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

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ERRATA And ADDENDA SHEET

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NOTE: Correct all copies of the applicable publication as specified below.

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
01	Pages viii, viii-a and x	Replace with new pages viii, viii-a and x.
02	Page 4-1	Replace with new page 4-1.
03	Pages 5-5, 5-6, 5-7 and 5-8	Replace with new pages 5-5, 5-5a, 5-6, 5-7, 5-8 and 5-8a.
04	Page 5-10	Replace with new page 5-10.
05	Page 5-32	Replace with new page 5-32.
06	Page 5-39	Replace with new pages 5-39 and 5-39a.
07	Pages 5-40 and 5-41	Replace with new pages 5-40, 5-41, 5-41a and 5-41b.
08	Page 5-42	Replace with new page 5-42.
09	Page 5-51	Replace with new pages 5-51 and 5-51a.
10	Page 5-58	Replace with new pages 5-58 and 5-58a.
11	Page 5-59	Replace with new page 5-59.
12	Pages 5-59a and 5-59b	Insert new pages 5-59a and 5-59b.
13	Pages 5-75a, 5-75b and 5-75c	Replace with new pages 5-75a, 5-75b and 5-75c.
14	Page 5-75d	Insert new page 5-75d.
15	Pages 5-76a, 5-76b and 5-76c	Replace with new pages 5-76a.1, 5-76a.2, 5-76b and 5-76c.
16	Page 5-76d	Insert new page 5-76d.
17	Page 5-76e	Insert new page 5-76e.
18	Pages 5-77a, 5-77b and 5-77c	Replace with new pages 5-77a, 5-77b and 5-77c.

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ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
19	Page 5-78	Replace with new page 5-78.
20	Pages 5-78a and 5-78b	Insert new pages 5-78a and 5-78b.
21	Page 5-79	Replace with new page 5-79.
22	Page 5-79a	Insert new page 5-79a.
23	Page 5-80	Replace with new pages 5-80, 5-81 and 5-82.
24	Pages A-i/A-ii, A-2, A-3, A-7, A-9, and A-11	Replace with new pages A-i/A-ii, A-2, A-3, A-3a, A-7, A-9 and A-11.
25	Appendix B	Insert new Appendix B.

LIST OF ILLUSTRATIONS (Continued)

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LIST OF ILLUSTRATIONS (Continued)

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LIST OF TABLES (Continued)

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4. STEADY-STATE HYDRAULIC MODELS

Core steady-state thermal-hydraulic analyses are performed using a model of the reactor core. This model includes hydraulic descriptions of orifices, lower tieplates, fuel rods, fuel rod spacers, upper tieplates, the fuel channel and core bypass flow paths. The orifice, lower tieplate, fuel rod spacers, upper tieplate and, where applicable, holes in the lower tieplate are hydraulically represented as being separate, distinct local losses of zero thickness. The fuel channel cross section is represented by a square section with enclosed area equal to the unrodded cross sectional area of the actual fuel channel.

For thermal-hydraulic analysis, General Electric uses one of the fuel assembly designs documented in Reference 4-9 in place of any non-GE assembly present in the core. The General Electric-supplied bundle which has a fuel rod geometry and enrichment most like that for a non-General Electric-supplied bundle is used in place of that non-General Electric-supplied bundle. Chosen replacement bundle designations for the reference cycle are given in the reference cycle supplement. Additional information is presented in Appendix B. 2/83

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors (References 4-1, 4-2, 4-3). The components of bundle pressure drop considered are friction, local, elevation and acceleration (see Subsections 4.1 through 4.4). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass paths documented in Subsection 4.5. The remainder passes through the orifice in the fuel support (experiencing a pressure loss), where more flow is lost through

The rod-by-rod R-factor distributions used for the bounding statistical analysis are summarized in Reference 5-25. The basis for the additive constants used to determine the P8x8R R-factor is documented in Reference 5-2.

Results of the analyses show that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition if the MCPR for P8x8R reloads is 1.07 or greater.

5.2 MCPR OPERATING LIMIT CALCULATIONAL PROCEDURE

A plant-unique MCPR operating limit is established for General Electric-supplied 9/79
fuel to ensure that the fuel cladding integrity safety limit for that fuel is not violated for any moderate frequency transient. This operating requirement is obtained by addition of the absolute, maximum CPR value for the most limiting transient from rated conditions postulated to occur at the plant to the fuel cladding integrity safety limit.

Core-wide rapid pressurization events (Turbine Trip w/o Bypass, Load Rejection w/o Bypass, Feedwater Controller Failure) are analyzed using the system model documented in References 5-27 and 5-28. The ODYN code contains a one-dimensional representation of the reactor core which is coupled to the recirculation and control system model. The integrated model is based on one-dimensional reactor kinetics, multi-noded thermal-hydraulic and heat transfer relationships, and mechanical kinetic equations of the equipment. ODYN contains a refined reactor core description and a detailed steamline model to simulate pressure dynamics during a transient. Improvements made in ODYN are 2/83
documented in Reference 5-29. The improvements consist of a new model used in the calculation of mass flux through the safety/relief valves and the replacement of the coupled five-equation thermal-hydraulics model with a similar model in which the momentum equation is decoupled. The improved version yields the same results as the previous version, within expected numerical uncertainty (Reference 5-30). Further updating of ODYN was performed to correct a minor coding error (Reference 5-35). For the slower core-wide transients, Loss of Feedwater Heating is analyzed using either the steady-state three-dimensional BWR Simulator Code (Reference 5-7) or the REDY Transient model

(References 5-3, 5-4 and 5-5), as described in Reference 5-31. A worse, usually maximum, power condition is assumed with thermally limited fuel conditions. The philosophy with respect to using the equipment performance components of the transient models for design and safety evaluations is to consider the performance of key components at their adverse tolerances. Circuitry delays in the reactor protection system as well as other key equipment circuit delays are assumed at the maximum specified values. The speed of all the control rod drives following a scram is assumed to be at the plant technical specification value. Field data have shown considerable conservatism in this key component performance. The setpoints for the safety/relief valves both in the safety and relief function and for pressure scram are assumed at their specified limits. The assumed setpoints used in the analysis are given in Table 5-3. Other equipment performance such as relief and safety valve opening characteristics, recirculation pump drive train inertia, and main steam line isolation valve closing times are all assumed to be at adverse tolerances.

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End-of-cycle (EOC) conditions for nuclear data are used (except where specific exposure dependent evaluations are performed, Subsection 5.2.2) to provide a varying level of conservatism associated with core exposure aspects. The nuclear data which are re-evaluated for each reload analysis are the scram reactivity function, void reactivity function and Doppler reactivity function. These parameters are calculated within the ODYN code for each cycle, as described in References 5-27 and 5-28. The basis for using EOC Doppler and void reactivity functions in the analyses is outlined in Reference 5-6. A discussion of the above significant parameters follows.

Scram reactivity is the worth of control rods as a function of time or position following the scram signal. The scram reactivity insertion is normally lowest at the EOC (all rods out condition) because there are no stubbed rods to insert negative reactivity more quickly than the remaining blades of the control rod bank. The scram reactivity function is related to dynamic performance when expressed (as plotted) in the form of $\Delta k/k\beta$. While the scram reactivity characteristic is uniquely determined during evaluations of rapid pressurization events, the scram reactivity used for slow transients (Loss of Feedwater Heating) is a typical reload curve (see Reference 5-25). A factor of 0.8 is applied for use in the analyses for added conservatism. This curve is applied generically for slow transients because these transients are insensitive to the rate of negative reactivity insertion during a scram. This insensitivity occurs because the rate of positive reactivity insertion from a reduction in core void fraction is very slow compared to the time required for complete control rod insertion. 2/83

The void reactivity coefficient is an important parameter, not only in transient analysis, but also in core stability. The core average void coefficient must be negative; however, it must not be so negative as to yield such a strong positive reactivity feedback during void collapse events that core and vessel limits are threatened. Conversely, events with void increase must produce sufficient negative feedback to maintain operation within safety limits. A transient index which is used to assess the void reactivity characteristics is the dynamic void coefficient. This parameter is defined as the core physics void coefficient multiplied by the average full power void fraction and divided by the delayed neutron fraction.

The presence of U-238 and, ultimately, Pu-240 contributes to yield a strong negative Doppler coefficient. This coefficient provides instantaneous negative reactivity feedback to any fuel temperature rise, either gross or local. The magnitude of the Doppler coefficient is not dependent on gadolinium position or concentration in any bundle because gadolinium has very little effect on the resonance group flux or on U-238 content of the core. The core physics

Doppler coefficient is divided by the delayed neutron fraction to define a dynamic Doppler coefficient (K_D), which most effectively correlates the dynamic response of the plant to the Doppler reactivity feedback.

Additional conservatism is introduced into the analysis to account for biases and uncertainties in the derivation of the nuclear data and its application to the REDY transient model. The conservatism factors used to account for these 2/83 biases and uncertainties are summarized in Table 5-4. These values, in effect, represent a maximum level of conservatism in the transient analysis upon which compliance to existing safety limits is generally based.

The reactor core behavior for General Electric-supplied fuel during the rod withdrawal error transient is calculated by doing a series of steady-state three-dimensional coupled nuclear thermal-hydraulic calculations using the three-dimensional BWR simulator (Reference 5-7). This approach assumes that the transient is very slow such that there is sufficient time for heat transfer and void redistribution to equilibrate and also that the neutron flux and heat flux are in equilibrium with each other. This calculation is achieved by maintaining a constant eigenvalue via increasing core average power as a control rod is withdrawn incrementally.

Descriptions of the transient events are given in Subsection 5.2.1. Inputs to these transients which are specific to the Oyster Creek plant are given in Tables 5-3 and 5-5. Transient inputs which are specific to the reference cycle are given in the reference cycle supplement. 1/80 Because the transient model establishes operating conditions, only licensing basis values are given in Table 5-5. Actual values used in the analyses will be within the tolerances shown in the table. The transient descriptions given in Subsection 5.2.1 are used as a basis for the typical analyses performed for plant reloads. Oyster Creek analyses will differ in certain aspects from the typical calculational procedure due to the utility selected margin improvement option of exposure-dependent limits. A description of this option and its effect upon the transient analysis is given in Subsection 5.2.2. ATWS pump trip is also assumed in the analysis of Oyster Creek.

The operating limit MCPR for General Electric fuel for rapid transients is calculated by using the SCAT computer program (Reference 5-8). Inputs to this program consist of the transient analysis results, steady-state flow distribution (from Section 4), bundle power, axial distribution, and gap conductance. The axial power distribution used in the analyses is given in Table 5-6. Nonvarying plant initial conditions for the GETAB analysis are given in Table 5-7. Reload-dependent plant initial conditions for the GETAB analysis and the resulting reload MCPR operating limit for General Electric fuel in the reference cycle are given in the reference cycle supplement. The initial MCPR assumed for transient analyses is usually greater than or equal to the GETAB operating limit. Figure 5-3 illustrates the effect of the initial MCPR on transient ΔCPR for a typical BWR core. This figure indicates that the change in ΔCPR is approximately 0.01 for a 0.05 change in initial MCPR. Therefore, nonlimiting GETAB transient analyses may be initiated from a MCPR below the operating limit because the higher operating limit MCPR more than offsets the increase in ΔCPR for the event. This may also be applied to limiting transients if the difference between the operating limit and the initial MCPR is small (0.01 or 0.02). Densification power spiking is not considered in establishing the MCPR operating limit. Justification for this is presented in Subsection 5.2.3.

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The deterministic ΔCPR value which results from ODYN/Improved SCAT evaluations (for all rapid pressurization transients) must be adjusted such that a 95/95 ΔCPR/ICPR licensing basis is calculated (i.e., 95% probability with 95% confidence that the safety limit will not be violated). The SER which describes these requirements and procedures is given in Reference 5-32. The method of applying these adjustment factors is given in Reference 5-33. Each utility has the choice of operating under one of the following basic options:

- Option A — Under Option A, an NRC-imposed factor of 1.044 is applied to the MCPR for each event to account for code uncertainties.

- Option B — Under Option B, General Electric - developed adjustment factors (Reference 5-32) are applied to the ΔCPR/ICPR ratio. These factors are a result of a statistical analysis of transient response based on an improved CRD scram insertion time distribution. The

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factors are a function of plant type and event. Since these adjustment factors take credit for conservatism in the scram speed assumed for the transient analysis, each plant operating under Option B must demonstrate that its scram speed is within the distribution used in the statistical analysis. This conformance procedure is described in Reference 5-32 and will be incorporated in the technical specifications of each plant operating under Option B. 2/83

The adjusted MCPR values are given in the Supplemental Reload Submittal.

The operational limit MCPR must be increased for low flow conditions. This is because, in the BWR, power increases as core flow increases, which results in a corresponding lower MCPR. If the MCPR at a reduced flow condition were at the 100% power and flow MCPR operating limit, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Safety Limit MCPR given in Subsection 5.2.1. Therefore, the required operating limit MCPR is increased at reduced core flow rates by a flow factor, K_f, such that:

$$\begin{matrix}\text{Required MCPR} \\ \text{Operating Limit}\end{matrix} = K_f * \begin{matrix}\text{MCPR Operating Limit} \\ \text{@ 100\% core flow}\end{matrix}$$

The flow factor (K_f), as a function of the core flow rate, the flow control mode, and (in the manual flow control mode) the maximum possible core flow

will be a slow, exponential curve. An analysis to the linear approximation, however, will be conservative, since it overpredicts the power level for any given exposure.

In Reference 5-9, evaluations were made to 40% power level points on the linear curve. The results show that the pressure margins from the limiting pressurization transient and the MCPR operating limits exhibit a larger margin for each of these points than the EOC full power, full flow case. MAPLHGR limits for the full power, rated flow case is conservative for the coastdown period, since the power will be decreasing and rated core flow will be maintained (Subsection 5.5). Therefore, it can be concluded that the coastdown operation beyond full power operation is conservatively bounded by the analysis at the EOC conditions. In Reference 5-34 this conclusion is confirmed for analyses done with ODYN. 2/83

5.2.1 Transient Descriptions

Eight nuclear system parameter variations can pose potential deleterious effects to the Nuclear Steam Supply System. The parameter variations are as follows:

(1) Nuclear system pressure increase - threatens to rupture the reactor coolant pressure boundary from internal pressure. Also, a pressure increase collapses the voids in the moderator. This causes an insertion of positive reactivity which may result in exceeding the fuel cladding safety limits.

(2) Reactor vessel water (moderator) temperature decreases - results in an insertion of positive reactivity as density increases. Positive reactivity insertions threaten the fuel cladding safety limits because of higher power.

(3) Positive reactivity insertion - is possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten the fuel cladding safety limits because of higher power.

Event	Approximate Elapsed Time
(g) APRM 115.7% power signal scrams reactor (conservative; in startup mode, APRM scram would be operative + IRM).	
(h) Scram terminates accident.	$\leq$ 5 sec

To limit the worth of the rod which could be dropped, the rod worth minimizer system (RWM) is used below 10% power to enforce the rod withdrawal sequence. The RWM is programmed to follow the control rod sequences shown in Figure 5-25. 2/83

The rod drop accident design limit restricts peak enthalpies in excess of 280 cal/gm for any possible plant operation or core exposure.

5.5.1.2 Model Parameters Sensitivities

Although there are many input parameters to the RDA analysis, the resultant peak fuel enthalpy is most sensitive to the following input parameters:

(1) steady-state accident reactivity shape function;

(2) total control rod reactivity worth;

(3) maximum interassembly local power peaking factor P_F. P_F represents the maximum local peaking factor normalized over the four bundles surrounding the dropped control rod. Mathematically,

$$P_F = \text{Max } P_{L_i} \left\{ \frac{4\ BP_i}{\sum_{j=1}^{4} BP_j} \right\} \quad i = 1,2,3,4 \tag{5-14}$$

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5.5.2 Loss-of-Coolant Accident

This analysis of the Oyster Creek Nuclear Generating Station loss-of-coolant accident (LOCA) is provided 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 documentation contained in this section is intended to satisfy these requirements.

The general description of the General Electric (GE) LOCA evaluation models is contained in Reference 5-18. Applicability and approval for pre-pressurized reload fuel are given in Reference 5-25. Model changes are described in References 5-20 and 5-21, which were approved by the USNRC (Reference 5-19). The analysis utilizes the short-term thermal-hydraulic model (LAMB) and the transient critical power model (SCAT) in addition to the long-term thermal-hydraulic model (SAFE) and the core heat-up model (CHASTE).

This LOCA analysis differs from previous BWR/2 LOCA analyses in the following ways:

1. Core flow coastdown and core depressurization are now calculated with the LAMB and SCAT codes, as opposed to no modelling of coastdown. In order to use the LAMB computer code for a non-jet pump plant, LAMB inputs were developed to execute the code for a recirculation line break. These inputs reflect the geometry of the non-jet pump plant. The LAMB code only allows for two recirculation pump loops; therefore, the five loops were modeled such that the intact loops are combined into one loop, and the broken loop is modeled as the second loop. 2/83

2. For the large break region, dryout time is now calculated using the SCAT code, as opposed to assuming a set dryout time.

The Oyster Creek LOCA analysis reported here was performed as an independent, self-contained analysis similar to that performed as a lead plant analysis.

5.5.2.1 Input to Analysis

A list of the significant input parameters to the LOCA analysis is presented in Table 5-10.

5.5.2.2 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). 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.

5.5.2.3 SCAT ANALYSIS

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This code completes the transient short-term thermal-hydraulic calculation for large breaks. 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.

5.5.2.4 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

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break sizes. This code is used during the entire course of the postulated accident. SAFE calculates reactor system pressure, ECCS flows (which are pressure dependent), and hot fuel node uncovery time. Reflooding times are not calculated in non-jet pump analysis. All peak cladding temperatures are turned over by action of core spray heat transfer.

The SAFE results presented are:

- Water Level inside the Shroud.
- Reactor Vessel Pressure.

5.5.2.5 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 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 contained in CHASTE, which solves the transient heat transfer equations for the entire LOCA at a single axial plane in a single fuel assembly. Iterative applications of CHASTE determine the maximum permissible planar power required to satisfy the 10CFR50.46 acceptance criteria. 2/83

The CHASTE results presented are:

- Peak Cladding Temperature and Peak Local Oxidation versus Time.
- Peak Cladding Temperature and Peak Local Oxidation versus Break Area.
- Peak Cladding Temperature and Peak Local Oxidation versus Planar Average Exposure for the most limiting break size. 9/79
- Maximum Average Planar Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size.

5.5.2.6 Methods

In the following sections, it will be useful to refer to the methods used to analyze large and small breaks. These methods are described below.

5.5.2.6.1 Large Break Methods (LBM)

LAMB/SCAT/SAFE/CHASTE Break sizes: 0.3 $ft^2 \leq A \leq$ DBA. Heat transfer coefficients: Figures 5-20a, 5-20b, and 5-20c. SCAT is omitted for those break sizes which do not result in significant positive coastdown flow (40% DBA to DBA). Under these circumstances, the duration of nucleate boiling following the recirculation line break is calculated by CHASTE using the GE dryout correlation, which is based on no-flow conditions. This correlation relies on experimental data from many different test assemblies with various axial power shapes and different geometries whose purpose was to investigate the no-flow dryout phenomenon. The Ellion pool boiling heat transfer correlation is used from the time that the decay heat transfer from the dryout correlation ceases to apply until the time that the hot fuel core node is uncovered. 2/83

5.5.2.6.2 Small Break Methods (SBM)

SAFE/CHASTE Break sizes: $A \leq 0.3\ ft^2$. Heat transfer coefficients: nucleate boiling prior to core uncovery, core spray, and 1000 Btu/hr-ft^2-°F after rods wet. Peak cladding temperature and peak local oxidation are calculated in CHASTE.

5.5.2.7 Break Spectrum Calculations

For convenience in describing the LOCA phenomena, the break spectrum has been separated into two regions: (1) small breaks, and (2) large breaks. The potentially limiting single active failures considered are given in Table 5-13.

The selection of the transition break between Large Break Methods (LBM) and Small Break Methods (SBM) (i.e., 0.3 ft^2) is made in order to achieve similar coastdown flow, core uncovery, and rated spray time characteristics to those

exhibited at the transition break for jet-pump plants. The transition break has been analyzed with both the large and small break methods with the same single failure to allow comparison between the methods. The analysis of the transition break is shown in Figures 5-19c, 5-19d, and 5-22b.

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 severence 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 Emergency Condensor failure.

5.5.2.8 Large Break Analysis

In the large break region between the 60% and DBA break sizes, there exist two single failures (ADS and EC) which produce break spectrums which are nearly equal. This is due to the fact that in this break region both failures have no impact on the hot node uncovery time and the time of rated core spray. The DBA break is the limiting break in the large break region ($0.3 \leq A \leq$ DBA). This is due to the earliest core uncovery and no coastdown flow characteristics for the DBA. The DBA break with the Emergency **Condenser** single failure produces a PCT of 2°F higher than the DBA with the ADS single failure. Therefore, the EC single failure is considered the limiting failure. In the region between 60% DBA and 0.3 ft^2 the EC failure is more limiting than the ADS failure by an even larger margin.

In the range of 40% DBA to DBA, credit is not taken for coastdown flow due to the rapid decrease in core inlet flow calculated by the LAMB code for this region. In the region between 40% and 0.3 ft^2 break, coastdown flow has a significant impact in delaying boiling transition, and, therefore, the results of LAMB and SCAT are used to calculate PCT.

5.5.2.9 Small Break Analysis

In this region, the vessel depressurizes relatively slow. The EC is the most severe single failure in this region compared to the ADS failure because the 1 ADS single failure results in a later hot node uncovery time. The break

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spectrum in this region shows that the 0.1 ft^2 break is the most limiting break size. The limiting break in this region shifts from the previous 0.13 ft^2 to 0.10 ft^2 because of the use of the SBM. The previous worst break of 0.13 ft^2 was determined by the LBM without credit for coastdown core flow. The differences in these models cause the time between uncovery and rated core spray to have a greater effect than the higher decay heat at time of uncovery for 0.13 ft^2. The time between uncovery and rated spray is approximately 15 seconds greater for the 0.1 ft^2 break compared to the 0.13 ft^2 break. This larger time between these two events is responsible for the 0.10 ft^2 break being more limiting than all other break sizes in this region.

5.5.2.10 Break Spectrum Conclusions

A summary of the analytical results is given in Table 5-11. Table 5-12 lists the figures provided for this analysis. The MAPLHGR values for each General Electric fuel type available to Oyster Creek are given in Table 5-14. Peak cladding temperatures and oxidation fractions are also given in this table.

The results of the complete break spectrum show that two break sizes may be limiting depending on fuel exposure. The exposure-dependent limiting break is due to the combined effects of gap conductance and rod internal pressure. For fuel in the low exposure region ($\leq$1000 MWd/ST), the DBA break size is limiting because of the PCT limit of 2200°F. Because of a gradual improvement in gap conductance after 1000 MWd/ST, the small break (0.1 ft^2) becomes limiting due to PCT at the mid-exposure range (5000 MWd/ST to 15,000 MWd/ST). At higher exposure ($\geq$20,000 MWd/ST), as fission gas release increases, the DBA break becomes limiting due to peak local oxidation fraction.

The break spectrum summary curve is shown in Figure 5-24. This figure gives the maximum PCT and the maximum local oxidation, over all exposures, as a function of break size.

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5.5.3 Main Steamline Break Accident Analysis

The analysis of the main steamline break accident depends on the operating thermal-hydraulic parameters of the overall reactor (such as pressure) and overall factors affecting the radiological consequences (such as primary coolant activity). Insertion of reload fuel will not change the radiological consequences of this event. Analytical results supporting this are presented in Reference 5-26. 1/80

5.5.4 Loading Error Accident Calculational Methods

One of the events which has been evaluated in BWR safety analysis reports is fuel bundle loading error. A loading error in the core configuration is defined as: (1) a General Electric fuel bundle in an improper location (mislocation), or (2) a General Electric fuel bundle in an improper orientation (i.e., misoriented - rotated 90° or 180°).

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

(1) The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

(2) The identification boss on the fuel assembly handle points toward the adjacent control rod.

(3) The channel spacing buttons are adjacent to the control rod passage area.

5-15 R. C. Stirn, C. J. Paone, and R. M. Young, *Rod Drop Accident Analysis for Large Boiling Water Reactors*, Licensing Topical Report, July 1972 (NEDO-10527, Supplement 1).

5-16 R. C. Stirn, C. J. Paone, and J. M. Haun, *Rod Drop Accident Analysis for Large Boiling Water Reactors, Addendum No. 2, Exposure Cores*, Licensing Topical Report, January 1973 (NEDO-10527, Supplement 2).

5-17 Fuel Densification Effects on General Electric Boiling Water Reactor Fuel, August 1973, Supplement 6 to NEDM-10735.

5-18 *Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K*, January 1976 (NEDE-20566-P and NEDO-20566).

5-19 Letter, K. R. Goller (NRC) to G. G. Sherwood (GE), "Safety Evaluation for General Electric ECCS Evaluation Model Modifications", dated April 12, 1977.

5-20 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 Heat Code CHASTE05".

5-21 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) Evaluation Model Code REFLOOD05".

5-22 Letter, N. G. Trikouros (GPU) to J. F. Kilty (GE), dated November 13, 1975, No. S&L-3305, "Oyster Creek Nuclear Generating Station Revised Single Failure LOCA Analysis". 9/79

5-23 Letter, R. E. Engel to D. G. Eisenhut (NRC), "Fuel Assembly Loading Error", November 30, 1977.

5-24 Letter, D. G. Eisenhut (NRC) to R. E. Engel (GE), MFN-200-78, May 8, 1978.

5-25 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P*.

5-26 Attachment to Letter, S. Bartnoff to A. Giambusso, "Oyster Creek Nuclear Generating Station Docket Number 50-219 Loss-of-Coolant Accident Analysis Re-evaluation Additional Information," April 28, 1975. 1/80

5-27 "Qualification of the One-Dimensional Core Transient Model for BWR's", October 1978 (NEDO-24154, Vol. 1 and 2).

5-28 "Qualification of the One-Dimensional Core Transient Model for BWR's", October 1978 (NEDE-24154-P, Vol. 3). 2/83

5-29 Letter, J. F. Quirk (GE) to P. S. Check (NRC), "ODYN Improvements," September 25, 1981.

*Reference refers to the revision of NEDE-24011-P-A which is approved by the NRC as of the date the specific analysis is initiated.

5-30 Letter, J. F. Quirk (GE) to T. P. Speis (NRC), "ODYN Improvements", October 13, 1981.

5-31 Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Change in GE Methods for Analysis of Cold Water Injection Transients", September 30, 1980.

5-32 Letter, R. C. Tedesco (NRC) to G. G. Sherwood (GE), "Acceptance for Referencing General Electric Licensing Topical Report NEDO-24154/NEDE-24154P", February 4, 1981.

5-33 Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits", January 19, 1981.

5-34 Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "End of Cycle Coastdown Analyzed With ODYN/TASC", September 1, 1981.

5-35 Letter, H. C. Pfefferlen (GE) to D. G. Eisenhut (NRC), "Correction of ODYN Errors", June 8, 1982.

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Table 5-10 1/80

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

Plant Parameters:

Core Thermal Power	1969 MWt, which corresponds to 102% of rated power	
Vessel Steam Flow	7.4×10^6 lbm/h, which corresponds to 102% of rated power	2/83
Vessel Steam Dome Pressure	1020 psig	
Recirculation Line Break Area for Large Break	4.66 ft^2 (DBA), 1.0 ft^2, 0.3 ft^2	9/79 2/83
Recirculation Line Break Area for Small Break	0.3 ft^2, 0.10 ft^2	2/83
Number of Drilled Bundles	None	9/79

Fuel Parameters:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum CPR*
P8DRB239	P8x8R	13.4	1.57	1.30
P8DRB265L	P8x8R	13.4	1.57	1.30
P8DRB265H	P8x8R	13.4	1.57	1.30

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*To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.2745 (i.e., 1.30 divided by 1.02) for a bundle with an initial MCPR of 1.30.

Table 5-11

SUMMARY OF BREAK SPECTRUM RESULTS

• Break Size • Location • Single Failure	PCT (°F)	Peak Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
• 4.66 ft^2 • Recirculation Discharge • Emergency Condenser (LBM)	2200	17.0	*
• 1.0 ft^2 • Recirculation Discharge • Emergency Condenser (LBM)	2064	*	*
• 0.3 ft^2 • Recirculation Discharge • Emergency Condenser (LBM)	2123	*	*
• 0.3 ft^2 • Recirculation Discharge • Emergency Condenser (SBM)	2042	*	*
• 0.10 ft^2 • Recirculation Discharge • Emergency Condenser (SBM)	2200	*	0.40

*Less than limiting case

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Table 5-12

LOCA ANALYSIS FIGURE SUMMARY

	Large Break Model			Small Break Model	
	DBA	1.0 ft^2	0.3 ft^2	0.3 ft^2	0.1 ft^2
Water Level Inside Shroud and Reactor Vessel Pressure vs. Time	5-18a	5-18b	5-18c	5-18c	5-18d
Peak Cladding Temperature and Peak Local Oxidation vs. Time	5-19a.1/ 5-19a.2	5-19b	5-19c	5-19d	5-19e
Heat Transfer Coefficient vs. Time	5-20a	5-20b	5-20c	--	--
Normalized Core Average Inlet Flow vs. Time	5-21a	5-21b	5-21c	--	--
Minimum Critical Power Ratio vs. Time	--	5-22a	5-22b	--	--
Normalized Power vs. Time	5-23	--	--	--	--
Peak Cladding Temperature and Peak Local Oxidation vs. Break Area	5-24	--	--	--	--

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Table 5-13

SINGLE FAILURES CONSIDERED IN THE OYSTER CREEK LOCA ANALYSIS
(Reference 5-22)

Break Location	Single Failure	Available Systems
Recirculation Line	1 EC	2 CS + 0 EC + 5 ADS
	1 ADS	2 CS + 1 EC + 4 ADS
Feedwater and Steam Lines	1 EC	2 CS + 1 EC + 5 ADS
	1 ADS	2 CS + 2 EC + 4 ADS
Core Spray Line	1 EC	1 CS + 1 EC + 5 ADS
	1 ADS	1 CS + 2 EC + 4 ADS

EC = Emergency Condenser

CS = Core Spray

ADS = Automatic Depressurization System

Table 5-14a

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Oyster Creek

Fuel Type: P8DRB239

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	9.5	2198	0.095
1000	9.5	2198	0.095
5000	9.5	2194	0.085
10,000	9.5	2193	0.085
15,000	9.5	2194	0.085
20,000	9.0	2092	0.169
25,000	8.9	2048	0.169
30,000	8.9	2049	0.170
35,000	8.9	2050	0.170
40,000	8.8	2049	0.170

Table 5-14b

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Oyster Creek

Fuel Type: P8DRB265L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	9.5	2198	0.095
1000	9.5	2198	0.095
5000	9.5	2198	0.086
10,000	9.5	2198	0.086
15,000	9.5	2198	0.086
20,000	9.0	2081	0.170
25,000	8.8	2049	0.169
30,000	8.8	2045	0.170
35,000	8.8	2049	0.170
40,000	8.8	2050	0.170

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Table 5-14c

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Oyster Creek

Fuel Type: P8DRB265

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	9.5	2198	0.095
1000	9.5	2198	0.095
5000	9.5	2198	0.086
10,000	9.5	2198	0.086
15,000	9.5	2198	0.086
20,000	8.9	2078	0.170
25,000	8.8	2050	0.170
30,000	8.8	2044	0.166
35,000	8.8	2049	0.170
40,000	8.8	2050	0.170

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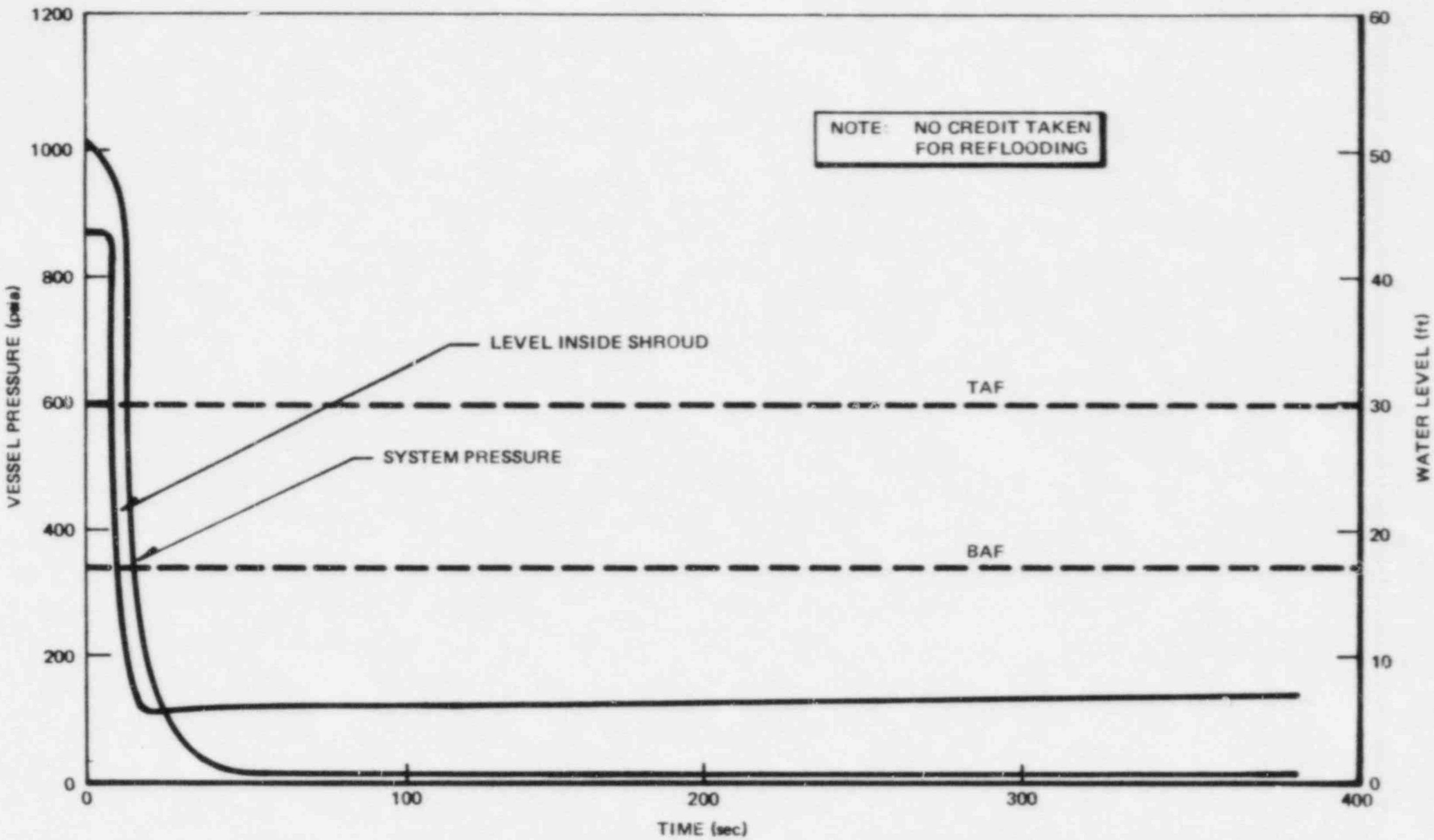

Figure 5-18a. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 4.66 ft^2 Recirculation Discharge Line Break, Emergency Condenser Failure

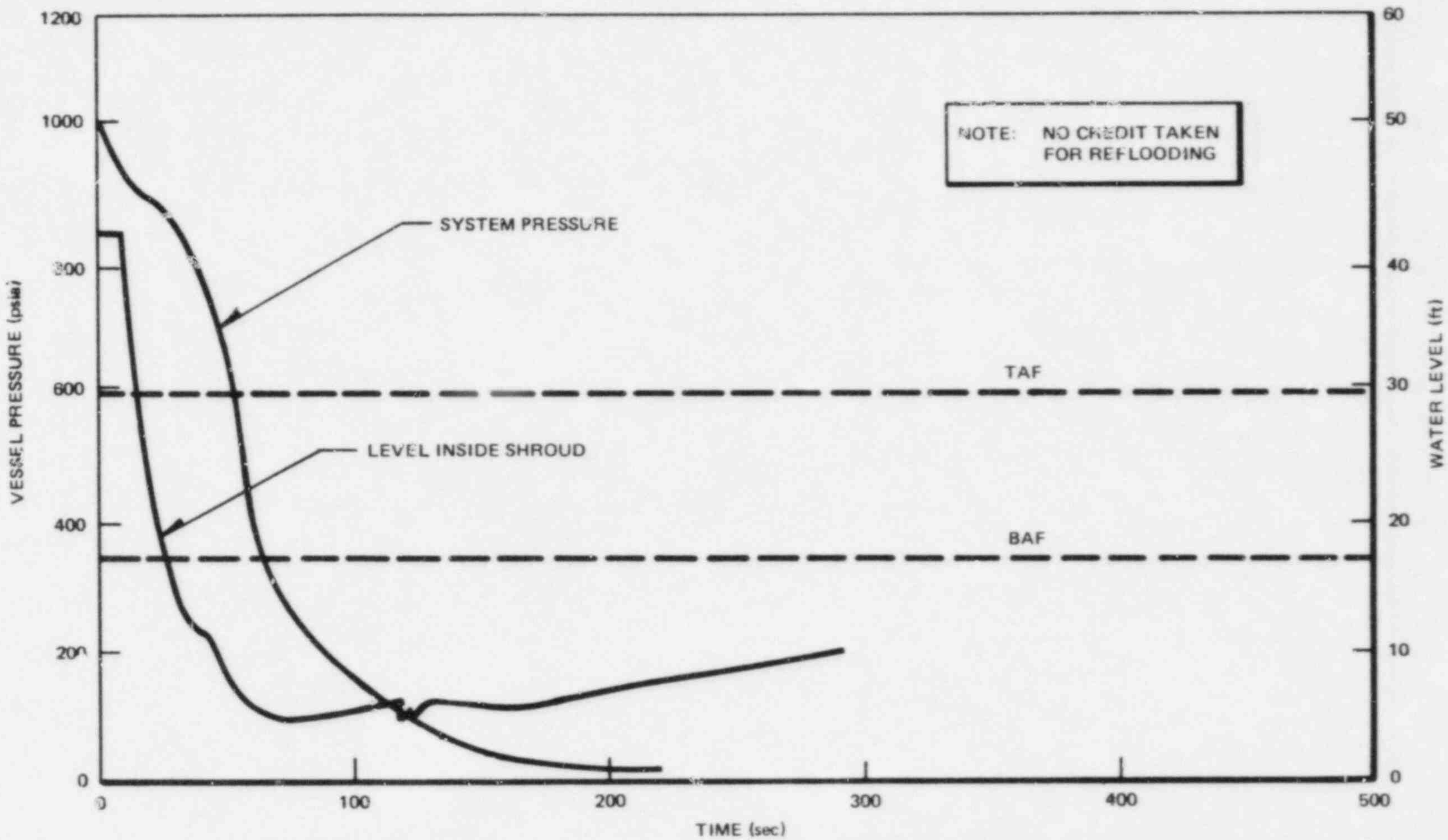

Figure 5-18b. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 1.0 ft^2 Recirculation Discharge Line Break, Emergency Condenser Failure

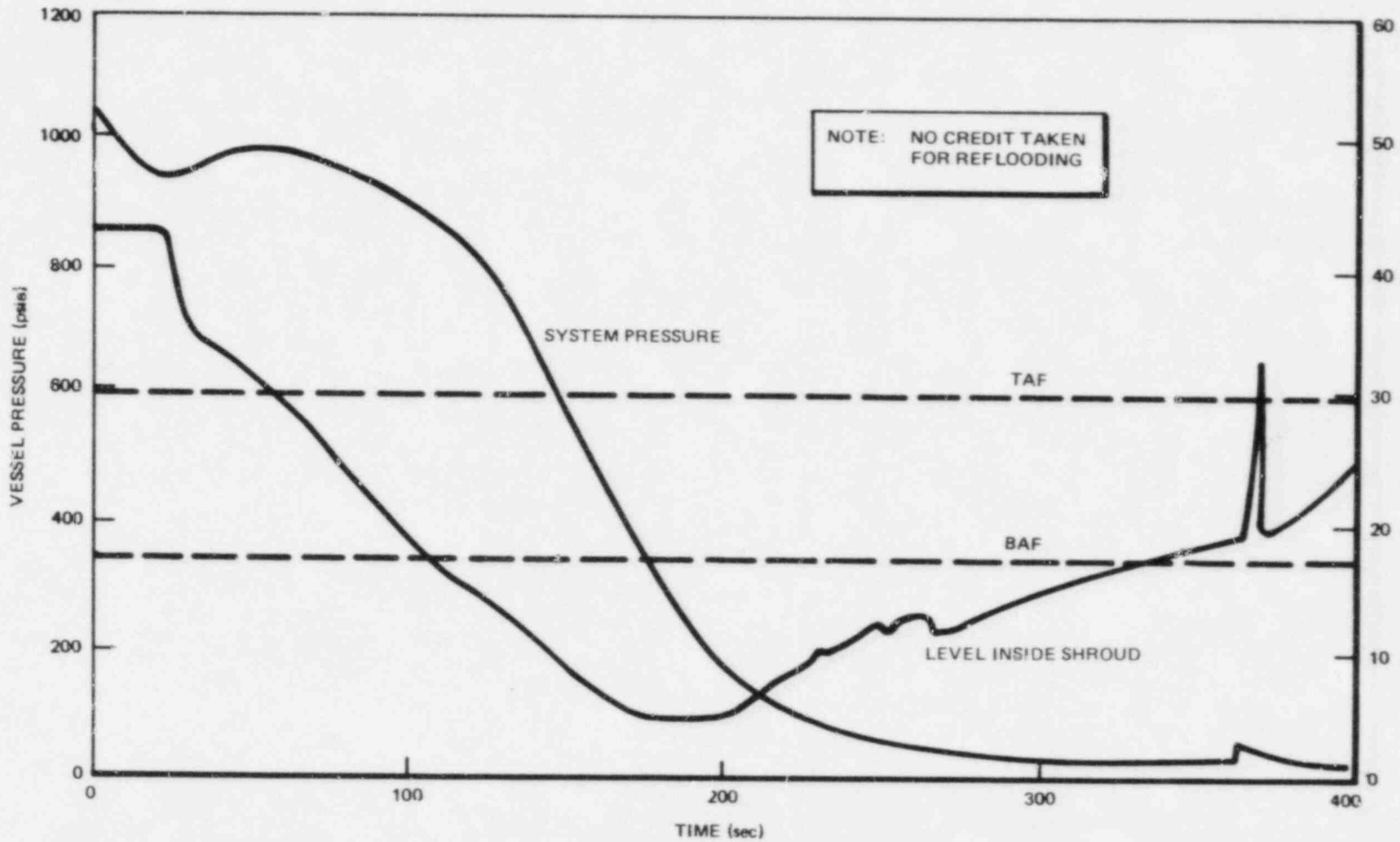

Figure 5-18c. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 0.3 ft^2 Recirculation Discharge Line Break, Emergency Condenser Failure

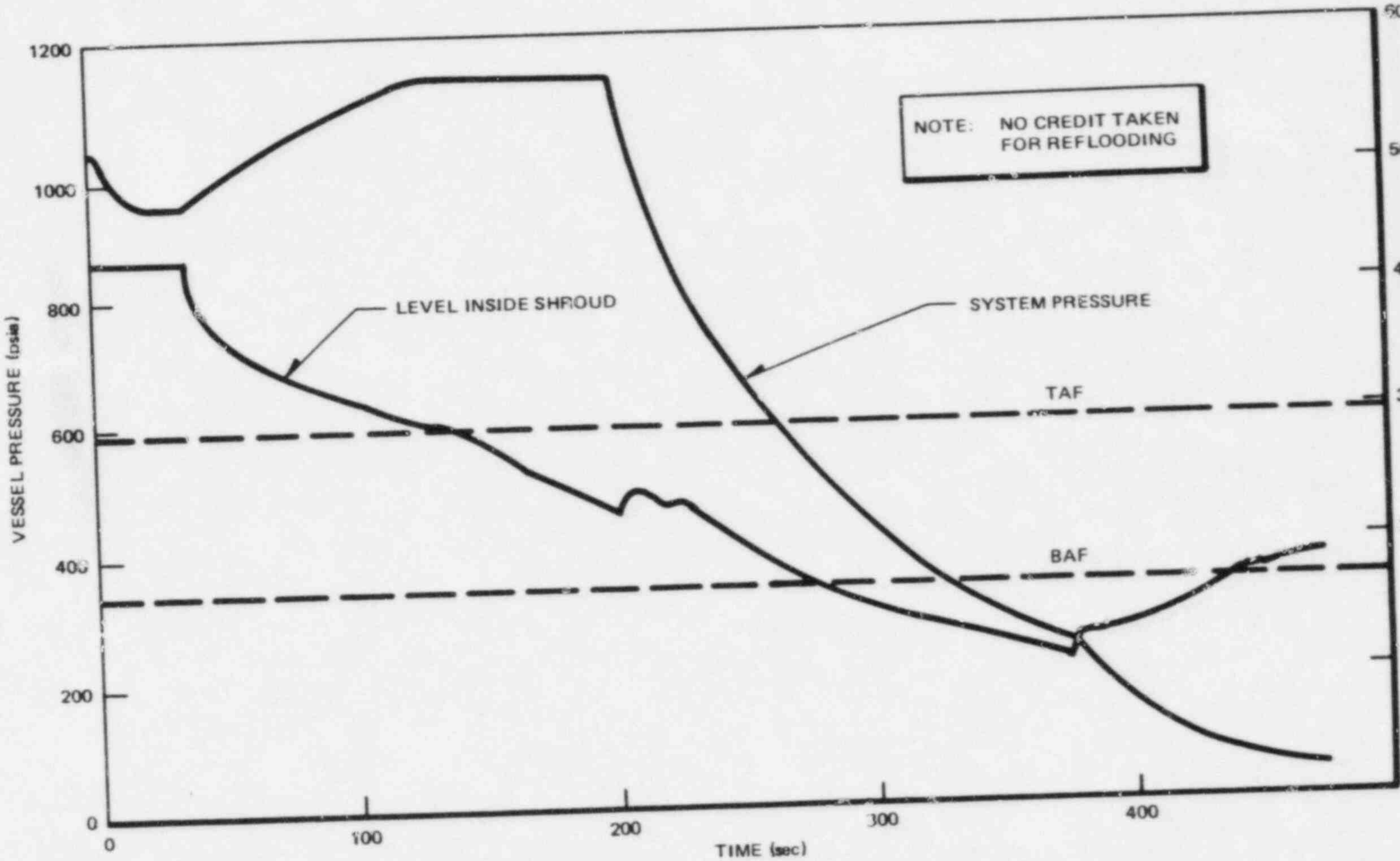

Figure 5-18d. Water Level Inside the Shroud and Reactor Vessel Pressure Following a 0.10 ft^2 Recirculation Discharge Line Break, Emergency Condenser Failure

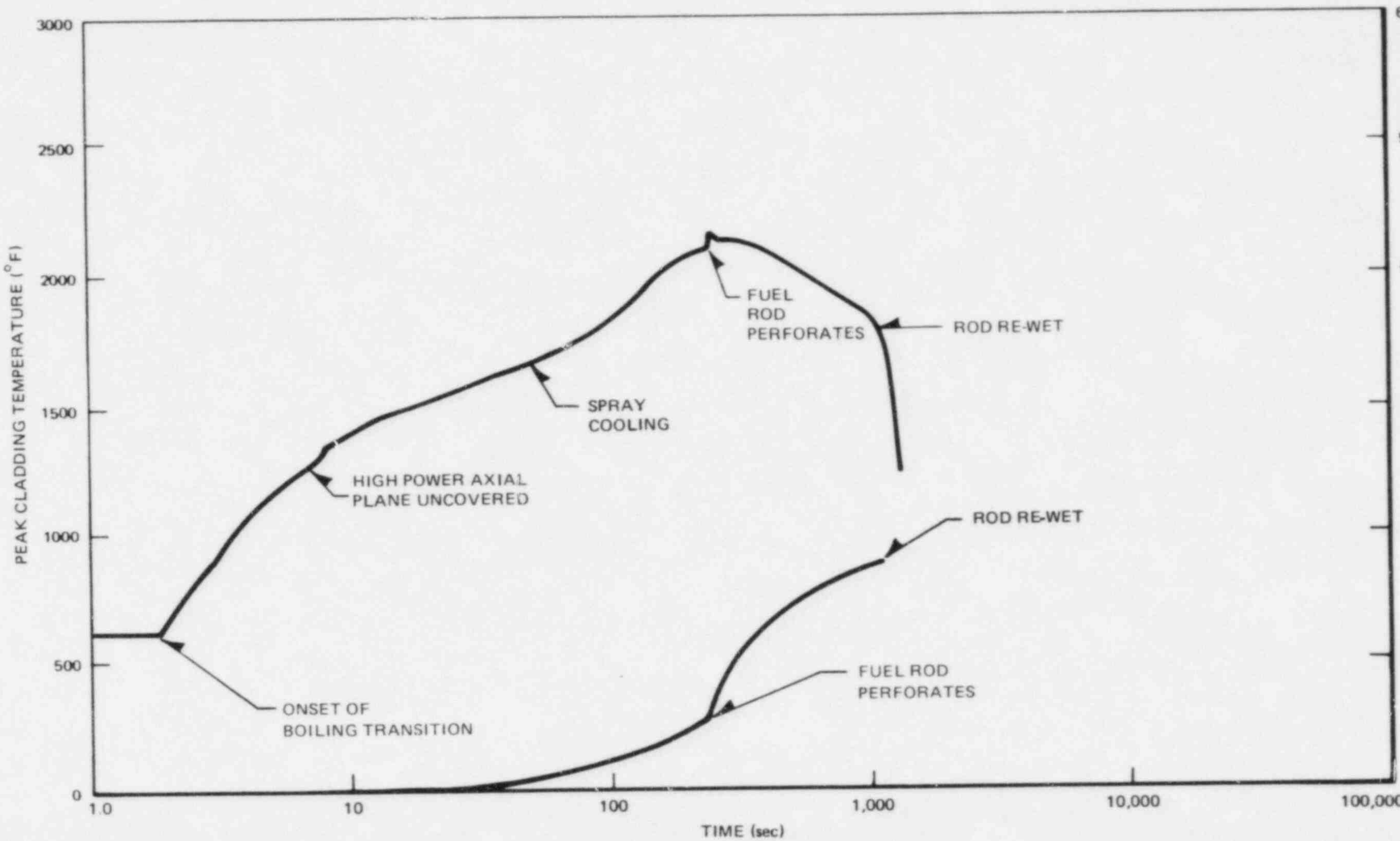

Figure 5-19a.1. Peak Cladding Temperature and Peak Local Oxidation Following a 4.66 ft^2 Recirculation Line Discharge Break, Emergency Condenser Failure (LBM) (E $\geq$ 20,000 MWd/ST)

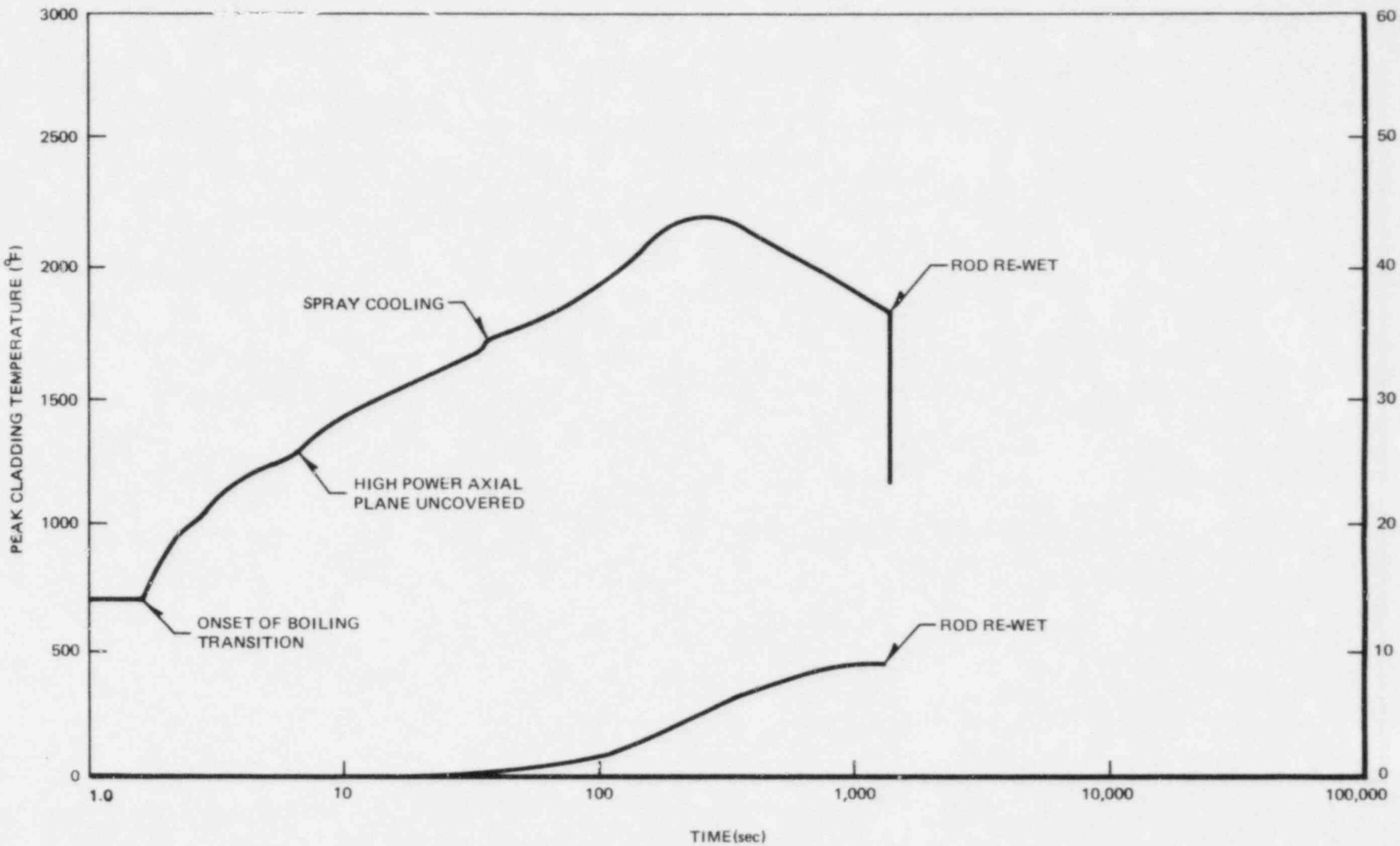

Figure 5-19a.2. Peak Cladding Temperature Following a 4.66 ft^2 Recirculation Line Discharge Break, Emergency Condenser Failure (LBM) ($E \leq 1000$ MWd/ST)

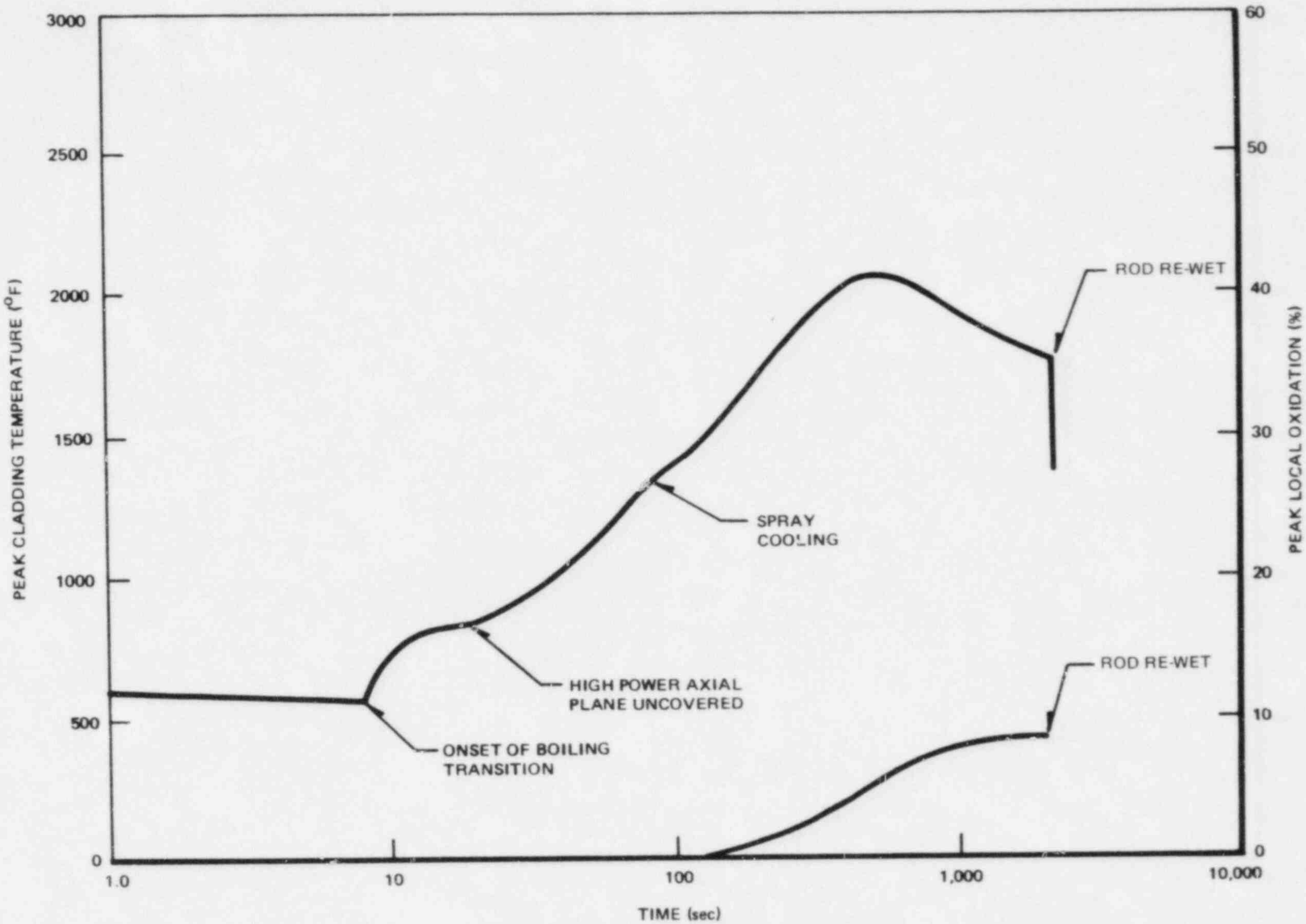

Figure 5-19b. Peak Cladding Temperature and Peak Local Oxidation Following a 1.0 ft^2 Recirculation Line Discharge Break, Emergency Condenser Failure (L[illegible]M)

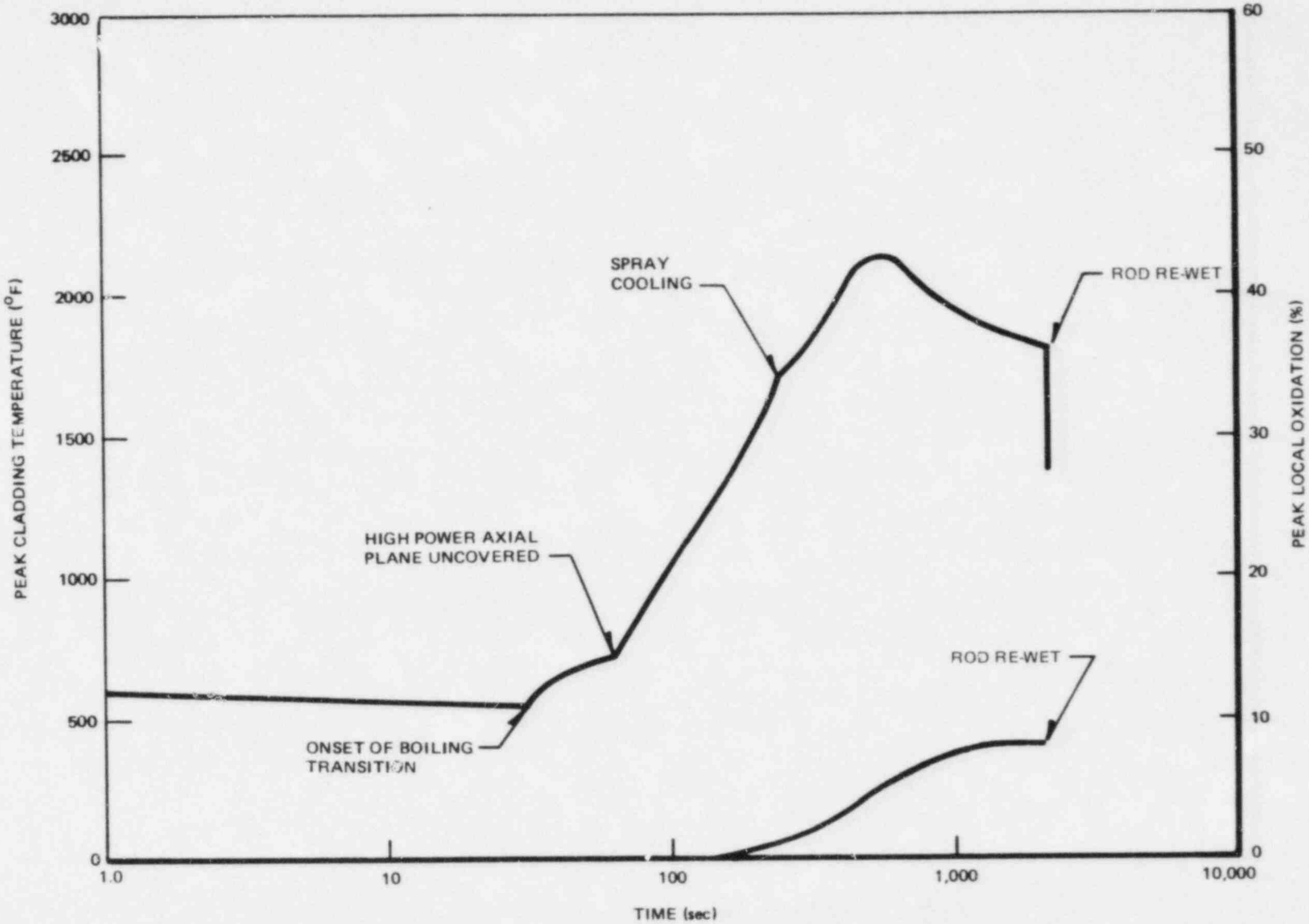

Figure 5-19c. Peak Cladding Temperature and Peak Local Oxidation Following a 0.3 ft^2 Recirculation Line Discarge Break, Emergency Condenser Failure (LBM)

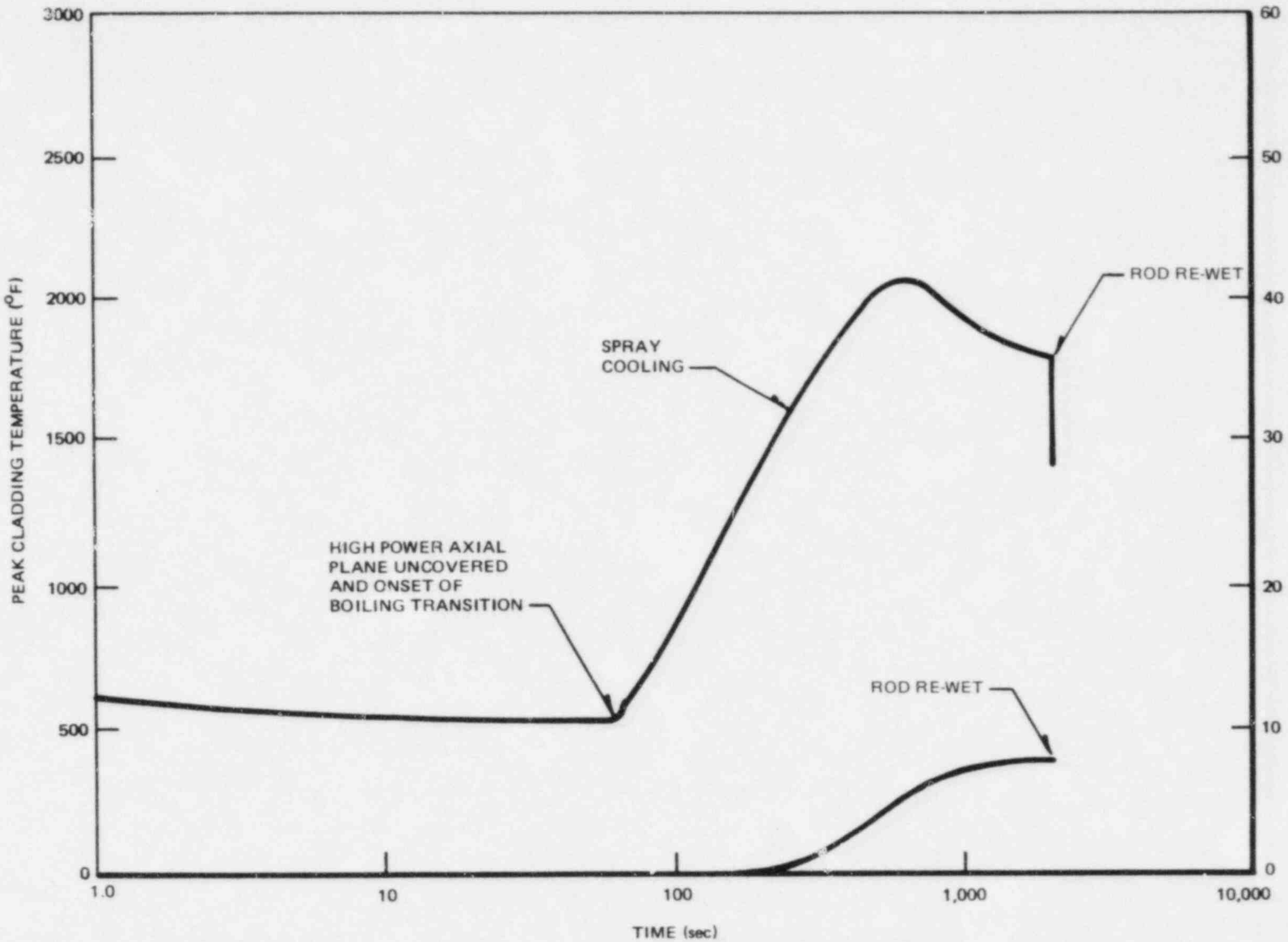

Figure 5-19d. Peak Cladding Temperature and Peak Local Oxidation Following a 0.3 ft^2 Recirculation Line Discharge Break, Emergency Condenser Failure (SBM)

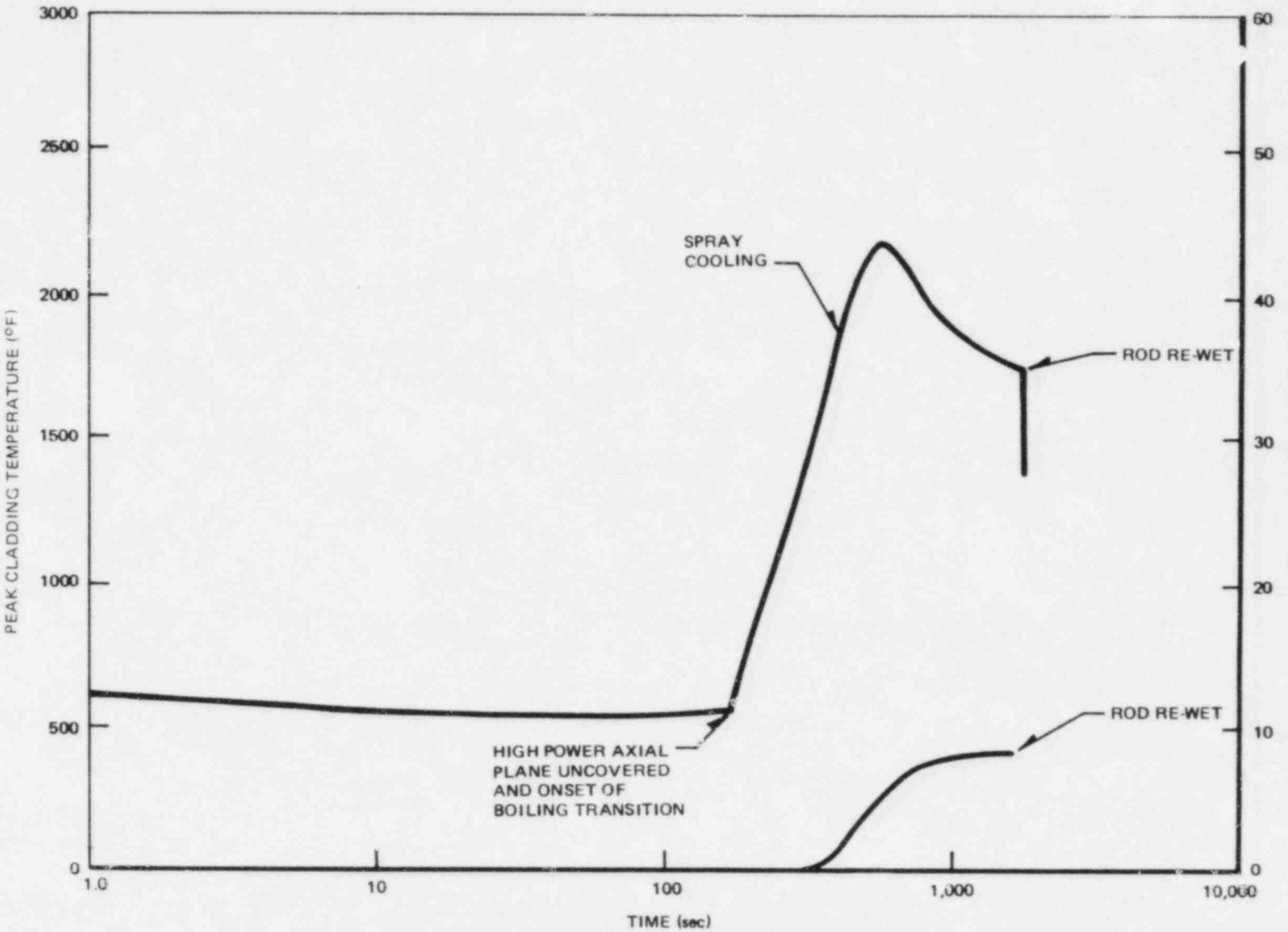

Figure 5-19e. Peak Cladding Temperature and Peak Local Oxidation Following a 0.10 ft^2 Recirculation Line Discharge Break, Emergency Condenser Failure (SBM)

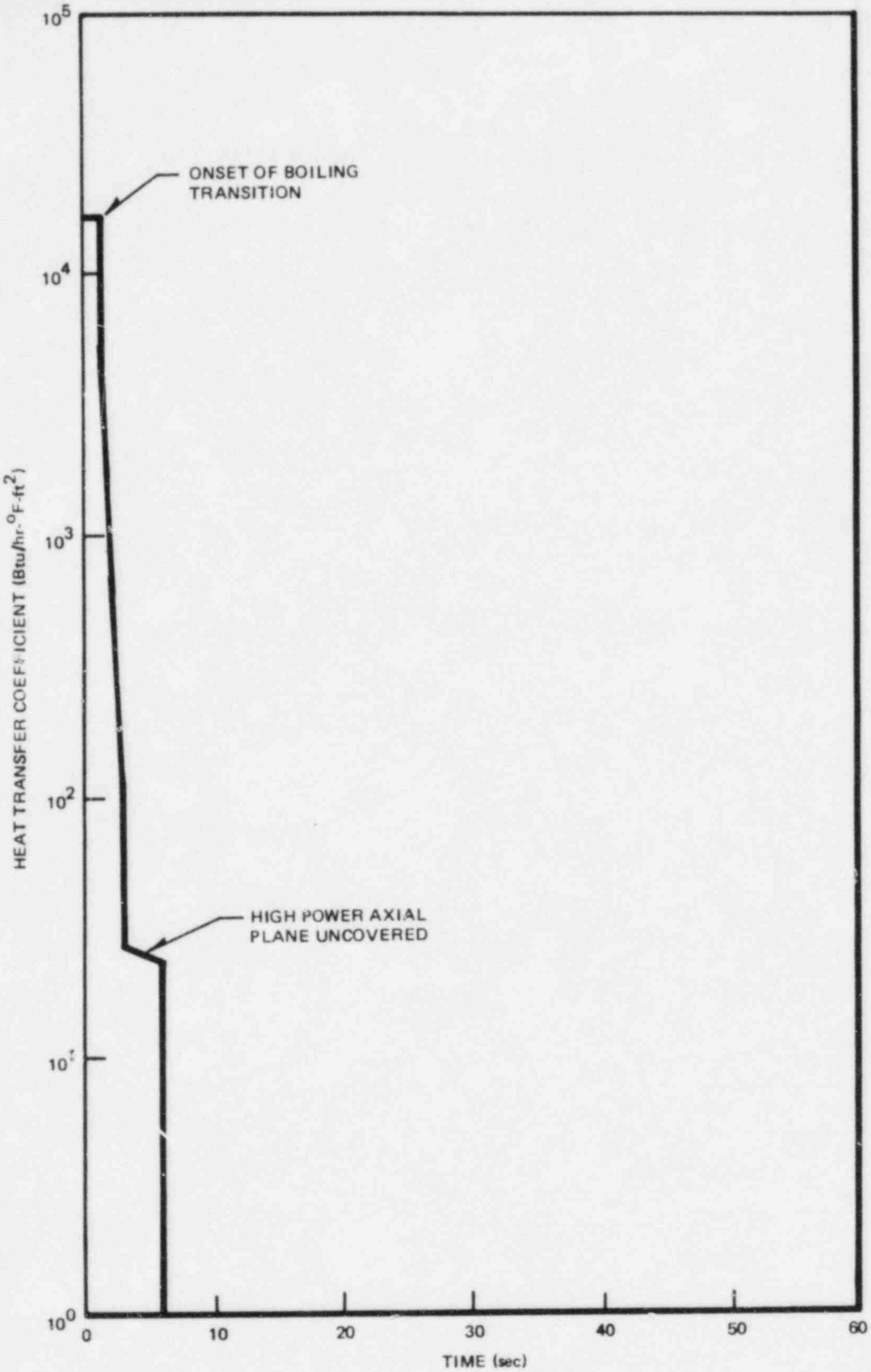

Figure 5-20a. Fuel Rod Convective Heat Transfer Coefficient at the High Power Axial Node for a (4.66) ft^2 Recirculation Line Discharge Break (LBM)

2/83

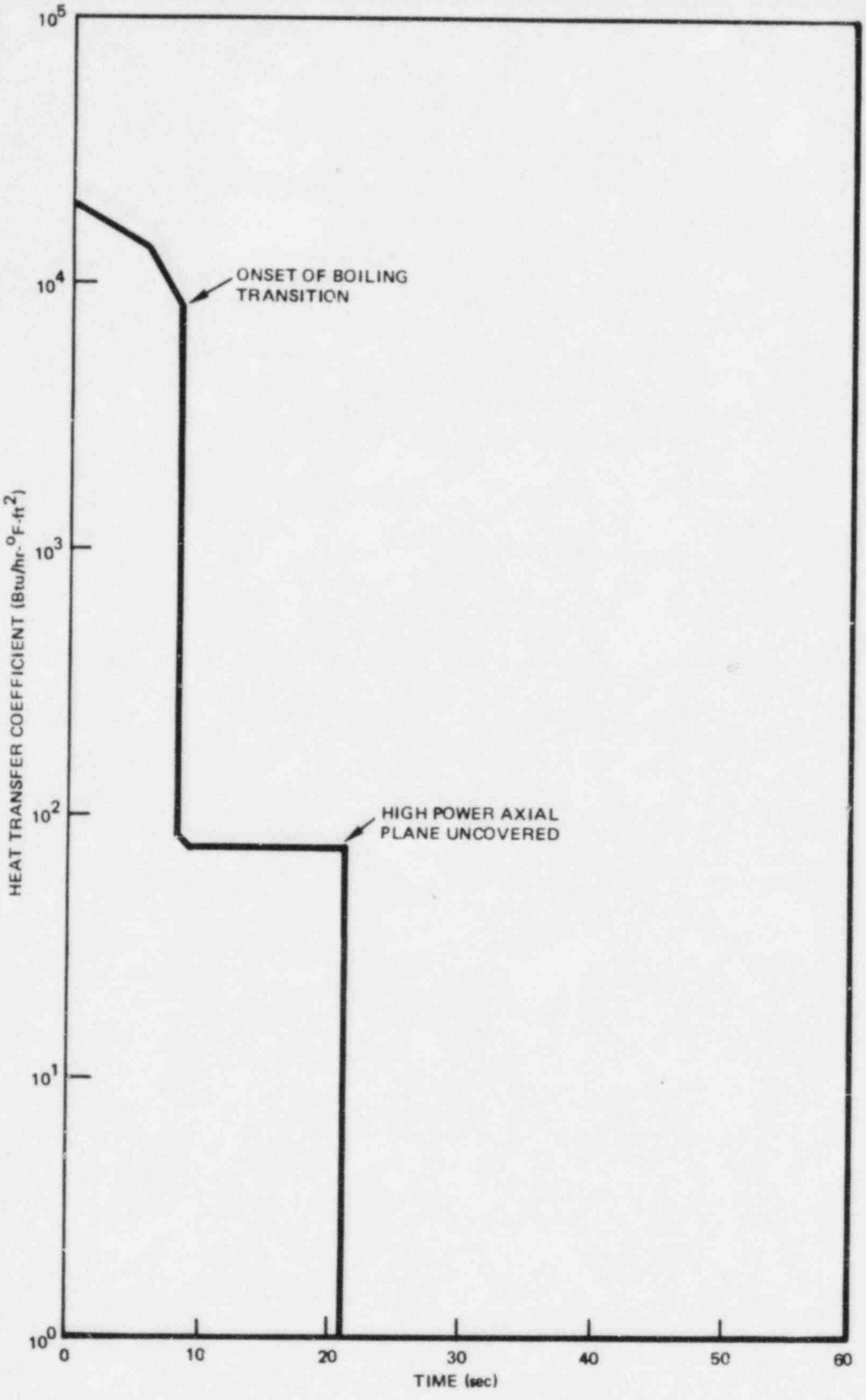

Figure 5-20b. Fuel Rod Convective Heat Transfer Coefficient at the Highest Power Axial Node for a 1.0 ft^2 Recirculation Line Discharge Break (LBM)

2/83

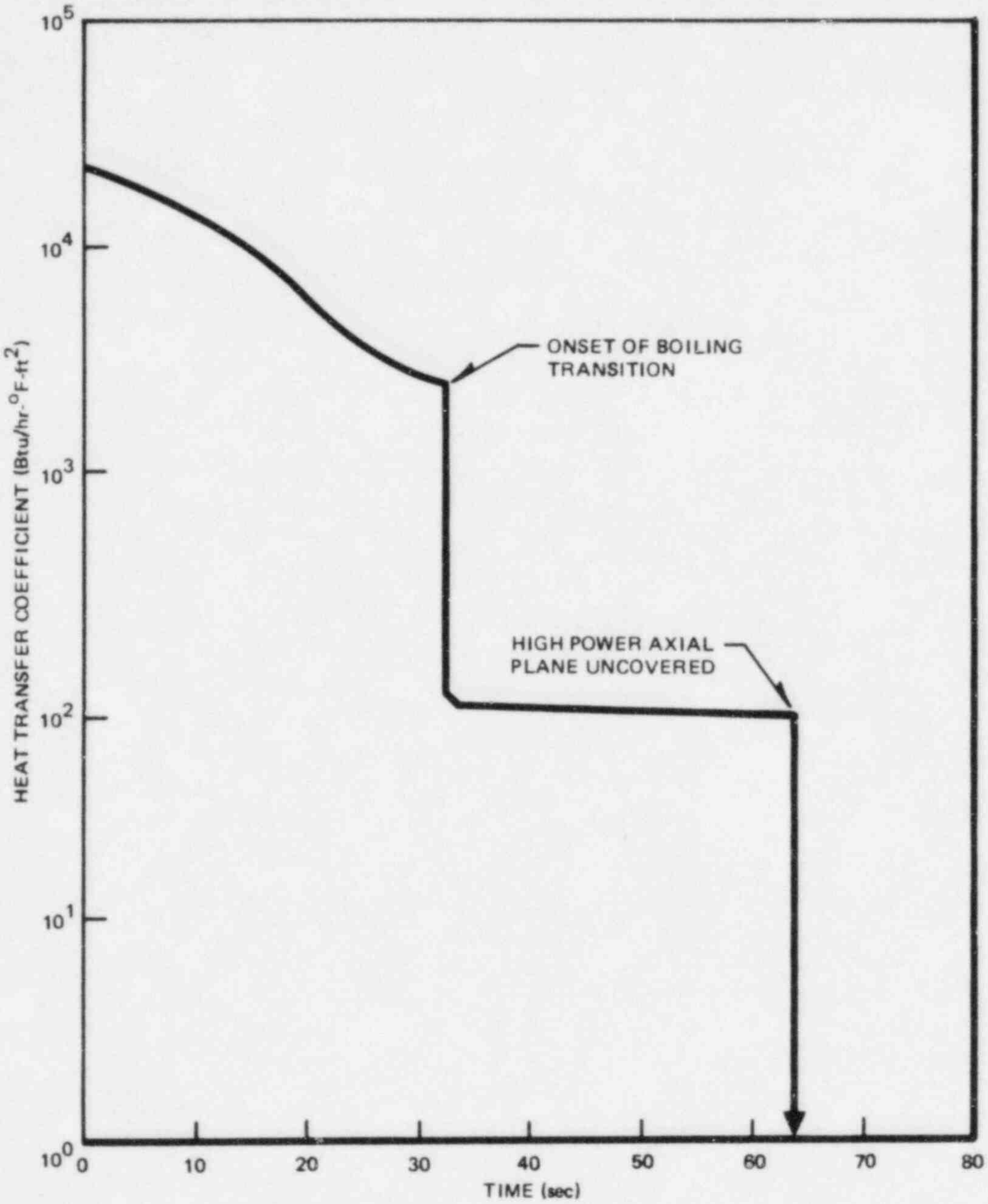

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Figure 5-20c. Fuel Rod Convective Heat Transfer Coefficient at the High Power Axial Node for a 0.3 ft^2 Recirculation Line Discharge Break (LBM)

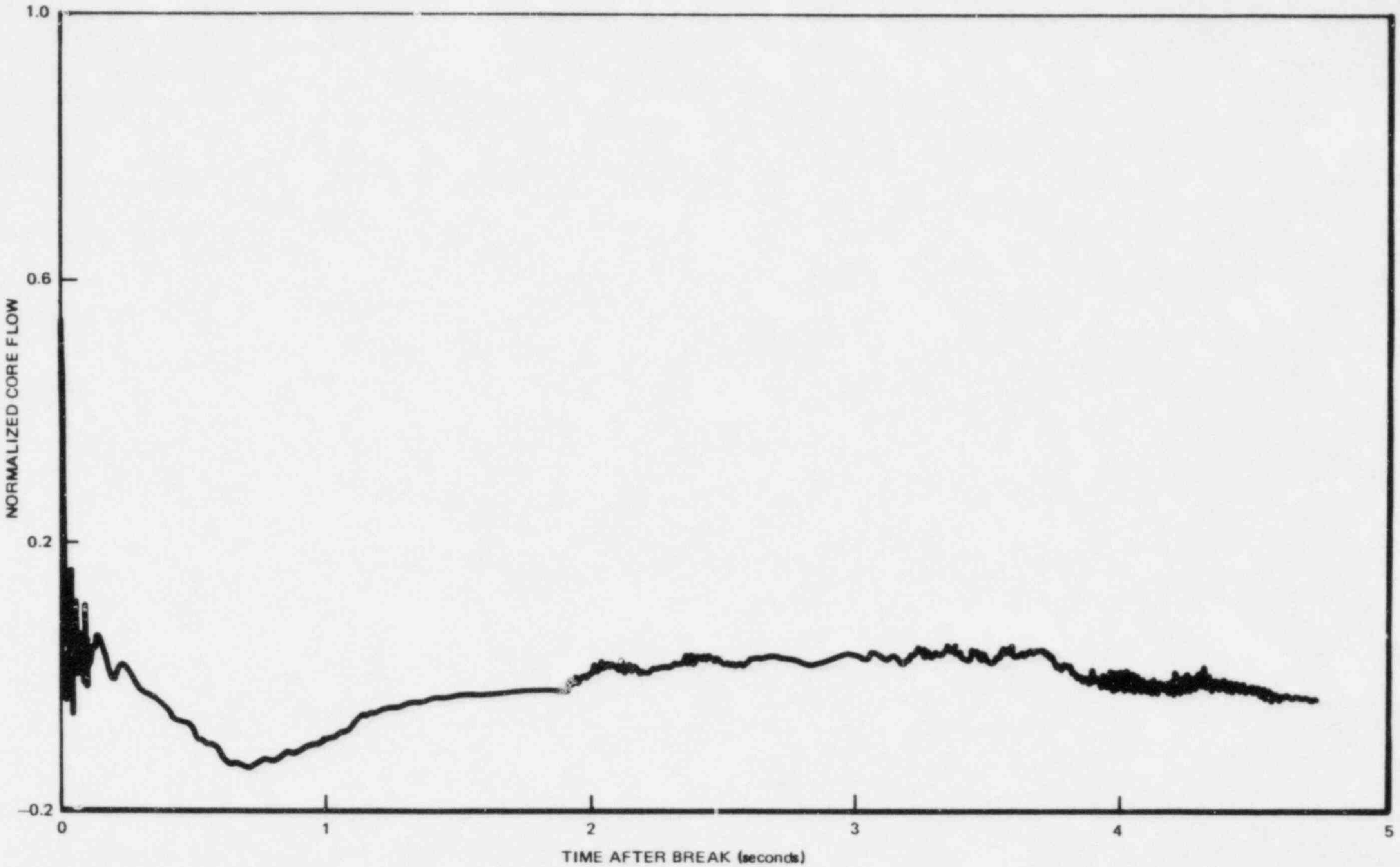


Figure 5-21a. Normalized Core Average Inlet Flow Following a Maximum Recirculation Line Discharge Break (4.66 ft^2)

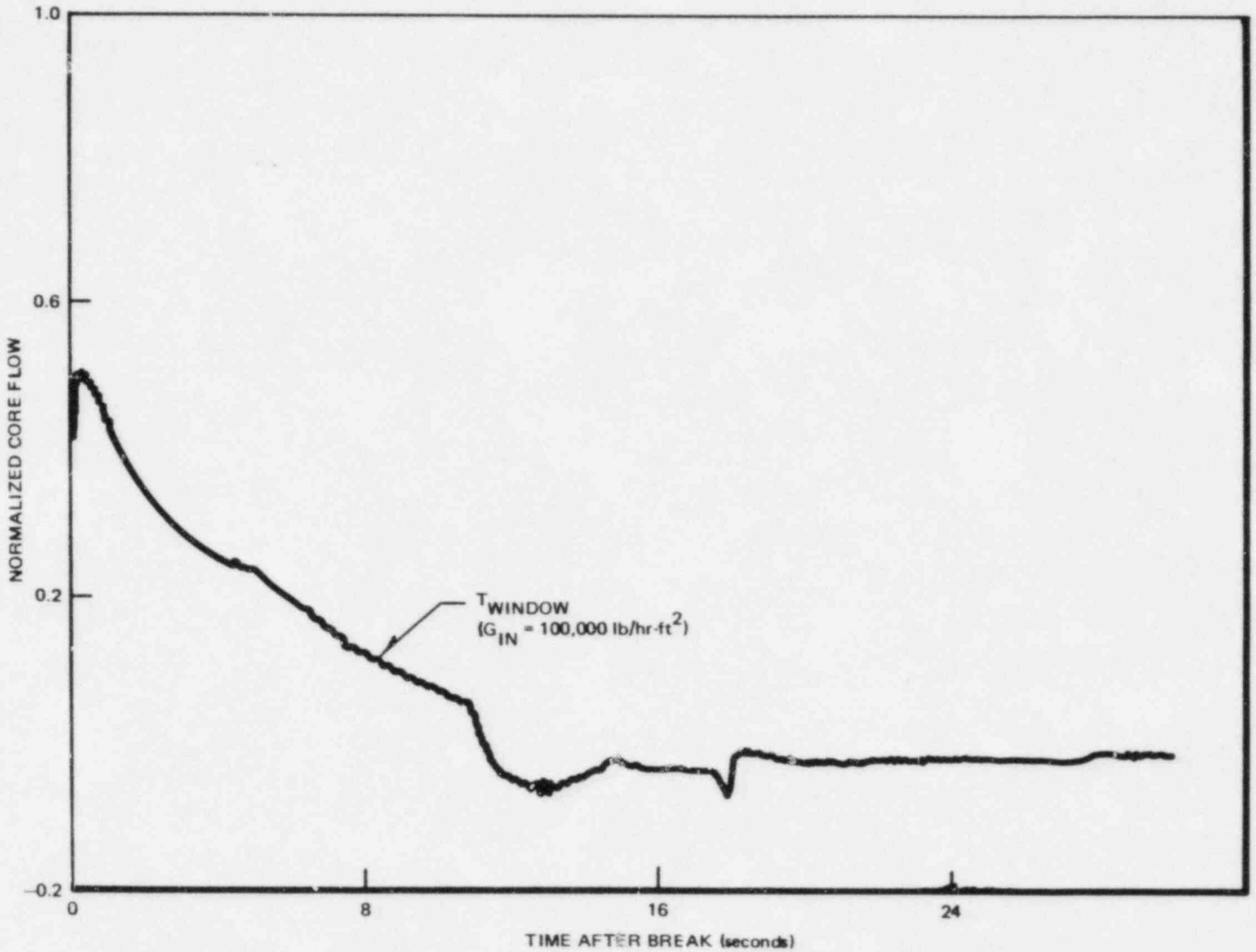


Figure 5-21b. Normalized Core Average Inlet Flow Following a Maximum Recirculation Line Discharge Break (1.0 ft^2)

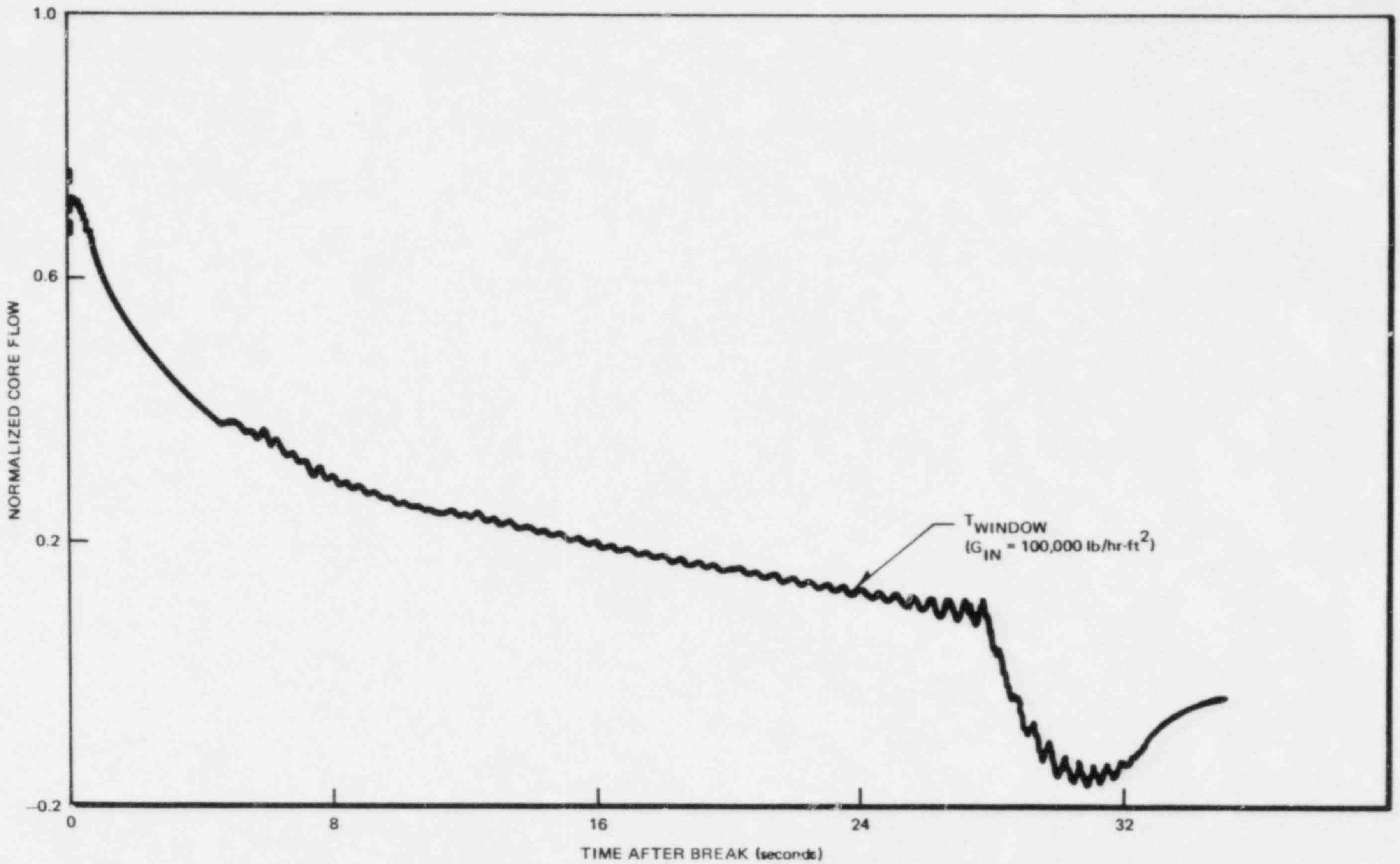


Figure 5-21c. Normalized Core Average Inlet Flow Following a Maximum Recirculation Line Discharge Break (0.3 ft^2)

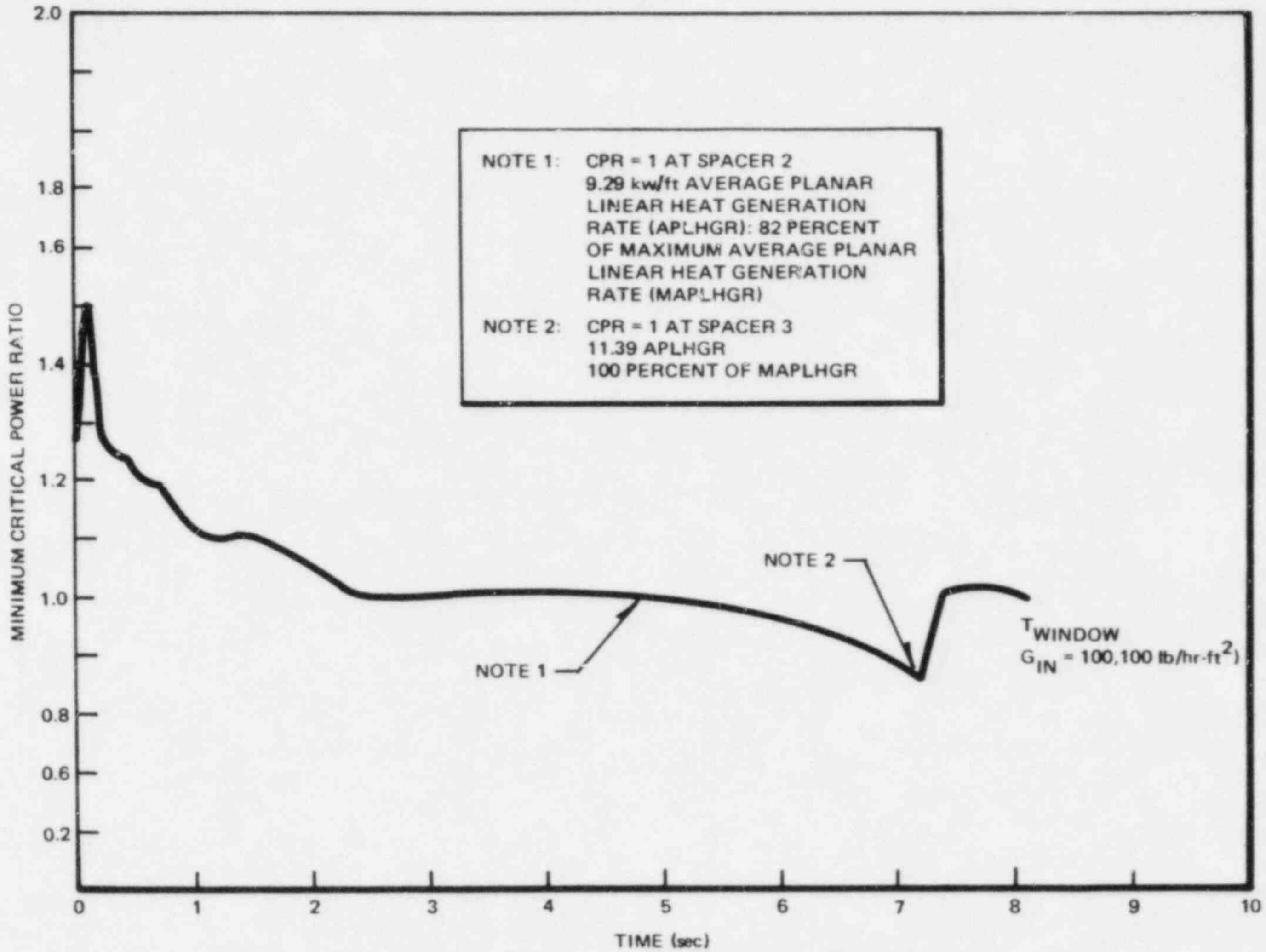


Figure 5-22a. Minimum Critical Power Ratio Following a 1.0 ft^2 Recirculation Line Discharge Break

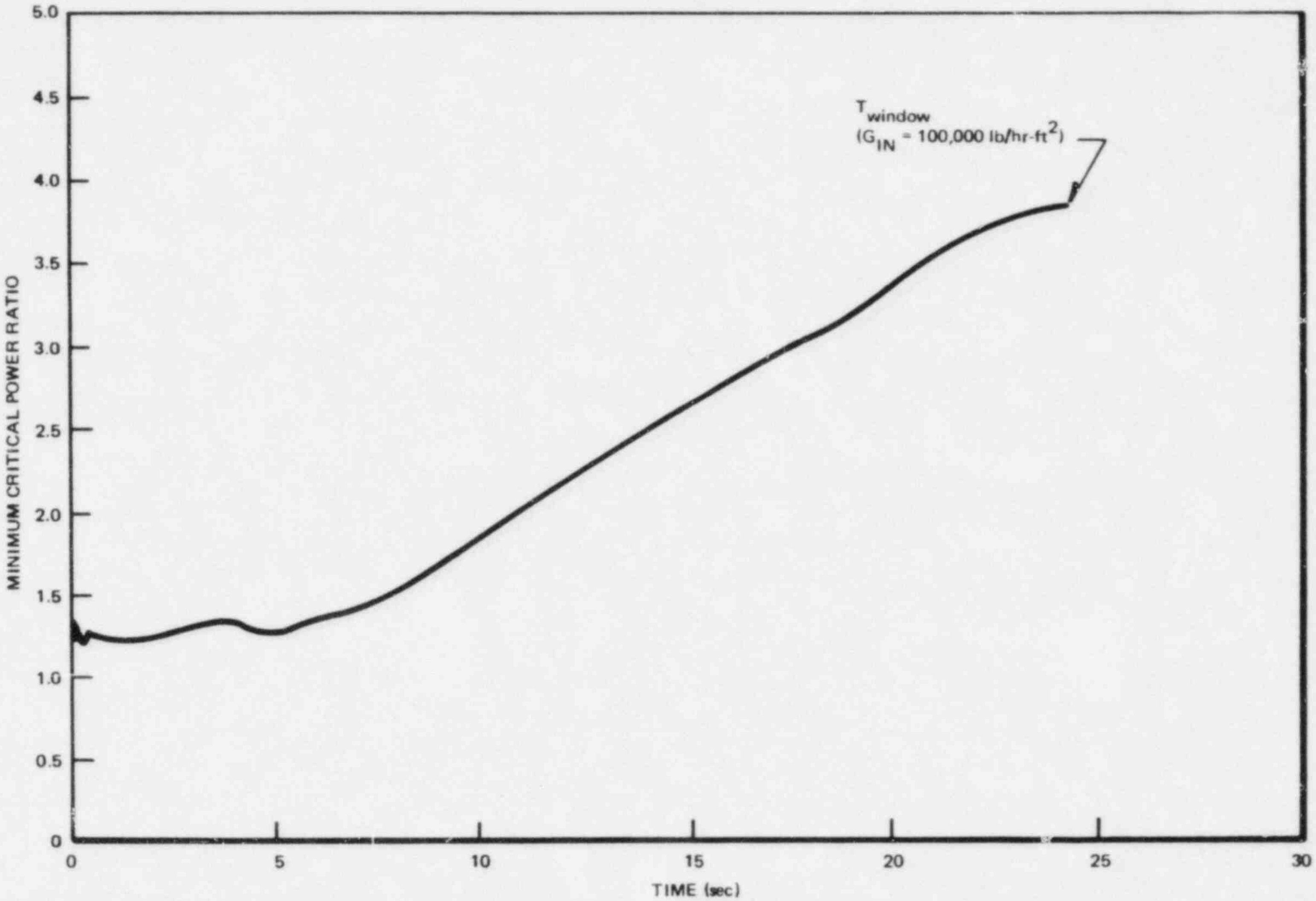


Figure 5-22b. Minimum Critical Power Ratio Following a 0.3 ft^2 Recirculation Line Discharge Break

2/83

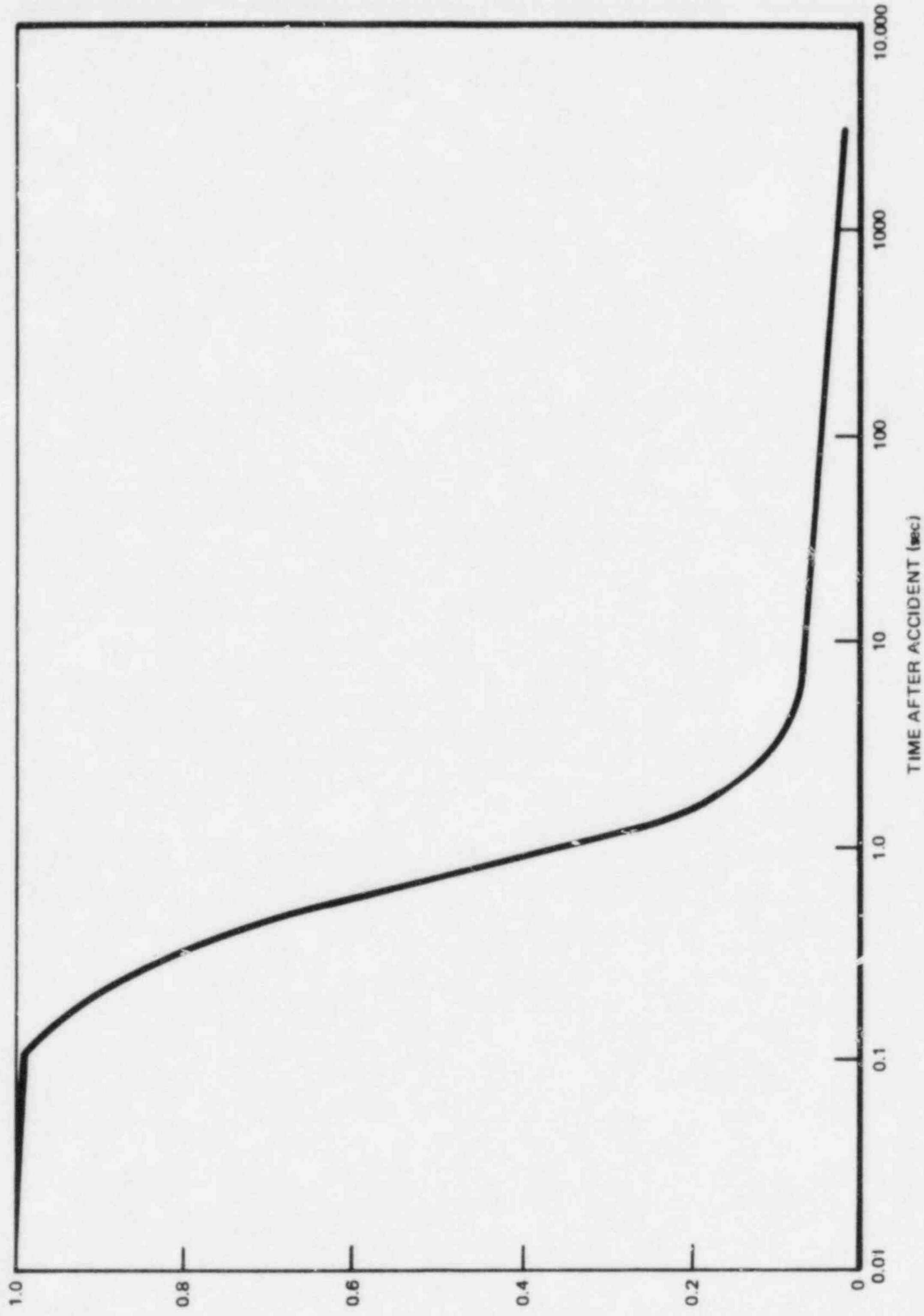


Figure 5-23. Normalized Power Versus Time

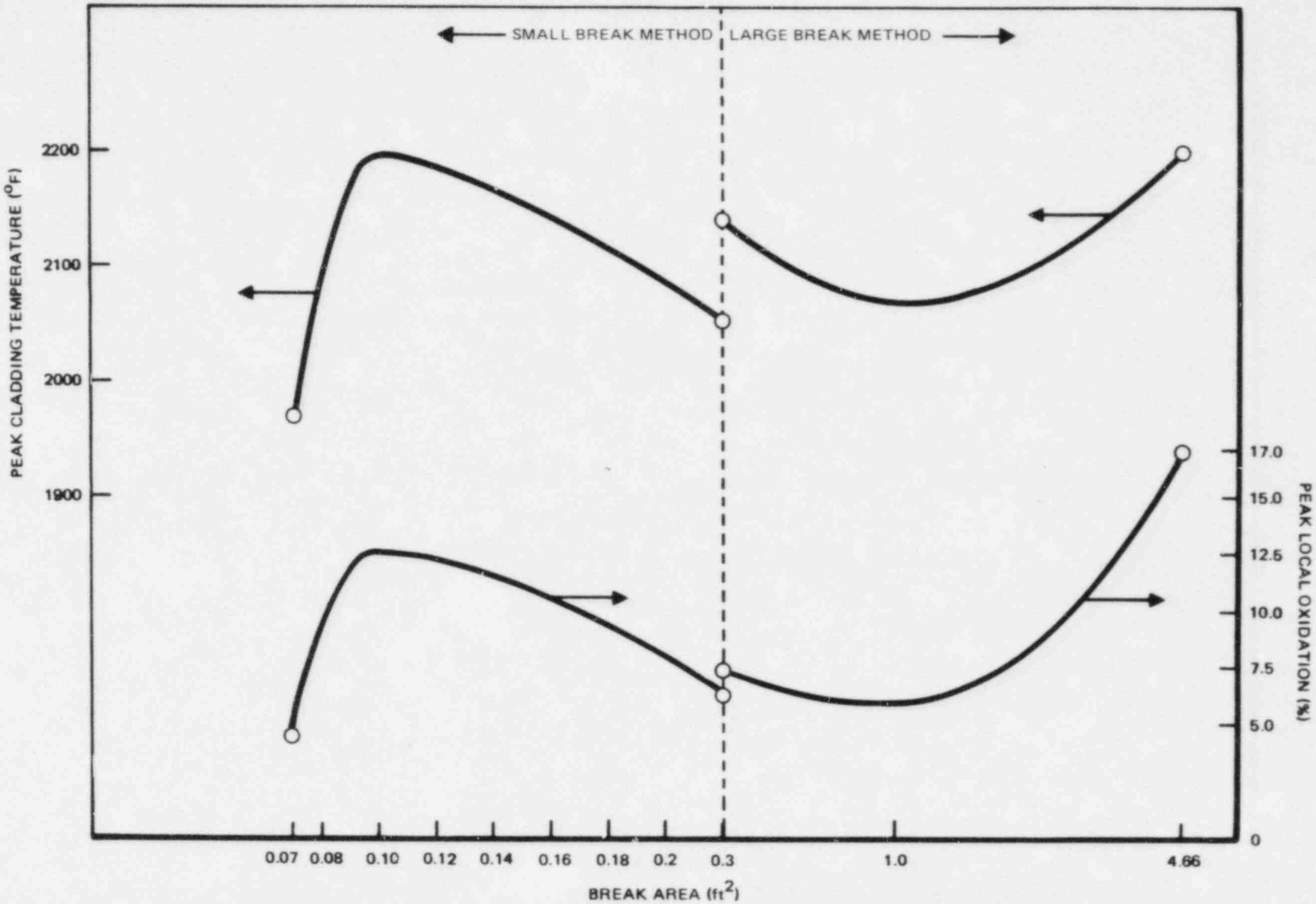
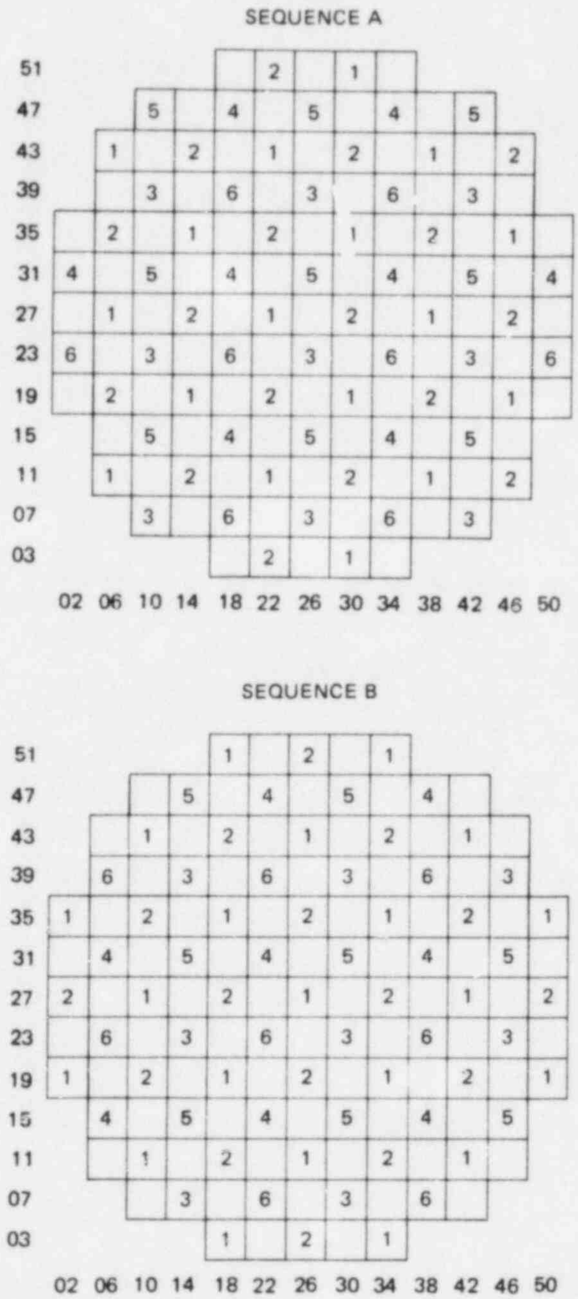


Figure 5-24. Peak Cladding Temperature and Local Peak Oxidation Versus Break Area, Recirculation Discharge Break, Emergency Condenser Failure



NOTE:

The maximum rod worth is determined by the Rod Drop Analysis, and the RWM is programmed to assure the 280 cal/gm limit is met. These groups may be subdivided to assure compliance with this limit.

Figure 5-25. Oyster Creek Control Rod Withdrawal Sequences 2/83

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APPENDIX A

OYSTER CREEK
REFERENCE CYCLE SUPPLEMENT

August 1980

Revised February 1983

2. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 and 3.3.2.1.2)

BOC k_{eff}	
Uncontrolled	1.111
Fully Controlled	0.946
Strongest Control Rod Out	0.987
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.000

3. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.043

4. TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 and 5.2)

	EOC
Void Coefficient N/A* (¢/% Rgo)	-6.48 / -8.10
Void Fraction (%)	35.92
Doppler Coefficient N/A (¢/°F)	-0.222/ -0.211
Average Fuel Temperature (°F)	1184
Scram Worth N/A ($)	-37.64 /-30.11
Scram Reactivity vs Time	Figure 2

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5. GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Fuel Design	EOC P8x8R	Ex8
Peaking factors (local, radial and axial)	1.20 1.711 1.40	1.28 1.624 1.40
R-Factor	1.051	1.098
Bundle Power (MWt)	5.744	5.462
Bundle Flow (10^3 lb/hr)	87.89	89.97
Initial MCPR	1.32	1.30

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*N = Nuclear Input Data
A = Used in Transient Analysis

6. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure (MWd/t)	$\hat{\phi}$ (% NBR)	Q/A (%)	ΔCPR P8x8R	ΔCPR Ex8	Figure
Turbine Trip without Bypass	EOC	517	119	0.25	0.23	Figure 3
Loss of 100°F Feedwater Heating	BOC to EOC	116	115	0.13		Figure 4
Feedwater Controller Failure	EOC	297	116	0.18	0.16	Figure 5

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7. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (5.2.1)

Limiting Rod Pattern: Figure 6

Reactor Power (%)	Rod Position (Feet Withdrawn)	ΔCPR P8x8R	MLHGR (kW/ft) P8x8R
104	4.0	0.08	16.56
105	5.5	0.14	16.62
106*	6.0	0.16	16.62
107	9.0	0.18	16.62
108	9.5	0.18	16.62
109	10.0	0.18	16.62

*Indicates APRM rod block setpoint selected.

8. CYCLE MCPR VALUES (5.2)

Non-pressurization Events

Exposure Range: BOC to EOC	P8x8R
Loss of Feedwater Heater	1.20
Fuel Loading Error	1.25
Rod Withdrawal Error	1.23

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Pressurization Events

Exposure Range: BOC to EOC

	Option A		Option B	
	P8x8R	Ex8	P8x8R	Ex8
Turbine Trip w/o Bypass	1.38	1.36	1.33	1.31
Feedwater Controller Failure	1.30	1.28	1.23	1.21

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9. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (No Scram)	1271	1320	Figure 7

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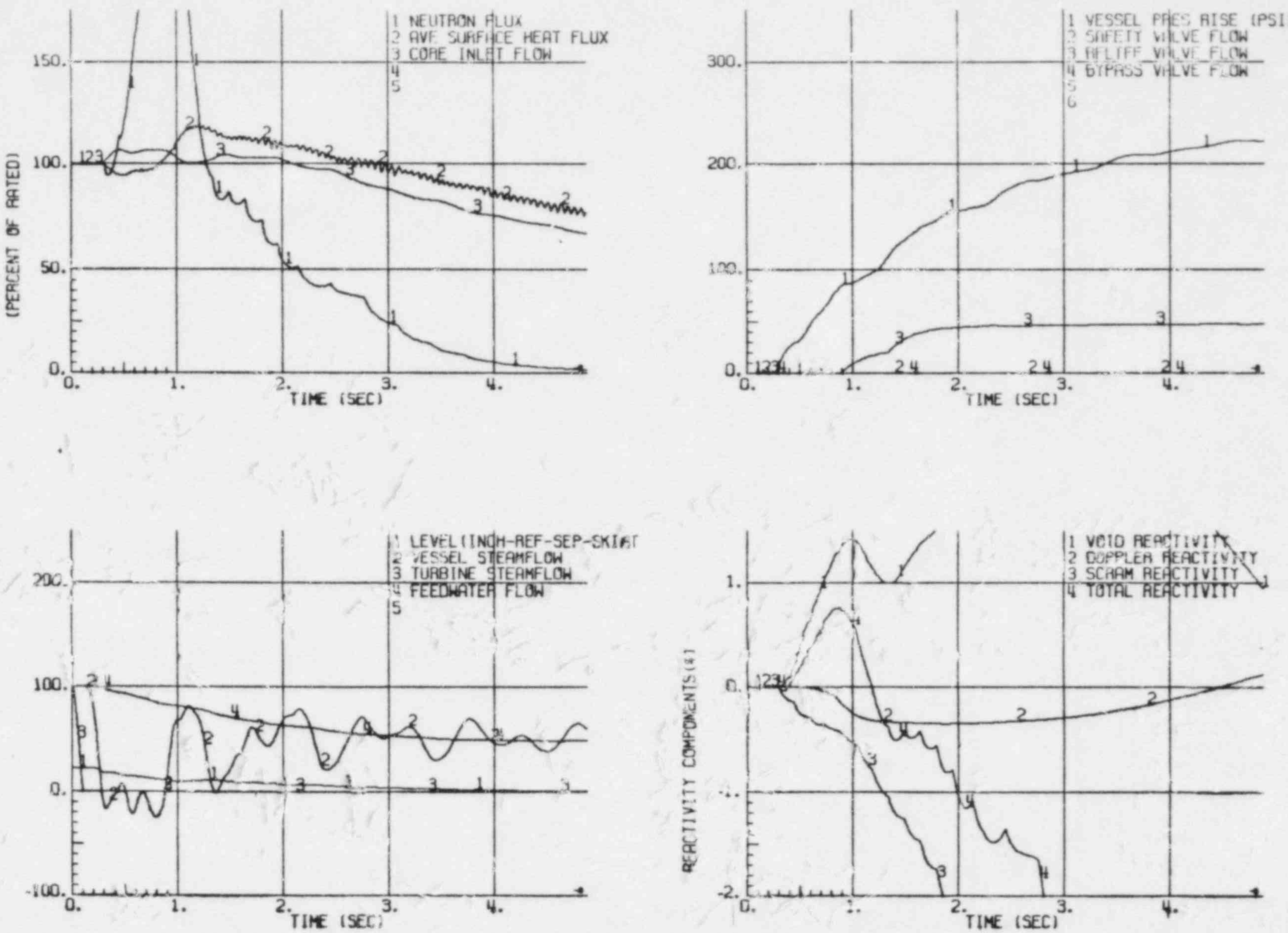

Figure 3. Plant Response to Turbine Trip without Bypass at EOC

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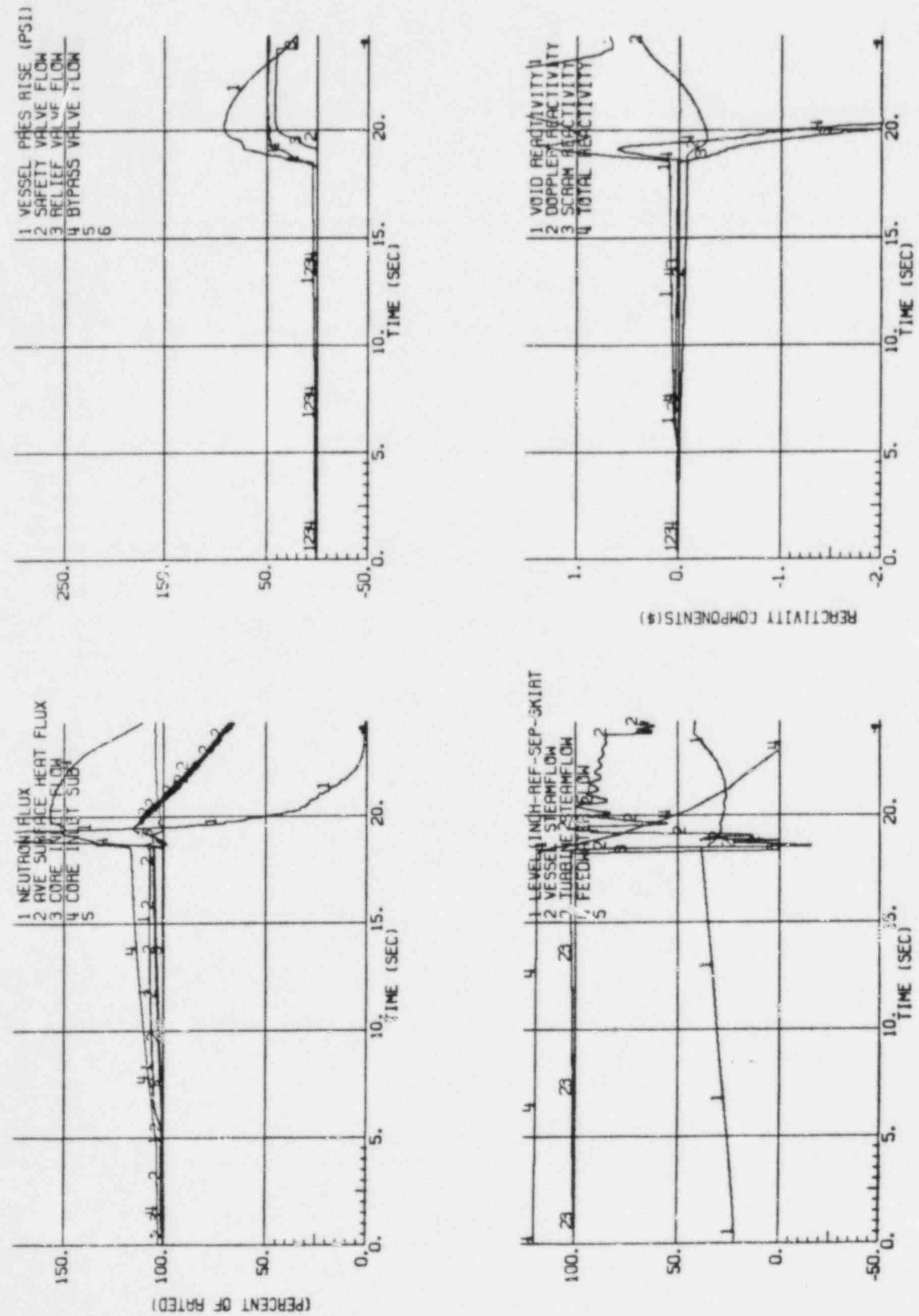

Figure 5. Plant Response to Feedwater Controller Failure at EOC

2/83

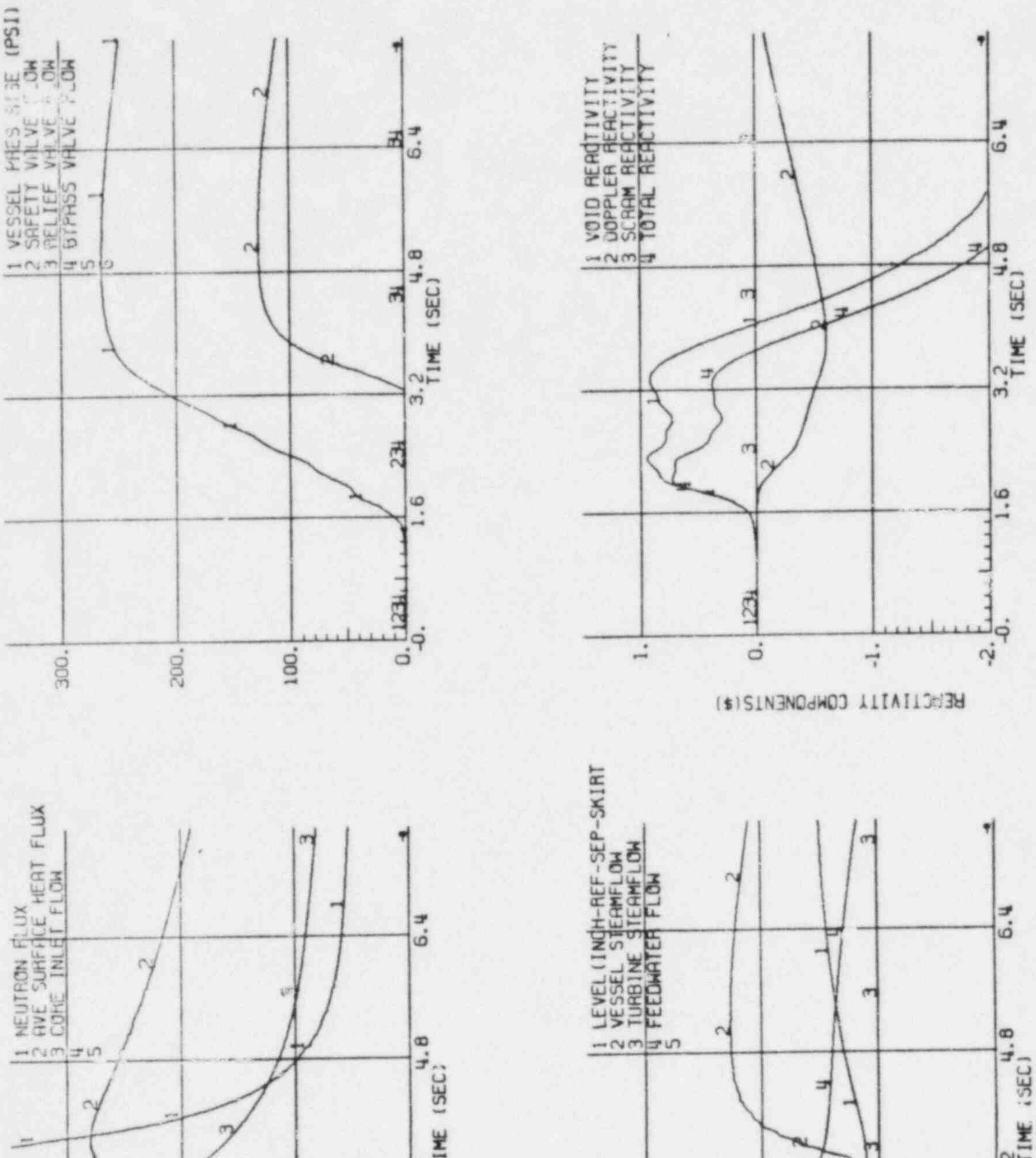

Figure 7. Plant Response to MSIV Closure

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APPENDIX B
THERMAL HYDRAULIC MODEL FOR
NON-GE FUEL
(EXXON 8x8 TYPE VB)

General Electric has developed a thermal hydraulic model for non-GE fuel (Exxon 8x8 Type VB) for use in transient analyses with a mixed core of GE and non-GE fuel. The thermal hydraulic model for the non-GE fuel design is based on the geometry of the non-GE fuel and pressure drop data for the non-GE fuel. Other thermal hydraulic characteristics of non-GE fuel were assumed to be identical GE fuel. The geometry of the non-GE fuel is tabulated in Table B-1 and the thermal hydraulic model assumptions are listed in Table B-2.

The data and assumptions used to develop the model were provided by GPU Nuclear. General Electric assumes no responsibility in determining the appropriateness of the modeling of the non-GE fuel, the applicability of GE methods to non-GE fuel or the usefulness of these results.

The transient results for non-GE fuel are reported in Appendix A along with the GE fuel. The calculation methods used are described in Section 5.

Table B-1

EXXON 8x8 FUEL*

Geometrical Dimensions

1. Fuel Cladding Dimensions	
a. Outside Diameter	0.5015 in.
b. Inside Diameter	0.4295 in.
2. Non-Fueled Rods	
a. Outside Diameter	0.5015 in.
b. Inside Diameter	N/A
c. Water Flow Rate at Rated Conditions	N/A
3. Channel Dimensions	
a. Inside Dimension	5.278 in.
b. Wall Thickness	0.080 in.
c. Corner Radius	0.380 in.
4. Fuel Pellet Diameter	0.4195 in.
5. Fuel Column Length	144 in.
6. Spacer Axial Location	
• Total 7 spacers	

*Oyster Creek FDSAR Amendment 76

Table B-2

THERMAL-HYDRAULIC MODEL ASSUMPTIONS

The non-GE bundle was assumed to be identical to a GE 8x8 bundle with a Mod 1 spacer design with respect to the following:

1. Non-spacer local pressure loss coefficients.
2. Friction pressure loss coefficients.
3. Two-phase multipliers.
4. Exposure dependent bypass leakage fraction.
5. Exposure dependent surface crudding.

The non-GE bundle was assumed to differ from a GE 8x8 bundle with a Mod 1 spacer design with respect to:

1. Bundle geometry.
2. Spacer loss coefficient.

The non-GE bundle was modeled with a spacer loss coefficient of 0.31, which was calculated to best fit the bundle pressure drop data supplied by GPU Nuclear.