ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Friday, December 18, 2009 11:41 AM
To:	Tesfaye, Getachew
Cc:	BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); BEELMAN Ronald J (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 221, FSAR Ch 6, Supplement 4
Attachments:	RAI 221 Supplement 4 Response US EPR DC_PUBLIC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 3 of the 32 questions of RAI No. 221 on June 17, 2009. Supplement 1 response to RAI No. 221 was sent on July 31, 2009 to address 4 of the remaining 29 questions. Supplement 2 response to RAI No. 221 was sent on August 27, 2009 to address 8 of the remaining 25 questions. Supplement 3 response to RAI No. 221 was sent on September 30, 2009 to address 3 of the remaining 17 questions. The attached file, "RAI 221 Supplement 4 Response US EPR DC_PUBLIC.pdf," provides technically correct and complete responses to 10 of the 14 remaining questions.

Since the response contains security-related sensitive information that should be withheld from public disclosure in accordance with 10 CFR 2.390, the attached file is a public version with the security-related sensitive information redacted. This email does not contain any security-related information. The unredacted SUNSI version is provided under separate email.

The following table indicates the respective pages in the response document, "RAI 221 Supplement 4 Response US EPR DC_PUBLIC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 221 — 06.02.01-15	2	6
RAI 221 — 06.02.01-21	7	15
RAI 221 — 06.02.01-26	16	20
RAI 221 — 06.02.01-28	21	23
RAI 221 — 06.02.01-30	24	31
RAI 221 — 06.02.01-32	32	32
RAI 221 — 06.02.01-33	33	34
RAI 221 — 06.02.01-34	35	38
RAI 221 — 06.02.01-42	39	40
RAI 221 — 06.02.01-44	41	41

A response to four questions cannot be provided at this time. The schedule for technically correct and complete responses to the remaining RAI No. 221 questions has been changed and is provided below:

Question #	Response Date
RAI 221 — 06.02.01-22	May 28, 2010
RAI 221 — 06.02.01-23	May 28, 2010
RAI 221 — 06.02.01-24	May 28, 2010
RAI 221 — 06.02.01-35	February 25, 2010

Since three of the remaining questions are regarding the safety-related doors and/or foils and dampers for which the responses depend on performance of the subcompartment analysis (OPEN ITEM), AREVA NP requests a telecon with NRC staff in January to gain clarity regarding the scope of the needed response with the goal of improving the response dates provided above.

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, September 30, 2009 4:19 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); BEELMAN Ronald J (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSAR Ch 6, Supplement 3

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 3 of the 32 questions of RAI No. 221 on June 17, 2009. Supplement 1 response to RAI No. 221 was sent on July 31, 2009 to address 4 of the remaining 29 questions. Supplement 2 response to RAI No. 221 was sent on August 27, 2009 to address 8 of the remaining 25 questions. The attached file, “RAI 221 Supplement 3 Response US EPR DC.pdf,” provides technically correct and complete responses to 3 of the 17 remaining questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 221 Question 06.02.01-18.

The following table indicates the respective pages in the response document, “RAI 221 Supplement 3 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 221 — 06.02.01-16	2	2
RAI 221 — 06.02.01-18	3	3
RAI 221 — 06.02.01-20	4	5

The schedule for technically correct and complete responses to the remaining questions remains unchanged and is provided below:

Question #	Response Date
RAI 221 — 06.02.01-15	December 17, 2009
RAI 221 — 06.02.01-21	December 17, 2009
RAI 221 — 06.02.01-22	December 17, 2009
RAI 221 — 06.02.01-23	December 17, 2009
RAI 221 — 06.02.01-24	December 17, 2009
RAI 221 — 06.02.01-26	December 17, 2009
RAI 221 — 06.02.01-28	December 17, 2009
RAI 221 — 06.02.01-30	December 17, 2009
RAI 221 — 06.02.01-32	December 17, 2009
RAI 221 — 06.02.01-33	December 17, 2009
RAI 221 — 06.02.01-34	December 17, 2009
RAI 221 — 06.02.01-35	December 17, 2009

RAI 221 — 06.02.01-42	December 17, 2009
RAI 221 — 06.02.01-44	December 17, 2009

Sincerely,

Ronda Pederson

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: WELLS Russell D (AREVA NP INC)
Sent: Thursday, August 27, 2009 6:19 PM
To: 'Getachew Tesfaye'
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); SLIVA Dana (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSAR Ch 6, Supplement 2

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 3 of the 32 questions of RAI No. 221 on June 17, 2009. Supplement 1 response to RAI No. 221 was sent on July 31, 2009 to address 4 of the remaining 29 questions. The attached file, “RAI 221 Supplement 2 Response US EPR DC.pdf,” provides technically correct and complete responses to 8 of the remaining 25 questions.

The following table indicates the respective pages in the response document, “RAI 221 Supplement 2 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 221 — 06.02.01-27	2	5
RAI 221 — 06.02.01-29	6	7
RAI 221 — 06.02.01-38	8	13
RAI 221 — 06.02.01-39	14	14
RAI 221 — 06.02.01-40	15	15
RAI 221 — 06.02.01-41	16	18
RAI 221 — 06.02.01-43	19	20
RAI 221 — 06.02.01-46	21	22

The schedule for technically correct and complete responses to the remaining questions remains unchanged and is provided below:

Question #	Response Date
RAI 221 — 06.02.01-15	December 17, 2009
RAI 221 — 06.02.01-16	September 30, 2009
RAI 221 — 06.02.01-18	September 30, 2009
RAI 221 — 06.02.01-20	September 30, 2009
RAI 221 — 06.02.01-21	December 17, 2009
RAI 221 — 06.02.01-22	December 17, 2009

RAI 221 — 06.02.01-23	December 17, 2009
RAI 221 — 06.02.01-24	December 17, 2009
RAI 221 — 06.02.01-26	December 17, 2009
RAI 221 — 06.02.01-28	December 17, 2009
RAI 221 — 06.02.01-30	December 17, 2009
RAI 221 — 06.02.01-32	December 17, 2009
RAI 221 — 06.02.01-33	December 17, 2009
RAI 221 — 06.02.01-34	December 17, 2009
RAI 221 — 06.02.01-35	December 17, 2009
RAI 221 — 06.02.01-42	December 17, 2009
RAI 221 — 06.02.01-44	December 17, 2009

Sincerely,

(Russ Wells on behalf of)
Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
New Plants Deployment
AREVA NP, Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, July 31, 2009 2:51 PM
To: 'Tesfaye, Getachew'
Cc: BEELMAN Ronald J (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSARCh. 6 , Supplement 1

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 3 of the 32 questions of RAI No. 221 on June 17, 2009. The attached file, “RAI 221 Supplement 1 Response US EPR DC.pdf” provides technically correct and complete responses to 4 of the remaining 29 questions and a revised schedule for the one partial response (RAI 221 — 06.02.01-38c) of the remaining 29 questions.

The following table indicates the respective pages in the response document, “RAI 221 Supplement 1 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 221 — 06.02.01-25	2	2
RAI 221 — 06.02.01-31	3	3
RAI 221 — 06.02.01-36	4	4
RAI 221 — 06.02.01-37	5	5

The schedule for technically correct and complete responses to the remaining questions has been changed and is provided below:

Question #	Response Date

RAI 221 — 06.02.01-15	December 17, 2009
RAI 221 — 06.02.01-16	September 30, 2009
RAI 221 — 06.02.01-18	September 30, 2009
RAI 221 — 06.02.01-20	September 30, 2009
RAI 221 — 06.02.01-21	December 17, 2009
RAI 221 — 06.02.01-22	December 17, 2009
RAI 221 — 06.02.01-23	December 17, 2009
RAI 221 — 06.02.01-24	December 17, 2009
RAI 221 — 06.02.01-26	December 17, 2009
RAI 221 — 06.02.01-27	August 27, 2009
RAI 221 — 06.02.01-28	December 17, 2009
RAI 221 — 06.02.01-29	August 27, 2009
RAI 221 — 06.02.01-30	December 17, 2009
RAI 221 — 06.02.01-32	December 17, 2009
RAI 221 — 06.02.01-33	December 17, 2009
RAI 221 — 06.02.01-34	December 17, 2009
RAI 221 — 06.02.01-35	December 17, 2009
RAI 221 — 06.02.01-38c	August 27, 2009
RAI 221 — 06.02.01-39	August 27, 2009
RAI 221 — 06.02.01-40	August 27, 2009
RAI 221 — 06.02.01-41	August 27, 2009
RAI 221 — 06.02.01-42	December 17, 2009
RAI 221 — 06.02.01-43	August 27, 2009
RAI 221 — 06.02.01-44	December 17, 2009
RAI 221 — 06.02.01-46	August 27, 2009

Sincerely,

Ronda Pederson

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, June 17, 2009 5:41 PM
To: 'Getachew Tesfaye'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); GUCWA Len T (EXT); BEELMAN Ronald J (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSARCh. 6 (Part 2 of 2)

Getachew,
Attached is response to RAI 221 (Part 2 of 2).

Ronda Pederson

ronda.pederson@areva.com

Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, June 17, 2009 5:09 PM
To: 'Getachew Tesfaye'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); GUCWA Len T (EXT); BEELMAN Ronald J (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSARCh. 6 (Part 1 of 2)

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 221 Response US EPR DC.pdf" provides technically correct and complete responses to 3 of the 32 questions.

The following table indicates the respective pages in the response document, "RAI 221 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 221 — 06.02.01-15	2	2
RAI 221 — 06.02.01-16	3	3
RAI 221 — 06.02.01-17	4	4
RAI 221 — 06.02.01-18	5	5
RAI 221 — 06.02.01-19	6	6
RAI 221 — 06.02.01-20	7	7
RAI 221 — 06.02.01-21	8	8
RAI 221 — 06.02.01-22	9	9
RAI 221 — 06.02.01-23	10	10
RAI 221 — 06.02.01-24	11	11
RAI 221 — 06.02.01-25	12	12
RAI 221 — 06.02.01-26	13	13
RAI 221 — 06.02.01-27	14	14
RAI 221 — 06.02.01-28	15	15
RAI 221 — 06.02.01-29	16	16
RAI 221 — 06.02.01-30	17	17
RAI 221 — 06.02.01-31	18	18
RAI 221 — 06.02.01-32	19	19
RAI 221 — 06.02.01-33	20	20
RAI 221 — 06.02.01-34	21	21
RAI 221 — 06.02.01-35	22	22
RAI 221 — 06.02.01-36	23	23
RAI 221 — 06.02.01-37	24	24
RAI 221 — 06.02.01-38	25	25
RAI 221 — 06.02.01-39	26	26
RAI 221 — 06.02.01-40	27	27

RAI 221 — 06.02.01-41	28	28
RAI 221 — 06.02.01-42	29	29
RAI 221 — 06.02.01-43	30	30
RAI 221 — 06.02.01-44	31	31
RAI 221 — 06.02.01-45	32	34
RAI 221 — 06.02.01-46	35	35

A complete answer is not provided for 29 of the 32 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 221 — 06.02.01-15	December 17, 2009
RAI 221 — 06.02.01-16	September 30, 2009
RAI 221 — 06.02.01-18	September 30, 2009
RAI 221 — 06.02.01-20	September 30, 2009
RAI 221 — 06.02.01-21	December 17, 2009
RAI 221 — 06.02.01-22	December 17, 2009
RAI 221 — 06.02.01-23	December 17, 2009
RAI 221 — 06.02.01-24	December 17, 2009
RAI 221 — 06.02.01-25	July 31, 2009
RAI 221 — 06.02.01-26	December 17, 2009
RAI 221 — 06.02.01-27	August 27, 2009
RAI 221 — 06.02.01-28	December 17, 2009
RAI 221 — 06.02.01-29	August 27, 2009
RAI 221 — 06.02.01-30	December 17, 2009
RAI 221 — 06.02.01-31	July 31, 2009
RAI 221 — 06.02.01-32	December 17, 2009
RAI 221 — 06.02.01-33	December 17, 2009
RAI 221 — 06.02.01-34	December 17, 2009
RAI 221 — 06.02.01-35	December 17, 2009
RAI 221 — 06.02.01-36	July 31, 2009
RAI 221 — 06.02.01-37	July 31, 2009
RAI 221 — 06.02.01-38c	July 31, 2009
RAI 221 — 06.02.01-39	August 27, 2009
RAI 221 — 06.02.01-40	August 27, 2009
RAI 221 — 06.02.01-41	August 27, 2009
RAI 221 — 06.02.01-42	December 17, 2009
RAI 221 — 06.02.01-43	August 27, 2009
RAI 221 — 06.02.01-44	December 17, 2009
RAI 221 — 06.02.01-46	August 27, 2009

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road

Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Tuesday, May 19, 2009 7:30 PM
To: ZZ-DL-A-USEPR-DL
Cc: Walton Jensen; Christopher Jackson; Jason Carneal; Joseph Colaccino; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 221 (2792), FSARCh. 6

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on April 14, 2009, and on May 15, 2009, you informed us that the RAI is clear with exception of Draft RAI Question 06.02.01-15, Part 33. After further evaluation, the staff has determined that Draft RAI Question 06.02.01-15, Part 33 is unnecessary and it is deleted. Additionally, per your request, RAI 221 has been renumbered to break up the single RAI question with 32 parts into 32 separate questions, i.e. Question 06.02.01-15, Part1-32 are now Questions 06.02.01-15 - Question 06.02.01-46. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

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

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 1051

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D31017E367F)

Subject: Response to U.S. EPR Design Certification Application RAI No. 221, FSAR Ch 6, Supplement 4
Sent Date: 12/18/2009 11:41:02 AM
Received Date: 12/18/2009 11:41:06 AM
From: Pederson Ronda M (AREVA NP INC)

Created By: Ronda.Pederson@areva.com

Recipients:
"BENNETT Kathy A (OFR) (AREVA NP INC)" <Kathy.Bennett@areva.com>
Tracking Status: None
"DELANO Karen V (AREVA NP INC)" <Karen.Delano@areva.com>
Tracking Status: None
"BEELMAN Ronald J (AREVA NP INC)" <Ronald.Beelman@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	16587	12/18/2009 11:41:06 AM
RAI 221 Supplement 4 Response US EPR DC_PUBLIC.pdf		253380

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

Response to

Request for Additional Information No. 221, Supplement 4

5/19/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 06.02.01 - Containment Functional Design
Application Section: 6.2.1, Technical Report ANP-10299P

QUESTIONS for Containment and Ventilation Branch 1 (AP1000/EPR Projects)
(SPCV)

Question 06.02.01-15:

The last paragraph of Section 2.2 on page 2-10 of ANP-10299P states that the convection foils provide the minimum free-flow cross-sectional area needed to mitigate a small break LOCA event. Provide analyses of potential design basis small break LOCA events within the US-EPR containment to justify this conclusion.

Response to Question 06.02.01-15:

A spectrum of small break loss of coolant accident (SBLOCA) breaks was analyzed, including a range of cold leg pump discharge, cold leg pump suction, and hot leg breaks, ranging from a three inch break to the transition break size of 0.5 ft^2. The SBLOCA analyses use conservative assumptions that maximize the mass and energy released from the reactor coolant system (RCS) to the containment atmosphere as described in U.S. EPR FSAR Tier 2, Section 6.2.1.3.1 (as revised in the Response to RAI 209, Supplement 1, Question 06.02.01-14). In addition, the analyses assume that all doors between different rooms in the containment remain closed during the entire transient.

In accordance with NUREG-0800, Section 6.2.1.3, the sources of stored and generated energy used in all loss of coolant accident (LOCA) analyses include:

- Reactor power (plus an appropriate calorimetric uncertainty).
- Decay heat.
- Stored energy in the core.
- Stored energy in the RCS fluid and metal, including the reactor vessel (RV) and internals.
- Metal-water reaction energy.
- Stored energy in the secondary system, including the steam generator (SG) tubing and secondary water.

Reactivity components are chosen to provide a conservative insertion of negative reactivity. An appropriate initial stored energy in the core is obtained by using a conservatively high initial fuel temperature. The RCS metal is modeled accurately with respect to its size, location, and composition. The SG secondary side metal mass that is in contact with RCS fluid is also explicitly modeled, and RELAP5 includes appropriate computation of the heat transfer across the SG tubes. The energy addition due to the metal-water reaction is calculated based on the correlation (Baker-Just) specified in 10 CFR 50, Appendix K.

To demonstrate that the free-flow, cross-sectional area of the convection foils is adequate to mitigate an SBLOCA, the three inch hot leg break was analyzed without crediting the rupture foils to open. Figure 06.02.01-15-1 through Figure 06.02.01-15-4 provide containment pressure and temperature response with and without credit for the rupture foils. The figures show that the long-term pressure and temperature response is similar for both cases and bounded by the large break loss of coolant accident (LBLOCA) response. In the short-term, the case that does not credit the rupture foils results in a local pressure rise adjacent to the break room. The pressure peak is less than 30 psia and is mitigated when the convection foils open.

FSAR Impact:

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

Figure 06.02.01-15-1—Containment Pressure - 3" HL Break with Rupture Foils

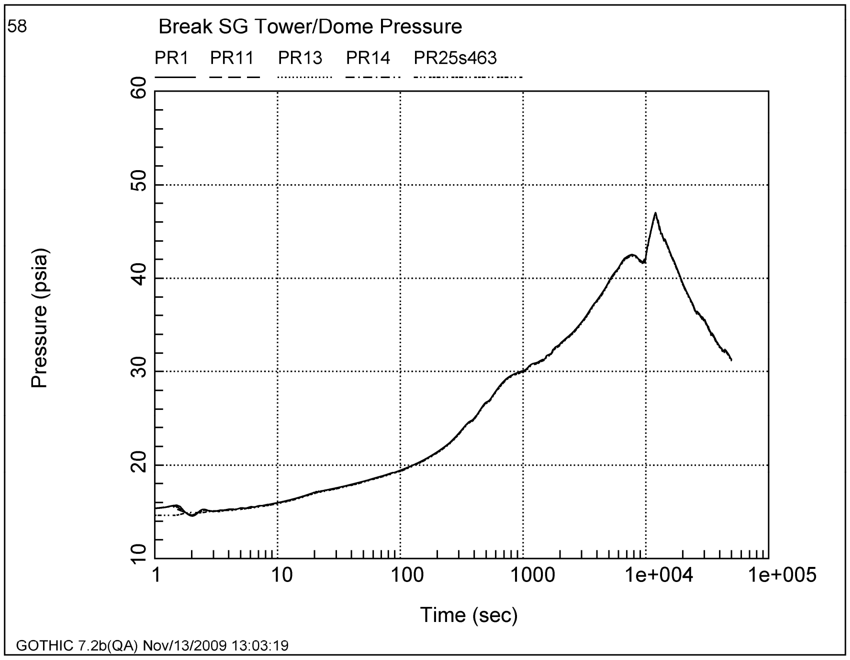


Figure 06.02.01-15-2—Containment Temperature – 3” HL Break with Rupture Foils

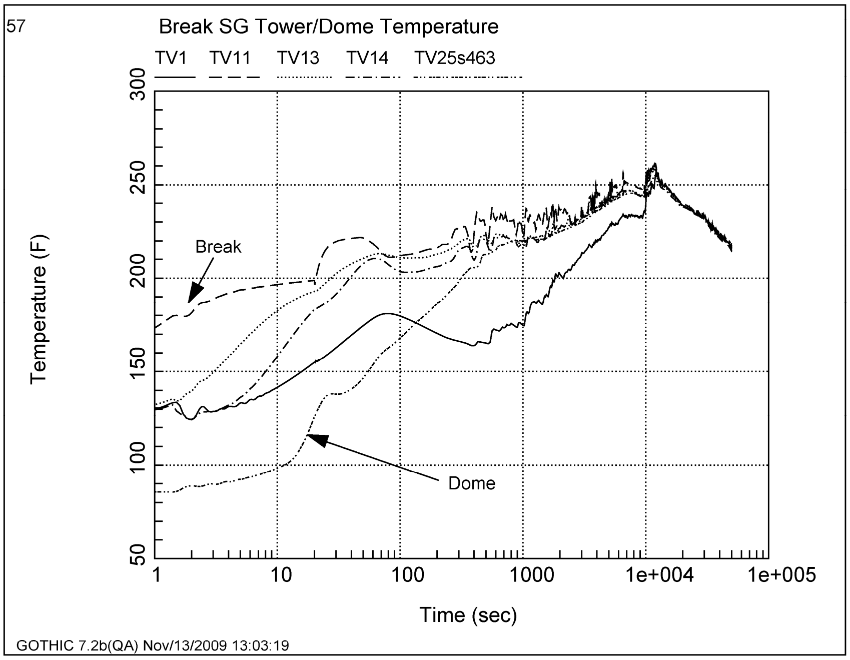


Figure 06.02.01-15-3—Containment Pressure - 3" HL Break – without Rupture Foils

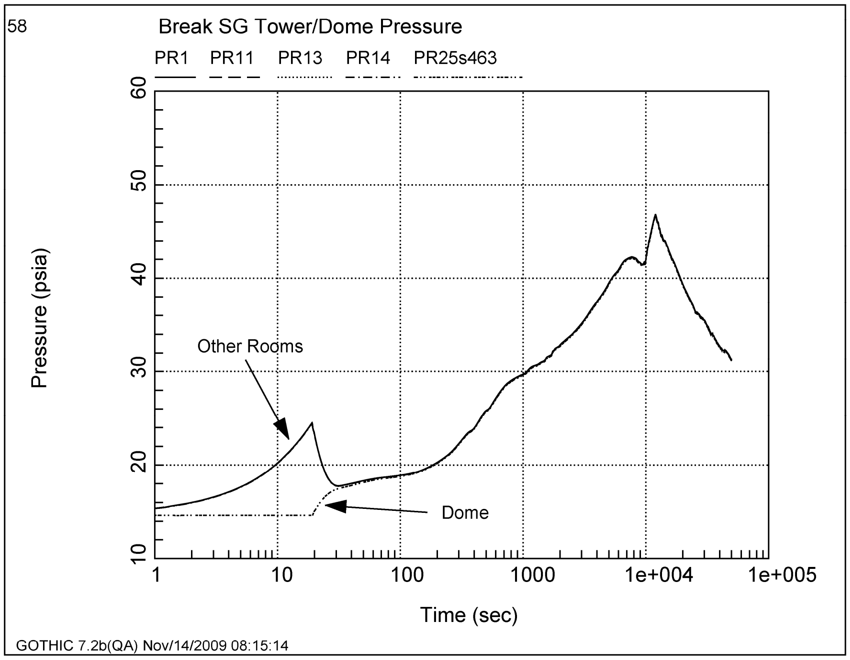
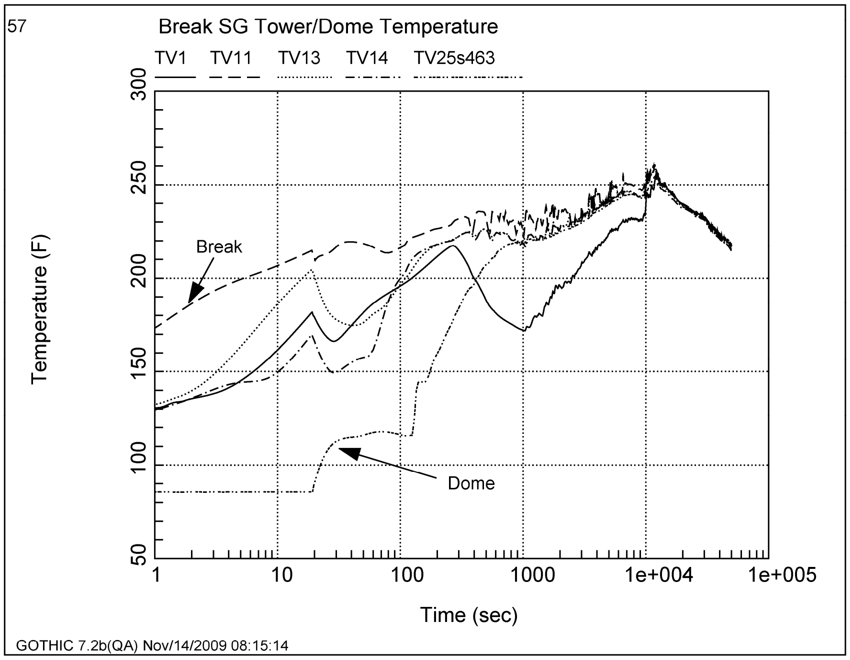


Figure 06.02.01-15-4—Containment Temperature - 3" HL Break – without Rupture Foils



Question 06.02.01-21:

Pages 6-117 and 6-118 of ANP-10299P describe doors between the accessible and non-accessible areas of the containment which are safety-related to prevent compartment over-pressurization. On page 6-120 non-safety-related access doors ("failure" junctions) are described. Provide a complete description of both the safety and non-safety related doors. Show their location and describe their operation. Provide analyses showing the effect on containment and compartment pressure if the doors function as designed and if they do not function.

Response to Question 06.02.01-21:

Table 06.02.01-21-1 lists the doors in the Containment Building, including their location and opening characteristics. Six of the doors are safety-related and are described in U.S. EPR FSAR Tier 2, Table 6.2.1-25 (as revised in the Response to RAI 209, Supplement 1, Question 06.02.01-14). The remaining doors are not classified as safety-related and are not assumed to open in either the main steam line break (MSLB) or loss of coolant accident (LOCA) analyses in the U.S.EPR FSAR. Sensitivity studies were completed where all of the doors, some of the doors, and none of the doors were allowed to open. These studies confirm that the limiting containment pressure results when all of the doors remain closed.

For the Response to RAI 209, Supplement 1, Question 06.02.01-14, analyses of breaks in the pressurizer compartment were completed to determine the number of doors that would have to open in order to vent the break effluent to the containment to maintain the containment pressure below the design limit. The results identified that at least five of the six safety-related doors identified in U.S. EPR FSAR Tier 2, Table 6.2.1-25 would need to operate and that the failure of one door was a less limiting single failure than eliminating one train of the emergency core cooling system (ECCS).

FSAR Impact:

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

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Table 06.02.01-21-1—U.S. EPR Analytical Inputs for Reactor Building Doors (8 Sheets)

Official Use Only - Security Sensitive Information - Withhold under 10 CFR 2.390

Question 06.02.01-26:

Section 5.3.1.5.6 of ANP-10299P discusses hot leg nozzle bypass and states that during later blowdown and early reflood phases, the thinner core barrel structure is cooled and the gaps grow to their maximum size. It is further stated that during the post-reflood phase the reactor vessel shell will cool and approach the pumped ECCS injection temperature but this may take several hours. The effect of hot leg injection is stated to have the effect of cooling portions of the core barrel faster than the reactor vessel shell and to keep the gaps open longer. Since the effect of hot leg nozzle gaps might have the effect of permitting hot leg injection to drain to the downcomer and thereby not be available for core mixing, provide an evaluation of this phenomenon showing its effect on steam condensation in the reactor vessel during hot leg injection.

Response to Question 06.02.01-26:

The introduction of emergency core coolant (ECC) into the upper plenum via hot leg injection cools the shroud at a faster rate than the reactor pressure vessel (RPV) thereby creating an intervening gap. The presence of these gaps during hot leg injection results in competing effects on core cooling. A portion of the relatively cold water from the ECCS hot leg injection may pass through these hot leg gaps bypassing the core and be directed toward the break, making it unavailable for core cooling and increasing core steaming (negative effect). On the other hand, steam from the upper plenum escaping through the gaps can condense in the downcomer, reducing the amount of steam reaching the containment (positive effect).

Another positive effect results when “warm” upper plenum water escapes through the gaps in the non-injected loops to the downcomer, resulting in cooler water coming up through the core. This loss from the upper plenum to the downcomer enhances ECC flow to the core resulting in additional core cooling. Consequently, the gaps in the non-injected loops are only a credit. The relative magnitude of these competing effects from the injected loop and non-injected loop gaps will determine the net consequence of the gaps to core cooling relative to the containment analysis.

A one-dimensional quasi steady-state analysis was performed to understand the effects of gap size and hot leg water level on core cooling. Gap size is the only time-varying parameter. Several cases were performed for variation in gap size and hot leg water level.

A distinction is made between the sizes of the gaps in the two loops with hot leg injection versus the size of the gaps in the two loops without hot leg injection.

The hot legs gaps in the reactor coolant system (RCS) loops without hot leg injection are relatively small because there is no direct cooling effect from the ECC associated with the hot leg injection flow. Based on thermal-stress analysis data for the hot leg injection, an estimate was made for the gap size on the two hot leg gaps in the non-injected loops prior to injection. This estimate was selected for all four gaps at the initiation of hot leg injection since all four gaps should have roughly the same size at this point.

Figure 06.02.01-26-1 shows gap data, from a thermal-stress analysis, as a function of time for a gap exposed to hot leg injection. A circumferential gradient in gap size exists after hot leg injection because the lower part of the gap is “cooler,” i.e., it is more directly exposed to the ECCS injection than the upper part, which is exposed to steam. For this analysis, a uniform circumferential gap size is assumed. Just prior to 5,400 seconds[1], there is a 3 mm gap size for the upper portion ("upper part of gap" curve at 5,400 seconds) compared to about a 2.3 mm gap size for the lower portion ("lower part of gap" curve at 5400 seconds). It is conservative to use a small gap size for the gaps in non-injected loops (core cooling credit decreases with decreasing size for these gaps); therefore, a non-injected loop gap size value of 2.3 mm is used from "lower part of gap" curve. Over time, the non-injected gap sizes will begin closing as the entire reactor vessel (RV) cools, but it is unclear how fast these gaps will close. Three sets of assumptions are made on the rate of change of non-injected gap size closure to attempt to bound these non-injected gap sizes (see Table 06.02.01-26-1 for the six cases):

1. Non-injected gap size remains at 2.3 mm over the entire transient (cases 1 through 3).
2. Non-injected gap size decreases at the same assumed linear rate as the injected-side gap size (linear approximation of 0.6 mm/hour gap closure rate) (cases 4 and 5).
3. Non-injected gap is completely closed at 9750 seconds (case 6) – conservative.

For the injected loop gaps, it is conservative to use the "lower part of gap" curve because it provides a larger average gap size. As gap size increases, both the cooling detriment and cooling credit effects of these gaps increase. Over most of the assumed range of hot leg level (except for lower levels) the detriment is larger than the credit. As gap size increases, the net detriment increases. Using the values for the "lower part of gap" for the injected loop hot leg gaps is conservative over most of the range of assumed hot leg levels. These "lower part of gap" values are approximated as closing at the rate of approximately 0.6mm/hour (based on the slope of "lower part gap" curve in Figure 06.02.01-26-1).

A gap closure rate of approximately 0.6mm/hr is linearly extrapolated to 13,000 seconds to get gap size values for cases 3 and 5.

For each set of gap sizes (each case), the hot leg water/steam interface level was varied from 25 percent to 100 percent (100 percent corresponding to hot leg filled with water). This analysis parameterizes hot leg level without considering the cause of hot leg water level change. As the water level rises in the hot leg, it consequently also rises in the upper plenum, implying less steam is being generated in the core. A 100 percent filled hot leg, for example, implies that the core is not producing any steam (and no flow path exists for the steam to reach containment).

For an assumed 50 percent hot leg water level, which is more realistic, a net steaming rate difference (due to presence of hot leg gaps) relative to that of the containment analysis is -0.7 lb_m/sec (gaps provide extra cooling to core) and 1.8 lb_m/sec for cases 3 and 5, respectively (see Table 06.02.01-26-1). These steaming rates compare with containment analysis steaming rates of 31 lb_m/sec at initiation of hot leg injection and 15.7 lb_m/sec at 13,000 seconds after the break, respectively.

Case 6 (see Table 06.02.01-26-1), which is the most conservative regarding non-injected gap size, shows the effects of taking no credit for the gaps on the non-injected loops (0 mm gap size, a cooling credit). This case, at 9,750 seconds after break initiation, (time at which gap size data ends) predicts an increase in steaming rate of 4.8 lbm/sec, at a 50 percent hot leg water level, adding about 18.5 percent steaming to the containment analysis steaming rate of 26

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[1] The calculation of the gap size is based on re-alignment of the ECCS at 90 minutes.

lb_m/sec at 9,801 seconds. Figure 06.02.01-26-2 and Figure 06.02.01-26-3 show the sensitivity of gap size and hot leg water level on core cooling due to the presence of gaps.

It is concluded that as the water level rises in the hot leg (and the upper plenum), the gap benefit/detriment ranges from slightly beneficial to slightly detrimental. Because there is only a slight detriment when the hot leg becomes significantly full, at which time steaming has almost halted, the loss of ECCS core cooling due to the presence of hot leg gaps is not significant. The worst case (case 6), at a 50 percent hot leg level, conservatively adds 4.8 lb_m/sec of steaming (an additional 19 percent steaming to the containment analysis prediction). Even at a hot leg level of 100 percent for case 6, the steaming rate is only 7.8 lb_m/sec. This value is unrealistic because, as a 100 percent hot leg level implies, there is no steaming in the core/upper plenum. The other cases, which are more realistic, show that at a hot leg level of 50 percent, the steaming rate increases are less than 12 percent above those of the containment analysis (see Table 06.02.01-26-1). In the containment analyses, there are conservatisms that will offset any negative effects of the hot leg gaps. For example, a very conservative upper plenum mixing efficiency was used in the containment analyses as discussed in Technical Report ANP-10299P.

FSAR Impact:

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

Table 06.02.01-26-1—Summary of Results

Case	Time after Break (seconds)	Injected Loop HL Gap Size (mm)	Non-Injected Loop HL Gap Size (mm)	Net Steaming Rate @ 50% HL Water Level (lb_m/sec)	Increase Over Containment Analysis Steaming Rate (no gaps)* (%)
1	5,400	4	2.3	0.9	2.9
2	9,750	3.5	2.3	0.1	0.4
3	13,000	3	2.3	-.7	-4.5
4	9,750	3.5	1.8	1.3	5.0
5	13,000	3	1.3	1.8	11.5
6	9,750	3.5	0	4.8	18.5

*Based on 31 lb_m/sec, 26 lb_m/sec, and 15.7 lb_m/sec at 5,501, 9,801, and 13,001 seconds, respectively, after the break.

Figure 06.02.01-26-1—Gap Between Reactor Vessel and Internals Close to the Outlet Nozzle

Figure 06.02.01-26-2: Net Core Cooling vs. Hot Leg Steam/Water Interface Level (non-injected gap sizes fixed)

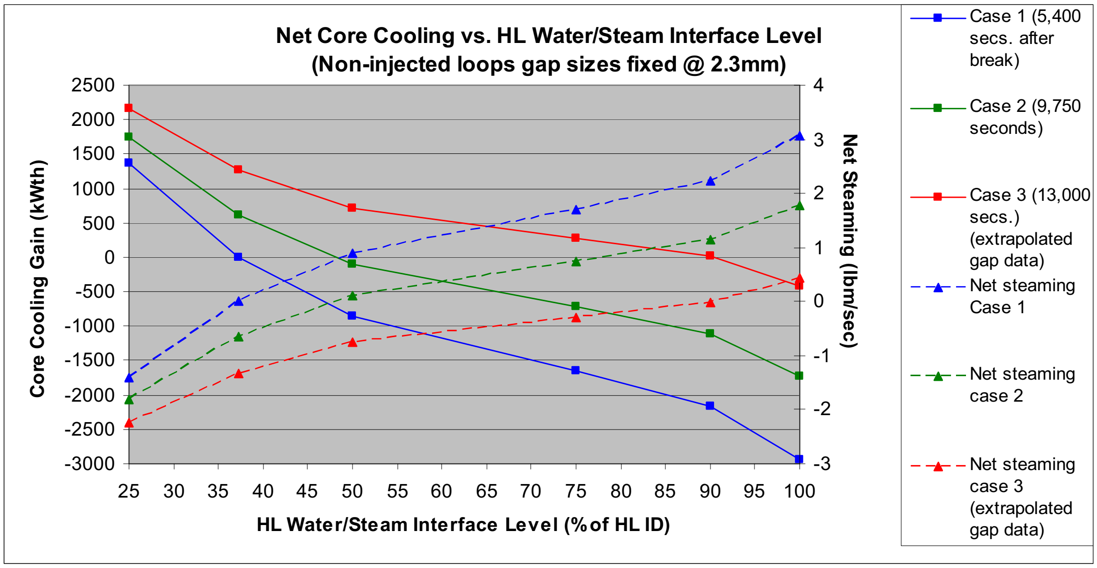
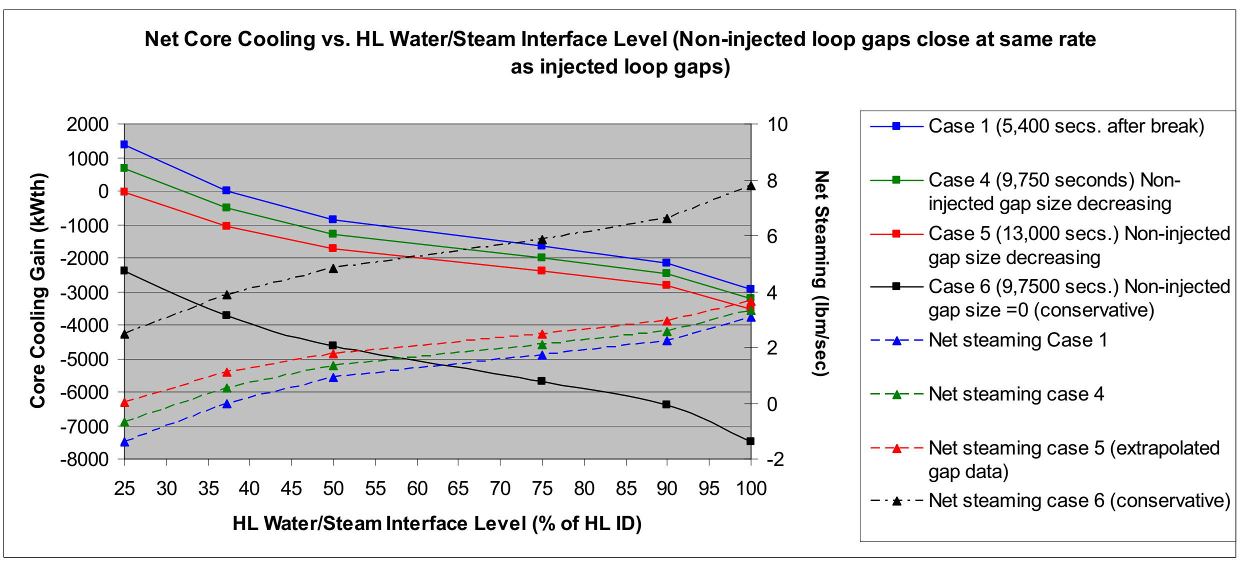


Figure 06.02.01-26-3: Net Core Cooling vs. Hot Leg Steam/Water Interface Level (non-injected loop gaps close at same rate as injected loop gaps)



Question 06.02.01-28:

For the US-EPR containment calculation in Section 8 of ANP-10299P using the new EM model, provide the heat transfer coefficient as a function of time that is calculated between the reactor vessel shell and the downcomer fluid and the temperature difference to the downcomer fluid as a function of time. Indicate the mode of heat transfer that is occurring associated with the differential temperature and heat transfer coefficient plots. Discuss the conservatism of the RELAP5 calculated heat transfer compared to the nucleate boiling assumption recommended by the SRP.

Response to Question 06.02.01-28:

The downcomer is modeled in RELAP5 by dividing it into five volumes: upper, inlet, two lower, and an extension. Three heat structures, nozzle shell, intermediate shell, and lower shell connecting to these volumes model the reactor pressure vessel (RPV). The major mode of heat transfer for the different sections is by nucleate boiling as long as the liquid phase is present.

For example, the parameters requested in this Question are provided for the RPV intermediate shell connected to the first section of the lower downcomer. Figure 06.02.01-28-1 shows the temperature difference between the RPV intermediate shell and the bulk fluid temperature. Figure 06.02.01-28-2 provides the heat transfer coefficient and mode number, showing that nucleate boiling is the major heat transfer mode in this section. Figure 06.02.01-28-3 shows that in the initial blowdown phase, the major mode of heat transfer is nucleate boiling. Based on the vapor fraction (Figure 06.02.01-28-4) and differential temperature conditions, single phase liquid convection, vapor condensation, and transition boiling exist for short durations during the initial blowdown phase, but the major heat transfer mode is nucleate boiling.

The same conclusion applies to other lower downcomer sections because more water is present. For sections above the intermediate section, especially the upper downcomer region, transition and film boiling is present because of the vapor conditions. Even in these sections, during the initial blowdown phase, the heat transfer is primarily due to nucleate boiling.

The heat transfer modes calculated by RELAP5 are consistent with the void fraction and differential temperature conditions in the different downcomer sections and meet the SRP recommendations. For the initial blowdown phase, RELAP5 uses nucleate boiling to calculate the heat transfer between the reactor vessel (RV) shell and downcomer fluid as long as the liquid phase is present. In the long term, nucleate boiling is the dominant heat transfer mode in the downcomer region.

FSAR Impact:

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

Figure 06.02.01-28-1—Temperature difference:- RPV intermediate shell – Lower downcomer

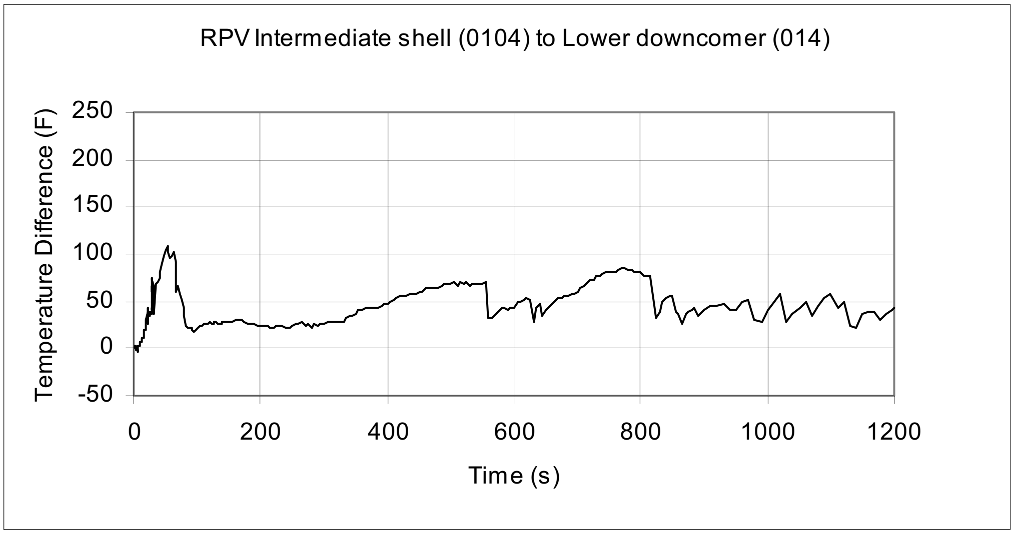


Figure 06.02.01-28-2—Heat transfer coefficient and mode:- RPV intermediate shell – Lower downcomer

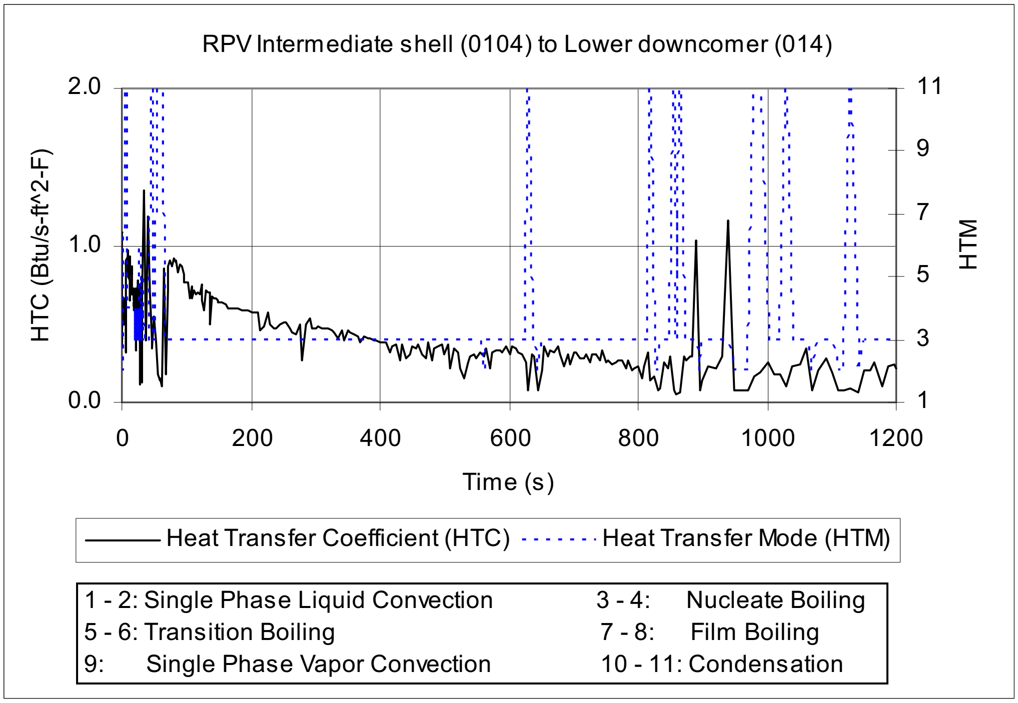


Figure 06.02.01-28-3—Heat transfer coefficient and mode:- RPV intermediate shell – Lower downcomer (40 seconds)

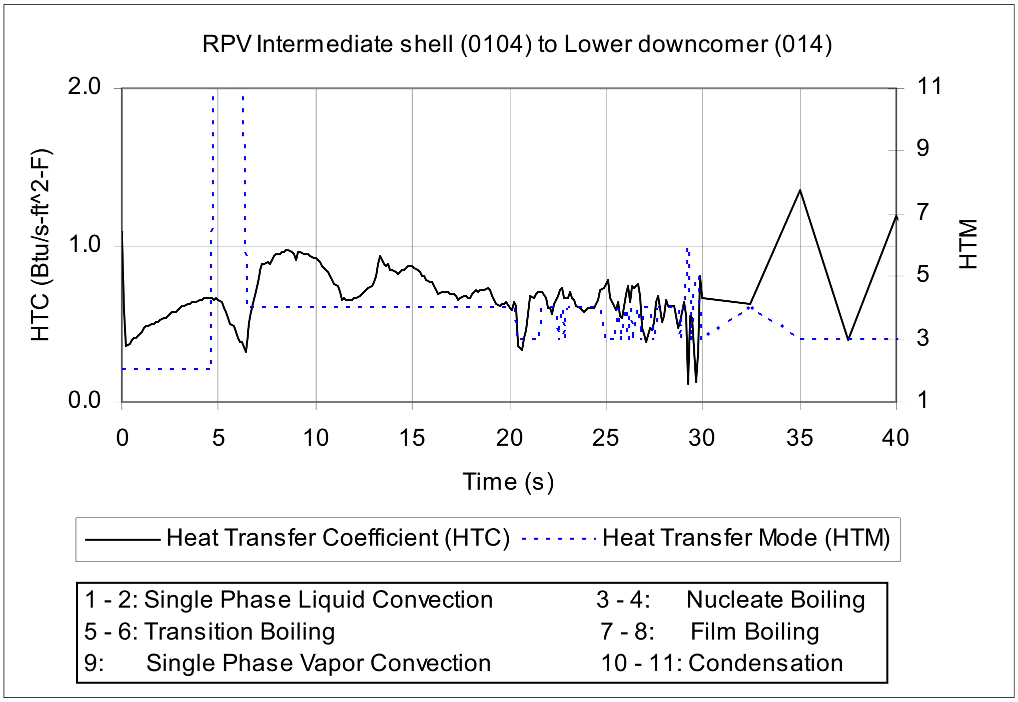
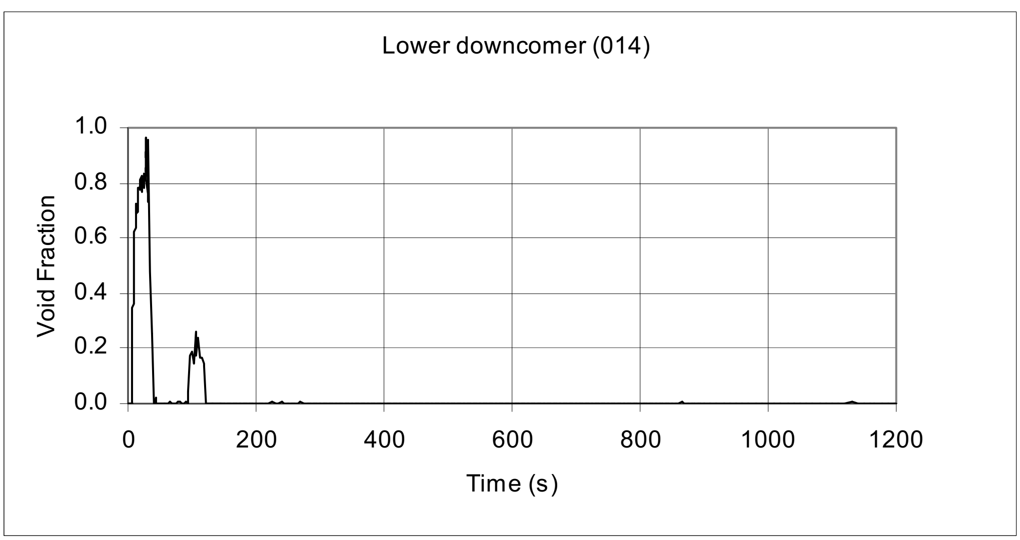


Figure 06.02.01-28-4—Void fraction:- Lower downcomer



Question 06.02.01-30:

Section 6.1.3 describes calculation of steam generator heat transfer by RELAP5-BW. Predictions for FLECHT-SEASET tests are presented which show good agreement when the test facility is modeled. The prediction of steam generator heat transfer appears to be a function of the steam and liquid flow rates within the steam generator tubes. For the US-EPR containment analysis with the new EM model in ANP-10299P show that with the predicted steam generator tube flow rates that the heat transfer predicted for the US-EPR is conservative based on comparison to experimental data.

Response to Question 06.02.01-30:

As part of the NRC/EPRI/Westinghouse FLECHT SEASET reflood and natural circulation test program, a series of separate effect tests were conducted on a U-tube steam generator (SG) (Reference 1). These tests measure the SG secondary side to primary side heat release under postulated inlet fluid conditions for a pressurized water reactor (PWR) loss of coolant accident (LOCA). The test results are presented in Reference 1. The FLECHT SEASET tests are primarily heat transfer tests as opposed to pressure drop tests.

The FLECHT-SEASET experimental test results show the appearance of a quench front inside the SG primary side tubes. The dispersed two-phase flow above the quench front provides enough heat transfer and precursory wall cooling so that the quench front advances up the tubes. An abrupt drop in the temperature at certain times confirms there is active heat transfer inside the tubes.

The secondary side of the test facility consisted of 32 active tubes. The outer tube diameter is 7/8 inch compared to 3/4 inch for the U.S. EPR design, and the tube thickness is 0.05 inch compared to 0.04 inch for the U.S. EPR design. The tube height of the FLECHT secondary side test facility is 35 ft compared to 38.42 ft for the tube height for the U.S. EPR SGs.

For the benchmarking analysis, the test data in Reference 1 were reviewed and the test data comparable to the U.S. EPR plant during reflood and post reflood conditions were selected for benchmarking. The initial and boundary condition for these tests are shown in Table 06.02.01-30-1.

Table 06.02.01-30-2 shows the test conditions and compares the major test parameters with U.S. EPR parameters. In the FLECHT SEASET tests, the steam/water flow mixture is kept constant as a boundary condition, whereas in the U.S. EPR design, the flow through the SG tubes varies with decay heat and the heat load from the secondary side to primary changes through the transient time. As a result, the heat transfer from the secondary to the primary side is relatively insensitive to the primary pressure range in the post reflood phase, being higher than the FLECHT tests.

The benchmarking results show that by using the Becker critical heat flux option on both sides of the SG tube sides, the SG stored energy is removed from the secondary to the primary side. The quench fronts in the tubes were conservatively predicted using the Becker correlation.

Figure 06.02.01-30-1 shows the secondary side fluid temperatures in the broken loop, both on the inlet and the outlet sides of the tubes, for the U.S. EPR at the entrance, middle, and top of the tubes.

Figure 06.02.01-30-2 through Figure 06.02.01-30-5 show the fluid temperature on the shell side for the benchmarked FLECHT SEASET tests at the entrance, middle, and top of the tube. For the FLECHT SEASET benchmarking tests, it was assumed that the inlet and outlet shell sides have uniform temperatures.

A comparison of FLECHT SEASET tests #21806 through #22314 in Table 06.02.01-30-2, and Figure 06.02.01-30-2 and Figure 06.02.01-30-4 shows that the heat transfer decreases as quality increases.

A comparison of the U.S. EPR results to FLECHT SEASET test #21909 shows that the mass flux in the U.S. EPR is less than two times the mass flux in test #21909, but the quench front in the U.S. EPR case progresses more than two times faster than in the FLECHT test. The quality in the U.S. EPR case is higher than that in test #21909. Comparing the U.S. EPR to test #21909 shows that the tube quench in the U.S. EPR occurs faster. As a result, the secondary side energy is removed at a relatively higher rate compared to FLECHT SEASET tests.

Faster transfer of energy from the secondary side of the SGs to the primary side is conservative for the containment pressure and temperature response.

References for Question 06.02.01-30:

1. EPRI FLECHT SEASET Steam Generator Separate-Effects Task, Data Analysis and Evaluation Report, EPRI NP-1461 Report No. 9, NUREG/CR-1534, February 1982.

FSAR Impact:

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

Table 06.02.01-30-1—Initial and Boundary Conditions for the Tests

Test #	Test 21806	Test 21909	Test 22314	Test 22415
Steam Flow, lb/s	0.100	0.100	0.248	0.213
Water flow, lb/s	0.400	0.847	0.252	0.402
Steam Temperature, ^{o}F	299	309	311	300
Water temperature, ^{o}F	262	264	260	256
Mixer Pressure, psig	29	30	28	29
Outlet Pressure, psig	25	25	25	25
SG Water Level, ft	32.7	34.0	34.3	33.8
SG Dome Temperature, ^{o}F	520	525	523	522
Inlet Mixer quality	0.2	0.105	0.495	0.345

Table 06.02.01-30-2—U.S. EPR and FLECHT SEASET Test Conditions

Cases	Primary Pressure (psia)	SG Inlet Quality	SG Inlet Mass Flux (lb/s-ft2)
U.S. EPR [1] Broken Loop	60 -70	~ 0.2 – 0.3	14 - 18
Test 21806	43.2	0.200	4.77
Test 21909	44.2	0.105	9.03
Test 22314	42.2	0.495	4.74
Test 22415	43.2	0.345	5.86

Notes:

1. The parameters correspond to cold leg pump suction break between 250 to 1000 seconds.

Figure 06.02.01-30-1: Broken Loop Steam Generator Secondary Fluid Temperature (Inlet Side) for U.S. EPR

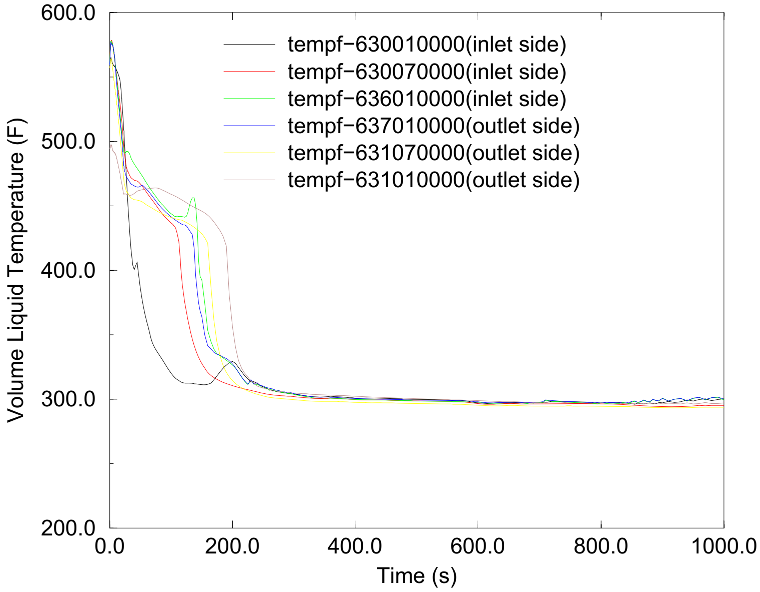


Figure 06.02.01-30-2: Secondary Side Fluid Temperature for Test 21806

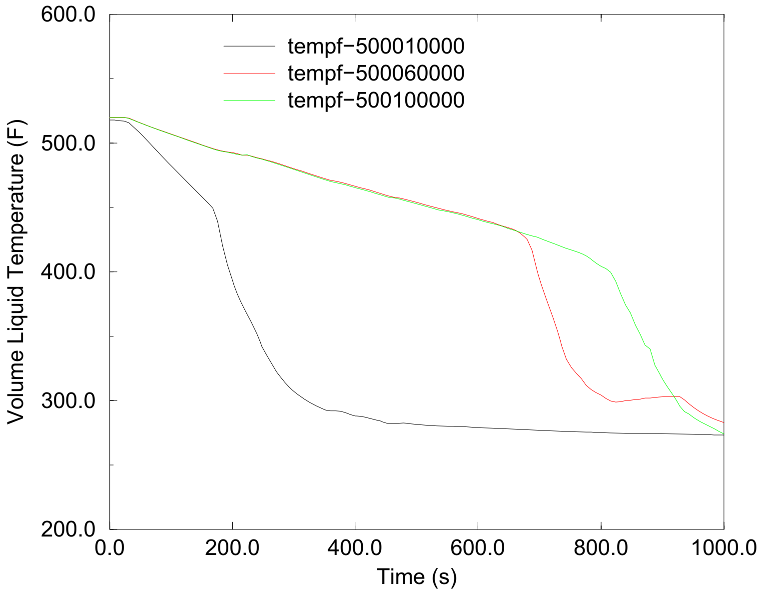


Figure 06.02.01-30-3: Secondary Side Fluid Temperature for Test 21909

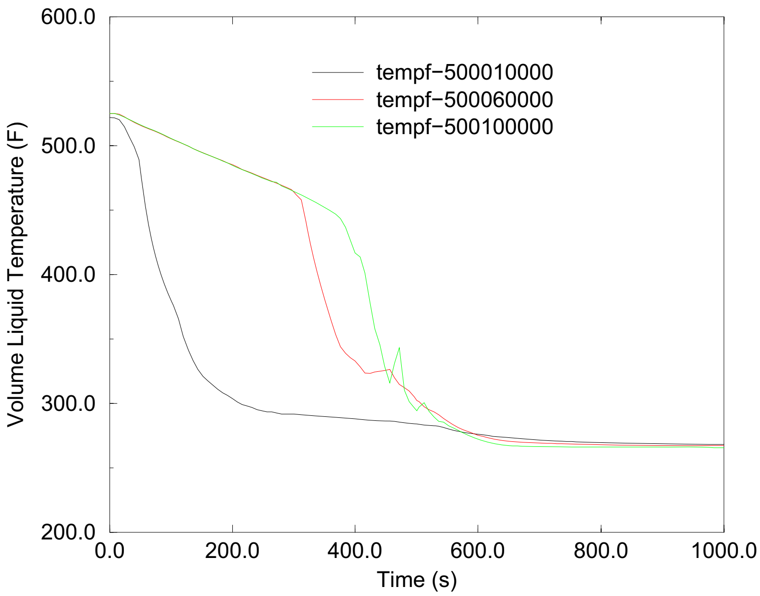


Figure 06.02.01-30-4: Secondary Side Fluid Temperature for Test 22314

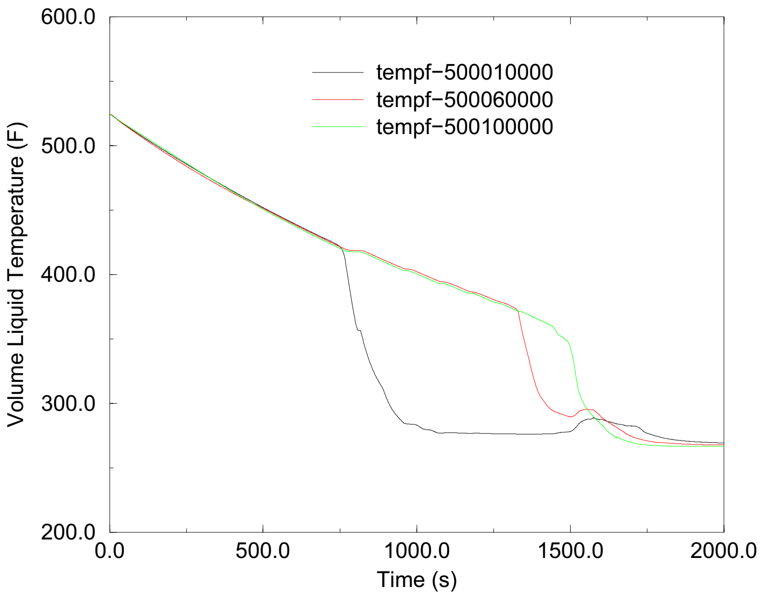
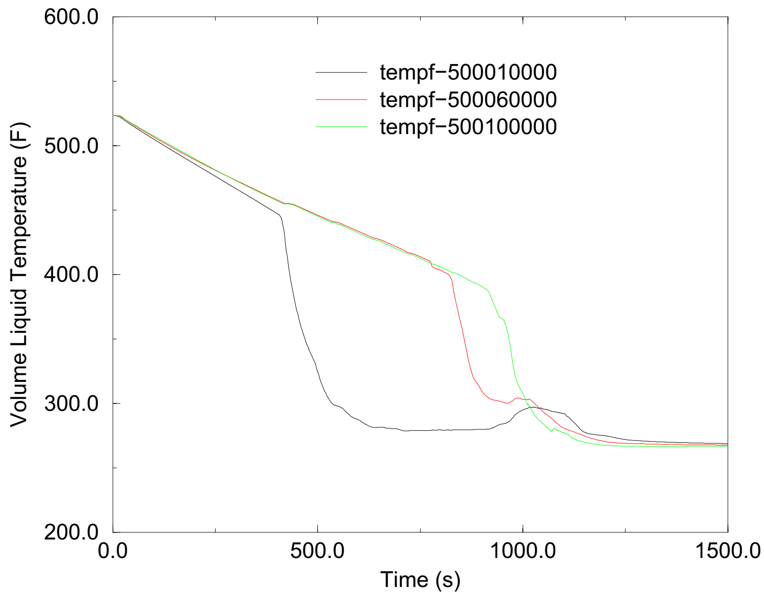


Figure 06.02.01-30-5: Secondary Side Fluid Temperature for Test 22415



Question 06.02.01-32:

The table on page 6-30 of ANP-10299P states that fuel cladding swelling and rupture was included in the RELAP5-BW prediction of mass and energy release. SRP 6.2.1.3 recommends that fuel swelling and rupture not be considered for containment mass and energy release calculations. Provide an assessment of the affect of any predicted cladding swelling and rupture on the calculated containment pressure for US-EPR and justify that the inclusion of these models leads to conservative results.

Response to Question 06.02.01-32:

The compliance matrix in ANP-10299P, Revision 2, page 8-32, "SRP 6.2.1.3 M&E for LOCA Sources of Energy," item 9 shows that the loss of coolant accident (LOCA) evaluation model utilizes a fuel cladding swelling and rupture model that includes provisions for the metal-water reaction energy addition. Fuel pin swelling and rupture models were included in the mass and energy release calculation to consider the energy addition due to the metal-water reaction.

The RELAP5-BW model used for the mass and energy release calculation considers an average core. Because an average core is modeled, the clad temperature and pressure differential across the cladding does not rise high enough during the event to cause swelling or rupture. The RELAP5 calculations show that the peak clad temperature is below 1000°F, providing considerable margin (more than 500°F) to the clad rupture temperature determined in the RELAP5 runs. Neither clad swelling nor rupture is predicted. As a result, calculated containment pressure of the U.S. EPR is unaffected by the fuel cladding swelling and rupture model.

FSAR Impact:

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

Question 06.02.01-33:

On October 31, 2008, Areva made a presentation to the NRC staff describing calculation of EPR reactor vessel mixing using the CATHARE 3D computer code. Provide a discussion comparing the reactor vessel mixing results from the CATHARE 3D analyses with the reactor vessel mixing analytical assumptions described in ANP-10299P. What value of η(mix) as defined in equation of 6-1 of the report would be predicted by CATHARE 3D? Provide a discussion of the use, validation and acceptance of CATHARE 3D in Europe.

Response to Question 06.02.01-33:

Based on the CATHARE 3D output from the NRC presentation on October 31, 2008, a mixing efficiency (η) of 68.9 percent was calculated following realignment of the low head safety injection (LHSI) to the hot legs. Following the NRC presentation, AREVA NP has presented results from additional CATHARE mixing analyses at NURETH-13 in Kanazawa City, Japan (Reference 1). The paper provides benchmarks to multiple tests and a comparison to the U.S. EPR model.

The CATHARE2 code is an advanced, best-estimate code used for nuclear power plant safety analyses (Reference 2). CATHARE2 describes two-phase flows using a two-fluid, six- equation model. The code input is provided through a flexible arrangement of component models that may be used to describe a full plant reactor coolant system (RCS) or a test facility using 0-D, 1-D, or 3-D modules. CATHARE2 applies an implicit numerical method for 0- and 1-D models, and semi-implicit in 3-D. Its qualification is based on physical validation through separate effect tests, components tests, and a large matrix of integral effect tests (Reference 3 through Reference 6). Development has been ongoing in France since 1979 by a joint effort of CEA, EDF, AREVA and IRSN.

The CATHARE2 3D module was developed to allow the calculation of 3-D thermal-hydraulic phenomena and is mainly used for the modeling of the pressurized water reactor (PWR) vessel. The assessment program of the 3-D module addresses situations of interest for safety analysis (Reference 3 through Reference 6) and is based on the following experiments:

- A simple 2-D configuration is represented in the PERICLES facility used to simulate core uncovery during small break loss of coolant accident (SBLOCA) conditions and core rewetting in large break loss of coolant accident (LBLOCA) conditions.
- Full-scale upper plenum test facility (UPTF) tests examining multi-dimensional downcomer refill phenomena in LBLOCA conditions, with countercurrent flow of liquid injected from accumulator or safety injection system (SIS), and vapor released from the core.
- LBLOCA system tests performed with the loss of fluid test (LOFT) facility, a 1/50- scaled nuclear core facility used to simulate LBLOCA transients including blowdown, refilling, and reflooding phases.

References for Question 06.02.01-33:

1. K.Abel, L. Brintet, C.K. Nithiandan, R. Martin, A. Benkert, “ Validation of CATHARE2 Code to Predict Core Thermal Mixing Using CCTF and SCTF Tests, “NURETH-13 Proceedings, Kanazawa City, Japan, October 2009.

2. I. Dor, et al., “Assessment of the CATHARE 3-D module for LBLOCA simulation,” NURETH-11 Proceedings, Avignon, France, October 2005.
3. M. Robert, M. Farvacque, et al., “CATHARE2 V2.5: a fully validated CATHARE2 version for various applications,” NURETH-10 Proceedings, Seoul, Korea, September 2003.
4. I. Dor, P. Bazin, P. Boudier (CEA/SSTH), "Assessment of the CATHARE 3-D module against LOFT test L2-5 and L2-6, “ICONE-13 Proceedings, Beijing, China, May 2005.
5. I. Dor, C. Morel, P. Bazin, P. Boudier “Assessment of the CATHARE 3-D module for LBLOCA simulation,” NURETH-11 Proceedings, Avignon, France, October 2005.
6. I. Dor, G. Lavialle, P. Germain, T. Mieusset (SSTH) "Investigation of Downcomer Level Evolution during Late Reflooding with CATHARE 3D Module," NURETH-12 Proceedings, Pittsburgh, Pennsylvania, U.S.A., September 2007.

FSAR Impact:

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

Question 06.02.01-34:

For the new EM containment M&E model described in Chapter 8 of ANP-10299P, RELAP5-BW is used to calculate the break flow to the containment until the containment loop seals are assumed to fill with water. After the loop seals are assumed to have filled, the GOTHIC code is utilized to calculate the break flow to the containment. Provide justification for the time selected for loop seal closing. Would an earlier time result in a higher containment pressure prediction?

Response to Question 06.02.01-34:

The sample problem, described in Technical Report ANP-10299P, was for cold leg pump suction (CLPS) large break loss of coolant accident (LBLOCA) and assumed loop seal formation at 1,200 seconds transient time. In the containment evaluation model (EM) mass and energy (M&E) methodology, RELAP5-BW is used to calculate the break flow to the containment prior to loop seal formation. After the loop seals are assumed to have filled, the GOTHIC code is used to calculate the break flow to the containment.

An earlier loop seal formation time increases the duration of an all steam flow discharge into containment. Closing the loop seals redirects the steam flowpath to the break with little or no condensation by the emergency core cooling system (ECCS). With earlier loop seal formation, the containment experiences pressurization over a longer time interval as a result of the prolonged steam discharge. This results in higher peak pressure prediction before low head safety injection (LHSI) realignment to hot leg injection at 3,600 seconds, which then suppresses core steaming, and reduces containment pressure.

Two independent approaches were used to justify the selection of 1,200 seconds as the loop seal formation time in the sample problem analysis. The first is a countercurrent flow limiting (CCFL) approach and the second is a venting calculation approach.

CCFL Approach

This evaluation determines how much the steam flow has to decrease to allow for the filling of the cross-over legs (horizontal suction leg) in the intact loops, leading to eventual loop seal plug. An empirical flooding correlation by Hewitt and Wallis (1963), as reported on pages 174-175 of Hsu and Graham (McGraw Hill, 1976), was used for this purpose.

The geometry where the empirical flooding correlation was applied is the vertical pipe leading to the reactor coolant pump (RCP), referred to as the CLPS upside loop seal. The diameter of the vertical tube is 2.57 ft. The gas flooding limit curve was calculated for a constant liquid injection entering at the top of the tube and directed downwards, representing the ECCS flow of one LHSI/medium head safety injection (MHSI) train and assumed in the sample problem analysis to be 440 lb_m/sec. The liquid density was assumed constant at 62 lb_m/ft^3.

Below the gas flooding limit curve, a stable counter-current annular flow (falling liquid film and upward gas or steam flow) is expected. This is an important limit because steam flows that lead to counter-current annular flow in the CLPS upside loop seal provide the requisite condition for the filling of the cross-over leg to obtain plugged loop seal.

The empirical flooding correlation is as follows:

$$\left(j_g^*\right)^{0.5} + \left(j_f^*\right)^{0.5} = C \qquad (1)$$

where C = 0.725 (sharp flanges, i.e., sharp edges)

= 0.88 (smooth flanges, i.e., rounded edges)

$$j_k^* = j_k\left[\frac{\rho_k}{gD(\rho_f - \rho_g)}\right]^{0.5} \qquad (2)$$

where j_k^* (k=f, g) is defined as the dimensionless liquid and gas superficial velocities. ρ_f and ρ_g are the liquid and gas densities, g is the acceleration due to gravity and D is the tube diameter. The text in Hsu and Graham denotes j_k^* (k=f, g) as u_k^+ (k=f, g). The equation for dimensionless superficial velocity as given in Hsu and Graham, page 173, differs slightly from equation (2) in the phasic density factor specified in the denominator. In Hsu and Graham, the phasic density factor in the denominator is given as ρ_f instead of $(\rho_f - \rho_g)$. This difference in the specification of phasic density factor in the denominator is negligible because,

$$\rho_f \approx (\rho_f - \rho_g)$$

An industrial scale pipe, such as the upside loop seal, will have sharp rather than smooth edges, and the value of C used in equation (1) is 0.725.

The gas flooding limit curve generated by equation (1) with C=0.725, and the other inputs regarding tube size (2.57 ft) and liquid flow (440 lbm/sec), is shown in Figure 06.02.01-34-1. The liquid flow at these conditions (440 lb_m/sec and 62 lb_m/ft^3) corresponds to a liquid superficial velocity of 1.363 ft/sec. In addition to the limit curve, this plot depicts the data point representing the RELAP5 calculated steam flow condition at 1,200 seconds, namely steam superficial velocity of 52 ft/sec at reactor coolant system (RCS) pressure of 43 psia.

Figure 06.02.01-34-1 shows that the data point representing the RELAP5 calculated steam flow condition at 1,200 seconds falls in the cocurrent flow regime. This indicates the flow pattern in the CLPS upside loop seal at 1,200 seconds favors “sweep-out” or entrainment of the liquid film by the steam flow originating in the core, preventing the ECCS liquid from reaching the cross-over leg and filling it.

Based on this evaluation, loop seal formation for the sample problem will occur much later than the assumed time of 1,200 seconds. Earlier loop seal formation results in conservative (higher) containment pressure prediction. Thus, the selection of 1,200 seconds as loop seal formation time in the sample problem is conservative.

Venting Calculation Approach

A calculation was performed to estimate the timing of loop seal formation in the intact loops based on assessment of the pressure difference across the CLPS upside loop seal as a function of several parameters. These parameters are: (a) time of transient which dictates the decay heat level and, hence amount of steaming, (b) elevation head and (c) friction and form losses for the flow traversing a single RCS loop from the top of the core to break location.

The criteria followed for estimation of the cut-off time at which loop seal formation occurs are as follows:

1. If the dynamic head resulting from the total steam flow (due to core decay heat, broken and intact loop SGs' sensible heat, and ECCS pump heat) exceeds the elevation head of a liquid column in one CLPS upside loop seal, then venting occurs through one or more intact loops in addition to the broken loop.
2. If the dynamic head resulting from the total steam flow (due to core decay heat, broken and intact loop SGs' sensible heat, and ECCS pump heat) does not exceed the elevation head of a liquid column in one CLPS upside loop seal, no venting occurs through one or more intact loops; the venting is through the broken loop.

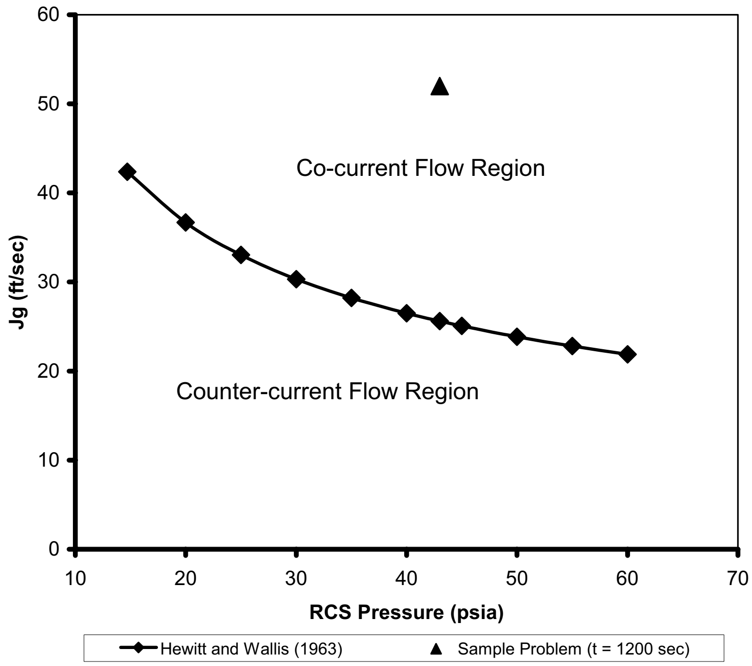
This mechanical energy balance was monitored based on the RELAP5 output for the sample problem, from 900 second to 2,000 seconds transient time. The decay heat assumed was based on 1.2 times the 1971 decay heat standard. The sensible heat rates in the SGs were relatively constant over this time interval. The friction and form losses were calculated for a single RCS loop based on the RELAP5 friction and form loss inputs for this flow path. The liquid/steam physical properties were taken to be constant and at saturation conditions corresponding to RCS pressure of 59 psia. An elevation head of the CLPS upside loop seal corresponding to a height of 8.357 ft was used.

The results of this venting calculation showed the cut-off time to be 1,910 seconds. The estimated transient time at which loop seal venting from the intact loops ceases, because of the onset of loop seal formation, is 1,910 seconds. Earlier loop seal formation results in conservative (higher) containment pressure prediction. Thus, the selection of 1,200 seconds as loop seal formation time in the sample problem is conservative.

FSAR Impact:

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

Figure 06.02.01-34-1—Comparison of Gas Flooding Limit Curve to Sample Problem Condition at Inlet to CL PS Upside Loop Seal



Question 06.02.01-42:

FSAR Section 5.2.3.4.3 documents that reflective metal insulation (RMI) will be used to cover the primary and secondary side system components. In Section 8.1.1 pf ANP-10299P the energy sources for mass and energy discharge are discussed but RMI is not included. During LBLOCA or MSLB, some of the RMI insulation is expected to be detached so that the sensible heat from this RMI would be directly discharged to the containment while remaining intact RMI would contribute to the heat added to the containment. Provide an assessment of the effect of the addition of RMI sensible heat to the containment peak pressure and temperature following a LBLOCA or MSLB.

Response to Question 06.02.01-42:

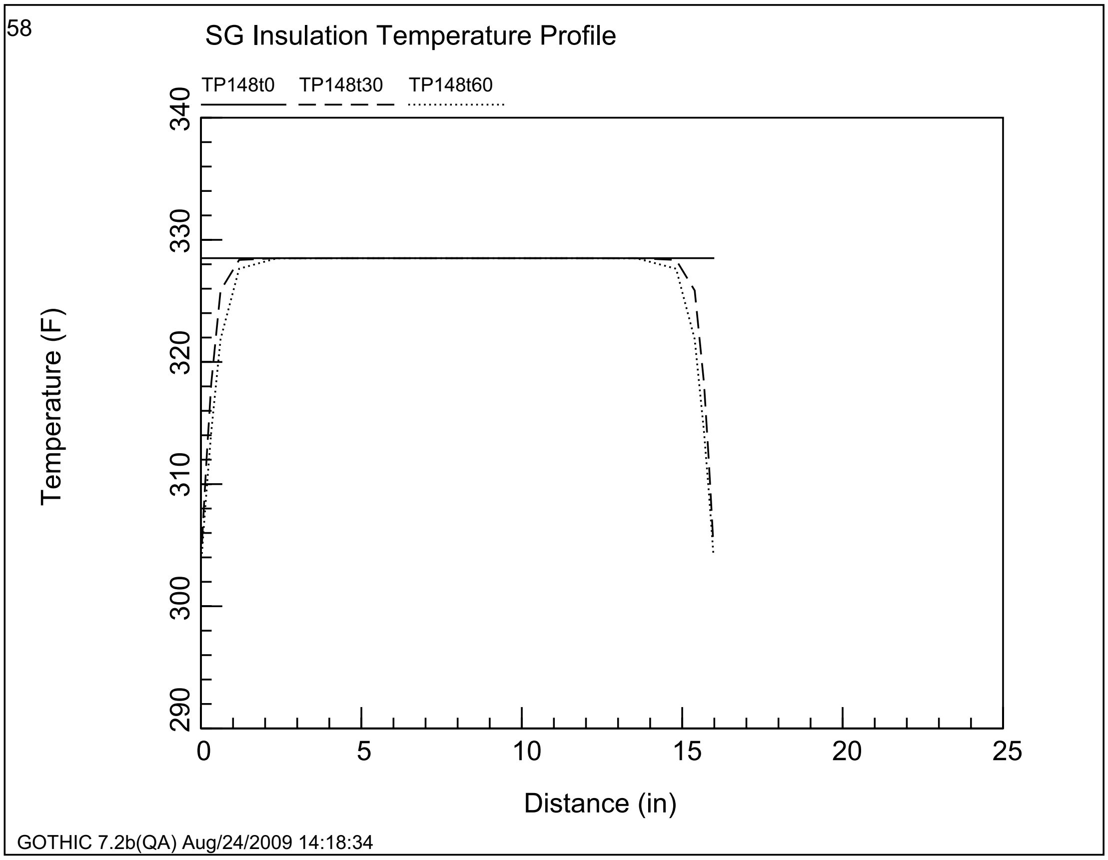
The total sensible heat held by the insulation is expected to be minor relative to the total energy released by a large break loss of coolant accident (LBLOCA) or a main steam line break (MSLB). The containment peak pressures and temperatures for a LBLOCA and MSLB occur within the first minute following the beginning of these events. The thermal lag in the insulation should allow only a fraction of the sensible heat to be released by the time of the peak containment pressure and temperature. Estimates of the insulation sensible heat have been calculated and were compared to the energy released by the LBLOCA and MSLB. For the cold-leg LBLOCA, a conservative estimate of the sensible heat available for release from the insulation resulted in a value less than 0.80 percent of the total energy released from the break up to the time of the short-term peak containment temperature and pressure. For the MSLB, the sensible heat available for release at the time of peak containment temperature and pressure was a negligible fraction of the total break energy release up to the time of the temperature and pressure peaks. For the long-term LBLOCA secondary containment pressure and temperature peaks, the insulation sensible heat is a negligible fraction of the total break energy release up to the time of these secondary peaks.

For the LBLOCA, a thermal-lag analysis for a typical sample of the reflective metallic insulation (RMI) was performed to demonstrate that a portion of the sensible heat remains in the insulation at the time of the short-term peak containment pressure and temperature. The thermal lag analysis was performed by adding an additional heat structure to a multi-volume GOTHIC model. The largest portion of the available insulation sensible heat is provided by the steam generator (SG) insulation. A typical sample of the SG insulation was used for the additional heat structure. Both sides of this heat structure were assumed to be in direct contact with the containment atmosphere. This LBLOCA GOTHIC case was then run to demonstrate that a significant fraction of the sensible heat remained in the insulation at the time of peak containment pressure and temperature. Figure 06.02.01-42-1 provides the insulation temperature profiles at 0, 30, and 60 seconds.

FSAR Impact:

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

Figure 06.02.01-42-1—Steam Generator Insulation Temperature Profiles



Question 06.02.01-44:

During the April 6 and 7 audit at Lynchburg, a discussion of the CONVECT system indicated that the pressurizer compartment is only compartment that remains closed post accident. No foils or dampers are provided to include this compartment in the post accident "one room" containment. Other components in this compartment are the pressurizer relief tank, CVCS tank and heat exchanger, core instrumentation, and at least 2 PARs. Confirm that this compartment remains isolated post accident. This question relates to ANP-10299P.

Response to Question 06.02.01-44:

Table 06.02.01-21-1 lists the doors in the Containment Building, including the location and any opening characteristics. Of the doors listed in Table 06.02.01-21-1, six are identified as being safety-related and provide venting between the pressurizer compartment and the accessible space in the containment. These doors are described in U.S. EPR FSAR Tier 2, Table 6.2.1-25 (as revised in the Response to RAI 209, Supplement 1, Question 06.02.01-14).

For the Response to RAI 209, Supplement 1, Question 06.02.01-14, analyses of breaks in the pressurizer compartment were completed to determine the number of doors that would have to open to vent the break effluent to the containment to maintain the containment pressure below the design limit. The results identified that at least five of the six safety-related doors in U.S. EPR FSAR Tier 2, Table 6.2.1-25 would need to operate and that the failure of one door was a less limiting single failure than eliminating one train of the emergency core cooling system (ECCS).

FSAR Impact:

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