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Attachment 5

ANP-3676NP, Revision 0, Surry Fuel-Vendor Independent Small Break LOCA Analysis Licensing Report

(Non-proprietary)

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

framatome

Surry Fuel-vendor Independent Small Break LOCA Analysis

ANP-3676NP
Revision 0

Licensing Report

July 2018

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
AFW	Auxiliary Feedwater
BOC	Beginning-of-Cycle
CFR	Code of Federal Regulations
CWO	Core Wide Oxidation
DC	Downcomer
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EM	Evaluation Model
EOC	End-of-Cycle
FVI	Fuel Vendor Independent
HHSI	High Head Safety Injection
LHGR	Linear Heat Generation Rate
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LSC	Loop Seal Clearing
MFW	Main Feedwater
MLO	Maximum Local Oxidation
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
PZR	Pressurizer
SBLOCA	Small Break Loss of Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
TT	Turbine Trip
W	Westinghouse
W3	Westinghouse 3-loop plant

1.0 INTRODUCTION

This report summarizes the fuel-vendor independent small break loss-of-coolant accident (SBLOCA) analysis for Surry Units 1 and 2. The purpose of the analysis is to provide the transient results which will support the demonstration of acceptable emergency core cooling system (ECCS) design performance against the 10 CFR 50.46 criteria. The analysis was performed in accordance with the Nuclear Regulatory Commission (NRC)-approved S-RELAP5 methodology described in Reference 1 and as supplemented by Reference 2. This report justifies the use of the methodology for a fuel-vendor independent (FVI) SBLOCA analysis.

The analyzed Surry plant is a 3-loop, Westinghouse (W)-designed pressurized water reactor (PWR) with a down-flow barrel baffle configuration and 15x15 fuel assemblies. The analysis supports operation at a core power level of 2597 MWt; an F_Q of 2.5 which includes uncertainties and K(z)=1; a radial peaking, $F_{\Delta H}$, of 1.70 which includes uncertainties; 7% steam generator (SG) tube plugging in each SG; and a total initial core bypass of 6.0%.

A complete spectrum of cold leg break sizes was considered, ranging from 1.0 inch diameter to 8.7 inch diameter. In addition, sensitivity studies were performed to consider delayed reactor coolant pump (RCP) trip, attached piping breaks, and ECCS fluid temperature.

2.0 SUMMARY OF RESULTS

The limiting Peak Clad Temperature (PCT) from the break spectrum occurs in the 2.6 inch break with a PCT of 1673°F. The maximum value from the break spectrum for the transient maximum local oxidation (MLO) is 1.43%. The transient MLO does not include the pre-transient oxidation which is dependent on cladding type. The maximum core-wide oxidation (CWO) is less than 0.06%.

Consistent with the additional prescriptions of the evaluation model (EM) supplement in Reference 2, a delayed RCP trip study, an attached pipe break study, and an ECCS temperature sensitivity study were performed. The delayed RCP trip study analyzed both cold leg and hot leg break spectrums with a trip time of five minutes after the loss of hot leg subcooling. The attached pipe break study analyzed a break in both the pumped safety injection (SI) line connection and the accumulator line. The ECCS temperature study analyzed the sensitivity to temperatures different than those prescribed in the break spectrum analysis. The conclusions of these studies support the break spectrum analysis as the licensing basis.

3.0 DESCRIPTION OF ANALYSIS

Section 3.1 of this report provides a brief description of the postulated SBLOCA event. Section 3.2 describes the analytical methods used in the analysis. That section contains a discussion of the application of the approved EM, the justification of the approved EM to fuel-vendor independent applications, any deviations, and compliance with the NRC's final Safety Evaluation (SE) of the EM. Section 3.3 presents a description of the analyzed Surry Units 1 and 2 and outlines the system parameters used in the SBLOCA analysis.

3.1 *Description of an SBLOCA Event*

The postulated SBLOCA is defined as a break in the Reactor Coolant System (RCS) pressure boundary for which the break area is up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP. This break location results in the largest amount of RCS inventory loss and the largest fraction of ECCS fluid 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 criteria (Reference 3).

The SBLOCA event progression develops in the following distinct phases: (1) subcooled depressurization (also known as blowdown), (2) natural circulation, (3) loop seal clearing, (4) core boil-off (5) core recovery and long-term cooling. The duration of each of these phases is break size and system dependent.

Following the break, the RCS rapidly depressurizes to the saturation pressure of the hot leg fluid. During the initial depressurization phase, a reactor trip is generated on low pressurizer pressure; the turbine is tripped on the reactor trip. The assumption of loss-of-offsite-power (LOOP) concurrent with the reactor scram results in RCP trip.

In the second phase of the transient, the RCS transitions to a quasi-equilibrium condition in which the core decay heat, leak flow, steam generator heat removal, and system hydrostatic head balance combine to control the core inventory. During this period, the RCPs are coasting down and the system drains top down with voids beginning to form at the top of the SG tubes and continuing to form in the reactor vessel upper head and at the top of the reactor vessel upper plenum region. The loop seals remain plugged during this phase, trapping vapor generated by the core in the RCS, and resulting in a low quality flow at the break.

The third phase in the transient is characterized by loop seal clearing. During this phase, the loop seal, which is liquid trapped in the RCP suction piping, can prevent steam from venting via the break. When a sufficient pressure difference between the reactor vessel upper head and downcomer is reached, liquid in the loop seal is displaced, clearing the loop seal, and allowing the trapped steam to be vented to the break. For a small break, the transient develops slowly, and liquid level in the RCS may drop to the loop seal level prior to establishing a steam vent. The core can become temporarily uncovered in this loop seal clearing process. Following loop seal clearing, the break flow transitions to primarily steam and the core recovers to approximately the cold leg elevation, as the pressure imbalances throughout the RCS are relieved.

The fourth phase is characterized as core boil-off. With the loop seal cleared, the venting of steam through the break causes a rapid RCS depressurization below the secondary pressure. As boiling increases in the core, the core mixture level decreases. The core mixture level will reach a minimum, in some cases resulting in deep core uncovery. The boil-off period of the transient ends when the core liquid level reaches this minimum. At this time, the RCS has depressurized to the point where ECCS flow into the reactor vessel matches the rate of boil-off from the core.

The last phase of the transient is characterized as core recovery and long-term cooling. The core recovery period extends from the time at which the core mixture level reaches a minimum in the core boil-off phase, until all parts of the core are quenched and covered by a low quality mixture. Core recovery is provided by pumped injection and passive accumulator injection when the RCS pressure decreases below the accumulator pressure.

The SBLOCA transient progression is dependent on the size of break and is typically broken into three different break size ranges. For break sizes towards the larger end of the break spectrum, significant primary system inventory loss results in larger primary system depressurization and rapid accumulator injection. For break sizes in the middle of the spectrum, the rate of inventory loss from the primary system is such 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 or to recover core liquid level on HHSI flow. This tends to maximize the heatup time of the hot rod and produce the maximum PCT and local cladding oxidation. For very small break sizes, the primary system pressure does not reach the accumulator injection pressure; however, primary system inventory loss is not significant and typically within the means of HHSI makeup capacity such that core uncovery is minimal if not precluded.

3.2 *Method of Analysis*

3.2.1 Approved Analytical Method

This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1 and as supplemented by Reference 2. This section describes the application of the approved methodology to the Surry plant. This application of the method to a non-Framatome fuel core is described in Section 3.2.2. Deviations from the method are described in Section 3.2.3. Compliance with the SE is described in Section 3.2.4.

The EMF-2328 SBLOCA evaluation model for event response of the primary and secondary systems and the hot fuel rod used in this analysis is based on the use of two computer codes. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50 (Reference 6), are incorporated.

Two computer codes were used in this analysis:

1. The RODEX2-2A code (References 4 and 5) 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.

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Representative system nodalization figures for a Westinghouse 3-loop plant are shown in Figure 3-1 (RCS), Figure 3-2 (Secondary System), and Figure 3-3 (Reactor Vessel). As noted in Figure 3-3, the upper plenum noding is variable.

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3.2.2 Fuel-vendor Independent Application

The foundation of the FVI-SBLOCA application is the NRC-approved SBLOCA methodology contained in EMF-2328 (Reference 1) and its supplement (Reference 2). The methodology incorporates the appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50 (Reference 6) and is applicable to W and CE-designed plants. It is typically used to analyze Framatome fuel products. A fuel-vendor independent application inherently must use representative fuel design and material characteristics. [

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for a 15x15 assembly, and non-fuel related plant-specific details.

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] The general SBLOCA event progression is described in Section 3.1. [

] Consequently, these inputs are all plant-specific.

[

] Important system parameters and initial conditions used in the analysis are given in Table 3-1, Table 3-2 and Table 3-3. [

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] The Framatome fuel product is designed so that it can be a

replacement to the resident fuel and therefore is functionally very similar to other vendor fuel designs. The application was reviewed in light of the current assembly design and

[

] the analysis herein is applicable to use for 15x15 fuel products with ZIRLO and Optimized ZIRLO cladding. Any changes in the fuel assembly design or cladding material would need to be evaluated for continued applicability of the method and analyses results.

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] The results of the SBLOCA analysis described herein can therefore be used to support Surry Units 1 and 2 with a non-Framatome core.

3.2.3 Deviations from Approved Analytical Method

A deviation from the modeling approach is applied to better represent the primary system mass distribution during a specific period of the SBLOCA transient:

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3.2.4 SE Compliance

The supplemented EMF-2328 method (Reference 1 and Reference 2) contains no restrictions. Except as indicated in Section 3.2.3, the analysis was performed in accordance with the approved methodology.

3.3 *Plant Description and Summary of Analysis Parameters*

Surry is a W-designed PWR with three loops. Each loop contains a hot leg, a U-tube SG, an RCP, and a cold leg. A pressurizer is connected to the hot leg of one of the loops. The reactor has a core power level of 2597 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 157 fuel assemblies. The ECCS contains two centrifugal charging/HHSI pumps, two low head safety injection (LHSI) pumps, and three accumulators.

The RCS was nodalized in the S-RELAP5 model with control volumes interconnected by flow paths or "junctions." The model includes three accumulators, a pressurizer, and three SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SG tube plugging level of 7% was modeled in each SG. Important system parameters and initial conditions used in the analysis are given in Table 3-1. 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 10 CFR 50 Appendix K (Reference 6).

The analysis assumed LOOP concurrent with reactor scram, which is based on the low pressurizer pressure reactor trip and includes delays as stated in Table 3-1. The assumption of LOOP results in RCP trip.

The single failure criterion required by 10 CFR 50 Appendix K (Reference 6) was satisfied by assuming the loss of one emergency diesel generator (EDG). Thus, this results in the loss of one HHSI pump, one LHSI pump and one motor-driven AFW pump. The initiation of the HHSI and LHSI systems were delayed by 40 seconds, following a safety injection actuation system (SIAS) activation. Table 3-2 and Table 3-3 show the minimum ECCS flow rates with EDG failure for HHSI and LHSI, respectively.

All three SGs receive AFW. The AFW flow rates were minimized and were delayed 60 seconds beyond the time of the AFW system initiation on low-low SG level. The input model includes the main steam lines from the SGs to the turbine control valve, as well as the inlet piping to the MSSVs. The MSSVs were set to open at their nominal setpoints plus a 3% tolerance.

The axial power shape for this analysis is shown in Figure 3-4. [

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Table 3-1 System Parameters and Initial Conditions

Parameter	Analysis Value
Reactor Power, MWt	2597 [1]
Axial Power Shape	Figure 3-4
Total Peaking Factor, F_Q	2.5 [1]
Radial Peaking Factor, $F_{\Delta H}$	1.70 [1]
RCS Flow Rate, gpm	265,500
Pressurizer Pressure, psia	2250
RCS Average Temperature, °F	581.6
Accumulator Pressure, psia	580.0
Accumulator Fluid Temperature, °F	110.0
Accumulator Water Volume, ft^3	965.0
SG Tube Plugging Level per SG, %	7
SG Secondary Pressure, psia	800
MSSV Lift Pressure and Tolerance	Nominal + 3% tolerance
MFW Temperature, °F	438.1
AFW Flow Rate per fed SG, gpm	233.3
AFW Temperature, °F	120.0
Pressurizer Pressure – Low Reactor Trip Setpoint (RPS), psia	1899.7
Reactor Trip Delay Time on Low Pressurizer Pressure[2], sec	2.0
Reactor Scram Delay Time, sec	0.0
SIAS Activation Pressurizer Pressure Setpoint, psia	1715.0
HHSI and LHSI Pump Delay Time on SIAS, sec	40.0
HHSI and LHSI Fluid Temperature, °F	62.5
Low-Low SG Level Setpoint, % Narrow Range Span	0.1
AFW Delay, sec	60.0

[1] Includes associated measurement uncertainty
[2] Includes scram delay

Table 3-2 HHSI Flow Rates

Pressure (psia)	Total Intact Flow (gpm)	Broken Flow (gpm)
0.0	253.2	146.6
14.7	253.2	146.6
64.7	250.3	144.9
114.7	247.3	143.2
214.7	241.5	139.8
514.7	223.9	129.7
1014.7	191.9	111.7
1264.7	173.7	101.1
1414.7	162.1	94.6
1731.7	131.0	76.2
2014.7	101.5	59.1
2114.7	89.2	52.5

Table 3-3 LHSI Flow Rates

Pressure (psia)	Total Intact Flow (gpm)	Broken Flow (gpm)
0.0	2015.8	1007.9
14.7	2015.8	1007.9
52.7	2015.8	1007.9
64.7	1850.0	925.0
69.7	1746.7	873.3
89.7	1401.3	700.6
114.7	919.5	459.7
139.7	370.6	185.3
149.7	246.1	123.0
154.7	102.7	51.3
154.8	0.0	0.0
2114.7	0.0	0.0

Figure 3-1 Generic W3 Plant Primary System Nodalization

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Figure 3-2 Generic W3 Plant Secondary System Nodalization

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Figure 3-3 Generic W3 Reactor Vessel Nodalization

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Figure 3-4 Axial Power Distribution

[ ]

Figure 3-5 [ ]

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4.0 ANALYTICAL RESULTS

4.1 *Break Spectrum Results*

The Surry break spectrum analysis for SBLOCA includes breaks of varying diameter up to 10% of the flow area for the cold leg. The break spectrum resolution follows that prescribed by the methodology and is refined to determine the limiting break size, identified as the case with the highest PCT, and the largest break size which depressurizes to a pressure just above the accumulator pressure. Figure 4-1 displays the PCT results as a function of break size. A summary of the results from each case of the break spectrum analysis is presented in Table 4-1. The event times for each case of the break spectrum are provided in Table 4-2. The limiting PCT case was determined to be the 2.6 inch break with a PCT of 1673°F. The 2.5 inch break results in the limiting transient MLO with a value of 1.43%. The largest break size which was resolved just above the accumulator setpoint is the 2.0 inch break.

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4.2 *Discussion of Limiting PCT Break Transient*

The limiting break from the break spectrum was determined to be the 2.6 inch break with a PCT of 1673°F. The transient progression is shown in Figure 4-2 through Figure 4-20. The cladding temperature at the PCT location is shown in Figure 4-2. The sequence of events is shown in Table 4-3. The break opens at 0.0 seconds. RCS depressurization (Figure 4-5) results in the low pressurizer setpoint being reached at 12.2 seconds. After the 2-second delay, the reactor trips (Figure 4-6) and the RCPs and turbine are assumed to trip coincidentally. The pressure in the secondary side begins to rise and is relieved via the MSSV (Figure 4-20). The low RCS pressure initiates the SI actuation signal at 25.6 seconds. Following ECCS startup delays, the HHSI begins to inject at 66 seconds (Figure 4-12).

During the initial blowdown period, the break flow is single-phase liquid (Figure 4-4). The RCS depressurization slows, temporarily settling just above the secondary side pressure. The AFW system actuates on a low SG level and begins to inject into all three steam generators at 84 seconds (Figure 4-19).

As the break flow rate is in excess of the HHSI injection, the RCS mass continues to decrease (Figure 4-7). The core begins to uncover at 333 seconds (Figure 4-9, Figure 4-10). The SG tubes drain at about 550 seconds (Figure 4-16). The broken loop seal clears at 593 seconds (Figure 4-15) and the break flow transitions to steam relief. The pressure imbalance across the RCS is relieved and the depressed core level recovers temporarily in a balance with the DC level (Figure 4-8). Approximately 400 seconds after the loop seal clears, the core mixture level begins to decrease again due to boil-off of the RV inventory and a clad temperature excursion begins (Figure 4-2).

Continued RCS depressurization leads to increased HHSI injection (Figure 4-12). However, during this period of the transient the mixture level remains low with poor cooling in the upper regions and the clad temperature excursion persists. The first accumulator injection occurs at 1662 seconds (Figure 4-14). The accumulator injection increases the DC level and the mixture level in the core increases. The cladding temperature excursion is terminated at 1785 seconds with a PCT of 1673°F. By about 2470 seconds, the core mixture level is fully recovered and the entire core is quenched. After this time, the HHSI is able to provide enough flow to stabilize the core cladding at low temperatures and maintain RCS and RV inventory.

4.3 *Additional Studies*

4.3.1 Delayed RCP Trip

The break spectrum analysis assume RCP trip coincident with reactor trip. For plants such as Surry 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 result in less inventory loss out the break early in the transient, but in the longer term could result in more overall inventory loss out the break. It has been postulated that tripping the pumps when the minimum RCS inventory occurs could cause a collapse of voids in the core, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher PCT.

Section II.K.3.5 of NUREG-0737 (Reference 7) calls for the analysis of a delayed RCP trip for SBLOCA analyses. This was followed with specific recommendations for manual RCP trip in References 8 and 9 as NRC Generic Letters applicable to Westinghouse and CE plants. Based on indications/conditions consistent with the Surry licensing basis and Emergency Operating Procedures, a spectrum of hot and cold leg breaks is analyzed to support the Surry RCP trip procedure. The assumed manual trip time in the analysis is five minutes after the loss of hot leg subcooling margin.

The studies demonstrated that the PCTs are over 150ºF less in the delayed RCP, cold leg break study and over 200ºF less in the delayed RCP, hot leg break study than limiting PCT in the break analysis. In conclusion, the severity of RCP trip delayed until 5 minutes after the loss of hot leg subcooling, with a break in either location, is less than that of the break spectrum analysis with RCP trip coincident with the reactor trip.

4.3.2 Attached Pipe Breaks

The ECCS must cope with ruptures of the main RCS piping and breaks in attached piping. To accomplish this, an evaluation is made of the ruptures in attached piping that also compromise the ability to inject emergency coolant into the RCS. When combined with a single failure, the ECCS capability is significantly compromised. The size of the rupture and the portion of ECCS lost directly to containment are dependent on the plant design. In order to assure acceptable ECCS performance, the scope of analysis includes accidents of this type.

Surry has a separate line for the accumulator and the pumped SI injection connected to each cold leg. The high head and low head system share a common short length of pipe before joining to the cold leg. Both the accumulator and SI line break are analyzed. The accumulator line break resulted in a PCT of 1292ºF and a transient MLO of 0.06%. The SI line break resulted in a PCT of 934ºF and transient MLO of less than 0.01%. The results are less limiting than those of the break spectrum analysis.

4.3.3 ECCS Temperature Sensitivity

Table 4-1 Summary of SBLOCA Break Spectrum Results

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Table 4-2 Event Times for Break Spectrum[4] (seconds)

Table 4-3 2.6 Inch Break – Sequence of Events

Event	Time (sec)
Break Opening	0.0
Low PZR Pressure Trip	12.2
Reactor Scram, RCP and Turbine Trip	14.2
SIAS Issued	25.6
HHSI Flow: Loop 1/2/3, Broken	66/66/66
AFW: SG 1/2/3	84/84/84
Core Uncovery	333
Loop Seal Clearing: Loop 3, Broken	593
Break Uncovery	595
Accumulator Flow: Loop 1/2/3, Broken	1662/1662/1662
PCT Time	1785
Loop Seal Clearing: Loop 2	2426
Loop Seal Clearing: Loop 1	2427
Approximate Core Quench	2470
Hot Rod Rupture Time	-
LHSI Flow: Loop 1/2/3, Broken	-/-/-

Figure 4-1 PCT vs. Break Size for Break Spectrum

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Figure 4-2 2.6 Inch Break – Cladding Temperature at PCT Node

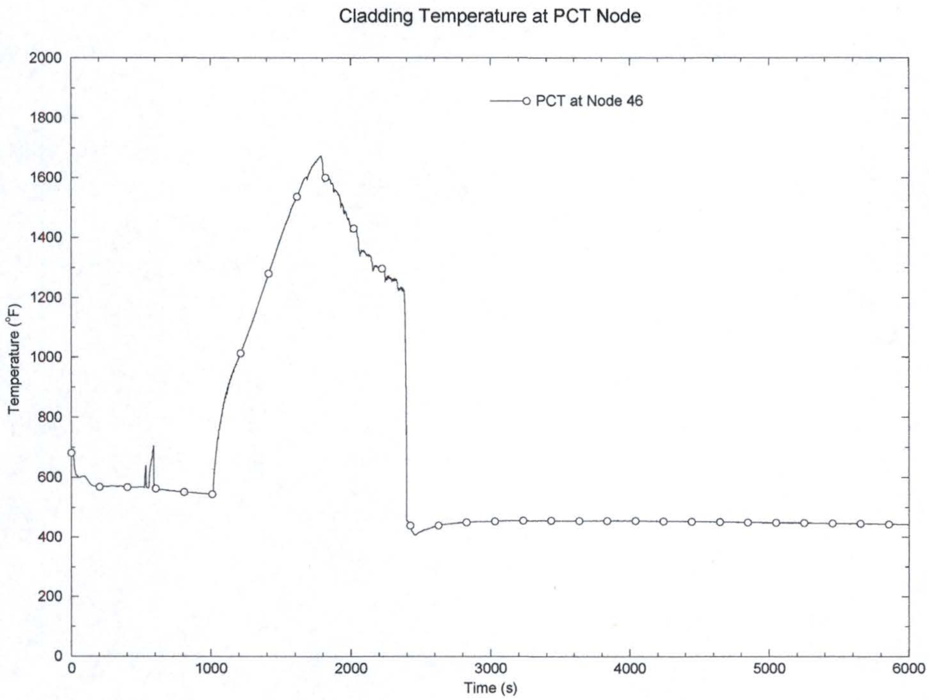

Figure 4-3 2.6 Inch Break – Break Flow Rate

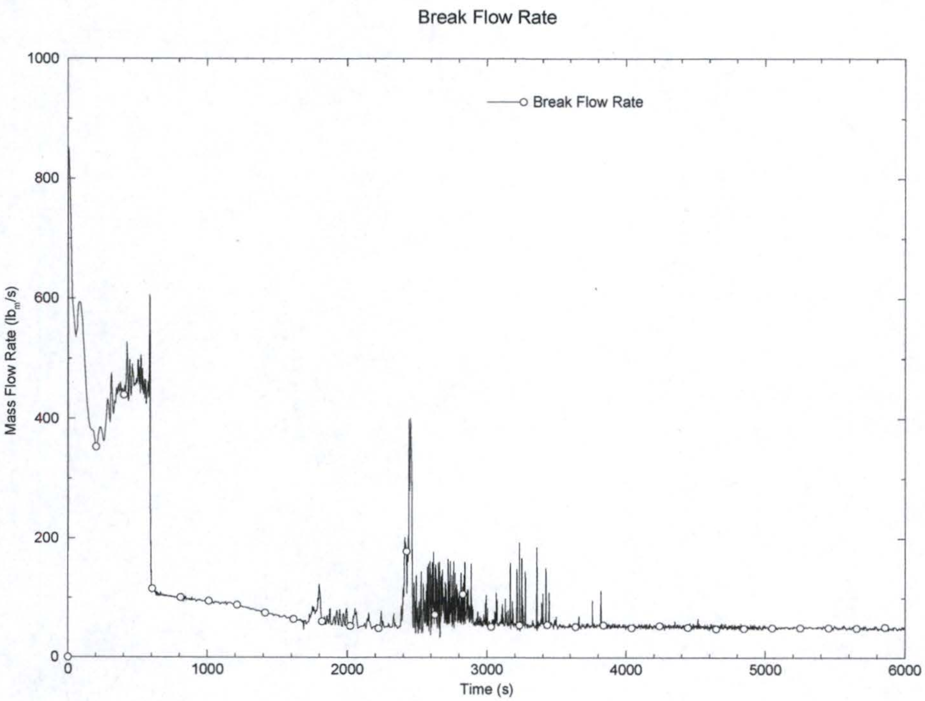

Figure 4-4 2.6 Inch Break – Break Void Fraction

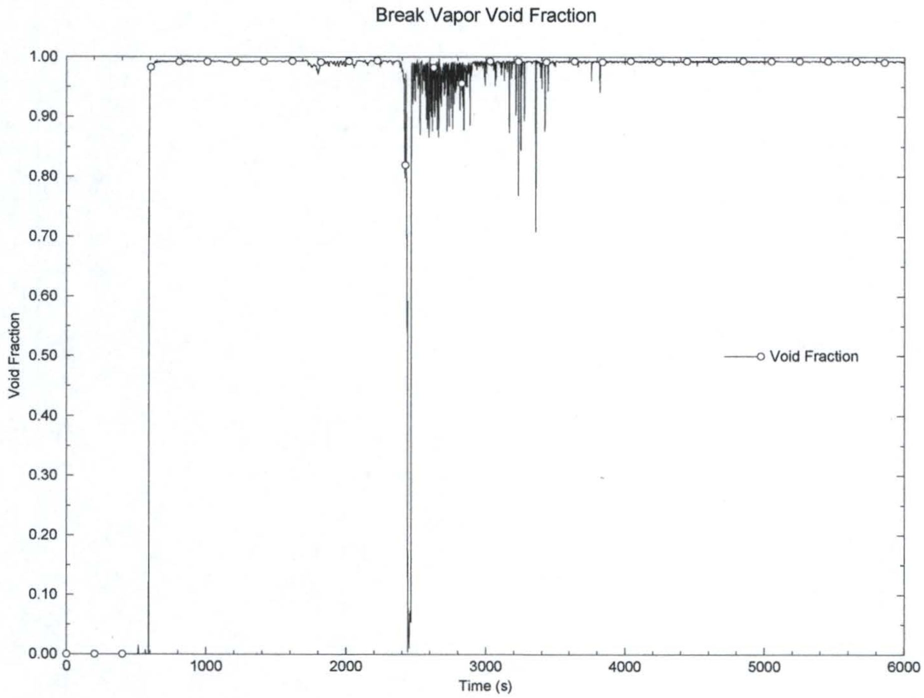

Figure 4-5 2.6 Inch Break – System Pressures

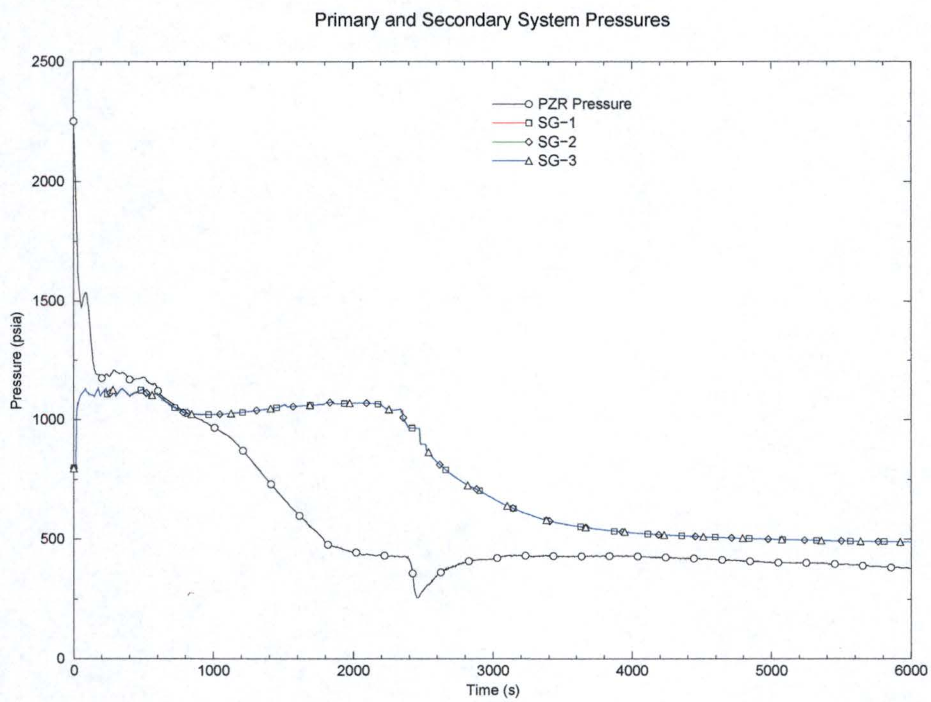

Figure 4-6 2.6 Inch Break – Reactor Power

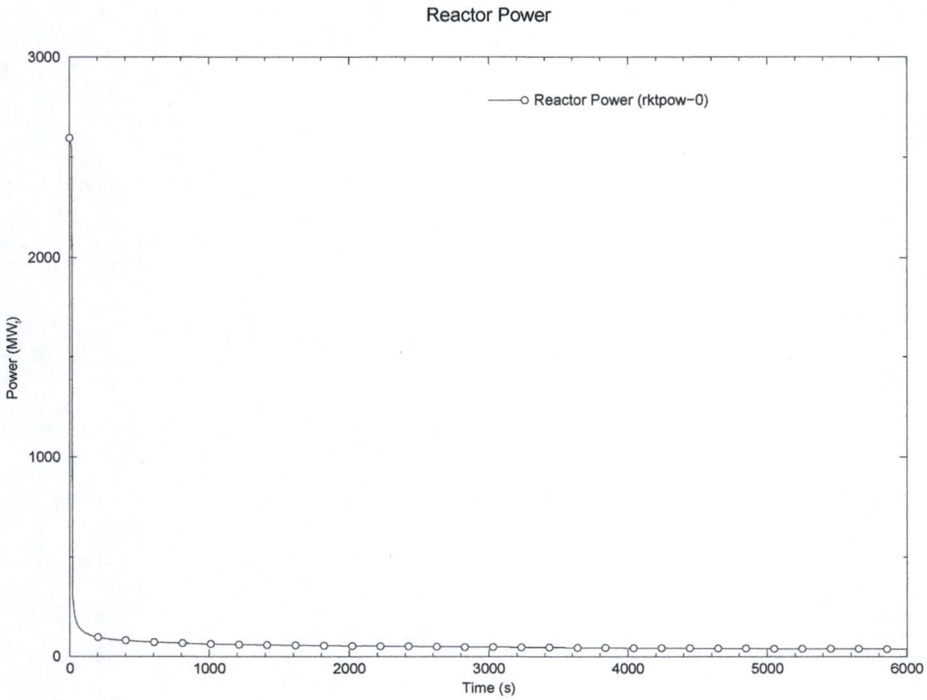

Figure 4-7 2.6 Inch Break – RCS and RV Masses

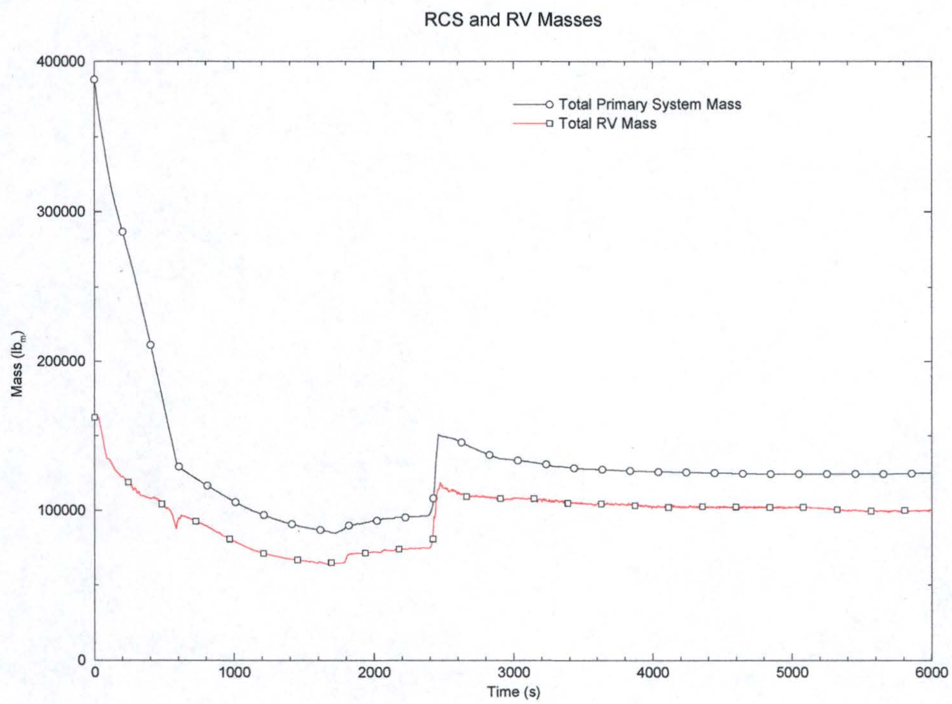

Figure 4-8 2.6 Inch Break – Downcomer Level

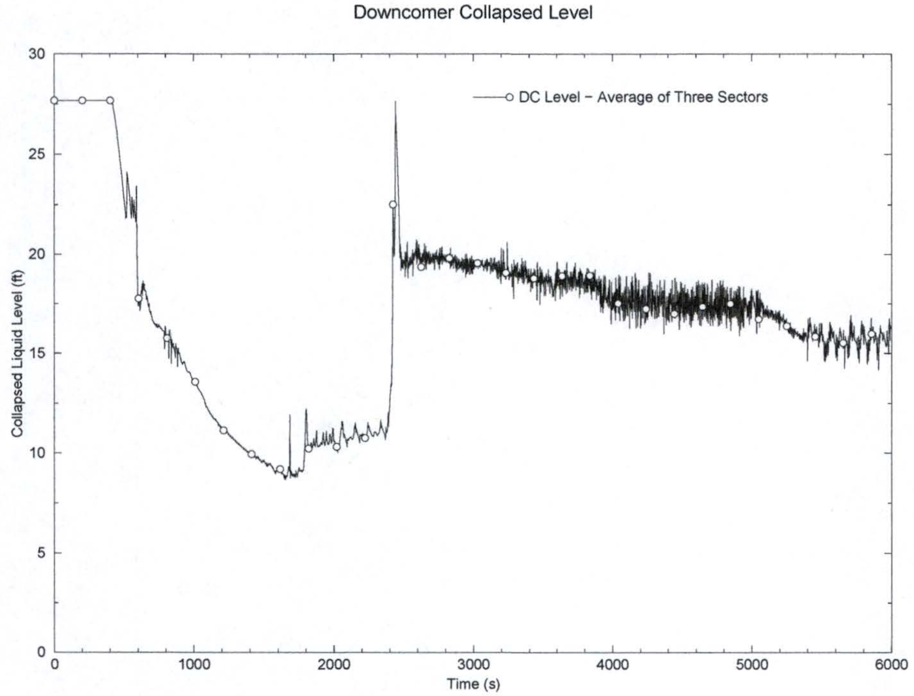

Figure 4-9 2.6 Inch Break – Hot Assembly Collapsed Level

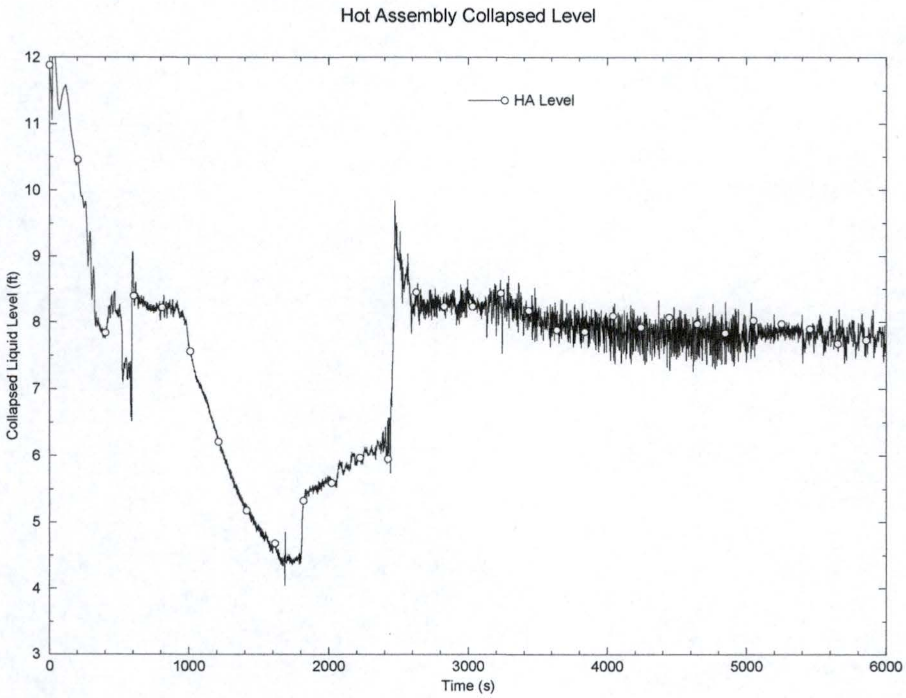

Figure 4-10 2.6 Inch Break – Hot Assembly Mixture Level

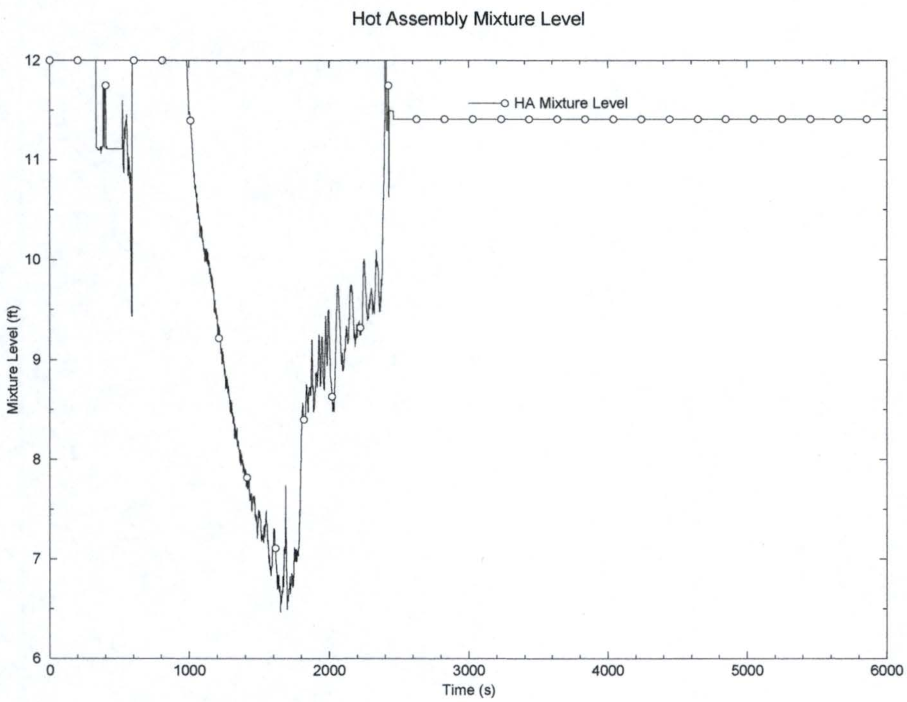

Figure 4-11 2.6 Inch Break – Cold Leg Mass Flow Rates

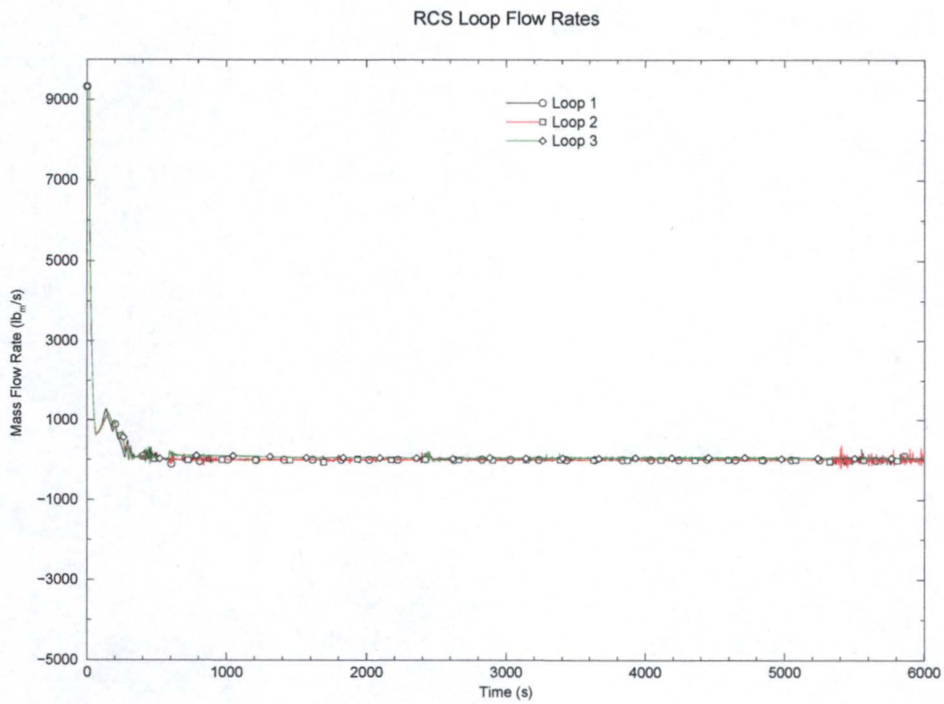

Figure 4-12 2.6 Inch Break – HHSI Mass Flow Rates

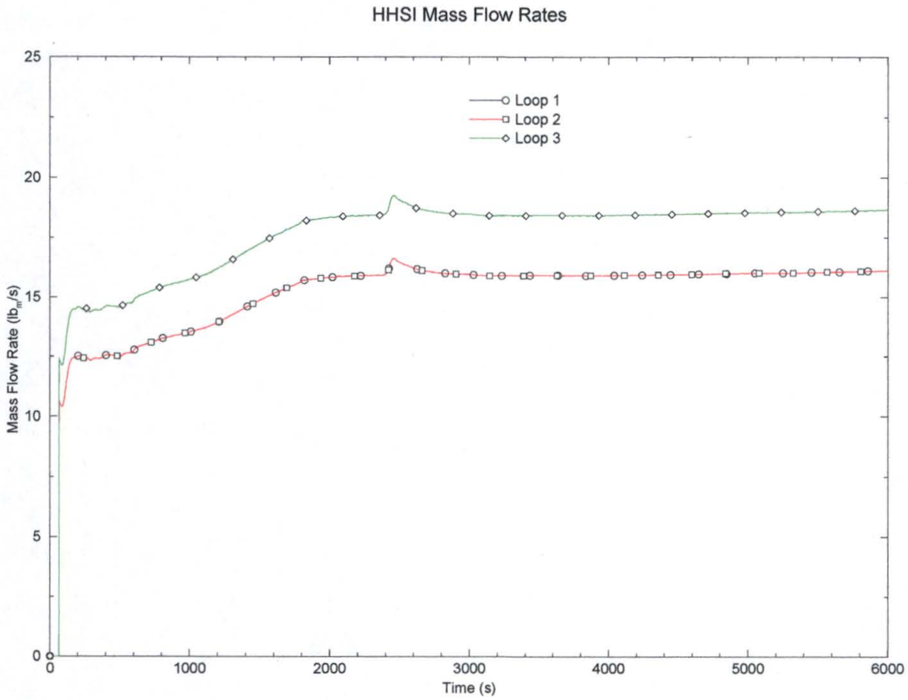

Figure 4-13 2.6 Inch Break – LHSI Mass Flow Rates

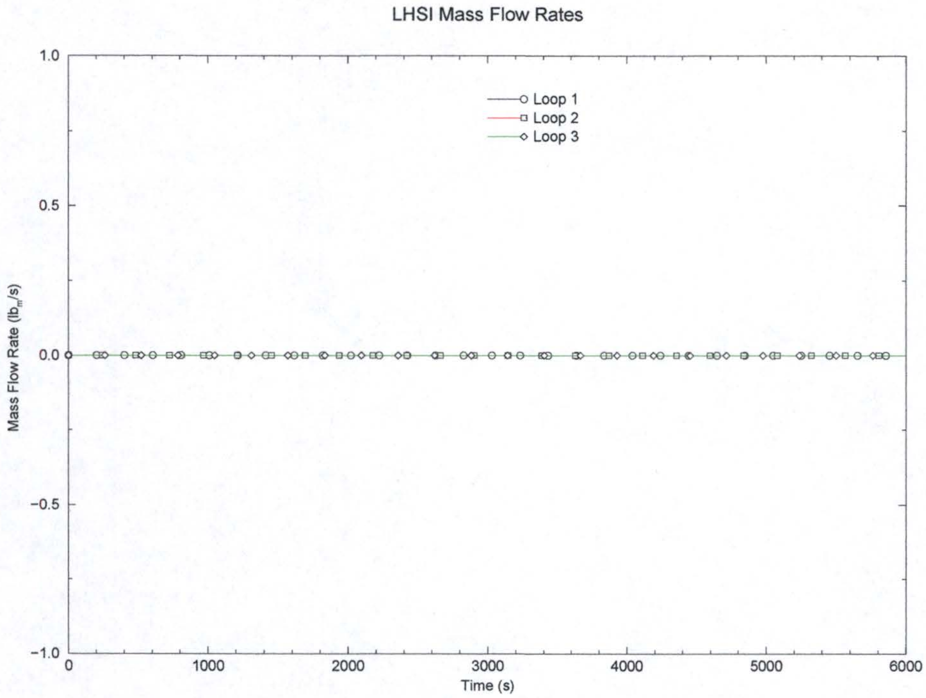

Figure 4-14 2.6 Inch Break – Accumulator Mass Flow Rates

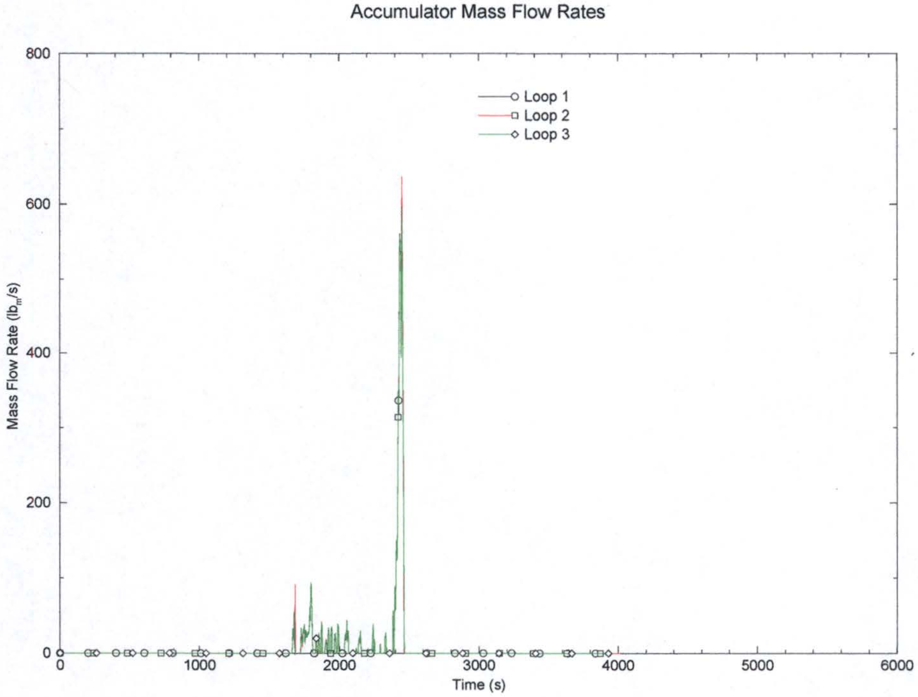

Figure 4-15 2.6 Inch Break – Loop Seal Upside Collapsed Levels

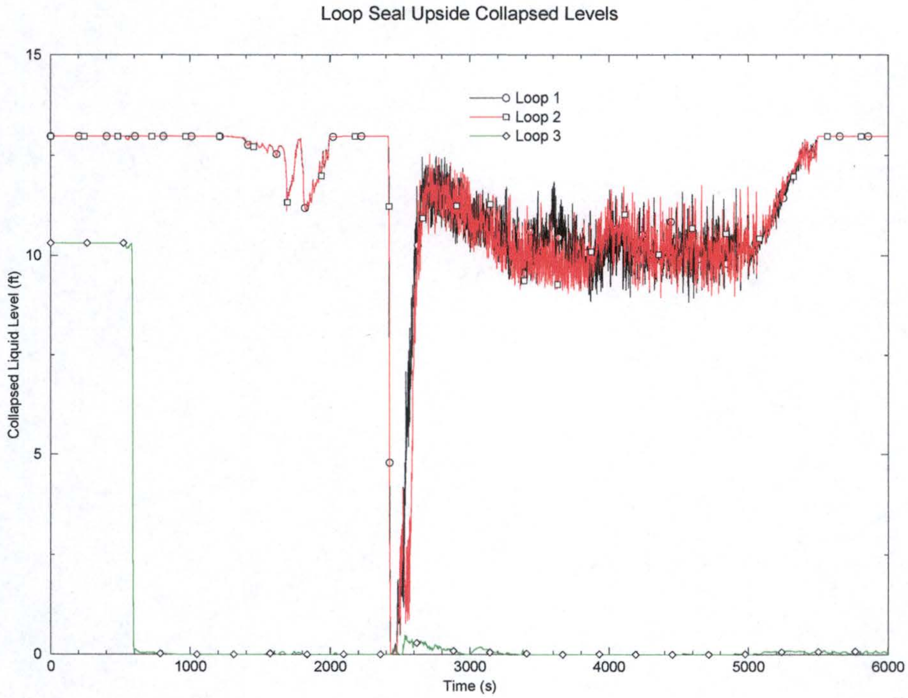

Figure 4-16 2.6 Inch Break – SG Upside Tube Collapsed Level

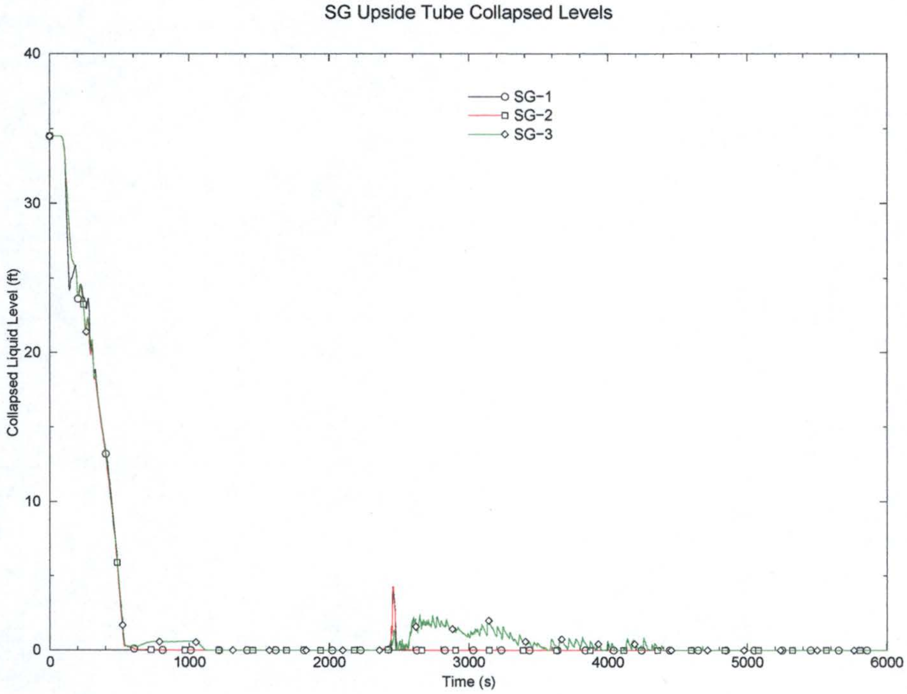

Figure 4-17 2.6 Inch Break – Secondary Mass

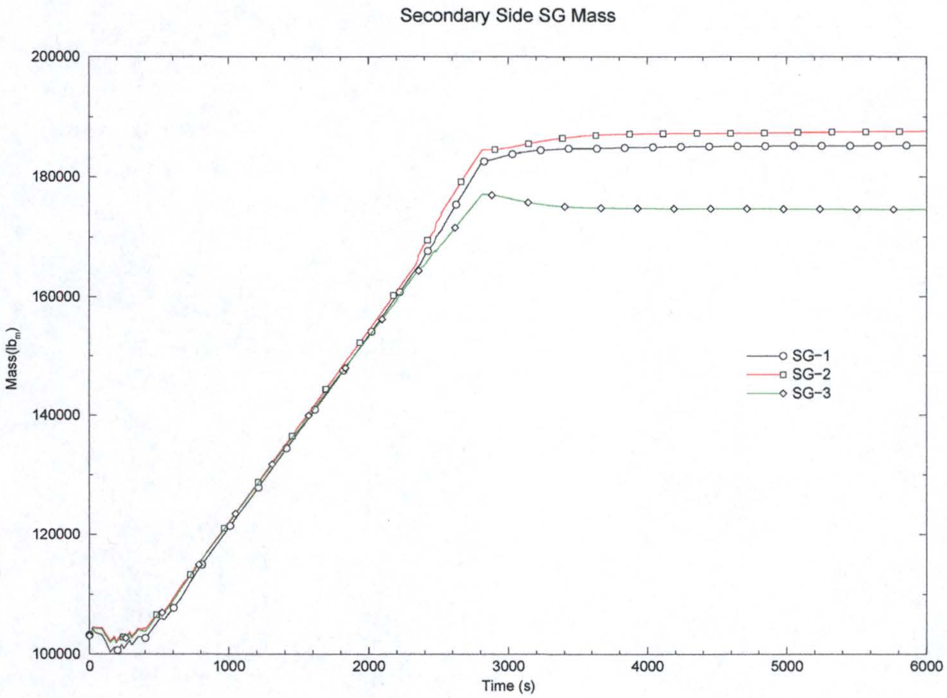

Figure 4-18 2.6 Inch Break – MFW Mass Flow Rates

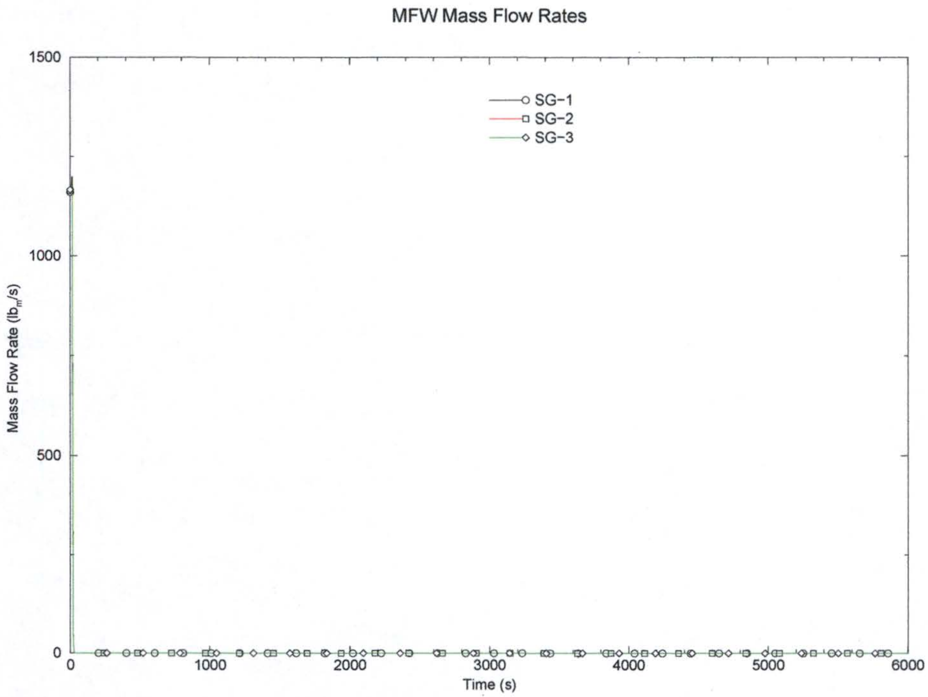

Figure 4-19 2.6 Inch Break – AFW Mass Flow Rates

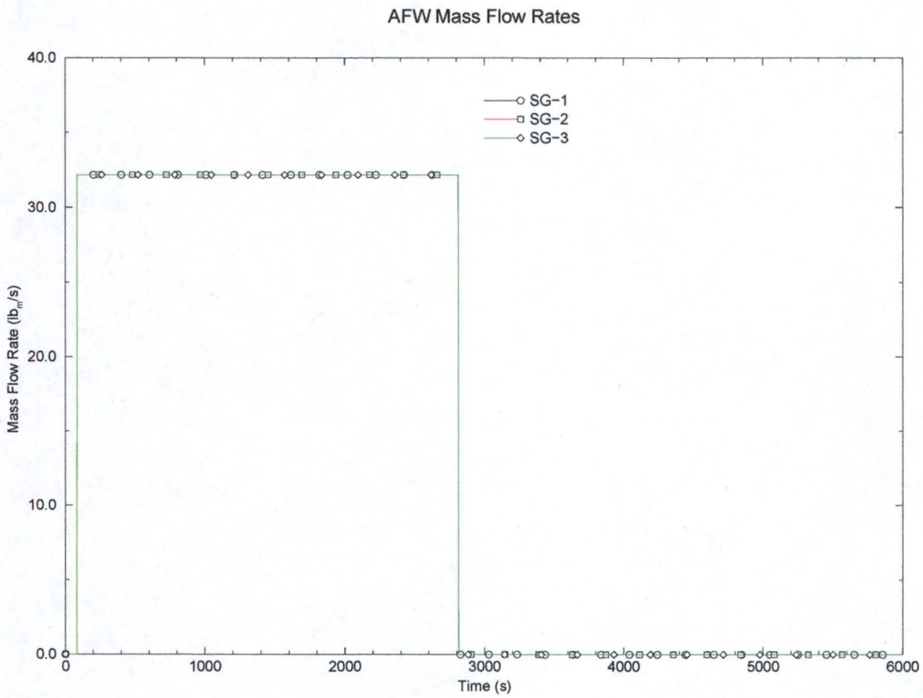

Figure 4-20 2.6 Inch Break – MSSV Mass Flow Rates

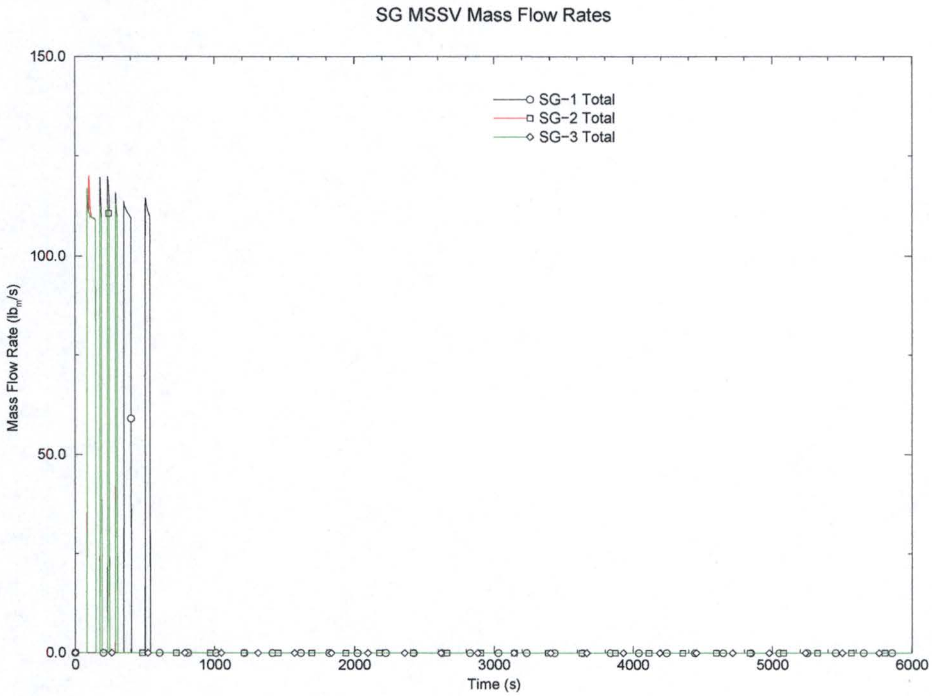

Figure 4-21 [ ]

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Figure 4-22 [ ]

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Figure 4-23 [

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