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NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

SECOND RELOAD SUBMITTAL

REVISION 1

FEBRUARY 8, 1974

Prepared By

Northern States
Power Company

Information Supplied By

General Electric
Company

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1. INTRODUCTION

This document provides the technical basis of the license submittal for the second reload of Northern States Power/Monticello Unit 1. Presented herein is a description of the new fuel and the results of the evaluation of the refueled core for the February, 1974 outage.

The fuel at the site available for loading at the outage will be 116 Reload-2 fuel bundles, which are 8x8 bundles with an average enrichment of 2.62 wt% of U-235. There will also be available f.r reinsert 7 initial core bundles discharged at the end of Cycle 1 with an average exposure of about 7400 MWd/t.

The objective of this outage is to remove the 44 temporary control curtains which remained in the core during Cycle 2 and to load the core so as to assure the availability of the plant at high power for approximately an annual cycle.

Sections 3, 4, 5 and 6 of this document, dealing with the subjects of reload fuel mechanical design and reloaded core thermal-hydraulic and nuclear characteristics, and safety analysis, present descriptions of design criteria, methods and results from design calculations and safety evaluations and represents complete information for the review of fuel assembly and core design.

A more detailed summary of General Electric experience with BWR Zircaloy-clad UO_2 pellet fuel, including recent production and development data, has been documented (see Reference 3).

3.4.4 Analysis of Fuel Densification Effects

This section presents results of the effects of densification in the 8x8 reload fuel as determined from application of the models described in Reference 7.

3.4.4.1 Power Spiking Analysis 8x8 Fuel Lattice

An analysis of potential local power spikes due to axial gaps in fuel pellet column for GE BWR's employing an 8x8 fuel lattice design has been performed. This analysis employs the same method and basic assumptions that were reported in Reference 7. Important aspects of this analysis are noted as follows:

1) The equation employed to calculate maximum gap size is that noted in Reference 7:

$$L = \left[\frac{.965 - \rho_i}{2} + 0.0025\right] L$$

where

L = maximum axial gap length

L = fuel column length

ρ_i = mean value of measured initial pellet density (immersion - 5%)

0.0025 = allowance for irradiation induced cladding growth and axial strain caused by fuel-clad mechanical interaction

2) The magnitude of the power spike versus gap size for fuel rods of the 8x8 design is shown in Figure 3-3 for normal operating conditions, and Figure 3-4 for cold zero void conditions.

FIGURE 3-3

8x8 POWER SPIKE VERSUS GAP SIZE - NORMAL OPERATION

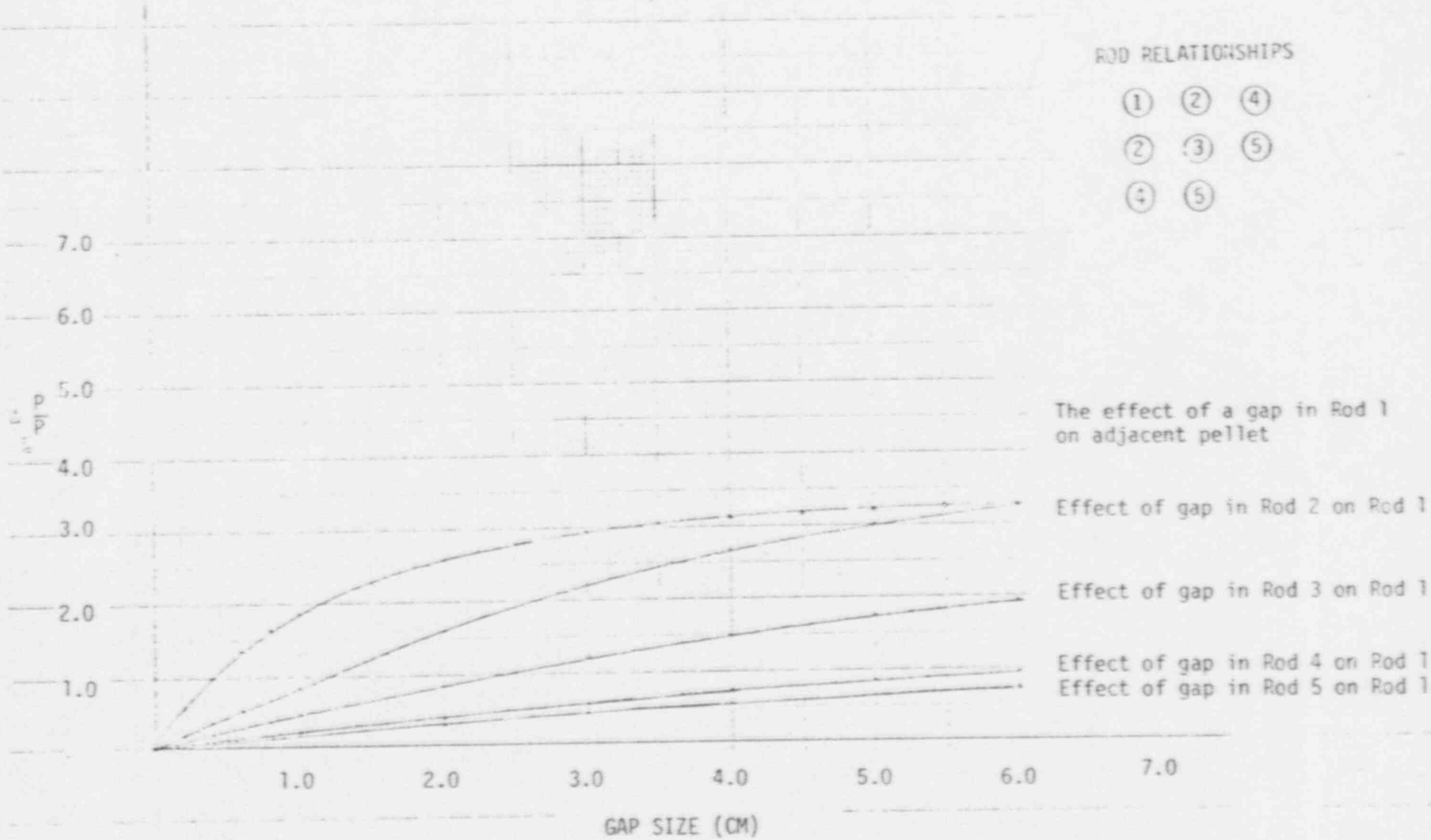

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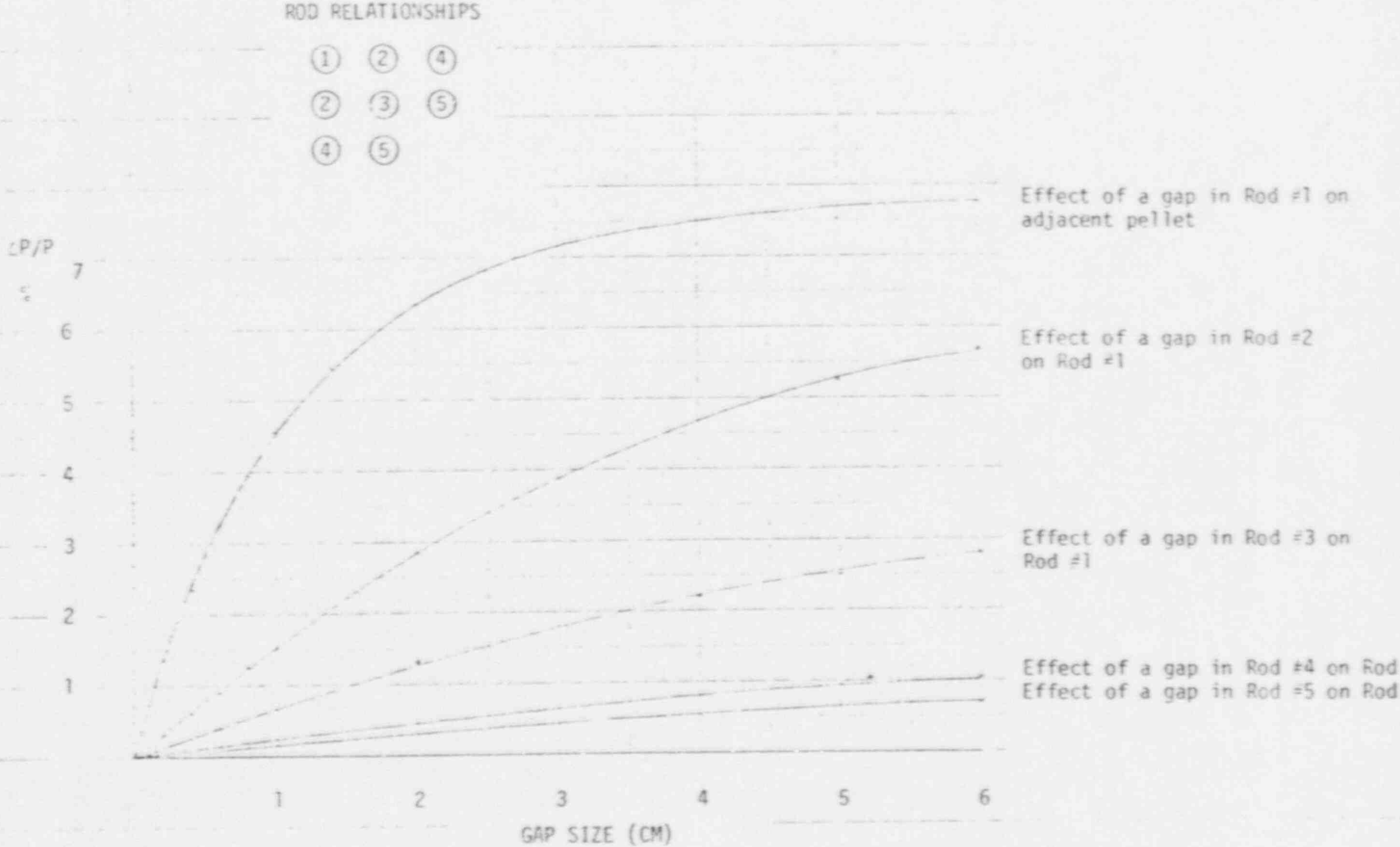
FIGURE 3-4

8x8 POWER SPIKE VERSUS GAP SIZE - COLD ZERO VOID CONDITION



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3) The core power histogram employed was for a 13.4 KW/ft maximum design linear heat operation rate; see Figure 3-5.

RESULTS

Normal Operation

The results from this analysis are shown in Figure 3-6 with initial fuel density as a parameter. The line shown for an initial fuel density of 95% T.D. is considered to be most representative considering current GE data on manufactured fuel pellet densities as a function of axial position, is the required margin which must be maintained during normal operation between the actual peak operating condition and the peak design LHGR; i.e., 13.4 KW/ft. Maintaining this margin will assure, with better than 95% confidence, that no more than one rod will exceed the design peak LHGR due to the random occurrence of power spikes resulting from axial fuel column gaps. Consistent with GE's position on densification, previously discussed in Reference 4 and its supplements, the results of this analysis are considered to be a very conservative representation of the power peaking penalty required to accommodate potential axial fuel column gaps during normal operating conditions in GE BWR's.

Accident Effects

Since the results of the power spiking analysis for normal operation will be utilized to limit bundle power to assure that the random occurrence of power spikes will not result in exceeding the design peak LHGR, it is not believed necessary to separately consider power spikes in the analysis of transients or accidents which have as an initial condition some form of normal operation. The control rod drop accident is unique in the respect that it begins at the cold condition, and is not affected by normal operating power level. Further, the existence of fuel column gaps can result in power

FIGURE 3-5

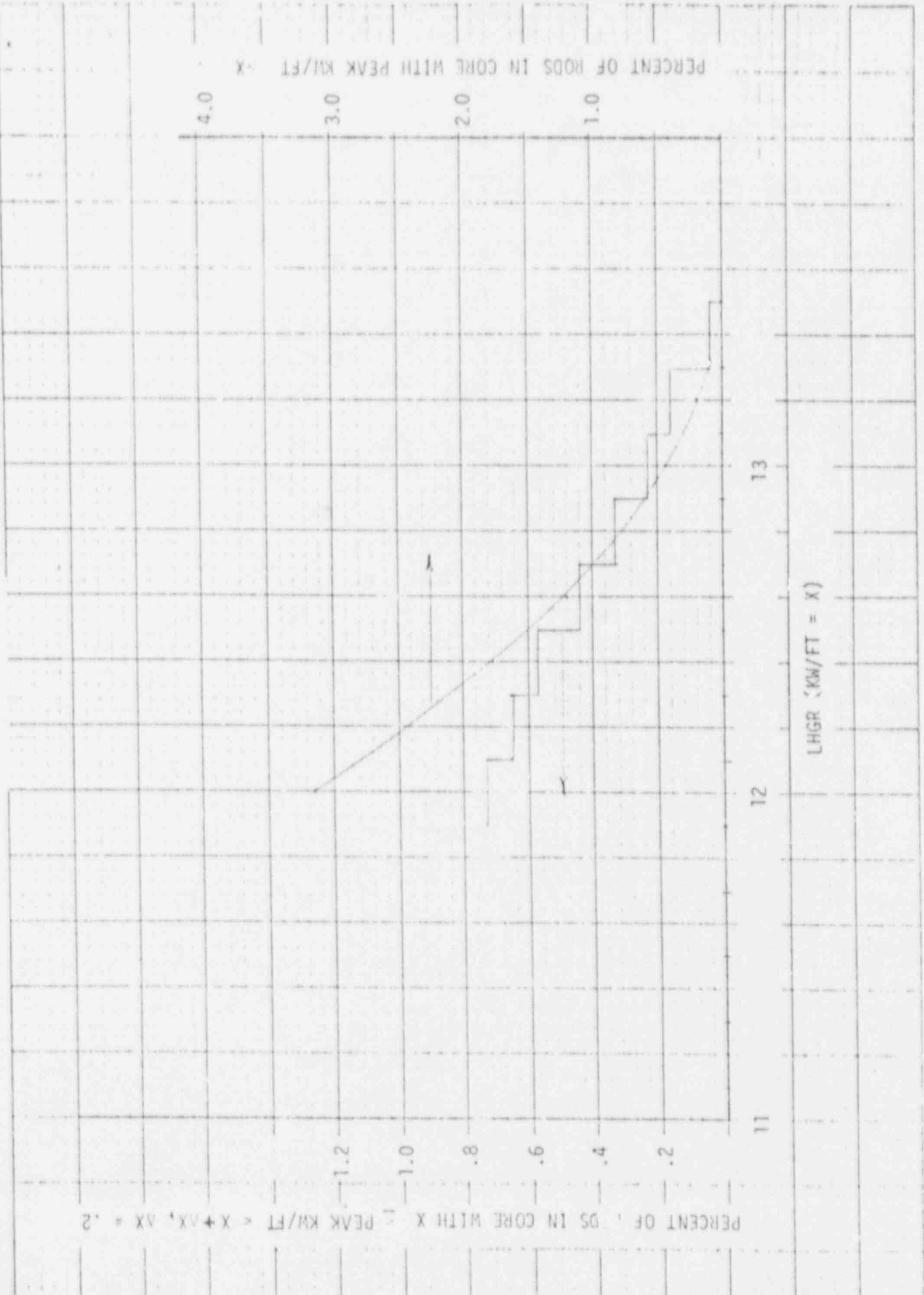

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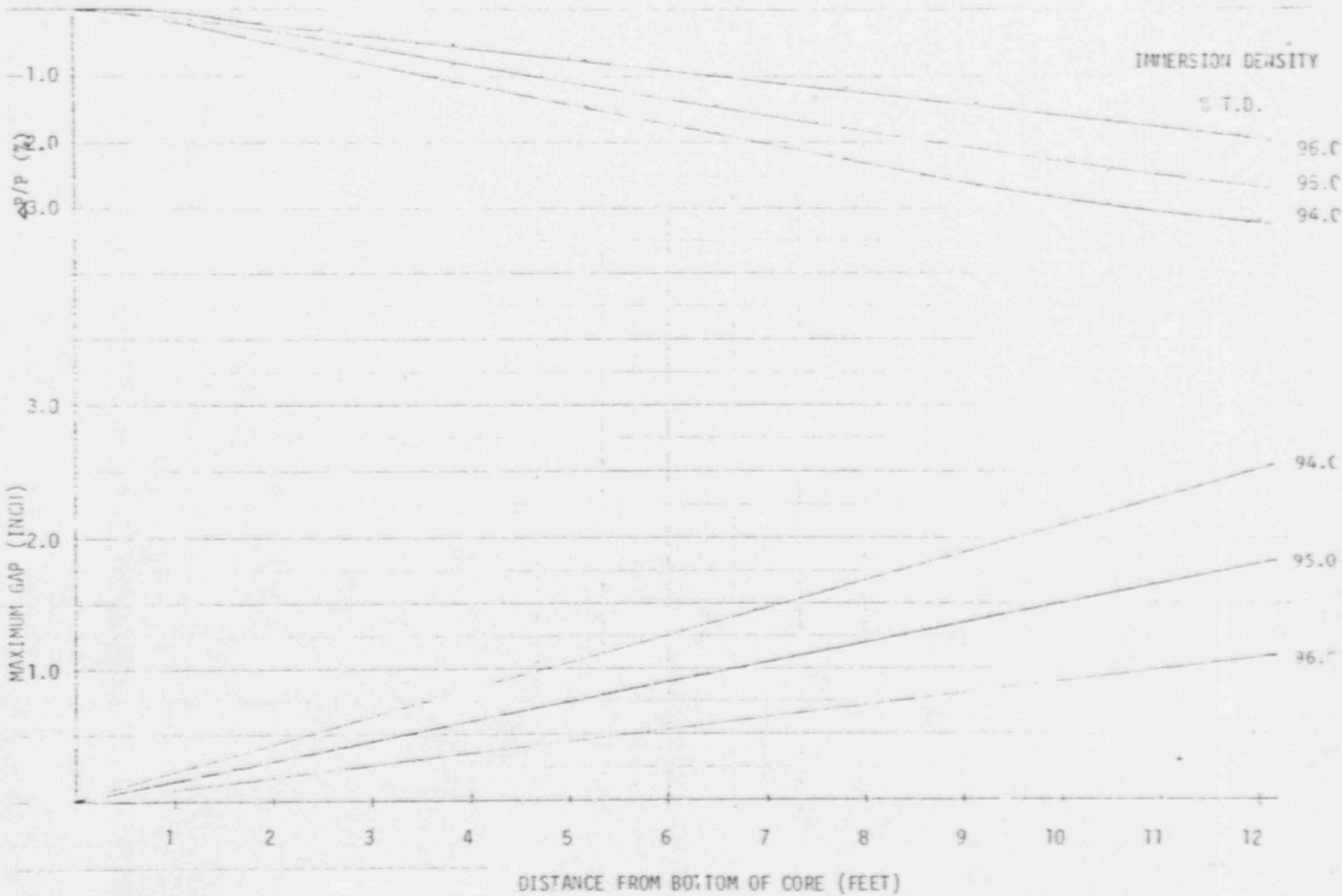
FIGURE 3-6

8x8 POWER SPIKE PENALTY VS AXIAL POSITION - NORMAL OPERATION



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spiking in the cold condition during a control rod drop which should thus be considered in the evaluation of this accident. For this purpose, a separate power spiking anlysis has been performed using the same assumptions as indicated above, but employing a power spike versus gap size calculated to occur in the cold condition with zero voids (Figure 3-4). This analysis was performed for a conservative maximum gap size calculated employing a pellet average immersion density of 94.5% T.D., and a position near the top of the core in order to maximize the power spiking effect. This analysis yielded a 99% probability that any given fuel rod would have a power spike of $\leq$5%.

3.4.4.2 Cladding Creep Collapse

Using the same conservative bases presented in References 7 and 8, the critical pressure ratio; i.e., ratio of collapse pressure to actual coolant pressure, was calculated. Figure 3-7 presents the clad mid-wall temperature versus time for the 8x8 reload fuel. No credit is taken for internal gas pressure due to released fission gas or volatiles. The interual pressure due to helium backfill at 1 atmosphere during fabrication is considered. The fuel chacteristics for creep collapse calculations are as follows:

Clad O.D., in.	0.493
Clad Thickness, in.	0.034 ± 0.003
Peak LHGR, KW/ft	13.4
Fast Flux >1 mev, n/cm^2-sec	4.37×10^{13}

Figure 3-8 gives the calculated critical pressure ratio. As evidenced by the curve, the calculated critical pressure ratio is always >1.0.

3.4.4.3 Increased Linear Heat Generation Rate

The following expression was employed to calculate the decrease in fuel column length due to densification in calculation of a penalty in linear heat generation rate:

FIGURE 3-7

CLADDING AVERAGE TEMPERATURE AT A FUEL COLUMN AXIAL GAP

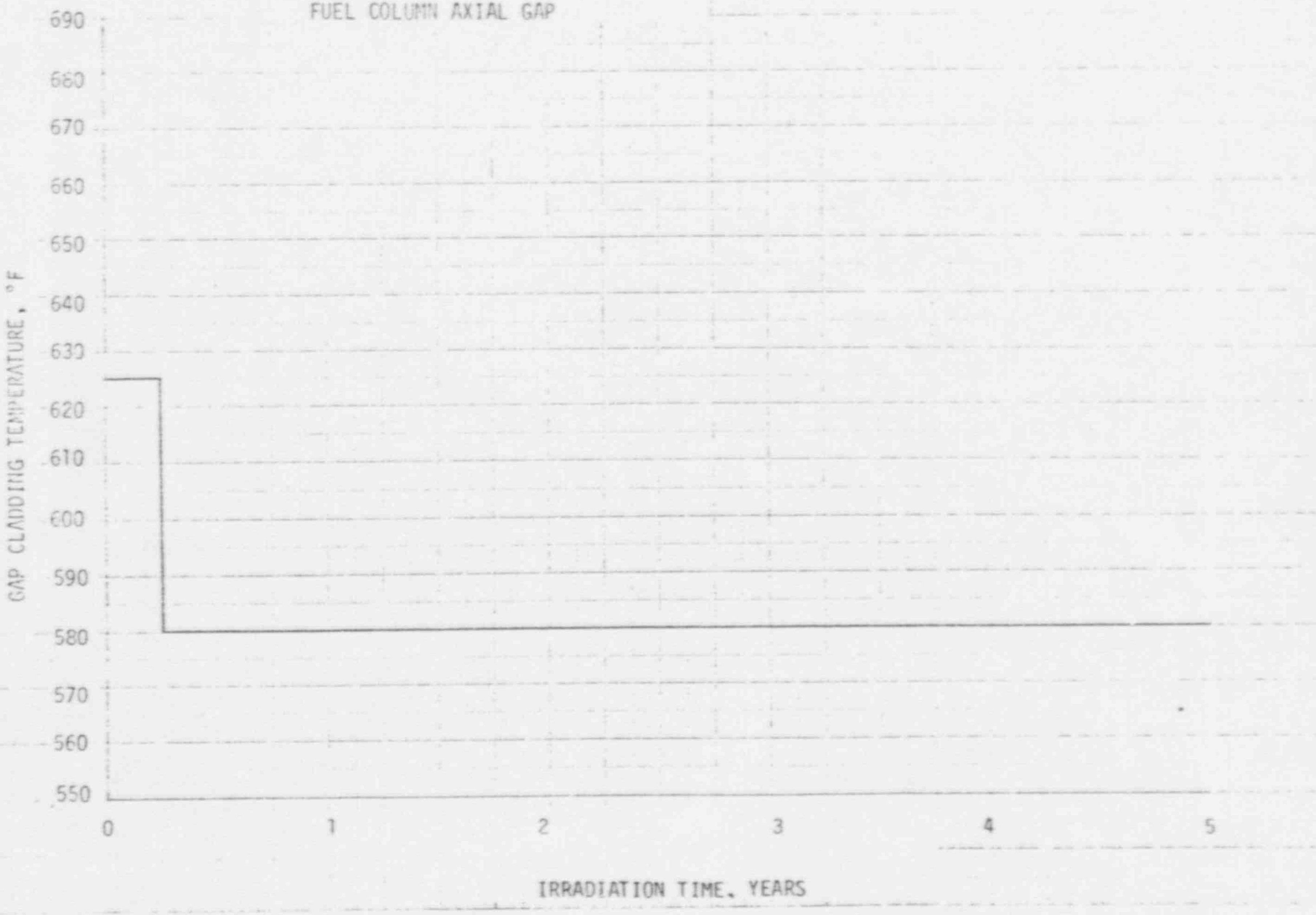
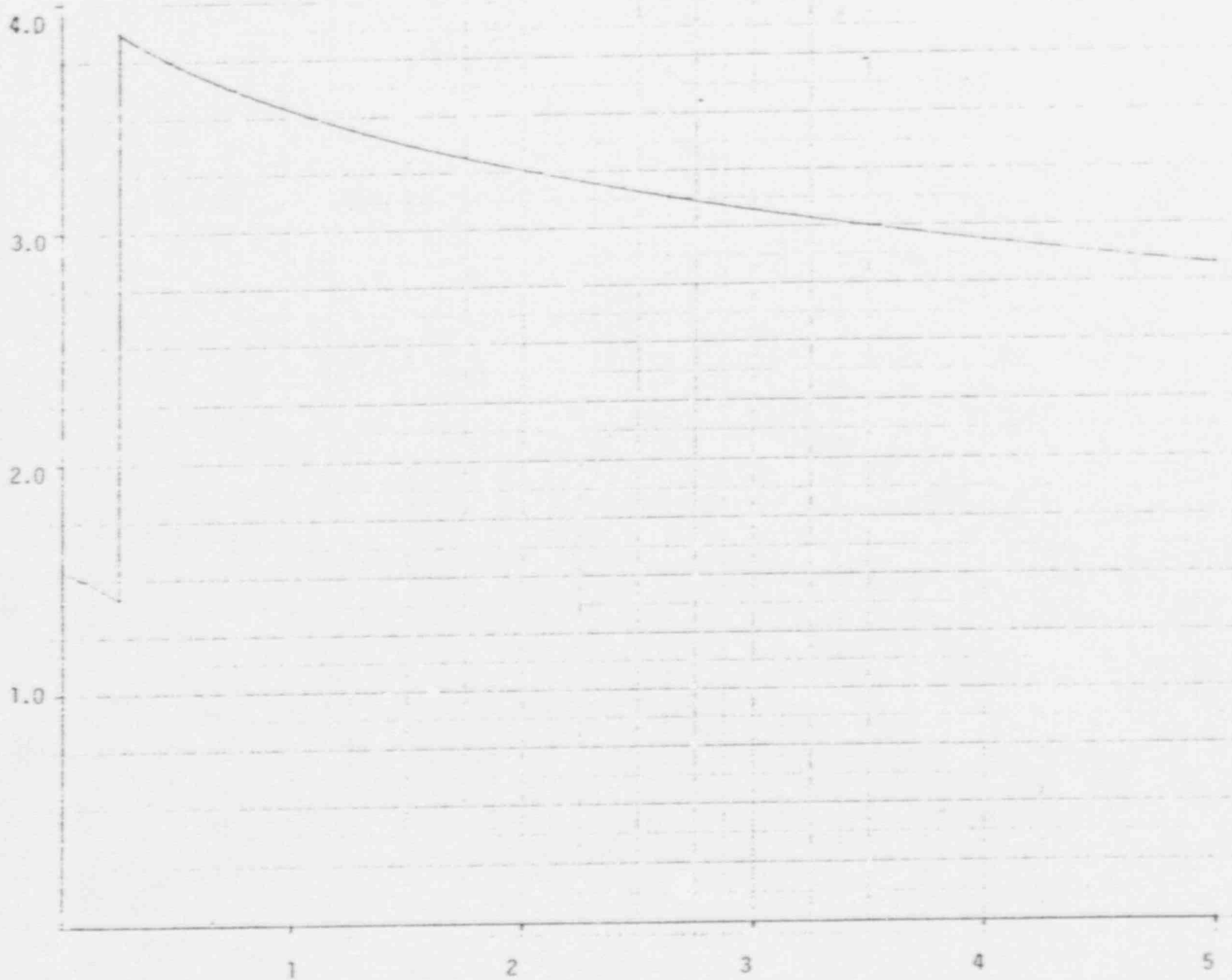


FIGURE 3-8

CLAD CRITICAL COLLAPSE PRESSURE RATIO VERSUS TIME
8x8 RELOAD FUEL



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$$\Delta L = \frac{0.965 - \rho_i}{2} L$$

where ΔL = decrease in fuel column length

L = fuel column length

ρ_i = mean value of measured initial pellet density (immersion - .5%)

The length reduction due to densification as calculated by the above equation requires knowledge of the mean immersion density (ρ_i) obtained from the QC data. A correction 0.5% T.D. is applied to convert the immersion density to a geometric density. The mean pellet immersion density for Monticello 8x8 fuel is 95.44% T.D. This results in

$$\frac{\Delta L}{L} = \frac{0.965 - (.9544 - 0.005)}{2} = \frac{0.0156}{2} \approx 0.008$$

or $\frac{\Delta L}{L} = 0.8\%$

Due to thermal expansion, an 8x8 pellet normally expands in going from the cold to hot condition, an amount equal to 1.2% for a pellet at 13.4 KW/ft. This increase in length from the cold to hot condition is not taken credit for in either design calculations or in the process of core performance analysis during reactor operation. The cold pellet length is assumed for these conditions.

Therefore, the decrease in pellet length due to densification is more than offset by pellet axial thermal expansion.

REFERENCES - SECTION 3

1. WAPD-TM-263, "Effects of High Burnup on Zircaloy-Clad, Bulk UO_2 Plate Fuel Element Samples," September 1962.

2. WAPD-TM-629, "Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished End UO_2 Pellets," July 1967.

3. Williamson, H. H., and Ditmore, D. C., "Experience with BWR Fuel Through September 1972," May 1972 (NFDO-10505).

4. Ditmore, D. C., and Elkins, R. B., "Densification Considerations in BWR Fuel Design and Performance," December 1972 (NEDM-10735).

5. Miner, M. A., "Cumulative Damage in Fatigue," Journal of Applied Mechanics, 12, Transactions of the ASME, 67, 1945.

6. O'Donnel, W. J., and Langer, B. F., "Fatigue Design Basis for Ziraloy Components," Nuclear Science and Engineering, 20, 1964.

7. Hinds, J. A. (General Electric) letter to V. A. Moore (USAEC), "Plant Evaluations With GEGAP-III," December 12, 1973.

8. NEDM-10735 Supplement 6, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," August 1973.

4. THERMAL-HYDRAULIC CHARACTERISTICS

4.1 FUEL ASSEMBLY HYDRAULIC ANALYSIS

4.1.1 Core Pressure Drop, Hydraulic Loads, and Correlations

The flow distribution to the fuel assemblies is calculated on the assumption that the pressure drop across all fuel assemblies is the same. This assumption has been confirmed by measurements of the flow distribution in modern boiling water reactor as reported in References 1 and 2. The components of bundle pressure drop considered are friction, local, elevation, and acceleration. 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.

4.1.1.1 Friction Pressure Drop

Friction pressure drop is calculated using the model relation

$$\Delta P_f = \frac{w^2}{2g\rho} \quad \frac{fL}{D_H A_{ch}^2} \quad \phi_{TPF} \; ;$$

where

ΔP_f = friction pressure drop, psi,
w = mass flow rate,
g = acceleration of gravity,
ρ = water density,
D_H = channel hydraulic diameter,
A_{ch} = channel flow area,
L = length,
f = friction factor, and
ϕ_{TPF} = two phase friction multiplier.

WIDE-WIDE CORNER

1	2	3	4	5	6	7	8
9	10	11	12	13	14	15	16
17	18	19	20	21	22	23	24
25	26	27	28	29	30	31	32
33	34	35	36	37	38	39	40
41	42	43	44	45	46	47	48
49	50	51	52	53	54	55	56
57	58	59	60	61	62	63	64

Figure 6-24. 8x8 Reload Fuel Rod Identification

Conformance With Interim Acceptance Criteria. In the analyses discussed above there have been no deviations from the evaluation model described in Appendix A, Part 2 of the AEC Interim Policy Statement.

Effects of ECCS Operation on the Core. The mechanical effects of ECCS operation on the core, reactor coolant system and ECCS are those associated with the thermal effect of injecting water into these systems which is cooler than these systems and components. These thermal stresses have been considered in the design of the core, reactor coolant system and ECCS.

There are no nuclear effects resulting from ECCS operation, since all control rods are inserted and the reactor remains subcritical during the injection of the cooler ECCS water.

There are no chemical additives in the ECCS water and therefore no chemical effects on the core, reactor coolant system or ECCS.

Lag Times. The system time delays assumed in the LOCA accident are as follows:

System	Maximum Allowable Time From Signal Receipt Until the Pumps Have Reached Rated Speed (sec)	Maximum Time Delay After Receipt of Signal Until All Valve Motion is Complete (sec)
HPCI	30	30
CS	30	30
LPCI	43	43
ADS	-	120

6.2.2.4.4 Densification Effects

Figure 6-24a provides plots of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and peak cladding temperature (PCT) versus exposure, for Monticello 8x8 reload fuel.

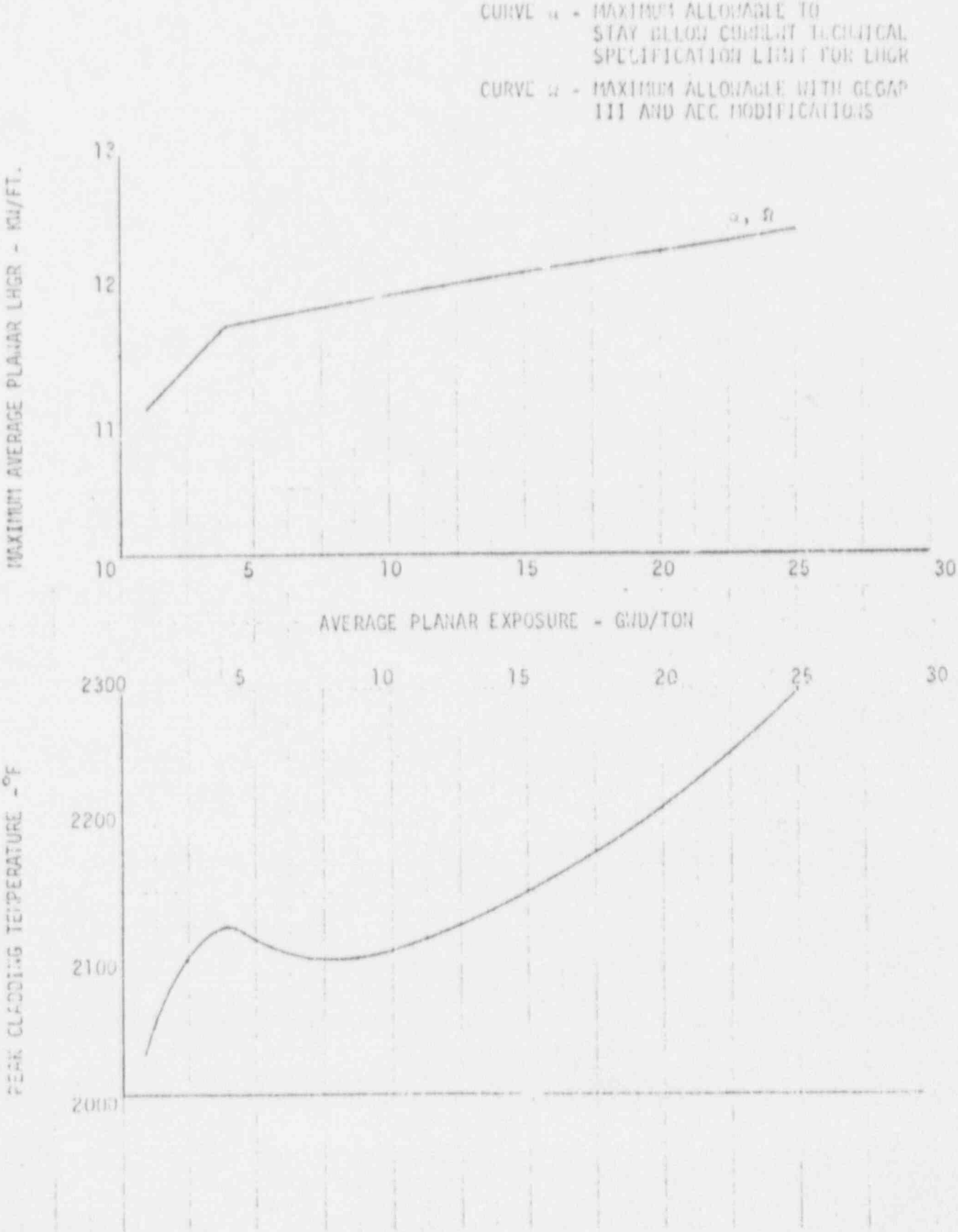
The LOCA analyses were performed using the approved Interim Acceptance Criteria Model with gap conductance values as calculated with the new GEGAP III model with AEC modifications.(17)

Figure 6-24a

MONTICELLO RELOAD 2 (8x8)

GEGAP III

MAPLHGR AND PCT VS EXPOSURE



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11. Monticello Nuclear Generating Plant, First Reload License Submittal, February 1973.

12. Millstone Unit 1, FSAR Amendment 14, Dkt. 50-245.

13. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel, Supplement 6, 7, and 8, Composite," August 1973 (NEDM-10735).

14. "Results of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics," February, 1973.

15. "Monticello - Safety Valve Setpoint Increase Analysis," Change request dated September 3, 1973.

16. "Monticello - Cycle 2 Scram Reactivity Considerations, Analyses and Modifications," October 1973.

17. Hinds, J. A. (General Electric) letter to V. A. Moore (USAEC), "Plant Evaluations With GEGAP III," December 12, 1973.

7. TECHNICAL SPECIFICATIONS

There are four areas of the Technical Specifications affected by the preceeding information. Changes made necessary by the reactor pressure relief system modifications discussed in Section 6.2.4 will be outlined in the forthcoming submittal on that subject. The formal request for Technical Specification changes will be a separate, subsequent submittal. Specifications affected by this submittal include the following:

Section 2 - The heat flux of a 7x7 fuel assembly operating up to 17.5 kw/ft results in a 3.08 total peaking factor. Changes should reflect the use of 8x8 fuel operating up to 13.4 kw/ft resulting in a 3.04 total peaking factor.

Section 3.3.C - The transient analysis (Section 6.2.4) was done based on a control rod scram time to 90% insertion of 3.5 seconds rather that 5.0 seconds as presently allowed. The Specification will be changed accordingly.

Section 3.5.K - The 8x8, R-2 fuel will have unique properties for consideration of postulated fuel densification phenomena. Since the AEC staff model requires the use of measured pellet theoretical density, this information can not be finalized until the fuel is fabricated.

Section 5.2 - The facility description states that fuel assemblies have 49 fuel rods each. This must be changed to allow the use of 8x8, 63 fuel rod assemblies.