAP5 input base deck and the other required input deck files to perform a confirmatory run of the following events:

a. The limiting increase in steam flow event,
b. The inadvertent Closure of an MSIV event,
c. CVCS Malfunction resulting in decreased RCS boron event,
d. RCCA ejection with rupture event,
e. SBLOCA event.

Provide S-RELAP5 source code and relevant documentation used for D3 analysis. Documentation should include, but not limited to, source code modification record, description of the altered numerical algorithm, benchmark and validation calculation, etc.

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.

Response to Question 07.08-41:

INPUT DECKS

Tables 07.08-41-1 and 07.08-41-2 describe the input data files for S-RELAP5 non-loss of coolant accident (LOCA) and small break loss of coolant accident (SBLOCA) calculations provided on the CD titled "U.S. EPR S-RELAP5 D3 Input Files."

Successive calculations that have led to the restart file (RSTPLT) used for D3 transient calculations are listed in the first part of these tables. The second part of these tables details the transient specific input files.

The following S-RELAP5 input data used for the D3 analyses are new and are not compatible with previous versions of the S-RELAP5 code. These new inputs were developed to obtain a more realistic representation of the reactor behavior and do not alter the algorithms of the code. Additional explanation of these new data will be provided in the Responses to RAI 413, Question 07.08-21 (response to be provided in RAI 413 Supplement 2) and Question 07.08-39, (previously provided within this response).

[ ]

S-RELAP5 Source Code and Relevant Documentation

Table 07.08-41-3 describes the S-RELAP5 version UAPR09-0 code elements, which are stored on the CD titled "S-RELAP5, UAPR09."

The significant code changes implemented between the version of S-RELAP5 used in the performance of safety analyses of U.S. EPR FSAR Tier 2, Chapter 15 (UAPR06-0) and the version of S-RELAP5 used for the performance of the D3 analyses of U.S. EPR FSAR Tier 2, Chapter 7 (UAPR09-0) are briefly described in Table 07.08-41-4.

FSAR Impact:

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

Table 07.08-41-1—NON-LOCA S-RELAP5 Input File Descriptions

File Name	Comments
Common Files to Establish U.S. EPR Geometry and Systems	
Directory: NON_LOCA/BASE_DECK	
Steady_state_BOC	Best Estimate BOC base deck
DAS Neut BOC (null transient)	Input for adding DAS and neutronics data for BOC cases
Steady_state_EOC	Best Estimate EOC base deck
DAS Neut EOC (null transient)	Input for adding DAS and neutronics data for EOC cases
Files for Event Initiation and Transient Analysis	
Directory: NON_LOCA/TRANSIENT	
a- Increase in steam flow	BOC, transient time card and initiation
b- MSIVC	EOC, transient time card and initiation
c- Boron dilution	BOC, transient time card and initiation
d- REA	BOC full break, transient time card and initiation

Table 07.08-41-2—SBLOCA S-RELAP5 Input File Descriptions

File Name	Comments
Common Files to Establish U.S. EPR Geometry and Systems	
Directory: SBLOCA/BASE_DECK	
SBLOCA_ss	Best Estimate EOC base deck
SBLOCA_null	Neutronics parameters, adjustment of the turbine steam flow scaling factor, reset of the RCP initial ΔP constants, mass flow calibration factors and thermal power (QTH) calibration factor (KCALTH).
SBLOCA_step	Neutronics parameters, block RCP automatic trips when Protection System unavailable, EBS modification, correction to the SGs Inlet Plenum Outlet Junction Δh.
Files for Event Initiation and Transient Analysis	
Directory: SBLOCA/TRANSIENT	
e- SBLOCA_transient	

Table 07.08-41-3—S-RELAP5 Code Elements Contained on CD

File Name	Comments
README	ReadMe file describing contents of CD and installation/build instructions
Directory uapr09/bin:	
relap5	UAPR09-0 executable file
STH2XT	Steam table file
Remainder of directories under uapr09	UAPR09 S-RELAP5 source and build files

Table 07.08-41-4—Changes in S-RELAP5 between DC and D3 Versions

Change Number	Changes in S-RELAP5
1	Correction of an error in the RELAP5 series Point Kinetics model identified by the Idaho National Laboratory Correction of an error in the RELAP5 series heat conduction model identified by the Idaho National Laboratory

Technical Report ANP-10284 Markups

A.3.7.3.2 Small Break LOCA

A SBLOCA event is defined as a break in the RCS pressure boundary that has an area of 0.5 ft^2 or less (~10 percent of cold leg pipe area). The most limiting SBLOCA is in the cold leg pipe at the discharge side of the RCPs. This break results in the largest inventory loss and the largest fraction of SIS fluid being ejected through the break. In turn, this produces the greatest degree of core uncovery and the longest fuel rod heatup time. Consequently, it poses the greatest challenge to the 10 CFR 50.46 criteria.

In the U.S. EPR FSAR analysis, PS initiates RT and actuates SI on low pressurizer pressure, for all cases. The U.S. EPR FSAR analysis ~~assumes~~ evaluates cases with and without LOOP. In the LOOP case, it is assumed that LOOP occurs coincident with RT, which also initiates EFW flow, when the SI signal is reached. ~~If~~ In the case where off-site power remains available~~LOOP does not occur~~, EFW is not initiated until a low SG level is reached.

In the case of an SWCCF in the PS, DAS initiates an RT on low hot leg pressure and actuates SI (i.e., MHSI) on low pressurizer pressure, providing protection equivalent to that described in the U.S. EPR FSAR scenario. MFW is available to provide decay heat removal. As discussed in Section A.2.1, under best estimate conditions, a single failure or preventative maintenance is not assumed. Thus, all EFW trains and SI trains are available.

For certain break sizes, the MSRTs are relied upon in the U.S. EPR FSAR analysis to depressurize the RCS to enable the injection of MHSI. In the case of an SWCCF in the PS, the MSRTs might not be available because actuation of the MSRT partial cooldown function is handled in the PS. The TBS is a normal operation control system that also has the capability of implementing the partial cooldown function and reducing secondary system pressures. After RT, the TBS controls SG pressure to a fixed setpoint; after an SI signal is generated by DAS on low pressurizer pressure, a programmed cooldown begins, similar to the MSRT partial cooldown. This function is initiated by DAS and is available during an SBLOCA, as long as the MSIVs remain open. The PS includes an MSIV closure signal on high containment pressure. The MSIVs would remain open if this feature fails as part of the SWCCF. In the event of a partial SWCCF of the PS where the MSIVs still close on high containment pressure, the TBS would not be available to provide for the partial cooldown.

After 30 minutes and before 60 minutes, if the hot leg pressure indication is below 275 psig, the operators will manually realign LHSI to the hot legs to suppress steaming in the core to prevent over-pressurization of the containment. This action also prevents boron precipitation. The ability to manually switch SI to the hot legs is available outside the PS and is available in the event of an SWCCF in the PS.

07.08-36

07.08-36

~~1 A review is ongoing that could affect use of the turbine bypass system during a small break LOCA event. This may require an additional DAS function or justification for manual operator action to open theMSRTs.~~

The PS provides an RCP trip on low RCP differential pressure to ensure that, during an SBLOCA, the RCPs are tripped early in the event. During an SBLOCA with RCPs running, a greater amount of inventory could be lost out the break than with RCPs tripped. After sufficient inventory is lost and the RCPs are tripped, a deeper core uncovery could result in a higher peak clad temperature (PCT). DAS does not include an RCP trip function. Thus, with an SWCCF in the PS, the RCPs continue operating, with the opportunity to be tripped (manually) at a later time. Manual RCP trip time sensitivity analyses are performed for a spectrum of break sizes, to determine the latest RCP trip time that gives acceptable PCT results (i.e., PCT less than2200°F).

The SBLOCA RCP trip time sensitivity analysis is performed for a spectrum of break sizes ranging from a ~~3~~1.0 inch ID break to the maximum small break of 10 percent pipe cross-sectional area, the 9.71 inch ID break. RCP trip times of 10, 60, 900, and 1800 seconds were assumed. The analysis is performed using best estimate assumptions, with an SWCCF in the PS. The key best estimate assumptions include four trains of SI (no single failure or preventative maintenance), offsite power available, best estimate decay heat, and TBS available for partial cooldown. To evaluate a partial SWCCF of the PS, a separate set of cases were studied where the TBS is not available as a result of MSIV closure on high containment pressure (PS feature functions to isolate the TBS).

The results of the sensitivity analysis with the TBS available indicate that the maximum PCT remains below the 10 CFR 50.46 criteria for the entire break spectrum, whether or not the RCPs are operating. Therefore, the timing of the RCP trip has little impact and manually tripping the RCPs is not required. In fact, only breaks at the large end of the SBLOCA spectrum result in fuel heatup and only for cases where the RCPs are tripped at 60 seconds or less.

~~For the situation of~~For a partial SWCCF of the PS where the TBS is not available, the results of the sensitivity analysis are similar. ~~, t~~The maximum PCT remains below the 10 CFR 50.46 criteria for the entire break spectrum, regardless of the RCP trip time. In this ~~case~~scenario, except for the large end of the break spectrum, decay heat is first removed through the ~~secondary~~ steam generators MSSVs until the loop seal clears~~s~~. ~~Upon loop seal clearing the break removes sufficient energy to depressurize the primary actuating MHSI.~~ Breaks 2.5 inch ID and larger clear the loops early and are able to depressurize the primary to the MHSI injection setpoint. As the RCS depressurizes further, the MHSI and LHSI flow overcome the break flow, verifying~~assuring~~ extended core cooling. The smaller the break, the longer it takes for the loop seal to clear. For breaks between 2.5 inch and 1.0 inch ID, ~~decay heat is removed for the most part through the steam generator MSSVs. T~~the primary pressure remains above the MHSI shutoff head until either EFW actuation or the loop seal clears. EFW actuation fills the secondary with cold water condensing steam and reduc~~inges~~es secondary pressure. The reduction in secondary pressure ~~in turn~~then reduces primary pressure, leading to MHSI injection. ~~If~~In cases where the effectiveness of the EFW to condense steam ~~were~~was reduced to zero, ~~in these cases~~ recovery occurs ~~once~~when the loop seal clears. Once the loop seal clears, sufficient energy is removed through the break to depressurize the

07.08-36

07.08-36

RCS to the MHSI actuation setpoint. For these breaks, injection from CVCS is sufficient to keep the core covered prior to MHSI injection. Actuation of MHSI recovers RCS inventory. For breaks around 1 inch, the loop seal may take several hours to clear. In this case, the operator will ~~need to~~ take manual control and cooldown through the MSRTs to reduce RCS pressure and actuate MHSI. There is sufficient time to manually initiate the cooldown ~~such~~so that the partial cooldown function is not required to be automated on DAS.

~~This~~ These ~~analysis~~ analyses demonstrate~~s~~ that the U.S. EPR design is adequate in addressing an SWCCF in the PS during SBLOCA events, including partial failures.