ATTACHMENT 9

ANP-2971(NP), Revision 1

Sequoyah Units 1 and 2 HTP Fuel S-RELAP5 Small Break LOCA Analysis

May 2011

(NON-PROPRIETARY VERSION)

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Revision 1

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HTP Fuel S-RELAP5
Small Break LOCA Analysis

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AREVA NP Inc.

AREVA

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Nature of Changes

Item	Page	Description and Justification
1	All	Revision 1. Changed the PCT result for the accumulator line break.

Table of Contents

This document contains a total of 55 pages.

List of Tables

List of Figures

Nomenclature

ADV	Atmospheric dump valve
AFAS	Auxiliary feedwater actuation signal
AFW	Auxiliary feedwater
AOR	Analysis of Record
BOC	Beginning-of-cycle
CCI	Centrifugal Charging Injection
CEA	Control Element Assembly
CFR	Code of Federal Regulation
CRGT	Control Rod Guide Tube
DC-HL	Downcomer – Upper Head
ECCS	Emergency core cooling system
EDG	Emergency diesel generator
EOC	End-of-cycle
EOP	Emergency Operating Procedure
HHSI	High head safety injection
HTP	High Thermal Performance
LHR	Linear heat rate
LOCA	Loss of coolant accident
LOOP	Loss of offsite power
LHSI	Low head safety injection
MFW	Main feedwater
MSIV	Main steam isolation valve
MSSV	Main steam safety valve
MWt	Mega-Watt thermal
NRC	Nuclear Regulatory Commission
PCT	Peak cladding temperature
PWR	Pressurized water reactor
RAI	Request for Additional Information
RCP	Reactor coolant pump
RCS	Reactor coolant system
RHR	Residual Heat Removal
SBLOCA	Small Break Loss-of-Coolant-Accident
SEQ	Sequoyah
SG	Steam generator
SGTP	Steam generator tube plugging
SI	Safety injection
SIAS	Safety injection actuation signal

1.0 Introduction

This report documents a SBLOCA analysis for the Sequoyah Units 1 & 2 (SEQ1 and SEQ2) plants for the transition from AREVA Mark-BW fuel to AREVA HTP fuel. This analysis was performed with the S-RELAP5 methodology (References [1] and [2]).

A spectrum of break sizes from 0.0055 (1-inch diameter) to 0.5217 ft^2 (9.78-inch diameter) was analyzed, including the reactor coolant pump trip sensitivity study and the attached piping break study to support the operation of SEQ1 and SEQ2.

This analysis supports plant operation at a power level of 3479 MWt (including measurement uncertainty), a F_Q of 2.65, a radial peaking factor of 1.7056 (1. 64 plus a 4% measurement uncertainty), and 15% steam generator tube plugging.

2.0 Summary of Results

A SBLOCA break spectrum analysis was performed for SEQ1 and SEQ2. The limiting PCT is 1470 °F for a break of 9.76-inch diameter pump discharge cold leg break.

This analysis was performed to demonstrate that the following acceptance criteria for Emergency Core Cooling Systems, as stated in 10 CFR 50.46(b)(1-4), have been met.

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

The break spectrum calculations conservatively assumed RCP trip at reactor trip and loss of offsite power. An evaluation of delayed RCP trip was performed since a delayed RCP trip can potentially produce more limiting results. Section 4.3 discusses the RCP trip sensitivity study. The results with respect to RCP trip would be "a trip within 2 minutes of loss of subcooling or SIAS actuation" does not result in a violation of 50.46(b).

3.0 Description of Analysis

Section 3.1 below provides a brief description of the postulated SBLOCA event. Section 3.2 describes the analytical models used in the analysis. Section 3.3 presents a description of SEQ1 and SEQ2 plant parameters and outlines the system parameters used in the SBLOCA analysis.

3.1 Description of SBLOCA Event

The postulated SBLOCA is defined as a break in the RCS pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location for the base break spectrum cases assuming LOOP is in the cold leg pipe on the discharge side of the RCP. This break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected out through the break. This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b)(1-4) criteria.

The SBLOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a Low Pressurizer Pressure signal. The SIAS occurs when the system has further depressurized. The capacity and shutoff head of the HHSI pumps are important parameters in the SBLOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is large enough that the HHSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the accumulator injection pressure (600 psig) or to recover core liquid level on HHSI flow. This tends to maximize the heatup time of the hot rod which produces the limiting PCT and local cladding oxidation. Core recovery for the limiting break begins when the SI flow that is retained in the RCS exceeds the mass flow rate out the break, followed by accumulator injection. For very small break sizes, the primary system pressure does not reach the accumulator injection pressure.

3.2 Analytical Models

The AREVA S-RELAP5 SBLOCA evaluation model for event response of the primary and secondary systems and hot fuel rod used in this analysis (Reference [1]) consists of two computer codes. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated. This methodology has been reviewed and approved by the NRC to perform SBLOCA analyses. The two AREVA computer codes used in this analysis are:

1. The RODEX2-2A code was used to determine the burnup-dependent initial fuel rod conditions for the system calculations.

2. The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response.

The gap conditions used to initialize S-RELAP5 are taken at EOC, consistent with an EOC top-peaked axial power distribution. The use of EOC fuel rod conditions along with an EOC power shape is bounding of BOC because (1) the gap conductance is higher at EOC, and (2) the power shape is more top-skewed at EOC. The initial stored energy, although higher at BOC, has a negligible impact on SBLOCA results since the stored energy is dissipated long before core uncovery.

Figures 3-1 through 3-3 show the RCS, Secondary, and Reactor Vessel nodalization.

Figure 3-1 S-RELAP5 SBLOCA Reactor Coolant System Nodalization

Figure 3-2 S-RELAP5 SBLOCA Secondary System Nodalization

Figure 3-3 S-RELAP5 SBLOCA Reactor Vessel Nodalization

3.3 Plant Description and Summary of Analysis Parameters

The plant analysis presented in this report is for a Westinghouse-designed PWR, which has four loops, each with a hot leg, a U-tube steam generator, and a cold leg with a RCP[1]. The RCS Also includes one pressurizer connected to a hot leg. The core contains (193) 17x17 AREVA fuel assemblies. The ECCS includes one charging and one accumulator/SI/RHR injection path per RCS loop. The SI and RHR feed into common headers which are connected to the accumulator lines. The charging pumps are also cross-connected. The limiting break location will be in a cold leg pump discharge location of piping. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e., RAS) for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the RCS, RV, pressurizer, and accumulator lines. The charging injection flows are connected to the RCS, and the SI and RHR injection flows are connected to the accumulator lines, consistent with the plant layout. A symmetric steam generator tube plugging level of 15-percent per steam generator was assumed.

MFW is injected into the downcomer of each SG. There are three AFW pumps, that is, two motor-driven and one turbine-driven pumps. The ECCS for the SBLOCA contains two HHSI pumps, four accumulators.

The RCS was nodalized in the S-RELAP5 model into control volumes interconnected by flow paths or "junctions." The model includes four accumulators, a pressurizer, and four SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SGTP level of 15% in each steam generator was assumed. The HHSI system was modeled to deliver the total SI flow asymmetrically to the broken loop and three intact loops in the S-RELAP5 model. The RHR system is included in the model.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.
The analysis assumed loss of offsite power concurrent with reactor scram on low pressurizer pressure. The single-failure criterion required by Appendix K was satisfied by assuming the loss of one EDG, which resulted in the disabling of one HHSI pump and a motor-driven AFW pump. This yields only one HHSI pump providing flow to the RCS. It also results in only one motor driven AFW pump being available. Minimum capability, degraded flow was assumed from the single operable HHSI pump. For Sequoyah, where charging pumps are considered safety grade, charging pump flow is included with the HHSI flow. Initiation of the HHSI system was delayed by 37-seconds beyond the time of SIAS. The 37-second delay represents the time required for diesel generator startup and switching, and the time it takes for the centrifugal charging pump cold leg injection valves to fully open.

[1] The RCPs are Westinghouse 93A type pumps. The homologous pump performance curves for this type of pump were input to the S-RELAP5 plant model.

The initiation of the motor-driven pump was delayed by 60-seconds. No credit is taken for the turbine-driven AFW pump.

The input model included details of both main steam lines from the SGs to the turbine control valve, including the MSSV inlet piping connected to the main steam lines. For this small area break, the primary and secondary pressure remained coupled at the MSSV actuation pressure setpoint.

Important system parameters and initial conditions used in the analysis are given in Table 3-1. The HHSI flow used in the analysis is shown in Table 3-2. The reactivity feedback tables (Table 3-3 and Table 3-4) are applicable for the HTP transition core designs.

The axial power shapes for this analysis is shown in Figure 3-4. Figure 3-5 compares the axial power shape at mid-node elevation for hot rod, hot assembly, inner and outer core used in the analysis.

Table 3-1 System Parameters and Initial Conditions

Parameter	Value
Reactor Power, MWt	3479[1]
Axial Power Shape	Figure 3-4
Radial Peaking Factor (includes uncertainty)	1.7056
Total Power Peaking Factor (includes uncertainty)	2.65
RCS Flow Rate, gpm	378400
Pressurizer Pressure, psia	2249.7
Core Inlet Coolant Temperature, °F	583
Accumulator Pressure, psia	614.7
Accumulator Fluid Temperature, °F	105
Accumulator Water Volume, ft^3	1050
SG Tube Plugging Level, %	15
SG Secondary Pressure, psia	877.4
MFW Temperature, °F	435.8
AFW Temperature, °F	120
Low SG Level AFAS Setpoint for harsh conditions, %NR Span	0
CCI, HHSI, and RHR Fluid Temperature, °F	110
Pressurizer Pressure – Low Reactor Trip Setpoint (RPS), psia	1859.70
Reactor Scram Delay Time on Low Pressurizer Pressure, sec	2
Scram CEA Holding Coil Release Delay Time, sec	0.5
SIAS Activation Setpoint Pressure, psia	1744.7
HHSI Pump Delay Time on SIAS, sec	37
MSSV lift pressures	1064.0 1077.0 1090.0 1103.0 1117.0

[1] Includes uncertainty.

Table 3-2 CCI, HHSI, & RHR Flows as Function of the Primary System Pressure

CCI Flow		HHSI Flow		RHR Flow	
Primary Pressure (psia)	Mass Flow Rate (lb_m/s)	Primary Pressure (psia)	Mass Flow Rate (lb_m/s)	Primary Pressure (psia)	Mass Flow Rate (lb_m/s)
2237	0.00	-	-		
1902	2.20	-	-		
1676	12.50	-	-		
1443	20.80	-	-		
1392	22.30	1392	0.00		
1220	27.30	1220	23.70		
1038	32.10	1038	37.80		
861	36.80	861	48.00		
675	41.20	675	56.10		
482	45.10	482	62.80		
291	49.30	291	69.30		
242	50.30	242	70.90		
194	51.40	194	72.50		
179	51.80	179	73.00	179	0.00
149	52.40	149	73.90	149	246.40
125	53.00	125	74.60	125	331.00
100	53.50	100	75.40	100	393.40
75	54.00	75	76.10	75	446.50
50	54.60	50	76.90	50	494.40
15	55.30	15	78.00	15	550.00

Table 3-3 Moderator Density Reactivity

Table 3-4 Doppler Reactivity

Figure 3-4 Axial Power Shape

Figure 3-5 Axial Power Distribution Comparison

4.0 Analytical Results

4.1 Results

The break spectrum should include breaks of varying diameter through the break that represents 10% of the cold leg flow area. Once the preliminary limiting break size (DLIM) has been identified, additional calculations should be run with diameters of DLIM±0.25 in, DLIM±0.50 in, and DLIM±0.75 in. The expectation is to

1. Cover the entire 0-10% small break area range; and
2. Demonstrate that the actual limiting break size has been identified within 0.25-inch diameter.

The current analysis takes into consideration breaks in the attached piping (accumulator line, charging line).

For the RCP trip sensitivity study, a hot leg break was analyzed in addition to the cold leg breaks. The break spectrum calculations were executed for breaks of 1.00, 2.00, 2.75, 3.00, 3.50, 4.00, 4.50, 4.75, 4.90, 4.95, 5.00, 5.05, 5.10, 5.12, 5.13, 5.14, 5.15, 5.20, 5.25, 5.50, 5.75, 6.00, 6.50, 7.00, 8.00, 8.50, 9.00, 9.75, 9.76, 9.77 and 9.78-inch diameter (the 9.76-inch diameter break corresponds to an area equal to 10% of the cold leg area). The results for the break spectrum calculations are presented in Table 4-1. Predicted event times are summarized in Table 4-2[1] for the break spectrum calculations.

The break spectrum calculations conservatively assumed RCP trip at reactor trip due to an assumed loss of offsite power at reactor trip. However, a delayed RCP trip can potentially produce more limiting results than tripping the RCPs at reactor trip. This delayed trip sensitivity is discussed in Section 4.3.

[1] The "*" in Table 4-2 indicates that the loop seals cleared after the accumulator injection occurred.

The key parameters for the limiting case (9.76-inch break) are shown in Figure 4-1 through Figure 4-24. The results of the analysis demonstrate the adequacy of the Emergency Core Cooling System (ECCS) to support the criteria given in 10 CFR 50.46(b) for SEQ1 and SEQ2 operating with AREVA supplied 17x17 HTP M5 clad fuel; as follows:

(1) *Peak cladding temperature:* The calculated limiting fuel element cladding temperature is 1470°F, which is less than the 2200° F limit criterion.

(2) *Maximum local cladding oxidation:* The calculated maximum local oxidation of the cladding is 0.17% which is less than the 17% limit of the criterion.

(3) *Maximum core-wide oxidation:* The calculated core-wide total oxidation is 0.0013%, which is less than the 1% limit of the criterion.

(4) *Coolable geometry:* The cladding remains amenable to cooling. None of the cases analyzed predicted hot rod rupture, hence no blockage is predicted to occur, which would degrade core cooling. Further, an M5 mechanical deformation analysis presented in Appendix F of Reference [4] is not changed by the fuel design changes. Both thermal and mechanical deformations of the fuel assemblies in the core have been assessed and the resultant deformations have been shown to maintain coolable core configurations. Therefore, the coolable geometry requirements of the criterion are met.

4.2 Discussion of Transient for 9.76-inch Break

The break opens at the initiation of the transient t=0 seconds and initiates a subcooled depressurization of the primary system. The Low Pressurizer Pressure Trip setpoint is reached within 0.4 seconds and within 3 seconds the reactor is tripped, the offsite power is lost, concurrent with the turbine trip, RCP trip, and Main Feedwater Pumps trip. After 7 seconds the Safety Injection Actuation Signal is issued on Low Pressurizer Pressure and at the same time the motor-driven Auxiliary Feedwater Pump actuation signal is issued.

The primary depressurization continues at a relatively high rate for the first approximately 20 seconds, after which it decreases significantly for approximately 60 seconds, while the pressure is relatively flat, and then remains relatively constant through the rest of the transient. The secondary side pressure increases in all four Steam Generators but not enough to reach the MSSV lift setpoint(s) and it stabilizes and remains almost constant through the rest of the transient (Figure 4-2). The single Auxiliary Feedwater Pump left running by the single-failure assumption starts delivering flow to the Steam Generators 2 and 4 around 66 seconds (Figure 4-9), but it is unable to increase the SG level above the lower boundary of the narrow range level span (Figure 4-11).

The continued depressurization causes the flashing of the liquid and the generation of steam bubbles that tend to accumulate to the high elevation points in the primary, e.g. the bend of the Steam Generator U-tubes. The liquid in the pressurizer drains rapidly and liquid in the core

starts to boil as the saturation pressure is reached and progressively the nodes in the hot assembly start voiding. As a result, the collapsed liquid level decreases rapidly at the beginning, but as the volume heat source decreases rapidly following the reactor trip, some of the steam bubbles formed initially collapse and the level briefly recovers around 10 seconds into the transient, after which it continues to decrease until the entire hot channel is completely void of liquid around approximately 50 seconds (Figure 4-17). At the same time, the hot rod mixture level remains to the top of the core (Figure 4-18) and remains there until around 45 seconds when it abruptly drops to the bottom. As downcomer level decreases (Figure 4-19), it reaches the point where the cold leg nozzles are uncovered and steam can start venting out to the cold legs (Figure 4-3 and Figure 4-4).

As the core is voided of liquid, the heat transfer conditions worsen (Figure 4-23) and reach the point where the hot rod is no longer properly cooled and starts the adiabatic heatup (Figure 4-24). Meanwhile, both Centrifugal Charging Injection and High Head Safety Injection start delivering fluid to the cold legs at 46 seconds (Figure 4-12), but the liquid injected cannot make up for the large amount of fluid lost to the break up to this point (Figure 4-14) and the cladding temperature keeps rising. As break uncovers around 86 seconds, the amount of liquid lost out the break decreases sharply and steam is vented out the break (Figure 4-3) releasing energy from the primary. The primary system pressure starts decreasing again and the loop seals begin to clear between 90 and 102 seconds (Figure 4-5). Seconds later, the accumulator injection begins (Figure 4-13) and the large amount of liquid added to the primary turns the transient around, liquid levels start rising, first in the downcomer (Figure 4-19), and later into the core (Figure 4-17). Cladding temperature reaches its peak of 1469.3 °F at 143 seconds, and then, as cooling conditions in the core starts improving due to the increased presence of liquid, it drops as core quenches.

4.3 RCP Trip Sensitivity Study

A Reactor Coolant Pump trip sensitivity study has been performed as required in Reference [3]. For plants that do not have an automatic RCP trip, a delayed RCP trip can potentially result in a more limiting condition than tripping the RCPs at reactor trip. Continued operation of the RCPs can result in earlier loop seal clearing with associated two-phase flow out the break, which would results in less inventory loss out the break early in the transient. It is possible that tripping the pumps at minimum RCS inventory could cause a collapse of core voids, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher PCT.

The TVA Emergency Operating Procedure No. E-1, "Loss of Reactor or Secondary Coolant" requires the RCPs to be tripped when:

(1) Primary system pressure is less than 1250 psig and at least one charging or one safety injection pump is running, or, when

(2) A Phase B isolation signal has been generated on high-high containment pressure (i.e. containment pressure ≥ 2.9 psig).

The nominal setpoint of 1250 psig is based on an instrument uncertainty of ±111 psi, thus the pump trip could occur at actual primary system pressures as high as 1361 psig or as low as 1139 psig.

The RCP trip sensitivity study will include those breaks for which the break uncovery time from the break spectrum case is less than the RCP trip time determined based on the EOPs. From the break spectrum transient study, this means breaks of 6.5-inch diameter and higher, i.e. 6.50, 7.00, 8.00, 9.00, 9.75, 9.76, 9.77 and 9.78-inch diameter.

4.3.1 Cold Leg Breaks RCP Trip Sensitivity Study

For the cold leg breaks RCP trip sensitivity study the selected breaks of 6.50, 7.00, 8.00, 9.00, 9.75, 9.76, 9.77 and 9.78-inch diameter are being re-analyzed with a delayed RCP trip. For these cases the RCP trip time based on EOPs is the same, 166.0 seconds.

A summary of results of the cold leg break RCP trip sensitivity runs are presented in Table 4-3. The results of the cold leg breaks RCP trip sensitivity study do not impact the limiting PCT of 1470°F from the SBLOCA break spectrum study.

4.3.2 Hot Leg Breaks RCP Trip Sensitivity Study

For the hot leg breaks RCP trip sensitivity study the same selected breaks of 6.50, 7.00, 8.00, 9.00, 9.75, 9.76, 9.77 and 9.78 inch diameter are being re-analyzed with a delayed RCP trip and a hot leg break. For these cases the RCP trip time based on EOPs is the same, 166.0 seconds.

A summary of results of the cold leg break RCP trip sensitivity runs are presented in Table 4-4. The results of the hot leg breaks RCP trip sensitivity study do not impact the limiting PCT of 1470°F from the SBLOCA break spectrum study.

4.4 Attached Piping LOCA Cases

Sensitivity calculations for breaks in the attached piping have also been performed. These consist of a double ended guillotine (DEG) break of the accumulator line and a DEG break of the charging line.

For the DEG break of the accumulator, the calculated PCT is 1386.1 °F. The results of the charging line break analysis showed that there is no heat up occurred, which is consistent with the conclusion of the Sequoyah FSAR analysis of this break. The results of these two breaks do not challenge the limiting PCT of 1470°F from the SBLOCA break spectrum study.

4.5 Results Comparison against AOR

The results from the current analysis break spectrum were compared to the analysis of record (AOR) as provided in Figure 4-25. In the AOR, the limiting break was the 2.75-inch diameter break, which resulted in a PCT of 1403 °F. In the current analysis, the 2.75-inch diameter break

resulted in a PCT of 1027 °F. The difference in results can be attributed to several factors of which the most important are enumerated below:

The most important difference comes from the axial power shape used by each of the analyses. The AOR uses a more conservative axial shape which puts more power in the top half of the core. As a result, the fuel nodes at the top of the core experience a faster liquid dry-out, less cooling and a prolonged adiabatic heatup which drives the PCT higher. Also, the $F_{\Delta H}$ values used are different – the AOR uses a very conservative value of 1.8928 (due to an axial peak fixed at 1.4 by the BAW-10168 (P)(A) methodology), while the current analysis complies with the EM requirements of using the Technical Specifications limit of 1.7056.

There are several important modeling differences that account for the different thermal-hydraulic behavior. The AOR model uses a lumped triple loop for modeling the intact loops, while the current analysis uses distinct loop models for each loop. This accounts for major differences in the system response with respect to steam venting and also plays a role in the loop seal clearing. Also, the AOR model for the break is very different using a "dummy" node connected to the bottom of the pipe, and a valve connected to this "dummy" node, while the S-RELAP5 model uses a trip valve directly connected to the centerline cold leg piping. This modeling difference is responsible for the differences in the vapor void fraction at the break, thus affecting the break uncovery time, the fluid mass lost out the break and the steam and energy venting out the break. This modeling difference plays a role in the significant difference in the primary pressure, which is lower in the current analysis from early into the transient and almost to the end, and, as a result, determines a higher pumped SI flow being delivered in the current analysis vs. the AOR. The differences in the SI flow being delivered and the fluid lost out the break account for the difference in reactor vessel mass and consequently the lower collapsed core liquid level in the AOR, the prolonged and deeper core uncovery and thus the higher PCT result in the AOR.

One other important difference is that the AOR model has been built with the original Steam Generators, while the current analysis uses the replacement Steam Generators. While the replacement Steam Generators are functionally identical to the original ones, there are differences in design that come into play and affect the transient behavior, with consequences on the loop seal clearing and steam venting.

Table 4-1 Summary of SBLOCA Break Spectrum Results

Break diameter (in)	1.00	2.00	2.20	2.40	2.60	2.75	2.90	3.00
Peak Clad Temperature (°F)	690.2	690.2	976.6	986.9	960.7	1027.0	884.5	1069.1
Time of PCT (sec)	1	1	1369	2856	2487	2004	2011	1832
PCT Elevation (ft)	10.125	10.125	10.625	10.625	11.125	10.625	11.125	11.125
Time of Rupture (sec)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Core Wide Oxidation (%)	0.0002	0.0002	0.0002	0.0004	0.0003	0.0004	0.0001	0.0006
Local Maximum Oxidation (%)	0.0004	0.0002	0.0036	0.0163	0.0158	0.0238	0.0055	0.0334

Break diameter (in)	3.25	3.50	3.75	4.00	4.25	4.50	4.75	4.80
Peak Clad Temperature (°F)	945.6	947.2	957.2	758.2	816.2	844.4	1246.9	852.0
Time of PCT (sec)	1580	1320	1035	387	340	294	547	255
PCT Elevation (ft)	11.125	11.125	11.125	10.625	10.625	10.625	11.125	10.625
Time of Rupture (sec)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Core Wide Oxidation (%)	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0009	0.0001
Local Maximum Oxidation (%)	0.0068	0.0048	0.0061	0.0002	0.0005	0.0007	0.0686	0.0008

Break diameter (in)	4.85	4.90	4.95	5.00	5.50	6.00	6.50	7.00
Peak Clad Temperature (°F)	868.8	889.3	894.5	904.1	1040.5	1056.1	1095.8	1214.4
Time of PCT (sec)	252	249	244	240	212	180	164	182
PCT Elevation (ft)	10.625	10.625	10.625	10.625	10.625	10.625	10.625	10.625
Time of Rupture (sec)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Core Wide Oxidation (%)	0.0001	0.0001	0.0001	0.0001	0.0002	0.0002	0.0002	0.0005
Local Maximum Oxidation (%)	0.0010	0.0013	0.0013	0.0015	0.0082	0.0107	0.0150	0.0474

Break diameter (in)	8.00	9.00	9.75	9.76	9.77	9.78
Peak Clad Temperature (°F)	1367.6	1341.5	1385.7	**1469.3**	1325.6	1379.4
Time of PCT (sec)	175	149	132	**143**	139	136
PCT Elevation (ft)	10.625	10.625	10.125	**10.625**	10.125	10.625
Time of Rupture (sec)	N/A	N/A	N/A	**N/A**	N/A	N/A
Core Wide Oxidation (%)	0.0009	0.0004	0.0007	**0.0013**	0.0003	0.0006
Local Maximum Oxidation (%)	0.1082	0.0812	0.1145	**0.1659**	0.0848	0.1184

Table 4-2 Sequence of Events for Break Spectrum Calculations, sec

Break diameter (in)	Break open (Trip 5)	Low PZR Pressure Trip (Trip 14)	RX, LOOP, RCPs, MFWP and Turbine Trip (Trip 1100)	Low PZR Pressure SIAS Trip (Trip 51)	HHSI Flow Begins	Motor driven AFWP on (Trip 1912)	Loop seal 1 clears	Loop seal 2 clears	Loop seal 3 clears	Loop seal 4 clears	Break uncovers	Accumulator injection begins	Minimum RV mass occurs	Hot Rod rupture occurs	PCT occurs	Time of core uncovery	Non-condensable gas at the break
1.00	0	225	227	240	684	231	-	-	-	-	-	-	4040	-	1	-	-
2.00	0	43	46	57	176	50	-	1370	-	-	1390	-	3993	-	1	-	-
2.20	0	34	36	47	152	40	-	1300	-	-	1138	-	3506	-	1369	1210	-
2.40	0	27	29	39	132	33	-	1038	-	-	936	-	2648	-	2856	980	-
2.60	0	21	23	33	116	27	-	872	-	-	880	3450	2268	-	2487	704	-
2.75	0	18	20	30	106	24	-	746	-	-	764	2690	1982	-	2004	738	-
2.90	0	15	17	27	64	21	-	694	-	-	670	2312	1938	-	2011	680	-
3.00	0	13	16	25	64	19	-	638	-	-	648	1878	1664	-	1832	624	-
3.25	0	10	13	22	60	16	-	566	-	-	550	1574	1578	-	1580	544	-
3.50	0	8	10	19	56	14	-	486	-	-	498	1262	1252	-	1320	276	-
3.75	0	6	9	17	54	12	-	412	-	-	422	1020	1026	-	1035	382	-
4.00	0	5	7	15	54	11	386	-	384	-	394	914	922	-	387	336	-
4.25	0	4	7	14	52	10	336	336	334	-	346	762	814	-	340	176	-
4.50	0	0.8	3	12	50	7	288	290	290	-	300	652	700	-	294	168	-
4.75	0	0.7	3	12	50	7	*	-	250	*	264	514	518	-	547	130	-
4.80	0	0.7	3	11	50	7	250	250	250	-	260	550	602	-	255	128	-
4.85	0	0.7	3	11	50	7	*	246	246	-	256	532	540	-	252	126	-
4.90	0	0.7	3	11	50	7	*	244	244	*	252	520	534	-	249	126	-
4.95	0	0.7	3	11	50	7	240	240	240	-	248	508	554	-	244	124	-
5.00	0	0.7	3	11	48	7	236	234	234	-	248	492	538	-	240	122	-
5.50	0	0.6	3	10	48	7	198	202	200	*	212	392	428	-	212	106	-
6.00	0	0.6	3	9	48	7	168	166	168	*	178	328	338	-	180	92	-
6.50	0	0.6	3	9	46	7	146	146	146	152	160	276	298	-	164	82	-
7.00	0	0.5	3	8	46	7	130	128	128	142	138	234	260	-	182	78	-
8.00	0	0.5	3	8	46	7	126	126	126	136	132	166	118	-	175	56	308
9.00	0	0.5	3	7	46	7	98	98	98	108	90	130	142	-	149	46	244
9.75	0	0.4	3	7	46	7	92	92	92	104	86	110	120	-	132	40	214
9.76	**0**	**0.4**	**3**	**7**	**46**	**7**	**90**	**92**	**92**	**102**	**86**	**110**	**130**	**-**	**143**	**40**	**220**
9.77	0	0.4	3	7	46	7	90	98	98	106	92	112	114	-	139	40	208
9.78	0	0.4	3	7	46	7	98	98	98	106	92	112	114	-	136	40	210

Table 4-3 Summary of Results for the Cold Leg Break RCP Trip Sensitivity

Break diameter (in)	6.50	7.00	8.00	9.00	9.75	9.76	9.77	9.78
Peak Clad Temperature (°F)	984.2	895.2	872.7	690.2	690.2	690.2	690.2	690.2
Time of PCT (sec)	244	247	209	1	1	1	1	1
PCT Elevation (ft)	10.625	10.625	10.625	10.125	10.125	10.125	10.125	10.125
Time of Rupture (sec)	-	-	-	-	-	-	-	-
Core Wide Oxidation (%)	0.0001	0.0001	0.0001	0.0000	0.0000	0.0000	0.0000	0.0000
Local Maximum Oxidation (%)	0.0048	0.0018	0.0011	0.0000	0.0000	0.0000	0.0000	0.0000

Table 4-4 Summary of Results for the Hot Leg Break RCP Trip Sensitivity Study

Break diameter (in)	6.50	7.00	8.00	9.00	9.75	9.76	9.77	9.78
Peak Clad Temperature (°F)	807.1	777.7	878.7	860.3	690.2	690.2	720.6	781.1
Time of PCT (sec)	349	210	216	197	1	1	167	163
PCT Elevation (ft)	10.625	10.625	10.625	10.125	10.125	10.125	10.125	10.125
Time of Rupture (sec)	-	-	-	-	-	-	-	-
Core Wide Oxidation (%)	0.0000	0.0000	0.0001	0.0000	0.0000	0.0000	0.0000	0.0000
Local Maximum Oxidation (%)	0.0004	0.0003	0.0011	0.0008	0.0001	0.0001	0.0001	0.0003

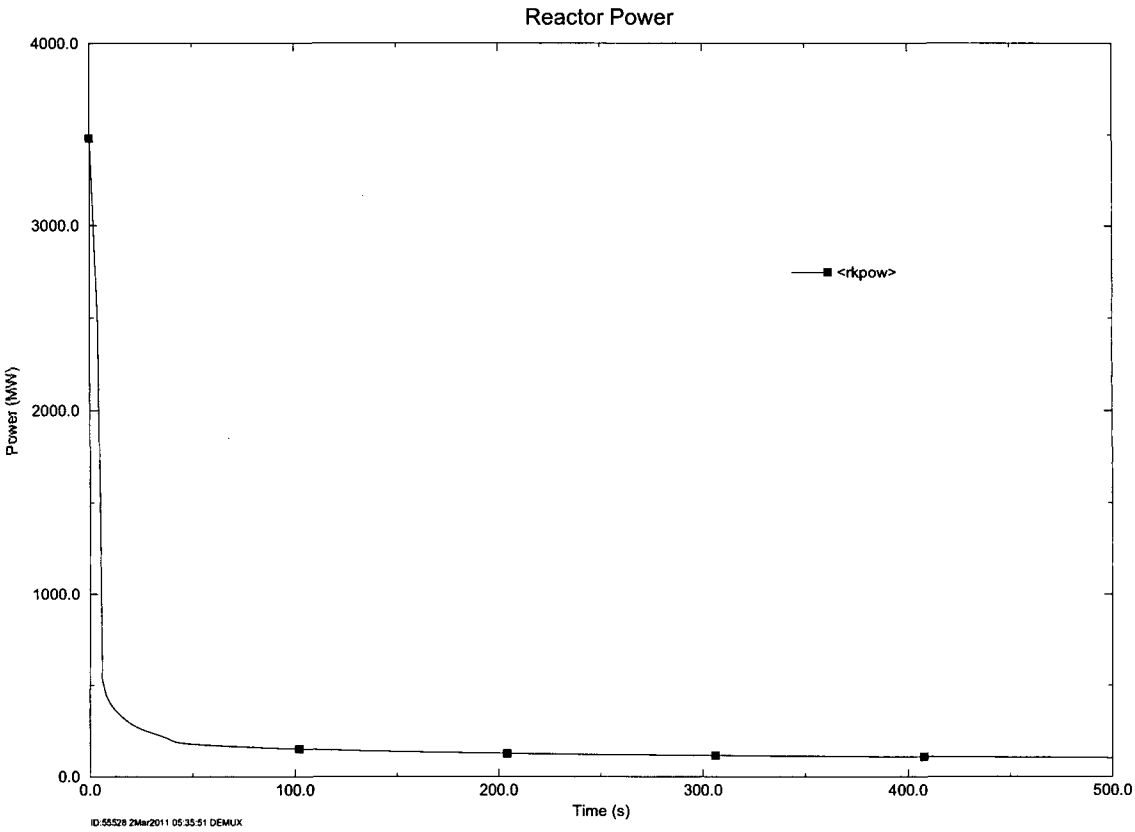

Figure 4-1 Reactor Power – 9.76-inch Break

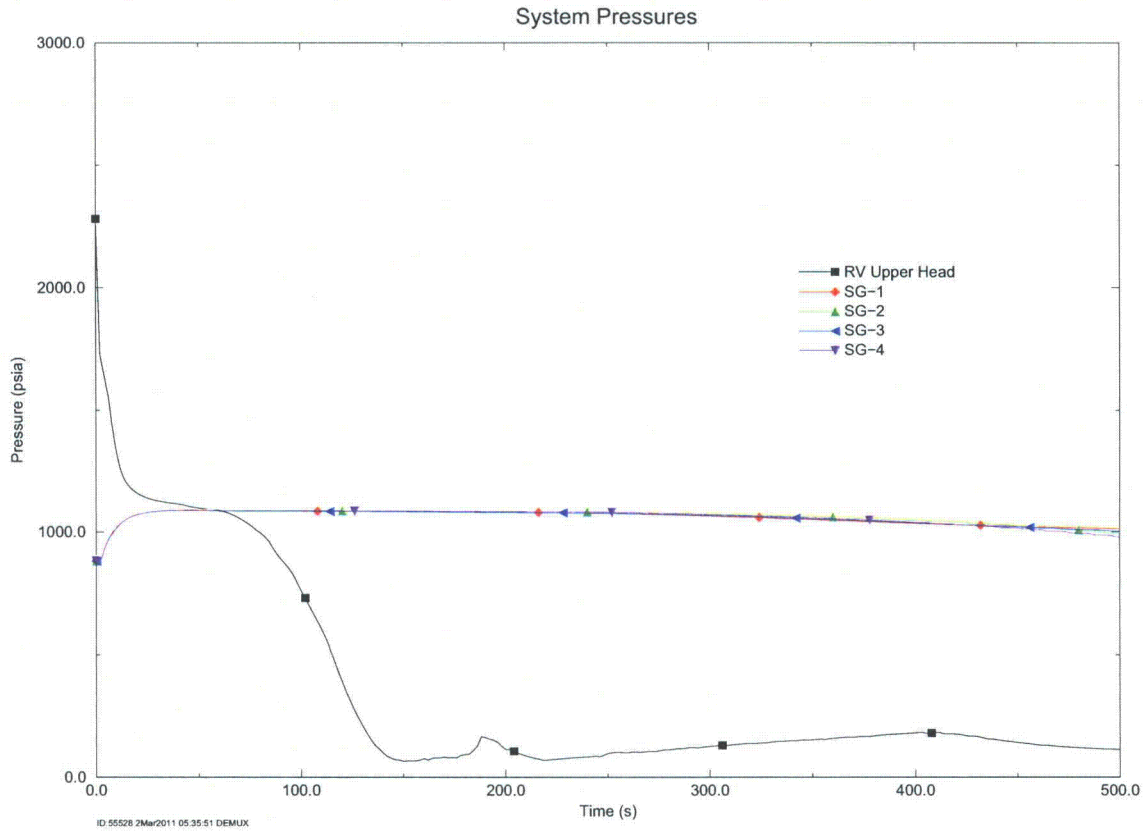

Figure 4-2 Primary and Secondary System Pressures – 9.76-inch Break

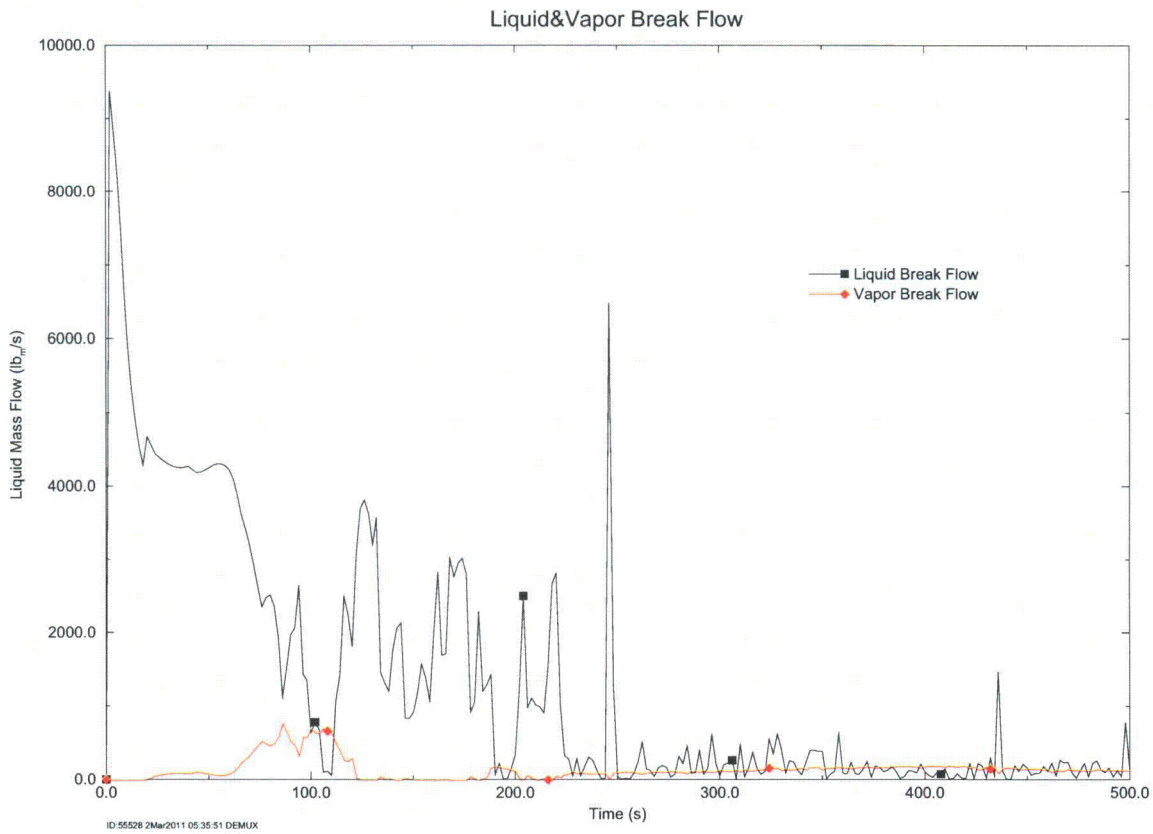

Figure 4-3 Liquid and Vapor Break Flow – 9.76-inch Break

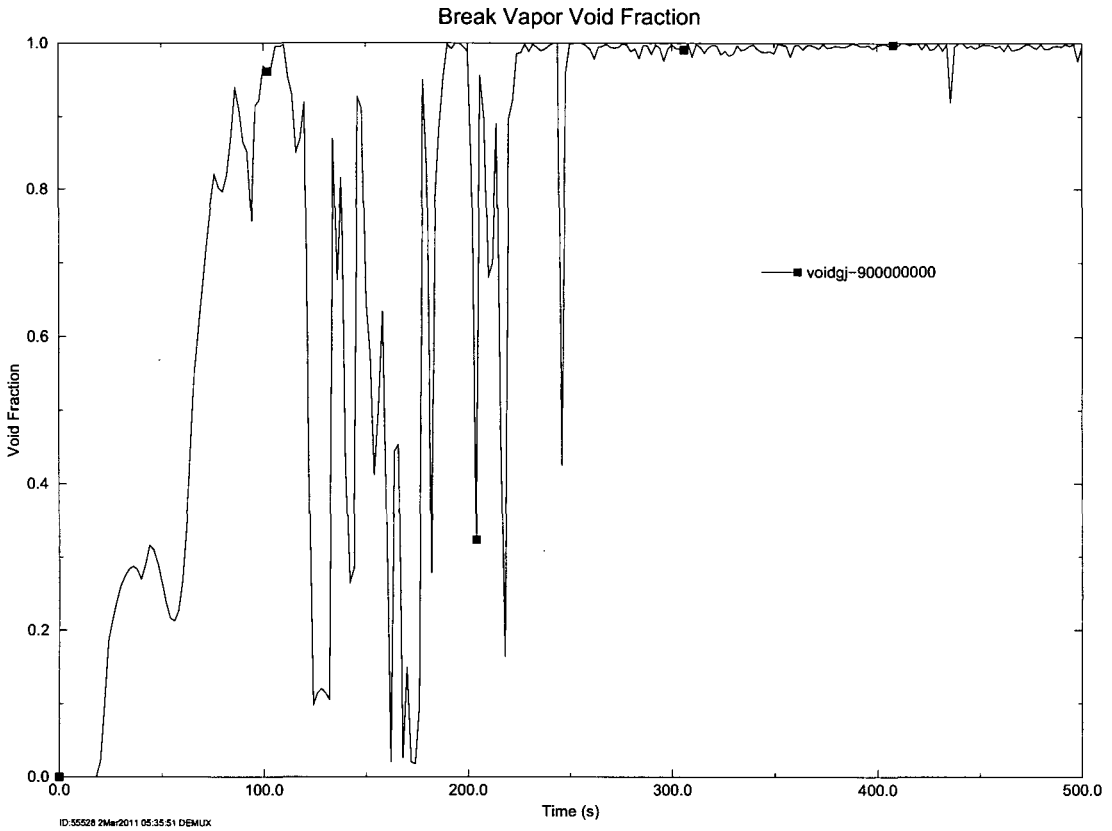

Figure 4-4 Vapor Void Fraction at the Break – 9.76 in Break

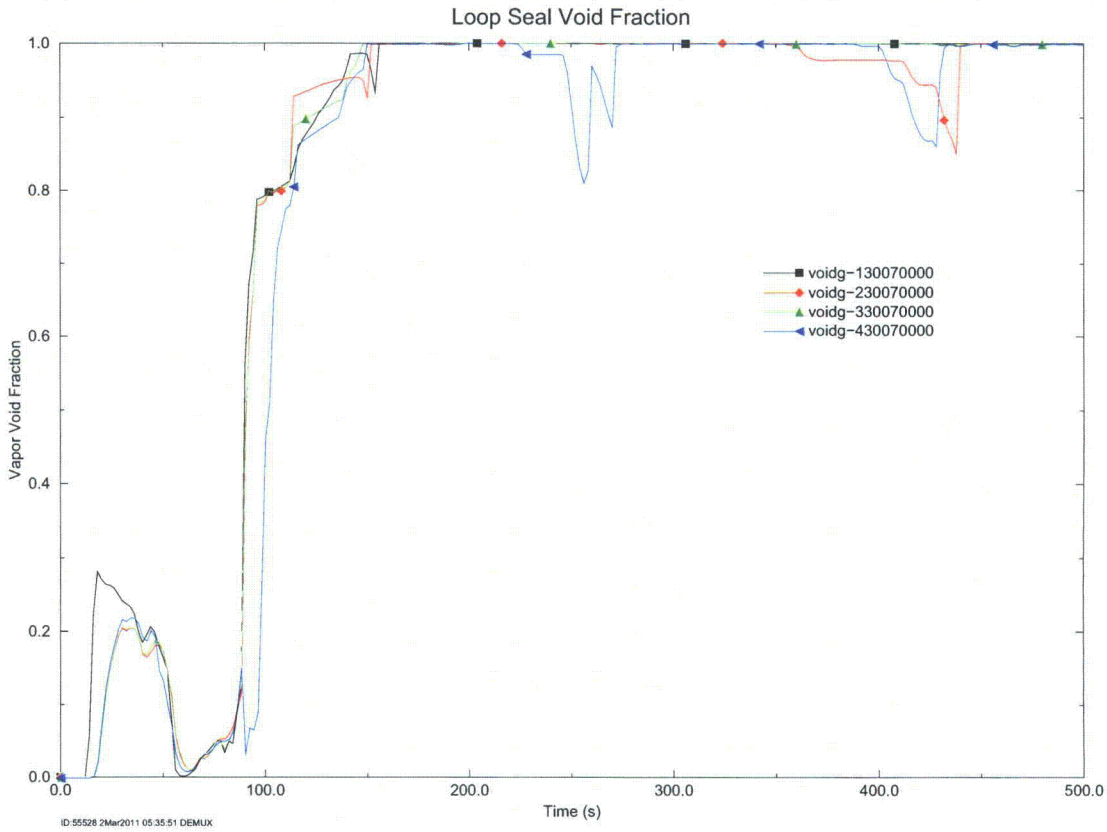

Figure 4-5 Loop Seal Void Fraction – 9.76-inch Break

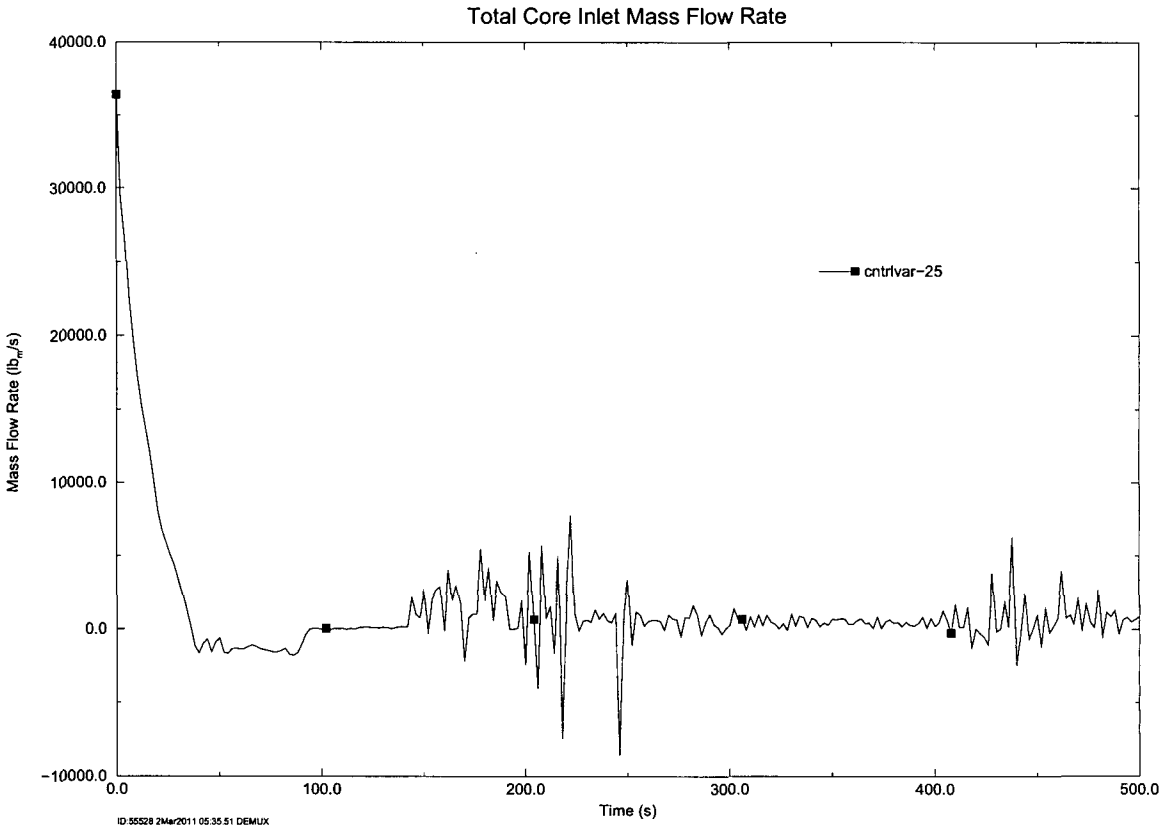

Figure 4-6 Total Core Inlet Mass Flow Rate – 9.76-inch Break

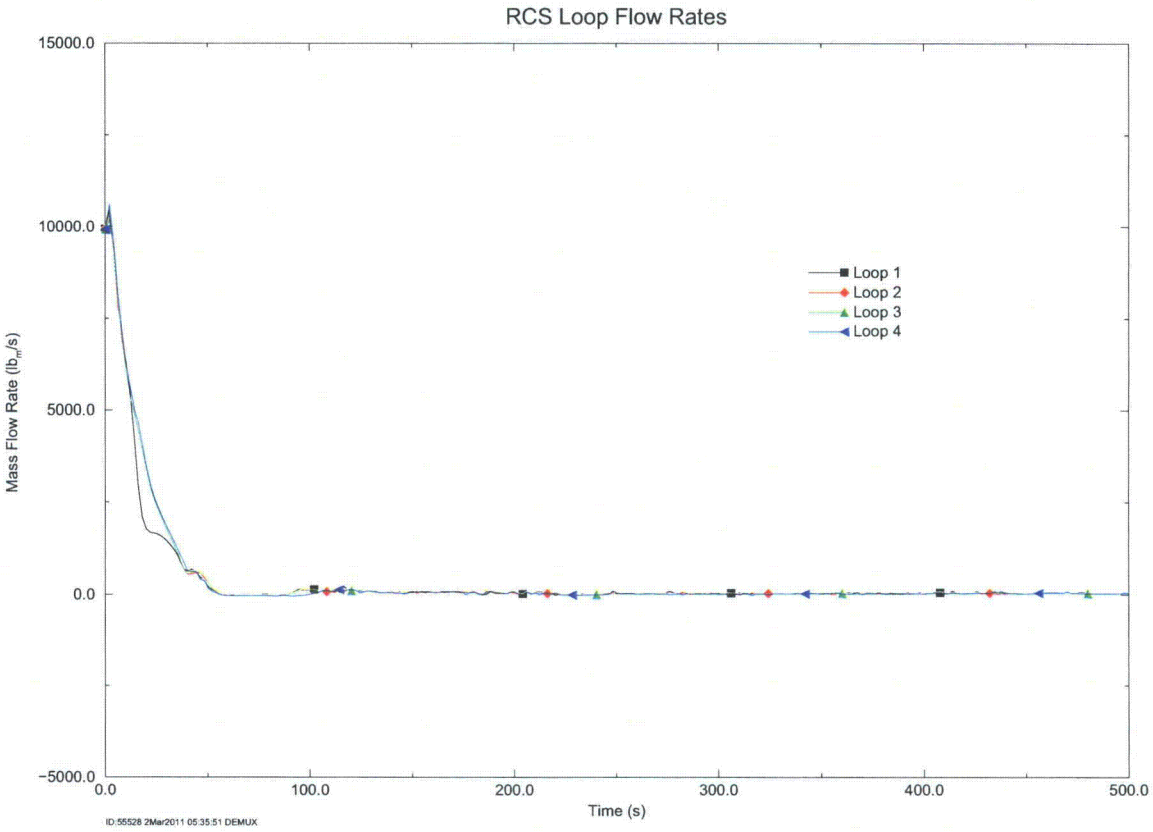

Figure 4-7 RCS Loop Flow Rate – 9.76-inch Break

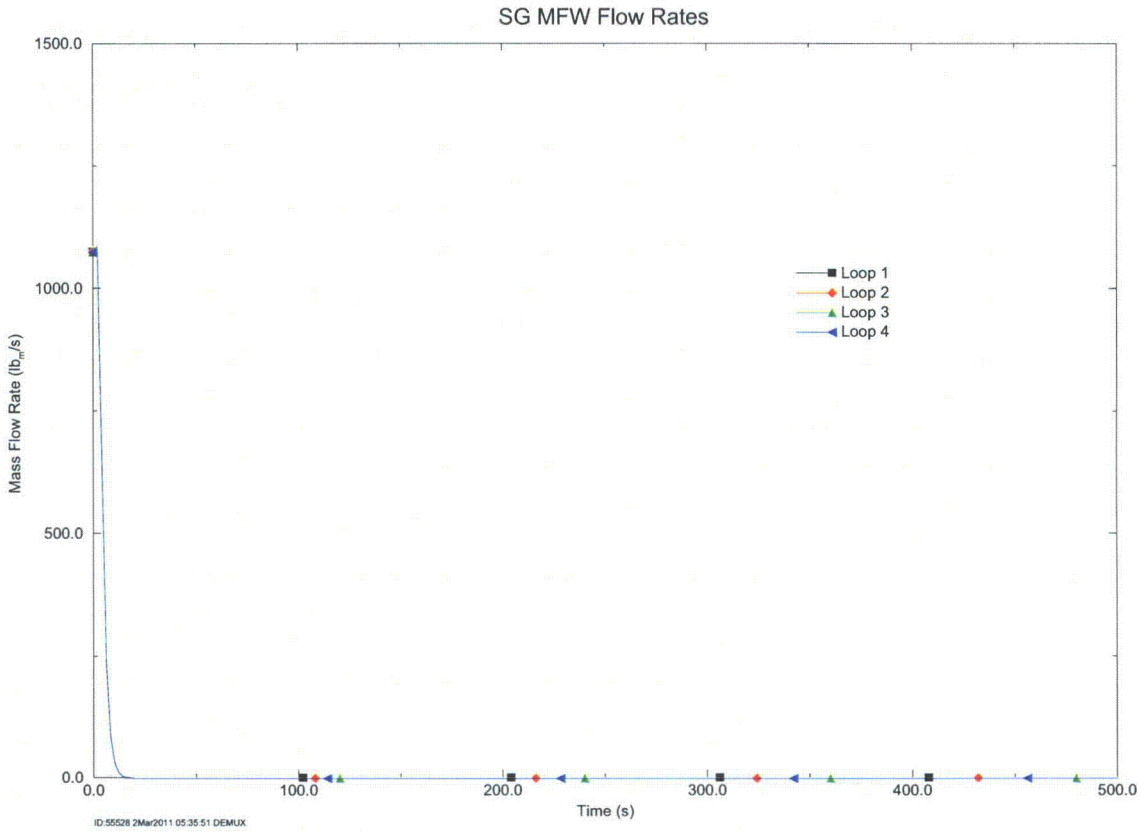

Figure 4-8 Steam Generator Main Feedwater Flow Rates – 9.76-inch Break

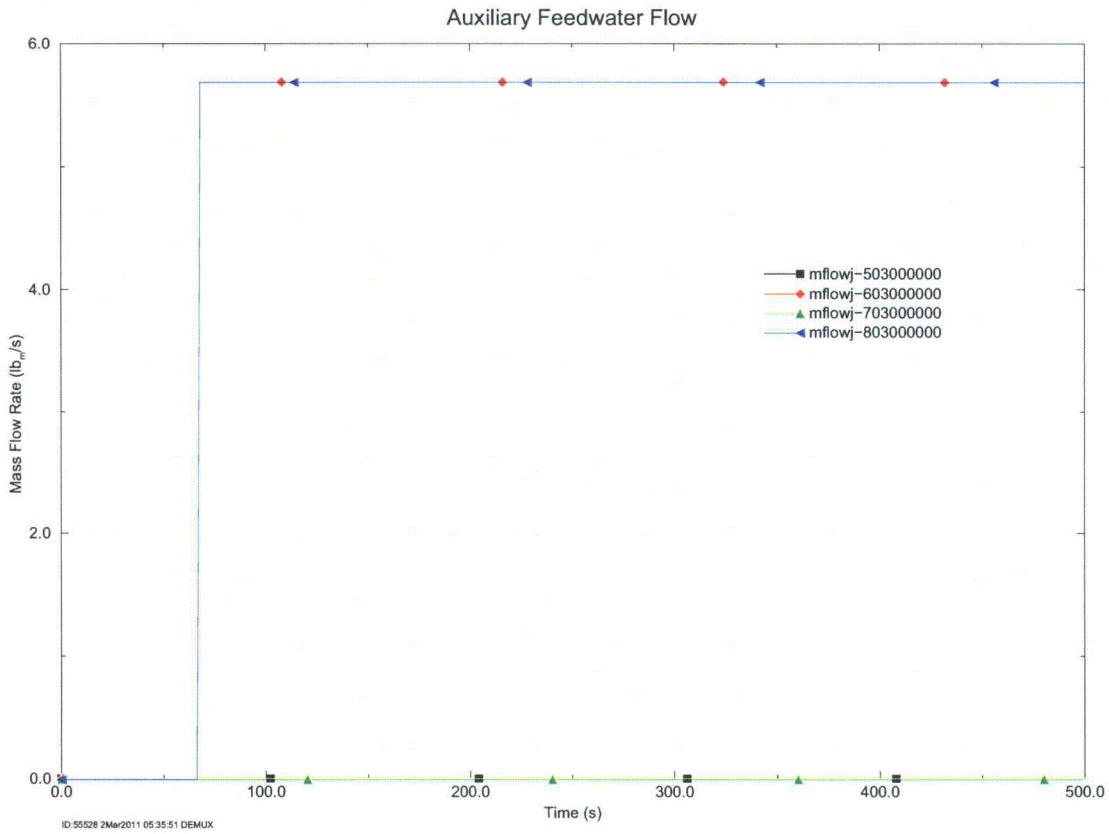

Figure 4-9 Auxiliary Feedwater Flow – 9.76 in Break

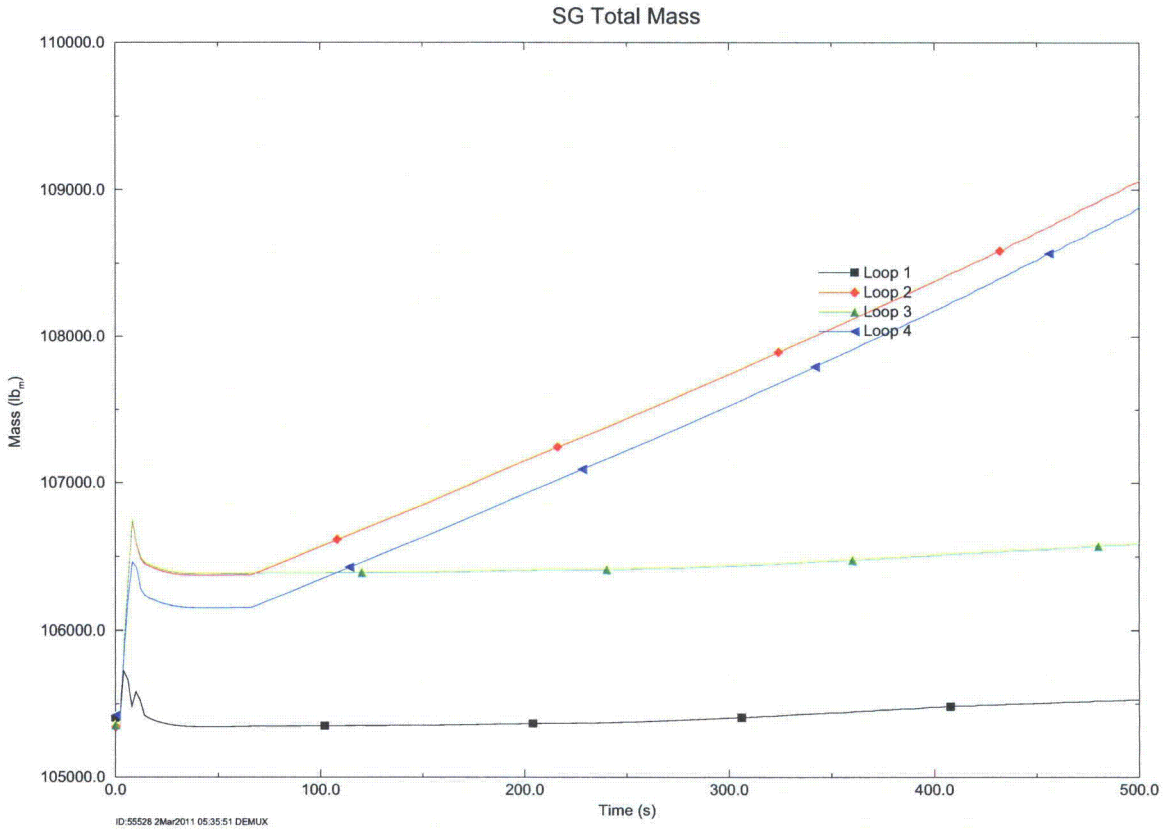

Figure 4-10 Steam Generator Total Mass – 9.76-inch Break

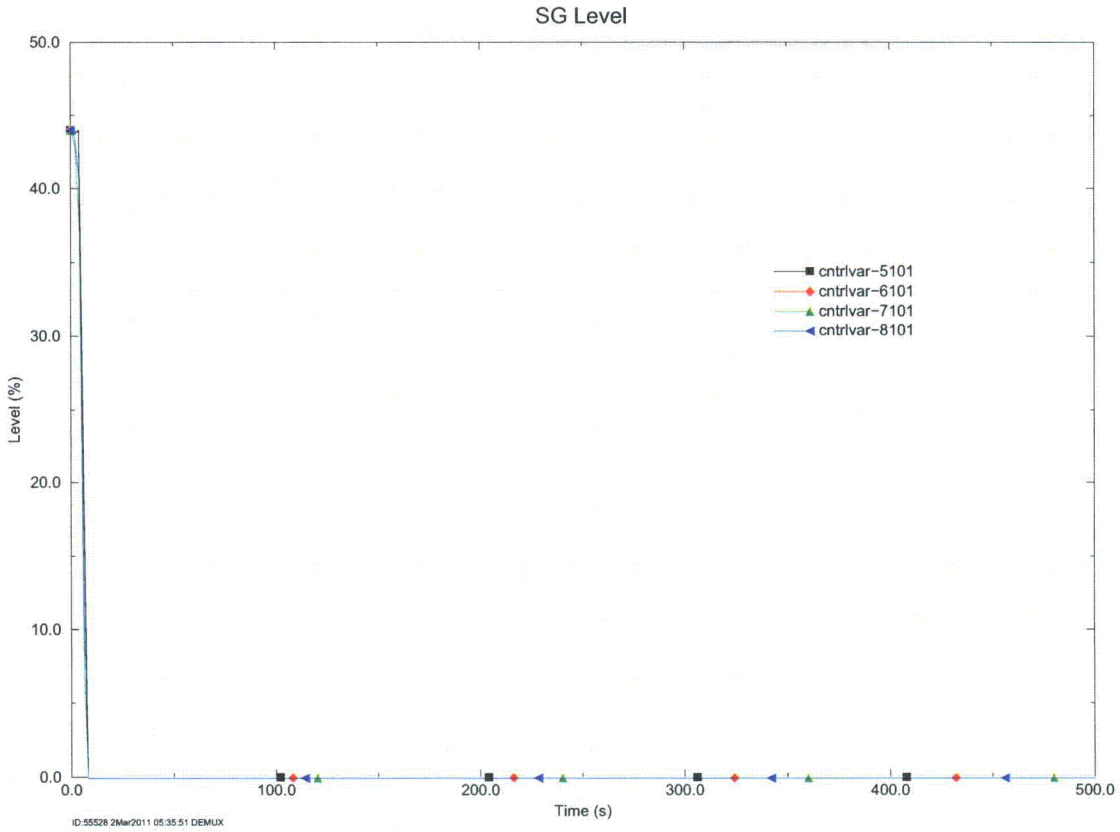

Figure 4-11 Steam Generator Level – 9.76 in Break

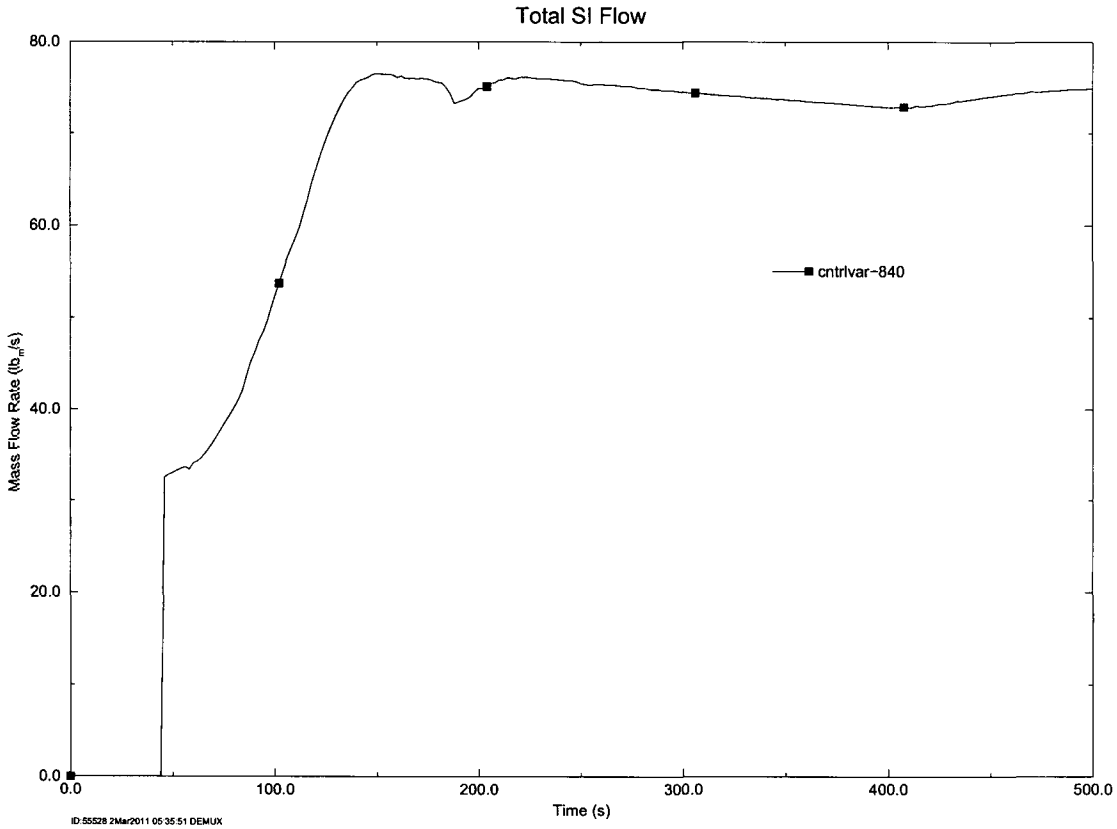

Figure 4-12 Total HHSI Mass Flow – 9.76-inch Break

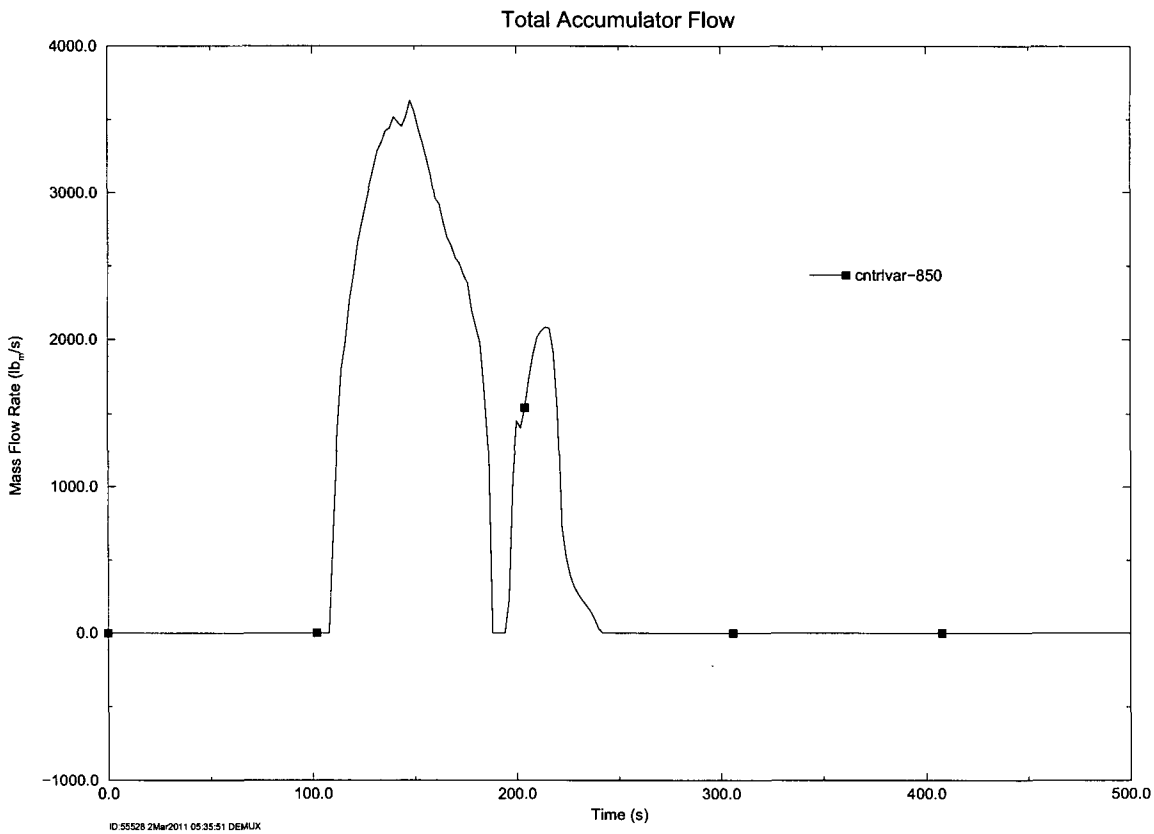

Figure 4-13 Total Accumulator Flow – 9.76-inch Break

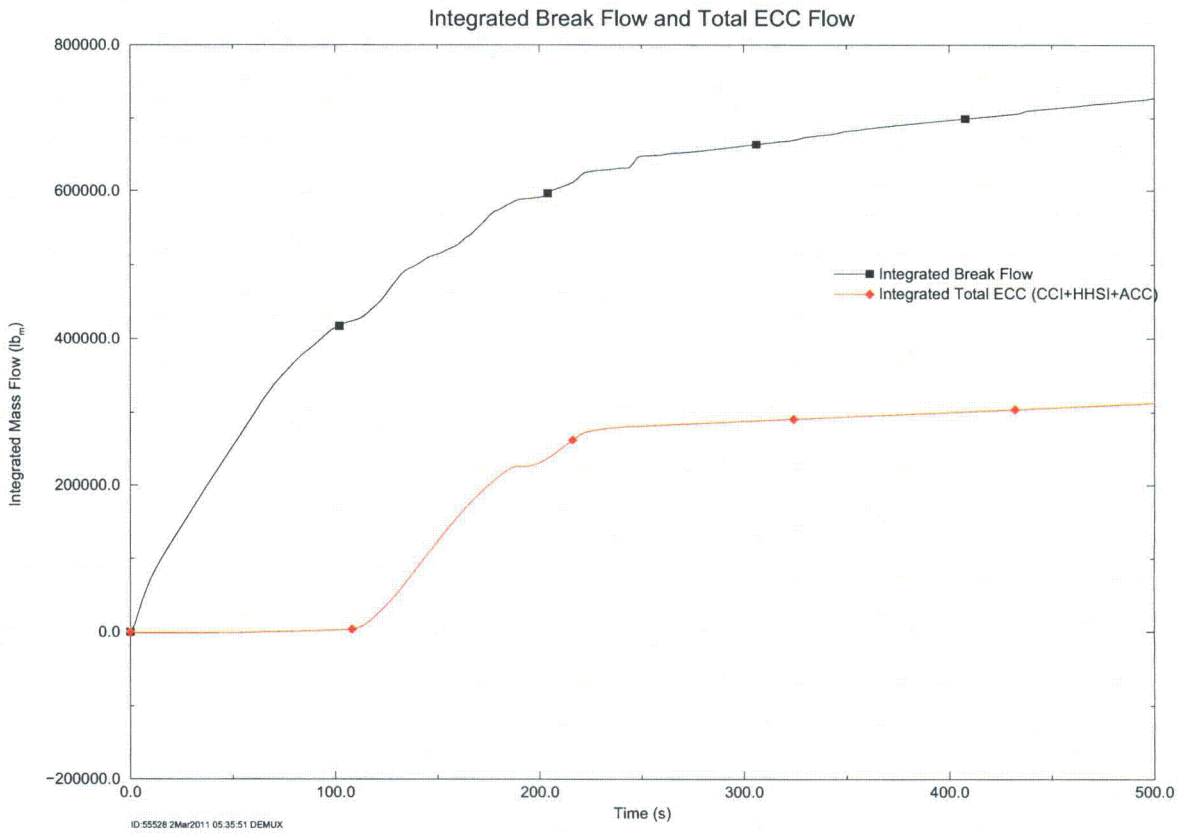

Figure 4-14 Integrated Break Flow and ECC Flow – 9.76 in Break

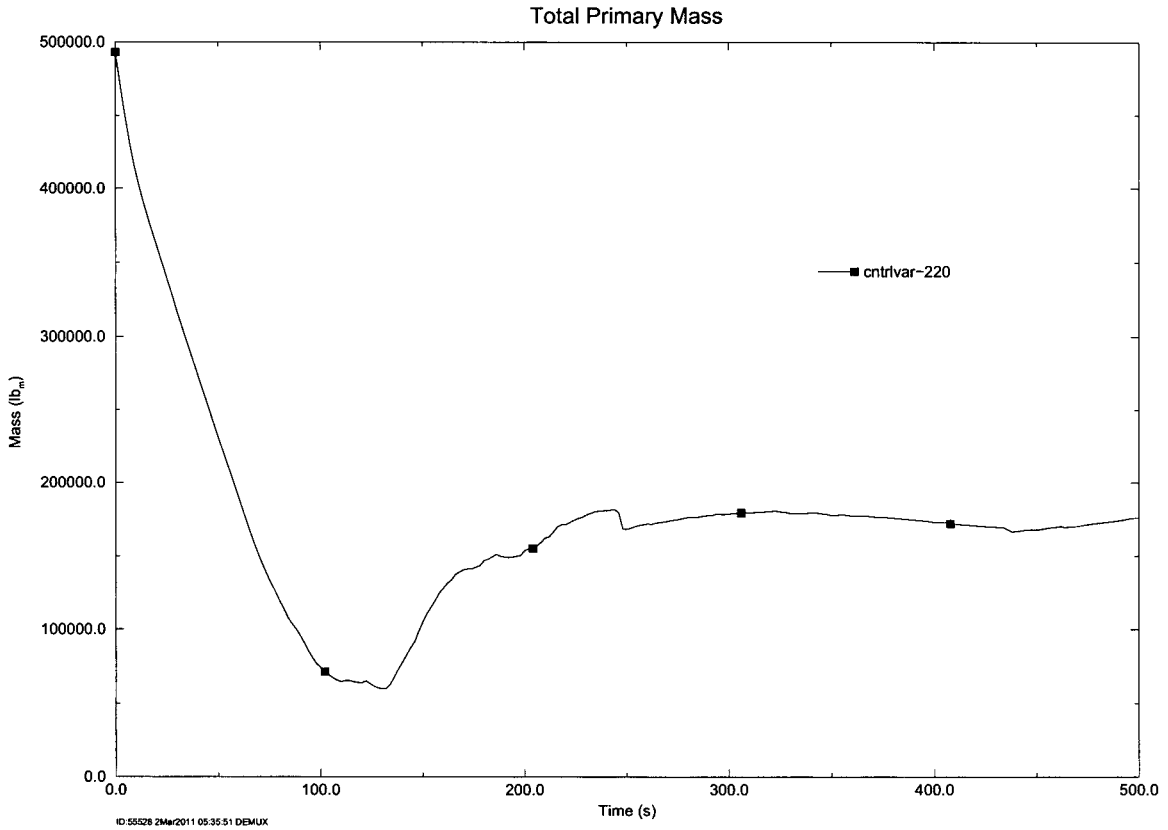

Figure 4-15 Total Primary Mass – 9.76-inch Break

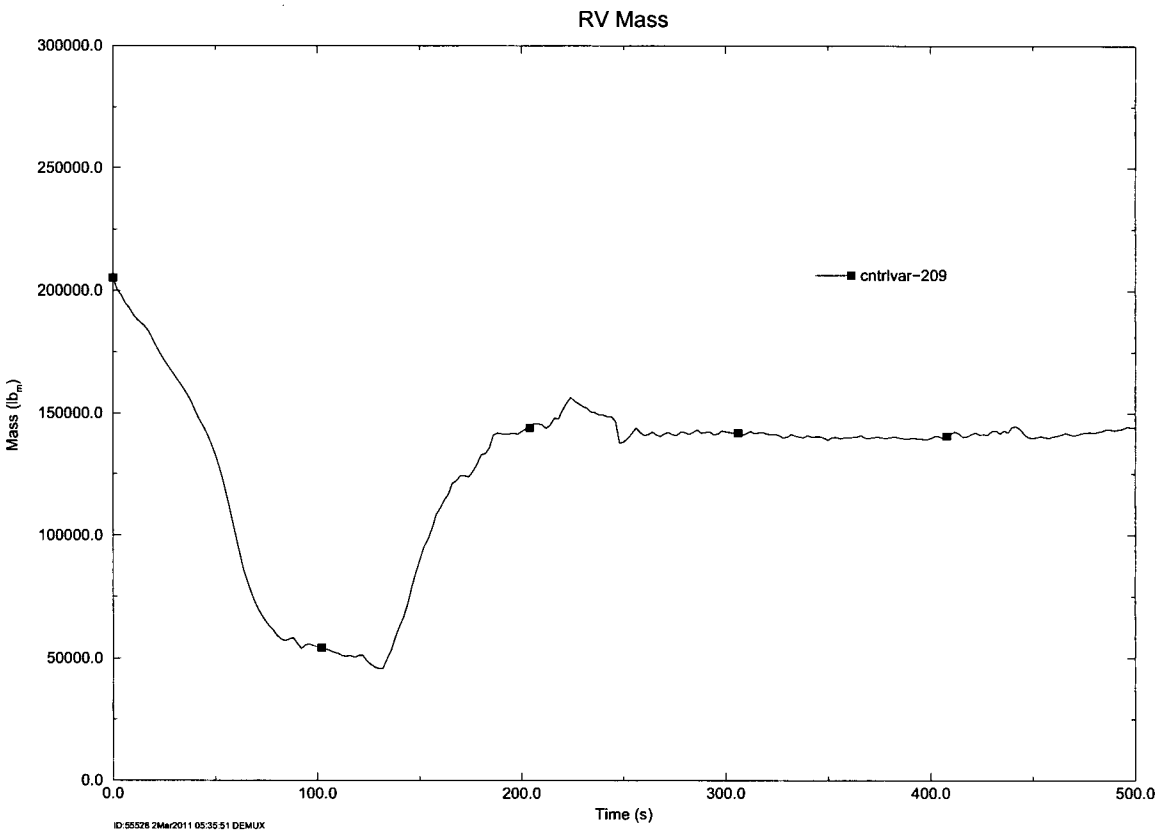

Figure 4-16 Reactor Vessel Mass – 9.76-inch Break

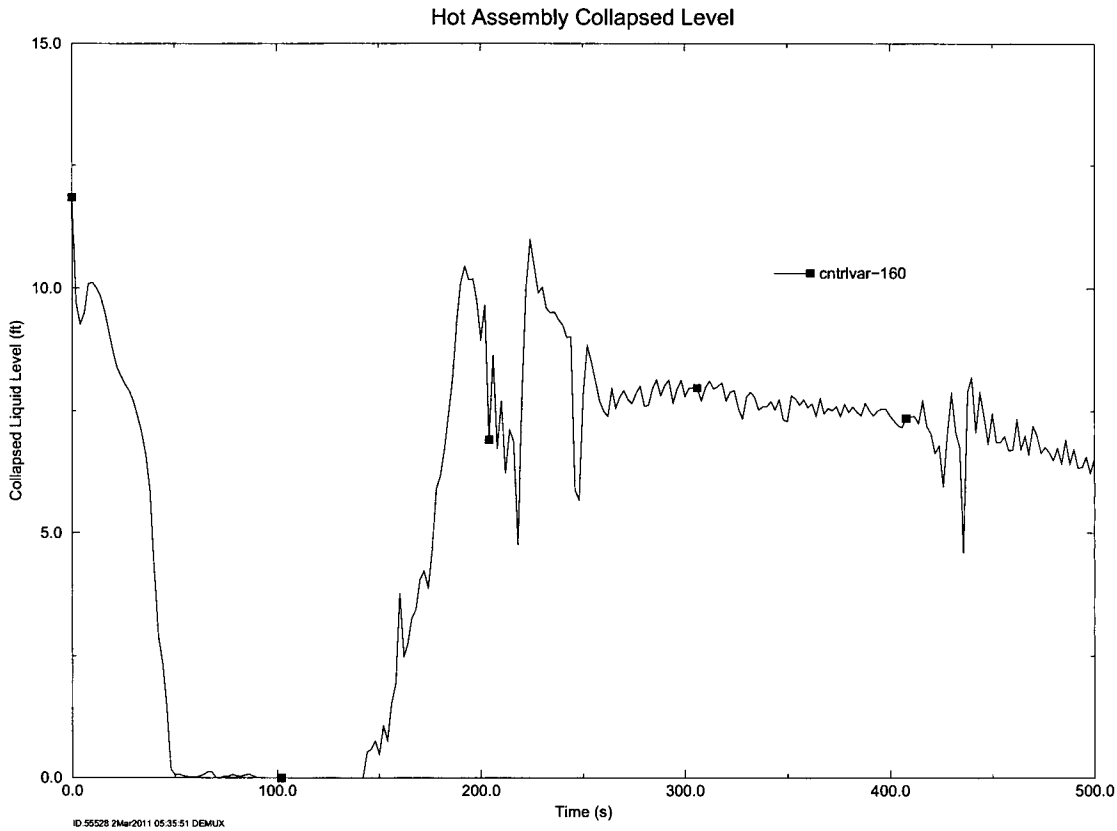

Figure 4-17 Hot Assembly Collapsed Liquid Level – 9.76-inch Break

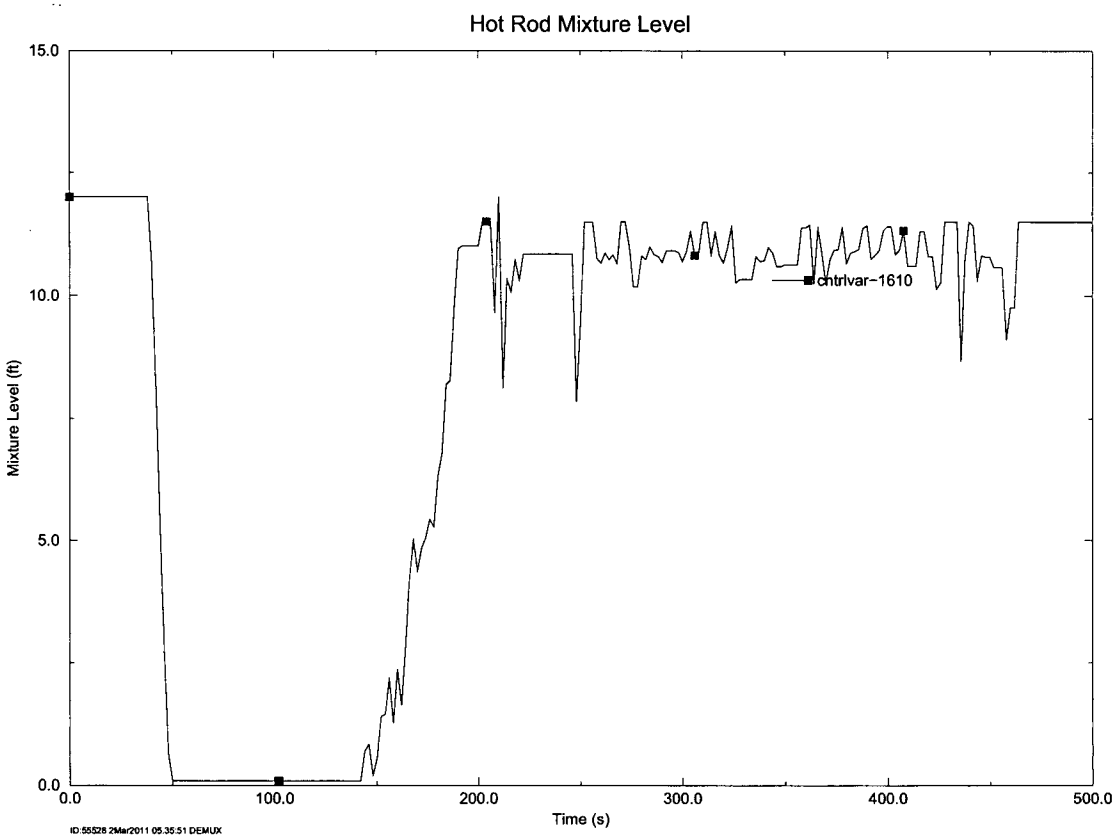

Figure 4-18 Hot Rod Mixture Level – 9.76 in Break

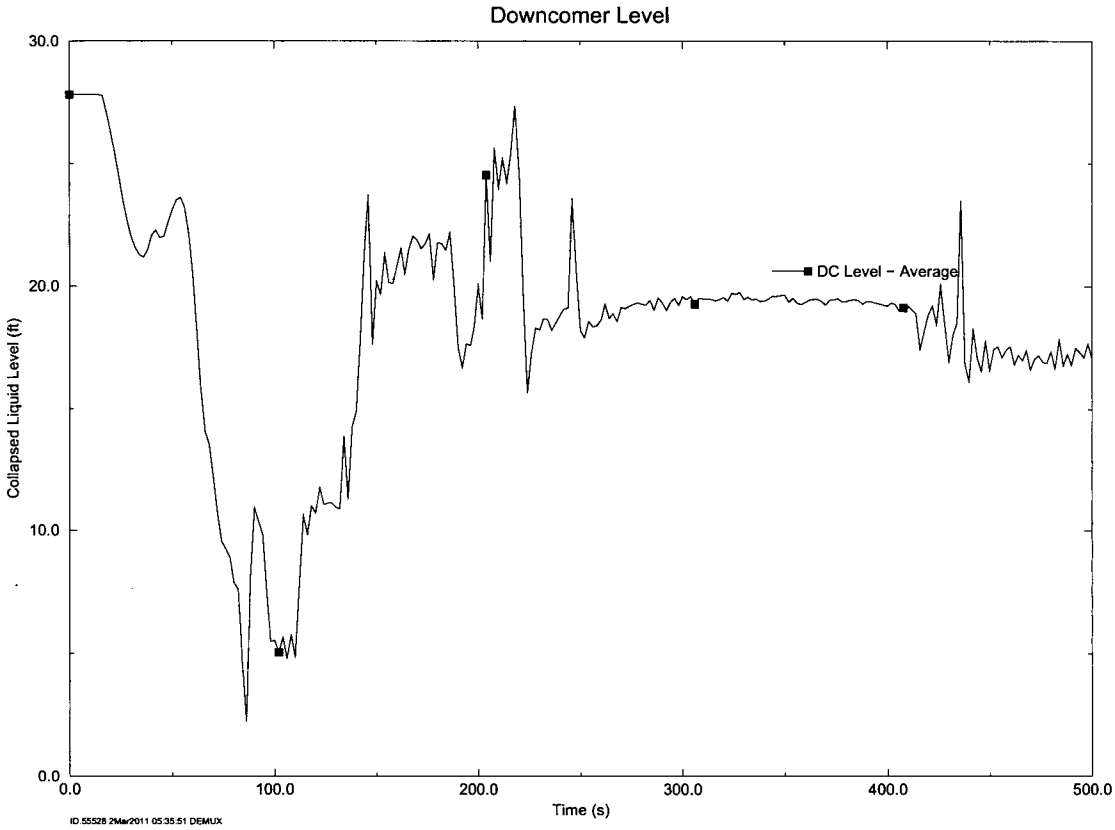

Figure 4-19 Downcomer Collapsed Liquid Level – 9.76 in Break

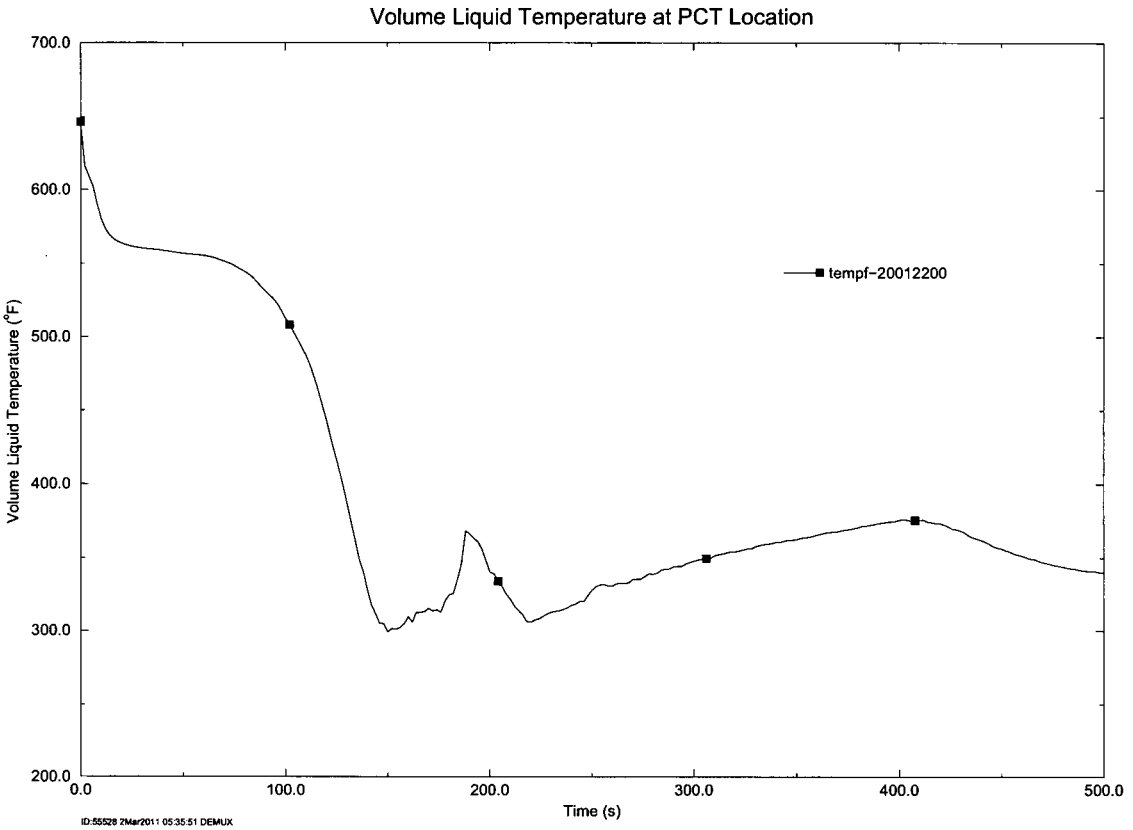

Figure 4-20 Volume Liquid Temperature at PCT Location – 9.76 in Break

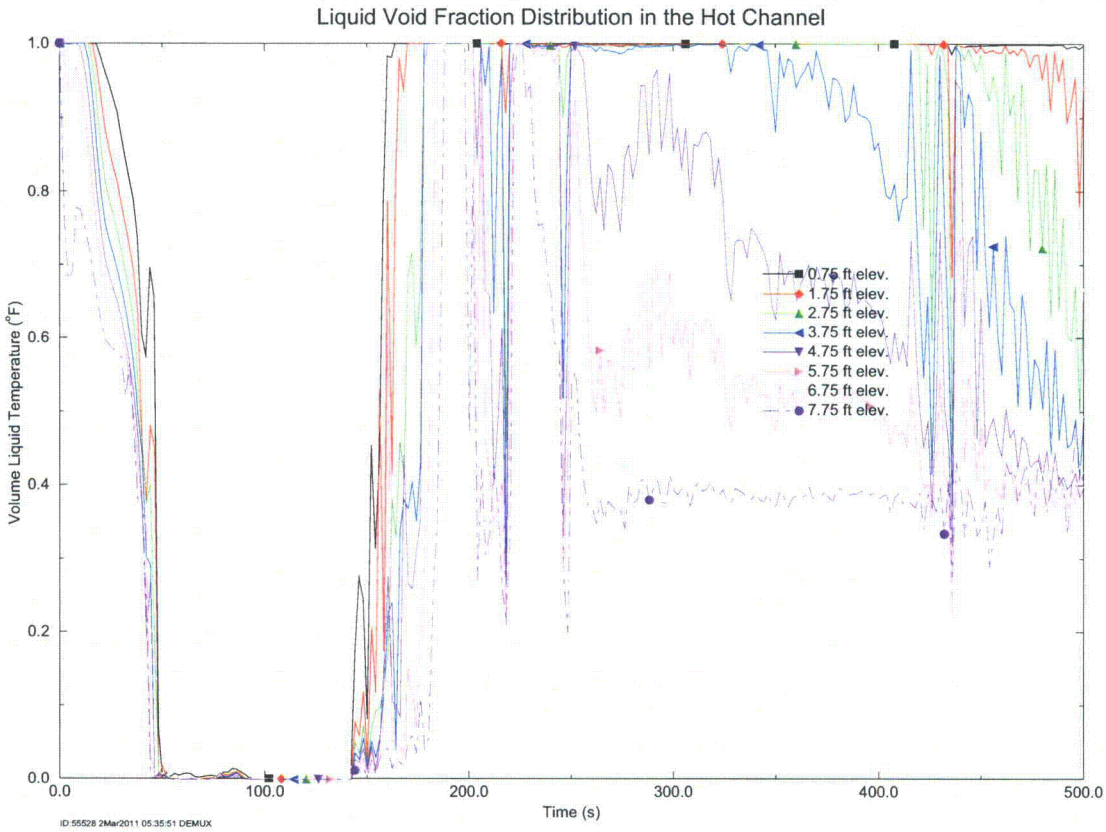

Figure 4-21 Liquid Void Fraction Distribution in the Hot Channel 0.75 – 7.75 ft Elevation – 9.76 in Break

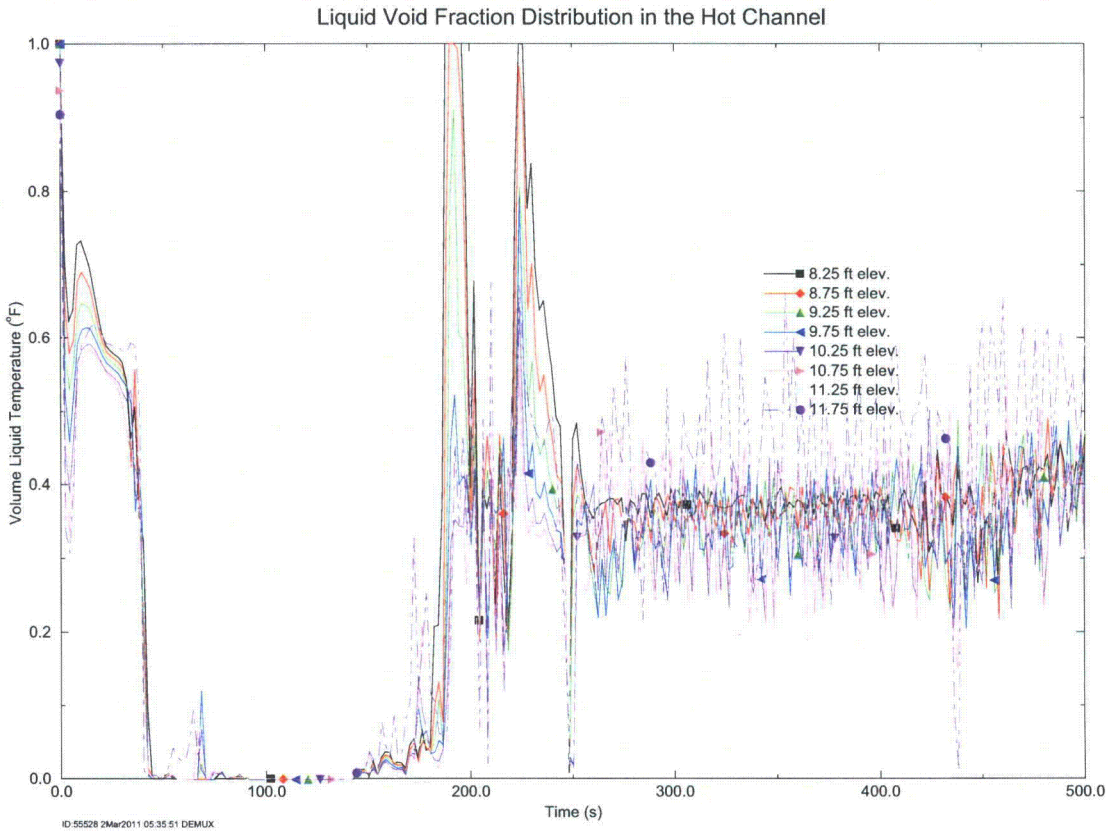

Figure 4-22 Liquid Void Fraction Distribution in the Hot Channel 8.25 – 11.75 ft Elevation – 9.76 in Break

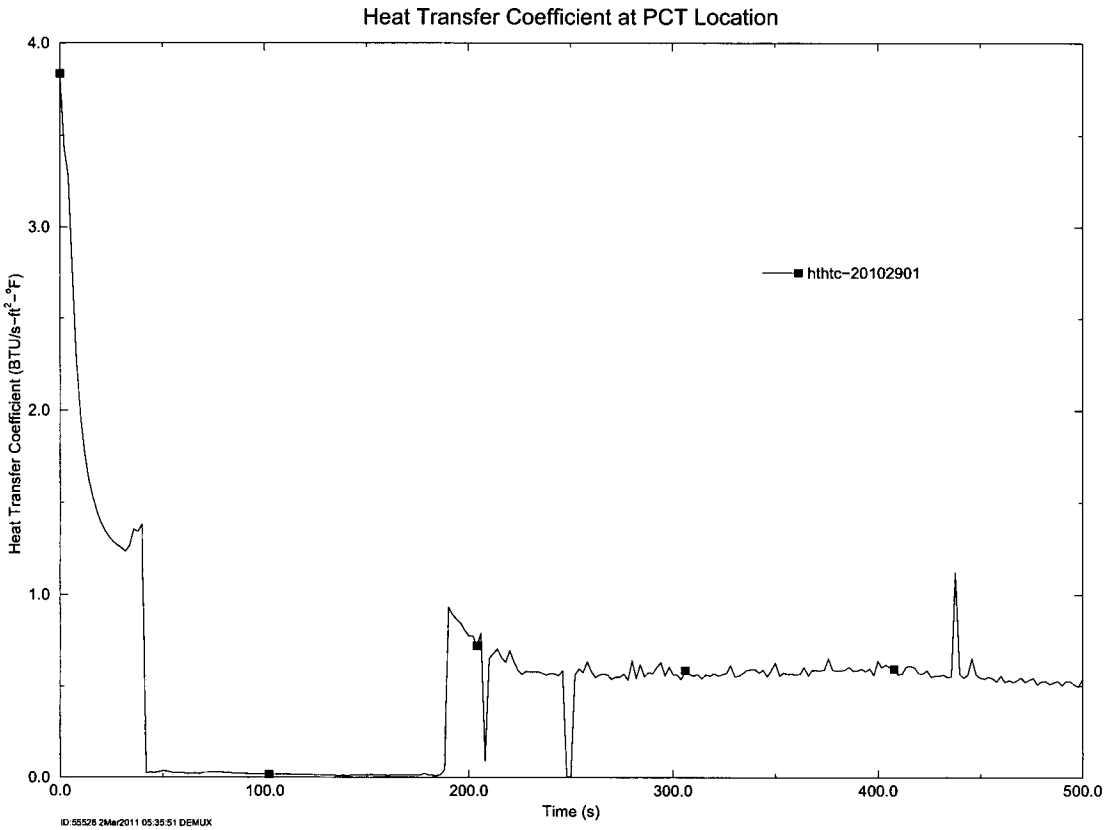

Figure 4-23 Heat Transfer Coefficient at PCT Location – 9.76-inch Break

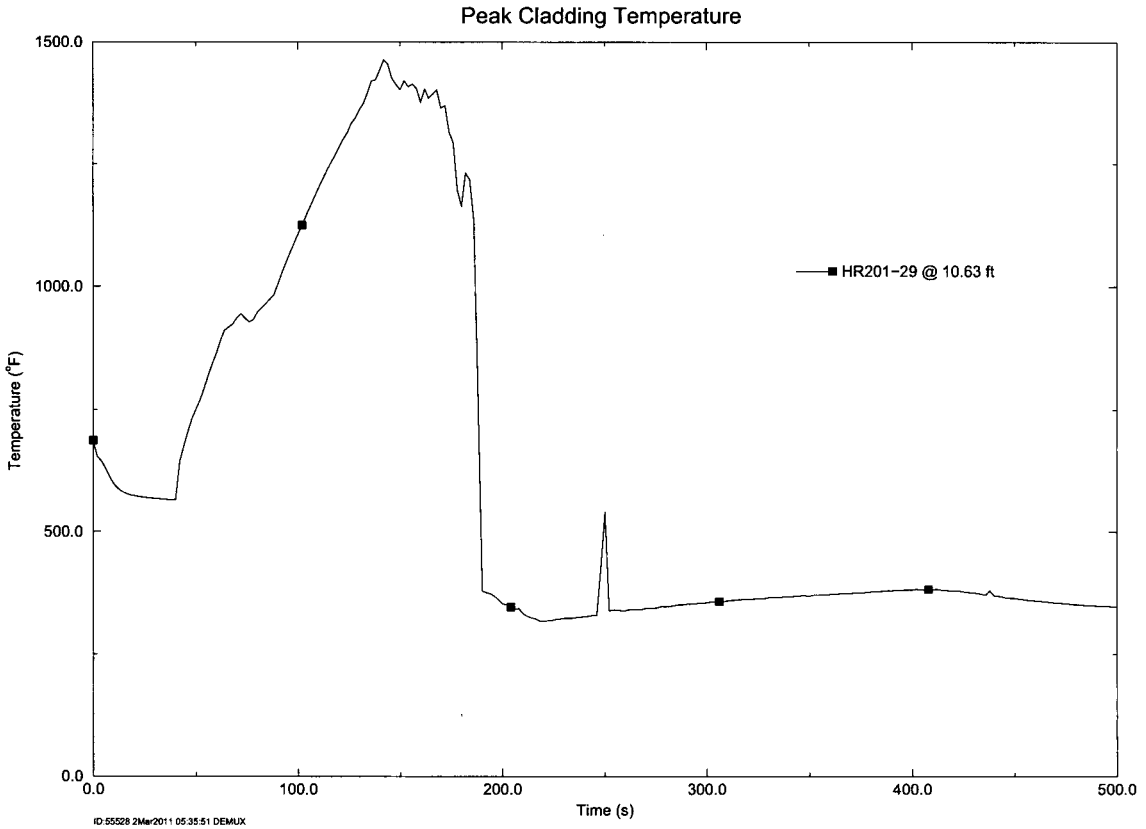

Figure 4-24 Peak Cladding Temperature at PCT Location – 9.76-inch Break

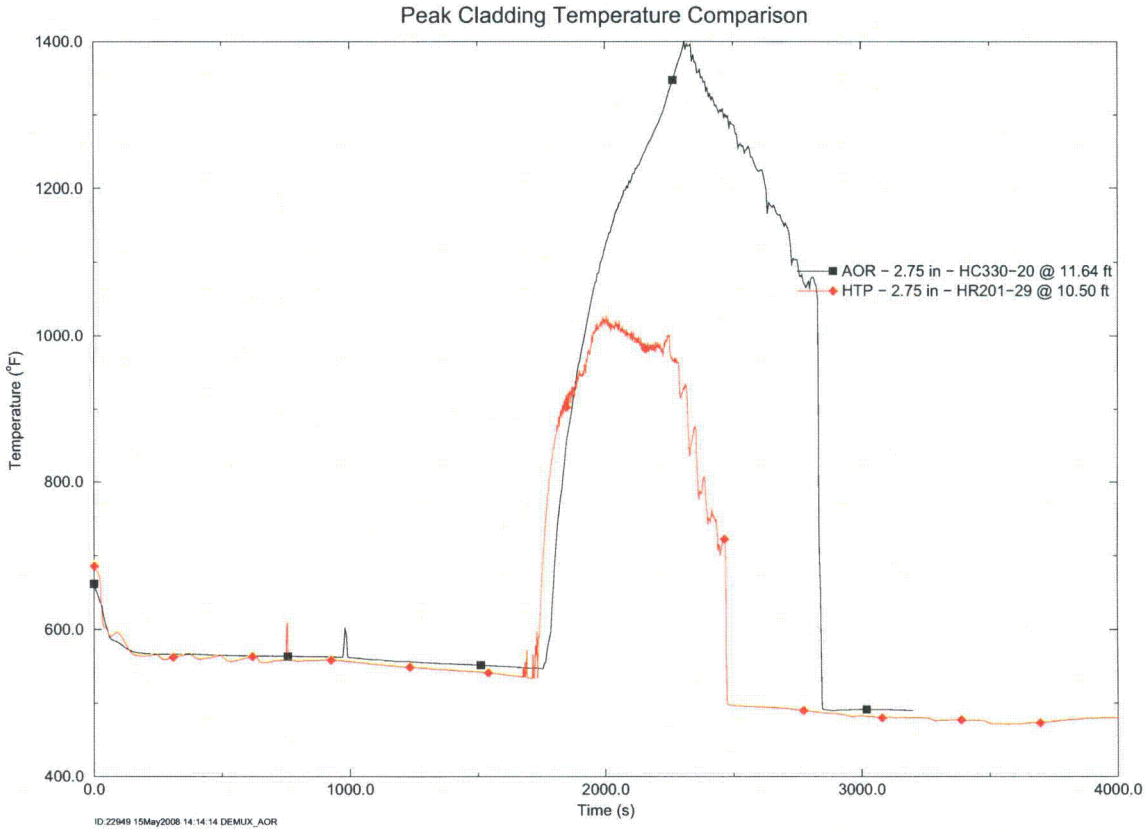

Figure 4-25 Comparison of PCTs – AOR vs. S-RELAP5 Analysis

5.0 References

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

2. AREVA NP Document BAW-10240(P)(A) Revision 0, *Incorporation of M5 Properties in Framatome ANP Approved Methods*, May 2004.

3. AREVA NP Document 51-9012330-001, *Guidelines for PWR Safety Analysis: Small Break LOCA Analysis Using S-RELAP5,* July 2007.

4. Topical Report BAW-10227P-A, Revision 1, *"Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," AREVA NP (June 2003).*