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ST. LUCIE UNIT 1 LOCA ANALYSIS USING THE EXEM/PWR ECCS MODEL

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The author wishes to express his appreciation to the individuals listed below for their efforts in performing various phases of the St. Lucie Unit 1 LOCA analyses as well as their comments and suggestions.

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1.0 INTRODUCTION AND SUMMARY

This report provides results of a LOCA ECCS analysis supporting St. Lucie Unit 1 reactor operation at 2700 MWt with Exxon Nuclear Company (ENC) supplied fuel. Contained within the report are results of (1) a LOCA break spectrum analysis, and (2) a limiting break recalculation using the GAPEX stored energy model and increased radial peaking. The break spectrum analysis uses the RODEX2 stored energy model. All analyses were performed using the EXEM/PWR ECCS evaluation model[1,2]. For the limiting break (0.4 DECLG case) with beginning-of-life (BOL) stored energy (GAPEX value) and end-of-cycle (EOC) fission gas release, the calculated peak clad temperature was 2059°F with the axial power peak at 70 percent of the core height and a peak linear heat generation rate of 15.30 kw/ft (102% of 15.00 kw/ft). The limiting break analysis was performed in conformance to Appendix K of 10 CFR 50, yield results which satisfy the NRC criteria specified in 10 CFR 50.46. The analysis applies only for Cycle 6 operation of St. Lucie Unit 1.

The LOCA break spectrum calculations included guillotine break configurations for double-ended cold leg pipe breaks (DECLG) with discharge coefficients of 1.0, 0.6 and 0.4. The split configuration breaks of the cold leg pipe were also calculated with a break area equal to twice the cross-sectional pipe area (DECLS, 10.01 ft^2), then with break areas of 6.01 and 0.8 square feet. The break spectrum analysis was performed for a core composed of ENC fuel at nominal beginning-of-life (BOL) conditions. The results of the spectrum analysis identified the limiting break to be the 0.4 DECLG case.

A detailed discussion of the break spectrum results is provided in Section 2.0. All of the calculations in the break spectrum were performed at a core power of 2754 MWt, which is 102 percent of rated power.

The break spectrum calculations were performed with a version of the RODEX2 code supplying the initial stored energy. The NRC is currently reviewing the RODEX2 code, with approval not expected in time for St. Lucie Unit 1 Cycle 6 startup. Therefore, ENC repeated the limiting break analysis using the GAPEX stored energy model previously approved by NRC. A combination of BOL maximum stored energy and EOL maximum fission gas release was used to bound Cycle 6 operation.

2.0 BREAK SPECTRUM ANALYSIS

2.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The analyses for large breaks specified by 10 CFR 50.46[3], "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors," is presented in this section. The results of the loss of coolant accident analysis are shown in Tables 2.1 and 2.2, and indicate compliance with the Acceptance Criteria. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are as described in XN-75-41, Volumes I and II, and supplements[1]; ENC EXEM/PWR model is described in XN-NF-82-20(P) and supplements[2]. The detailed system models are as given in the example problem report for a Combustion Engineering 2x4 PWR which is Supplement 3 of XN-NF-82-20(P).

For the purpose of loss-of-coolant accident (LOCA) analyses, a LOCA is defined as a hypothetical rupture of the Reactor Primary Coolant System piping, up to and including the double-ended rupture of the largest pipe in the Reactor Coolant System or of any line connected to that system up to the first closed valve.

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer lower pressure trip setpoint is reached. Reactor trip and scram were conservatively neglected for the large break analyses. A Safety Injection System signal is actuated when the appropriate setpoint (high containment pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.
2. Injection of borated water provides heat transfer from the reactor core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire Reactor Coolant System contains subcooled liquid which transfers heat from the core by forced convection cooling. After the break develops, the time to departure from nucleate boiling (DNB) is calculated consistent with Appendix K of 10 CFR 50[3]. Post-DNB core heat transfer (both transition and film boiling occurring) is also calculated in accordance with Appendix K of 10 CFR 50. As the core becomes uncovered, both turbulent and laminar forced convection to steam are considered as core heat transfer mechanisms.

When the Reactor Coolant System pressure falls below 230 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator ECC water bypasses the core and goes out through the break until the termination of bypass. This conservatism is consistent with Appendix K of 10 CFR 50.

2.2 THERMAL ANALYSIS

2.2.1 Method of Analysis

For breaks greater than 0.8 ft^2, the RELAP4-EM code in EXEM/PWR is used to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of flow out of the break. A specialized calculation (RELAP4-EM/HOT CHANNEL) is used to

calculate cladding temperatures using time dependent boundary conditions in the upper and lower plenum volumes from the basic blowdown analysis. Beyond the point of refill to the bottom of the core, a specialized calculation (REFLEX) is applied to determine the reflooding rate and system conditions. After end-of-bypass (EOBY), the program TOODEE2 is used to calculate peak clad temperatures.

2.2.2 Large Break LOCA Analysis Modeling

The St. Lucie Unit 1 nuclear power plant is a 2x4 Combustion Engineering pressurized water reactor with a dry containment. The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow paths or "junctions" as described in Supplement 3 of XN-NF-82-20(2). The nodalization in Figure 2.1 differs from the example problem nodalization in the broken loop cold leg region. The number of broken loop cold leg volumes have been reduced.

Five percent of the steam generator tubes were assumed to be uniformly plugged. The unbroken loop was assumed symmetrical and modeled the same as the broken loop except for the break nodalization and the pressurizer. Pump performance curves characteristic of the St. Lucie Unit 1 pumps were used in the analysis. System input parameters are given in Table 2.3.

The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The axial power profile used for the break spectrum analysis is a top skewed curve with the power peak above the core midplane.

The values for the primary coolant system core inlet temperatures and the steam generator secondary side pressure were set for the St. Lucie Unit 1 plant based upon information provided by the utility. The values of the core inlet temperature and the steam generator secondary side pressure are 549°F and 870 psig, respectively.

The containment backpressure for the analysis of the postulated LOCA was evaluated in accordance with the discussion presented in XN-75-41, Supplement 5, Section 4.6. A containment analysis was performed using the computer code CONTEMPT-LT, Version 22, modified as described in Supplement 5, Revision 1 of XN-75-41(1). The condensing heat transfer coefficient is modeled in accordance with Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation"(4). The containment parameters used in the containment analysis to determine the ECCS backpressure are presented in Table 2.4.

The reflood nodalization for the guillotine and split breaks in the break spectrum are shown in Figures 2.2 and 2.3. These nodalizations include a leakage path between the upper plenum and the upper downcomer.

2.3 BREAK SPECTRUM RESULTS

Using the EXEM/PWR codes, transient system behavior is determined by solving the governing conservation equations for mass, energy, and momentum. Energy transport, flow rates, and heat transfer are determined from appropriate correlations. Table 2.1 presents the timing and sequence of events as determined for the large break guillotine configuration with discharge coefficients of 1.0, 0.6 and 0.4 and the split break configuration with break areas of 10.01, 6.01, and 0.8 square feet.

The blowdown calculations for the break spectrum were initialized with the ENC fuel performance code RODEX2. Pending NRC approval of RODEX2, the limiting break in the spectrum was reanalyzed using GAPEX for fuel rod stored energy initialization. The fuel initialization using GAPEX encompasses the maximum fuel rod stored energy (BOL) and end-of-cycle (EOC) fission gas release for Cycle 6. The input radial peaking factor was also increased to bound measurement uncertainties while the peak liner heat generation rate was maintained at 15 kw/ft. The limiting break results which have been initialized with GAPEX and include the increased radial peaking factor are identified in Tables 2.1 and 2.2.

In general, the transient events occur slower for smaller discharge coefficients or break sizes. Table 2.2 presents the peak clad temperatures and maximum metal-water reaction results for the above spectrum of break cases. This range of break sizes was determined to include the limiting case for peak clad temperature.

The analysis of the loss-of-coolant accident is performed at 102% of 2700 MWt (2754 MWt). The core power and other parameters used in the analyses are given in Table 2.3. Since there is usually margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation.

For the result discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table 2.2 for each break size analyzed.

Figures 2.4 througn 2.22 present the results of the revised analysis for the limiting break (0.4 DECLG). Unless otherwise noted, zero time corresponds to the time of break initiation.

The maximum peak cladding temperature of 2059°F was calculated for the double-ended cold leg guillotine break configuration (C_D = 0.4) with a total linear heat generation rate of 15.30 kw/ft for ENC fuel (102% of 15.00 kw/ft). The maximum local metal-water reaction is less than 5% and the core-wide reaction is less than 1%, all well below the limits set by the criteria of 10 CFR 50.46.

ENC has performed numerous analyses and sensitivity studies on PWR systems using the ENC ECCS evaluation model. These studies have demonstrated the adequacy of the system nodalization used. In addition, these studies have shown that for transient conditions similar to those calculated for the St. Lucie Unit 1 reactor during the LOCA, the reactor coolant inlet pipe or cold leg is the worst break location.

NSSS vendor analyses have shown large breaks to be more limiting than small breaks for St. Lucie Unit 1 ECCS analyses.

Table 2.1 St. Lucie Unit 1 Large Break Events

Event	Time (seconds)						
	DECLG* (C_D=0.4)	DECLG (C_D=0.4)	DECLG (C_D=0.6)	DECLG (C_D=1.0)	1.0 DECLS (10.01 ft^2)	0.6 DECLS (6.01 ft^2)	0.08 DECLS (0.8 ft^2)
Start	0.	0.	0.	0.	0.	0.	0.
Initiate Break	0.05	0.05	0.05	0.05	0.05	0.05	0.05
Safety Injection Signal	1.20	1.20	0.98	0.82	0.86	0.91	5.20
Pressurizer Empties	8.95	8.95	8.85	8.80	9.10	9.25	9.25
Accumulator Injection, Intact Loops	22.58	22.65	18.50	16.65	16.65	17.33	99.70
End-of-Bypass	28.01	27.03	24.27	21.98	21.21	21.96	103.53
Safety Injection Flow, SIS	31.20	31.20	30.98	30.98	30.82	30.86	35.20
Start of Reflood	45.58	44.42	42.07	39.78	38.87	39.61	120.57
Peak Clad Temperature Reached	161.5	156.5	159.1	161.6	156.0	153.1	420.5

*Initial Stored Energy calculated with GAPEX code and radial peaking factor increased.

Table 2.2 St. Lucie Unit 1 Large Break Results

Event	DECLG* (C_D=0.4)	DECLG (C_D=0.4)	DECLG (C_D=0.6)	DECLG (C_D=1.0)	1.0 DECLS (10.01 ft^2)	0.6 DECLS (6.01 ft^2)	0.08 DECLS (0.8 ft^2)
Peak Cladding Temperature	2059	1770	1730	1730	1719	1697	1663
Peak Temperature Location, ft.	9.22	9.22	9.22	9.22	9.22	8.97	8.97
Local Zr/H_2O Reaction (Max.), % at 300 sec.	4.0	1.5	1.4	1.4	1.3	1.2	1.0
Local Zr/H_2O Location, ft.	8.7	9.22	9.22	9.22	9.22	8.47	8.97
Total H_2 Generation, % of total Zr Reacted	<1%	<1%	<1%	<1%	<1%	<1%	<1%
Hot Rod Burst Time, sec.	39.8	52.7	63.9	63.9	64.1	71.1	309.5
Hot Rod Burst Location, ft.	8.0	8.5	8.5	8.7	8.7	8.7	8.2
Linear Heat Generation Rate, kw/ft at BOCREC	.7290	.7318	.7396	.7462	.7514	.7469	.6012

*Initial Stored Energy Calculated with GAPEX code and radial peaking factor increased.

Table 2.3 St. Lucie Unit 1 PWR Data

Primary Heat Output, MWt	2700*
Primary Coolant Flow, lbm/hr	1.394×10^8
Primary Coolant Volume, ft^3	19,214**
Operating Pressure, psia	2250
Inlet Coolant Temperature, ^oF	549
Reactor Vessel Volume, ft^3	4402
Pressurizer Volume, Total, ft^3	1500
Pressurizer Volume, Liquid, ft^3	800
Accumulator Volume, Total, ft^3 (one of four)	2020
Accumulator Volume, Liquid, ft^3	1090
Accumulator Pressure, psia	230
Steam Generator Heat Transfer Area, ft^2 (one of two)	74,722
Steam Generator Secondary Flow, lbm/hr	5.899×10^6
Steam Generator Secondary Pressure, psia	885
Reactor Coolant Pump Head, ft	280
Reactor Coolant Pump Speed, rpm	886
Moment of Inertia, lbm-ft^2/rad	101,900
Cold Leg Pipe, I.D., in.	30
Hot Leg Pipe, I.D., in.	42
Pump Suction Pipe, I.D., in.	30

* Primary Heat Output used in RELAP4-EM Model - 1.02 x 2700 = 2754 MWt.
** Includes total accumulator and pressurizer volume.

Table 2.3 St. Lucie Unit 1 PWR Data (Cont.)

Fuel Assembly Rod Diameter, in*	.440
Fuel Assembly Rod Pitch, in*	.580
Fuel Assembly Pitch, in*	8.180
Fueled (Core) Height, in*	136.7
Fuel Heat Transfer Area, ft^2	50,117
Fuel Total Flow Area, ft^2	53.19
Steam Generator Tube Plugging (Assumed Uniform)	5%

* Enc Fuel Parameters

Table 2.4 St. Lucie Unit 1 Dry Containment Data

Containment Physical and Thermal Parameters

Net Free Volume	2.511×10^6 ft^3
Enclosure Building Temperature	38°F
Initiation Time for:	
Spray Flow	30.0 sec
Fan Coolers	30.0 sec
Containment Initial Conditions:	
Temperature	90°F
Pressure	14.6 psia
Relative Humidity	100%
Containment Spray Water:	
Temperature	55°F
Flow Rate (Total, 2 pumps)	6750 gpm

Fan Air Cooler Capacity (Total, 4 coolers)

Vapor Temperature (°F)	Capacity (Btu/hr)
60	0.
120	5.00×10^7
180	1.06×10^8
220	1.67×10^8
264	3.00×10^8

Thermal Conductivity and Volumetric Heat Capacity

Materials	Thermal Conductivity (Btu/hr-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Steel	25.9	53.6
Structural Concrete	1.0	34.2
Stainless Steel	9.8	54.0
Galvanizing	64.0	40.6

Table 2.4 St. Lucie Unit 1 Dry Containment Data (Cont.)

Containment Passive Heat Sinks

	Description	Material	Thickness	Surface Area Ft^2
1.	Containment Shell	Steel	.1171 ft	86,700
2.	Miscellaneous Concrete	Concrete	1.5 ft	87,751
3.	Floor Slab	Concrete	20 ft	12,682
4.	Galvanized Steel	Zinc Steel	0.00059 ft 0.014 ft	130,000
5.	Carbon Steel	Steel	0.031	25,000
6.	Stainless Steel	Steel	0.038 ft	22,300
7.	Miscellaneous Steel	Steel	0.063 ft	40,000
8.	Miscellaneous Steel	Steel	0.021 ft	41,700
9.	Miscellaneous Steel	Steel	0.177 ft	7,000
10.	Imbedded Steel	Steel Concrete	0.071 ft 7.0 ft	18,000

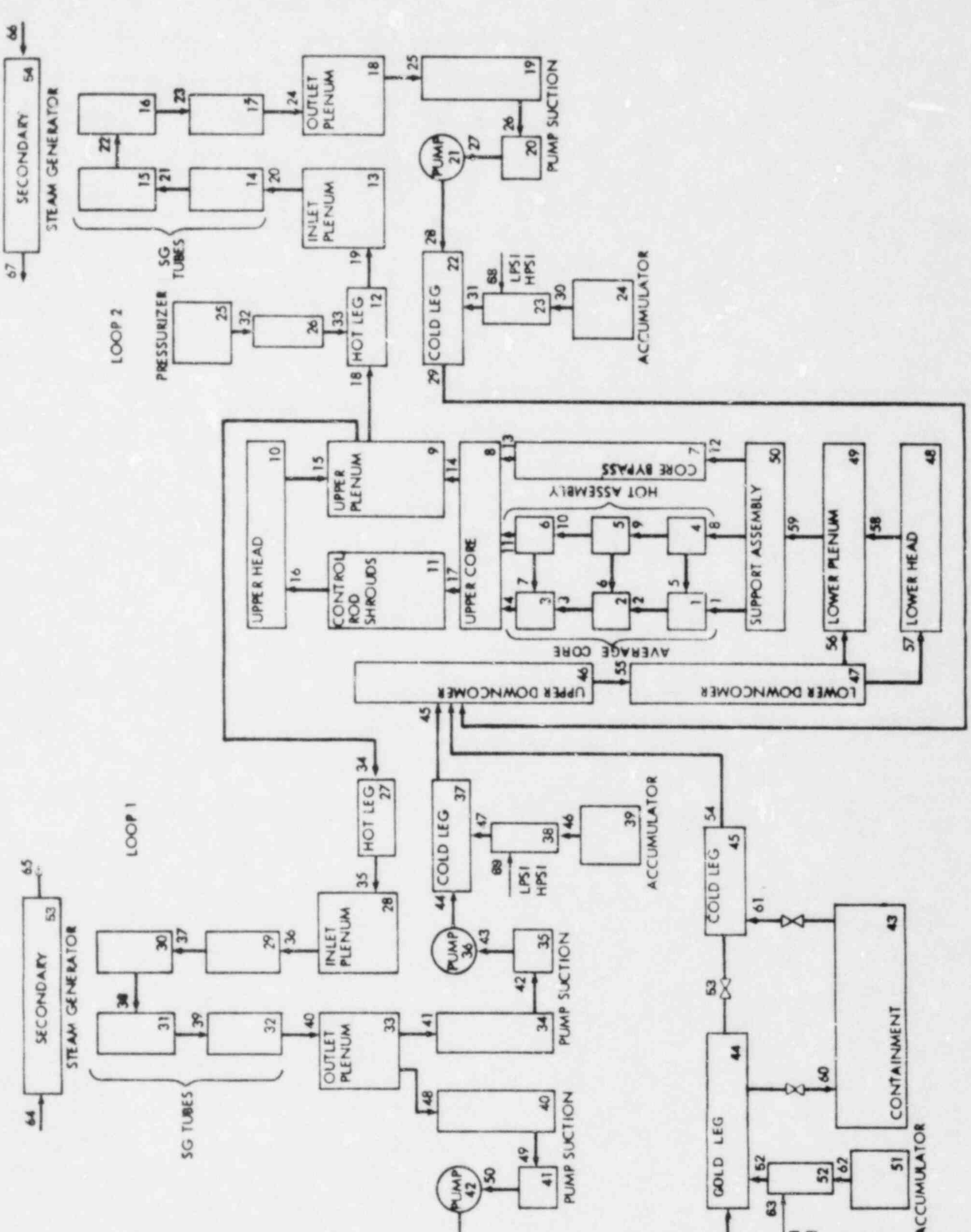

Figure 2.1 RELAP4-EM Blowdown System Nodalization For St. Lucie Unit 1

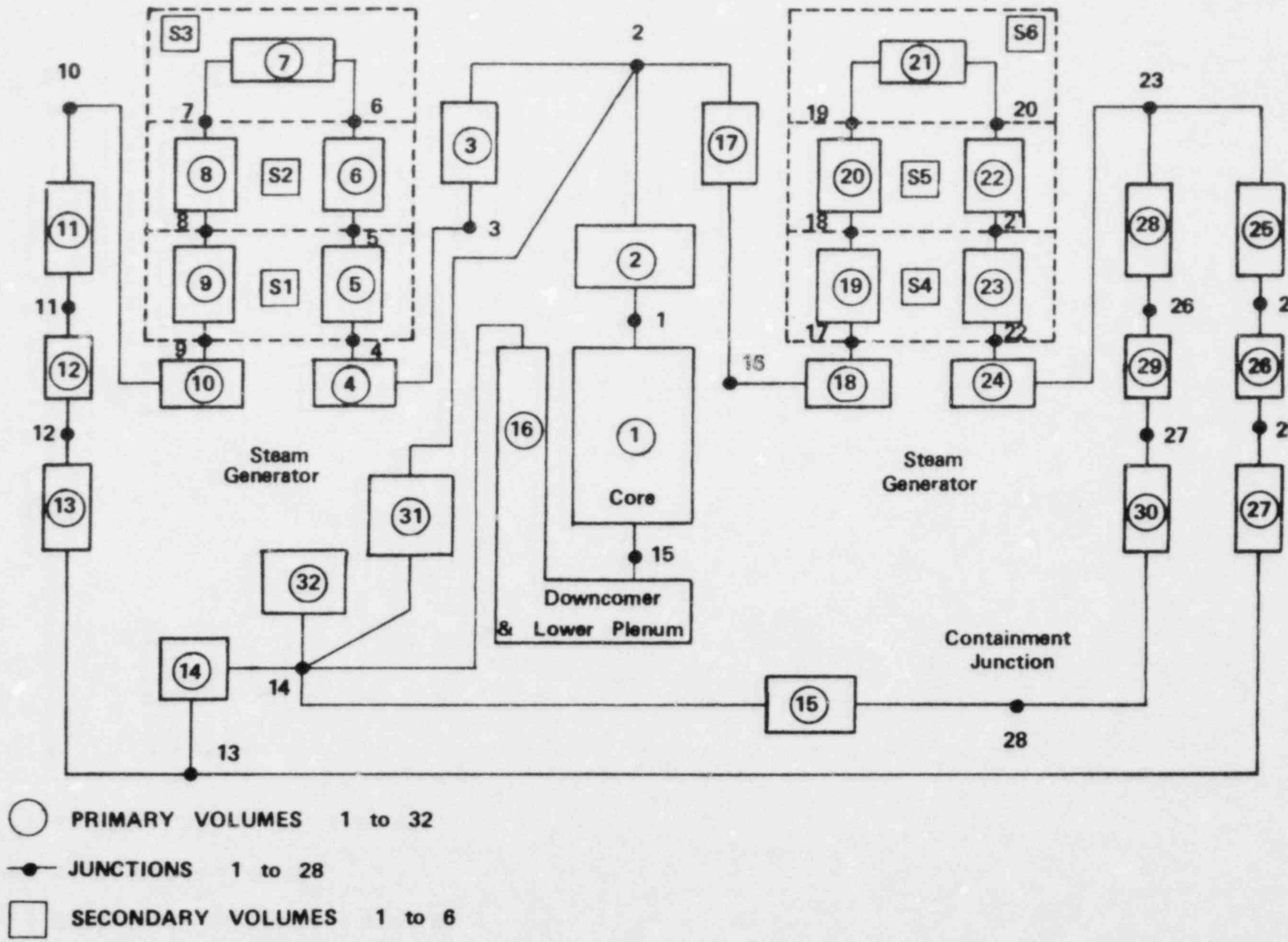

Figure 2.2 Reflood Nodalization For Guillotine Breaks For St. Lucie Unit 1

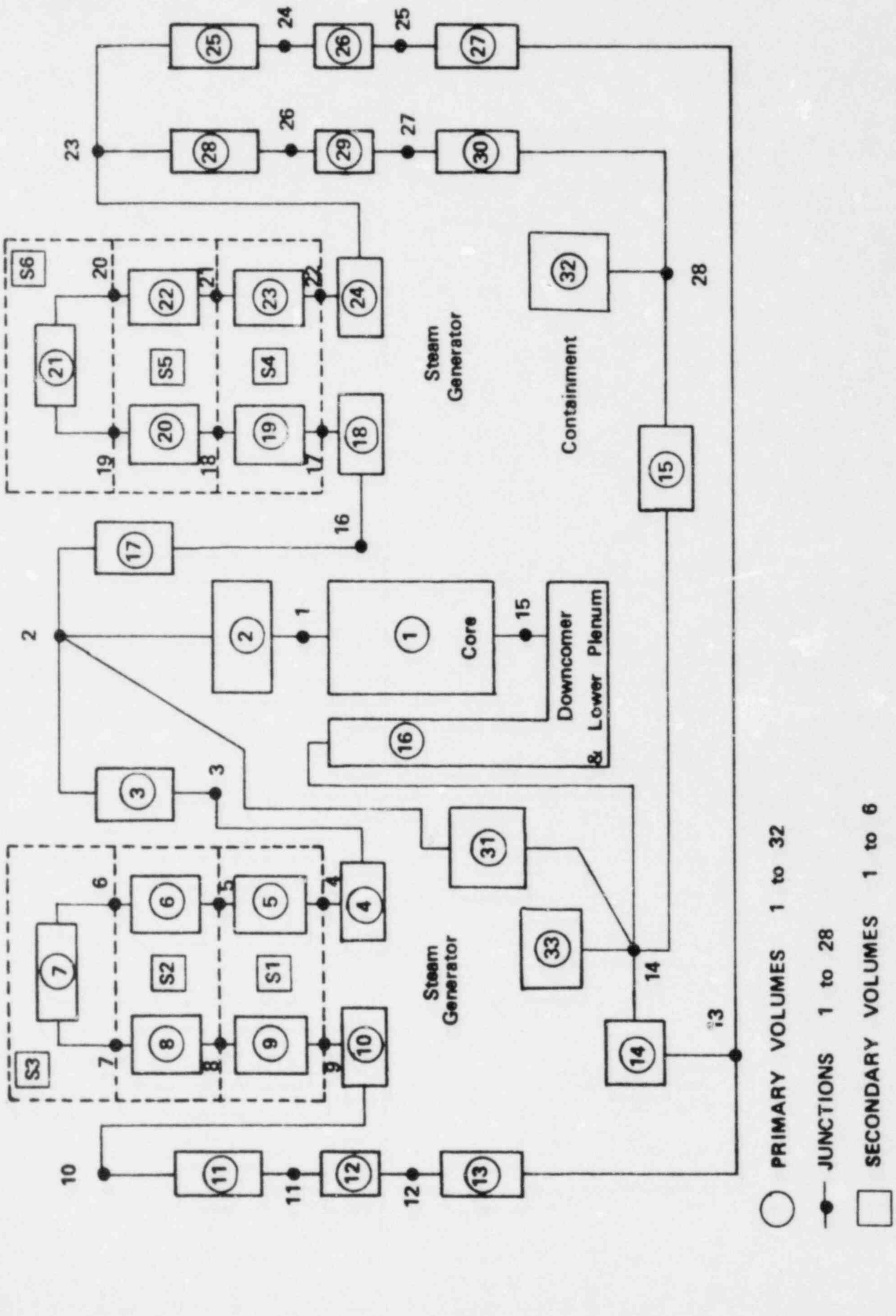

Figure 2.3 Reflood Nodalization For Split Breaks For St. Lucie Unit 1

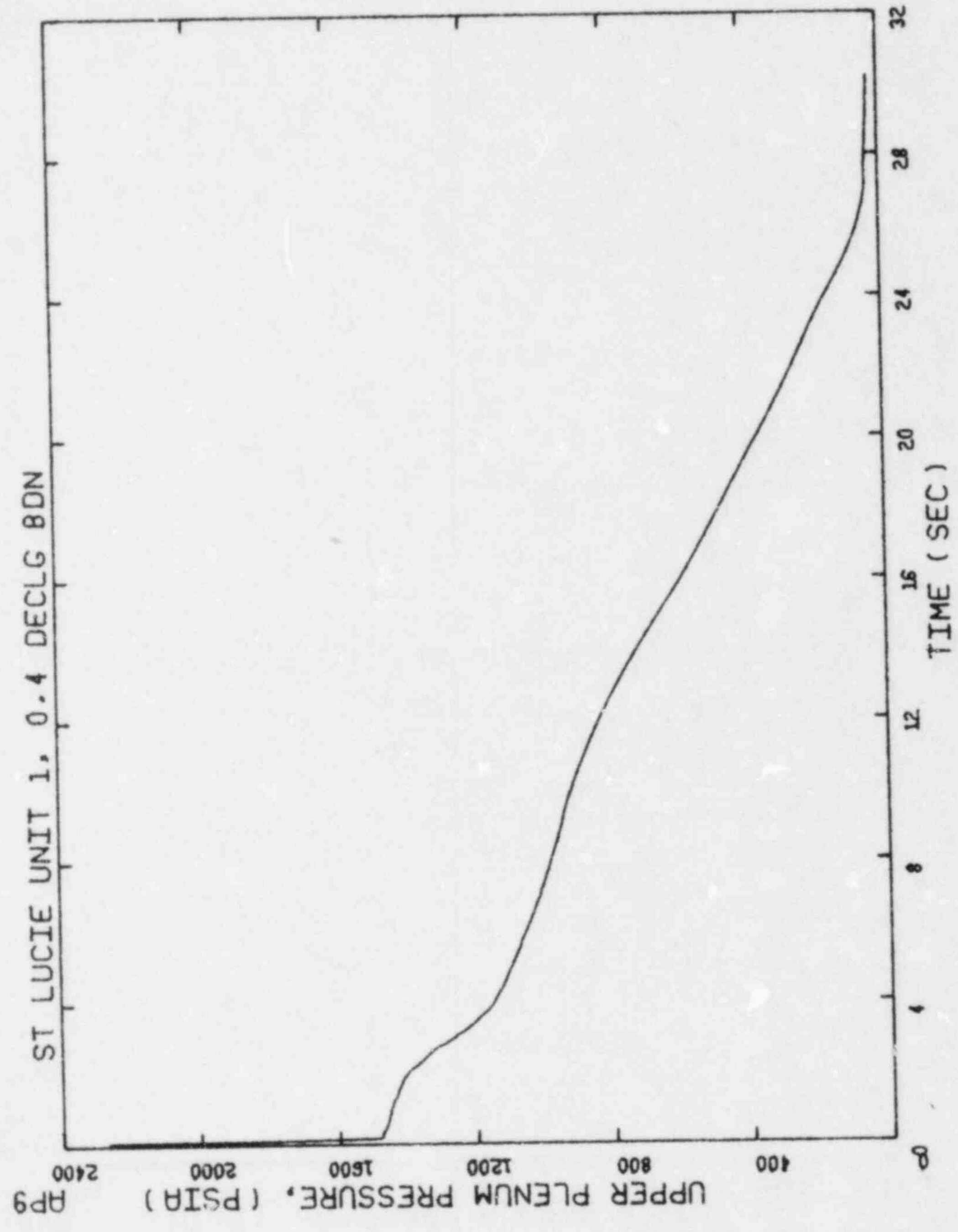

Figure 2.4 Blowdown System Pressure, 0.4 DECLG Break

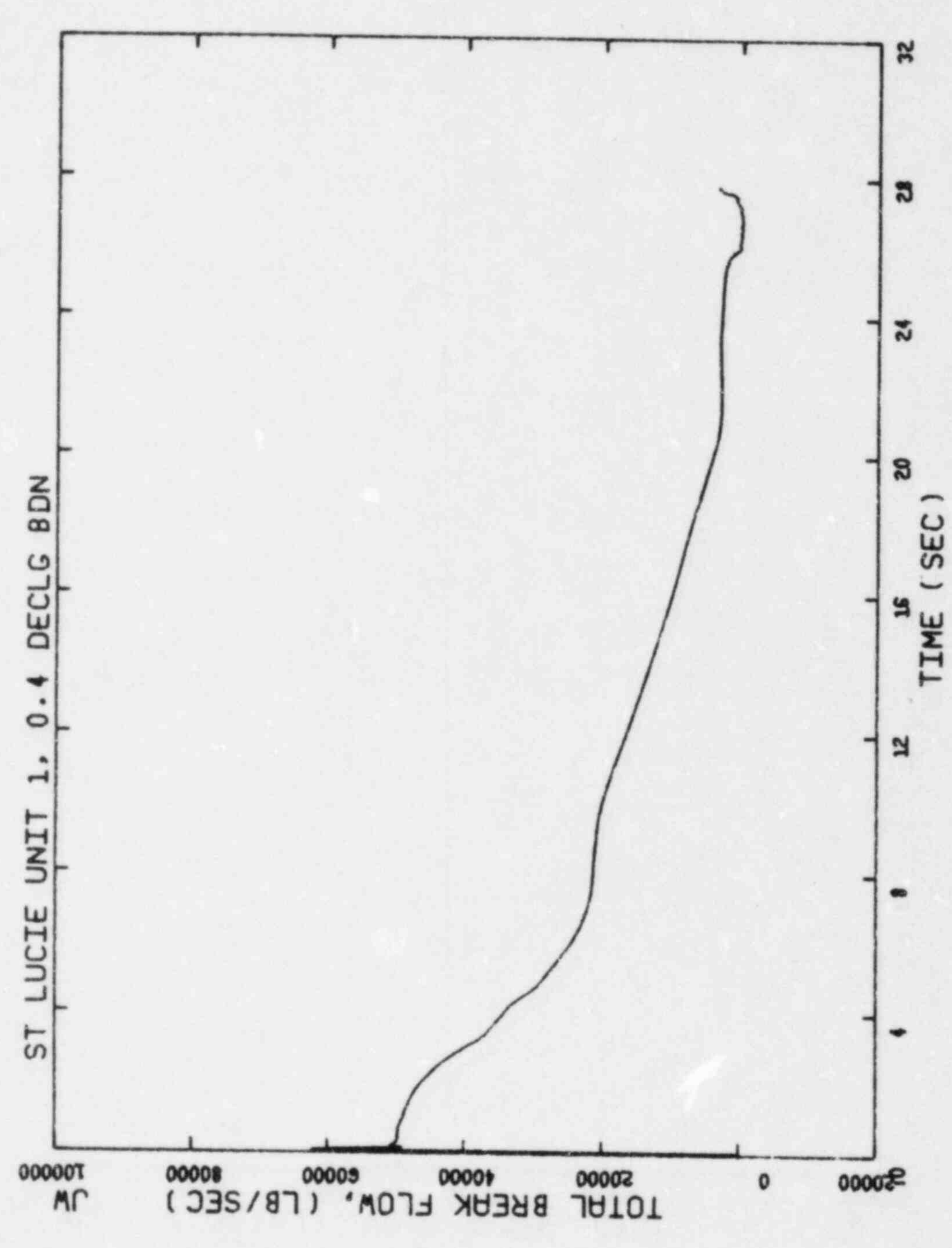

Figure 2.5 Blowdown Total Break Junction Flow Rate, 0.4 DECLG Break

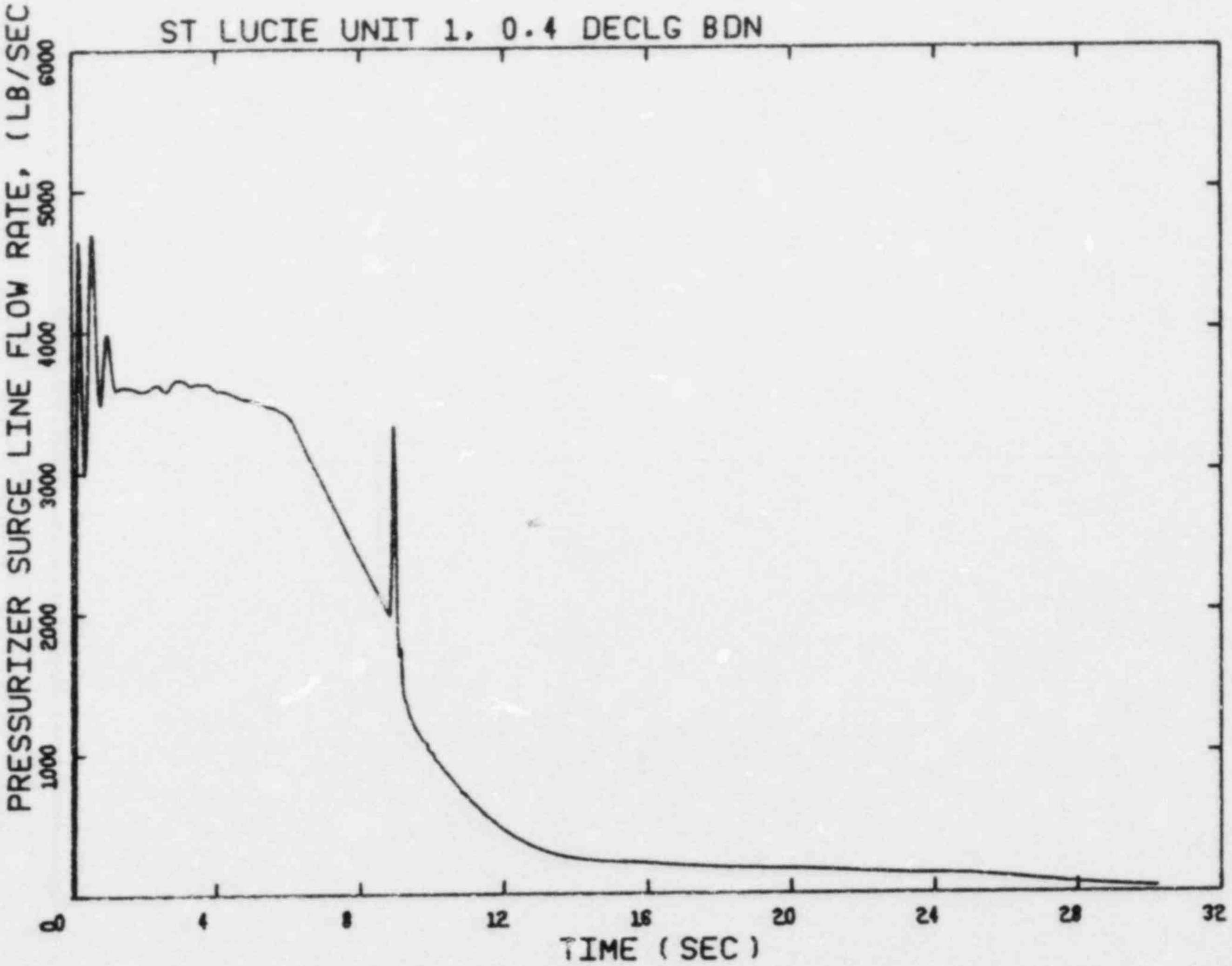

Figure 2.6 Blowdown Pressurizer Surge Line Flow Rate, 0.4 DECLG Break

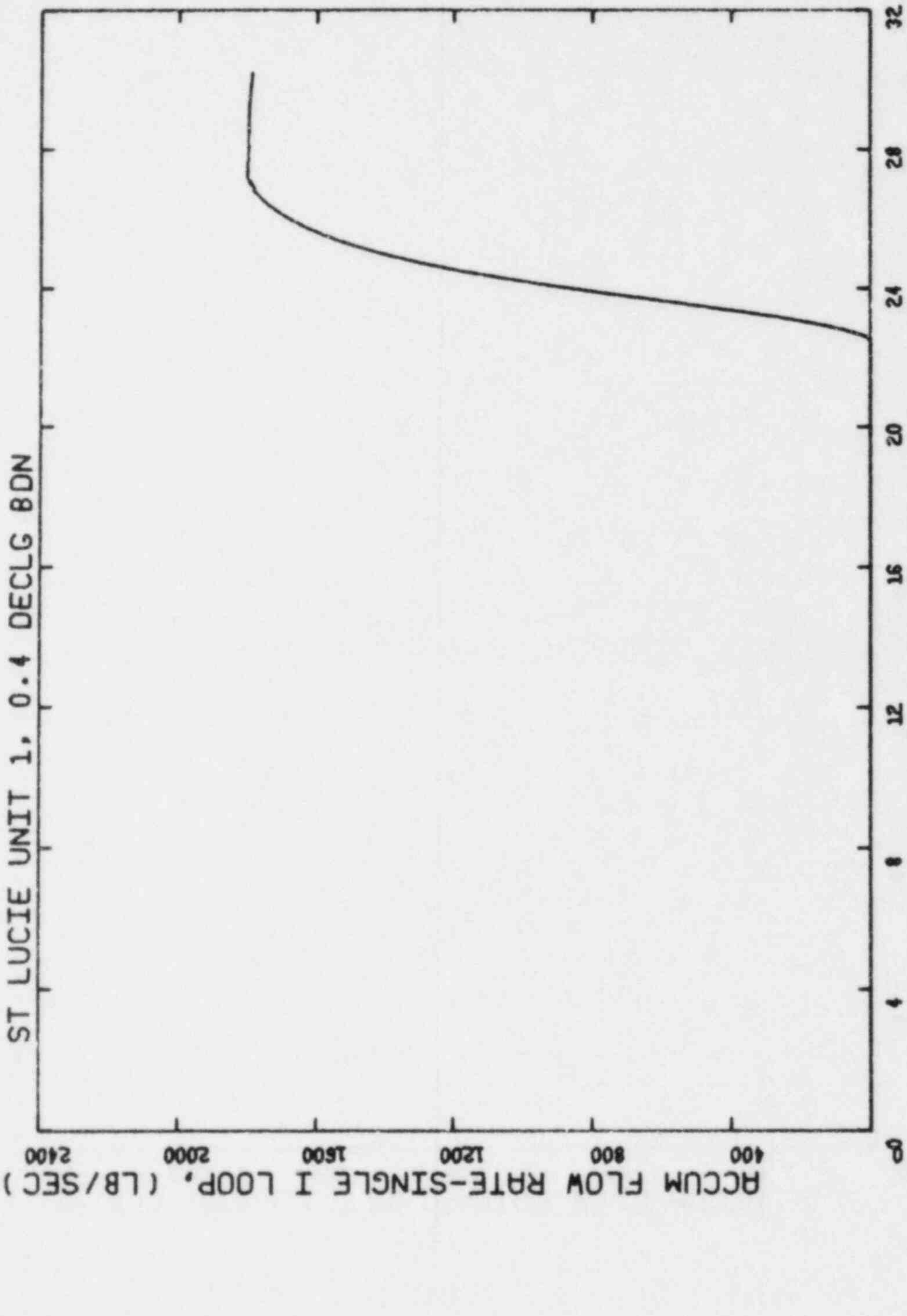

Figure 2.7 Double Intact Loop Accumulator Flow Rate, 0.4 DECLG Break

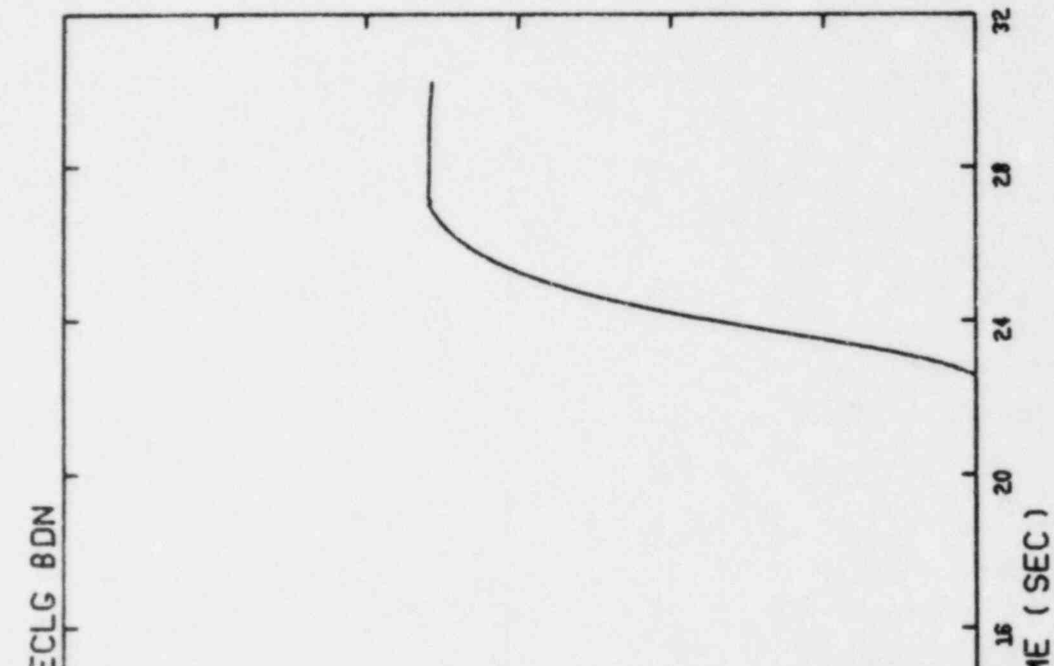

Figure 2.8 Single Intact Loop Accumulator Flow Rate, 0.4 DECLG Break

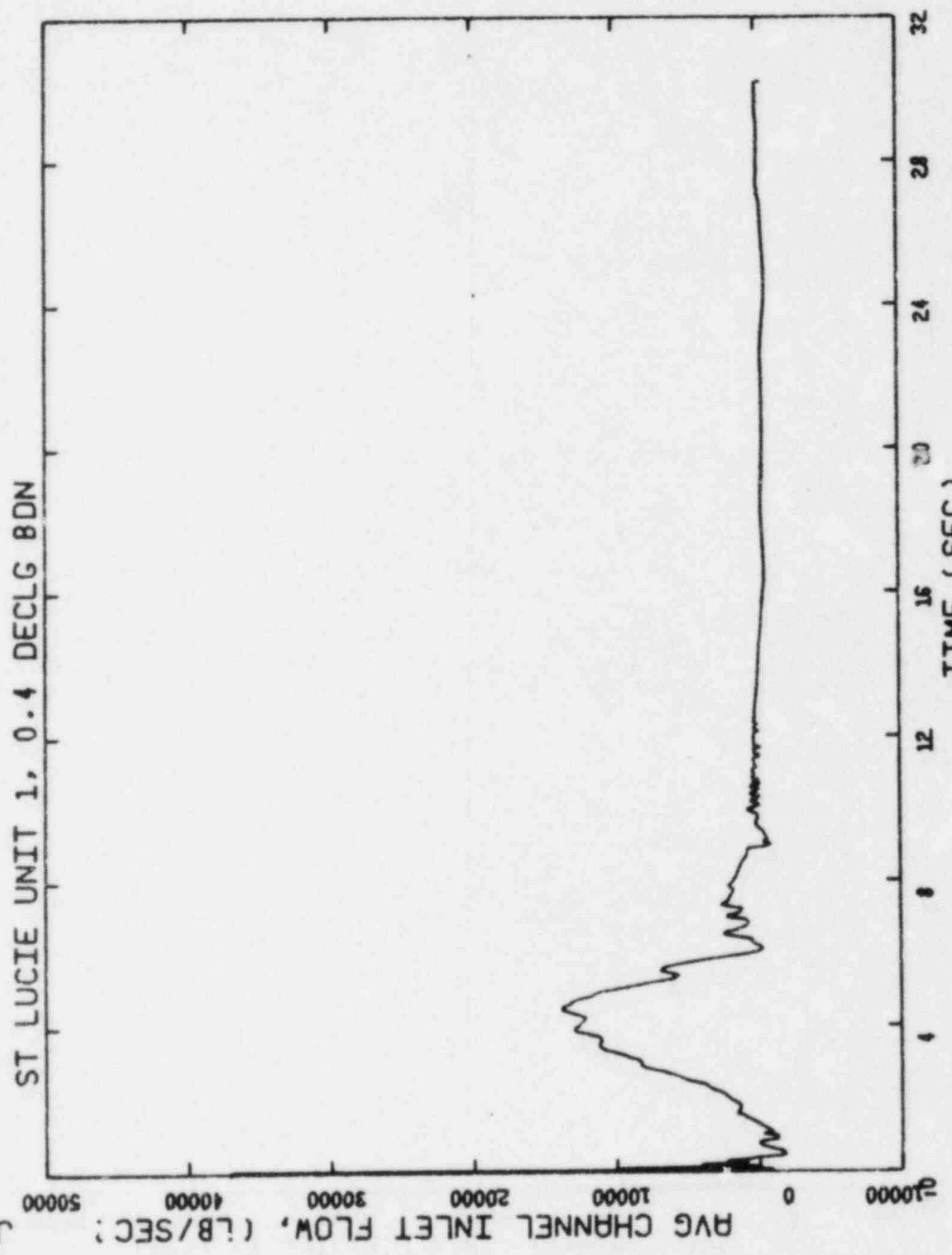

Figure 2.9 Blowdown Average Channel Inlet Flow Rate, 0.4 DECLG Break

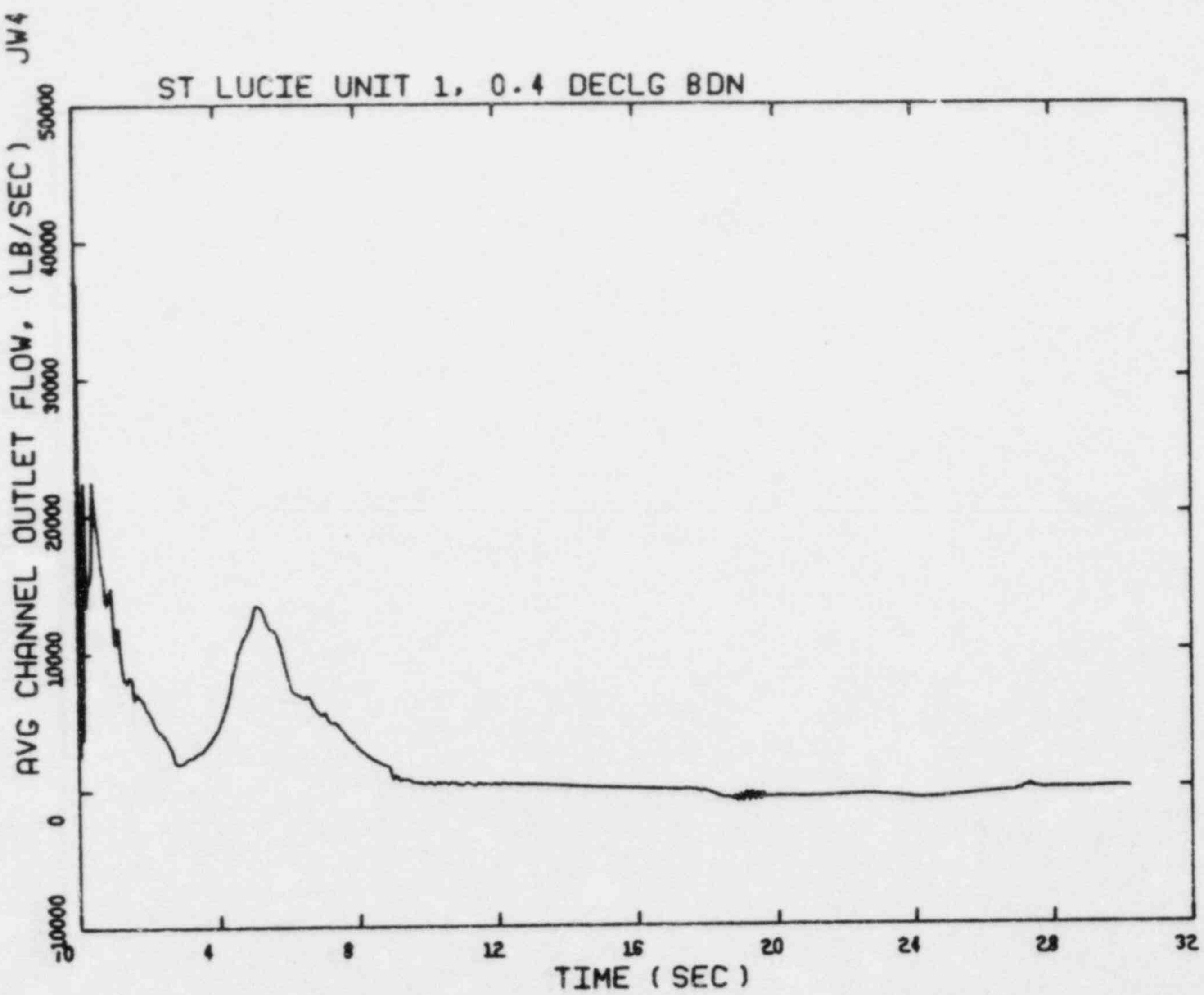

Figure 2.10 Blowdown Average Channel Outlet Flow Rate, 0.4 DECLG Break

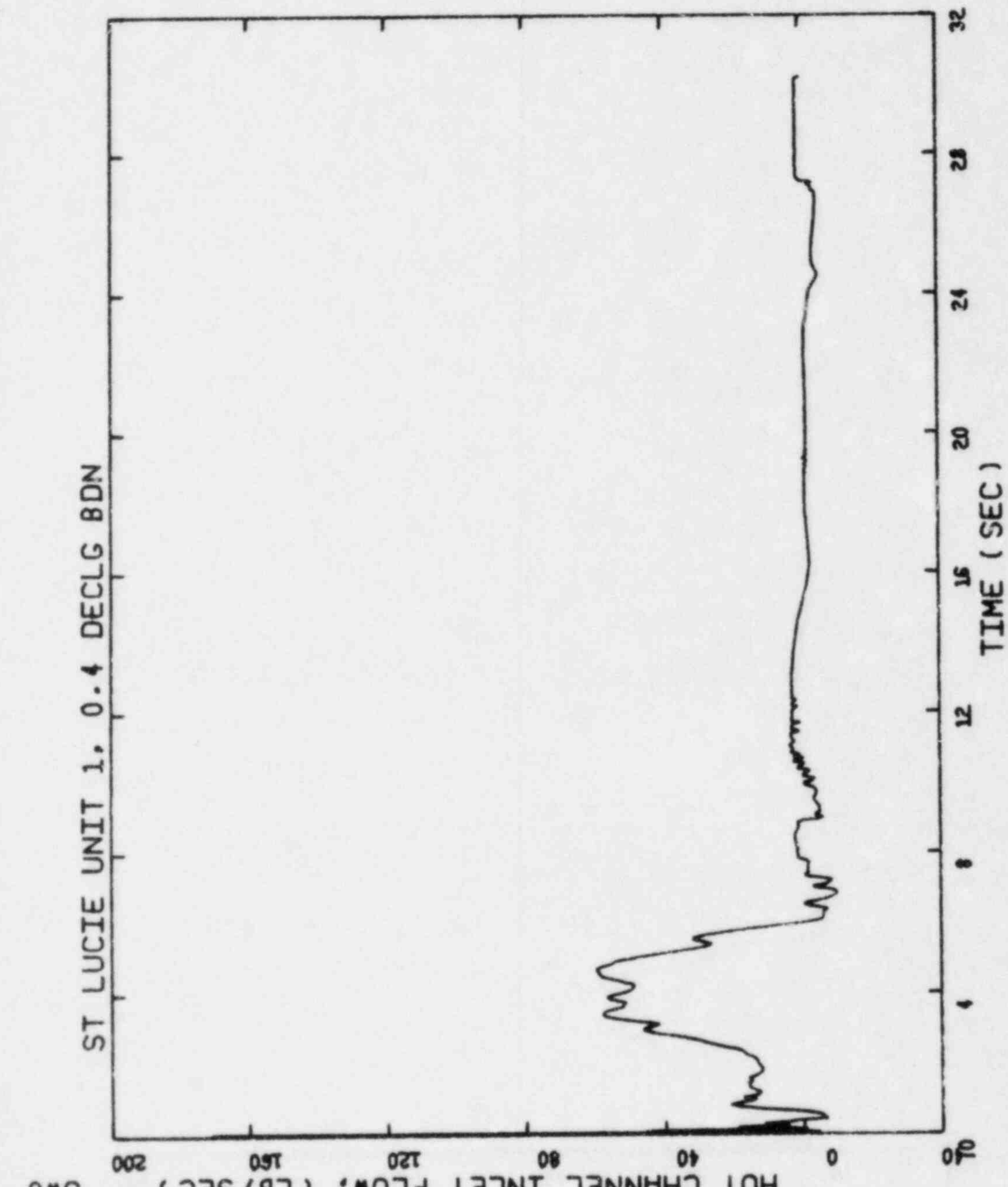

Figure 2.11 Blowdown Hot Channel Inlet Flow Rate, 0.4 DECLG Break

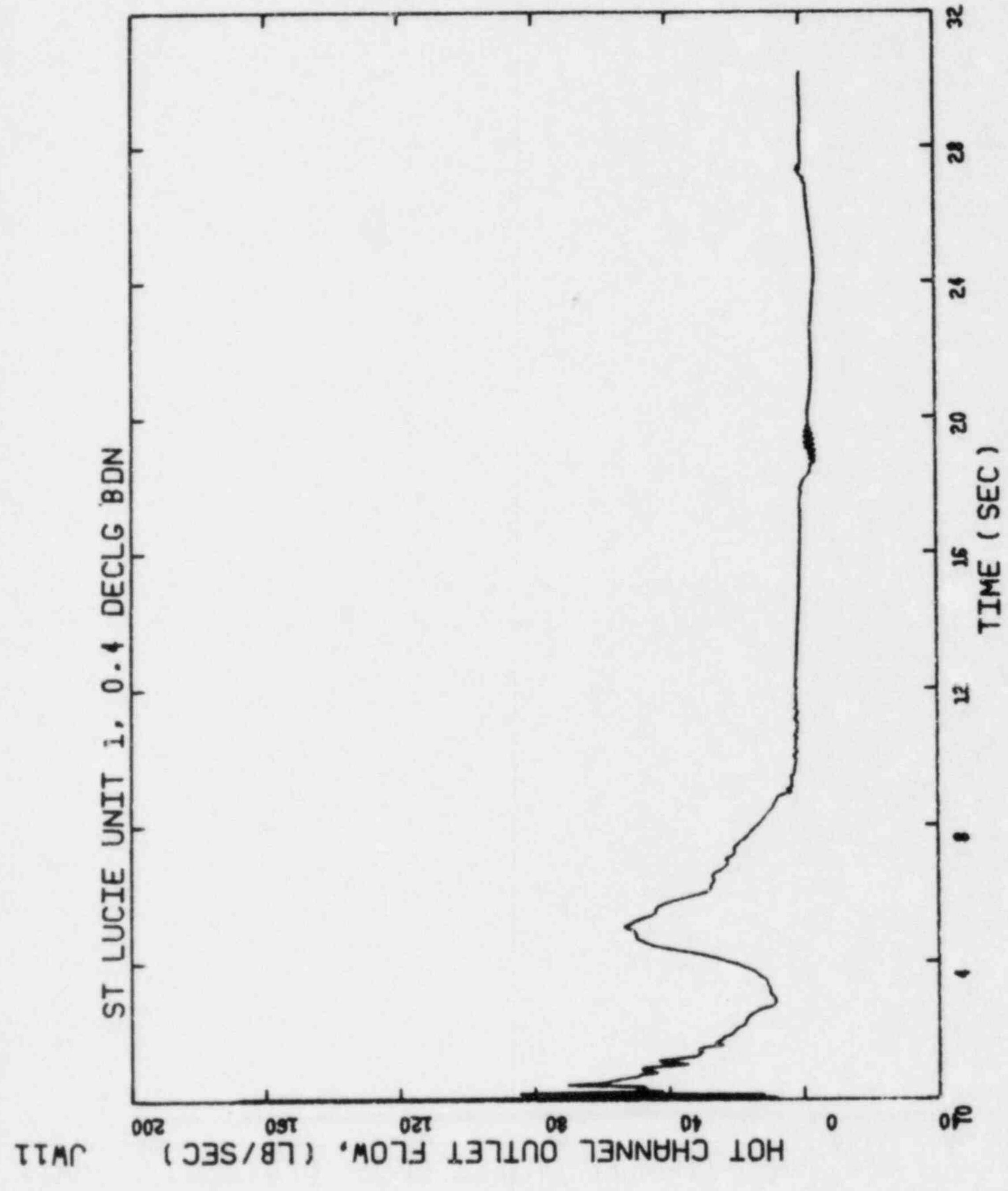

Figure 2.12 Blowdown Hot Channel Outlet Flow Rate, 0.4 DECLG Break

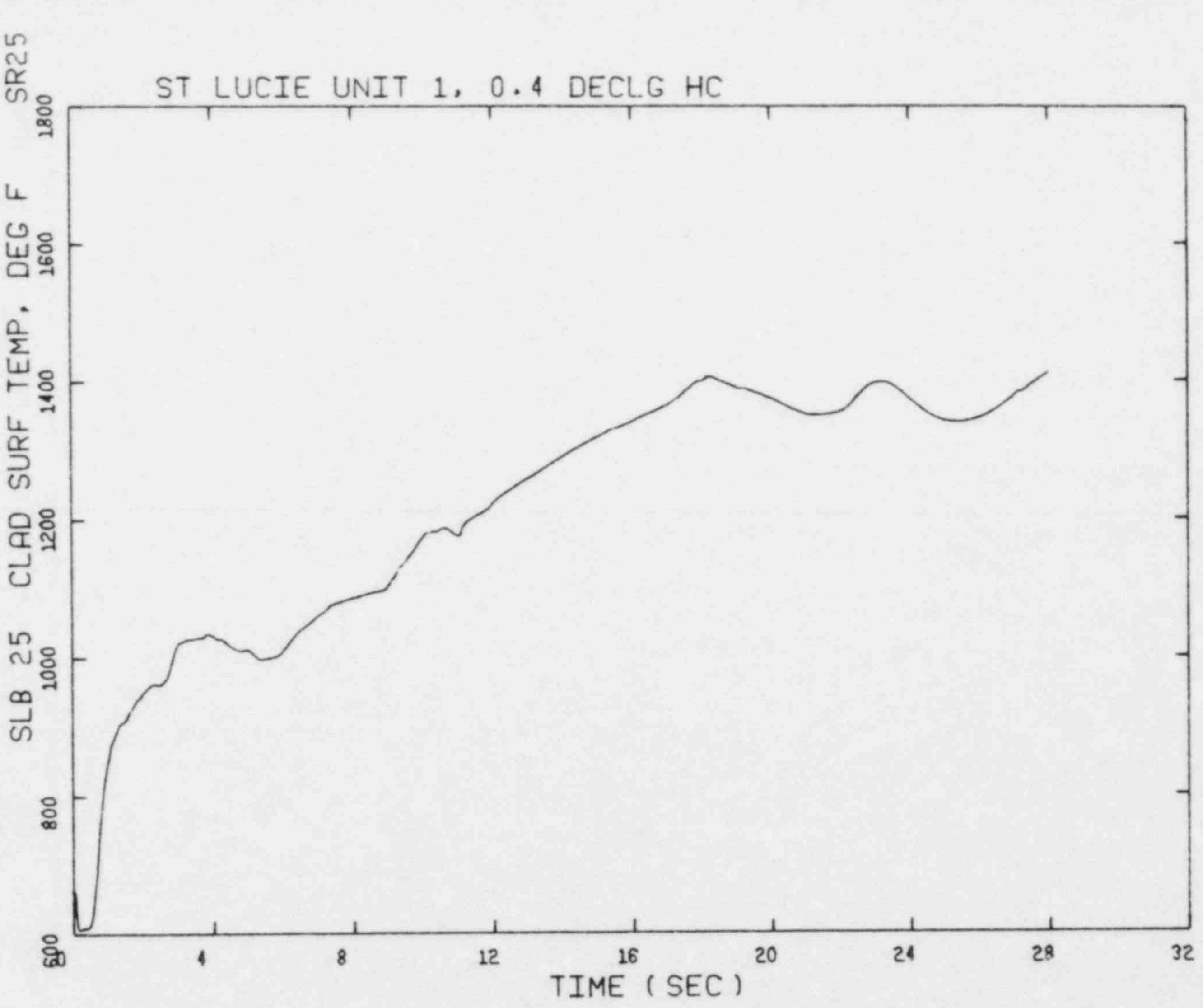

Figure 2.13 Blowdown Hot Rod Cladding surface Temperature, Node 25, 0.4 DECLG Break

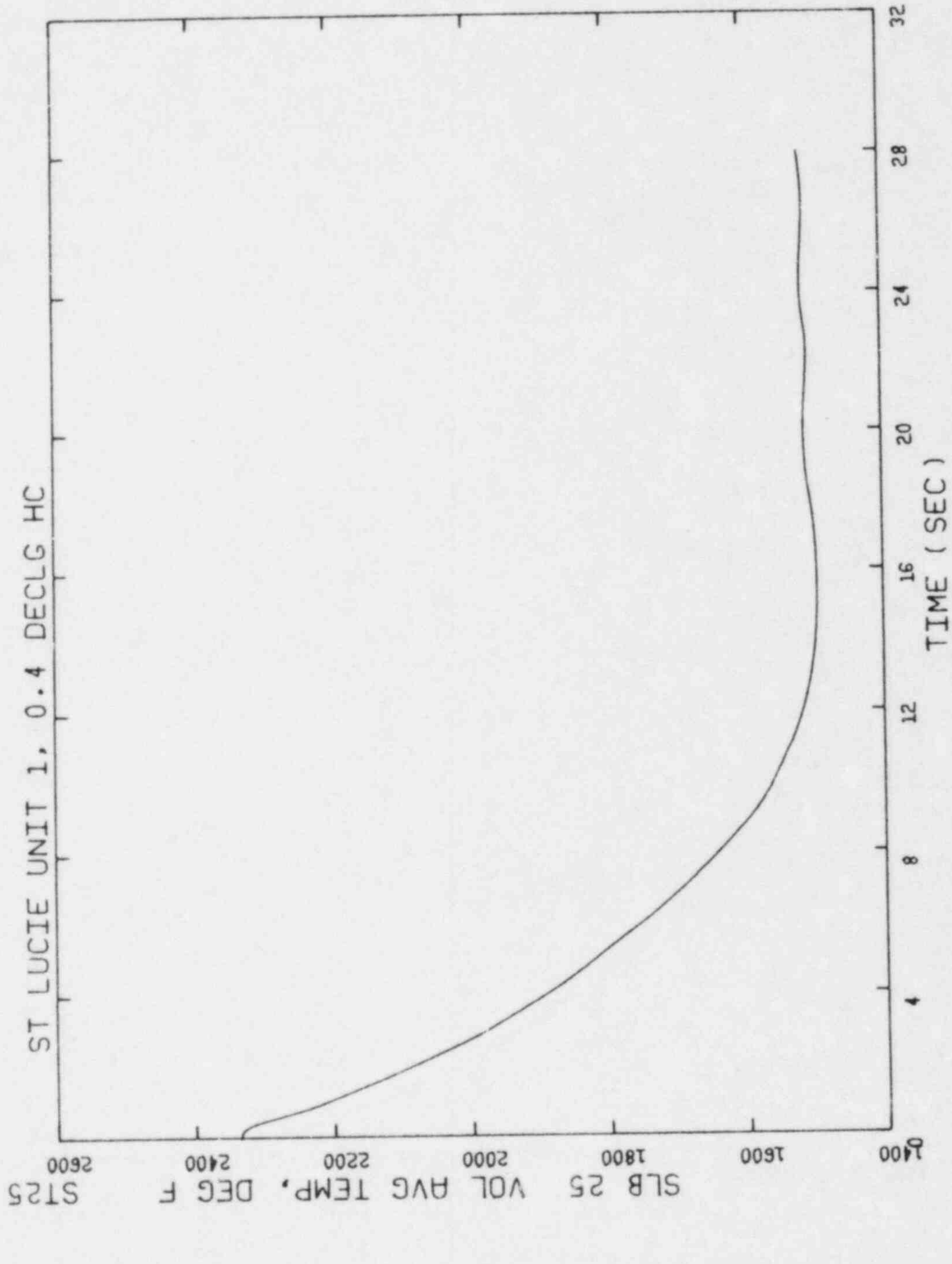

Figure 2.14 Blowdown Hot Rod Volumetric Average Temperature, Node 25, 0.4 DECLG Break

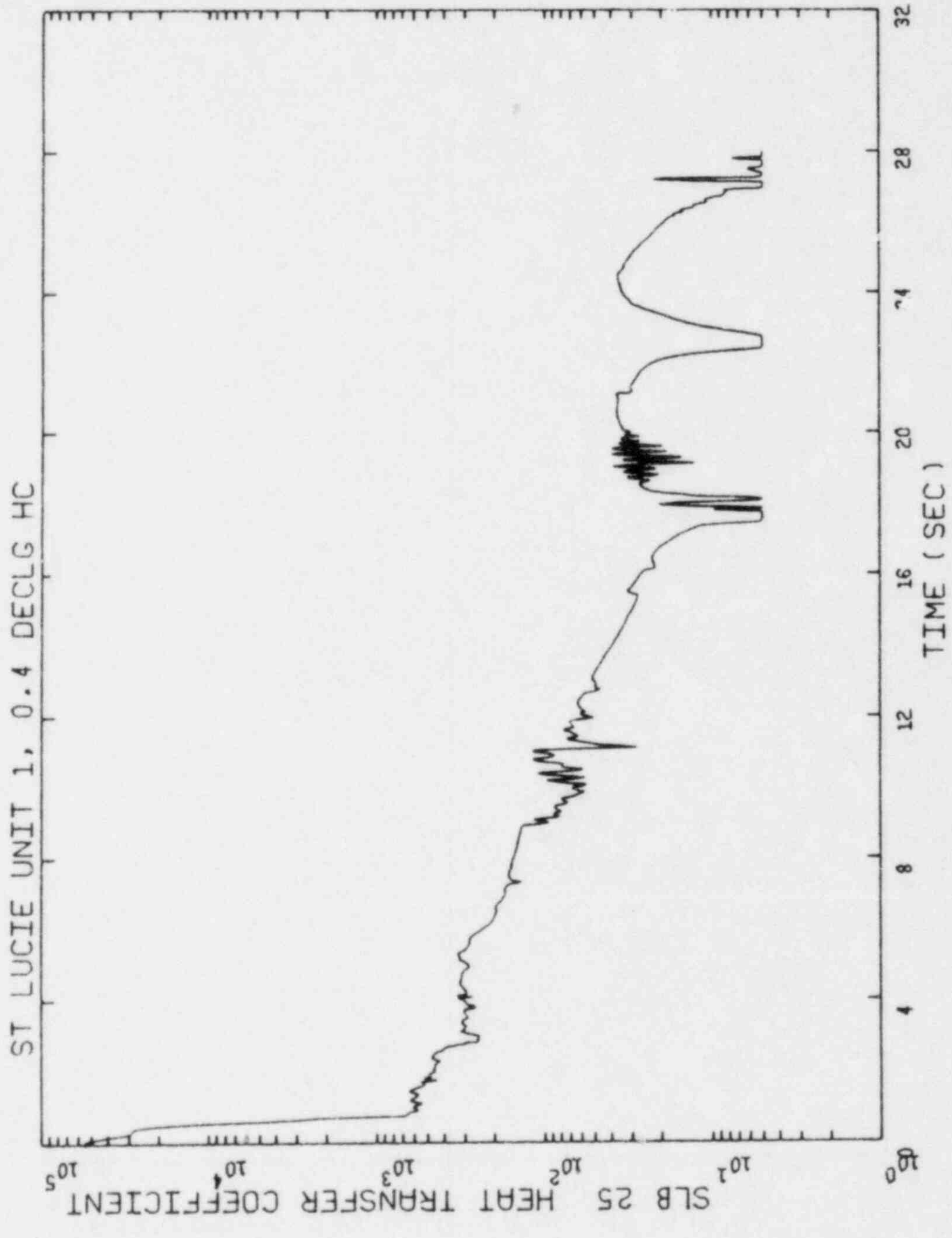

Figure 2.15 Hot Rod Blowdown Heat Transfer Coefficient, Node 25, 0.4 DECLG Break

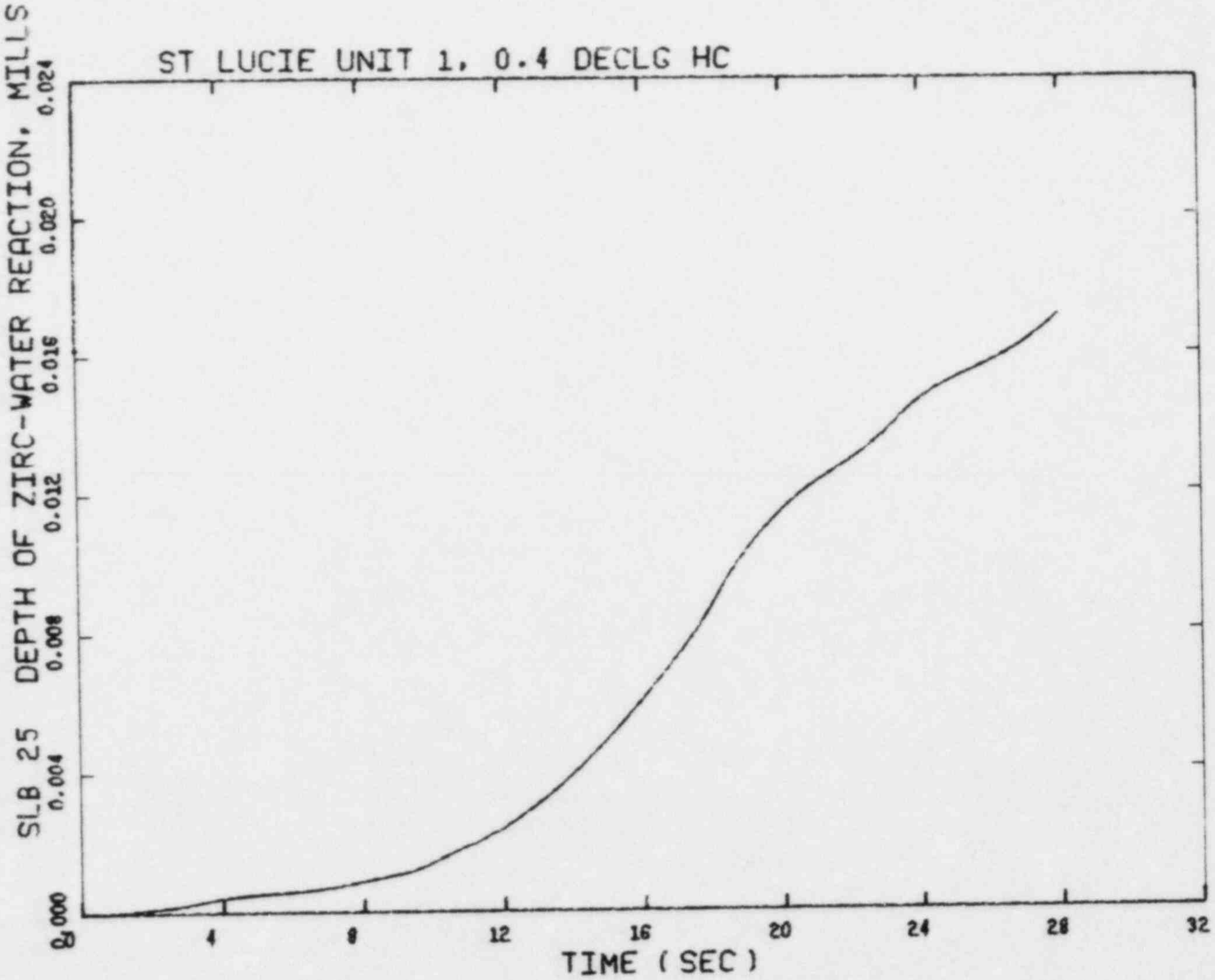

Figure 2.16 Hot Rod Blowdown Depth Of Zirconium - Water Reaction, Node 25, 0.4 DECLG Break

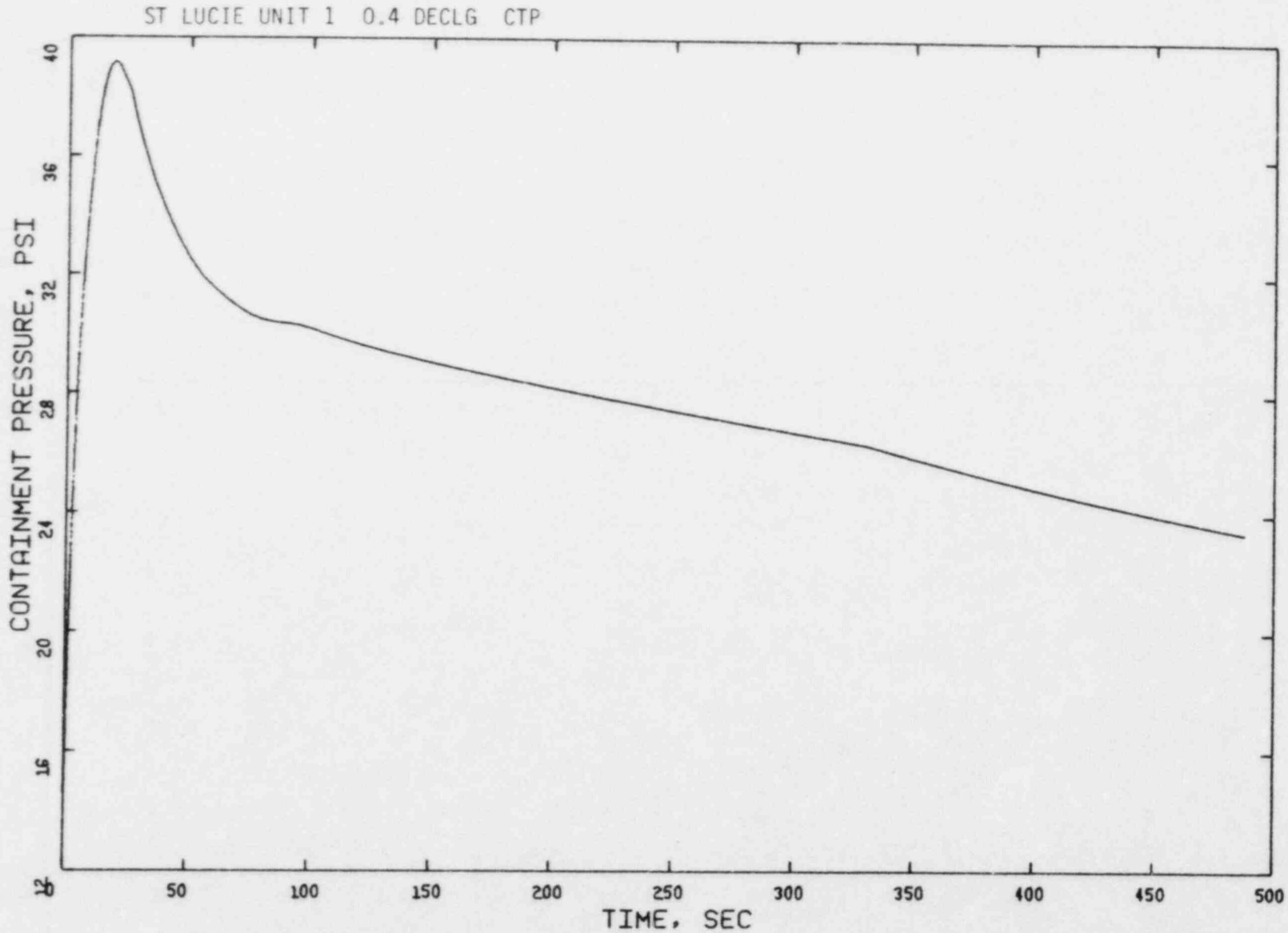

Figure 2.17 Containment Backpressure Versus Time, 0.4 DECLG Break

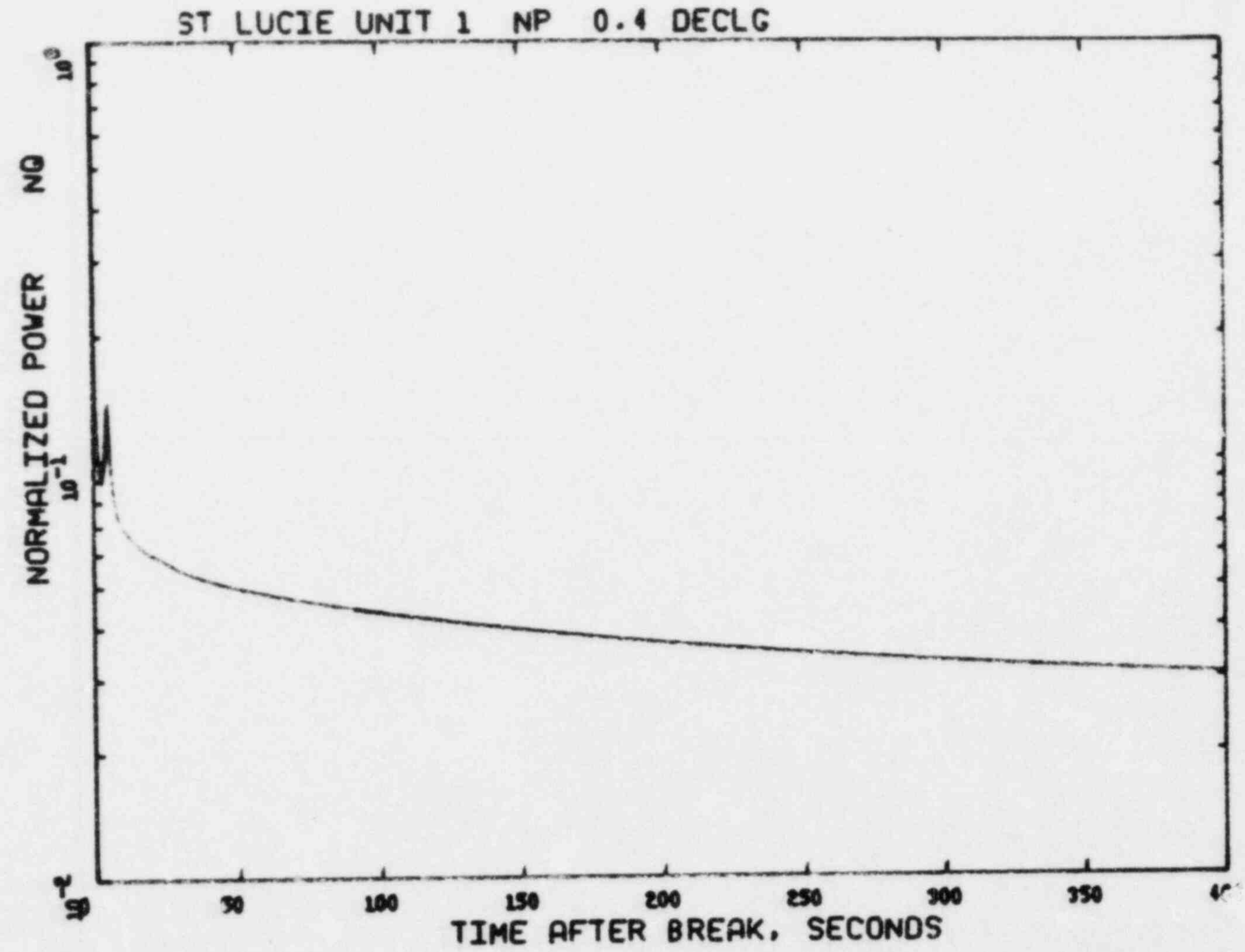

Figure 2.18 Normalized Power Versus Time, 0.4 DECLG Break

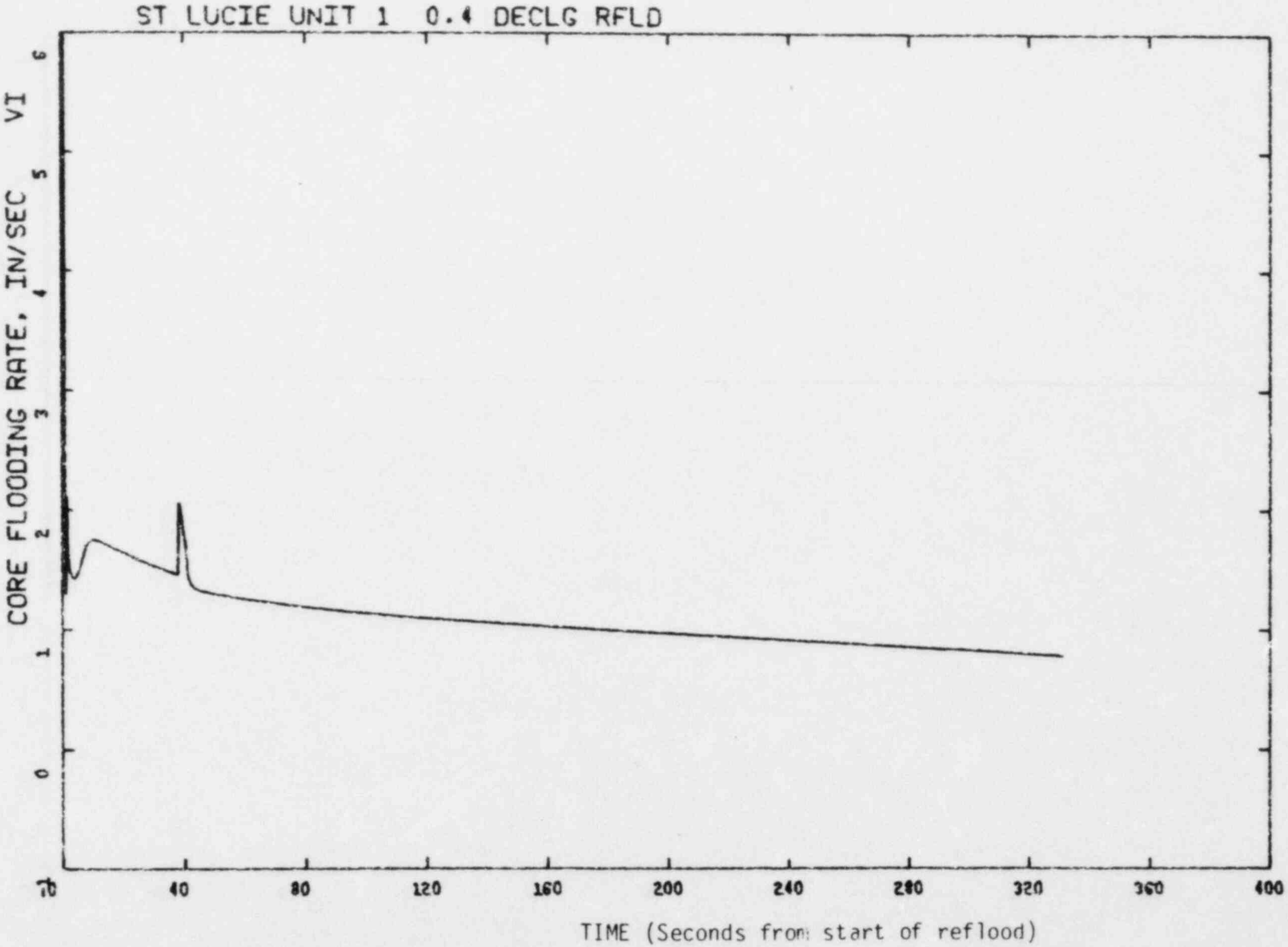

Figure 2.19 Reflood Core Flooding Rate, 0.4 DECLG Break

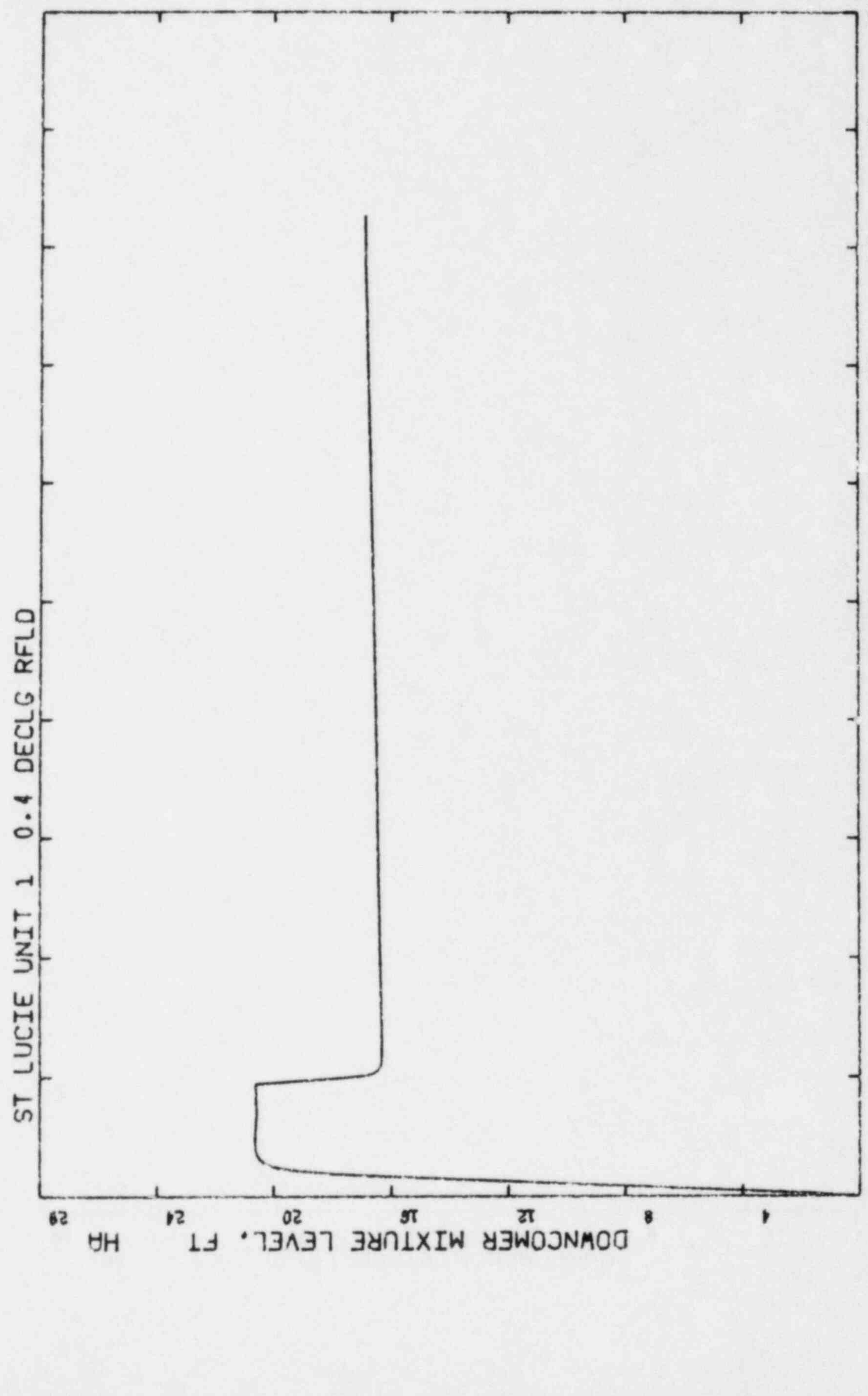

Figure 2.20 Reflood Downcomer Mixture Level, 0.4 DECLG Break

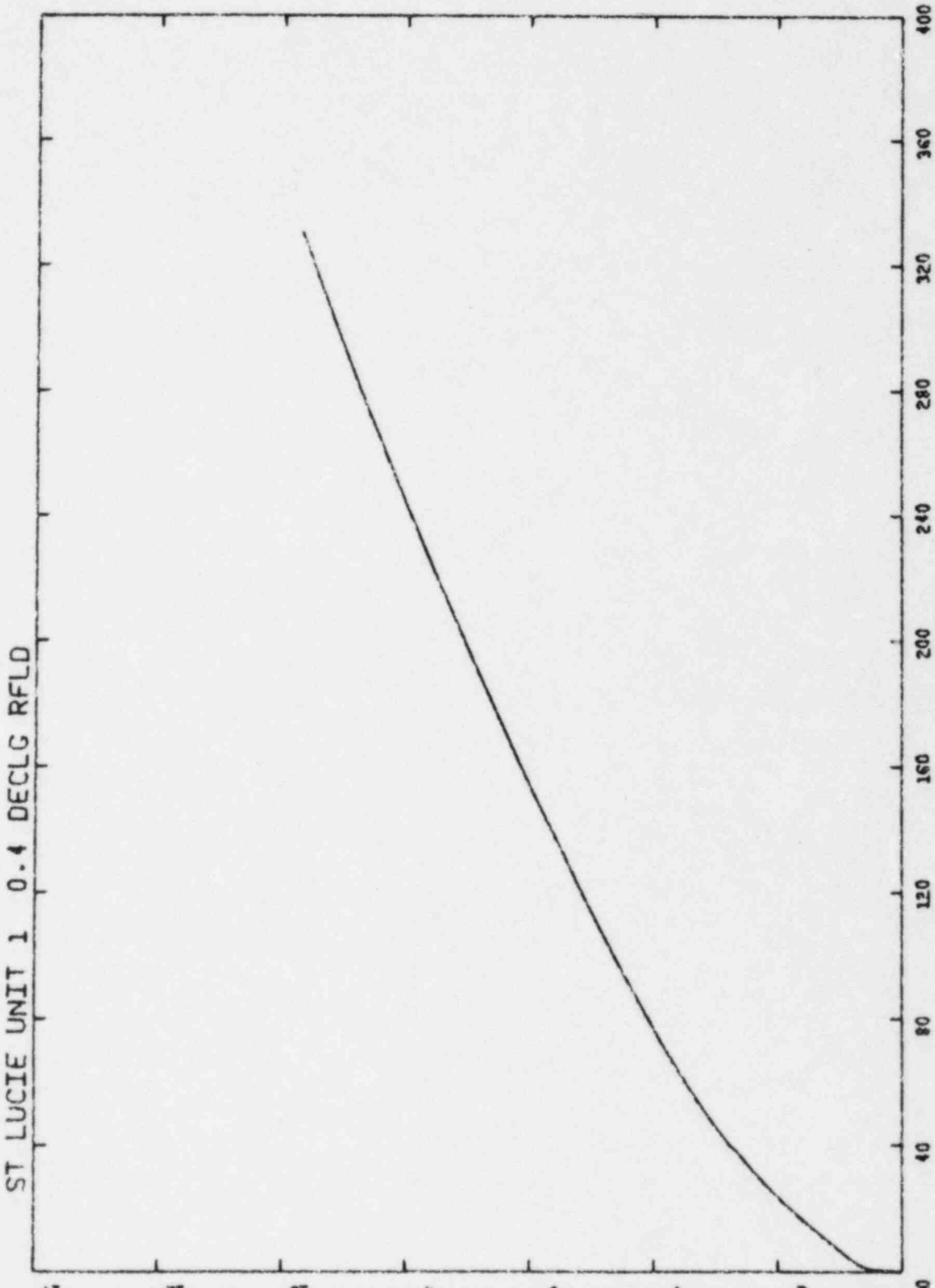

Figure 2.21 Reflood Core Mixture Level, 0.4 DECLG Break

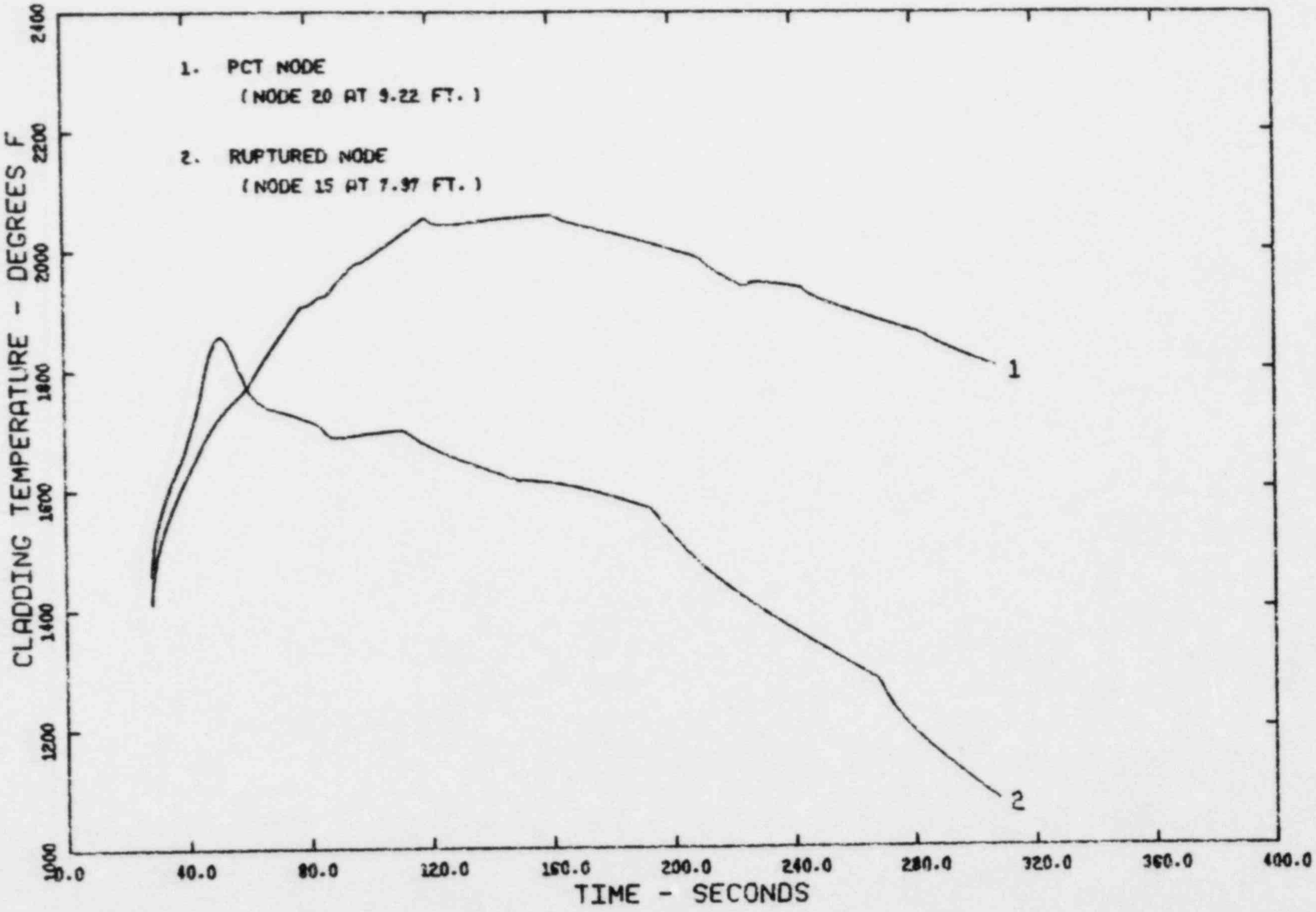

Figure 2.22 TOODEE2 Calculated Cladding Surface Temperature, 0.4 DEGLG Break

3.0 CONCLUSIONS

For breaks up to and including the double-ended severance of a reactor coolant pipe, the St. Lucie Unit 1 Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10 CFR 50.46 with the Cycle 6 core, with the results described herein and for ENC reload fuel. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200°F limit.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.
3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The system long term cooling capabilities provided for previous cores remain applicable for ENC fuel.

These Acceptance Criteria are satisfied if the St. Lucie Unit 1 reactor is operated at 2700 MWt within the maximum LHGR of 15.00 kw/ft.

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ST. LUCIE UNIT 1 LOCA ANALYSIS USING

THE EXEM/PWR ECCS MODEL

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