D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the primary sensor to verify the proper instrument channel response, alarm, and/or initiating action.

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, accuracy, and response time to a known value (s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip. Response time is not part of the routine instrument calibration but will be checked once per cycle.

G. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

H. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

I. Minimum Critical Heat Fluz and Power Ratios

1. Minimum Critical Heat Flux Ratio (MCHFR) - The lowest in-core ratio of critical heat flux (that heat flux which results in transition boiling) to the actual heat flux.

2. Minimum Critical Power Ratio (MCPR) - The lowest in-core ratio of critical power (that power which causes some point in the assembly to experieince the onset of transition boiling) to the bundle power.

J. Mode - The reactor mode is that which is established by the mode-selector switch.

K. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

L. Operating - Operating means that a system or component is performing its required functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
I. Recirculation System	I. Recirculation System
1. Except as specified in 3.5.I.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.	1. Once per month, when irriated fuel is in the reactor with reactor coolant temperature greate. than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cros tie valves cannot be opened using the normal control switch.
2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.	2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

Bases Continued 3.5:

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Deleted

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq. ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed with two pump operation to reduce the potential break area.

The cross tie valve is allowed to be open during one pump operation. With only one pump, rated power cannot be achieved. Under these conditions, the expected peak clad temperature during a loss of coolant accident is less than that for two pump operation with the cross tie valve closed.

Bases 4.5:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel, which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation will initially be functionally tested once per month until a trend is established and thereafter according to Figure 4.1 (see Section 3.1/4.1) with an interval not greater than three months. The pumps and motor-operated valves are tested each month to assure their operability. The combination of a simulated automatic actuation test each refueling cycle and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

3.0 LIMITING CONDITIONS FOR OPERATIONS	4.0 SURVEILLANCE REQUIREMENTS

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11-1. If at any time during steady state power operation it is determined that the limiting value for APLHGR is being exceeded, action shall be taken immediately to restore operation to within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

B. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_{d} \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{LT}\right) \right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft for 7x7 fuel

= 13.4 kw/ft for 8x8 fuel

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= 0.026 for 7x7 fuel

= 0.021 for 8x8 fuel

LT = Total core length = 12 ft

L = Axial position above bottom core

If at any time during steady state power operation it is determined that the limiting value of LHGR is being exceeded, action shall be taken immediately to restore operation to within prescribed limits.

B. Local LHGR

The local LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% of rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be ≥ 1.19. If at any time during steady state power operation it is determined that the limiting value for MCPR is being exceeded, action shall be taken immediately to restore operation within the prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined weekly during reactor power operation at ≥ 25% rated thermal power.

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

It is recognized that APLHGR is a calculated parameter that is not continually monitored and alarmed directly during core power distribution changes. If at the time of the calculation it is found that the limits are being exceeded, there is always an action which will return the average planar LHGR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Therefore, the only way to have a reportable Abnormal Occurrence is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the average planar LHGR to within prescribed limits.

B. Local LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and axial gaps between core bottom and top and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

It is recognized that LHGR is a calculated parameter that is not continually monitored and alarmed directly during core power-distribution changes. If at the time of the calibration it is found that the limits are being exceeded, there is always an action which will return the LHGR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Therefore, the only way to have a reportable Abnormal Occurrence is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the LHGR to within prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the worst steady state MCPR to be 1.19. This value is chosen such that the onset of transition boiling is avoided in operational transients. Since lower values of MCPR have not been considered in the ECCS evaluation, operation at that condition is not permitted.

It is recognized that MCPR is a calculated parameter that is not continually monitored and alarmed directly during core power distribution and thermal-hydraulic changes. If at the time of the evaluation it is found that the limits are being exceeded, there is always an action which will return the MCPR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Therefore, the only way to have a reportable Abnormal Occurrence is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the MCPR to within prescribed limits.

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff)

3. Communication: V A Moore to I S Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Monticello Nuclear Generating Plant Loss-Of-Coolant Accident Analysis Conformance with 10 CFR 50 Appendix K, August 1974," L O Mayer (NSP) to J F O'Leary, August 20, 1974.

The APLHGR and local LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences.

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this condition, plant operating experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The weekly requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The APLHGR and local LHGR are more sensitive to local power changes than MCPR. They will also generally reach limiting values prior to MCPR reaching its limiting value. In this way APLHGR and local LHGR serve as early indicators of the MCPR behavior. Therefore, it is consistent to require a calculation of APLHGR and local LHGR on a more frequent basis than MCPR.

Figure 3.11.1-A

Limiting Average Planar Linear Heat Generation Rate Versus Planar Average Exposure - Monticello Initial Core 7x7 Fuel

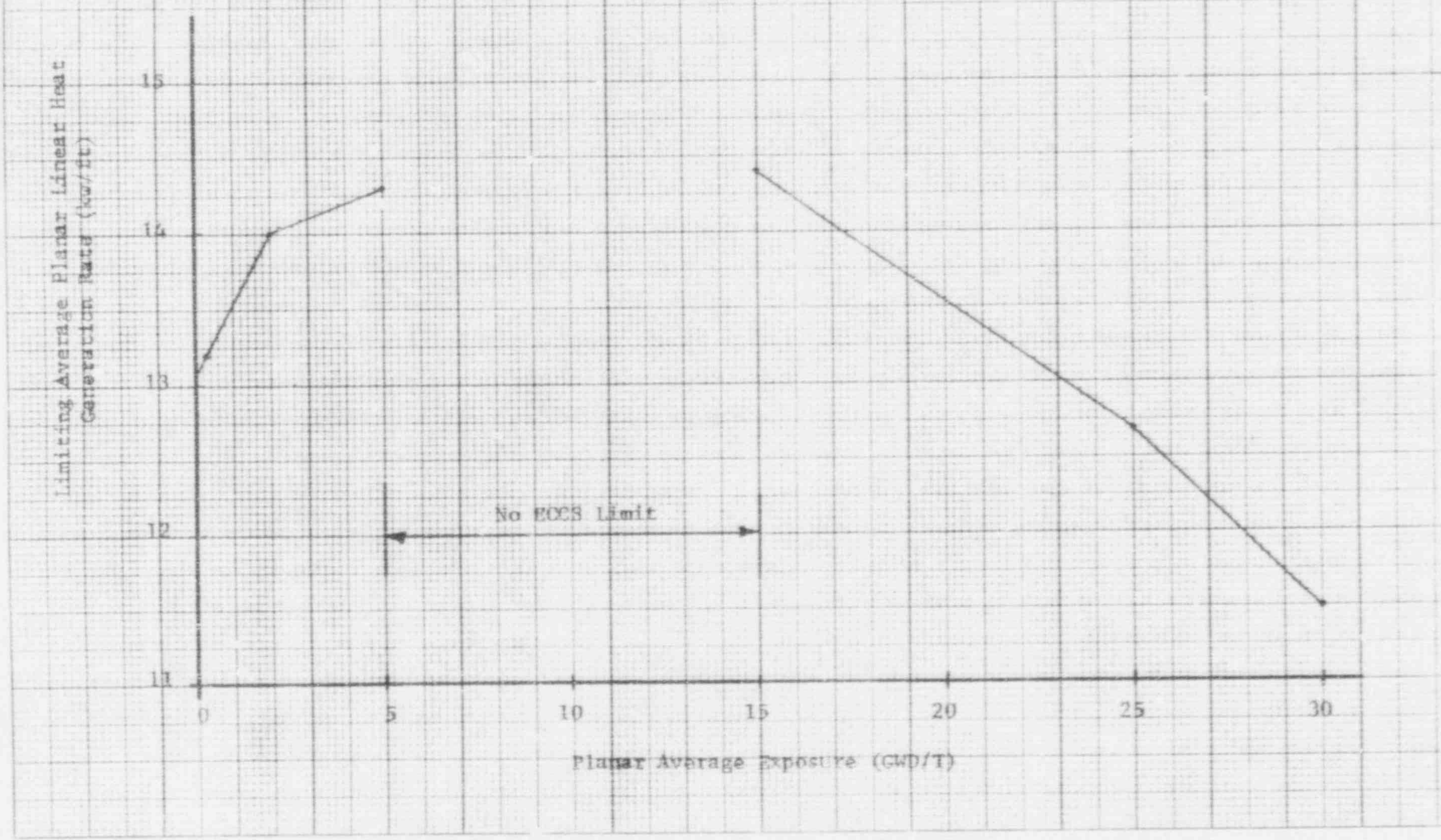


Figure 3.11.1-B

Limiting Average Planar Linear Heat Generation Rate Versus Planar Average Exposure - Monticello Reload 1 (Generic B) 7x7 Fuel

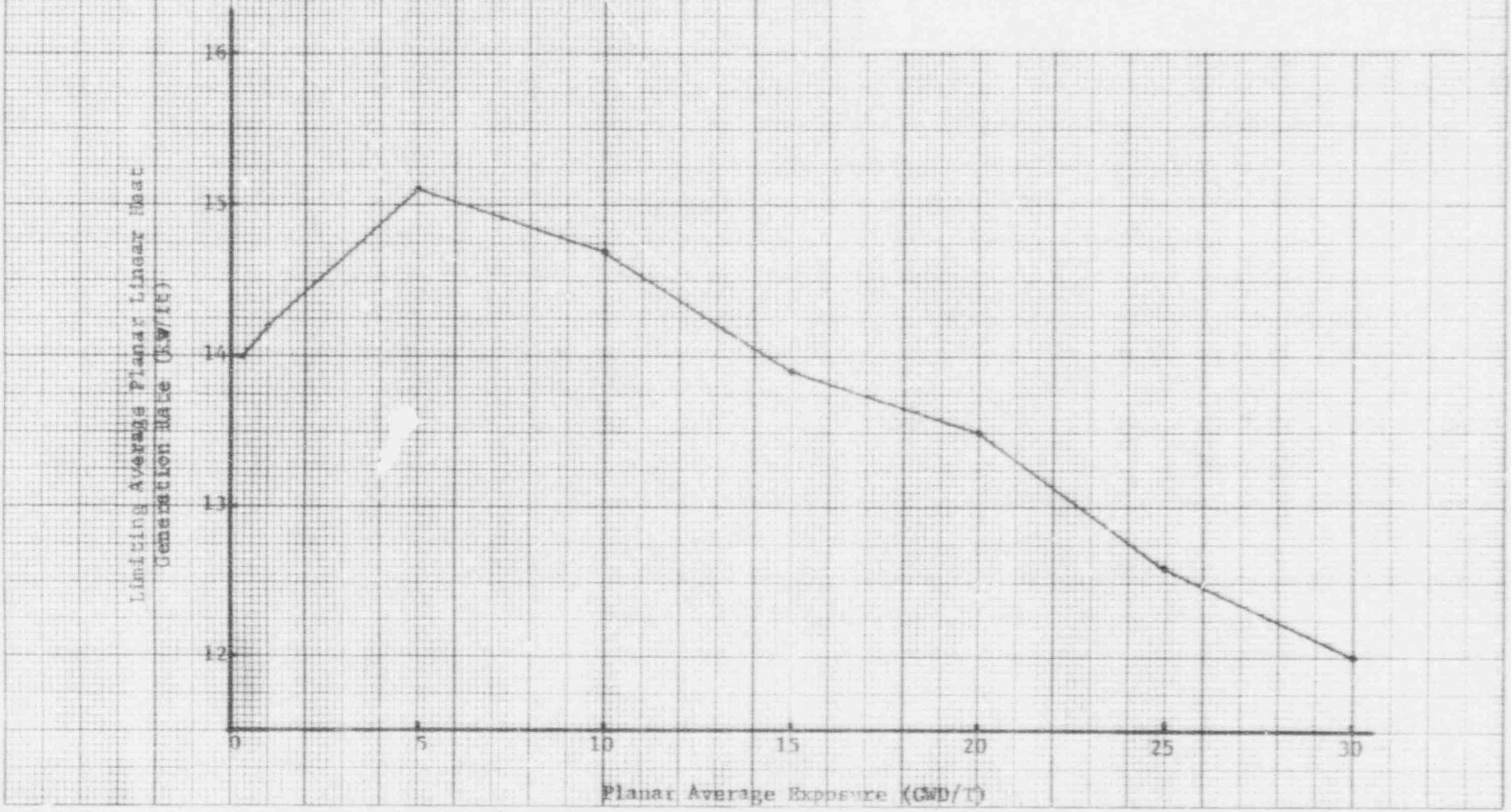


Figure 3.11.1-C

Limiting Average Planar Linear Heat Generation Rate Versus Planar Average Exposure - Monticello Reload 2 (8D262) Fuel

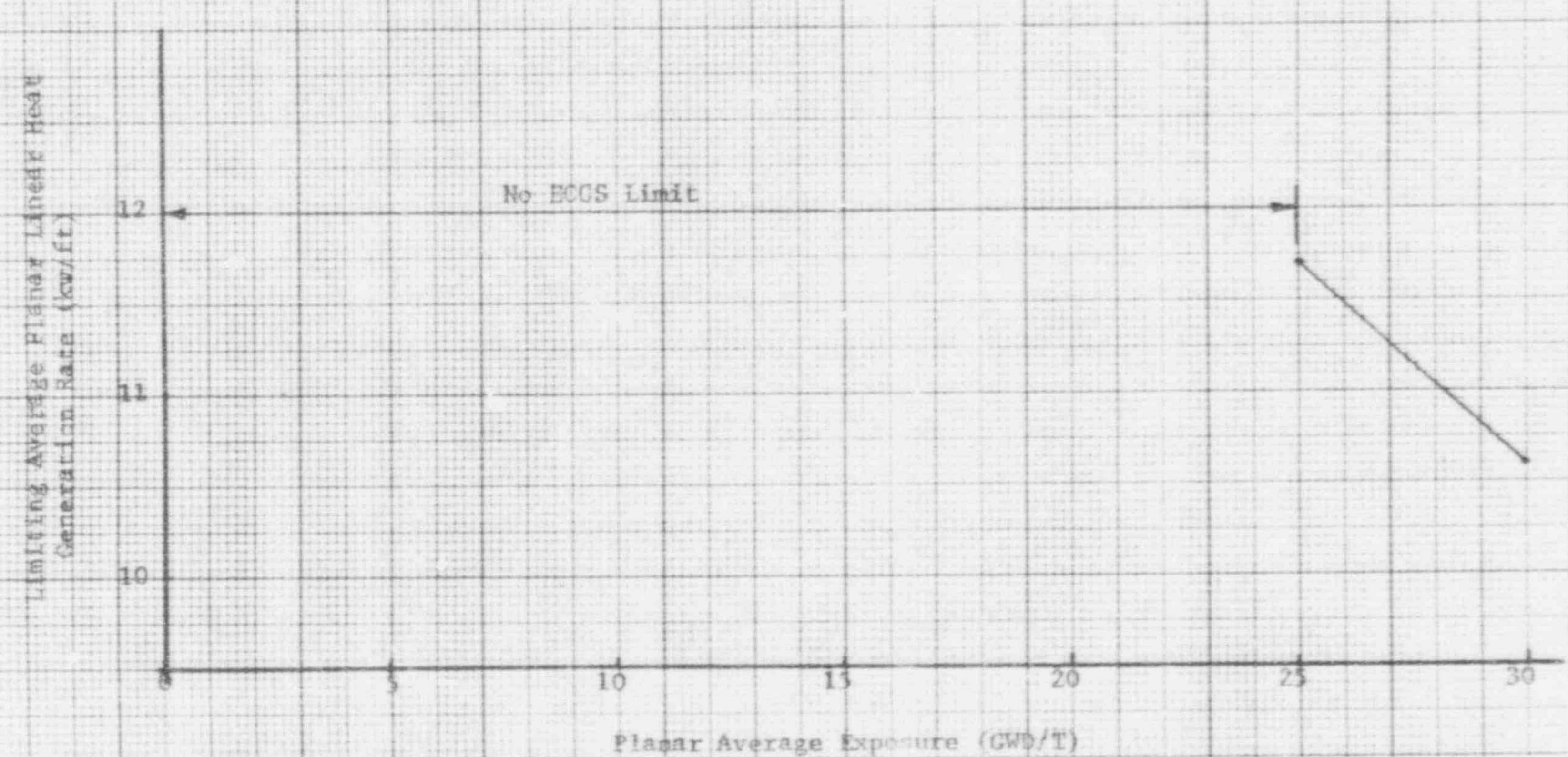


EXHIBIT C

MONTICELLO NUCLEAR GENERATING PLANT
LOSS-OF-COOLANT ACCIDENT ANALYSES
CONFORMANCE WITH 10CFR 50
APPENDIX K

AUGUST 1974

Discussion

Presented in the following document are the results of the loss-of-coolant accident analysis of the Monticello Nuclear Power Station. The analysis was performed using General Electric calculational models which are consistent with the requirements of Appendix K of 10 CFR part 50. A complete discussion of each code employed in the analysis is presented in Section II of Reference 1.

Input to the Analysis

A list of the significant plant input parameters to the loss-of-coolant accident 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 % Licensed core power

Vessel Steam Output . . . . $6,913 \times 10^6$ Lbm/h which corresponds to 102 % of Licensed core power

Vessel Steam Dome Pressure . . . . 1040 psia

Design Basis Recirculation Line Break Area . . . . 3.9 ft^2

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
Initial Core	7x7	17.5	1.57	1.19
Reload 1	7x7	17.5	1.57	1.19
Reload 2	8x8	13.4	1.57	1.19

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

Results of the Analysis

The results of the analysis are presented in the order in which they are calculated. The presentation of the results is divided into four major portions according to the model from which the output is obtained. These portions are:

A. Calculated by the Short-Term Thermal-Hydraulic Model (LAMB)

B. Calculated by the Transient Critical Power Model (SCAT)

C. Calculated by the Long-Term Thermal-Hydraulic Model (SAFE)

D. Calculated by the Core Heatup Model (CHASTE)

A summary of the results is presented in Table 2. At the MAPLHGR* employed in the analysis, the most severe pipe break yields a calculated peak cladding temperature less than or equal to 2200°F, a calculated maximum local metal-water reaction less than or equal to 17% and a calculated core-wide metal-water reaction less than or equal to 1%. Compliance with the 10CFR50.46 criteria for coolable geometry and long-term cooling has been shown in Reference 1. The reactor is therefore fully in conformance with 10CFR50.46 and Appendix K with operations at the MAPLHGR used in the analysis. Details of the MAPLHGR values used as a function of fuel exposure are given in Figure(s) D5 for each fuel type in the reactor. These values, if more limiting than other design parameters, represent limits for operation to ensure conformance with 10CFR50.46 and Appendix K.

*Maximum (Bundle) Average Planar Linear Heat Generation Rate

Table 2

SUMMARY OF RESULTS OF THE MONTICELLO NUCLEAR POWER STATION APPENDIX K LOSS-OF-COOLANT ACCIDENT ANALYSIS

Single Failure	Design Basis Break PCT(°F)	Maximum Metal-Water Reaction Core Average	Maximum Metal-Water Reaction Local	Highest Temperature Intermediate Break PCT(°F)	Highest Temperature Intermediate Break Break Area (ft^2)
LPCI Inj. Valve Failure (a)	2200	0.3 %	7%		
HPCI (b)				1700	0.07

(a) Systems Available - 2CS + HPCI + ADS

A. APPENDIX K SHORT-TERM THERMAL-HYDRAULIC ANALYSIS

General Description of the LAMB Code

The LAMB code is a model which is used to analyze the short-term thermodynamic and thermo-hydraulic behavior of the coolant in the vessel during a postuled loss-of-coolant accident. In particular, LAMB predicts the core flow, core inlet enthalpy and core pressure during the blowdown prior to the end of lower plenum flashing (~ 20 seconds). For a detailed description of the model and a discussion regarding sources of input to the model refer to the "LAMB Code Documentation" portion of Section II of Reference 1.

Results of LAMB Analysis

Presented in this section are results of the loss-of-coolant accident analysis which are calculated by LAMB. These results include:

Parameter	Figure
Core Average Inlet Flow Rate (Normalized to unity at the beginning of the accident) following a Design Basis Accident	A-1
Core Inlet Enthalpy following a Design Basis Accident	A-2
Core Average Pressure following a Design Basis Accident	A-3

These results are input to the SCAT code discussed in Section B.

B. APPENDIX K TRANSIENT CRITICAL POWER ANALYSIS

General Description of the SCAT Code

The SCAT code is used to evaluate the short-term thermal-hydraulic response of the coolant in the core during a postulated loss-of-coolant accident. In particular the convective heat transfer process in the thermally limiting fuel bundle is analyzed during the transient. For a detailed description of the model and a discussion regarding sources of input to the model refer to the "SCAT Code Documentation" portion of Section II of Reference 1.

Results of the SCAT Analysis

Presented in this section are results of the loss-of-coolant accident analysis which are calculated by SCAT. These results include:

Parameter	Figure
Minimum Critical Power Ratio following a Design Basis Accident	B-1
Convective Heat Transfer Coefficient following a Design Basis Accident	B-2

These results are used as input to the CHASTE code discussed in Section D.

C. APPENDIX K LONG TERM THERMAL-HYDRAULIC ANALYSIS

General Description of SAFE Code

The SAFE code is a model which is used to analyze the long-term thermodynamic behavior of the coolant in the vessel during both small and large breaks. Since the calculational procedure of the loss-of-coolant accident analysis differs depending on whether or not a break is classified as "small" or "large", it is appropriate to distinguish between the two classifications of breaks. A small break is defined as that size break for which nucleate boiling heat transfer exists in the core until the heat fluxes are below the pool boiling critical power condition. This occurs approximately 20 to 25 seconds after the break. For small breaks, core heatup is therefore based solely on the uncovery and recovery of the fuel and the duration of spray cooling all of which are predicted by the SAFE core. For the "large" break analysis, the LAMB and SCAT codes are employed to determine the time of boiling transition and the post-boiling-transition convective heat transfer during the blowdown. The SAFE code calculates the uncovery and reflooding of the fuel and the duration of spray cooling.

For a detailed description of the model and a discussion regarding sources of input to the model refer to the "SAFE Code Documentation" portion of Section II of Reference 1.

Results of the SAFE Analysis

Presented in this section are results of the loss-of-coolant accident analysis which are calculated by SAFE. These results include:

Parameter	Figure
Water Level inside the Shroud and Reactor Vessel Pressure following a Design Basis Accident	C-1
Fuel Rod Convective Heat Transfer Coefficient following a Design Basis Accident	C-2
Water Level inside the Shroud and Reactor Vessel Pressure following a Small Break of the Recirculation Line	C-3

D. APPENDIX K CORE HEATUP ANALYSIS

General Description of CHASTE Code

The transient thermal response of the core to a loss-of-coolant accident calculated by CHASTE can generally be broken down into four stages: (1) fuel pin temperature redistribution; (2) fuel rod bundle temperature redistribution; (3) metal-water reaction heatup; and (4) core standby cooling system effects. Phenomena occurring during these stages that are considered in the analysis are described below.

Fuel Pin Temperature Redistribution

Following a reactor shutdown, a large heat source is still available within the core in the form of sensible heat in the fuel. This is represented by the temperature profile in the fuel rod. Initially, the temperature profile is steep because of the

high power generation rates during normal operation. Following the shutdown, the sensible heat in the fuel will be redistributed by thermal conduction within the fuel and cladding and by convection and radiation in the gap between fuel and cladding, with the amount of heat removed being dependent on surface conditions. At the end of three or more fuel time constants (fuel thermal time constant is about 8 to 10 seconds), the radial temperature profile in the fuel pin is almost flat, consistent with the low fission product decay power generation.

Fuel Rod Bundle Temperature Redistribution

As the cladding temperature increases and the core coolant void fraction approaches unity, radiant heat transmission between rods and the channel wall tends to equalize the temperature of all rods at a given axial position. The total energy in the core continues to increase during this period due to continuing fission product decay.

Metal-Water Reaction Heatup

The fuel pin cladding is made of an alloy of Zircaloy, which reacts with steam at high temperatures. The zircaloy-steam checmical reaction rate is exothermic and strongly dependent upon the reaction temperature. The temperature dependence is exponential and the rate of reaction becomes significant at cladding temperatures in the range of 2200°F or higher.

Emergency Core Cooling System (ECCS) Effects

Redundant emergency core cooling systems performance for a given LOCA is dependent upon the conditions of the accident. The core cooling systems will provide sufficient cooling to prevent excessive cladding heatup. The primary purpose of the core heatup analysis is to determine the effectiveness of the emergency core cooling systems.

For a detailed description of the CHASTE model and a discussion regarding sources of input to the model refer to the "CHASTE Code Documentation" portion of Section II of Reference 1.

Results of CHASTE Analysis

Presented in this section are results of the loss-of-coolant accident analysis which are calculated by CHASTE. These results include:

Parameter	Figure
Peak Cladding Temperature following a Design Basis Accident	D-1
Peak Cladding Temperature following a small Break of the Recirculation Line	D-2
Peak Cladding Temperature versus Break Area	D-3
Peak Cladding Temperature versus Planar Average Exposure	D-4
Maximum Average Planar Linear Heat Generation versus Planar Average Exposure	D-5

Figure D-4 shows the calculated peak cladding temperature as a function of

exposure if the fuel bundle is operated at the average planar linear heat generation rate plotted in Figure D-5. For discussion purposes, Figures D-4 and D-5 can be separated into three ranges of average planar exposure:

1. No ECCS Limit

   In many cases, (that is, over a range of planar average exposures), the fuel has the capability to operate at the peak technical specification linear heat generation rate (Table 1) and the calculated peak cladding temperatures during the postulated accident are less than 2200°F. In general, this occurs early in the life of the fuel when the gap conductance is high and the internal fuel rod pressure is low. Over this exposure range, no maximum ALHGR is plotted in Figure D-5 and the fuel can be operated without a power restriction in conformance with Appendix K. The resulting maximum temperatures and maximum oxidation percentages are plotted in Figure D-4.

2. ECCS Limit Applies

   In this exposure range, the calculated temperature would exceed 2200°F if the fuel were operated at the peak technical specification LHGR, so it is necessary to reduce the average planar power to meet Appendix K requirements. The resulting maximum average planar linear heat generation rate (MAPLHGR) is plotted in Figure D-5 and the resulting temperatures (2200°F) and oxidation percentages are shown in Figure D-4.

3. Fuel Depletion Limits Apply

   After the initial cycle of operation, depleted fuel assemblies operate at lower power relative to fresh reload fuel. Consideration of this effect has been included in the GEGAP-III gap conductances, which are input as initial conditions in analysis of the postulated LOCA by employing conservative estimates of the maximum duty of the individual fuel rods in the highest power assembly (see Section 4.2 of Reference 2). For consistency with the GEGAP-III calculation of gap conductance and rod fission gas inventory, the GEGAP-III assumed exposure history for the peak power rod in the bundle was employed to determine the peak LHGR late in life. The average planar power was determined from the peak LHGR and the highest local peaking factor. The average planar power used in the CHASTE calculation is shown in Figure D-5, and the resulting temperatures and oxidation percentages are shown in Figure D-4. The apparent limit imposed by this part of the curve should be of no practical concern because it is improbable that any particular fuel type would be capable of operating at or above the presented power level. The planar power used in the analysis is, however, shown as a limit because a conformance calculation has not been made in excess of that value.

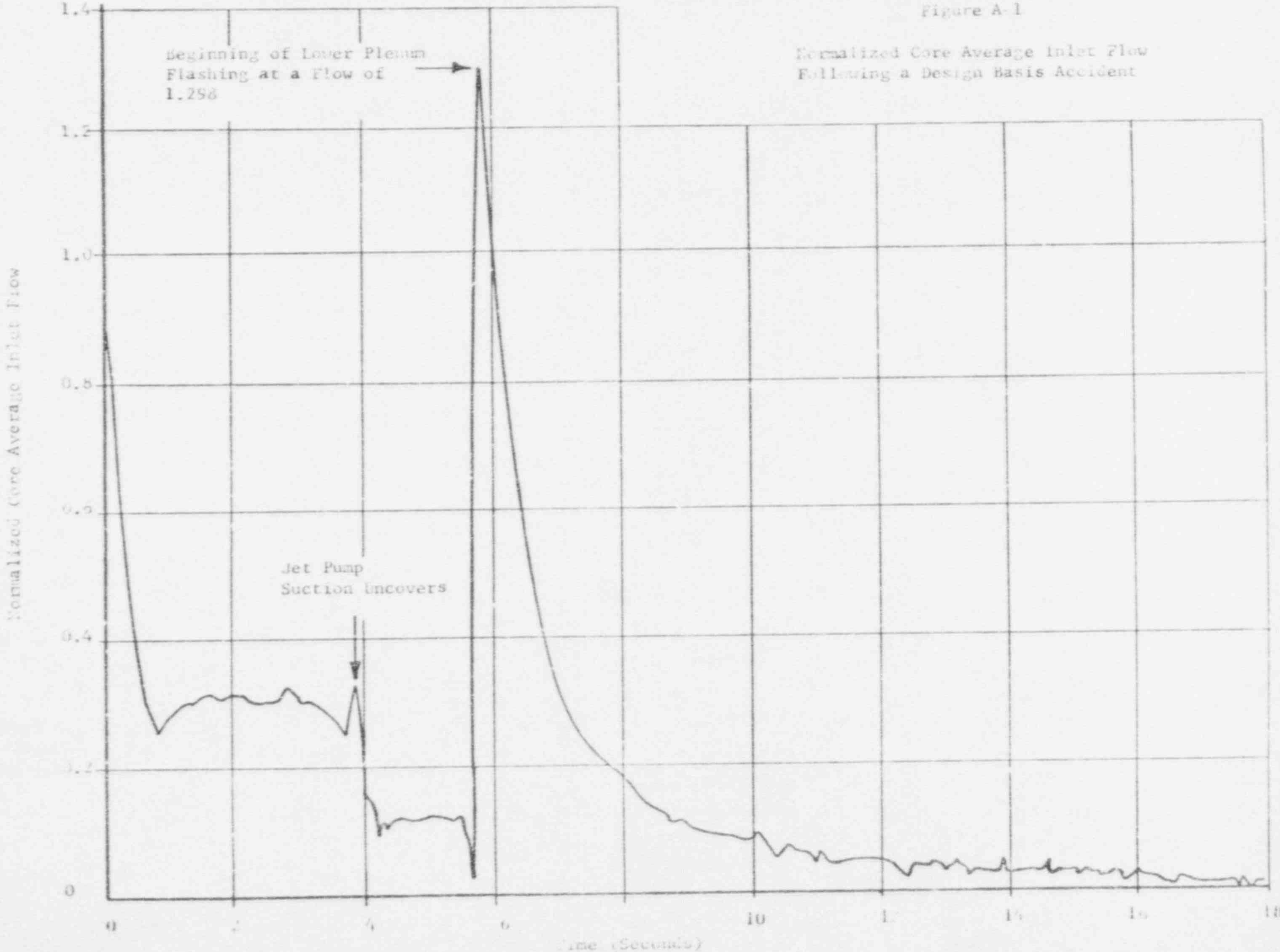
REFERENCES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDO-20566 (to be issued)

2. GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods, NEDO-20181, November 1973.

Figure A-1

Normalized Core Average Inlet Flow Following a Design Basis Accident



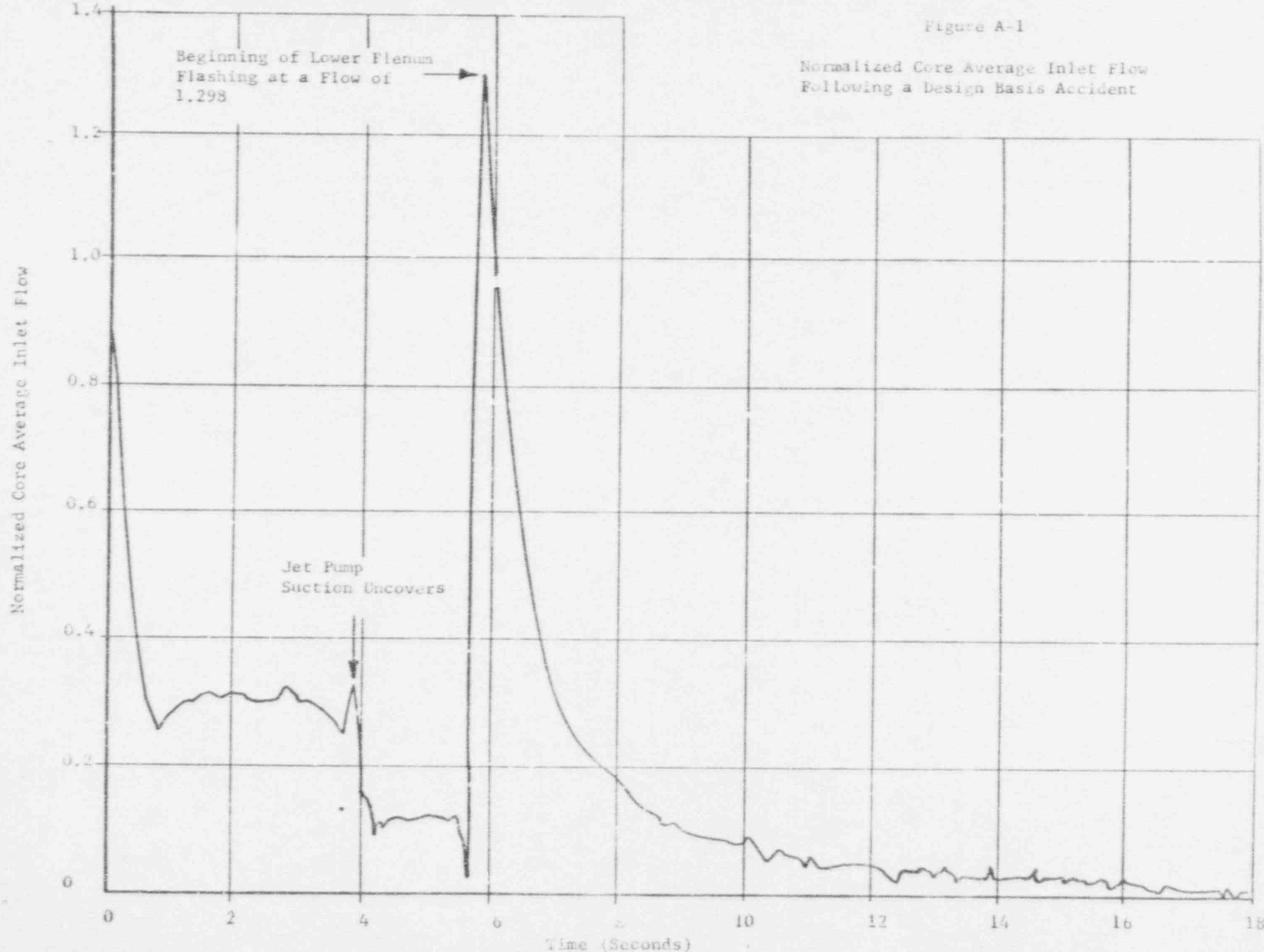


Figure A-1

Normalized Core Average Inlet Flow
Following a Design Basis Accident

Figure A-2

Core Inlet Enthalpy Following
a Design Basis Accident

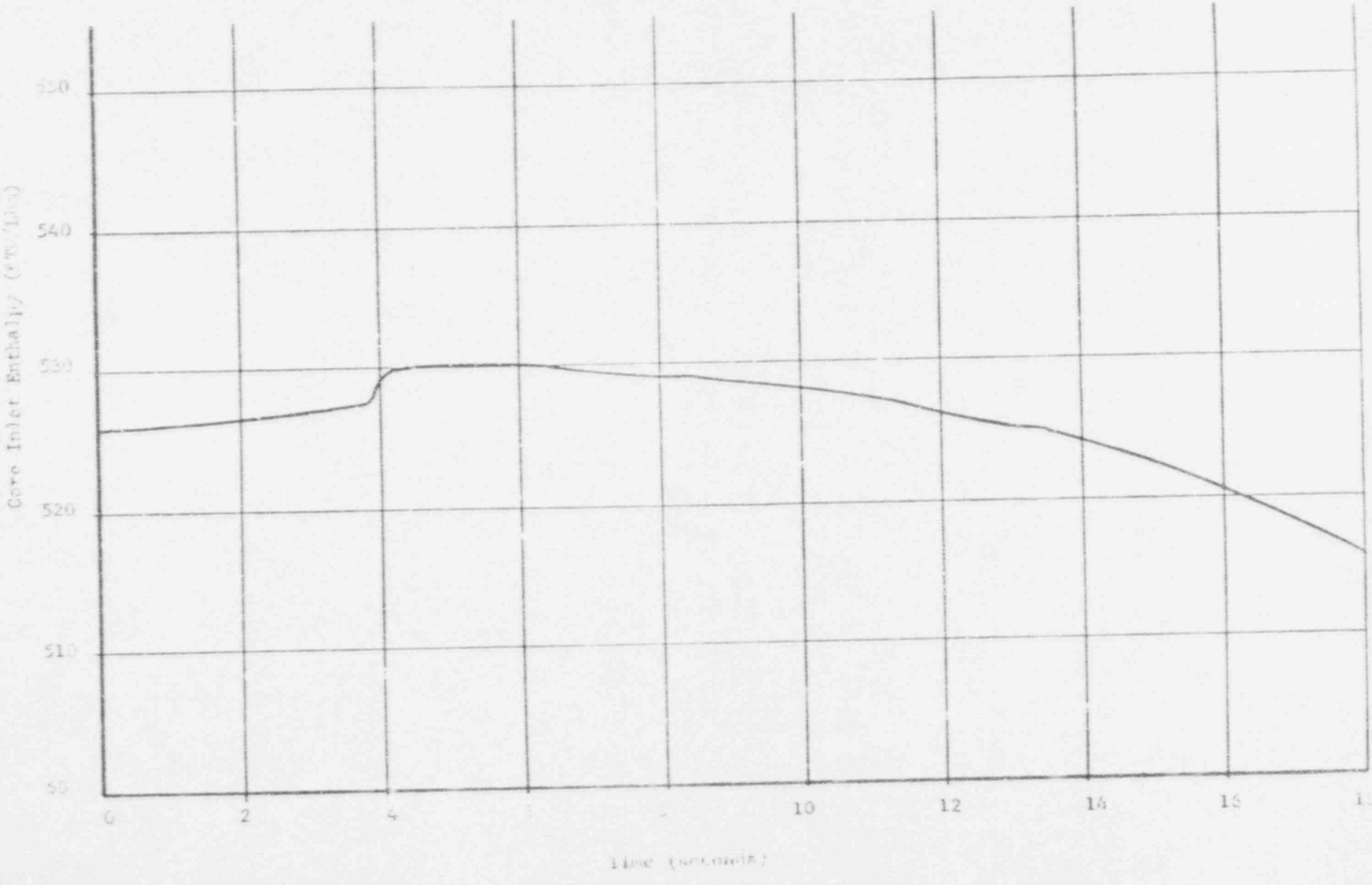


Figure 4-3

Core Average Pressure
Following a Design Basis Accident

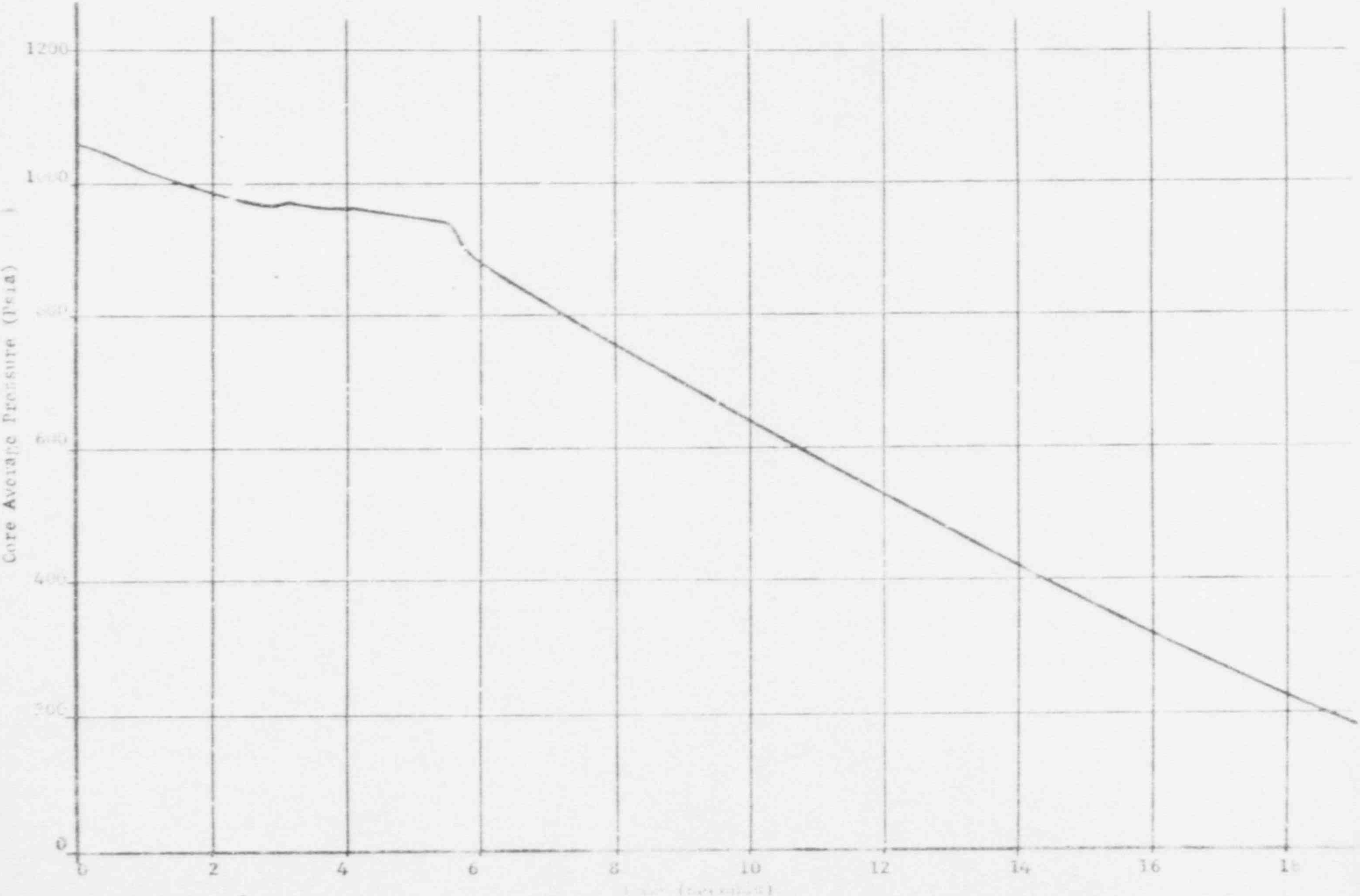


Figure _-1 (7x7 Fuel)

Minimum Critical Power Ratio
Following a Design Basis Accident

Note 1: CPR = 1.0 at Spacer 2 with
APLHGR = 12 kw/ft which is
[illegible]% of MAPLHGR

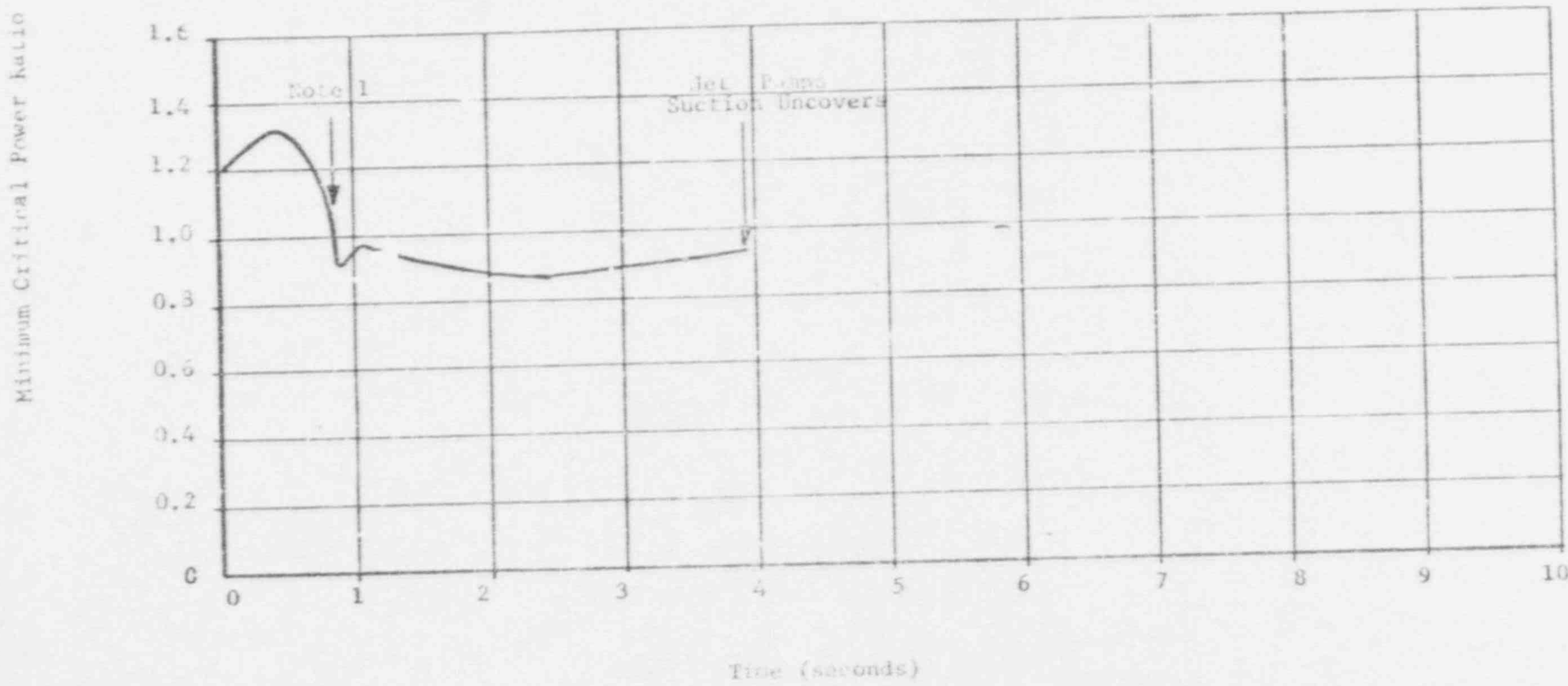


Figure 2-1 (8x8 Fuel)

Minimum Critical Power Ratio
Following a Design Basis Accident

Note 1: CPR = 1.0 at Spacer 2 with APLHGR = 9.34 kw/ft which is 85% of MAPLHGR

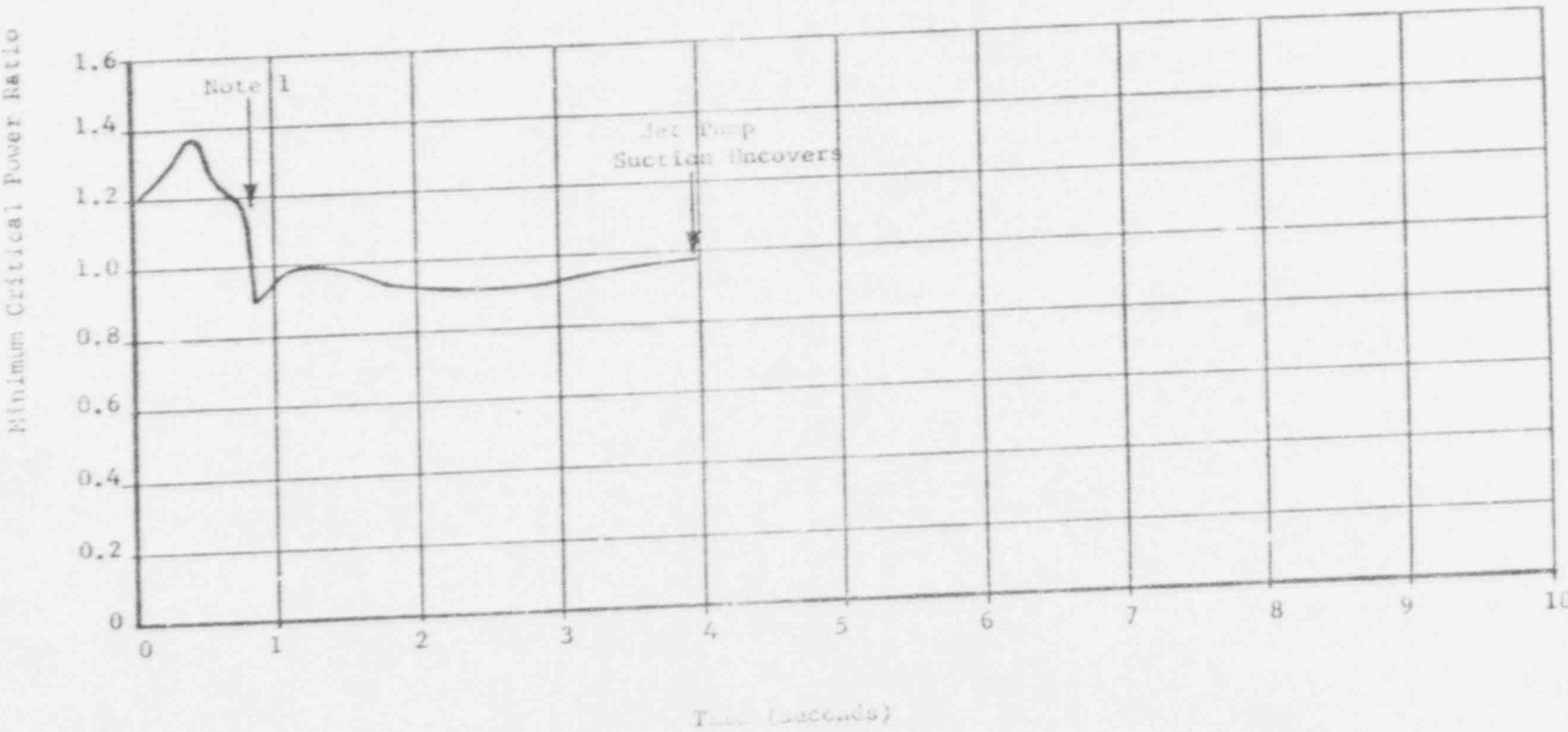


Figure C-2

Fuel Rod Convective Heat Transfer Coefficient
Following a Design Basis Accident-High Power Axial Node

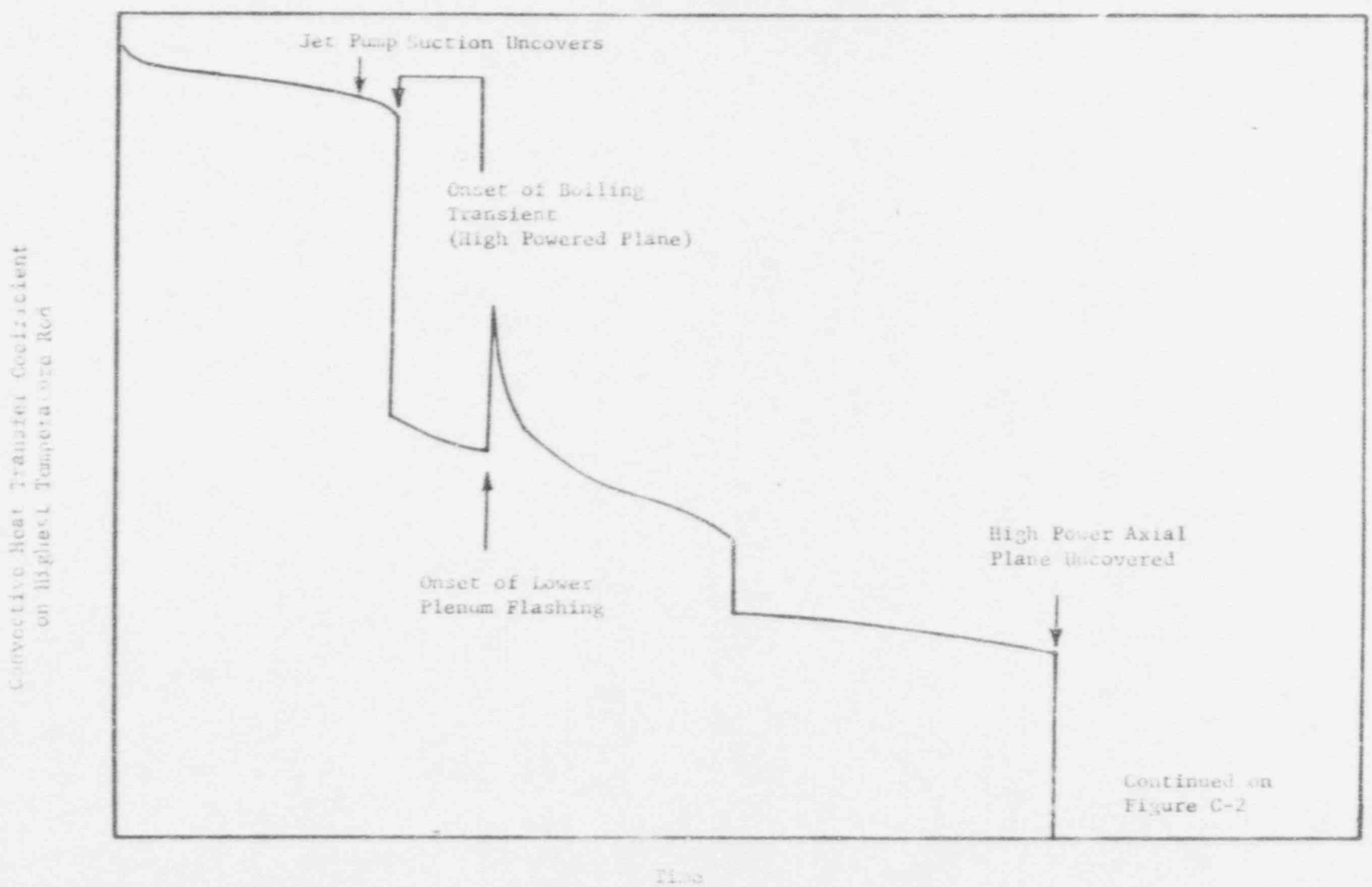


Figure C-1

Water Level Inside the Shroud and Reactor Vessel Pressure
Following a Design Basis Accident

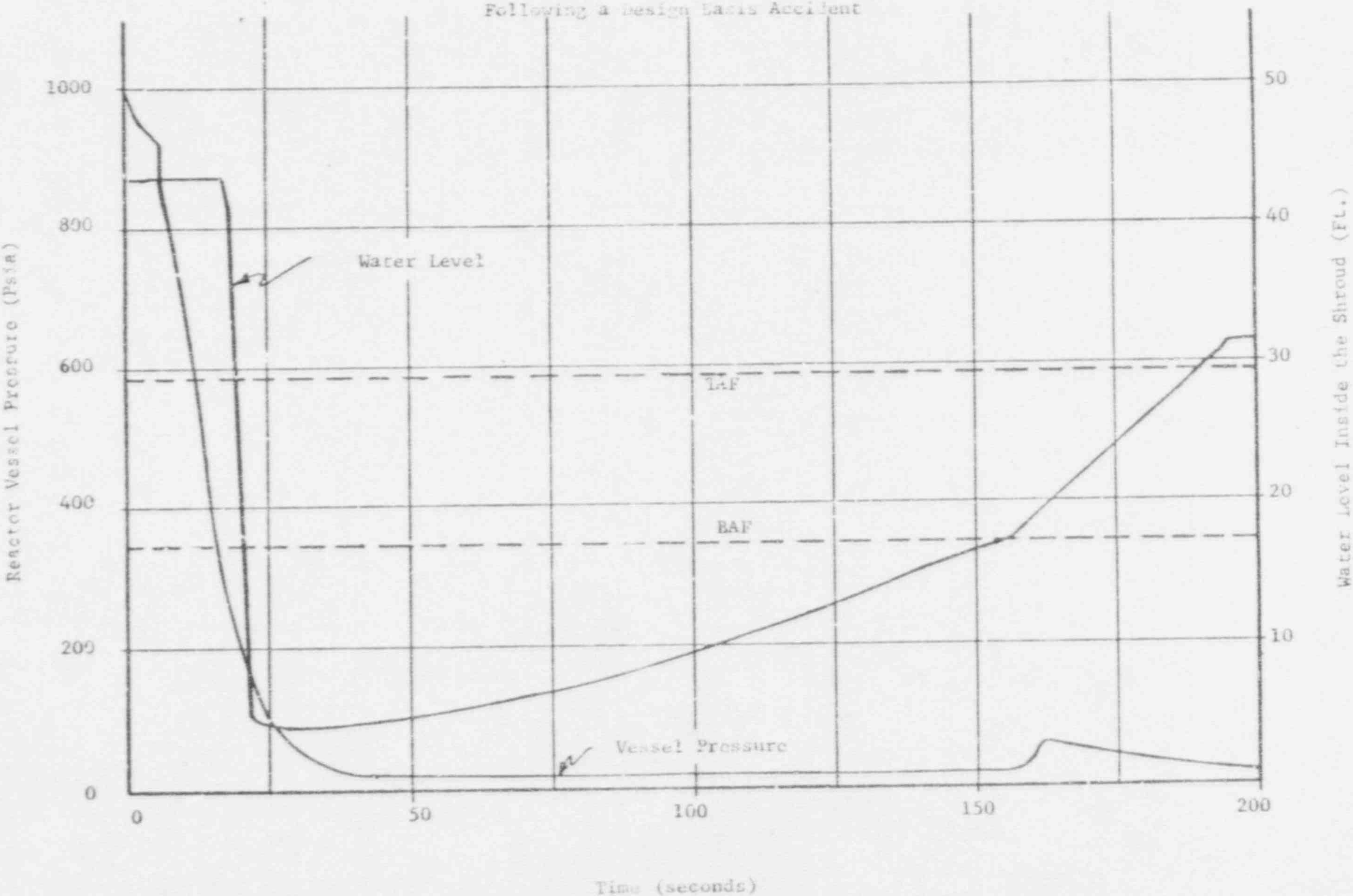


Figure C-2

Fuel Rod Convective Heat Transfer Coefficient
Following a Design Basis Accident-High Power Axial Node

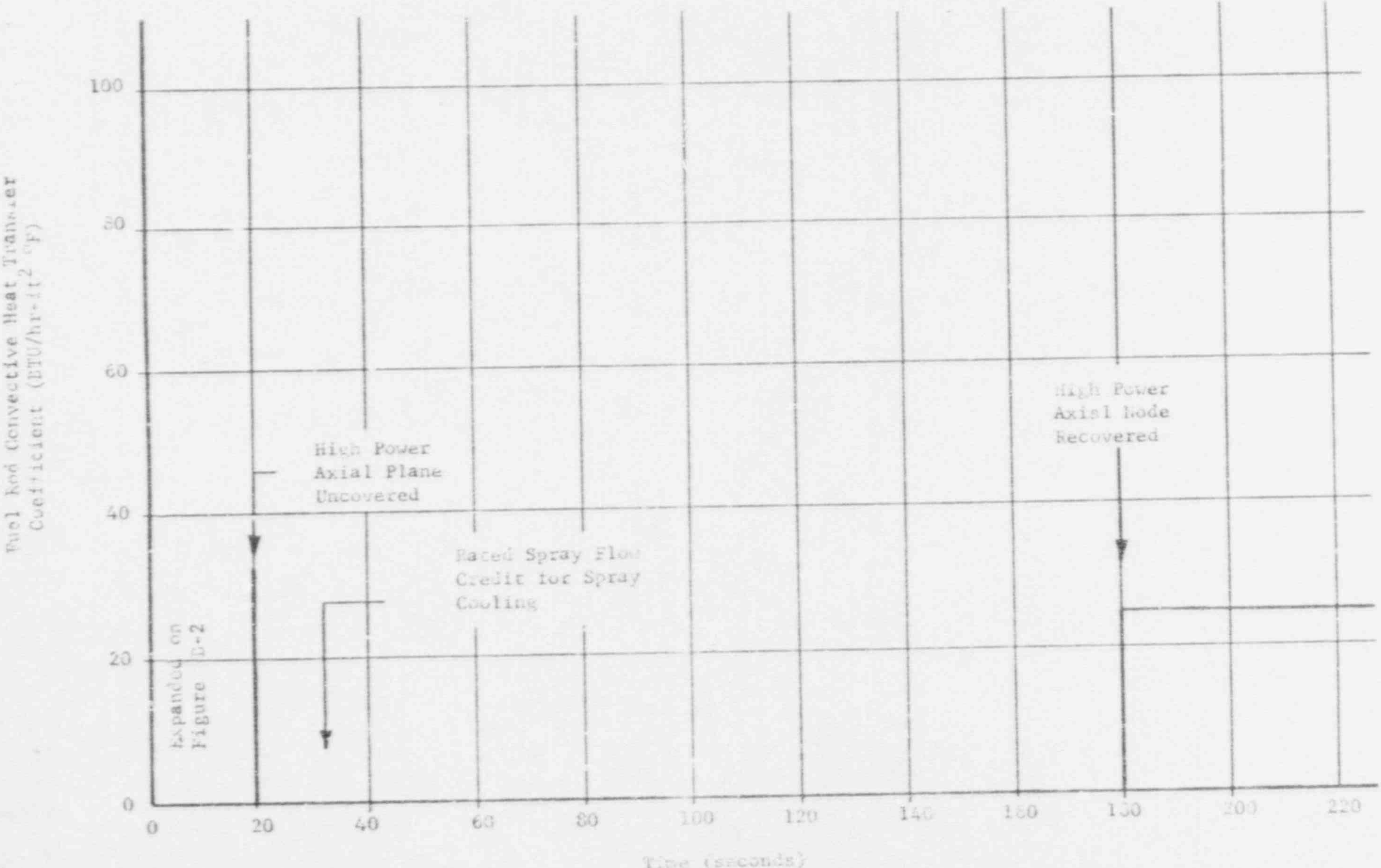


Figure C-3

Water Level Inside the Shroud and Reactor Vessel Pressure
Following a Small Break of the Recirculation Line
(HPCI System Failure; 0.1 ft.^{2} Break Area; 4 LPCI + 2 CS + ADS Available)

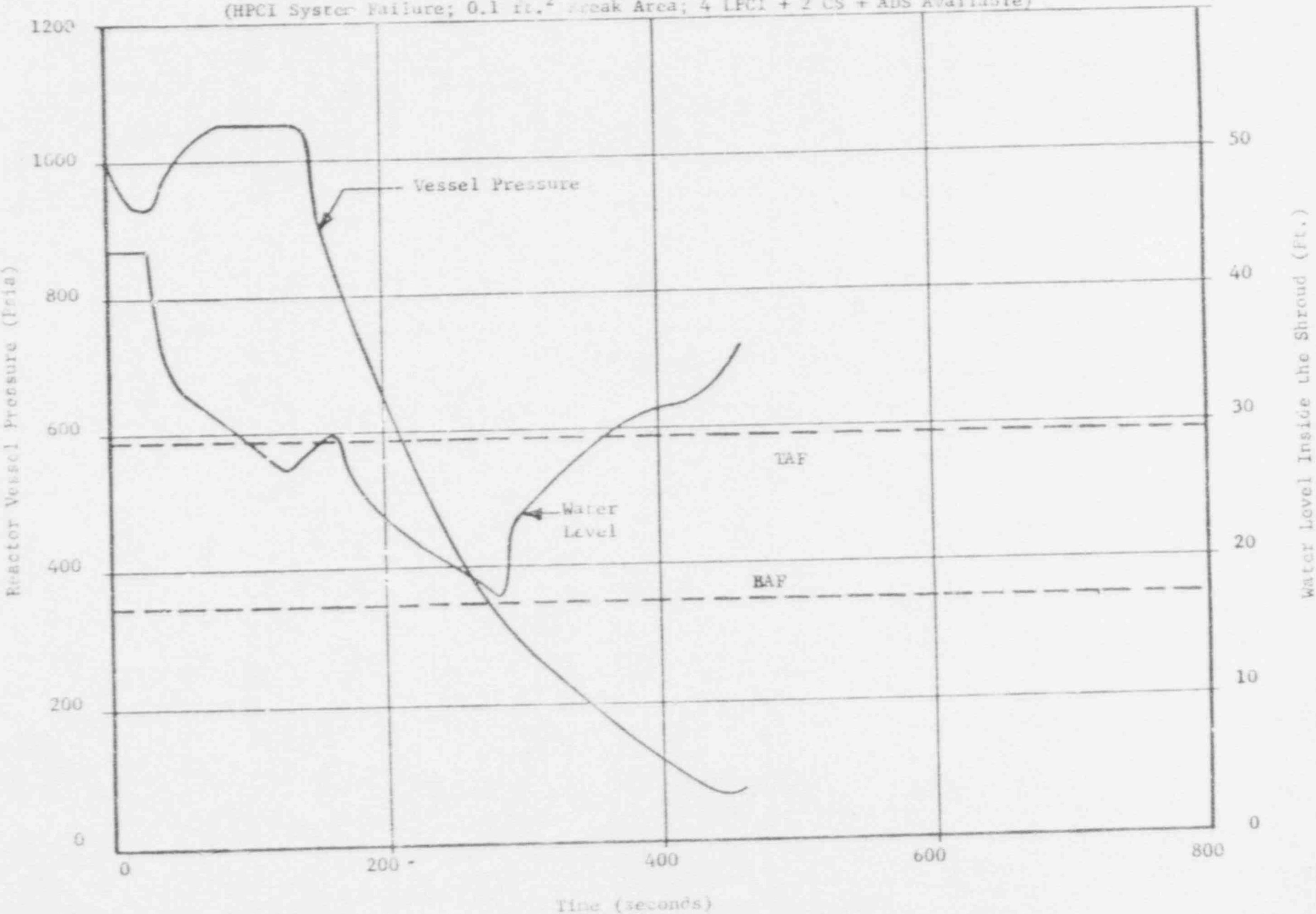


Figure D-1

Peak Cladding Temperature Following a Design Basis Accident

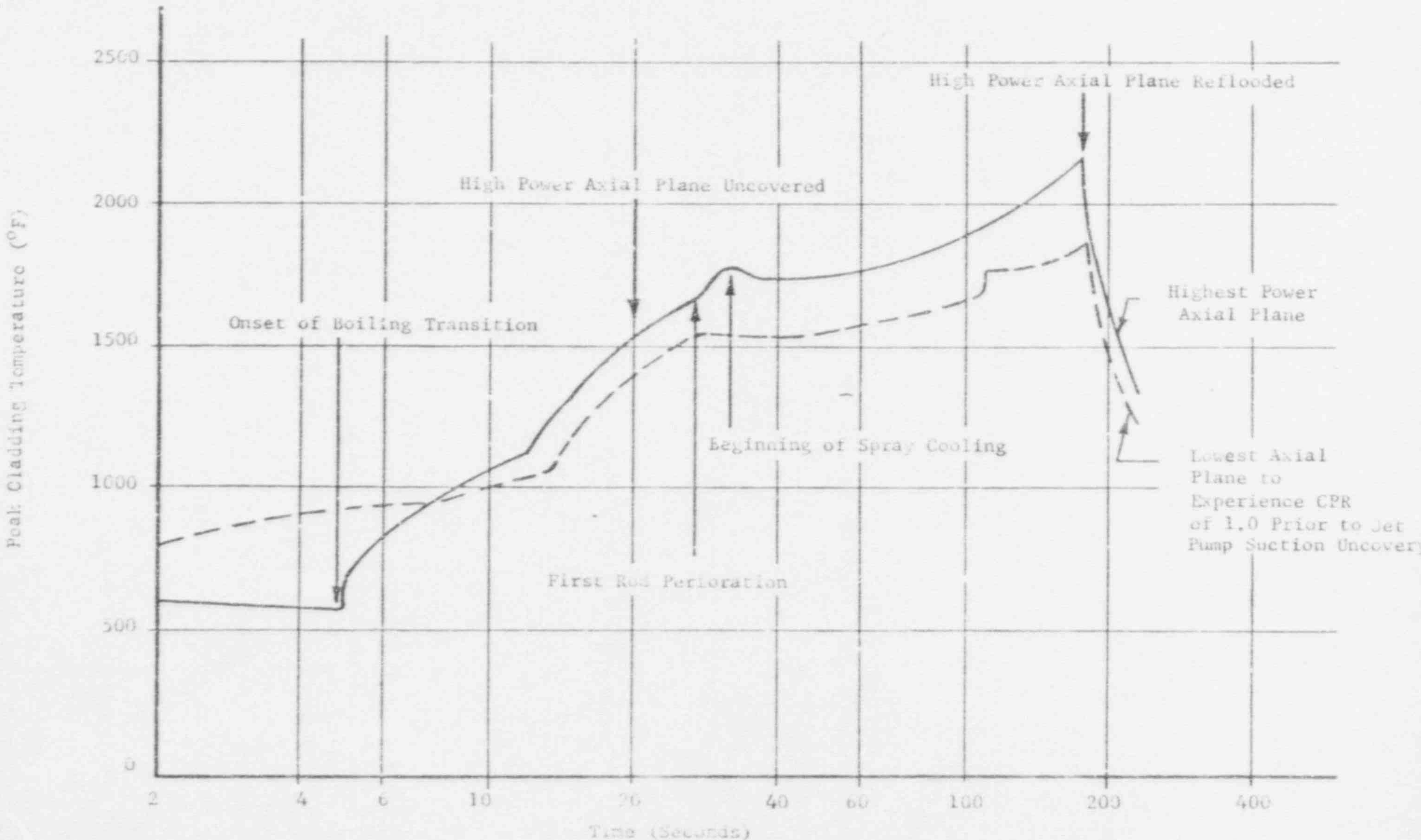


Figure D-2

Peak Cladding Temperature Following a Small
Break of the Recirculation Line-Highest Temperature Axial Node
(HPCI System Failure; 0.1 ft.^{2} Break Area; 4 LPCI + 2 CS + ADS Available)

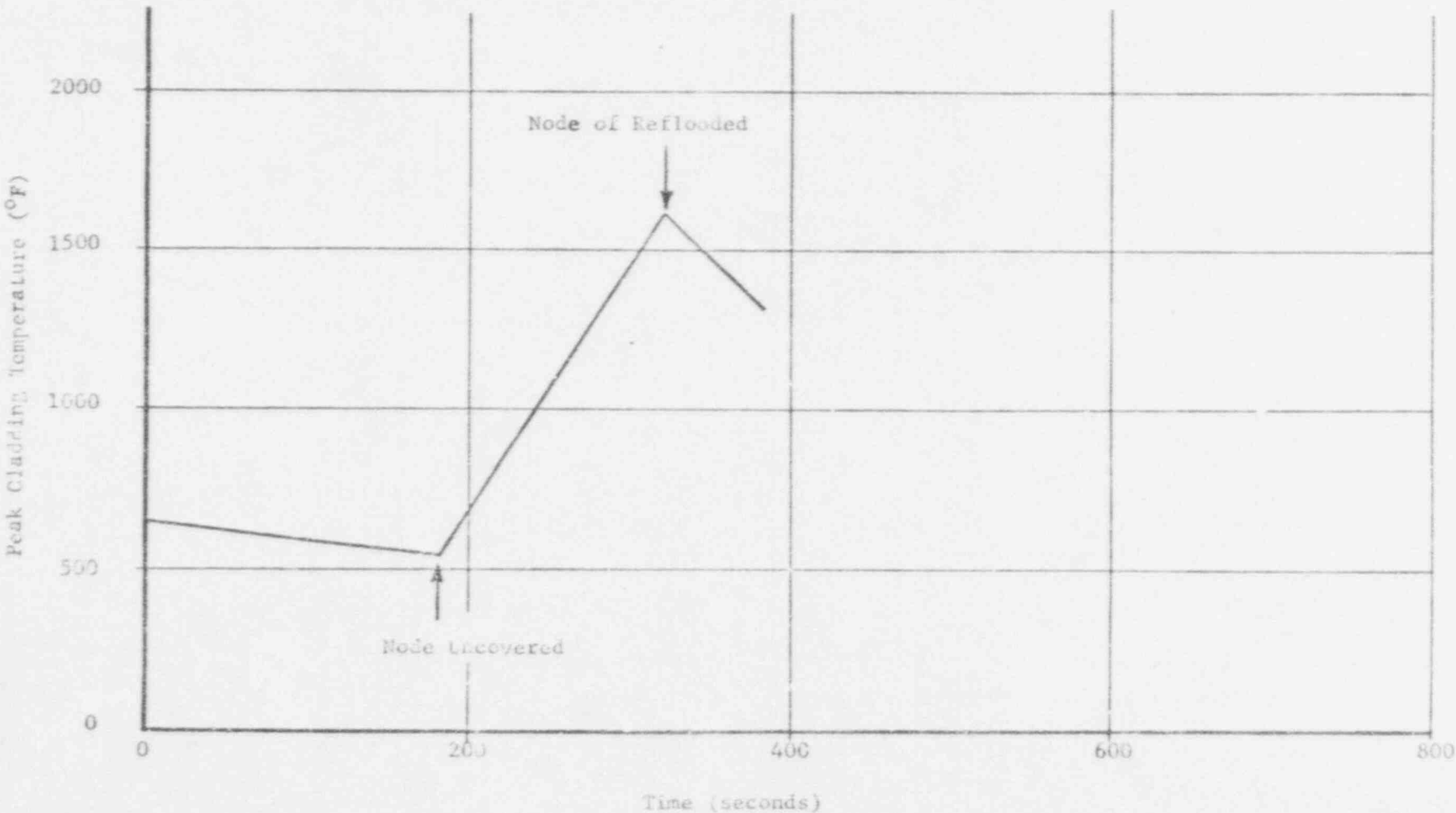


Figure D-3

Peak Cladding Temperature Versus Break Area

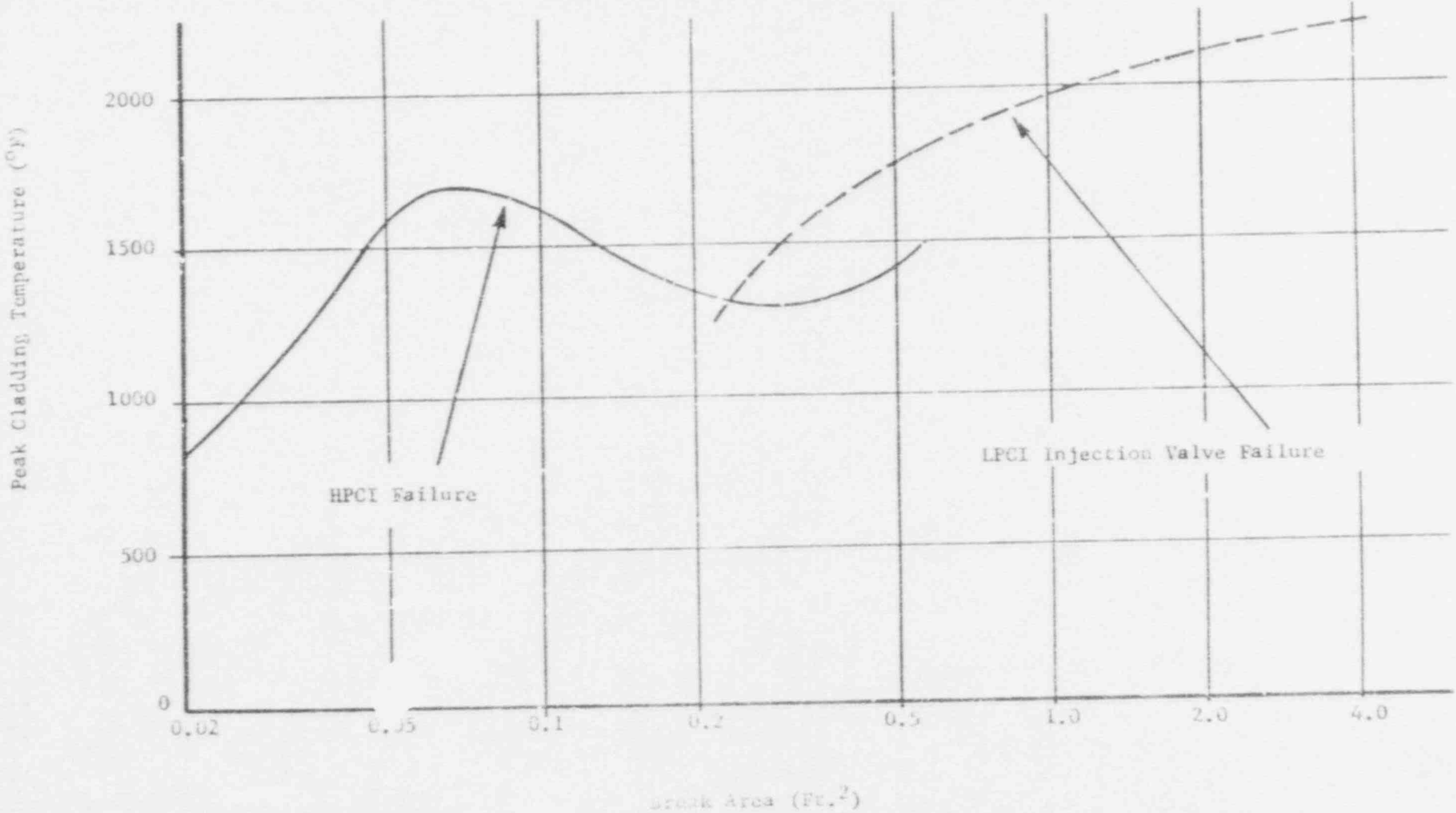


Figure D-4A

Peak Cladding Temperature Versus Planar Average Exposure - Initial Core 7x7 Fuel

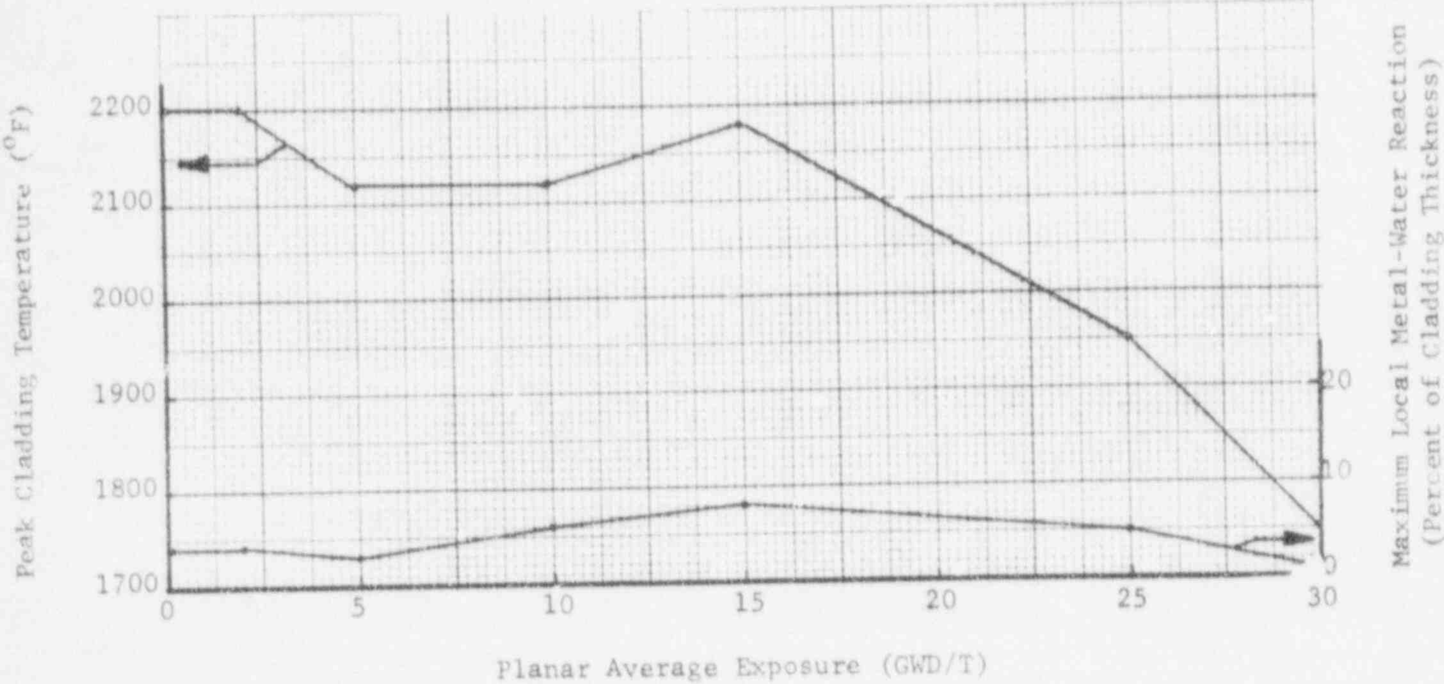


Figure D-5A

MAPLHGR Versus Planar Average Exposure

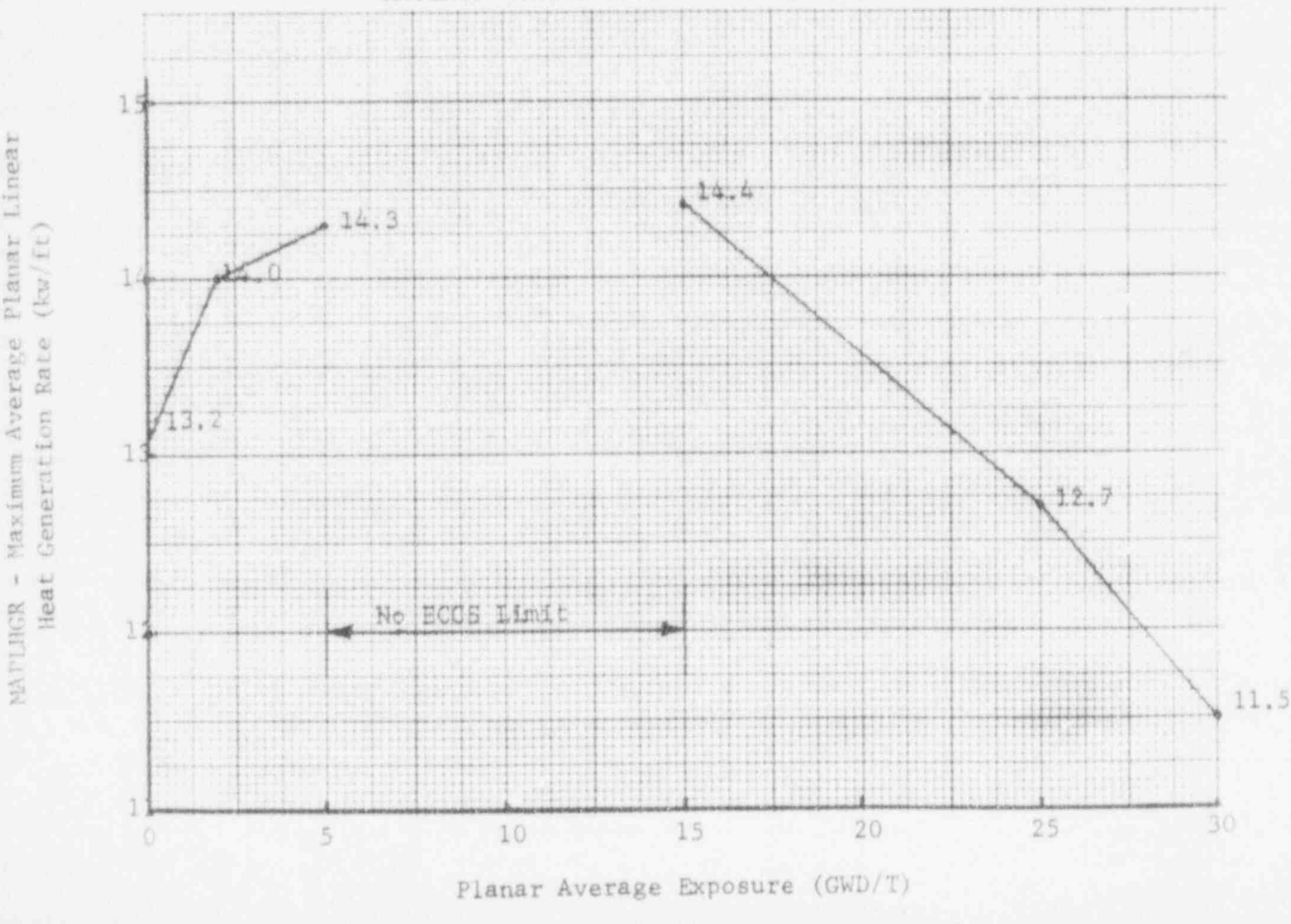


Figure D-4B

Peak Cladding Temperature Versus Planar Average Exposure - Reload 1 (Generic B) 7x7 Fuel

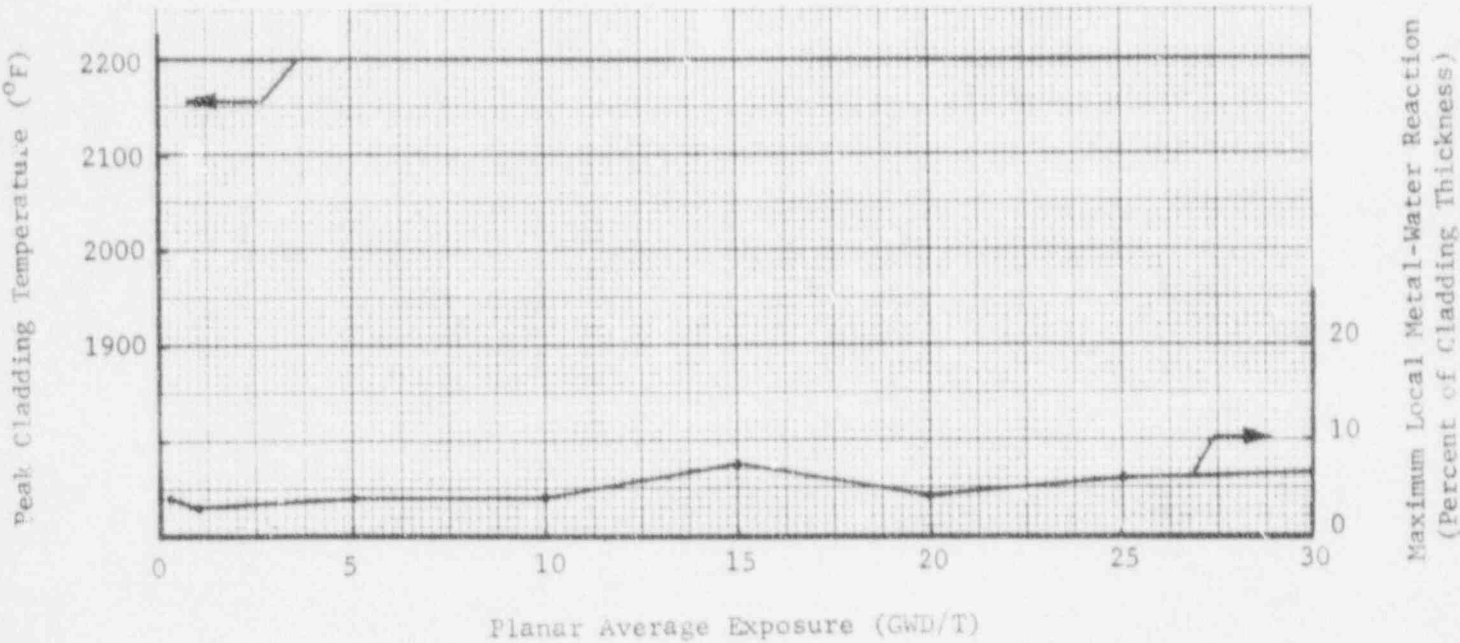


Figure D-5B

MAPLHGR Versus Planar Average Exposure

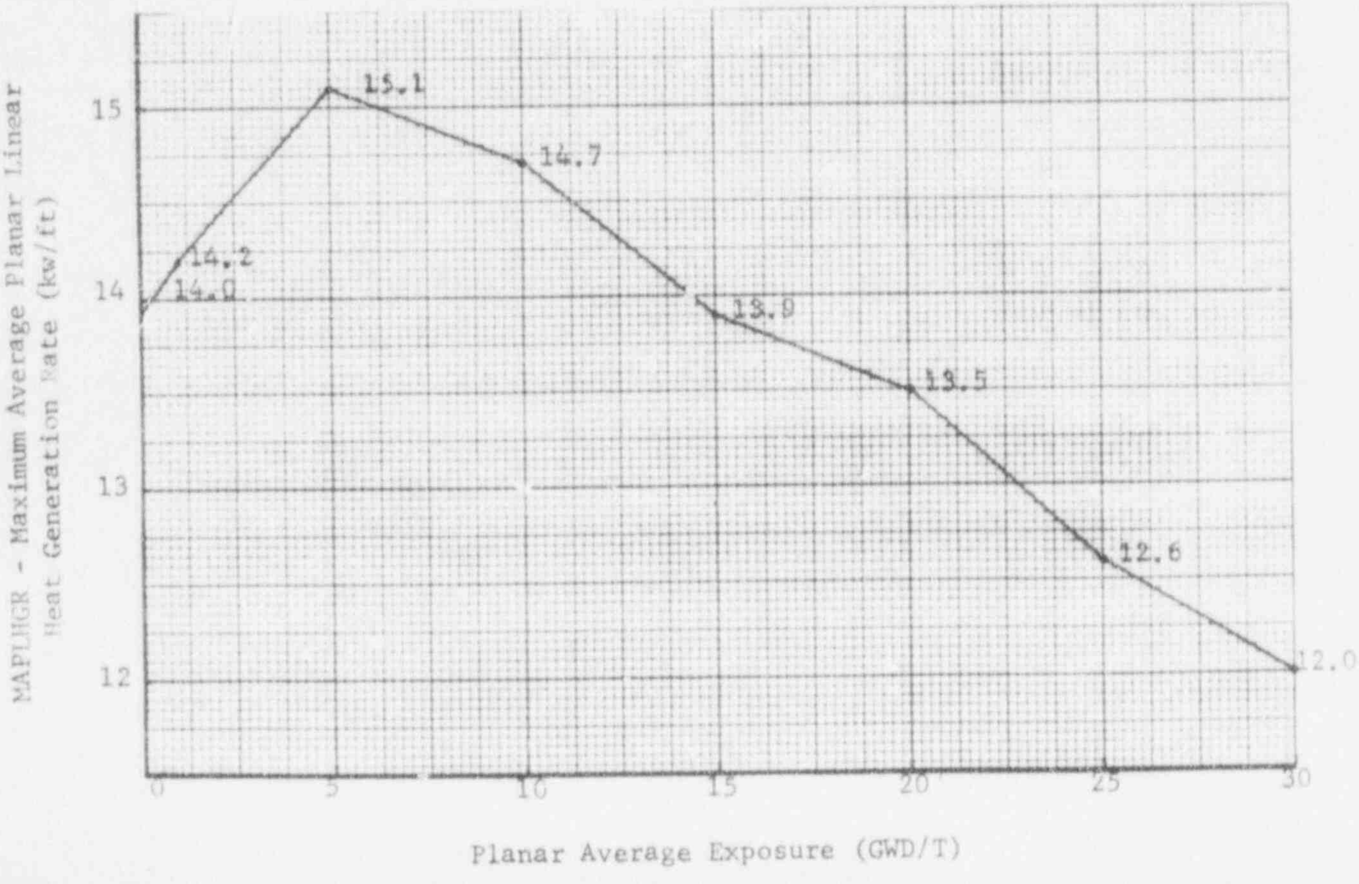


Figure D-4C

Peak Cladding Temperature Versus Planar Average Exposure - Reload 2 (8D262) Fuel

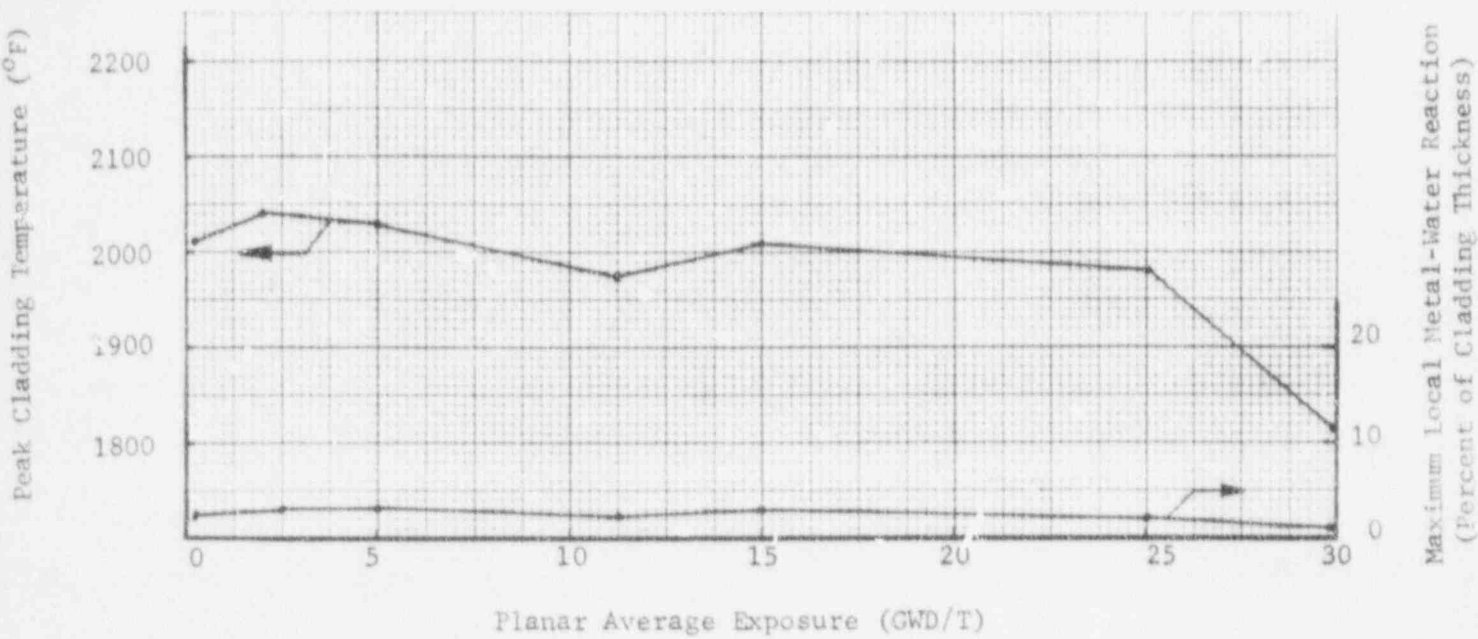


Figure D-5C

MAPLHGR Versus Planar Average Exposure