NEDO-24088-2
REVISION 2
77NED360
CLASS I
MAY 1985

LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR BROWNS FERRY NUCLEAR PLANT UNIT 2

GENERAL ELECTRIC

NEDO-24088-2
Revision 2
77NED360
Class I
May 1985

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR

BROWNS FERRY NUCLEAR PLANT

UNIT 2

NUCLEAR ENERGY BUSINESS OPERATIONS • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

Please Read Carefully

The only undertaking of General Electric Company respecting information in this document are contained in the contract between the Tennessee Valley Authority and General Electric Company and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than the Tennessee Valley Authority or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS (Continued)

1. INTRODUCTION

The purpose of this document is to provide the results of the loss-of-coolant accident (LOCA) analysis for the Browns Ferry Nuclear Plant Unit 2 (BF-2). 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 BF-2 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 BF-2, which is a BWR/4 with 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. The difference 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. Input changes have been made in the new analysis which are essentially an upgrading of the input parameters 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 BF-2 is James A. FitzPatrick Nuclear Power Plant. The justification for categorizing BF-1 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	3440 MWt, which corresponds to 105% of rated steam flow
Vessel Steam Output	14.05×10^6 lbm/h, which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks - Discharge	1.9 ft^2 (DBA) 1.3 ft^2 (66% DBA)
- Suction	4.2 ft^2
Number of Drilled Bundles	744

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. IC Type 1 & 3	7 x 7	18.5	1.5	1.2
B. IC Type 2	7 x 7	18.5	1.5	1.2
C. 8DB274L	8 x 8	13.4	1.4	1.2
D. 8DB274H	8 x 8	13.4	1.4	1.2
E. 8DRB284L	8 x 8	13.4	1.4	1.2
F. P8DRB284L	8 x 8	13.4	1.4	1.2
G. P8DRB265H	8 x 8	13.4	1.4	1.2

*To account for the 2% uncertainty in the 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 Linear Heat Generation Rate (MAPLHGR) versus 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 BF-2 core are presented in Tables 4A through 4G.]

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 $Btu/hr\text{-}ft^2\text{-}°F$ 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 RESULTS

• Break Size • Location • Single Failure	PCT (°F)	Peak Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
• 1.3 ft^2 (66% DBA) • Recirc Discharge • LPCI Injection Valve	2128(1)	Note 2	0.12
• 1.9 ft^2 (DBA) • Recirc Discharge • LPCI Injection Valve	2086(1)	Note 2	Note 3
• 4.2 ft^2 • Recirc Suction • LPCI Injection Valve	2151(1)	5.7%	Note 3

1. PCT from CHASTE.
2. Less than most limiting break (5.7%).
3. Less than most limiting case (0.12%).

Table 3

LOCA ANALYSIS FIGURE SUMMARY

	Maximum Suction Break (LPCI Injection Valve Failure) (4.2 ft^2)	Maximum Discharge Break (LPCI Injection Valve Failure) (1.9 ft^2)	Limiting Discharge Break (LPCI Injection Valve Failure) (66% DBA)
Water Level Inside Shroud and Reactor Vessel Pressure	1a	1b	1c
Peak Cladding Temperature	2a	2b	2c
Heat Transfer Coefficient	3a	3b	3c
Core Average Inlet Flow	4a	4b	4c
Minimum Critical Power Ratio	5a	5b	5c
Peak Cladding Temperature of Highest Powered Plane Experiencing Boiling Transition	2a		
Uncovered Time Versus Break Area	6a,b		

Table 4A

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2

Fuel Type: Initial Core - Type 1 & 3

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	15.0	1926	0.009
1,000	15.1	1902	0.008
5,000	16.0	1975	0.011
10,000	16.3	2047	0.015
15,000	16.1	2151	0.055
20,000	15.4	2136	0.054
25,000	14.2	2035	0.037
30,000	13.1	1922	0.023
35,000	11.8	1821	0.015
40,000	10.5	1640	0.003

Table 4B

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 Fuel Type: Initial Core - Type 2

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	15.6	1973	0.010
1,000	15.5	1956	0.009
5,000	16.2	1973	0.010
10,000	16.5	2063	0.016
15,000	16.5	2143	0.057
20,000	15.8	2119	0.055
25,000	14.5	2005	0.038
30,000	13.3	1886	0.024
35,000	11.9	1782	0.015
40,000	10.6	1615	0.003

Table 4C

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 Fuel Type: 8DB274L

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1673	0.004
1,000	11.3	1681	0.004
5,000	11.9	1744	0.005
10,000	12.1	1755	0.005
15,000	12.2	1777	0.005
20,000	12.1	1778	0.005
25,000	11.6	1737	0.004
30,000	10.9	1666	0.003
35,000	9.9	1577	0.003
40,000	9.3	1520	0.003

Table 4D

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 Fuel Type: 8DB274H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.1	1667	0.004
1,000	11.2	1670	0.004
5,000	11.8	1730	0.005
10,000	12.1	1756	0.005
15,000	12.2	1778	0.005
20,000	12.0	1775	0.005
25,000	11.5	1733	0.004
30,000	10.9	1667	0.003
35,000	10.0	1579	0.003
40,000	9.3	1522	0.003

Table 4E

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 Fuel Type: 8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1747	0.005
1,000	11.3	1746	0.005
5,000	11.8	1804	0.006
10,000	12.0	1815	0.006
15,000	12.0	1827	0.007
20,000	11.8	1814	0.006
25,000	11.2	1759	0.005
30,000	10.8	1708	0.004
35,000	10.2	1651	0.003
40,000	9.5	1574	0.002

Table 4F

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 | Fuel Type: P8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1738	0.005
1,000	11.3	1740	0.005
5,000	11.8	1792	0.006
10,000	12.0	1801	0.006
15,000	12.0	1809	0.006
20,000	11.8	1801	0.006
25,000	11.2	1745	0.005
30,000	10.8	1684	0.004
35,000	10.2	1620	0.003
40,000	9.5	1546	0.002
45,000	8.8	1472	0.001

Table 4G

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-2 Fuel Type: P8DRB265H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.5	1755	0.005
1,000	11.6	1762	0.005
5,000	11.9	1795	0.006
10,000	12.1	1812	0.006
15,000	12.1	1821	0.007
20,000	11.9	1813	0.007
25,000	11.3	1755	0.005
30,000	10.7	1687	0.004
35,000	10.2	1623	0.003
40,000	9.6	1551	0.002

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 BF-2 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 Core Power in REFLOOD - The core power in REFLOOD was corrected to 102% of rated power.

   d. Corrected guide tube thermal resistance.

   e. 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 assumption 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. <u>Input Change</u>

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

   b. Core Power - The core power in REFLOOD has been corrected to 102% of rated.

2. <u>Model Change</u>

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

   b. Suction Line Friction in Discharge Valve Closure Assumption - Took credit for friction due to irreversible losses in the suction line.

   c. Modify Recirculation Discharge Valve Closure Assumption - Assume the valve does not close for a discharge break.

Note: This analysis takes credit for flow through holes in fuel lower tieplate. Since the previous analysis, alternate flow paths have been incorporated in fuel lower tieplates.

6. CONCLUSIONS

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. This is the case for a break occurring in either the suction or discharge piping. For a break in the discharge piping, this failure results in no LPCI flow, and for a suction line failure, LPCI flow is minimized.

Comparison of the calculated PCT's for the maximum size break in the suction and discharge piping determines which is the DBA and which is the second most limiting location. For BF-2, the suction break is the most limiting location (Table 3) for 7x7 fuel. 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. The break that results in the longest period during which the hot node remains uncovered usually 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).

6.1 RESULTS FOR 7x7 FUEL

Figures 6a and 6b show the variation with discharge and suction break size of the calculated time the hot node remains uncovered for Browns Ferry Unit 2. The shape of the discharge break curve is very similar to the lead plant, showing a maximum hot node uncovered time at a break less than the DBA (i.e., 66% DBA). However, owing to the geometric differences of Browns Ferry Unit 2 relative to the lead plant (larger peripheral bypass area and smaller vessel volume to break area ratio), the reflooding times for all discharge break sizes from 1.0 ft^2 to the DBA are considerably lower than that of the lead plant. This resulted in the suction break being more limiting than any of the discharge breaks.

The determination of the suction DBA being the most limiting break was based on the reasoning discussed above and the procedure used for the lead plant. From Figure 6a the 66% discharge DBA was determined to be the most limiting discharge break smaller than the discharge DBA. Also, from Figure 6b the suction DBA was determined to be the most limiting suction break. A CHASTE calculation was performed to compare the PCT for the 66% discharge DBA, the discharge DBA, and the suction DBA.* The suction DBA was determined to result in the highest PCT of the three break sizes and, hence, was determined to be the most limiting break. The results for 7x7 fuel are presented in Tables 4A and 4B.

*The CHASTE calculation for the discharge DBA used the discharge DBA LAMB/SCAT and SAFE/REFLOOD results. The CHASTE calculation for the suction DBA used the suction DBA LAMB/SCAT and SAFE/REFLOOD results. In accordance with the conservative approach used for the lead plant, the 80% discharge DBA LAMB/SCAT results were used with the 66% discharge DBA SAFE/REFLOOD results to determine the 66% discharge DBA results.

The DBA (the complete severence of the recirculation suction piping) results are shown on Figures 1a through 5a. These results have changed very little from the previous analysis because with LPCI flow available the flood time is relatively insensitive to parameters that affect CCFL.

The second most limiting location for the LOCA is the recirculation discharge line. The results of the maximum break in this piping are shown on Figures 1c through 5c. These results have changed very little from the previous analysis because the decrease in flow through the bundles, due to increased CCFL resulting from the new pressure assumption, is offset by the flow through the lower tie plate holes.

The single failure evaluation showing the remaining ECCS following an assumed failure and the effects of a single failure or operator error that cause 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.

6.2 RESULTS FOR 8x8 FUEL

For 8x8 fuel bundles, the worst break is the 66% discharge break rather than the 100% DBA suction break, which is the worst for 7x7 fuel. This difference is due to the higher stored energy of the larger fuel rods in the 7x7 fuel. The PCT of the 7x7 fuel is more sensitive to early boiling transition, while the PCT of the 8x8 fuel is more sensitive to the total uncovered time. Since the core configuration has changed to all 8x8 fuel (8x8, 8x8R, and P8x8R) the ECCS analysis will be virtually identical to that of the Browns Ferry 3 analysis reported in Reference 13. Although large PCT margin exists, some adjustments in MAPLHGRs were made to be consistent with fuel thermal-mechanical heat flux limits. Analyses were performed for all of the 8x8 fuel types for the 66% DBA discharge break, and the results are presented in Tables 4C through 4G.

6.3 APPLICABILITY

1. Single-Loop Operation

   This analysis is valid for operation with one recirculation loop out-of-service with the following MAPLHGR reduction factors applied:

Fuel Type	MAPLHGR Multiplier
7x7	0.70
8x8	0.83
8x8R/P8x8R	0.82

2. Safety Relief Valve Out-of-Service

   This analysis is valid for operation with two safety relief valves out-of-service.

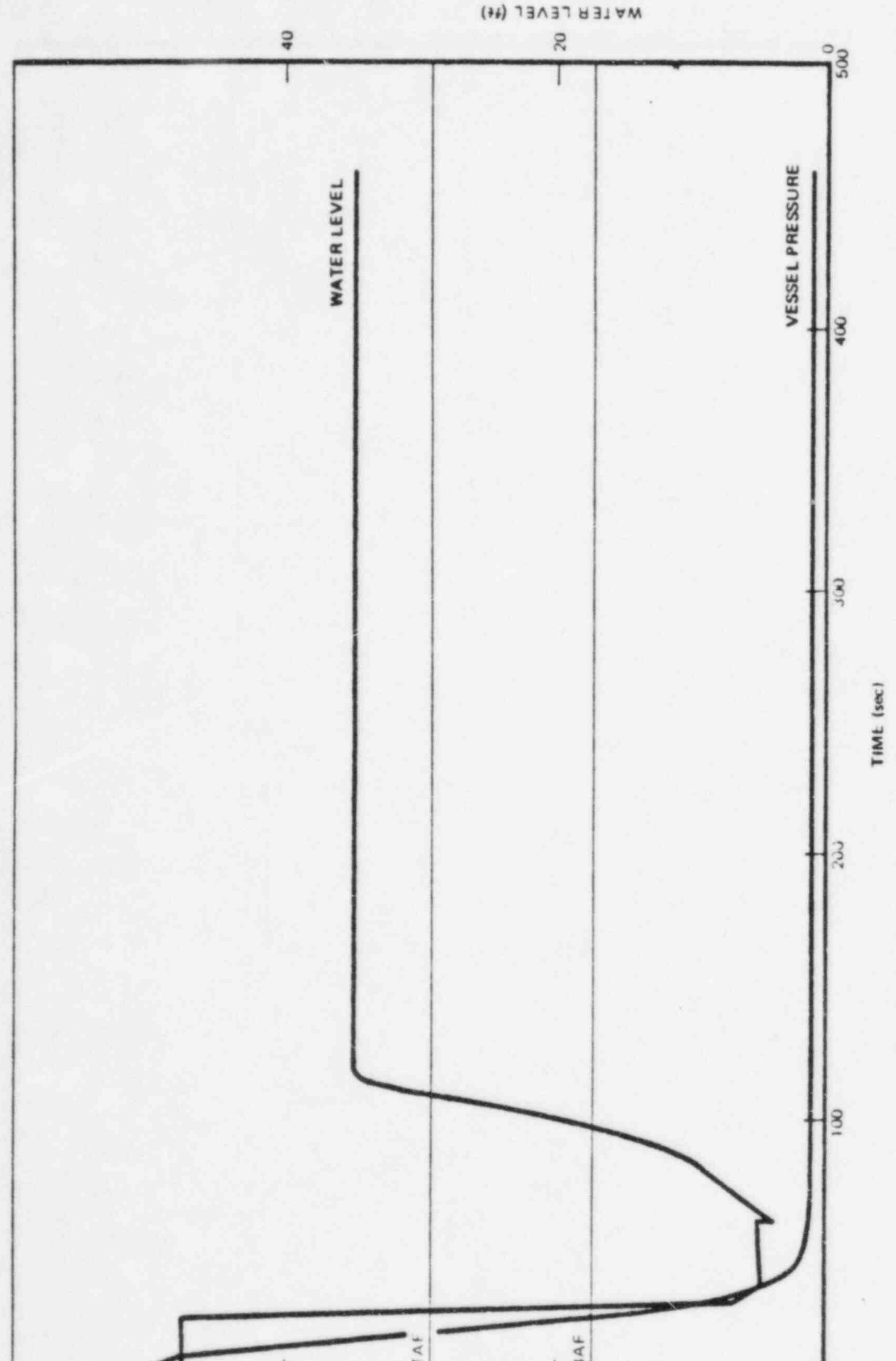

Figure 1a. Water Level Inside the Shroud and Reactor Vessel Pressure Following Maximum Recirculation Line Suction Break, LPCI Injection Valve Failure Break Area = 4.2 ft^2 (DBA)

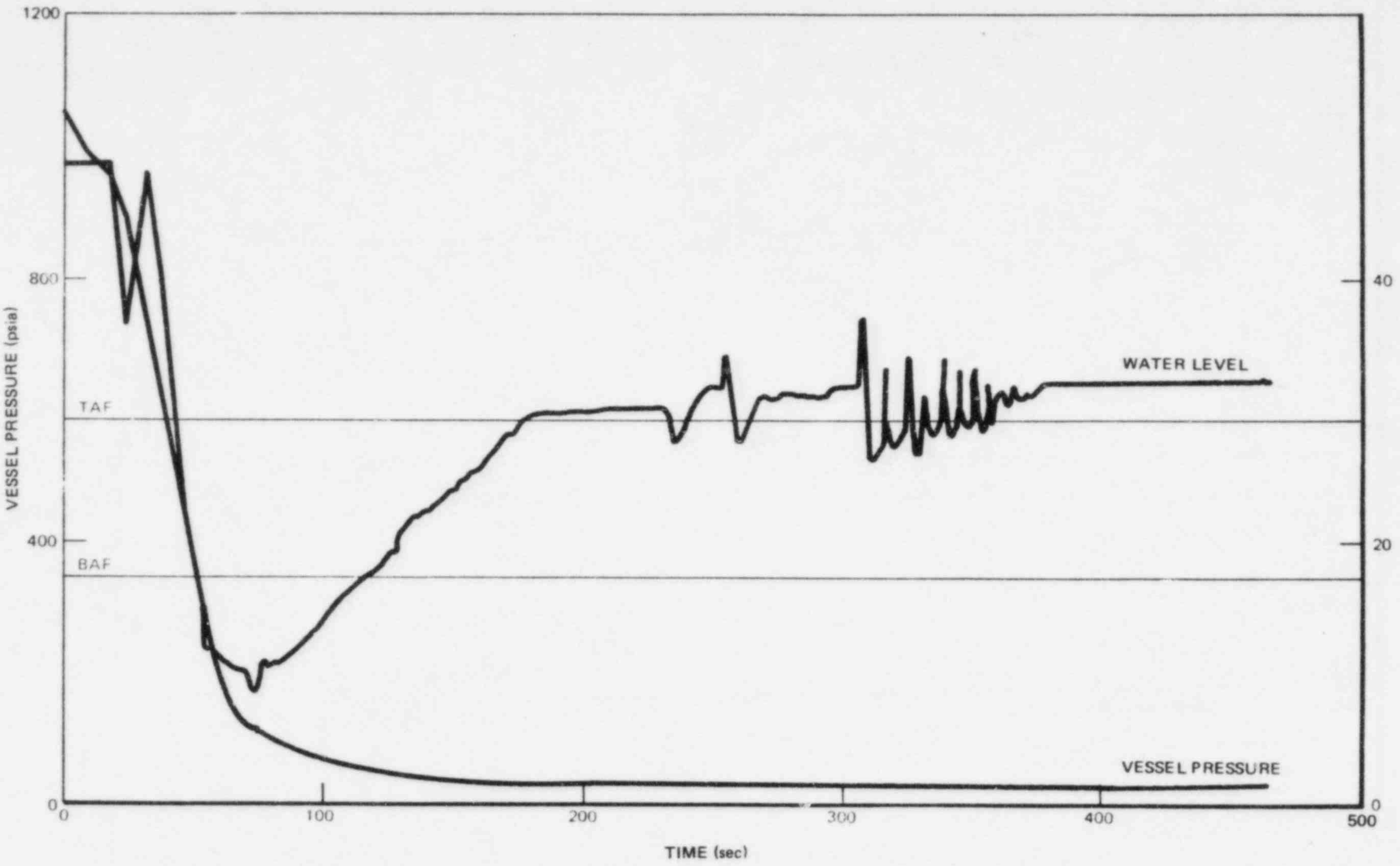

Figure 1b. Water Level Inside the Shroud, and Reactor Vessel Pressure Following a Maximum Recirculation Line Discharge Break. LPCI Injection Valve Failure, Break Area = 1.9 ft^2 (LBM)

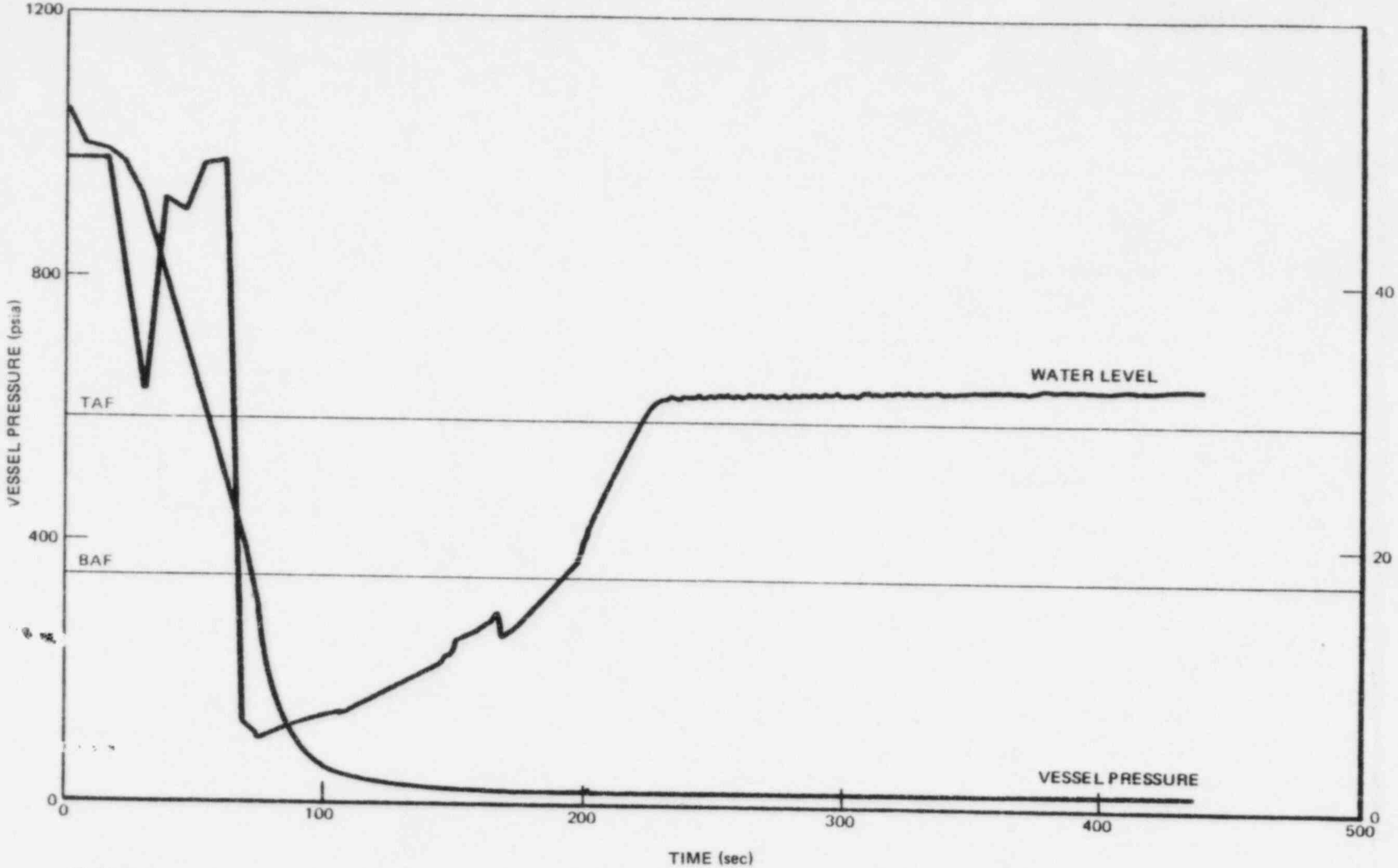

Figure 1c. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Recirculation Line Discharge Break. LPCI Injection Valve Failure, Break Area = 1.3 ft^2 (66% DBA) (LBM)

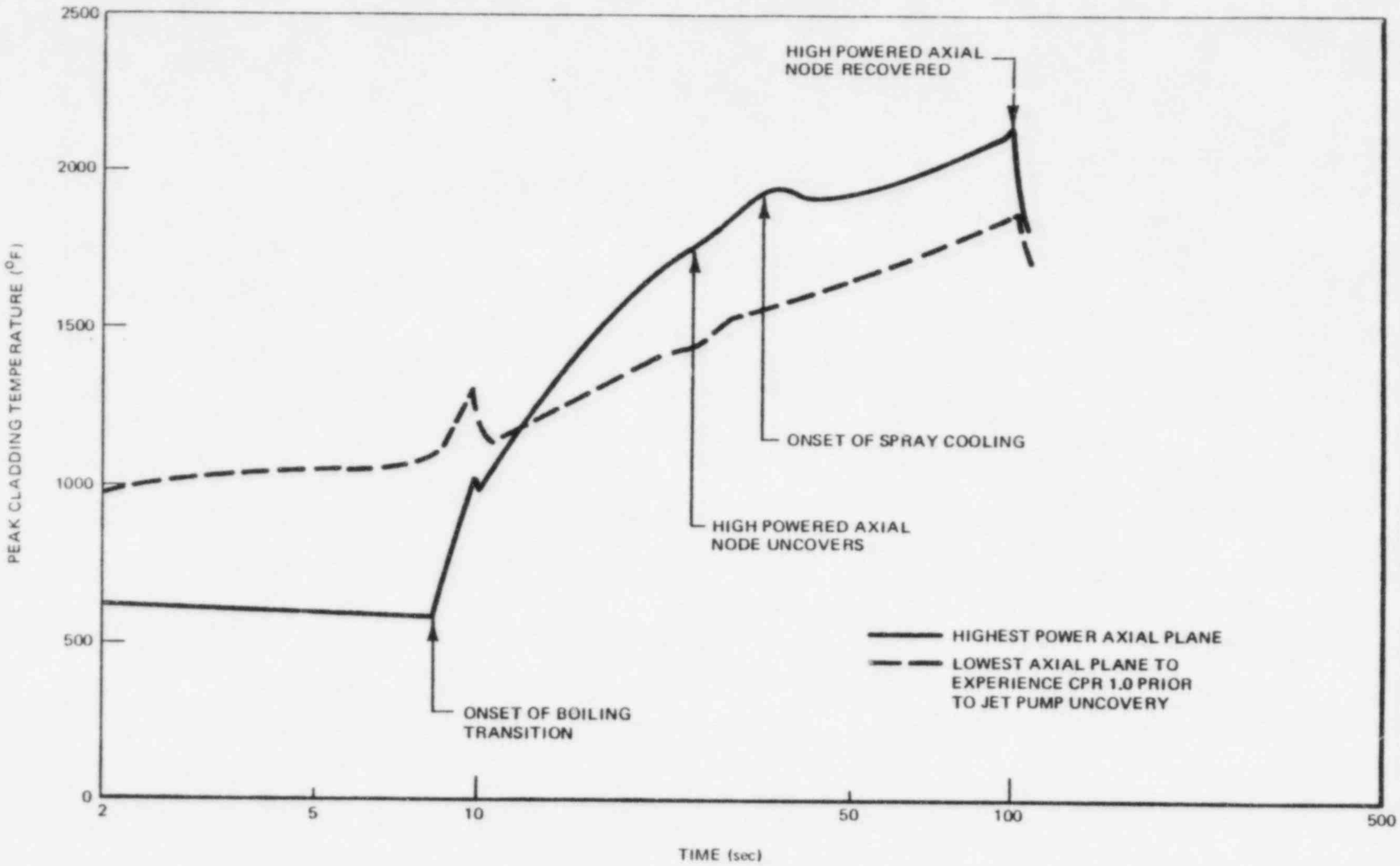

Figure 2a. Peak Cladding Temperature Following a Maximum Recirculation Line Suction Break, LPCI Injection Valve Failure, Break Area = 4.2 ft^2 (LBM)

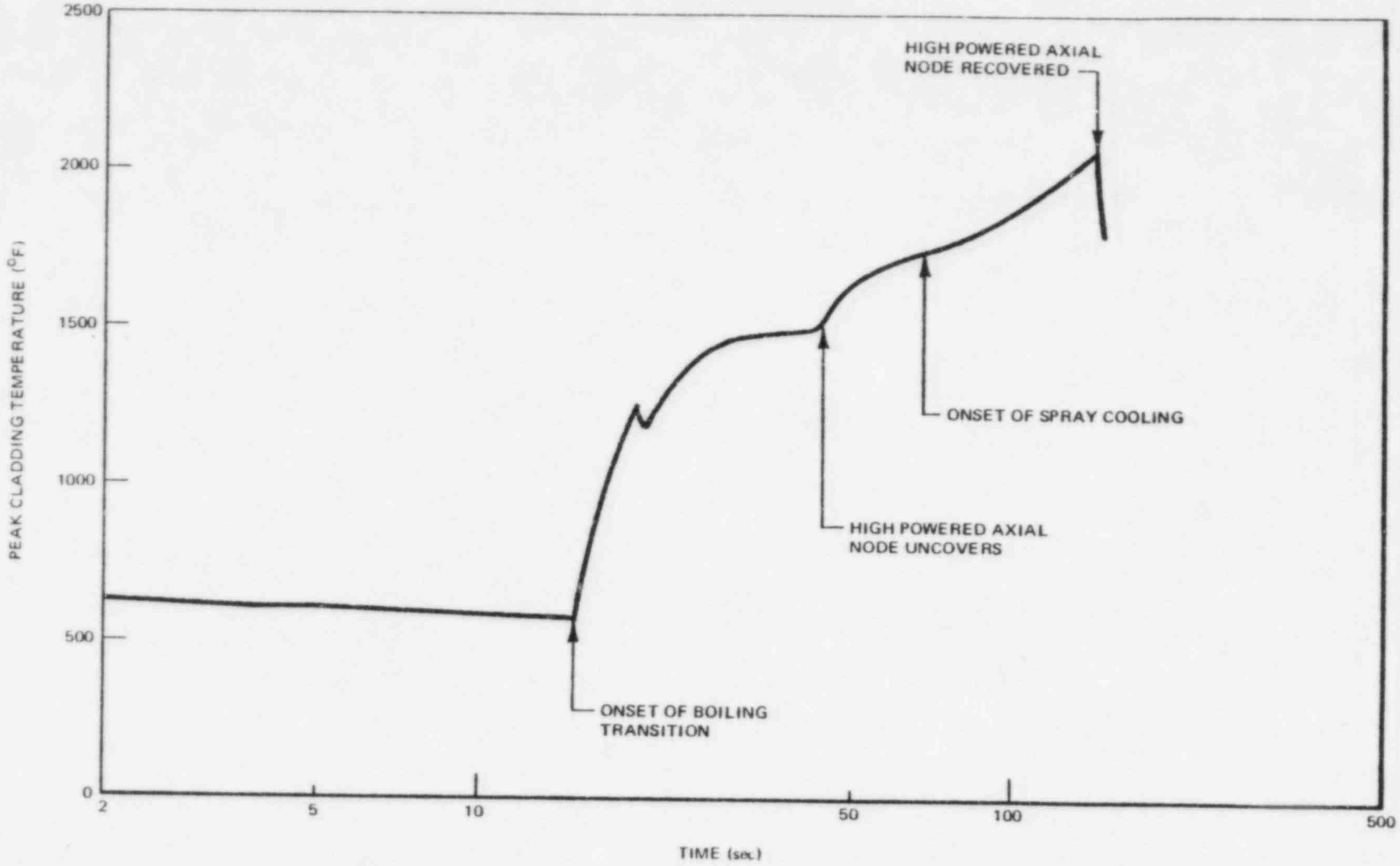

Figure 2b. Peak Cladding Temperature Following a Maximum Recirculation Line Discharge Break, LPCI Injection Valve Failure, Break Area = 1.9 ft^2 (LBM)

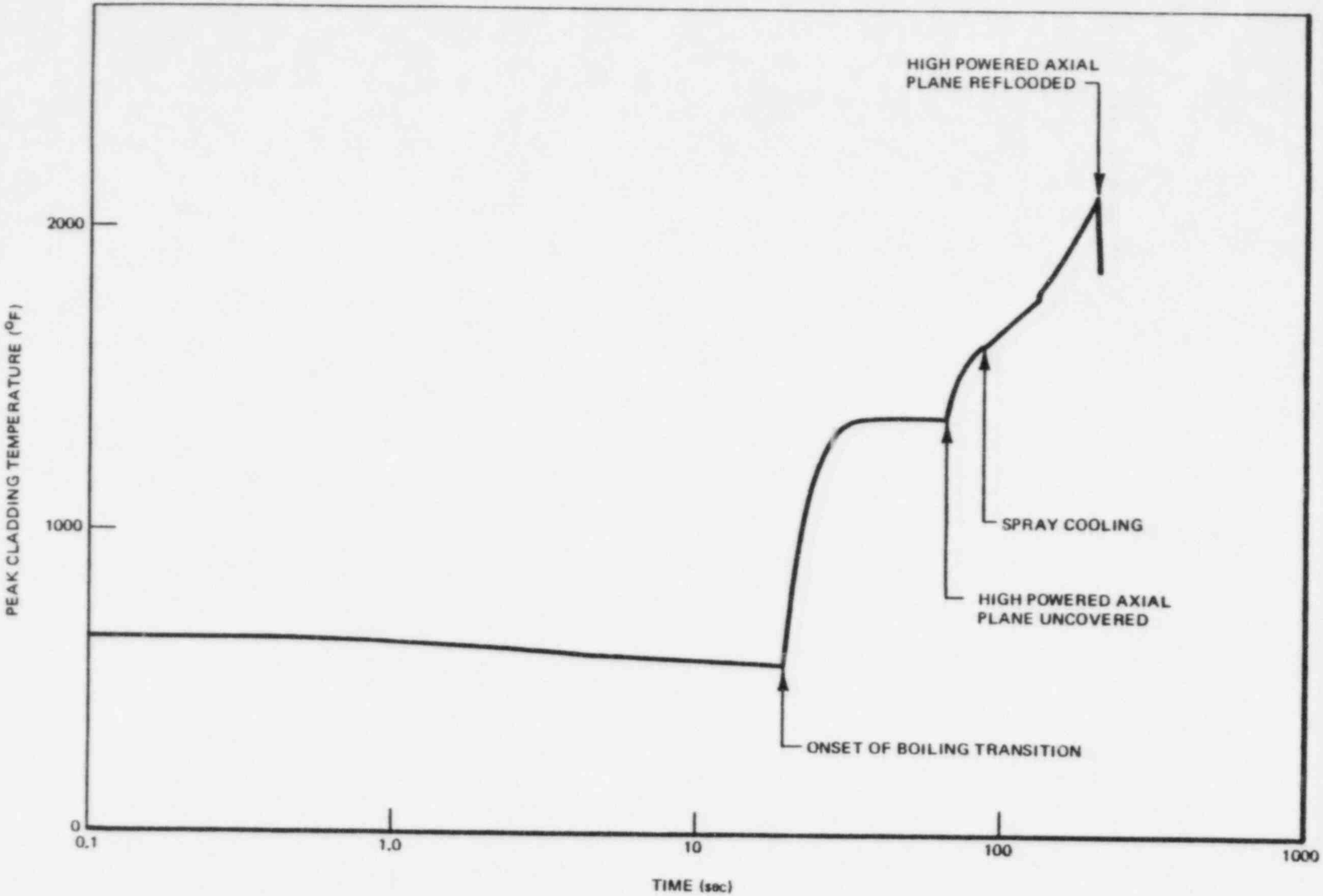

Figure 2c. Peak Cladding Temperature Following a Recirculation Line Discharge Break, LPCI Injection Valve Failure Break Area = 1.3 ft^2 (66% DBA) (LBM)

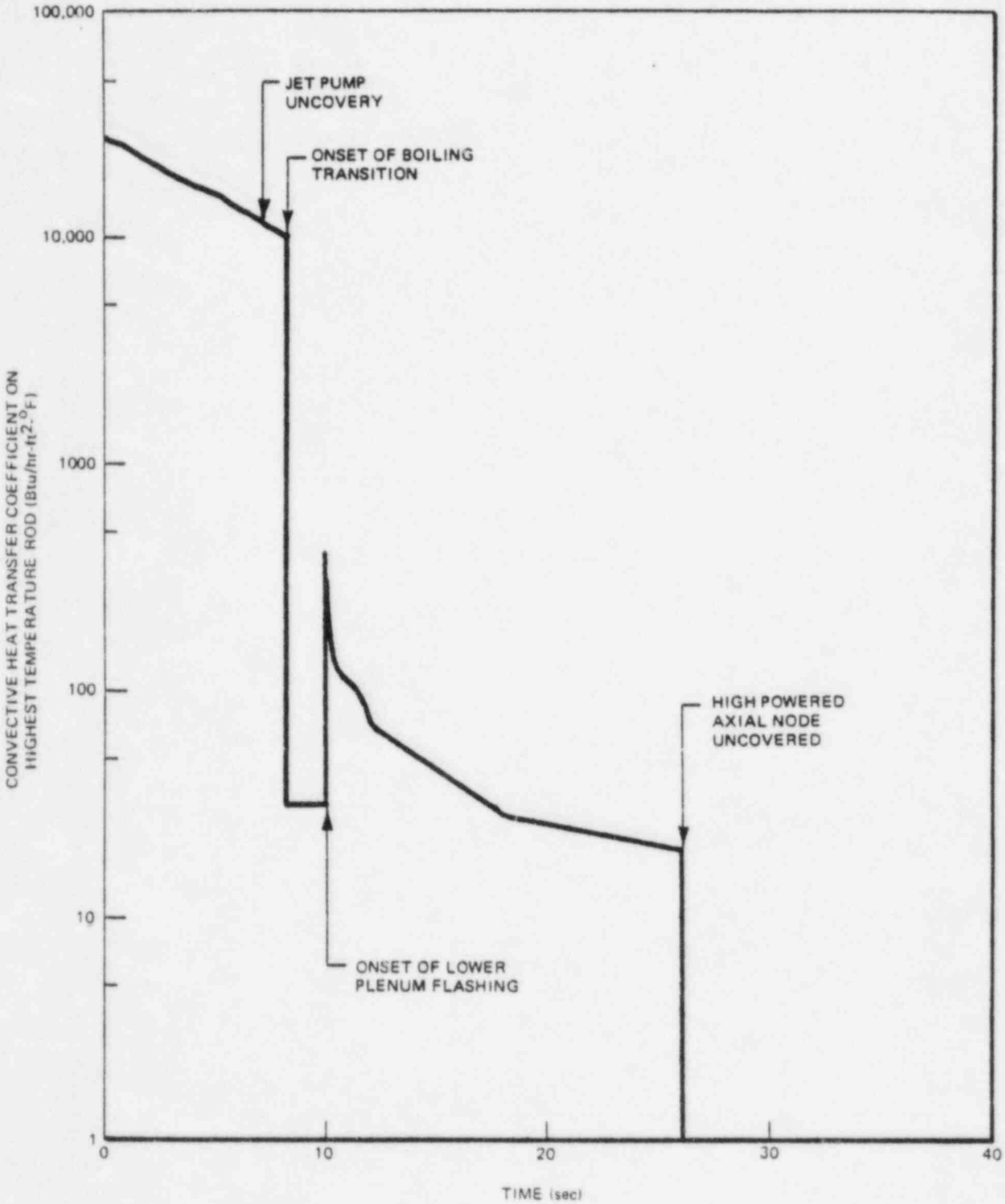

Figure 3a. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node Following a Maximum Recirculation Line Suction Break, LPCI Injection Valve Failure, Break Area = 4.2 ft^2

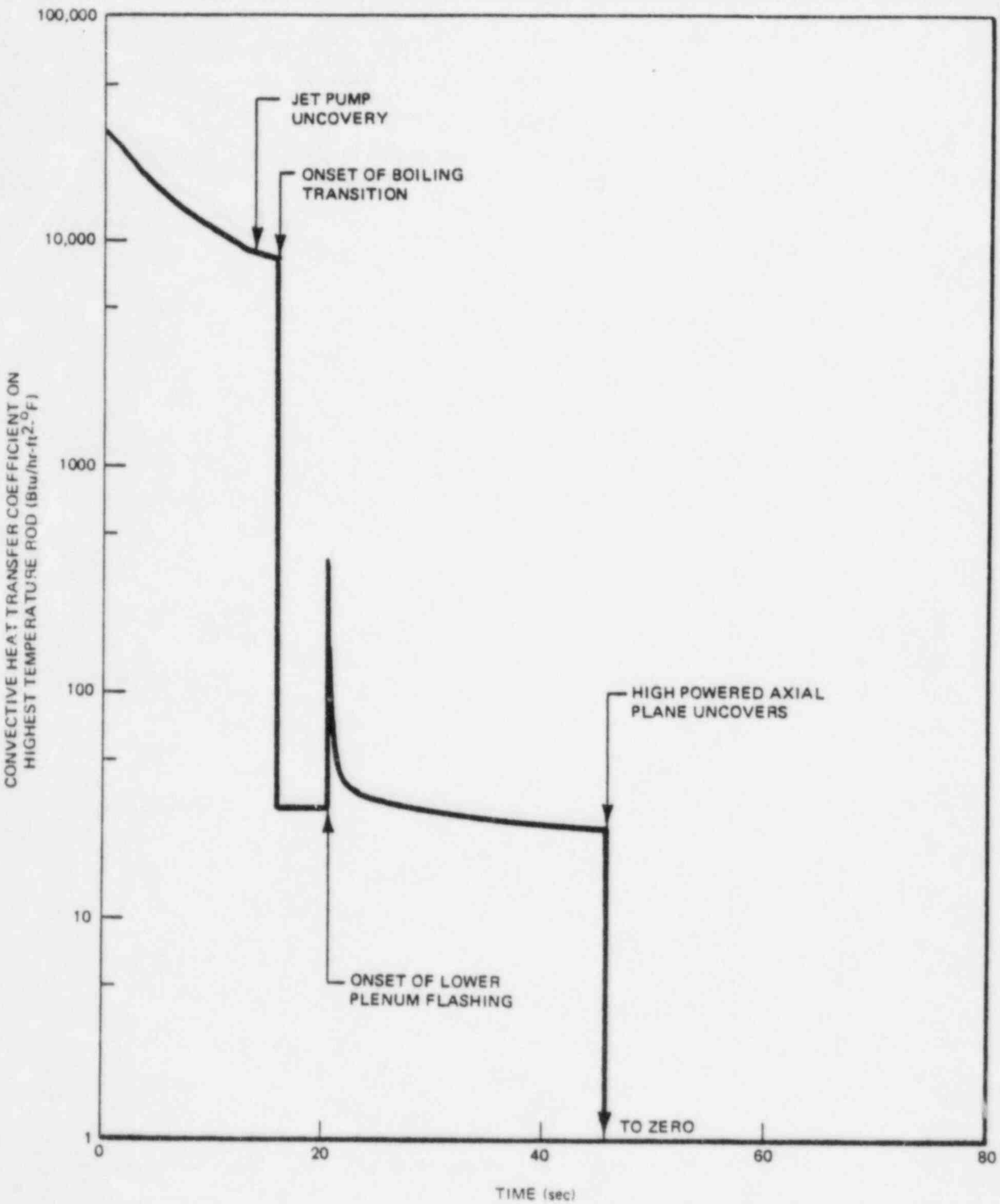

Figure 3b. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node Following a Maximum Recirculation Line Discharge Break, LPCI Injection Valve Failure, Break Area = 1.9 ft^2

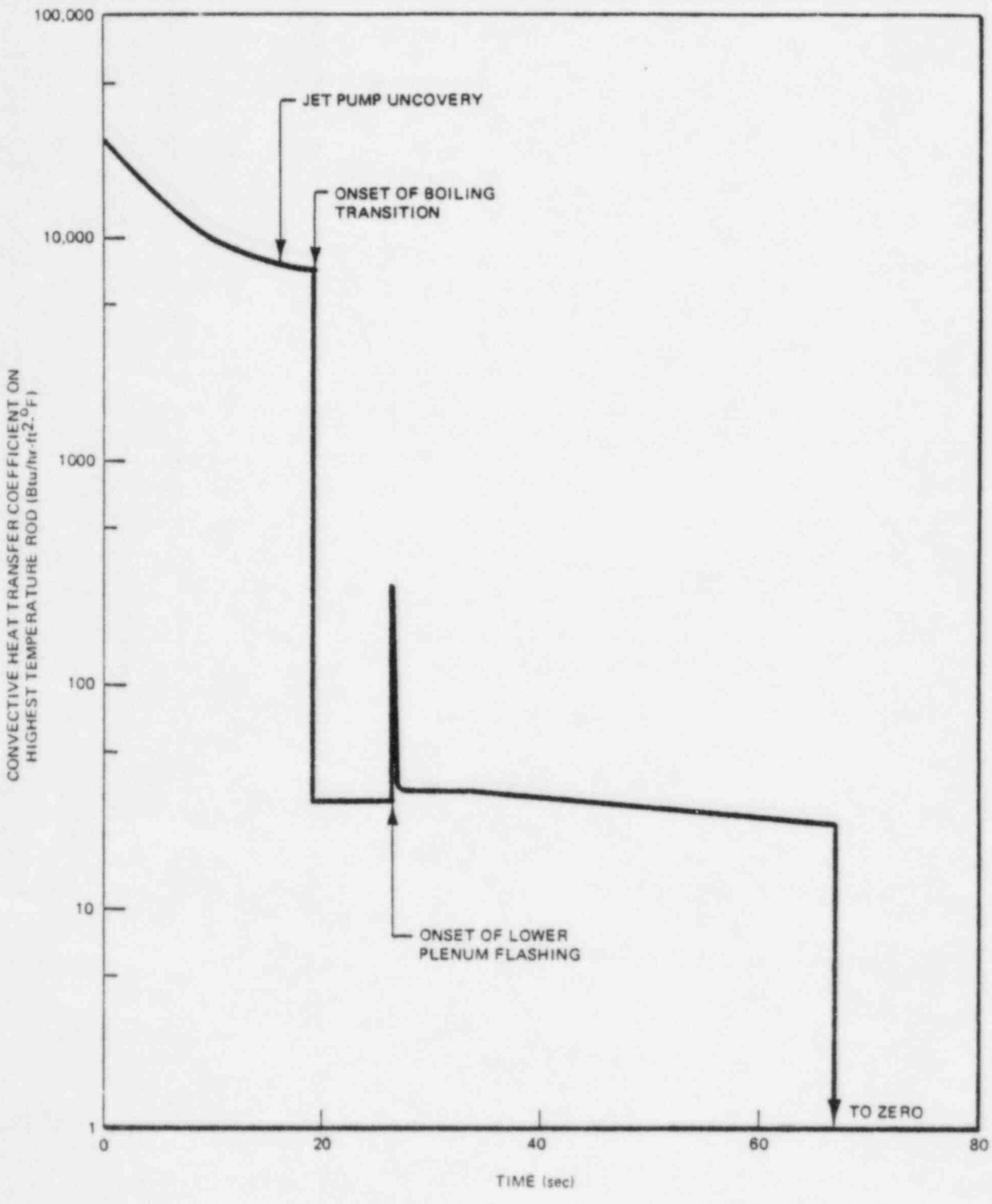

Figure 3c. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node Following a Recirculation Line Discharge Break, LPCI Injection Valve Failure, Break Area = 1.3 ft^2 (66% DBA)

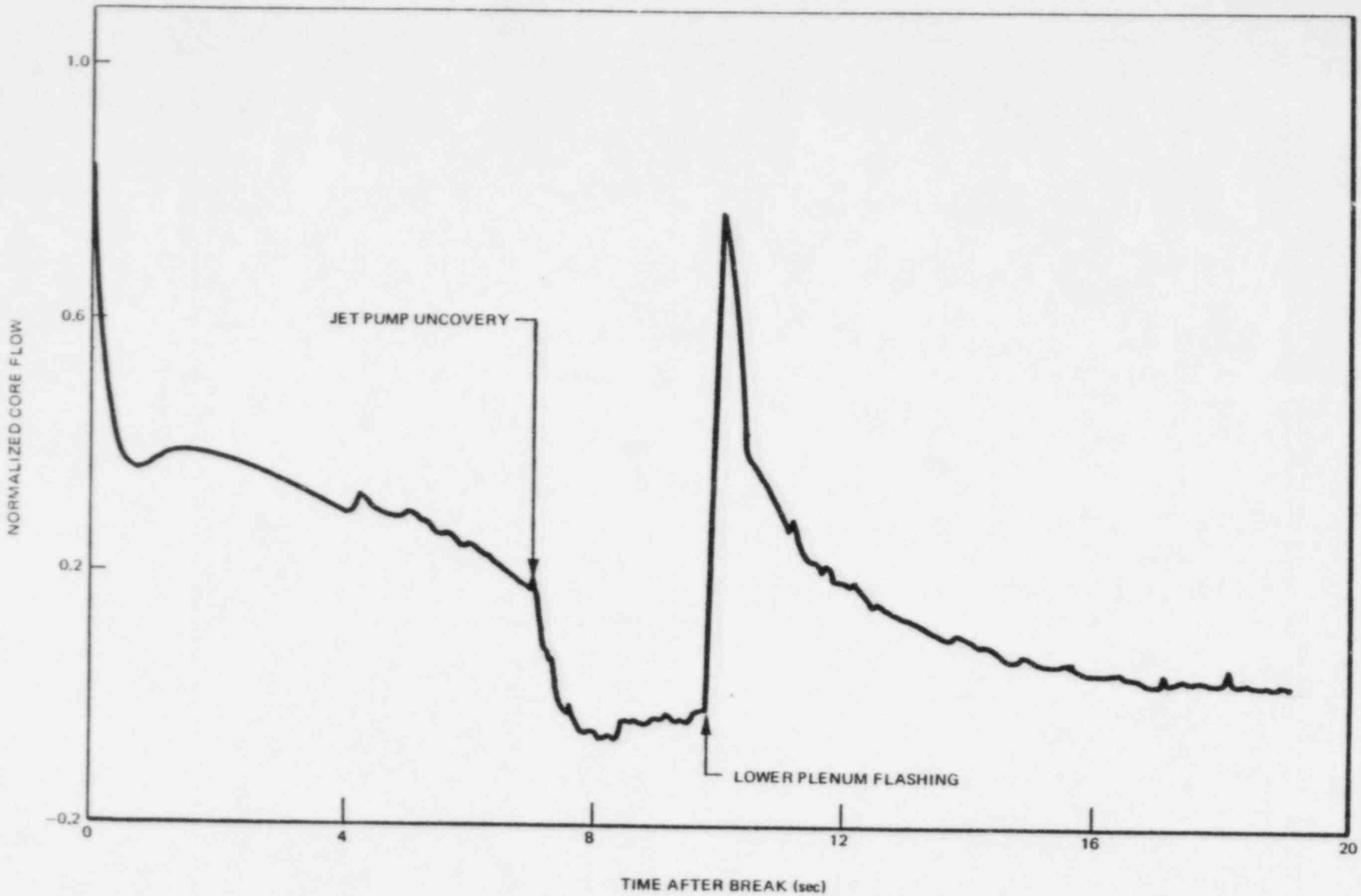

Figure 4a. Normalized Core Average Inlet Flow Following a Maximum Recirculation Line Suction Break, Break Area = 4.2 ft^2

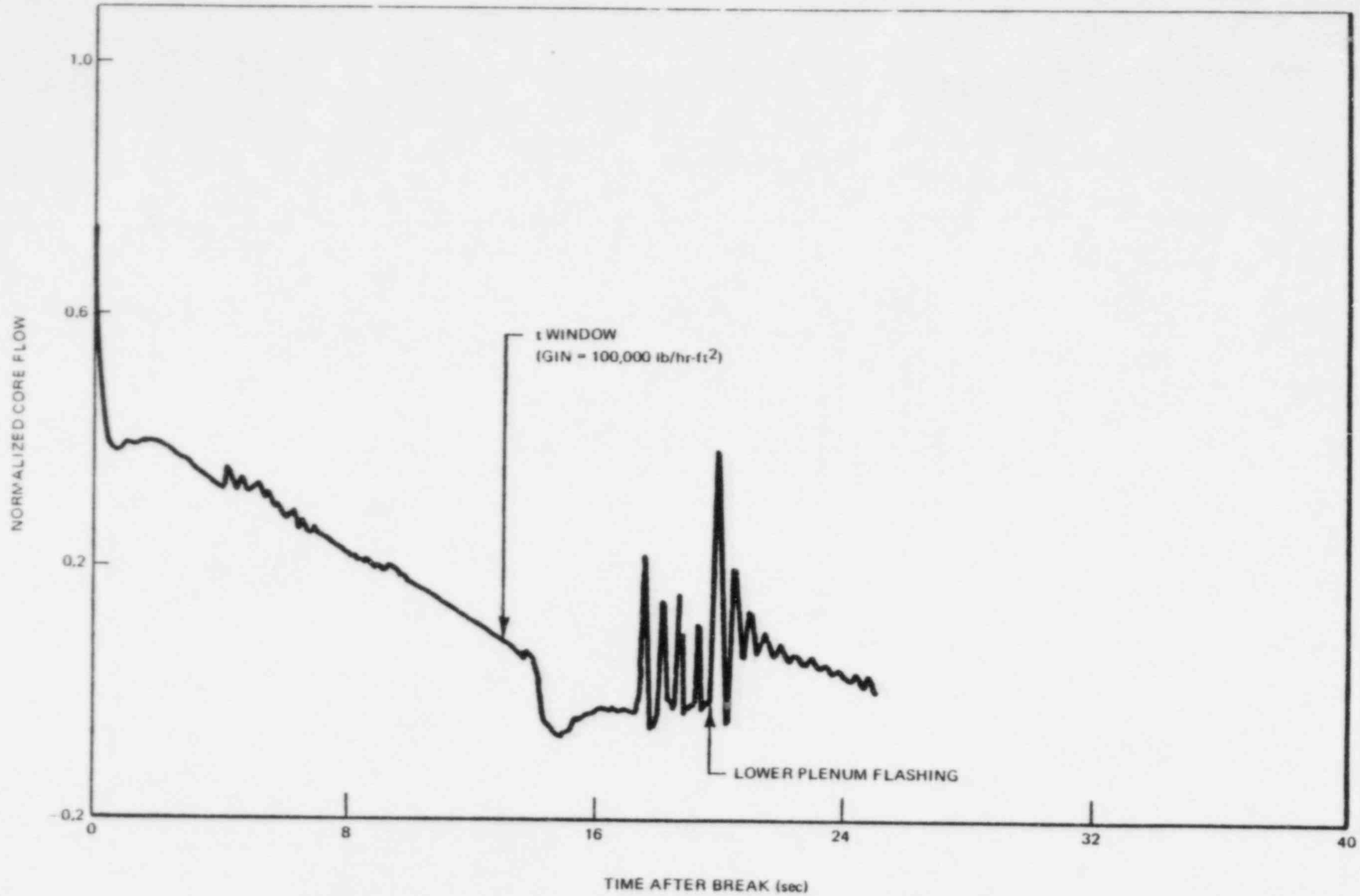

Figure 4b. Normalized Core Average Inlet Flow Following a Maximum Recirculation Line Discharge Break, Break Area = 1.9 ft^2

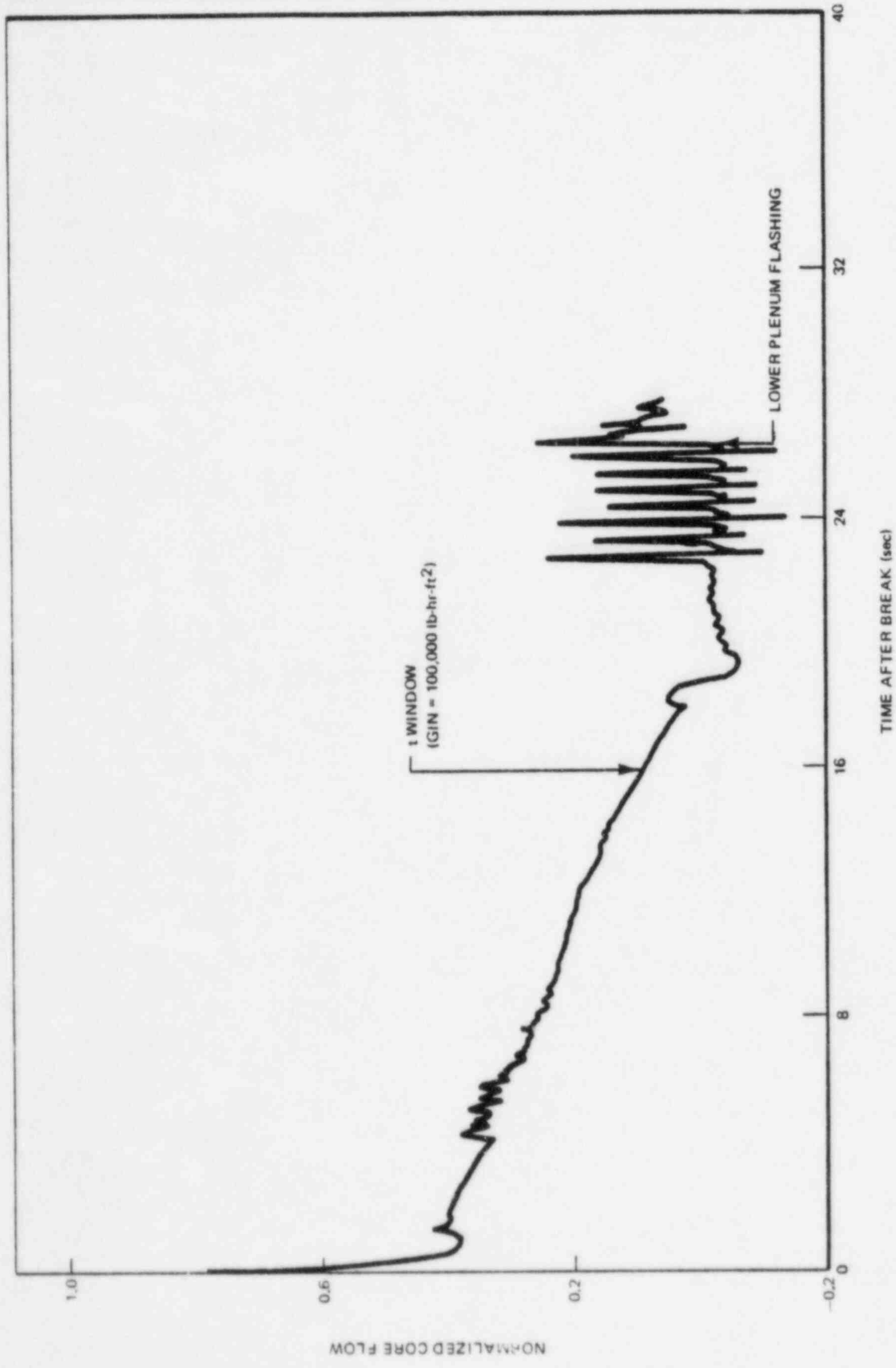

Figure 4c. Normalized Core Average Inlet Flow Following a Recirculation Line Discharge Break (80% DBA)

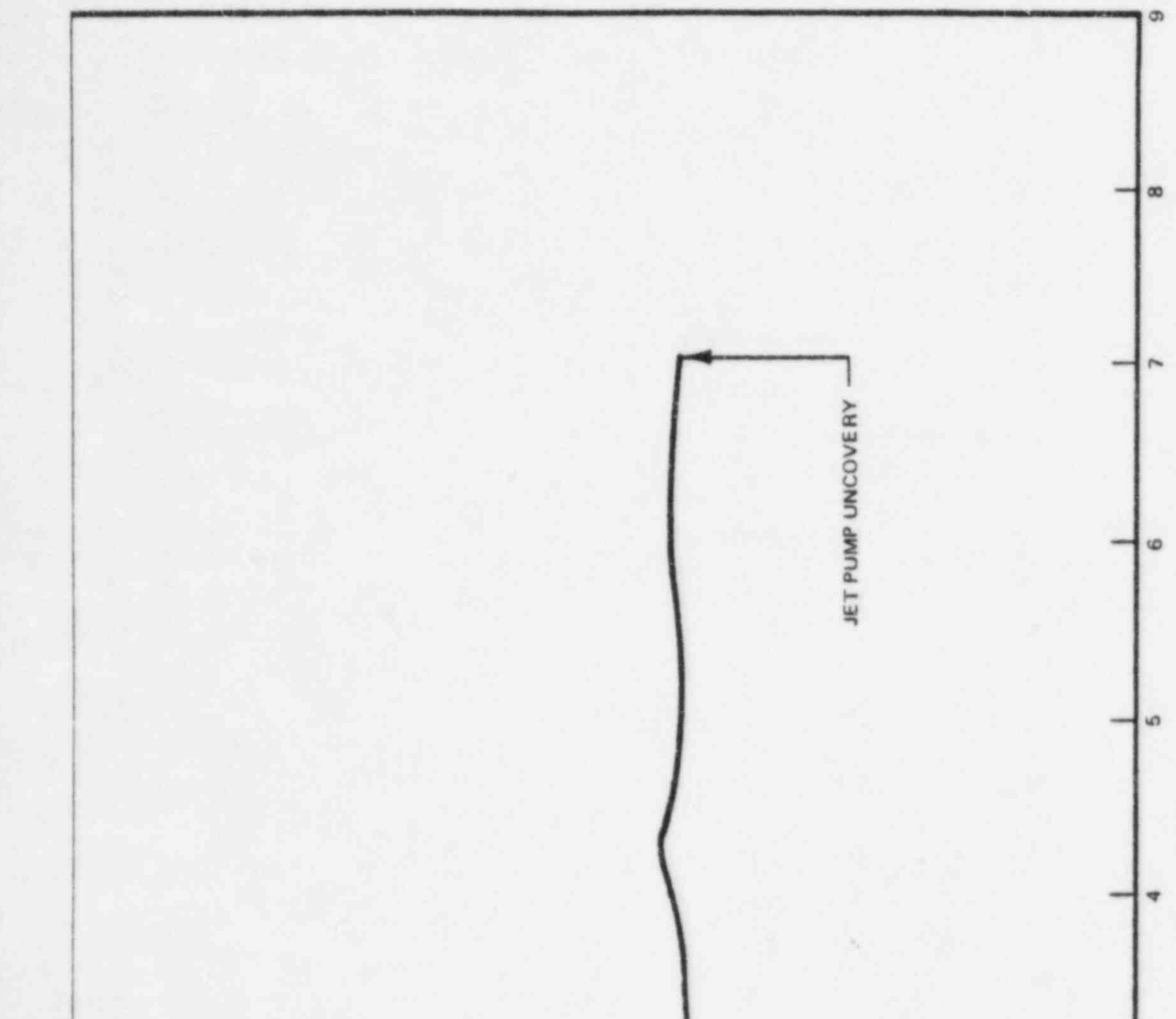

Figure 5a. Minimum Critical Power Ratio Following a Maximum Recirculation Line Suction Break, Break Area = 4.2 ft^2

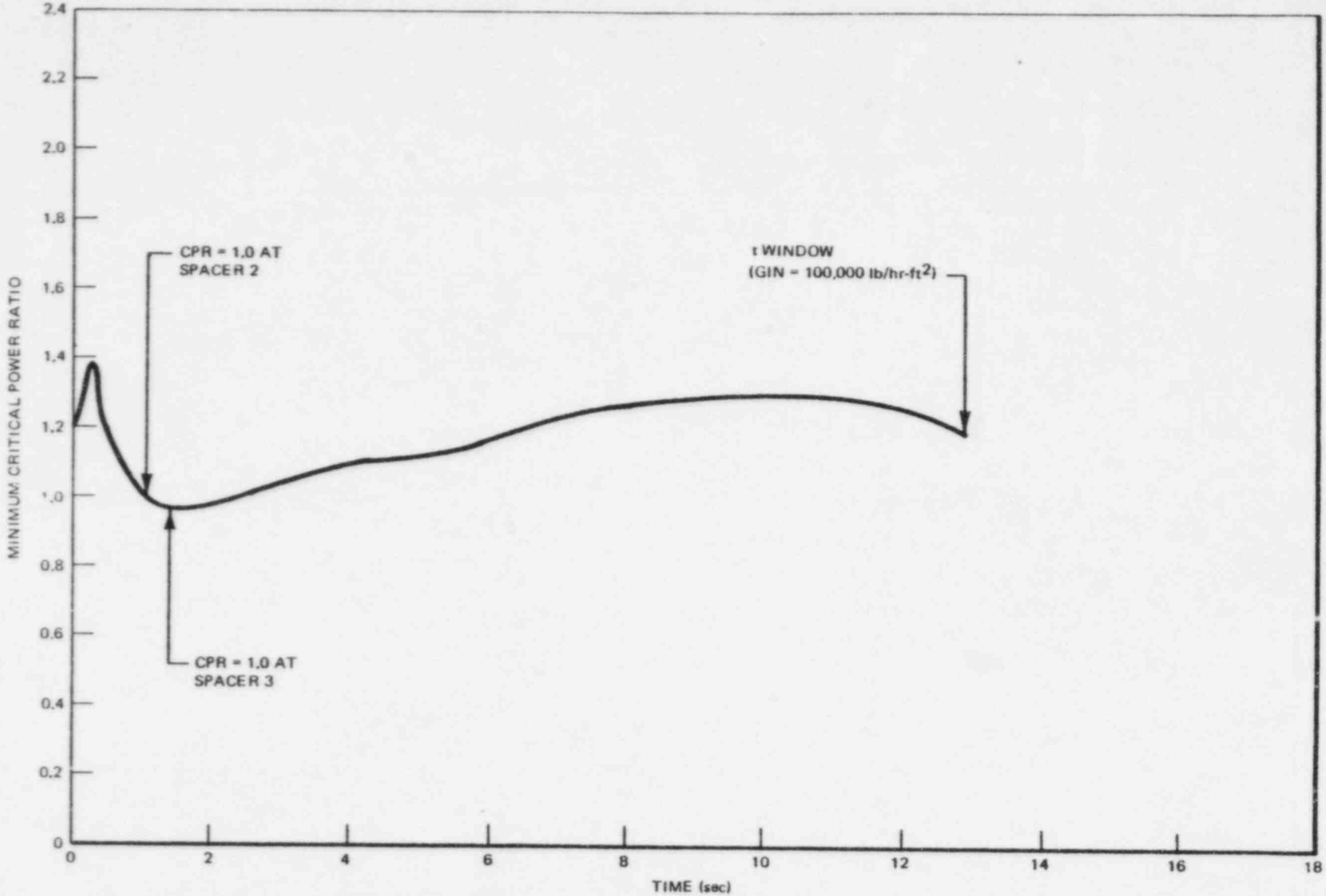

Figure 5b. Minimum Critical Power Ratio Following a Maximum Recirculation Line Discharge Break, Break Area = 1.9 ft^2

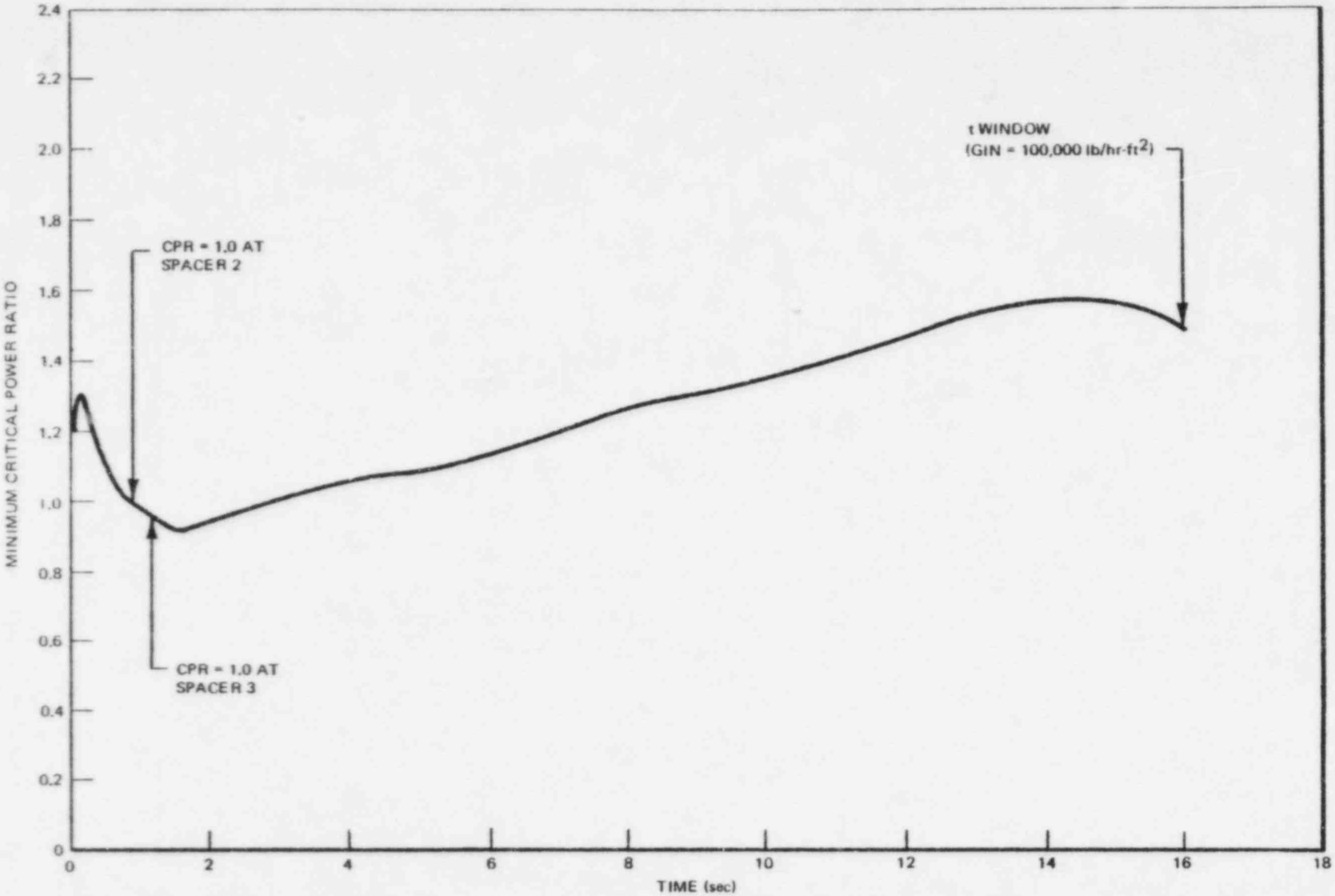

Figure 5c. Minimum Critical Power Ratio Following a Recirculation Line Discharge Break (80% DBA)

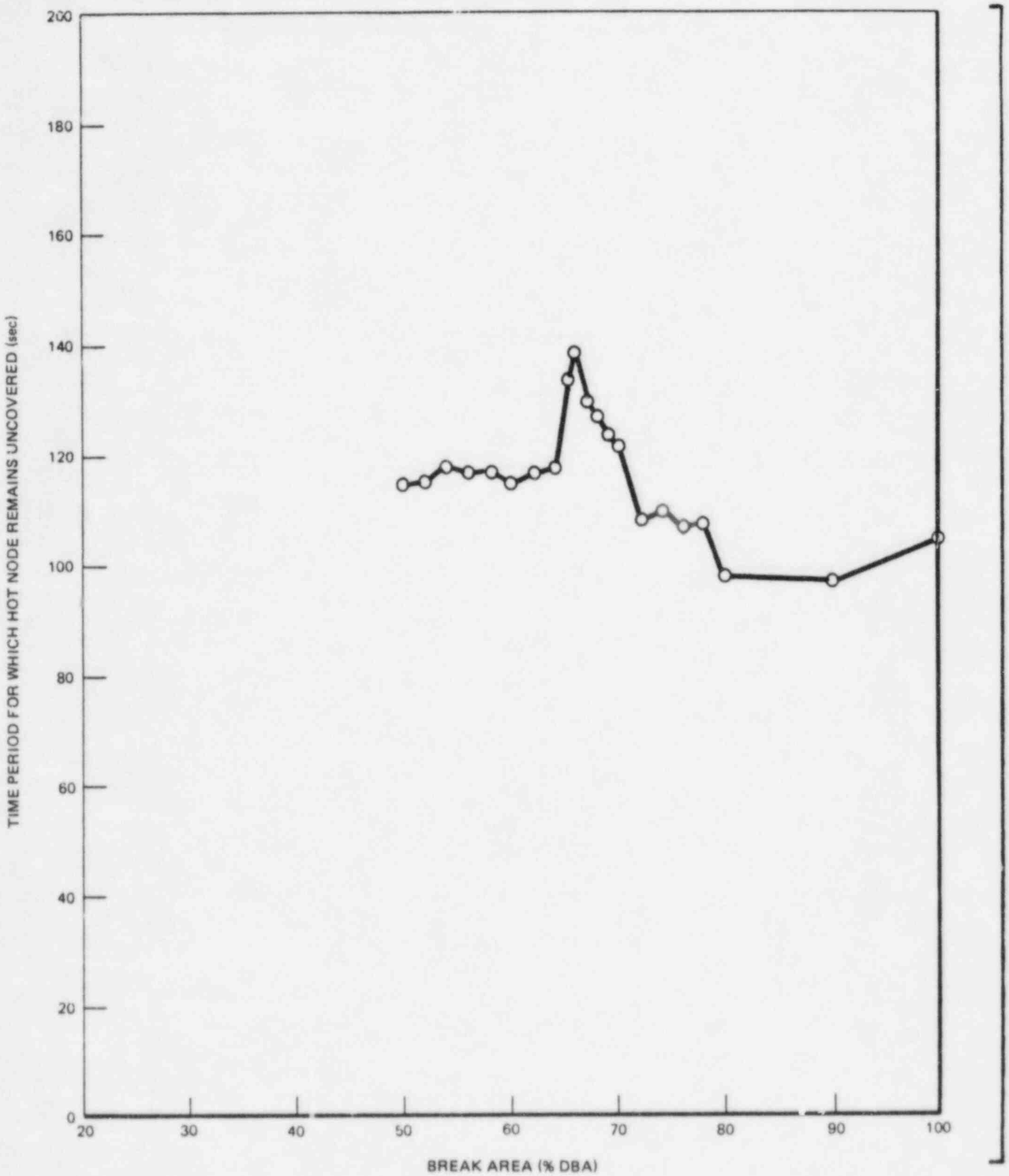

Figure 6a. Variation With Break Area of Time for Which Hot Node Remains Uncovered (Discharge)

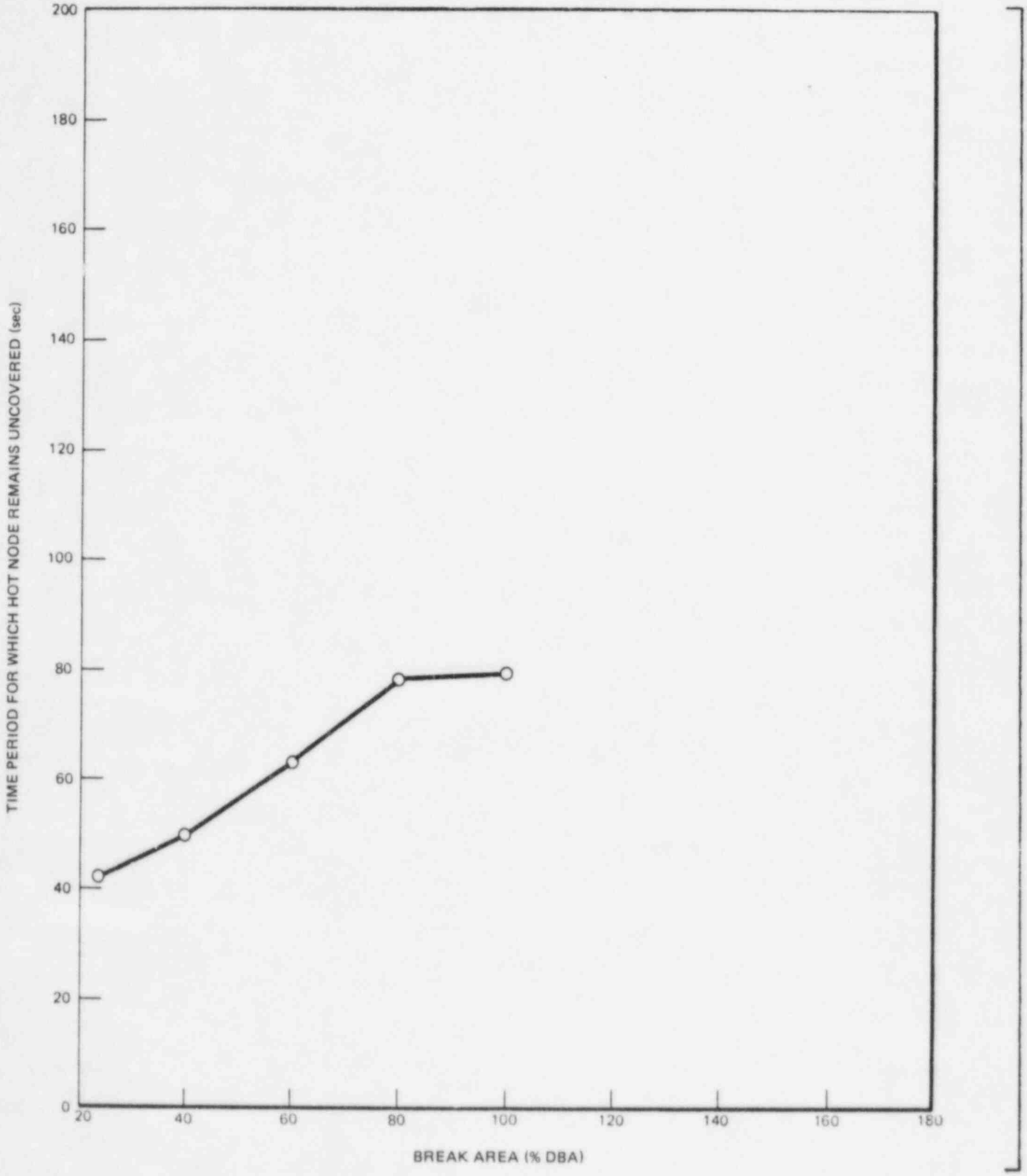

Figure 6b. Variation with Break Area of Time for Which Hot Node Remains Uncovered (Suction)

7. REFERENCES

1. Letter, A. Schwencer (NRC) to Godwin Williams, Jr (TVA), "Re: Browns Ferry Nuclear Plant, Units Nos. 1 and 2," dated March 11, 1977.

2. Letter, Darrel 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 10CRF50 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 December 20, 1974.

4. "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," letter from K. R. Goller (NRC) to G. G. Sherwood (GE), dated April 12, 1977.

5. Letter, A. J. Levine (GE) to D. F. Ross (NRC) dated January 27, 1977, "General Electric (GE) Loss of Coolant Accident (LOCA) Analysis Model Revisions - Core Heatup Code CHASTE05."

6. Letter, A. J. Levine (GE) to D. B. Vassallo (NRC), dated March 14, 1977, "Request for Approval for Use of Loss of Coolant Accident (LOCA) Evaluations Model Code REFLOOD05."

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

8. "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations," Supplement 2, NEDE-21156-2, January 1977.

9. Letter, R. Engel (GE) to V. Stello (NRC), "Answers to NRC Questions on NEDE-21156-2," January 24, 1977.

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

11. Letter, George T. Berry (PASNY) to Robert W. Reid (NRC), "James A. FitzPartick Nuclear Power Plant ECCS Analysis Docket No. 50-333," dated July 29, 1977.

12. "Emergency Core Cooling System Analysis Appendix K Requirements," NEDO-20973, dated August, 1975.

13. "LOCA Analysis for Browns Ferry Nuclear Plant Unit 3," NEDO-24194A, July 1979.

GENERAL ELECTRIC