XN-NF-84-72

H. B. ROBINSON UNIT 2
LARGE BREAK LOCA-ECCS ANALYSIS
WITH INCREASED ENTHALPY RISE FACTOR

JULY 1984

RICHLAND WA 99352

EXXON NUCLEAR COMPANY, INC.

XN-NF-84-72
Issue Date: 7/12/84

H. B. ROBINSON UNIT 2

LARGE BREAK LOCA-ECCS ANALYSIS

WITH INCREASED ENTHALPY RISE FACTOR

Prepared by: T. Tahvili 7/10/84
T. Tahvili
PWR Safety Analysis

Reviewed by: C. E. Leach 7/10/84
C. E. Leach, Project Manager
PWR Safety Analysis

Concur: W. V. Kayser 7/10/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: R. B. Stout for
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Approve: R. B. Stout 12 JULY 84
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: George A. Sofer 11 JUL 84
G. A. Sofer, Manager
Fuel Engineering & Technical Services

naa

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

In 1975, Exxon Nuclear Company (ENC) performed a Loss-of-Coolant Accident (LOCA) analysis for H.B. Robinson Unit 2[1], using the ENC WREM-based PWR ECCS Evaluation Model[2]. A followup analysis in December 1976[3] identified the double-ended cold leg guillotine break with a discharge coefficient of 0.8 as the most limiting break. The analysis assumed 6% steam generator tube plugging (SGTP) with a total linear heat generation rate (LHGR) of 13.43 kw/ft, corresponding to a total peaking (F_Q^T) of 2.2 at 102% of rated power. In September 1980 and August 1981, additional analyses were performed[4,5] to support the operation of the plant with 10% and 15% steam generator tube plugging for an LHGR and F_Q^T of 13.43 kw/ft and 2.2, respectively. In addition, an analysis was performed at 15% steam generator tube plugging which supported an LHGR of 14.16 kw/ft and an F_Q^T of 2.32. Additional analyses were performed in 1982 for SGTP of 20% and in 1983 for SGTP of 30%, at reduced power and temperature[6,7]. The 1983 analysis used the ENC WREM-IIA PWR ECCS evaluation model[8,9,10,11] with the NUREG-0630 clad swelling and rupture model and the revised EXEM/PWR steam cooling model[12]. Table 2.2 summarizes the ENC licensing history for the H.B. Robinson Unit 2 plant.

This report presents the results of a LOCA ECCS analysis performed for the previously identified limiting break, of 0.8 DECLG with peak rod exposures up to 49 MWD/kg. This analysis was performed as a result of the decision by Carolina Power and Light Company (CP&L) in 1983 to: (1) replace the H.B. Robinson Unit 2 steam generators, (2) implement a low radial leakage fuel management scheme in order to reduce vessel fluences and

thereby alleviate concerns about thermal shock, and (3) to increase the peak assembly discharge exposure for the H.B. Robinson fuel to 44 MWD/kgU. To implement the low radial leakage fuel management scheme, the total nuclear enthalpy rise ($F^T_{\Delta H}$) was increased to 1.65. The analysis was performed with an LHGR, including the 1.02 factor for power uncertainty, of 14.16 kw/ft, corresponding to a total power peaking factor of 2.32 (F_Q^T). The analysis is applicable for up to 6% steam generator tube plugging with the reactor operating at 100% power, 2300 MWt.

2.0 SUMMARY

The calculational basis and results are summarized in Table 2.1. The maximum calculated peak cladding temperature (PCT) is 2042°F, occurring at 60 seconds into the accident at a location 6.0 feet from the bottom of the active core, with a total metal-water reaction less than one percent. The results of the analyses show that within the limits established, the H.B. Robinson Nuclear Reactor, operating at the rated power level of 2300 MWt, and with steam generator tube plugging up to 6%, satisfy the criteria specified by 10 CFR 50.46[13] for peak rod burnups less than 49 MWD/kgU.

Table 2.1 H. B. Robinson Unit 2

LOCA-ECCS Analysis Results

Calculational Basis	
License Core Power, MWt	2300
Power Used for Analysis, MWt**	2346
Peak Linear power for Analysis, kW/ft**	14.16
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise, Nuclear, $F_{\Delta H}^T$	1.65
Steam Generator Tube Plugging (%)	6.00

Analysis Results	2 MWD/kgU Peak Rod Exposure	9 MWD/kgU Peak Rod Exposure	49 MWD/kgU Peak Rod Exposure
Peak Clad Temperature (PCT), ^{o}F	2042	1815	1785
Peak Clad Temperature Reached, (sec)	60	139	139
Peak Clad Temperature Location, ft.	6.0	8.5	8.5
Local Zr/H_2O Reaction (max.), %*	4.65	1.93	1.72
Local Zr/H_2O Location, ft. from Bottom	5.25	5.25	5.25
Total H_2 Generation, % of total Zr Reacted	<1	<1	<1
Hot Rod Burst Time, sec.	39.9	45.7	45.2
Hot Rod Burst Location, ft.	6.0	6.0	6.0

* Computer value at 380 seconds.

** Including 1.02 factor for power uncertainties.

Table 2.2 ENC Licensing History of H. B. Robinson Unit 2

Plant Parameters	XN-NF-75-41, XN-NF-76-54 1765 & 1976	XN-NF-80-43, XN-NF-81-54 1980 & 1981			XN-NF-82-18 1982	XN-NF-82-18(P) Supplement 2 1983	Current XN-NF-84-72 1984
F_Q^T	2.2	2.2	2.2	2.32	2.32	2.32	2.32
$F_{\Delta H}^T$	1.55	1.55	1.55	1.55	1.60	1.60	1.65
LHGR (kw/ft)	13.43	13.43	13.43	14.16	12.04	12.04	14.16
% of rated power	102	102	102	102	87	87	102
Primary coolant flow (GPM/Loop)	89965	89965	89965	89965	82700	80000	88330
Vessel T_{ave} (^oF)	579.5	579.5	579.5	579.5	537.1	537.1	579.5
SG Tube Plugging (%)	6	10	15	15	20	30	6
Break Type	Break Spectrum	Limiting Break 0.8 DECLG	Limiting Break 0.8 DECLG	Limiting Break 0.8 DECLG	Limiting Break 0.8 DECLG	Limiting Break 0.8 DECLG	Limiting Break 0.8 DECLG

3.0 LIMITING BREAK LOCA ANALYSIS

This report documents the results of the LOCA-ECCS analysis performed for H.B. Robinson Unit 2 with a steam generator tube plugging up to 6%. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are described in the ENC WREM models[2], and the Emergency Core Cooling System Evaluation Model Updates: WREM-II[14], WREM-IIA[10], and EXEM/PWR[12].

A LOCA break spectrum analysis was performed and reported in XN-76-54[3]. The limiting LOCA break was determined to be a large double-ended guillotine break of the cold leg, with a discharge coefficient of 0.8 (0.8 DECLG). The analyses performed and reported herein used the following LOCA/ECCS models:

(1) Fuel Rod Model - The RODEX2[15] stored energy and fission gas release model in place of the previously approved GAPEX[16] model.

(2) Blowdown Model - The RELAP4-EM code with NUREG-0630 clad swelling and rupture model[12], and fuel rod model consistant with RODEX2 gap conductance model[17].

(3) Reflood Model - The REFLEX code with the EXEM/PWR core outlet enthalpy model[12] and the ENC WREM carry rate fraction (CRF) correlation[2].

(4) Heatup Model - The TODDEE2 code with the EXEM/PWR steam cooling model[12], the NUREG-0630 clad swelling and rupture model[12], and the WREM heat transfer correlations[2].

(5) All other model revisions documented in XN-NF-82-20(P), Revision 1[12].

3.1 LOCA ANALYSIS MODEL

The Exxon Nuclear Company EXEM/PWR ECCS evaluation model[12] was used to perform the analyses. This model consists of the following computer codes: RODEX2[15] for initial rod stored energy and internal fuel rod gas inventory; RELAP4-EM[17,18] for the system blowdown and hot channel blowdown calculations; CONTEMPT-LT/22 as modified in CSB 6-1[19] for computation of containment backpressure; REFLEX[10,12] for computation of system reflood; and TOODEE2[12,20,21] for the calculation of final fuel rod heatup.

The H.B. Robinson Unit 2 nuclear reactor is a three-loop Westinghouse pressurized water reactor with 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-75-57[1]. The system nodalization is depicted in Figure 3.1. The single failure is assumed to be the loss of one of three HPSI pumps in addition to a loss of one LPSI pump. The reactor coolant pump performance characteristic curves are the Westinghouse pump curves built into the RELAP4 code. Six percent of the tubes in each steam generator are assumed to be 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 3.1. Figure 3.2 shows the REFLEX nodalization in the reflood calculation of the H.B. Robinson Unit 2.

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 3.3, with a maximum axial peaking factor of 1.365, corresponding to a total peaking factor of 2.32, and $F^T_{\Delta H}$ 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 3.4. The analysis of the loss-of-coolant accident is performed at 102 percent of rated power. The fuel design parameters are shown in Table 3.2.

Three cases of LOCA-ECCS calculations were performed with input which bounds the fuel history up to 49,000 MWD/kgU peak rod exposure. The most limiting fuel conditions were determined and used in each calculation. Decay power, internal rod pressure and the fission gas releases were highest at EOL (third 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 were used in the LOCA-ECCS analyses over the exposure ranges shown.

3.2 RESULTS

Table 3.3 presents the timing and sequence of events as determined for the large guillotine break with a discharge coefficient of 0.8. Figures 3.5 through 3.34 present plotted results for system blowdown analysis. Unless

otherwise noted on the figures, time zero corresponds to the time of break initiation. Figures 3.14 through 3.28 present results for the hot channel blowdown calculations. Figure 3.35 presents calculated containment back-pressure time history. Figure 3.36 shows the normalized power calculation results. The reflood calculation results are shown in Figures 3.37 through 3.40.

The maximum peak cladding temperature (PCT) calculated for the 0.8 DECLG break occurs at 2 MWD/KgU and is 2042°F (Figure 3.41). The maximum local metal-water reaction is 4.65 after 380 seconds, and the total core metal-water reaction is less than 1%. The PCT location is at an elevation of 6.0 feet from the bottom of active core. For the 9 MWD/kgU exposure, the calculated PCT is 1815°F (Figure 3.42) occurring at 139 seconds at an elevation of 8.5 feet relative to the bottom of the active core. For the EOL case, the calculated PCT is 1785°F occurring at 139 seconds at 8.5 feet elevation relative to the bottom of the active core (Figure 3.43).

Table 3.1 H. B. Robinson Unit 2 System Data

Primary Heat Output, MWt	2346*
Primary Coolant Flow, lbm/hr	100.3×10^6
Primary Coolant Volume, ft^3	9768**
Operating Pressure, psia	2,250.
Inlet Coolant Temperature, ^oF	546.2
Reactor Vessel Volume, ft^3	3660
Pressurizer Volume, Total, ft^3	1300
Pressurizer Volume, Liquid, ft^3	780
Accumulator Volume, Total, ft^3 (each of three)	1200
Accumulator Volume, Liquid, $ft^{(3)}$	825
Accumulator Trip Point Pressure, psia	615
Steam Generator Heat Transfer Area, $ft^{(2)}$	40,859**
Steam Generator Secondary Flow, lbm/hr	3.37×10^6
Steam Generator Secondary pressure, psia	800
Reactor Coolant Pump Head, ft	264
Reactor Coolant Pump Speed, rpm	1180
Moment of Inertia, lbm-ft^2/rad	70,000
Cold Leg Pipe, I.D., in	27.5
Hot Leg Pipe, I.D., in	29.0
Pump Suction Pipe, I.D., in	31.09

* Primary Heat Output used in RELAP4-EM Model = 1.02 x 2300 = 2346 MWt.

** Includes 6% SG tube plugging.

Table 3.2 Fuel Design Parameters

Parameter	ENC Fuel
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
Enriched UO_2, in.	132
Upper Blanket, in.	6.0
Lower Blanket, in.	6.0
Cell Water/Fuel Ratio	1.76
Rod Pitch	0.563

Table 3.3 H.B. Robinson Unit 2 LOCA/ECCS Event Table for Limiting Break (0.8 DECLG and 2 MWD/kgU Exposure)

Event	Time (seconds)
Start	0.0
Initiate Break	0.1
Safety Injection Signal	0.6
Accumulator Injection, Broken Loop	3.1
Pressurizer Empties	9.0
Accumulator Injection, Intact Loop	12.0
End-of-Bypass	22.57
Safety Pump Injection, HPSI	25.6
Safety Pump Injection, LPSI (Broken)	32.17
Start of Reflood	45.79
Accumulators Empty	48.76
Safety Pump Injection, LPSI (Intact)	48.87
Peak Clad Temperature Reached (sec)	60.0

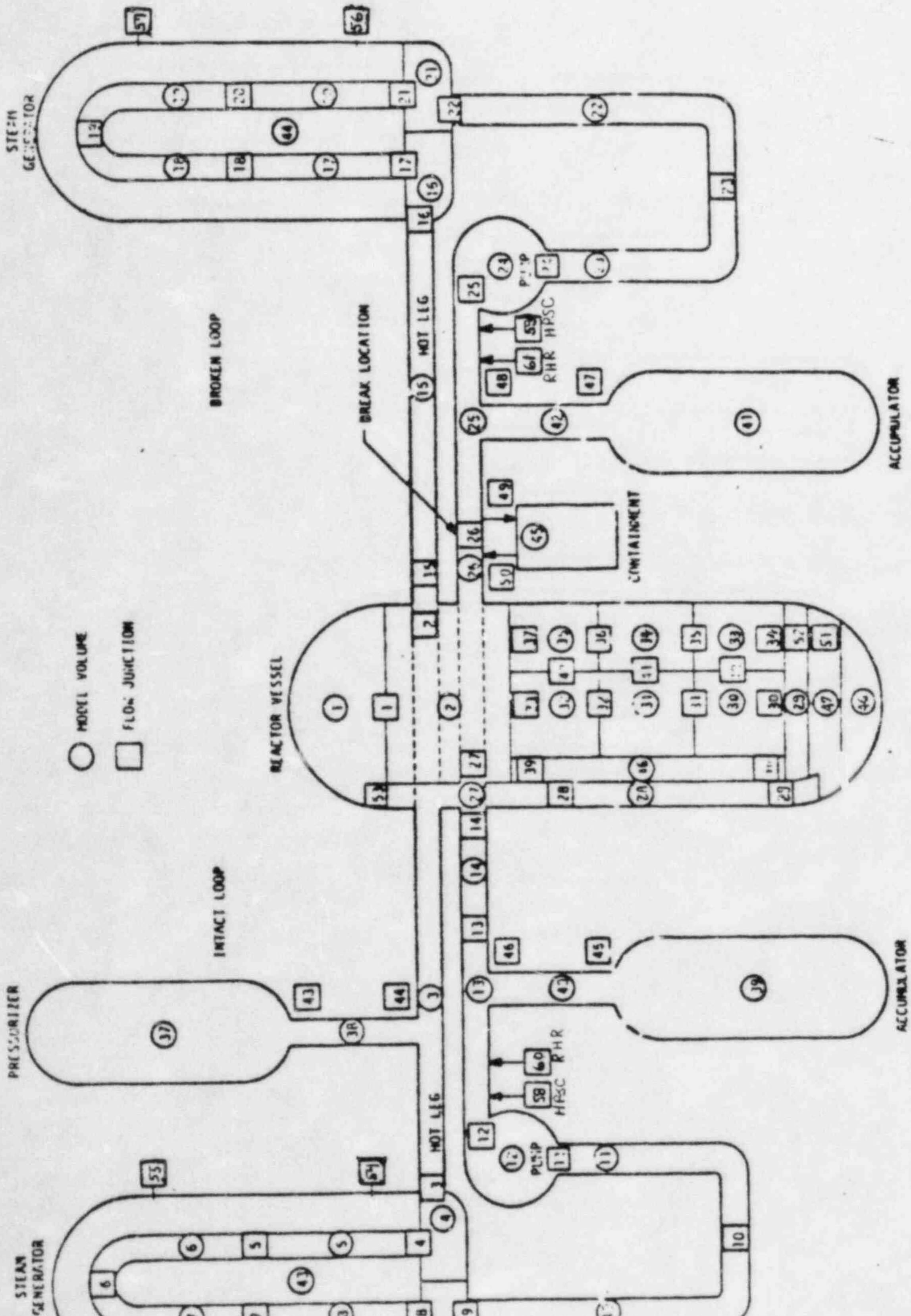


Figure 3.1 Blowdown System Nodalization

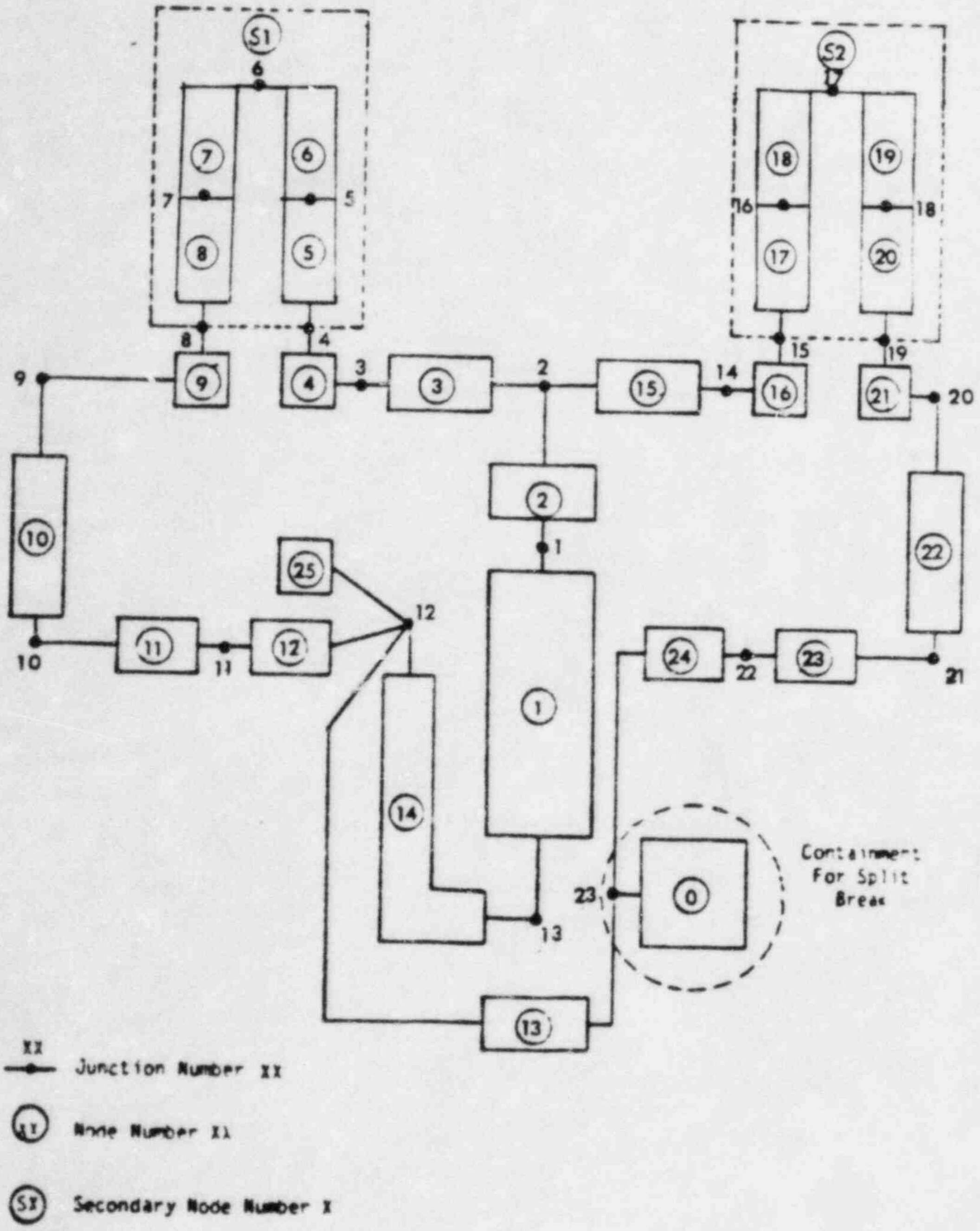
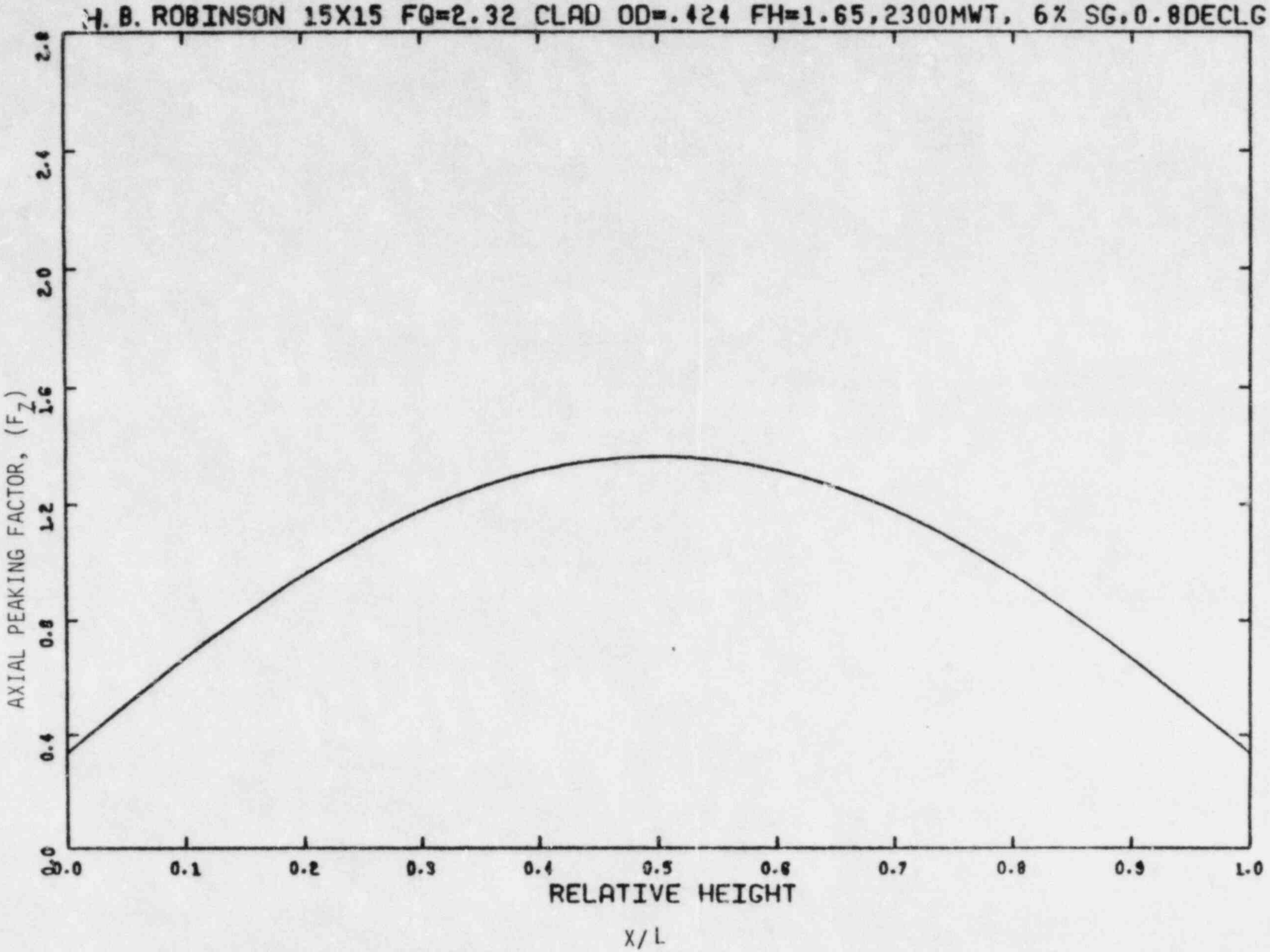


Figure 3.2 REFLEX Nodalization


H. B. ROBINSON 15X15 FQ=2.32 CLAD OD=.424 FH=1.65,2300MWT, 6% SG,0.8DECLG
AXIAL PEAKING FACTOR, (F_Z)
0.0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
RELATIVE HEIGHT
X/L

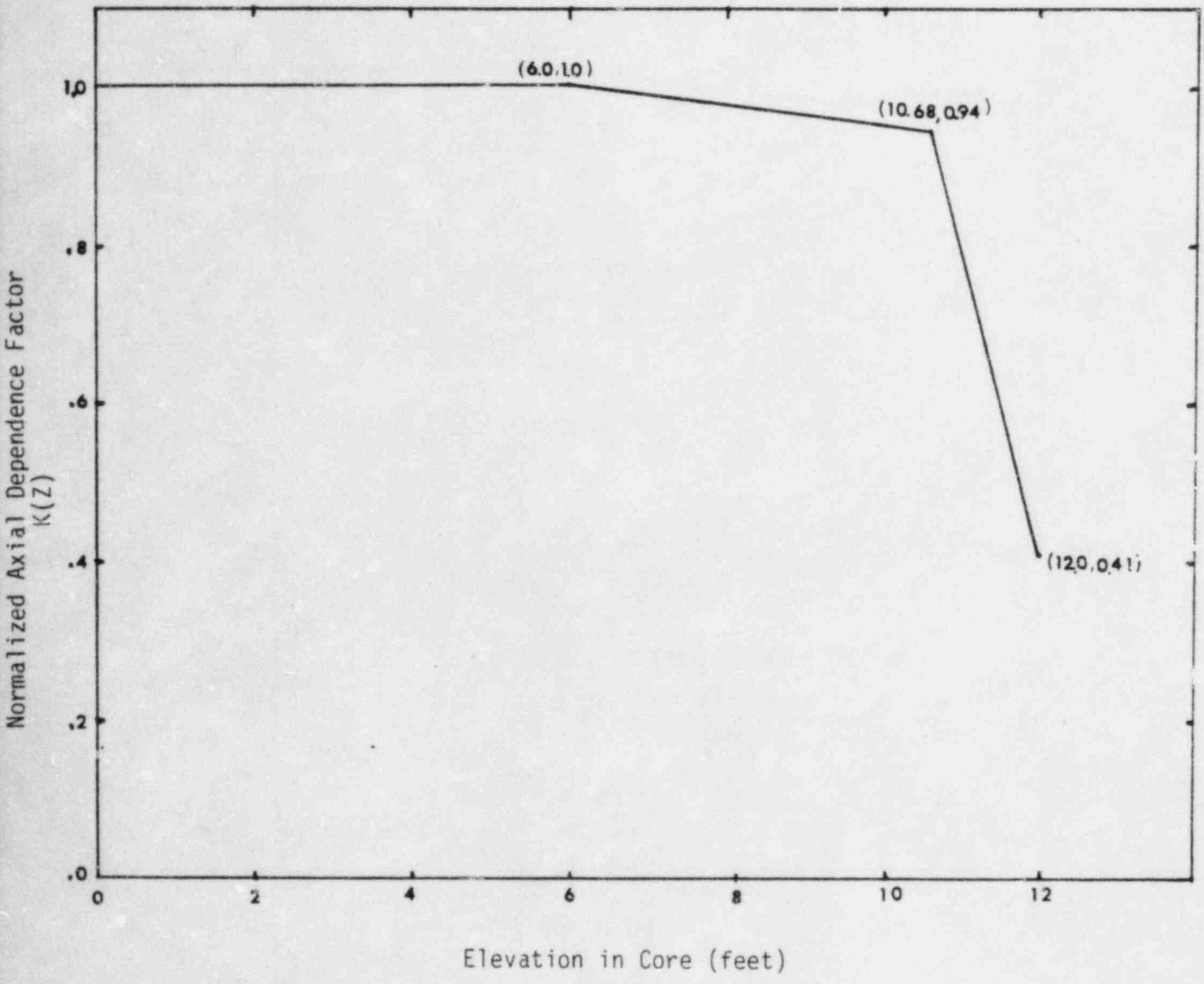


Figure 3.4 Normalized Axial Dependence Factor for F_q^T=2.32 versus Elevation ($F_{\Delta H}^T$=1.65)

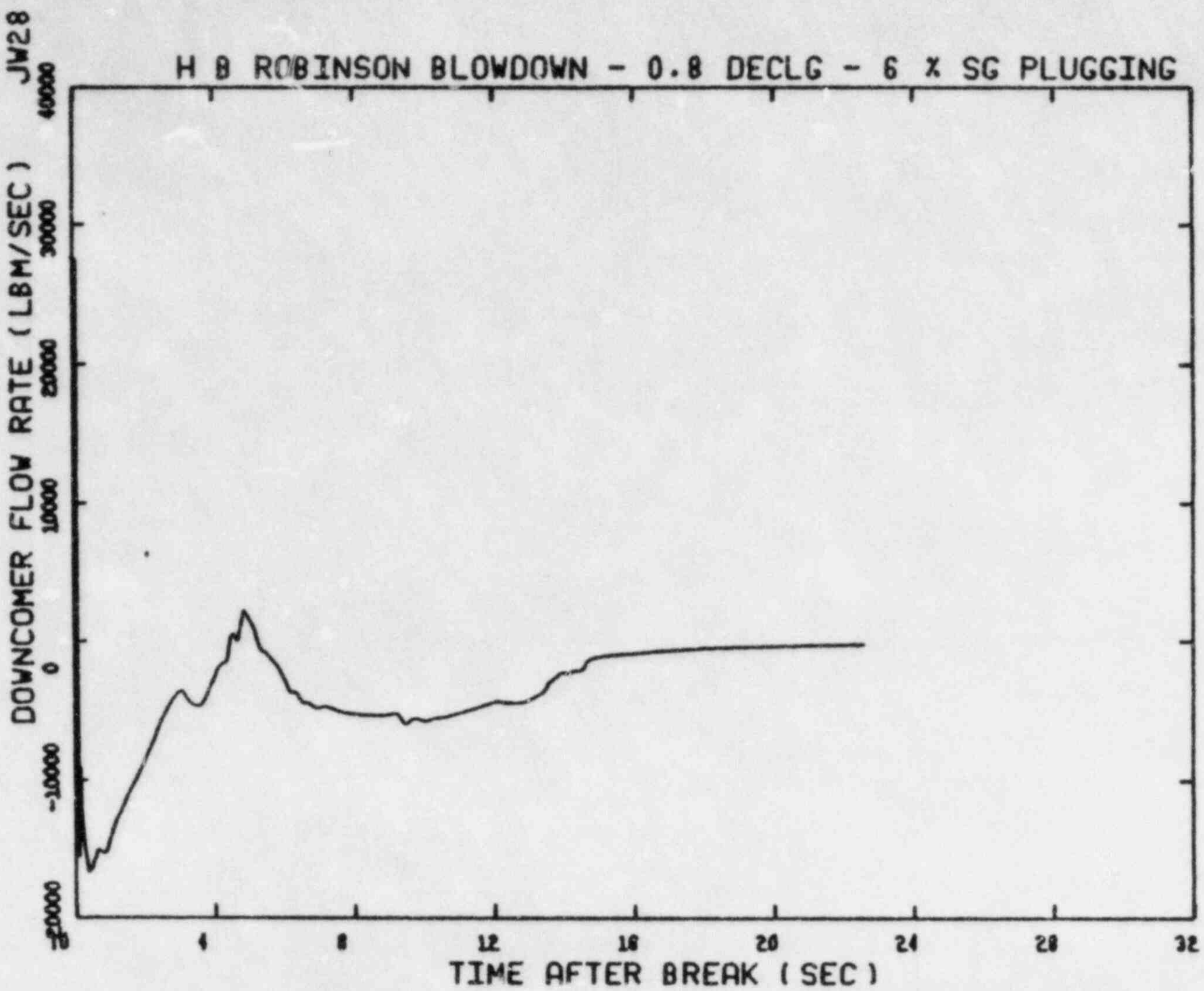


Figure 3.5 Downcomer Flow Rate During Blowdown Period, 0.8 DECLG Break

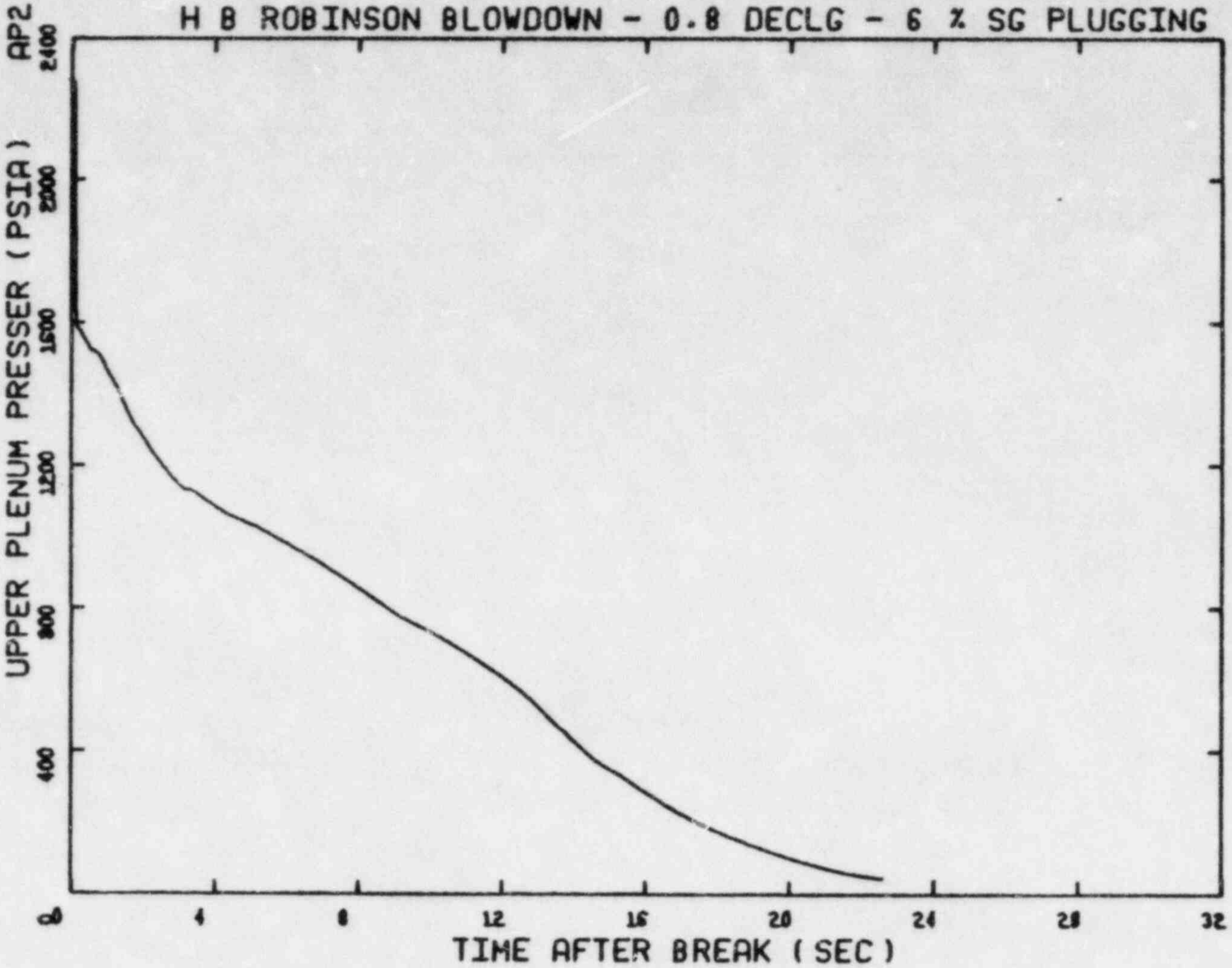


Figure 3.6 Upper Plenum Pressure During Blowdown Period, 0.8 DECLG Break

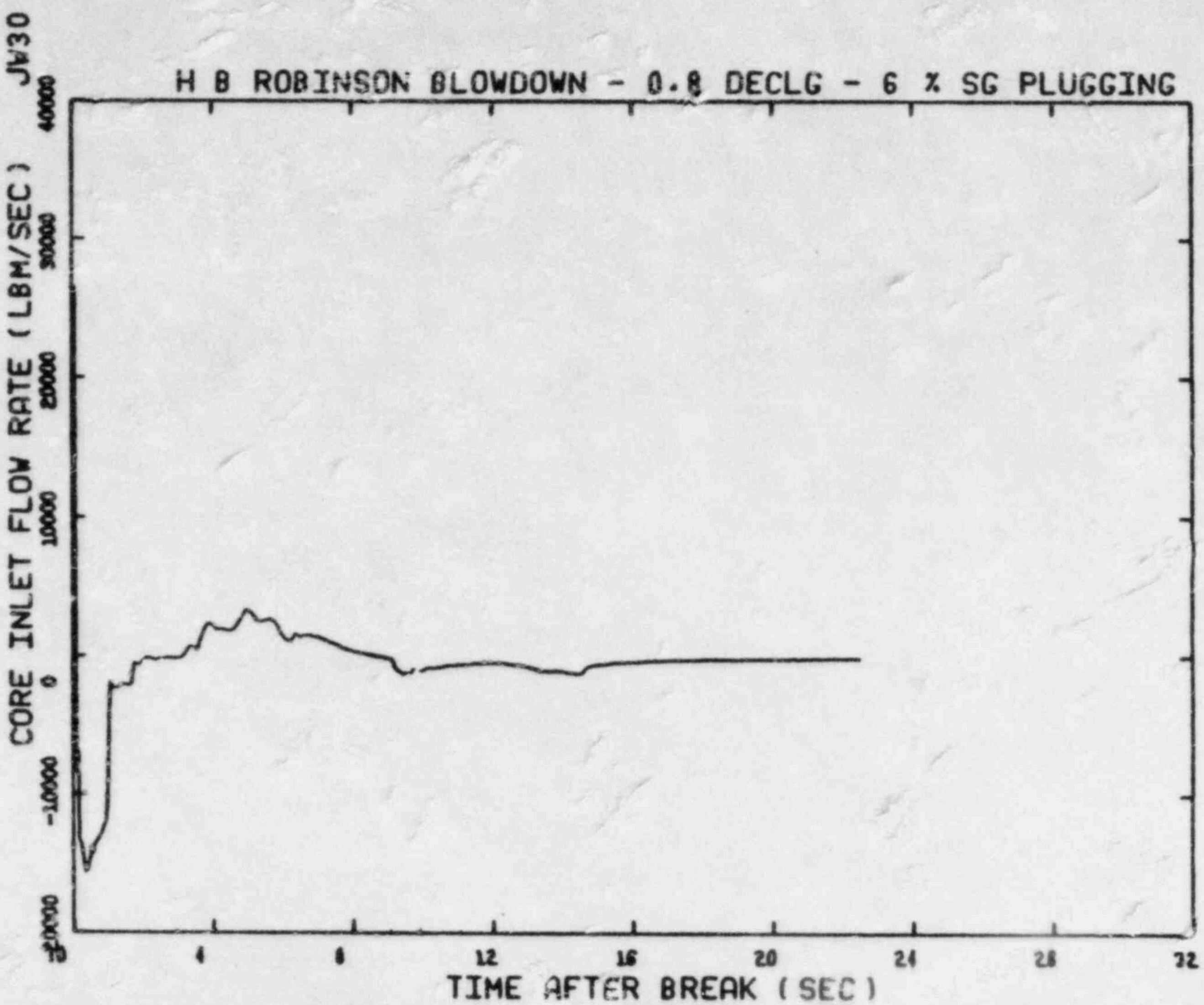


Figure 3.7 Average Core Inlet Flow During Blowdown Period, 0.8 DECLG Break

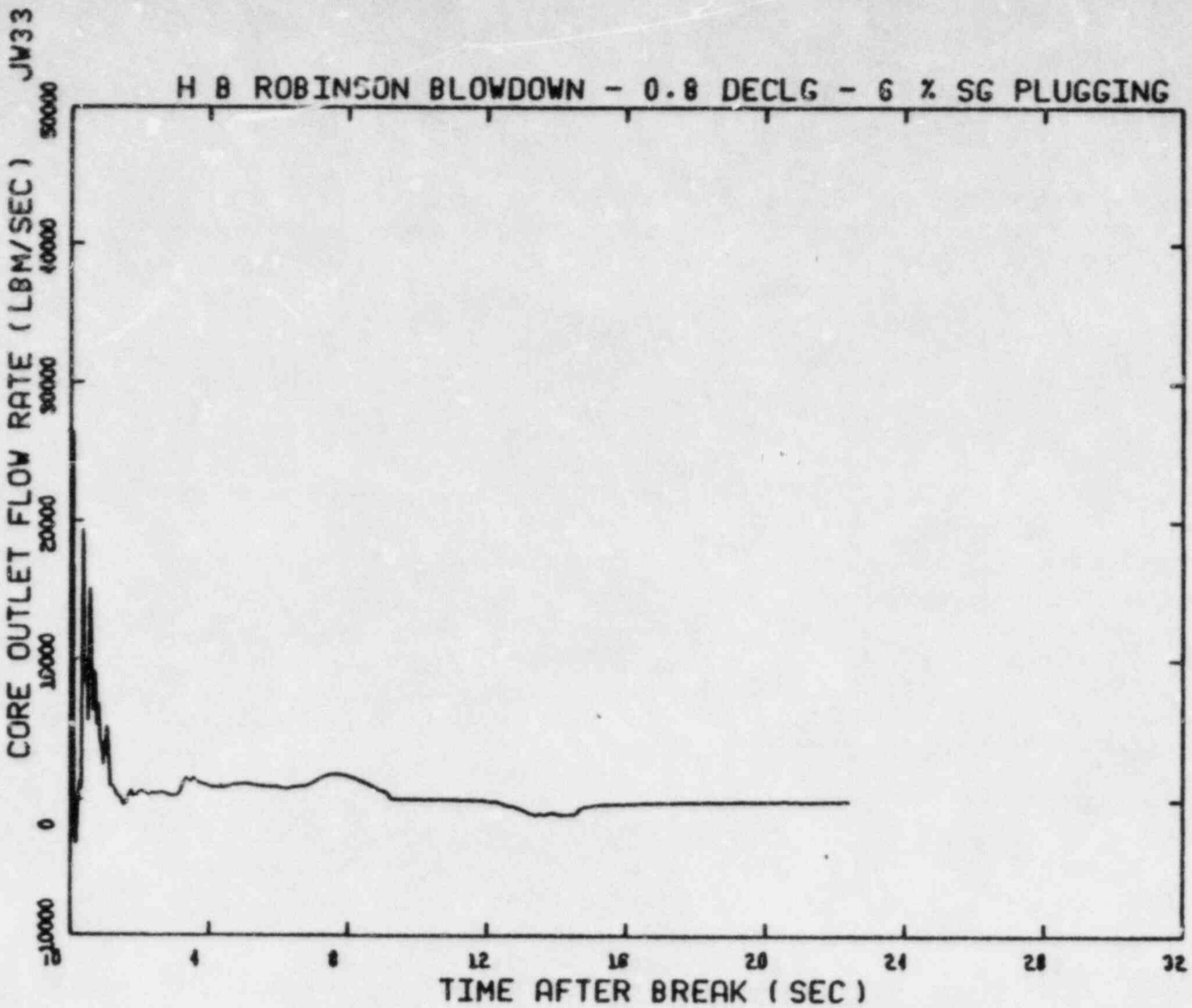


Figure 3.8 Average Core Outlet Flow During Blowdown Period, 0.8 DECLG Break

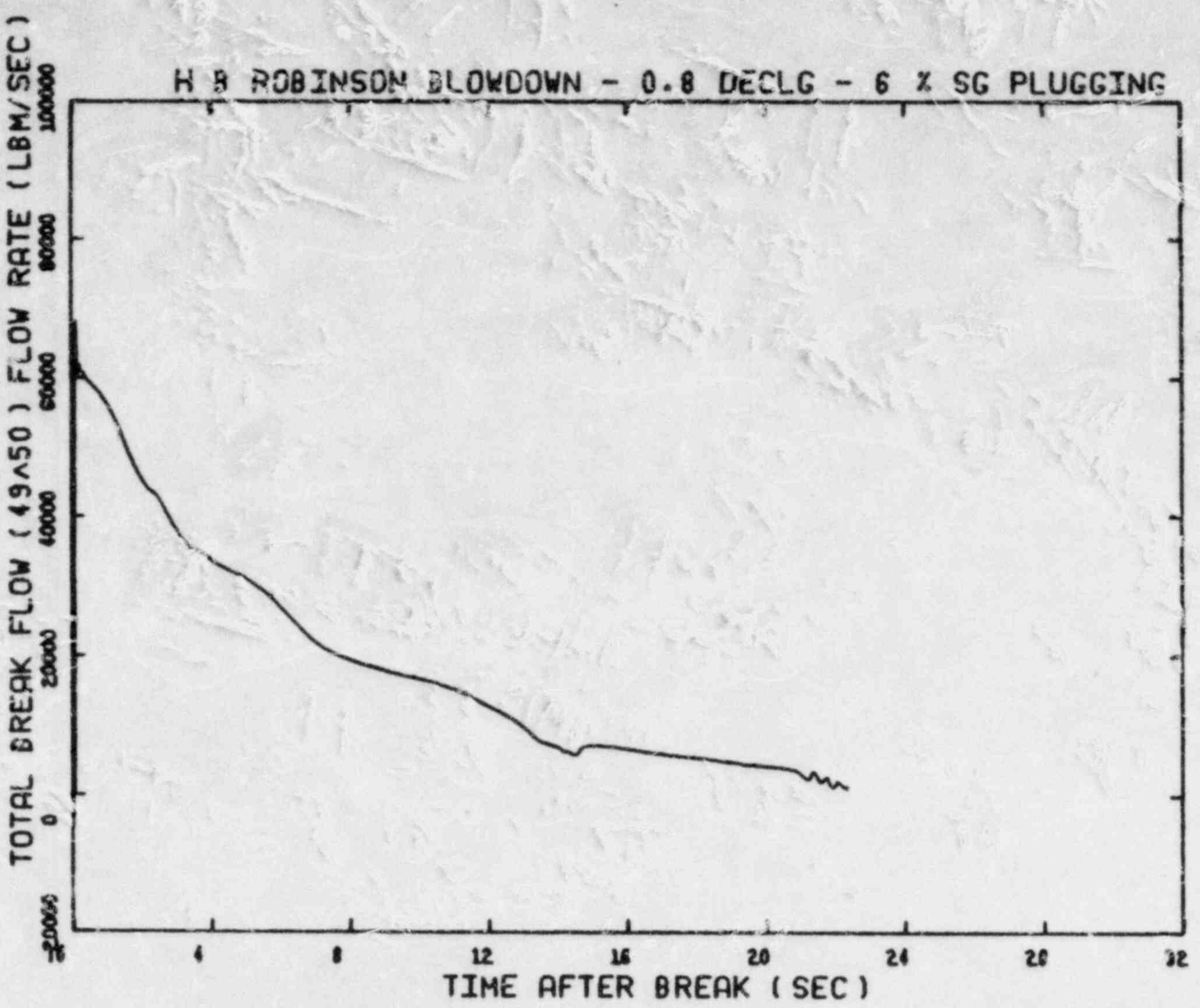


Figure 3.9 Total Break Flow During Blowdown Period, 0.8 DECLG Break

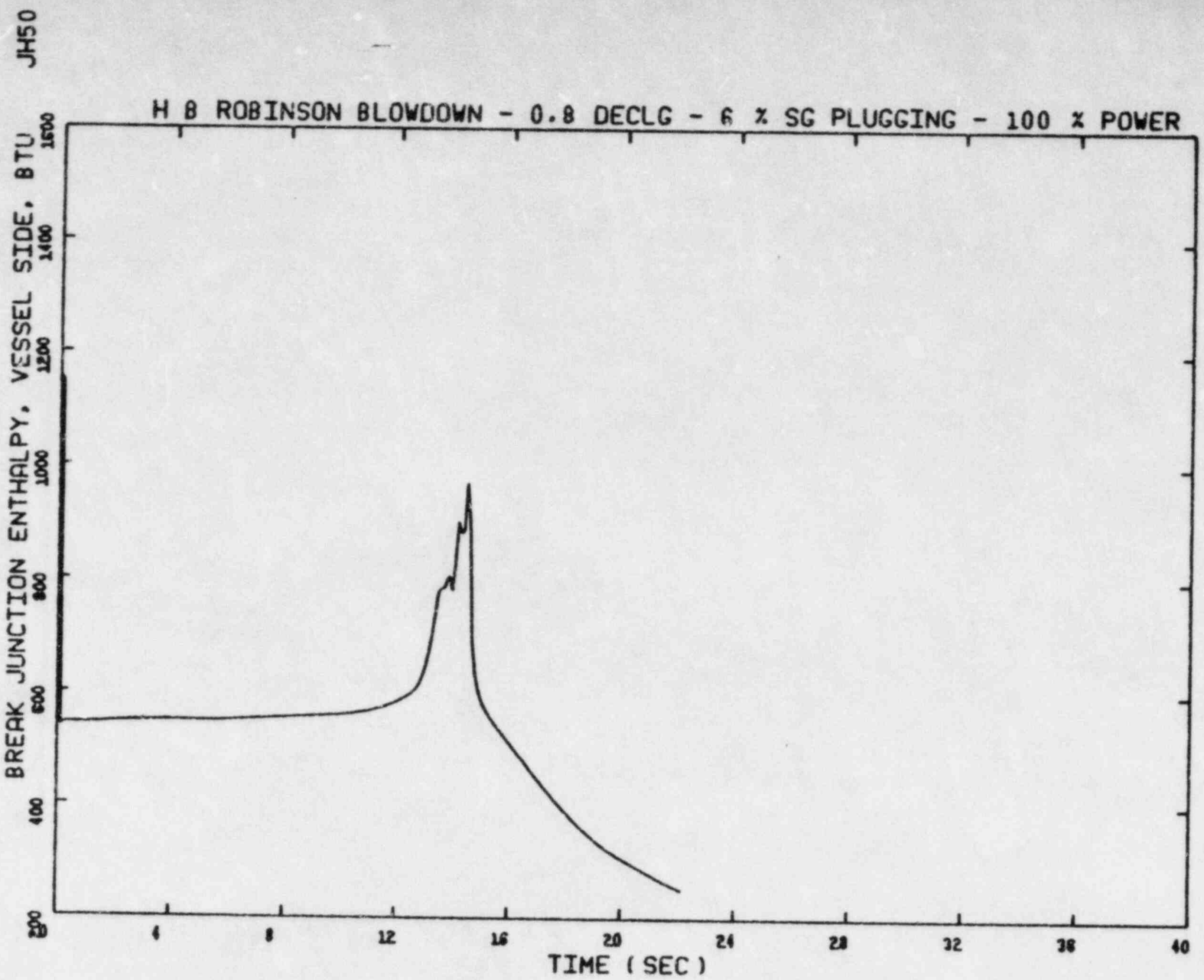


Figure 3.10 Break Flow Enthalpy During Blowdown Period, Vessel Side, 0.8 DECLG Break

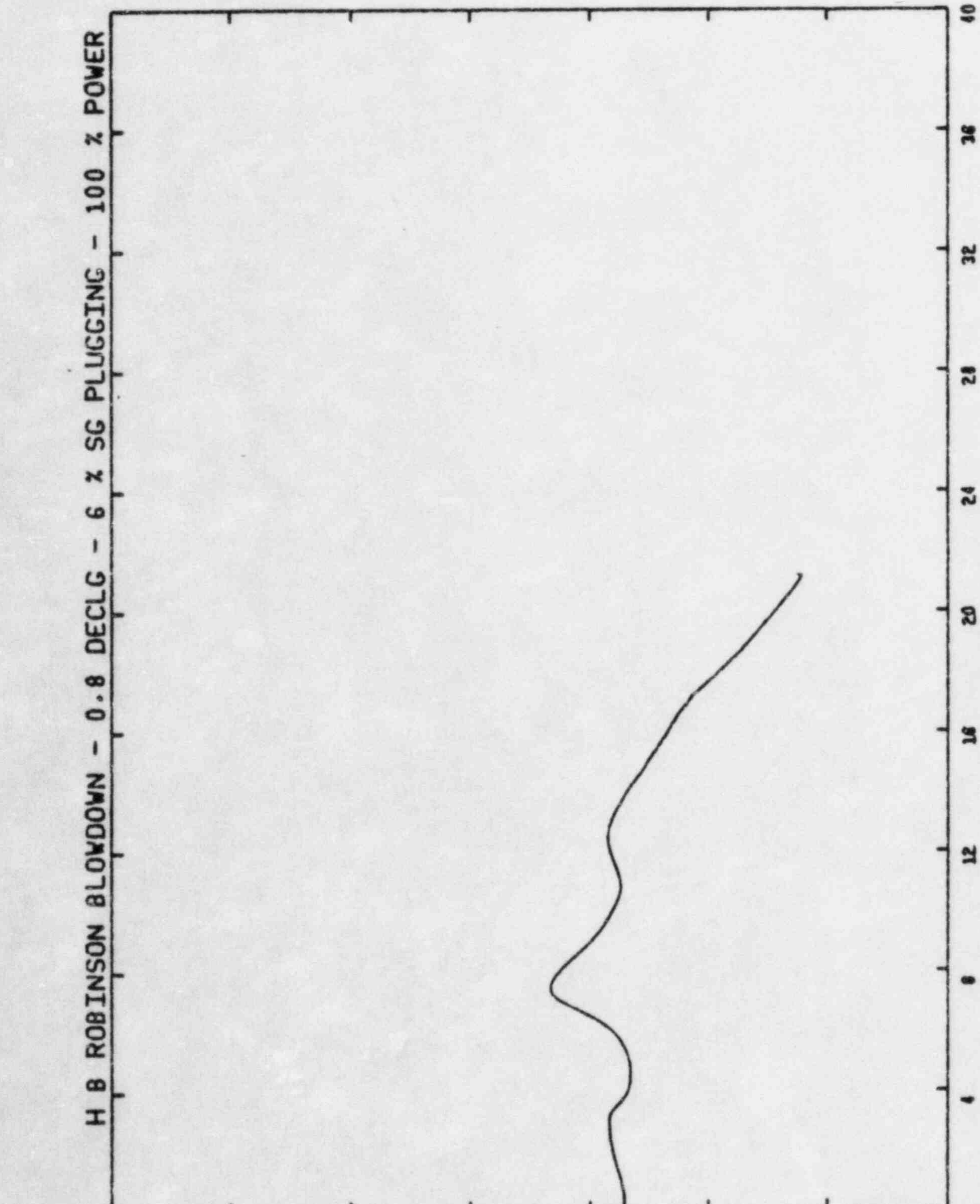


Figure 3.11 Break Flow Enthalpy During Blowdown Period, Pump Side, 0.8 DECLG Break

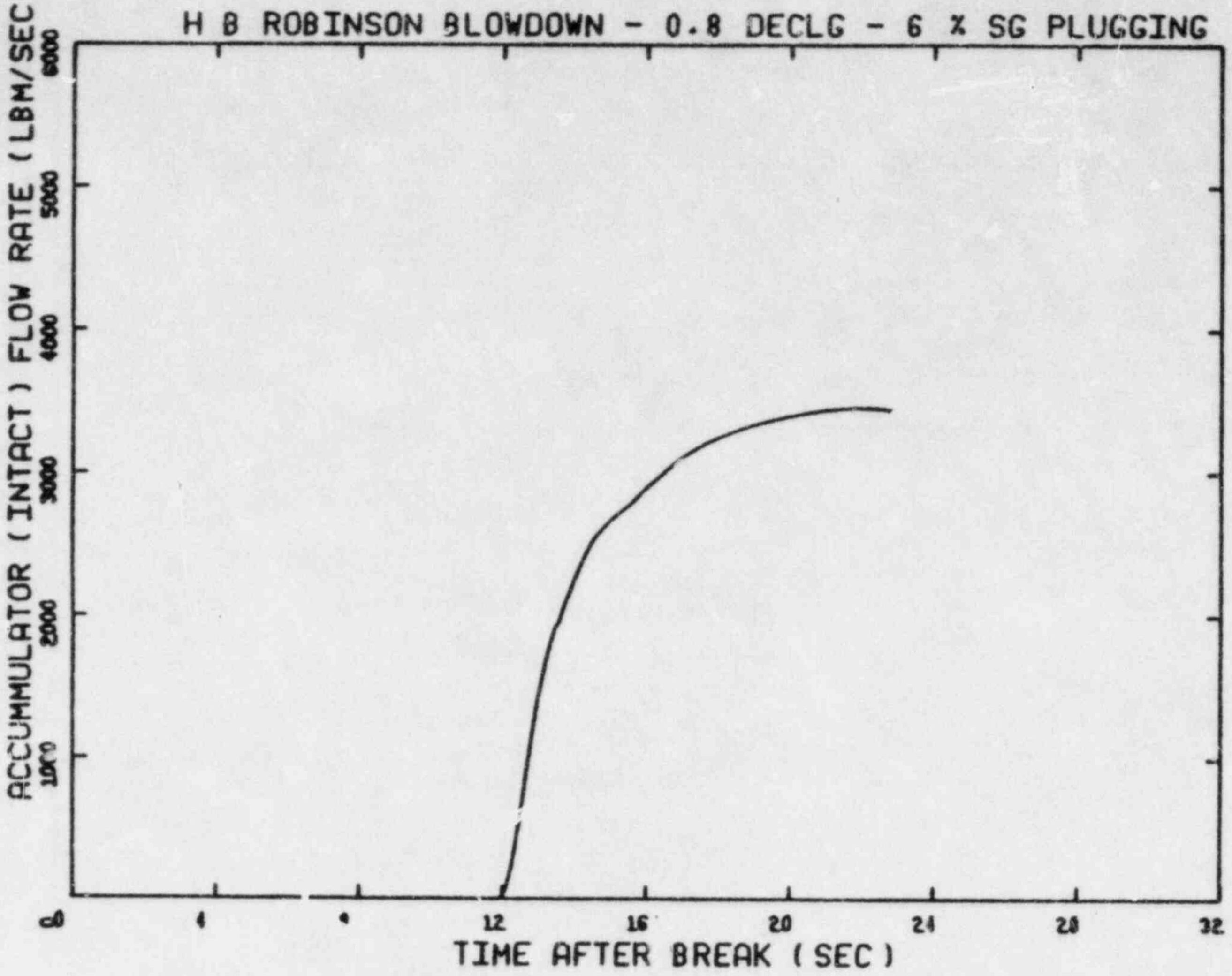


Figure 3.12 Flow from Intact Loop Accumulator During Blowdown Period, 0.8 DECLG Break

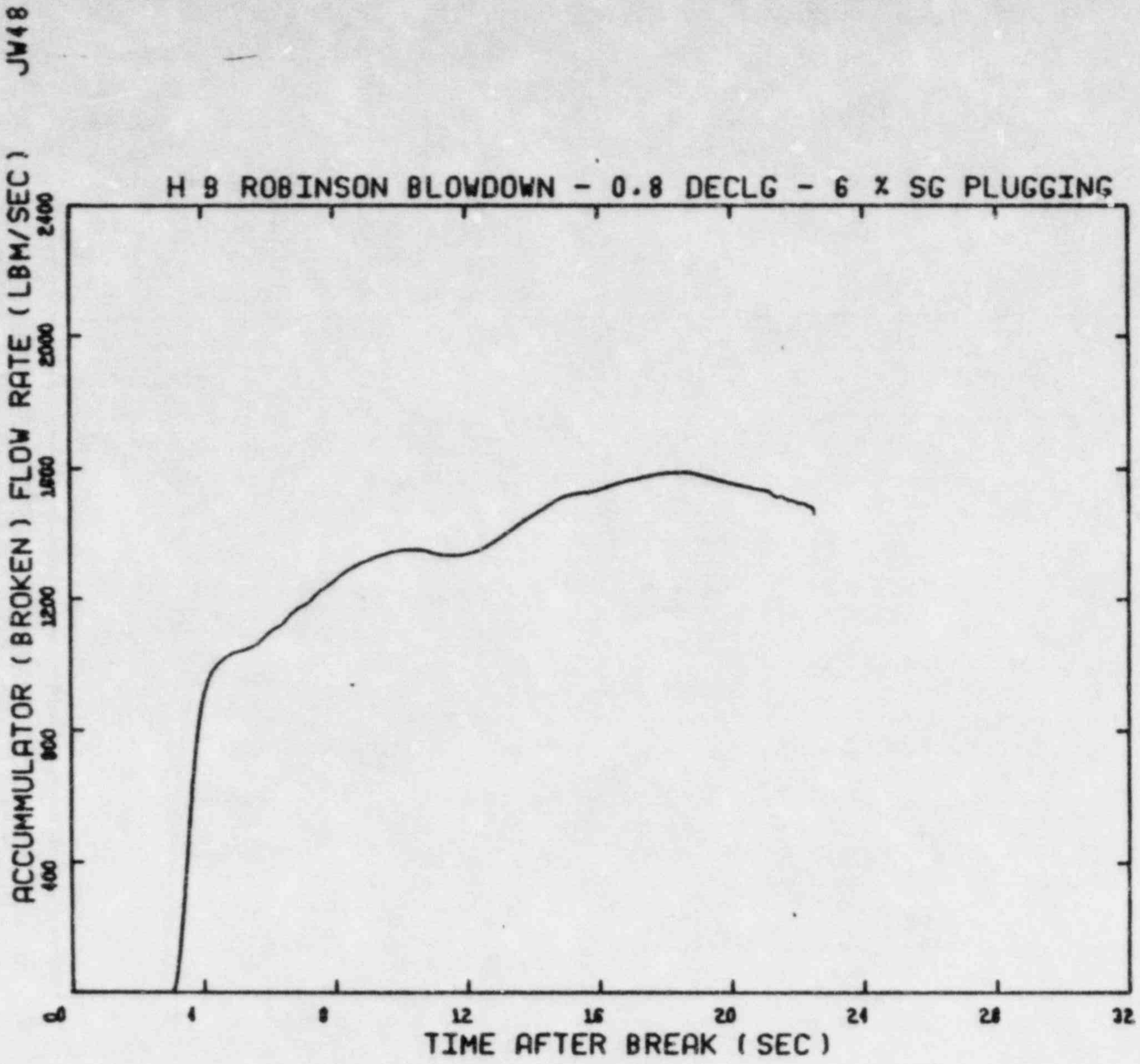


Figure 3.13 Flow From Broken Loop Accumulator During Blowdown Period, 0.8 DECLG Break

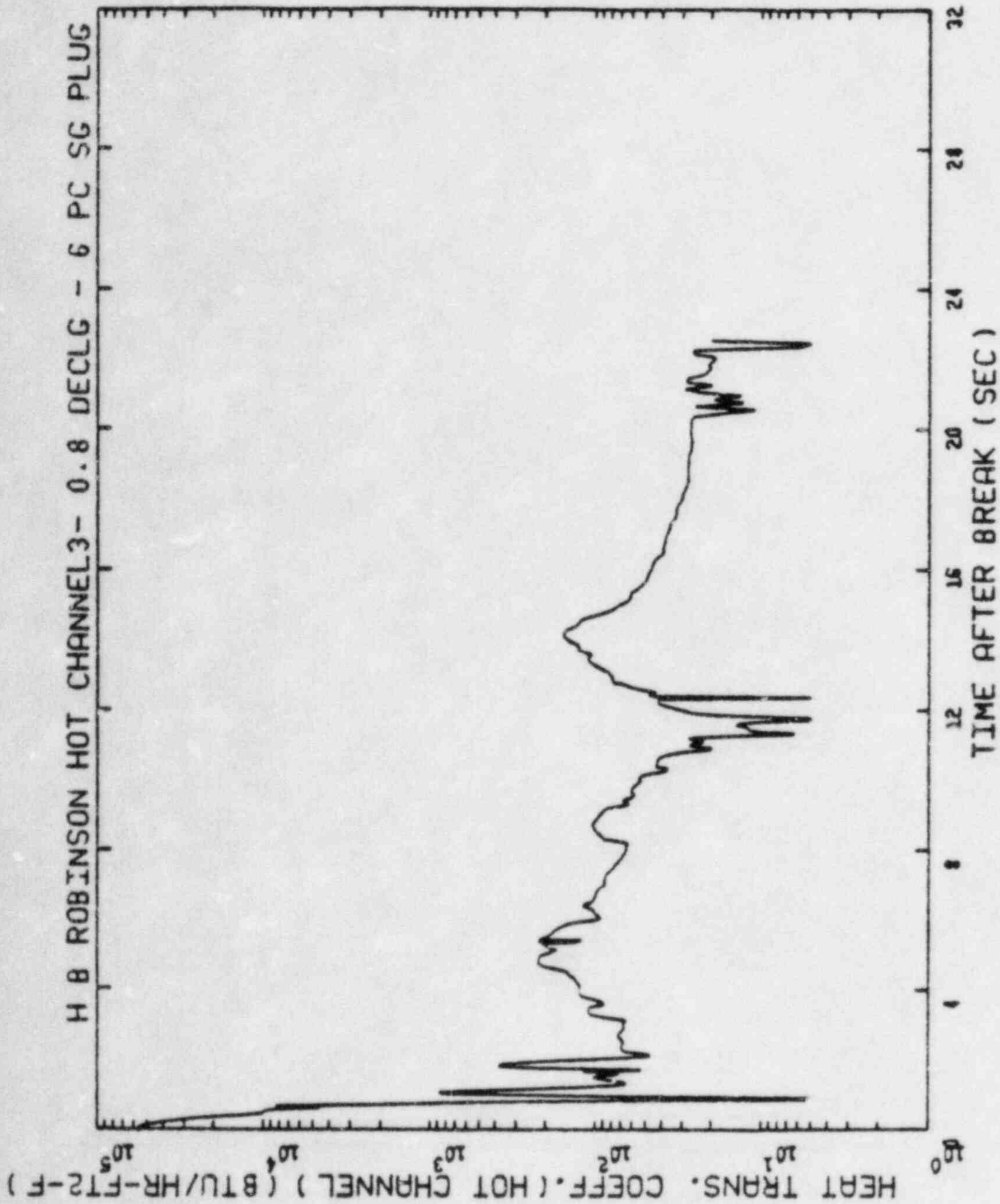


Figure 3.14 Heat Transfer Coefficient During Blowdown Period at PCT Node, 0.8 DECLG Break, 2 MWD/KgU

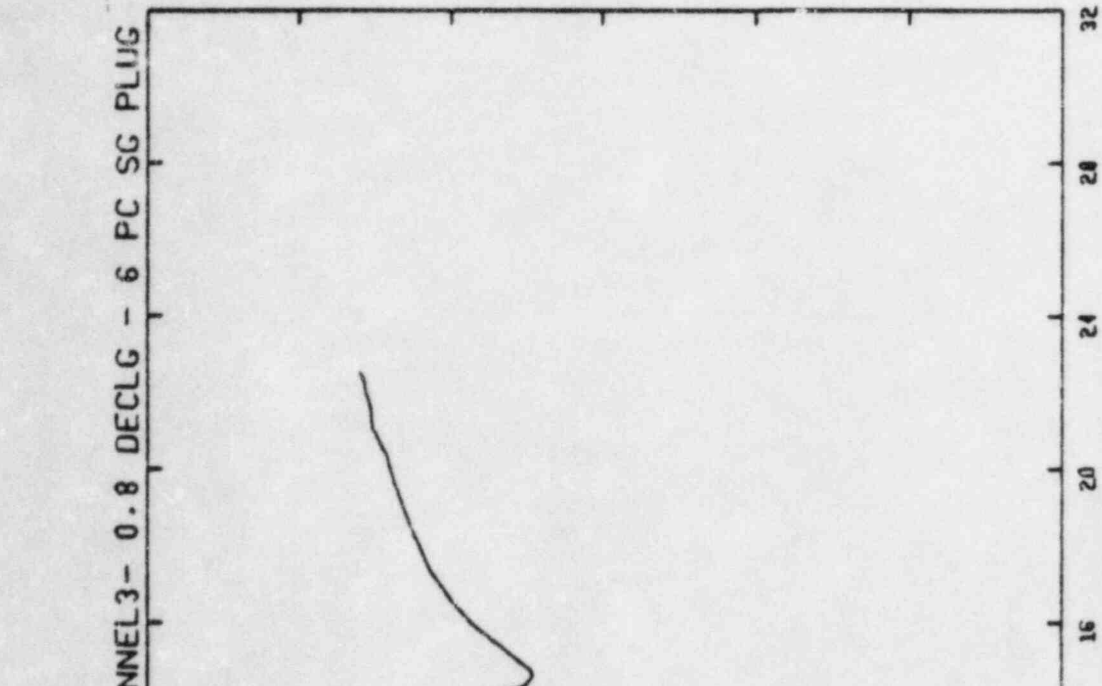


Figure 3.15 Clad Surface Temperature during Blowdown Period
PCT Node, 0.8 DECLG Break, 2 MWD/KgU

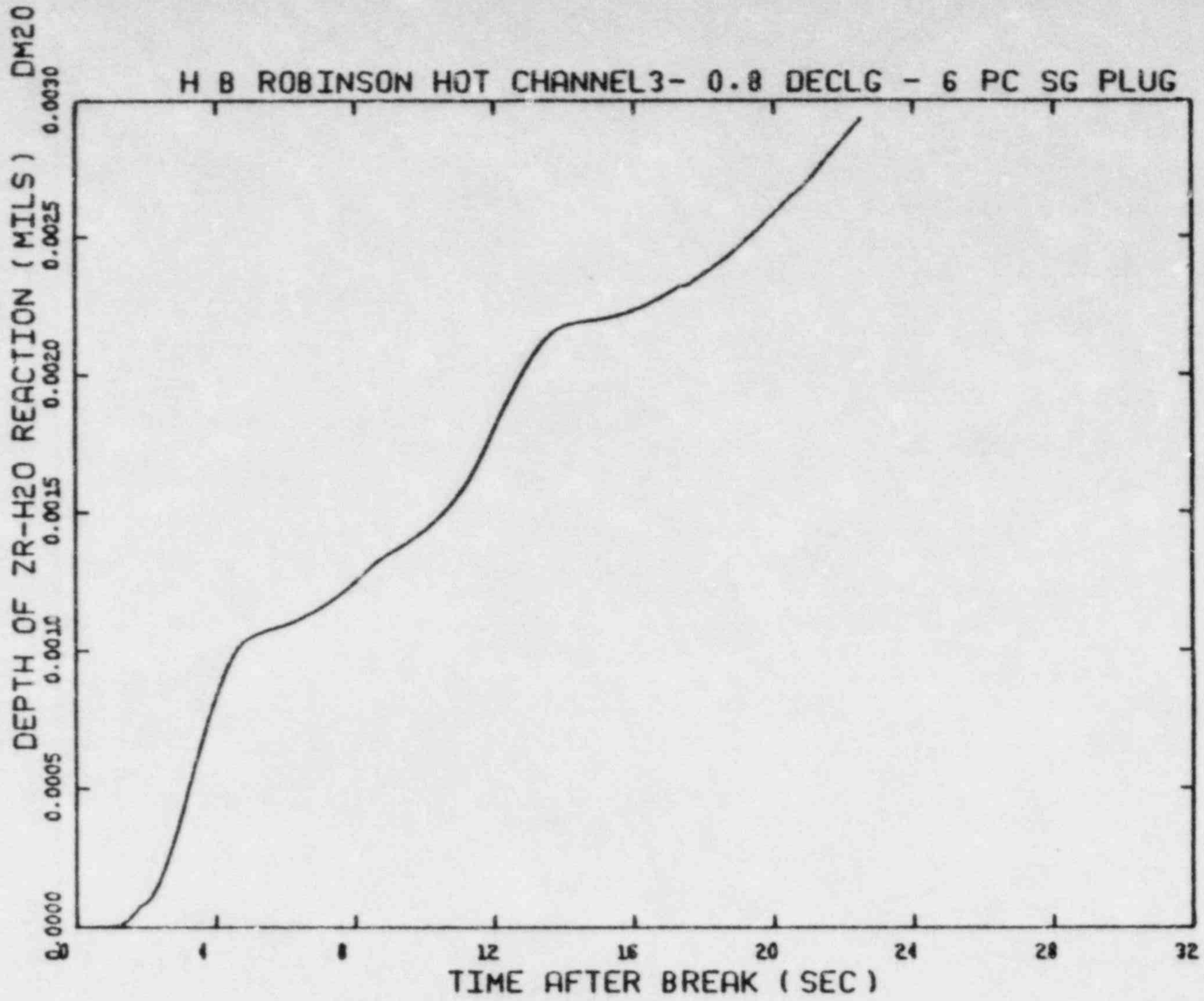


Figure 3.16 Depth of Metal-Water Reaction During Blowdown Period at PCT Node, 0.8 DECLG Break, 2 MWD/KgU

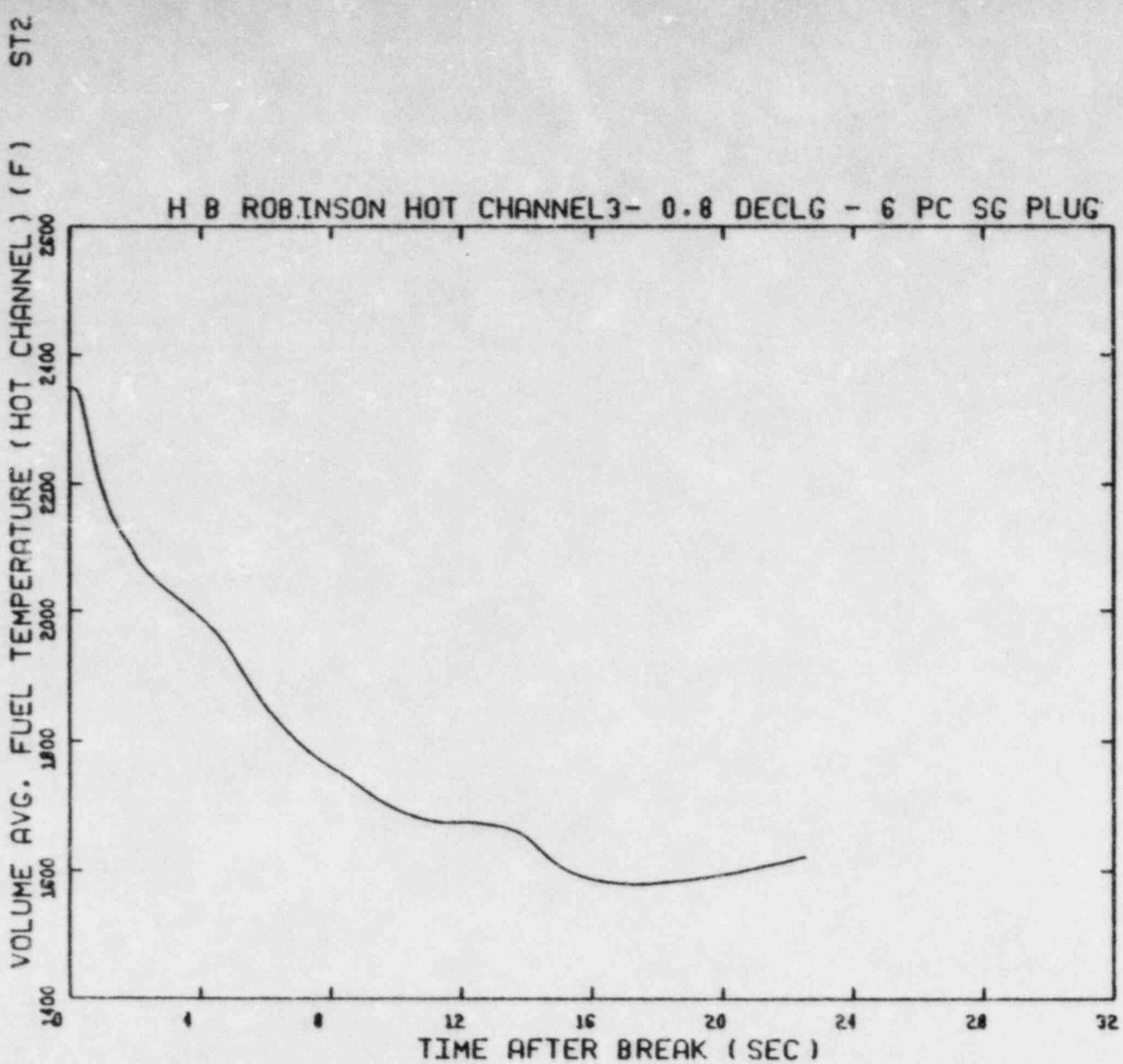


Figure 3.17 Average Fuel Temperature During Blowdown Period at PCT Node, 0.8 DECLG Break, 2 MWD/KgU

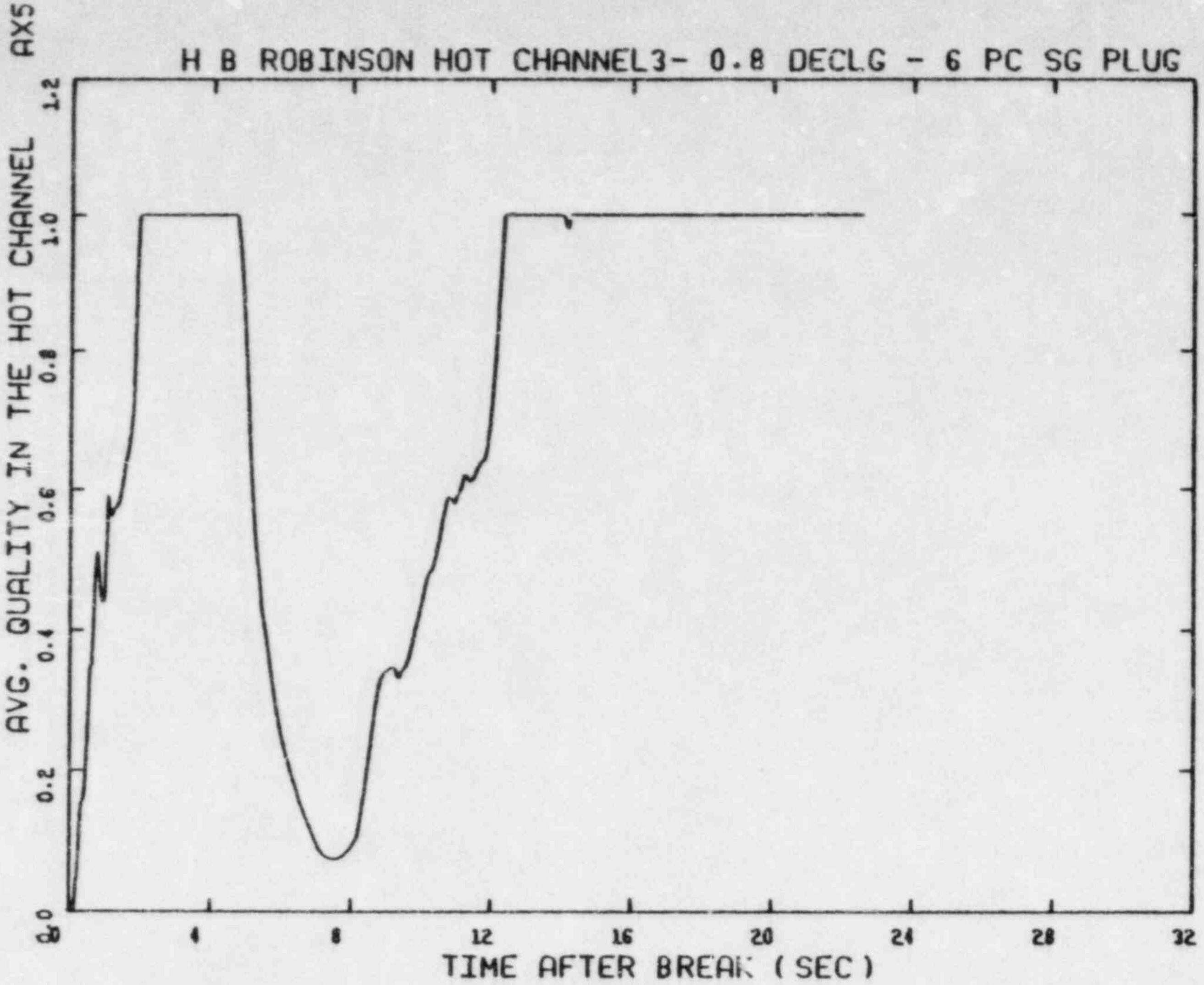


Figure 3.18 Hot Channel Average Quality, Center Volume
0.8 DECLG Break, 2 MWD/kgU

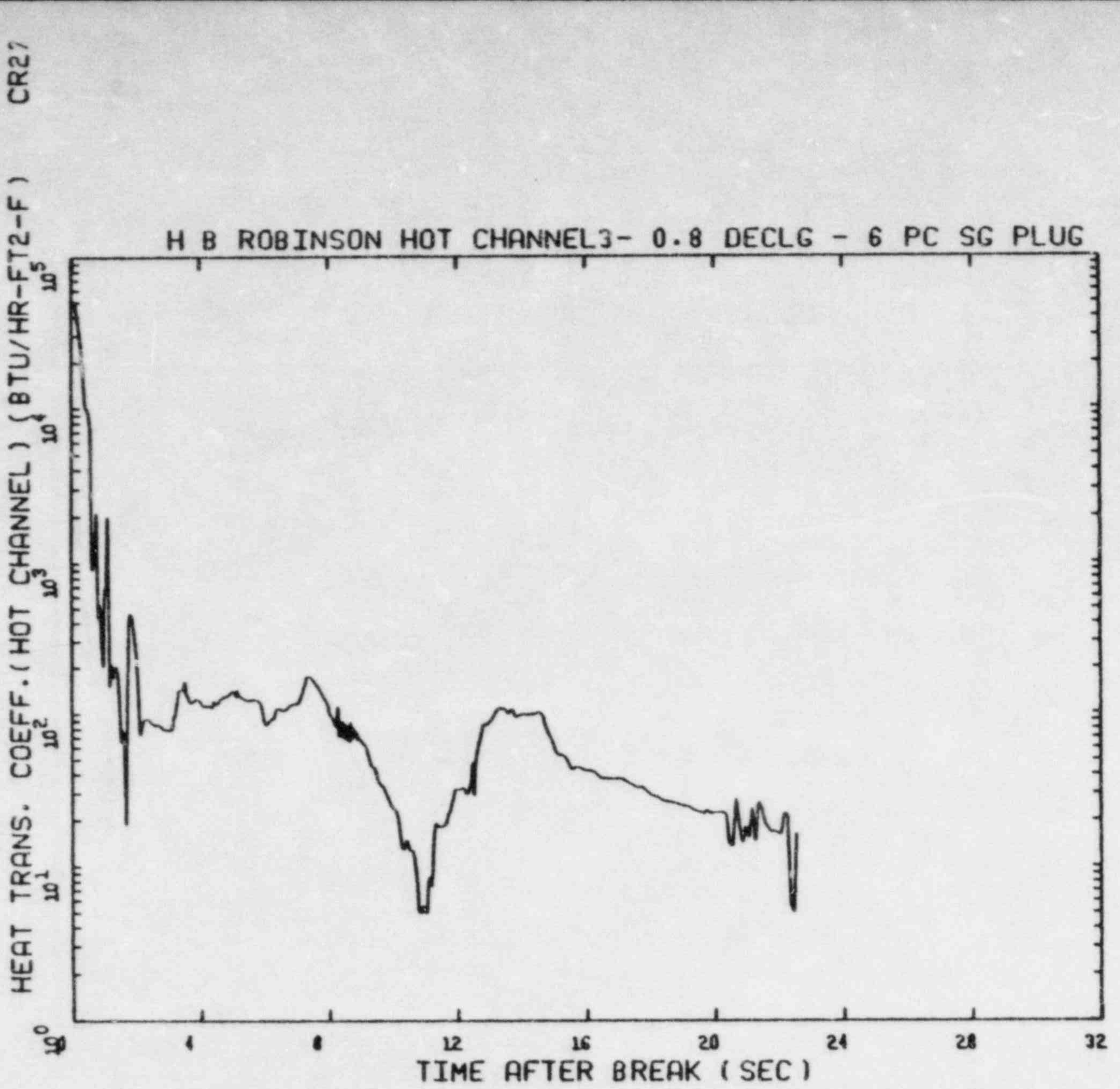


Figure 3.19 Heat Transfer Coefficient During Blowdown Period at PCT Node, 0.8 DECLG Break, 9 MWD/KgU

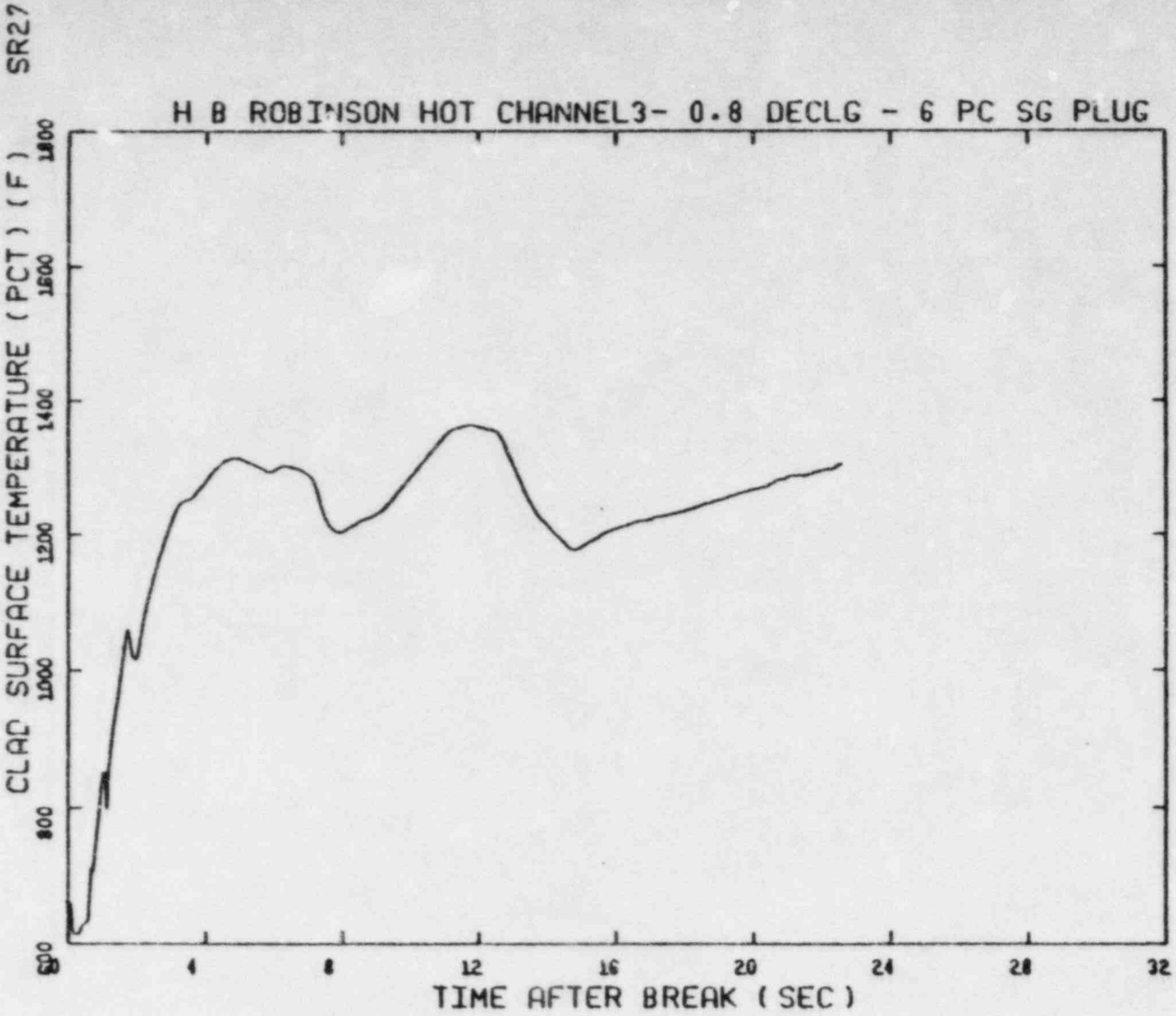


Figure 3.20 Clad Surface Temperature During Blowdown Period
PCT Node, 0.8 DECLG Break, 9 MWD/KgU

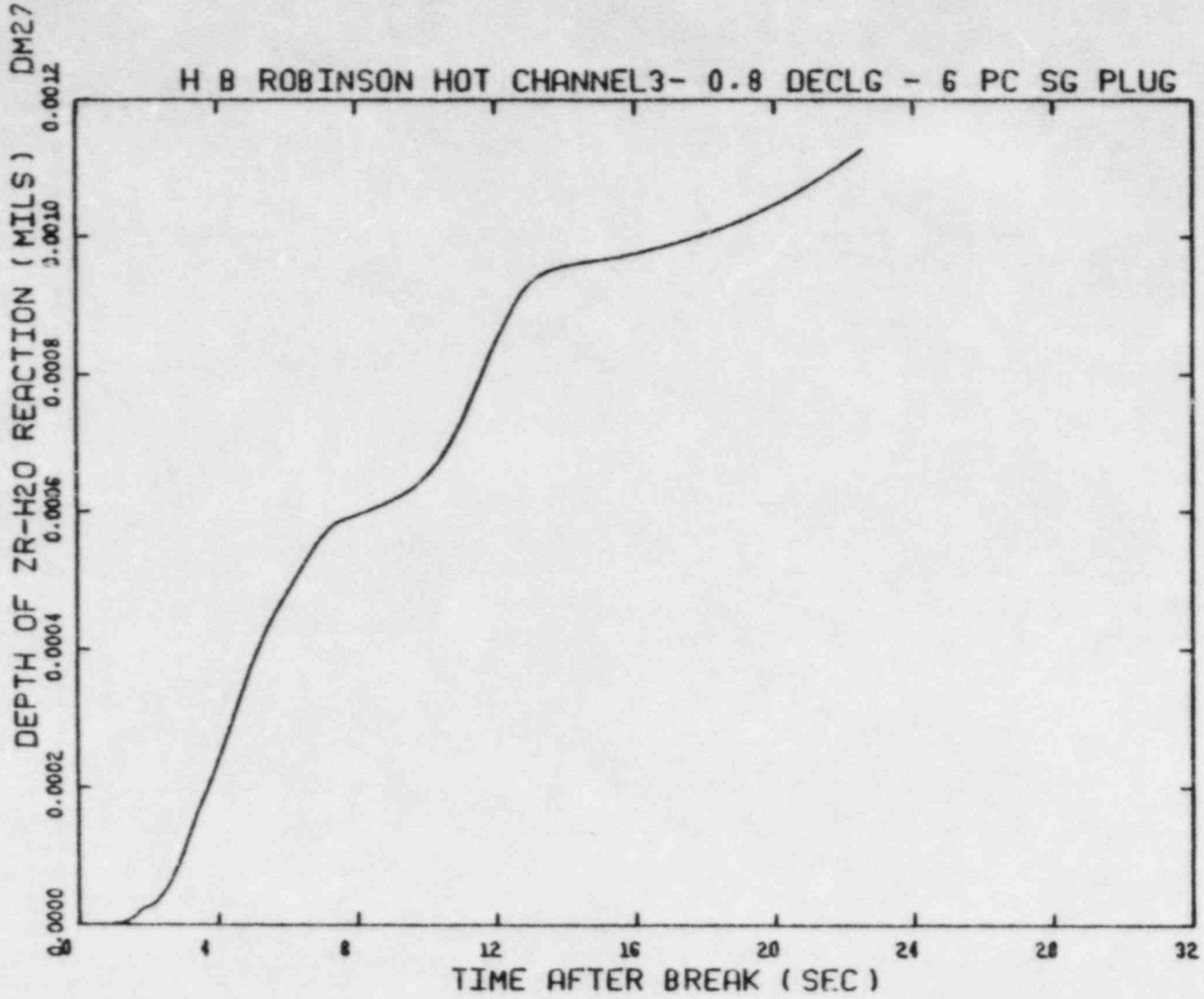


Figure 3.21 Depth of Metal-Water Reaction During Blowdown Period at PCT Node, 0.8 DECLG Break, 9 MWD/KgU

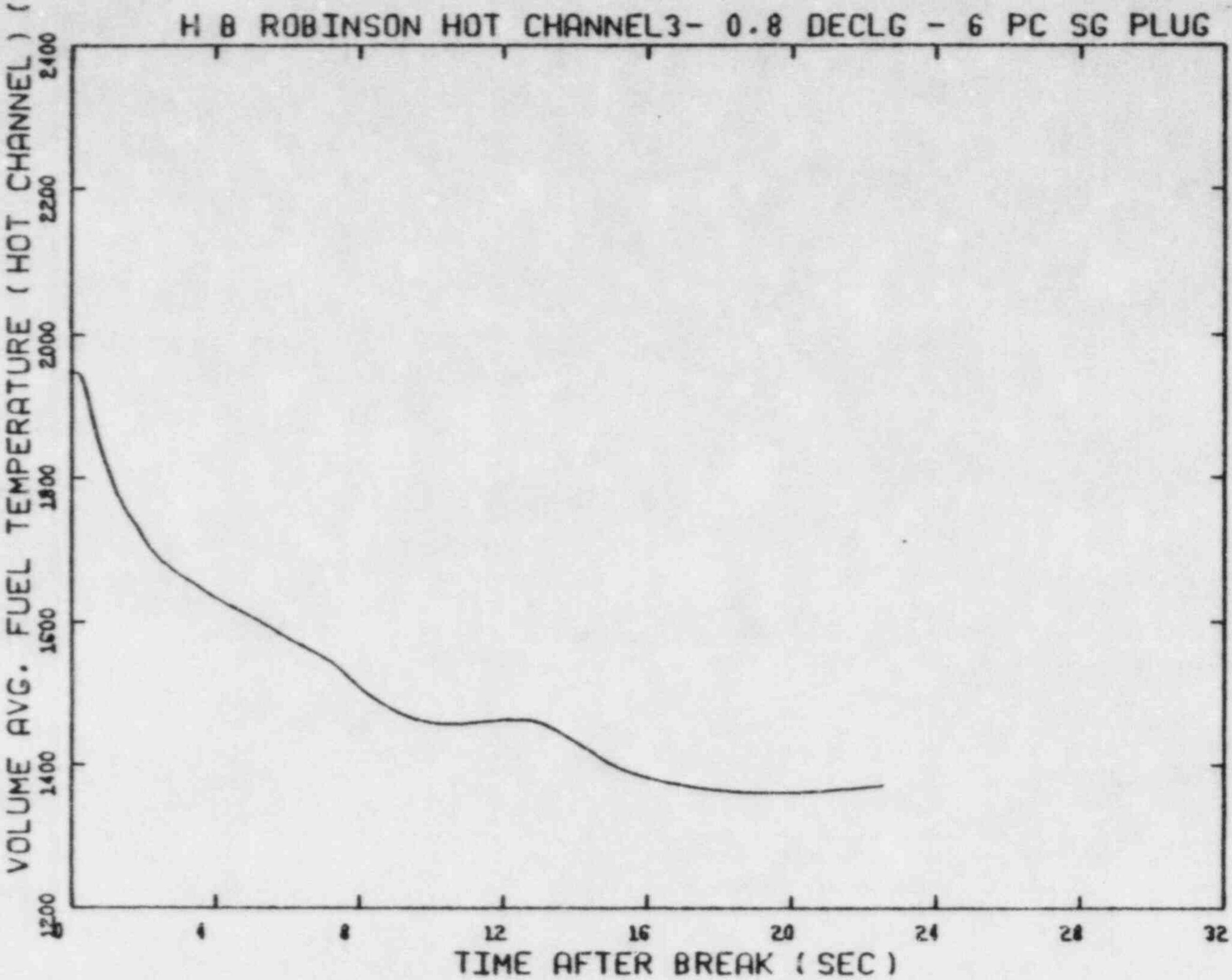


Figure 3.22 Average Fuel Temperature During Blowdown Period at PCT Node, 0.8 DECLG Break, 9 MWD/KgU

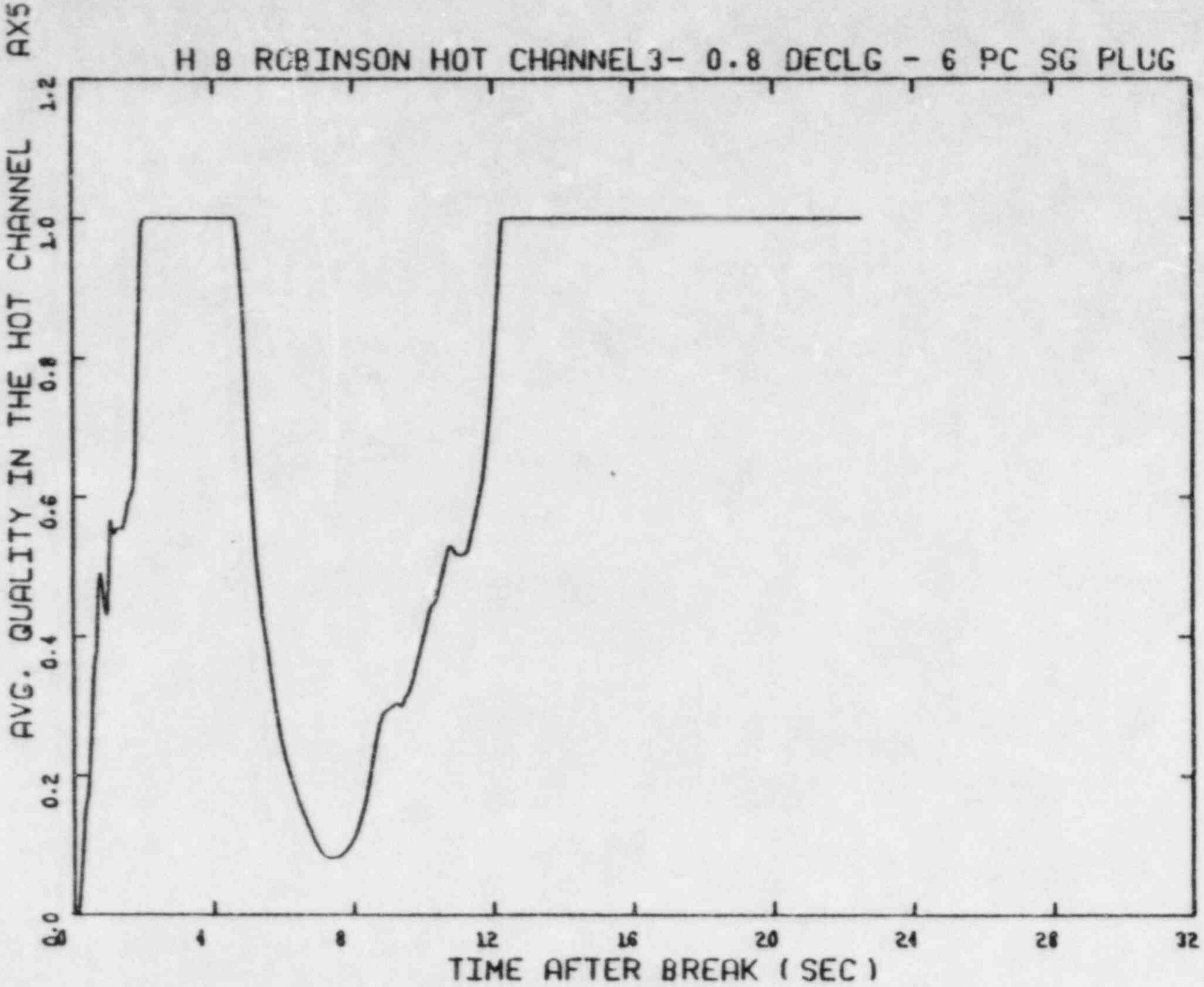


Figure 3.23 Hot Channel Average Quality, Center Volume
0.8 DECLG Break, 9 MWD/kgU

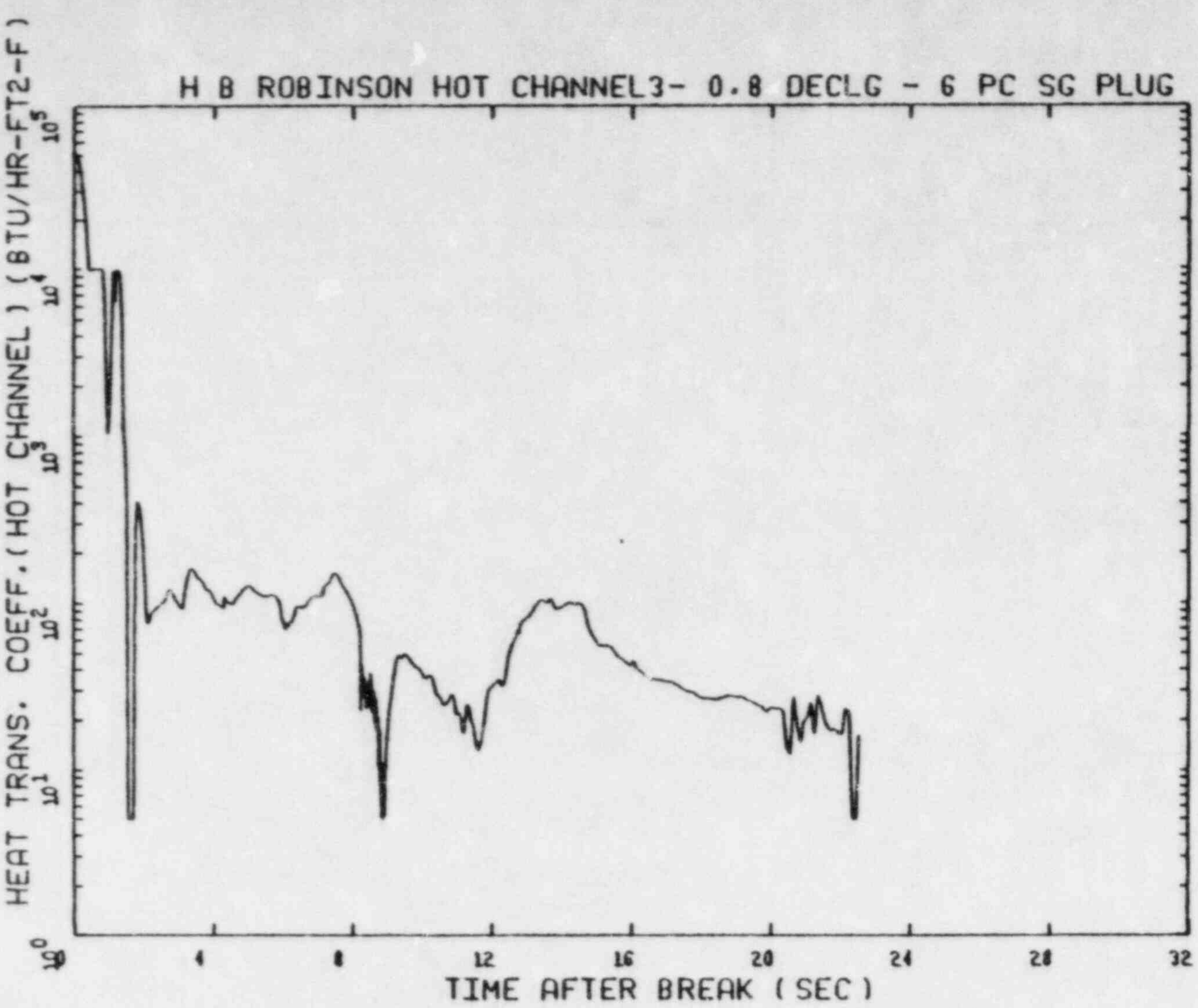


Figure 3.24 Heat transfer Coefficient During Blowdown Period at PCT Node, 0.8 DECLG Break, 49 MWD/kgU

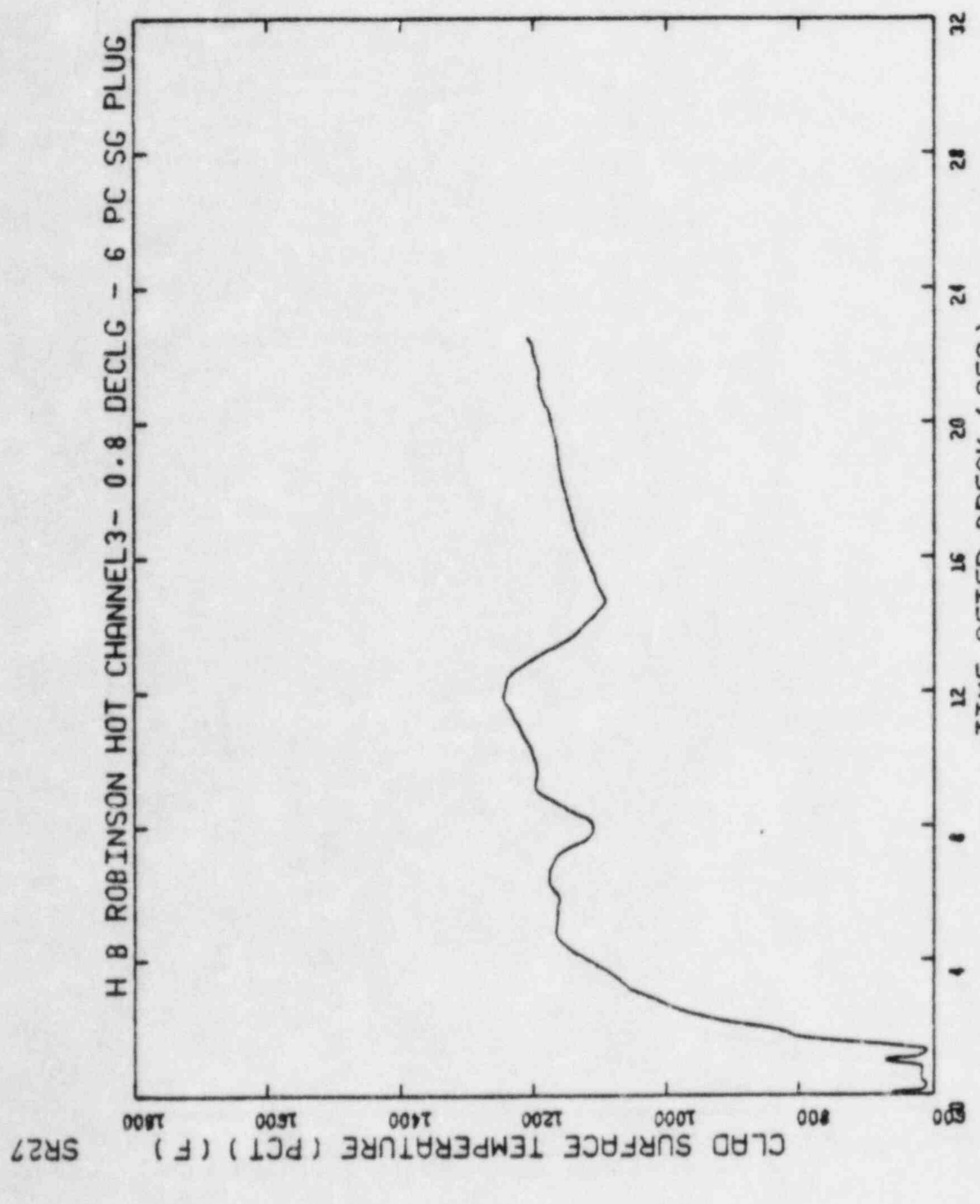


Figure 3.25 Clad Surface Temperature During Blowdown Period at PCT Node, 0.8 DECLG Break, 49 MWD/kgU

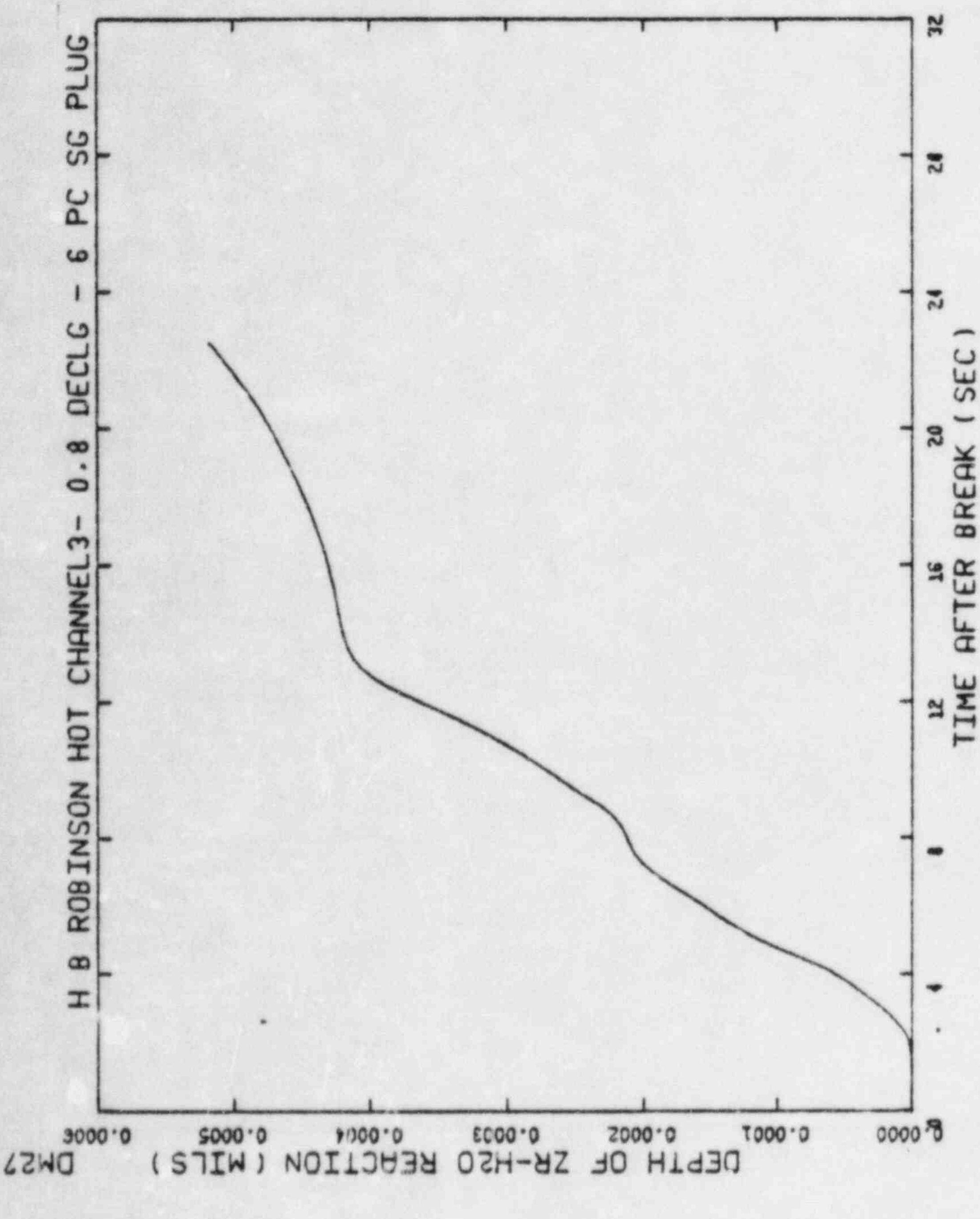


Figure 3.26 Depth of Metal-Water Reaction During Blowdown Period at PCT Mode, 0.8 DECLG Break, 49 MWD/kgU

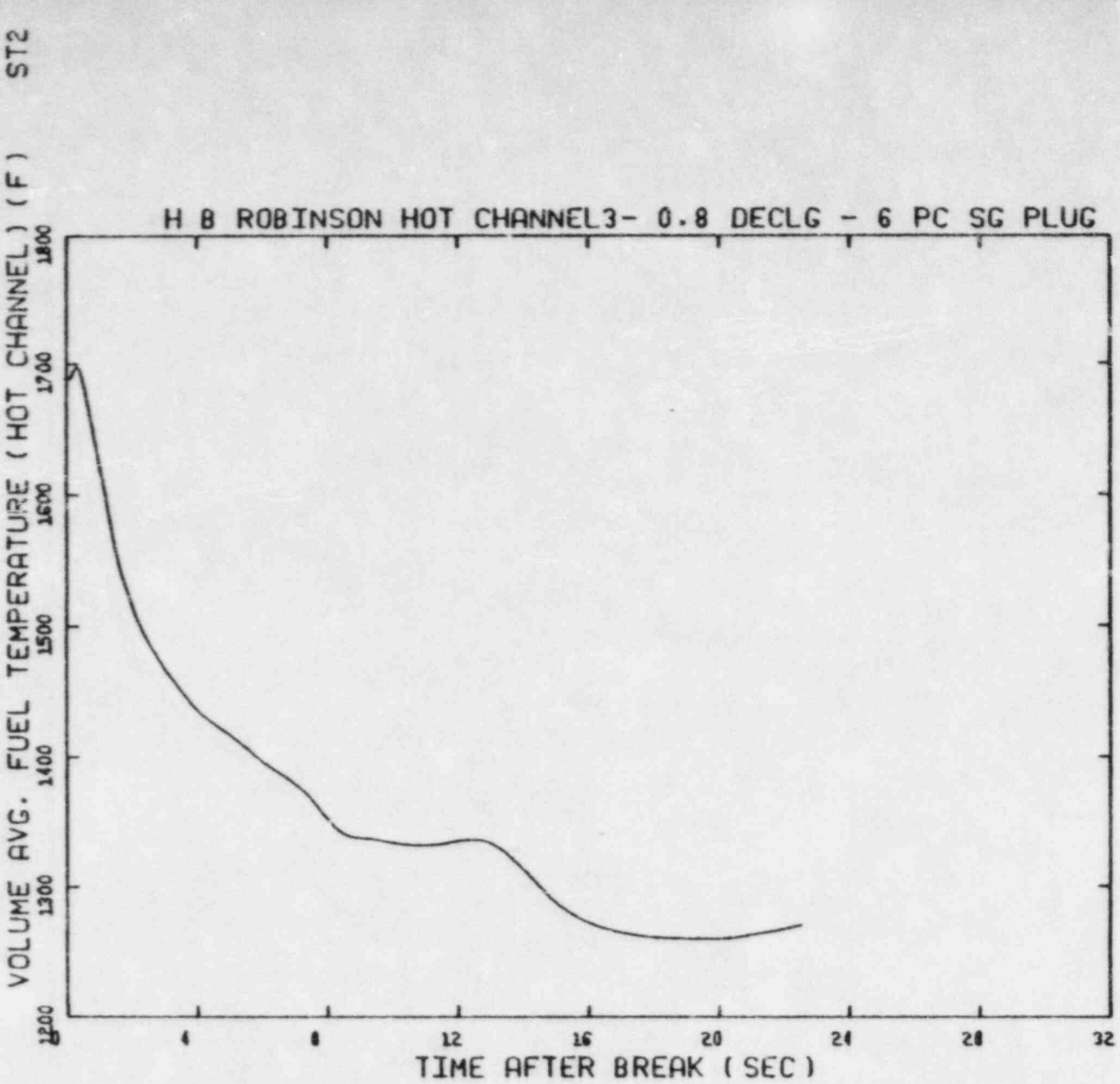


Figure 3.27 Average Fuel Temperature During Blowdown Period at PCT Node, 0.8 DECLG Break, 49 MWD/kgU

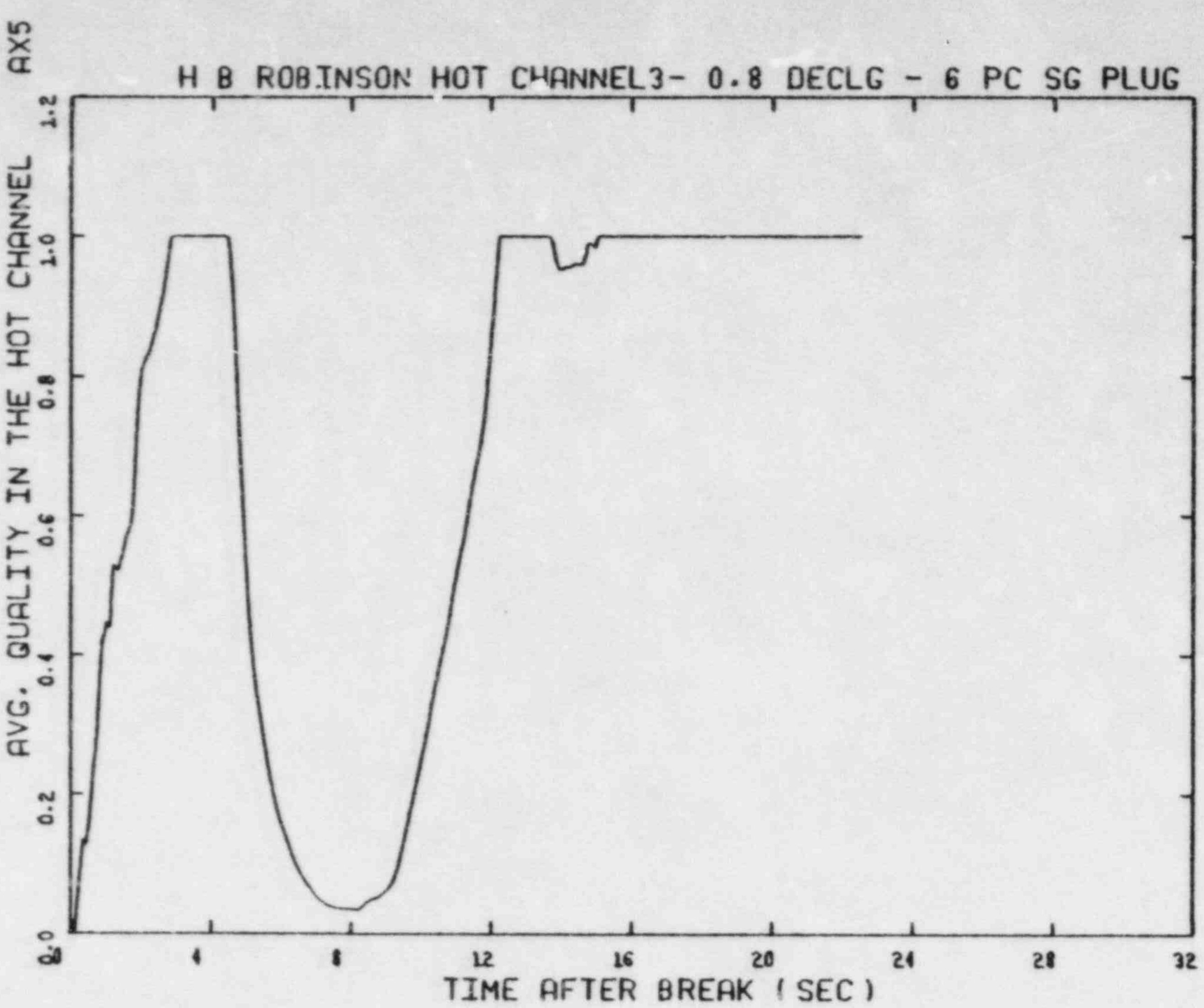


Figure 3.28 Hot Channel Average Quality, Center Volume
0.8 DECLG Break, 49 MWD/kgU

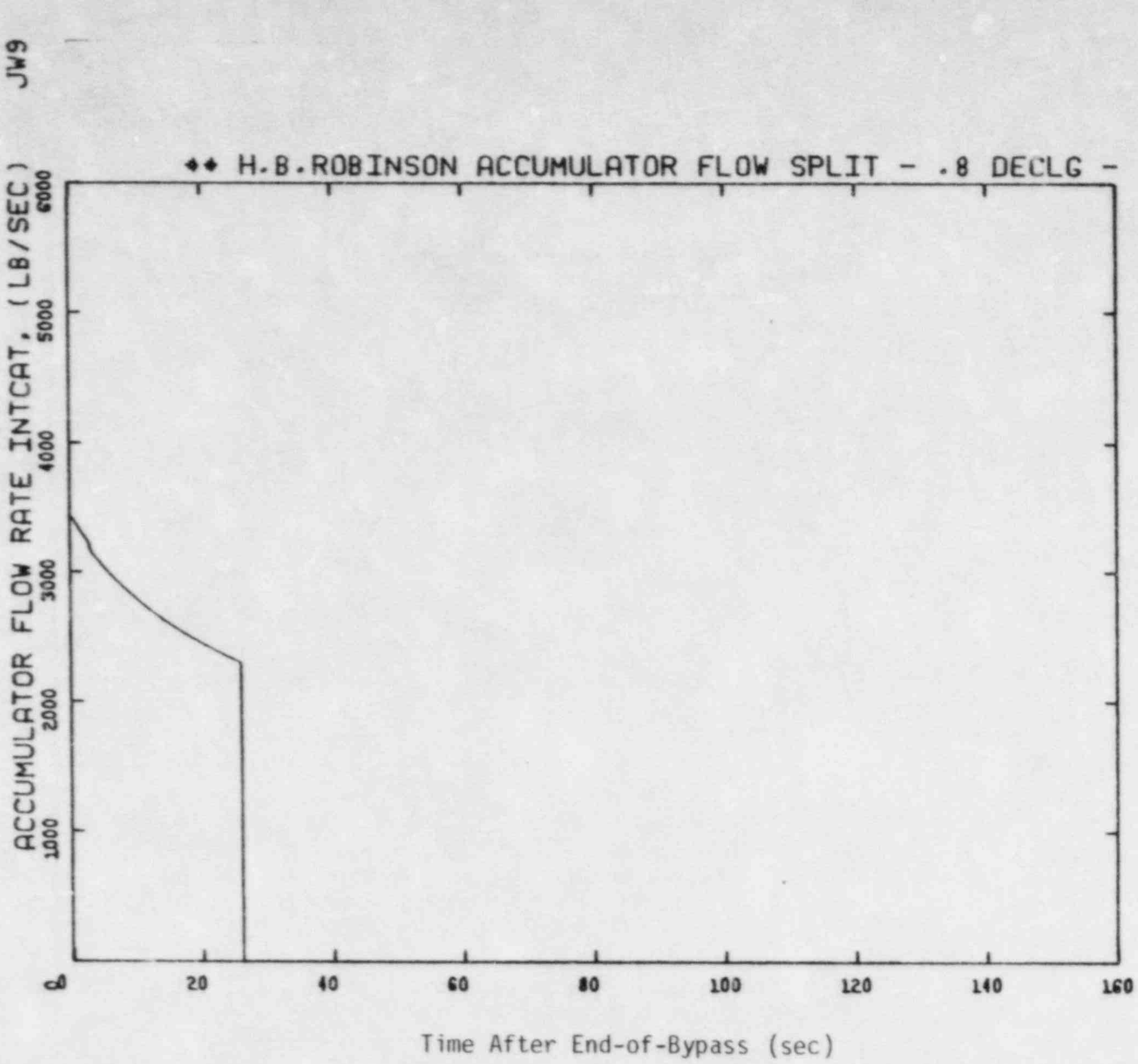


Figure 3.29 Accumulator (Intact) Flow During Refill and Reflood Periods, 0.8 DECLG Break

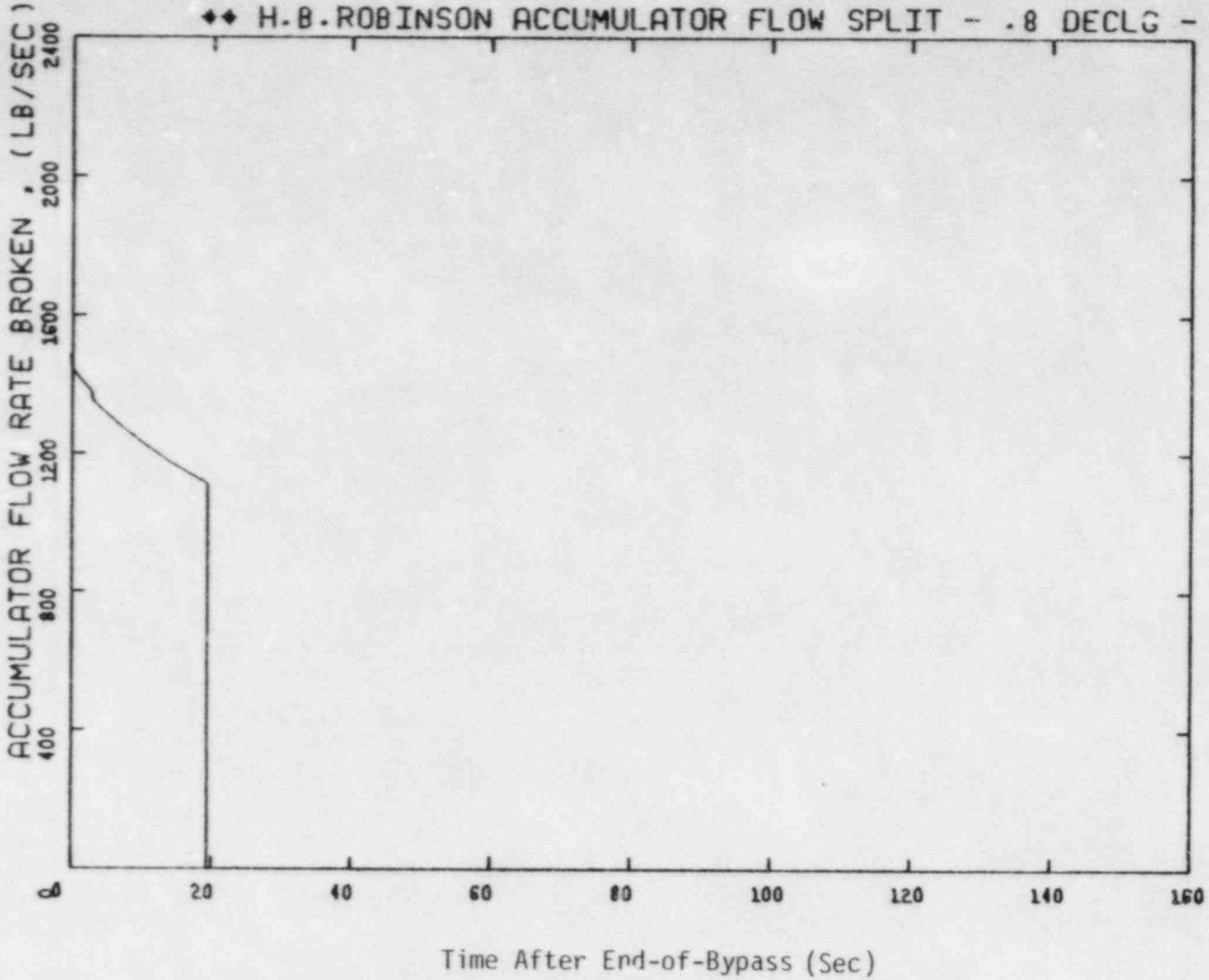


Figure 3.30 Accumulator (Broken) Flow During Refill and Reflood Periods, 0.8 DECLG Break

JWS

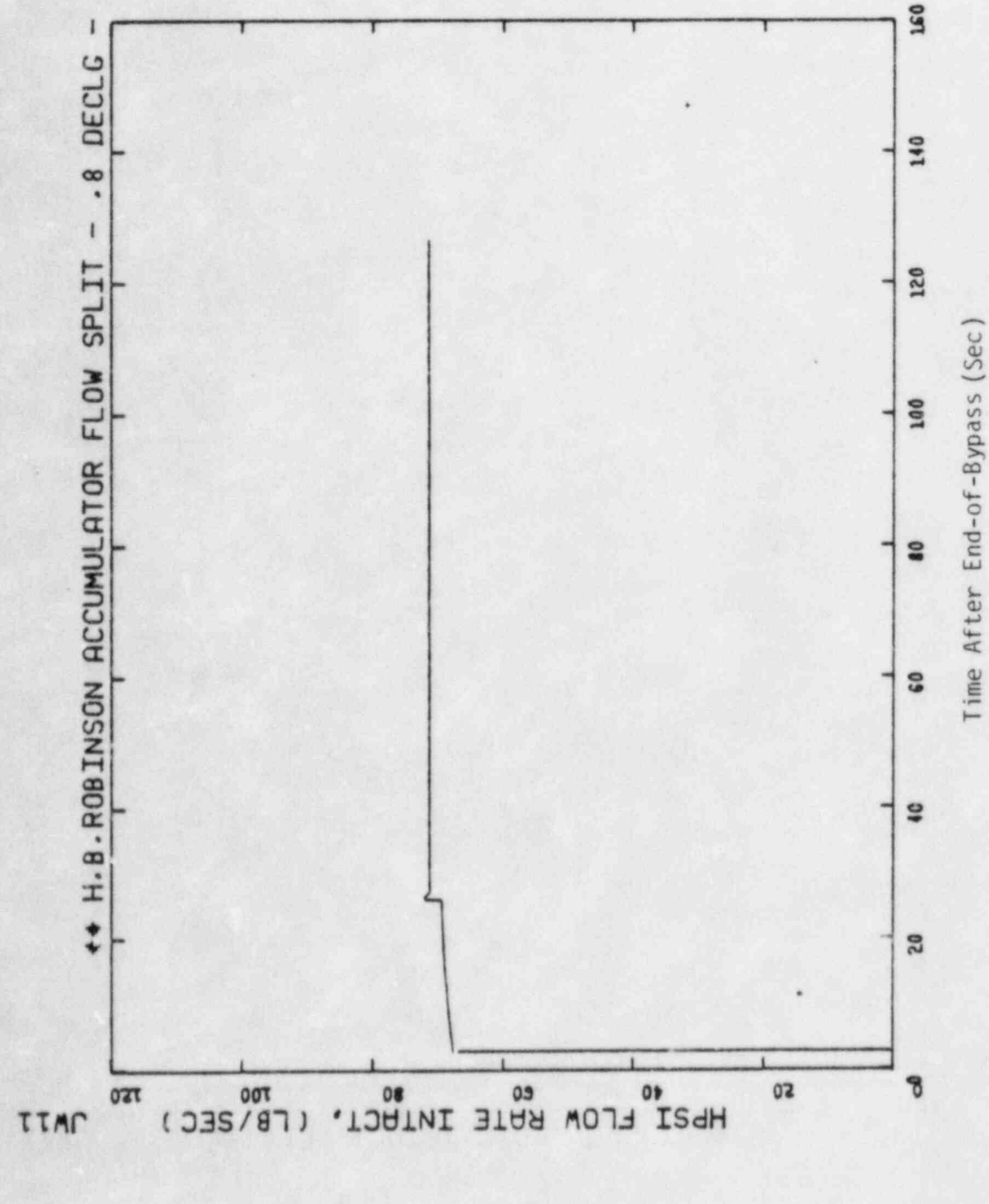


Figure 3.31 HPSI (Intact) Flow During Refill and Reflood Periods, 0.8 DECLG Break

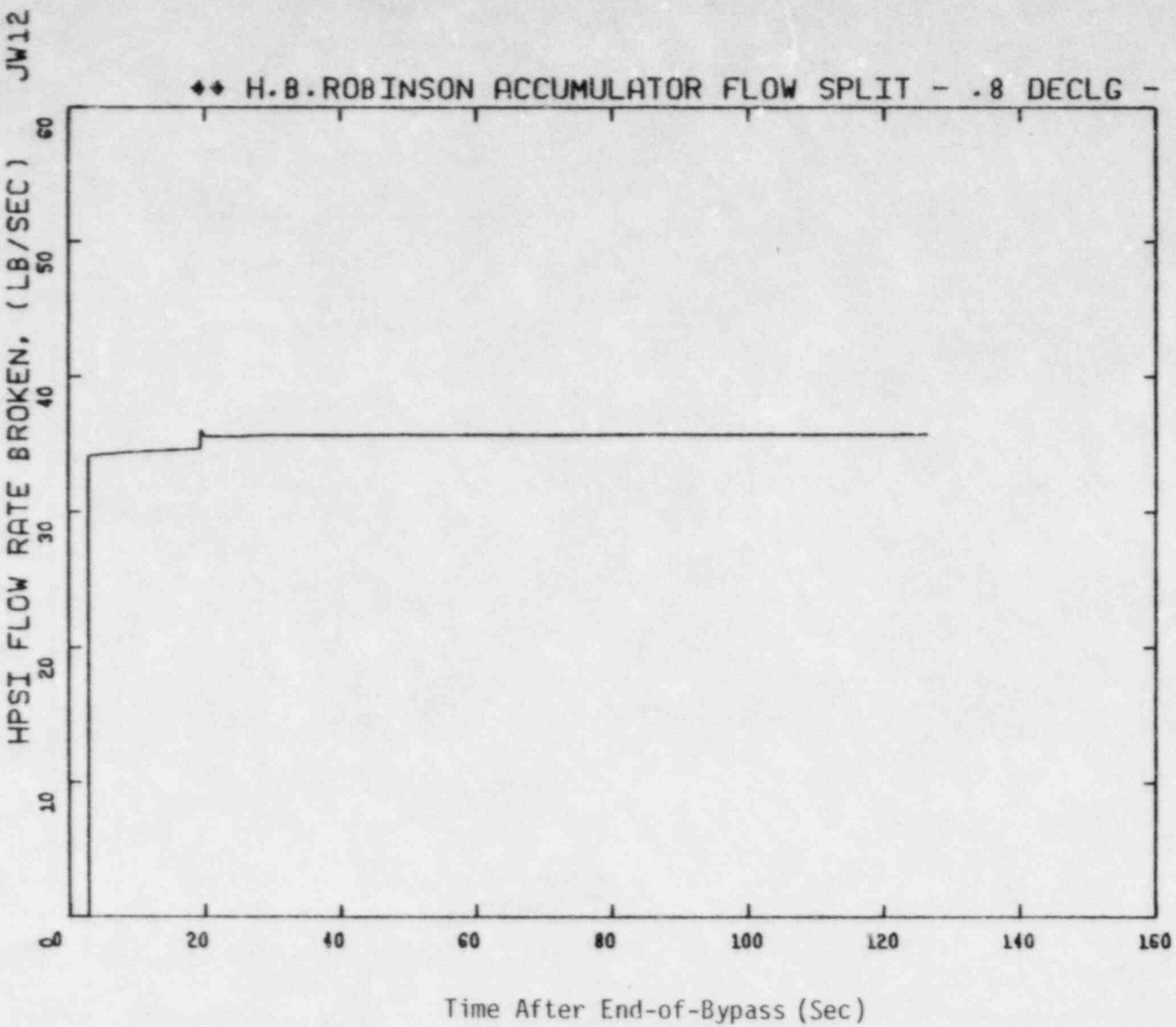


Figure 3.32 HPSI (Broken) Flow During Refill and Reflood Periods, 0.8 DECLG Break

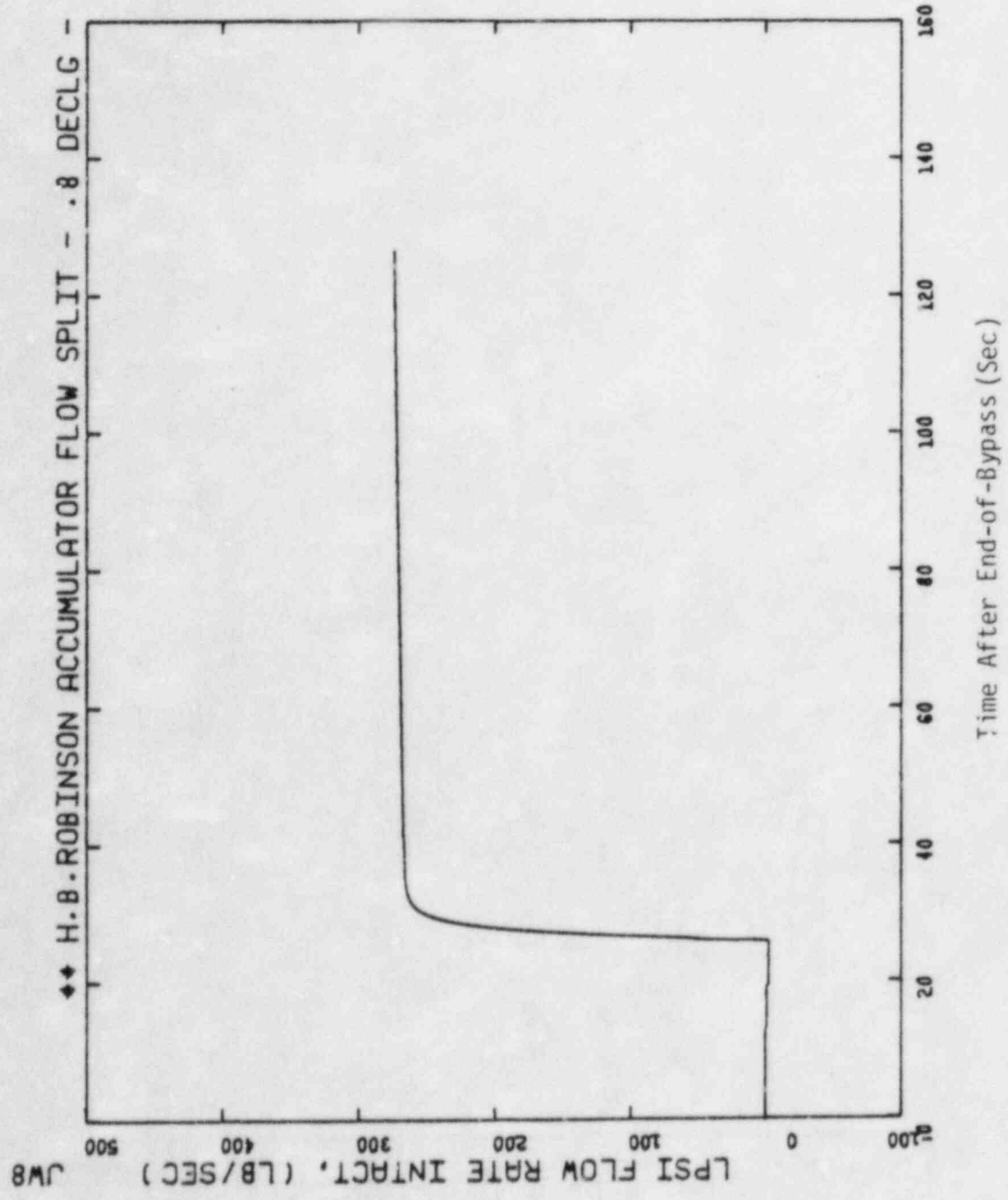


Figure 3.33 LPSI (Intact) Flow During Refill and Reflood Periods, 0.8 DECLG Break

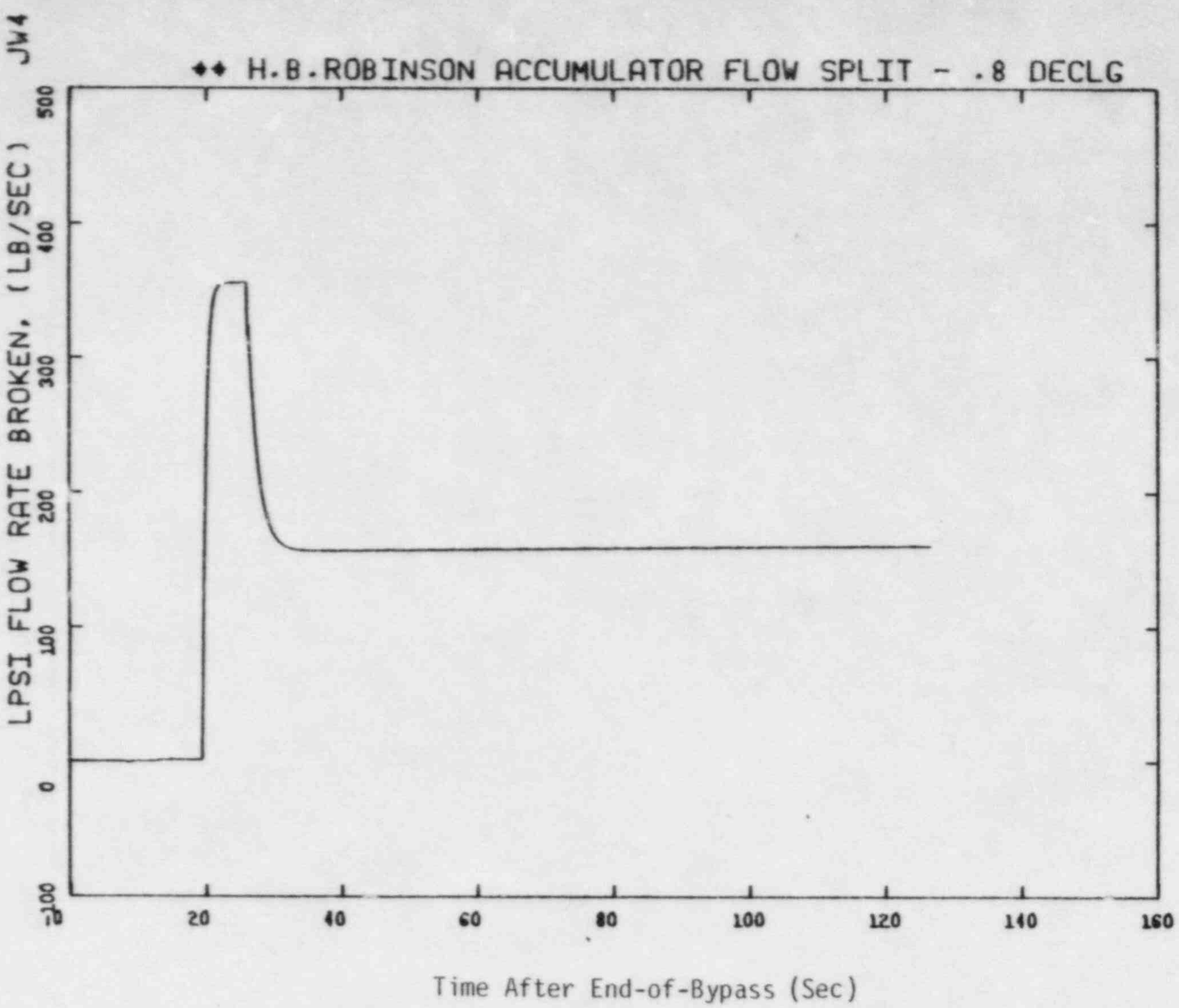


Figure 3.34 LPSI (Broken) Flow During Refill and Reflood Periods, 0.8 DECLG Break

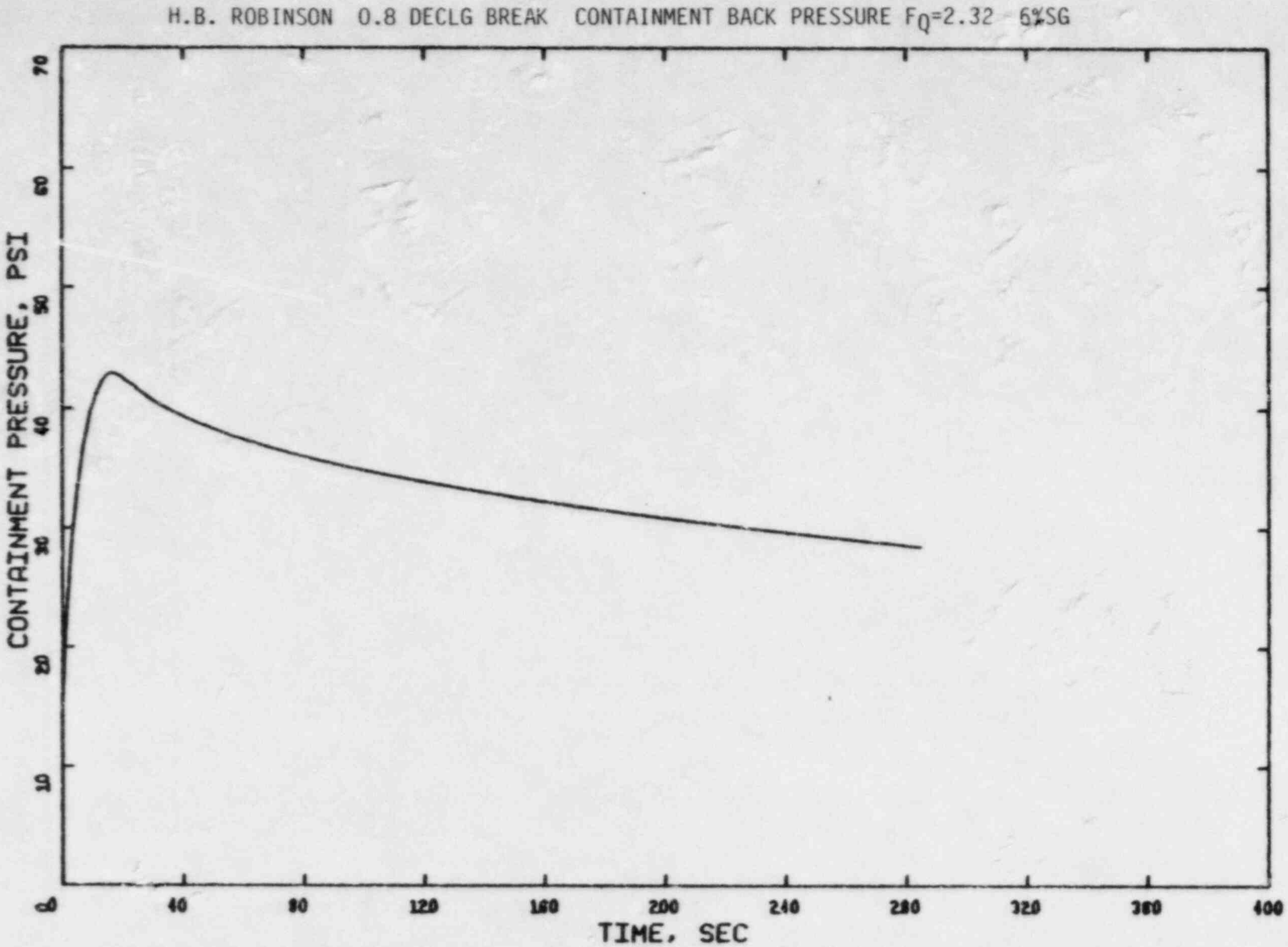


Figure 3.35 Containment Back Pressure, 0.8 DECLG Break

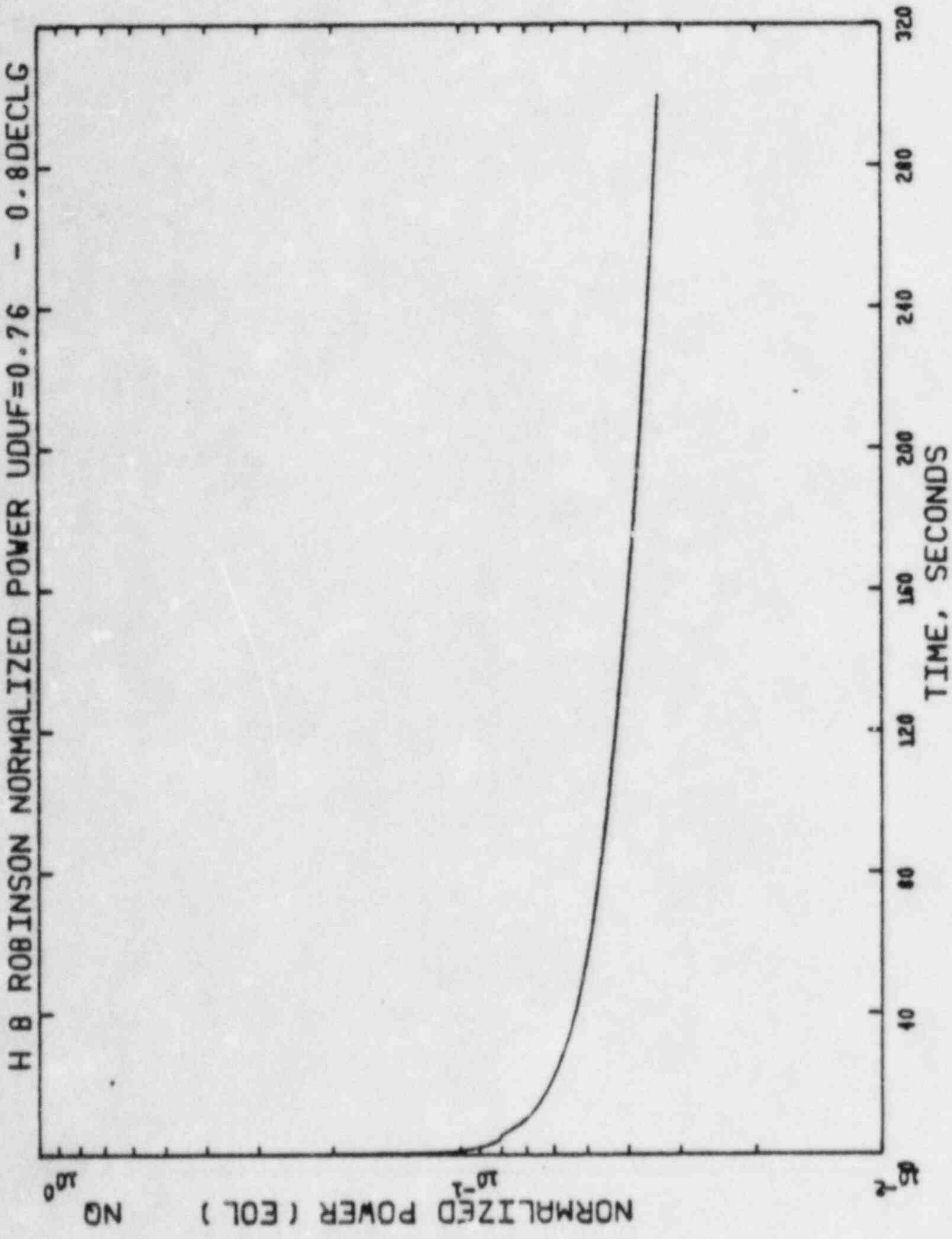


Figure 3.36 Normalized Power, 0.8 DECLG Break

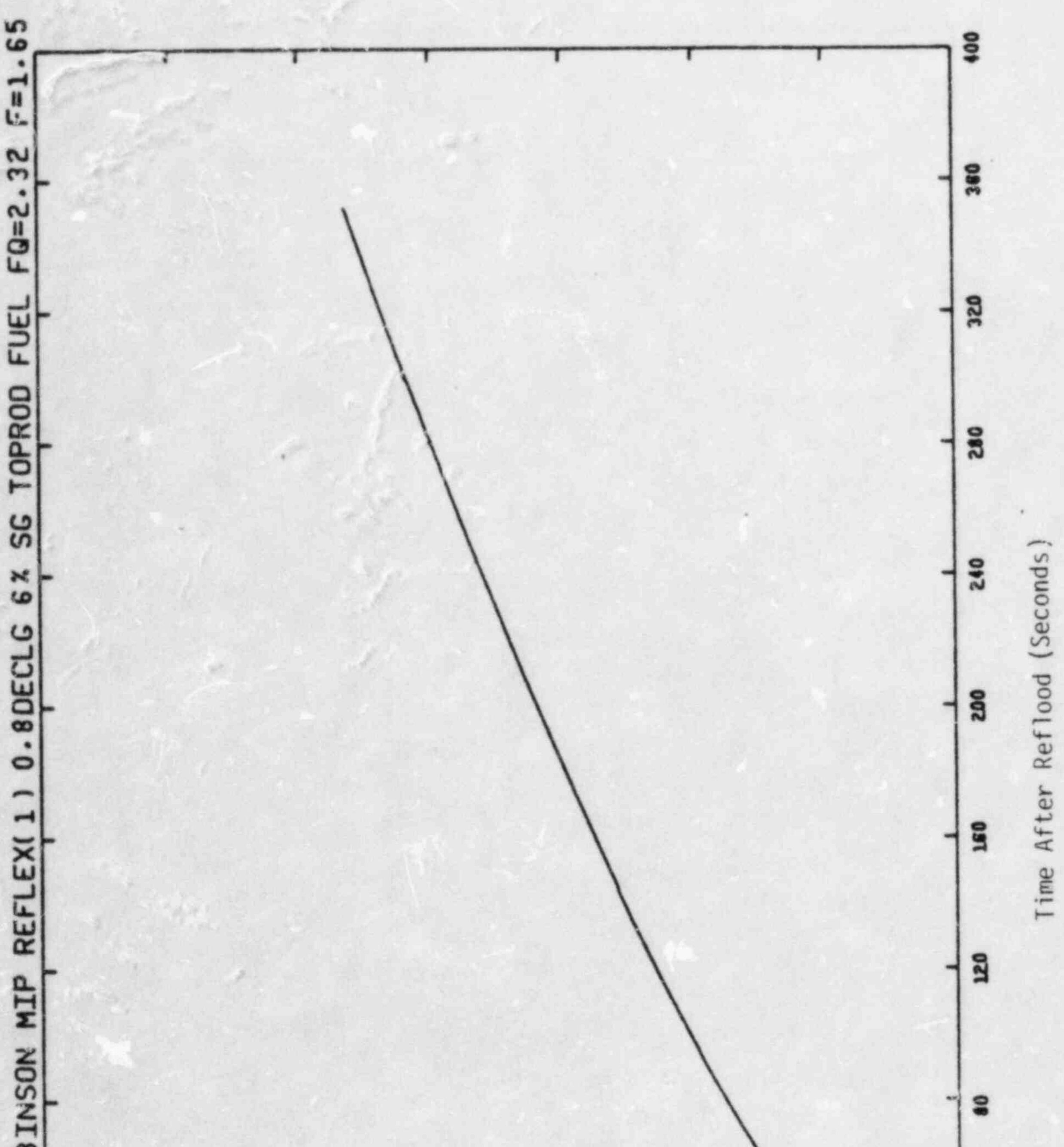


Figure 3.37 Reflood Core Mixture Level, 0.8 DECLG Break

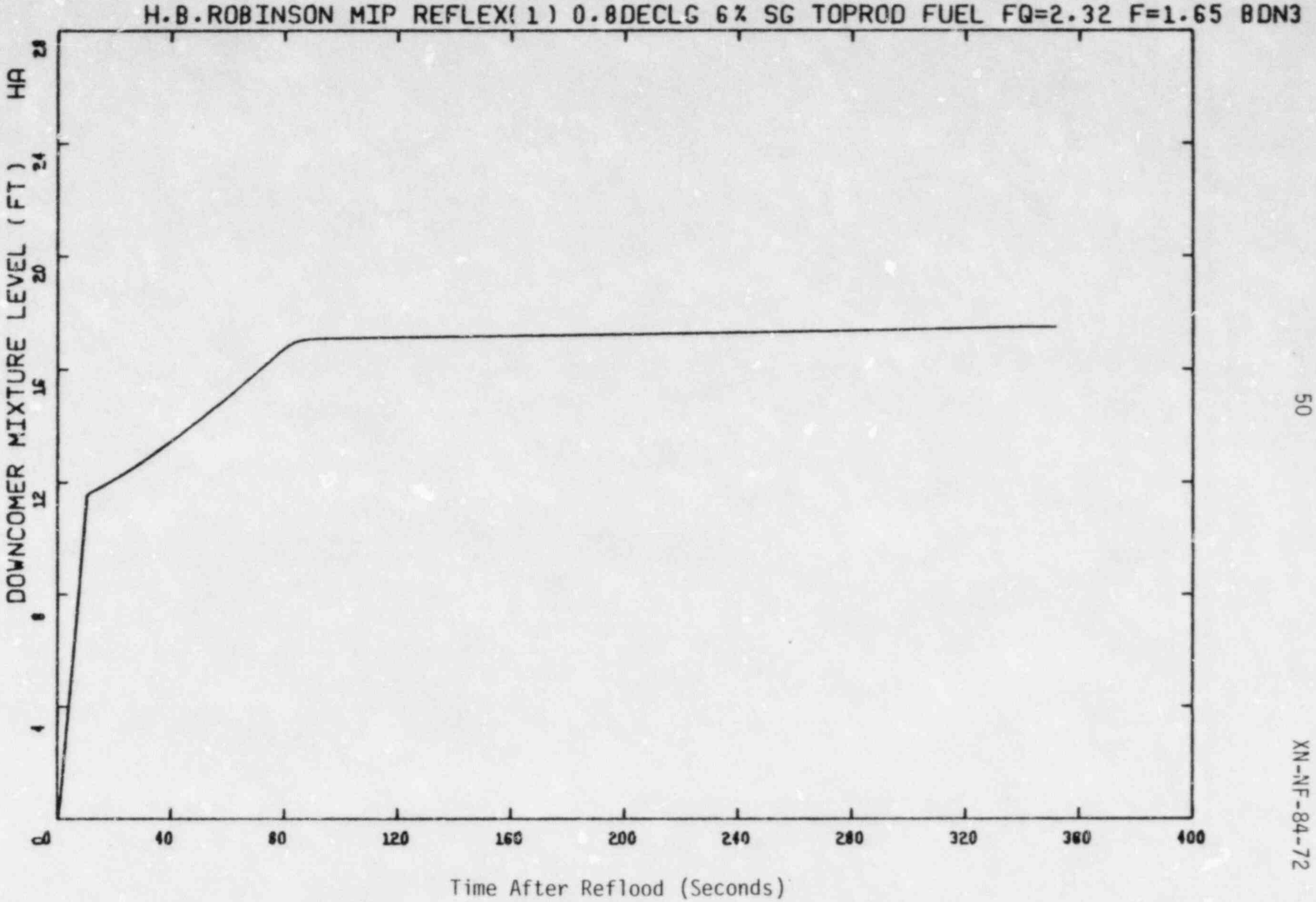


Figure 3.38 Reflood Downcomer Mixture Level, 0.8 DECLG Break

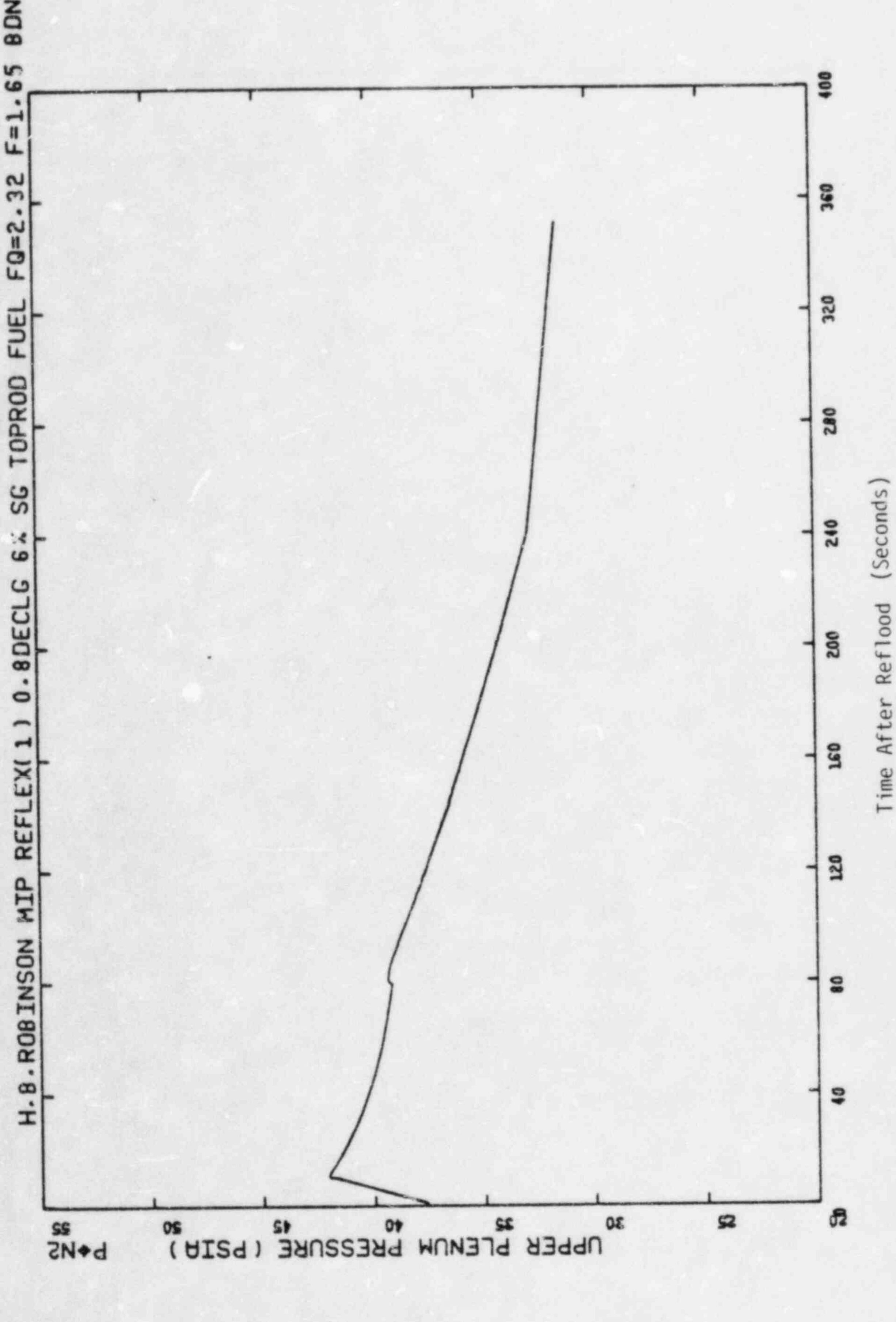


Figure 3.39 Reflood Upper Plenum Pressure, 0.8 DECLG Break

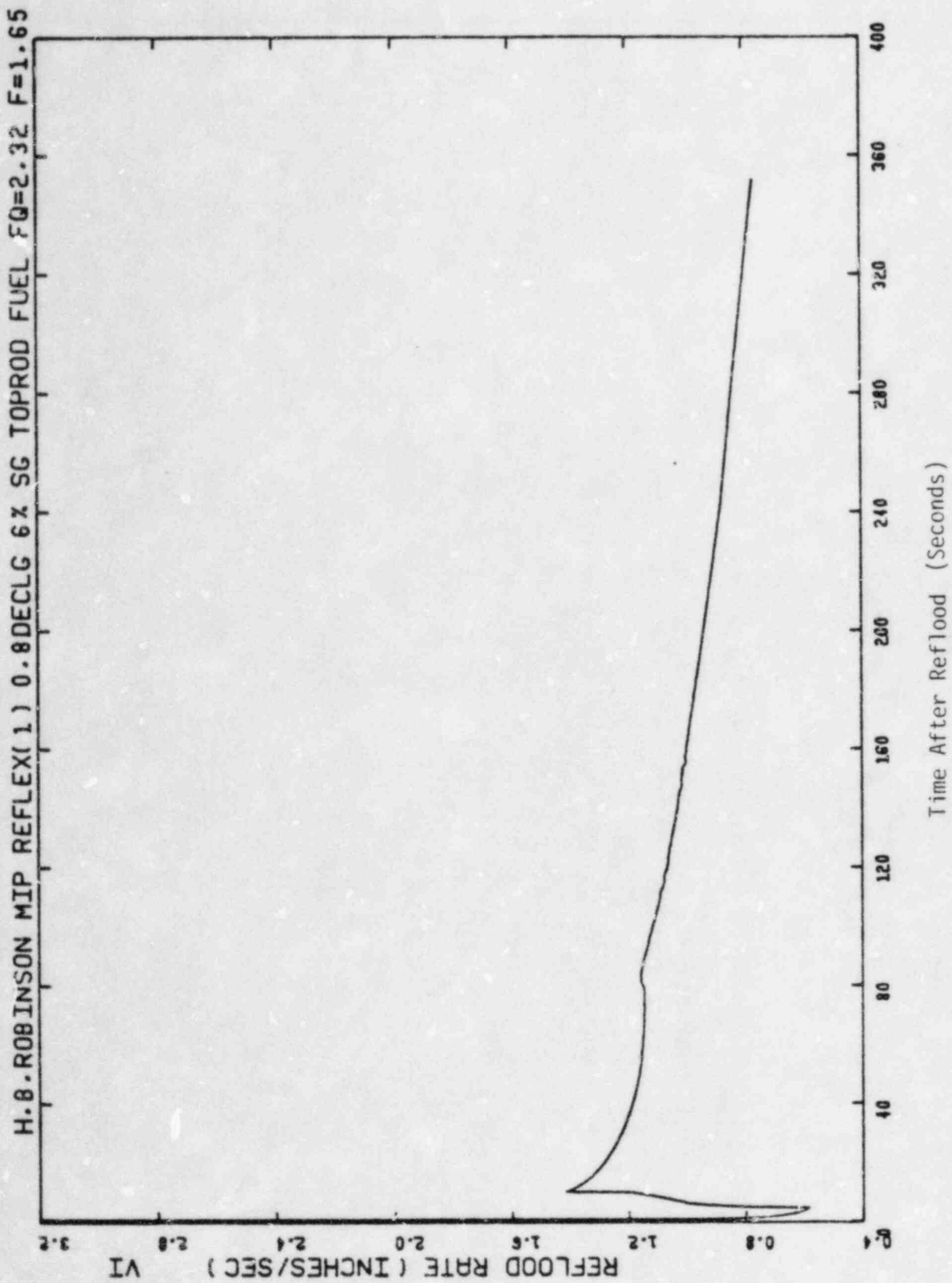


Figure 3.40 Core Flooding Rate, 0.8 DECLG Break

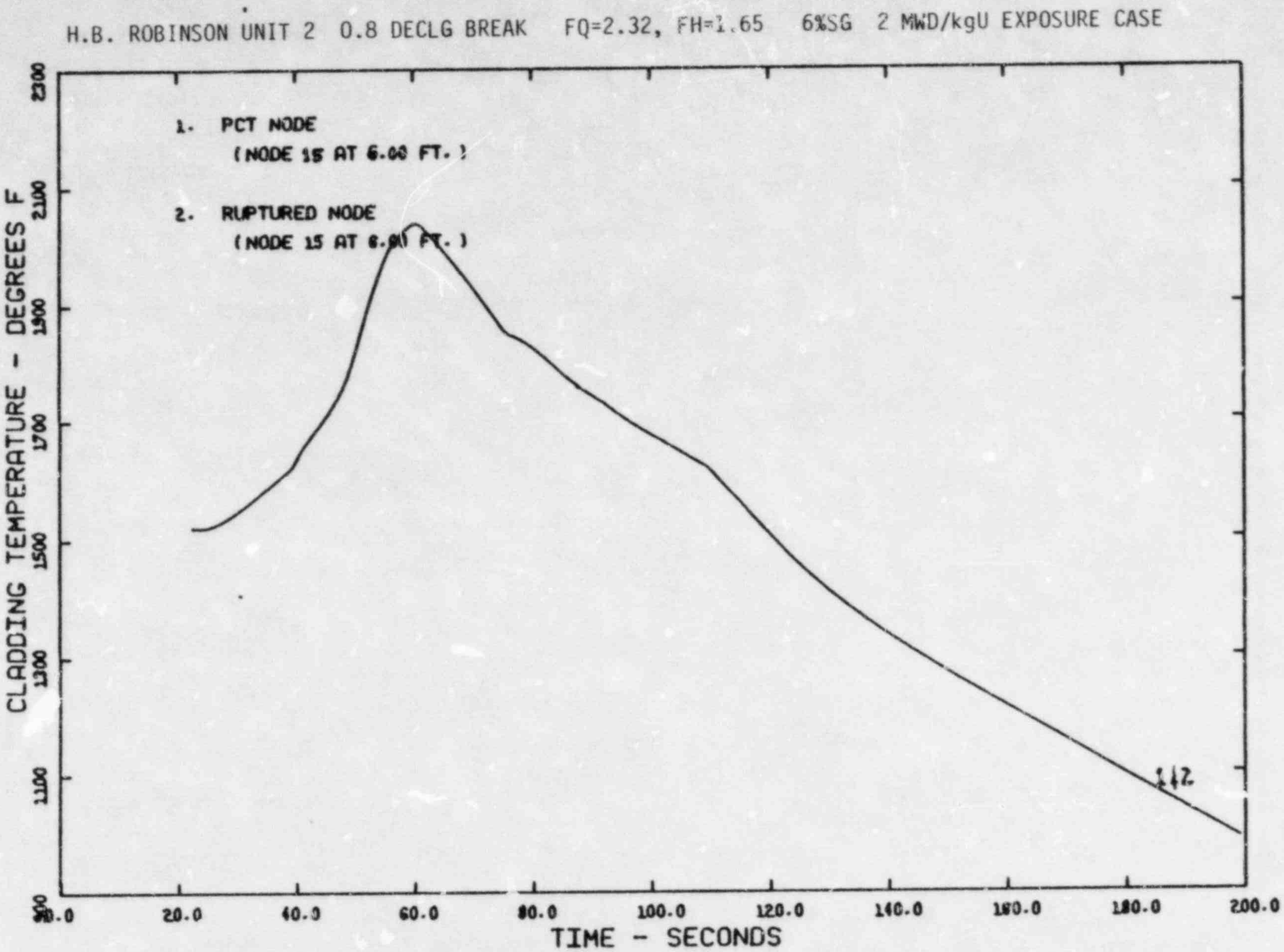


Figure 3.41 TOODEE2 Cladding Temperature vs. Time, 0.8 DECLG Break, 2 MWD/kgU Exposure

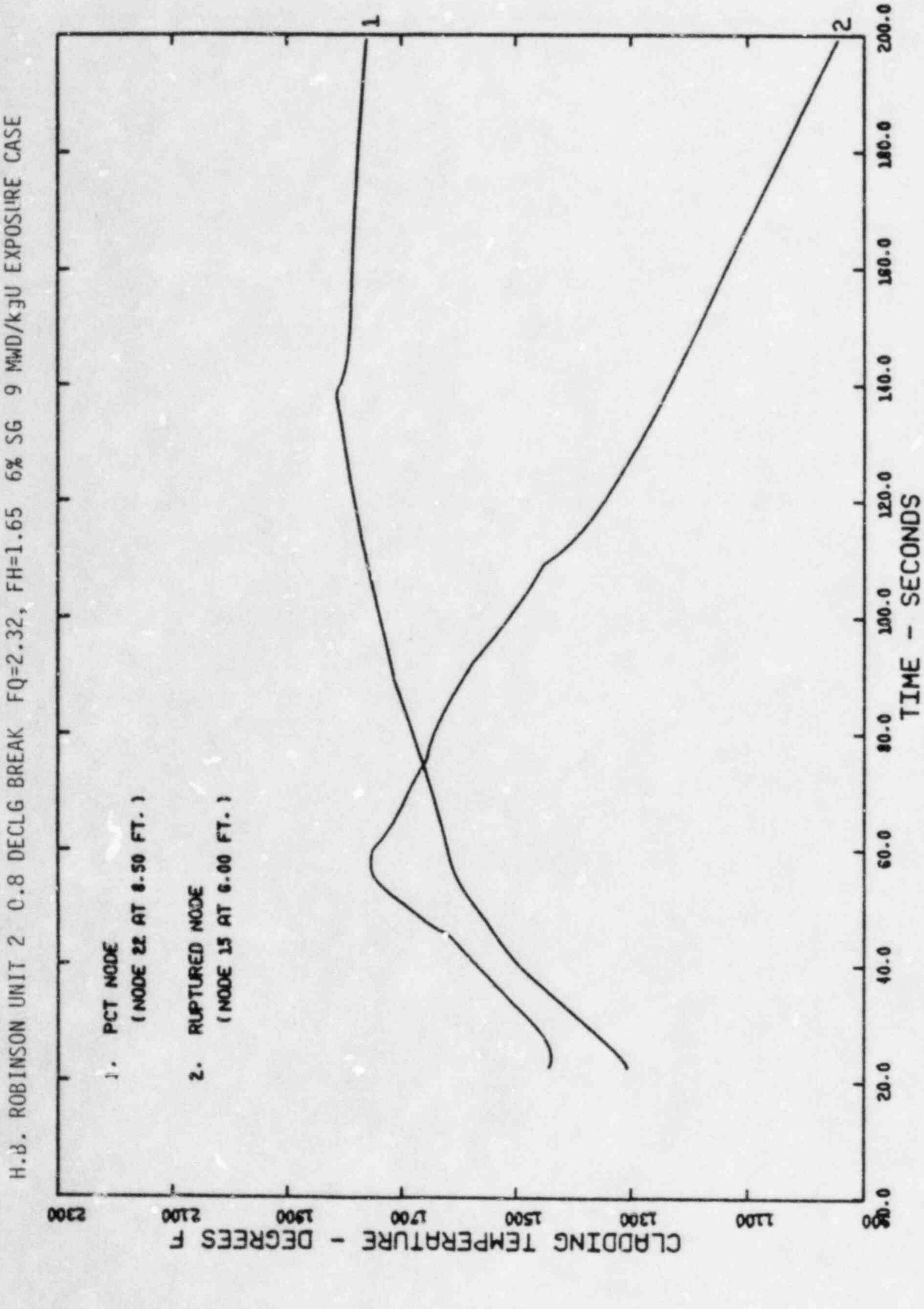
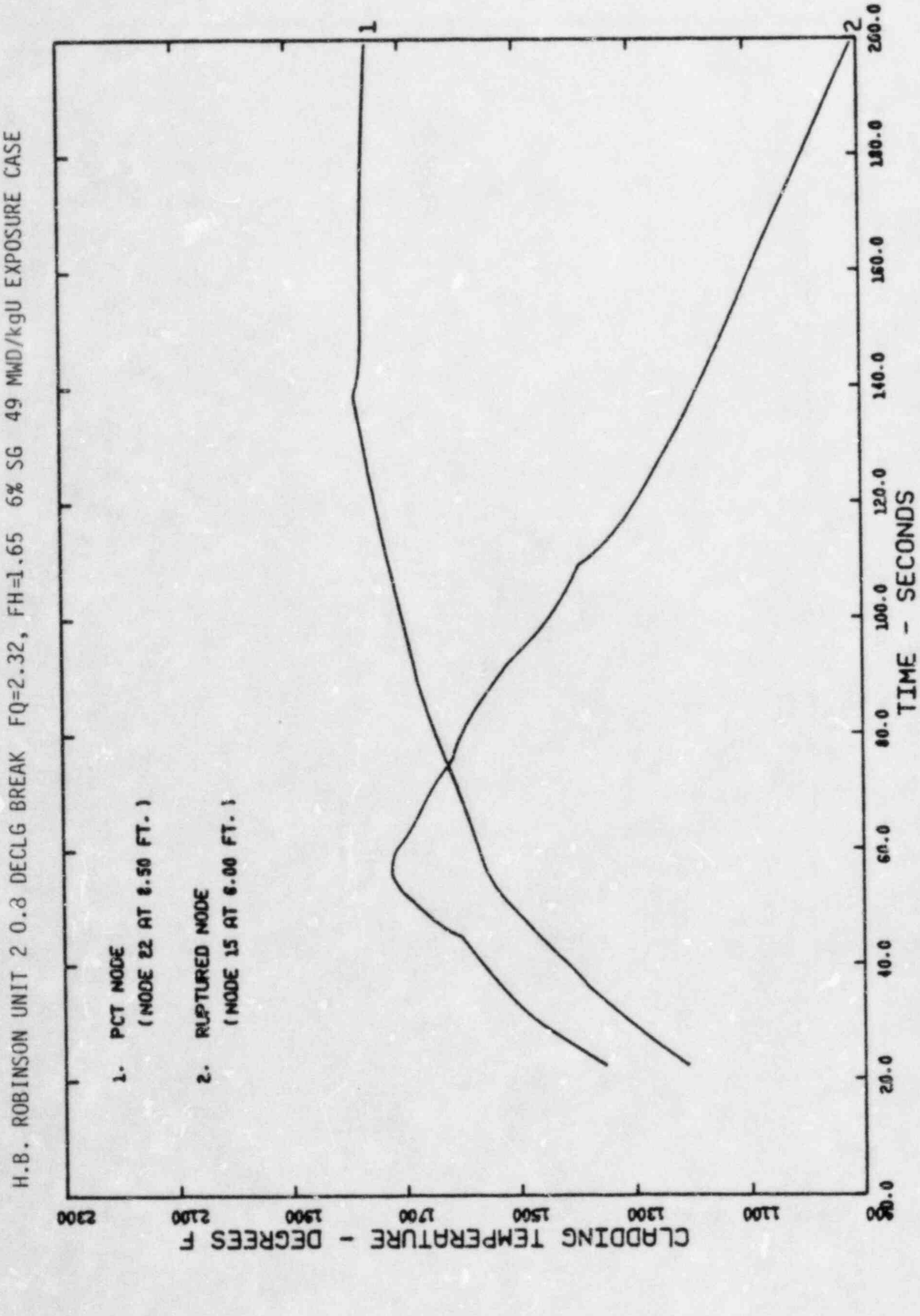


Figure 3.42 TOODEE2 Cladding Temperature vs. Time, 0.8 DECLG Break, 9 MWD/kgU Exposure


H.B. ROBINSON UNIT 2 0.8 DECLG BREAK FQ=2.32, FH=1.65 6% SG 49 MWD/kgU EXPOSURE CASE
1. PCT NODE
(NODE 22 AT 8.50 FT.)
2. RUPTURED NODE
(NODE 15 AT 6.00 FT.)
CLADDING TEMPERATURE - DEGREES F
TIME - SECONDS
2300
2100
1900
1700
1500
1300
1100
900
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
1
2

4.0 CONCLUSION

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Emergency Core Cooling System for H.B. Robinson Unit 2 will meet the Acceptance Criteria as presented in 10 CFR 50.46, with the 2.32 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^{o}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.

5.0 REFERENCES

1. "H.B. Robinson Unit No. 2 LOCA Analyses Using the ENC WREM-Based PWR ECCS Evaluation Model (September 26, 1975 Version)," XN-75-57, October 1975, and Revision 1, November 1975, Exxon Nuclear Company, Richland, WA.

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

3. "LOCA Analyses for H.B. Robinson Unit No. 2 Using WREM-Based PWR ECCS Evaluation Model with Reduced LPSI Flow, Steam Generator Plugging and Increased Upper Head Temperature," XN-76-54, Exxon Nuclear Company, Richland, WA, December 1976.

4. "ECCS and PTS Analyses for H.B. Robinson Unit No. 2 Reactor with 5%, 10% and 15% Steam Generator Tube Plugging," XN-NF-80-43, Exxon Nuclear Company, Richland, WA, August 1981.

6. "ECCS and Plant Transient Analyses for H.B. Robinson Unit 2 Reactor Operating at Reduced Primary Temperature," XN-NF-82-18, Exxon Nuclear Company, Richland, WA, March 1982.

7. "ECCS and Plant Transient Analyses for H.B. Robinson Unit 2 Reactor Operating at Reduced Primary Temperature; Supplement 1: Analysis of the Seized Primary Coolant Pump Accident for Asymmetric Steam Generator Tube Plugging," Exxon Nuclear Company, Richland, WA, March 1983.

8. Letter, G.F. Owsley (ENC) to D.F. Ross (NRC), "Description of RELAP4-EM ENC28B," dated October 30, 1978.

9. Letter, Thomas A. Ippolito (NRC) to Warren S. Nechodom (ENC), "ENC-EM Update Evaluation," March 1979.

10. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Updated WREM-IIA," XN-NF-78-30(A), Exxon Nuclear Company, Richland, WA, May 1979.

11. Letter, Thomas A. Ippolito (NRC) to W.S. Nechodom (ENC), "Topical Report Evaluation," dated March 30, 1979.

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

13. "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.

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

15. "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," XN-NF-81-58(P), Revision 2, Exxon Nuclear Company, February 1983.

16. "GAPEXX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," XN-73-25, Exxon Nuclear Company, Richland, WA, August 1973.

17. "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," XN-NF-82-20(P), Supplement 2 to Rev. 1, Exxon Nuclear Company, Richland, WA, June 1984.

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

19. "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," XN-NF-77-25(A), Exxon Nuclear Company, Richland, WA, September 1978.

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

21. "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," XN-NF-82-07(P), Exxon Nuclear Company, Richland, WA, March 1982.

XN-NF-84-72

Issue Date: 7/12/84

H. B. ROBINSON UNIT 2

LIMITING BREAK LOCA-ECCS ANALYSIS

WITH INCREASED ENTHALPY RISE FACTOR

Distribution

MJ Ades

GJ Busselman

JC Chandler

RE Collingham

GC Cooke

LJ Federico

SE Jensen

WV Kayser

CE Leach

MR Killgore

GA Sofer

IZ Stone

RB Stout

T Tahvili

NRC/CP&L/TJ Helbling (82)

Document Control (5)

SINGLE FAILURE ANALYSIS

TABLE 1: SUMMARY OF THE RESULTS

	Transients	Worst Single Failure		Transients Not Analyzed	Comment
			15.1.1	Feedwater Malfunctions that result in a decrease in feedwater temperature	This increase in heat removal by the secondary is not severe enough to drop Reactor Coolant System pressure to the SI setpoint. Reactor trip and turbine trip prevent drastic cooldown of reactor coolant system.
			15.1.2	Feedwater System Malfunctions that result in an increase in feedwater flow	This increase is heat removal by the secondary is not severe enough to drop Reactor Coolant System pressure to the SI setpoint. Reactor trip and turbine trip prevent drastic cooldown of the reactor coolant system.
15.1.3	Excessive Increase in Secondary Steam Flow	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			
			15.1.4	Inadvertent Opening of SG Relief or PORV	Bounded by Excessive Increase in Secondary Steam Flow (15.1.3) and hand calculations.
15.1.5	Main Steamline Break with Loss of Offsite Power	For fuel thermal limits: failure of one of two diesel generators to start For containment integrity for MSLB inside containment: failure of one of two diesel generators to start			
	Main Steamline Break with Offiste Power available	For fuel thermal limits: failure of one of three SI pumps For offsite dose for MSLB outside containment: continued normal feedwater injection at reduced flow			

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure		Transients Not Analyzed	Comment
15.2.1	Loss of External Electric Load	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.2.2	Turbine Trip	Bounded by Loss of Load (15.2.1)
			15.2.3	Loss of Condenser Vacuum and other events resulting in Turbine Trip	Bounded by Loss of Load (15.2.1)
			15.2.4	Inadvertent Closure of MSIV's	Bounded by Loss of Load (15.2.1)
			15.2.6	Loss of Non-Emergency AC Power to Station Auxiliaries	Bounded by Complete Loss of Flow (15.3.1) and Loss of Normal Feedwater Flow (15.2.7)
15.2.7	Loss of Normal Feedwater Flow	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.2.8	Feedwater System Pipe Break	Bounded by Steamline Break (15.1.5)
15.3.1	Loss of Forced Primary Coolant Flow	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
15.3.3	Reactor Coolant Pump Shaft Seizure	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.3.4	Reactor Coolant Pump Broken Shaft	Bounded by Shaft Seizure (15.3.3)
15.4.1	Uncontrolled RCCA Withdrawal from Subcritical or Low Power Startup Condition	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			
15.4.2	Uncontrolled RCCA Withdrawal from Power	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure		Transients Not Analyzed	Comment
15.4.3	RCCA Misoperation	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			
			15.4.4	Startup of an Inactive Coolant Loop at Incorrect Temperature	Power operation with less than three loops is not allowed
15.4.6	CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant				The operational modes of refueling and startup are analyzed to show that adequate time exists to secure inadvertent boron dilution before criticality occurs. No ESF systems are involved.
			15.4.7	Inadvertent Loading of a Fuel Assembly into an Improper Location	Administrative procedures preclude occurrence of this event.
15.4.8	Spectrum of RCCA Ejection Accidents	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.5.1	Inadvertent Operation of ECCS	Shutoff head of high pressure SI pumps is 1500 psia < 1750 psia trip setpoint pressure.
			15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Effect on Reactor Coolant System Pressure is completely mitigated by the Reactor Protection System and relief valves.
15.6.1	Inadvertent Opening of Pressurizer Safety or PORV	Failure of one of three SI pumps to deliver flow			
			15.6.2	Loss of Reactor Coolant from Rupture of Small Pipes or from Cracks in Large Pipes which actuate the ECCS	Bounded by large break LOCA (15.6.5)

(3610NH/pgp)

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure	Transients Not Analyzed	Comment
15.6.3	Steam Generator Tube Rupture	Steam driven Auxiliary Feedwater Pump fails to deliver flow		
15.6.5	LOCA	Failure of one diesel generator to start		

(3610NH/pgp)

ATTACHMENT 12

XN-NF-84-68 (P)

H. B. ROBINSON UNIT 2
RADIOLOGICAL ASSESSMENT
OF POSTULATED ACCIDENTS

A F F I D A V I T

STATE OF Washington)
ss.
COUNTY OF Benton)

I, Richard B. Stout, being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Exxon Nuclear Company, Inc. ("ENC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with ENC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the document XN-NF-84-68(P) entitled "H.B. Robinson Unit 2 Radiological Assessment of Postulated Accidents" referred to as "Document". Information contained in this Document has been classified by ENC as proprietary in accordance with the control system and policies established by ENC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by ENC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to Carolina Power and Light Company and the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of ENC and would be helpful to competitors of ENC when competing with ENC.

7. The information contained in the Document is considered to be proprietary by ENC because it reveals certain distinguishing aspects of radiological assessment procedures which secure competitive advantage to ENC for fuel design optimization and improved marketability, and includes information utilized by ENC in its business which affords ENC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into radiological assessent procedures and would result in substantial harm to the competitive position of ENC.

9. The Document contains proprietary information which is held in confidence by ENC and is not available in public sources.

10. In accordance with ENC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside ENC only as required and under suitable agreement providing for non-disclosure and limited use of the information.

11. ENC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. This Document provides information which reveals the radiological assessment procedures developed by ENC over the past several years. ENC has invested thousands of dollars and many man-years of effort in developing the BWR thermal hydraulic analysis methods revealed in the Document. Assuming a competitor had available the same background data and incentives as ENC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as ENC.

13. Based on my experience in the industry, I do not believe that the background data and incentives of ENC's competitors are sufficiently similar to the corresponding background data and incentives of ENC to reasonably expect such competitors would be in a position to duplicate ENC's proprietary information contined in the Documents.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

Richard B. Stout

SWORN TO AND SUBSCRIBED
before me this 10 day of
July, 1984.

Susan E. Backus
NOTARY PUBLIC