ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Wednesday, February 11, 2009 4:53 PM
To:	Getachew Tesfaye
Cc:	NOXON David B (AREVA NP INC); DELANO Karen V (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 2
Attachments:	RAI 133 Supplement 2 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 7 of the 17 questions of RAI No. 133 on December 8, 2008. AREVA NP submitted Supplement 1 to the response on January 30, 2009 to address 3 of the 17 questions.

The attached file, "RAI 133 Supplement 2 Response US EPR DC.pdf" provides technically correct and complete responses to 4 of the remaining 7 questions.

The following table indicates the respective pages in the response document, "RAI 133 Supplement 2 Response US EPR DC.pdf" that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-232	2	6
RAI 133 — 19-233	7	15
RAI 133 — 19-238	16	17
RAI 133 — 19-240	18	23

The schedule for technically correct and complete responses to the remaining 3 questions is unchanged and provided below.

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

Sincerely,

Ronda Pederson
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From: WELLS Russell D (AREVA NP INC)
Sent: Friday, January 30, 2009 4:21 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 (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 1

Getachew,

The proprietary and non-proprietary versions of the response to RAI No. 133, Supplement 1 are submitted via AREVA NP Inc. letter, "Response to U.S. EPR Design Certification Application RAI No. 133, Supplement 1 " NRC 09:008, dated January 30, 2009. The enclosure to that letter provides technically correct and complete responses to 3 of the remaining 10 questions in RAI No. 133. An affidavit to support withholding of information from public disclosure, per 10CFR2.390(b), is provided as an enclosure to that letter.

The schedule for technically correct and complete responses to the remaining questions in RAI No. 133 is provided below:

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-232 (c)	February 13, 2009
RAI 133 — 19-233	February 13, 2009
RAI 133 — 19-238	February 13, 2009
RAI 133 — 19-240 (2)	February 13, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

Sincerely,

(Russ Wells on behalf of)
Ronda Pederson
ronda.pederson@areva.com
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From: WELLS Russell D (AREVA NP INC)
Sent: Monday, December 08, 2008 6:43 PM
To: 'Getachew Tesfaye'
Cc: 'John Rycyna'; 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. 133, FSAR Ch 19

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 133 Response US EPR DC.pdf" provides technically correct and complete responses to 7 of the 17 questions.

The following table indicates the respective pages in the response document, “RAI 133 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 133 — 19-230	2	2
RAI 133 — 19-231	3	3
RAI 133 — 19-232	4	4
RAI 133 — 19-233	5	5
RAI 133 — 19-234	6	6
RAI 133 — 19-235	7	7
RAI 133 — 19-236	8	8
RAI 133 — 19-237	9	9
RAI 133 — 19-238	10	10
RAI 133 — 19-239	11	12
RAI 133 — 19-240	13	45
RAI 133 — 19-241	46	47
RAI 133 — 19-242	48	48
RAI 133 — 19-243	49	49
RAI 133 — 19-244	50	50
RAI 133 — 19-245	51	52
RAI 133 — 19-246	53	79

A complete answer is not provided for 10 of the 17 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 133 — 19-230	June 19, 2009
RAI 133 — 19-232 (c)	February 13, 2009
RAI 133 — 19-233	February 13, 2009
RAI 133 — 19-236	January 30, 2009
RAI 133 — 19-237	January 30, 2009
RAI 133 — 19-238	February 13, 2009
RAI 133 — 19-240 (2)	February 13, 2009
RAI 133 — 19-242	January 30, 2009
RAI 133 — 19-243	June 19, 2009
RAI 133 — 19-244	March 6, 2009

(Russ Wells on behalf of)

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Thursday, November 06, 2008 8:42 PM
To: ZZ-DL-A-USEPR-DL

Cc: Edward Fuller; Theresa Clark; Hanh Phan; Hossein Hamzehee; Lynn Mrowca; John Rycyna; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 133 (1456), FSARCh. 19

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 29, 2008, and discussed with your staff on November 6, 2008. Draft RAI Questions 19-229 was deleted and Draft RAI Questions 19-238 was modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information wiwasll impact the published schedule.

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

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 211

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D31A9F79B)

Subject: Response to U.S. EPR Design Certification Application RAI No. 133, FSAR Ch 19, Supplement 2
Sent Date: 2/11/2009 4:53:26 PM
Received Date: 2/11/2009 5:02:02 PM
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MESSAGE	6513	2/11/2009 5:02:02 PM
RAI 133 Supplement 2 Response US EPR DC.pdf		345287

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
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Response to

Request for Additional Information No. 133, Supplement 2

11/07/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

Question 19-232:

(Follow-up to Question 19-84d) The response to Question 19-84d discusses the results of sensitivity cases that were investigated for source term calculations.

a. In discussions of the “effect of the SAHRS” (severe accident heat removal system), “isolation failure size,” and “small leakage,” the text refers to Figures 19-84d-1 through 19-84d-6 that are missing in the response. Please, provide these figures.

b. In discussions under “small leakage,” the containment leak rate for 2- and 3-inch failure diameter (Figure 19-84-9d [SIC]), the response indicates that the flow rate for the 3-inch diameter case is conservatively calculated as (9/4) x (the leak rate for a 2-inch diameter). However, the figure illustrating the results show an inconsistency, namely, where the leak rate is lower for the 3-inch diameter case as compared to that for the 2-inch diameter break. Please, explain this apparent inconsistency.

c. In addition, in discussing the effects of molten corium-to-concrete interaction (MCCI), the response refers to Figures 19-84d-11 and 13 depicting lower values of SrO releases for cases involving MCCI as compared to accidents where MCCI is prevented. Please explain the phenomenological processes causing this apparent anomaly.

Response to Question 19-232:

Response to Question 19-232a:

This question was answered in the Response to RAI 133.

Response to Question 19-232b:

This question was answered in the Response to RAI 133.

Response to Question 19-232c:

Additional sensitivity cases have been performed to investigate the apparent anomaly.

When large amounts of MCCI were examined, the basemat thickness was increased from 0.1 meters to 5 meters and flooding was disabled. Because the basemat is defined in MAAP as a heat sink, the change from 0.1 meters to 5 meters incurred a large increase in the size of the nodalization array that represents the basemat. This alteration in array size influenced the calculation of some of the events calculated during the transient. This was confirmed by the examination of four cases (permutations of 0.1 meter or 5 meter basemat and passive flooding allowed or not) where the transients did not exhibit the same traces until the time when the corium enters the spreading area.

Because the transients are the same until melt plug failure, the MAAP calculation was modified to remove the influence that the basemat thickness has on the events of the transient prior to the time when the corium enters the spreading area. Therefore, the MAAP runs for the 1 square meter isolation failure area were performed with the identical inputs for the basemat up to the point were the corium enters the spreading area. At this point, the case without flooding (where MCCI was expected) was restarted with the 5 meter basemat, while the case with flooding was allowed to continue.

The impact of the thicker basemat on the calculational time step was confined to the time when MCCI was expected. The 1 inch failure area cases were treated similarly.

Figure 19-232c-1 and Figure 19-232c-2 show revised SrO releases for cases involving MCCI compared to accidents where MCCI is prevented. The plots show higher values of SrO release for cases involving MCCI, which is the expected result and removes the anomaly found previously.

The updated MAAP run for ST1.8 in Table 19-232c-1 shows a decrease in the source term for Iodine (FREL (2)), Cesium (FREL(6)), and Tellurium (FREL(3) and FREL(11)). Although the increase in FREL(11) is a large percentage change, its absolute value is small, and it does not significantly impact the overall decrease in the Tellurium source term.

The updated MAAP runs for ST1.8b in Table 19-232c-2 shows a decrease in the source term for Iodine (FREL (2)), a modest increase in the source term from Cesium (FREL(6)), and a decrease in the source term from Tellurium (FREL(3) and FREL(11)). The source term for SrO (FREL(4)) increases by an order of magnitude.

These changes have a negligible effect on risk, based on the following:

- The change in Iodine, Cesium and Tellurium source term is minor and would not change the categorization of release category as “large;” therefore large release frequency (LRF) is unchanged.
- The affected release categories (with MCCI) have a cumulative frequency of approximately 8E-10 per year. This is equivalent to approximately 0.2 percent of core damage frequency. A small increase in source term for these release categories would have a negligible impact on the overall source term.
- Although the increase in source term for SrO is significant (about one order of magnitude), the frequency weighted contribution of this increase will not make a significant change to the offsite consequences from the U.S. EPR.

FSAR Impact:

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

Table 19-232c-1—Comparison of Source Term Results for MAAP Run ST1.8 (RC 205 and 304)

		FREL(1)	FREL(2)	FREL(3)	FREL(4)	FREL(5)	FREL(6)	FREL(7)	FREL(8)	FREL(9)	FREL(10)	FREL(11)	FREL(12)
ST1.8 (RC205/ 304)	Source Term Release Fraction	9.78E-01	5.71E-02	2.89E-02	4.05E-03	9.81E-03	3.63E-02	6.08E-03	2.96E-04	5.32E-04	9.33E-02	4.44E-08	1.17E-07
	Updated Release Fraction	9.94E-01	4.66E-02	2.46E-02	1.90E-03	6.76E-03	3.44E-02	6.31E-03	1.17E-04	2.14E-04	7.39E-02	3.15E-06	8.63E-08
	Percent Change	2	-18	-15	-53	-31	-5	4	-60	-60	-21	6985	-27

Table 19-232c-2—Comparison of Source Term Results for MAAP Run ST1.8b (RC 203 and 302)

		FREL(1)	FREL(2)	FREL(3)	FREL(4)	FREL(5)	FREL(6)	FREL(7)	FREL(8)	FREL(9)	FREL(10)	FREL(11)	FREL(12)
ST1.8b RC 203/ 302	Source Term Release Fraction	8.87E-01	5.31E-02	2.40E-02	1.37E-04	6.75E-03	2.80E-02	2.21E-03	1.50E-05	2.36E-04	1.58E-01	8.53E-06	2.63E-05
	Updated Release Fraction	9.34E-01	4.38E-02	2.38E-02	1.94E-03	6.79E-03	3.24E-02	6.42E-03	1.19E-04	2.17E-04	7.33E-02	5.98E-06	9.41E-08
	Percent Change	5	-18	-1	1323	1	16	191	696	-8	-54	-30	-100

Figure 19-232c-1—Mass Fraction of SrO Released to Environment – Cases with 1m² Isolation Failure Area With & Without MCCI

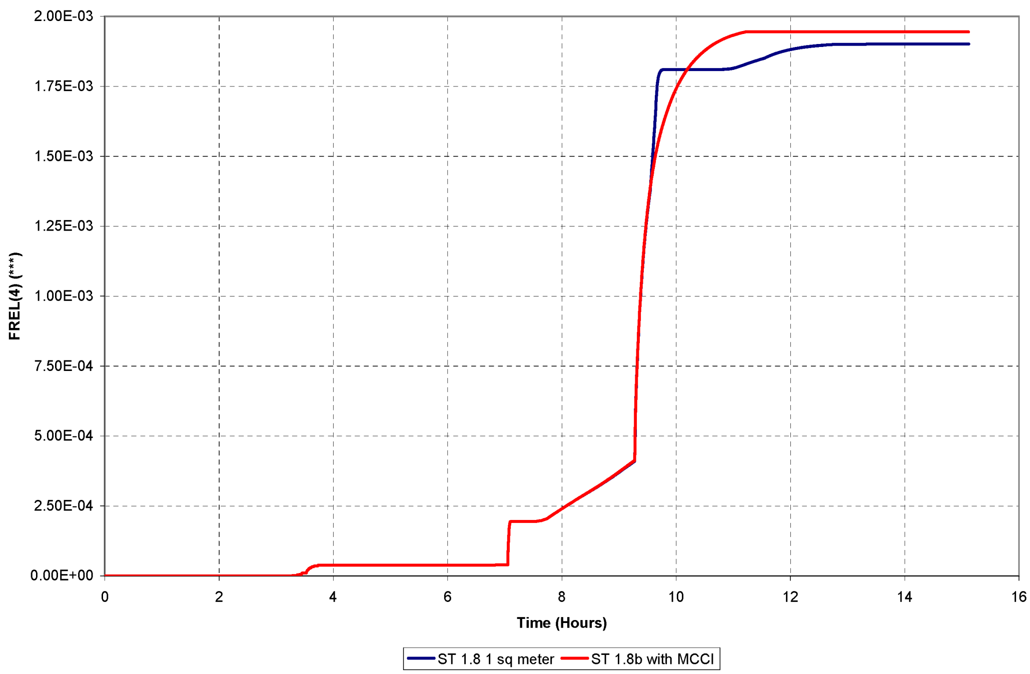
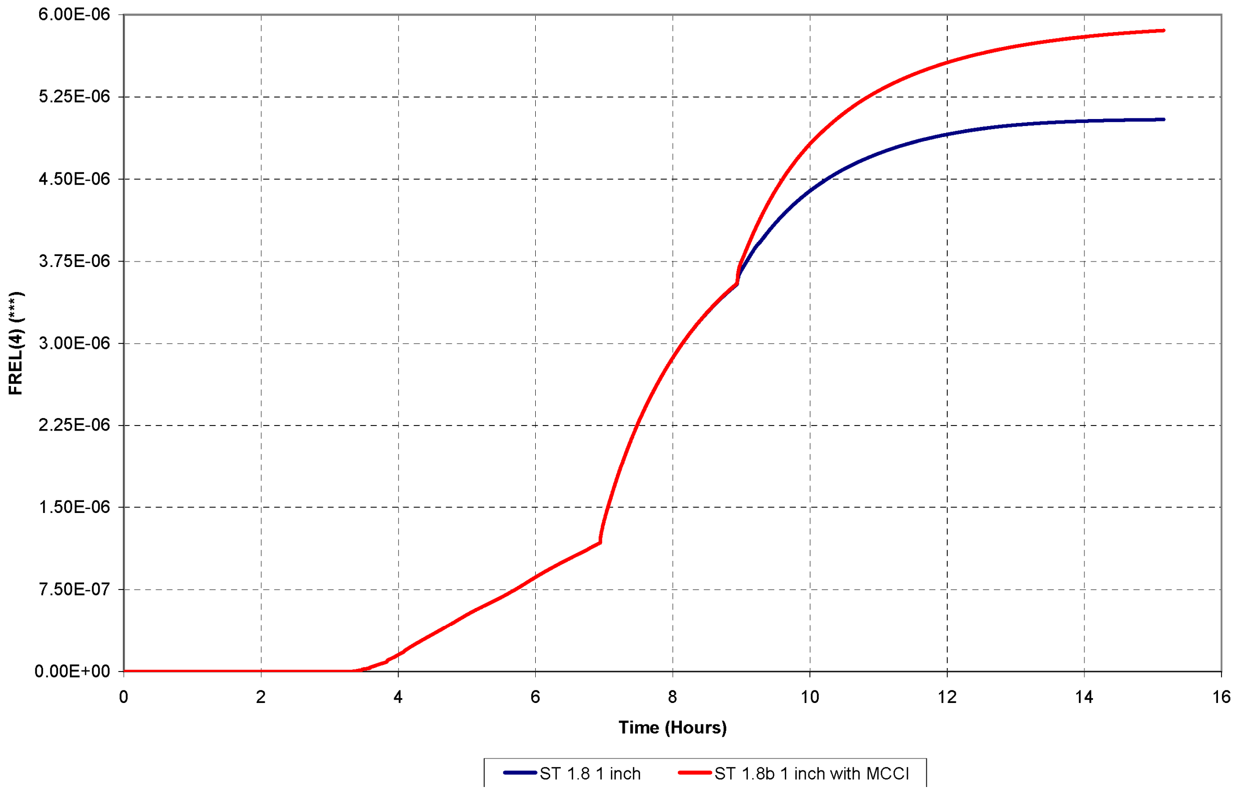


Figure 19-232c-2—Mass Fraction of SrO Released to Environment – Cases with 1 Inch Failure Area With & Without MCCI



Question 19-233:

(Follow-up to Question 19-84e) The response to Question 19-84e discusses the assumptions applicable to analysis of steam generator tube rupture (SGTR) source terms (i.e., Release Category 702). The response confirms that the analysis assumes the failure of a single tube, whereas the results of a preliminary sensitivity study by AREVA shows the increase in fission product source term under the postulated condition of more than a single tube rupture. Please address the likelihood for multiple SGTRs and provide associated source terms.

Response to Question 19-233:

The possibility of multiple SGTRs is investigated as follows:

- Additional modular accident analysis program (MAAP) cases with creep-induced SGTR are performed for different number of tube ruptures.
- Based on the results of the MAAP runs, the maximum number of tubes susceptible to fail is evaluated as the number of ruptures that achieve successful primary depressurization.
- The likelihood of a certain number of tube failures is estimated with a Poisson distribution, for each applicable core damage end state (CDES).
- The source term is calculated based on the results for each individual MAAP case and the probabilities calculated.

A series of representative MAAP cases with multiple creep-induced SGTRs following a 2 inch cold leg loss of coolant accident (LOCA) with station blackout (SBO) were analyzed. Runs with 1, 2, 5, 10, and 100 tubes ruptures were performed.

Figure 19-233-1 shows the primary pressure for each case. The 100 tube ruptures case shows a significant depressurization of the primary system following the ruptures. Therefore, the rupture of 100 tubes precludes any further induced SGTR.

Conditional probability of a given number of tube ruptures

To address the source term resulting from multiple SGTRs, it is necessary to estimate the likelihood of multiple ruptures. A simplified method using the Poisson distribution is used.

Precedent exists for estimating the probability of multiple tube failure by the use of a Poisson distribution. For example, Reference 1 states:

> "Assuming that each tube contains exactly one crack leads to the probability P_f of having a randomly chosen tube failed. Following the Poisson distribution, and taking advantage of the large number of cracked tubes N, the probability of having **i** tubes failed follows as:
>
> $$\mathbf{p(i) = (N\,P_f)^i \,/\, i!\ \ e^{-N\,P_f};\quad i=0, 1, 2, \ldots N} \qquad (1)$$
>
> The probability of having at least one out of N tubes failed is then:
>
> $$\mathbf{P(i>=1) = 1 - e^{-N\,Pf}} \qquad (2)$$
>
> Accordingly, two or more tubes will fail with a probability of:

$$\mathbf{P\ (\ i>=2\ )\ =\ 1-\ (\ 1+N\ P_f\ )\ e^{-N\ P_f}} \tag{3}$$

To obtain an estimate of the conditional probability of different numbers of ruptured tubes due to thermally induced creep, the following assumptions are used:

- Tubes within a single MAAP steam generator (SG) tube node are exposed to the same temperature, pressure, and stress conditions.
- Ruptures occur in the hot side tube node. In addition, hot leg rupture is more likely to happen before SGTR, and the resulting depressurization precludes subsequent failures leading to a relatively low probability of SGTR.
- One tube rupture does not significantly change the conditions to which the other tubes are exposed.
- The probabilities of induced SGTR calculated in the induced reactor coolant system (RCS) rupture phenomenological evaluation for each CDES correspond to the probability of induced creep rupture of one or more tubes (i.e., P(i>=1) in the equations).

The U.S. EPR has a total of 5,980 tubes per SG and the number of tubes vulnerable (N in Equations 1- 3) is:

N = (total number of tubes in one SG) * (Fraction of tubes receiving "outflow" from the natural circulation gas flow pattern)

This gives:

N = 5980 * 0.5 = 2990 tubes / SG

The Poisson distribution parameter (N^*P_f) for this situation can be evaluated by determining P_f using Equation 2 above.

Table 19-233-1 summarizes the probability of failure Pf calculated using Equation 2 for all the CDES (see U.S. EPR FSAR Tier 2, Table 19.1-16 for CDES definitions) with non-zero probability of having at least 1 SGTR.

Using the values of N and Pf and Equation 1, the Poisson distribution representing the probability of **i** or more tubes failing is calculated and the results are presented in Table 19-233-2 and Figure 19-233-2.

Source Term

Approach:

Severe accidents involving SGTR are commonly grouped into three categories:

1. Those caused by spontaneous SGTRs that result in core damage.
2. Those caused by another type of accident (e.g., steam line break) with induced tube ruptures prior to core damage.
3. Those that progress to core damage prior to induced tube rupture.

Categories 1 and 2 involve SGTR prior to core damage and are accounted for in the Level 1 quantification; Category 3 involves SGTR as a result of core damage and is considered in the Level 2 quantification.

Evaluation:

Release Category RC 702 is the result of the three categories of severe accident sequences defined above. Each of these sequences has different conditional probabilities of multiple tube ruptures:

- Category 1: Level 1 initiating event spontaneous SGTR, only one tube rupture is postulated.
- Category 2: Level 1 pressure-induced SGTR, the distribution is 50 percent one tube rupture, 50 percent two to ten tube ruptures (see the Response to RAI 2, Question 19-35).
- Category 3: Level 2 creep-induced SGTR, the Poisson distribution is used.

For each phenomenon contributing to RC 702, the fractional contribution to the total (internal, fire, and flood) release frequency is multiplied by the probability of a given number of tube ruptures. The result is a distribution of the frequencies of multiple SGTRs for RC 702.

Table 19-233-3 summarizes the contributions to RC 702 from the multiple SGTR distributions considered from either Level 1 or Level 2.

Table 19-233-4 summarizes the source term obtained for multiple tube ruptures for three fission products: Iodine, Cesium, and Tellurium. These fission products are used in the large release frequency (LRF) definition and are used here to illustrate how the likelihood of multiple tubes ruptures influences the final source term.

When grouping together multiple tubes ruptures ranges, the source term was assigned as follows:

- For the Level 2 creep-induced SGTR, the source term corresponds to the larger number of tube ruptures.
- For the Level 1 pressure-induced SGTR, the source term corresponds to a weighted average of the applicable MAAP outputs.

Table 19-233-5 shows the source terms resulting from multiple tube ruptures weighted by their proportional contributions to RC 702 for Iodine, Cesium, and Tellurium. For each of these fission products, the sum of partial source terms from single and multiple tubes ruptures results in the RC 702 total source term.

The source terms resulting from single and multiple tube ruptures and associated with all nine fission product groups defined in U.S. EPR FSAR Tier 2, Chapter 19 (U.S. EPR FSAR Tire 2, Table 19.1-20) are presented in Table 19-233-6.

The increase in RC 702 source terms is about a factor of 1.5 when considering multiple versus single SGTR for Iodine, Cesium, and Tellurium. Other fission product groups show higher increases up to a factor of 1.9. The relatively low probability of a large number of tubes ruptures leads to a moderate increase in the RC 702 source terms.

Conclusion

The sensitivity analysis of RC 702 source term to multiple tube ruptures shows an increase by a factor within a range of 1.5 to 1.9 compared to source terms obtained from a single tube rupture. This does not impact the definition of LRF because RC 702 is already characterized as LRF. The methodology used to obtain the results presented in this analysis carry several conservative assumptions:

- Likelihood of multiple tube ruptures: The probability Pf of rupture of any given steam generator tube is conservatively kept constant in the Poisson distribution used.
  - The depressurization following any given number of tube ruptures is not accounted for. Instead, the pressure is assumed to be equal to its initial value, even after a number of tube ruptures.
  - This depressurization leads to a lower probability Pf of a given tube rupturing. Instead, Pf is kept constant.
  - At initial conditions of pressure and temperature, hot leg rupture is more likely to occur than SGTR (Reference 2). If a rupture of a number of tubes does not significantly depressurize the primary system, the initial conditions are unchanged and hot leg rupture is more likely to occur before any further tube rupture. In this analysis, no credit is taken for hot leg rupture.
- Source term evaluation.
  - The source terms assigned to a range of multiple tube ruptures were those obtained for the upper bound of the range. For example: for three to five tubes ruptured, the source term was the one obtained for five tubes.

Given the conservatism applied in this analysis and the moderate increase in the source term for RC 702, no modification of the FSAR is necessary.

References:

1. Cizelj and Mavko, “On the Efficiency of Dedicated Maintenance of Steam Generator Tubes,” Annual Meeting of the Nuclear Society of Slovenia, September 1994.
2. EPRI TR-106194-V1, “Steam Generator Management Project: Risks from Severe Accidents Involving Steam Generator Tube Leaks or Ruptures, Volume 1: Risk Assessment,” Final report, October 1997.

FSAR Impact:

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

Table19-233-1—Probabilities of any given SGT failure for the applicable CDES

CDES	Probability of at least 1 SGTR	Probability of a given tube failure Pf
SSD/SLD (0.6” LOCA)	2.86E-01	1.13E-04
SSD/SLD (2” LOCA)	8.40E-01	6.14E-04
TRD	4.00E-04	1.34E-07

Table 19-233-1—Probability distributions for multiple SGTR based on a Poisson distribution

CDES	1 SGTR	2 SGTR	3-5 SGTR	6-10 SGTR	>10
SSD/SLD (0.6” LOCA)	8.41E-01	1.42E-01	1.73E-02	5.32E-06	3.97E-13
SSD/SLD (2” LOCA)	3.49E-01	3.20E-01	3.18E-01	1.34E-02	3.77E-06
TRD	1.00E+00	2.00E-04	2.67E-08	1.42E-20	2.63E-42

Table 19-233-3—Frequencies of multiple tube ruptures

TOTAL (internal, fire and flood) contributions to RC 702	Frequencies of multiple tubes ruptures= FV* conditional probability		Contribution to RC 702
Level 1 Initiating Event SGTR	1 tube	2.75E-09	51.2%
Level 1 pressure-induced SGTR	1 tube	8.46E-10	15.8%
	2-10 tubes	8.46E-10	15.8%
Level 2 creep-induced SGTR	1 tube	4.35E-10	8.1%
	2 tubes	2.56E-10	4.8%
	3-5 tubes	2.26E-10	4.2%
	6-10 tubes	9.40E-12	0.2%
	> 10 tubes	2.63E-15	0.0%

Table 19-233-4—Source term from multiple tube ruptures for I, Cs and Te

TOTAL (internal, fire and flood) contributions to RC 702	Number of tubes ruptured	Source term from MAAP			
		Iodine Release %	Cesium Release %	Tellurium Release %	source term reference
Level 1 Initiating Event SGTR	1 tube	8.4	8.7	13.9	MAAP run 1 tube
Level 1 pressure-induced SGTR	1 tube	8.4	8.7	13.9	MAAP run 1 tube
	2-10 tubes	27.7	30.3	46.7	33% of 2, 5 and 10 tubes
Level 2 creep-induced SGTR	1 tube	8.4	8.7	13.9	MAAP run 1 tube
	2 tubes	14.9	16.1	25.1	MAAP run 2 tubes
	3-5 tubes	28.8	31.0	49.4	MAAP run 5 tubes
	6-10 tubes	39.4	43.8	65.7	MAAP run 10 tubes
	> 10 tubes	48.9	54.0	89.0	MAAP run 100 tubes

Table 19-233-5—RC 702 source term weighted by the probability of multiple tube ruptures for I, Cs and Te

TOTAL (internal, fire and flood) contributions to RC 702	Number of tubes ruptured	Iodine Release %	Cesium Release %	Tellurium Release %
Level 1 Initiating Event SGTR	1 tube	4.3	4.5	7.1
Level 1 pressure-induced SGTR	1 tube	1.3	1.4	2.2
	2-10 tubes	4.4	4.8	7.4
Level 2 creep-induced SGTR	1 tube	0.7	0.7	1.1
	2 tubes	0.7	0.8	1.2
	3-5 tubes	1.2	1.3	2.1
	6-10 tubes	0.1	0.1	0.1
	> 10 tubes	0.0	0.0	0.0
Total Source term		12.7	13.5	21.2

Table 19-233-6—RC 702 source terms from single and multiple SGTR

Source term for RC 702	XE/KR %	I %	Ce %	Te %	Sr %	Ru %	La %	Ce %	Ba %
Single SGTR	11.0	8.4	8.7	13.9	1.2	9.6	0.0	0.2	5.4
Multiple SGTR	16.4	12.7	13.5	21.2	2.2	14.8	0.1	0.4	8.6
Source term ratio multiple to single SGTR	1.5	1.51	1.55	1.52	1.9	1.5	1.8	1.7	1.6

Figure 19-233-1—Primary pressure (Pa) for multiple SGTR runs

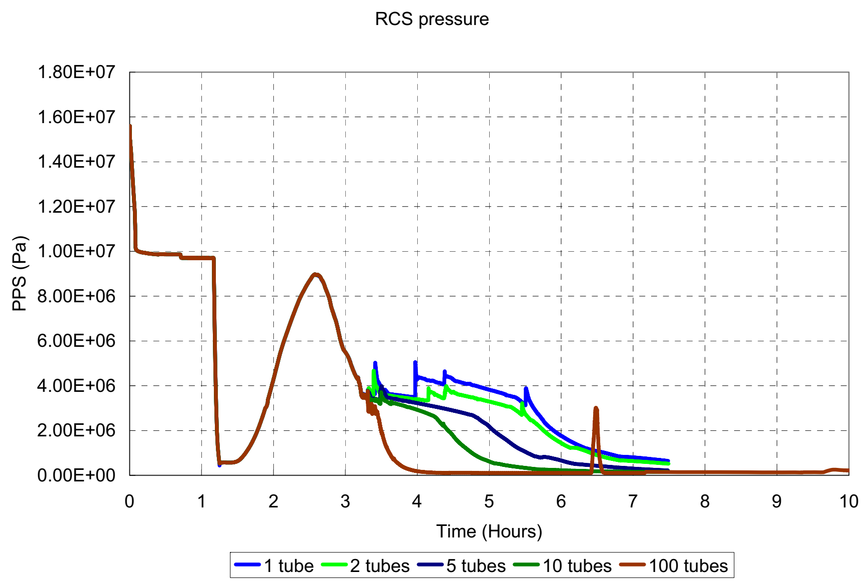
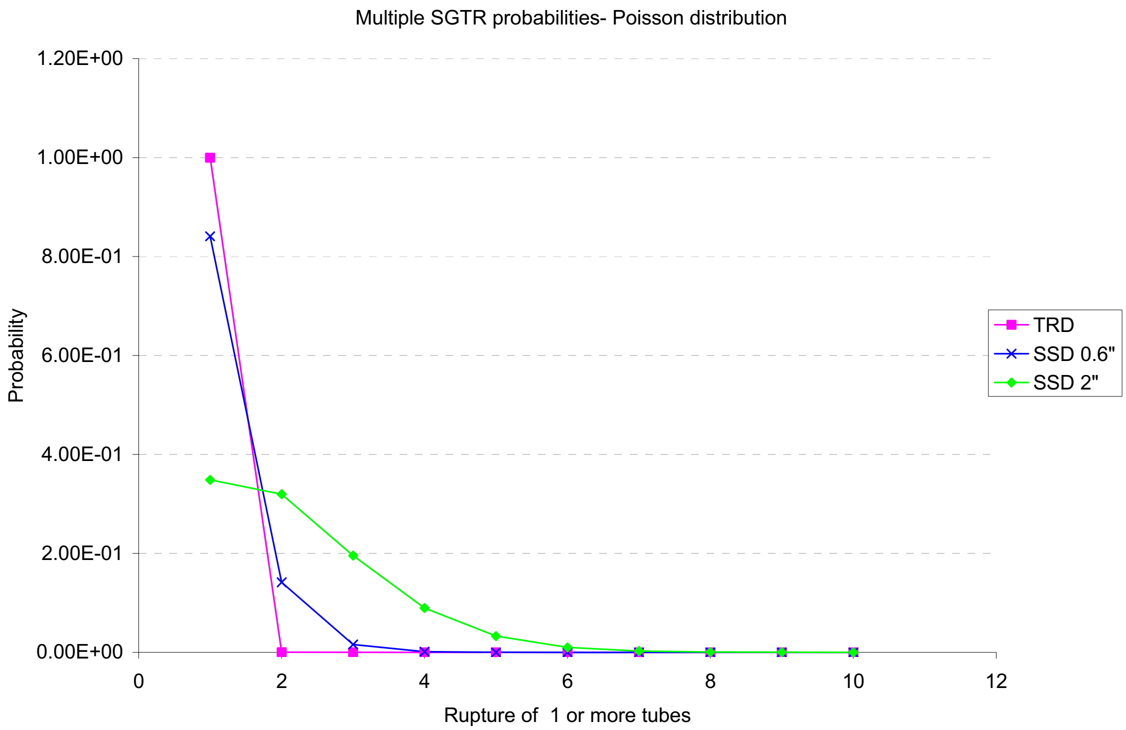


Figure 19-233-2—Multiple SGTR probabilities given at least 1 tube rupture-Poisson distribution



Question 19-238:

(Follow-up to Question 19-121f) The response to Question 19-121f indicates that if the released water/steam through the safety relief valve following a SGTR accident were to be injected (through a specially designed system) back into the containment, the resulting containment pressure would be comparable with containment loads for other severe accident scenarios that have been analyzed (see Figure 19-121-1 and 19-121-3). In fact, the associated loads both in terms of pressure and water loading on the containment should be similar to that of a small steam line break, with a known release path to a specially designed in-containment quench/scrub system. Therefore, at least from the safety point of view, this appears to be a viable alternative that should have been considered as part of the SAMDA cost benefit studies. Please perform an analysis of this SAMDA to determine if it merits implementation into the design, or explain why it is not necessary to do so.

Response to Question 19-238:

In order to determine if venting the main steam safety valves (MSSV) into containment merit implementation, the steam generator tube rupture (SGTR) mitigation strategy, and the design change implementation cost are evaluated. The SGTR mitigation strategy and the design change implementation cost are discussed in the following sections.

SGTR Mitigation Strategy

The U.S. EPR has mitigation features for a SGTR. During a SGTR event, a high main steam line radiation signal or high-high water level signal is used to identify the affected steam generator (SG). To isolate the affected SG, a partial cooldown occurs; the main steam isolation valve (MSIV), main feedwater isolation valve (MFWIV), and SG blowdown isolation valves are closed; and the emergency feedwater (EFW) for the affected SG is isolated. These actions verify that no other sources add to the uncontrolled water level rise and minimize the spread of contaminated fluid. Also, the main steam relief train (MSRT) setpoint for the affected SG is increased to 1,436 psia to minimize the release of contaminated fluid through the MSRT.

The U.S. EPR has the capability to dump the steam to three condenser shells through the main steam bypass valves. With a reactor trip (RT), the combined capacity of the main steam bypass valves sufficiently prevents MSRT or MSSV actuation following a turbine trip or full load rejection. Due to radiological concerns during a SGTR event, it is preferred to steam to the condenser, if available, through the main steam bypass valves instead of the MSRT. A likely scenario for the condenser to be unavailable is during a loss of offsite power (LOOP) event. If the condenser is unavailable, increasing the setpoint of the MSRT will minimize the contaminated release.

The SGTR mitigation strategy for the U.S. EPR was developed to minimize the offsite release through steaming to the condenser instead of the MSRT and by increasing the MSRT setpoint to 1,436 psia on the affected SG.

Implementation Cost

Based on the Response to RAI 6, Question19-121d, the cost to implement an average design change requires a minimum of three months of engineering support. Engineering support involves two engineers working full-time at a rate of $100/hour on the design change. Venting the MSSVs into containment is a significant design change and requires more than three months of engineering support, which places the cost over $100,000. The overall implementation cost, including equipment costs, is higher. The maximum benefit (elimination of all potential risk) is approximately $70,300 with a seven percent discount rate and 2008 replacement power costs. The maximum benefit is less than the overall implementation cost of the design change of venting MSSVs to containment, and venting the MSSVs into containment is not cost beneficial for the U.S. EPR.

Conclusion

The severe accident mitigation design alternatives (SAMDA) candidate, venting MSSVs into containment, does not merit implementation in the U.S. EPR due to the mitigation strategy for SGTR and the implementation cost of the design change. In the SAMDA analysis, the screening of this SAMDA will be changed from not applicable to excessive implementation cost. A revision to the AREVA NP Environmental Report Standard Design Certification (ANP-10290) will be provided by March 30, 2009.

FSAR Impact:

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

Question 19-240:

(Follow-up to Question 19-79)

1. Please confirm the relationship for the stress and Larson-Miller parameter given in Table 19-79-1 of the response to RAI 19-79. Specifically, should LMP be multiplied by the fitting parameter bfit or added to it?
2. More information is needed to clarify the response to question 19-79a, as summarized below.
   - Since cases where the secondary side is depressurized dominate the risk in current PWRs, please run a variation of the MAAP base case with degraded tubes where the steam generators are depressurized and report the damage fractions and the times of predicted hot leg failure, tube failure, and when the core outlet temperature reaches 650 C. Also provide plots of the steam generator tube temperatures.
   - If there are any relevant PRA sequences that would include the 2" diameter breaks, please identify them, and state their core damage frequencies. For MAAP case 1.1i (depressurized SGs), tube failure is estimated to occur about 25 minutes after the core outlet temperature reaches 650 C (11918 sec. vs. 10440 sec.). How does this relatively short time interval affect potential severe accident management high-level actions and strategies? What would be the tube failure times for variations of this case where tubes are damaged, in particular for 1/2, 2/3, and 3/4 through-wall volumetric degradation (MAAP parameter FERSGT=0.5, 0.33 and 0.25). Such damage can realistically be expected to occur from foreign object wear above the top of the tube sheet.
3. The response to Question 19-79e states that system-related top events in the containment event trees are used to model the status of containment isolation, safety injection, severe accident heat removal, and other systems. But FSAR Section 19.1.4.2.1.3 states that the CDES are not, however, directly transferred to Level 2 CETs. Rather, each individual end state is transferred through an intermediate event tree, referred to as CDES link event tree, prior to transfer to a Level 2 CET. Please provide the details of these link event trees, including diagrams, tables of grouping and quantifications, how they are applied to each of the CDESs and how they are linked to the Level 2 CETs that are shown in FSAR Appendix 19C.

Response to Question 19-240:

Response to 19-240, 1:

This question was answered in the Response to RAI 133.

Response to Question 19-240, 2:

MAAP base case with steam generator tube thinning

A variation of the base station blackout (SBO) modular accident analysis program (MAAP) case is run with secondary side depressurized and 50 percent steam generator (SG) wall thinning. Table 19-240(2)-1 shows relevant timings for this run. MAAP predicts hot leg rupture to occur at about 3.8 hours and SG tube rupture after four hours. Even with tube wall thinning, hot leg rupture still precludes SG tube rupture (SGTR).

Figure 19-240(2)-1 shows SG tube creep damage fractions for the unbroken (FCRUHT) and broken (FCRBHT) loops. Figure 19-240(2)-2 shows SG gas and metal tube temperatures for the unbroken and broken loops.

PRA sequences associated with 2 inch LOCAs

The relevant probabilistic risk assessment (PRA) sequences are the 2 inch breaks loss of coolant accidents (LOCAs) and 2 inch reactor coolant pump (RCP) seal LOCAs (a 2 inch seal LOCA designates a seal LOCA with a flow equivalent to a 2 inch break LOCA) with a depressurized secondary side. These sequences are part of the core damage end states (CDES) SLD and SL1D (small LOCA with secondary depressurized) and SSD and SS1D (seal LOCA with secondary depressurized).

The total (internal events, fires, and floods) core damage frequency (CDF) for these CDES is:

- CDES SLD: 5.6E-09/yr.
- CDES SL1D: 2.4E-09/yr.
- CDES SSD: 3.2E-09/yr.
- CDES SS1D: 1.4E-08/yr.

The cumulative CDF for these four CDES is 2.5E-08/yr, which includes internal events, fires, and floods at power.

The range of break sizes included in the definition of small breaks is 0.6 to 3 inches. For the purpose of the Level 2 phenomenological evaluations, it is assumed that half of these breaks are on the lower end of the range (a representative 0.6 inch break diameter assumed) while the other half is on the higher end (a representative 2 inch break diameter assumed).

Therefore, the total CDF of sequences with a 2 inch break or 2 inch-equivalent seal LOCA and secondary side depressurized is 0.5 x 2.5E-08/yr = 1.3E-08/yr.

Impact of short time response on severe accident high level management actions

The severe accident management guidelines for the U.S. EPR will be developed based on the Operating Strategies for Severe Accidents (OSSA) framework, as explained in U.S. EPR FSAR Tier 2, Section 19.2.5.

The OSSA action applicable to the prevention of induced SGTR is the depressurization of the reactor coolant system (RCS) via the primary depressurization system (PDS) valves. Primary depressurization leads to a low pressure difference between the primary and the secondary and effectively precludes induced SGTR. As discussed in U.S. EPR FSAR Tier 2, Section 19.2.5.3, primary depressurization is an immediate action (i.e., it would be performed simultaneously with or immediately following the transition from emergency operating procedures (EOPs) to OSSA).

Therefore, timely operator response performed within the OSSA framework prevents the occurrence of an induced SGTR, including in the specific case discussed.

As stated in U.S. EPR FSAR Tier 2, Section 19.1.5.2, the results of severe accident MAAP calculations are used as an input for the technical bases of the development of the severe accident mitigation guidelines (SAMGs).

The Level 2 PRA considers the small time available in the case of a 2 inch LOCA with secondary depressurized by defining a specific operator action OPF-L2-DEPRESS-25M with a 25 minute time window.

Impact of steam generator tube thinning on time to rupture

Variations of MAAP case 1.1i (2 inch LOCA with secondary side depressurized) are run with SG tube thinning of 50 percent, 67 percent, and 75 percent. The corresponding times of tube rupture are shown in Table 19-240(2)-2, along with the time interval between the core outlet temperature reaching 650°C (1202°F) and the tube rupture. The time interval decreases from 25 minutes to approximately 19 minutes in case 1.1i. The time to rupture does not significantly vary between different values of the wall thinning. This change in time interval would not affect the conclusion that an immediate depressurization performed within the OSSA framework would be an adequate response to prevent induced SGTR.

Response to 19-240, 3:

This question was answered in the Response to RAI 133.

FSAR Impact:

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

Table 19-240(2)-1—Relevant Severe Accident Timings for base case station blackout with secondary side depressurized and degraded steam generator tube walls

MAAP Case Characteristics	Core Outlet Temperature 650°C	Time Steam Generator Tube Rupture	Hot Leg Rupture
Station blackout, secondary side depressurized, 50% steam generator tube thinning	2.75 hr	4.06 hr (unbroken loop) 4.18 hr (broken loop)	3.79 hr (unbroken loop) 3.80 hr (broken loop)

Table 19-240(2)-2—Relevant Severe Accident Timings for Variations of MAAP Case 1.1i with different wall thinning values

Wall thinning	Time of Core Outlet Temperature 650°C (entry in OSSA)	Time of Steam Generator Tube Rupture	Time Interval between OSSA entry and SGTR
50%	2.88 hr	3.20 hr	19.2 min
67%	2.88 hr	3.19 hr	18.6 min
75%	2.87 hr	3.17 hr	18.0 min

Figure 19-240(2)-1—Steam Generator Tube Damage Fractions for SBO scenario with secondary depressurized and 50% tube wall thinning

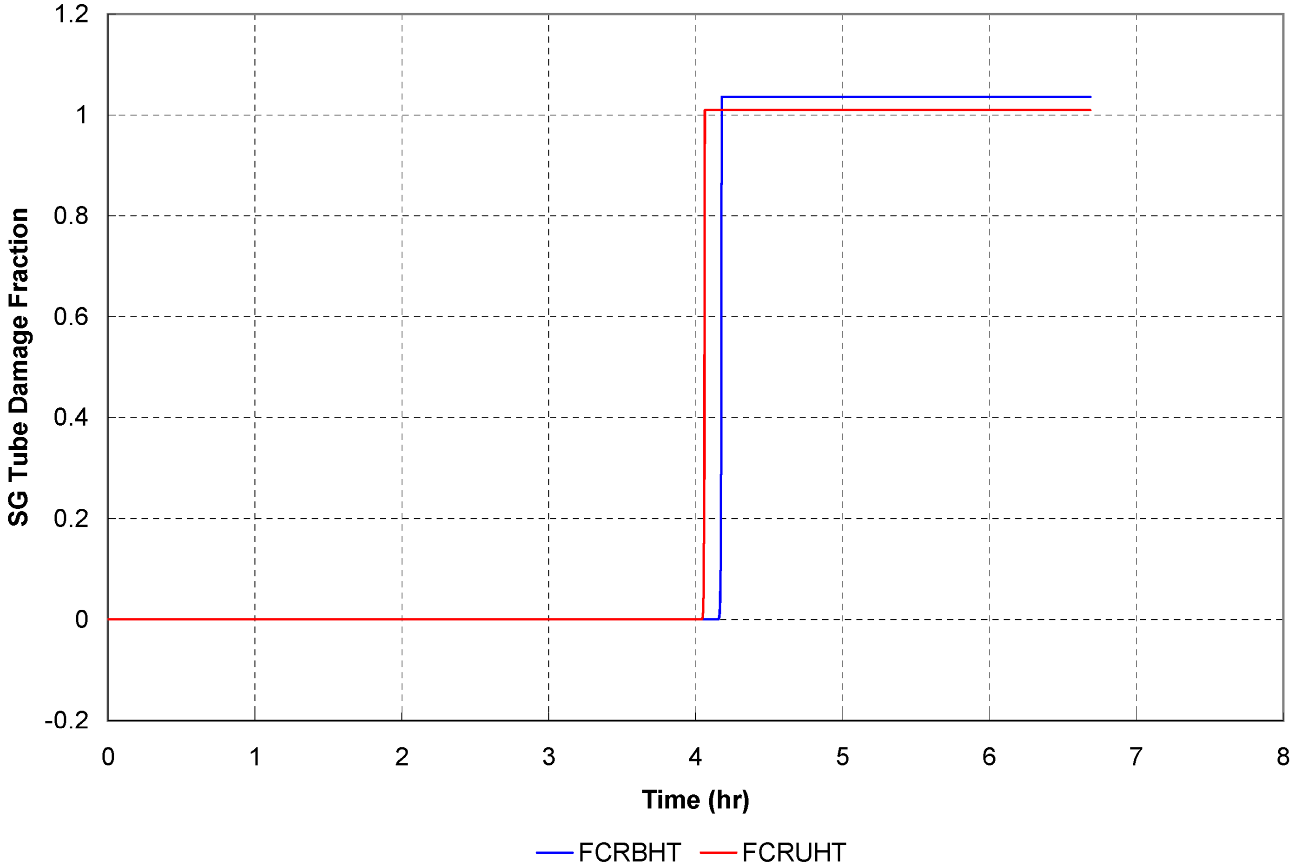


Figure 19-240(2)-2—Steam Generator Tube Temperatures for SBO scenario with secondary depressurized and 50% tube wall thinning