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ANALYSIS OF AXIAL POWER DISTRIBUTION LIMITS
FOR THE PALISADES NUCLEAR REACTOR
AT 2530 MWt: SENSITIVITY STUDIES

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1.0 INTRODUCTION

LOCA ECCS analyses presented in XN-NF-77-24[1], XN-NF-78-16[2] and XN-NF-81-34[3] provide portions of the licensing bases[4] for the Palisades reactor. The analyses were performed for operation of the Palisades reactor at 2530 MWt, with 4175 steam generator tubes plugged, with a nominal core inlet temperature of 536.5°F, and with a nominal pressurizer pressure of 2060 psia. This document presents the results of sensitivity studies to the XN-NF-78-16 analyses which were performed to determine the impact on the LOCA limits for increased steam generator tube plugging (5000 total tubes plugged), increased nominal core inlet temperature (545°F), and increased nominal pressurizer pressure (2150 psia).

Results of a large break spectrum analysis were performed in 1977 for the operation of the Palisades reactor at 2530 MWt and reported in XN-NF-77-24[1]. The limiting break was identified to be a guillotine break in the pump discharge lire with a Moody discharge coefficient of 0.6 (the 0.6 DEG/PD break). The analyses were performed for the Exxon Nuclear Company (ENC) reload batch E fuel design using the ENC WREM-II PWR Evaluation Model[5,6,7]. The linear heat generation rate (LHGR) used in the analysis was 14.68 kw/ft (14.39 kw/ft times a 1.02 multiplier for power uncertainty), corresponding to a total peaking of 2.64 (F^T_Q) with an assembly radial peaking of 1.50 and a local bundle peaking of 1.224. The analyses assumed 4175 steam generator tubes were plugged, a 536.5°F core inlet temperature and the pressurizer pressure at 2060 psia.

Results were presented in XN-NF-78-16[2] which showed that the allowable LHGR decreased linearly as the axial power peak moved toward the top of the core. For a power shape peaked at 0.6 of core height, the LHGR used in the analysis was 15.28 Kw/ft (14.98 kw/ft times a 1.02 multiplier for power uncertainty), corresponding to a total peaking of 2.76 (F^T_Q) with an assembly radial peaking of 1.45 and a local bundle peaking of 1.224. The reduction in assembly radial peaking in the 1977 analysis(1) from 1.50 to the 1.45 value permitted the total peaking to be increased from 2.64 to 2.76.

Results of an exposure analysis were presented in XN-NF-81-34(3) which showed that the allowed LHGR limits for the ENC batch H fuel design is reduced at high burnup due to the effects that the higher fission gas releases have on the LOCA response of the fuel rod. High fission gas release resulted in greater ballooning of the fuel rod which resulted in a steam cooling heat transfer penalty during the reflood heatup portion of the LOCA transient. For rod average burnups less than 27.5 MWD/kg, the LHGR limit used in the analysis remained at the previous limit of 15.28 kw/ft, corresponding to a total peaking limit of 2.76.

2.0 SUMMARY

Three sensitivity studies have been performed to determine the effect of changed operating conditions in the Palisades reactor on the LOCA response for the ENC reload batch E fuel design. The analyses were performed using the ENC WREM-II PWR ECCS Evaluation Model. The analyses considered variations in operating conditions to those previously reported in XN-NF-78-16(2) for the 0.6 DEG/PD break with the axial peak located at 60 percent of core height and with a peak linear heat generation rate of 15.28 kw/ft. Changes in steady-state operating conditions relative to the base case included:

a. The temperature of the coolant entering the core was increased 8.5°F (545°F versus 536.5°F).

b. The pressure in the pressurizer was increased 90 psia (2150 psia versus 2060 psia).

c. An additional 825 steam generator tubes were plugged (total 5000 tubes versus 4175 tubes).

Results of the analyses, along with initial operating conditions, are given in Table 2.1. Differences in Peak Cladding Temperature (PCT) from the base case (PCT = 2081°F)(3) are given. In all cases the PCT results are within limits required by 10 CFR 50.46(8); PCT $\leq$2200°F, local metal oxidation $\leq$17%, and core average cladding oxidation $\leq$1%. Increasing the number of steam generator tubes plugged results in an increase in PCT as does increasing the pressurizer pressure. Increasing the core inlet temperature results in a decrease in PCT.

In general, the change in PCT due to increased steam generator tube plugging and coolant inlet temperature arises from their effect on the reflood rate. Increasing the number of steam generator tubes plugged increases the resistance to flow through the steam generators, resulting in a decrease in flooding rate. With higher coolant inlet temperature, more energy is released to the containment resulting in higher containment pressure which increases the flooding rate because of less steam binding in the primary recirculation loops during reflood. Peak clad temperature is strongly affected by reflood rate since the water flooding the core must remove the core energy during a postulated LOCA. The change in PCT with increased pressurizer pressure is due to a change in core thermal hydraulics during blowdown. For the 2150 psia case, the flow in the core was degraded sufficiently to result in poorer heat transfer during blowdown and ultimately a higher PCT.

TABLE 2.1

Results of LOCA ECCS Sensitivity Studies

Pressurizer Pressure	Core Inlet Temperature	Number Steam Generator Tubes Plugged	Difference in PCT From Base
2060 psia	536.5°F	4175	0°F
2060 psia	536.5°F	5000	+25°F
2060 psia	545.0°F	4175	-18°F
2150 psia	536.5°F	4175	+54°F

3.0 ANALYSIS RESULTS AND DISCUSSION

This report presents the results of three sensitivity studies which have been performed to determine the impact on the LOCA performance of ENC reload fuel to changes in operating conditions at the Palisades reactor. Changes in operating conditions considered in the analyses included:

a. The temperature of the core inlet coolant was increased 8.8°F.

b. The pressure in the pressurizer was increased 90 psia.

c. The number of steam generator tubes plugged was increased by 825 tubes.

The results are compared to a base case analysis presented in XN-NF-78-16[2]. The base case analysis assumed operation of the Palisades reactor at 2530 MWt with:

a. A core inlet coolant termperature of 536°F.

b. The pressurizer pressure set at 2060 psia.

c. 4175 steam generator tubes plugged

d. A peak linear heat generation rate of 15.28 Kw/ft (14.98 kw/ft times a 1.02 multiplier for power uncertainty), corresponding to a total peaking (F^T_Q) of 2.76 with an assembly radial peaking of 1.45 and a local bounding peaking of 1.224.

e. ENC batch E fuel at BOL fuel conditions

f. The axial power peaked located at 0.6 of core height.

3.1 LOCA ANALYSES MODEL

The analytical techniques used are in compliance with Appendix K of 10 CFR 50[8], and are as described in XN-75-41, Volumes I and II, and supplements[5], with ENC WREM-II model updates as described in XN-76-27[6].

The revised ENC nucleate boiling lockout as described in XN-76-44[7] was also used in the blowdown analysis.

The ENC WREM-II ECCS evaluation model[5,6,7] was used to perform the analyses. The model consists of the following computer codes: GAPEXX[9] code for initial rod stored energy and internal gas inventory; RELAP4-EM[10] for the system blowdown, hot channel blowdown and reflood calculations; CONTEMPT-LT22 as modified in CSB6-1[11] for computation of containment back pressure; and TOODEE2[12] 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 "junctions". The system nodalization is depicted in Figure 3.1. The pump performance characteristics of Combustion Engineering pump were used in the analysis[13]. Asymmetric steam generator tube plugging is assumed such that Loop 1 has the greater plugging. 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 for the base case are given in Table 3.1. Systems parameters used in the sensitivity study are given in Table 3.2. Fuel design data are given in Table 3.3 for ENC reload E, F, G, H, I, and J.

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 2.2 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.

3.2 IDENTIFICATION OF CAUSE 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 te 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 correspondinq 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(8). 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 262.5 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.

3.3 RESULTS

Table 3.4 presents the timing and sequence of events for the base case and three sensitivity analyses. In general, the timing for major events are comparable. Results of the hot channel analysis for the base case are plotted in Figure 3.3 to 3.8. Blowdown, hot channel, reflood and heatup plots for the 545°F inlet temperature case are shown in Figure 3.9 to 3.27. Blowdown, hot channel, reflood and heatup plots for the 2150 psia pressurizer case are shown in Figures 3.28 to 3.42. Blowdown, hot channel, reflood and heatup plots for the increased steam generator tests plugging case are shown in Figure 3.47 to 3.61.

Difference in PCTs for the four cases are small and are shown in Table 2.1. The PCT decreased with increased inlet coolant temperature primarily due to greater energy released to the containment which resulted in a higher system pressures during reflood and higher reflood rates. The PCT increased with increased steam generator tube plugging primarily due to the reduction in reflood rates resulting from the increased flow resistance in the steam generator. The PCT increased with increased pressurizer pressure primarily due to a degradation of flow in the core during the blowdown portion of the transient.

Table 3.1

Palisades System Data: Base Case

Primary Heat Output, MWt		2530.
Primary Coolant Flow, Mlbm/hr		124.0
Primary Coolant Volume, ft^3		10,530.
Design Pressure, psia		2060.
Inlet Coolant Temperature, ^oF		536.5
Reactor Vessel Volume, ft^3		4790.
Pressurizer Volume, Total, ft^3		1500.
Pressurizer Volume, Liquid, ft^3		800.
Accumulator Volume, Total ft^3 (each of four)		2011.
Accumulator Volume, Liquid, ft^3		1150.
Accumulator Pressure, psia		215.
Steam Generator Heat Transfer Area, ft^2	Loop 1	5.8×10^4
	Loop 2	6.4×10^4
Steam Generator Tubes Plugged	Loop 1	2407
	Loop 2	1768
Steam Generator Secondary Flow, lbm/hr		11.2×10^6
Steam Generator Secondary Pressure, psia		735.
Reactor Coolant Pump Head, ft		240. - 245.
Reactor Coolant Pump Speed, rpm		880.
Moment of Inertia, lbm-ft^2/rad		98,000.
Cold Leg Pipe, I.D., in.		30.
Hot Leg Pipe, I.D., in.		42.
Pump Suction Pipe, I.D., in.		30.

Table 3.2

Palisades System Data Used in Sensitivity Studies

	Base	2150 psia Case	545°F Case	Tube Plugging Case
Core Inlet Temperature (°F)	536.5	536.5	545.0	536.5
Pressurizer Pressure (psia)	2060	2150	2060	2060
Steam Generator Plugging (No. Tubes)				
Loop 1	2407	2407	2407	2820
Loop 2	1768	1768	1768	2180
Steam Generator Inside Heat Transfer Area (ft^2)				
Loop 1	5.8×10^4	5.8×10^4	5.8×10^4	5.4×10^4
Loop 2	6.4×10^4	6.4×10^4	6.4×10^4	6.0×10^4
Primary Coolant Flow (Mlb/hr)	124.3	123.1	125.3	122.5

Table 3.3

Fuel Design Data

Fuel Design	Reload Designs H, I, J	Reload Designs E, F, G
Cladding Data	0.417	0.415
Fuel Assembly Rod Pitch, in.	0.550	0.550
Fuel Assembly Pitch, in.	8.485	8.485
Fueled (Core) Height, in.	131.8	131.8
Fuel Heat Transfer Area, ft^2	50878	50630
Core Total Flow Area, ft^2	56.76	57.20

Table 3.4

PALISADES LARGE BREAK EVENT TIMES

E FUEL AT BOL AND 2530 MWT

Event	Time (seconds)			
	0.6 DEG/PD			
	Base Case	2150 psia Case	545°F Inlet	5000 SG Tubes Plugged
Start	0.0	0.0	0.0	0.0
Initiate break	0.1	0.1	0.1	0.1
Safety Injection Signal	0.8	0.8	0.8	0.8
Accumulator Injection, Broken Leg	13.0	13.3	13.5	13.0
Accumulator Injection, Intact Leg	16.7	16.6	16.5	16.5
Accumulator Injection, Intact Loop	16.7	16.6	16.6	16.5
End-of-Blowdown (Break Flow Reversal)	20.9	20.5	20.5	20.6
End-of-Bypass	25.56	25.45	25.44	25.43
Bottom of Core Recovery	44.27	43.76	43.77	43.74
Accumulator Empty, Intact Leg	74.1	73.7	74.3	74.4
Accumulator Empty, Intact Loop	74.3	73.9	74.6	74.6
Safety Pump Injection HPIS	21.8	21.8	21.8	21.8
Safety Pump Injection LPIS	28.8	28.8	28.8	28.8
Peak Clad Temperature Reached	241.6	206.6	218.0	231.0

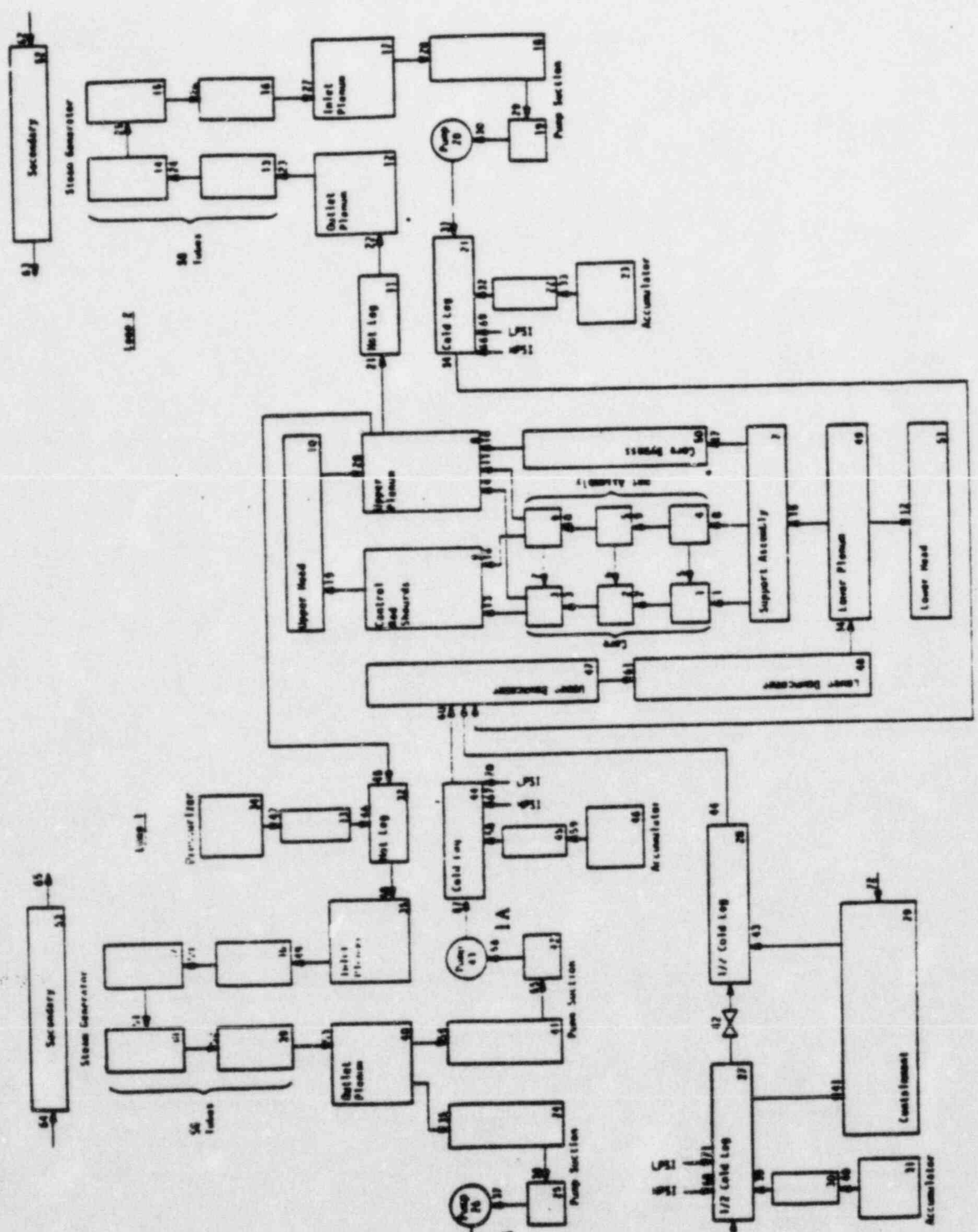

Figure 3.1 RELAP4/EM Blowdown System Nodalization For Palisades Nuclear Plant

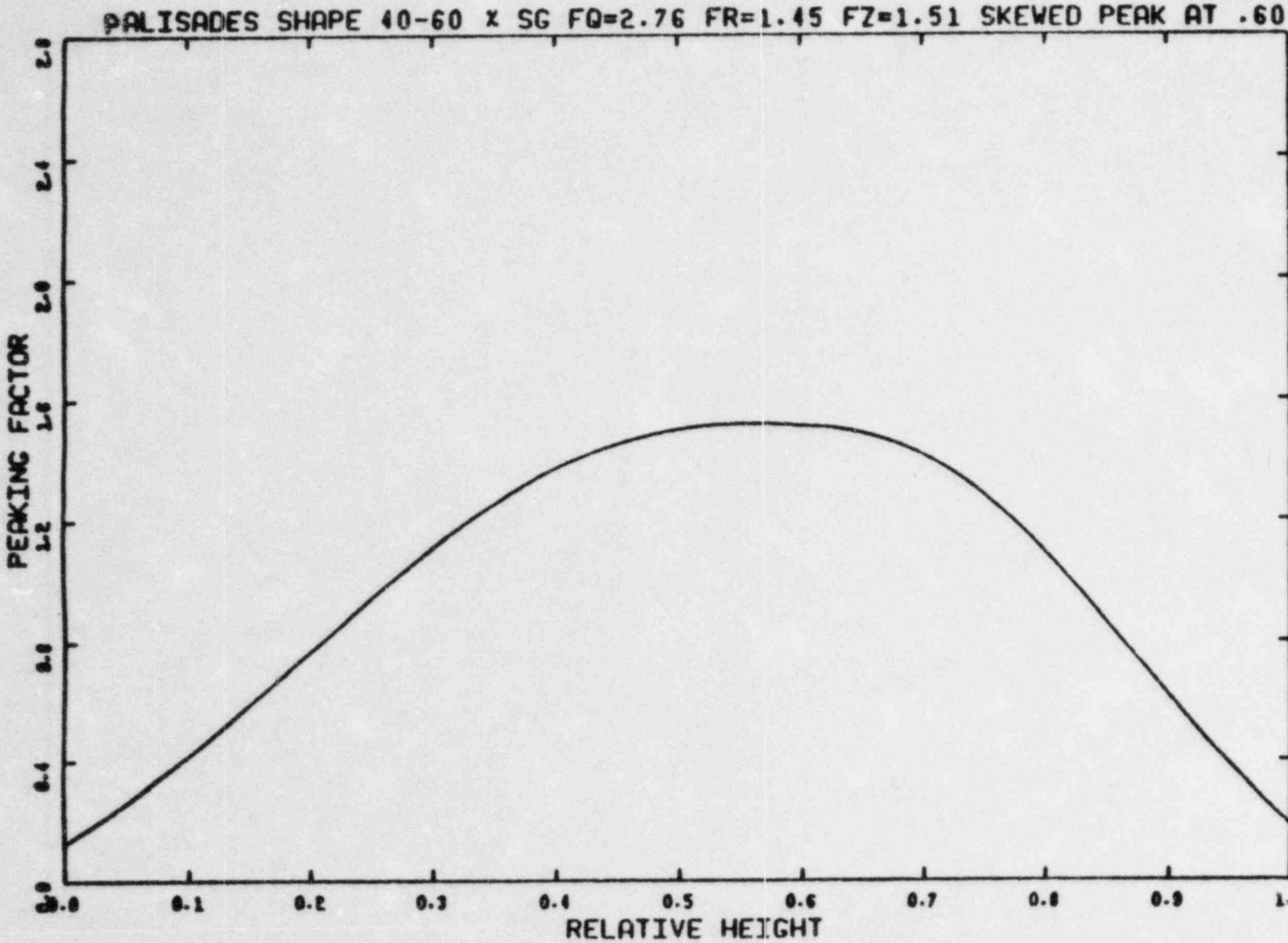

Figure 3.2 Axial Power Profile With Peak at X/L=0.6 and with Skewing Factor of 1.0

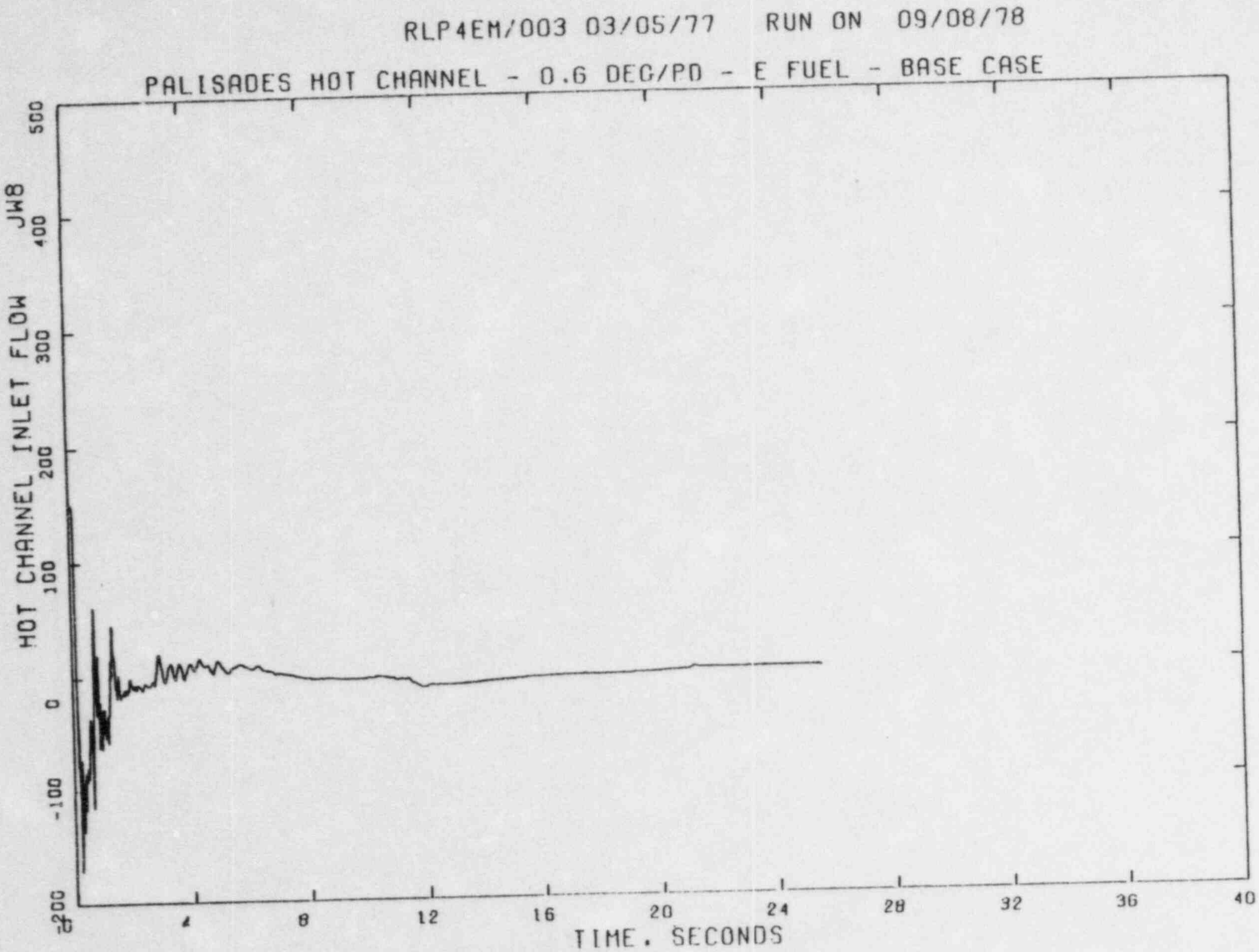

Figure 3.3 Hot Assembly Inlet Flow During
Blowdown, Base Case

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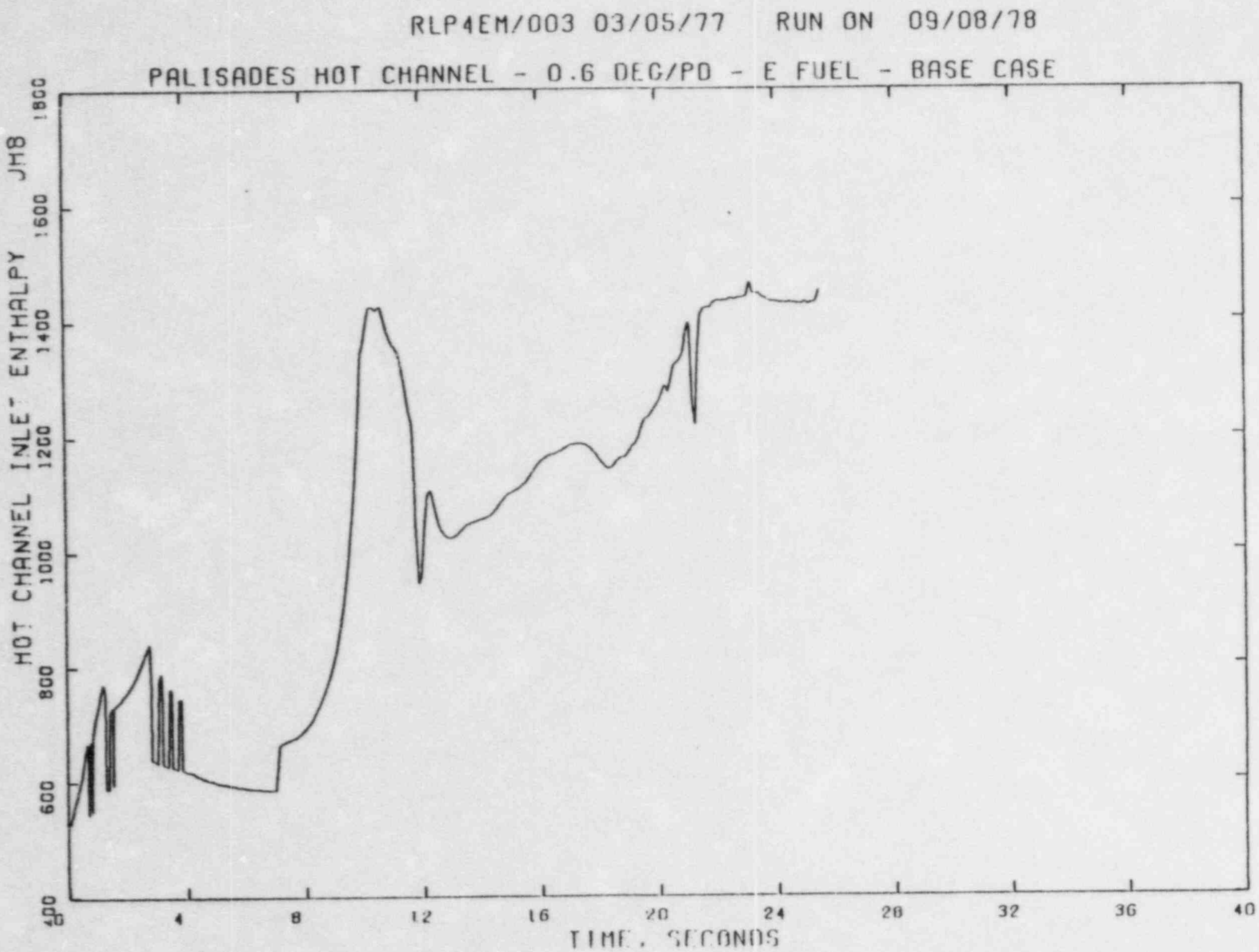

Figure 3.4 Hot Assembly Inlet Enthalpy During Blowdown,
Base Case

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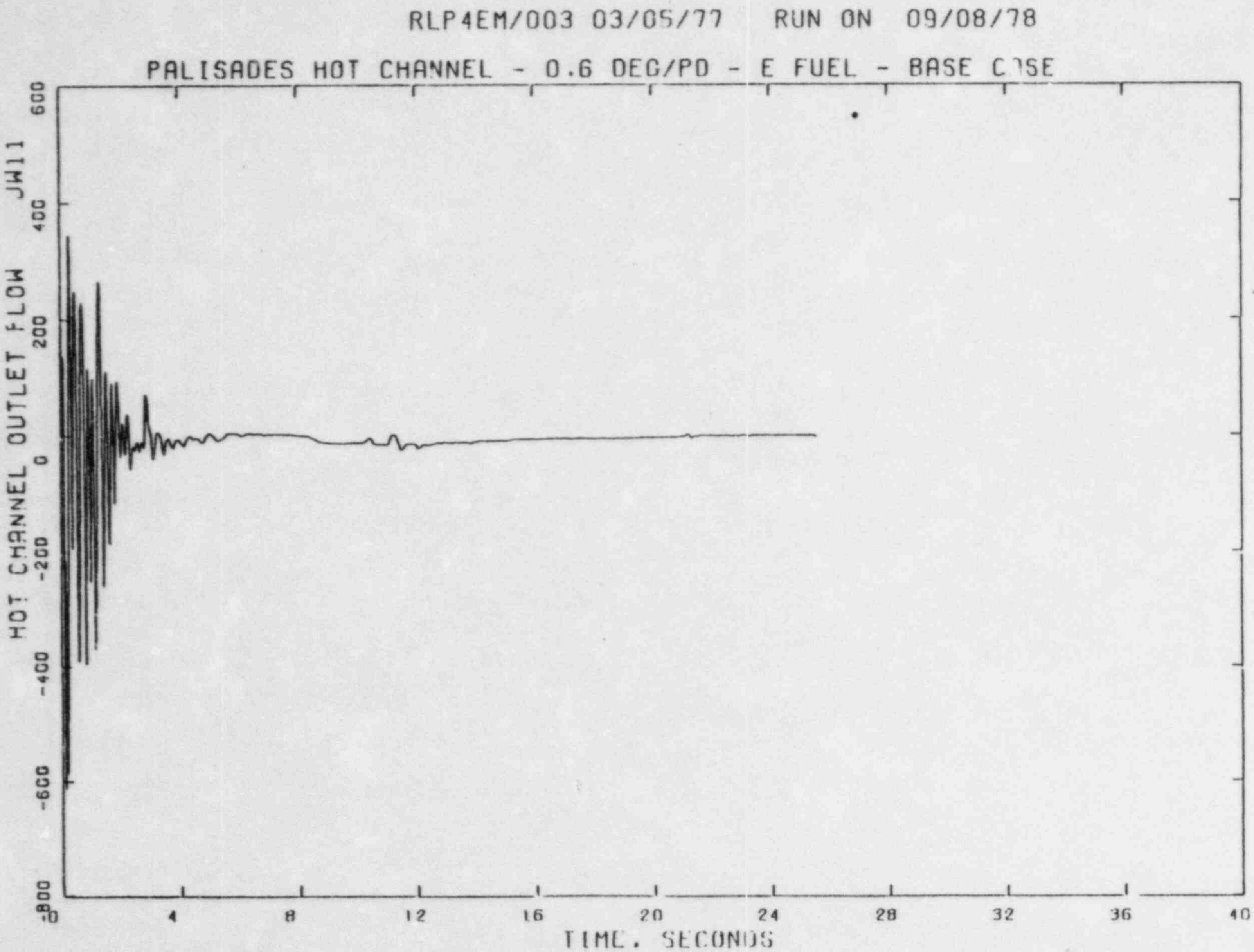

Figure 3.5 Hot Assembly Outlet Flow During Blowdown, Base Case

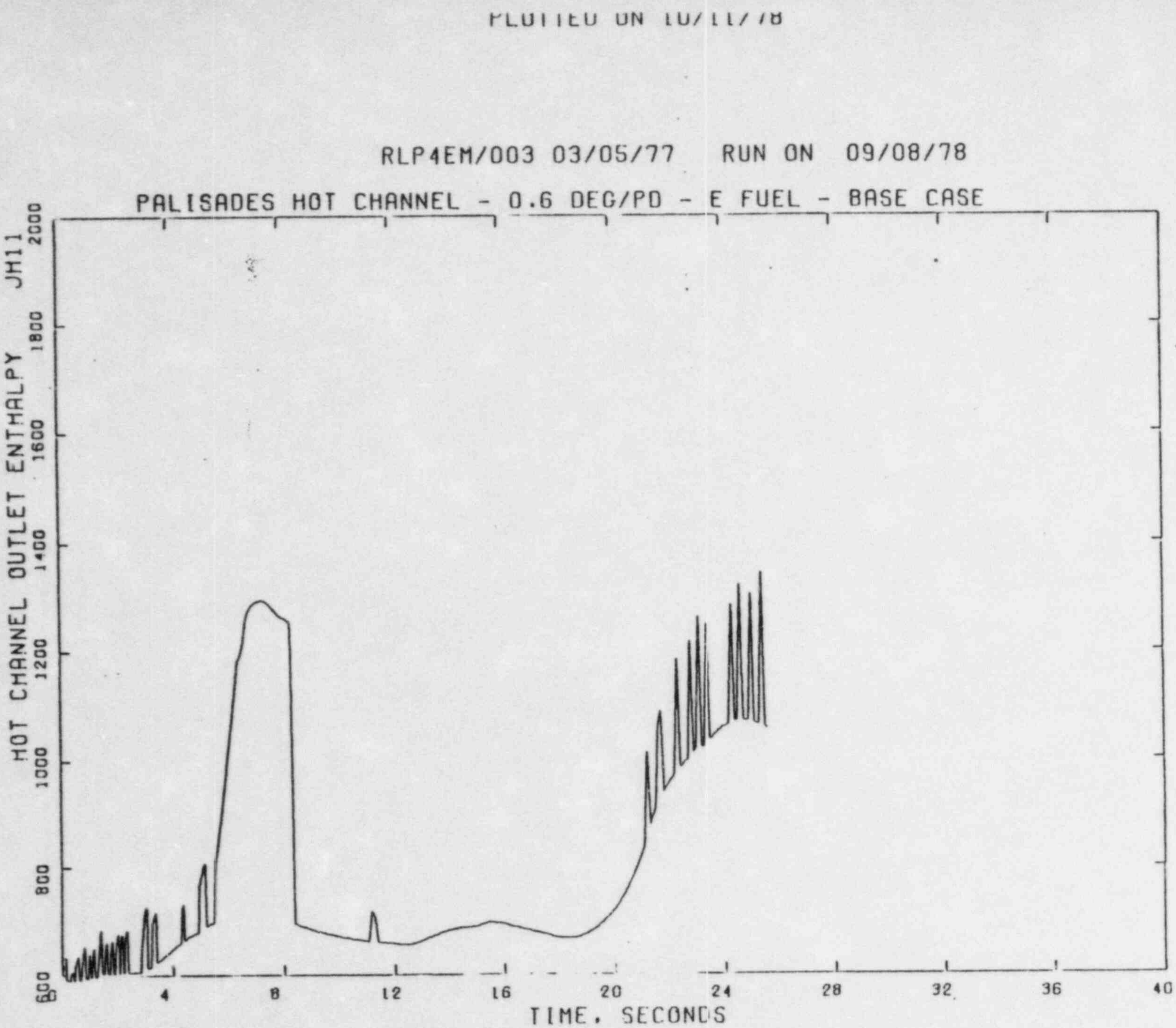

Figure 3.6 Hot Assembly Outlet Enthalpy During Blowdown, Base Case

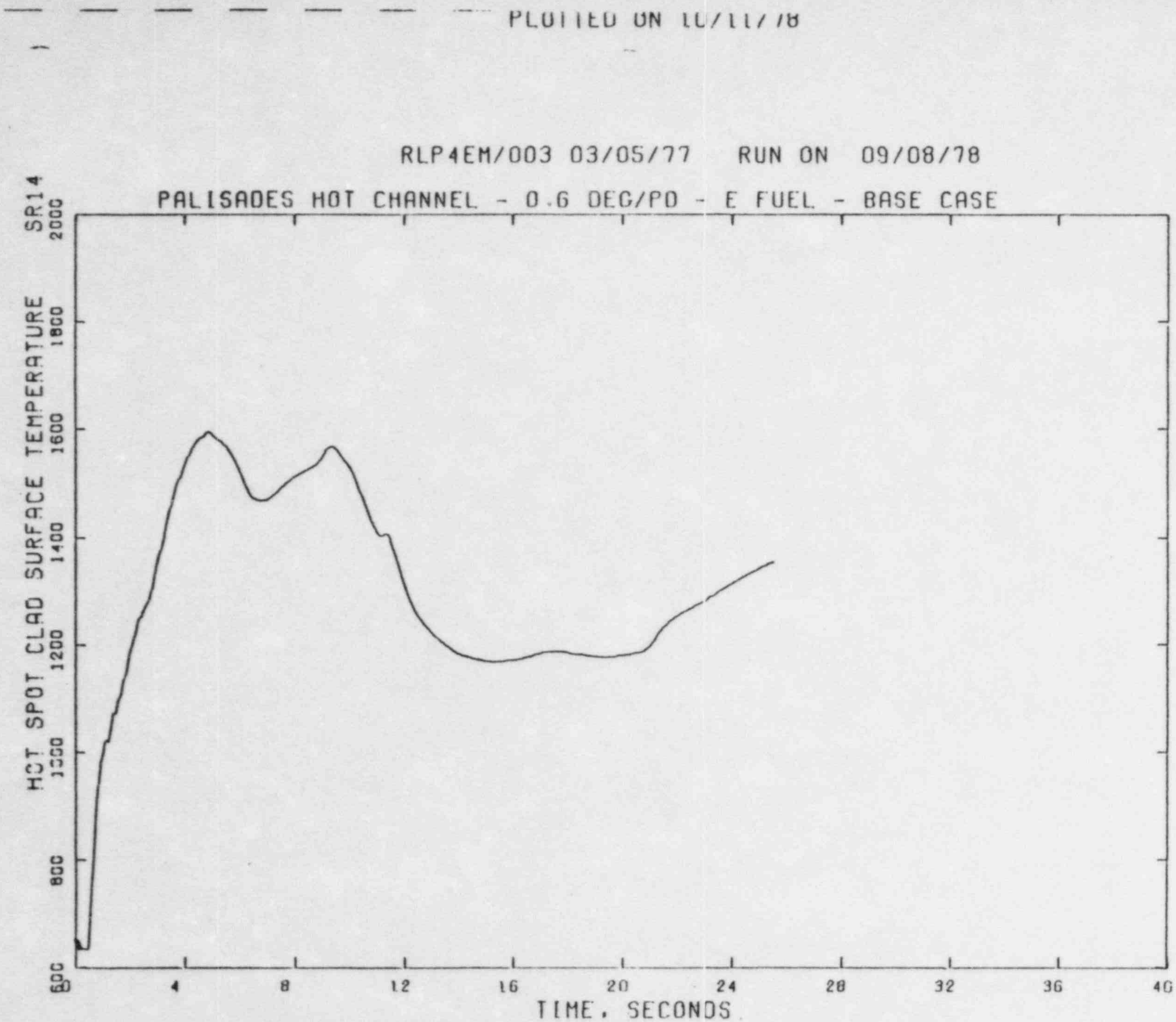

Figure 3.7 Hot Rod Clad Surface Temperature at Peak Power Location During Blowdown, Base Case

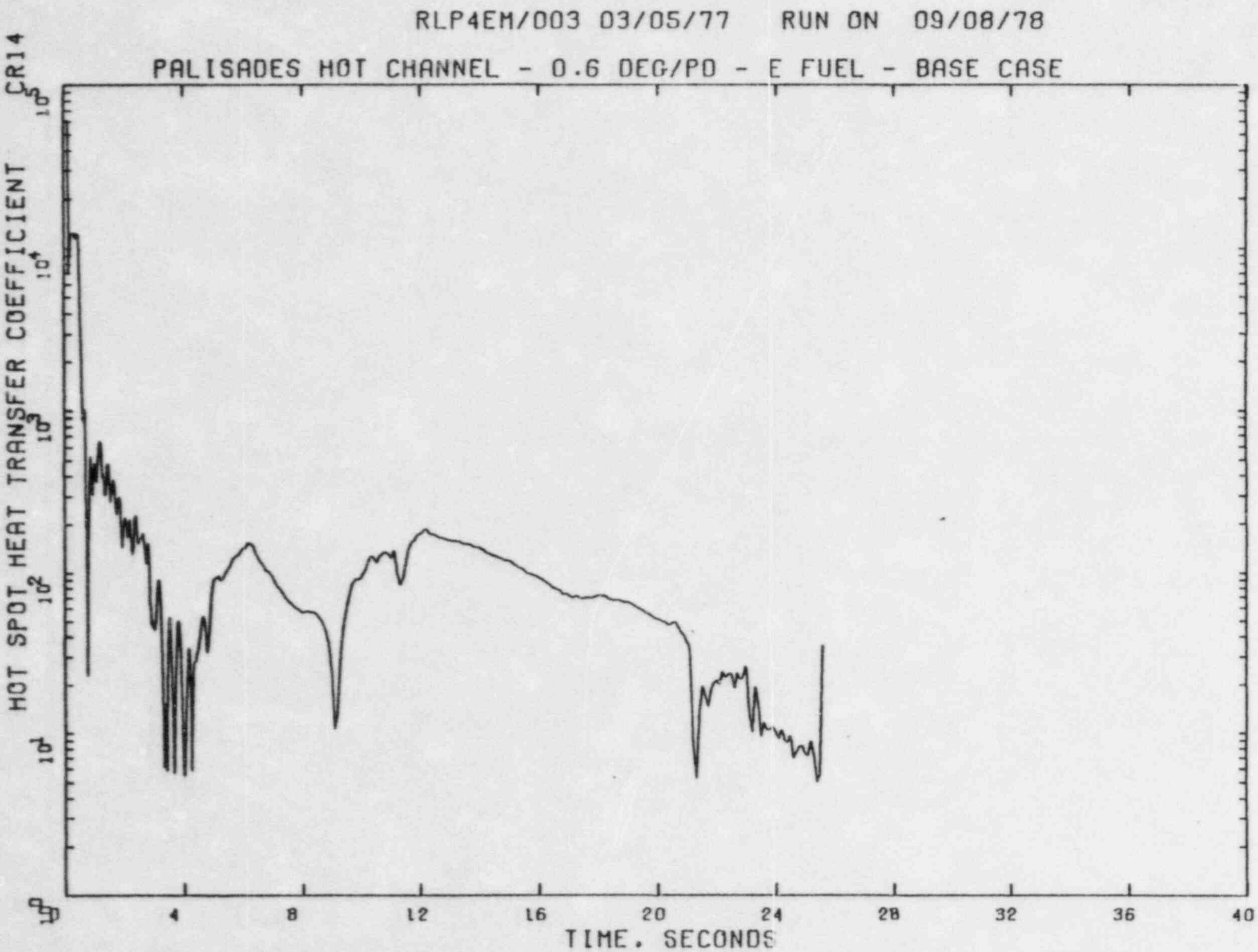

Figure 3.8 Hot Rod Heat Transfer Coefficient at Peak Power Location During Blowdown, Base Case

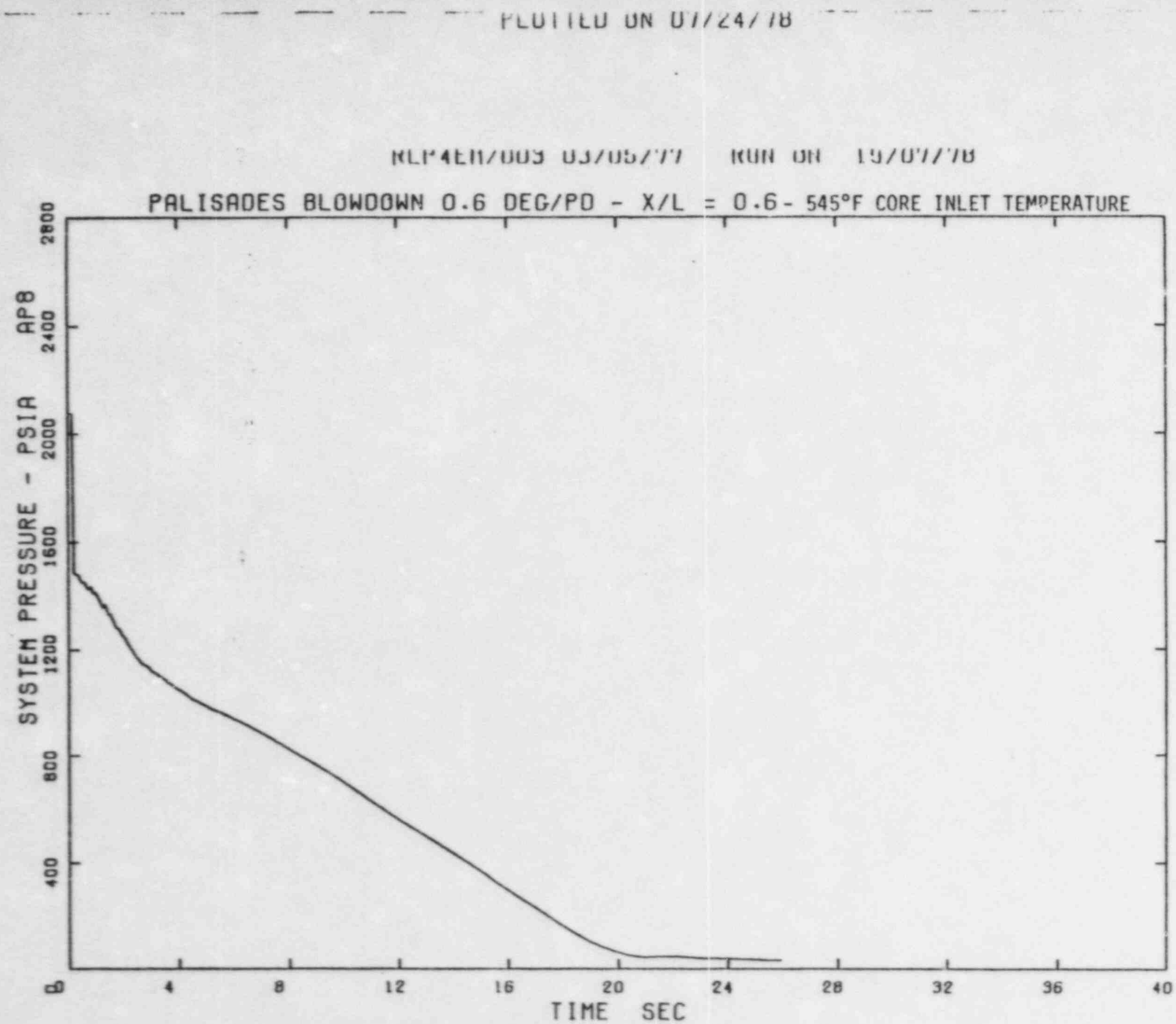

Figure 3.9 Upper Plenum Pressure During Blowdown, 545^{o}F Core Inlet Temperature Case

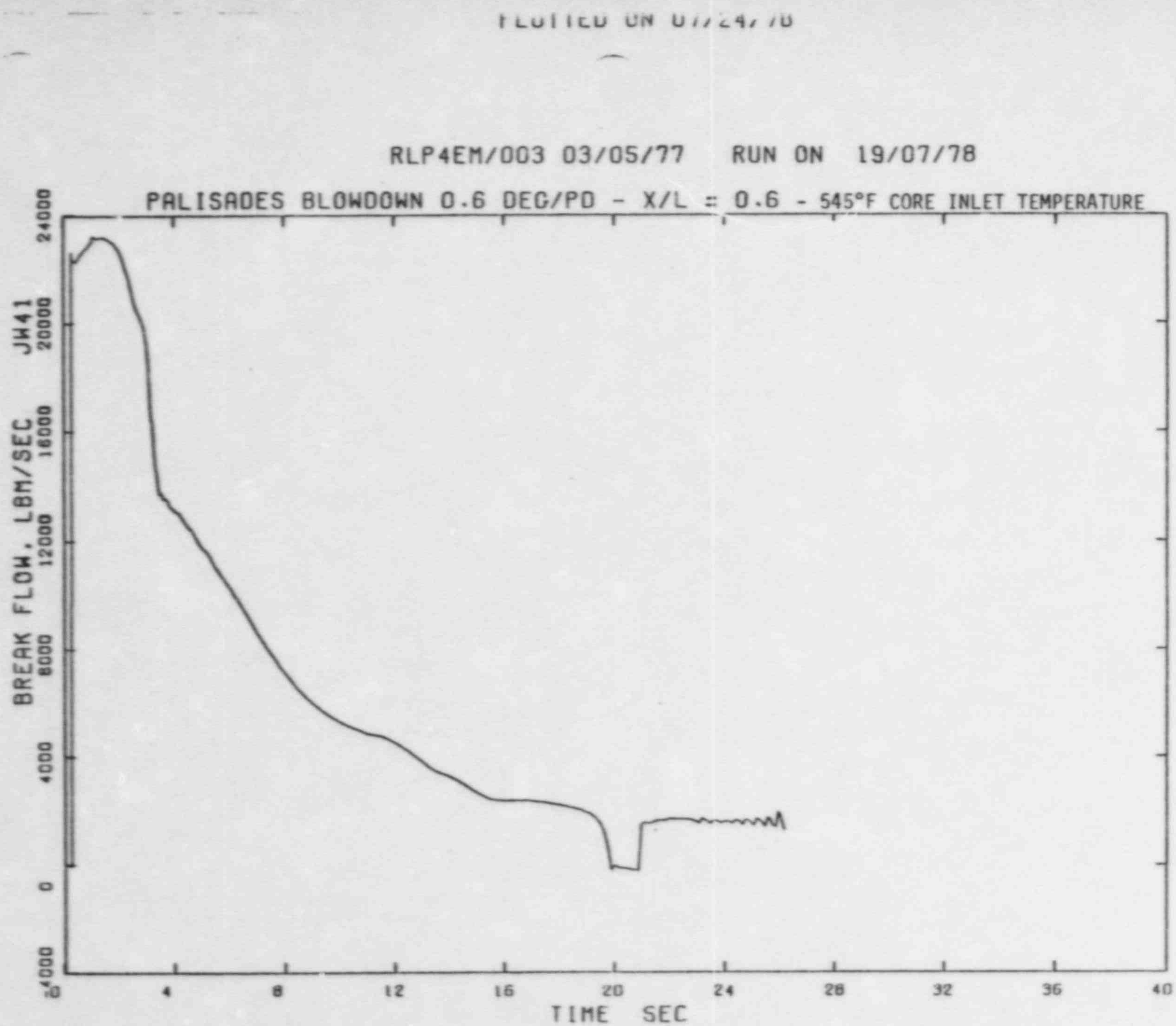

Figure 3.10 Pump Side Break Flow During Blowdown, 545°F Core Inlet Temperature Case

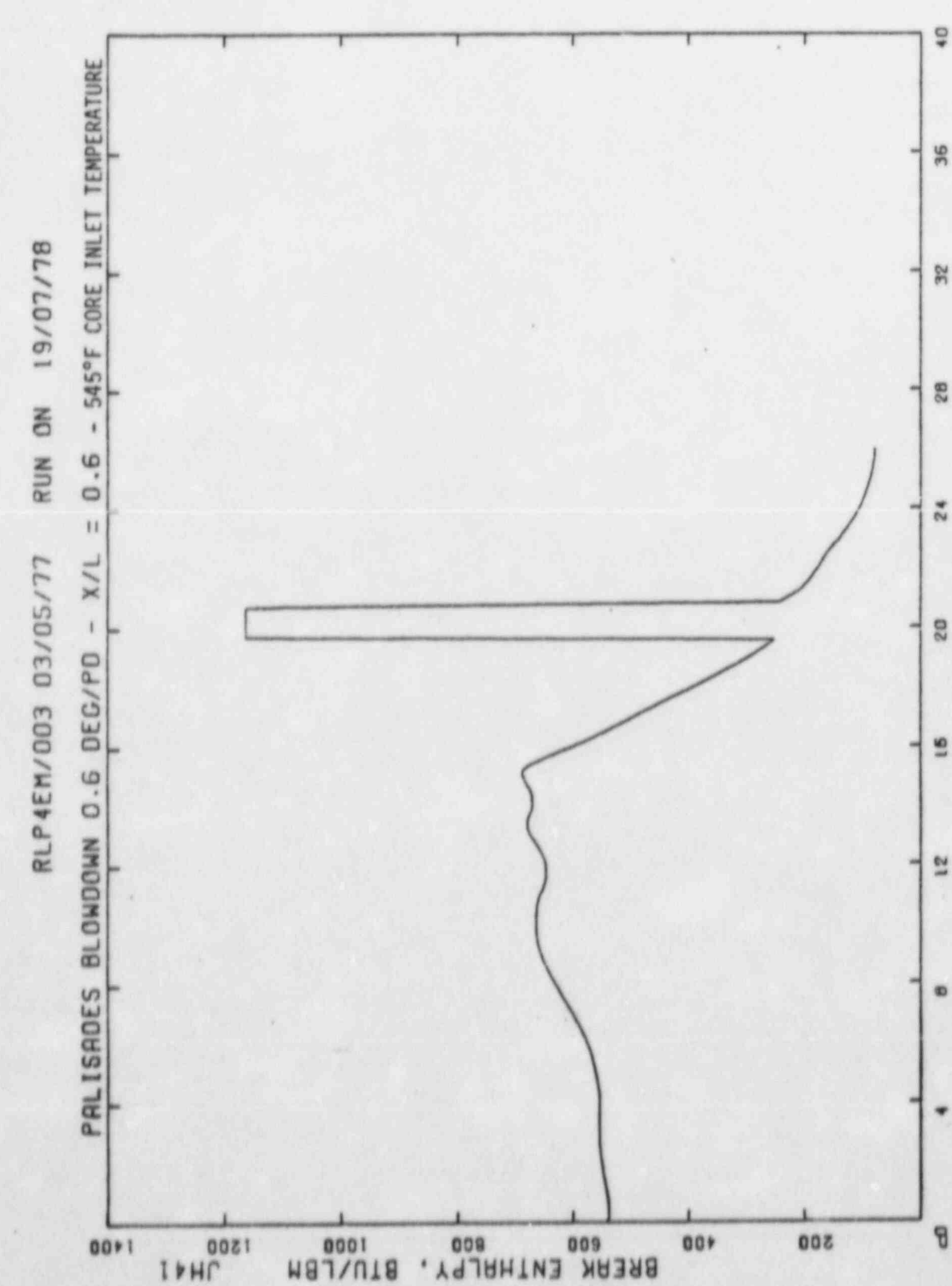

Figure 3.11 Pump Side Break Enthalpy During Blowdown, 545°F Core Inlet Temperature Case

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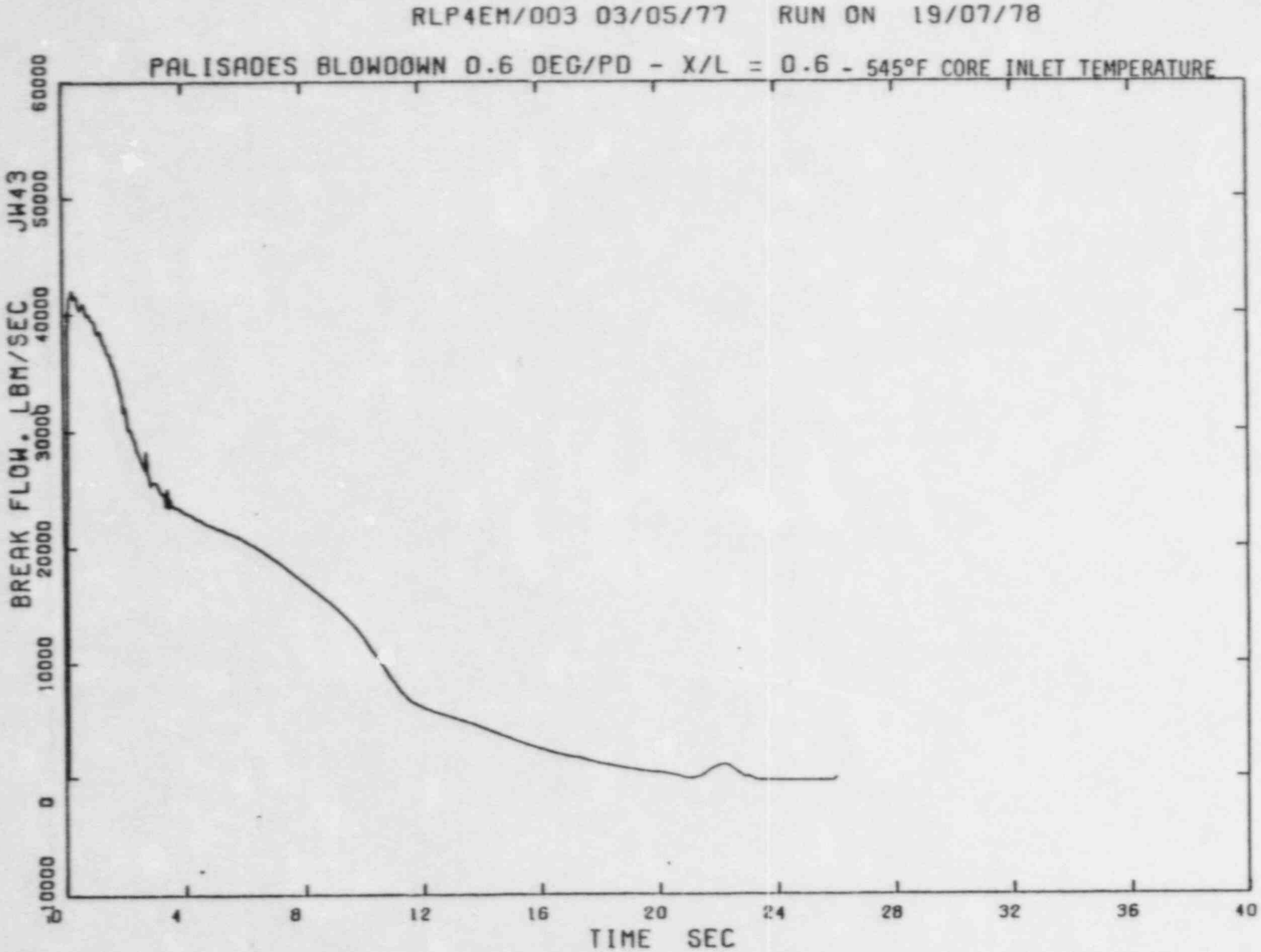

Figure 3.12 Vessel Side Break Flow During Blowdown, 545°F Core Inlet Temperature Case

PLOTTED ON 07/24/78

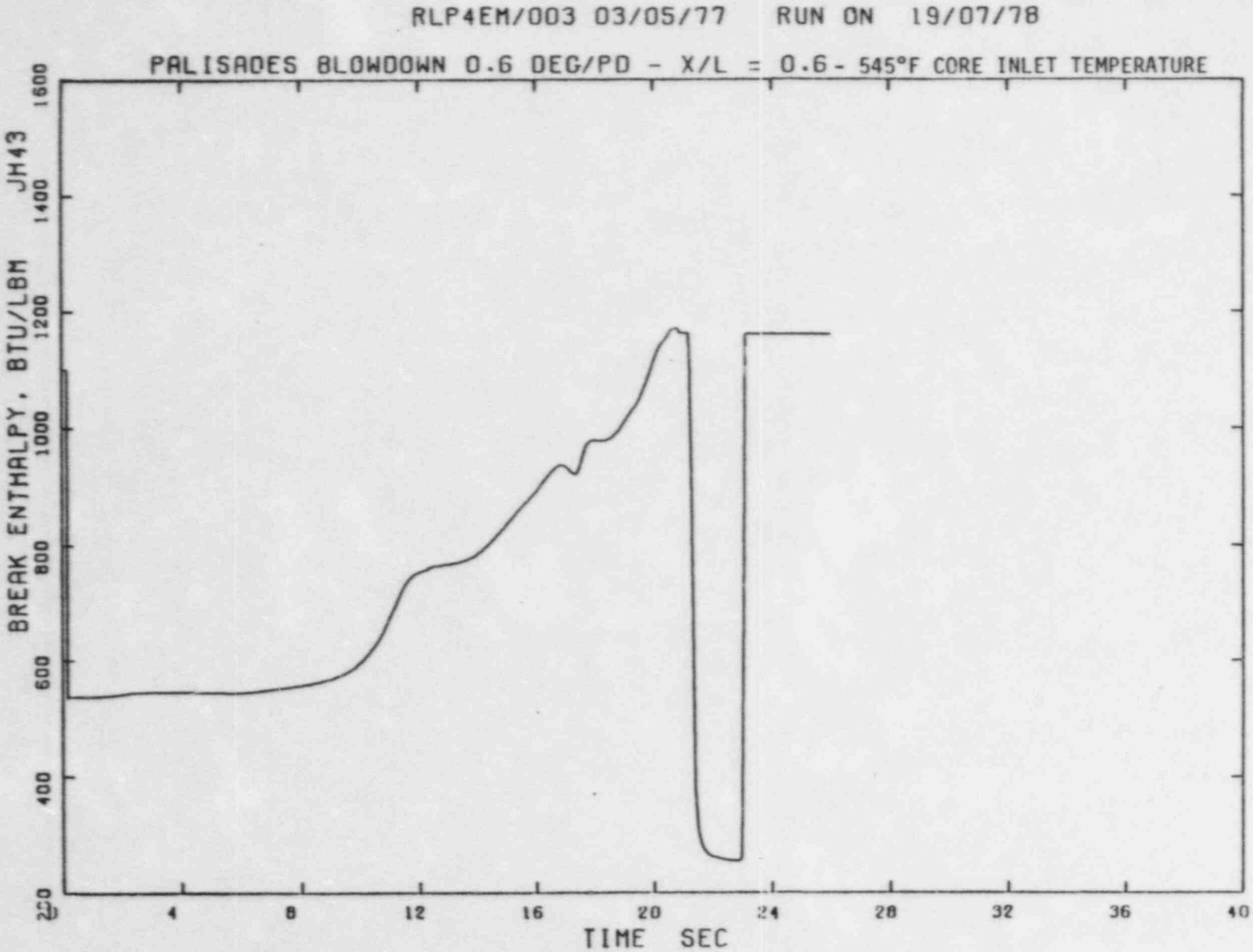

Figure 3.13 Vessel Side Break Enthalpy During Blowdown, 545°F Core Inlet Temperature Case

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PALISADES HOT CHANNEL - 0.6 DEG/PD - E FUEL - 545 CORE INLET TEMPERATURE

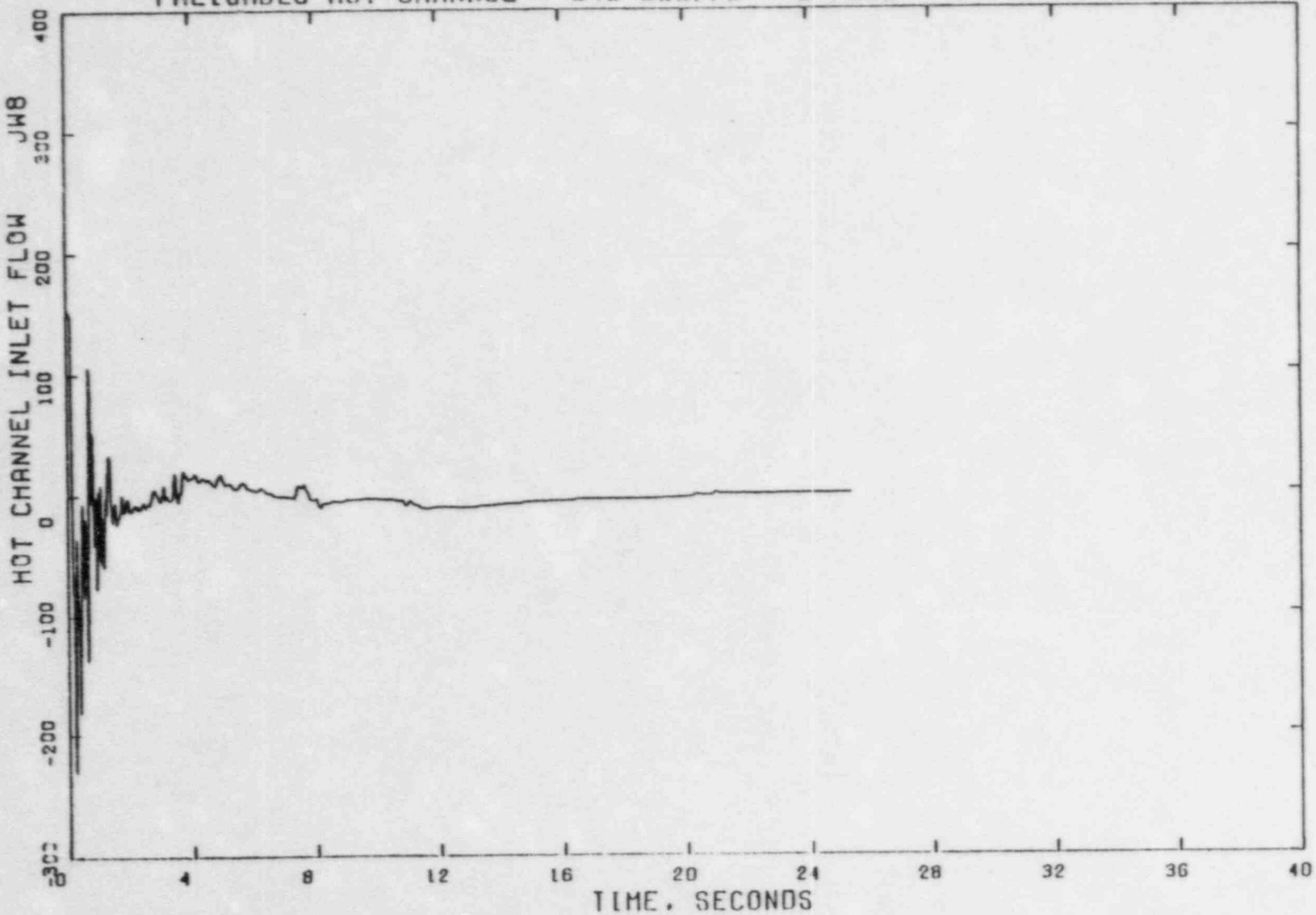

Figure 3.14 Hot Assembly Inlet Flow During Blowdown, 545°F Core Inlet Temperature Case

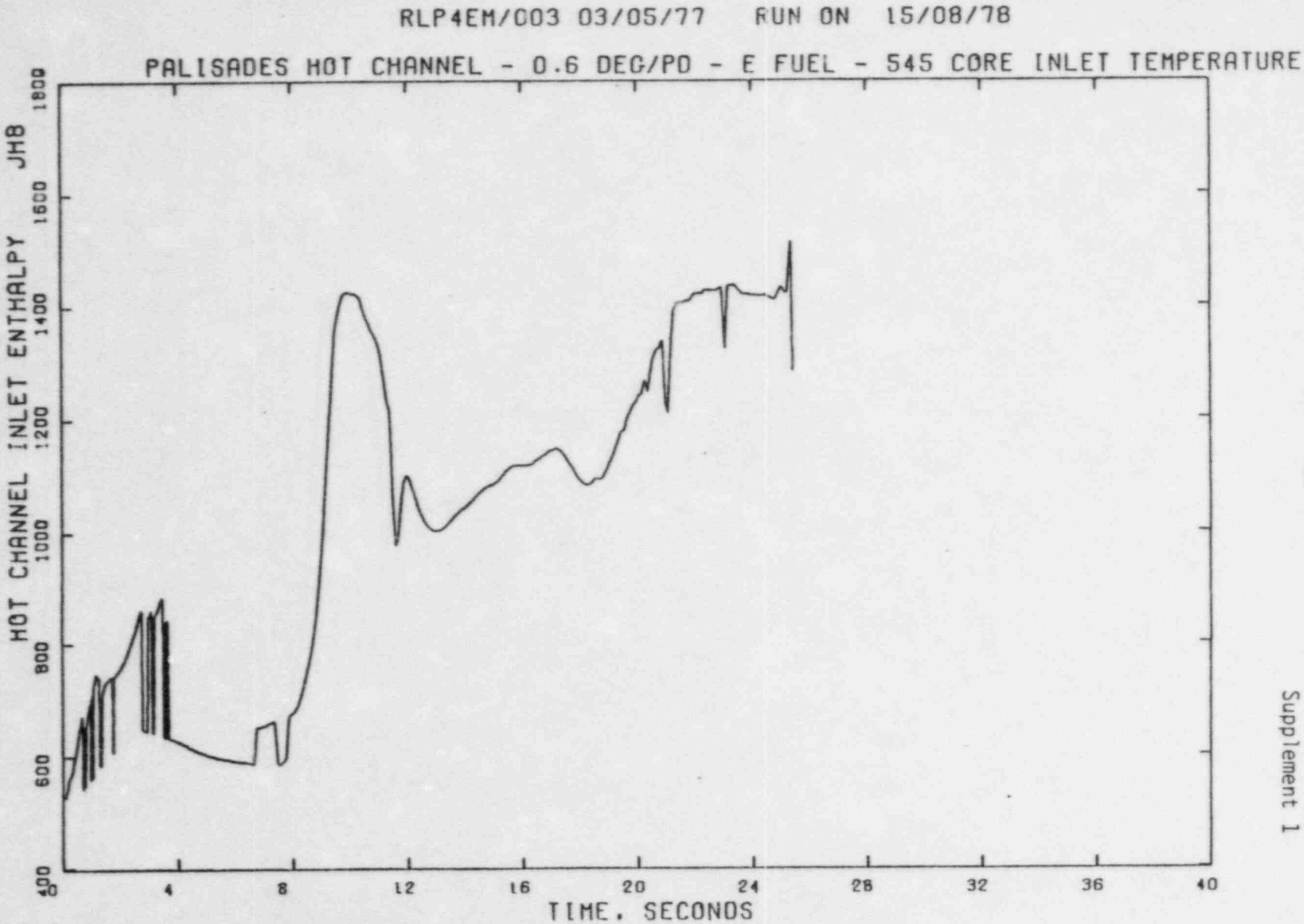

Figure 3.15 Hot Assembly Inlet Enthalpy During Blowdown, 545°F Core Inlet Temperature Case

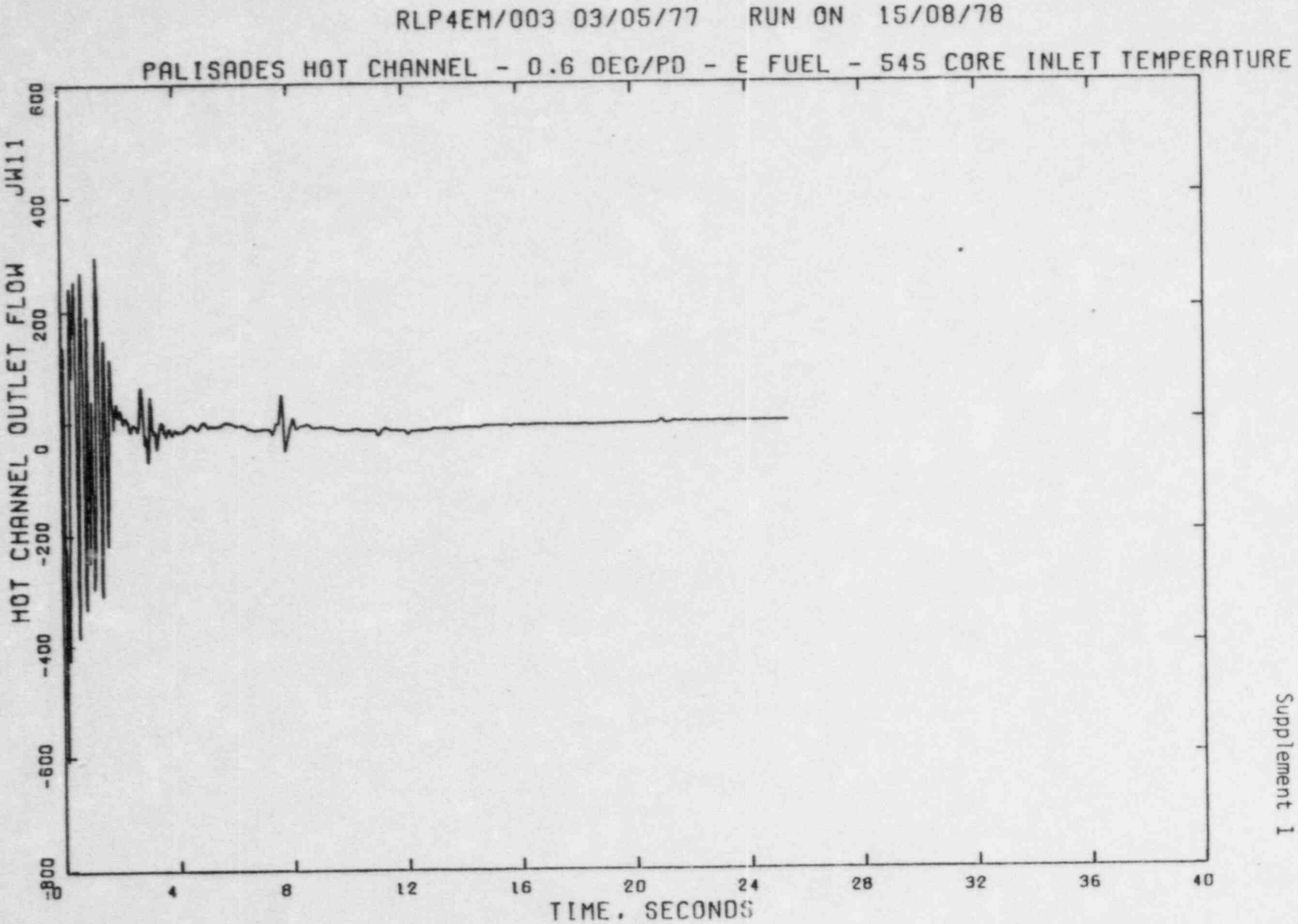

Figure 3.16 Hot Assembly Outlet Flow During Blowdown,
545^{o}F Core Inlet Temperature Case

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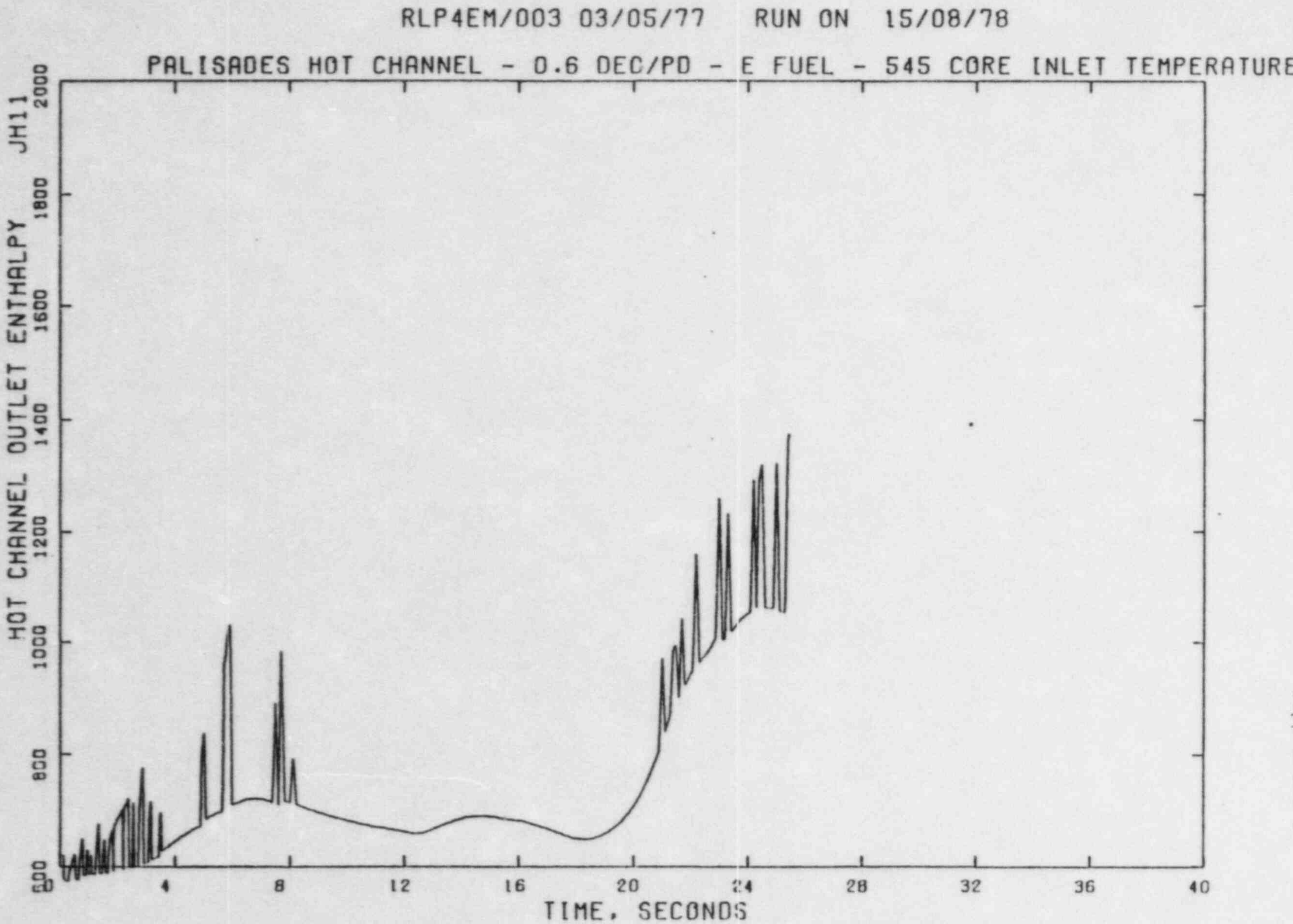

Figure 3.17 Hot Assembly Outlet Enthalpy During Blowdown, 545°F Core Inlet Temperature Case

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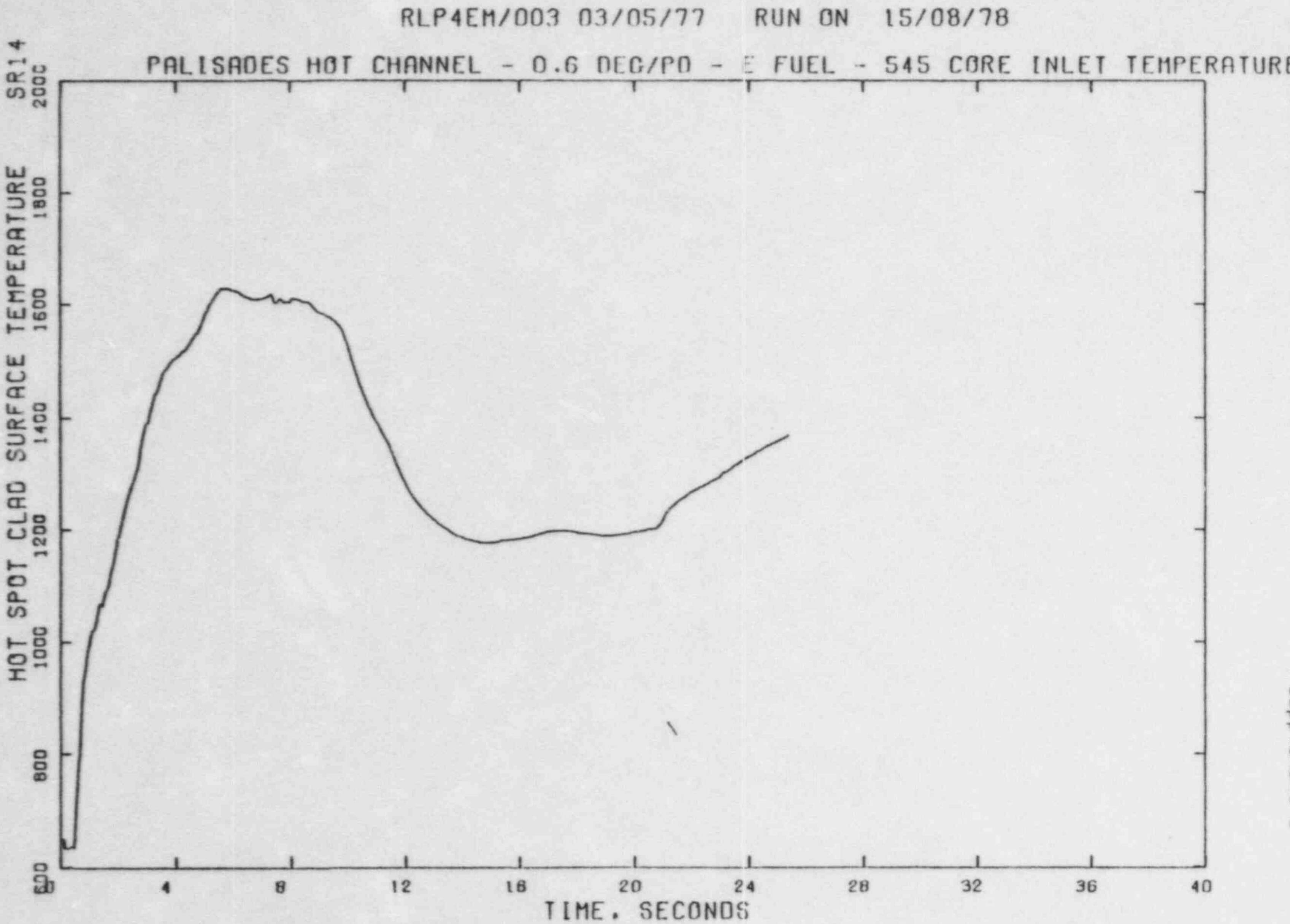

Figure 3.18 Hot Rod Clad Surface Temperature at Peak Power Location During Blowdown, 545°F Core Inlet Temperature Case

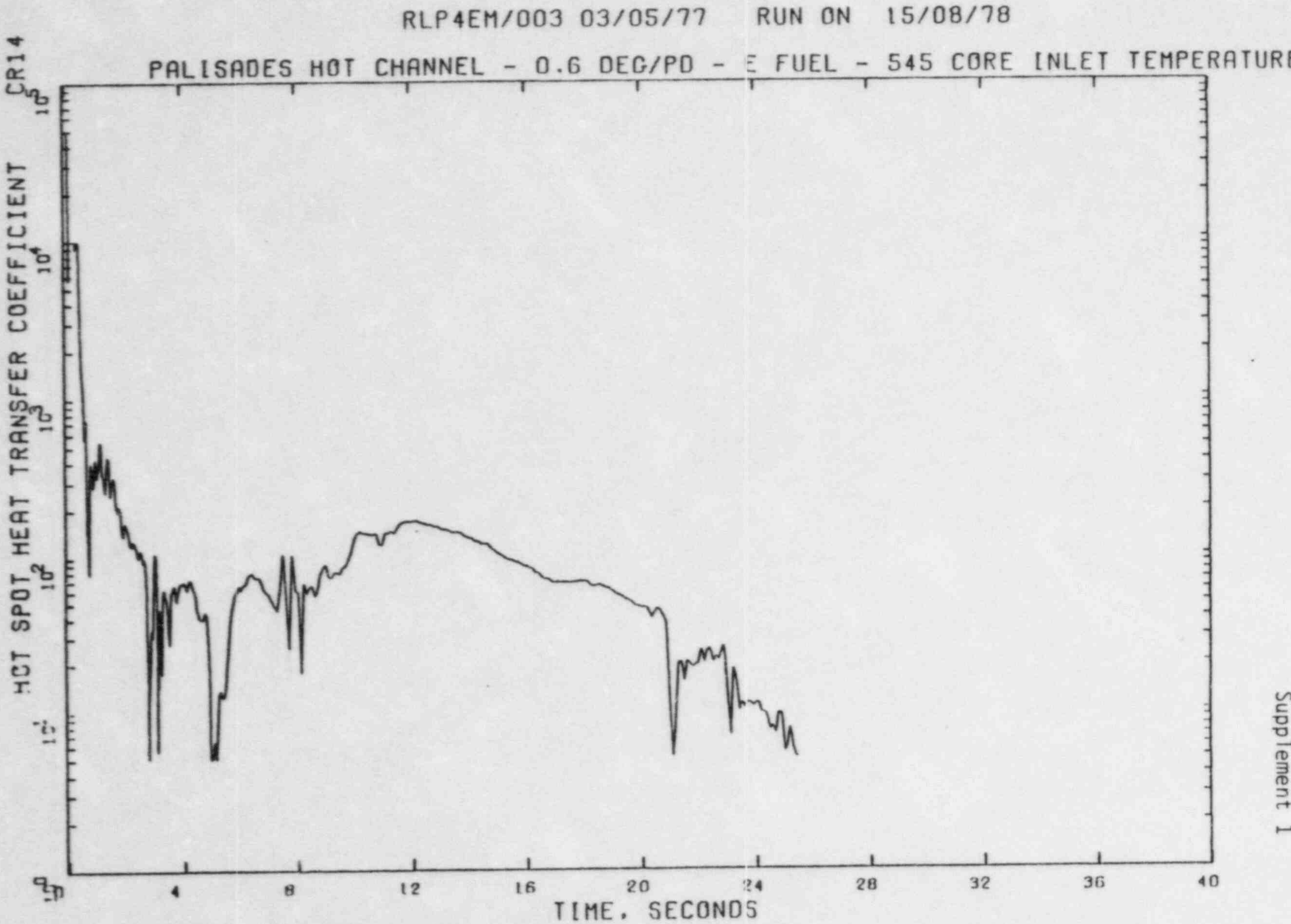

Figure 3.19 Hot Rod Heat Transfer Coefficient at Peak Power Location During Blowdown, 545°F Core Inlet Temperature Case

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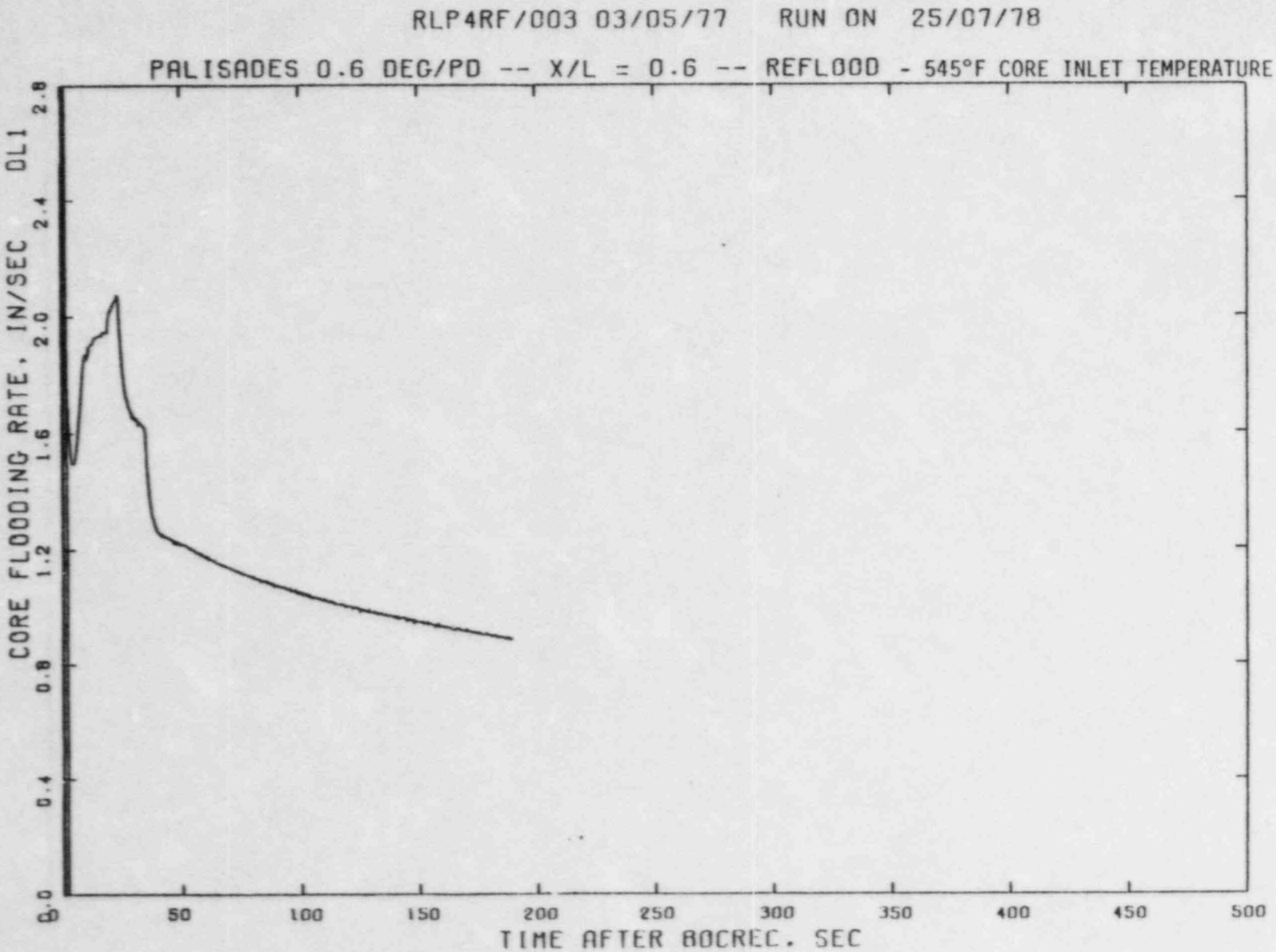

Figure 3.20 Core Reflood Rate During Reflood,
545°F Core Inlet Temperature Case

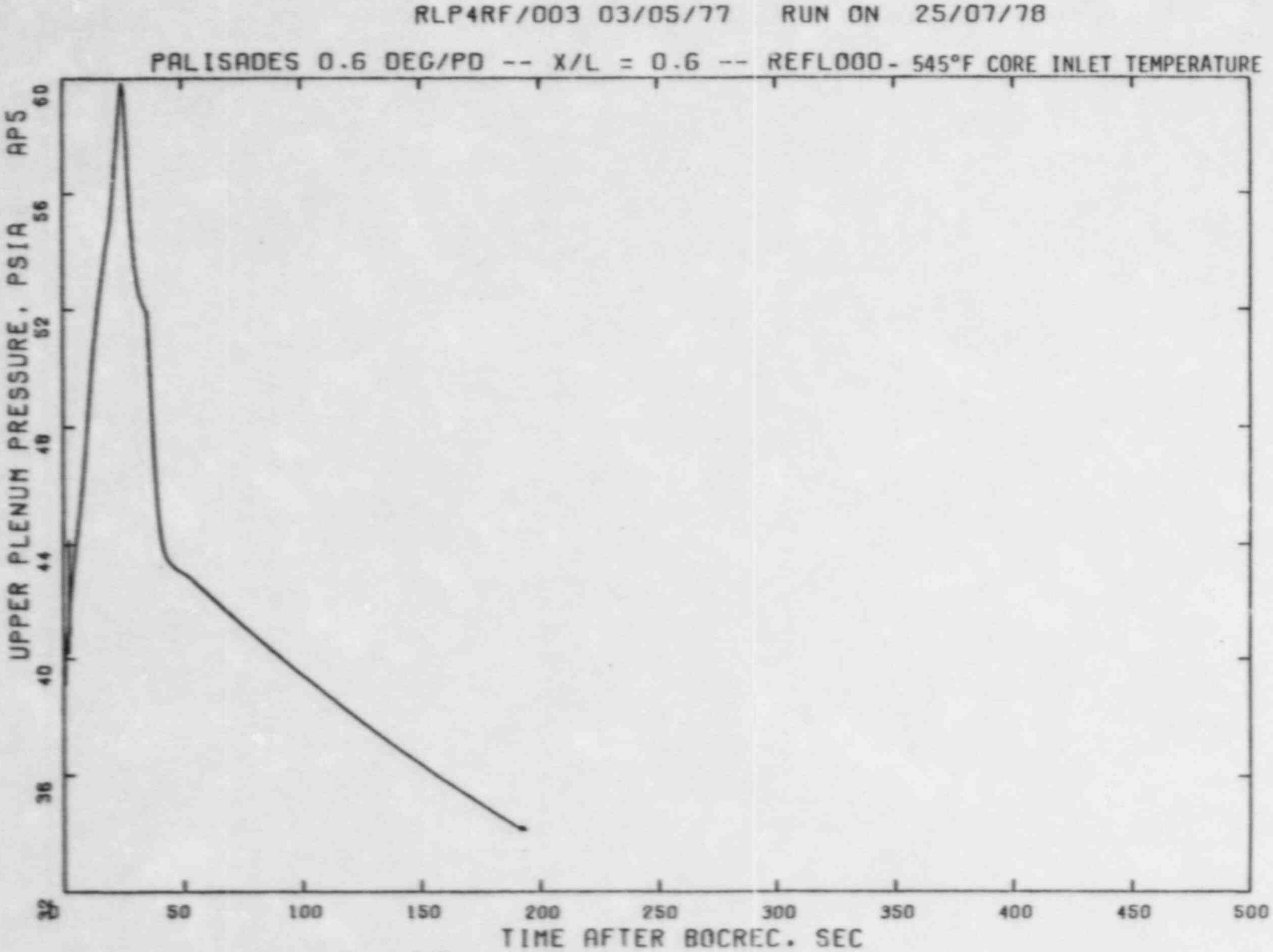

Figure 3.21 Upper Plenum Pressure Curing Reflood,
545°F Core Inlet Temperature Case

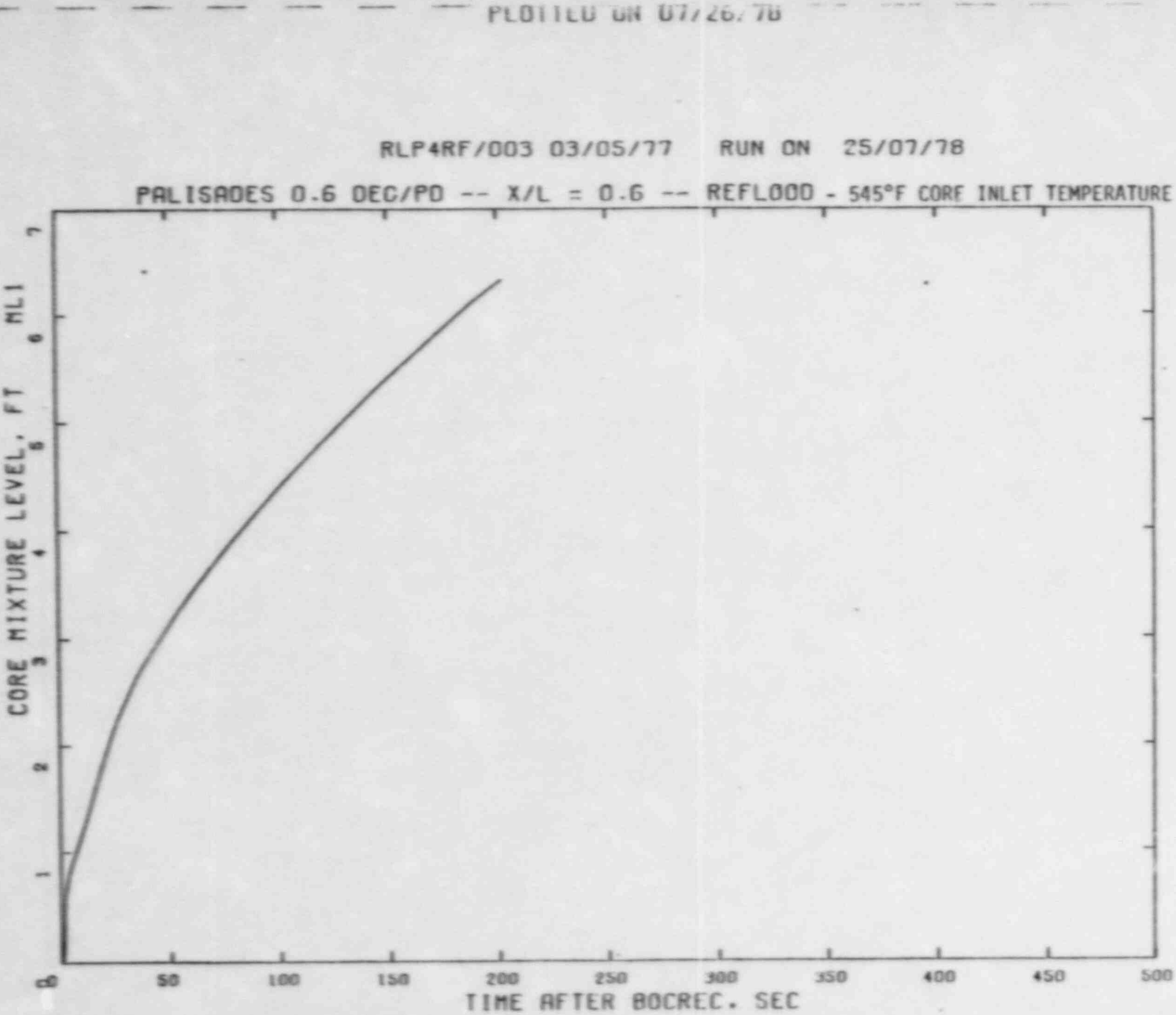

Figure 3.22 Collapsed Liquid Level in Core During Reflood, 545°F Core Inlet Temperature Case

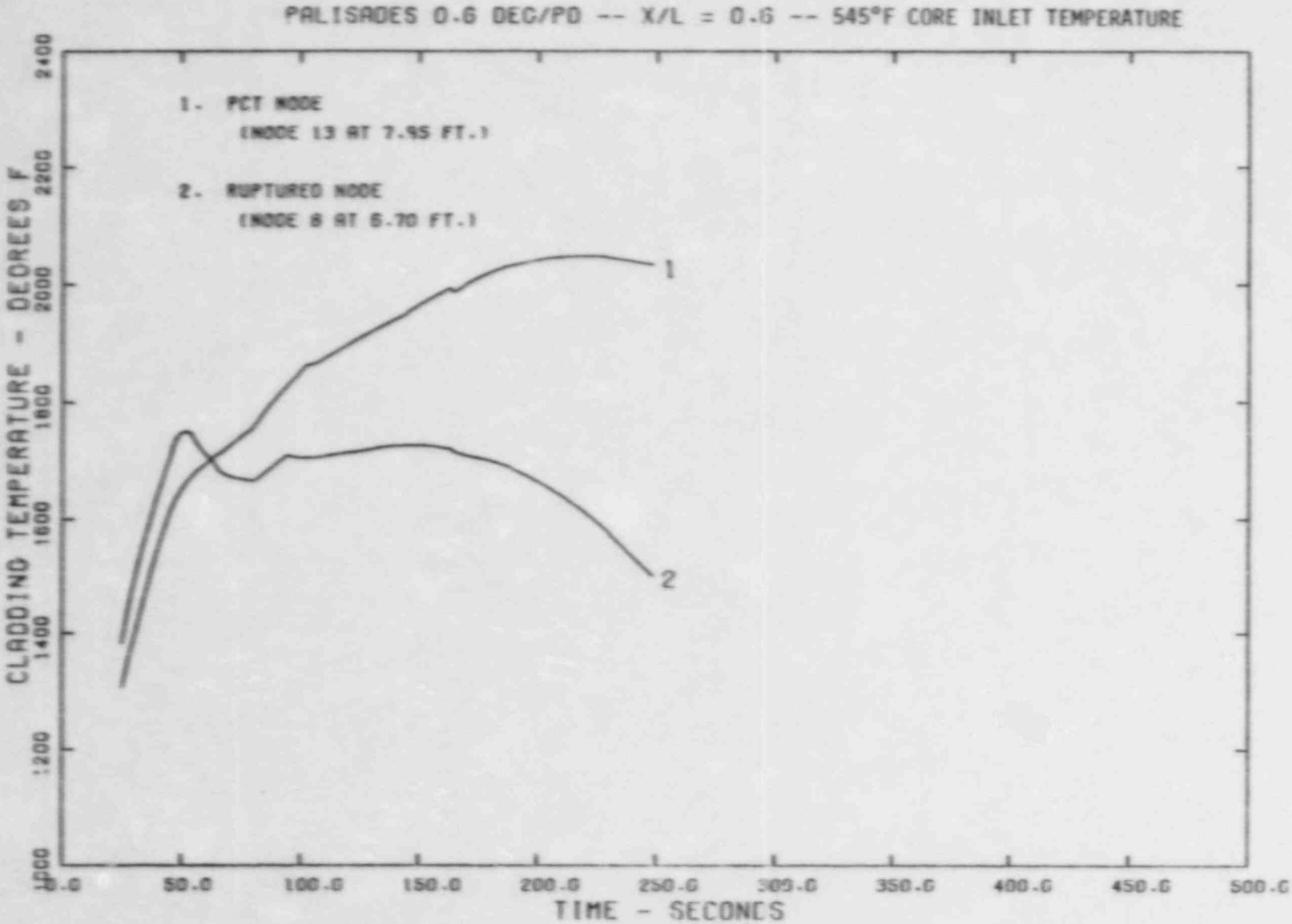

Figure 3.23 Cladding Temperatures at PCT and Rupture Locations, 545°F Core Inlet Temperature Case

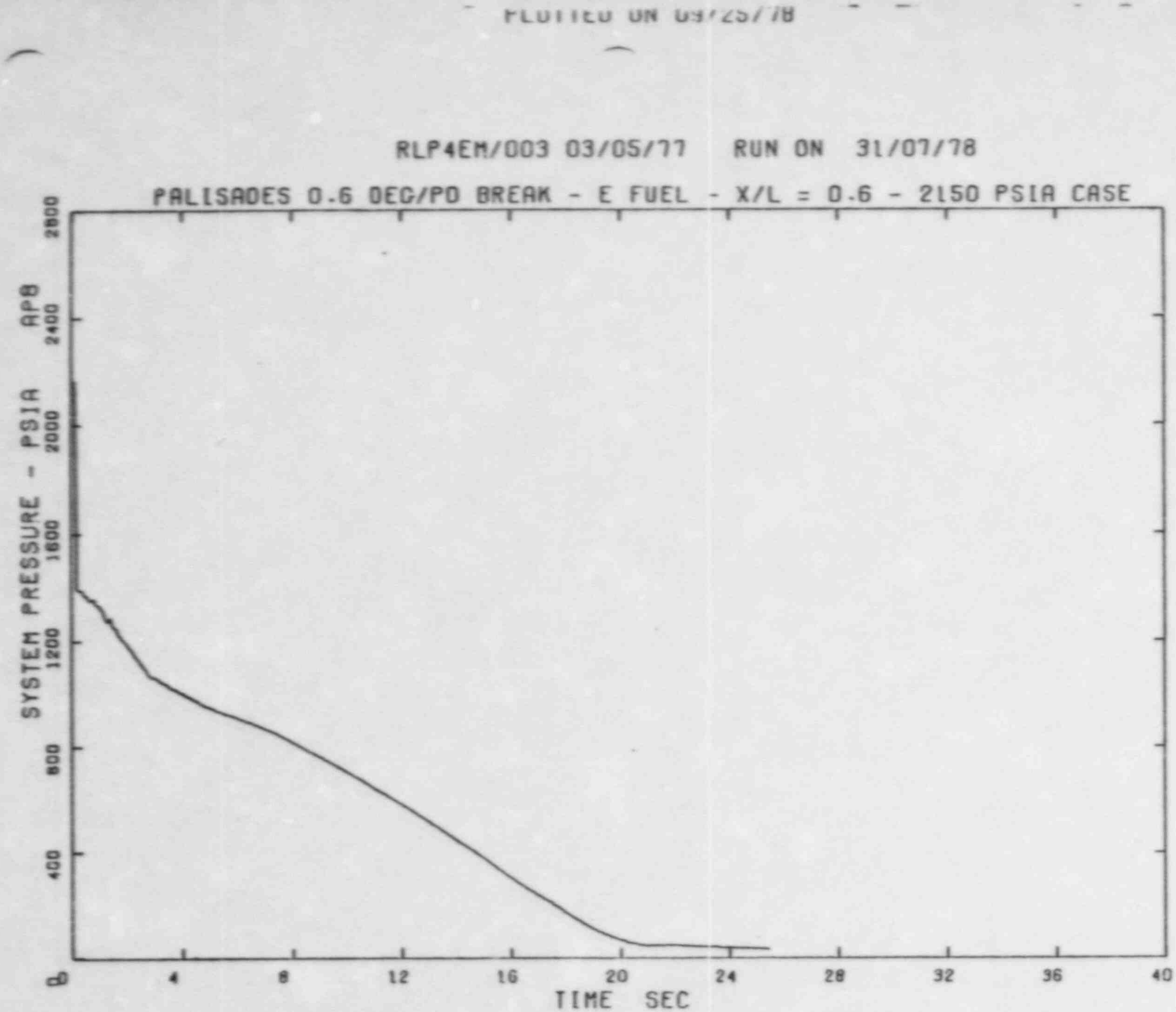

Figure 3.24 Upper Plenum Pressure During Blowdown,
2150 psia Pressurizer Pressure Case

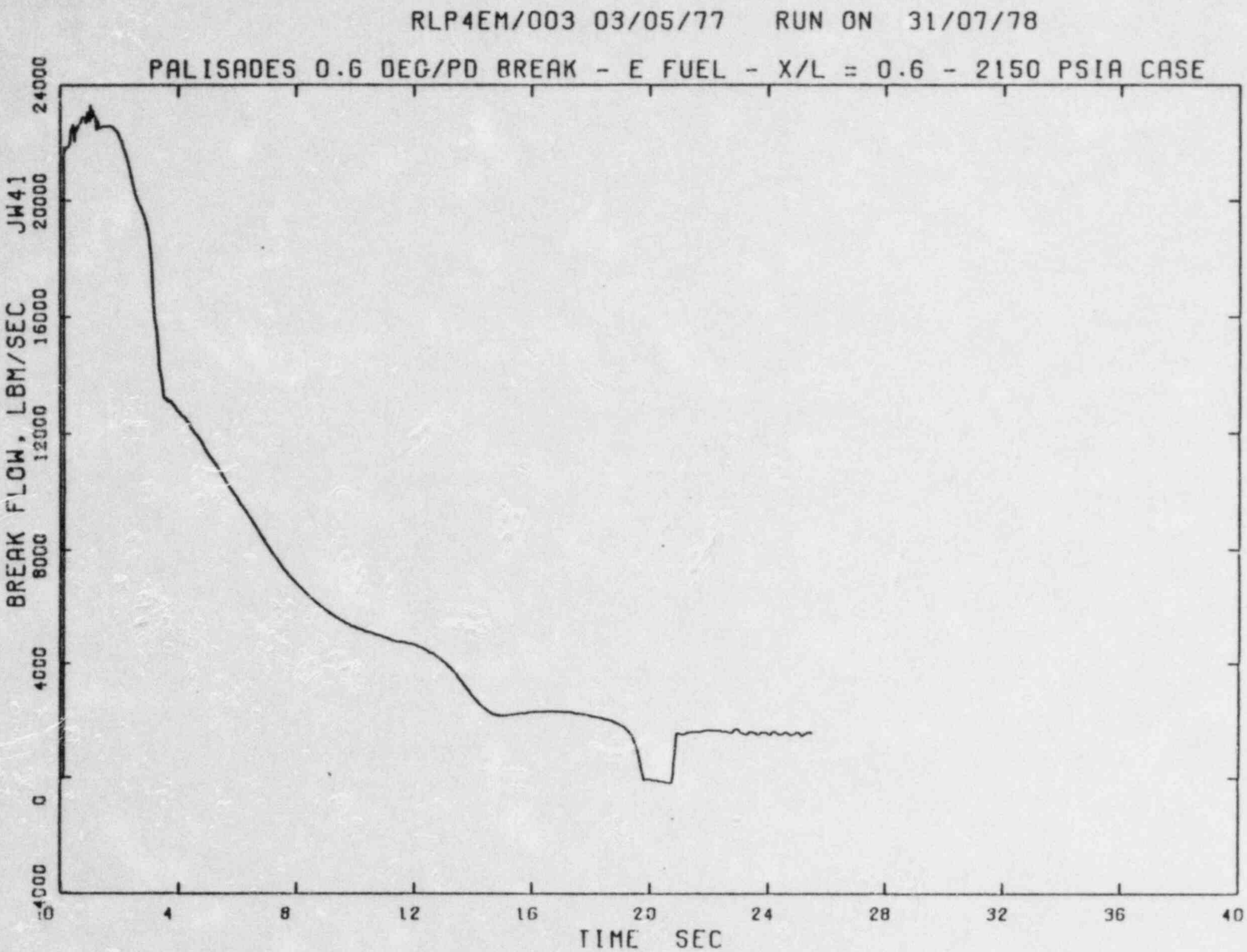

Figure 3.25 Pump Side Break Flow During Blowdown, 2150 psia Pressurizer Pressure Case

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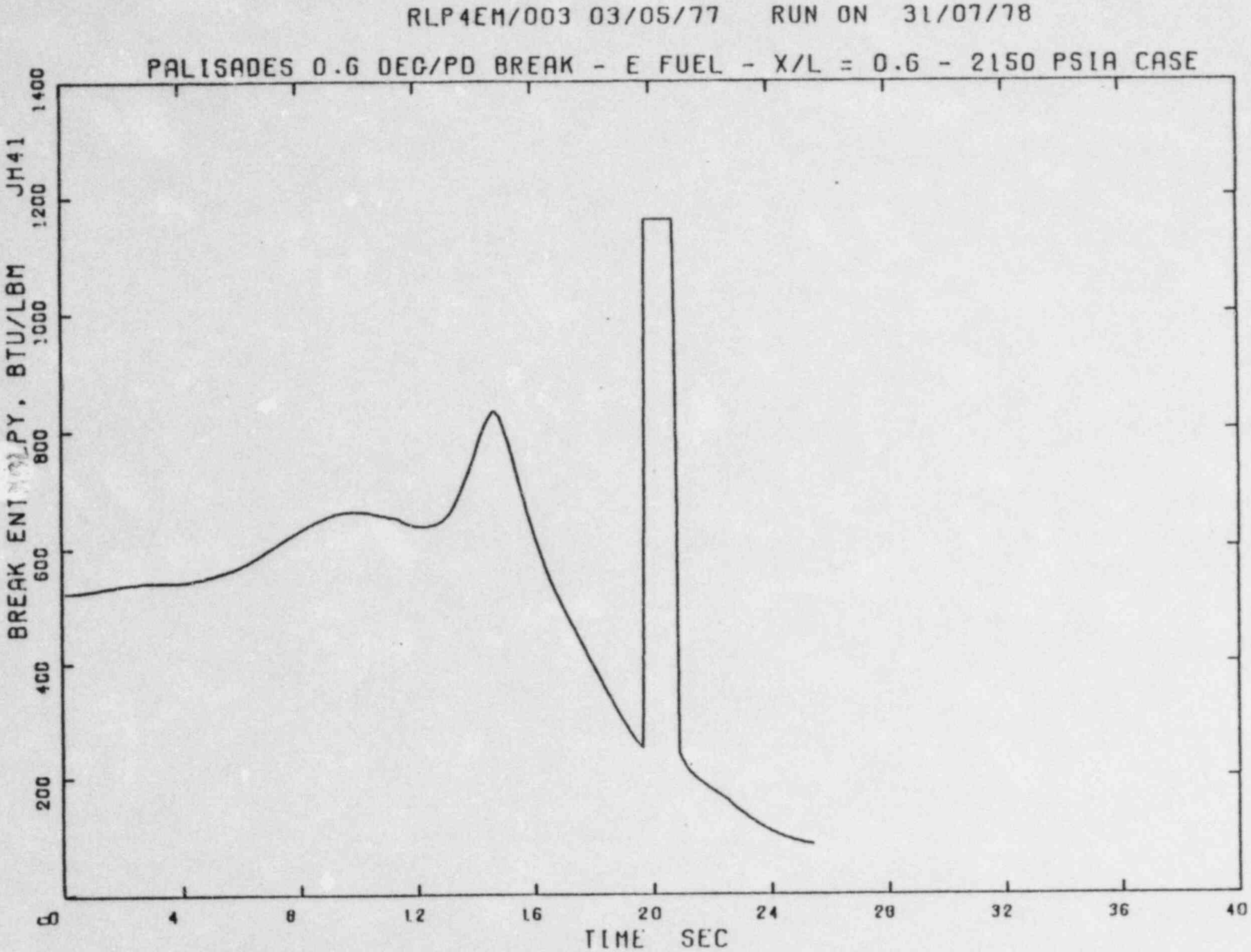

Figure 3.26 Pump Side Break Enthalpy During Blowdown,
2150 psia Pressurizer Pressure Case

PLOTTED ON 09/25/78

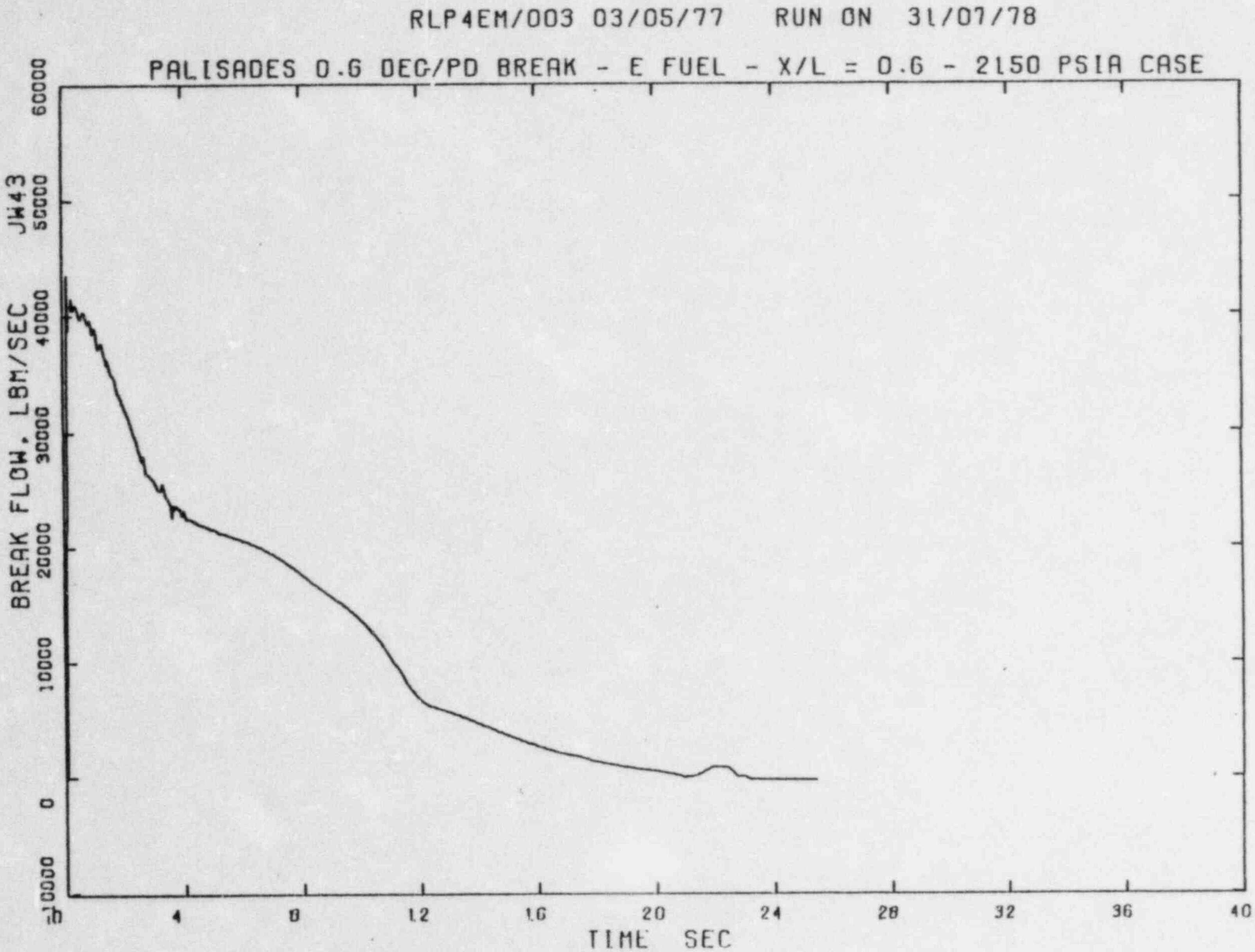

Figure 3.27 Vessel Side Break Flow During Blowdown, 2150 psia Pressurizer Pressure Case

PLOTTED ON 03/25/78

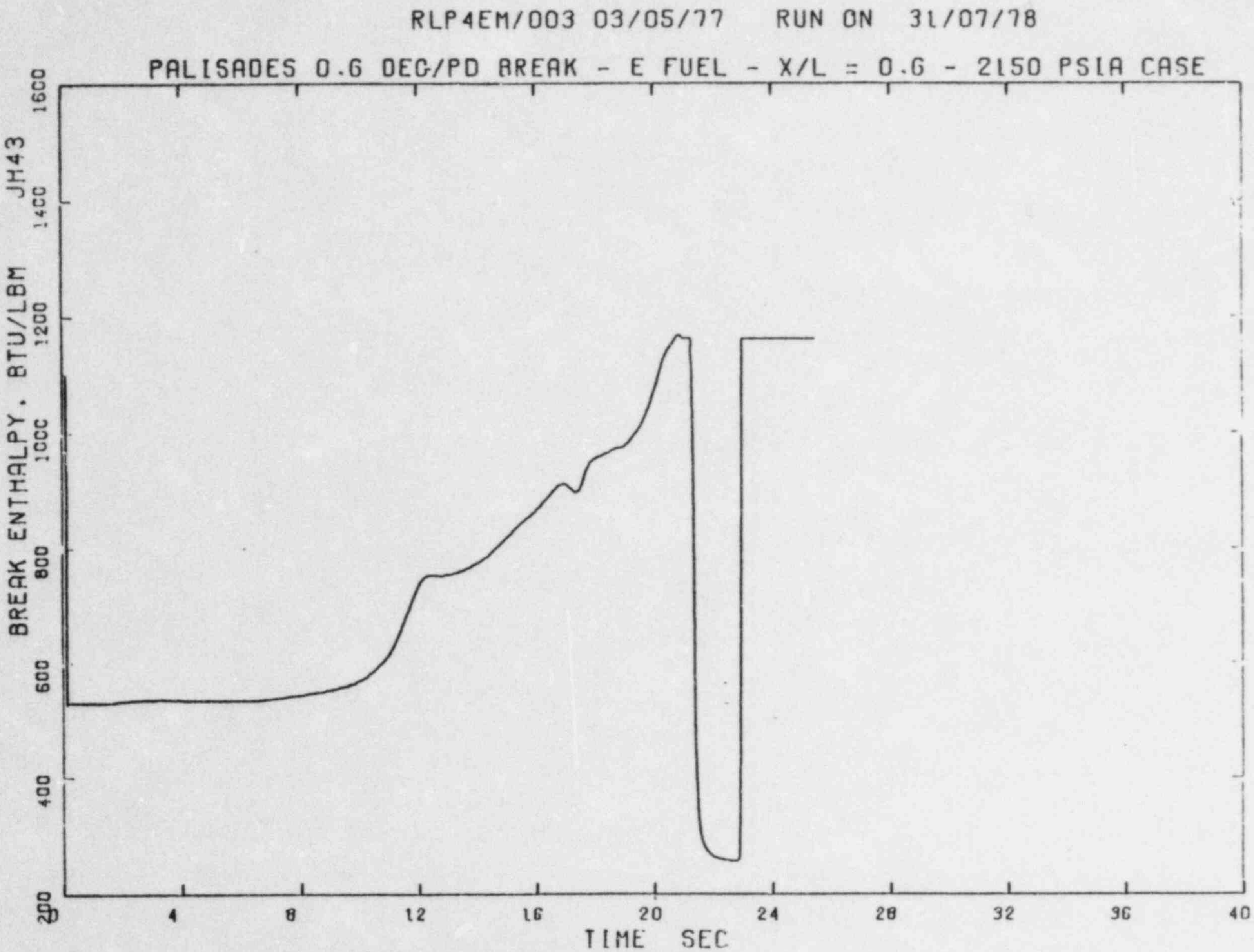

Figure 3.28 Vessel Side Break Enthalpy During Blowdown,
2150 psia Pressurizer Pressure Case

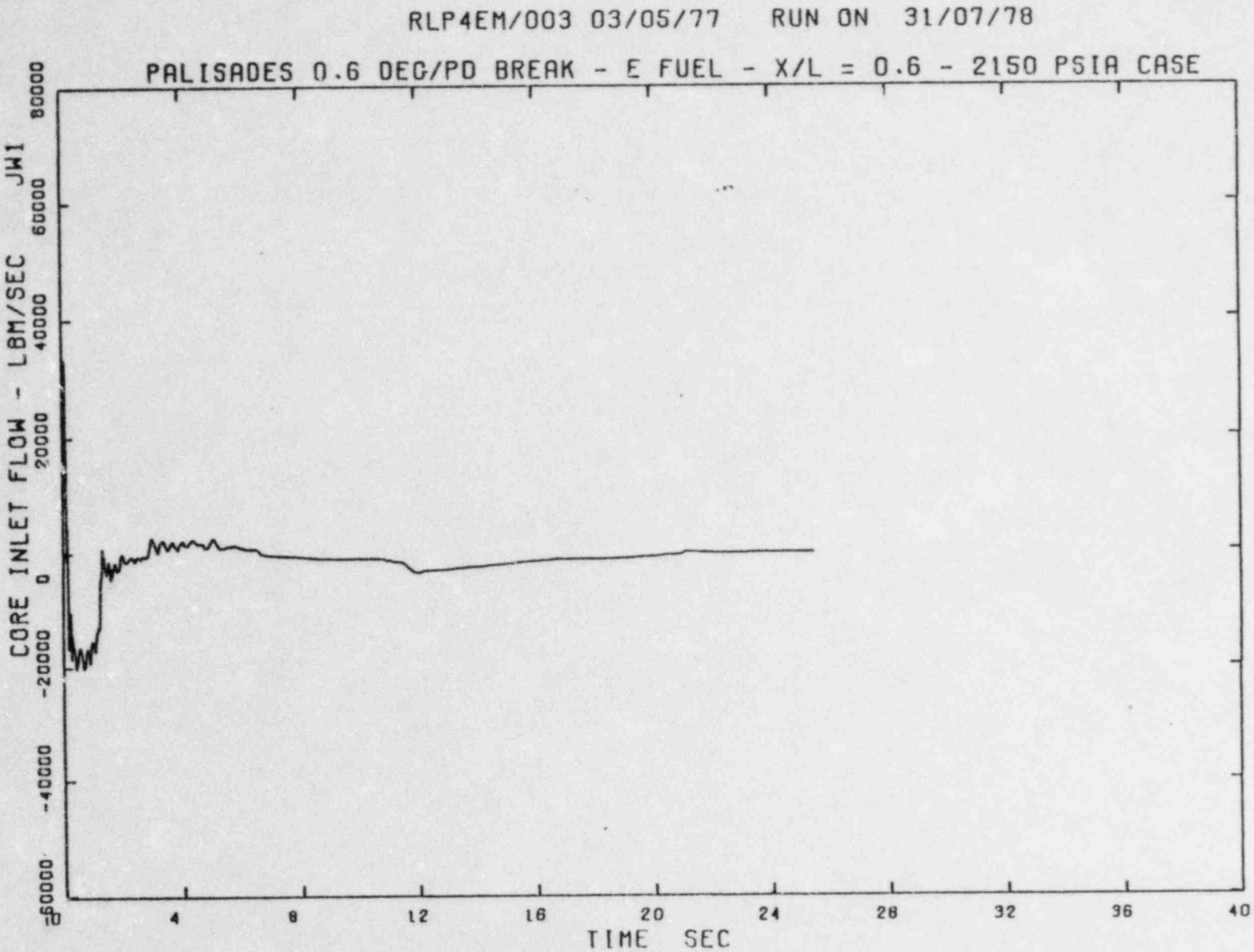

Figure 3.29 Average Core Inlet Flow During Blowdown, 2150 psia Pressurizer Pressure Case

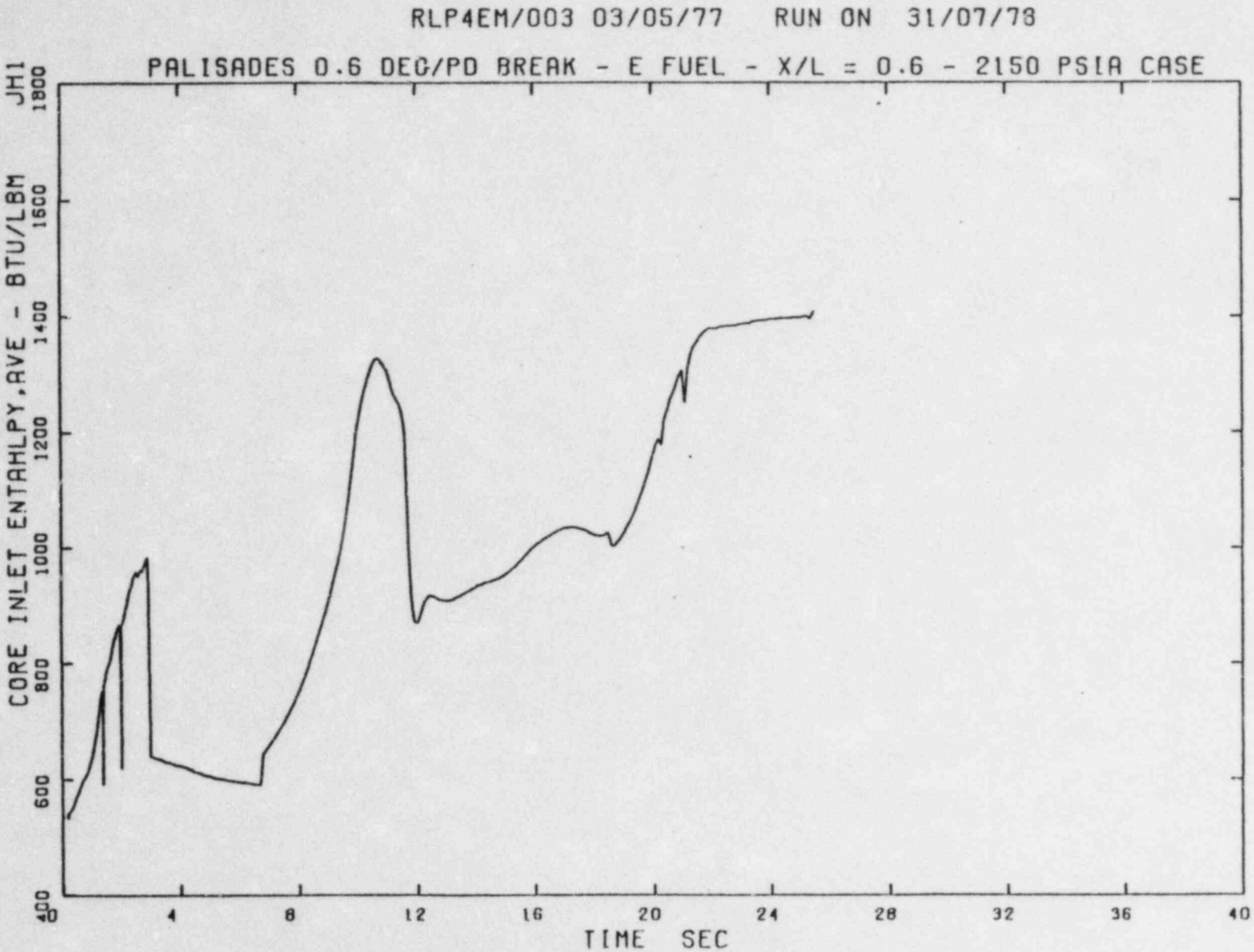

Figure 3.30 Average Core Inlet Enthalpy During Blowdown, 2150 psia Pressurizer Pressure Case

PLOTTED ON 03/25/78

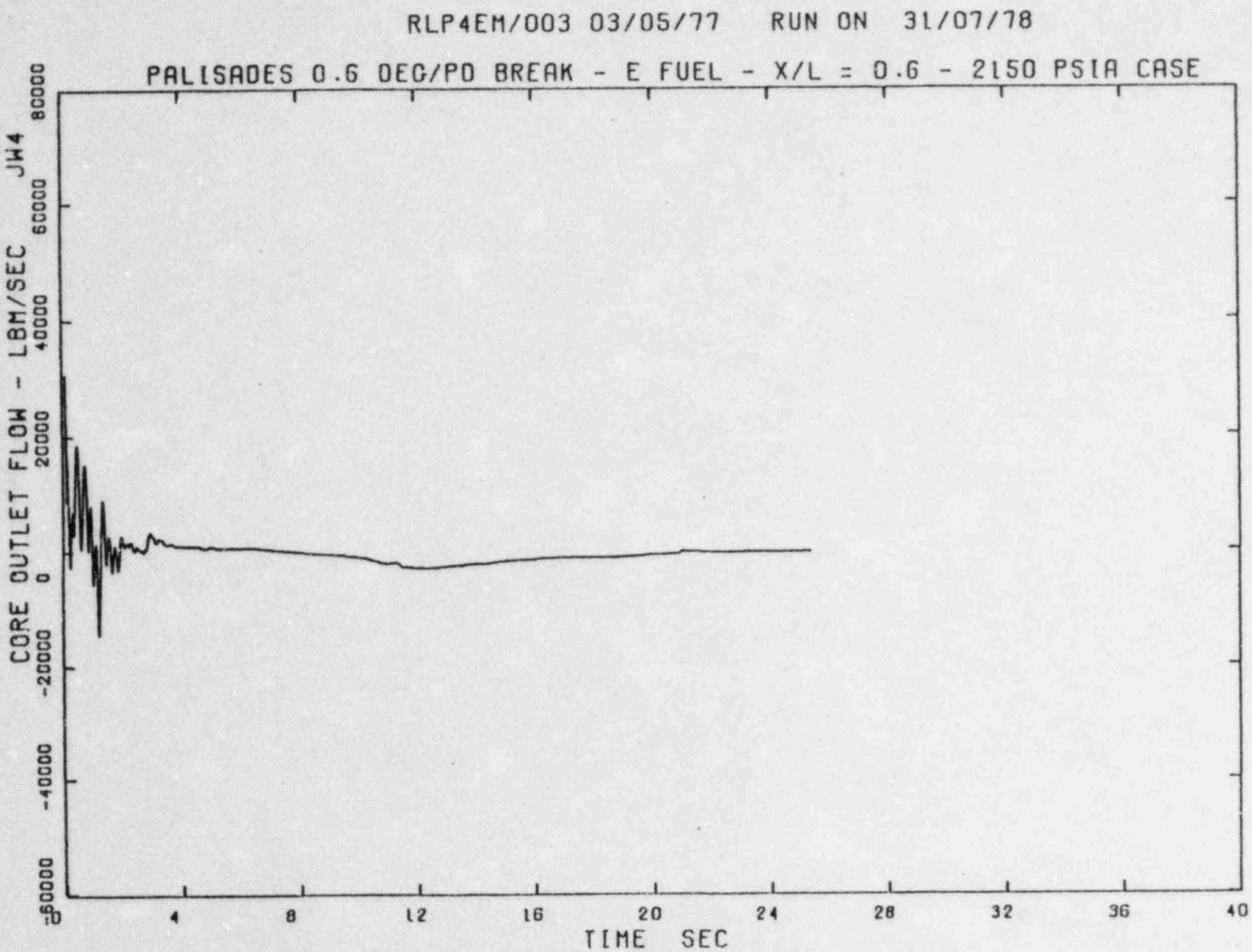

Figure 3.31 Average Core Outlet Enthalpy During Blowdown,
2150 psia Pressurizer Pressure Case

PLOTTED ON 09/25/78

RLP4EM/003 03/05/77 RUN ON 31/07/78

PALISADES 0.6 DEG/PD BREAK - E FUEL - X/L = 0.6 - 2150 PSIA CASE

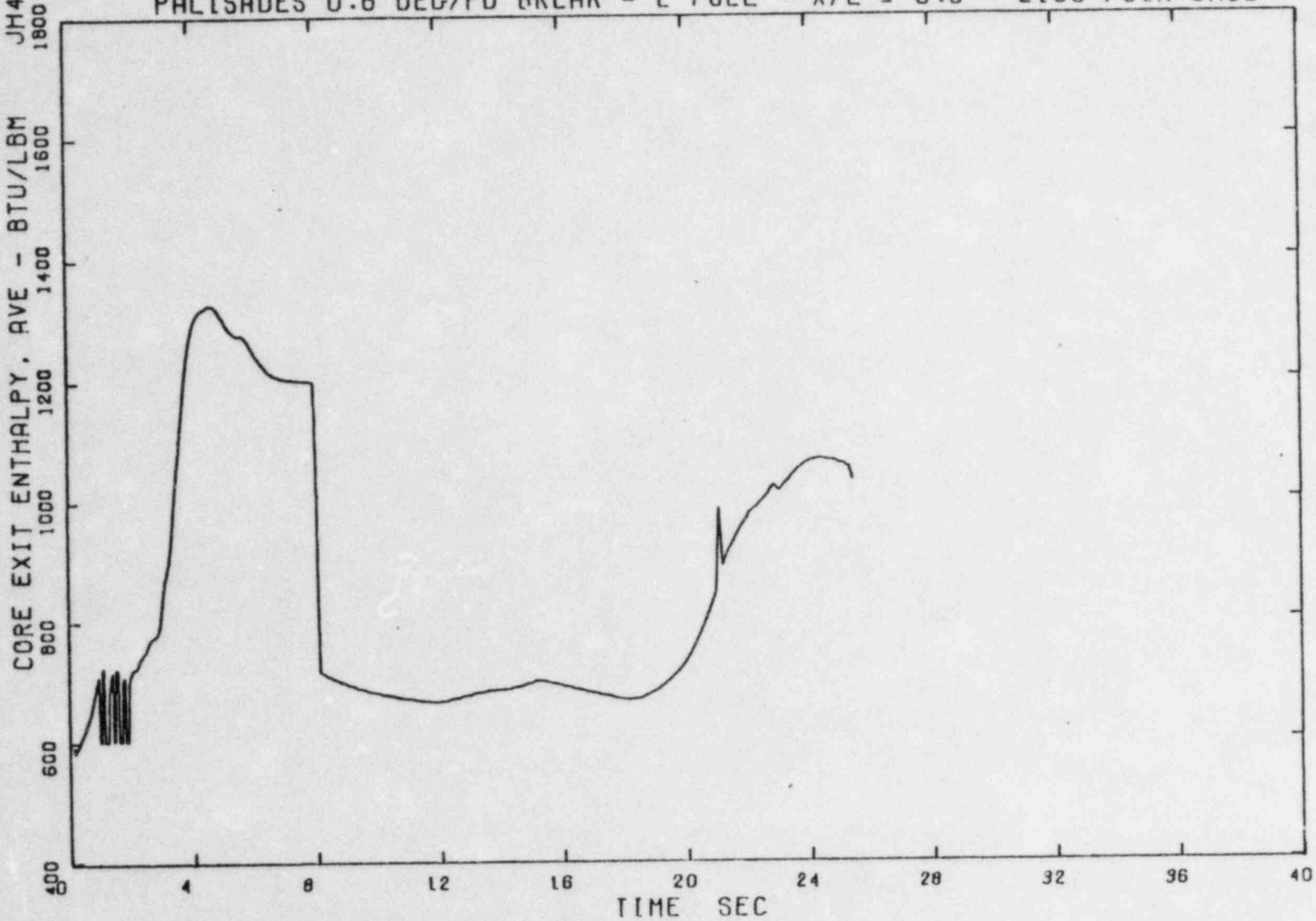

Figure 3.32 Average Core Outlet Enthalpy During Blowdown, 2150 psia Pressurizer Pressure Case

PLOTTED ON 10/11/78

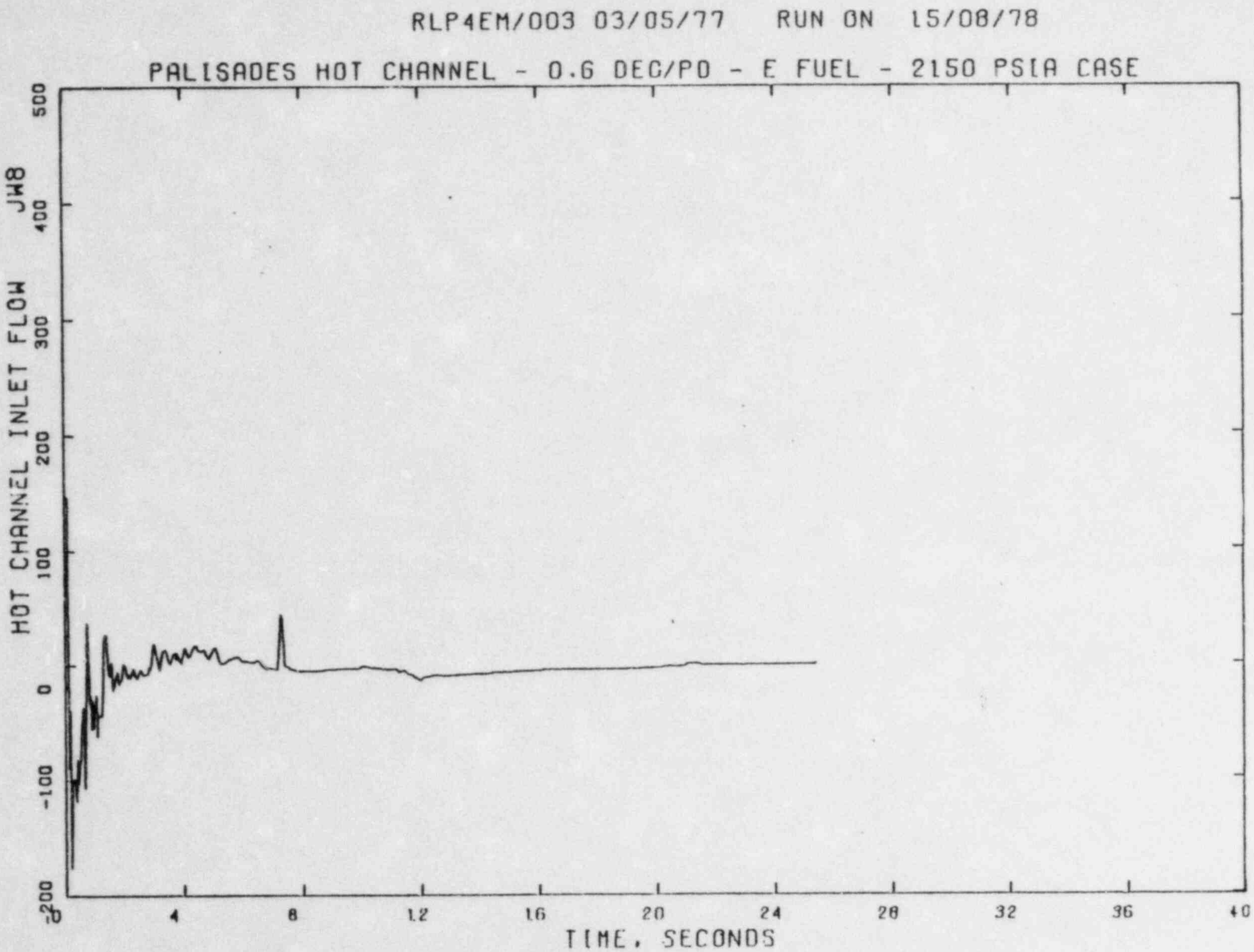

Figure 3.33 Hot Assembly Inlet Flow During Blowdown, 2150 psia Pressurizer Pressure Case

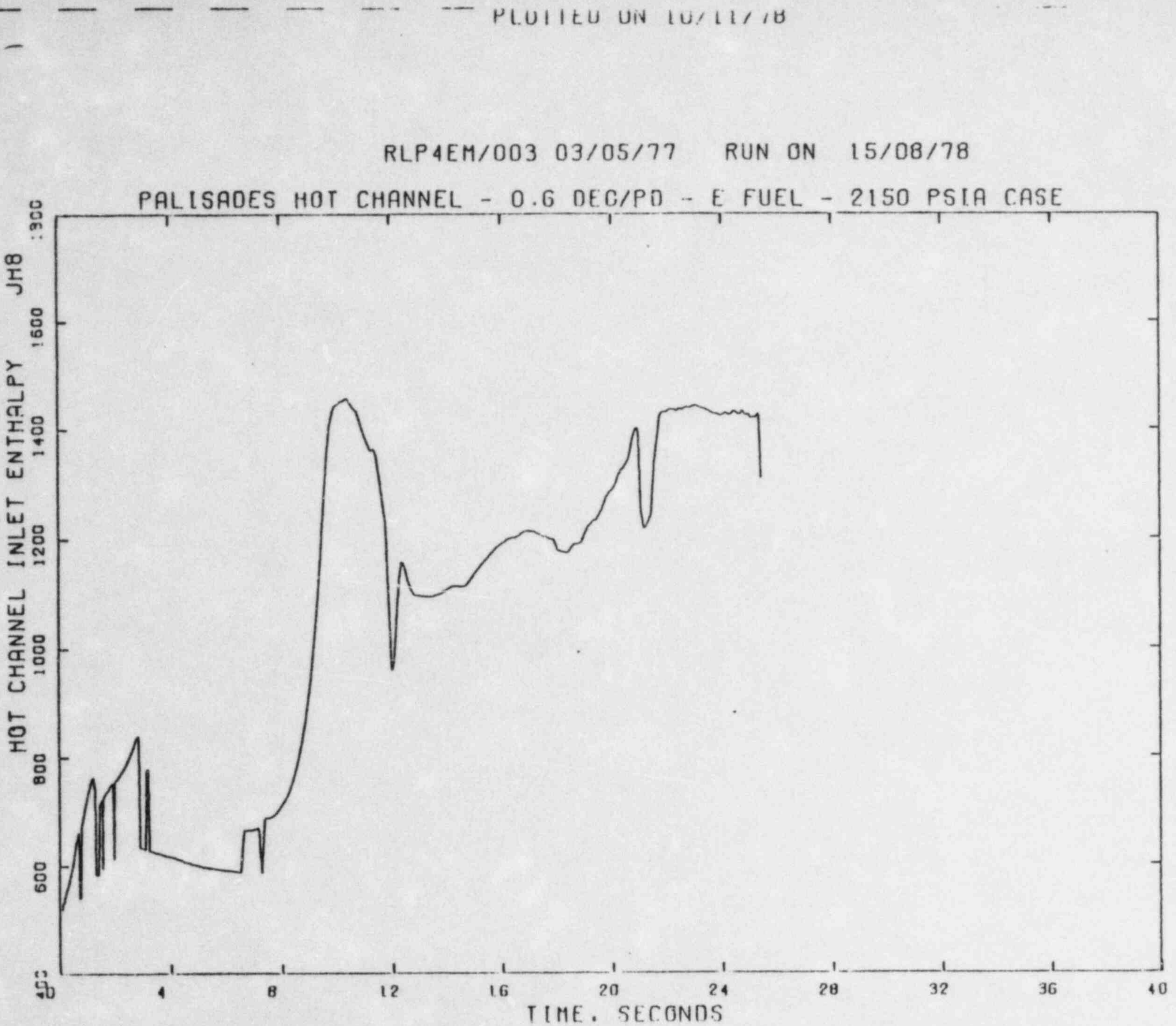

Figure 3.34 Hot Assembly Inlet Enthalpy During Blowdown,
2150 psia Pressurizer Pressure Case

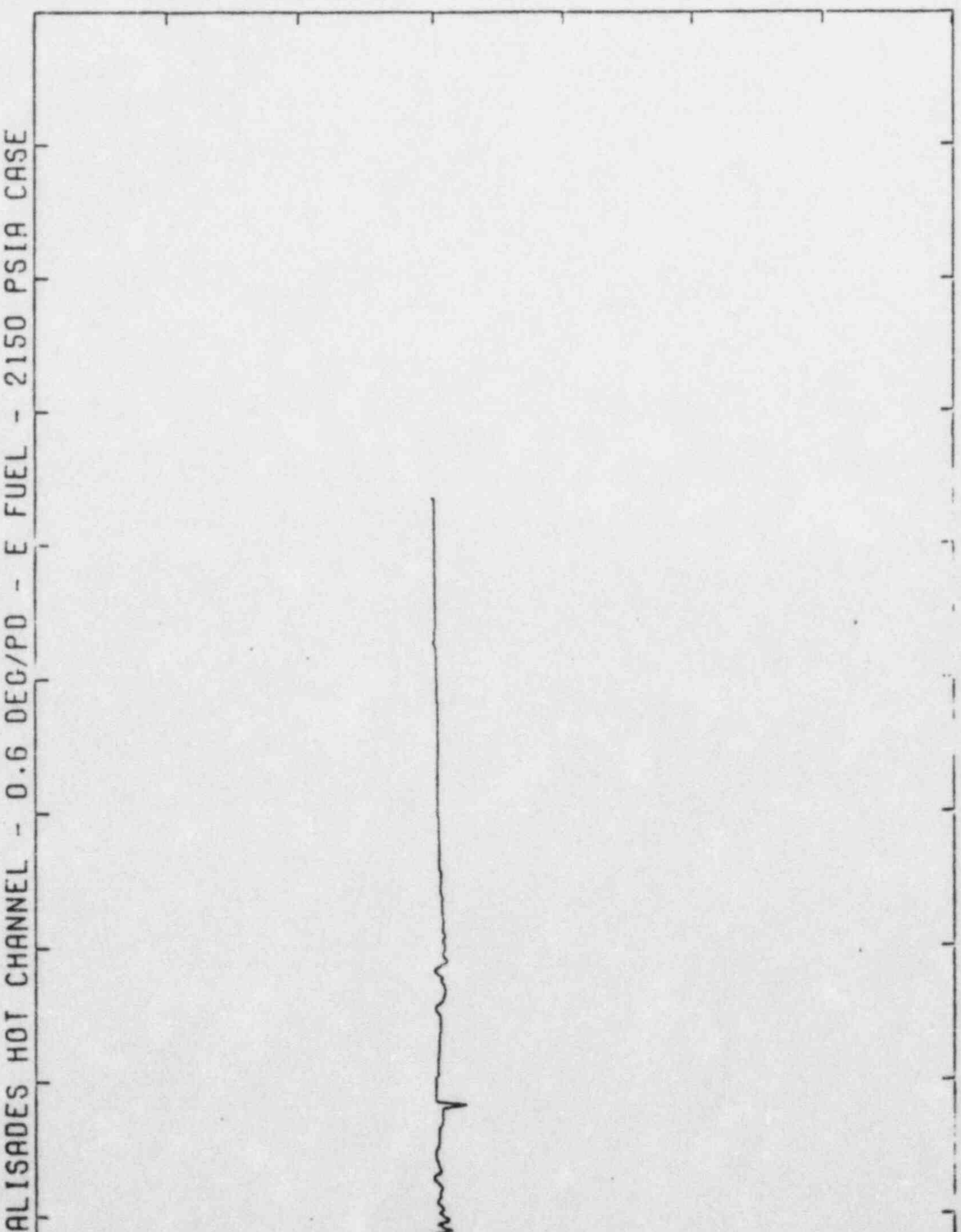

Figure 3.35 Hot Assembly Outlet Flow During Blowdown,
2150 psia Pressurizer Pressure Case

PLOTTED ON 10/11/78

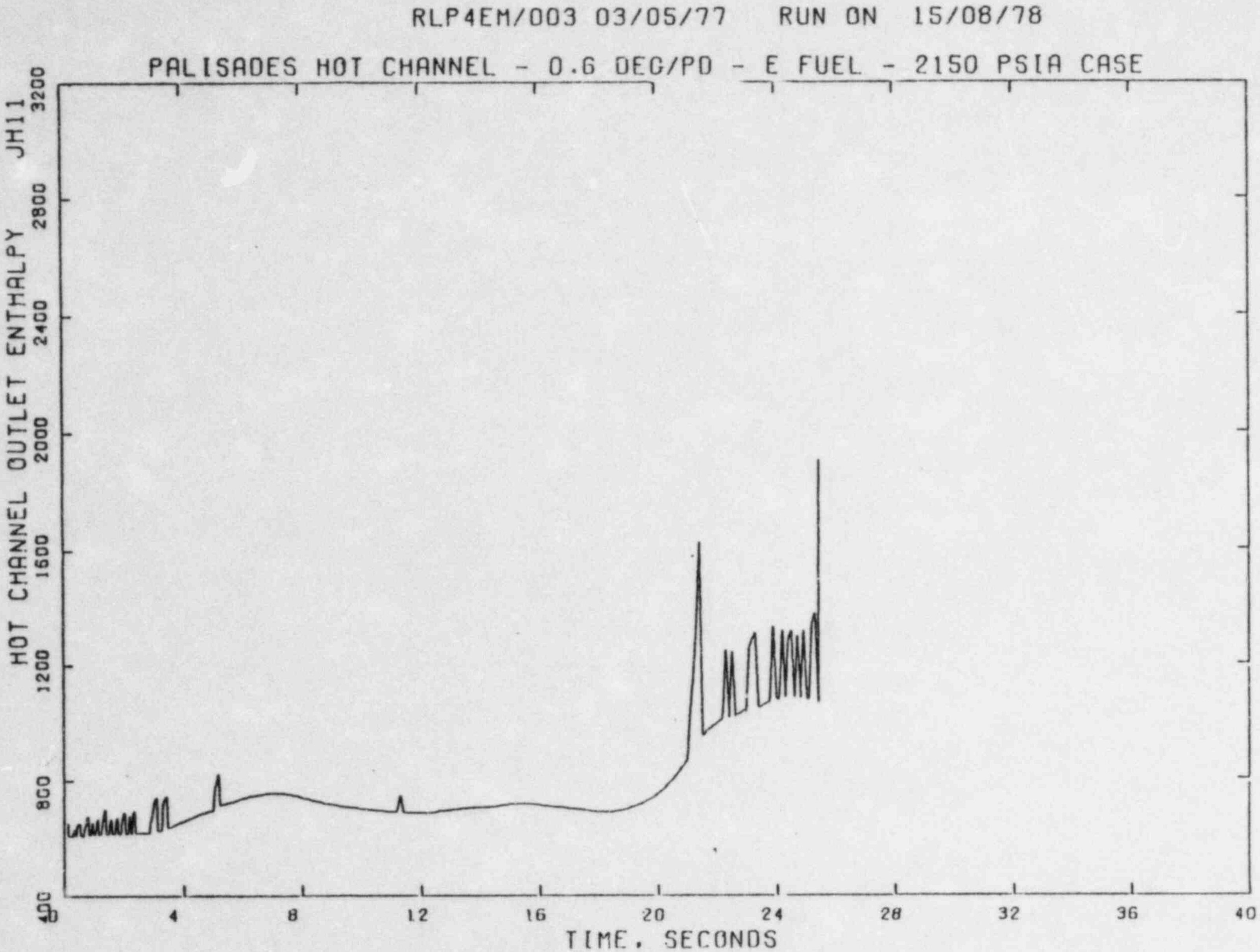

Figure 3.36 Hot Rod Clad Surface Temperature at Peak Power Location During Blowdown, 2150 psia Pressurizer Pressure Case

PLOTTED ON 10/11/78

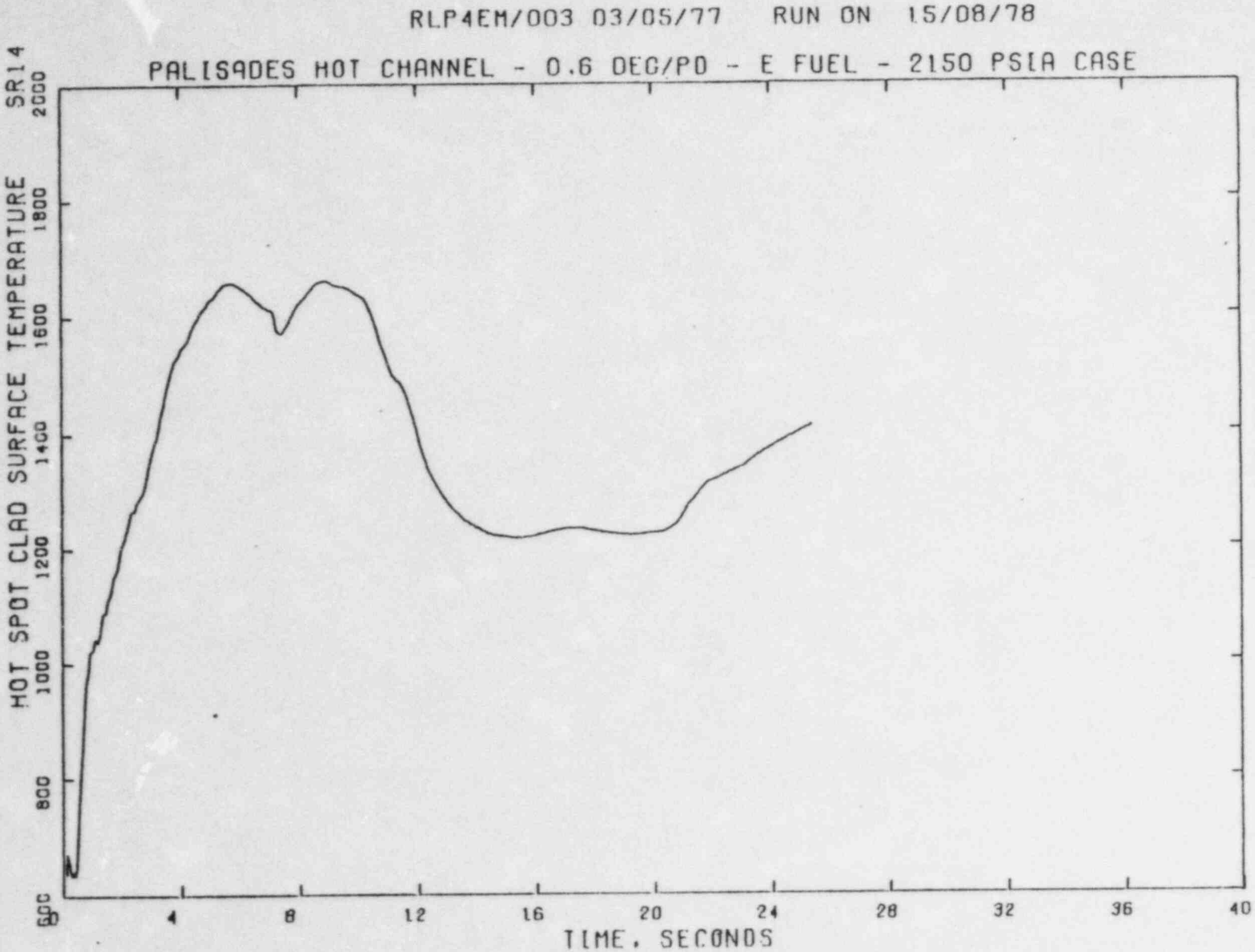

Figure 3.37 Hot Rod Clad Surface Temperature at Peak Power Location
During Blowdown, 2150 psia Pressurizer Pressure Case

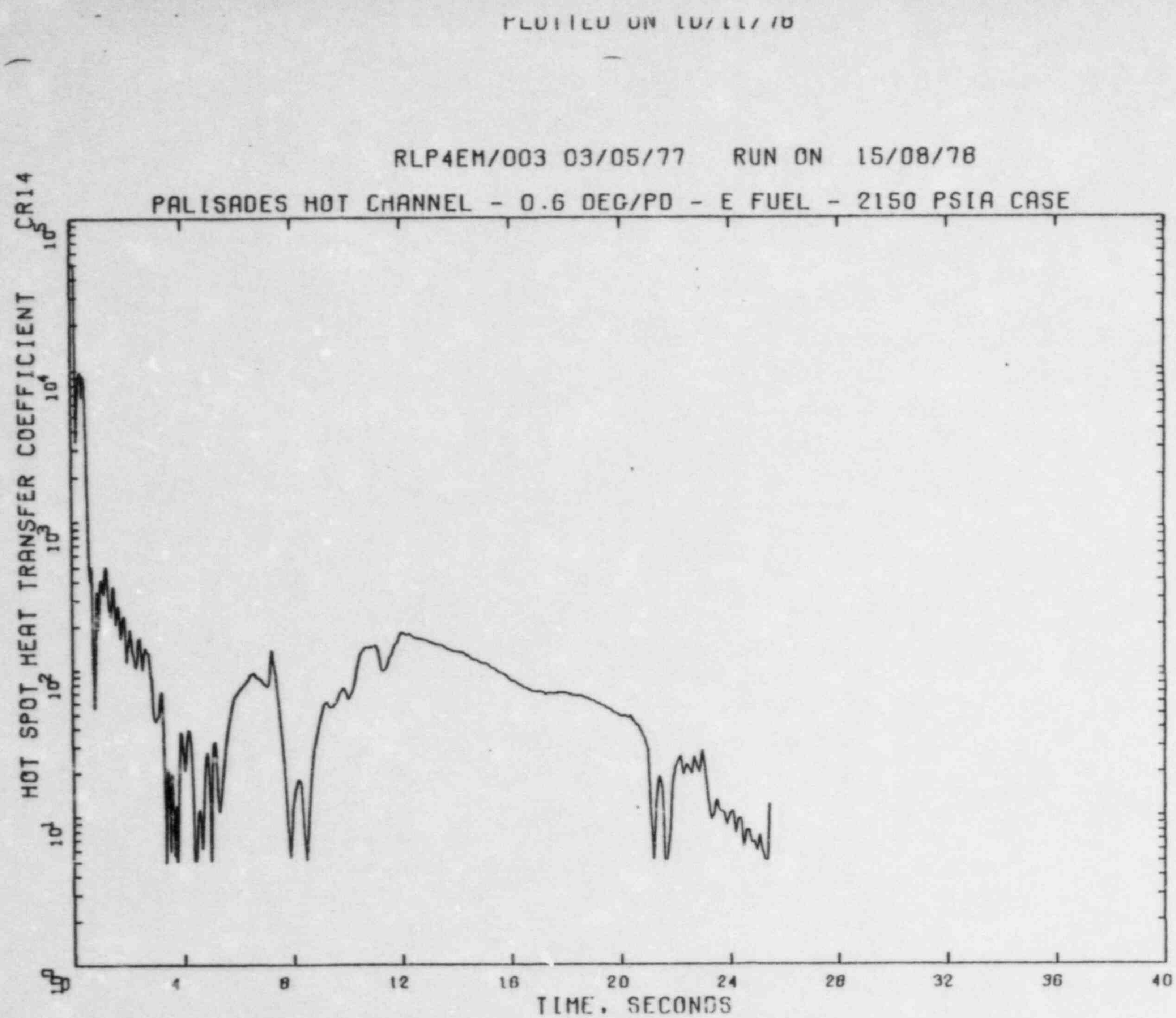

Figure 3.38 Hot Rod Heat Transfer Coefficient At Peak Power Location During Blowdown, 2150 psia Pressurizer Pressure Case

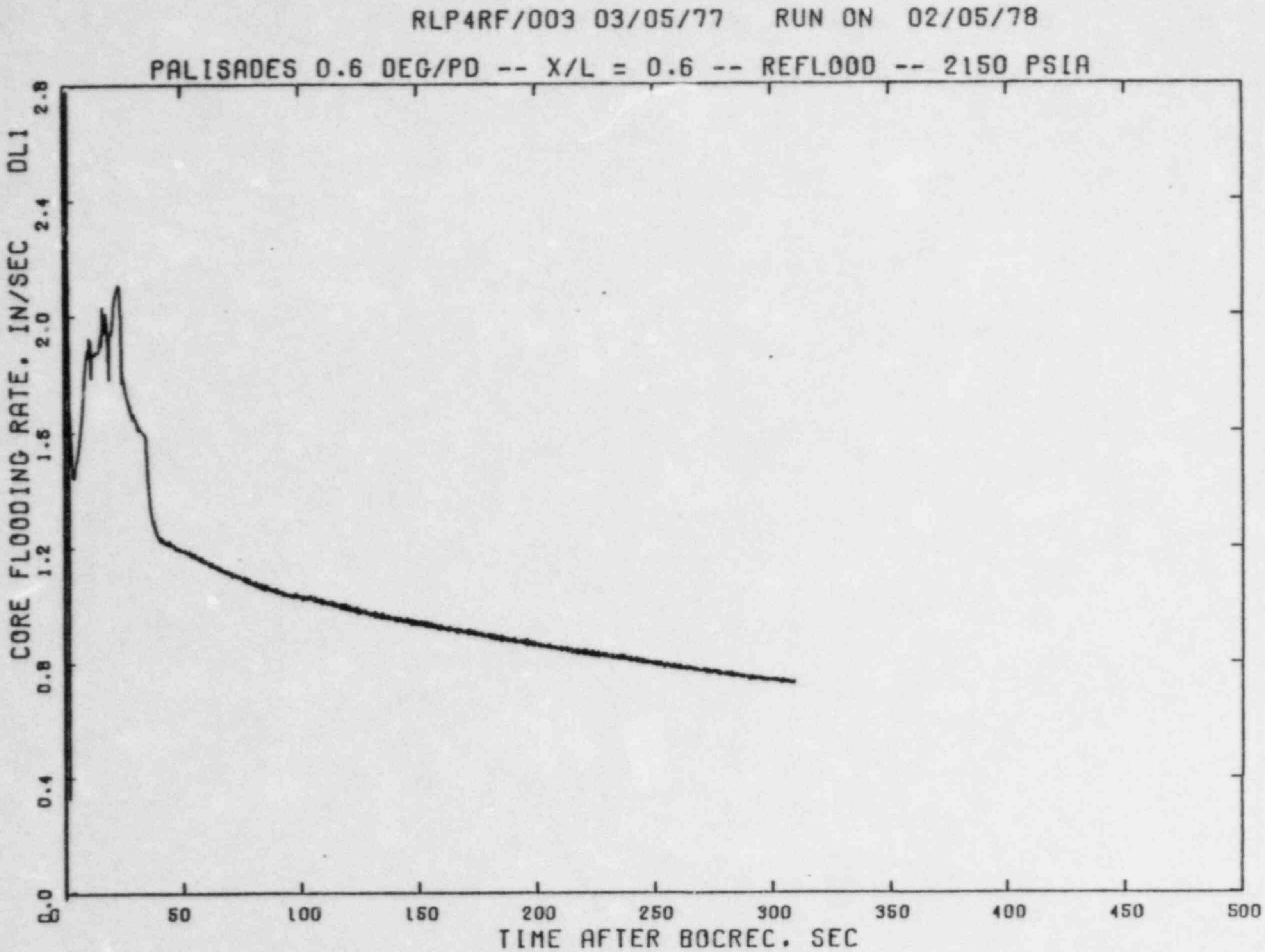

Figure 3.39 Core Reflood Rate During Reflood,
2150 psia Pressurizer Pressure Case

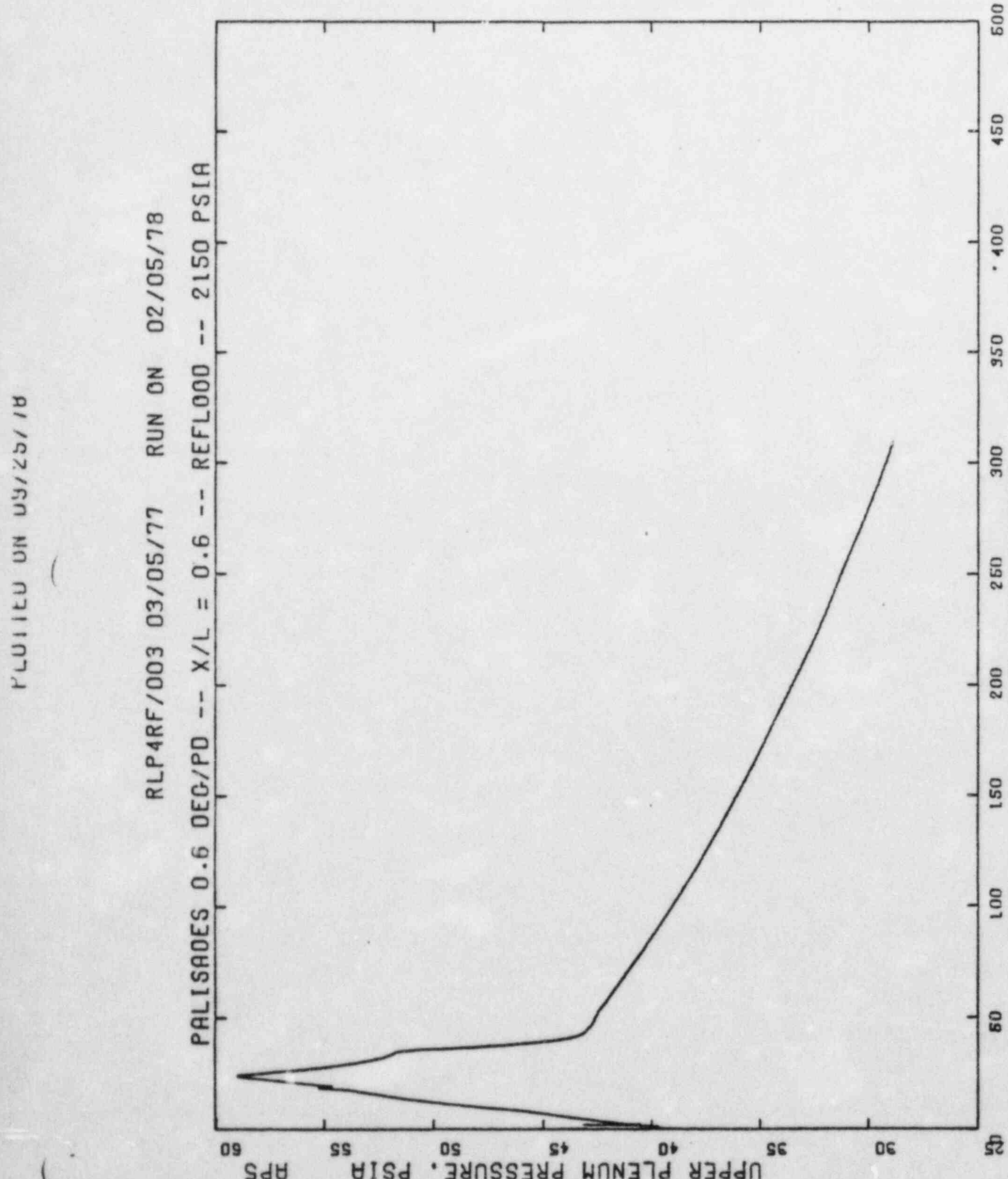

Figure 3.40 Upper Plenum Pressure During Reflood,
2150 psia Pressurizer Pressure Case

PLOTTED ON 09/25/78

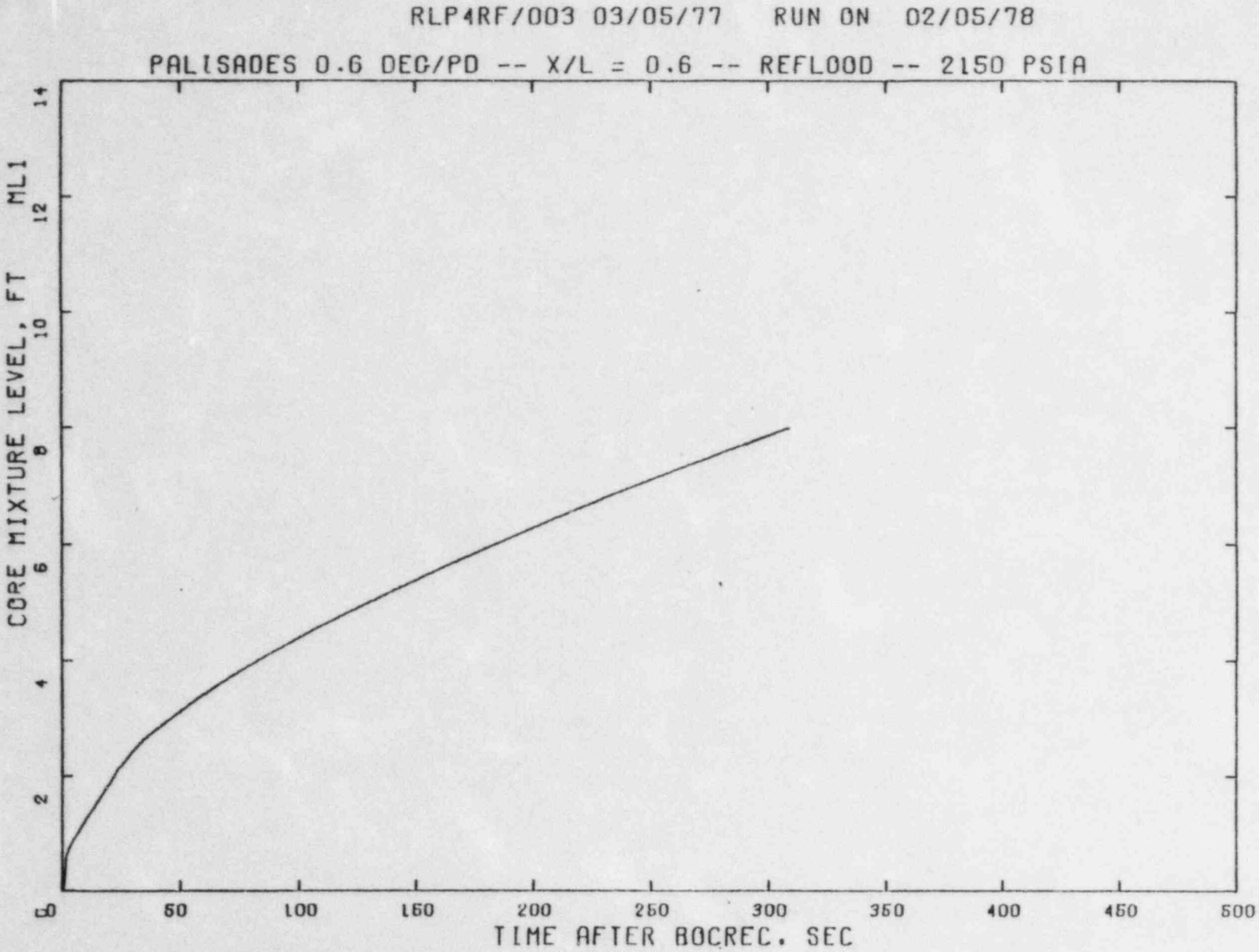

Figure 3.41 Collapsed Liquid Level in Core During Reflood, 2150 psia Pressurizer Pressure Case

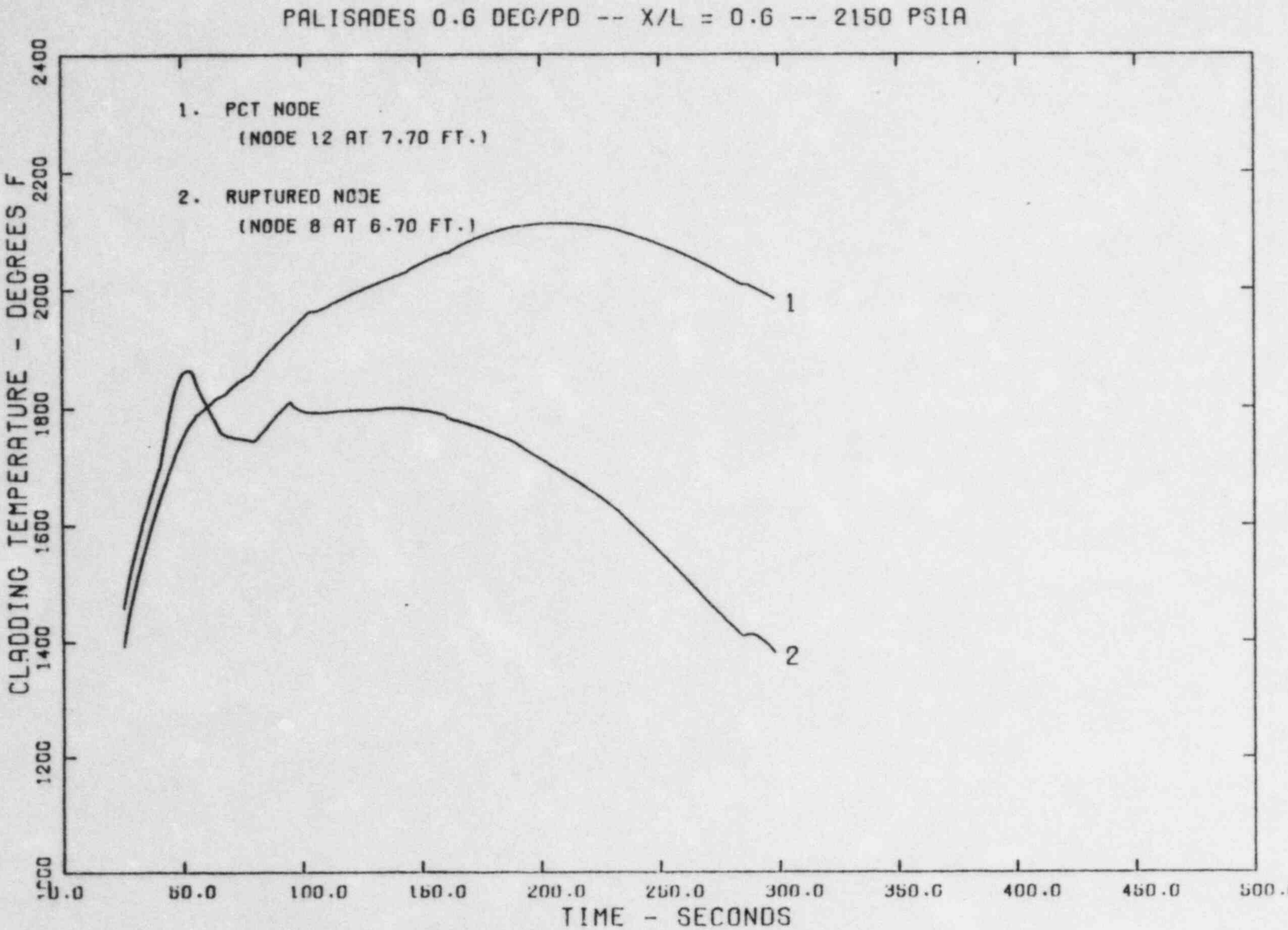

Figure 3.42 Cladding Temperature at PCT and Rupture Locations, 2150 psia Pressurizer Pressure Case

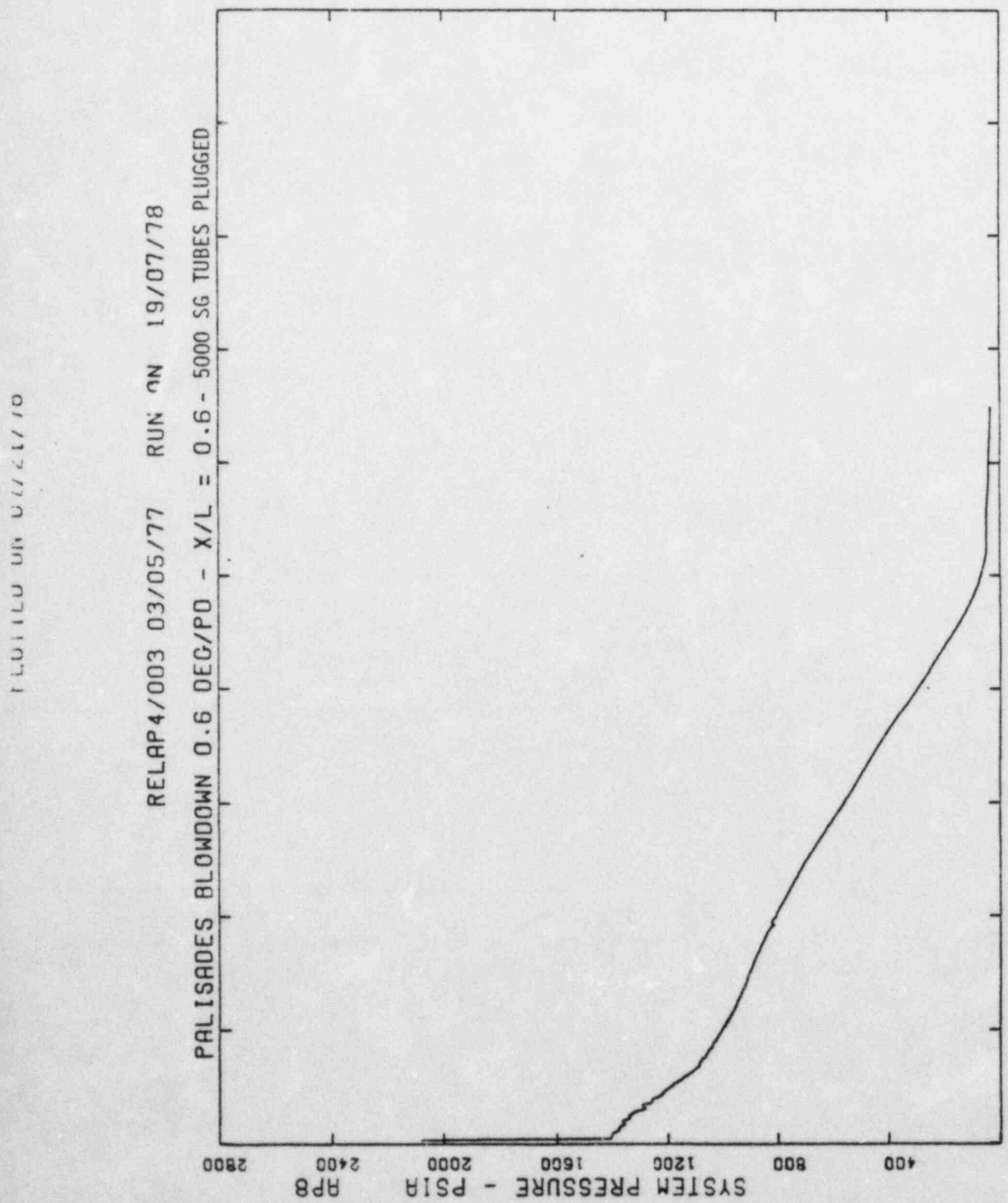

Figure 3.43 Upper Plenum Pressure During Blowdown, 5000 Tubes Plugged Case

PLOTTED ON 07/21/78

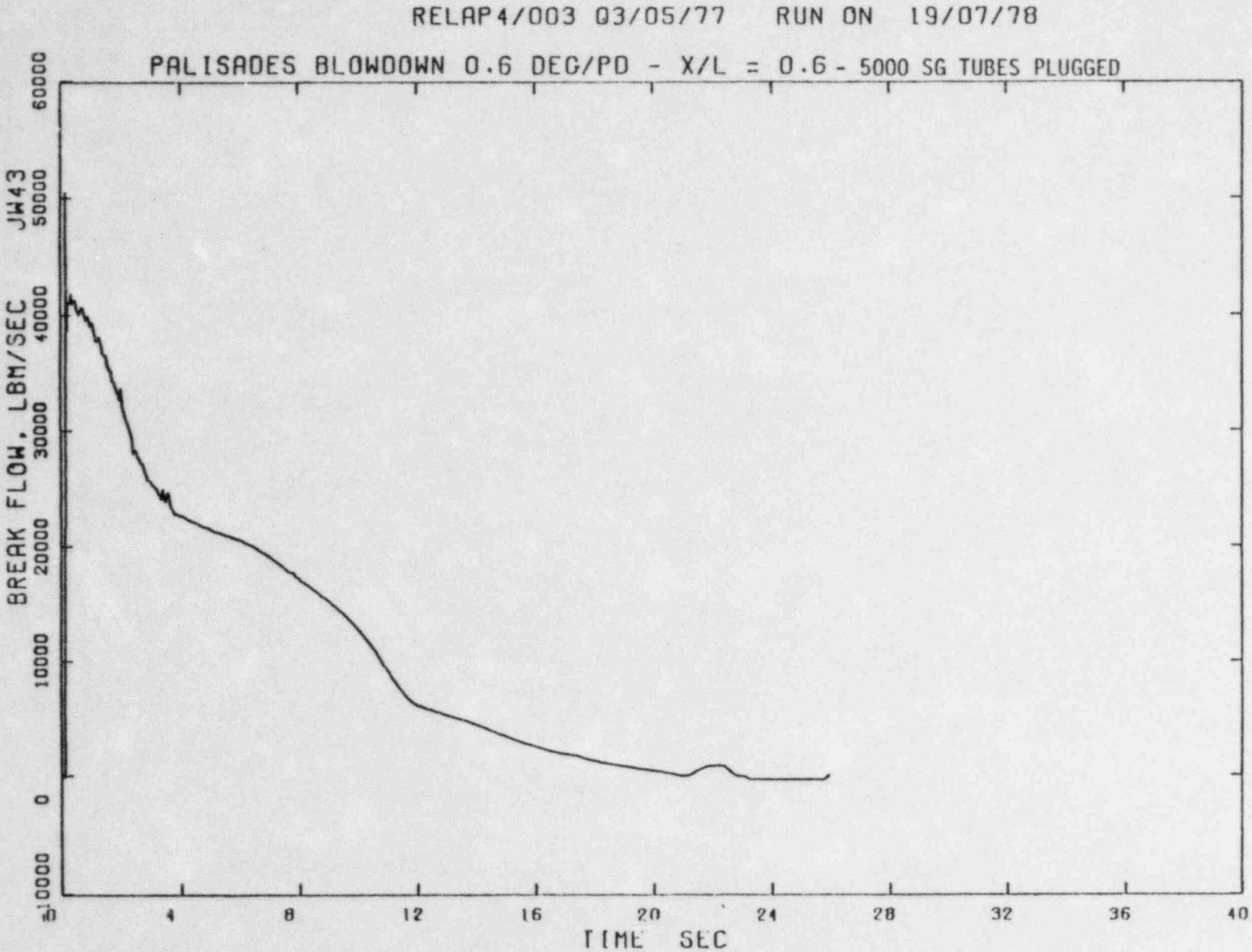

Figure 3.44 Pump Side Break Flow During Blowdown,
5000 Tubes Plugged Case

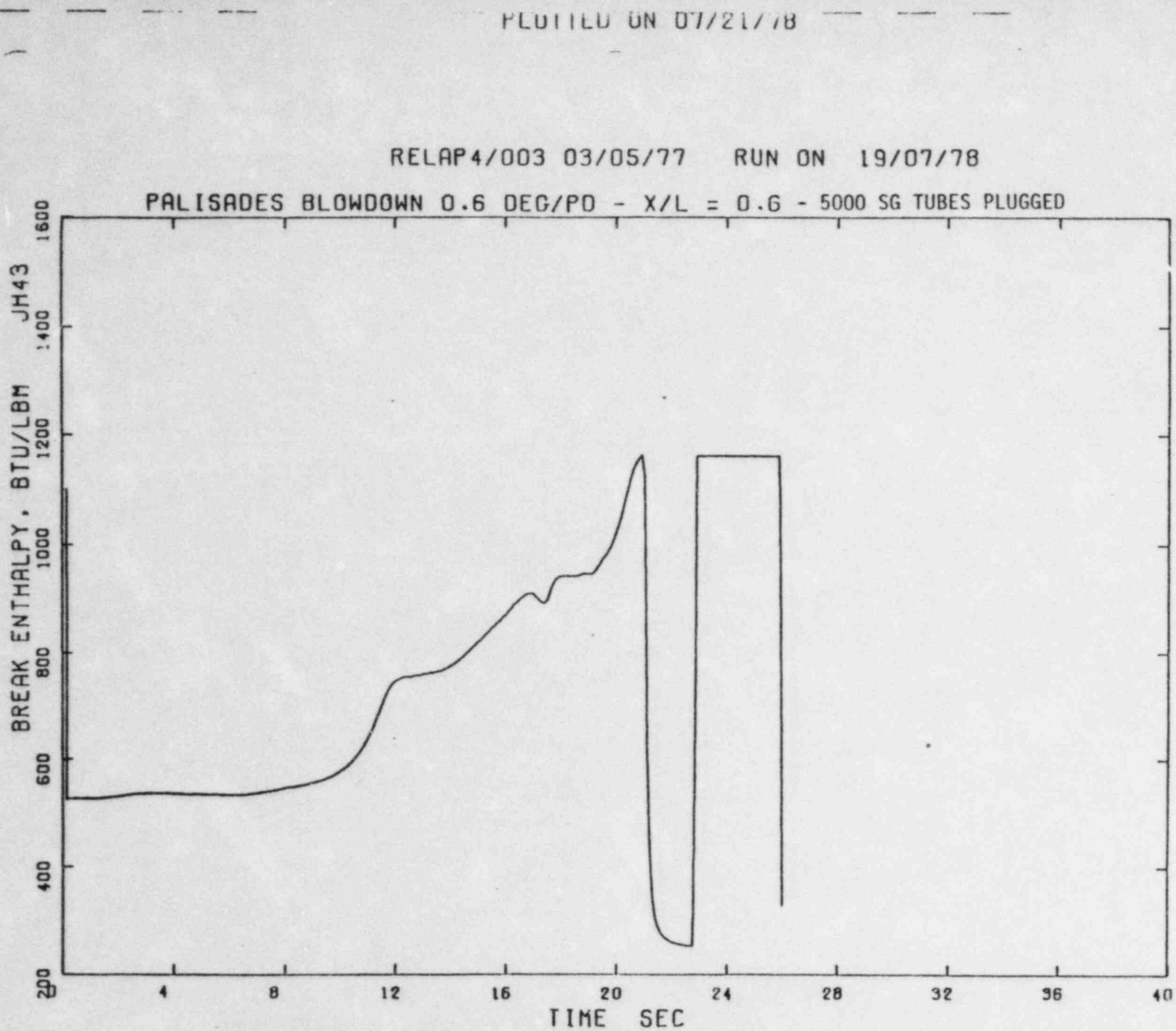

Figure 3.45 Pump Side Break Enthalpy During Blowdown,
5000 Tubes Plugged Case

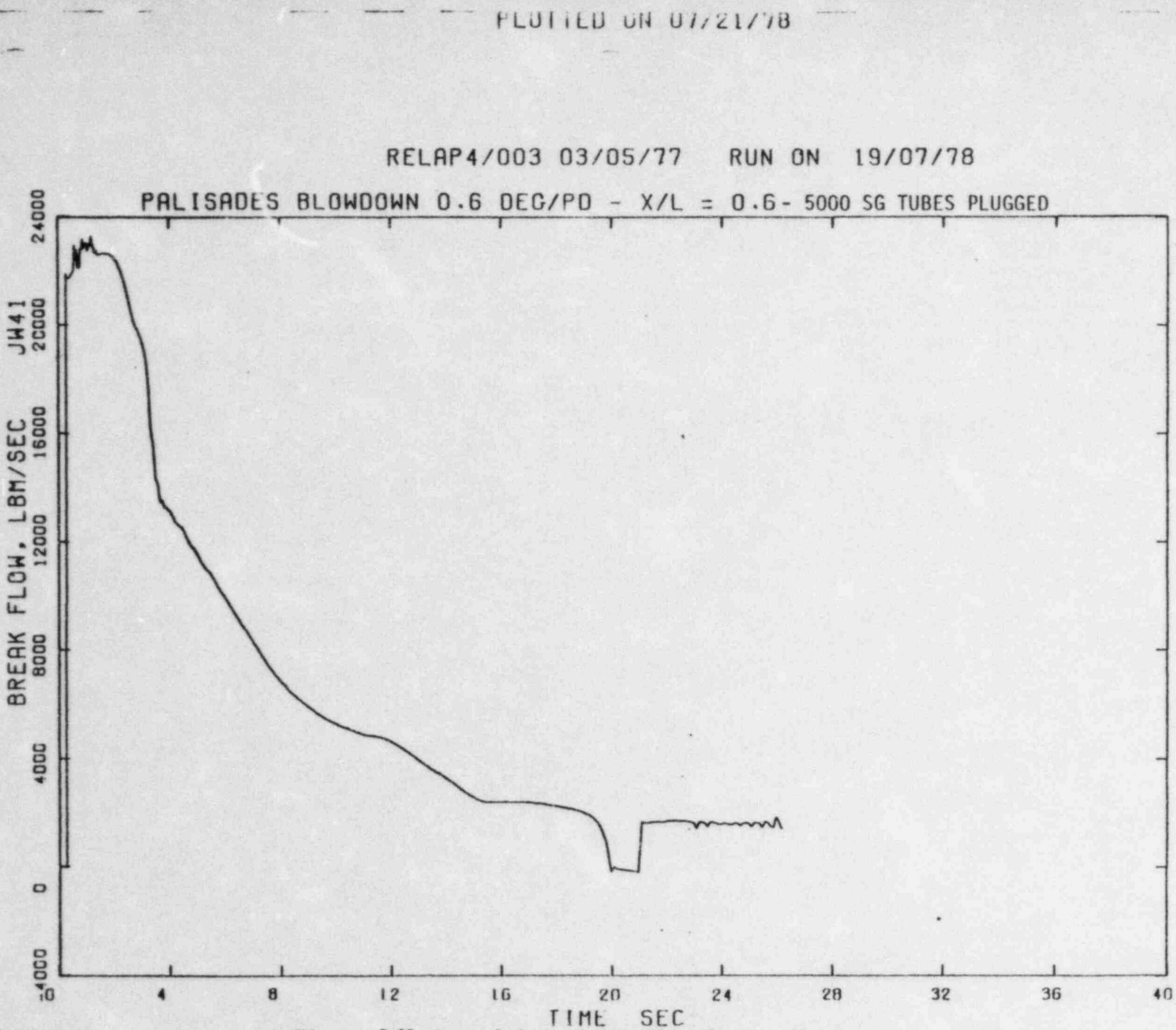

Figure 3.46 Vessel Side Break Flow During Blowdown, 5000 Tubes Plugged Case

PLOTTED ON 07/21/78

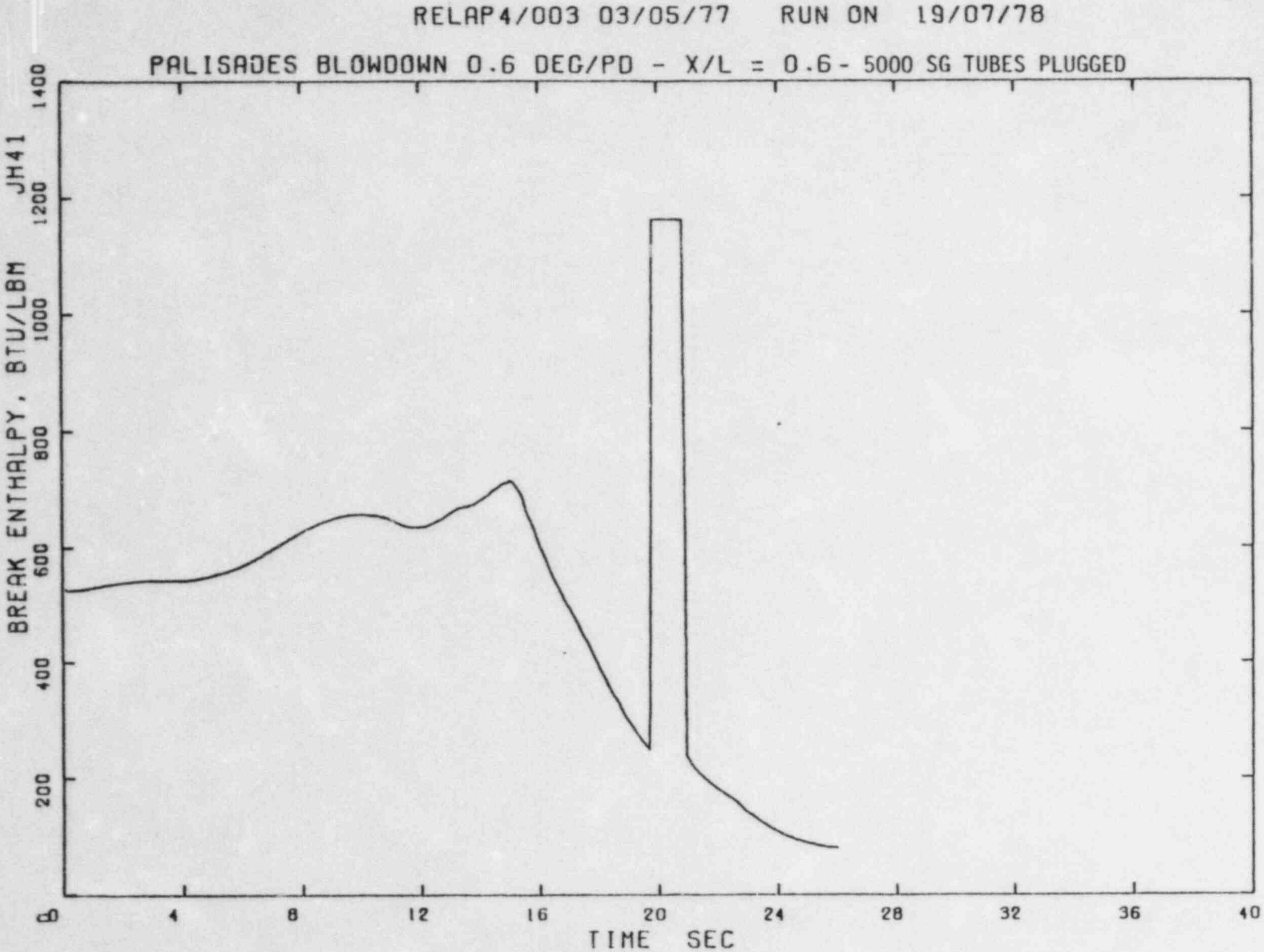

Figure 3.47 Vessel Side Break Enthalpy During Blowdown, 5000 Tubes Plugged Case

PLOTTED ON 07/21/78

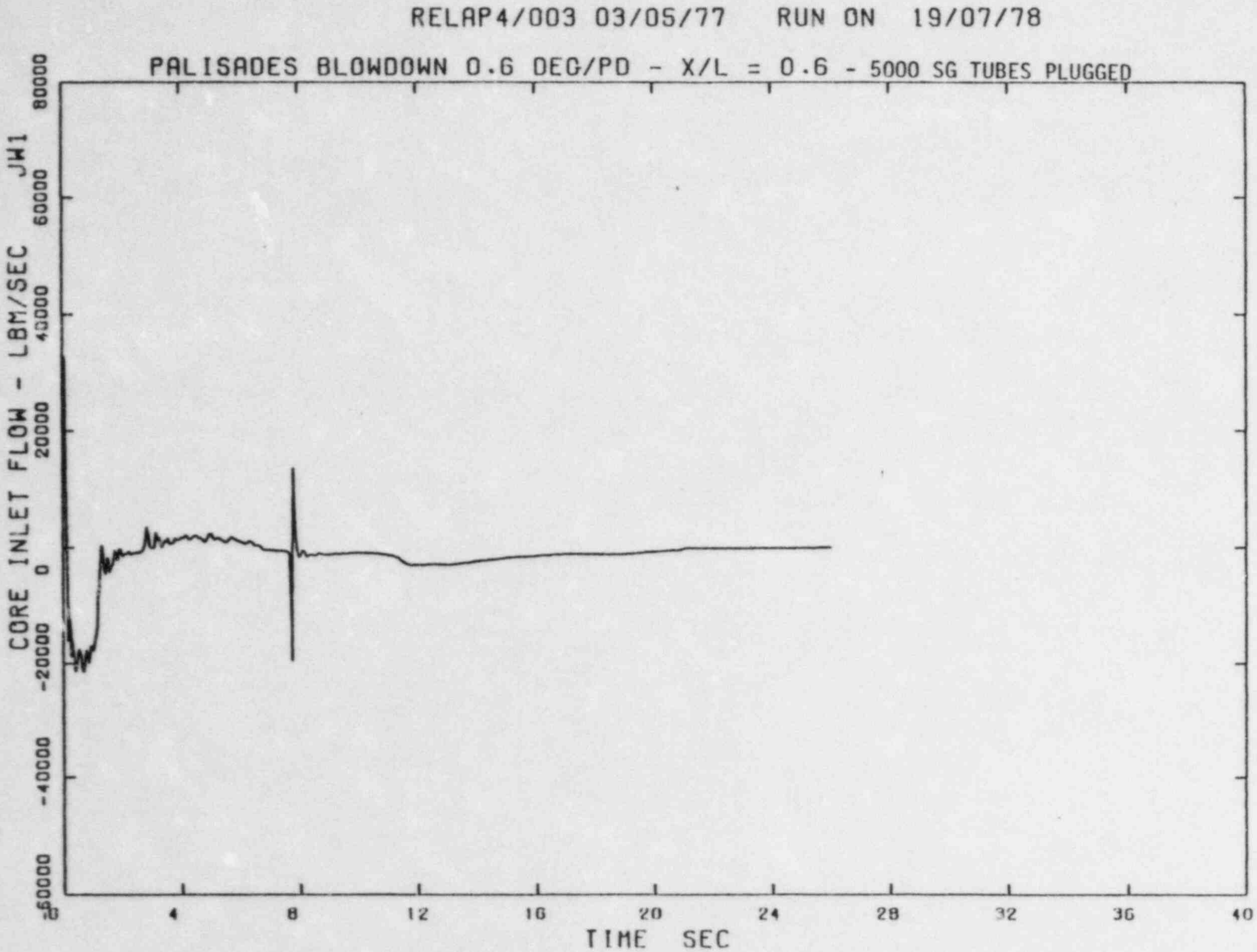

Figure 3.48 Average Core Inlet Flow Dring Blowdown,
5000 Tubes Plugged Case

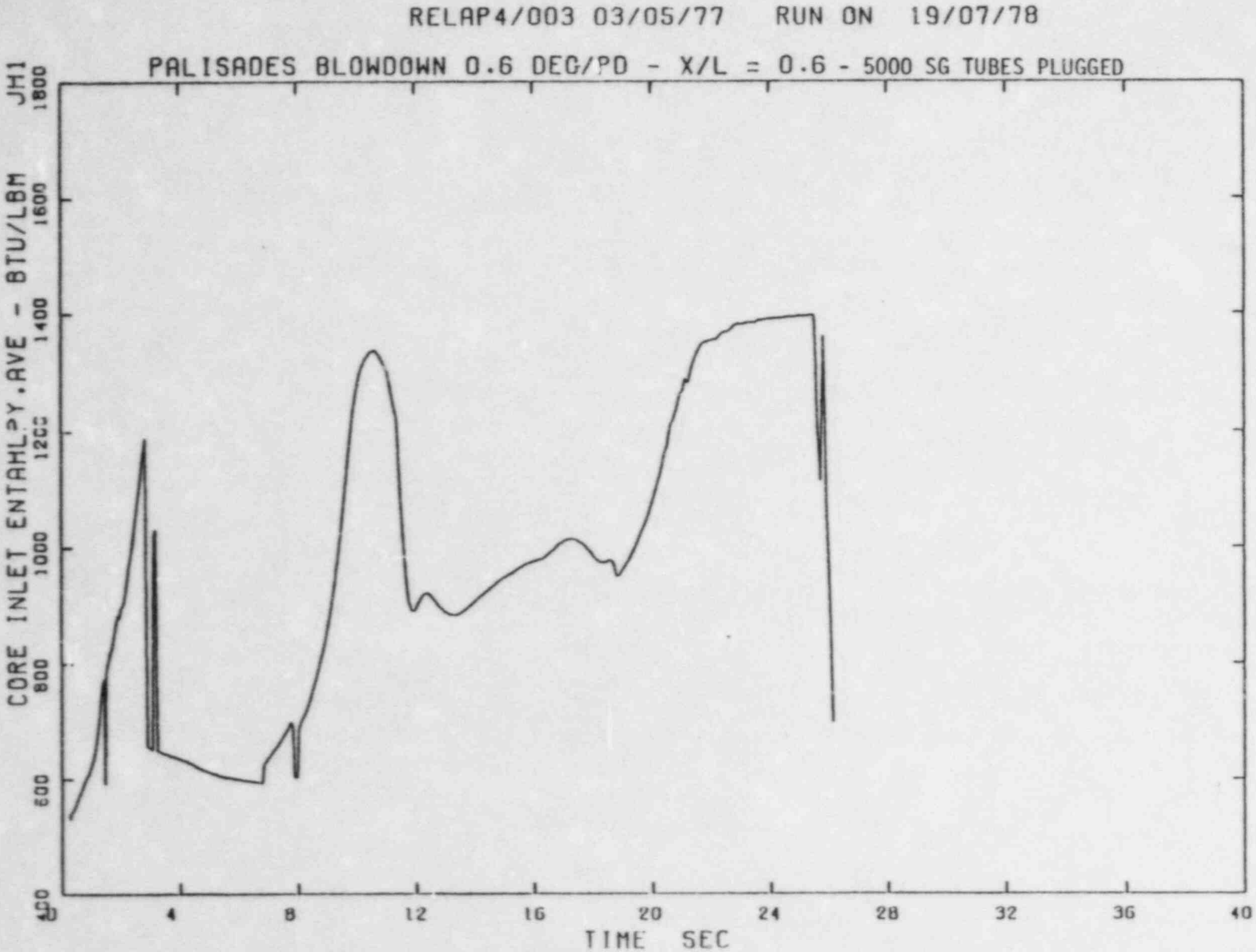

Figure 3.49 Average Core Inlet Enthalpy During Blowdown, 5000 Tubes Plugged Case

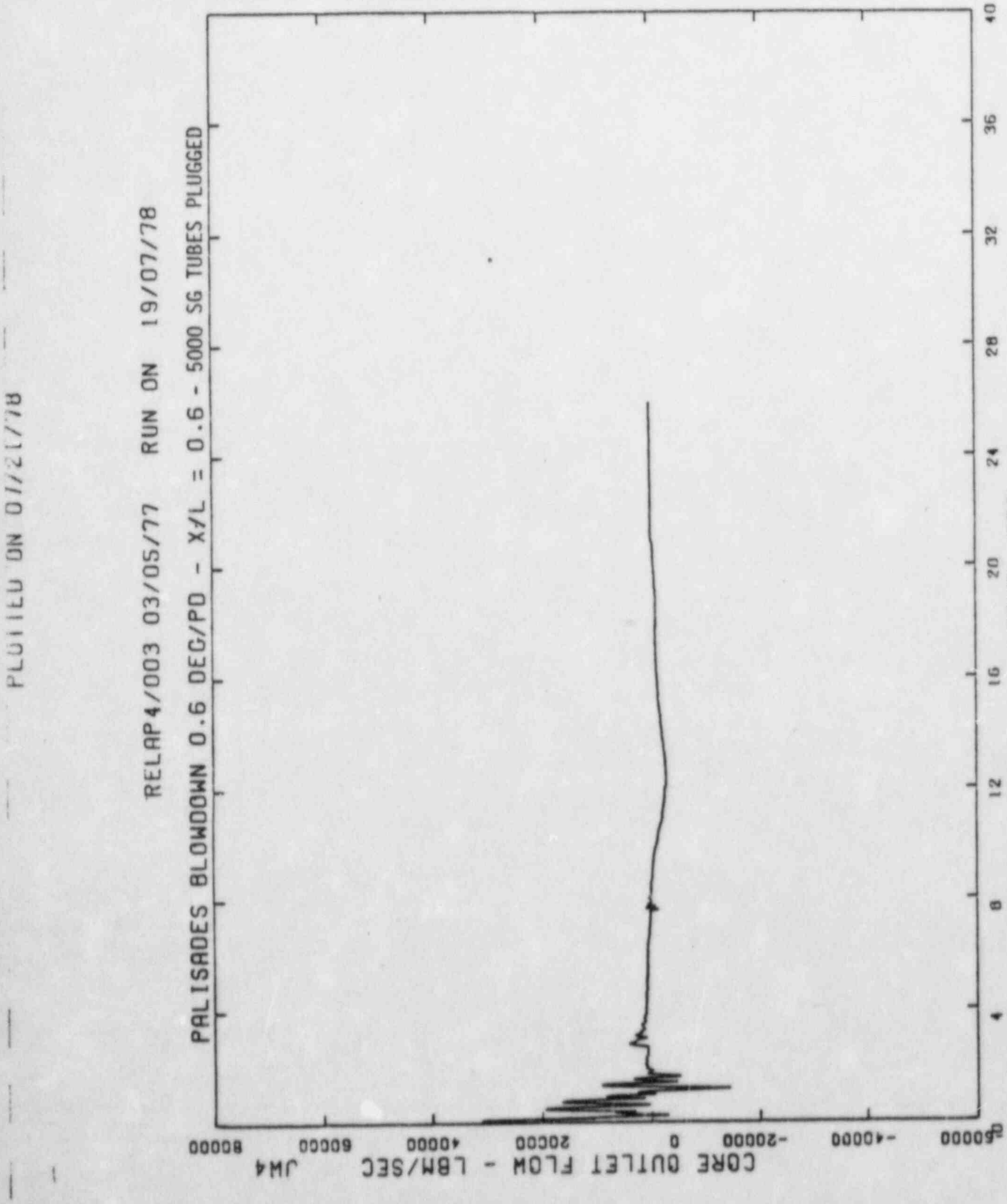

Figure 3.50 Average Core Outlet Flow During Blowdown,
5000 Tubes Plugged Case

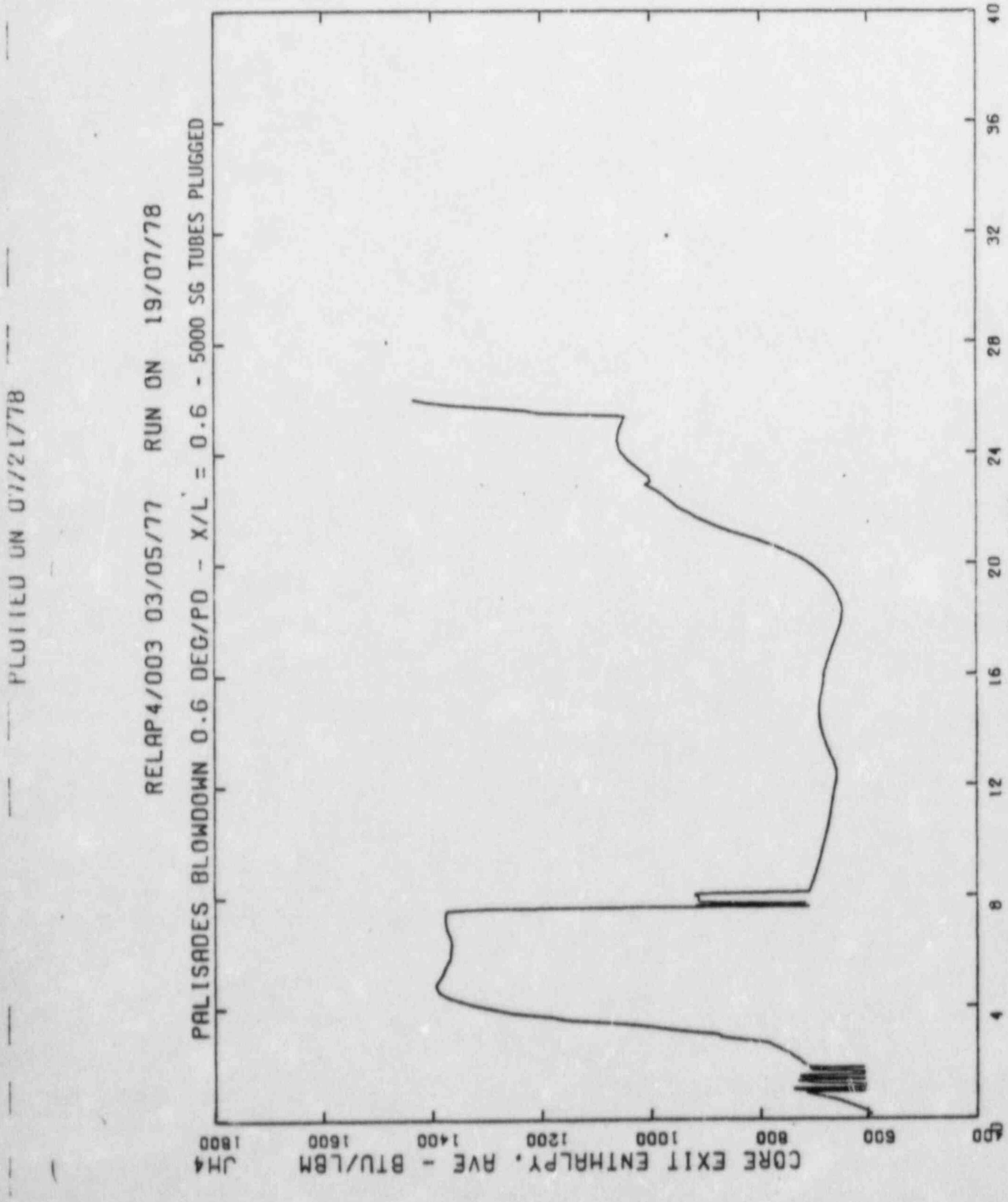

Figure 3.51 Average Core Outlet Enthalpy During Blowdown, 5000 Tubes Plugged Case

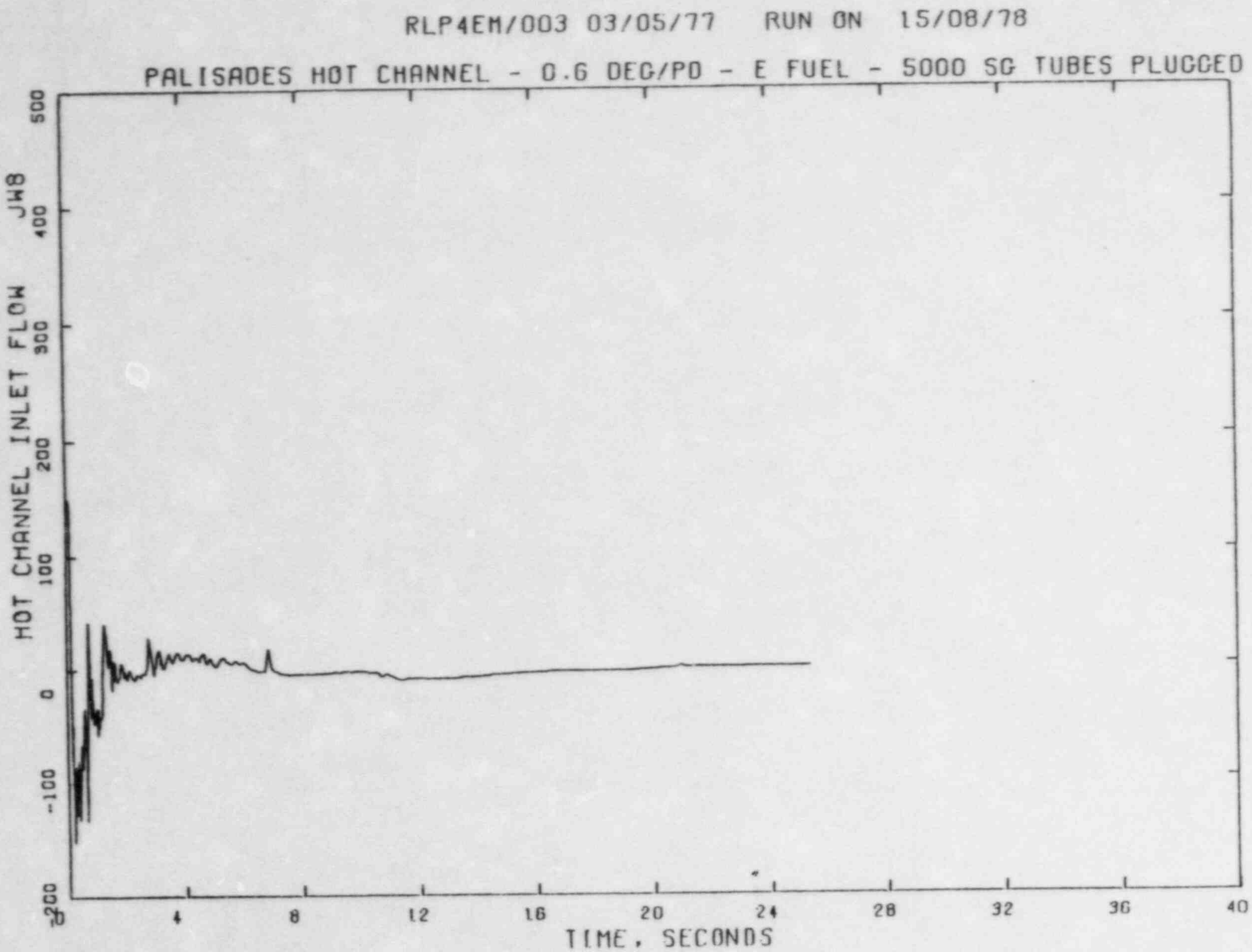

Figure 3.52 Hot Assembly Inlet Flow During Blowdown,
5000 Tubes Plugged Case

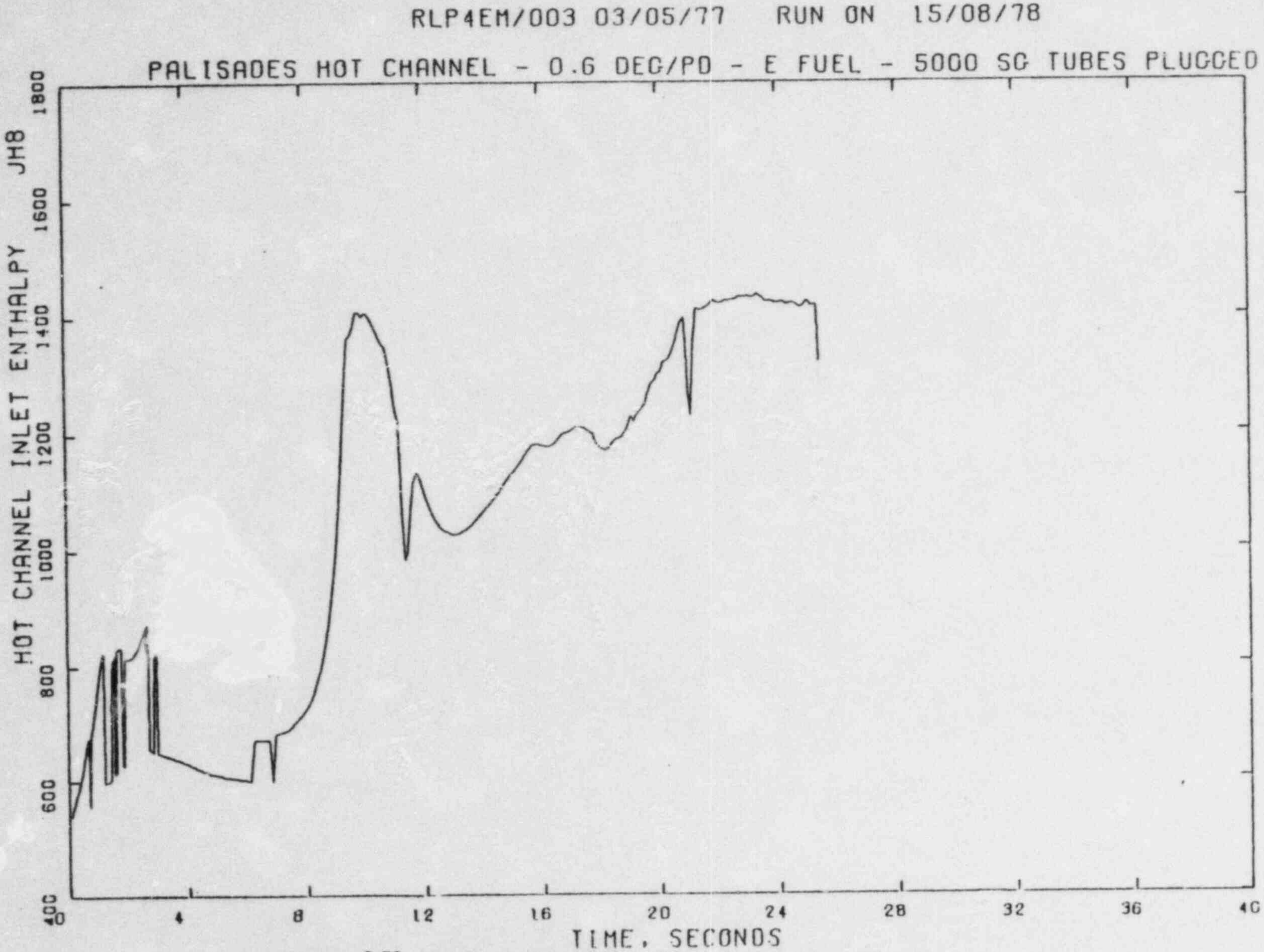

Figure 3.53 Hot Assembly Inlet Enthalpy During Blowdown, 5000 Tubes Plugged Case

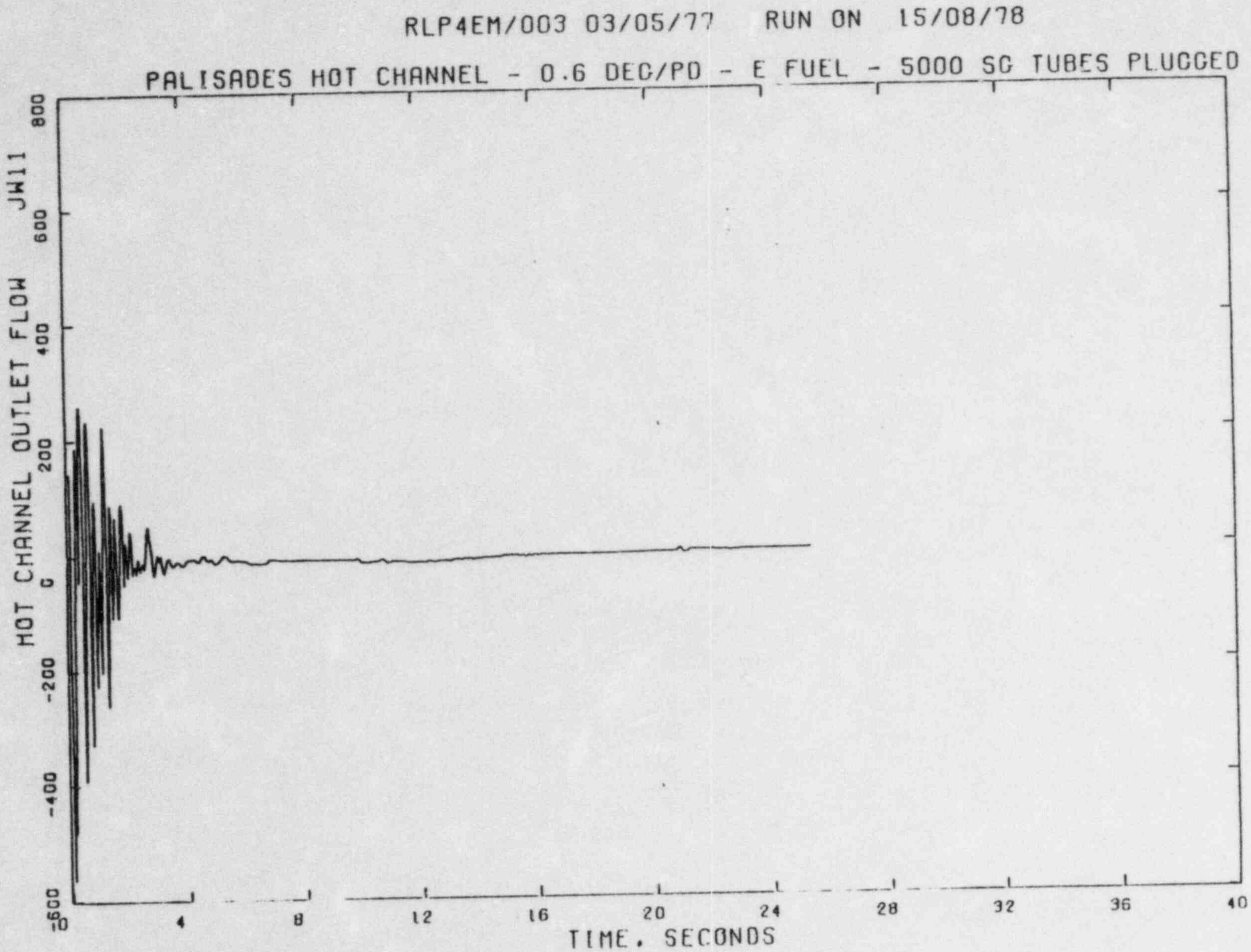

Figure 3.54 Hot Assembly Outlet Flow During Blowdown, 5000 Tubes Plugged Case

PLOTTED ON 10/13/78

RLP4EM/003 03/05/77 RUN ON 15/08/78

PALISADES HOT CHANNEL - 0.6 DEG/PD - E FUEL - 5000 SG TUBES PLUGGED

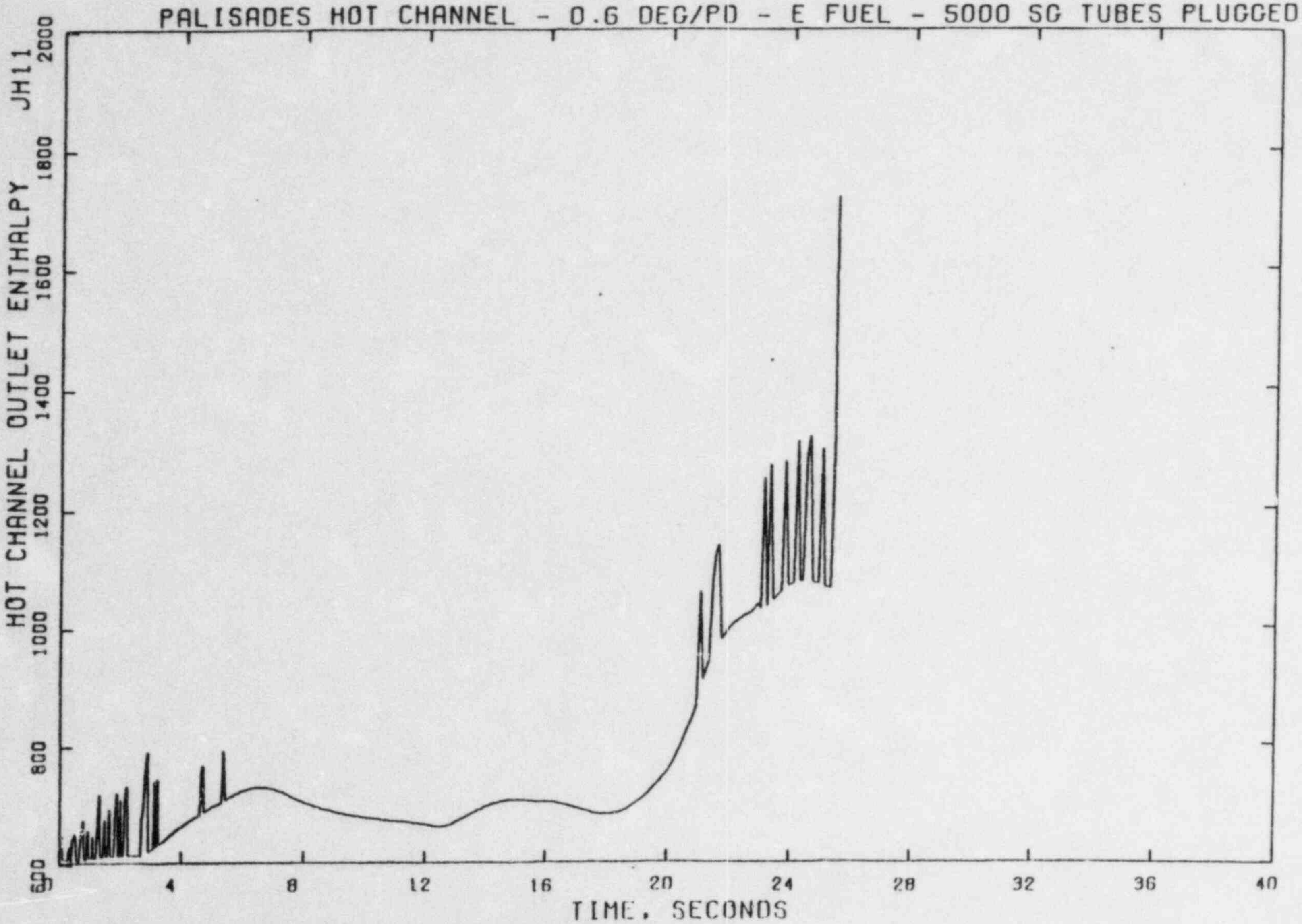

Figure 3.55 Hot Assembly Outlet Enthalpy During Blowdown, 5000 Tubes Plugged Case

PLOTTED ON 10/11/78

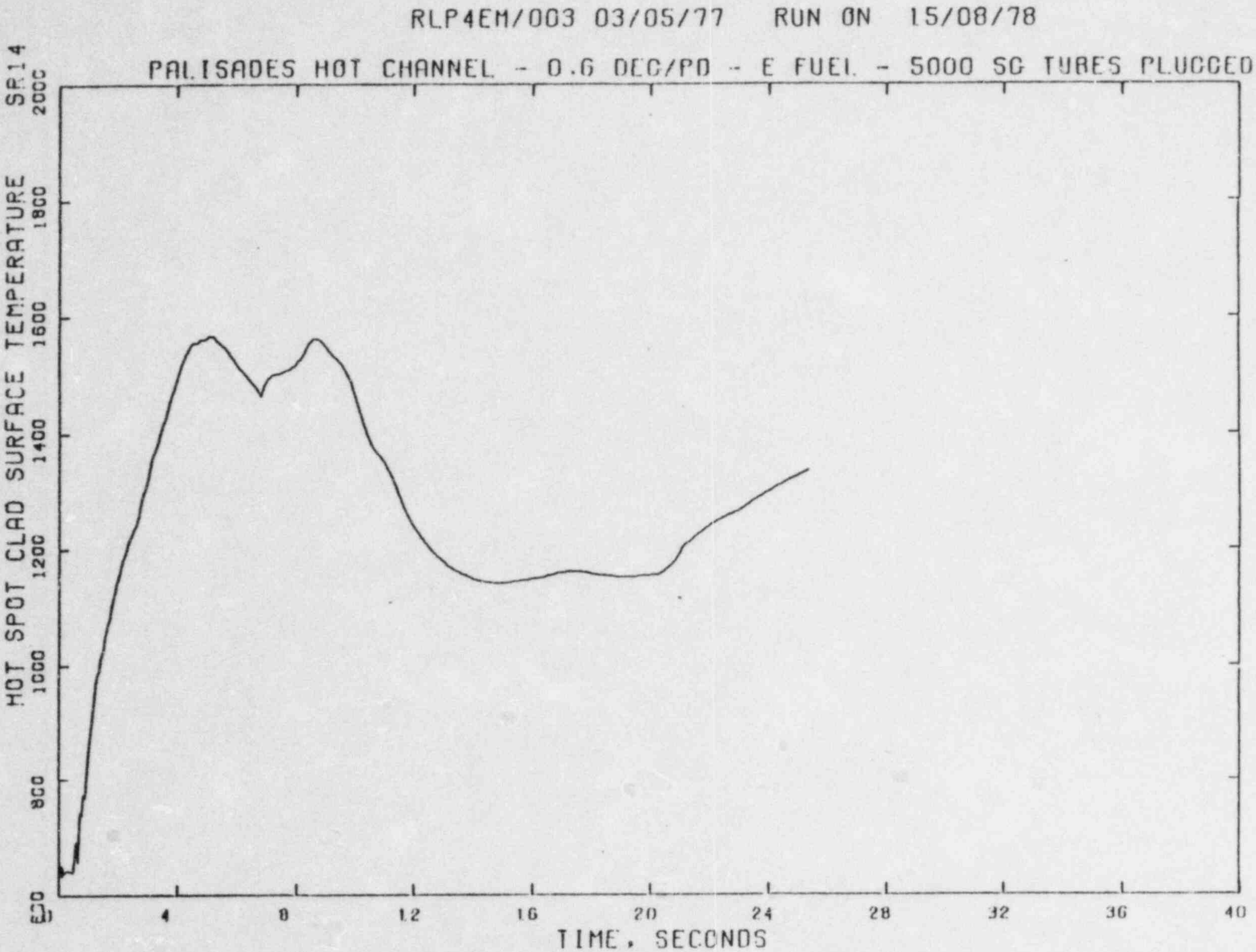

Figure 3.56 Hot Rod Clad Surface Temperature at Peak Power Location
During Blowdown, 5000 Tubes Plugged Case

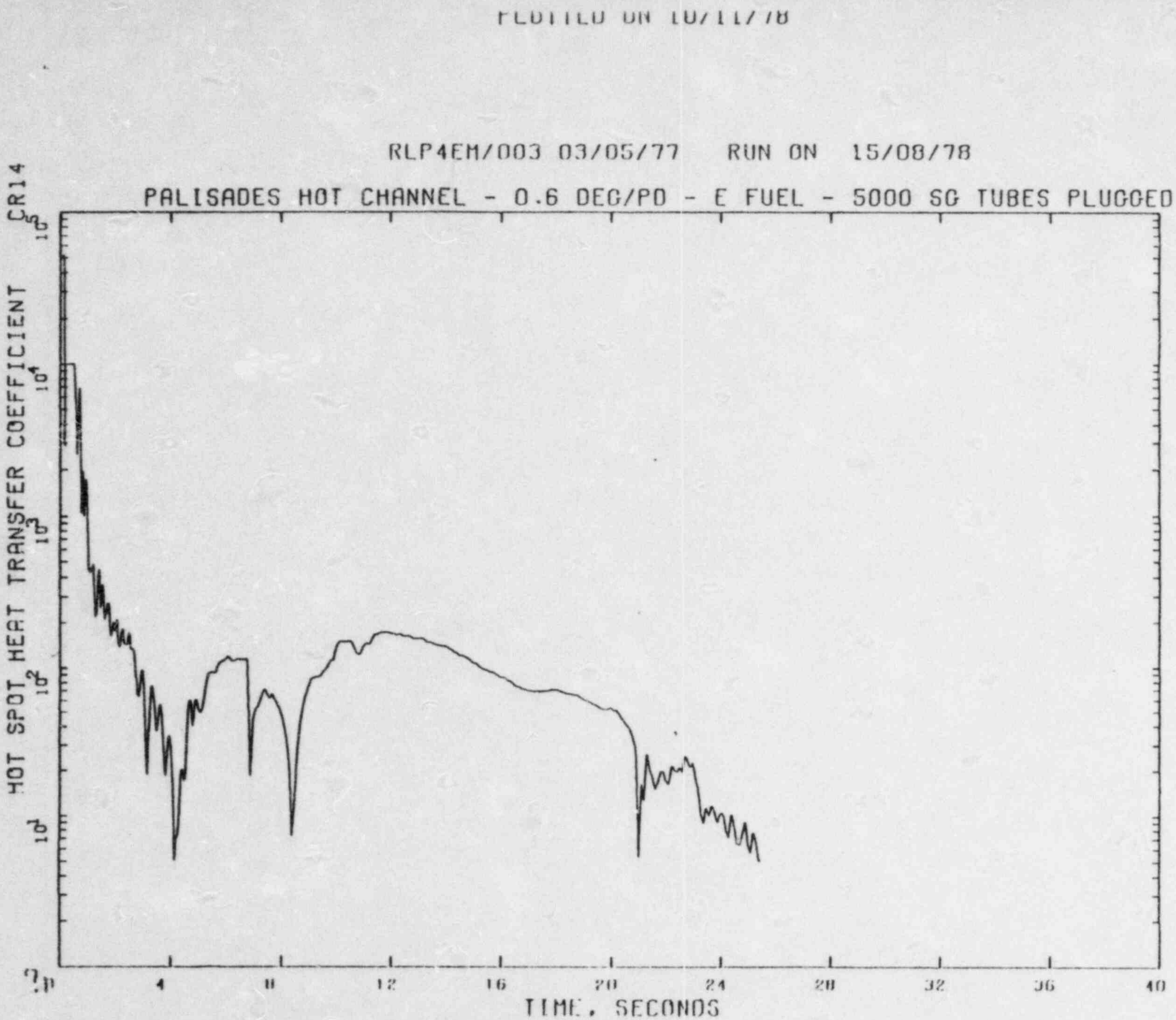

Figure 3.57 Hot Rod Heat Transfer Coefficient at Peak Power Location During Blowdown, 5000 Tubes Plugged Case

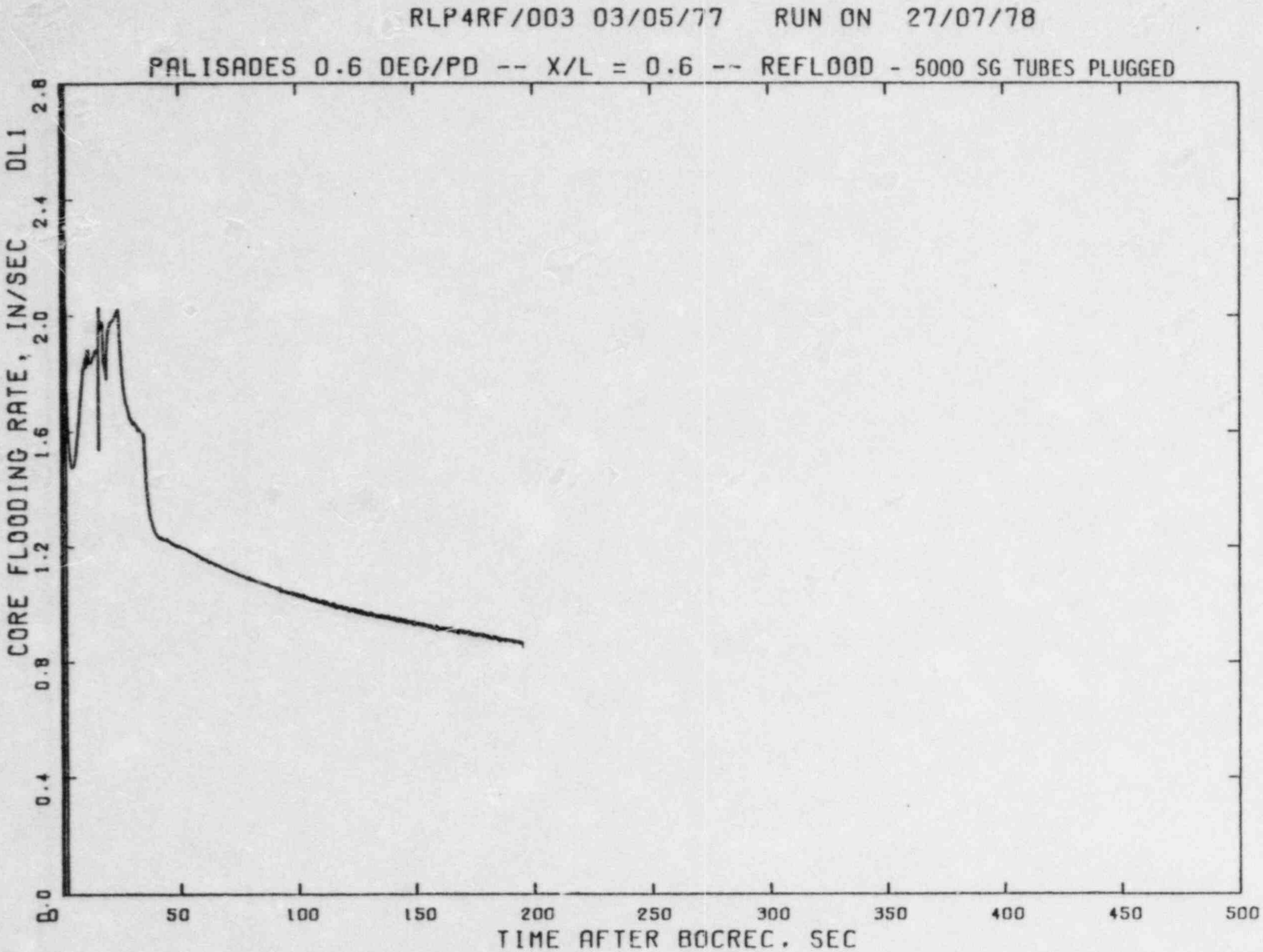

Figure 3.58 Core Reflood Rate During Reflood, 5000 Tubes Plugged Case

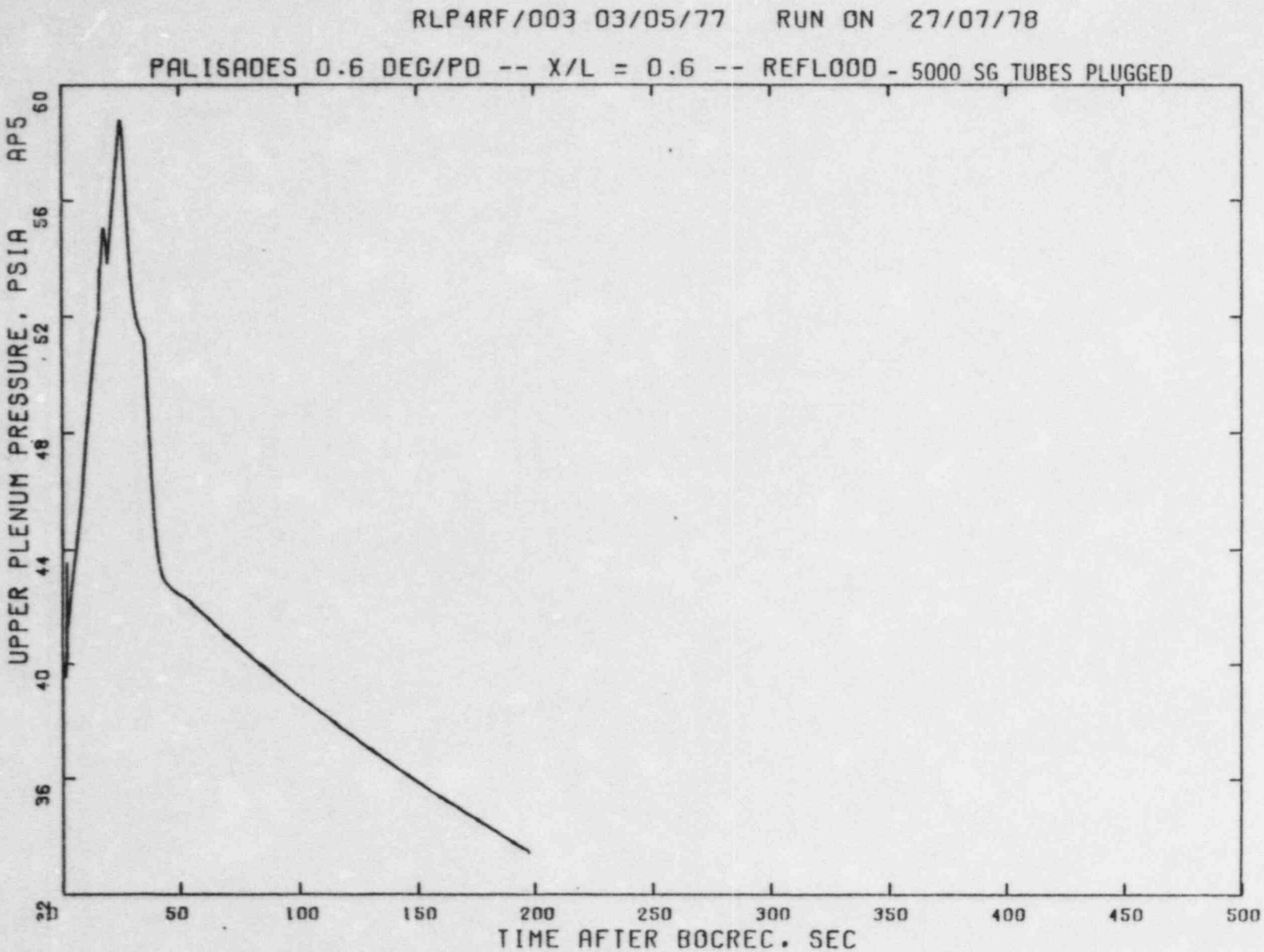

Figure 3.59 Upper Plenum Pressure During Reflood, 5000 Tubes Plugged Case

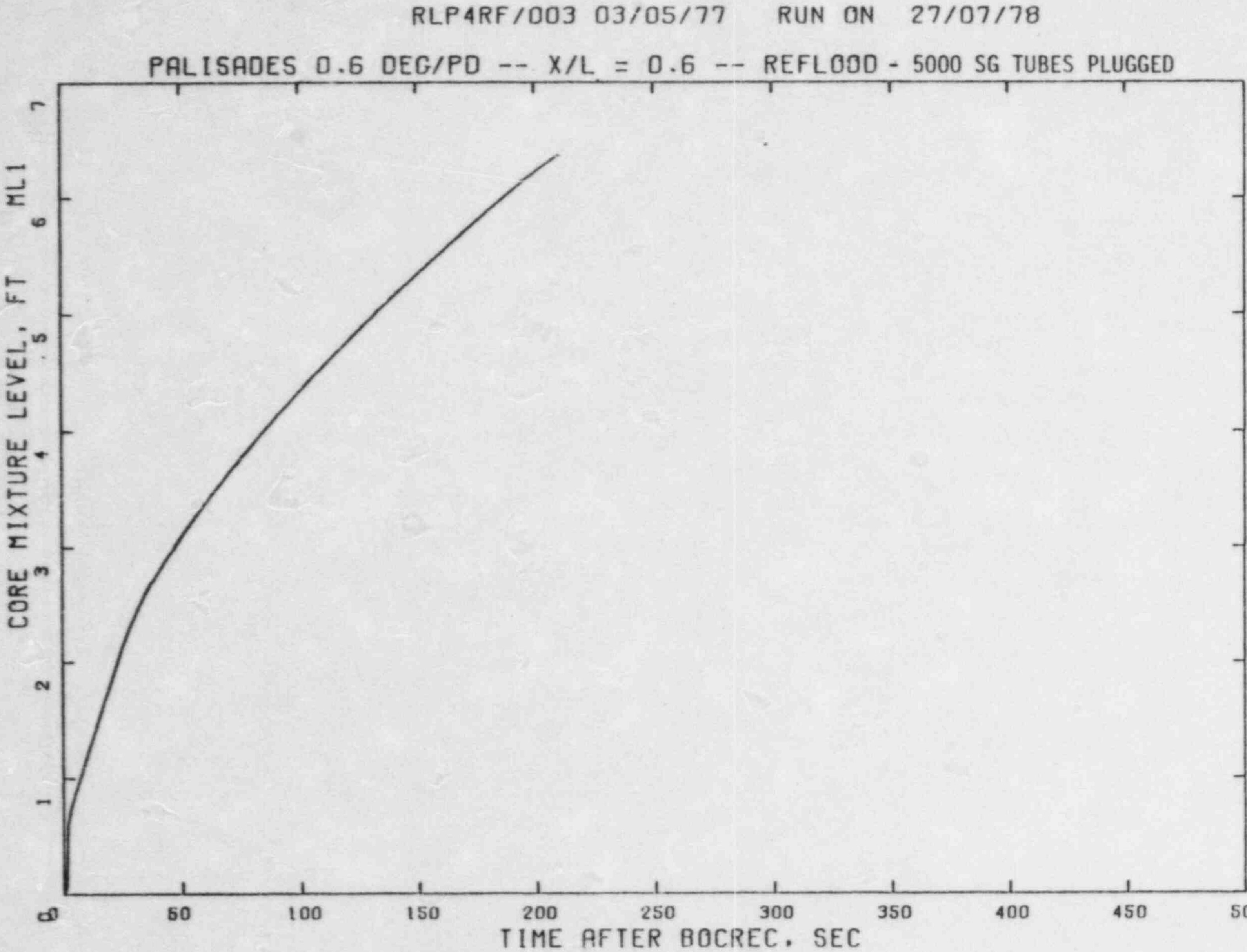

Figure 3.60 Collapsed Liquid Level in Core During Reflood,
5000 Tubes Plugged Case

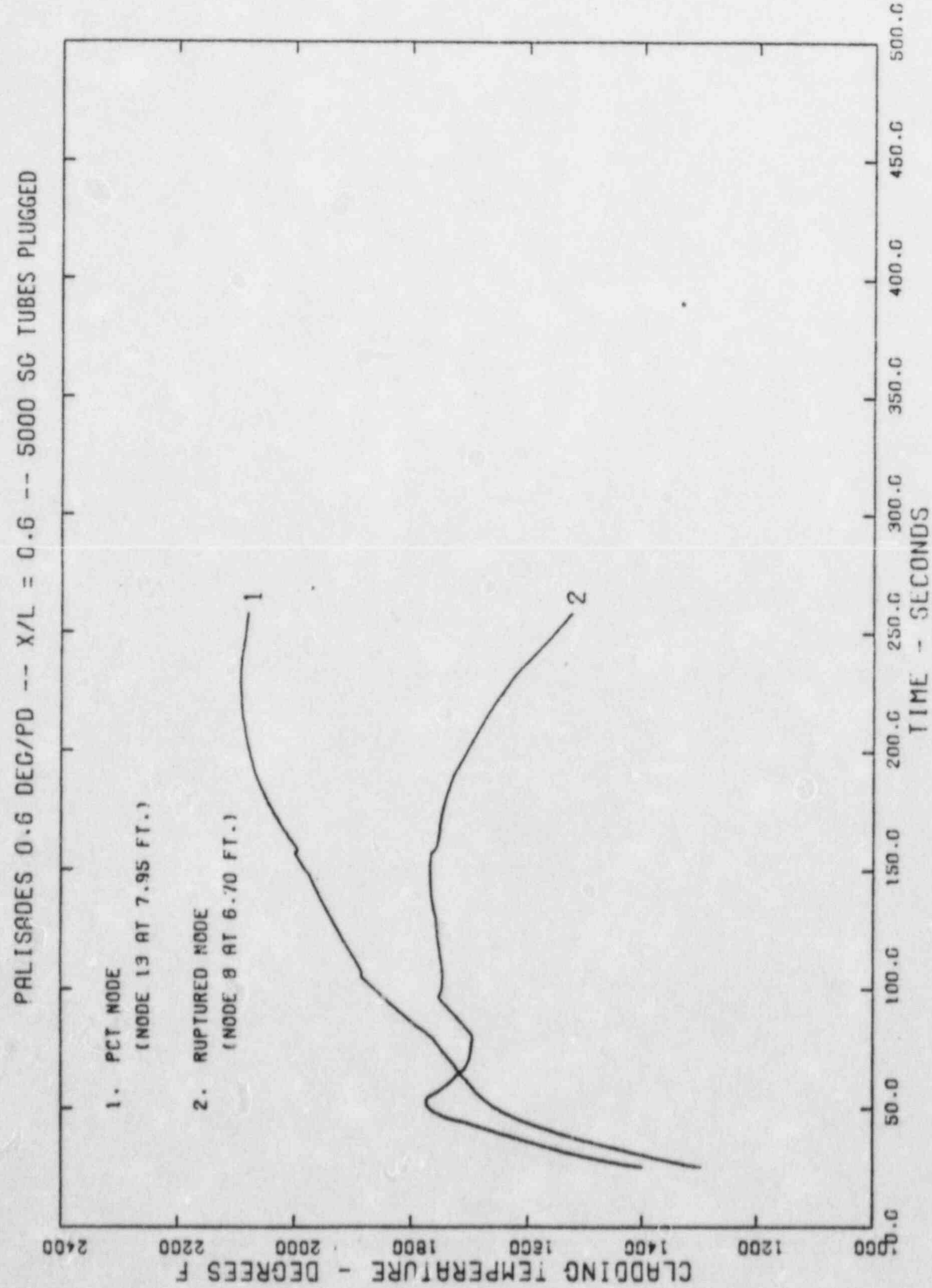

Figure 3.61 Cladding Temperature at PCT and Rupture Location, 5000 Tubes Plugged Case

4.0 CONCLUSION

For the changes in operating conditions reported in this document, the analyses demonstrate that the impact on LOCA limits are small. Used in conjunction with an approved LOCA ECCS analysis at fixed operating conditions, the result of the sensitivity study will provide a basis for making small changes in operating conditions such that 10 CFR 50.46 Acceptance Criteria are satisfied. 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.

5.0 REFERENCES

1. "LOCA Analysis for Palisades at 2530 MWT using the ENC WREM-II PWR ECCS Evaluation Model", XN-NF-77-24, Exxon Nuclear Company, July 1977.

2. "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MWT", XN-NF-78-16, Exxon Nuclear Company, June 1978.

3. "Palisades Cycle 5 Reload Fuel Safety Analysis Report", XN-NF-81-34(P), Exxon Nuclear Company, May 1981.

4. "Palisades Technical Specifications", Amendment No. 68, December 8, 1981, pages 3-103 to 3-110.

5. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model, XN-75-41, Exxon Nuclear Company:

    a. Volume I, July 1975
    b. Volume II, August 1975
    c. Volume III, Revision 2, August 1975
    d. Supplement 1, August 1975
    e. Supplement 2, August 1975
    f. Supplement 3, August 1975
    g. Supplement 4, August 1975
    h. Supplement 5, Revision 1, October 1975
    i. Supplement 6, October 1975
    j. Supplement 7, November 1975.

6. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II", Exxon Nuclear Company.
    a. July 1976
    b. Supplement 1, September 1976
    c. Supplement 2, November 1976.

7. "Revised Nucleate Boiling Lockout for ENC WREM-Based ECCS Evaluation Models", XN-76-44, Exxon Nuclear Company, September 1976.

8. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50, U.S. Code of Federal Regulations [47FR55409, August 8, 1980].

9. "GAPEX, A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients", XN-73-25, Exxon Nuclear Company, August 13, 1973.

10. U.S. Nuclear Regulatory Commission Letter, T.A. Ippolito (NRC) to W.S. Nechodom (ENC), "SER for ENC RELAP4-EM Update", March 1979.

11. U.S. Nuclear Regulatory Commission, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," Branch Technical Position CSB 6-1.

12. G.N. Lauben, "TOODEE2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program," NRC Report NUREG-75-057, May 1975.

13. "C-E/EPRI Two-Phase Pump Performance Program", Quarterly Report Number 1, C-E Power Systems, January 1 to April, 1975.

XN-NF-78-16
Supplement 1
Issue Date: 4/20/84

ANALYSIS OF AXIAL POWER DISTRIBUTION LIMITS
FOR THE PALISADES NUCLEAR REACTOR
AT 2530 MWt: SENSITIVITY STUDIES

DISTRIBUTION

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