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Class I
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LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT

FOR

MONTICELLO NUCLEAR GENERATING PLANT

BOILING WATER REACTOR PROJECTS DEPARTMENT • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF ILLUSTRATIONS

1. INTRODUCTION

The purpose of this document is to provide the results of the loss-of-coolant accident (LOCA) analysis for the Monticello Nuclear Generating Plant (Monticello). The analysis was performed using approved General Electric (GE) calculational models.

This reanalysis 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 is 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. Recently approved model changes (Reference 4) are described in References 5 and 6. These model changes are employed in the new REFLOOD and CHASTE computer codes which have been used in this analysis. In addition, a model which takes into account the effects of drilling alternate flow path holes in the lower tieplate of the fuel bundle and the use of such fuel bundles in a full or partial core loading is described in References 7, 8, and 9. This model was also approved in Reference 4. Also included in the reanalysis are current values for input parameters based on the LOCA analysis reverification program being carried out by GE. The specific changes as applied to Monticello are discussed in more detail in later sections of this document.

Plants are separated into groups for the purpose of LOCA analysis (Reference 10). 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 non-lead plant analysis for Monticello, which is a BWR/3 group of plants 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. The difference in input paramenters 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. Input changes have been made in the new analysis which are essentially an upgrading of the input paramenters to the computer codes. Thus, the lead plant concept is still valid for this evaluation.

2. LEAD PLANT SELECTION

Lead plants are selected and analyzed in detail to permit a more 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.

The lead plant for Monticello is Quad Cities. The justification for categorizing Monticello in this group of plants and the lead plant analysis for this group is presented in Reference 11.

3. INPUT TO ANALYSIS

A list of the significant plant input parameters to the LOCA analysis is presented in Table 1.

Table 1

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

Plant Parameters:

Core Thermal Power	1703 MWt, which corresponds to 102% of rated core power
Vessel Steam Output	6.91 x 10^6 lbm/h, which corresponds to 102% of rated core power
Vessel Steam Dome Pressure	1040 psia
Recirculation Line Break Area for Large Breaks - Suction	1.6 ft^2 (40% DBA), 4.0 ft^2 (DBA)
Number of Drilled Bundles	0

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.	8D219	8 x 8	13.4	1.57	1.2
B.	8D250	8 x 8	13.4	1.57	1.2
C.	8D262	8 x 8	13.4	1.57	1.2

*To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

4. LOCA ANALYSIS COMPUTER CODES

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 for 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.

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.

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

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 requested 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 greater 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 methods

4.5 RESULTS 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 Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size

A summary of the analytical results is given in Table 2. Table 3 lists the figures provided for this analysis. The MAPLHGR values for each fuel type in the Monticello core are presented in Tables 4A through 4C.

4.6 METHODS

In the following sections, it will be useful to refer to the methods used to analyze DBA, large breaks, a d 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 $Btu/hr\text{-}ft^2\text{-}°F$ after recovery, core spray when appropriate. Peak cladding temperature and peak local oxidation are calculated in non-DBA-REFLOOD. Break sizes: $A \leq 1.0\ ft^2$.

Table 2

SUMMARY OF BREAK SPECTRUM RESULTS

• Break Size • Location • Single Failure	PCT (°F)	Peak Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
• 1.6 ft^2 (40% DBA) • Recirc Suction • LPCI Injection Valve	2200[(1)]	3.4	0.23
• 4.0 ft^2 (DBA) • Recirc Suction • LPCI Injection Valve	2095[(1)]	Note 2	Note 3

1. PCT from CHASTE
2. Less than most limiting break (3.4%)
3. Less than most limiting break (0.23%)

Table 3

LOCA ANALYSIS FIGURE SUMMARY — NON-LEAD PLANT

	Large Break Methods	
	Limiting Suction Break (LPCI Injection Valve Failure) (1.6 ft^2) (40% DBA)	Maximum Suction Break (LPCI Injection Valve Failure) (4.0 ft^2) (DBA)
Water Level Inside Shroud and Reactor Vessel Pressure	1a	1b
Peak Cladding Temperature	2a	2b
Heat Transfer Coefficient	3a	3b
Core Average Inlet Flow	4a	4b
Minimum Critical Power Ratio	5a	5b
Peak Cladding Temperature of the Highest Powered Plane Experiencing Boiling Transition	2a	
Variation with Break Area of Time for Which Hot Node Remains Uncovered	6	

Table 4A

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Monticello   Fuel Type: 8D219

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.7	2199	0.033
1,000	10.7	2199	0.033
5,000	10.8	2200	0.033
10,000	10.7	2196	0.033
15,000	10.7	2199	0.033
20,000	10.6	2194	0.033
25,000	10.6	2200	0.034
30,000	10.2	2138	0.028

Table 4B

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Monticello   Fuel Type: 8D250

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
300	10.6	2195	0.033
1,000	10.7	2198	0.033
5,000	10.7	2195	0.033
10,000	10.8	2194	0.032
15,000	10.7	2197	0.033
20,000	10.6	2196	0.033
25,000	10.6	2198	0.033
30,000	10.6	2199	0.034

Table 4C

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Monticello

Fuel Type: 8D262

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.6	2197	0.033
1,000	10.7	2195	0.033
5,000	10.7	2196	0.033
10,000	10.8	2197	0.033
15,000	10.7	2199	0.033
20,000	10.7	2198	0.033
25,000	10.6	2196	0.033
30,000	10.6	2198	0.034

5. DESCRIPTION OF MODEL AND INPUT CHANGES

This section provides a general description of the input and model changes as they relate to the break spectrum calculations. It provides a general background so that the more specific calculated results shown in subsequent sections can be more easily understood, particularly as they relate to how well trends observed in specific lead plant break spectrum analyses can be applied to the general nonlead plant case. The most limiting break size results are not discussed in this context (except to the extent that they affect the shape of the break spectrum) because detailed limiting break size calculational results will be presented for each plant.

The majority of the input and model changes primarily affect the amount of ECCS flow entering the lower plenum as a result of the counter current flow limiting (CCFL) effect. These changes as applied to Monticello are listed below.

1. Input Changes

    a. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

    b. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

    c. Corrected guide tube thermal resistance.

    d. Correct heat capacity of reactor internals head nodes.

2. Model Change

    a. Core CCFL pressure differential = 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

    b. Incorporate NRC pressure transfer assumption - The asumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

1. Input Change

   a. Break Areas - The DBA break area was calculated more accurately.

2. Model Change

   a. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

6. CONCLUSIONS

The LOCA analysis results in accordance with the requirements of Reference 2 for non-lead plants are presented in Figures 1a through 5a for the limiting suction break (40% DBA) and Figures 1b through 5b for the maximum suction break (DBA).

The characteristics that determine which is the most limiting break area at the DBA location 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 6 shows the variation with break size of the calculated time the hot node remains uncovered for Monticello. Based on these results the 40% DBA was determined to be the break that results in the highest calculated PCT in the 1.0 ft^2 to DBA region. The determination of the 40% DBA being the most limiting break was based on the reasoning discussed above and the procedure used for the lead plant. The 40% DBA was determined to be the most limiting break smaller than the DBA from Figure 6. Then a CHASTE calculation was performed to compare the PCT for the DBA and the 40% DBA. The 40% DBA was determined to result in a higher PCT compared to the DBA and, hence, was determined to be the most limiting break.

The conservative approach of using the 60% DBA LAMB/SCAT results with the 40% DBA SAFE/REFLOOD results for calculations for the 40% DBA was used in all calculations for the analysis to determine the MAPLHGR's in Tables 4A through 4C.

The DBA (the complete severence of the recirculation discharge piping) results are shown on Figures 1b through 5b. The most significant change in these results from the previous analysis is that the reflooding time decreases from approximately 330 seconds to approximately 260 seconds. This is due to the input and model changes described in Section 5.

The single failure evaluation showing the remaining ECCS following an assumed failure and the effects of a single failure or operator error that causes any manually controlled, electrically operated valve in the ECCS to move to a position that could adversely affect the ECCS are presented in Reference 12.

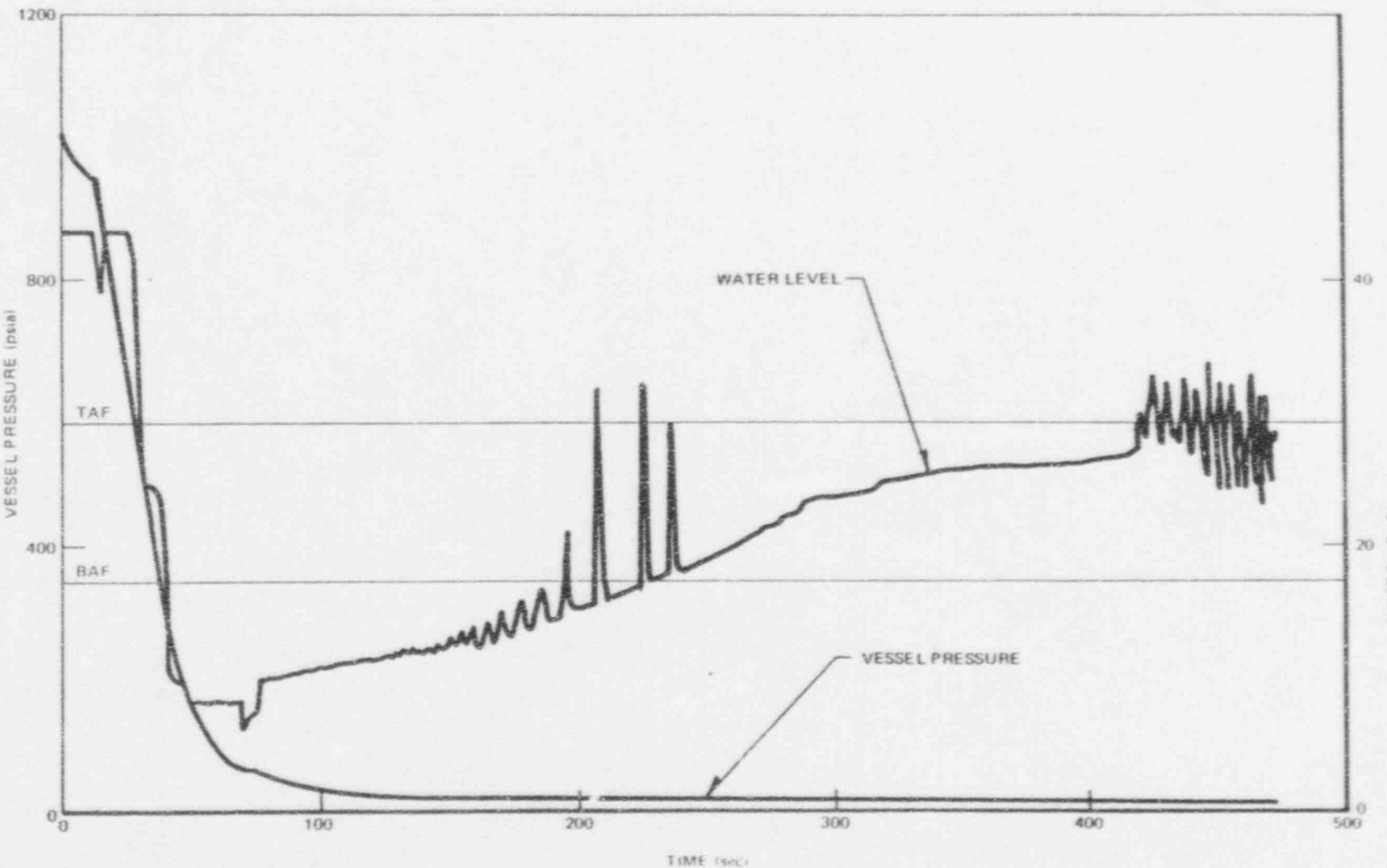

Figure 1a. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 1.6 ft^2 Recirculation Line Suction Break, LPCI Injection Valve Failure (40% DBA) (LBM)

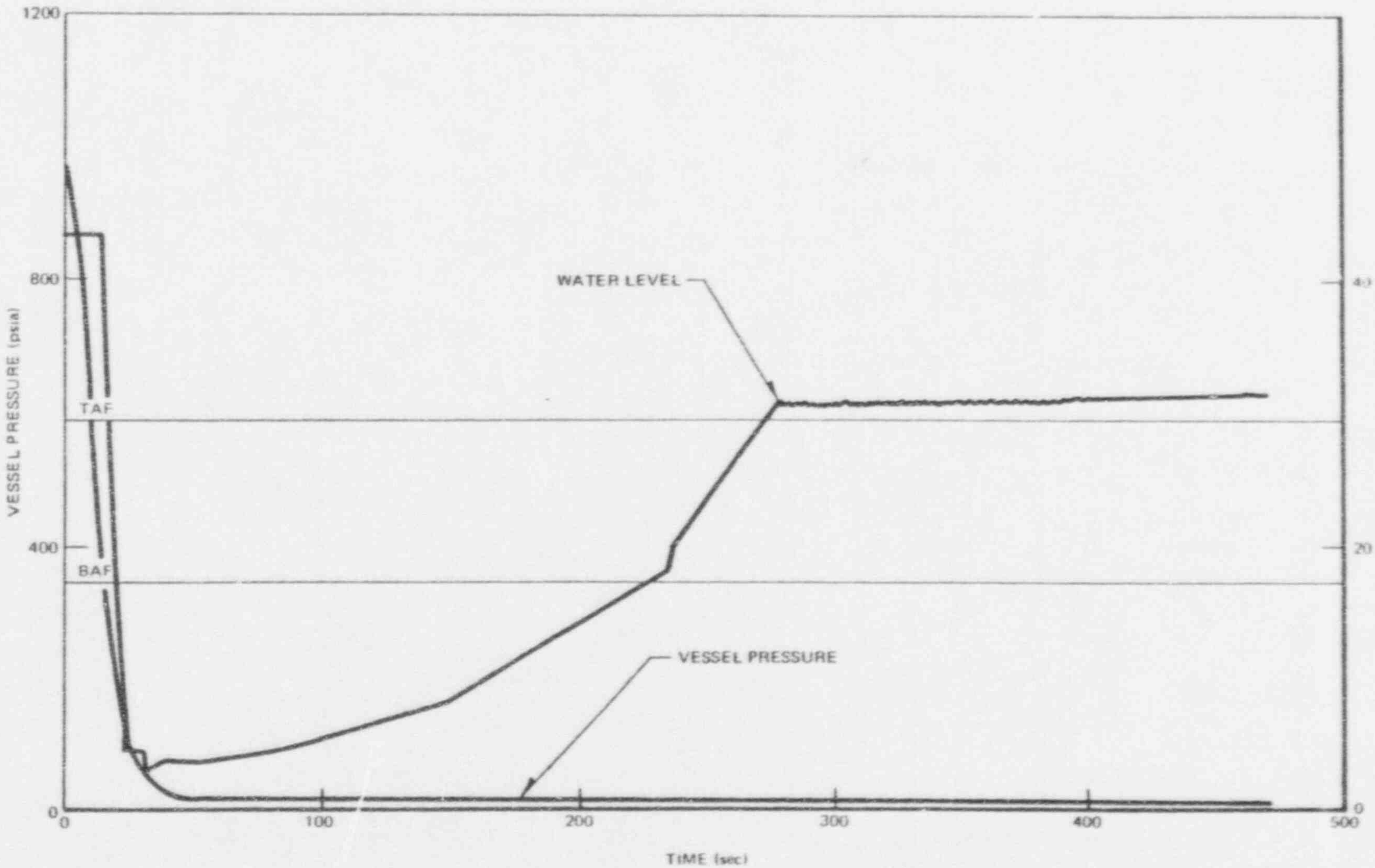

Figure 1b. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 4.0 ft^2 Recirculation Line Suction Break, LPCI Injection Valve Failure, (DBA)

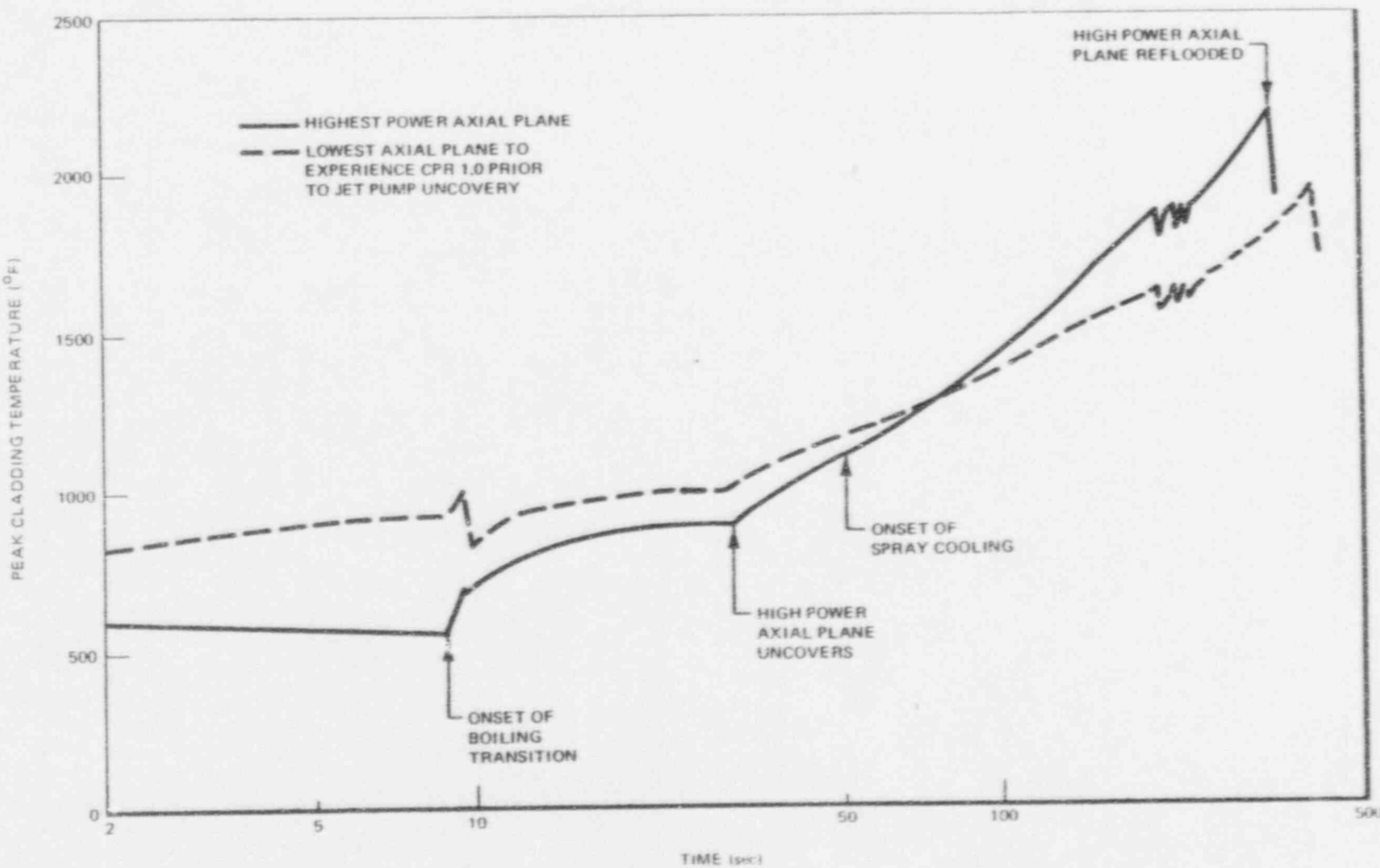

Figure 2a. Peak Cladding Temperature Following a 1.6 ft^2 Recirculation Line Suction Break, LPCI Injection Valve Failure, Break Area = (40% DBA) (LBM)

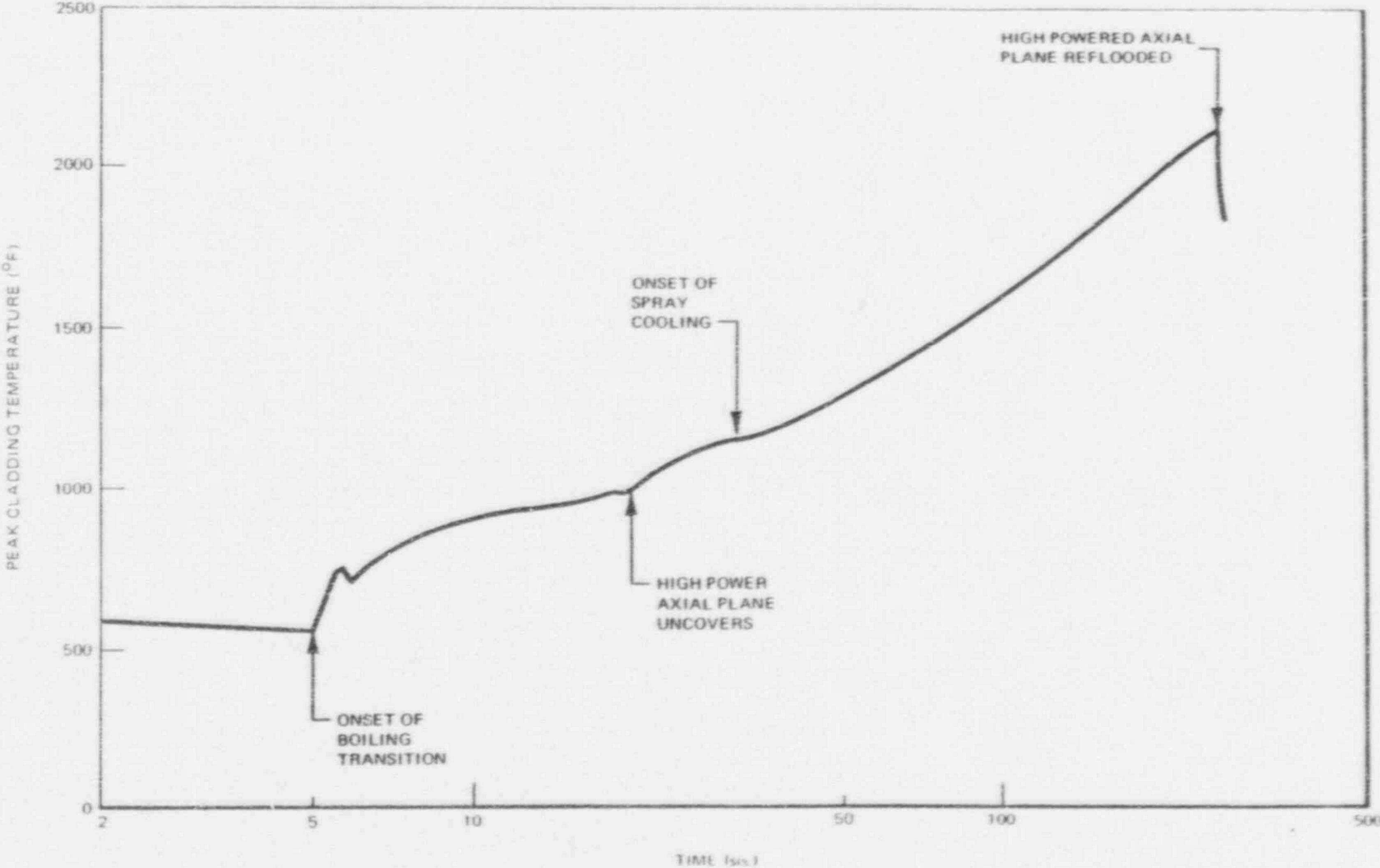

Figure 2b. Peak Cladding Temperature Following a 4.0 ft^2 Recirculation Line Suction Break, LPCI Injection Valve Failure, (DBA)

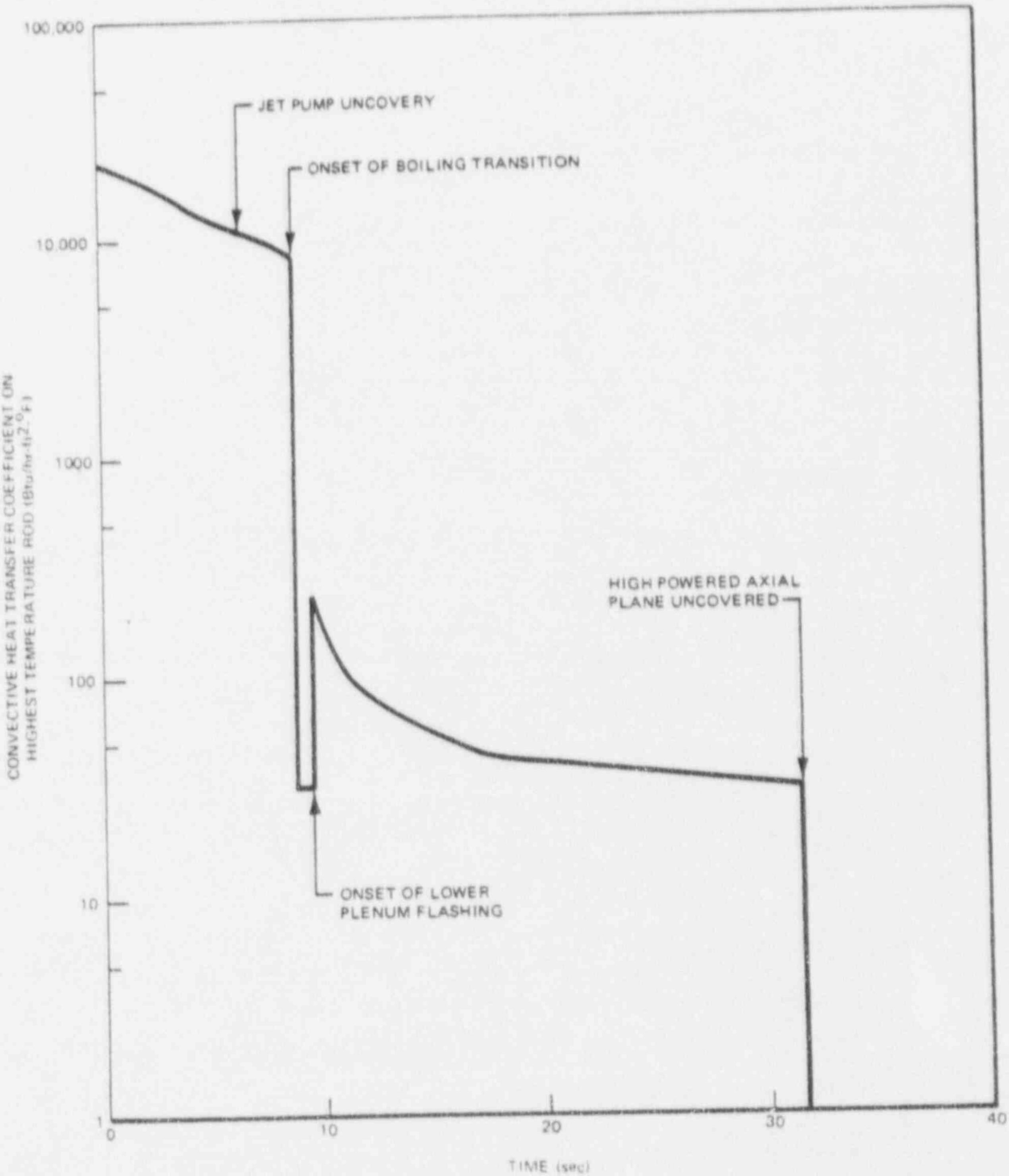

Figure 3a. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node Following a 1.6 ft^2 Reci·culation Line Suction Break, LPCI Injection Valve Failures, (40% DBA)

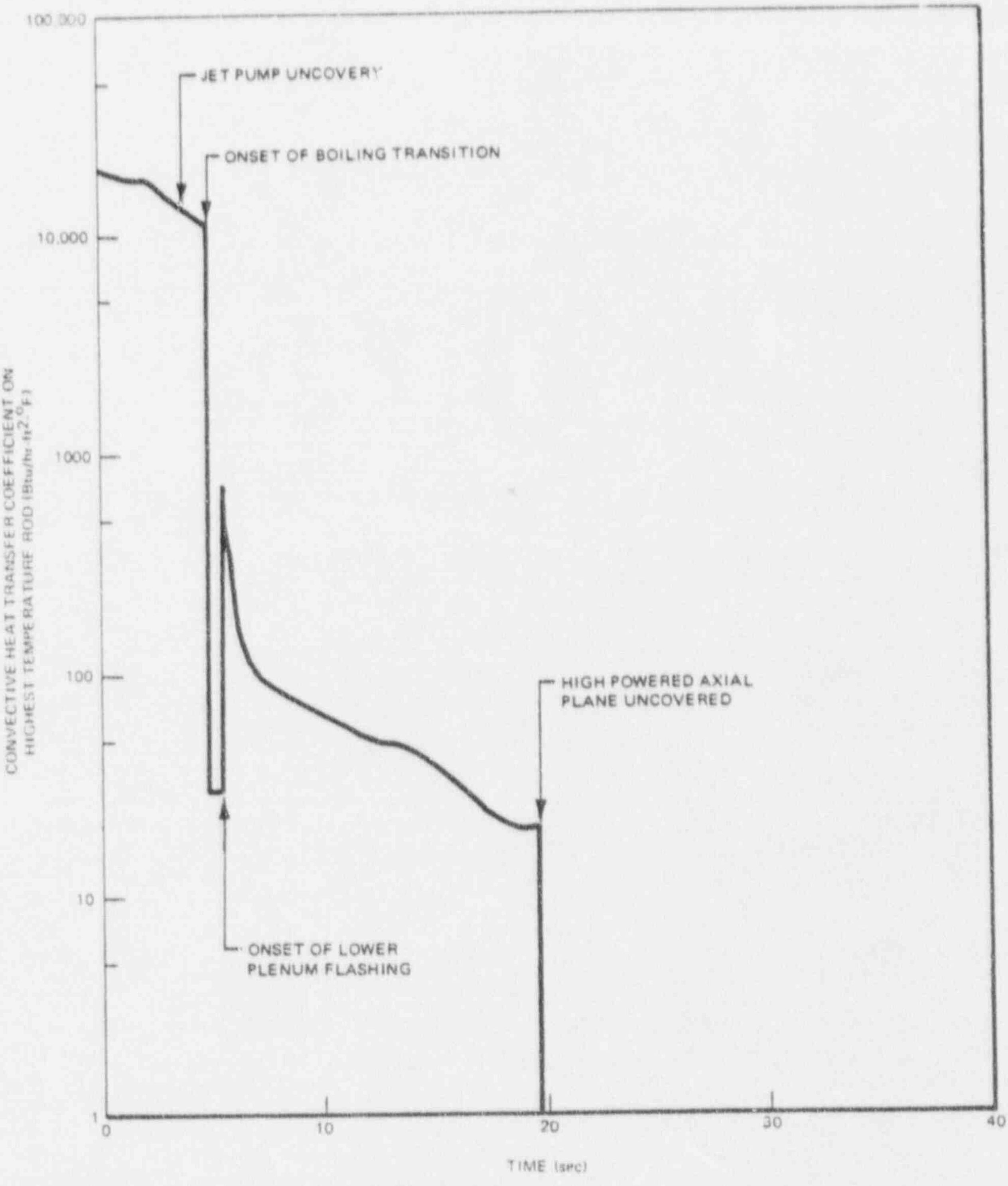

Figure 3b. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node for a 4.0 ft^2 Recirculation Line Suction Break, LPCI Injection Valve Failure, (DBA)

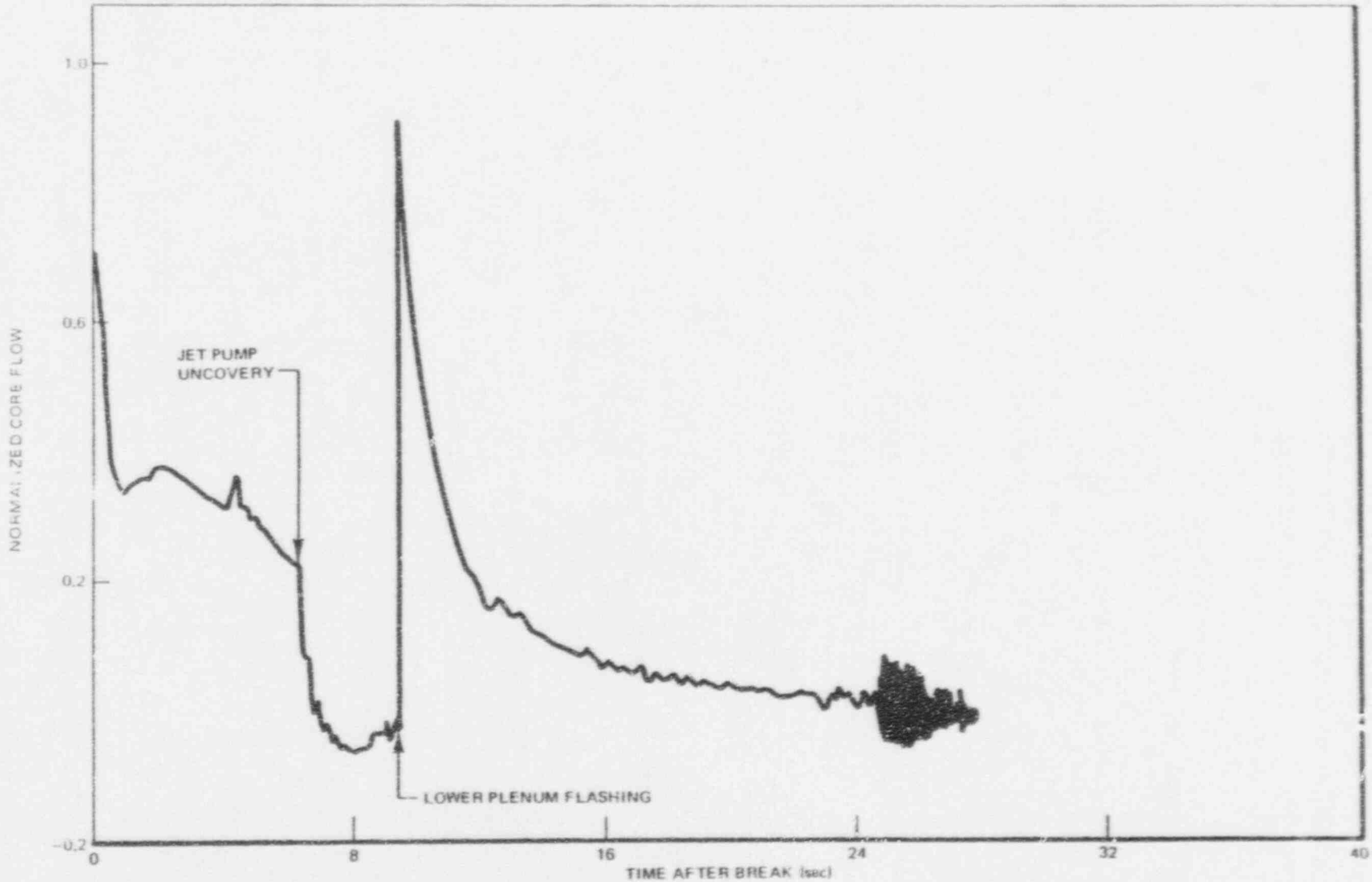

Figure 4a. Normalized Core Average Inlet Flow Following a 2.4 ft^2 Recirculation Line Suction Break, (60% DBA)

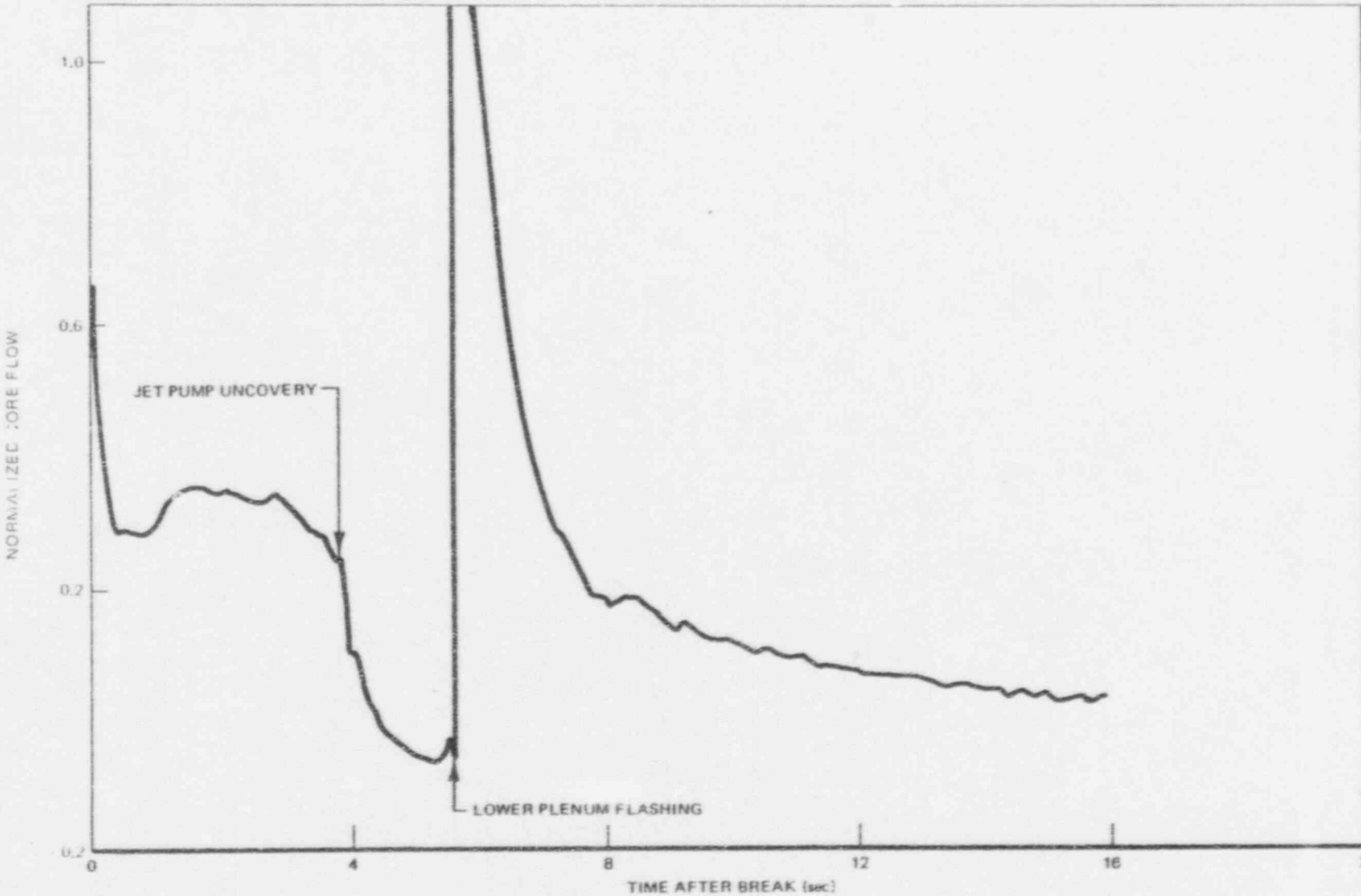

Figure 4b. Normalized Core Average Inlet Flow Following a 4.0 ft^2 Recirculation Line Suction Break (L.A)

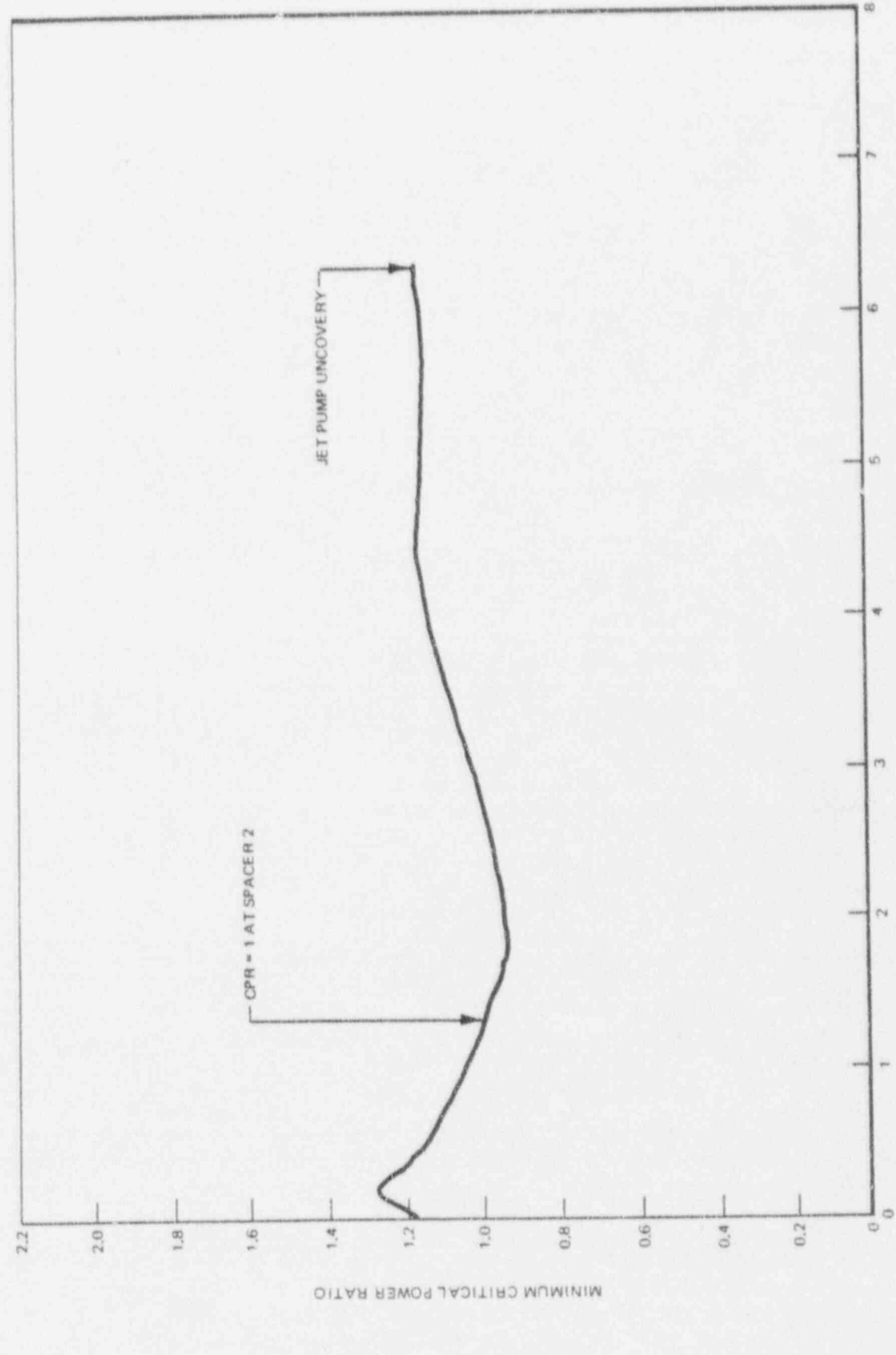

Figure 5a. Minimum Critical Power Ratio Following a 2.4 ft^2 Recirculation Line Suction Break (60% DBA)

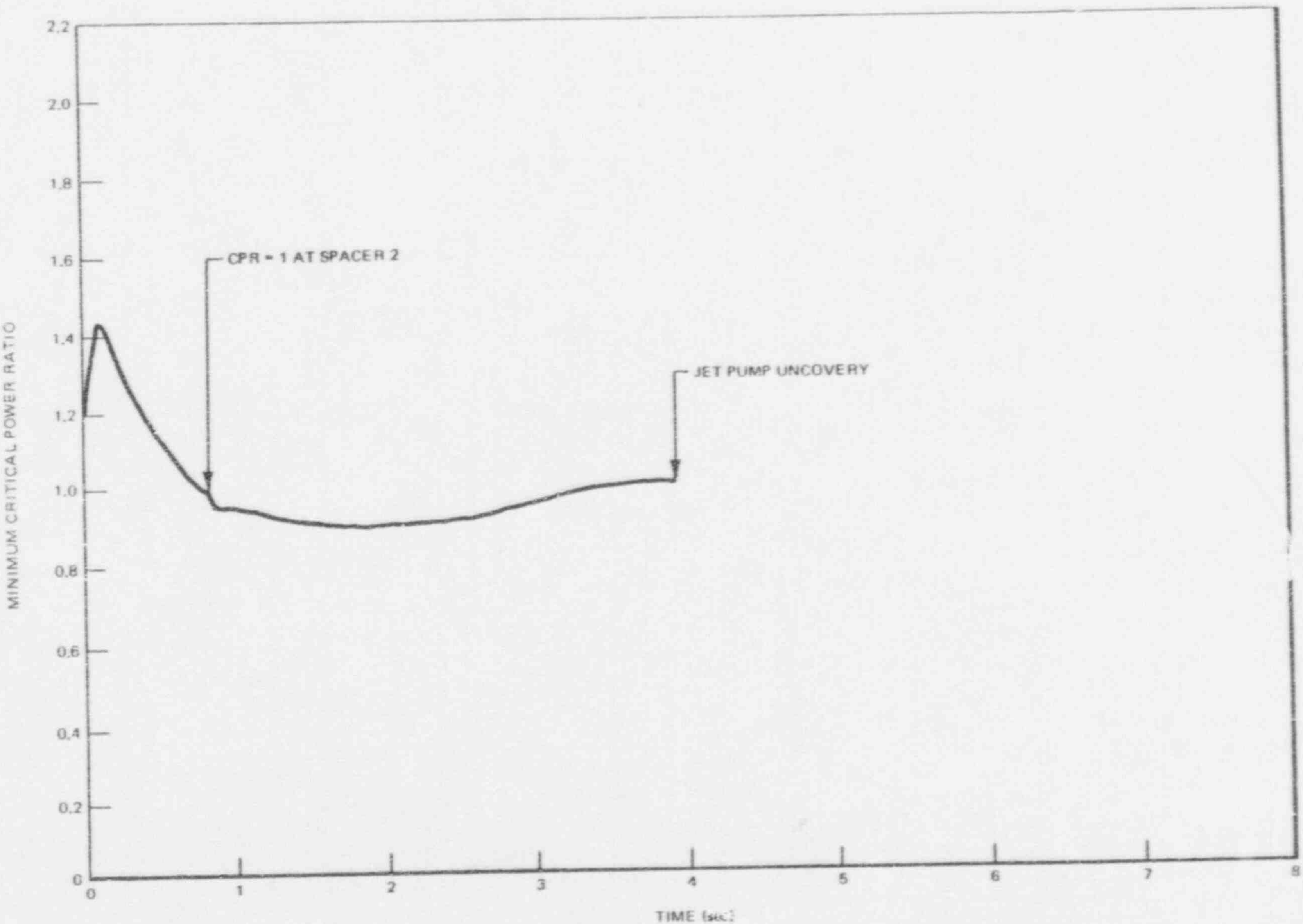

Figure 5b. Minimum Critical Power Ratio Following a 4.0 ft^2 Recirculation Line Suction Break (DBA)

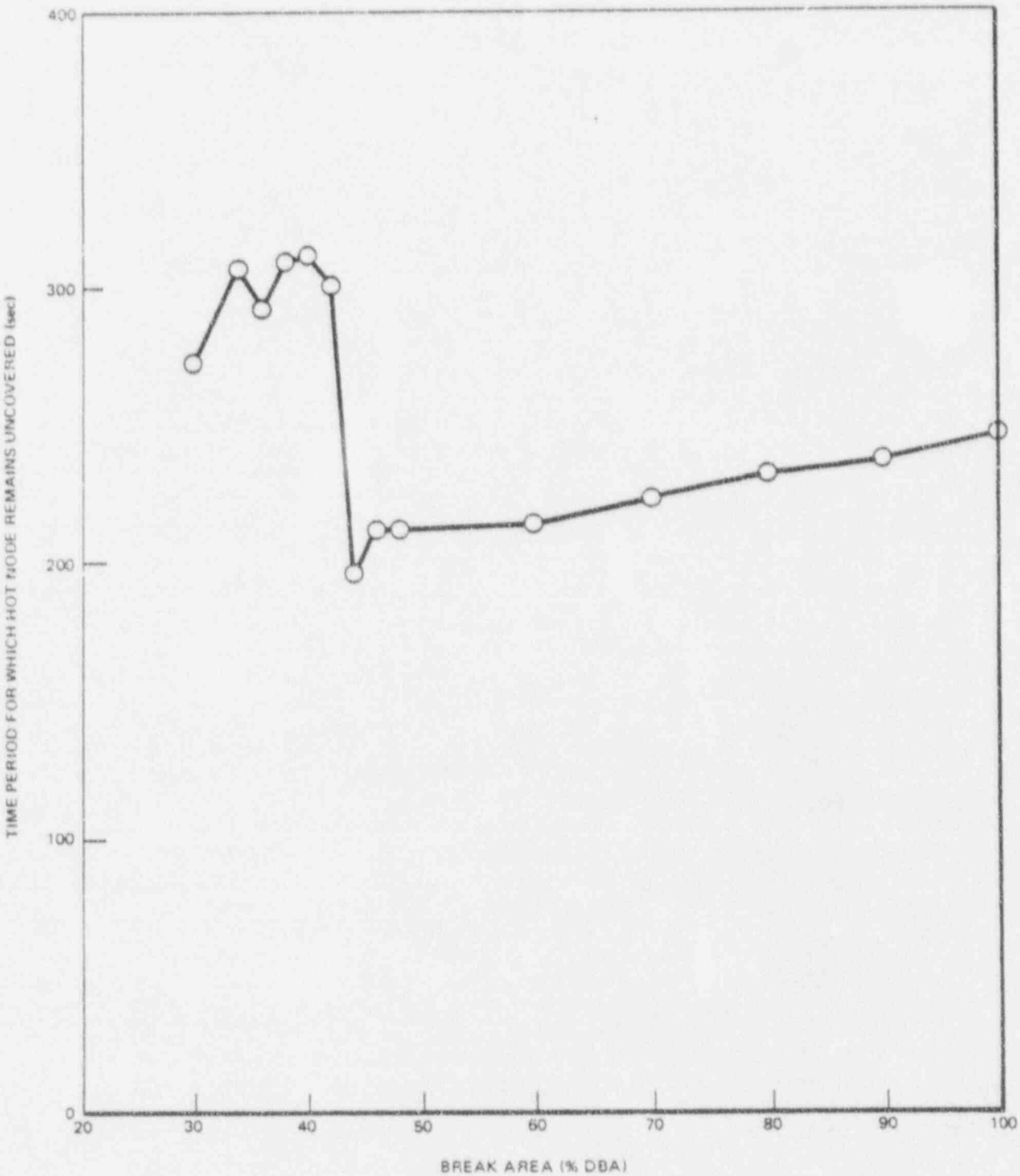

Figure 6. Variation with Break Area of Time for Which Ho. Node Remains Uncovered

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7. "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations," Supplement 1, NEDE-21156-1, September 1976.

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