EXHIBIT D

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PRAIRIE ISLAND UNITS 1 AND 2
LIMITING BREAK LOCA-ECCS ANALYSIS
WITH INCREASED ENTHALPY RISE FACTOR

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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 Prairie Island Units 1 and 2 nuclear reactors. The analyses assume a reactor operating power of 1683 MWt (includes 2% power uncertainty), and use of Exxon Nuclear Company's (ENC's) TOPROD fuel. The calculations were made for the double-ended cold leg guillotine break, with a discharge coefficient of 0.4 (0.4 DECLG) identified in the previous analyses as the most limiting break.(1,2,3,4)

The analyses were performed using the EXEM/PWR ECCS evaluation model(5), with the RODEX2 computer model for evaluating the rod stored energy and fission gas release(6). The EXEM/PWR ECCS evaluation model includes the NRC fuel swelling and flow blockage model, NUREG-0630.(15) The analyses are applicable up to a five percent (5%) steam generator (SG) tube plugging, and maximum peak pellet exposure limit of 55,000 MWD/MTM. The allowable linear heat generation rate, including the 1.02 factor for power uncertainty, was 14.76 kW/ft, corresponding to a total power peaking factor of 2.28 (F_Q^T), and total enthalpy rise of 1.65 ($F_{\Delta H}^T$). The peaking limits are applicable over the entire exposure.

The analyses were performed assuming an entire core with TOPROD fuel. With respect to a LOCA, the TOPROD fuel design is more limiting than previous ENC XN-1 and XN-2 reload fuel designs in Prairie Island Units 1 and 2. This is due to the increased core flow area which reduces core reflood rates in the LOCA analysis for TOPROD fuel and results in higher PCTs. This analysis is therefore applicable to the XN-1 and XN-2 fuel designs for peak pellet burnups less than 55,000 MWD/MTM.

The calculational basis and results are summarized in Table 1.1. The maximum calculated peak cladding temperature (PCT) is 2199°F, occurring at 191 seconds into the accident at a location 9.37 feet from the bottom of the active core, with a total metal-water reaction less than one percent. The 2199°F PCT includes a 1°F temperature addition due to the use of NRC interim upper plenum injection (UPI) model[7] as modified by Westinghouse[8]. The results of the analyses show that within the limits established, the Prairie Island Nuclear Reactors operating at the stated power level, and with steam generator tube plugging up to 5%, satisfy the criteria specified by 10 CFR 50.46[9].

Table 1.1 Prairie Island Units 1 and 2
TOPROD LOCA-ECCS Analysis Results

Analysis Results	0 - 15000 MWD/MTM Peak Pellet Exposure	15000 - 55000 MWD/MTM Peak Pellet Exposure
Peak Clad Temperature (PCT), °F*	2199	2080
ΔPCT for UPI, °F	1	8
Time of PCT, sec.	191	249
Peak Clad Temperature Location, ft.	9.37	9.37
Local Zr/H_2O Reaction (max.), %**	6.38	4.37
Local Zr/H_2O Location, ft. from bottom	9.37	9.37
Total H_2 Generation, % of total Zr Reacted	<1.0	<1.0
Hot Rod Burst Time, sec.	30.20	38.80
Hot Rod Burst Location, ft.	6.0	6.25

Calculational Basis	
License Core Power, MWt	1650
Power Used for Analysis, MWt***	1683
Peak Linear Power for Analysis, kW/ft***	14.76
Total Peaking Factor, F_Q^T	2.28
Enthalpy Rise, Nuclear, $F_{\Delta H}^T$	1.65
Steam Generator Tube Plugging (%)	5.00

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* Including ΔPCT for UPI.
** Computer value at 380 seconds.
*** Including 1.02 factor for power uncertainties.

2.0 LIMITING BREAK LOCA ANALYSIS

This report provides LOCA-ECCS analyses performed for Prairie Island Units 1 and 2 with a steam generator tube plugging up to 5%. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are described in the ENC WREM models[10], and the Emergency Core Cooling System Evaluation Model Updates: WREM-II[18], WREM-IIA[14] and EXEM/PWR[5].

A LOCA break spectrum analysis was performed and reported in XN-NF-78-46[1]. The limiting LOCA break was determined to be a large double-ended guillotine break of the cold leg, with a discharge coefficient of 0.4 (0.4 DECLG). The analyses performed and reported herein for the 0.4 DECLG break consider:

(1) A revised stored energy model RODEX2[6] in place of the previously applied GAPEX[11] model.

(2) The NRC upper plenum injection (UPI) interim model, developed by the NRC Staff[7] and modified by Westinghouse[8].

(3) Updates to the latest Prairie Island Units 1 and 2 application to reflect all model revisions and documented in XN-NF-82-20(P), Revision 1[5].

(4) The FLECHT/ENC2 WREM-II[18] heat transfer coefficient multipliers.

2.1 LOCA ANALYSIS MODEL

The Exxon Nuclear Company EXEM/PWR ECCS evaluation model[5] was used to perform the analyses. This model consists of the following computer codes: RODEX2[6] 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[17] for computation of

containment backpressure; REFLEX[5,15] for computation of system reflood; and TOODEE2[5,15,16] for the calculation of final fuel rod heatup.

The Prairie Island nuclear reactor is a two-loop Westinghouse pressurized water reactor with an upper plenum injection and dry containment. The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow-paths or "junctions" as described in XN-NF-77-25(A)[17]. The system nodalization is depicted in Figure 2.1. The pump performance characteristic curves are supplied by the NSSS vendor. Five percent of the steam generator tubes are assumed to be plugged in each generator. 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 chopped cosine axial power profile used for the analyses is shown in Figure 2.2, with a maximum axial peaking factor of 1.342, corresponding to a total peaking factor of 2.28, and $F_{\Delta H}^{T}$ of 1.65. The $F_Q{}^T$ determined with this axial profile in combination with the current K(Z) function developed originally by the NSSS vendor is used to define the envelope for $F_Q{}^T$, where the K(Z) curve is limited by large break LOCAs. Where small break LOCAs are limited, the K(Z) curve was modified such that Linear Heat Generation Rates (LHGRs) were determined by the NSSS vendor analyses. The K(Z) curve is represented in Figure 2.3. The analysis of the loss-of-coolant accident is performed at 102 percent of rated power. The fuel design parameters are shown in Table 2.2.

Two cases of LOCA-ECCS calculations were performed with input which bounds the fuel history up to 55,000 MWD/MTM peak pellet exposure. The most limiting fuel conditions from beginning-of-life to 15,000 MWD/MTM (first case), and from 15,000 MWD/MTM to end-of-life (second case) were determined and used in each calculation. Decay power, internal rod pressure and the fission gas releases are highest at EOL, while the stored energy is calculated to be highest at lower exposure (~2 MWD/KgU). The combination of highest stored energy, rod pressure, and decay power was used to bound the LOCA-ECCS analysis over the exposure ranges shown.

The small rod diameter for ENC TOPROD fuel, as compared to other fuel designs in the Prairie Island reactors, results in a larger core flow area. The larger core flow area decreases the core flooding rates, which results in higher PCTs. Furthermore, the 55,000 MWD/MTM exposure limit considered in this analysis encompasses the exposure limits expected for the previous ENC XN-1 and XN-2 fuel designs operating in Prairie Island units. Therefore, the LOCA-ECCS analyses reported in this document bound the previous Prairie Island ENC fuel designs.

2.2 RECOMMENDATIONS FOR OPERATION WITH VARIABLE $F_{\Delta H}^{T}$

The analysis reported in Reference 4 had analyzed the Prairie Island LOCA-ECCS for a total peaking factor of 2.32 (F_Q^T) with $F_{\Delta H}^{T}$ of 1.55. The current analysis is performed with a total peaking factor of 2.28 and $F_{\Delta H}^{T}$ of 1.65. Since the change in F_Q^T as a function of $F_{\Delta H}^{T}$ between the previous and the current analysis is small, it is recommended that a linear interpolation be used between the two total enthalpy rises ($F_{\Delta H}^{T}$) of 1.55 and 1.65 to evaluate the total peaking limit (F_Q^T); see Figure 2.4. Since the LOCA analysis is

performed to bound expected operation within the F_Q^T and $F_{\Delta H}^T$ limits, the analysis supports operation of the Prairie Island reactors if all rods or bundles are within these limits, irrespective of the number of rods or bundles at these limits.

To assure compliance to Appendix K criteria, K(Z) curves have been developed which define the total peaking limits as a function of the location of the axial peaking. In this analysis, a K(Z) curve has been developed (Figure 2.3) which is applicable when the total enthalpy rise ($F_{\Delta H}^T$) is 1.65 and F_Q^T is 2.28. The current Prairie Island Units 1 and 2 Technical Specification K(Z) was developed for $F_{\Delta H}^T$ of 1.55 with F_Q^T of 2.32 (Figure 2.5). The two K(Z) curves differ slightly. To simplify the monitoring of F_Q^T as a function of axial locations for $F_{\Delta H}^T$ less than or equal to 1.65, it is recommended that a new K(Z) curve (Figure 2.6) be incorporated in the Prairie Island Technical Specification. The new curve has been developed to conservatively bound the values shown in Figures 2.3 and 2.5, and is identical to Figure 2.5.

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 0.4. Comparison of these results with the previous LOCA-ECCS analysis for a TOPROD fuel shows very slight change in the event times. Figures 2.7 through 2.15 present plotted results for system blowdown analysis. Unless otherwise noted on the figures, time zero corresponds to the time of break initiations. Figures 2.16 through 2.27 present results for the hot channel blowdown calculations. Figures 28 through 33 present the accumulators, HPSI and LPSI

flows during the refill and reflood periods of LOCA transient. Figure 2.34 presents calculated containment backpressure time history. Figures 2.34 and 2.35 show the normalized power calculation results. The reflood calculation results are shown in Figures 2.37 through 2.44.

The maximum peak cladding temperature (PCT) calculated for the 0.4 DECLG break is 2199°F (Figure 2.45) and is calculated to occur at fuel rod exposures less than 15,000 MWD/MTM. This value includes a 1°F temperature addition associated with the use of the NRC interim upper plenum injection (UPI) model as modified by Westinghouse. The maximum linear heat generation rate is 14.76 kW/ft (F_Q^T=2.28) for ENC TOPROD fuel. The maximum local metal-water reaction in this case is 6.38% at 380 seconds, and the total core metal-water reaction is less than 1%. The PCT location is at an elevation of 9.37 feet from the bottom of active core. At fuel rod exposures between 15,000 and 55,000 MWD/MTM (EOL), the maximum PCT is calculated to be 2080°F (Figure 2.46) including an 8°F for UPI effect, occurring at 9.37 feet elevation relative to the bottom of the active core. The local metal-water reaction is 4.37% at 380 seconds, with a total metal-water reaction of less than 1%.

Table 2.1 Prairie Island Units 1 and 2 System Data

Primary Heat Output, MWt	1650*
Primary Coolant Flow, lbm/hr	6.82×10^7
Primary Coolant Volume, ft^3	10,247.** †
Operating Pressure, psia	2,250.
Inlet Coolant Temperature, ^oF	530.
Reactor Vessel Volume, ft^3	2364.
Pressurizer Volume, Total, ft^3	1000.
Pressurizer Volume, Liquid, ft^3	600.
Accumulator Volume, Total, ft^3 (each of two)	2000.
Accumulator Volume, Liquid, ft^3	1250.
Accumulator Trip Point Pressure, psia	714.7
Steam Generator Heat Transfer Area, ft^2	48,925.†
Steam Generator Secondary Flow, lbm/hr	3.54×10^6
Steam Generator Secondary Pressure, psia	724.7
Reactor Coolant Pump Head, ft	277.
Reactor Coolant Pump Speed, rpm	1190.
Moment of Inertia, lbm-ft^2/rad	78,000.
Cold Leg Pipe, I.D., in	27.5
Hot Leg Pipe, I.D., in	29.0
Pump Suction Pipe, I.D., in	31.0

* Primary Heat Output used in RELAP4-EM Model = 1.02 x 1650 = 1683 MWt.
** Includes total accumulator and pressurizer volumes.
† Includes 5% SG tube plugging.

Table 2.2 Fuel Design Parameters

Parameter	ENC Standard	TOPROD
Cladding, O.D., in.	0.426	0.417
Cladding, I.D., in.	0.364	0.358
Cladding Thickness, in.	0.031	0.0295
Pellet O.D., in.	0.3565	0.3505
Diametral Gap, in.	0.0075	0.0075
Pellet Density, % TD	94.0	94.0
Active Fuel Length, in.	144.0	144.0
Enriched UO_2, in.	144.0	132.0
Upper Blanket, in.	-	6.0
Lower Blanket, in.	-	6.0
Cell Water/Fuel Ratio	1.67	1.79
Rod Pitch	0.556	0.556

Table 2.3 Prairie Island Units 1 and 2 TOPROD LOCA-ECCS Analysis Results, Event Times

Event	Time (sec.)
Start	0.00
Break Initiation	0.05
Safety Injection Signal	0.65
Accumulator Injection, Broken Loop	4.80
Accumulator Injection, Intact Loop	8.70
End-of-Bypass	21.05
Safety Injection Flow	25.60
Start of Reflood	36.80
Accumulator Empties, Broken Loop	39.95
Accumulator Empties, Intact Loop	43.95
Peak Clad Temperature Reached, 0-15,000 MWD/MTM Case	191.0
Peak Clad Temperature Reached, 15,000 MWD/MTM to EOL Case	249.0

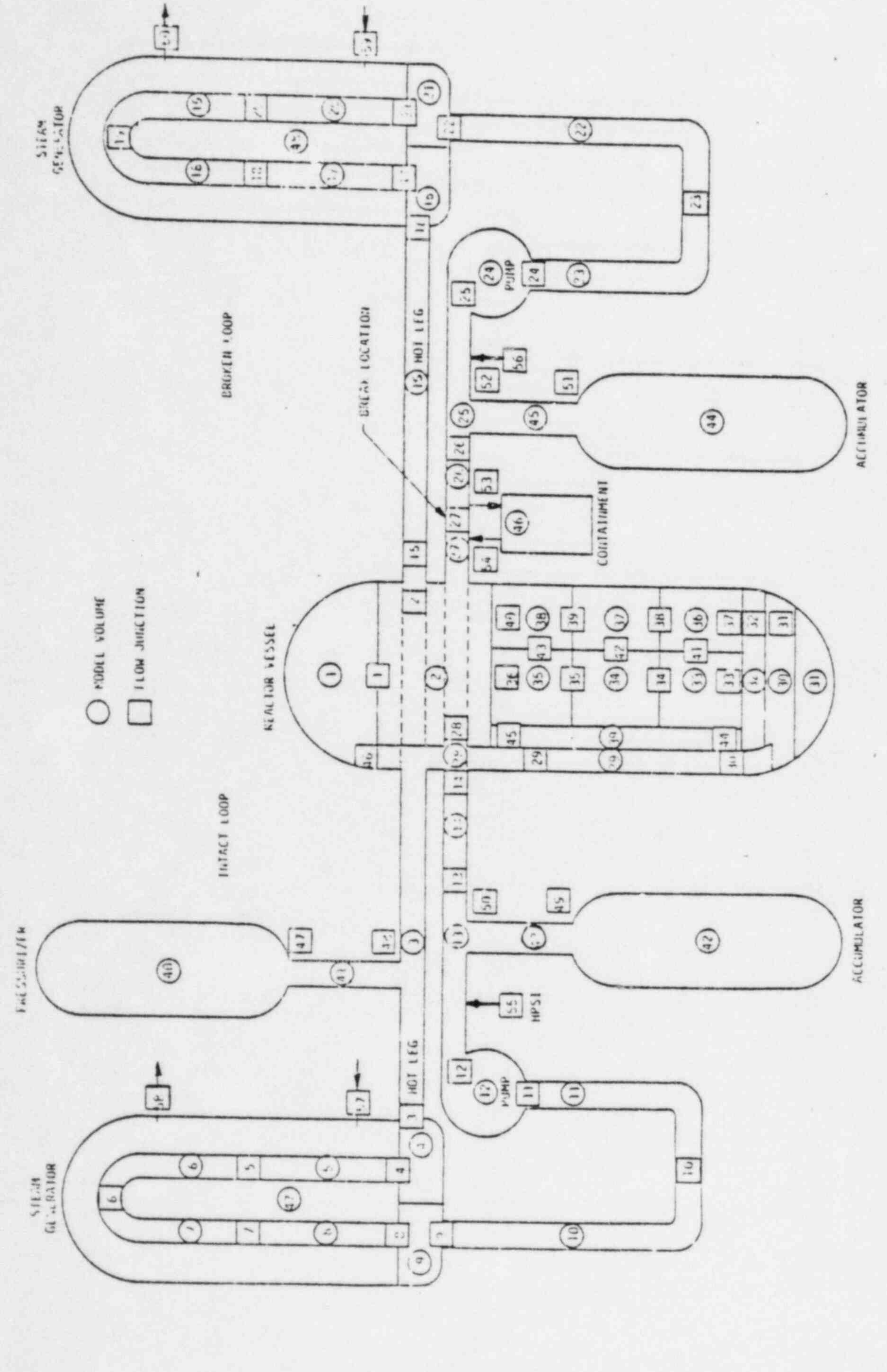

FIGURE 2.1 RELAP4/EM BLOWDOWN SYSTEM NODALIZATION FOR PRAIRIE ISLAND UNIT 1 AND 2

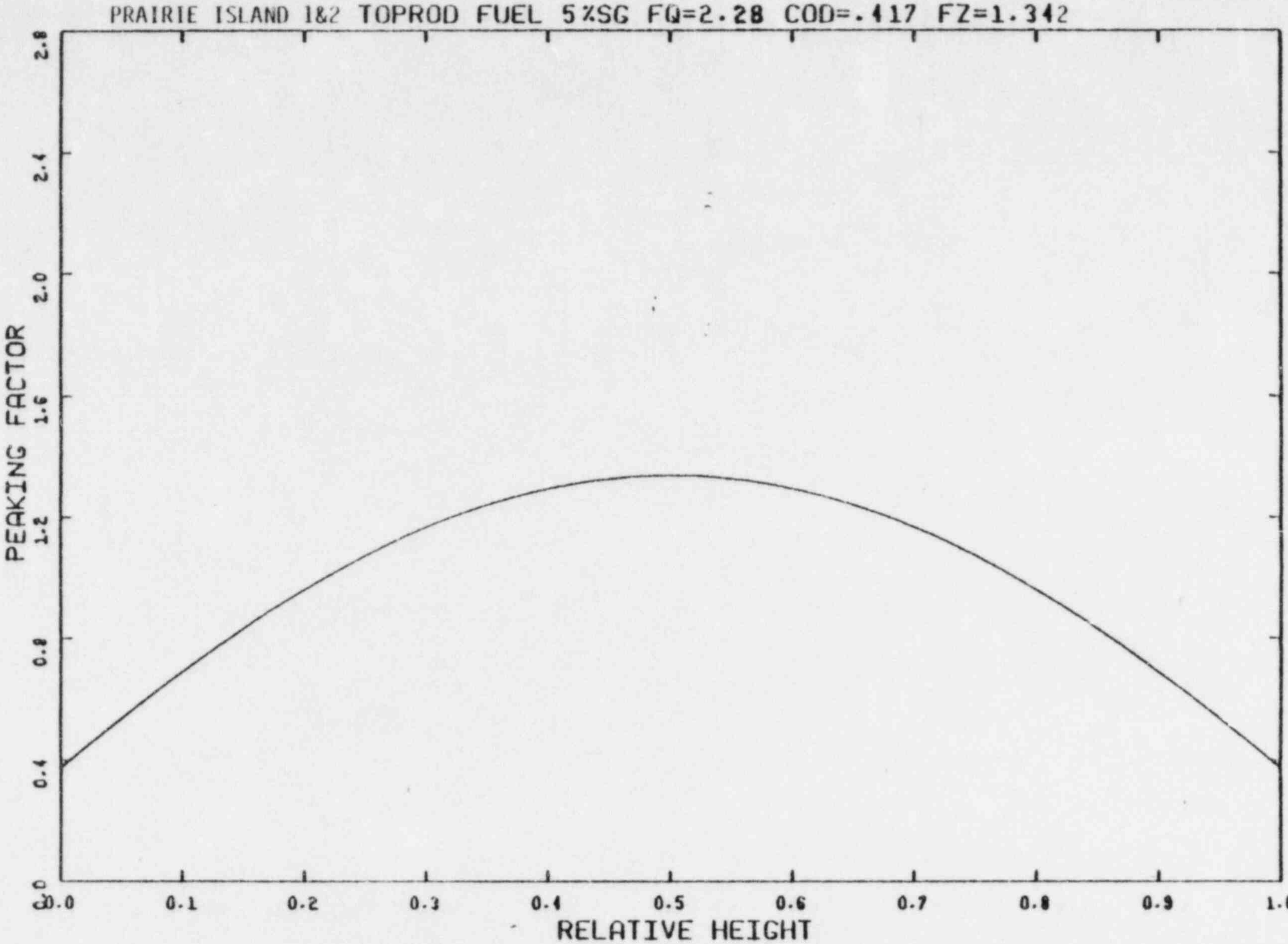

Figure 2.2 Axial Peaking Factor versus Rod Length, 0.4 DECLG Break

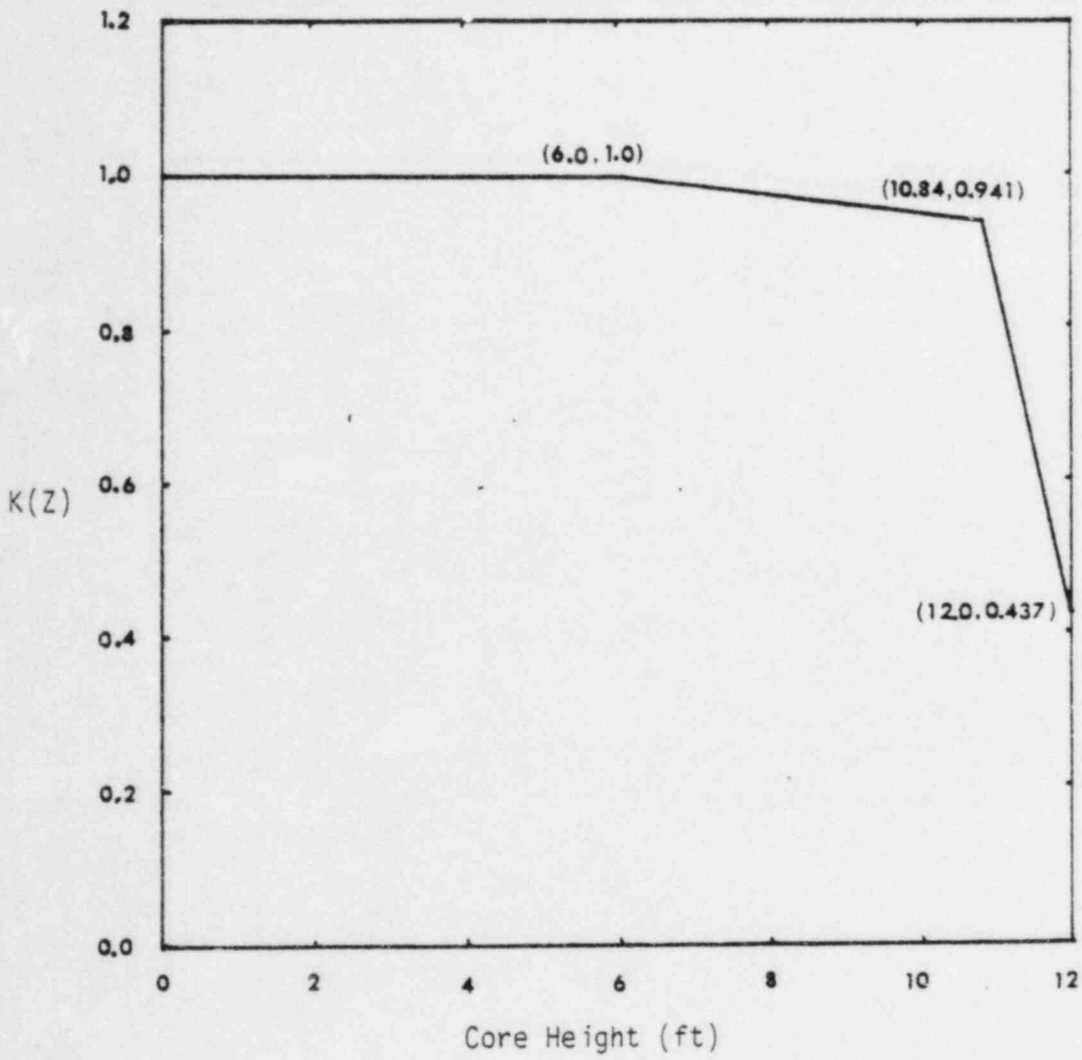

Figure 2.3 Hot Channel Factor Normalized Operating Envelope for F_Q=2.28 with $F_{\Delta H}$ = 1.65

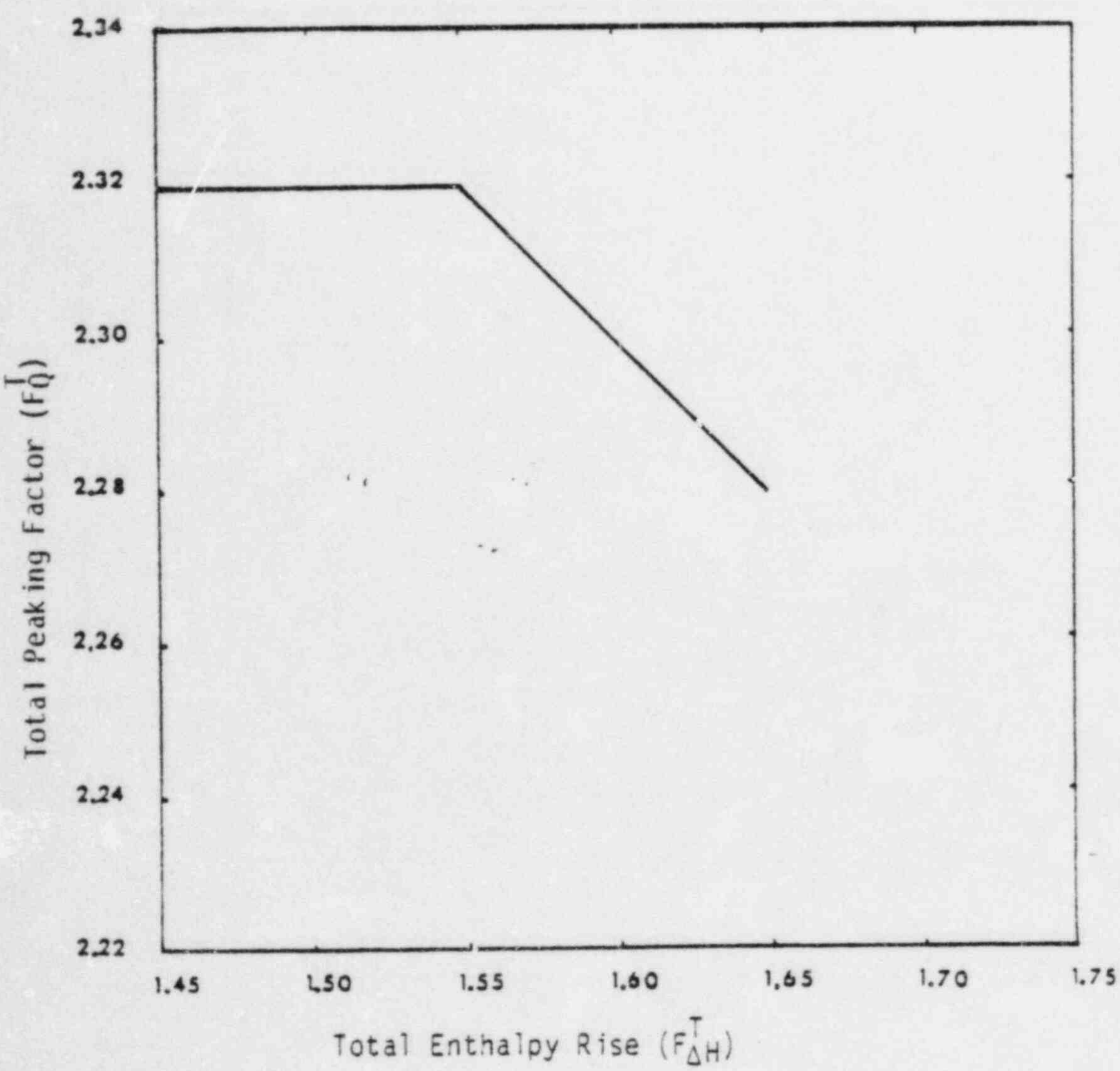

Figure 2.4 Enthalpy Rise Dependent Total Peaking Factor

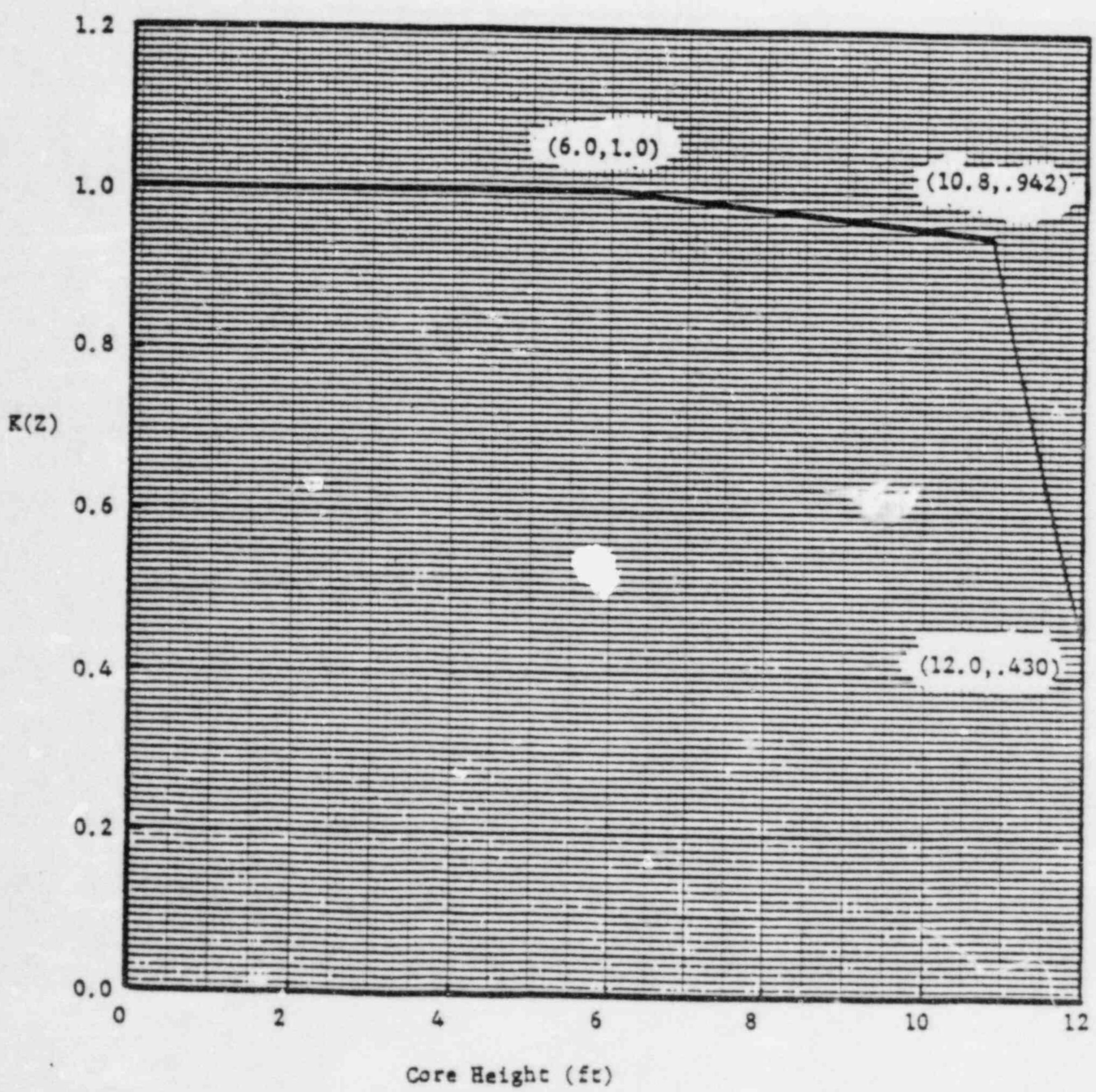

Figure 2.5 HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE FOR $F_Q = 2.32$ with $F_{\Delta H} \leq 1.55$

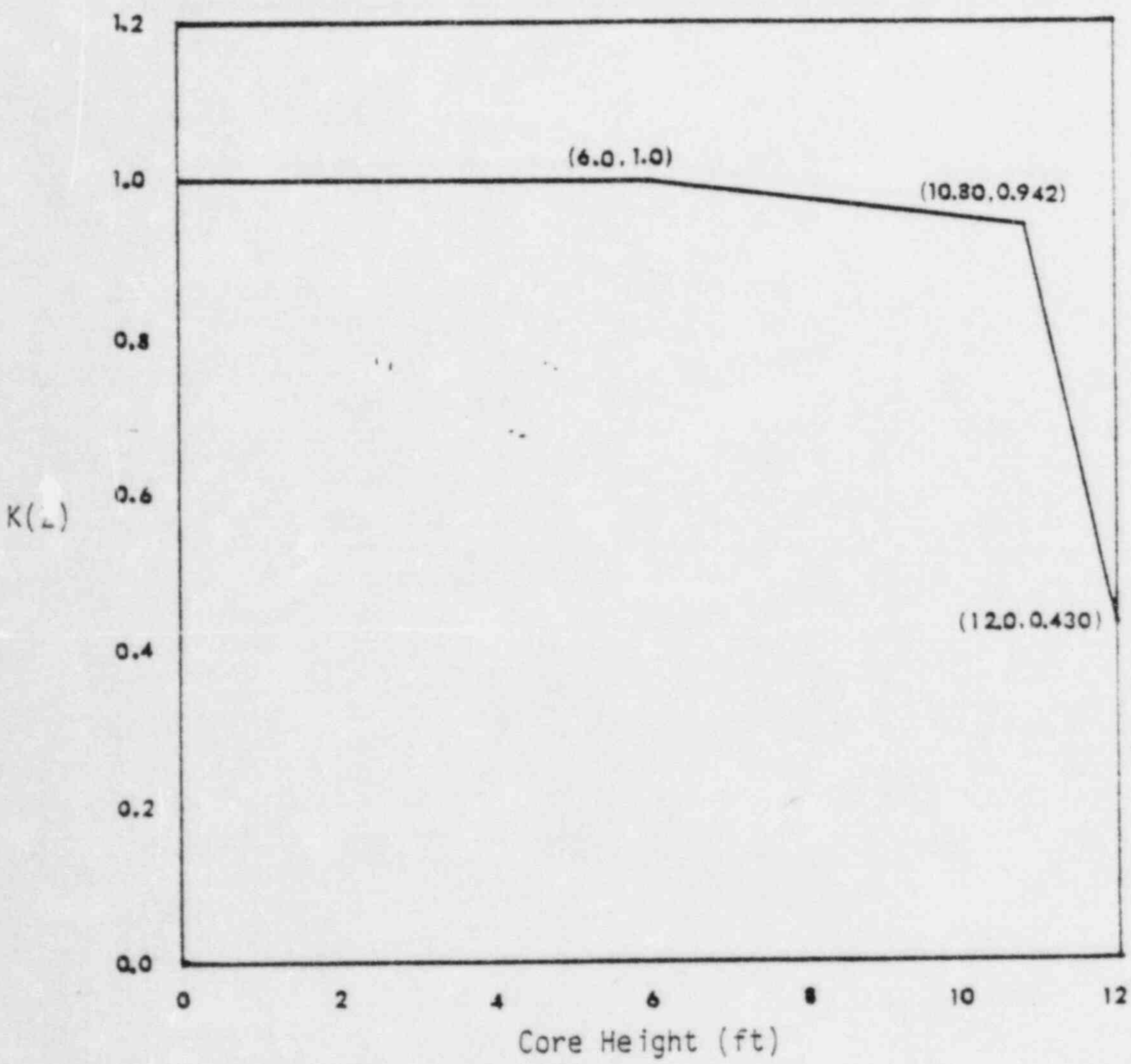

Figure 2.6 Hot Channel Factor Normalized Operating Envelope for F_Q with $F_{\Delta H} \leq 1.65$

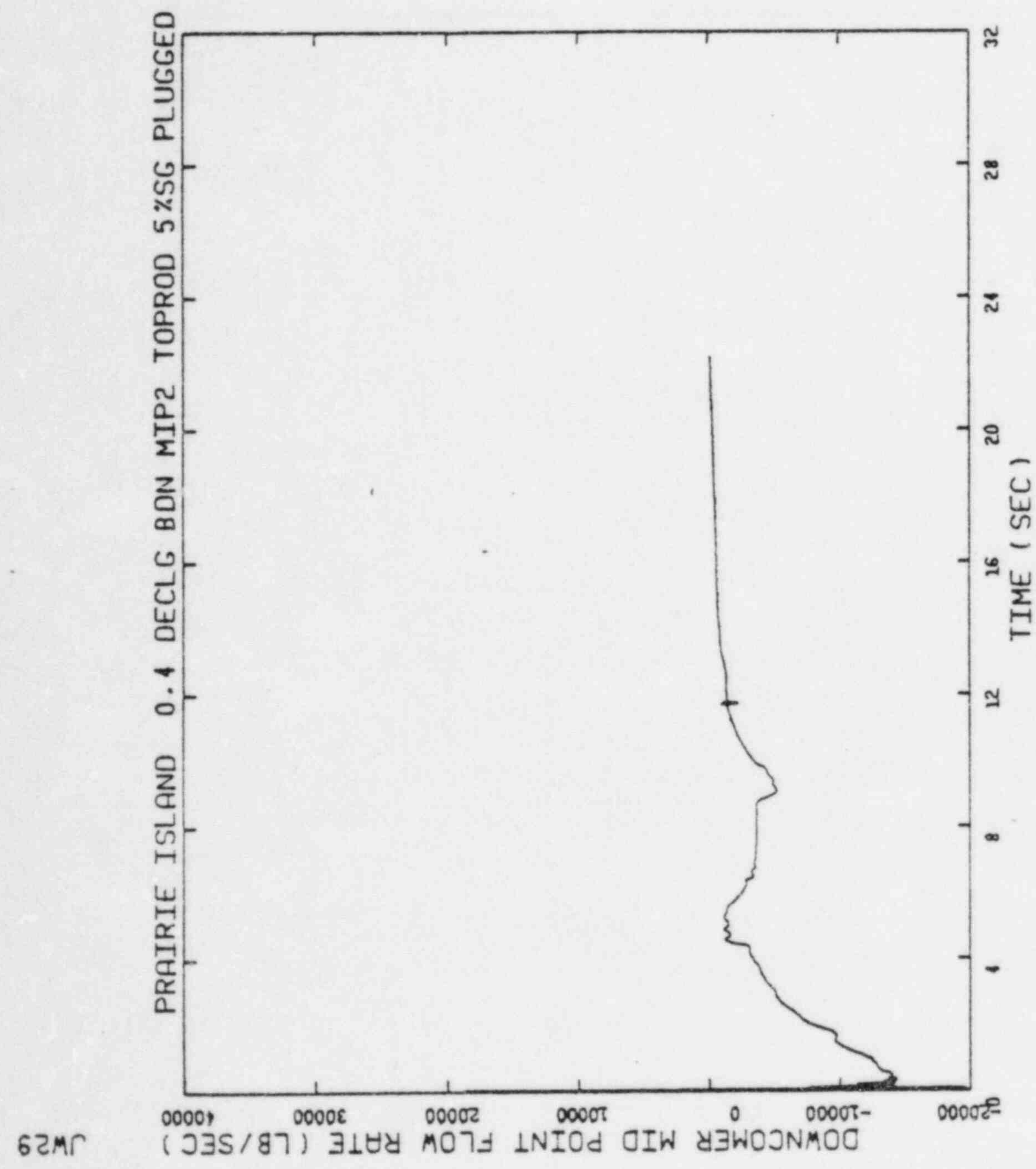

Figure 2.7 Downcomer Flow Rate During Blowdown Period, 0.4 DECLG Break

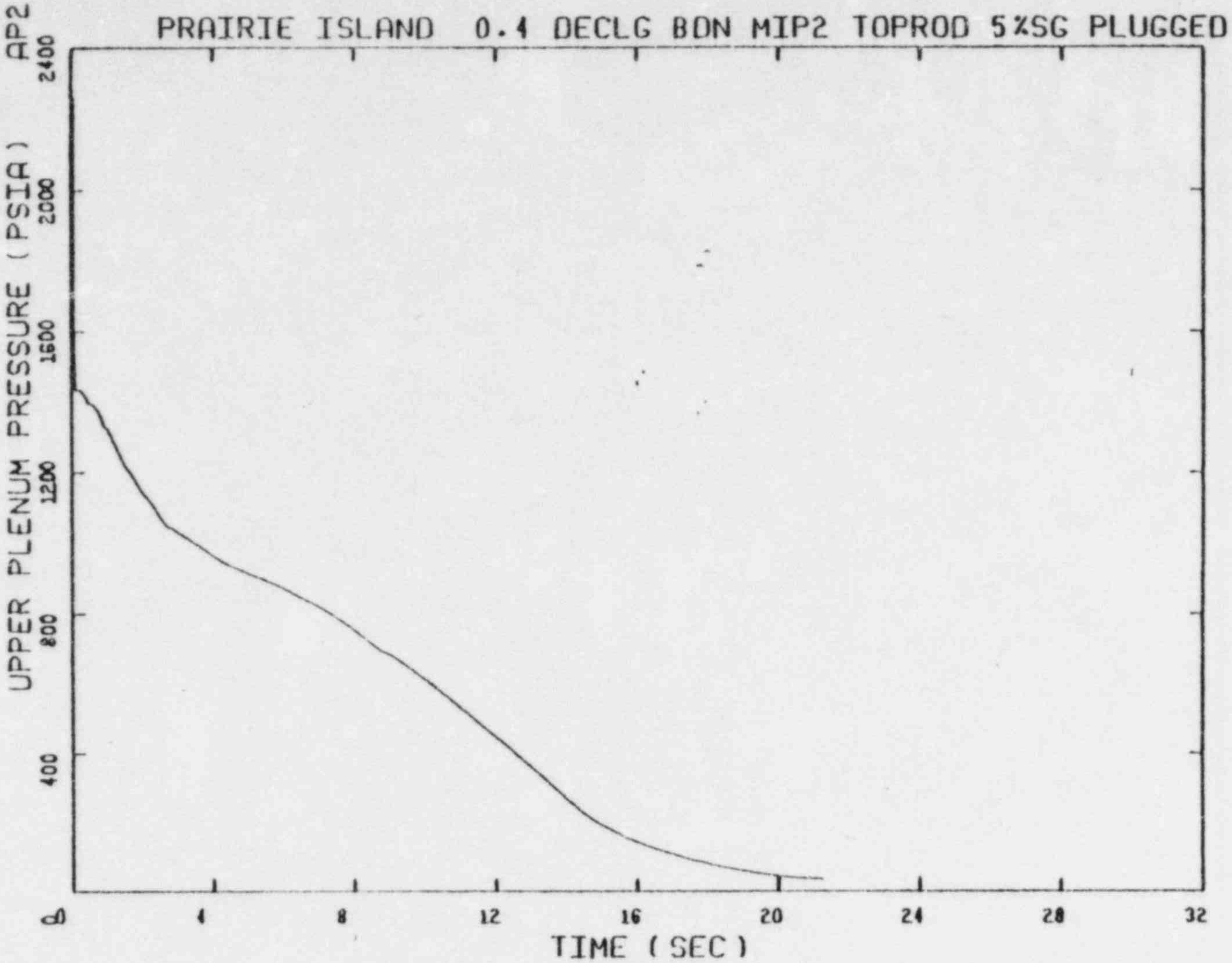

Figure 2.8 Upper Plenum Pressure During Blowdown Period, 0.4 DECLG Break

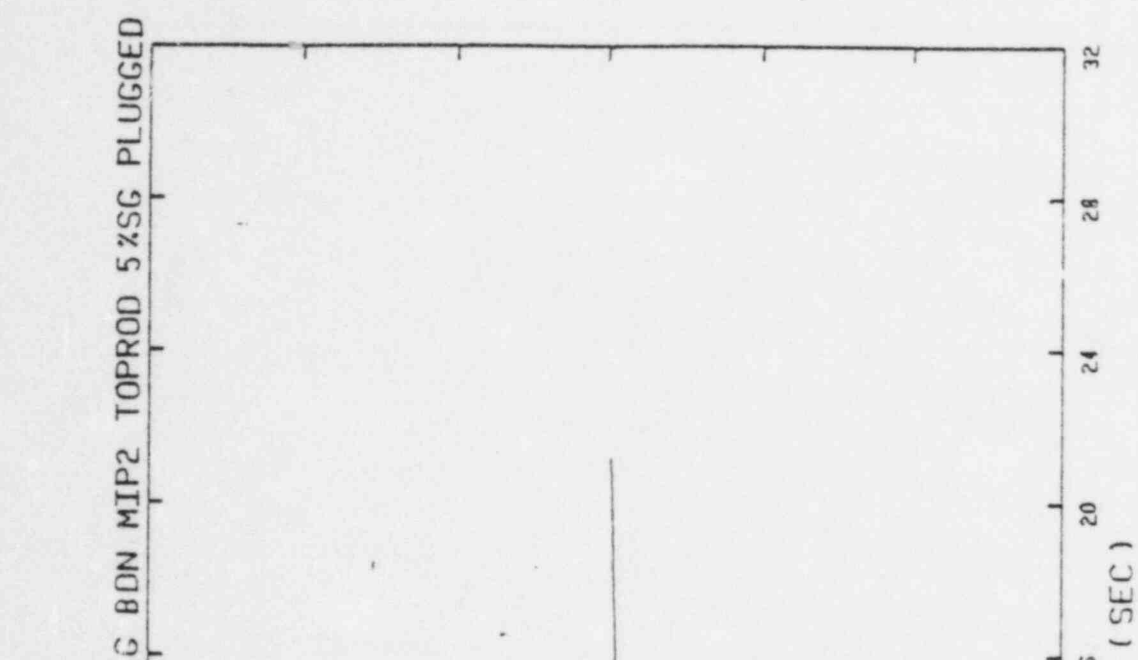

Figure 2.9 Average Core Inlet Flow During Blowdown Period, 0.4 DECLG Break

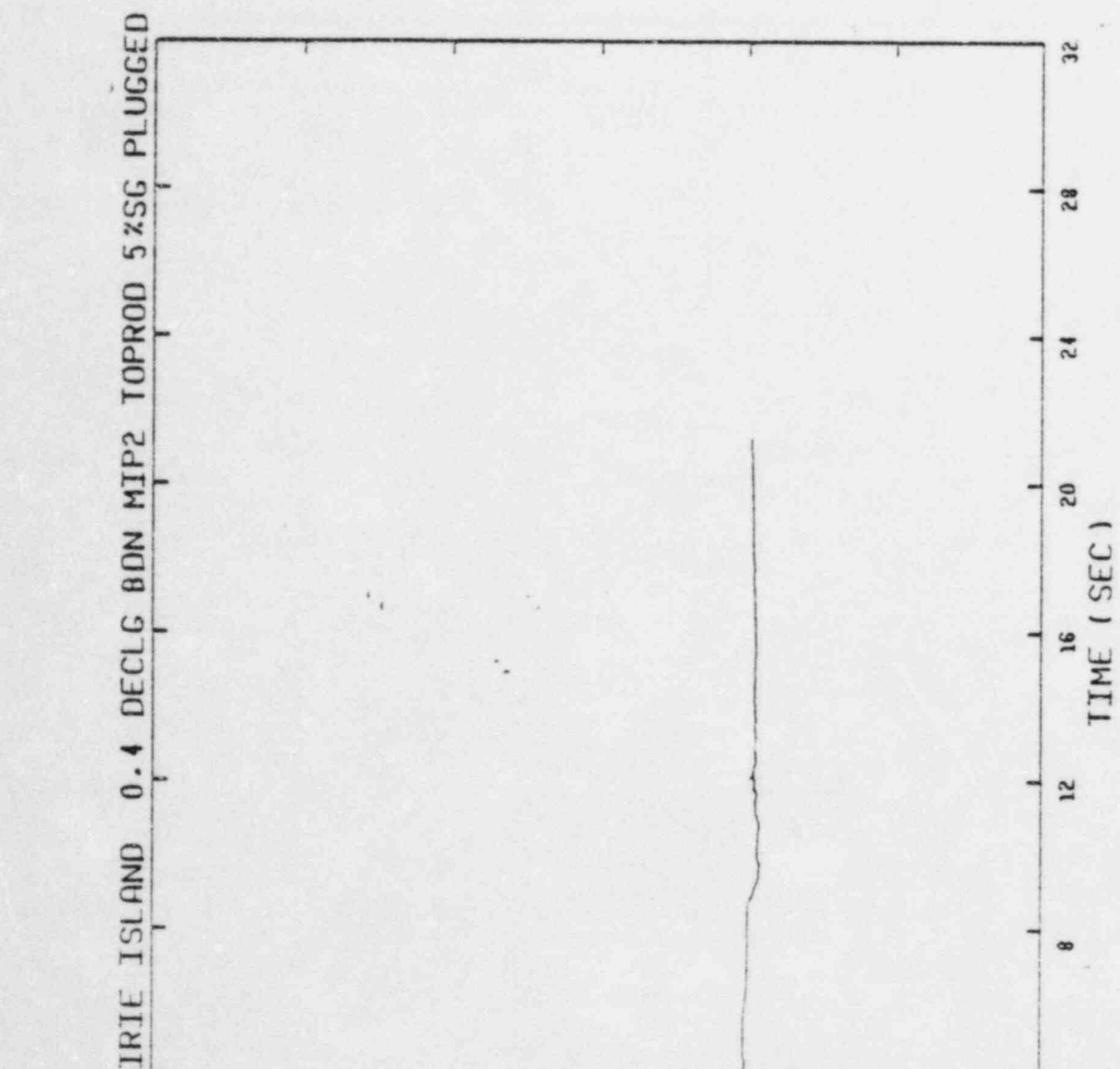

Figure 2.10 Average Core Outlet Flow During Blowdown Period, 0.4 DECLG Break

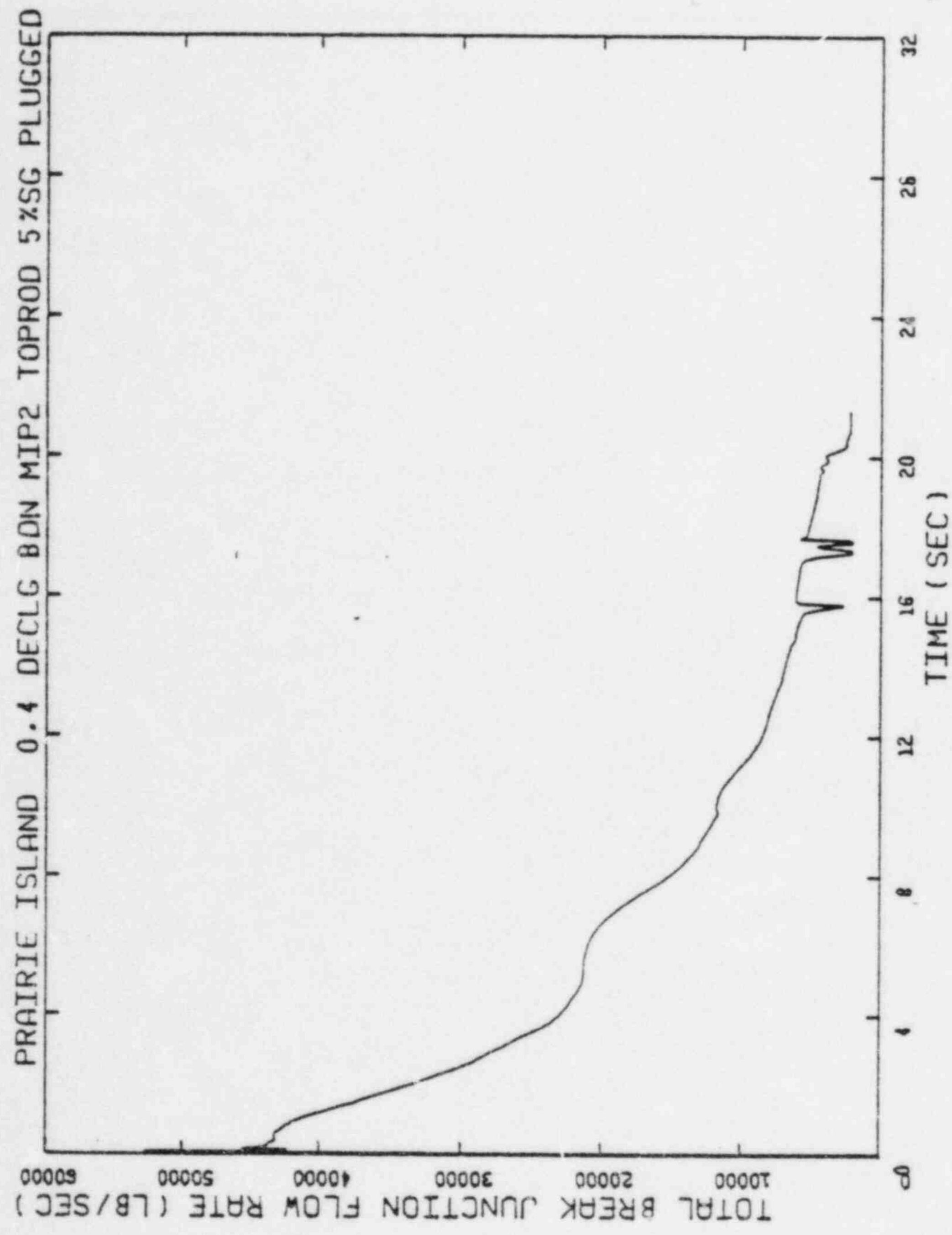

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Figure 2.11 Total Break Flow During Blowdown Period, 0.4 DECLG Break

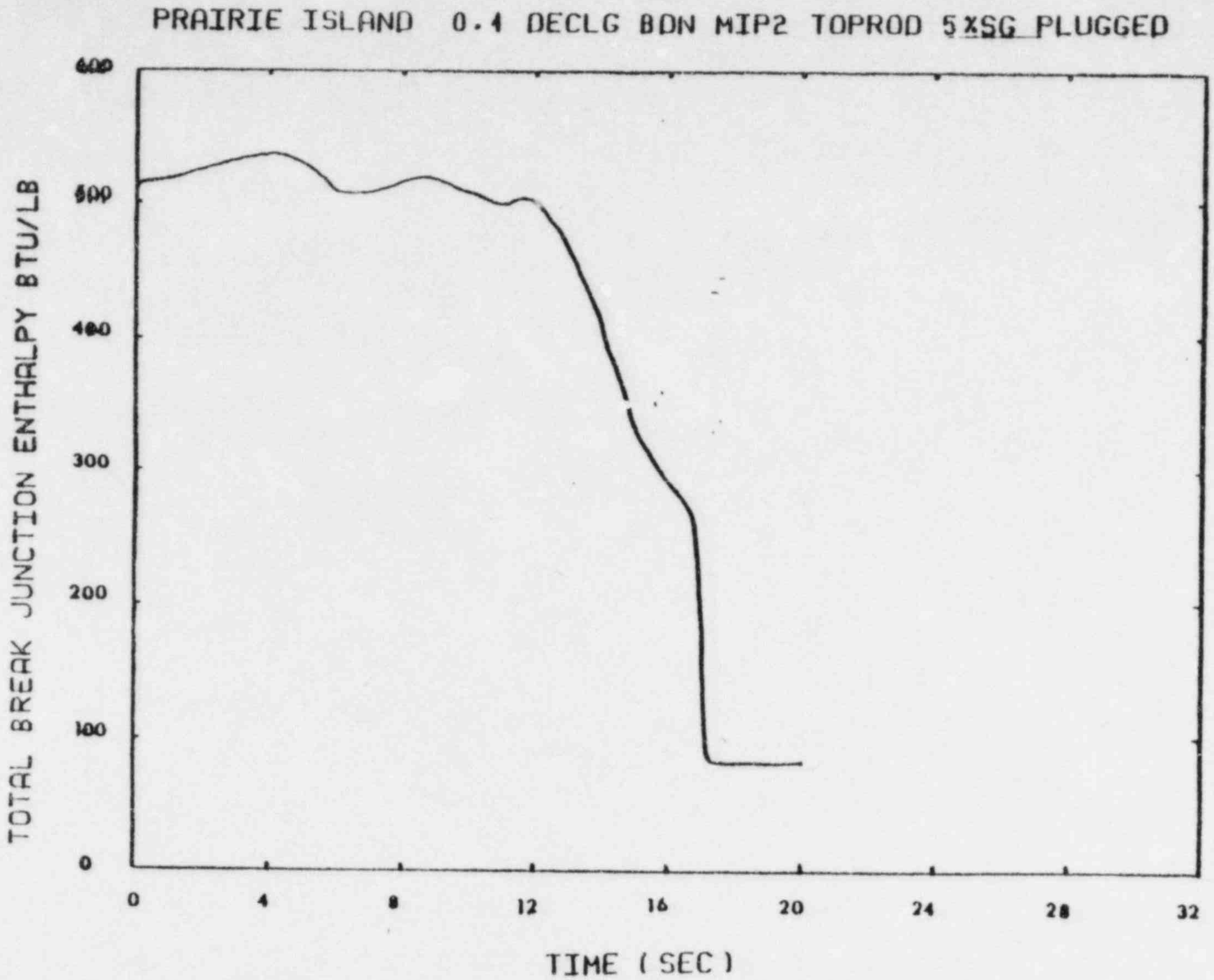

Figure 2.12 Break Flow Enthalpy during Blowdown Period, 0.4 DECLG Break

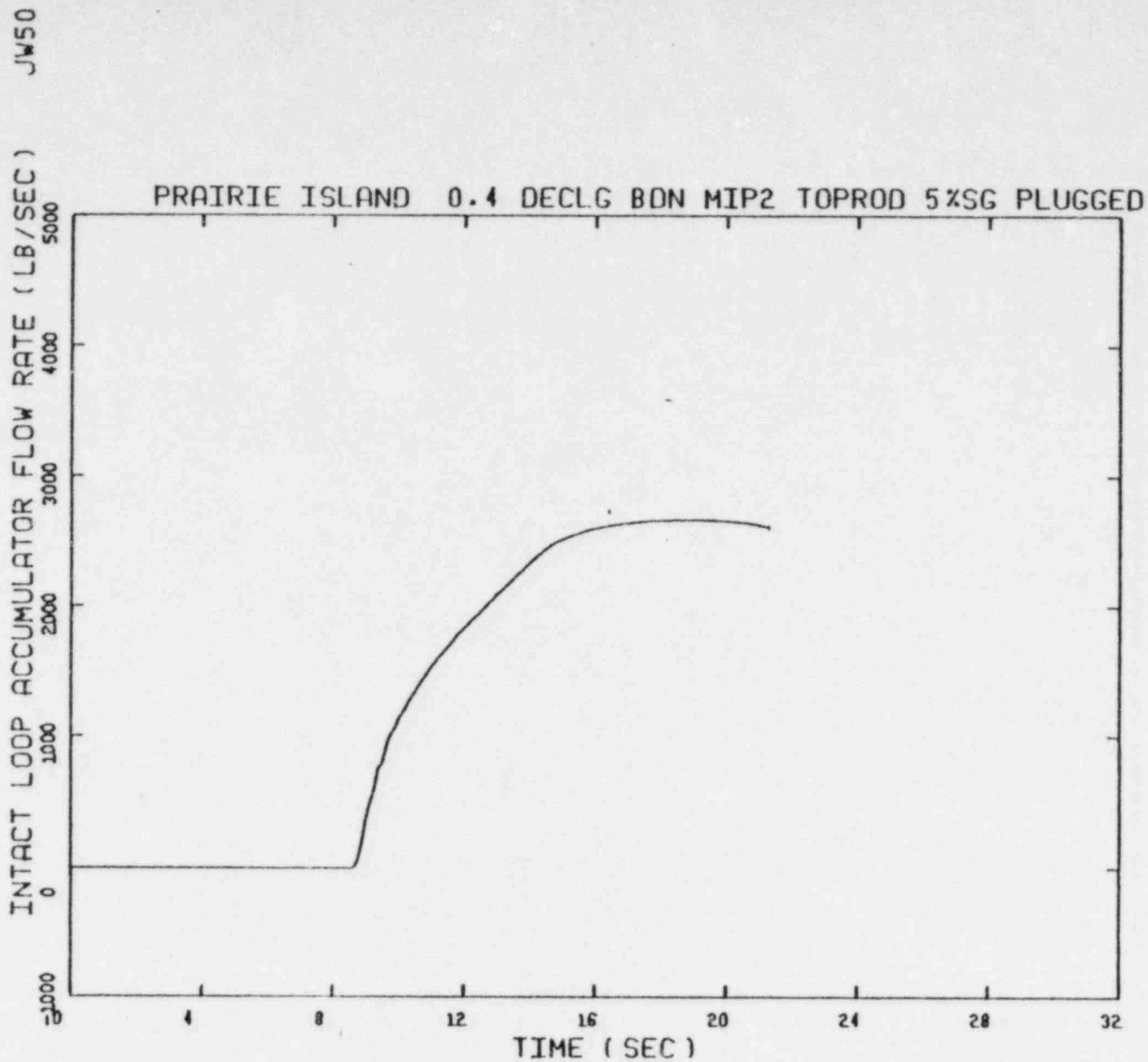

Figure 2.13 Flow from Intact Loop Accumulator During Blowdown Period, 0.4 DECLG Break

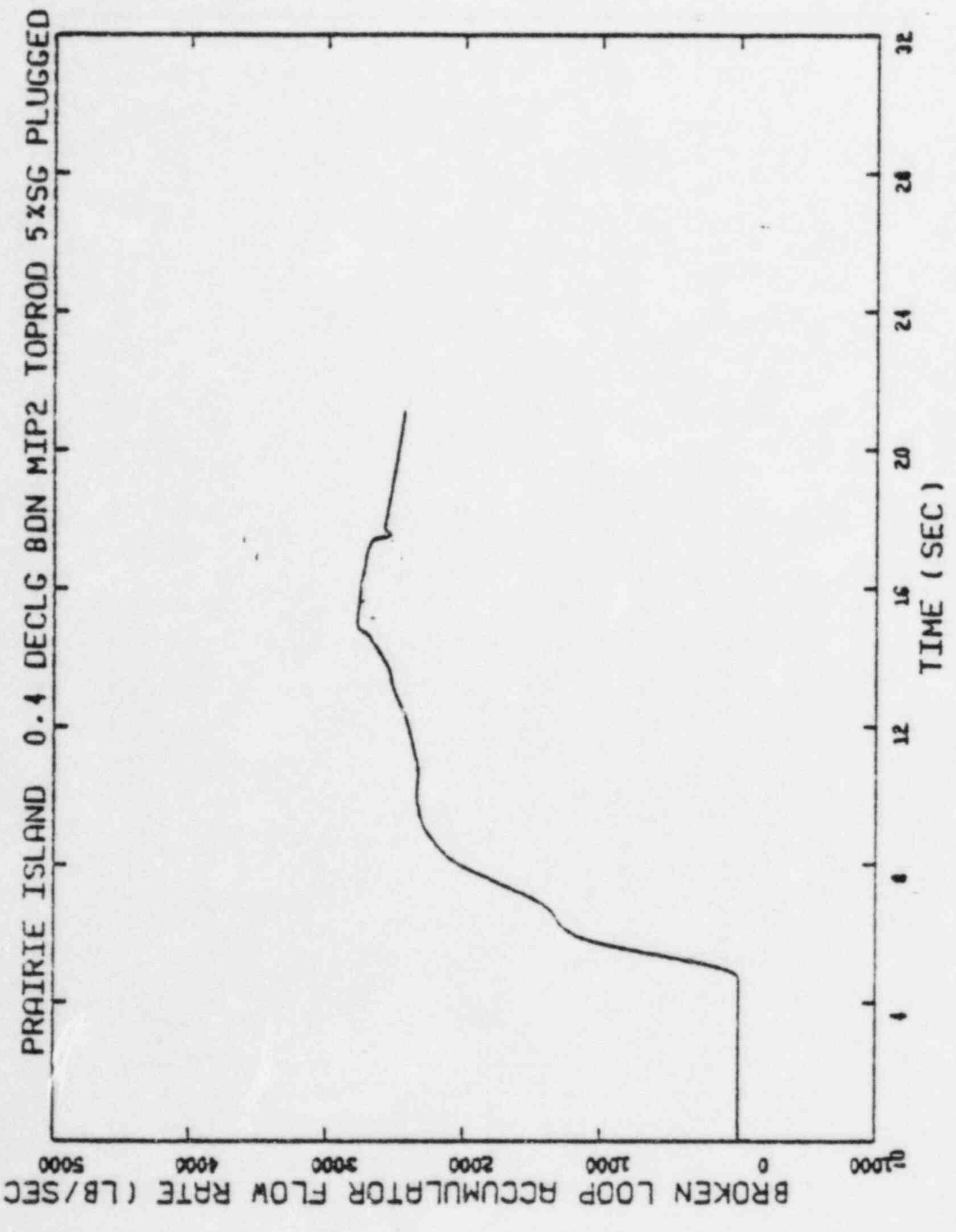

Figure 2.14 Flow from Broken Loop Accumulator During Blowdown Period, 0.4 DECLG Break

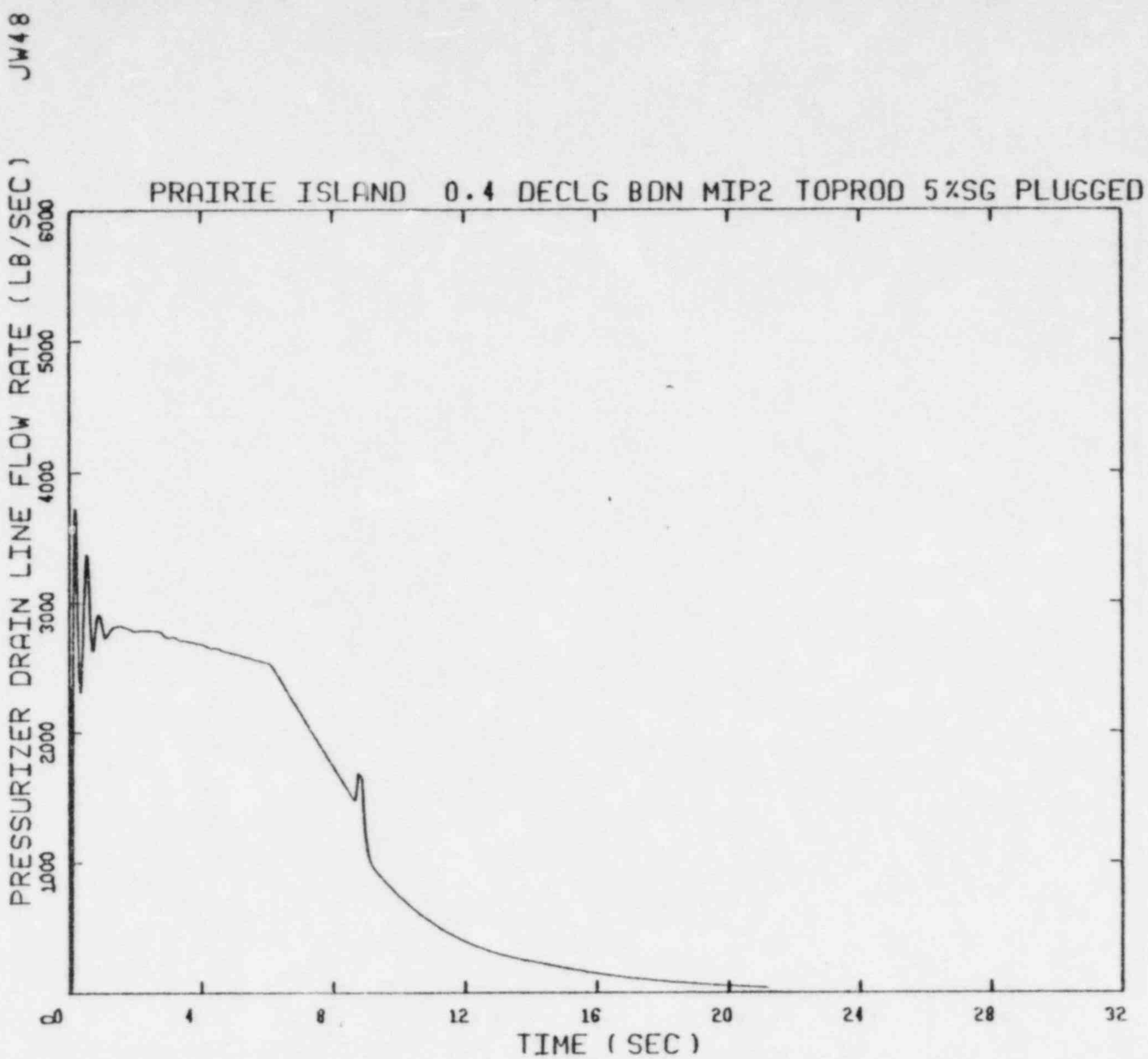

Figure 2.15 Pressurizer Surge Line Flow During Blowdown Period, 0.4 DECLG Break

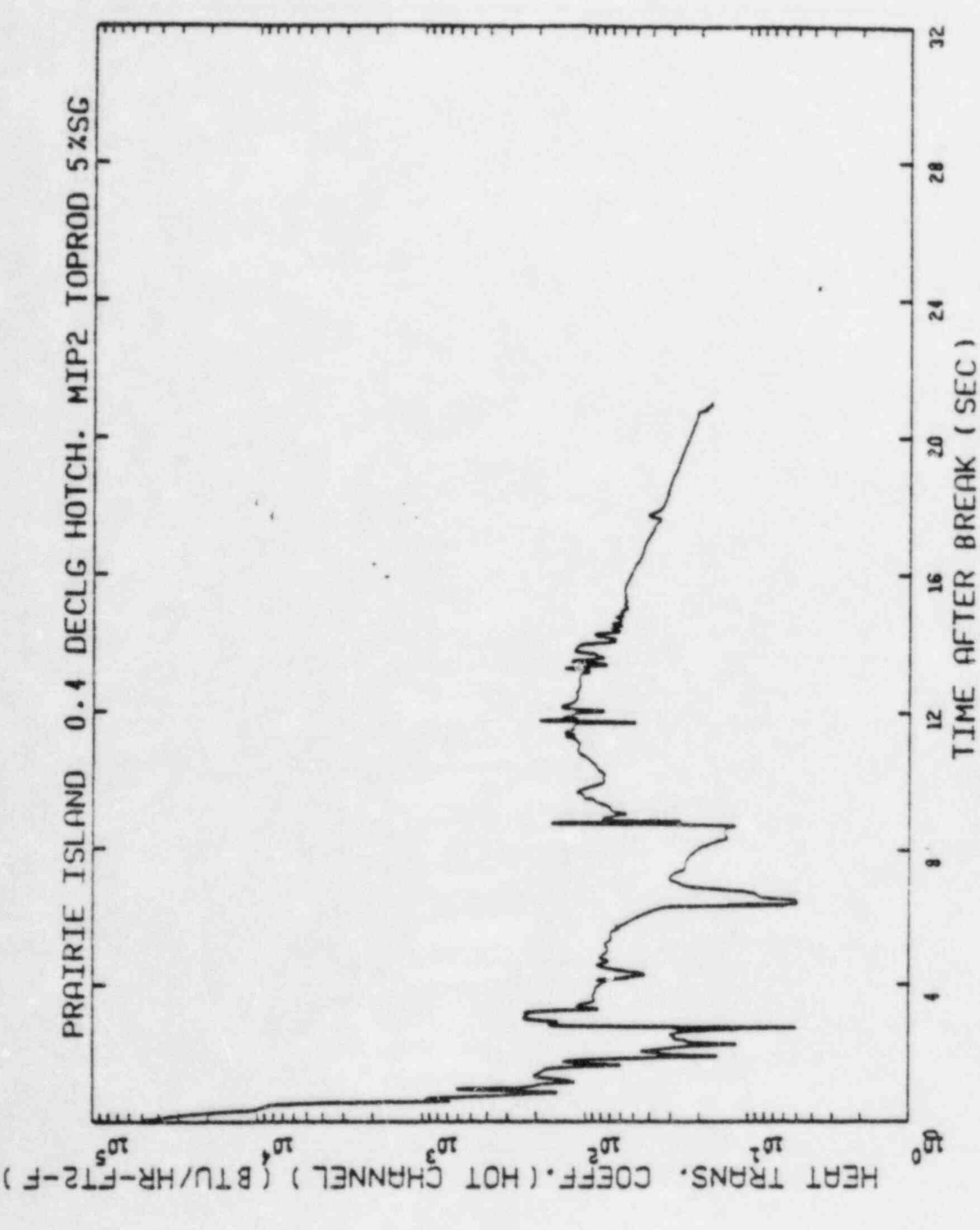

Figure 2.16 Heat Transfer Coefficient During Blowdown Period at PCT Node, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

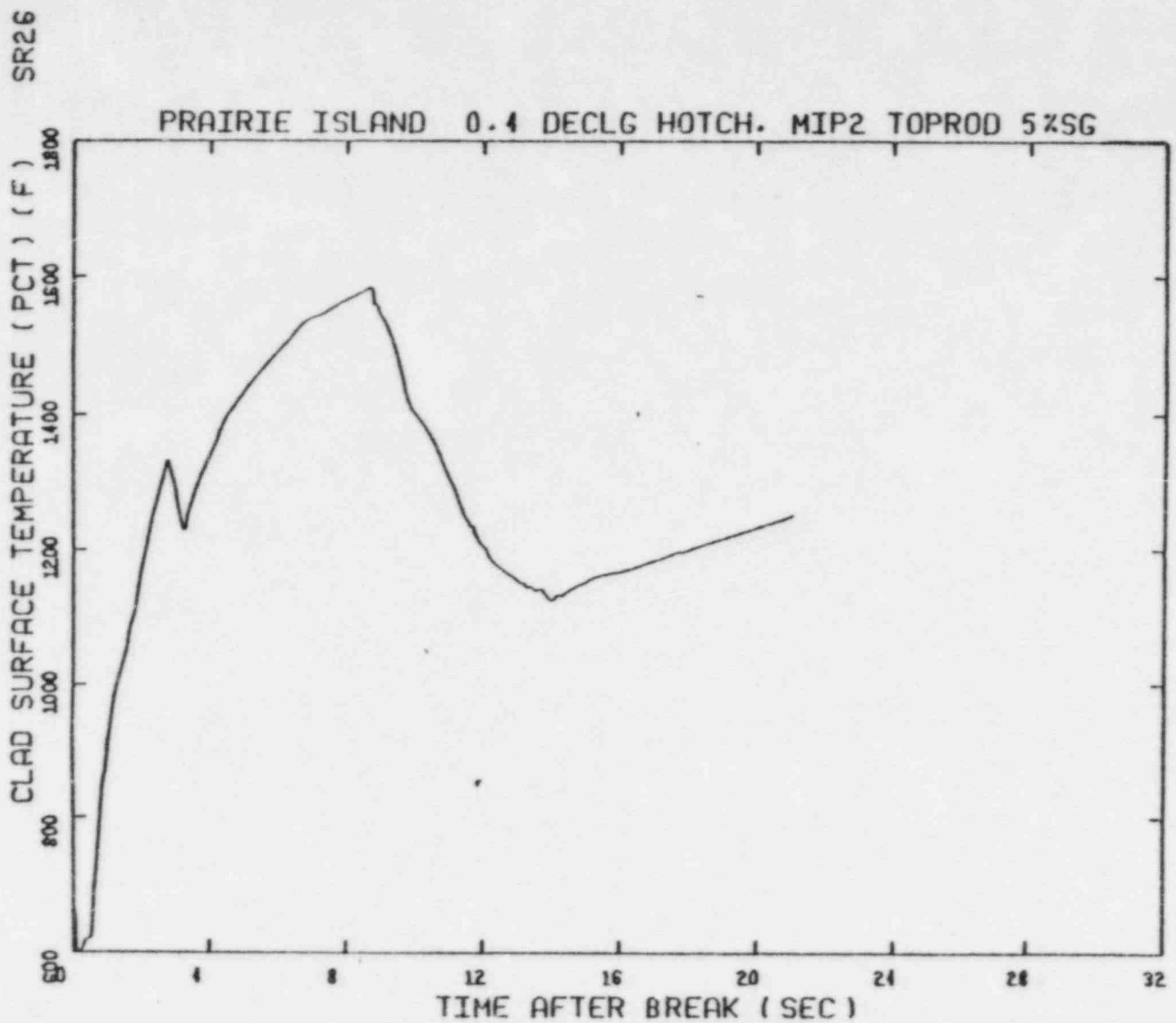

Figure 2.17 Clad Surface Temperature during Blowdown Period at PCT Node, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

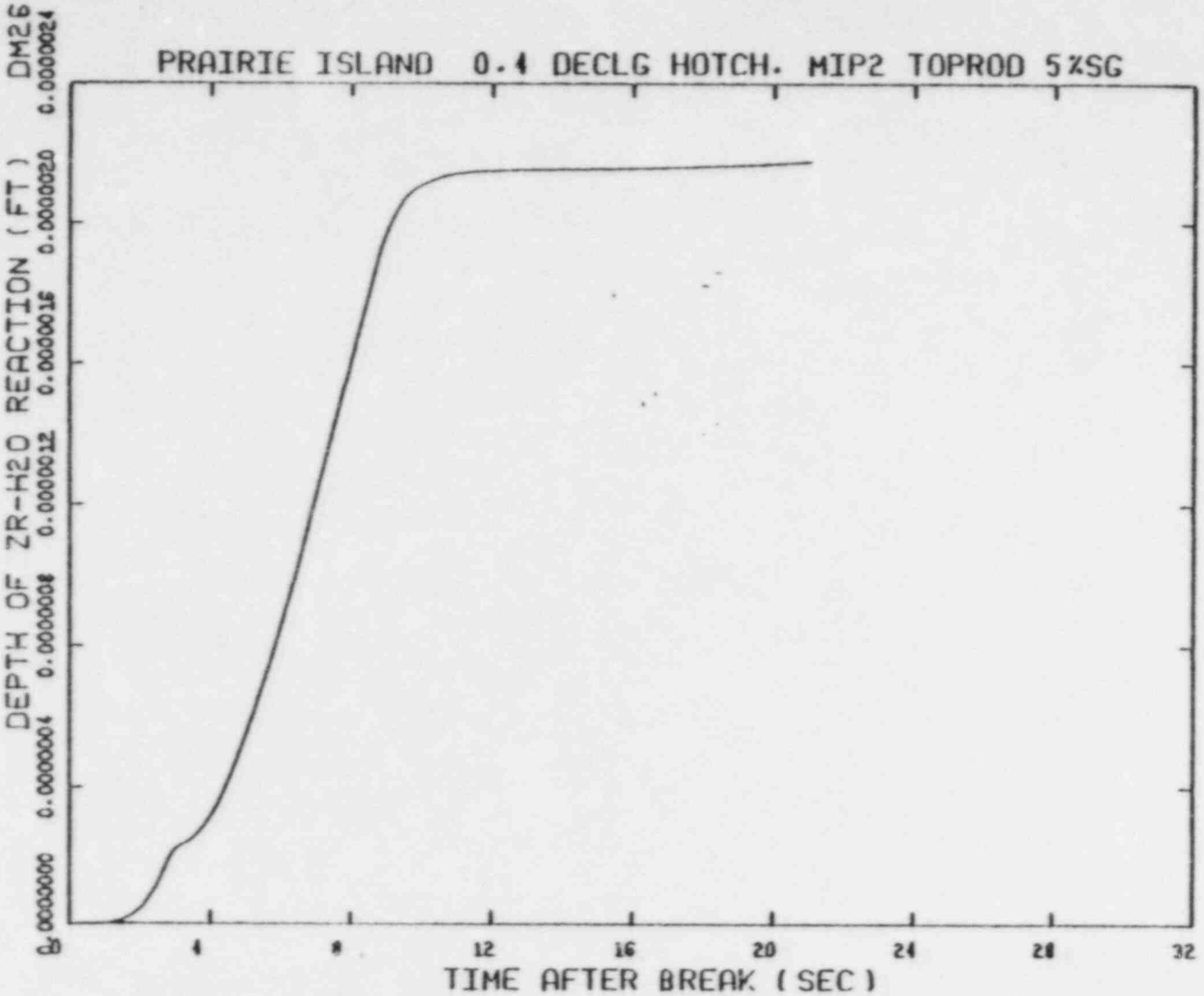

Figure 2.18 Depth of Metal-Water Reaction during Blowdown Period at PCT Node, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

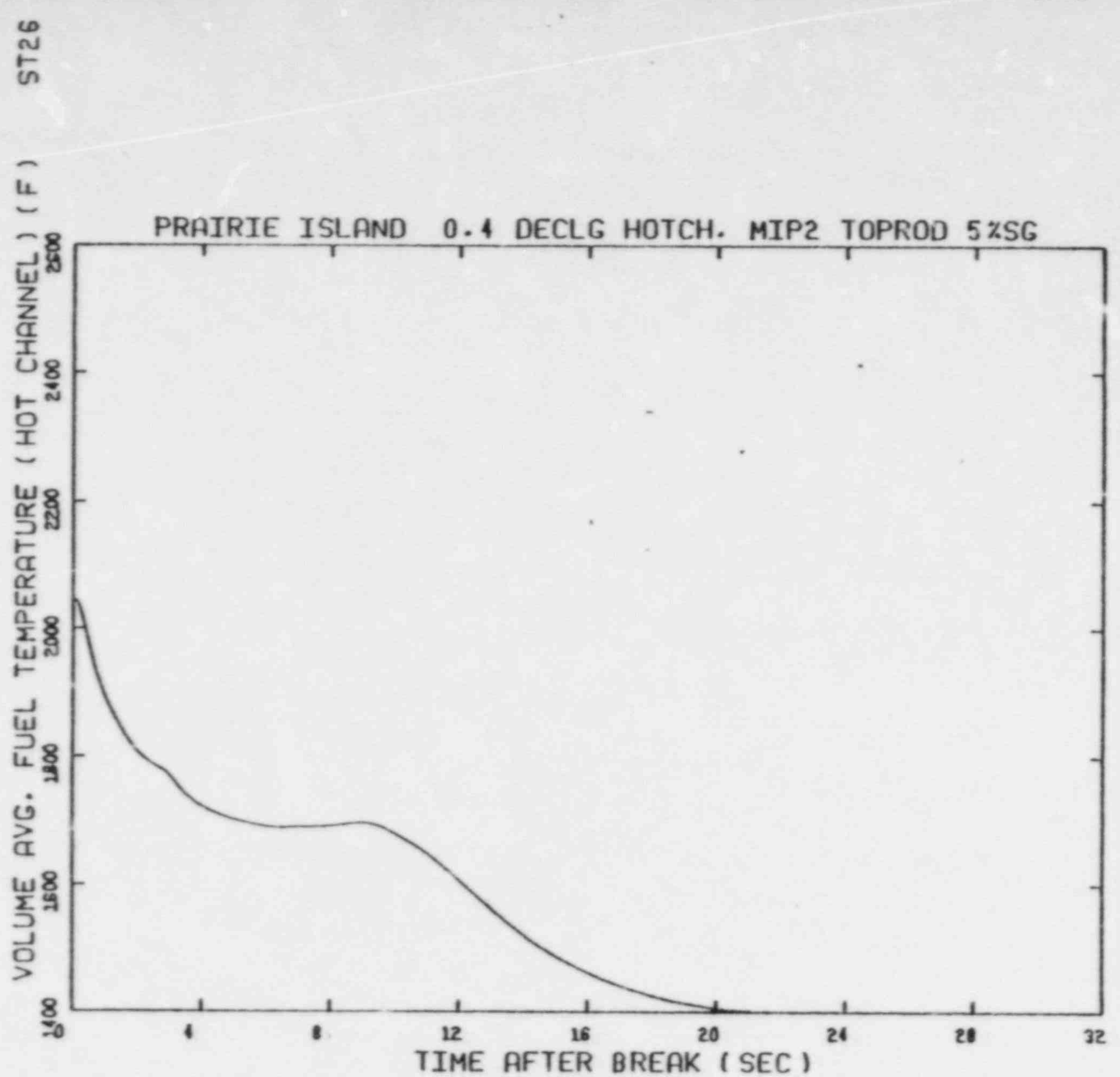

Figure 2.19 Average Fuel Temperature during Blowdown Period at PCT Location, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

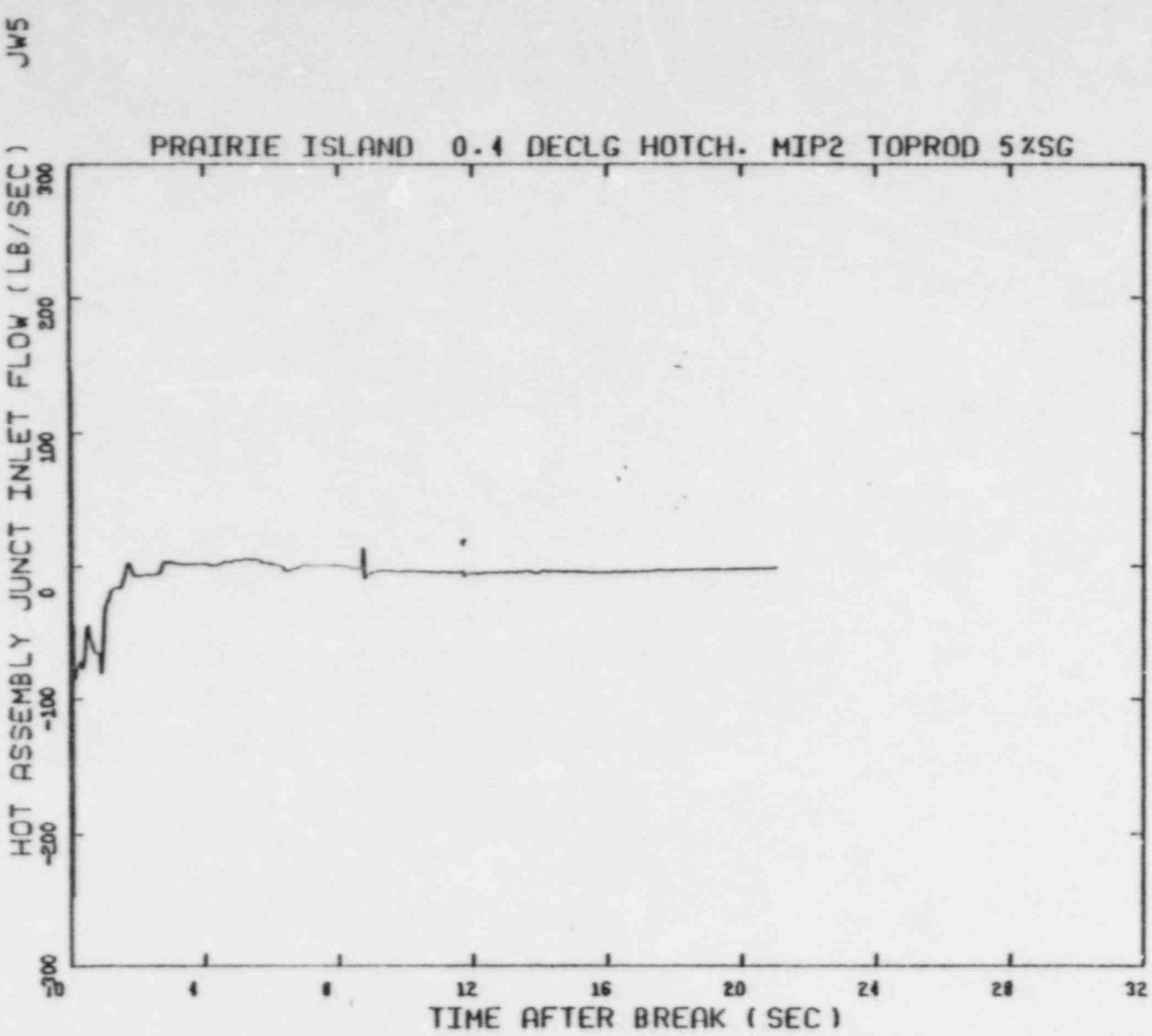

Figure 2.20 Hot Assembly Inlet Flow during Blowdown Period, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

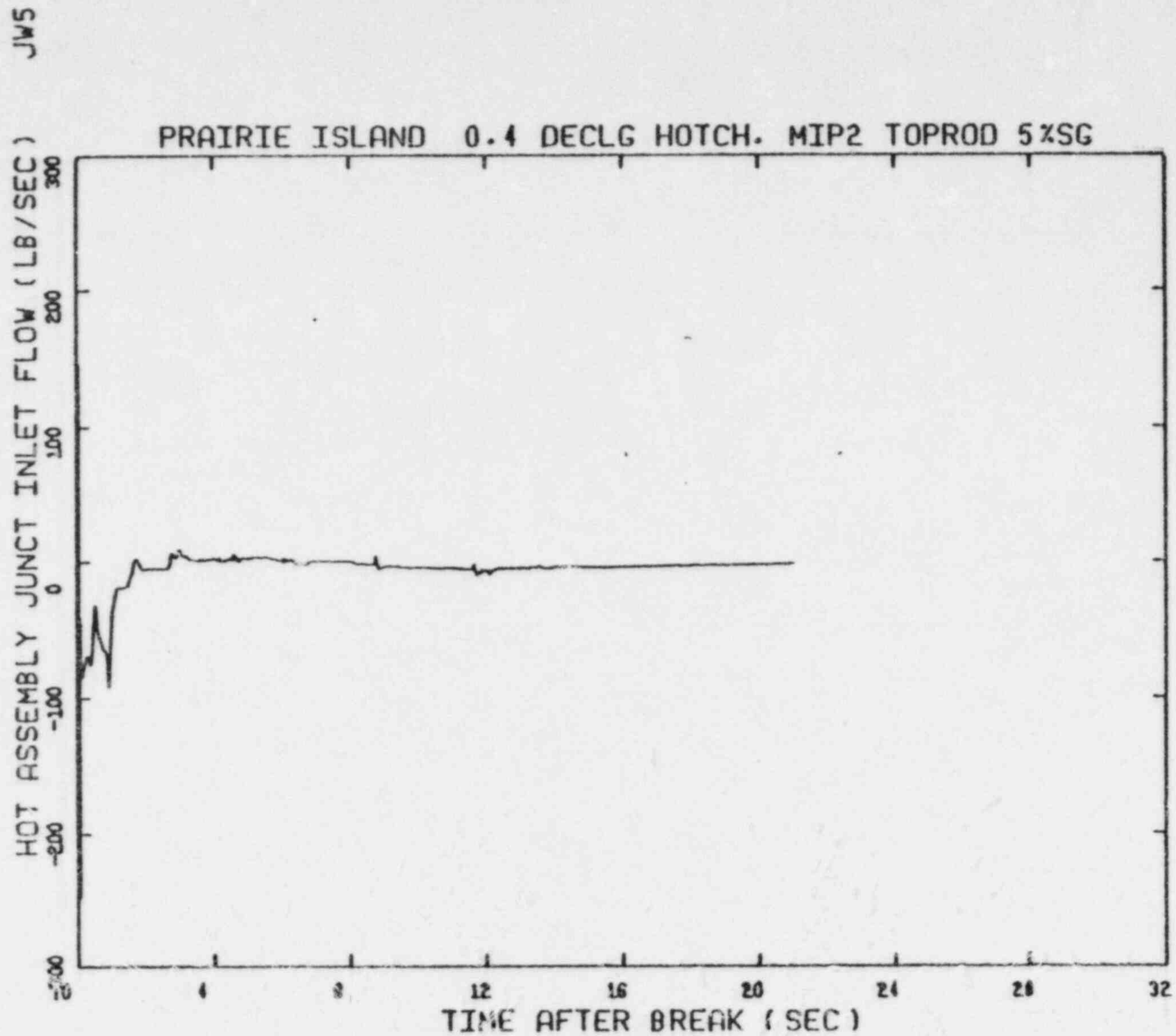

Figure 2.21 Hot Assembly Inlet Flow during Blowdown Period, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

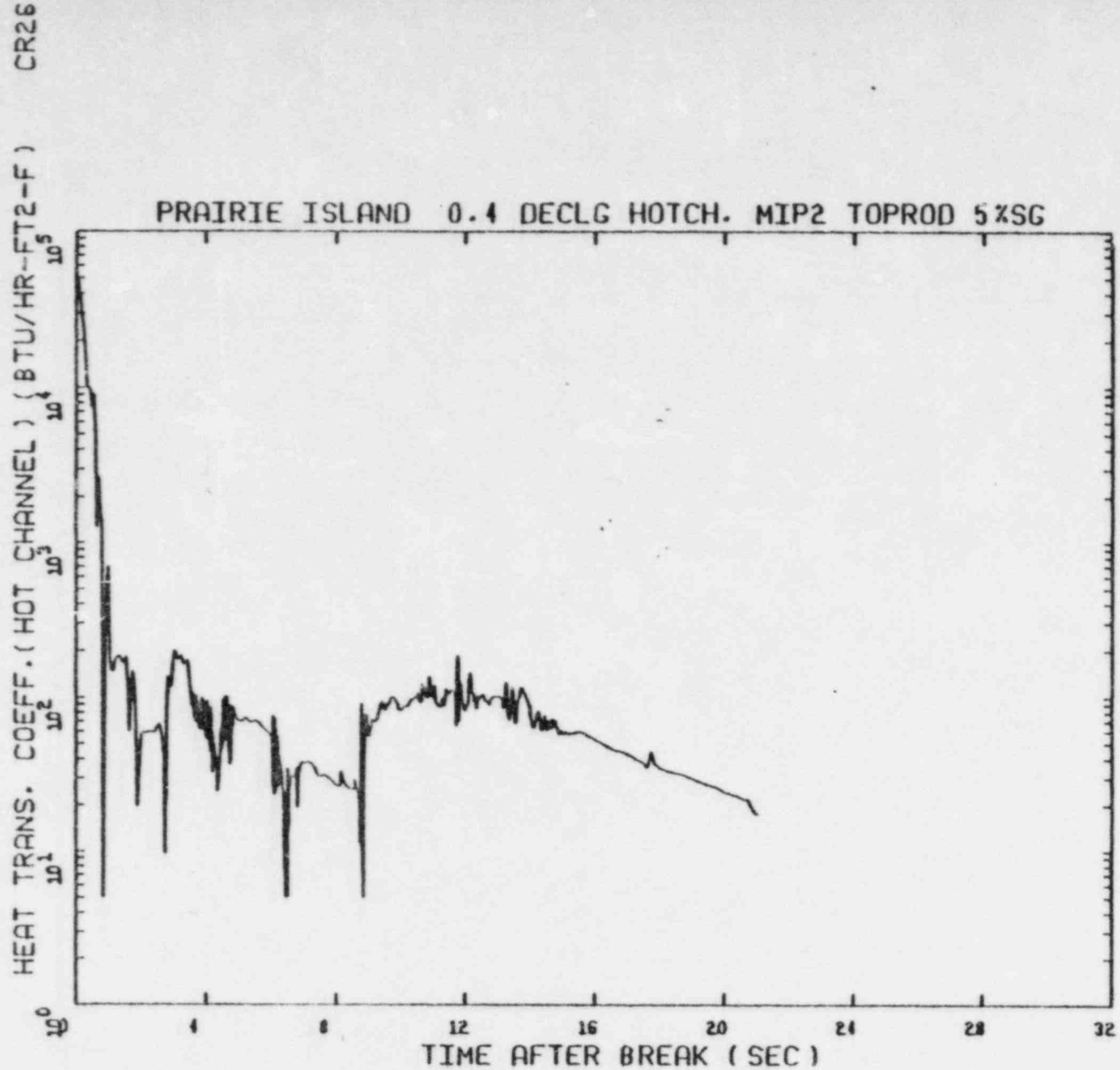

Figure 2.22 Heat Transfer Coefficient during Blowdown Period at PCI Node, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

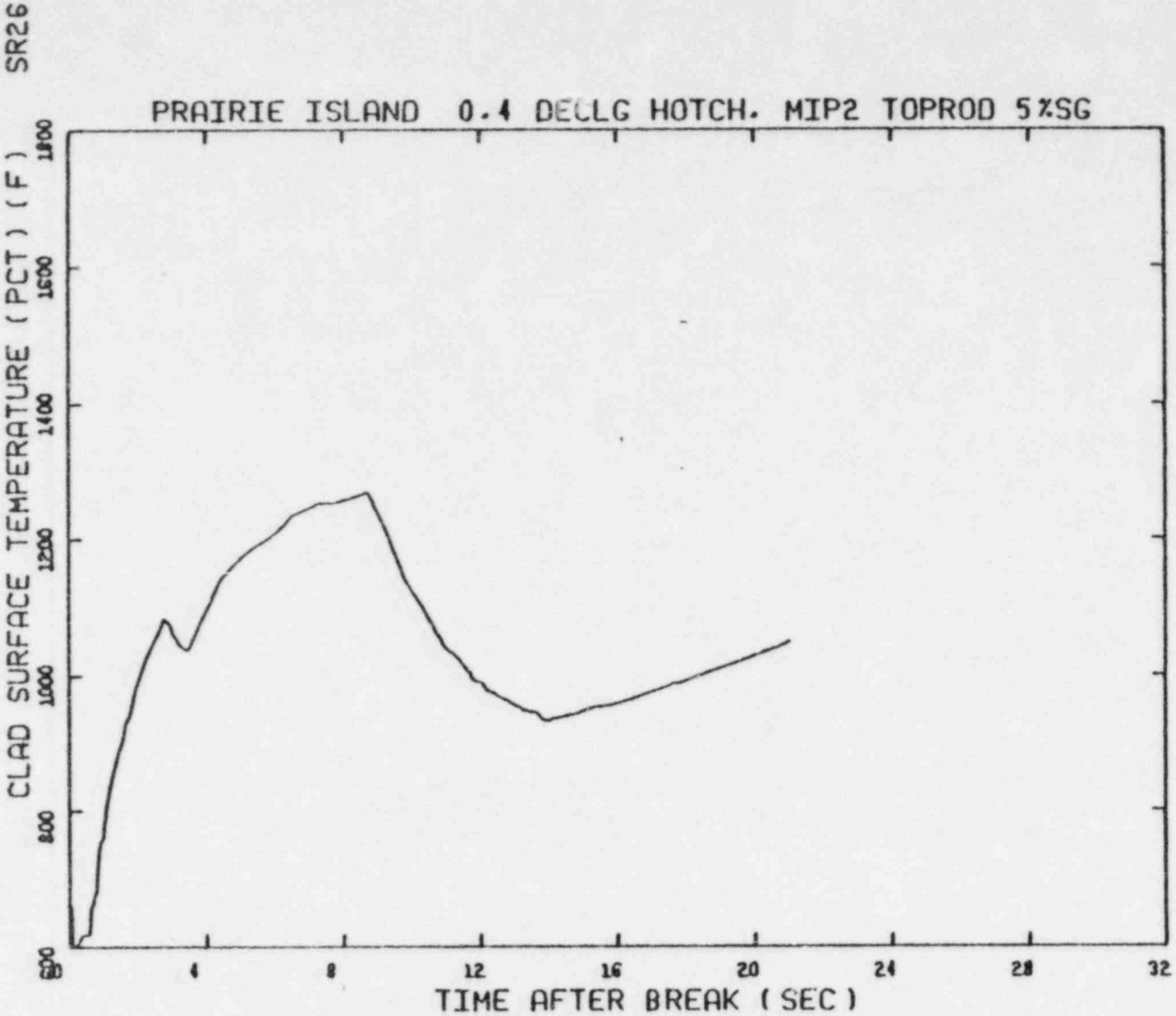

Figure 2.23 Clad Surface Temperature during Blowdown Period at PCT Node, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

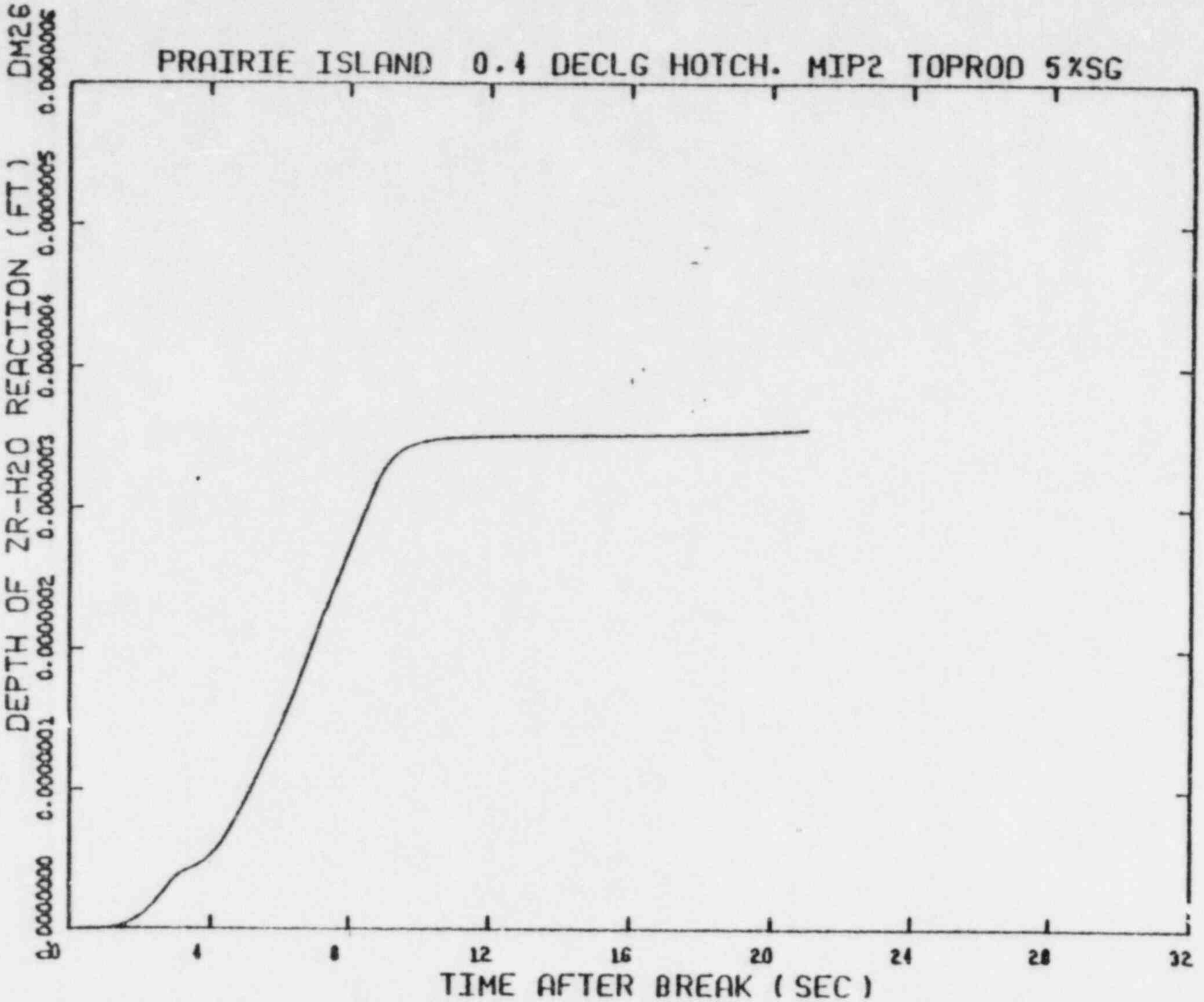

Figure 2.24 Depth of Metal-Water Reaction during Blowdown Period at PCT Node, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

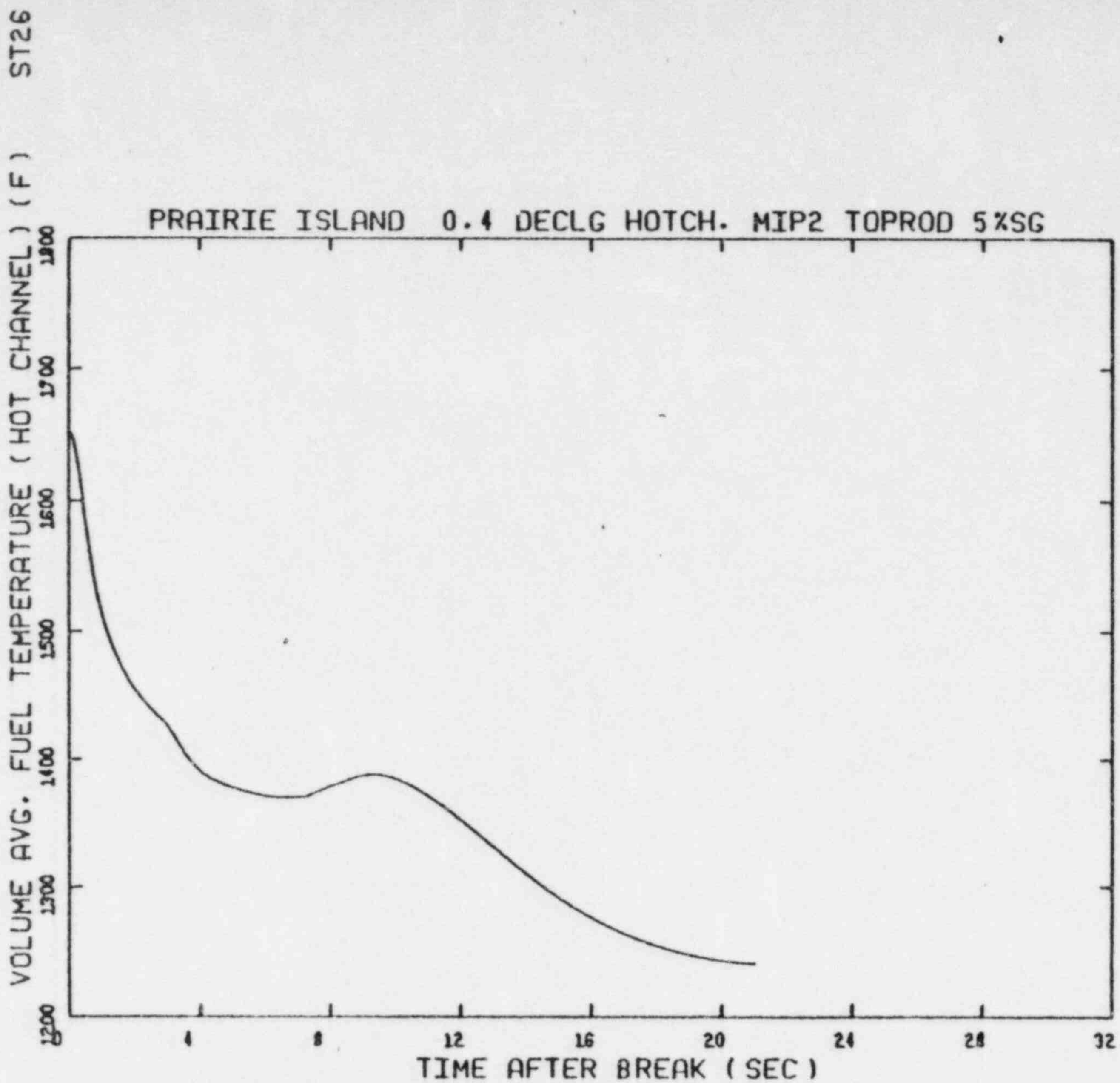

Figure 2.25 Average Fuel Temperature during Blowdown Period at PCT Location, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

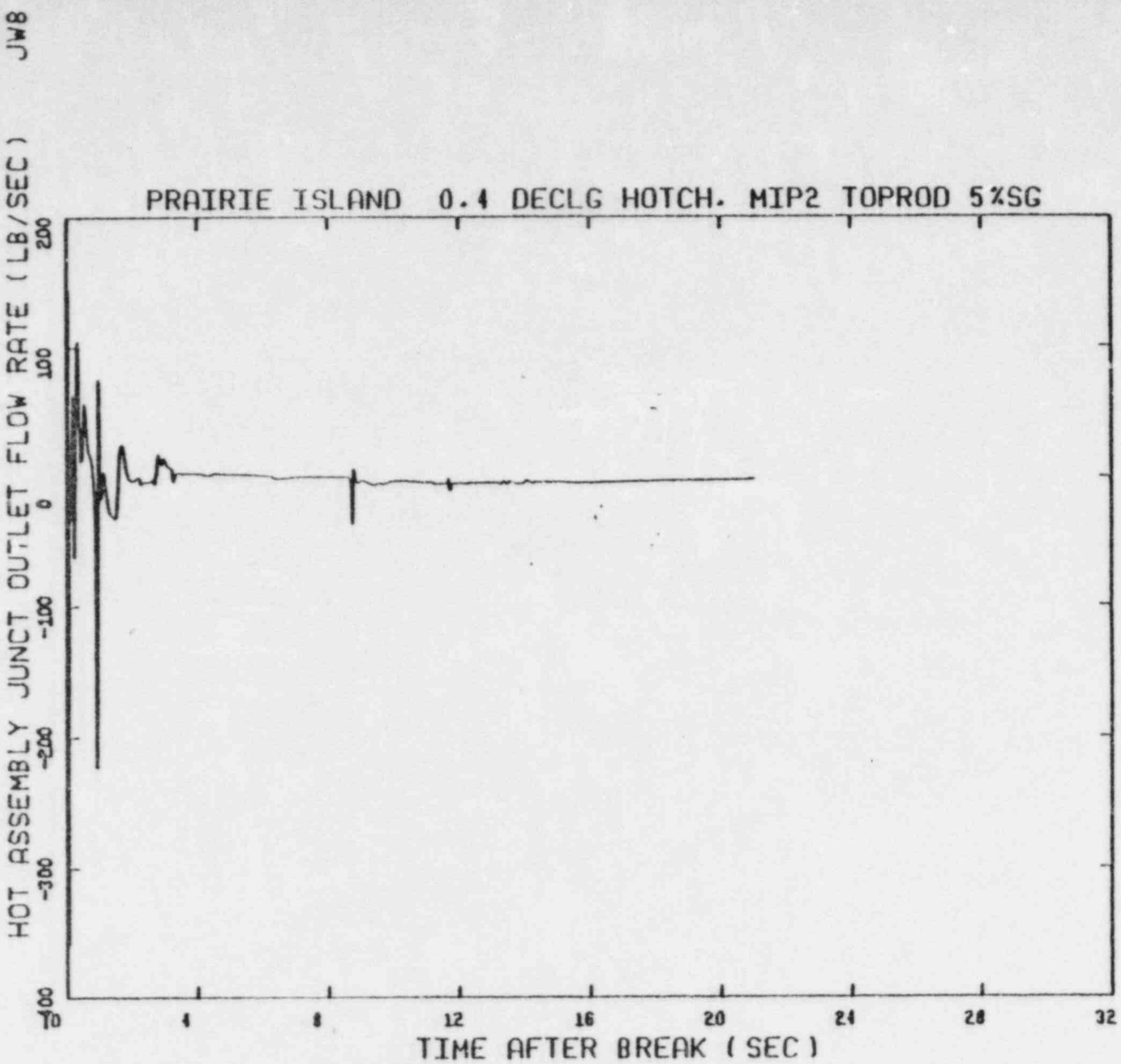

Figure 2.26 Hot Assembly Outlet Flow during Blowdown Period, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

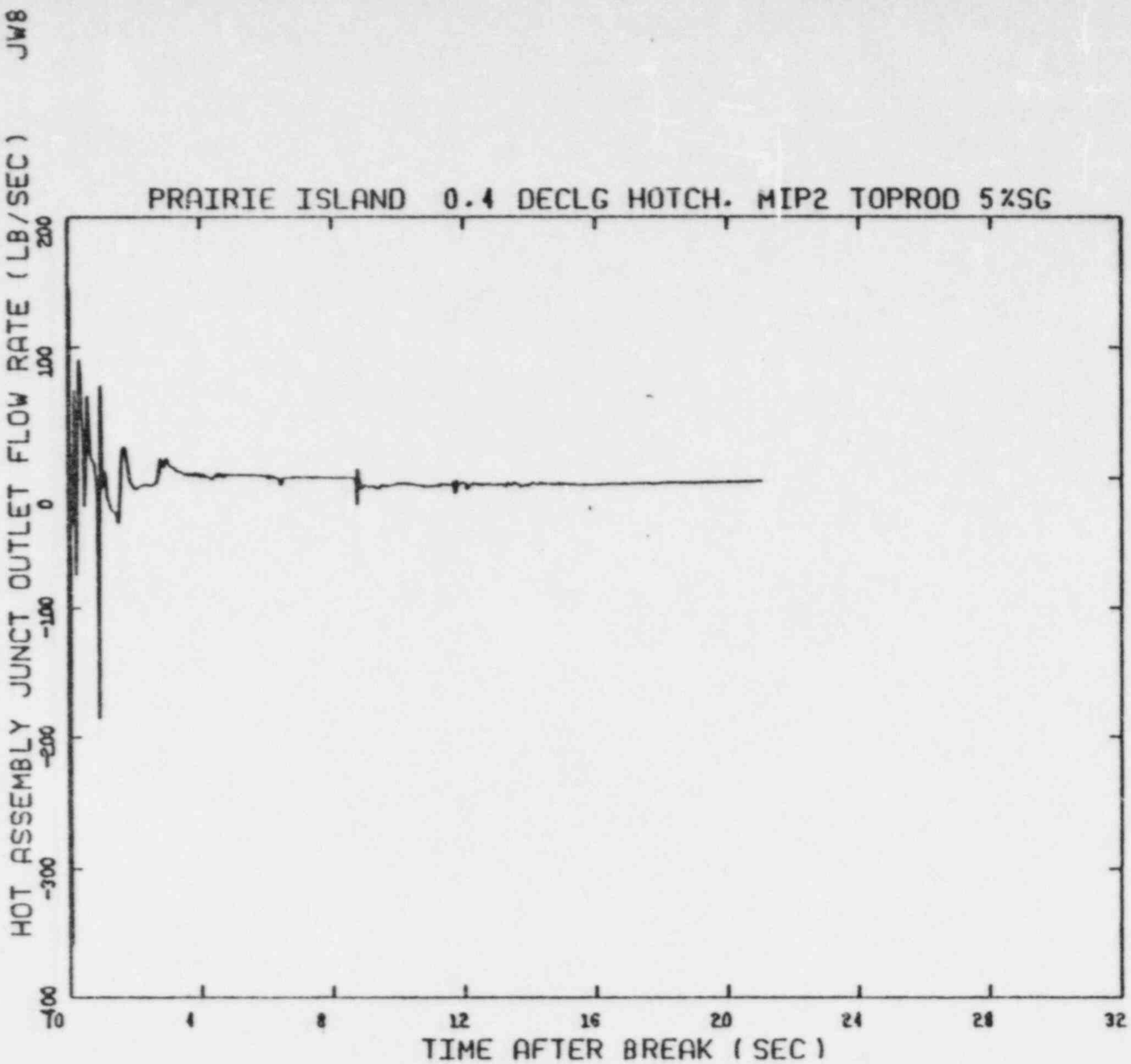

Figure 2.27 Hot Assembly Outlet Flow during Blowdown Period, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

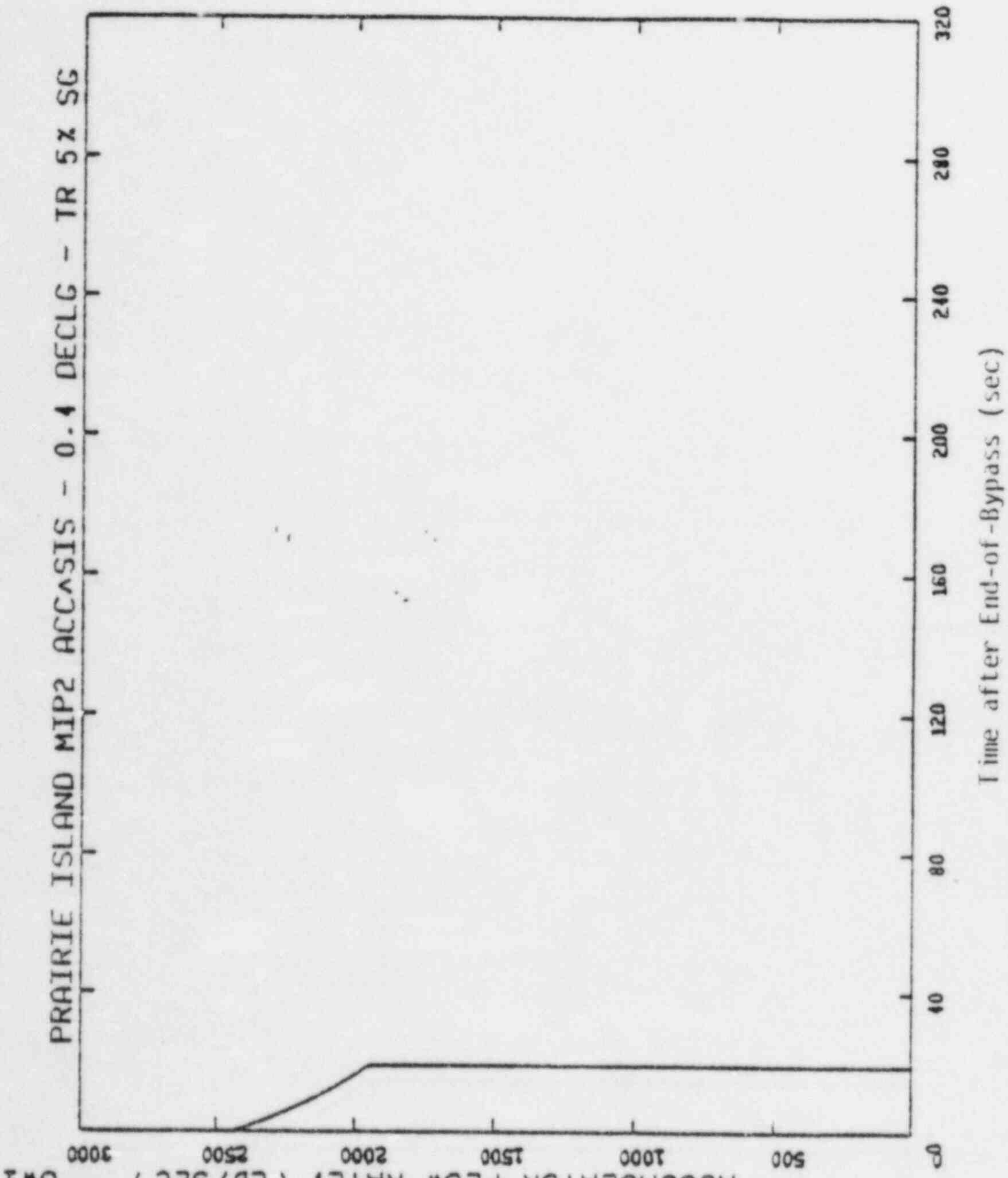

Figure 2.28 Accumulator Flow during Refill and Reflood Periods, Broken Loop, 0.4 DECLG Break

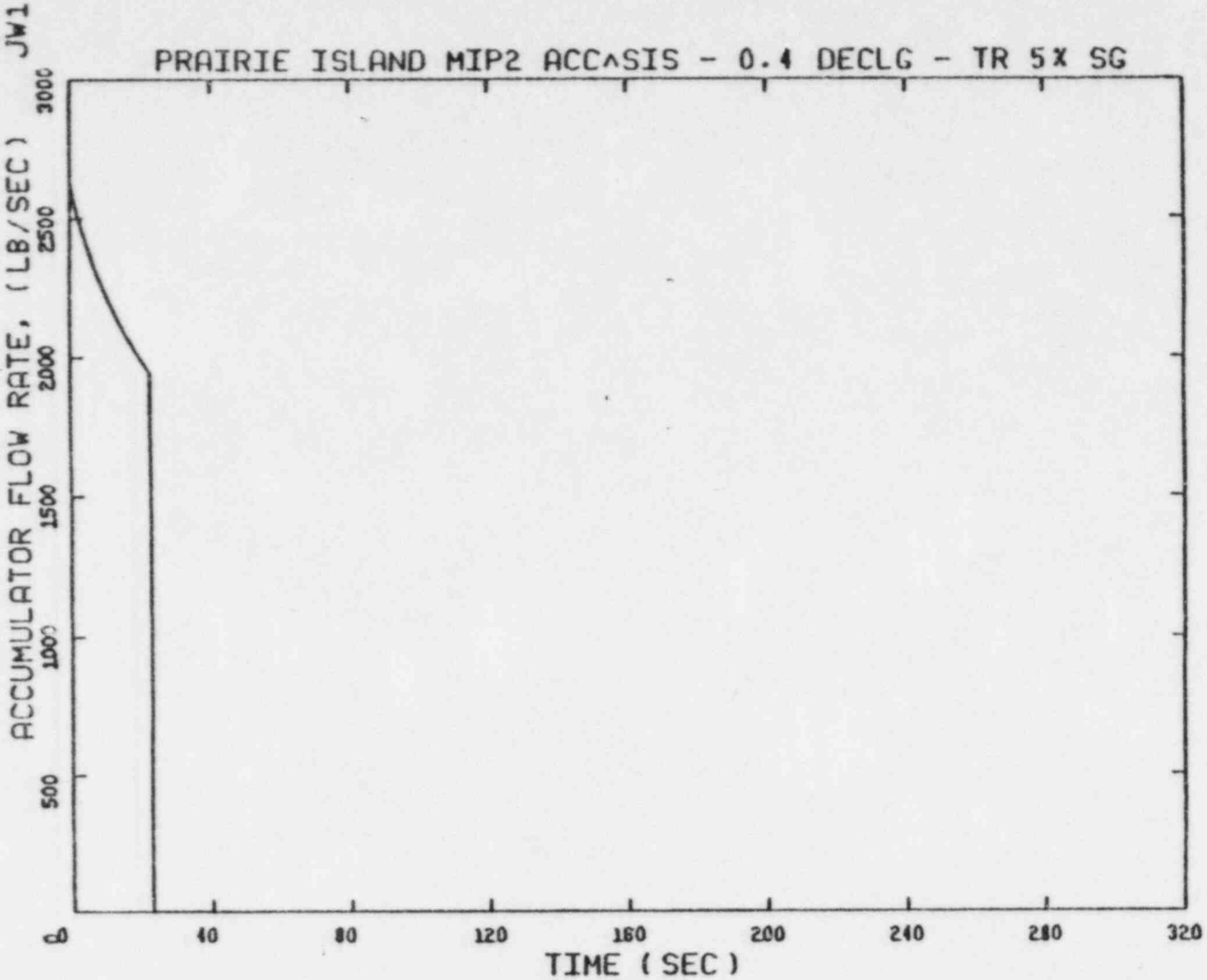

Figure 2.29 Accumulator Flow during Refill and Reflood Periods, Intact Loop, 0.4 DECLG Break

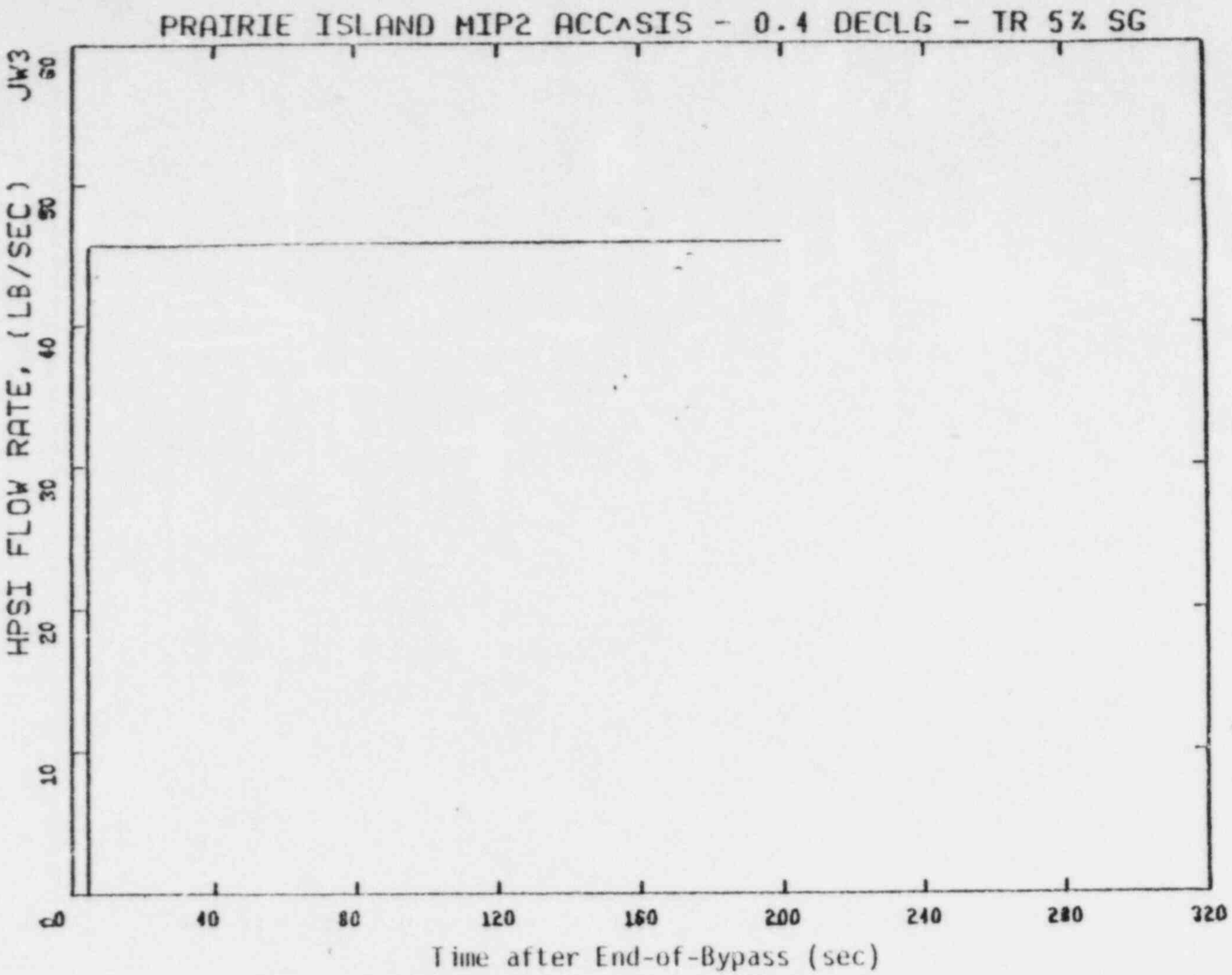

Figure 2.30 HPSI Flow during Refill and Reflood Periods, Broken Loop, 0.4 DECLG Break

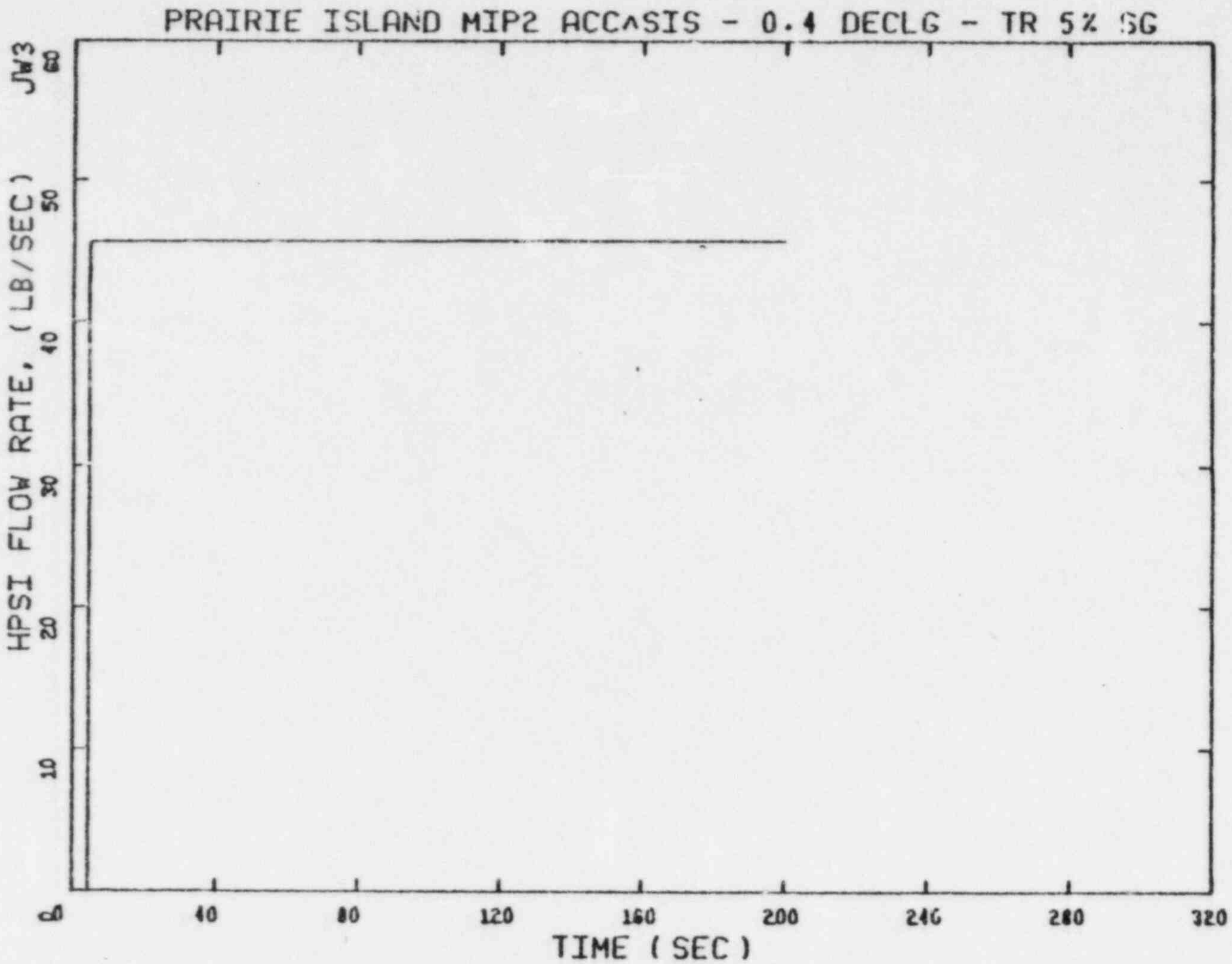

Figure 2.31 HPSI Flow during Refill and Reflood Periods, Intact Loop, 0.4 DECLG Break

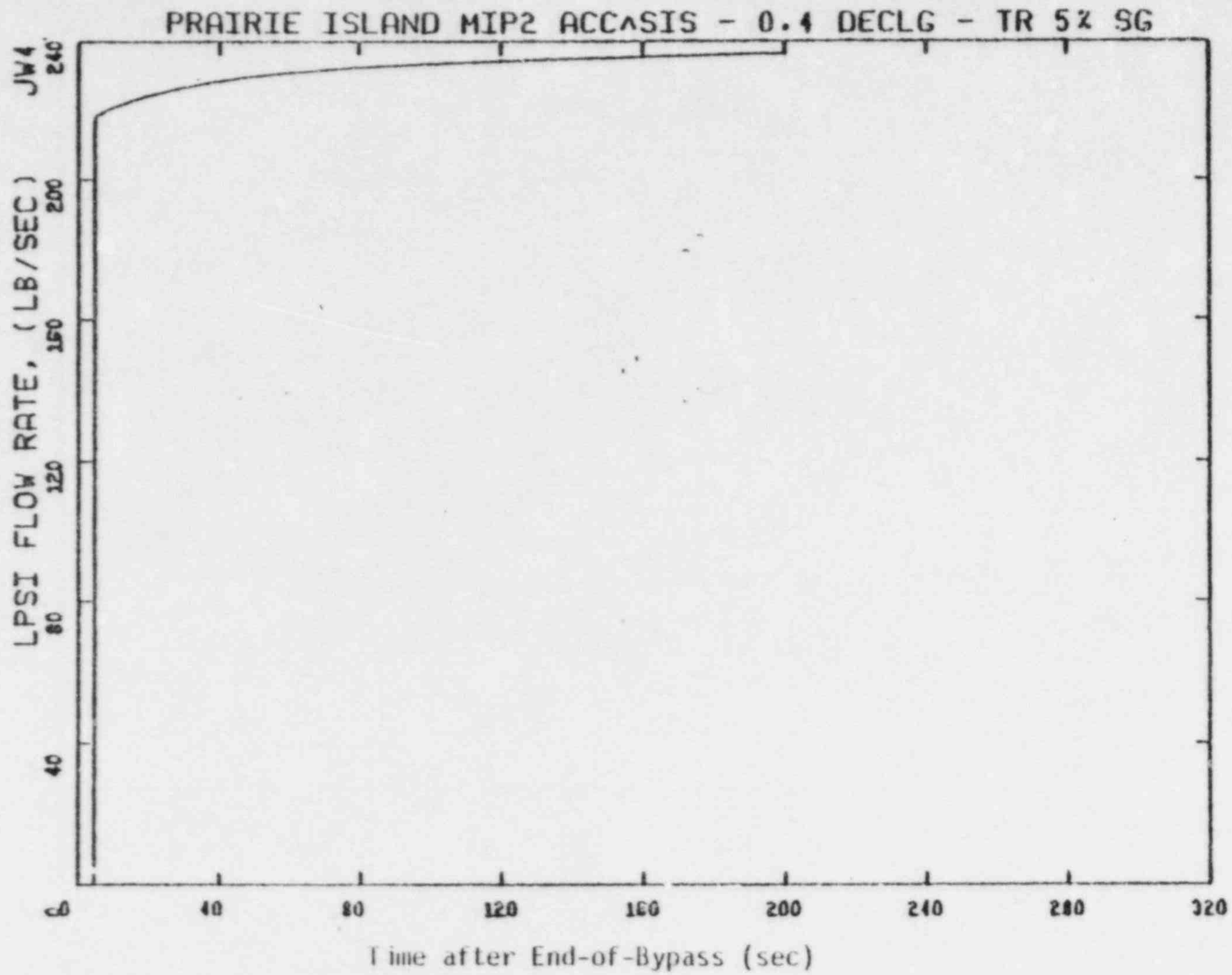

Figure 2.32 LPSI Flow during Refill and Reflood Periods, Broken Loop, 0.4 DECLG Break

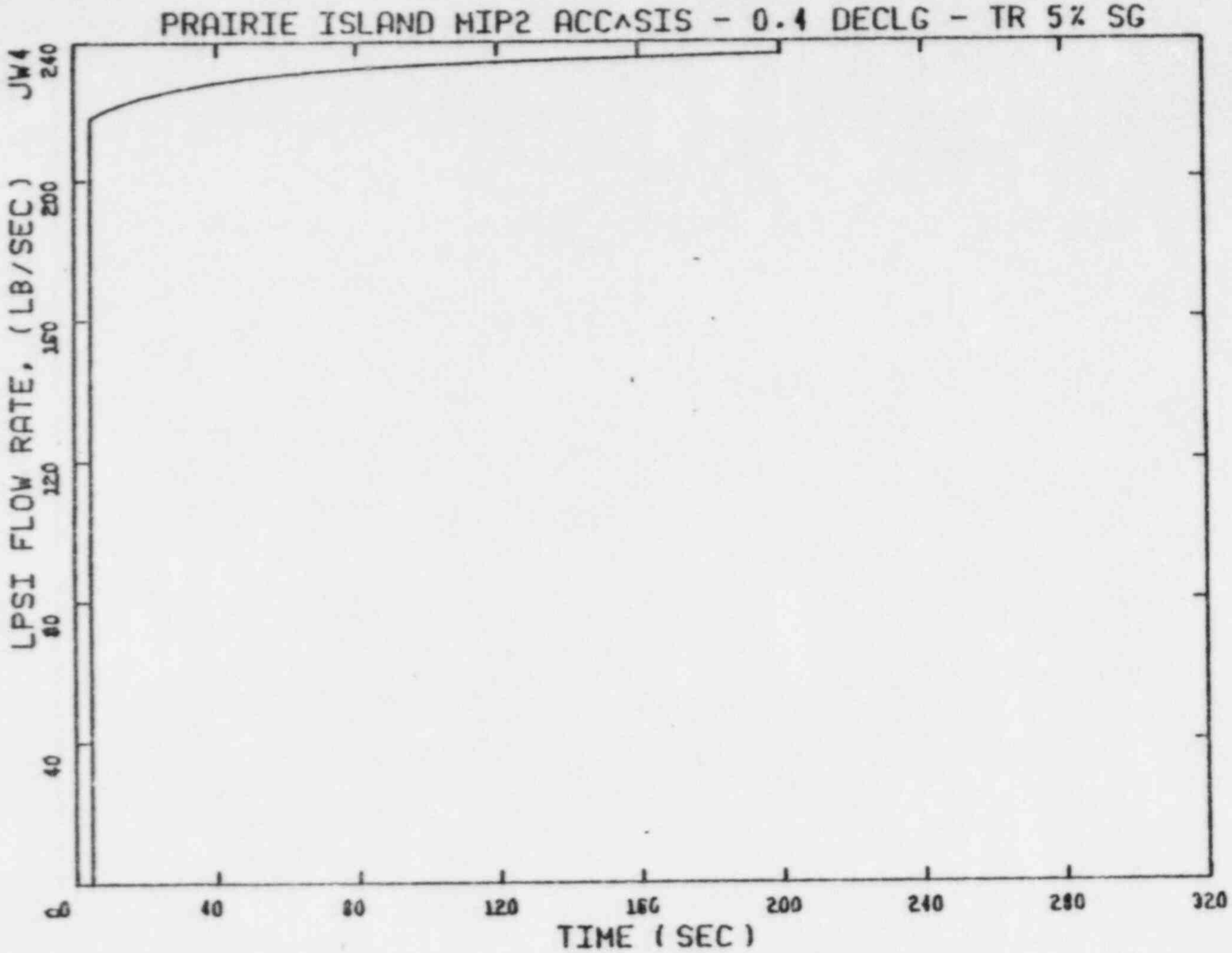

Figure 2.33 LPSI Flow during Refill and Reflood Periods, Intact Loop, 0.4 DECLG Break

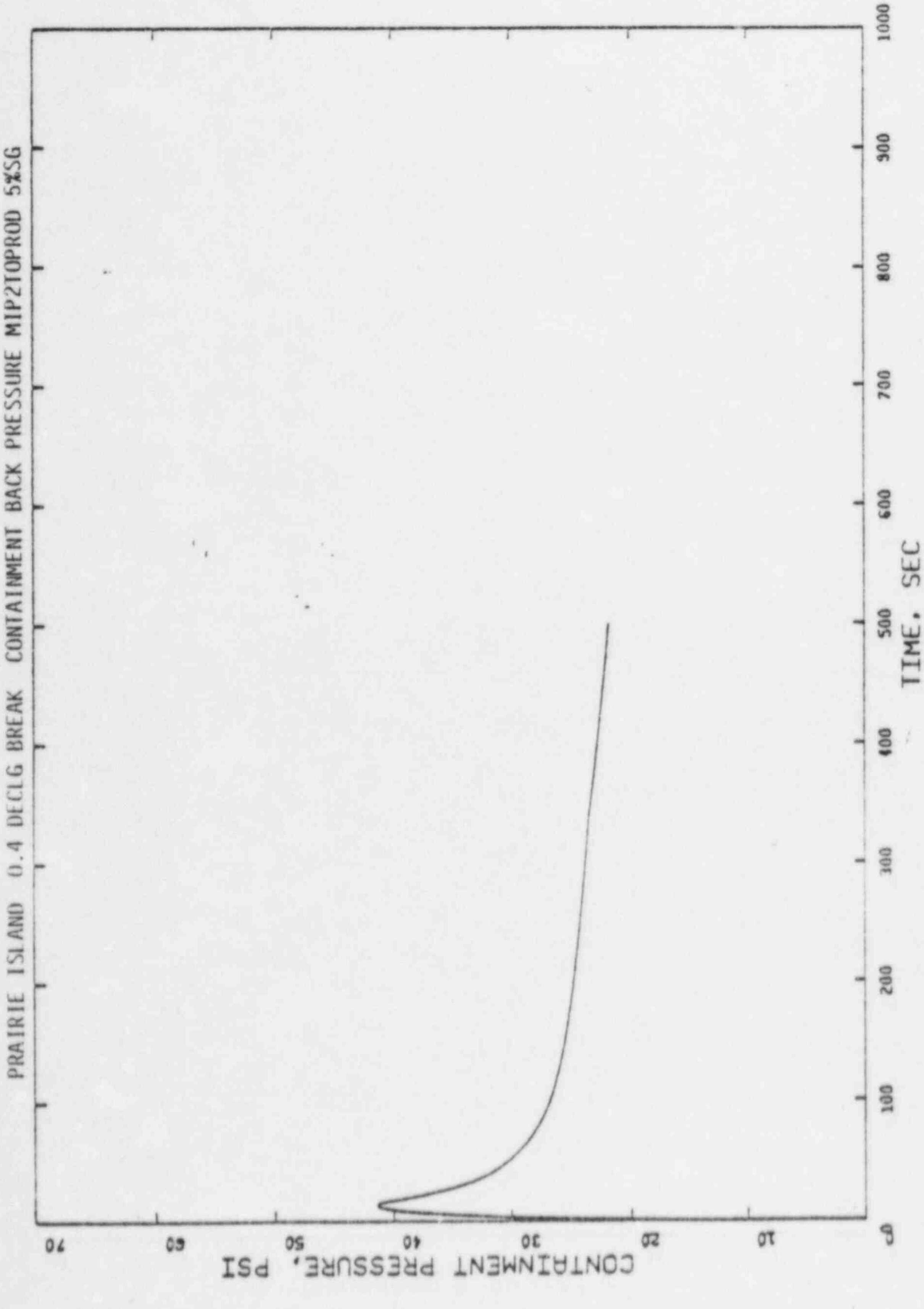

Figure 2.34 Containment Back Pressure, 0.4 DECLG Break

Figure 2.35 Normalized Power, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

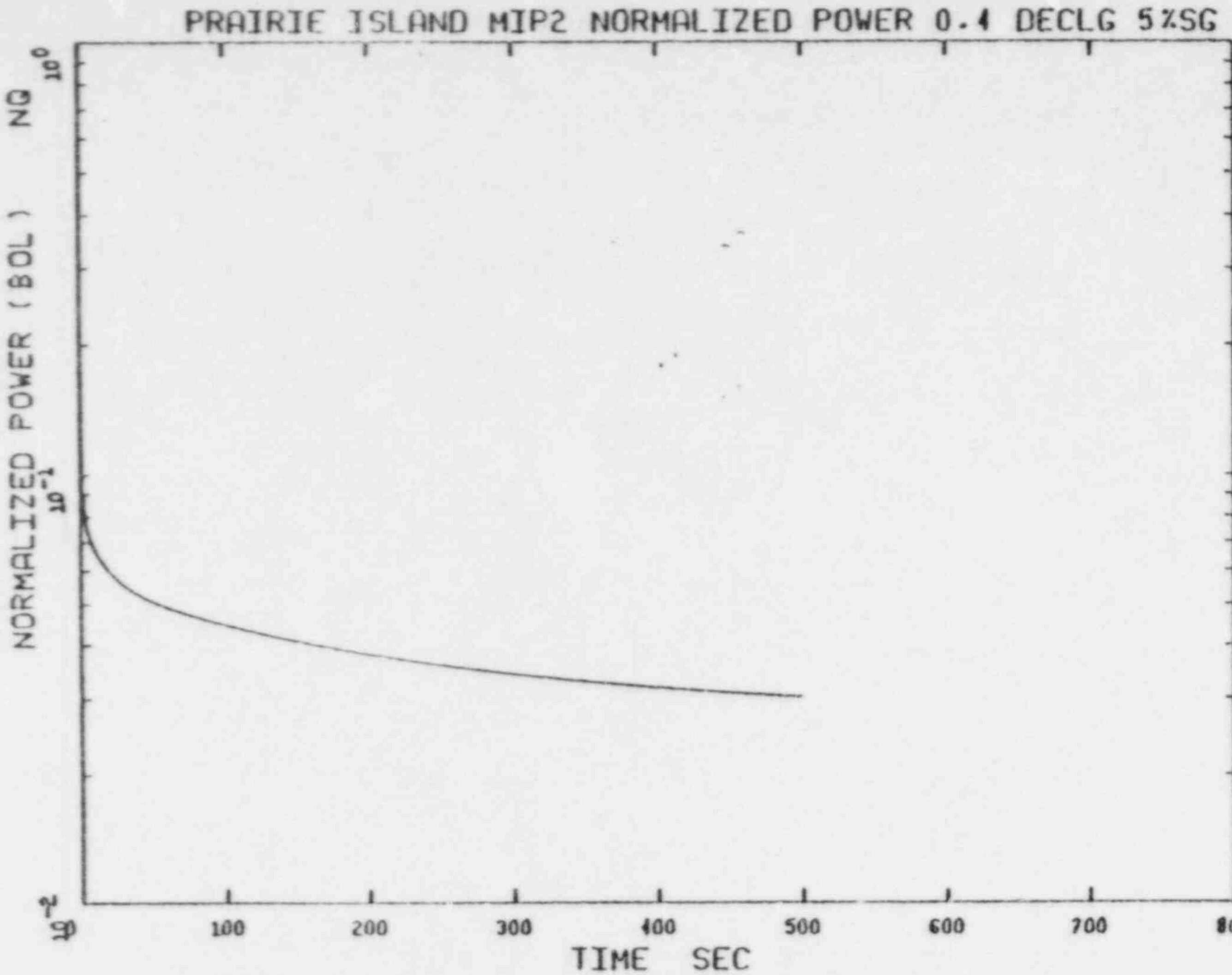

Figure 2.36 Normalized Power, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

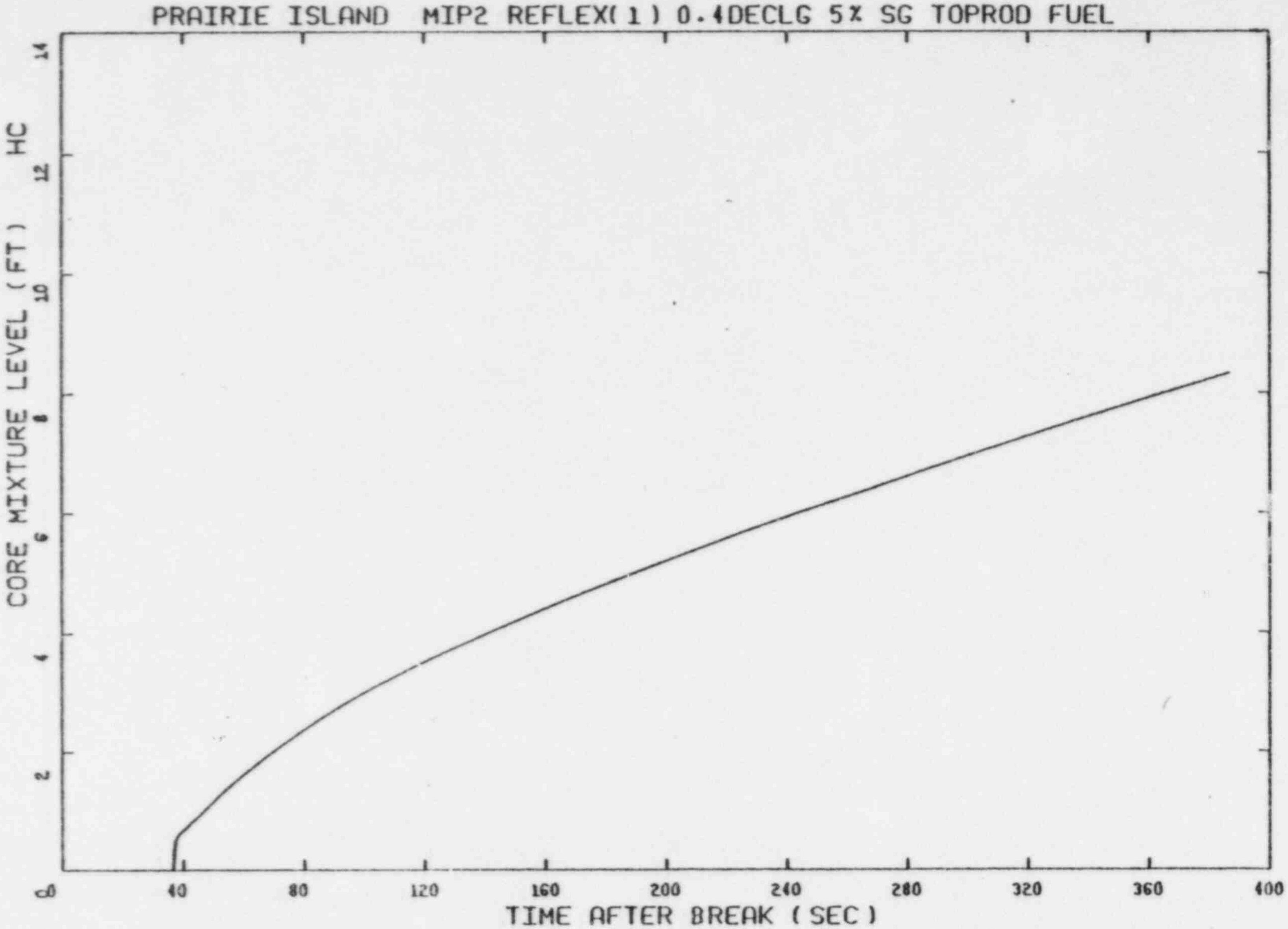

Figure 2.37 Reflood Core Mixture Level, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

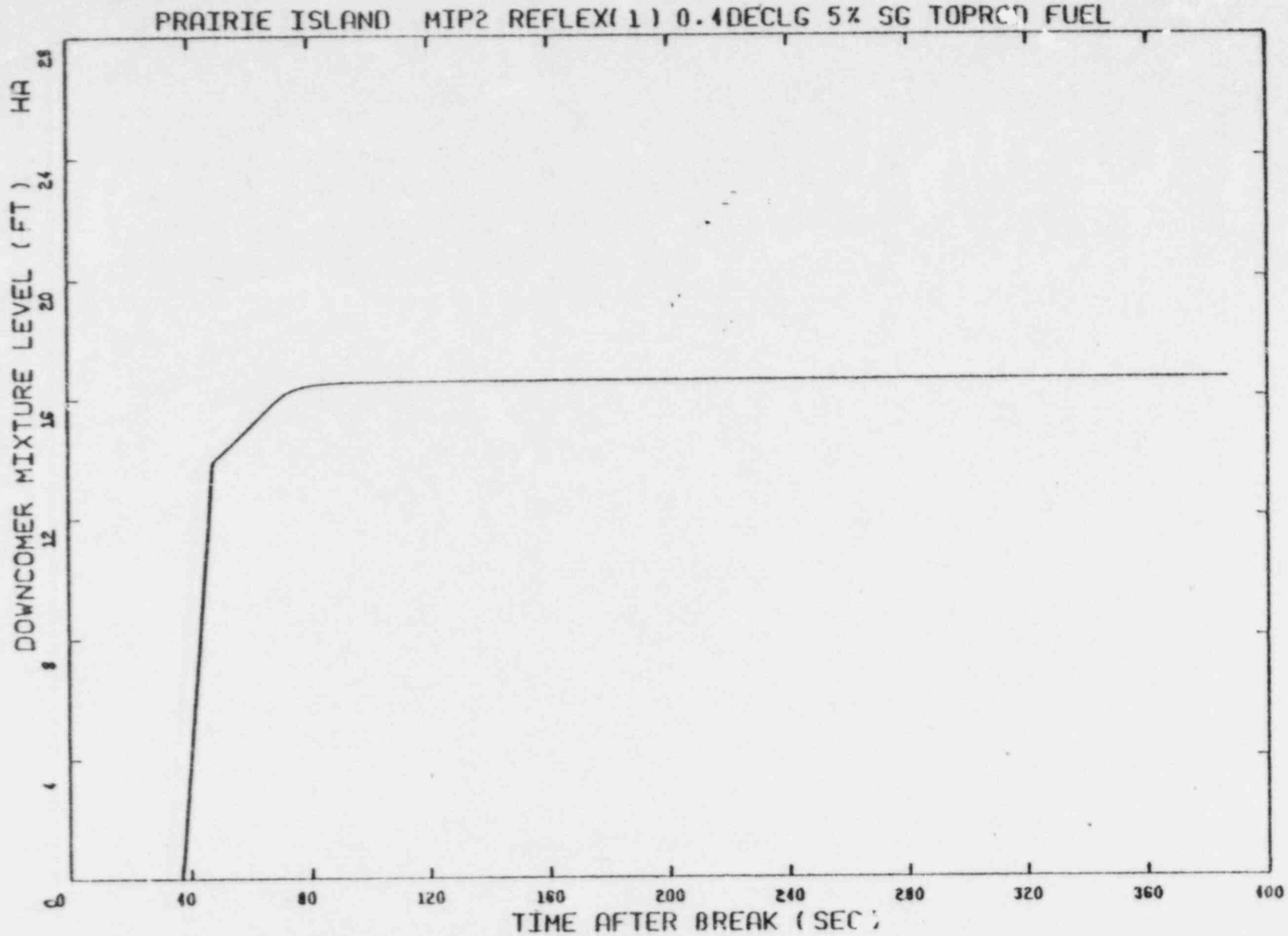

Figure 2.38 Reflood Downcomer Mixture Level, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

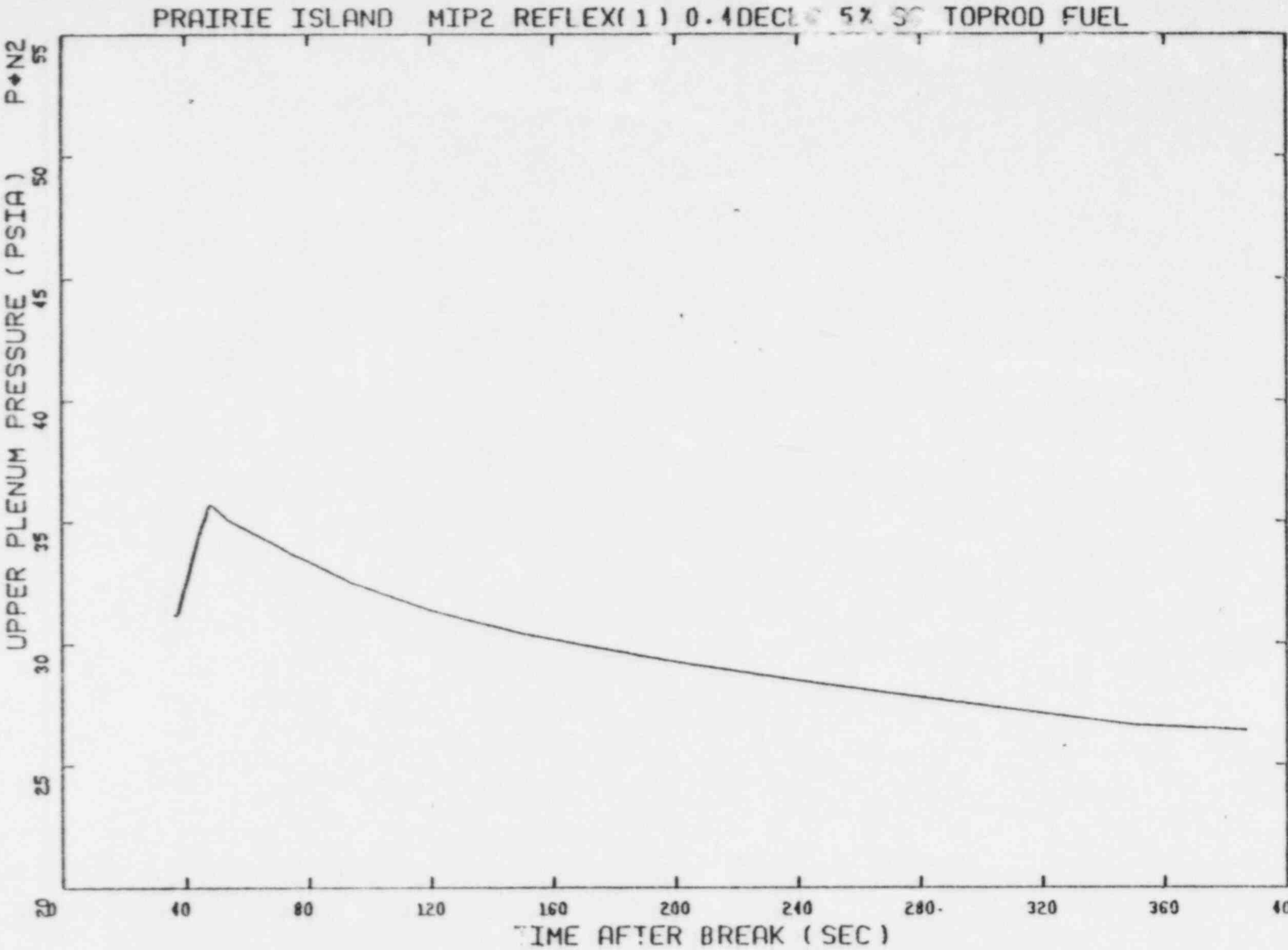

Figure 2.39 Reflood Upper Plenum Pressure, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

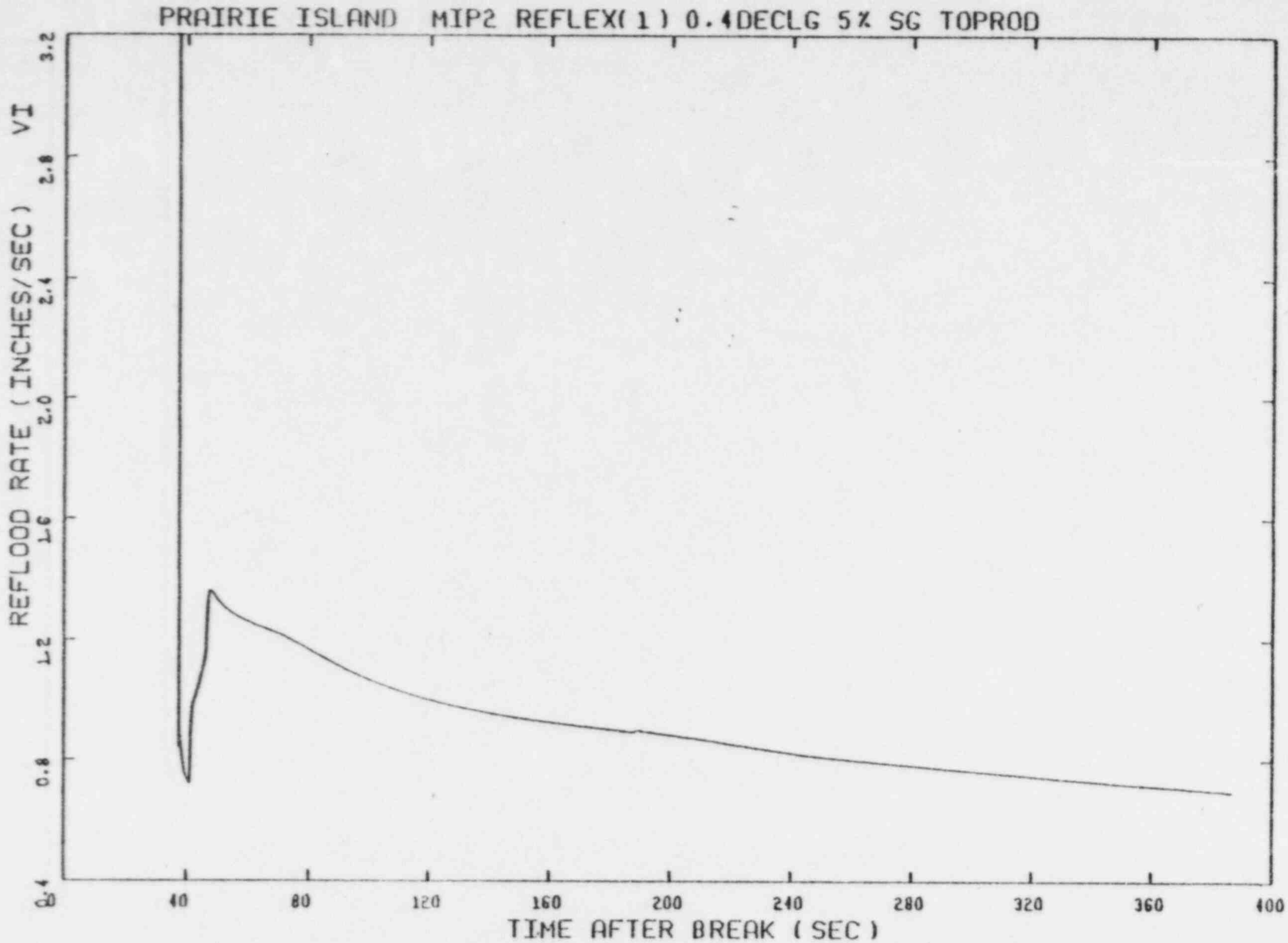

Figure 2.40 Core Flooding Rate, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

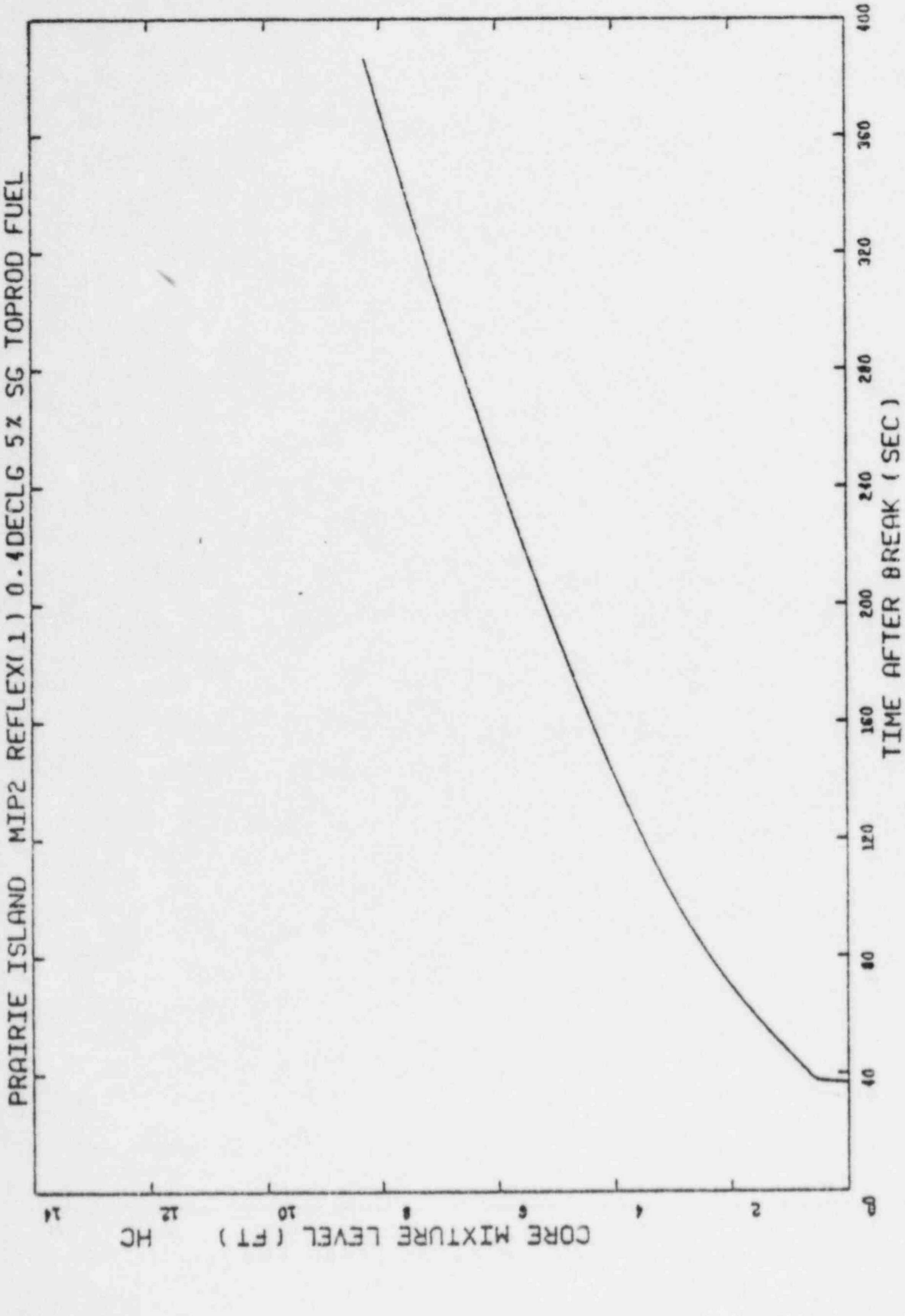

Figure 2.41 Reflood Core Mixture Level, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

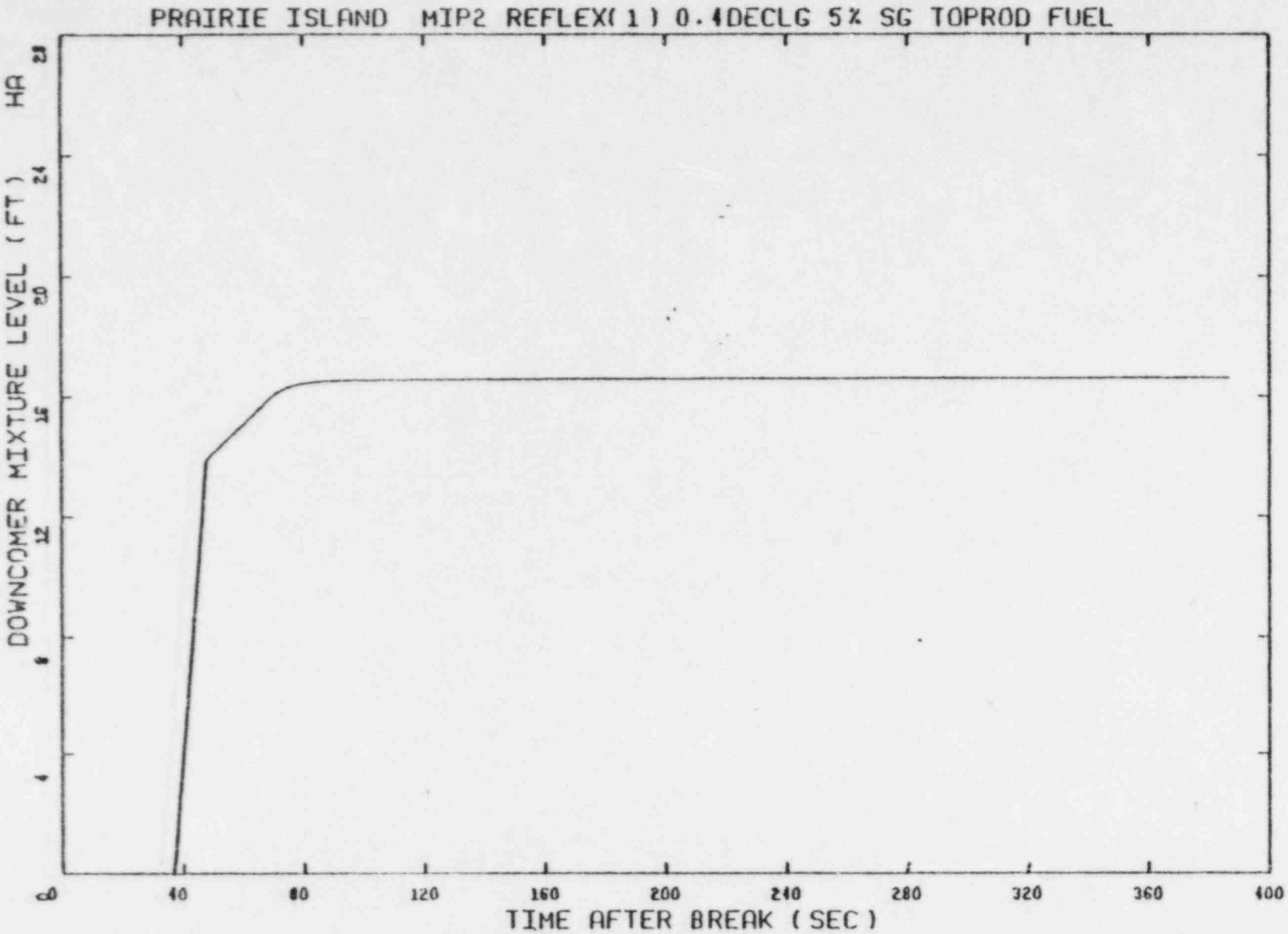

Figure 2.42 Reflood Downcomer Mixture Level, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

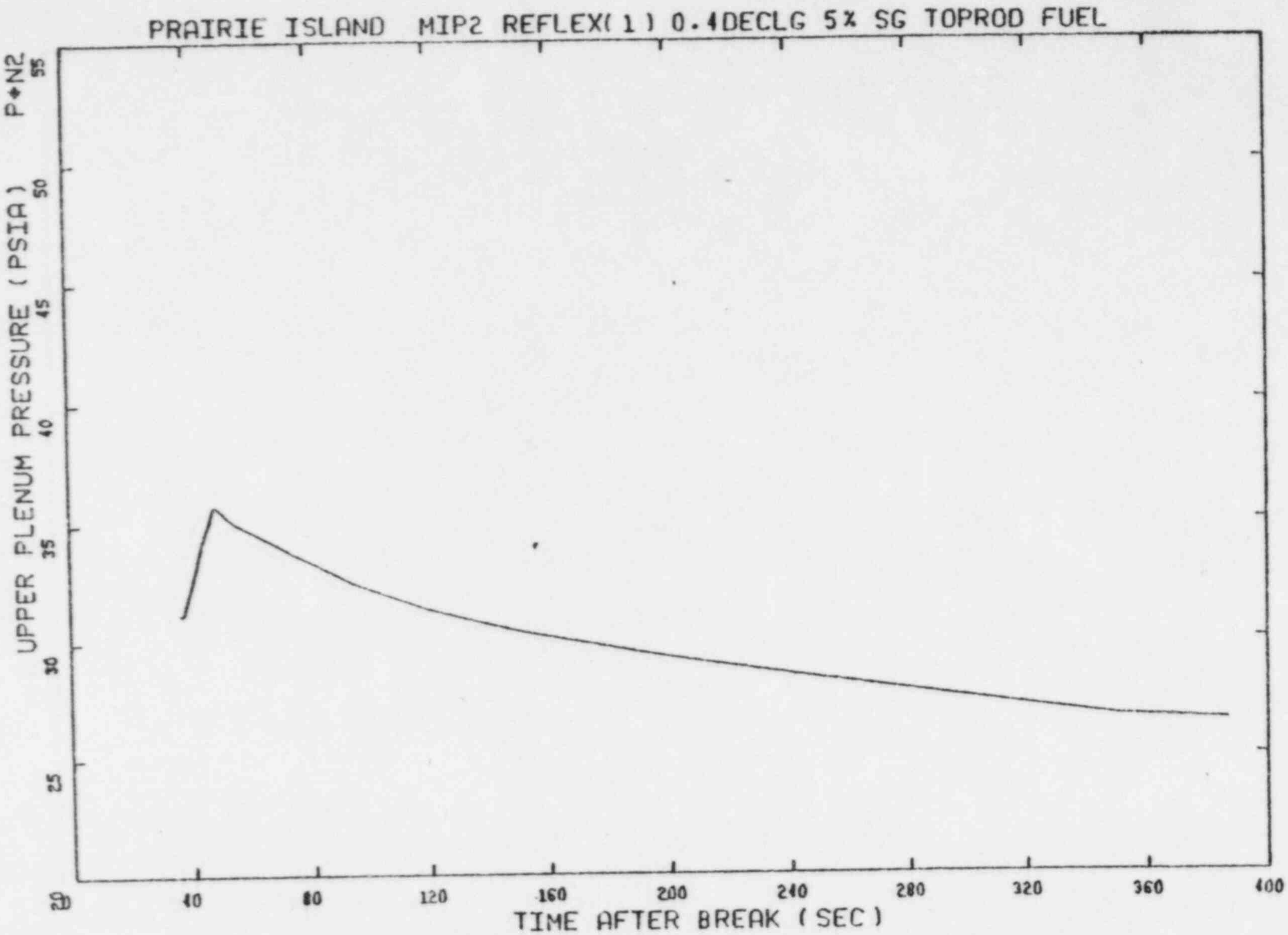

Figure 2.43 Reflood Upper Plenum Pressure, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

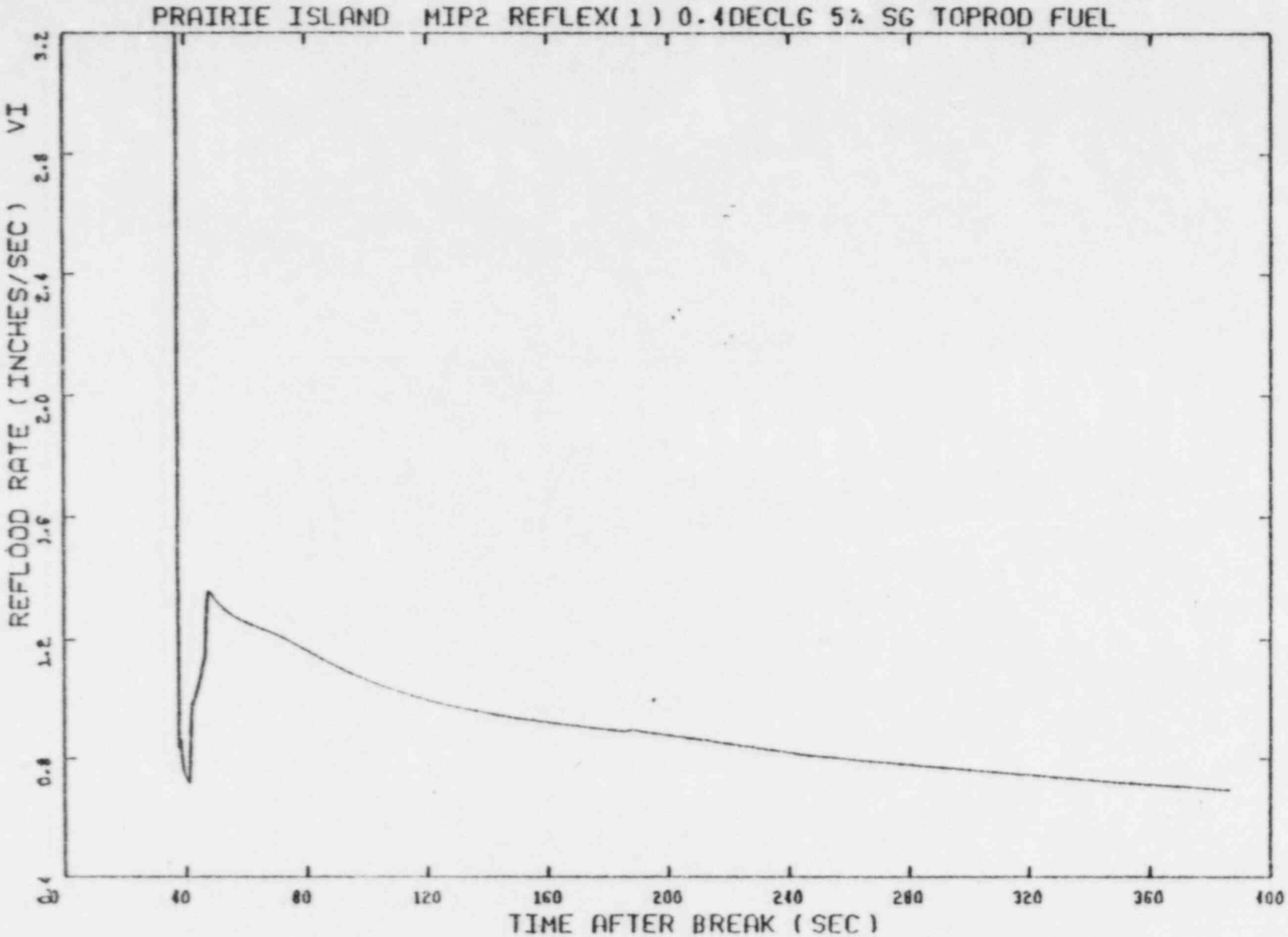

Figure 2.44 Core Flooding Rate, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

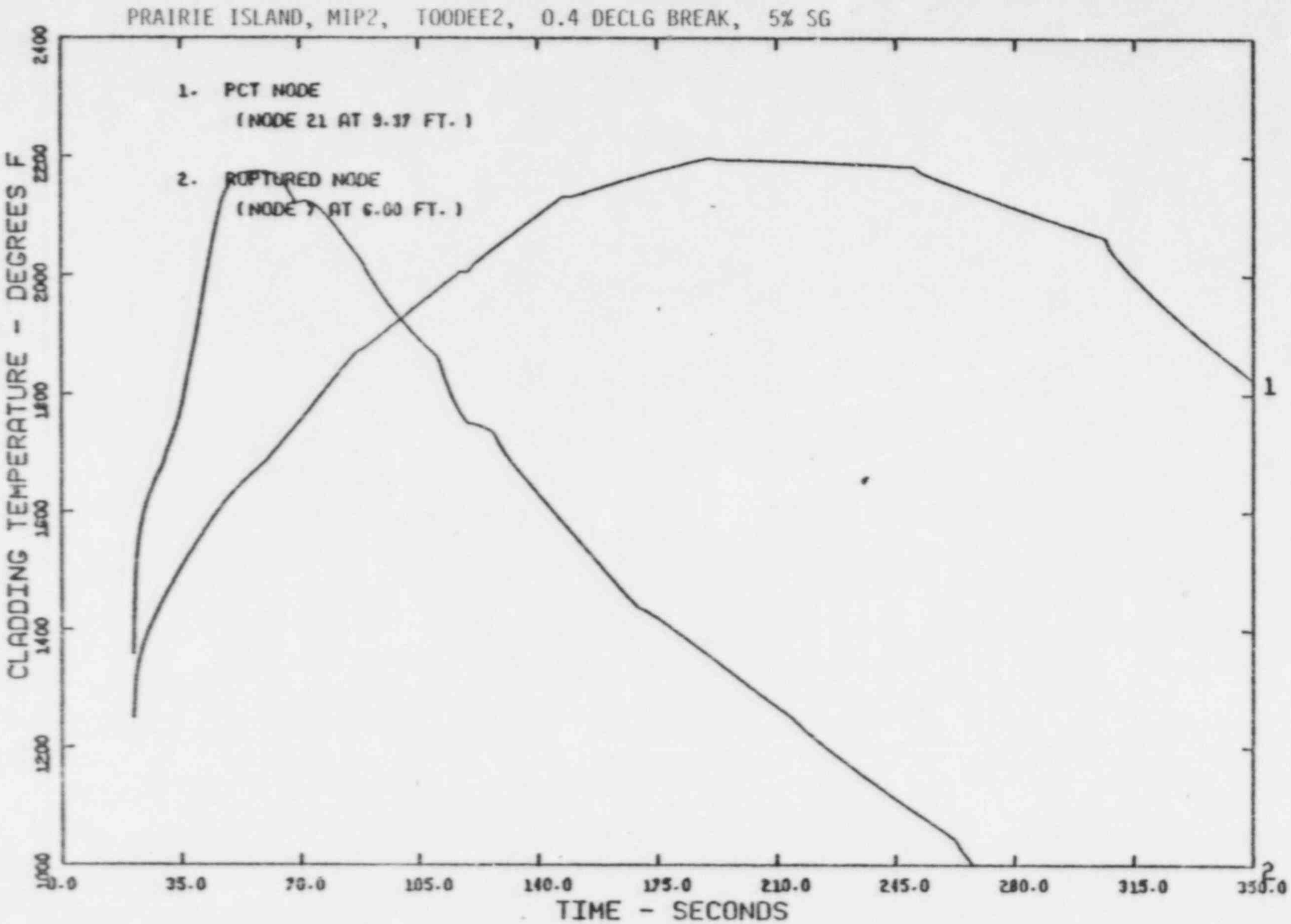

Figure 2.45 TOODEE2 Cladding Temperature vs Time, 0.4 DECLG Break, 0-15,000 MWD/MTM Case

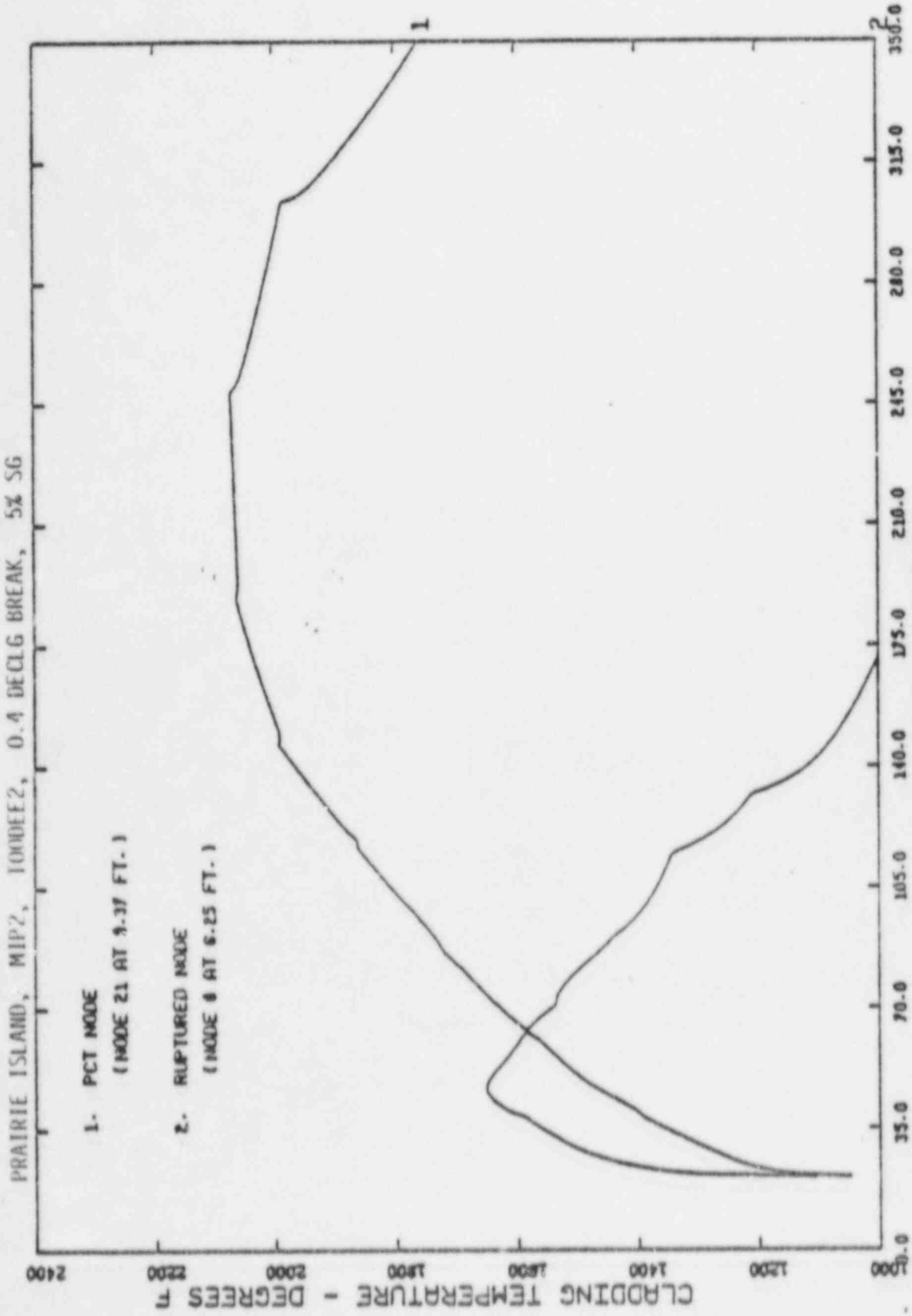

Figure 2.46 TOODEE2 Cladding Temperature vs Time, 0.4 DECLG Break, 15,000 MWD/MTM to EOL Case

3.0 CONCLUSION

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Emergency Core Cooling System for both Prairie Island units will meet the Acceptance Criteria as presented in 10 CFR 50.46, with the 2.28 F_Q^T and 1.65 $F_{\Delta H}^T$ limits. 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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