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Revision 0

Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology Technical Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification

Contents

Page

1.0 INTRODUCTION 1-1
2.0 SBLOCA 2-1
2.1 SBLOCA Methodology 2-1
2.2 SBLOCA SG Nodalization Studies 2-4
2.2.1 Description of Models used in Nodalization Studies 2-4
2.2.2 Results of Nodalization Studies 2-6
2.2.3 Conclusions 2-7
3.0 NON-LOCA 3-1
3.1 Non-LOCA SG Nodalization Studies 3-1
3.1.1 Description of Models used in Nodalization Studies 3-1
3.1.2 Sensitivity Study Results 3-3
3.1.3 Conclusions 3-4
4.0 REFERENCES 4-1
APPENDIX A
A.1. Introduction A-4
A.2. S-RELAP5 Model A-4
A.3. Initial Conditions and Other Input Parameters A-6
A.4. Break Spectrum A-7
A.5. 6.5-Inch Break (Limiting Case) A-8
A.6. Results/Summary/Conclusion A-10
A.7. References A-12

List of Tables

Page

Table 2-1 Comparison of PCTS for the Axial Nodalization Study 2-8

Table 2-2 Comparison of PCTS for the Radial Nodalization Study 2-8

Table A-1 Input Parameters A-13

Table A-2 Safety Classified I&C Signals Considered for SBLOCA Analysis A-14

Table A-3 Break Spectrum - Peak Cladding Temperature A-15

Table A-4 Limiting Break Sequence of Events A-16

List of Figures

Figure 2-1 Comparison between Measured and Calculated Peak Temperatures from BETHSY Test 9.1b. 2-9

Figure 2-2 Comparison between Measured and Calculated Core Collapsed Liquid Level from BETHSY Test 9.1b. 2-10

Figure 2-3 Comparison between Measured and Calculated Peak Temperatures from Semiscale Test S-UT-8 2-11

Figure 2-4 Comparison between Measured and Calculated Core Collapsed Liquid Level from Semiscale Test S-UT-8 2-12

Figure 2-5 Base Case Reactor Coolant System Nodalization 2-13

Figure 2-6 Base Case Secondary System Nodalization 2-14

Figure 2-7 Axial Nodalization Study - Reactor Coolant System Nodalization 2-15

Figure 2-8 Axial Nodalization Study - Secondary System Nodalization 2-16

Figure 2-9 Radial Nodalization Study - Reactor Coolant System Nodalization 2-17

Figure 2-10 Radial Nodalization Study – Secondary System Nodalization 2-18

Figure 2-11 Axial Nodalization Study -SG-1 Hot Leg Mass for a 6.5 inch Break 2-19

Figure 2-12 Axial Nodalization Study - SG-1 Upside Collapsed Liquid Level for a 6.5 inch Break 2-20

Figure 2-13 Axial Nodalization Study - SG-1 Downside Collapsed Liquid Level for a 6.5 inch Break 2-21

Figure 2-14 Axial Nodalization Study - SG-1 Inlet Flowrate for a 6.5 inch Break 2-22
Figure 2-15 Axial Nodalization Study - SG-1 Outlet Flowrate for a 6.5 inch Break 2-23
Figure 2-16 Axial Nodalization Study - Loop SG-1 Tube Heat Transfer for a 6.5 inch Break 2-24
Figure 2-17 Axial Nodalization Study - Pressurizer Pressure for a 6.5 inch Break 2-25
Figure 2-18 Axial Nodalization Study - Loop 1 Secondary Pressure for a 6.5 inch Break 2-26
Figure 2-19 Axial Nodalization Study - RCS Fluid Mass for a 6.5 inch Break 2-27
Figure 2-20 Axial Nodalization Study - Reactor Vessel Fluid Mass for a 6.5 inch Break 2-28
Figure 2-21 Axial Nodalization Study - Hot Assembly Collapsed Liquid Level for a 6.5 inch Break 2-29
Figure 2-22 Axial Nodalization Study - PCT for a 6.5 inch Break 2-30
Figure 2-23 Radial Nodalization Study - SG-1 Hot Leg Mass for a 6.5 inch Break 2-31
Figure 2-24 Radial Nodalization Study - SG-4 Hot Leg Mass for a 6.5 inch Break 2-32
Figure 2-25 Radial Nodalization Study - SG-1 Downside Collapsed Liquid Level for a 6.5 inch Break 2-33
Figure 2-26 Radial Nodalization Study - SG-1 Inlet Flow for a 6.5 inch Break 2-34
Figure 2-27 Radial Nodalization Study - SG-1 Outlet Flowrate for a 6.5 inch Break 2-35
Figure 2-28 Radial Nodalization Study - Loop 1 SG Tube Heat Transfer for a 6.5 inch Break 2-36
Figure 2-29 Radial Nodalization Study - Pressurizer Pressure for a 6.5 inch Break 2-37
Figure 2-30 Radial Nodalization Study - Loop 1 Secondary Pressure for a 6.5 inch Break 2-38
Figure 2-31 Radial Nodalization Study - RCS Fluid Mass for a 6.5 inch Break 2-39
Figure 2-32 Radial Nodalization Study - Reactor Vessel Fluid Mass for a 6.5 inch Break 2-40

Figure 2-33 Radial Nodalization Study - Hot Assembly Collapsed Liquid Level for a 6.5 inch Break 2-41
Figure 2-34 Radial Nodalization Study - PCT for a 6.5 inch Break 2-42
Figure 3-1 Base Model Steam Generator Nodalization 3-5
Figure 3-2 Case 2 Steam Generator Nodalization 3-6
Figure 3-3 Case 3 Steam Generator Nodalization 3-7
Figure 3-4 Case 1 Turbine Trip – Maximum RCS Pressure 3-8
Figure 3-5 Case 1 Turbine Trip – Reactor Power 3-9
Figure 3-6 Case 1 Turbine Trip – Maximum Steam Generator Pressure 3-10
Figure 3-7 Case 1 Turbine Trip – MSIV1 Flow Rate 3-11
Figure 3-8 Case 1 Turbine Trip – RCS Flow Rate 3-12
Figure 3-9 Case 2 Turbine Trip – Maximum RCS Pressure 3-13
Figure 3-10 Case 2 Turbine Trip – Reactor Power 3-14
Figure 3-11 Case 2 Turbine Trip – Maximum Steam Generator Pressure 3-15
Figure 3-12 Case 2 Turbine Trip – MSIV1 Flow Rate 3-16
Figure 3-13 Case 2 Turbine Trip – RCS Flow Rate 3-17
Figure 3-14 Case 3 Turbine Trip – Maximum RCS Pressure 3-18
Figure 3-15 Case 3 Turbine Trip – Reactor Power 3-19
Figure 3-16 Case 3 Turbine Trip – Maximum Steam Generator Pressure 3-20
Figure 3-17 Case 3 Turbine Trip – MSIV1 Flow Rate 3-21
Figure 3-18 Case 3 Turbine Trip – RCS Flow Rate 3-22
Figure 3-19 Case 3 Loss of Feed Flow – Reactor Power 3-23
Figure 3-20 Case 3 Loss of Feed Flow – Total Steam Generator Power 3-24
Figure 3-21 Case 3 Loss of Feed Flow – Core Mass Flow Rate 3-25
Figure 3-22 Case 3 Loss of Feed Flow – Maximum Steam Generator Pressure 3-26
Figure 3-23 Case 3 Loss of Feed Flow – Maximum RCS Pressure 3-27
Figure 3-24 Case 3 Loss of Feed Flow – Steam Generator No. 1 NR Level 3-28
Figure 3-25 Case 3 Loss of Feed Flow – Steam Generator No. 2 NR Level 3-29
Figure 3-26 Case 3 Loss of Feed Flow – Steam Generator No. 3 NR Level 3-30

Figure 3-27 Case 3 Loss of Feed Flow – Steam Generator No. 4 NR Level 3-31
Figure 3-28 Case 3 Loss of Feed Flow – Steam Generator No. 1 & No. 3 Total Mass 3-32
Figure 3-29 Case 3 Loss of Feed Flow – Steam Generator No. 2 & No. 4 Total Mass 3-33
Figure 3-30 Case 3 Loss of Feed Flow – Pressurizer Level 3-34
Figure 3-31 Case 3 Increase in Steam Flow – Steam Generator No. 1 Dome Pressure 3-35
Figure 3-32 Case 3 Increase in Steam Flow – Reactor Power 3-36
Figure 3-33 Case 3 Increase in Steam Flow – TCV Flow Rate 3-37
Figure 3-34 Case 3 Increase in Steam Flow – Reactor Vessel Lower Plenum Pressure 3-38
Figure 3-35 Case 3 Increase in Steam Flow – MSIV1 Flow Rate 3-39
Figure 3-36 Case 3 Increase in Steam Flow – RCS Flow Rate 3-40
Figure 3-37 Case 3 Increase in Steam Flow – Steam Bypass Flow Rate 3-41
Figure 3-38 Case 3 Increase in Steam Flow – Steam Line No. 2 Pressure 3-42
Figure 3-39 Case 3 Increase in Steam Flow – RCS Cold Leg No. 1 Temperature 3-43
Figure 3-40 Case 3 Increase in Steam Flow – Steam Generator No. 1 Main Feedwater Flow Rate 3-44
Figure A-1 U.S. EPR Vessel Nodalization A-17
Figure A-2 U.S. EPR Steam Generator Nodalization A-18
Figure A-3 U.S. EPR Loop Nodalization A-19
Figure A-4 Nodalization For S-RELAP5 ECC Model A-20
Figure A-5 Reactor Power for 6.5 inch Break A-21
Figure A-6 Primary and Secondary System Pressure for 6.5 inch Break A-22
Figure A-7 MSRT Flow for 6.5 inch Break A-23
Figure A-8 Break Flow Rate for 6.5 inch Break A-24
Figure A-9 Loop Seal Void Fraction for 6.5 inch Break A-25
Figure A-10 ECCS Flow for 6.5 inch Break A-26
Figure A-11 MFW Flow Rate for 6.5 inch Break A-27
Figure A-12 Steam Generator Mass Inventory for 6.5 inch Break A-28

Figure A-13 Steam Generator Level for 6.5 inch Break....A-29
Figure A-14 Steam Generator Steam Line Pressure for 6.5 inch Break....A-30
Figure A-15 Steam Generator Steam Line Pressure and Cooldown Curve for 6.5 inch Break....A-31
Figure A-16 Total RCS and RV Mass Inventories for the 6.5 inch Break....A-32
Figure A-17 Hot Assembly Collapsed Liquid Level for 6.5 inch Break....A-33
Figure A-18 MHSI Flow (Loops 1 and 4) for 6.5 inch Break....A-34
Figure A-19 LHSI Flow (Loops 1 and 4) for 6.5 inch Break....A-35
Figure A-20 Vapor and Clad Temperature for the Hot Node for a 6.5 inch Break....A-36

Nomenclature

Acronym	Definition
CCFL	Counter Current Flow Limit
DC	Downcomer
ECC	Emergency Core Cooling System
EOC	End of Cycle
EOL	End of Life
EFW(S)	Emergency Feedwater (System)
HL	Hot Leg
I&C	Instrumentation and Controls
LHSI	Low Head Safety Injection
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LP	Lower Plenum
MFW(S)	Main Feedwater (System)
MHSI	Medium Head Safety Injection
MSIV	Main Steam Isolation Valve
MSRCV	Main Steam Relief Control Valve
MSRT	Main Steam Relief Train
NR	Narrow Range
PCT	Peak Clad Temperature
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RT	Reactor Trip
SBLOCA	Small Break LOCA
SG	Steam Generator
SIS	Safety Injection System
TCV	Turbine Control Valve
USNRC	United States Nuclear Regulatory Commission

1.0 INTRODUCTION

The AREVA NP Inc. codes and methods used for analyses of the U.S. EPR are discussed in the topical report ANP-10263PA (Reference 1). During the NRC review and approval of this report, AREVA NP Inc. committed to perform additional sensitivity studies related to the use of S-RELAP5 for Small Break LOCA (SBLOCA) and non-LOCA analyses.

This technical report contains the results and conclusions of these studies and a summary of the changes made to the U.S. EPR SBLOCA evaluation model. Section 2 addresses SBLOCA methodology changes and sensitivity studies that were described in the response to RAI 5 (Reference 1). Section 3 addresses non-LOCA sensitivity studies that were described in the response to RAI 8 (Reference 1). Appendix A contains the SBLOCA sample problem that was rerun using the SBLOCA methodology of Section 2.

2.0 SBLOCA

2.1 *SBLOCA Methodology*

The SBLOCA methodology is presented in Section 4.0 of ANP-10263PA. As a first step in performing the sensitivity studies, SBLOCA analysis trends and results were examined and the following methodology changes were incorporated:

- Modeling of the Downcomer-Hot Leg (DC-HL) Gaps: The SBLOCA methodology described in ANP-10263PA models the DC-HL bypass paths explicitly. They are now included in the control rod guide tube bypass flow path.

  The change is made based on information for the U.S. EPR that the DC-HL gaps tend to close during the depressurization of the RCS following a SBLOCA.

- Preventing Top-Down Quench in Hot Channel: Initial SBLOCA studies for the U.S. EPR exhibited the non-physical draining of water into the hot channel from the upper plenum. Because this could reduce peak clad temperature (PCT) a modeling change is made to incorporate a high reverse k-factor at the outlet of the hot channel (multiplying the existing k-factor by 1000). This effectively prevents the non-physical behavior in the hot channel without affecting overall core flow dynamics.

- RODEX2 Input: The methodology described in ANP-10263PA uses the RODEX2 code to determine the initial hot fuel rod conditions for SBLOCA analysis based on the power history for normal operation that produces the highest EOL gas pressure. A sensitivity study performed for the U.S. EPR demonstrates that PCT is not sensitive to the power history chosen. Therefore, a representative power history is used in RODEX2 and fuel rod conditions that represent a bounding high hot rod burnup at EOC are used in the S-RELAP5 calculation. This approach is consistent with AREVA's response (Reference 1) to Request for Additional Information Number 15 on ANP-10263PA, that states,

"The actual power history used in the RODEX2 code to predict fuel rod conditions at EOC does not have a significant effect on the PCT."

- Analysis of Non-LOOP, RC Pump Trip SBLOCA Scenarios: The SBLOCA methodology described in ANP-10263PA directs the analyst to assume offsite power is lost coincident with turbine trip (reactor scram) because it produces higher PCTs than when offsite power is retained. It does not require the analysis of SBLOCA scenarios in which offsite power is not lost. The U.S. EPR differs from current plants in that the RC pumps are tripped automatically on low RC pump Delta P in combination with a safety injection signal, rather than low saturation margin. Because of this design difference, the spectrum of SBLOCA cases analyzed for the U.S. EPR includes cases without a loss of offsite power (LOOP).
- Isolation of Accumulators when Non-condensables are Detected at the Accumulator Exit: The SBLOCA methodology described in ANP-10263PA allows non-condensable gases to be injected into the RCS when the accumulators empty. The presence of non-condensable gases at the break junction causes the S-RELAP5 code to switch from the Moody break flow model to the homogeneous break flow model. Since 10 CFR 50, Appendix K, requires the use of the Moody critical flow model, the accumulators are isolated as soon as non-condensable gases are detected at their nozzles. This prevents the switch from the Moody break flow model to the homogeneous break flow model.
- Low Head Safety Injection (LHSI) and Addition of ECCS Piping: The LHSI system and associated cross-connect related piping are modeled for the U.S. EPR to enable the evaluation of the entire break size spectrum of small breaks and the analysis of long-term phenomena.
- Modeling of Hot Leg and Counter Current Flow Limit (CCFL) at the Steam Generator (SG) Tube Inlets for Drainage During Reflux Cooling: Several changes in modeling are made to better predict drainage of the hot leg and SG tubes during reflux cooling. First, the S-RELAP5 SBLOCA methodology

described in ANP-10263PA shows the hot legs modeled with four nodes in which the first three nodes represent the horizontal portion of the hot leg and the fourth node represents the elbow and the SG inlet nozzle. This arrangement forces the S-RELAP5 flow regime map to change from horizontal to vertical in the hot leg. To remove this anomaly in the U.S. EPR SBLOCA model, the fourth node in the hot leg (elbow plus SG inlet nozzle) is divided into two nodes, such that the fourth node now represents the elbow and the fifth node represents the SG inlet nozzle. This causes the new fourth node (the elbow) to be in horizontal stratified flow during the reflux cooling period. This promotes draining of the hot leg and produces core level and flow behavior representative of that observed in experimental data.

Along with the nodalization change, the CCFL model (Wallis with m = 1.0 and c = 0.8) is applied at the SG tube inlets. This prevents premature draining of the SG tubes until the conditions of steam and liquid flow permit it.

The preceding changes are benchmarked against the Semiscale S-UT-8 and BETHSY 9.1b tests presented in the original S-RELAP5 SBLOCA methodology report (EMF-2328(P)(A)) (Reference 2). In both these cases, the hot leg nodalization is similar to the plant model in that the SG inlet nozzle is modeled explicitly as a separate volume with the appropriate angle of inclination.

The results from the BETHSY Test 1b assessment are shown in Figures 2-1 and 2-2. Figure 2-1 shows the comparison between measured and calculated peak core temperatures and Figure 2-2 shows the comparison between measured and calculated core collapsed liquid levels. Both comparisons show the calculated core temperature excursion and level depression occurring approximately 100s earlier than measured. This behavior is due to the effects of the inclusion of CCFL at the tube sheet inlet. In this application of CCFL, slightly less bypass flow from the upper head to the downcomer occurs just before break uncovery. This flow causes some delay in the break uncovery and sufficiently more mass is ejected from the primary system causing the earlier timings previously

mentioned. These effects are restricted to the 2" break from the BETHSY facility since break uncovery is determined by upper head to downcomer bypass flow instead of loop seal clearing. Otherwise, these results are similar to previous results.

The results from the Semiscale Test S-UT-8 assessment are shown in Figures 2-3 and 2-4. Figure 2-3 shows the comparison between measured and calculated core temperature at the hot location and Figure 2-4 shows the comparison between measured and calculated core collapsed liquid levels. The comparison period for this assessment is prior to loop seal clearing, from 0 to 300s. The calculated temperature excursion occurs earlier in time than the measurement; however, this is a characteristic of this S-RELAP5 benchmark. Likewise, the core uncovery also occurs earlier than measured. Again, this is expected behavior from this system benchmark. Overall, there is little difference between these results and those from previous calculations. S-RELAP5 captures the important phenomena and is adequately conservative.

2.2 *SBLOCA SG Nodalization Studies*

Sensitivity studies were performed to verify the adequacy of the steam generator nodalization for SBLOCA analyses. The sensitivity studies were based on three calculations: (1) the base U.S. EPR model, (2) a SG axial nodalization sensitivity, including an economizer heat structure, and (3) a SG radial nodalization in the tube region sensitivity. All three calculations incorporated all of the methodology/modeling changes described in Section 2.1 of this report.

2.2.1 *Description of Models used in Nodalization Studies*

2.2.1.1 *Base Model*

[ ]

2.2.1.2 *Axial Nodalization Sensitivity Study*

2.2.1.3 *Radial Nodalization Sensitivity Study*

[ ]

2.2.2 *Results of Nodalization Studies*

2.2.2.1 *Axial Nodalization Study Results*

Calculations were performed with the revised steam generator axial nodalization for the 4.0., 6.0, 6.5, 7.0, and 8.0 inch-diameter break sizes and compared to the results from the base model. The PCTs presented in Table 2-1 compare well between the base model and the axial nodalization study model, particularly in the region near the limiting break size.

Figures 2-11 through 2-22 compare key parameters between the base model and the axial nodalization study model for the limiting 6.5 in. break size. These comparisons show there is no significant difference in the thermal hydraulic behavior and system response between the two cases. It is concluded that the base model steam generator axial nodalization is acceptable for U.S. EPR SBLOCA design certification analysis and that additional detail is not required.

2.2.2.2 *Radial Nodalization Study Results*

Calculations were performed for the 4.0, 6.0, 6.5, 7.0, and 8.0 inch-diameter break sizes with radial nodalization in the tube region of the steam generator and compared to the results from the base model. The PCTs presented in Table 2-2 compare well between the base model and the radial nodalization study model, particularly in the region near the limiting break size.

Figures 2-23 through 2-34 compare key parameters between the base model and the radial nodalization study model for the limiting 6.5 in break size. The comparisons show there is no significant difference in the thermal hydraulic behavior and system response between the two cases. It is concluded that the base model is acceptable for U.S. EPR SBLOCA design certification analysis and that radial nodalization of the steam generator tube region is not required.

2.2.3 *Conclusions*

Based on the results of the two sensitivity studies discussed in Section 2.2.2, the overall conclusion is that the steam generator nodalization used in the U.S. EPR SBLOCA base model is acceptable for use in the U.S. EPR SBLOCA design certification analysis. The SBLOCA sample problem supplied as Appendix C in ANP-10263PA (Reference 1) has been re-analyzed using the base model described in Section 2.2.1.1. The results appear as Appendix A of this report.

Table 2-1 Comparison of PCTs for the Axial Nodalization Study

	PCT (°F)	
Break Size (in.)	**Base Model**	**Axial Nodalization Study Model**
4.0	1088	1209
6.0	1125	1189
6.5	1638	1674
7.0	1587	1575
8.0	1469	1464

Table 2-2 Comparison of PCTs for the Radial Nodalization Study

	PCT (°F)	
Break Size (in.)	**Base Model**	**Radial Nodalization Study Model**
4.0	1088	1088
6.0	1125	1156
6.5	1638	1632
7.0	1587	1571
8.0	1469	1454

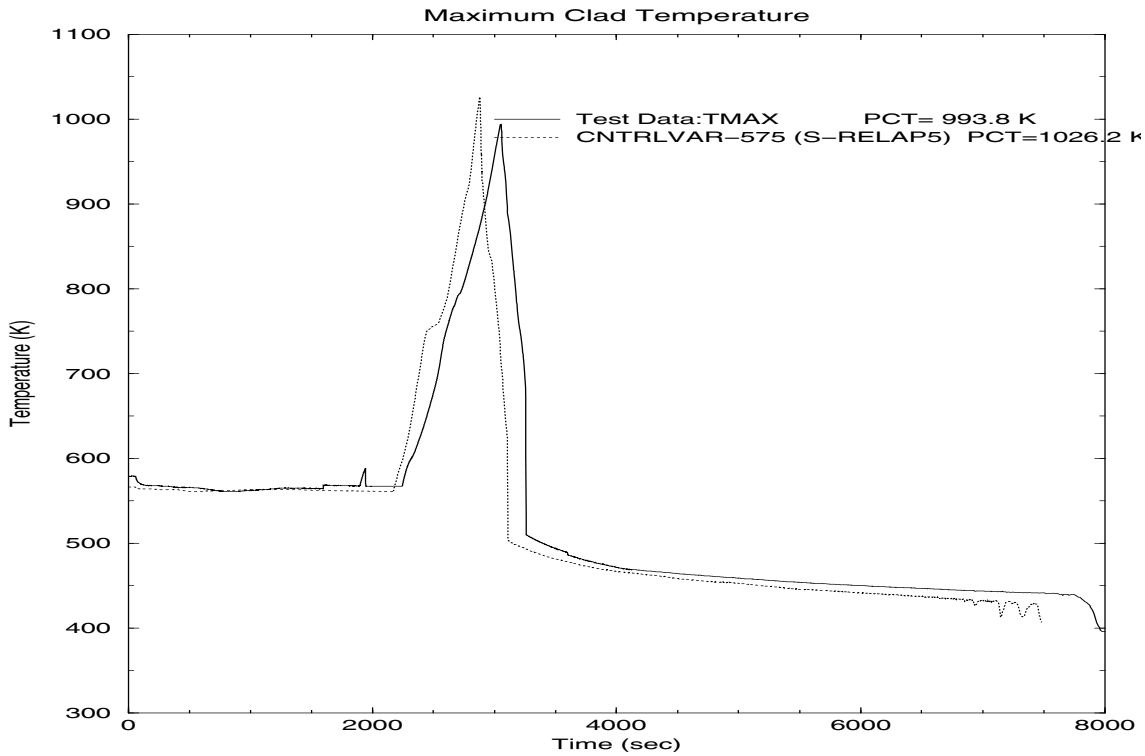

Figure 2-1 Comparison between Measured and Calculated Peak Temperatures from BETHSY Test 9.1b

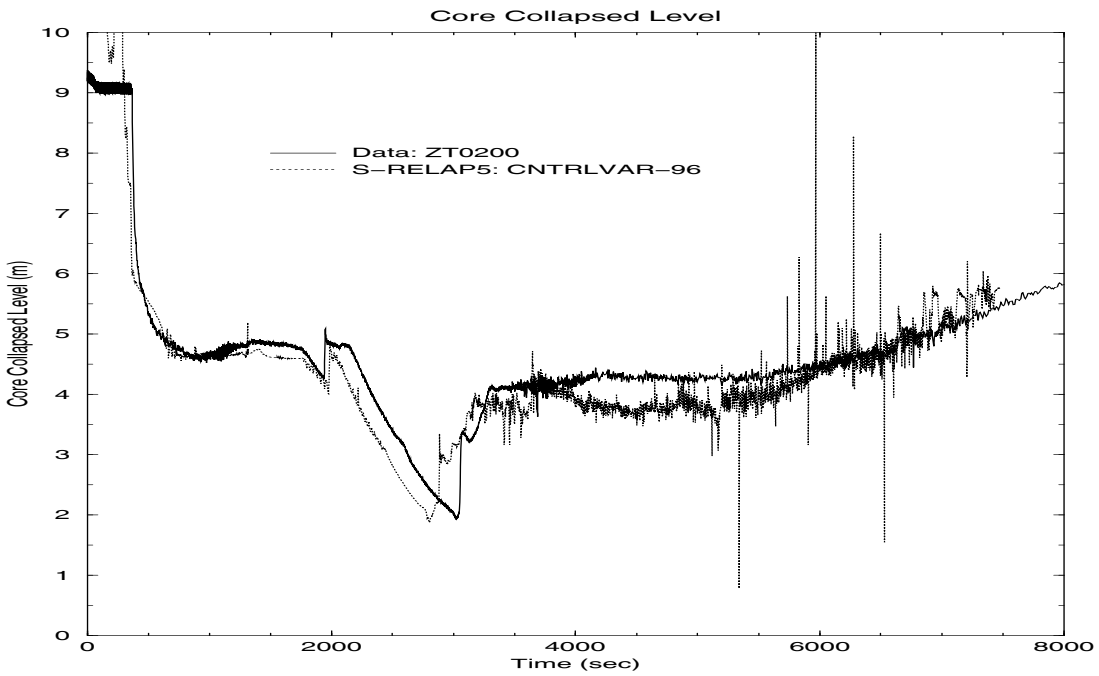

Figure 2-2 Comparison between Measured and Calculated Core Collapsed Liquid Level from BETHSY Test 9.1b

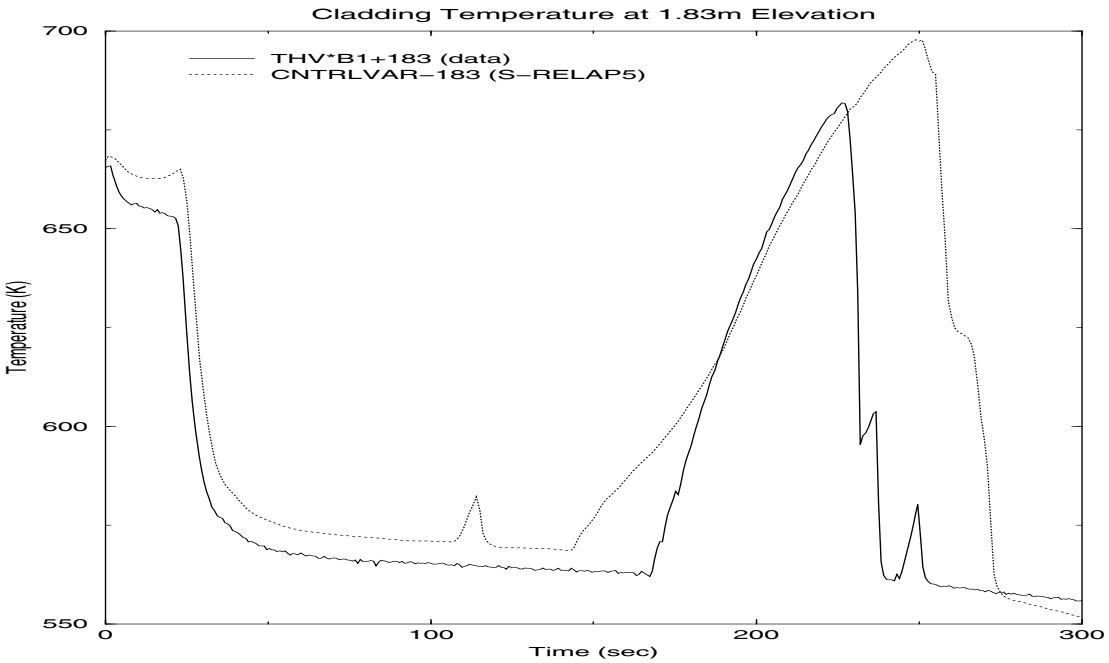

Figure 2-3 Comparison between Measured and Calculated Peak Temperatures from Semiscale Test S-UT-8

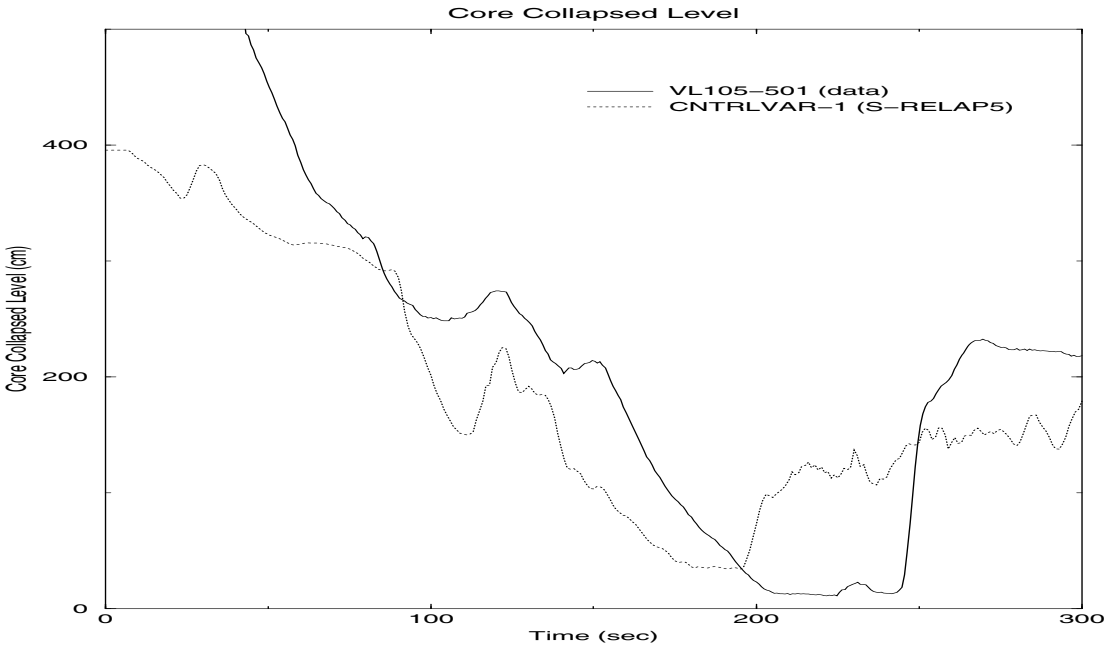

Figure 2-4 Comparison between Measured and Calculated Core Collapsed Liquid Level from Semiscale Test S-UT-8

Figure 2-5 Base Case Reactor Coolant System Nodalization

Figure 2-6 Base Case Secondary System Nodalization

Figure 2-7 Axial Nodalization Study - Reactor Coolant System Nodalization

Figure 2-8 Axial Nodalization Study - Secondary System Nodalization

Figure 2-9 Radial Nodalization Study - Reactor Coolant System Nodalization

Figure 2-10 Radial Nodalization Study – Secondary System Nodalization

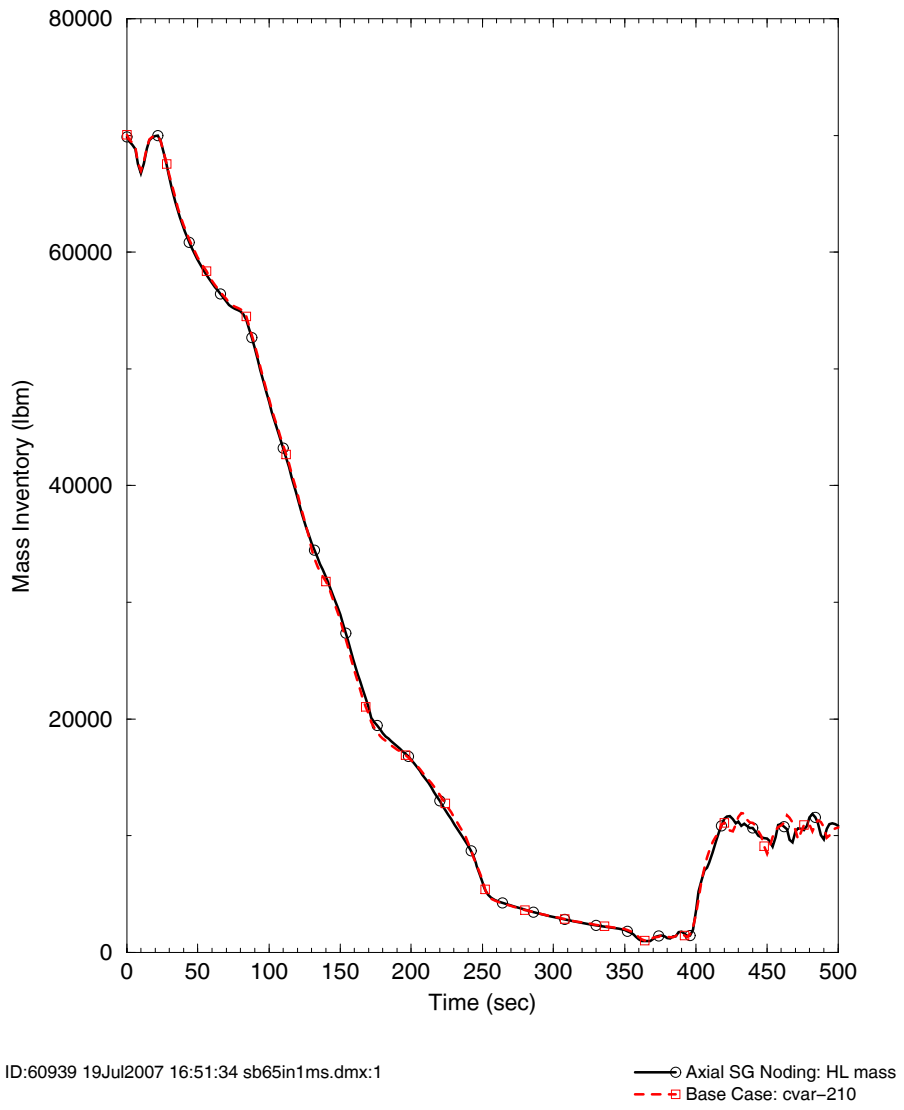

Figure 2-11 Axial Nodalization Study - SG-1 Hot Leg Mass for a 6.5 inch Break

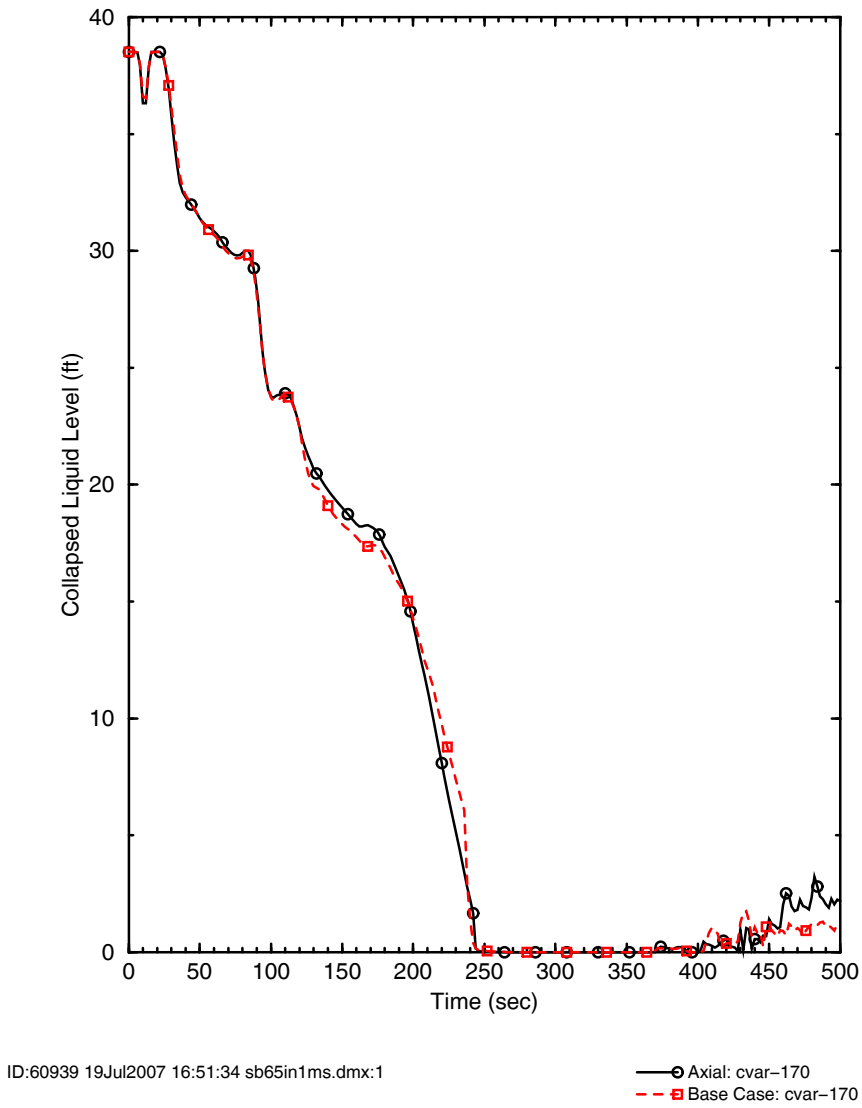

Figure 2-12 Axial Nodalization Study - SG-1 Upside Collapsed Liquid Level for a 6.5 inch Break

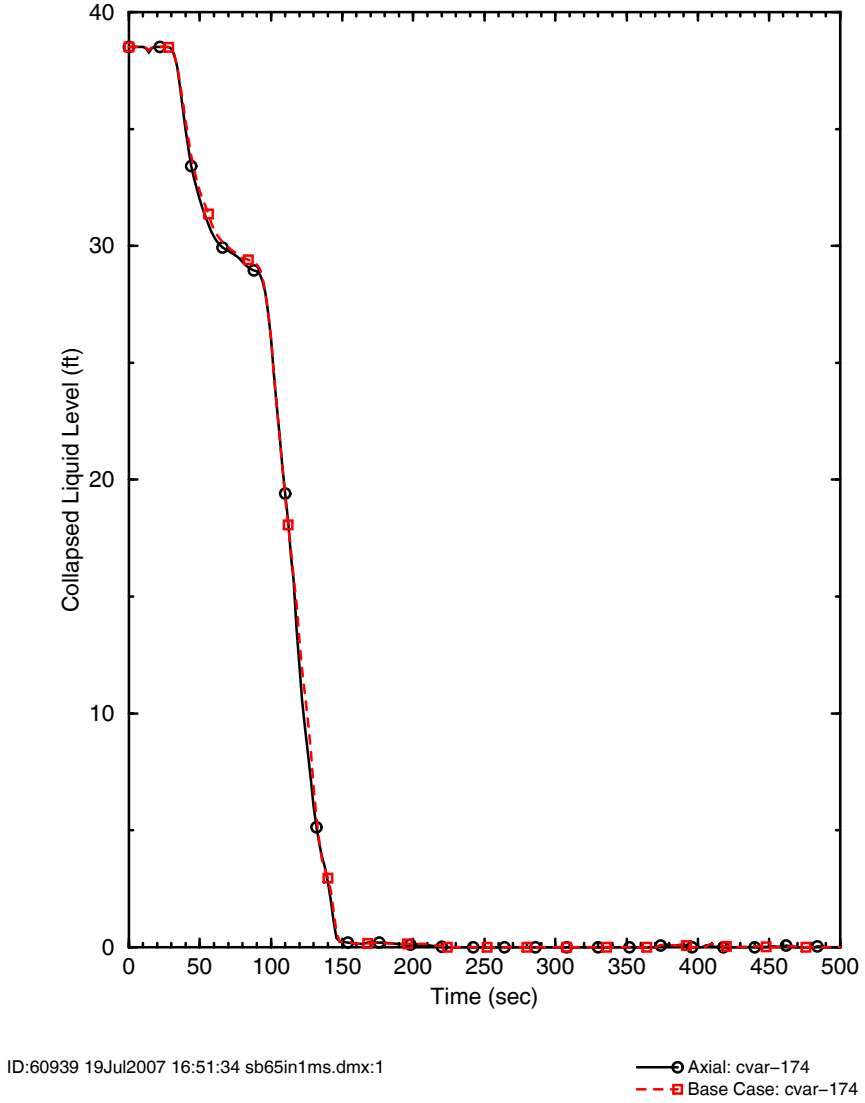

Figure 2-13 Axial Nodalization Study - SG-1 Downside Collapsed Liquid Level for a 6.5 inch Break

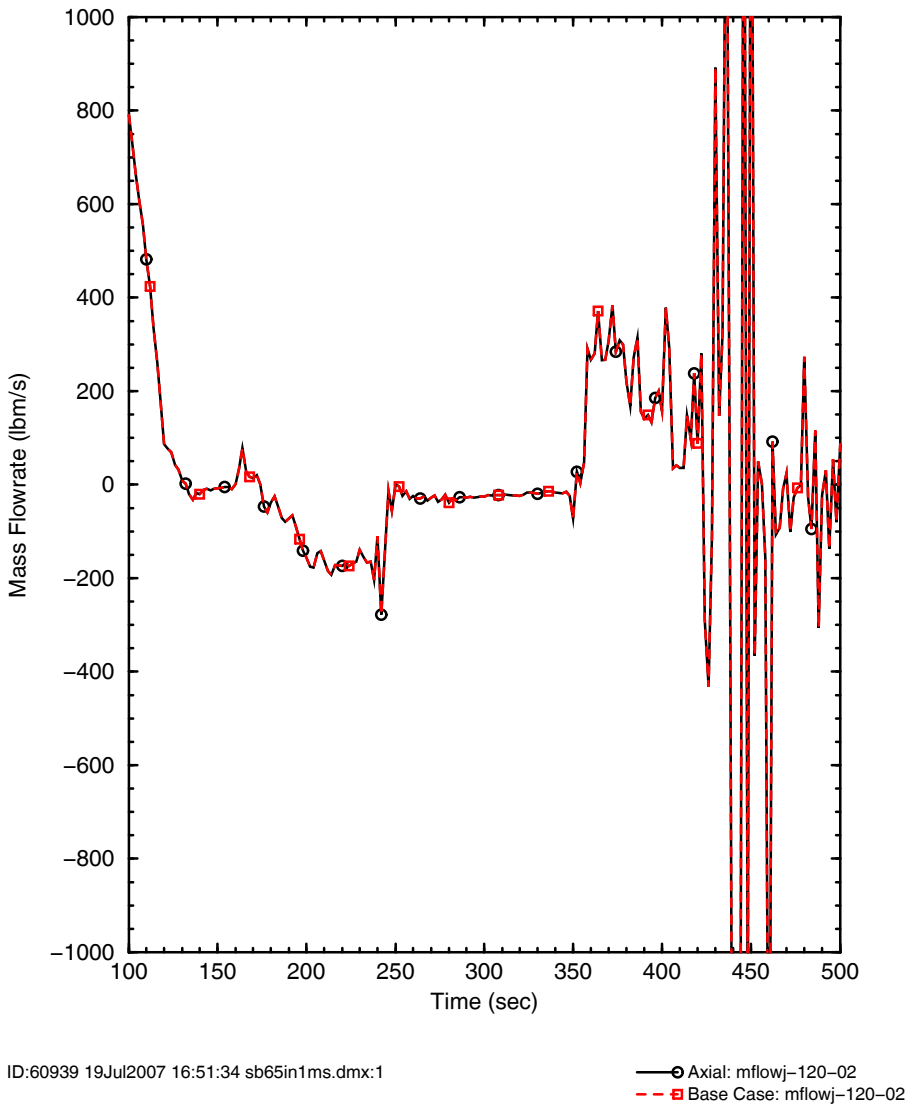

Figure 2-14 Axial Nodalization Study - SG-1 Inlet Flowrate for a 6.5 inch Break

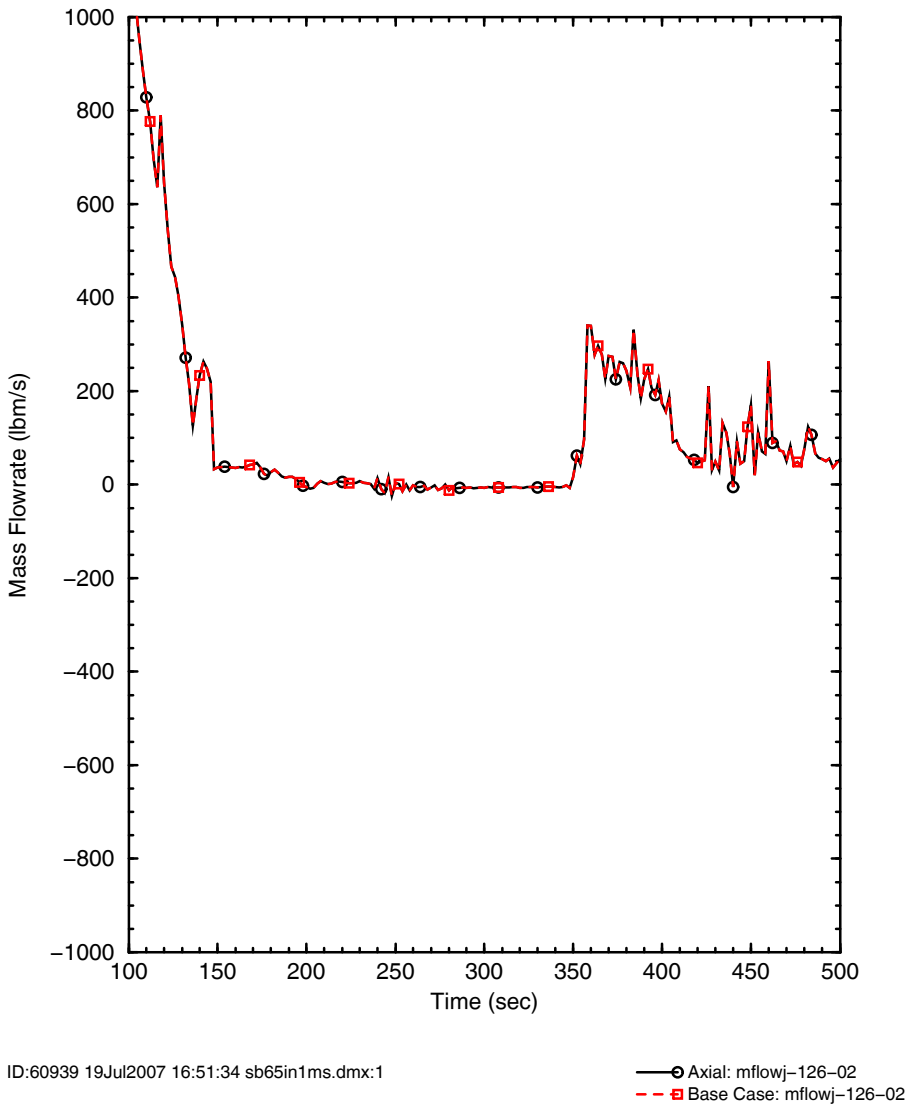

Figure 2-15 Axial Nodalization Study - SG-1 Outlet Flowrate for a 6.5 inch Break

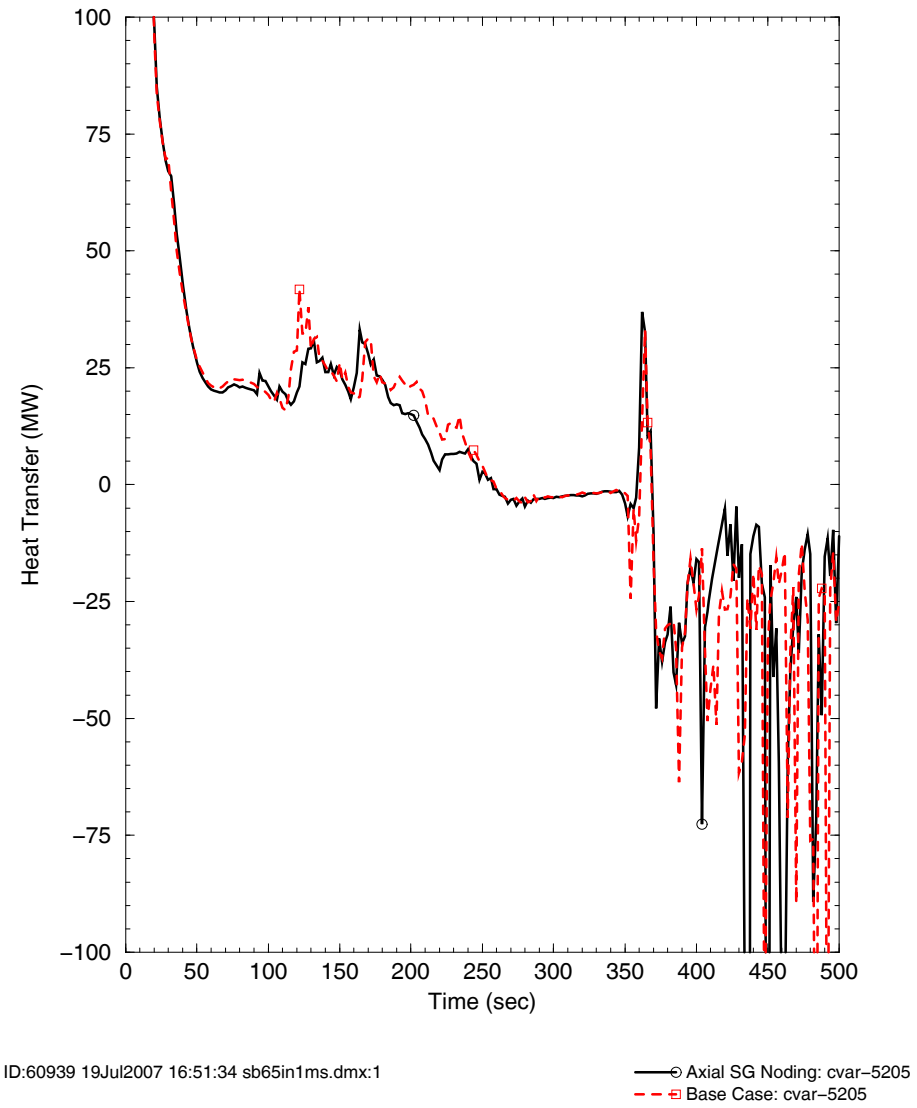

Figure 2-16 Axial Nodalization Study - Loop SG-1 Tube Heat Transfer for a 6.5 inch Break

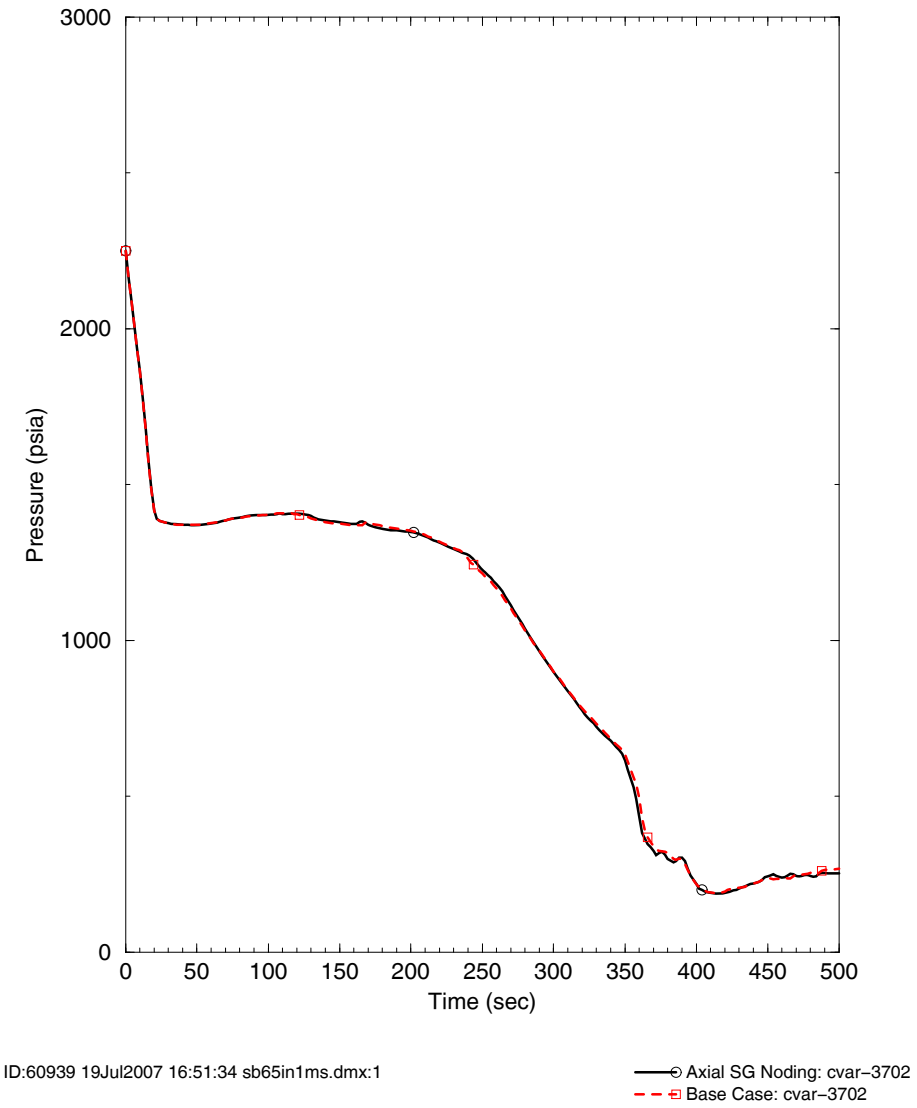

Figure 2-17 Axial Nodalization Study - Pressurizer Pressure for a 6.5 inch Break

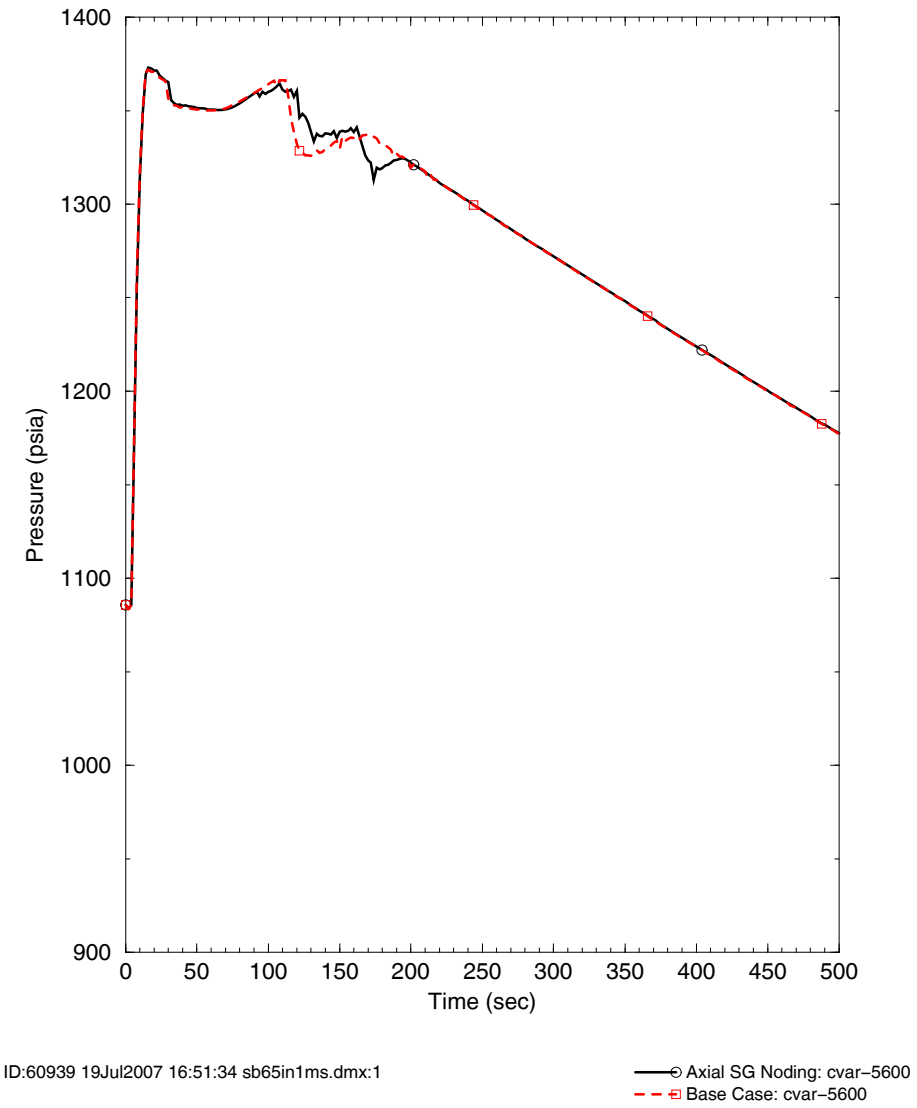

Figure 2-18 Axial Nodalization Study - Loop 1 Secondary Pressure for a 6.5 inch Break

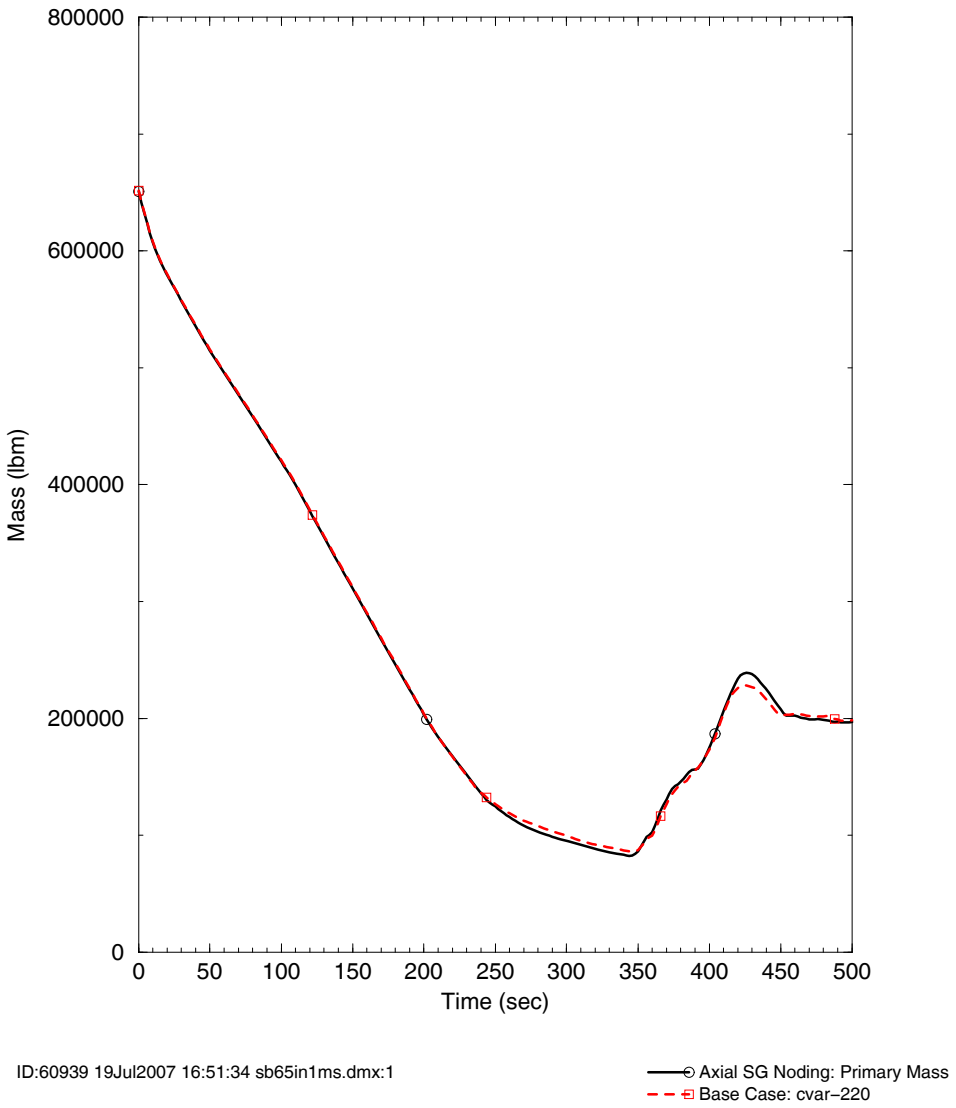

Figure 2-19 Axial Nodalization Study - RCS Fluid Mass for a 6.5 inch Break

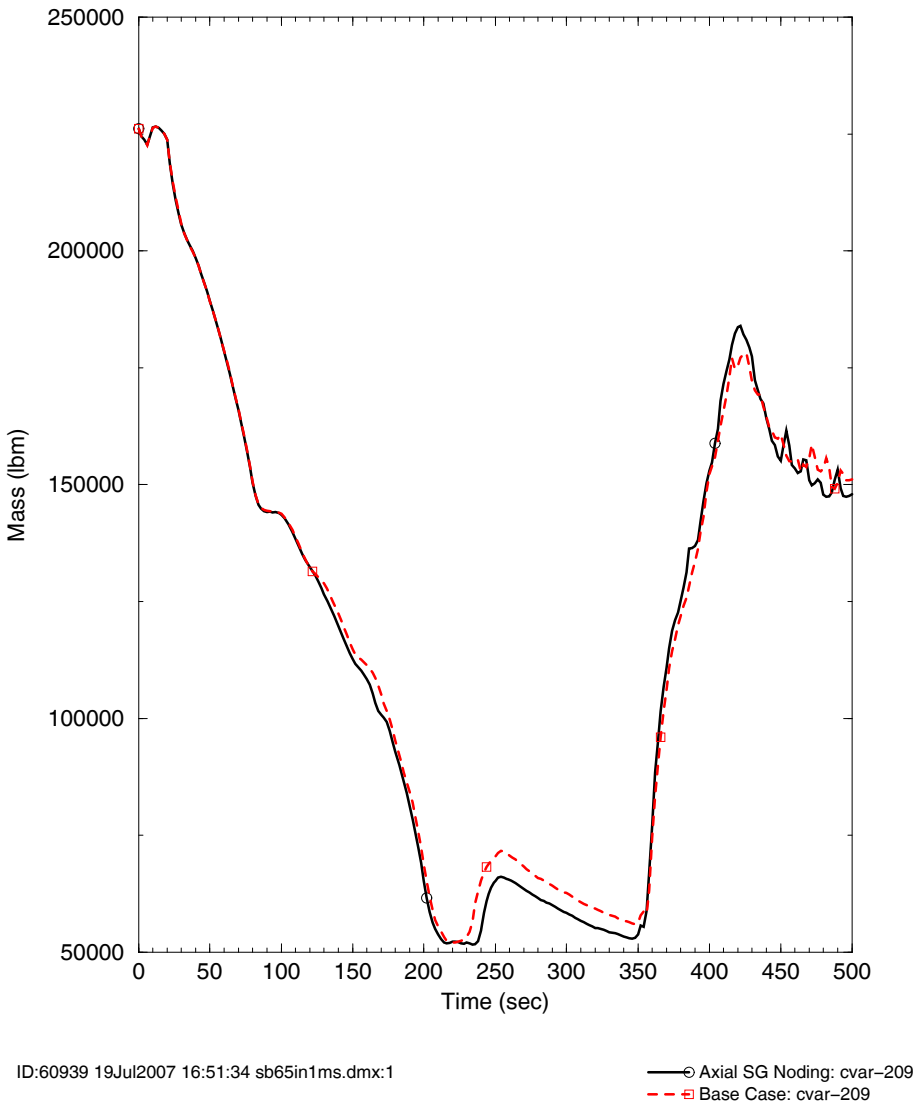

Figure 2-20 Axial Nodalization Study - Reactor Vessel Fluid Mass for a 6.5 inch Break

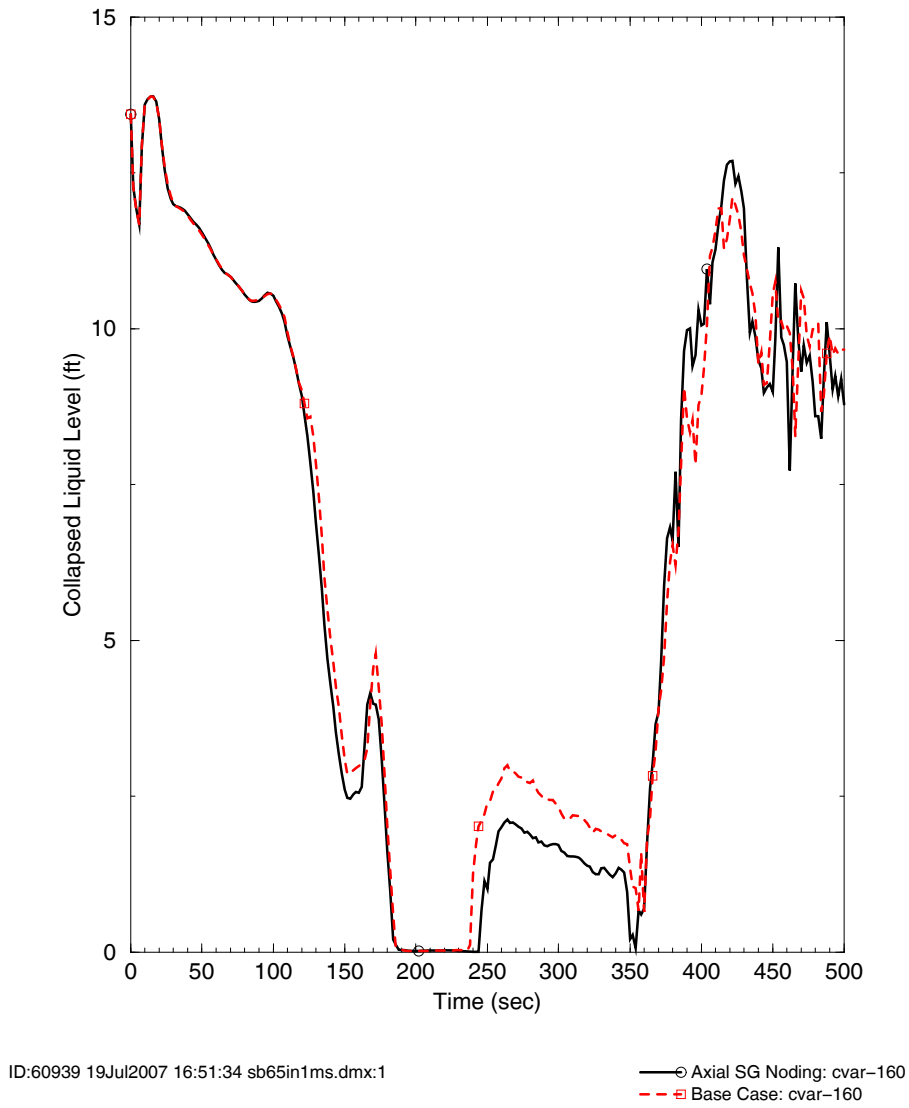

Figure 2-21 Axial Nodalization Study - Hot Assembly Collapsed Liquid Level for a 6.5 inch Break

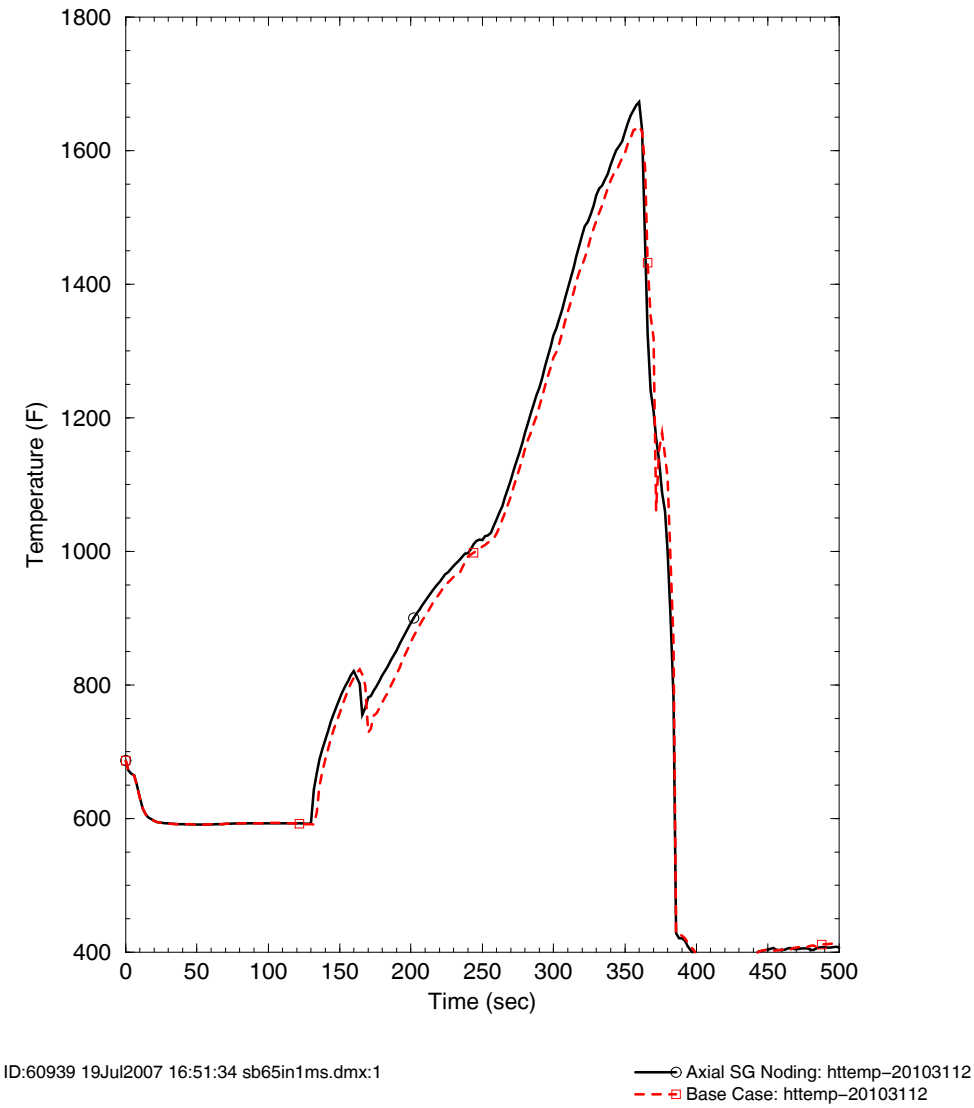

Figure 2-22 Axial Nodalization Study - PCT for a 6.5 inch Break

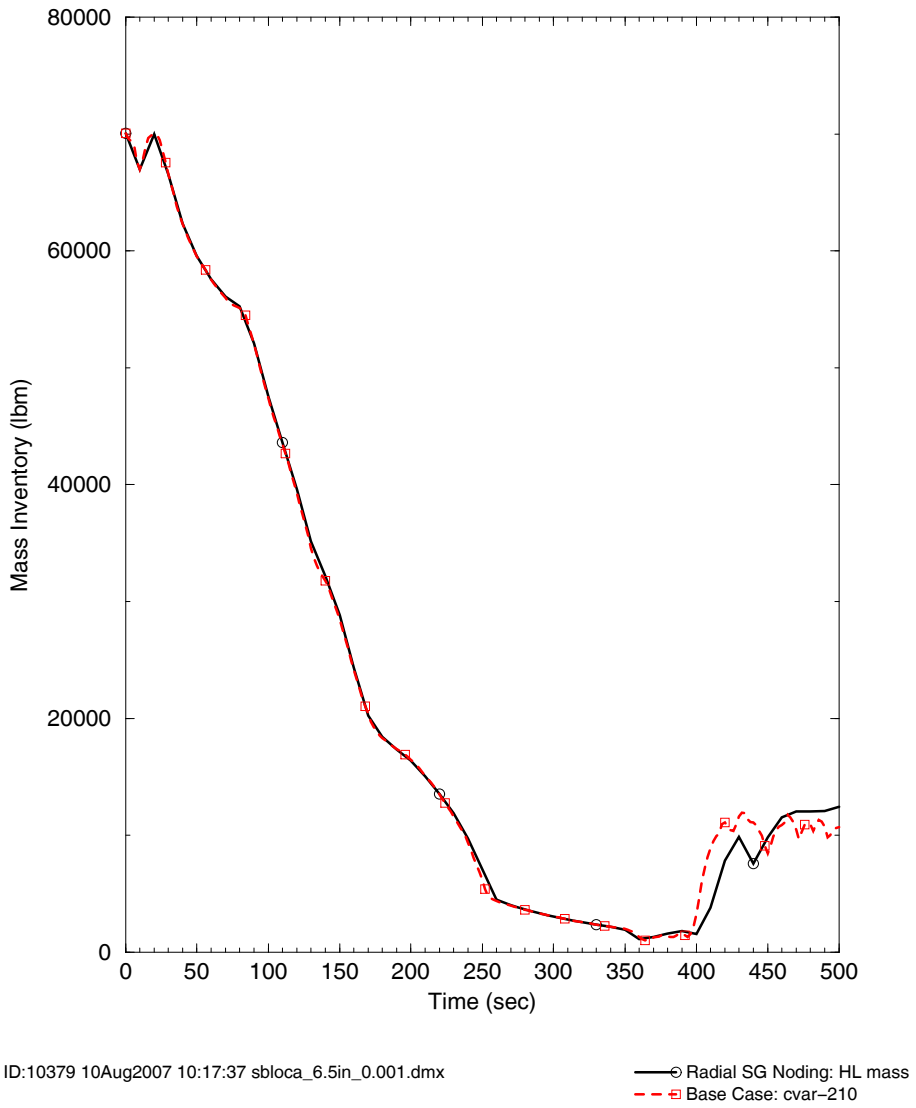

Figure 2-23 Radial Nodalization Study - SG-1 Hot Leg Mass for a 6.5 inch Break

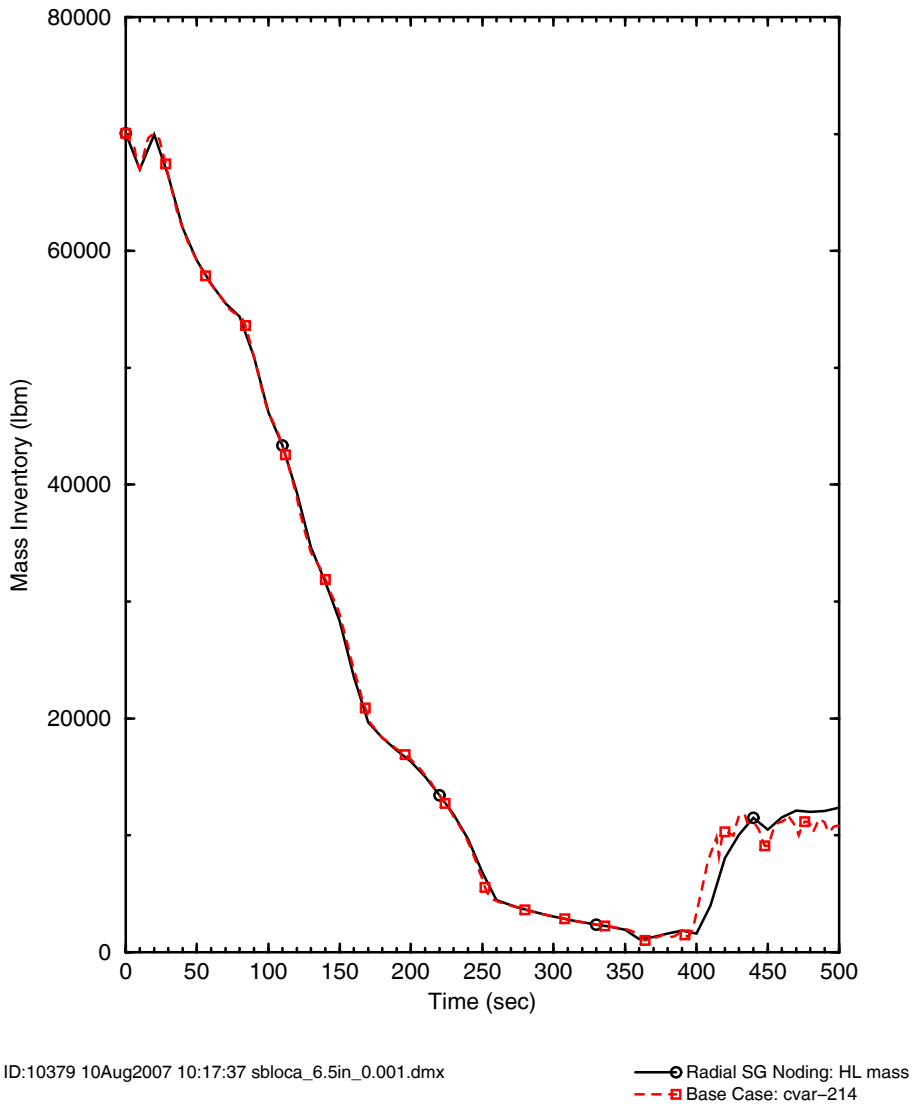

Figure 2-24 Radial Nodalization Study - SG-4 Hot Leg Mass for a 6.5 inch Break

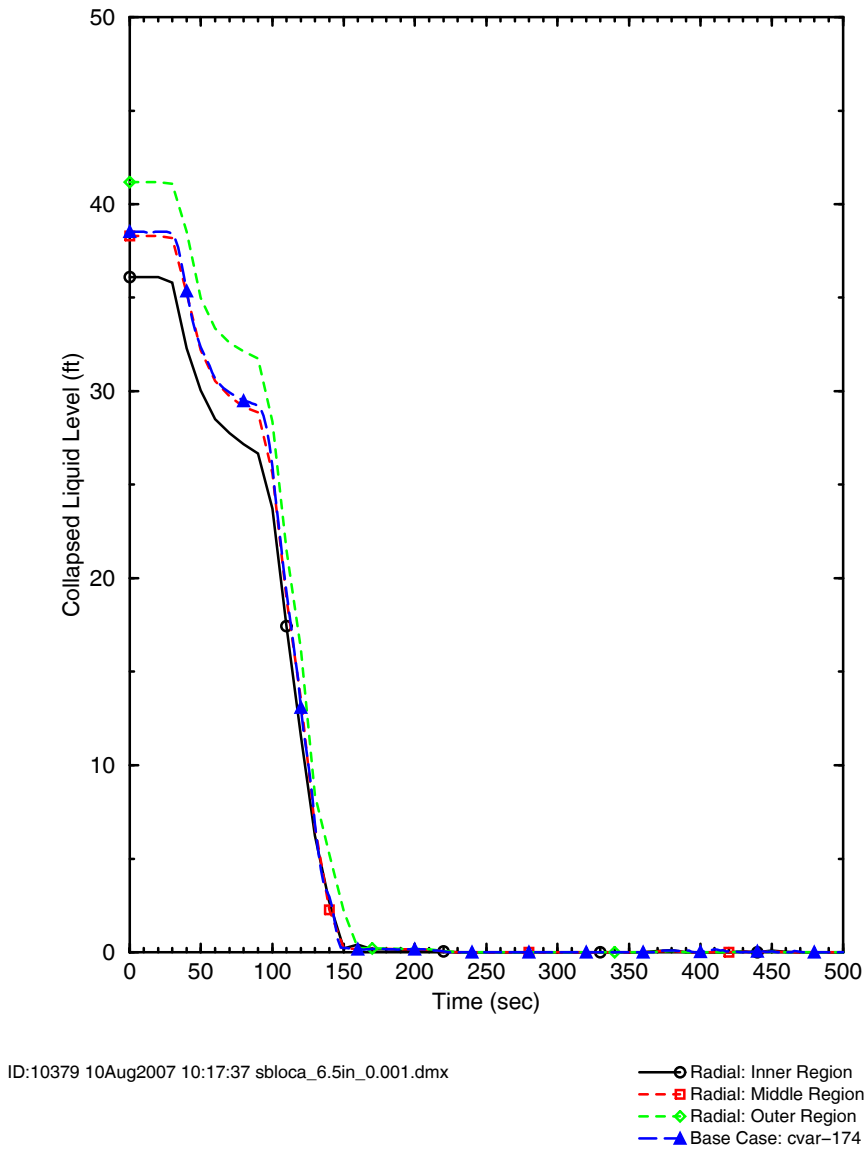

Figure 2-25 Radial Nodalization Study - SG-1 Downside Collapsed Liquid Level for a 6.5 inch Break

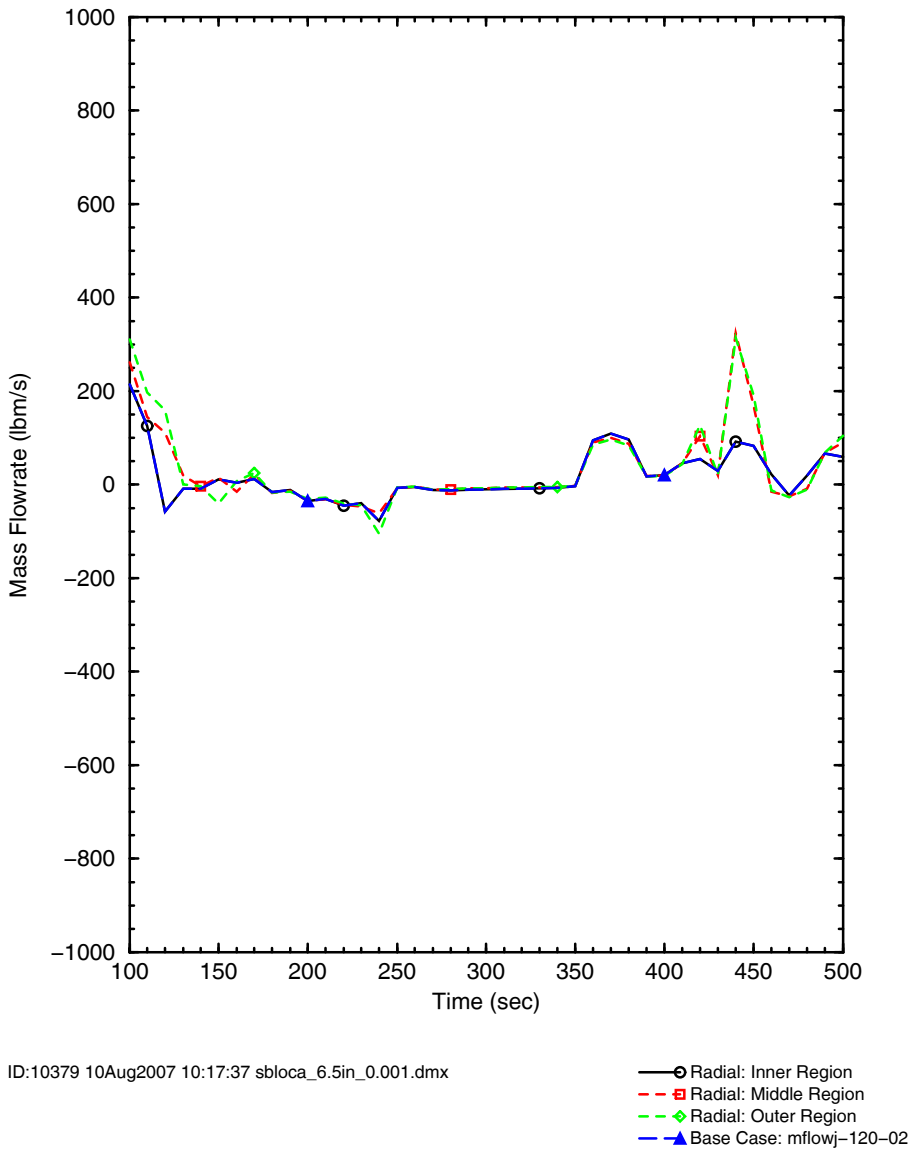

Figure 2-26 Radial Nodalization Study - SG-1 Inlet Flow for a 6.5 inch Break

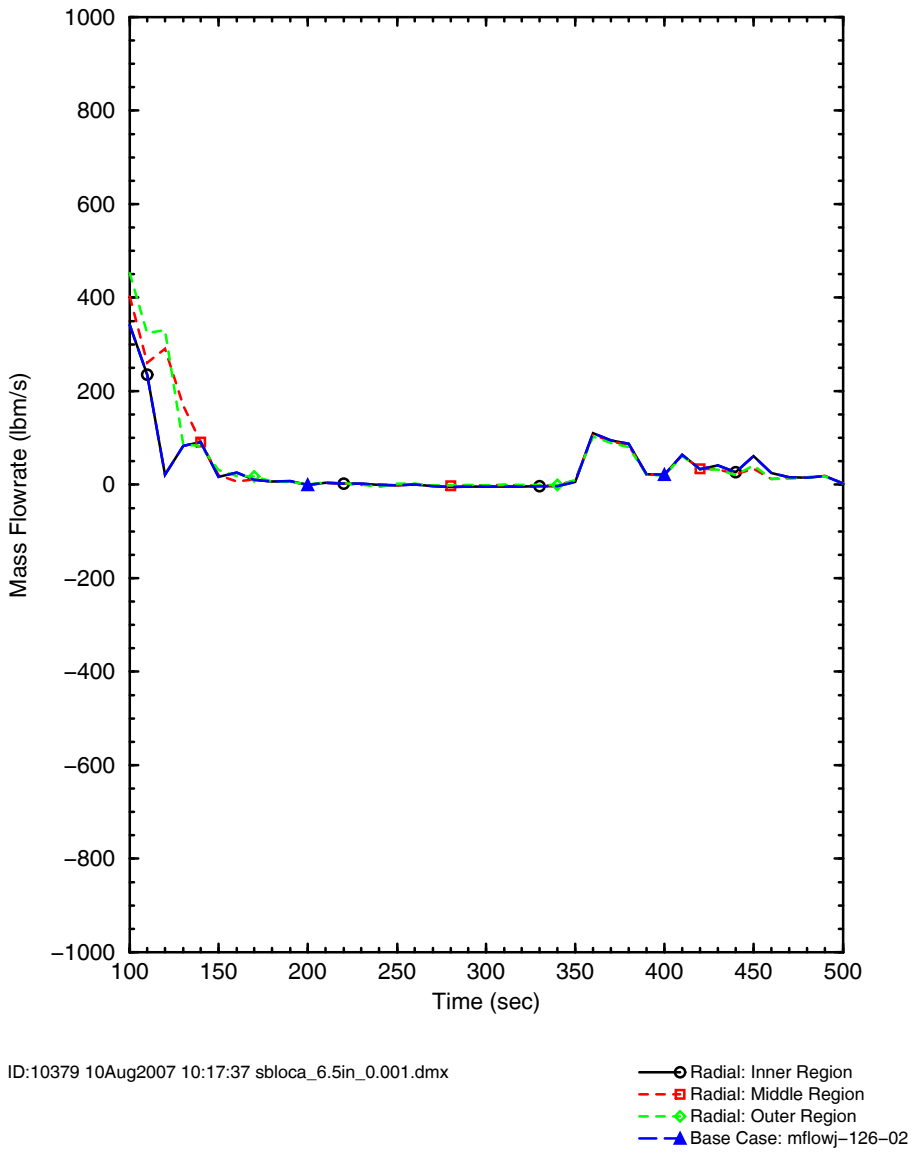

Figure 2-27 Radial Nodalization Study - SG-1 Outlet Flowrate for a 6.5 inch Break

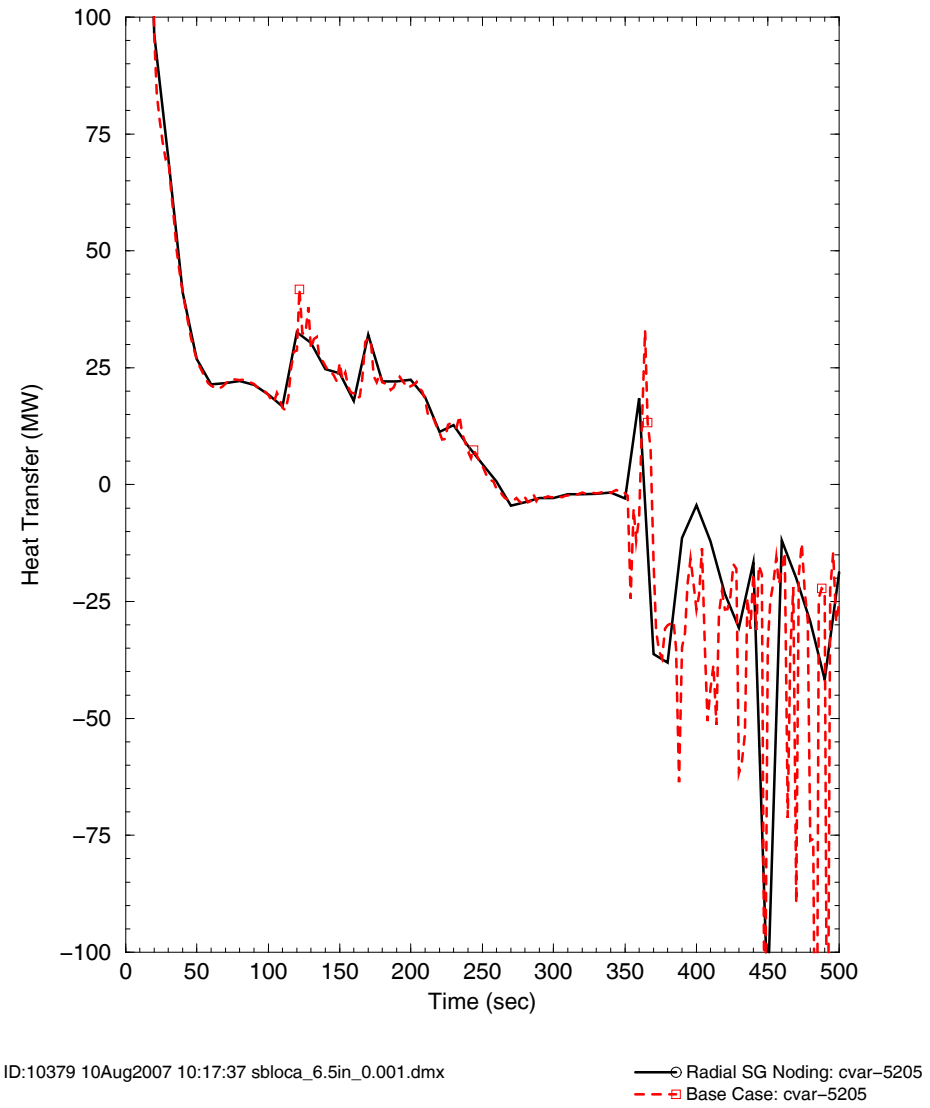

Figure 2-28 Radial Nodalization Study - Loop 1 SG Tube Heat Transfer for a 6.5 inch Break

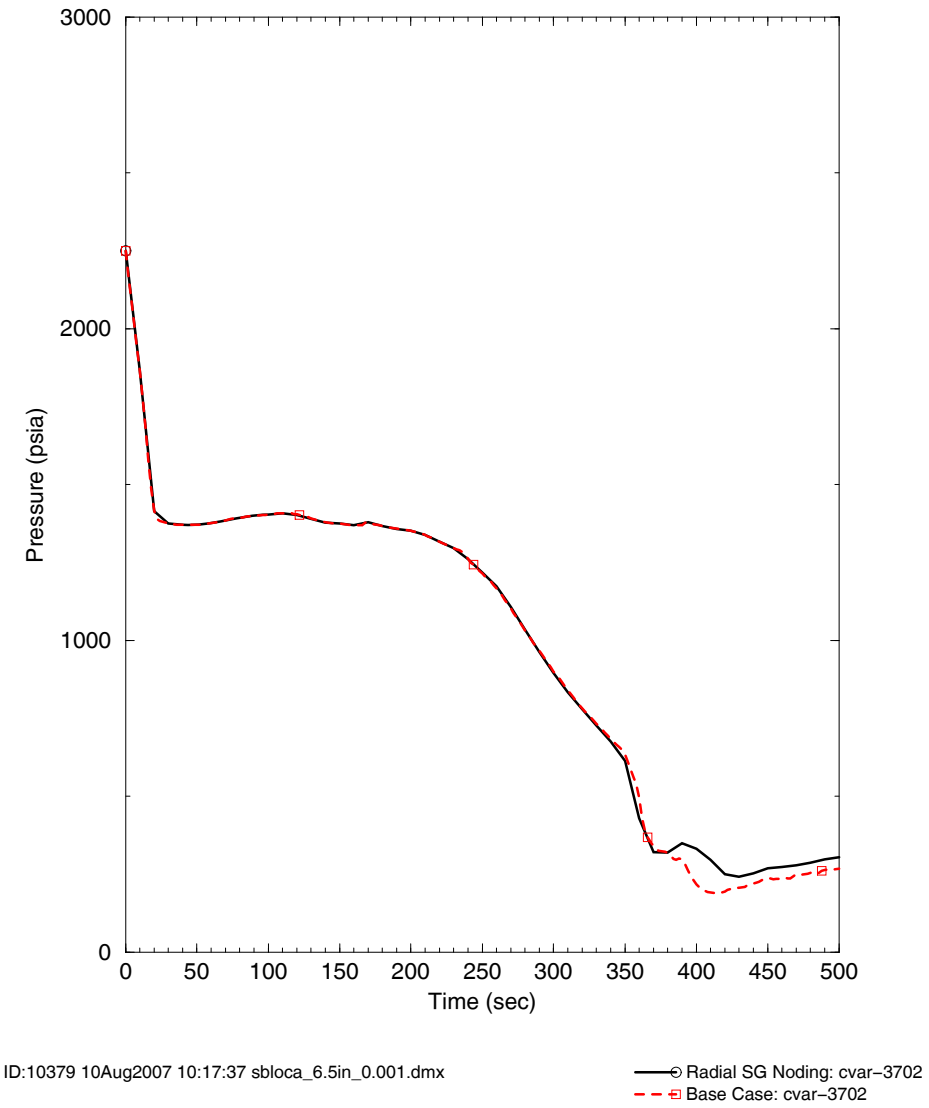

Figure 2-29 Radial Nodalization Study - Pressurizer Pressure for a 6.5 inch Break

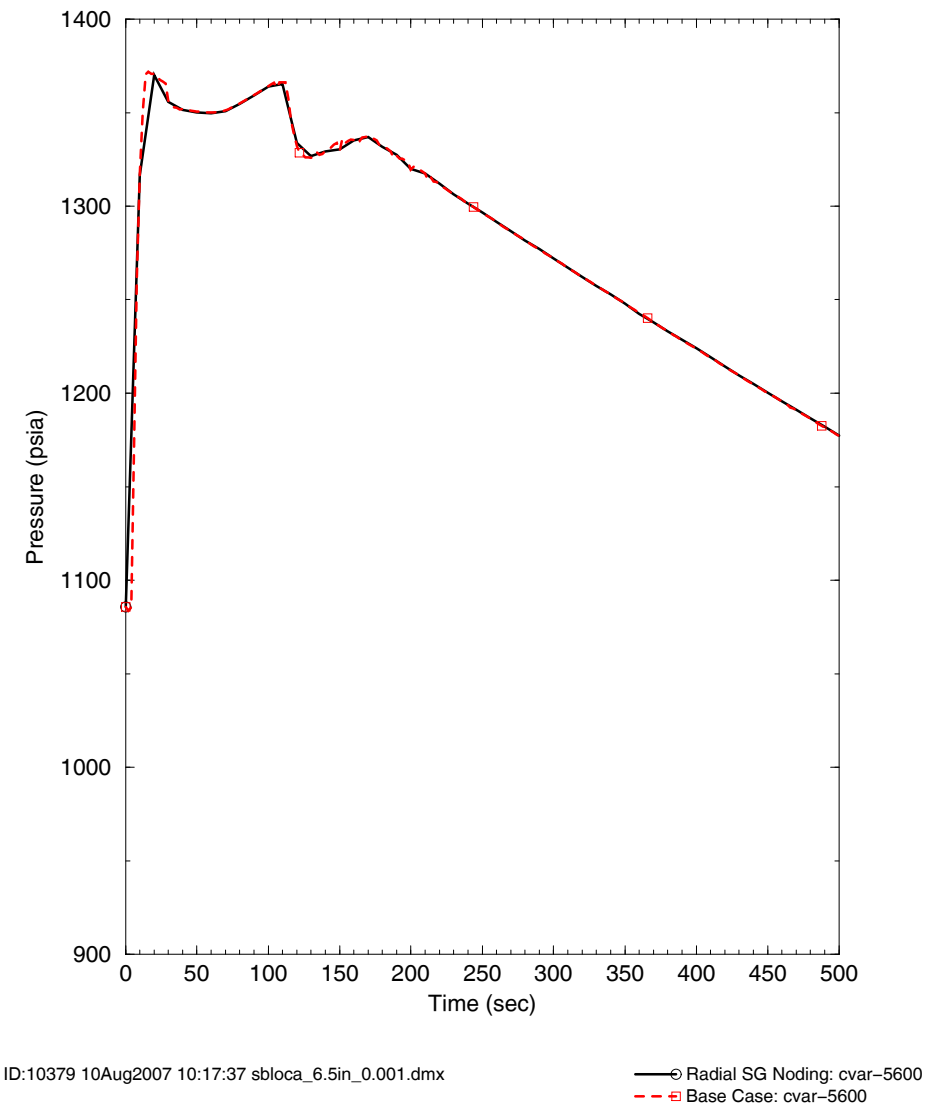

Figure 2-30 Radial Nodalization Study - Loop 1 Secondary Pressure for a 6.5 inch Break

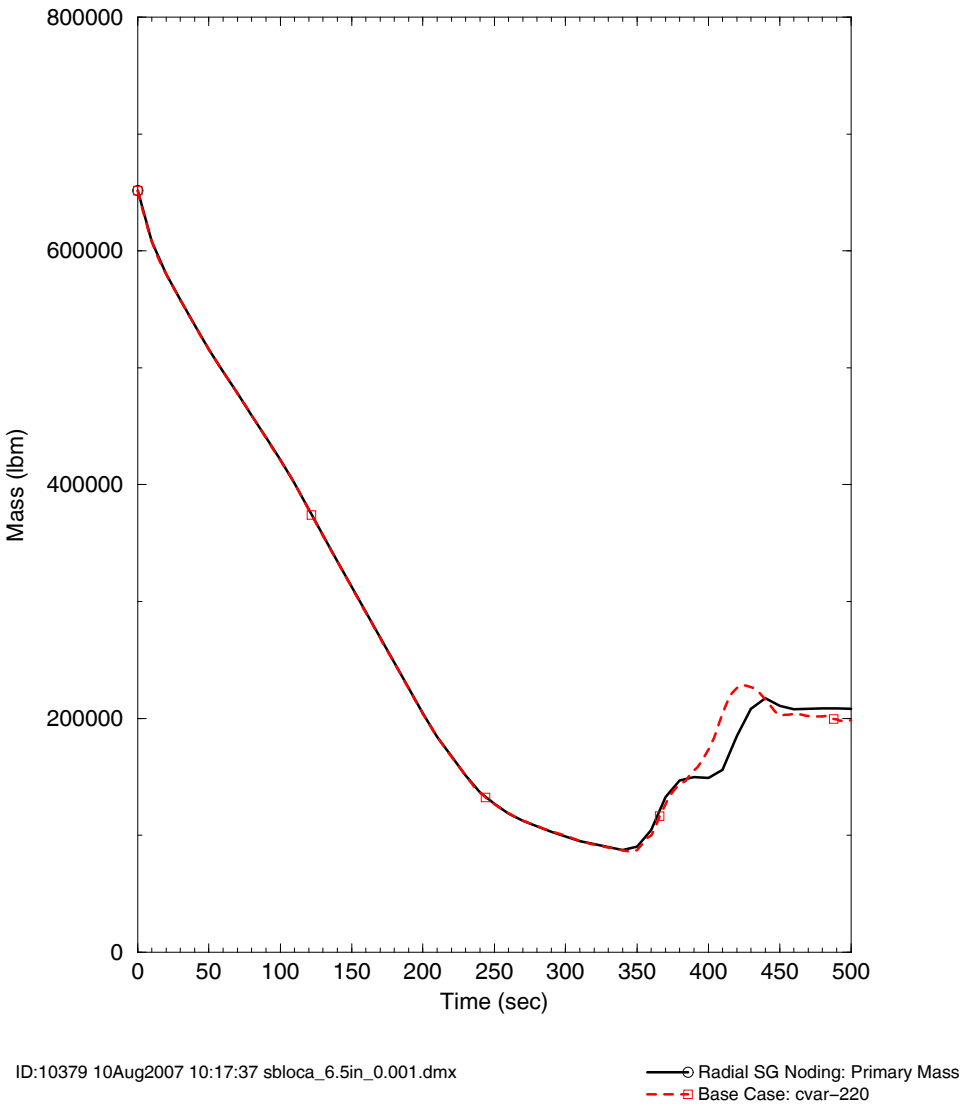

Figure 2-31 Radial Nodalization Study - RCS Fluid Mass for a 6.5 inch Break

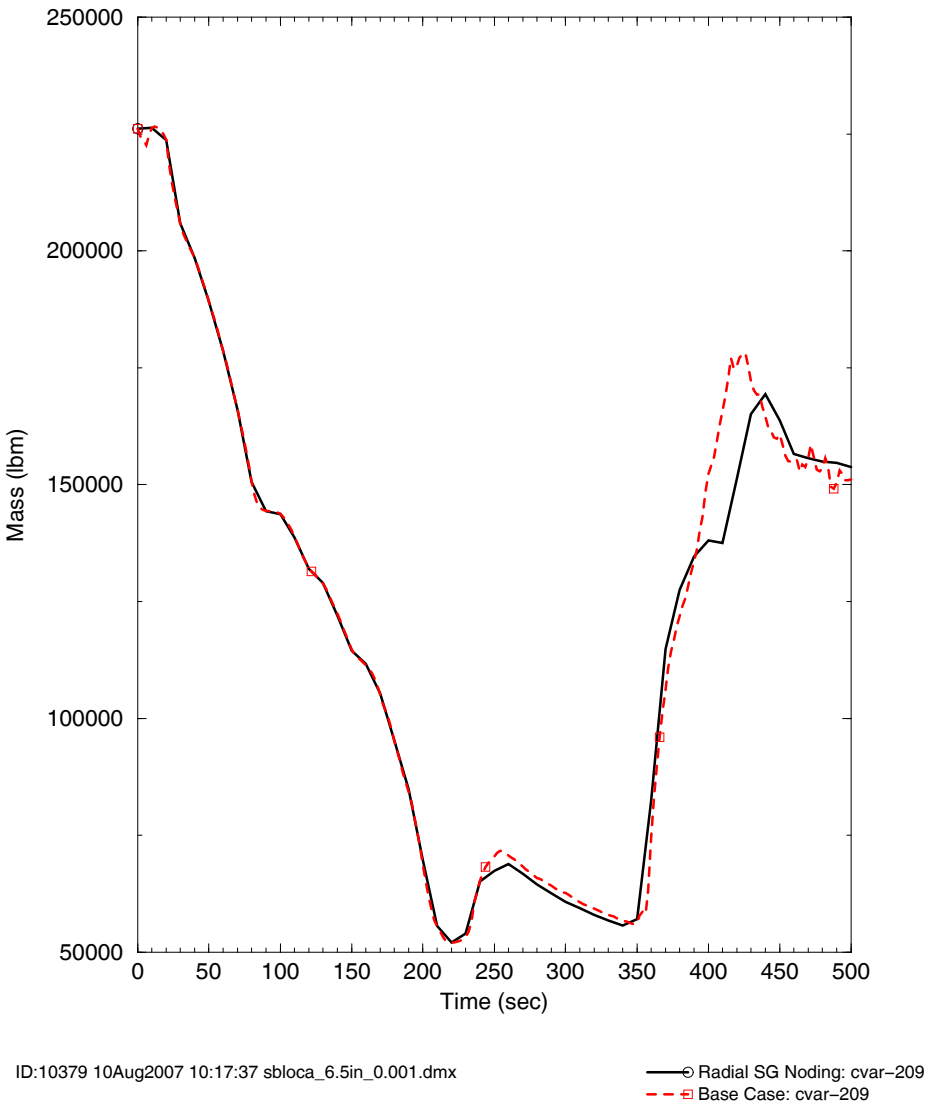

Figure 2-32 Radial Nodalization Study - Reactor Vessel Fluid Mass for a 6.5 inch Break

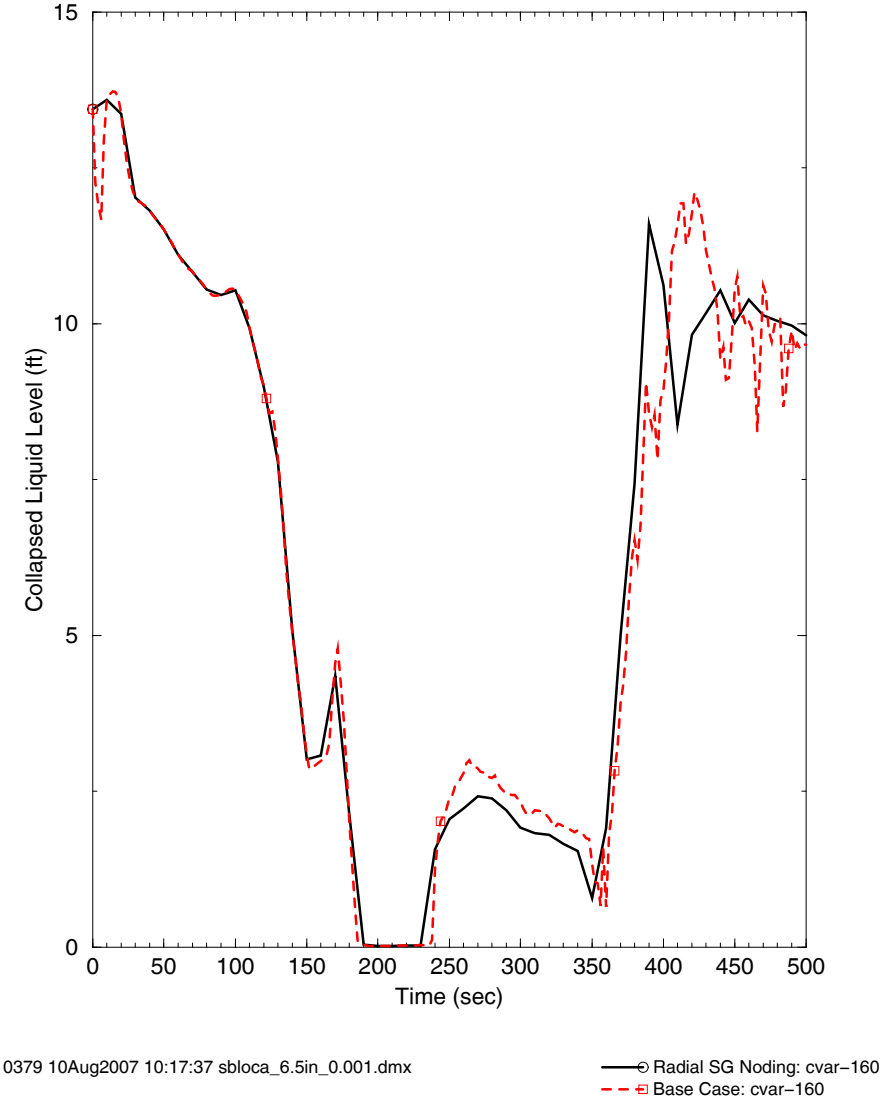

Figure 2-33 Radial Nodalization Study - Hot Assembly Collapsed Liquid Level for a 6.5 inch Break

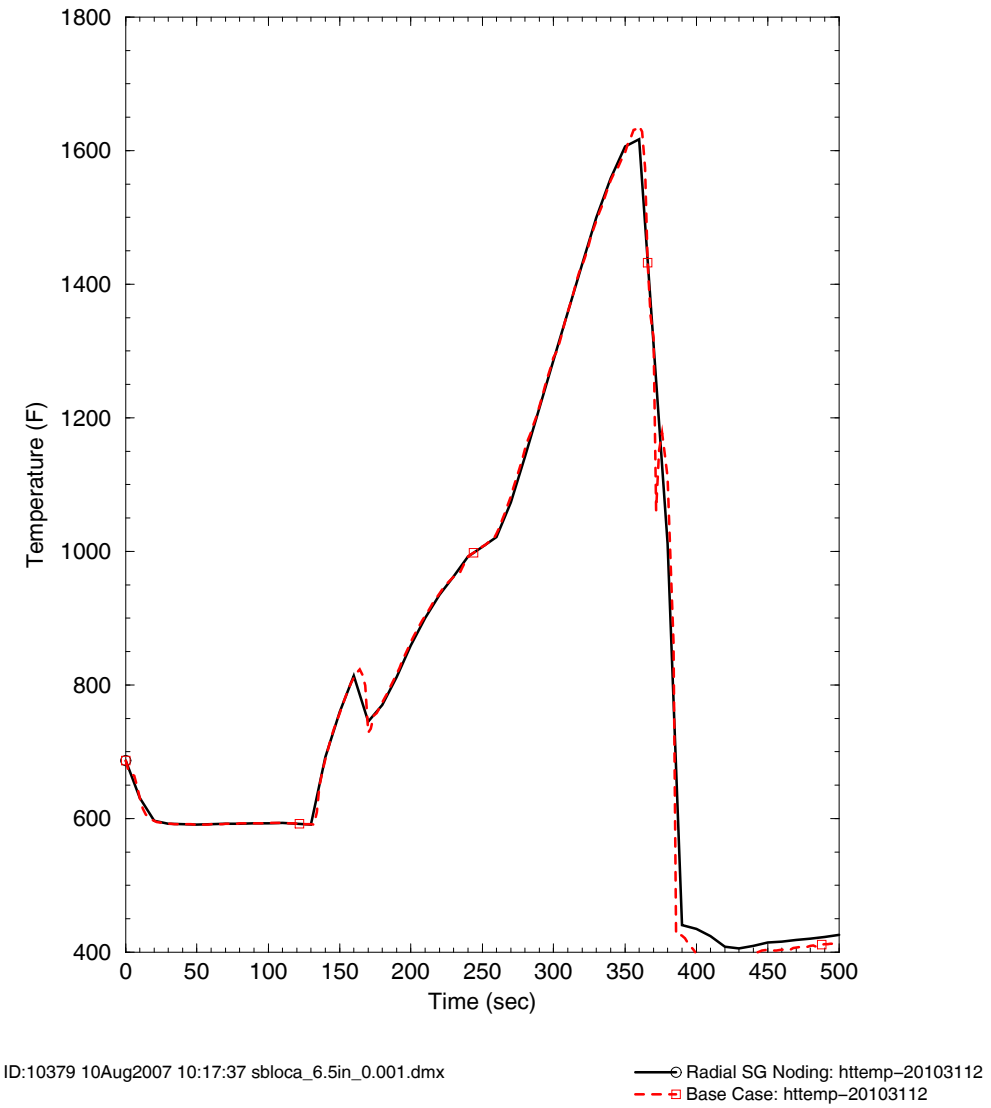

Figure 2-34 Radial Nodalization Study - PCT for a 6.5 inch Break

3.0 NON-LOCA

3.1 *Non-LOCA SG Nodalization Studies*

A series of sensitivity studies were performed to verify the adequacy of the steam generator nodalization for non-LOCA analyses. The three steam generator nodalizations used for the sensitivity studies are described below. The steam generator nodalizations were applied to all four steam generators; however, only the changes to the loop 1 steam generator are provided in the nodalization descriptions. It should be noted that the emergency feedwater (EFW) junction connection location for these studies differs from the location used in the sample problem presented in ANP-10263PA. The EFW junction connection location was changed to better represent the actual EFW spray ring location for the U.S. EPR steam generators.

3.1.1 *Description of Models Used in Nodalization Studies*

3.1.1.1 *Base Model*

[ ]

3.1.1.2 *Case 1*

[ ]

[ ]

3.1.1.3 *Case 2*

[ ]

3.1.1.4 *Case 3*

[ ]

The following sensitivity studies were performed with the three steam generator nodalizations:

Steam Generator Nodalization Sensitivity Study Case Matrix

	Turbine Trip	Loss of Feed Flow	Increase in Steam Flow
Case 1	**X**		
Case 2	**X**		
Case 3	**X**	**X**	**X**

3.1.2 *Sensitivity Study Results*

3.1.2.1 *Turbine Trip Results*

The Turbine Trip transient results for the three steam generator nodalizations are presented in Figure 3-4 through Figure 3-18. The calculations show that the three types of steam generator nodalizations produce results very similar to the base case results.

The base case peak reactor coolant system (RCS) pressure was 2759.1 psia. The calculated peak RCS pressures for Cases 1, 2, and 3 are 2759.1, 2760.6, and 2760.4 psia respectively. The base case peak steam generator pressure was 1485.1 psia. Peak steam generator pressures for Cases 1, 2, and 3 are 1484.1, 1487.0, and 1487.2 psia, respectively. Thus, the RCS and steam generator peak pressures change little as a result of the changed steam generator nodalizations.

3.1.2.2 *Loss of Feed Flow Results*

The Case 3 Loss of Feed Flow transient results are provided in Figure 3-19 through Figure 3-30. The Case 3 results for the early portion of the transient (<40 seconds) are very similar to the base case results. The difference in results between the Case 3 Loss of Feed Flow case and the base case during the later portion of the transient (>40 seconds) can be attributed to the earlier reactor trip for Case 3. Figure 3-24 through Figure 3-27 show the calculated narrow range steam generator level results versus the base case results for all four steam generators. The base case results show that all four steam generators have nearly the same level transients. The base case steam generator levels decrease smoothly until about 40 seconds at which point there is a “knee” in the level curve. The comparable Case 3 results show that between 30 and 40 seconds only one steam generator (SG3) exhibits a small “knee” while steam generators 1, 2, and 4 continue to decrease without the knee. The lower level in these steam generators is responsible for the earlier reactor trip. Earlier reactor trip leads to a more benign transient for the Case 3 steam generator nodalization.

The conclusion from these results is that the consistency of results before and after reactor trip and the conservatism of the base case model in predicting time of reactor trip support use of the base nodalization for the loss of feed flow and similar transients.

3.1.2.3 *Increase in Steam Flow Results*

The Case 3 Increase in Steam Flow transient results are provided in Figure 3-31 through Figure 3-40. The Case 3 Increase in Steam Flow transient results show that the primary system parameters (power, RCS flow, RCS pressures, and RCS temperatures) are little changed from the base case results.

3.1.3 *Conclusions*

The three steam generator nodalization studies for Turbine Trip provided results that differed insignificantly from the base case.

The Loss of Feed Flow and Increase in Steam Flow events were also analyzed using the combined Case 3 nodalization. For the Loss of Feed Flow event, the increase in the number of steam generator nodes in the downcomer region for Case 3 resulted in a difference in the calculation of the NR steam generator level. As a result, Case 3 predicted in an earlier reactor trip on low steam generator level. Other than this shift in event time, the Case 3 results are very similar to the base results. Therefore, it can be concluded that the steam generator nodalization used for the base case is adequate and provides a slightly more conservative result than the Case 3 nodalization for this event.

The results for the Case 3 Increase in Steam Flow event are very similar to the base case results; therefore, the sensitivity calculation supports use of the base steam generator nodalization for U.S. EPR non-LOCA transient analyses.

The sensitivity cases showed no significant differences relative to the base case for the Turbine Trip and Increase in Steam Flow events. In addition, the base model is conservative for the Loss of Feed Flow event. Therefore, the base steam generator nodalization is acceptable for non-LOCA applications.

Figure 3-1 Base Model Steam Generator Nodalization

Figure 3-2 Case 2 Steam Generator Nodalization

Figure 3-3 Case 3 Steam Generator Nodalization

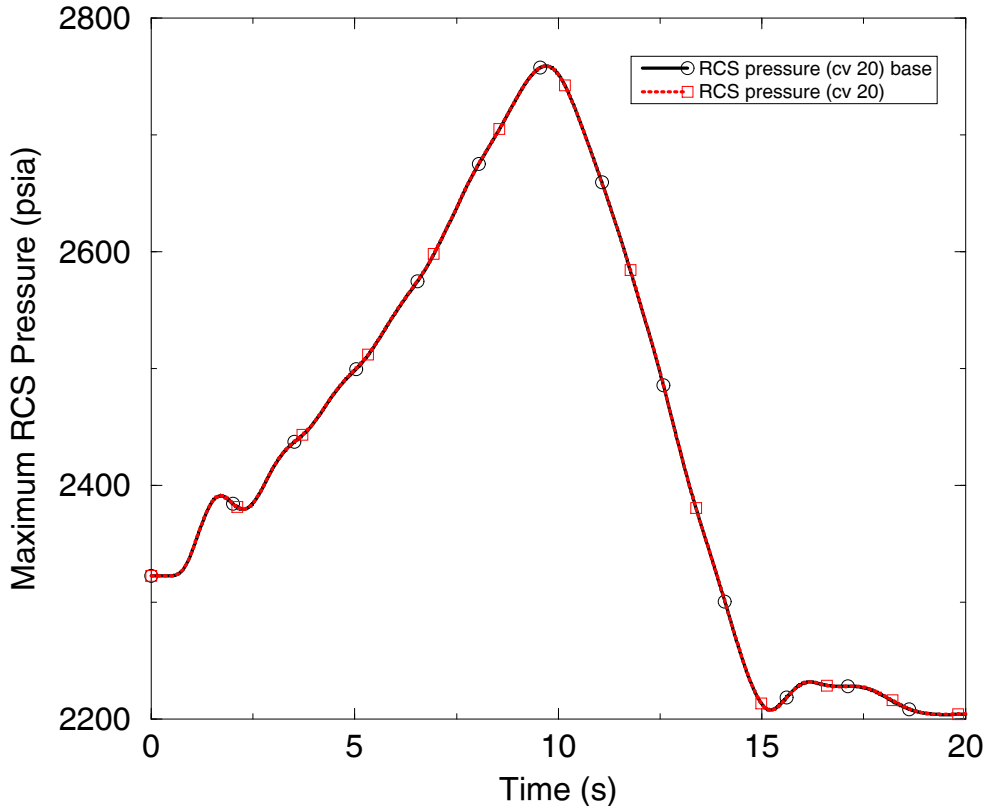

Figure 3-4 Case 1 Turbine Trip – Maximum RCS Pressure

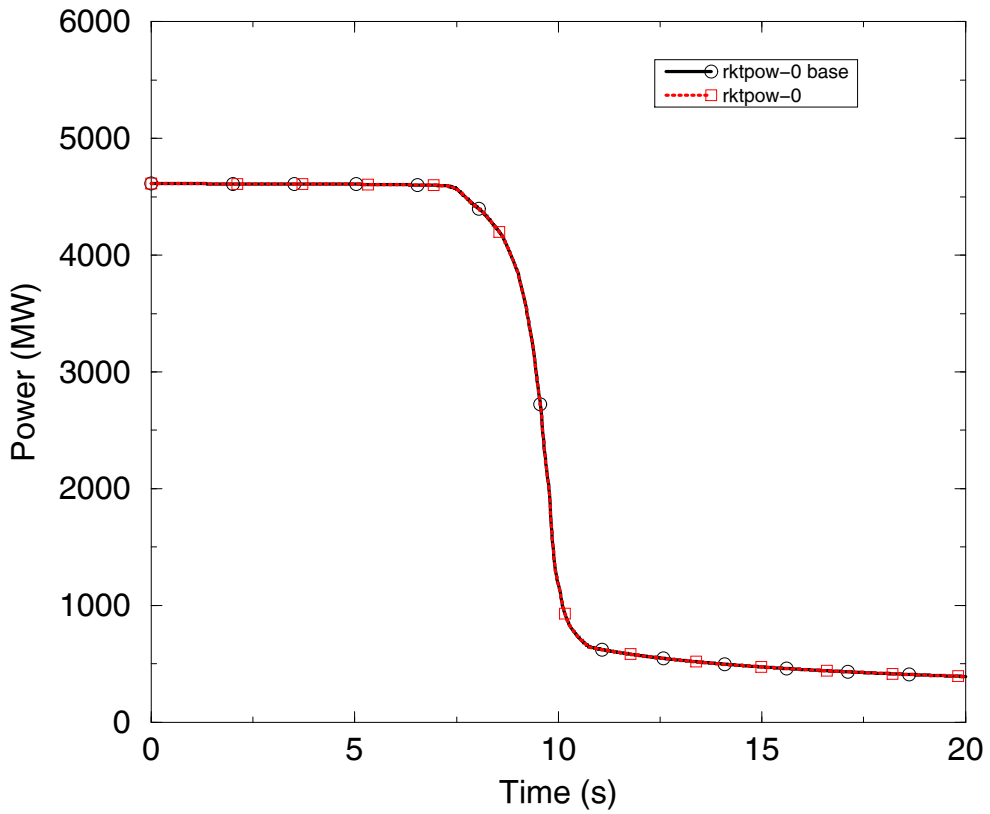

Figure 3-5 Case 1 Turbine Trip – Reactor Power

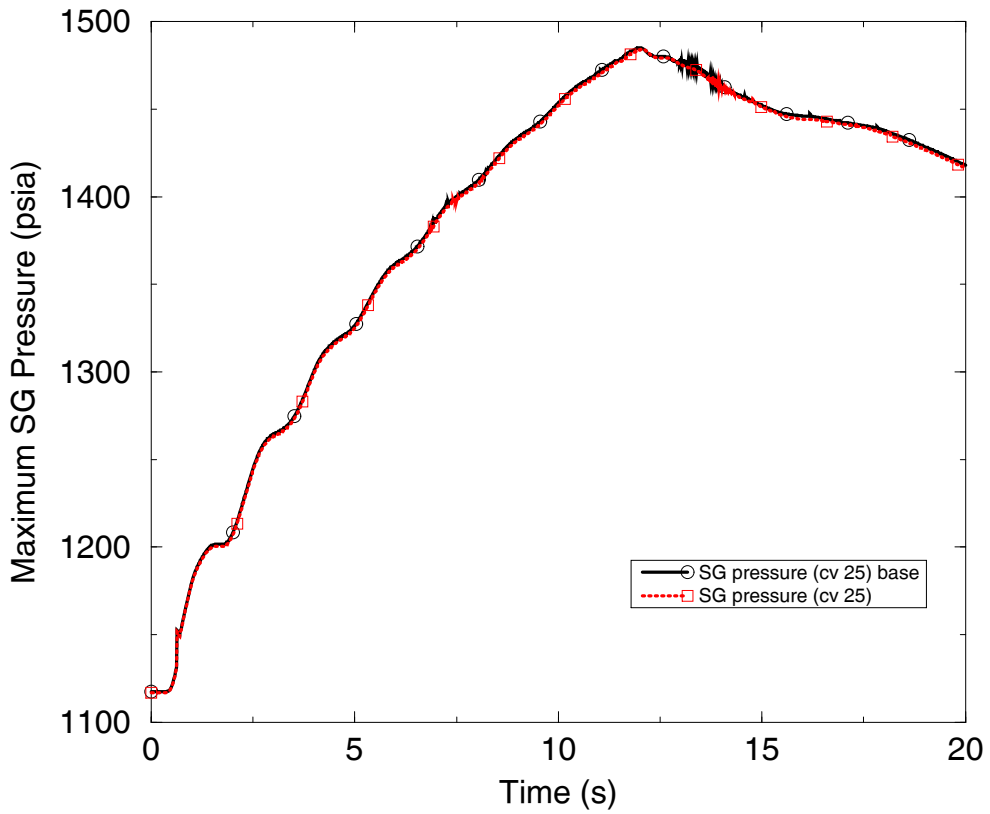

Figure 3-6 Case 1 Turbine Trip – Maximum Steam Generator Pressure

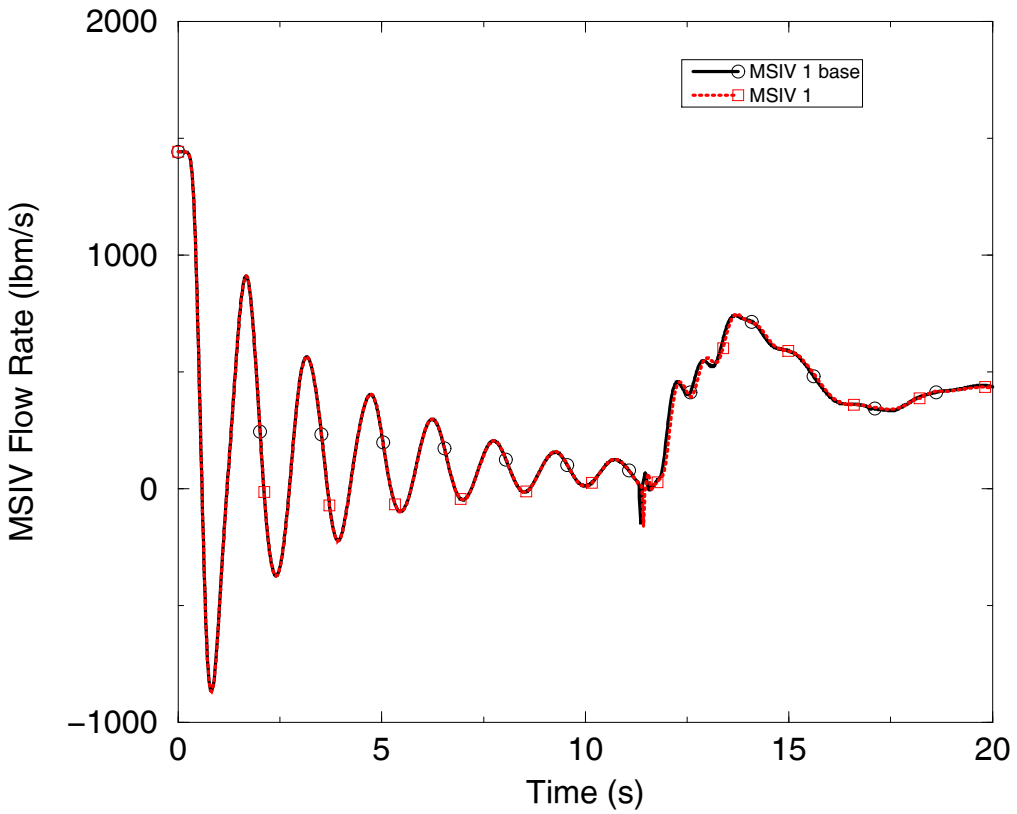

Figure 3-7 Case 1 Turbine Trip – MSIV1 Flow Rate

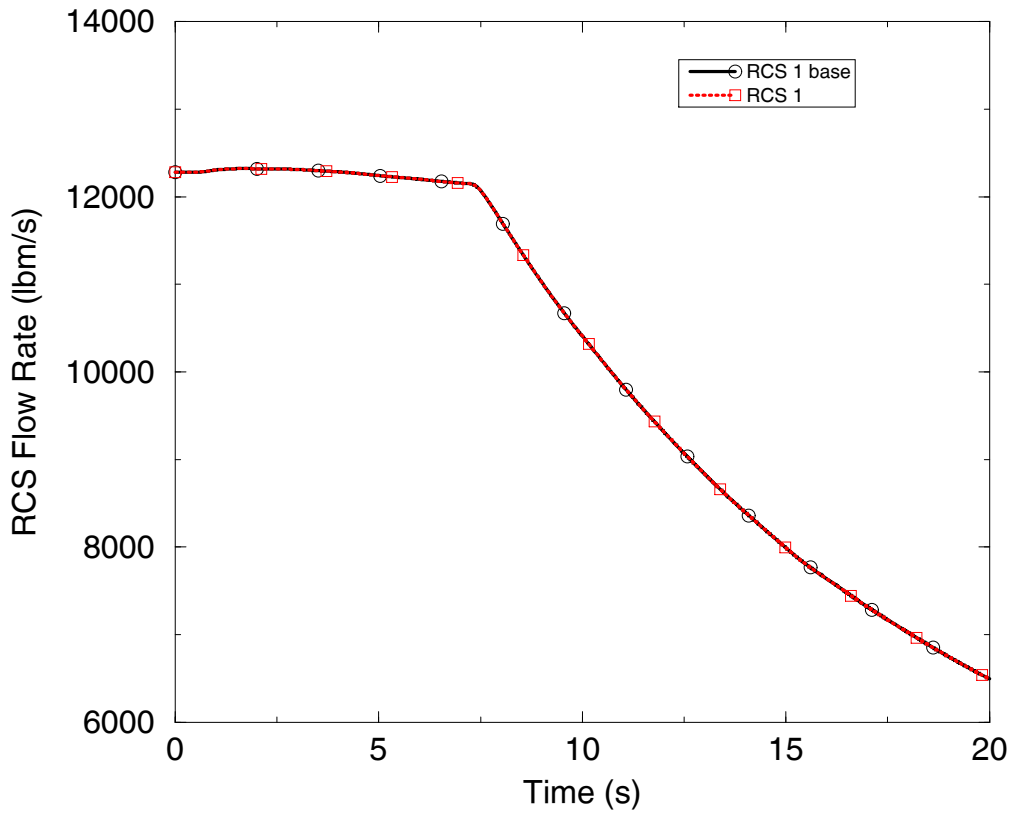

Figure 3-8 Case 1 Turbine Trip – RCS Flow Rate

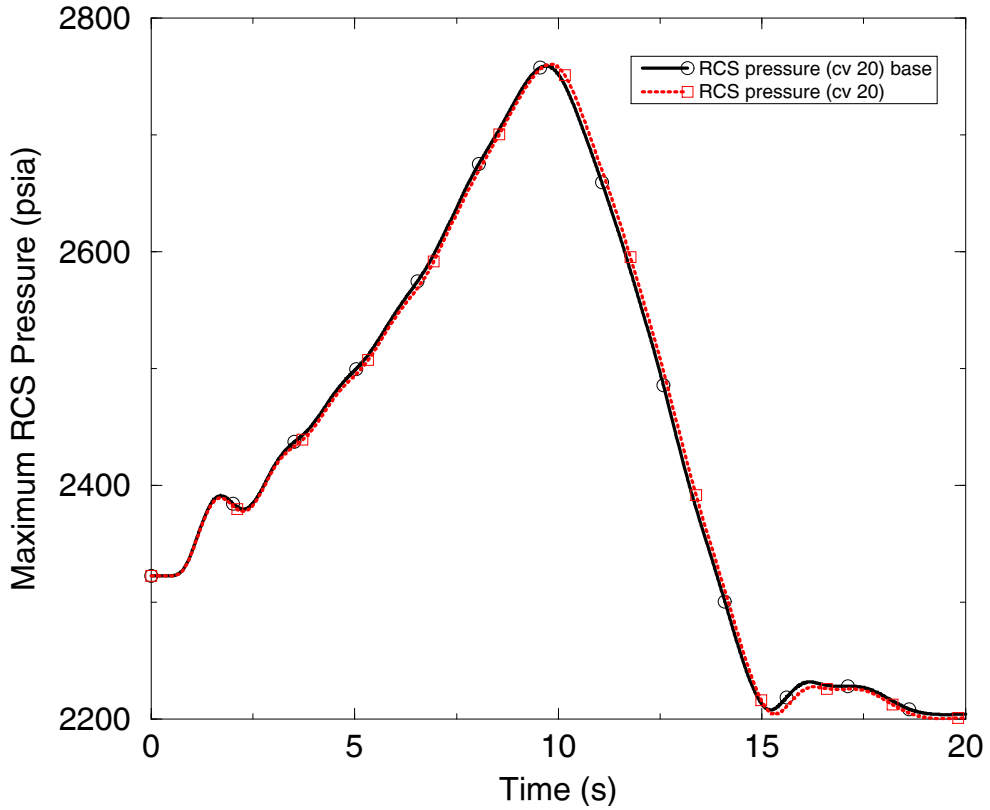

Figure 3-9 Case 2 Turbine Trip – Maximum RCS Pressure

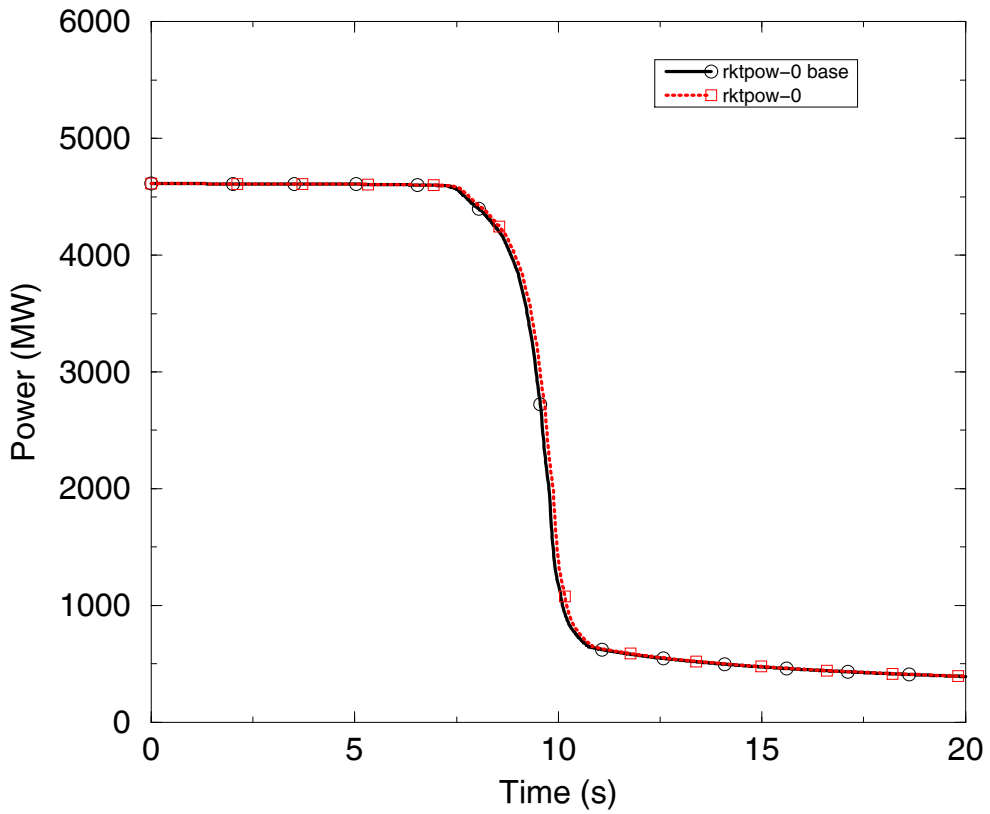

Figure 3-10 Case 2 Turbine Trip – Reactor Power

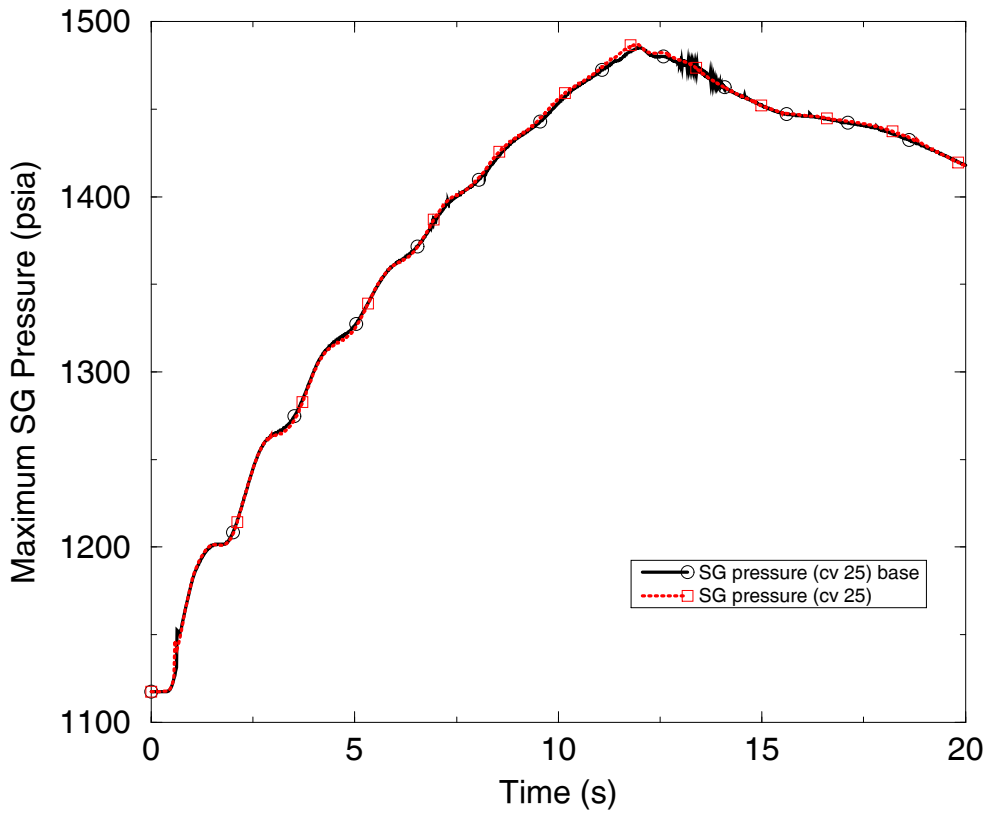

Figure 3-11 Case 2 Turbine Trip – Maximum Steam Generator Pressure

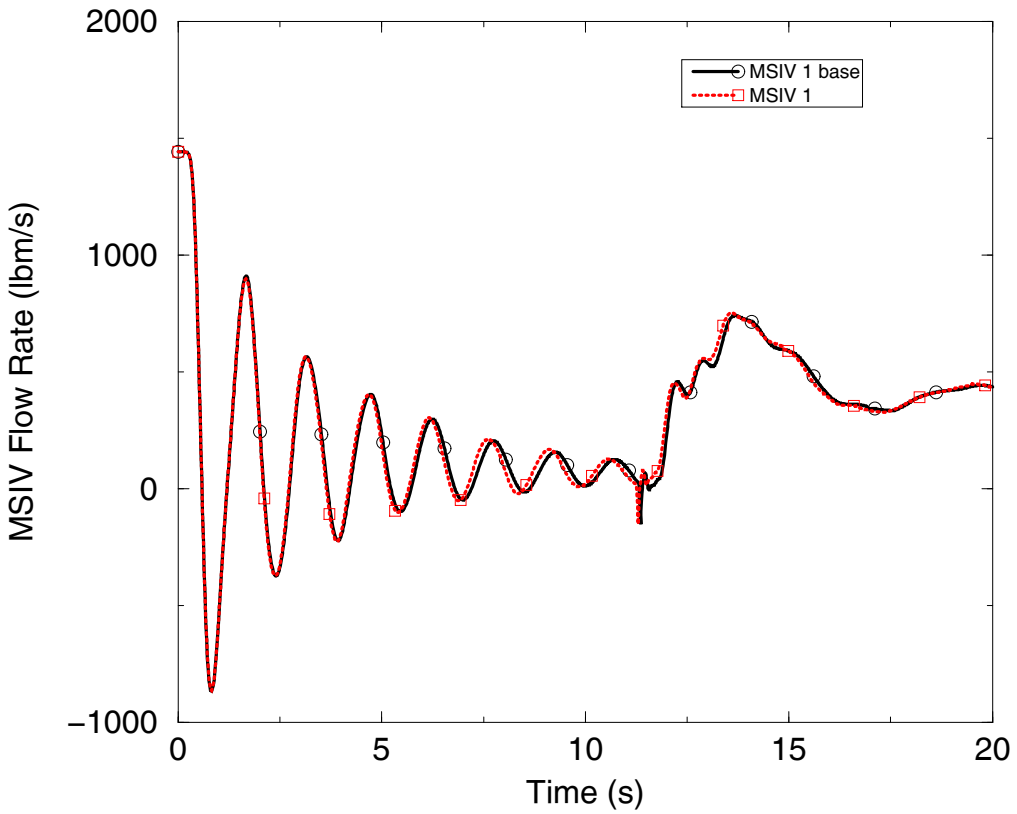

Figure 3-12 Case 2 Turbine Trip – MSIV1 Flow Rate

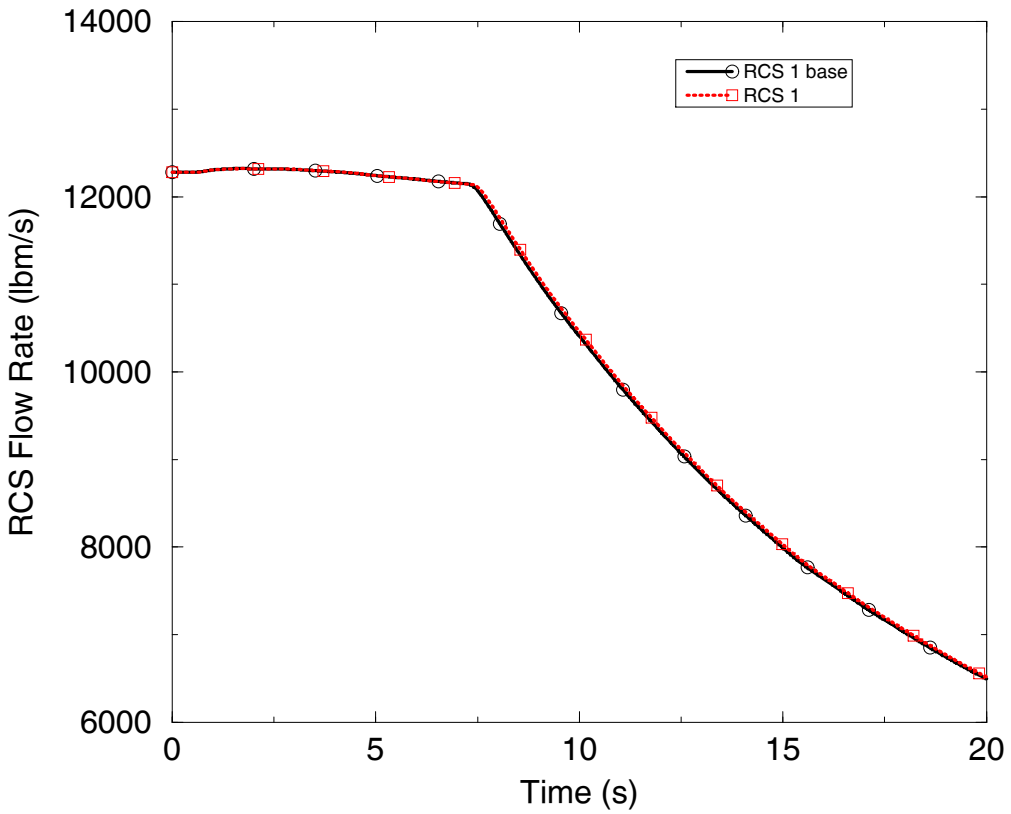

Figure 3-13 Case 2 Turbine Trip – RCS Flow Rate

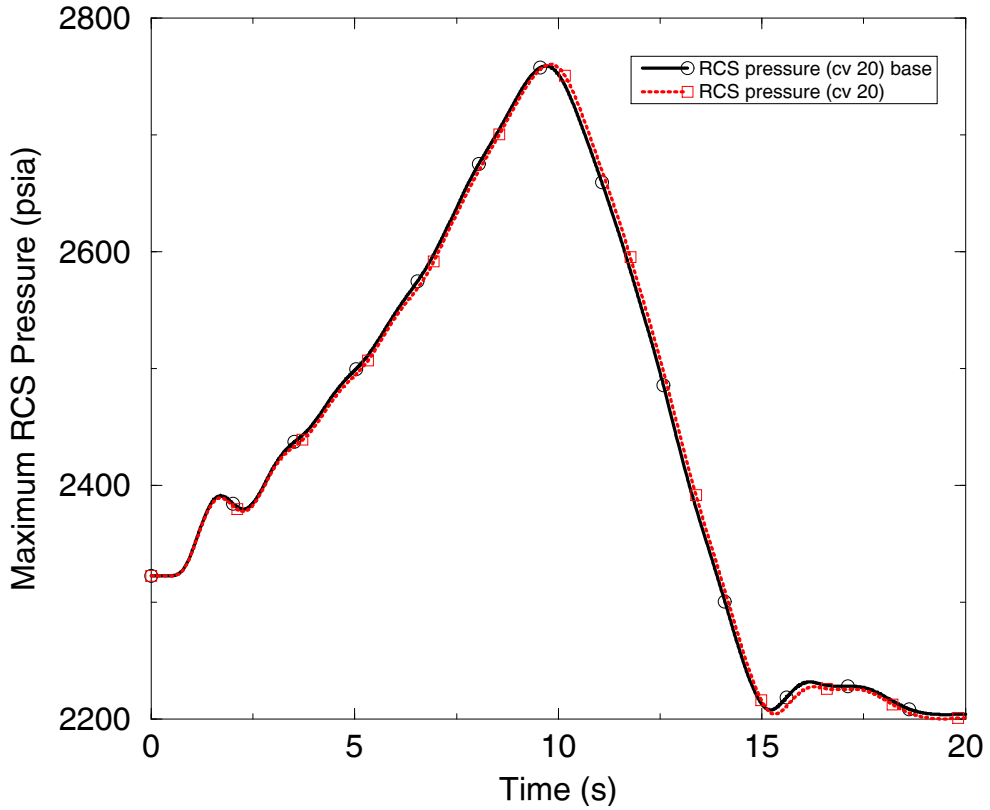

Figure 3-14 Case 3 Turbine Trip – Maximum RCS Pressure

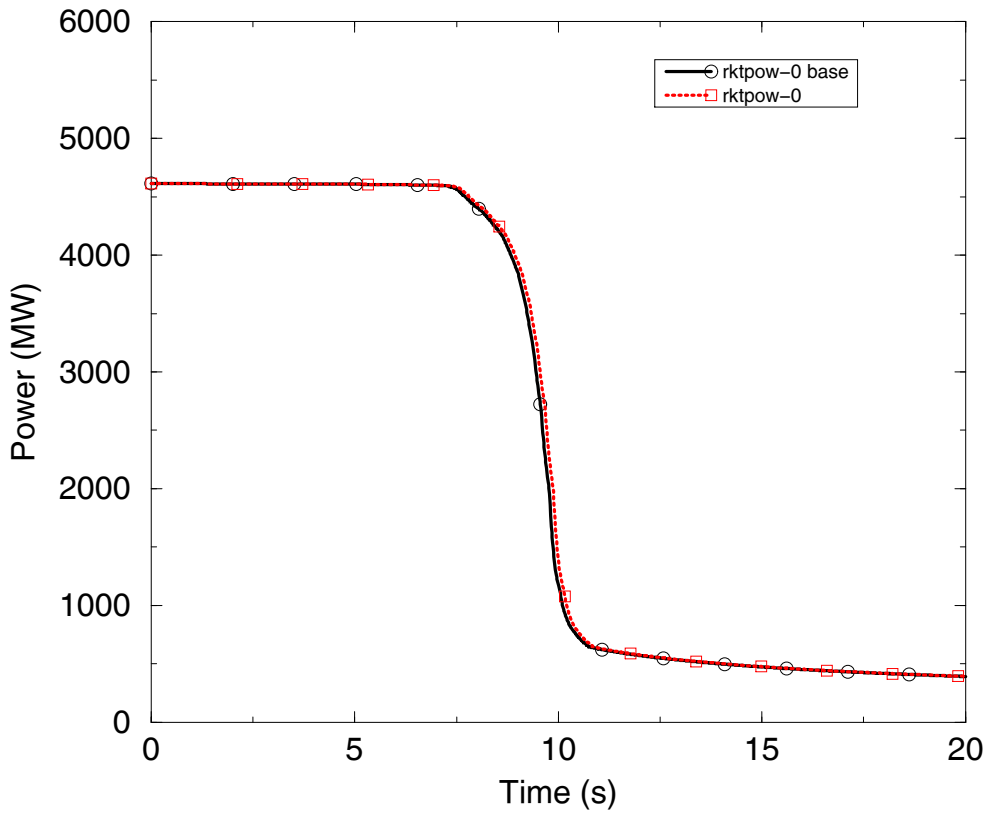

Figure 3-15 Case 3 Turbine Trip – Reactor Power

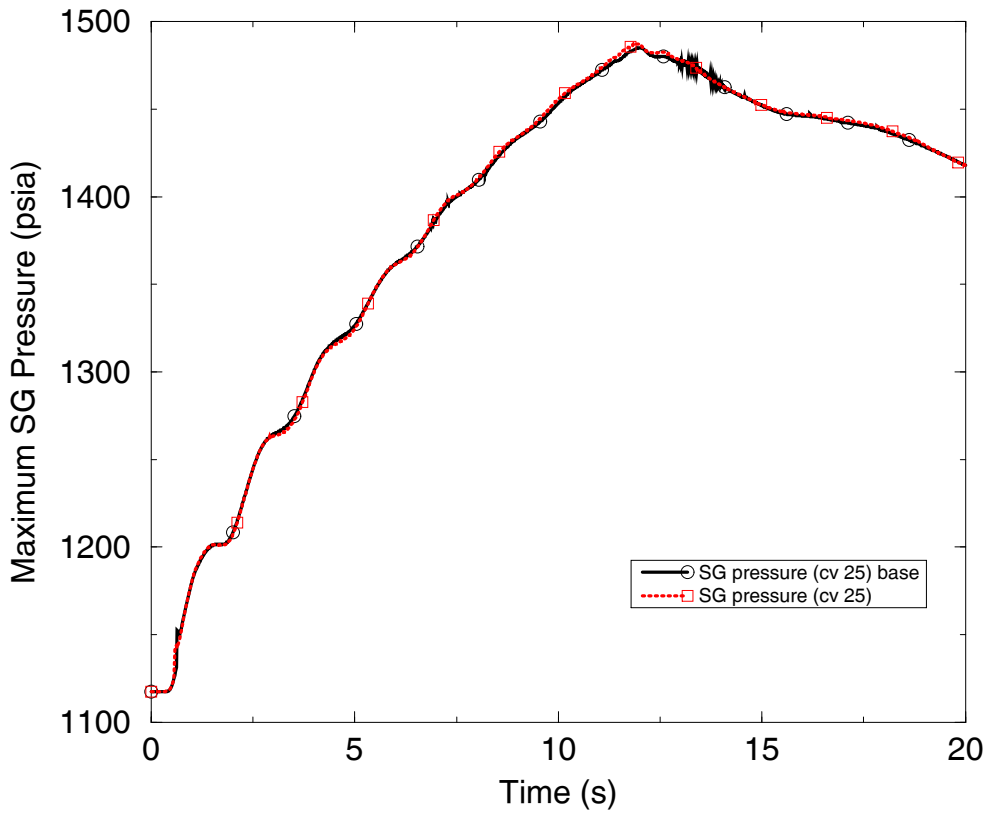

Figure 3-16 Case 3 Turbine Trip – Maximum Steam Generator Pressure

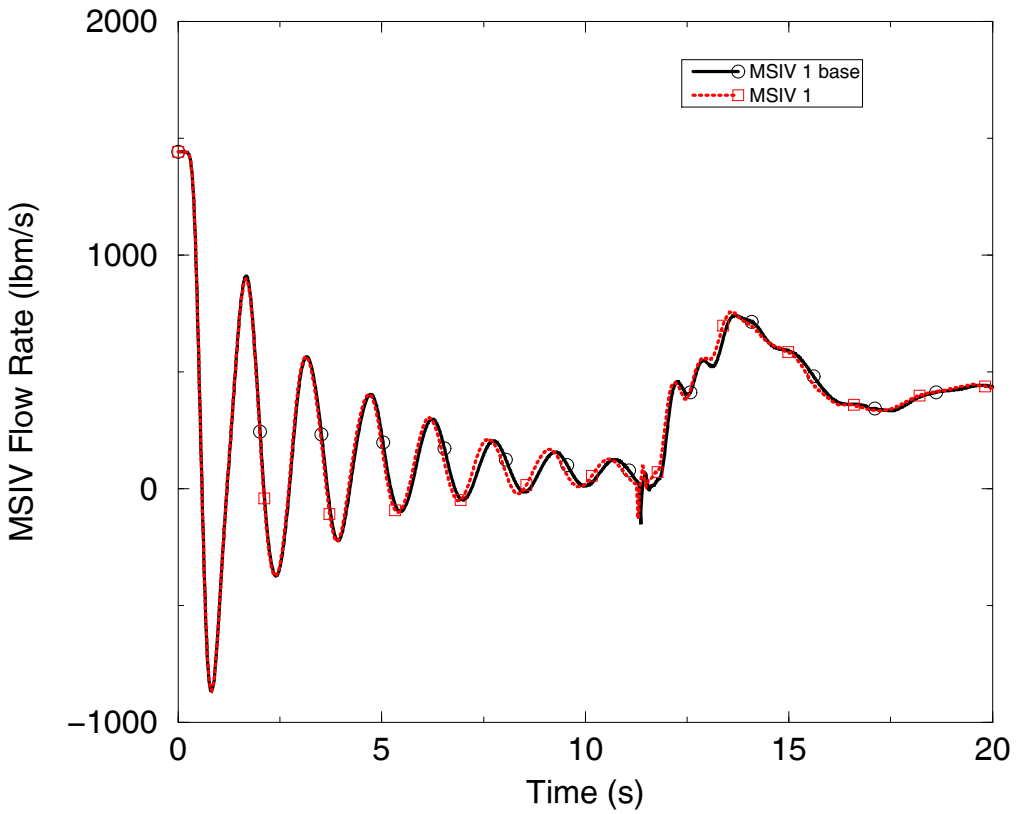

Figure 3-17 Case 3 Turbine Trip – MSIV1 Flow Rate

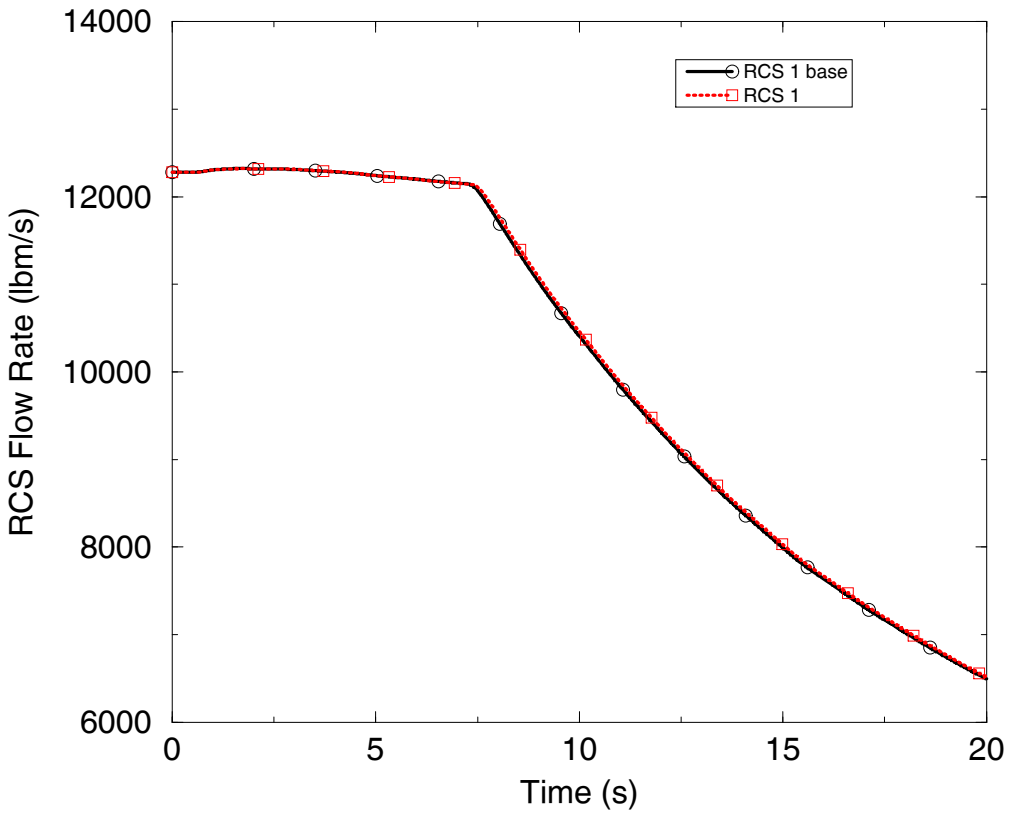

Figure 3-18 Case 3 Turbine Trip – RCS Flow Rate

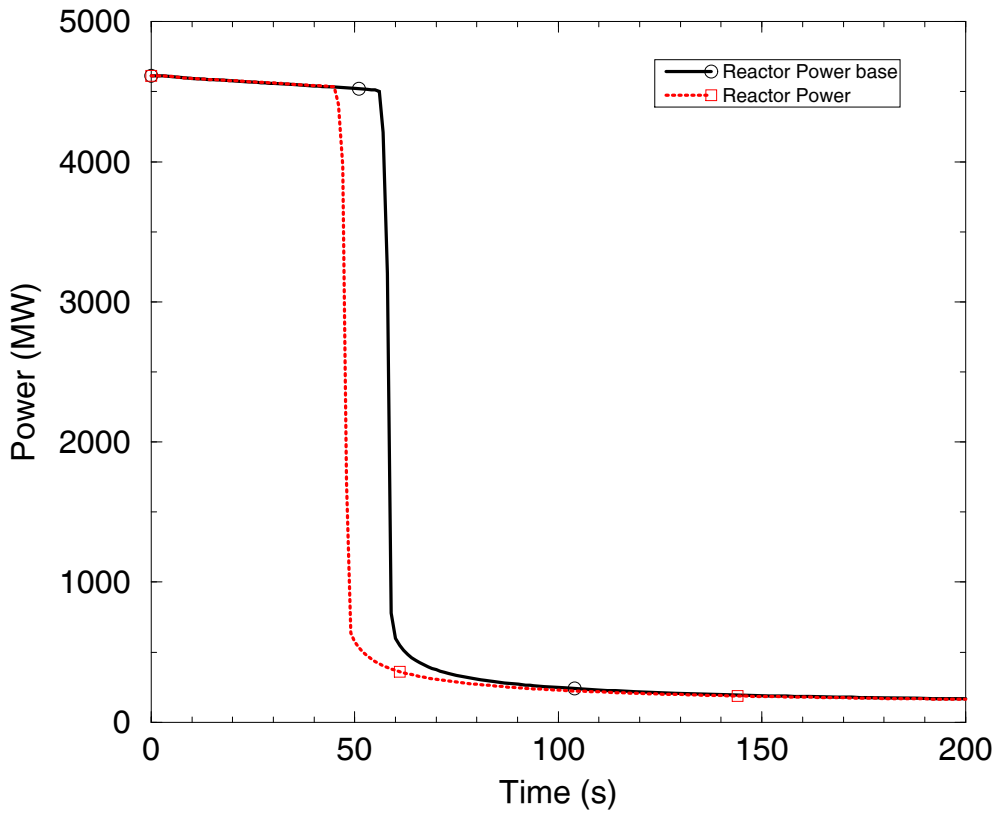

Figure 3-19 Case 3 Loss of Feed Flow – Reactor Power

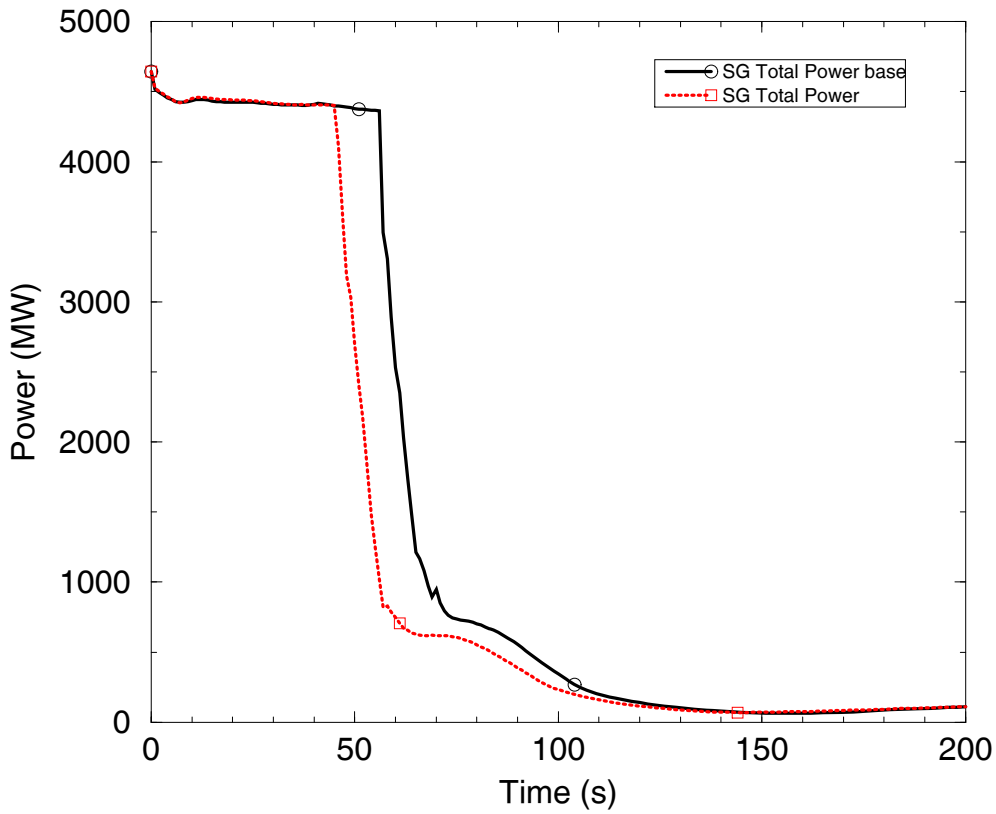

Figure 3-20 Case 3 Loss of Feed Flow – Total Steam Generator Power

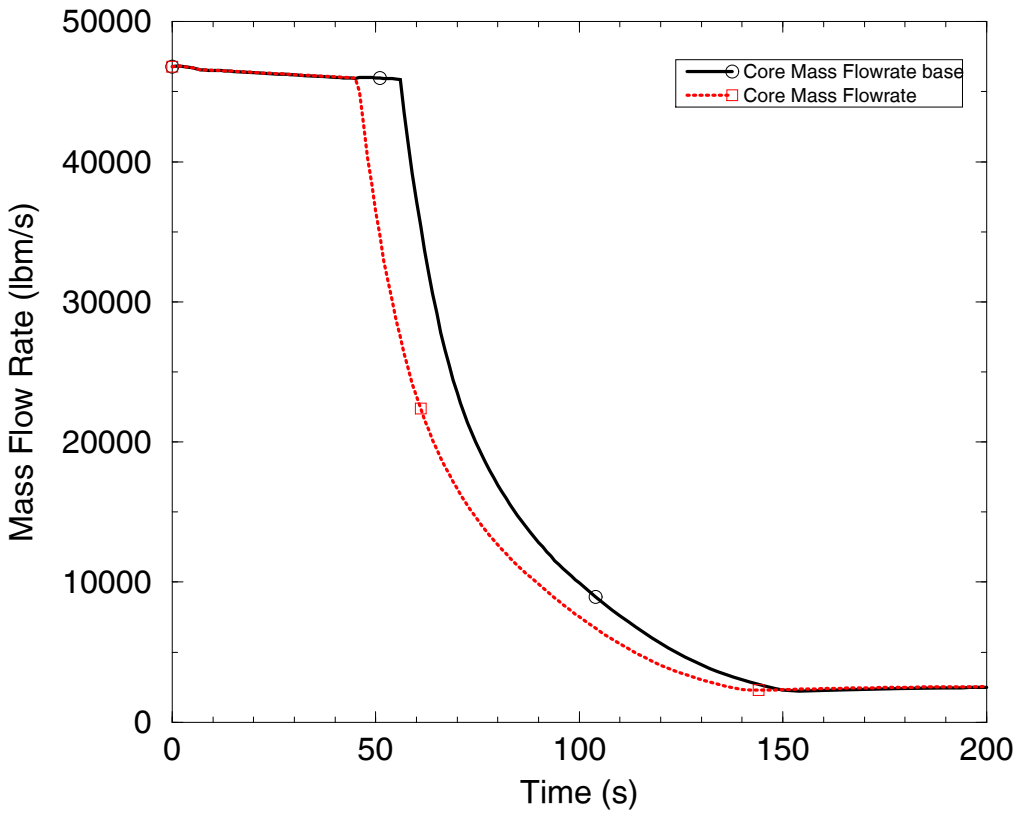

Figure 3-21 Case 3 Loss of Feed Flow – Core Mass Flow Rate

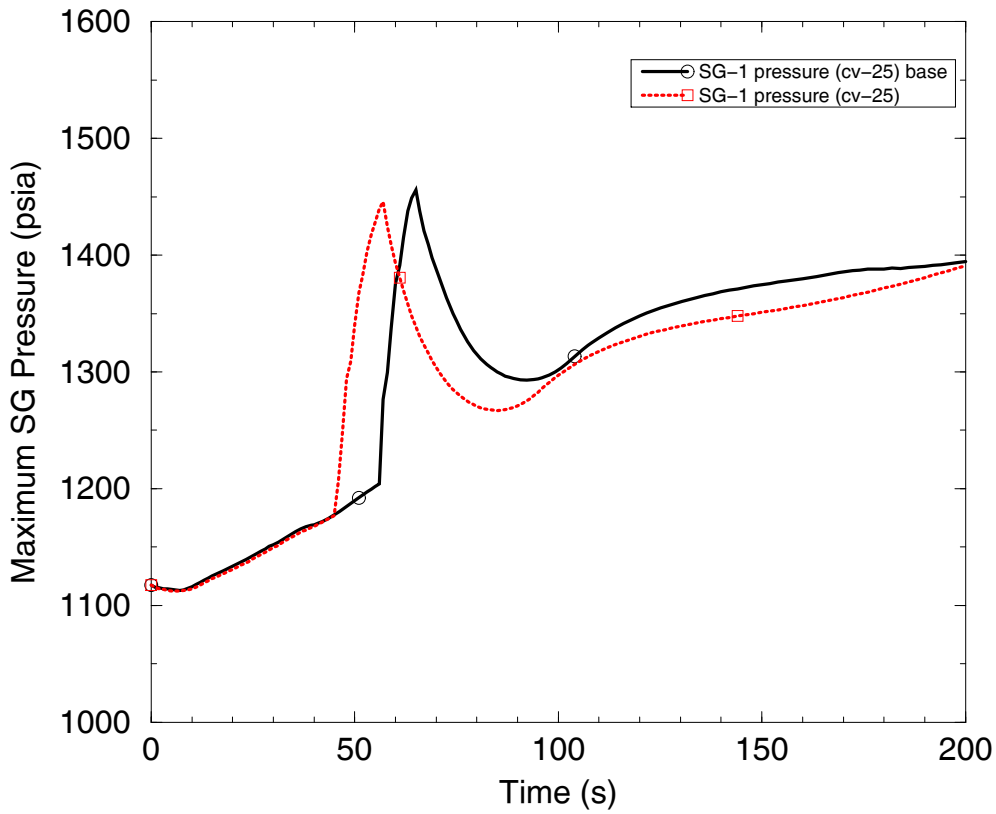

Figure 3-22 Case 3 Loss of Feed Flow – Maximum Steam Generator Pressure

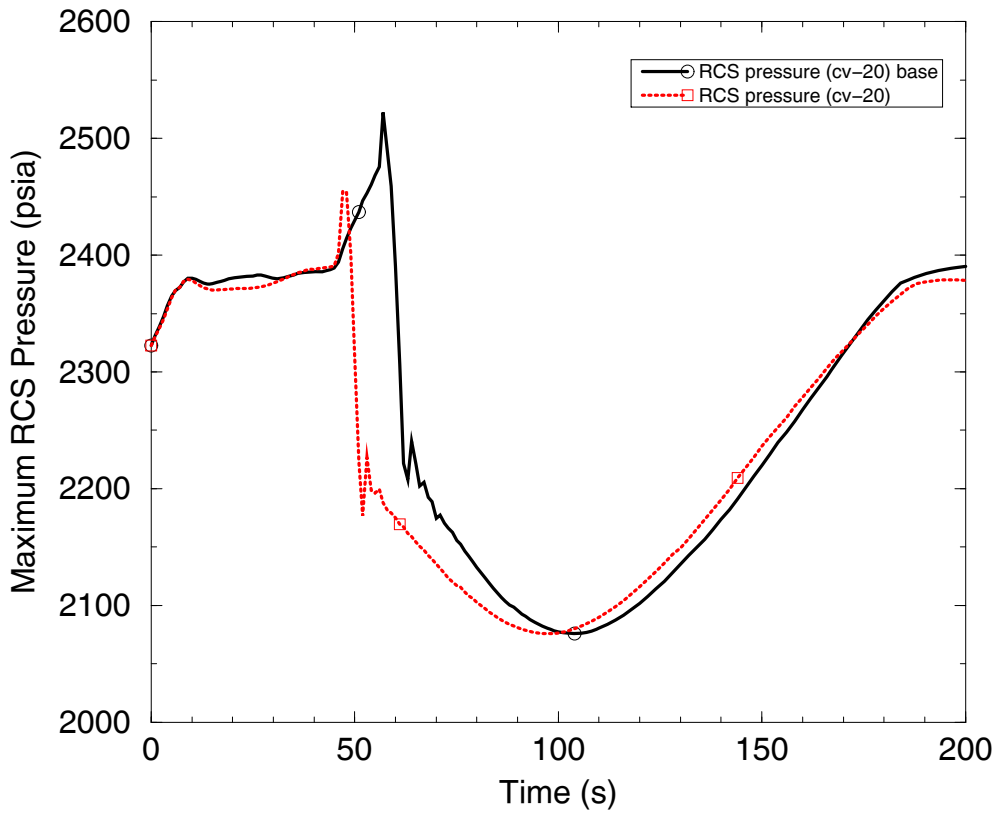

Figure 3-23 Case 3 Loss of Feed Flow – Maximum RCS Pressure

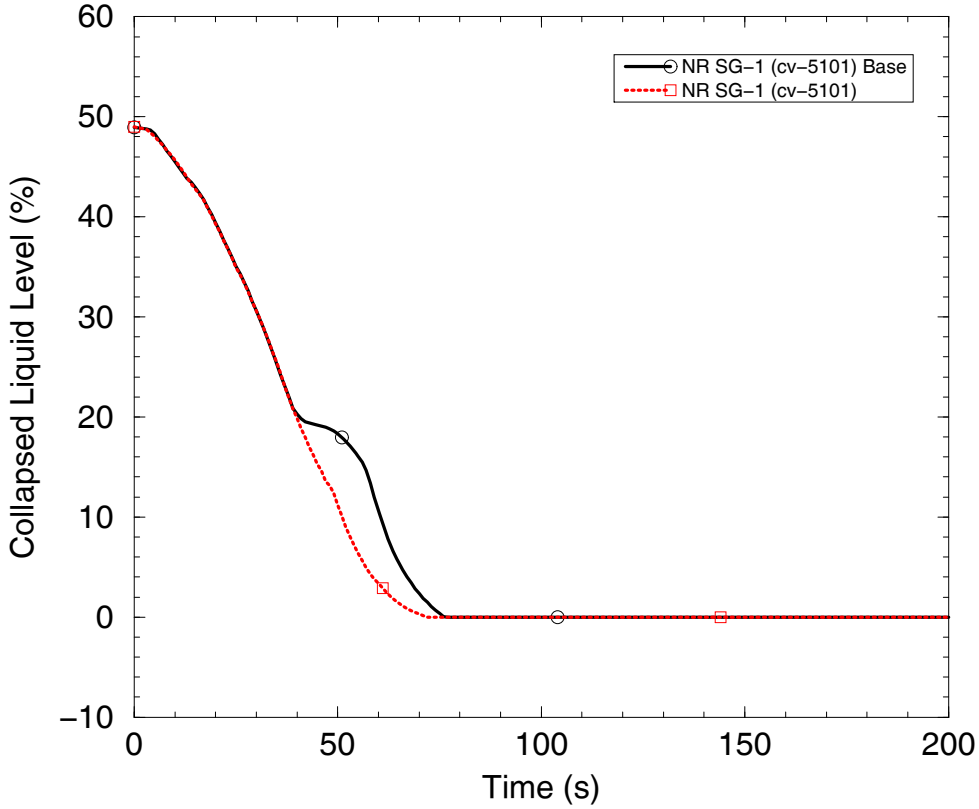

Figure 3-24 Case 3 Loss of Feed Flow – Steam Generator No. 1 NR Level

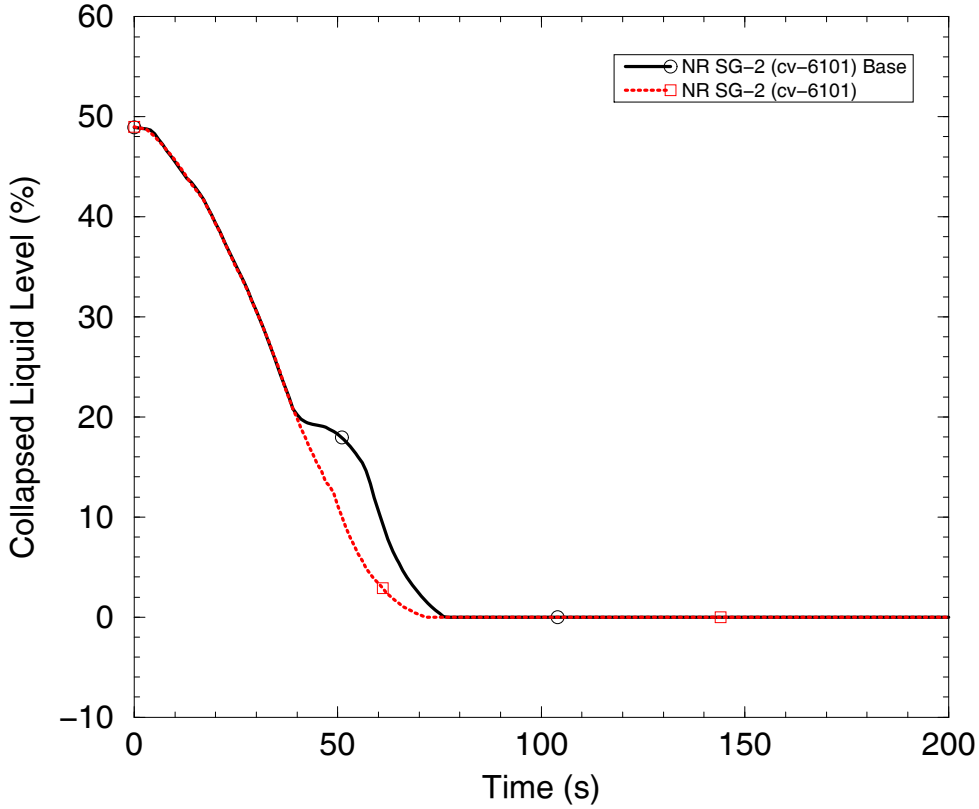

Figure 3-25 Case 3 Loss of Feed Flow – Steam Generator No. 2 NR Level

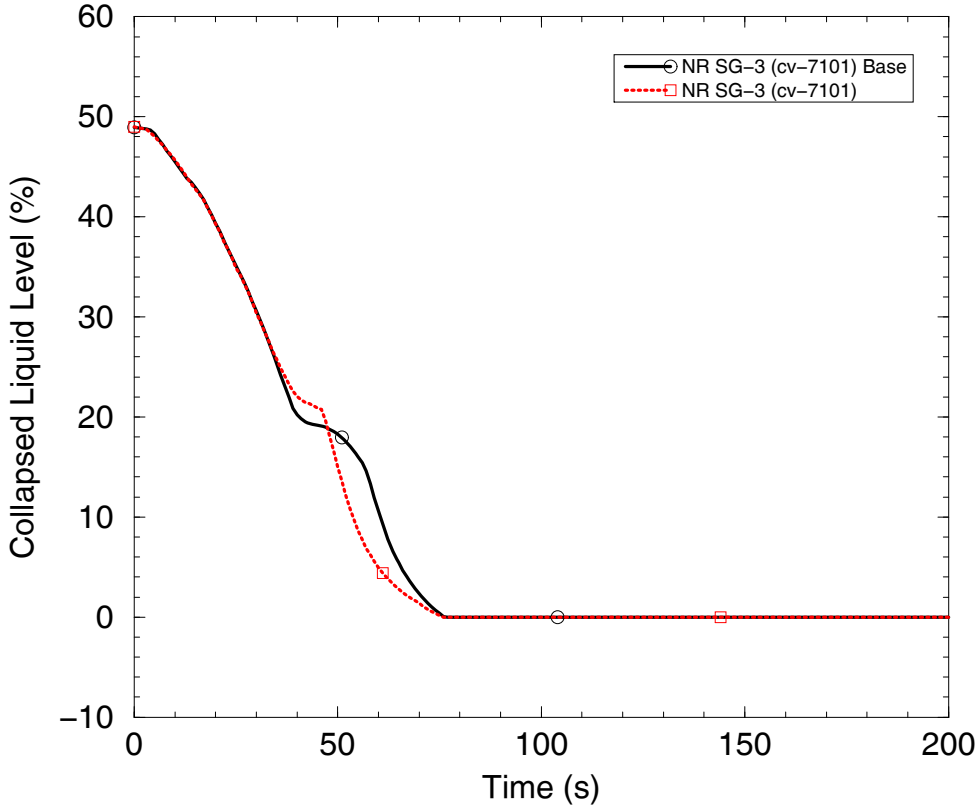

Figure 3-26 Case 3 Loss of Feed Flow – Steam Generator No. 3 NR Level

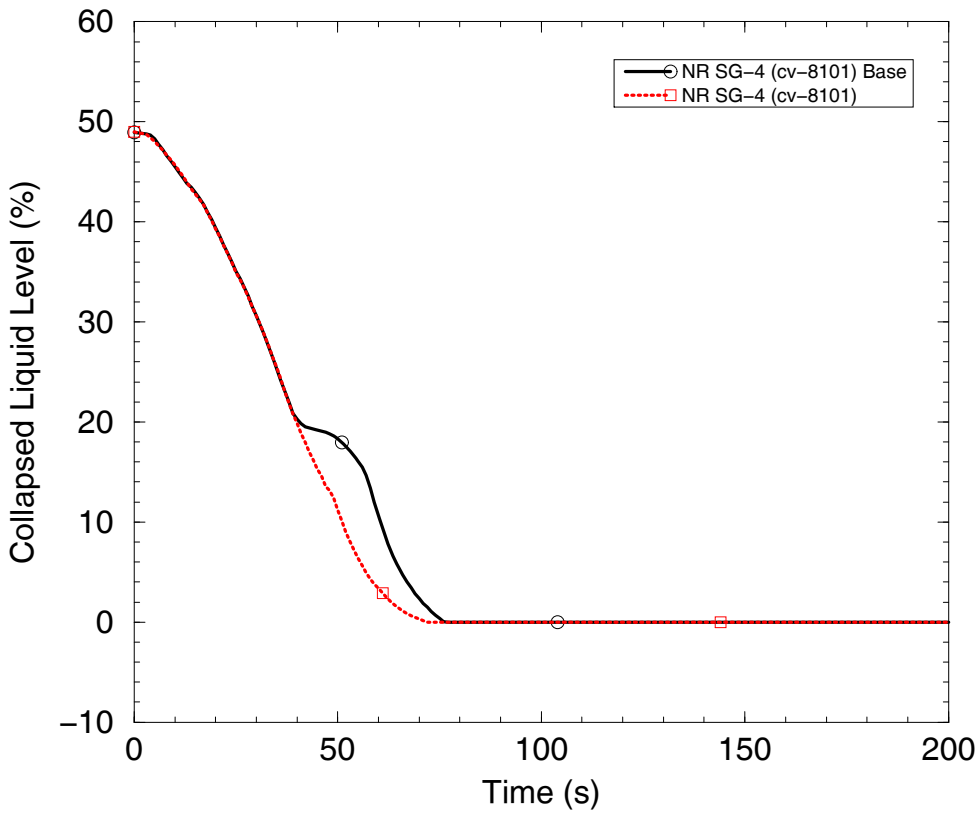

Figure 3-27 Case 3 Loss of Feed Flow – Steam Generator No. 4 NR Level

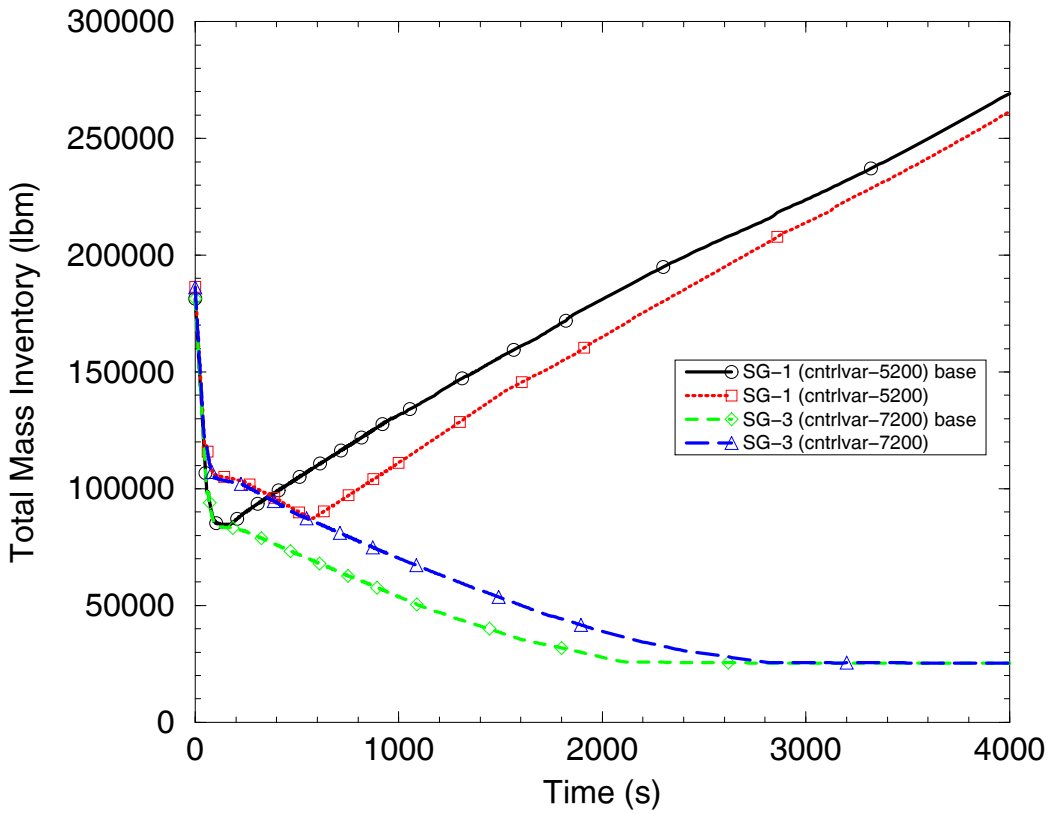

Figure 3-28 Case 3 Loss of Feed Flow – Steam Generator No. 1 & No. 3 Total Mass

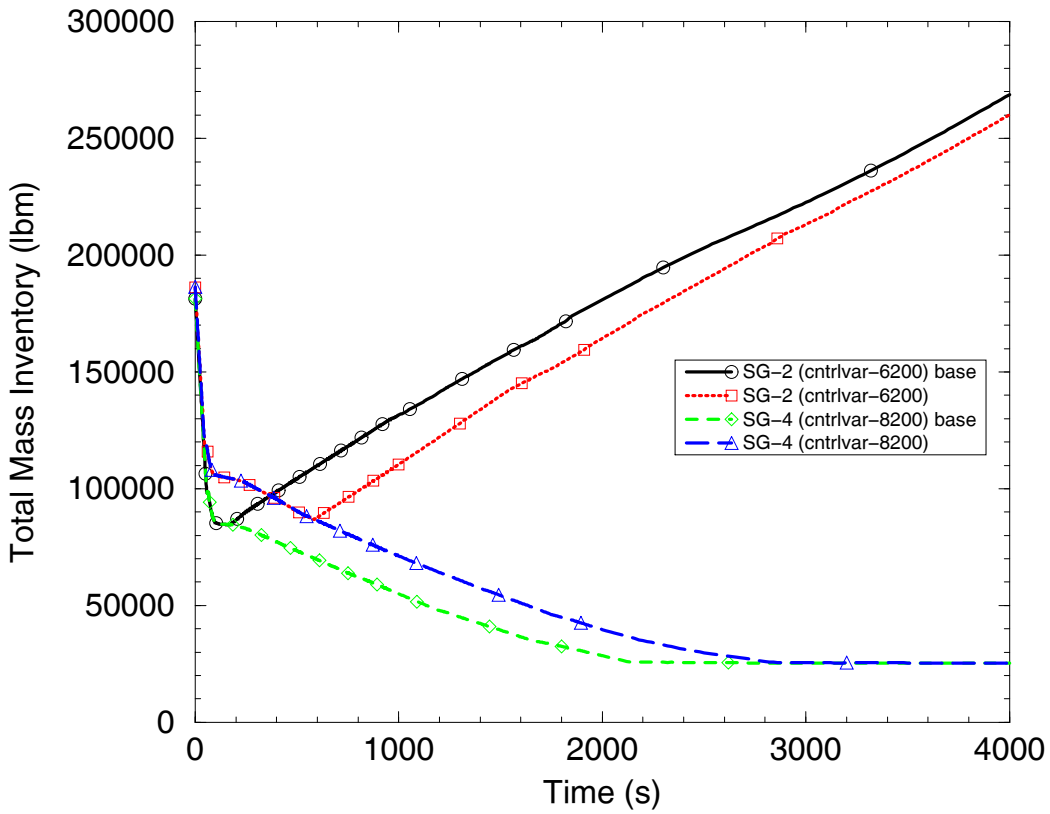

Figure 3-29 Case 3 Loss of Feed Flow – Steam Generator No. 2 & No. 4 Total Mass

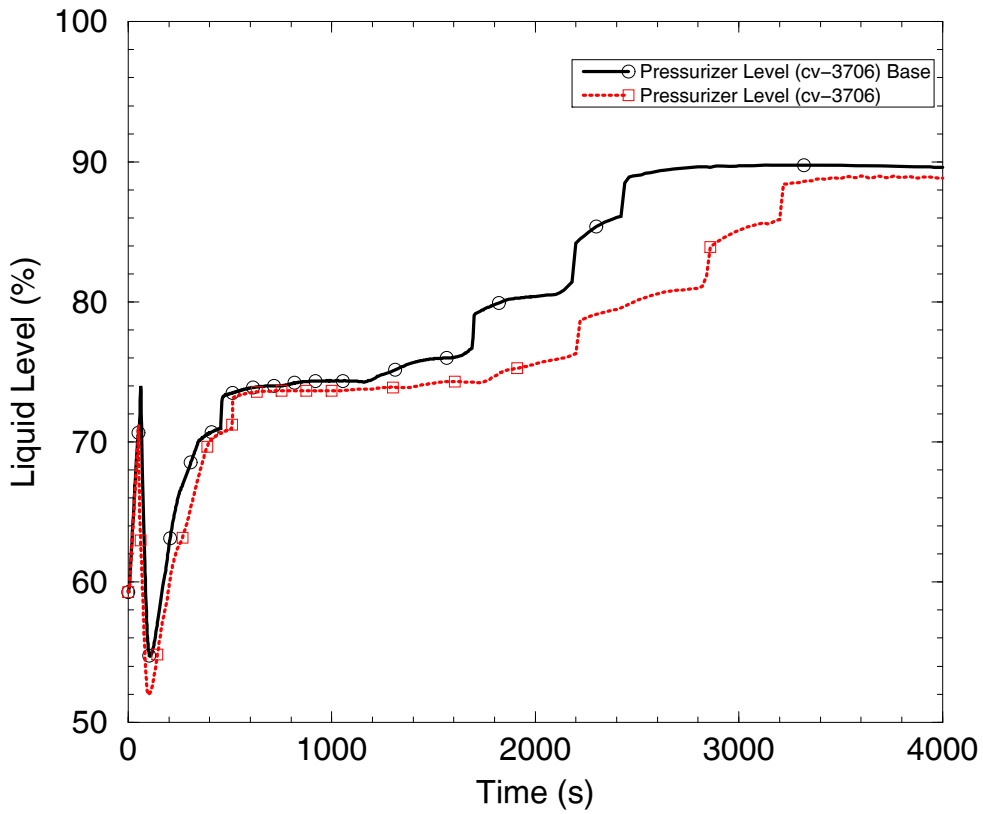

Figure 3-30 Case 3 Loss of Feed Flow – Pressurizer Level

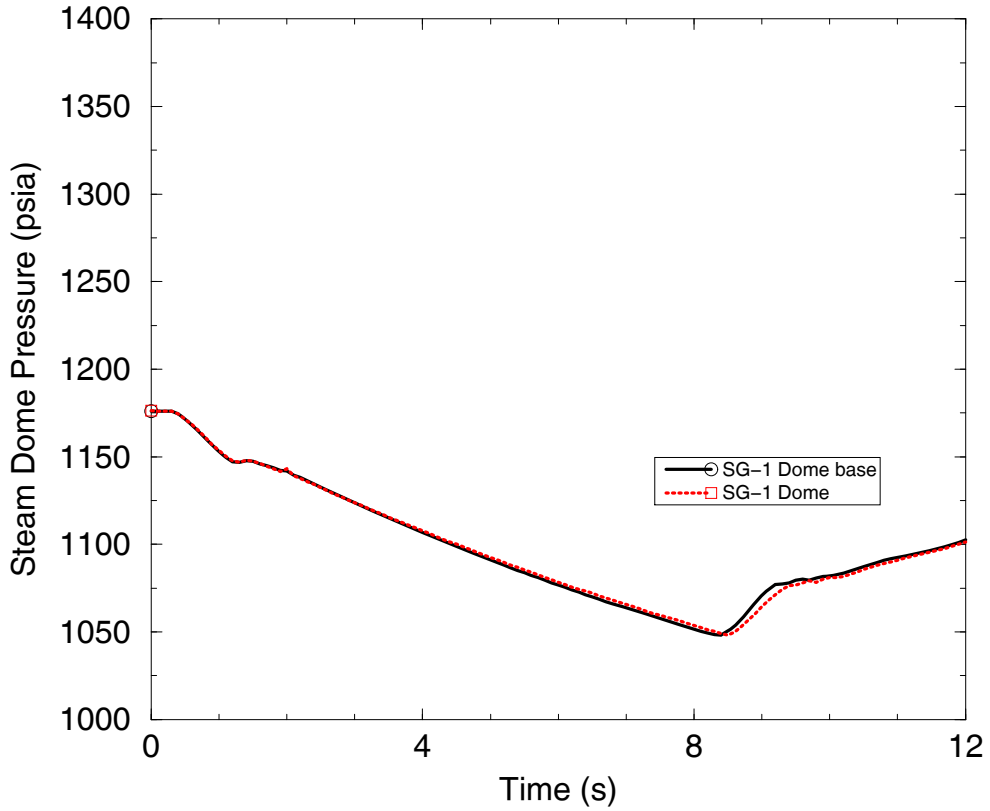

Figure 3-31 Case 3 Increase in Steam Flow – Steam Generator No. 1 Dome Pressure

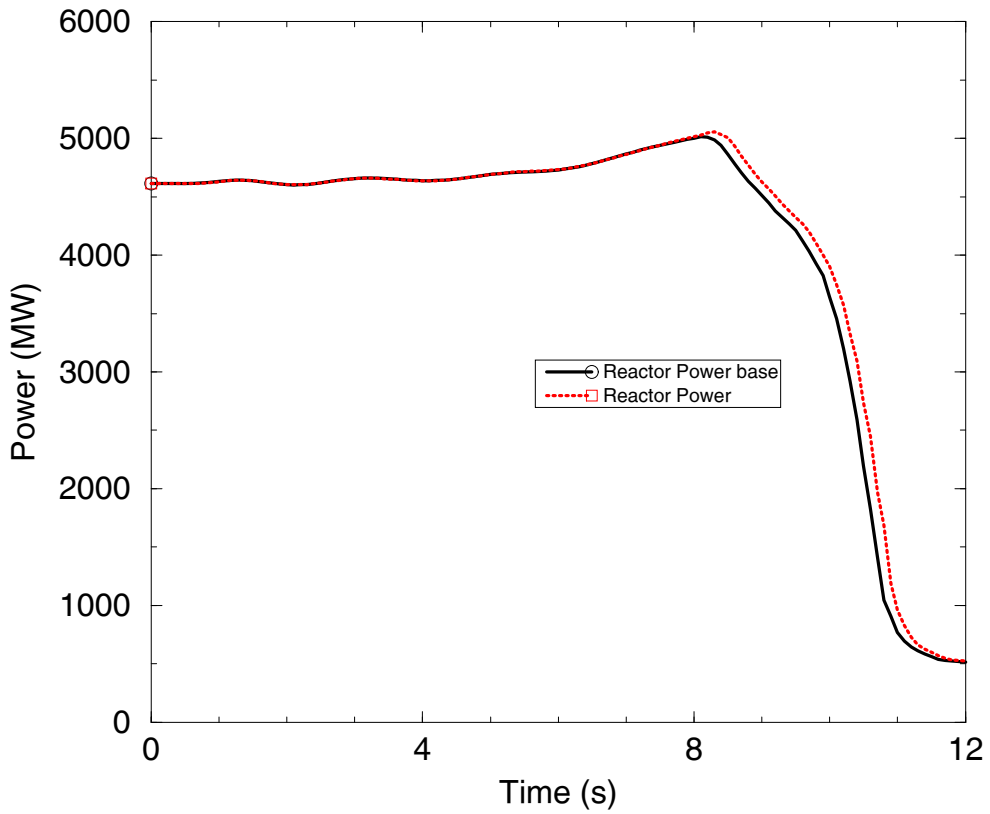

Figure 3-32 Case 3 Increase in Steam Flow – Reactor Power

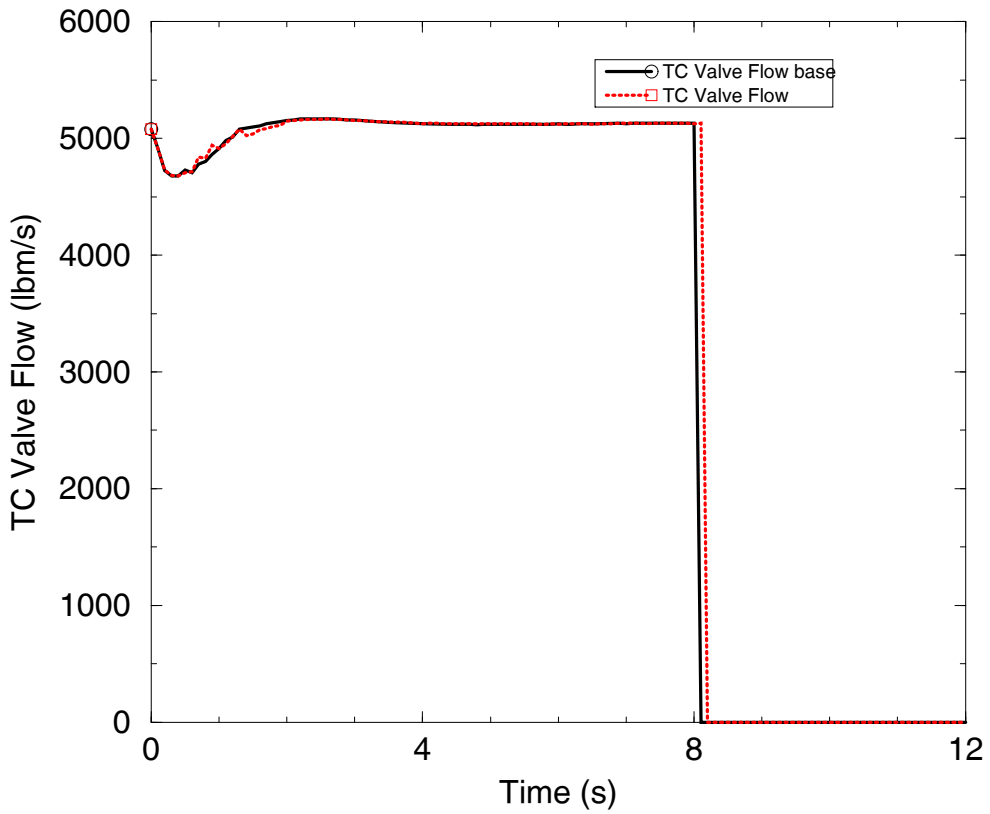

Figure 3-33 Case 3 Increase in Steam Flow – TCV Flow Rate

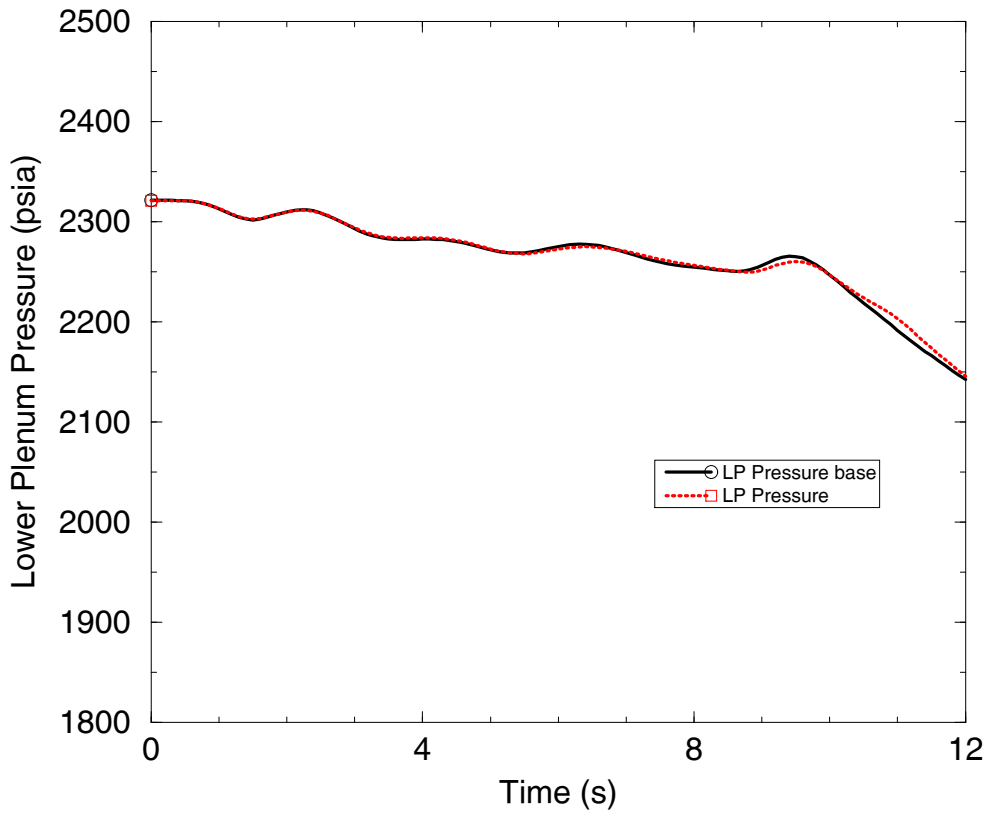

Figure 3-34 Case 3 Increase in Steam Flow – Reactor Vessel Lower Plenum Pressure

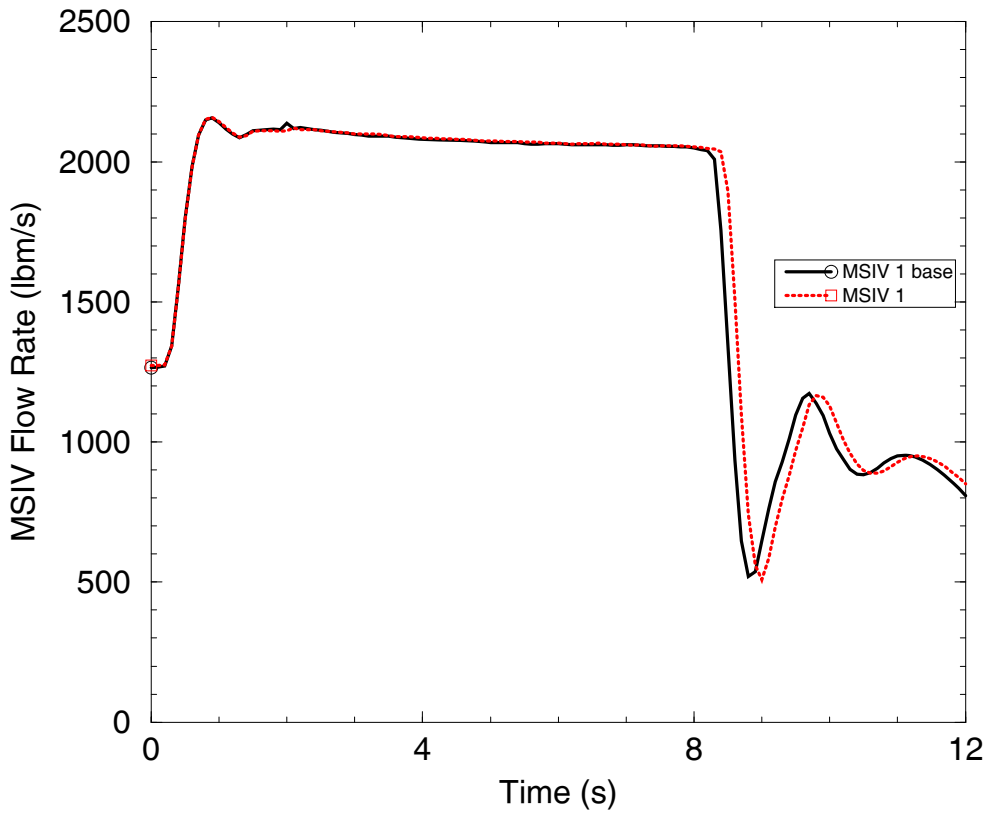

Figure 3-35 Case 3 Increase in Steam Flow – MSIV1 Flow Rate

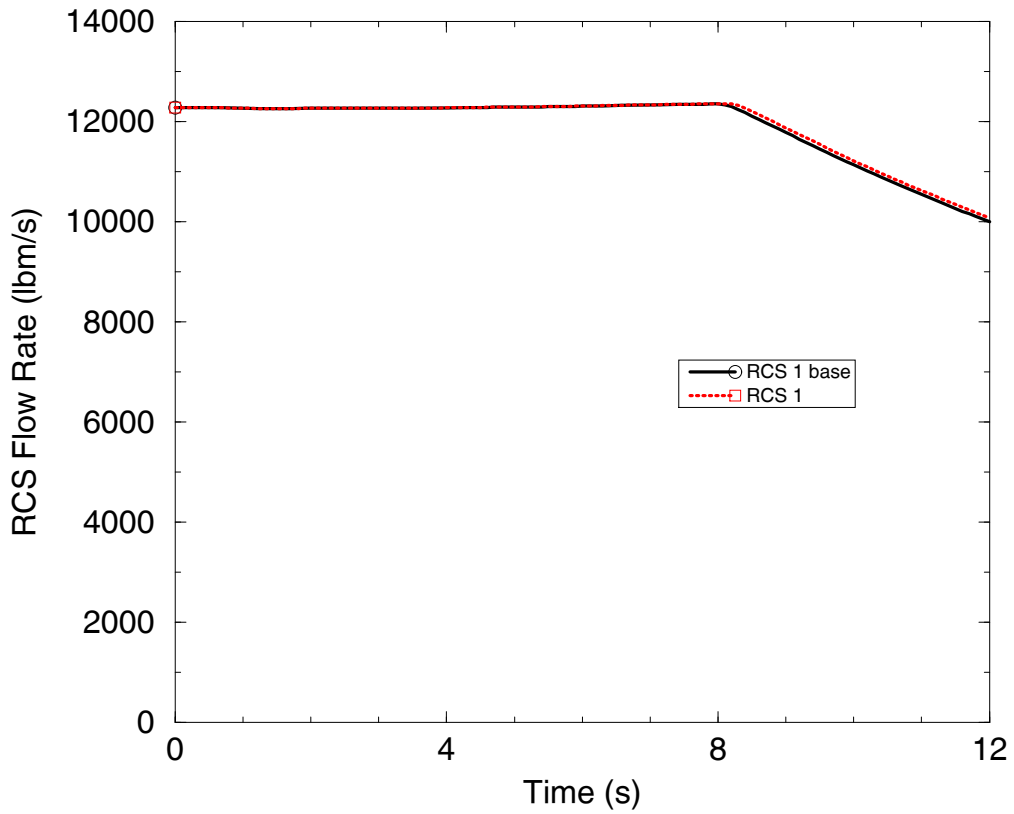

Figure 3-36 Case 3 Increase in Steam Flow – RCS Flow Rate

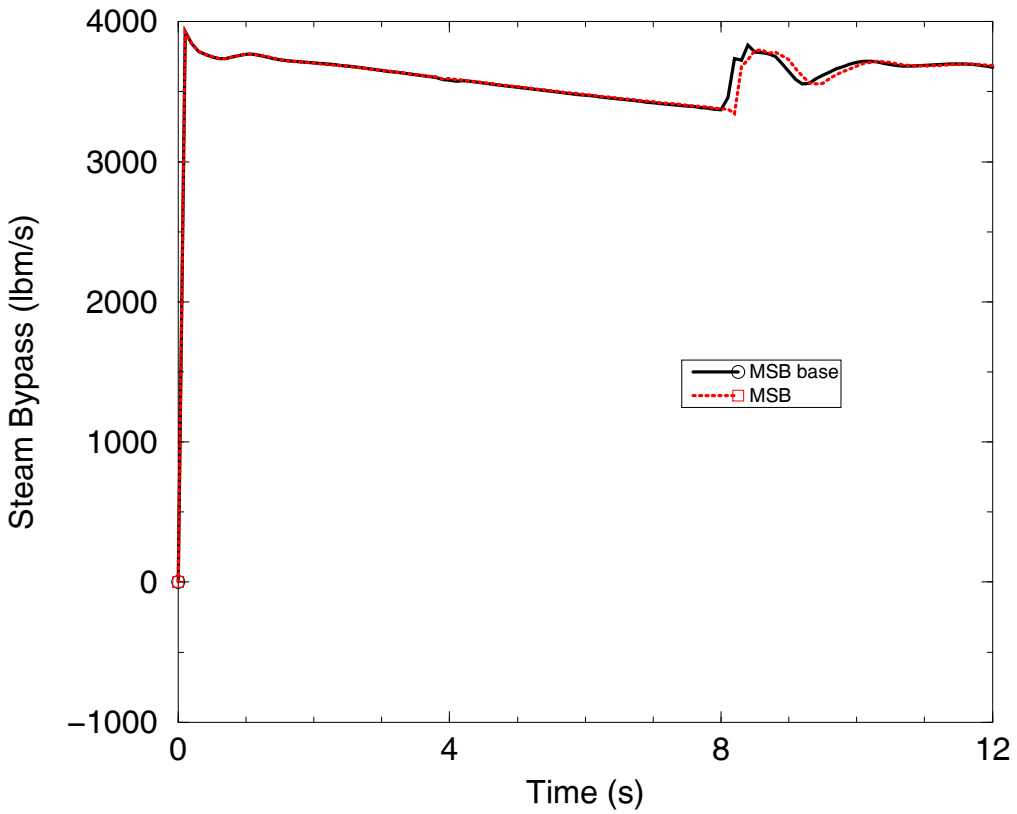

Figure 3-37 Case 3 Increase in Steam Flow – Steam Bypass Flow Rate

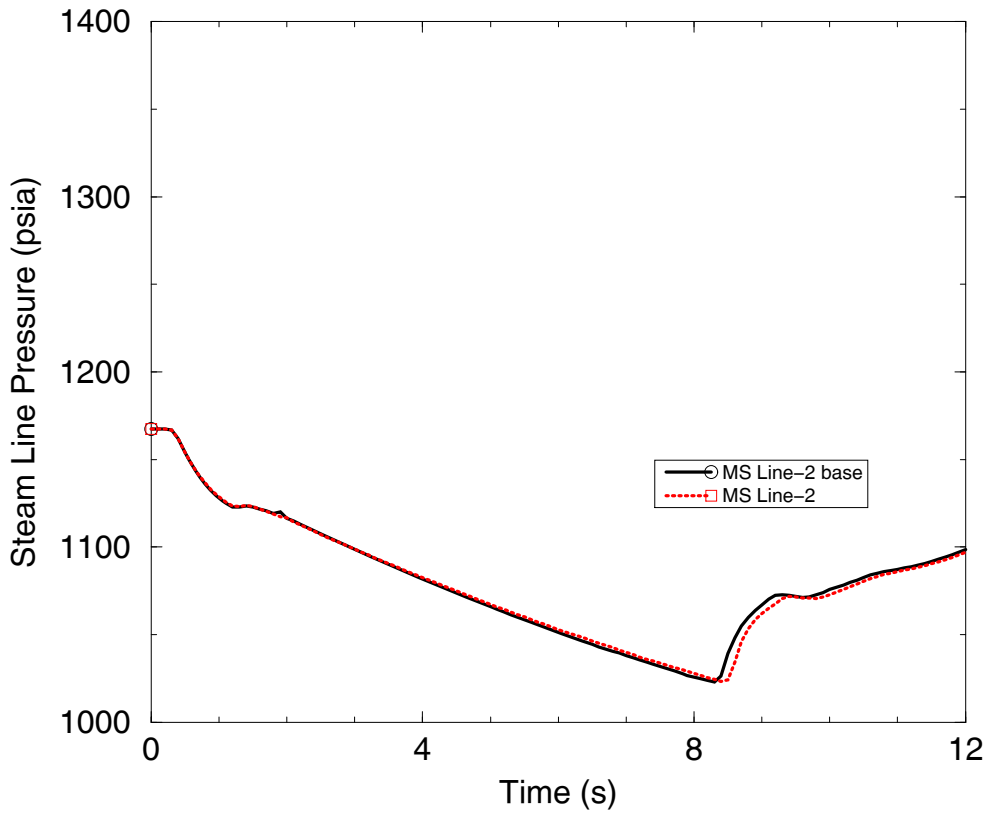

Figure 3-38 Case 3 Increase in Steam Flow – Steam Line No. 2 Pressure

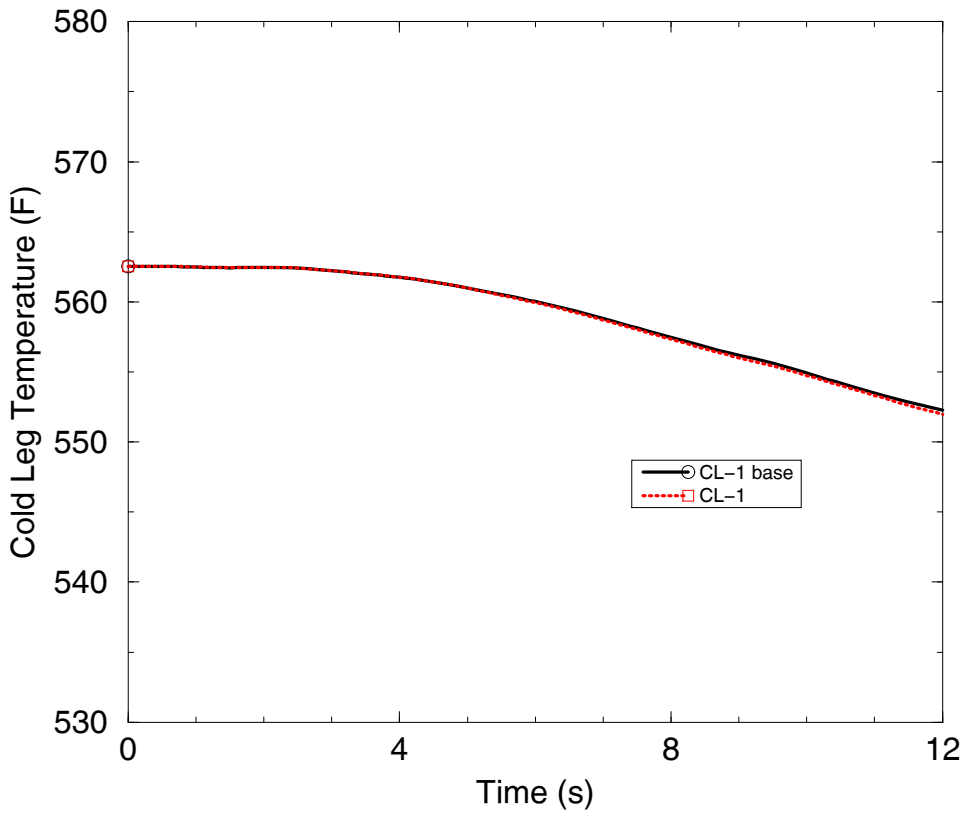

Figure 3-39 Case 3 Increase in Steam Flow – RCS Cold Leg No. 1 Temperature

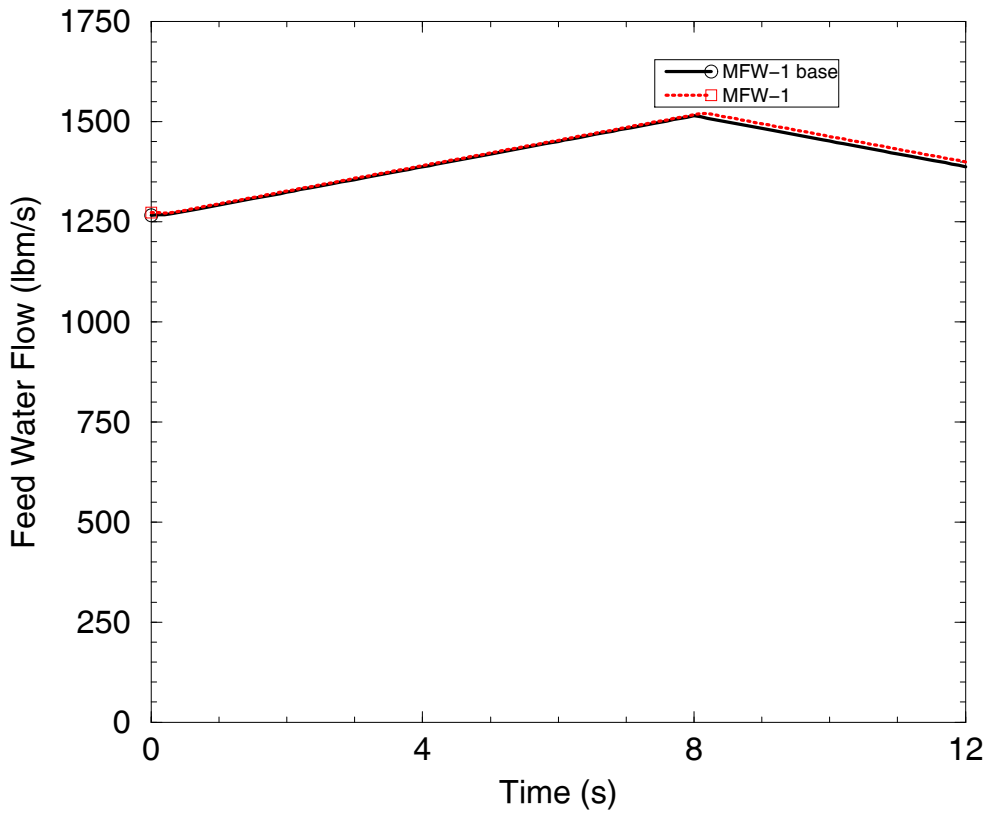

Figure 3-40 Case 3 Increase in Steam Flow – Steam Generator No. 1 Main Feedwater Flow Rate

4.0 REFERENCES

1. "Codes and Methods Applicability Report for the U.S. EPR," Revision 9, ANP-10263PA, August 2007.

2. "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," EMF-2328(P)(A) Revision 0, March 2001.

Appendix A

SMALL BREAK LOCA SAMPLE CALCULATION

TABLE OF CONTENTS

Page

A.1. Introduction A-4
A.2. S-RELAP5 Model A-4
A.3. Initial Conditions and Other Input Parameters A-6
A.4. Break Spectrum A-7
A.5. 6.5-Inch Break (Limiting Case) A-8
A.6. Results/Summary/Conclusion A-10
A.7. References A-12

LIST OF TABLES

Table A-1 Input Parameters A-13
Table A-2 Safety Classified I&C Signals Considered for SBLOCA Analysis A-14
Table A-3 Break Spectrum- Peak Cladding Temperature A-15
Table A-4 Limiting Break Sequence of Events A-16

LIST OF FIGURES

Figure A-1 U.S. EPR Vessel Nodalization A-17
Figure A-2 U.S. EPR Steam Generator Nodalization A-18
Figure A-3 U.S. EPR Loop Nodalization A-19
Figure A-4 Nodalization for S-RELAP5 ECC Model A-20
Figure A-5 Reactor Power for 6.5 Inch Break A-21
Figure A-6 Primary and Secondary System Pressure for 6.5 Inch Break A-22
Figure A-7 MSRT Flow for 6.5 inch Break A-23
Figure A-8 Break Flow Rate for 6.5 Inch Break A-24
Figure A-9 Loop Seal Void Fraction for 6.5 Inch Break A-25
Figure A-10 ECCS Flow for 6.5 Inch Break A-26
Figure A-11 MFW Flow Rate for 6.5 Inch Break A-27
Figure A-12 Steam Generator Mass Inventory for 6.5 Inch Break A-28
Figure A-13 Steam Generator Level for 6.5 Inch Break A-29
Figure A-14 Steam Generator Steam Line Pressure for 6.5 Inch Break A-30

Figure A-15 Steam Generator Steam Line Pressure and Cooldown Curve for 6.5 Inch Break....A-31

Figure A-16 Total RCS and RV Mass Inventories for the 6.5 Inch Break....A-32

Figure A-17 Hot Assembly Collapsed Liquid Level for 6.5 Inch Break....A-33

Figure A-18 MHSI Flow (Loops 1 and 4) for 6.5 Inch Break....A-33

Figure A-18 MHSI Flow (Loops 1 and 4) for 6.5 Inch Break....A-34

Figure A-19 LHSI Flow (Loops 1 and 4) for 6.5 Inch Break....A-35

Figure A-20 Vapor and Clad Temperature for the Hot Node for a 6.5 Inch Break....A-36

A.1. INTRODUCTION

This appendix presents the application of the S-RELAP5 SBLOCA methodology (Reference A-1) to the U.S. EPR. The U.S. EPR has four loops with four hot-legs, four cold-legs, and four vertical U-tube steam generators. The reactor vessel contains a downcomer, upper and lower plena, and a reactor core with 241 fuel assemblies. The hot-legs connect the reactor vessel with the vertical U-tube steam generators. Feedwater is injected into the downcomer of each steam generator. The Main Steam Relief Trains (MSRTs) are used to depressurize the SG secondary side at a rate corresponding to 180 ^{o}F/hr following receipt of a Safety Injection (SI) signal and, thereby, help cool and depressurize the primary system in a LOCA event. There are four EFW pumps; all are motor driven. The SIS contains four MHSI pumps, four accumulators, and four LHSI pumps.

A.2. S-RELAP5 MODEL

The reactor coolant system of the plant is modeled in S-RELAP5 as a network of control volumes interconnected by flow paths. The model includes four accumulators, a pressurizer, and four steam generators in which both the primary and secondary sides are modeled. Loops are modeled explicitly to provide an accurate representation of the plant. Figure A-1 and Figure A-2 are nodalization diagrams for the reactor vessel and secondary system. Figure A-3 presents the nodalization of the loops. Figure A-1 was modified from the original submittal by eliminating the downcomer to hot legs bypass. This bypass was eliminated because it was demonstrated that the bypass could close during a SBLOCA event. Figure A-2 was modified by modeling the SG top of the boiler section as two volumes and lowering the location of the EFW injection to the actual injection location. This representation is considered more appropriate to the actual plant geometry. Figure A-3 was modified from the original submittal by adding another node in the hot legs. The modification involved splitting the last node in the hot leg (at the inlet to the SG plenum) into two nodes, one representing the hot leg bend, and the other

representing the SG nozzle. This representation is considered more appropriate to the actual plant geometry. The ECCS injection scheme is presented in Figure A-4.

Decay heat is determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K to 10 CFR Part 50.

The calculations assume a loss of off-site power concurrent with reactor scram. The single failure criterion required by Appendix K is satisfied by assuming the failure of one diesel generator. This causes the disabling of one MHSI pump, one LHSI pump and one motor-driven EFW pump. In addition, one SIS train is assumed to be off line for service, leaving active only two MHSI pumps, two LHSI pumps and two emergency feedwater pumps. All four accumulators are assumed to inject.

EFW is actuated on the combination of LOOP and SI signal or on SG low wide-range level. The two active trains of MHSI are assumed to inject into Loop 4 (the broken loop) and into Loop 1 (the intact loop adjacent to the broken loop). The adjacent loop was chosen because it provides the greatest opportunity for injected ECCS to flow directly to the break and bypass the core.

The main feedwater (MFW) system and the reactor coolant pump (RCPs) are assumed to be tripped-off at reactor scram, coincident with an assumed LOOP.

The axial power shape used is a conservatively top-skewed, End of Cycle (EOC) shape. The power peak occurs at a normalized distance of 0.8542. The power in the hot rod is assumed at the technical specification peaking limits for the U.S. EPR. The sample problem assumes peaking limits of F_q = 2.6 and $F_{\Delta H}$ = 1.7.

[

]

The RCPs are modeled to reflect how the fluid is distributed vertically in the pumps to incorporate the physical elevation of the impeller into the model and to allow the fluid in the pump volutes (the pump discharge regions) to overflow into the discharge cold-legs. The pump model consists of a PUMP component representing the region from the pump suction to the impeller outlet and a BRANCH component representing the pump volute. The flow area of the PUMP component is set to that of the pump suction piping.

Following receipt of a SI signal, the SG MSRT system provides a controlled secondary system depressurization from 1414.7 psia (1384.7 +30 psia) (to 900 psia (870 psia + 30 psi uncertainty) at a rate corresponding to 180 ^{o}F/h.

Steam generator tube plugging is set to 5% symmetrically.

The core is modeled with a two-dimensional component having 28 axial nodes and three radial nodes. The fuel pin model utilizes a radial nodalization having eight intervals in the pellet. This is for compatibility with the RODEX2 fuel code, which is used to generate the EOC burn-up conditions that are input to S-RELAP5. The Baker-Just metal water reaction correlation is used for all fuel rod heat structures. The rupture model was invoked for the hot rod.

The limiting case was identified via a break spectrum analysis.

A.3. INITIAL CONDITIONS AND OTHER INPUT PARAMETERS

Table A-1 provides a listing of the initial conditions for the S-RELAP5 analysis of the SBLOCA event.

Table A-2 presents the safety classified I&C signals considered for this SBLOCA analysis. Degraded conditions are used to establish an uncertainty offset for signals unless the action occurs before degraded conditions occur. Mitigating systems are

assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. The RPS scram trip credited in the SBLOCA analysis is generated on low pressurizer pressure or low hot leg pressure. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

A.4. BREAK SPECTRUM

The sample problem analysis for the SBLOCA covers a range of breaks located in the pump discharge cold-leg. The analysis includes break diameters ranging from 2.0 inches to 8.0 inches in 0.5 inch increments, in addition to a maximum break having a 9.71 inch diameter (approximately ten percent of the cross-sectional area of the cold leg).

In addition, a break in the ECCS line is analyzed. For this break, in addition to the loss of ECCS trains due to single failure and preventive maintenance, one ECCS train (consisting of one MHSI, one LHSI, and one accumulator) injects into the broken ECCS line, which spills into containment.

The 6.5-inch break produced the limiting PCT. (See Table A-3)

The 6.5-inch break causes a longer period of core uncovery than do the 6.0-inch and 7.0-inch breaks. The rate of depressurization to the accumulator discharge pressure for the 6.5-inch break is slower than that for the 7.0 inch break; and, therefore, the amount of time the hot rod remains uncovered is greater.

The 2.0-inch and 2.5-inch breaks are small enough that MHSI flow is sufficient to prevent core uncovery. The 2.5 inch break depressurizes to the accumulator injection setpoint, however the cladding temperature turns around on MHSI injection only. All other breaks depressurize to the accumulator injection setpoint. The analyses are terminated when the PCT turns around, the vessel downcomer is filled to the elevation of the cold-leg nozzles with a two-phase mixture, and the core collapsed level and

vessel inventory show a steady or increasing trend. For the smaller breaks (less than 6 inch) the recovery occurs without LHSI injection. For the other larger small breaks, the primary depressurizes to the shutoff head of the LHSI pumps.

The 7.0-inch break behaves similarly to the 6.5-inch break in that it recovers as soon as accumulator discharge begins. For breaks larger than 7.0 inch, the accumulators empty, hence the accumulators were disconnected when noncondensibles were detected at the accumulator exit.

A.5. 6.5-INCH BREAK (LIMITING CASE)

Table A-4 presents the sequence of events for the 6.5 inch break case. Cladding temperatures rise steadily from about 700 seconds to about 1350 seconds, when cladding quench occurs.

Selected parameters are presented in Figures A-5 through A-20. The reactor trip (RT) occurs at 4.493 seconds due to low hot-leg pressure (Figure A-5). All RCPs are tripped due to LOOP at RT.

After the initiation of the break, the primary pressure drops rapidly to the saturation pressure of the hottest reactor coolant loop (Figure A-6). The depressurization is halted at about 25 seconds as the energy removal by the SGs and through the break is less than the energy generated in the core.

Initially, the secondary side pressure increases rapidly due to the closing of the turbine stop valves at the time of LOOP. This pressure increase is limited by the opening of the MSRIVs at about 114 seconds (Figure A-7) which results in a drop in secondary pressure of about 43 psia. Due to the large SG pressure decrease, the main steam relief control valve (MSRCV) closes at 134 seconds. At about 170 seconds the MSRCV reopens when the SG pressure intersects the cooldown curve. From 170 seconds to the end of the transient, the MSRCV valve modulates to depressurize the secondary

side at a rate corresponding to 180 ^{o}F/hr. At about 255 seconds the primary pressure drops below the secondary pressure (Figure A-6).

After the break uncovers at about 250 seconds, the rate of energy removal out of the break increases and the break flow transitions from two-phase to steam (Figure A-8) causing the RCS depressurization rate to increase. Loops 2 and 3 clear at 234 and 237 seconds, respectively followed by loop 1 and 4 at 360 and 362 seconds which creates a flow path for the steam to vent out of the break (Figure A-9). The depressurization continues until the accumulator flow begins at about 346 seconds (Figure A-10) at which time the pressure increases slightly (Figure A-6). This break is more limiting than the 6 inch break because the clearing of loops 1 and 4 is delayed by about 90 seconds between the 6 inch and 6.5 inch break.

The SG main feedwater flow is isolated at 4.494 seconds on RT (Figure A-11).

In the beginning of the transient, as the core decay heat cannot be removed by the break alone, the SGs are important in providing the RCS heat removal. The flow through the MSRCV does not result in the SG mass dropping before EFW starts to inject (Figure A-12). The mass drops in SG 2 and 3, in which there is no EFW injection. Same as the mass inventory, the level decreases in SG 2 and 3 and increases in SG 1 and 4 (Figure A-13). The EFW pumps for SGs 1 and 4 begin injecting at 76.807 seconds on SI and LOOP signals. After 170 seconds, the secondary pressure continues to decrease (Figure A-14) as the MSRCVs modulate to depressurize the secondary side at a rate corresponding to 180°F/hr. The SG pressure history and the MSRTs pressure setpoint corresponding to the cooldown rate of 180°F/hr is shown in Figure A-15.

The RCS inventory, and consequently the collapsed hot assembly liquid level, fall rapidly as primary fluid is lost out of the break (Figures A-16 and A-17). At about 246 seconds two MHSI pumps begin to inject into Loops 1 and 4 reducing the net rate of coolant inventory loss (Figure A-18). However, the inventory lost out the break exceeds

that supplied by the MHSI pumps, resulting in RCS net inventory loss and core uncovery. A PCT of 1638°F occurs at 360.26 seconds (Figure A-20)

The large quantity of water supplied by the accumulators terminates the net loss of primary coolant inventory, recovering the core level and ultimately quenching the core. As the RCS depressurizes further, the MHSI and LHSI (Figure A-19) overcome the break flow, providing extended, adequate RCS coolant inventory.

The analysis is terminated at 1000 seconds, at which time RCS and RV inventories are stable and the decay heat is being removed by the combination of the MHSI, LHSI and break flow.

A.6. RESULTS/SUMMARY/CONCLUSION

AREVA NP's, NRC-approved, SBLOCA evaluation methodology was used to perform a SBLOCA sample problem analysis for the U.S. EPR plant design. The analyses covered a spectrum of breaks in the pump discharge cold leg piping and one ECCS piping break. The analyses assumed the worst-case failure of a diesel generator that takes out one entire train of SIS injection and EFW. Additionally, it assumed another train of SIS injection was unavailable because of preventive maintenance. EOC fuel conditions were assumed as calculated by RODEX2. Peak cladding temperature was calculated for the hot rod using S-RELAP5.

The analysis of the limiting case SBLOCA demonstrates that the acceptance criteria are met as follows:

- A PCT of 1638°F was calculated for the limiting case. This is below the 2200°F PCT limit specified in 10 CFR 50.46(b)(1).
- The total cladding oxidation at the PCT location is 0.383 percent for the limiting case. This is below the 17 percent limit specified in 10 CFR 50.46(b)(2).
- The hydrogen generated in the core during the SBLOCA by cladding oxidation, 0.00897 percent, is below the one percent limit specified in 10 CFR 50.46(b)(3).

- The calculation shows that the core retains a coolable geometry. Thus, the coolable geometry criterion in 10 CFR 50.46 (b)(4) is satisfied.

  The analysis also demonstrated that SBLOCA events in the U.S. EPR exhibit all the phenomena encountered in a U.S. PWR plant. The limiting break exhibits all five phases of a small break LOCA: blowdown, natural circulation, loop seal clearance, boil off and core recovery. As such, the U.S. EPR behaves in the same manner as other U.S. PWRs during a SBLOCA event.

The overall conclusion is that the current SBLOCA evaluation methodology is applicable to the U.S. EPR. The U.S. EPR is similar to other U.S. PWRs for which the methodology was used successfully to analyze SBLOCA events.

A.7. REFERENCES

A-1. "PWR Small-Break LOCA Evaluation Model, S-RELAP5 Based," AREVA NP Report EMF-2328 (P)(A), Revision 0, March 2001.

Table A-1 Input Parameters

Parameter	U.S. EPR Value
Core power	4590.0 MWth + 0.48 % = 4612 MWth
Axial power shape and power peaking	An EOC top skewed
Loop flow rate/per loop	119,692 gpm
Reactor vessel Average temperature	594°F
Primary pressure (Pressurizer Pressure)	2250 psia
Initial Pressurizer Liquid Level	54.3%
Secondary pressure	1103.2 psia, consistent with the plugging level
Initial SG Level	49 % NR
SG Secondary Side Inventory	181480 lbm/SG
Accumulator pressure	652.7 psia
Accumulator Volume	Accumulator Liquid Volume: 1324.3 ft^3, Accumulator Total Volume: 1942.3 ft^3
Accumulator temperature	90.5°F
MHSI fluid temperature	122°F
SG tube plugging[1]	5%
Minimum EFW Flow	400 gpm
EFW Temperature	122°F
EFW-Start Time	Offsite-power-available cases: 15 sec after signal LOOP cases: 60 sec after signal
Single Failure Assumption (see note 1)	Loss of 1 EDG + 1 EDG in Preventive Maintenance

[1] Performing the SBLOCA analysis at higher tube plugging levels is conservative because the initial primary system mass inventory is reduced.

Table A-2 Safety Classified I&C Signals Considered for SBLOCA Analysis

Signal	Setpoint and Time Delay
Reactor Trip, on low PZR pressure	2005-55 = 1950 psia Degraded conditions 55 psi Delay = 1.2[2] seconds
Reactor Trip, on low hot leg pressure	2005-75 = 1930 psia Degraded conditions 75 psi Delay = 1.2[3] seconds
Turbine Trip, on RT signal	Trip Delay = 1.2 +0.15 seconds
Beginning of Rod Insertion	1.2 seconds after RT signal (0.9 I&C + 0.3 seconds for RT breaker opening and gripper release), 3.5 seconds for complete rod insertion (seismic conditions)
MFW Isolation	Assumed at closure of the turbine valve.
Safety Injection, on low-low PZR pressure	1667.9 - 55 = 1612.9 psia Degraded conditions 55 psi Time delay: 1.5 seconds plus 15 seconds w/o LOOP for SI delivery 40 seconds with LOOP includes EDG loading
RCP trip	RCP trip on LOOP at the time of scram
RCP trip w/o LOOP	ΔP across RCP (pump trip) 80%-5% =75% Degraded conditions (5%) Time delay; 3.9 seconds[4]
MSRT opening and pressure control,(partial cooldown signal simultaneously with SI signal)	1384.7 + 30[5] psia before beginning of partial cooldown. 870.0 + 30 psia after the end of partial cooldown. Time delay: 1.5 seconds
EFWS initiation	EFWS Actuation on LOOP + SIS Actuation 15 sec w/o LOOP for EFW delivery 60 sec with LOOP (includes EDG loading)
EFW Initiation on Low SG Level	40% WR -11% (11% degraded conditions) Delay:1.5 seconds plus 15 seconds w/o LOOP for EFW delivery 60 seconds with LOOP (includes EDG loading)

[2] For reactor trip functions the time delay is from the time the value is sensed at the sensor until the stationary gripper releases. It includes sensor delay, I&C delay (0.9sec), and the delay for the trip breakers to open and the stationary gripper to release (0.3 sec).
[3] For reactor trip functions the time delay is from the time the value is sensed at the sensor until the stationary gripper releases. It includes sensor delay, I&C delay (0.9sec), and the delay for the trip breakers to open and the stationary gripper to release (0.3 sec).
[4] The delay includes 3 seconds for bus supply breaker trip delay
[5] The non-degraded condition is used because the instrumentation is located outside containment

Table A-3 Break Spectrum - Peak Cladding Temperature

Break Diameter (in)	Break Area (ft^2)	PCT (^oF)	Time of PCT(sec)
2.0	0.0218	No Heatup	N/A
2.5	0.0341	1042	5000.2
3.0	0.0491	917	2986.4
3.5	0.0668	949	1837.9
4.0	0.0873	1088	1222.3
4.5	0.1104	1223	908.13
5.0	0.1364	1085	679.28
5.5	0.165	1199	548.34
6.0	0.1963	1125	459.28
6.5	**0.2304**	**1638**	**360.26**
7.0	0.2673	1587	305.39
7.5	0.3068	1479	266.43
8.0	0.3491	1469	234.27
Break in Acc line 8.4993	0.394	1459	208.6
Max Break 9.71	0.5143	1416	161.98

Table A-4 Limiting Break Sequence of Events

Time (sec)	Event
0	Begin Analysis
0	Break Opened
4.493	Reactor Tripped on Low Hot Leg Pressure
4.494	RCPs Tripped
16.807	SIS Signal on Low Pressurizer Pressure
76.807	Emergency Feedwater Initiated (Loop 1 and 4)
114	MSRIV opens
134	MSRCV closes (faster SG depressurization)
170	MSRCV reopens to control SG depressurization at a rate of 180 ^{o}F/hr
234	Loop Seal Clearing – Loop 2
237	Loop Seal Clearing – Loop 3
246	Broken Loop MHSI Delivery Began (Loop 4)
246	Intact Loop MHSI Delivery Began (Loop 1)
250	Break uncovery
346	Accumulator Injection (Loop 1, 2, and 3 and 4 respectively)
360	Loop Seal Clearing – Loop 1
360.26	PCT Occurred (1638, node #31)
362	Loop Seal Clearing – Loop 4
380	Broken Loop LHSI Delivery Began (Loop 4)
380	Intact Loop LHSI Delivery Began (Loop 1)
1000	Transient Calculation Terminated

Figure A-1 U.S. EPR Vessel Nodalization

Figure A-2 U.S. EPR Steam Generator Nodalization

Figure A-3 U.S. EPR Loop Nodalization

Figure A-4 Nodalization for S-RELAP5 ECC Model

Figure A-5 Reactor Power for 6.5 Inch Break

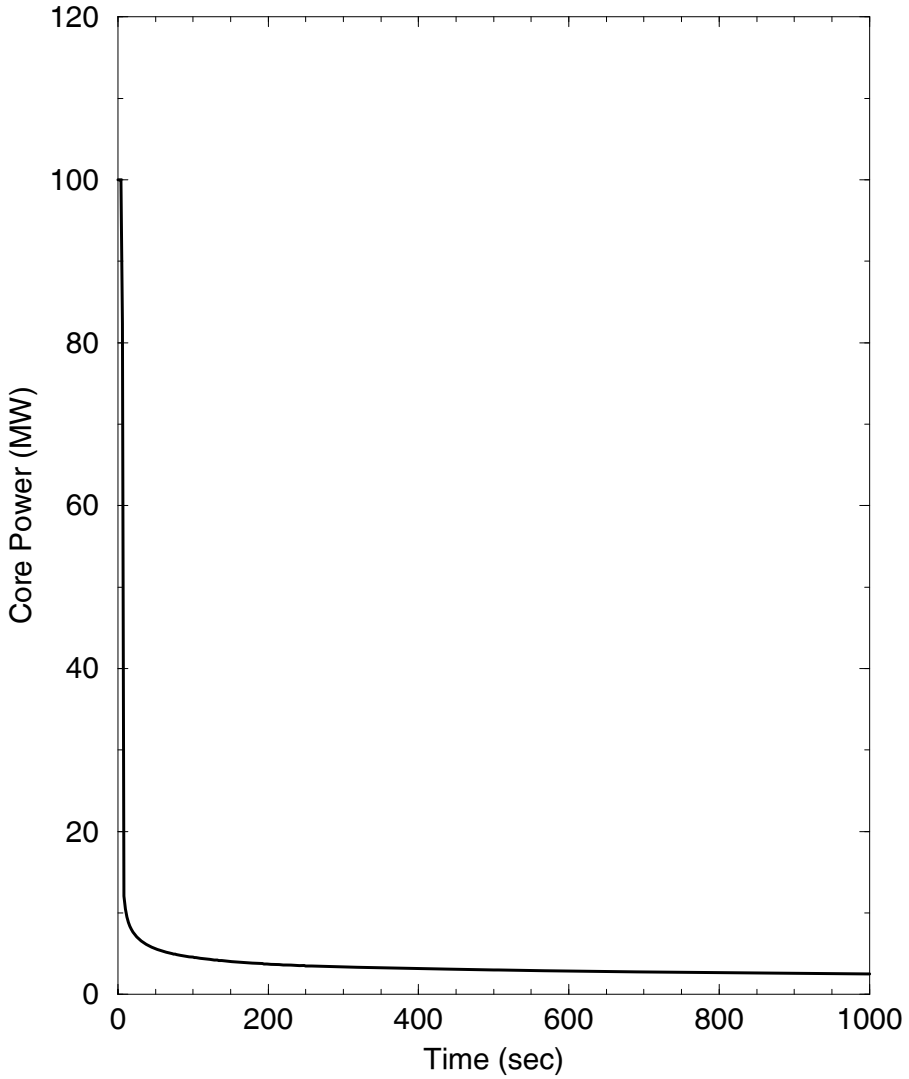

Figure A-6 Primary and Secondary System Pressure for 6.5 Inch Break

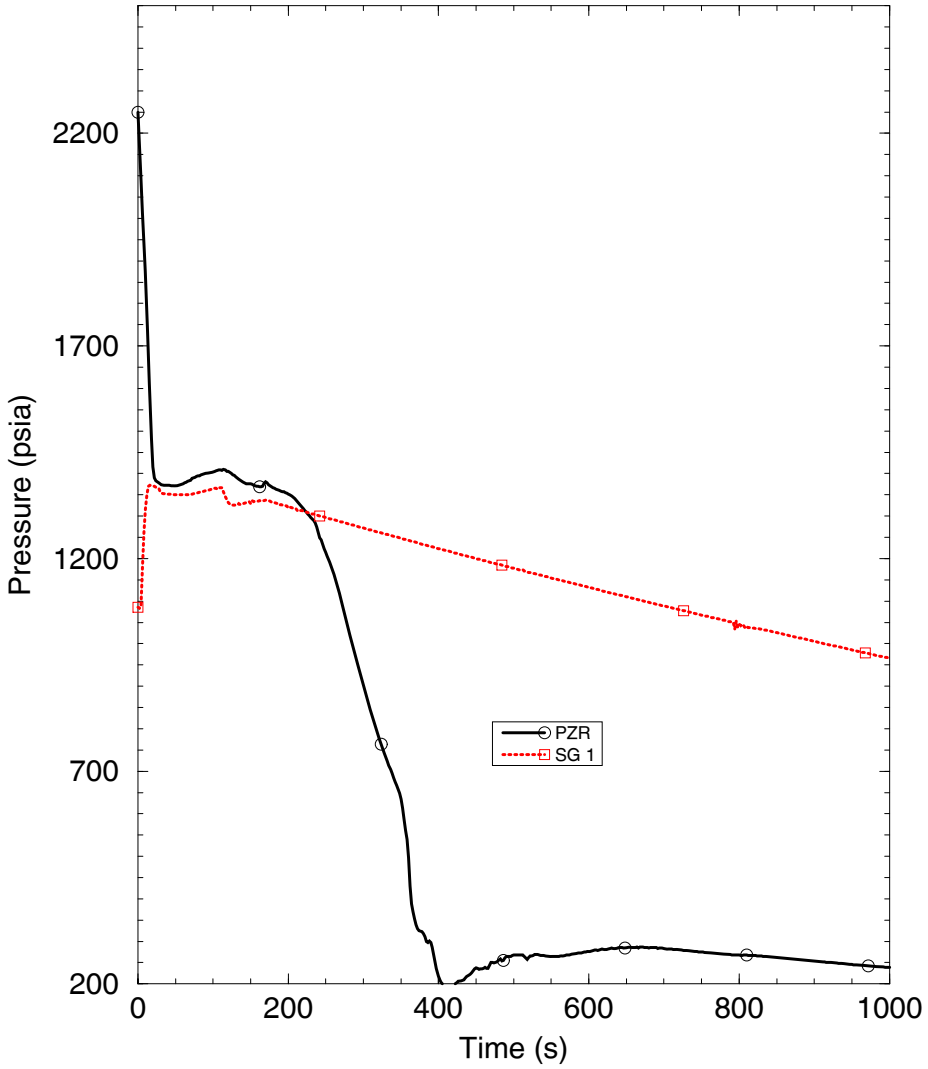

Figure A-7 MSRT flow for 6.5 inch Break

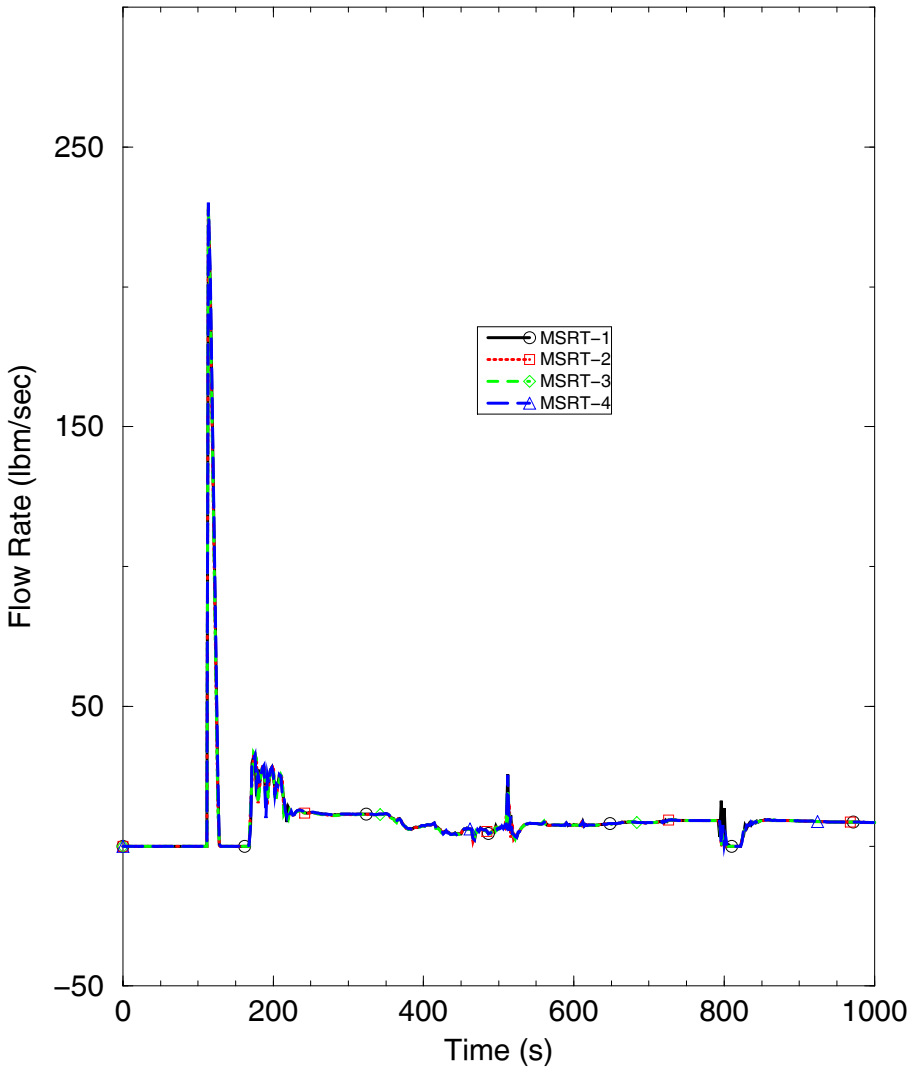

Figure A-8 Break Flow Rate for 6.5 Inch Break

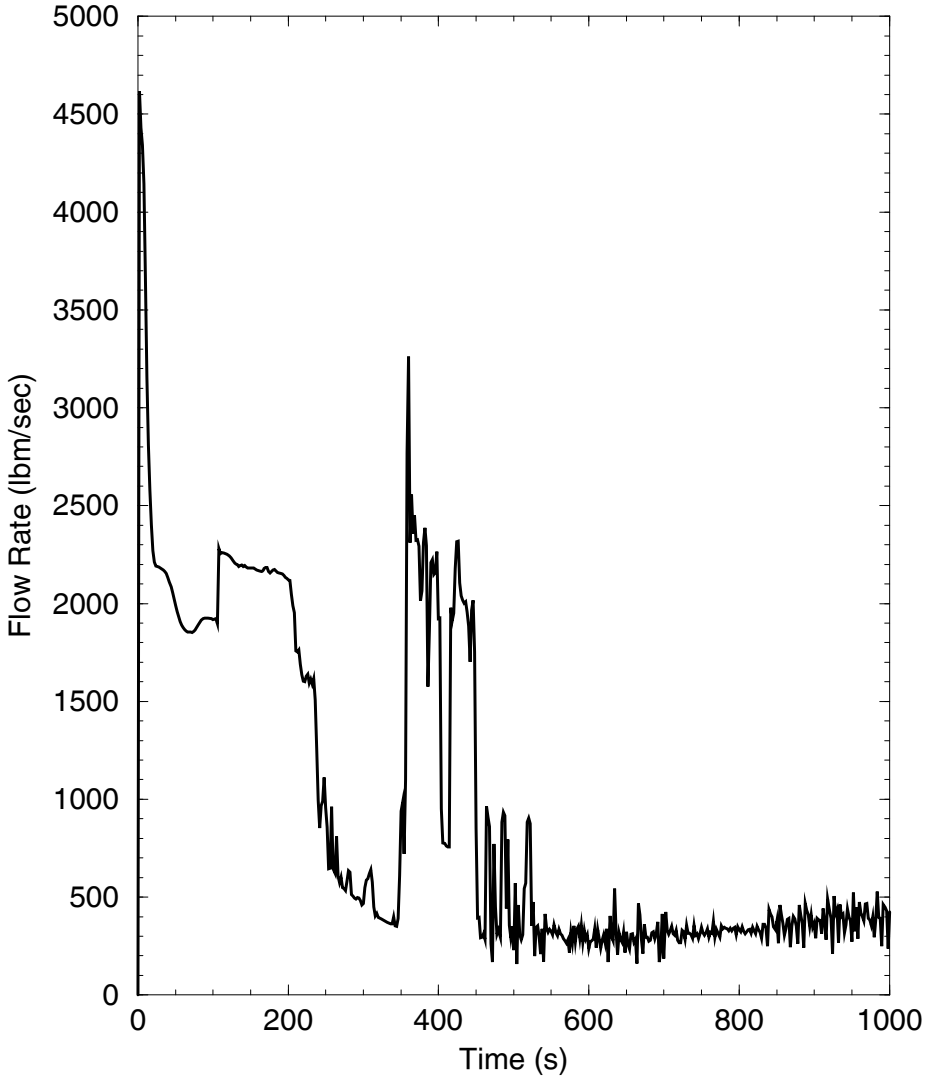

Figure A-9 Loop Seal Void Fraction for 6.5 Inch Break

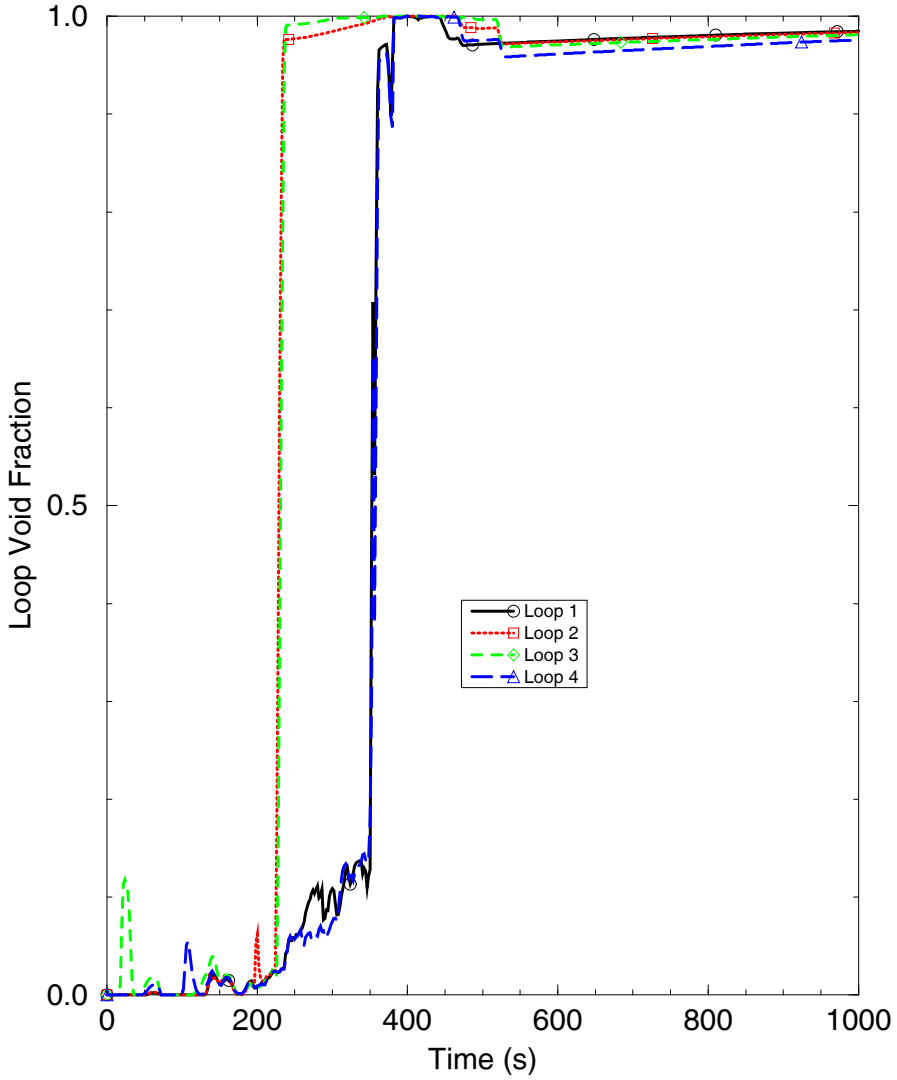

Figure A-10 ECCS Flow for 6.5 Inch Break

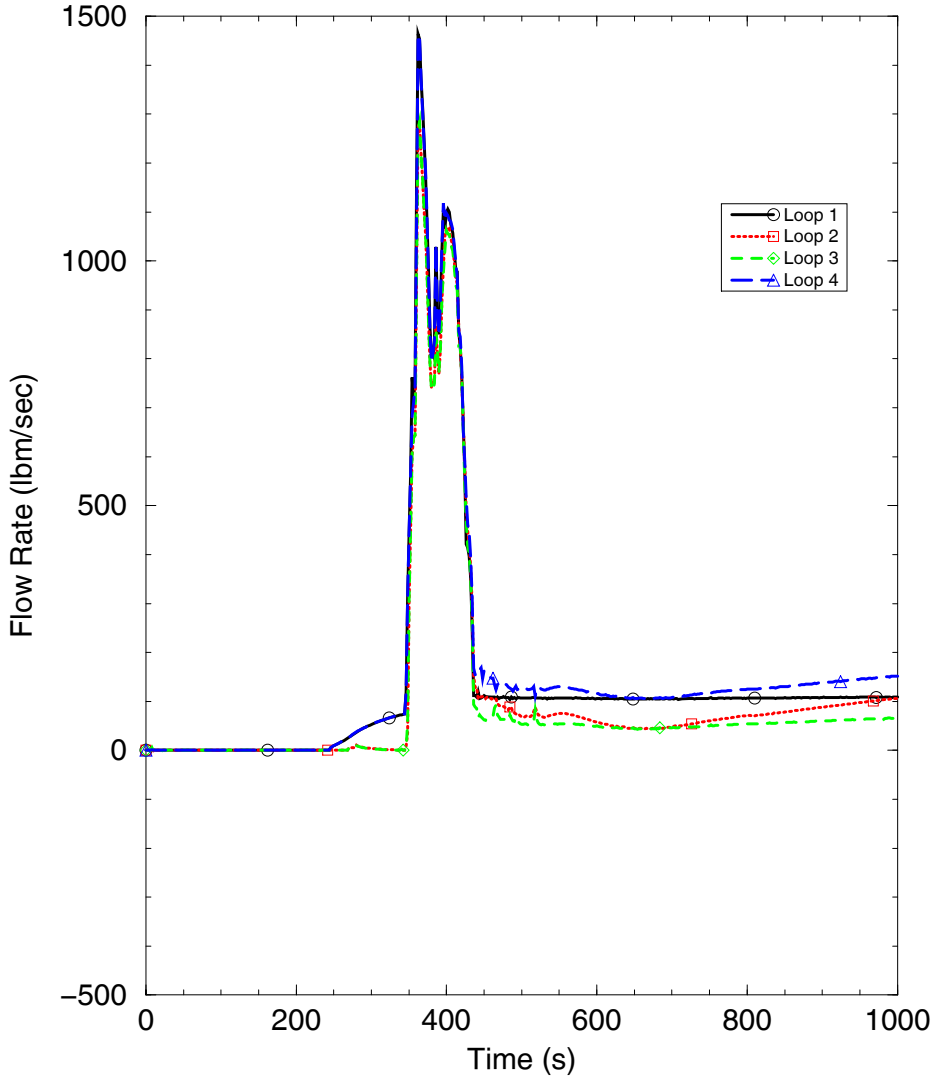

Figure A-11 MFW Flow Rate for 6.5 Inch Break

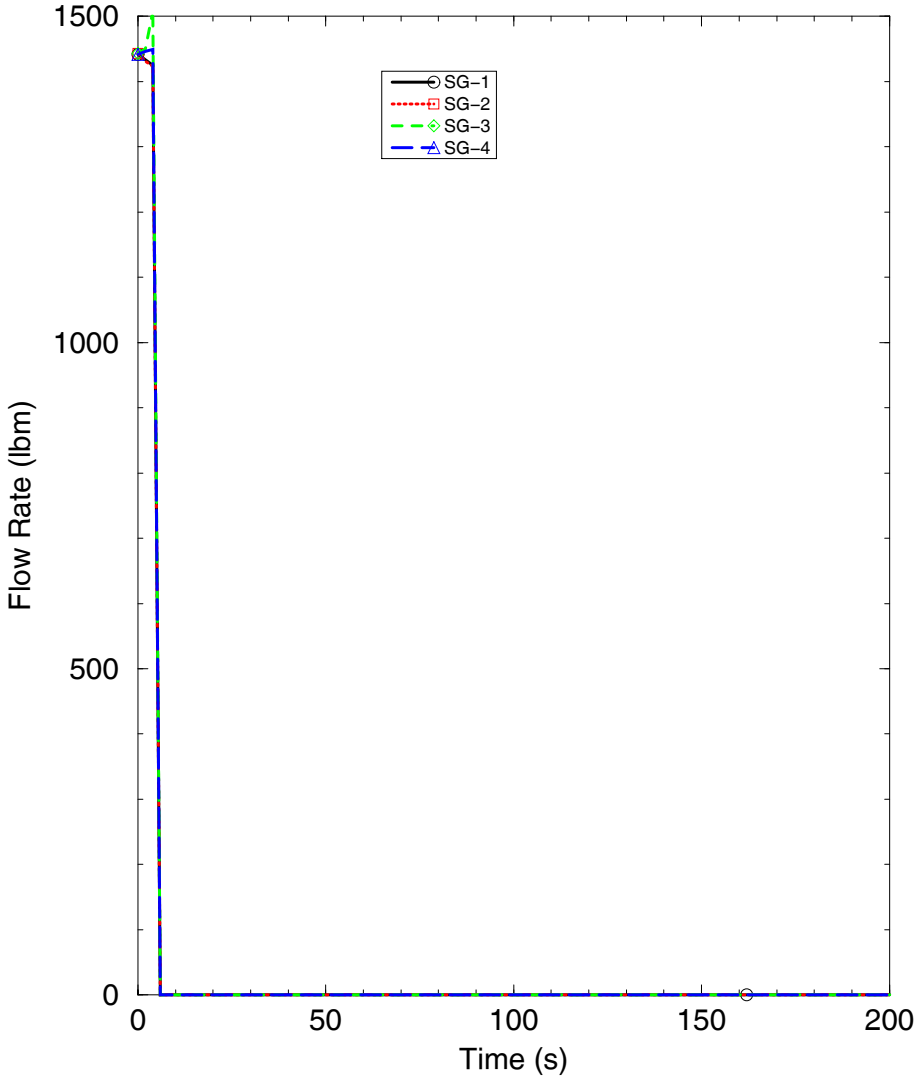

Figure A-12 Steam Generator Mass Inventory for 6.5 Inch Break

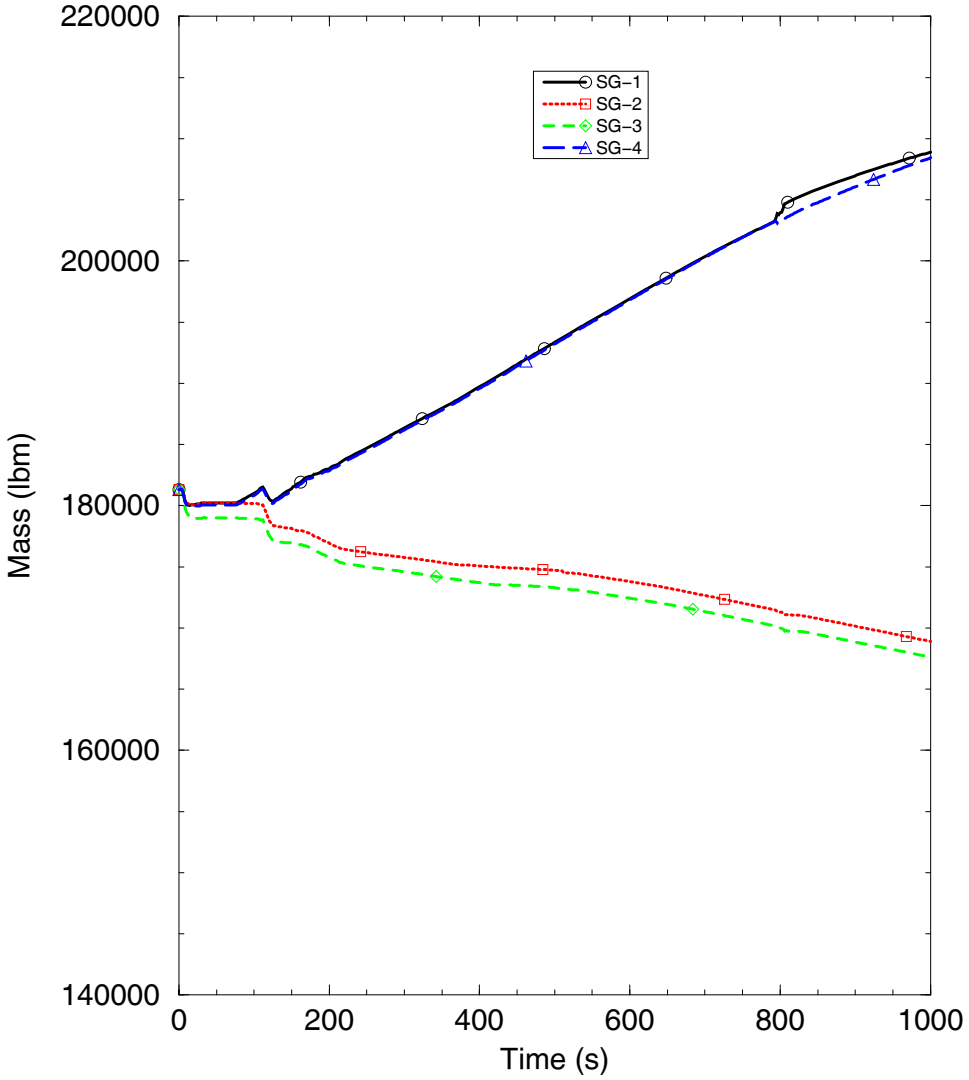

Figure A-13 Steam Generator Level for 6.5 Inch Break

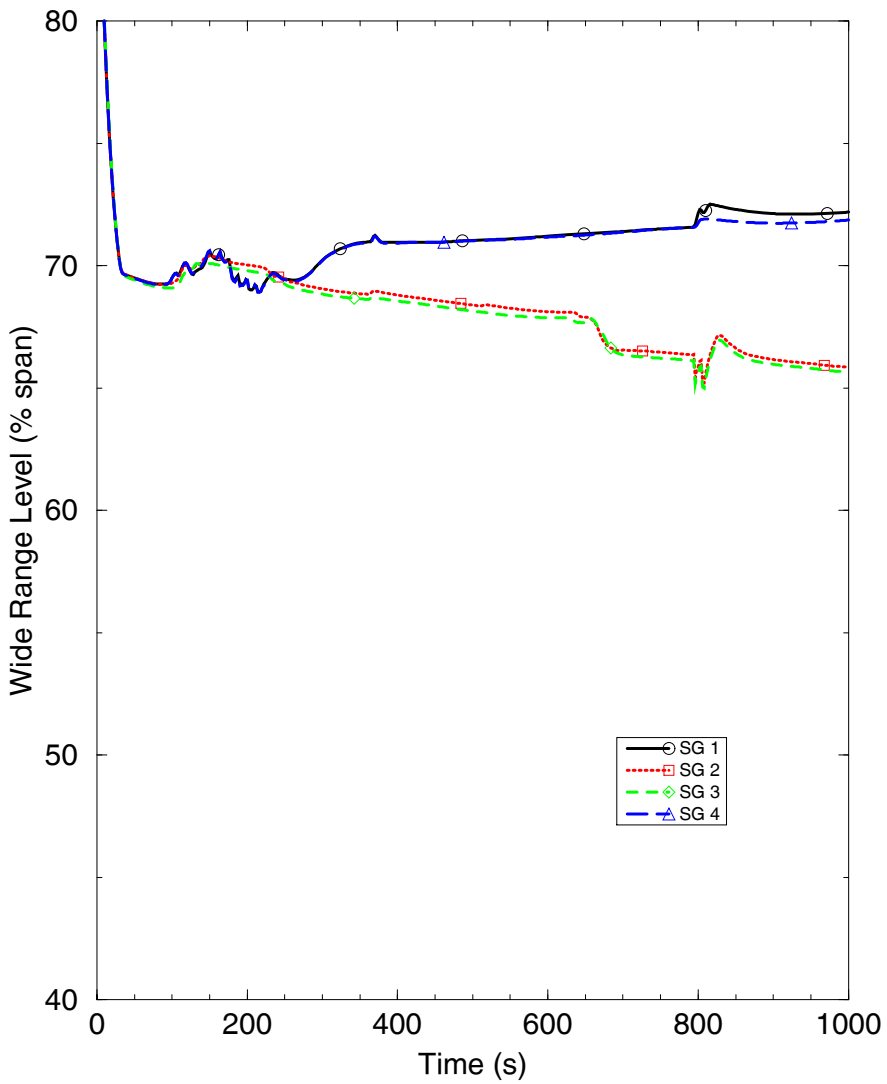

Figure A-14 Steam Generator Steam Line Pressure for 6.5 Inch Break

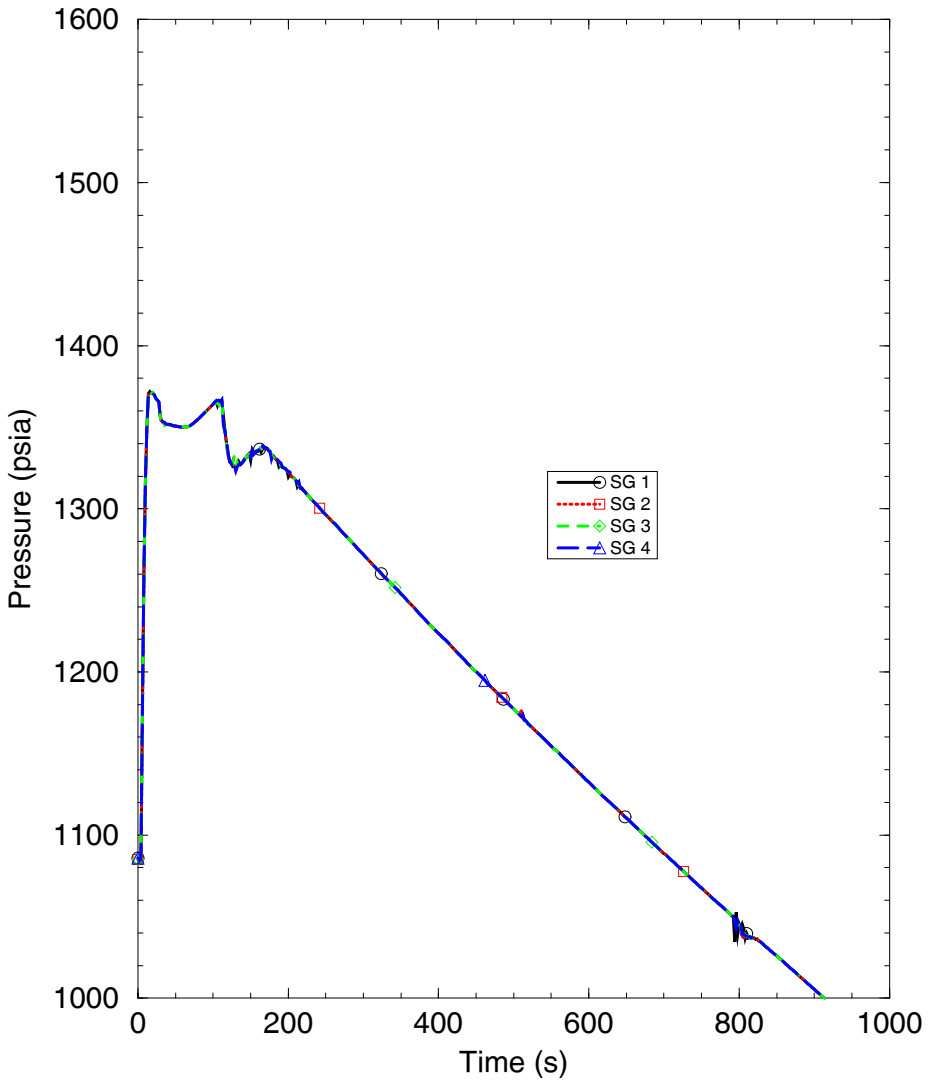

Figure A-15 Steam Generator Steam Line Pressure and Cooldown Curve for 6.5 Inch Break

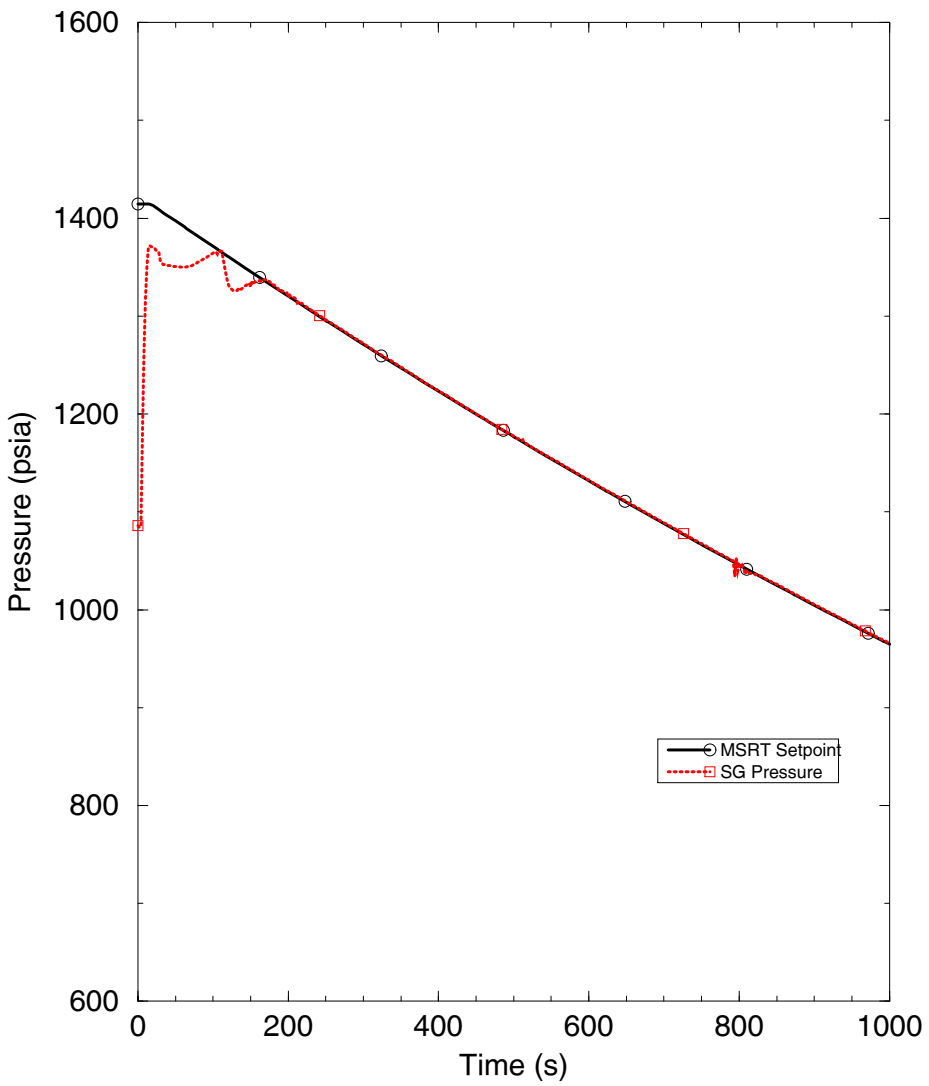

Figure A-16 Total RCS and RV Mass Inventories for the 6.5 Inch Break

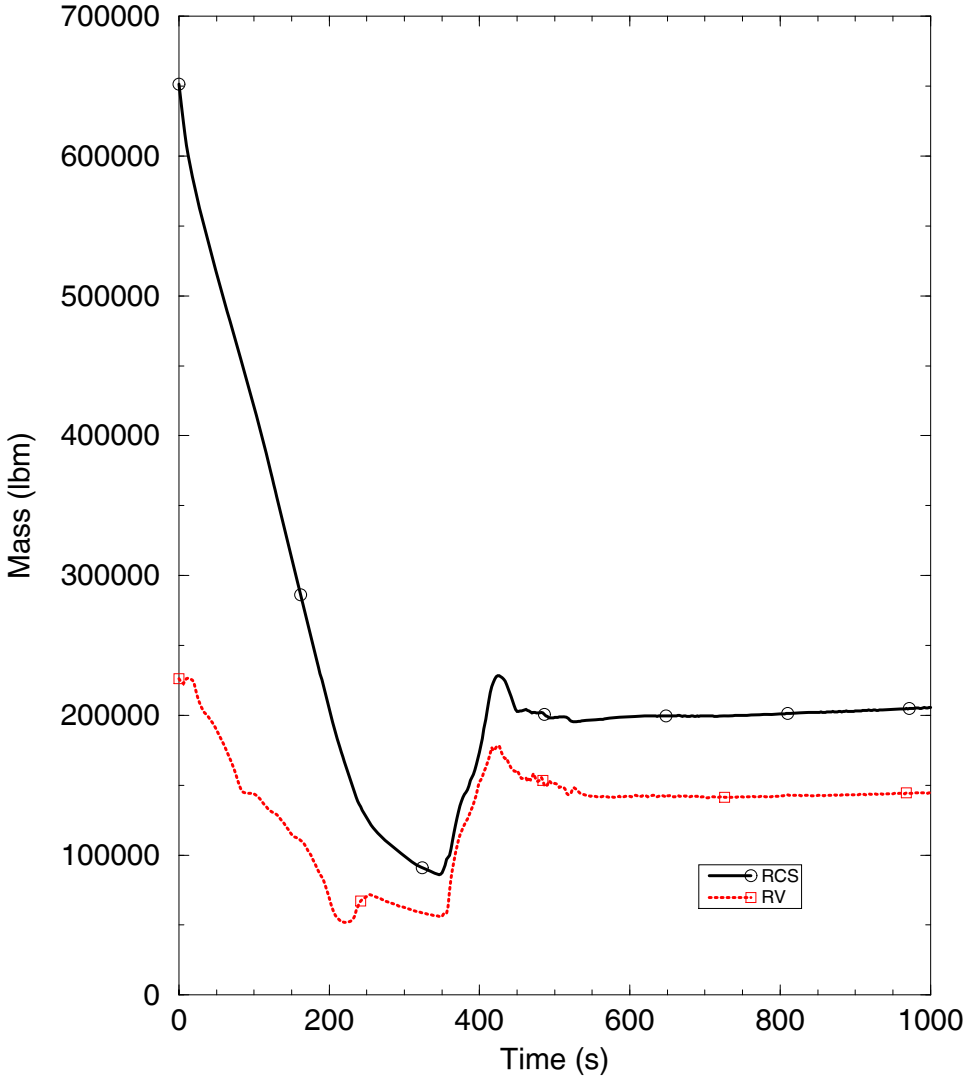

Figure A-17 Hot Assembly Collapsed Liquid Level for 6.5 Inch Break

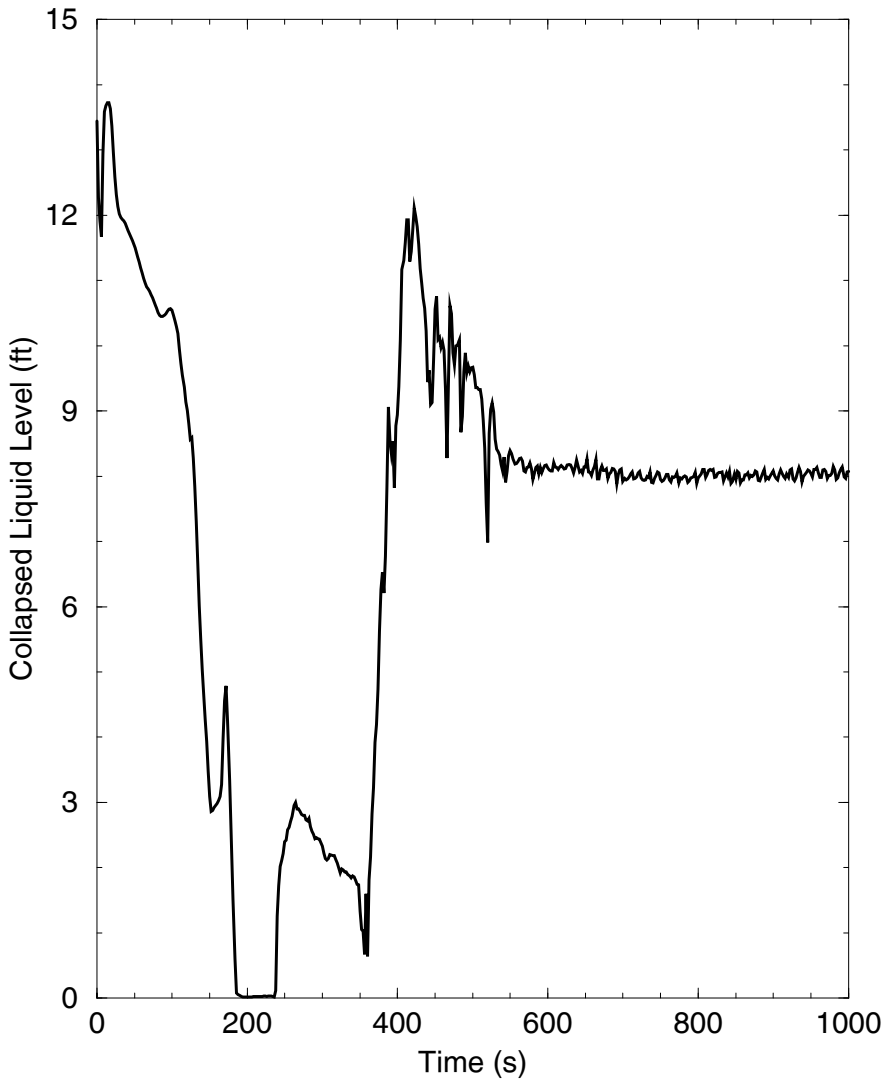

Figure A-18 MHSI Flow (Loops 1 and 4) for 6.5 Inch Break

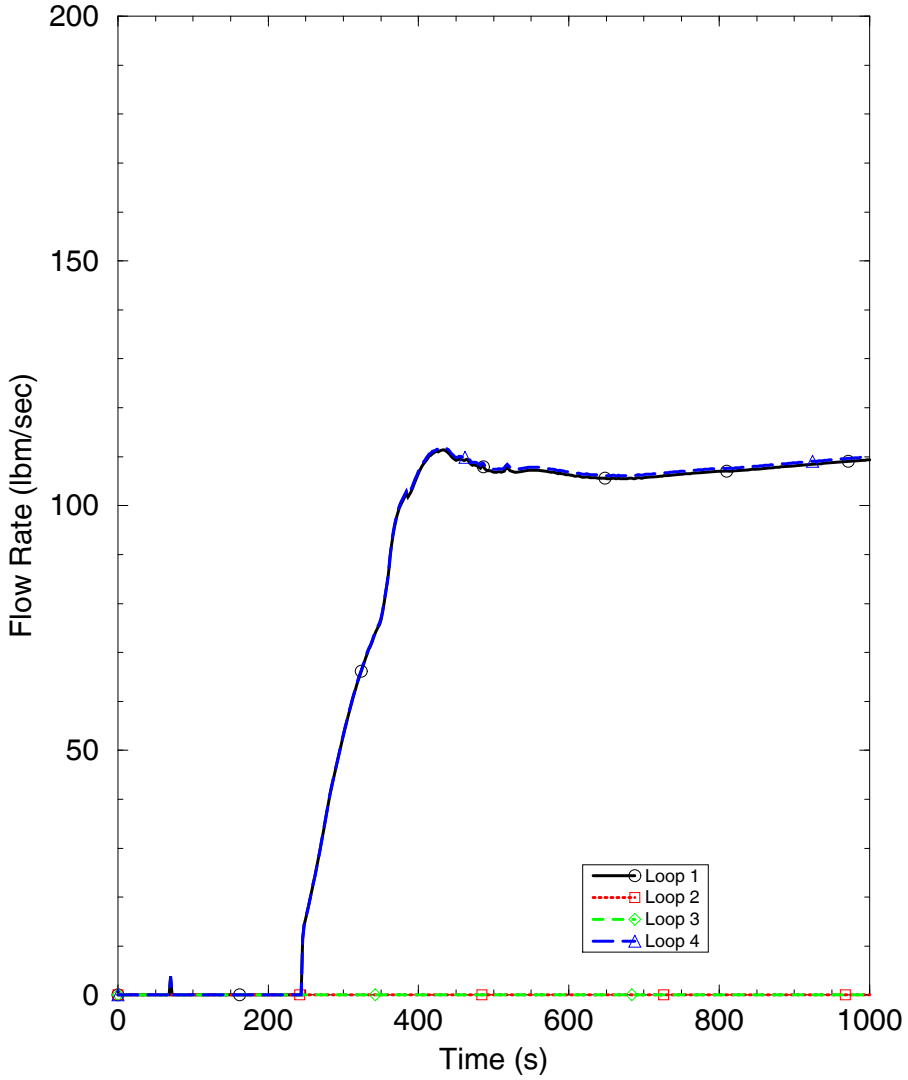

Figure A-19 LHSI Flow (Loops 1 and 4) for 6.5 Inch Break

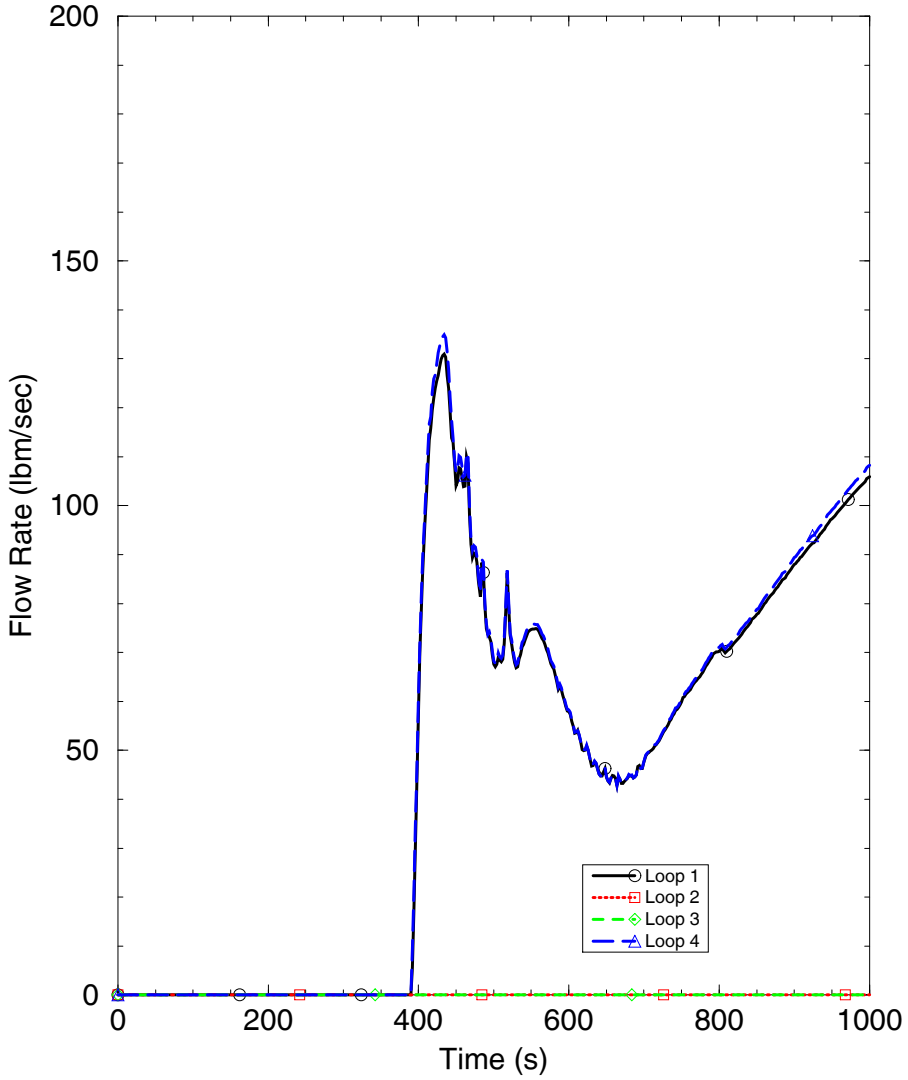
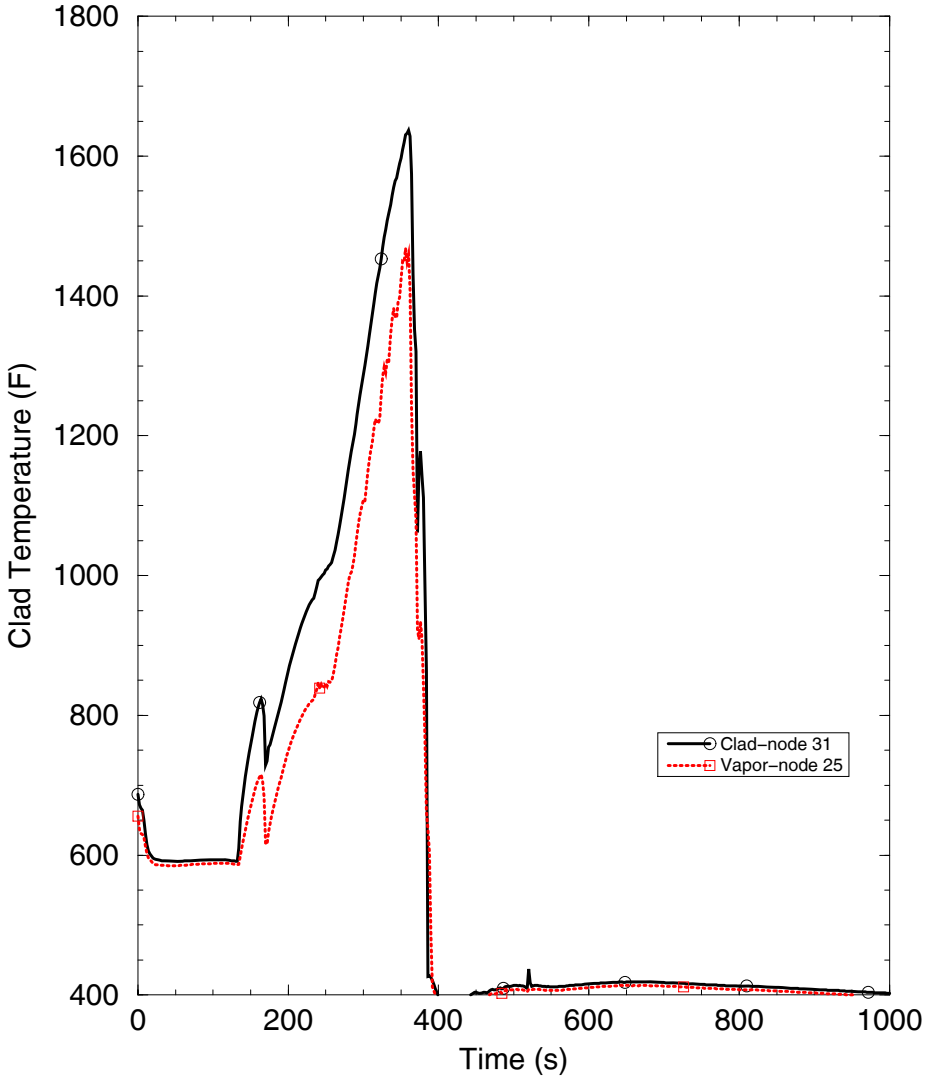
Figure A-20 Vapor and Clad Temperature for the Hot Node for a 6.5 Inch Break