Canova, Michael

From: Canova, Michael
Sent: Wednesday, December 22, 2010 9:58 AM
To: ArevaEPRDCPEm Resource
Subject: FW: Non-PROPRIETARY DRAFT Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7, <AE>
Attachments: RAI 413 Supplement 3 Response US EPR DC (Public) - DRAFT.pdf

---

From: RYAN Tom (AREVA) [mailto:Tom.Ryan@areva.com]
Sent: Wednesday, December 22, 2010 9:56 AM
To: Carneal, Jason
Cc: PANNELL George (AREVA); BUDZIK Dennis (AREVA); Canova, Michael; BRYAN Martin (EXTERNAL AREVA); Tesfaye, Getachew; SLOAN Sandra (AREVA); BROWNSON Doug (AREVA)
Subject: RE: Non-PROPRIETARY DRAFT Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7,

Jason – due to this RAI 413 draft being on the agenda for the 1/4/10 Public Meeting, I am sending you the Non-Proprietary (Public) version.

Please let me know if you have any questions,

Thank you,

Tom Ryan
Project Engineer
Regulatory Affairs, New Plants

AREVA NP
An AREVA and Siemens company
7207 IBM Drive - CLT2B
Charlotte, NC 28262
Phone: 704-805-2643, Cell : 704-292-5627
Fax: 434-382-6657

From: BRYAN Martin (External RS/NB)
Sent: Friday, December 17, 2010 4:55 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); PANNELL George (CORP/QP); HALLINGER Pat (EXT); WILLIFORD Dennis (RS/NB); BUDZIK Dennis (EP/PE); Carneal, Jason; Canova, Michael
Subject: PROPRIETARY DRAFT Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7,

Getachew,

To support the final response dates for nine questions in RAI 413, a draft response to RAI 413 questions 07.08-21, 07.08-24, 07.08-27, 07.08-28, 07.08-30, 07.08-31, 07.08-35, 07.08-37, and 07.08-40 is attached.
Because AREVA NP believes some of this material in the response is proprietary, an affidavit is attached to request withhold from public disclosure in accordance with 10 CFR 2.390. Let me know if the staff has questions of if the response can be submitted as final.

Thanks,

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

From: BRYAN Martin (External RS/NB)
Sent: Monday, December 13, 2010 8:40 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); PANNELL George (CORP/QP)
Subject: Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7, Supplement 2

Getachew,

AREVA NP provided a schedule for technically complete and correct responses to the questions in RAI 413 on September 08, 2010. Supplement 1 response to RAI No. 413 was sent on November 19, 2010, to provide a revised schedule.

To provide additional time to interact with the NRC a revised schedule is provided below (bolded dates have changed).

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	**January 28, 2011**
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	**January 28, 2011**
RAI 413 07.08-19	**February 22, 2011**
RAI 413 07.08-20	**January 28, 2011**
RAI 413 07.08-21	January 28, 2011
RAI 413 07.08-22	**January 28, 2011**
RAI 413 07.08-23	**January 28, 2011**
RAI 413 07.08-24	January 28, 2011
RAI 413 07.08-25	**January 28, 2011**
RAI 413 07.08-26	**February 22, 2011**
RAI 413 07.08-27	**January 28, 2011**
RAI 413 07.08-28	**January 28, 2011**
RAI 413 07.08-29	February 22, 2011
RAI 413 07.08-30	January 28, 2011
RAI 413 07.08-31	January 28, 2011
RAI 413 07.08-32	February 22, 2011
RAI 413 07.08-33	**January 28, 2011**
RAI 413 07.08-34	**January 28, 2011**

RAI 413 07.08-35	January 28, 2011
RAI 413 07.08-36	**January 28, 2011**
RAI 413 07.08-37	January 28, 2011
RAI 413 07.08-38	**January 28, 2011**
RAI 413 07.08-39	**January 28, 2011**
RAI 413 07.08-40	January 28, 2011
RAI 413 07.08-41	**January 28, 2011**
RAI 413 07.08-42	March 15, 2011

Sincerely,

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

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

Getachew,

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

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

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

Sincerely,

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

Getachew,

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

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

Question #	Start Page	End Page
RAI 413 07.08-10	2	2
RAI 413 07.08-11	3	3
RAI 413 07.08-12	4	4
RAI 413 07.08-13	5	5
RAI 413 07.08-14	6	6
RAI 413 07.08-15	7	7
RAI 413 07.08-16	8	8
RAI 413 07.08-17	9	9
RAI 413 07.08-18	10	10

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

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

Question #	Response Date
RAI 413 07.08-10	March 15, 2011
RAI 413 07.08-11	March 15, 2011
RAI 413 07.08-12	March 15, 2011
RAI 413 07.08-13	March 15, 2011
RAI 413 07.08-14	March 15, 2011
RAI 413 07.08-15	December 17, 2010
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	December 17, 2010
RAI 413 07.08-19	January 28, 2011
RAI 413 07.08-20	December 17, 2010
RAI 413 07.08-21	January 28, 2011
RAI 413 07.08-22	December 17, 2010
RAI 413 07.08-23	December 17, 2010
RAI 413 07.08-24	January 28, 2011
RAI 413 07.08-25	December 17, 2010
RAI 413 07.08-26	December 17, 2010
RAI 413 07.08-27	December 17, 2010
RAI 413 07.08-28	December 17, 2010
RAI 413 07.08-29	January 28, 2011
RAI 413 07.08-30	January 28, 2011
RAI 413 07.08-31	January 28, 2011

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

Sincerely,

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

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

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

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 07.08-21

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 Tavg 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 07.08-21

Purpose of S-RELAP5 heat transfer model modification

The purpose of the S-RELAP5 heat transfer modification is to provide a more realistic computation of steam generator heat transfer.

Previous S-RELAP5 computations have used a variety of input choices to increase steam generator heat transfer to achieve target secondary side pressure during steady-state initialization. Typical input choices have included an increase in steam generator tube area and the use of a small hydraulic diameter to increase the steam generator heat transfer. It is desired to use more realistic input options to achieve the target secondary side pressure.

Description of the changes

S-RELAP5 includes two new input options that improve steam generator heat transfer. They are:

1. An option for "T_{ave}"; and,

2. An option to use the "Inayatov" heat transfer multiplier.

Both options improve the accuracy of heat transfer computation in steam generators and aid in achieving the target secondary side pressure. While an adjustment in heat transfer is still needed to get the target secondary side pressure, it is done by applying the LIQHTC heat transfer coefficient multiplier available in S-RELAP5 input.

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 actually assigned to the nodal boundary (exit), so, if the flow is from the left-to-right, T_K is assigned to j and T_L is assigned to j+1.

A more accurate temperature for computing wall heat transfer is to use an average of the assigned temperatures for the fluid node facing the heat structure. That change is implemented as an option in S-RELAP5 for best estimate non-LOCA analysis.

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:

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

To include the Inayatov multiplier in the Dittus-Boelter heat transfer correlation the above correlation is modified as follows:

$$h = \max\left[0.023\left(\frac{P_1P_2}{D^2}\right)^{0.5}\mathrm{Re}_f^{0.8}\,\mathrm{Pr}_f^{0.4}, 7.86\right]\frac{k_f}{D_h}$$

The Inayatov multiplier $(P_1P_2/D^2)^{0.5}$ is taken from RELAP5/MOD3.3. P_1 and P_2 are the tube pitches and D is the tube outer tube diameter. The multiplier range is from 1.1 to 1.6. The U.S. EPR steam generator has a uniform triangular tube array, so $P_1 = P_2 = P$. The Inayatov multiplier therefore becomes P/D. S-RELAP5 has an input option for P/D.

The Inayatov multiplier is applied in the same way to two-phase forced convection using the Chen correlation (macro part).

LIQHTC Multiplier

LIQHTC is a liquid heat transfer coefficient multiplier that is part of the S-RELAP5 Code Uncertainty Analysis input. LIQHTC multiplies the single phase heat transfer coefficient and also the nucleate boiling heat transfer coefficient macro term. LIQHTC is selected manually for use in the steady-state initialization to achieve the desired secondary side pressure. LIQHTC continues to be applied during the transient calculation.

<u>Validation approach</u>

The validation computations demonstrate the improved accuracy of the T_{ave} option. S-RELAP5 can produce results consistent with the theoretical solution and independent nodal solutions using T_{exit} and T_{ave}.

All validation computations use a simple steam generator consisting of a single tube and a tube wall. By defining boundary conditions on the tube wall and at the inlet and exit of the channel, it is possible to consider three types of validation computations.

1. Simple Steam Generator

   The steam generator is modeled with a constant shell side temperature to represent T_{sat} and a constant tube side heat transfer coefficient. This steam generator has a theoretical solution from which it is possible to assess the accuracy of the T_{exit} and T_{ave} fluid temperature definitions. The steam generator also has nodal (finite difference) solutions for the T_{exit} and T_{ave} fluid temperature definitions that have a direct relationship to the S-RELAP5 solutions. This simple modeling is also used to define input files for S-RELAP5 for a variety of situations including the T_{exit} and T_{ave} options and with forward and reverse flow. The fluid temperatures computed by S-RELAP5 are compared to the theoretical and nodal solutions in Excel.

2. Single-Phase Forced Convection

   The steam generator is modeled as in Item 1, except that the single-phase heat transfer coefficient is computed by using the S-RELAP5 heat transfer correlations. The shell (right) side wall temperature is constant and the tube (left) side water temperature decreases by heat transfer across the tube wall. The heat transfer coefficient computation also includes the Inayatov multiplier selected by input option. The single-phase heat transfer coefficient computed by S-RELAP5 is compared to its independent computation in Excel.

3. Two-Phase Forced Convection

   The steam generator is modeled as in Item 1, except roles of the shell and tube are switched. The shell (right) side boils and the two-phase heat transfer coefficient is computed on the shell side by using the S-RELAP5 heat transfer correlation for two-phase forced convection. The heat transfer coefficient computations also include the Inayatov multiplier. The tube (left) side wall temperature is a constant and greater than T_{sat} on the shell (right) side, thus, the shell-side water boils. The two-phase heat transfer coefficients computed by S-RELAP5 are compared to its independent computation in Excel.

Mathematical Solutions for a Simple Steam Generator

First consider the theoretical solution for a simple steam generator. Then consider the nodal solutions using the T_{exit} and T_{ave} definitions.

<u>Theoretical Solution</u>

The theoretical solution for heat transfer in a simple steam generator is based on the following:

- Hot water, at a constant flow rate, drives the heat transfer process across a tube wall where the outer surface is constant at T_{sat}. There is no subcooled region on the T_{sat} side of the wall.
- The circular tube wall is arbitrarily thin to eliminate the thermal resistance of the wall.
- The tube-side heat transfer coefficient is constant.

The heat transfer process for this simple steam generator has a theoretical solution that can be defined from first principals. The solution is expressed as:

$$(T - T_{sat}) = (T_{in} - T_{sat})e^{-ax}$$

where

$$a = \frac{hP_h}{mC}$$

<u>Nodal Solution Using T_{exit}</u>

A finite difference solution for the liquid temperature in the same simple steam generator, using T_{exit}, produces a recursive relationship that can be expressed as:

$$T_j - T_{sat} = \frac{1}{(1 + a\Delta x)}(T_{j-1} - T_{sat})$$

Given the inlet temperature, the nodal temperature T_j can be advanced along the channel.

<u>Nodal Solution Using T_{ave}</u>

A finite difference solution for the liquid temperature in the same simple steam generator, using T_{ave}, produces a recursive relationship that can be expressed as:

$$T_j - T_{sat} = \frac{\left(1 - \frac{a\Delta x}{2}\right)}{\left(1 + \frac{a\Delta x}{2}\right)}(T_{j-1} - T_{sat})$$

Given the inlet temperature, the nodal temperature T_j can be advanced along the channel.

S-RELAP5 Model of Simple Steam Generator

S-RELAP5 is used to model the simple steam generator as described above. It is directly applicable to the theoretical and nodal solutions considered in Excel. The modeling consists of a flow channel (pipe component) with volumes placed at the each end for application of pressure and temperature boundary conditions. Those volumes are connected to the flow channel by junctions at each end. The inlet junction defines the flow boundary condition.

Heat is transferred between the fluid channel and the tube wall heat structures. The wall is cylindrical, thin and has high thermal conductivity to eliminate its resistance to heat transfer. The heat transfer coefficient is held constant. The outer surface of the wall is also held at a constant temperature. Figure 07.08-21-1 shows the nodal arrangement.

Validation summary results

The following plots present representative results from the three types of validation computations.

Simple Heat Exchanger

Figure 07.08-21-3 shows the theoretical solution and the T_{exit} solutions from S-RELAP5 and the corresponding T_{exit} nodal solution from Excel. The solutions using T_{exit} are in agreement, but, they are both above the temperatures from the theoretical solution.

Figure 07.08-21-4 shows the results of the same computation but using T_{ave} in S-RELAP5 and in the corresponding nodal solution from Excel. The overlay of the three solutions shows the improvement when using T_{ave}. The plots also show the excellent agreement among all three computations. The same excellent agreement is found for reverse flow and when switching the tube and shell sides.

The agreement between the independent computations done in Excel for the temperature in a simple steam generator and the results of the S-RELAP5 computations validate the implementation of the T_{ave} option in S-RELAP5.

Single-Phase Forced Convection Validation

This case considers the following:

- The single-phase forced convection heat transfer on the tube (left) side is computed by using the S-RELAP5 heat transfer correlations presented above.
- The shell (right) side tube wall temperature is a constant.
- An Inayatov multiplier of P/D = 1.5 is selected for this validation computation.

The shell (right) side wall temperature is a constant. The heat transfer across the tube wall from the tube side to the shell side decreases the temperature of the liquid on the tube (left) side.

Excel performs an independent calculation of the heat transfer coefficient including an Inayatov multiplier of 1.5. The heat transfer coefficient is uniform because of uniform flow rate and fluid properties.

Figure 07.08-21-5 shows the heat transfer coefficient from S-RELAP5 and from Excel. The heat transfer from S-RELAP5 is essentially a constant along the length as expected. The Excel value is very close to the result computed in S-RELAP5.

The independent computations in Excel validate the implementation of the Inayatov multiplier applied to the single-phase heat transfer coefficient computed in S-RELAP5.

Two-Phase Forced Convection Validation

This case considers the following:

- S-RELAP5 computes the two-phase forced convection heat transfer coefficients on the shell (right) side.

- Set shell (right) side temperature to T_{sat}.
- Set tube (left) side temperature greater than T_{sat} to transfer heat to the shell side.
- Set the Inayatov multiplier to 1.5 on the shell side. The value of 1.5 is arbitrary.

The two-phase heat transfer coefficient is defined by the Chen correlation. The Chen correlation for boiling heat transfer consists of "macro" and "micro" heat transfer coefficients terms as follows:

$$q'' = h_{mac}\left(T_w - T_f\right) + h_{mic}\left(T_w - T_{sat}\right)$$

The forced convection or "macro" part of the Chen correlation is based on the Dittus-Boelter correlation, and, with the Inayatov multiplier, it is written as:

$$h_{mac} = 0.023\left(\frac{P_1 P_2}{D^2}\right)^{0.5} \mathrm{Re}_f^{0.8}\,\mathrm{Pr}_f^{0.4}\,\frac{k_f}{D_h}F$$

The F-factor and the "micro" part of the Chen correlation are not modified by the Inayatov multiplier. The F-factor is defined as:

$$F = \begin{cases} 1.0 & \chi_{tt}^{-1} \le 0.1 \\ 2.35\left(\chi_{tt}^{-1} + 0.213\right)^{0.736} & \chi_{tt}^{-1} > 0.1 \end{cases}$$

For homogeneous two-phase flow the Martinelli parameter is defined as:

$$\chi_{tt}^{-1} = \left(\frac{X}{1-X}\right)^{0.9}\left(\frac{\rho_f}{\rho_g}\right)^{0.5}\left(\frac{\mu_g}{\mu_f}\right)^{0.1}$$

Note that the inverse of the Martinelli parameter is just notation and that an inverse is not actually done. The parameter is defined in terms of steam quality and the fluid properties as shown.

The two-phase forced convection heat transfer coefficient is independently computed in Excel. The Excel computation mimics the S-RELAP5 computation by computing the two-phase mixture enthalpy rise and steam quality along the heated channel. The components of the Chen correlation are computed as a function of quality. The combined macro and micro heat transfer coefficients are reported along with the S-RELAP5 coefficients in Figure 07.08-21-6. The heat transfer coefficient increases with distance from the inlet as the vapor quality increases. This agreement is very good given the complexities of the two-phase heat transfer computation and the changing fluid properties with pressure along the flow path in the S-RELAP5 computation. The Excel computation uses uniform fluid properties representative of those from taken from the corresponding S-RELAP5 case.

The independent computations in Excel validate the implementation of the Inayatov multiplier in the two-phase heat transfer coefficient computed in S-RELAP5.

Nomenclature

C	Specific heat capacity, BTU/(lbm-°F)
D	Diameter, ft
D_h	Hydraulic diameter, ft
F	Chen correlation F-Factor
h	Heat transfer coefficient, BTU/(sec-ft^2-°F)
k	Thermal conductivity, BTU/(sec-ft-°F)
$\dot{m}$	Mass flow rate, lbm/sec
P	Tube pitch, ft
P_h	Heated perimeter, ft
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

Subscripts

Ave	Average
Exit	Exit
F	fluid, liquid
In	Inlet
J	Nodal index
Mac	Macro
Mic	Micro
Sat	saturation
W	Wall

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

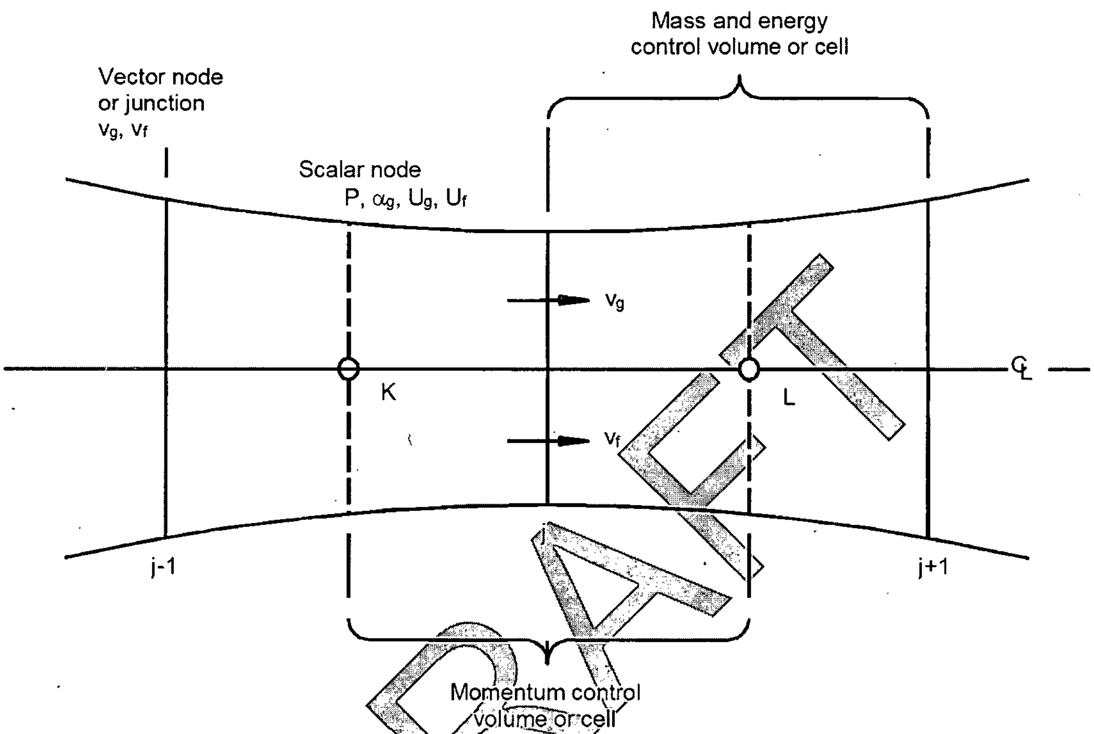


Figure 07.08-21-2—S-RELAP5 Nodal Model for Simple Steam Generator

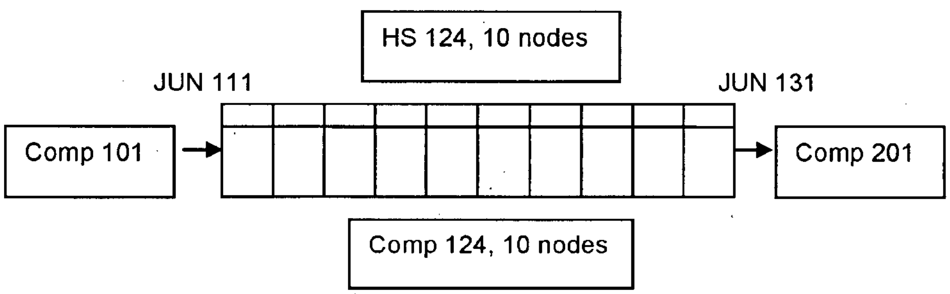


Figure 07.08-21-3 — Steam Generator Solutions using T_{exit}

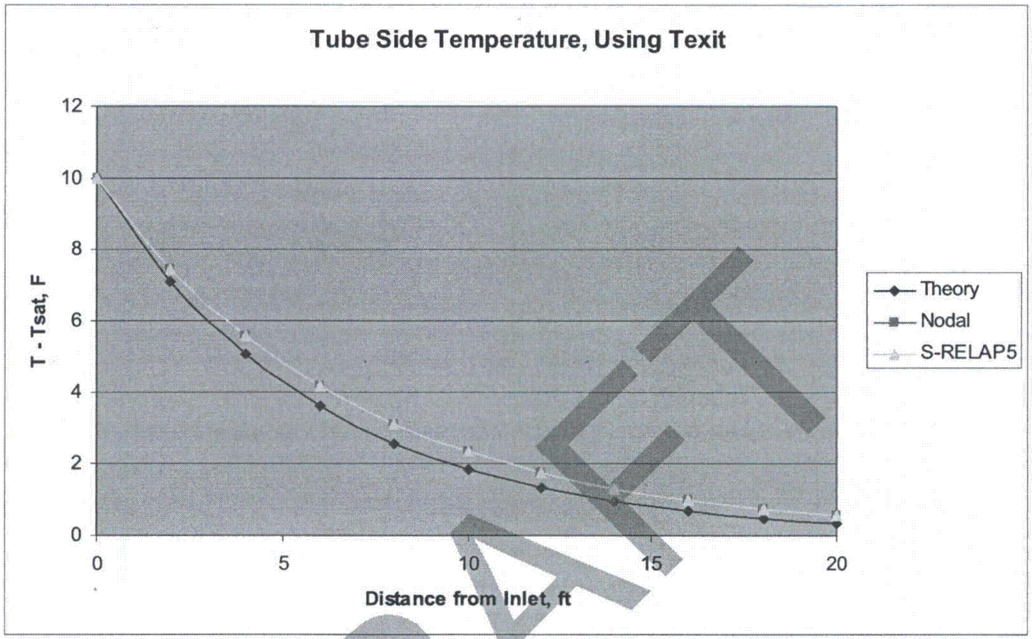


Figure 07.08-21-4 — Steam Generator Solutions using T_{ave}

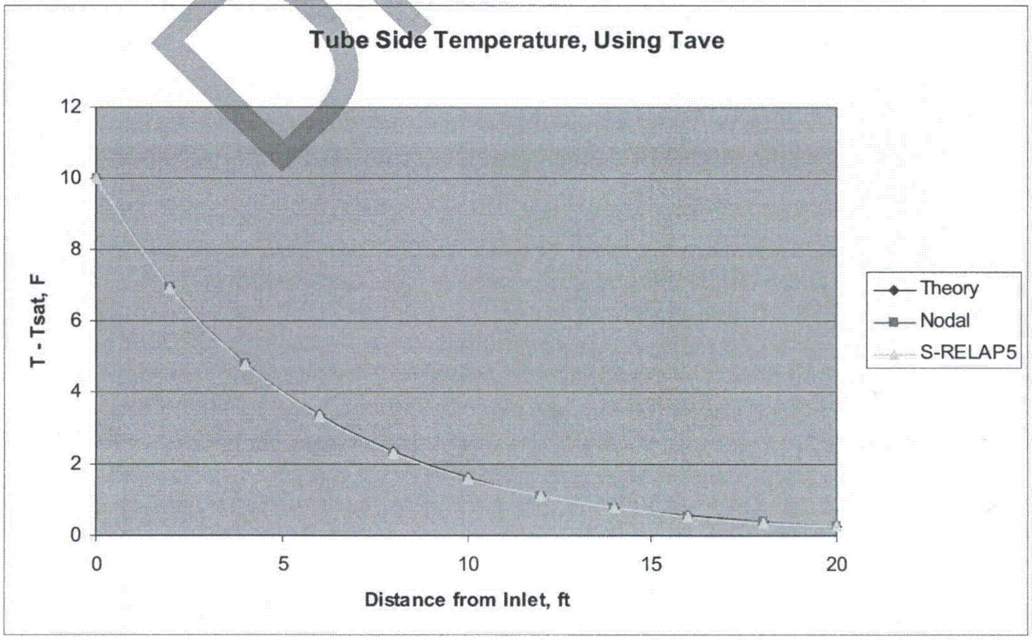


Figure 07.08-21-5 — Single-Phase Heat Transfer Coefficients

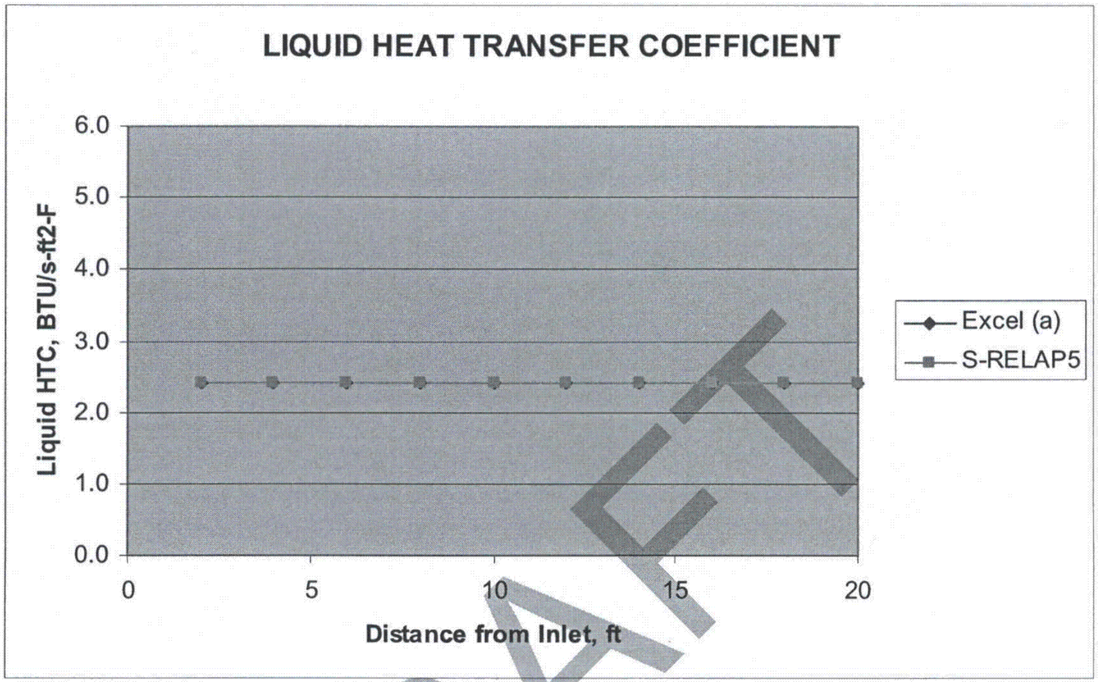
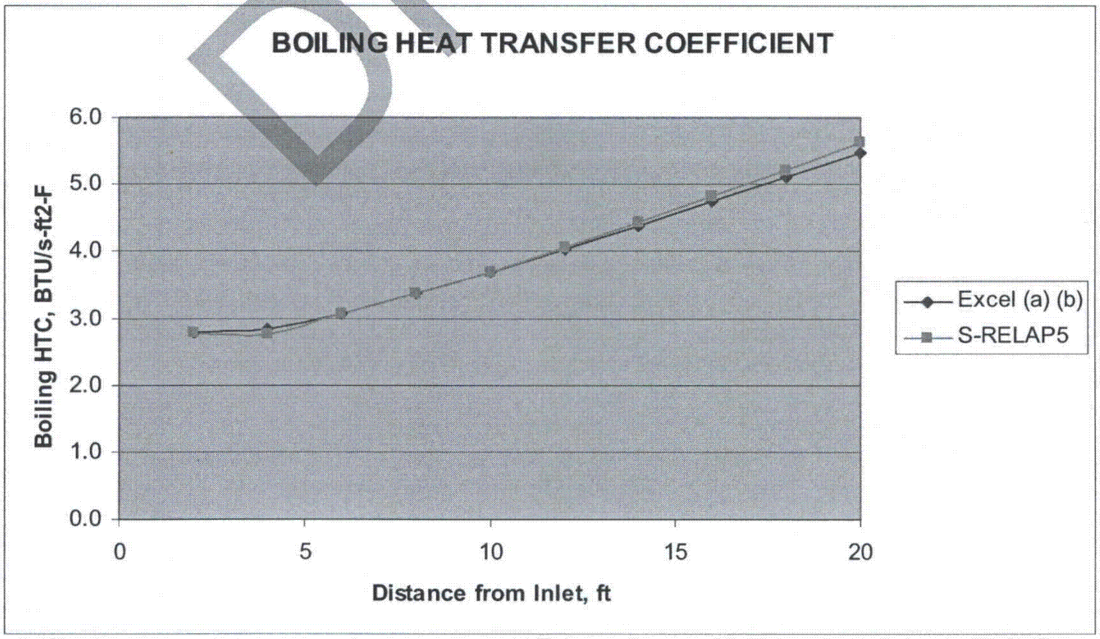


Figure 07.08-21-6 — Two-Phase Heat Transfer Coefficients



FSAR Impact:

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

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):

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

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:

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

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

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

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:

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

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:

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

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

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):

Steam Generator Level Transient Response

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

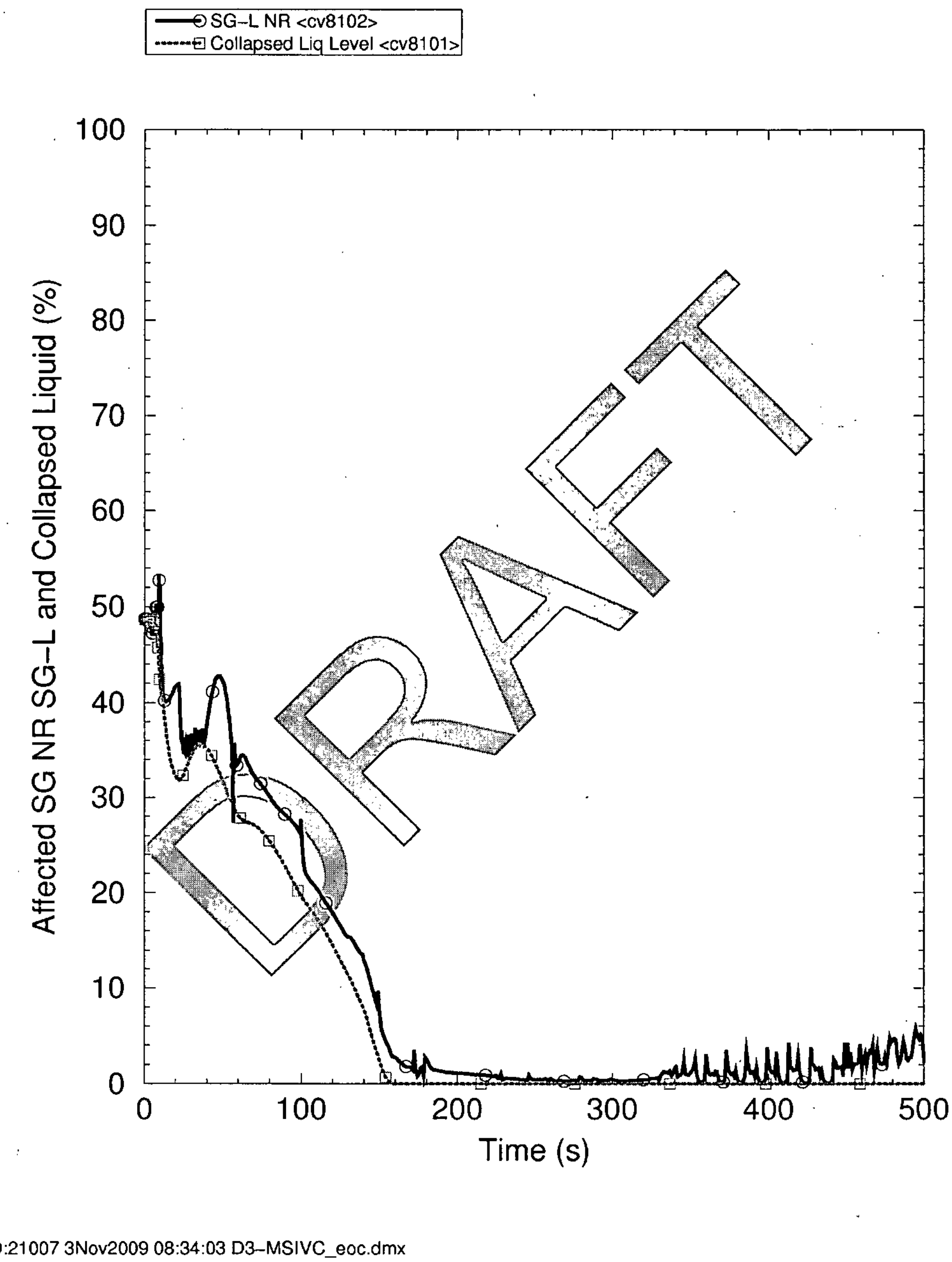


Figure 07.08-24-3 — Reactor Power - D3 MSIV Closure

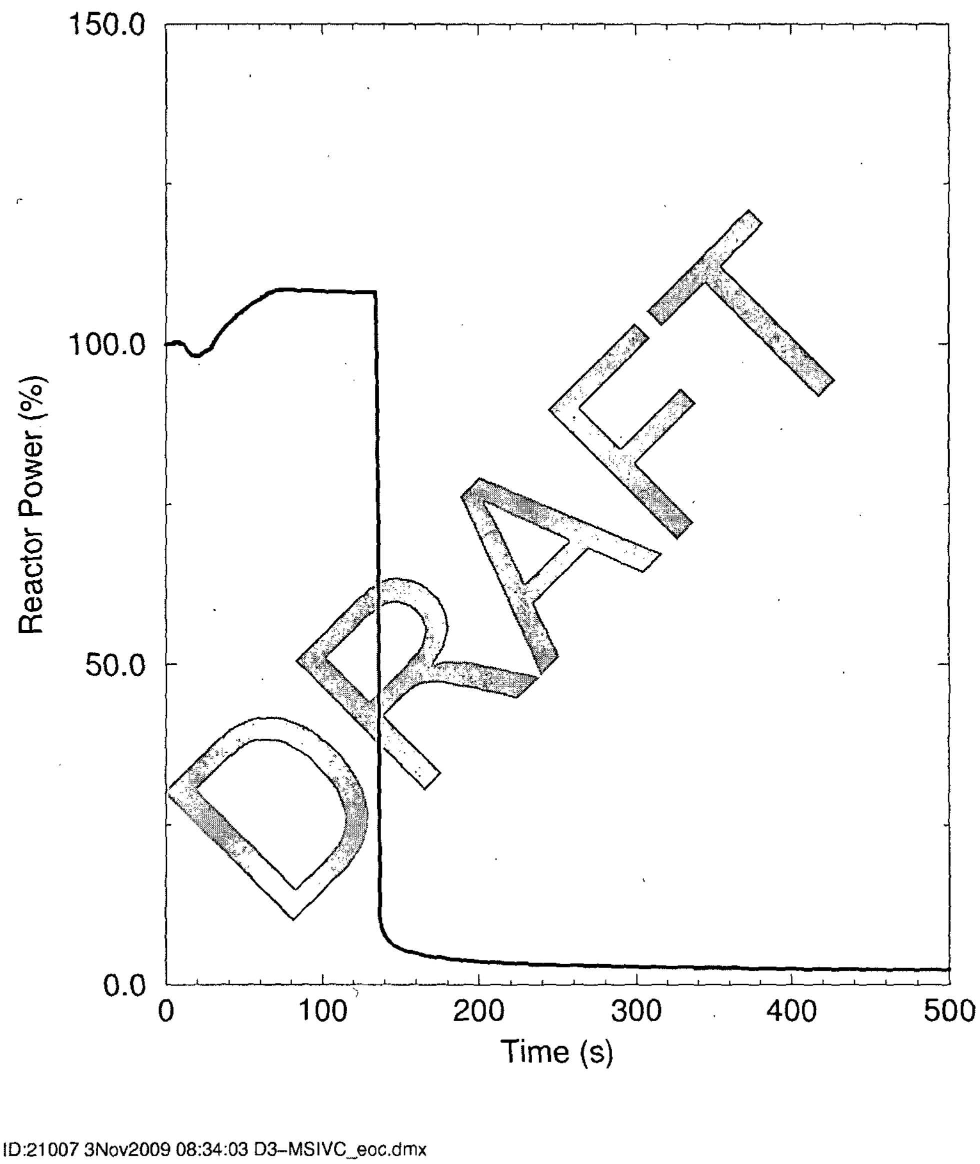


Figure 07.08-24-4 — SG-4 WR Level - D3 MSIV Closure

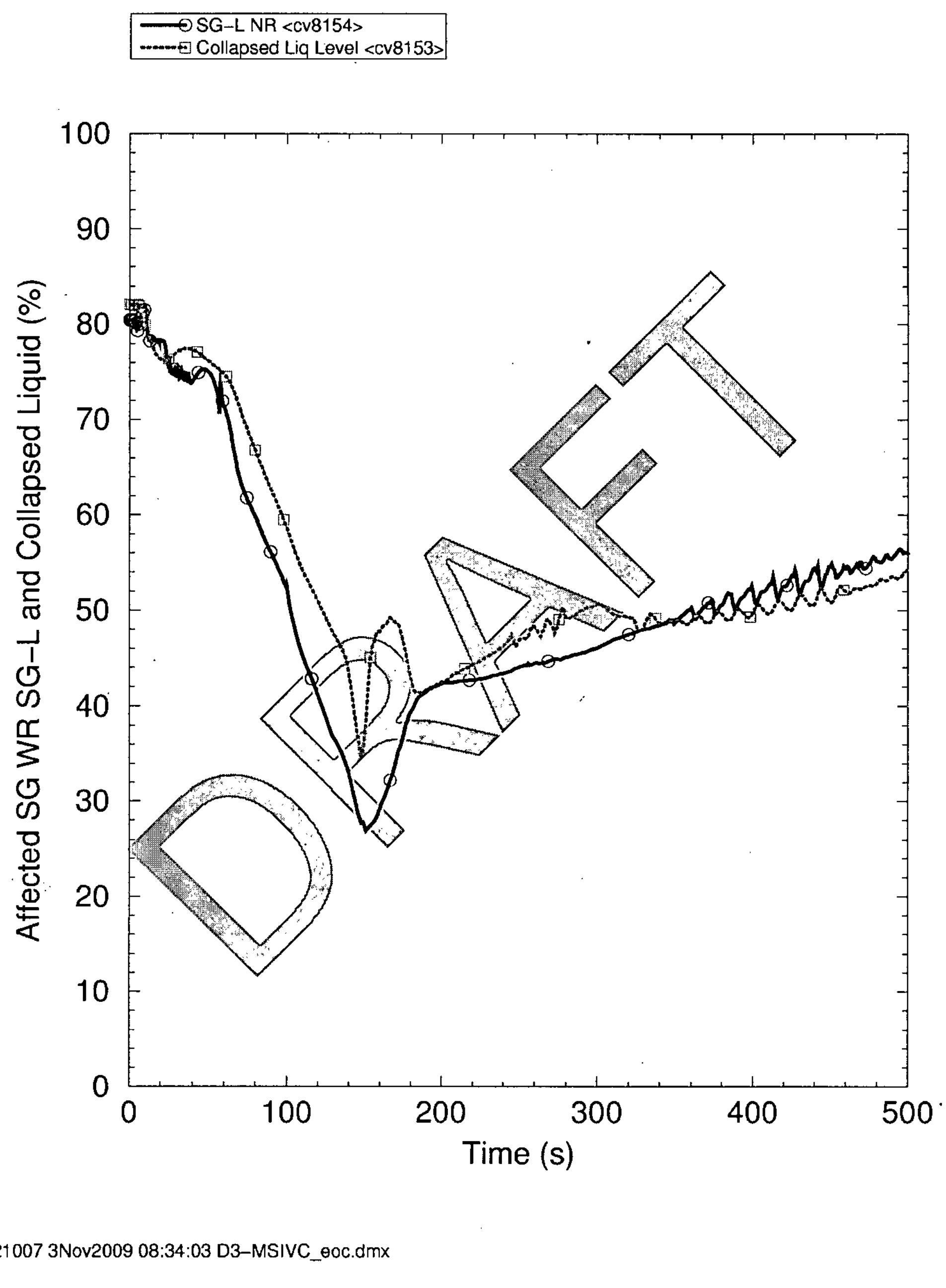
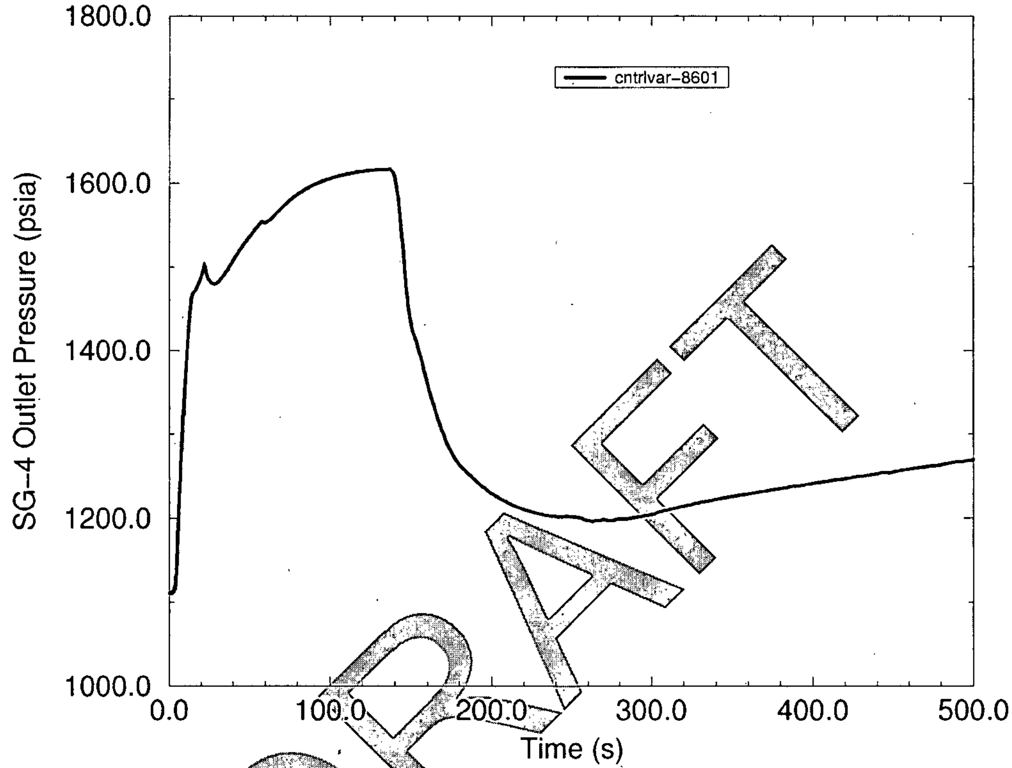


Figure 07.08-24-5 — SG-4 Outlet Pressure - D3 MSIV Closure



FSAR Impact:

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

Question 07.08-27

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 FΔH (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.

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

FSAR Impact:

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

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.

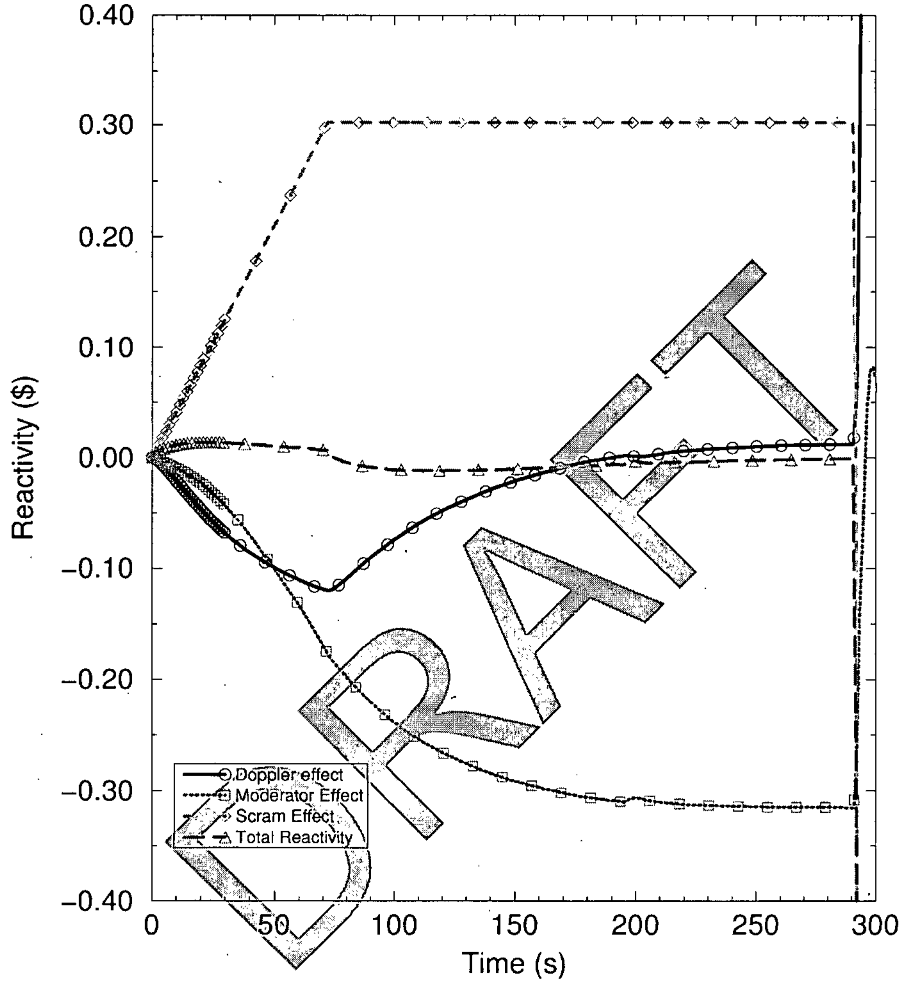
Additional Key Differences Relevant to the Uncontrolled Bank Withdrawal Event

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.

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



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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

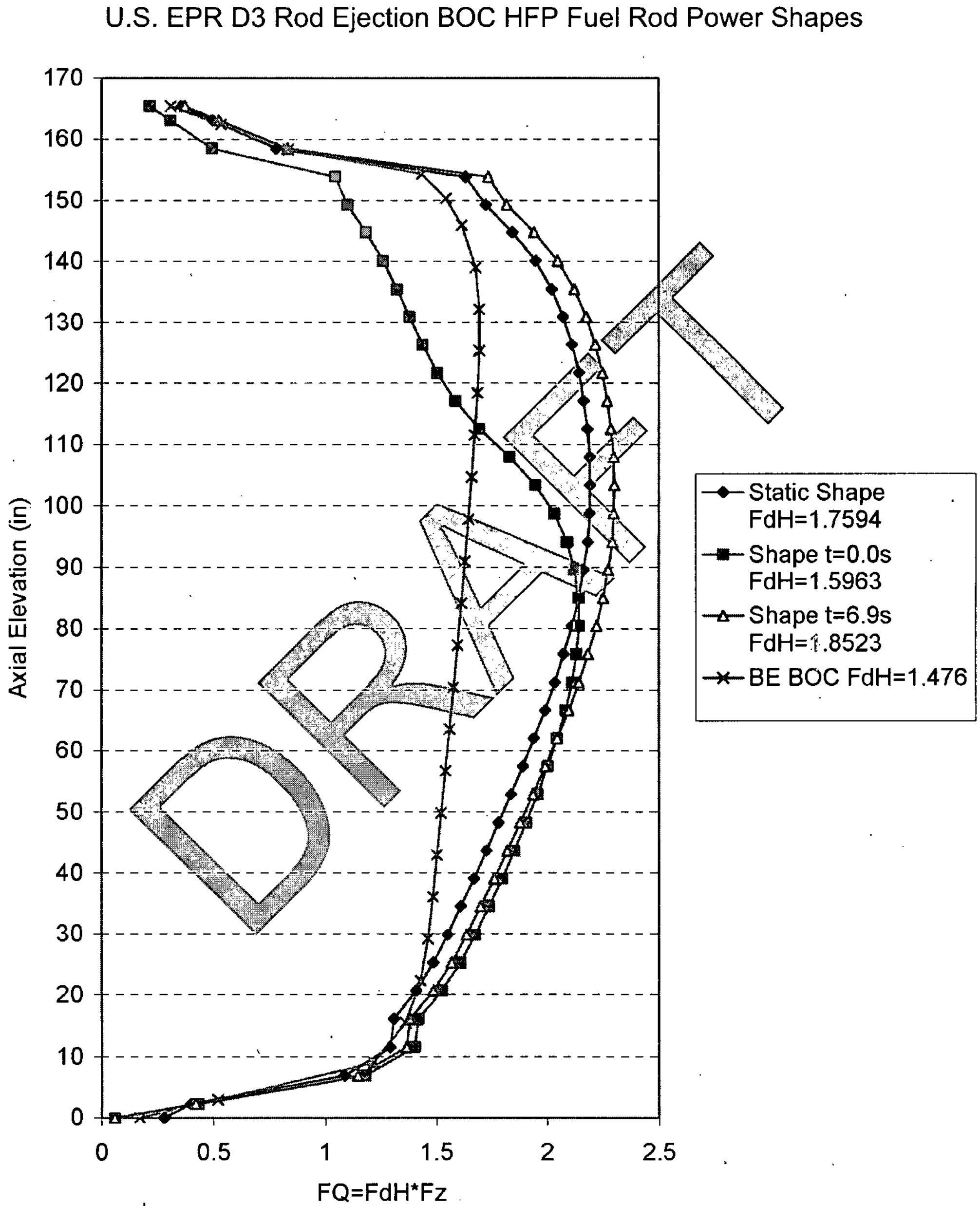


Figure 07.08-31-2 — Fraction of Core Power Response to Rod Ejection from BOC HFP

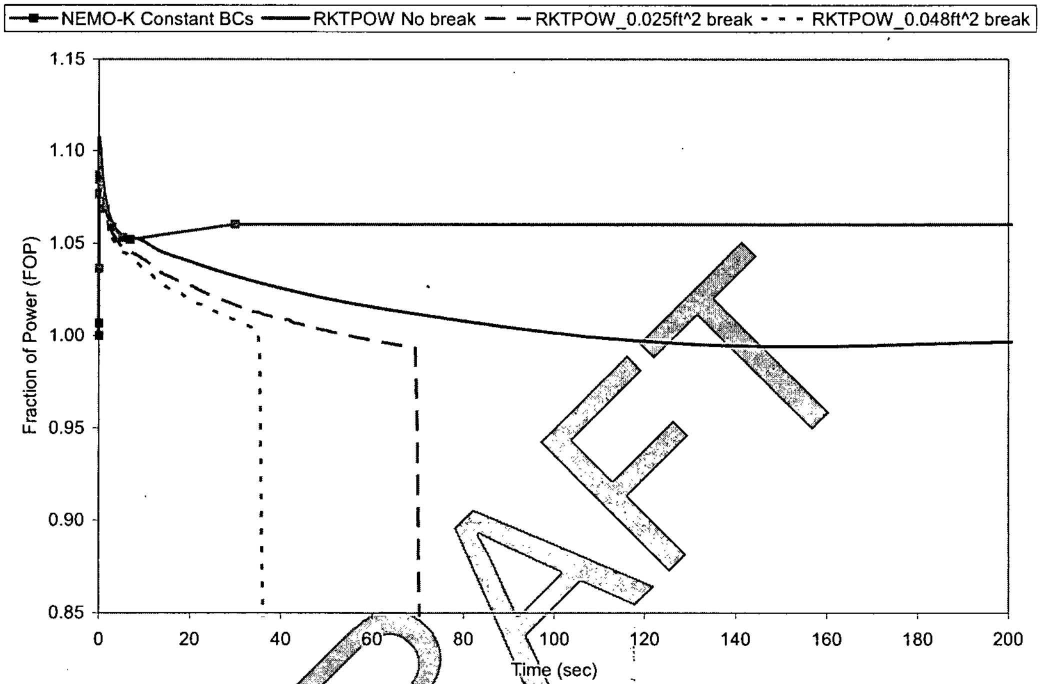


Figure 07.08-31-3— Core Exit Pressure and Inlet Temperature Responses to Rod Ejection from BOC HFP

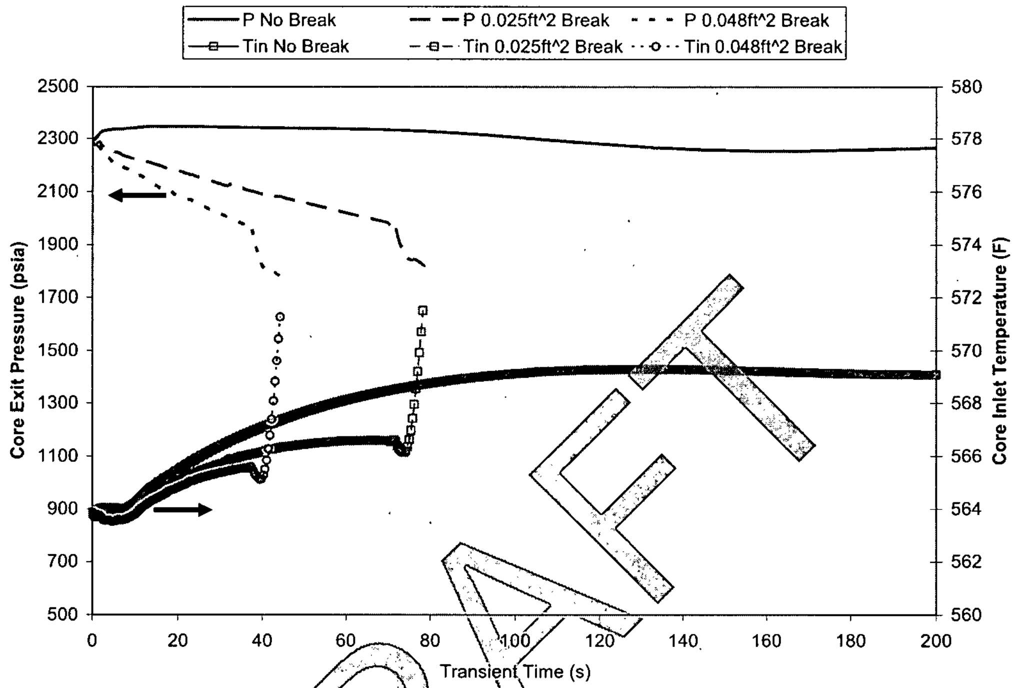


Figure 07.08-31-4 — MDNBR/SAFDL Response to Rod Ejection from BOC HFP

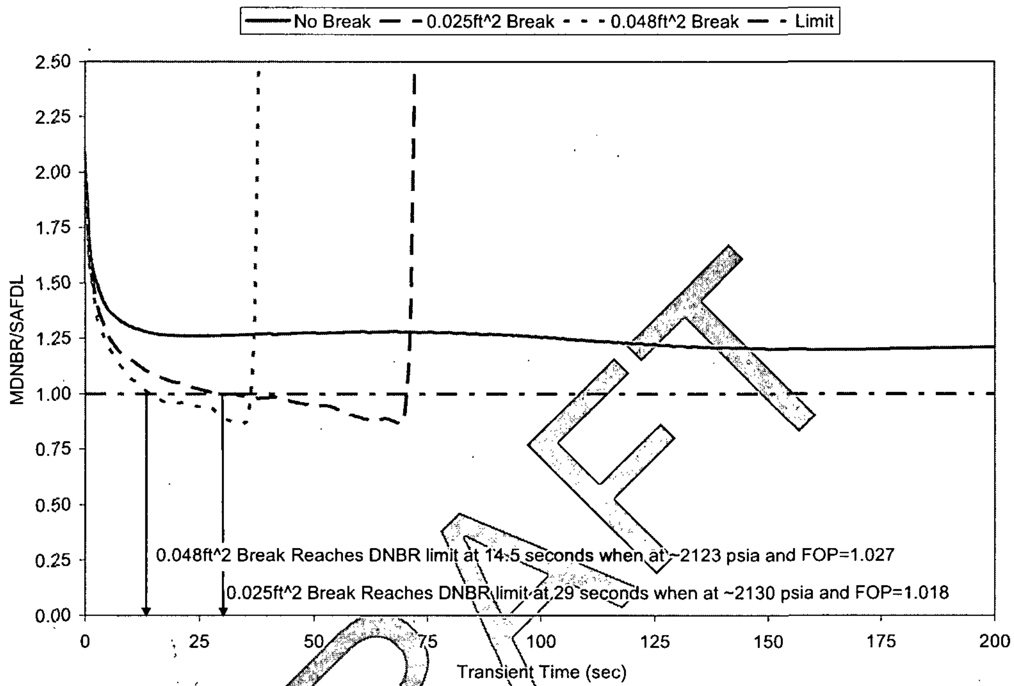


Figure 07.08-31-5 — Peak Fuel and Cladding Temperature Responses to Rod Ejection from BOC HFP

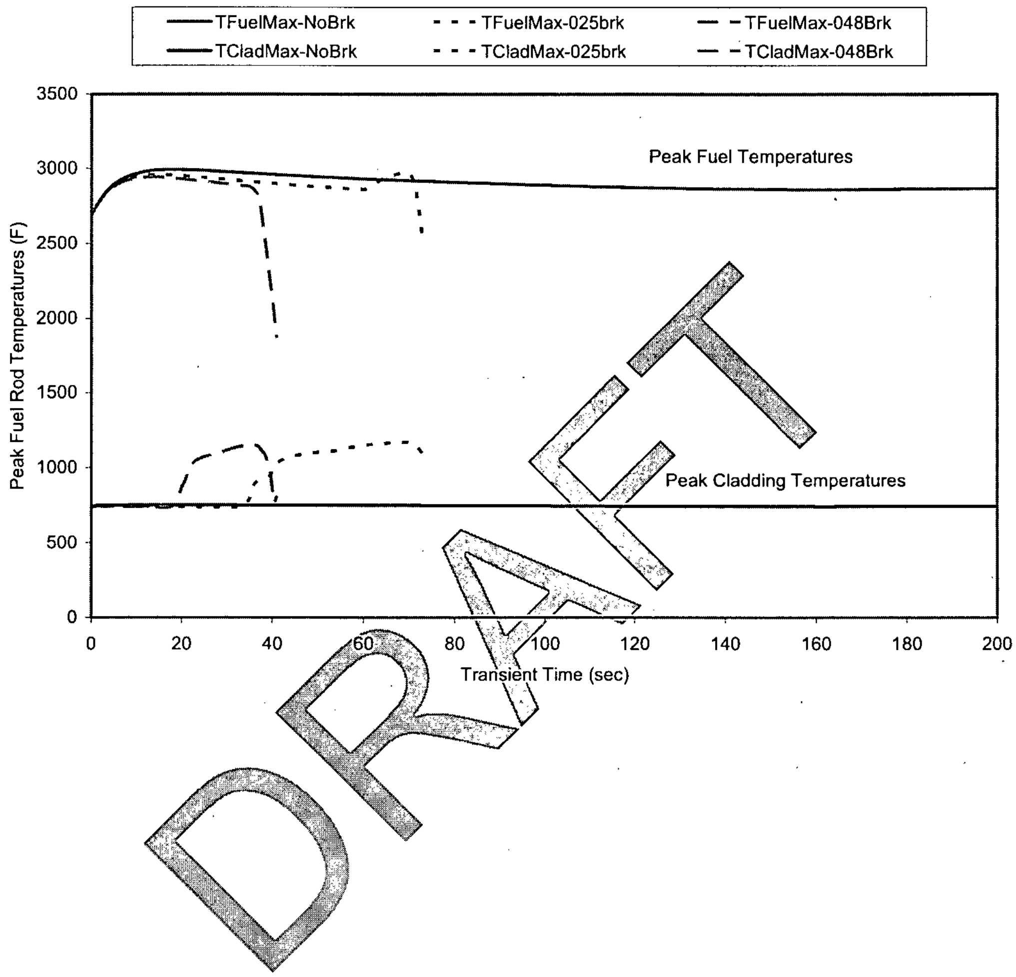
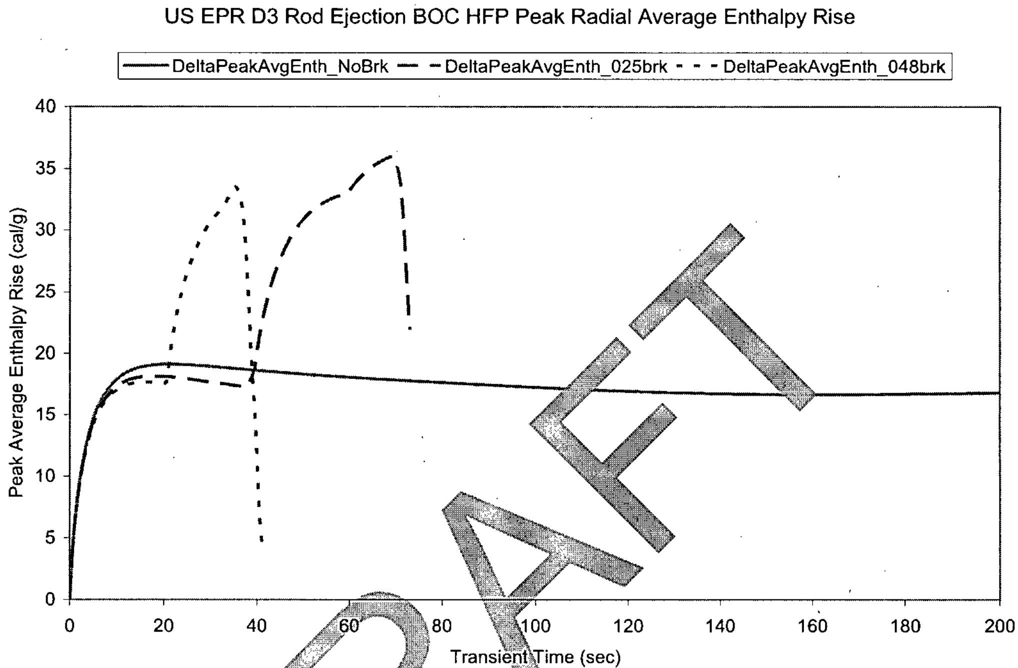


Figure 07.08-31-6 — Peak Average Enthalpy Rise Responses to Rod Ejection from BOC HFP



FSAR Impact:

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

Question 07.08-35

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 07.08-35

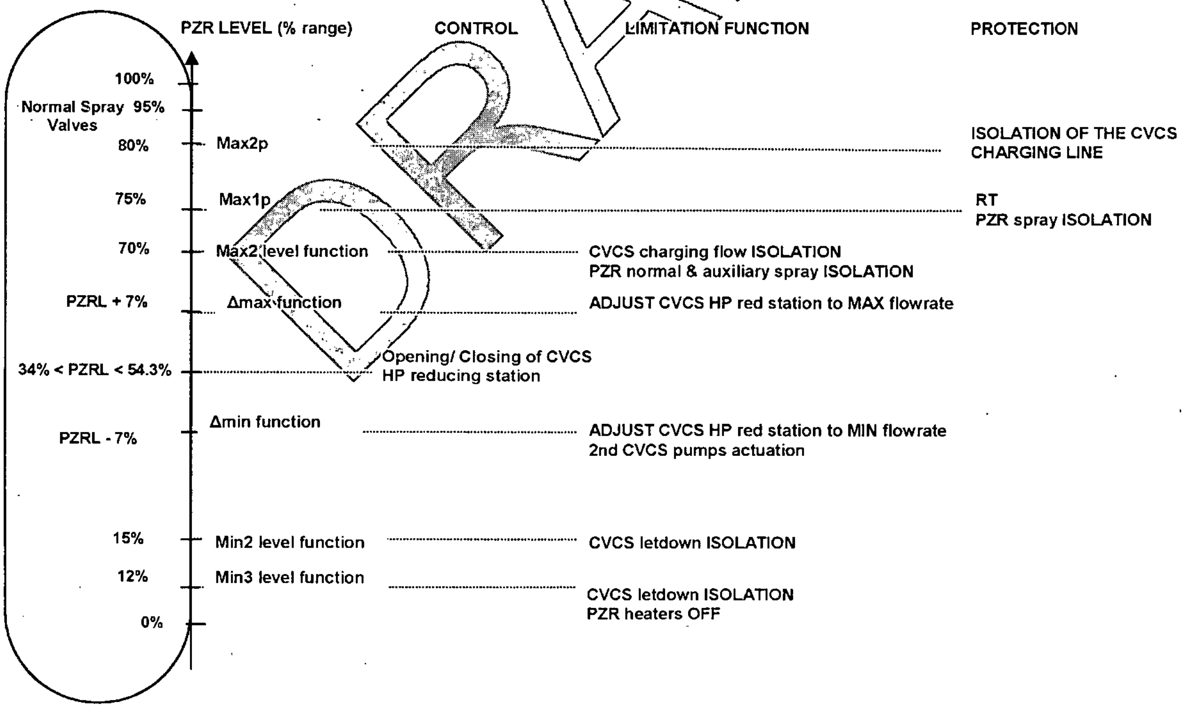
The chemical and volume control system (CVCS) malfunction that increases reactor coolant system (RCS) inventory event results from a spurious actuation, either by a control system or operator action, of the CVCS that adds fluid to the RCS without letdown. At steady-state full power operation, one charging pump of the CVCS is operating and the CVCS control system maintains the pressurizer level through control of the letdown valve position. The CVCS malfunction is assumed to isolate letdown and start 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 and CVCS isolation. Both these features are part of the protection system

(PS). For the diversity and defense-in-depth (D3) analysis, with a software common cause failure (SWCCF) in the PS, neither the reactor trip nor the CVCS isolation occurs. The discussion provided in Section A.3.6.2 of ANP-10304, Revision 1 conservatively estimates that it will take 24 minutes for the pressurizer to fill solid if the CVCS malfunction continues unabated. 24 minutes is sufficient time for the operator to recognize and terminate the charging flow before the pressurizer overfills. Under this scenario there are several indications/opportunities that would alert the operator that the pressurizer is filling uncontrollably. The operator will first see a deviation alarm when pressurizer level exceeds the normal pressurizer level control band. The VCT tank level will decrease 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, there exists a pressurizer level limitation function, separate from the PS process automation system. This function is intended to improve plant availability by avoiding reactor trip and other safety function actuations for events that lead to increasing level in the pressurizer. As illustrated in Figure 07.08-35-1, the limitation function will isolate charging and pressurizer spray when the pressurizer level reaches 70 percent. Therefore, the pressurizer level limitation function will terminate this event well before filling the pressurizer. This automatic feature will actuate approximately 8.5 minutes after event initiation.

Figure 07.08-35-1 — Pressurizer Level Setpoints



FSAR Impact:

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

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 07.08-40-2 — 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

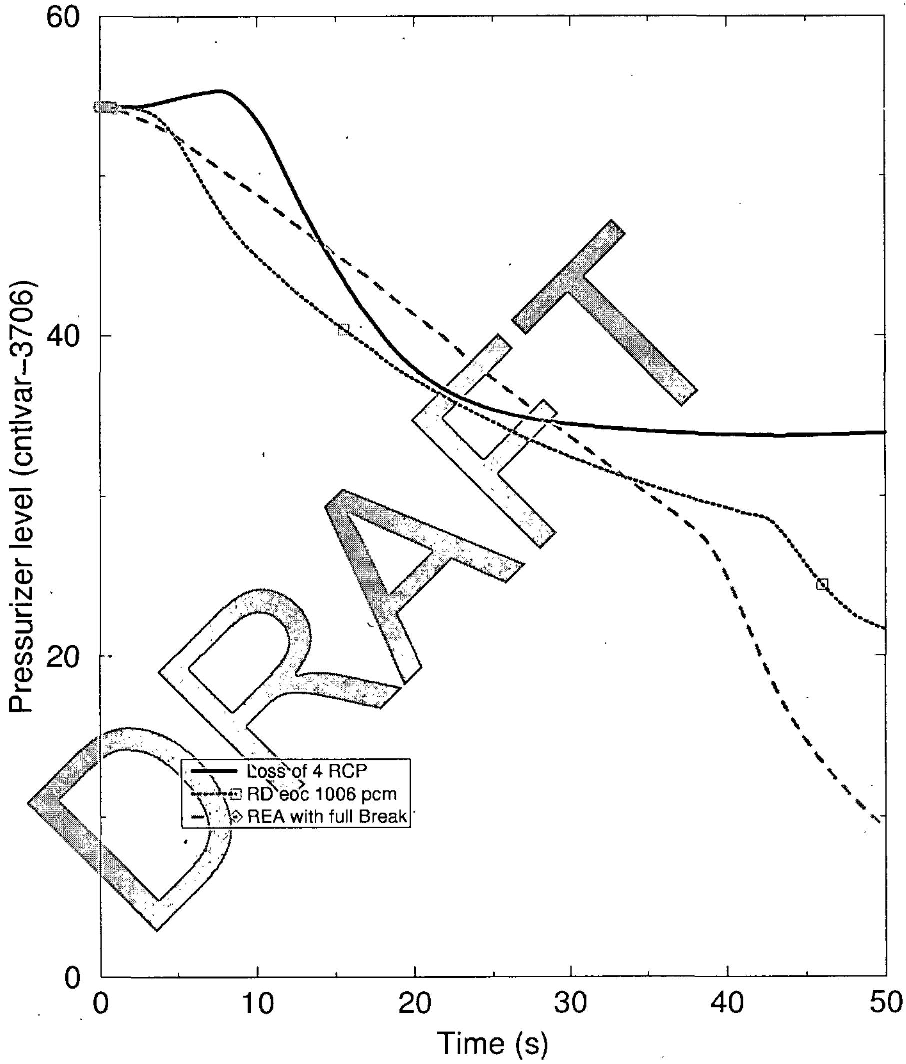


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Figure 07.08-40-6 — Pressurizer Pressure Response for Various Events

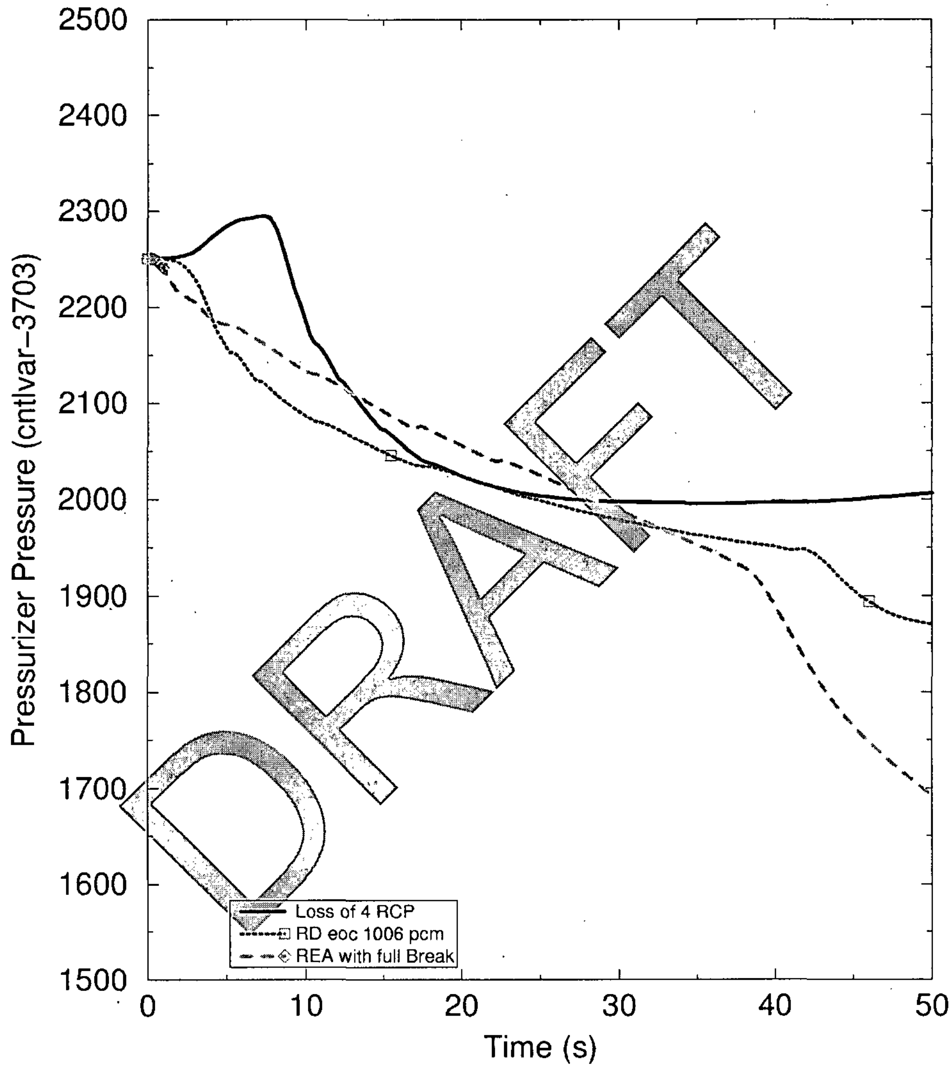

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Figure 07.08-40-7 — Mode 1 RCS Pressure Control

FSAR Impact:

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