ArevaEPRDCPEm Resource

From: DUNCAN Leslie E (AREVA NP INC) [Leslie.Duncan@areva.com]
Sent: Thursday, February 25, 2010 7:02 PM
To: Tesfaye, Getachew
Cc: DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); ROMINE Judy (AREVA NP INC); GUCWA Len T (EXT); BRYAN Martin (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 266, FSAR Ch 6, Supplement 4
Attachments: RAI 266 Supplement 4 Response US EPR DC.pdf

Getachew,

AREVA NP, Inc. provided a response to 1 of the 15 questions of RAI No. 266 on October 12, 2009. Supplement 1 response to RAI No. 266 was sent on November 4, 2009 to provide a response schedule for Questions 06.02.01.02-2, 06.02.01.02-3, 06.02.01.02-4. Supplement 2 response to RAI No. 266 was sent on December 10, 2009 to address 1 of the remaining questions. Supplement 3 response to RAI No. 266 was sent on December 18, 2009 to address 2 of the remaining questions. The attached file, “RAI 266 Supplement 4 Response US EPR DC.pdf,” provides a technically correct and complete response to 6 of the remaining 11 questions and a revised schedule for responding to Question 06.02.01.04-4.

The following table indicates the respective pages in the response document, “RAI 266 Supplement 4 Response US EPR DC.pdf,” that contain AREVA NP’s responses to the subject questions.

Question #	Start Page	End Page
RAI 266 — 06.02.01-48	2	3
RAI 266 — 06.02.01.03-2	4	19
RAI 266 — 06.02.01.03-3	20	20
RAI 266 — 06.02.01.04-2	21	29
RAI 266 — 06.02.01.04-3	30	30
RAI 266 — 06.02.01.04-4	31	31
RAI 266 — 06.02.01.04-5	32	37

The schedule for responding to the remaining RAI No. 266 questions is provided below. The schedule for responding to Question 06.02.01.04-4 was revised while the schedule for providing technically correct and complete responses to the remaining RAI No. 266 questions is unchanged.

Question #	Response Date
RAI 266 — 06.02.01.02-2	May 5, 2010
RAI 266 — 06.02.01.02-3	May 5, 2010
RAI 266 — 06.02.01.02-4	May 5, 2010
RAI 266 — 06.02.01.04-4	May 5, 2010
RAI 266 — 06.02.02-33	June 30, 2010

Sincerely,

Les Duncan
Licensing Engineer
AREVA NP Inc.
An AREVA and Siemens Company
Tel: (434) 832-2849
Leslie.Duncan@areva.com

From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, December 18, 2009 3:59 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. 266, FSAR Ch 6, Supplement 3

Getachew,

AREVA NP, Inc. provided a response to 1 of the 15 questions of RAI No. 266 on October 12, 2009. Supplement 1 response to RAI No. 266 was sent on November 4, 2009 to provide a response schedule for Questions 06.02.01.02-2, 06.02.01.02-3, 06.02.01.02-4. Supplement 2 response to RAI No. 266 was sent on December 10, 2009 to address 1 of the remaining questions. The attached file, "RAI 266 Supplement 3 Response US EPR DC.pdf," provides a technically correct and complete response to 2 of the remaining 13 questions.

The following table indicates the respective pages in the response document, "RAI 266 Supplement 3 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 266 — 06.02.01.04-6	2	2
RAI 266 — 06.02.01.04-7	3	3

The schedule for technically correct and complete responses to the remaining RAI No. 266 questions remains unchanged and is provided below.

Question #	Response Date
RAI 266 — 06.02.01-48	February 25, 2010
RAI 266 — 06.02.01.02-2	May 5, 2010
RAI 266 — 06.02.01.02-3	May 5, 2010
RAI 266 — 06.02.01.02-4	May 5, 2010
RAI 266 — 06.02.01.03-2	February 25, 2010
RAI 266 — 06.02.01.03-3	February 25, 2010
RAI 266 — 06.02.01.04-2	February 25, 2010
RAI 266 — 06.02.01.04-3	February 25, 2010
RAI 266 — 06.02.01.04-4	February 25, 2010
RAI 266 — 06.02.01.04-5	February 25, 2010
RAI 266 — 06.02.02-33	June 30, 2010

Sincerely,

Ronda Pederson
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From: Pederson Ronda M (AREVA NP INC)
Sent: Thursday, December 10, 2009 6:06 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. 266, FSAR Ch 6, Supplement 2

Getachew,

The response to RAI No. 266, Supplement 2, is submitted via AREVA NP Inc. letter, "Response Supplement 2 to U.S. EPR Design Certification Application RAI No. 266," NRC 09:123, dated December 10, 2009.

AREVA NP considers the information in the requested proprietary data files for the response to RAI No. 266 Question 06.02.01 - 47 submitted via that letter to be proprietary in their entirety and thus no non-proprietary version is provided. An affidavit to support withholding of information from public disclosure, per 10 CFR 2.390(b), is provided as an enclosure to that letter. The 2-page response document, alone, does not contain any proprietary information.

The following table indicates the respective page in the response document, "RAI 266 Supplement 2 Response US EPR DC.pdf," that contains AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 266 — 06.02.01 - 47	2	2

A response to Question 06.02.01.03 – 2 cannot be provided at this time. The schedule for technically correct and complete responses to the remaining RAI No. 266 questions has been changed and is provided below.

Question #	Response Date
RAI 266 — 06.02.01-48	February 25, 2010
RAI 266 — 06.02.01.02-2	May 5, 2010
RAI 266 — 06.02.01.02-3	May 5, 2010
RAI 266 — 06.02.01.02-4	May 5, 2010
RAI 266 — 06.02.01.03-2	February 25, 2010
RAI 266 — 06.02.01.03-3	February 25, 2010
RAI 266 — 06.02.01.04-2	February 25, 2010
RAI 266 — 06.02.01.04-3	February 25, 2010
RAI 266 — 06.02.01.04-4	February 25, 2010
RAI 266 — 06.02.01.04-5	February 25, 2010
RAI 266 — 06.02.01.04-6	December 18, 2009
RAI 266 — 06.02.01.04-7	December 18, 2009
RAI 266 — 06.02.02-33	June 30, 2010

Sincerely,

Ronda Pederson

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From: WELLS Russell D (AREVA NP INC)
Sent: Wednesday, November 04, 2009 3:20 PM
To: 'Getachew Tesfaye'
Cc: Pederson Ronda M (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. 266, FSAR Ch 6, Supplement 1
Getachew,

AREVA NP, Inc. provided a response to 1 of the 15 questions of RAI No. 266 on October 12, 2009. As indicated in our response, a schedule for the response to Questions 06.02.01.02-2, 06.02.01.02-3, 06.02.01.02-4, would be provided by November 5, 2009. Accordingly, the schedule for the response to these questions is provided in the attached file, RAI 266 Supplement 1 Response US EPR DC.pdf" and reflected in the below table.

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

Question #	Response Date
RAI 266 — 06.02.01-47	December 10, 2009
RAI 266 — 06.02.01-48	February 25, 2010
RAI 266 — 06.02.01.02-2	May 5, 2010
RAI 266 — 06.02.01.02-3	May 5, 2010
RAI 266 — 06.02.01.02-4	May 5, 2010
RAI 266 — 06.02.01.03-2	December 10, 2009
RAI 266 — 06.02.01.03-3	February 25, 2010
RAI 266 — 06.02.01.04-2	February 25, 2010
RAI 266 — 06.02.01.04-3	February 25, 2010
RAI 266 — 06.02.01.04-4	February 25, 2010
RAI 266 — 06.02.01.04-5	February 25, 2010
RAI 266 — 06.02.01.04-6	December 18, 2009
RAI 266 — 06.02.01.04-7	December 18, 2009
RAI 266 — 06.02.02-33	June 30, 2010

Sincerely,

(Russ Wells on behalf of)
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From: Pederson Ronda M (AREVA NP INC)
Sent: Monday, October 12, 2009 6:38 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. 266, FSAR Ch. 6

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 266 Response US EPR DC.pdf" provides technically correct and complete responses to 1 of the 15 questions.

The following table indicates the respective pages in the response document, "RAI 266 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 266 — 06.02.01-47	2	2
RAI 266 — 06.02.01-48	3	3
RAI 266 — 06.02.01.02-2	4	4
RAI 266 — 06.02.01.02-3	5	5
RAI 266 — 06.02.01.02-4	6	6
RAI 266 — 06.02.01.03-2	7	7
RAI 266 — 06.02.01.03-3	8	8
RAI 266 — 06.02.01.04-2	9	9
RAI 266 — 06.02.01.04-3	10	10
RAI 266 — 06.02.01.04-4	11	11
RAI 266 — 06.02.01.04-5	12	12
RAI 266 — 06.02.01.04-6	13	13
RAI 266 — 06.02.01.04-7	14	14
RAI 266 — 06.02.02-33	15	15
RAI 266 — 06.02.02-34	16	16

A complete answer is not provided for 14 of the 15 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 266 — 06.02.01-47	December 10, 2009
RAI 266 — 06.02.01-48	February 25, 2010
RAI 266 — 06.02.01.02-2	Schedule to be provided by November 5, 2009
RAI 266 — 06.02.01.02-3	Schedule to be provided by November 5, 2009
RAI 266 — 06.02.01.02-4	Schedule to be provided by November 5, 2009
RAI 266 — 06.02.01.03-2	December 10, 2009
RAI 266 — 06.02.01.03-3	February 25, 2010
RAI 266 — 06.02.01.04-2	February 25, 2010
RAI 266 — 06.02.01.04-3	February 25, 2010
RAI 266 — 06.02.01.04-4	February 25, 2010
RAI 266 — 06.02.01.04-5	February 25, 2010
RAI 266 — 06.02.01.04-6	December 18, 2009
RAI 266 — 06.02.01.04-7	December 18, 2009
RAI 266 — 06.02.02-33	June 30, 2010

Sincerely,

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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Thursday, September 10, 2009 9:10 AM
To: ZZ-DL-A-USEPR-DL
Cc: Jensen, Walton; Jackson, Christopher; Snodderly, Michael; Carneal, Jason; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 266(3408,3443,3444,3445,3446), FSAR Ch. 6
Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on August 3, 2009, and discussed with your staff on August 13, 2009. RAI Questions 06.02.01-47, 06.02.01-48, 06.02.01.04-2, 06.02.01.04-5, and 06.02.01.04-7 were revised as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

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

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 1178

Mail Envelope Properties (F322AA625A7A7443A9C390B0567503A1019C6CED)

Subject: Response to U.S. EPR Design Certification Application RAI No. 266, FSAR Ch 6, Supplement 4
Sent Date: 2/25/2010 7:01:46 PM
Received Date: 2/25/2010 7:01:50 PM
From: DUNCAN Leslie E (AREVA NP INC)

Created By: Leslie.Duncan@areva.com

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RAI 266 Supplement 4 Response US EPR DC.pdf		308644

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

Response to

Request for Additional Information No. 266, Supplement 4

9/10/2009

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 06.02.01 - Containment Functional Design
SRP Section: 06.02.01.02 - Sub-compartment Analysis
SRP Section: 06.02.01.03 - Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)
SRP Section: 06.02.01.04 - Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures
SRP Section: 06.02.02 - Containment Heat Removal Systems

Application Section: FSAR Chapter 6

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

Question 06.02.01-48:

This question relates to negative containment pressure analysis in FSAR Section 6.2.1.1.1. FSAR Section 6.2.1.1.1 lists 5 potential events which cause negative pressure across the containment wall: 1. a sudden containment temperature reduction, 2. removal of IRWST inventory, 3. HVAC pulldown of containment pressure, 4. post-accident cooldown, and 5. post-severe accident cooldown. The greatest amount of pressure reduction, calculated for the sudden containment temperature reduction, was calculated to be 2.92 psi. The design negative pressure is 3.0 psi. Therefore the maximum temperature reduction was concluded to be within the design. For the sudden containment temperature reduction, drop from 122 ^{O}F to 59 ^{O}F was assumed. The minimum allowed containment temperature was stated to be 59 ^{O}F.

AREVA did not explain how the minimum allowed containment temperature was determined or explain the actions which would be taken to prevent the containment temperature from decreasing below 59 ^{O}F. AREVA did not explain the basis of assuming an initial relative humidity of 70% rather than 100% which would be more conservative for this calculation. Since the reduction in containment pressure might take place over a period of time, the operational staff might have time to mitigate the event. AREVA did not provide any mitigation procedures or evaluate their effectiveness. The staff requires additional information in order to close this issue.

AREVA did not describe the event that can result in the sudden temperature reduction from 122°F to 59°F. Describe the event. Can this event be prevented from occurring by interlocks and/or administrative controls? Provide justification for the assumed initial relative humidity of 70%. Furthermore, provide the mitigation procedures available to the operators to mitigate containment negative pressure events.

In response to RAI #104, 14.03-1a.1 AREVA stated that the screening approach using discipline checklists based on SRP 14.3 guidance did not identify safety-significant design features for a negative pressure inside containment. AREVA further stated that no design basis events described in the U.S. EPR FSAR result in a negative pressure inside containment. Therefore, ITAAC for a negative pressure inside containment are not included in U.S. EPR FSAR Tier 1. Provide justification for these statements in view of the statements made in FSAR Section 6.2.1.1.1 which discusses design basis events which might cause a negative containment pressure.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01-48:

The U.S. EPR design does not have an automatic containment spray system or containment air coolers for design basis accident (DBA) mitigation. Thus, the U.S. EPR design is not susceptible to inadvertent actuation of those systems, or the potential for damage because of the rapid reduction of the containment internal pressure that would result from such an inadvertent actuation. The severe accident heat removal system (SAHRS) described in U.S. EPR FSAR Tier 2, Section 19.2, includes a manually actuated containment spray system dedicated to severe accident mitigation. This system is not used for DBAs. Because the SAHRS is manually aligned and manually actuated, it is not subject to a single failure that could cause inadvertent actuation of containment spray, thereby eliminating the need to analyze for this event. The five events that are listed in the U.S. EPR FSAR Tier 2, Section 6.2.1.1.1 are not credible events and therefore were removed from this section of the FSAR. The changes were included in the FSAR markups that were provided with the Response to RAI 209, Question 06.02.01-14.

FSAR Impact:

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

Question 06.02.01.03-2:

This question relates to mass and energy release analysis for postulated loss-of-coolant accidents in accordance with SRP Section 6.2.1.3. In RAI #82 6.02.01.03-p the staff requested that AREVA provide justification for decreasing the core decay heat multiplier from 1.2 to 1.1 in the mass and energy release calculations for the long term post reflood phase. In the containment mass and energy release calculations for the EPR, AREVA uses the 1971 ANS standard with a 1.20 multiplier for the first 1000 seconds. After that a multiplier of 1.10 is used. AREVA presented a comparison of the decay heat from the 1971 ANS standard using the 20 and 10 percent multipliers with the 1979 ANS standard using a 2σ multiplier. The staff has recently accepted containment mass and energy release calculations using the1979 ANS standard with a 2σ multiplier. The 1979 ANS standard has a more extensive data base than the 1971 standard and is considered to be the more accurate. AREVA concluded that the core decay heat used in the mass and energy calculations for the EPR containment analysis is conservative compared with the 1979 ANS standard with a 2σ multiplier and is therefore acceptable. AREVA extended the comparison to 10,000 seconds. The staff obtained a similar result for the first 10,000 seconds. At times greater than 20,000 the staff found the 1979 ANS standard with a 2σ multiplier to be the more conservative. In order to resolve this issue, demonstrate that the decay heat model used in the containment mass and energy calculations is conservative. In application of the 1979 ANS decay heat standard or later standards justify the input assumptions selected. These include assumptions for actinide decay, actinide production factor, multiplier to account for neutron capture activation, fissions per initial fissile atom and power history. The effect of these assumptions is discussed in NRC IN 96-39.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.03-2:

The following response is provided in two parts. The first part (part (a)) was provided to the NRC staff during the audit held on July 14 and 15, 2009. This information addresses the question: "…demonstrate that the decay heat model used in the containment mass and energy calculations is conservative." The second part (part (b)) addresses the question: "In application of the 1979 ANS decay heat standard or later standards justify the input assumptions selected."

Part (a)

The 1994 ANS standard (Reference 2) has a more extensive database than the 1979 standard (Reference 1) and is considered to be more accurate. The 1994 ANS standard also has a higher decay heat than the 1979 standard, as indicated in Table 06.02.01.03-2-1. The 1994 ANS standard is conservative with respect to the 1979 standard for the decay heat energy released during a loss-of-coolant accident (LOCA). The 1994 ANS standard is therefore used for comparison to the U.S. EPR decay heat energy released to the containment, which is based on the 1971 standard (Reference 3).

Because the industry generally accepts the position that the 1971 decay heat standard has conservative biases (too much decay heat energy), the first step in addressing this question is to compare the 1971 and the 1994 standards on a best-estimate basis. That is, no uncertainties are applied to the results, ORIGEN is used to assess the actinide heavy elements, and equilibrium U.S. EPR fuel cycles are used to assess:

- Isotopic fission energy production rates.
- Ratio of U-238 capture rate to the total fission rate for actinide production (the *R* value).
- Transmutations of fission products due to neutron capture (the "ψ" factor for $G(t, T)$ in References 1 and 2).

Figure 06.02.01.03-2-1 shows the comparison. The 1971 standard is conservative with decay heat values that are higher than those associated with the 1994 standard.

The actinide decay heat rates for both the 1971 and the 1994 standards are identical. The differences shown in Figure 06.02.01.03-2-1 are caused by the relative isotopic fission energy production rates in the 1994 standard. These rates reflect an equilibrium U.S. EPR fuel cycle design.

The equilibrium design has an average beginning of cycle burnup on the order of 20,000 MWD/MTU. At this burnup level, a decay heat model would show that ten percent of the fission energy is coming from fast fissions (U-238); thirty percent from Pu, with fifteen percent of the Pu being Pu-241; and the remainder of the fission energy, sixty percent, coming from U-235.

Figure 06.02.01.03-2-1 shows the conservative nature of the 1971 standard in comparison to the 1994 standard and confirms the industry assessment of decay heat (i.e., the 1971 standard has conservative biases). To further address this question, the comparison needs to be appropriate for LOCA analyses. Thereby, the "2σ" uncertainties were applied to the Figure 06.02.01.03-2-1 values. These uncertainties reflect a 95 percent level of confidence that at least 95 percent of the data will be within the bounds of the mean decay heat value plus the product of the confidence factor ($K = \pm 2$) and standard deviation (95 / 95).

Figure 06.02.01.03-2-2 reflects the application of 95 / 95 uncertainty values to the values in Figure 06.02.01.03-2-1.

The procedures for applying the uncertainties are consistent with the respective standards. For the 1971 standard, the 95 / 95 uncertainty values (1.20 / 1.10) are only applied to the fission product decay heat. The U-239 and Np-239 actinides are biased with a high production factor representative of burnups in the 40,000 to 60,000 MWD/MTU range; however, they have no 95 / 95 uncertainty value applied. For the 1994 standard, not only are the 95 / 95 uncertainty values applied as required, but additional uncertainty values are applied for the actinide production factor, *R*, and the transmutation of fission products due to neutron capture, $G_{MAX}(t, T)$ (actinides continue to be biased with a high production factor).

Figure 06.02.01.03-2-2 shows that the 1971 standard is conservative with decay heat values that are higher than those associated with the 1994 standard. The data in Figure 06.02.01.03-2-2 shows more decay heat energy before 1,000 seconds than Figure 06.02.01.03-2-1 shows. However, after 1,000 seconds, the Figure 06.02.01.03-2-2 data indicate that not only does the 1971 LOCA decay heat data trend toward the 1994 data, but the 1994 values are slightly higher in the neighborhood of 50,000 to 100,000 seconds.

The 95 / 95 uncertainty factors are the reason for the differences between Figures 06.02.01.03-2-1 and 06.02.01.03-2-2. These uncertainty factor differences are shown in Figure 06.02.01.03-2-3.

Figure 06.02.01.03-2-2 shows that there are some instances where the data associated with the 1994 standard has higher decay heat than the 1971 standard; however, the differences are too small to be important. The limiting conditions associated with energy release to the containment following a LOCA are related to the integral of the data in Figure 06.02.01.03-2-2. The figure indicates that the data from the 1971 standard are more conservative.

The data in Figures 06.02.01.03-2-1 and 06.02.01.03-2-2 for the equilibrium U.S. EPR cycle do not support the NRC staff observation stated in this question. To address the question, a decay heat model for the beginning-of-life (BOL) was needed. At the BOL, which is the beginning of the first cycle, there is no burnup. With no burnup, a conservative 1994 decay heat model would have isotopic fission energy production rates of ten percent coming from fast fission (U-238), and the remaining ninety percent coming from U-235. Figure 06.02.01.03-2-4 provides a comparison of decay heat between the 1971 best-estimate model and the BOL 1994 model.

The data for the 1971 standard and that for the BOL 1994 standard are essentially identical, with the exception of the difference that occurred due to an error identified by the developer of the 1971 data.

Applying uncertainties to the 1994 BOL model that are consistent with those shown in Figure 06.02.01.03-2-3 produced the behavior that the NRC staff questioned. This question notes that at 10,000 seconds the 1971 standard had decay heat values that are slightly higher than those in the 1979 standard (and later ones). However, at times greater than 20,000 seconds, the NRC staff found that the data for the 1971 standard was not conservative in comparison to the data for the later standards. To resolve this issue it is necessary to demonstrate that the decay heat model used in the containment mass and energy calculations is conservative.

Figure 06.02.01.03-2-5 compares the 1971 model with the uncertainties (1.20 / 1.10) to the 1994 model with positive – two-sided 95 / 95 uncertainties. These uncertainties were determined in exactly the same manner as described previously for Figure 06.02.01.03-2-2. In addition to the uncertainties required by the 1994 standard, there are additional uncertainties for (a) actinides and (b) fission product transmutation due to neutron capture. Thus, like the equilibrium model, the BOL model shown in Figure 06.02.01.03-2-5 contains more uncertainties than recommended by the standard.

While the two sets of data in Figure 06.02.01.03-2-5 look identical to the two sets in Figure 06.02.01.03-2-2, the detail between 1,000 and 100,000 seconds indicates some difference. Figure 06.02.01.03-2-6 contains exactly the same data as Figure 06.02.01.03-2-5. However, the Figure 06.02.01.03-2-6 time scale has been reduced to clearly observe the non-conservative data from the 1971 standard versus the data from the 1994 standard.

At 10,000 seconds the 1971 standard has higher decay heat values and is conservative compared to the 1994 standard. Beyond 30,000 seconds, however, the 1971 standard has lower decay heat values. The lower values are questionable with respect to conservative mass and energy releases to the containment following a LOCA (See Figure 06.02.01.03-2-6 for the trend).

It is not the standards' decay heat data at each instant of time that is directly applicable to mass and energy releases to the containment following a LOCA. It is the integral of the standards' decay heat energy as a function of time that produces the limiting conditions in the containment.

Consequently, the data curves shown in Figure 06.02.01.03-2-5 need to be integrated over time to determine if the 1971 standard has an overall lower release of decay heat energy.

The functional form of the decay heat curves shown in Figure 06.02.01.03-2-5 is a sum of exponential terms that are equivalent to multiple pseudo-isotopes. If the decay heat and time variables are transformed to natural logarithmic functions, then as shown in the 1971 standard, a single constant is sufficient to represent the data over increments of 1,000 seconds or more. With a single constant representing discrete time increments, the decay heat curves may be analytically integrated using multiple unique increments and a piece-wise summation over the increments. Figure 06.02.01.03-2-7 shows the integrated – average data from the 1971 and 1994 standards for the period between 1,000 and 100,000 seconds.

Comparing Figures 06.02.01.03-2-6 and 06.02.01.03-2-7 shows that while specific data points in the 1971 standard are less than those in the 1994 standard during the time periods between 30,000 and 100,000 seconds, the 1971 integrated – average decay heat energy released following a LOCA is greater than that for the 1994 standard.

As shown in Figure 06.02.01.03-2-6, it is possible to create BOL conditions where the discrete decay heat data at times greater than 20,000 seconds in the 1971 standard are not as high as values from other standards with 95 / 95 uncertainties. However, considering the decay heat energy released to the containment following a LOCA, the total energy, integrated from the 1971 data, is higher and thereby more conservative than the total energy from later standards. The integrated – average comparison is shown in Figure 06.02.01.03-2-7. Moreover, after BOL, which begins around the middle of the first cycle and extends to the equilibrium cycle, the integrated 1971 data gives a higher total energy than any data from later standards and is thereby more conservative.

Part (b)

In the second part of this question the NRC staff requested that the input assumptions used in the evaluation be explained. The assumptions are to include:

- Power history.
- Neutron capture in fission products.
- Fissions per initial fission atom.
- Actinide decay.
- Actinide production factor.

<u>Power History</u>: The total decay heat from both the fission products and the actinides are a function of the power history. Thus, the explanation of the input begins with the power history. The data for the 1979 and later standards rely heavily on instantaneous fission pulses. The pulse data is integrated over an operation period T (FP) to convert the data to a useful form for long-term reactor operation. The coefficients associated with the integrated expressions, F (t, T) were used to compute the decay heat. The standards recommend that infinite operation {T (FP) = 1×10^{13} seconds} be considered for safety-related decay heat evaluations.

Infinite operation was used for the decay heat energy released to the containment from the 1971 standard. Consequently, it was also used for the comparisons discussed above. While the standards define the decay heat energy in terms of millions of electron volts (MeV), the MeV values are related to fractional values for each respective isotope. The isotopic fission energy is consistent with the decay heat data and core analyses. The respective weight of each isotope contributing to the fission energy is determined from the power history for the U.S. EPR fuel cycle analyses.

The power history associated with the actinides is the same as that associated with the fission products. The difference is the bias introduced into the actinide decay heat to provide a conservative burnup history for actinide production. The burnup history for an equilibrium cycle would have approximately three batches of fuel with incremental burnups on the order of 20,000 MWD/MTU. Thus, the core would have a beginning of cycle burnup associated with 0 (batch 1), 20,000 (batch 2), and 40,000 (batch 3) MWD/MTU, giving an average of 20,000 MWD/MTU. At the end of cycle the burnup associated with each batch would be 20,000 (batch 1), 40,000 (batch 2), and 60,000 (batch 3) MWD/MTU, giving an average of 40,000 MWD/MTU. The actinide bias for the equilibrium cycle represents a burnup history associated with 40,000 to 60,000 MWD/MTU. The corresponding bias for BOL represents a burnup history associated with 20,000 to 40,000 MWD/MTU.

Neutron Capture in Fission Products $G(t, T)$: The fission product decay heat data in the 1979 Standard relies heavily on instantaneous fission pulses. This is different from the data relied on for the 1971–1973 Standard (Reference 3). The 1971–1973 data applied summation calculations to long-term (thousands of seconds) calorimetric measurements that were supported by gamma-ray measurements. The calorimetric techniques had the advantage of representing the total decay heat from burned fuel. The disadvantage of the techniques was that it was difficult to obtain accurate data for periods less than 1,000 seconds.

To increase the accuracy of the data in the time intervals important to LOCA, the measurement techniques were changed. The newer data was developed from instantaneous fission pulses. The gamma ray measurements from the pulses of the fuel material could be obtained in time periods of seconds or less. The beta measurements required a longer period but could be analytically evaluated to be consistent with the gamma ray measurements. The precision associated with the random error in the new data was small enough to be useful for LOCA safety analyses.

While the pulse data improved the precision of the decay heat over the short-term, there was the disadvantage that the total decay heat from burned fuel was incomplete when long-term operational periods were considered. To include long-term operational periods $\{T(FP)\}$, the pulse data was integrated over time to include the burnup associated with reload cycles. However, the integral could not represent the neutron capture in fission products $\{G(t, T)\}$ during operation. To include the effects of capture reactions, summation calculations were performed. These calculations represent the effects of the pulse data with no fission product transmutations, and also represent transmutations due to capture reactions. The fission product transmutations produced more decay heat than the pulse without transmutations. To include the additional decay heat, the variable $G(t, T)$ was applied to the standards (References 1 and 2). This variable is composed of values that are greater than 1.0. $G(t, T)$ is multiplied by the standards' long-term fission product decay heat to represent fission product transmutations resulting from capture reactions.

The standards' expression for $G(t, T)$ is:

$$G(t, T) = 1.0 + (3.24 \times 10^{-6} + 5.23 \times 10^{-10} \times t) \times T^{0.4} \times \psi \qquad (1)$$

The time of decay is t; the time of operation is $T(G)$, and ψ is a variable representing different rates of transmutation (fissions per initial fission atom).

$$0.0 \leq T(G, \text{seconds}) \leq 1.2614 \times 10^{8} \qquad 1.0 \leq \psi \leq 3.0$$

The $G(t, T)$ input for the neutron capture in fission products was developed from tabular values for $G_{MAX}(t, T)$ specified in the standards. $G_{MAX}(t, T)$ is represented by Equation (1) with $T(G) = 1.2614 \times 10^{8}$ seconds and $\psi = 3.0$. Equation (1) however produces slightly different values than those in the standards' tables. To have consistency between Equation (1) and the table, Equation (1) is used in conjunction with the standards' tabular values. The parenthetical expression is the solution to the time dependent decay function when $T(G)$ and ψ are maximum values. The parenthetical expression shows that $G(t, T)$ is a linear function of decay time, t. Therefore, points in time that are not in the standards' tabular values are accurately determined by linear interpolation of the values in the tables.

Evaluation of ψ (Fissions per Initial Fission Atom): The standards provide a means of calculating ψ by considering the fissions per initial fission atom. The standards' references to ψ, however, discuss ORIGEN summation calculations with and without fission product transmutations as being the basis for the ψ values. Considering the ORIGEN methods for determining $G(t, T)$ and ψ, the values of ψ can be evaluated using the (Appendix K) data in the 1971–1973 Standard (Reference 3). The (Appendix K) decay heat values in the 1971–1973 Standard include the effects of $G(t, T)$ for long-term burnup of the fuel material. This evaluation of ψ is shown by the results in Figure 06.02.01.03-2-8.

The BOL model is based on fission energies representative of 10 percent fast fission (U-238) and 90 percent U-235. The fractions of isotopic fission rates and operational time can affect comparisons of calculated results to the measured data. Nonetheless, it is the ψ value in the $G(t, T)$ component of the model that is being sought. The Figure 06.02.01.03-2-8 comparison of model results to the 1971–1973 Standard's data indicates that the model results are within the measurement uncertainty, which is represented by a relative value of $\pm$ 0.05 (1.05). Demonstrating that the model results are within the measurement uncertainty indicates that $\psi = 1.0$ produces accurate results.

Actinide Decay: There are two actinides, U-239 and Np-239, which produce the majority of the actinide decay heat. The original actinide model presented in the 1971–1973 Standard was based on discharge burnups of 30,000 MWD/MTU. At this low burnup level it was only necessary to consider U-239 and Np-239. The decay heat from the other actinides was negligible. Moreover, there was no uncertainty applied to the respective actinides. The U.S. EPR fuel can accommodate burnups of 60,000 MWD/MTU. At these high burnup levels it is prudent to consider all actinides in the decay heat model even though the 1979 and later standards have not added this as a requirement. In addition, the fission product uncertainties are no longer adequate to cover the actinide uncertainties. AREVA has applied unique actinide uncertainties to the 1979 and later standards. The models include ORIGEN analyses to determine total actinide decay heat.

The production of the total actinide decay heat energy is based on the results of the decay rates from all the actinides in ORIGEN. The decay steps are sufficiently fine to allow linear interpolation of points represented in the figures that were not included in the ORIGEN data points.

<u>Actinide Production Factor</u>: The actinide production factor, *R*, in the standards' equations (References 1 and 2) can be represented by the ratio of captures in U-238 to the total fissions. This ratio, *R*, is edited from the U.S. EPR fuel design analyses. The ORIGEN results are applied to the equations in the standards to assess the conservative bias.

The best-estimate *R* value is 0.548 at the BOL. However, to provide appropriate decay heat uncertainties for safety analyses, unique uncertainties have been applied to the U-239 and Np-239 decay heat, as well as the total decay heat from the sum of all other actinides (ξ). The positive 95 / 95 uncertainty factors for the actinides are:

$$\text{U-239 (95 / 95 Factor)} = 1.20$$

$$\text{Np-239 (95 / 95 Factor)} = 1.22$$

$$\sum_{Actinides} \xi_{Actinides} \text{ (95 / 95 Factor)} = 1.40$$

These 95 / 95 uncertainty factors are applied to the actinide decay heat resulting in a conservatively biased R value of 0.668 at BOL.

The best-estimate R value is 0.761 for the model of the equilibrium cycle. With the 95 / 95 uncertainty factors noted above applied to the actinide decay heat, the conservatively biased R value is 0.940 for the model of the equilibrium cycle.

References:

1. ANSI/ANS-5.1, "Decay Heat Power in Light Water Reactors", American Nuclear Society, 1979.
2. ANSI/ANS-5.1, "Decay Heat Power in Light Water Reactors", American Nuclear Society, 1994.
3. ANS 5.1, American Nuclear Society Proposed Standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", American Nuclear Society, October, 1971, Revised October, 1973.

FSAR Impact:

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

Table 06.02.01.03-2-1—Comparison of 1979 Versus 1994 Standard

Time (Seconds)	1979 (U-235)	1994 (U-235)
1	.065159	.065010
10	.049046	.049090
100	.031918	.031938
1,000	.019533	.019558
10,000	.009811	.009831
100,000	.005009	.005029
	1979 (Pu)	**1994 (85% Pu-239) (15% Pu-241)**
1	.054480	.054846
10	.042598	.042891
100	.029376	.029240
1,000	.018236	.018070
10,000	.009023	.009025
100,000	.004939	.005003
	1979 (U-238)	**1994 (U-238)**
1	.087027	.085897
10	.060197	.056855
100	.034157	.034673
1,000	.019420	.020103
10,000	.009499	.009725
100,000	.004921	.004968

Figure 06.02.01.03-2-1—Decay Heat Comparison

(Best-Estimate – No Uncertainties)
(Equilibrium = 10 % U-238 – 30 % Pu – 60 % U-235)

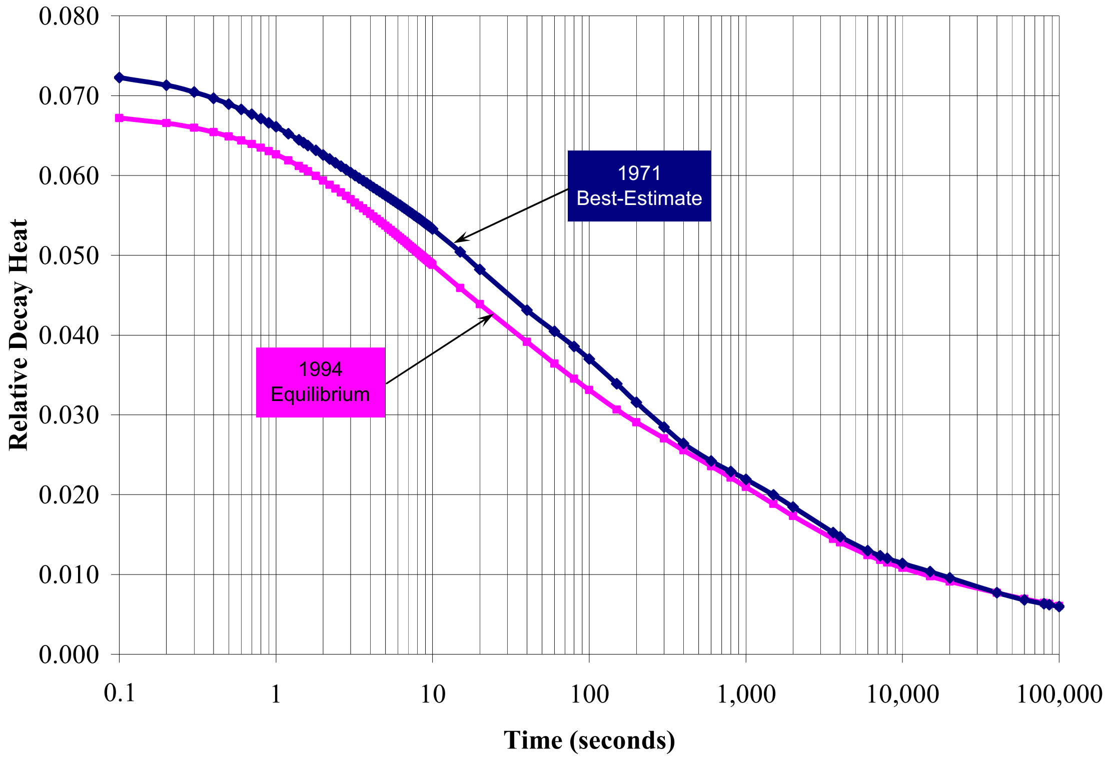


Figure 06.02.01.03-2-2—Decay Heat Comparison

(With 95 / 95 Uncertainties)

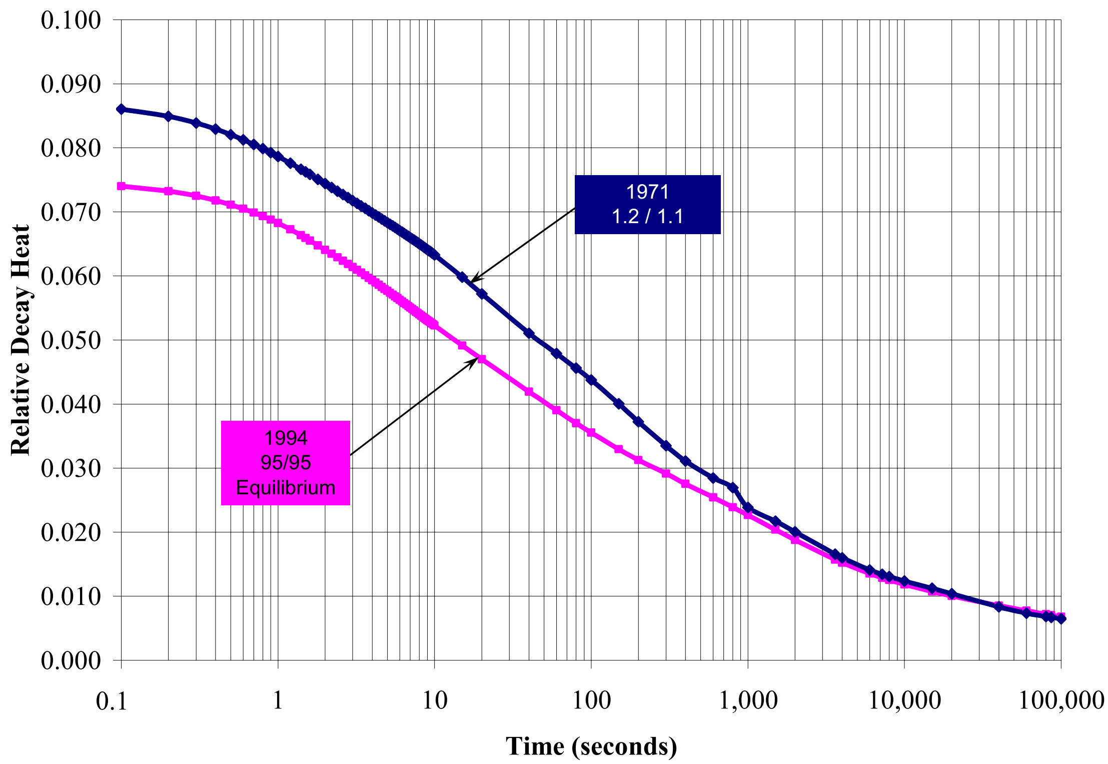


Figure 06.02.01.03-2-3—Uncertainty Factor Comparison
(Positive Two-Sided 95 / 95 Values)

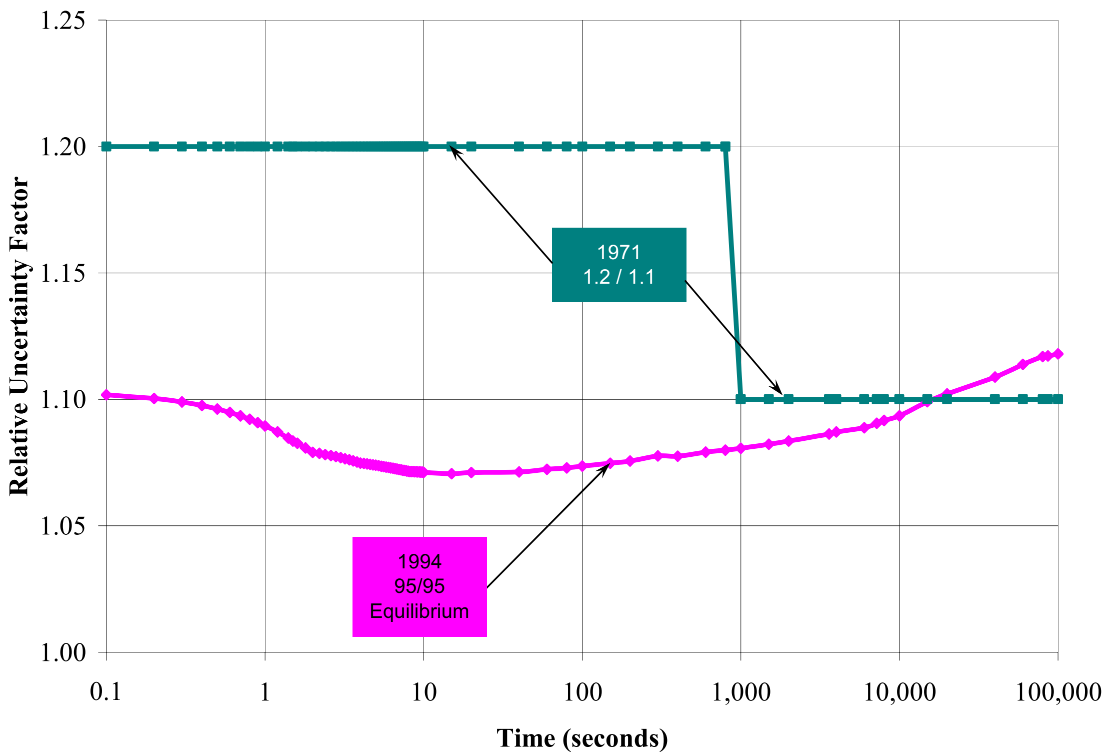


Figure 06.02.01.03-2-4—Decay Heat Comparison

(Best-Estimate – No Uncertainties)
(BOL = 10 % U-238 – 0 % Pu – 90 % U-235)

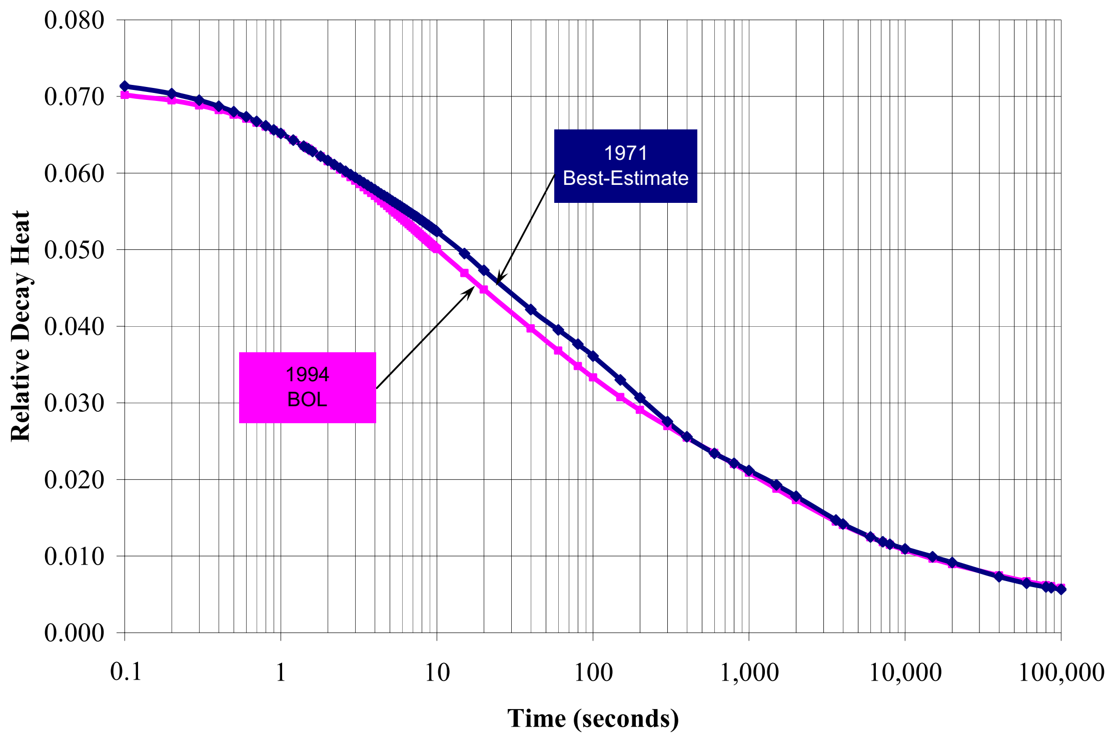


Figure 06.02.01.03-2-5—Decay Heat Comparison
(With 95 / 95 Uncertainties)

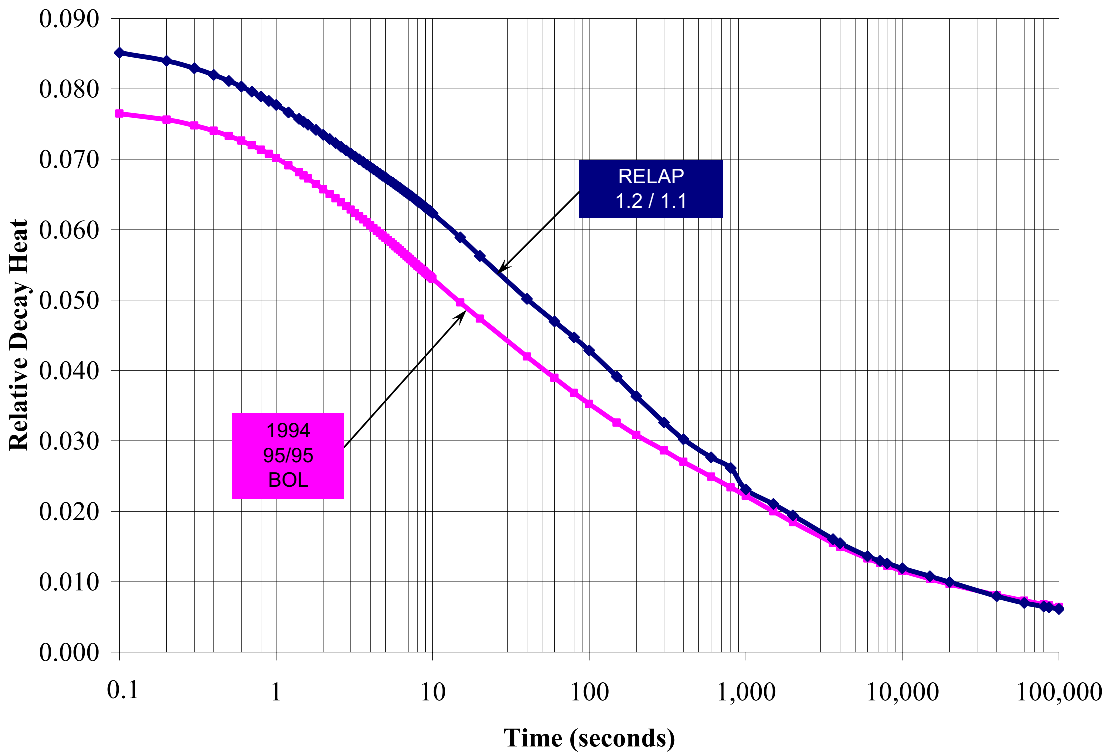


Figure 06.02.01.03-2-6—Decay Heat Comparison

**(With 95 / 95 Uncertainties)
(1,000 to 100,000 Seconds)**

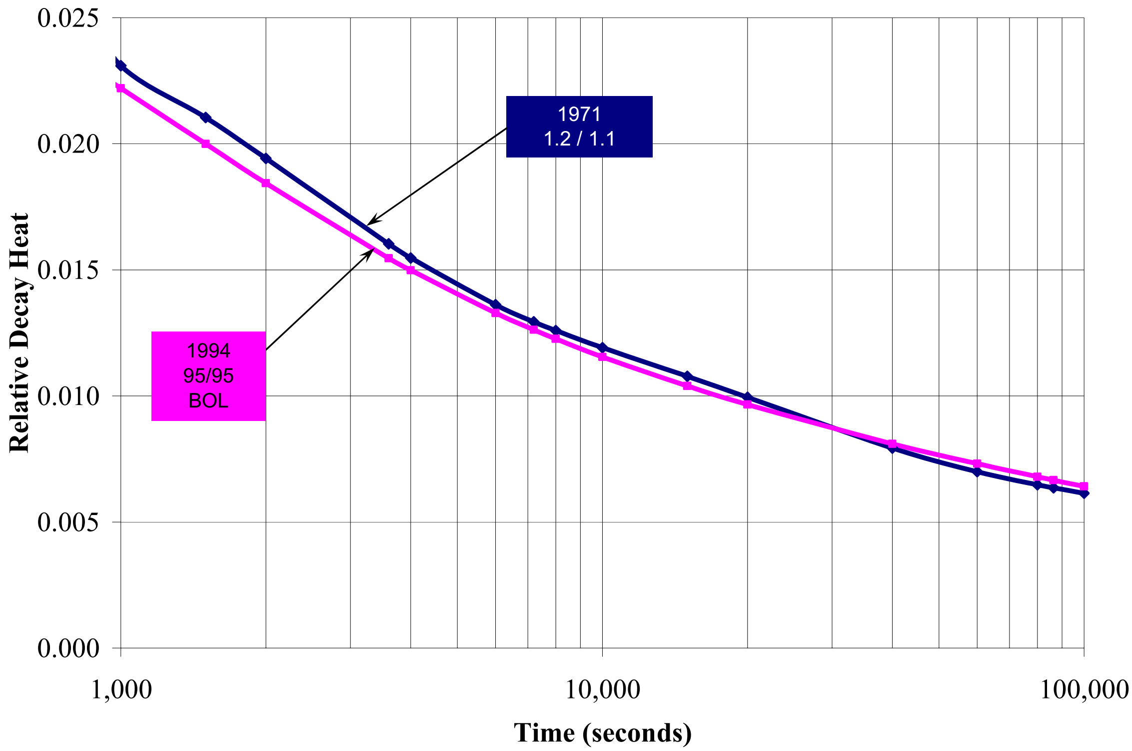


Figure 06.02.01.03-2-7—Integrated–Average Decay Heat Comparison

(With 95 / 95 Uncertainties)
(1,000 to 100,000 Seconds)

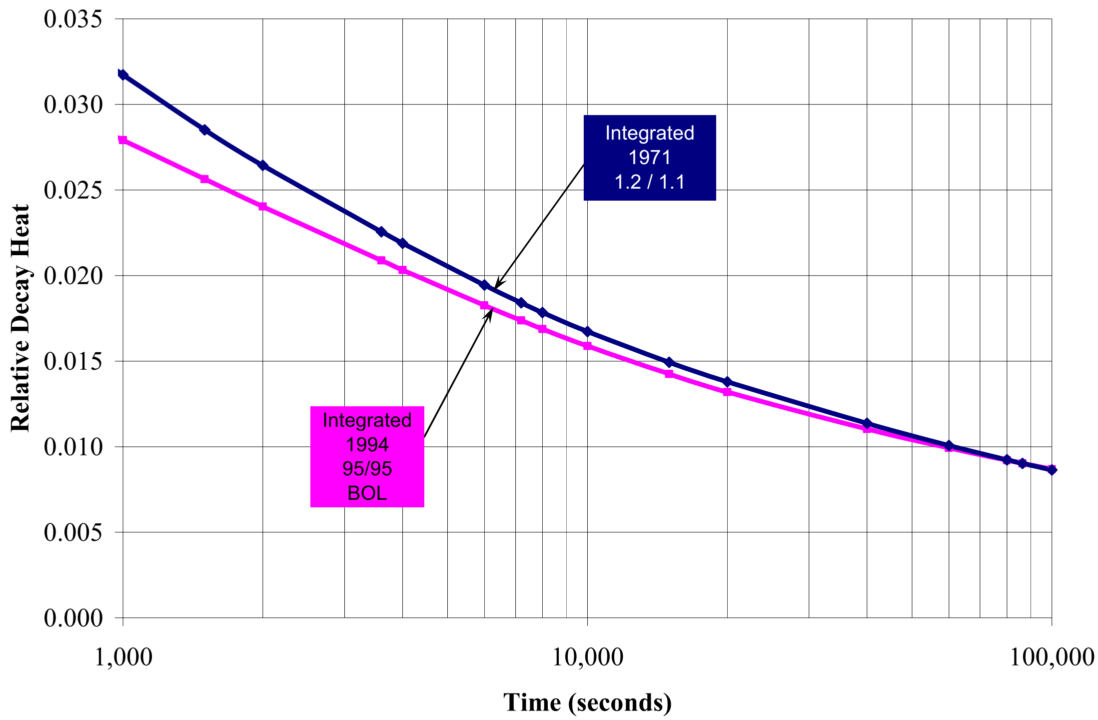
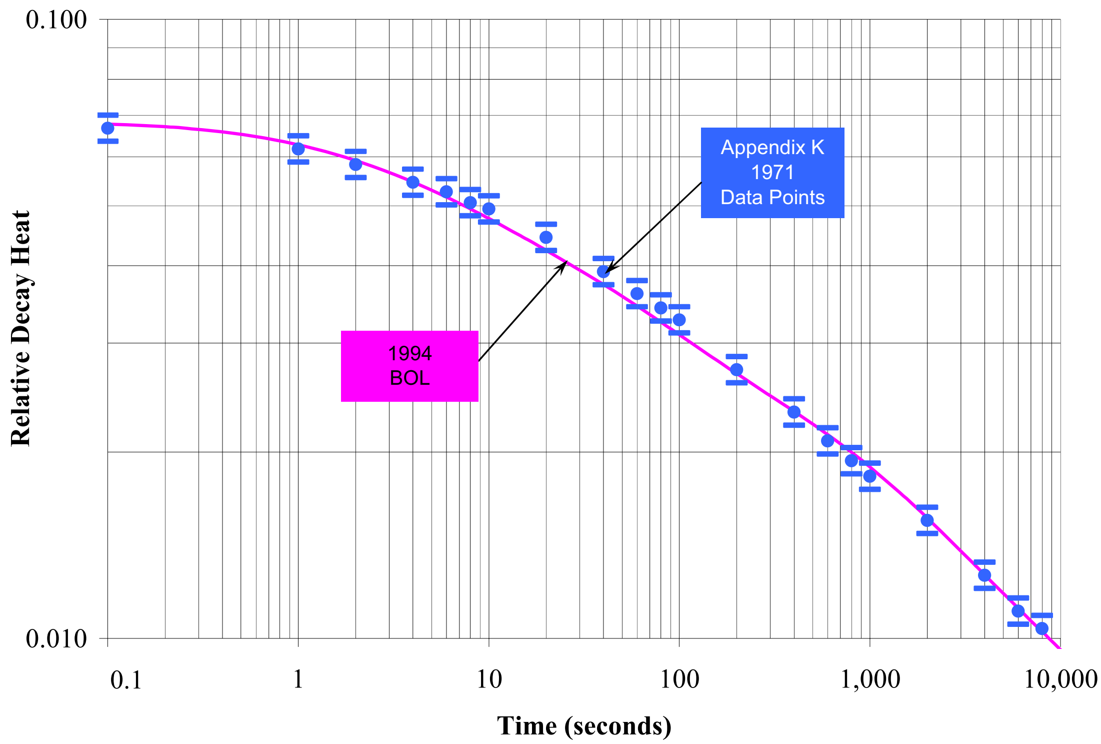


Figure 06.02.01.03-2-8—Comparison Between Model Results and Data (Only Fission Product Decay Heat – No Actinides – No Model Uncertainties)



Question 06.02.01.03-3:

This question relates to mass and energy release analysis for postulated loss-of-coolant accidents in accordance with SRP Section 6.2.1.3. RAI #82 6.02.01.03-1c is related to flow oscillations (chugging) in the reactor core calculated by RELAP5-BW. In the response to this RAI (Sup. 3), AREVA provided Figure 6.02.01.03-1-1 giving the inlet core flow from the original FSAR model and from a revised model. The revised models showed the core flow to be much smoother. Describe the changes in the RELAP5-BW model which produced the smoother core flow.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.03-3:

The changes considered to have the most significant impact on reducing the core flow oscillations include:

1. The original model has all four accumulators modeled to inject 140ºF water directly into the lower plenum to conservatively shorten the duration of the refill phase. The revised model modeled the accumulators so that three discharged into the downcomer at 140ºF and only one discharged directly into the lower plenum, but as saturated liquid at containment pressure. This approach reduced the potential for condensation in the lower plenum and, thereby, lessened flow oscillations at the core inlet.
2. The original model uses the Biasi-Zuber critical heat flux option for the steam generator (SG) tube wall of the primary side and the Becker critical heat flux option for SG tube wall of the secondary side. The revised model uses the Becker critical heat flux option on both sides of the SG tube wall. FLECHT-SEASET benchmarking tests showed that using the Becker critical heat flux option on both sides of SG tube wall greatly improves the heat transfer rate from the SG secondary side to the primary side, and produces a higher steam discharge rate to the containment. As a result, this option was used for both sides of the SG tube walls.
3. The original model allows steam backflow from containment to the reactor coolant system by setting the break junction back loss coefficient to 1.5. The revised model prevents containment backflow by setting the break junction back loss coefficient to an arbitrarily large number. This change reduces steam condensation at the break area, and therefore “smoothes” steam flow to the containment.

FSAR Impact:

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

Question 06.02.01.04-2:

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In response to RAI #82 6.02.01.04-1a Sup.1 AREVA described, in general, the heat transfer models in the RELAP5/MOD2-B&W computer code. AREVA did not designate which heat transfer correlations were used in the main steam line break mass and energy release calculations. The most significant locations of heat transfer are the heat transfer from the primary system to the two phase mixture in the affected steam generator, the reversed heat transfer from the unaffected steam generators to the primary system, and heat transfer from the core. Demonstrate that these processes are conservative for the US-EPR limiting MSLB case. SRP 6.2.1.4 recommends that calculation of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer. Demonstrate that nucleate boiling was assumed for the limiting case or provide justification for other assumptions.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.04-2:

The limiting main steam line break (MSLB) case, EFW20d41, was examined to demonstrate the heat transfer modes in the MSLB analysis. This case is the double-ended guillotine (DEG) break at 20 percent reactor power and without crediting emergency feedwater. Heat transfer data was taken at four different locations along the steam generator tubes for both the affected and unaffected steam generators. Data was also taken at three different locations along the active core for both the triple and single loop sides. The heat transfer modes are defined in Table 06.02.01.04-2-1.

In the affected steam generator the primary mode of heat transfer is nucleate boiling until it dries out and single phase vapor convection occurs. The heat transfer oscillates between transition boiling and film boiling for a period of time before eventually drying out. During the transition and film boiling interval the void fraction approaches 0.999 and the heat transfer is insignificant due to the negligible amount of liquid present. As the water level in the steam generator decreases, the heat transfer mode in each location transitions to single phase vapor convection. The bottom location takes 900 seconds to dry out and switch to single phase vapor convection. This is shown in Figures 06.02.01.04-2-1 and 06.02.01.04-2-2. The calculation of heat transfer to water in the faulted steam generator meets Standard Review Plan 6.2.1.4 recommendations of nucleate boiling heat transfer.

The initial mode of heat transfer is also nucleate boiling in the unaffected steam generator. When the main steam isolation valve is closed, the pressure increases causing the voids in the steam generator liquid to collapse and some condensation to briefly occur. This can be seen in Figures 06.02.01.04-2-3 and 06.02.01.04-2-4. After the voids collapse, the liquid in the steam generator is subcooled and the heat transfer mode is single phase liquid convection. The liquid temperature eventually rises close to the saturation temperature so that subcooled nucleate boiling occurs.

The active core only transfers heat by single phase liquid convection and maintains a void fraction of 0 throughout the transients. The heat transfer mode and void fraction in the core is shown in Figures 06.02.01.04-2-5 and 06.02.01.04-2-6.

FSAR Impact:

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

Table 06.02.01.04-2-1—Description of Heat Transfer Modes

Mode	Description
1	Single phase liquid convection at critical or supercritical pressure
2	Single phase liquid convection at subcritical pressure
3	Subcooled nucleate boiling
4	Saturated nucleate boiling
5	Subcooled transition film boiling
6	Saturated transition film boiling
7	Subcooled film boiling
8	Saturated film boiling
9	Single phase vapor convection
10	Condensation when void fraction equals one
11	Condensation when void fraction is less than one

Figure 06.02.01.04-2-1—Heat Transfer Modes at the Steam Generator Tubes in the Faulted Steam Generator for the EFW20d41 Case

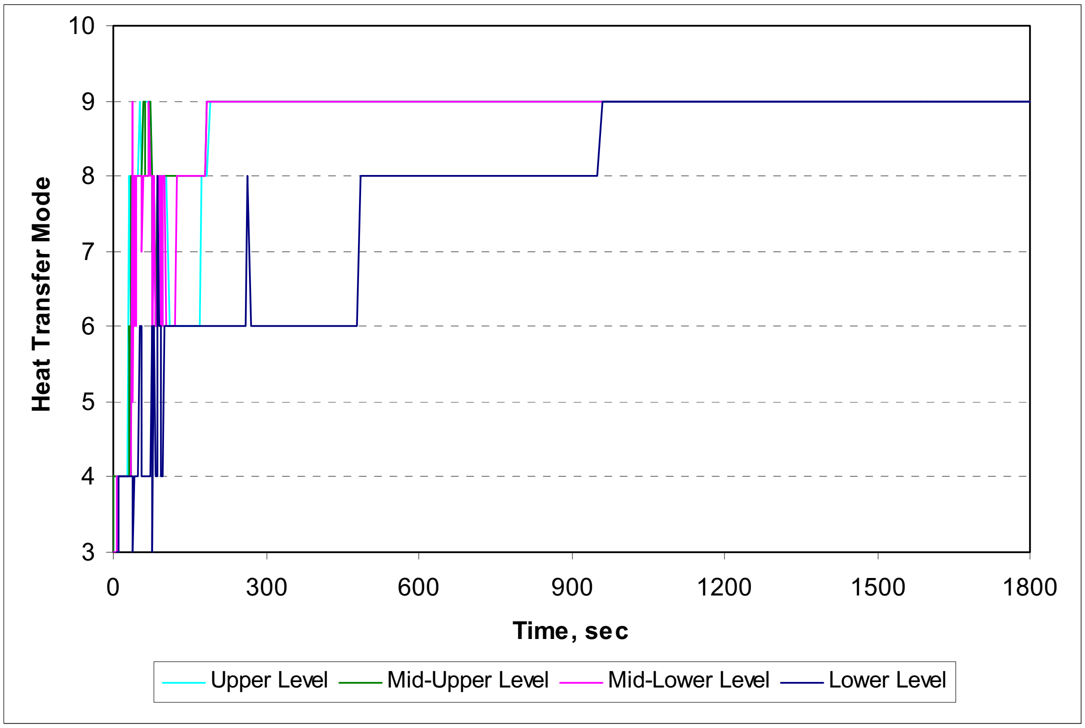


Figure 06.02.01.04-2-2—Void Fractions at the Steam Generator Tubes in the Faulted Steam Generator for the EFW20d41 Case

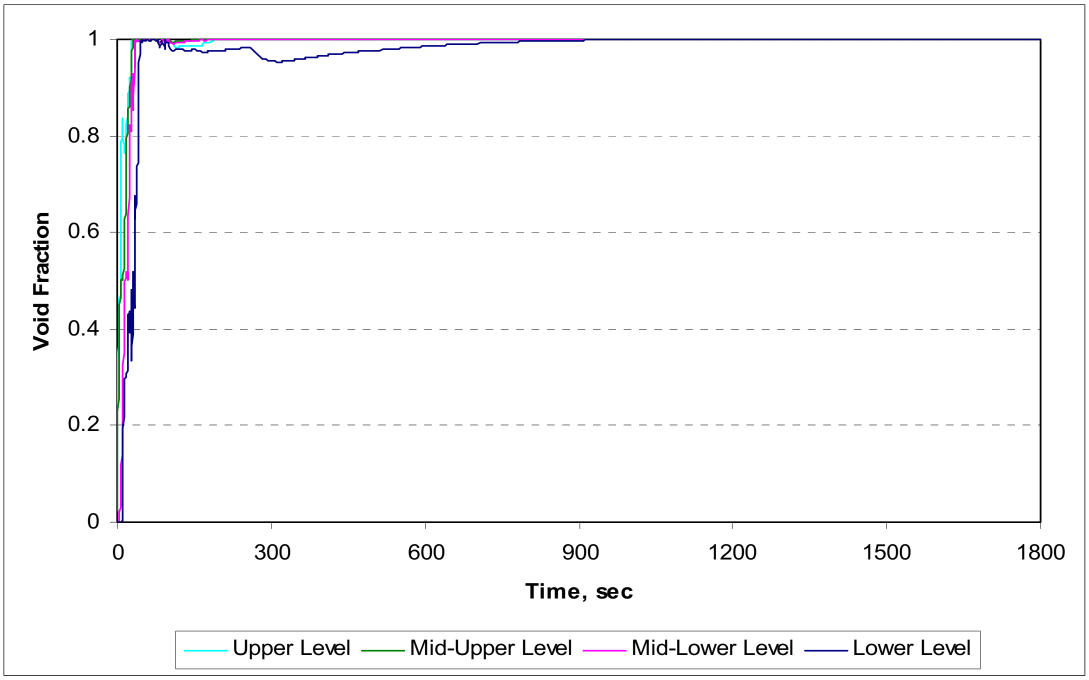


Figure 06.02.01.04-2-3—Heat Transfer Modes at the Steam Generator Tubes in the Intact Steam Generator for the EFW20d41 Case

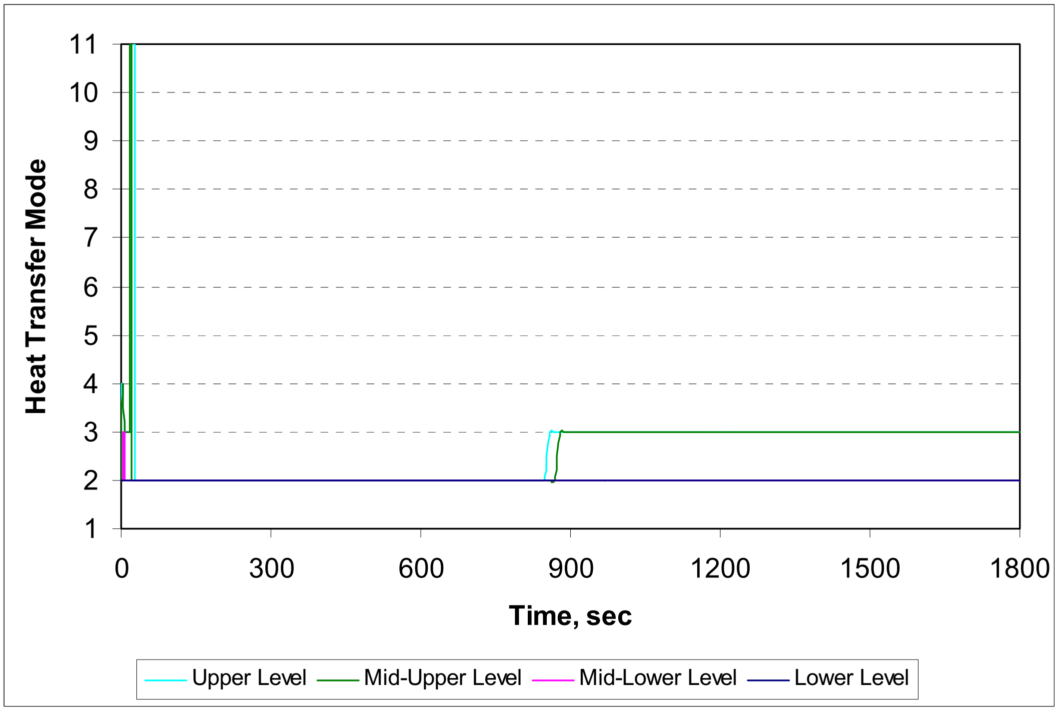


Figure 06.03.02.04-2-4—Void fractions at the Steam Generator Tubes in the Intact Steam Generator for the EFW20d41 Case

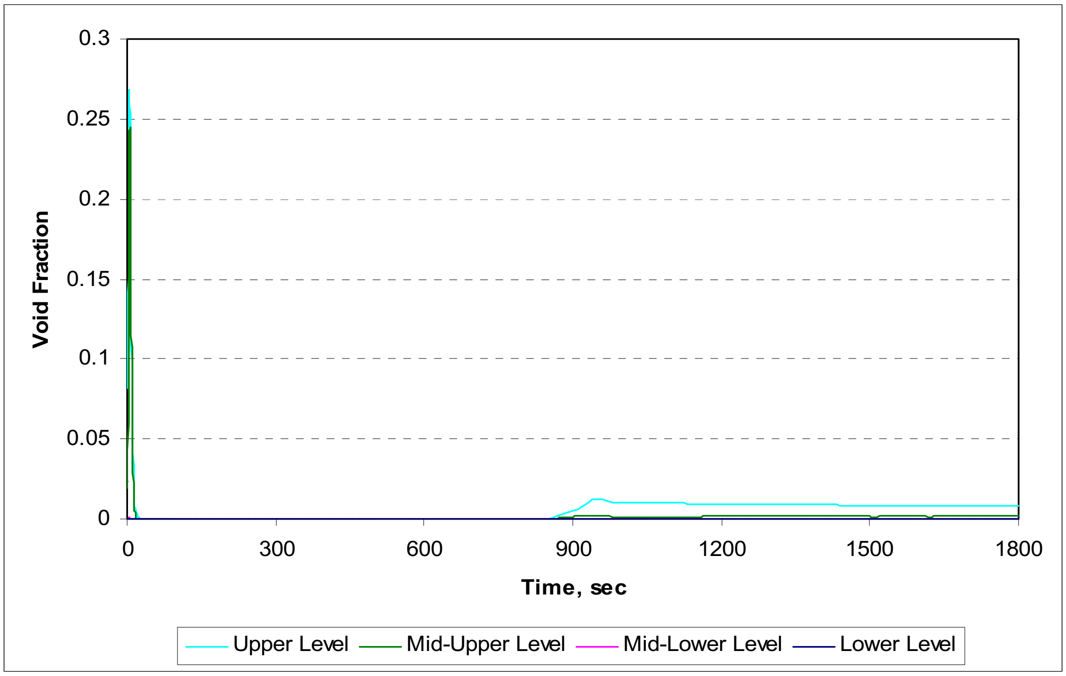


Figure 06.02.01.04-2-5—Heat Transfer Mode at the Active Core for the EFW20d41 Case

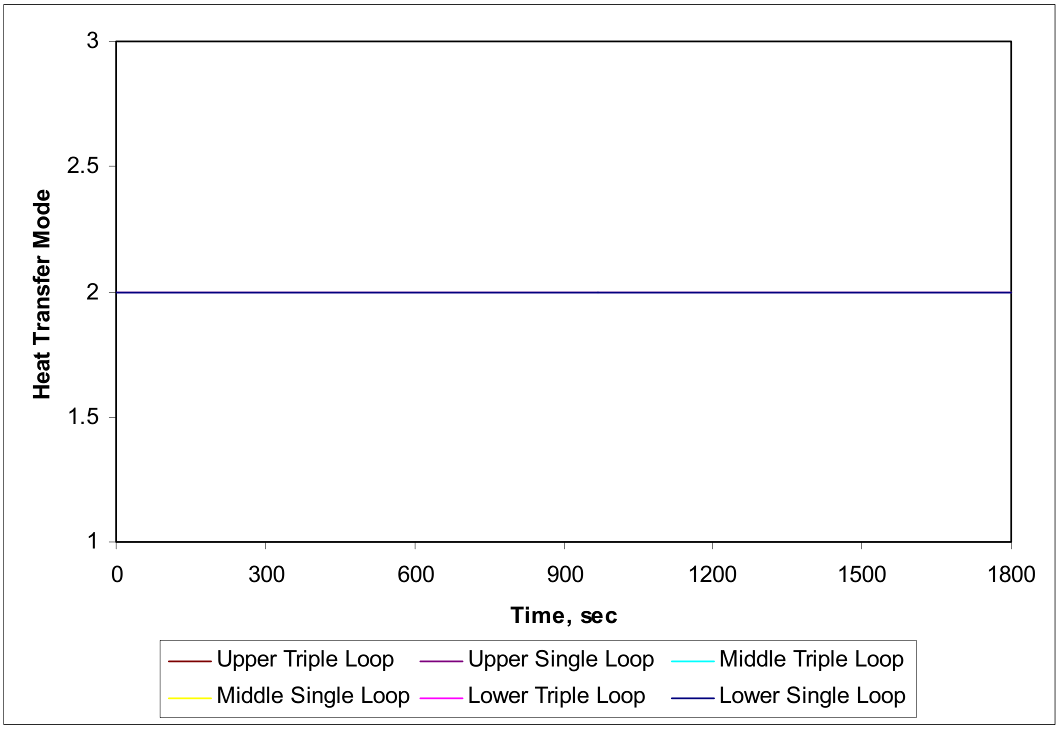
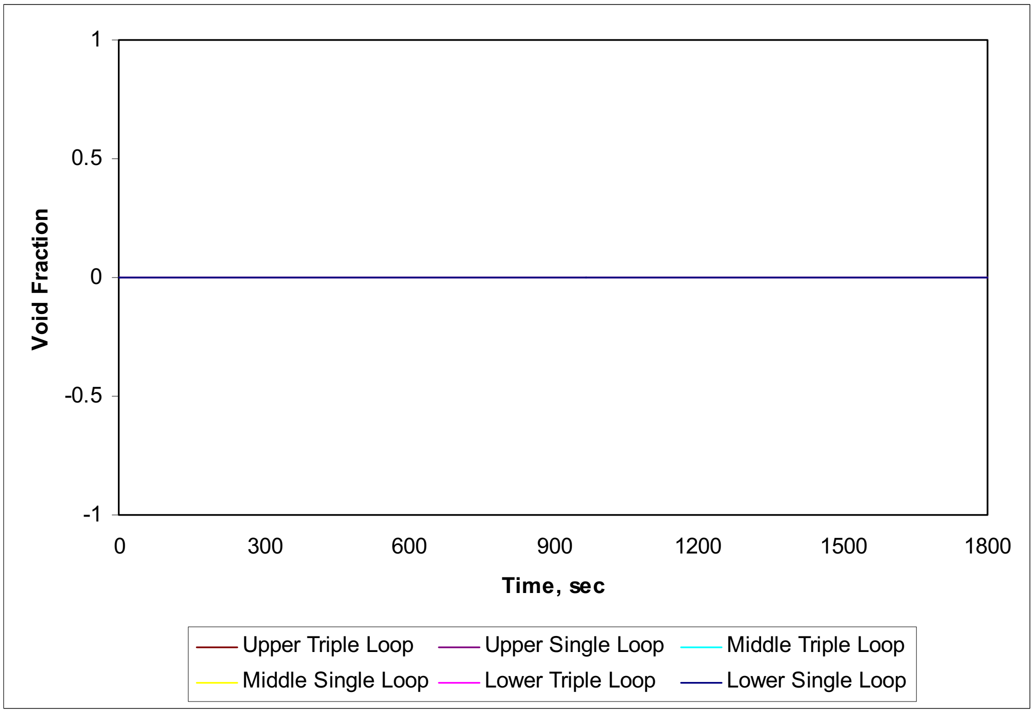


Figure 06.02.01.04-2-6—Void Fractions at the Active Core for the EFW20d41 Case



Question 06.02.01.04-3:

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In RAI #82 6.02.01.04-1c the NRC staff requested that AREVA identify the decay heat model that was used for the main steam line break analyses and provide justification that the model is conservative for containment analysis. SRP 6.2.1.4 does not address decay heat. However the SRP states that among the energy sources which should be considered is the energy transferred from the primary coolant to the water in the affected steam generator during blowdown. Additional decay heat will increase the reactor system temperature which will cause additional heat to flow to the affected steam generator. This energy source should therefore be made conservative in the safety analysis. Demonstrate that decay heat model used for evaluating the containment response to a main steam line break is conservative. The staff has accepted the 1979 ANS decay heat standard with a 2σ multiplier. If the 1979 ANS decay heat standard or a later standard is used for analysis, provide and justify the input assumptions selected. These include assumptions for actinide decay, actinide production factor, multiplier to account for neutron capture activation, fissions per initial fissile atom and power history. The effect of these assumptions is discussed in NRC IN 96-39.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.04-3:

The decay heat model used in the analysis is 1.2 times the values for infinite operating time in the 1971 ANS Standard plus decay of actinides. This decay heat modeling is conservatively high because return to power does not occur prior to the peak of containment pressure in the limiting main steam line break (MSLB) case.

FSAR Impact:

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

Question 06.02.01.04-4:

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. In RAI #82 6.01.02.04-1d the staff questioned the assumptions used to calculate reactor power following a main steam line break and the effect of these assumptions on containment analysis. In the response AREVA provided a curve of reactor power vs. time for the MSLB case calculated to produce the highest calculated containment temperature and pressure. This was for a postulated double ended break with the reactor at an initial power level of 50%. Offsite power was assumed to remain available so that the reactor coolant pumps would continue to operate. The break flow was assumed to be limited by the area of the flow restrictors in the steam generator nozzles to 1.4 ft2. The RELAP5/Mod2-B&W computer code was used to model the reactor system. No return to power was calculated even though the most reactive control rod was assumed to be stuck and to not enter the core following reactor trip.

AREVA also evaluated the consequences of a main steam line break in FSAR Section 15.1.5 to determine the potential for reactor core damage. In these evaluations the S-RELAP5 computer code was used. For the limiting break a return to power was calculated which reached a maximum of 23.14% over a period of approximately 200 seconds. This energy generation is sufficient, if considered in the containment analysis to have a considerable effect on the calculated containment temperature and pressure. In the response to RAI #34 15.01.05-2, AREVA provided the results from the sensitivity study of postulated steam line breaks for which the potential for core damage was evaluated. Initial power levels of 100%, 60%, 25% and 0% were investigated using S-RELAP5. The core was calculated to return to power generation following reactor trip regardless of the initial power level. The staff understands that part of the reason that return to power was calculated for the Chapter 15 analyses but not for the Chapter 6 analysis, was that a much higher control rod shutdown margin was assumed for the Chapter 6 analyses as compared with the Chapter 15 analyses. Provide analyses of the containment response to postulated main steam line breaks for which the core physics assumptions are consistent or conservative as compared to those which the staff is reviewing to support the main steam line break analyses in FSAR chapter 15.

The above questions are follow-up questions to previously issued RAIs and the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.04-4:

A response to this question will be provided by May 5, 2010.

Question 06.02.01.04-5:

This question relates to conservativeness of the secondary system break mass and energy release calculations in FSAR Section 6.2.1.4. The staff understands that following a postulated main steam line break, if the liquid fraction in the steam separator region of the RELAP5-BW model exceeds a threshold value then entrained liquid is assumed to exit the break and to flow into the containment. If the liquid fraction in the steam separator region of the RELAP5-BW model is less than a second threshold value, any calculated entrained liquid is treated as steam. SRP 6.2.1.4 states that if liquid entrainment is assumed, experimental data should be provided to support the predictions. In the past the staff has accepted MSLB calculations with liquid entrainment based on model validation applying full scale steam generator test data. Provide experimental validation of the RELAP5-BW MSLB methodology or provide new analyses for which no liquid entrainment is assumed.

The above question is a follow-up question to the containment audit held in Lynchburg on July 14 and 15, 2009.

Response to Question 06.02.01.04-5:

The main steam line break (MSLB) analyses have been completed in which no liquid release to containment occurred during the event. These analyses demonstrate that the AREVA methods are conservative with respect to liquid release for containment response.

AREVA methods distinguish between liquid entrainment and liquid swell. Entrained liquid occurs when liquid drops swept out the break are caused by high steam velocity, whereas liquid release due to swell is caused by rapid voiding of the steam generator (SG) inventory which pushes water out the break. The comparative effect of steam-only release versus liquid and steam release is best demonstrated in those cases in which the greatest liquid is released. The amount of liquid released in an MSLB event increases with increasing break size, and liquid released in an MSLB event increases with decreasing initial power level.

Therefore, the greatest liquid release occurs during the double-ended guillotine (DEG) break at a low initial power level, as shown in Figure 06.02.01.04-5.1. The containment response from these cases shows that the DEG break from 20 percent power is the limiting case for containment pressure (see Figure 06.02.01.04-5.2). Furthermore, the treatment of the liquid release using the AREVA method causes the limiting MSLB case of any break size from any power level to be the DEG break from 20 percent power. Therefore, the 20 percent power DEG case represents not only the limiting MSLB case with respect to liquid release, but also the limiting MSLB case.

The DEG break from 20 percent power case was modified to allow steam-only release of the break flow. The other inputs and boundary conditions are unchanged from the case which allows both liquid and steam release of the break flow. Figure 06.02.01.04-5.3 shows that the modification to the input deck is successful in preventing the liquid from flowing out the break by preventing liquid swell or entrainment in the steam generator dome.

Figure 06.02.01.04-5.4 provides the integrated mass release for the 20 percent power DEG case for the liquid and steam release case and the steam-only release case. The liquid and steam release case exhibits rapid mass release initially due to the presence of liquid in the break flow compared to the steam-only release case. The integrated mass curve for both cases flattens due to the isolation of main feedwater and no emergency feedwater causing SG dryout. A separate sensitivity study demonstrates that the lack of emergency feedwater is conservative for the limiting MSLB DEG case. The modification to the input deck to produce the steam-only release causes RELAP5MOD2B-W to calculate a slightly larger secondary system mass error. This mass error causes a slight difference in the integrated mass shown in Figure 06.02.01.04-5-4 between the liquid and steam release case and the steam-only release case. However, the mass error is small compared to the total release, and the error increases the integrated mass release which will increase the containment response for the steam-only release case. Figure 06.02.01.04-5-5 provides the integrated energy release of the 20 percent power DEG with liquid and steam release case and the steam-only release case, which exhibit the same behavior as the integrated mass release.

Figure 06.02.01.04-5-6 provides the containment pressure response for the 20 percent power DEG case with liquid and steam release case and steam-only release case. The peak containment pressure in the liquid and steam release case is significantly greater than the peak containment pressure in the steam-only release case. The rapid mass and energy release due to the presence of liquid in the break flow is the dominant effect in the comparison study.

In addition to the 20 percent initial power case, the study was performed on the 0 percent power and 40 percent initial power DEG break cases to show that the comparison is well bounded. Table 06.02.01.04-5-1 summarizes the containment response for the MSLB entrainment study, and demonstrates that the AREVA method, which allows liquid and steam release, is conservative compared to steam only release.

FSAR Impact:

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

Table 06.02.01.04-5-1—Containment Pressure Results

Initial Power (% power)	Peak Pressure (psia)	
	Liquid and Steam	Steam Only
0	60.9	58.6
20	66.3	56.5
40	64.1	54.6

Figure 06.02.01.04-5-1—SG Dome Liquid Fraction

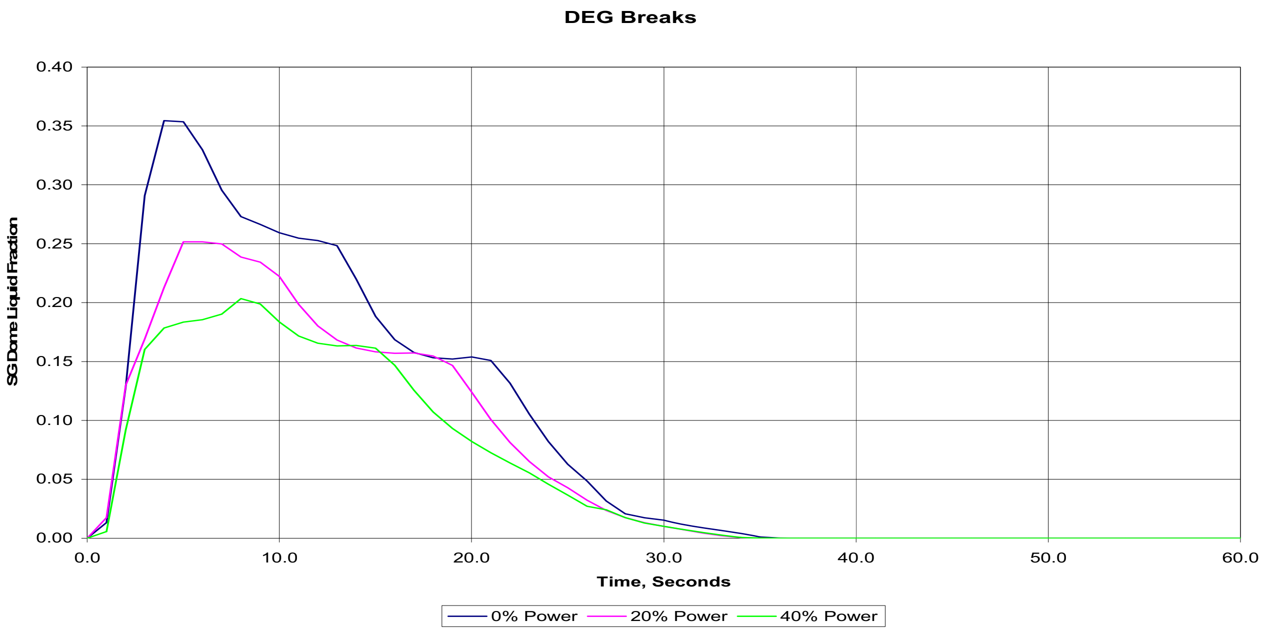


Figure 06.02.01.04-5-2—Containment Pressure

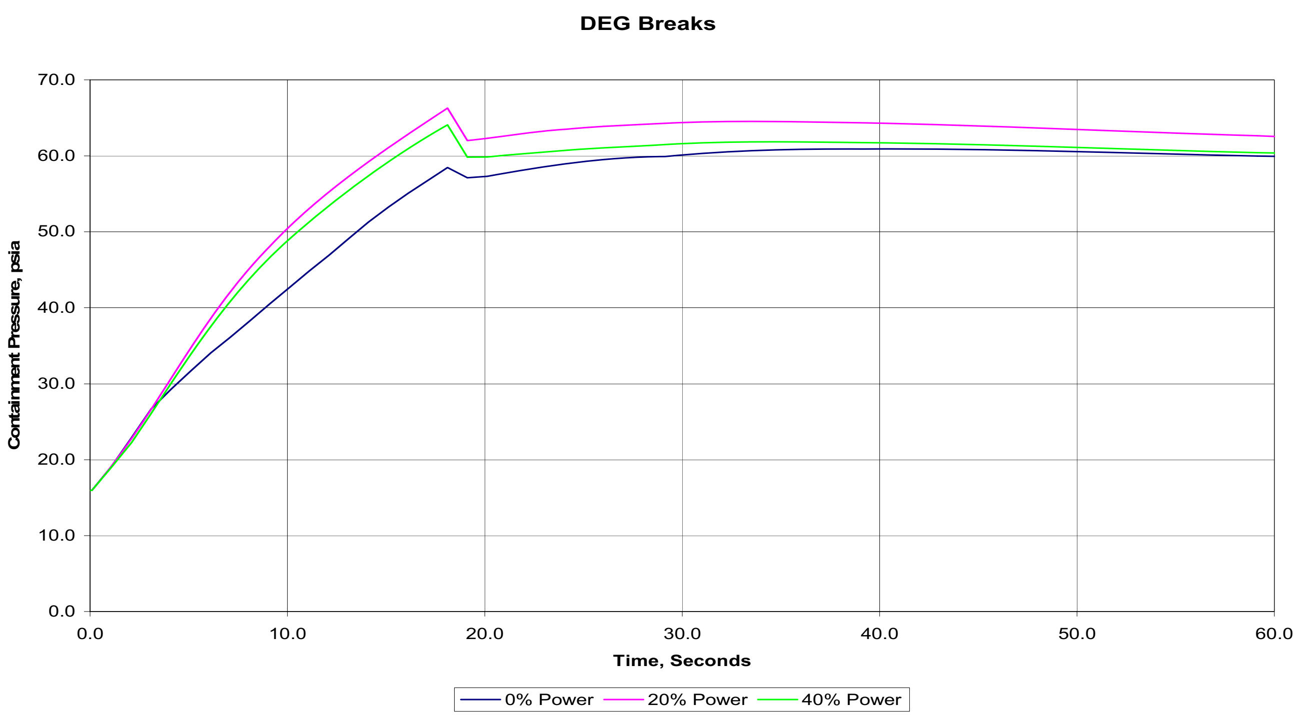


Figure 06.02.01.04-5-3—SG Dome Liquid Fraction

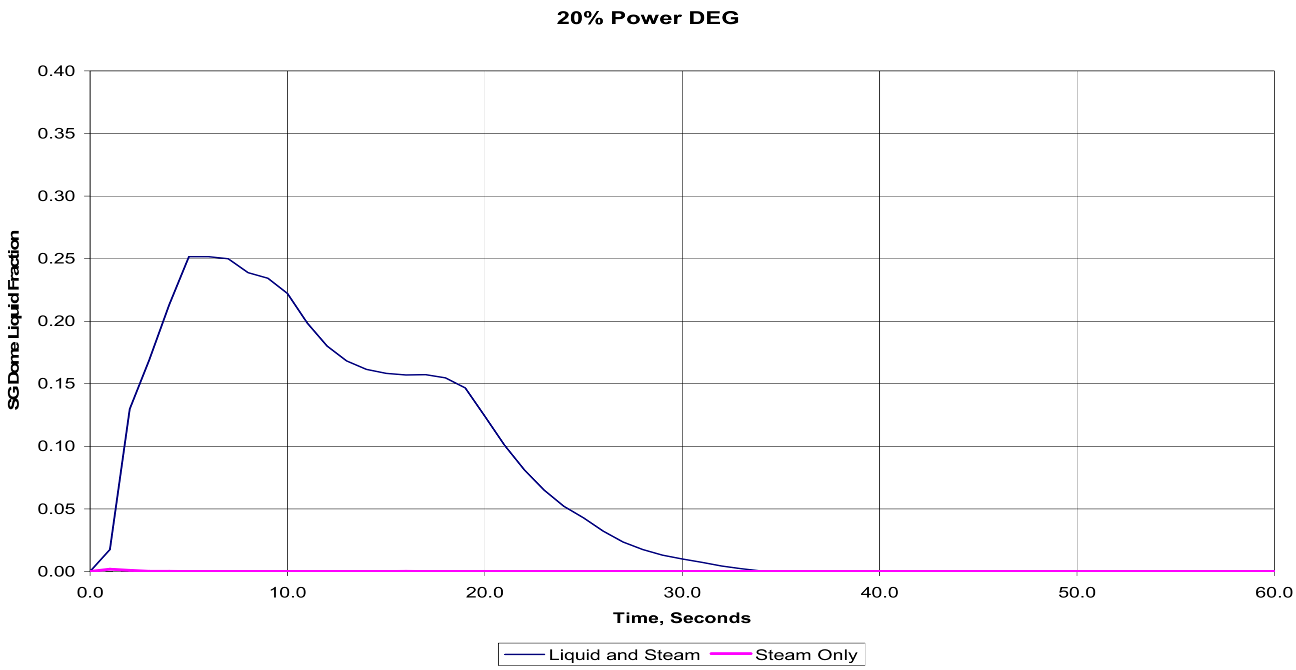


Figure 06.02.01.04-5-4—Integrated Break Mass

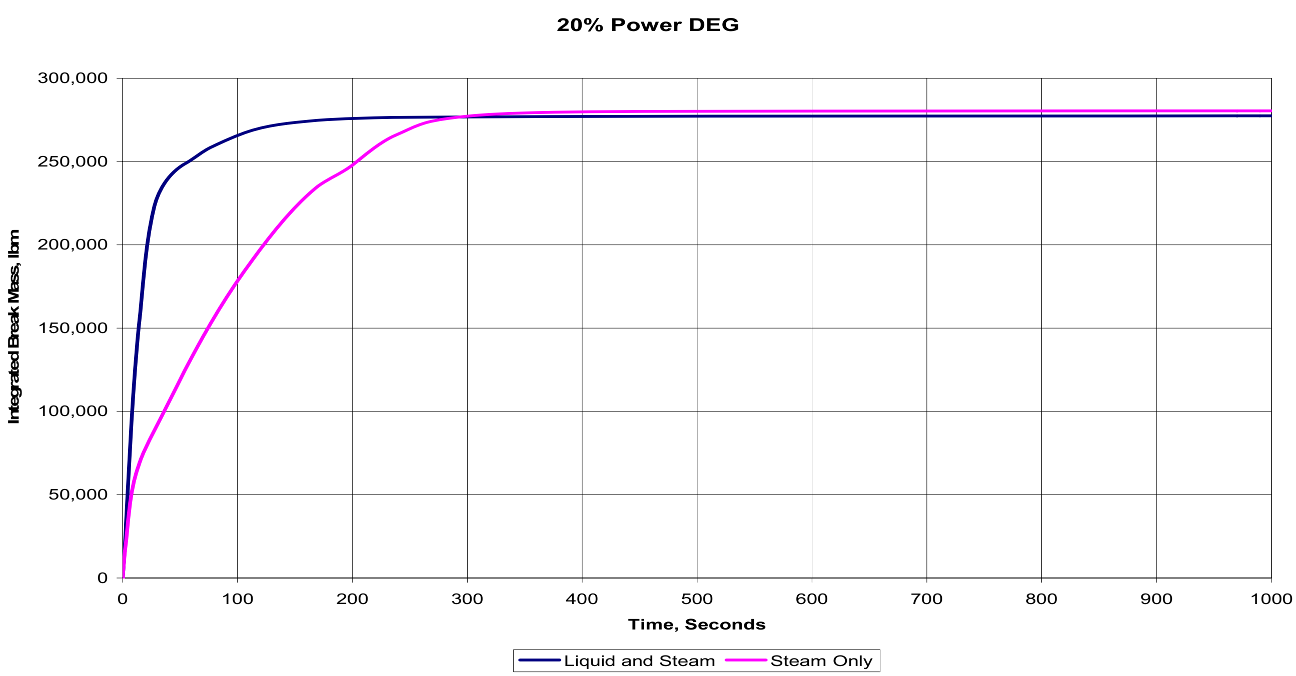


Figure 06.02.01.04-5-5—Integrated Break Energy

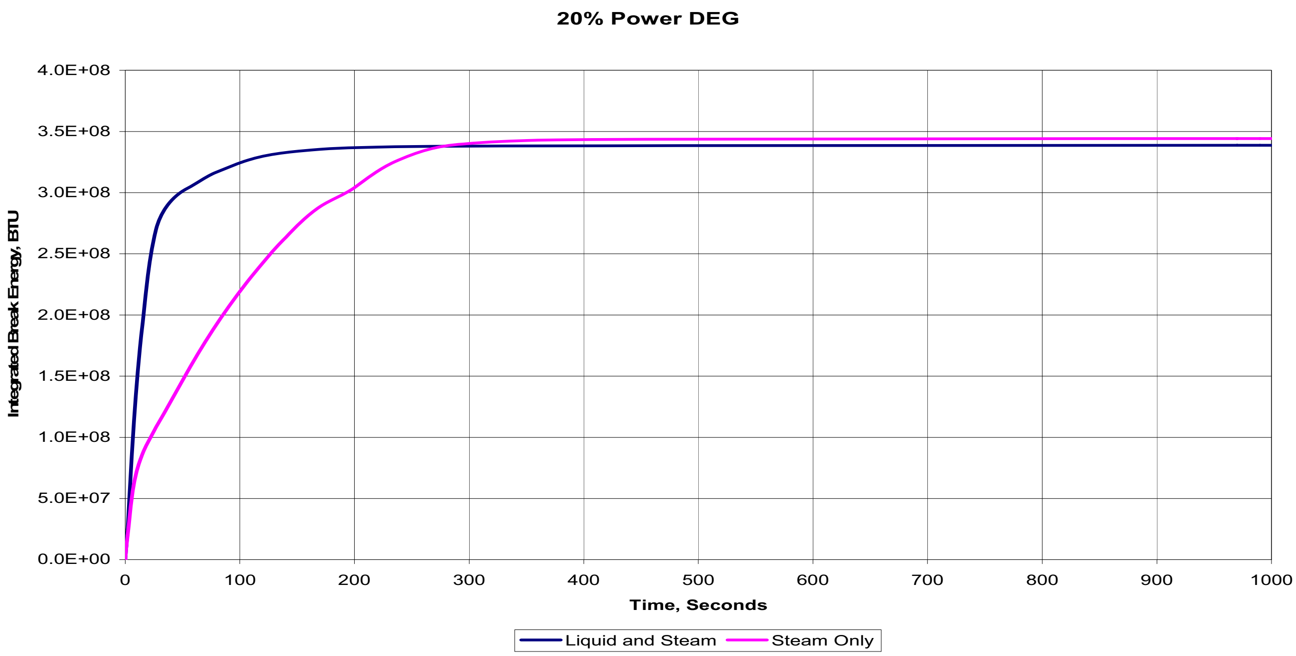


Figure 06.02.01.04-5-6—Containment Pressure Comparison