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ADVANCED NUCLEAR FUELS CORPORATION

PALISADES LARGE BREAK LOCA/ECCS ANALYSIS WITH INCREASED RADIAL PEAKING

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PALISADES LARGE BREAK LOCA/ECCS

ANALYSIS WITH INCREASED RADIAL PEAKING

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August 1988

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PALISADES LARGE BREAK LOCA/ECCS ANALYSIS WITH INCREASED RADIAL PEAKING

1.0 INTRODUCTION

This document presents the results of a large break loss-of-coolant accident (LOCA) analysis for the Palisades plant operating with Advanced Nuclear Fuels Corporation (ANF) fuel. The primary purpose of the analysis was to support an increase in the total radial peaking factor from 1.77 to 1.83. The analysis was performed at a total radial peaking factor of 1.92 to bound potential future increases in the total radial peaking factor. The analysis supports a maximum LHR of 15.28 kW/ft and a modification in the axial LHR limit curve shown in Figure 3.23-1 of the technical specifications. The analysis also provides justification for removal of Figure 3.23-2 (allowable LHR as a function of burnup) and Figu . 3.23-3 (allowable LHR as a function of peak power location for interior and narrow water gap fuel rods) from the technical specifications. The analysis was performed for the Palisades plant operating at 2581 MWt (2530 MWt plus 2% uncertainty) and a maximum average steam generator tube plugging level of 29.3% with up to 4.5% asymmetry.

Numerous changes have occurred in the ANF LOCA methodology since the previous licensing calculations were performed for the Palisades plant. Therefore, the scope of this analysis includes a mini-break spectrum analysis. Calculations were performed for a 0.4, 0.6, and 0.8 double-ended cold leg guillotine break (DECLG) at the pump discharge to verify the previously determined 0.6 DECLG limiting break size[1]. The analysis also includes calculations at the limiting break size for both a BOC axial power shape peaked at a relative core height of 0.6 and an EOC axial power shape peaked at a relative core height of 0.8. The calculations conservatively used the maximum fuel stored energy near BOC where maximum densification occurs. Justification is provided to support operation with ANF fuel up to a bundle average exposure of 52,500 MWd/MTU with regard to the large break LOCA.

2.0 SUMMARY OF RESULTS

The results of the analysis verified the 0.6 DECLG break as the limiting break size. The analysis demonstrates that the 10 CFR 50.46(b) criteria are satisfied for the Palisades plant with the axially dependent power peaking limit curve shown in Figure 2.1. The analysis supports a maximum LHR of 15.28 kW/ft up to a relative core height of 0.6 and a LHR of 14.75 kW/ft at a relative core height of 0.8. The analysis supports a total radial peaking factor of 1.92 and a maximum average steam generator tube plugging level of 29.3% with up to 4.5% asymmetry. Results of the analysis for both the BOC and EOC axial profiles at the limiting 0.6 DECLG break size are shown in Table 2.1. The peak cladding temperature was calculated to be 1914°F for the BOC profile and 2114°F for the EOC profile. The analysis supports Cycle 8 operation and is intended to support operation for future cycles.

TABLE 2.1 SUMMARY OF RESULTS FOR 0.6 DECLG LIMITING BREAK SIZE

	BOC Stored Energy BOC Axial Shape X/L = 0.6	BOC Stored Energy EOC Axial Shape X/L = 0.9
Peak LHR (kW/ft)	15.28	14.75
Hot Rod Burst		
- Time (Sec)	41.77	41.37
- Elevation (ft)	7.4	8.9
- Channel Blockage Fraction	0.31	0.34
Peak Cladding Temperature		
- Temperature (°F)	1913.7	2114.2
- Time (Sec)	52.47	57.57
- Elevation (ft)	7.4	8.9
Metal-Water Reaction		
- Local Maximum (%)	2.23	4.14
- Elevation of Local Max. (ft)	7.4	8.9
- Hot Pin Total (%)	0.46	0.47
- Core Maximum (%)	<1.0*	<1.0**

* At 350 Seconds

** At 200 Seconds

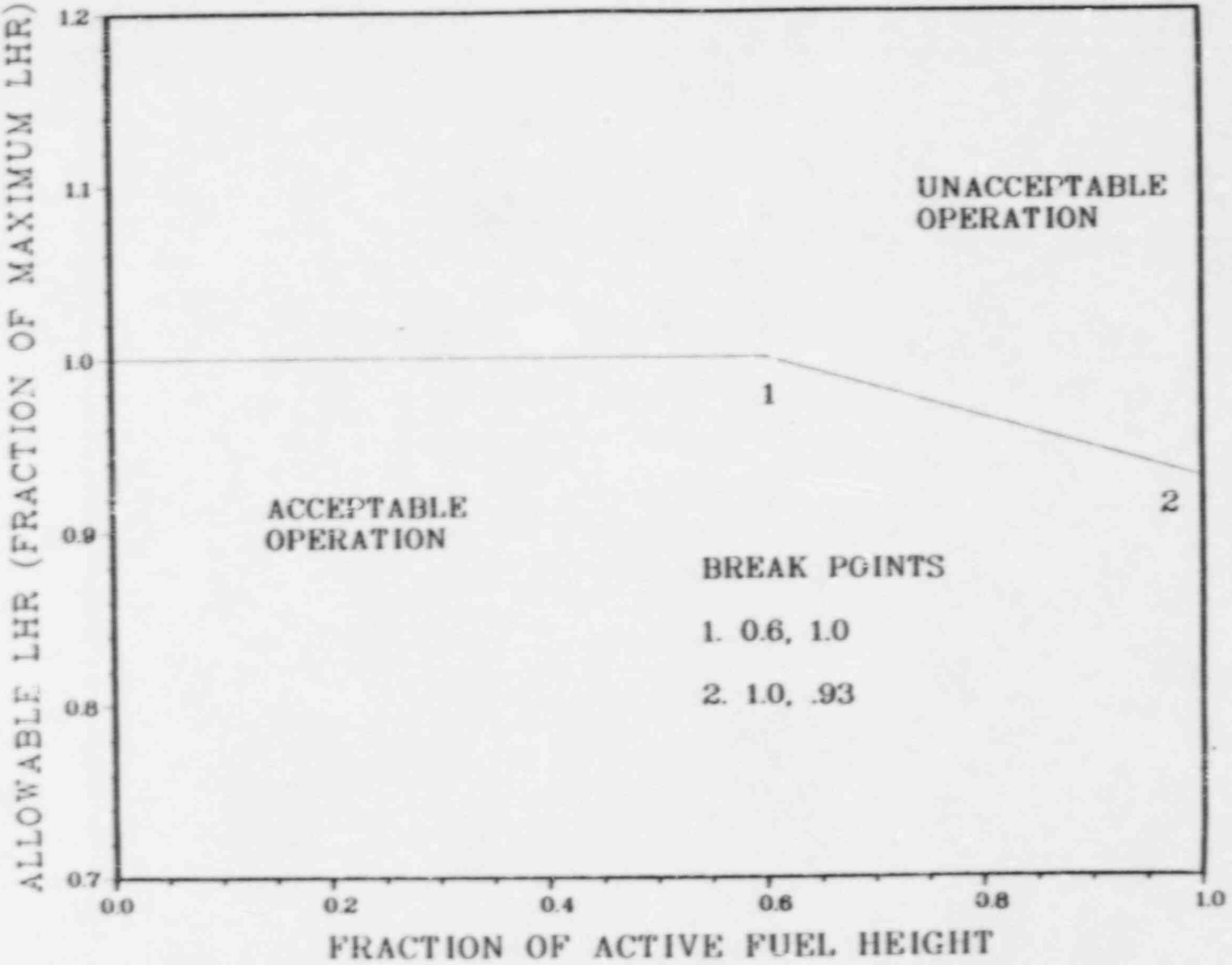

FIGURE 2.1 ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION

3.0 ANALYSIS

Section 3.1 of this report provides a description of the postulated large break loss-of-coolant transient. Section 3.2 describes the methodology and major assumptions used in the analysis. Section 3.3 provides a description of the Palisades plant and a summary of the system, parameters used in the analysis. Section 3.4 provides a summary of the results of the mini-break spectrum calculations. Section 3.5 summarizes the results of the limiting EOC axial power shape and section 3.6 provides justification for the burnup independence of the LHR limit for ANF fuel.

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. The LOCA is assumed to result from an earthquake and is co-incident with loss-of-offsite power. 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 Injection System (HPSI) and Low Pressure Injection System (LPSI) pumps, SIS flow is injected into the cold legs.

Single failure criteria is met by assuming that one HPSI pump and one LPSI pump are not available for operation. When the system pressure falls below the accumulator pressure, flow from the accumulators 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.

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.

3.2 Description of Analytical Models

The ANF EXEM/PWR evaluation model[2] was used to perform the analysis. This evaluation model consists of the following computer codes:

1) RODEX2[3] 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 the EXEM/PWR models.

3.3 Plant Description and Summary of Analysis Parameters

The Palisades 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 was 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 accumulators, a pressurizer, and two steam generators with both primary and secondary sides of the steam generators modeled. The high pressure safety injection (HPSI) and residual heat removal (LPSI) pumps were modeled as fill junctions at the accumulator lines, with conservative flow rates given as a function of system back-pressure. The pump performance curves are 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 29.3% was assumed with an asymmetric steam generator tube plugging of 4.5%. 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 3.3.1.

3.4 Break Spectrum Results

A mini-break spectrum study was performed to confirm the previously determined 0.6 DECLG break as the limiting break size since numerous changes have occurred in the ANF LOCA methodology since the previous licensing calculations were performed for the Palisades Plant. Calculations were performed for 0.4, 0.6, and 0.8 DECLG break sizes with an axial power shape peaked at a relative core height of 0.6. Also, ANF methodology previously and currently shows that split breaks are less limiting the guillotine breaks. Therefore, split break calculations were not included in this analysis. System blowdown calculations were first performed to the end-of-bypass (EOBY) to confirm the 0.6 DECLG as the limiting break size. Fuel and cladding temperatures between the 0.4 and 0.6 DECLG break sizes were fairly close at the end-of-bypass such that it was not conclusive that the 0.6 DECLG break was the limiting break. Therefore, calculations were performed through the refill and reflood periods for these two break sizes. The results of the break spectrum study are shown in Table 3.4.1. The 0.6 DECLG break size is confirmed as 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 1914 °F. Thus, a maximum LHR of 15.28 kW/ft is supported up to a relative core height of 0.6. Calculated event times for the 0.4 DECLG break are shown in Table 3.4.2. Calculated event times for the 0.6 DECLG break are shown in Table 3.4.3.

3.5 Axial Shape Study Results

An EOC (top-skewed) axial power shape was analyzed to define the axially dependent LHR limit curve shown in Figure 2.1. The axial power shape was peaked at a relative core height of 0.8 with an LHR of 14.75 kW/ft. The axial shape was selected from those axial shapes allowed by T_{inlet} LCO barn. A BOC fuel stored energy was conservatively used in conjunction with this axial shape. The results for the EOC shape are shown in Table 2.1. The PCT was calculated to be 2114°F. Plots of parameters depicting calculations for the limiting 0.6 DECLG break and the EOC shape are shown in Figures 3.5.1 through 3.5.26.

3.6 Exposure Limits

The results of previous exposure analyses for the Palisades plant(1) required a reduction in the LHR limit at high exposures. This was a result of the use of the previous ANF fuel rod code, GAPEX. Exposure calculations have been performed with the current EXEM/PWR methodology using RODEX2 for two plants with a maximum bundle average exposure of 52,500 MWd/MTU. The current ANF methodology predicts maximum fuel storage energy to occur near BOC where maximum densification occurs. Closure of the fuel-cladding gap at higher exposures significantly reduces the fuel stored energy. At high exposures, gap closure significantly outweighs the effect of higher concentrations of fission gases which tend to reduce the gap conductance and increase fuel stored energy. Also, the reduced stored energy at high exposures outweighs any adverse effects of increased rod internal pressure at high exposures. Thus, the peak cladding temperature will be lower at high exposures than for the limiting case reported in Section 3.5 which assumes a BOC fuel stored energy. Since this phenomena is fuel related rather than system related, the exposure study results for other plants are applicable to the Palisades plant. Thus, the LHR limit is independent of exposure up to a maximum bundle average exposure of 52,500 MWd/MTU.

TABLE 3.3.1 PALISADES SYSTEM ANALYSIS PARAMETERS

Primary Heat Output, MWt	2530*
Primary Coolant Flow Rate, lbm/hr	1.203×10^8 (318,770 gpm)
Primary Coolant System Volume, ft^3	8808**
Operating Pressure, psia	2060
Inlet Coolant Temperature (hottest loop), °F	544
Reactor Vessel Volume, ft^3	4782
Pressurizer Total Volume, ft^3	1504
Pressurizer Liquid Total, ft^3	803
Accumulator Total Volume, ft^3 (one of four)	2011
Accumulator Liquid Volume, ft^3	1116
Accumulator Pressure, psia	215
Accumulator Fluid Temperature, °F	90
Total Number of Tubes per Steam Generator	8519
Steam Generator Tube Plugging	33.8 - 24.8 % split
Number of Tubes Plugged (33.8 % SGTP)	2878
Number of Tubes Plugged (24.8 % SGTP)	2114
Steam Generator Secondary Side Heat Transfer Area, 33.8% SGTP, ft^2	48,661
Steam Generator Secondary Side Heat Transfer Area, 24.8% SGTP, ft^2	55,245
Steam Generator Secondary Flow Rate, lbm/hr (47-53% power split)	5.241×10^6 (33.8% SGTP) 5.949×10^6 (24.8% SGTP)
Steam Generator Secondary Pressure, psia	730
Steam Generator Feedwater Enthalpy, Btu/lbm	414

* Primary Heat Output used in RELAP4-EM Model - 1.02 x 2530 = 2580.6 MWt.

** Includes pressurizer total volume and 29.3% average SGTP.

TABLE 3.3.1 PALISADES SYSTEM ANALYSIS PARAMETERS (CONTINUED)

Reactor Coolant Pump Rated Head, ft	260
Reactor Coolant Pump Rated Torque, ft-lbf	32,530
Reactor Coolant Pump Rated Speed, rpm	880
Reactor Coolant Pump Moment of Inertia, lbm-ft^2	98,000
Containment Volume, ft^3	1.64×10^6
Containment Temperature, °F	90
SIS Fluid Temperature, °F	70
HPSI Delay Time, Sec.	27.0
LPSI Delay Time, Sec.	28.0

TABLE 3.4.1 PALISADES BREAK SPECTRUM ANALYSIS RESULTS

	DECLG 0.4 X/L = 0.6	DECLG 0.6 X/L = 0.6	DECLG 0.8 X/L = 0.6
Peak LHR (kW/ft)	15.28	15.28	15.28
EOBY Time (Sec)	24.77	19.17	16.74
Fuel Average Temperature at EOBY (°F)	1424.3	1429.1	1386.8
Cladding Temperature at EOBY (°F)	1250.9	1245.1	1176.8
Hot Rod Burst			
- Time (Sec)	48.77	41.77	
- Elevation (ft)	7.4	7.4	
- Channel Blockage Fraction	0.32	0.32	
Peak Cladding Temperature			
- Temperature (°F)	1851.0	1913.7	
- Time (Sec)	57.57	52.47	
- Elevation (ft)	7.4	7.4	
Metal Water Reaction			
- Local Maximum, (%)	1.93	2.23	
- Elevation of Local Max. (ft)	7.4	7.4	
- Hot Pin Total (%)	0.43	0.46	
- Core Maximum (%)	<1.0*	<1.0*	

* At 350 Sec

TABLE 3.4.2 CALCULATED EVENT TIMES FOR 0.4 DECLG BREAK

Event	Time (Sec.)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.81
Pressurizer Empties	12.6
Accumulator Injection Begins, Broken Loop	18.5
Accumulator Injection Begins, Single Intact Loop	20.3
Accumulator Injection Begins, Double Intact Loop	20.3
End-of-Bypass (EOBY)	24.77
Start of Reflood	44.41
Peak Cladding Temperature is Reached (X/L = 0.6)	57.57
Accumulators Empty, Broken Loop	74.74
Accumulators Empty, Single Intact Loop	77.41
Accumulators Empty, Double Intact Loop	78.35

TABLE 3.4.3 CALCULATED EVENT TIMES FOR 0.6 DECLG BREAK

Event	Time (Sec.)
Start	0.0
Break is Fully Open	0.05
Safety Injection Signal	0.62
Accumulator Injection Begins, Broken Loop	11.70
Pressurizer Empties	12.26
Accumulator Injection Begins, Single Intact Loop	15.65
Accumulator Injection Begins, Double Intact Loop	15.65
End-of-Bypass (EOBY)	19.17
Start of Reflood	27.70
Peak Cladding Temperature is Reached (X/L = 0.6)	52.47
Peak Cladding Temperature is Reached (X/L = 0.8)	57.57
Accumulators Empty, Broken Loop	68.9
Accumulators Empty, Single Intact Loop	72.85
Accumulators Empty, Double Intact Loop	73.75

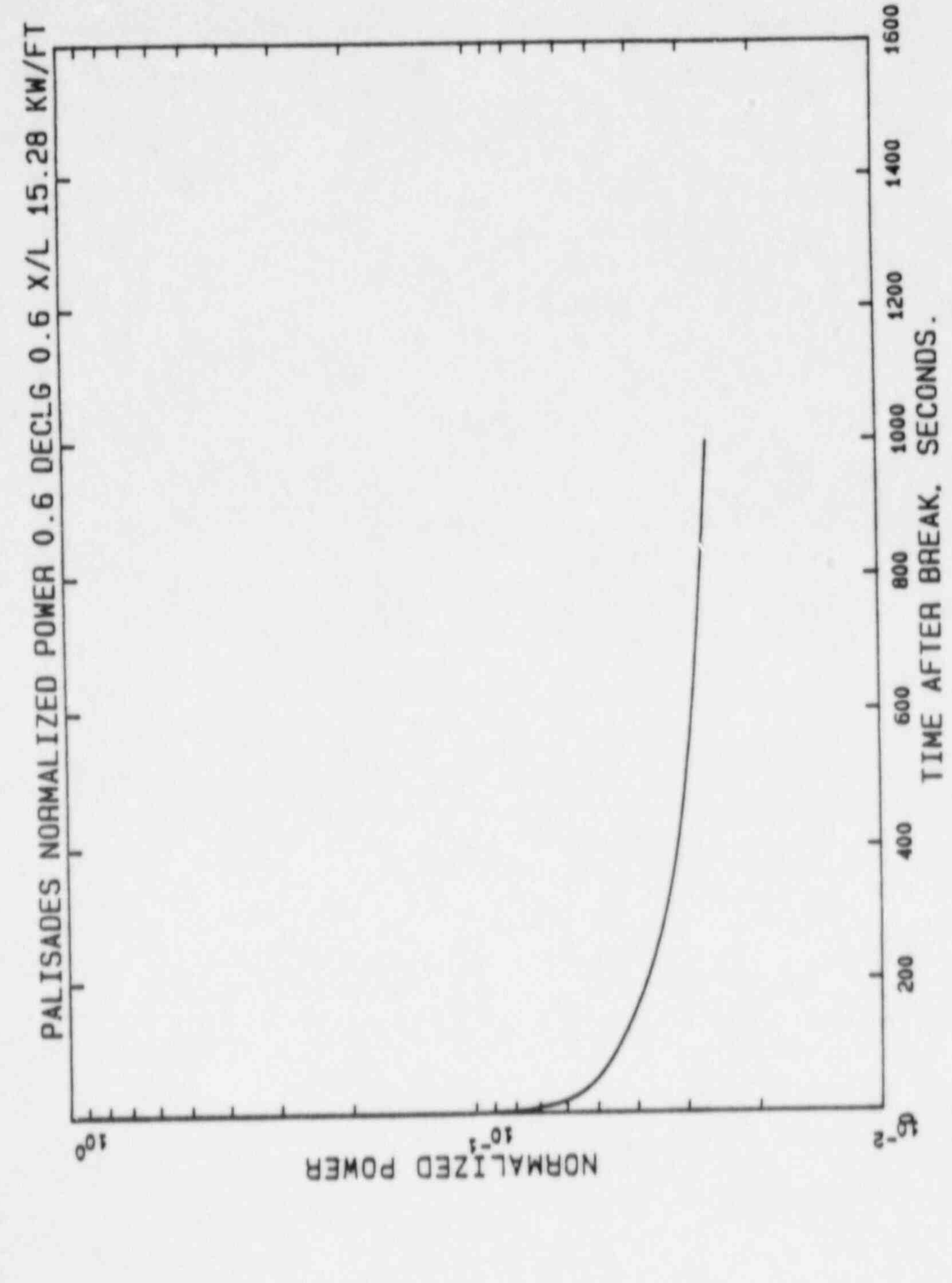

FIGURE 3.5.1 NORMALIZED POWER (EOC), 0.6 DECLG BREAK

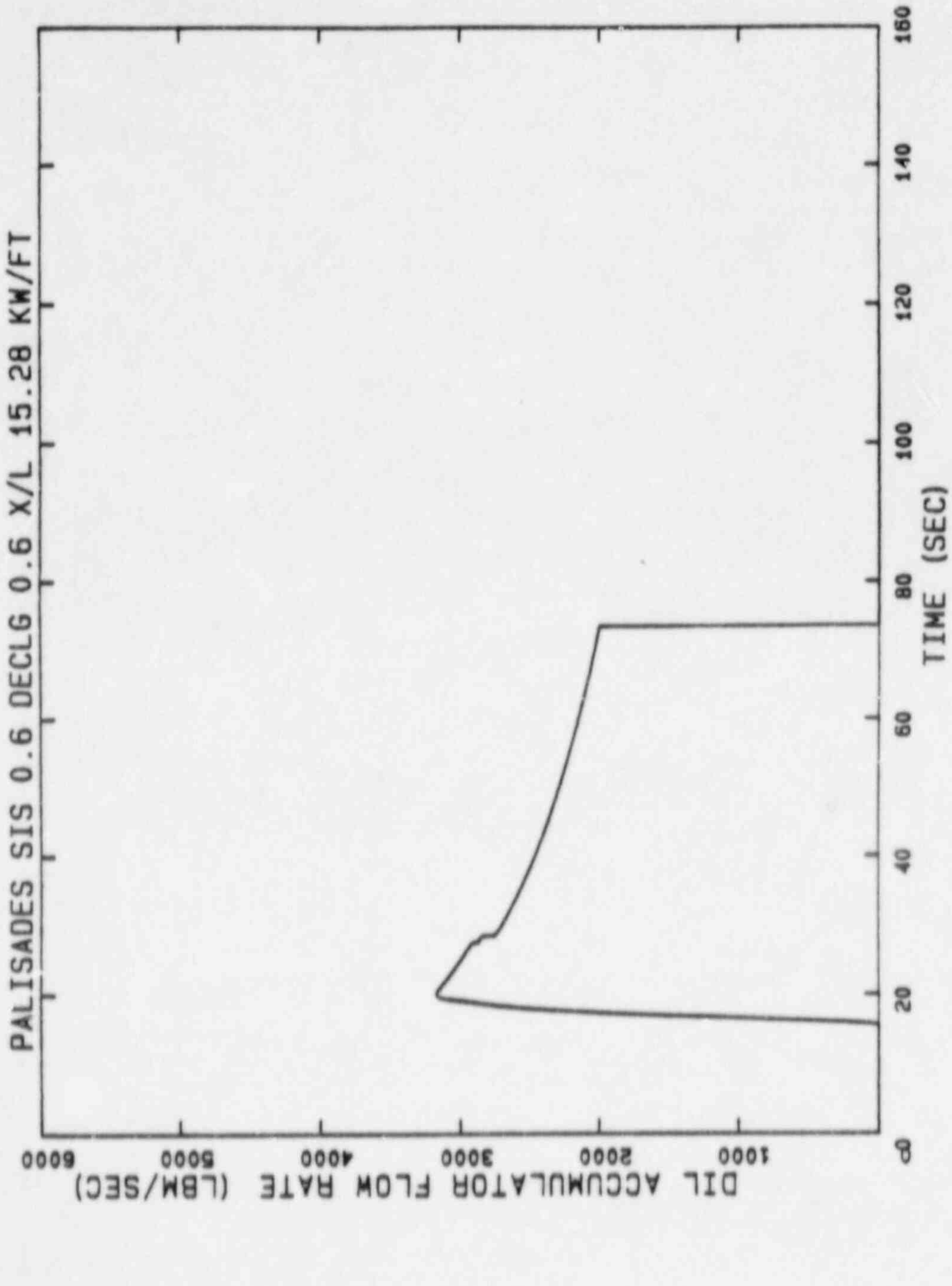

FIGURE 3.5.2 DOUBLE INTACT LOOP ACCUMULATOR FLOW RATE, 0.6 DECLG BREAK

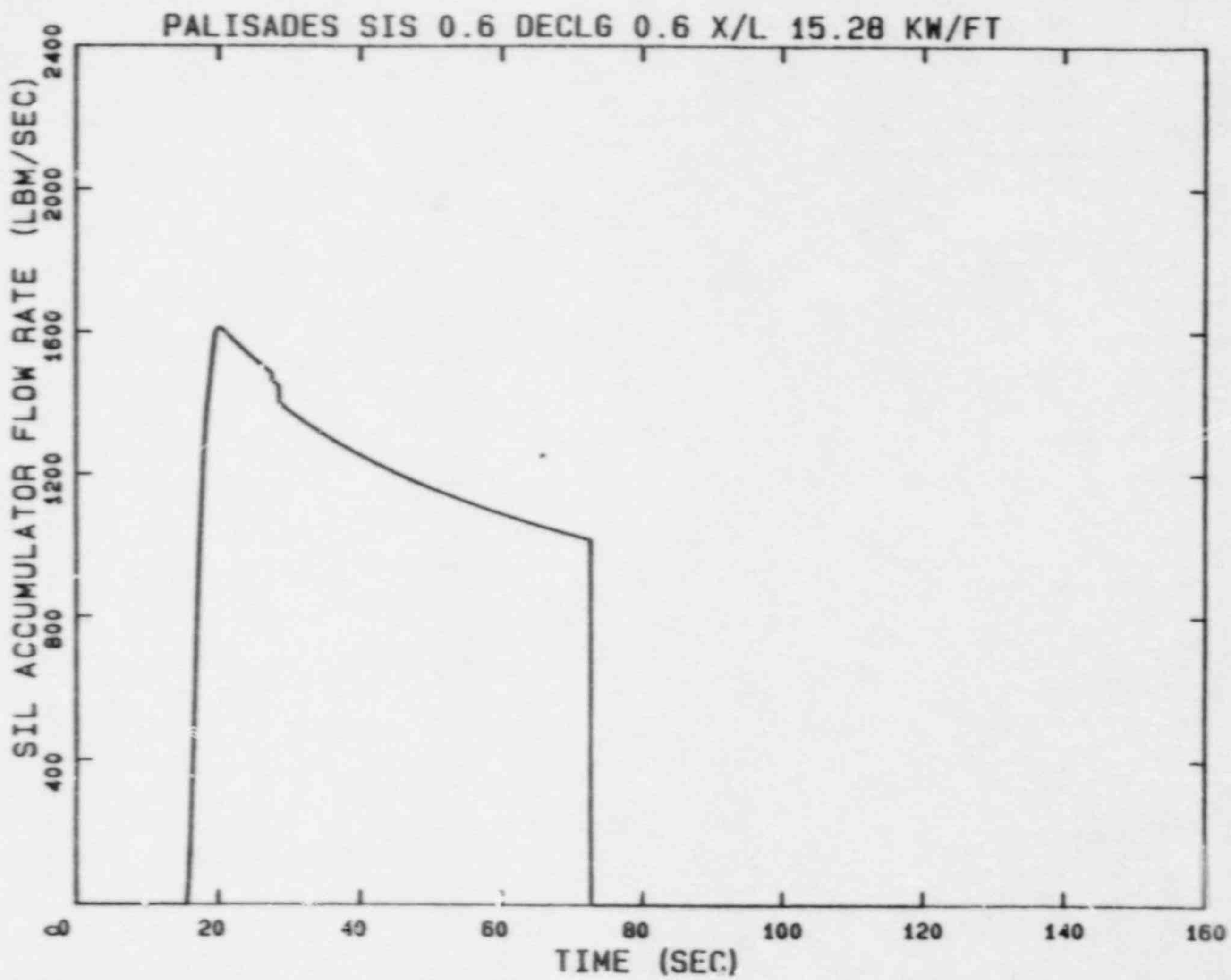

FIGURE 3.5.3 SINGLE INTACT LOOP ACCUMULATOR FLOW RATE, 0.6 DECLG BREAK

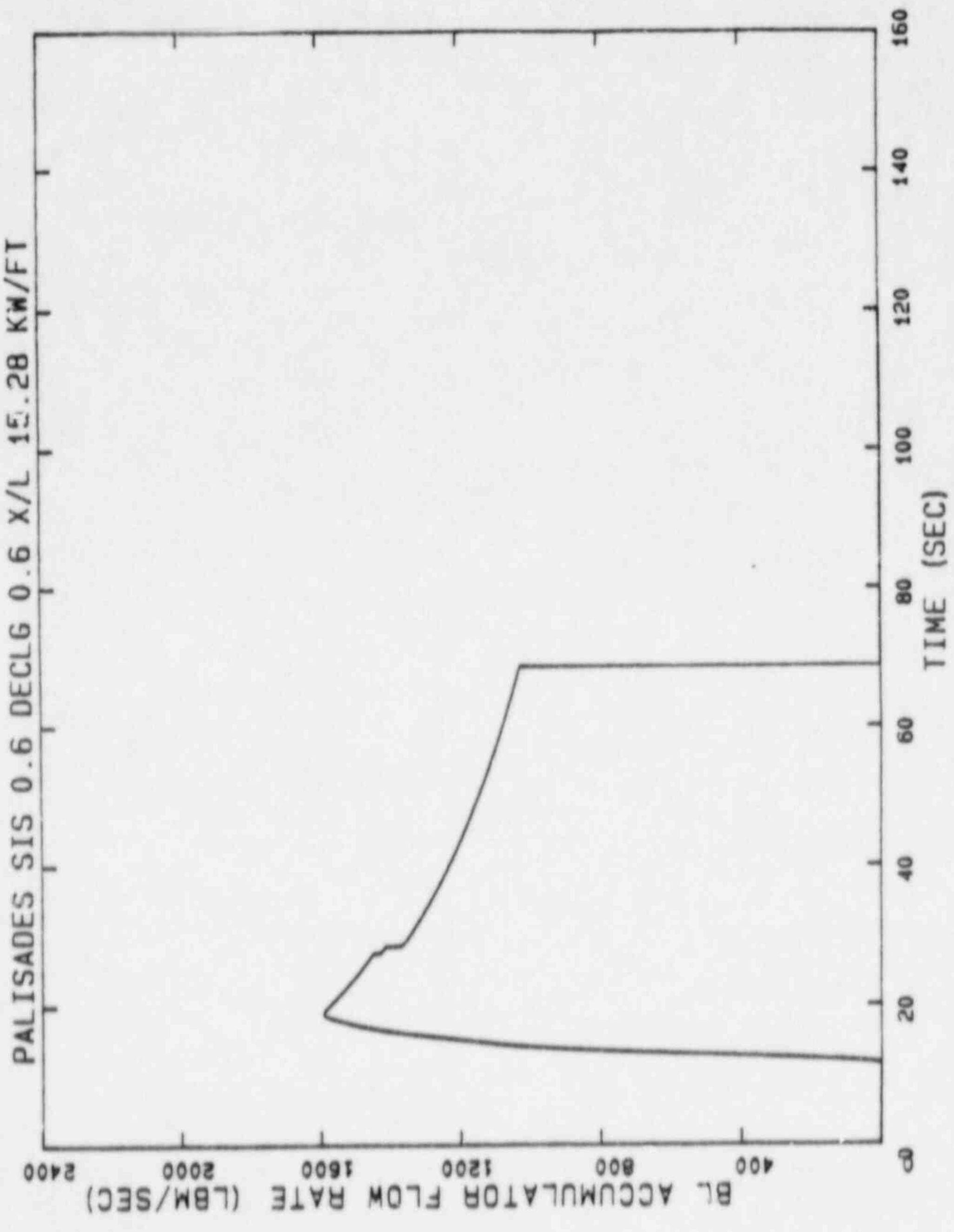

FIGURE 3.5.4 BROKEN LOOP ACCUMULATOR FLOW RATE, 0.6 DECLG BREAK

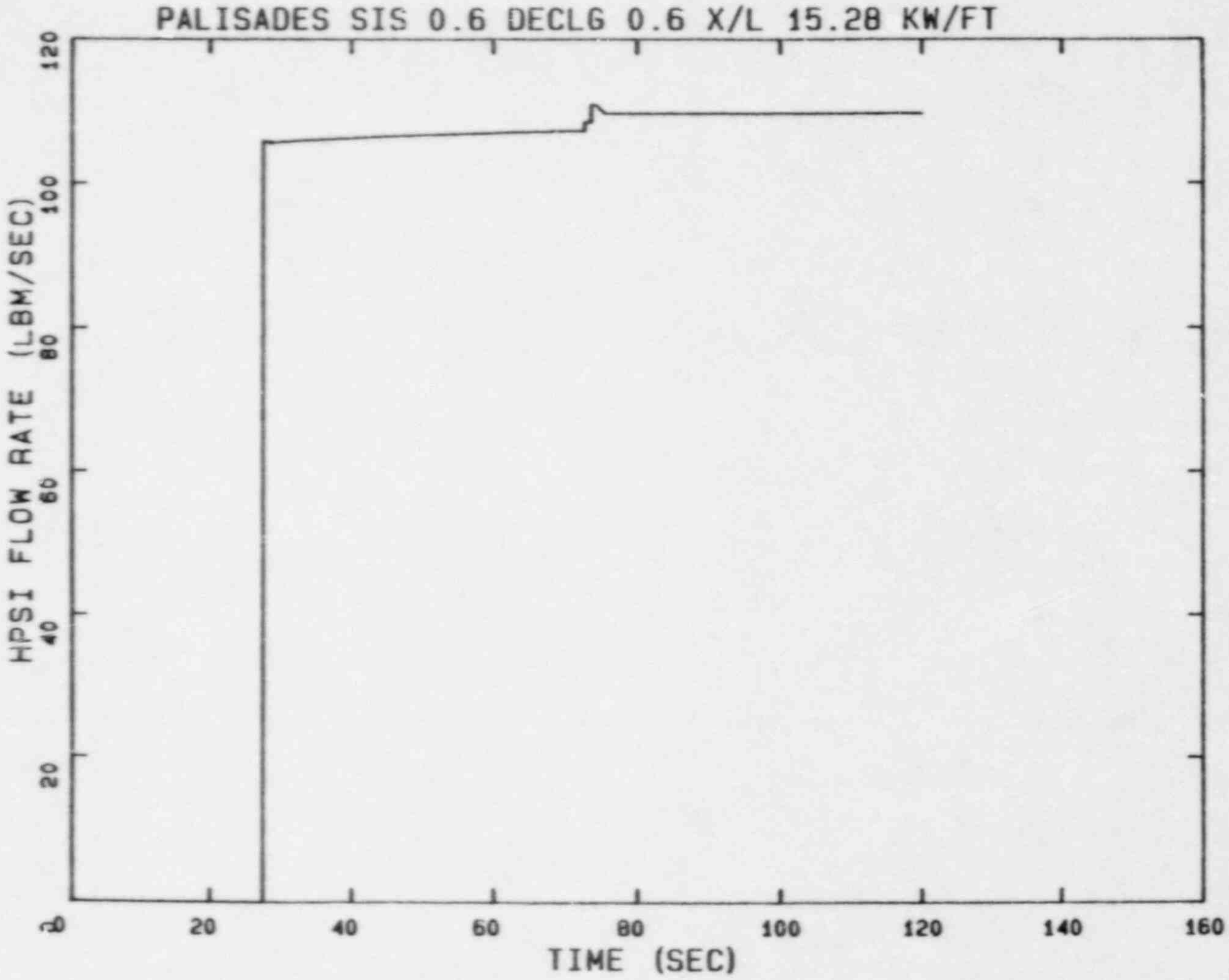

FIGURE 3.5.5 TOTAL INTACT LOOP HPSI FLOW RATE, 0.6 DECLG BREAK

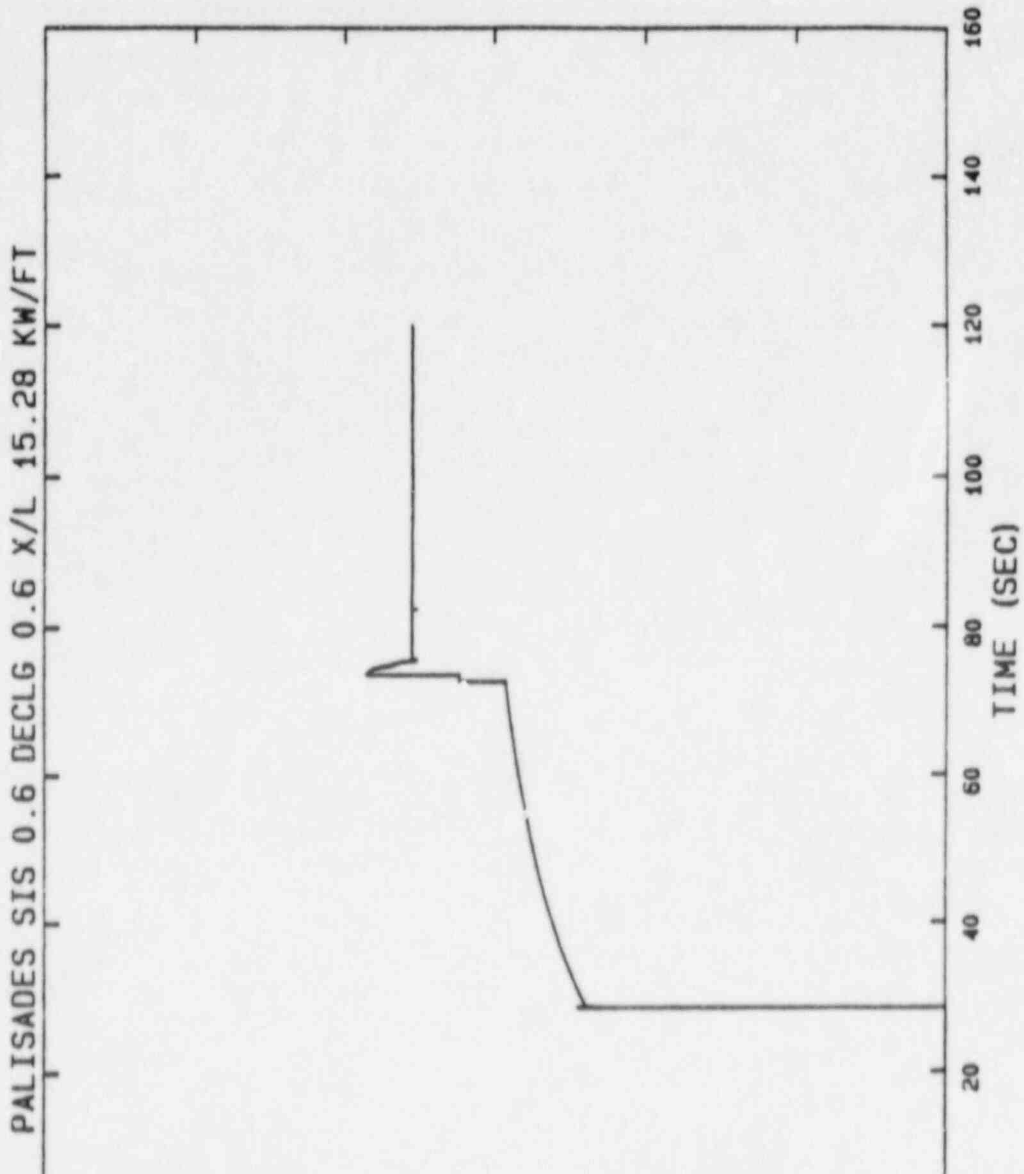

FIGURE 3.5.6 TOTAL INTACT LOOP LPSI FLOW RATE, 0.6 DECLG BREAK

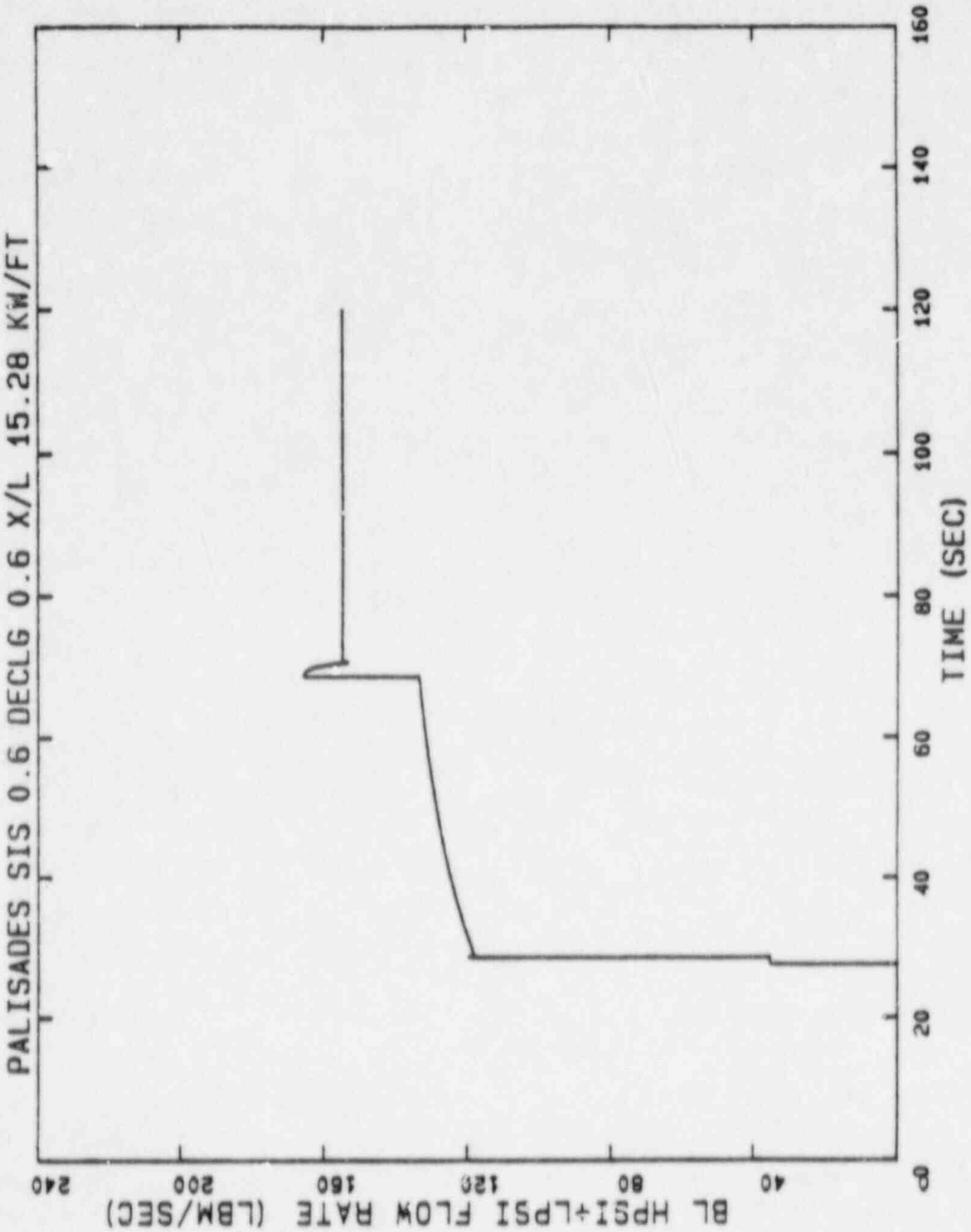

FIGURE 3.5.7 BROKEN LOOP SIS FLOW RATE, 0.6 DECLG BREAK

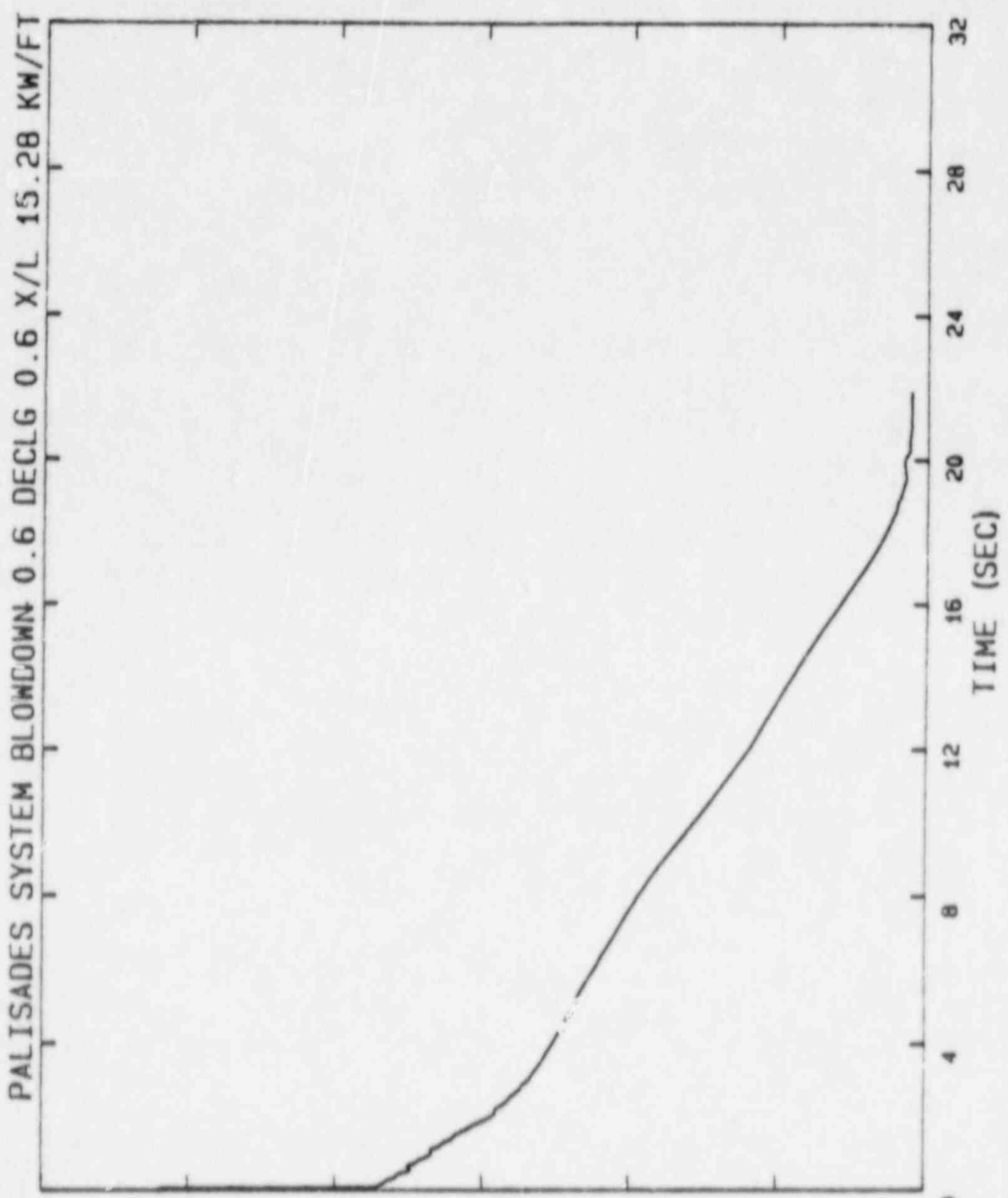

FIGURE 3.5.8 UPPER PLENUM PRESSURE DURING BLOWDOWN, 0.6 DECLG BREAK

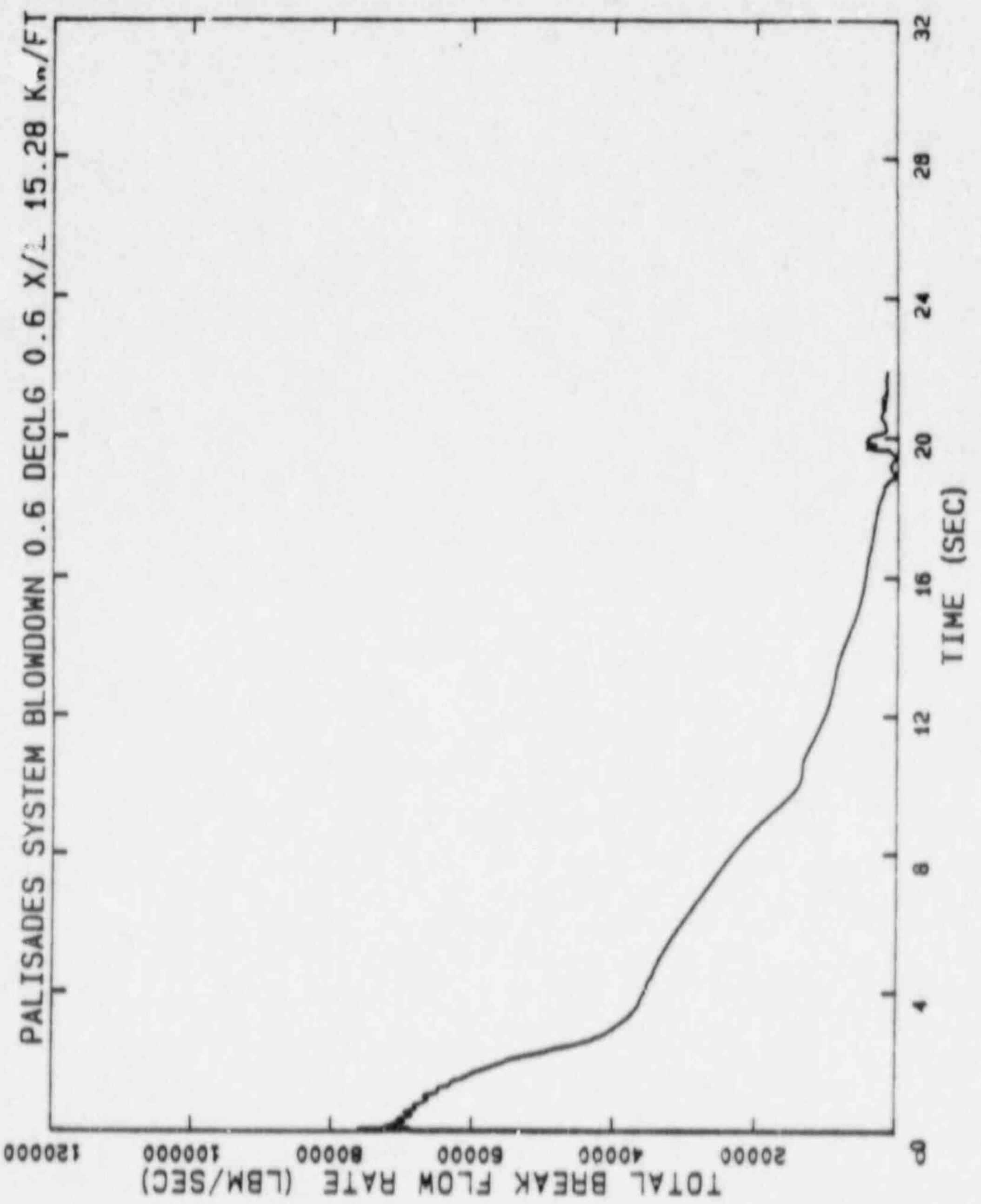

FIGURE 3.5.9 TOTAL BREAK FLOW RATE DURING BLOWDOWN, 0.6 BREAK

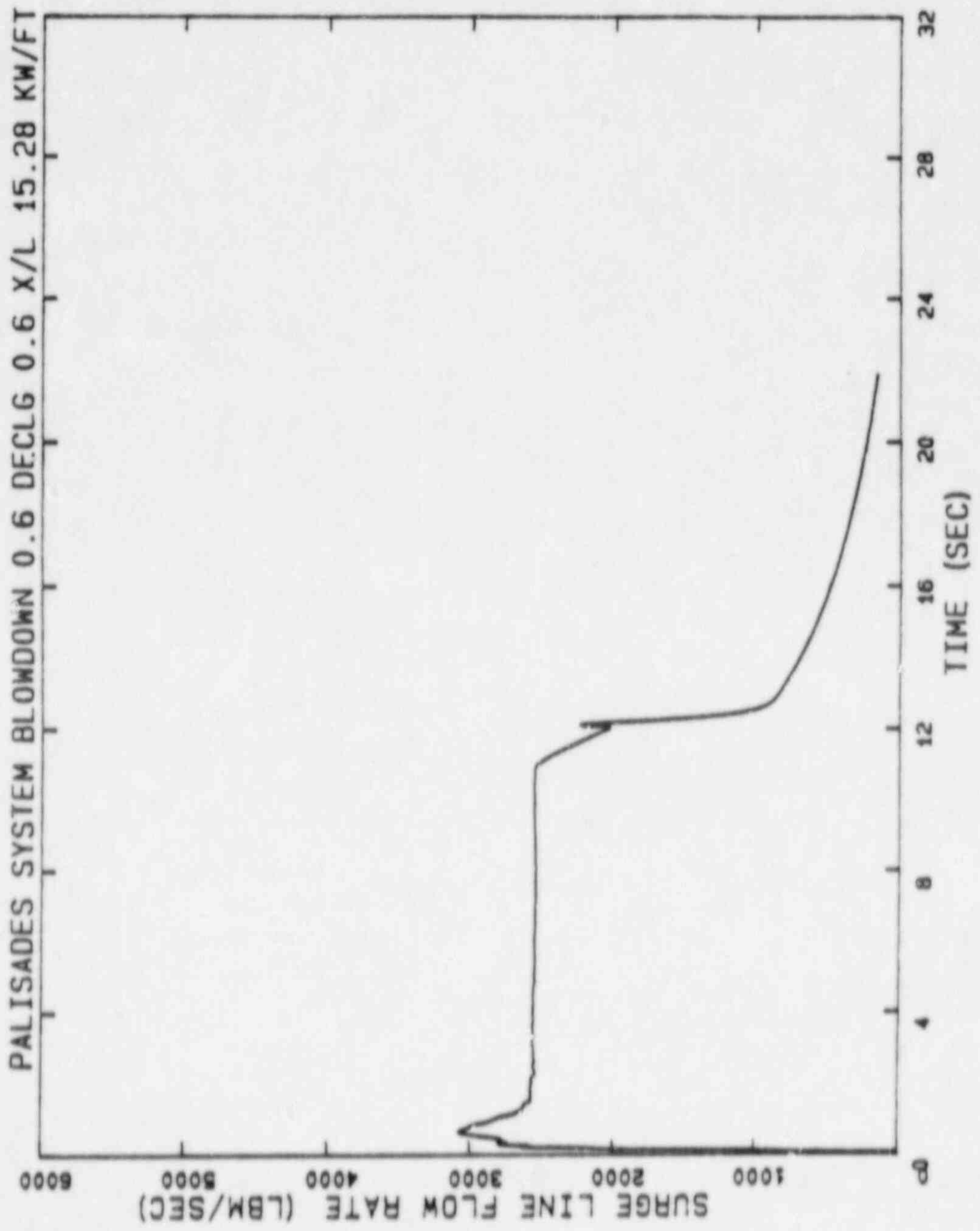

FIGURE 3.5.10 PRESSURIZER SURGE LINE FLOW RATE DURING BLOWDOWN, 0.6 DECLG BREAK

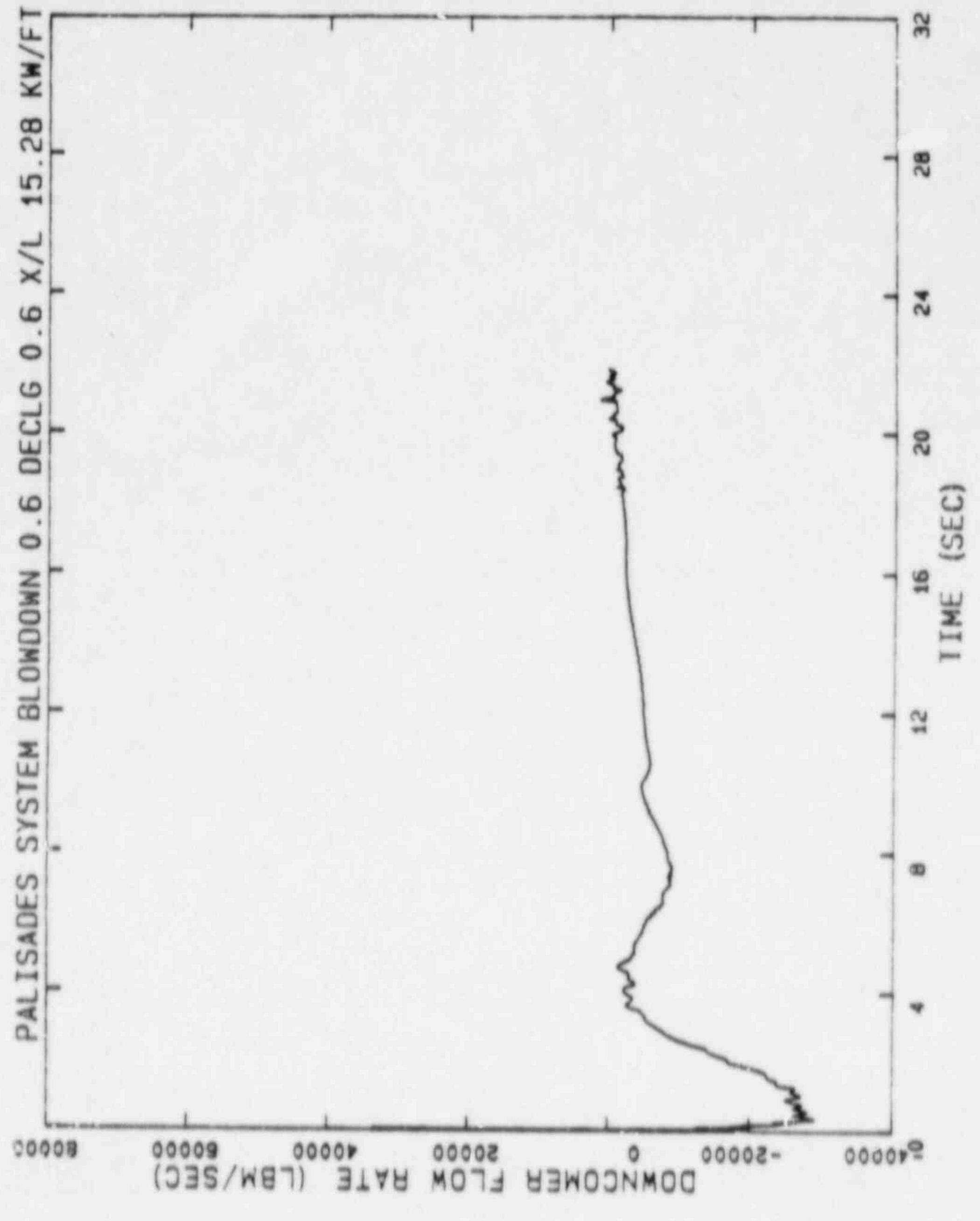

FIGURE 3.5.11 DOWNCOMER FLOW RATE DURING BLOWDOWN, 0.6 DECLG BREAK

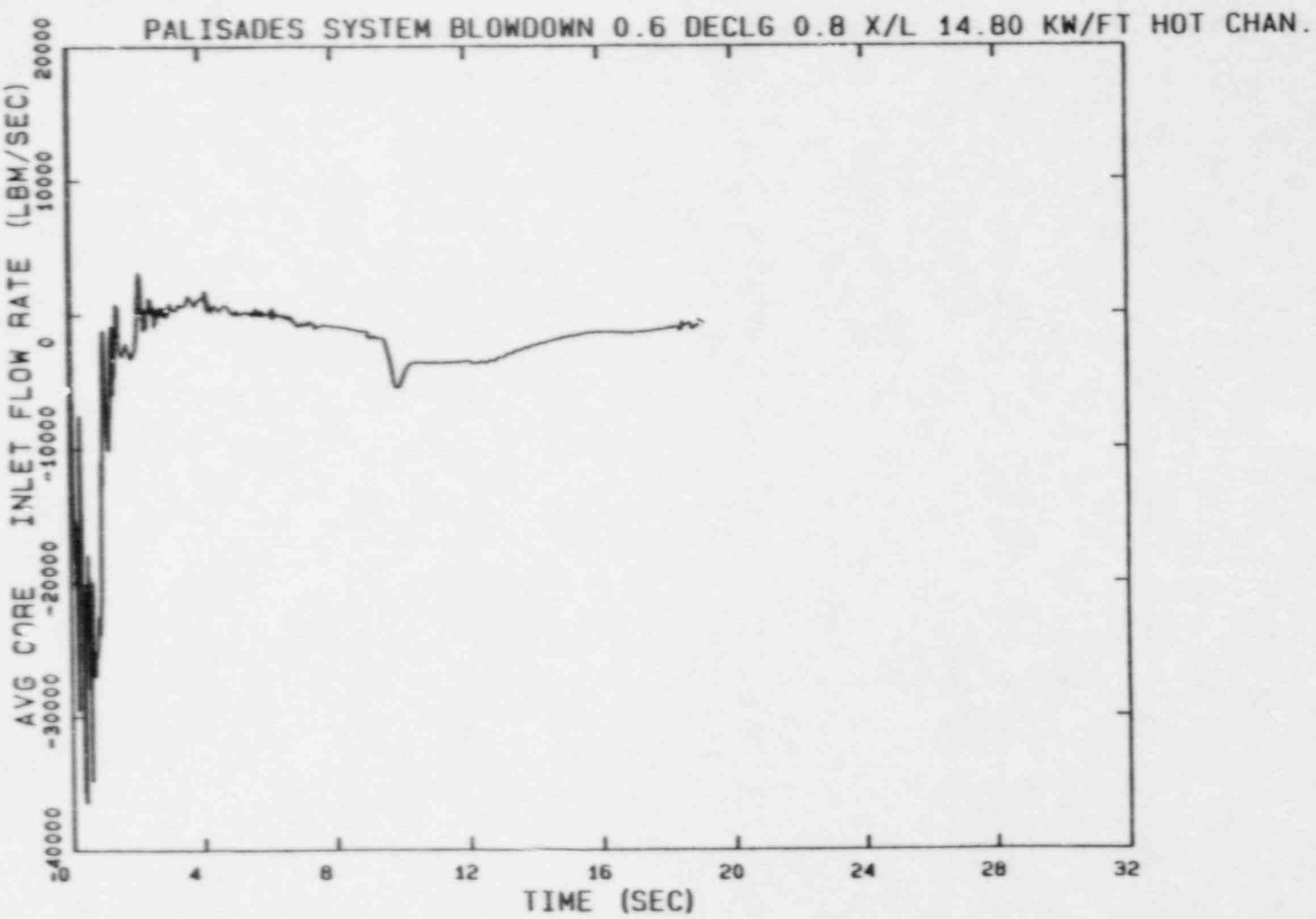

FIGURE 3.5.12 AVERAGE CORE INLET FLOW RATE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L = 0.8

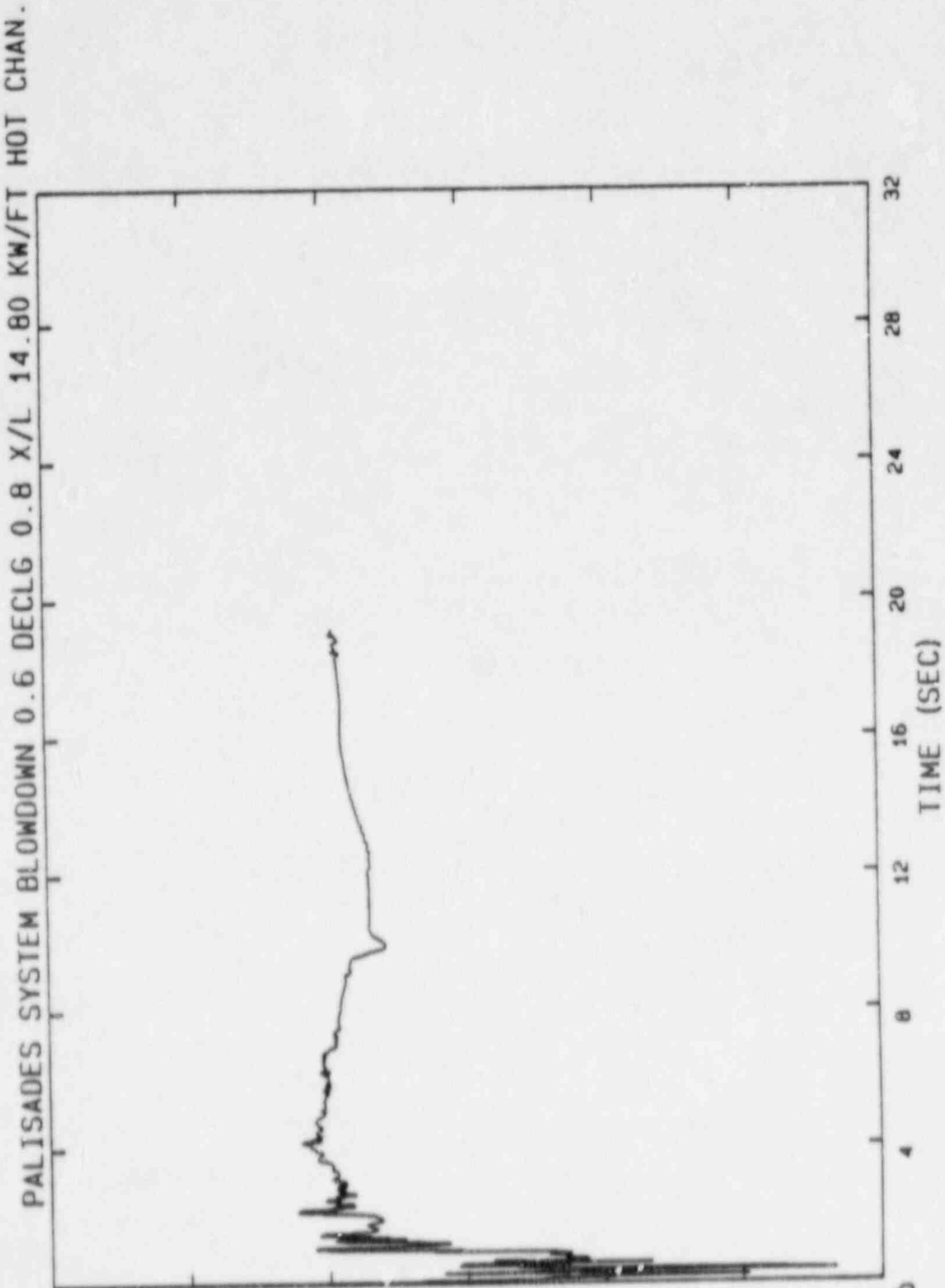

FIGURE 3.5.13 HOT CHANNEL INLET FLOW RATE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L - 0.8

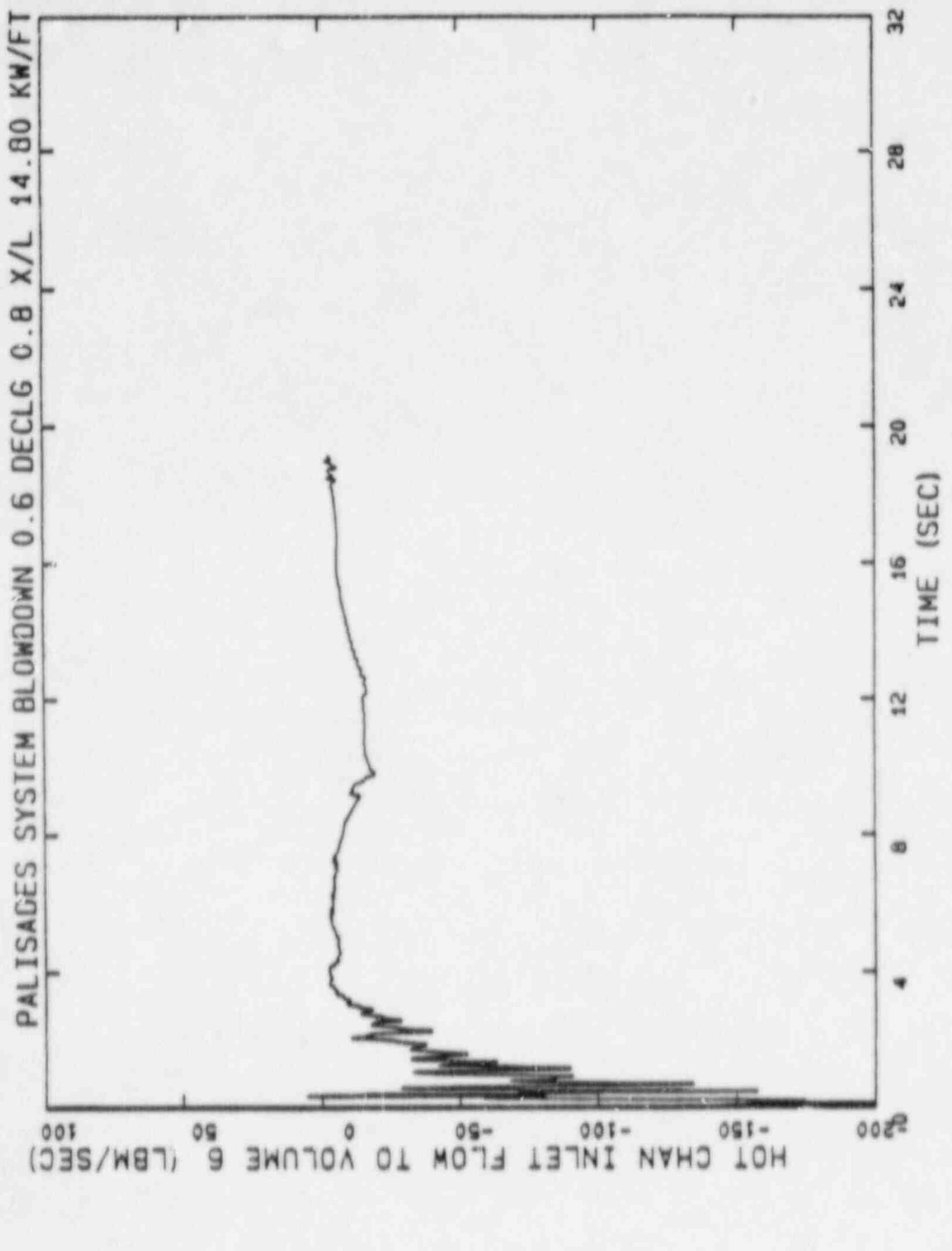

FIGURE 3.5.14 HOT VOLUME INLET FLOW RATE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L - 0.8

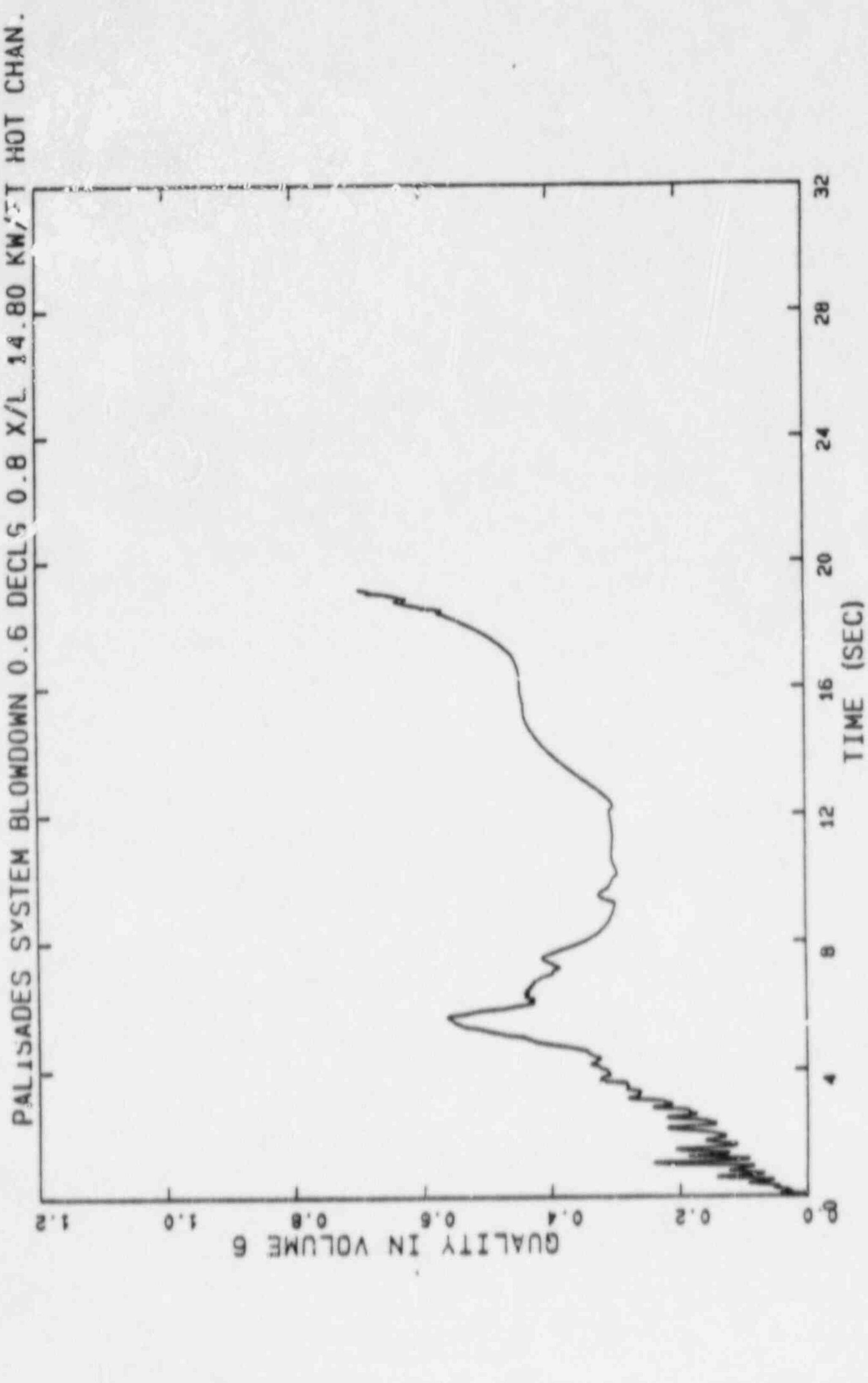

FIGURE 3.[illegible].15 HOT NODE FLUID QUALITY DURING BLOWDOWN, 0.6 DECLG BREAK, X/L = 0.8

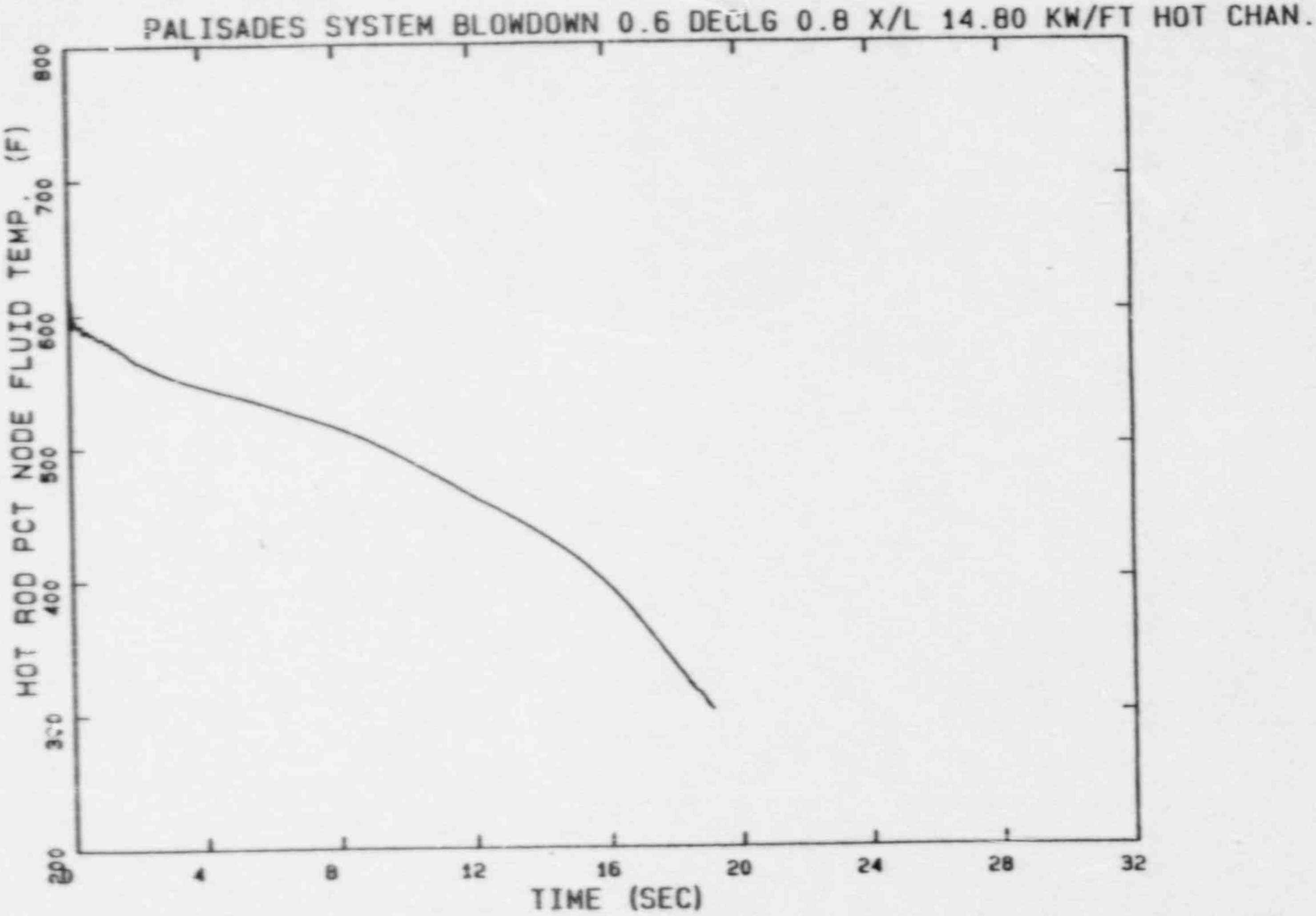

FIGURE 3.5.16 PCT NODE FLUID TEMPERATURE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L - 0.8

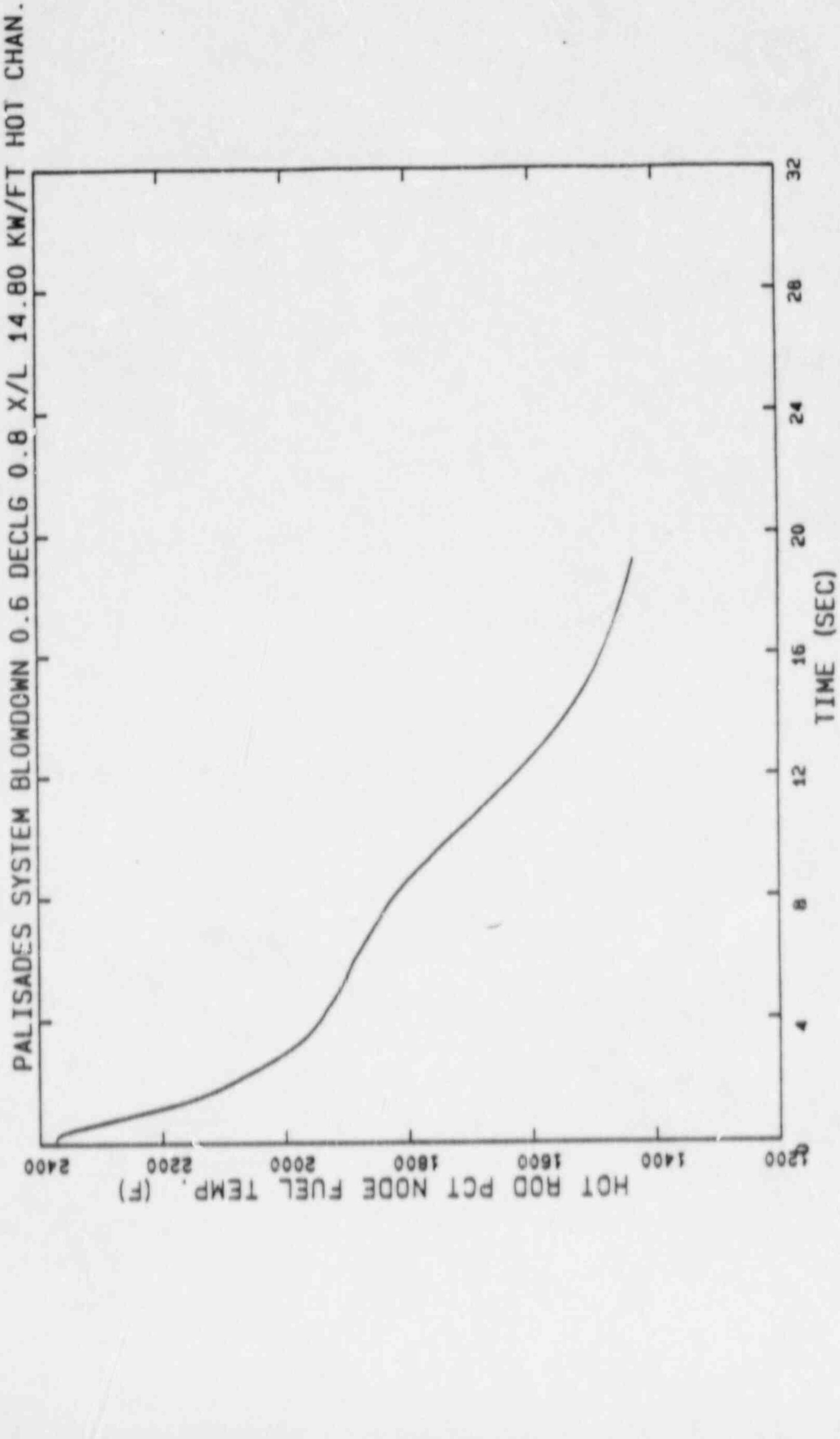

FIGURE 3.5.17 PCT NODE FUEL AVERAGE TEMPERATURE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L = 0.8

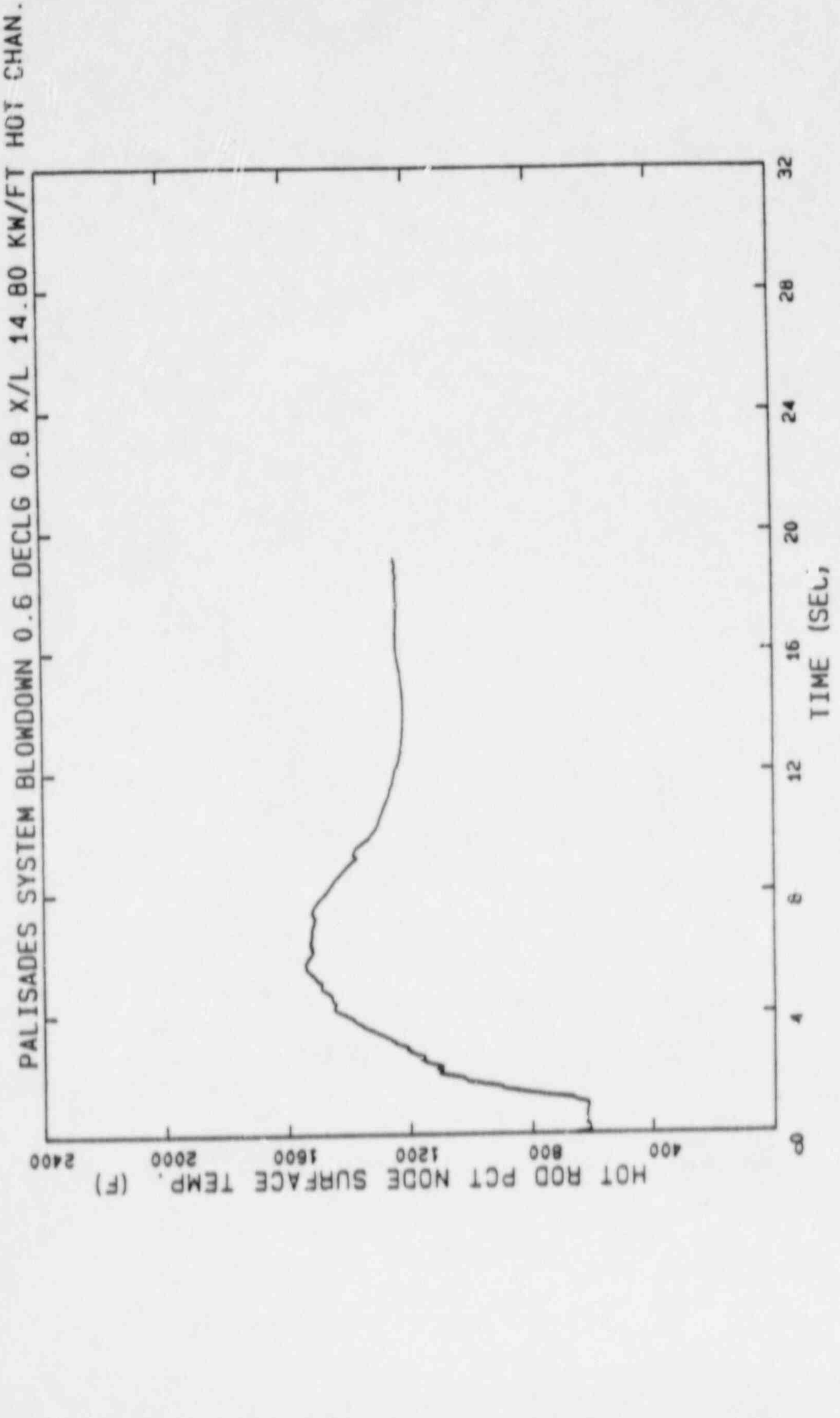

FIGURE 3.5.18 PCT NODE CLADDING TEMPERATURE DURING BLOWDOWN, 0.6 DECLG BREAK, X/L = 0.8

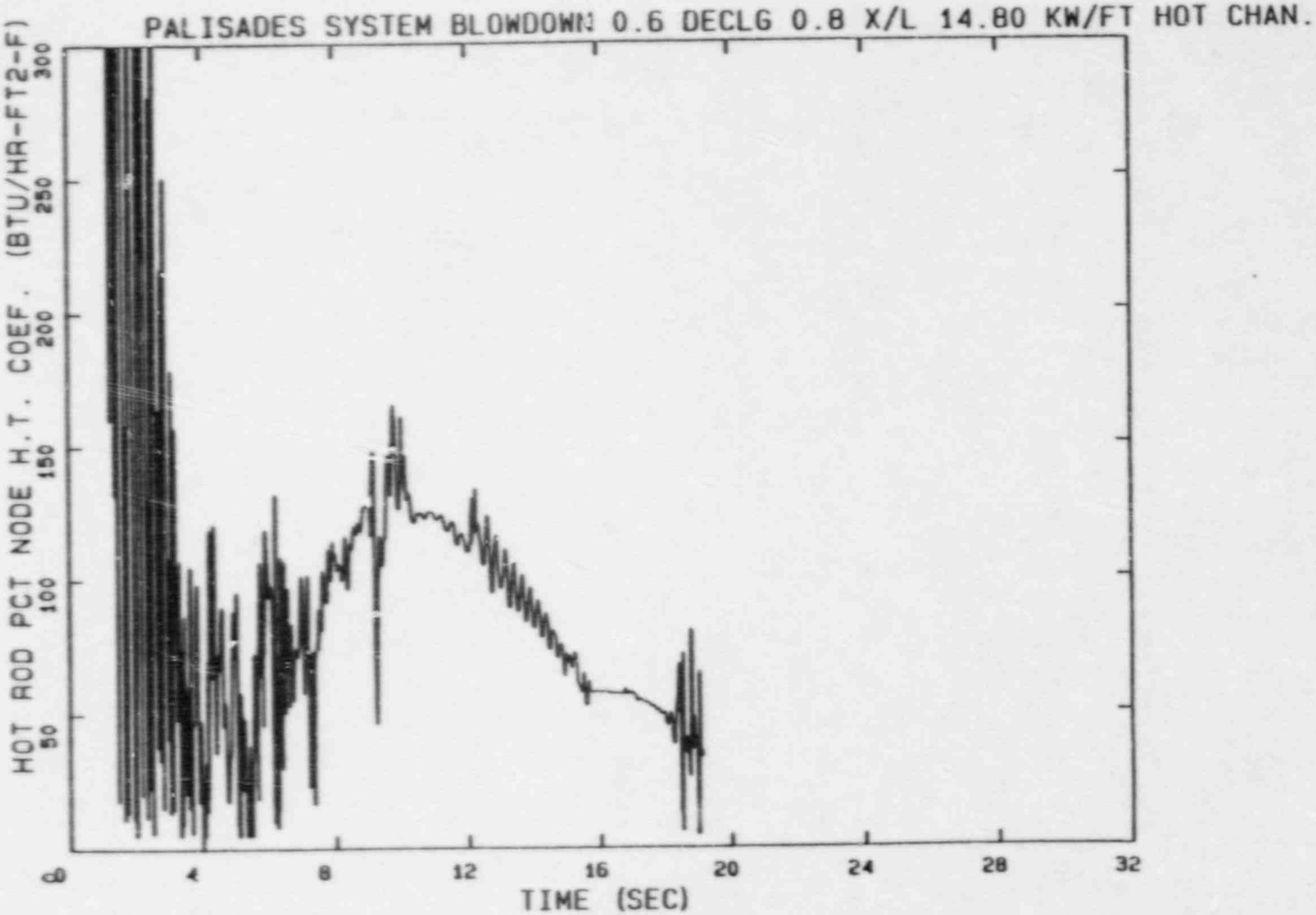

FIGURE 3.5.19 PCT NODE HEAT TRANSFER COEFFICIENT DURING BLOWDOWN, 0.6 DECLG BREAK, X/L - 0.8

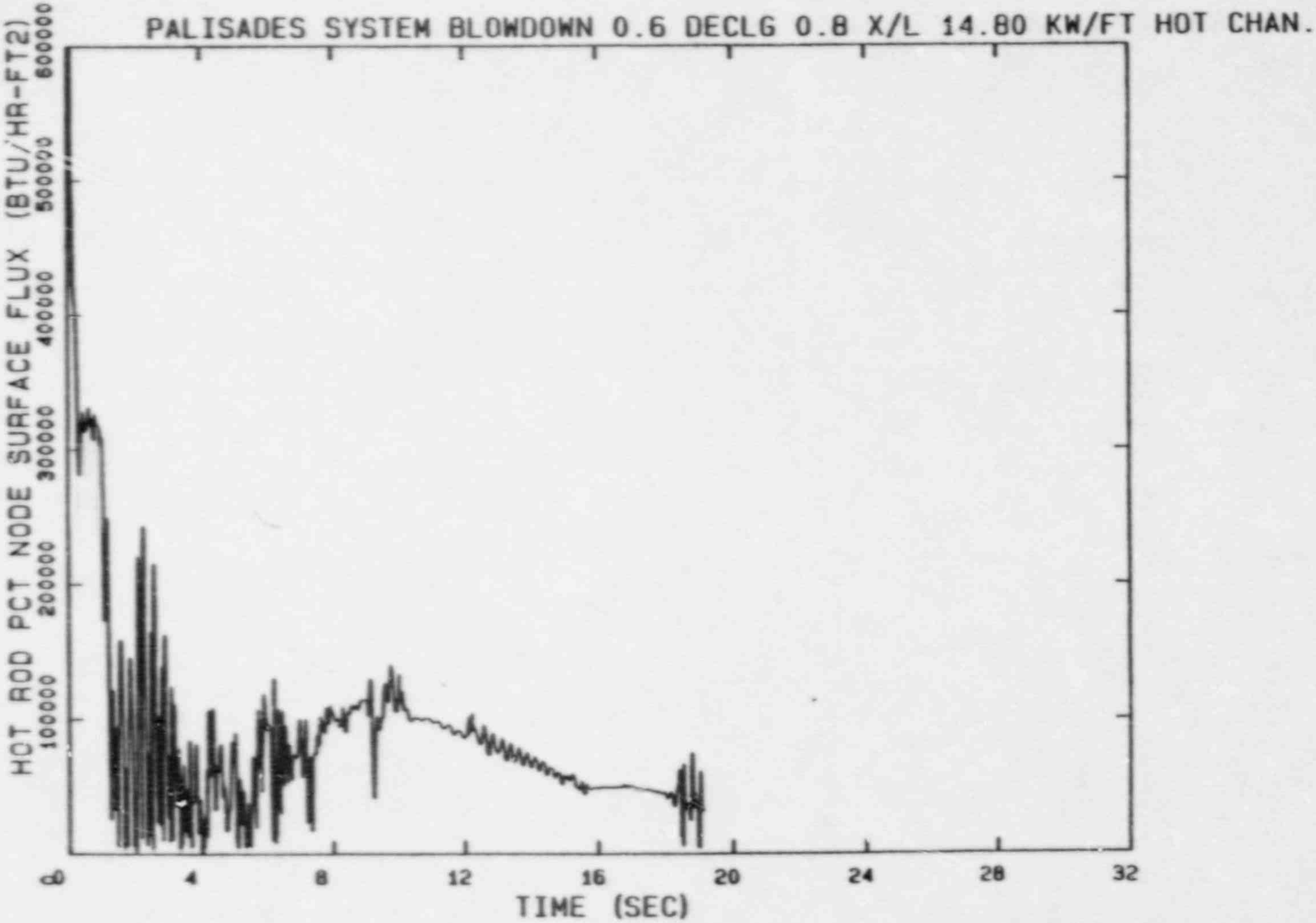

FIGURE 3.5.20 PCT NODE HEAT FLUX DURING BLOWDOWN, 0.6 DECLG BREAK, X/L = 0.8

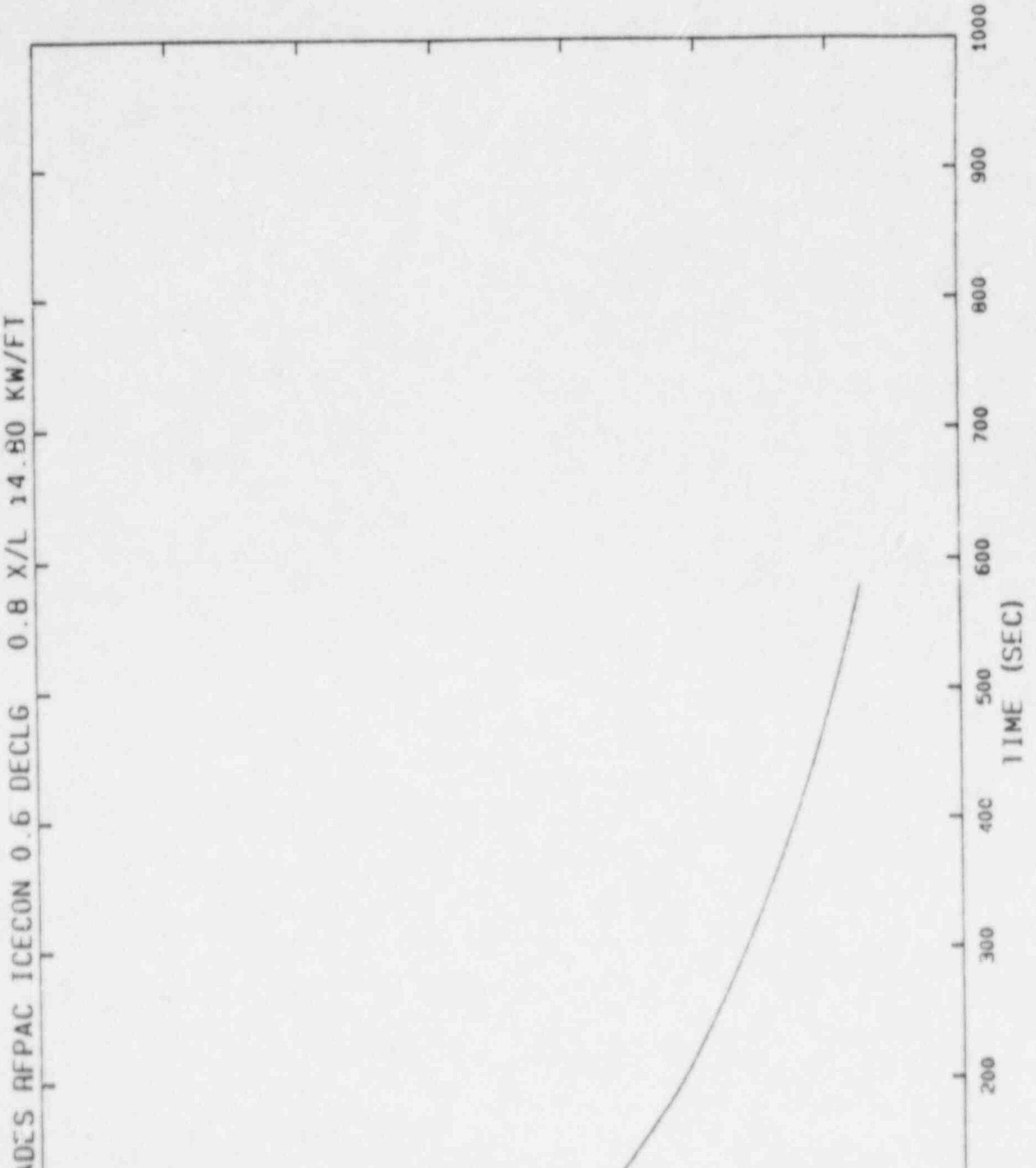

FIGURE 3.5.21 CONTAINMENT PRESSURE, 0.6 DECLG BREAK, X/L = 0.8

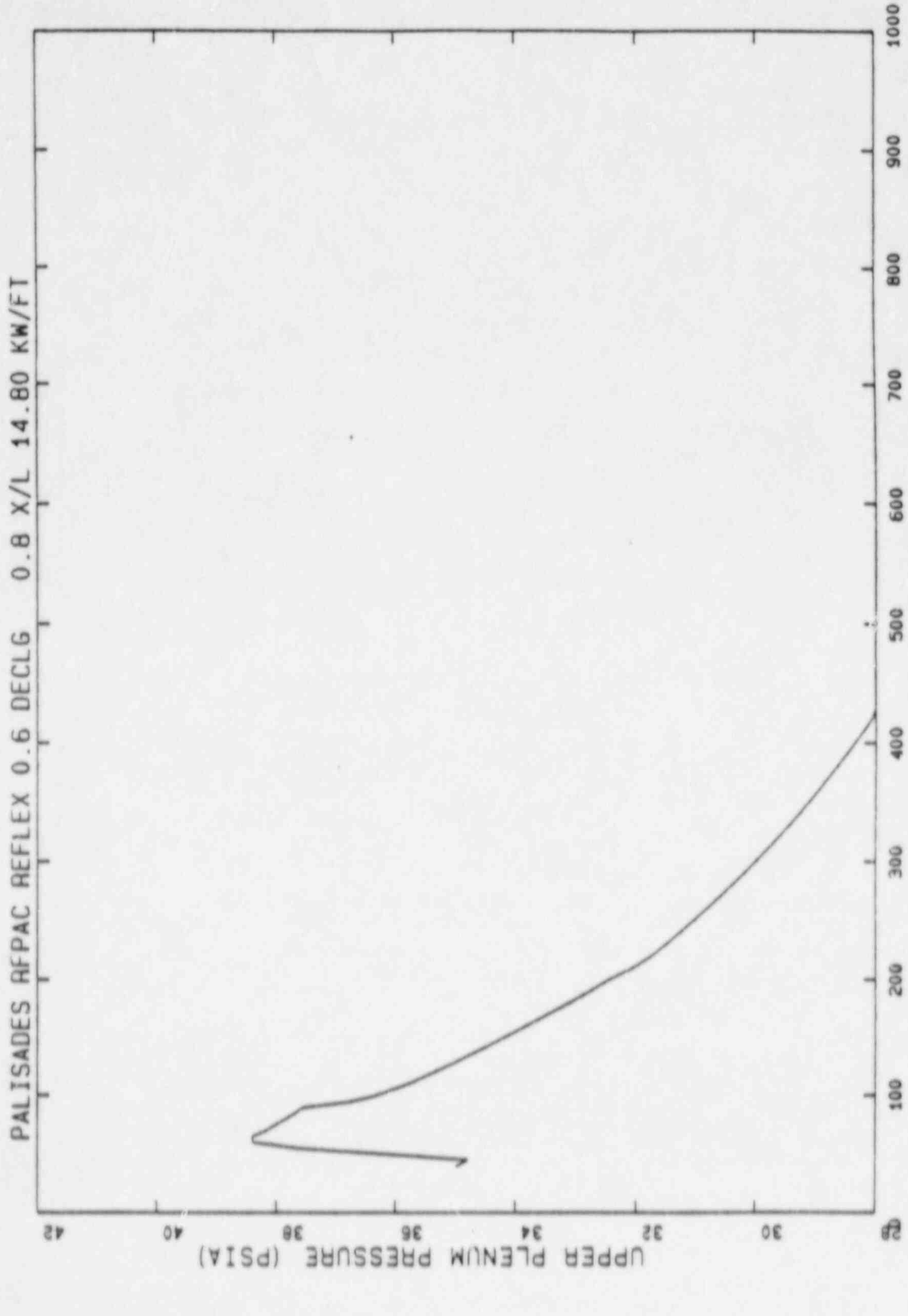

FIGURE 3.5.22 UPPER PLENUM PRESSURE AFTER EOBY, 0.6 DECLG BREAK, X/L = 0.8

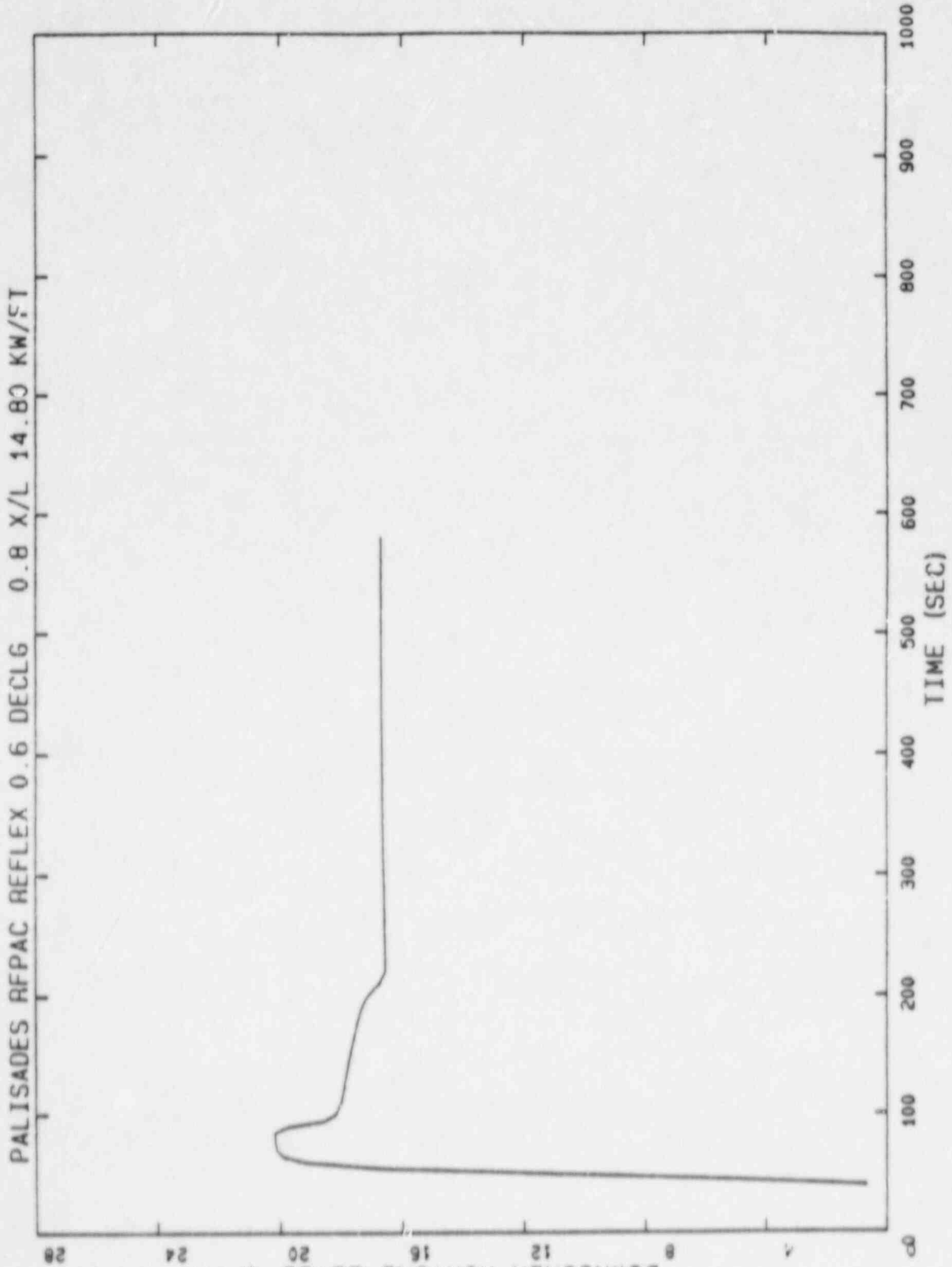

FIGURE 3.5.23 DOWNCOMER MIXTURE LEVEL AFTER EOBY, 0.6 DECLG BREAK, X/L = 0.8

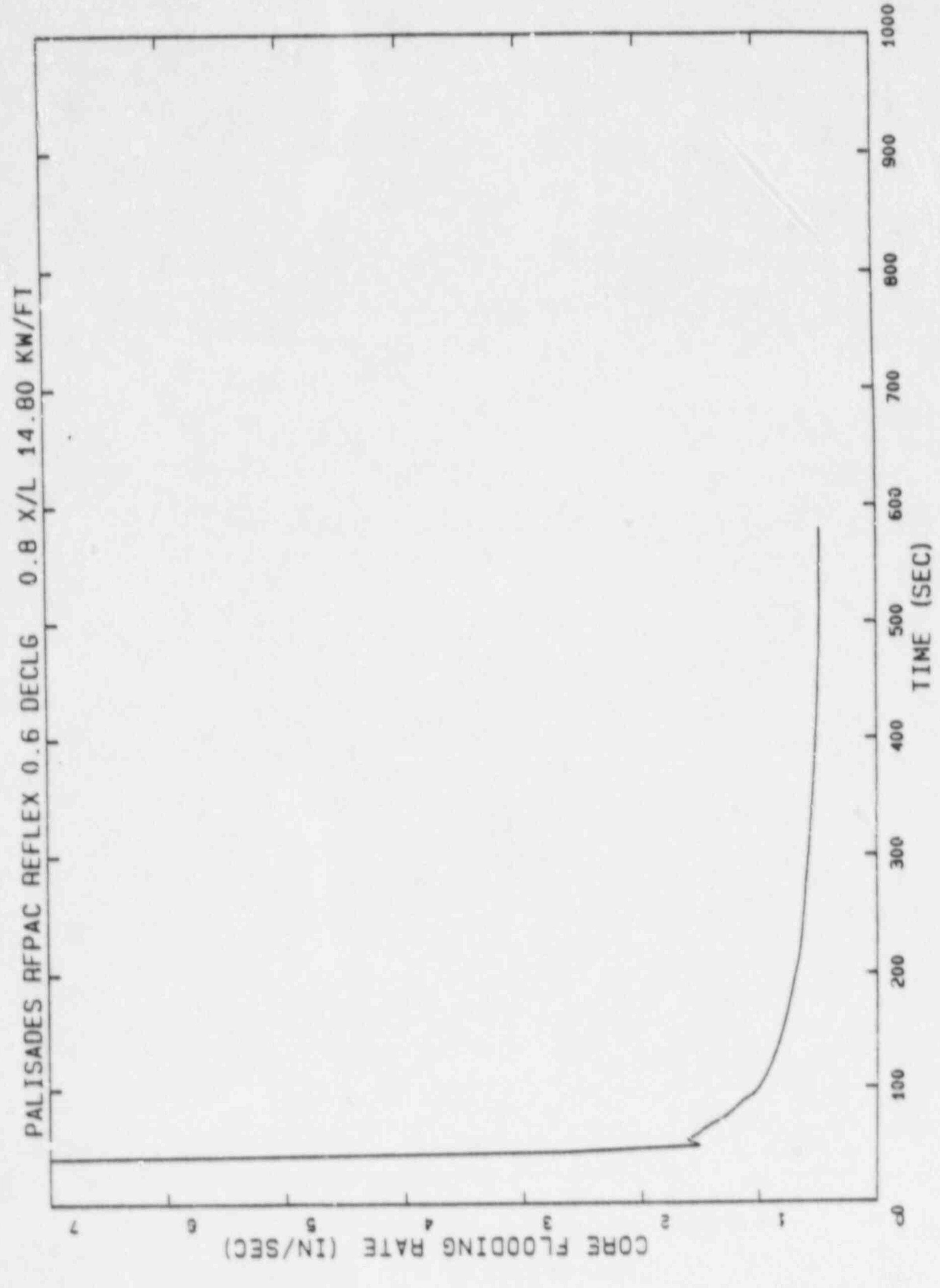

FIGURE 3.5.24 CORE FLOODING RATE AFTER EOBY, 0.6 DECLG BREAK, X/L = 0.8

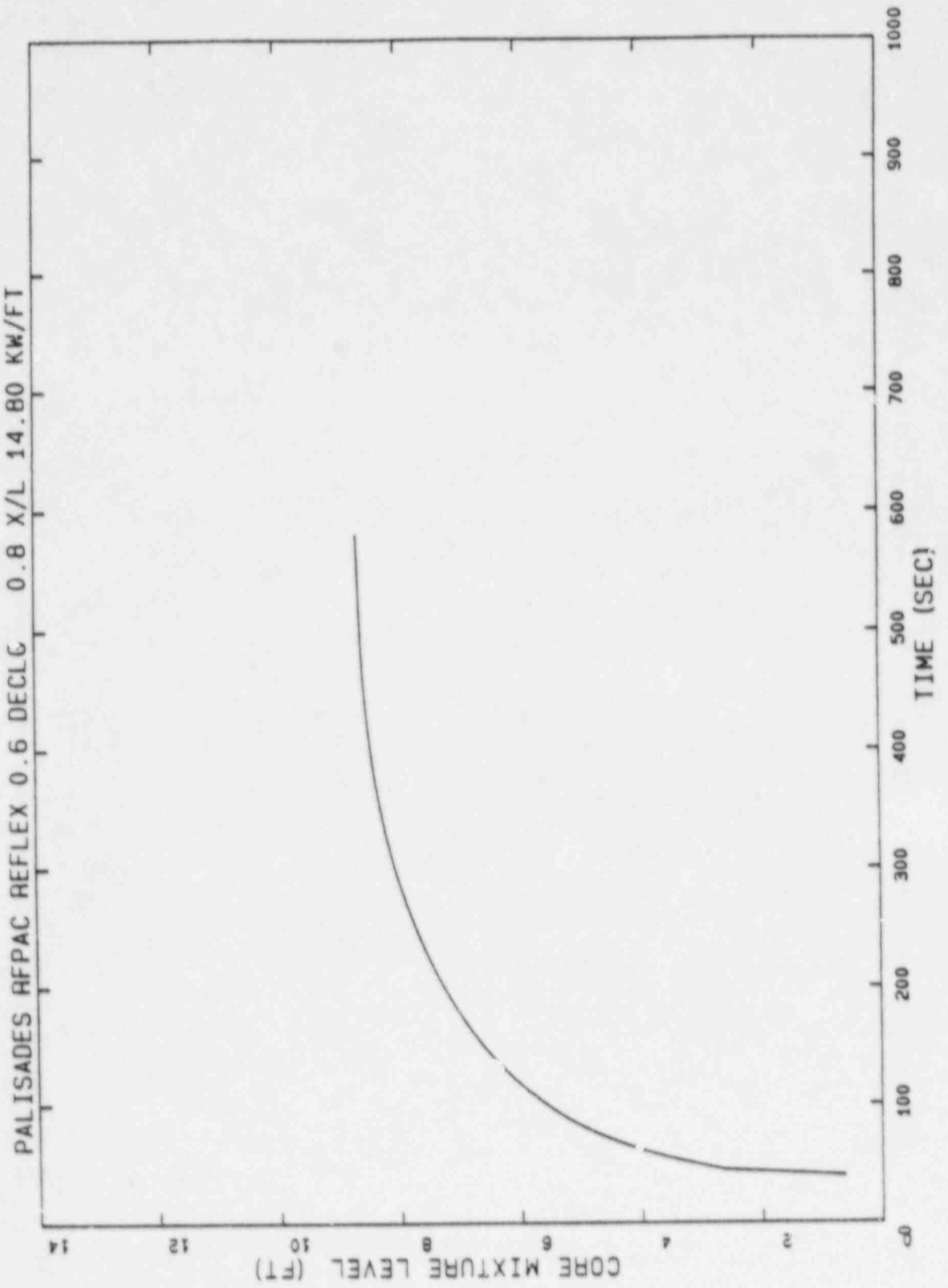

FIGURE 3.5.25 CORE MIXTURE LEVEL AFTER EOBY, 0.6 DECLG BREAK, X/L - 0.8

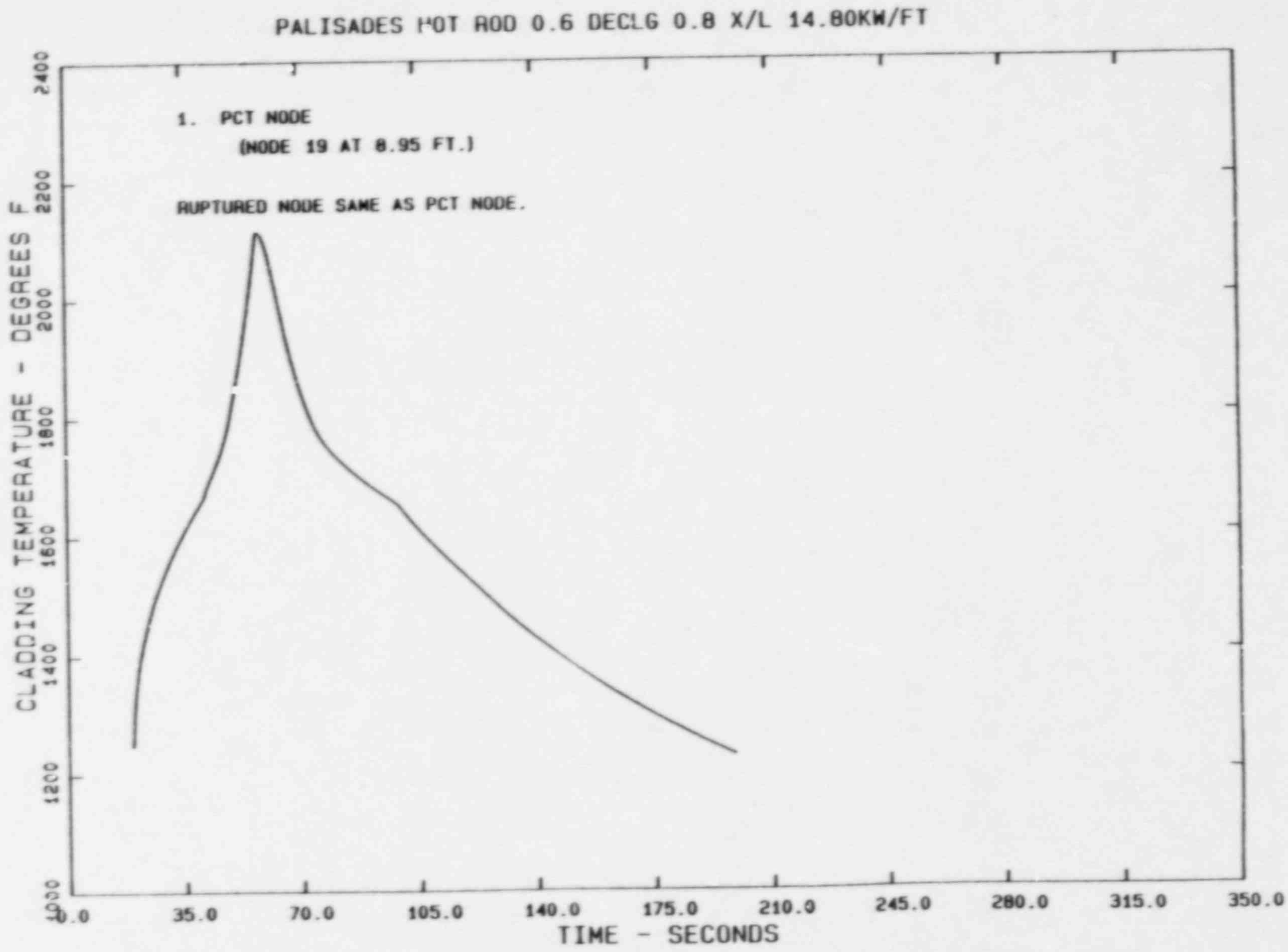

FIGURE 3.5.26 PCT NODE CLADDING TEMPERATURE AFTER EOBY, 0.6 DECLG BREAK, X/L = 0.8

4.0 CONCLUSIONS

The analysis with the current EXEM/PWR models for the Palisades plant confirms the 0.6 DECLG break size as the limiting break size. The analysis supports operation of the Palisades plant at a power level of 2530 MWt, an increase in the total radial peaking factor from 1.77 to 1.83, and an average steam generator tube plugging level of 29.3% with a maximum asymmetry of 4.5%. The analysis supports a peak LHR of 15.28 kW/ft with the axially dependent power peaking limit shown in Figure 2.1. The analysis supports Cycle 8 operation and is intended to support operation for future cycles.

Operation of the Palisades plant with ANF 15x15 fuel at or below the LHR limits 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 the Palisades plant.

5.0 REFERENCES

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2 Dennis M. Crutchfield (USNRC Asst. Director division of PWR Licensing-B) "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.

3. RODEX2: Fuel Rod Thermal Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 1, and Supplements 1-4, Exxon Nuclear Company, Richland, WA 99352, February 1983.

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PALISADES LARGE BREAK LOCA/ECCS
ANALYSIS WITH INCREASED RADIAL PEAKING

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