Response to

Request for Additional Information No. 30 Supplement 1, Revision 0

7/29/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.06.05 – Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application Section: FSAR Ch 15
SRSB Branch

Question 15.06.05-1:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," dated November 2007.

Please provide the following information for the EPR Design:

a. Loop friction and geometry pressure losses from the core exit to the steam generators through the cold legs to the inlet nozzle of the reactor vessel. Also, provide the locked rotor RCP k-factor. Please provide the mass flow rates, flow areas, k-factors, hydraulic diameters and coolant temperatures for the pressure losses provided (upper plenum, hot legs, SGs, suction legs, RCPs, discharge legs, and all exit/inlet nozzles). Please also provide the loss from each of the intact cold legs through the annulus to a single broken cold leg.

b. Mixing volume void fraction at two hrs in fig. 2-19 for the limiting large break LOCA.

c. LHSI head flow curve

d. elevation of the top of the core, bottom elevation of the suction leg piping, bottom elevation of the discharge legs, and bottom elevation of the downcomer

Response to Question 15.06.05-1:

a. The U.S. EPR realistic large break loss-of-coolant accident (RLBLOCA) input model uses the following numbering convention:

Components	Description
001 - 099	Reactor Vessel Components
100 - 159	Primary Loop 1 Components
200 - 259	Primary Loop 2 Components
300 - 359	Primary Loop 3 Components
360 - 399	Surge Line and Pressurizer Components
400 - 459	Primary Loop 4 Components
495 - 499	Break Components
500 - 599	Steam Generator 1 Secondary Components
600 - 699	Steam Generator 2 Secondary Components
700 - 799	Steam Generator 3 Secondary Components
800 - 899	Steam Generator 4 Secondary Components
910 - 999	ECCS Components

Realistic Large Break LOCA

Figure 15.06.05-1-1 shows the RLBLOCA reactor coolant system (RCS) primary loop nodalization diagram, Figure 15.06.05-1-2 shows the reactor vessel nodalization diagram, and Figure 15.06.05-1-3 shows the emergency core cooling system (ECCS) nodalization diagram.

Table 15.06.05-1-1 provides the flow areas, hydraulic diameters, and liquid temperatures for the reactor vessel and Loop 1 primary hydrodynamic volumes.

Table 15.06.05-1-2 lists component mass flow rates, flow areas, and the k-factors that are used in pressure loss calculations for the reactor vessel and Loop 1 primary junction components.

The temperatures and mass flowrates in Table 15.06.05-1-1 and Table 15.06.05-1-2 correspond to the ANP-10288P RLBLOCA case steady-state values.

Small Break LOCA

Figure 15.06.05-1-4 shows the small break loss-of-coolant accident (SBLOCA) RCS primary loop nodalization diagram and Figure 15.06.05-1-5 shows the reactor vessel nodalization diagram. The ECCS nodalization shown in Figure 15.06.05-1-3 also applies to the SBLOCA.

Table 15.06.05-1-3 provides the flow areas, hydraulic diameters, and liquid temperatures for the reactor vessel and Loop 1 primary hydrodynamic volumes.

Table 15.06.05-1-4 lists component mass flow rates, flow areas, and the k-factors that are used in pressure loss calculations for the reactor vessel and Loop 1 primary junction components.

The temperatures and mass flowrates in Table 15.06.05-1-3 and Table 15.06.05-1-4 correspond to the ANP-10288P SBLOCA case steady-state values.

Locked Rotor Reactor Coolant Pump K-factor

The homologous pump head curve for the U.S. EPR reactor coolant pumps (RCP) is shown in Figure 15.06.05-1-6.

The K-factor for the RCP locked rotor scenario is given by the following formula:

$$K = (2gA^2 K_{LR} H_r) / Q_r^2$$

Where

A = pump flow area = **[ ]** ft^2

K_{LR} = pump head curve at zero speed = **[ ]** (from Figure 15.06.05-1-6)

H_r = rated pump head = 349.2 ft

Q_r = rated pump flow = 2.7816E+02 ft^3/s

K = **[ ]**

b. See response to RAI 30 Question 15.06.05-6 in NRC:08:059 (August 28, 2008).

c. Information was provided on a CD transmitted to the NRC by AREVA NP letter NRC:06:063 (ADAMS Accession No. ML070160274), (December 21, 2006), CD directory Non-LOCA Base Deck; file eprdc_x_connect_S-RELAP5_input.

d. Relative elevation information was provided to the NRC in AREVA NP letter NRC:06:063, (ADAMS Accession No. ML070160274), (December 21, 2006).

FSAR Impact:

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

Table 15.06.05-1-1—U.S. EPR RLBLOCA Primary Loop Hydrodynamic Volume Parameters
(6 Sheets)

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(6 Sheets)

Table 15.06.05-1-2—U.S. EPR RLBLOCA Primary Loop Junction Parameters (14 Sheets)

Table 15.06.05-1-2—U.S. EPR RLBLOCA Primary Loop Junction Parameters (14 Sheets)

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Table 15.06.05-1-3—U.S. EPR SBLOCA Primary Loop Hydrodynamic Volume Parameters
(6 Sheets)

Table 15.06.05-1-3—U.S. EPR SBLOCA Primary Loop Hydrodynamic Volume Parameters
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Table 15.06.05-1-4—U.S. EPR SBLOCA Primary Loop Junction Parameters (12 Sheets)

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Table 15.06.05-1-4—U.S. EPR SBLOCA Primary Loop Junction Parameters (12 Sheets)

Figure 15.06.05-1-1—U.S. EPR RLBLOCA Primary System Nodalization

Figure 15.06.05-1-2—U.S. EPR RLBLOCA Reactor Vessel Nodalization

Figure 15.06.05-1-3—U.S. EPR ECCS Nodalization

Figure 15.06.05-1-4—U.S. EPR SBLOCA Primary Loop Nodalization

Figure 15.06.05-1-5—U.S. EPR SBLOCA Reactor Vessel Nodalization

Figure 15.06.05-1-6—U.S. EPR Reactor Coolant Pump Homologous Head Curve

Question 15.06.05-5:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," dated November 2007.

Vapor exiting the two-phase surface in the core during the long term contains boric acid. What happens to the boric acid in the vapor as it passes through the steam generators to reach the break? Please discuss the plate-out effects on the RCS internals and the impact on long term cooling.

Response to Question 15.06.05-5:

Until the steam generator secondary side cools down, droplets either evaporate in the flow stream, on the tube surface, or they pass through the steam generator. If the droplets evaporate on the tube surface, the boric acid will plate-out there. If the droplets dry out in the flow stream, the steam will transport the boric acid out of the steam generator tubes to the break. The distribution of boric acid in the steam generator tubes depends on the energy content available in the steam generator's secondary water and tube materials.

The plate-out distribution in the steam generator can be evaluated from the steam generator energy source term distribution. Based on the stored heat capacity of the secondary water and tube metal, the expected plate-out thickness during the cooldown can be calculated. The secondary water will stratify, as the cooling is from the bottom where the droplets and steam enter the tube bundle. The calculated mass of droplet water evaporated equals the energy removed to cool the unit tube cell from 540°F to 212°F, assuming the steam is not superheated as it passes through the steam generator; that is, the steam generator cell energy goes into droplet evaporation on the tube surfaces. The boric acid in the water droplets that evaporate to cool the unit cell would plug about 12 percent of the tube flow area at any location in the steam generator if the droplets were at the boron precipitation limit of 38,500 ppm at 212°F. Based on this conservative distribution model and regardless of the amount of entrainment as a function of time, the steam generator tubes would not be significantly blocked due to boric acid plate-out on the tube surface.

For significant plate-out to occur on the upper internals, an energy source term would be needed. The water leaving the core region is a two-phase mixture and saturated at the core exit pressure. If the water droplets deposit on the upper internals surface and evaporate, the boron concentration in the concentrating volume of water would decrease. The analysis does not consider this decrease in the boron precipitation. As in the steam generator evaluation, the energy content of the upper internals is not sufficient to produce a boric acid film that would cause a significant flow area reduction. If the water droplets were to impinge on a surface that had boric acid deposited on it after the surface cools down, the water would tend to dissolve any deposits because the droplets are below the solubility limit of boric acid for more than two hours post-LOCA.

FSAR Impact:

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

Question 15.06.05-8:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," dated November 2007.

Please describe the tests used to validate the two-phase level swell and boric acid precipitation models.

Response to Question 15.06.05-8:

Level Swell

The S-RELAP5 code has been benchmarked against numerous system and separate effects tests, as described in AREVA document EMF-2103(P)(A), “Realistic Large Break Loss-of-Coolant Accident Methodology for Pressurized Water Reactors” (ADAMS Accession No. ML030760312), April 9, 2003.

S-RELAP5 was used to calculate the water content in the core region that starts at the bottom of the core support plate and ends at the bottom of the hot leg nozzles inside the core barrel. No credit was taken for water from above the bottom of the outlet nozzles to the bottom of the top hat; though this water would also participate in the core region volume. Also, no credit was taken for water in the reactor vessel lower head.

Many tests have been used to verify the S-RELAP5 code relative to level swell, void fraction, or mass retained in the core region: FLECHT Skewed, FLECHT-SEASET, Thermal Hydraulic Test Facility (THTF) level swell at Oak Ridge National Laboratory (ORNL), General Electric (GE) level swell, and FRIGG Loop, are the most notable. Other important tests include those conducted at the Slab Core Test Facility (SCTF), the Cylindrical Core Test Facility (CCTF), and the Upper Plenum Test Facility (UPTF).

The S-RELAP5 analyses of the FLECHT Skewed and FLECHT-SEASET test (low pressure test, representative of a large break loss of coolant accident (LBLOCA)) under-predict the mass inventory in the test section or core region (see Figures 4.61 through 4.69 in EMF-2103(P)(A) Volume 2. EMF-2103(P)(A) Volume 2, page 4-27, concludes:

> “The calculated mass accumulation generally is less than measured. Most of the mass accumulation occurs early in the transient as the lower half of the test section is filled. Once the water accumulation reaches the high power mid-plane region of the test bundle, the water accumulation becomes a balance between injected water entering and entrained and evaporated water leaving.”

Other tests that support the void fraction in the core include the THTF level swell tests at ORNL, the GE level swell, and the FRIGG Loop tests reported in Figures 4.16 through 4.31 of EMF-2103(P)(A) Volume 2. The carry-out from the core to the upper plenum was investigated by analyzing the UPTF tests (see Figures 4.118 and 4.119 in EMF-2103(P)(A) Volume 2). In all cases, S-RELAP5 over-predicted the carry-out to the upper plenum. These tests are at high pressure and are more representative of very small break LOCAs (SBLOCA).

Upper plenum inventory or level was investigated in three test facilities: UPTF, CCTF, and FLECHT-SEASET (see Figures 4.154 through 4.163 in EMF-2103(P)(A) Volume 2). In general, S-RELAP5 under-predicted the upper plenum inventory until the time at which the core concentrating region volume was established for the boron precipitation analysis.

Additional level swell information is available in the RAI response incorporated into EMF-2103(P)(A) Volume 1. The following question and response are on page 151.

> **NRC:02.062 Attachment 1 Question 71a:** Core 3-D Flow and Void Distribution (Page 4-85, EMF 2103). Comparison to the THTF and GE Level Swell data, for example, are high pressure tests and do not represent PWR reflood conditions. The GE data does not apply to rod bundle drag. On the other hand, specific FLECHT boil-off or reflood data are applicable to voids in bundles at low pressure (FLECHT-SEASET Test 35658, for example).
>
> **Response 71a**: In addition to the high pressure void distribution comparisons for the THTF and GE level swell tests, void distribution comparisons have been made for low pressure reflood tests. The void distribution comparisons for FLECHT-SEASET and FLECHT SKEWED reflood tests are represented by the calculated and measured differential pressures between 72 and 84 in. displayed in Figures 3.3.71 to 3.3.79 of EMF-2102(P) Revision 0. Note particularly that code data agreement is excellent after the region between 72 and 84 in. is completely quenched. This demonstrates the wet-wall (pre-CHF) interphase friction model is applicable and adequate for both high pressures (THTF Tests) and low pressures. The void distribution above the quench front is dominated by the dry-wall (post-CHF) interphase friction model. The differential pressure plots show that (1) the calculated liquid fraction far above the quench front is lower than the measured and (2) the calculated liquid fraction near the quench front is higher than the data for injection rates below 1.5 in/sec and lower than the data for injection rates higher than 3.0 in/sec. This void distribution pattern is consistent with the clad temperature history at around the 78 in. elevation.

In conclusion, the S-RELAP5 analyses of a wide variety of tests indicate that the quantity of water in the core concentrating region is under-predicted at the time selected to calculate the volume of water to be used in the boron precipitation analysis. In particular, the FLECHT Skewed and FLECHT-SEASET test (low pressure test) analyses demonstrate that S-RELAP5 under-predicts the water inventory out to 200 seconds for the LBLOCA.

Boric Acid Precipitation Model

The boron model in S-RELAP5 was not used; S-RELAP5 was only used to determine the concentrating water volume at 200 seconds, and then later to show that the switch to simultaneous injection for the 4.1 ft^2 LBLOCA would result in dilution of the boron concentration accumulated before the switch. U.S. EPR FSAR, Tier 2, Section 15.6.5.4.1.2 will be revised to clarify that the LBLOCA case analyzed was for the 4.1 ft^2 case. The precipitation model is based on the mixed mean boron concentration of the reactor coolant system (RCS), extra borating system (EBS), accumulators, and in-containment refueling water storage tank (IRWST) that gives the highest concentration. The inlet concentration is not decreased as a function of accumulation in the core region. The precipitation concentration is based on the 212°F mixing limit for injection from the IRWST, when the IRWST solution is at the technical specification limit of 59°F and the mixed mean boron concentration is at 1929 ppm. To remain conservative, the model neglects:

- The increase of water inventory with time in the concentrating region.
- The subcooling of the water entering the core concentrating region.
- Boron solubility in steam.
- Droplet entrainment leaving the core concentrating region and removing boron.
- Plate-out on the upper internals surfaces in the core concentrating region due to droplet evaporation.
- Increased solubility of boron due to other solutes.
- Increase in boiling temperature due to boric acid concentration.
- Water in the upper plenum above the bottom of the outlet nozzles.

The decay heat is based on the draft ANS 5.1 1971 standard with a multiplier of 1.2 for the fission product decay heat, as approved for the existing fleet of operating plants.

See also the response to RAI No. 30, Question 15.06.05-10 for combination of inlet concentration based on pumped flow rates.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.6.5.4.1.2 will be revised as described in the response and indicated on the enclosed markup.

Question 15.06.05-10:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," date November 2007.

Page 2-7 describes four sources of boric acid in the EPR design. Since following a large break LOCA very little liquid will remain in the vessel, combining the low concentration core boric acid with the other sources does not appear appropriate. Please discuss the effect of excluding the initial RCS concentration from the boric acid calculations. Also computing a mixed mean boric acid concentration to determine precipitation time does not take into account that some of the higher concentration sources could inject at much higher flow rates. As such, please show the concentration vs. time in Fig. 2-19 assuming that the EBS injects alone followed by only the IRWST as sources.

Response to Question 15.06.05-10:

The maximum flow rate of the extra borating system (EBS) is 111 gpm, with both trains injecting. At this water injection rate, the EBS can supply only a fraction of the water being evaporated. It takes about 2.9 hours to pump the inventory of the EBS pools into the reactor coolant system (RCS) through the RCS cold leg emergency core cooling system (ECCS) nozzles using the medium head safety injection (MHSI) and low head safety injection (LHSI) systems.

If the RCS water is not considered in the mixing, then the water entering the core region is a function of the various systems' pumping capacity. Boron concentrations for several pump combinations are shown in Table 15.06.05-10-1.

The design basis analysis is based on the automatic start of two MHSI and two LHSI trains. The EBS, although safety grade, is started manually by the operator. With only the accumulators, the LHSI, and MHSI injecting, the maximum concentration in the injected water is 1900 ppm. This is less than the mixed mean generated by mixing all of the water available, and is the most likely scenario because MHSI and LHSI start automatically. If the operators also start one or two EBS trains, then other mixture combinations may be generated.

Potential pumped mixture combinations are presented in Table 15.06.05-10-1. If the concentrations are a mixture of the boron concentration of the in-containment refueling water storage tank (IRWST) and the EBS pools with the flows established by the pumps, then the effective concentration entering the RCS through the ECCS cold leg inlet nozzle could increase approximately 0.1% if the operators started one EBS pump immediately. However, this increase is compensated in the core region by starting at the system concentration of 955 ppm, rather than the 1929 ppm mixed mean concentration. The most that the operators could inject is two trains of the EBS with the entering concentration of 1984 ppm. The operators would still have more than two hours to switch LHSI injection to the hot legs if they started two EBS trains, as shown in Figure 15.06.05-10-1.

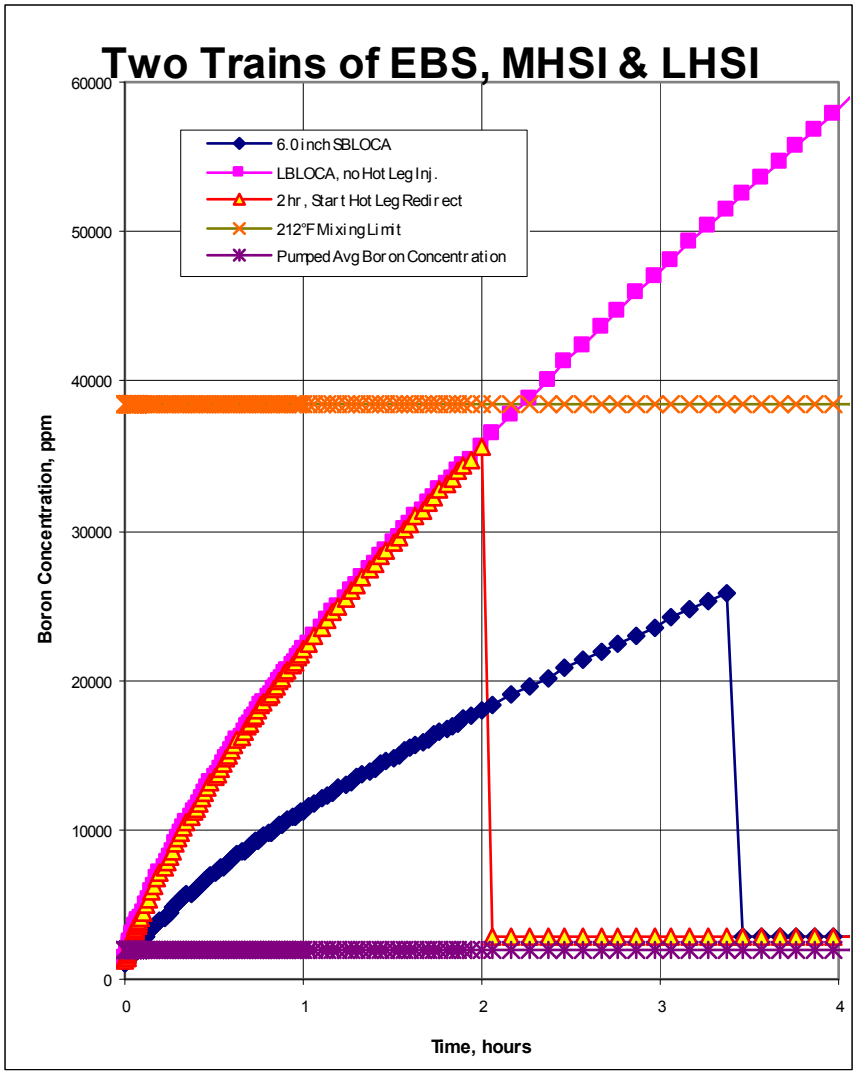
FSAR Impact:

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

Table 15.06.05-10-1—Mixture Combinations

No. of EBS Pumps (55.4 gpm per pump)	**No. of MHSI Pumps** (1053 gpm per pump)	**No. of LHSI Pumps** (2471 gpm per pump)	**Total Flow** (gpm)	**Concentration** (ppm)	**Deviation** (%)
2	0	0	111	7300	Inadequate Core Cooling
1	2	2	7103	1942	0.07
2	2	2	7159	1984	2.80

Figure 15.06.05-10-1—Boron Concentration vs. Time



Question 15.06.05-11:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," dated November 2007.

Discuss the means by which the boric acid from the high concentration EBS tanks can be injected into the RCS? Could this be the sole source of injection once the accumulators empty? Please explain

Response to Question 15.06.05-11:

The extra borating system (EBS) is a safety-grade system that requires operator action to start the injection into the reactor coolant system (RCS) cold legs. The EBS pool volume is such that it would take the EBS pumps, operating at their maximum flow rate, about 2.9 hours to pump all of the water inventory into the RCS. The EBS contains sufficient boron to keep the reactor subcritical during cooldown at the RCS cooldown rate of 90°F/h using the main steam relief train (MSRT). Operation of the EBS will be included in plant operating procedures and the emergency operating procedures.

For loss of coolant accidents (LOCA) that empty the accumulators, all four trains of the medium head safety injection (MHSI) and the low head safety injection (LHSI) systems would have to fail for the EBS injection pumps to be the only source of water injection into the RCS. Thus, it is not postulated that the EBS could be the only solution injection source. For the very small breaks, the continued cooldown of the RCS using the MSRT limits boron concentration build-up and dilution within acceptable limits. The MSRT does this by reducing RCS pressure until the MHSI and the LHSI (for the larger, small breaks) provide adequate injection to overcome the break flow. The continued cooldown allows the RCS to return to single phase natural circulation.

FSAR Impact:

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

Question 15.06.05-13:

The staff submits the following Request for Information (RAI) based on the review of ANP-10288P entitled "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," dated November 2007.

The switch to simultaneous injection is identified to be 2 hrs post-LOCA. Is this switch time to be identified in the EOPs? Since the precipitation limit is reached about 10 minutes later, please demonstrate that the switch can be performed in less than 10 minutes.

Also, at 2 hrs post-LOCA, what is the entrainment rate in the hot legs due to steaming from the core at the lowest achievable RCS pressure following the limiting LBLOCA? Please justify and demonstrate that the hot leg injection is not entrained into the hot legs and steam generators at 7200 seconds. Please also justify the maximum elevation of the two-phase level in the hot leg at this time so that the appropriate vapor velocity (used in the entrainment calculation) in the upper portion of the hot leg can be determined.

Response to Question 15.06.05-13:

The time to switch to simultaneous injection will be addressed in the emergency operating procedures (EOP) and will be consistent with the safety analysis case. The analyses described in Technical Report ANP-10288P are conservative relative to the boron concentration in the core region. However, even with the conservatisms, the operators would have up to two hours to switch to simultaneous injection. Because the switch can be made from the control room, the switch takes less than 10 minutes.

An analysis of a large break loss of coolant accident (LBLOCA) with medium head safety injection (MHSI) and low head safety injection (LHSI) trains injecting into loops 2 and 3 shows a slight drop in the collapsed liquid level in the core with the switch of about half of the LHSI flow to the hot leg as simultaneous injection. The analysis also shows that the penetration of the hot leg injection into the upper plenum and down through the reactor core will prevent accumulation of boron in the core region within about 200 seconds after simultaneous injection is initiated. The recirculation in the core region following the switch to simultaneous injection increases with the amount of water carried up through the hotter regions and down through the colder outer lower power region. The 2000 seconds for the LHSI switch used in the analyses is 200 seconds more than the 1800 seconds minimum elapsed time for a credited operator action.

A realistic LBLOCA case was extended to about 6000 seconds, with simultaneous injection starting at 2000 seconds. Because decay heat decreases with time, an early start increases the likelihood that the injection water will be carried into the steam generators. With half of the injected LHSI water going into the hot leg, the water begins to drain back into the upper plenum within 100 seconds. The water entering the upper plenum is mixed in the two-dimensional upper plenum model and is carried down through the outer core low power region. Some of the water that is heated with the mixing and down-flow through the core goes into the lower plenum and then into the lower head and into the downcomer. In addition, some energy is carried out with the backflow of water from the core concentrating region, reducing the quantity of steam generated in the core region. This reduces the quantity of steam that must be vented around the loops. The void fraction in the reactor coolant system (RCS) pipe entry node of the hot legs that is receiving simultaneous injection goes from about 60 percent to about 80 percent. As the steam flow decreases with time, the void fraction starts to decrease in that node at about 5000

seconds. The void fraction in the RCS pipe entry node of the loops that are not receiving LHSI injection in the hot leg increases much less, holding at about 60 percent, and starts to decrease more rapidly around 5000 seconds.

FSAR Impact:

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

Question 15.06.05-21:

According to FSAR Figure 15.6-83 the highest PCTs for non-LOOP cases are the 3½" break as well as the 5½" - 6½" breaks. The SG nodding study presented in ANP-10291P showed that PCT for 4" break increased 121 K when more axial nodes were used. If this same result applied to non-LOOP cases, a different size break might then be the limiting case. Please justify not using detailed SG axial nodalization as given in ANP-10291P for the nodalization used in the FSAR analyses.

Response to Question 15.06.05-21:

An analysis of small break loss of coolant accident (SBLOCA) cases with no loss of offsite power (LOOP) and using a detailed steam generator (SG) axial model shows consistent peak cladding temperature (PCT) and system behavior when compared to the FSAR non-LOOP SBLOCA cases using a coarser SG model. Thus, the SG nodalization used for the SBLOCA U.S. EPR FSAR analysis is adequate.

A mini-SBLOCA spectrum analysis was performed using the detailed SG nodalization model presented in ANP-10291P, "Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology Technical Report," November 2007. Table 15.06.05-21-1 summarizes the break spectrum results for PCT and loop seal clearing times for the two sets of SG models. The results show that the limiting case for the non-LOOP break spectrum using the detailed SG nodalization changed from 6.0 inch to 6.5 inch; however, it still remained in the 5.5 inch to 6.5 inch region and it did not switch to a smaller size break.

Comparison plots for the 6.0 inch break and 6.5 inch break show consistent behavior between the two models, for PCT (Figure 15.06.05-21-1 and Figure 15.06.05-21-2), reactor coolant system and reactor vessel mass inventories (Figure 15.06.05-21-3 and Figure 15.06.05-21-4), and pressurizer (PZR) / secondary pressures (Figure 15.06.05-21-5 and Figure 15.06.05-21-6).

FSAR Impact:

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

Table 15.06.05-21-1—Non-LOOP Break Spectrum Results

Break Size (in)	**Case**	**PCT** (°F)	**PCT Time** (sec)	**PCT Node**	**Loop Seal Clearing Times** (sec)			
					Loop 1	**Loop 2**	**Loop 3**	**Loop 4**
3.5	FSAR	1504	1469.90	33	1480	725	725	1480
	Detailed SG Model	1376	1517.60	33	1900	1500	1500	700
4.0	FSAR	1505	1084.60	33	1100	550	550	1100
	Detailed SG Model	1499	1083.00	33	1100	550	550	1100
5.5	FSAR	1572	521.25	33	520	310	310	520
	Detailed SG Model	1365	531.46	33	300	300	300	300
6.0	FSAR	1585	428.02	33	426	280	280	280
	Detailed SG Model	1604	424.48	33	430	270	270	270
6.5	FSAR	1577	358.17	31	360	240	240	240
	Detailed SG Model	1627	351.60	31	350	250	350	250
7.0	FSAR	1524	305.71	31	215	215	215	215
	Detailed SG Model	1494	310.15	31	220	220	220	220

Figure 15.06.05-21-1—PCT for the 6.0 Inch Breaks

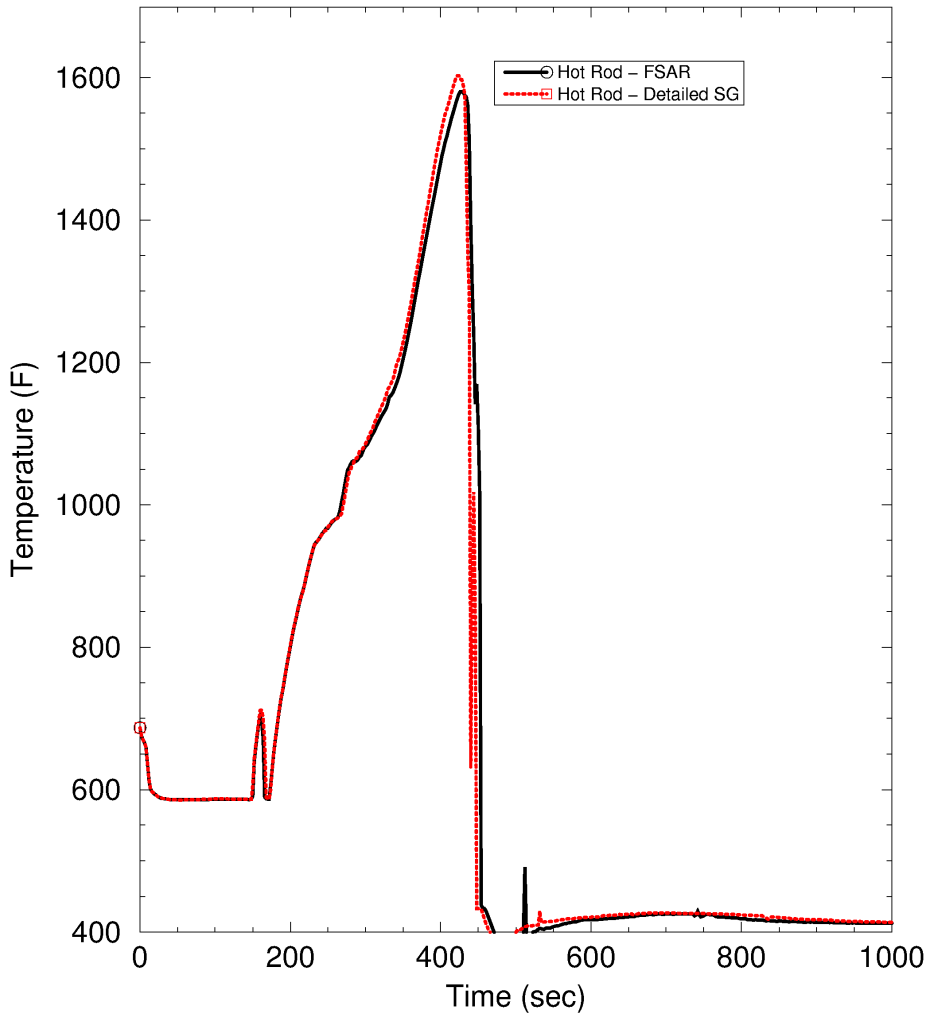


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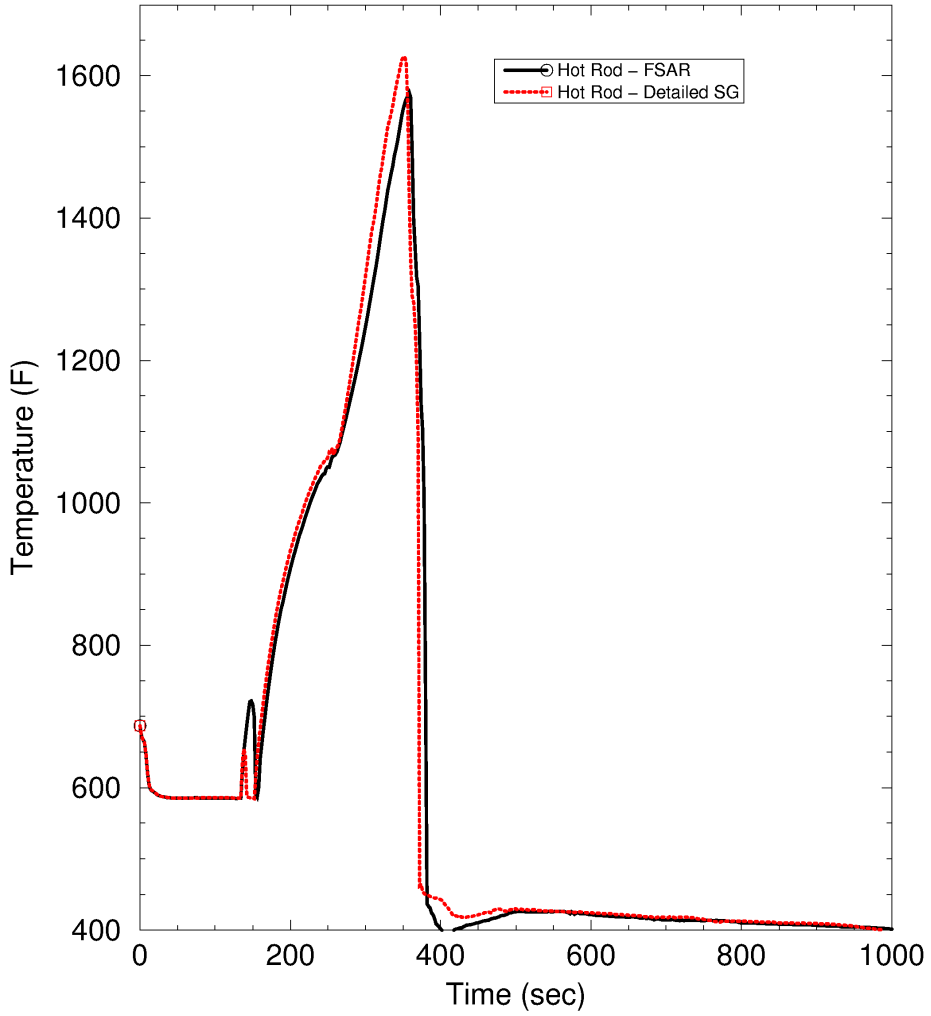
Figure 15.06.05-21-2—PCT for the 6.5 Inch Breaks



ID:08844 2Aug2007 19:42:24 6_5in_001_pump.dmx:1

Figure 15.06.05-21-3—Mass Inventory for the 6.0 Inch Breaks

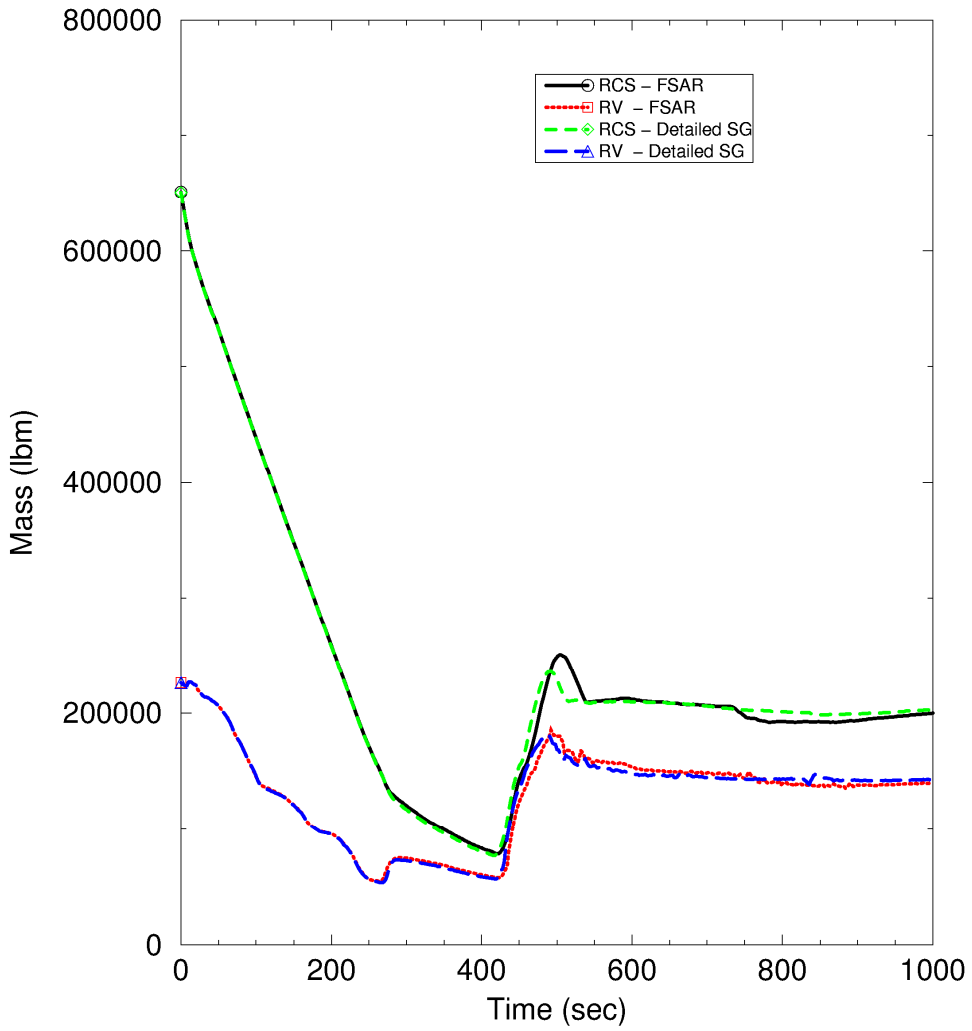


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Figure 15.06.05-21-4—Mass Inventory for the 6.5 Inch Breaks

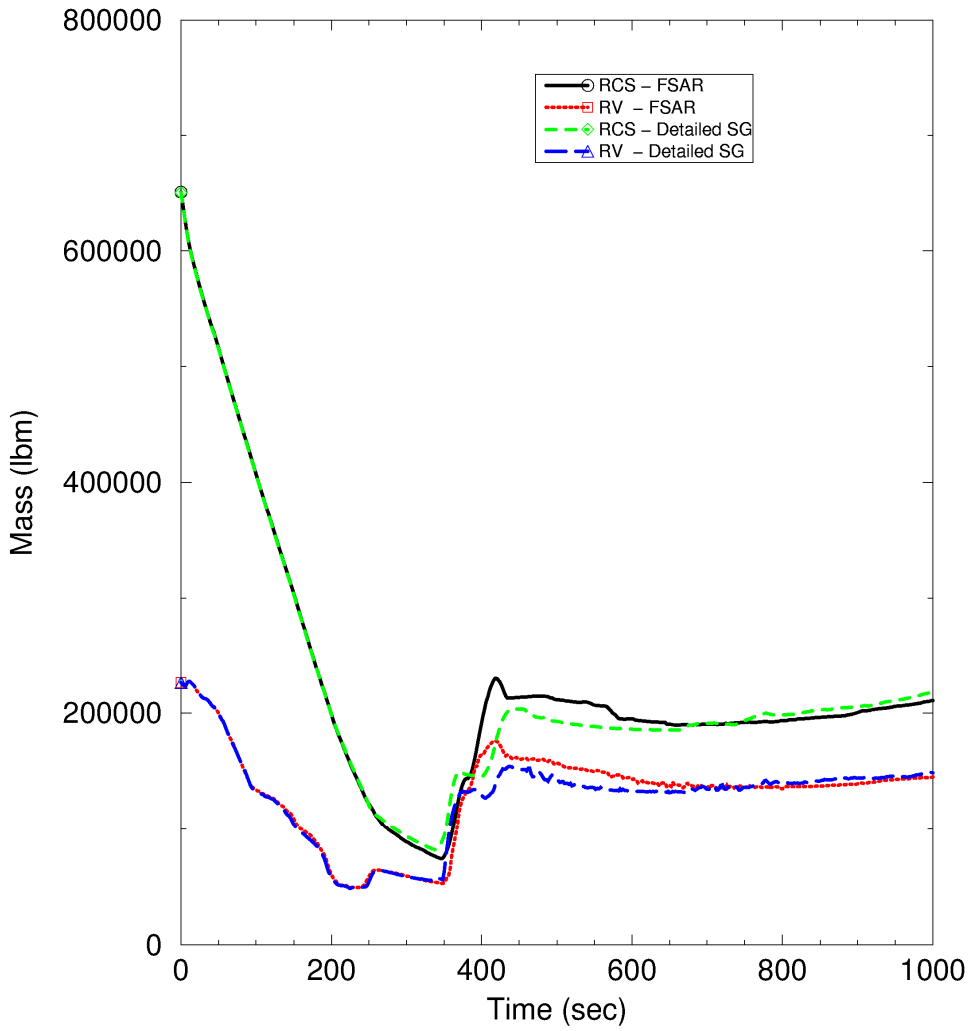

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Figure 15.06.05-21-5—Primary and Secondary Pressure for the 6.0 Inch Break

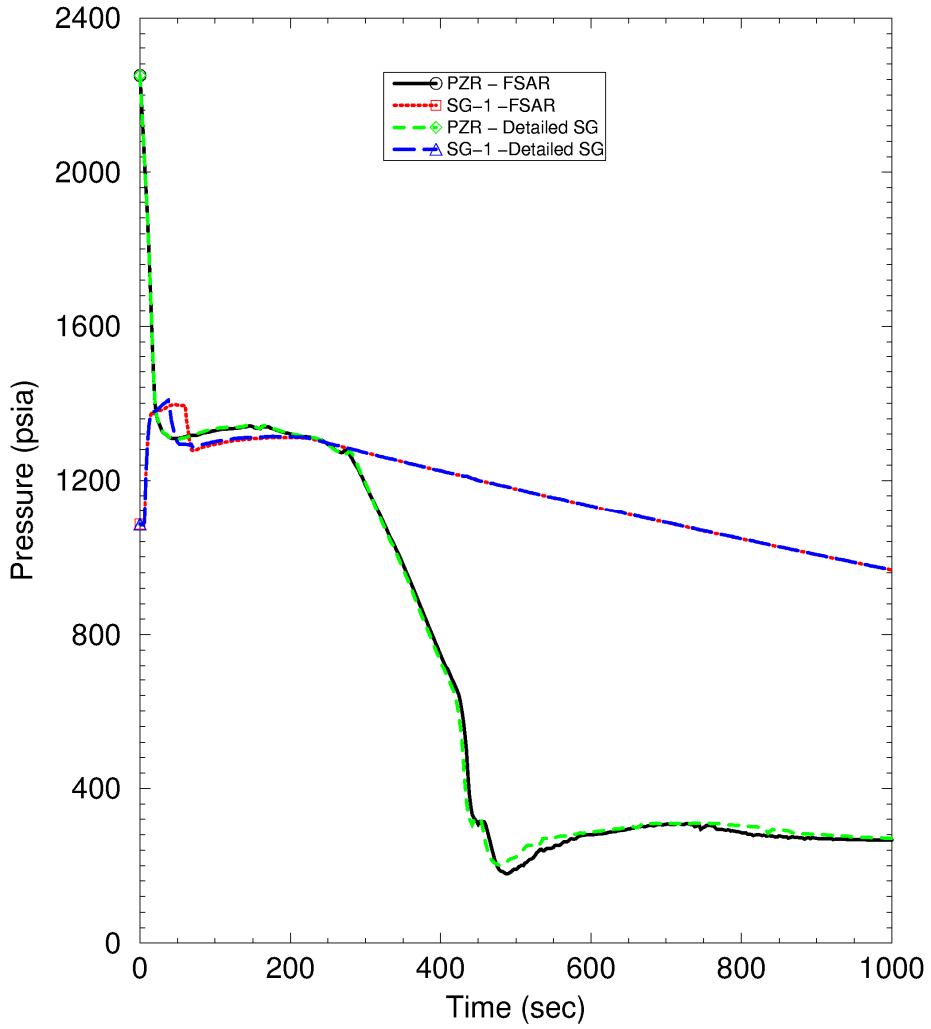

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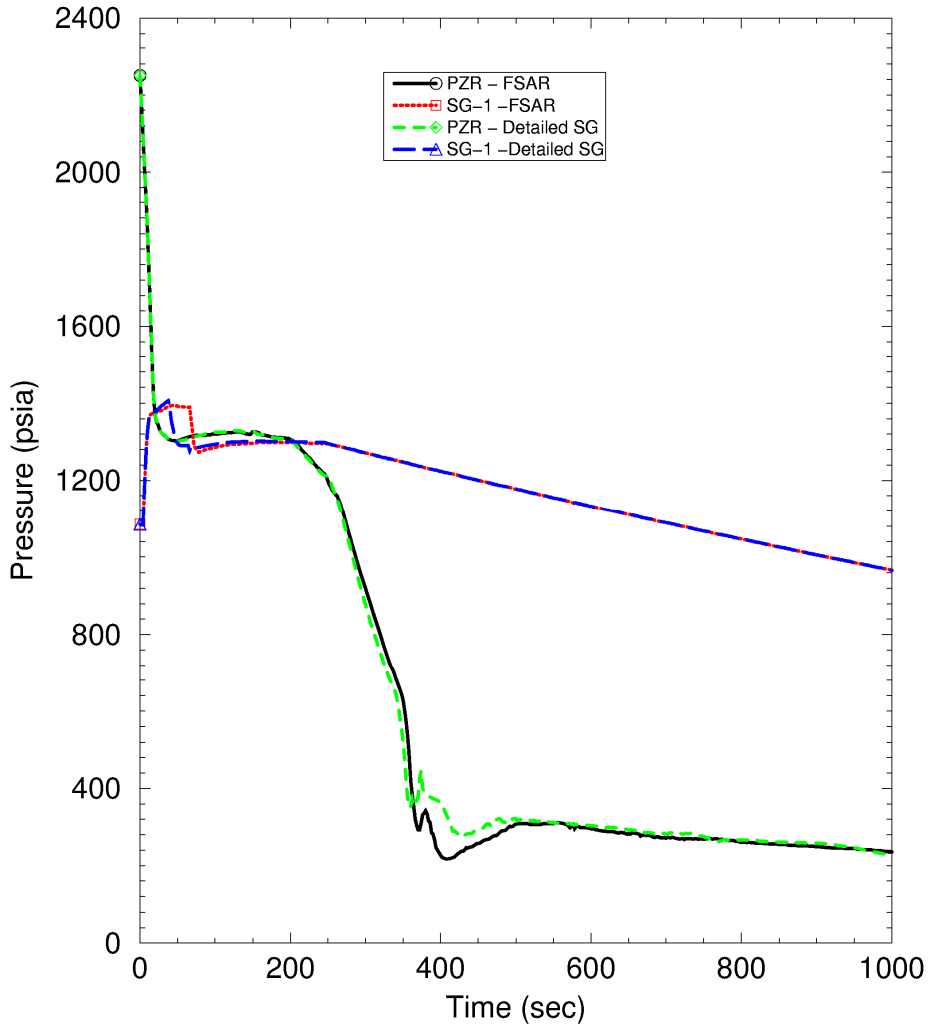
Figure 15.06.05-21-6—Primary and Secondary Pressure for the 6.5 Inch Breaks



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Question 15.06.05-22:

Please provide the methodology to determine that the single failure of an EDG is worse than the failure of the MSRCV in the broken loop SG.

Response to Question 15.06.05-22:

A small break loss of coolant accident (SBLOCA) single failure analysis showed that the failure of the broken loop steam generator (SG) main steam relief train (MSRT) to open, allows for higher primary system inventory and core level, and results in lower peak cladding temperature (PCT) than the failure of one emergency diesel generator (EDG) to start. Thus, the failure of one EDG to start is worse than the failure of the broken loop MSRT to open.

A single failure (SF) analysis was performed for the limiting SBLOCA case, the 6.5 inch break at reactor coolant pump (RCP) discharge, with loss of offsite power (LOOP) assumed at reactor trip. The sensitivity analysis assumed that the MSRT in SG- 4 (broken loop) is unavailable for heat removal due to the failure of the main steam relief isolation valve (MSRIV) to open. The failure of the MSRIV to open is assumed instead of the main steam relief control valve (MSRCV) failing closed. This assumption is bounding, because the MSRCV is normally opened and it will allow some energy removal before it closes, whereas the MSRIV is normally closed. Therefore, for this sensitivity case, three safety injection (SI) trains and three emergency feedwater (EFW) trains are available for recovery (with one emergency diesel generator (EDG) in preventive maintenance). Table 15.06.05-22-1 and Figures 15.06.05-22-1 through 15.06.05-22-3 present a comparison between the MSRT-4 SF case and the limiting SBLOCA case presented in the U.S. EPR FSAR, which assumes the failure of one EDG to start as the SF.

A PCT of 1551°F is calculated for the MSRT-4 SF case, which is bounded by the PCT of 1638°F obtained for the one EDG SF case (Figure 15.06.05-22-1). The MSRT-4 SF case allows for a larger amount of cooling SI water to enter the primary system (three SI trains versus two SI trains), resulting in a higher mass inventory present in the core (Figure 15.06.05-22-2). The primary / pressurizer (PZR) pressure decouples from the secondary pressure before the PCT is reached (Figure 15.06.05-22-3). Later the unavailability of one SG for heat removal becomes less significant. The loop-clearing history is similar for both SF cases, with the clearing sequence slightly different but with all loops clearing before 360 seconds for each case.

An additional MSRT-4 SF case with a smaller break size, 4.0 inch inner diameter, was analyzed. Smaller breaks depressurize slower and thus they rely longer on the SGs to remove primary energy and depressurize to the medium head safety injection (MHSI) injection setpoint. Table 15.06.05-22-1 and Figures 15.06.05-22-4 through 15.06.05-22-6 present a comparison between the 4.0 inch MSRT-4 SF case and the 4.0 inch EDG SF case. Figure 15.06.05-22-4 shows that a boil-off PCT does not occur for this break size. This is attributed to the larger core inventory (Figure 15.06.05-22-5) caused by the increased SI flow injection, combined with less inventory lost. The PCT of 755°F occurred prior to the loops seal clearing and it remains bounded by the base 4.0 inch case, which had a PCT of 1088°F.

FSAR Impact:

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

Table 15.06.05-22-1: Break Spectrum Results

Break Size (in)	Single Failure **Assumption**	**PCT** (°F)	**PCT Time** (sec)	**PCT Node**	**Loop Seal Clearing Times** (sec)			
					Loop 1	**Loop 2**	**Loop 3**	**Loop 4**
6.5	One EDG	1638	360.0	31	360	235	235	360
	MSRT-4	1551	358.4	33	350	250	350	250
4.0	One EDG	1088	1222.3	31	1260	502	502	1260
	MSRT-4	722	490.3	31	1150	500	650	1150

Figure 15.06.05-22-1—PCT for the 6.5 Inch Breaks

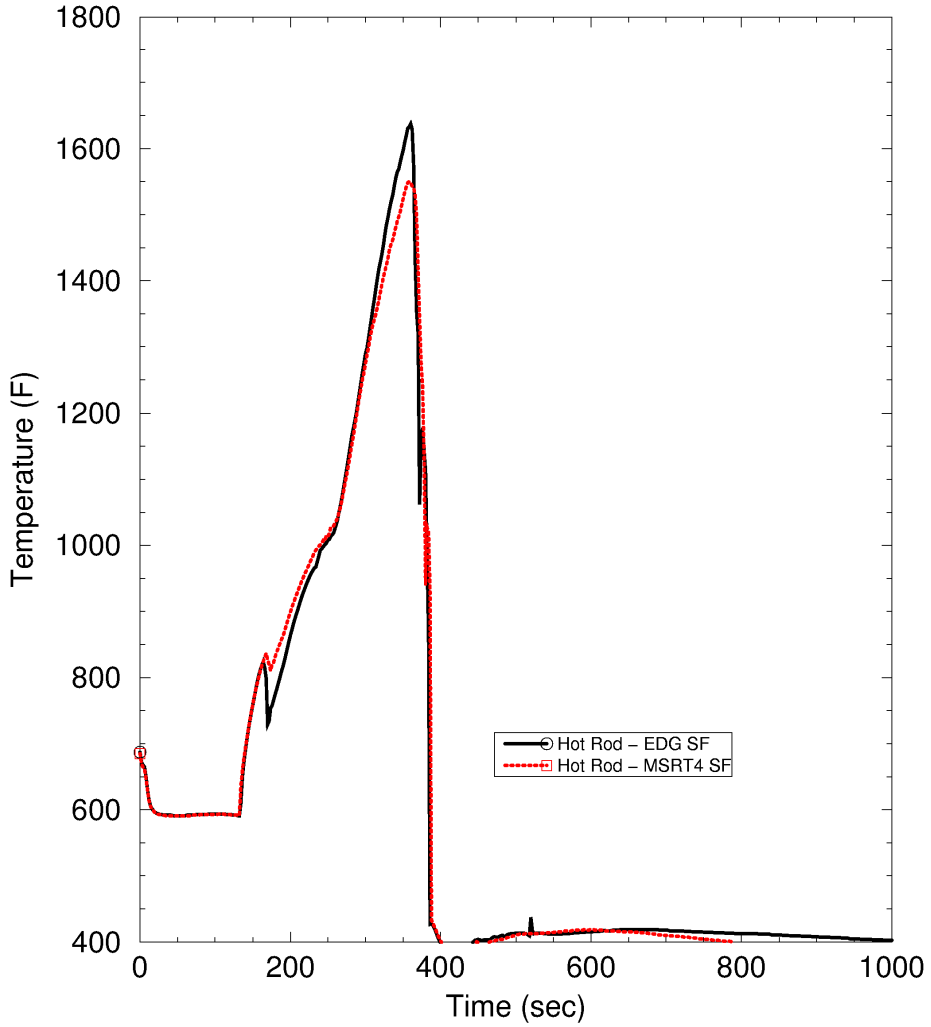

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Figure 15.06.05-22-2—Hot Assembly Mass Inventory for the 6.5 Inch Breaks

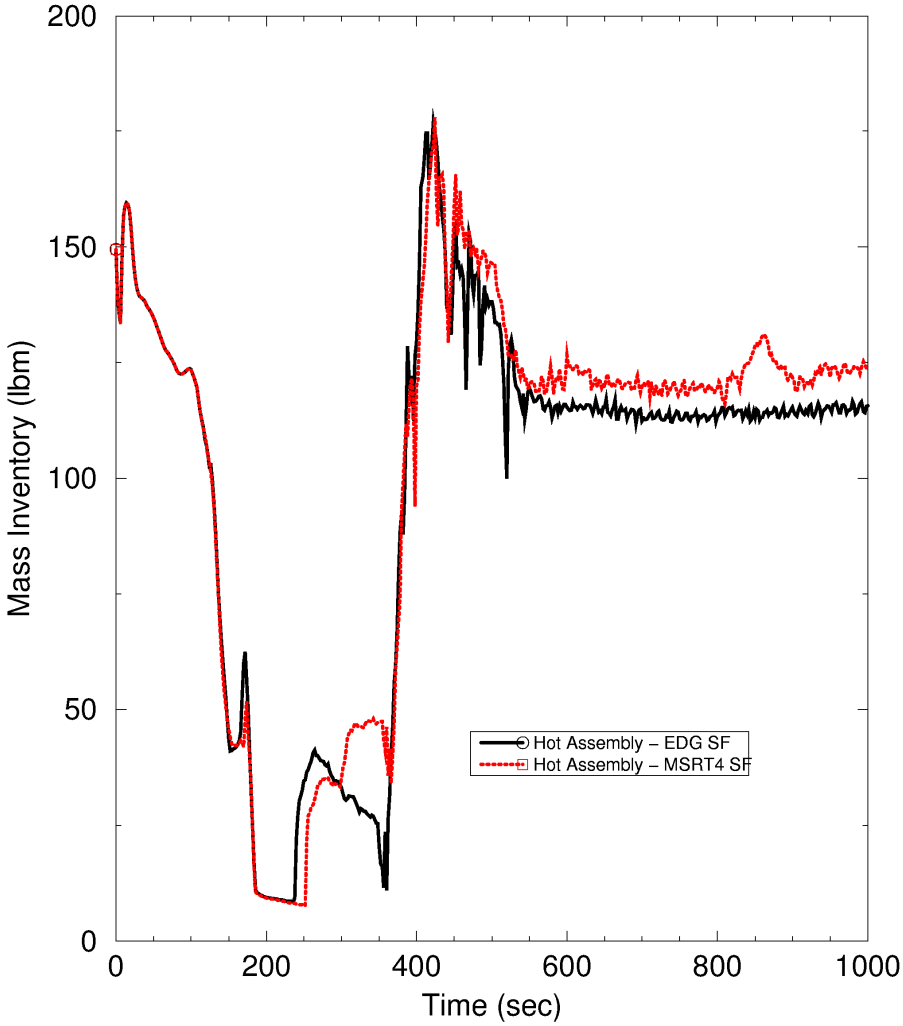

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Figure 15.06.05-22-3—Primary and Secondary Pressure for the 6.5 Inch Breaks

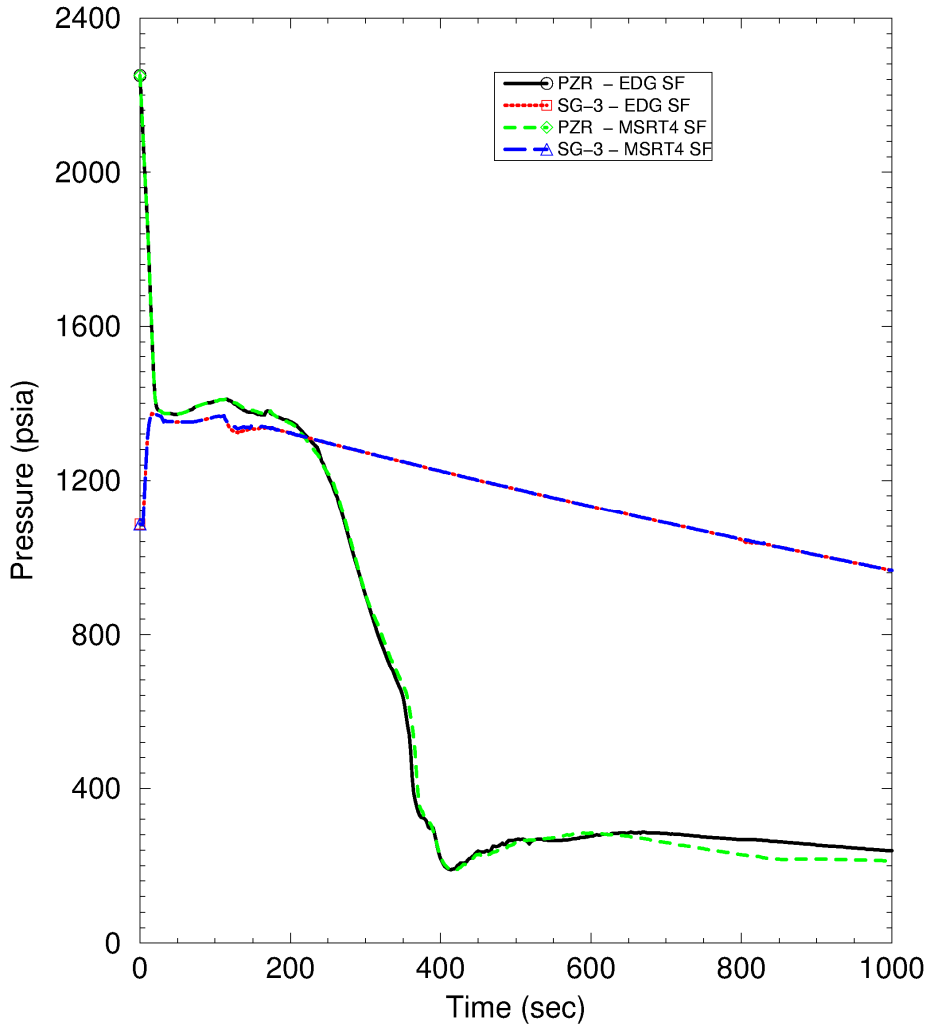

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Figure 15.06.05-22-4—PCT for the 4.0 Inch Breaks

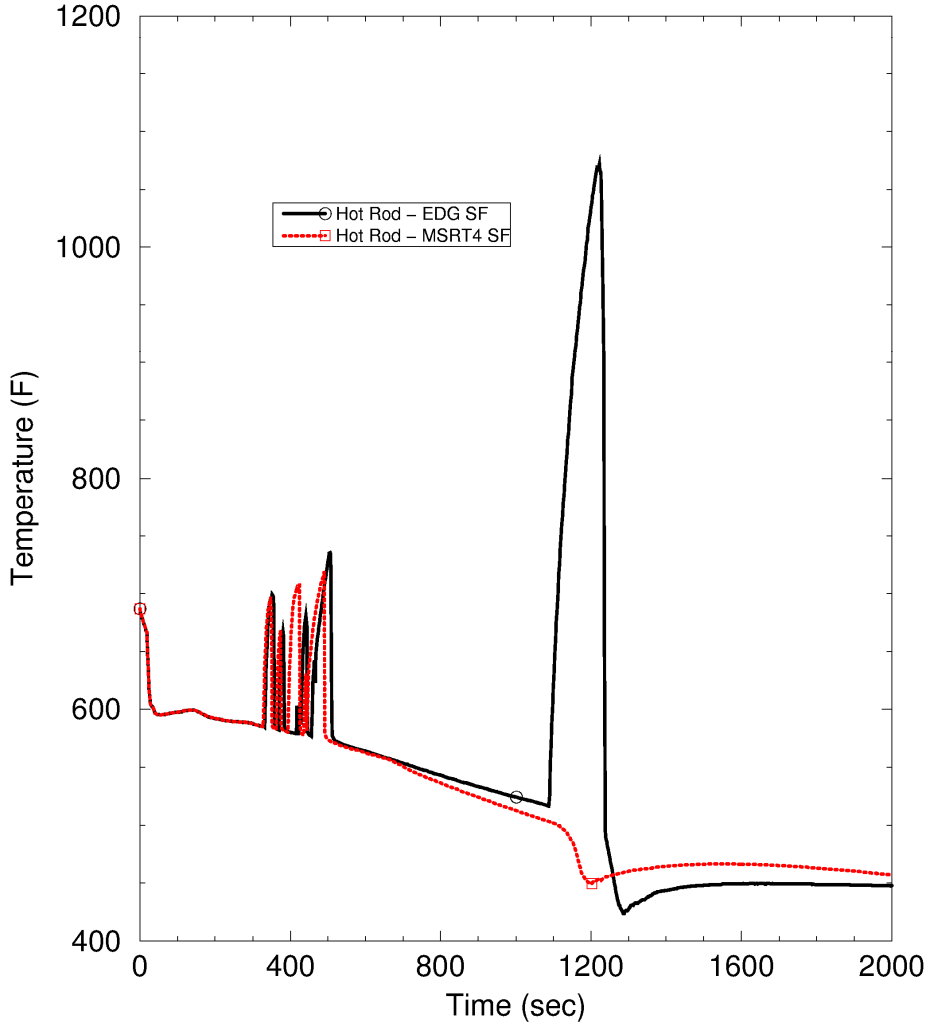

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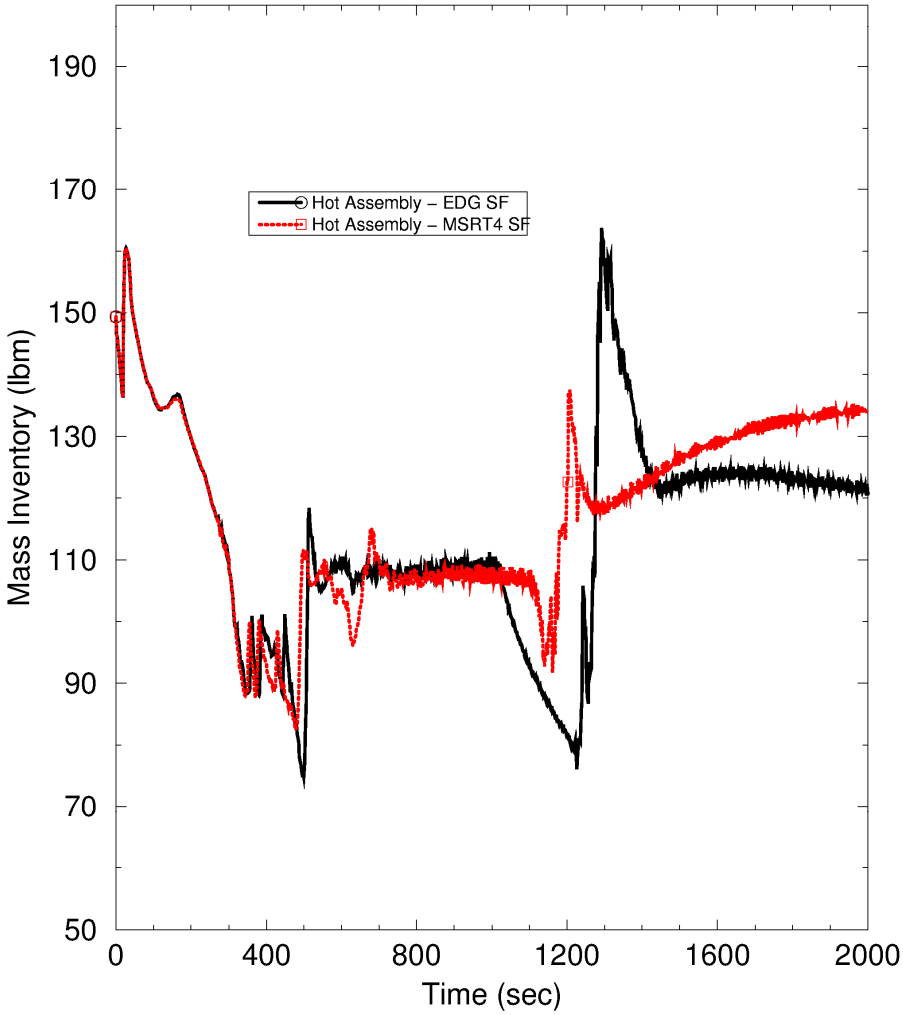
Figure 15.06.05.22-5—Hot Assembly Mass Inventory for the 4.0 Inch Breaks



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Question 15.06.05-23:

According to ANP-10291P the accumulators in the SBLOCA simulation are isolated when non-condensable gases are detected at their nozzles. Please justify the conservatism of this assumption, particularly with respect to long term cooling.

Response to Question 15.06.05-23:

The analysis of small break loss of coolant accident (SBLOCA) cases showed that the non-condensable gases entering the primary system after the accumulators emptied, but were not isolated, travel from the safety injection system injection locations around the downcomer annulus and out the break without being entrained in the steam generator (SG) tubes. For these cases, the peak cladding temperature (PCT) occurred before the accumulators emptied and the non-condensable gases entered the primary system. Thus, the short- and long-term cooling for these SBLOCA cases would not be impacted by the potential penetration of non-condensable gases into the primary system.

The SBLOCA analysis presented in the U.S. EPR FSAR, Tier 2, Section 15.6.5 assumes the accumulators isolate when non-condensable gases are detected at the accumulators' exit nozzles. This assumption is made in order to comply with 10 CFR 50, Appendix K, which requires the use of Moody critical flow at the break and because the S-RELAP5 code switches from the Moody model to the homogeneous model once non-condensable gases are present. This assumption is made only for break sizes from 7.5 inch inside diameter to the maximum size small break (10 percent of the cold leg pipe area).

Analyses of SBLOCA cases with no accumulator isolation showed that after the accumulators empty, the non-condensable gases enter the primary system and travel from the safety injection (SI) locations, around the downcomer annulus, and out the break without coalescing inside the SG tubes. An analysis of the 8.0 inch SBLOCA without isolation displays this behavior. Figure 15.06.05-23-1 shows the PCT occurs prior to 300 seconds while the accumulators are still injecting (Figure 15.06.05-23-2). Non-condensable gases travel to the break at approximately 800 seconds and between 1400 and 1500 seconds (Figure 15.06.05-23-3). The presence of non-condensable gases is observed at these times in all cold legs nodes (Figure 15.06.05-23-4, Figure 15.06.05-23-5, Figure 15.06.05-23-6, and Figure 15.06.05-23-7). However, non-condensable gases are not present in the reactor vessel upper regions (Figure 15.06.05-23-8) or in the steam generators tubes (Figure 15.06.05-23-9, Figure 15.06.05-23-10, Figure 15.06.05-23-11, and Figure 15.06.05-23-12), where they would be expected to coalesce. A similar system response was also predicted for the maximum SBLOCA case (10 percent of the cold leg pipe area) with no accumulator isolation.

SBLOCA cases smaller than 2.5 inch do not empty the accumulator, as the low head safety injection and medium head safety injection hold the primary system pressure high. The intermediate SBLOCA cases, in the 4.0 inch region, do not empty the accumulator before natural circulation is established. The results show steam flow around the loops, from the upper plenum to the break through the SG tubes, is sufficient to prevent a back-flow of non-condensable gases from the injection nozzles to the steam generators. If the accumulators empty and non-condensable gases penetrate the primary system in the long term, these breaks are large enough to remove the decay heat without relying on the steam generators.

FSAR Impact:

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

Figure 15.06.05-23-1—PCT for the 8.0 Inch Break

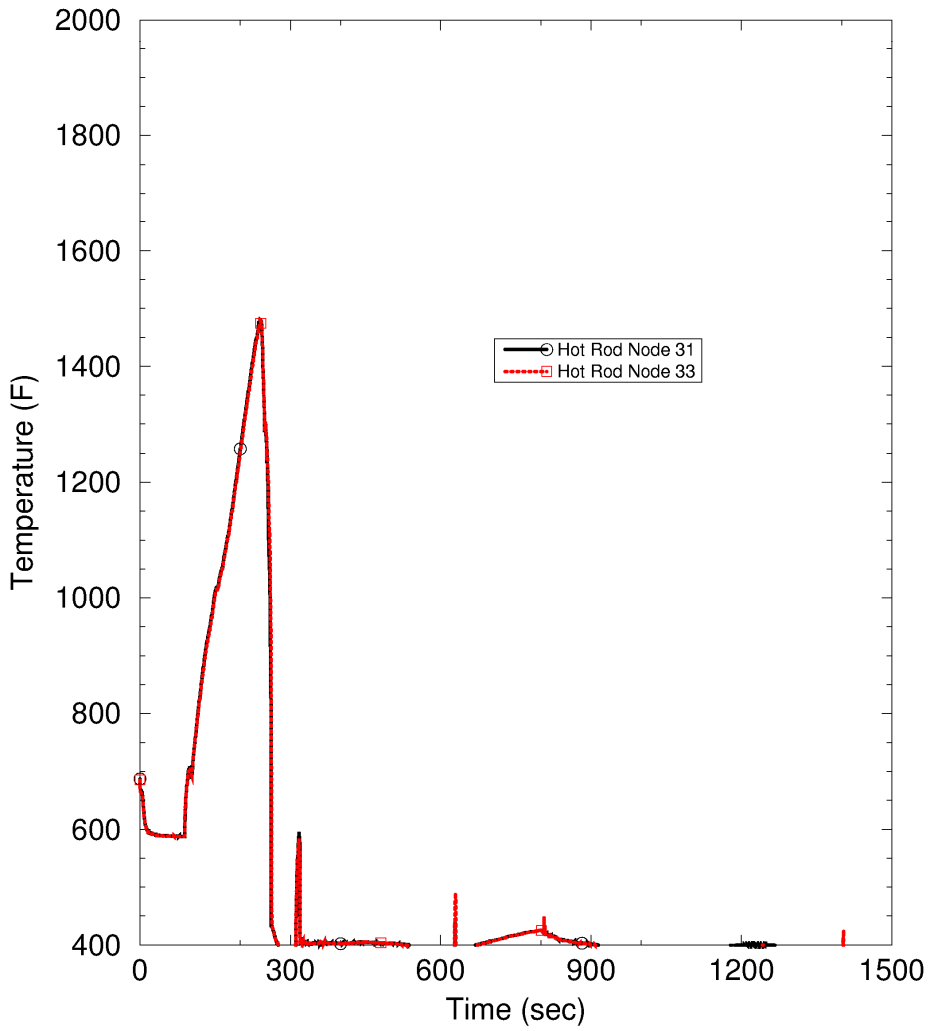

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Figure 15.06.05-23-2—Accumulator Flow Rate for the 8.0 Inch Break

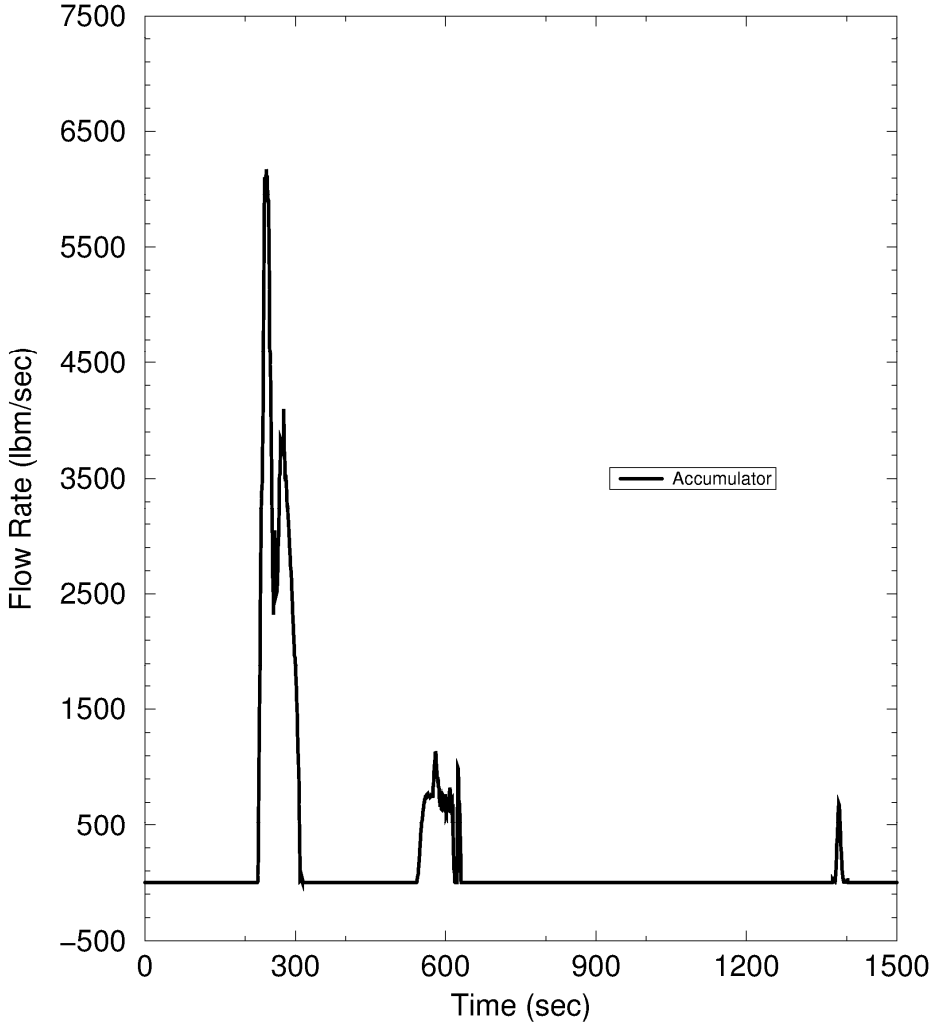

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Figure 15.06.05-23-3— Break Junction Non-Condensable Quality for the 8.0 Inch Break

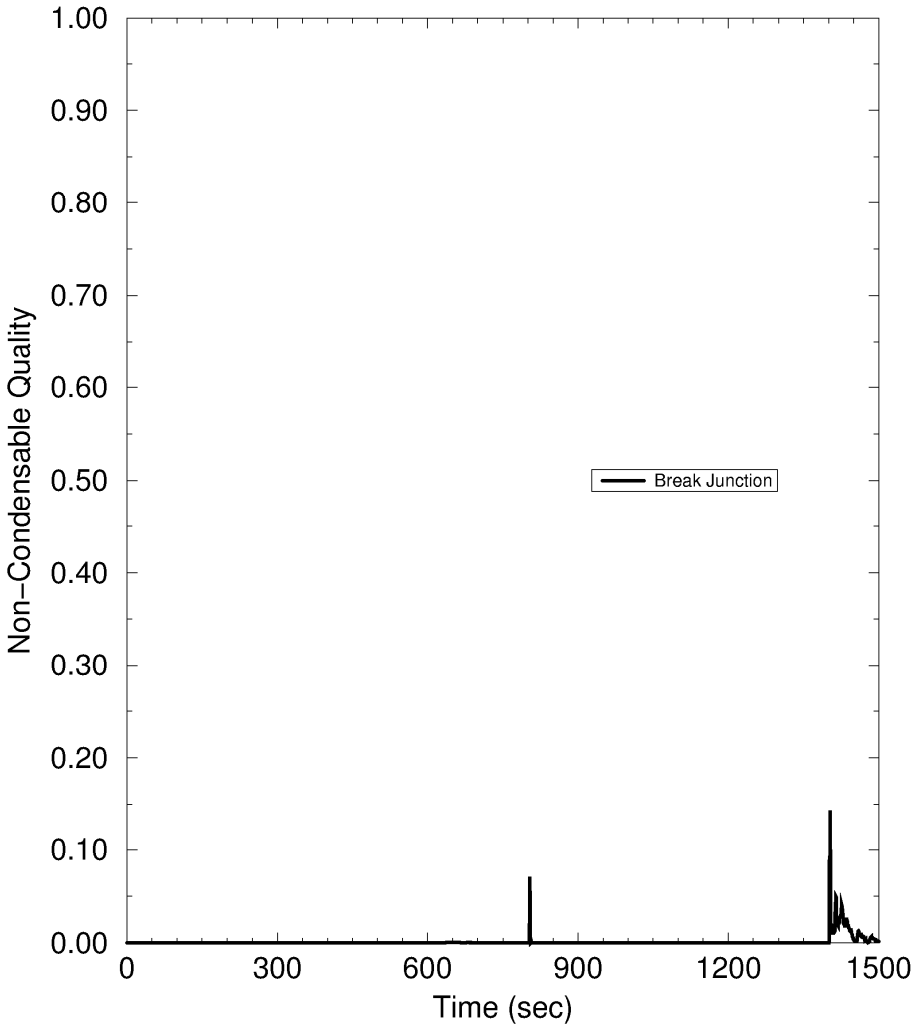

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Figure 15.06.05-23-4—Cold Leg-1 Non-Condensable Quality for the 8.0 Inch Break

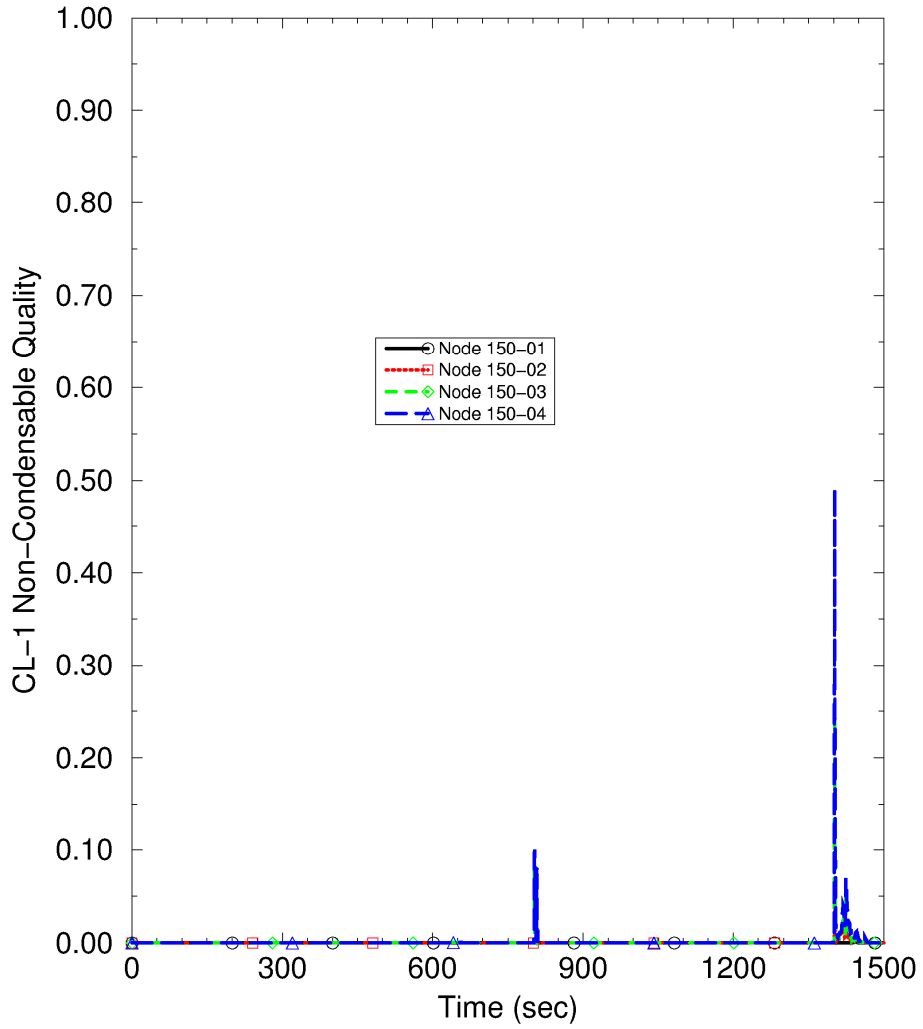

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Figure 15.06.05-23-5—Cold Leg-2 Non-Condensable Quality for the 8.0 Inch Break

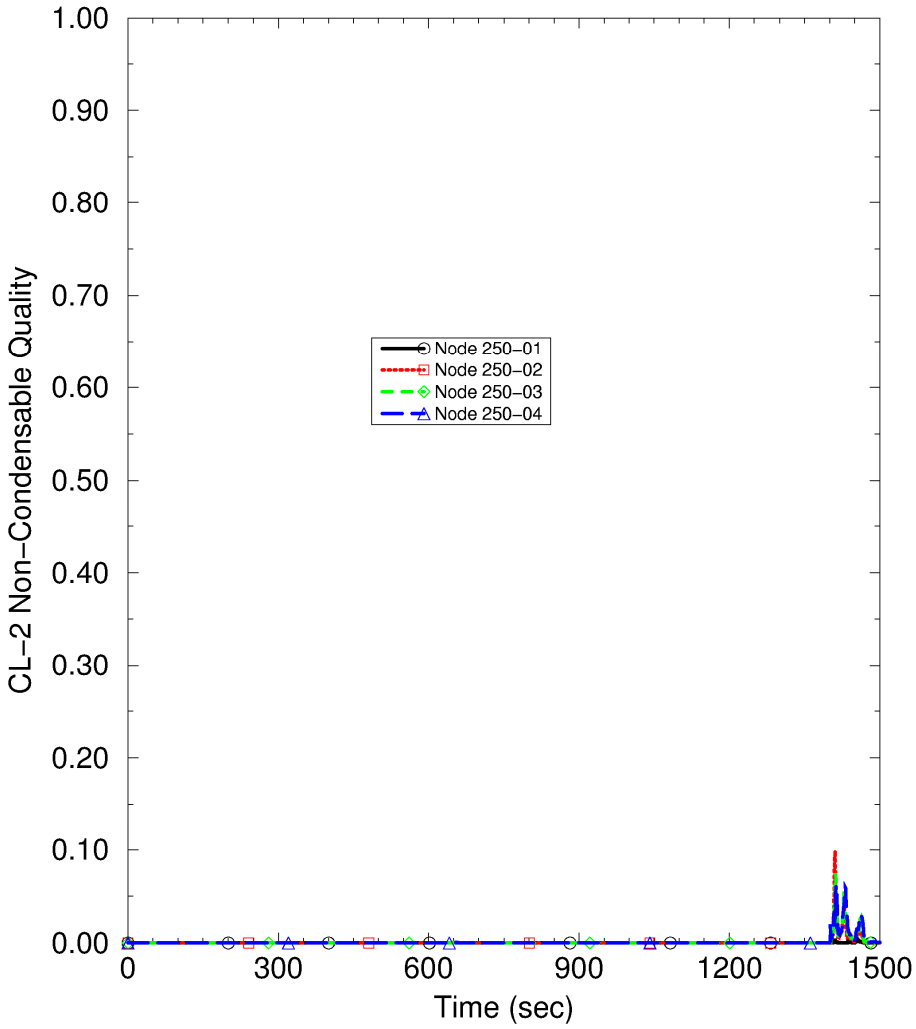

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Figure 15.06.05-23-6—Cold Leg-3 Non-Condensable Quality for the 8.0 Inch Break

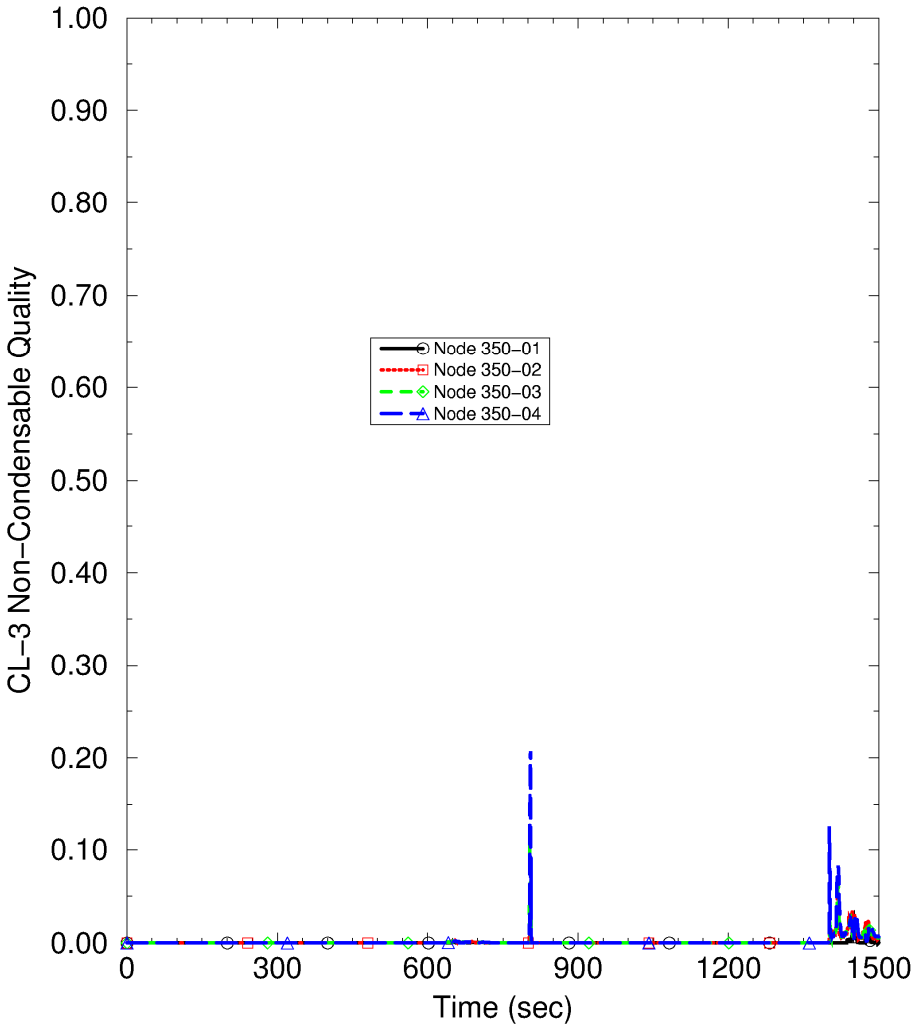

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Figure 15.06.05-23-7—Cold Leg-4 Non-Condensable Quality for the 8.0 Inch Break

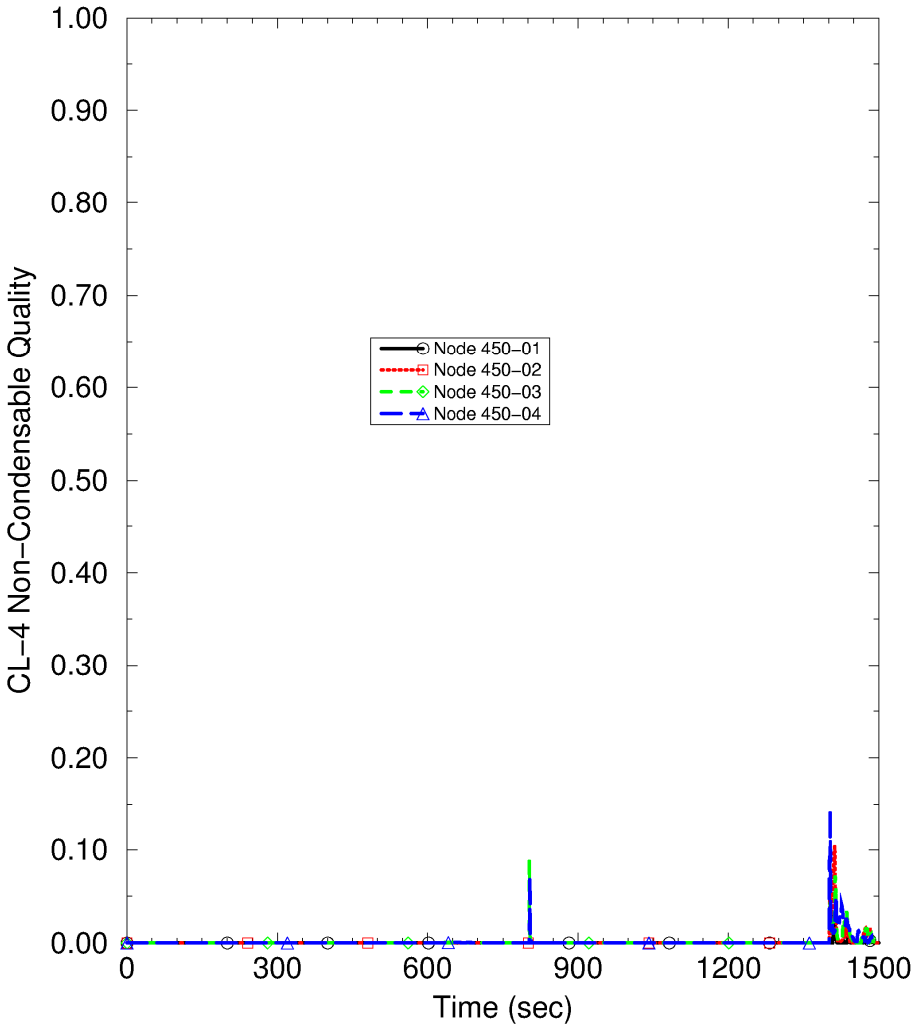

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Figure 15.06.05-23-8—Reactor Vessel (RV) Non-Condensable Quality for the 8.0 Inch Break

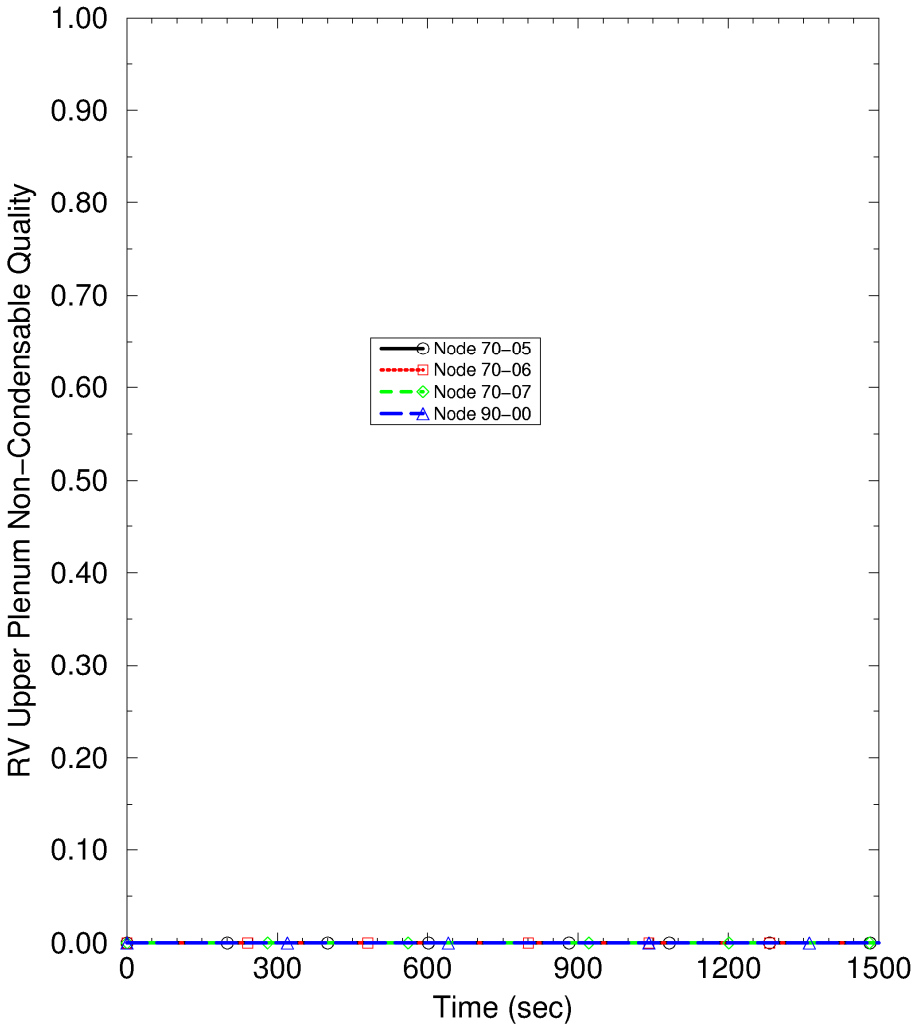

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Figure 15.06.05-23-9—SG-1 Tubes Non-Condensable Quality for the 8.0 Inch Break

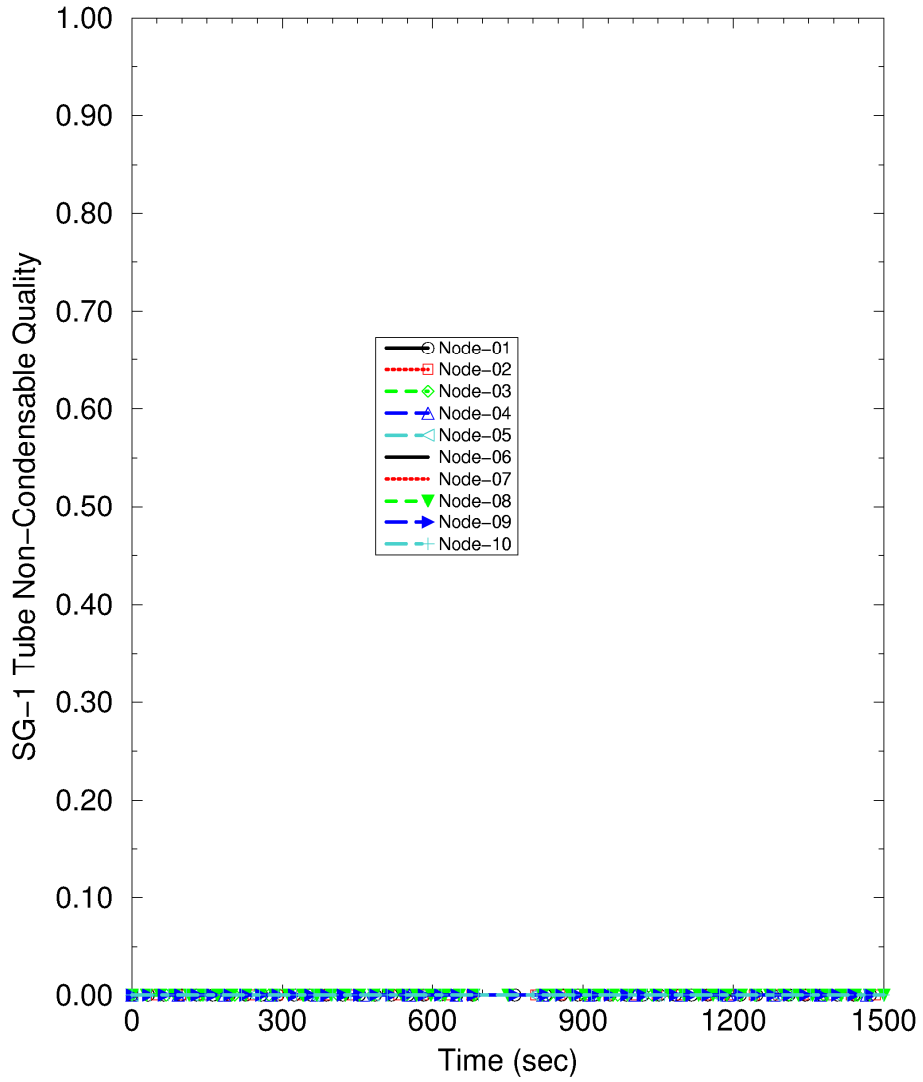

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Figure 15.06.05-23-10—SG-2 Tubes Non-Condensable Quality for the 8.0 Inch Break

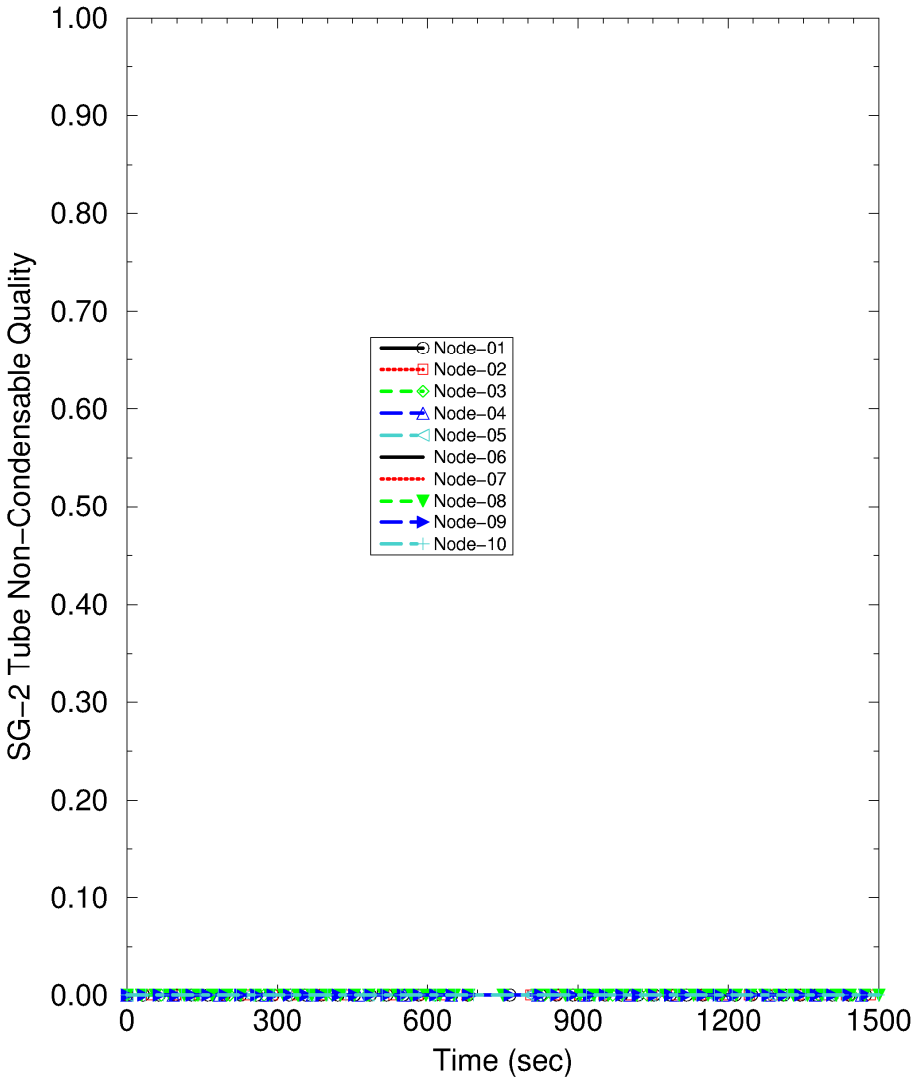

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Figure 15.06.05-23-11—SG-3 Tubes Non-Condensable Quality for the 8.0 Inch Break

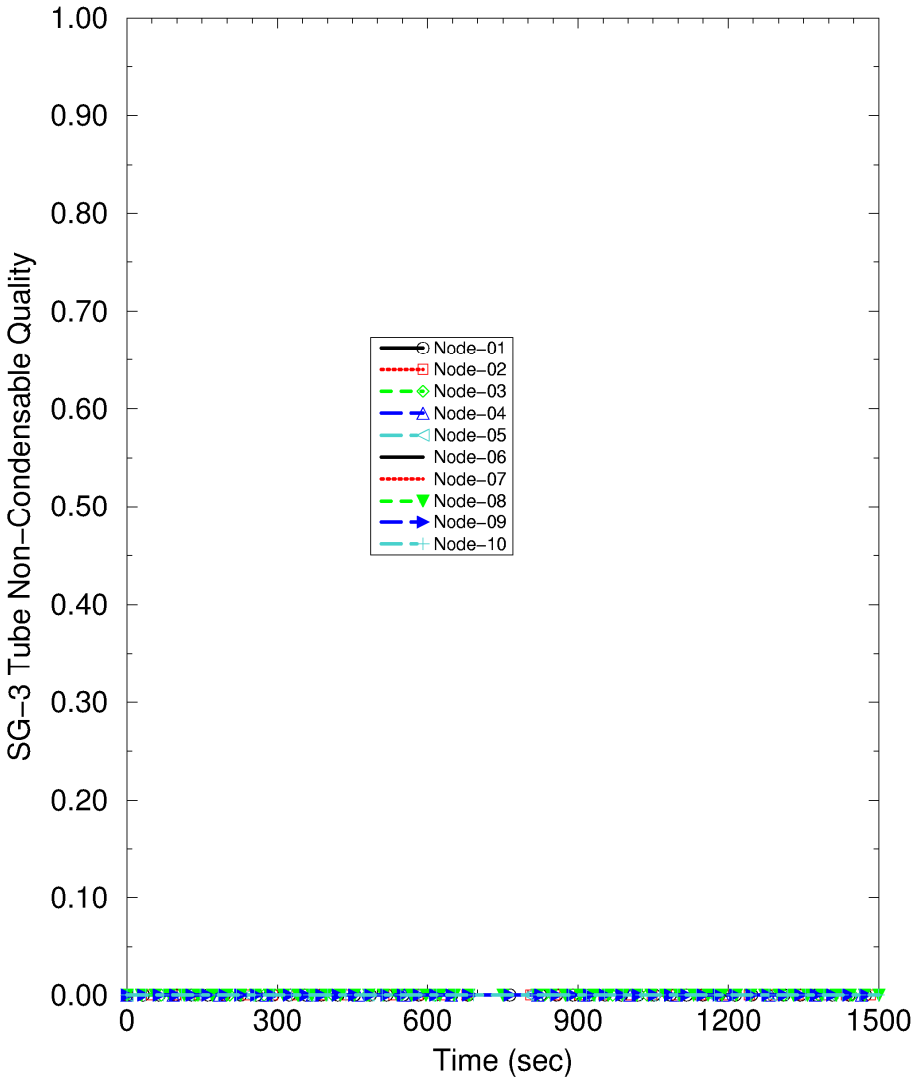

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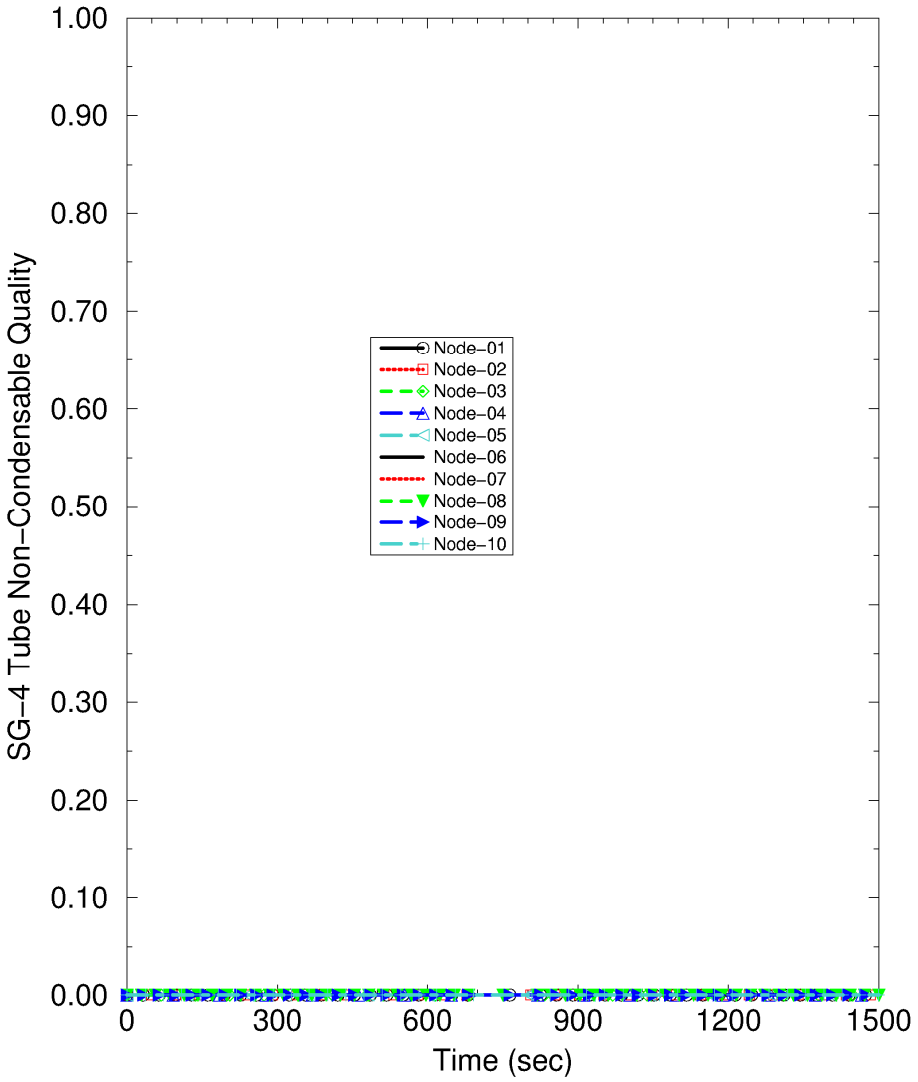
Figure 15.06.05-23-12—SG-4 Tubes Non-Condensable Quality for the 8.0 Inch Break



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Question 15.06.05-25:

Please provide the methodology for determining the various break locations (e.g., bottom, side, and top) in the RCP discharge leg.

Response to Question 15.06.05-25:

The small break loss of coolant accident (SBLOCA) analysis was performed in accordance with the approved AREVA NP Appendix K SBLOCA methodology and the methodology guidelines, which assume that the break junction is centrally located. The NRC-approved AREVA NP SBLOCA methodology in EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001, and ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR," August 2007, uses the Moody critical flow model to calculate break flow, as required by 10 CFR 50, Appendix K. As implemented, the Moody critical flow model uses a fixed slip ratio of 1.0 which yields homogeneous flow (equal phase velocities). As a consequence, this Appendix K-required model calculates the same critical flow rate at the break regardless of break orientation.

FSAR Impact:

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