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MILLSTONE UNIT 2 LARGE BREAK LOCA/ECCS ANALYSIS

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MILLSTONE UNIT 2 LARGE BREAK LOCA/ECCS ANALYSIS

Prepared by

R. C. Gottula

R. C. Gottula, Team Leader
PWR Safety Analysis
Licensing & Safety Engineering
Fuel Engineering & Technical Services

Contributors:

T. Guidotti (Numerical Applications)
R. D. Hentzen (EI International)

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1.0 INTRODUCTION

This document presents the results of a large break loss-of-coolant accident (LOCA) analysis for the Millstone Unit 2 reactor. The analysis was performed to support operation with a mixed core or a core containing only ANF fuel. Break spectrum calculations were performed to determine the limiting break size. The break spectrum calculations included 0.4, 0.6, 0.8, and 1.0 double-ended-cold-leg-guillotine (DECLG) break sizes and 0.4, 0.8, and 1.0 double-ended-cold-leg-split (DECLS) break sizes. Calculations were also performed which considered axial power shapes, exposure, and full power operation with a reduced primary coolant temperature of up to 12°F.

2.0 SUMMARY OF RESULTS

The results of the analysis indicated the limiting break size was the 0.6 DECLG break. The Peak Cladding Temperature (PCT) for the limiting case was calculated to be 2163°F. The PCT for a 12°F reduction in primary coolant temperature was calculated to be 2176°F.

The analysis supports full power operation at 2754 MWt (2700 MWt plus 2% uncertainty) with an average steam generator tube plugging of 23.5% with a maximum asymmetry of 5.9%. The analysis supports assembly average exposures of up to 52,500 MWd/MTU. The analysis also supports operation at full power, with a primary coolant T_{ave} reduction of up to 12°F. The analysis demonstrates that the 10 CFR 50.46(b) criteria are satisfied for the Millstone Unit 2 reactor with an axial and exposure independent LHR of 15.1 kW/ft as shown in Figure 2.1.

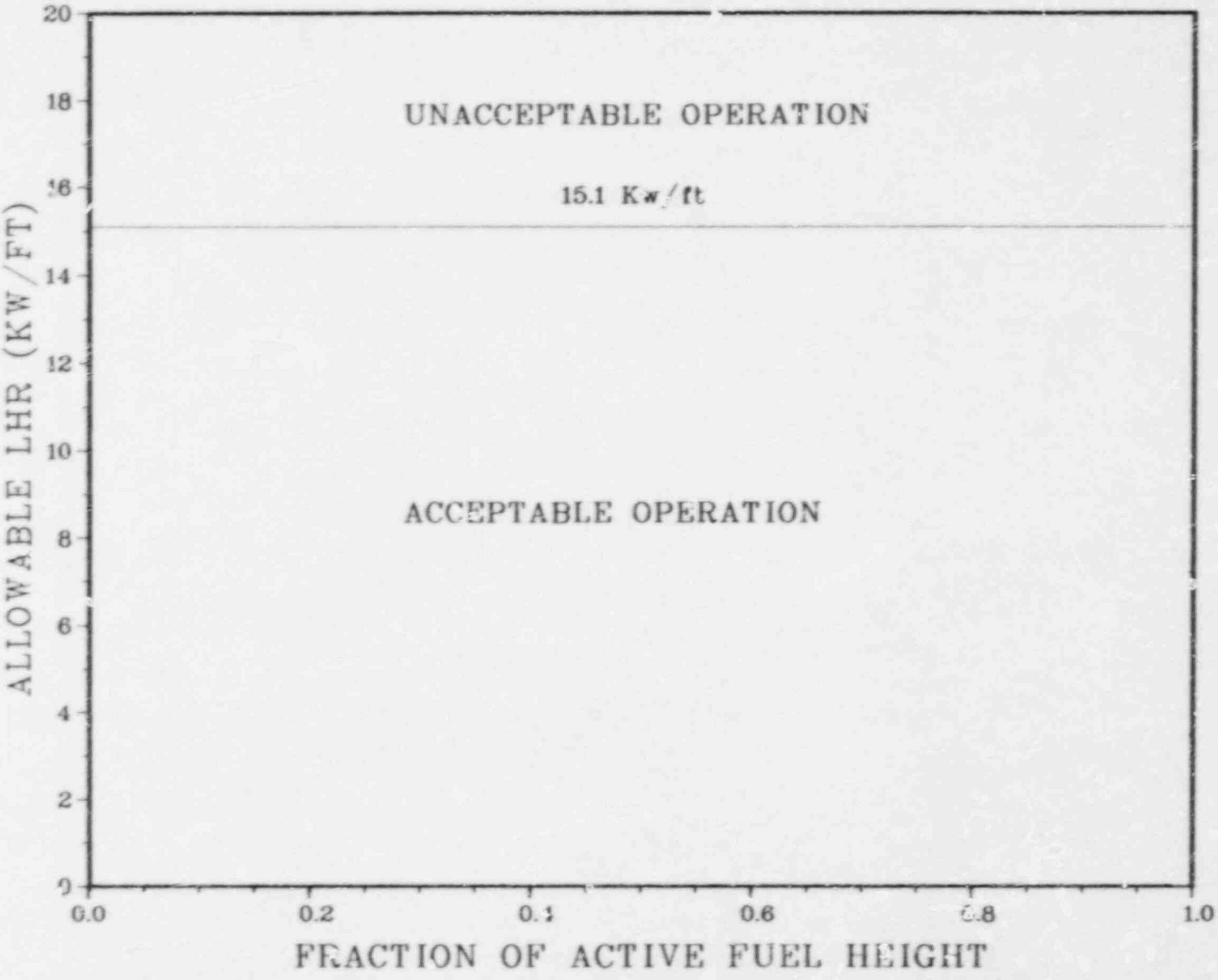

Figure 2.1 Allowable LHR as a Function of Axial Location

3.0 ANALYSIS

The purpose of the LBLOCA analysis is to demonstrate that the criteria stated in 10 CFR 50.46(b) are met. The criteria are:

1. The calculated peak fuel element cladding temperature does not exceed the 2200°F limit.

2. The amount of fuel element cladding which reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the core.

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 limit of 17% is not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Section 3.1 of this report provides a description of the postulated large break loss-of-coolant transient. Section 3.2 describes the analytical models used in the analysis. Section 3.3 provides a description of the Millstone Unit 2 plant and a summary of the system parameters used in the LOCA analysis. Section 3.4 provides a summary of the results of the break spectrum calculations. Section 3.5 summarizes the results of the axial power shape study. Section 3.6 summarizes the results of the exposure analysis. Section 3.7 summarizes the results of an analysis to support full power operation with a reduced primary coolant temperature of 12°F.

3.1 Description of LBLOCA Transient

A loss-of-coolant accident (LOCA) is defined as the rupture of the Reactor Coolant System primary piping up to and including a double-ended guillotine break. The limiting break occurs on the pump discharge side of a cold leg pipe. Loss-of-offsite power is assumed to occur co-incident with the LOCA. Primary coolant pump coastdown occurs co-incident with the loss-of-offsite power. Following the break, depressurization of the reactor coolant system, including the pressurizer, occurs. A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. Reactor trip and scram are conservatively neglected in the LOCA analysis. Early in the blowdown, the reactor core experiences flow reversal and stagnation which causes the fuel rods to pass through critical heat flux (CHF). Following CHF, the fuel rods dissipate heat through the transition and film boiling modes of heat transfer. Rewet is precluded during blowdown by Appendix K of 10 CFR 50.

A Safety Injection System (SIS) signal is actuated when the appropriate setpoint (high containment pressure) is reached. Due to loss-of-offsite power, a time delay for startup of diesel generators and SIS pumps is assumed. Once the time delay criteria is met and the system pressure falls below the shutoff head of the High Pressure Safety Injection (HPSI) pumps and Low Pressure Safety Injection (LPSI) pumps, SIS flow is injected into the cold legs. Single failure criteria is met by assuming that one diesel fails. This results in the loss of one HPSI pump and one LPSI pump. When the system pressure falls below the Safety Injection Tank (SIT) pressure, flow from the Safety Injection Tanks is injected into the cold legs. Flow from the Emergency Core Cooling System (ECCS)is assumed to bypass the core and flow to the break until the end-of-bypass (EOBY) is predicted to occur (sustained downflow in the downcomer). Following EOBY, ECCS flow fills the downcomer and lower plenum until the liquid level reaches the bottom of the core (beginning-of-core-recovery or BOCREC time). During the refill period, heat is transferred from the fuel rods by radiation heat transfer.

The reflood period begins at BOCREC time. ECCS fluid fills the downcomer and provides the driving head to move coolant through the core. As the mixture level moves up the core, steam is generated. Steam binding occurs as the steam flows through the intact and broken loop steam generators and pumps. The pumps are assumed to have a locked rotor (per Appendix K of 10 CFR 50) which tends to reduce the reflood rate. The fuel rods are eventually cooled and quenched by radiation and convective heat transfer as the quench front moves up the core. The reflood heat transfer rate is predicted through experimentally determined heat transfer and carry-over rate fraction correlations.

3.2 Description of Analytical Models

The ANF EXEM/PWR evaluation model[1] was used to perform the analysis. This evaluation model consists of the following computer codes:

(1) RODEX2[2] for computation of initial fuel stored energy, fission gas release, and gap conductance;

(2) RELAP4-EM for the system and hot channel blowdown calculations;

(3) CONTEMPT/LT-22 as modified in accordance with NRC Branch Technical Position CSB 6-1 for computation of containment back pressure;

(4) REFLEX for computation of system reflood; and

(5) TOODEE2 for the calculation of fuel rod heatup during the refill and reflood portions of the LOCA transient.

The quench time, quench velocity, and carryover rate fraction (CRF) correlations in REFLEX, and the heat transfer correlations in TOODEE2 are based on ANF's Fuel Cooling Test Facility (FCTF) data.

The governing conservation equations for mass, energy, and momentum transfer are used along with appropriate correlations consistent with Appendix K of 10 CFR 50. The reactor core in RELAP4 is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback, and with actinide and decay heating as required by Appendix K. Appropriate conservatisms specified by Appendix K of 10 CFR 50 are incorporated in all of the EXEM/PWR models.

3.3 Plant Description and Summary of Analysis Parameters

The Millstone Unit 2 nuclear power plant is a Combustion Engineering (CE) designed pressurized water reactor which has two hot leg pipes, two U-tube steam generators, and four cold leg pipes with one recirculation pump in each cold leg. The plant utilizes a large dry containment. The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow paths or "junctions". The two cold legs connected to the intact loop steam generator were assumed to be symmetrical and were modeled as one intact cold leg with appropriately scaled input. The model considers four Safety Injection Tanks, a pressurizer, and two steam generators with both primary and secondary sides of the steam generators modeled. The HPSI and LPSI pumps were modeled as fill junctions at the SIT lines, with conservative flows given as a function of system back-pressure. The pump performance curves were characteristic of CE pumps. The reactor core was modeled radially with an average core and a hot assembly as parallel flow channels, each with three axial nodes. A steam generator tube plugging level of 23.5% was assumed with a maximum asymmetry of 5.9%. The break was conservatively assumed to have occurred in the most highly plugged loop since this results in more steam binding during reflood and a higher peak cladding temperature.

Values for system parameters used in the analysis are given in Table 3.1.

3.4 Break Spectrum Results

Calculations were performed for 0.4, 0.6, 0.8 and 1.0 DECLG break sizes and 0.4, 0.8, and 1.0 DECLS break sizes with an axial power shape peaked at a relative core height of 0.6 to determine the limiting break. The break spectrum calculations were performed at a peak LHR of 15.6 kW/ft. Beginning-of-Cycle (BOC) stored energy (where maximum densification occurs at a hot rod average burnup of about 2000 MWd/MTU) was conservatively used in all of the break spectrum calculations. Calculations were performed through the blowdown, refill, and reflood periods of the LOCA transient for all break sizes. The results of the break spectrum study are shown in Table 3.2. Calculated event times for the various break sizes are given in Tables 3.3 through 3.9. The results indicated the 0.6 DECLG break size to be the limiting break. The peak cladding temperature (PCT) for the 0.6 DECLG break with an axial power shape peaked at a relative core height of 0.6 was calculated to be 1892°F. Thus, a maximum LHR of 15.6 kW/ft is supported up to a relative core height of 0.6.

3.5 Axial Shape Study Results

Axial shape and exposure studies were performed at the limiting 0.6 DECLG break size. Conservative combinations of stored energy and axial power shapes were analyzed to bound the effects of axial power shape and exposures from BOC to EOC. In addition to the axial shape which was used in the break spectrum study, peaked at a relative core height of 0.6, bounding axial shapes representative of MOC and EOC were determined from projected axial shapes for equilibrium and xenon oscillation transient conditions. Two conservative combinations of stored energy and axial shape were analyzed at a peak LHR of 15.1 kW/ft. They are (1) a BOC stored energy (where maximum densification occurs) combined with a MOC shape peaked at a relative core height of 0.81, and (2) a MOC stored energy combined with an EOC shape peaked at a relative core height of 0.9. The results of these two calculations are shown in Table 3.10. Case 1 was determined to be the limiting case with a PCT of 2163°F. These calculations support a peak LHR of 15.1 kW/ft that is independent of axial position and exposure over the exposure range of BOC to EOC. Plots of parameters depicting the calculations for case 1 at the limiting 0.6 DECLG break size are shown in Figures 3.1 through 3.25.

3.6 Exposure Study Results

The calculations described in Section 3.5 support exposures out to EOC. An additional calculation was performed to support assembly average exposures up to 52,500 MWd/MTU. This one additional calculation was sufficient to support assembly average exposures up to 52,500 MWd/MTU for the following reasons. Fuel rod stored energy decreases with exposure (after peak stored energy occurs at maximum densification near BOC) out to exposures of about 40,000 MWd/MTU due to closure of the pellet-cladding gap. Beyond exposures of about 40,000 MWd/MTU, the stored energy increases again due to fission gas release to the gap, but is still significantly less than the stored energy at MOC. The rod internal pressure increases beyond exposures of about 20,000 MWd/MTU which tends to decrease the time of cladding rupture and also increase the pellet-cladding gap which retards heat release during blowdown. Thus, stored energy and rod internal pressure at an exposure of 52,500 MWd/MTU bound exposures between MOC and 52,500 MWd/MTU. The PCT calculated at an assembly exposure of 52,500 MWd/MTU was compared to the PCT for Case 2 described in Section 3.5 to verify that the PCT for exposures greater than EOC are less than PCTs calculated for exposures from BOC to EOC. The PCT at an exposure of 52,500 MWd/MTU was calculated to be 1839°F. Therefore, a peak LHR of 15.1 kW/ft is supported for assembly average exposures up to 52,500 MWd/MTU.

3.7 Reduced T_{ave} Operation Results

Calculations were performed to support full power operation with a reduced primary coolant temperature of up to 12°F. A 12°F reduction in primary temperature has a small effect on the blowdown hydraulics and a small adverse effect on containment pressure, reflood rate, and PCT. The lower initial coolant enthalpy results in a slightly lower containment pressure which results in a lower reflood rate and a higher PCT. Calculations were performed to estimate the reduced containment pressure and its effect on reflood rate and PCT. Calculations were performed for the limiting case (0.6 DECLG break, BOC stored energy with an MOC axial power shape) to determine the change in PCT due to the reduced primary coolant temperature. The PCT was calculated to be 2176°F. All 10 CFR 50.46(b) criteria are met. A reduction in primary coolant temperature greater than 12°F is acceptable as long as there is a corresponding decrease in power level.

Table 3.1 Millstone Unit 2 System Analysis Parameters

Primary Heat Output, MWt	2700*
Primary Coolant Flow Rate, lbm/hr	1.28×10^8 (340,000 gpm)
Primary Coolant System Volume, ft^3	10,510**
Operating Pressure, psia	2250
Inlet Coolant Temperature, °F	549
Reactor Vessel Volume, ft^3	4538
Pressurizer Total Volume, ft^3	1500
Pressurizer Liquid Total, ft^3	800
SIT Total Volume, Ft^3 (one of four)	2019
SIT Liquid Volume, ft^3	1150.5
SIT Pressure, psia	238.5
SIT Fluid Temperature, °F	106.8
Total Number of Tubes per Steam Generator	8519
Steam Generator Tube Plugging	29.4 - 17.6% split
Number of Tubes Plugged (29.4% SGTP) (Broken Loop)	2500
Number of Tubes Plugged (17.6% SGTP) (Double Intact Loop)	1500
Steam Generator Secondary Side Heat Transfer Area, 29.4% SGTP, ft^2	63,370
Steam Generator Secondary Side Heat Transfer Area, 17.6% SGTP, ft^2	73,898
Steam Generator Secondary Flow Rate, lbm/hr (48-52% power split)	5.75×10^6 (29.4% SGTP) 6.233×10^6 (17.6% SGTP)
Steam Generator Secondary Pressure (broken loop), psia	830.1
Steam Generator Secondary Pressure (intact loop), psia	850.4
Steam Generator Feedwater Temperature, °F	435

* Primary Heat Output used in RELAP4-EM Model - 1.02 x 2700 = 2754 MWt.

** Includes pressurizer total volume and 23.5% average SGTP.

Table 3.1 Millstone Point 2 System Analysis Parameters (Cont.)

Reactor Coolant Pump Rated Head, ft	271.8
Reactor Coolant Pump Head, ft (DIL)	230.38*
Reactor Coolant Pump Head, ft (SIL,BL)	233.0*
Reactor Coolant Pump Rated Torque, ft-lbf	31,560
Reactor Coolant Pump Rated Speed, rpm	892
Initial Reactor Coolant Pump Speed, rpm	874.2*
Reactor Coolant Pump Moment of Inertia, $lbm\text{-}ft^2$	100,000
Containment Volume, ft^3	1.938×10^6
Containment Temperature, °F	108.1
SIS Fluid Temperature, °F	72.8
HPSI Delay Time, sec	30.0
LPSI Delay Time, sec	50.0

* Values used in RELAP4 for initialization.

Table 3.2 Millstone Unit 2 Break Spectrum Analysis Results

	0.4 DECLG X/L=0.6	0.6 DECLG X/L=0.6	0.8 DECLG X/L=0.6	1.0 DECLG X/L=0.6	0.4 DECLS X/L=0.6	0.8 DECLS X/L=0.6	1.0 DECLS X/L=0.6
Peak LHR (kW/ft)	15.6	15.6	15.6	15.6	15.6	15.6	15.6
Hot Rod Burst							
- Time (sec)	49.31	40.75	39.36	39.53	57.44	42.55	40.32
- Elevation (ft)	6.97	7.72	7.72	6.97	7.72	7.72	7.72
- Channel Blockage Fraction	0.25	0.3	0.26	0.25	0.3	0.3	0.28
Peak Cladding Temperature							
- Temperature (°F)	1815.8	1892.3	1881.7	1869.6	1685.2	1761.2	1787.5
- Time (sec)	72.01	5[illegible].15	56.36	66.03	71.09	65.55	64.42
- Elevation (ft)	7.72	7.72	7.72	7.72	7.97	7.97	7.97
Metal Water Reaction							
- Local Maximum (%)	1.34	2.53	2.44	1.71	0.90	1.30	1.46
- Elevation of Local Max. (ft)	6.97	7.72	7.72	6.97	7.72	7.72	7.72
- Hot Pin Total (%)	0.44	0.52	0.52	0.51	0.26	0.35	0.38
- Core Maximum (%)	<1.0*	<1.0*	<1.0*	<1.0*	<1.0*	<1.0*	<1.0*

* At 200 seconds.

Table 3.3 Calculated Event Times for 0.4 DECLG Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.90
Pressurizer Empties	9.19
SIT Injection Begins, Broken Loop	20.2
SIT Injection Begins, Single Intact Loop	21.6
SIT Injection Begins, Double Intact Loop	21.6
End-of-Bypass (EOBY)	26.11
Start of Reflood	39.7
SIT empties, Broken Loop	57.8
SIT empties, Double Intact Loop	57.95
SIT empties, Single Intact Loop	58.75
Peak Cladding Temperature is Reached (X/L=0.6)	72.01

Table 3.4 Calculated Event Times for 0.6 DECLG Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.73
Pressurizer Empties	8.6
SIT Injection Begins, Broken Loop	14.7
SIT Injection Begins, Single Intact Loop	16.75
SIT Injection Begins, Double Intact Loop	16.75
End-of-Bypass (EOBY)	21.05
Start of Reflood	34.73
SIT empties, Broken Loop	52.0
SIT empties, Double Intact Loop	53.0
SIT empties, Single Intact Loop	53.4
Peak Cladding Temperature is Reached (X/L=0.6)	58.15

Table 3.5 Calculated Event Times for 0.8 DECLG Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	[illegible].63
Pressurizer Empties	8.6
SIT Injection Begins, Broken Loop	11.25
SIT Injection Begins, Single Intact Loop	14.95
SIT Injection Begins, Double Intact Loop	14.95
End-of-Bypass (EOBY)	18.36
Start of Reflood	32.54
SIT empties, Broken Loop	49.45
SIT empties, Double Intact Loop	51.35
SIT empties, Single Intact Loop	51.6
Peak Cladding Temperature is Reached (X/L=0.6)	56.36

Table 3.6 Calculated Event Times for 1.0 DECLG Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.58
Pressurizer Empties	8.6
SIT Injection Begins, Broken Loop	8.85
SIT Injection Begins, Single Intact Loop	14.5
SIT Injection Begins, Double Intact Loop	14.5
End-of-Bypass (EOBY)	18.33
Start of Reflood	32.04
SIT empties, Broken Loop	47.8
SIT empties, Double Intact Loop	50.85
SIT empties, Single Intact Loop	51.1
Peak Cladding Temperature is Reached (X/L=0.6)	66.03

Table 3.7 Calculated Event Times for 0.4 DECLS Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.77
Pressurizer Empties	9.1
SIT Injection Begins, Broken Loop	18.45
SIT Injection Begins, Single Intact Loop	18.6
SIT Injection Begins, Double Intact Loop	18.6
End-of-Bypass (EOBY)	22.29
Start of Reflood	36.[illegible]
SIT empties, Double Intact Loop	[illegible]
SIT empties, Single Intact Loop	[illegible]
SIT empties, Broken Loop	[illegible]
Peak Cladding Temperature is Reached (X/L=0.6)	71.09

Table 3.8 Calculated Event Times for 0.8 DECLS Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.51
Pressurizer Empties	8.6
SIT Injection Begins, Broken Loop	11.25
SIT Injection Begins, Single Intact Loop	12.15
SIT Injection Begins, Double Intact Loop	12.15
End-of-Bypass (EOBY)	15.05
Start of Reflood	28.77
SIT empties, Broken Loop	49.05
SIT empties, Double Intact Loop	49.15
SIT empties, Single Intact Loop	49.35
Peak Cladding Temperature is Reached (X/L=0.6)	65.55

Table 3.9 Calculated Event Times for 1.0 DECLS Break

Event	Time (Sec)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.5
Pressurizer Empties	8.6
SIT Injection Begins, Broken Loop	9.8
SIT Injection Begins, Single Intact Loop	11.75
SIT Injection Begins, Double Intact Loop	11.75
End-of-Bypass (EOBY)	14.72
Start of Reflood	28.46
SIT empties, Broken Loop	47.95
SIT empties, Double Intact Loop	48.85
SIT empties, Single Intact Loop	49.05
Peak Cladding Temperature is Reached (X/L=0.6)	64.42

Table 3.10 Summary of Results for 0.6 DECLG Limiting Break Size

	BOC Stored Energy BOC Axial Shape X/L = 0.6	BOC Stored Energy MOC Axial Shape X/L = 0.81	MOC Stored Energy EOC Axial Shape X/L = 0.9	60,000 MWd/MTU Hot Rod Average Burnup X/L = 0.9
Peak LHR (kW/ft)	15.6	15.1	15.1	15.1
Hot Rod Burst				
- Time (sec)	40.75	35.55	37.15	46.95
- Elevation (ft)	7.72	9.22	10.22	9.97
- Channel Blockage Fraction	0.3	0.28	0.33	0.4
Peak Cladding Temperature				
- Temperature (°F)	1892.3	2163.3	2108.0	1839.2
- Time (sec)	58.15	61.75	67.05	291.05
- Elevation (ft)	7.72	9.22	10.22	10.72
Metal-Water Reaction				
- Local Maximum (%)	2.53	5.84	5.73	2.18
- Elevation of Local Max. (ft)	7.72	9.22	10.22	10.72
- Hot Pin Total (%)	0.52	0.57	0.82	0.24
- Core Maximum (%)	<1.0*	<1.0*	<1.0*	<1.0**

* At 200 seconds.

** At 350 seconds.

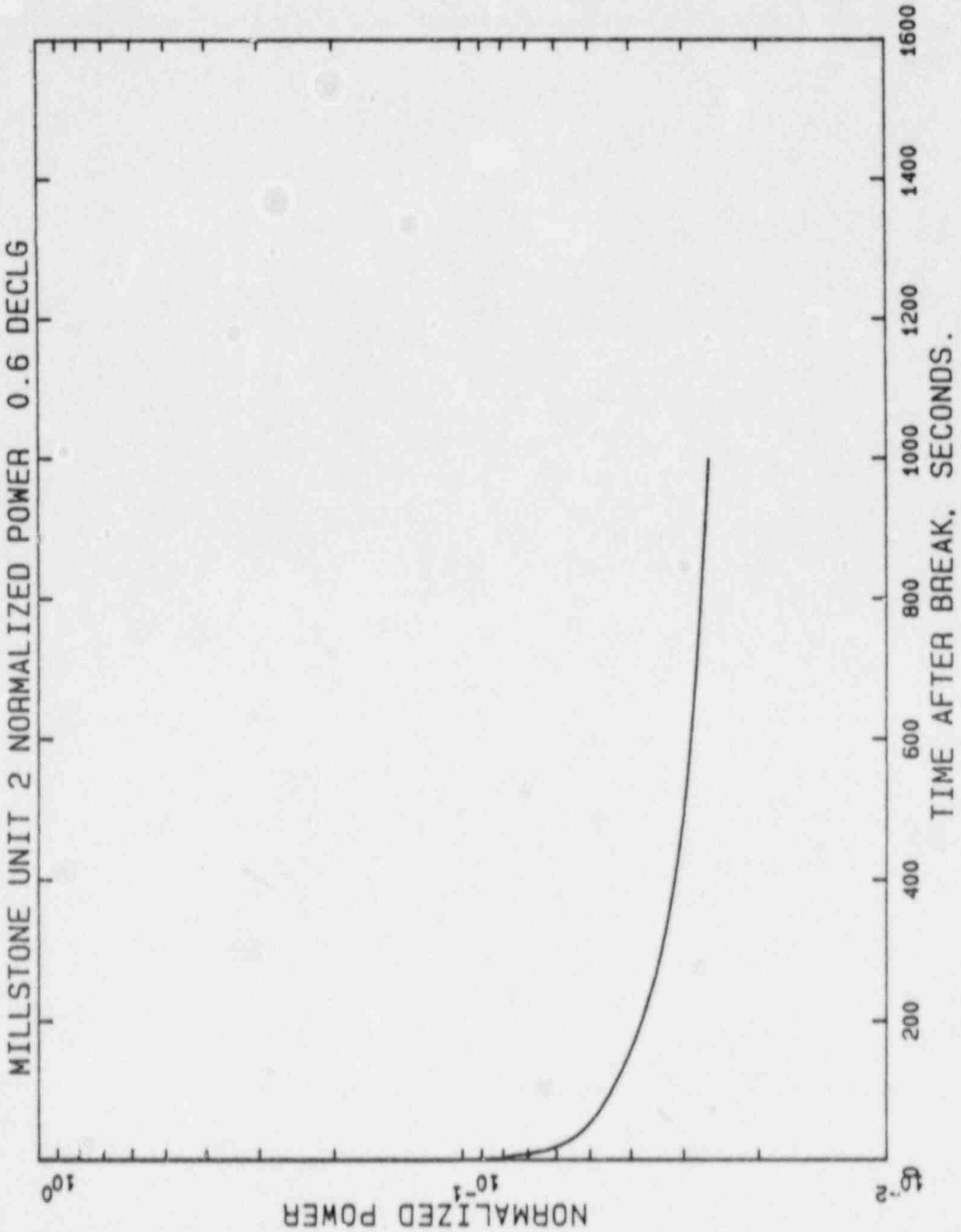

Figure 3.1 Normalized Power (EOC), 0.6 DECLG Break

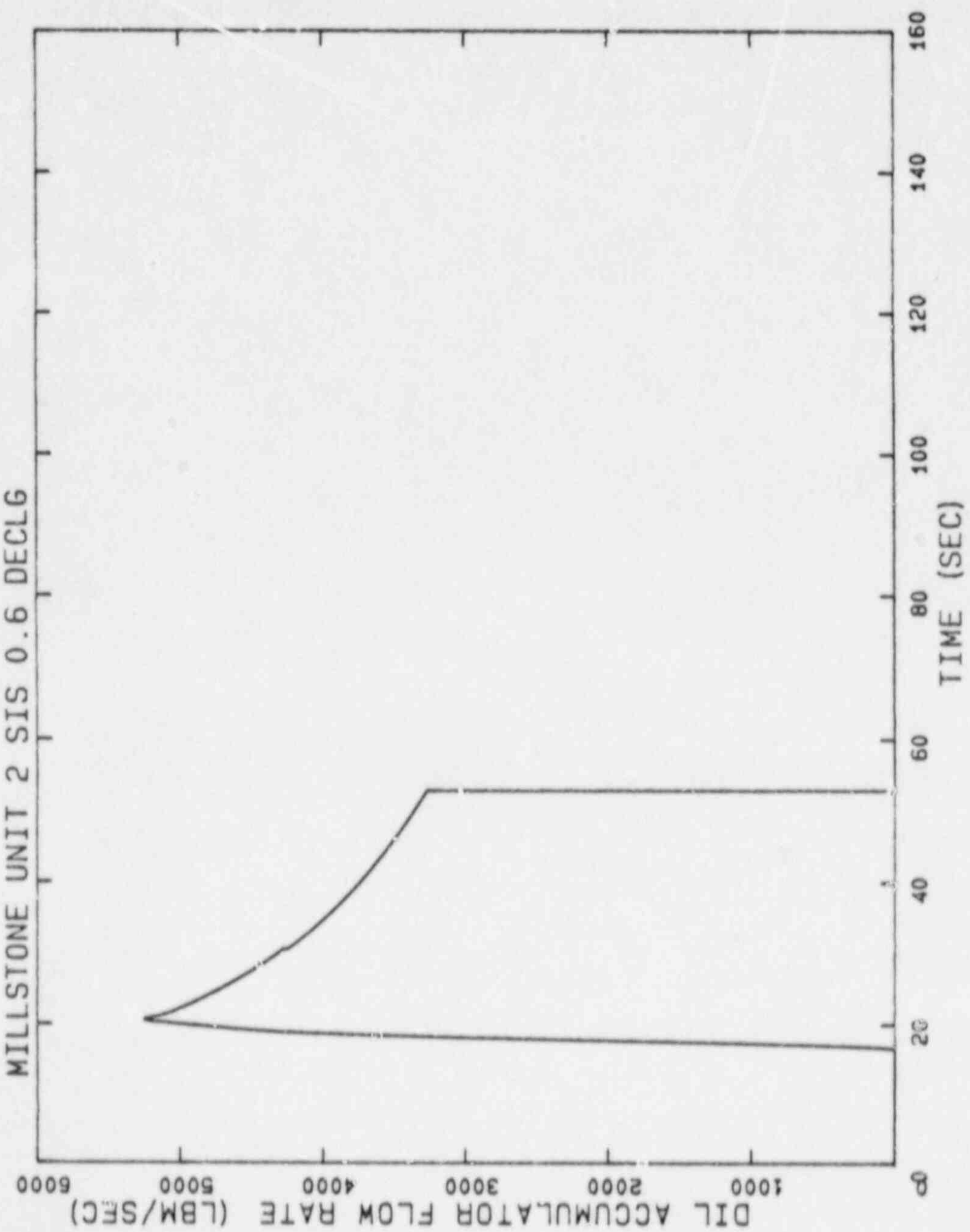

Figure 3.2 Double Intact Loop Accumulator Flow Rate, 0.6 DECLG Break

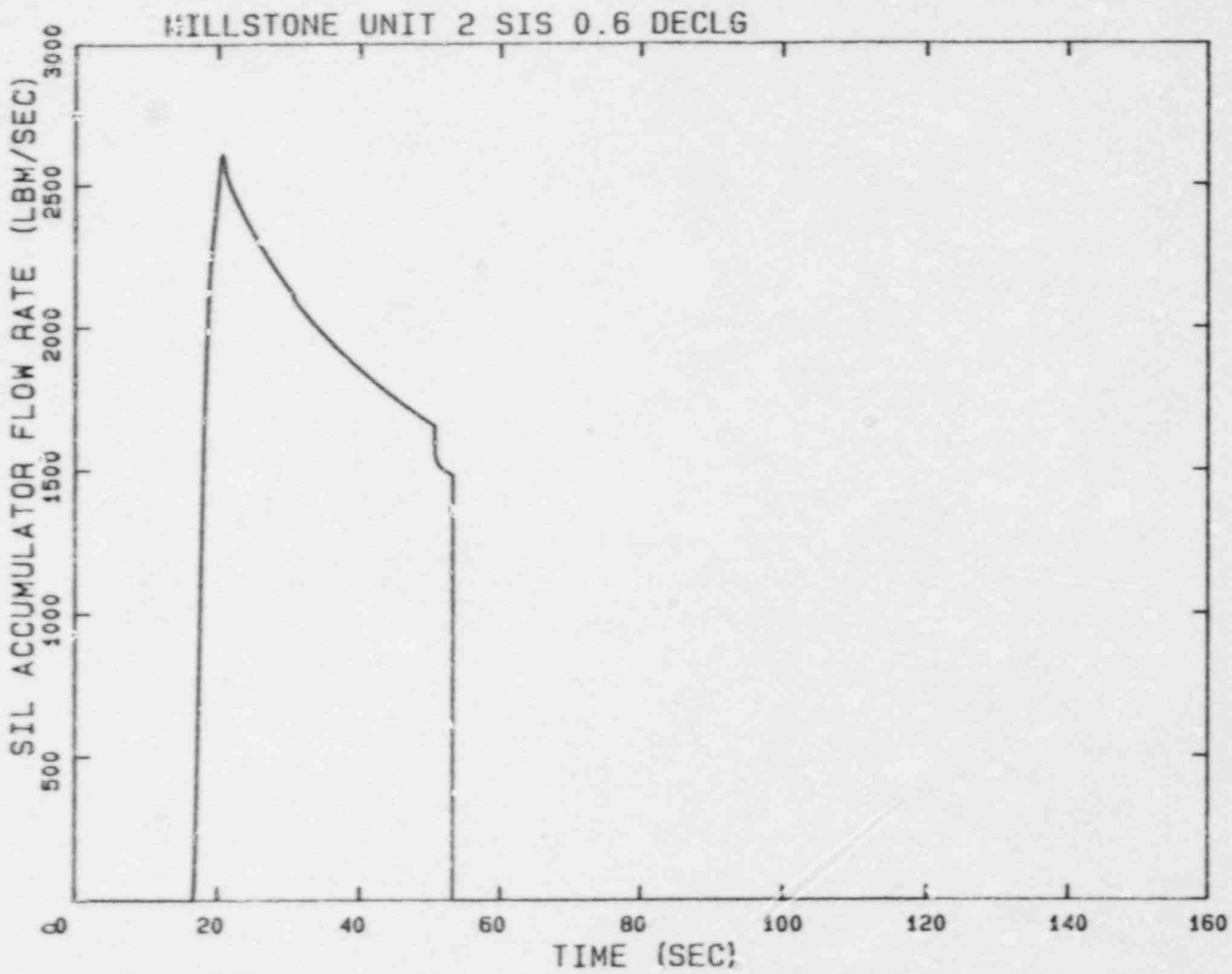

Figure 3.3 Single Intact Loop Accumulator Flow Rate, 0.6 DECLG Break

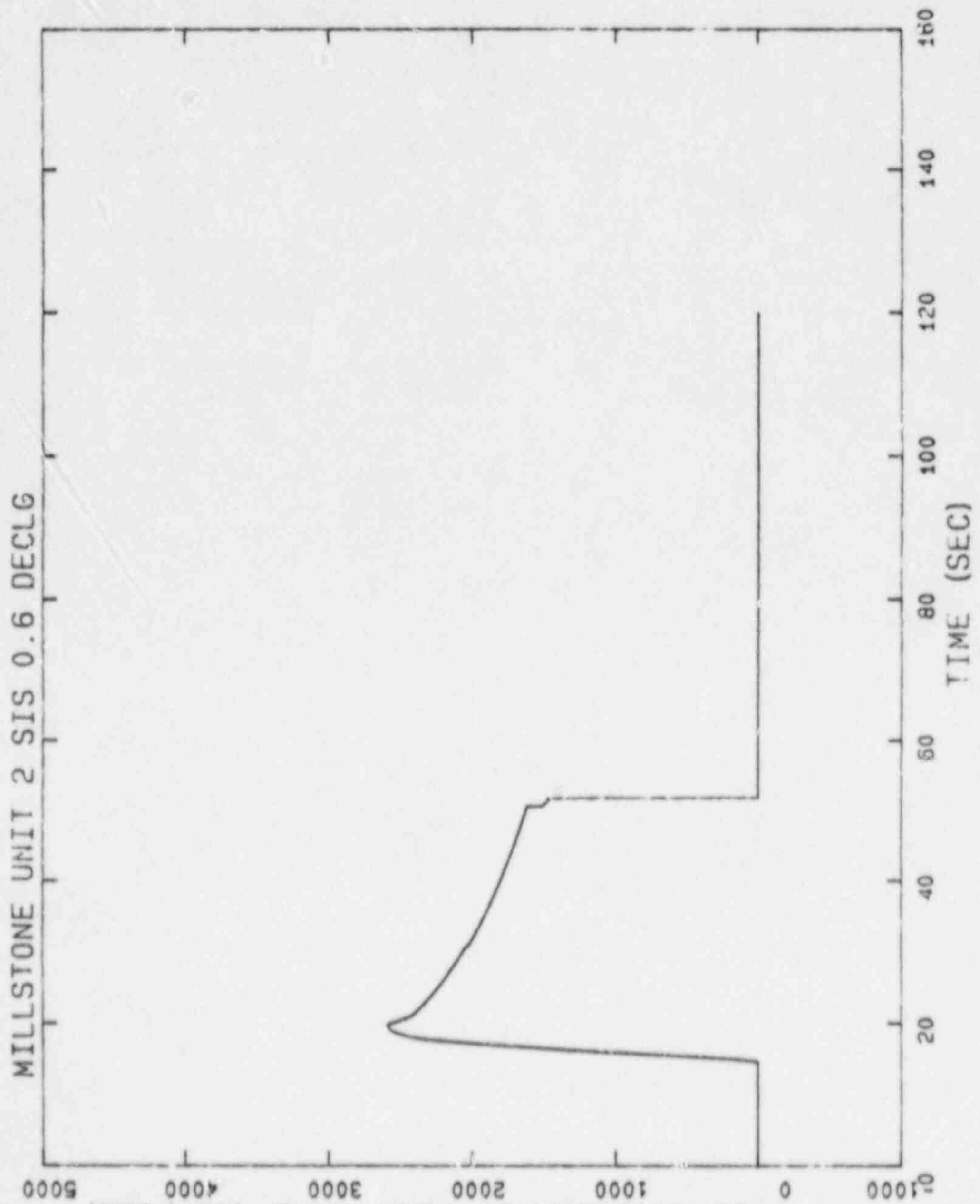

Figure 3.4 Broken Loop Accumulator Flow Rate, 0.6 DECLG Break

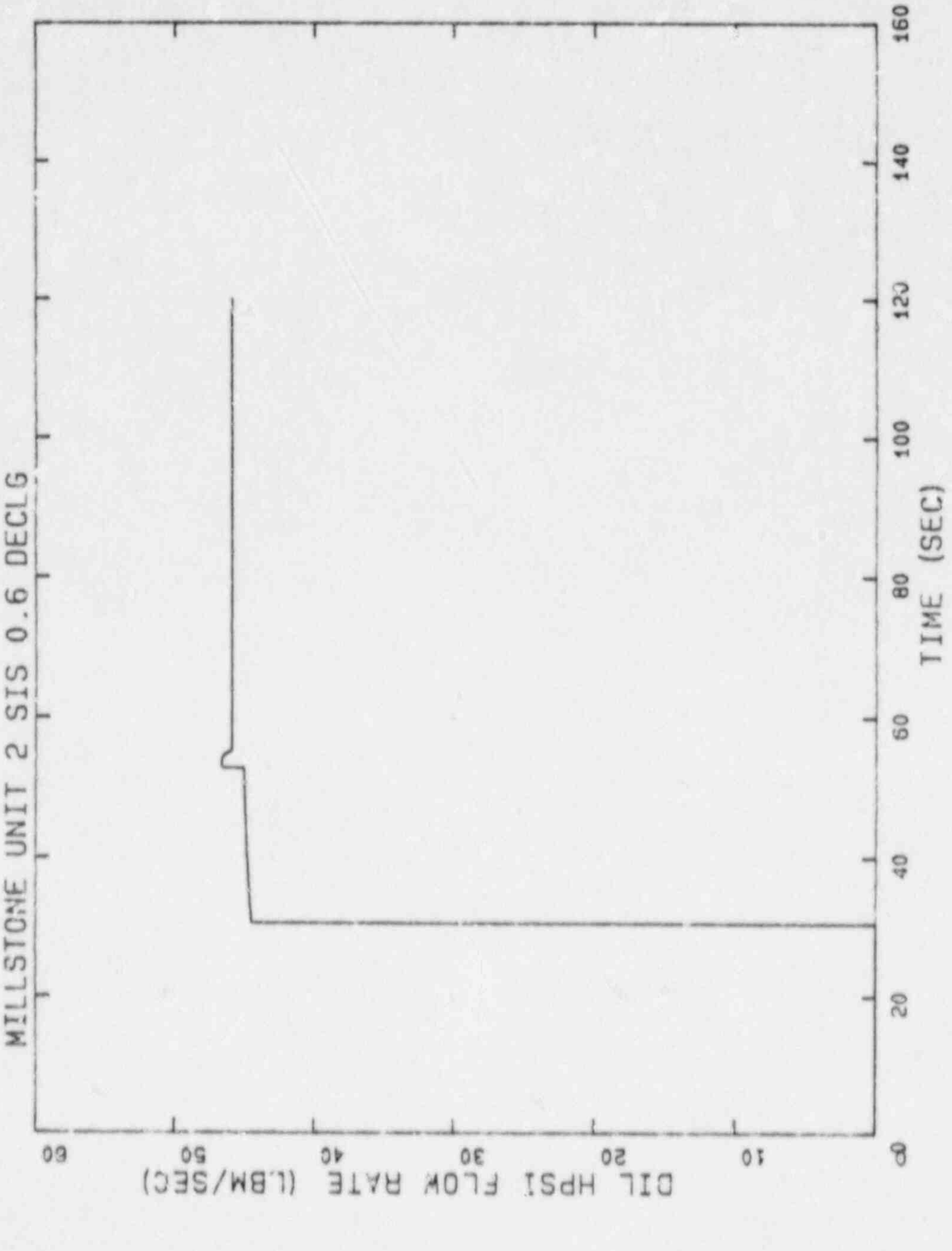

Figure 3.5 Double Intact Loop HPSI Flow Rate, 0.6 DECLG Break

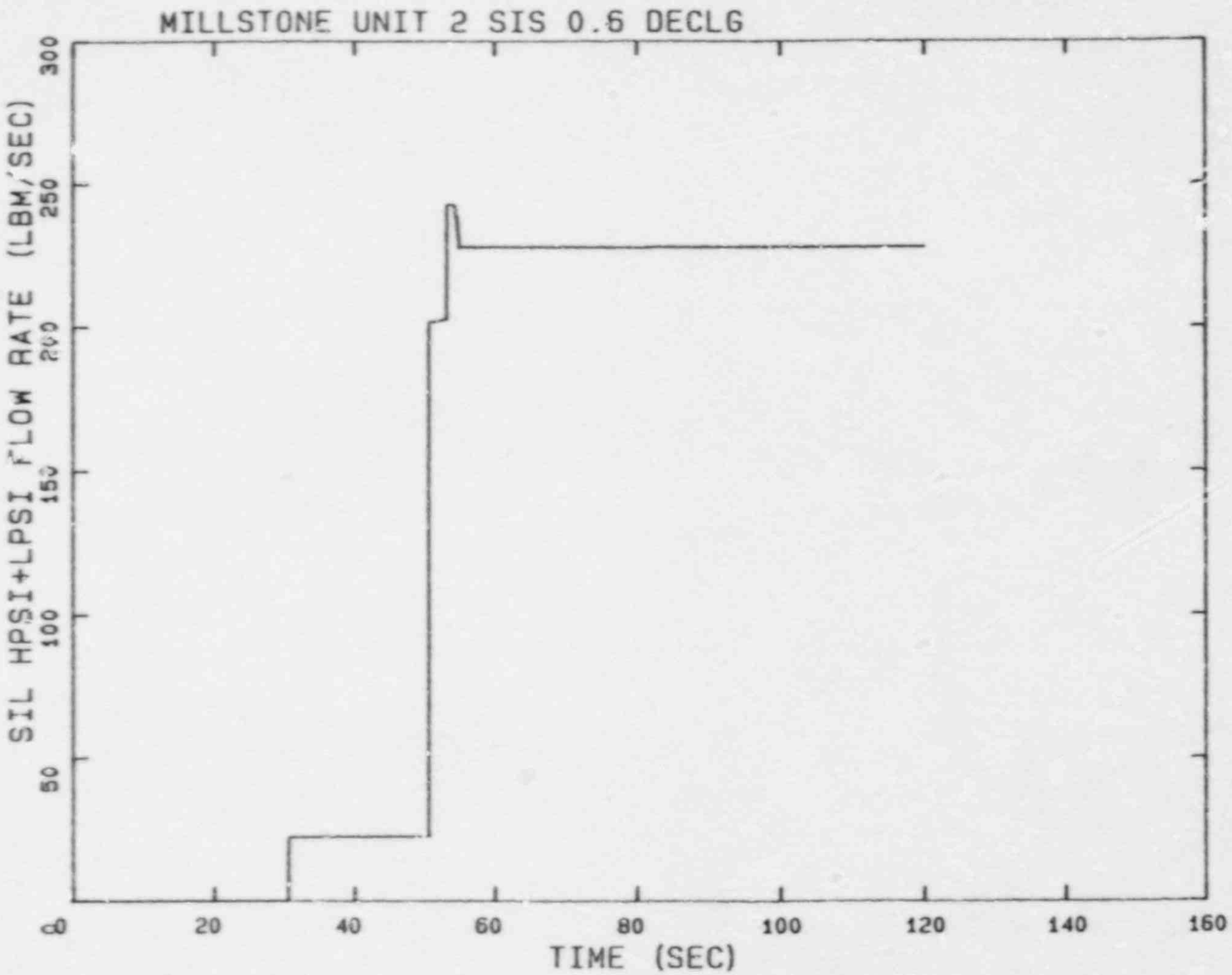

Figure 3.6 Single Intact Loop SIS Flow Rate, 0.6 DECLG Break

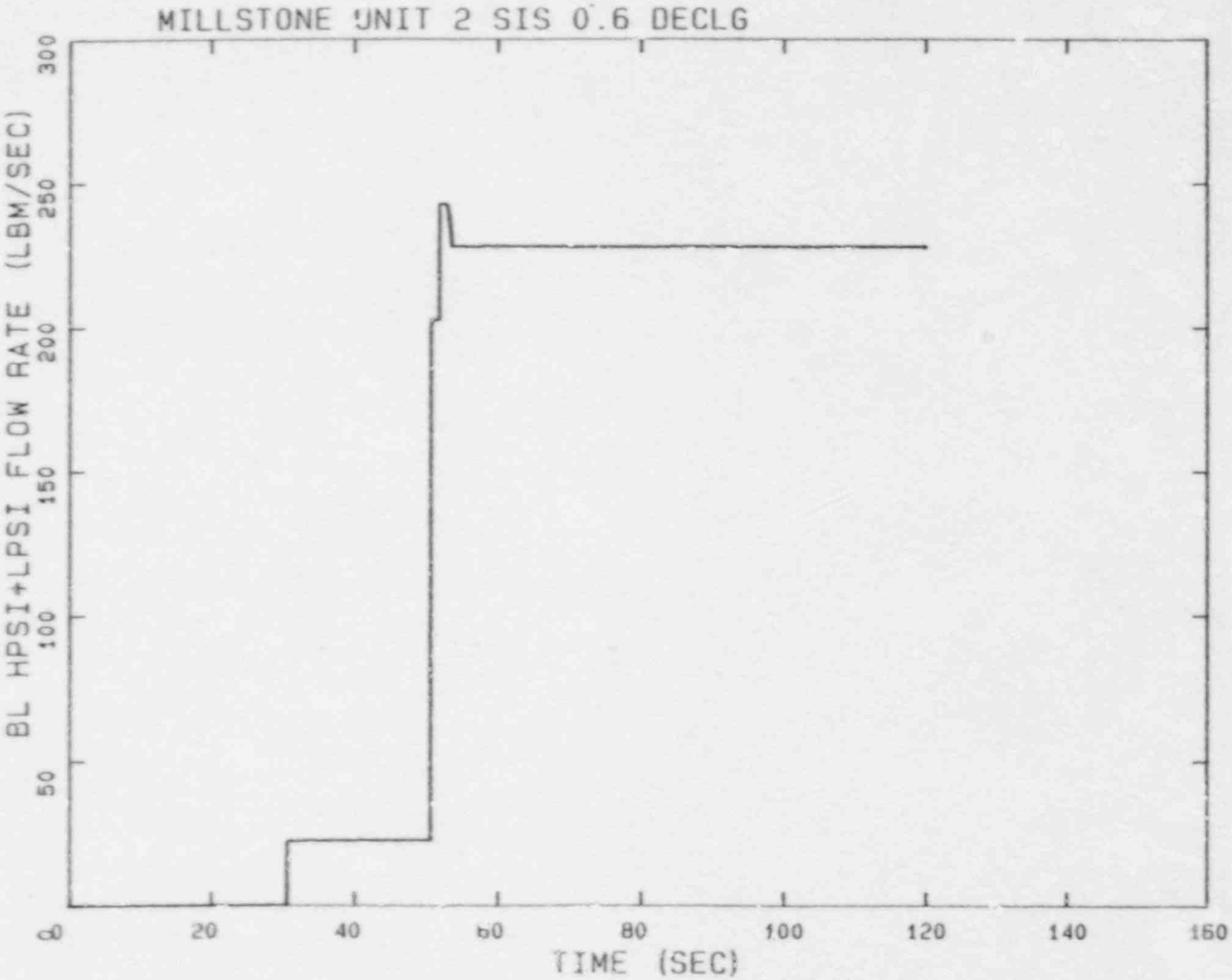

Figure 3.7 Broken Loop SIS Flow Rate, 0.6 DECLG Break

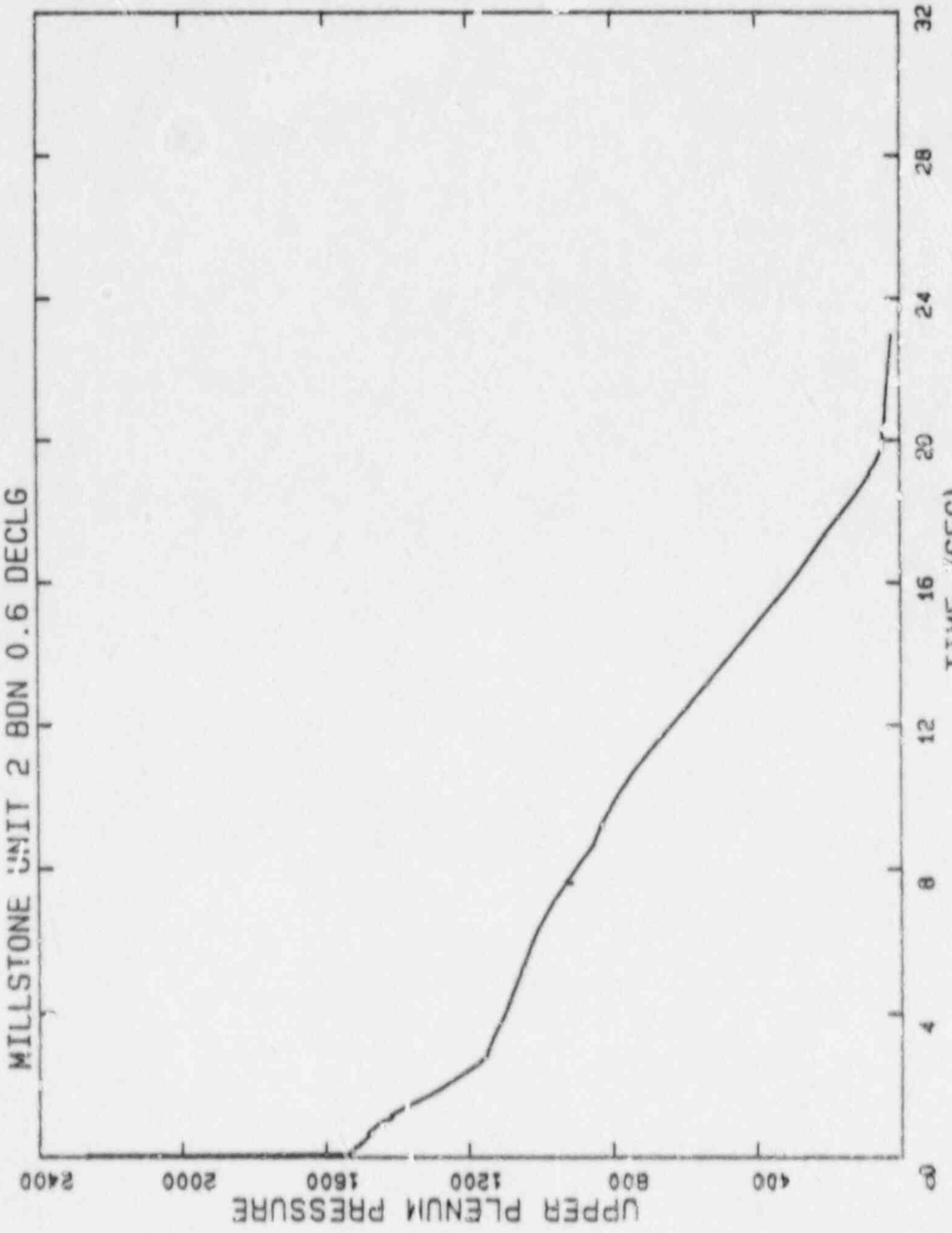

Figure 3.8 Upper Plenum Pressure during Blowdown, 0.6 DECLG Break

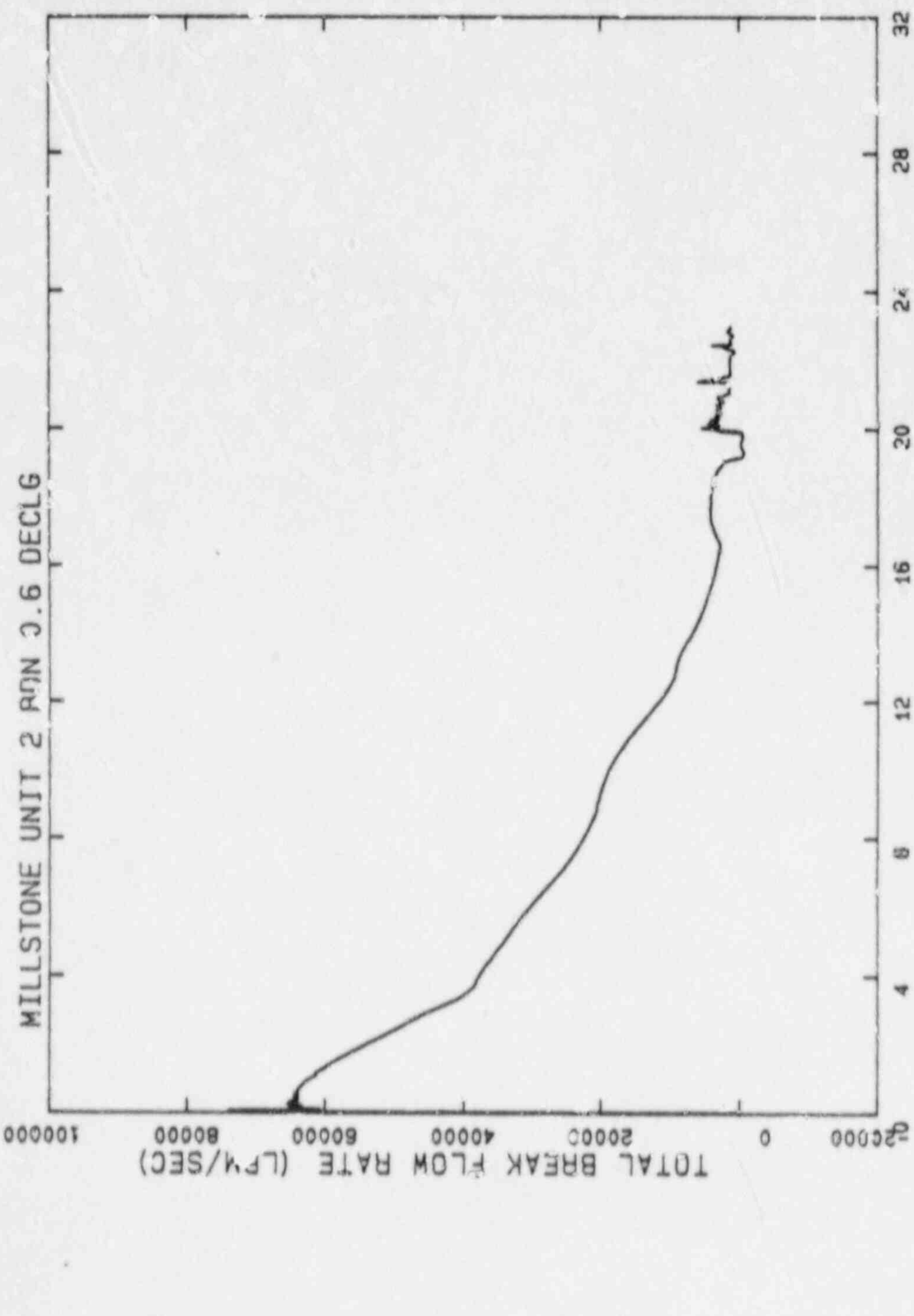

Figure 3.5 Total Break Flow Rate during Blowdown, 0.6 DECLG Break

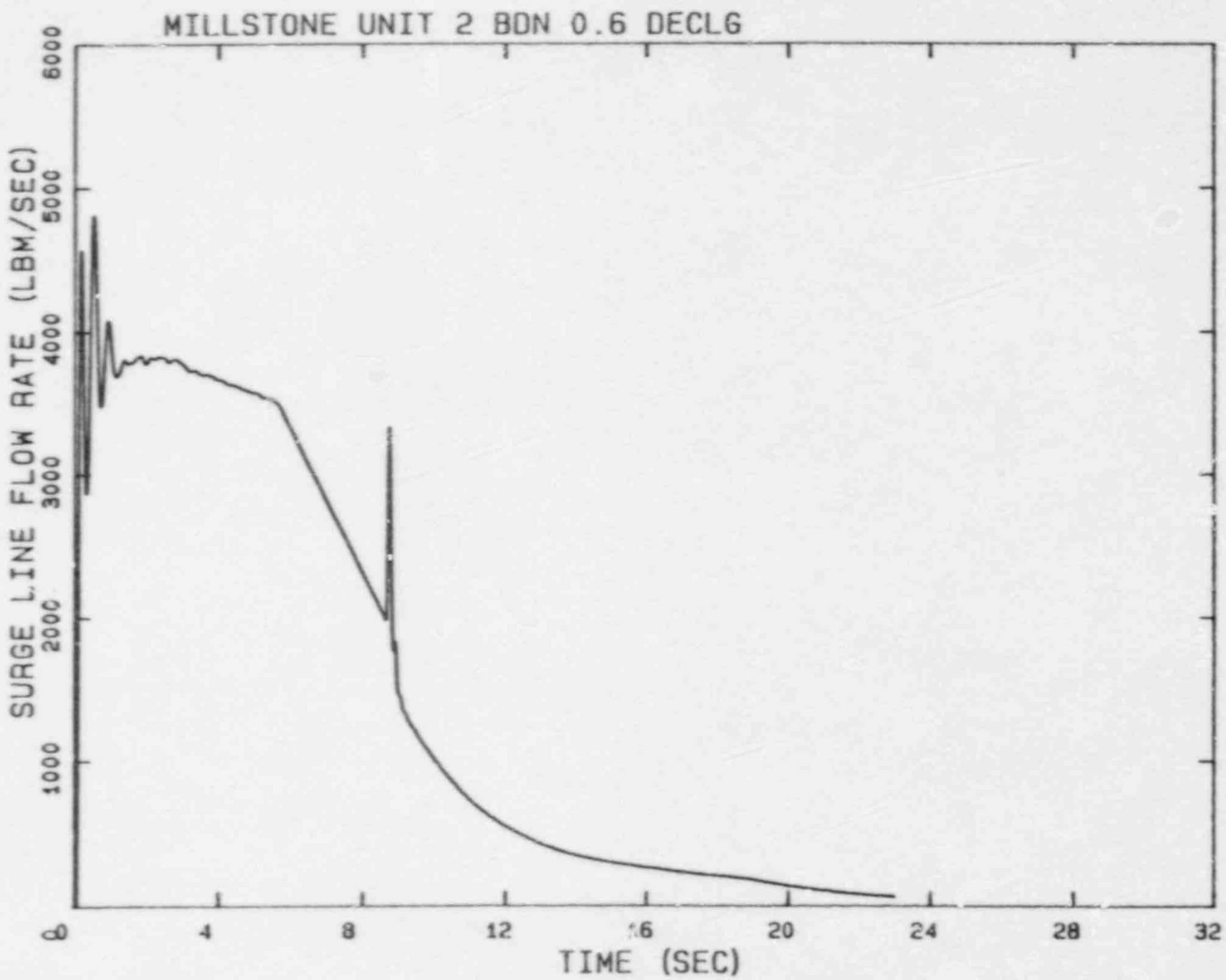

Figure 3.10 Pressurizer Surge Line Flow Rate during Blowdown, 0.6 DECLG Break

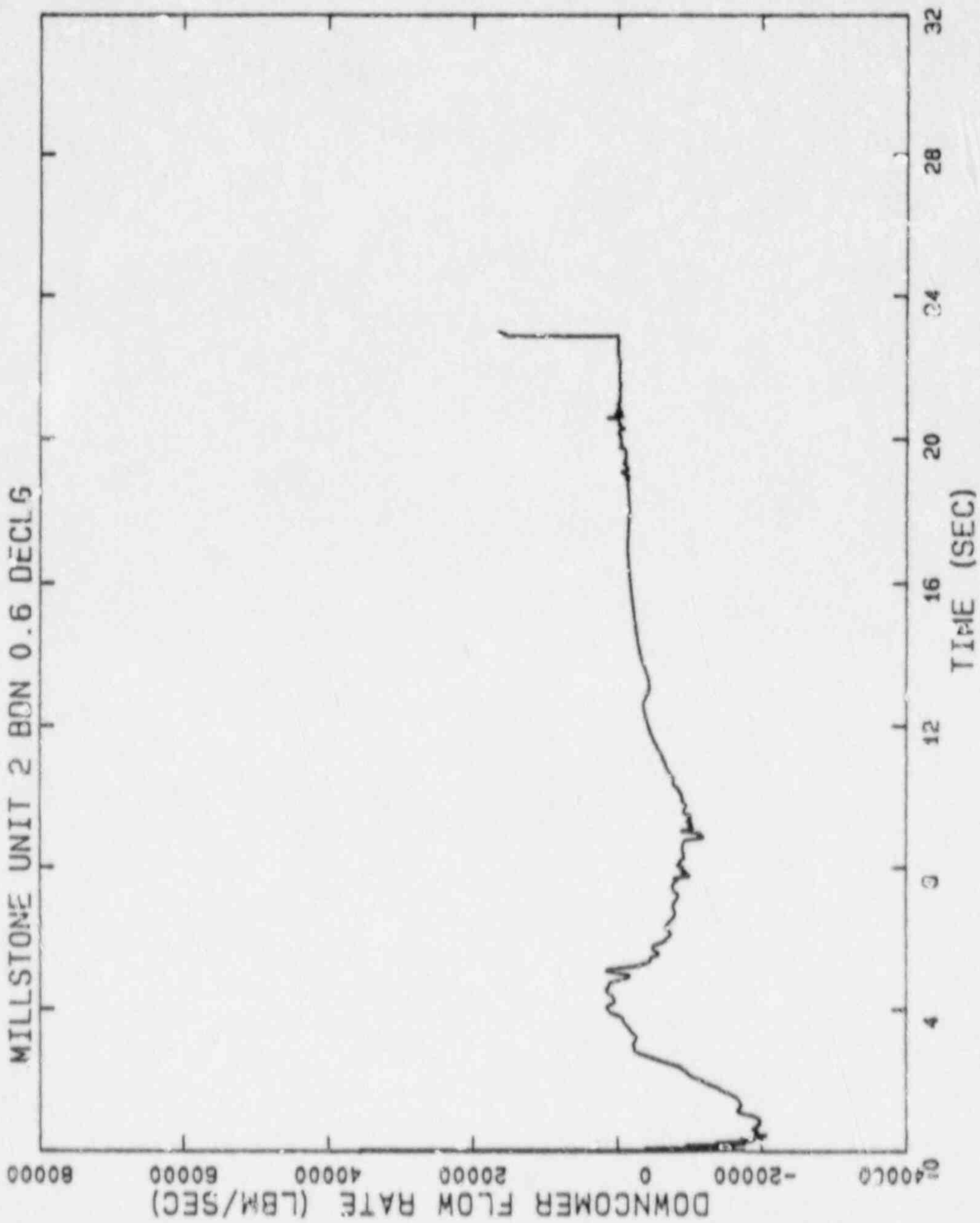

Figure 3.11 Downcomer Flow Rate during Blowdown, 0.5 DECLG Break

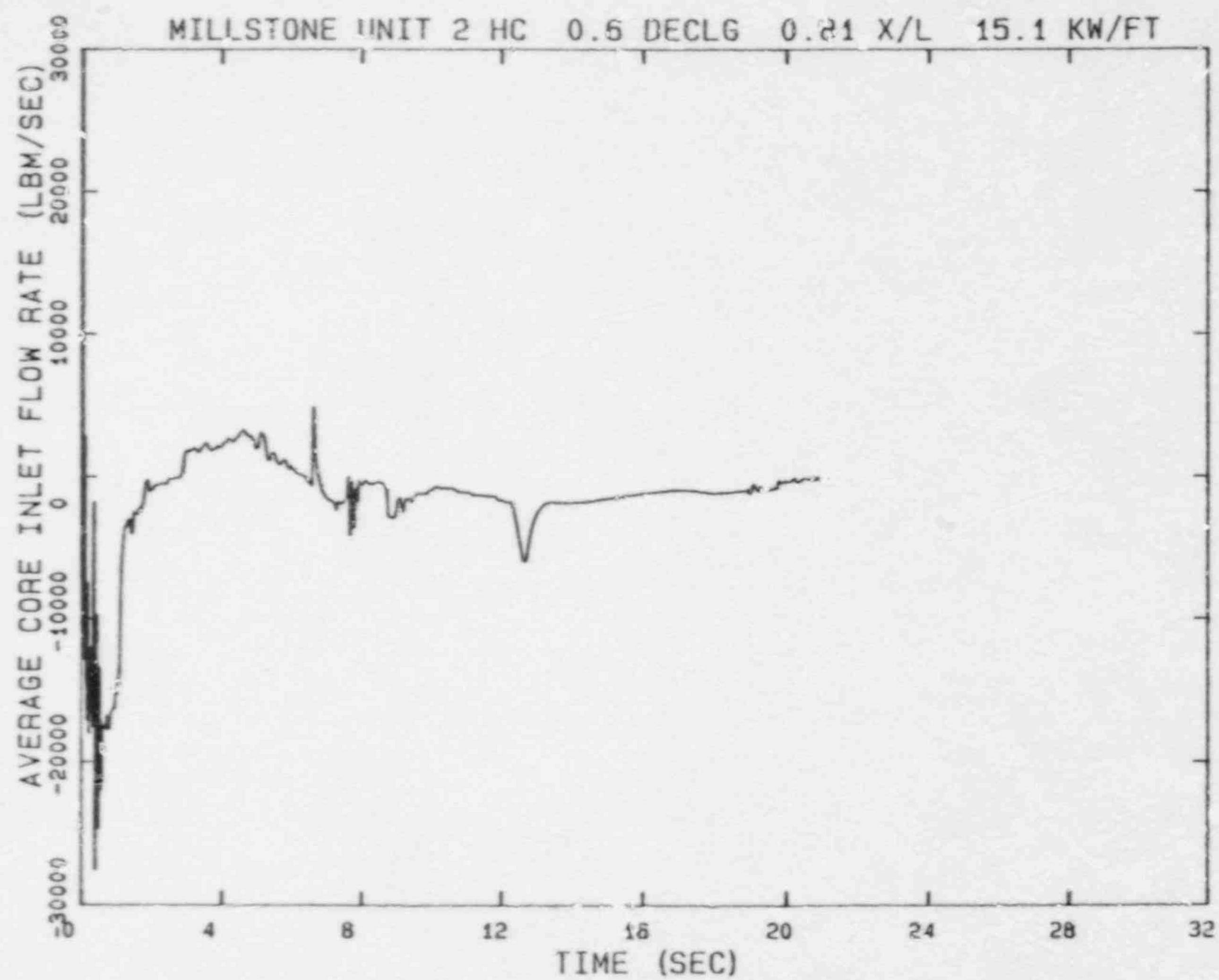

Figure 3.12 Average Core Inlet Flow Rate during Blowdown, 0.6 DECLG Break, X/L = 0.81

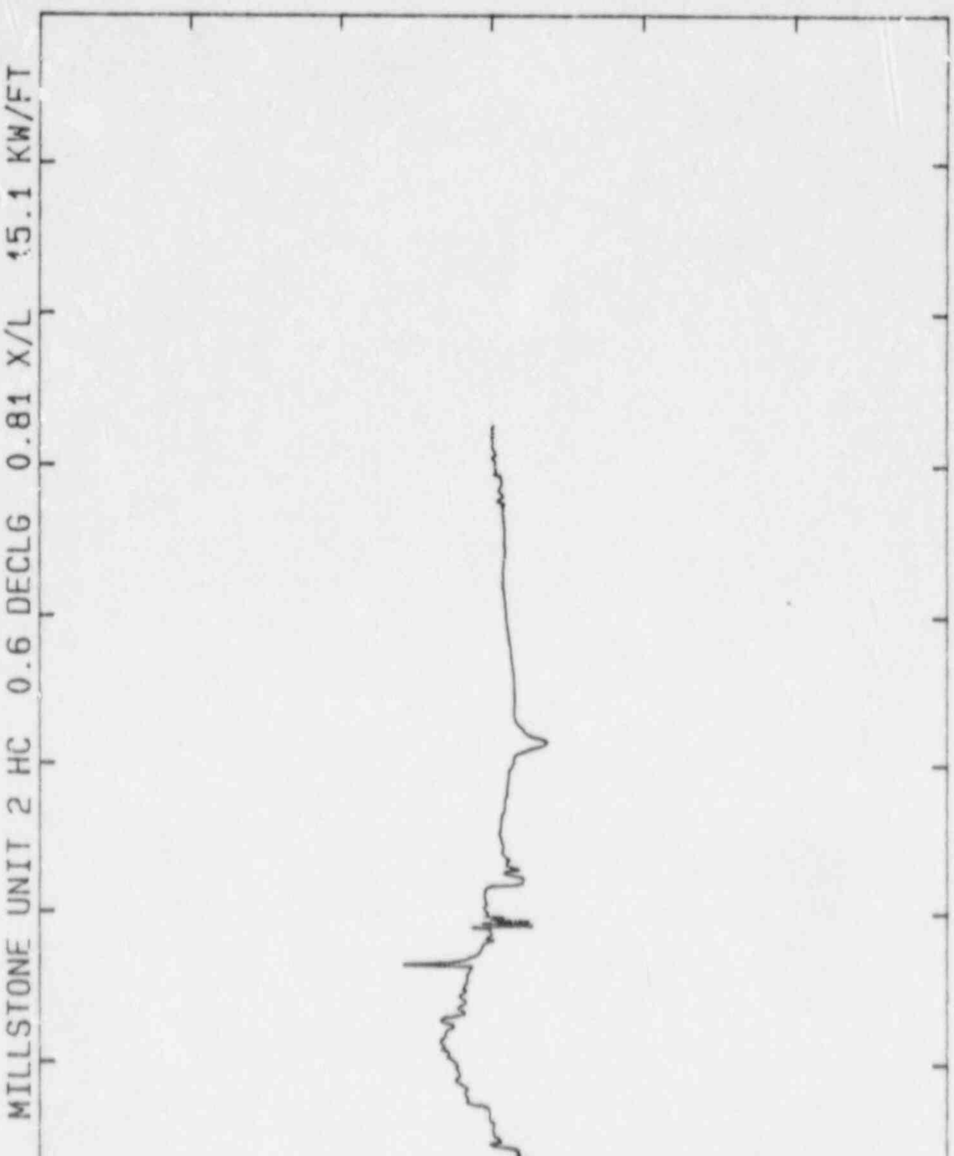

Figure 3.13 Hot Channel Inlet Flow Rate during Blowdown, 0.6 DECLG Break, X/L = 0.81

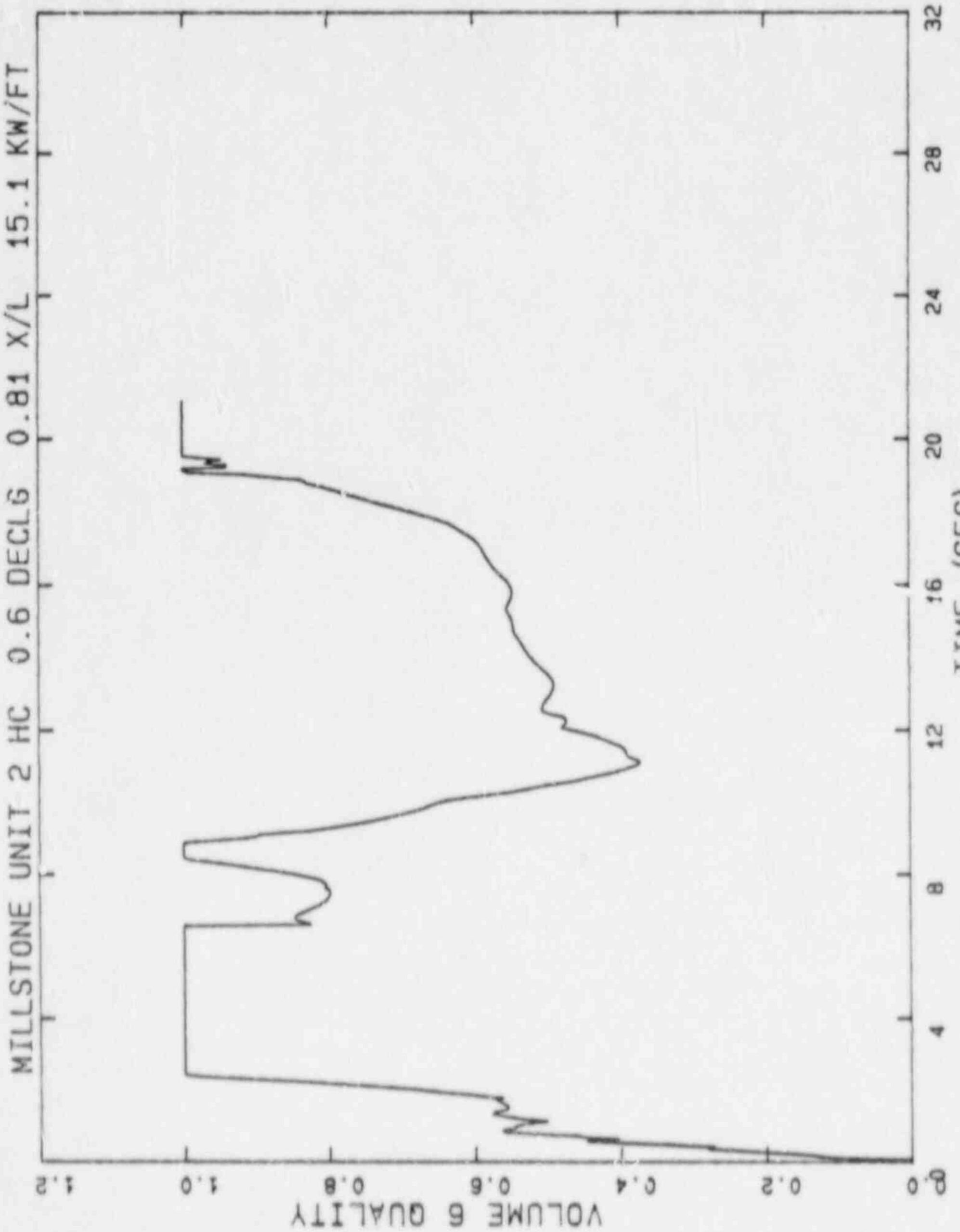

Figure 3.14 PCT Node Fluid Quality during Blowdown, 0.6 DECLG Break, X/L = 0.81

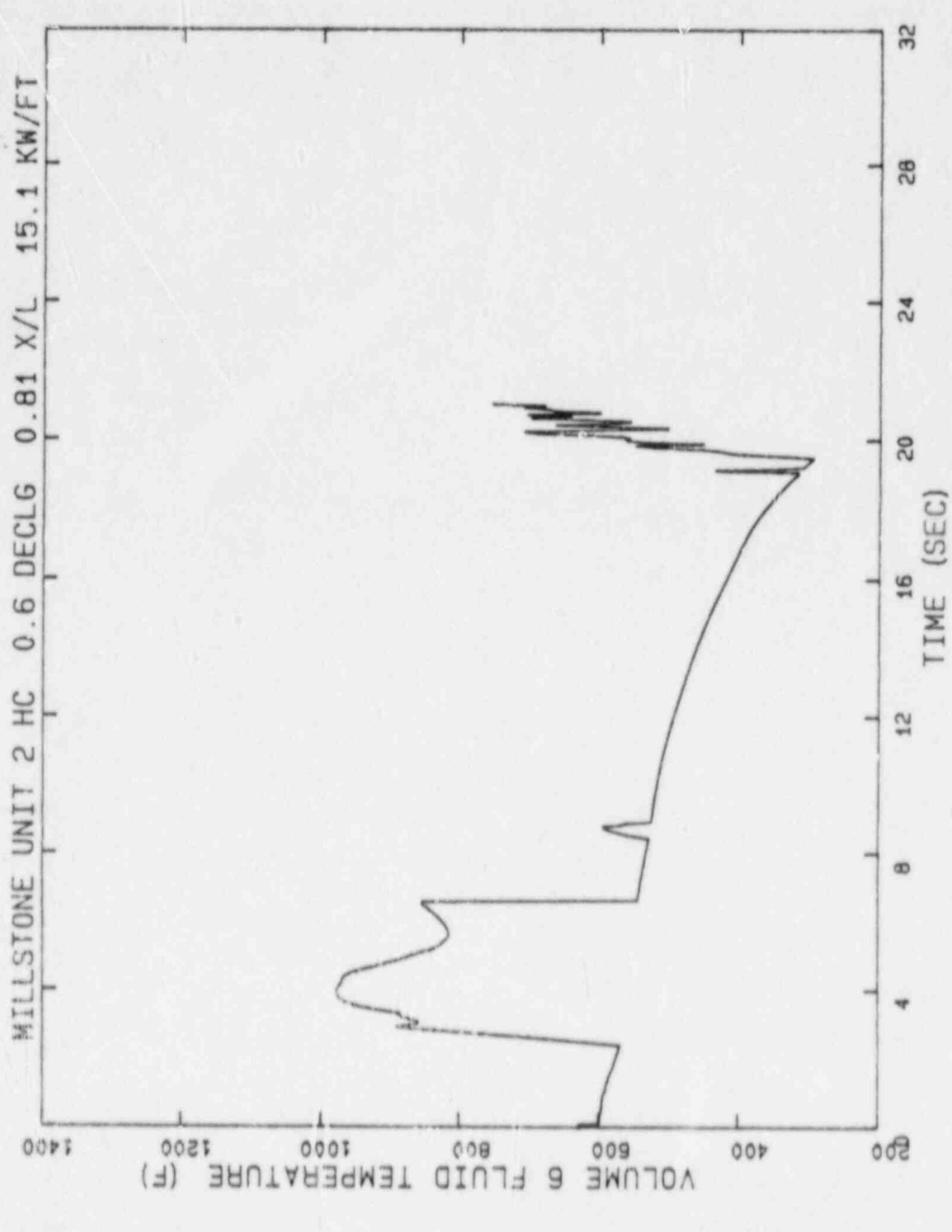

Figure 3.15 PCT Node Fluid Temperature during Blowdown, 0.6 DECLG Break, X/L = 0.81

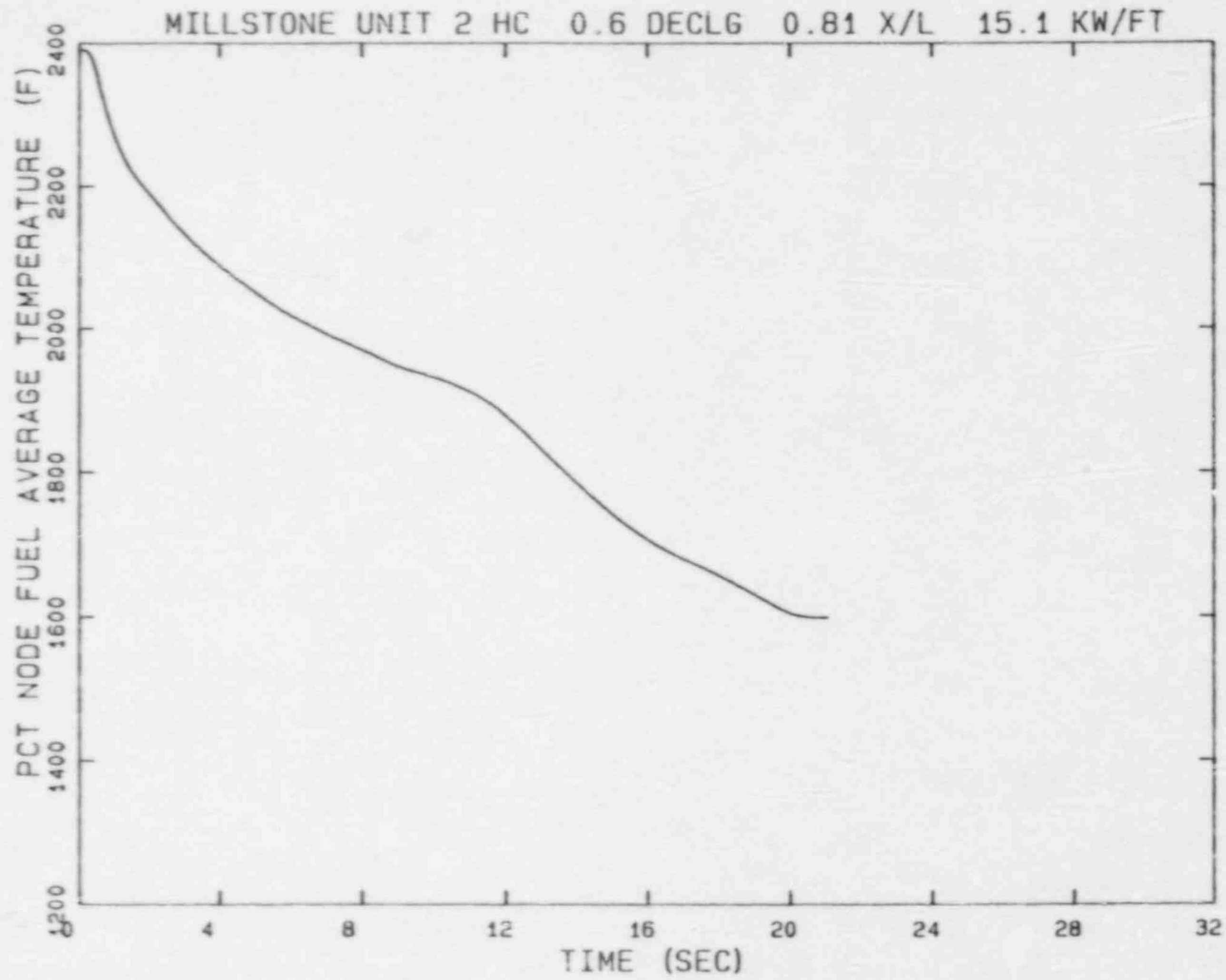

Figure 3.16 PCT Node Fuel Average Temperature during Blowdown, 0.6 DECLG Break, X/L = 0.81

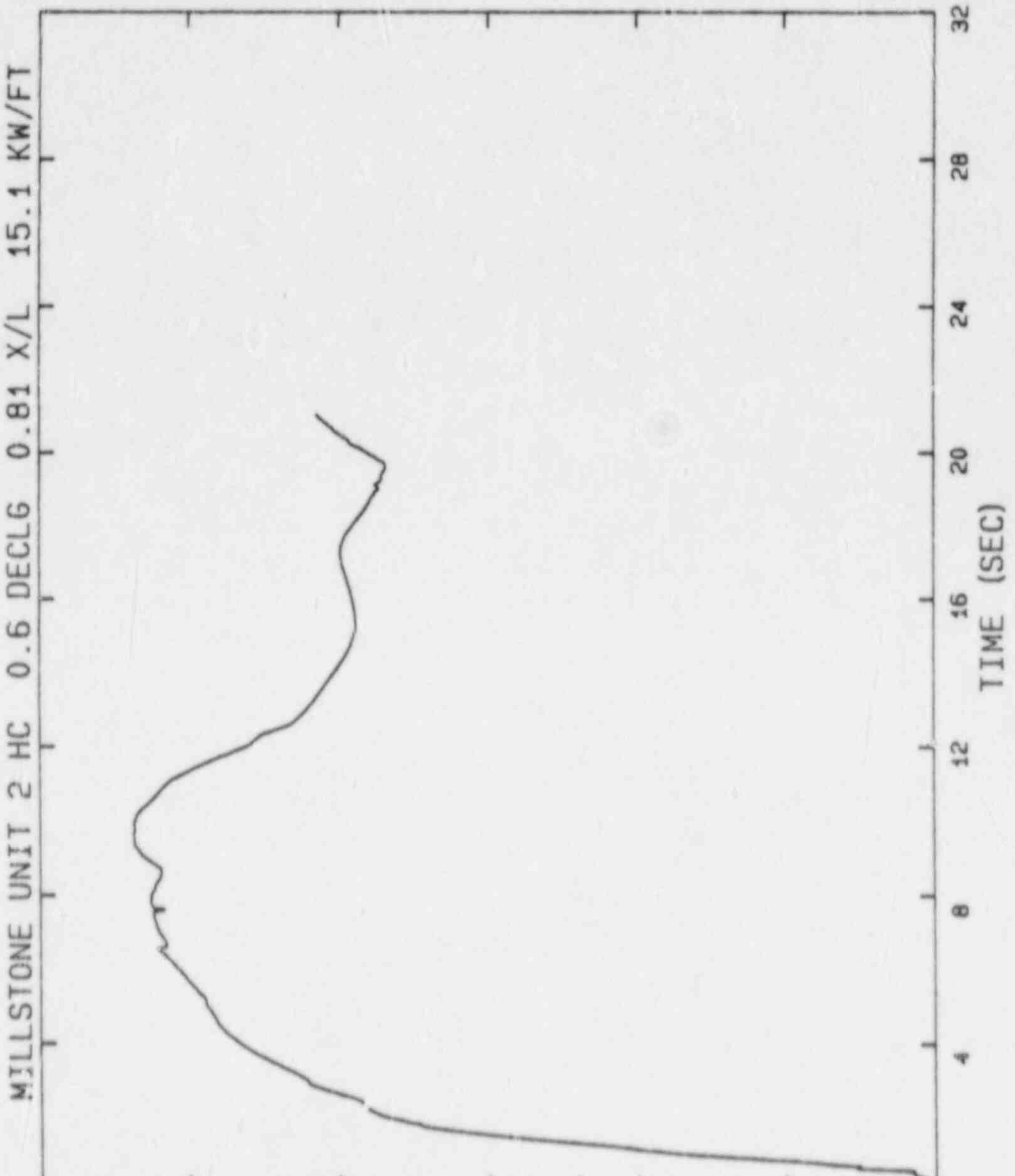

Figure 3.17 PCT Node Cladding Temperature during Blowdown, 0.6 DECLG Break, X/L = 0.81

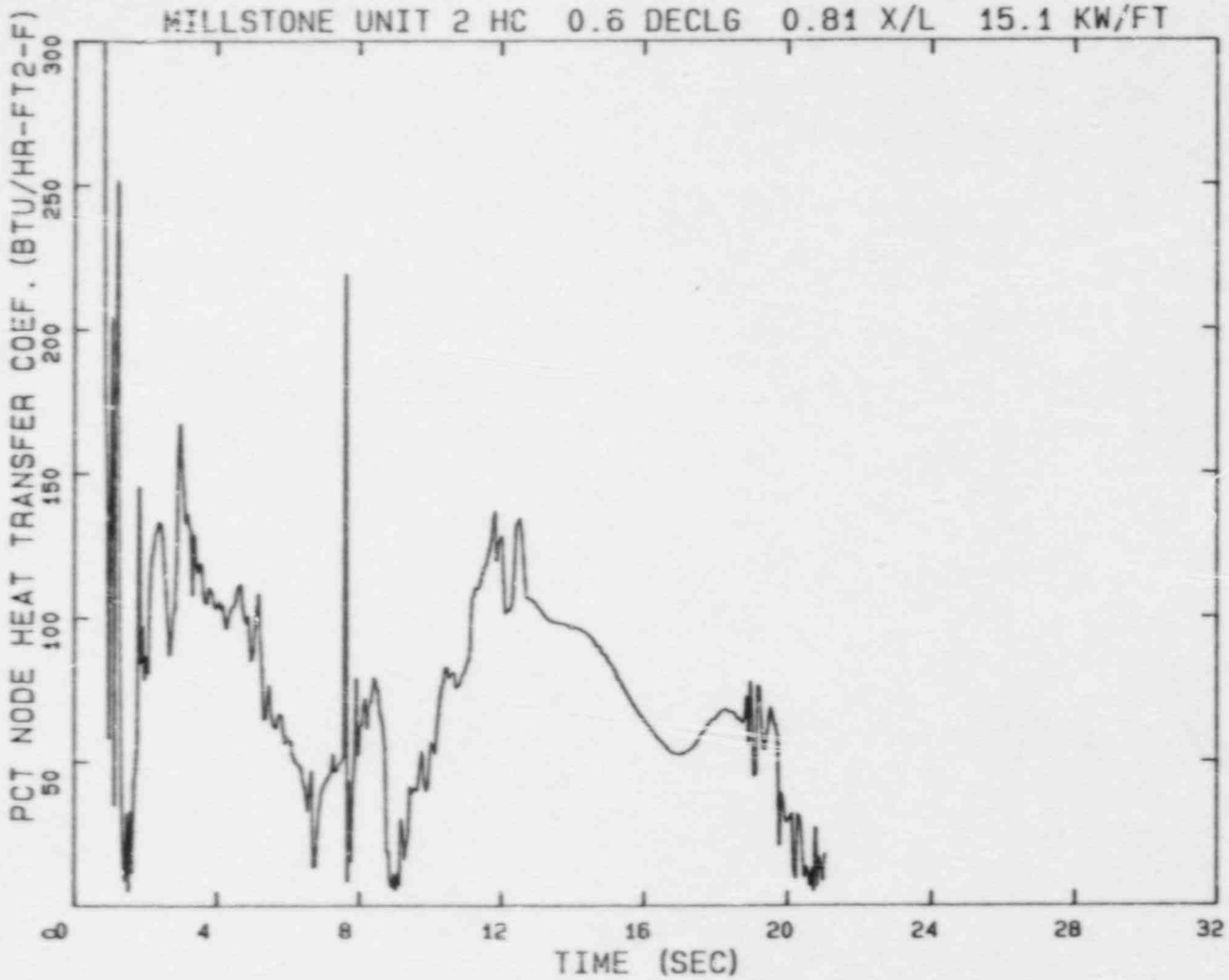

Figure 3.18 PCT Node Heat Transfer Coefficient during Blowdown, 0.6 DECLG Break, X/L = 0.81

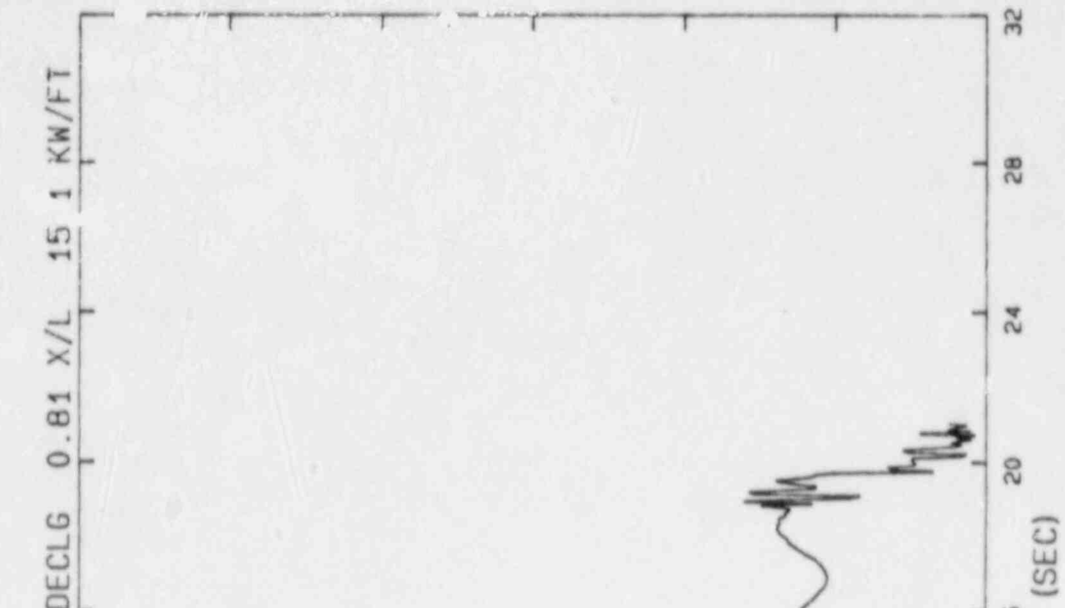

Figure 3.19 PCT Node Heat Flux during Blowdown, 0.6 DECLG Break, X/L = 0.81

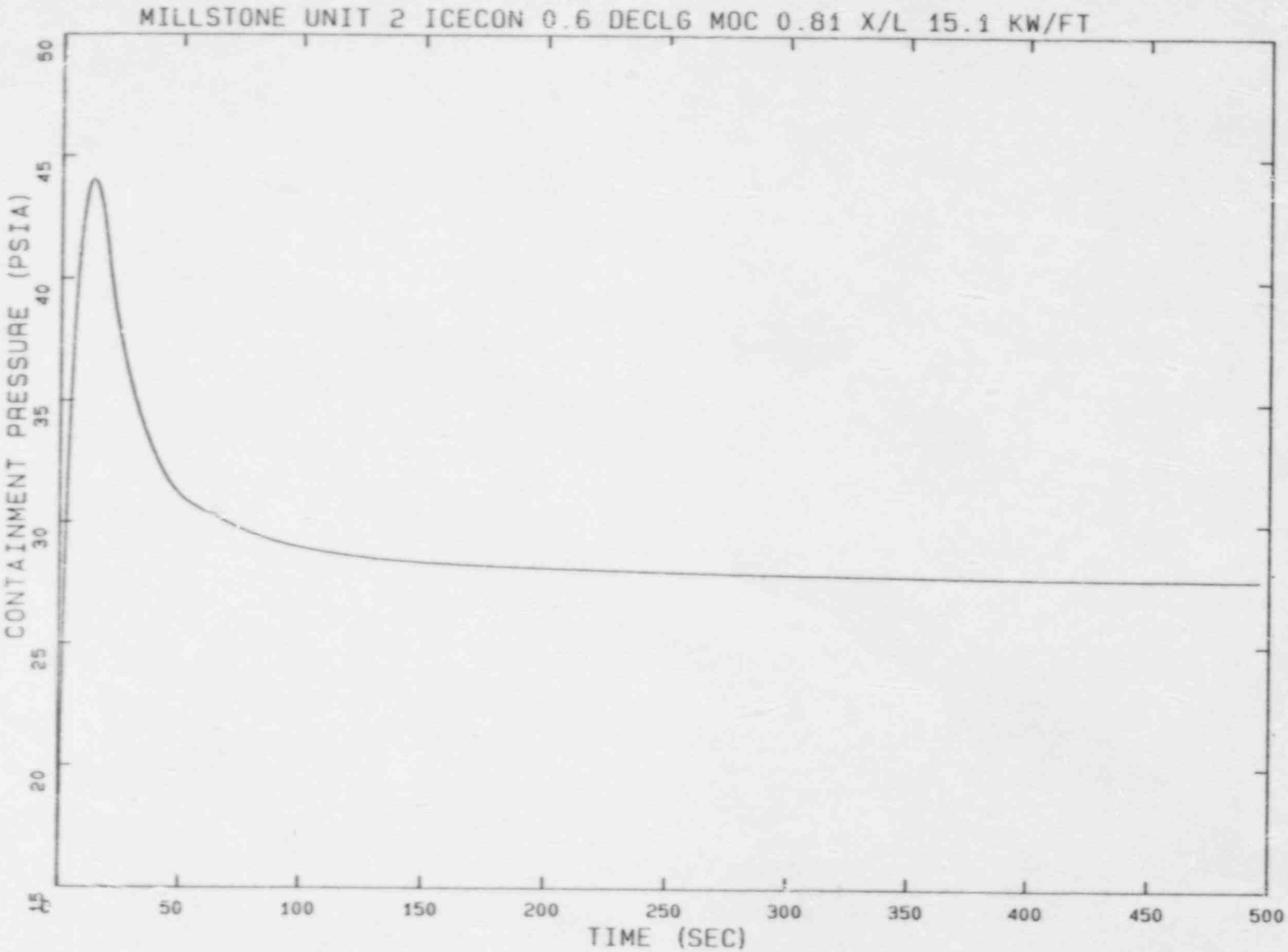

Figure 3.20 Containment Pressure, 0.6 DECLG Break, X/L = 0.81

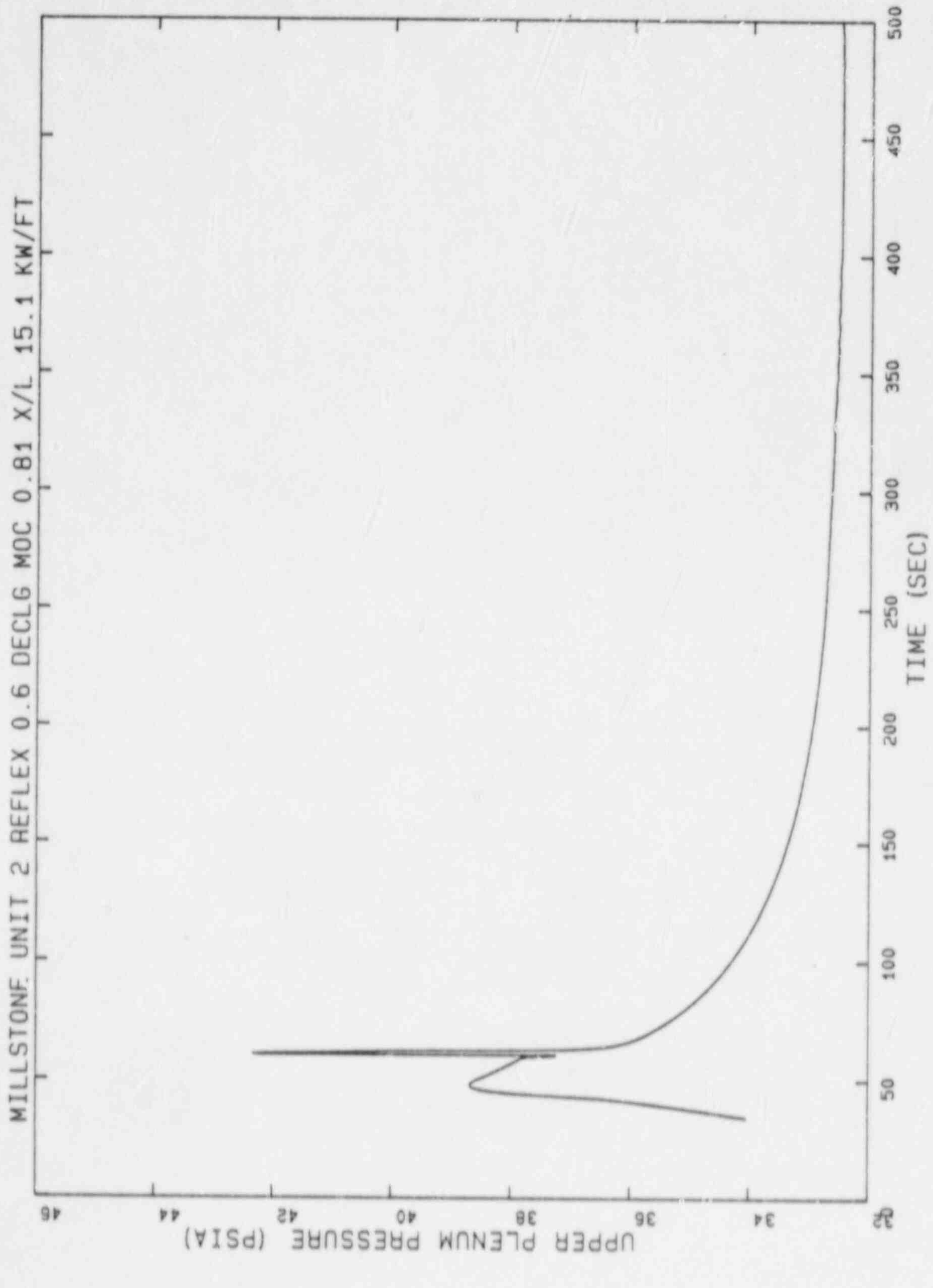

Figure 3.21 Upper Plenum Pressure after EOBY, 0.6 DECLG Break, X/L = 0.81

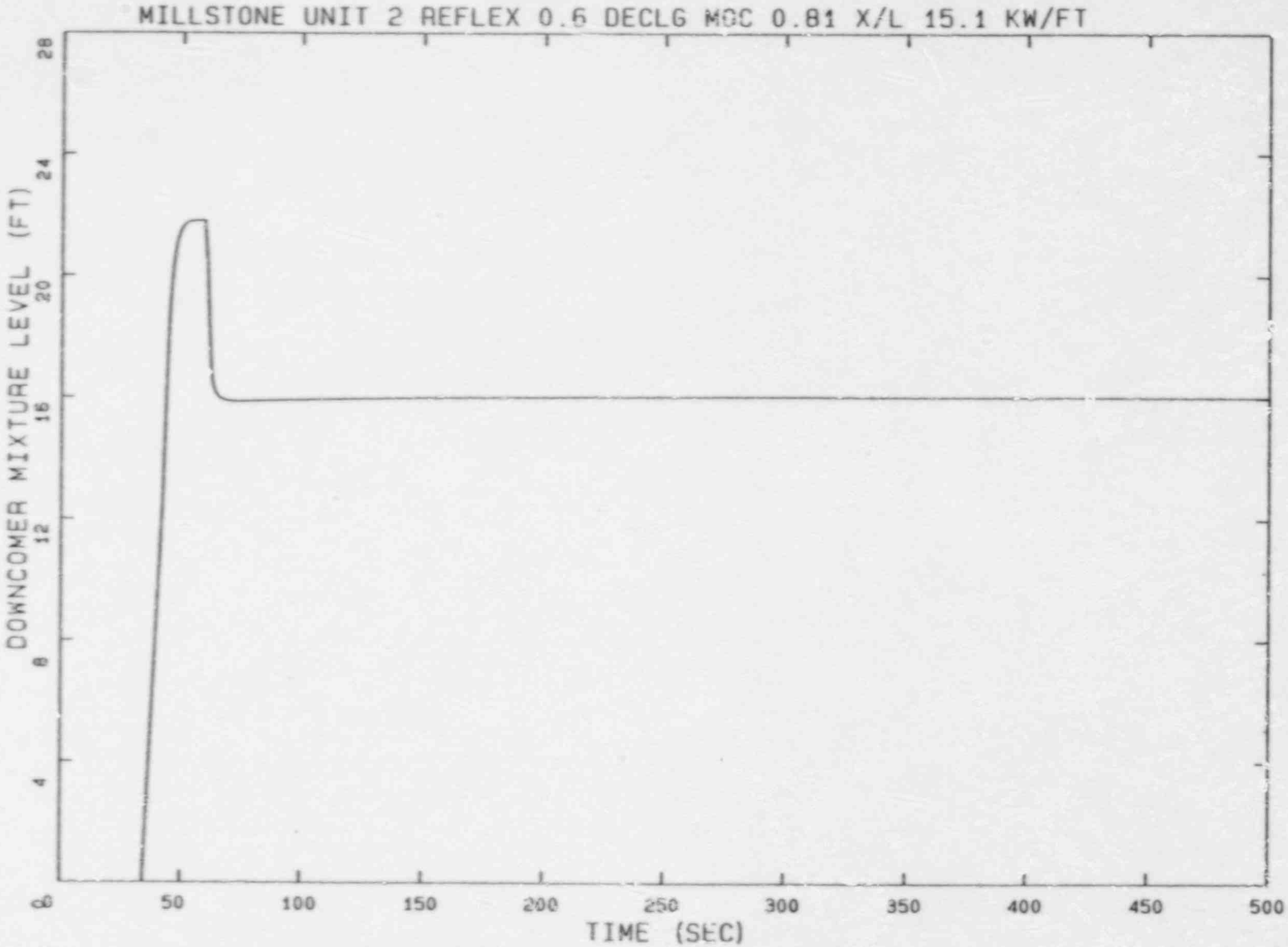

Figure 3.22 Downcomer Mixture Level after EOBY, 0.6 DECLG Break, X/L = 0.81

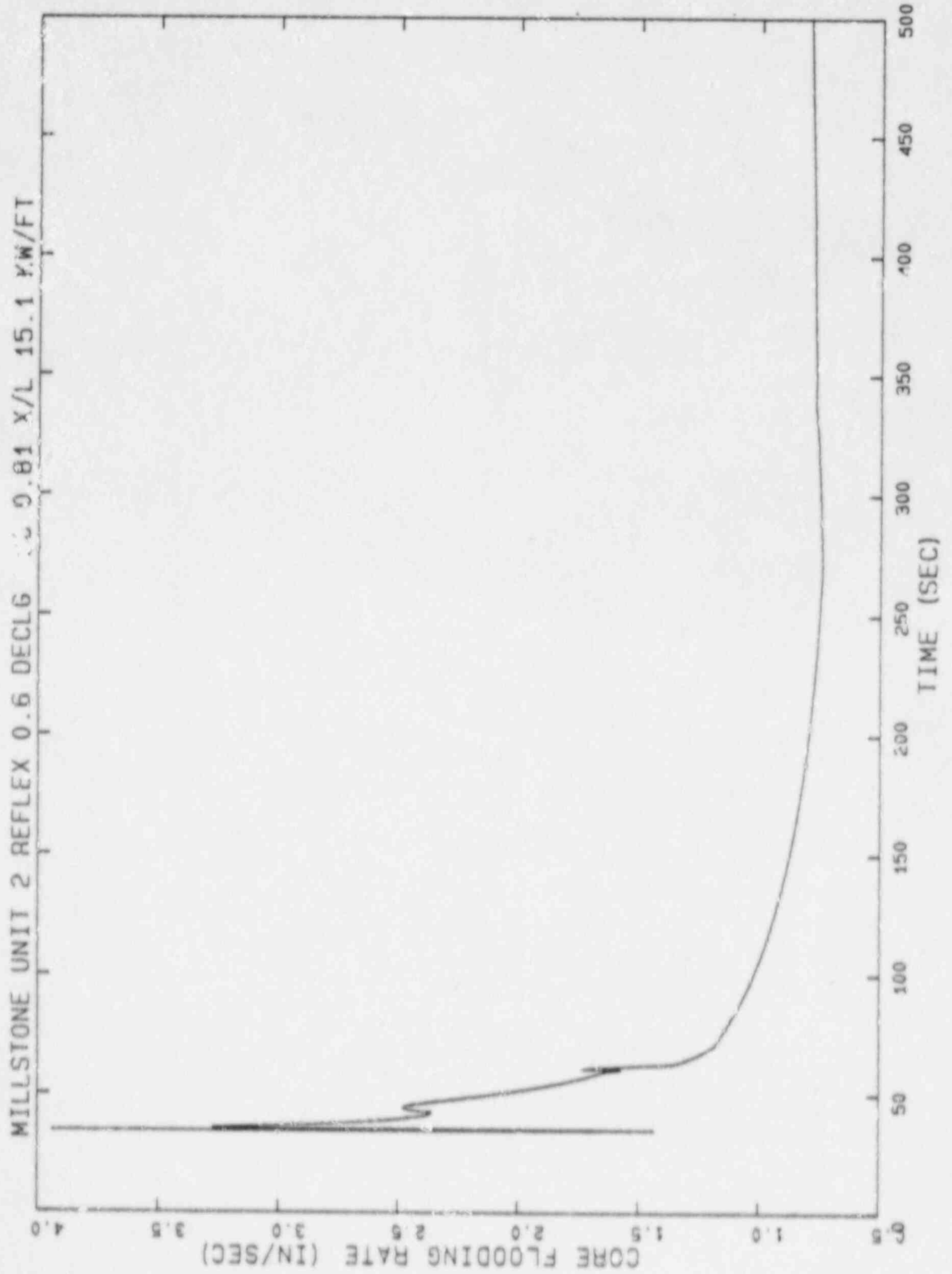

Figure 3.23 Core Flooding Rate after EOBY, 0.6 DECLG Break, X/L = 0.81

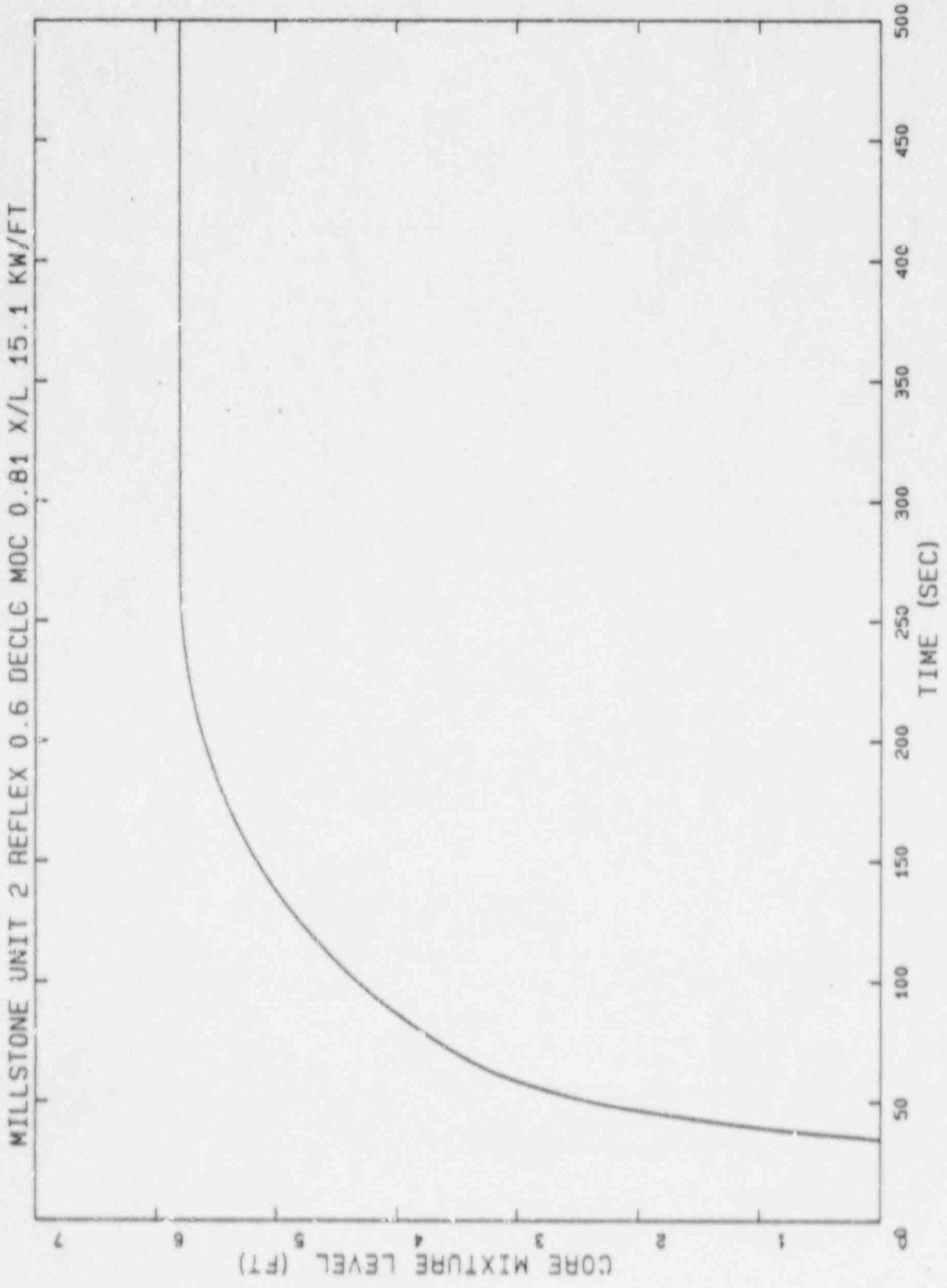

Figure 3.24 Core Mixture Level after EOBY, 0.6 DECLG Break, X/L = 0.81

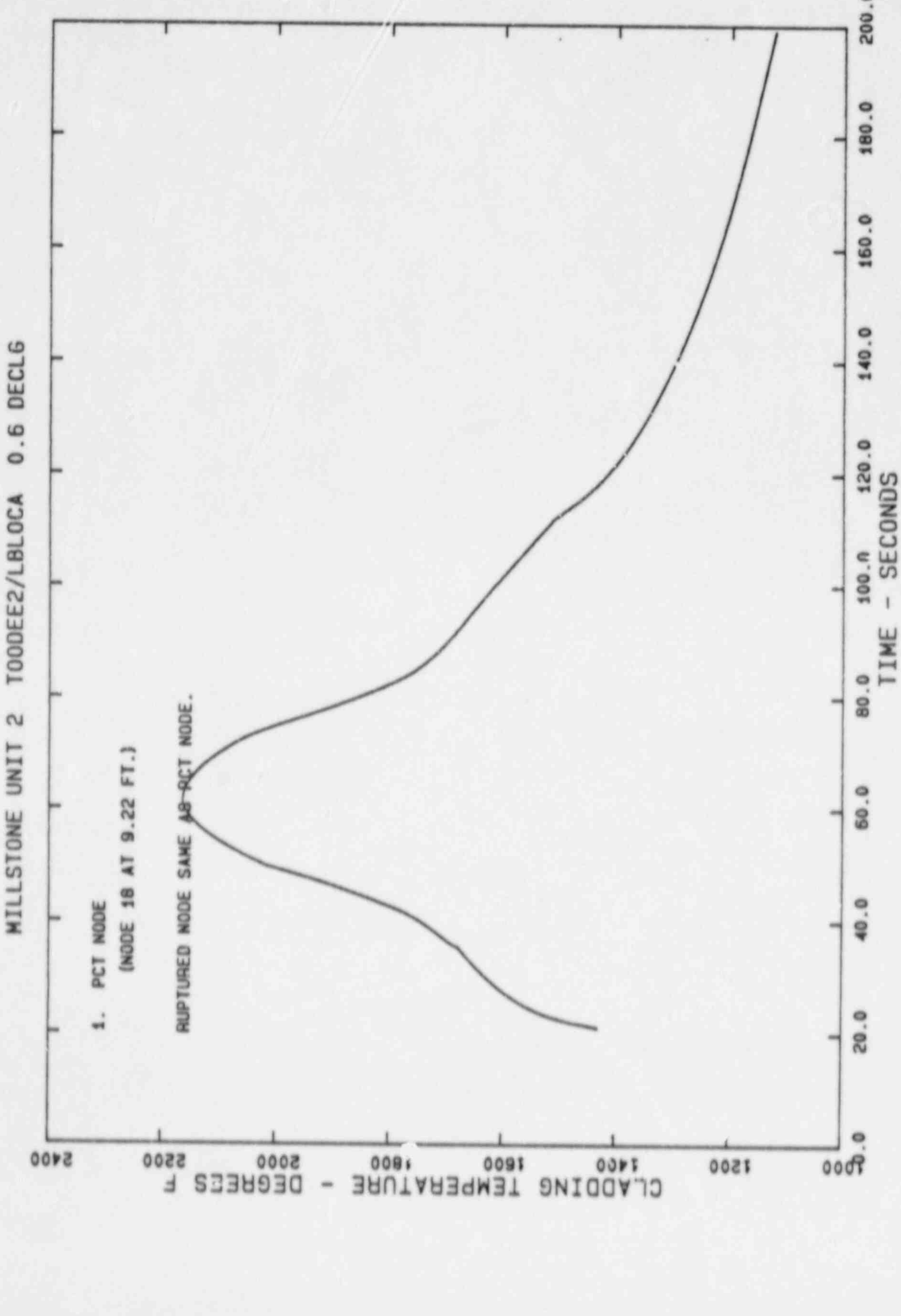

Figure 3.25 PCT Node Cladding Temperature after EOBY, 0.6 DECLG Break, X/L = 0.81

4.0 CONCLUSIONS

The results of the large break LOCA analysis for Millstone Unit 2 showed the 0.6 DECLG break size to be the limiting break with current EXEM/PWR models. The analysis supports operation of Millstone Unit 2 at a power level of 2700 MWt and an average steam generator tube plugging level of 23.5% with a maximum asymmetry of 5.9%. The analysis supports a peak LHR of 15.1 kW/ft with an axial and exposure independent power peaking limit as shown in Figure 2.1. The analysis supports assembly average exposures of up to 52,500 MWd/MTU. The analysis supports full power operation with a reduced primary coolant temperature of up to 12°F. A reduction in primary coolant temperature greater than 12°F is acceptable as long as there is a corresponding decrease in power level. The analysis supports Cycle 10 operation and is intended to support operation for future cycles.

Operation of Millstone Unit 2 with ANF 14x14 fuel at or below the LHR limit shown in Figure 2.1 assures that the NRC acceptance criteria (10 CFR 50.46(b)) for Loss-of-Coolant Accident pipe breaks up to and including the double-ended severance of a reactor coolant pipe will be met with the emergency core cooling system for Millstone Unit 2.

5.0 REFERENCES

(1) Letter, Dennis M. Crutchfield (USNRC Asst. Director division of PWR Licensing-B) to Gary M. Ward (ENC Manager, Reload Licensing), "Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports", dated July 8, 1986.

(2) XN-NF-81-58(P)(A), Revision 1, and Supplements 1-4, "RODEX2: Fuel Rod Thermal Mechanical Response Evaluation Model." Exxon Nuclear Company, February 1983.