AREVA

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

Response to U.S. EPR Design Certification Application RAI 413, Supplement 9

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 (4772), 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 (4772), FSAR Ch. 7, Supplement 1" 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 (4772), FSAR Ch. 7, Supplement 2" December 13, 2010.

Ref. 5: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 3" January 28, 2011.

Ref. 6: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 4" February 1, 2011.

Ref. 7: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 5" February 23, 2011.

Ref. 8: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 6" February 24, 2011.

Ref. 9: E-mail, Russ Wells (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 7" March 2, 2011.

Ref. 10: E-mail, Martin Bryan (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 413 (4772), FSAR Ch. 7, Supplement 8" March 15, 2011.

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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). AREVA NP provided a schedule for technically complete and correct responses to the questions in RAI 413 on September 08, 2010 (Reference 2). Supplement 1 to RAI No. 413 was sent on November 19, 2010, to provide a revised schedule (Reference 3). Supplement 2 to RAI No. 413 was sent on December 13, 2010, to provide a revised schedule (Reference 4). Supplement 3 to RAI No. 413 was sent on January 28, 2011, to provide technically correct and complete responses to seven questions (Questions 07.08-24, 07.08-30, 07.08-31, 07.08-36, 07.08-37, 07.08-40, 07.08-41) (Reference 5). Supplement 4 to RAI No. 413 was sent on February 1, 2011, to provide technically correct and complete responses to seven questions (Questions 07.08-15, 07.08-18, 07.08-20, 07.08-23, 07.08-25, 07.08-33, 07.08-34) (Reference 6). Supplement 5 to RAI No. 413 was sent on February 23, 2011, to provide technically correct and complete responses to five questions (Questions 07.08-10, 07.08-11, 07.08-12, 07.08-14, 07.08-16) (Reference 7). Supplement 6 to RAI No. 413 was sent on February 24, 2011, to provide a revised schedule (Reference 8). Supplement 7 to RAI No. 413 was sent on March 2, 2011, to provide technically correct and complete response to one question (Question 07.08-39) (Reference 9). Supplement 8 to RAI No. 413 was sent on March 15, 2011, to provide technically correct and complete responses to two of the remaining 13 questions (Questions 07.08-17 and 07.08-42) (Reference 10).

The attached file, "RAI 413 Supplement 9 Response US EPR DC - Proprietary.pdf" provides technically correct and complete responses to ten of the remaining 11 questions. AREVA NP considers some of the material contained in the enclosed response to be proprietary information. As required by 10 CFR 2.390(b), an affidavit is attached to support the withholding of the proprietary information from public disclosure. Proprietary and nonproprietary versions of the enclosure to this letter are provided.

The following table indicates the respective pages in the in the response document, "RAI 413 Supplement 9 Response US EPR DC - Proprietary.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 413 07.08-19	2	12
RAI 413 07.08-21	13	24
RAI 413 07.08-22	25	27
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RAI 413 07.08-32	47	52
RAI 413 07.08-35	53	55
RAI 413 07.08-38	56	59

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

Question #	Response Date
RAI 413 07.08-13	April 21, 2011

If you have any questions related to this submittal, please contact me by telephone at 434-832-2369 or by e-mail to sandra.sloan@areva.com.

Sincerely,

Darrell Gardner
for

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

Enclosure

cc: G. Tesfaye
Docket No. 52-020

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Pedro Salas. I am Acting Manager, Corporate 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 "RAI 413 Supplement 9 Response U.S. EPR DC - Proprietary," 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 paragraphs 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.

SUBSCRIBED before me this 23rd

day of March, 2011.

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2011
Registration No. 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 9

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 07.08-19:

For each of the following, provide a description of how the best-estimate initial condition assumptions used in the D3 analyses were generated for:

a. Core average axial and core radial power distributions and the reactivity coefficient curves,

b. The core average axial power distributions used in the D3 analyses,

c. The reactivity coefficient curves (MTC, Doppler),

d. The hot channel factors $F_{\Delta h}$ and F_Q used in the D3 analyses,

e. The scram reactivity curves and the total scram reactivity worth ($\Delta\rho$ or pcm),

f. The non-fuel-related reactor core parameters affecting DNBR, including T_{inlet}, reactor system pressure, and total core flow, and

g. An assessment of the initial steady-state DNBR operating margin with the best-estimate assumptions and comparison to the expected Technical Specification (TS 3.2.3) DNBR Limiting Condition for Operation (LCO) value.

For each of the above b through f, describe or provide a comparison of each of the parameters to their respective design values as contained in the FSAR and justify any deviations. Also, state whether the same values of the above parameters are used consistently for all the D3 analyses.

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 07.08-19:

The methods used to provide power distributions for the U.S. EPR FSAR analysis are described in Topical Report 10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR." [ ] LYNXT model for the U.S. EPR FSAR analyses, shown in Table 07.08-19-1, is used as described in Topical Report 10287P, Section C.1, and supported by Topical Report 10287P, RAI 8 and RAI 9. Events and power distributions combinations protected by an incore trip are initially scaled to start at the departure from nucleate boiling (DNB) limiting condition for operation (LCO), linear power density (LPD), LCO, or the radial peaking (FΔH) LCO, whichever occurs first. Using these methods, the power distributions will be equally limiting if the event is covered by the low DNBR or high LPD protection functions. Figure 07.08-19-2 shows an axial flux shape (AFS) representative of a limiting condition for both low DNBR and high LPD.

Event and power distribution combinations protected by initial steady state margins are evaluated to determine the degradation in DNBR or LPD where an incore trip does not occur. For the U.S. EPR FSAR, the limiting transient event protected by the initial steady state margin is the complete loss of forced reactor coolant flow (see the Response to RAI 413, Question 07.08-26). Figure 07.08-19-2 shows the limiting axial power distribution used to determine the limiting ΔDNBR.

Part c of this response discusses the reactivity coefficient curves for moderator and Doppler. Part e of this response discusses the scram reactivity curves.

b) The core average axial power distributions used in the D3 analyses are shown in Figure 07.08-19-2 and described in part a of this response. Following the methodology in Topical Report 10287P, the axial power distributions will be equally limiting if the event is covered by the low DNBR or high LPD protection functions for the U.S. EPR FSAR. Figure 07.08-19-2 shows an AFS representative of a limiting condition for both low DNBR and high LPD.

c) [

] Figure 07.08-19-4

compares the BOC and EOC moderator temperature coefficient (MTC) values to the values applied in the U.S. EPR FSAR. [

]

[

] Figure 07.08-19-5 compares the BOC and EOC Doppler reactivity curves to the values applied in the U.S. EPR FSAR. The U.S. EPR FSAR values in Figure 07.08-19-5 are constant Doppler temperature coefficients (DTC) of -1.17 pcm/°F at BOC and -1.85 pcm/°F at EOC. They are the most positive DTC for BOC and most negative DTC for EOC in U.S. EPR FSAR Tier 2, Table 15.0-6. [

]

d) The best estimate radial power distribution $F_{\Delta H}$ and axial power distribution are selected through the criteria described in part a of this response. [

] In comparison, the U.S. EPR FSAR analysis applies the transient methodology described in Topical Report 10287P.

For the U.S. EPR FSAR analysis, the core design uses an F_Q technical specification limit of 2.60. The U.S. EPR FSAR analysis applies technical specification $F_{\Delta H}$ limits of 1.70 at or above 100 percent of rated power linearly increased to 2.295 at 0 percent of rated power. These limits are acceptable to Topical Report 10287P methodology, which uses numerous power distributions to perform the U.S. EPR FSAR DNB and LPD analyses. [

]

e) [

] to develop a scram worth as a function of time for D3 analyses illustrated in Figure 07.08-19-6.

The scram reactivity curve applied for the U.S. EPR FSAR analyses employs a bounding relative reactivity worth as a function of time. This relative scram worth curves was used with the bounding scram worth (least worthy) from the 18-month and 24-month core designs corresponding to burnup and power level as shown in U.S. EPR FSAR Tier 2, Table 15.0-6.

Figure 07.08-19-6 compares the respective HFP BOC and EOC scram worths of -6161 pcm and -7353 pcm for the U.S. EPR FSAR to the scram worth curve applied in the D3 analysis. Figure 07.08-19-6 demonstrates the conservatism of the scram worth curve applied for the U.S. EPR FSAR analyses.

f) The initial non-fuel-related core input parameters affecting DNBR include core inlet temperature, reactor exit pressure, and core inlet mass flux. Table 07.08-19-3 compares these initial conditions for the D3 DNBR analyses, which are extracted from the S-RELAP5 output, to the U.S. EPR FSAR values.

[ ] and the RCS thermal design flow of 478,768 gpm for DNBR calculations in the U.S. EPR FSAR. U.S. EPR FSAR Tier 2, Section 4.4.4.2.1 states that the core bypass flow applied in the U.S. EPR FSAR analysis was a maximum of 5.5 percent. [ ]

[ ]

The pressure difference is negligible regarding the initial DNBR.

g) [ ] The expected technical specification DNBR limiting condition for operation LCO for zero failed SPNDs is 2.50. [ ] For perspective, the bounding ΔDNBR used in the DNBR LCO development is 0.60 in DNBR.

FSAR Impact:

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

Table 07.08-19-1—Radial Power Distributions in LYNXT

Table 07.08-19-2—Power Distribution Peaking Factors

Burnup		F	F
D3 BOC			
D3 MOC			
D3 EOC	1.425		
FSAR Limits	Max of 1.70 at HFP	Derived	Max of 2.60

Table 07.08-19-3—Non-Fuel Related Initial Conditions Affecting DNBR, U.S. EPR FSAR v. D3

T-H Parameter	U.S. EPR FSAR	D3
Core Exit Pressure (psia)		
Core Inlet Temperature (°F)		
Core Inlet Mass Flux ($Mlb_m/hr\text{-}ft^2$)		

Figure 07.08-19-1—Maximum Rod $F_{\Delta h}$ vs. Burnup

[ ]

Figure 07.08-19-2—Axial Flux Shapes, U.S. EPR FSAR v. D3

Figure 07.08-19-3—17-Channel, 20-Rod 1/8th Core LYNXT Model

Notes:

1. Bold Print = Rod Heat Structure
2. Regular Print = Coolant Channel

Figure 07.08-19-4—Moderator Reactivity Curves, U.S. EPR FSAR v. D3

Figure 07.08-19-5—Doppler Reactivity Curves, U.S. EPR FSAR v. D3

Figure 07.08-19-6—Scram Reactivity Curves, U.S. EPR FSAR v. D3

Question 07.08-21:

Provide additional explanation of the changes made to S-RELAP5 as utilized for the D3 analysis presented in ANP-10304 Rev 1, including the following:

a. A description of Heat Transfer modifications (i.e., fluid temperature, Inayatov multiplier, LIQHTC) made to the S-RELAP5 code and the purpose of the change, and

b. The validation basis for T_{avg} change made to the S-RELAP5 code.

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 Appendix A to ANP-10304 Rev 1 states that minor changes to the S-RELAP5 code were made to reflect improved heat transfer in the steam generator secondary system.

Response to Question 07.08-21:

Purpose of S-RELAP5 heat transfer model modification

The S-RELAP5 heat transfer modification provides a more realistic computation of steam generator (SG) heat transfer.

Previous S-RELAP5 computations have used different input choices to increase SG heat transfer to achieve target secondary side pressure during steady state initialization. Typical input choices include an increase in SG tube area and using a small hydraulic diameter to increase the SG heat transfer. Using more realistic input options achieves the target secondary side pressure.

Description of the changes

S-RELAP5 includes two new input options that improve SG heat transfer:

1. An option for "T_{ave}".
2. An option to use the "Inayatov" heat transfer multiplier.

Both options improve the accuracy of heat transfer computation in SGs and aid in achieving the target secondary side pressure. Because an adjustment in heat transfer is needed to achieve the target secondary side pressure, the LIQHTC heat transfer coefficient multiplier available in S-RELAP5 input is applied.

Option to Use Average Fluid Temperature for Wall Heat Transfer

The original S-RELAP5 heat transfer computation from the wall to the fluid is based on the fluid nodal temperature adjacent to the surface of the heat structure. Figure 07.08-21-1 shows two scalar nodes that are the fluid control volumes for mass and energy. Temperatures are shown at scalar cell centers, with T_K on the left side and T_L on the right. The heat transfer from a heat

structure adjacent to Node L uses T_L, but that temperature is assigned to the nodal boundary (exit). If the flow is from left-to-right, T_K is assigned to j and T_L is assigned to j+1.

[ ]

Option to Include Inayatov Multiplier

Single-phase, turbulent, forced convection heat transfer coefficient in S-RELAP5 is based on the Dittus-Boelter correlation as follows:

$$h = \max\left[0.023\,\mathrm{Re}_f^{0.8}\,\mathrm{Pr}_f^{0.4}, 7.86\right]\frac{k_f}{D_h}$$

[ ]

LIQHTC Multiplier

[ ]

Validation approach

The validation computations demonstrate the improved accuracy of the T_{ave} option. [ ]

[ ]

FSAR Impact:

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

Table 07.08-21-1—Acronym List

Nomenclature		Subscripts	
C	Specific heat capacity, BTU/(lbm-°F)	Ave	Average
D	Diameter, ft	Exit	Exit
D_h	Hydraulic diameter, ft	F	fluid, liquid
F	Chen correlation F-Factor	In	Inlet
h	Heat transfer coefficient, BTU/(sec-ft^2-°F)	J	Nodal index
k	Thermal conductivity, BTU/(sec-ft-°F)	Mac	Macro
m	Mass flow rate, lbm/sec	Mic	Micro
P	Tube pitch, ft	Sat	saturation
P_h	Heated perimeter, ft	W	Wall
Pr	Prandtl number		
q"	Heat flux, BTU/(sec-ft^2)		
Re	Reynolds number		
T	Temperature, °F		
x	Axial distance, ft		
X	Steam quality, mass fraction		
μ	Viscosity, lbm/(ft-sec)		
ρ	Density, lbm/ft^3		
X_{tt}^{-1}	Martinelli Parameter		

Figure 07.08-21-1—S-RELAP5 One-Dimensional Nodalization Scheme

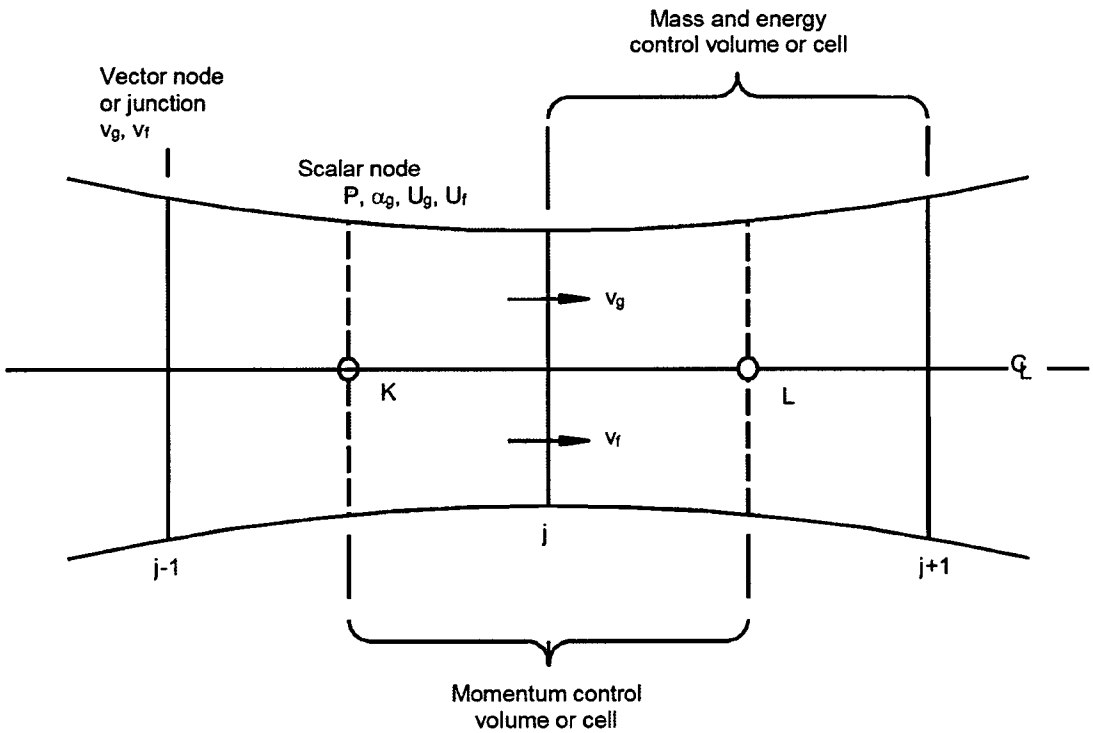


Figure 07.08-21-2—S-RELAP5 Nodal Model for Simple Steam Generator

[ ]

Figure 07.08-21-3—Steam Generator Solutions using T_{exit}

Figure 07.08-21-4—Steam Generator Solutions using T_{ave}

Figure 07.08-21-5—Single-Phase Heat Transfer Coefficients

Figure 07.08-21-6—Two-Phase Heat Transfer Coefficients

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.

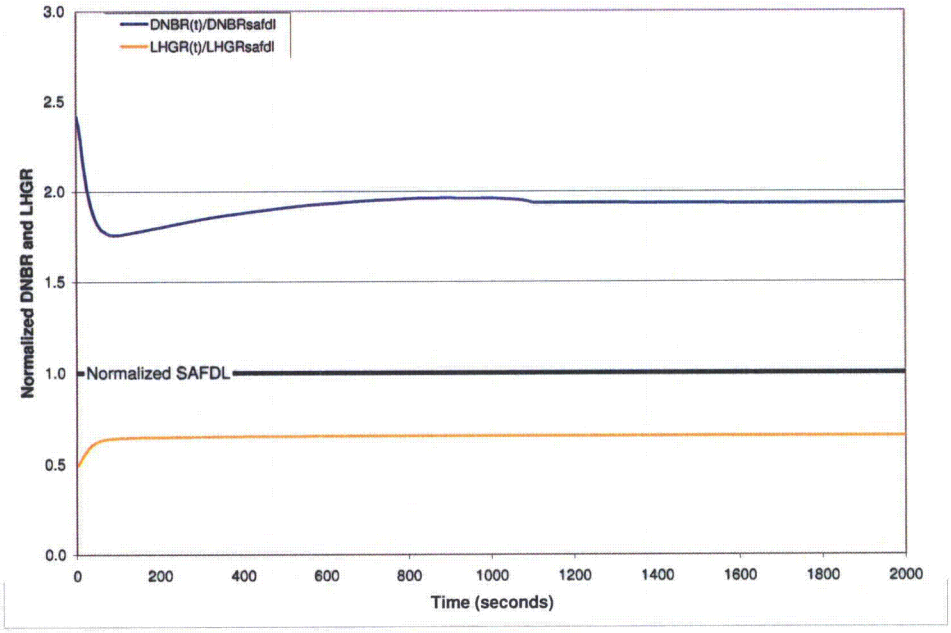
For the increase in steam flow analysis in the U.S. EPR FSAR Tier 2 Chapter 15, the uncertainties applicable to the Low DNBR and High LPD reactor trips are applied as described in the the Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report, ANP-10287P-000. To make sure the DNBR 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.

FSAR Impact:

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

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



Question 07.08-26

Provide the following additional information regarding the Complete Loss of Forced Coolant Flow as described in ANP-10304 Rev 1 Section A.3.4.2:

a. A comparison of the RCS flow coastdown (normalized flow versus time from t=0 sec to 5 sec) for the FSAR Tier 2, Chapter 15 analysis versus the D3 analysis presented in Section A.3.4.2 of ANP-10304 Rev 1,

b. A plot of core average heat flux (normalized core average heat flux versus time from t=0 sec to 10 sec) for the D3 analysis,

c. A plot of DNBR (DNBR normalized to the SAFDL) versus time (t=0 sec to 5 sec) for the D3 analysis, and

d. A table that lists the sequence of events for the Complete Loss of Forced RCS Flow, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and turbine trip for the D3 analysis.

In addition, explain the differences in initial DNBR margin between the FSAR Tier 2, Chapter 15 analysis and the D3 analysis, i.e., identify the best-estimate assumptions and assess their beneficial effects on DNBR.

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.3.4.2 provides an engineering analysis of the Complete Loss of Forced RCS Flow event (AOO). The D3 analysis shows that reactor trip actuation occurs on a DAS Low RCS Flow (Two Loops) signal, which is somewhat later that the PS reactor trip actuation on Low RCP Speed (DAS does not have a Low RCP Speed reactor trip function). The Complete Loss of Forced Coolant Flow is a rapidly occurring event with a calculated minimum DNBR occurring at around 3 seconds, and it is the limiting DNBR AOO. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-26

c) Figure 07.08-26-1 compares the reactor coolant system (RCS) flow coastdown (normalized flow versus time) during the complete loss of flow event used in the U.S. EPR FSAR Tier 2, Chapter 15 analysis and in the diversity and defense-in-depth (D3) analysis. In the U.S. EPR FSAR Tier 2, Chapter 15 analysis, the reactor trip (RT) setpoint is reached at 1.5 seconds and the Δ departure from nucleate boiling ratio (DNBR) maximum at 3.0 seconds.

d) Figure 07.08-26-2 shows the normalized core average heat flux for the D3 analysis.

e) Figure 07.08-26-3 shows the DNBR normalized to the specified acceptable fuel design limits (SAFDL) versus time.

f) Table 07.08-26-1 shows the sequence of events for the D3 analysis of the complete loss of RCS flow event.

The diverse actuation system (DAS) low RCS loop flow RT is delayed by 2.85 seconds regarding the low reactor coolant pump (RCP) speed trip in the U.S. EPR FSAR Tier 2, Chapter 15 analysis. However, the application of best-estimate conditions affecting both the initial DNBR, and the DNBR performance through the event offsets the trip delay. As a result, the transient is less limiting than the case presented in U.S. EPR FSAR Tier 2, Section 15.3.2.

The Response to RAI 413, Question 07.08-19 discusses the differences between the power distributions used in the D3 analyses and the power distributions in U.S. EPR FSAR Tier 2, Chapter 15. When compared to the power distributions used in the U.S. EPR FSAR Tier 2, Chapter 15 analysis, the best-estimate conditions show a significant reduction in Fq (50 percent) and FΔH (15 percent) (see Table 07.08-26–2). The result shows a significant improvement in initial DNBR performance (see Table 07.08.26-2).

The DNBR performance during the transient is improved as a result of the following:

I. Best-estimate flow and core bypass – The initial flow assumed in the U.S. EPR FSAR Tier 2, Chapter 15 analysis is based on thermal design flow with a design core bypass fraction (values presented in Table 07.08-26-2).

II. SCRAM reactivity – The Response to RAI 413, Question 07.08-23 discusses the differences in the SCRAM reactivity curves. Regarding the U.S. EPR FSAR Tier 2, Chapter 15 analysis, these differences illustrate that under best estimate-assumptions, SCRAM reactivity is greater at any given time following a RT, which results in a lower power-to-flow ratio throughout the event. These results benefit DNBR performance.

III. Moderator feedback – The U.S. EPR FSAR Tier 2, Chapter 15 complete loss of flow analysis does not credit the effects of moderator feedback, which results in the core power level remaining constant until the RT is initiated. A negative, nonzero moderator coefficient of reactivity in the D3 analysis results in an immediate and continuous reduction in core power until the time of RT because the RCS temperature increases following the loss of the four RCPs. Figure 07.08-26–2 shows the core average heat flux data, which illustrates this power reduction.

U.S. EPR Technical Report ANP-10304 will be revised to clarify the Complete Loss of Forced Coolant Flow in the U.S. EPR D3 analysis as described in this response. To support submittal of complete and consistent information, and considering multiple RAI responses and design changes communicated to the NRC staff, U.S. EPR Technical Report ANP-10304 revisions described in this response will be submitted with the Response to RAI 413, Question 07.08-13.

FSAR Impact:

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

Table 07.08-26–1—Complete Loss of RCS FLOW – Sequence of Events

Event	Time (s)
Trip RCPs 1, 2, 3, 4	0.00
DAS Low RCS Loop Flow Setpoint reached	3.74
DAS Low RCS Loop Flow signal	4.24
Rods begin to drop	4.65
DNBR minimum	5.00
TT Trip	5.24

Table 07.08-26–2—Complete Loss of RCS FLOW – Comparison U.S. EPR FSAR Tier 2, Chapter 15 versus D3

Parameters	D3, DNBR Evaluation	U.S. EPR FSAR Tier 2, Section 15.3.2
Initial reactor power (MWt)	4590	4612
Average RCS temperature (°F)	594	594±4
Initial PZR pressure (psia)	2250	2250±50
Initial RCS loop flow rate (gpm)	124,741	119,692
Rods begin to drop (sec)	4.65	1.8
Fq	[ ]	2.6
FΔH	[ ]	1.70
Scram (pcm) BOC, HFP EOC, HFP	[ ]	6161($10.35) 7353 ($14.28)
Core bypass fraction (%)	[ ]	5.5
Initial DNBR (normalized to SAFDL)	**2.41**	**1.36**

Figure 07.08-26–1—Complete Loss of RCS Flow – RCS Flow Coastdown Comparison

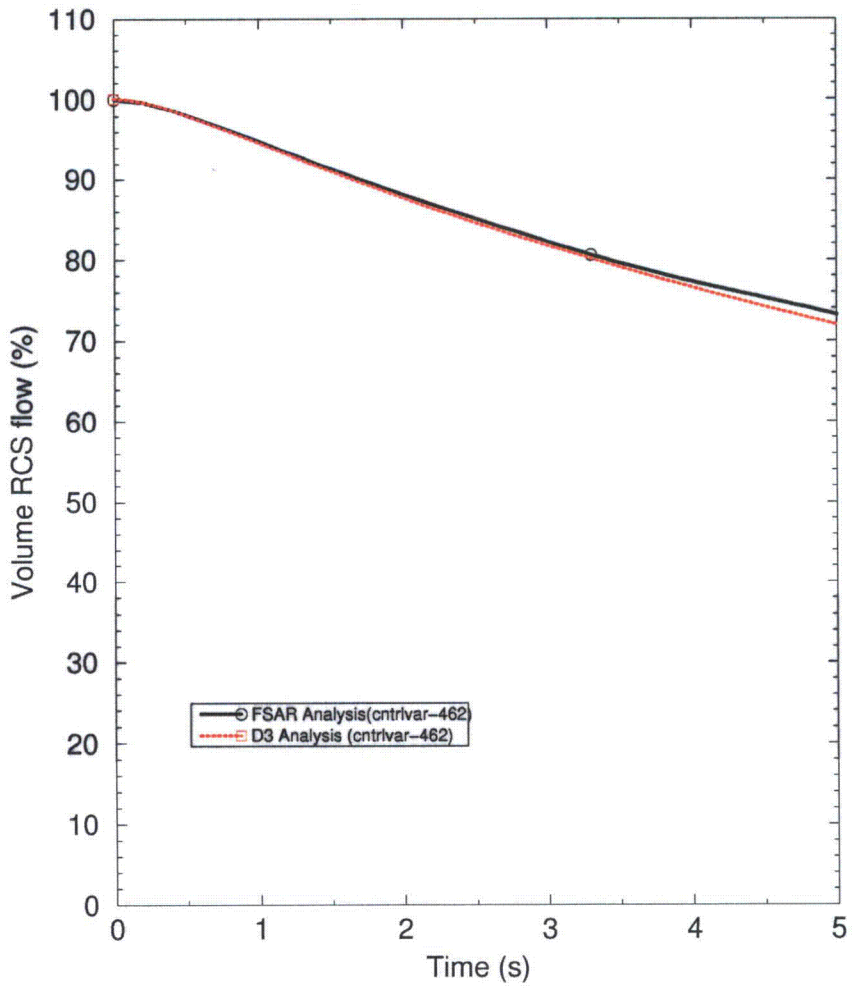


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Figure 07.08-26–2—Complete Loss of RCS Flow – Core Average Heat Flux (D3 analysis)

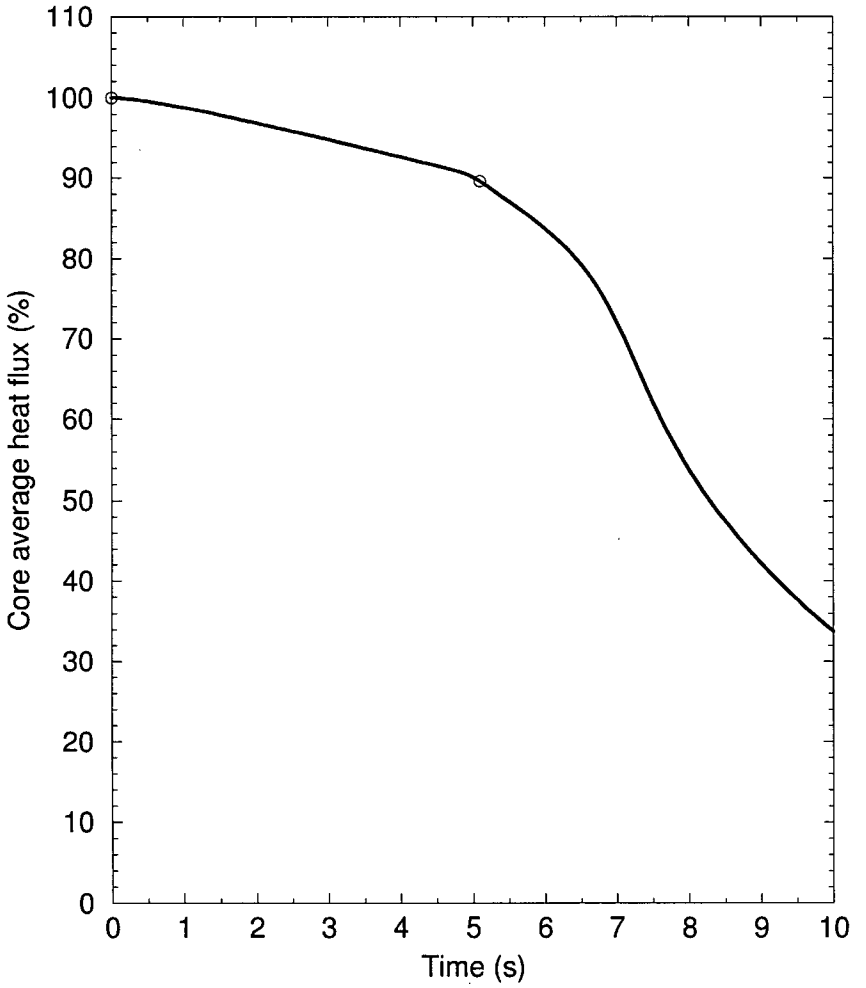

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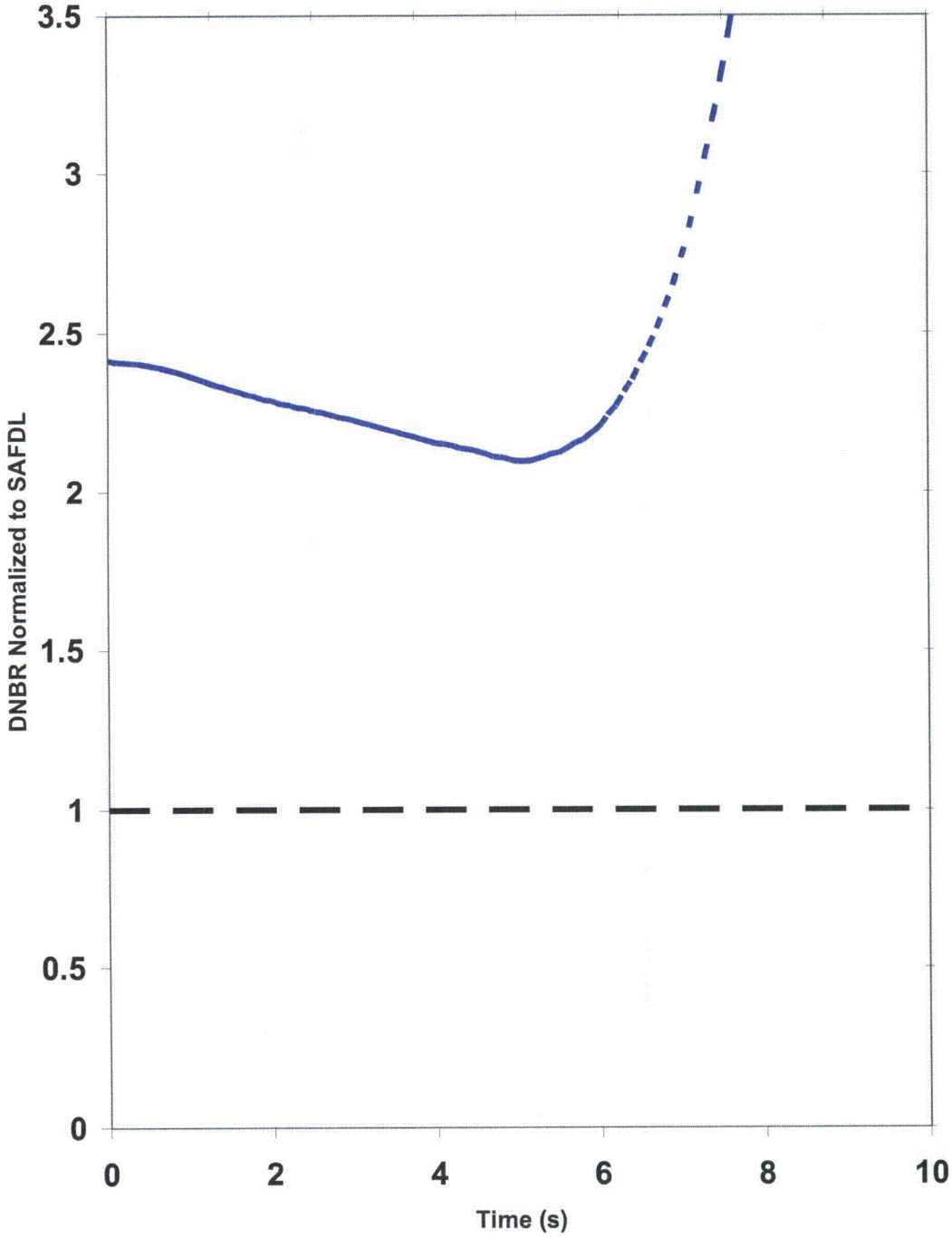
Figure 07.08-26–3—Complete Loss of RCS Flow – Plot of Minimum DNBR Normalized to SAFDL



Question 07.08-27

For the RCP Rotor Seizure event described in ANP-10304 Rev 1 Section A.3.4.3, provide the following additional information:

a. A comparison of the RCS flow coastdown rate assumed for D3 evaluation versus the flow coastdown rate shown FSAR Figure 15.3-9;

b. The calculated initial (t=0 seconds) and minimum DNBR for the RCP Rotor Seizure event as analyzed in both Section 15.3.3 of the FSAR and the D3 analysis;

c. A table that lists the sequence of events for the RCP Rotor Seizure event, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and turbine trip.

In addition, explain the differences in initial DNBR margin between the FSAR Chapter 15 analysis and the D3 analysis, i.e., identify the best estimate assumptions and assess their beneficial effects on DNBR.

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.3.4.3 provides an engineering argument for the RCP Rotor Seizure event (PA), stating that DAS will actuate a reactor trip on Low-Low RCS Flow (one loop), and therefore provide protection comparable to the PS as shown in the FSAR Section 15.3.3 analysis. The FSAR analysis shows that the DNBR SAFDL is exceeded, resulting in fuel damage. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-27

The reactor coolant pump (RCP) rotor seizure event was not specifically analyzed for the diversity and defense-in-depth (D3) assessment, but was evaluated by a quantitative comparison with U.S. EPR FSAR Tier 2, Chapter 15 analysis using best estimate assumptions. The overall conclusion from this comparison was that the U.S. EPR FSAR Tier 2, Chapter 15 analysis is bounding and that fuel failure fractions and offsite consequences would be less for D3 as a result of the use of best estimate assumptions.

In the D3 assessment the potential detrimental effect of the lower diverse actuation system (DAS) low-low reactor coolant system (RCS) flow trip setpoint and longer time delay was offset by the use of best estimate assumptions. The best estimate assumptions were all beneficial in the comparison demonstrating, in overwhelming fashion, that the U.S. EPR FSAR Tier 2, Chapter 15 analysis is bounding. This is illustrated below by discussing the U.S. EPR FSAR Tier 2, Chapter 15 analysis assumptions with a comparison to the corresponding best estimate assumption.

The U.S. EPR FSAR Tier 2, Chapter 15 system and core response for the rotor seizure event is evaluated with the S-RELAP5 and LYNXT computer codes. It was conservatively predicted that eight percent of the core experiences departure from nucleate boiling (DNB)-induced cladding failure. Radiological analysis of this event assumed a bounding fuel failure fraction of 9.5 percent. As presented in U.S. EPR FSAR Tier 2, Table 15.3-1—Decrease in Reactor Coolant System Flow Rate Events – Key Input Parameters, the initial conditions are biased to achieve conservative results. Specifically, the parameters power, pressure, and RCS flow rate, are chosen to penalize the departure from nucleate boiling ratio (DNBR) evaluation. Table 07.08.27-1 presents the comparison between parameters used in U.S. EPR FSAR Tier 2, Section 15.3 and those assumed in the D3 assessment.

Although the low-low RCS flow trip would be delayed in the D3 analysis (DAS setpoint) as compared to the U.S. EPR FSAR Tier 2, Chapter 15 analysis, this delay is only a fraction of a second as a result of the steep flow decrease from the seized rotor shown in U.S. EPR FSAR Tier 2, Figure 15.3-9. The flow to the inlet of the core in the D3 analysis would not be impacted as much because the RCPs in the unaffected loops would continue to run. In the U.S. EPR FSAR Tier 2, Chapter 15 analysis, loss of offsite power (LOOP) was assumed upon reactor trip at which time the RCPs in the unaffected loops coast down. The initial flow in the RCS is higher under best estimate conditions. The initial flow assumed in the U.S. EPR FSAR Tier 2, Chapter 15 analysis is based on thermal design flow with a design core bypass fraction. Overall these flow effects are conservatively judged to be a slight penalty on DNB performance.

The initial power distributions in the U.S. EPR FSAR Tier 2, Chapter 15 analysis as compared to best estimate conditions result in a significant reduction in Fq (50 percent) and $F\Delta H$ (15 percent) as illustrated in Table 07.08-27–1. The Response to RAI 413, Question 07.08-19 provides an overview of the differences between the power distributions used in the D3 analyses as compared to the power distributions within the U.S. EPR FSAR Tier 2, Chapter 15. Overall, this results in a significant improvement in initial DNB performance as illustrated in Table 07.08.27-1.

The Response to RAI 413, Question 07.08-23 provides an overview of the differences in the SCRAM reactivity curves which illustrate that, under the best estimate assumptions, with respect to the U.S. EPR FSAR Tier 2, Chapter 15 analysis, SCRAM reactivity is greater at any given time following a reactor trip which results in a lower power-to-flow ratio throughout the event which is beneficial with respect to DNBR performance.

This discussion demonstrates that the credited best estimate items for D3 more than offset the effects of a delayed reactor trip on DAS, and that the DNB performance is equal to or better than in the U.S. EPR FSAR Tier 2, Chapter 15 analysis. Therefore, the U.S. EPR design is adequate in addressing a software common cause failure in the protection system (PS) during RCP rotor seizure and shaft break events.

FSAR Impact:

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

Table 07.08-27-1—D3 Rotor Seizure Event Parameters– Comparison FSAR Tier 2, Chapter 15 versus D3

Parameters	D3, DNBR Evaluation	FSAR Tier 2, Section 15.3.3 DNBR evaluation
Initial reactor power (MWt)	4590	4612
Average RCS temperature (°F)	594	594±4
Initial PZR pressure (psia)	2250	2250±50
Initial RCS loop flow rate (gpm)	124,741	119,692
Low-low flow trip setpoint (time delay)	44% (1.30 sec)	50% (1.05 sec)
Flow to core Inlet	RCPs in 3 unaffected loops continue	Impacted by LOOP (no RCPs available)
Fq		2.6
FΔH		1.70
Scram (pcm) BOC, HFP EOC, HFP		 6161($10.35) 7353 ($14.28)
Core bypass fraction (%)		5.5
Initial DNBR (normalized to SAFDL)	2.41	1.30

Question 07.08-28

For the Uncontrolled RCCA Withdrawal at Power event described in ANP-10304 Rev 1 Section A.3.5.2, provide the following additional information:

a. The calculated initial (t=0 seconds) and minimum DNBR for the Uncontrolled RCCA Withdrawal at Power event, and

b. A table that lists the sequence of events for the Uncontrolled RCCA Withdrawal at Power event, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and positioning of the RCCA bank being withdrawn.

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 Uncontrolled RCCA Withdrawal at Power event is analyzed assuming full power initial conditions at both BOC and EOC conditions, and with the RCCAs inserted to the Technical Specification Power Dependent Insertion Limit (PDIL). The time in cycle life-time (BOC vs. EOC) affects moderator temperature coefficient and control rod bank worth. The FSAR Section 15.4.2 analysis states that the reactor system is protected by PS Low DNBR, High LPD, Excore High Rate of Change, High Core Power Level, and High Pressurizer Level reactor trip functions, none of which are provided by DAS. The D3 engineering analysis provided in ANP-10304 Rev 1 Section A.3.5.2, however, shows that DAS actuates a reactor trip on Low SG Level and that reactor power peaks at approximately 108 percent. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to 07.08-28

Item 07.07-28 (a):

The departure from nucleate boiling ratio (DNBR) normalized to the specified acceptable fuel design limit (SAFDL) values are given rather than the calculated values to facilitate direct comparison to the results presented for the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power transient in U.S. EPR FSAR Tier 2, Figure 15.4-54.

For the diversity and defense-in-depth (D3) Uncontrolled RCCA Withdrawal at Power transient analysis, the lowest DNBR is reached during the beginning of cycle (BOC) case. The transient is simulated by an insertion of reactivity corresponding to the withdrawal of the Bank D from the PDIL to the all rods out position at the maximum RCCA extraction speed. The initial DNBR normalized to the specified acceptable fuel design limit (SAFDL) is 2.41 and the minimum DNBR normalized to the SAFDL value reached during the transient is 2.00 at 287.4 seconds.

Item 07.07-28 (b):

Table 07.08-28-1 presents the sequence of events for the Uncontrolled RCCA Withdrawal at Power transient.

Table 07.08-28-1 — RCCA Withdrawal at Power – Sequence of Events

Event	Time (sec)
Bank D Withdrawal from PDIL beginning	0.0
Bank D withdrawal from PDIL end	72.0
DAS low SG level delay (RT signal)	290.1
DAS RT with delay (rod release for scram)	290.5
Minimum DNBR	288.0
DAS turbine trip (TT) with delay	291.1

The Uncontrolled RCCA Withdrawal at Power transient is shown to be more challenging in the U.S. EPR FSAR Tier 2, Section 15.4.2 evaluation than in the D3 analysis. This is because of differences between the U.S. EPR FSAR Tier 2, Chapter 15 and D3 analyses that impact the initial DNBR and linear power density (LPD) margins, the use of best estimate core physics parameters, and other modeling differences that contribute to the overall event progression and severity.

Initial DNBR and LPD Margin

The Response to RAI 413, Question 07.08-19 will provide an overview of the differences between the best estimate assumptions used in the 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 uncontrolled RCCA withdrawal at power, as compared to the U.S. EPR FSAR Tier 2, Section 15.4.2 analysis, include the following:

Power Distributions

The power distribution applied for the D3 Uncontrolled RCCA Withdrawal at Power DNB and LPD analysis is a representative nominal power distribution for beginning of cycle (BOC) conditions. These best estimate power distributions will be described in the Response to RAI 413, Question 07.08-19, Part b.

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

To initiate the Uncontrolled RCCA Withdrawal at Power transient for the U.S. EPR FSAR Tier 2, Section 15.4.2 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 15.1-17, the power distribution was scaled from a nominal FDH of 1.42 to 1.47 (3.5 percent increase) to initiate the transient.

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. This linear heat rate is compared to the limiting linear heat rate corresponding to fuel centerline melt or one percent clad strain.

For the Uncontrolled RCCA Withdrawal at Power analysis in the U.S. EPR FSAR Tier 2, Section 15.4.2, the uncertainties applicable to the Low DNBR and High LPD reactor trips are applied as described in the Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report, ANP-10287P-000. To provide reasonable assurance that 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 provide reasonable assurance that the LPD is less than the LPD corresponding to fuel centerline melt or one percent clad strain with 95 percent probability and a 95 percent confidence level.

Additional Key Differences Relevant to the Uncontrolled Bank Withdrawal Event

In the U.S. EPR FSAR Tier 2, Section 15.4.2 analysis the moderator is at its maximum value (zero, as shown U.S. EPR FSAR Tier 2, Figure 15.4-7) to minimize the moderator feedback. The moderator and Doppler coefficients for U.S. EPR FSAR Tier 2, Section 15.4.2 and D3 analyses are compared in the Response to RAI 413, Question 07.08-23. The Uncontrolled RCCA Withdrawal at Power causes an RCS temperature increase. Because the best estimate moderator coefficient in the D3 analysis is negative, the response to the RCS temperature increase is an insertion of negative reactivity which compensates the reactivity increase due to the RCCA withdrawal as shown in Figure 07.08-28-1.

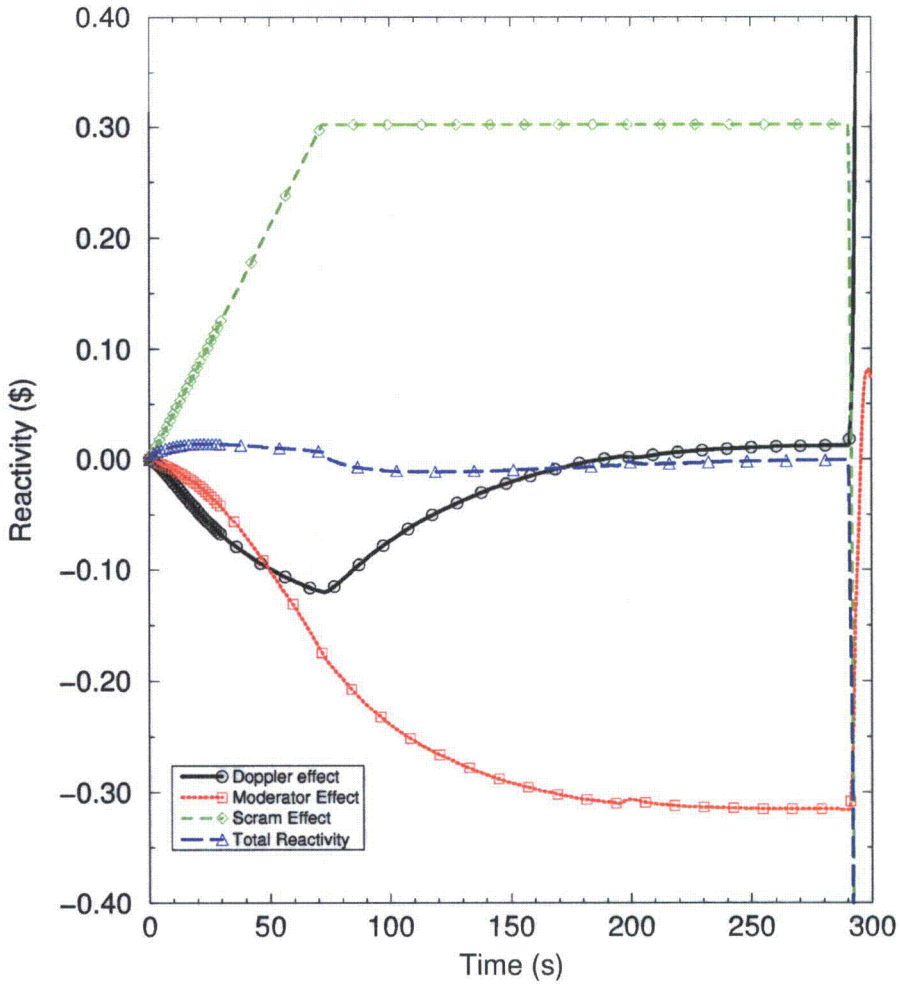
Another key difference in the modeling of this event is the fact that in the U.S. EPR FSAR Tier 2, Section 15.4.2 analysis the reactivity addition due to the RCCA withdrawal was assumed to continue until the reactor trip was initiated while the D3 analysis maintained the reactivity addition magnitude so that the overall reactivity addition from the withdrawal was constrained to the worth of the length of the control rods withdrawn from the PDIL.

The negative reactivity contribution of the moderator and the termination of positive reactivity insertion upon full withdrawal of the RCCA contribute to the transient in a way that limits the peak power achieved and to turn power downward prior to the reactor trip, as shown Figure A.3.5-1 of ANP-10304, Revision 1. This reduces the DNBR degradation during the event compared to that of the U.S. EPR FSAR Tier 2, Section 15.4.2 analysis. As a result, the U.S. EPR FSAR uncontrolled RCCA withdrawal analysis reached a peak power of about 120 percent with a minimum DNBR value normalized to the SAFDL of 1.11 while the D3 analysis reached a peak power of approximately 108 percent with a minimum DNBR value normalized to the SAFDL of 2.0. Both analyses demonstrate adequate margin to the safety limit.

FSAR Impact:

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

Figure 07.08-28–1—RCCA Withdrawal at Power Event – Reactivity



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Question 07.08-29

For the Dropped RCCA event described in ANP-10304 Rev 1 Section A.3.5.3, provide the following additional information:

a. A description of the response of RCSL during the limiting Dropped RCCA event and its affect t on the calculated results of the transient,

b. The calculated initial (t=0 seconds) and minimum DNBR for the limiting Dropped RCCA event, and

c. An explanation of why the drop of a single RCCA is not more limiting than the drop of an RCCA bank, considering that the drop of a single RCCA results in a more severe radial peaking pattern (nonsymmetric core condition) with a similar return to full power conditions. Provide the calculated radial peaking for both the RCCA bank drop and single RCCA drop to support the explanation.

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.

In the FSAR analysis, the PS actuates a reactor trip on Low DNBR, which is unavailable in DAS. The limiting case, as reported in Section A.3.5.3, is the drop of a full bank of RCCAs from the Power Dependent Insertion Limit (PDIL) position. The D3 analysis of this event shows that following the drop of the RCCA bank, the reactor power returns to full power with no reactor trip actuation. The applicant reports that the best estimate conditions offset the failure of the PS trip functions, and no fuel failures occur. In addition, the D3 analysis assumes that RCSL is automatically controlling RCCAs, which worsens the transient. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-29

a) ANP-10304 Revision 1, Section A.3.5.3 analyzes three cases. Two of these cases are initiated from the all rods out (ARO) condition, and the third is initiated with the controlling bank at the power dependent insertion limit (PDIL). The rod cluster control assembly (RCCA) bank drop events are characterized by an initial decrease in reactor coolant system (RCS) temperature. The average coolant temperature (ACT) function in the reactor control, surveillance and limitation (RCSL) system will withdraw the controlling RCCA bank to restore nominal RCS temperature. Cases initialized with the controlling RCCA bank fully withdrawn (i.e., ARO) are not impacted by the ACT function in RCSL.

   For the limiting case, the drop of Bank A at end of cycle (EOC) conditions initialized with the controlling RCCA bank at the power-dependent insertion limit (PDIL), the RCSL system will automatically initiate a withdrawal of the controlling RCCA bank in order to restore RCS temperature. Following the initial decrease in core power, a return to power occurs, and the associated overshoot causes the maximum power level reached to be 102 percent. Figure 07.08-29-1, Figure 07.08-29-2, and Figure 07.08-29-3 show the controlling bank position

adjusted by the automatic ACT control function in RCSL as a function of time through the transient, the reactor power level, and the RCS average temperature during the transient.

b) The limiting case of RCCA drop is the insertion of 468 pcm at EOC from PDIL without a reactor trip (RT). The initial departure from nucleate boiling ratio (DNBR) normalized to the specified acceptable fuel design limit (SAFDL) is 2.66, and the minimum DNBR normalized to SAFDL is 1.63. The improvement in the initial DNBR margin resulting from the use of best-estimate conditions is shown by comparing the 2.66 initial DNBR normalized to the SAFDL value for the D3 analysis to the 1.38 initial DNBR normalized to the SAFDL value for U.S. EPR FSAR Tier 2, Chapter 15.

c) In each of the dropped RCCA and dropped RCCA bank events, the core power distribution is perturbed so that it leads to an increase in the magnitude of the radial power peak. Because each of these events have a similar return to full power, the DNBR performance is dominated by the increase in radial peaking that results from the drop of the RCCA or RCCAs. Table 07.08-29–1 shows the calculated radial peaking factors for the largest single RCCA drop and for the drop of the RCCA Bank A for two times in life in two different fuel cycles. Comparing the radial ($F\Delta H$) augmentation factors shows that the peaking factors for the RCCA bank drop are consistently larger. While the larger magnitudes of negative reactivity insertion from the RCCA bank drop cases lead to larger decreases in core inlet temperature than single RCCA drop cases, the impact on DNBR performance is small when compared to the increase in the radial power peak.

U.S. EPR Technical Report ANP-10304 will be revised to clarify the Dropped RCCA event in the U.S. EPR D3 analysis as described in this response. To support submittal of complete and consistent information, and considering multiple RAI responses and design changes communicated to the NRC staff, U.S. EPR Technical Report ANP-10304 revisions described in this response will be submitted with the Response to RAI 413, Question 07.08-13.

FSAR Impact:

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

Table 07.08-29-1—Radial Power Peaking Factors (FDH) for Single RCCA Drop and RCCA Bank A Drop

Condition	Single RCCA Maximum F∆H Augmentation	RCCA Bank A F∆H Augmentation
Cycle 1 BOC PDIL		
Cycle 1 EOC PDIL		
Equilibrium Cycle BOC PDIL		
Equilibrium Cycle EOC PDIL		

Figure 07.08-29–1—D3 Rod Drop Event – EOC 468 pcm – D Bank RCCA Extraction by Average Coolant Temperature Control

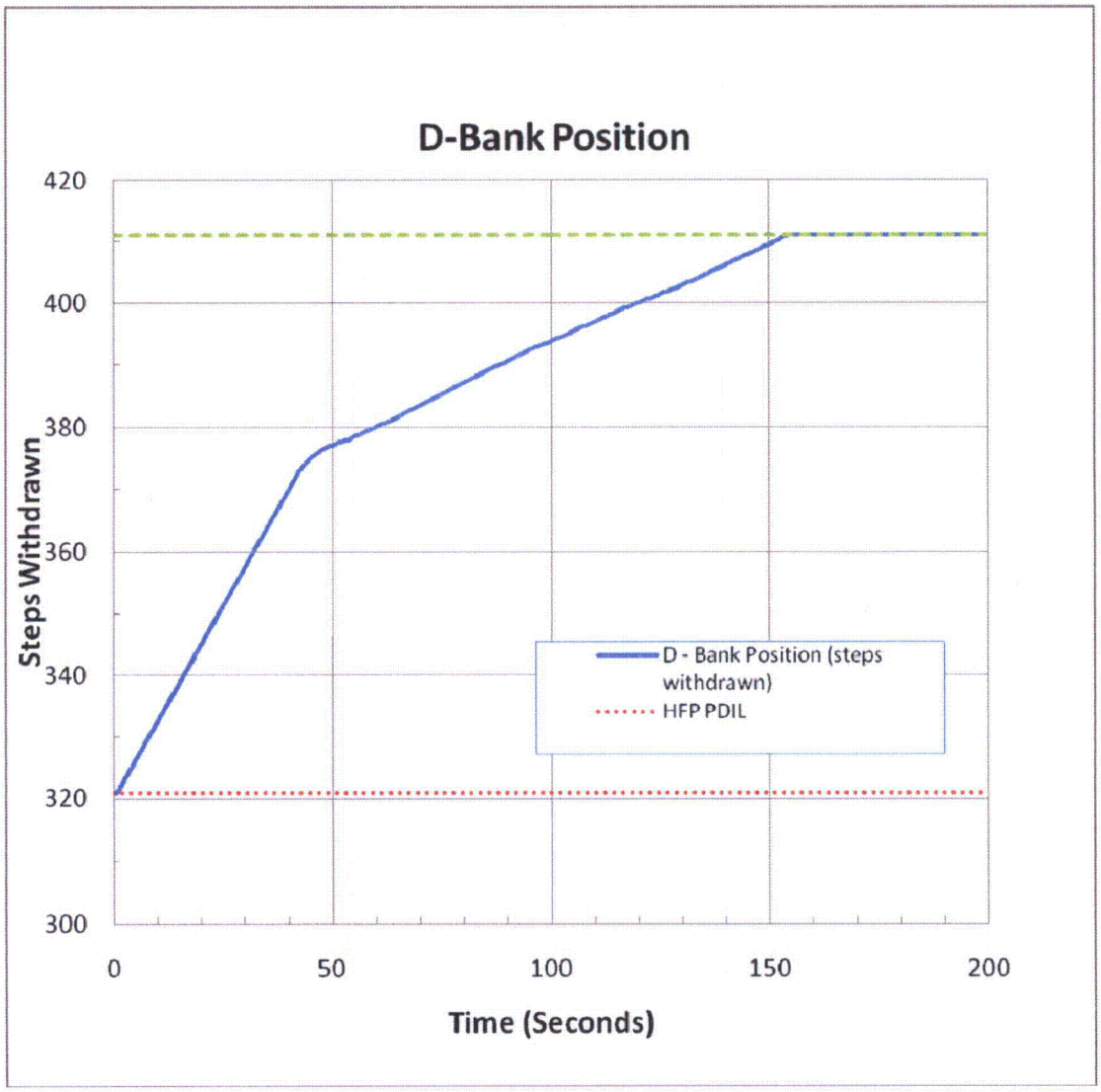

Figure 07.08-29–2—D3 Rod Drop Event – EOC 468 pcm –Reactor Power Level

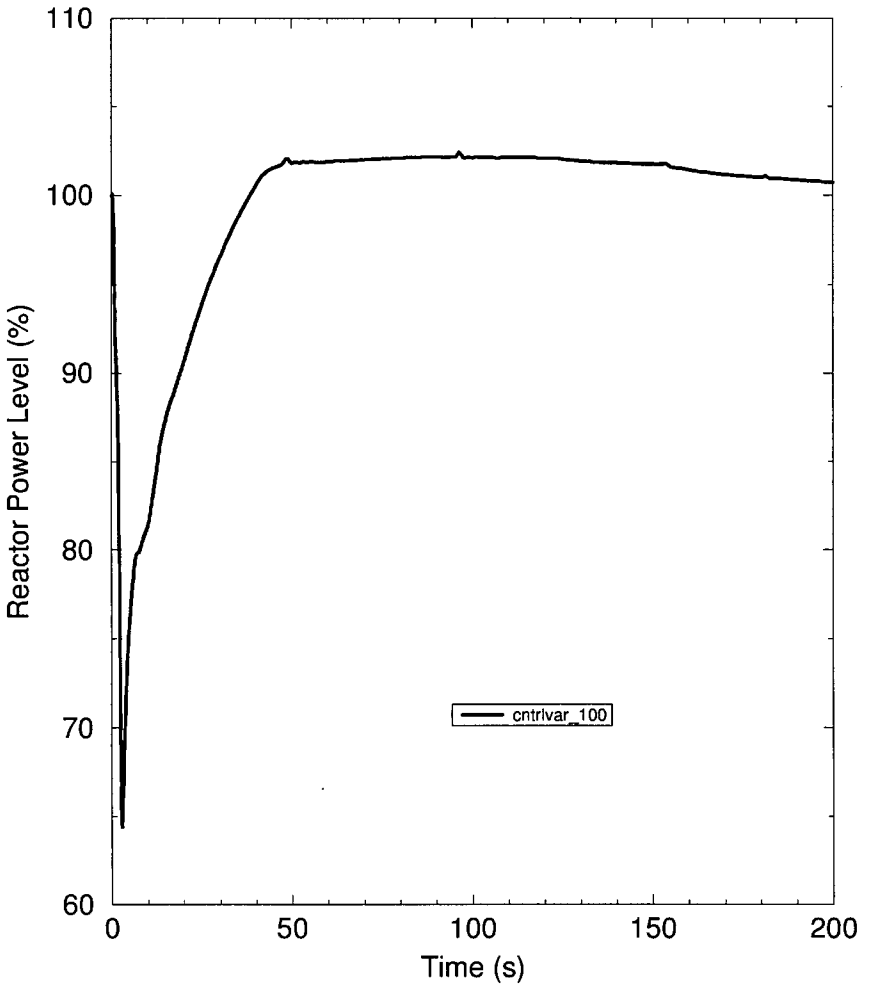

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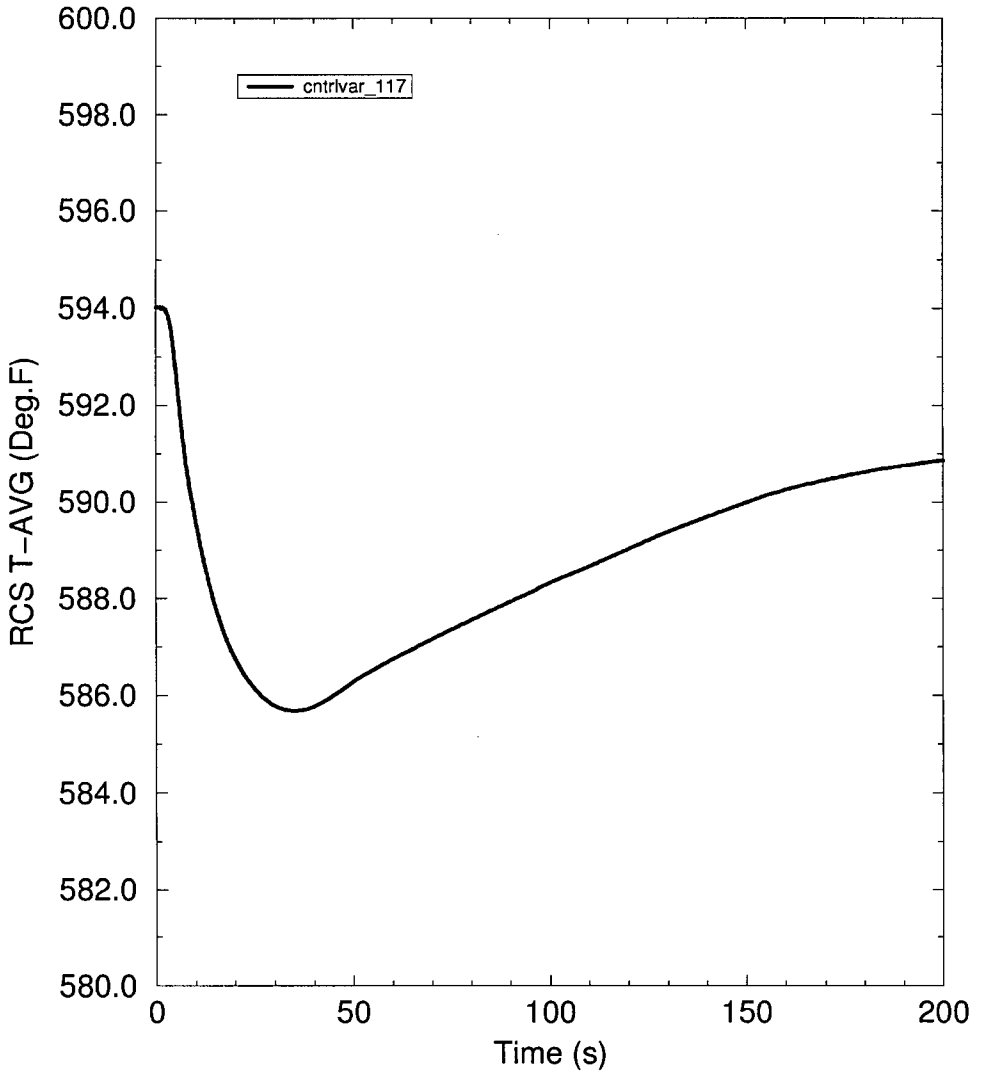
Figure 07.08-29-3—D3 Rod Drop Event – EOC 468 pcm – RCS Average Temperature



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Question 07.08-32:

For the Steam Generator Tube Rupture (SGTR) event described in ANP-10304 Rev 1 Section A.3.7.2, provide the following additional information:

a. An explanation of why the D3 analysis of the SGTR results in SG overfill as compared to the FSAR Section 15.6.3 analysis which does not show SG overfill,

b. The design reference basis for main steam line loading with solid fill at a hydrostatic pressure of 1.25 design pressure, and

c. An assessment of the possibility of steam line water hammer occurrence during the SG overfill portion of the SGTR 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.3.7.2 of ANP-10304 Rev 1 reports that the Steam Generator Tube Rupture (SGTR) event results in an overfill of the affected steam generator and discharge of liquid into the main steam line, whereas the SG does not overfill in FSAR Section Chapter 15 SGTR analysis. With essentially the same manual controls available and credited for the FSAR analysis and D3 analysis, plus availability of the DAS MFW isolation on SG High Level, it is unclear why the SGTR with PS SWCCF results in SG overfill. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-32:

The original evaluation of the diversity and defense-in-depth (D3) steam generator tube rupture (SGTR) event was made by estimating the steam generator (SG) level response. No operator action was credited for isolation of the main feedwater (MFW) valves in this evaluation. Instead, the rate of SG level increase from the U.S. EPR FSAR Tier 2 analysis was extended for the amount of time that it would take the SG level control system to respond. This estimation resulted in a level of 90 percent wide range when the valves reached full closure. A D3-specific analysis was performed with S-RELAP5 to more precisely evaluate the SG overfill behavior in a best estimate SGTR event. The best estimate SGTR analysis credits the following automatic control systems:

- Pressurizer level control: Charging increases and the second pump is started with falling pressurizer level. Letdown isolation occurs on low-low pressurizer level.
- Reactor coolant system (RCS) pressure control: Pressurizer heaters increase power to compensate for falling RCS pressure.
- SG level control: Normal SG level is 49 percent narrow range (NR), and the full load MFW control valve (FLCV) operates to maintain within a control band around the setpoint. Following a reactor trip (RT), the SG level setpoint is temporarily reduced to 34 percent. The setpoint increases with time after the RT. The protection system (PS) would close the full load control valve and isolation valve on RT check back signal, but this function is not available for the D3 analyses. The normal interlock transition from FLCV control to the lower

load control valves works as designed and the low load control valve (LLCV) (<5-20 percent reactor power) or very low load control valve (VLLCV) (<5 percent reactor power) operate to maintain SG level at the programmed setpoint. After the affected SG is isolated, no further FW is introduced to that SG.

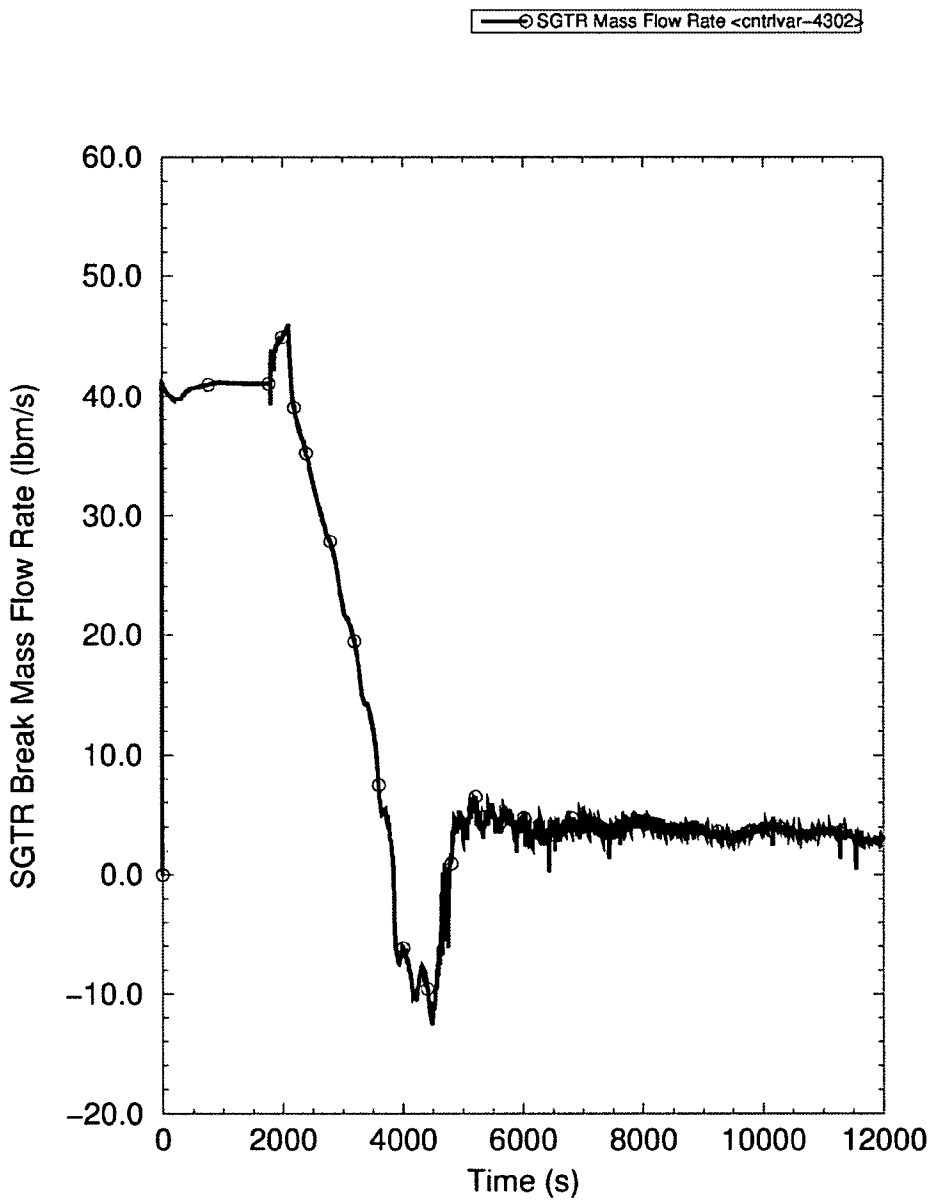
In the D3 SGTR analysis, the operation of the chemical and volume control system (CVCS) system compensates for the break flow (see Figure 07.08-32-1). The pressurizer pressure (see Figure 07.08-32-2) remains above the setpoint, and an automatic RT does not occur prior to 30 minutes. At 30 minutes, the operator performs a manual RT. After RT, the operator performs the actions necessary to isolate the affected SG. In an isolated SG, MFW, emergency feedwater (EFW), and SG blowdown are isolated, the main steam relief train (MSRT) setpoint is raised above the safety injection (SI) shutoff head, and the main steam isolation valve (MSIV) is closed. The operator then manually initiates a cooldown with the unaffected SGs and performs a concurrent RCS depressurization to allow the RCS to be connected to the residual heat removal system (RHRS). The maximum level, 84 percent wide range (58 percent narrow range), is reached at 63 minutes (see Figure 07.08-32-3 and Figure 07.08-32-4). Because the SG does not overfill throughout the event, an evaluation of main steam line loading and main steam line water hammer is not necessary.

Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment," will be revised to include the information requested in this RAI. The revision to Technical Report ANP-10304 will be transmitted separately.

FSAR Impact:

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

Figure 07.08-32-1—D3 SGTR Event – Break Flow



ID:45781 15Feb2011 18:04:27 SGTR_rst2.dmx

Figure 07.08-32-2—D3 SGTR Event – Pressurizer Pressure

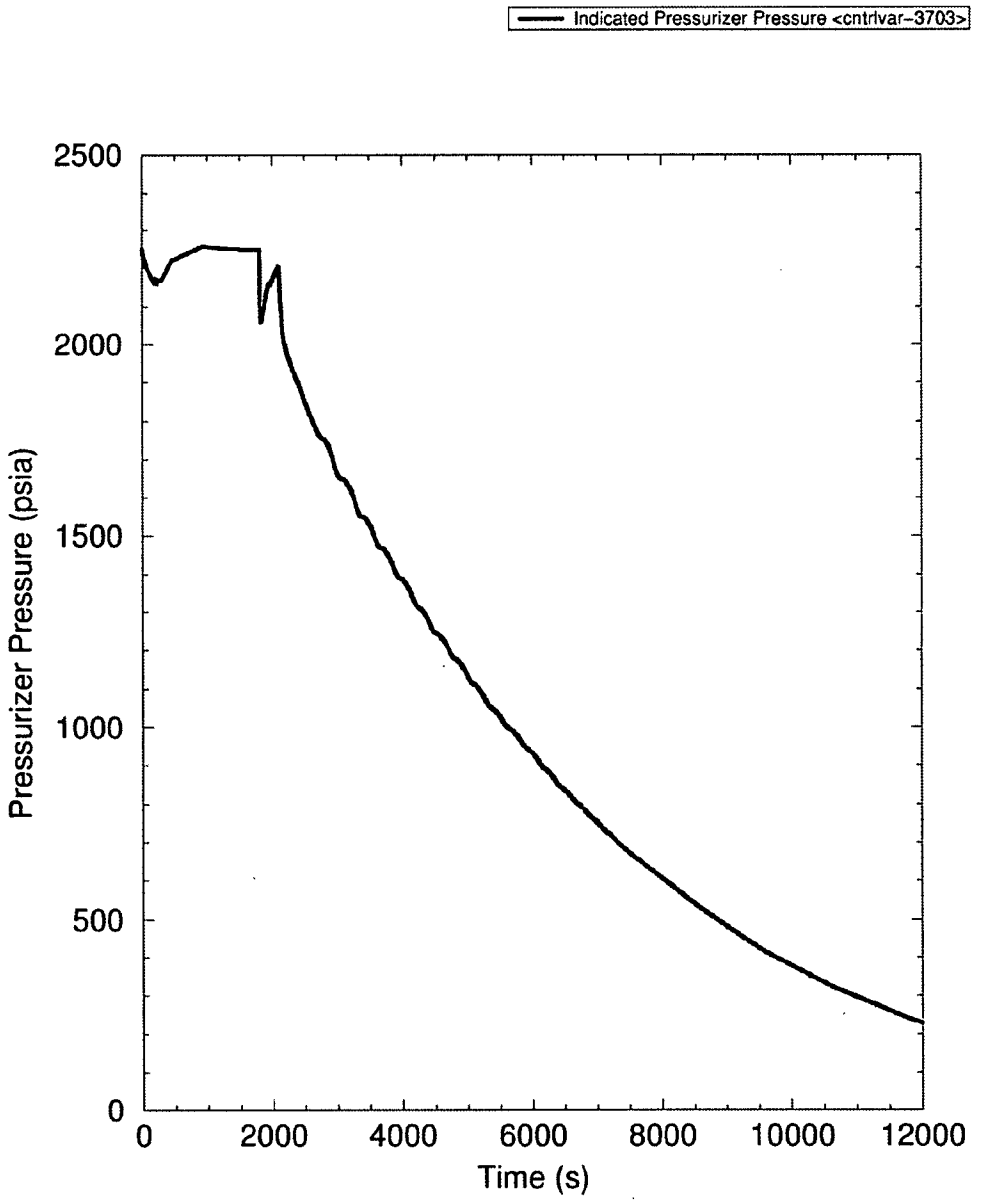


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Figure 07.08-32-3—D3 SGTR Event – SG Level, Wide Range

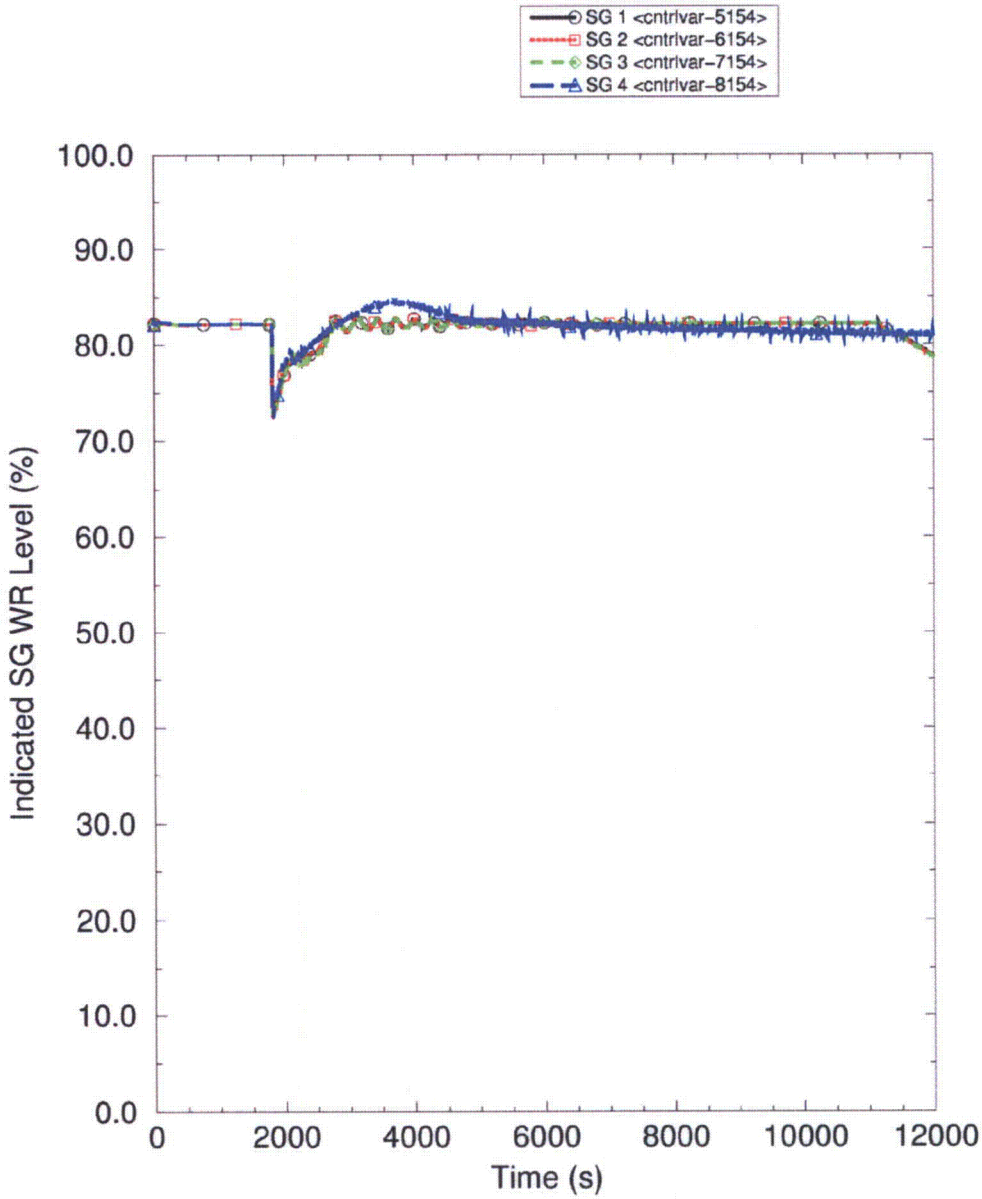

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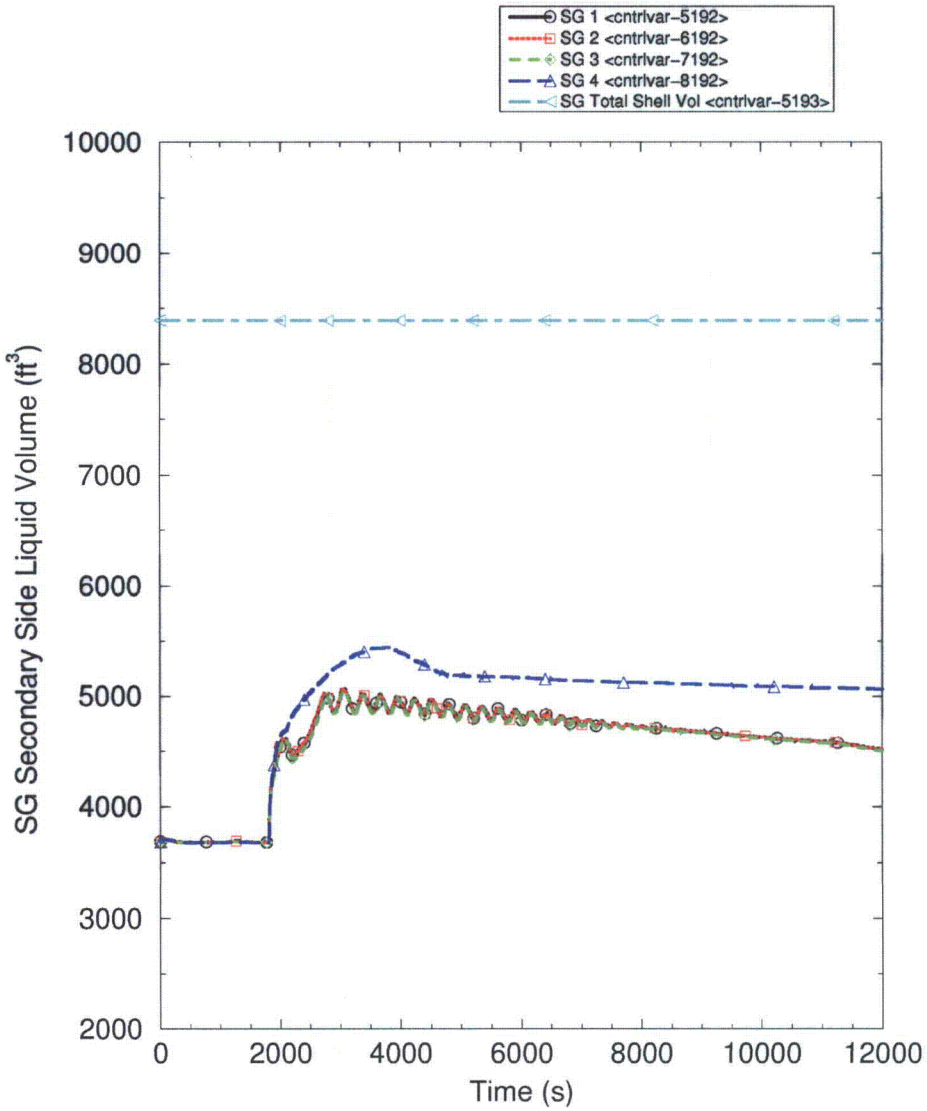
Figure 07.08-32-4—D3 SGTR Event – SG Volume



ID:45781 15Feb2011 18:04:27 SGTR_rst2.dmx

Question 07.08-35:

Identify the credited diverse means to address the loss of the PS initiated safety functions of RT on a high pressurizer level and CVCS isolation on high pressurizer level, as discussed in ANP-10304, Revision 1, Section A.3.6.2," CVCS Malfunction that Increases RCS Inventory," in accordance with the D3 policy stated in SRM to SECY-93-087, Point 3. If the credited diverse means are manual actuations, 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. A description of the operator action sequence, starting with recognition of the increasing RCS inventory event and ending with isolation of CVCS,
b. The time within which the operator can accomplish the required actions to isolate CVCS and terminate the event, 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.

For the Increase in RCS Inventory event with SWCCF, as described in Section A.3.6.2 of ANP-10304 Rev 1, the transient does not terminate on Pressurizer High Level as is the case with the FSAR Chapter 15 analysis. In the FSAR analysis, the pressurizer high level causes a reactor trip and CVCS isolation. For the D3 analysis, neither the reactor trip nor the CVCS isolation occurs, and the pressurizer fills solid. Although Section A.3.6.2 of ANP-10304 Rev 1 states that pressurizer PSRVs are capable of relieving water, thus ensuring that the RCS pressure boundary is maintained, sufficient indication and procedures should be available to ensure that the plant operations personnel recognize and terminate the event in a timely manner. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-35:

The chemical and volume control system (CVCS) malfunction that increases reactor coolant system (RCS) inventory event results from a spurious actuation, either by a control system or operator action, of the CVCS that adds fluid to the RCS without letdown. At steady-state full power operation, one charging pump of the CVCS is operating, and the CVCS maintains the pressurizer level through control of the letdown valve position. The CVCS malfunction isolates letdown and starts the second charging pump. Two charging pumps inject fluid to the RCS from the volume control tank (VCT) without letdown.

In the U.S. EPR FSAR Tier 2, Chapter 15 analysis of this event, the pressurizer high level causes a reactor trip (RT) and CVCS isolation. These features are components of the

protection system (PS). For the diversity and defense-in-depth (D3) analysis, with a software common cause failure (SWCCF) in the PS, the RT and the CVCS isolation does not occur. Technical Report ANP-10304, Revision 1, Section A.3.6.2 conservatively estimates that it takes 24 minutes for the pressurizer to fill solid if the CVCS malfunction continues unabated. Twenty-four minutes is sufficient time for the operator to recognize and terminate the charging flow before the pressurizer overfills. Under this scenario, there are indications/opportunities that alert the operator that the pressurizer is filling uncontrollably. The operator sees a deviation alarm when the pressurizer level exceeds the normal pressurizer level control band. The VCT tank level decreases, resulting in the automatic start of the boron and reactor water makeup pumps.

This assessment is conservative because systems that would be available for event mitigation are not credited. In the U.S. EPR design, a pressurizer level limitation function exists in the process automation system (PAS), which is separate from the PS. This function improves plant availability by avoiding a RT and other safety-related function actuations for events that lead to increasing levels in the pressurizer. Figure 07.08-35-1 shows that the limitation function isolates charging and pressurizer spray when the pressurizer level reaches 70 percent. The pressurizer level limitation function will terminate this event before filling the pressurizer. This automatic feature will actuate approximately 8.5 minutes after event initiation.

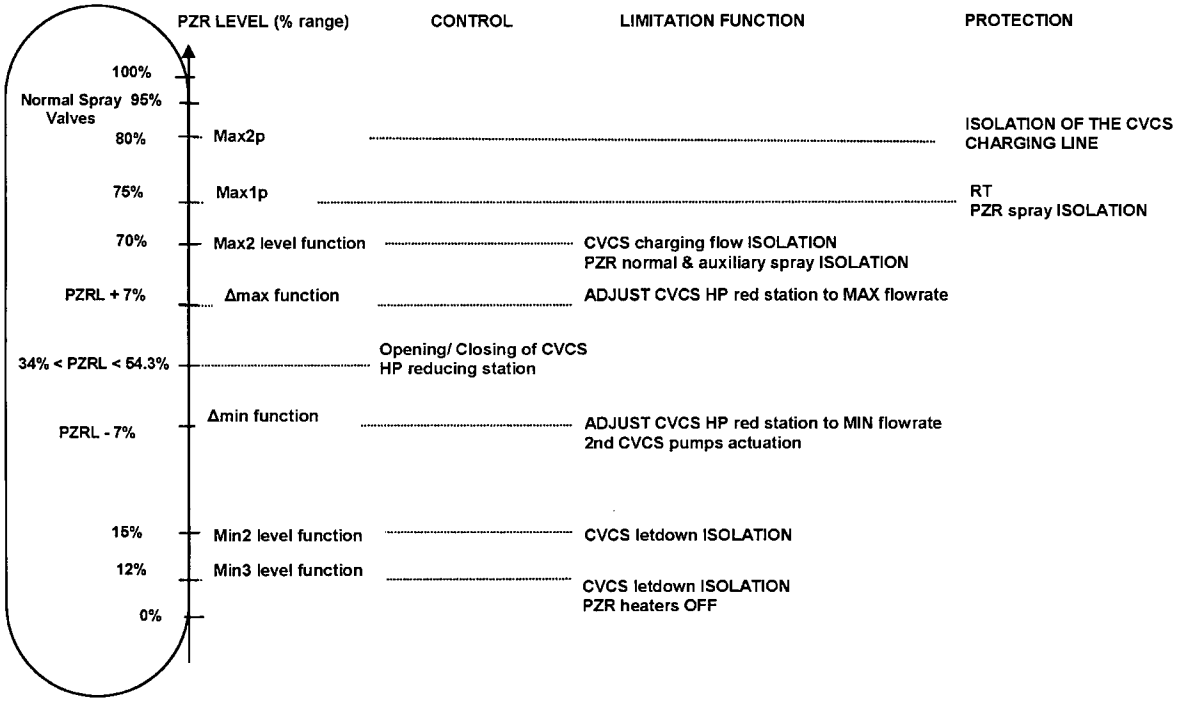
With credit for the pressurizer level limitation system, there is no need for operator action for this event. Charging flow would be automatically isolated when pressurizer level reaches 70 percent. Specific procedures are not required for this event.

Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment," will be revised to include the information requested in this RAI. The revision to Technical Report ANP-10304 will be transmitted separately.

FSAR Impact:

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

Figure 07.08-35-1—Pressurizer Level Setpoints



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 Question 07.08-38:

The temperature decalibration factors are determined by an independent adjoint DORT (Discrete Ordinates Transport) calculation. This adjoint calculation uses the Oak Ridge National Laboratory (ORNL) programs GIP and 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 25°F temperature variation around the nominal inlet coolant temperature. Because of the uncertainty on the exact location of the excore detector, the factor is calculated for three different locations of the excore detector. The temperature decalibration factor, DF, for the locations is calculated to be 0.51 percent/°F.

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

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

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

When the temperature decreases, like during the increased steam flow event (see Technical Report ANP-10304, Revision 1, Figure A.3.2-2), the correction $T(°F) \times DF\left(\frac{\%}{°F}\right)$ is negative.

The indicated reactor power is lower than the current reactor power, and the reactor trip (RT) on high neutron flux is delayed as stated in Technical Report ANP-10304, Section A.3.2.3.

Formula (1) is incorrect. The correct formula for utilization 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 resulting from the change in downcomer temperature depends on the current reactor power. This dependence was omitted from the previous equation.

A AREVA NP corrective action process establishes 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 Technical Report 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) RT setpoint (see Technical Report ANP-10304, Figure A.3.2-1). No other RT is actuated.

According to the numerical simulation in this response, the revised indicated power would be lower because of the negative ΔT, and the conditions would be farther from the RT conditions. The transient analysis would not be impacted by this correction. However, between the end of cycle (EOC) and the beginning of cycle (BOC), reactivity kinetics conditions can lead to the stabilization of the indicated neutron flux signal just under the RT setpoint. A transient calculation was performed with these conditions for the moderator temperature coefficient (MTC) and fuel Doppler coefficient (DTC). Figure 07.08-38-1 shows the indicated power and the nuclear power. For this case, Figure 07.08-38-2 shows the corresponding departure from nucleate boiling ratio (DNBR) and the linear power density (LPD) margin.

If the decalibration coefficient calculated for a 25°F, variation is not applicable to the predicted 34.5°F variation during the increased steam flow event. In an extreme case, the revised indicated power would be higher than previously calculated, and the DAS high neutron flux RT 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 Technical Report ANP-10304, Section A.3.2.3.

Numerical simulation :

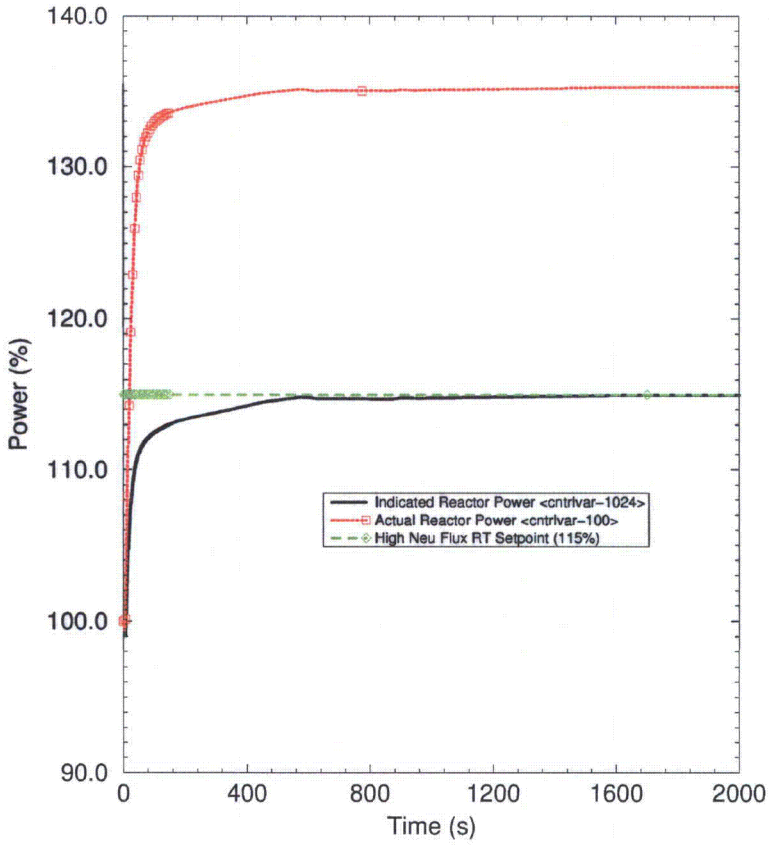
The calculation with the approximate final conditions of the increased steam flow event (Reactor power = 131 percent and ΔT = -34.5°F), shown in Technical Report ANP-10304, Figures A.3.2-1 and A.3.2-2, gives 113.41 percent for the current indicated power and 107.95 percent for the revised indicated power (the RT setpoint is 115 percent).

Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment," will be revised to include the information requested in this RAI. The revision to Technical Report ANP-10304 will be transmitted separately.

FSAR Impact:

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

Figure 07.08-38-1—D3 - Increase in Steam Flow - Case Stabilizing under the High Neutron Flux Setpoint - Power



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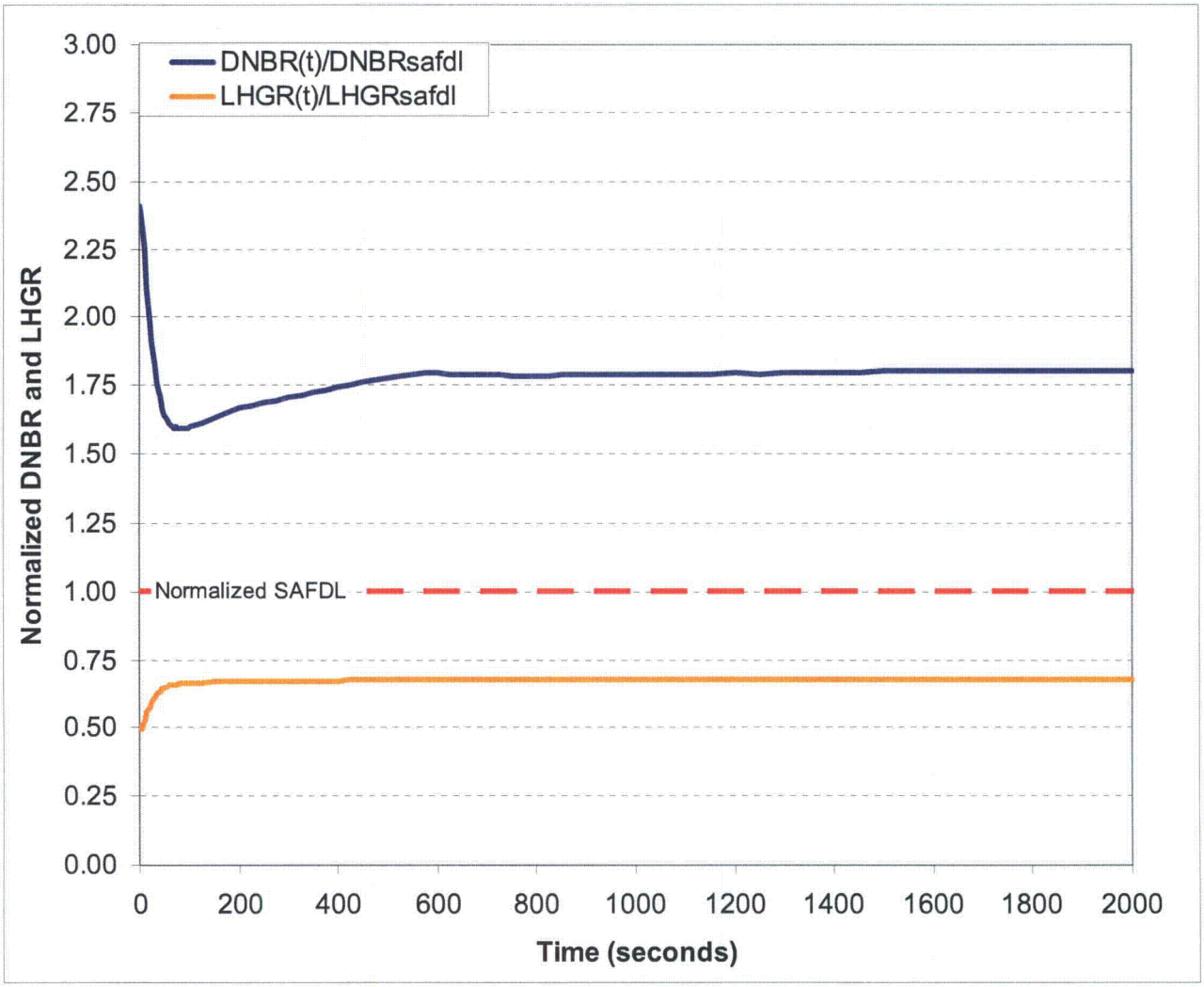
Figure 07.08-38-2—D3 - Increase in Steam Flow - Case Stabilizing under the High Neutron Flux Setpoint - DNBR and Linear Heat Genetation Rate vs. Limits