St. Lucie Unit 2 Fuel Transition
Small Break LOCA Summary Report
ANP-3345NP, Revision 1

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AREVA

St. Lucie Unit 2 Fuel Transition Small Break LOCA Summary Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	Page 3-5	First paragraph proprietary marks have been removed. This paragraph does not contain proprietary information.
2	Section 4	Page numbering for entire section is corrected. Section 4 now starts at page 4-1. The Table of Contents, List of Tables, and List of Figures are also updated accordingly.

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Nomenclature

Acronym	Definition
ADV	Atmospheric dump valve
AFAS	Auxiliary feedwater actuation signal
AFW	Auxiliary feedwater
AOR	Analysis of record
AREVA	AREVA Inc.
BOC	Beginning-of-cycle
CE	Combustion Engineering
CEA	Control element assembly
CFR	Code of Federal Regulations
CRGT	Control rod guide tube
DC-HL	Downcomer – Hot Leg
DC-UH	Downcomer – Upper Head
ECCS	Emergency Core Cooling System
EDG	Emergency diesel generator
EM	Evaluation model
EOC	End-of-cycle
EOP	Emergency Operating Procedure
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
RAI	Requested for Additional Information
RCP	Reactor coolant pump
RCS	Reactor Coolant System
PCT	Peak cladding temperature
PWR	Pressurized water reactor
PZR	Pressurizer
SBLOCA	Small break loss-of-coolant accident
SER	Safety Evaluation Report
SG	Steam generator
SGTP	Steam generator tube plugging
SI	Safety injection
SIAS	Safety injection actuation signal
SIT	Safety injection tank

1.0 INTRODUCTION

The purpose of this report is to summarize the small break loss-of-coolant accident (SBLOCA) analysis performed with AREVA 16x16 HTP fuel with M5®[1] cladding for the St. Lucie Unit 2 plant. This document provides input to the License Amendment Request (LAR) in support of the transition to AREVA fuel in St. Lucie Unit 2. The SBLOCA analysis was performed in accordance with AREVA's S-RELAP5 SBLOCA methodology (References 1 and 2) and the additional considerations discussed in Section 3.2.

A complete spectrum of cold leg break sizes was considered, ranging from 2.0-inch diameter (0.022 ft^2) to 9.49-inch diameter (0.491ft^2) range. In addition, a sensitivity study was performed to consider attached piping break sensitivity.

The analysis supports plant operation at a core power level of 3029.06 MWt (including measurement uncertainty), a peak linear heat rate (LHR) of 13.0 kW/ft, a radial peaking factor of 1.65 (1.81 including uncertainty and an augmentation factor) and a steam generator tube plugging level of 20% with ±4% asymmetry.

A bounding total SIT line and check valve loss coefficient value is used in the SBLOCA analysis for each loop, including both major and minor loss components.

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1 M5 is a registered trademark of AREVA Inc.

2.0 SUMMARY OF RESULTS

A SBLOCA break spectrum analysis was performed for St. Lucie Unit 2 to 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 3), 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 peak cladding temperature (PCT) is 1926°F for a 2.70-inch diameter (0.040 ft^2) cold leg pump discharge break. The total maximum local oxidation is less than 9.0%, including a pre-transient oxidation of 2.3925% and transient maximum oxidation of less than 6.0%. The maximum core-wide oxygen generation is less than 0.2%. The results of the analysis demonstrate the adequacy of the ECCS to support the criteria given in 10 CFR 50.46(b).

In addition to the break spectrum analysis, a sensitivity study was performed to consider a break in an attached pipe. The break in an attached piping sensitivity study was performed with a 10.126-inch diameter (double-ended guillotine) break in the safety injection tank (SIT) line. The PCT calculated for this case was 1451°F. The SIT line break results are non-limiting compared to the break spectrum results.

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 St. Lucie Unit 2 plant 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 reactor coolant pump (RCP). This break location results in the largest amount of 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 3).

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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 safety injection actuation signal (SIAS) occurs when the system has further depressurized. The capacity and shutoff head of the HPSI pumps are important parameters in the SBLOCA analysis. For the limiting break size, 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. 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 injection of SIT flow. For very small break sizes, the primary system pressure does not reach the SIT injection pressure.

The SBLOCA event develops in the following distinct phases: (1) subcooled depressurization, (2) loop saturation and loop flow coastdown, (3) loss of loop circulation and reflux mode cooling, (4) loop seal clearing and core refill and (5) long-

term cooling provided by high and low head safety pump and SIT injections. The SBLOCA development phases are outlined in Table 3-1.

Following the break, the RCS rapidly depressurizes to the saturation pressure of the hot leg fluid. During the initial depressurization phase, reactor trip occurs on low pressurizer pressure, and 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[1].

In the second phase of the transient, the reactor coolant pumps coastdown. In this phase, natural circulation flows are sufficient to provide continuous core heat removal via the steam generators. However, mass continues to be lost to the break during this period.

The third phase in the transient is characterized as a period of loop draining that results in the loss of RCS flows. During this period, the core decay heat removal is provided via reflux boiling. The RCS stabilizes at an equilibrium pressure above the steam generator secondary side pressure. The system reaches a quiescent state in which the core decay heat, break flow, steam generator heat removal, and system hydrostatic head balance combine to control the core inventory.

The fourth phase in the transient is characterized by loop seal clearing and core recovery. The RCS inventory continues to decrease. Prior to loop seal clearing, liquid trapped in the reactor coolant pump suction piping can prevent steam from venting via the break. For a small break, the transient develops slowly, and liquid level in the reactor coolant system may descend to the loop seal level prior to establishing a steam vent. The core can become temporarily uncovered in this loop seal clearing process.

Once the loop seal clears, venting of steam through the break causes a rapid RCS depressurization below the secondary pressure and boiling in the core increases. The

[1] Tripping the reactor coolant pumps at the time of SCRAM instead of time zero is:
- A small delay relative to the time of loop seal uncovery for the limiting cases.
- Expected to be slightly conservative, due to the additional loss of primary system inventory through the break.

depressurization also promotes an increase in ECCS flows and the mass loss through the break decreases substantially as a result of phase change. These occurrences combine to cause either the core to uncover and heat up if ECCS flow is still not sufficient to offset the inventory lost out the break or an increase in RCS liquid inventory, preventing core uncovery, when ECCS flow is greater than the inventory lost out the break.

The last phase of the transient is characterized as a long-term cooling period during which the RCS inventory control is provided by the Emergency Core Cooling System. Pumped injection continues and the passive SIT injection occurs when the RCS pressure decreases below the SIT tank pressure. Long-term RCS inventory and decay heat removal are successfully controlled in this manner.

3.2 Analytical Models

The AREVA S-RELAP5 SBLOCA evaluation model (References 1) for event response of the primary and secondary systems and the hot fuel rod used in the analysis is based on the use 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 the analysis are:

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.

The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated into the SBLOCA EM methodology (Reference 1). This methodology has been reviewed and approved by the NRC to perform SBLOCA analyses. However, several modeling differences from the current SBLOCA EM methodology (Reference 1) were included to make the model consistent with the modifications requested in NRC Requests for Additional Information (RAIs).

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System nodalization details are shown in Figure 3-1 (RCS), Figure 3-2 (Secondary System), and Figure 3-3 (Reactor Vessel).

3.3 Plant Description and Summary of Analysis Input Parameters

St. Lucie Unit 2 is a Combustion Engineering (CE)-designed Pressurized Water Reactor (PWR) with two hot legs, four cold legs, and two vertical U-tube steam generators (SGs). The reactor has a core power of 3029.06 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plena, and a reactor core containing 217 fuel assemblies. The hot legs connect 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-driven. The ECCS contains two HPSI pumps, four SITs, and two low pressure safety injection (LPSI) pumps.

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 SGTP level of 20% in each steam generator was modeled, which bounds an average SGTP level of up to 20% with an asymmetry of ±4%. Important system parameters and initial conditions used in the analysis are given in Table 3-2.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heat as prescribed by Appendix K of 10 CFR 50.

The analysis assumed 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 single failure criterion required by Appendix K of 10 CFR 50 was satisfied by assuming the loss of one emergency diesel generator (EDG). Thus, a single HPSI and LPSI pump were assumed to be available. The charging system is a safety related system and supported by the remaining EDG, therefore flow from the charging system was credited in the analysis. Initiation of the HPSI and LPSI systems was delayed by 30 seconds beyond the time of SIAS. The 30-second delay represents the time required for diesel generator startup and switching. The charging system is delayed 330 seconds following actuation of SIAS.

The HPSI system was modeled to deliver the SI flow symmetrically to all four loops (Table 3-3). The LPSI system was modeled to deliver the total SI flow asymmetrically to the broken loop (Loop 2B) and one intact loop (Loop 2A) (Table 3-4). The HPSI and LPSI flow are modeled differently, asymmetric verses symmetric, due to the influence the single failure criterion has on the LPSI system. When assuming single failure of an EDG, valves are not opened to allow LPSI flow to Loops 1A and 1B.

The disabling of a motor-driven AFW pump, due to the single failure criterion, leaves one motor-driven pump and the turbine-driven pump available. The initiation of the motor-driven AFW pump was delayed 330 seconds beyond the time of the auxiliary feedwater actuation signal (AFAS) indicating low SG level (4.0% narrow range). The turbine-driven AFW pump was not credited in the analysis. The AFW flow is directed to the SG attached to the broken loop. Although not significant, a sensitivity study performed with AFW directed into the SG attached to the intact loop produced a lower PCT.

The input model included details of both main steam lines from the SGs to the turbine control valve, including the main steam safety valve (MSSV) inlet piping connected to the main steam lines. The MSSVs were set to open at their nominal setpoints plus 3% tolerance for the first bank and 2% tolerance for the second bank.

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

3.4 SER Compliance

A spectrum of cold leg break sizes from 0.022 ft^2 (2.0-inch diameter) to 0.491 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. In addition, to support the operation of St Lucie Unit 2 with M5® cladding, a sensitivity study was performed to consider attached piping break sensitivity. There is no other SER requirement or restriction on EMF-2328 (Reference 1).

Table 3-1 Small break LOCA Development Phases

SBLOCA Phase >>>>	(1) Subcooled depressurization	(2) Loop saturation and loop flow coastdown	(3) Loss of loop circulation and reflux mode cooling	(4) Loop seal clearing and core refill	(5) Long-term cooling
Primary Description	RCS depressurization, reactor trip, turbine trip, loss of offsite power	hot leg saturates, RCS flow coasts down	RCS boils down and flow is interrupted by void formation	loop seal clears, steam is vented to break	core covered and cooled by ECCS injection
Break Characteristics	subcooled discharge	subcooled / saturated liquid discharge	subcooled / saturated liquid discharge	saturated / superheated vapor discharge	subcooled / saturated liquid discharge
RCS Flow	forced flow and coastdown	coastdown to natural circulation	stagnant	steam flow to break	pool boiling
RCS Heat Removal	forced convection via steam generators	forced convection via steam generators	reflux condensation in steam generators, break flow	boiling and break flow	boiling and break flow
RCS Pressure	rapid depressurization	rapid depressurization	pressure plateaus just above secondary pressure	rapid depressurization below secondary pressure	slow depressurization
ECCS Injection	none	none	initiates	continuous, potential SIT injection	continuous, potential SIT injection
Core Level	covered	covered	covered or uncovered[1]	covered or uncovered[2]	core recovery

1 Depending on the loop seal elevation with respect to the top of the active core.

2 Depending on the break size.

Table 3-2 System Parameters and Initial Conditions

Reactor Power, MWt	3029.06[1]
Peak LHR, kW/ft	13.0
Radial Peaking Factor (1.65 plus uncertainty & augmentation factor)	1.81
RCS Flow Rate, gpm	370000
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	554
SIT Pressure, psia	499.7
SIT Fluid Temperature, °F	124.5
Average SG Tube Plugging Level, %	20
SG Secondary Pressure, psia	846-849
MFW Temperature, °F	436
AFW Flow Rate per SG, gpm	255
AFW Temperature, °F	110
AFW Delay Time, sec	330
Low SG Level AFAS Setpoint, %	4
HPSI & LPSI Fluid Temperature, °F	104
Charging System Delay Time, sec	330
Reactor Trip - Low Pressurizer Pressure Setpoint, psia	1810
Reactor Trip Delay Time on Low Pressurizer Pressure, sec	1.15
SCRAM CEA Holding Coil Release Delay Time, sec	0.74
SIAS Activation Setpoint Pressure for Harsh Conditions, psia	1638
HPSI & LPSI Pump Delay Time on SIAS, sec	30
MSSV Lift Pressures and Tolerances	Nominal + 3% (Bank 1 Valves) Nominal + 2% (Bank 2 Valves)

1 Includes 0.3% measurement uncertainty

Table 3-3 HPSI Flow Rate versus RCS Pressure

RCS Pressure (psia)	Loop 1A (gpm)	Loop 1B (gpm)	Loop 2A (gpm)	Loop 2B (gpm)
1063.1	0.0	0.0	0.0	0.0
1062.6	10.6	10.6	10.6	10.6
1062.0	21.3	21.3	21.3	21.3
1045.8	31.9	31.9	31.9	31.9
1009.7	42.5	42.5	42.5	42.5
954.3	53.1	53.1	53.1	53.1
883.3	63.8	63.8	63.8	63.8
800.7	74.4	74.4	74.4	74.4
708.4	85.0	85.0	85.0	85.0
603.7	95.6	95.6	95.6	95.6
476.6	106.3	106.3	106.3	106.3
307.6	116.9	116.9	116.9	116.9
148.1	124.3	124.3	124.3	124.3
125.5	125.2	125.2	125.2	125.2
117.9	125.5	125.5	125.5	125.5
94.3	126.4	126.4	126.4	126.4
59.9	127.7	127.7	127.7	127.7
12.4	129.3	129.3	129.3	129.3
0.0	129.8	129.8	129.8	129.8

Table 3-4 LPSI Flow Rate versus RCS Pressure

RCS Cold Leg Pressure (psia)	Intact Loop 1 1A (gpm)	Intact Loop 2 1B (gpm)	Intact Loop 3 2A (gpm)	Broken Loop 2B (gpm)
125.5	0.0	0.0	0.0	0.0
117.9	0.0	0.0	240.0	240.0
94.3	0.0	0.0	560.0	560.0
59.9	0.0	0.0	880.0	880.0
12.4	0.0	0.0	1200.0	1200.0
0.0	0.0	0.0	1267.2	1267.2

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

Figure 3-4 Axial Power Shape

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) (Reference 3) for St. Lucie Unit 2 operating with AREVA supplied 16x16 HTP 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. Section 4.3 describes the sensitivity study for a break in the attached piping.

4.1 Results for Break Spectrum

The St. Lucie Unit 2 cold leg pump discharge 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.

The results for the break spectrum calculations are presented in Table 4-1. Time sequence evolution for each break size analysis is reported in Table 4-2. Figure 4-1 shows the calculated PCTs as a function of break size. The limiting break size was determined to be a 2.70-inch diameter (0.040-ft^2) break.

4.2 Discussion of Transient for Limiting Break

The results for the limiting break size (2.70-inch) are shown in Figure 4-2 through Figure 4-16. The following discussion pertains to the limiting case.

The primary pressure decreased immediately after break initiation (Figure 4-3). When the primary pressure reached the low pressurizer pressure trip setpoint at 28 seconds the reactor is tripped (Figure 4-2). Within approximately 1 second after reactor trip, reactor SCRAM occurs, which occurs coincident with the loss of offsite power, causing turbine to trip, RCP trip and MFW pump trip (Figure 4-7 and Figure 4-8). As MFW to the

SGs is ramped down and steam flow to the main steam isolation valves (MSIVs) are closed, the pressure in the SGs increase for approximately 10 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 200 seconds. Significant inventory is lost out the break until the break transitions from expelling liquid to steam at approximately 600 seconds (Figure 4-4 and Figure 4-5). Around this time, the broken leg loop seal begins to clear and the remaining intact loops remain plugged (Figure 4-6).

Prior to loop seal clearing, the core uncovers below top of active fuel (Figure 4-15). Since 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 at 596 seconds into the transient. This causes an abrupt level drop in the downcomer region and a small core level recovery. 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.

From approximately 1000 to 1400 seconds, the primary side pressure is only slightly higher than the secondary side pressure (Figure 4-3), limiting the primary side heat removal. Broken loop steam generator AFW is initiated at approximately 936 seconds (Figure 4-9), providing an increasing SG level (Figure 4-10). However, the small pressure difference between the primary and secondary side limits the AFW's ability to remove heat from the primary side. At approximately 1500 seconds, the primary pressure reduces below the secondary side pressure, ending heat removal by the secondary side.

Meanwhile, the HPSI system becomes available at 68 seconds into the transient but does not began to inject water into the primary system cold legs until 664 seconds into the transient (Figure 4-9). However, HPSI flow does not provide sufficient inventory at this time to offset the large amount of RCS inventory lost out the break. As effective

cooling is lost in the core, and the fuel rods begin to heat up at approximately 1350 seconds (Figure 4-16). Hot rod rupture occurs at 2004 seconds. The fuel continues to heat up until the maximum PCT of 1926°F is reached at 2200 seconds. SIT injection begins just prior to reaching the PCT at 2194 seconds, providing sufficient cooling to turn the PCT over. The PCT occurs at approximately 4 inches below the top of active fuel.

For this break size, although LPSI is available, it did not inject (Figure 4-12) due to the slow primary system depressurization. The continued supply of HPSI supports an increasing reactor vessel inventory (Figure 4-14) through event termination.

In conclusion, the limiting PCT break spectrum case is a 2.70-inch diameter cold leg pump discharge break. The PCT of this case is 1926°F. The maximum local oxidation is less than 6% and the maximum core-wide oxidation is less than 0.2%. The total maximum local oxidation is less than 9%, including a pre-transient oxidation of 2.3925%.

4.3 Attached Piping Sensitivity Study

Although breaks in the SIT attached piping are not typically PCT limiting, they do result in reduced ECCS flows available to mitigate the event since one SIT inventory spills to containment, in addition to the reduced SI flow due to single failure assumption. Therefore, an analysis of the limiting break size and location in an attached piping was performed. For St. Lucie Unit 2, the limiting break location and size for an attached piping is considered a double-ended guillotine break of an SIT line (10.126-inch diameter). The break was located in the SIT line connected to Loop 2B.

The calculated PCT was 1451°F, which is less limiting than the maximum PCT of the break spectrum analysis. The maximum local oxidation thickness is 0.1404% and the maximum core-wide oxidation thickness is 0.0017% which are both less limiting than the maximum obtained in the break spectrum analysis. The minimal HPSI and LPSI flows in the analysis are sufficient to prevent a subsequent heatup after the initial quench from the SIT discharge.

Table 4-1 Summary of SBLOCA Break Spectrum Results

Break diameter (in)	2.00	2.50	2.60	**2.70**	2.80	3.00
Break Area (ft^2)	0.022	0.034	0.037	**0.040**	0.043	0.049
Peak Clad Temperature (°F)	1587	1726	1814	**1926**	1896	1823
Time of PCT (sec)	3752	2615	2456	**2200**	1997	1659
Time of Rupture (sec)	--	2443	2219	**2004**	1848	1654
Transient Local Maximum Oxidation (%)	1.3586	3.6120	4.3352	**5.7788**	5.0864	3.5589
Total Local Maximum Oxidation (%)[1]	3.7511	6.0045	6.7277	**8.1713**	7.4789	5.9514
Core Wide Oxidation (%)	0.0955	0.1273	0.1382	**0.1664**	0.1388	0.0903
PCT Elevation (ft)	10.77	11.02	11.02	**11.02**	11.02	11.02

Break diameter (in)	3.50	4.00	4.50	5.00	6.00	7.00
Break Area (ft^2)	0.067	0.087	0.110	0.136	0.196	0.267
Peak Clad Temperature (°F)	1694	1578	1283	1077	1017	1080
Time of PCT (sec)	1074	802	649	527	145	131
Time of Rupture (sec)	1057	--	--	--	--	--
Transient Local Maximum Oxidation (%)	1.2501	0.3964	0.0683	0.0126	0.0065	0.0116
Total Local Maximum Oxidation (%)[1]	3.6426	2.7889	2.4608	2.4051	2.3990	2.4041
Core Wide Oxidation (%)	0.0335	0.0121	0.0011	0.0002	0.0001	0.0002
PCT Elevation (ft)	10.77	10.52	10.27	10.27	10.02	10.02

Break diameter (in)	8.00	9.00	9.49
Break Area (ft^2)	0.349	0.442	0.491
Peak Clad Temperature (°F)	1459	1469	1520
Time of PCT (sec)	169	148	136
Time of Rupture (sec)	--	--	--
Transient Local Maximum Oxidation (%)	0.1539	0.1569	0.1621
Total Local Maximum Oxidation (%)[1]	2.5464	2.5494	2.5546
Core Wide Oxidation (%)	0.0017	0.0021	0.0026
PCT Elevation (ft)	10.02	10.02	10.02

[1] Includes the Pre-transient Oxidation of 2.3925%.

Table 4-2 Sequence of Events for the SBLOCA Break Spectrum

Break diameter (in)	PCT (°F)	Break opens	Low PZR Pressure Trip	Reactor SCRAM, RCP & Turbine Trip	SIAS issued	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-condens. gas at the break
2.00	1587	0	50	52	63	93	93	1114	--	940	--	--	--	1090	1272	--	3368	--	3752	2386	--
2.50	1726	0	32	34	43	73	73	782	--	930	--	--	--	670	782	2748	2218	2443	2615	1510	--
2.60	1814	0	30	32	40	70	70	724	--	932	--	--	--	626	724	2452	2100	2219	2456	378	--
2.70	**1926**	**0**	**28**	**29**	**38**	**68**	**68**	**664**	**--**	**936**	**--**	**--**	**--**	**596**	**668**	**2194**	**1978**	**2004**	**2200**	**354**	**--**
2.80	1896	0	26	28	35	65	65	604	--	948	--	--	--	566	634	1992	1854	1848	1997	328	--
3.00	1823	0	22	24	32	62	62	536	--	1668	--	--	--	504	564	1652	1624	1654	1659	288	--
3.50	1694	0	17	19	25	55	55	412	--	--	--	--	--	380	444	1064	1068	1057	1074	208	--
4.00	1578	0	13	15	21	51	51	322	--	--	--	--	--	296	306	896	902	--	802	162	--
4.50	1283	0	11	13	17	47	47	266	--	--	--	--	260	242	258	640	644	--	649	128	--
5.00	1077	0	10	12	15	45	45	222	--	--	--	--	208	198	216	514	518	--	527	102	--
6.00	1017	0	8	10	13	43	43	146	--	--	--	--	142	138	160	326	330	--	145	76	--
7.00	1080	0	8	10	12	42	42	104	--	--	--	--	112	104	128	232	236	--	131	54	--
8.00	1459	0	8	9	11	41	41	68	--	--	100	102	--	80	106	164	166	--	169	48	--
9.00	1469	0	7	9	11	41	41	52	198	--	68	86	70	60	84	132	134	--	148	40	--
9.49	1520	0	7	9	10	40	40	46	142	--	58	68	64	56	82	120	122	--	136	38	--

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

Figure 4-2 Reactor Power – 2.70 inch Break

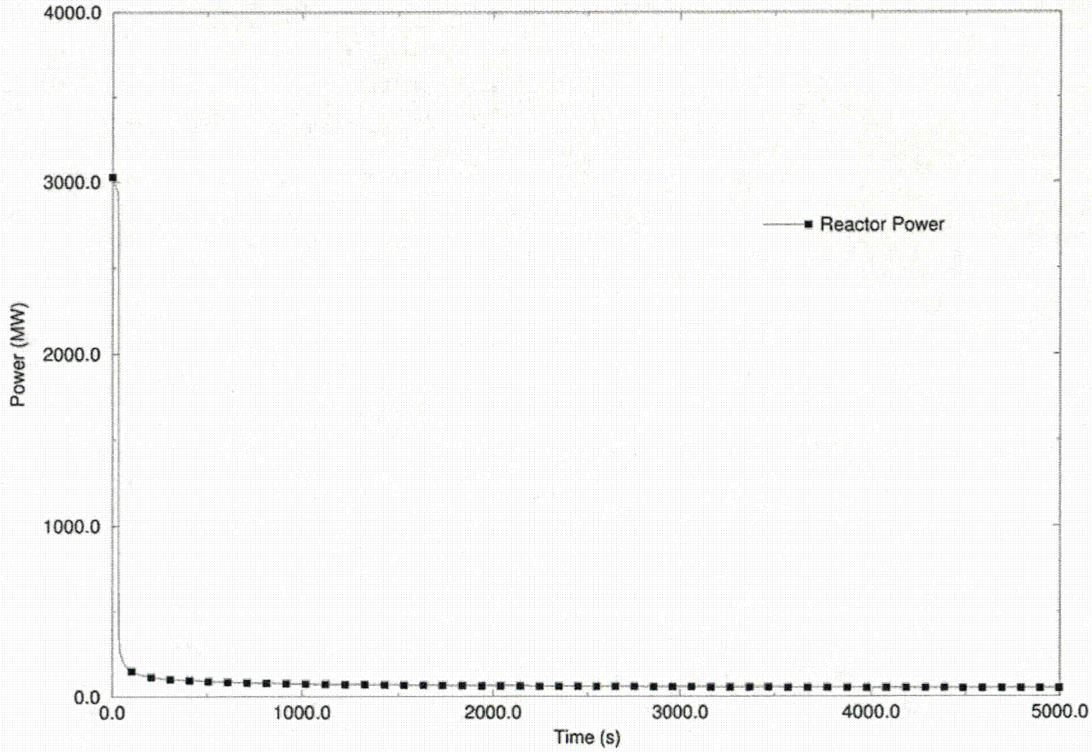


Figure 4-3 Primary and Secondary System Pressures – 2.70 inch Break

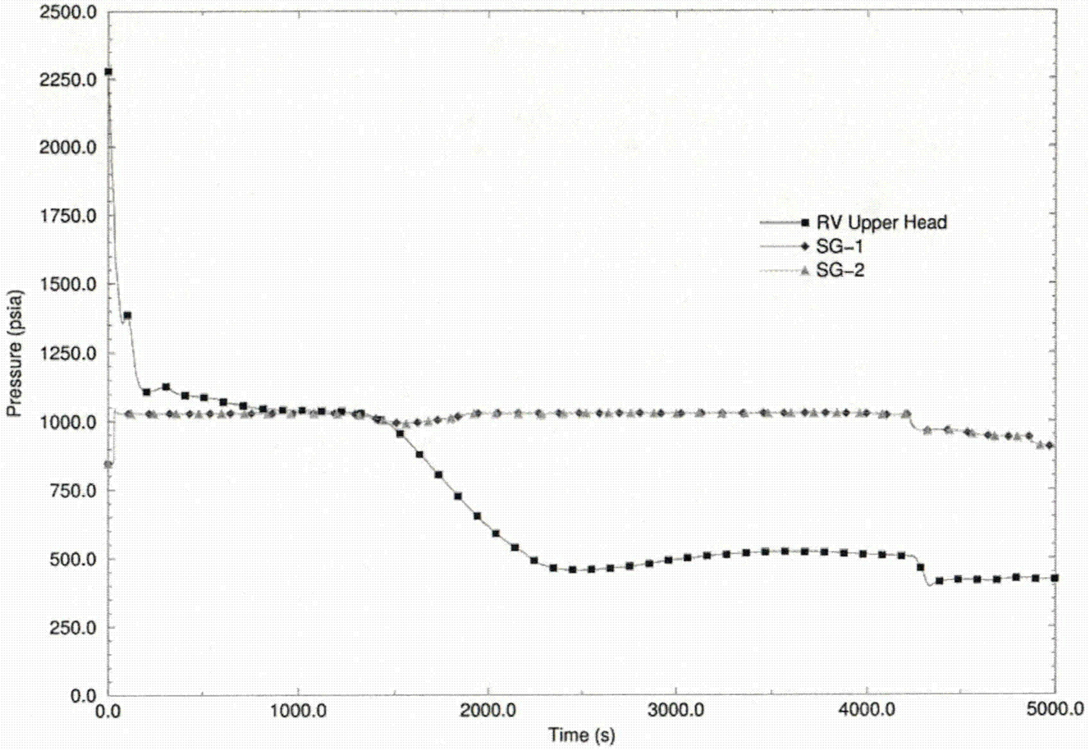


Figure 4-4 Break Void Fraction – 2.70 inch Break

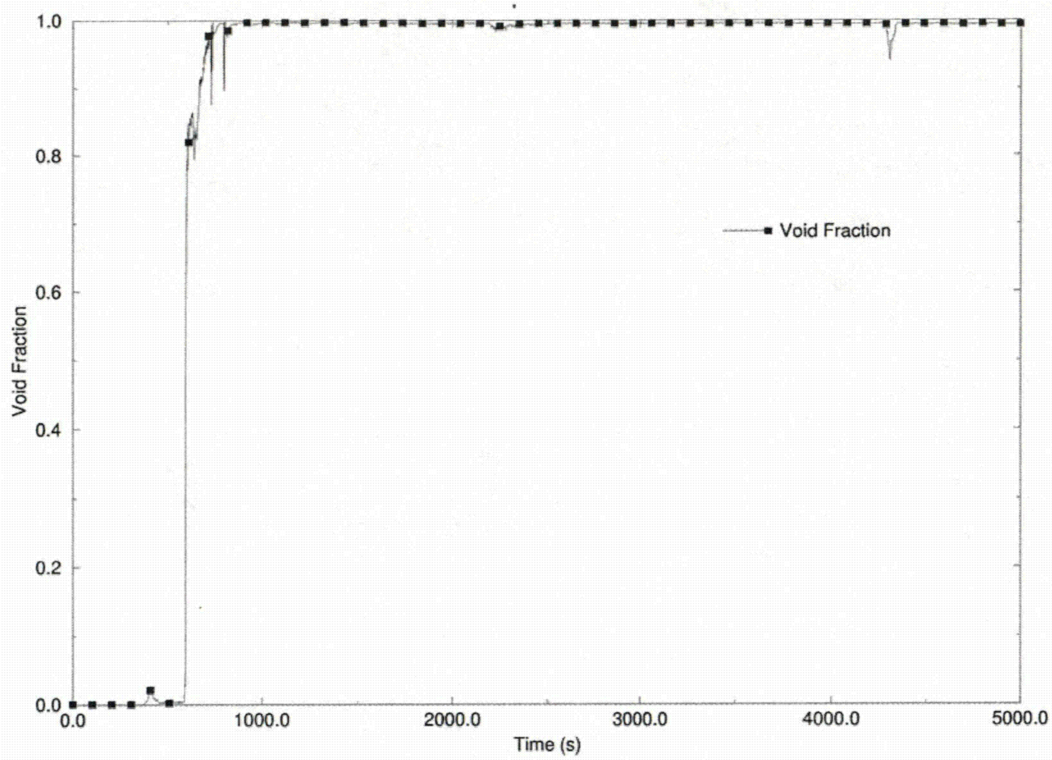


Figure 4-5 Break Mass Flow Rate – 2.70 inch Break

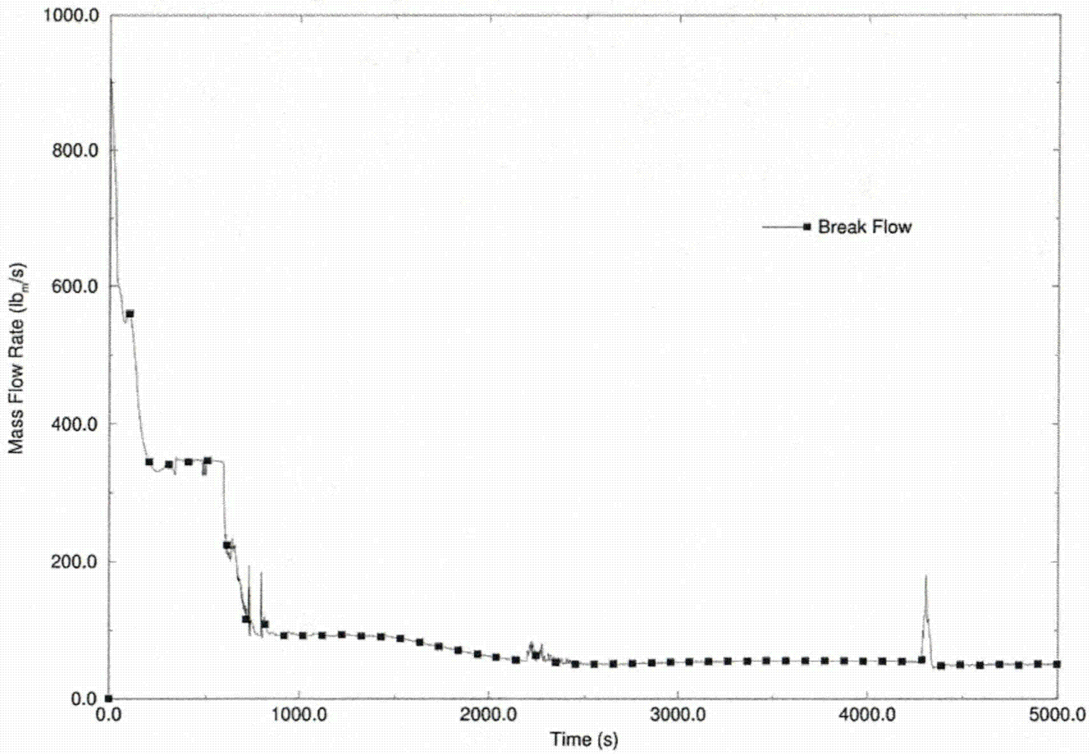


Figure 4-6 Loop Seal Void Fraction – 2.70 inch Break

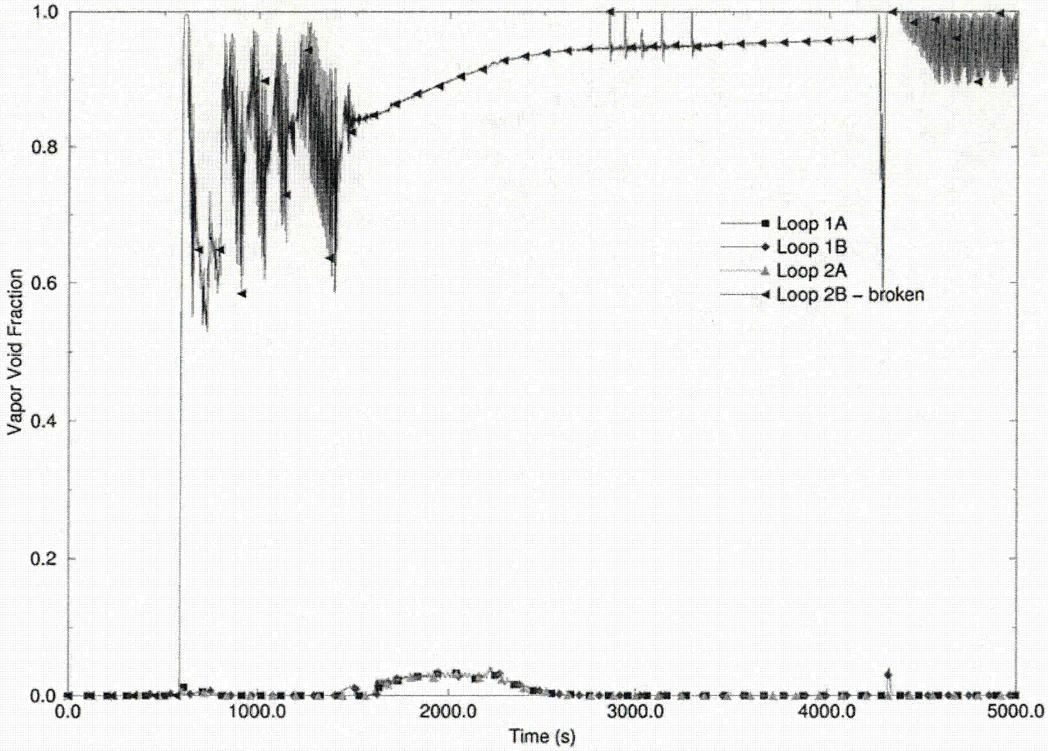


Figure 4-7 RCS Loop Mass Flow Rate – 2.70 inch Break

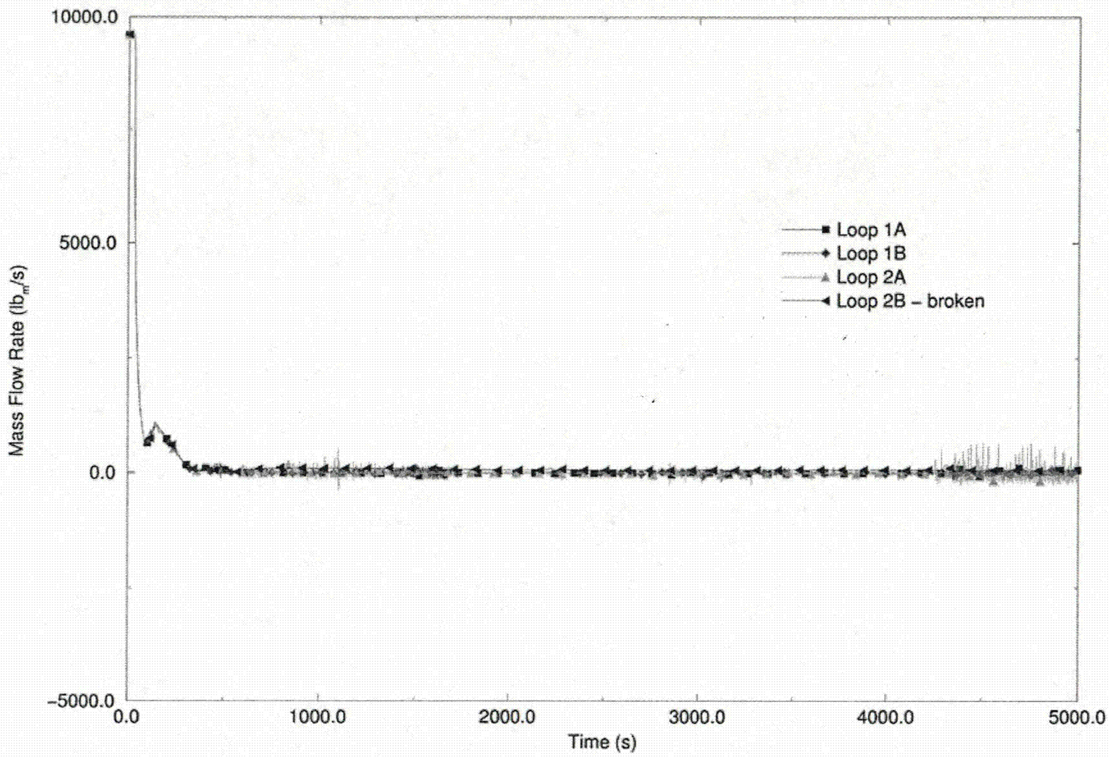


Figure 4-8 Main Feedwater Mass Flow Rate – 2.70 inch Break

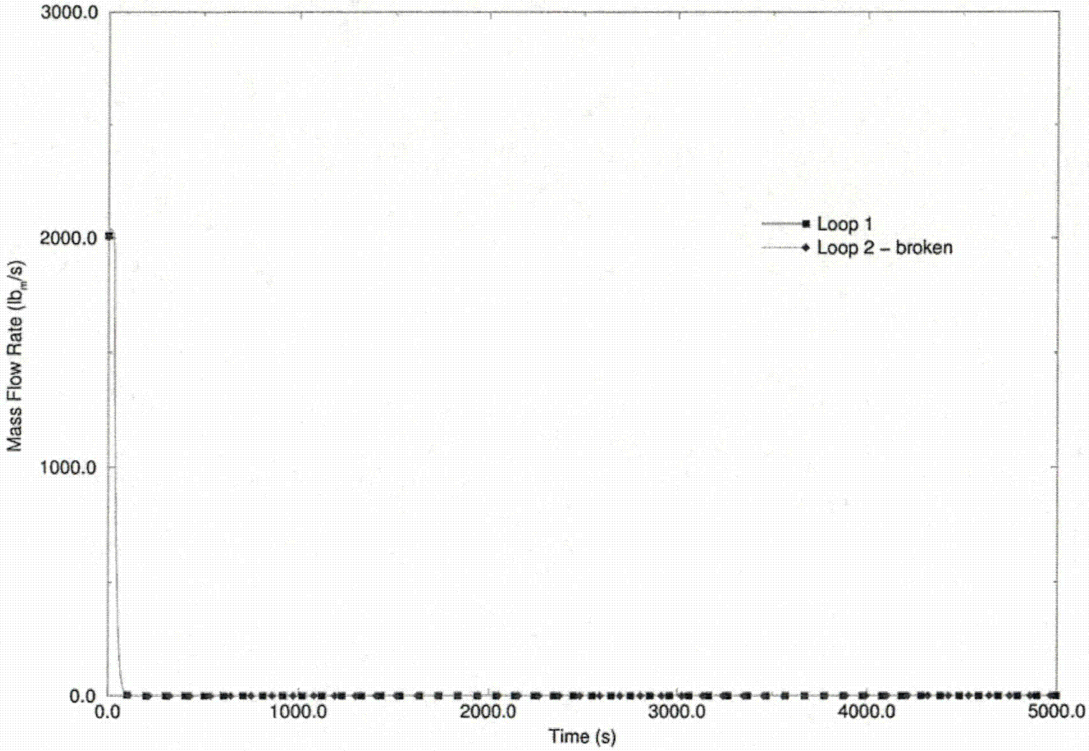


Figure 4-9 Auxiliary Feedwater Mass Flow Rate – 2.70 inch Break

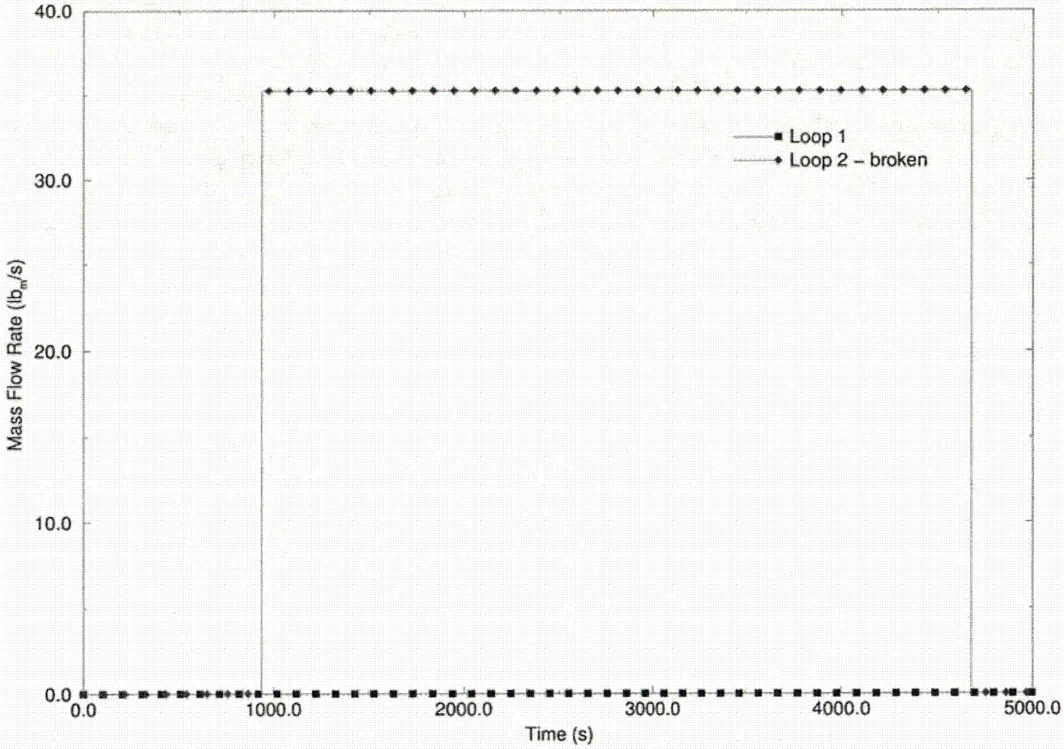


Figure 4-10 Steam Generator Total Mass – 2.70 inch Break

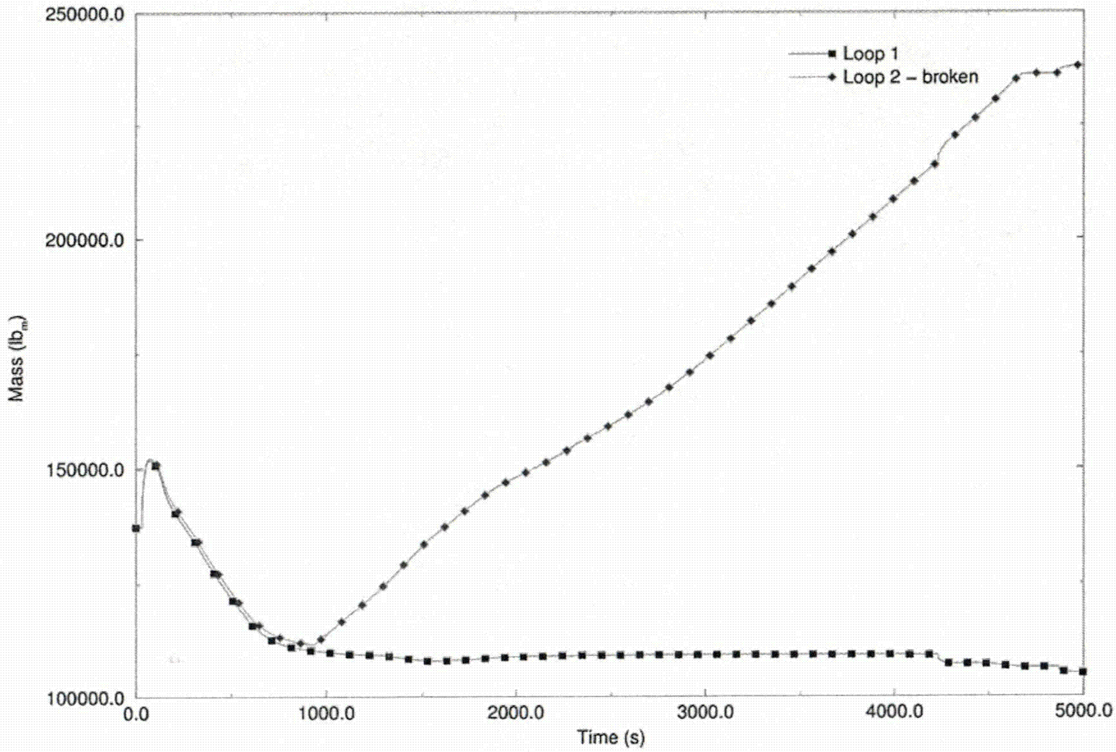


Figure 4-11 High Pressure Safety Injection Mass Flow Rates – 2.70 inch Break

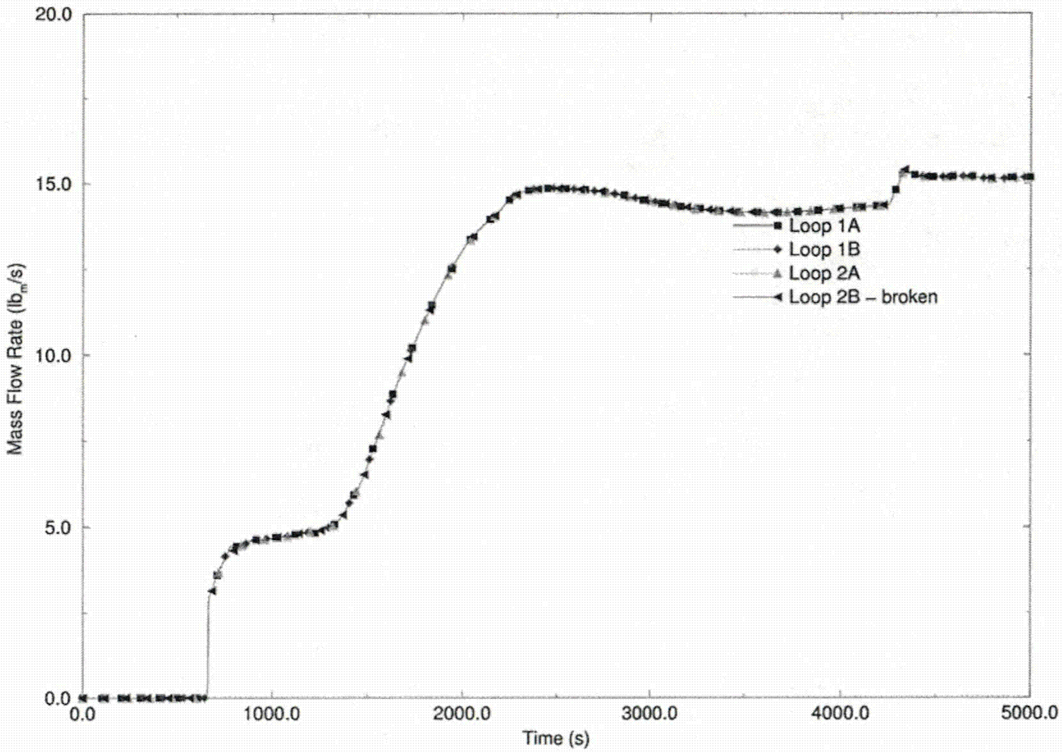


Figure 4-12 Low Pressure Safety Injection Mass Flow Rates – 2.70 inch Break

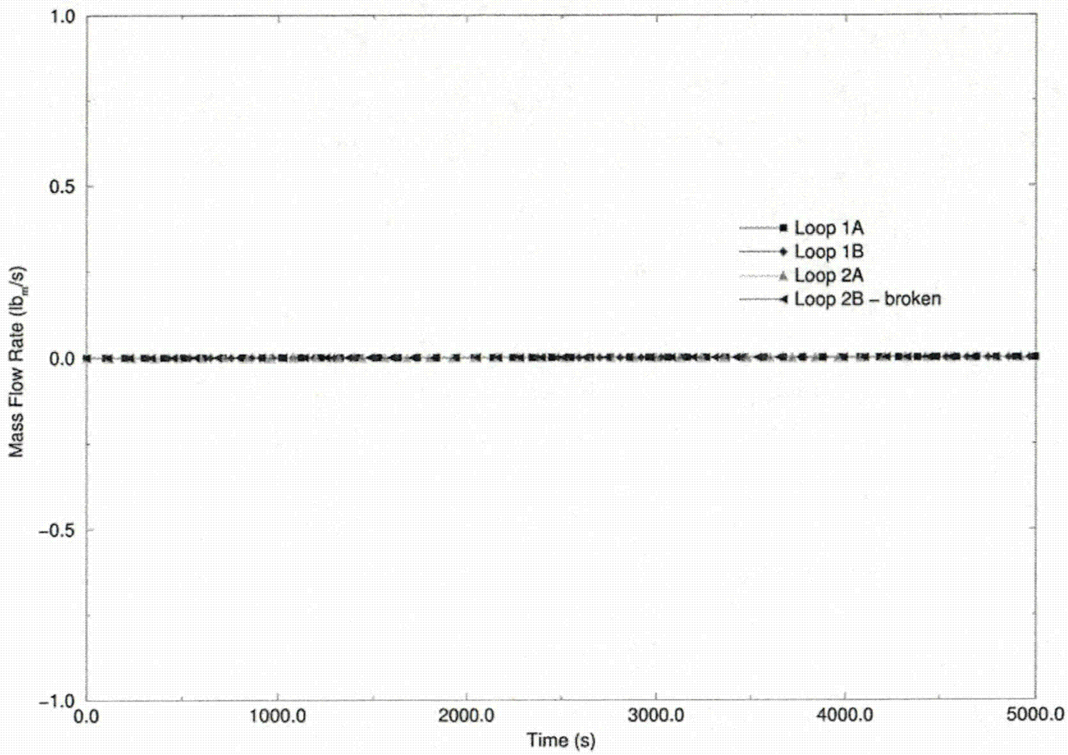


Figure 4-13 Safety Injection Tank Mass Flow Rates – 2.70 inch Break

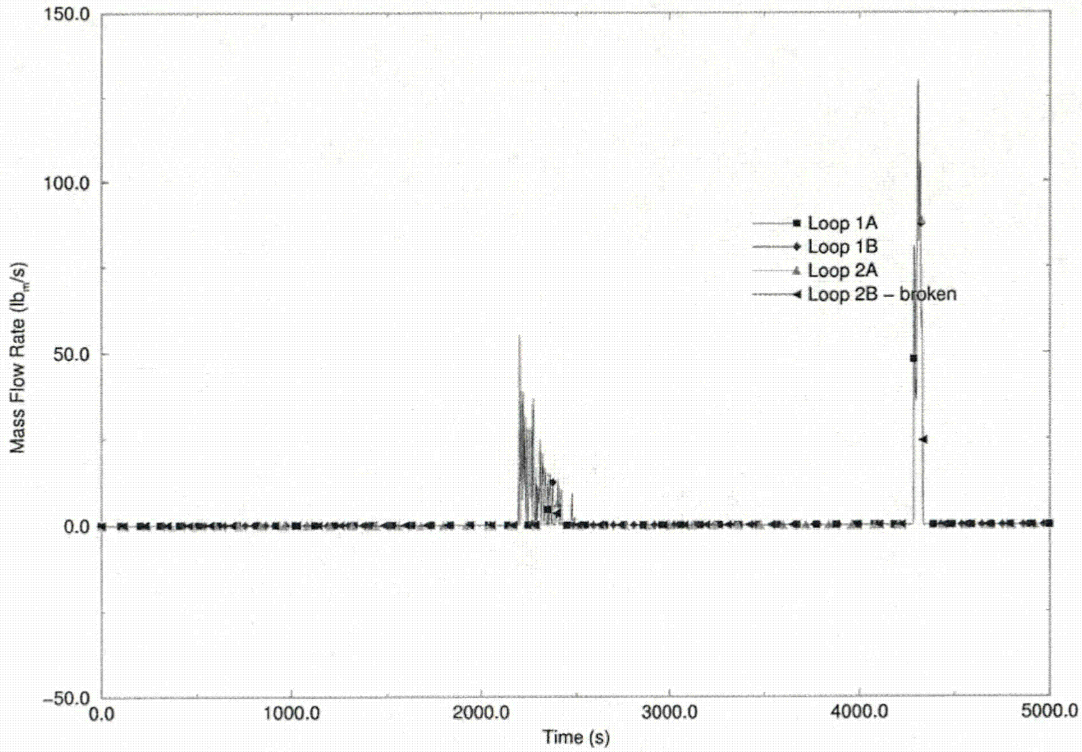


Figure 4-14 Reactor Vessel Mass Inventory – 2.70 inch Break

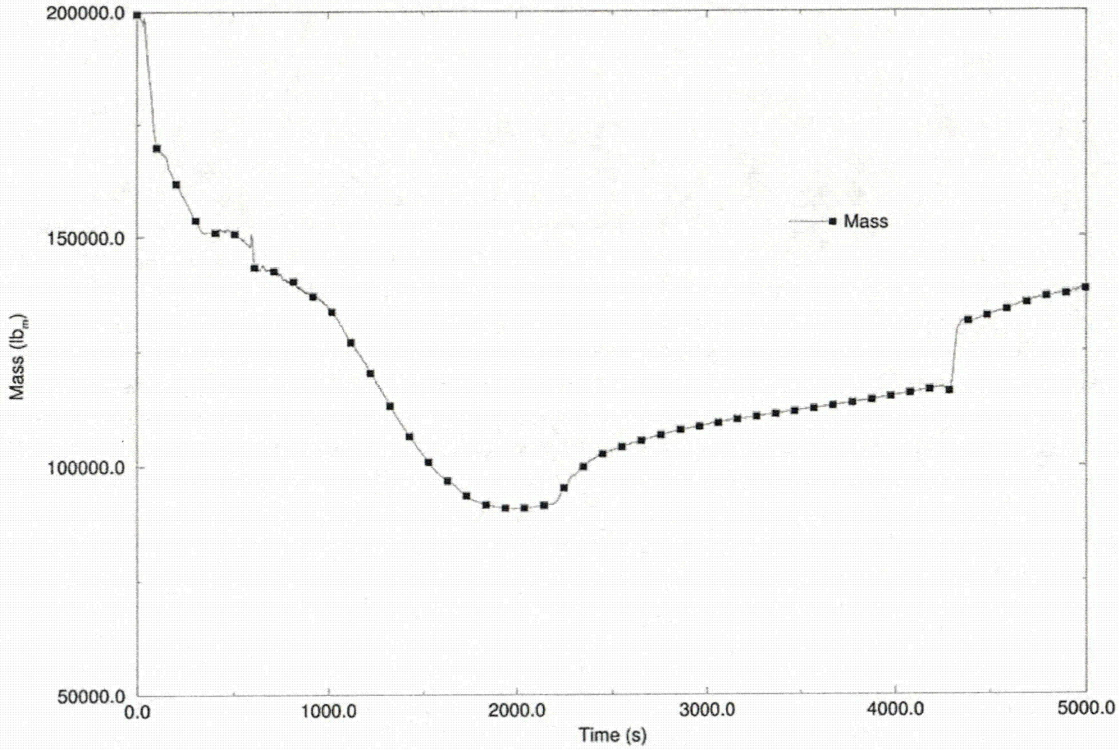


Figure 4-15 Hot Assembly Mixture Level – 2.70 inch Break

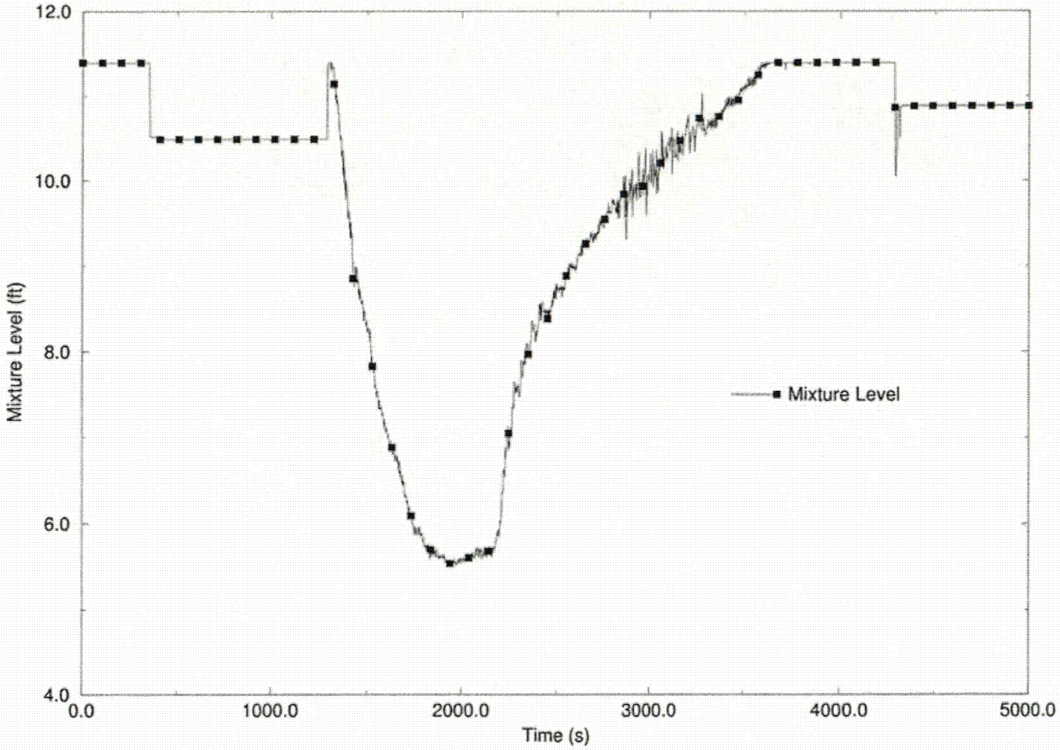
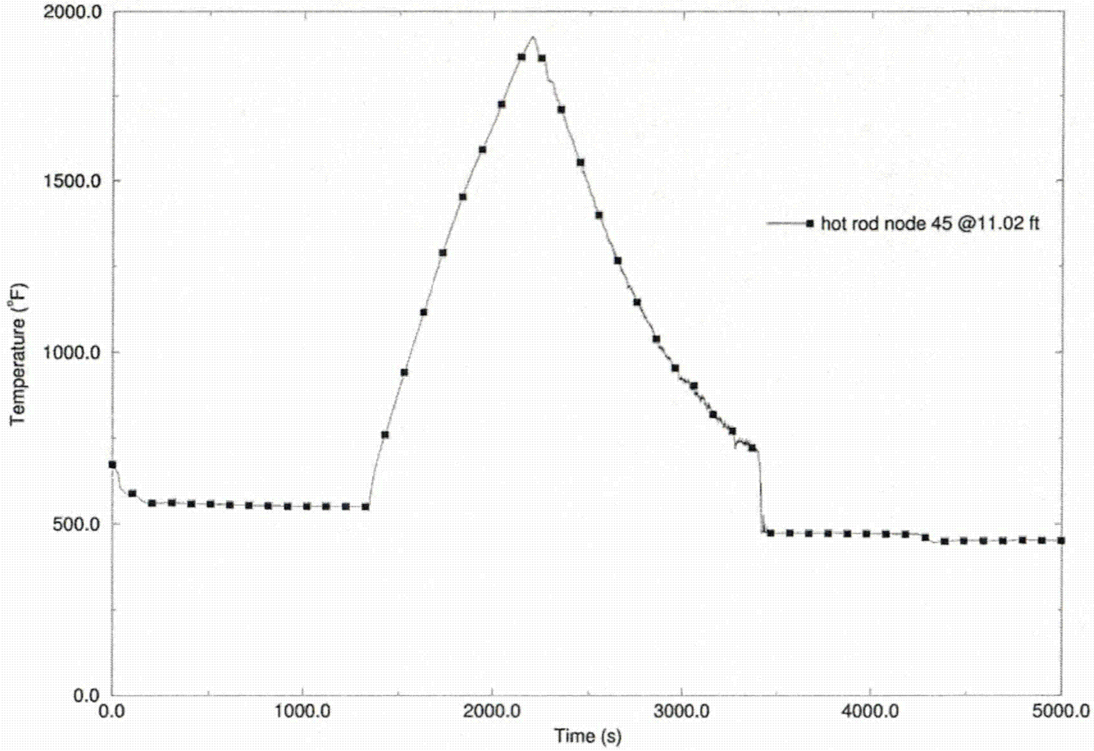


Figure 4-16 Hot Spot Cladding Temperature – 2.70 inch Break



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

3. 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.

4. 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.

5. 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.

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