XN-NF-84-31
REVISION 1

KEWAUNEE HIGH BURNUP SAFETY ANALYSIS: LIMITING BREAK LOCA AND RADIOLOGICAL CONSEQUENCES

OCTOBER 1984

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

XN-NF-84-31
Revision 1

Issue Date: 10/1/84

KEWAUNEE HIGH BURNUP SAFETY ANALYSIS:

LIMITING BREAK LOCA AND RADIOLOGICAL CONSEQUENCES

Prepared by: Mark S. Stricker
M. S. Stricker
PWR Safety Analysis

Concur: 9/26/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: RA Copeland 26 Sep 84
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Approve:
R. B. Stout, Manager
Licensing & Safety Engineering

Concur: 9/28/84
J. N. Morgan, Manager
Proposals & Customer Services Engineering

Approve:
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

Without derogating from the foregoing, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

1.0 INTRODUCTION AND SUMMARY

This document presents analytical results for a postulated large break loss-of-coolant accident (LOCA) and assessment of radiological consequences of accidents for the Kewaunee reactor operated with ENC fuel up to a fuel rod burnup of 49,000 MWD/MTM. The analyses assume a reactor operating power of 1683 MWt (1650 MWt plus 2% power uncertainty), and use of Exxon Nuclear Company's (ENC's) 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 LOCA analyses were performed for a full core of ENC fuel 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 to a five percent (5%) average steam generator (SG) tube plugging, and maximum peak rod average exposure of 49,000 MWD/MTM. The allowable linear heat generation rate for the entire exposure range (including the 1.02 factor for power uncertainty) is 14.76 kW/ft, corresponding to a total power peaking factor of 2.28 (F_Q^T), and nuclear enthalpy rise of 1.55 ($F_{\Delta H}^T$).

The calculational basis and results are summarized in Table 1.1. The maximum calculated peak cladding temperature (PCT) is 2011°F, occurring at 260 seconds into the accident at a location 8.88 feet from the bottom of the active core, with a total metal-water reaction less than one percent. The 2011°F PCT includes a 51°F temperature correction to allow for 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 Kewaunee nuclear reactor satisfies the criteria specified by 10 CFR 50.46[8] for operation at the rated system power level and with the steam generator tube plugging up to 5%.

For breaks up to and including the double-ended severance of a reactor cold leg coolant pipe, the Emergency Core Cooling System for the Kewaunee unit will meet the Acceptance Criteria as presented in 10 CFR 50.46, with the 2.28 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.

The results of the radiological consequences analysis are given in Section 3.0. The analysis was performed in accordance with the methodology specified in "Assessment of Potential Radiological Consequences for High Exposure Fuel."[18] The postulated LOCA and fuel handling accidents were analyzed for maximum assembly average exposures to 49,000 MWD/MTM. This revision contains updated dose predictions for a Fuel Handling Accident (FHA) at Kewaunee. The previous analysis used a version of RODEX2 fuel performance computer code that incorrectly calculated the isotopic release fractions using the ANS 5.4 fission gas release model. The error in RODEX2 was corrected and a reanalysis of the radiological consequence of an FHA in the auxiliary building was performed. The new dose predictions are reported in Table 3.1 and are well below 10 CFR 100 guidelines. The results show that the radiological consequences of a LOCA or a fuel handling accident involving ENC high burnup fuel are well below 10 CFR 100 dose limits of 300 and 25 rem for the thyroid and whole body, respectively. Specifically, the 2 hour thyroid and whole body doses received following a LOCA are 10.9 and 1.8 rem, respectively; the LOCA 30 day thyroid and whole body doses are 3.8 and 1.8 rem, respectively; the 2 hour thyroid and whole body doses following a fuel handling accident are 8.3 and 1.7 rem, respectively.

Table 1.1 Kewaunee LOCA-ECCS Analysis Results

Analysis Results	0-15000 MWD/MTM Peak Average Rod Exposure	15000-49000 MWD/MTM Peak Average Rod Exposure
Peak Clad Temperature (PCT), ^oF***	1865	2011
ΔPCT for UPI, ^oF	-18	51
Time of PCT, sec.	100	260
Peak Clad Temperature Location, ft.	7.25	8.88
Local Zr/H_2O Reaction (max.), %*	2.3	3.2
Local Zr/H_2O Location, ft. from bottom	7.94	8.88
Total H_2 Generation, % of total Zr reacted	< 1.0	< 1.0
Hot Rod Burst Time, sec.	39	40.6
Hot Rod Burst Location, ft.	6	6
Calculational Basis		
License Core Power, MWt	1650	1650
Power Used for Analysis, MWt**	1683	1683
Peak Linear Power for Analysis, kW/ft**	14.76	14.76
Total Peaking Factor, F_Q^T	2.28	2.28
Enthalpy Rise, Nuclear, $F_{\Delta H}^T$	1.55	1.55
Steam Generator Tube Plugging (%)	5.00	5.00

* Computer value at 380 seconds

** Including 1.02 factor for power uncertainties

*** Includes ΔPCT for UPI

2.0 LIMITING BREAK LOCA ANALYSIS

This report provides the results of a LOCA-ECCS analysis performed for Kewaunee with total 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 for a similar Westinghouse two-loop plant, with results 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 Kewaunee 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. 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 Kewaunee 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 as 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 1.428, corresponding to a total peaking factor F_Q^T of 2.28, and $F_{\Delta H}^T$ of 1.55. The F_Q^T determined using this axial power profile in conjunction with the current K(Z) function developed by the NSSS vendor is used to define the operating envelop for F_Q^T where the K(Z) curve is limited by large break LOCA. Where small break LOCA is limiting, the K(Z) curve is modified such that the Linear Heat Generation Rates (LHGRs) are determined by the NSSS vendor analysis. The modified K(Z) function is shown in Figure 2.35. 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 LOCA-ECCS calculations were performed with input which bounds the fuel history up to 49,000 MWD/MTM peak power rod average 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.

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 ENC 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 initiation. Figure 2.10 presents calculated containment backpressure time history. Figures 2.11 through 2.22 present results for the hot channel blowdown calculations. Figure 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 2011°F (Figure 2.34). This value includes a 51°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 the ENC fuel. The maximum local metal-water reaction in this case is 3.2% after 260 seconds, and the total core metal-water reaction is less than 1%. The PCT location is at an elevation of 8.88 feet from the bottom of active core. For the exposure up to 15,000 MWD/MTM, the PCT is 1865°F (Figure 2.33) including -18°F for UPI effect, occurring at 7.25 feet elevation relative to the bottom of the active core. The local metal-water reaction is 2.3%, with a total metal-water reaction of less than 1%.

Table 2.1 Kewaunee System Data

Primary Heat Output, MWt	1650*
Primary Coolant Flow, lbm/hr	6.82×10^7
Operating Pressure, psia	2,250
Inlet Coolant Temperature, ^oF	534
Reactor Vessel Volume, ft^3	2406
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 Secondary Heat Transfer Area, ft^2	48,925**
Steam Generator Secondary Flow, lbm/hr	3.56×10^6
Steam Generator Secondary Pressure, psia	750
Reactor Coolant Pump Head, ft (Design)	277
Reactor Coolant Pump Speed, rpm (Design)	1190
Moment of Inertia, lbm-ft^2/rad	80,000
Cold Leg Pipe, I.D., in	27.5
Hot Leg Pipe, I.D., in	29
Pump Suction Pipe, I.D., in	31

* Primary Heat Output used in RELAP4-EM Model = 1.02 x 1650 = 1683 MWt.
** Includes 5% SG tube plugging.

Table 2.2 Fuel Design Parameters

Cladding, O.D., in.	0.424
Cladding, I.D., in.	0.364
Cladding Thickness, in.	0.030
Pellet O.D., in.	0.3565
Diametral Gap, in.	0.0075
Pellet Density, % TD	94.0
Active Fuel Length, in.	144.0
Rod Pitch	0.556

Table 2.3 Kewaunee LOCA-ECCS Analysis Results, Event Times

Event	Time (sec.)
Start	0.00
Break Initiation	.05
Safety Injection Signal	.65
Accumulator Injection, Broken Loop	4.8
Accumulator Injection, Intact Loop	8.8
End-of-Bypass	22.7
Safety Injection Flow	25.7
Start of Reflood	38.0
Accumulator Empties, Intact Loop	43.1
Peak Clad Temperature Reached -	
49,000 MWD/MTM	260.0
15,000 MWD/MTM	100.0

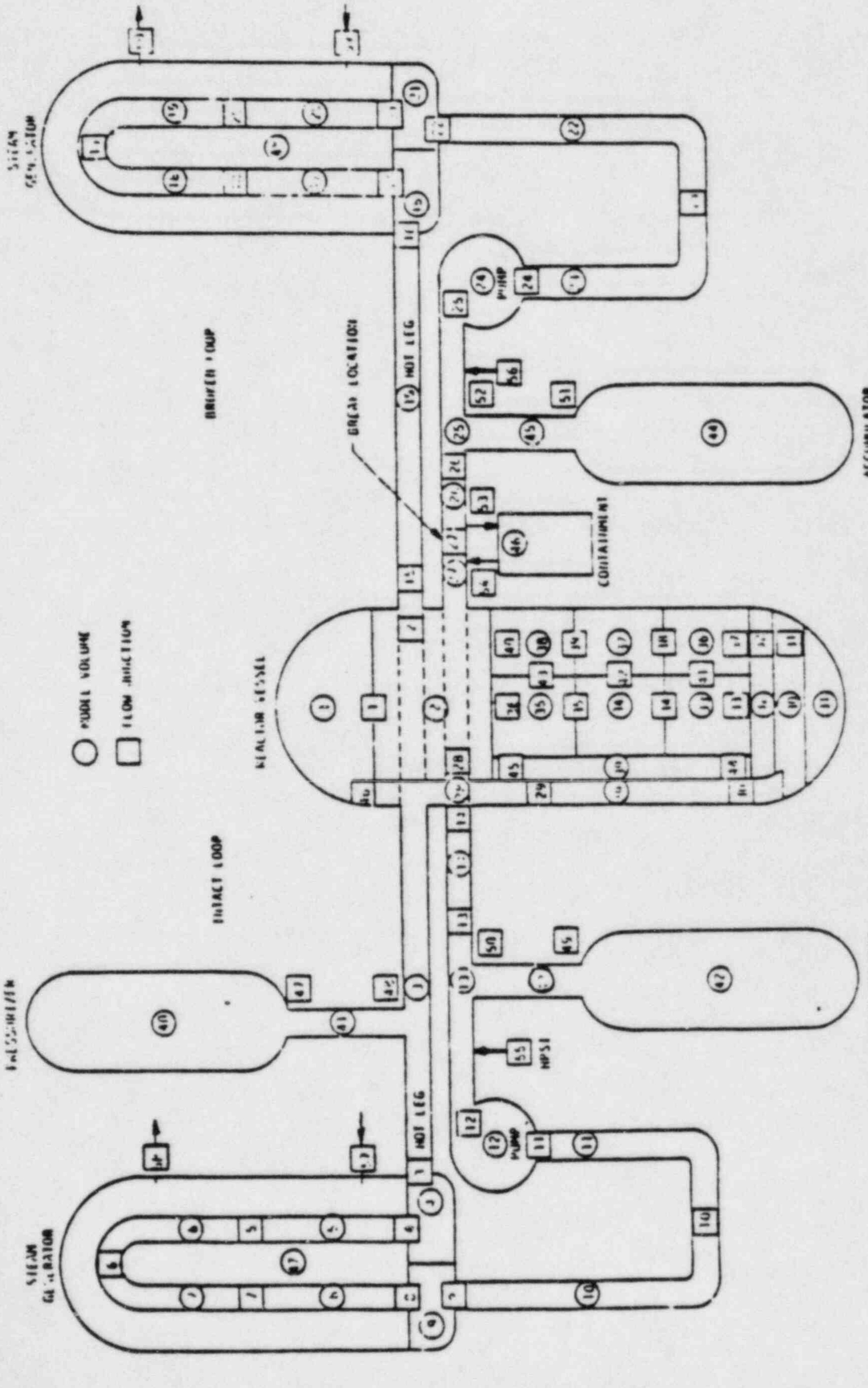

Figure 2.1 RELAP4/EM Blowdown System Nodalization for Kewaunee

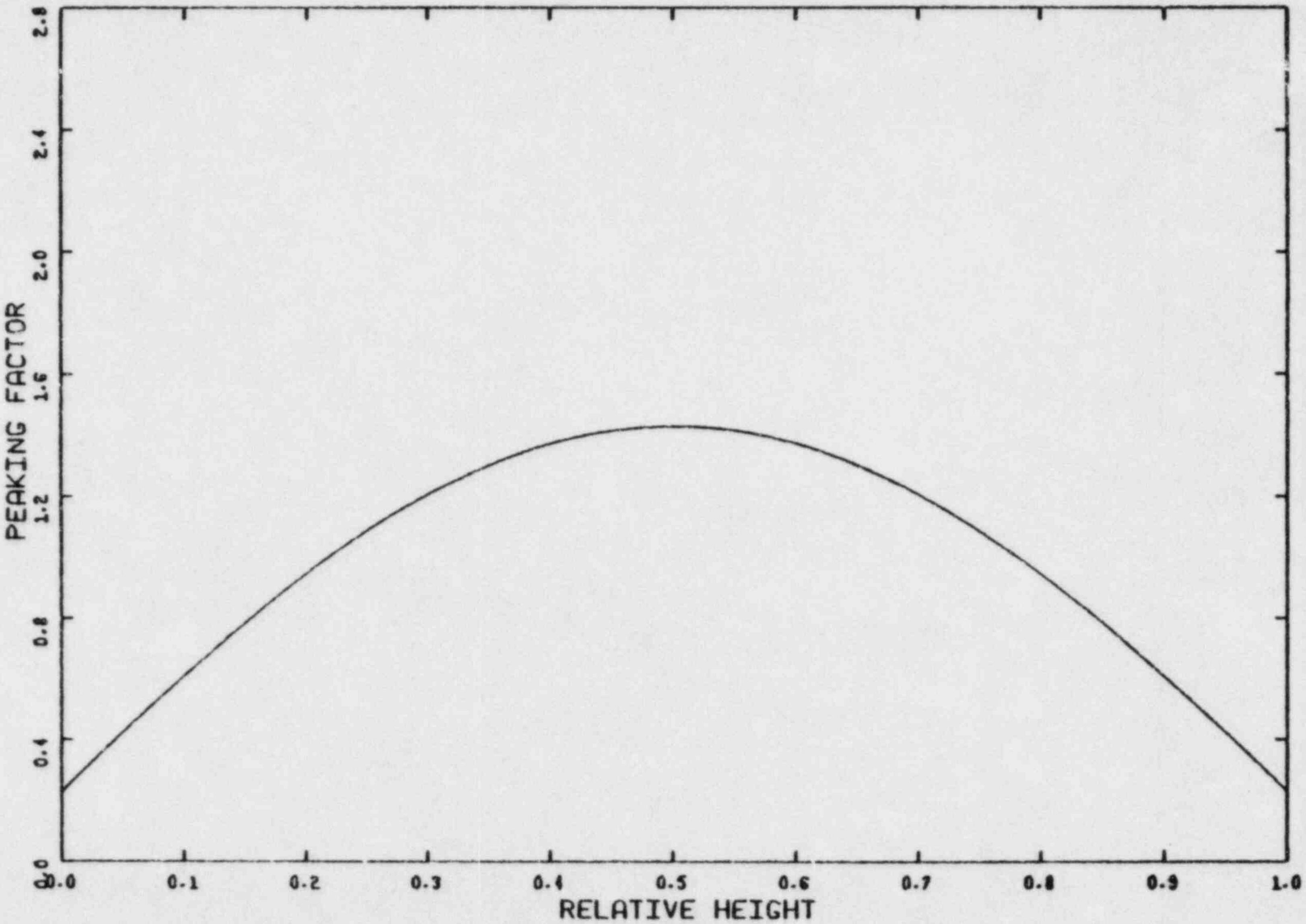

Figure 2.2 Axial Peaking Factor versus Rod Length
0.4 DECLG Break

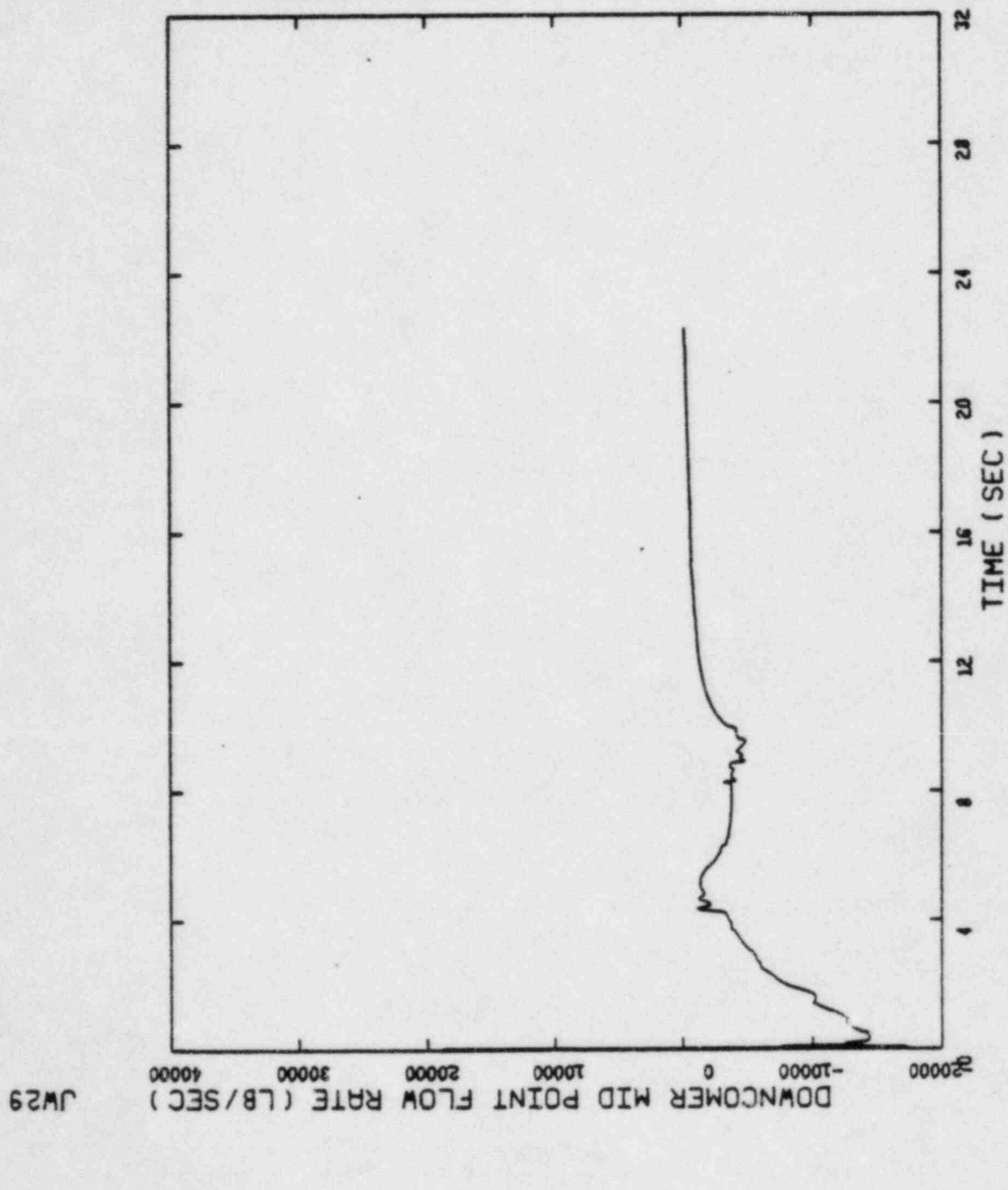

Figure 2.3 Downcomer Flow Rate, 0.4 DECLG Break

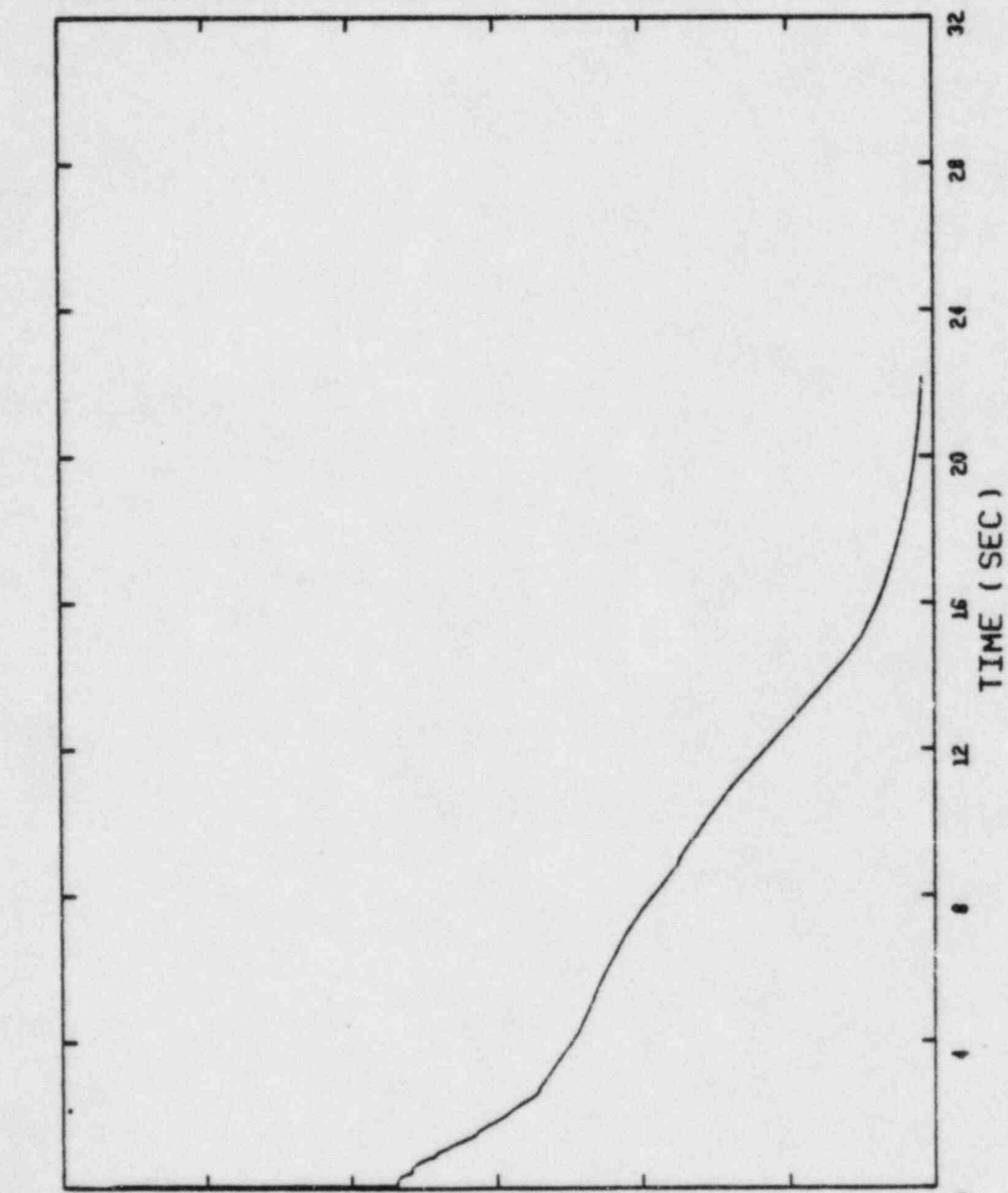

Figure 2.4 Upper Plenum Pressure, 0.4 DECLG Break

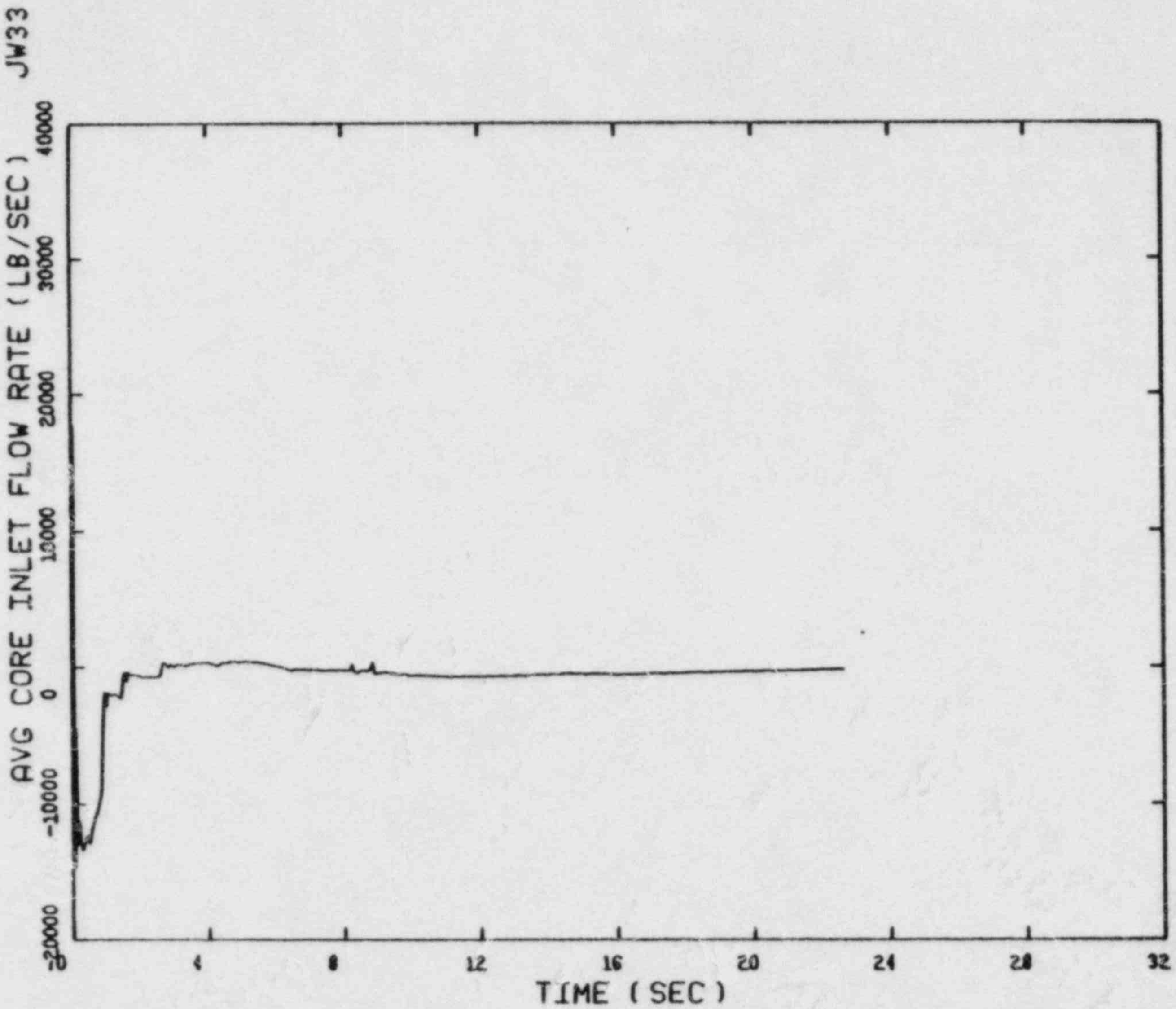

Figure 2.5 Average Core Inlet Flow, 0.4 DECLG Break

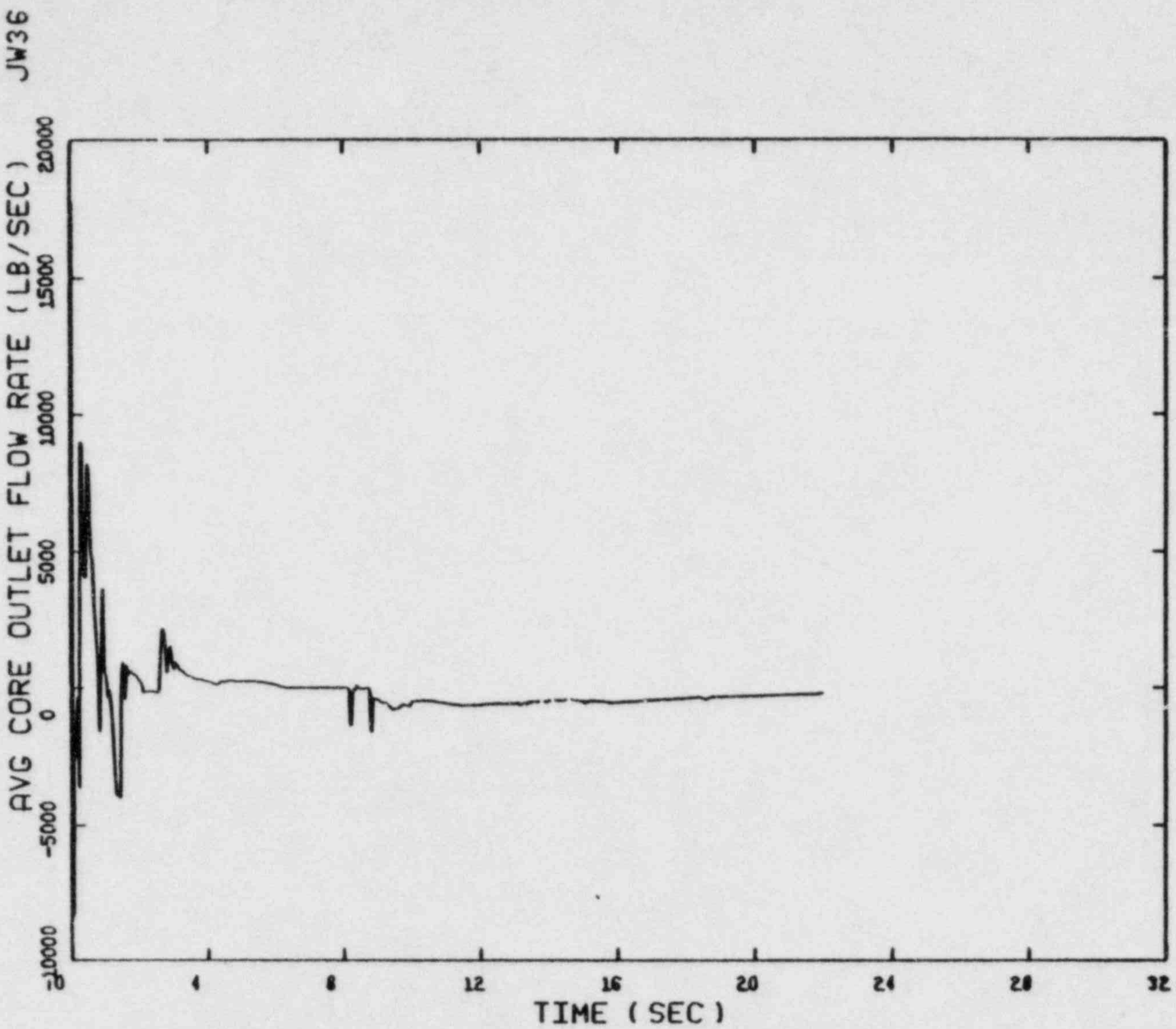

Figure 2.6 Average Core Outlet Flow, 0.4 DECLG Break

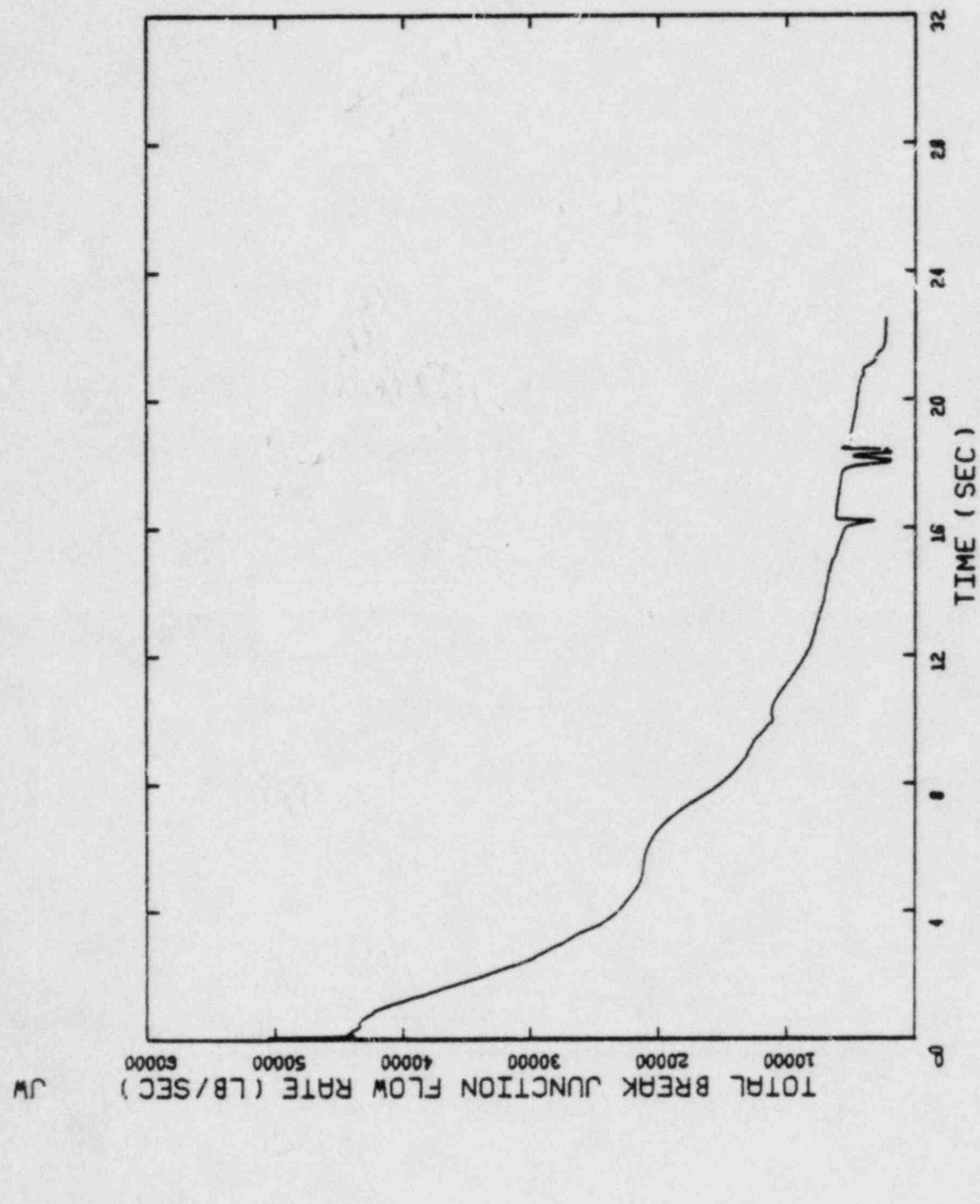

Figure 2.7 Total Break Flow, 0.4 DECLG Break

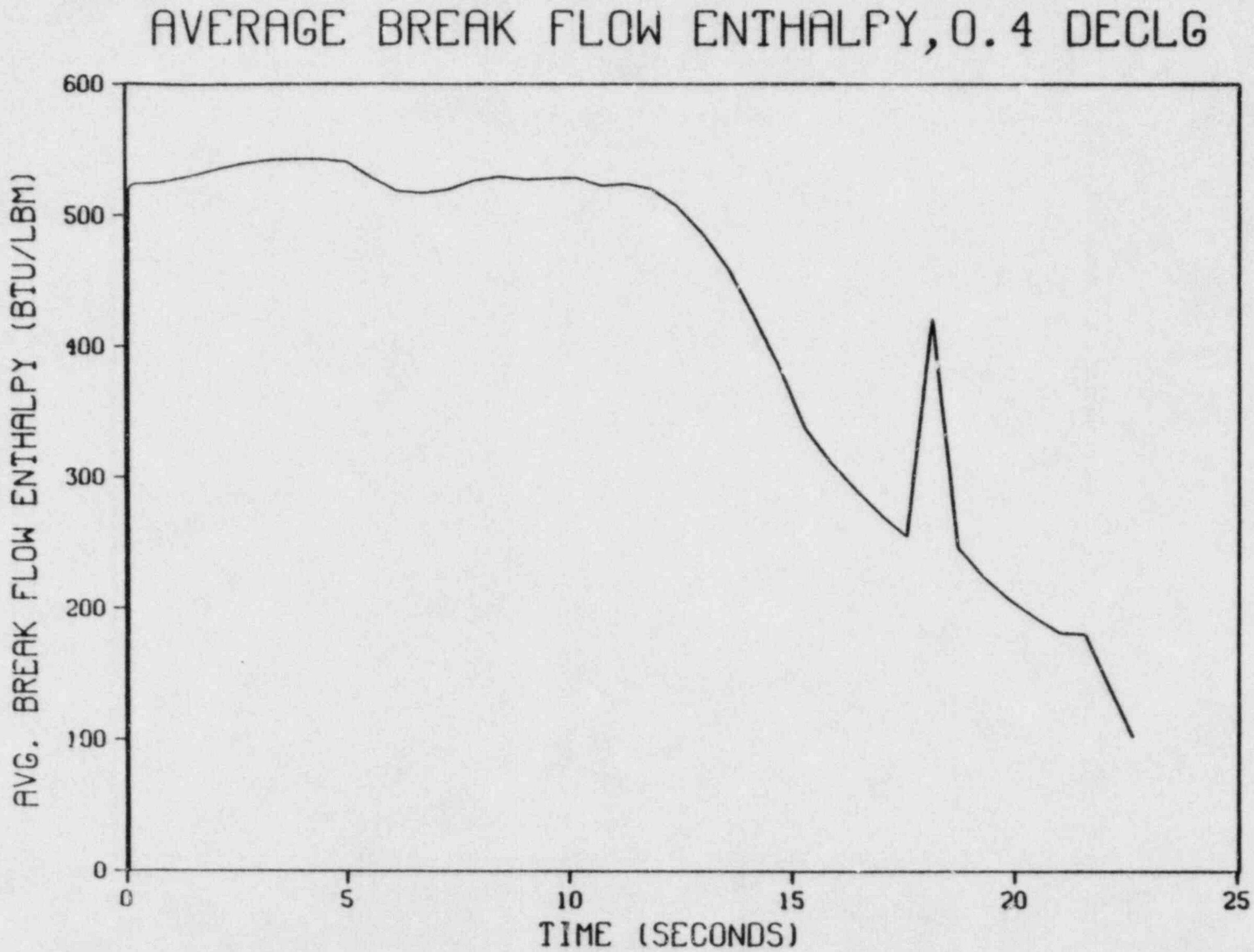

Figure 2.8 Average Break Flow Enthalpy, 0.4 DECLG

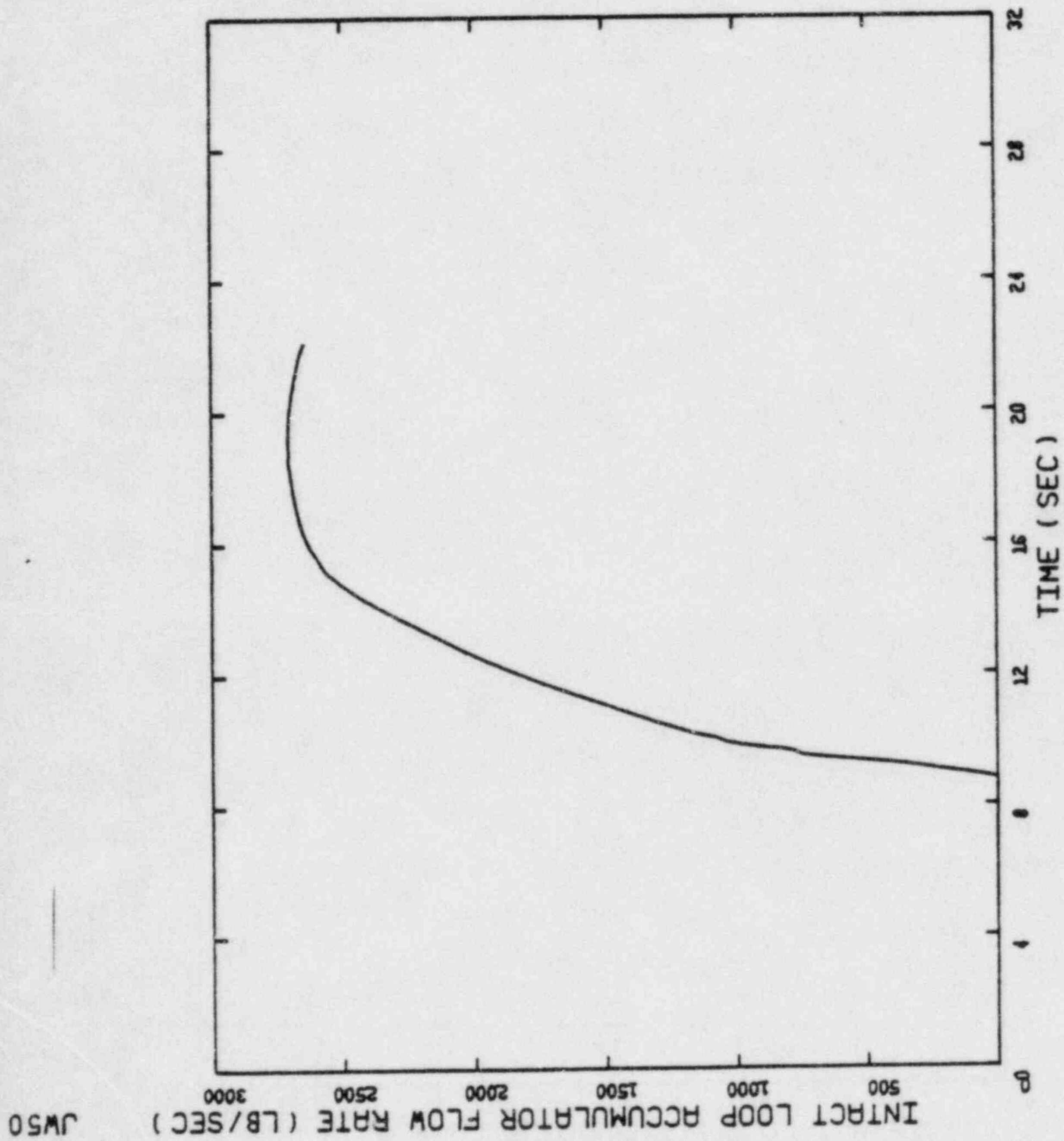

Figure 2.9 Flow From Intact Loop Accumulator, 0.4 DECLG Break

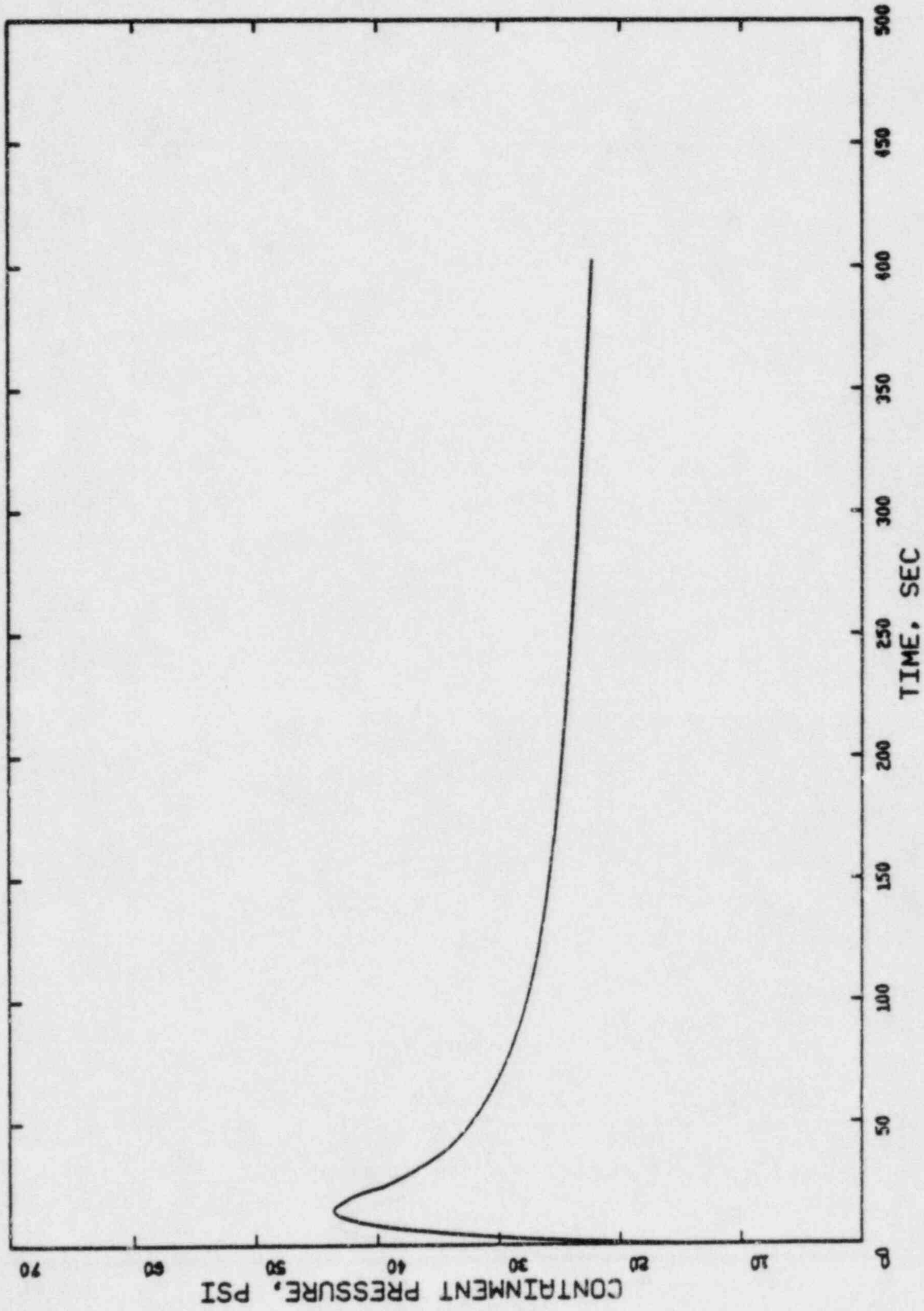

Figure 2.10 Containment Back Pressure, 0.4 DECLG Break

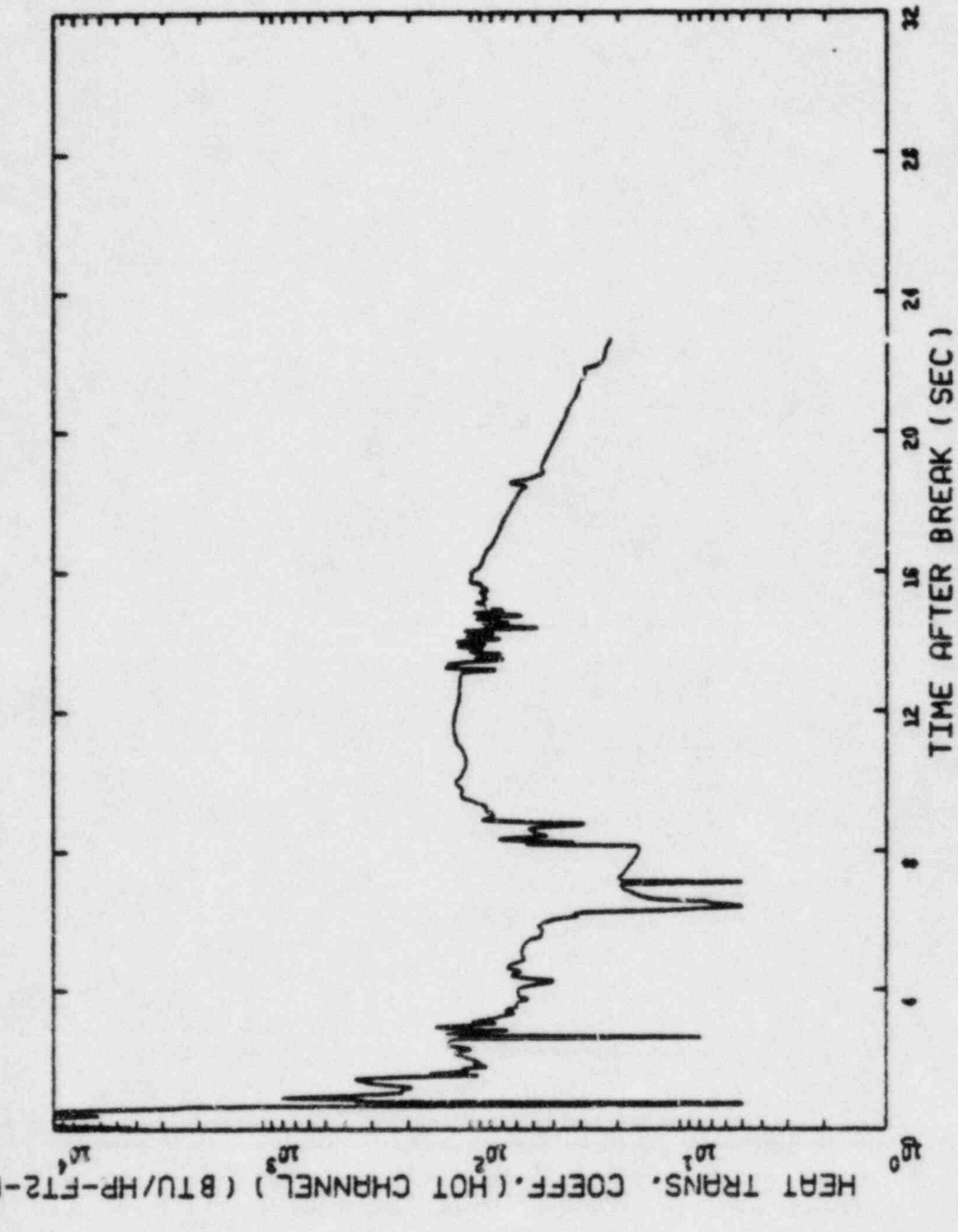

Figure 2.11 Hot Channel Heat Transfer Coefficient, 0.4 DECLG Break, 0-15,000 MWD/MTM

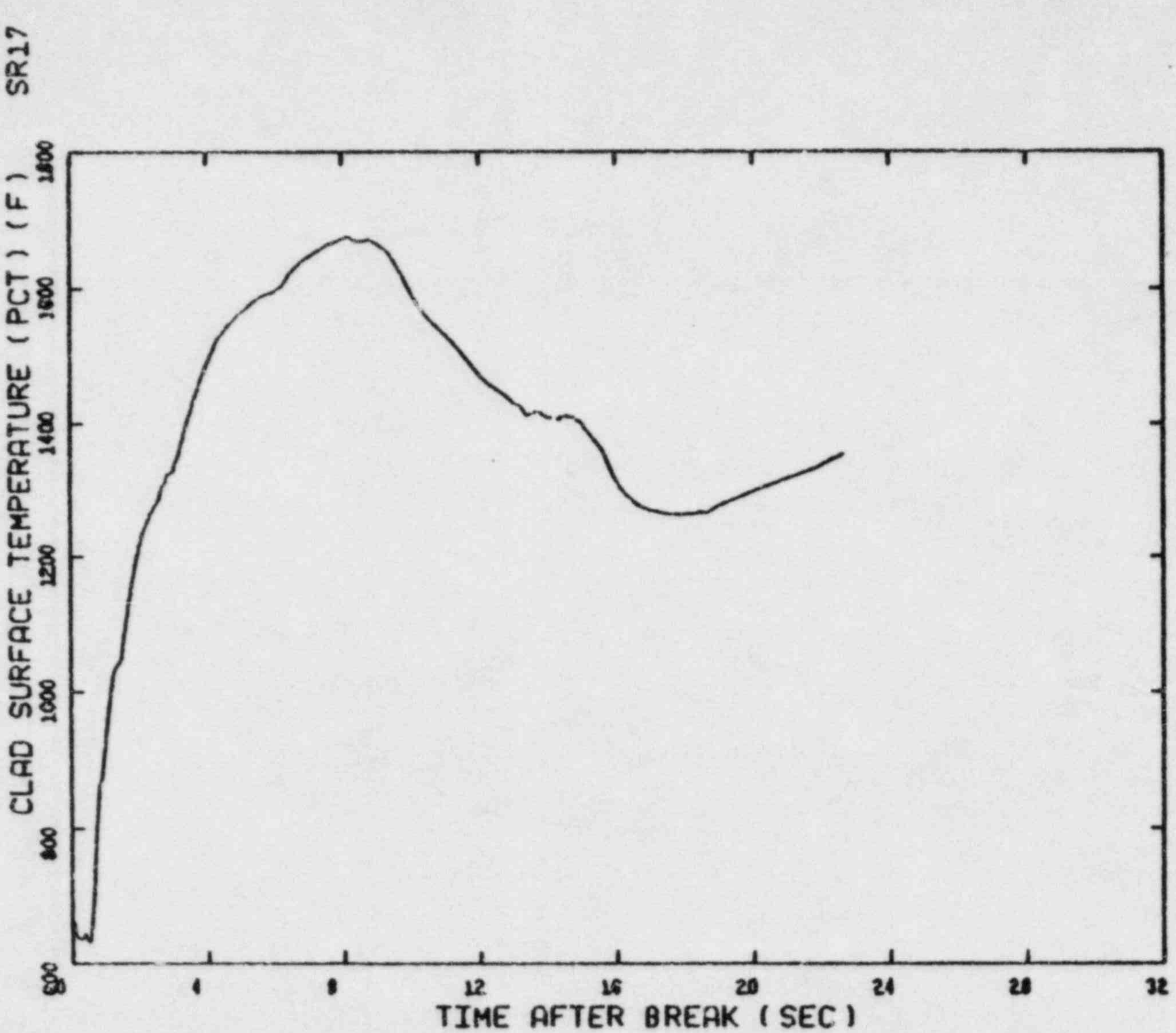

Figure 2.12 Clad Surface Temperature, 0.4 DECLG Break, 0-15,000 MWD/MTM

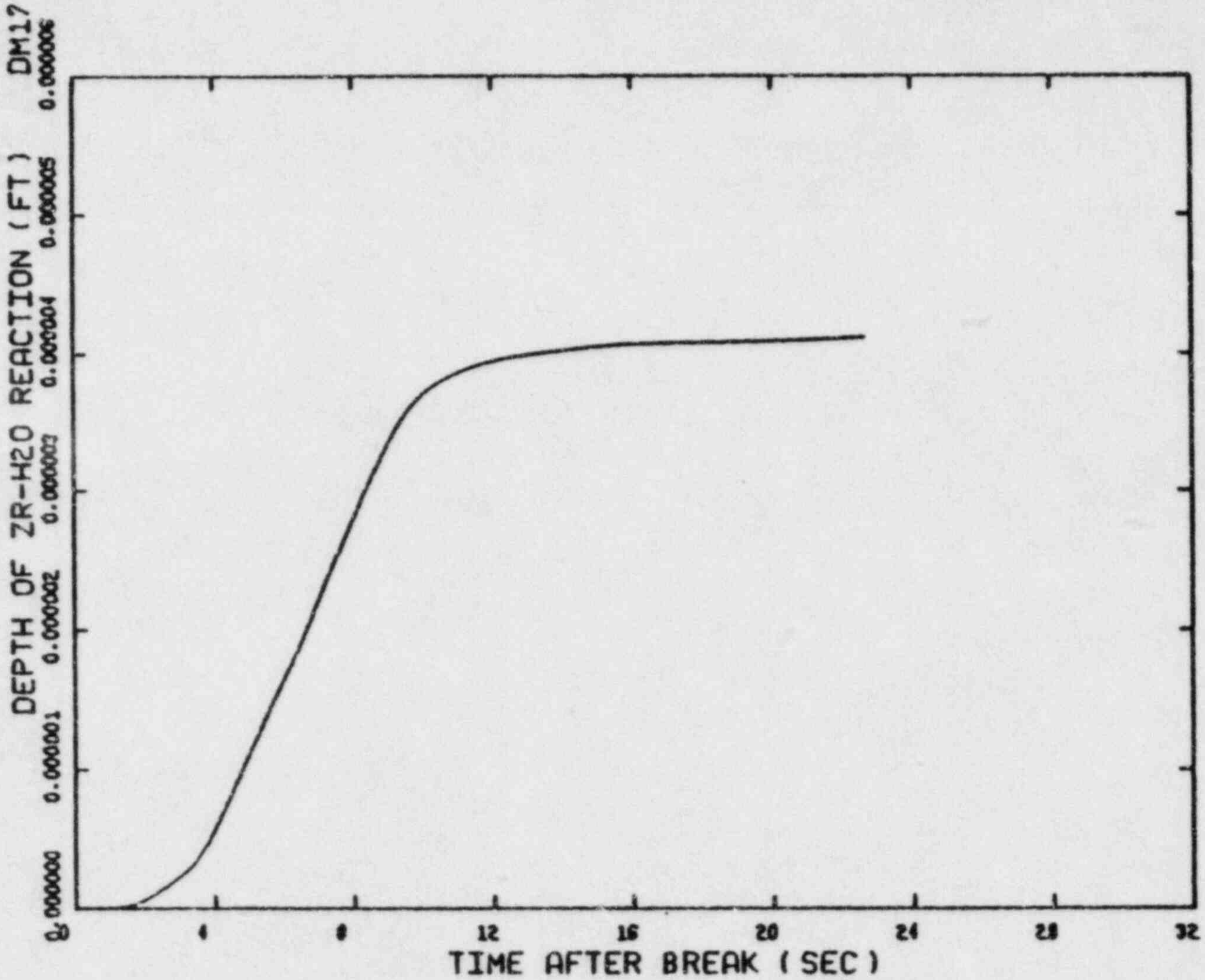

Figure 2.13 Depth of Metal-Water Reaction, 0.4 DECLG Break, 0-15,000 MWD/MTM

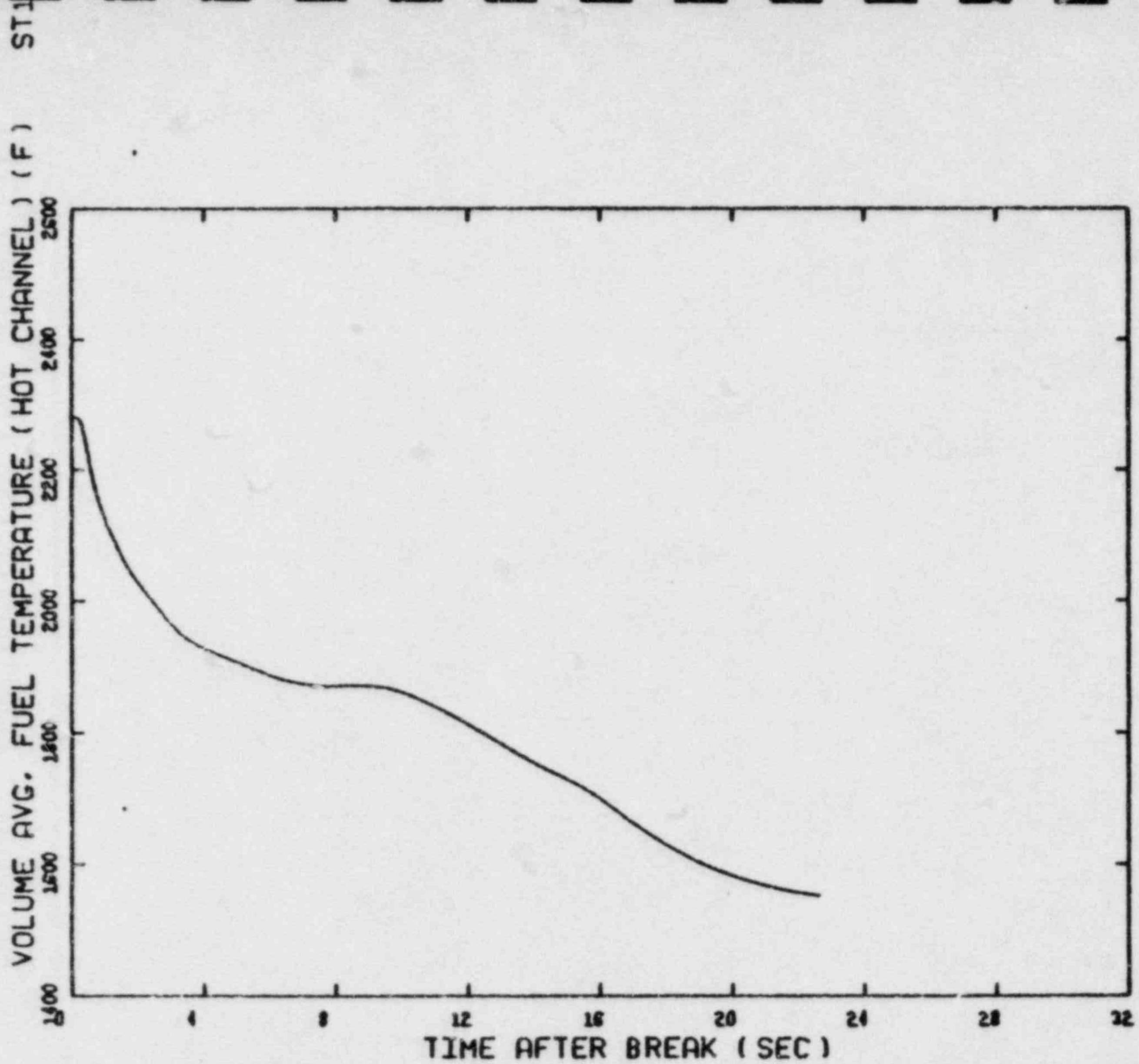

Figure 2.14 Hot Channel Average Fuel Temperature, 0.4 DECLG Break, 0-15,000 MWD/MTM

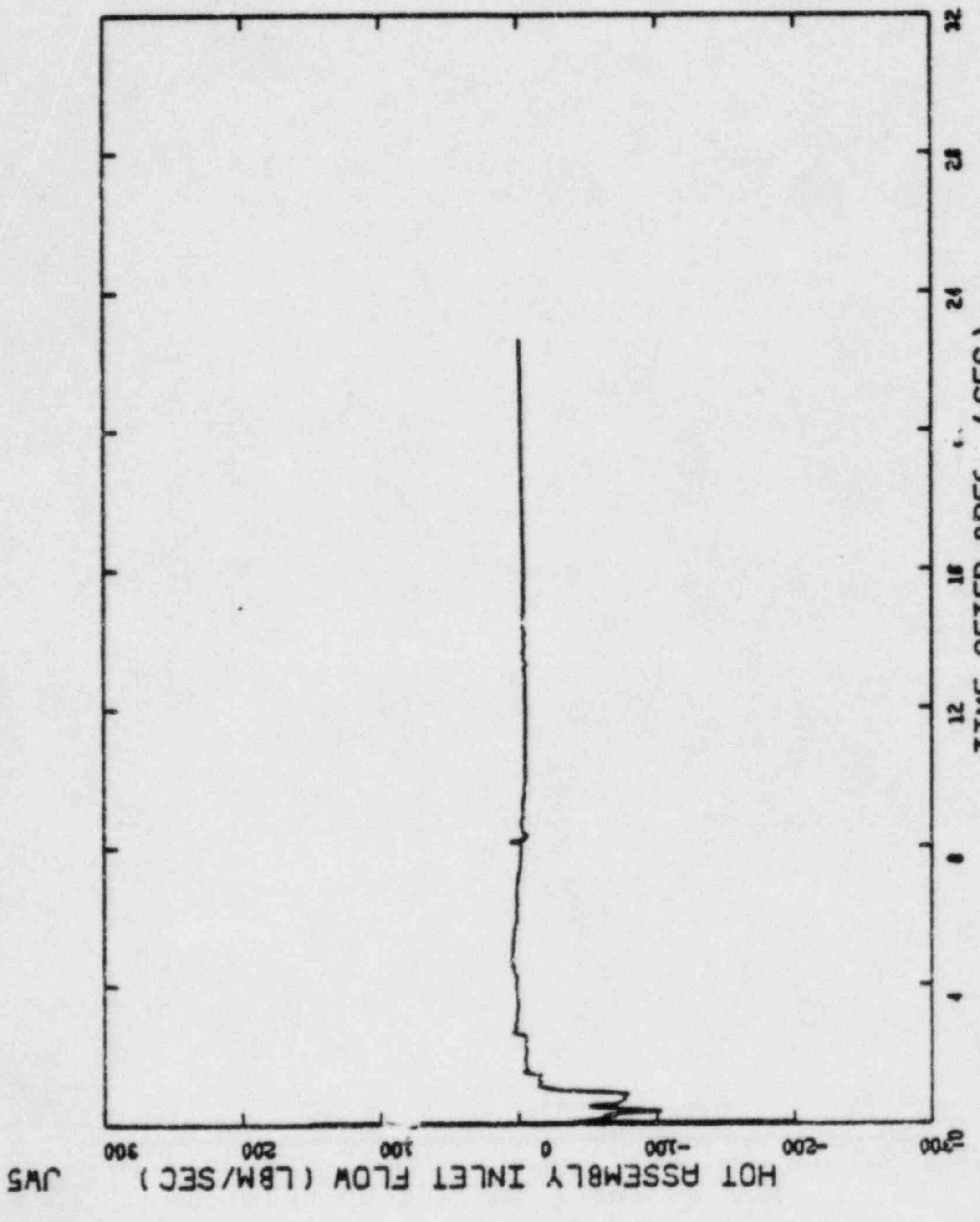

Figure 2.15 Hot Assembly Inlet Flow,
0.4 DECLG Break, 0-15,000 MWD/MTM

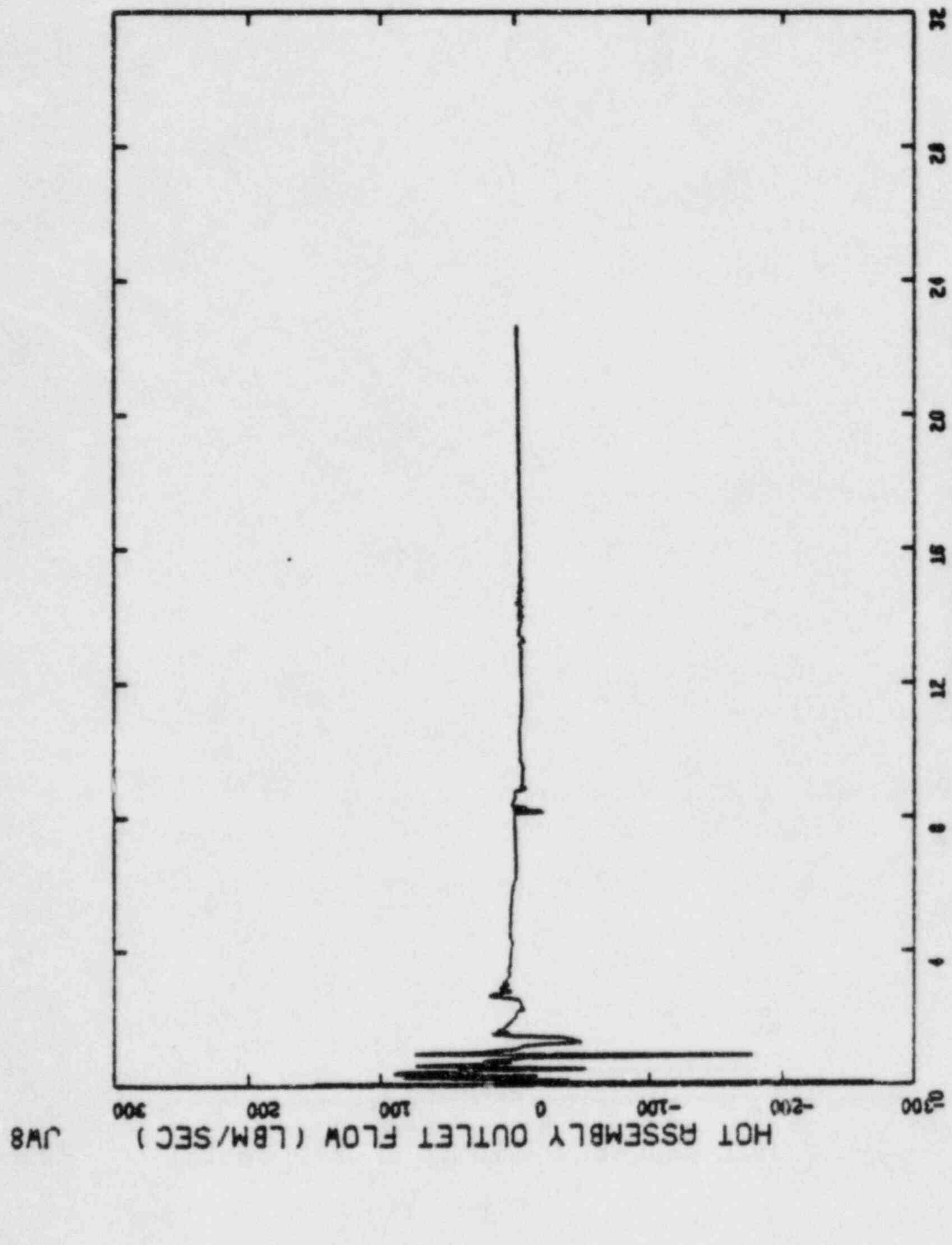

Figure 2.16 Hot Assembly Outlet Flow, 0.4 DECLG Break, 0-15,000 MWD/MTM

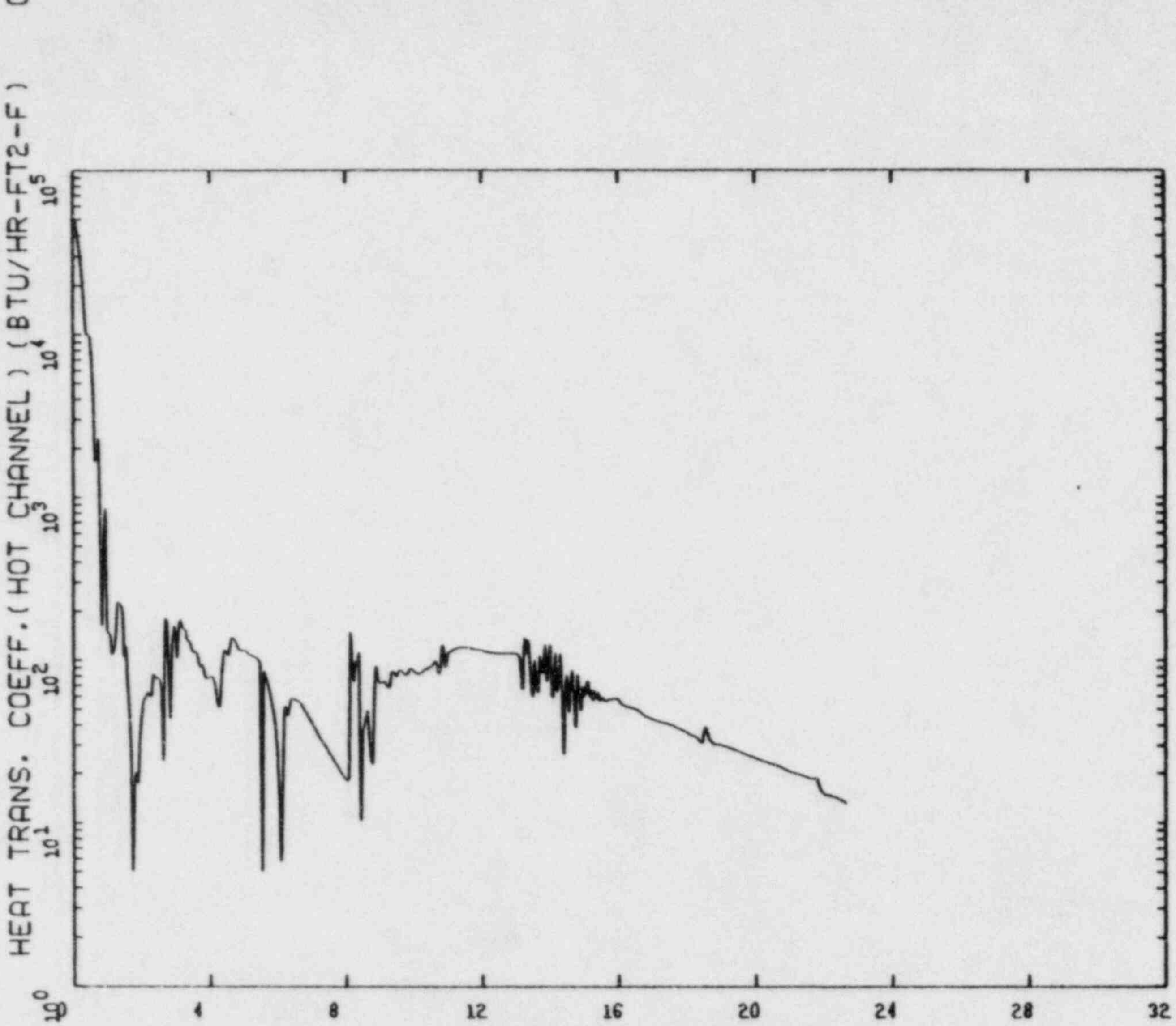

Figure 2.17 Hot Channel Heat Transfer Coefficient, 0.4 DECLG Break, 49,000 MWD/MTM

CR2

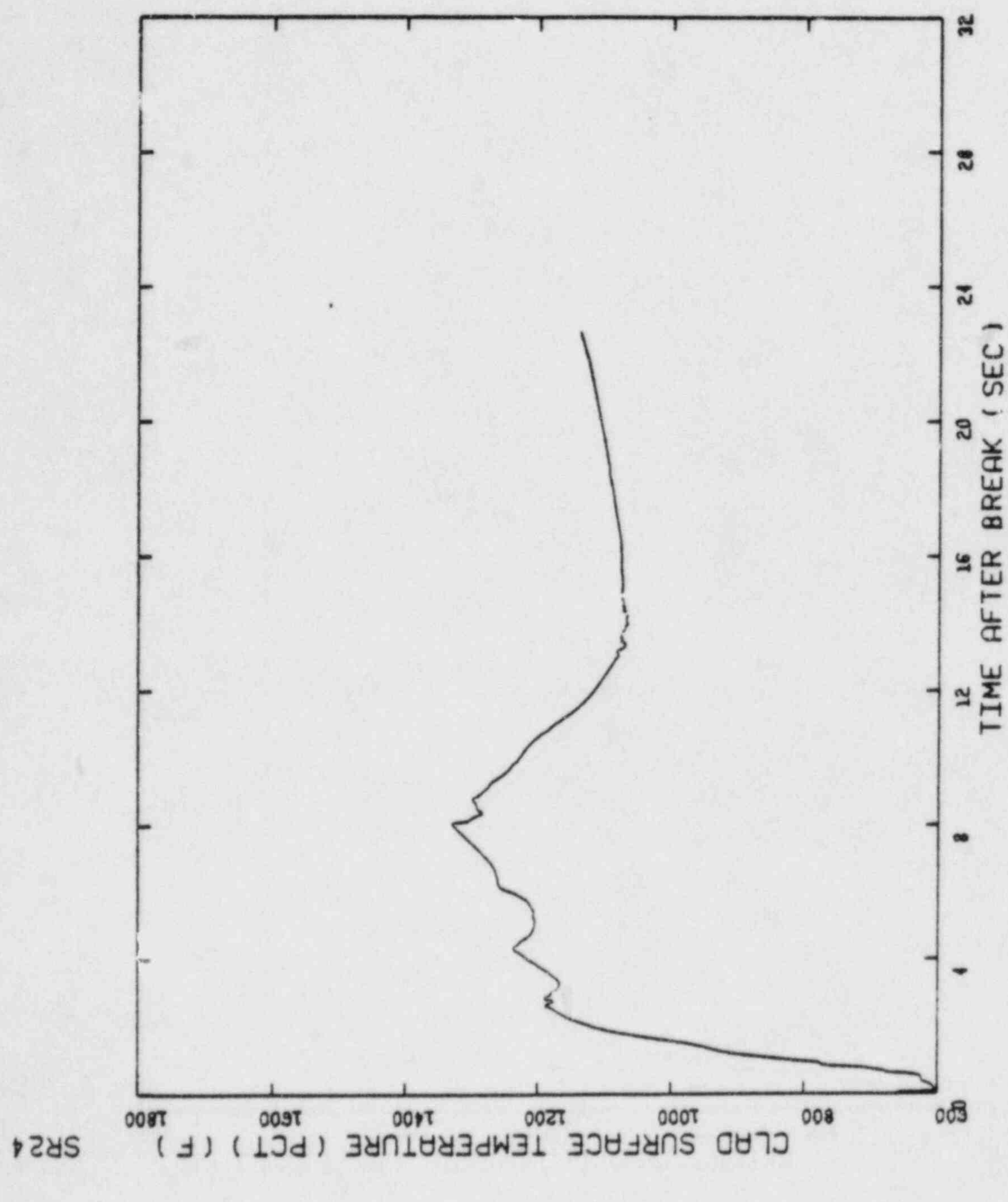

Figure 2.18 Clad Surface Temperature,
0.4 DECLG Break, 49,000 MWD/MTM

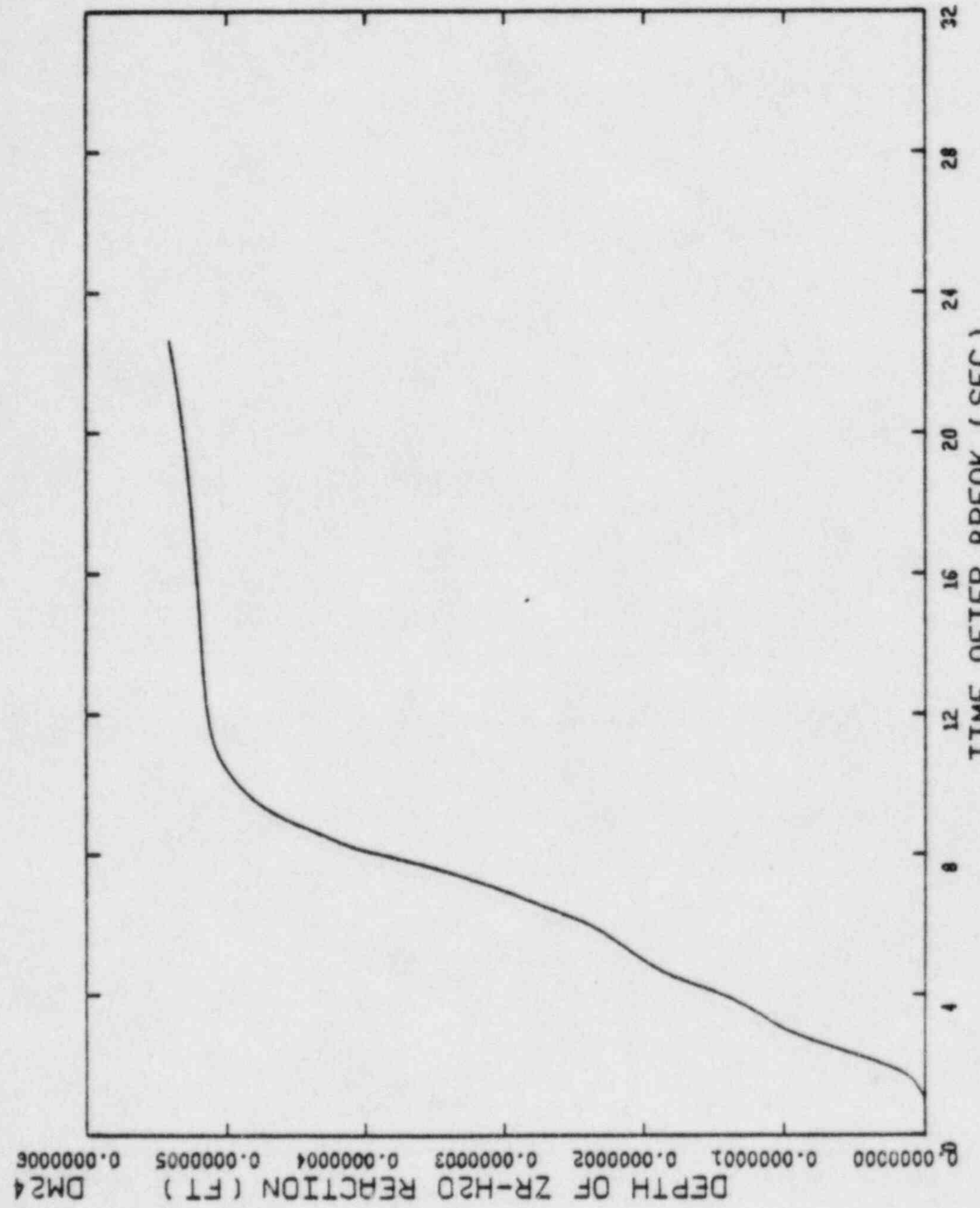

Figure 2.19 Depth of Metal-Water Reaction, 0.4 DECLG Break, 49,000 MWD/MTM

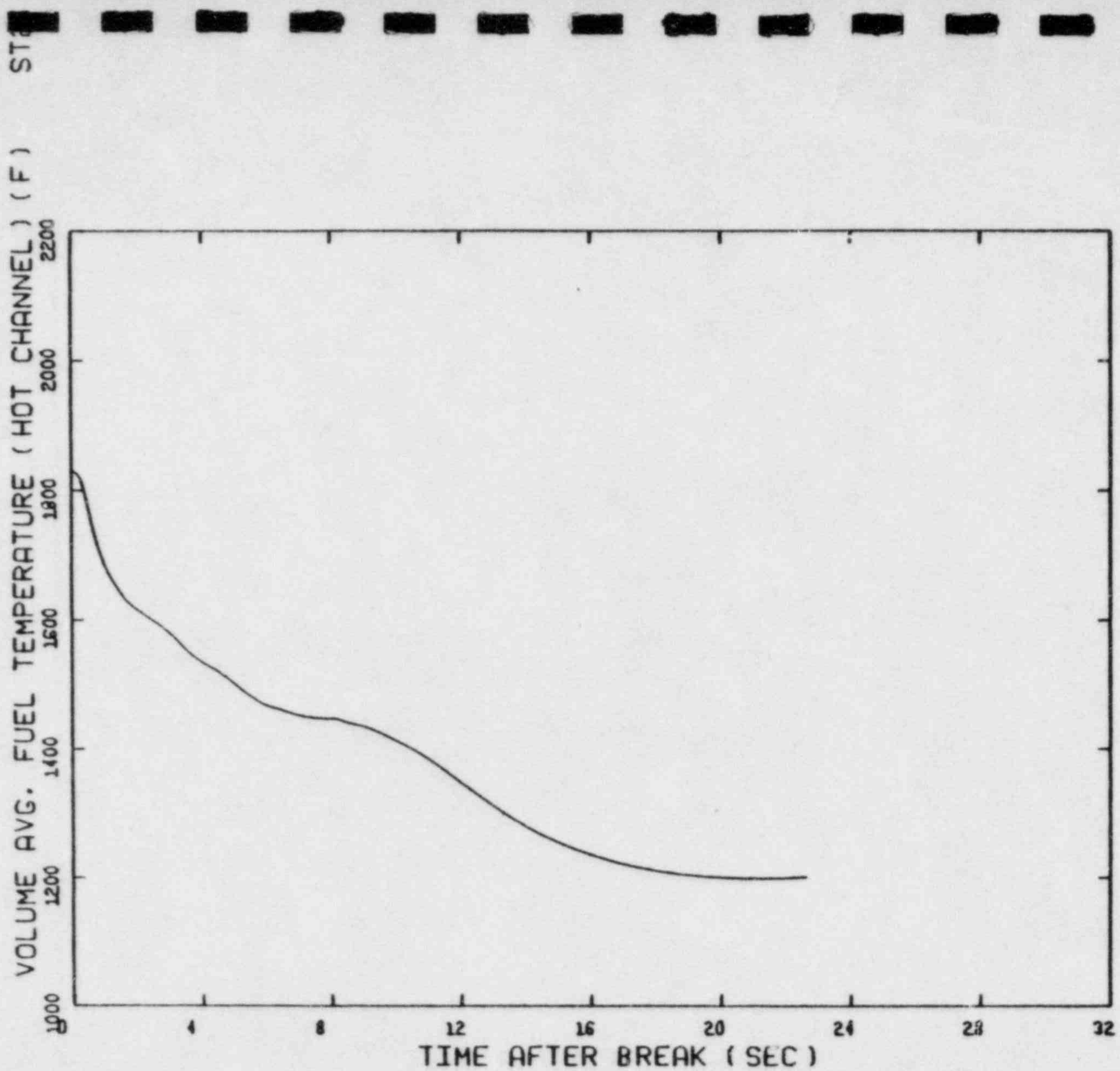

Figure 2.20 Hot Channel Average Fuel Temperature, 0.4 DECLG Break, 49,000 MWD/MTM

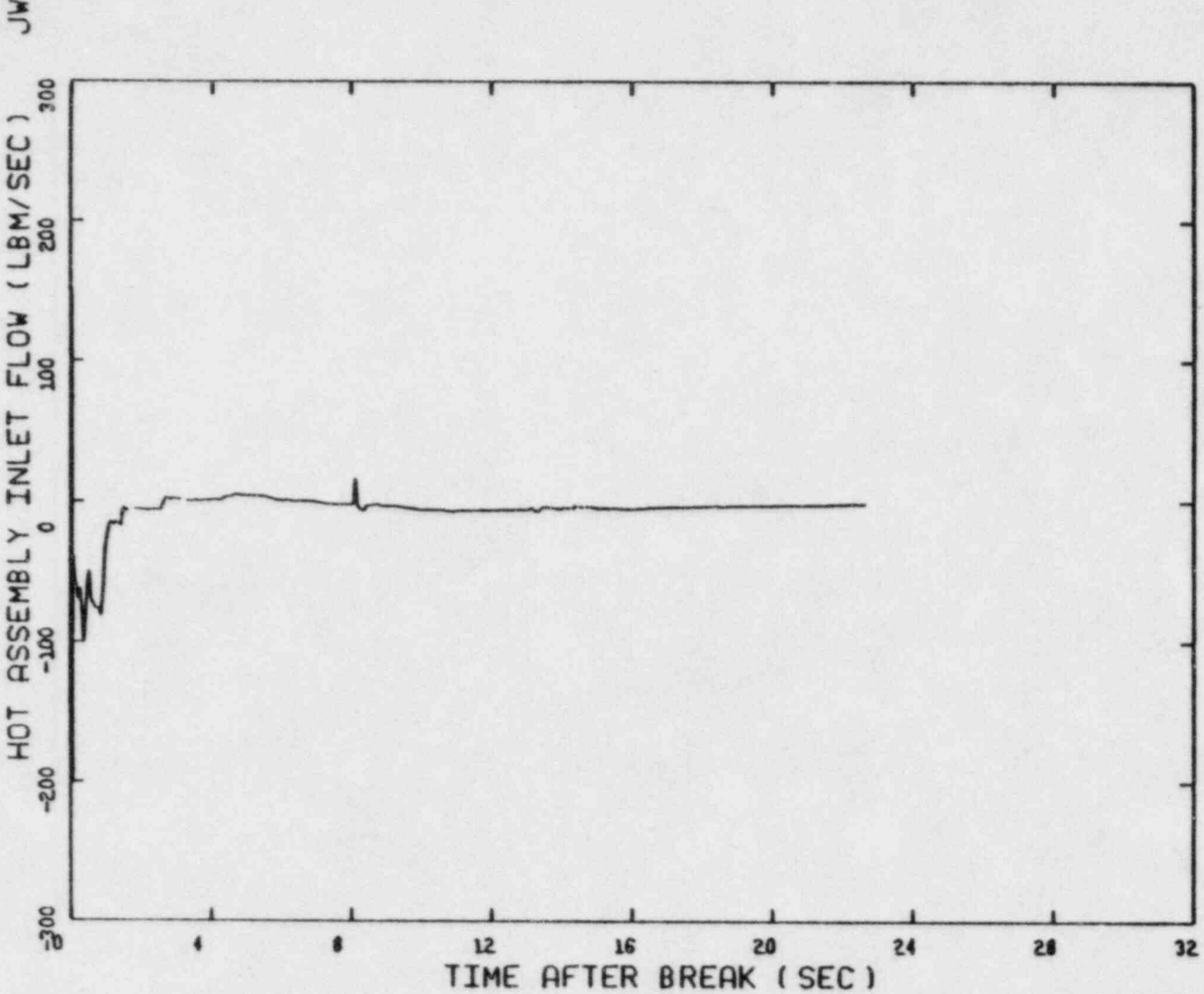

Figure 2.21 Hot Assembly Inlet Flow, 0.4 DECLG Break, 49,000 MWD/MTM

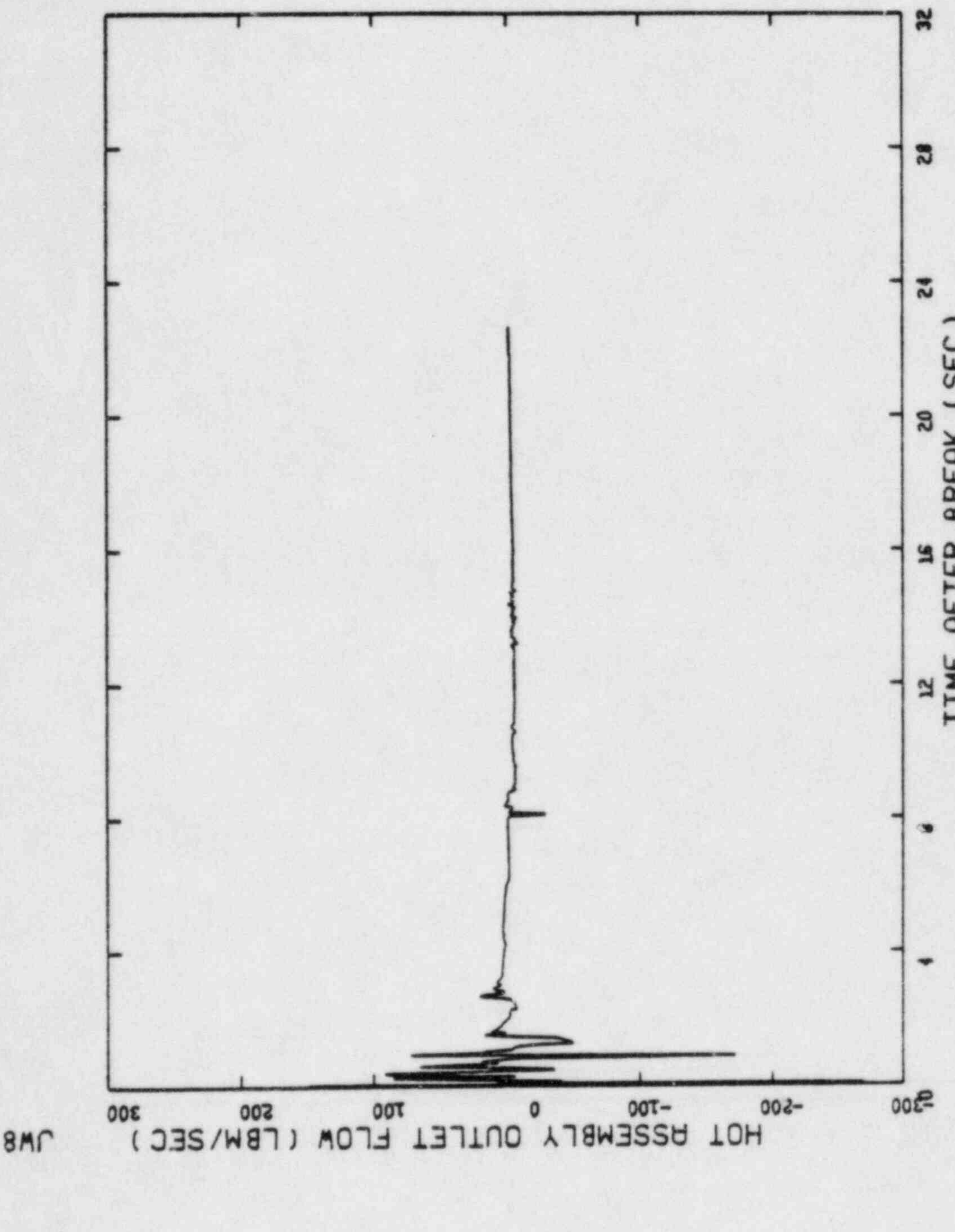

Figure 2.22 Hot Assembly Outlet Flow,
0.4 DECLG Break, 49,000 MWD/MTM

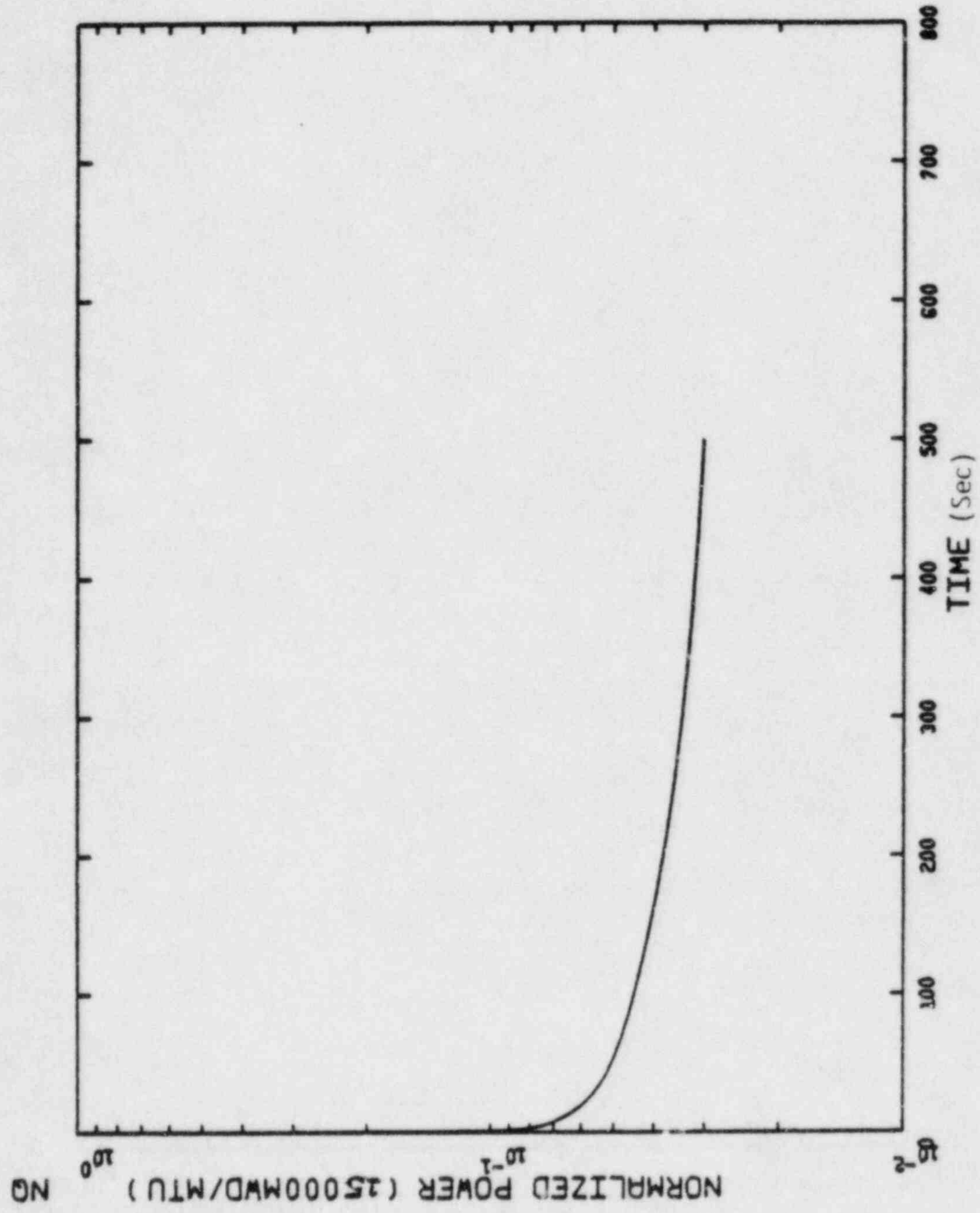

Figure 2.23 Normalized Power, 0.4 DECLG Break, 0-15,000 MWD/MTM

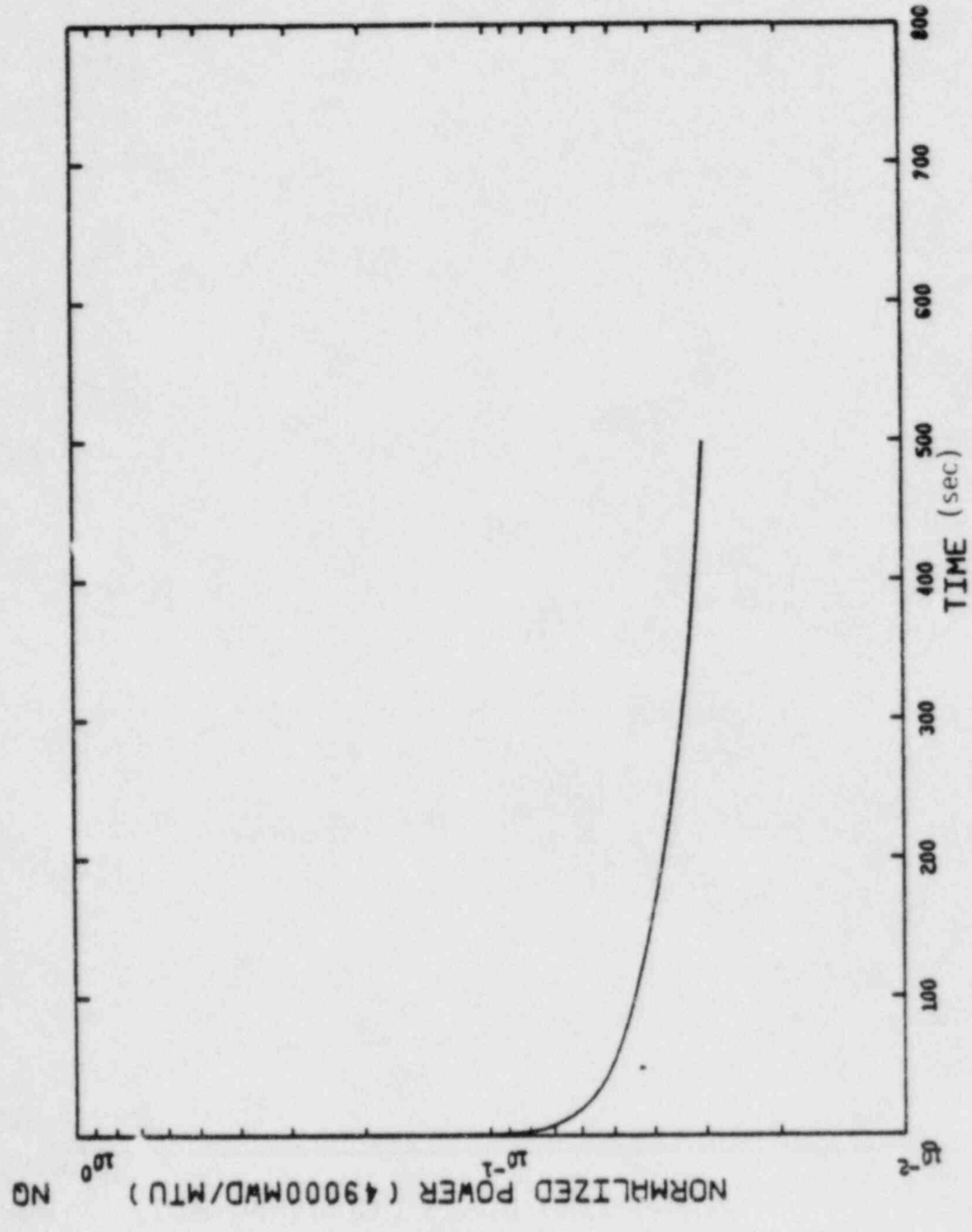

Figure 2.24 Normalized Power, 0.4 DECLG Break, 49,000 MWD/MTM

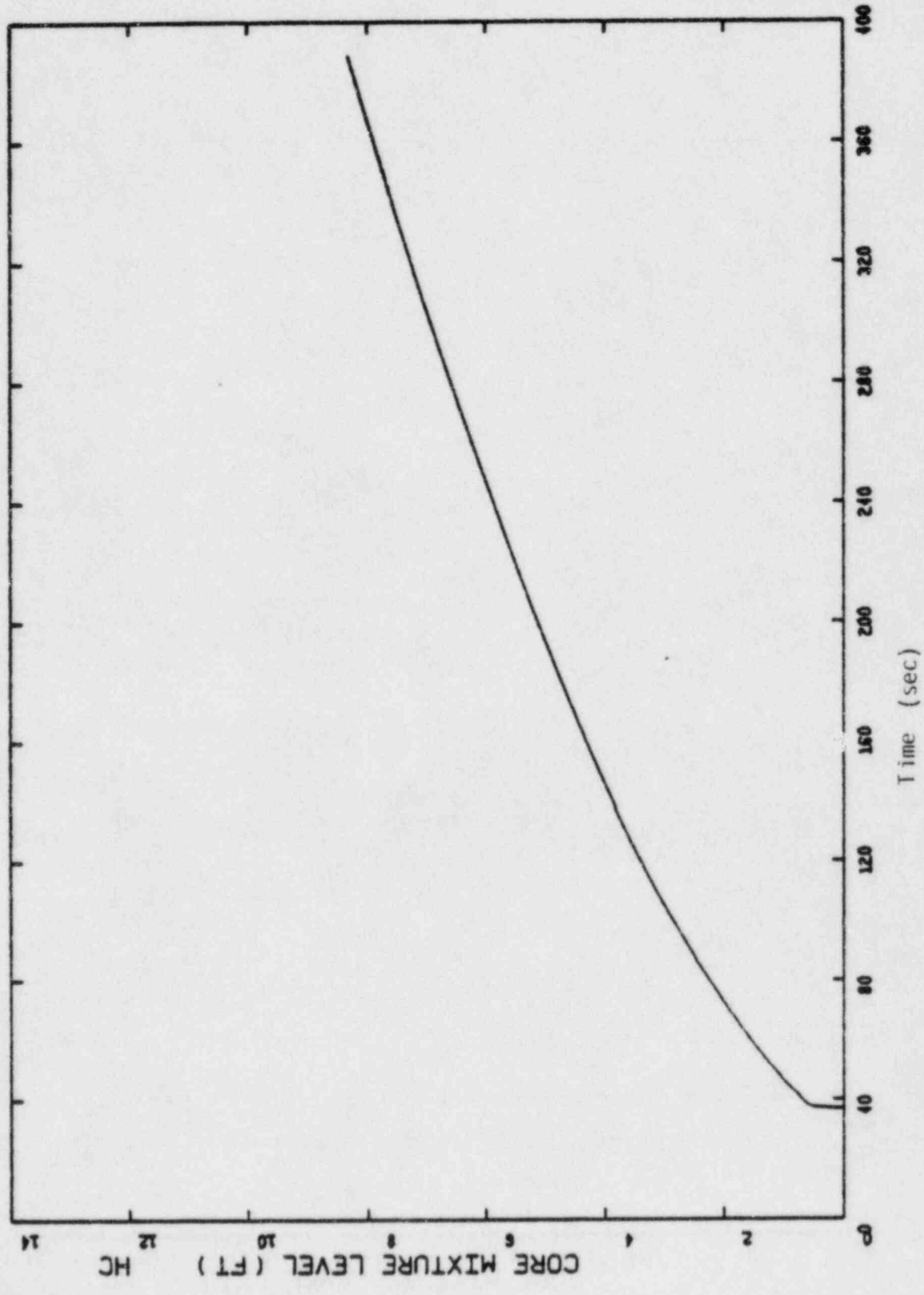

Figure 2.25 Reflood Core Mixture Level, 0.4 DECLG Break, 0-15,000 MWD/MTM

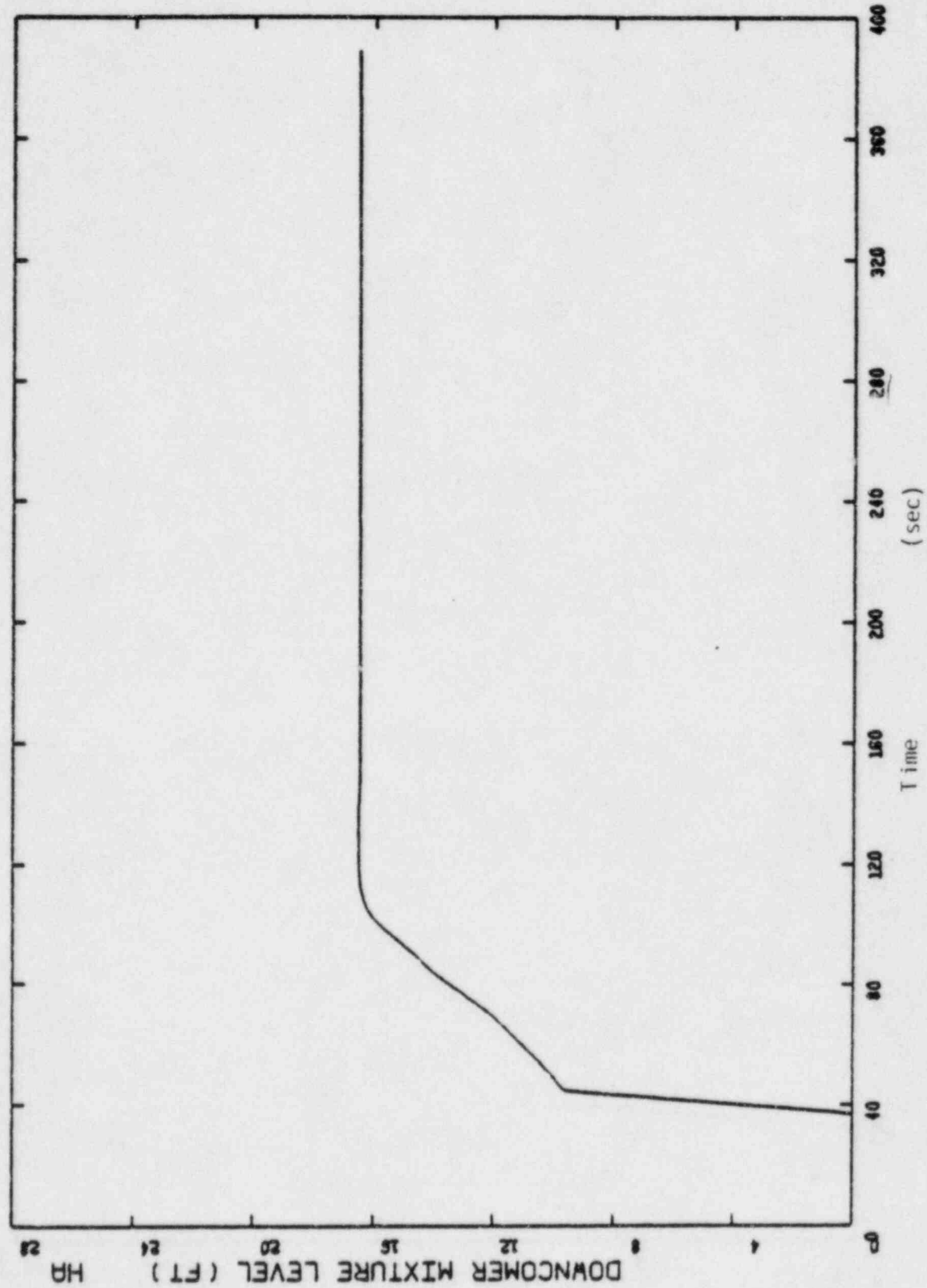

Figure 2.26 Reflood Downcomer Mixture Level,
0.4 DECLG Break, 0-15,000 MWD/MTM

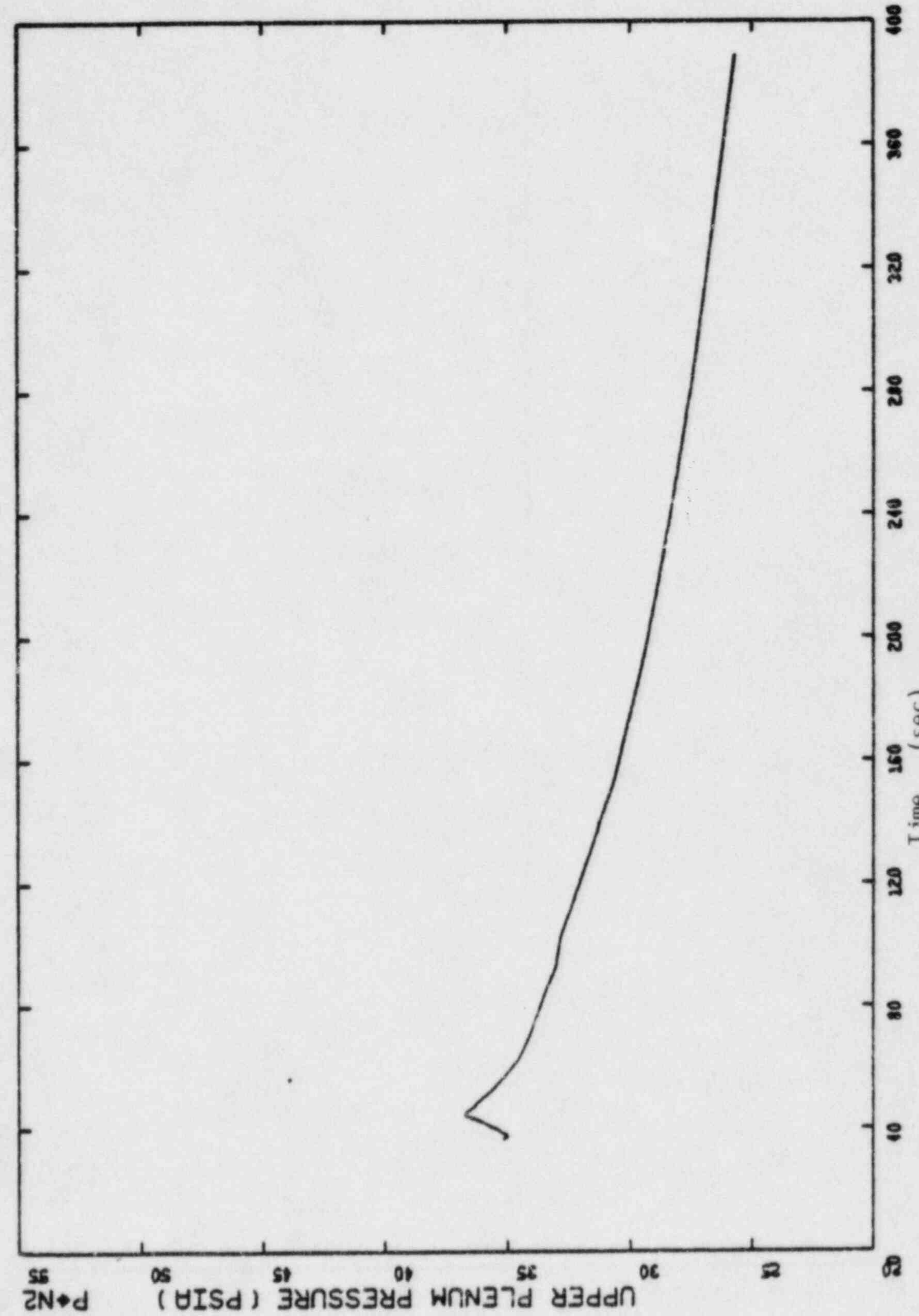

Figure 2.27 Reflood Upper Plenum Pressure, 0.4 DECLG Break, 0-15,000 MWD/MTM

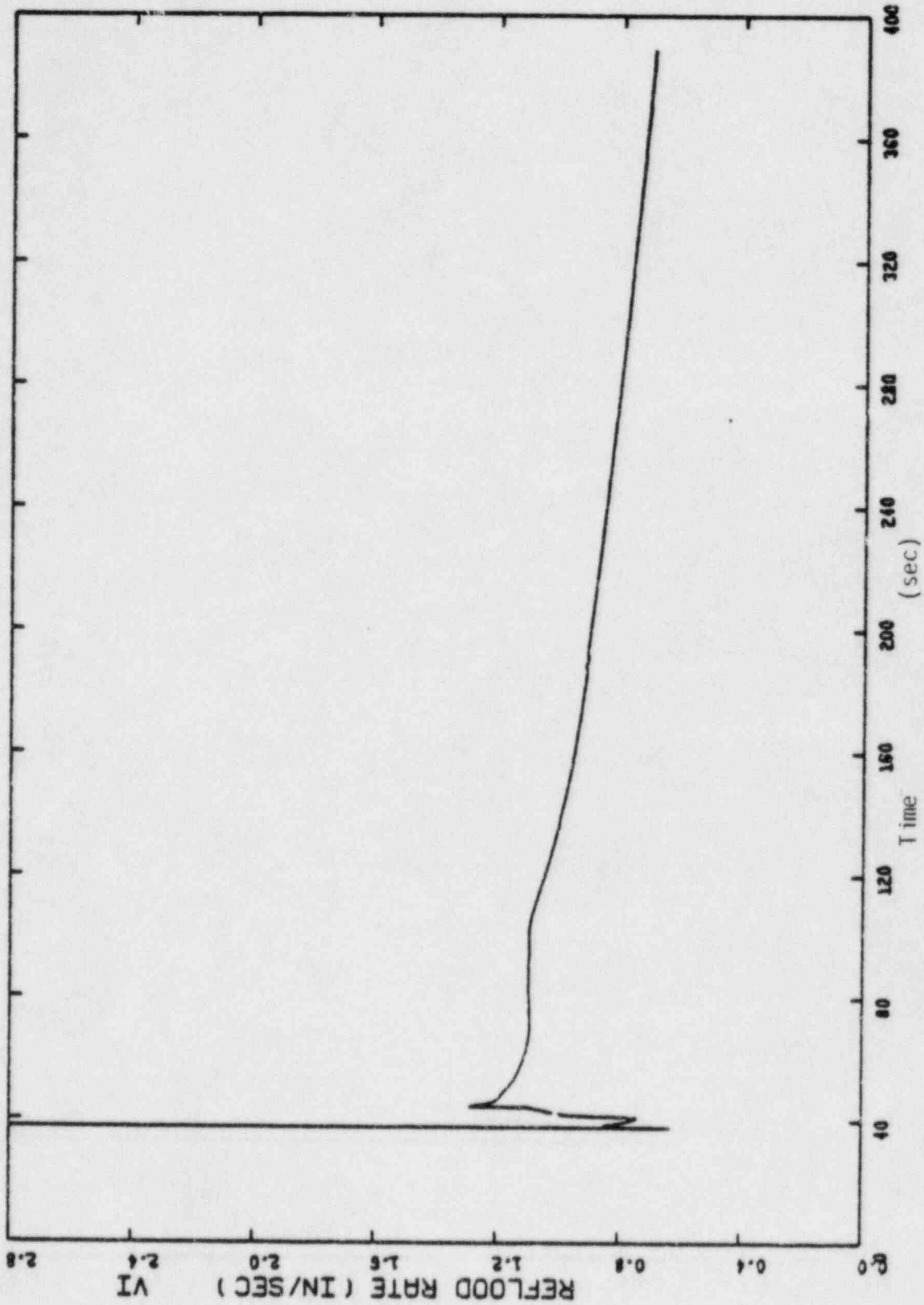

Figure 2.28 Core Flooding Rate,
0.4 DECLG Break, 0-15,000 MWD/MTM

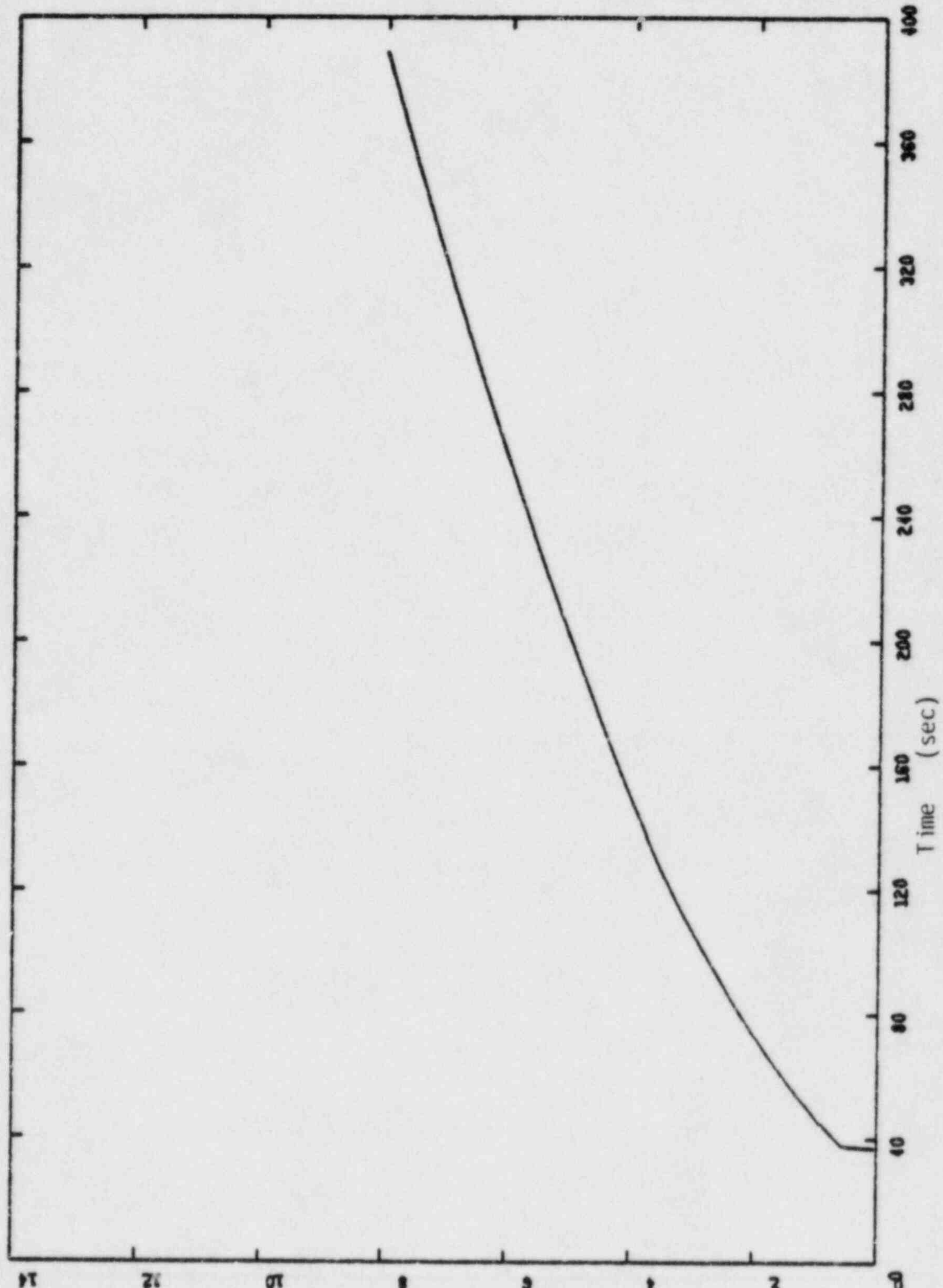

Figure 2.29 Reflood Core Mixture Level
0.4 DECLG Break, 49,000 MWD/MTM

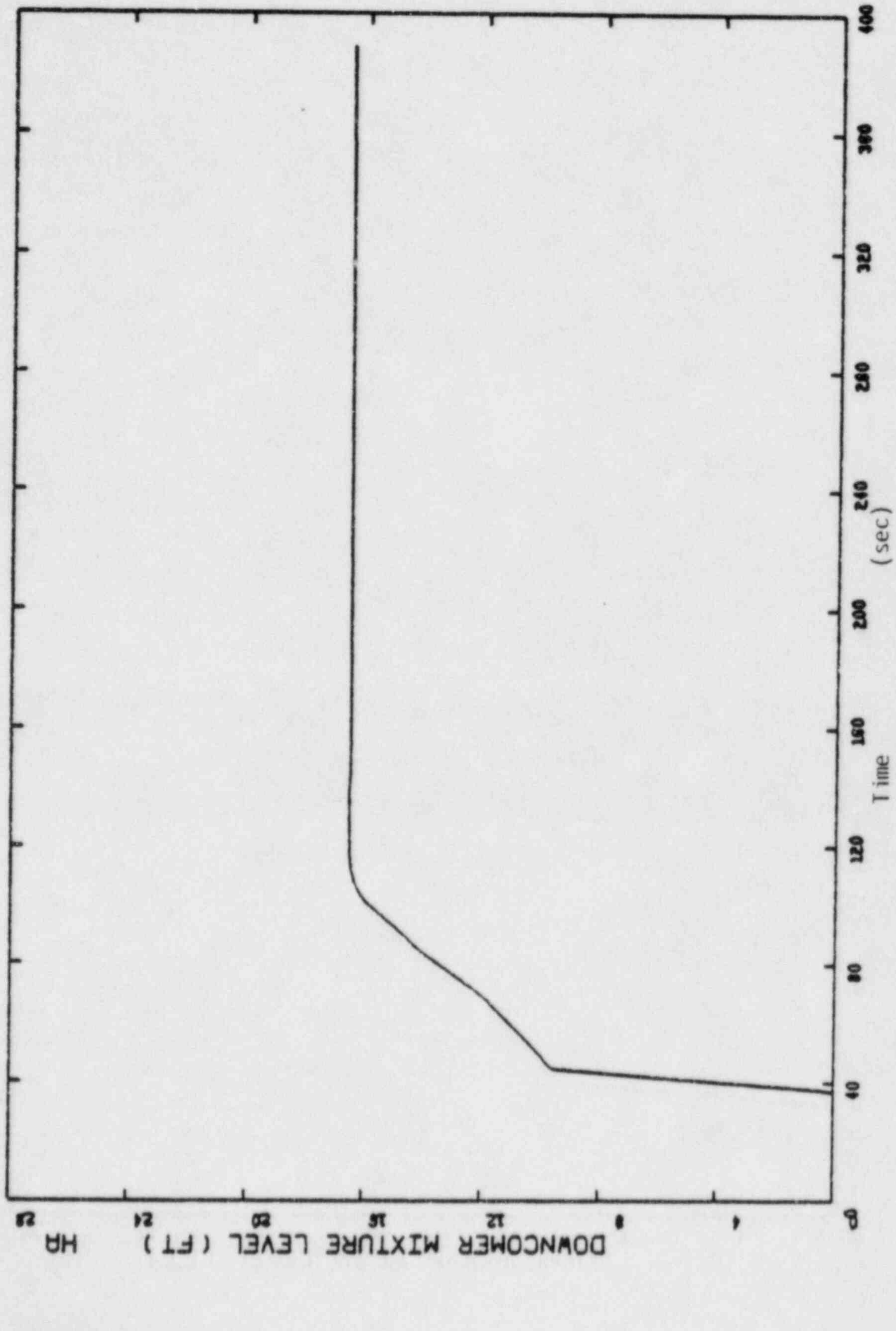

Figure 2.30 Reflood Downcomer Mixture Level, 0.4 DECLG Break, 49,000 MWD/MTM

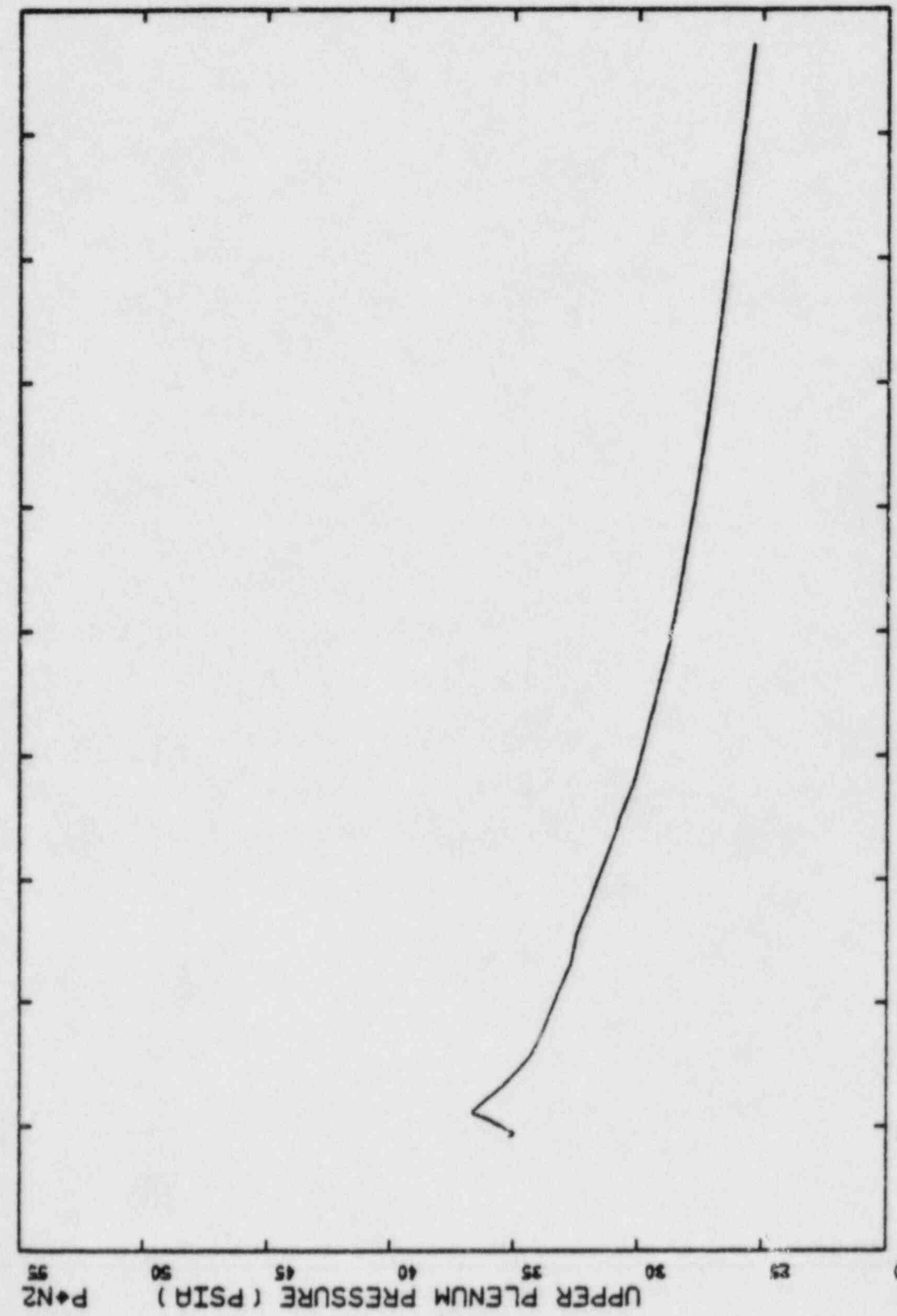

Figure 2.31 Reflood Upper Plenum Pressure, 0.4 DECLG Break, 49,000 MWD/MTM

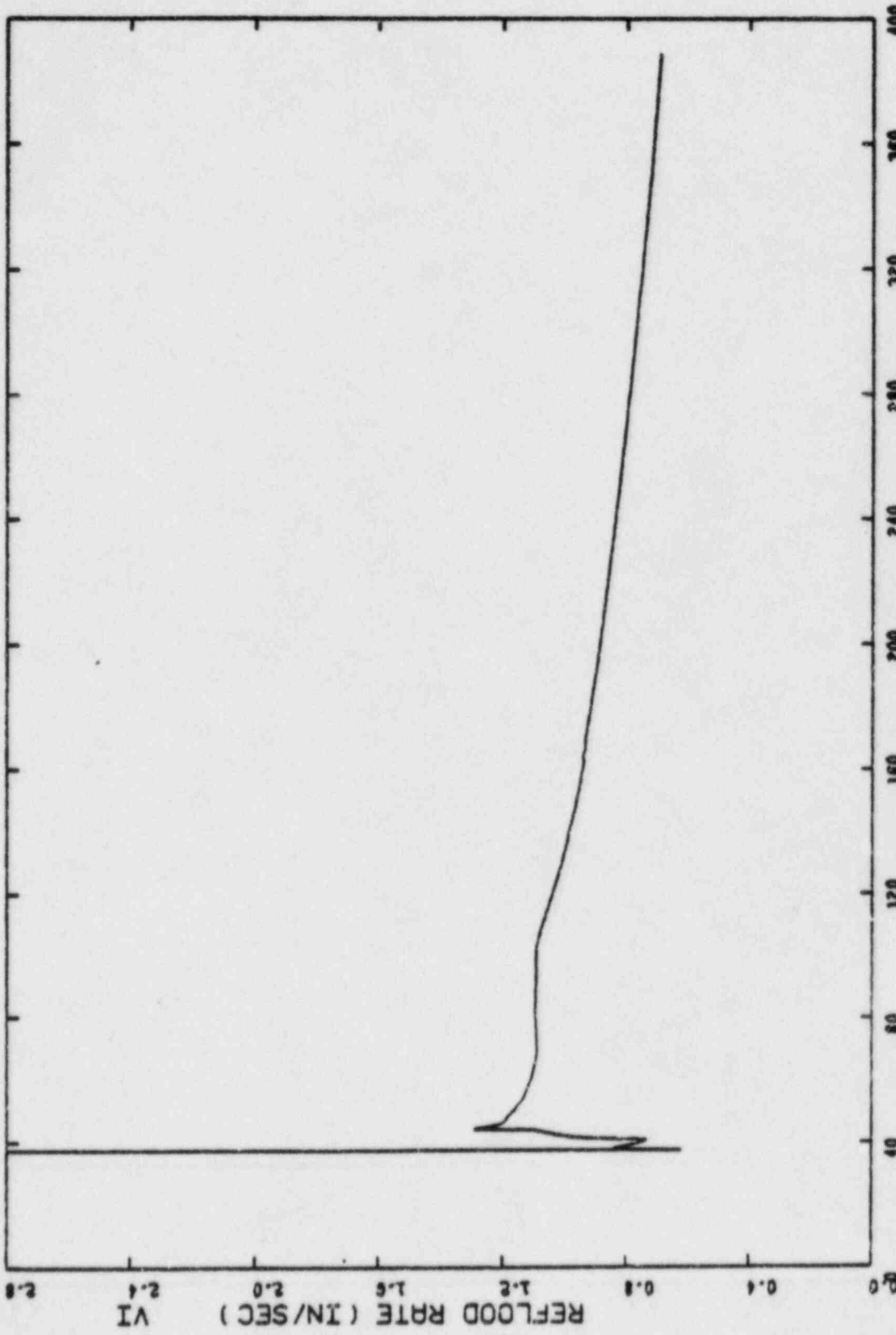

Figure 2.32 Core Flooding Rate, 0.4 DECLG Break, 49,000 MWD/MTM

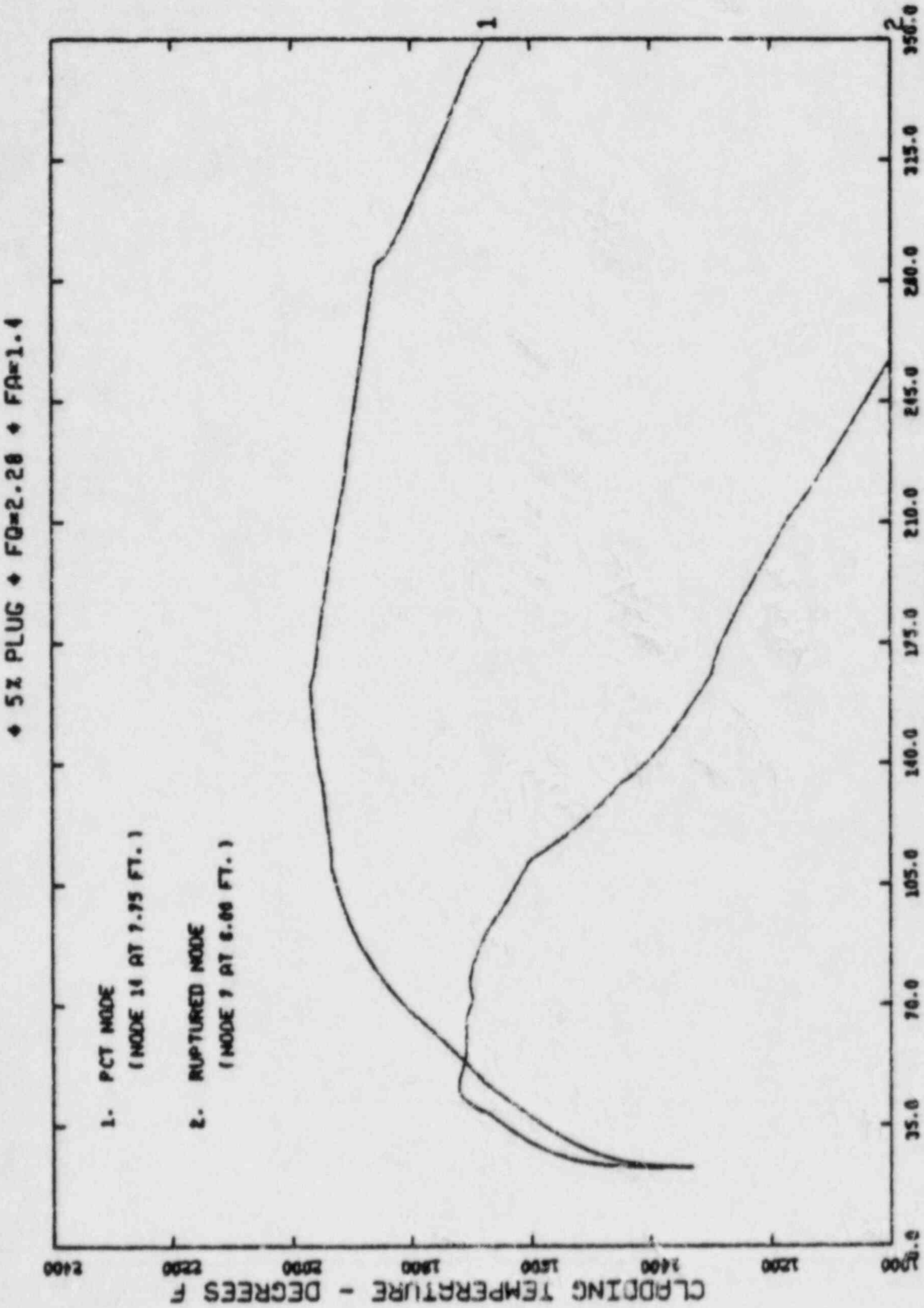

Figure 2.33 TOODEE2 Cladding Temperature vs. Time, 0.4 DECLG Break, 0-15,000 MWD/MTM

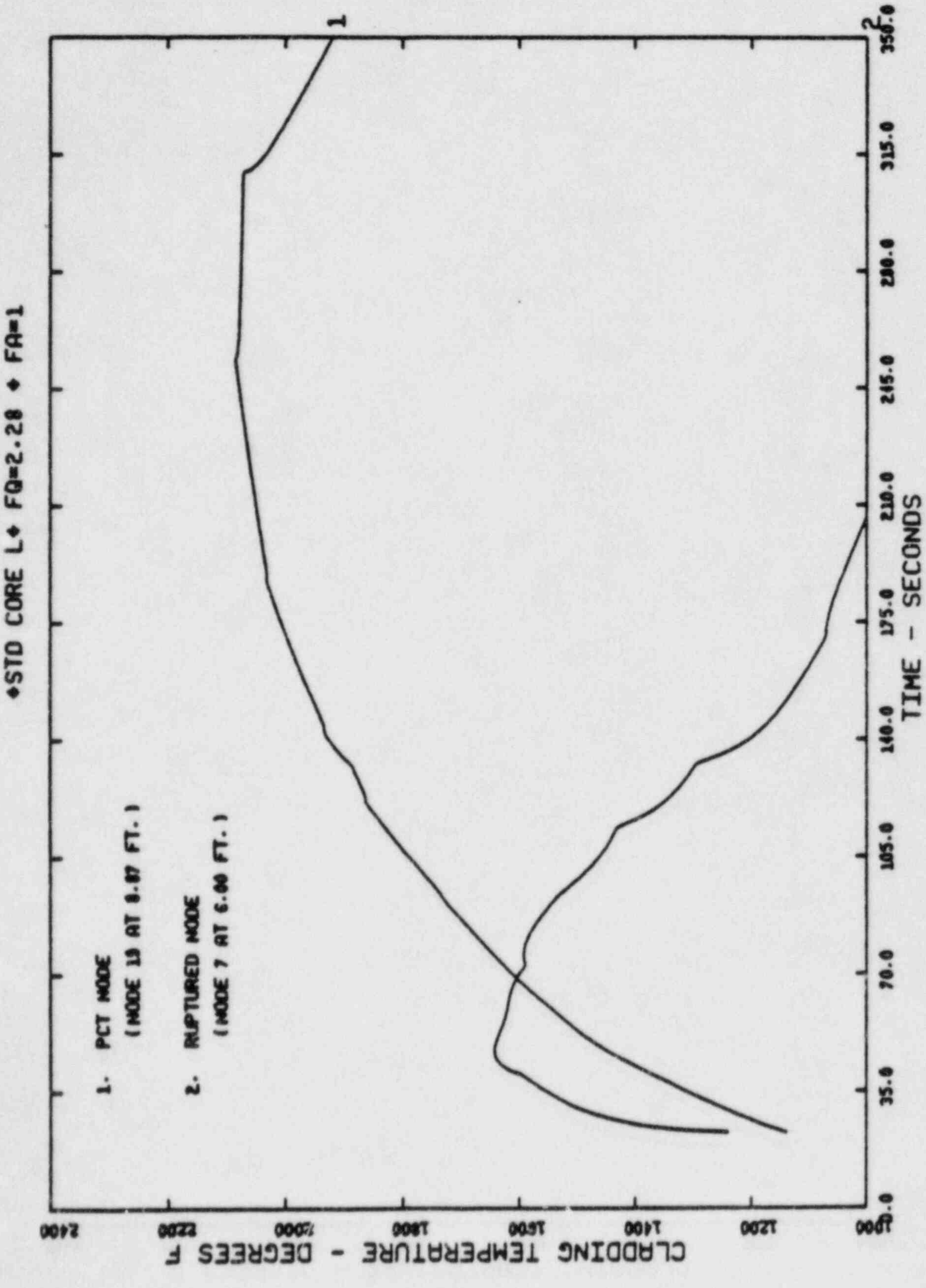

Figure 2.34 TOODEE2 Cladding Temperature vs. Time, 0.4 DECLG Break, 49,000 MWD/MTM

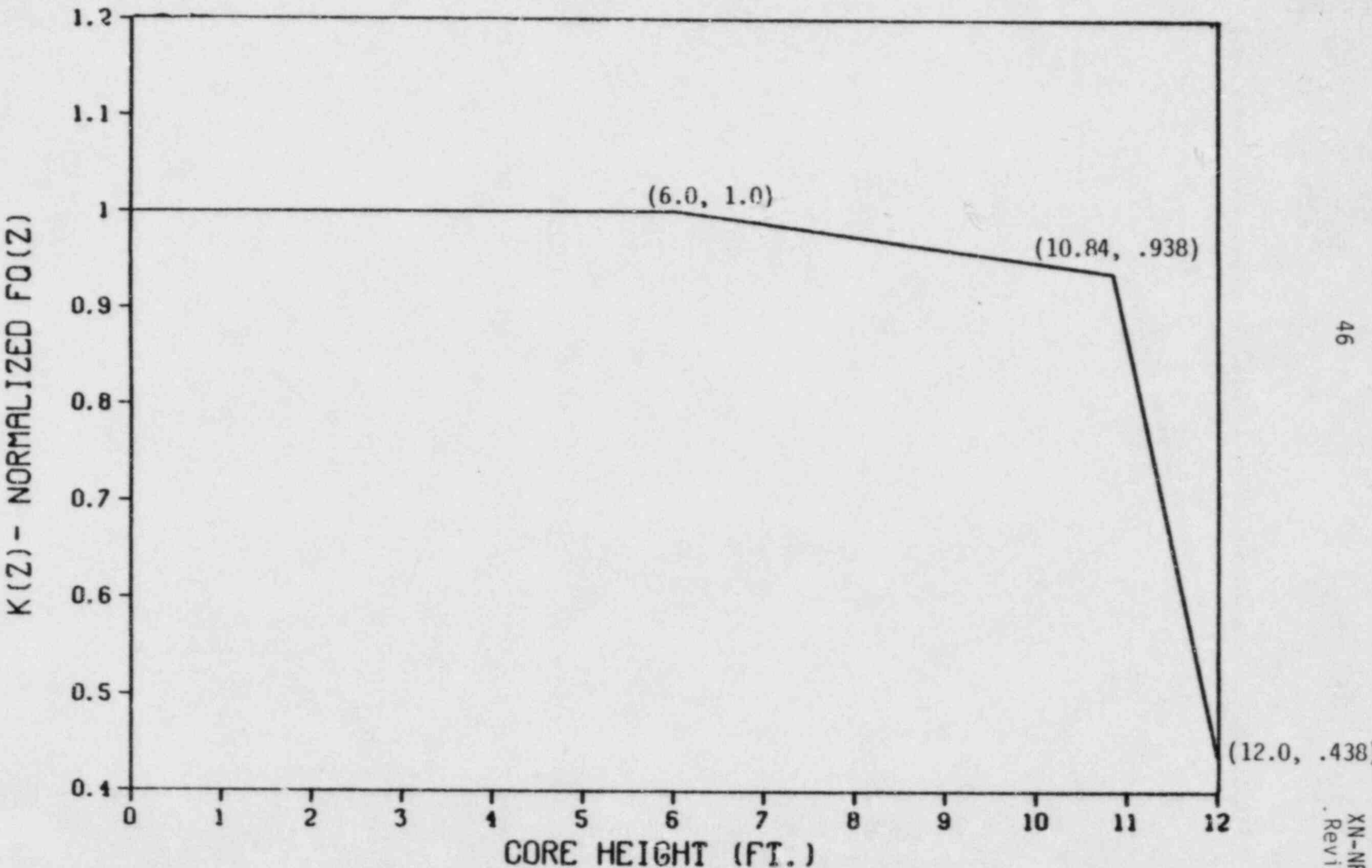

Figure 2.35 Hot Channel Factor Normalized Operating Envelope,

F_Q = 2.28 with $F_{\Delta H}^T$ = 1.55, K(Z) Function

3.0 RADIOLOGICAL CONSEQUENCES

3.1 SUMMARY

This section describes the application of the radiological release methodology discussed in Reference 18 to high burnup fuel in the Kewaunee nuclear power plant. The analysis was performed to conservatively envelop operation to a maximum assembly average exposure of 49,000 MWD/MTM.

Results show that the whole body and thyroid doses received from the postulated loss-of-coolant accident (LOCA) and fuel handling accident (FHA) for high burnup fuel are conservatively calculated to be well below values prescribed by NRC guidelines, 10 CFR 100. Calculated thyroid and whole body doses received from LOCA and fuel handling accidents are summarized in Table 3.1.

3.2 MODEL APPLICATION

To evaluate the radiological consequence of accidents involving high burnup fuel, two postulated accident scenarios are considered. The first accident considered is a postulated double-ended primary coolant pipe rupture LOCA. This accident was postulated to result in the depletion of the coolant inventory and extensive core damage. It is conservatively assumed that the LOCA resulted in cladding failure in all fuel rods throughout the reactor core, with the subsequent release of the core fission product inventory to the containment atmosphere.

The second accident considered is a fuel handling accident in which the gap fission product inventory of the highest burnup, highest power assembly is released as a result of faulty post-irradiation fuel assembly manipulation in the spent fuel pool. It is conservatively assumed that the cladding is breached on all fuel rods within the assembly such that the isotopic release represents the total assembly gap fission product inventory. The accident is assumed to occur 100 hours after reactor shutdown. This time represents the minimum delay between reactor shutdown and removal of fuel from the reactor core. Radioactive decay of nuclides is allowed to occur during this delay time.

To evaluate the impact of high burnup fuel on offsite radiological release relative to previously determined dose rates, ENC methodology requires analyses for: (1) a base case that models fuel operation consistent with prior analyses[3]; and (2) a case that models the proposed fuel operation. This is performed for both the LOCA and fuel handling accidents. Table 3.2 outlines the parameters used for both of the postulated accidents. Modeling assumptions and radiological dose results are reported in Reference 3 for both of the prior analyses. The LOCA analyses assumed reactor operation for 500 days at 1721.4 MWt. The reactor was modeled to operate at 1683 MWt for 850 days to obtain the desired average core exposure for high burnup fuel for the present LOCA analysis. The respective fuel assembly exposures for the base and high burnup cases for the fuel handling accident were 37,000 and 49,000 MWD/MTM. These elements provide the basis for determination of the radiological transport terms to be used in the high burnup analyses.

There are two primary calculational steps required to evaluate the radiological consequences of postulated accidents. The first step involves the determination of the isotopic composition and activity of the fission products available for release at a specific time. The ORIGEN[19] computer code is used to determine the isotopic composition and activity for both the existing and high burnup fuel designs under conditions simulating both a LOCA and a fuel handling accident. The LOCA simulation assumes a fission product mixture representative of end-of-cycle conditions, while the fuel handling simulation assumed a fission product mixture representative of peak powered assembly end-of-life conditions 100 hours after reactor shutdown. The RODEX2[5] thermal-mechanical fuel response code is used to determine fission gas fractions released to the fuel rod gap during the irradiation period for a limiting fuel assembly. The ANS-5.4[20-23] gas release model is used to evaluate release fractions.

The second step in evaluating the radiological consequences of the postulated accidents is the calculation of the biological impact of a release in terms of whole body and thyroid offsite doses. Included in the biological impact assessment is the transport of radionuclides from source to receptor. The transport characteristics which represent dispersion, deposition, and filtering efficiency do not change between base and high burnup cases. These characteristics are independent of fuel exposure being a property of the plant and its environs. These are determined in accordance with the described methodology.[18]

Both the whole body and thyroid dose calculations are dependent on specific isotopic composition and activity. Thyroid doses are calculated using the DACRIN-III[24] computer code. Whole body doses are calculated based on an energy weighted summation of all isotopes over the period of exposure. Whole body and thyroid doses were evaluated for 2 hour and 30 day exposure times for the LOCA case and 2 hour exposure for the fuel handling accident. In both accidents, the 2 hour dose was calculated at the site boundary, while the 30 day dose resulting from a LOCA was evaluated at the Low Population Zone boundary in accordance with 10 CFR 100 guidelines.

Decontamination factors (DF) decrease the isotopic inventory available for release to the atmosphere. According to the methodology outlined in Reference 3 for the LOCA evaluation, 25% of the halogen gases and 100% of the noble gases are assumed to be released to the atmosphere. For the fuel handling accident, 0.2% of the halogen gases and 100% noble gases are assumed to be released to the atmosphere. Decontamination factors for other elements are taken from Reference 25.

Biological doses are calculated using the following equation:

$$D_{HB} = (D_{FSAR})(R) \qquad (3.2.1)$$

where D_{HB} = dose received from high burnup fuel for a postulated event

D_{FSAR} = dose reported for prior analysis

R = ratio accounting for the change in dose due to high burnup relative to the dose for the base case fuel

Each ratio is evaluated for 2 hour and 30 day exposures following a LOCA event and for a 2 hour exposure due to a fuel handling accident. Doses for base case fuel are given in Reference 3 for LOCA and fuel handling accidents. Tables 3.3 and 3.4 indicate the isotopes and isotopic abundance considered in both the present and reference analyses for LOCA and fuel handling events, respectively. The base case activities in these tables are as calculated for this analysis for the reference conditions using the present methodology.

3.3 RESULTS AND DISCUSSION

Results of ORIGEN evaluations for base case fuel and high burnup fuel at end-of-cycle conditions for the average core are given in Table 3.3. Table 3.3 shows the isotopes considered in the previous radiological release analysis along with the isotopes accounted for in the present analysis. The activities contained in the fuel matrix of each fuel rod were used in order to calculate thyroid and whole body doses that result from the postulated LOCA event.

Table 3.4 gives the gap activity for the base case and high burnup fuel assembly case evaluated 100 hours after reactor shutdown. These activities were used to determine the radiological consequences of a fuel handling accident. Also shown in Table 3.4 are the gap release fractions for both fuel designs.

Table 3.5 contains the value of the dose proportionality factor defined in Equation 3.2.1 for both the LOCA and fuel handling accidents. The thyroid dose factor for the LOCA event is slightly less than 1.0 indicating that the 2 hour and 30 day thyroid dose received as a result of a LOCA involving high burnup fuel would be slightly less compared to that of the base case fuel design. The concentration of Iodine, the primary constituent of the thyroid dose, is less for high burnup fuel relative to base case fuel due to a slightly smaller mass of Uranium in the core.

Whole body dose ratios given in Table 3.5 indicate that the whole body dose received from high burnup fuel assemblies after a LOCA event is higher than that for the base case fuel design. This is a direct consequence of the increased number of isotopes considered in the present analysis relative to the previous analysis. The 30 day LOCA whole body dose ratio is higher than the 2 hour dose ratio because of the decay of short-lived isotopes. The remainder is composed of long-lived isotopes that reflect the level of core burnup.

The dose proportionality ratios for the fuel handling accident are also given in Table 3.5. The ratios for both the thyroid and whole body doses are significantly higher than corresponding LOCA dose ratios. This is due to differences in fission gas release fractions between high burnup and base case fuel designs.

Using the ratios defined in Equation 3.2.1 and quantified in Table 3.5, biological doses of high burnup fuel can be calculated for postulated LOCA and fuel handling accidents. For means of comparison, Table 3.1 shows dose results for the referenced case.[3] Table 3.1 shows the 2 hour thyroid and whole body doses received after a LOCA event are 10.9 and 1.8 rem, respectively; the 30 day thyroid and whole body doses are 3.8 and 1.8 rem, respectively; the 2 hour doses received subsequent to a fuel handling accident in the auxiliary building are 8.3 and 1.7 rem, respectively for high burnup fuel. In all cases the biological dose received from these accidents does not exceed 10 CFR 100 guidelines of 300 and 25 rem for thyroid and whole body doses, respectively.

This present application of the Exxon Nuclear radiological release methodology is considered to conservatively predict doses relative to the expected consequences of severe accidents. Conservatisms that were added include:

- Both the LOCA and fuel handling accidents assumed 100% fuel rod failure in the core and fuel assembly, respectively.
- The failed assembly modeled in the fuel handling accident analysis was assumed to be the peak power limiting assembly.
- The ANS 5.4 fission gas release model was used to conservatively calculate gas release fractions.
- A conservatively bounding power history was used to maximize fission gas release to the pellet-clad gap.
- Reduction in concentration of soluble Iodine due to containment spray was conservatively neglected.

Table 3.1 Thyroid and Whole Body Doses for Postulated LOCA and Fuel Handling Accidents

Event	Exposure Time	Organ	Dose (rem) Base Case Fuel	Dose (rem) High-Burnup Fuel
* LOCA	2 hr	Thyroid	11.0	10.9
		Wh. Body	1.6	1.8
* LOCA	30 day	Thyroid	3.8	3.8
		Wh. Body	1.2	1.8
** FHA in Aux. Bldg.	2 hr	Thyroid	4.0	8.3
		Wh. Body	0.35	1.7

10 CFR 100 Dose Guidelines: Thyroid Dose = 300 rem
Wh. Body Dose = 25 rem

* Based on core fission product inventory.
** Based on gap fission product inventory.

Table 3.2 Operating Parameters for Postulated LOCA and Fuel Handling Accidents

	Base Case Fuel	High-Burnup Fuel
Core Power (MWt)	1721.4	1683*
Average EOC Core Exposure(MWD/MTU)	18070	30000
Max. Assembly EOL Exposure(MWD/MTU)	37000	49000

* 1650 MWt times 2% power uncertainty.

Table 3.3 Core Activities for LOCA Radiological Release Analysis

Isotope	Core Activity (Ci) Base Case Fuel **	High-Burnup Fuel	DF +
I-129	*	1.60	0.25
I-131	4.81E+07	4.84E+07	0.25
I-132	6.91E+07	6.87E+07	0.25
I-133	9.34E+07	9.04E+07	0.25
I-134	1.07E+08	1.03E+08	0.25
I-135	8.34E+07	8.00E+07	0.25
Total I	4.01E+08	3.91E+08	
Kr- 85	3.33E+05	5.02E+05	1.00
Kr- 85m	1.24E+07	1.06E+07	1.00
Kr- 87	2.43E+07	2.04E+07	1.00
Kr- 88	3.52E+07	2.98E+07	1.00
Kr- 89	*	3.71E+07	1.00
Total Kr	7.22E+07	9.84E+07	
Xe-131m	*	3.50E+05	1.00
Xe-133	9.10E+07	8.99E+07	1.00
Xe-133m	2.25E+06	2.19E+06	1.00
Xe-135	1.93E+07	1.74E+07	1.00
Xe-135m	2.51E+07	2.42E+07	1.00
Xe-137	*	8.77E+07	1.00
Xe-138	*	8.45E+07	1.00
Total Xe	1.38E+08	3.06E+08	

Table 3.3 Continued,

Isotope	Base Case Fuel ** (Core Activity (Ci))	High-Burnup Fuel (Core Activity (Ci))	DF +
Cs-134	*	1.11E+07	0.20
Cs-134m	*	3.36E+06	0.20
Cs-135	*	1.08E+01	0.20
Cs-136	*	2.98E+06	0.20
Cs-137	*	4.70E+06	0.20
Cs-138	*	8.50E+07	0.20
Cs-139	*	8.36E+07	0.20
Cs-140	*	7.68E+07	0.20
Cs-141	*	5.47E+07	0.20
Cs-142	*	4.34E+07	0.20
Cs-143	*	2.23E+07	0.20
Total Cs	*	3.88E+08	
Te-125m	*	1.37E+05	0.0005
Te-127	*	3.99E+06	0.0005
Te-127m	*	8041E+05	0.0005
Te-129	*	1.88E+07	0.0005
Te-129m	*	3.19E+06	0.0005
Te-131	*	4.24E+07	0.0005
Te-131m	*	7.23E+06	0.0005
Te-132	*	6.64E+07	0.0005
Te-133	*	2.57E+07	0.0005
Te-133m	*	7.00E+07	0.0005
Te-134	*	9.08E+07	0.0005
Te-135	*	7.86E+07	0.0005
Total Te	*	4.08E+08	
Totals	6.11E+08	1.59E+09	

* Isotopes not used in the Kewaunee FSAR radiological release analysis.

\+ Decontamination factors for I, Kr and Xe are from Reference 3. Decontamination factors for Cs and Te are from Reference 25.

** Isotopic activity for the base case fuel was re-calculated with ENC methodology but using assumptions consistent with Reference 3.

Table 3.4 Fuel Assembly and Gap Activities for Fuel Handling Accident Radiological Release Analysis (100 hr after shutdown)

	Base Case Fuel**			High-Burnup Fuel			
Isotope	Assembly Activity (Ci)	Gap Fraction	Gap Activity (Ci)	Assembly Activity (Ci)	Gap Fraction	Gap Activity (Ci)	DF+
I-131	4.41E+05	.02083	9.19E3	4.59E+05	.02553	1.17E4	0.002
I-132	3.56E+05	.00666	2.37E3	3.70E+05	.00822	3.04E3	0.002
I-133	4.29E+04	.01316	5.65E2	4.37E+04	.01619	7.08E2	0.002
I-135	3.16E+01	.00748	2.36E-1	3.23E+01	.00923	2.98E-1	0.002
Total I	8.40E+05		1.21E+04	8.73E+05		1.54E+04	
Kr-85	4.84E+03	.0339	1.64E2	5.75E+03	.0412	2.37E2	1.00
Total Kr	4.84E+03		1.64E+02	5.75E+03		2.37E+02	
Xe-131m	*		*	4.74E+03	.02509	1.19E2	1.00
Xe-133	7.75E+05	.01004	7.78E3	7.95E+05	.01236	9.83E3	1.00
Xe-133m	1.18E+04	.01596	1.88E2	1.21E+04	.01960	2.37E2	1.00
Xe-135	1.42E+03	.00244	3.46E0	1.44E+03	.00301	4.33E0	1.00
Total Xe	7.88E+05		7.97E3	8.13E+05		1.02E+04	
Cs-134	*		*	2.10E+05	.054	1.13E4	0.20
Cs-136	*		*	4.11E+04	.04652	1.91E3	0.20
Cs-137	*		*	6.09E+04	.054	3.29E3	0.20
Total Cs	*		*	3.12E+05		1.65E+04	

Table 3.4 Continued,

Isotope	Base Case Fuel** Assembly Activity (Ci)	Base Case Fuel** Gap Fraction	Base Case Fuel** Gap Activity (Ci)	High-Burnup Fuel Assembly Activity (Ci)	High-Burnup Fuel Gap Fraction	High-Burnup Fuel Gap Activity (Ci)	DF+
Te-125m	*		*	2.15E+03	.20728	4.46E2	0.0005
Te-127	*		*	3.71E+04	.03243	1.20E3	0.0005
Te-127m	*		*	1.24E+04	.23917	2.97E3	0.0005
Te-129	*		*	2.62E+04	.01194	3.13E2	0.0005
Te-129m	*		*	4.07E+04	.23256	9.46E3	0.0005
Te-131	*		*	1.76E+03	.00725	1.28E1	0.0005
Te-131m	*		*	9.64E+03	.05475	5.28E2	0.0005
Te-132	*		*	3.59E+05	.08225	2.95E4	0.0005
Total Te	*		*	4.89E+05		4.44E4	
Totals	1.63E+06		2.02E04	2.49E+06		8.67E04	

* Isotopes not used in the Kewaunee FSAR radiological release analysis.

\+ Decontamination factors for I, Kr and Xe are from Reference 3. Decontamination factors for Cs and Te are from Reference 25.

** Isotopic activity for the base case fuel was re-calculated with ENC methodology but using assumptions consistent with Reference 3.

Table 3.5 Conversion Ratios for LOCA and Fuel Handling Accident Dose Calculations using Eqn. 3.2.1 for High-Burnup Fuel

Event	Exposure Time	Organ	R
* LOCA	2 hr	Thyroid	0.99
		Wh. Body	1.15
* LOCA	30 day	Thyroid	1.01
		Wh. Body	1.52
** FHA	2 hr	Thyroid	2.08
		Wh. Body	4.75

* Based on core fission product inventory.
** Based on gap fission product inventory.

4.0 REFERENCES

1. XN-NF-78-46, "ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 Using ENC WREM-IIA PWR Evaluation Model," November 1978.

2. XN-NF-79-1, "ECCS Analysis for Kewaunee Using ENC WREM-IIA PWR Evaluation Model," January 1979.

3. Kewaunee Nuclear Generating Plant Final Safety Analysis Report.

4. XN-NF-82-20(P), Revision 1, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," August 1982; Supplement 1, March 1982; and Supplement 2, March 1982.

5. XN-NF-81-58(A), Revision 2 & Supps. 1 and 2, "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984.

6. U.S. Nuclear Regulatory Commission, Safety Evaluation Report on Interim ECCS Evaluation Model for Westinghouse Two-Loop Plants, Analysis Branch, Division of System Safety, Office of Nuclear Reactor Regulation, November 1977.

7. Letter, L.O. Mayer to Director of Nuclear Reactor Regulation, February 24, 1978 (Docket No. 50-282 and 50-306).

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; Federal Register, Volume 39, Number 3, January 4, 1974.

9. XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," July 1975, and Supplements and Revisions thereto.

10. XN-73-25, "GAPEXX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," August 13, 1973.

11. Letter, T.A. Ippolito (USNRC) to W.S. Nechodom (ENC), "SER for ENC RELAP4-EM Update," March 1979.

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

13. XN-NF-78-30(A), "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA," May 1979.

14. XN-NF-82-07(P)(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, August 1982.

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

16. XN-NF-77-25(A), "Exxon Nuclear Company ECCS Evaluation of a 2-Loop Westinghouse PWR with Dry Containment Using the ENC WREM-II ECCS Model - Large Break Example Problem," September 1978.

17. XN-76-27, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II," July 1976; Supplement 1, September 1976; and Supplement 2, November 1976.

18. XN-NF-719, "Assessment of Potential Radiological Consequences for High Exposure Fuel," August 1983.

19. M.J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laborary, May 1973.

20. R.A. Lorenz, "ANS-5.4 Fission Gas Release Model III - Low Temperature Release," ANS Topical Meeting on Light Water Reactor Fuel Performance, (1979).

21. L.D. Noble, "ANS-5.4 Fission Gas Release Model I - Noble Gases at High Temperatures," ANS Topical Meeting on Light Water Reactor Fuel Performance, (1979).

22. "Status Report: ANS-5.4 Fuel Plenum Gas Activity (N218)(Fission Product Release from UO_2 Fuel)," (1977), (Available from S.E. Turner, Southern Science Applications, P.O. Box 10, Dunedim, FL 33528).

23. W.N. Rausch and F.E. Panisko, "ANS-5.4: A Computer Subroutine for Predicting Fission Gas Release," NUREG/CR-1213 (1980).

24. J.R. Houston, et al., "DACRIN: A Computer Program for Calculating Organ Doses from Acute or Chronic Radionuclide Inhalation," Battelle-Pacific Northwest Laboratories, BNWL-B-389/UC-41 (December 1974, Reissued April 1976).

25. R.I. Scherplez and A.E. Desrosiers, "Doses Received While Crossing a Plume of Radioactive Material Released During an Accident at a Nuclear Power Plant," Health Physics, Vol. 43, No. 2, p. 187, August 1982.

XN-NF-84-31
Revision 1

Issue Date: 10/1/84

KEWAUNEE HIGH BURNUP SAFETY ANALYSIS:

LIMITING BREAK LOCA AND RADIOLOGICAL CONSEQUENCES

Distribution

J. C. Chandler

R. A. Copeland

L. J. Federico

S. E. Jensen

W. V. Kayser

M. R. Killgore

T. R. Lindquist

T. E. Millsaps

L. C. O'Malley

R. B. Stout

M. S. Stricker

T. Tahvili

P. J. Valentine

Wisconsin Public Service/LC O'Malley (60)

Document Control (5)