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PALISADES LOCA-ECCS ANALYSIS FOR 2125 MWt OPERATION AND 50% STEAM GENERATOR TUBE PLUGGING

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EXXON NUCLEAR COMPANY, Inc.

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1.0 INTRODUCTION AND SUMMARY

This document presents analytical results for a postulated large break loss-of-coolant accident (LOCA), performed for the Palisades nuclear reactor. The analyses assume: (a) a reactor operating power of 2167.7 MWt (2125.2 MWt + 2% power uncertainty) which is 84% of the rated power, (b) an increase in steam generator tube plugging to an average of 50% (40% in Loop 1 and 60% in Loop 2), and (c) an all Exxon Nuclear Company core (Batch J fuel design). The calculations were made for the double-ended cold leg guillotine break located at the pump discharge with a discharge coefficient of 0.6 (0.6 DEG/PD) which was previously identified as the most limiting break[1,2,3].

The analyses were performed using the EXEM/PWR ECCS evaluation model[4], with the RODEX2 computer model[5] for evaluating the rod stored energy and fission gas release. The EXEM/PWR ECCS evaluation model includes the NRC fuel swelling and flow blockage model, contained in NUREG-0630[6]. The analyses are applicable up to an average of 50 percent steam generator (SG) tube plugging and maximum peak pellet exposure limit of 47,000 MWD/MTM. The allowable linear heat generation rate was 12.84 kW/ft (12.59 Kw/ft times 1.02 multiplier for power uncertainty), corresponding to a total power peaking factor of 2.76 ($F_Q{}^T$), and radial peaking factor of 1.45 with a local peaking factor of 1.224. The axial power distribution is based on a skewed profile with a peak magnitude of 1.51 located at 60 percent of active core height[2].

The calculational basis and results are summarized in Table 1.1. The maximum calculated peak cladding temperature (PCT) is 1567°F, occurring 160 seconds into the accident at a location 8.20 feet from the bottom of the active core, with the maximum local metal-water reaction less than one

percent. The results of the analyses show that 10 CFR 50.46[7] and Appendix K criteria are satisfied for operation of the Palisades Nuclear Power Station within the established peaking limits, at the stated power level, and with 50% average steam generator tube plugging.

Table 1.1 Palisades Nuclear Reactor LOCA-ECCS Analysis Results

Analysis Results	Peak Pellet Exposure
Peak Clad Temperature (PCT)	1567
Time of PCT, sec.	160
Peak Clad Temperature Location, ft.	8.20
Local Zr/H_2O Reaction (max.), %*	0.59
Local Zr/H_2O Location, ft. from bottom	7.97
Total H_2 Generation, % of total Zr Reacted	<1.0
Calculational Basis	
License Core Power, MWt	2125.2
Power Used in Analysis, MWt**	2167.7
Peak Linear Power for Analysis, kW/ft**	12.84
Total Peaking Factor, F_Q^T	2.76
Radial Peaking Factor F_R	1.45
Local Peaking Factor F_L	1.224
Peak Axial power	1.51
Z/L	0.6
Skewing Factor	1.0
Steam Generator Tube Plugging	40% Loop 1, 60% Loop 2

* Computed value at 330 seconds.
** Including 1.02 factor for power uncertainties.

2.0 LIMITING BREAK LOCA ANALYSIS

This report provides LOCA-ECCS analyses performed for Palisades nuclear reactor with an average steam generator tube plugging of up to 50%. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are described in the ENC WREM models[8], and the Emergency Core Cooling System Evaluation Model Updates: WREM-II[9], and EXEM/PWR[4].

A LOCA break spectrum analysis was performed and reported in XN-NF-77-24[1]. The limiting LOCA break was determined to be a large double-ended guillotine break of the pump discharge, with a discharge coefficient of 0.6 (0.6 DEG/PD). The current analysis assumes 84% of the rated power, reduced primary pressure and flow and an average steam generator tube plugging level of 50%. The secondary system pressure and flow are correspondingly reduced. Table 2.1 shows the new system parameters for reduced power operation. The analyses performed and reported herein for the 0.6 DEG/PD break used:

(1) The revised stored energy model RODEX2[5] in place of the previously applied GAPEX[11] model.

(2) Applicable EXEM/PWR models[4].

(3) The ENC/NUREG-0630 clad swelling and rupture model[6].

(4) The REFLEX code for reflood calculations[10].

(5) The FLECHT/ENC2 WREM[8] heat transfer coefficient multipliers.

2.1 LOCA ANALYSIS MODEL

The Exxon Nuclear Company EXEM/PWR ECCS evaluation model[4] was used to perform the analyses. This model consists of the following computer codes: RODEX2[5] code for initial rod stored energy and internal fuel rod gas inventory; RELAP4-EM[12] for the system blowdown and hot channel blowdown

calculations; CONTEMPT-LT/22 as modified in CSB 6-1(13) for computation of containment back pressure; REFLEX(4,6) for comparison of system reflood; and TOODEE2(4,6,14) for the calculation of final fuel rod heatup.

The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow-paths or "junctons". The system nodalization is depicted in Figure 2.1 while the REFLEX nodalization is shown in Figure 2.2. The pump performance characteristic of Combustion Engineering pump were used with analysis.(1) Asymmetric steam generator tube plugging is assumed such that Loop 1 is 60% plugged and Loop 2 is 40% plugged. The break is assumed to have occurred in the most highly plugged loop since this results in higher peak clad temperatures. The transient behavior was determined from the governing conservation equations for mass, energy, and momentum. Energy transport, flow rates, and heat transfer are determined from appropriate correlations. System input parameters are given in Table 2.1.

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 analyses is shown in Figure 2.3 with a maximum axial peaking factor of 1.51 peaked at 60% of the core height, corresponding to a total peaking factor of 2.76, and a radial peaking of 1.45 with local peaking factor of 1.224.

The ECCS calculations were performed with input which bounds the fuel history up to 47000 MWD/MTU. The most limiting fuel conditions from beginning-of-life to end-of-life exposures were determined and used in the calculations. Internal rod pressure and decay power are highest at EOL for

the rod while stored energy is highest at about 2000 MWD/MTM. This combination of highest stored energy, highest rod pressure, and higher decay power were used in the analysis to bound operation of the Palisades reactor for the entire fuel exposure history.

2.2 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

For the purpose of LOCA analyses, a loss of coolant accident is defined as a rupture of the Reactor Primary Coolant System piping 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 low 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 enhances 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 is calculated, consistent with Appendix K of 10 CFR 50.[7] Thereafter, the core heat transfer is unstable, with both transition and film

boiling occurring. 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 215 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 again consistent with Appendix K of 10 CFR 50.

2.3 RESULTS

Table 2.3 presents the timing and sequence of events as determined for the large guillotine break with a discharge coefficient of 0.6 located at pump discharge. Comparison of these results with the previous LOCA-ECCS analysis[1,2,3] for E, G, and H fuel loadings show an improvement in the LOCA-ECCS analysis due to the reduction in the LHGR. Figures 2.4 through 2.18 present plotted results for system blowdown analysis. Unless otherwise noted on the figures, time zero corresponds to the time of break initiations. Figures 2.19 through 2.22 present results for the hot channel blowdown calculations. Figures 2.23 through 2.29 present the accumulators, HPSI and LPSI flows during the refill and reflood periods of LOCA transient. Figure 2.30 presents calculated containment backpressure time history. Figure 2.31 shows the normalized power calculation results. The reflood calculation results are shown in Figures 2.32 through 2.35.

The maximum peak cladding temperature (PCT) calculated for the 0.6 DEG/PD break is 1567^{o}F, Figure 2.36, which is applicable for the entire fuel exposure. The maximum linear heat generation rate is 12.84 kW/ft ($F_Q{}^T$=2.76). The maximum local metal-water reaction is less than 1%. The PCT location is at an elevation of 8.20 feet from the bottom of active core.

Table 2.1 Input Data For Reduced Power Operation

Reactor Power (MWt)	2167.7	
Primary System Pressure (psia)	1950	
Average Coolant Temperature (^{o}F)	564	
Reactor Coolant Flow (Mlbm/hr)	99	
Average S.G. Tube Plugging (%)	50	
	Loop 1	Loop 2
S. G. Tube Plugging (%)	60	40
S.G. Primary Fluid Volume (ft^3)	403.7	605.6
S.G. Primary Flow Area (ft^2)	7.9	11.9
S.G. Heat Transfer Area (inside) (ft^2)	32396	48596
S.G. Heat Transfer Area (outside) (ft^2)	37152	55728
Secondary Pressure (psia)	600	600

Table 2.2 Nominal Fuel Design Parameters (Batch J Design)

Parameter	Dimension
Cladding, O.D., in.	0.417
Cladding, I.D., in.	0.358
Cladding Thickness, in.	0.0295
Pellet O.D., in.	0.350
Diametral Gap, in.	0.0080
Pellet Density, % TD	94.0
Active Fuel Length, in.	131.8
Enriched UO_2, in.	131.8
Rod Pitch, in.	0.55
No. of Fuel Rods/Assembly	208
No. of Fuel Assemblies	204

Table 2.3 Palisades Large Break Event Times - 0.6 DEG/PD Break

Event	Time (sec)
Start Transient	0.0
Initiate Break	0.1
Safety Injection Signal	C.83
Accumulator Injection, Broken Leg	10.80
Accumulator Injection, Intact Leg	17.20
Accumulator Injection, Intact Loop	17.20
End-of-Blowdown (Break Flow Reversal)	20.20
Safety Pump Injection HPSI	21.83
End-of-Bypass (Downcomer Flow Reversal)	25.97
Safety Pump Injection LPSI	28.83
Bottom of Core Recover	43.42
Start of Reflood	44.45
Accumulator Empty, Broken Leg	68.61
Accumulator Empty, Intact Leg	75.41
Accumulator Empty, Intact Loop	75.66
Peak Clad Temperature Reached	160

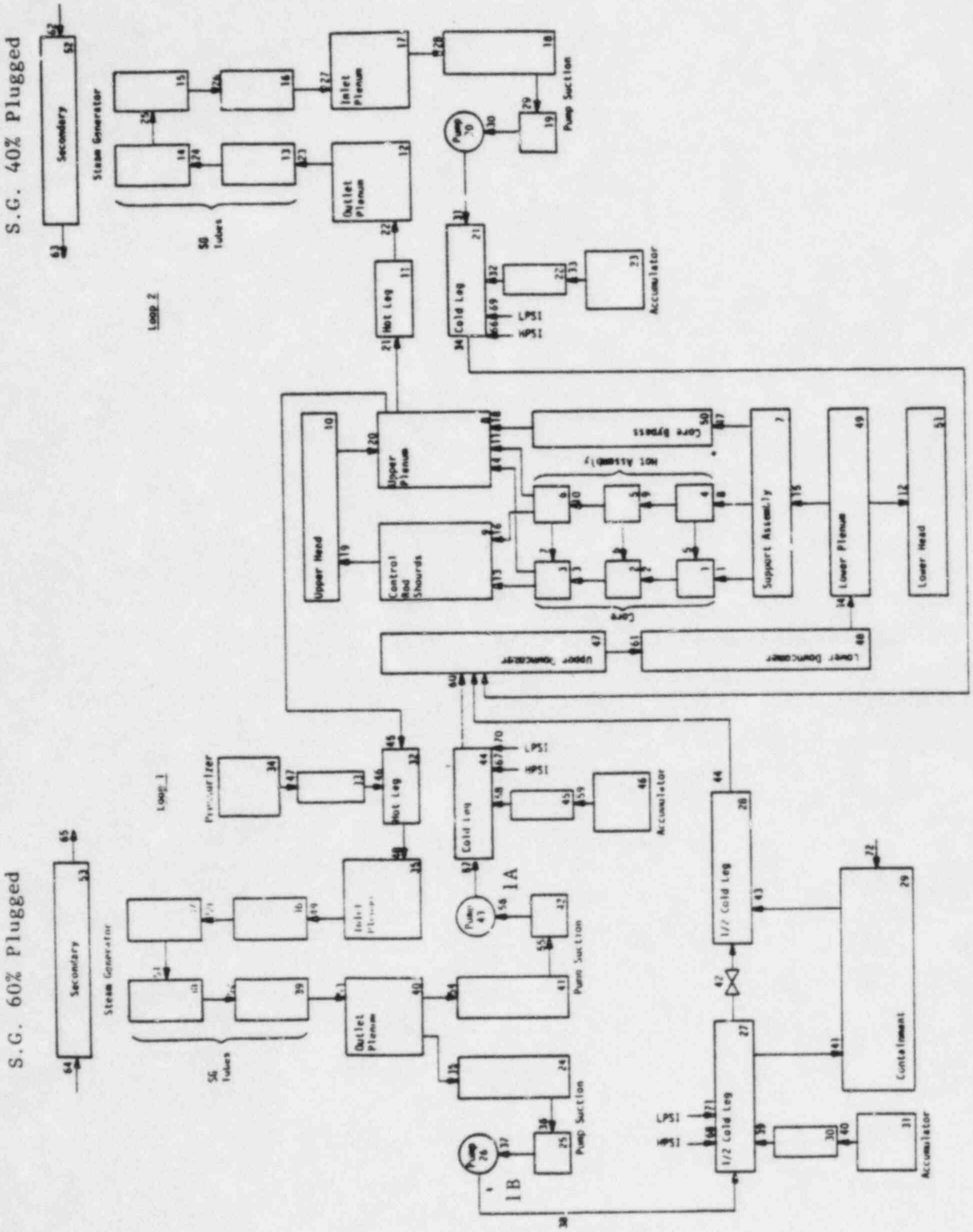


Figure 2.1 RELAP4/EM Blowdown System Nodalization For Palisades Nuclear Plant

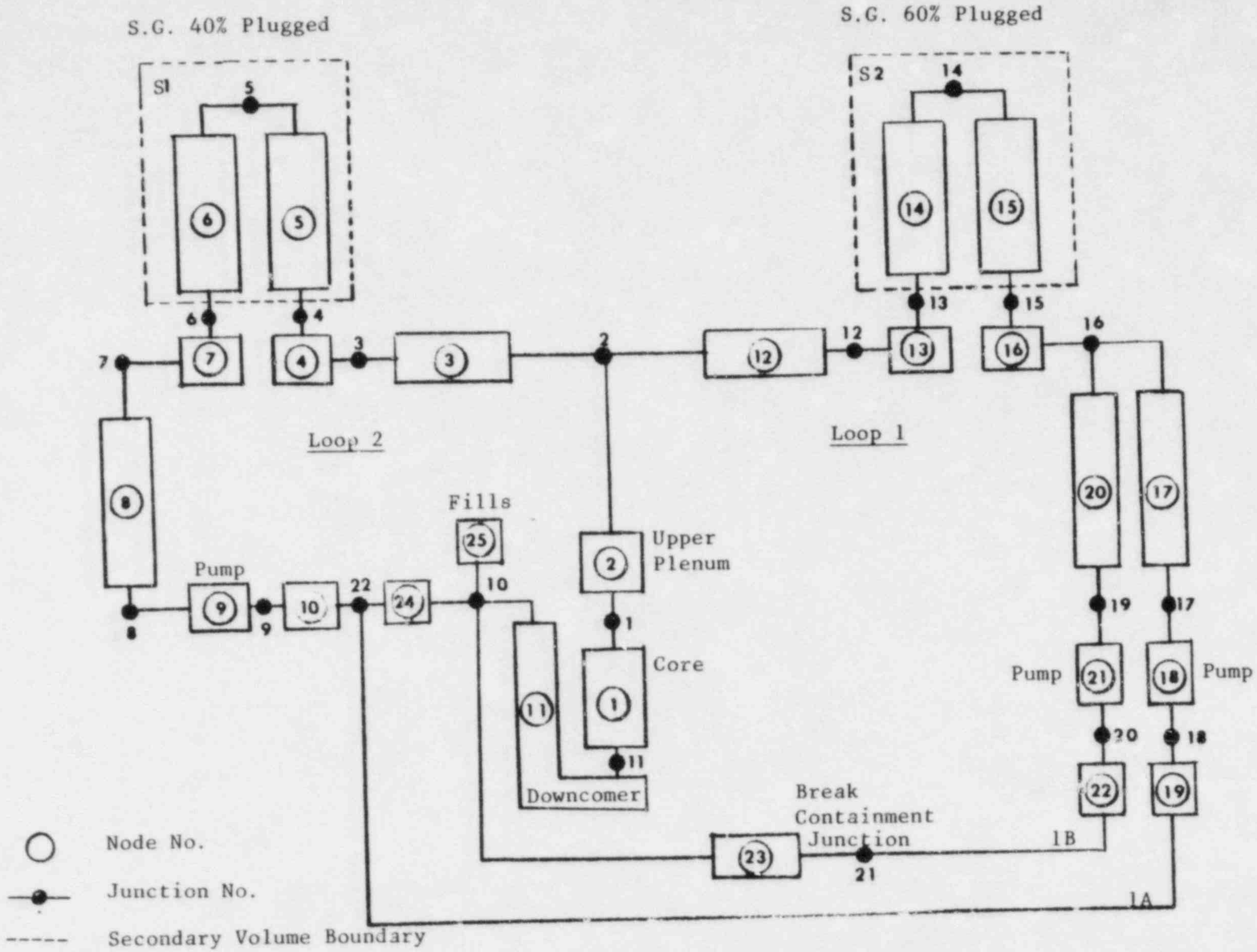


Figure 2.2 REFLEX Nodalization EXEM/PWR, Palisades Nuclear Plant

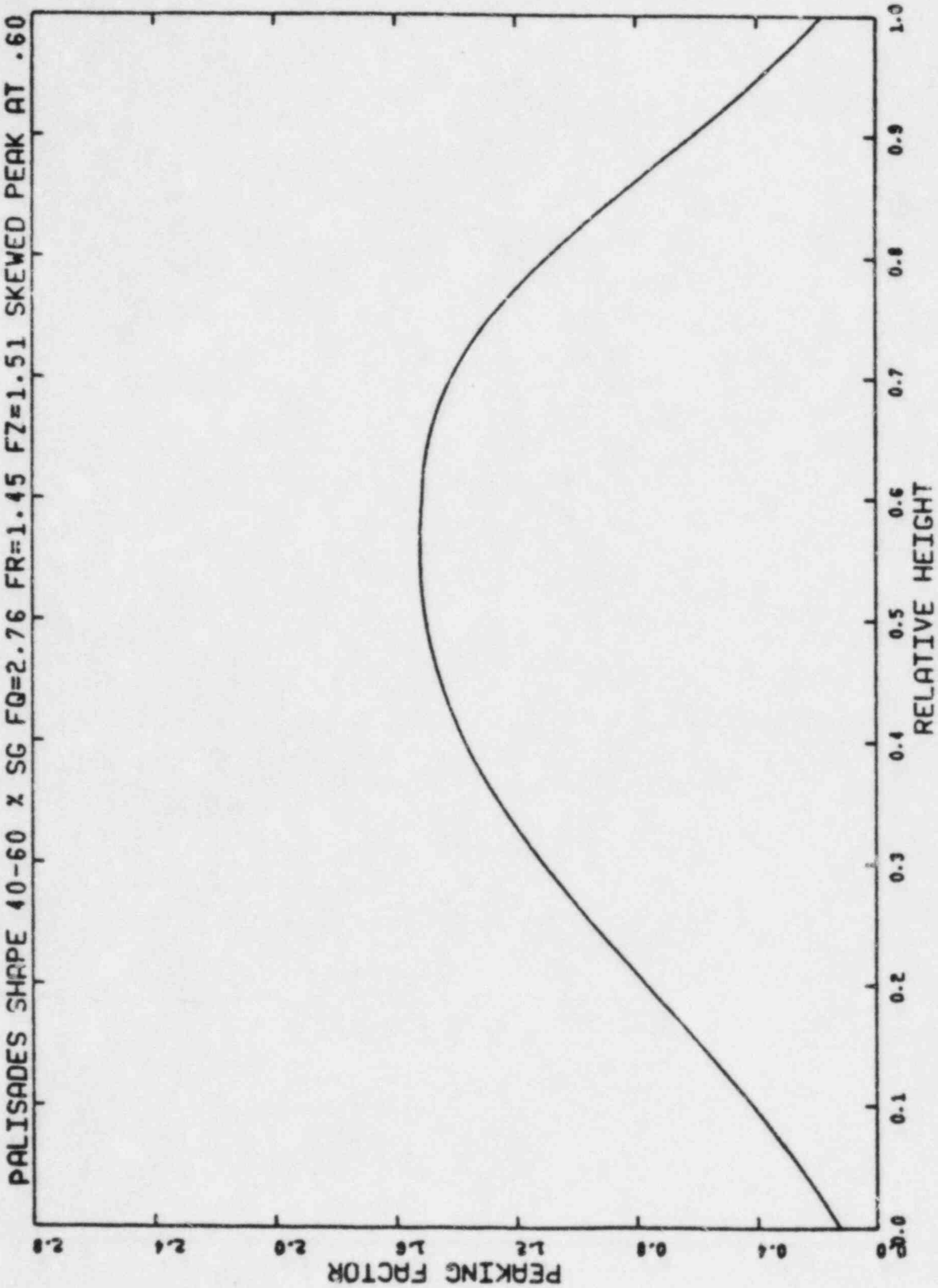


Figure 2.3 Axial Power Profile With Peak at X/L=0.6 and with Skewing Factor of 1.0

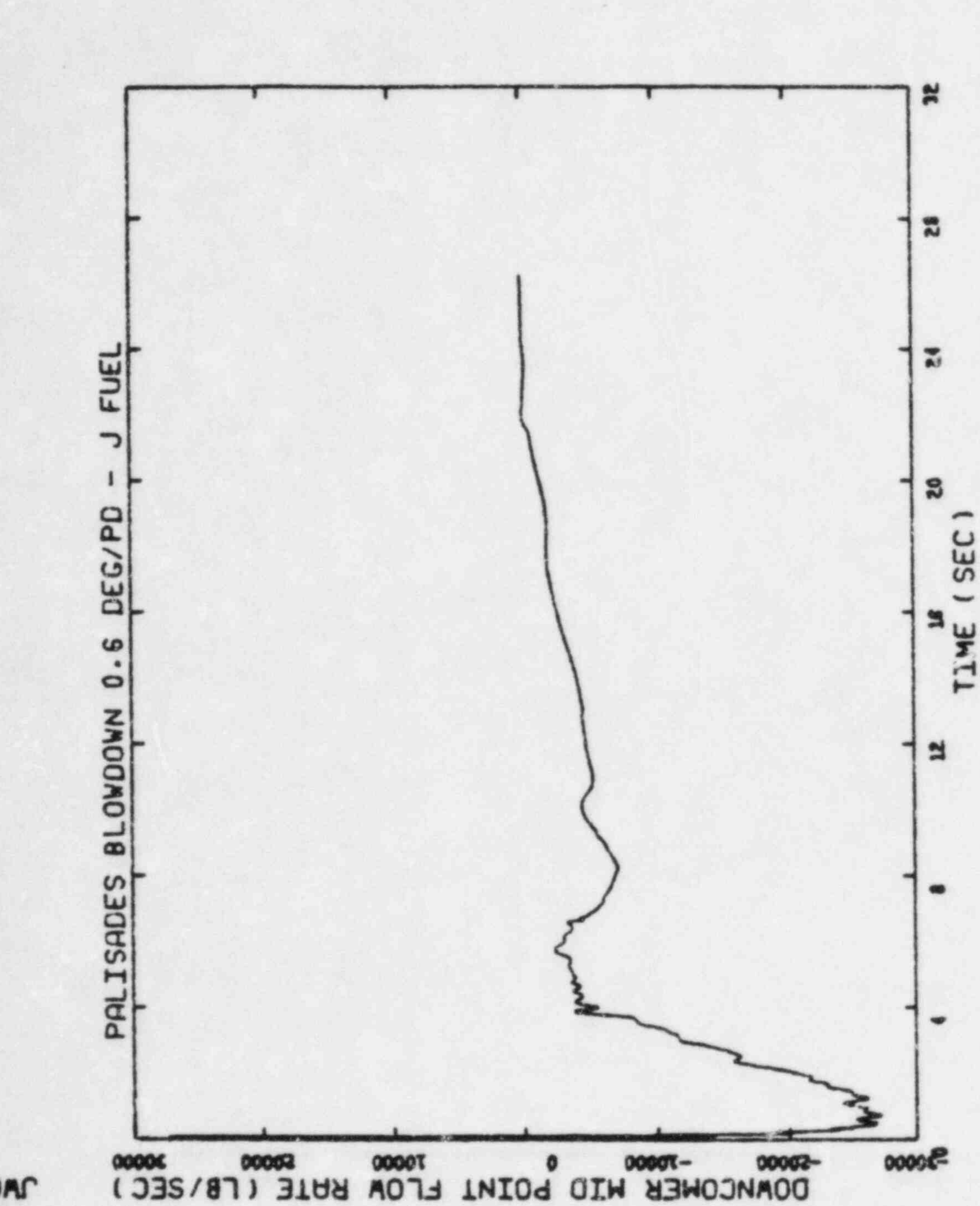


Figure 2.4 Downcomer Flow Rate During Blowdown Period, 0.6 DEG/PD Break

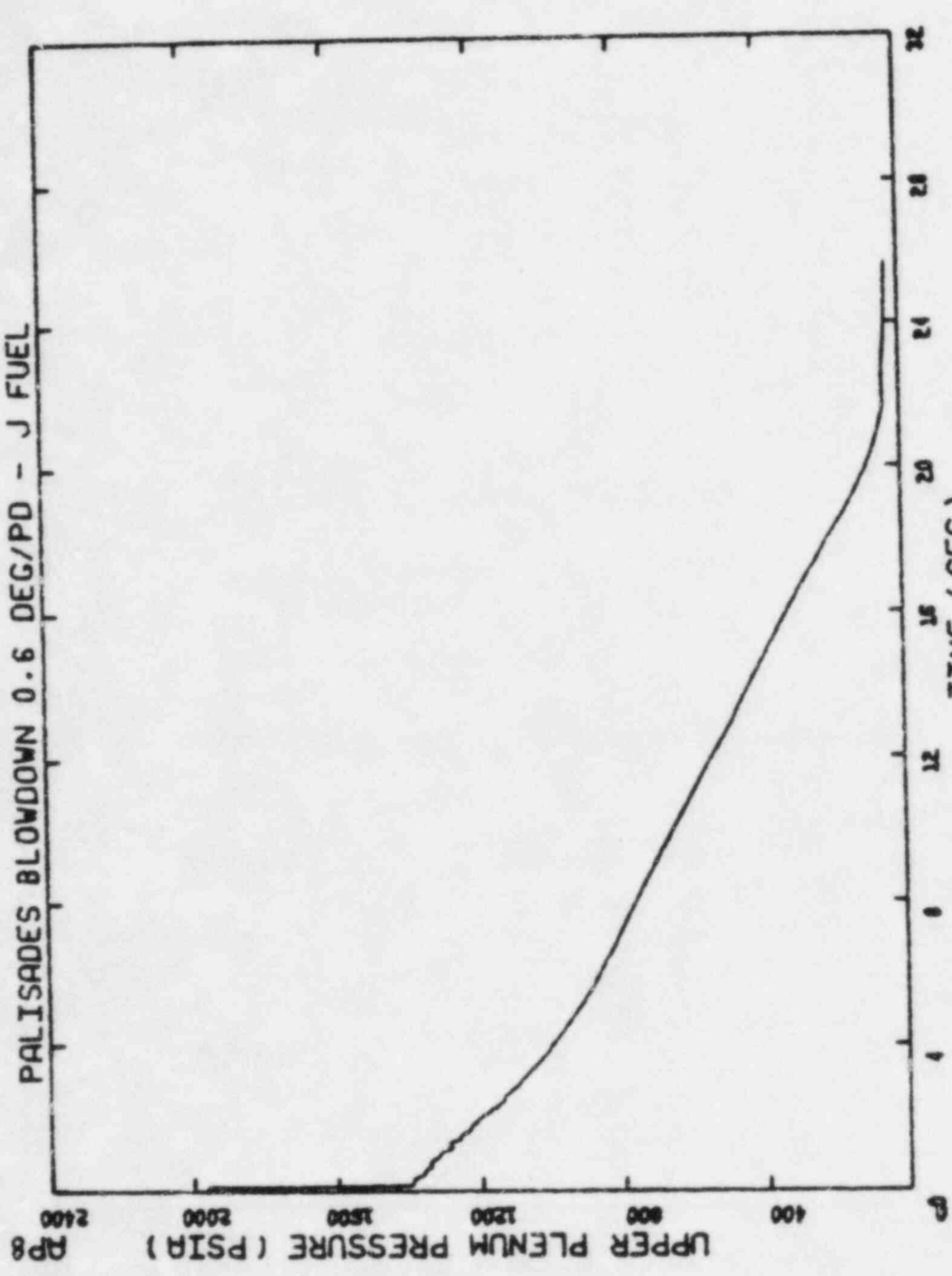


Figure 2.5 Upper Plenum Pressure During Blowdown Period, 0.6 DEG/PD Break

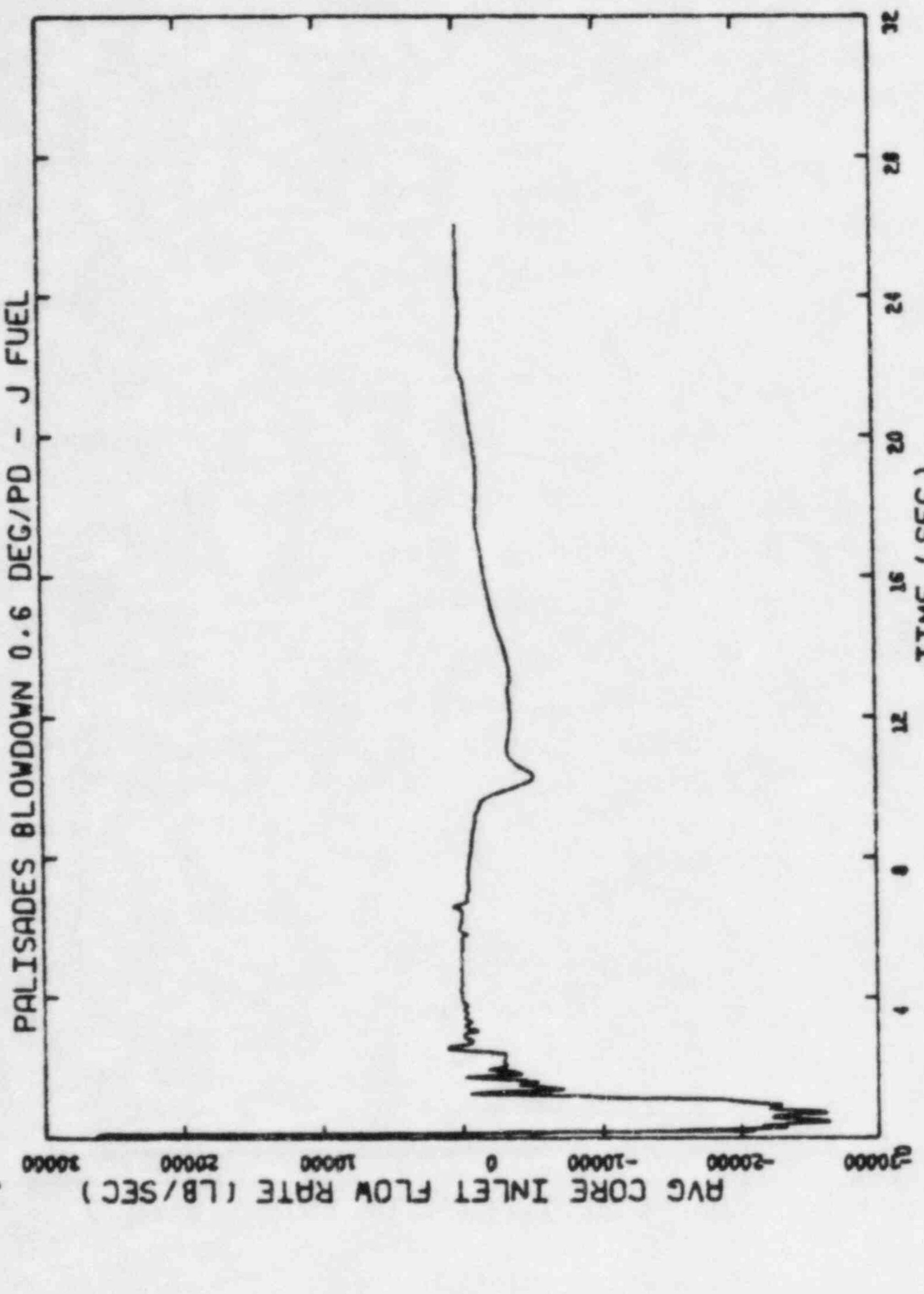


Figure 2.6 Average Core Inlet Flow During Blowdown Period, 0.6 DEG/PD Break

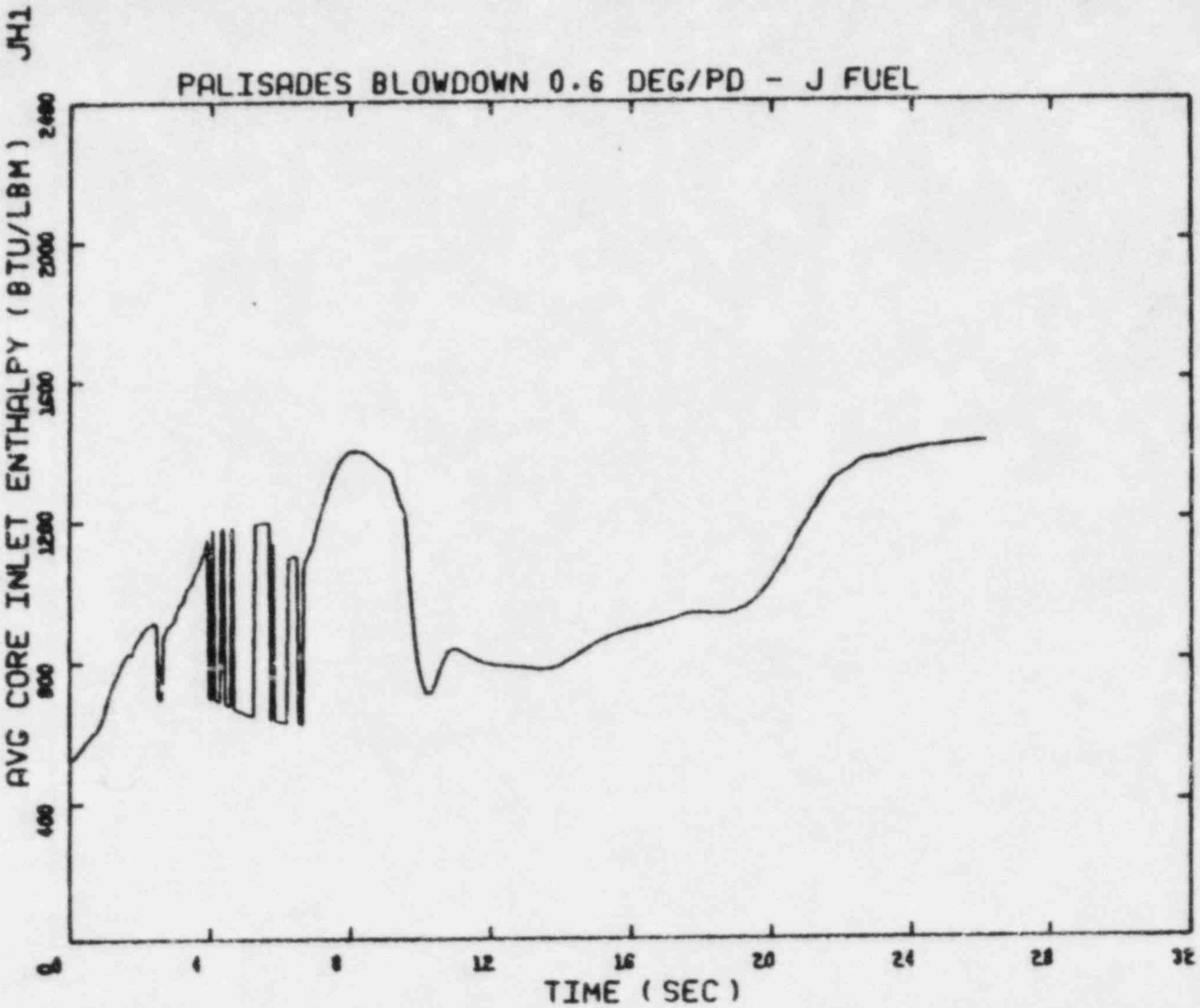


Figure 2.7 Average Core Inlet Enthalpy During Blowdown Period,
0.6 DEG/PD Break

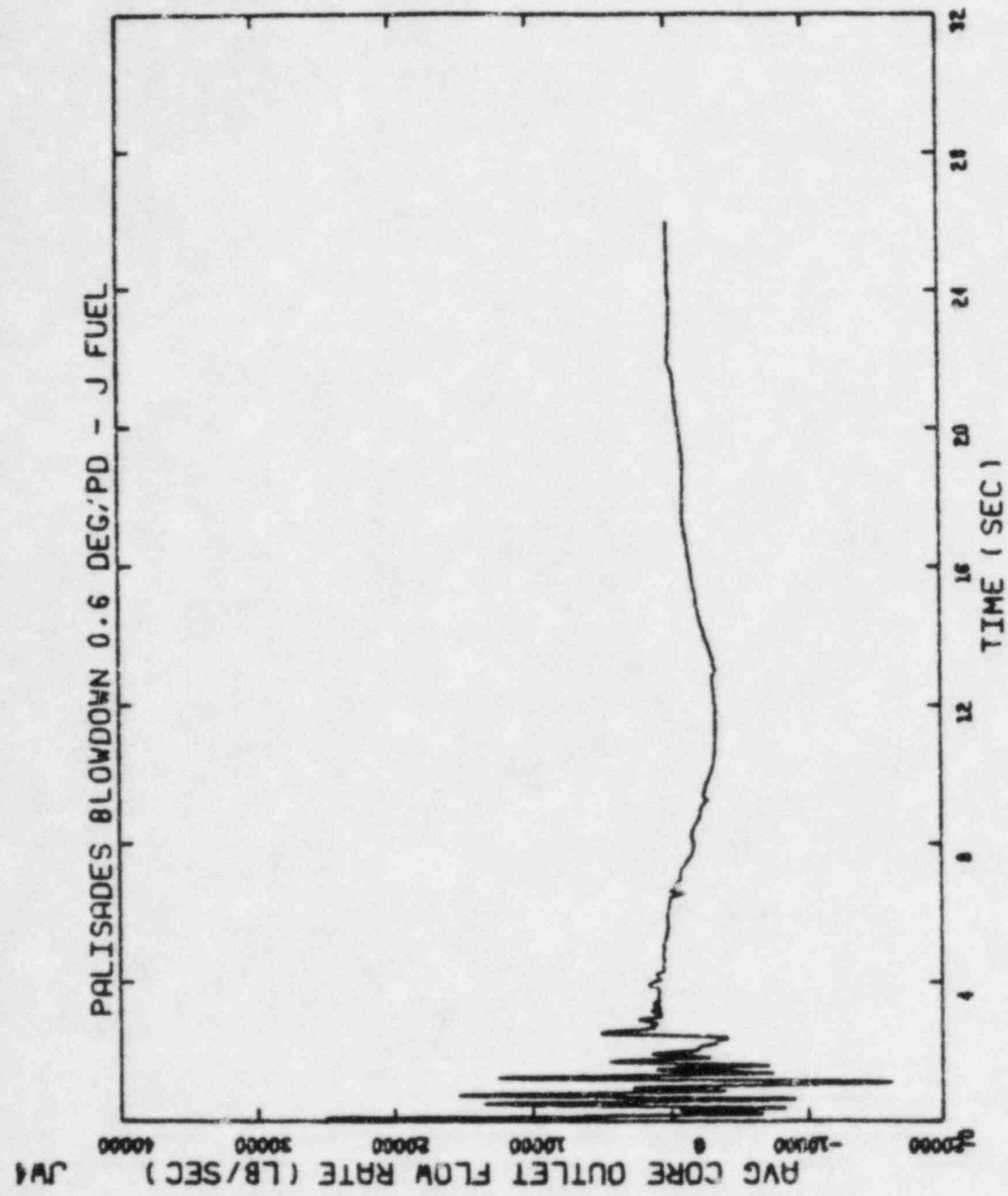


Figure 2.8 Average Core Outlet Flow During Blowdown Period, 0.6 DEG/PD Break

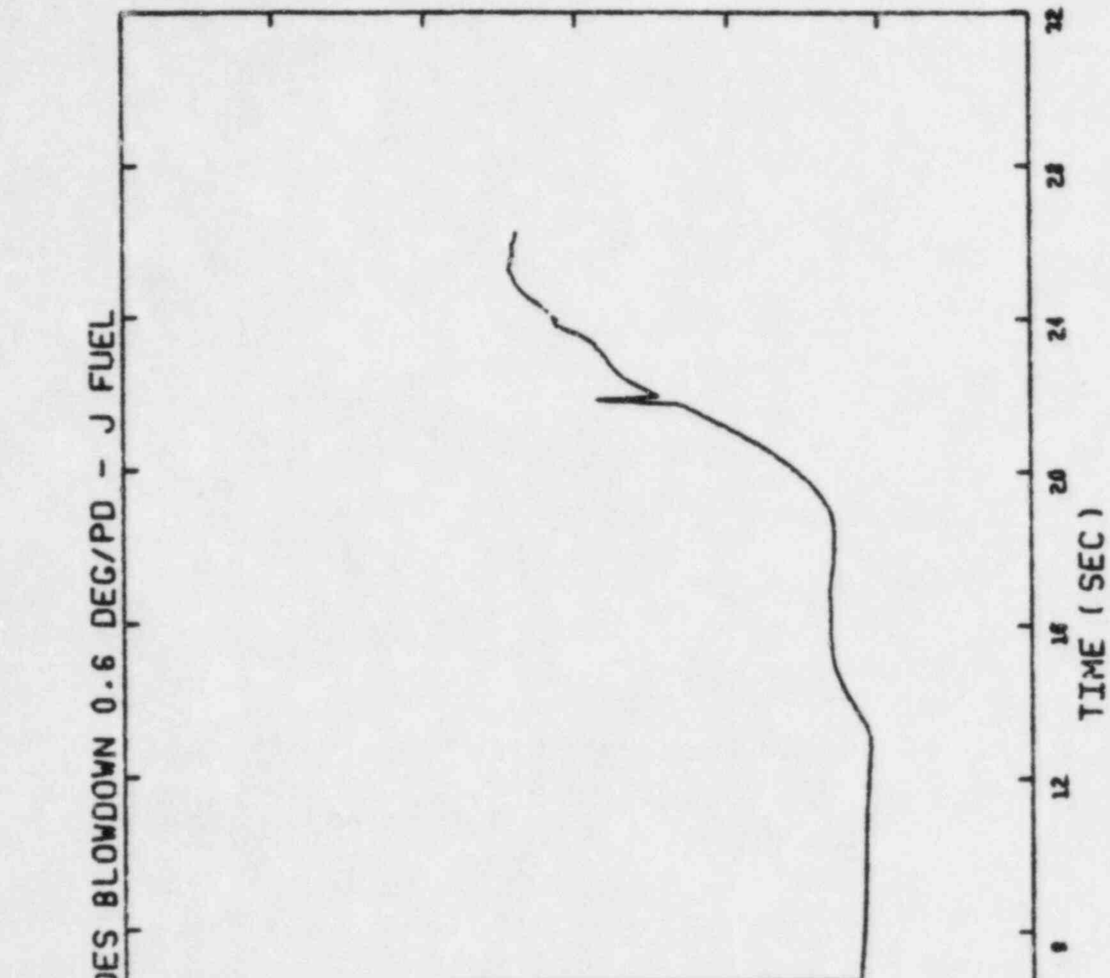


Figure 2.9 Average Core Outlet Enthalpy During Blowdown period, 0.6 DEG/PD Break

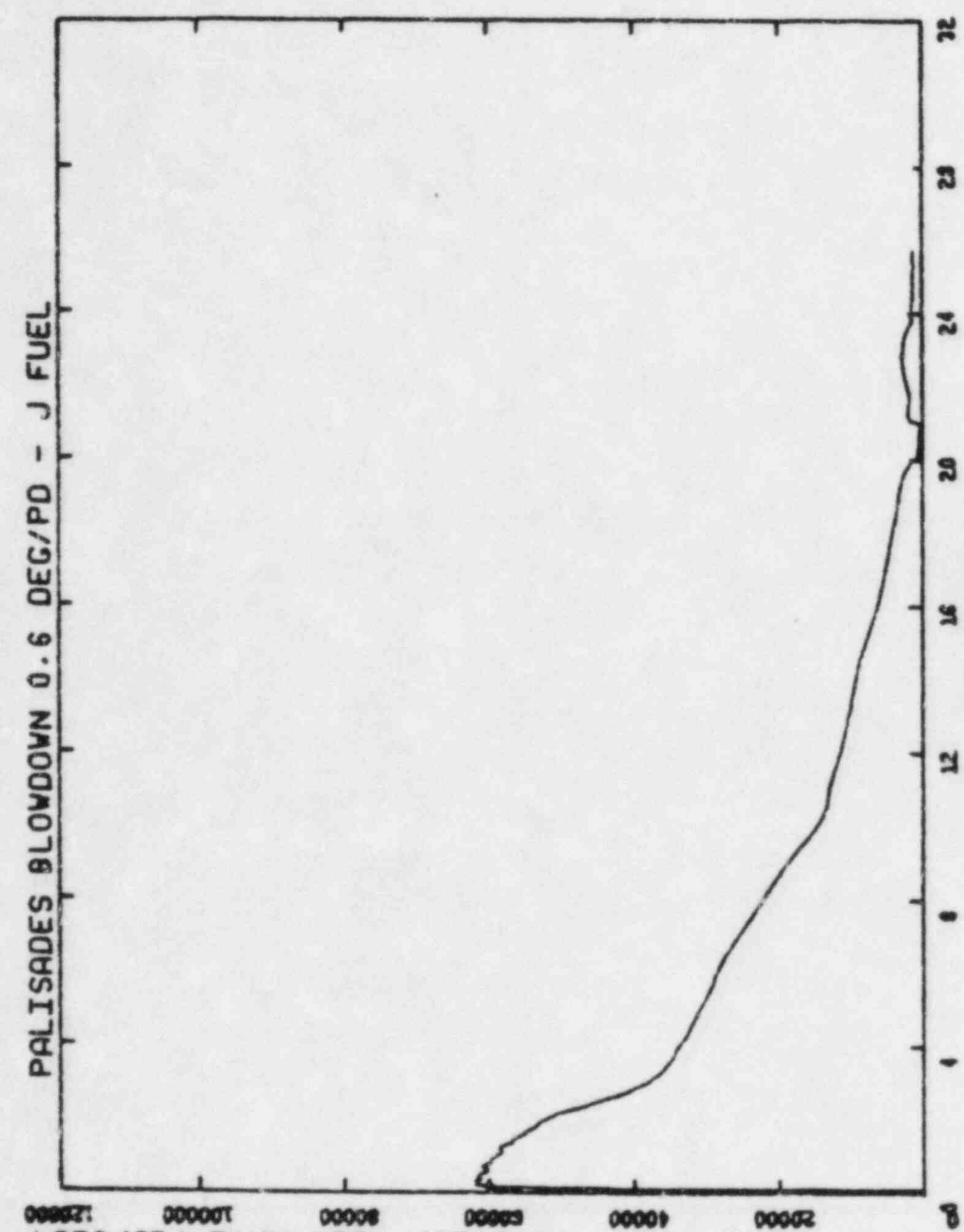


Figure 2.10 Total Break Flow During Blowdown Period, 0.6 DEG/PD Break

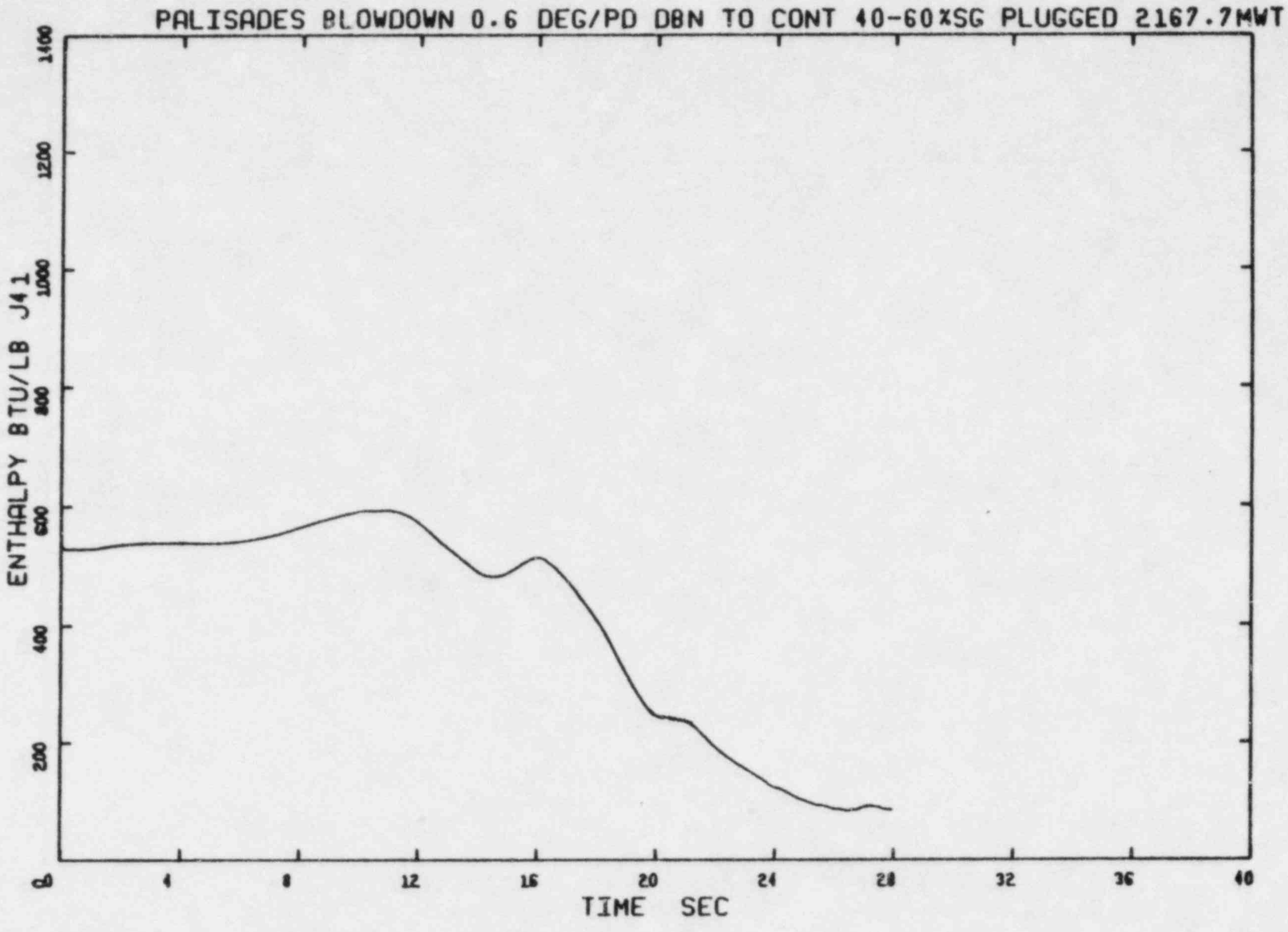


Figure 2.11 Vessel Side Break Enthalpy During Blowdown Period, 0.6 DEG/PD Break

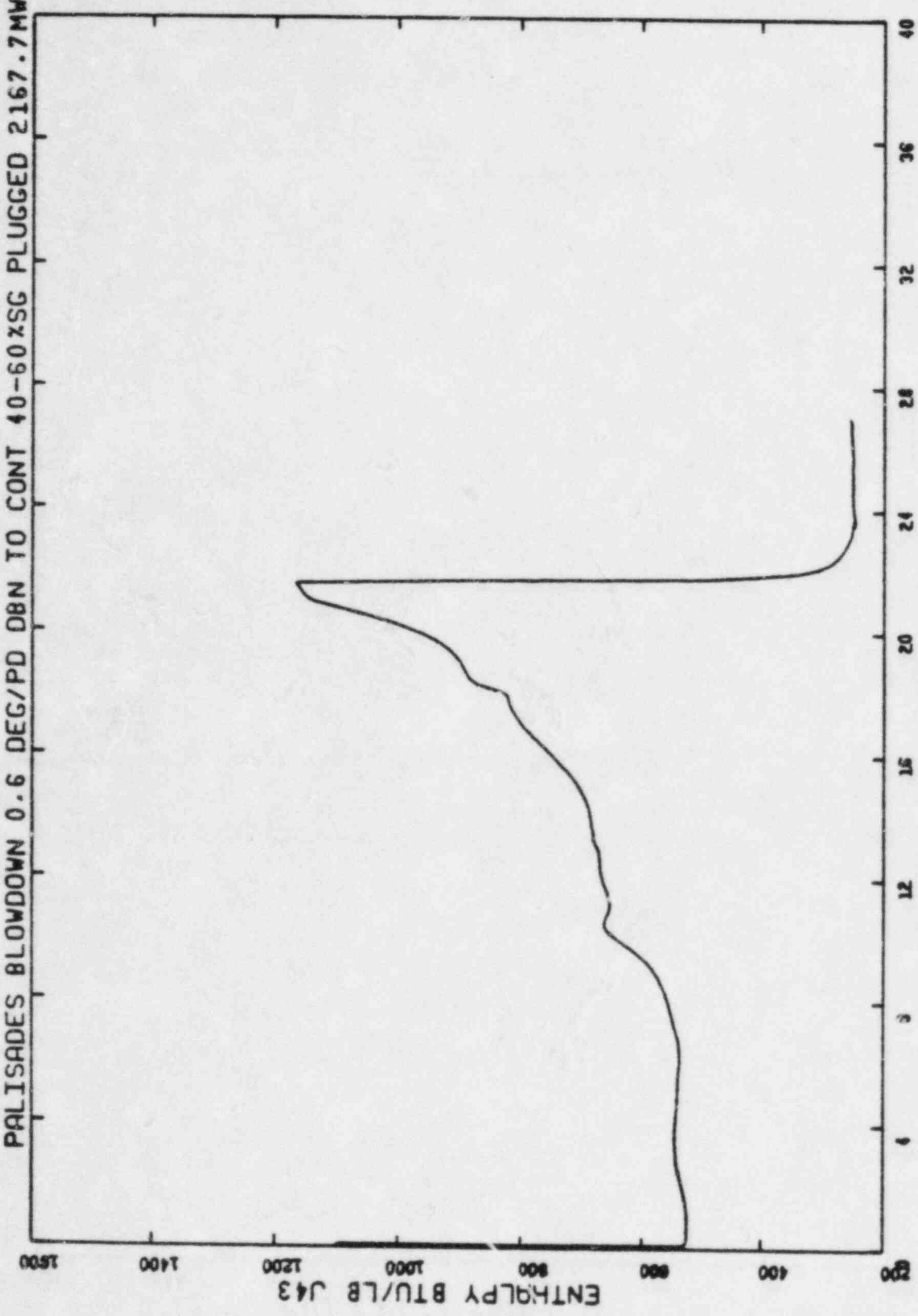


Figure 2.12 Pump Side Break Enthalpy During Blowdown Period, 0.6 DEG/PD Break

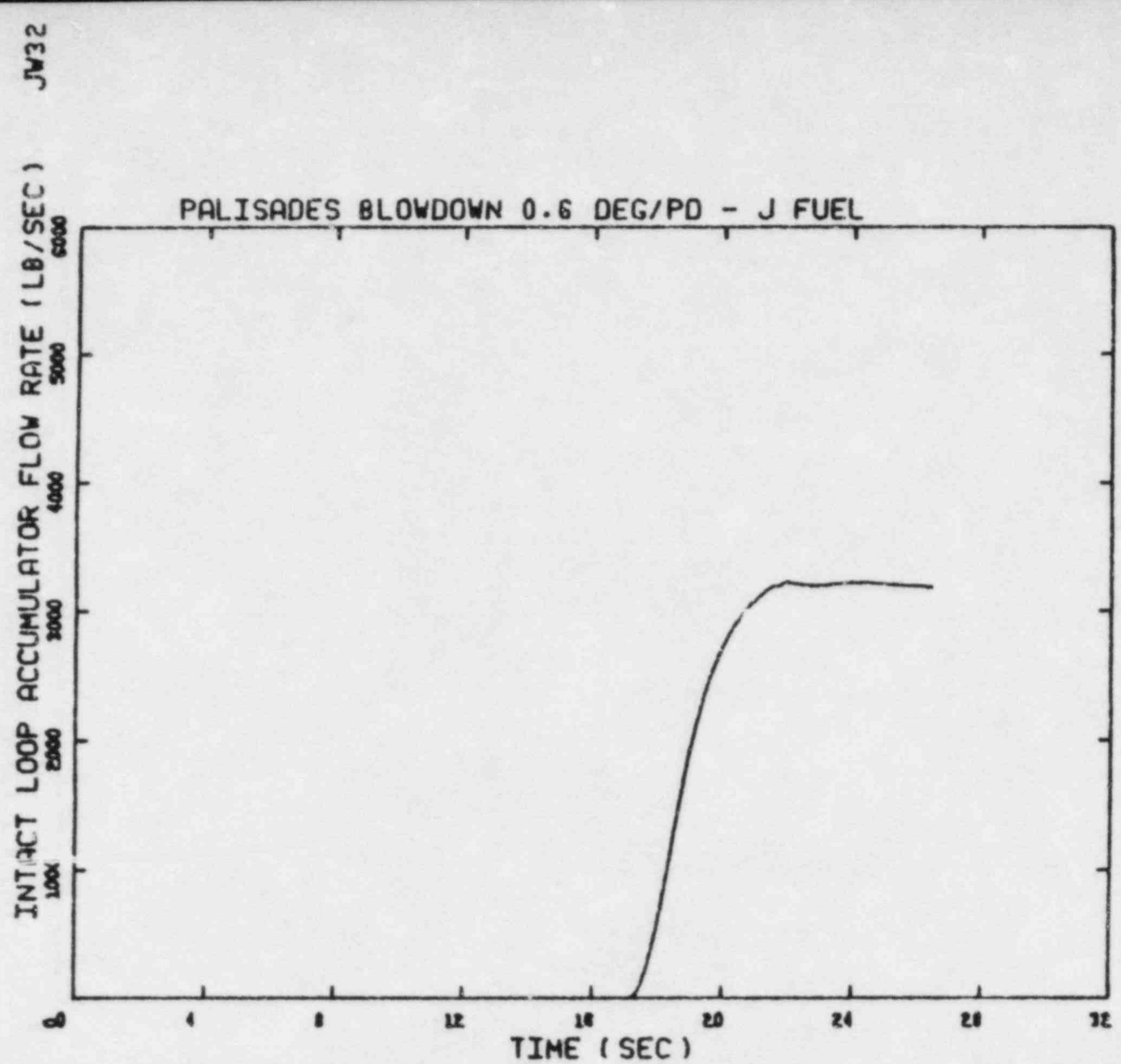


Figure 2.13 Flow From Two Intact Accumulators During Blowdown Period, 0.6 DEG/PD Break

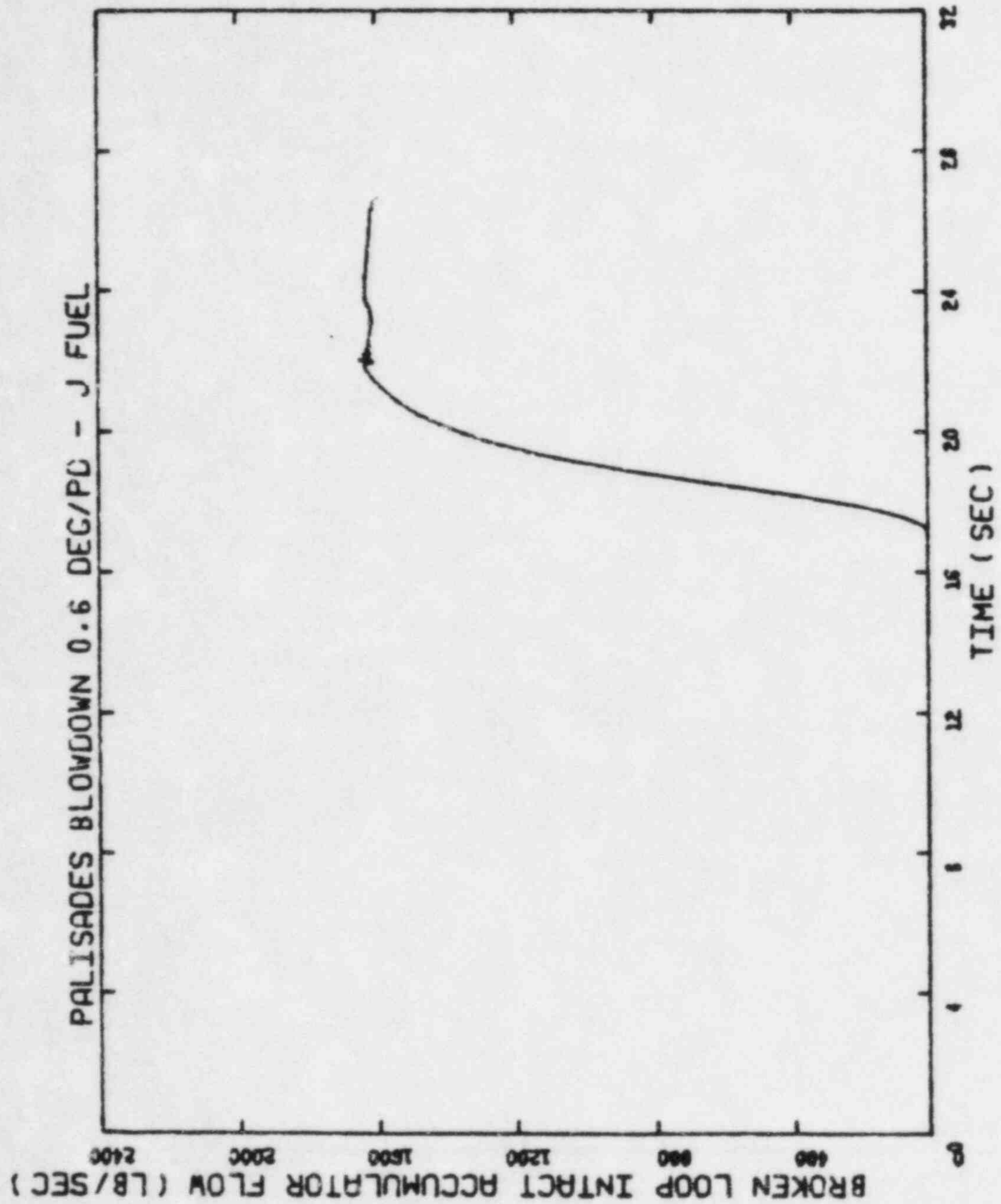


Figure 2.14 Flow From Broken Side Intact Accumulator During Blowdown Period, 0.6 DEG/PD Break

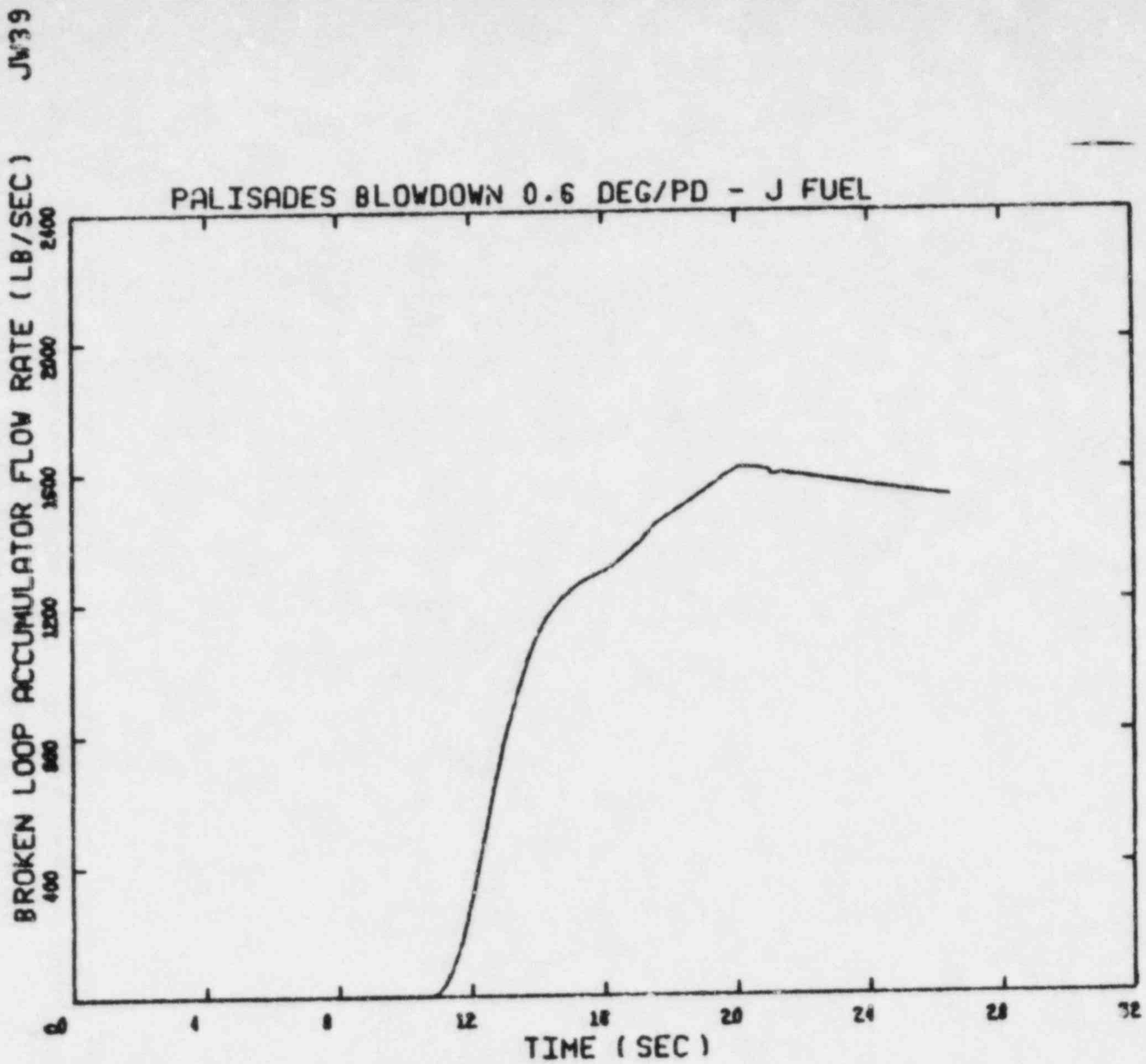


Figure 2.15 Flow from Broken Loop Accumulator During Blowdown Period, 0.6 DEG/PD Break

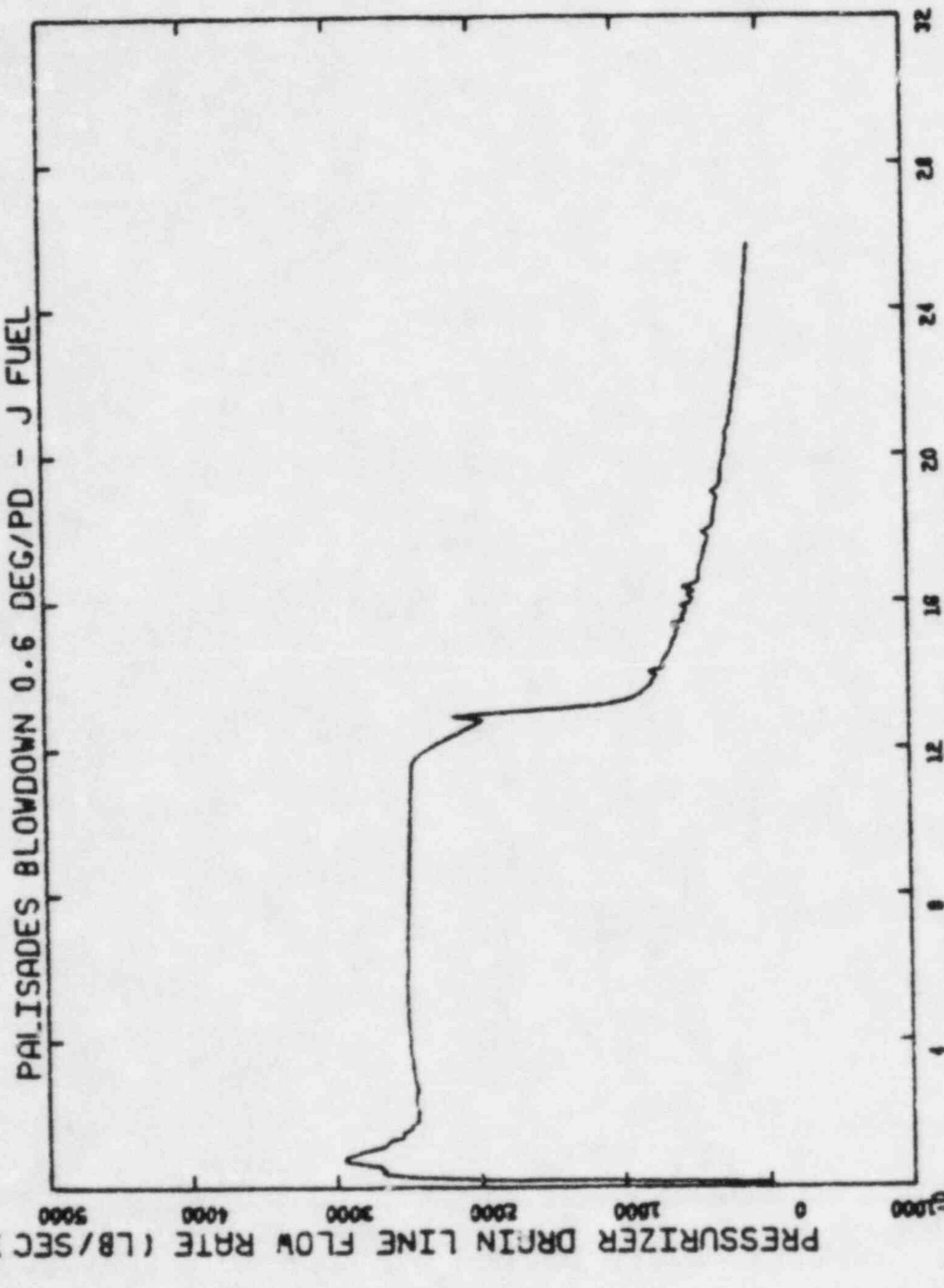


Figure 2.16 Pressurizer Surgeline Flow During Blowdown Period, 0.6 DEG/PD Break

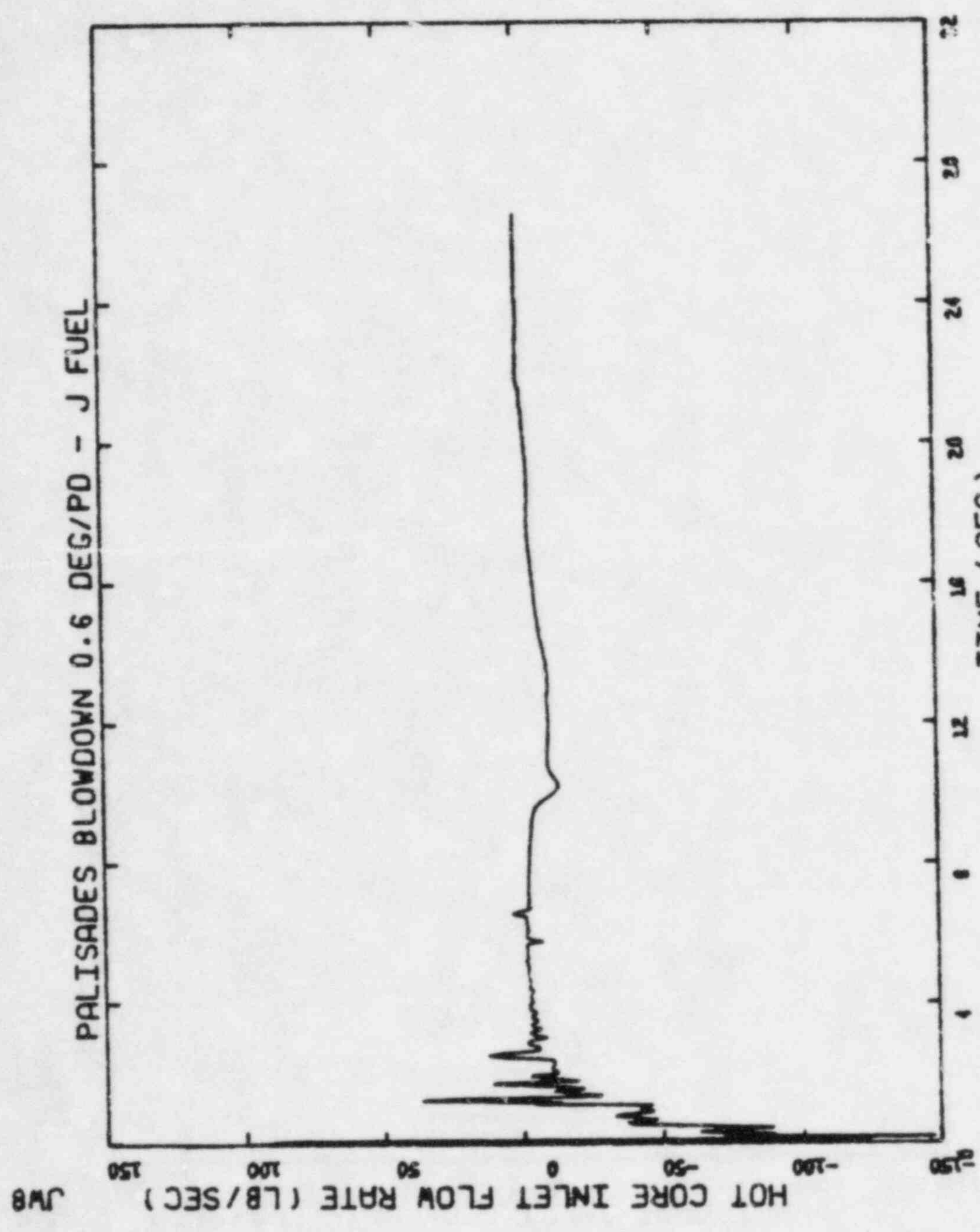


Figure 2.17 Hot Assembly Inlet Flow During Blowdown Period, 0.6 DEG/PD Break

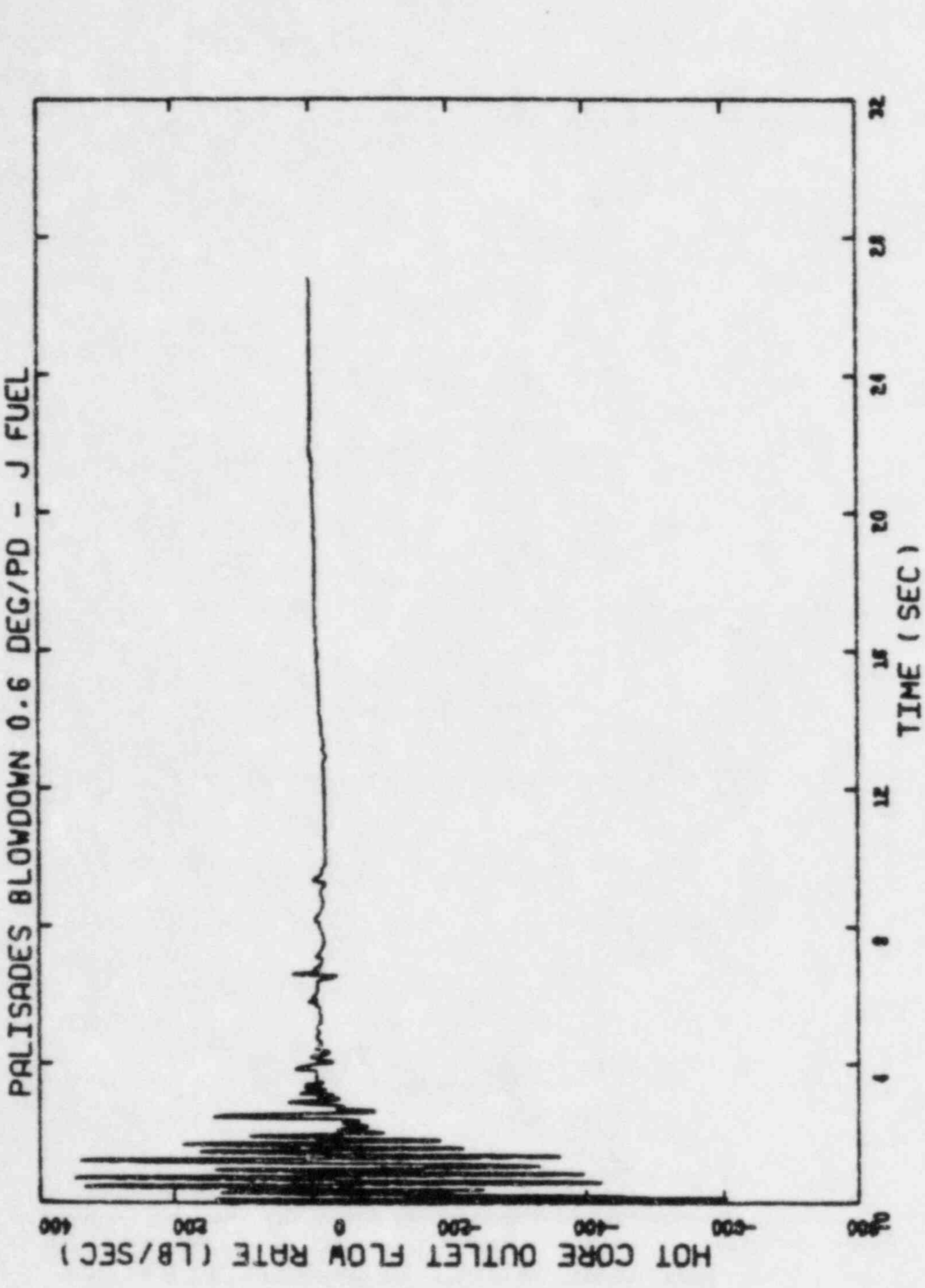


Figure 2.18 Hot Assembly Outlet Flow During Blowdown Period, 0.6 DEG/PD Break

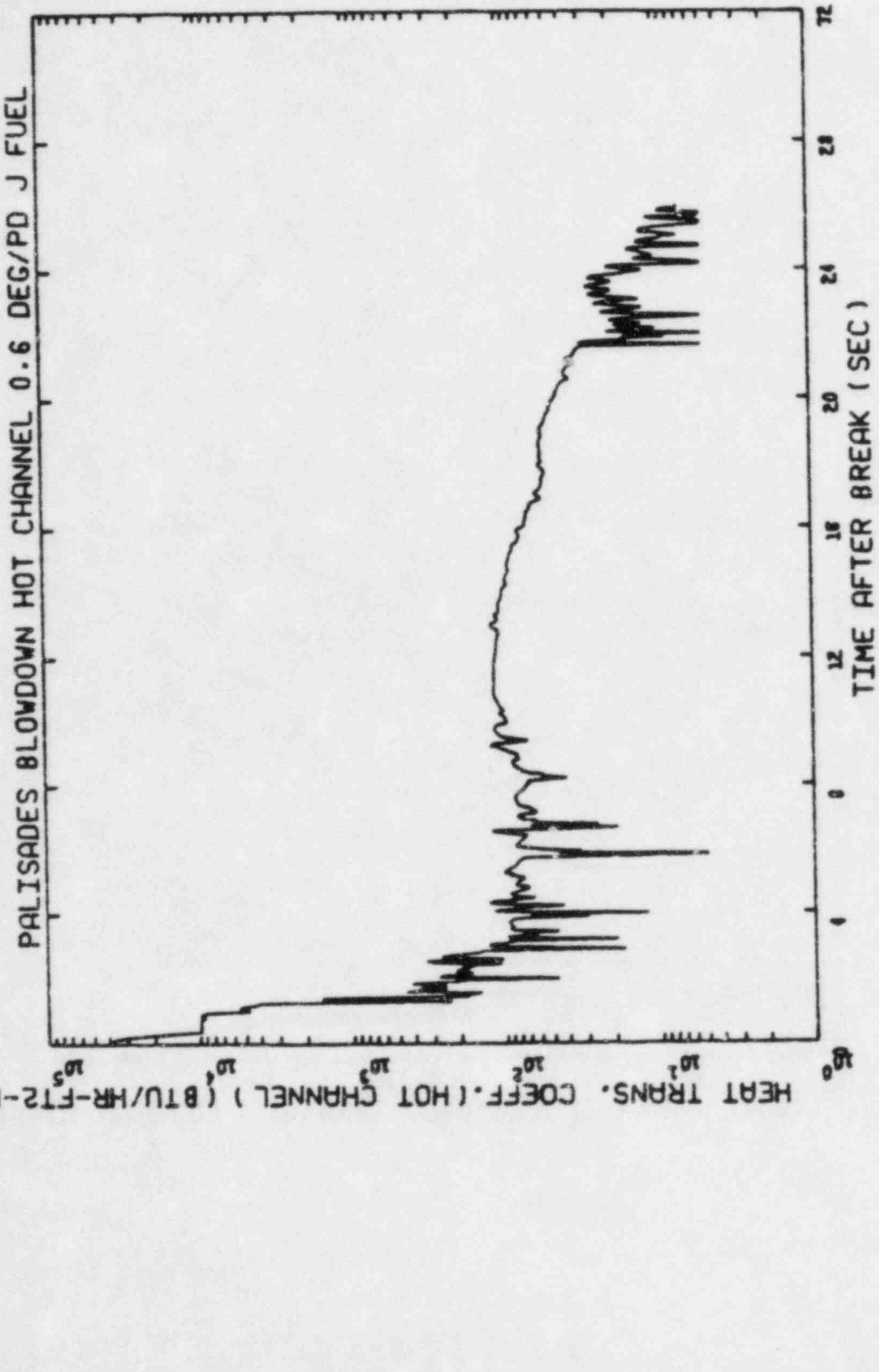


Figure 2.19 Heat Transfer Coefficient During Blowdown Period at PCT Location, 0.6 DEG/PD Break

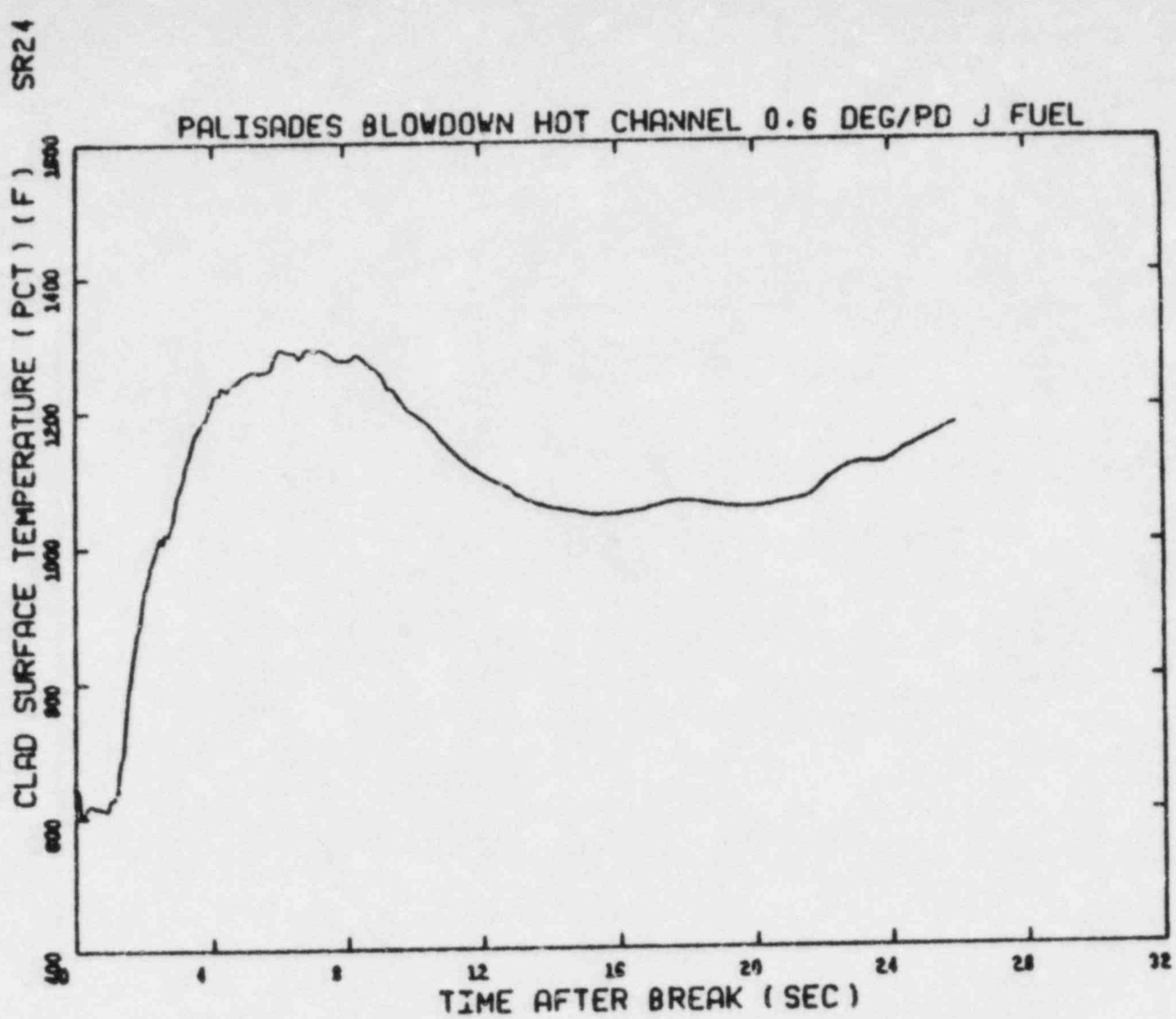


Figure 2.20 Clad Surface Temperature During Blowdown Period at PCT Location, 0.6 DEG/PD Break

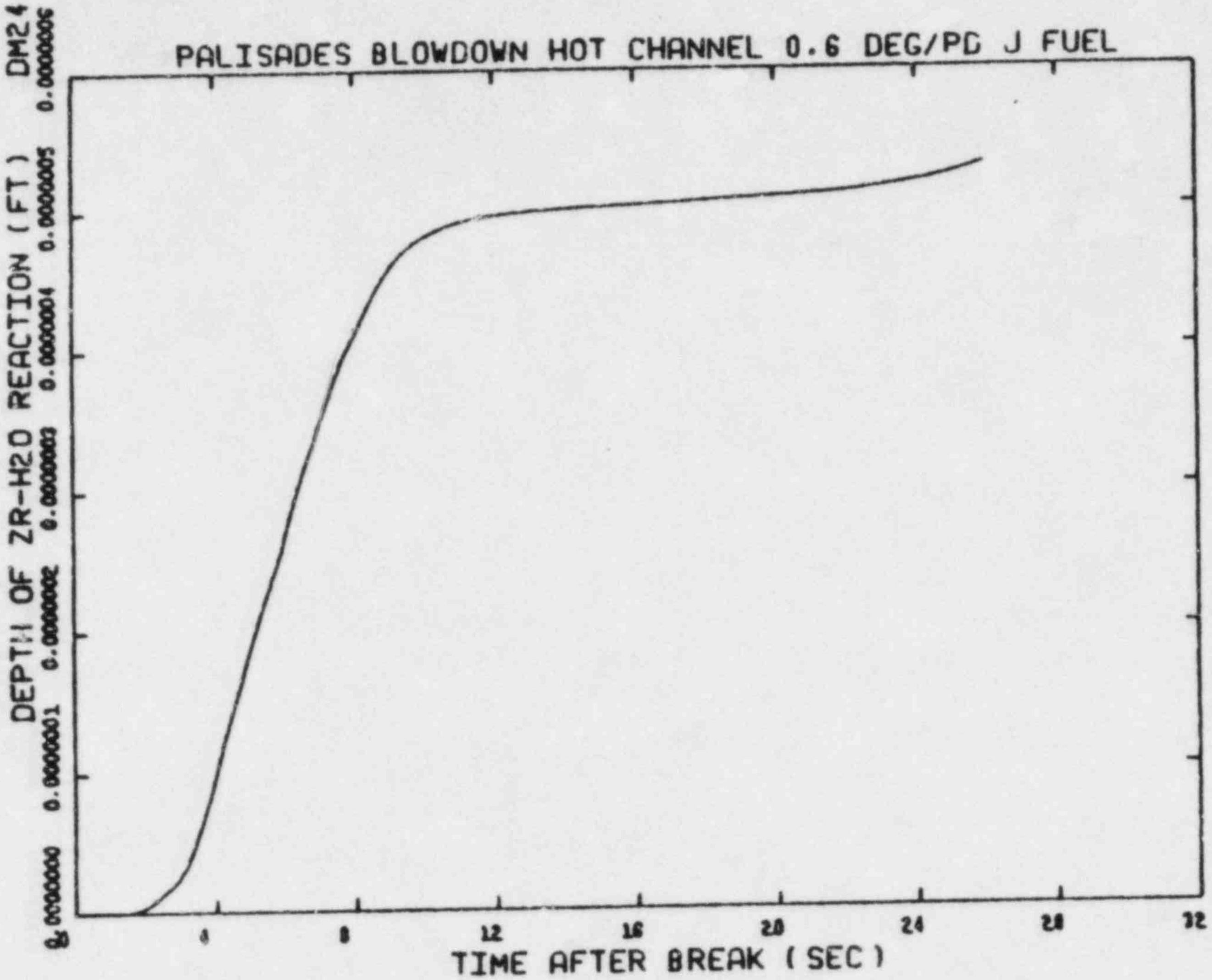


Figure 2.21 Depth of Metal-Water Reaction During Blowdown Period at PCT Location, 0.6 DEG/PD Break

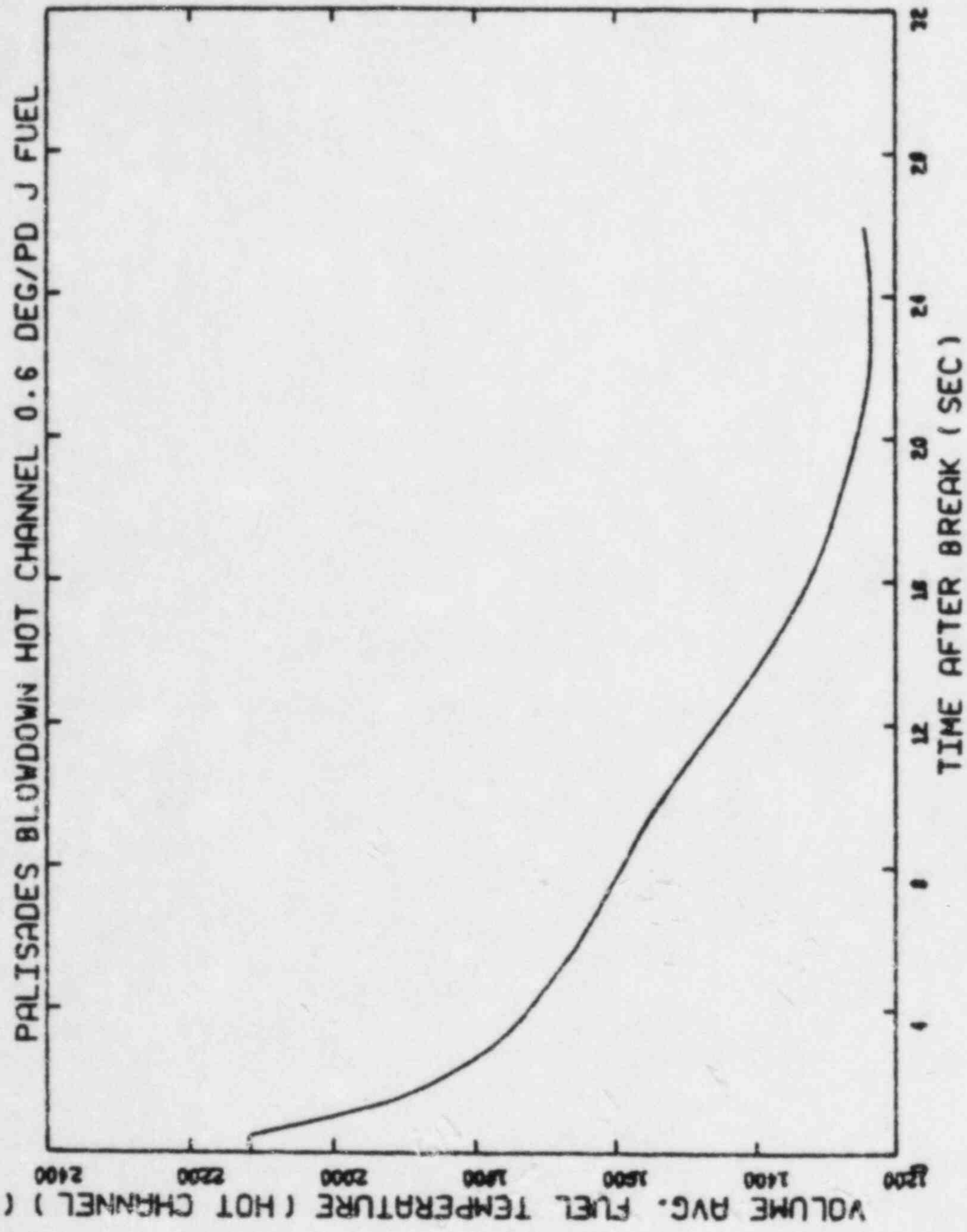


Figure 2.22 Average Fuel Temperature During Blowdown Period at PCT Location, 0.6 DEG/PD Break

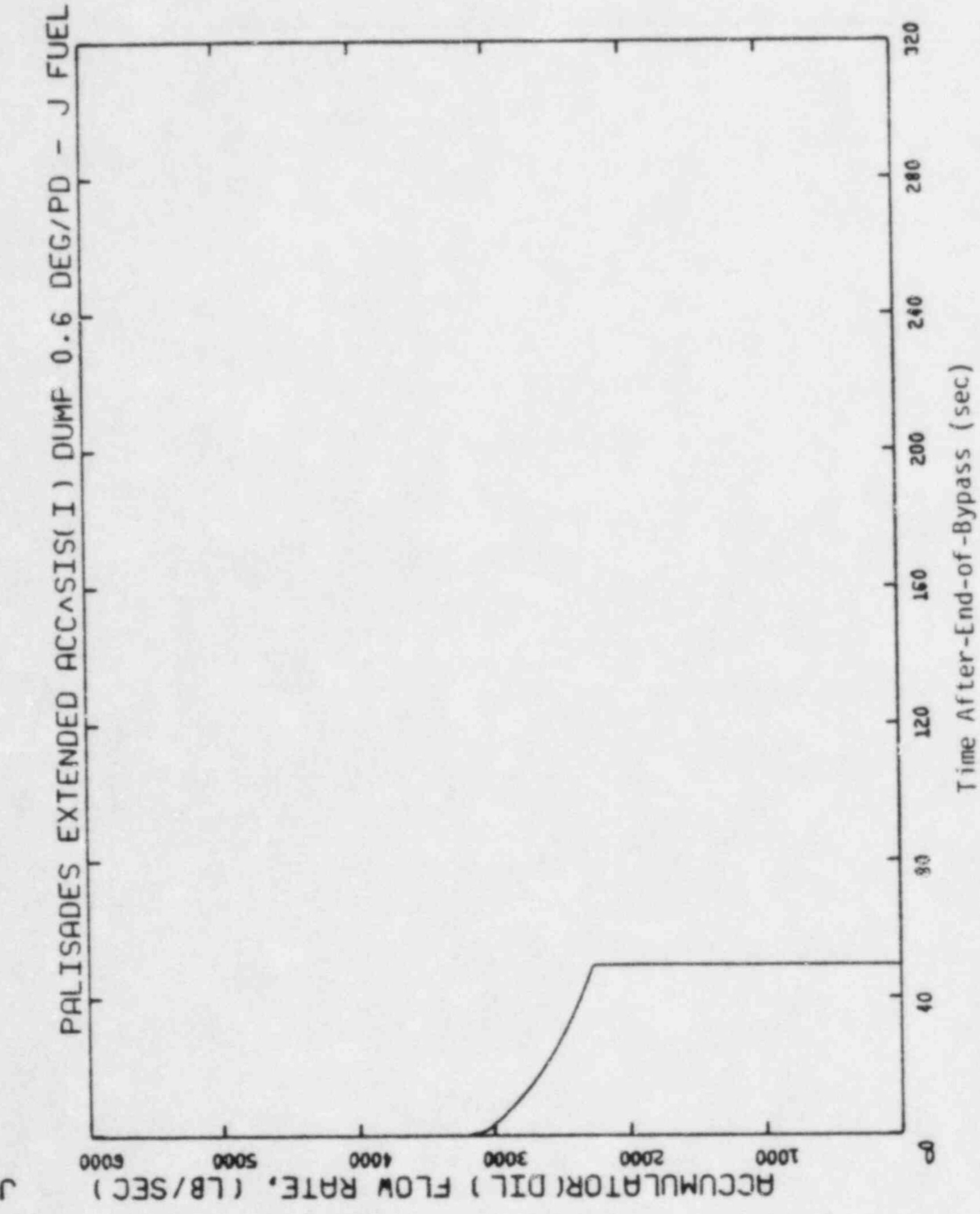


Figure 2.23 Two Intact Loop Accumulator Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

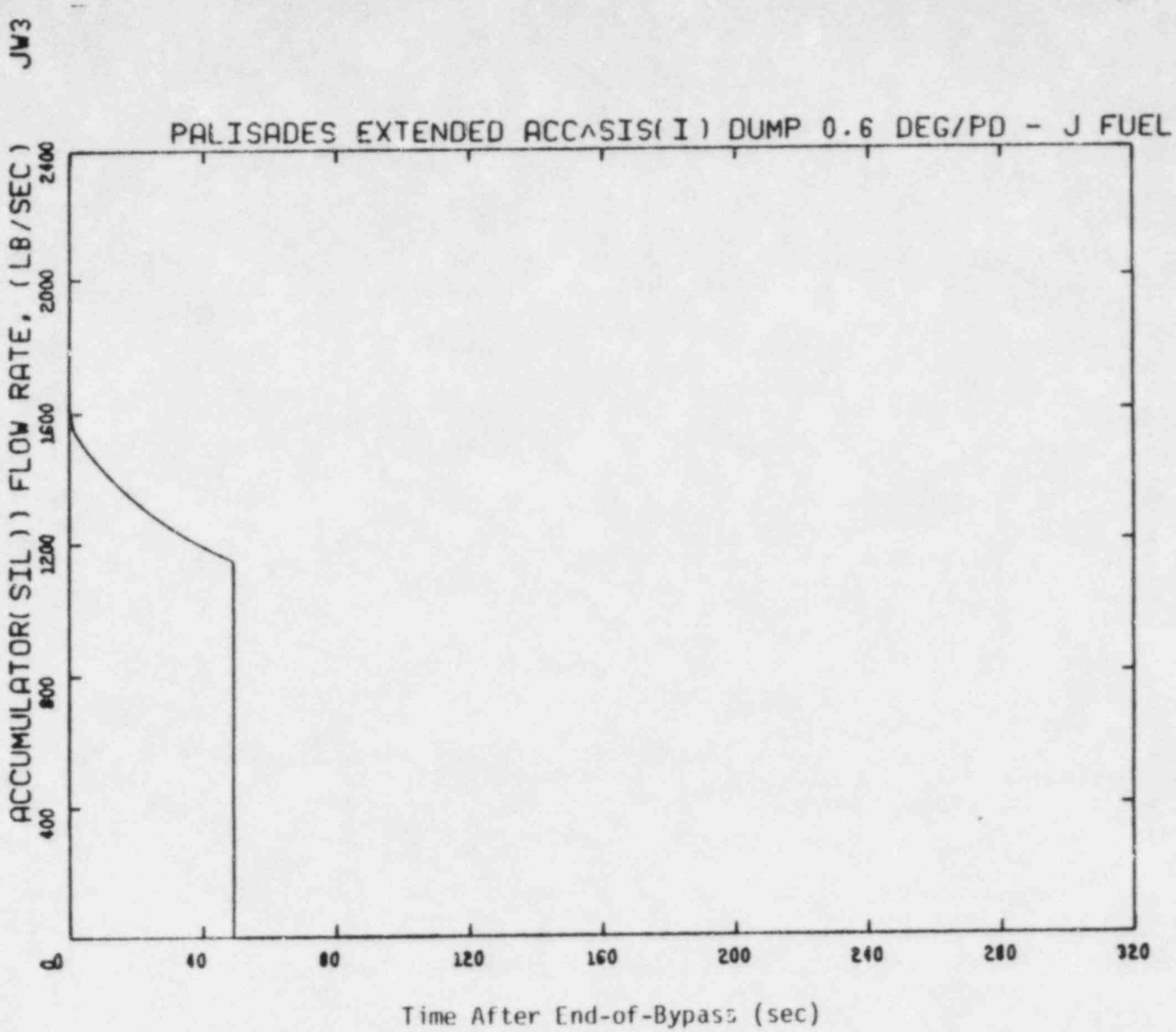


Figure 2.24 Single Intact Loop Accumulator Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

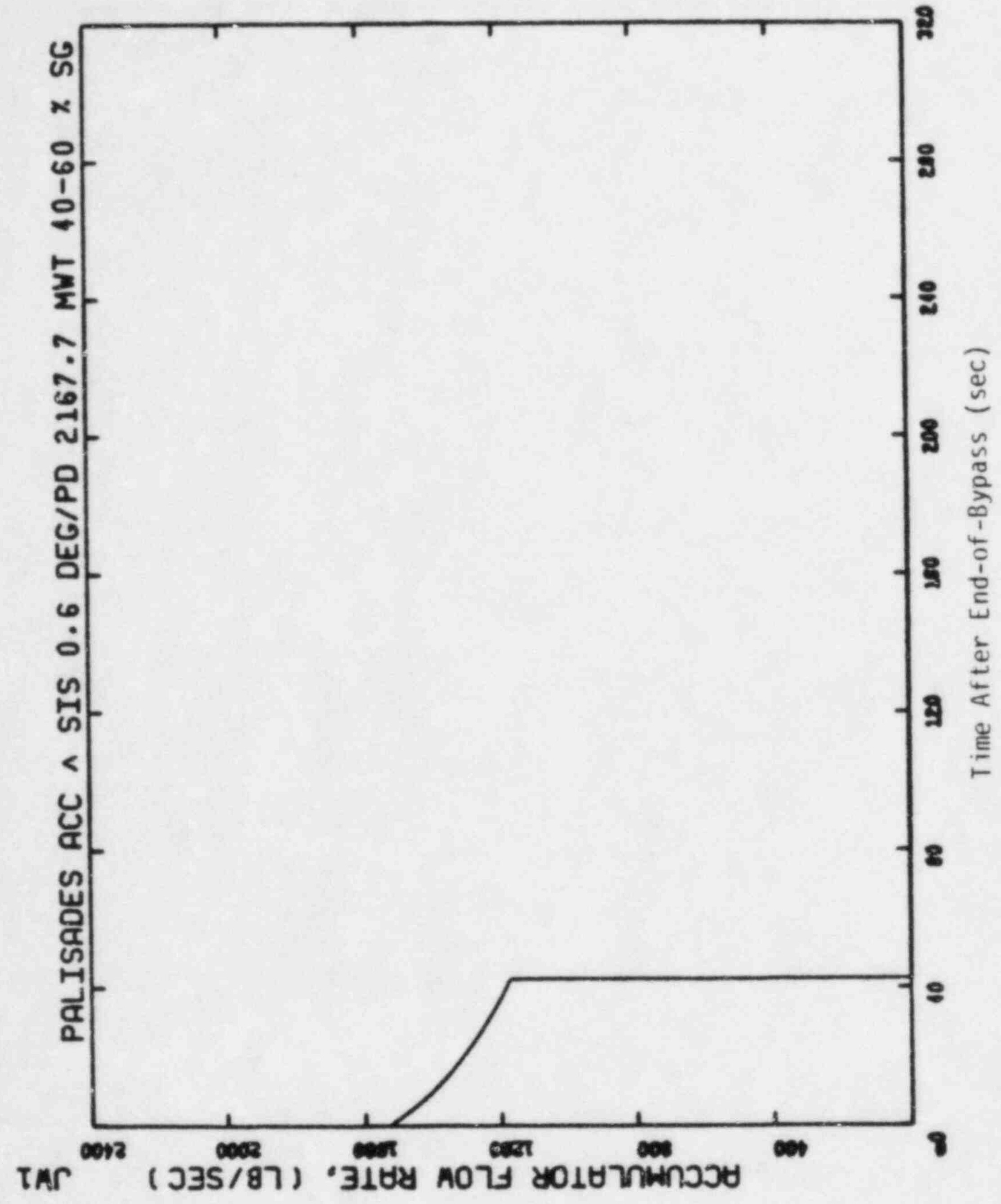


Figure 2.25 Broken Loop Accumulator Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

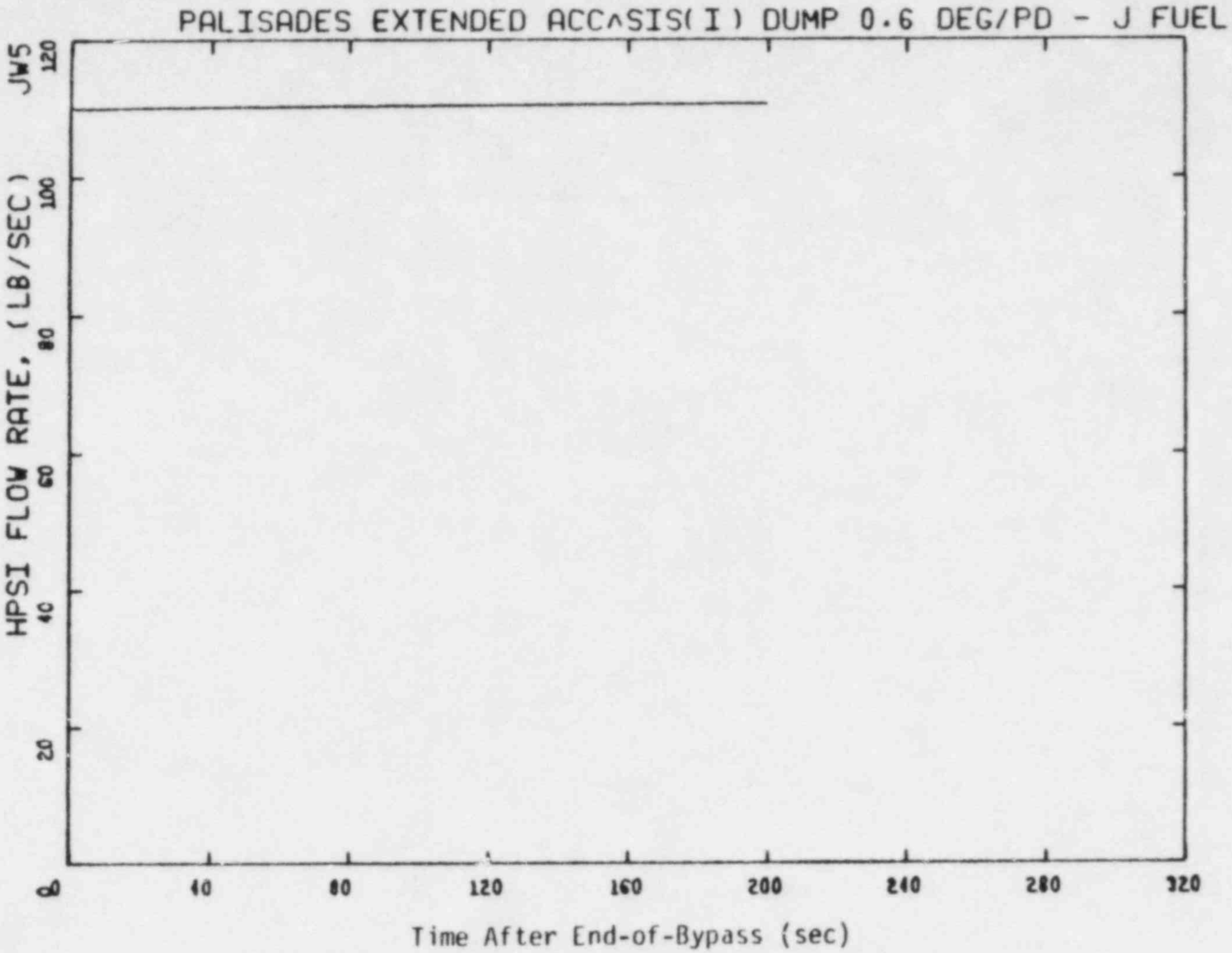


Figure 2.26 Intact Loops HPSI Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

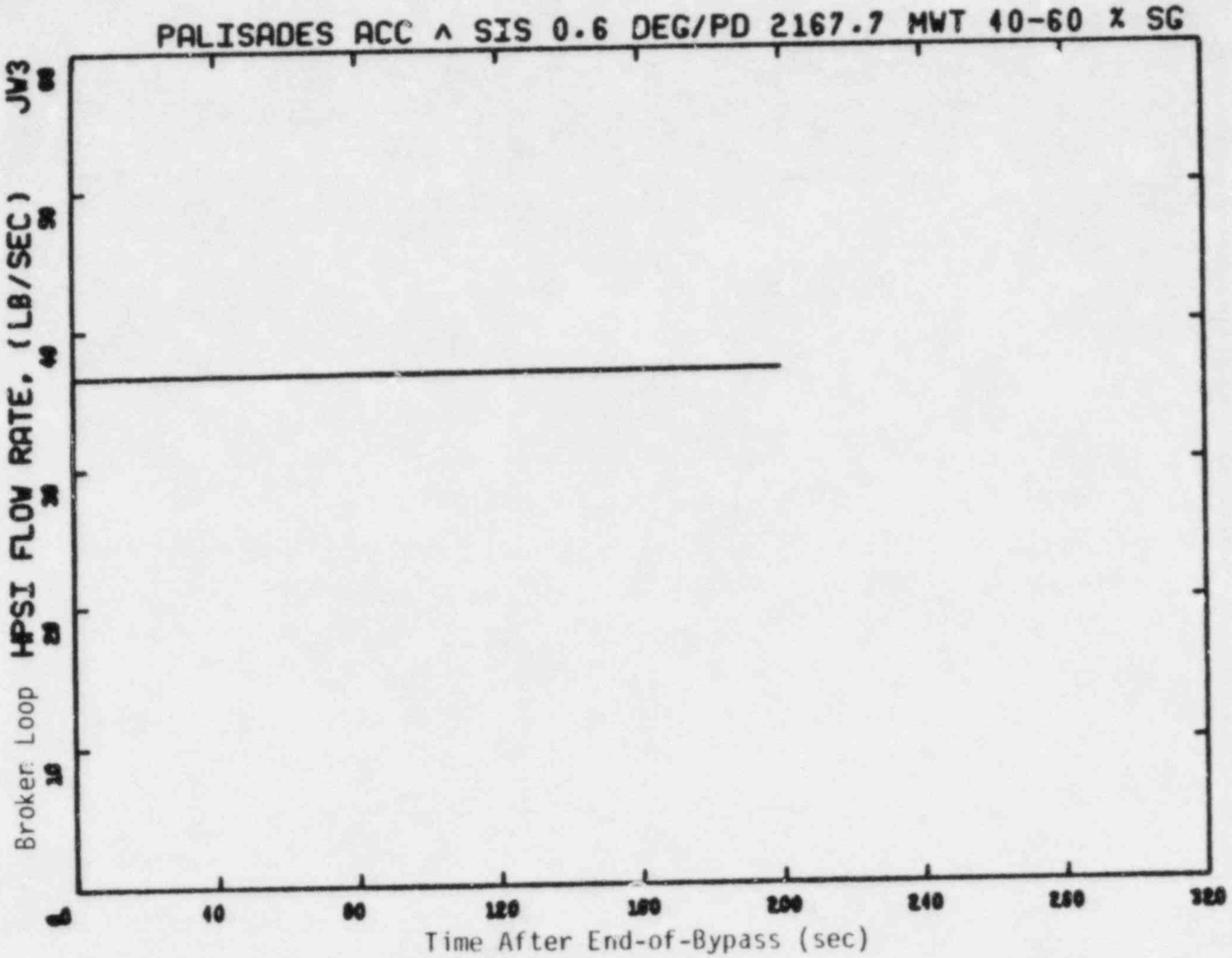


Figure 2.27 Broken Loop HPSI Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

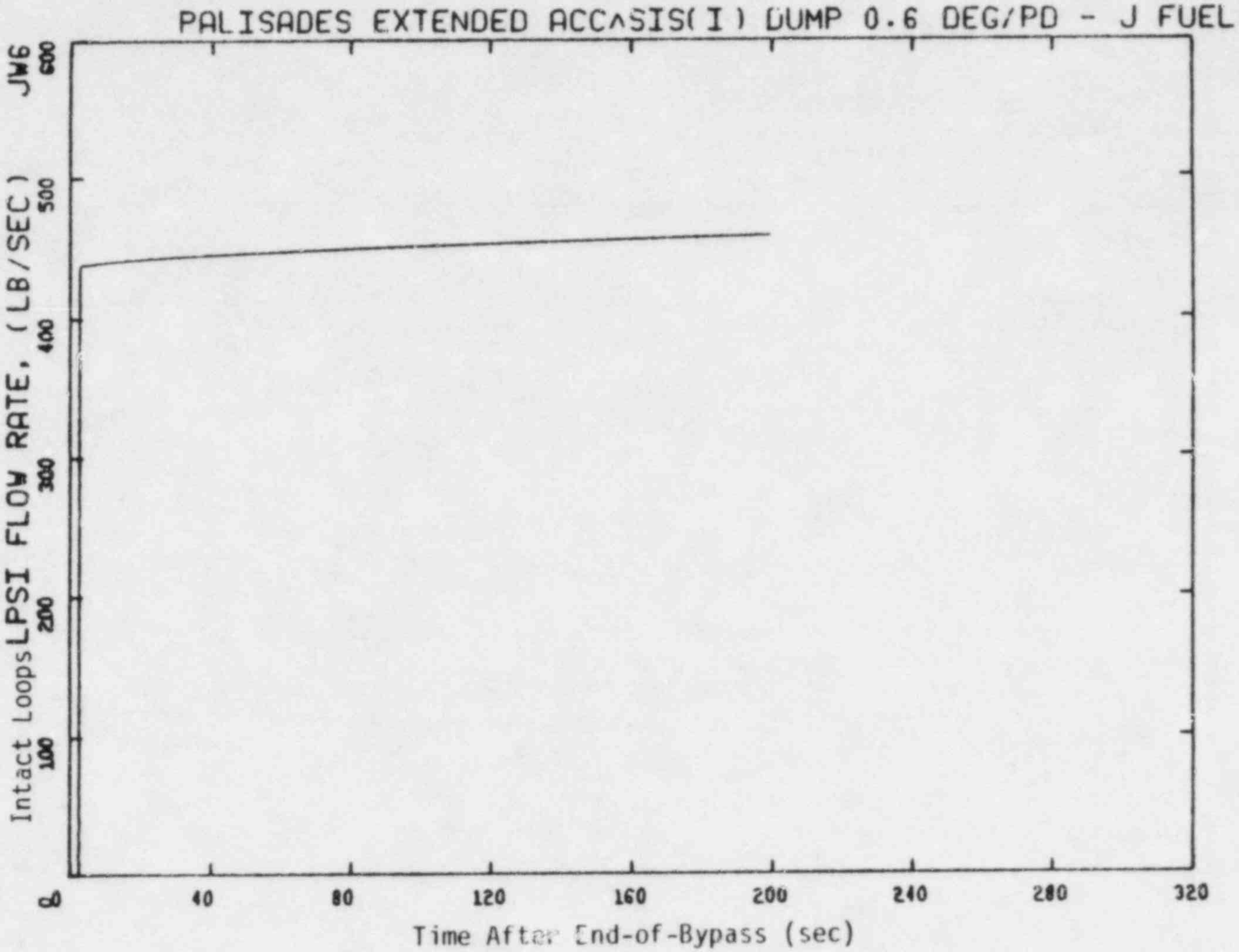


Figure 2.28 Intact Loops LPSI Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

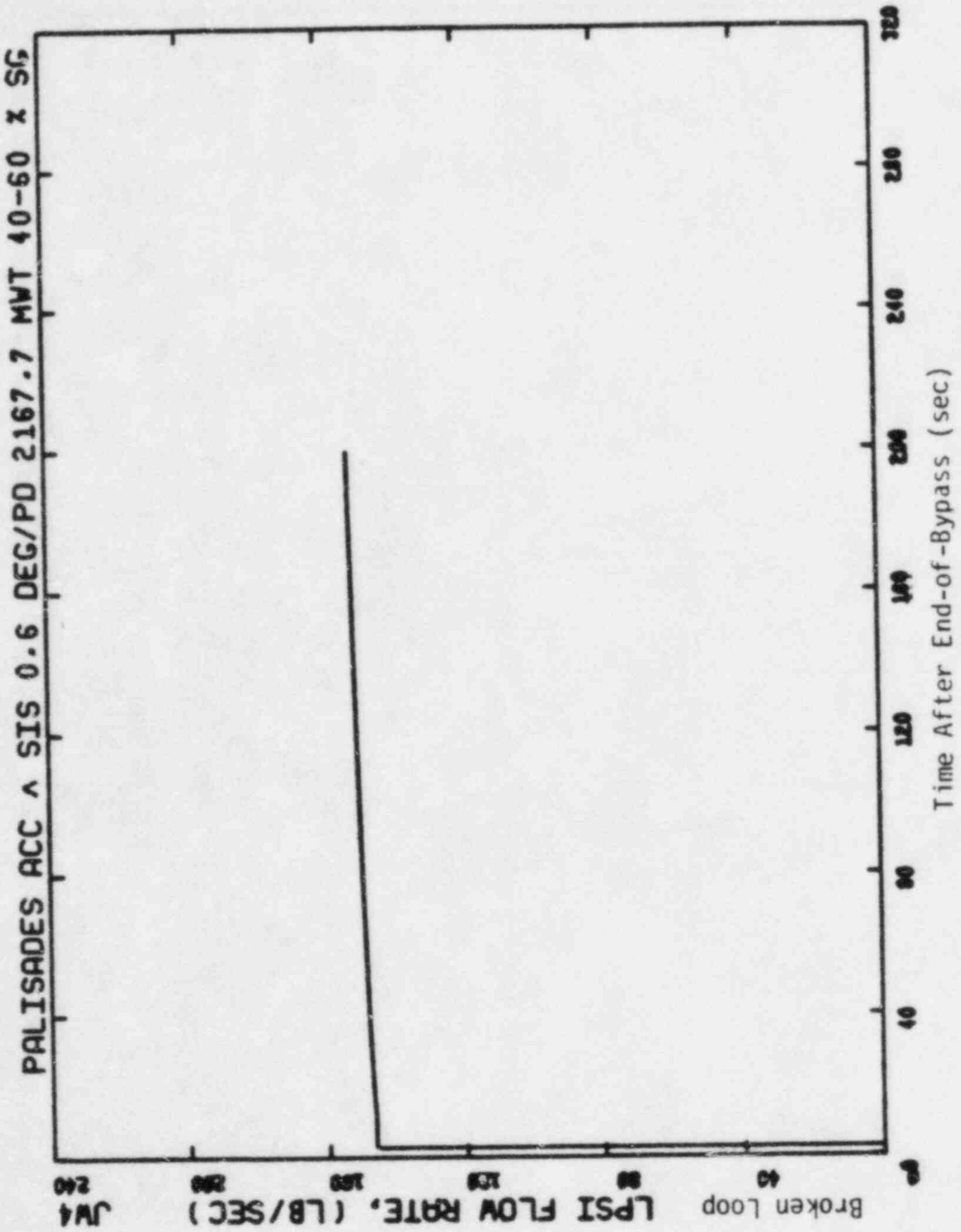


Figure 2.29 Broken Loop LPSI Flow During Refill and Reflood Periods, 0.6 DEG/PD Break

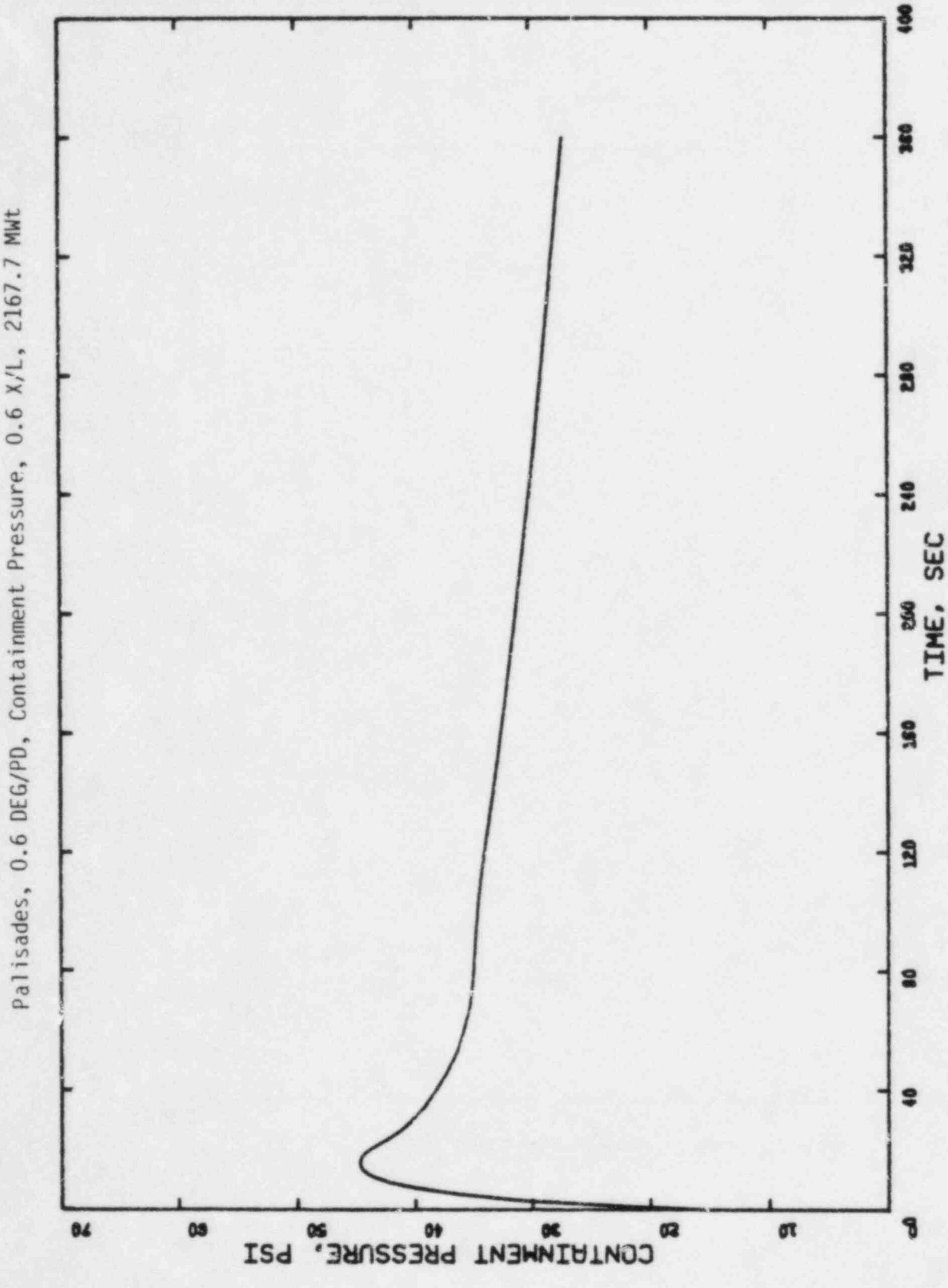


Figure 2.30 Containment Back Pressure, 0.6 DEG/PD Break

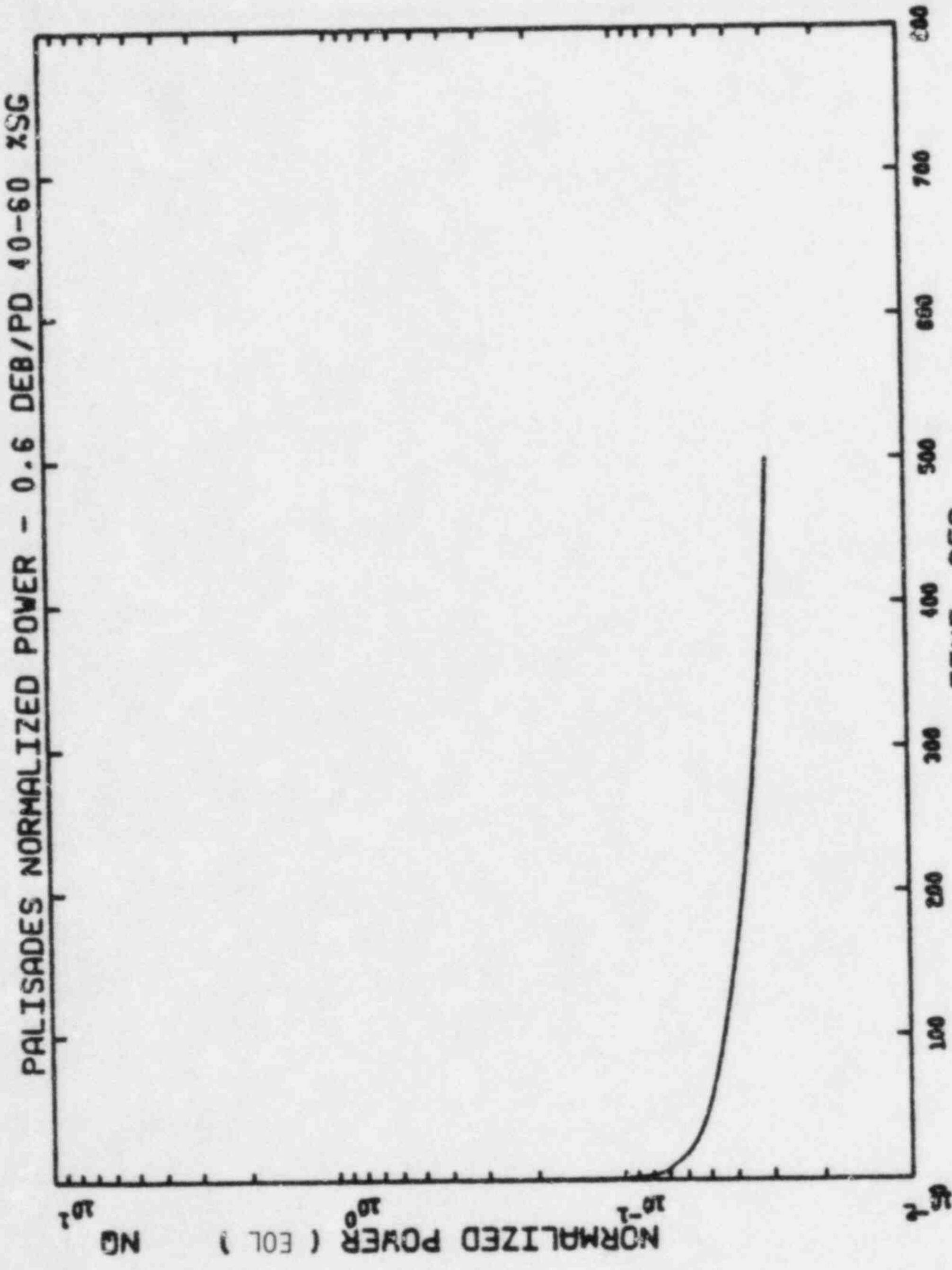


Figure 2.31 Normalized Power, 0.6 DEG/PD Break

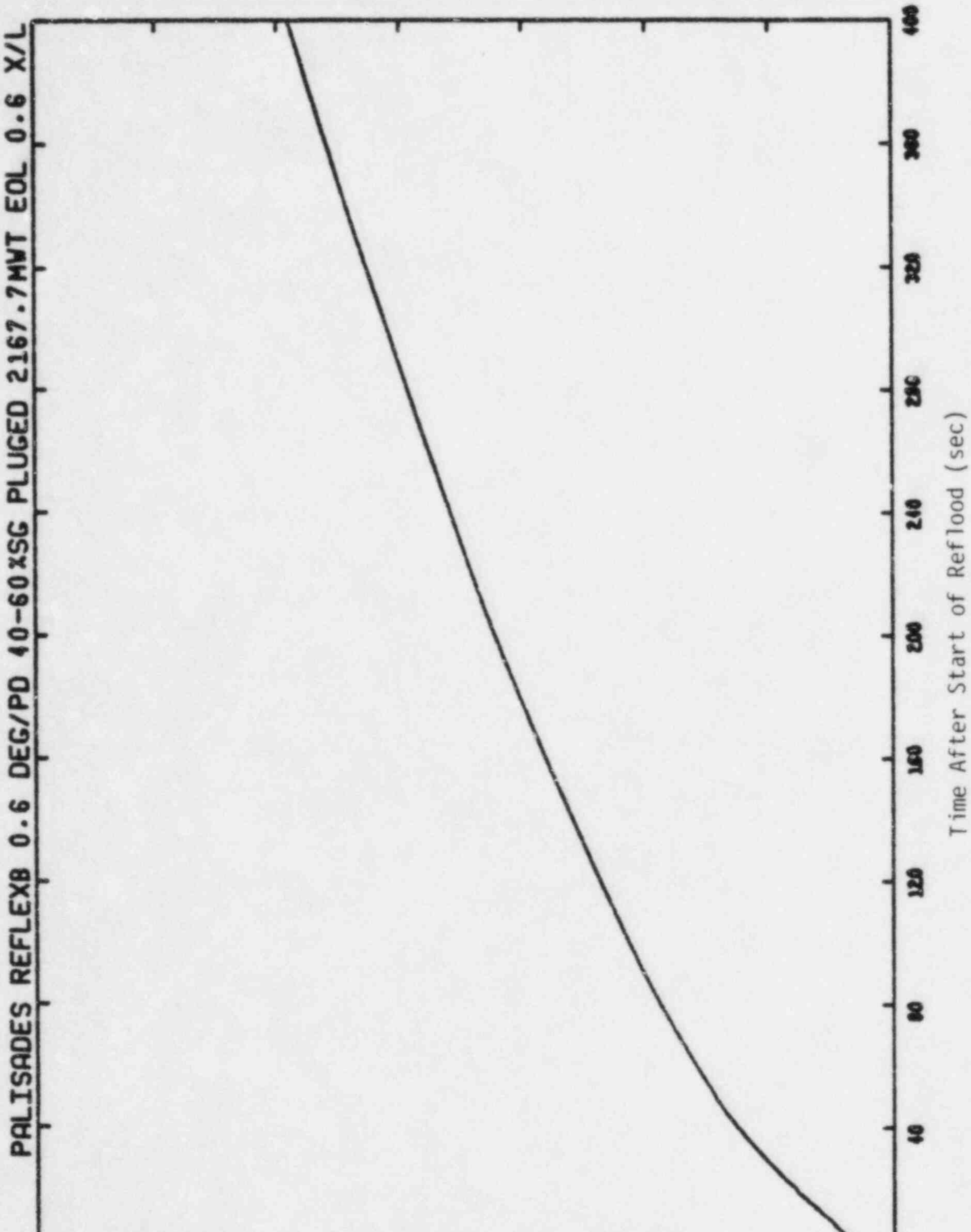
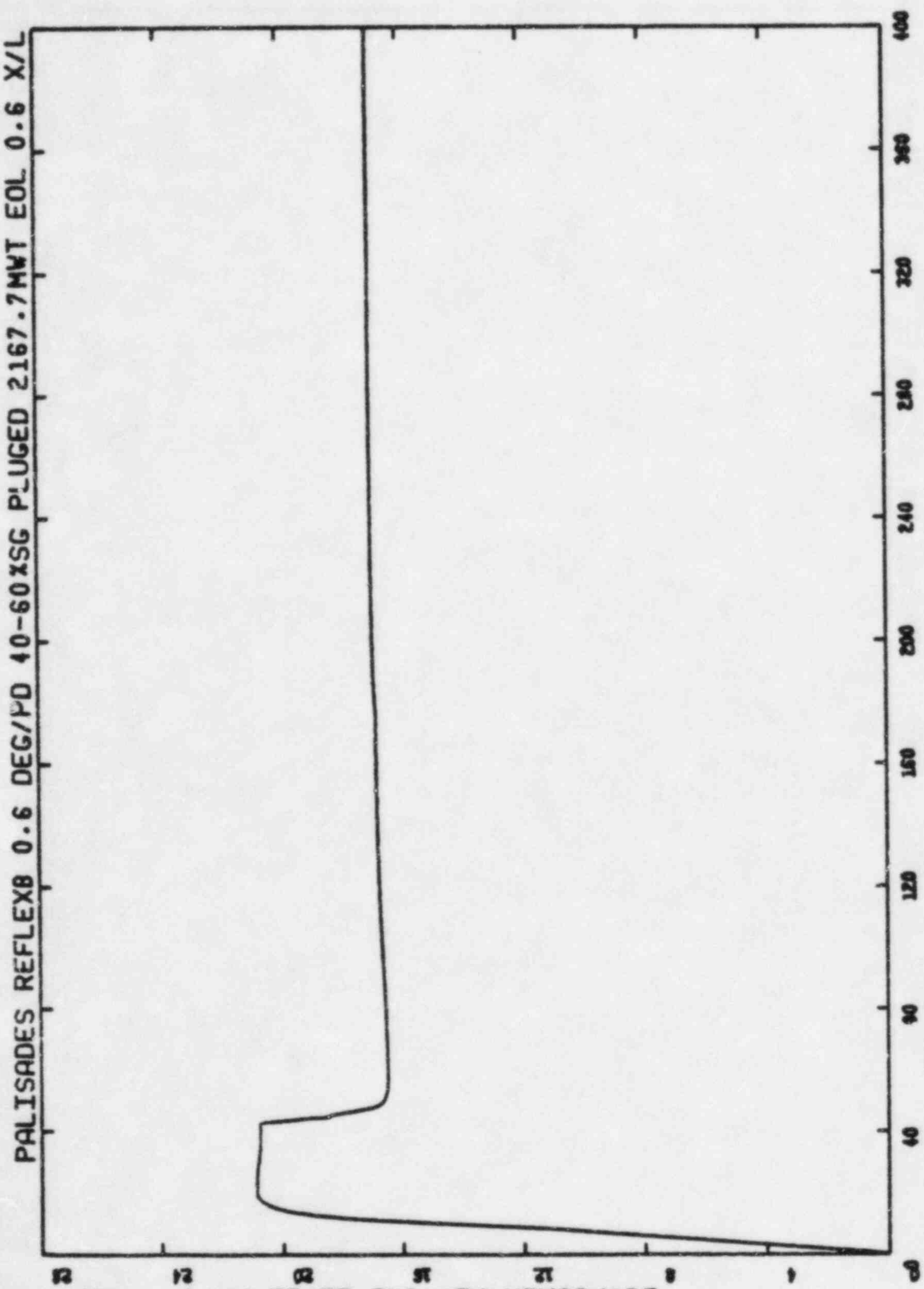


Figure 2.32 Reflood Core Mixture Level, 0.6 DEG/PD break



Time After Start of Reflood (sec)

Figure 2.33 Reflood Downcomer Mixture Level, 0.6 DEG/PD Break

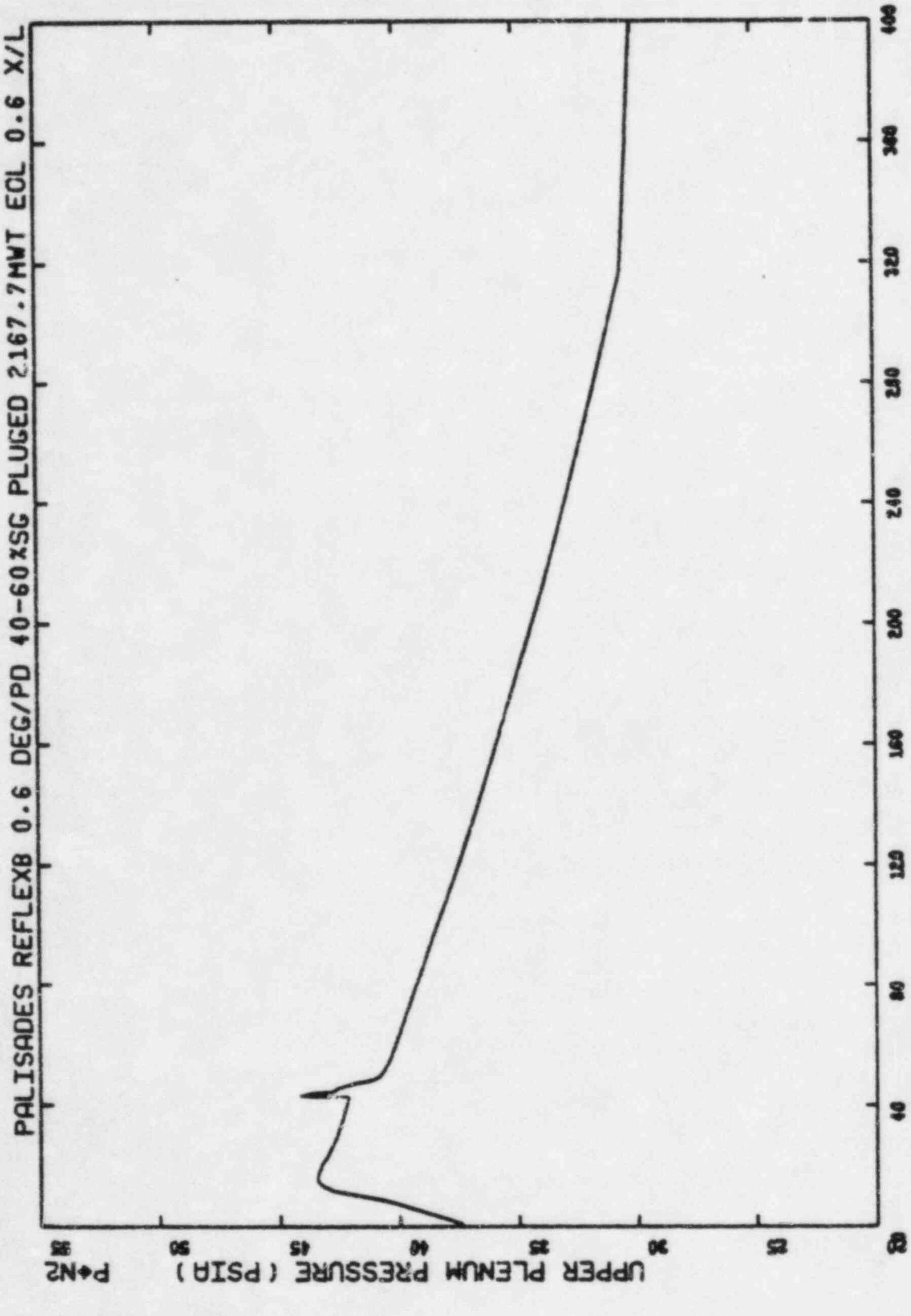


Figure 2.34 Reflood Upper Plenum Pressure, 0.6 DEG/PD Break

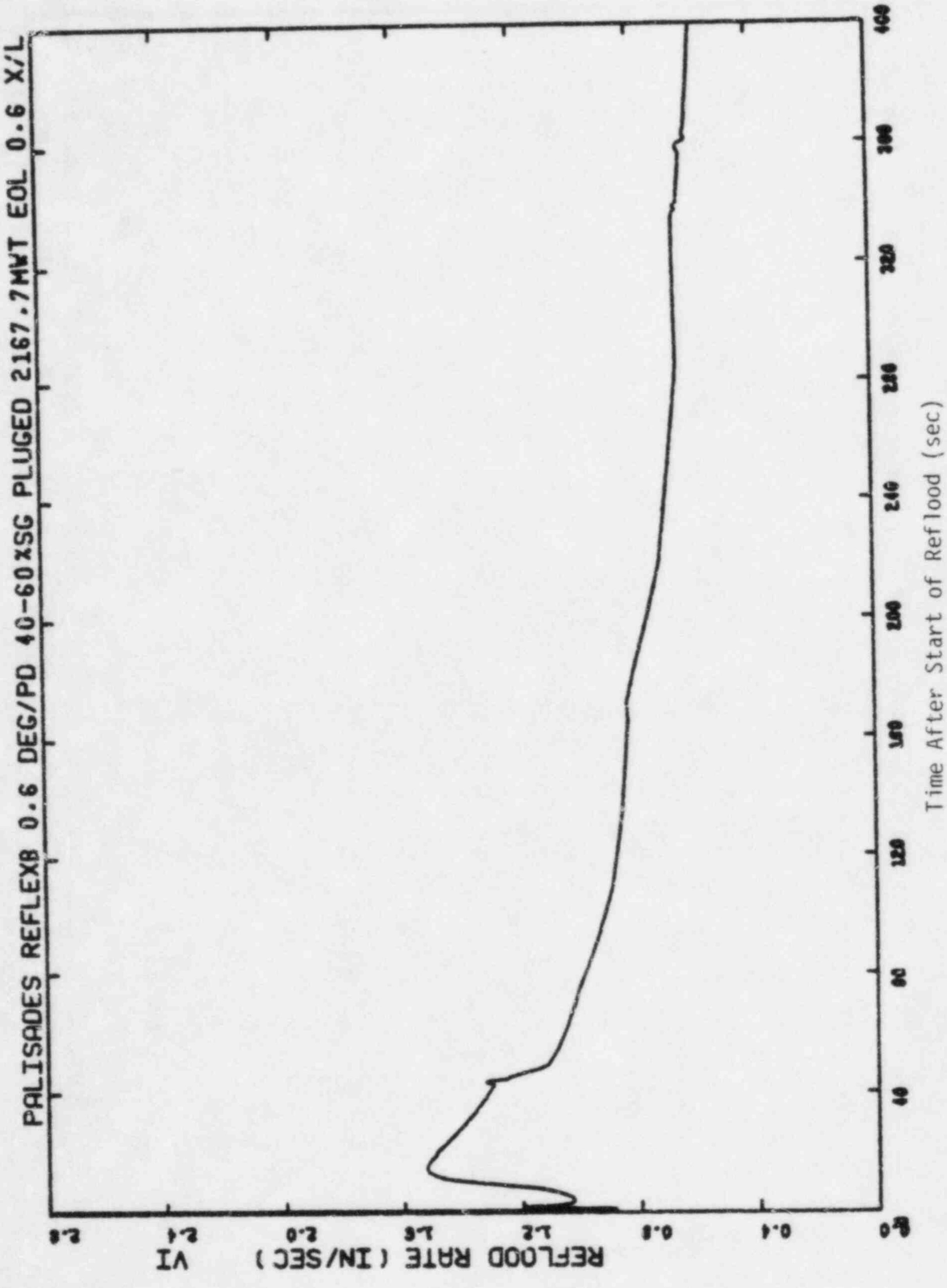


Figure 2.35 Core Flooding Rate, 0.6 DEG/PD Break

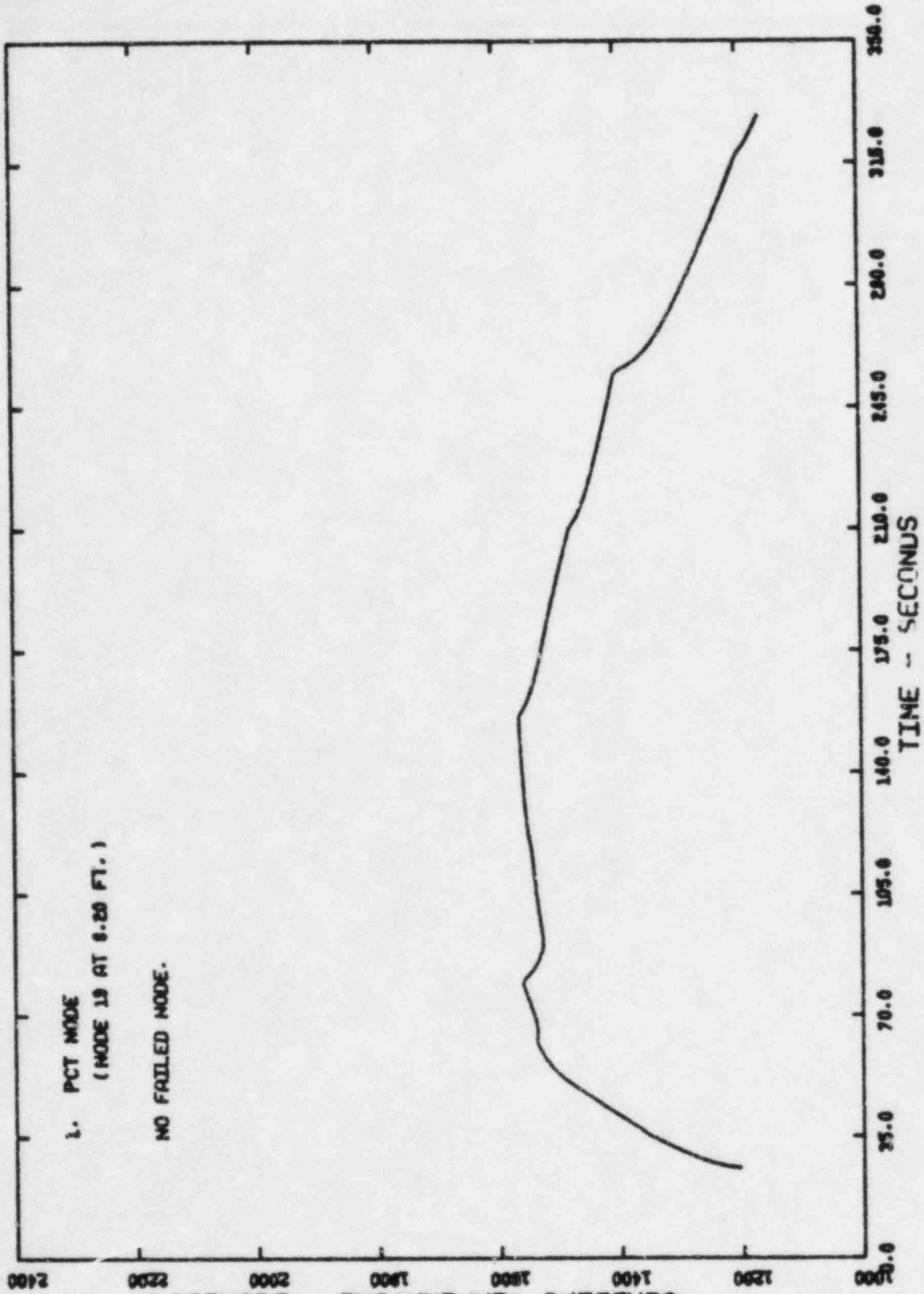


Figure 2.36 TOODEE2 Cladding Temperature vs. Time, 0.6 DEG/PD Break

3.0 CONCLUSION

For pipe breaks up to and including the double-ended severance of a reactor coolant pipe, the Emergency Core Cooling System for the Palisades nuclear reactor will meet the Acceptance Criteria as presented in 10 CFR 50.46 when calculated according to the requirements of Appendix K, with a total peaking factor of 2.76 ($F_Q{}^T$) and the operating parameters shown on Table 2.1. The criteria are as follows:

(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 percent 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 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.

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