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MILLSTONE UNIT 2 M5® UPGRADE, SMALL BREAK LOCA ANALYSIS LICENSING REPORT

(NON-PROPRIETARY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

Millstone Unit 2 M5® Upgrade, Small Break LOCA Analysis

ANP-3315NP
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Licensing Report

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AREVA Inc.

Nature of Changes

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

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Nomenclature

Acronym	Definition
AFAS	Auxiliary Feedwater Actuation Signal
AFW	Auxiliary Feedwater
AREVA	AREVA Inc.
BOC	Beginning-of-Cycle
CE	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
DC	Downcomer
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOC	End-of-Cycle
HMP™	High Mechanical Performance Spacer Grid
HTP™	High Thermal Performance Spacer Grid
HPSI	High Pressure Safety Injection
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
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
SER	Safety Evaluation Report
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection actuation signal
SIT	Safety Injection tank

1.0 INTRODUCTION

This report summarizes the small break loss-of-coolant accident (SBLOCA) analysis for Millstone Nuclear Plant Unit 2. The purpose of the SBLOCA analysis is to support the upgrade to the AREVA Advanced Combustion Engineering (CE) 14 x 14 Fuel Design with M5®[1] cladding for the Millstone Nuclear Plant Unit 2 core design. This analysis was performed in accordance with the Nuclear Regulatory Commission (NRC)-approved S-RELAP5 methodology described in Reference 1 and as modified by Reference 2.

Millstone Nuclear Plant Unit 2 is a 2x4-loop, Combustion Engineering (CE)-designed pressurized water reactor (PWR). The AREVA Advanced CE14 Fuel Design with M5® cladding for Millstone Nuclear Plant Unit 2 consists of a 14x14 CE array with HTP™[2] intermediate grids and a lower HMP™[2] grid. The fuel assembly will include a Zirc-4 MONOBLOC™[2] guide tube design, M5® fuel rod design and FUELGUARD™[2] debris-resistant lower tie-plate design.

A complete spectrum of cold leg break sizes was considered, ranging from 2.0 inch diameter to 9.49 inch diameter. In addition, sensitivity studies were performed to consider delayed reactor coolant pump (RCP) trip sensitivity, attached piping break sensitivity and safety injection low fluid temperature sensitivity.

The analysis supports plant operation at a core power level of 2754 MWt (including measurement uncertainty), a peak linear hear rate (LHR) of 15.1 kW/ft, a radial peaking factor of 1.854 (including uncertainty), and 5.87% steam generator tube plugging.

[1] M5 is a registered trademark of AREVA.
[2] HTP, HMP, MONOBLOC and FUELGUARD are trademarks of AREVA.

2.0 SUMMARY OF RESULTS

A SBLOCA break spectrum analysis was performed for Millstone Nuclear Plant Unit 2 using the NRC-approved AREVA method (Reference 1) as modified by Reference 2. The analysis results demonstrate that the following acceptance criteria for Emergency Core Cooling Systems (ECCS), as stated in 10 Code of Federal Regulations (CFR) 50.46(b)(1-4) (Reference 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 limiting PCT is 1707°F for a 3.78-inch diameter cold leg pump discharge break. The total maximum local oxidation is less than 6%, including a pre-transient oxidation of 2.3% and transient maximum oxidation of 3.6%. The maximum core-wide oxidation is less than 0.04%. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 4).

In addition to the cold leg pump discharge break spectrum analysis, three sensitivity studies were performed to consider a delayed RCP trip, break in an attached pipe and low fluid temperature safety injection. The results of the delayed RCP trip sensitivity demonstrated that there is at least 2 minutes for operators to trip all four RCPs after subcooling margin is lost in the cold leg pump suction in order to meet the 10 CFR 50.46(b)(1-4) criteria (Reference 4). The break in an attached piping sensitivity study

was performed with a 10.5-inch diameter break in the safety injection tank (SIT) line. The result of the SIT break was a PCT of 1239°F, which is less limiting than the cold leg pump discharge limiting break. The last sensitivity study performed was on reducing the SIT and safety injection (SI) fluid temperatures to approximate nominal temperatures as opposed to the maximum temperatures used in the spectrum analysis. The result of the sensitivity study confirmed that the SIT and SI fluid temperatures used in the break spectrum are conservative.

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 models used in the analysis. Section 3.3 presents a description of the Millstone Nuclear Plant Unit 2 plant parameters and outlines the system parameters used in the SBLOCA analysis. Section 3.4 describes the Safety Evaluation Report (SER) compliance.

3.1 *Description of SBLOCA Event*

The postulated SBLOCA is defined as a break in the Reactor Coolant System (RCS) pressure boundary for which the 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(b)(1-4) criteria (Reference 4).

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 a loss-of-offsite-power concurrent with the reactor SCRAM results in reactor coolant pump 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 reactor coolant pumps are coasting down and the system drains top down with voids beginning to form at the top of the steam generator (SG) tubes and continuing to form in the reactor vessel upper head and at the top of the reactor vessel upper plenum region. Also, 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 reactor coolant pump suction piping, can prevent steam from venting via the break. When the maximum pressure difference between the reactor vessel upper head and downcomer is reached, loop seal upflow is pushed, clearing the loop seal, and the trapped steam can be vented to the break. For a small break, the transient develops slowly, and liquid level in the reactor coolant system 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 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. During this time, the PCT occurs. Core recovery is

provided by pumped injection and passive SIT injection when the RCS pressure decreases below the SIT 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 to the SITs, which provides sufficient inventory in time to limit the core uncovery and clad heatup, thus hot rod heatup is typically not limiting. For break sizes in the middle of the spectrum, the rate of inventory loss from the primary system is such that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the SIT injection pressure or to recover core liquid level on HPSI flow. This tends to maximize the heatup time of the hot rod which produces the maximum PCT and local cladding oxidation. The limiting break case will either exhibit core recovery with the HPSI pumped injection alone while the primary system pressure remains barely above the SIT injection setpoint, or core recovery from SIT injection after an extended period of uncovery. For very small break sizes, the primary system pressure does not reach the SIT injection pressure; however, primary system inventory loss is not significant and typically within the means of HPSI makeup capacity such that core uncovery is minimal if not precluded.

3.2 *Analytical Methods*

The AREVA S-RELAP5 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 7), are incorporated. This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1 and as modified by Reference 2.

The two AREVA computer codes used in this analysis are:

1. The RODEX2-2A code (References 5 and 6) 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.

Representative system nodalization figures are shown in Figure 3-1 (RCS), Figure 3-2 (Secondary System), and Figure 3-3 (Reactor Vessel). Since these figures are representative, they may not contain all of the model details specific to Millstone Unit 2. For example, the charging system is not simulated in the SBLOCA analysis; therefore, the charging system noding diagram shown in Figure 3-1 is not used.

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*

Millstone Nuclear Plant Unit 2 is a CE-designed PWR with two hot legs, four cold legs, and two vertical U-tube SGs. The reactor has a core power of 2754 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 217 fuel assemblies. The hot legs connect to the reactor vessel with the vertical U-tube steam generators. Main feedwater (MFW) is injected into the downcomer of each SG. There are three AFW pumps, two motor-driven and one turbine (steam)-driven. The ECCS contains two HPSI pumps, two LPSI pumps, and four SITs.

The RCS was nodalized in the S-RELAP5 model with control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two 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 5.87% 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 Appendix K.

The analysis assumed a loss-of-offsite power concurrent with reactor SCRAM, which is based on the low pressurizer pressure reactor trip and includes delays for Reactor Protection System (RPS) circulation and Control Element Assembly (CEA) coil delay. The assumption of loss-of-offsite concurrent with reactor SCRAM results in RCP trip.

Tripping the reactor coolant pumps at the time of SCRAM instead of time zero is reasonable since a small delay relative to the time of loop seal uncovery for the limiting case is considered to be conservative due to the additional loss of primary system inventory through the break.

The single failure criterion required by 10 CFR 50 Appendix K (Reference 7) was satisfied by assuming the loss of one emergency diesel generator (EDG). Thus, this

results in the loss of one HPSI pump, one LPSI pump and one motor-driven AFW pump. The initiation of the HPSI and LPSI systems were delayed by 25 and 45 seconds, respectively, following safety injection actuation system (SIAS) activation.

Table 3-2 and Table 3-3 show the minimum ECCS flow rates with diesel generator failure for HPSI and LPSI, respectively. The HPSI system was modeled to deliver the highest SI flow to both legs of the broken loop (Loop 2A and Loop 2B (broken loop)). The LPSI system was modeled to deliver the highest SI flow to the broken loop (Loop 2B) and the lower SI flow to one of the intact legs in the intact loop (Loop 1A). Although the charging system is considered safety grade, it was not modeled in the analysis.

The disabling of a motor-driven AFW pump leaves one motor-driven pump and the turbine-driven pump available. The initiation of the motor-driven pump was delayed 240 seconds beyond the time of the auxiliary feedwater actuation signal (AFAS) indicating low SG level (0.0% narrow range). The turbine-driven AFW pump was not credited in the analysis.

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. The MSSVs were set to open at their nominal setpoints plus 3% tolerance.

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

3.4 *SER Compliance*

A spectrum of cold leg break sizes from 0.02182 ft^2 (2.0-inch diameter) to 0.49120 ft^2 (9.49-inch diameter, 10% of cold leg pipe area) was analyzed. This satisfies the limitation placed on EMF-2328 (Reference 1), that the methodology is acceptable for modeling transients where the break flow area is less than or equal to 10% of the cold

leg flow area. There is no other SER requirement or restriction on EMF-2328 (Reference 1).

In addition, the reactor coolant pump trip sensitivity study, attached piping break study and safety injection low liquid temperature study have been performed to support the operation of Millstone Nuclear Plant Unit 2 with M5® cladding.

Table 3-1 System Parameters and Initial Conditions

Reactor Power, MWt	2754[1]
Axial Power Shape	Figure 3-4
Peak LHR, kW/ft	15.1
Radial Peaking Factor (1.69 plus uncertainties)	1.854
RCS Flow Rate, gpm	360,000
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	549
SIT Pressure, psia	214.7
SIT Fluid Temperature, °F	120
SIT Water Volume, ft^3	1135
Maximum SG Tube Plugging Level per SG, %	5.87
SG Secondary Pressure, psia	880
MFW Temperature, °F	435
AFW Flow Rate per SG, gpm	72
AFW Temperature, °F	70
Low-Low SG Level Setpoint, % Narrow Range Span	0
AFW Delay, sec	240
HPSI and LPSI Fluid Temperature, °F	140
Pressurizer Pressure – Low Reactor Trip Setpoint (RPS), psia	1700
Reactor Trip Delay Time on Low Pressurizer Pressure, sec	0.9
SCRAM CEA Holding Coil Release Delay Time, sec	0.5
SIAS Activation Pressurizer Pressure Setpoint (Harsh Environment Conditions), psia	1500
HPSI Pump Delay Time on SIAS, sec	25
LPSI Pump Delay Time on SIAS, sec	45
MSSV Lift Pressure and Tolerance	Nominal + 3%

[1] Includes 2.0% measurement uncertainty

Table 3-2 High Pressure Safety Injection Flow Rates for Cold Leg Breaks

RCS Cold Leg Pressure (psia)	Loop 1A (lbm/s)	Loop 1B (lbm/s)	Loop 2A (lbm/s)	Loop 2B (lbm/s)
14.7	19.555	19.418	19.828	19.828
50	19.281	19.144	19.418	19.418
100	18.871	18.871	19.008	19.008
150	18.461	18.461	18.461	18.461
200	18.050	18.050	18.050	18.050
300	17.093	17.093	17.093	17.093
500	14.905	14.905	14.905	14.905
700	12.581	12.581	12.581	12.581
900	9.709	9.709	9.709	9.709
1000	8.068	8.068	8.068	8.068
1050	7.111	7.111	7.111	7.111
1100	5.880	5.880	5.880	5.880
1150	4.102	4.102	4.102	4.102
1190	2.051	2.051	2.051	2.051
1204	0.0	0.0	0.0	0.0

Table 3-3 Low Pressure Injection Flow Rates for Cold Leg Breaks

RCS Cold Leg Pressure (psia)	Intact Loop 1A (lbm/s)	Intact Loop 1B (lbm/s)	Intact Loop 2A (lbm/s)	Broken Loop 2B (lbm/s)
14.7	179.683	0.0	0.0	187.204
50	159.171	0.0	0.0	166.008
100	123.617	0.0	0.0	129.224
150	70.971	0.0	0.0	74.663
200	0.0	0.0	0.0	0.0

Figure 3-4 Axial Power Distribution Comparison

4.0 ANALYTICAL RESULTS

The analysis results demonstrate the adequacy of the ECCS to support the criteria given in 10 CFR 50.46(b)(1-4) for Millstone Nuclear Plant Unit 2 operating with AREVA supplied Advanced CE14 Fuel with M5® cladding.

Section 4.1 describes the SBLOCA break spectrum for the cold leg break. Section 4.2 describes the event for the limiting break size. Section 4.3 discusses the delayed RCP trip sensitivity study. Section 4.4 discusses the attached piping break sensitivity study. Section 4.5 discusses the safety injection low fluid temperature sensitivity study.

4.1 *Results for Break Spectrum*

The Millstone Nuclear Plant Unit 2 break spectrum analysis for SBLOCA includes breaks of varying diameter up to 10% of the flow area for the cold leg. The spectrum includes a wide enough range of break sizes from 2.0 inch diameter to 9.49 inch diameters to establish a PCT trend. Additional break sizes are performed with a smaller break interval once the potential limiting break size is determined to confirm the limiting break size. Figure 4-1 shows the calculated PCTs for these breaks. For the break spectrum analysis, RCP trip is assumed to occur on reactor SCRAM.

The results of the cold leg SBLOCA break spectrum analysis are presented in Table 4-1. The predicted event times for the break spectrum are provided in Table 4-2. The limiting break size was determined to be 3.78-inch diameter (0.07793 ft^2), resulting in a PCT of 1707°F.

4.2 *Discussion of Transient for Limiting Break*

The break opens at t=0 seconds and initiates a subcooled depressurization of the primary system. The low pressurizer pressure trip setpoint is reached at 19 seconds

and within 2 seconds the reactor is tripped then scrammed, the offsite power is lost, coincident with the turbine trip, RCP trip, and MFW pump trip (Figure 4-2, Figure 4-11, Figure 4-12 and Table 4-2). The SIAS is issued at 27 seconds on low pressurizer pressure. As MFW to the SGs is ramped down, the pressure in the SGs increase for approximately 30 seconds until MSSV inlet reaches the lowest opening pressure setpoint. This provides core heat removal in the early stages of the transient.

The primary system depressurization continues at a relatively fast rate for the first 125 seconds as fluid rushes out of the break (Figure 4-3). The primary side pressure continues to decrease, reaching that of the secondary side at approximately 300 seconds, thus ending SG secondary side inventory reduction and producing SG secondary side condensation (Figure 4-14). Shortly thereafter, the broken leg loop seal clears (356 seconds, Figure 4-6), and the primary side pressure reaches the saturation pressure of the fluid in the hot legs and reactor vessel upper plenum (366 seconds), which results in the fluid beginning to flash to steam as demonstrated in the horizontal loop seals and break void fractions in Figure 4-6 and Figure 4-5, respectively.

Prior to loop seal clearing in the broken leg, the core uncovers about 4 feet below the top of the active fuel (Figure 4-9, Figure 4-20 and Figure 4-21). As there is no loop flow, a large amount of steam is generated and accumulated in the core by the decay heat power until enough pressure is built to blow the upflow leg of the loop seal in the broken leg around 356 seconds into the transient. This causes an abrupt level drop in the downcomer region (Figure 4-8) with a simultaneous core recovery (Figure 4-9). As the broken leg clears, the plant then enters a fairly slow boil-off phase where mass is lost out the break, and the primary system continues to empty. All intact loops remained plugged for the duration of the transient.

As liquid drains out of the loop piping, the break flow transitions from liquid to two phase flow, and then to steam. The break flow becomes primarily steam around 366 seconds resulting in a reduced mass flow rate out of the break (Figure 4-4) and an increase in the depressurization rate of the primary system (Figure 4-3). The liquid level in the

reactor vessel continues to drop until the reactor vessel reaches a minimum level at 1386 seconds (Figure 4-10).

Although HPSI flow to the primary system cold legs began at approximately 62 seconds into the transient (Figure 4-16), it does not provide sufficient inventory at this time to offset the large amounts lost out the break at this time. As effective cooling is lost in the core, the fuel rods begin to heat up at approximately 800 seconds (Figure 4-22). The fuel continues to heat up until the maximum PCT of 1707°F is reached at 1824 seconds. Fuel rod rupture does occur for the hot rod, but the calculated blockage factor indicates that the channel around the hot rod is not completely blocked and that all other channels in the core are also not completely blocked. Therefore, the hot rod and all other channels in the core are amenable to cooling.

For this break size, the HPSI flow is eventually sufficient to compensate for the rate of inventory loss out of the break. At the time of the PCT, the primary system has depressurized to a pressure slightly above the SIT pressure and the LPSI shut-off head. SIT injection begins at 4580 seconds (Figure 4-18), followed by LPSI injection 56 seconds later (Figure 4-17), resulting in no influence on PCT turnaround.

The downcomer level (Figure 4-8) and the reactor vessel inventory (Figure 4-10) start slowly increasing at approximately 1400 seconds. The onset of SIT and LPSI injection helps the reactor vessel levels to step up, ensuring core recovery and long term core cooling.

In conclusion, the limiting PCT break spectrum case is a 3.78-inch diameter cold leg break. The PCT of this case is 1707°F. The maximum local oxidation is 3.6% and the maximum core-wide oxidation is less than 0.04%. The total maximum local oxidation is less than 6%, including a pre-transient oxidation of 2.3%. The hot rod resulted in rupture, but remained amenable to cooling. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 4).

4.3 *RCP Trip Sensitivity Study*

For plants such as Millstone Nuclear Plant Unit 2 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 SCRAM. 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. Therefore, an RCP trip sensitivity for both the cold and hot leg breaks was performed with delayed time following loss of subcooling margin to demonstrate 10 CFR 50.46(b)(1-4) criteria (Reference 4). This manual RCP trip study was performed consistent with the Combustion Engineering Owners Group guidelines described in Generic Letter 86-06 (Reference 8) where compliance with 10 CFR 50.46 is demonstrated when operator action to trip the RCPs is taken within 2 minutes after the RCP trip criterion is reached using the 10 CFR 50 Appendix K (Reference 7) method.

Also, consistent with Generic Letter 86-06, additional delayed RCP trip sensitivity studies were performed to find out the maximum delay time for operator action under a more realistic scenario. Best-estimate assumptions were applied using the same model as the Appendix K RCP analysis with relaxation in two areas: decay heat multiplier reduction from 1.2 to 1.0 and critical break flow model change from Moody to the Homogeneous Equilibrium Model. Relaxing these conservative parameters gives an analysis which is more representative of a realistic response. A range of RCP trip delay times was examined for both hot and cold leg break locations. The results demonstrated that longer delay times could be accommodated and still meet the 10 CFR 50.46 criteria.

Section 4.3.1 discusses the cold leg breaks RCP trip sensitivity study. Section 4.3.2 discusses the hot leg breaks sensitivity study.

4.3.1 Cold Leg Breaks RCP Trip Sensitivity Study

For the cold leg break RCP trip sensitivity study, the spectrum of break sizes from 2.0 inches to 9.49 inches in diameter were re-analyzed with a 2-minute RCP trip delay following loss of subcooling margin in the cold leg pump suction. The limiting cold leg break size for this sensitivity study was a 4.02-inch diameter break, producing a PCT of 1644°F. Therefore, the cold leg breaks RCP trip sensitivity study has shown that there is at least 2 minutes to trip all four RCPs after loss of subcooling margin in order to meet the 10 CFR 50.46(b)(1-4) criteria. In addition, relaxation of Appendix K assumptions demonstrated that longer delay times of up to 15 minutes after losing subcooling at the RCP suction could be accommodated and still meet the 10 CFR 50.46 criteria for a spectrum of cold leg break cases.

4.3.2 Hot Leg Breaks RCP Trip Sensitivity Study

For the hot leg break RCP trip sensitivity study, a spectrum of hot leg break sizes from 2.0 inches to 9.49 inches in diameter were analyzed with a 2-minute RCP trip delay following loss of subcooling margin in the cold leg pump suction. The limiting hot leg break size for this sensitivity study was the 5.0-inch diameter break, producing a PCT of 1575°F. Therefore, the hot leg breaks RCP trip sensitivity study has shown that there is at least 2 minutes to trip all four RCPs after loss of subcooling margin in order to meet the 10 CFR 50.46(b)(1-4) criteria. In addition, relaxation of Appendix K assumptions demonstrated that longer delay times of up to 10 minutes after losing subcooling at the RCP suction could be accommodated and still meet the 10 CFR 50.46 criteria for a spectrum of hot leg break cases.

4.4 *Attached Piping Break Sensitivity Study*

Although breaks in the attached piping are not typically PCT limiting, they do result in reduced ECCS flows available to mitigate the event. Therefore, an analysis of the limiting break size and location in an attached piping was performed. For Millstone Nuclear Plant Unit 2, the limiting break location and size for an attached piping is considered a double-ended guillotine break of an SIT line. The break was located in the SIT line connected to Loop 2B.

For the double-ended guillotine break in the SIT, the calculated PCT is 1239°F, which is bounded by the limiting PCT of the break spectrum. The minimal HPSI and LPSI flow rates modeled were sufficient to prevent a subsequent heatup after the initial quench from the SIT discharge.

4.5 *Safety Injection Low Fluid Temperature Sensitivity Study*

Table 4-1 Summary of SBLOCA Break Spectrum Results

Break diameter (in)	2.00	3.00	3.60	3.70	3.75	3.76
Break Area (ft^2)	0.02182	0.04909	0.07069	0.07467	0.07670	0.07711
Peak Clad Temperature (°F)	1135	1384	1472	1501	1599	1651
Time of PCT (sec)	4330	2451	2008	1978	1892	1857
Time of Rupture (sec)	--	--	--	--	1772	1654
Transient Local Maximum Oxidation (%)	0.0864	0.4249	0.6647	0.7543	1.9966	2.6718
Total Local Maximum Oxidation (%)[1]	2.337	2.676	2.915	3.005	4.247	4.922
Core Wide Oxidation (%)	0.0042	0.0111	0.0124	0.0139	0.0254	0.0314
PCT Elevation (ft)	10.52	10.52	10.77	10.77	11.02	11.02

Break diameter (in)	**3.78**	3.785	3.79	3.90	4.02	4.40
Break Area (ft^2)	**0.07793**	0.07814	0.07834	0.08296	0.08814	0.10559
Peak Clad Temperature (°F)	**1707**	1690	1476	1606	1596	1581
Time of PCT (sec)	**1824**	1852	2035	1954	1746	1284
Time of Rupture (sec)	**1563**	1591	--	1852	1684	1270
Transient Local Maximum Oxidation (%)	**3.5273**	3.2474	0.6689	1.3276	1.1046	0.6721
Total Local Maximum Oxidation (%)[1]	**5.778**	5.498	2.920	3.578	3.355	2.923
Core Wide Oxidation (%)	**0.0396**	0.0368	0.0119	0.0175	0.0162	0.0110
PCT Elevation (ft)	**11.02**	11.02	10.77	11.02	11.02	10.52

Break diameter (in)	4.60	4.80	5.00	5.30	5.50	6.00
Break Area (ft^2)	0.11541	0.12566	0.13635	0.15321	0.16499	0.19635
Peak Clad Temperature (°F)	1621	1637	1615	1591	1398	1535
Time of PCT (sec)	1076	939	832	699	674	501
Time of Rupture (sec)	1047	907	814	691	--	--
Transient Local Maximum Oxidation (%)	0.7874	0.8477	0.6456	0.5072	0.1142	0.2316
Total Local Maximum Oxidation (%)[1]	3.038	3.098	2.896	2.758	2.365	2.482
Core Wide Oxidation (%)	0.0108	0.0111	0.0091	0.0075	0.0020	0.0051
PCT Elevation (ft)	10.77	10.77	10.52	10.52	10.27	10. 27

Break diameter (in)	7.00	8.00	9.00	9.49
Break Area (ft^2)	0.26725	0.34907	0.44179	0.49120
Peak Clad Temperature (°F)	1510	1540	1399	1437
Time of PCT (sec)	353	258	202	180
Time of Rupture (sec)	--	--	--	--
Transient Local Maximum Oxidation (%)	0.2133	0.2247	0.1097	0.1367
Total Local Maximum Oxidation (%)[1]	2.464	2.475	2.360	2.387
Core Wide Oxidation (%)	0.0049	0.0046	0.0018	0.0024
PCT Elevation (ft)	10. 27	10. 27	10.02	10.02

[1] Includes the pre-transient oxidation of 2.2507%

Table 4-2 Sequence of Events for Break Spectrum (seconds)

Break diameter (in)	PCT (°F)	Break opens	Low PZR Pressure Trip	Reactor Scram, RCP and Turbine Trip	SIAS issued	AFAS Available	HPSI available	LPSI available	HPSI Flow Begins	LPSI Flow Begins	Motor driven AFW on, Initial	Loop seal 1A clears	Loop seal 1B clears	Loop seal 2A clears	Loop seal 2B clears	Break uncovers	SIT injection begins	Minimum RV mass occurs	Hot Rod rupture occurs	PCT occurs	Time of core uncovery	Non-condensable gas at the break
2.00	1135	0	66	67	79	90	104	124	242	--	516	--	--	--	1088	1100	--	4116	--	4330	3030	--
3.00	1384	0	29	30	38	54	63	83	98	--	418	--	--	--	512	552	--	2082	--	2451	280	--
3.60	1472	0	20	22	29	46	54	74	68	--	414	--	--	--	384	394	5700	1638	--	2008	208	--
3.70	1501	0	19	21	27	46	52	72	64	--	412	--	--	--	364	374	5102	1572	--	1978	200	--
3.75	1599	0	19	20	27	46	52	72	64	--	422	--	--	--	360	370	4660	1458	1772	1892	190	--
3.76	1651	0	19	20	27	46	52	72	62	--	420	--	--	--	358	368	4638	1428	1654	1857	188	--
3.78	**1707**	**0**	**19**	**20**	**27**	**44**	**52**	**72**	**62**	--	**408**	--	--	--	**356**	**366**	**4580**	**1386**	**1563**	**1824**	**186**	**--**
3.785	1690	0	19	20	27	44	52	72	62	--	408	--	--	--	356	366	4534	1396	1591	1852	186	--
3.79	1476	0	18	20	26	44	51	71	62	--	408	944	--	1950	1330	374	--	1680	--	2035	186	--
3.90	1606	0	18	19	25	44	50	70	58	--	434	--	366	2340	1176	350	1944	1598	1852	1954	192	--
4.02	1596	0	17	18	24	42	49	69	56	--	430	700	--	1250	1278	340	1738	1452	1684	1746	184	--
4.40	1581	0	14	16	22	40	47	67	48	--	442	600	--	1230	954	292	1278	1230	1270	1284	154	--
4.60	1621	0	13	15	20	40	45	65	46	--	514	250	250	--	478	272	1068	1072	1047	1076	144	--
4.80	1637	0	13	14	19	40	44	64	46	956	--	--	480	--	574	260	934	936	907	939	136	--
5.00	1615	0	12	13	19	38	44	64	44	848	--	--	430	710	506	260	824	830	814	832	128	--
5.30	1591	0	11	13	18	38	43	63	44	714	--	--	290	250	224	232	692	698	691	699	118	--
5.50	1398	0	11	12	17	38	42	62	42	692	--	436	204	270	190	212	670	674	--	674	112	--
6.00	1535	0	10	12	16	38	41	61	42	514	--	--	186	180	180	188	496	500	--	501	102	--
7.00	1510	0	9	11	14	38	39	59	40	360	--	258	264	180	112	146	350	352	--	353	74	--
8.00	1540	0	9	10	13	218	38	58	40	264	--	--	98	116	86	114	254	256	--	258	56	--
9.00	1399	0	9	10	13	--	38	58	38	206	--	70	86	66	102	84	198	202	--	202	46	288
9.49	1437	0	9	10	12	--	37	57	38	184	--	74	76	60	84	78	178	180	--	180	42	--

Figure 4-1 Peak Cladding Temperature versus Break Size (SBLOCA Break Spectrum)

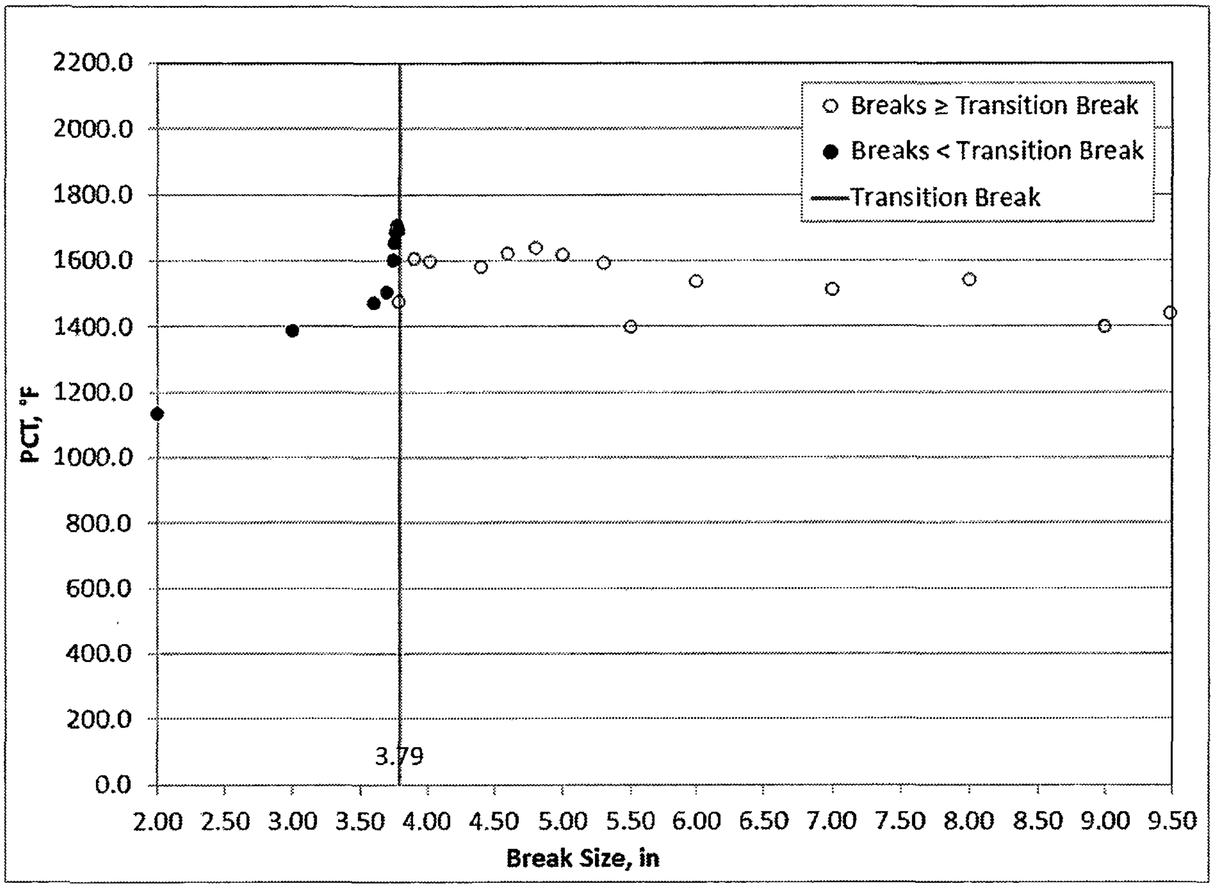


Figure 4-2 Reactor Power – 3.78-inch Break

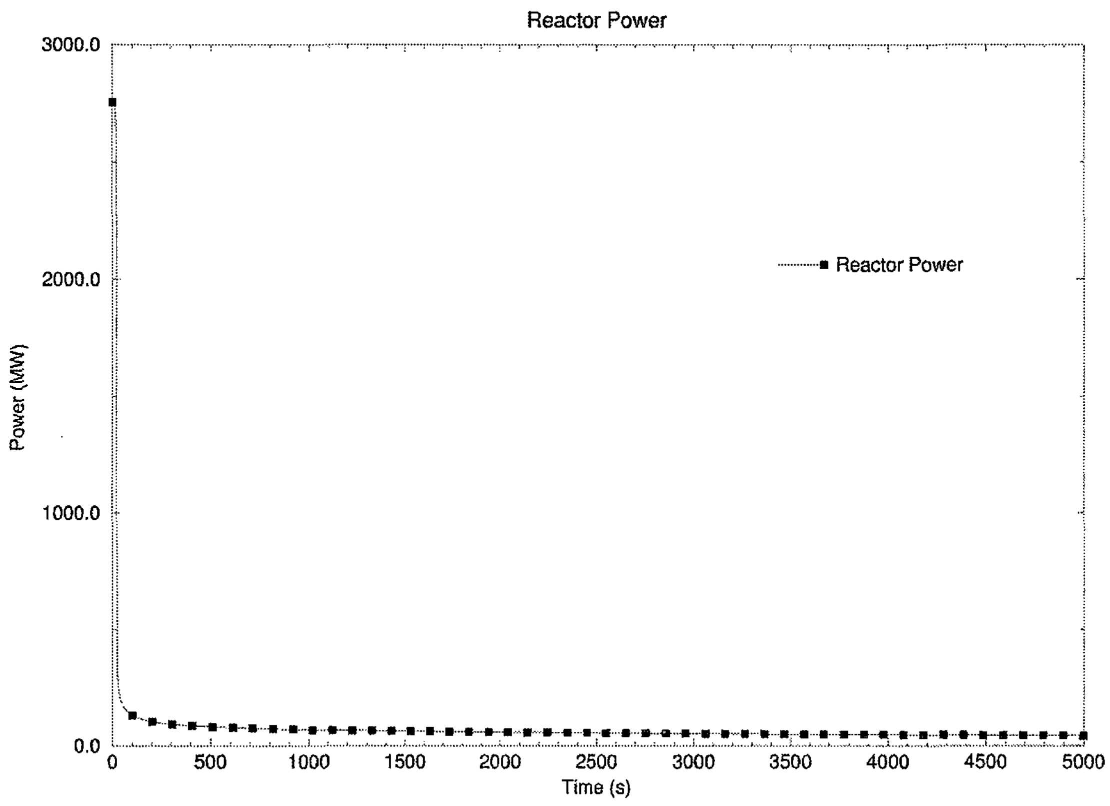


Figure 4-3 Primary and Secondary System Pressures – 3.78-inch Break

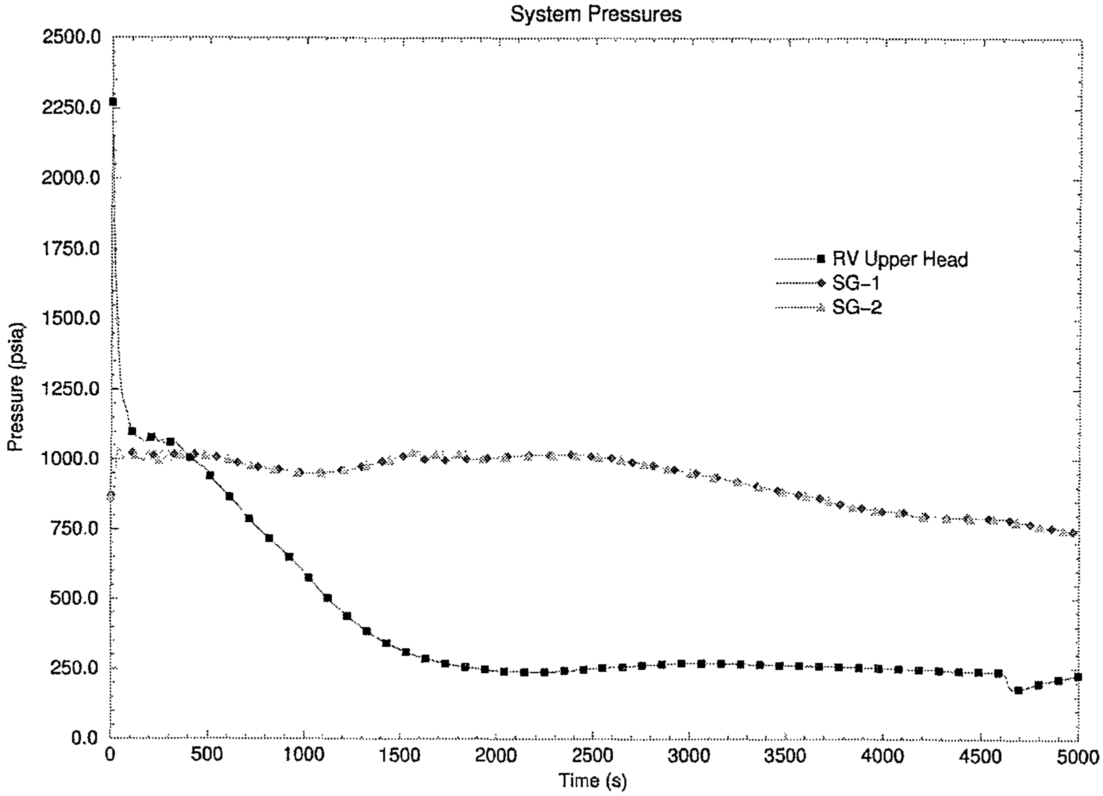


Figure 4-4 Break Mass Flow Rate – 3.78-inch Break

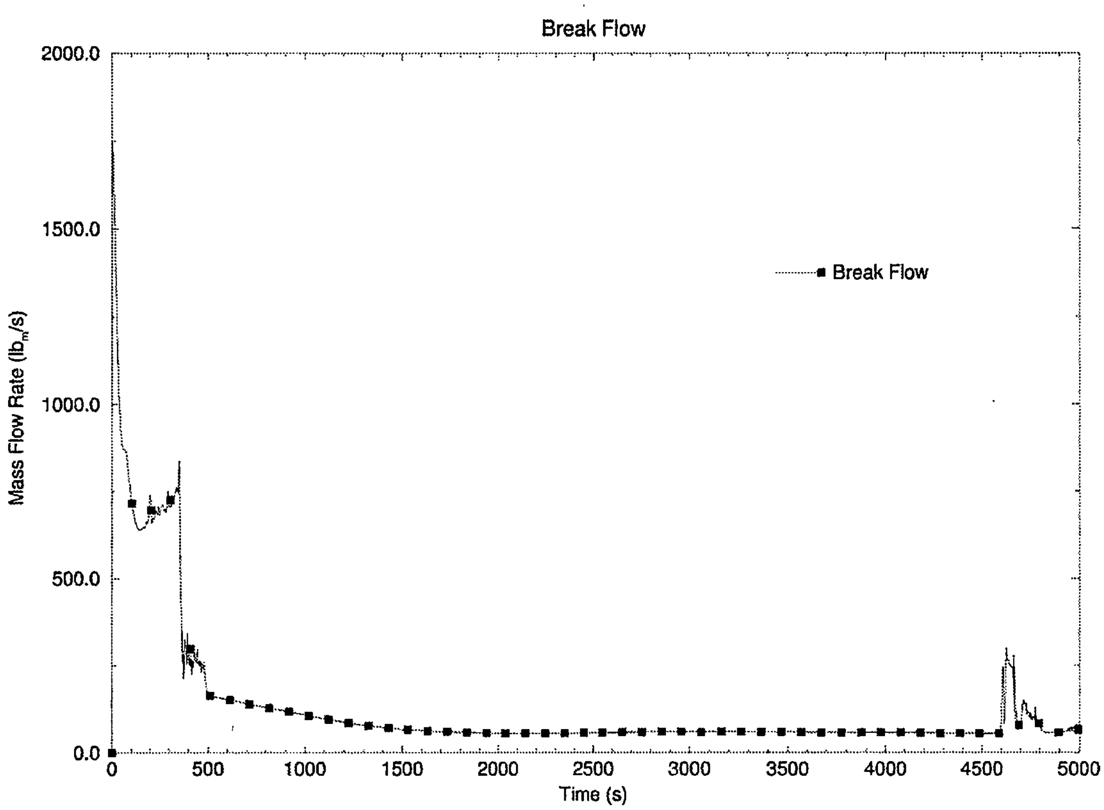


Figure 4-5 Break Vapor Void Fraction – 3.78-inch Break

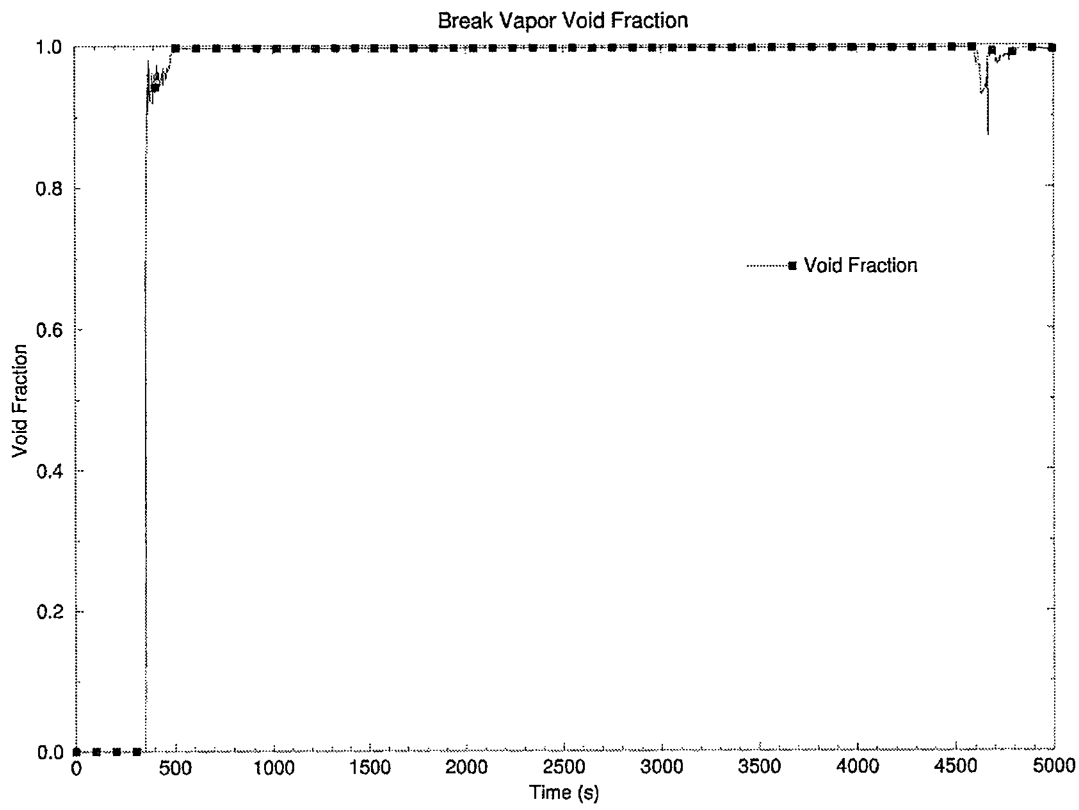


Figure 4-6 Loop Seal Void Fraction – 3.78-inch Break

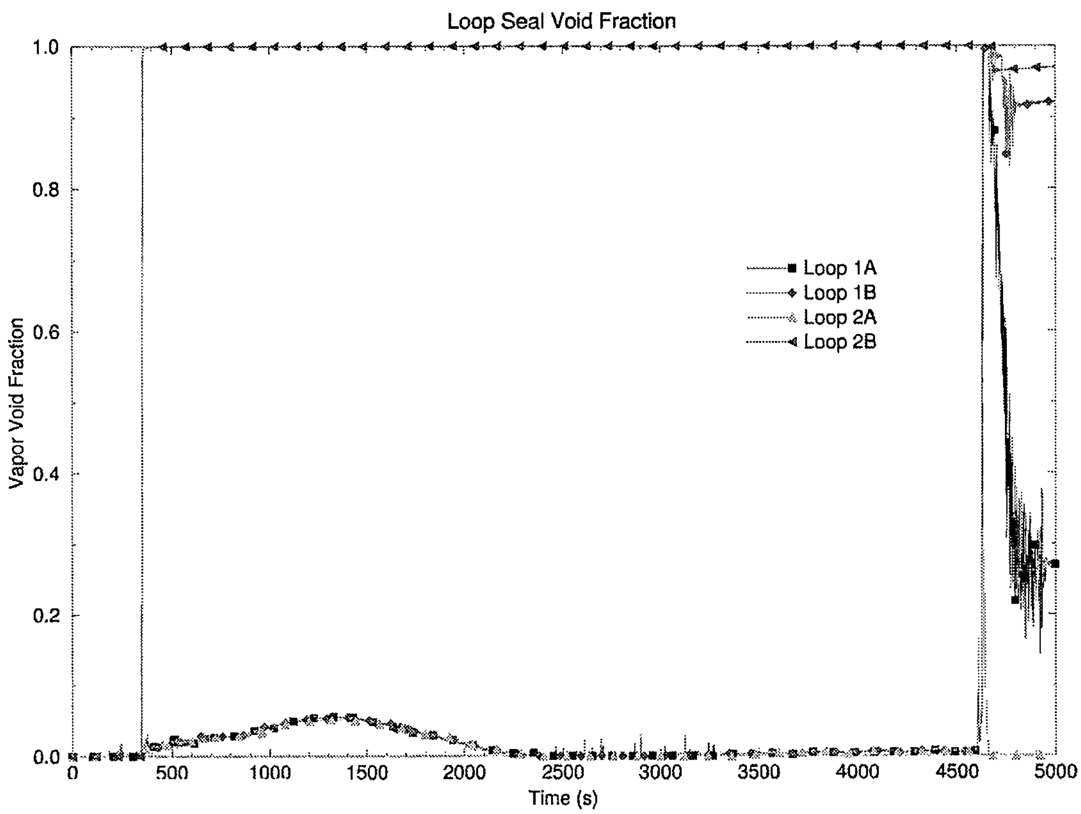


Figure 4-7 Total Core Inlet Mass Flow Rate – 3.78-inch Break

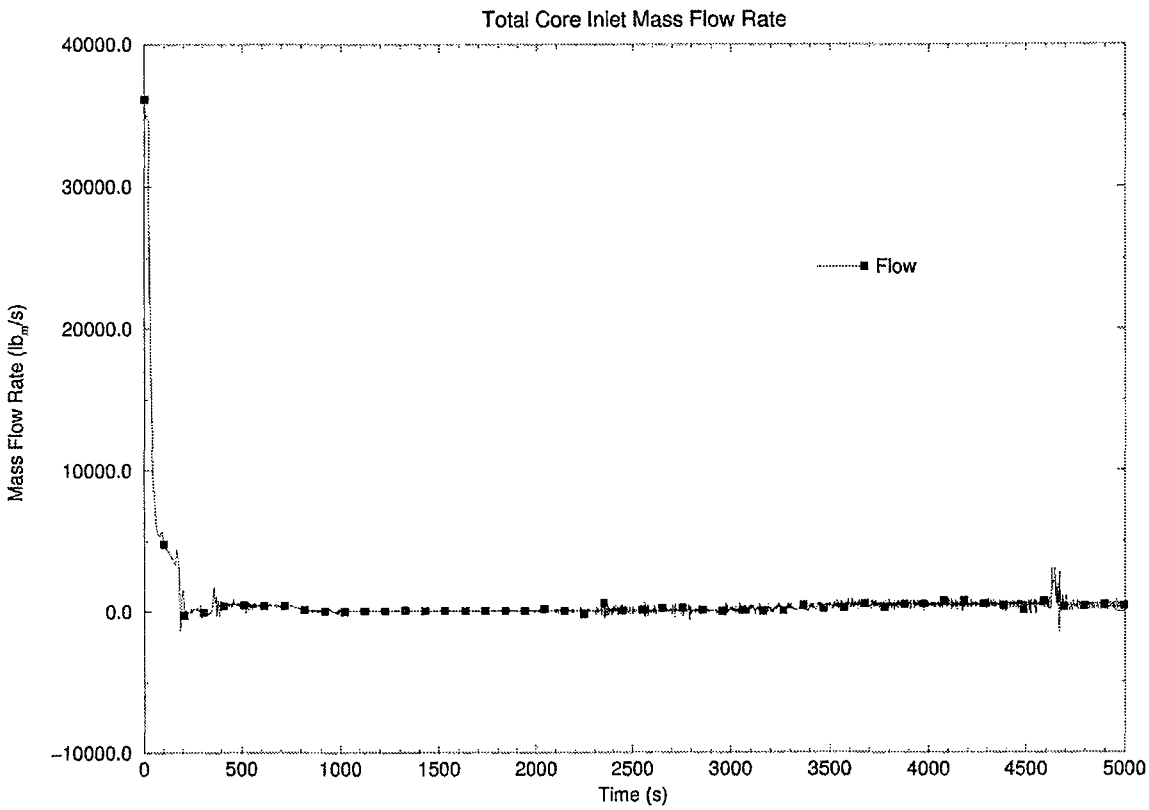


Figure 4-8 Downcomer Collapsed Liquid Level – 3.78-inch Break

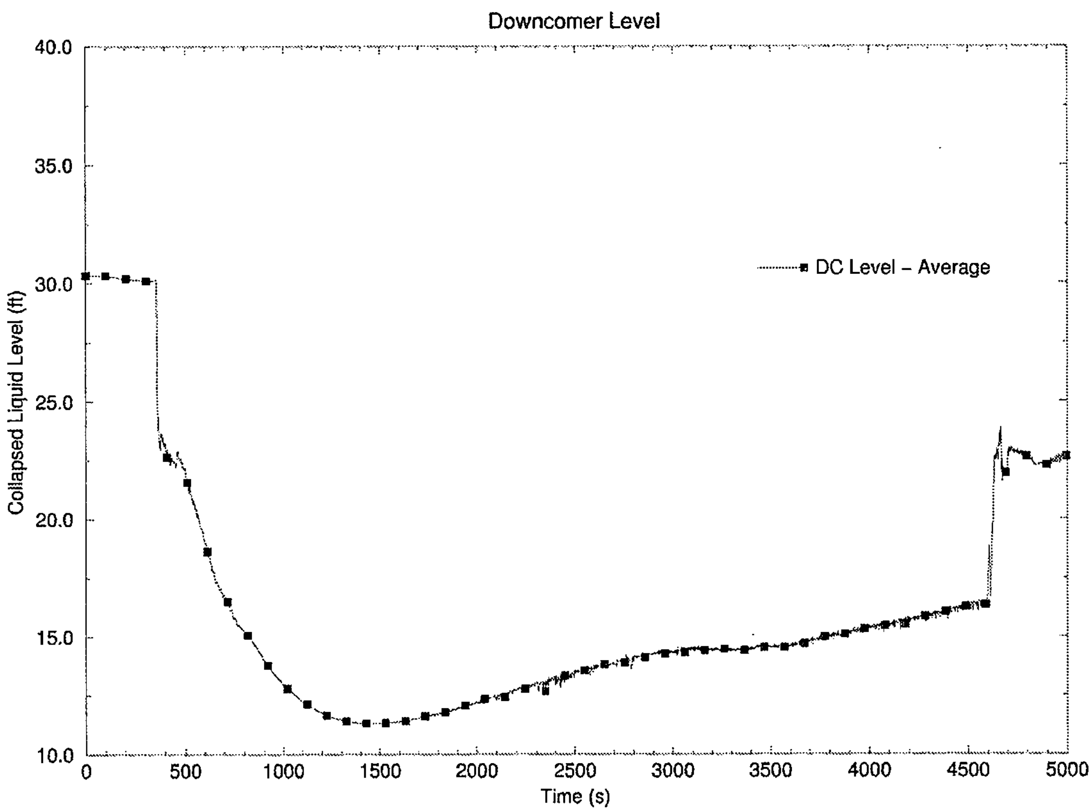


Figure 4-9 Inner and Outer Core Collapsed Liquid Level – 3.78-inch Break

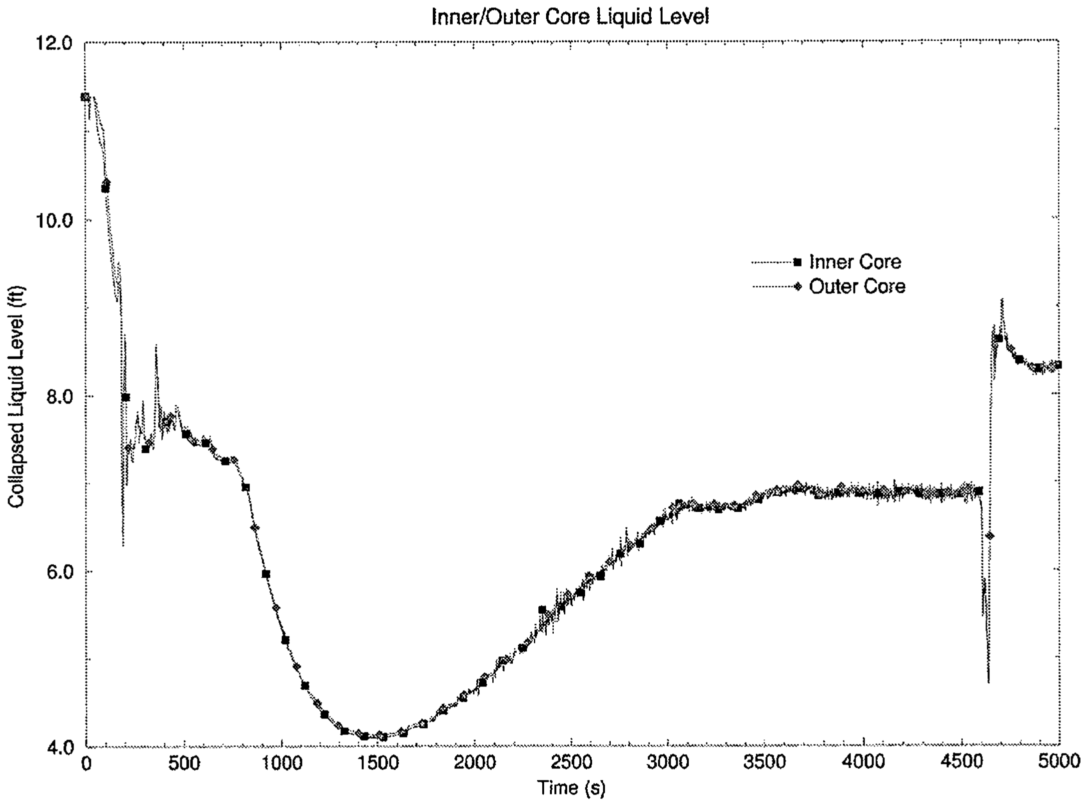


Figure 4-10 Reactor Vessel Mass – 3.78-inch Break

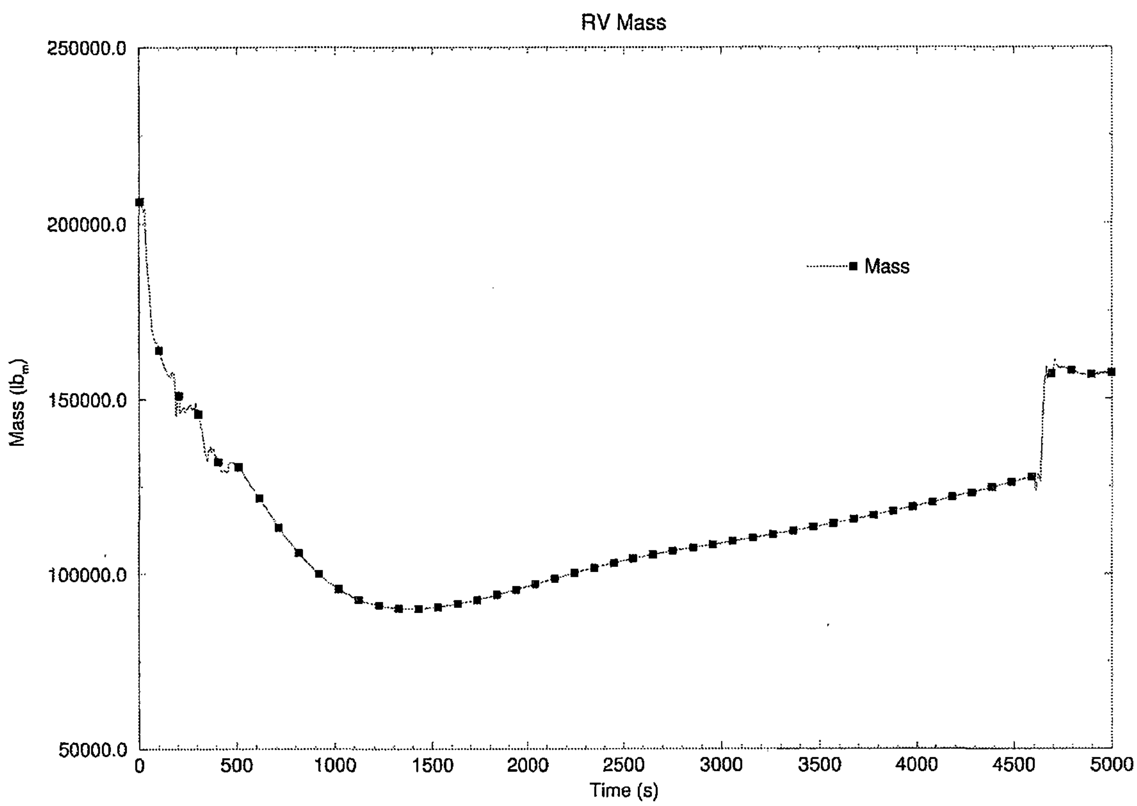


Figure 4-11 RCS Loop Mass Flow Rates – 3.78-inch Break

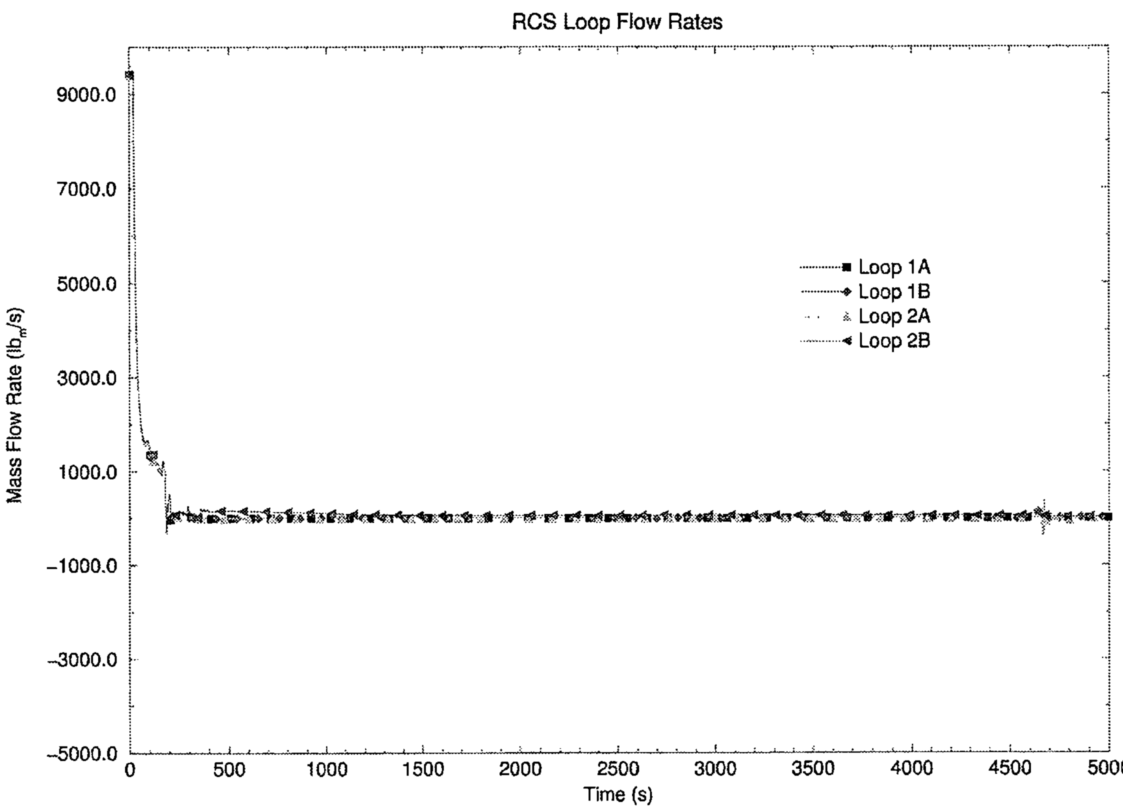


Figure 4-12 Steam Generator Main Feedwater Mass Flow Rates – 3.78-inch Break

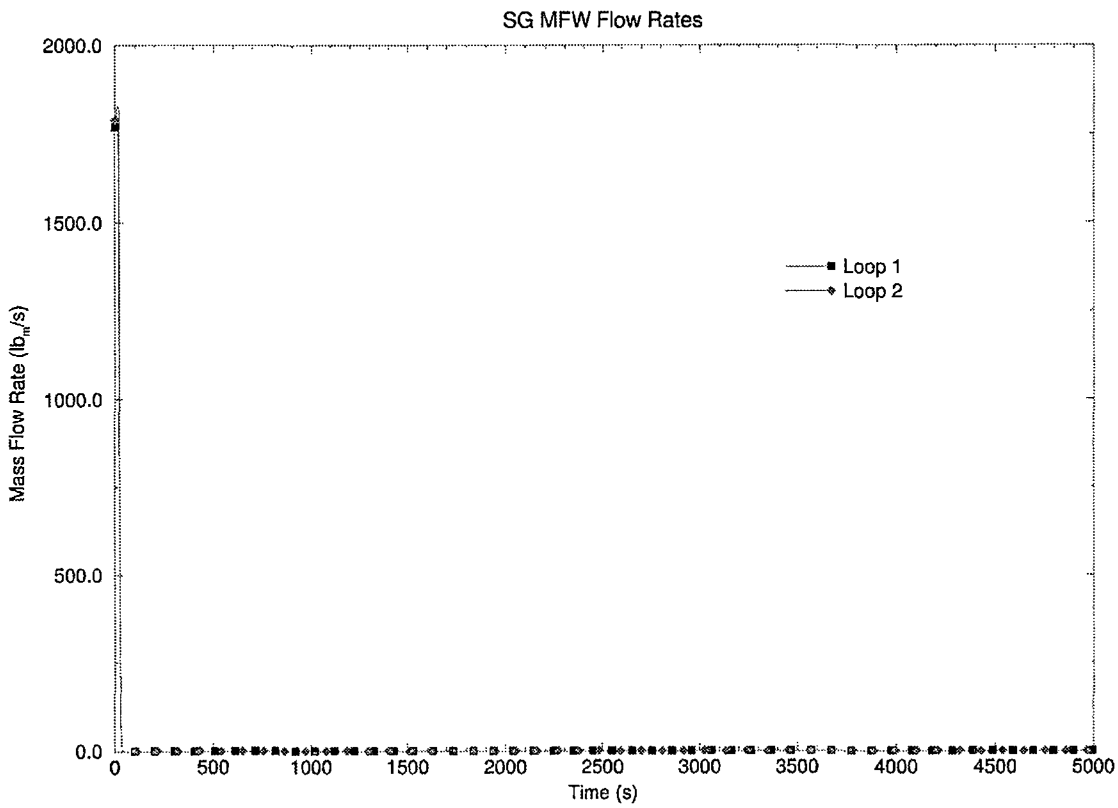


Figure 4-13 Steam Generator Auxiliary Feedwater Mass Flow Rates – 3.78-inch Break

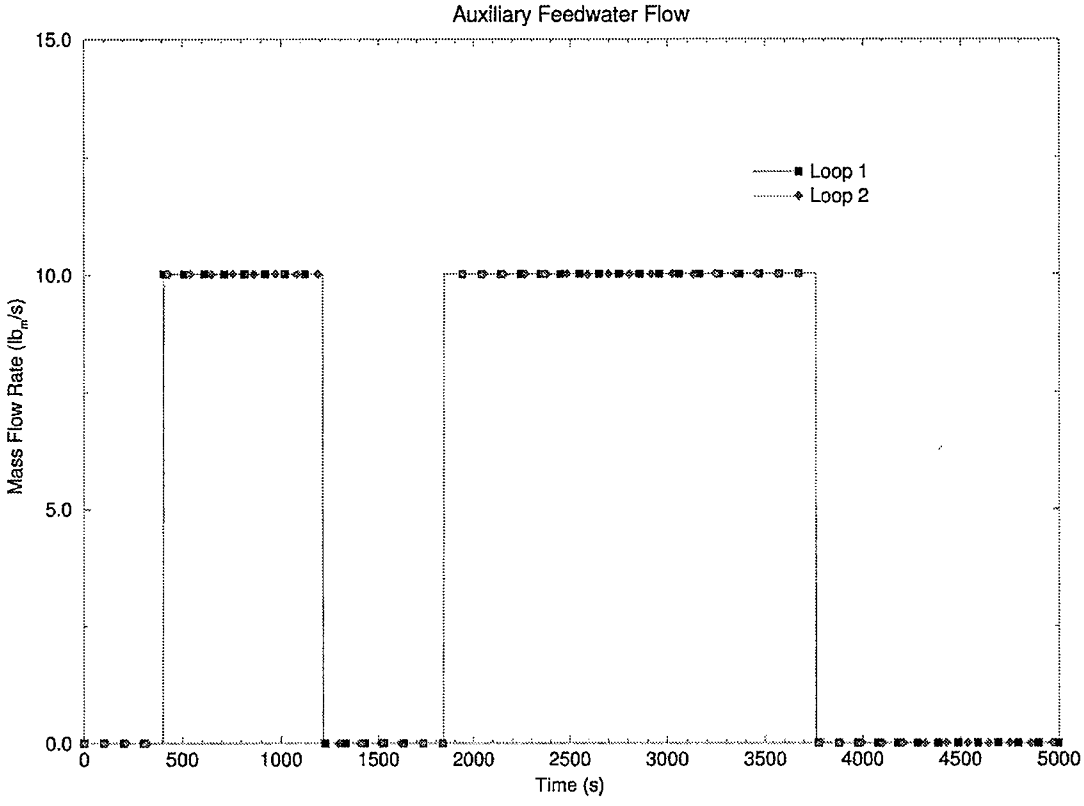


Figure 4-14 Steam Generator Total Mass – 3.78-inch Break

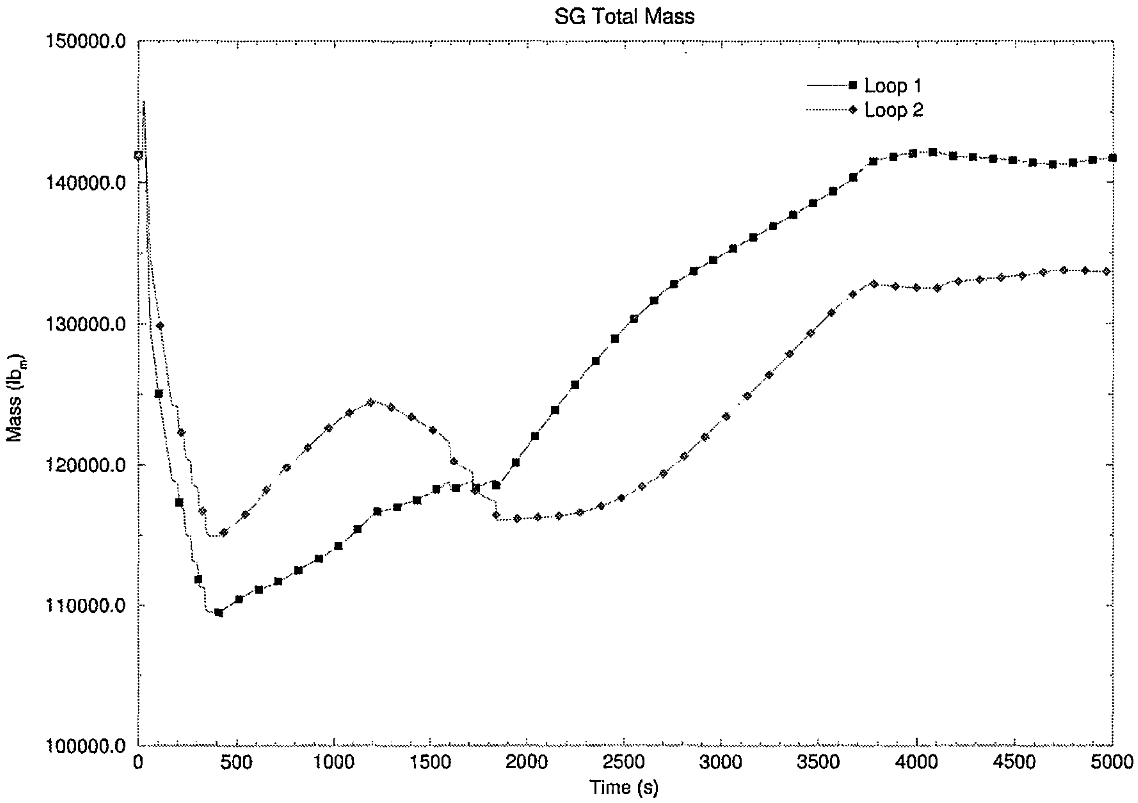


Figure 4-15 Steam Generator Narrow Range Level % – 3.78-inch Break

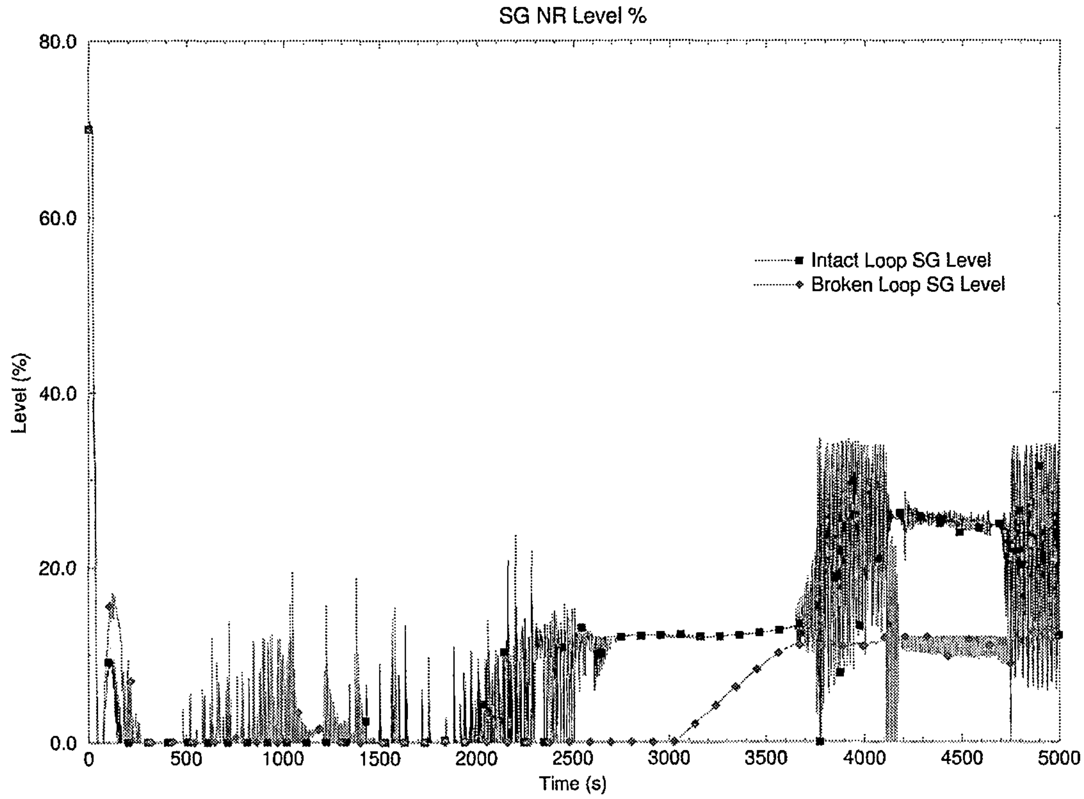


Figure 4-16 High Pressure Safety Injection Mass Flow Rates – 3.78-inch Break

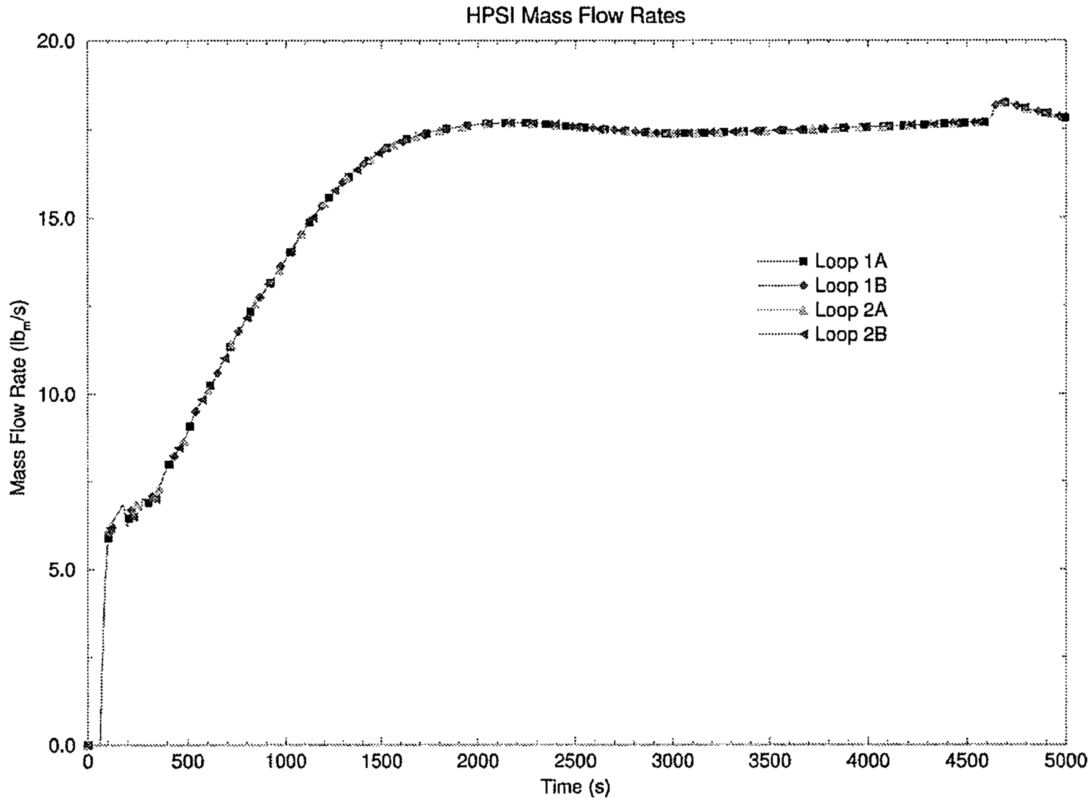


Figure 4-17 Low Pressure Safety Injection Mass Flow Rates – 3.78-inch Break

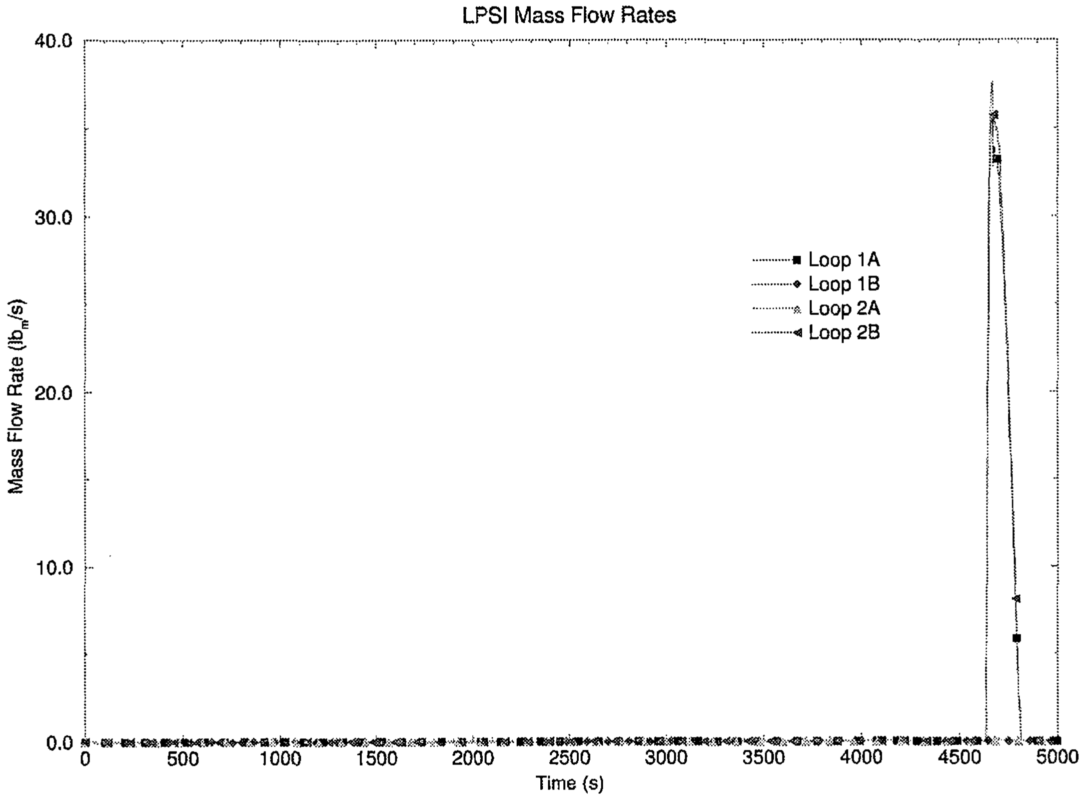


Figure 4-18 Safety Injection Tank Mass Flow Rates – 3.78-inch Break

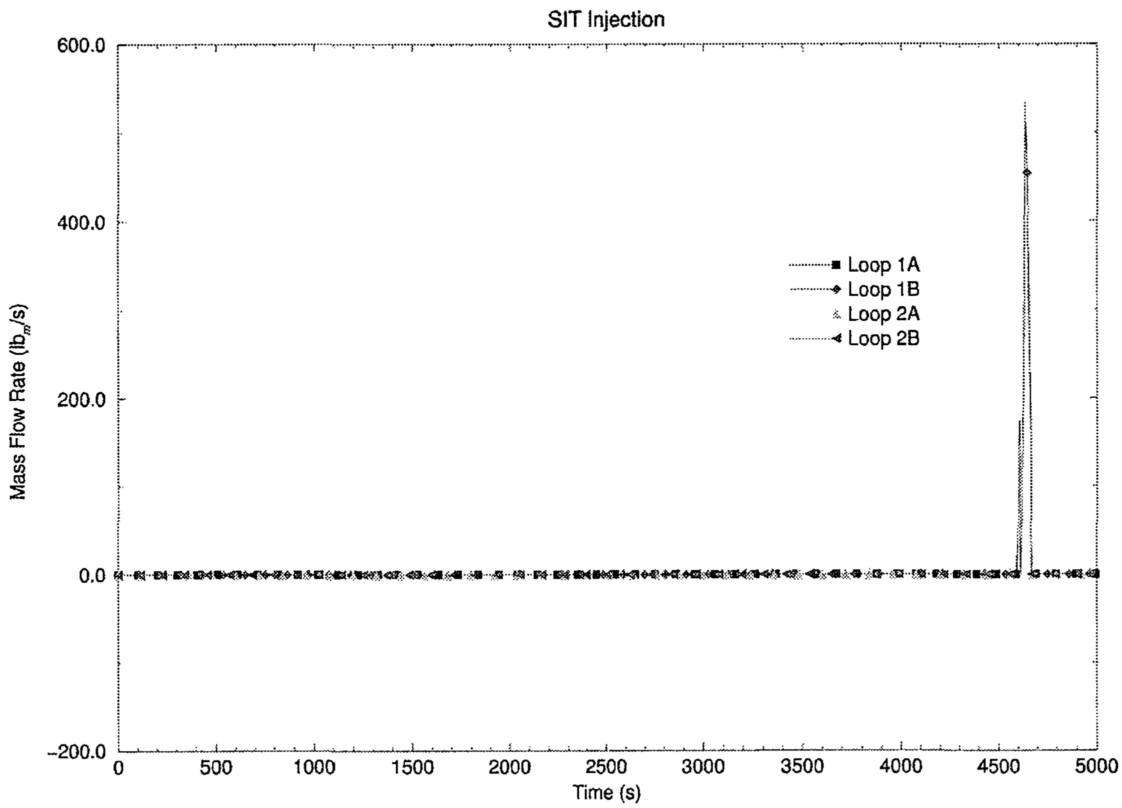


Figure 4-19 Integrated Break Flow and ECCS Flow – 3.78-inch Break

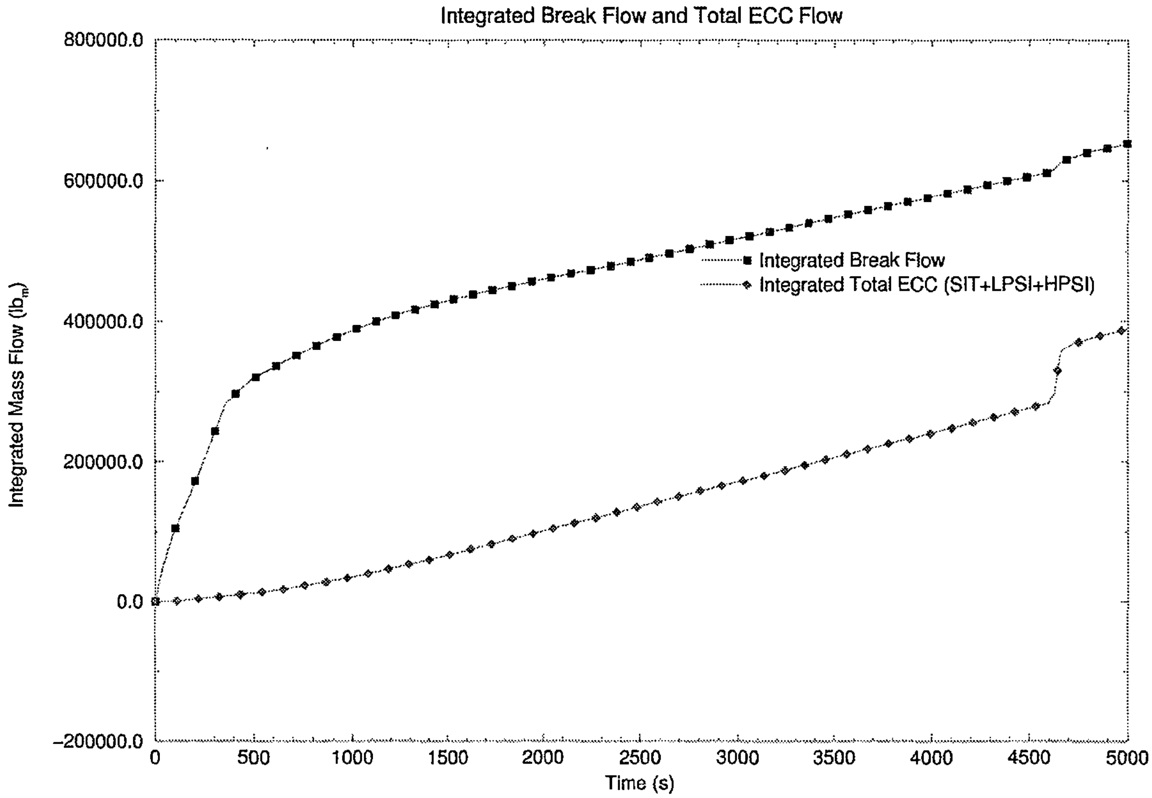


Figure 4-20 Hot Assembly Collapsed Liquid Level – 3.78-inch Break

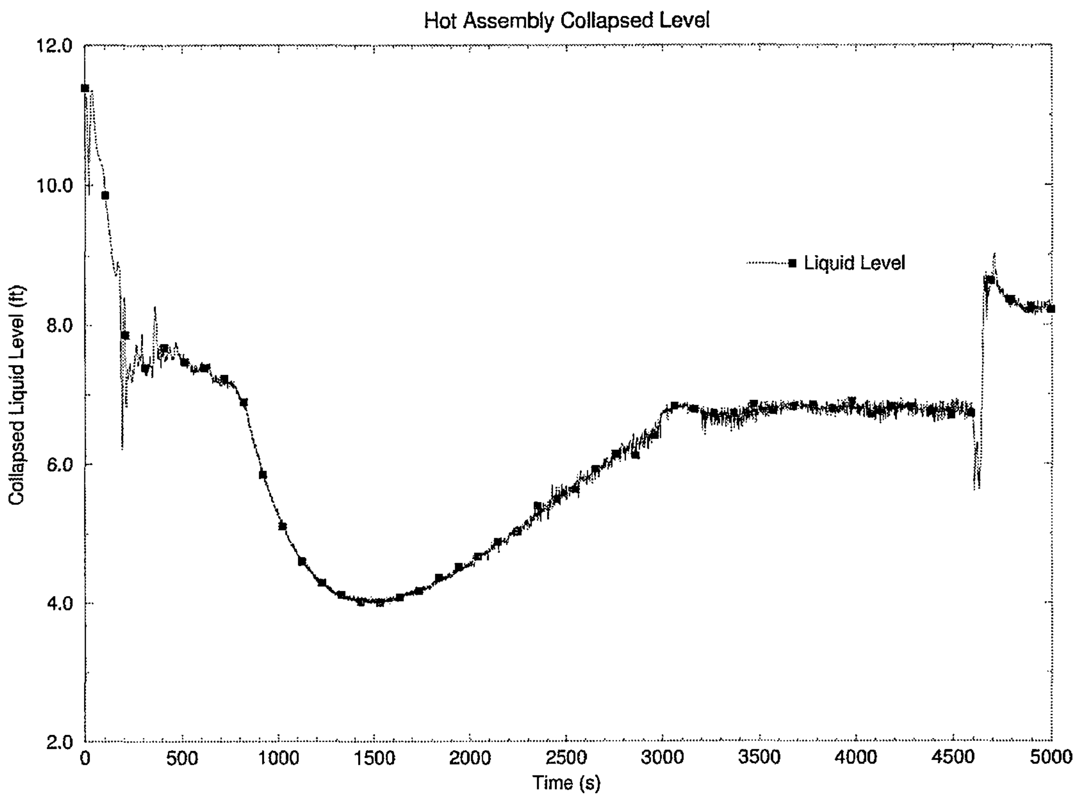


Figure 4-21 Hot Assembly Mixture Level – 3.78-inch Break

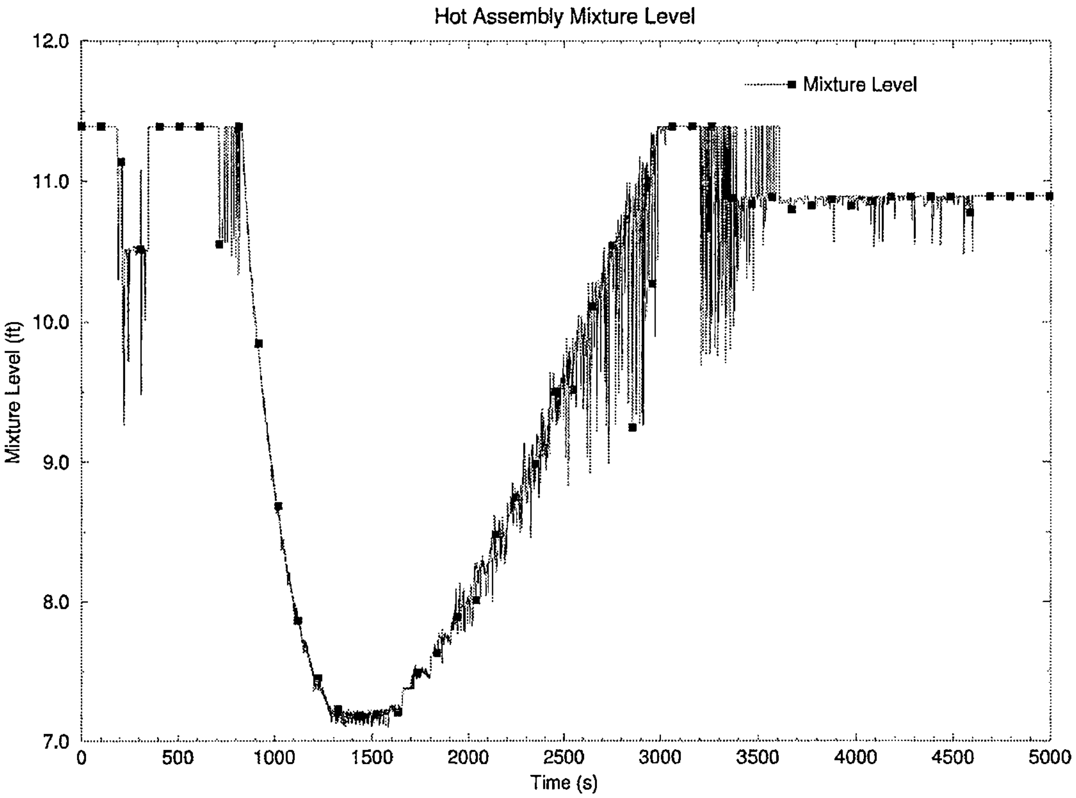
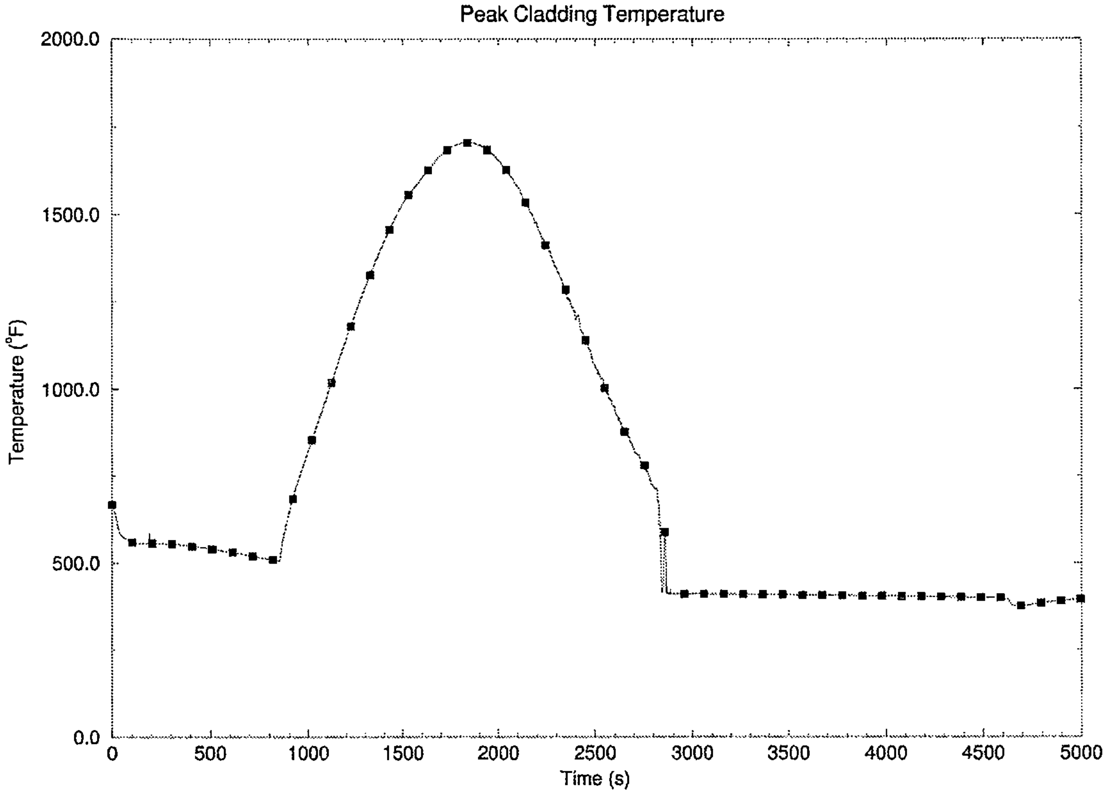


Figure 4-22 Peak Cladding Temperature at PCT Location (11.02 ft) – 3.78-inch Break



5.0 REFERENCES

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

2. AREVA Inc. Topical Report EMF-2328(P) Revision 0, Supplement 1, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, March 2012.

3. AREVA Inc. Topical Report BAW-10240(P)(A) Revision 0, *Incorporation of M5® Properties in Framatome ANP Approved Methods*, May 2004.

4. Code of Federal Regulations, Title 10, Part 50, Section 46, *Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors,* January 2010.

5. AREVA Inc. Topical Report XN-NF-81-58(P)(A) Revision 2, Supplements 1 and 2, *RODEX2 FUEL Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

6. AREVA Inc. Topical Report ANF-81-58(P)(A) Revision 2, Supplements 3 and 4, *RODEX2 FUEL Thermal-Mechanical Response Evaluation Model*, Advanced Nuclear Fuels Corporation, June 1990.

7. Code of Federal Regulations, Title 10, Part 50, Appendix K, *ECCS Evaluation Models,* March 2000.

8. Nuclear Regulatory Commission Generic Letter 86-06, Subject: *Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter No. 86-06)*, Mary 29, 1986.

Attachment 5

AREVA APPLICATION FOR WITHHOLDING AND AFFIDAVIT

**DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2**

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the Licensing Report ANP-3315P, Revision 0, entitled, "Millstone Unit 2 M5® Upgrade, Small Break LOCA Analysis," dated April 2015 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c) through 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 24th day of April, 2015.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018