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CLASS I
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APPENDIX A

LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT FOR DUANE ARNOLD ENERGY CENTER (LEAD PLANT)

GENERAL ELECTRIC

NEDO-21082-03
Class I
June 1984
Appendix A

LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT

FOR

DUANE ARNOLD ENERGY CENTER

(LEAD PLANT)

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

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LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS (Continued)

A.1 INTRODUCTION

The purpose of this document is to provide the results of the loss-of-coolant accident (LOCA) analysis for the Duane Arnold Energy Center (DAEC). The analysis was performed using approved General Electric (GE) calculational models.

This analysis of the plant LOCA is provided in accordance with the NRC requirement (Reference 1) and to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46. The objective of the LOCA analysis contained herein is to provide assurance that the most limiting break size, break location, and single failure combination has been considered for the plant. The required documentation for demonstrating that these objectives have been satisfied are given in Reference 2. The documentation contained in this report is intended to satisfy these requirements.

The general description of the LOCA evaluation models is contained in Reference 3, with a detailed description provided in Reference 4.

Plants are separated into groups for the purpose of LOCA analysis (Reference 5). Within each plant group there will be a single lead plant analysis which provides the basis for the selection of the most limiting break size yielding the highest peak cladding temperature (PCT). Also, the lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. The remainder of the plants in that group will have non-lead plant analyses referenced to the lead plant analysis. This document contains the lead plant analysis for DAEC which is a BWR 4 with loop selection logic (plants that have not incorporated the low pressure coolant injection (LPCI) system modification) group plant and is consistent with the requirements outlined in Reference 2.

The same models and computer codes are used to evaluate all plants. Changes to these models will cause changes in phenomenological responses that are similar within any given plant group. Differences in input parameters are not expected to result in significantly different results for the plants within a given group. Emergency core cooling system (ECCS) and geometric differences between plant groups may result in different responses for different groups but within any group the responses will be similar. Thus, the lead plant concept is still valid for this evaluation.

A.2 LEAD PLANT SELECTION

Lead plants are selected and analyzed in detail to permit a comprehensive review and eliminate unnecessary calculations. This constitutes a generic analysis for each plant of that type which can be referenced in subsequent plant submittals.

Based on the criteria given in Reference 5, all operating General Electric Boiling Water Reactors (BWR) have been divided into groups. A lead plant was selected for each group whose LOCA response would be representative of the entire group. The three main criteria for selecting the lead plants are:

(1) typical blowdown and reflood characteristics;
(2) typical reactor thermal power; and
(3) number of plants.

The first of these is important because it establishes that the shape of the break spectrum will be typical of plants in the class. The second and third are important because they establish the degree to which the lead plant analysis can be considered "generic".

The duration of nucleate boiling is the best measure of blowdown heat transfer. It is determined largely by the ratio of downcomer volume to break area. Shorter periods of nucleate boiling during blowdown result in higher PCT's since less stored energy is removed.

Reflood time is the best measure of overall emergency core cooling effectiveness. It is determined primarily by the complement of ECCS equipment available, and the number of bundles with alternate flow path holes. Shorter reflood times result in lower PCT's since the fuel has less time to heat up after nucleate boiling is lost.

Based on these considerations, the first criteria is the primary reason for differentiating between the various groups of plants. The balance between heat transfer during blowdown and reflood time is of primary importance in determining how plants respond to the LOCA.

The reactor thermal power is used to determine the lead plant from a large number of plants in a group to permit a more comfortable extrapolation of the generic results to other reactor sizes where no plant is clearly identified by the first criterion.

The number of similar plants is used as a criterion for selecting lead plants since it maximizes the number of plants that the lead plant analysis is directly applicable to.

As a result of the above criteria, all currently operating jet-pump GE-BWR's have been separated into three groups for the LOCA analysis. The three groups are identified as BWR/3, BWR/4 with loop selection logic, and BWR/4 with LPCI modification. The basis for selecting DAEC as the lead plant for the BWR/4 with loop selection logic group is discussed in the following paragraph.

Since the last lead plant analysis some of the plants which were previously in this group have installed the LPCI modification. Only two domestic operating BWR/4 plants have not installed the LPCI modification. The previous lead plant for this group was FitzPatrick which has since installed the LPCI modification. Of the two remaining plants in this group, there is no technical reason to select one above the other. Table 1 shows a comparison of the important LOCA/ECCS parameters for both plants remaining in this group. Based on Criterion 1 there is no significant difference between these two plants. Since there are only two plants remaining in this group Criteria 2 and 3 are not applicable. Thus, DAEC has been selected as lead plant for this group since it was scheduled for completion of the ECCS analysis first.

A third plant, Pilgrim, has been added to this group. Although Pilgrim is classified as a BWR/3, its design incorporates finger springs on the fuel and alternate flow path holes in the fuel bundle lower tieplates, and it incorporates the same ECCS design as the BWR/4 plants with loop selection logic.

This difference between Pilgrim and the other BWR/3's results in Pilgrim having a significantly reduced reflood time. Thus, Pilgrim is closer in ECCS performance to a BWR/4 than a BWR/3 and has been included with the BWR/4's.

A.3 INPUT TO ANALYSIS

A list of the significant plant input parameters to the LOCA analysis is presented in Table 2.

A.4 LOCA ANALYSIS COMPUTER CODES

A.4.1 Results of the LAMB Analysis

This code is used to analyze the short-term blowdown phenomena for large postulated pipe breaks (breaks in which nucleate boiling is lost before the water level drops and uncovers the active fuel) in jet pump reactors. The LAMB output (core flow as a function of time) is input to the SCAT code calculation of blowdown heat transfer.

The LAMB results presented are:

- Core Average Inlet Flow Rate (normalized to unity at the beginning of the accident) following a Large Break.

A.4.2 Results of the SCAT Analysis

This code completes the transient short-term thermal-hydraulic calculation for large breaks in jet pump reactors. The GEXL correlation is used to track the boiling transition in time and location. The post-critical heat flux heat transfer correlations are built into SCAT which calculates heat transfer coefficients for input to the core heatup code, CHASTE.

The SCAT results presented are:

- Minimum Critical Power Ratio following a Large Break.
- Convective Heat Transfer Coefficient following a Large Break.

A.4.3 Results of the SAFE Analysis

This code is used primarily to track the vessel inventory and to model ECCS performance during the LOCA. The application of SAFE is identical for all break sizes. The code is used during the entire course of the postulated accident, but after ECCS initiation, SAFE is used only to calculate reactor system pressure and ECCS flows, which are pressure dependent.

The SAFE results presented are:

- Water Level inside the Shroud (up to the time REFLOOD initiates) and Reactor Vessel Pressure

A.4.4 Results of REFLOOD Analysis

This code is used across the break spectrum to calculate the system inventories after ECCS actuation. The models used for the design basis accident (DBA) application ("DBA-REFLOOD") was described in a supplement to the SAFE code description transmitted to the USNRC December 20, 1974. The "non-DBA REFLOOD" analysis is nearly identical to the DBA version and employs the same major assumptions. The only differences stem from the fact that the core may be partially covered with coolant at the time of ECCS initiation and coolant levels change slowly for smaller breaks by comparison with the DBA. More precise modeling of coolant level behavior is thus required principally to determine the contribution of vaporization in the fuel assemblies to the counter current flow limiting (CCFL) phenomenon at the upper tieplate. The differences from the DBA-REFLOOD analysis are:

(1) The non-DBA version calculates core water level more precisely than the DBA version in which great precision is not necessary.

(2) The non-DBA version includes a heatup model similar to but less detailed than that in CHASTE, designed to calculate cladding temperature during the small break. This heatup model is used in calculating vaporization for the CCFL correlation, in calculating swollen level in the core, and in calculating the peak cladding temperature.

The REFLOOD results presented are:

- Water Level inside the Shroud
- Peak Cladding Temperature and Heat Transfer Coefficient for breaks calculated with small break models

A.4.5 Result of the CHASTE Analysis

This code is used, with suitable inputs from the other codes, to calculate the fuel cladding heatup rate, peak cladding temperature, peak local cladding oxidation, and core-wide metal-water reaction for large breaks. The detailed fuel model in CHASTE considers transient gap conductance, clad swelling and rupture, and metal-water reaction. The empirical core spray heat transfer and channel wetting correlations are built into CHASTE, which solves the transient heat transfer equations for the entire LOCA transient at a single axial plane in a single fuel assembly. Iterative applications of CHASTE determine the maximum permissible planar power where required to satisfy the requirements of 10CFR50.46 acceptance criteria.

The CHASTE results presented are:

- Peak Cladding Temperature versus time
- Peak Cladding Temperature versus Break Area
- Peak Cladding Temperature and Peak Local Oxidation versus Planar Average Exposure for the most limiting break size
- Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size

A summary of the analytical results is given in Table 3. Table 4 lists the figures provided for this analysis. The MAPLHGR values for each fuel type presently in the DAEC core are presented in Tables 5a through 5d.

A.4.6 Methods

In the following sections, it will be useful to refer to the methods used to analyze DBA, large breaks, and small breaks. For jet-pump reactors, these are defined as follows.

a. DBA Methods. LAMB/SCAT/SAFE/DBA-REFLOOD/CHASTE.
   Break size: DBA.

b. Large Break Methods (LBM). LAMB/SCAT/SAFE/ non-DBA REFLOOD/CHASTE,
   Break sizes: $1.0\ ft.^2 \leq A < DBA$.

c. Small Break Methods (SBM). SAFE/non-DBA REFLOOD,
   Heat transfer coefficients: nucleate boiling prior to core uncovery, 25 $Btu/hr\text{-}ft^2\text{-}^oF$ after recovery, core spray when appropriate. Peak cladding temperature and peak local oxidation are calculated in non-DBA-REFLOOD.
   Break sizes: $A \leq 1.0\ ft^2$.

A.5 BREAK SPECTRUM CALCULATIONS

For convenience in describing the LOCA phenomena, the break spectrum has been separated into three regions: small breaks, intermediate breaks and large breaks. The selection of the break sizes to be included in each region is dependent on the most limiting single failure and the ECCS evaluation method used. The potentially limiting single active failures considered in establishing the various break regions are given in Table 6.

The small break region is defined as that portion of the break spectrum where the high pressure coolant injection (HPCI) is the most limiting single failure. In this region, the "small break methods" are used.

The intermediate break region is defined as that portion of the break spectrum up to the transition break where the LPCI injection valve is the most limiting single failure. The transition break is defined as the 1.0 ft^2 break size. This break size has been chosen in order to be consistent with previous analyses thus allowing for a better comparison between the old and new analyses. The calculational techniques employed in the SBM are intended to conservatively model small breaks only. As the break size increases ($\gtrsim$ 1.0 ft^2) the SBM becomes overly conservative and does not appropriately describe some of the phenomena (e.g., radiation heat transfer, blowdown heat transfer). The transition break has been analyzed with both the large and small break methods with the same single failure to allow a comparison between the methods. The analysis of the transition break is shown in Figures 1b through 5b for the large break method and Figures 1c and 2c for the small break methods. In the intermediate break region, small break methods are used.

The large break region is defined as that portion of the break spectrum between the transition break and the DBA. The DBA is defined as the complete severance of the largest pipe in that portion of the system which yields the highest peak cladding temperature when the most limiting single failure is assumed. The most limiting single failure in this region is the failure of the LPCI injection valve. In the large break region, large break methods are used. For the DBA, the DBA methods are used.

A.5.1 Large Break Analysis

In this region, the vessel depressurizes rapidly and the HPCI has an insignificant effect on the event. Consequently, failure of the core spray or LPCI is more severe. Analyses have demonstrated that failure of the LPCI is the most severe failure among the low pressure ECCS because unlike the core spray which must pass through the CCFL regions at the top of the core, LPCI is injected into the lower plenum through the jet pumps. Thus, the LPCI injection valve is the worst single failure in the large break region.

The characteristics that determine which is the most limiting break are:

(a) the calculated hot node reflooding time,

(b) the calculated hot node uncovery time, and

(c) the time of calculated boiling transition.

The time of calculated boiling transition increases with decreasing break size, since jet pump suction uncovery (which leads to boiling transition) is determined primarily by the break size for a particular plant. The calculated hot node uncovery time also generally increases with decreasing break size, as it is primarily determined by the inventory loss during the blowdown. The hot node reflooding time is determined by a number of interacting phenomena such as depressurization rate, counter current flow limiting and a combination of available ECCS.

The period between hot node uncovery and reflooding is the period when the hot node has the lowest heat transfer. Hence, the break that results in the longest period during which the hot node remains uncovered results in the highest calculated PCT. If two breaks have similar times during which the hot node remains uncovered, then the larger of the two breaks will be limiting as it would have an earlier boiling transition time (i.e., the larger break would have a more severe LAMB/SCAT blowdown heat transfer analysis).

Figure 8 shows the variation with break size of the calculated time the hot node remains uncovered for DAEC. Based on these calculations, the DBA was determined to be the break that results in the highest calculated PCT in the 1.0 ft^2 to DBA region. Confirmation that this is the most limiting break over the entire break spectrum is shown in Figure 7.

The DBA results are presented in Figures 1a through 5a, 6, and 8.

A.5.2 Small Break Analysis

In this region the vessel depressurizes relatively slowly (or not at all, depending on the break size) because the break is small. HPCI is the most severe equipment failure in this region because its loss results in a loss of ECCS delivery capability at high pressure. With HPCI available, the core remains covered for longer periods of time than for cases with a single ADS valve failed or for cases with low-pressure core spray or LPCI failures.

For the BWR/4 plants with loop selection logic, the remaining ECCS assuming an HPCI failure are 2 low pressure core spray and 4 LPCI pumps. With all of the LPCI flow directly available to the lower plenum, the reflooding time is rather insensitive to the small changes in CCFL (which affect the delivery of core spray to the lower plenum) that result from the model and input changes. Therefore, the uncovered time for the fuel in this region of the break spectrum is only slightly altered when new inputs and models are applied. The change in PCT from the previous ECCS analysis (Reference 6) is due, primarily, to a different heatup rate during the period of fuel uncovery. Any change in MAPLHGR calculated from the limiting break is fed back into the small break heatup calculation by a change in the power of the hot rod. The change in PCT due to this power change is approximately proportional to the change in MAPLHGR.

For DAEC, the limiting break size (0.07 ft^2) has been identified as a result of the several sizes considered as shown on Figure 7. The results of the 0.07 ft^2 analysis are shown on Figures 1e and 2e.

For all BWR/4's with loop selection logic, the small break temperatures are on the order of 1500°F (specific results for DAEC are shown on Figure 7) and since MAPLHGR will seldom increase by more than about 15%, the maximum expected increase in small break temperature is about 15% of (1500 - 500°F), or about 150°F.

This trend will be experienced by all plants in this group because:

(1) For small breaks with LPCI flow, the core spray flow into the lower plenum through the alternate flow path holes contributes

a very small amount to the total reflooding flow. Thus, the effect of the alternate flow path is overshadowed by the LPCI flow.

(2) The small break has characteristically a linear heatup. Thus, the effect of a change in reflooding time can be accurately predicted from a previous analysis.

(3) The change in the slope of the heatup curve is directly proportional to the change in power (i.e., MAPLHGR).

A.5.3 Intermediate Break Analysis

In this region, the vessel depressurizes rather rapidly through the break and the high-pressure delivery capability of HPCI is less significant than it is for smaller breaks. Consequently, failure of low pressure core spray or LPCI is more severe. Analyses have demonstrated that the LPCI is the most severe failure among the low pressure ECCS because, unlike core spray which must pass through the CCFL regions at the top of the core, LPCI is injected into the lower plenum through the jet pumps. Thus, the LPCI injection valve failure, which results in no LPCI being available is the worst single failure in the intermediate break region.

For DAEC, the limiting break size (0.8 ft^2) in this region has been identified as a result of the several sizes considered as shown on Figure 7. The results of the 0.8 ft^2 analysis are shown on Figures 1d and 2d.

Throughout the entire intermediate break spectrum ($\sim$0.3 ft^2 to 1.0 ft^2) there is a similar reduction in the PCT as evidenced by the 1.0 ft^2 analysis (Figure 7).

This same trend will be experienced by all plants covered by this lead plant analysis.

A.6 CONCLUSIONS

The results of the analysis demonstrate that the ECCS will perform its function in an acceptable manner and that the ECCS acceptance criteria of 10CFR50.46 are met.

REFERENCES

1. Letter, George Lear (NRC) to Duane Arnold (IEL&P), "Re: Duane Arnold Energy Center," dated March 11, 1977.

2. Letter, Darrell G. Eisenhut (NRC) to E.D. Fuller (GE), "Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants," June 30, 1977.

3. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDO-20566 (Draft), submitted August 1974, and General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter G.L. Gyorey (GE) to Victor Stello, Jr. (NRC), dated Decmber 20, 1974.

4. General Electric Standard Application for Reactor Fuel (U.S. Supplement), NEDE-24011-P-A-6-US, April 1983.

5. Letter, G.L. Gyorey (GE) to V. Stello, Jr., dated May 12, 1975, "Compliance with Acceptance Criteria of 10CFR50.46."

6. Duane Arnold Energy Center Loss-of-Coolant Accident Analyses Conformance with 10CFR50 Appendix K (Jet Pump Plant), June 1975.

Table 1

BWR/4 WITH LOOP SELECTION LOGIC IMPORTANT LOCA/ECCS PARAMETERS

Parameter	DAEC	Pilgrim	Browns Ferry 3
Power, MWt	1691	1998	3293
Vessel Id, in.	183	224	251
Recirc Line ID, in.	22	28	28
Number of Fuel Bundles	368	580	764
Number of Drilled Fuel Bundles	368	428	764
Fuel Design (Operating Reactors)	8x8	8x8	8x8
ECCS Available, DBA	2CS	2CS	2CS
ECCS Available, Small Break	2CS 4LPCI ADS	2CS 4LPCI ADS	2CS 4LPCI ADS

Table 2

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	1691 MWt, which corresponds to 102% of rated steam flow
Vessel Steam Output	7.344×10^{6} lbm/h, which corresponds to 102% of rated steam flow
Vessel Steam Dome Pressure	**1055 psia**
Recirculation Line Break Area for Large Breaks	**(DBA) 2.51 ft^2, 1.0 ft^2**
Recirculation Line Break Area for Small Breaks	**1.0 ft^2, 0.8 ft^2,** 0.07 **ft^2**
Number of Drilled Bundles	368

FUEL PARAMETERS:

Fuel Type	**Fuel Bundle Geometry**	**Peak Technical Specification Linear Heat Generation Rate (kW/ft)**	**Design Axial Peaking Factor**	**Initial Minimum Critical Power Ratio**
A. P8DPB289/ BP8DRB289	8x8	13.4	1.4	1.2
B. P8DRB299/ BP8DRB299	8x8	13.4	1.4	1.2
C. P8DRB284H/ BP8DRB284H	8x8	13.4	1.4	1.2
D. P8DRB301L/ BP8DRB301L	8x8	13.4	1.4	1.2

Table 3

SUMMARY OF BREAK SPECTRUM RESULTS

o Break Size o Location o Single Failure	PCT (°F)	Peak Local Oxidation %	Core-Wide Metal-Water Reaction %
o $2.51\ ft^2$ (DBA) o Recirc suction o LPCI Injection Valve	1959	1.1	0.08
o $1.0\ ft^2$ (LBM) o Recirc Suction o LPCI Injection Valve	1765	Note 1	Note 2
o $1.0\ ft^2$ (SBM) o Recirc suction o LPCI Injection Valve	1358	Note 1	Note 2
o $0.8\ ft^2$ o Recirc Suction o LPCI Injection Valve	1142	Note 1	Note 2
o $0.07\ ft^2$ o Recirc Suction o HPCI	1566	Note 1	Note 2

NOTES:

1. Less than DBA (1.1%)
2. Less than DBA (0.08%)

Table 4

LOCA ANALYSIS FIGURE SUMMARY - LEAD PLANT

	Large Break Method		Small Break Method		
	Limiting Break - DBA - (LPCI Inj. Valve Failure) (2.51 ft^2)	Transition Break (LPCI Inj. Valve Failure) (1.0 ft^2)	Transition Break (LPCI Inj. Valve Failure) (1.0 ft^2)	Limiting Break (LPCI Inj. Valve Failure) (0.8 ft^2)	(HPCI Failure) (0.07 ft^2)
Water Level Inside Shroud and Reactor Vessel Pressure	1a	1b	1c	1d	1e
Peak Cladding Temperature	2a	2b	2c	2d	2e
Heat Transfer Coefficient	3a	3b	2c	2d	2e
Core Average Inlet Flow	4a	4b			
Minimum Critical Power Ratio	5a	5b			
Peak Cladding Temperature of the Highest Powered Plane Experiencing Boiling Transition	2a				
Normalized Power	6				
Peak Cladding Temperature vs. Break Area	7				
Hot Node Uncovered Time vs. Break Area	8				

Table 5a

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: DAEC

BUNDLE TYPE: P8DRB301L/BP8DRB301L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.5	1947	0.011
1000	11.5	1936	0.011
5000	11.9	1927	0.011
10000	12.3	1935	0.011
15000	12.4	1959	0.011
20000	12.2	1941	0.011
25000	11.3	1854	0.008
35000	9.9	1679	0.004
45000	8.7	1559	0.002

Table 5b

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: DAEC

BUNDLE TYPE: P8DPB289/BP8DRB289

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1915	0.010
1000	11.2	1906	0.010
5000	11.8	1923	0.010
10000	12.0	1914	0.009
15000	12.1	1926	0.010
20000	11.9	1921	0.010
25000	11.4	1863	0.008
30000	10.8	1786	0.006
35000	10.3	1712	0.005
40000	9.6	1646	0.004
45000	8.9	1580	0.003

Table 5c

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: DAEC

BUNDLE TYPE: P8DRB284H/BP8DRB284H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1912	0.010
1000	11.2	1900	0.010
5000	11.7	1909	0.010
10000	12.0	1921	0.010
15000	12.0	1926	0.010
20000	11.8	1918	0.010
25000	11.1	1837	0.007
30000	10.4	1744	0.005
35000	9.8	1674	0.004
40000	9.1	1608	0.003
45000	8.5	1541	0.002

Table 5d

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: DAEC

BUNDLE TYPE: P8DRB299/BP8DRB299

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.9	1874	0.009
1000	11.0	1869	0.009
5000	11.5	1874	0.009
10000	12.2	1929	0.010
15000	12.3	1950	0.011
20000	12.1	1948	0.011
25000	11.5	1890	0.009
30000	11.0	1807	0.007
35000	10.3	1725	0.005
40000	9.7	1661	0.004
45000	9.0	1601	0.003

Table 6
SINGLE-FAILURE EVALUATION

The following table shows the single, active failures considered in the ECCS performance evaluation.

Assumed Failure	Suction Break Systems Remaining
LPCI Injection Valve	ADS, 2 CS, HPCI
Diesel Generator (D/G)	ADS, 1 CS, HPCI, 2 LPCI
HPCI	ADS, 2 CS, 4 LPCI
One ADS Valve	ADS minus one, 2 CS, HPCI, 4 LPCI

Other postulated failures are not specially considered because they all result in at least as much ECCS capacity as one of the above assumed failures.

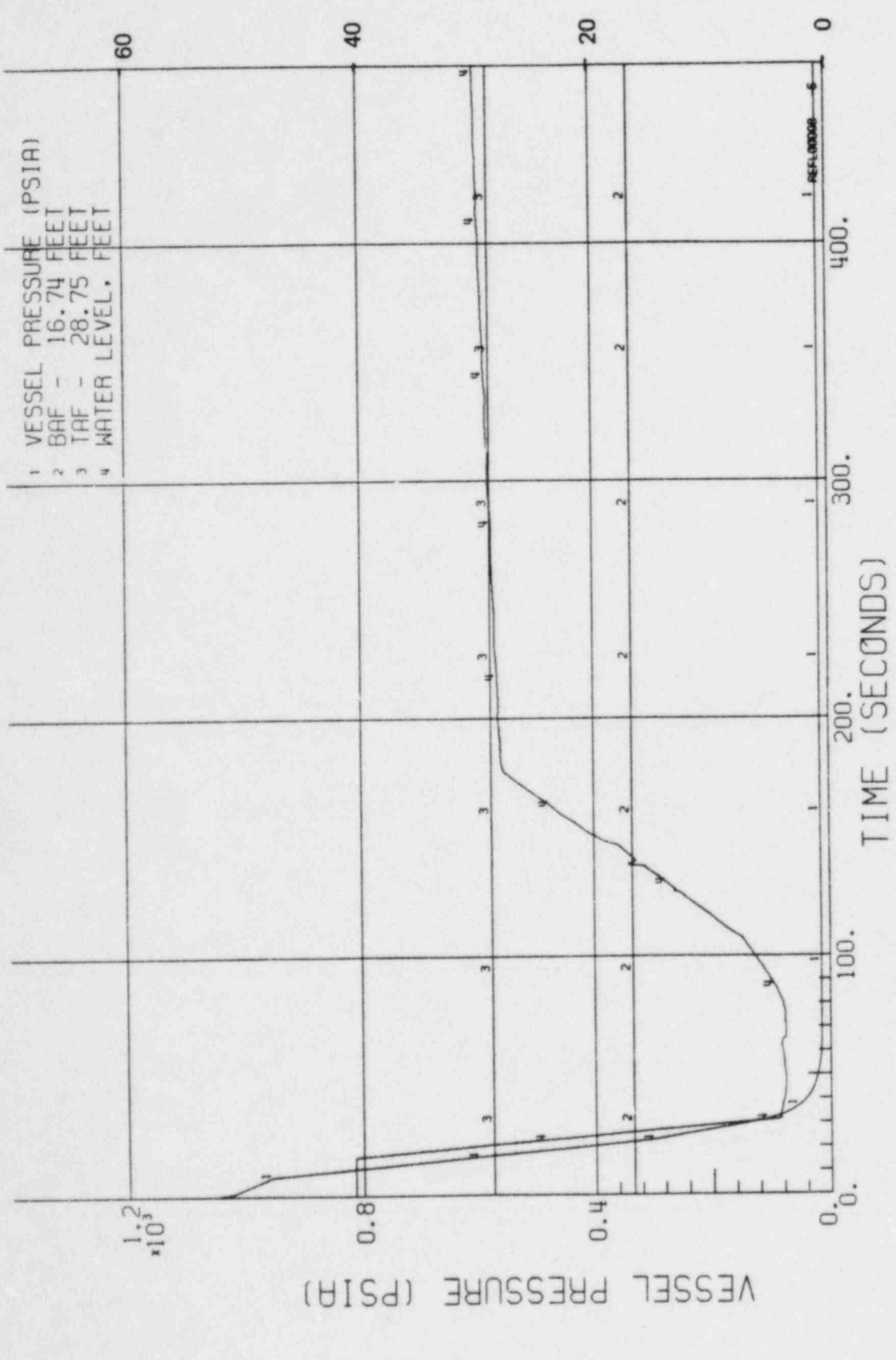

Figure 1a. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Design Basis Accident, LPCI Injection Valve Failure

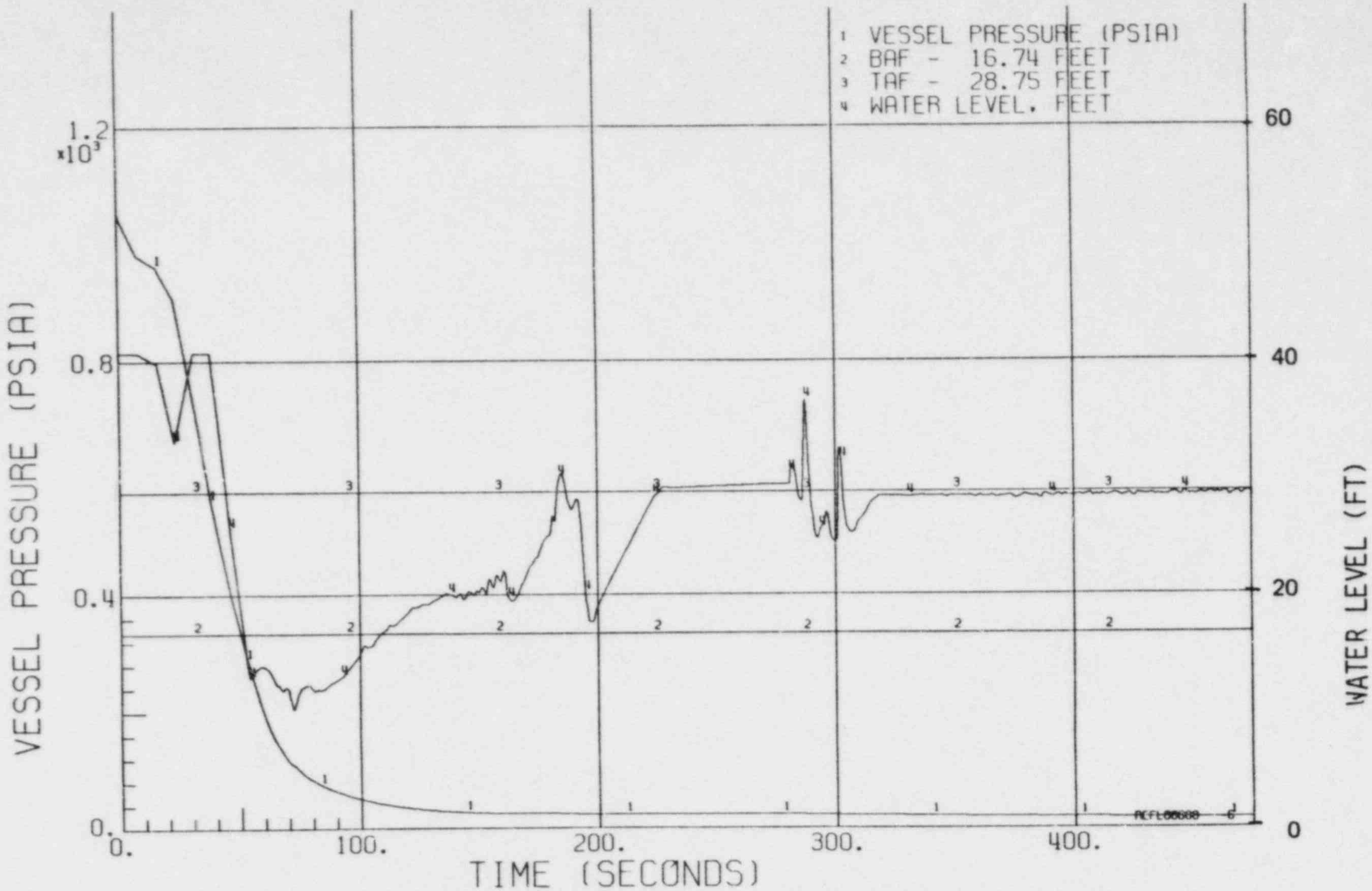

Figure 1b. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Break of the Recirculation Line, LPCI Injection Valve Failure, Break Area = 1.0 ft^2 (LBM)

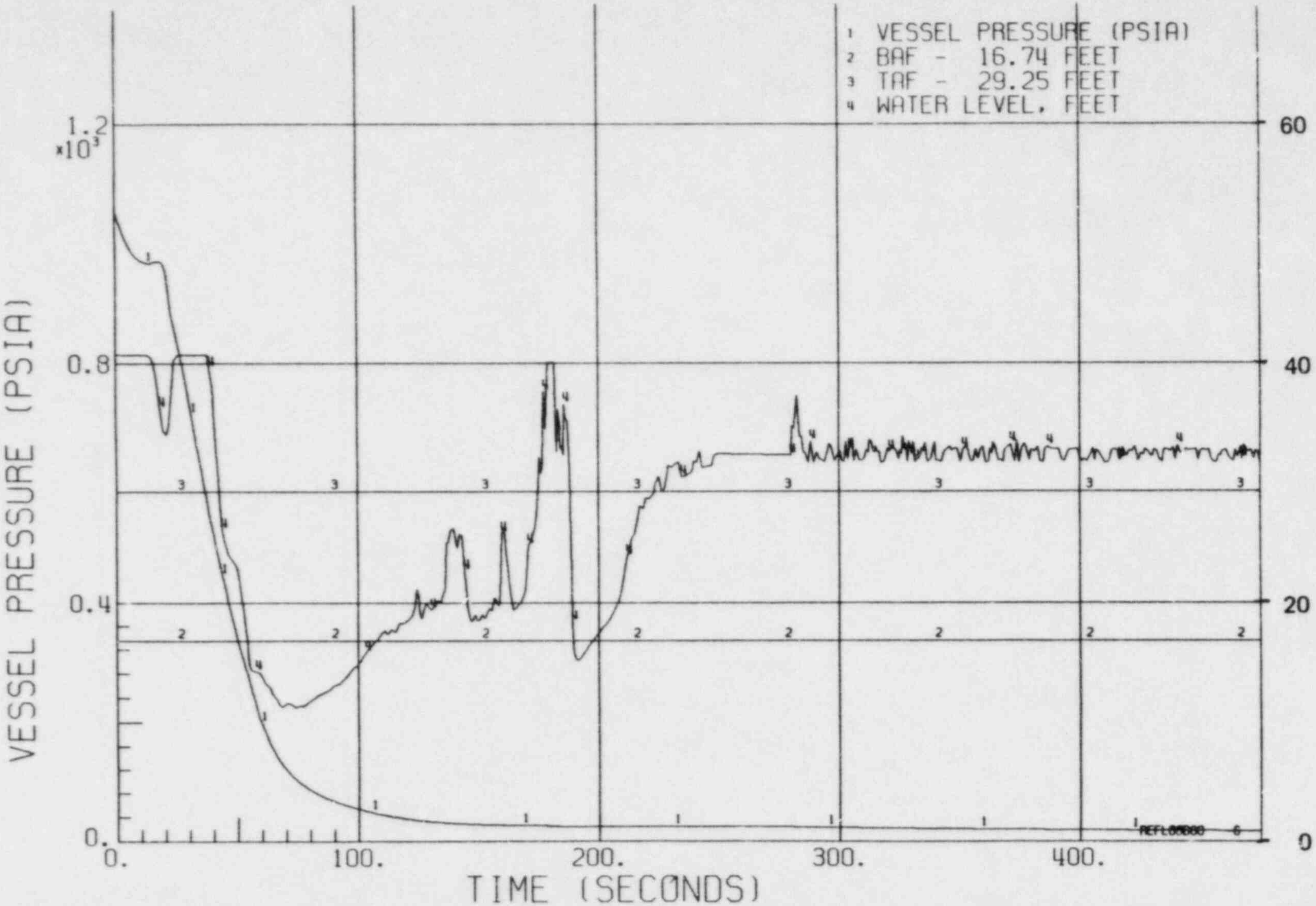

Figure 1c. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Small Break of the Recirculation Line, LPCI Injection Valve Failure, Break Area = 1.0 ft^2 (SBM)

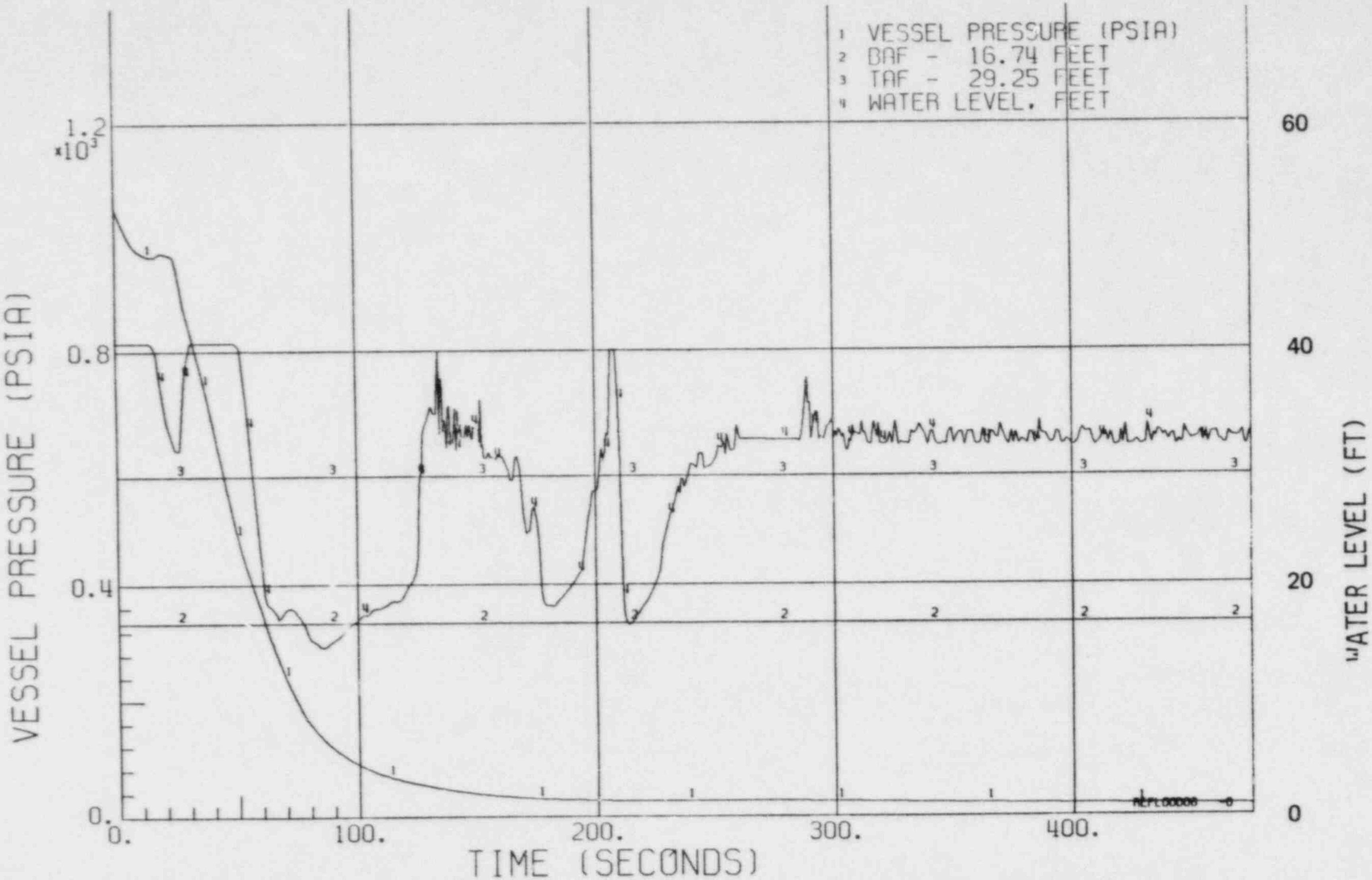

Figure 1d. Water Level Inside the Shroud and Reactor Vessel Pressure Following Small Break of the Recirculation Line, LPCI Injection Valve Failure, Break Area = 0.8 ft^2 (SBM)

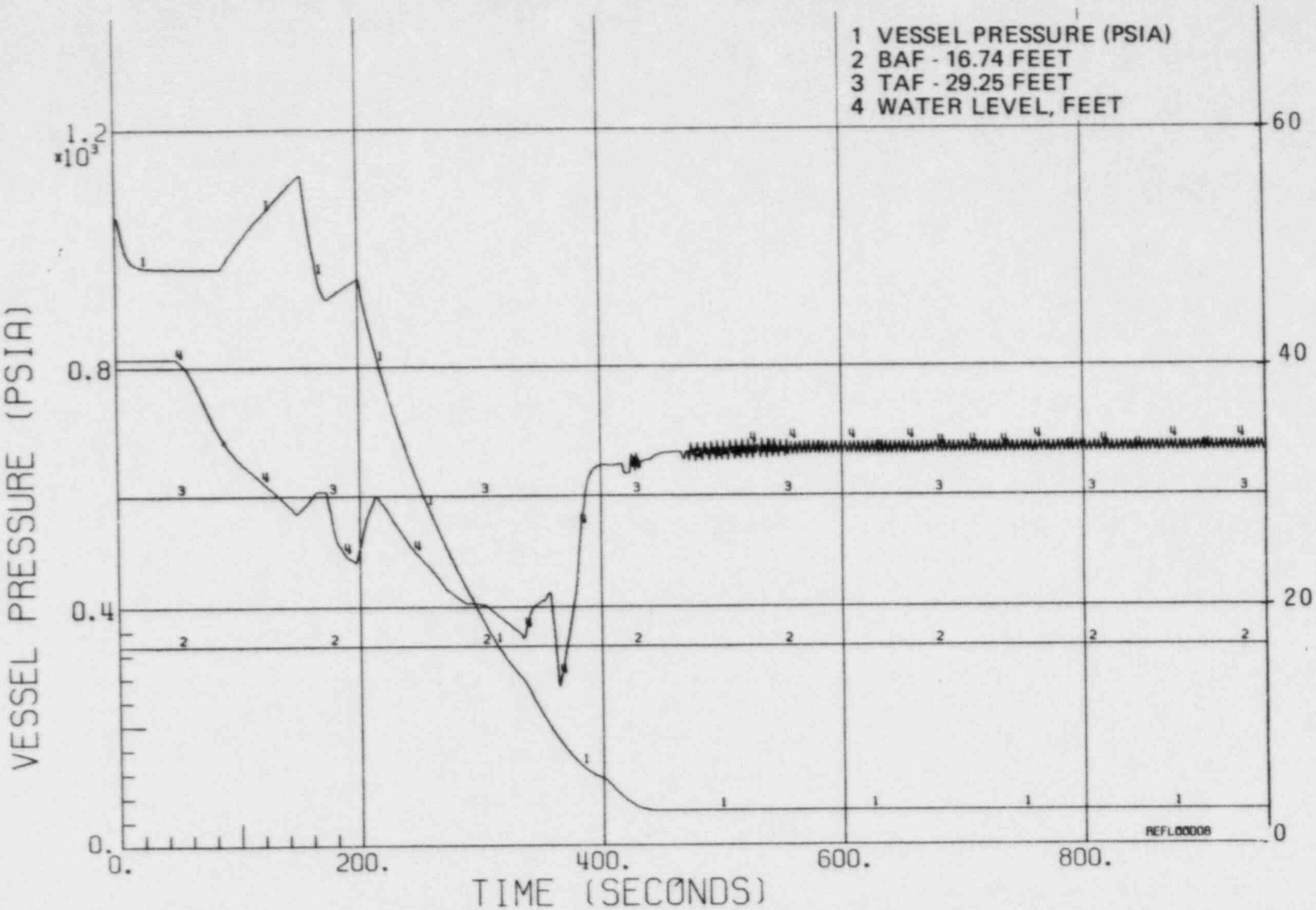

Figure 1e. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Small Break of the Recirculation Line, HPCI Failure, Break Area = 0.07 ft^2 (SBM)

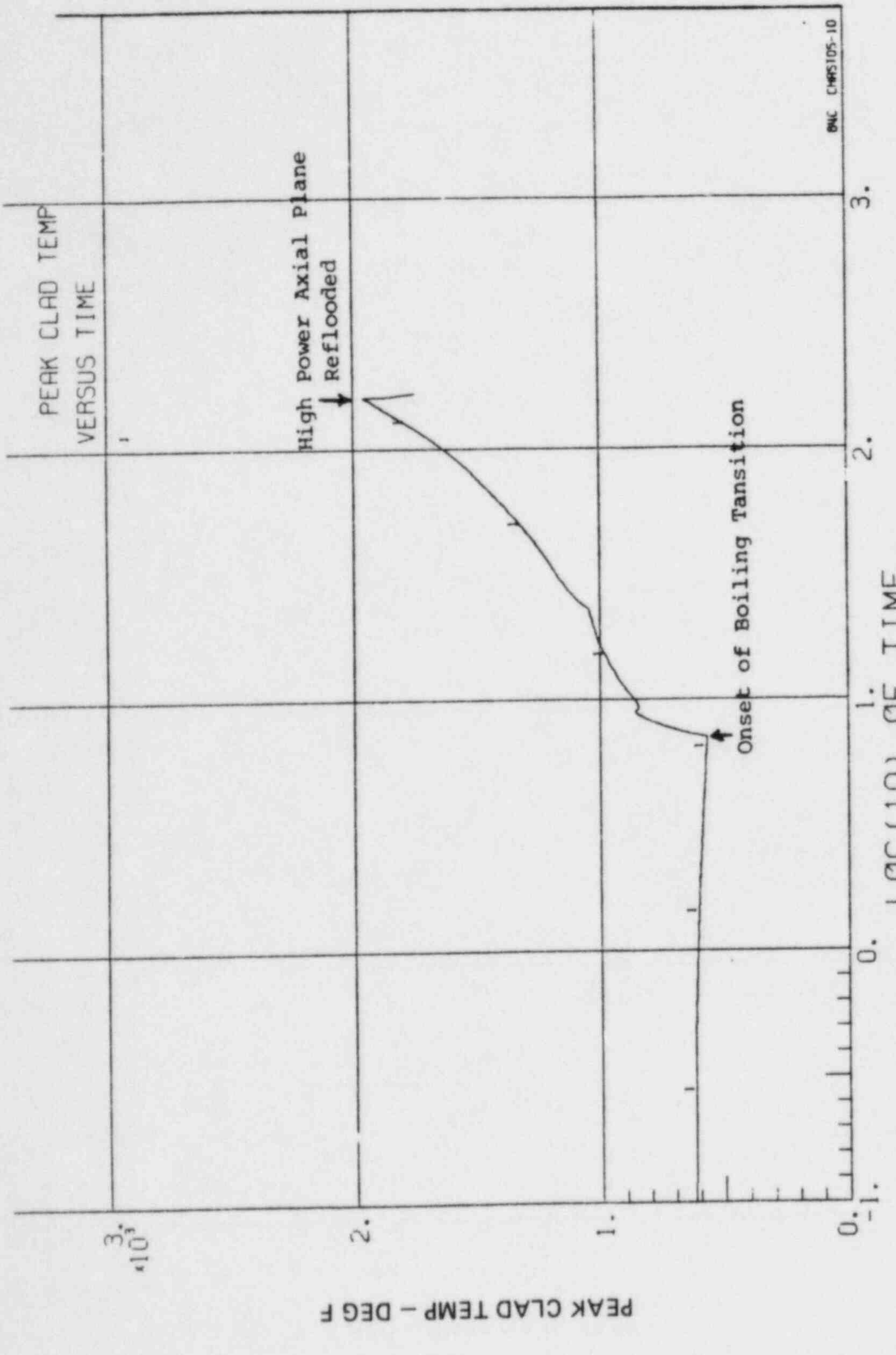

Figure 2a. Peak Cladding Temperature Following a Design Basis Accident, LPCI Injection Valve Failure

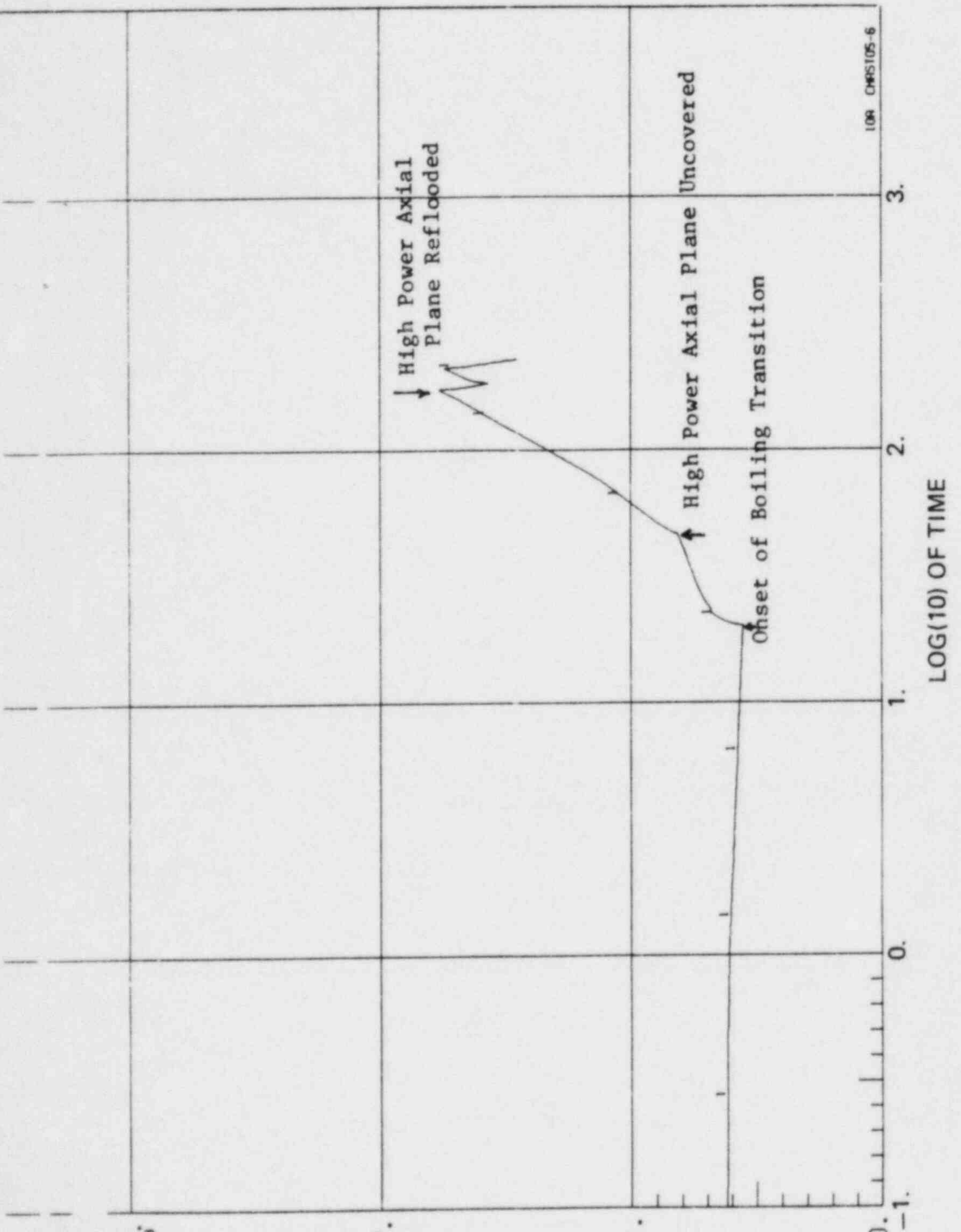

Figure 2b. Peak Cladding Temperature Following a 1 ft^2 Break, LPCI Injection Valve Failure (LBM)

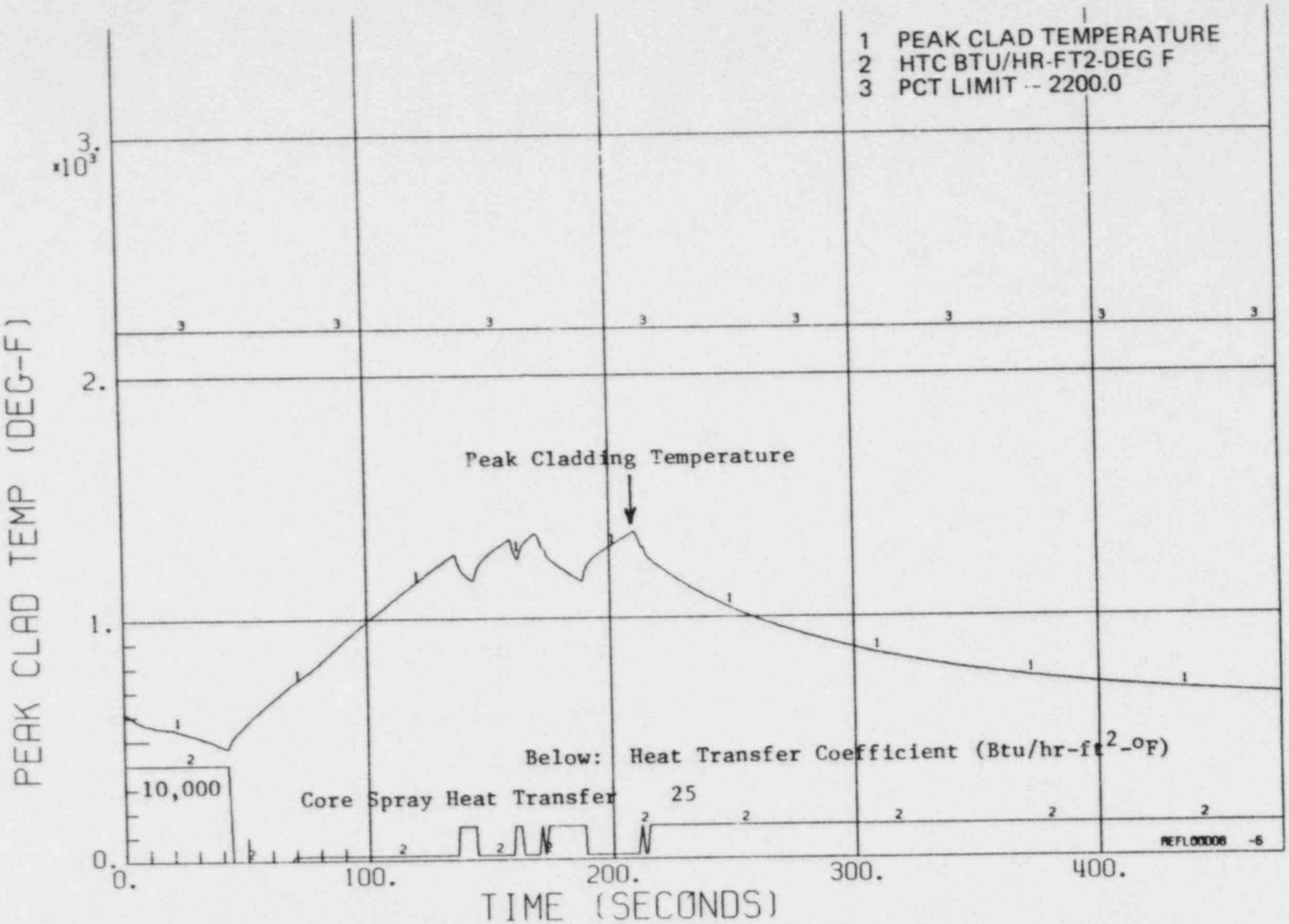

Figure 2c. Peak Cladding Temperature Following a Small Break of the Recirculation Line, LPCI Injection Valve Failure, Break Area = 1.0 ft^2 (SBM)

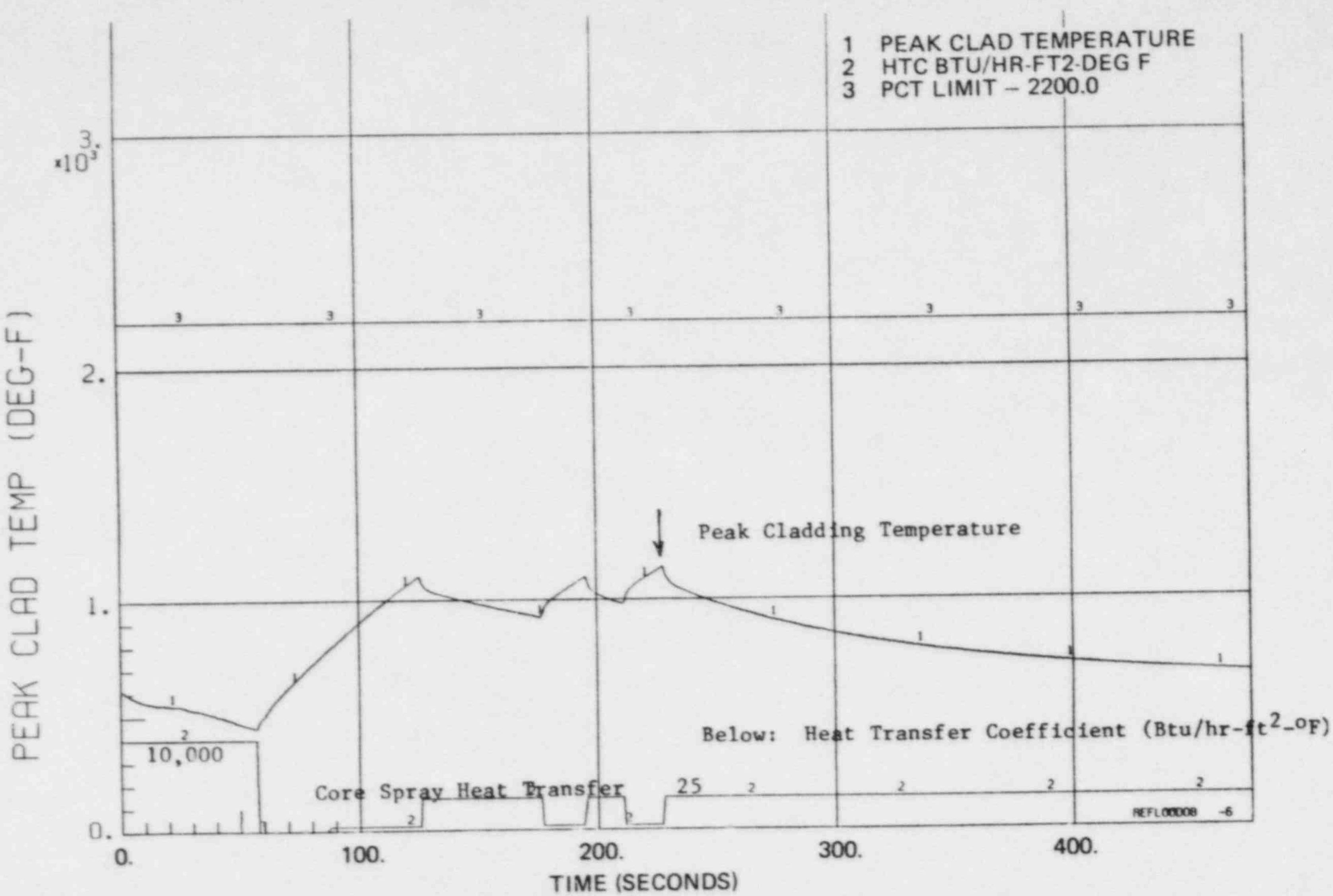

Figure 2d. Peak Cladding Temperature Following a Small Break of the Recirculation Line, LPCI Injection Valve Failure, Break Area = 0.8 ft^2 (SBM)

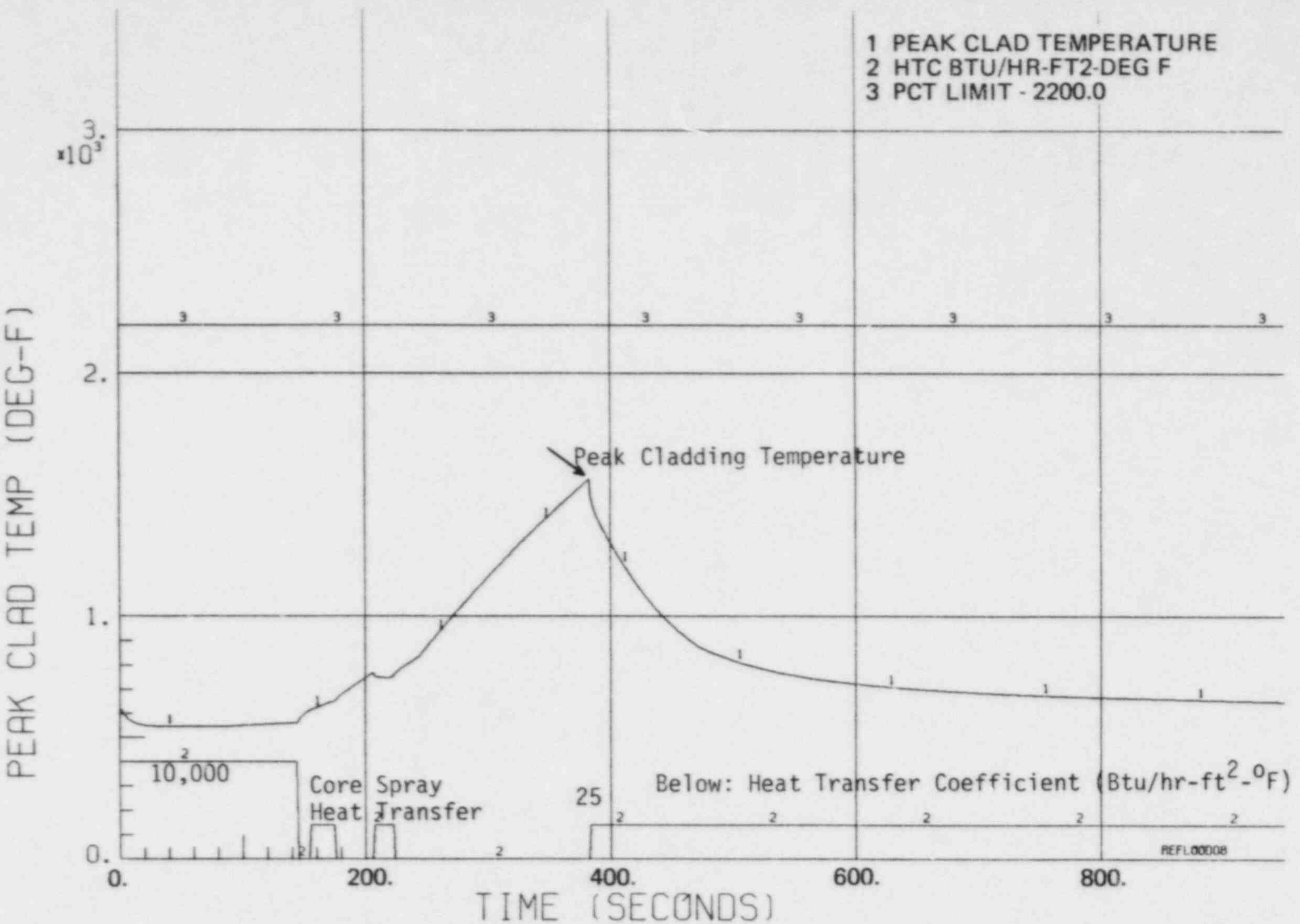

Figure 2e. Peak Cladding Temperature Following a Small Break of the Recirculation Line, HPCI Failure, Break Area = 0.07 ft^2 (SBM)

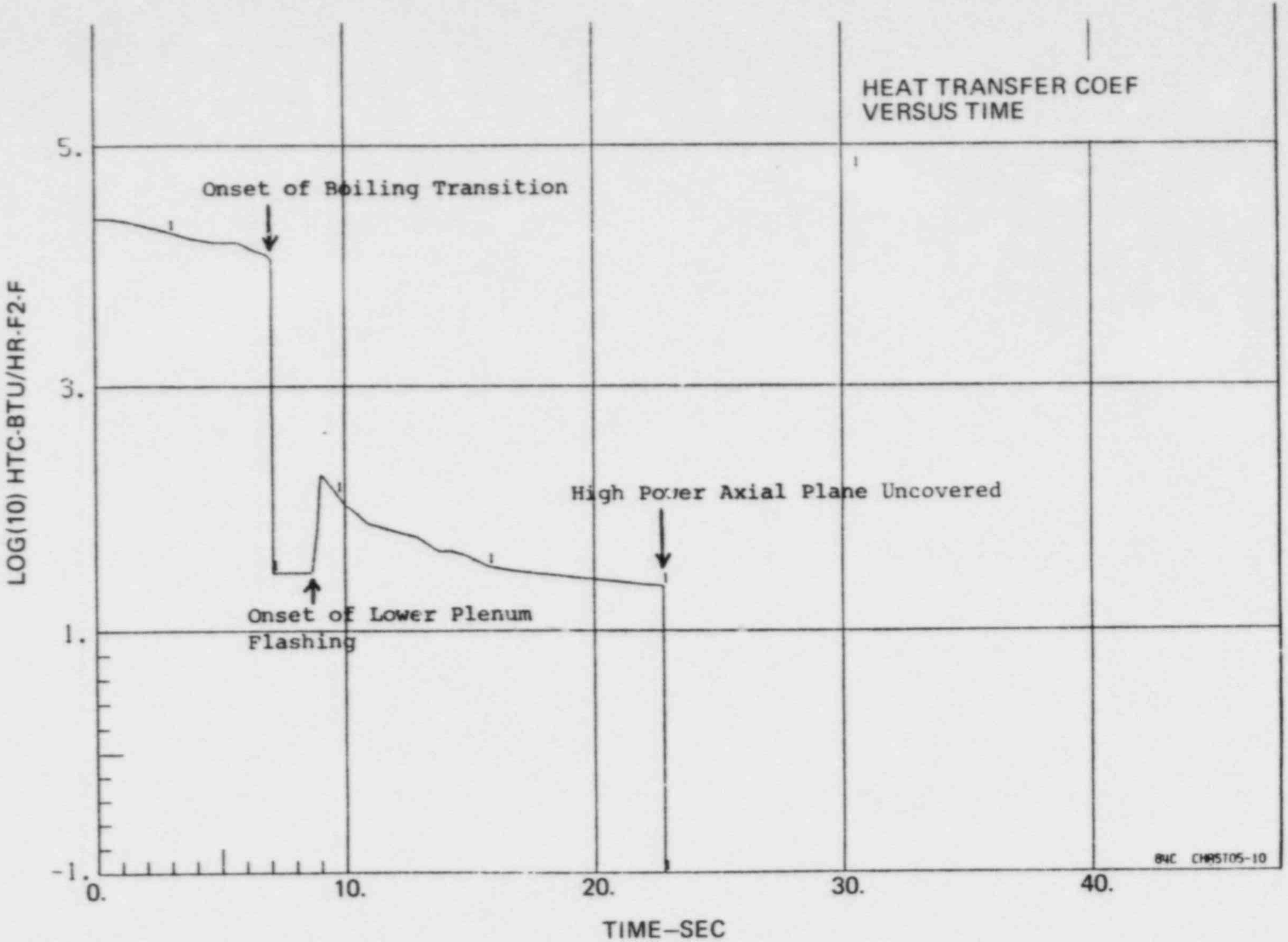

Figure 5a. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node for DBA

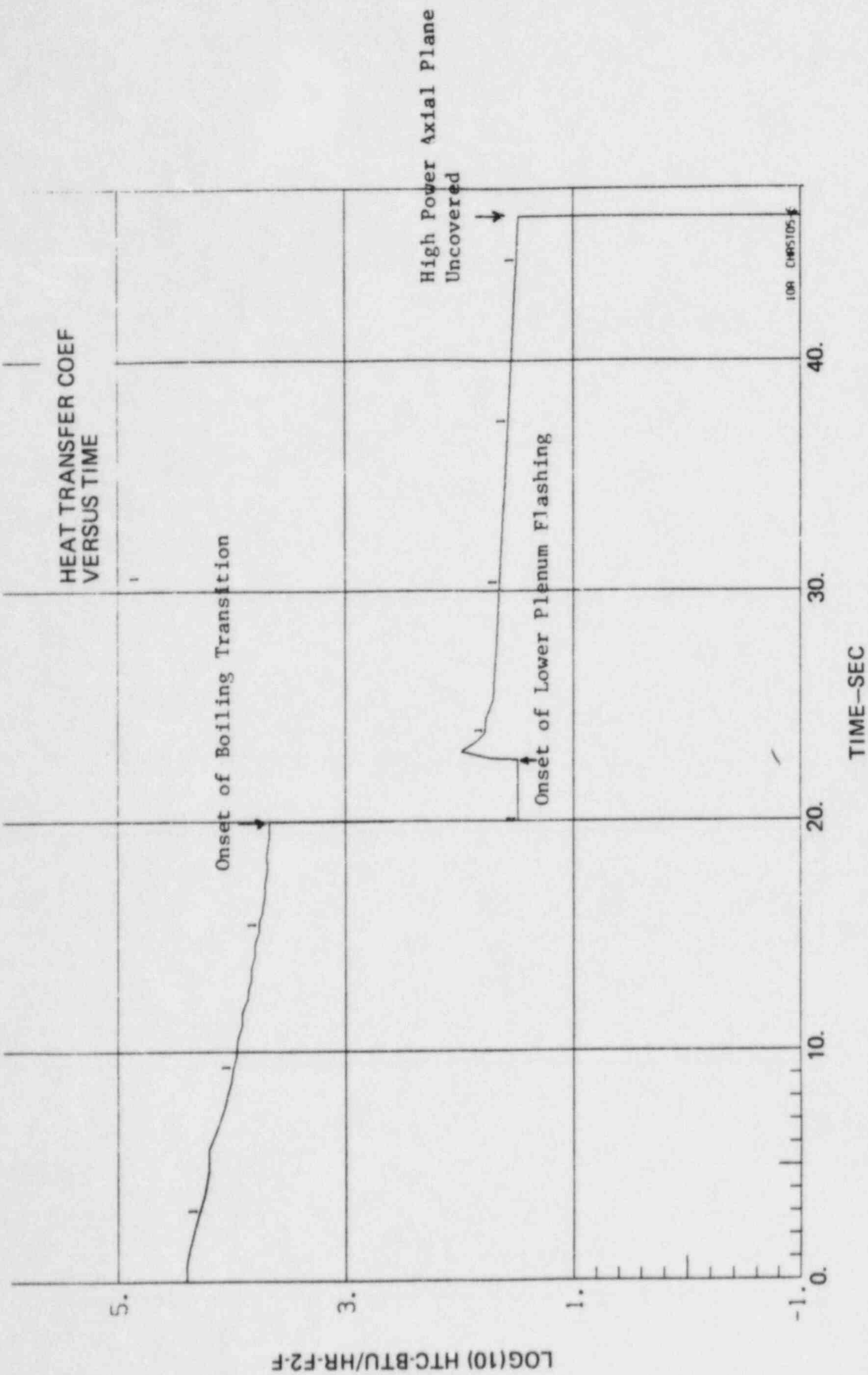

Figure 3b. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node for 1.0 ft^2 Break (LRM)

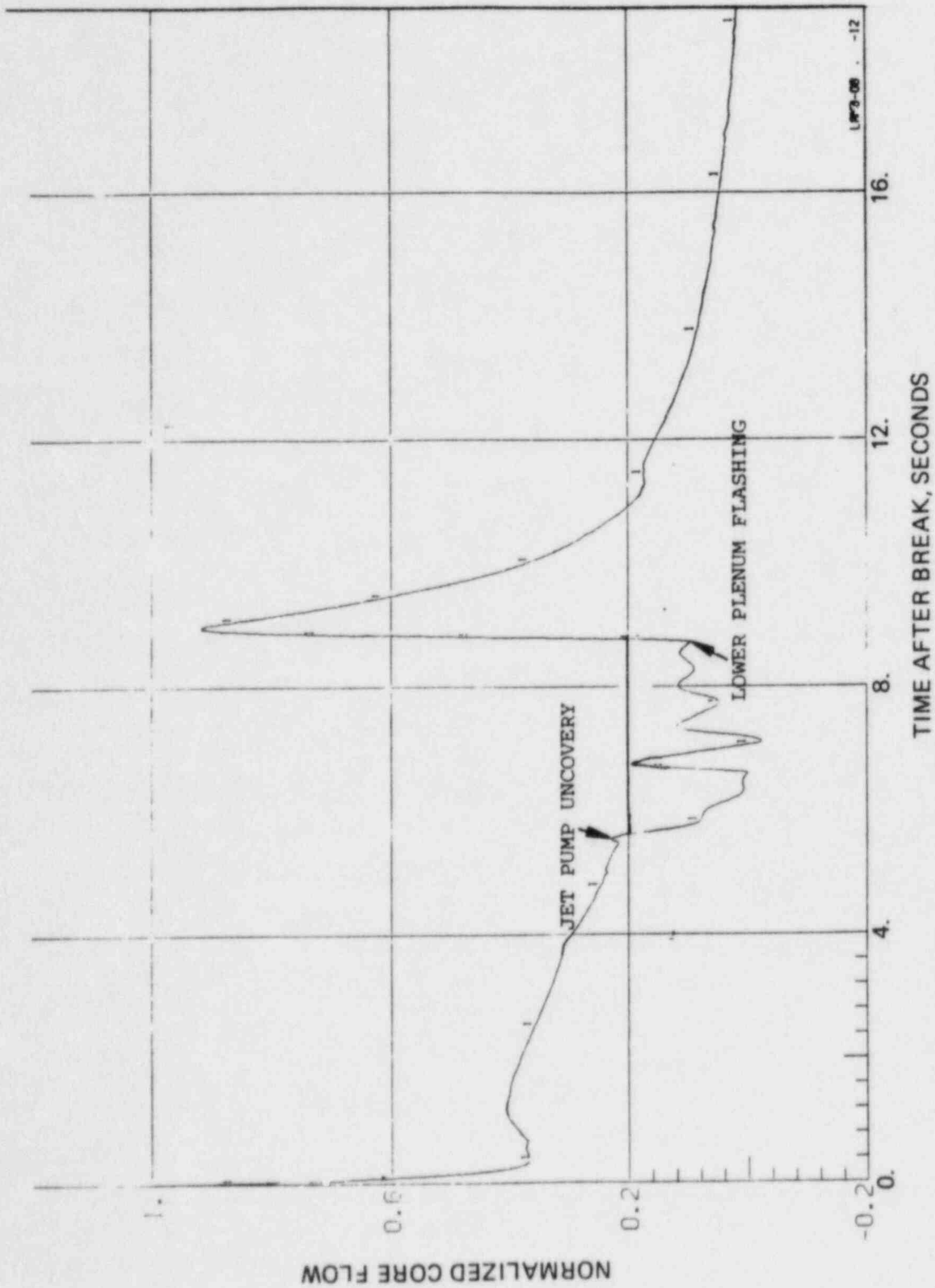

Figure 4a. Normalized Core Average Inlet Flow Following a Design Basis Accident

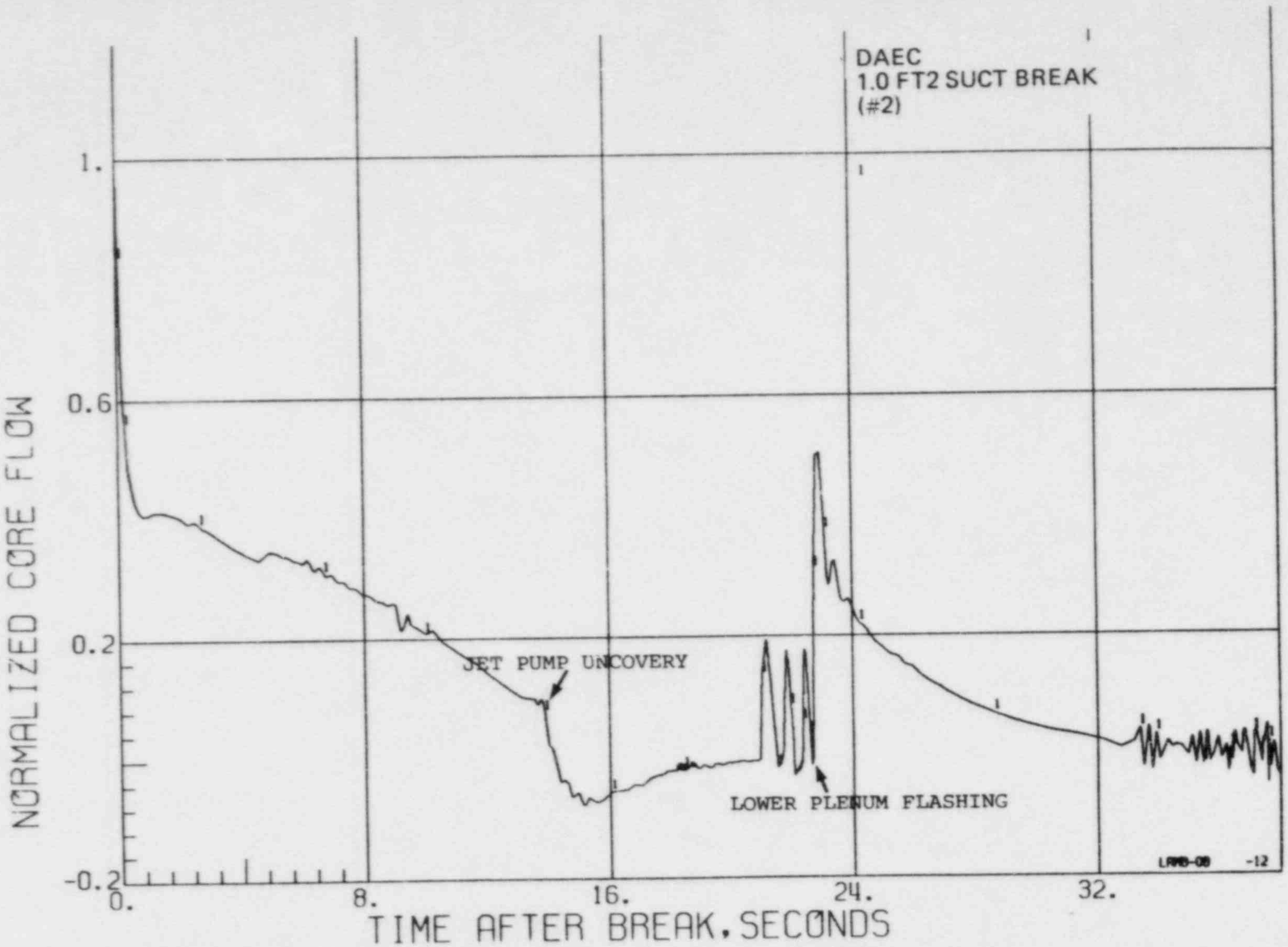

Figure 4b. Normalized Core Average Inlet Flow Following a 1.0 ft^2 Break (LBM)

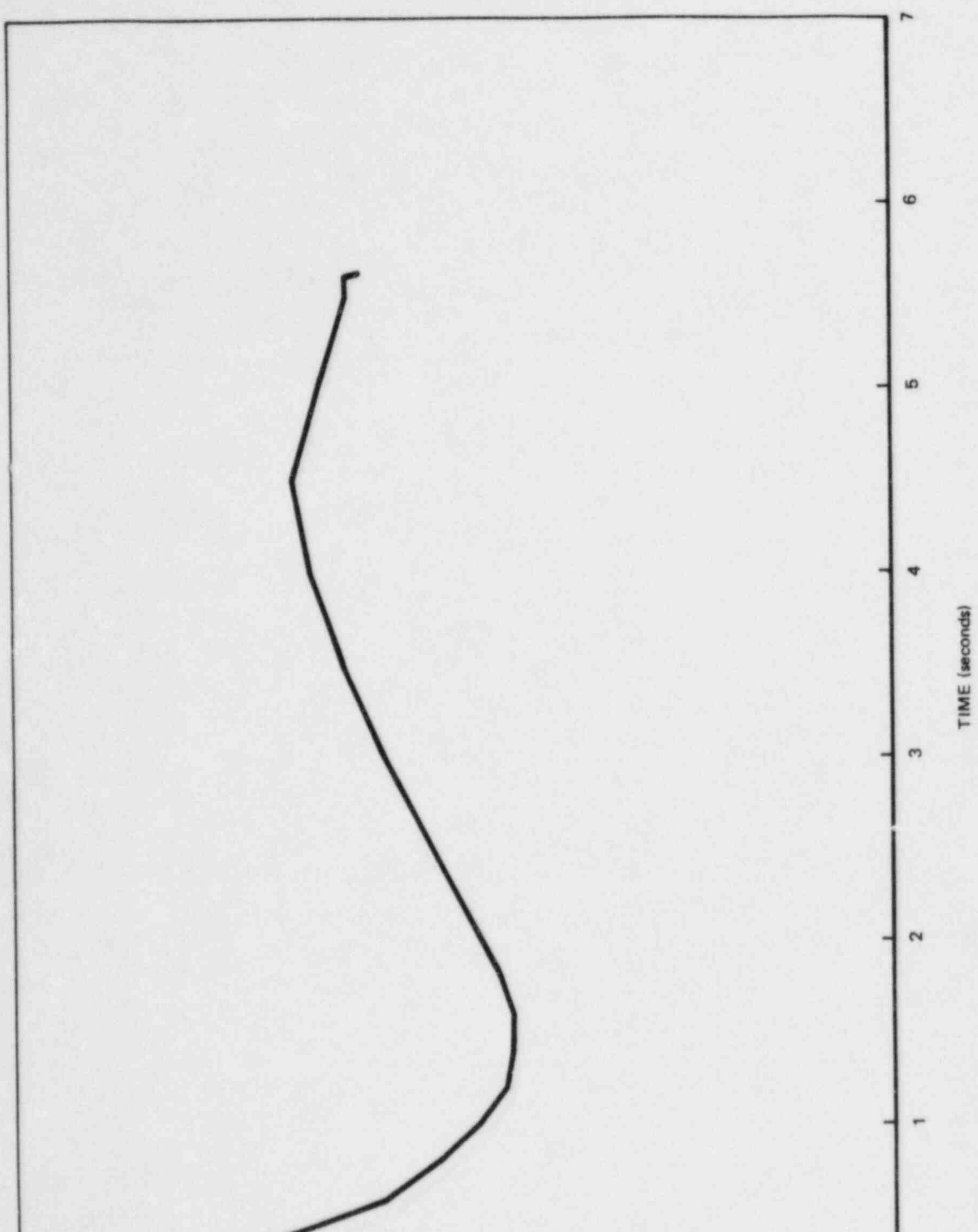

Figure 5a. Minimum Critical Power Ratio Following a DBA

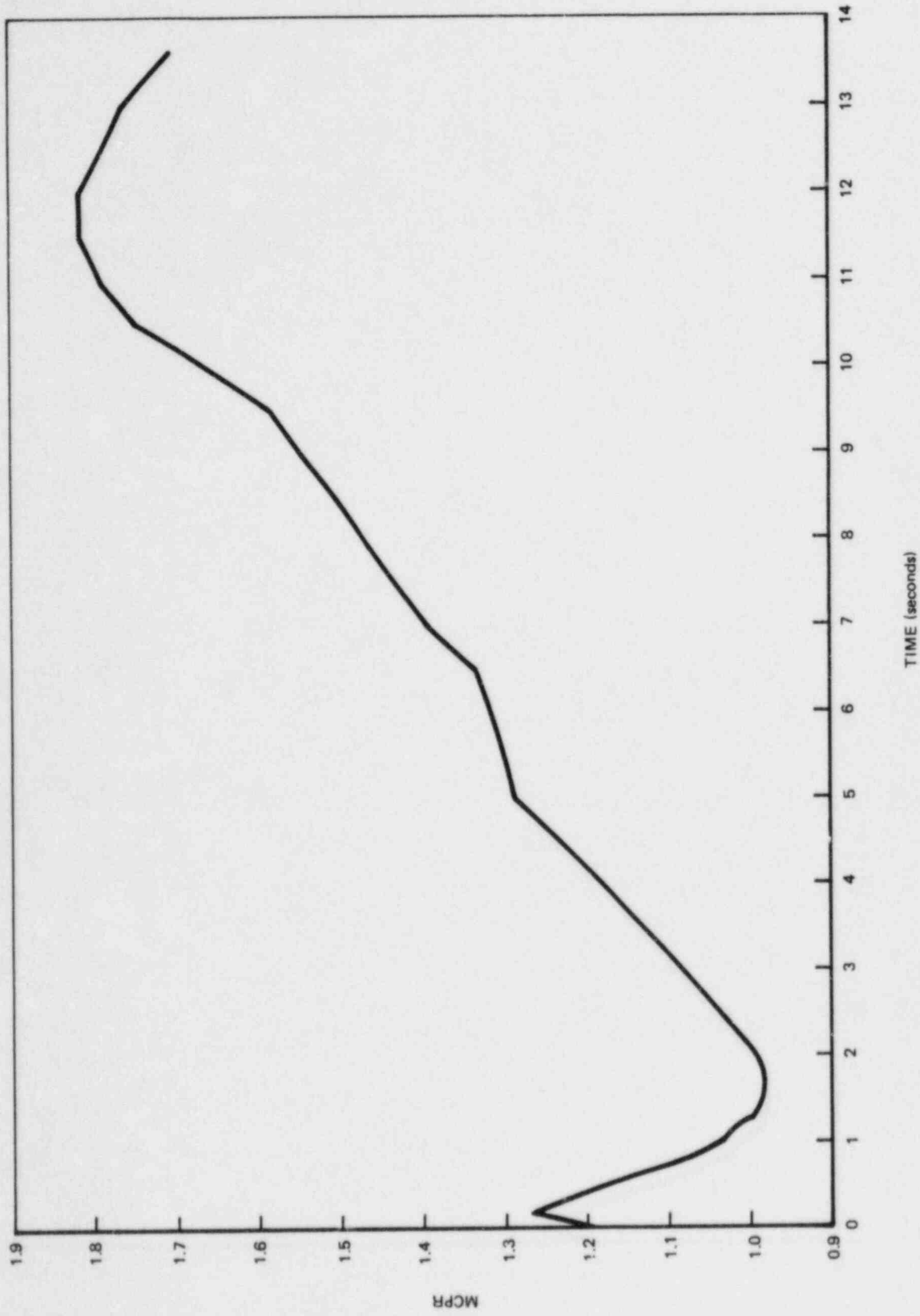

Figure 5b. Minimum Critical Power Ratio Following a 1 ft^2 Break (LBM)

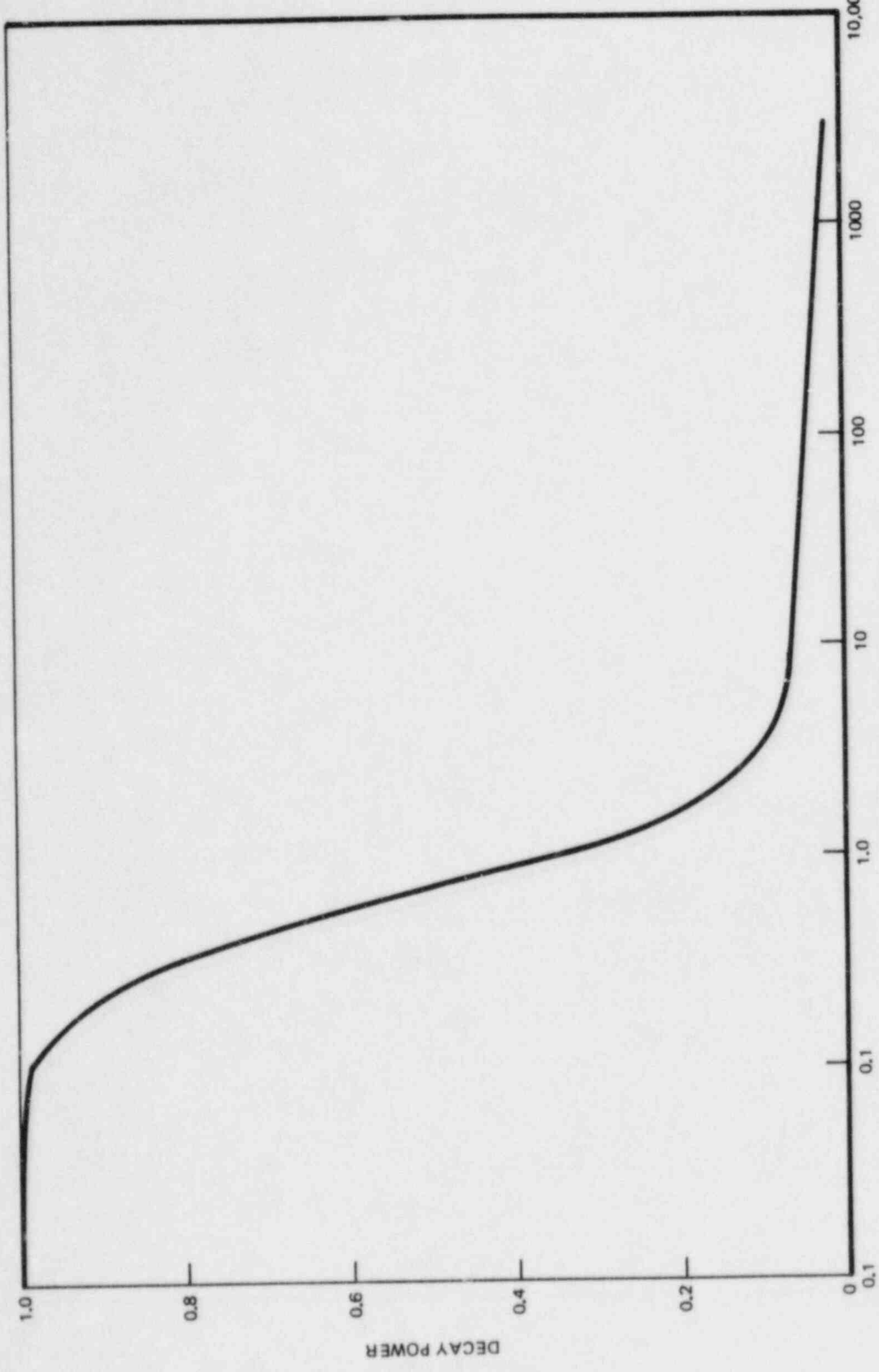

Figure 6. Normalized Power Versus Time

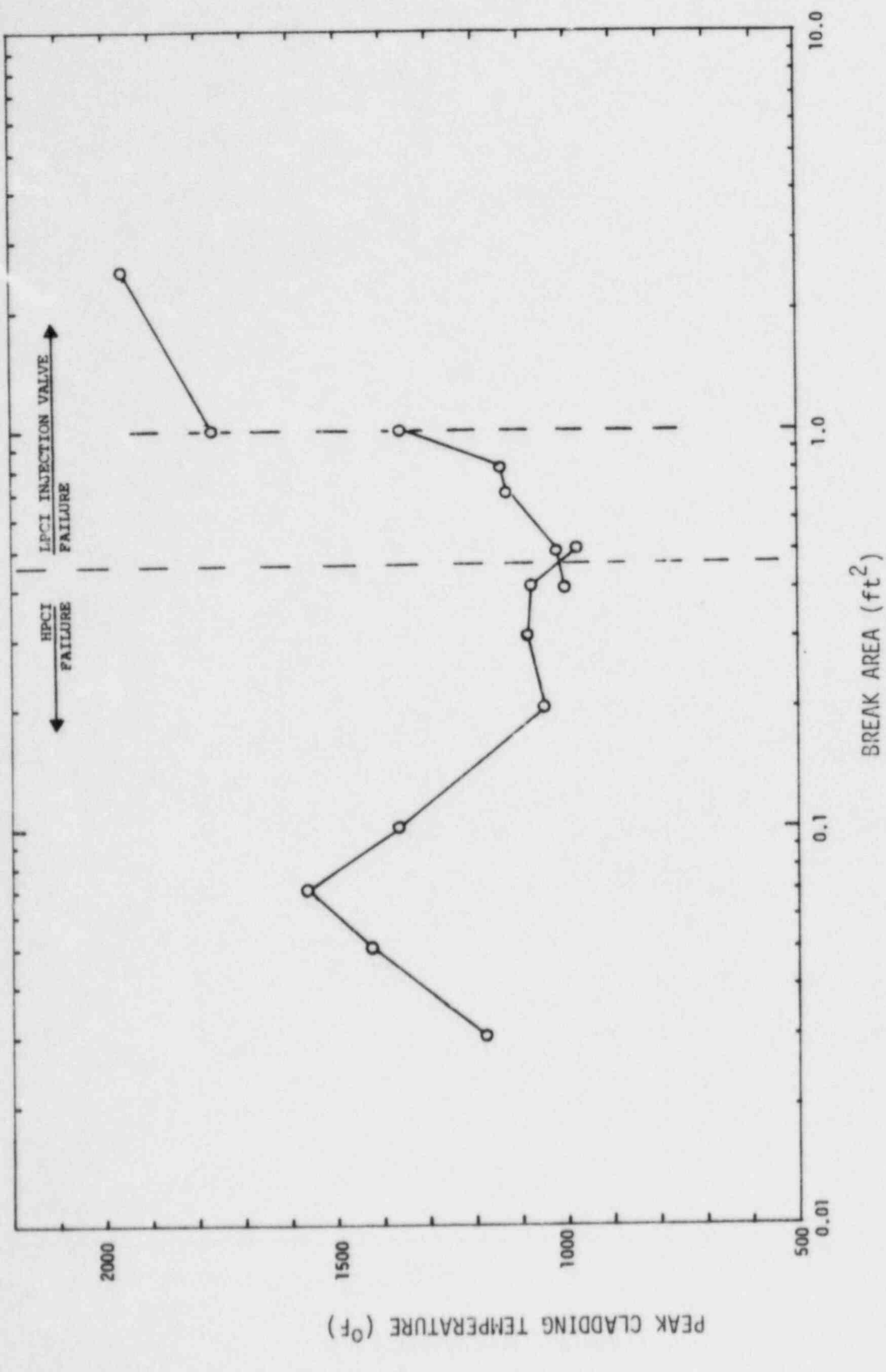

Figure 7. Peak Cladding Temperature Versus Break Area

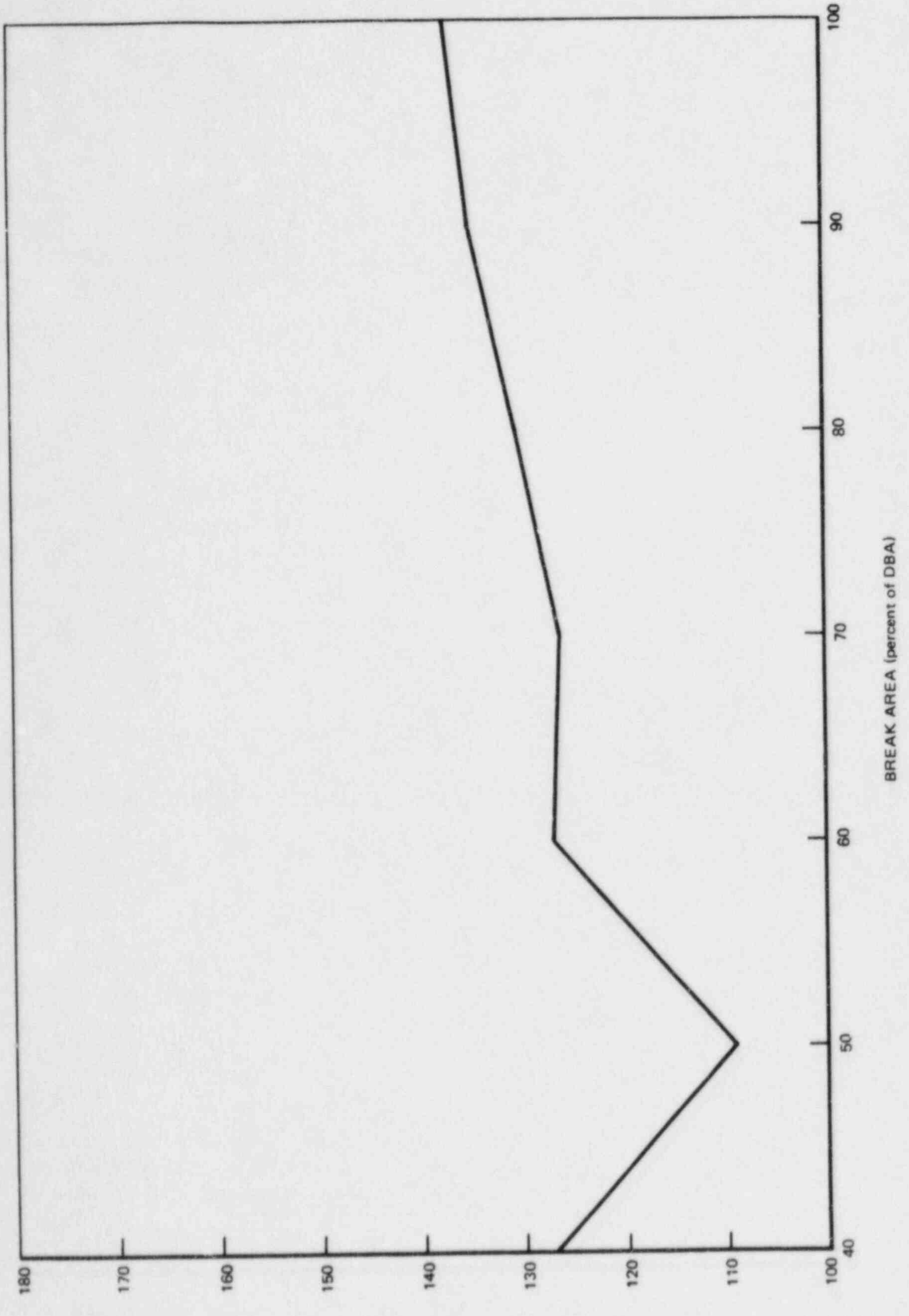

Figure 8. Variation with Break Area of Time for Which Hot Node Remains Uncovered

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