EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST DATED AUGUST 4, 1975

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

Pursuant to 10CFR50.59 the holders of Provisional Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications.

This exhibit makes reference to the following documents:

i. Monticello Nuclear Generating Plant
Technical Specifications, Appendix A

ii. Monticello Nuclear Generating Plant
License Amendment Request dated August 20, 1974

iii. December 27, 1974, Order for Modification of License for the Monticello Nuclear Generating Plant

iv. Monticello Nuclear Generating Plant
License Amendment Request Dated March 12, 1975

1. AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

PROPOSED CHANGE

This specification currently exists as APLHGR limit figures in Reference i (page 108C), in the proposed changes in Reference ii (pages 189H, 189I, and 189J of Exhibit B) and in Reference iii (Figures A-1 through A-4). The attached APLHGR limit figures (pages 189H through 189L of attached Exhibit B) are proposed to replace the limits stated in each of the three above referen...s. The remaining portions of Reference ii are not affected.

NOTE: Three fuel types are considered in References i and ii. The fourth fuel type currently in use at Monticello was not limited by ECCS considerations using the model described in Reference ii. Because of changes in the ECCS model required by the AEC staff, Reference iii imposed a restriction on this fourth fuel type also. The proposed changes shown in the attached Exhibit B include APLHGR limits for the four fuel types currently in use at Monticello as well as a fifth type soon to be loaded in the Monticello reactor.

REASON FOR CHANGE

These proposed changes are provided in accordance with the requirements of Reference iii. They are the result of calculations using an ECCS model modified to incorporate the changes required by that reference. The attached Exhibit C discusses the revised ECCS model and presents the results.

2. BASES 3.11 (page 189F of Reference iv, Exhibit B)

PROPOSED CHANGES

This page supersedes page 189F of Reference iv; it includes two minor changes. The second line of this page states that an initial operating MCPR using the ECCS calculations was 1.19. That value should be changed to 1.18. Reference 5 of page 189F should be changed as indicated in the attached Exhibit B.

REASON FOR CHANGE

Exhibit C uses an initial operating MCPR in the calculation which is slightly different than that used in Reference ii and reported in Reference iv. The assumed initial condition for the ECCS calculation is well below the operating limit discussed in the paragraph under change.

The change to Reference 5 on page 189F corrects an error in Reference iv.

3. SPECIFICATION 3.11.C (page 189K of Reference iv, Exhibit B)

PROPOSED CHANGE

Change the page number of Figure 3.11.2 from 189K (as presented in Reference iv Exhibit B) to 189M as shown in the attached Exhibit B.

REASON FOR CHANGE

The insertion of additional pages in item 1 above requires this change to arrange figures sequentially.

EXHIBIT B

LICENSE AMENDMENT REQUEST DATED AUGUST 4, 1975

Exhibit B, attached, consists of newly prepared pages for the Appendix A Technical Specifications as listed below. These pages incorporate the proposed changes.

<u>PAGES</u>

189 F
189 H
189 I
189 J
189 K
189 L
189 M

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss of coolant accident to be 1.18 for all fuel types. The Operating MCPR Limit of 1.41 for 8x8 fuel and 1.33 for 7x7 fuel is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit of 1.06 (T.S.2.1.A) applicable to all fuel types is maintained in the event of the most limiting abnormal operational transient.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

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. Whenever the limit is exceeded the monitored value will be documented and available for review, audit and inspection of plant operations. The only way to violate the Limiting Condition for Operation 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.

5. "General Electric BWR Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1, November, 1974.

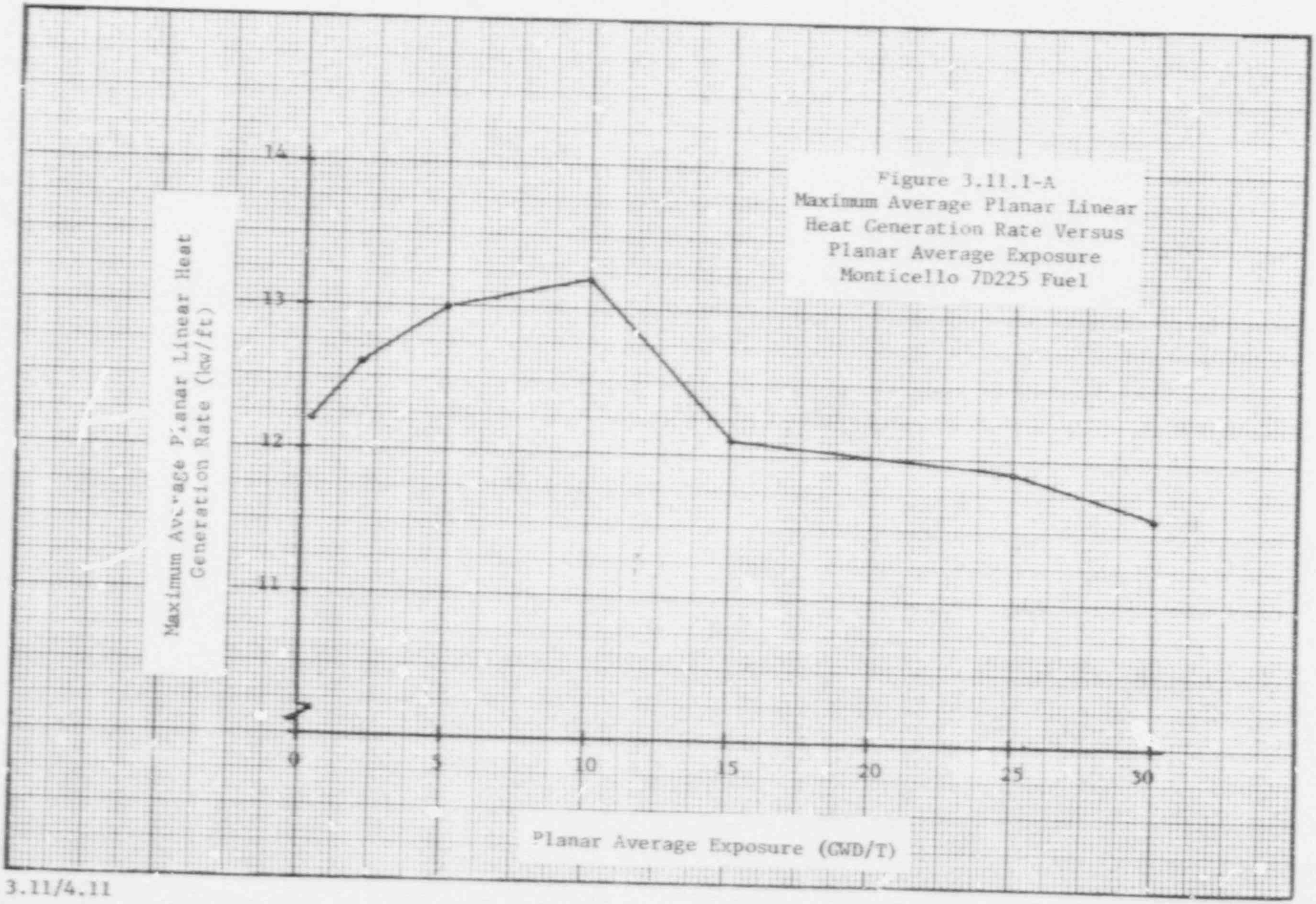

Figure 3.11.1-A
Maximum Average Planar Linear
Heat Generation Rate Versus
Planar Average Exposure
Monticello 7D225 Fuel

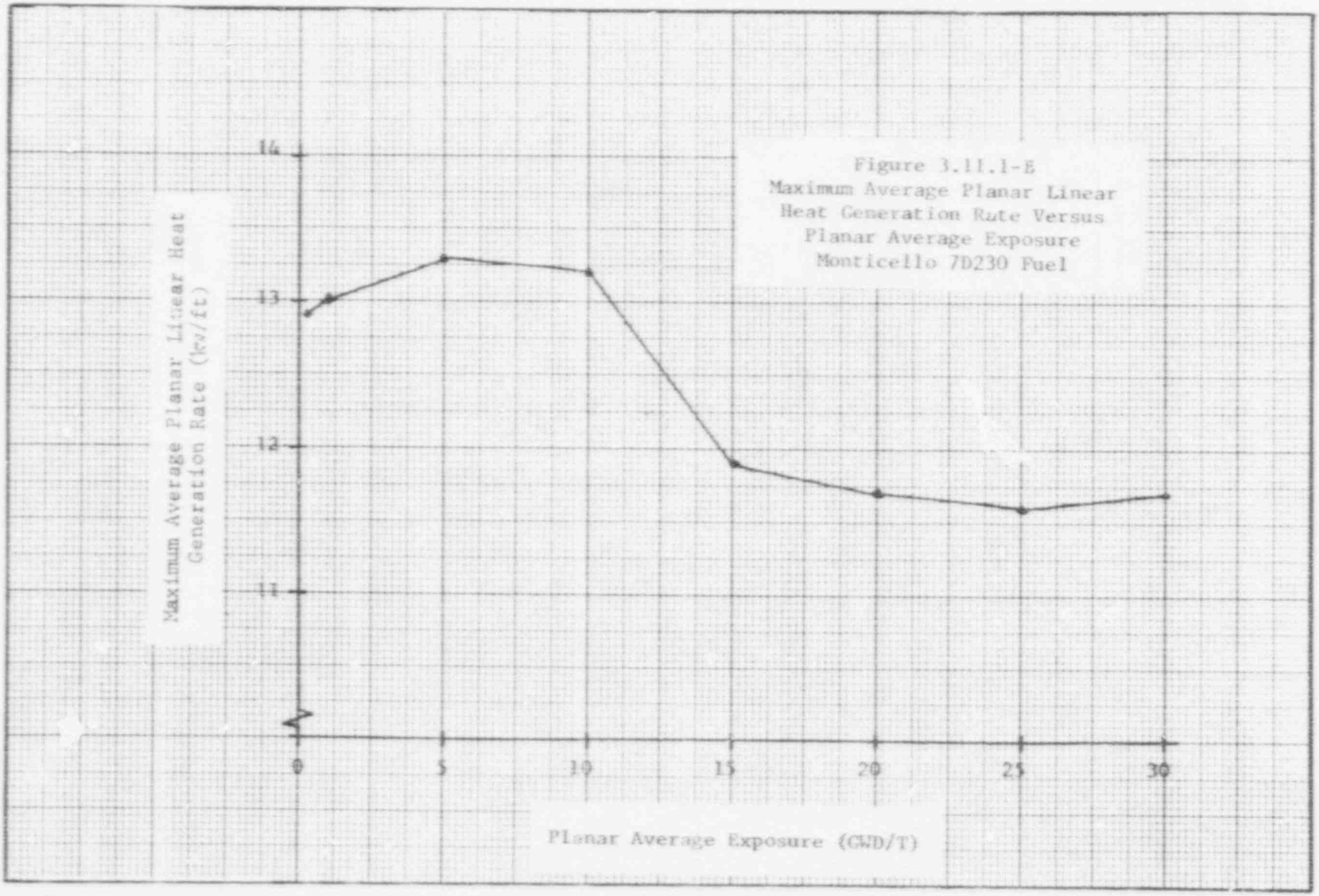

Figure 3.11.1-E
Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure Monticello 7D230 Fuel

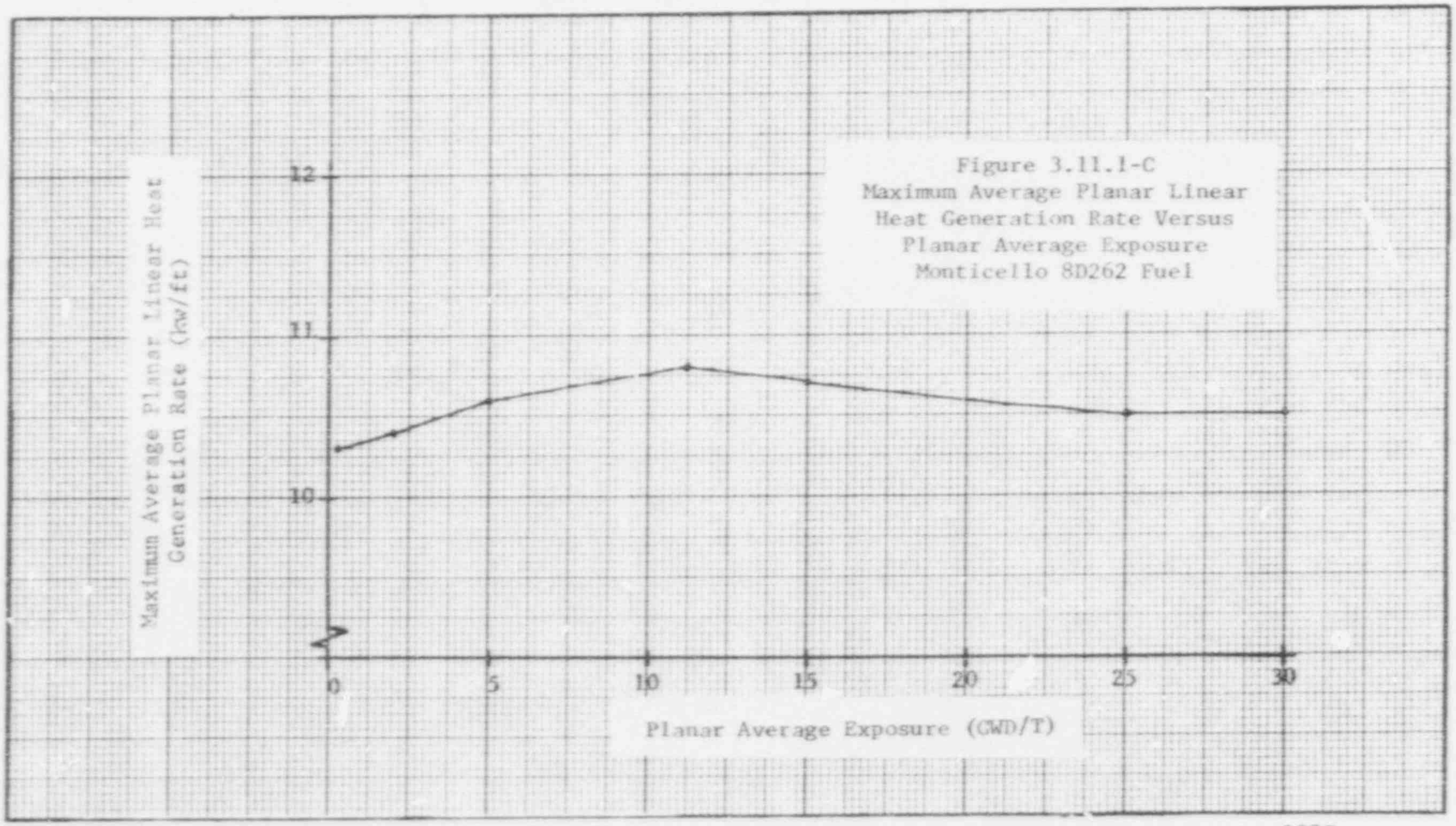

Figure 3.11.1-C
Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure
Monticello 8D262 Fuel

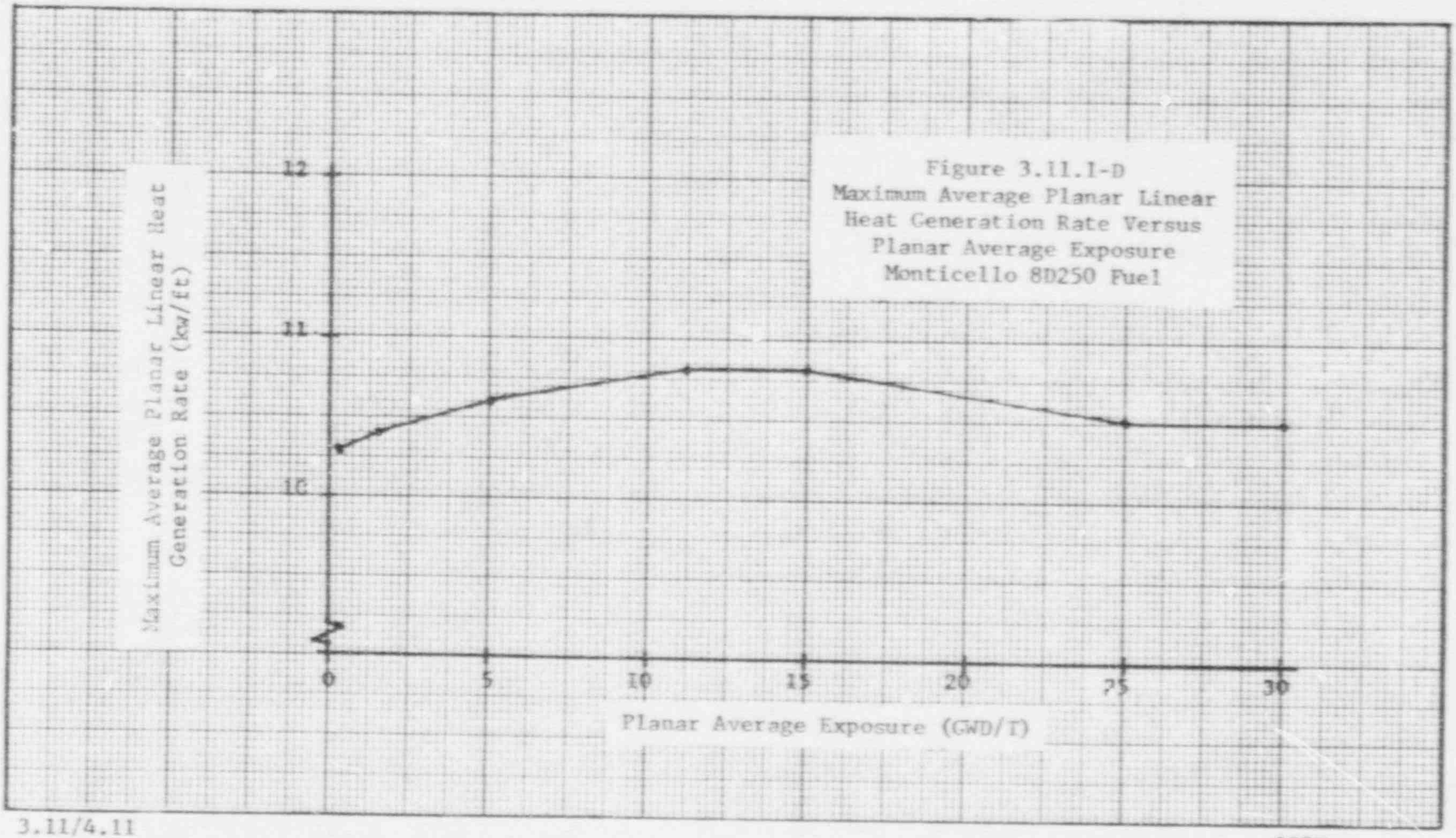

Figure 3.11.1-D
Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure Monticello 8D250 Fuel

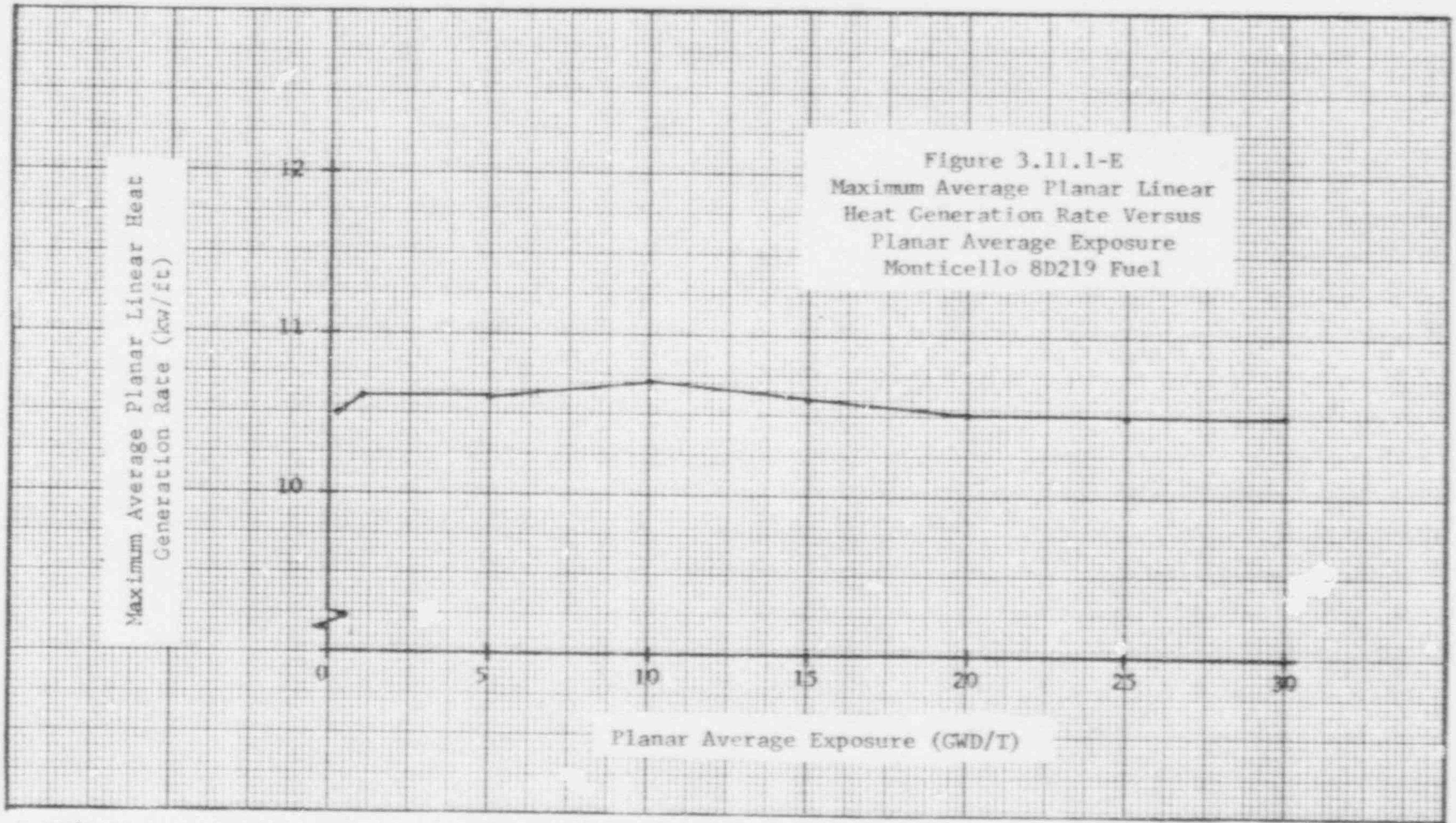

Figure 3.11.1-E
Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure
Monticello 8D219 Fuel

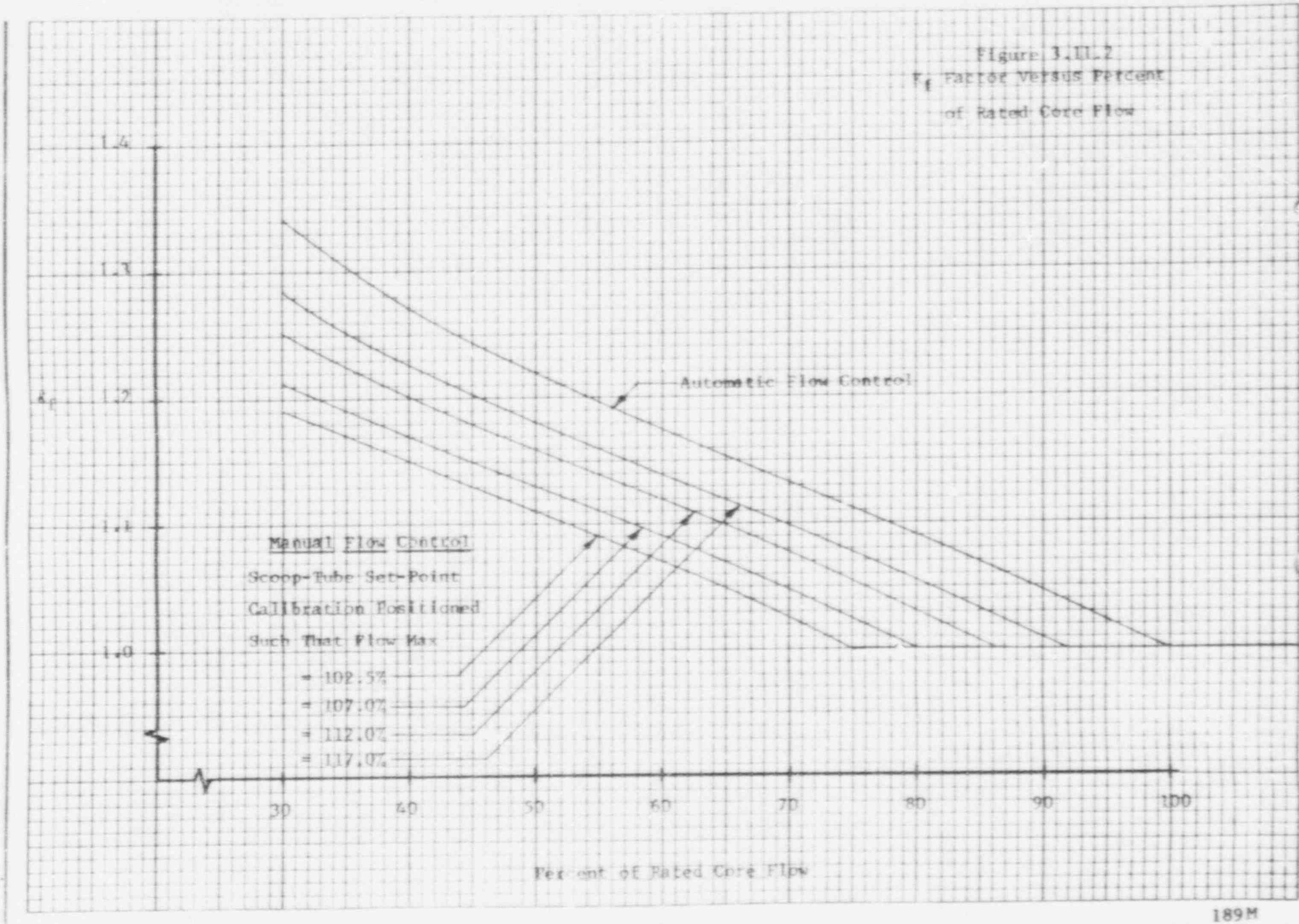

Figure 3.11.2
K_f Factor Versus Percent
of Rated Core Flow

EXHIBIT C

MONTICELLO

NUCLEAR GENERATING PLANT

CONFORMANCE WITH 10 CFR 50

APPENDIX K

(JET PUMP PLANT)

AUGUST

1975

DISCUSSION

Presented in the following document are the results of the loss-of-coolant accident analysis of the Monticello Nuclear Generating Plant. 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 Reference 1.

Between August and December, 1974, General Electric and the USAEC worked together to resolve differences in interpretation of Appendix K and to consider additional phenomena in the evaluation models. As a result, the models used in the present analysis differ from those used in previous submittals in the following respects:

(1) The new analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in the Figures;

(2) Fission product decay is computed assuming an energy release rate of 200 MeV/Fission;

(3) Pool film boiling is assumed after nucleate boiling is lost during the flow stagnation period;

(4) The effects of core spray entrainment and counter-current flow limiting are included in the reflooding calculation.

In addition, there have been a few other minor improvements to the computer codes which individually and jointly have a small effect on the calculated results. The figures in this submittal reflect these changes, as well as the four major changes enumerated above.

In the analysis of the break spectrum of this reactor, a range of break sizes was studied, with a range of single failures being considered for each break size. A list of the single failures considered for each break size is shown in the lead plant submittal referenced herein. That list is applicable to the analysis of this reactor.

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 INPUTS PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR MONTICELLO

PLANT PARAMETERS:

Core Thermal Power.................... 1703 MWt which corresponds to 102% of licensed core power*

Vessel Steam Output............. 6.913×10^6 Lbm/h which corresponds to 102 % of rated steam flow

Vessel Steam Dome Pressure............ 1040 psia

Design Basis Recirculation Line Break Area for Large Breaks 3.9 ft^2 1.0 ft^2

Recirculation Line Break Area for Small Breaks 1.0 ft^2 0.07 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 7D225	7 x 7	17.5	1.57	1.18
Reload 1 7D230	7 x 7	17.5	1.57	1.18
Reload 2 8D262	8 x 8	13.4	1.57	1.18
Reload 3 8D250	8 x 8	13.4	1.57	1.18
Reload 4 8D219	8 x 8	13.4	1.57	1.18

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

*This power level equals the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its maximum (technical specification) linear heat generation rate.

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 Hydraulics Model (LAMB)

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

C. Calculated by the Long-Term Thermal Hydraulics 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 operation at the MAPLHGR used in the analysis. These values, if more limiting than other design parameters, represent limits for operation to ensure conformance with 10CFR50.46 and Appendix K.

The peak cladding temperatures as a function of time are shown in Figure D-1. Other parameters relevant to the analysis are shown in the attached figures and are described in subsequent paragraphs.

Results for guillotine severances of a main steam line, a feedwater line, and a core spray line are presented in Reference 2.

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*Maximum (Bundle) Average Planar Linear Heat Generation Rate

TABLE 2

APPENDIX K RESULTS FOR MONTICELLO

Break Size Location Single Failure		PCT(°F)	Peak Local Oxidation %	Core-Wide Metal-Water Reaction %
DBA ANALYSIS(1)				
3.9 ft^2 (DBA) Recirc Suction LPCI Injection Valve		2200(1)	8.7%	0.5
BREAK SPECTRUM ANALYSIS(3)				
3.9 ft^2 (DBA) Recirc Suction LPCI Injection Valve		2200(1)	8.7%	0.5
1.0 ft^2 Recirc Suction LPCI Injection Valve	Large Break Methods	1670(1)	< 1	-
	Small Break Methods	1690(2)	< 1	-
0.07 ft^2 Recirc Suction HPCI		1430(2)	< 1	-

Notes:

(1) CHASTE - large break methods
(2) Non-DBA reflood
(3) For other breaks in spectrum see lead plant analysis, Reference 2. For justification of selection of lead plant, see Reference 3.

A. Appendix K Short-Term Thermal Hydraulics Analysis

General Description of the LAMB Code

The LAMB code is a model which is used to analyze the short-term thermodynamics and thermal hydraulics behavior of the coolant in the vessel during a postulated 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 Reference 1.

Results of the LAMB Analysis

Presented in the section are results of the loss-of-coolant accident analysis which are calculated by LAMB. Table 3 lists the figures provided for all the analyses. These results include the following:

Parameter	Figure
Core Average Inlet Flow Rate (Normalized to unity at the beginning of the accident)	
-- Following a Design Basis Accident	A-1a
-- Following a 1.0 Sq. Ft. Break	A-1d
Core Inlet Enthalpy	
-- Following a Design Basis Accident	A-2a
-- Following a 1.0 Sq. Ft. Break	A-2d
Core Average Pressure	
-- Following a Design Basis Accident	A-3a
-- Following a 1.0 Sq. Ft. Break	A-3d

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 hydraulics 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 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. Table 3 lists the figures provided for all the analyses. These results include the following:

Parameter	Figure
Minimum Critical Power Ratio	
-- Following a Design Basis Accident, 8x8	B-1a-1
-- Following a Design Basis Accident, 7x7	B-1a-2
-- Following a 1.0 Sq. Ft. Break, 8x8	B-1d
Convective Heat Transfer Coefficient	
-- Following a Design Basis Accident	B-2
-- Following a 1.0 Sq. Ft. Break	B-2

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

C. Appendix K Long-Term Thermal Hydraulics 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 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 code. For the "large" break analysis, the LAMB and SAFE codes are employed to determine the time of boiling transition and the post-boiling transition convective heat transfer coefficient during the blowdown. The SAFE code calculates the uncovery and re-flooding of the fuel and the duration of spray cooling.

The SAFE analytical model has been expanded and refined to consider explicitly the following phenomena:

(1) Counter-current flow limiting (CCFL) in the fuel bundles and in the core bypass region, of ECCS water injected over the core;

(2) Entrainment and loss of ECCS water injected over the core; and

(3) Filling of discrete volumes (control rod guide tubes, core bypass and lower plenum) which were previously taken together.

Calculation of these effects is presently external to the SAFE code: the calculational logic will eventually be incorporated in the SAFE code.

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. Table 3 lists the figures provided for all the analyses. These results include the following:

Parameter	Figure
Water Level Inside Shroud	
-- Following a Design Basis Accident (LPCI Inj. Valve Failure)	C-1
-- Following a 1.0 Sq. Ft. Large Break (LPCI Inj. Valve Failure)	C-1
-- Following a 1.0 Sq. Ft. Small Break (LPCI Inj. Valve Failure)	C-2
-- Following a 0.07 Sq. Ft. Small Break (HPCI Failure)	C-2
Reactor Vessel Pressure	
-- Following a Design Basis Accident (LPCI Inj. Valve Failure)	C-1
-- Following a 1.0 Sq. Ft. Large Break (LPCI Inj. Valve Failure)	C-1
-- Following a 1.0 Sq. Ft. Small Break (LPCI Inj. Valve Failure)	C-2
-- Following a 0.07 Sq. Ft. Small Break (HPCI Failure)	C-2

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 Zircaloy, which reacts with steam at high temperatures. The zircaloy steam chemical reaction rate is exothermic and stongly 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.

A break spectrum analysis has been performed using the CHASTE code showing that the most limiting (highest calculated) peak clad temperature is associated with the design basis accident. The conclusion of this analysis is applicable to this plant. The analysis has been documented in the Quad Cities Station Special Report 15, Supplement C (Docket No. 50-254).

For each submittal of a construction permit, operating license, or reload license, the DBA peak cladding temperature, peak local oxidation, and a MAPLHGR is determined for each fuel type of interest. For calculational convenience in some cases, the rod-to-rod power distribution is assumed to be flat and the least favorable exposure is assumed in determining gap conductance. Calculation of the results under these conditions conservatively represents the results at all exposures. The code application is described, briefly, as follows:

A. For jet-pump plants a LAMB calculation is performed. In mixed cores, full-core LAMB calculations are performed for 7x7 and 8x8 fuel and the more restrictive of the two is used in the SCAT input.

B. For jet-pump plants, SCAT calculations are performed for 7x7 fuel and 8x8 fuel, as appropriate.

C. A SAFE and a DBA-REFLOOD calculation is performed, assuming the fuel to be the most predominant type of bundle in the core (7x7 or 8x8).

D. CHASTE calculations are performed for each fuel type (which in a given reactor may include several 7x7 fuel types and several 8x8 fuel types) at several exposure points.

The MAPLHGR, peak cladding temperature and maximum local oxidation variations with exposure for each fuel type are the results of these calculations.

Results of the CHASTE Analysis

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

Parameter	Figure
Peak Cladding Temperature	
-- Following a Design Basis Accident	D-1
-- Following a 1.0 Sq. Ft. Large Break	D-1
-- Following a 1.0 Sq. Ft. Small Break	D-2
-- Following a 0.07 Sq. Ft. Small Break	D-2
Peak Cladding Temperature and Local Peak Oxidation versus Break Area	D-3
Peak Cladding Temperature and Local Peak Oxidation versus Planar Exposure	
-- Initial Core Fuel (7D225)	D-4a
-- Reload 1 Fuel (7D230)	D-4b
-- Reload 2 Fuel (8D262)	D-4c
-- Reload 3 Fuel (8D250)	D-4d
-- Reload 4 Fuel (8D219)	D-4e

Parameter		Figure
Maximum Average Planar Linear Heat Generation Rate versus Planar Exposure.		
----Initial Core Fuel	(7D225)	D-5a
----Reload 1 Fuel	(7D230)	D-5b
----Reload 2 Fuel	(8D262)	D-5c
----Reload 3 Fuel	(8D250)	D-5d
----Reload 4 Fuel	(8D219)	D-5e

Figures D-4 show the calculated peak cladding temperature as a function of exposure if the fuel bundles are operated at the average planar heat generation rate plotted in figure D-5. Figures D-5 show the average planar linear heat generation rate (APLHGR) as a function of exposure if the fuel bundles are limited to the most restrictive of:

1. The 10CFR50.46 limits of 2200°F PCT
2. The 10CFR50.46 limits of 17% Local Metal Water Reaction
3. The 10CFR50.46 limits of 1% Core-wide metal-water reaction or,
4. The maximum design linear heat generation rate (LHGR) for the fuel and the peaking factor limits.

The APLHGR values indicated in figures D-5 impose restrictions to the operation of the reactor in the exposure range where the peak clad temperature (PCT) is 2200°F or the maximum local metal-water reaction is 17% in figures D-4. Outside this exposure range the calculation was performed on the basis of maximum design basis LHGR for the fuel and these limits are more restrictive than those of Appendix K.

Monticello

TABLE 3

KEY TO FIGURES

	LARGE BREAK METHOD			INTERMEDIATE BREAK		SMALL BREAK				
						Worst Sm. Brk.	Add'l Sm. Brk.			
	DBA	.80 DBA	.60 DBA	1.0 ft^2 (1) Large Break Methods	1.0 ft^2 (1) Small Break Methods	0.07 ft^2 Suction HPCI Fail		Core Spray Line	Feedwater Line	Main Steam Line
Core Avg. Inlet Flow	A-1a	A-1b*	A-1c*	A-1d	--	--	--	--	--	--
Core Inlet Enthalpy	A-2a	A-2b*	A-2c*	A-2d	--	--	--	--	--	--
Core Avg. Pressure	A-3a	A-3b*	A-3c*	A-3d	--	--	--	--	--	--
Min. Critical Power Ratio	B-1a	B-1b*	B-1c*	B-1d	--	--	--	--	--	--
Convective Heat Trans. Coefficient	B-2	B-2b*	B-2c*	B-2	D-2	D-2	D-2c*	D-2d*	D-2e*	--
Water Level Inside Shroud & Reactor Vessel Pressure	C-1	C-1b*	C-1c*	C-1	C-2	C-2	C-2c*	C-2d*	C-2e*	C-2f*
Peak Cladding Temperature	D-1	D-1b*	D-1c*	D-1	D-2a	D-2b	D-2c*	D-2d*	D-2e*	--
Break Spectrum	D-3									
Peak Cladding Temp & Max Oxidation vs. Exposure	D-4									
MAPLHGR	D-5									

*SEE QUAD CITIES BWR 3 LEAD PLANT ANALYSIS

SINGLE FAILURE STUDY ON ECC SYSTEM MANUALLY CONTROLLED ELECTRICALLY OPERATED VALVES

The effects of a single failure or operator error that causes any manually controlled, electrically operated valve in the ECC System to move to a position that could adversely affect the ECCS has been studied. The purpose of this evaluation is to determine that any such malfunction does not affect the ECCS more than the results of the worst single failure which is reported in the LOCA calculations performed in accordance with 10CFR50 Appendix K.

The results of the break spectrum analysis show the single failure which results in the maximum calculated peak clad temperature (PCT). For any other single failure to be more significant, its effect on the ECCS must be greater than this single failure. Therefore, a study was made to determine if the malfunction of a manually controlled, electrically operated valve by some unknown cause or by an operator improperly positioning a control switch could affect the ECCS more severely than this failure.

In accordance with appropriate IEEE standards, the ECC System valves are electrically assigned to different divisions of power supply. The effect of an operator improperly actuating a single switch on the control panel is to cause only a single valve to move to an incorrect position. For the operator error of actuating a single switch of the ADS System, the system valves are not actuated. However, the consequences of a malfunction which causes one ADS valve to inadvertently open has been noted.

The summary of the ECCS Valve Single Failure Analysis is provided in the attached Table 4. Comparing the effects of the single valve failure noted in Table I with the results of the Appendix K LOCA analysis, it can be seen that these failures are not more severe than those reported. The single failures considered for the ECCS analysis are presented in Table 5 .

TABLE 4

MONTICELLO

ECCS SINGLE VALVE FAILURE ANALYSIS

SYSTEM	VALVE(S)	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPENED	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN BASIS LOCA
Core Spray	Suction		X	Negate use of one core spray loop
	Injection(s)	X	X	Negate use of one core spray loop
	Test Return	X		Negate use of one core spray loop
High Pressure Coolant Injection	Condensate Suction		X	Utilize Suppression Pool Water
	Suppression Pool Suction Valve	X		Utilize Condensate Storage Tank water
	Suppression Pool Test Return	X		Partial loss of flow due to flow to suppression pool
	Injection(s)	X	X	Negate HPCI
	Turbine Inlet(s)	X	X	Negate HPCI
Low Pressure Coolant Injection	Injection(s)	X	X	Negate use of LPCI
	Minimum Flow	X		Partial flow loss in one loop due to flow to suppression pool
	Cross Tie		X	No LPCI fix: Negate on LPCI Loop (two pumps per loop)
	Test Return	X		No consequence
	HX Bypass		X	Reduce Flow due to HX Pressure Drop
	Pump Suction		X	Negate one loop
Automatic Depressurization System	One Relief Valve	X		Vessel depressurizes faster, increases rate of HPCI injection (assuming the failure of a single ADS valve to open does not affect the results because the effects on small breaks is insignificant with HPCI in operation)

TABLE 5

SINGLE FAILURES CONSIDERED

FOR ECCS ANALYSIS

PLANT	SINGLE FAILURE	REMAINING ECCS
BWR/3	LPCI Injection Valve	2 CS + HPCI + ADS
MONTICELLO	HPCI	2 CS + CPCI + ADS

(Suction Break)

Reference Plant Analysis

The lead plant for this product line BWR is Quad Cities 2. (2)

The 60% DBA, 80% DBA analyses, additional Small Break analyses, Core Spray line break, Feedwater line break, and Main Steam line break analyses for the lead plant are applicable to this plant and are hereby incorporated by reference (3).

REFERENCES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, NED-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 to Victor Stello, Jr., dated, December 20, 1974.

2. Quad Cities Station Special Report No. 15, Supplement C, Unit 2 and Attachment A (Proprietary information).

3. Letter, G. L. Gyorey to V. Stello, "Compliance with Acceptance Criteria of 10CFR50.46," May 12, 1975.

FIGURE A-1.a

NORMALIZED CORE AVERAGE INLET FLOW
FOLLOWING A DESIGN BASIS ACCIDENT

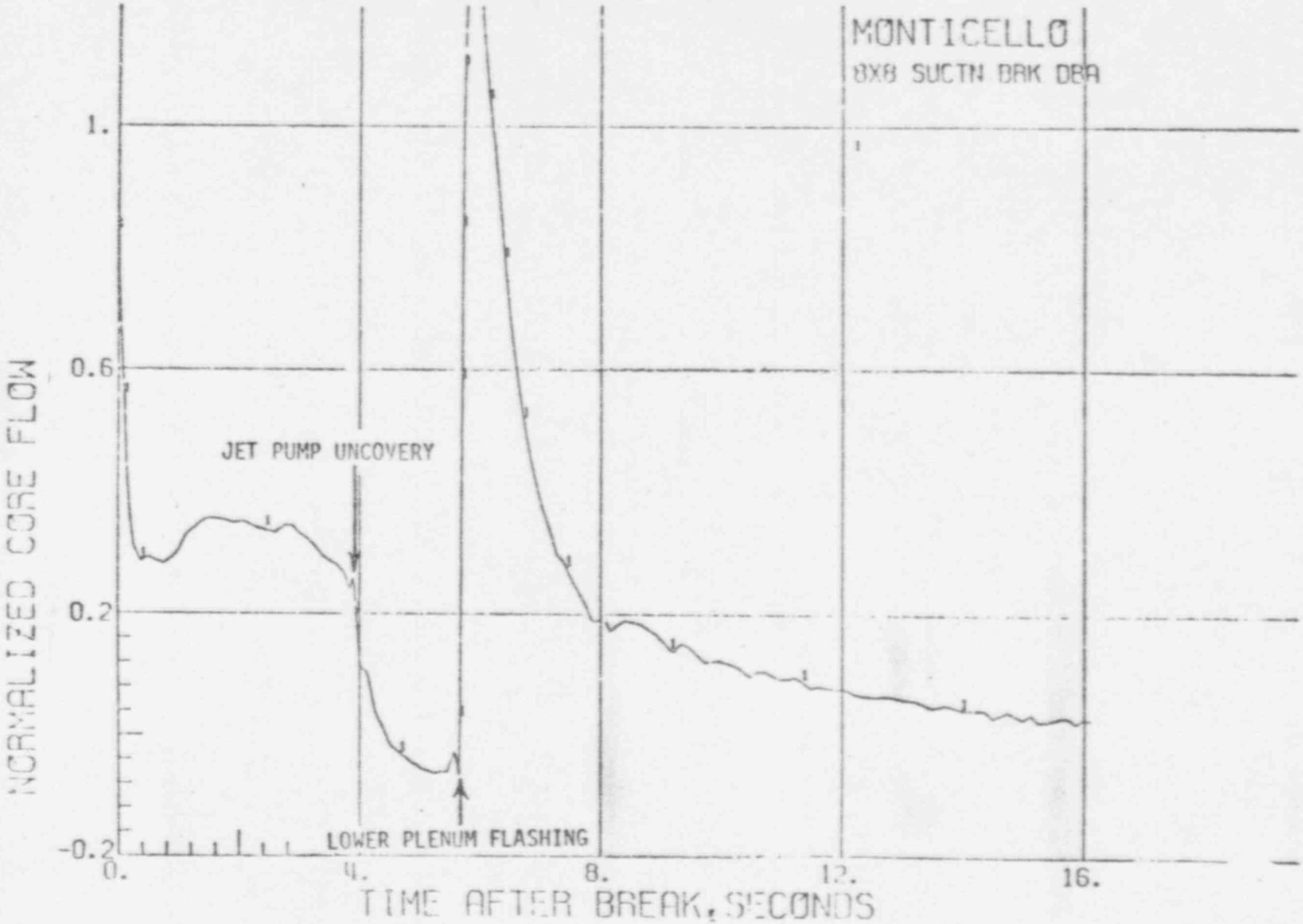


FIGURE A-1d

NORMALIZED CORE AVERAGE INLET FLOW
FOLLOWING A 1sq. ft. BRK

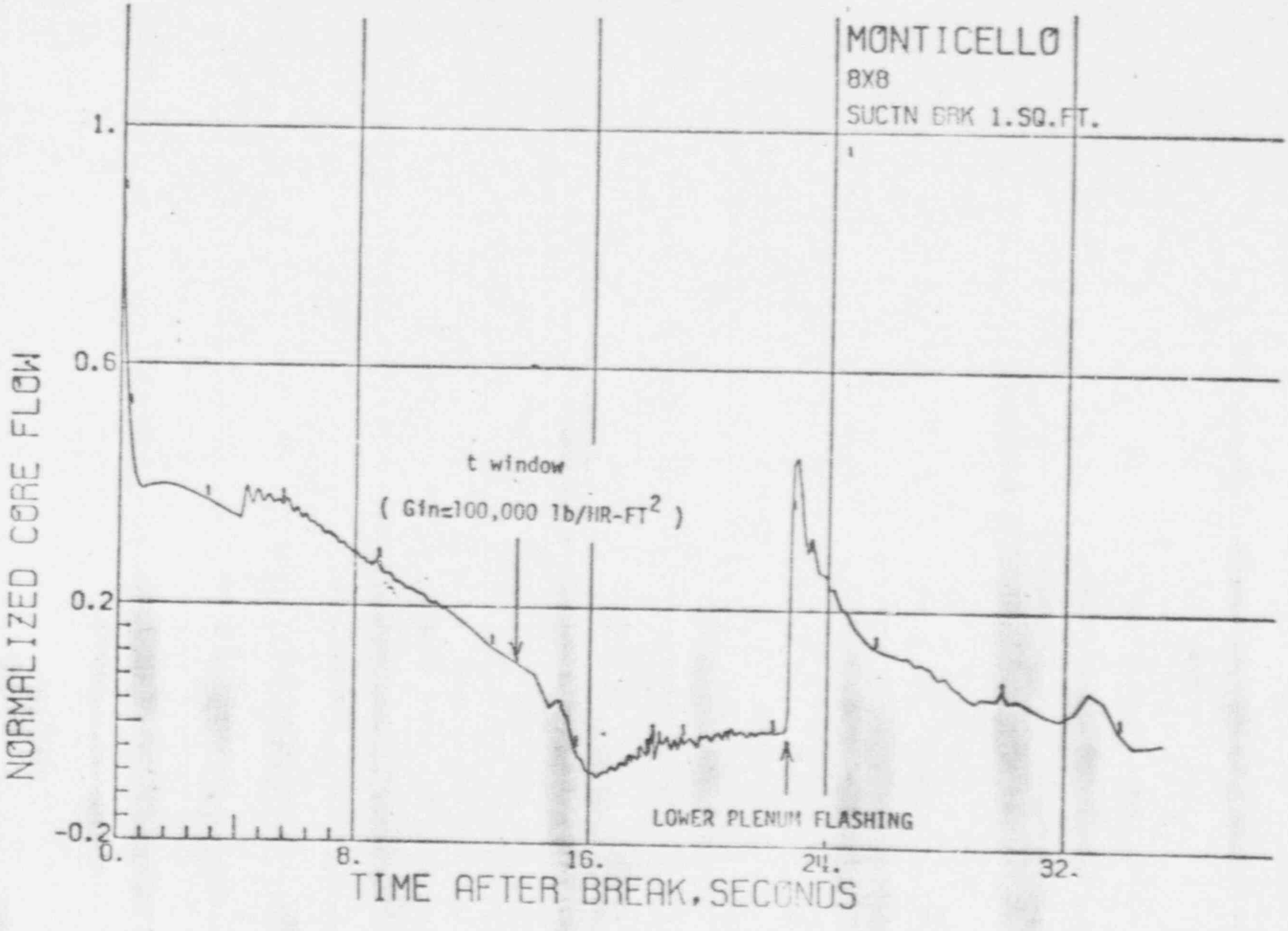


FIGURE A-2a

CORE INLET ENTHALPHY FOLLOWING
A DESIGN BASIS ACCIDENT

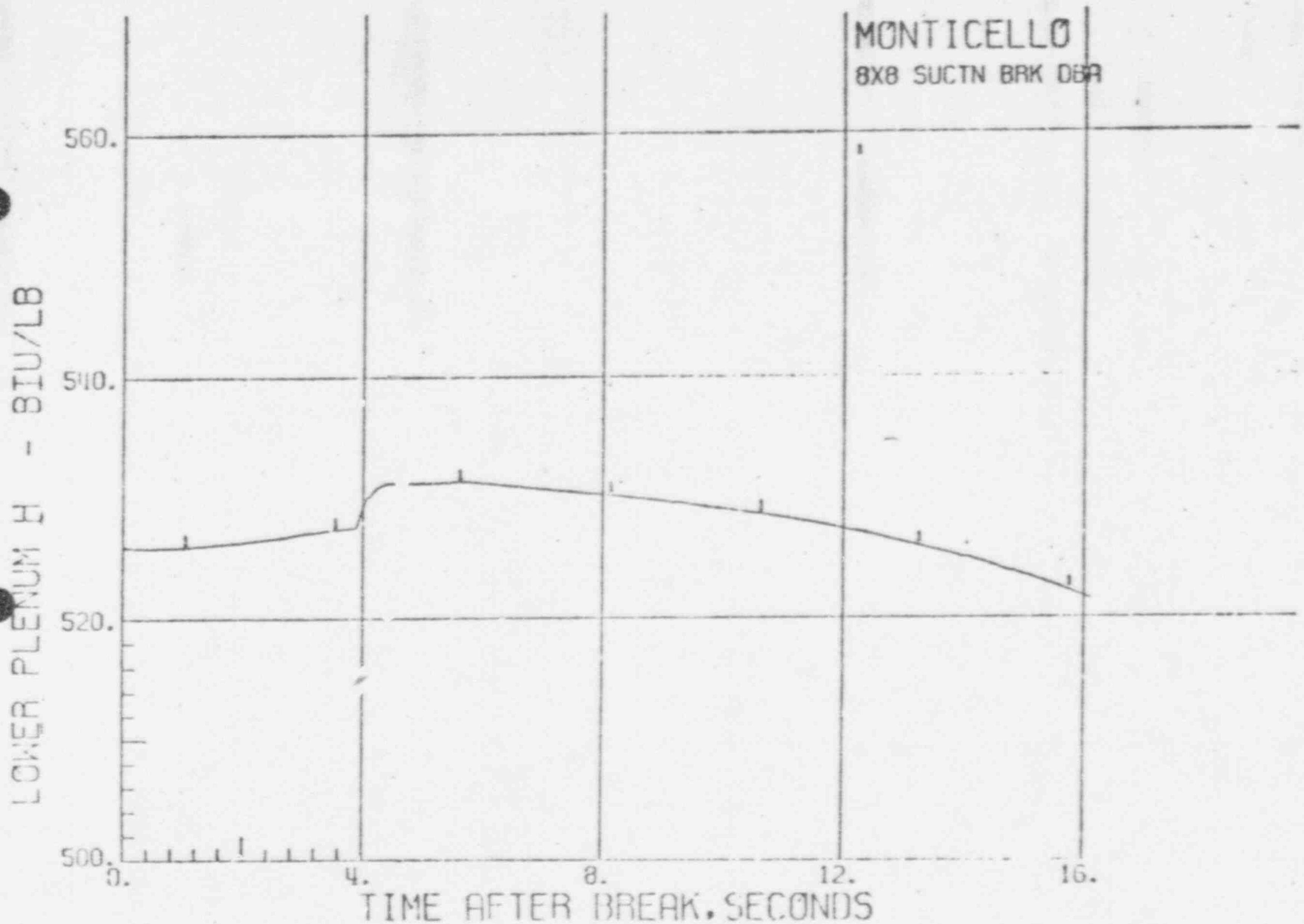


FIGURE A-3a

CORE AVERAGE PRESSURE
FOLLOWING A DESIGN BASIS ACCIDENT

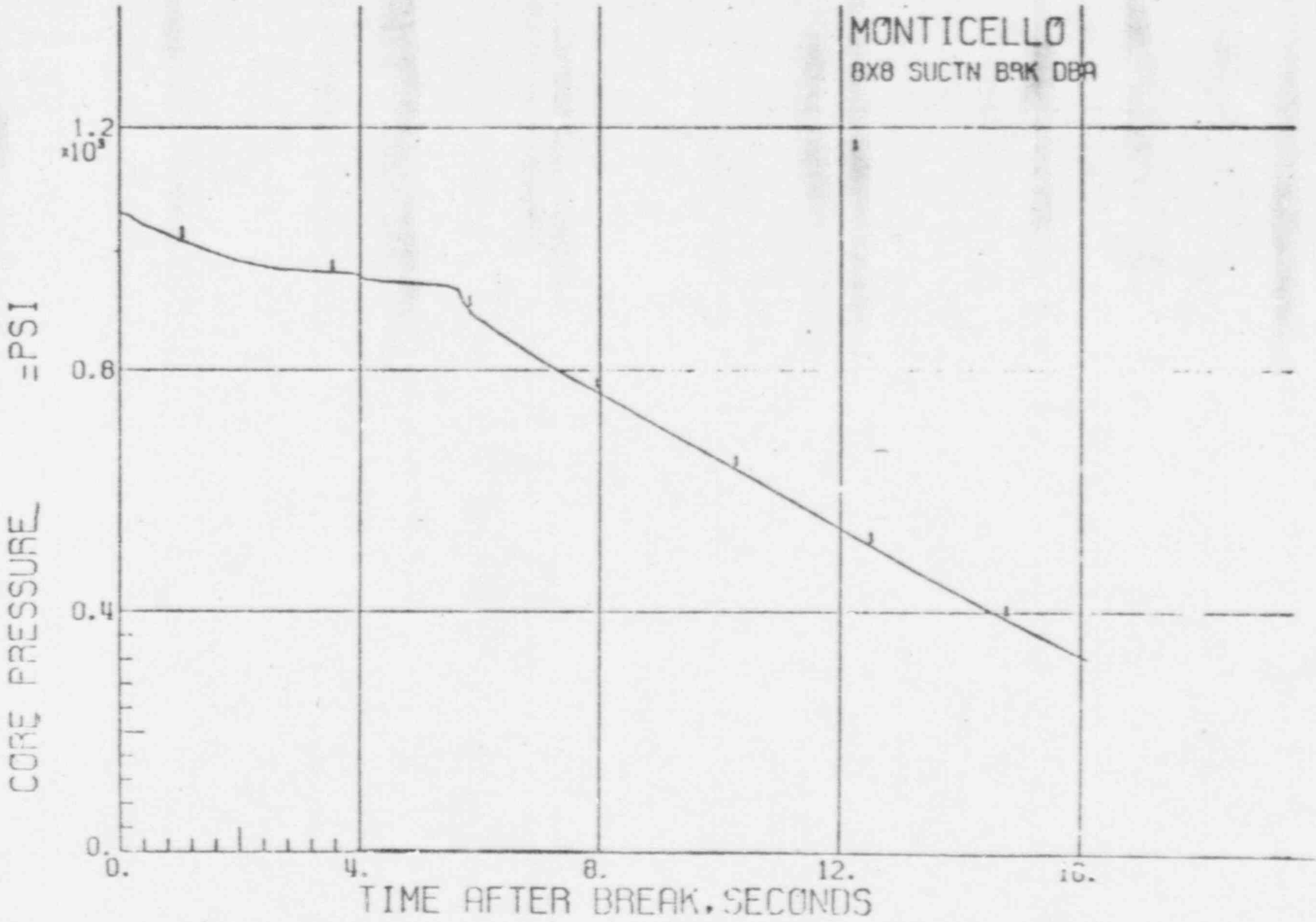
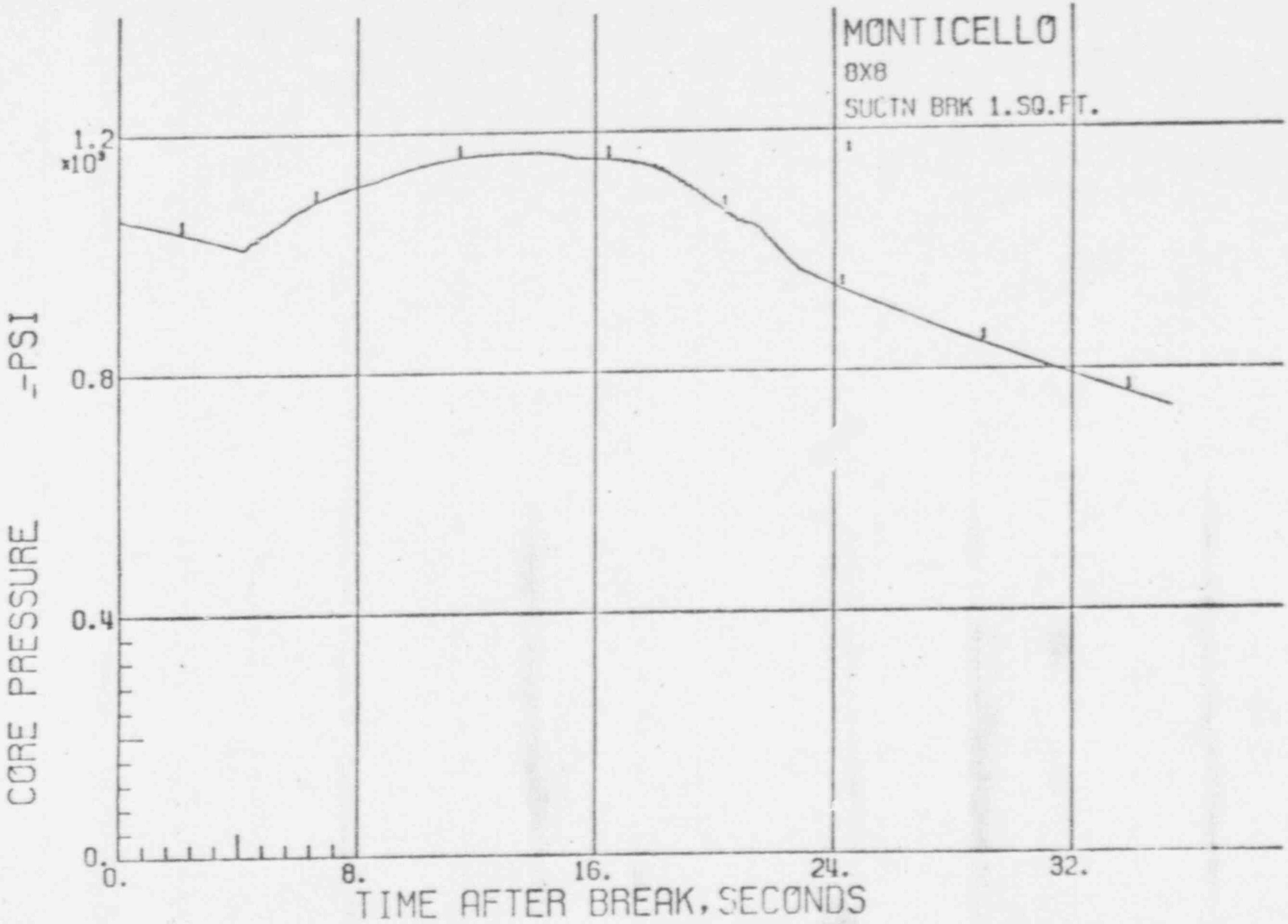


FIGURE A-3d

CORE AVERAGE PRESSURE
FOLLOWING A 1sq. ft. BRK



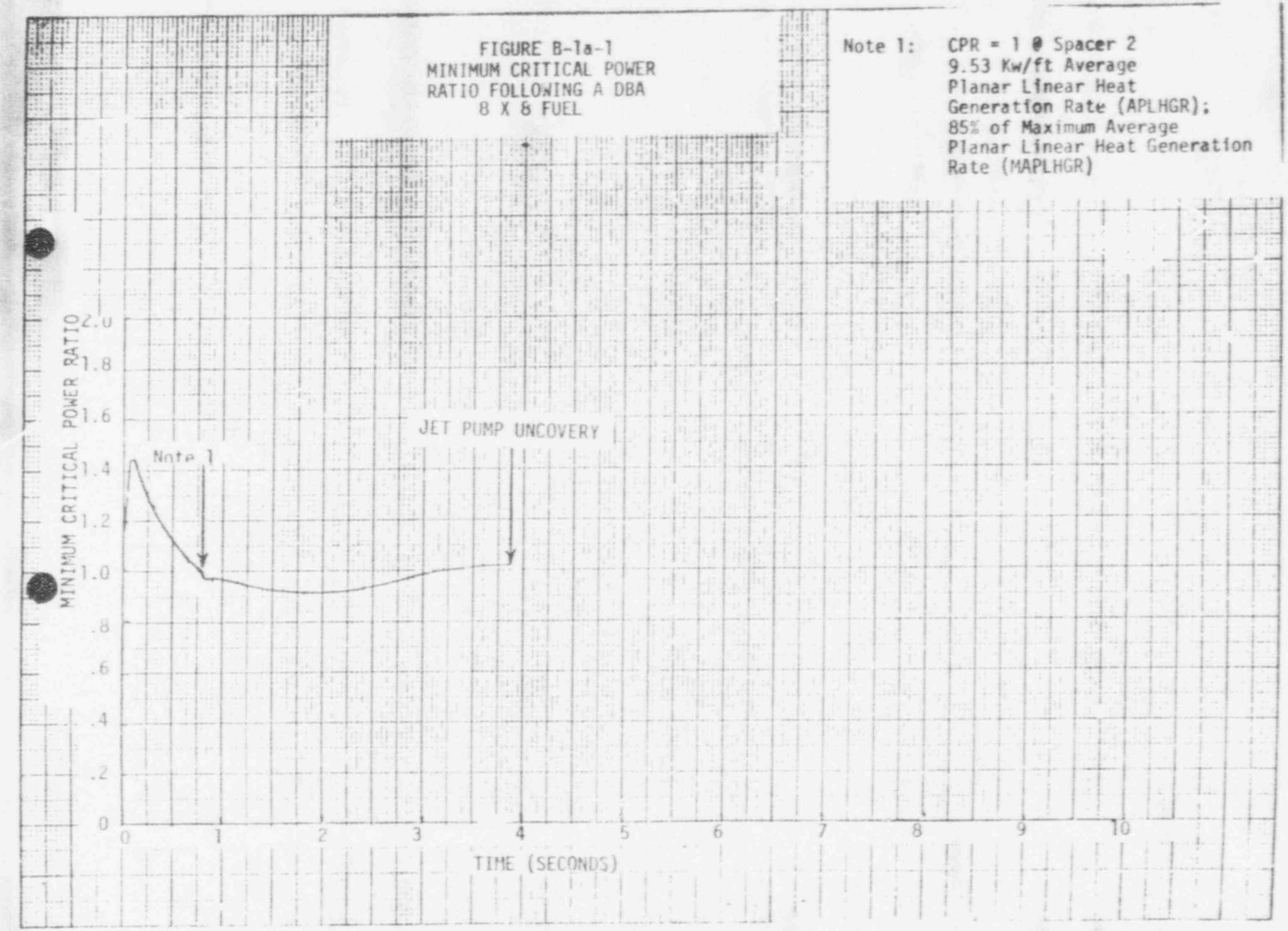
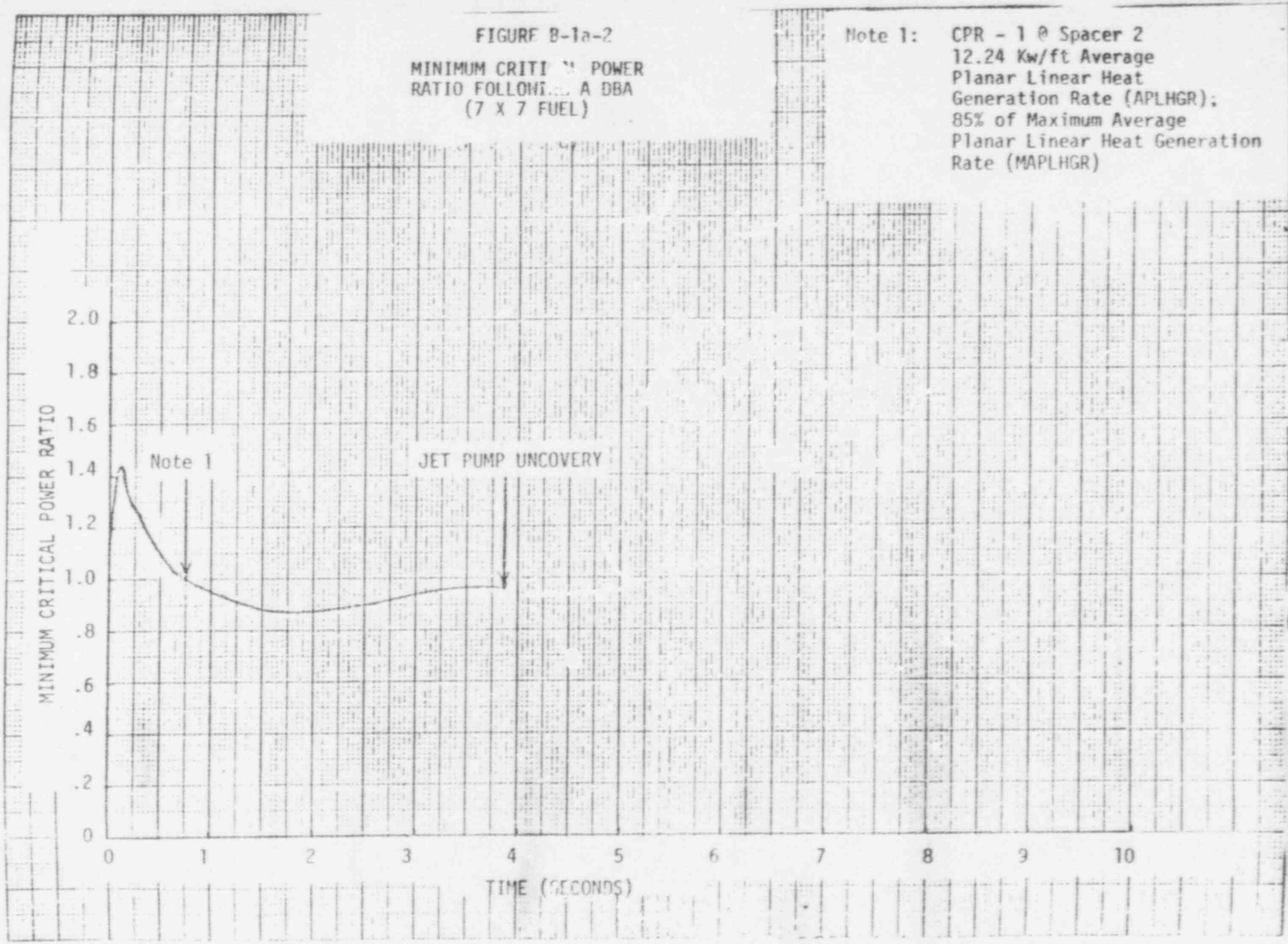


FIGURE B-1a-1
MINIMUM CRITICAL POWER RATIO FOLLOWING A DBA
8 X 8 FUEL

FIGURE B-1a-2

MINIMUM CRITICAL POWER RATIO FOLLOWING A DBA (7 X 7 FUEL)



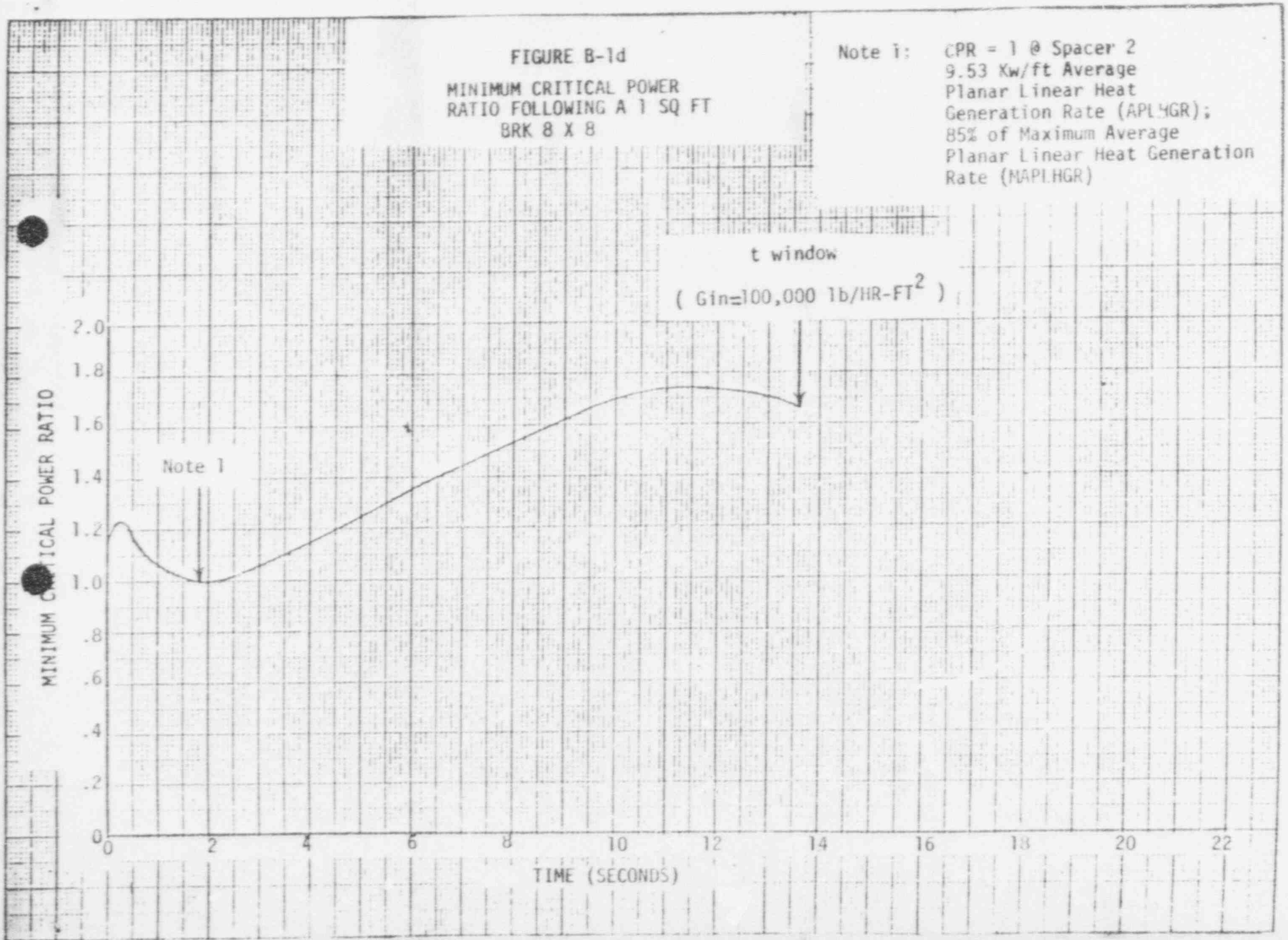
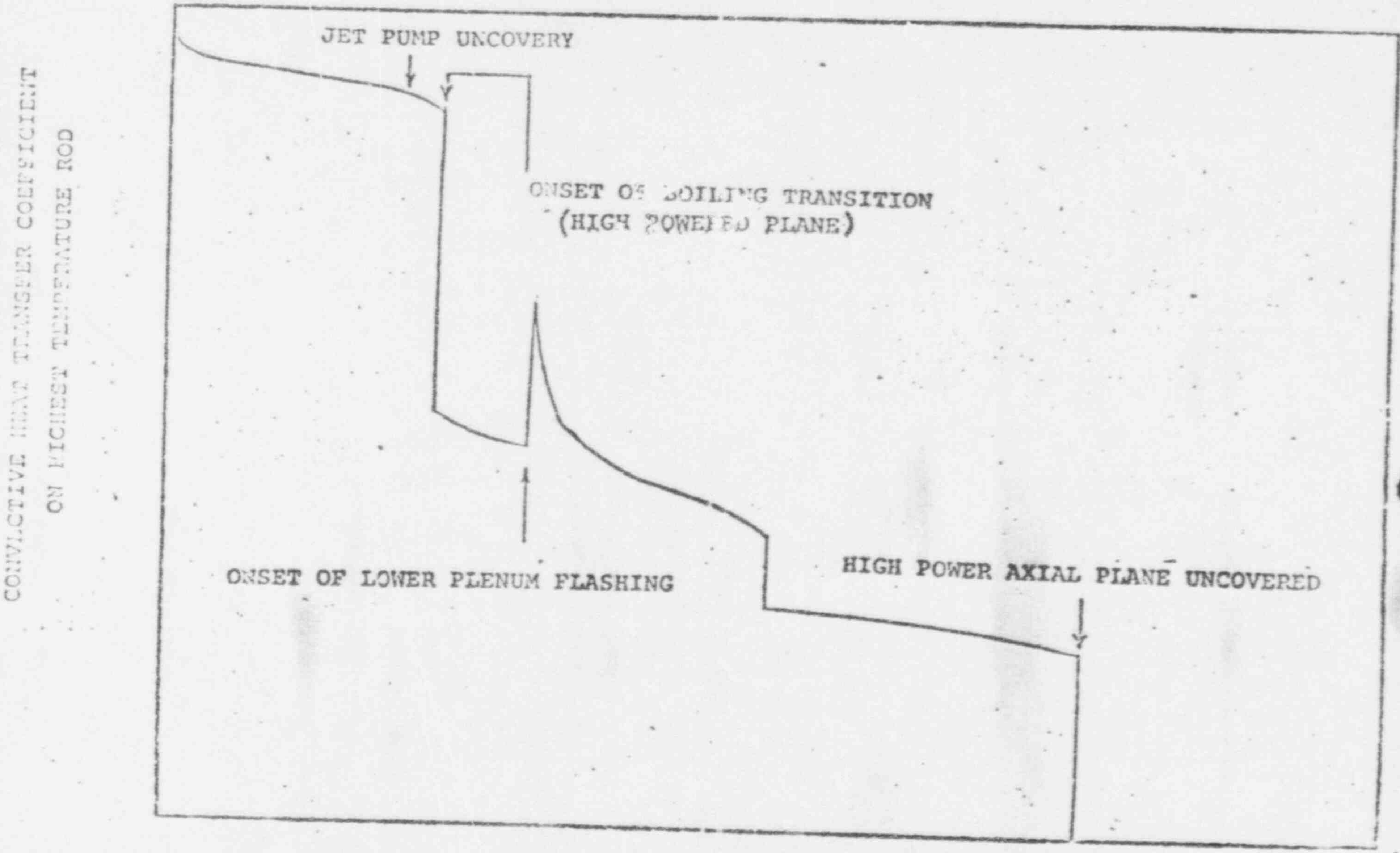


FIGURE B-1d
MINIMUM CRITICAL POWER RATIO FOLLOWING A 1 SQ FT BRK 8 X 8
Note 1: CPR = 1 @ Spacer 2
9.53 Kw/ft Average Planar Linear Heat Generation Rate (APLHGR);
85% of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
t window
(Gin=100,000 lb/HR-FT2)
Note 1
MINIMUM CRITICAL POWER RATIO
0
.2
.4
.6
.8
1.0
1.2
1.4
1.6
1.8
2.0
0
2
4
6
8
10
12
14
16
18
20
22
TIME (SECONDS)

FIGURE B-2

FUEL ROD CONVECTIVE HEAT TRANSFER COEFFICIENT DURING BLOWDOWN AT THE HIGH POWER AXIAL NODE



FIGU C-1

WATER LEVEL INSIDE THE SHROUD AND REACTOR VESSEL PRESSURE FOLLOWING A DESIGN BASIS ACCIDENT

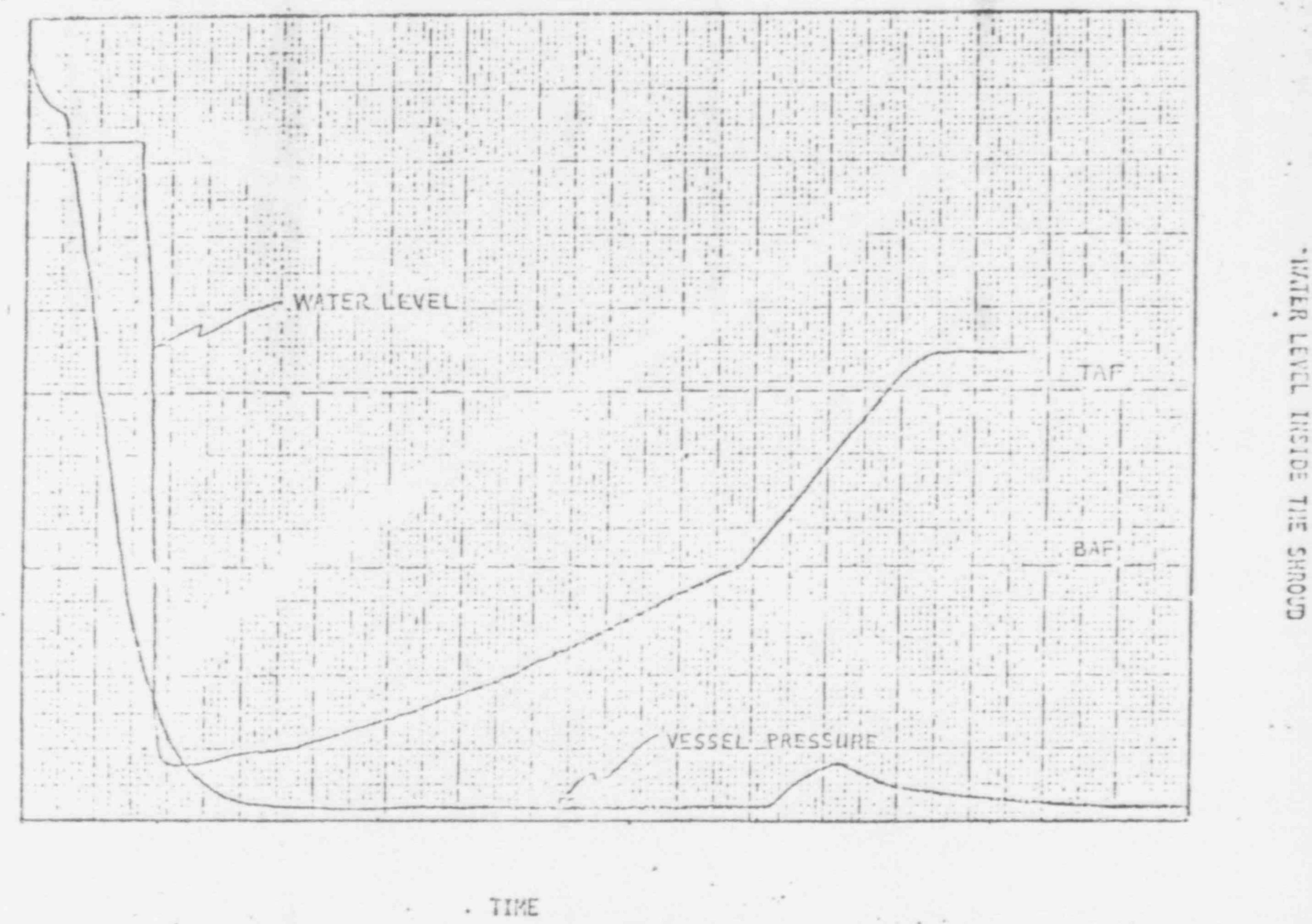


FIGURE C-2

WATER LEVEL INSIDE THE SHROUD AND REACTOR VESSEL PRESSURE FOLLOWING A SMALL BREAK OF THE RECIRCULATION LINE

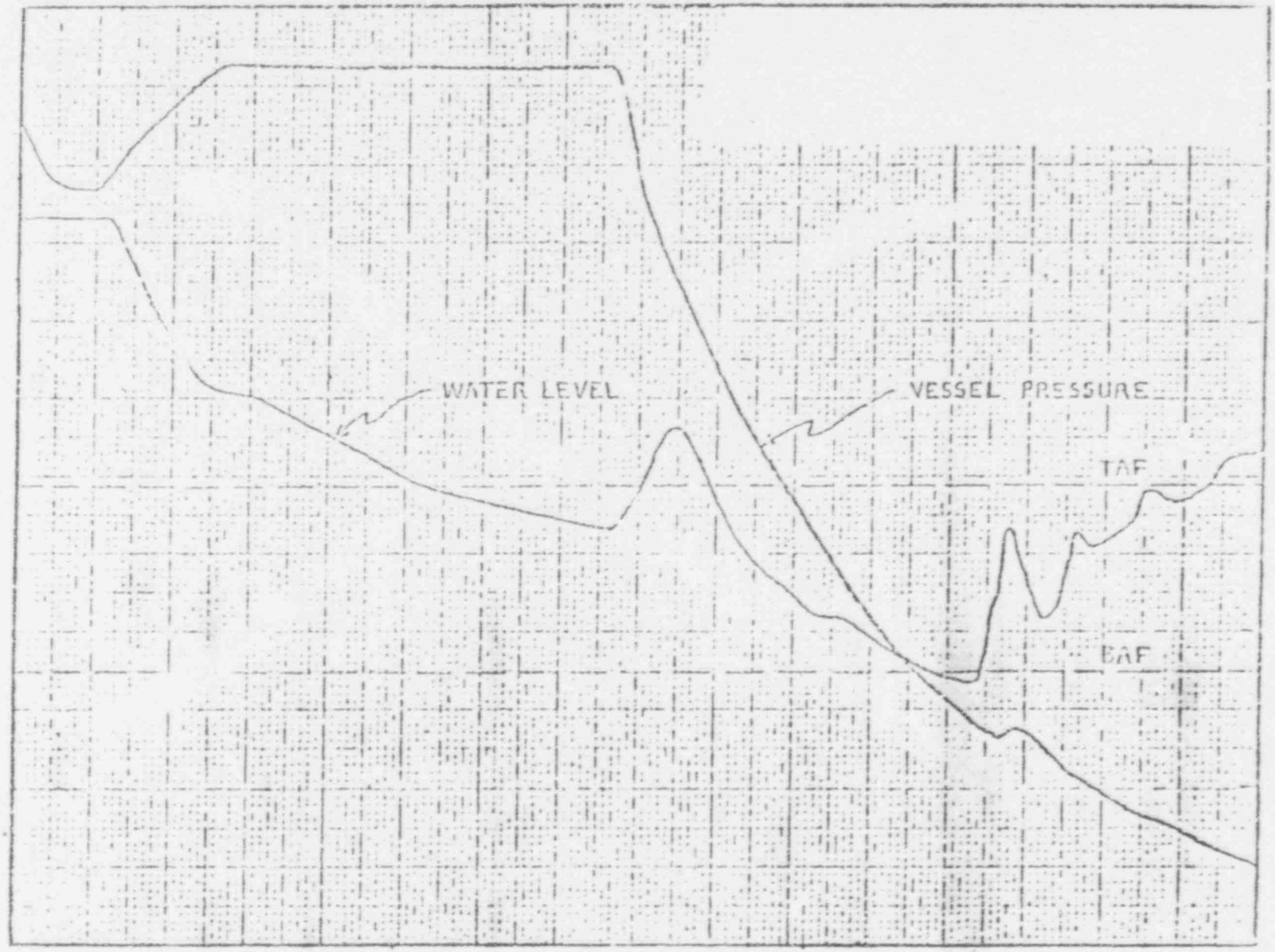


FIGURE D-1

PEAK CLADDING TEMPERATURE FOLLOWING A DESIGN BASIS ACCIDENT

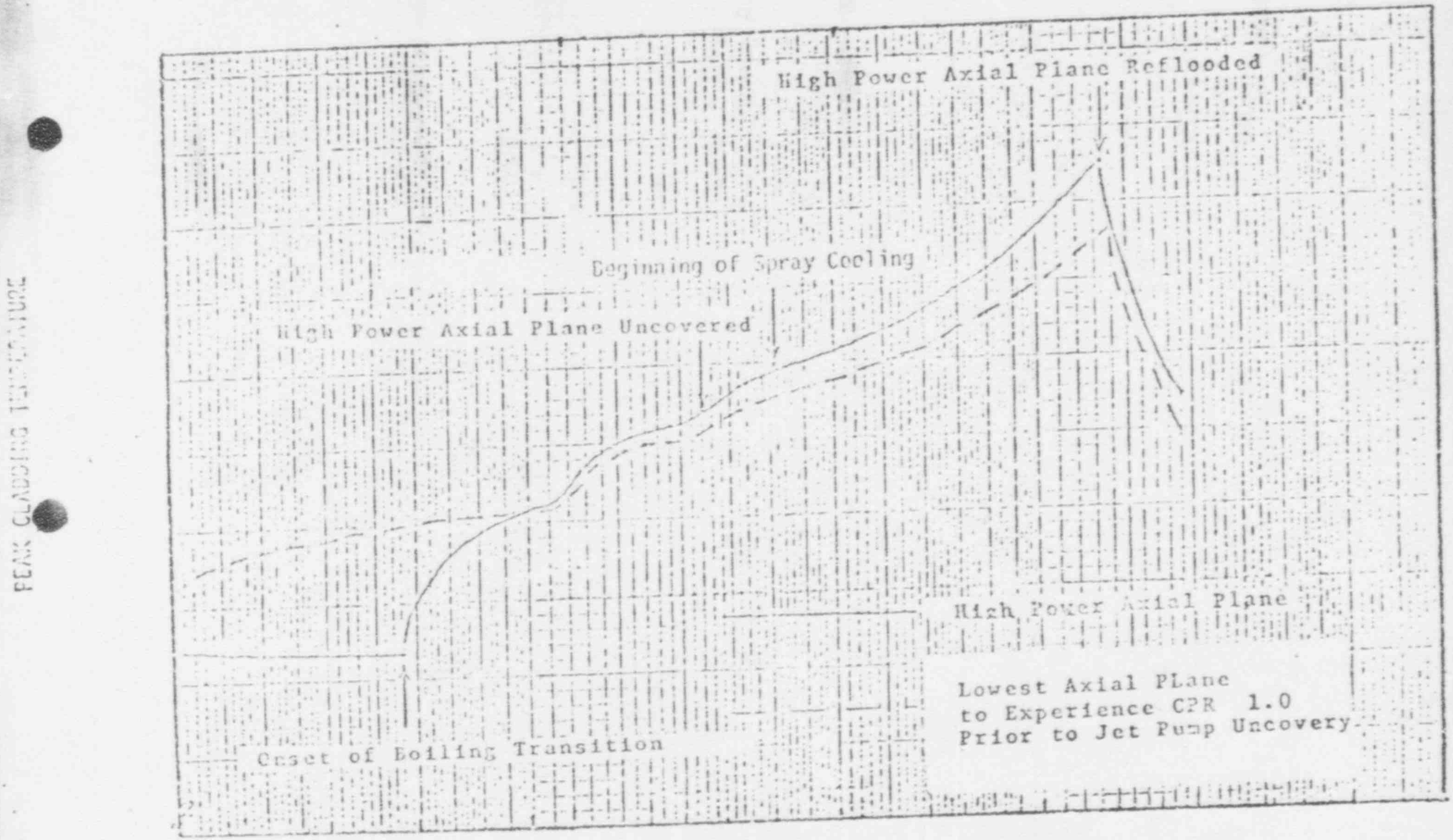


FIGURE D-2

PEAK CLADDING TEMPERATURE FOLLOWING A SMALL BREAK OF THE RECIRCULATION LINE

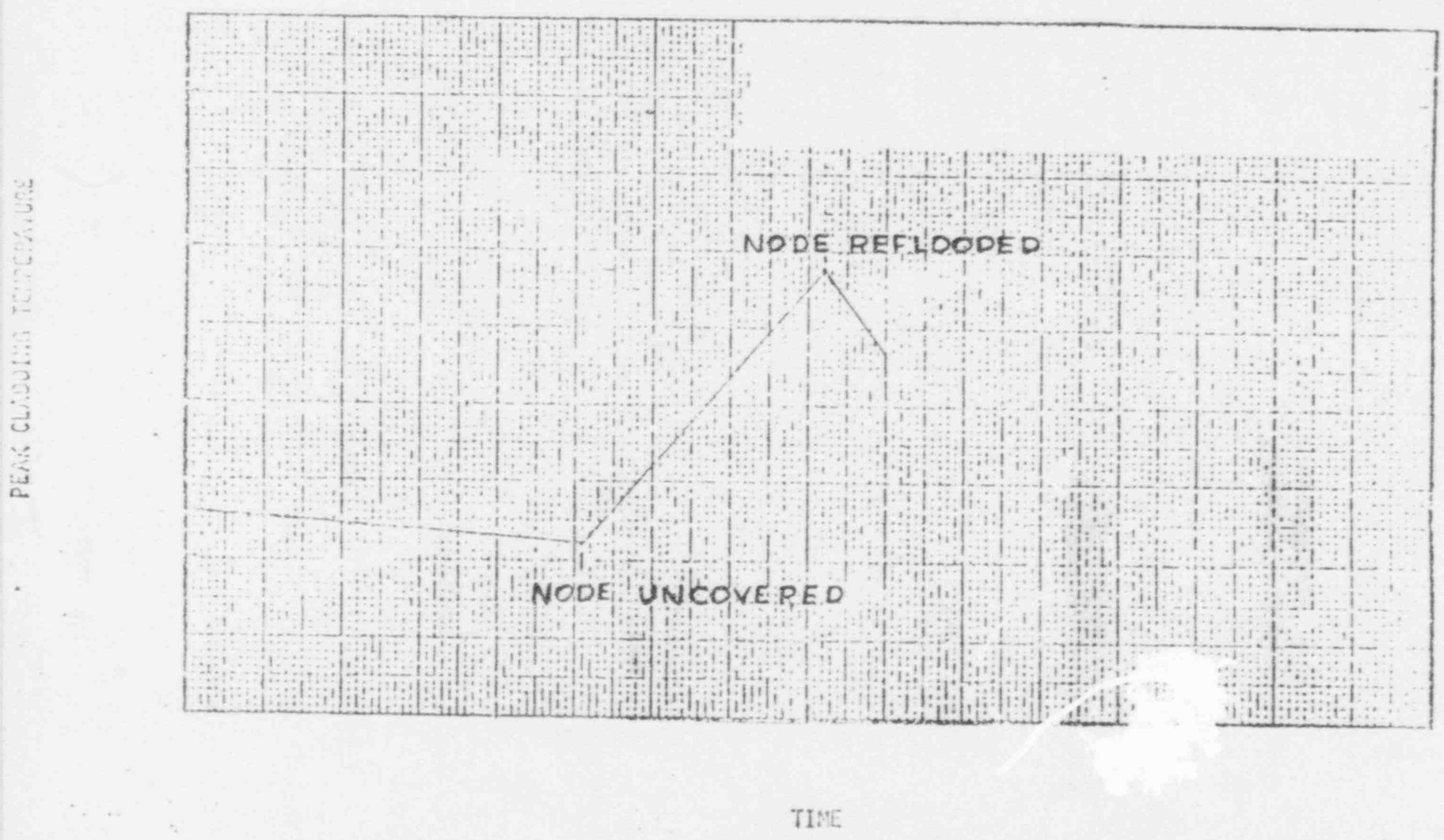


FIGURE D-3

PEAK CLADDING TEMPERATURE AND
LOCAL PEAK OXIDATION
VERSUS BREAK AREA

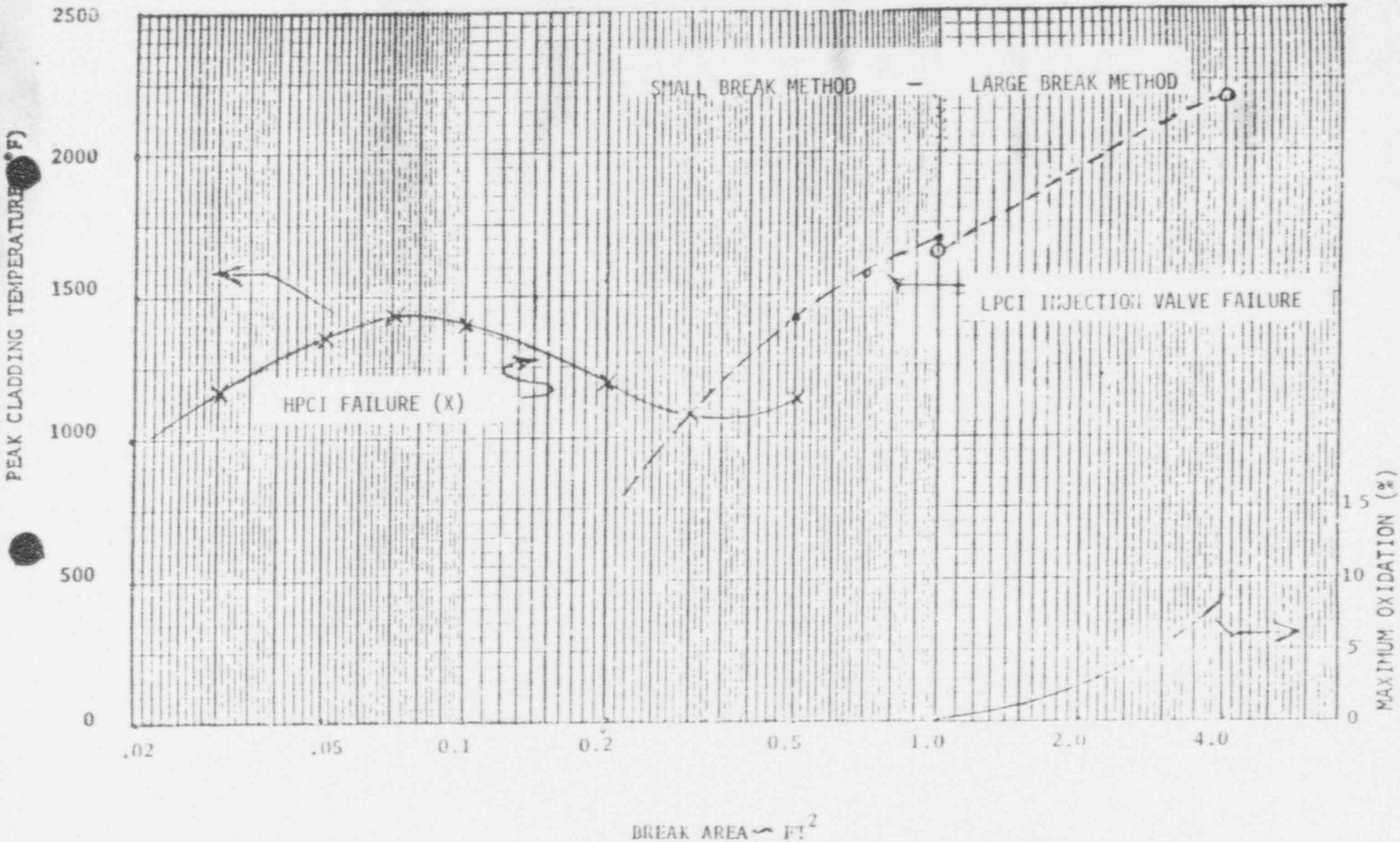
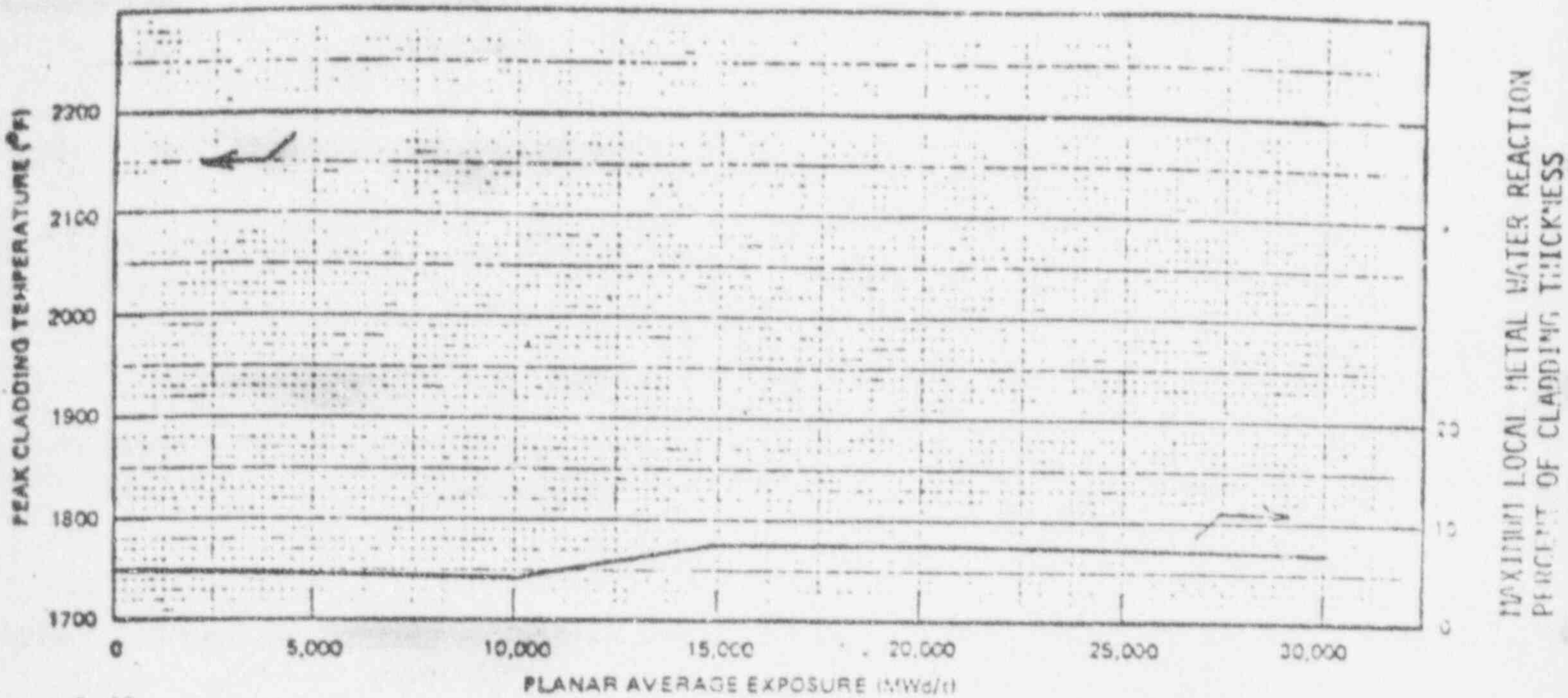
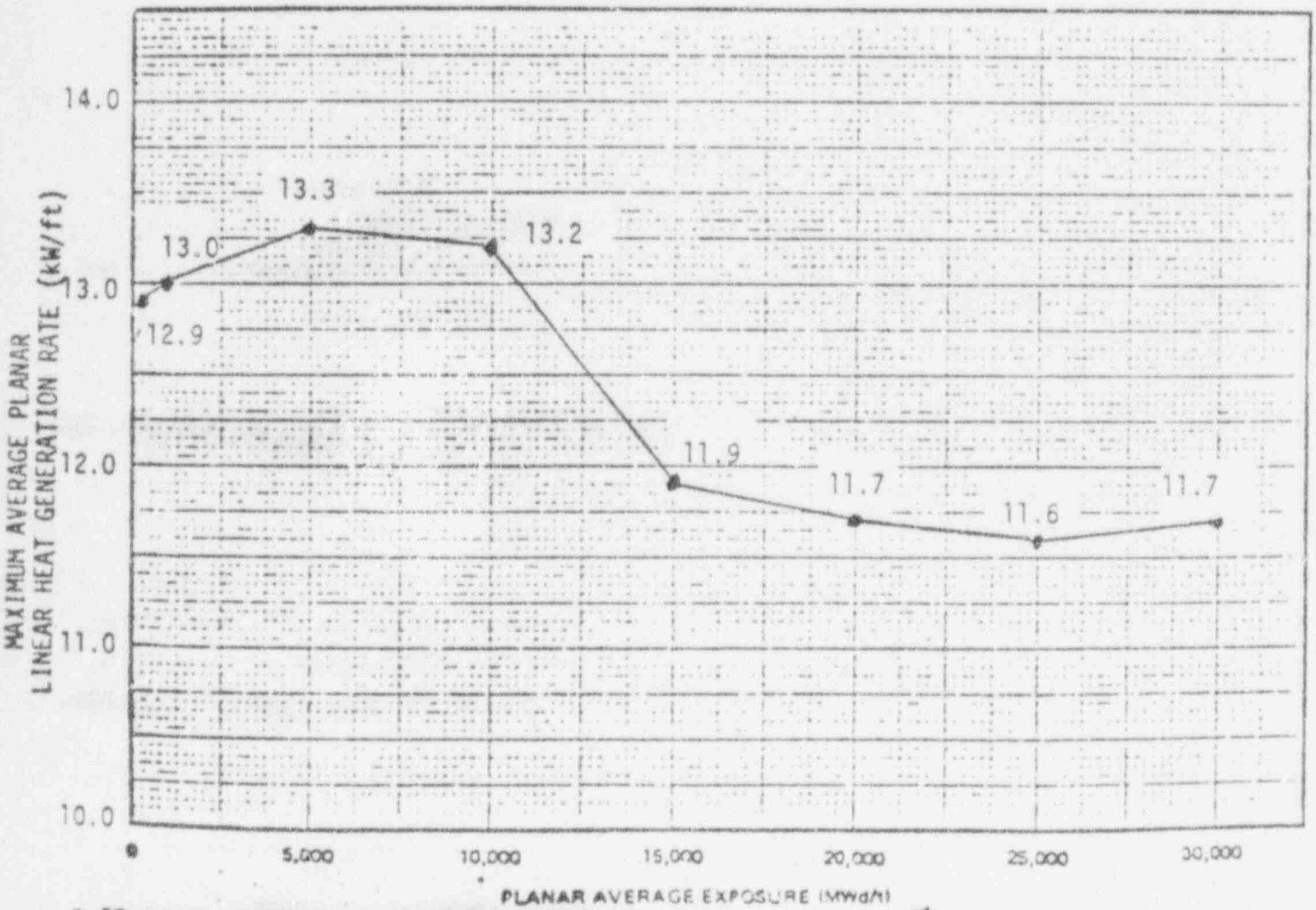


FIGURE D-4B & D-5b



D-4B

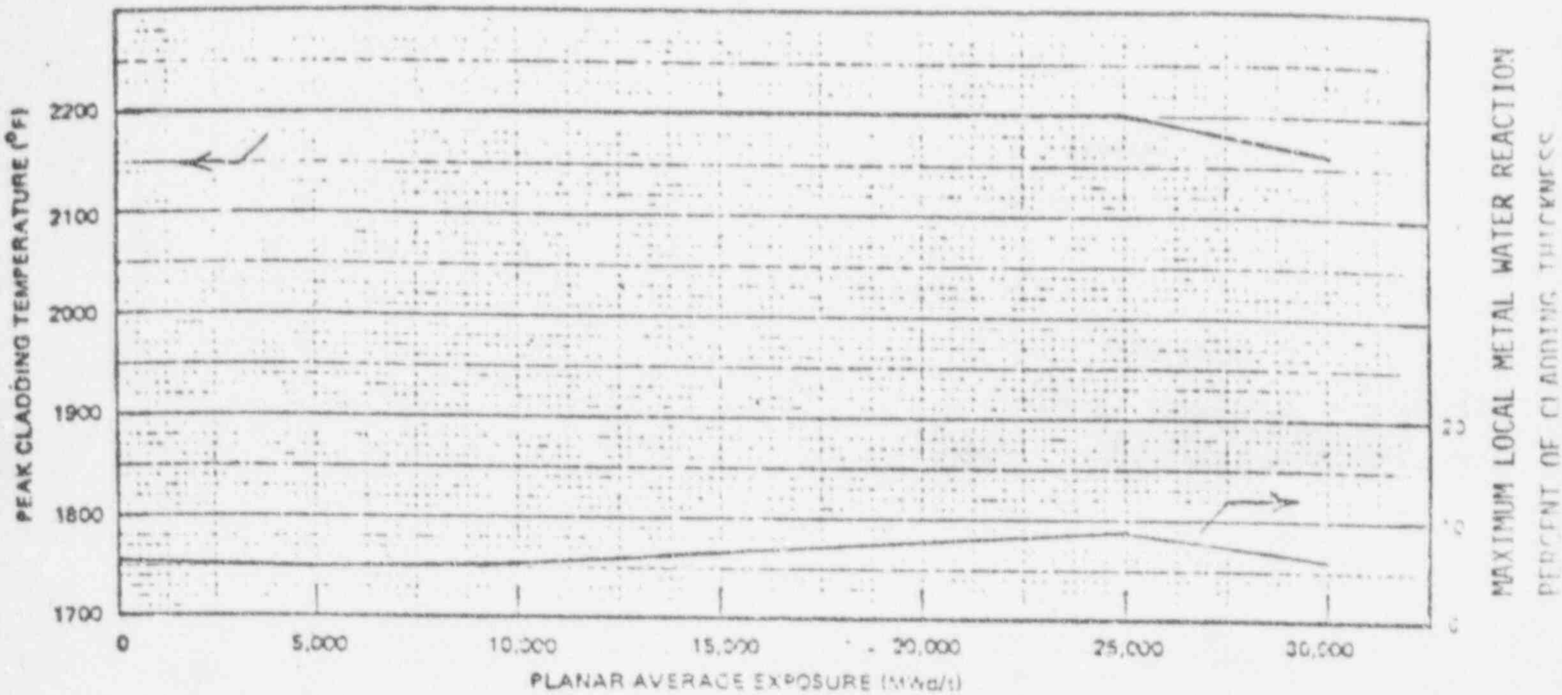
PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE



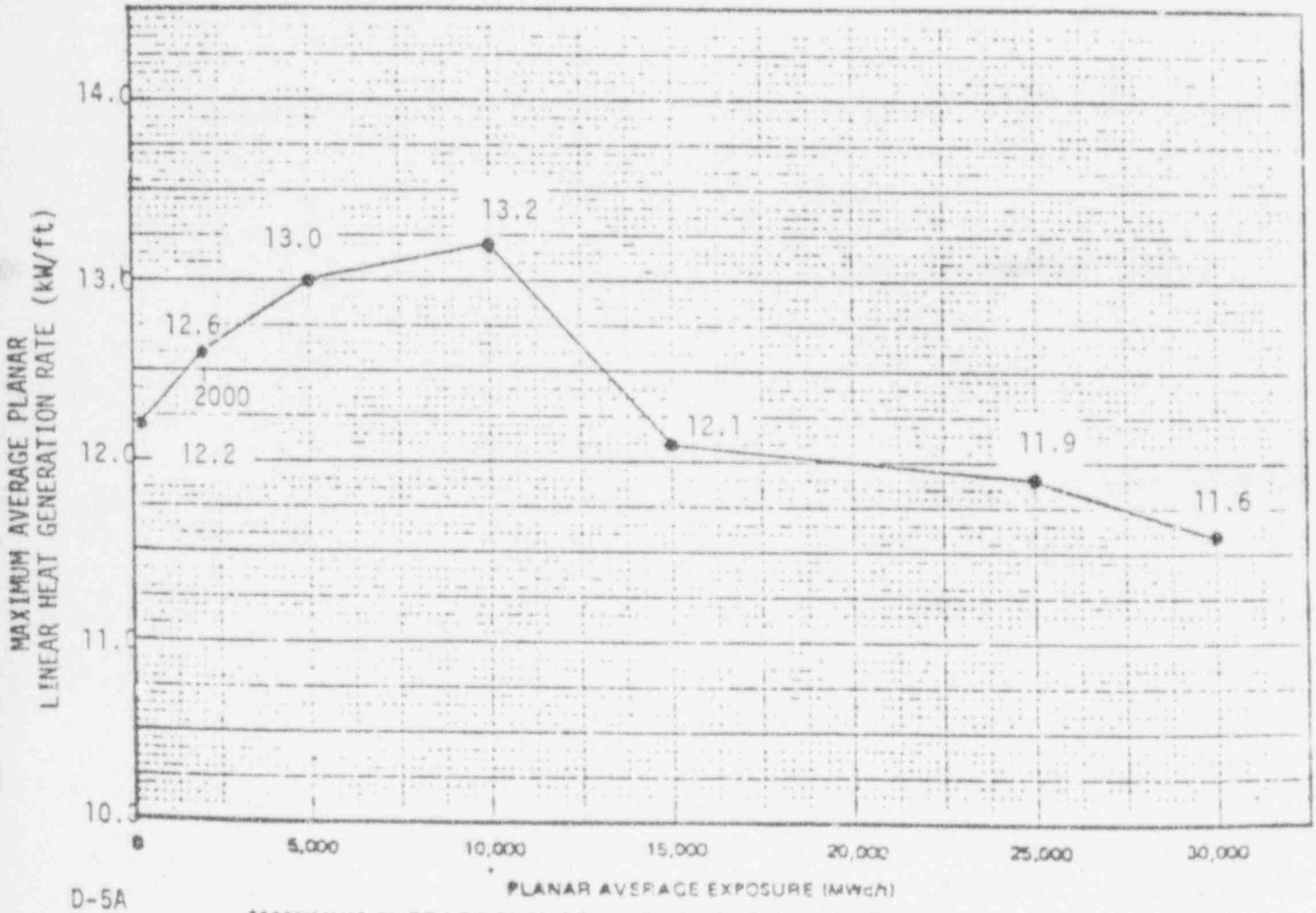
D-5B

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FIGURE D-4A & D-5A



D-4A PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE



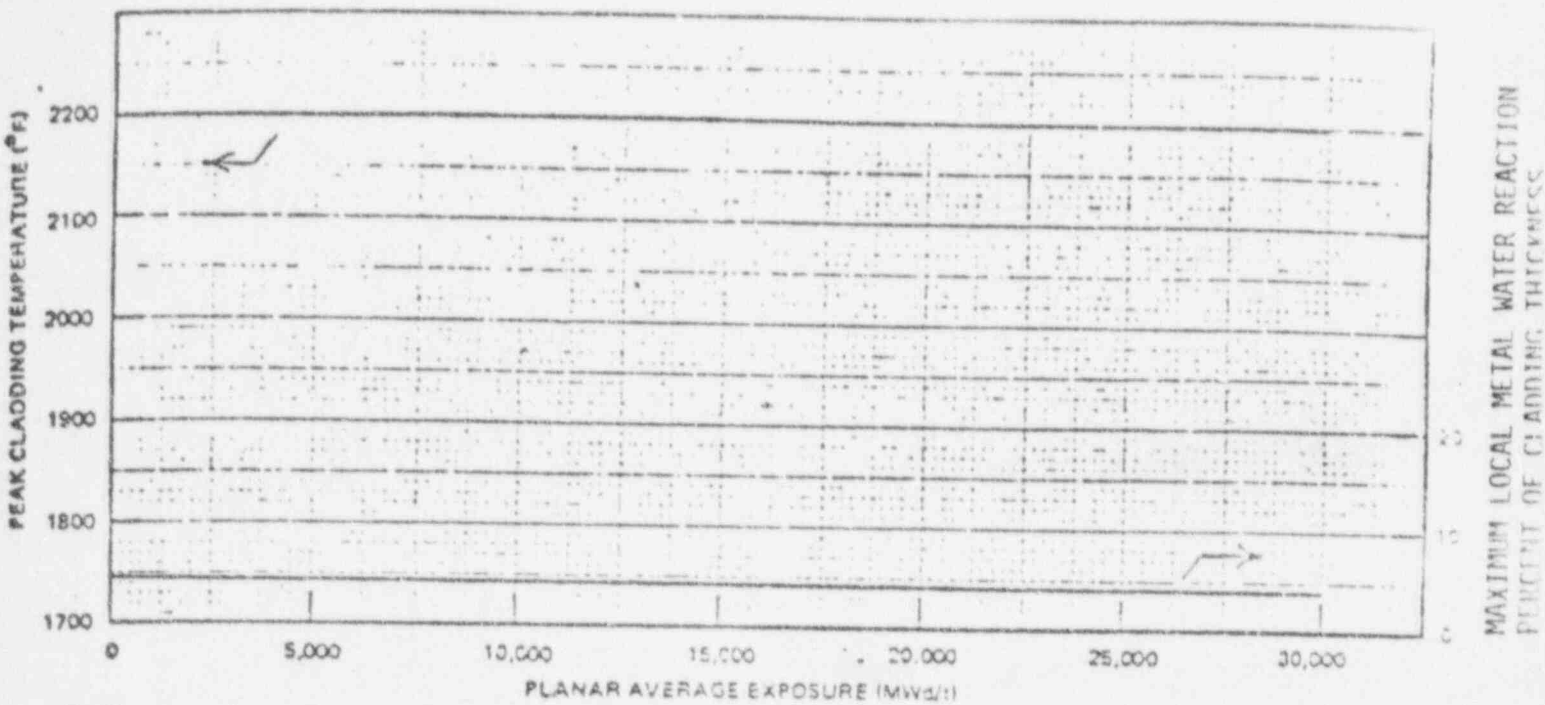
D-5A

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

MONTICELLO

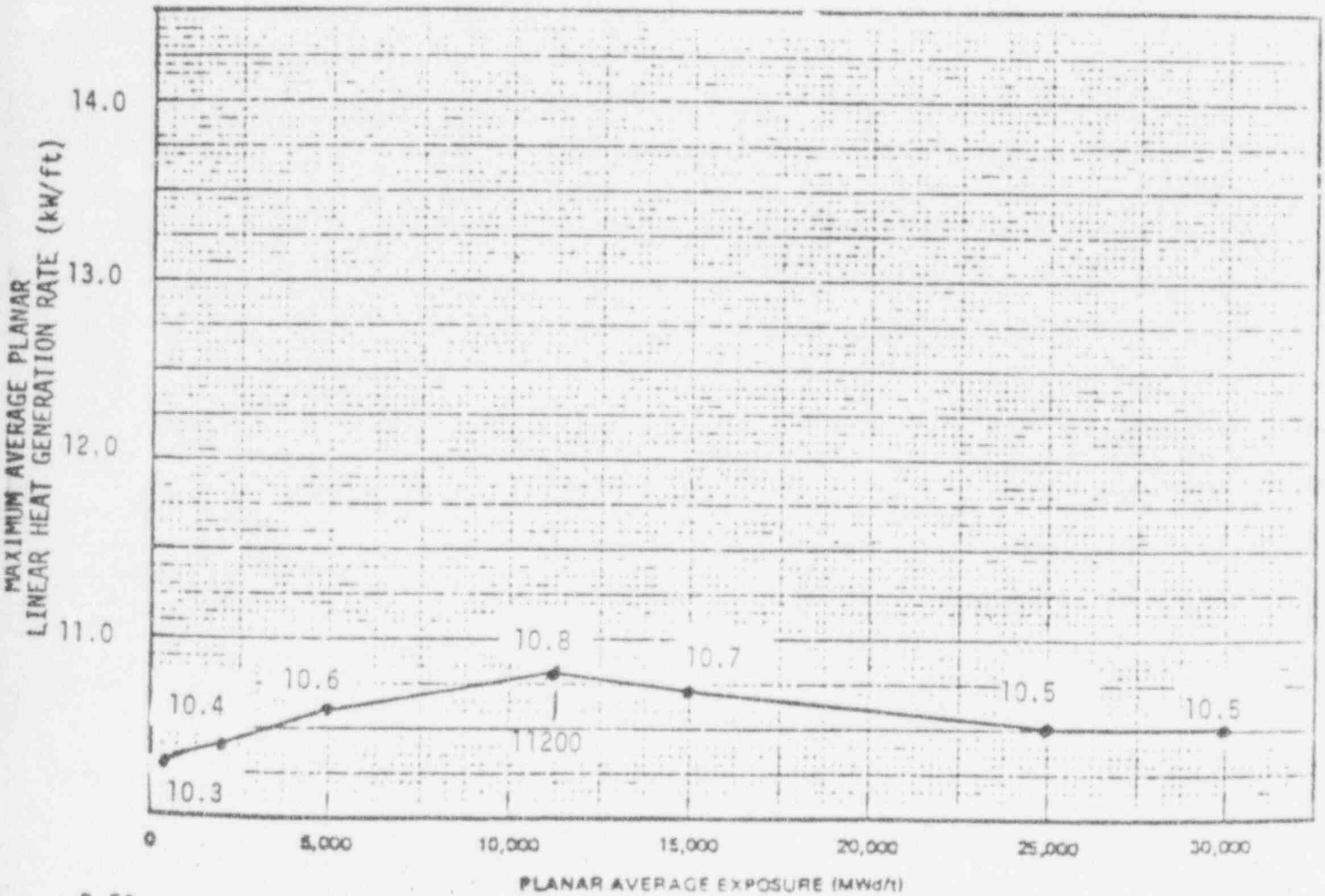
INITIAL CORE 7D225

FIGURE D-4C & 5C



D-4C

PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE

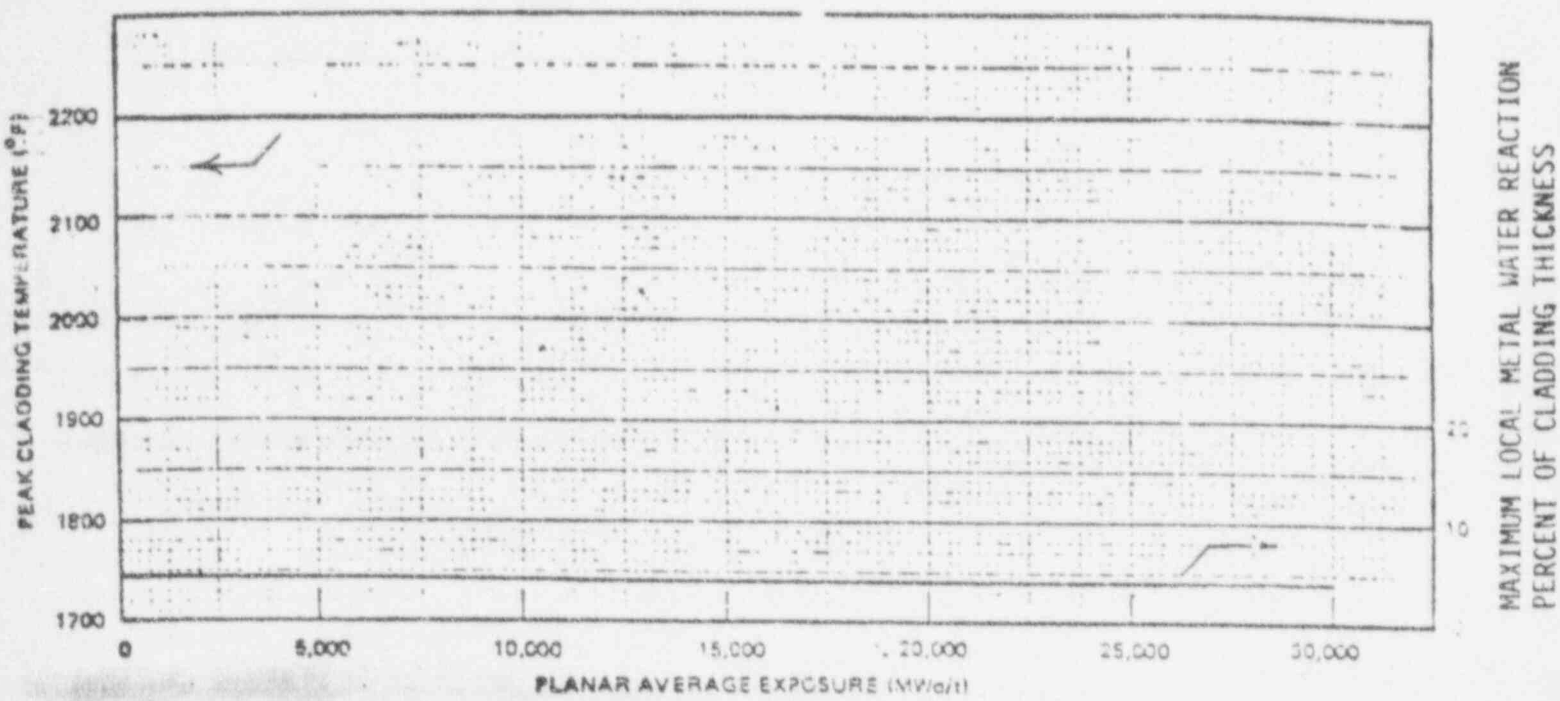


D-5C

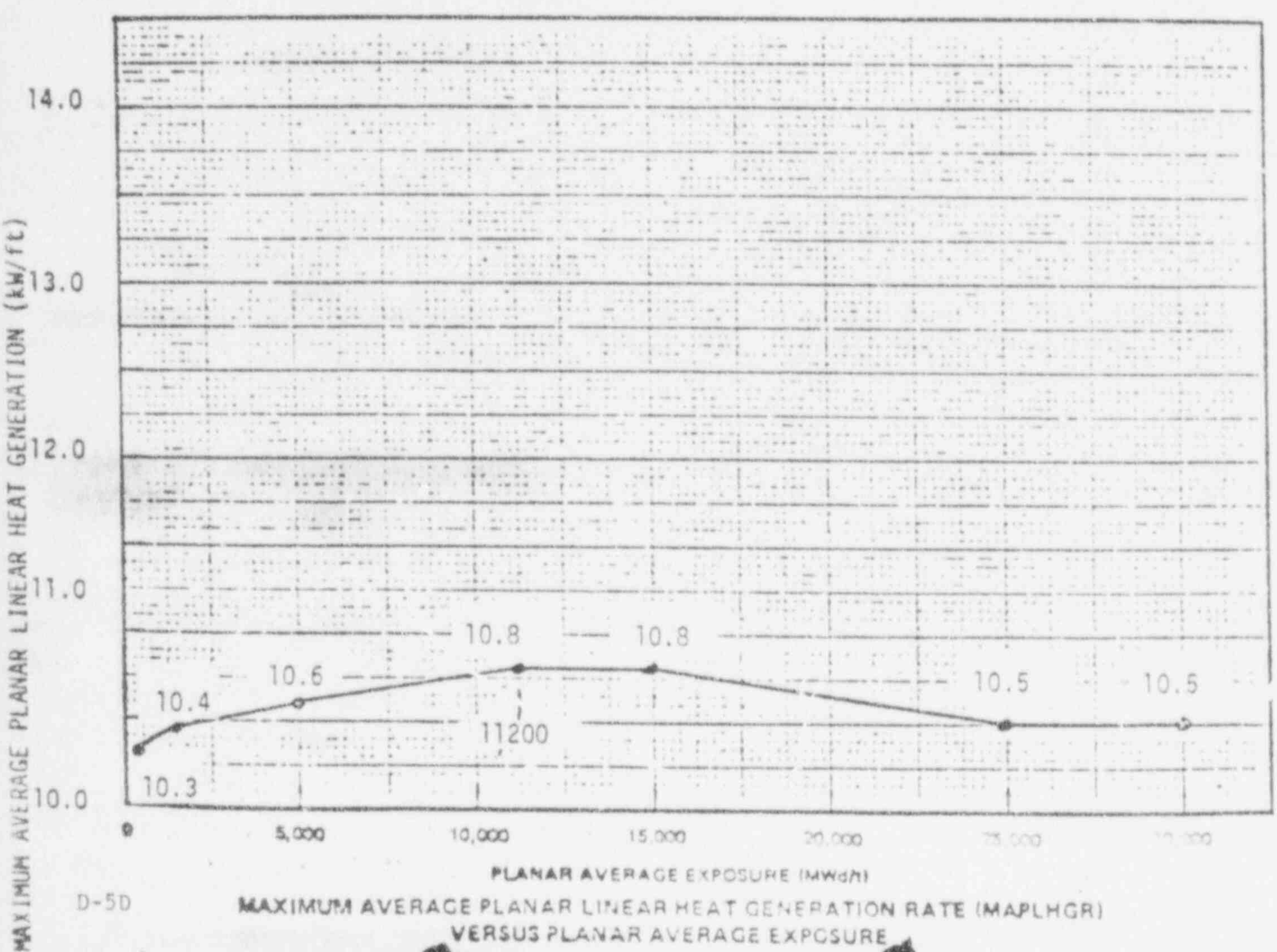
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

MONTICELLO
RELOAD 2 2D262

FIGURE D-4D & D-5D

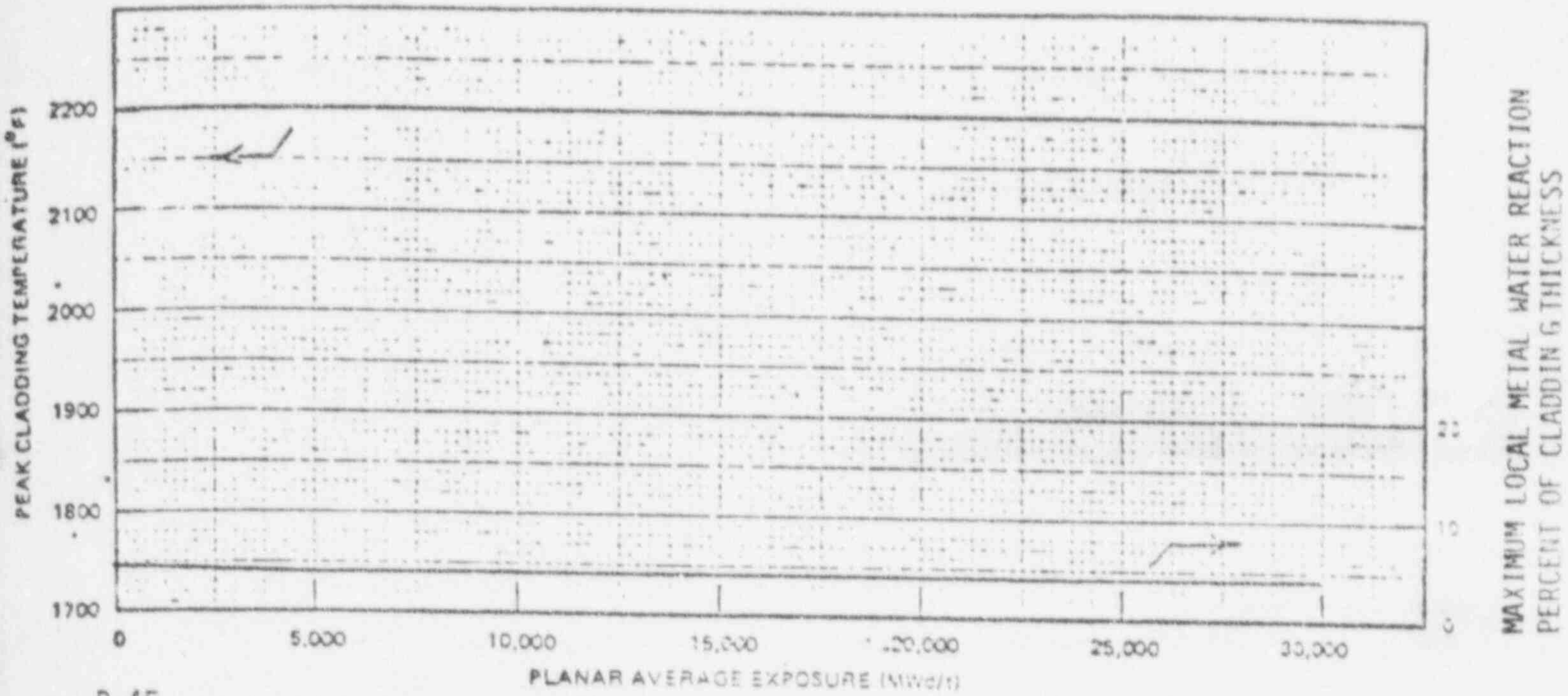


D-4D PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE

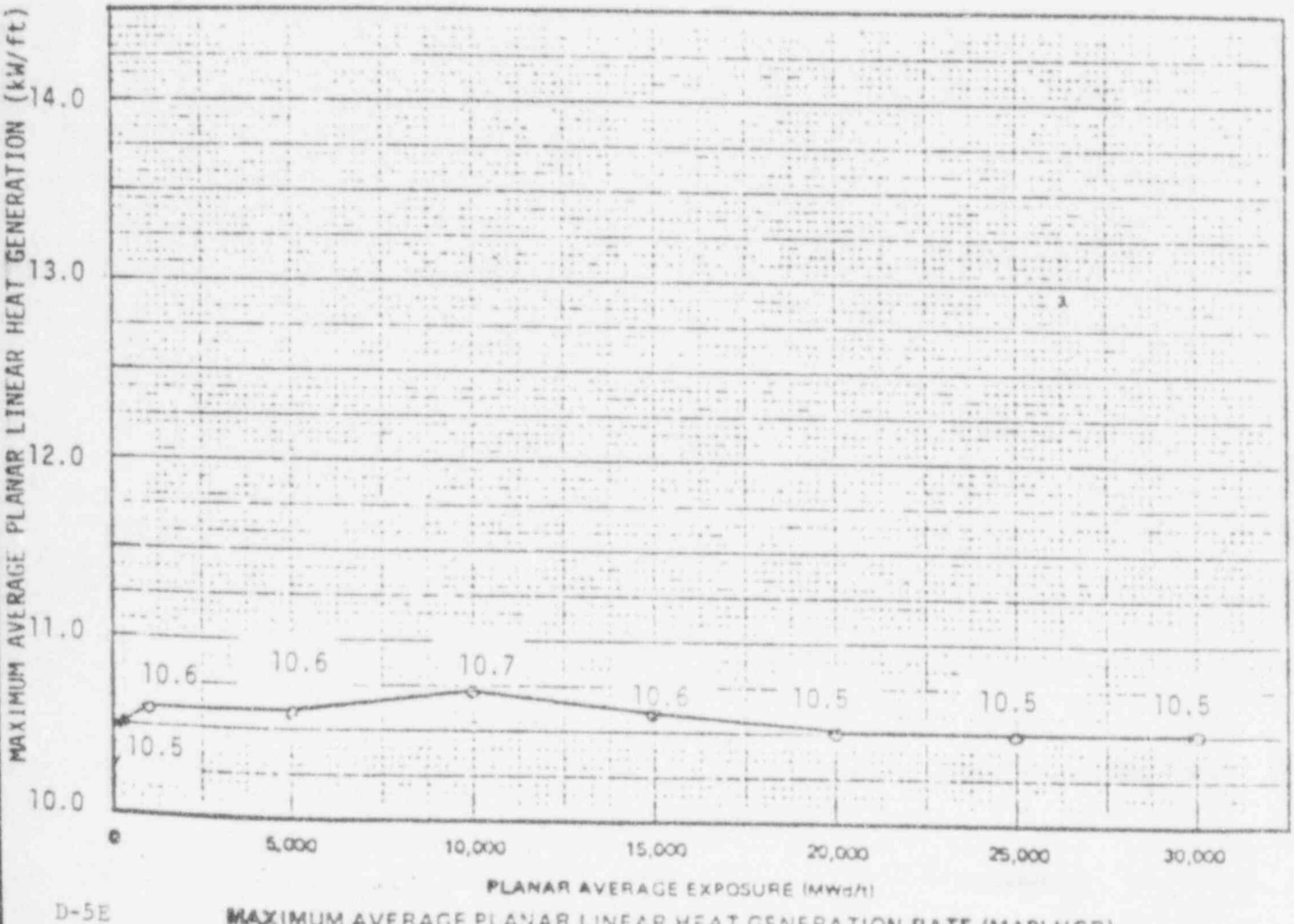


D-5D MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FIGURE D-4E & D-5E



D-4E PEAK CLADDING TEMPERATURE VERSUS PLANAR AVERAGE EXPOSURE



D-5E MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

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