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St. Lucie Unit 1 Small Break LOCA Analysis with 30% Steam Generator Tube Plugging

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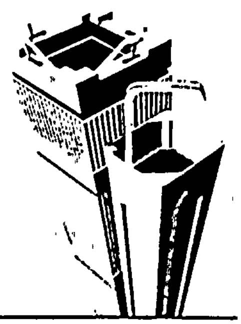

Siemens Power Corporation

Nuclear Division

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St. Lucie Unit 1 Small Break LOCA Analysis with 30% Steam Generator Tube Plugging

Prepared by:

J. C. Hibbard, Sr. Engineer
PWR Safety Analysis
Nuclear Engineering

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1.0 OBJECTIVE AND CONCLUSIONS

The objective of this Small Break Loss-of-Coolant-Accident (SBLOCA) analysis is to support an increased Steam Generator Tube Plugging (SGTP) and reduced Reactor Coolant System (RCS) flow rate for St. Lucie Unit 1 at full power (Reference 1). The SGTP is increased from 25% to 30% with an asymmetry of ±7%, and the primary coolant flow rate is reduced from 355,000 gpm to 345,000 gpm. A nominal power level of 2700 MWt is supported by the analysis. In the Reference 2 analysis, the 0.10 ft^2 break was identified as the limiting break size. Increasing SGTP by 5% will not change the limiting break size. Therefore, only the 0.10 ft^2 break size was analyzed to support the increased SGTP and reduced RCS flow rate. The break spectrum, Reactor Coolant Pump (RCP) trip delay sensitivity, and SGTP asymmetry calculations presented in Reference 2 continue to be the licensing basis SBLOCA analysis.

This analysis was performed using Siemens Power Corporation (SPC) SBLOCA methodology (References 3 and 4). The SPC SBLOCA methodology has been generically approved by the NRC for PWR licensing applications. The increased SGTP SBLOCA analysis for St. Lucie Unit 1 demonstrates that 10 CFR 50.46 criteria are satisfied.

2.0 SUMMARY OF RESULTS

A SBLOCA analysis was performed for St. Lucie Unit 1 to support a nominal full power of 2700 MWt, an increase in SGTP from 25% to 30% with an asymmetry of ±7% and a reduction in RCS flow rate from 355,000 gpm to 345,000 gpm. The analysis supports a radial peaking factor (F_r) of 1.75 (1.89 with uncertainties), and a maximum Linear Heat Generation Rate (LHGR) of 15 kW/ft. These values support the current Technical Specification limits. This analysis was performed to demonstrate that the acceptance criteria for Emergency Core Cooling Systems (ECCS), as stated in 10 CFR 50.46, have been met. The 10 CFR 50.46 criteria are:

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.
5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The Reference 2 SBLOCA analysis identified the 0.1 ft^2 break to be the limiting break size. For the increased SGTP and reduced RCS flow rate analysis, the limiting Peak Cladding Temperature (PCT) for the 0.1 ft^2 break was calculated to be 1804°F.

All 10 CFR 50.46 criteria have been demonstrated to be met for the limiting case:

1. The limiting PCT was calculated to be 1804°F.
2. The local cladding oxidation was calculated to be 0.02 times the total cladding thickness before oxidation.
3. The core-wide metal-water reaction was calculated to be less than 0.01 times the total cladding thickness before oxidation.
4. The core remains amenable to cooling by staying within the local oxidation criteria.
5. Flow rates from the ECCS ensure that the core temperatures have been reduced to acceptably low values and long term cooling has been established.

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 1 plant and outlines the system parameters used in the SBLOCA analysis.

3.1 Description of SBLOCA Event

The postulated SBLOCA is defined as a break in the Pressurized Water Reactor (PWR) pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe at the discharge side of the RCP. This break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected out through the break. This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46 criteria.

The SBLOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a Thermal Margin/Low Pressure (TM/LP) trip signal. The Safety Injection Actuation Signal (SIAS) occurs when the system has further depressurized. The capacity and shutoff head of the High Pressure Safety Injection (HPSI) pumps are important parameters in the SBLOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is large enough that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the Safety Injection Tank (SIT) pressure. 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 combined intact loop Safety Injection (SI) flow rate and SIT flow rate exceed the mass flow rate out the break.

3.2 Analytical Models

The SPC SBLOCA evaluation model for event response of the plant and hot fuel rod used in this analysis (References 3 and 4) consists of three 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 three SPC computer codes used in this analysis are:

1. The RODEX2 code was used to determine the initial fuel stored energy and gap conditions for the initialization of the system blowdown and hot rod response calculations.

2. The SPC version of RELAP5/MOD2 (ANF-RELAP) was used to model the primary system and secondary side of the steam generators throughout the event. The governing conservation equations for mass, energy, and momentum transfer are used along with appropriate correlations consistent with Appendix K of 10 CFR 50.

3. The TOODEE2 code was employed to model the behavior of the hot rod during the entire event. TOODEE2 uses thermal-hydraulic boundary conditions from the ANF-RELAP system calculation.

Initial fuel rod temperatures, corresponding to beginning-of-cycle (BOC) peak stored energy conditions, were used to initialize both ANF-RELAP and TOODEE2. Initial cladding and fuel pellet dimensions, plenum gas inventory and composition, and effective plenum volume were taken at end-of-cycle (EOC) conditions, and were used to initialize TOODEE2. This conservative combination of BOC and EOC conditions bounds operation of the fuel over the entire span of the fuel cycle.

3.3 Plant Description and Summary of Analysis Parameters

St. Lucie Unit 1 is a Combustion Engineering (CE) designed two-by-four loop PWR with two hot legs, four cold legs, and two vertical U-tube steam generators. The reactor has a rated core power of 2700 MWt. The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 217 fuel assemblies. The hot legs connect the reactor vessel with the vertical U-tube steam generators. Feedwater is injected into the downcomer of each steam generator. There are three auxiliary feedwater 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 reactor coolant system of the plant was nodalized in the ANF-RELAP model into control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two steam generators with both primary and secondary sides modeled. All the loops were modeled explicitly to provide an accurate representation of the plant. A steam generator tube plugging level of 30% was assumed. Both the HPSI system and the LPSI system flows were delivered to the respective cold legs in the ANF-RELAP model.

The heat generation rate in the ANF-RELAP reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.

The analysis assumed loss of offsite power concurrent with reactor scram. The single failure criterion required by Appendix K was satisfied by assuming the loss of one diesel generator, which resulted in the disabling of one HPSI pump, one LPSI pump and one motor-driven auxiliary feedwater pump. Initiation of the HPSI system was delayed 30 seconds beyond the time of SIAS. The 30-second delay represents the time required for diesel generator startup and switching. The disabling of a motor-driven auxiliary feedwater pump would leave one motor-driven pump and the turbine-driven pump available. The initiation of the motor-driven pump was delayed 305 seconds beyond the time of the Auxiliary Feedwater Actuation Signal (AFAS) indicating low steam generator level, while the turbine-driven pump was actuated at 600 seconds after break initiation.

In the analysis, the RCPs were conservatively assumed to be tripped at reactor scram, coincident with the loss of offsite power. The effect of delaying the RCP trip was assessed in Reference 2. The conclusion that tripping pumps at reactor scram is conservative continues to be valid for the increased SGTP and reduced RCS flow rate SBLOCA analysis.

In the Reference 2 analysis, a sensitivity calculation was performed to investigate the effect of an asymmetric distribution of plugged tubes. The conclusion that the SBLOCA analysis is not sensitive to SGTP asymmetry continues to be valid for the increased SGTP and reduced RCS flow rate SBLOCA analysis.

Important system parameters and initial conditions used in the analysis are given in Table 3.1.

Table 3.1

System Parameters and Initial Conditions

Primary Heat Output, MWt	2700[a]
Primary Coolant Flow, gpm	345,000
Primary Coolant System Volume, ft^3	10319[b]
Operating Pressure, psia	2250
Inlet Coolant Temperature, °F	549
Reactor Vessel Volume, ft^3	4522
Pressurizer Total Volume, ft^3	1500
SIT Volume, ft^3 (each of four)	2020
SIT Liquid Volume, ft^3	1090
SIT Pressure, psia	214.7
SIT Fluid Temperature, °F	110
Total Number of Tubes per Steam Generator	8485
Number of Tubes Plugged per Steam Generator	2546 (30%)
Secondary Flow Rate / Steam Generator, lbm/sec	1667
Steam Generator Secondary Pressure, psia	770
Steam Generator Feedwater Temperature, °F	435
SI Fluid Temperature, °F	100
Reactor Scram Low Pressure Setpoint, psia	1807
SIAS Activation Setpoint Pressure, psia	1520
HPSI Pump Delay Time on SIAS, sec	30
Main Steam Safety Valve Setpoint Pressure, psia	
Bank 1	1060[c]
Bank 2	1102[c]

(a) Primary heat output used in ANF-RELAP model 1.02 x 2700 = 2754 MWt.

(b) Includes pressurizer total volume and 30% SG tube plugging.

(c) Includes 3% tolerance and 3% accumulation.

Table 3.1

System Parameters and Initial Conditions (continued)

HPSI and LPSI Flow Rate Versus RCS Pressure

RCS Pressure (psia)	Total HPSI Flow[a] (lbm/sec)	Total LPSI Flow[a][b] (lbm/sec)
1129.0	0.0	0.0
1125.0	3.59	0.0
1115.0	9.12	0.0
1015.0	28.73	0.0
815.0	47.79	0.0
615.0	61.05	0.0
315.0	76.66	0.0
165.6	82.98	0.0
159.9	83.22	132.60
135.5	84.26	270.73
92.4	86.06	408.86
28.7	88.76	546.99
14.7	89.36	546.99

5.25 lbm/sec charging pump flow. All of the charging flow was assumed to go to the broken loop, with one half of the flow going to the intact cold leg and the remaining half going to the broken cold leg.

(a) 1/4 of the flow is distributed to each of the cold legs.

(b) LPSI flow did not actuate in the analysis.

4.0 ANALYSIS RESULTS

Increased SGTP and reduced RCS flow rate SBLOCA calculations were performed for the limiting break size (0.10 ft^2) identified in Reference 2. Predicted event times for the 0.10 ft^2 break are summarized in Table 4.1. Results from the corresponding TOODEE2 hot rod response calculation are presented in Table 4.2.

The system response for the ANF-RELAP calculation is shown in Figures 4.1 through 4.8. The TOODEE2-predicted hot rod cladding temperature response for the PCT location and rupture location is presented in Figure 4.9.

The primary and secondary pressure responses are shown in Figure 4.1. The primary pressure decreased immediately after break initiation. Reactor scram occurred when the primary pressure reached the TM/LP trip setpoint of 1807 psia. The secondary pressure increased rapidly after break initiation as steam generator isolation took place. The secondary pressure continued to increase until the Main Steam Safety Valves (MSSVs) opened, causing the secondary pressure to stabilize. At approximately 270 seconds, liquid was expelled from the loop seal piping, allowing steam to flow directly to the break, which caused the primary pressure to decrease more rapidly.

The break flow rate is shown in Figure 4.2. Single phase liquid flow begins at the initiation of the break and continues to about 270 seconds. The drop in flow rate at 270 seconds is due to the transition from single phase liquid flow to steam flow which occurs due to loop seal clearing. The increase in the break flow rate at approximately 1211 seconds was caused by the initiation of SIT flow, as shown in Figure 4.7.

The collapsed liquid level in the core is shown in Figure 4.5. The level drops from the time of initiation of the event until approximately 300 seconds as a void profile is established in the core due to steam generated by the removal of decay heat. The level then remains relatively stable until 600 seconds when the decreasing liquid inventory causes the level to begin to decrease. Core uncovery begins at approximately 700 seconds, as shown by the increasing steam temperature in Figure 4.8. The level continues to fall until SIT flow is activated at 1211 seconds at which time the core level begins to recover.

The total HPSI flow rate is shown in Figure 4.6. Flow begins at 67 seconds and increases as primary system pressure decreases.

Flow from the SITs, shown in Figure 4.7, begins at 1211 seconds and adds sufficient liquid to the core to terminate the heatup of the core, as shown in Figure 4.9. Pressure equilibrating cross-tie piping in the SITs does not affect the flow rates from the SITs because the pressures in all four cold legs are not significantly different throughout the transient.

The reactor vessel fluid mass, shown in Figure 4.4, declines rapidly after event initiation. After loop seal clearing at approximately 270 seconds, the amount of mass in the reactor vessel continues to decline but at a reduced rate. The minimum reactor vessel mass occurs at 1211 seconds. At this time SIT flow begins, increasing reactor vessel inventory, which terminates the cladding heatup. The system slightly repressurizes due to increased vapor generation in the core, terminating the SIT flow. HPSI flow continues to increase the inventory in the reactor vessel while steam continues to vent through the loop seals to the break. Cladding temperatures continue to decrease as reactor vessel inventory rises and long term cooling is established.

The PCT was calculated to be 1804°F.

Table 4.1

Sequence of Events for 30% SGTP SBLOCA Calculations
(0.10 ft^2 Break)

	Time (sec)
Event initiation	0.0
Reactor and RCP Trip	13.4
MSSVs Open	20
SIAS + 30 sec delay	66.6
HPSI injection starts	66.6
Loop Seals Clear Intact loop, cold leg 1A Intact loop, cold leg 1B Broken loop, cold leg 2A (intact) Broken loop, cold leg 2B (broken)	 NA ~270 ~270 NA
MSSVs Close	276
Auxiliary Feedwater (AFW) On Motor-Driven AFW Turbine-Driven AFW	 322 600
Beginning of Core Heatup	~700
SIT Injection Starts	1211
PCT Occurs	1214
End Of Calculation	1800

Table 4.2

30% SGTP SBLOCA Calculation Results
(0.10 ft^2 Break)

Hot Rod Burst	
Time (sec)	1111
Elevation (ft)	10.0
Channel Blockage Fraction, %	48.7
Peak Cladding Temperature	
Temperature (°F)	1804
Time (sec)	1214
Elevation (ft)	10.5
Metal-Water Reaction	
Local Maximum (%)	1.96
Elevation of Local Max. (ft)	10.0
Hot Pin Average (%)	0.25
Core Wide (%)	< 1

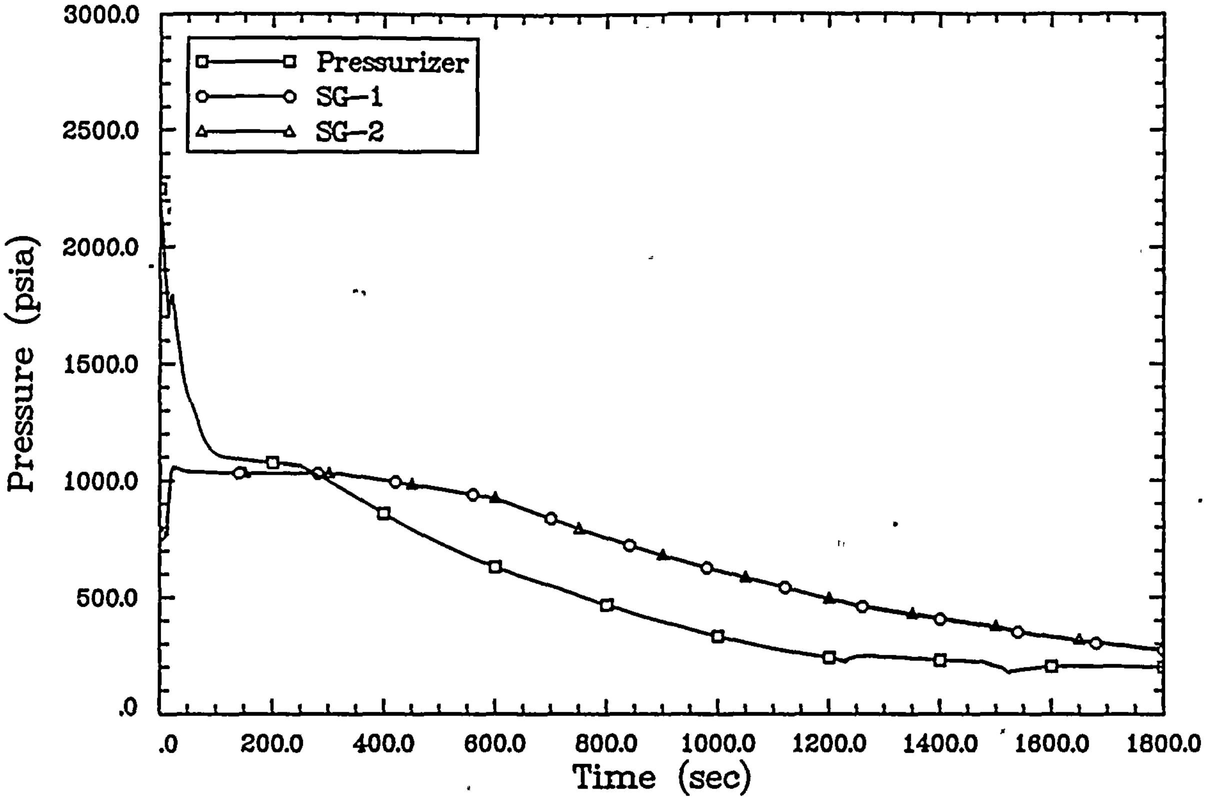

Figure 4.1

System Pressures for 0.10 ft^2 Break

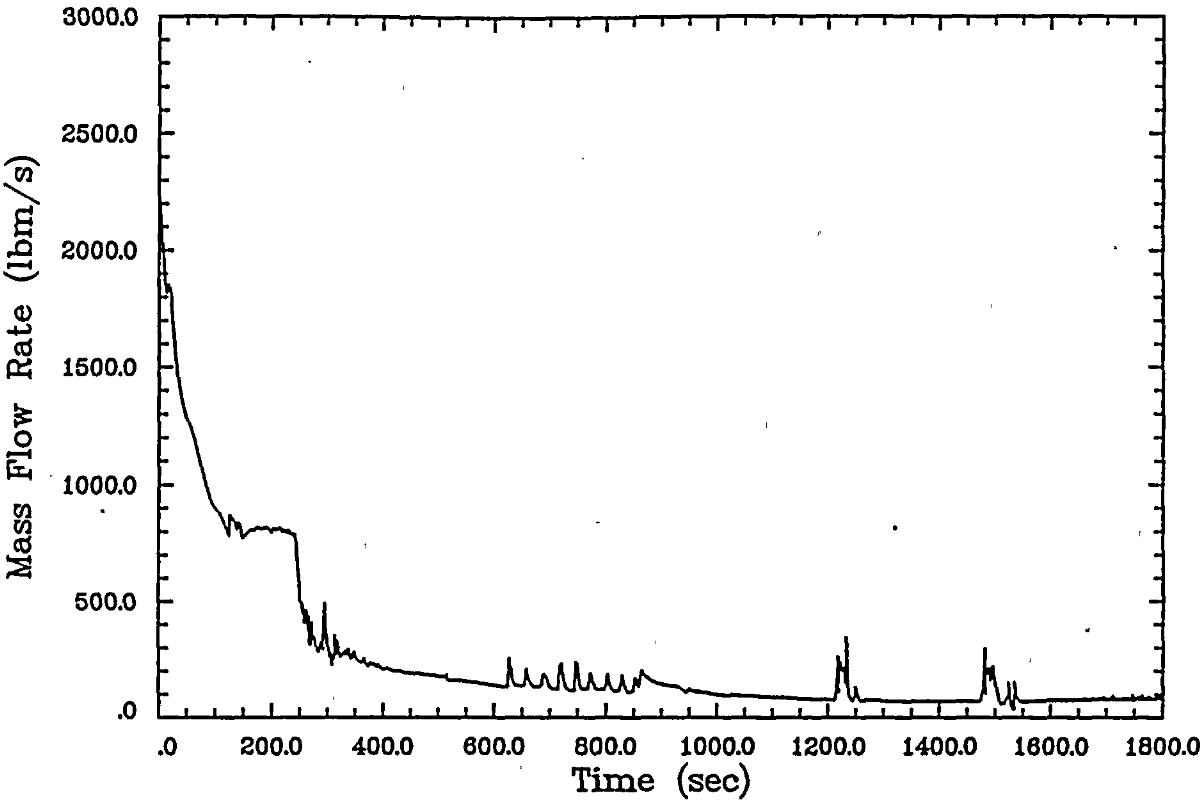

Figure 4.2

Break Flow Rate for 0.10 ft^2 Break

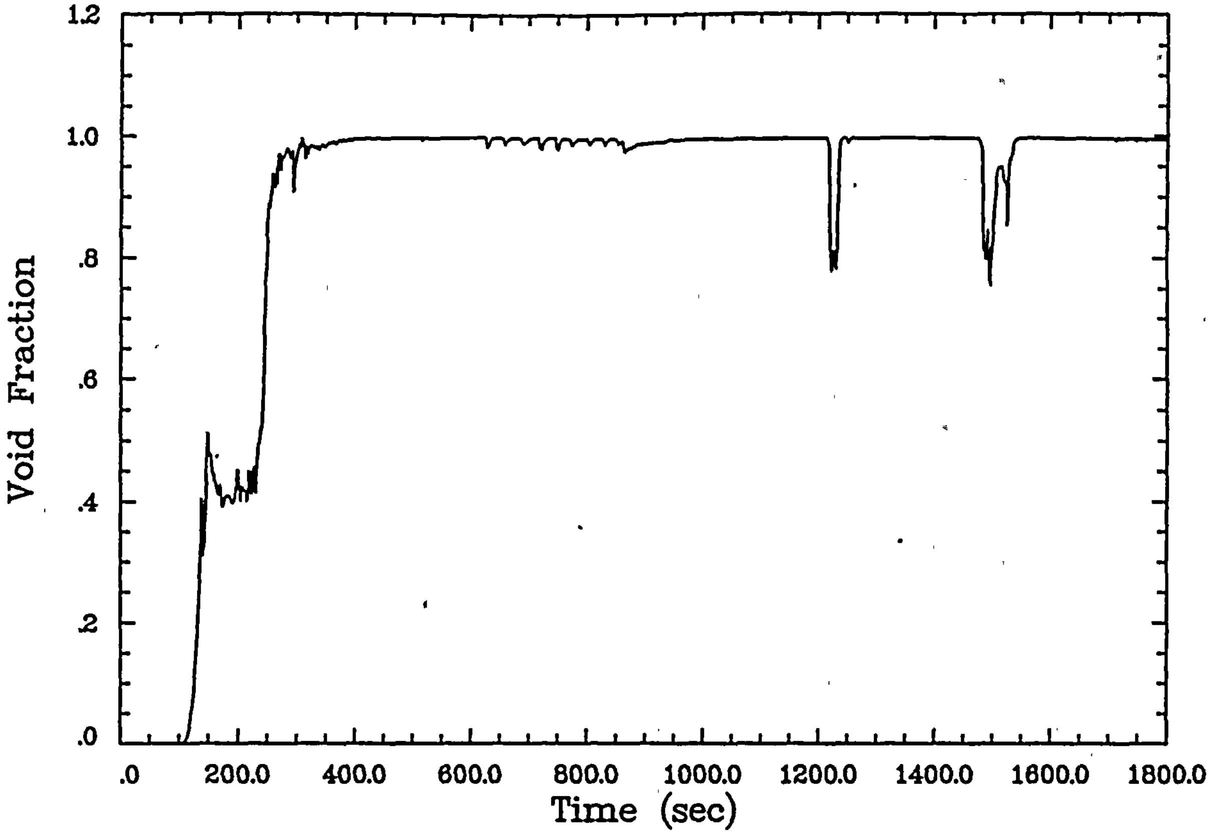

Figure 4.3

Void Fraction at the Break for 0.10 ft^2 Break

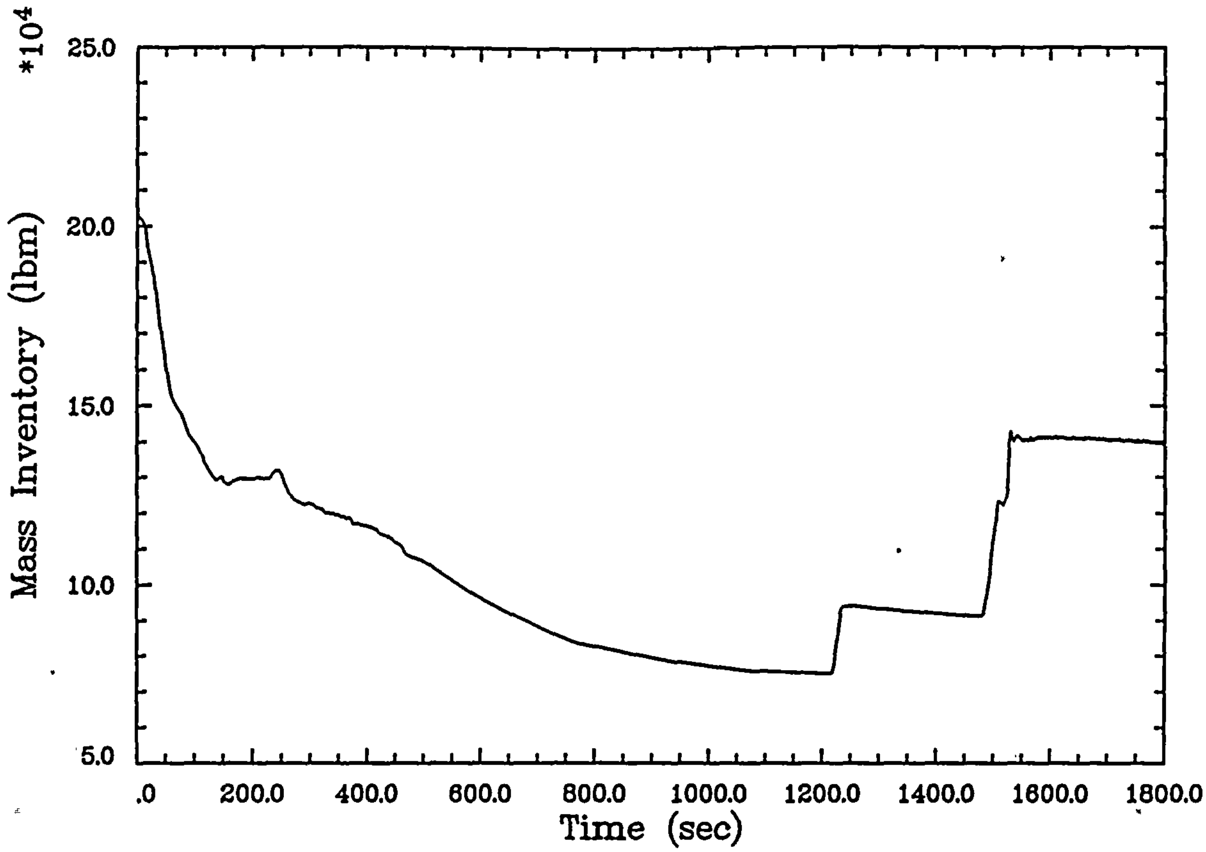

Figure 4.4

Reactor Vessel Mass Inventory for 0.10 ft^2 Break

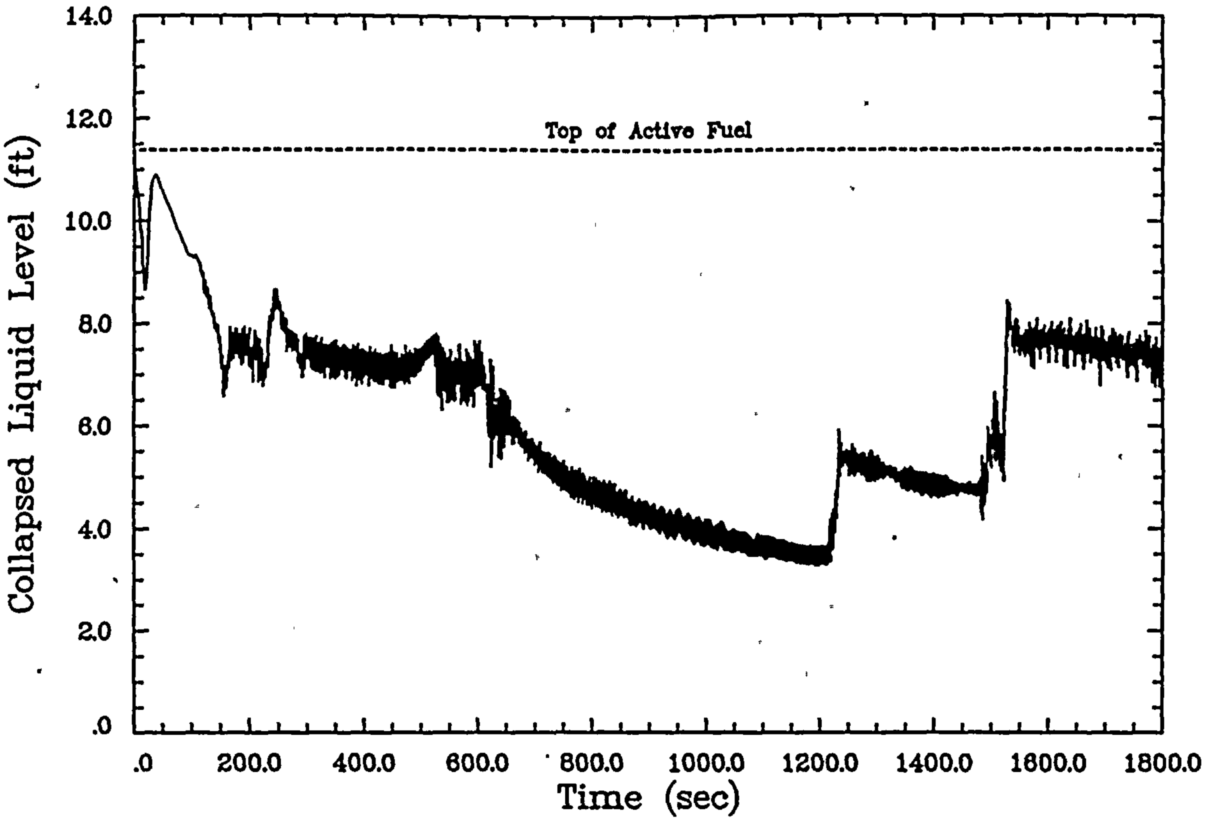

Figure 4.5

Core Collapsed Liquid Level for 0.10 ft^2 Break

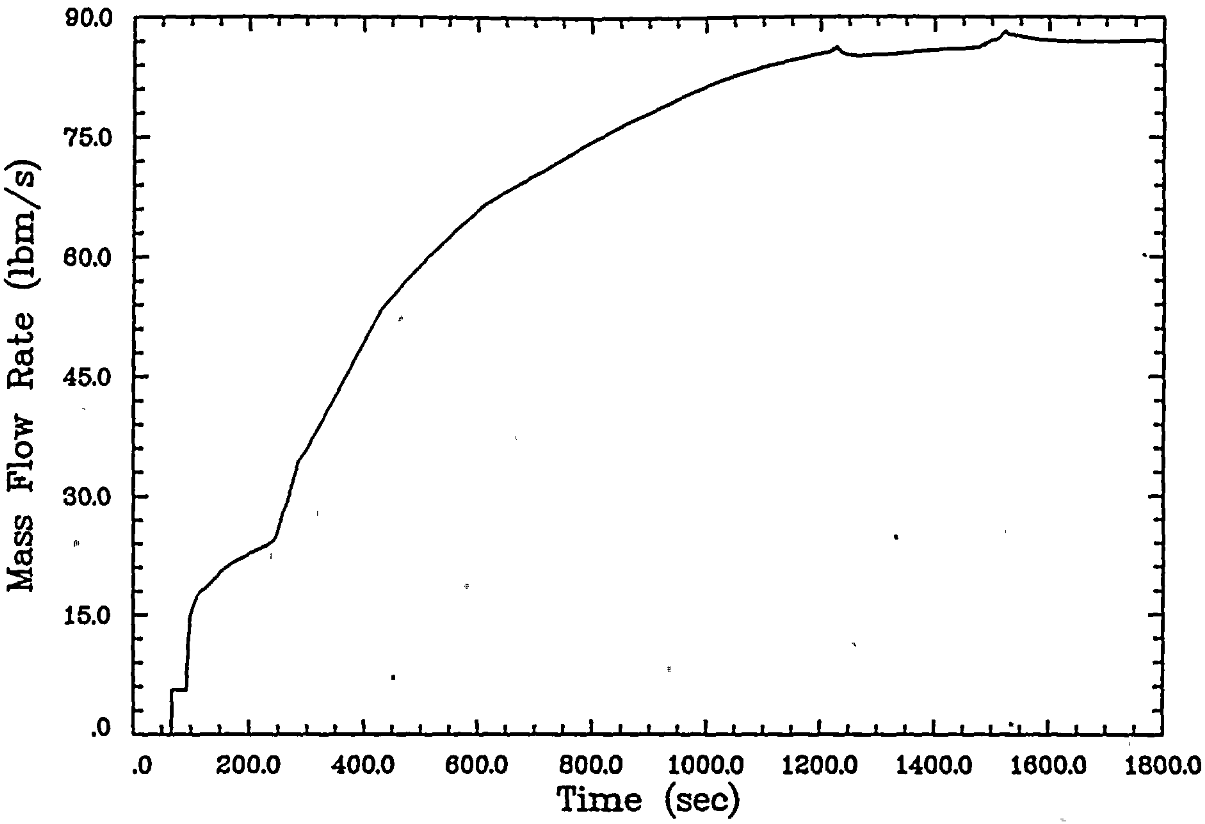

Figure 4.6

Total HPSI Flow Rate for 0.10 ft^2 Break

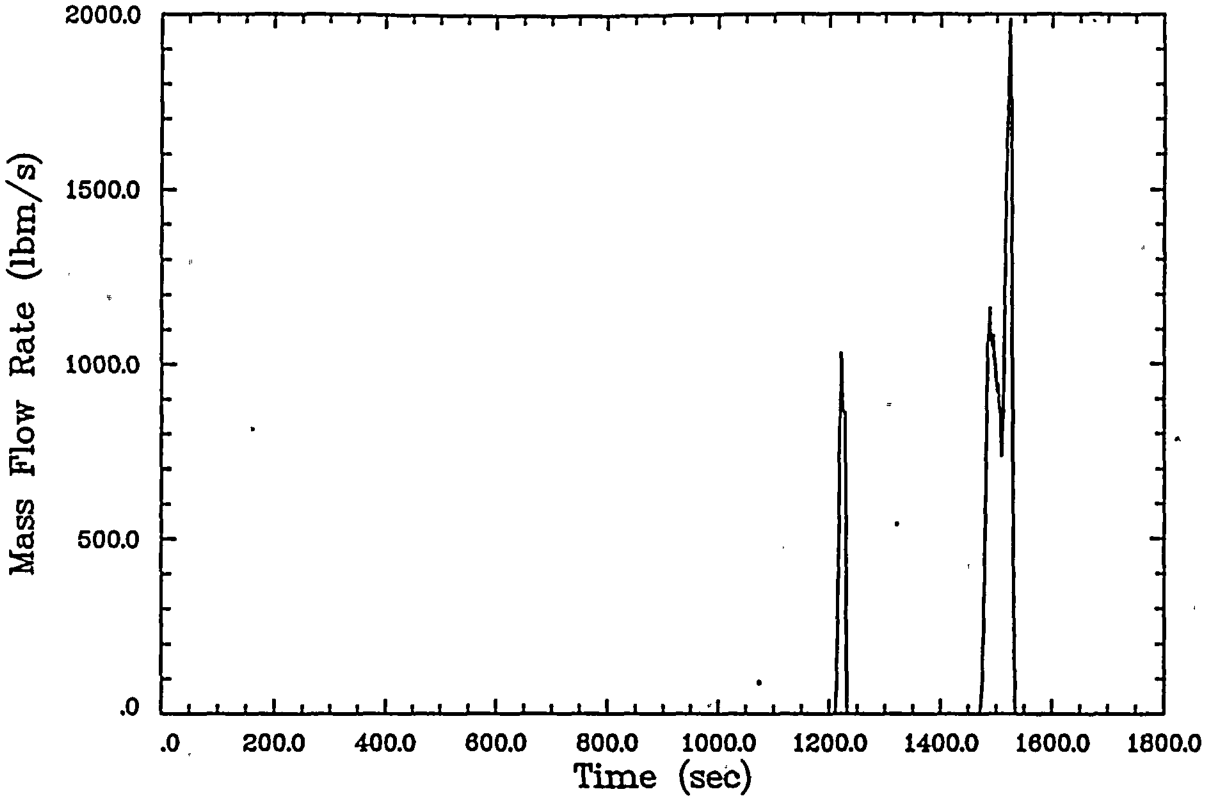

Figure 4.7

Total SIT Flow Rate for 0.10 ft^2 Break

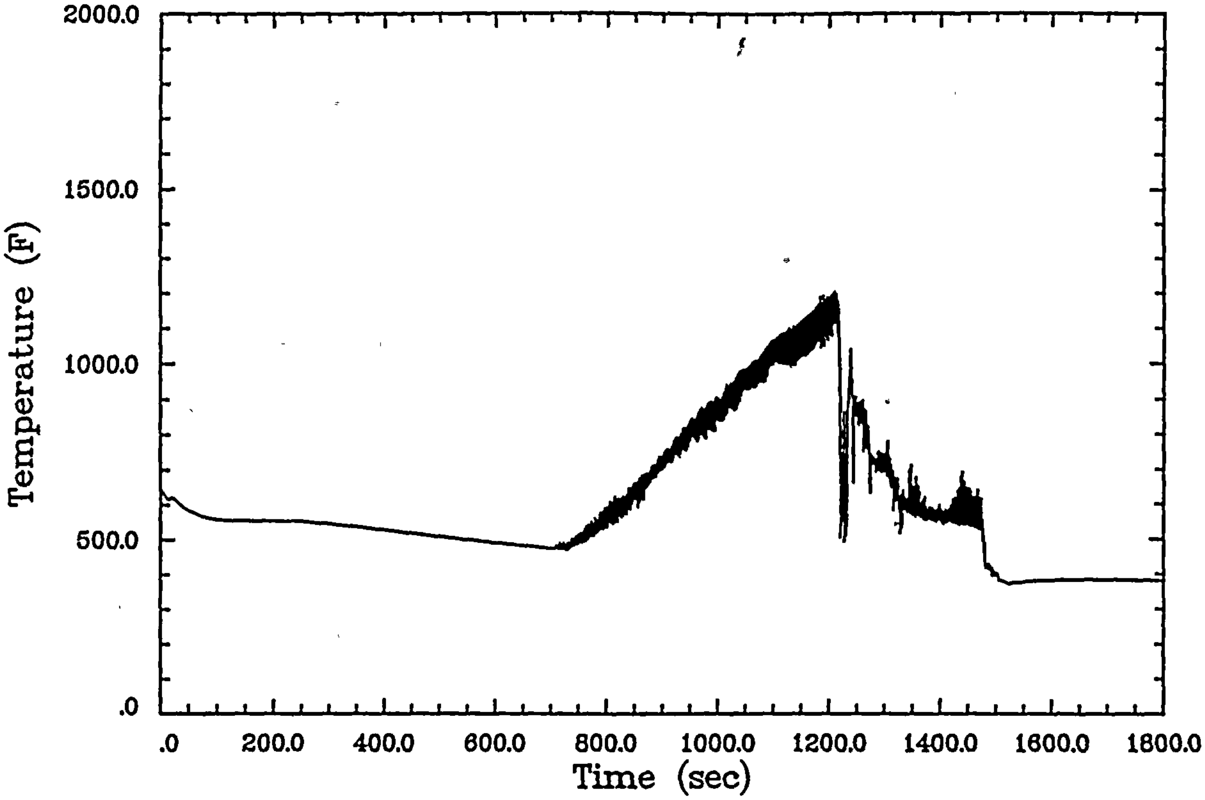

Figure 4.8

Fluid Temperature at PCT Node for 0.10 ft^2 Break

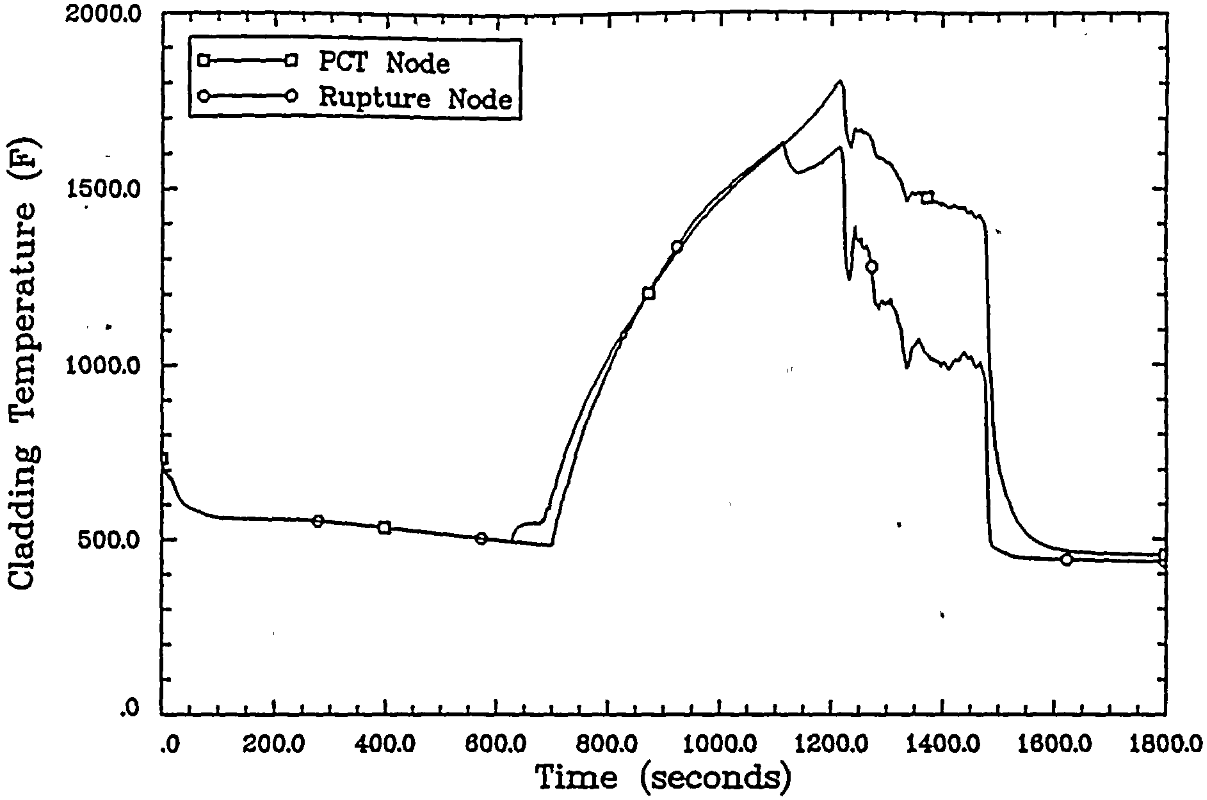

Figure 4.9

Hot Rod Temperature Response for 0.10 ft² Break

5.0 REFERENCES

1. *St. Lucie Unit 1 Chapter 15 Event Review and Analysis for 30% Steam Generator Tube Plugging*, EMF-96-135, May 1996.

2. *Siemens Power Corporation - Nuclear Division St. Lucie Unit 1 Small Break LOCA Analysis*, EMF-92-148, Revision 1, May 1994.

3. *Exxon Nuclear Company Evaluation Model-EXEM PWR Small Break Model*, XN-NF-82-49(P)(A), Revision 1, Supplement 1 and Correspondence, December 1994.

4. *Exxon Nuclear Company Evaluation Model-EXEM PWR Small Break Model*, XN-NF-82-49(P), Revision 1, June 1986.

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