XN-NF-83-38

Issue Date: 5/27/83

# PRAIRIE ISLAND UNITS 1 AND 2
# LIMITING BREAK LOCA-ECCS ANALYSIS
# USING EXEM/PWR


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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]

The analyses were performed using the EXEM/PWR ECCS evaluation model[4], with the RODEX2 computer model for evaluating the rod stored energy and fission gas release[5]. The EXEM/PWR ECCS evaluation model includes the NRC fuel swelling and flow blockage model, NUREG-0630.[14] 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 15.02 kW/ft, corresponding to a total power peaking factor of 2.32 ($F_Q^T$), and nuclear enthalpy rise of 1.55 ($F_{\Delta H}^T$) for 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 2142°F, occurring at 196 seconds into the accident at a location 9.13 feet from the bottom of the active core, with a total metal-water reaction less than one percent. The 2142°F PCT includes a 10°F temperature addition due to the use of NRC interim upper plenum injection model[6] as modified by Westinghouse[7]. 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[8].

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), $^{o}F$ | 2091 | 2142 |
| $\Delta$PCT for UPI, $^{o}F$ | 10 | 10 |
| Time of PCT, sec. | 191 | 196 |
| Peak Clad Temperature Location, ft. | 8.88 | 9.13 |
| Local $Zr/H_2O$ Reaction (max.), %* | 4.68 | 5.60 |
| Local $Zr/H_2O$ Location, ft. from bottom | 8.63 | 9.13 |
| Total $H_2$ Generation, % of total Zr Reacted | <1.0 | <1.0 |
| Hot Rod Burst Time, sec. | 30.19 | 30.19 |
| Hot Rod Burst Location, ft. | 6.0 | 6.0 |

| Calculational Basis | |
|---|---|
| License Core Power, MWt | 1650 |
| Power Used for Analysis, MWt** | 1683 |
| Peak Linear Power for Analysis, kW/ft** | 15.02 |
| Total Peaking Factor, $F_Q^T$ | 2.32 |
| Enthalpy Rise, Nuclear, $F_{\Delta H}^T$ | 1.55 |
| Steam Generator Tube Plugging (%) | 5.00 |

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* 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[9], and the Emergency Core Cooling System Evaluation Model Updates: WREM-II[17], WREM-IIA[13] and EXEM/PWR[4].

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[5] in place of the previously applied GAPEX[10] model.

(2) The NRC upper plenum injection (UPI) interim model, developed by the NRC Staff[6] and modified by Westinghouse[7].

(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[4].

### 2.1 LOCA ANALYSIS MODEL

The Exxon Nuclear Company EXEM/PWR ECCS evaluation model[4] was used to perform the analyses required. This model consists of the following computer codes: RODEX2[5] code for initial rod stored energy and internal fuel rod gas inventory; RELAP4-EM[11] for the system blowdown and hot channel blowdown calculations; CONTEMPT-LT/22 as modified in CSB 6-1[16] for computation of containment backpressure; REFLEX[4,14] for computation of system reflood; and TOODEE2[4,14,15] 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)(16). The system nodalization is depicted in Figure 2.1. The unbroken loops were assumed symmetrical and modeled as one intact loop with appropriately scaled input. The pump performance characteristic curves are supplied by the NSSS vendor. Five percent of the steam generator tubes are assumed to be uniformly plugged. 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 1.453, corresponding to a total peaking factor of 2.32, and $F^T_{\Delta H}$ of 1.55. This axial power profile in configuration with the current K(Z) function developed by the NSSS vendor will not be used to define operating envelop for $F_Q$. 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 were highest at EOL (second case) for the hot rod, while stored energy was calculated to be highest at lower exposure (first case). 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. In addition, 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 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 2.3 through 2.9 present plotted results for system blowdown analysis. Unless otherwise noted on the figures, time zero corresponds to the time of break initiations. Figure 2.10 presents calculated containment backpressure time history. Figures 2.11 through 2.22 present results for the hot channel blowdown calculations. Figures 2.23 and 2.24 show the normalized power calculation results. The reflood calculation results are shown in Figures 2.25 through 2.32.

The maximum peak cladding temperature (PCT) calculated for the 0.4 DECLG break at the EOL is 2142°F (Figure 2.34). This value includes a 10°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 15.02 kW/ft ($F_Q^T$=2.32) for ENC TOPROD fuel. The maximum local metal-water reaction in this case is 5.60% after 380 seconds, and the total core metal-water reaction is less than 1%. The PCT location is at an elevation of 9.13 feet from the bottom of active core. For the exposure up to 15,000 MWD/MTM, the PCT is 2091°F (Figure 2.33) including a 10°F for UPI effect, occurring at 8.88 feet elevation relative to the bottom of the active core. The local metal-water reaction is 4.68%, 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 | $6.82 \times 10^7$ |
| Primary Coolant Volume, $ft^3$ | 10,309.** |
| 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 | $3.54 \times 10^6$ |
| 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, Intact Loop | 8.70 |
| Accumulator Injection, Broken Loop | 4.80 |
| End-of-Bypass | 21.14 |
| Safety Injection Flow | 25.60 |
| Start of Reflood | 36.89 |
| Accumulator Empties, Intact Loop | 44.09 |
| Peak Clad Temperature Reached | 196.00 |

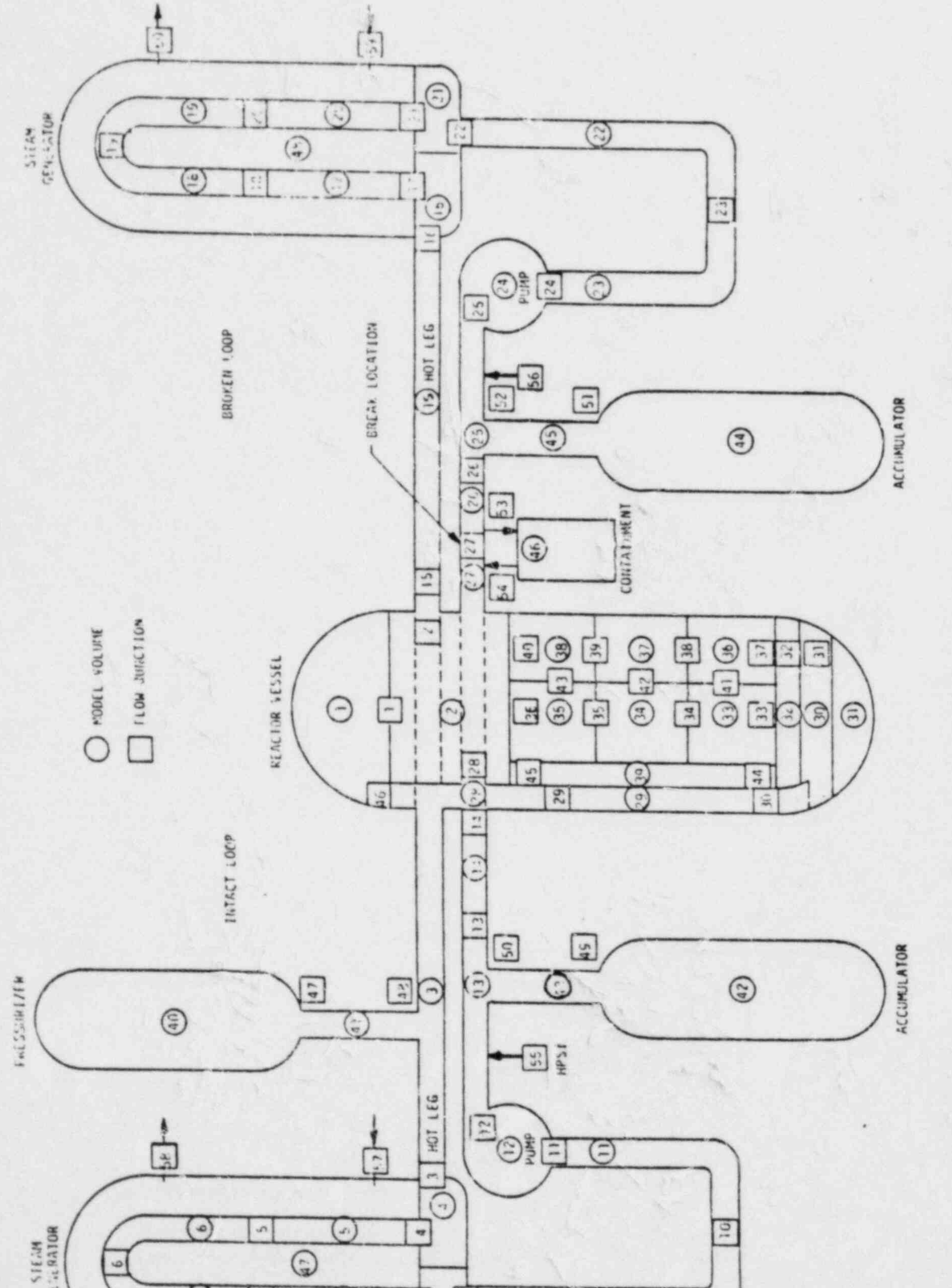


FIGURE 2.1 RELAP4/EM BLOWDOWN SYSTEM NODALIZATION FOR PRAIRIE ISLAND UNIT 1 AND 2

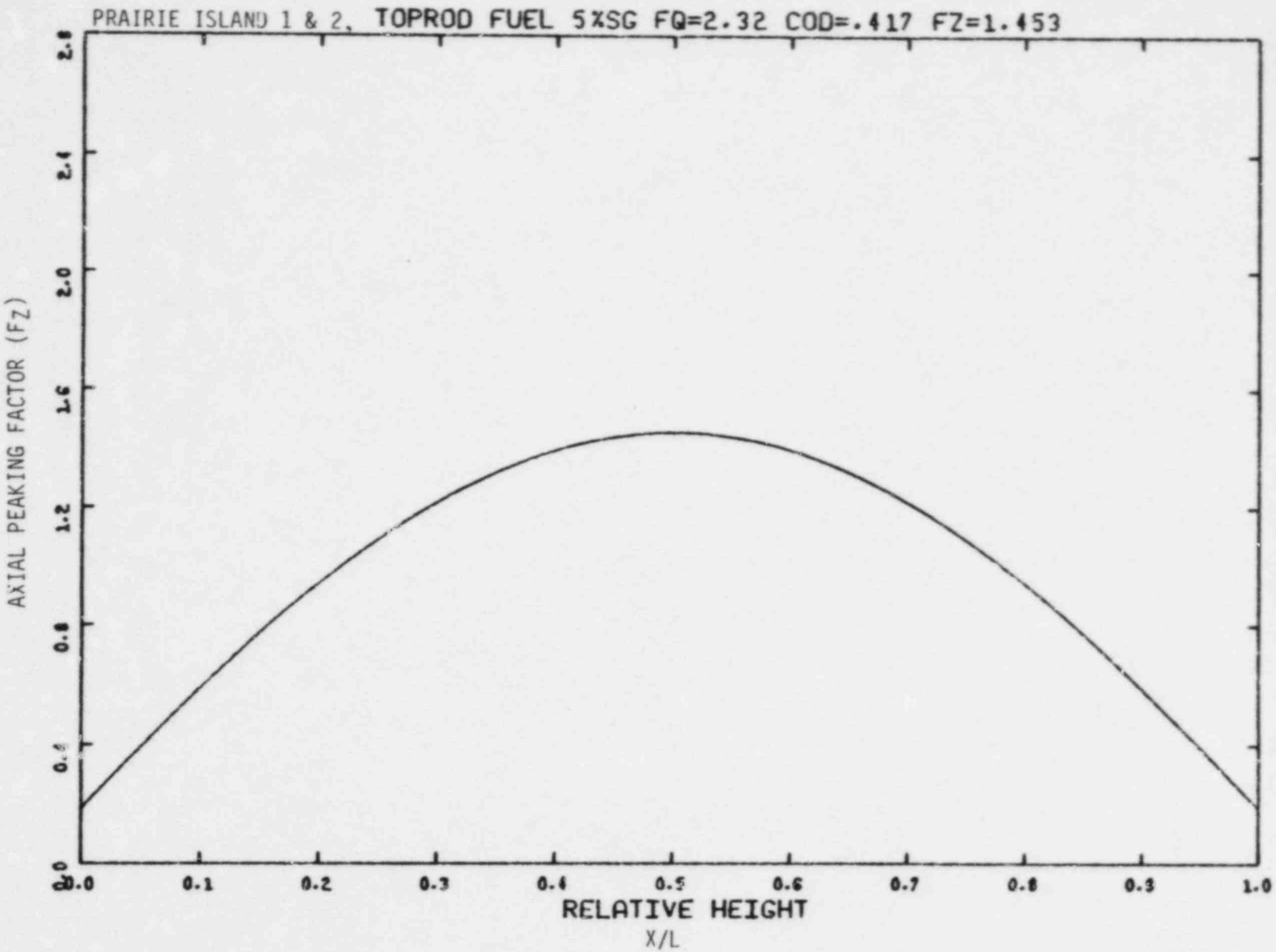


Figure 2.2 Axial Peaking Factor versus Rod Length, 0.4 DECLG Break

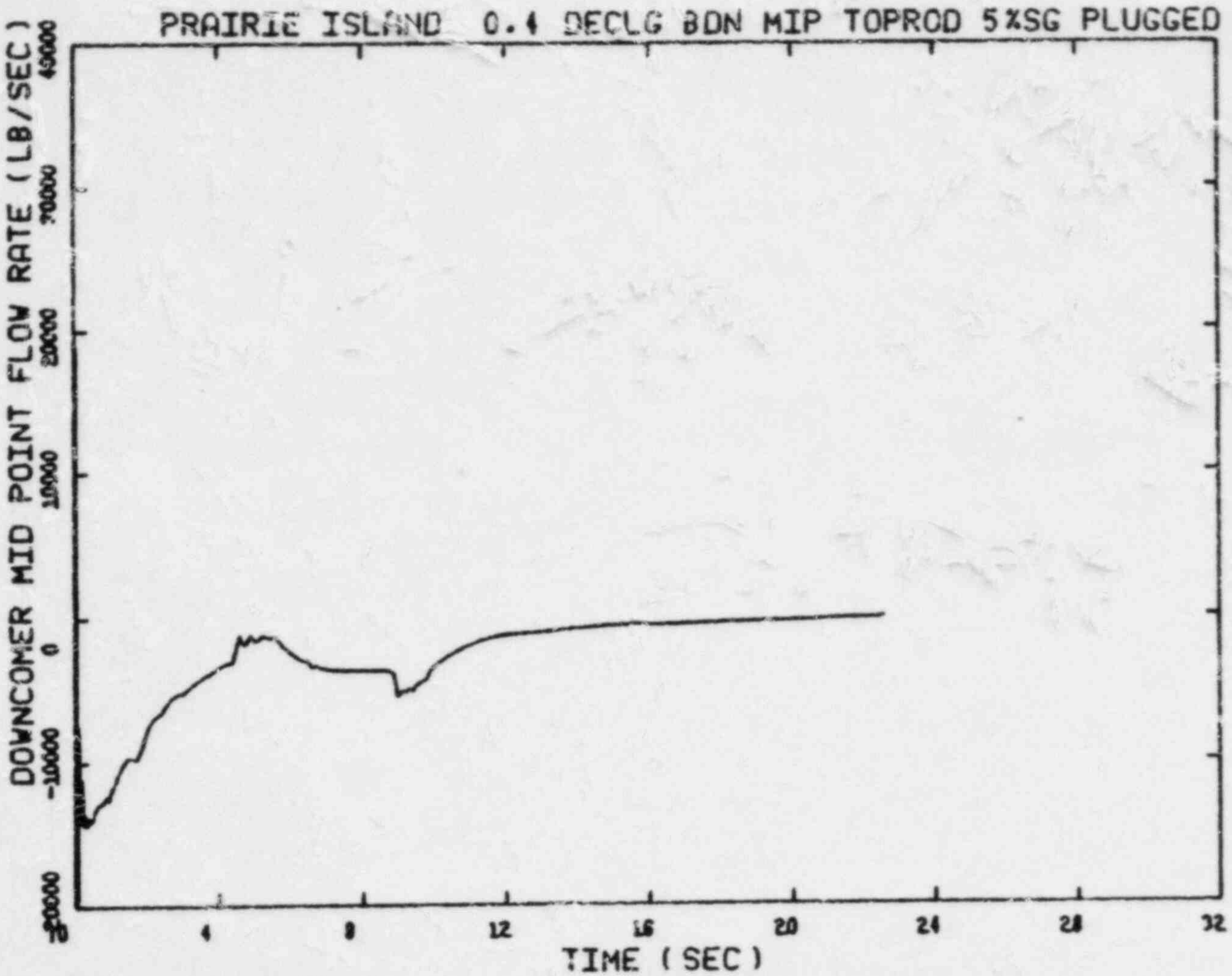


Figure 2.3 Downcomer Flow Rate, 0.4 DECLG Break

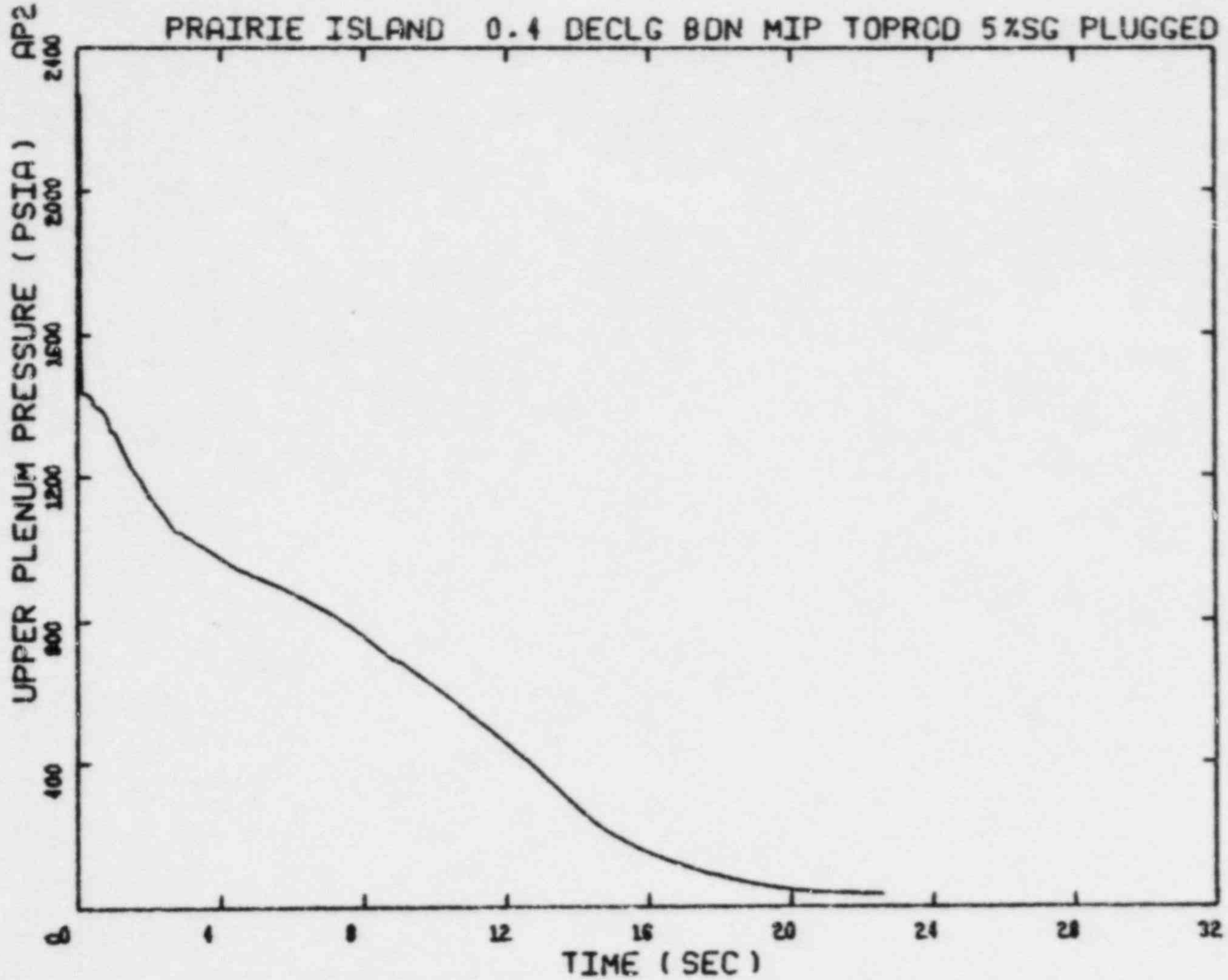


Figure 2.4 Upper Plenum Pressure, 0.4 DECLG Break

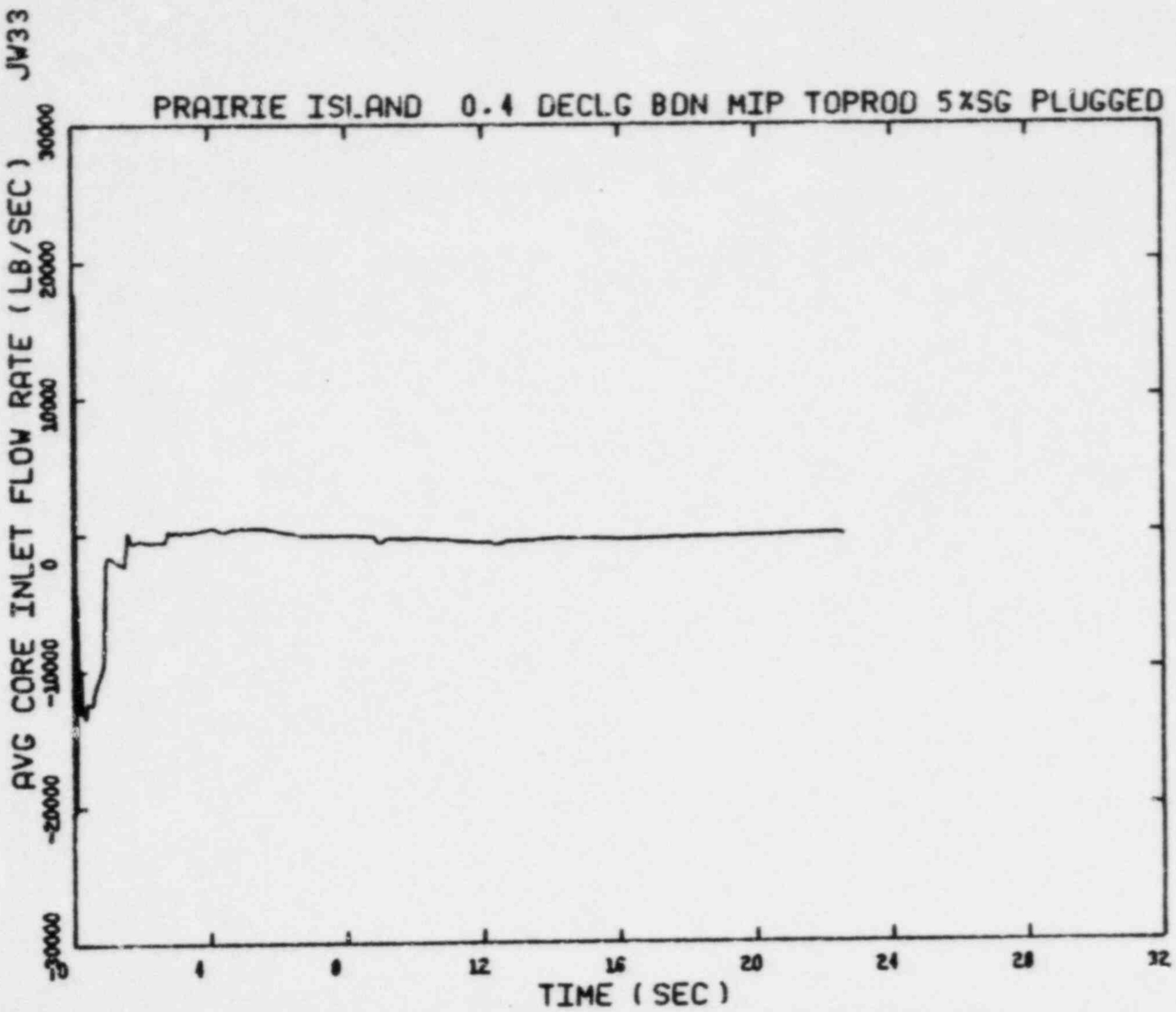


Figure 2.5 Average Core Inlet Flow, 0.4 DECLG Break

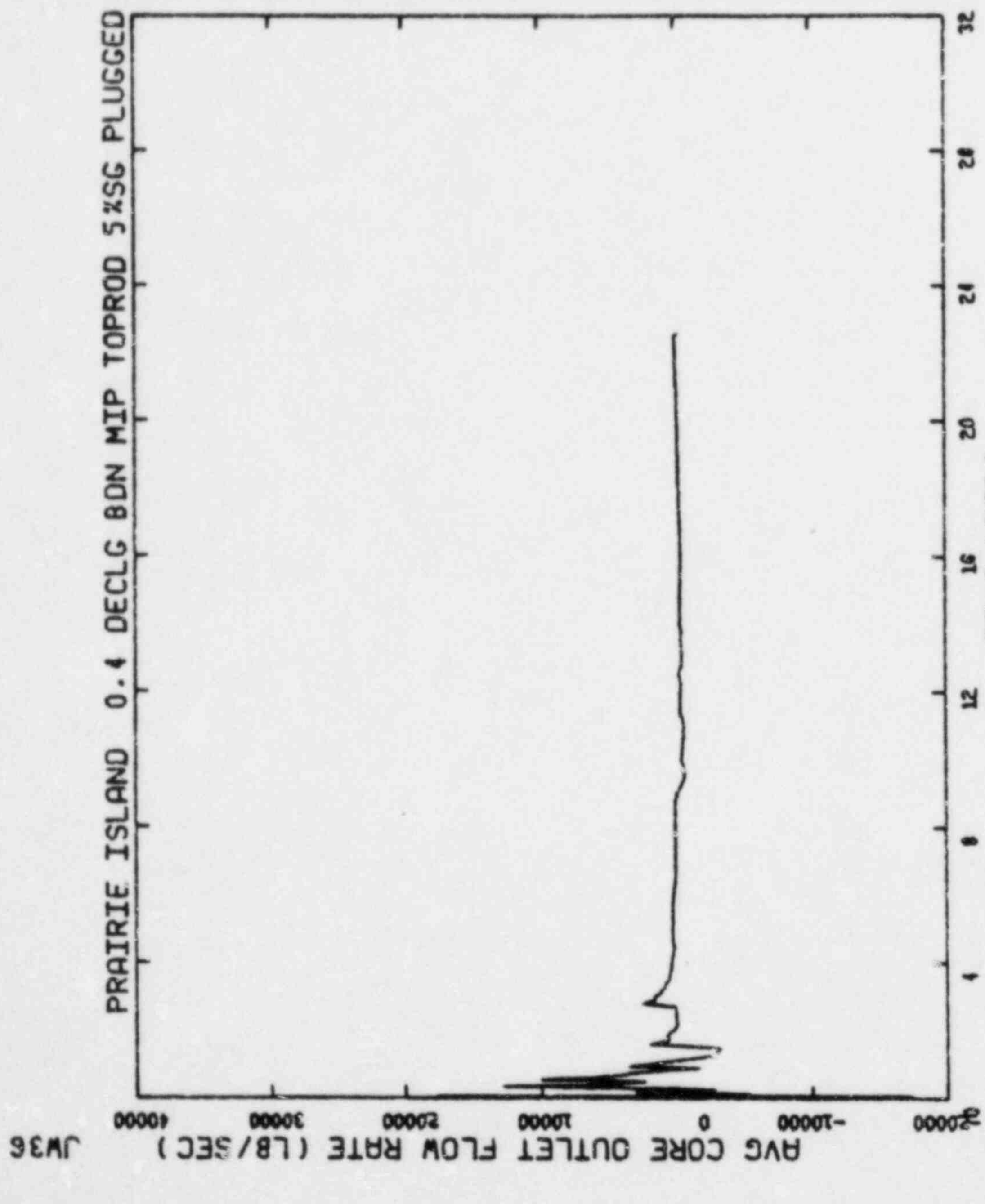


Figure 2.6 Average Core Outlet Flow, 0.4 DECLG Break

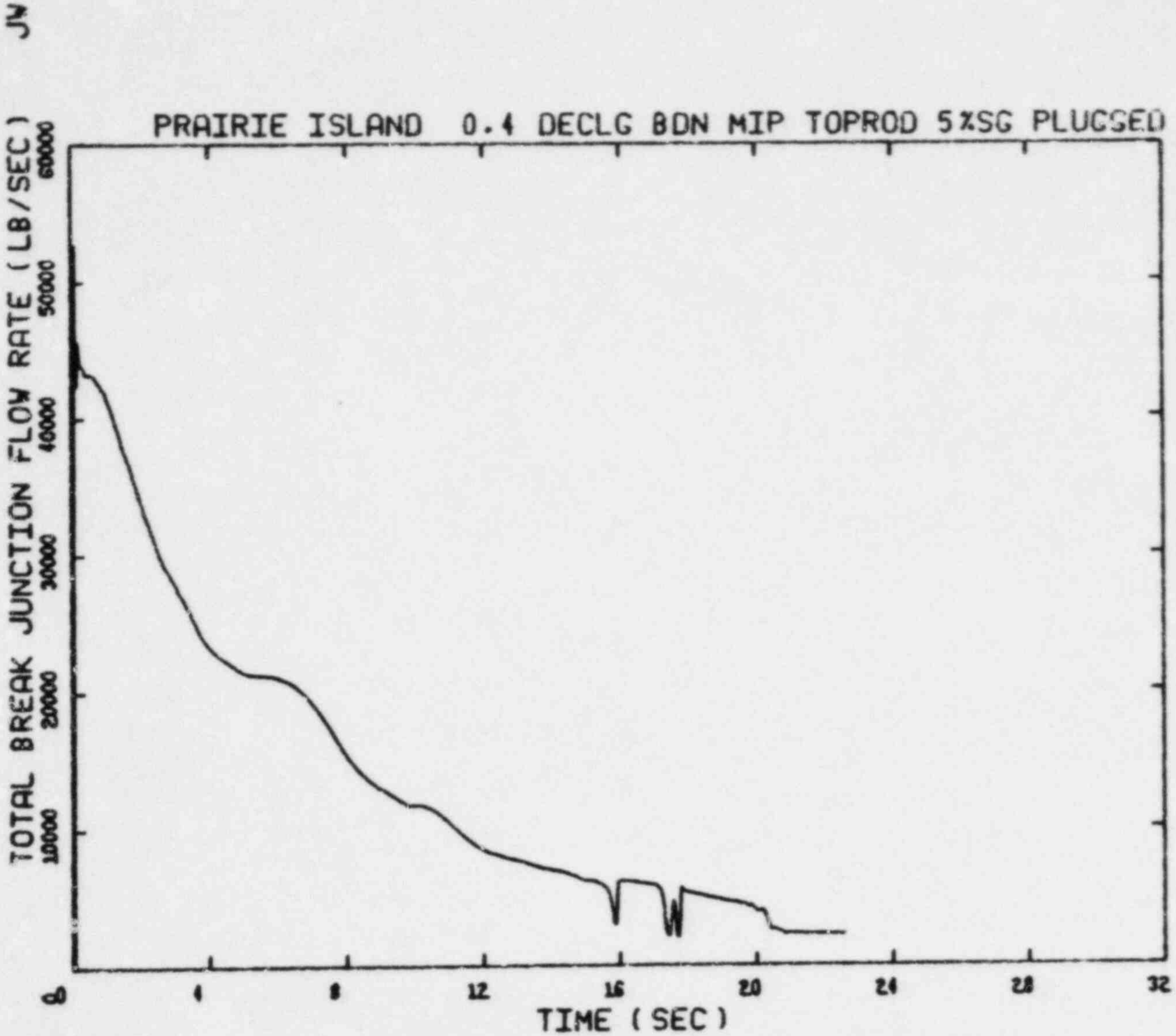


Figure 2.7 Total Break Flow, 0.4 DECLG Break

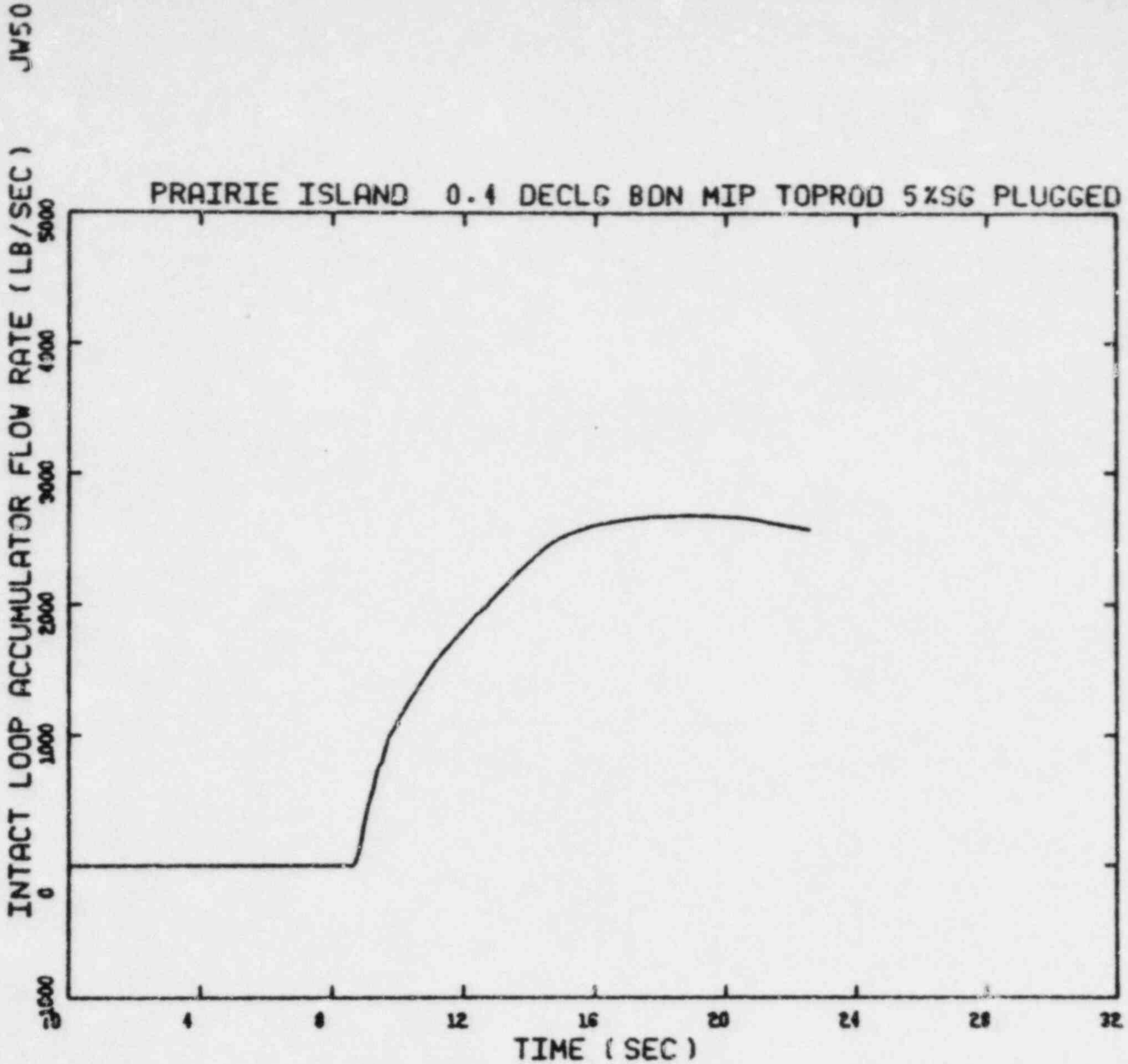


Figure 2.8 Flow from Intact Loop Accumulator, 0.4 DECLG Break

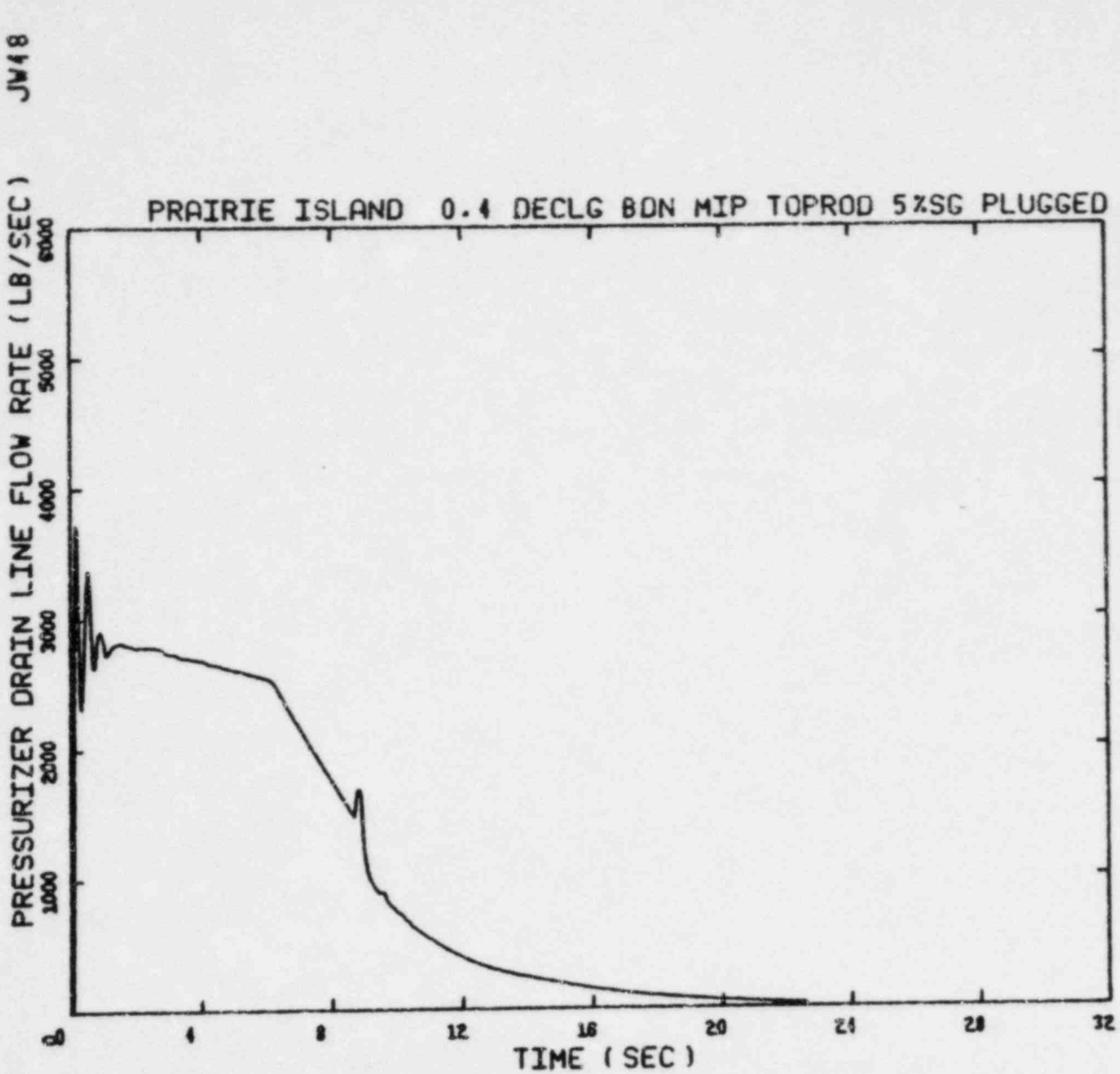


Figure 2.9 Pressurizer Surge Line Flow, 0.4 DECLG Break

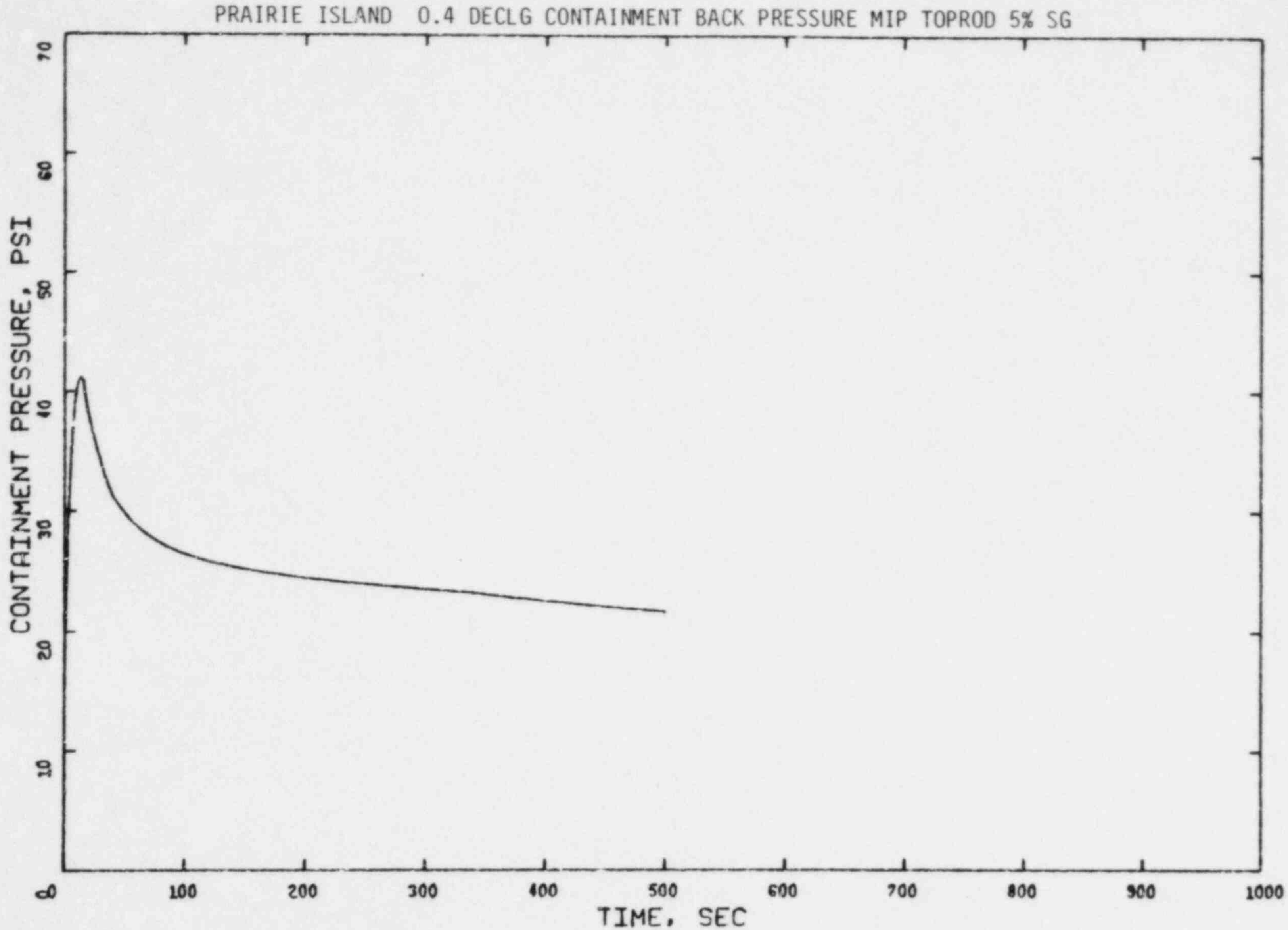


Figure 2.10 Containment Back Pressure, 0.4 DECLG Break

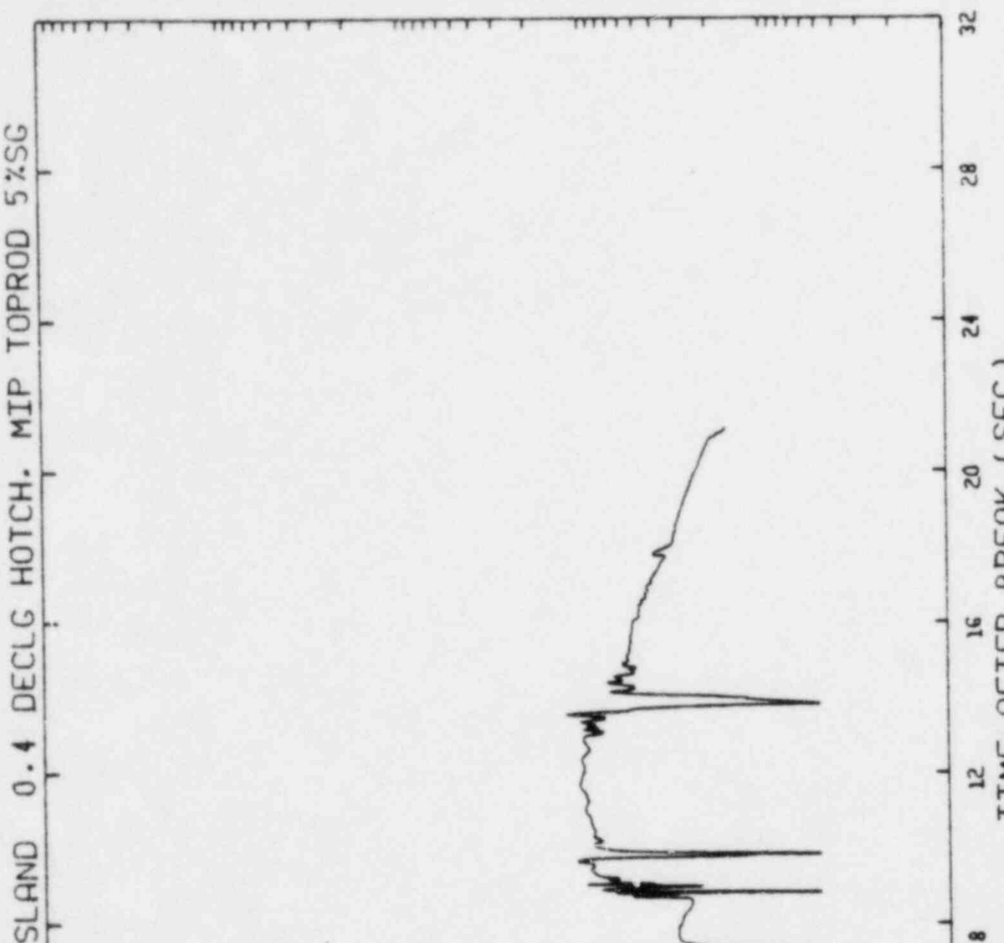


Figure 2.11 Hot Channel Heat Transfer Coefficient, 0.4 DECLG Break, 0-15,000 MWD/MTM

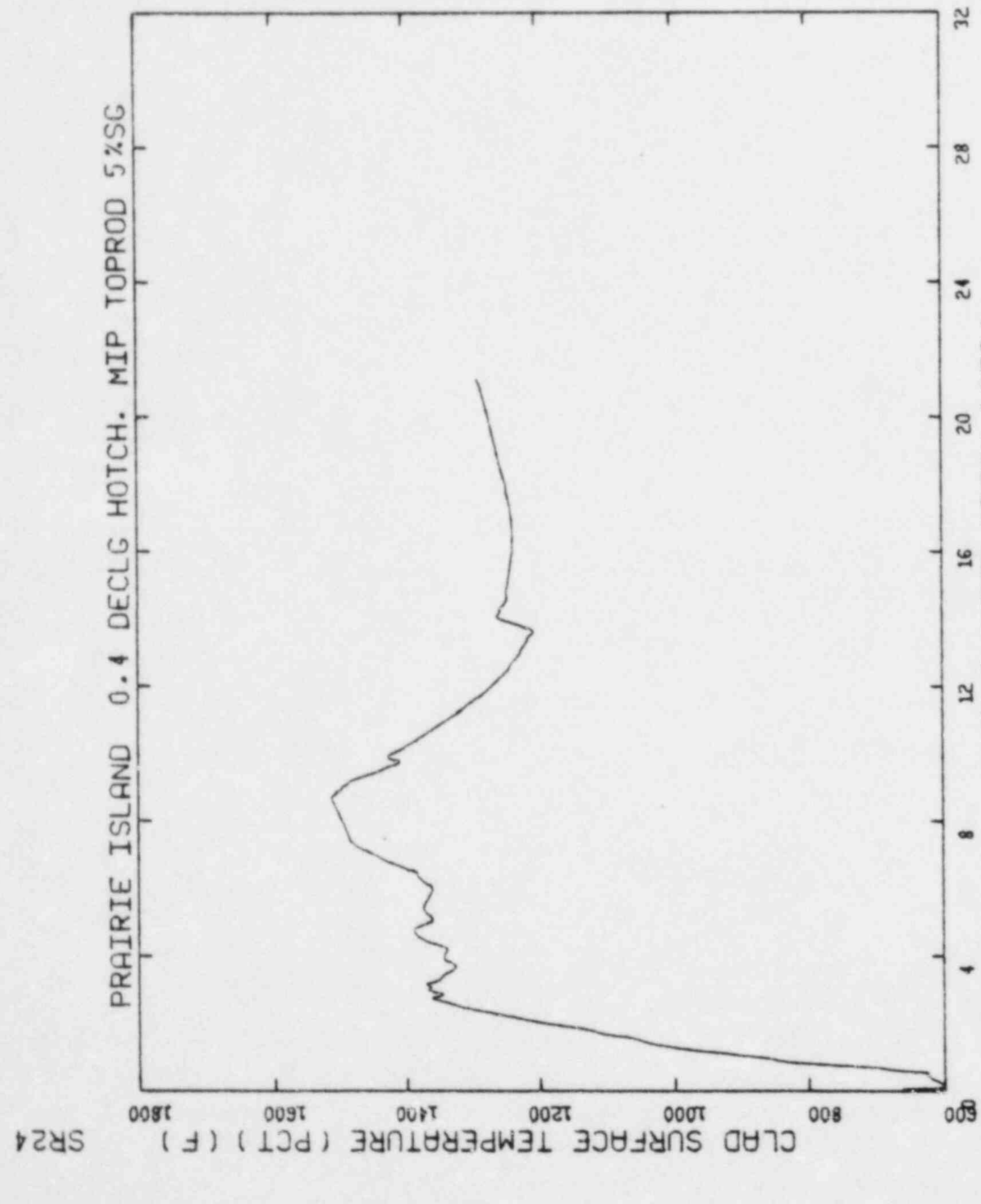


Figure 2.12 Clad Surface Temperature, 0.4 DECLG Break, 0-15,000 MWD/MTM

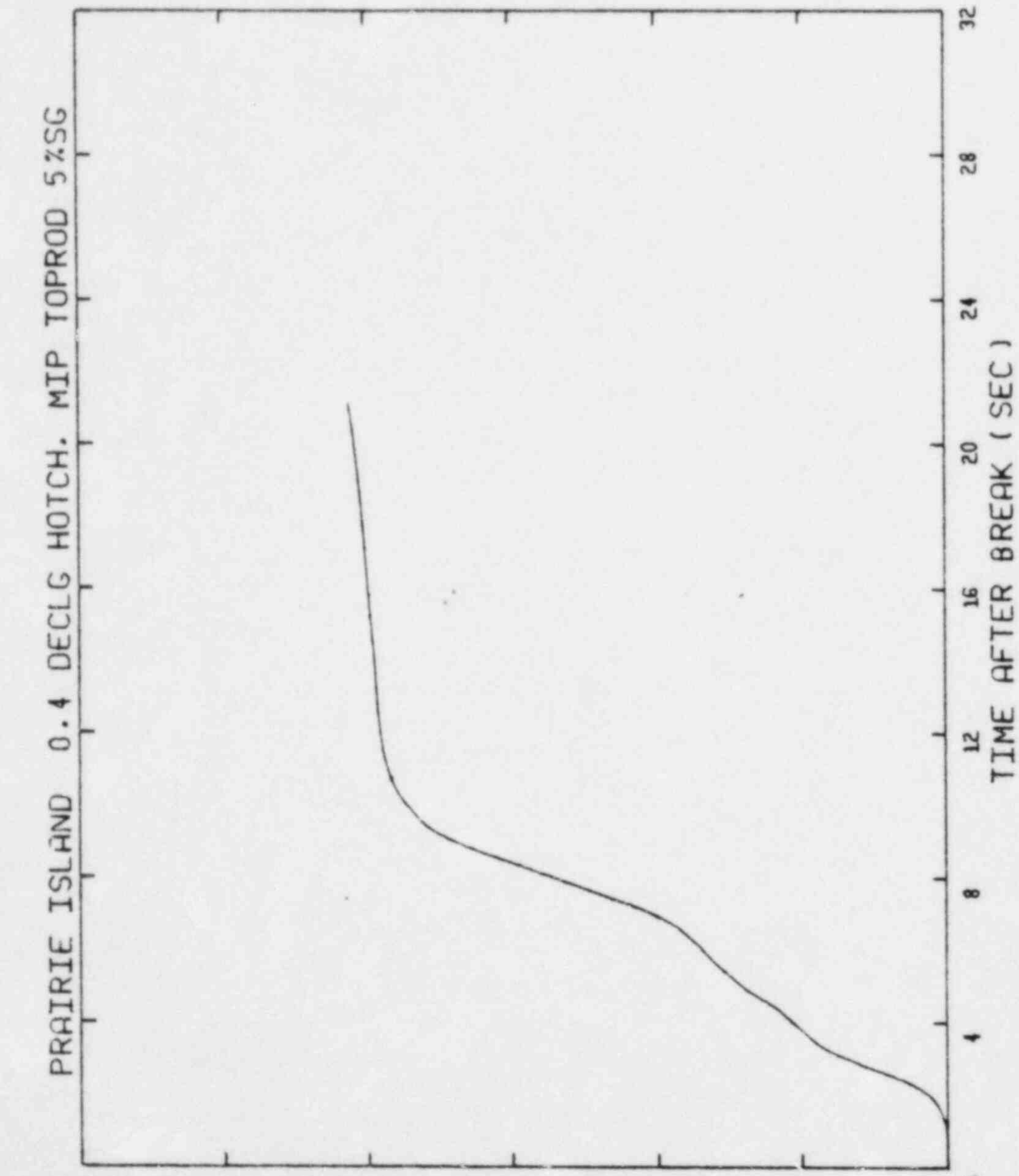


Figure 2.13 Depth of Metal-Water Reaction, 0.4 DECLG Break, 0-15,000 MWD/MTM

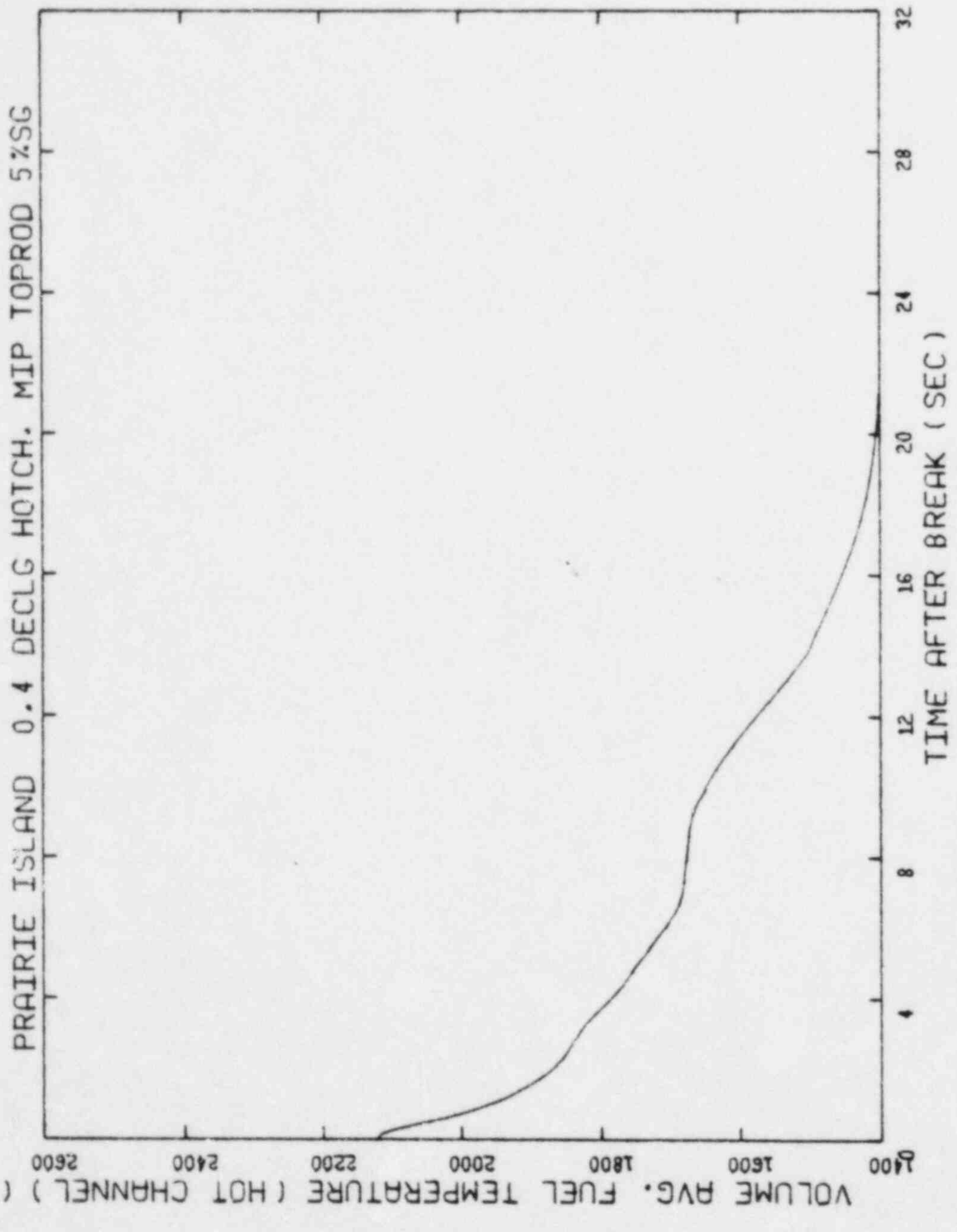


Figure 2.14 Hot Channel Average Fuel Temperature, 0.4 DECLG Break, 0-15,000 MWD/MTM

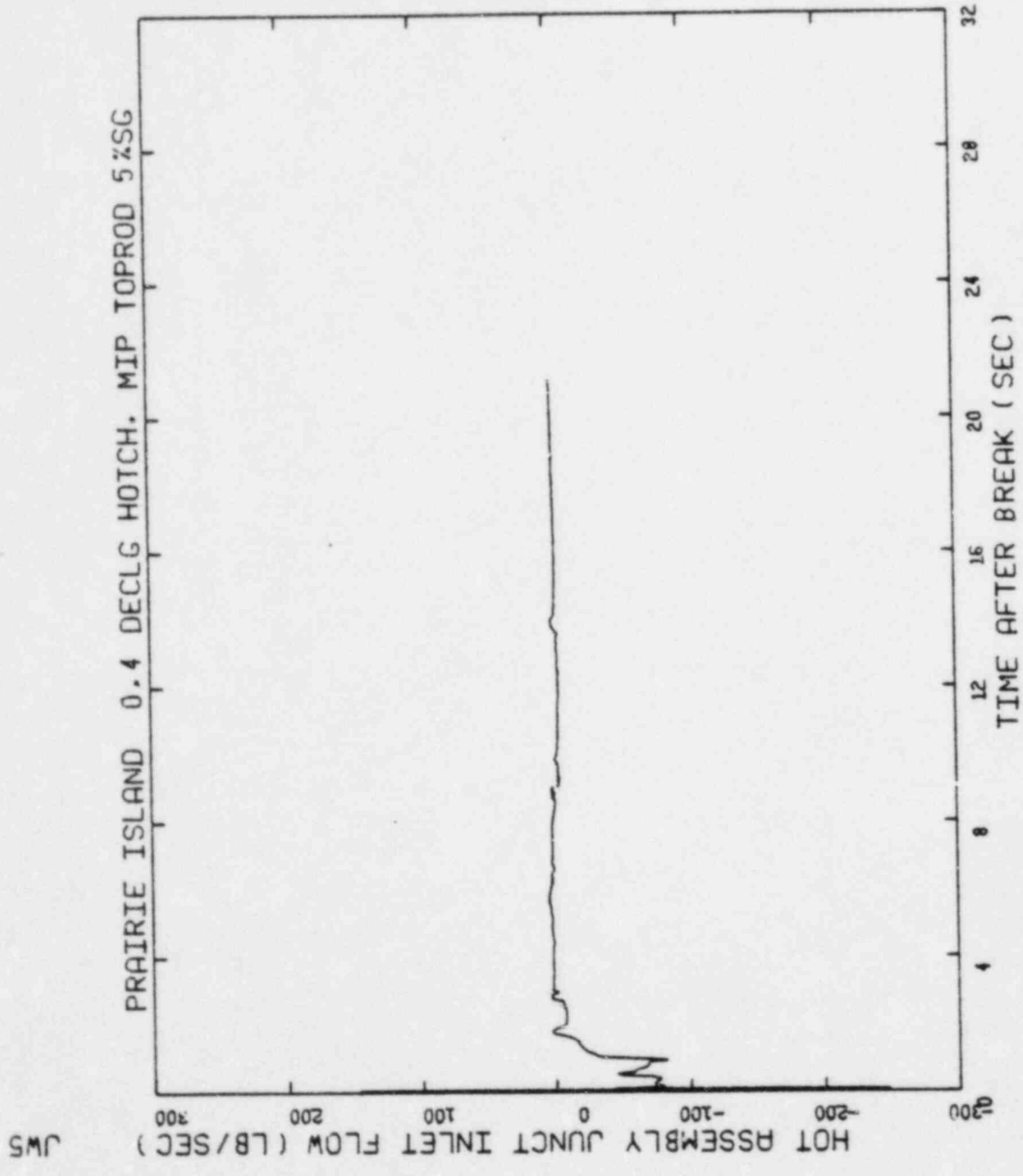


Figure 2.15 Hot Assembly Inlet Flow, 0.4 DECLG Break, 0-15,000 MWD/MTM

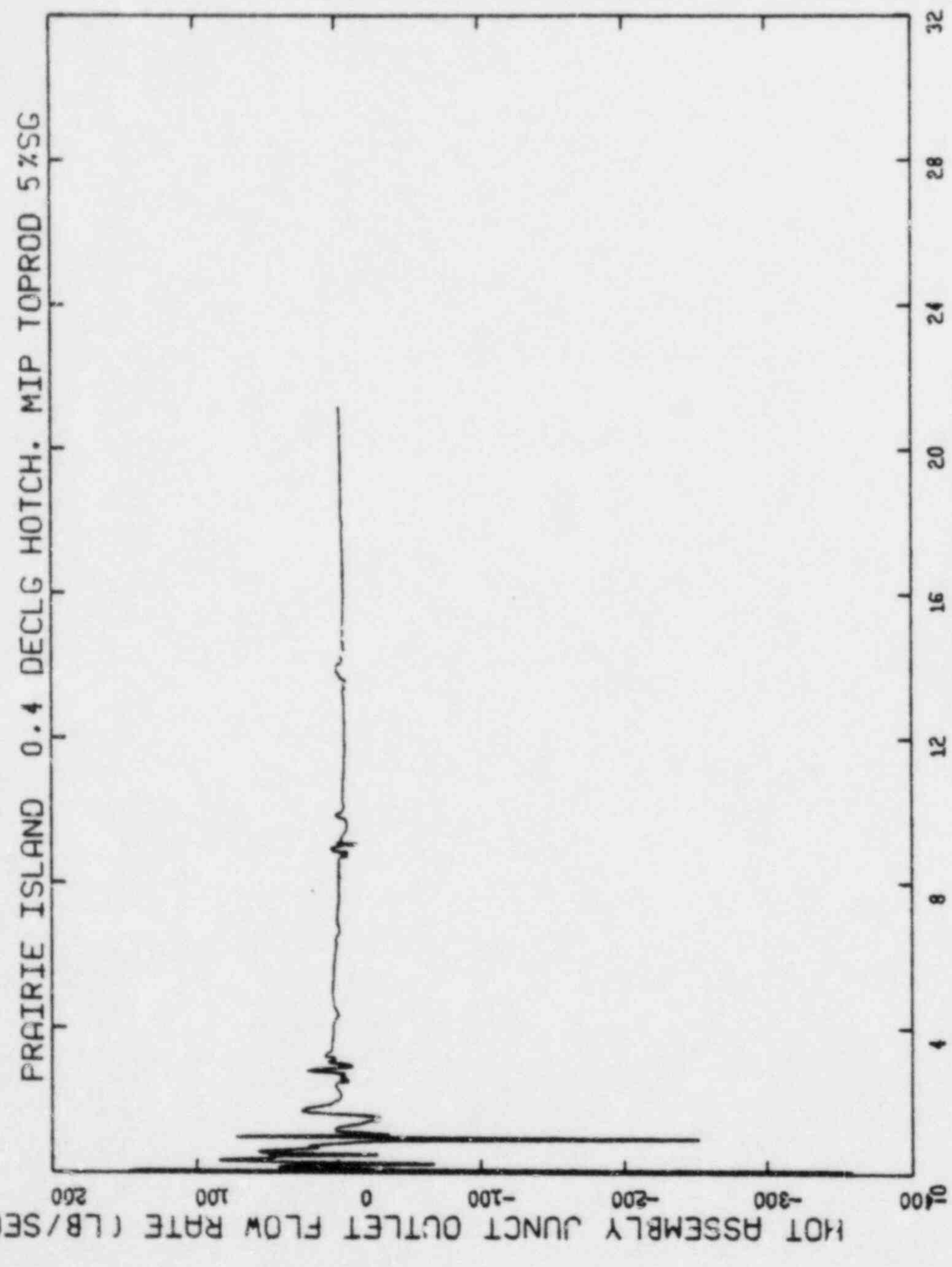


Figure 2.16 Hot Assembly Outlet Flow, 0.4 DECLG Break, 0-15,000 MWD/MTM

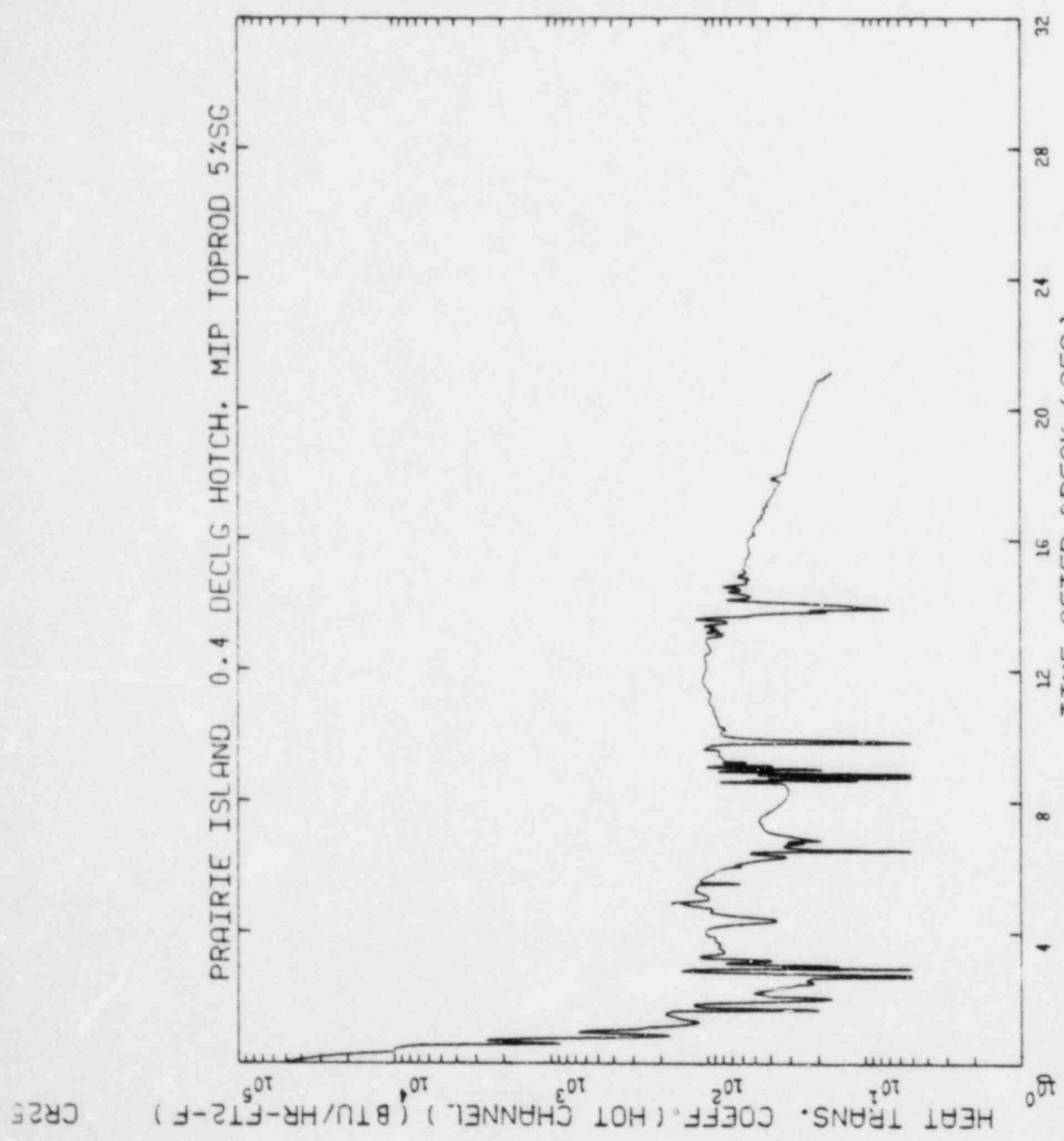


Figure 2.17 Hot Channel Heat Transfer Coefficient, 0.4 DECLG Break, EOL

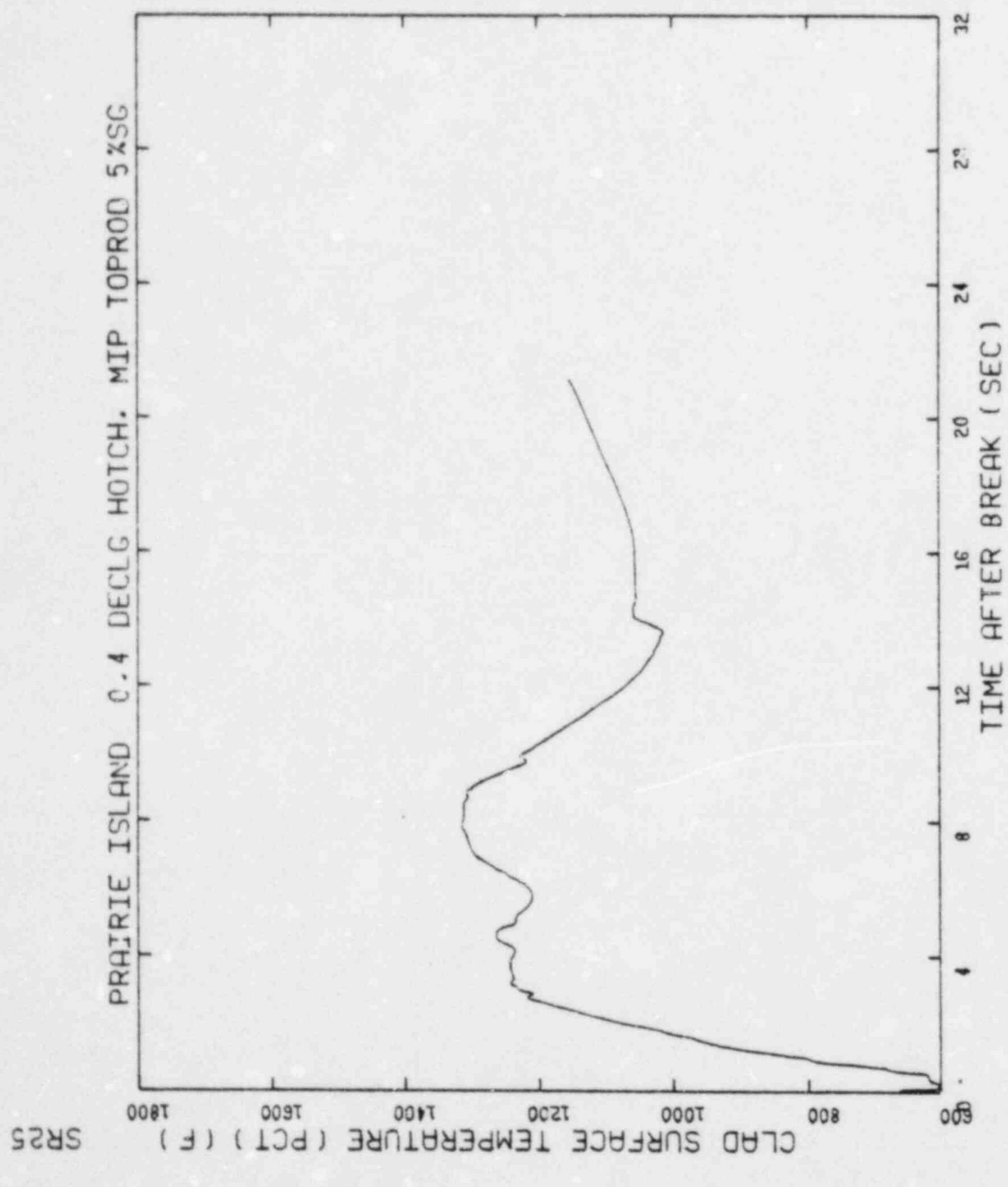


Figure 2.18 Clad Surface Temperature, 0.4 DECLG Break, EOL

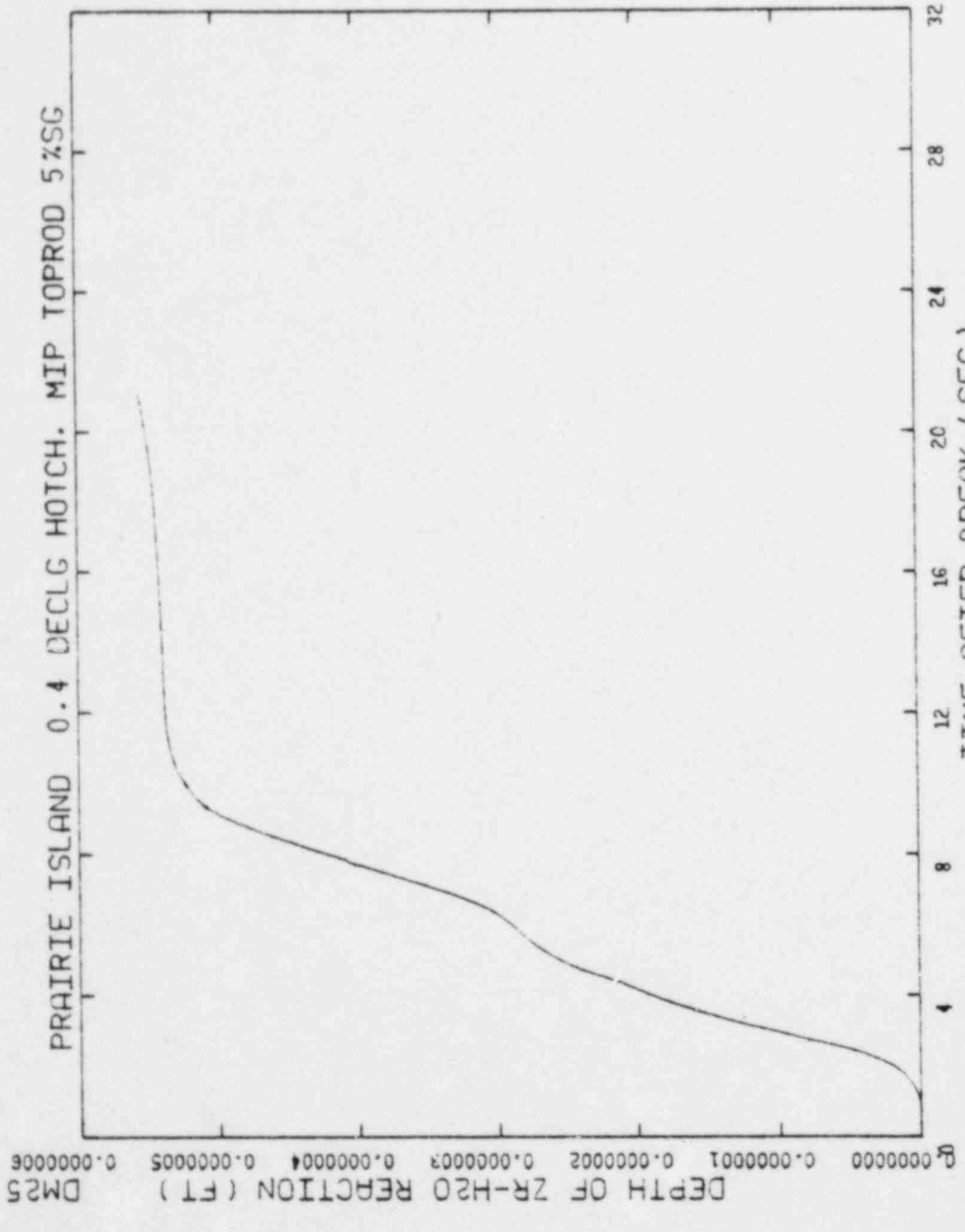


Figure 2.19 Depth of Metal-Water Reaction
0.4 DECLG Break, EOL

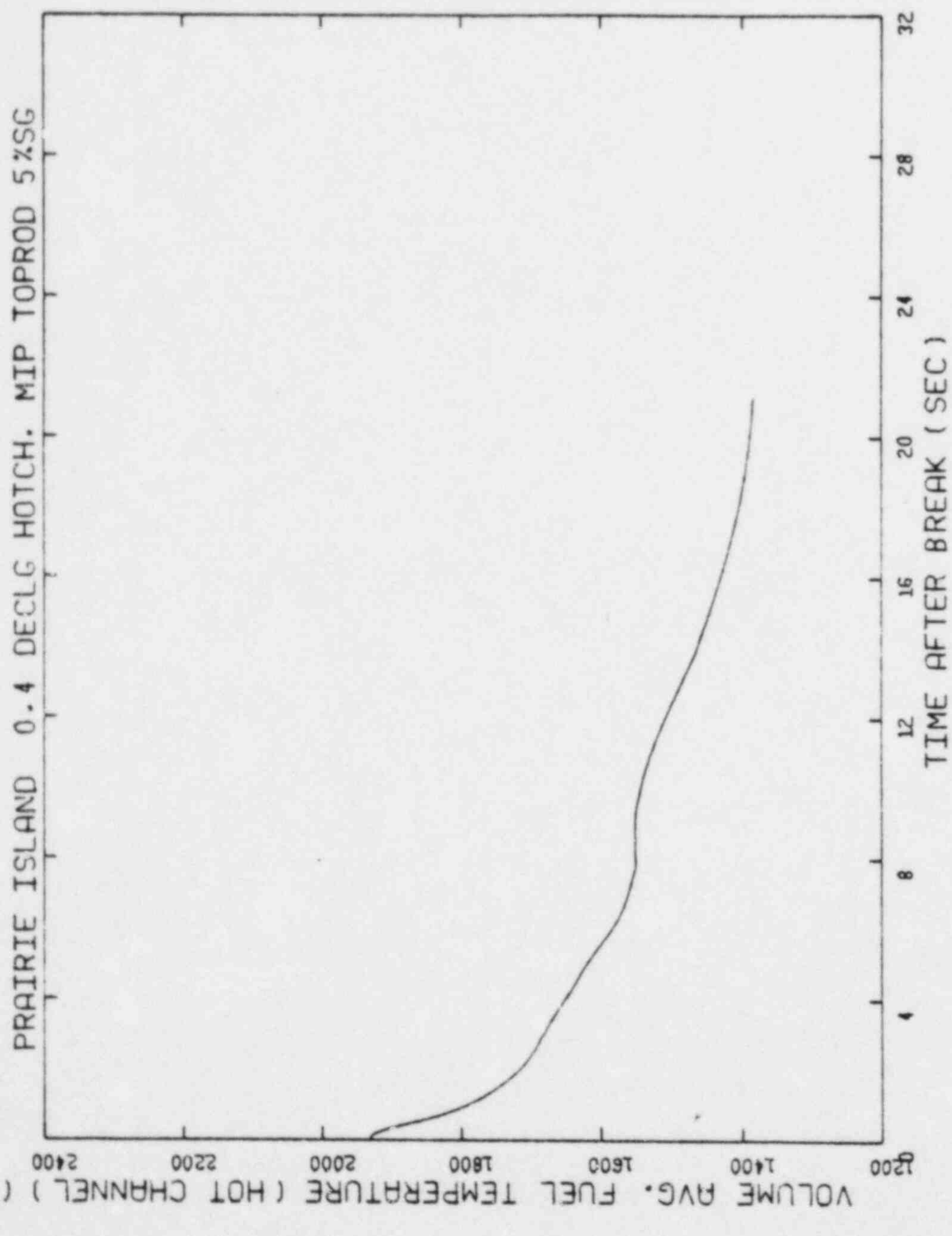


Figure 2.20 Hot Channel Average Fuel Temperature,
0.4 DECLG Break, EOL

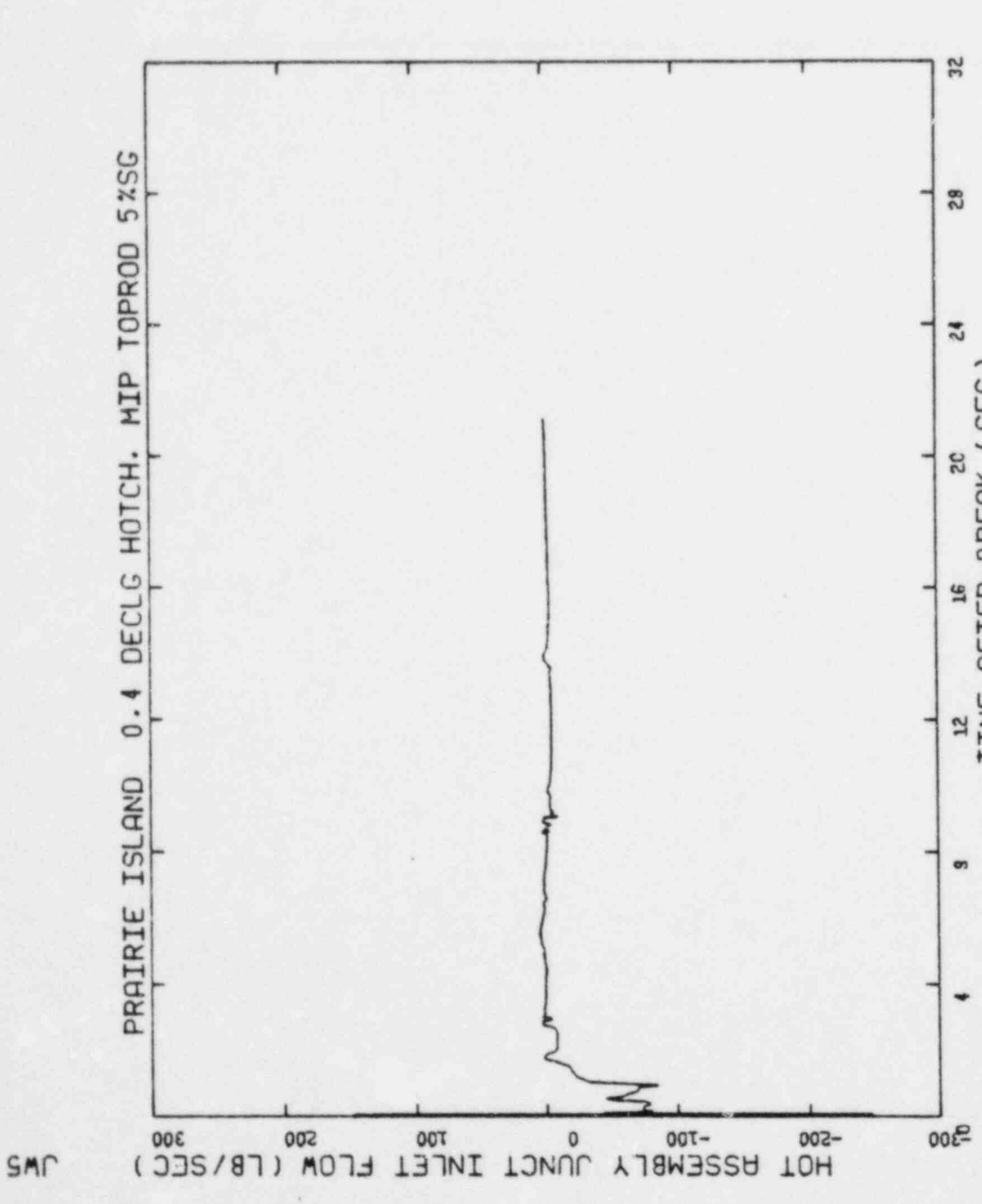


Figure 2.21 Hot Assembly Inlet Flow, 0.4 DECLG Break, EOL

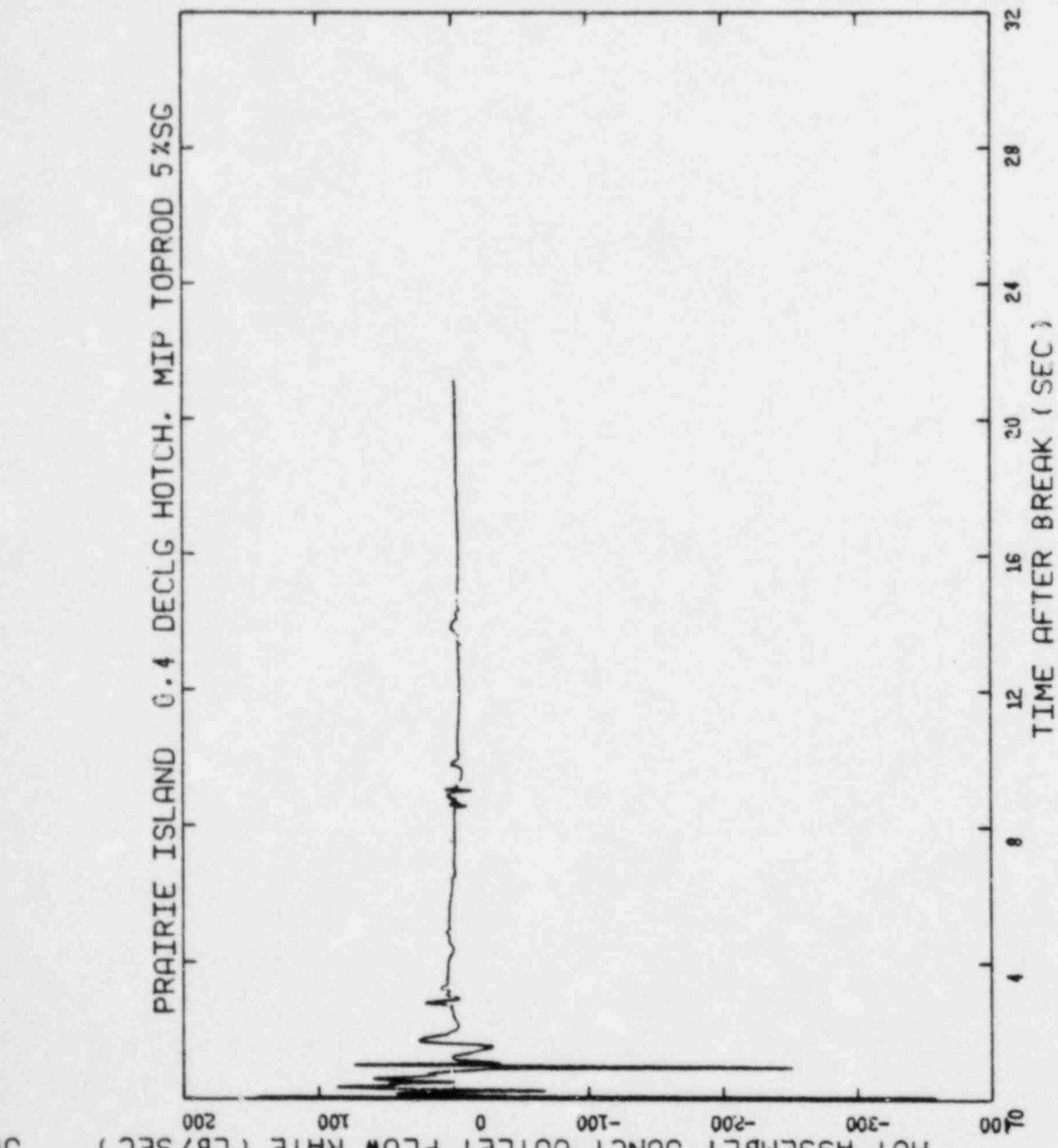


Figure 2.22 Hot Assembly Outlet Flow, 0.4 DECLG Break, EOL

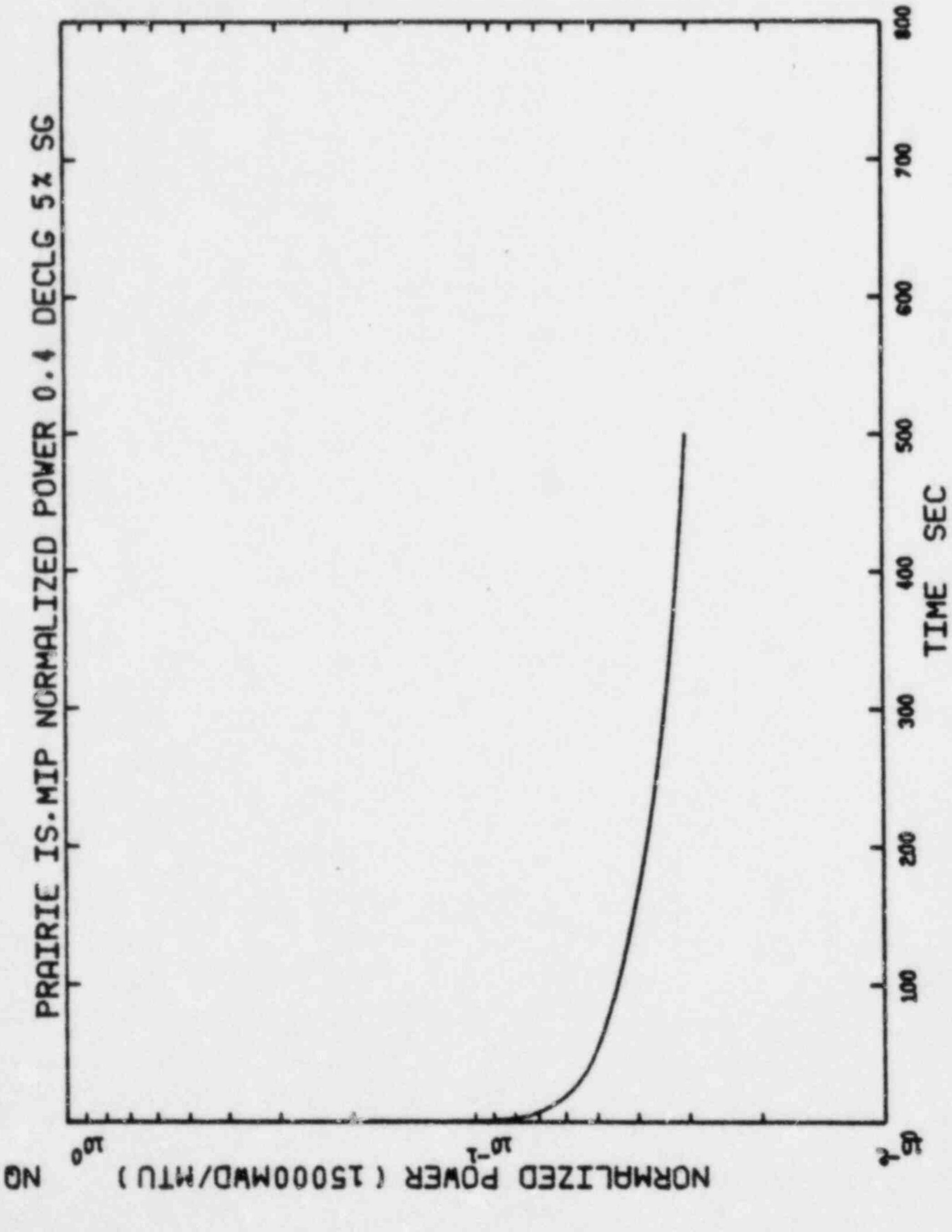


Figure 2.23 Normalized Power, 0.4 DECLG Break, 0-15,000 MWD/MTM

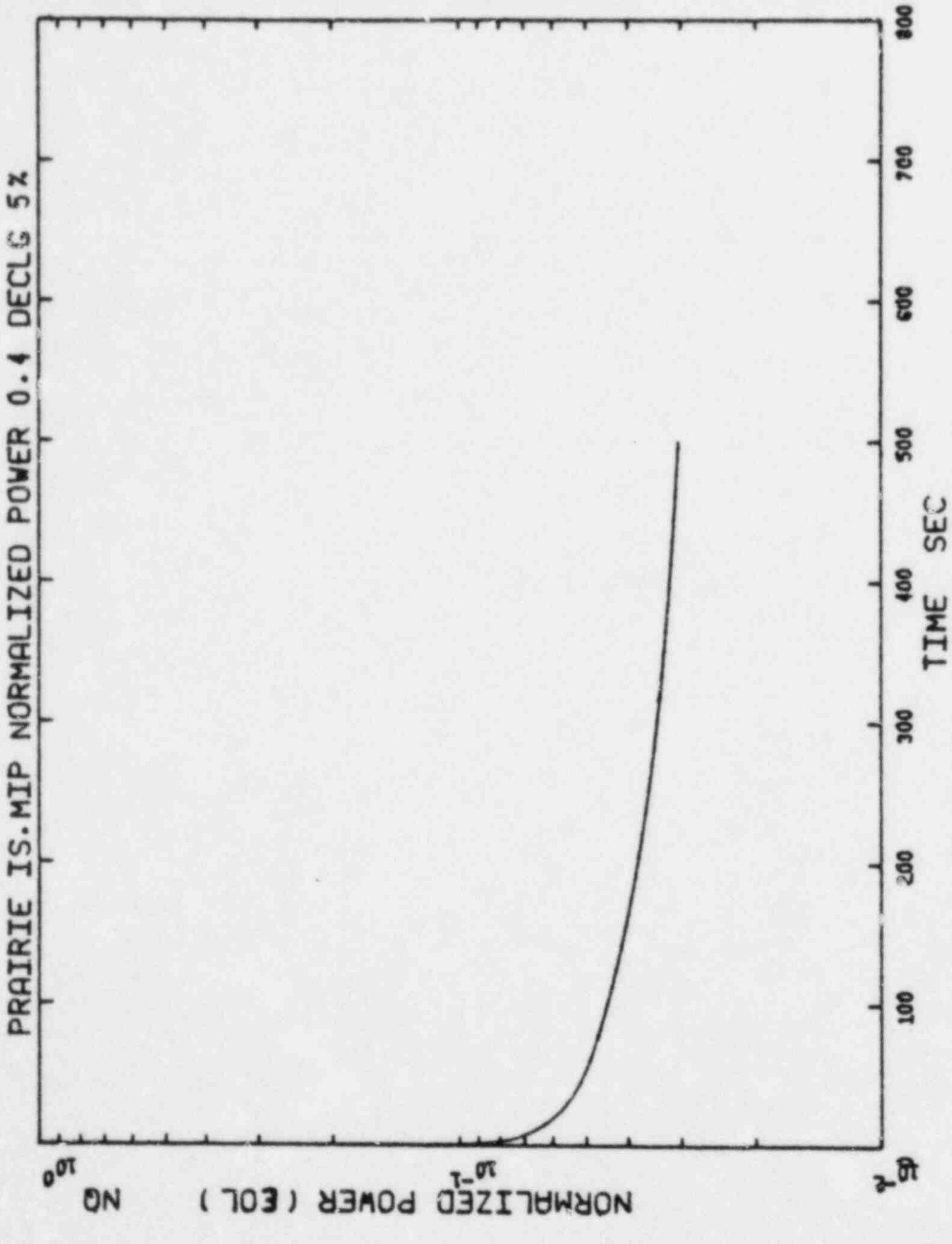


Figure 2.24 Normalized Power, 0.4 DECLG Break, EOL

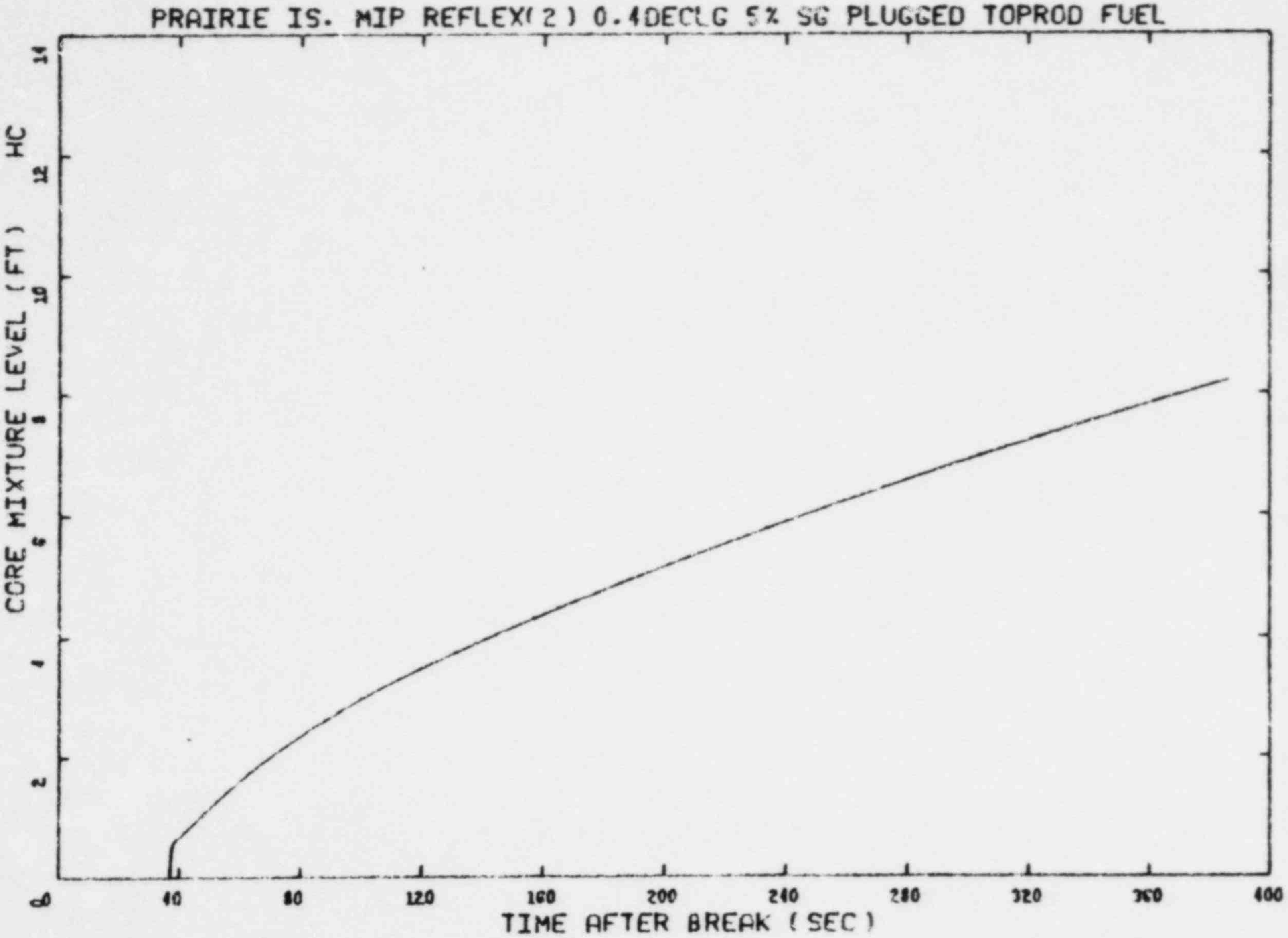


Figure 2.25 Reflood Core Mixture Level
0.4 DECLG Break, 0-15,000 MWD/MTM

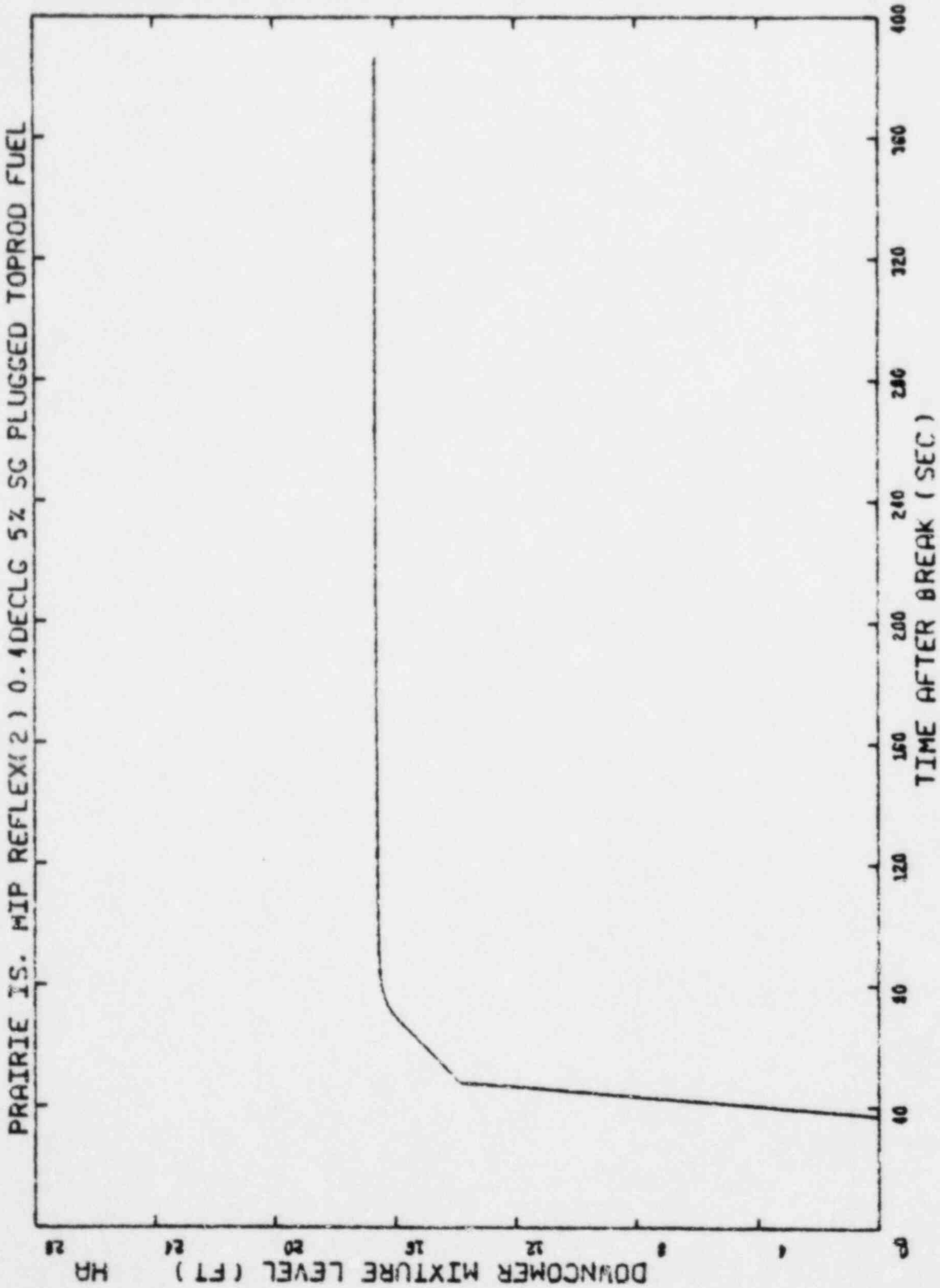


Figure 2.26 Reflood Downcomer Mixture Level, 0.4 DECLG Break, 0-15,000 MWD/MTM

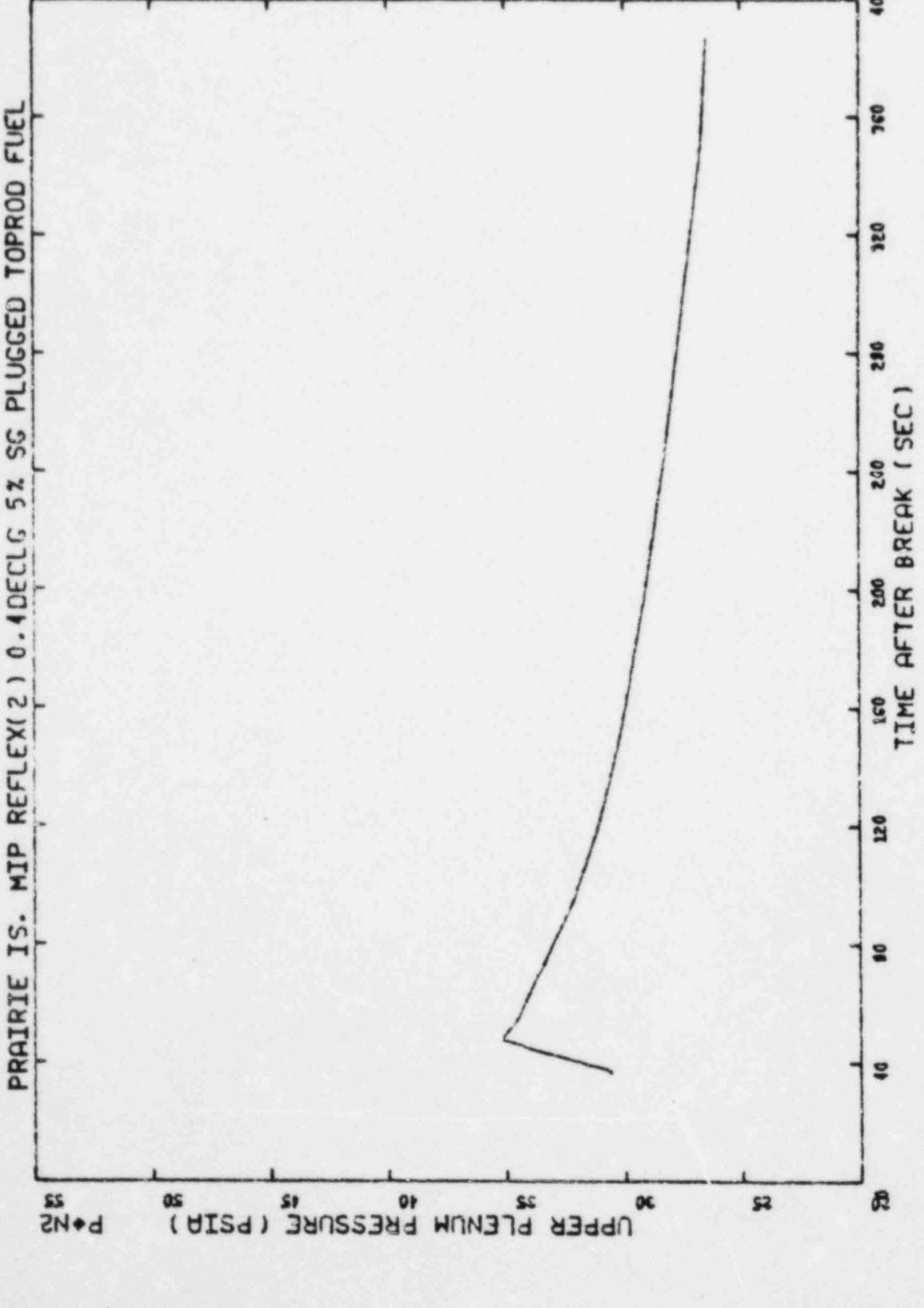


Figure 2.27 Reflood Upper Plenum Pressure, 0.4 DECLG Break, 0-15,000 MWD/MTM

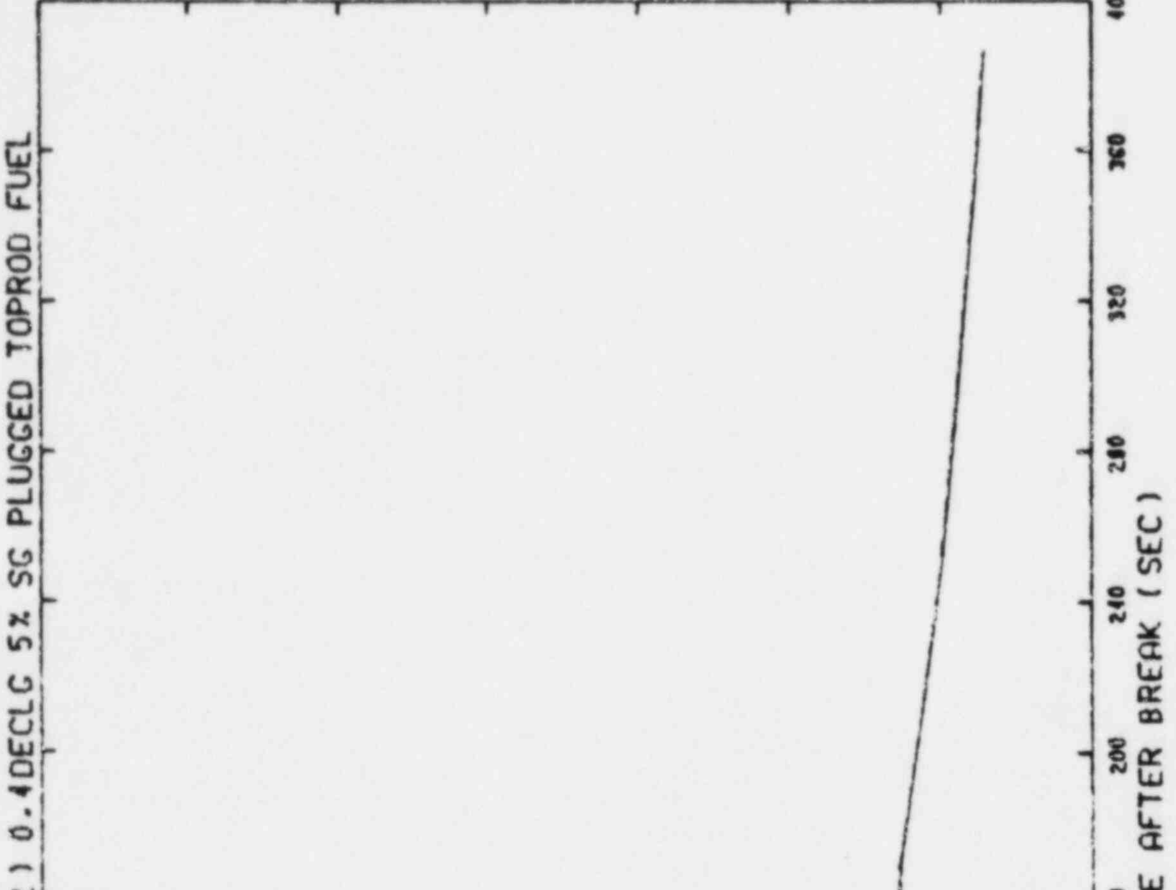


Figure 2.28 Core Flooding Rate
0.4 DECLG Break, 0-15,000 MWD/MTM

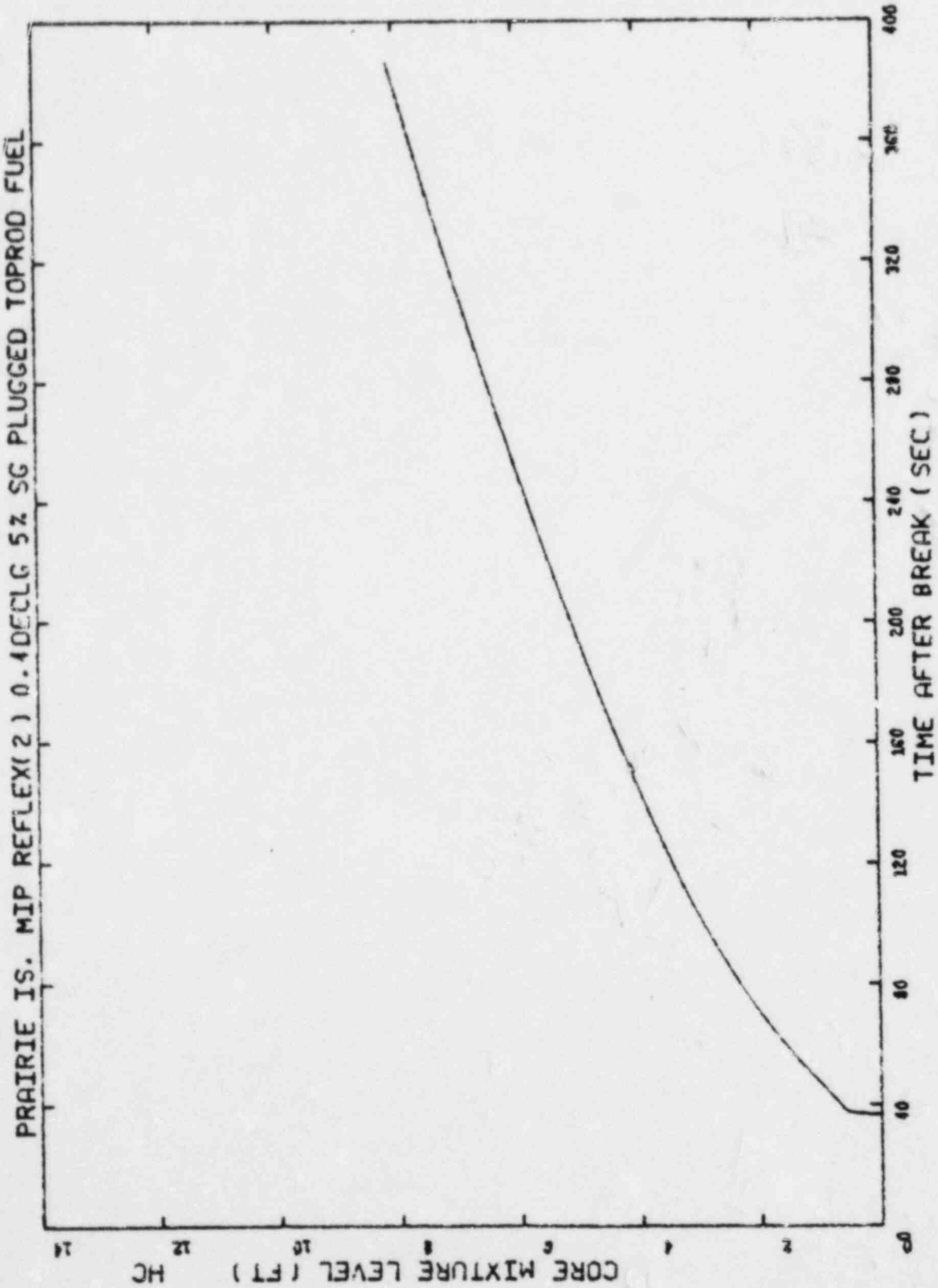


Figure 2.29 Reflood Core Mixture Level, 0.4 DECLG Break, EOL

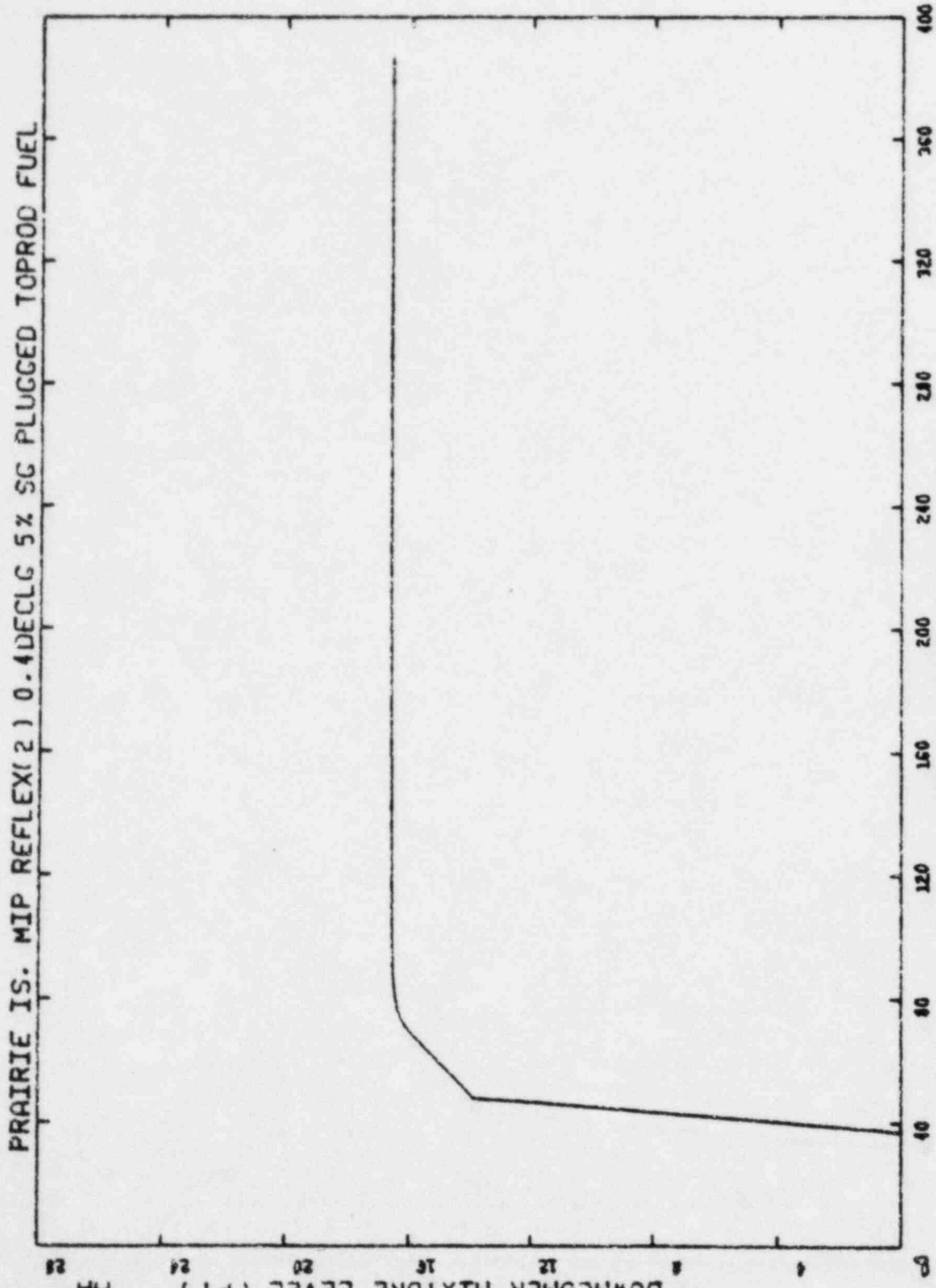


Figure 2.30 Reflood Downcomer Mixture Level, 0.4 DECLG Break, EOL

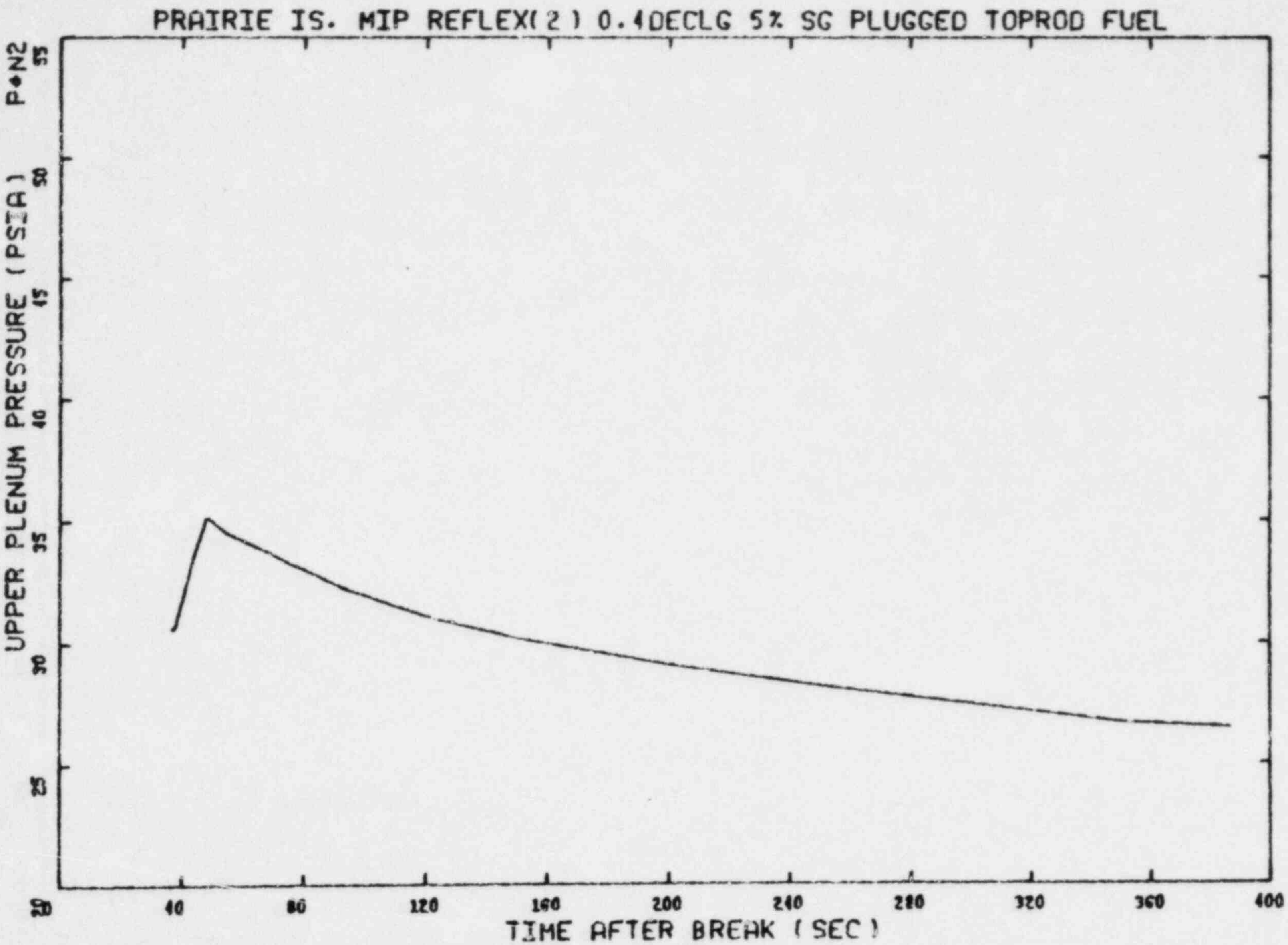


Figure 2.31 Reflood Upper Plenum Pressure
0.4 DECLG Break, EOL

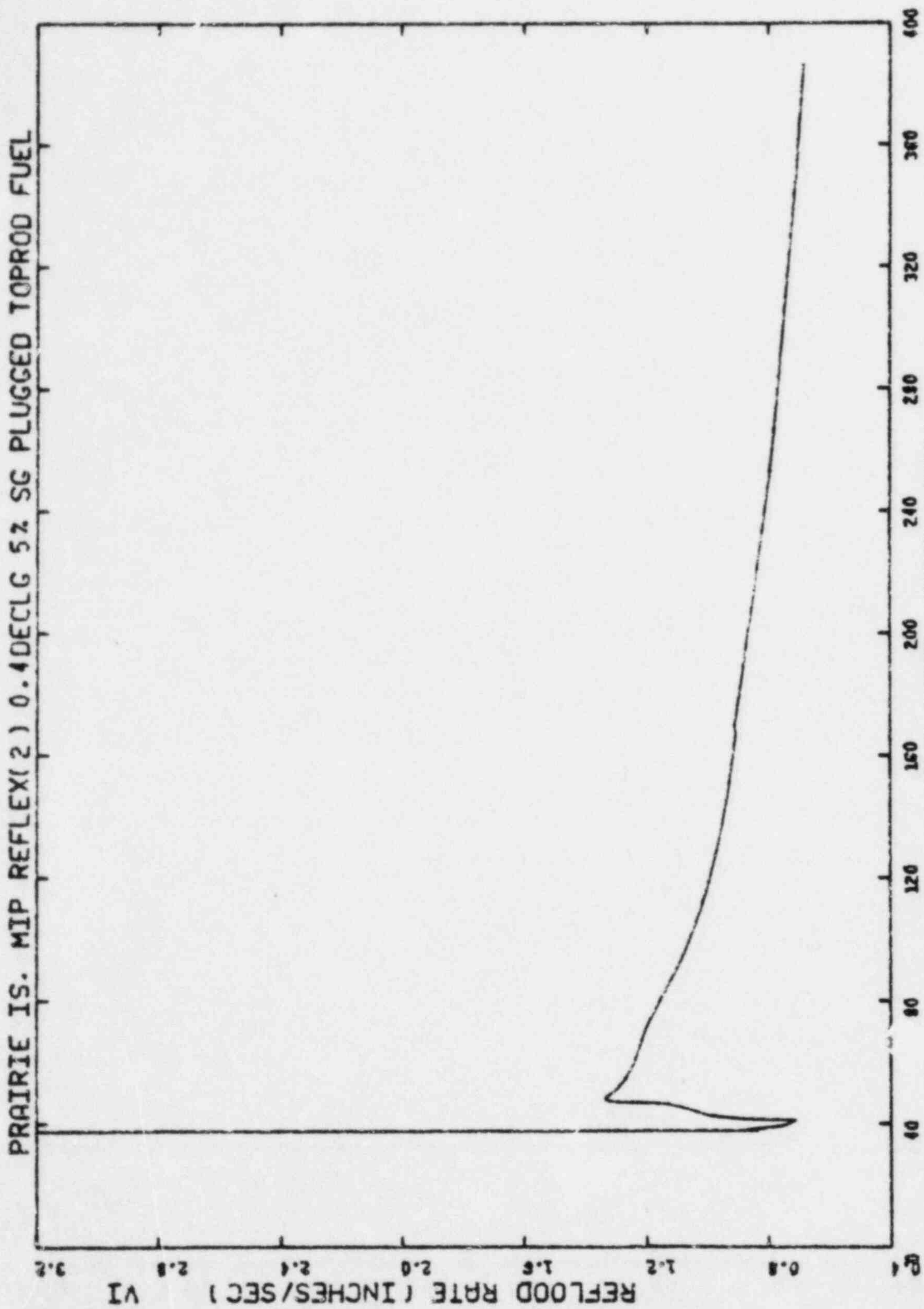


Figure 2.32 Core Flooding Rate, 0.4 DECLG Break, EOL

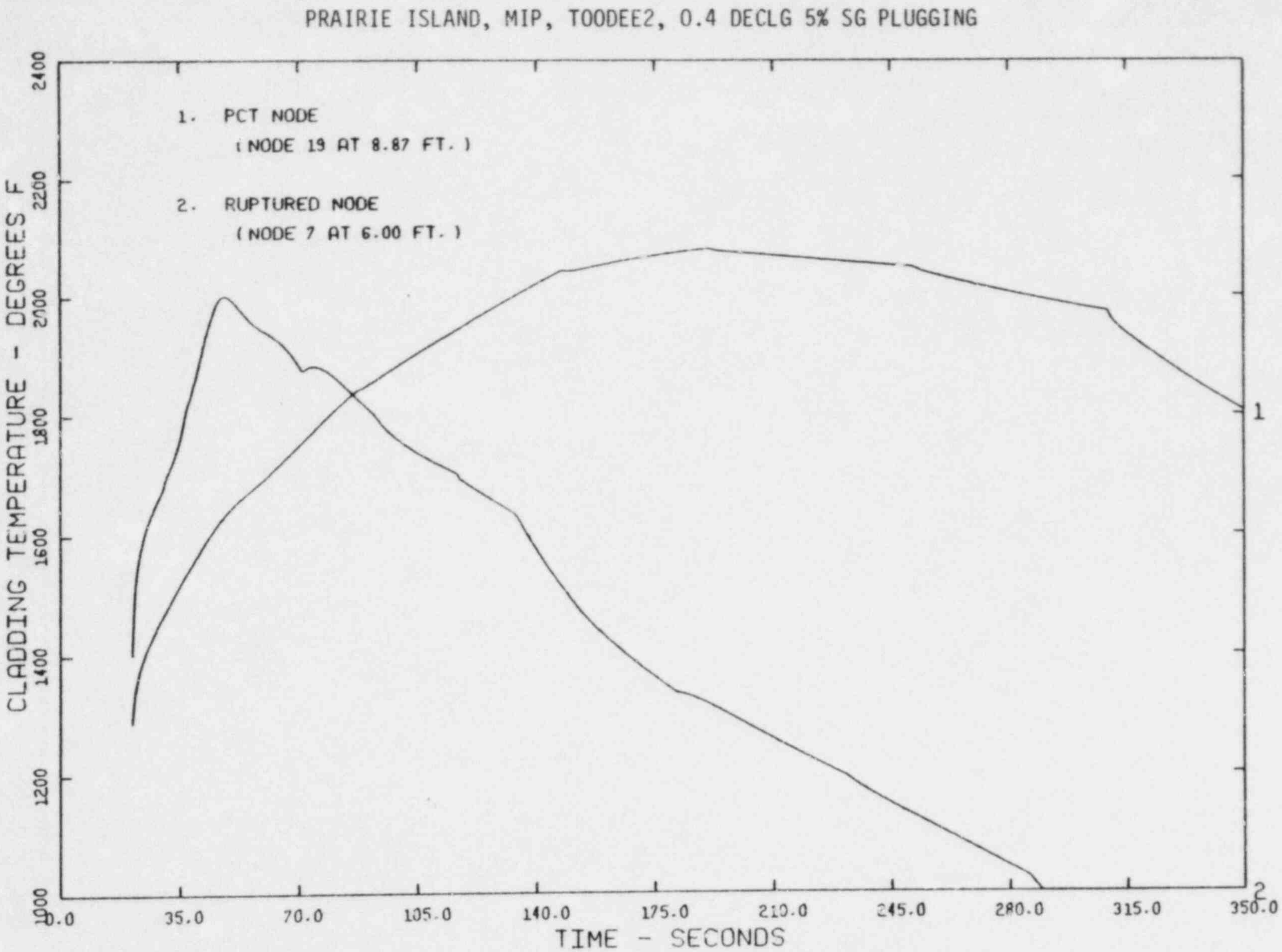


Figure 2.33 TOODEE2 Cladding Temperature vs Time
0.4 DECLG Break, 0-15,000 MWD/MTM

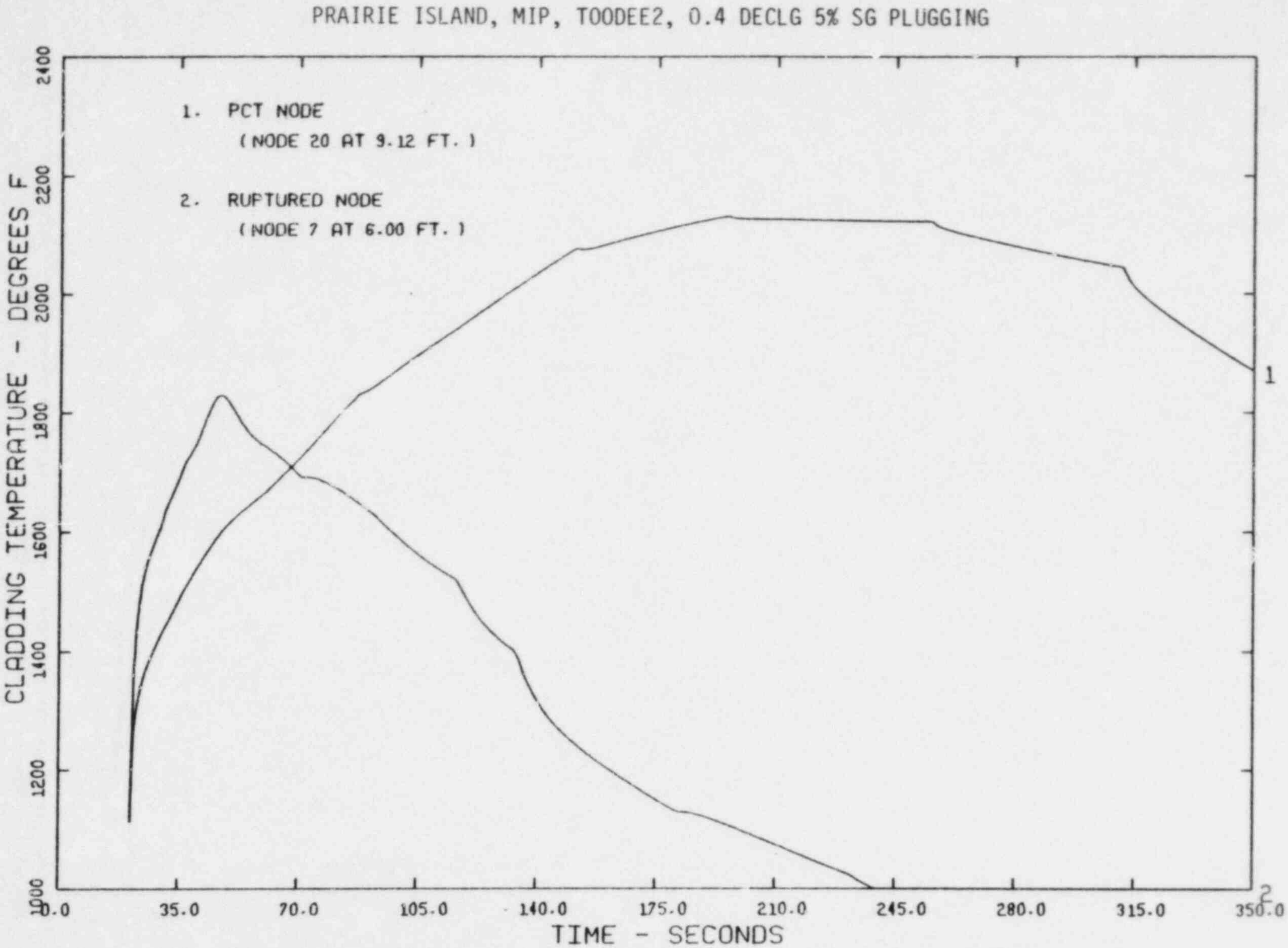


Figure 2.34 TOODEE2 Cladding Temperature vs Time
0.4 DECLG Break, EOL

## 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 2.32 $F_Q^T$ and 1.55 $F_{\Delta H}^T$ 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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XN-NF-83-38

Issue Date: 5/27/83

PRAIRIE ISLAND UNITS 1 AND 2

LIMITING BREAK LOCA-ECCS ANALYSIS

USING EXEM/PWR

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