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March 1974

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Oconee 3

FUEL DENSIFICATION REPORT

(Nonproprietary Version of BAW-1399)

Babcock & Wilcox

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ABSTRACT

In June of 1973, BAW-10055, Rev. 1, was filed with the AEC in accordance with the guidelines set forth in the AEC report, "Technical Report on Densification of Light Water Reactor Fuels," dated November 14, 1972. This revision incorporated the answers to additional questions from the AEC Staff concerning generic items on fuel densification.

In October of 1973, B&W filed an additional report, BAW-10079, "Operational Parameters for B&W Rodded Plants," which sets forth the core operating parameters for B&W rodded plants. This report established the loss-of-coolant accident (LOCA) basis for determining the maximum allowable heat rate and outlined the analysis used to determine plant operating restrictions owing to the postulated effects of fuel densification. Questions relating to individual plants (as-built data, etc.) are answered in individual reports which are filed for each plant.

This report, along with the appendix, presents an analysis of the effects of fuel densification on the fuel for Oconee 3 and supports the safe operation of that unit at the rated power level of 2568 MWt.

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1. INTRODUCTION

This report documents the effects of postulated fuel densification for the Oconee 3 core as calculated in accordance with guidelines set forth in the AEC report of November 14, 1972. The application of these guidelines to the results presented in this report is discussed fully in B&W's proprietary topical report BAW-10055, Rev. 1. "Fuel Densification Report." Further considerations as presented in BAW-10079, "Operational Parameters for B&W Rodded Plants," were also taken into account.

The analysis of Oconee 3 is limited to an examination of the first fuel cycle. Babcock & Wilcox now has operating plant data on the Oconee 1 fuel, and there are no signs of fuel densification after 75 EFPD. It is expected that data from other pressurized water reactors (PWRs) now operating with prepressurized fuel will allow relaxation of the current guidelines. Before the completion of the first cycle, a supplementary report will be filed for Oconee 3 to cover three full cycles of operation at 2568 MWt.

2. CONCLUSIONS

Based on the analysis performed for Oconee 3, which utilized the methods given in BAW-10055, Rev. 1, and BAW-10079, the following conclusions are made even if the fuel pellets are assumed to densify to 96.5% of their theoretical density:

1. The cladding will not collapse because all B&W fuel rods are pressurized.

2. The mechanical performance of B&W fuel rods will not be impaired.

3. The interim acceptance criteria for the emergency core cooling system (ECCS) will not be violated.

4. The reactor can be safely operated at the rated power level of 2568 MWt with the reactor protection system (RPS) setpoints outlined herein. These modifications ensure that the thermal design criteria are not exceeded.

5 The modifications to the RPS are a reduction in the overpower trip setpoint, from 114 to 112% of rated power, and a minor reduction in allowable imbalance limits as shown in Figure 3.3-3.

3. RESULTS

This section of the report covers four main topics: thermal analysis, nuclear analysis, safety analysis, and mechanical analysis. The thermal analysis section considers protection of the fuel melt and DNBR criteria. The nuclear analysis section considers thermal design criteria, imbalance trip limits, and core operational limits. The safety analysis section reanalyzes all postulated accidents analyzed in the Oconee 3 FSAR assuming that densification occurs. The mechanical analysis section contains the input summary and results for cladding creep and collapse, cladding stresses, and fuel pellet irradiation swelling. Since complete as-built data were not available for this analysis, the most conservative values from the specification are used in each analysis.

3.1. Power Spike Model

The AEC guidelines outlined in "Technical Report on Densification of Light Water Reactor Fuels," November 14, 1972, have been used to determine the maximum axial gap as a function of core height. The probability values (F_K) given in the same report (Table 4.2.A, column 4) have been used in calculating the power spike factor. This factor, as calculated in section 2 of BAW-10055, Rev. 1, is applicable to individual reactors. The maximum gap size versus axial position is shown in Figure 3.1-1, and the power spike factor versus axial position is shown in Figure 3.1-2. These figures also show the initial and final theoretical densities (TDI, TDF) used in the calculations. These data form the basis for the analyses in this report.

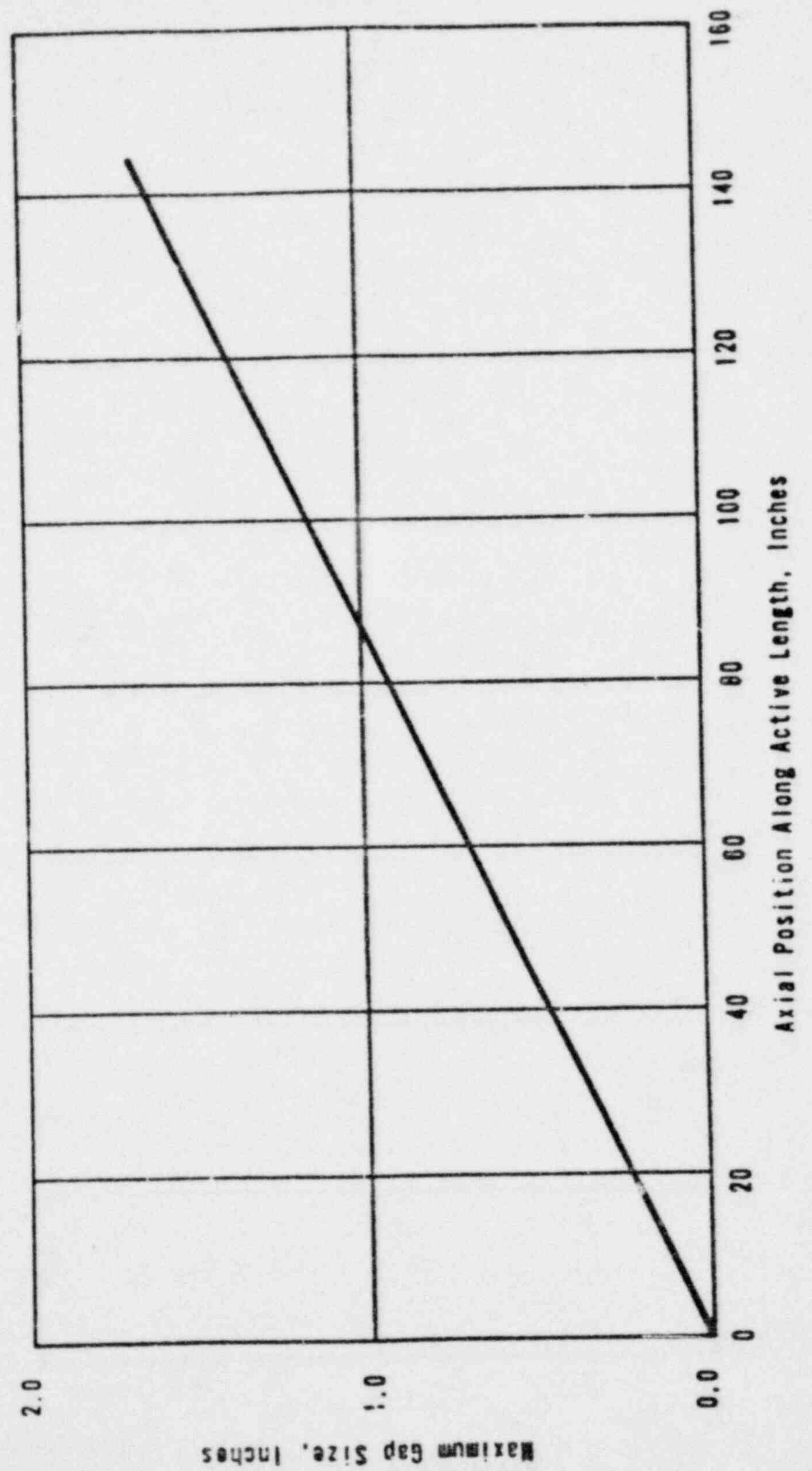

Figure 3.1-1. Maximum Gap Size Vs Axial Position

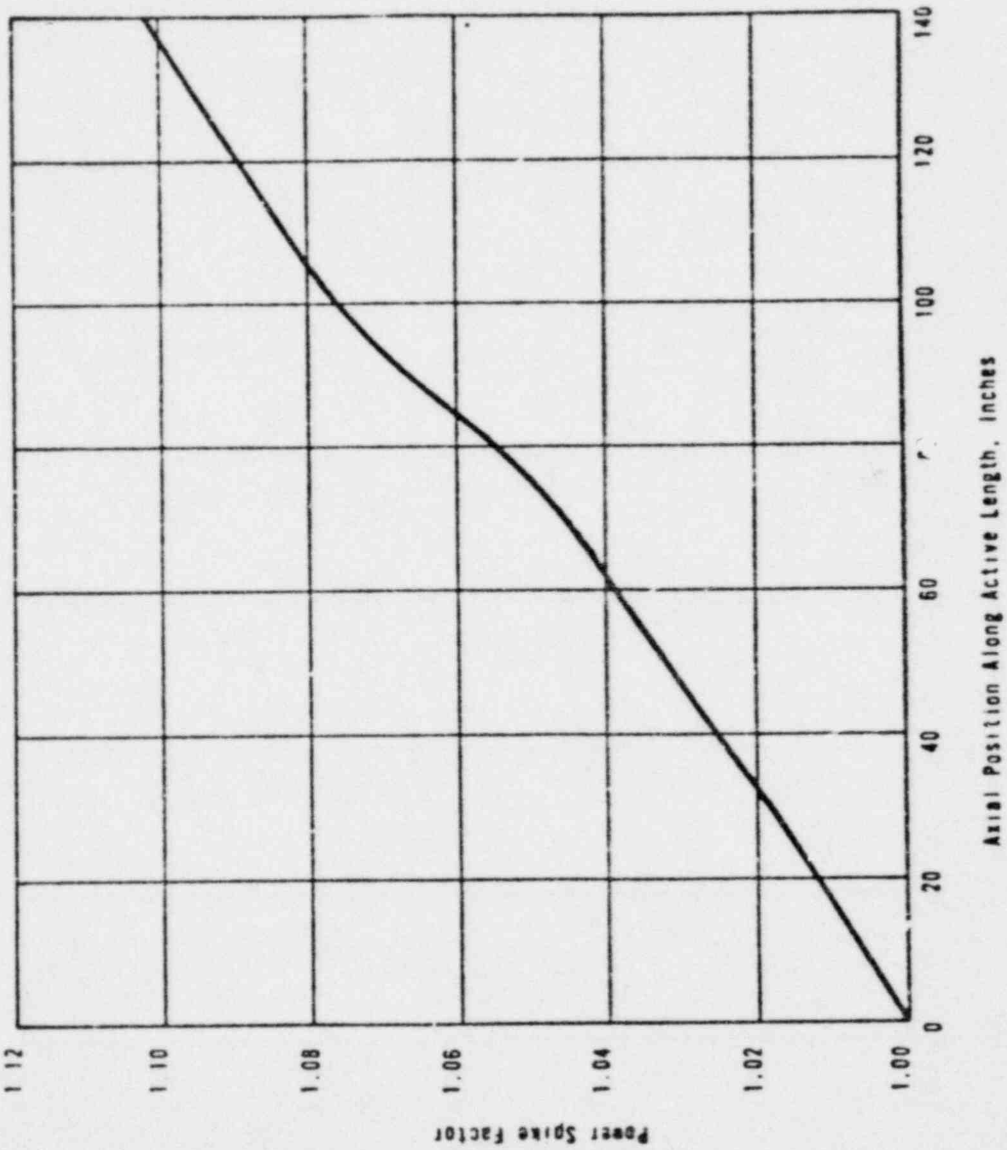

Figure 3.1-2. Power Spike Factor Vs Axial Position

3.2. Thermal Analysis

3.2.1. Fuel Temperature Analysis

Utilizing the analysis established in BAW-10055, Rev. 1 plus modifications as requested by DOL, a fuel-to-cladding cold diametral gap of 12.45 mils after densification was analyzed. The results of this analysis are presented in Tables 3.2-1 and 3.2-2 and in Figures 3.2-1 through 3.2-3.

The modifications are as follows:

1. TAFY* thermal code
   a. No fuel restructuring.
   b. A 25% reduction in gap conductance.
2. Inputs to TAFY
   a. Most conservative specification data used for fuel density and diameter and for cladding ID (Table A-1).

3.2.2. DNBR Analysis

The thermal effects due to densification can be divided into two categories: (1) the result of the reduced stack height and (2) the combined result of the reduced stack height with the power spike superimposed. Thermal effects are then imposed on calculations of the minimum departure from nucleate boiling ratio (DNBR) used to set thermal design limits.

The reduced active length was calculated to be 139.94 inches, which represents a reduction of 4.06 inches from the nominal active length of 144.0 inches. The most conservative specification information given in the appendix was used in calculating this densified active length.

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*See note at end of Table 3.2-3.

The axial flux shape that gave the maximum change in DNBR from the original design value was an outlet peak with a core offset of +11.8%. The spike magnitude and the maximum gap size used in the analysis are 1.100 and 1.65 inches, respectively. The results of the two effects are summarized in Table 3.2-3 in terms of percentage change in minimum hot channel DNBR and peaking margin.

3.2.3. Summary

This analysis assumes that densification and associated phenomena will affect the hot channel, which has the most limiting thermal-hydraulic characteristics in the core. Both the fuel temperature analysis and the DNBR analysis were conducted independently with the respective most conservative specification values. In addition, the power spike is assumed to be located at the hot channel position that minimized DNBR. The resultant loss in DNBR of 4.4% results in a DNBR of 1.48 at 114% of 2568 MWt. This is equivalent to a 2.1% loss in allowable power peaking. The inclusion of control rod insertion limits as well as the reduction of the overpower from 114% to 112% of 2568 MWt compensates for this loss. The plant can then function at the full core rated power level without violating the design criteria for DNBR and/or centerline fuel melting. The allowable power shapes and the new offset limits are discussed in section 3.3.

Table 3.2-1. Fuel Temperatures at Low Power Density

Density, % TD	Cold gap, mils	kW/ft	Gap coeff, Btu/h-ft^2-°F	Surface fuel temp, F	Average fuel temp, F	Maximum fuel temp, F
96.5	12.45	6.0	680	977	1337	1733

Table 3.2-2. Fuel Temperatures at High Power Density

Density, % TD	Cold gap, mils	kW/ft	Gap coeff, Btu/h-ft^2-°F	Surface fuel temp, F	Average fuel temp, F	Maximum fuel temp, F
96.5	12.45	18.9	965	1483	3126	4849

Table 3.2-3. Effects of Fuel Densification on DNBR and Power Margin at 114% of 2568 MWt

	Densified active length			Densified active length and power spike		
Axial power shape	DNBR (W-3)	%Δ DNB	%Δ Margin	DNBR (W-3)	%Δ DNB	%Δ Margin
Outlet peak with +11.8% core offset	1.50	-2.8	-1.3	1.48	-4.4	-2.1

NOTE

B&W topical report BAW-10044 describes the TAFY computer program. The code has been used as described in the analysis of fuel densification except for the following:

> The option in the code for no restructuring of fuel has been used in the analysis presented here in accordance with DOL's interim evaluation of TAFY.

Figure 3.2-1. Maximum Fuel Temperature Vs Linear Heat Rate

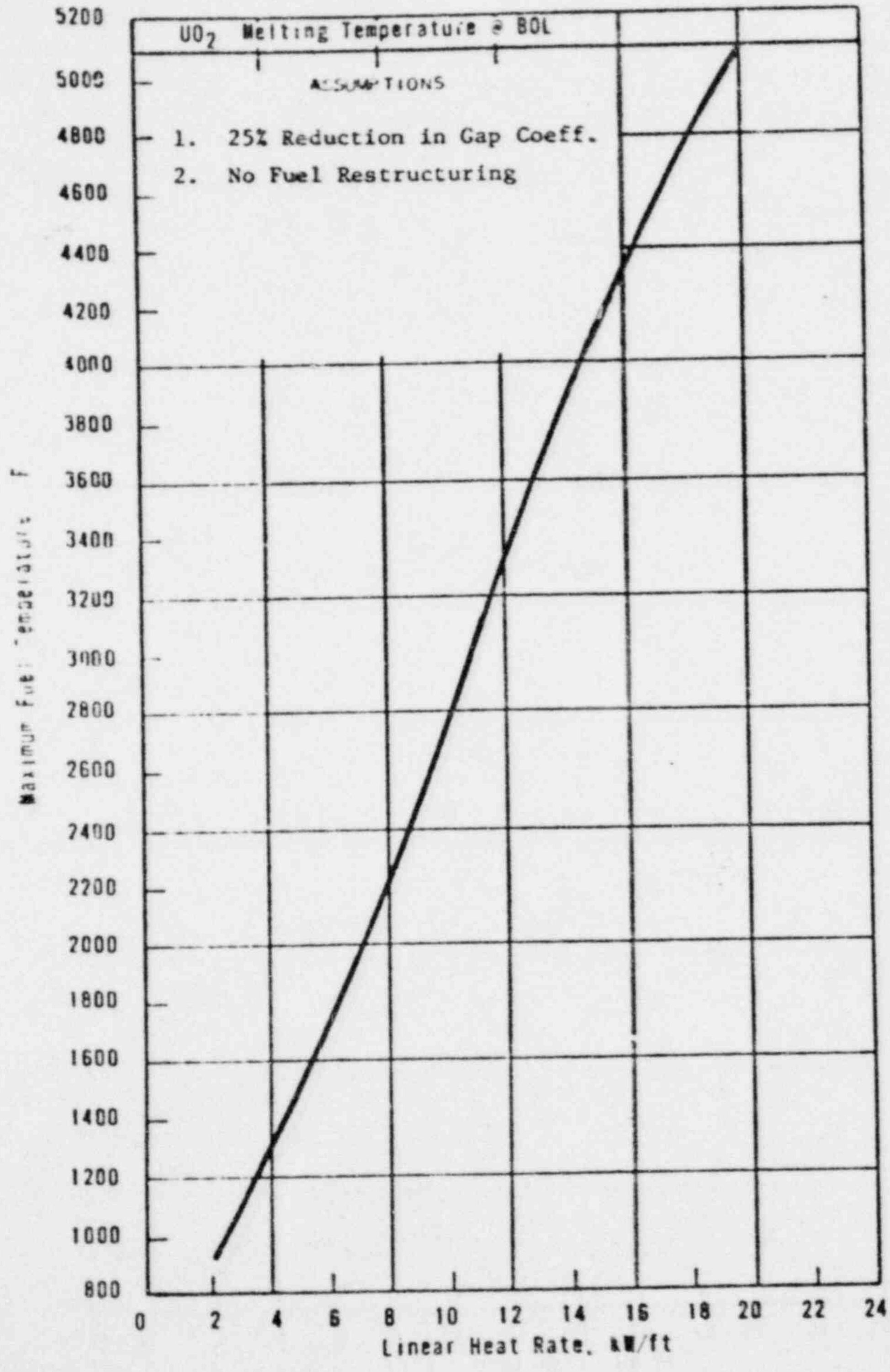


Figure 3.2-2. Average Fuel Temperature Vs Linear Heat Rate

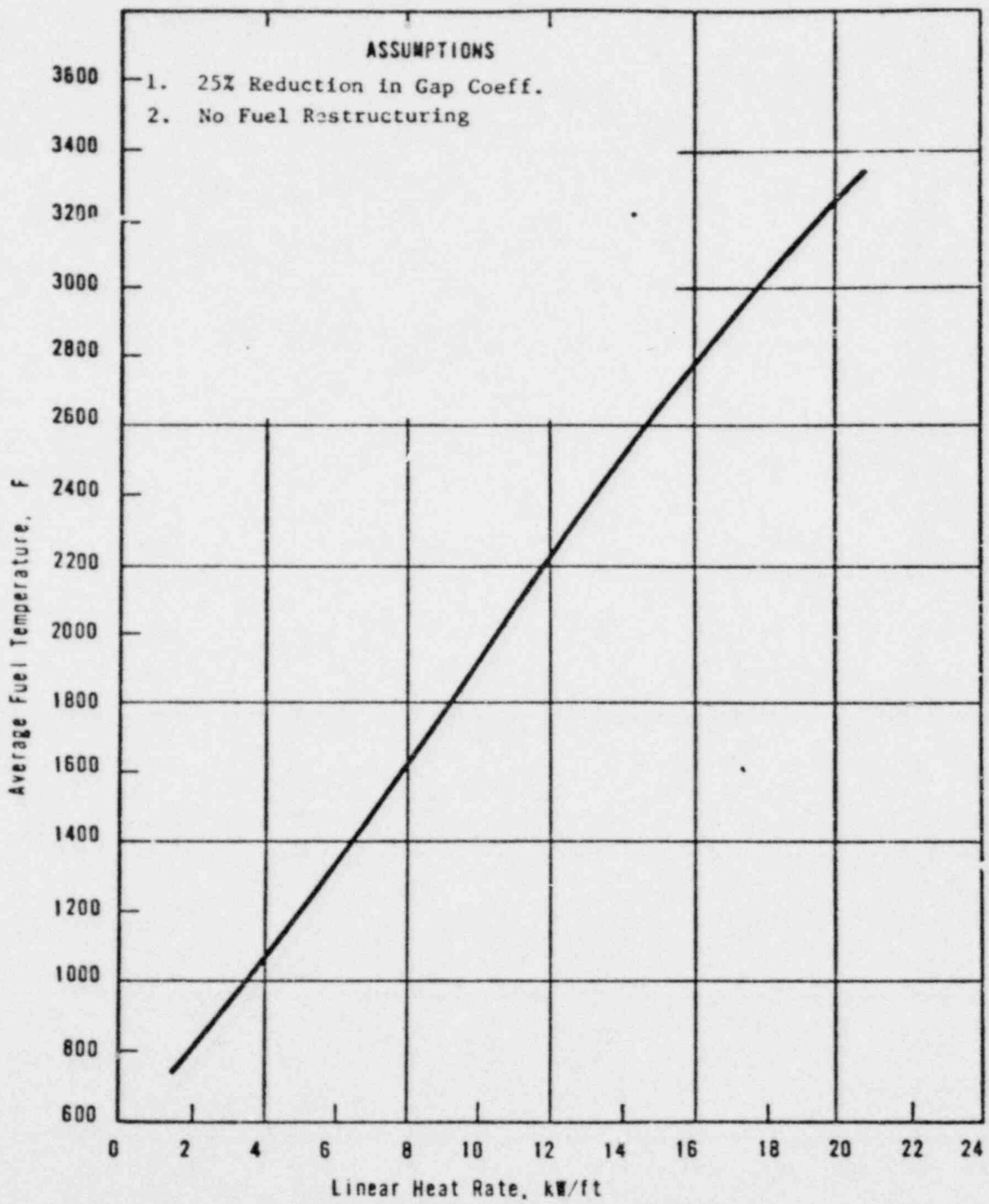
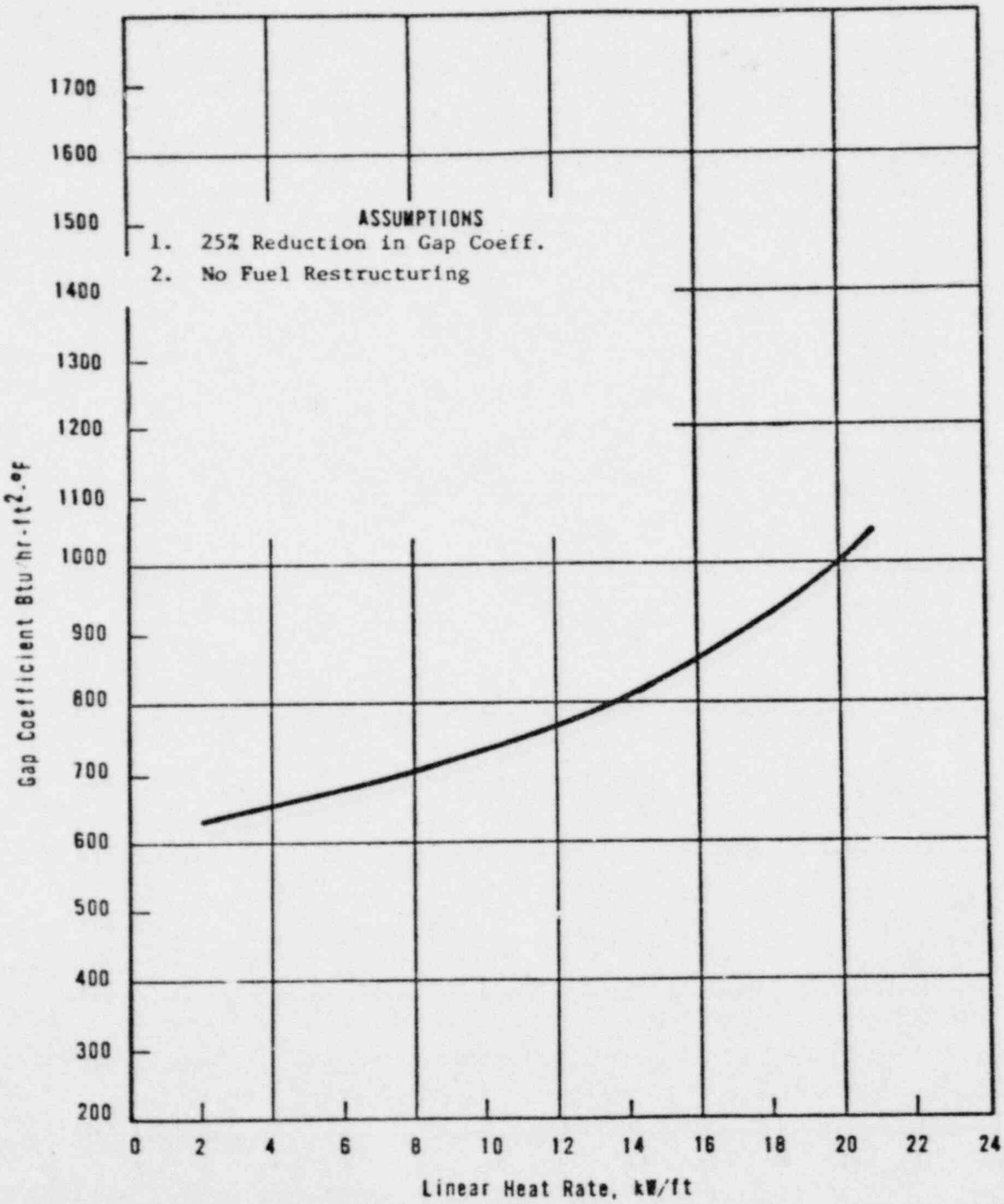


Figure 3.2-3. Gap Coefficient Vs Linear Heat Rate



3.3. Nuclear Analysis

3.3.1. Reactor Protection System

The safe operation of a reactor core requires an extensive analysis of power distributions resulting from the various modes of plant operation. The primary considerations and results of this analysis are as follows:

1. Assurance that thermal criteria are not exceeded; i.e., specified minimum DNBRs and centerline fuel temperatures may not be violated.
2. Definition of imbalance limits to prevent adverse power peaks that would exceed the foregoing criteria.
3. Definition of core operational limits and recommended operating procedures to prevent unnecessary reactor trips.

The complete maneuvering study entails a combined nuclear-thermal analysis of the power distributions. This section describes the methods and criteria used in developing the RPS setpoints and in modifying the setpoints required to account for postulated densification effects.

3.3.2. Analysis of Power Distributions Before Densification

The three-dimensional PDQ07 code with thermal feedback effects is used to analyze power distributions. This analysis determines power distributions for all modes of reactor operation except accidents and other rapid transients. The design power transient (100-30% power and return to 100% at peak xenon) is analyzed throughout core life. The fuel cycle and transient analyses determine power distributions for normal equilibrium and transient conditions, respectively. The extremes of core operation, such as control rod bank insertion beyond normal limits and maloperation of axial power shaping rods, are also examined. The extreme control rod bank conditions define the limits for the imbalance protection system.

3.3.2.1. Correlation of Power Peaks to Thermal Design Criteria

The power peaks from PDQ cases are corrected for calculational uncertainty and are analyzed to determine the margin to the thermal criteria: centerline fuel melt and departure from nucleate boiling (DNB). The margin to centerline fuel melting is defined as

$$\text{Fuel melt margin} = \left(\frac{\text{Max allowable peak}}{\text{Max calculated peak}} - 1\right) 100\%.$$

The maximum allowable peak is defined as the pointwise power that yields centerline fuel melting:

$$\text{Max allowable peak} = \frac{22.2\ \text{kW/ft}}{5.66\ \text{kW/ft} \times 1.014 \times \text{FOP}}$$

where
22.2 kW/ft = fuel melt limit,
5.66 kW/ft = average heat rate at 2568 MWt,
1.014 = hot channel factor,
FOP = fraction of power.

The maximum calculated peak is the largest total peak from the PDQ power maps increased by a factor of 1.075 to account for calculational uncertainty.

The determination of DNB margin requires a more complex analysis. DNBR is a function of peak location, magnitude of the power peak component parts (radial and axial), and other core parameters. To arrive at true DNB conditions, each power distribution is analyzed explicitly. From the PDQ power distribution, the maximum calculated total peak is obtained and adjusted for uncertainty. The DNB margin is then defined as

$$\text{DNB margin} = \left(\frac{\text{Allowable total peak}}{\text{Max calculated total peak}} - 1\right) 100\%.$$

The basis for the allowable total peak is the reference design DNBR at design conditions, or a 1.30 DNBR associated with the protection system envelope, or a quality limit based on model applicability, whichever is most limiting.

3.3.2.2. Offset-Margin Relationship

Core offset, a measure of the axial power imbalance, is defined as the fraction of total core power in the top half of the core minus the fraction of total core power in the bottom half of the core:

$$\text{Offset} = \frac{\text{Power (top)} - \text{power (bottom)}}{\text{Power (top)} + \text{power (bottom)}}.$$

The relationship between hot channel power peaks (i.e., thermal margins) and core offset defines the protection system setpoints. Power imbalance is the primary signal to the protection system for flux shape protection. The maneuvering analysis defines the relationship between core imbalance and thermal margin.

Limiting offsets are determined to prevent the violation of thermal criteria for all operating conditions and power levels. To yield the imbalance trip envelope, the limiting offset values are corrected for potential instrumentation errors, imbalance detection bias, and calibration. The imbalance trip envelope defines the range of allowable operational imbalance and ensures that 1.3 DNBR and/or the central fuel melting limit will not be exceeded. Figure 3.3-1 presents the trip setpoints based on these criteria. The overpower trip setpoint shown in Figure 3.3-1 is controlling for overpower transients, whereas, the solid horizontal line is the trip for loss of flow transients.

3.3.3. Analysis of Power Distributions With Densification Effects

3.3.3.1. RPS Considerations

Provision for possible fuel densification requires modification of the imbalance trip system for two reasons: (1) the fuel melt (kW/ft) criterion change, and (2) an additional power spike is included in the reactor power distributions. Since the power spike factor is a function of axial position, the appropriate power spike factor is used to increase each PDQ peak to account for potential densification.

The modified offset limits with fuel densification effects included are presented in Figure 3.3-2 and are compared with the previous offset limits. The primary differences between the two sets of calculated limits are as follows:

1. The DNBR loss of -4.4% results in a peaking margin loss of -2.1%.

2. The central fuel melting limit changes from 22.2 kW/ft before densification to 20.15 kW/ft.

3. A 4.1-inch decrease in fuel column length increases the nominal heat rate at 2568 MWt from 5.66 kW/ft before densification to 5.82 kW/ft after densification.

4. The local power spike factor is applied to the calculated power distributions.

5. The overpower limit in the imbalance protection system is redefined as 112% of 2568 MWt. The effect of the reduced overpower limit is one-to-one for local heat rate and approximately two-to-one for DNBR.

The trip setpoints are obtained from the calculated offset limits by adjusting for potential electronic errors and offset measurement bias by the out-of-core detectors. The error-adjusted limits for densified fuel are shown in Figure 3.3-3. The imbalance trip points and overpower trip provide operating flexibility with assurance that thermal criteria are not exceeded. Furthermore, potential relaxation of these limits may be realized as B&W obtains operational data and experience with Oconee 1 and 2.

3.3.3.2. ECCS Considerations

ECCS calculations have resulted in an axial-dependent kW/ft limit as shown in Figure 3.3-4. (See section 3.4.2.2 for further information.)

The maximum operating heat rates are maintained lower than this limit by imposing restrictions on certain core operating parameters. The maximum allowable heat rate and the maximum expected heat rate for Oconee 3 are compared in Figure 3.3-4.

The derivation of the operating restrictions is fully described in BAW-10079, which includes consideration of the following operating parameters:

1. Fuel depletion.
2. Control rod position.
3. Axial power imbalance.
4. Transient xenon.
5. Quadrant power tilt.

Appropriate controls will be provided to ensure that the LOCA heat rate limits are not exceeded during plant operation.

3.3.4. Summary

Fuel densification and associated design limit changes have required modifications to the technical specifications. The power peaking margin loss of 2.1% from the DNB analysis, the lower fuel melting limits, and the additional power spike factor have been compensated by a 2% reduction in design overpower and by more stringent offset limits. The revised technical specifications allow operation at 100% power with assurance that thermal criteria, with all densification effects included, are not exceeded. The modifications are summarized and compared with the previous system in Table 3.3-1.

Table 3.3-1. Modifications to Reactor Protection System Setpoints and Design Parameters

A. Imbalance System

Parameter	Previous system	Modified system (densification)
1. Fuel melt limit, kW/ft	22.2	20.15
2. DNB peaking margin penalty, %	—	2.1
3. Nominal heat rate, kW/ft	5.66	5.82
4. Overpower, % of 2568 MWt	114	112
5. Offset limits at rated power		
a. Positive offset	+49	+34
b. Negative offset	-56	-36
6. Trip setpoints at rated power		
a. Positive imbalance	+22	+15
b. Negative imbalance	-33	-15
7. Spike factor	None	1.00 to 1.101
8. Nuclear power peak uncertainty	1.075	1.075

B. Power Peaking Control — LOCA kW/ft Limit

A series of operating restrictions as given in BAW-10079 has been imposed on plant operation to limit the peak linear heat rate to less than the axially dependent LOCA kW/ft limit. These will be factored into the technical specifications as was done for the Oconee 1 and 2 application.

Figure 3.3-1. Trip Setpoints Vs Axial Imbalance Without Densification Effects

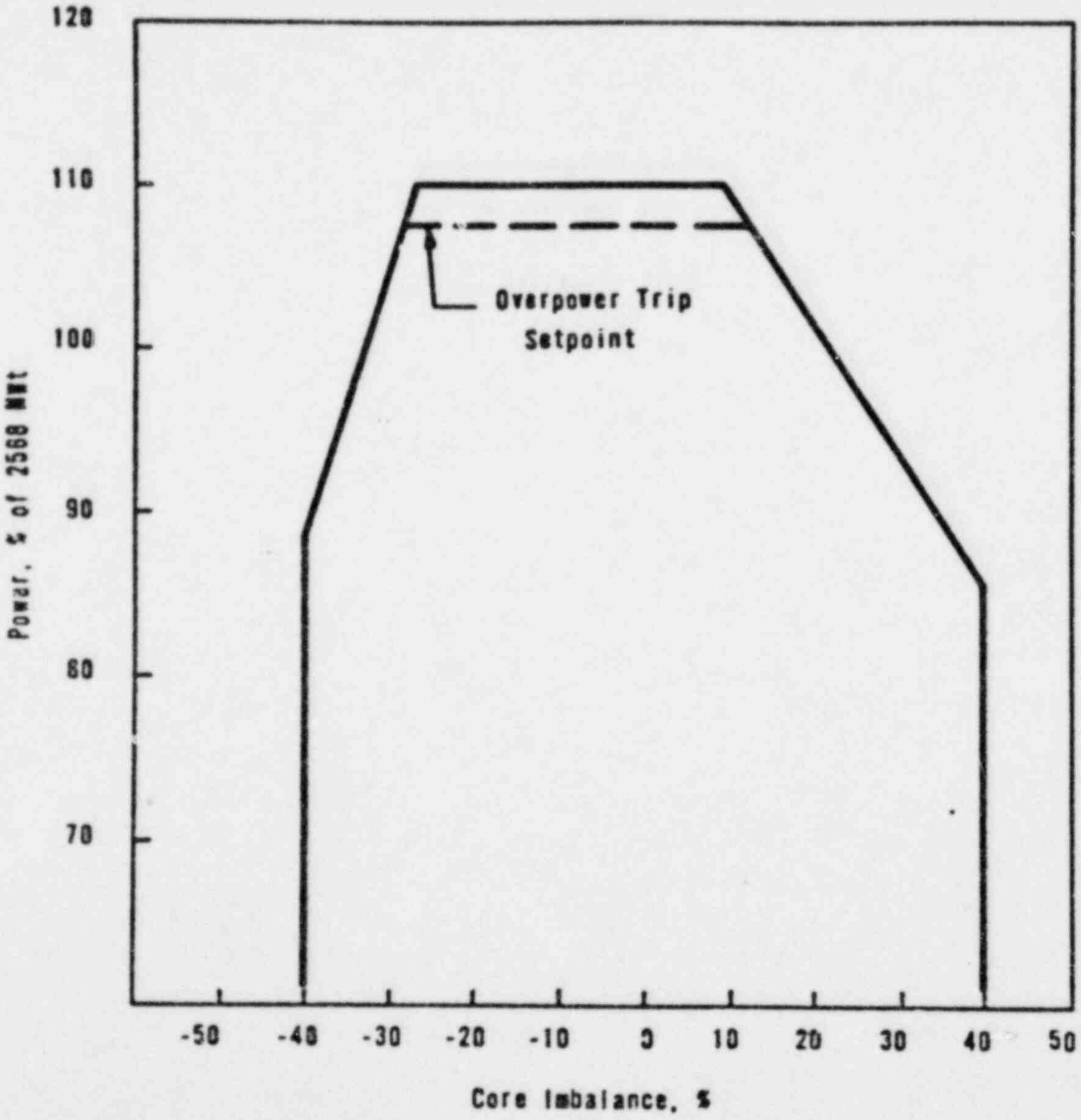


Figure 3.3-2. Calculated Offset Limits Vs Power

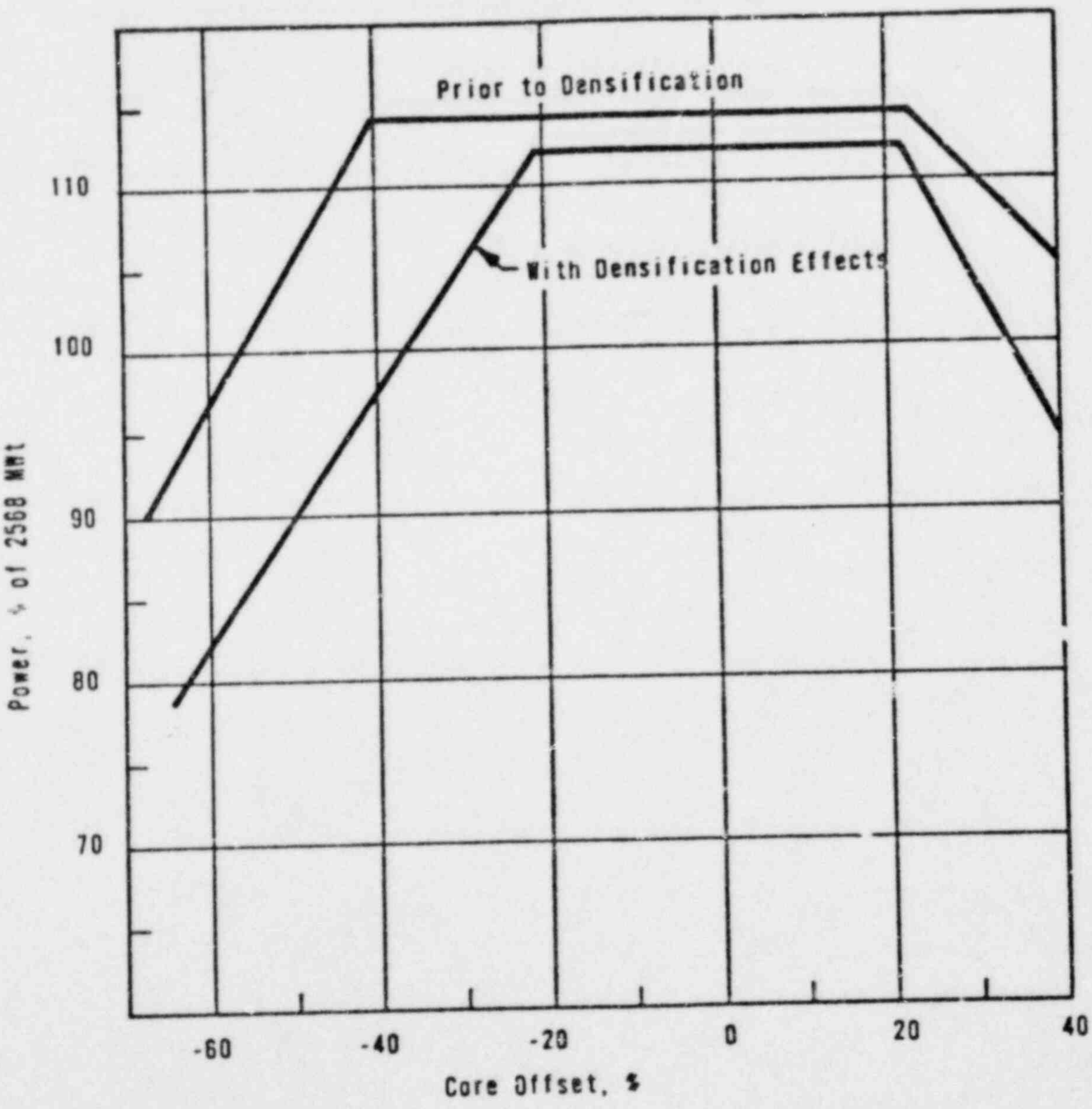


Figure 3.3-3. Trip Setpoints Vs Axial Imbalance
With Densification Effects

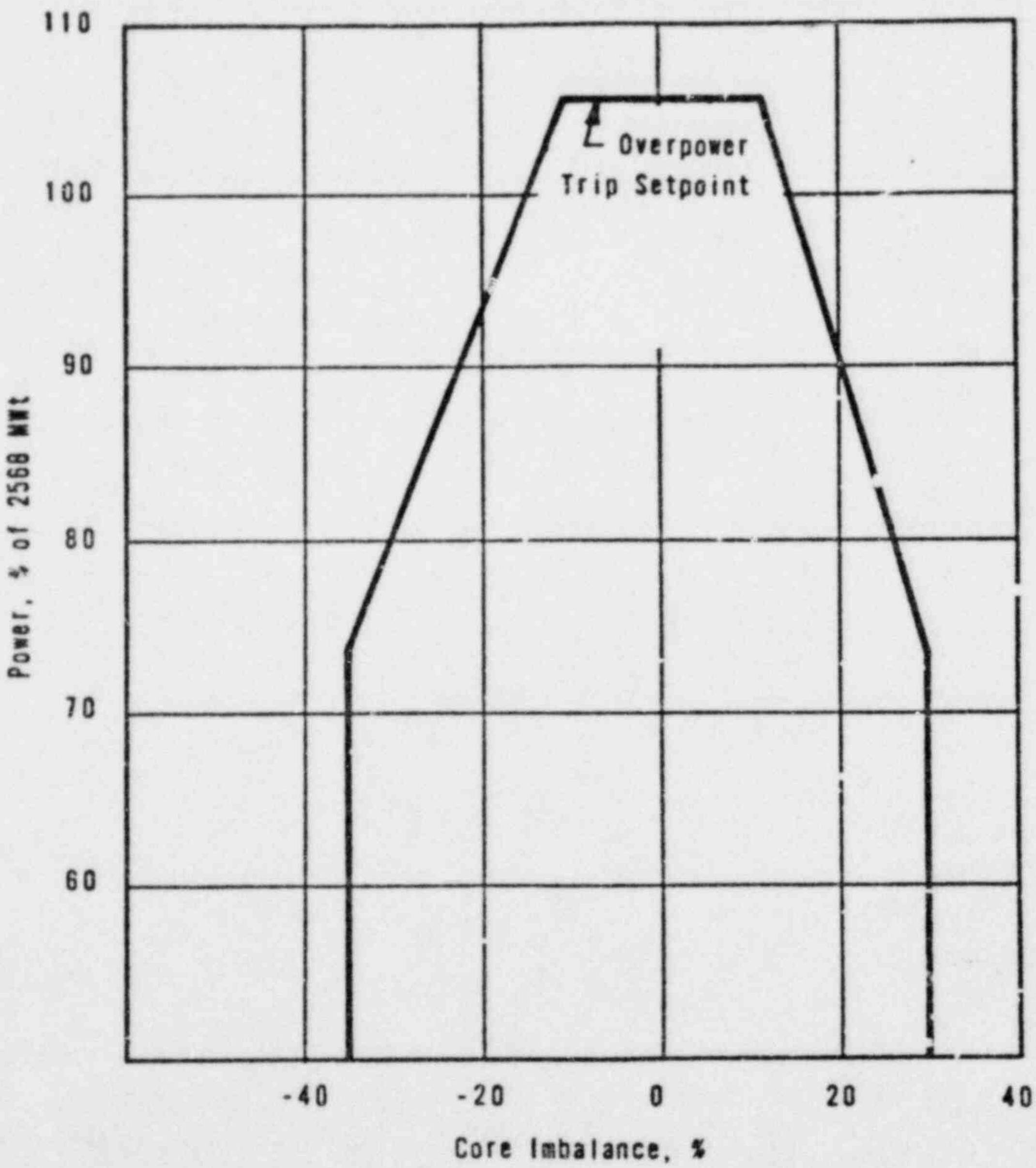
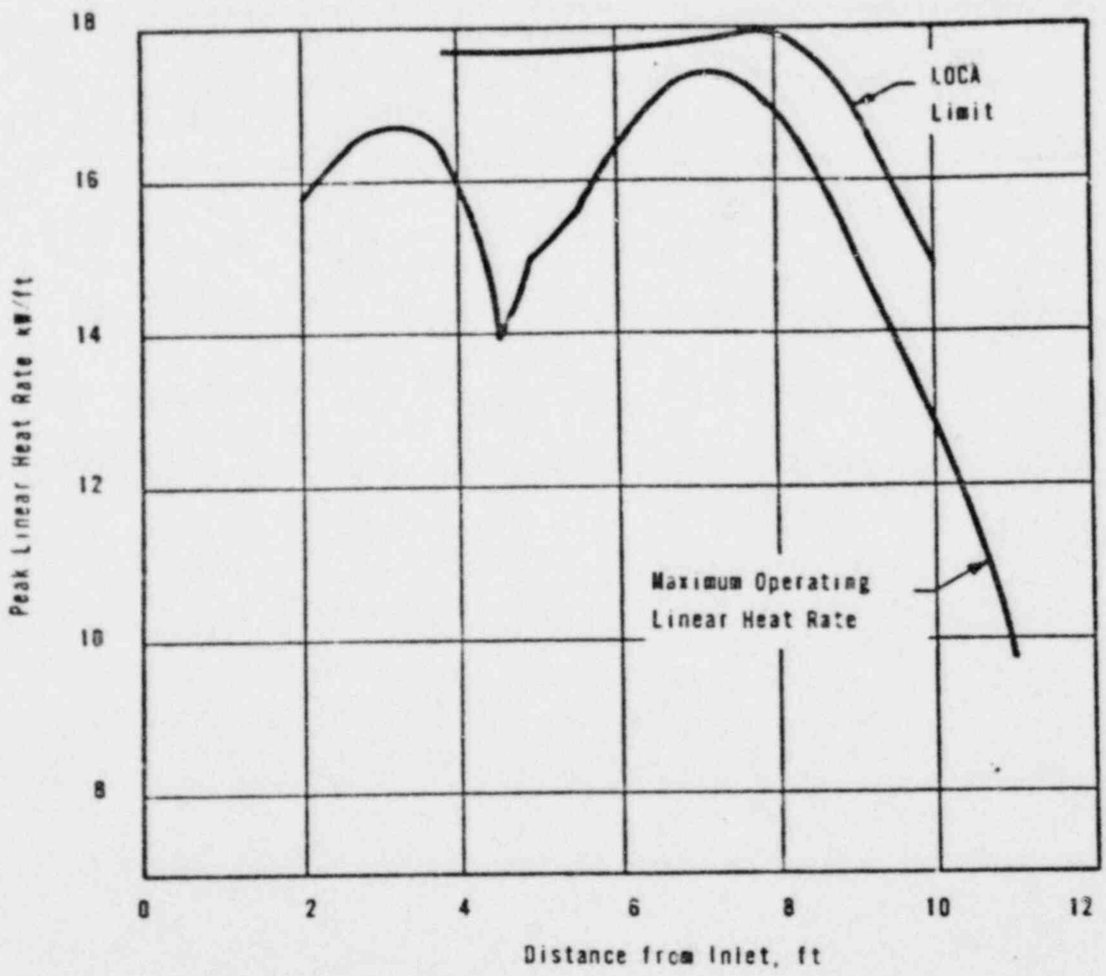


Figure 3.3-4. Envelope of Maximum Operating Linear Heat Rates as Function of Axial Position



3.4. Safety Analysis

3.4.1. General Safety Analysis

3.4.1.1. Introduction

The significant effects of fuel densification are an increase in maximum fuel temperature and a slight increase in average heat flux due to shrinkage of the pellet stack length. In addition, spikes in the neutron power can occur due to gaps in the fuel. These combined effects will lead to a slightly decreased initial DNBR for the accident calculations presented in the Oconee 3 FSAR. For overpower transients such as rod withdrawal, the effects are offset by a reduction in the overpower trip setpoint. The parameters used in the analysis are the same as those used in the FSAR analysis. The changes in fuel geometry and higher fuel temperature will lead to slightly more negative values of the moderator and Doppler coefficients; however, to maintain conservatism the original values were used. All calculations were made for BOL conditions.

3.4.1.2. Reactivity Insertion Transients

The rod withdrawal was not recalculated since for all combinations of parameters, including the simultaneous withdrawal of all rods in the core, the peak thermal power attained during the transient is always less than the 112% design thermal power level; therefore, the 1.3 limit on DNBR is maintained for this transient.

The startup of an inactive loop was not considered in the analysis since the maximum thermal power achieved during the transient is much less than 100% and occurs after full flow is reached. The rod drop accident results in an initial decrease in power which is followed by a return to 100% power. Since it has been shown previously that neither the withdrawal nor the drop of a single control element will perturb the flux shape sufficiently to exceed design conditions at 112%, such occurrences still do not present any thermal problems. The moderator dilution accident results in reactivity insertion rates that are very slow, and the accident is terminated by the high pressure trip well before power reaches the 112% design thermal power level. Therefore, the 1.3 limit on DNBR is maintained.

The ejection of the maximum technical specification value of rod worth (0.65%) from the core, considering the effects of fuel densification, has been analyzed. The basic assumptions for the calculations of the plant parameters are the same as presented in the Oconee 3 FSAR. Figure 3.4-1 shows the neutron power fraction, pressure, and core average heat flux fraction for the ejection of a 0.65% $\Delta k/k$ control rod at beginning of core life. The neutron power reaches about 710% prior to inward rod motion which occurs at about 0.6 second after which the power decays to a value of about 30%. The pressure increases to about 2465 psia due to the increased energy transfer to the coolant, then decreases later on in the transient. Table 3.4-1 shows the important assumptions for the thermal analysis. Figure 3.4-2 shows the axial power distribution used for the thermal analysis. Figure 3.4-3 shows the fuel and cladding temperature at the point of maximum temperature during the transient. It is seen that the fuel temperature reaches centerline melting at about 0.8 second after the peak neutron power. The gap coefficient used was 669 Btu/h-ft^2-°F; this is an effective gap value chosen to match the TAFY steady-state fuel temperature. Figure 3.4-3 also shows the cladding temperature, clad-to-moderator heat transfer coefficient and DNB ratio as a function of time. The DNB ratio reached 1.3 at about 0.4 second after which the maximum cladding temperature reached was 1560F, a value well below the assumed limit of 2300F. It can be seen from the plot of film coefficient versus time that the film boiling heat transfer coefficient reaches a low value of 450 Btu/h-ft^2-°F at about 0.35 second and remains low for several seconds; however, the clad temperature decreases after about 2.2 seconds due to the decreased neutron power. A parameter study was performed to determine the percentage of fuel pins that would experience a DNBR less than or equal to 1.3. It was determined that for the rod worth analyzed (0.65% $\Delta k/k$), about 28% of the pins would exhibit a DNBR of 1.3 or lower. The maximum hot spot fuel enthalpy was found to be about 147 cal/gm.

Secondary system accidents resulting in a power increase occur at or near end of life (EOL) when a highly negative moderator coefficient exists. Since more DNB margin exists at EOL, these secondary accidents, such as a steam line break, are not expected to cause thermal limits that are more severe than those presented in the FSAR. The FSAR analysis of secondary system accidents, such as steam generator tube ruptures and loss of electric power, is unchanged since the thermal power remains the same or decreases during the transients and, therefore, does not increase the potential for reaching design limits.

3.4.1.3. Loss of Coolant Flow

The loss-of-coolant flow accident has been analyzed under initial conditions that represent the most conservative that can occur in the core with densified fuel. The case considered is a balanced power peak case with the power spike placed as shown in Figure 3.4-2. The other parameters normally considered in the coastdown calculations remain unchanged from the FSAR values. Figure 3.4-4 shows power, flow, and the calculated core average heat flux fractions for a four-pump coastdown initiated from 102%. Figure 3.4-5 shows the calculated DNBR and film coefficient as a function of time. The gap conductance used for this calculation was 669 Btu/h-ft^2-°F. The fuel and cladding temperature is not shown since there was no increase in these parameters, because the DNBR for this accident did not go below the criterion value of 1.3. It is therefore concluded that no fuel damage will occur.

An analysis has been performed for the locked rotor accident with the assumptions presented in Table 3.4-1. The power distribution was assumed to be a 1.5 cosine with a power spike located as shown in Figure 3.4-2. Figure 3.4-6 shows the power, flow, and calculated core average heat flux fractions. The pressure was assumed to be constant at 2135 psig. The initial power level for this accident was 102% of 2568 MWt. Trip occurs at about 0.9 second. Figure 3.4-7 shows the maximum fuel temperature versus time. The fuel temperature is affected very little since the power rises only slightly. Figure 3.4-7 also shows the maximum cladding temperature and the DNB ratio. It is seen that the DNBR reaches the criterion value of 1.3 at about 0.9 second after which the cladding temperature increases to a value of 1390F which occurs 4.0 seconds after the initiation of the accident.

3.4.2. LOCA Analysis

3.4.2.1. Introduction

The maximum allowable linear heat generation rate for a typical B&W rodded plant accounting for fuel densification is established in previous fuel densification reports and in BAW-10079, "Operational Parameters for B&W Rodded Plants," which forms the basis for this section of the report.

The effectiveness of the emergency core cooling system (ECCS) for B&W's 177-fuel assembly, vent valve plants during a postulated LOCA was evaluated as specified in Part 4, Appendix A of the AEC Interim Policy Statement. Calculations were made by using the CRAFT computer code during the blowdown period, the REFLOOD code during the vessel refill portion of the transient, and the THETA1-B code for the fuel rod heatup. The results of these analyses and the general methods and assumptions used in B&W's evaluation model are reported in topical report BAW-10034, Rev 3, and in the respective applicant's FSARs. Both analyses were performed without assumed fuel densification effects.

3.4.2.2. Effects of Fuel Densification

The LOCA analyses established the 8.55-ft^2 split in the cold leg pipe at the pump dishcarge as the break size and location resulting in the highest calculated cladding temperature. The consequences of this design basis accident (DBA) with the added restrictions imposed by the postulated fuel densification phenomena have been investigated. Three of the most influential restrictions are as follows:

1. Power spikes assumed to occur in gaps between fuel pellets.

2. Increase in the average linear heat rate due to the assumed reduction in the fuel pellet stack height.

3. A 25% reduction in B&W's fuel pellet gap conductance model as specified by the AEC's preliminary evaluation of the analytical method.

These restrictions, when incorporated in the B&W evaluation model, increase the core average fuel temperature at the start of the LOCA analysis; however, in the earlier analysis (BAW-10034, Revision 3), for conservative purposes, a higher initial core temperature was used rather than the value that resulted from fuel densification. The limiting break size and location does not change due to fuel densification effects.

When the cladding temperature response for the DBA was calculated, the restrictions due to fuel densification were incorporated into B&W's evaluation model, and a maximum linear heat rate was calculated for which a peak cladding temperature of 2300F resulted. Initially, the flux shape, resulting from the design power maneuver for each plant, was used to establish the maximum allowable heat rate. This transient had the largest peaking factors at any time in life. In this analysis, an equivalent radial multiplier was applied over the entire length of the pin instead of imposing a power spike only at the location of the peak axial power. This procedure leads to a conservative evaluation of the peak cladding temperature.

However, the results presented in the fuel densification reports before preparation of the Crystal River 3 report, were calculated by assuming a negative moderator coefficient. Consistent with the analyses and method presented in BAW-10079, this report uses a zero moderator coefficient. The sensitivity of the maximum allowable heat rate (LOCA limit) to this parameter was studied in BAW-10079, for Oconee 2, which is very similar to Oconee 3, and is presented in Figure 3.4-8. (For additional information, see BAW-10079, section 2.2.)

To further demonstrate the safe full-power operation of B&W nuclear plants, the sensitivity of the LOCA limit to the axial position of the power peak was also investigated in BAW-10079. This study utilized a zero moderator coefficient and an axial power peaking factor of 1.7 at various points from an elevation of 4 to 10 feet. This peaking factor was conservative due to operating restrictions placed on B&W reactors, which preclude the existence of peaking factors of this magnitude. For additional conservatism, the most conservative dimensions were used to determine the stored energy values used in the calculations. The results of this analysis are shown graphically in Figure 3.4-9. The calculations showed that the allowable heat rate is essentially constant up to the 8-foot elevation. Beyond this elevation, a gradual decrease is observed owing to the degraded heat transfer during the reflood portion of the LOCA.

The locus of points generated by this analysis defines the allowable heat rate versus axial position at rated power for Oconee 2 and ensures that the LOCA criteria specified in the interim policy statement are met.

Calculations conducted for Oconee 3 ensure that the LOCA limits in Figure 3.4-9 (generated for Oconee 2) are both adequate and conservative for Oconee 3.

Table 3.4-1 Thermal Data Input for Safety Analysis

Active fuel length, in.	139.9
Fuel pellet diameter, in.	0.365
Fuel cladding thickness, in.	0.0265
Gap coefficient, Btu/h-ft^2-°F	669
Film coefficient	Variable[a]
Hot channel factors	
Overall power factor (F_q)	1.0107
Local heat flux factor (F_q'')	1.0137
Flow area reduction factor	0.98
Assumed DNB	1.30
DNB correlation used	W-3
Errors	
T - inlet, F	+2
Pressure, psi	-65
Flux trip setpoint, %	+6.5

[a] After a DNBR of 1.3, the Bishop, Sandburg, Tong correlations were used for both transition and film boiling.

Figure 3.4-1. Pressure, Power, and Flux Vs Time for Densified Fuel, Rod Ejection Accident

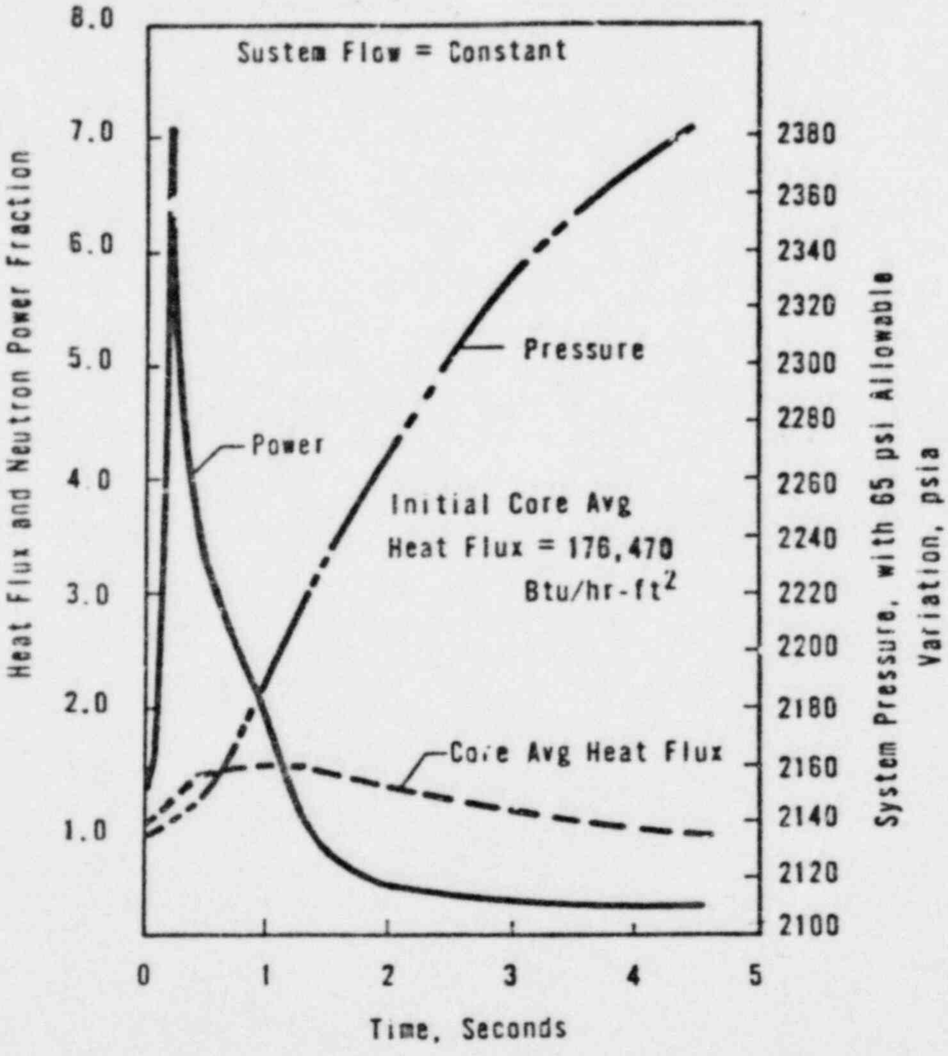


Figure 3.4-2. Slumped and Spiked Axial Flux Shape

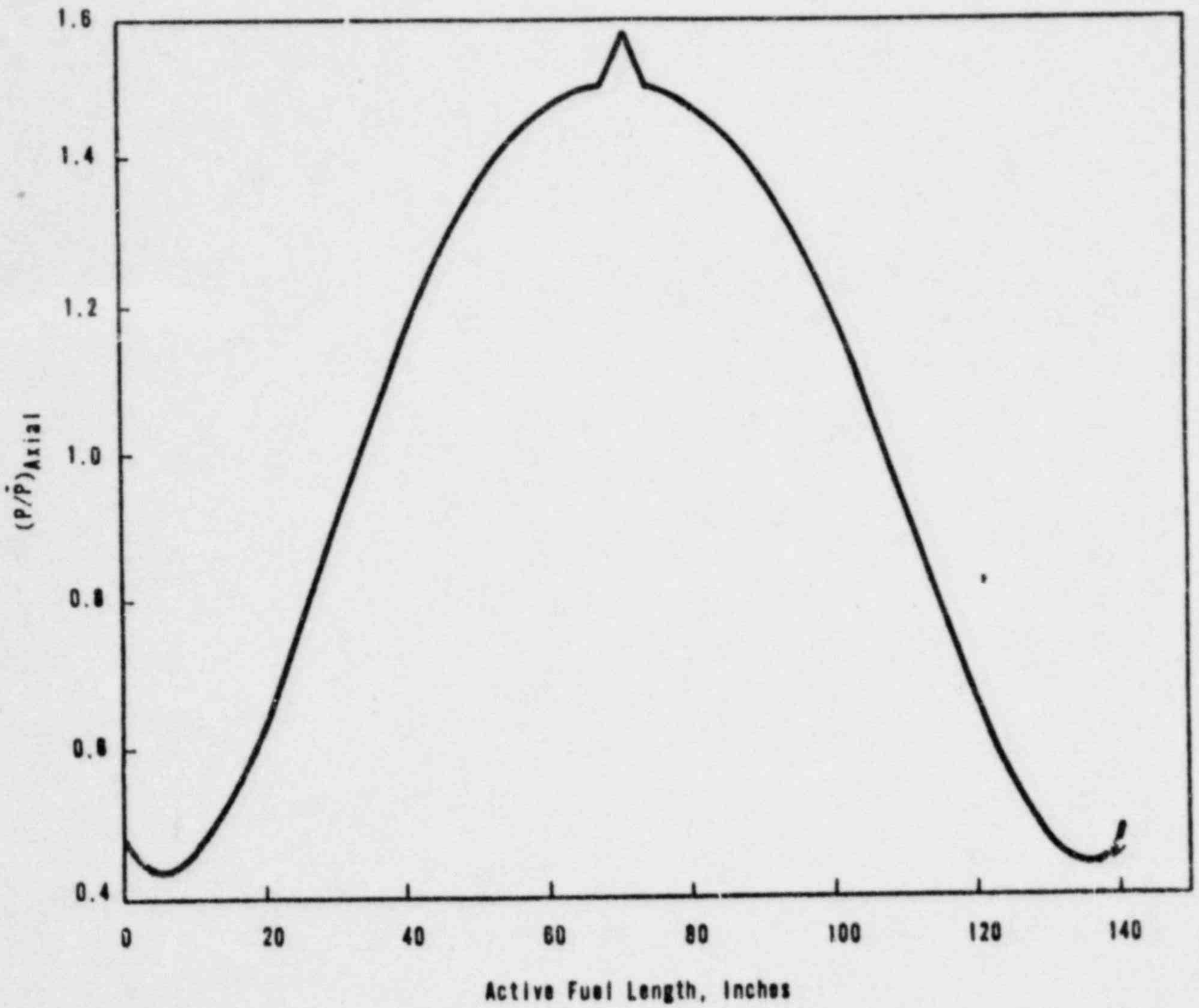


Figure 3.4-3. DNBR, Fuel and Cladding Temperatures, and Film Coefficient Vs Time for Rod Ejection Accident

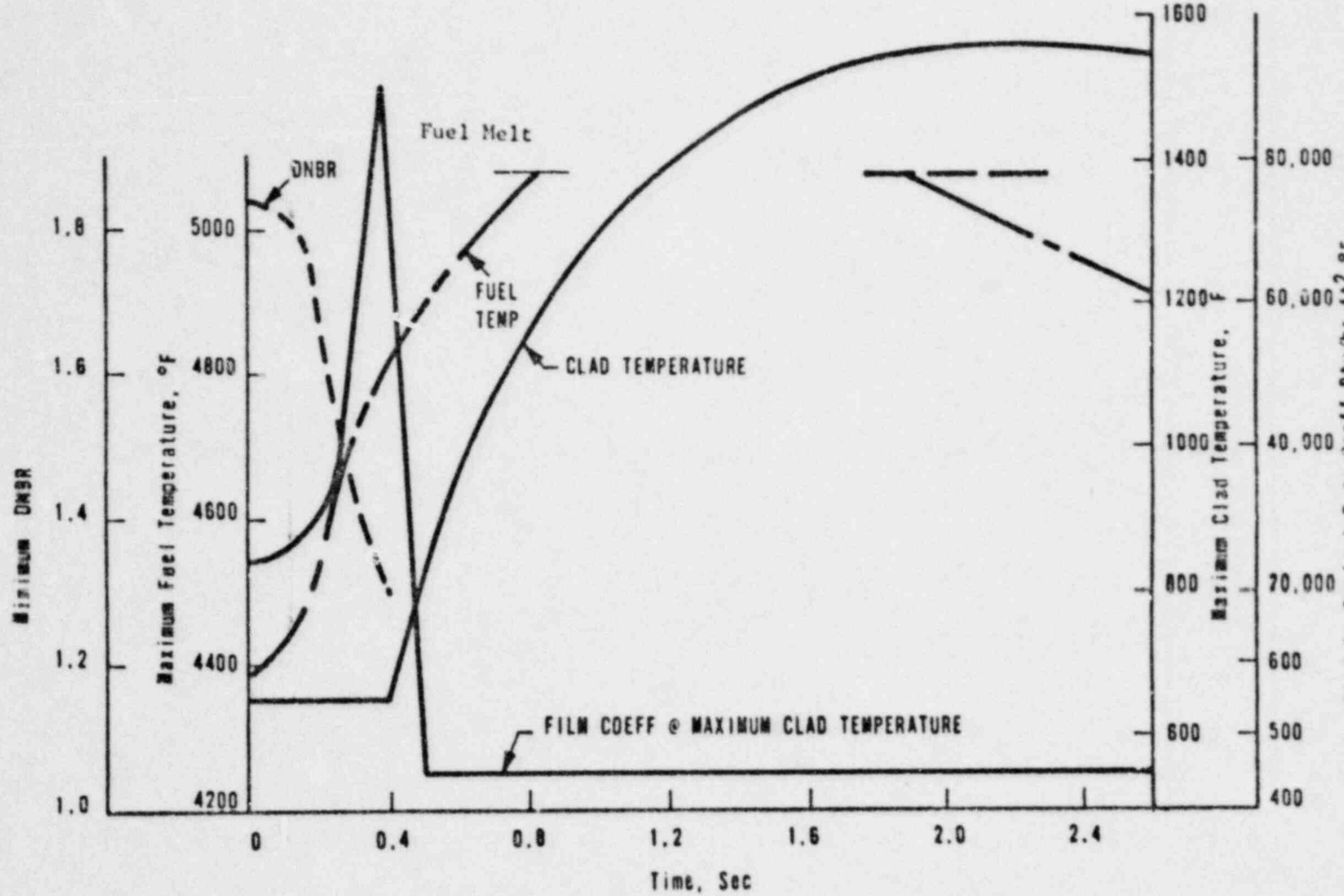


Figure 3.4-4. Power, Flow, and Flux Vs Time for Densified Fuel, Four-Pump Coastdown

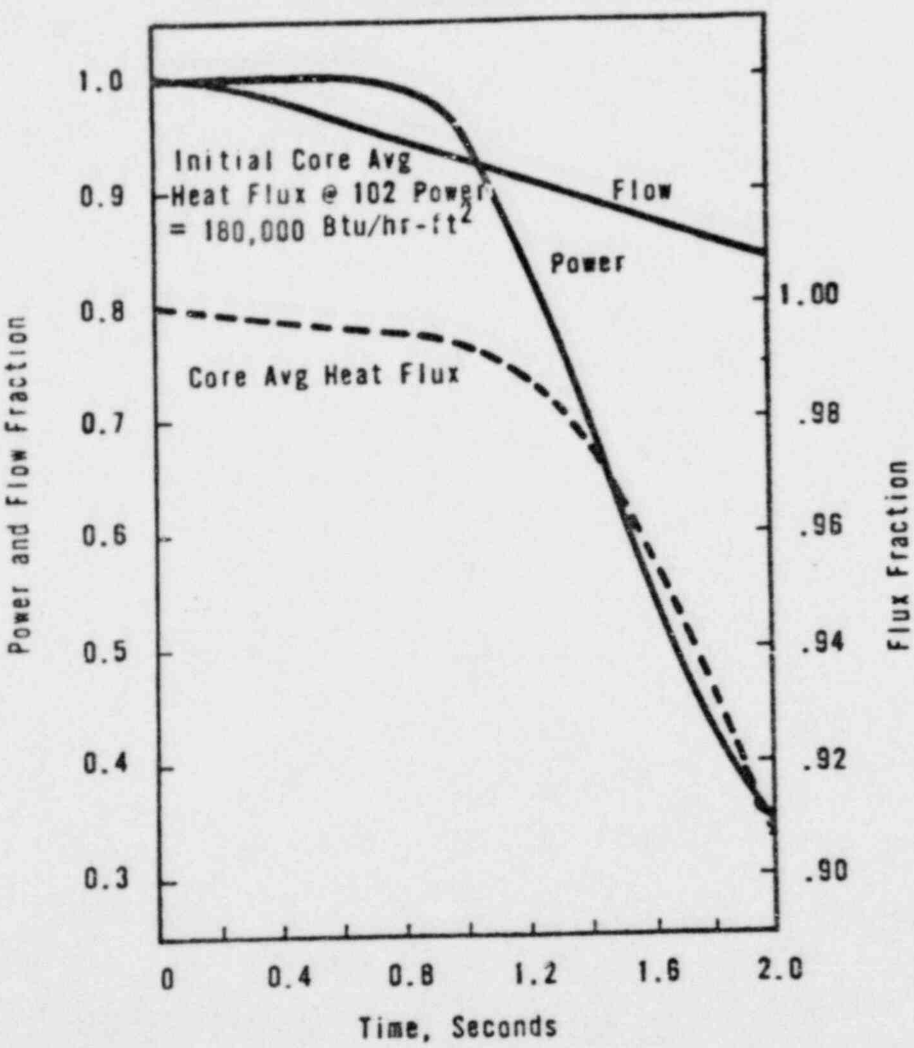


Figure 3.4-5. DNBR and Film Coefficient Vs Time for Densified Fuel, Four-Pump Coastdown

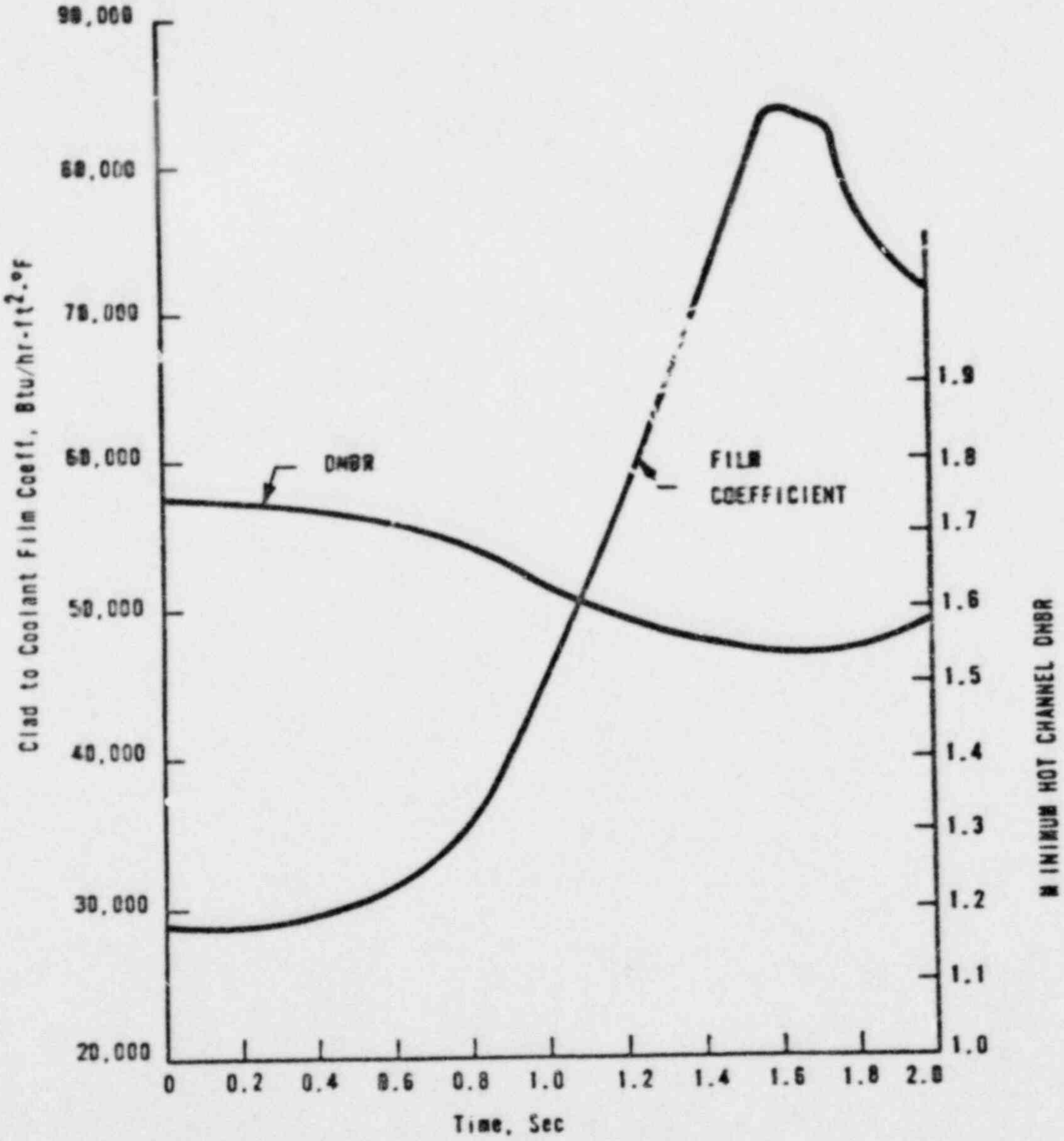


Figure 3.4-6. Power, Flow, and Flux Vs Time for Densified Fuel, Locked Rotor Accident

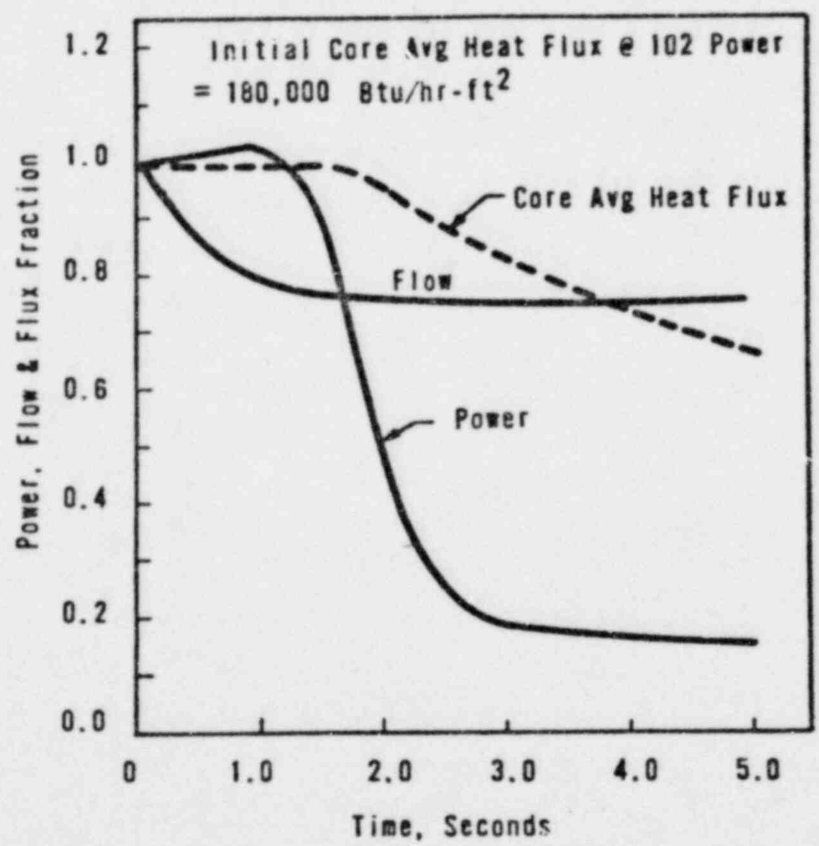


Figure 3.4-7. Cladding and Fuel Temperatures and DNBR Vs Time for Densified Fuel, Locked Rotor Accident

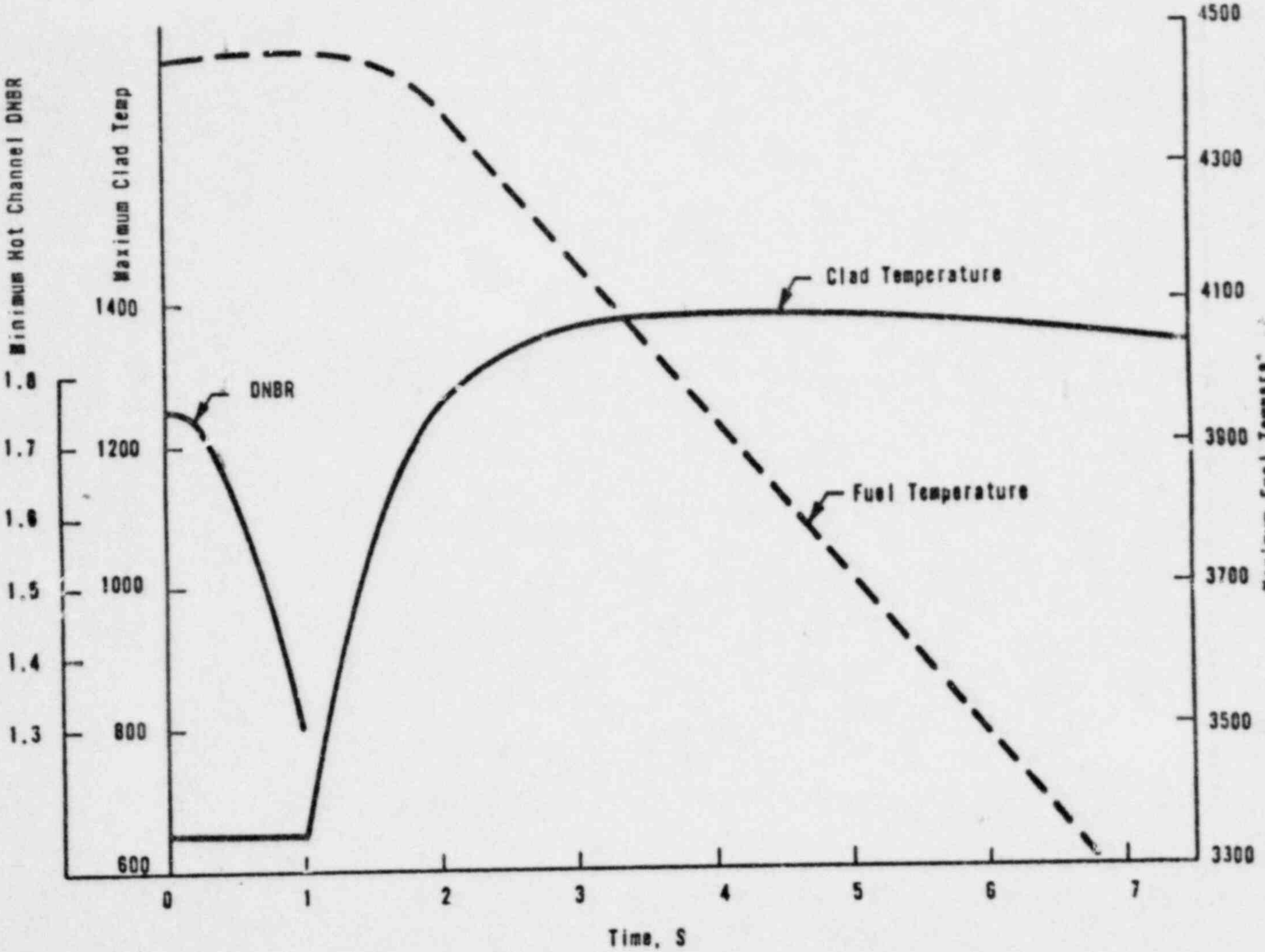


Figure 3.4-8. Sensitivity of Allowable Peak Linear Heat Rate to Moderator Coefficient (Peak 5 Feet From Core Inlet)

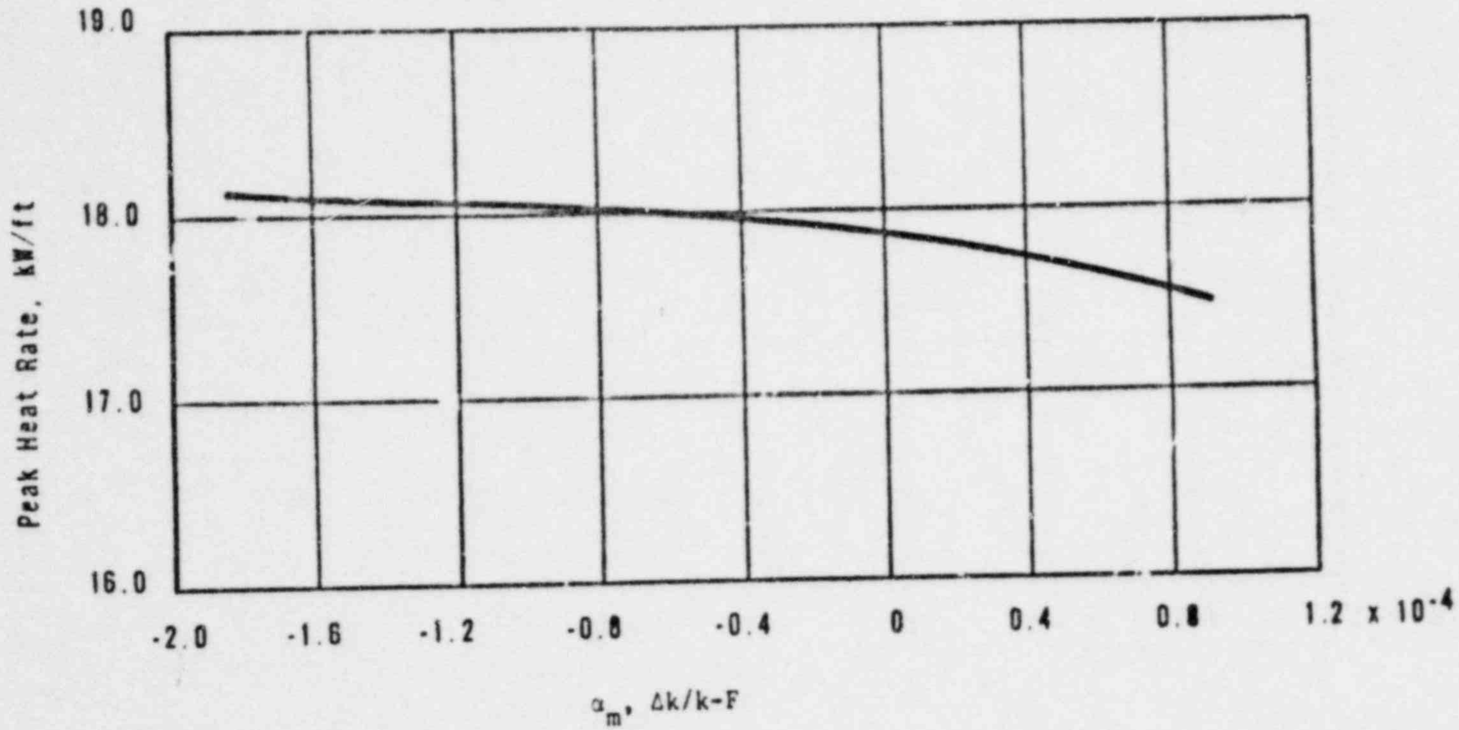
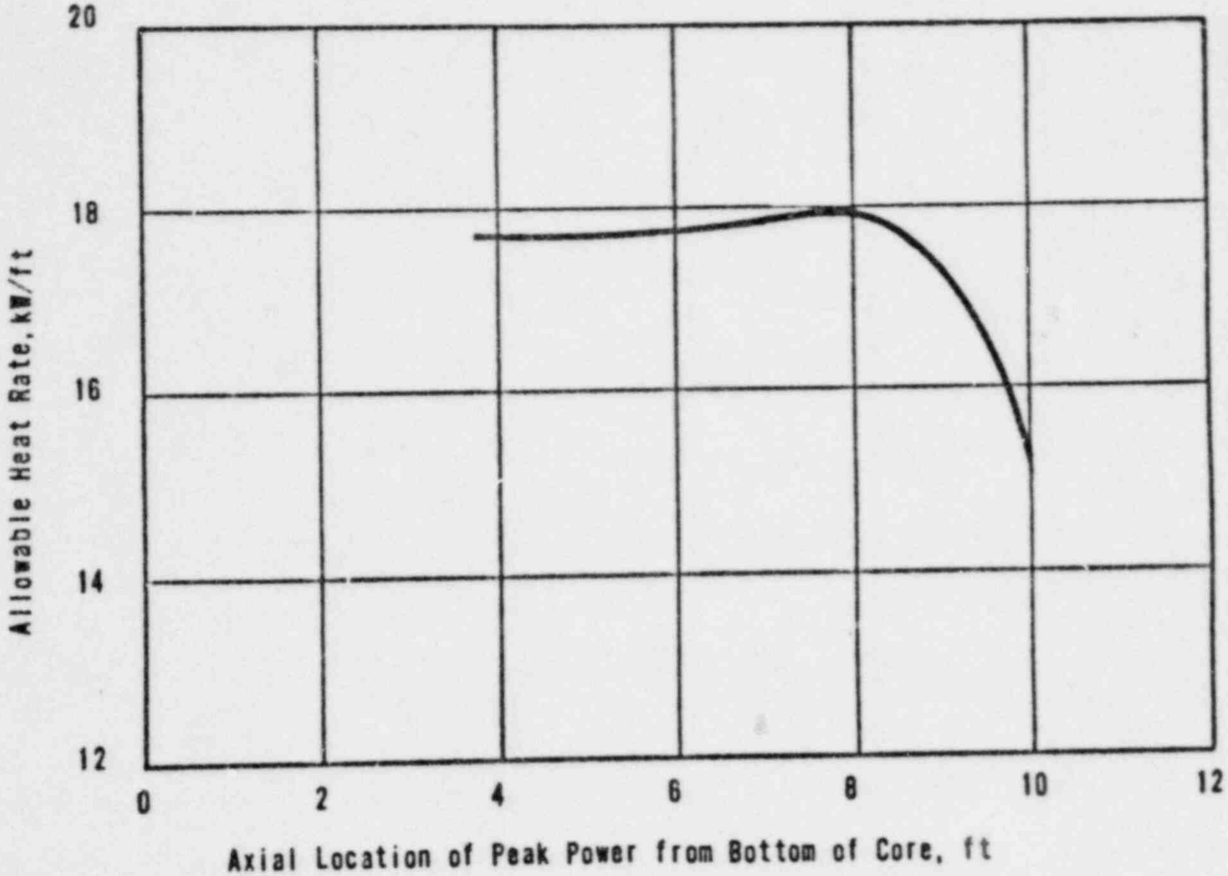


Figure 3.4-9. Axially Dependent Linear Heat Rate



3.5. Mechanical Analysis of Oconee 3 Fuel

3.5.1. Cladding Collapse

Results

1. Predicted time-to-collapse [ ] efph.

3.5.2. Cladding Stress

Results

1. Table 3.5-1 lists maximum cladding circumferential stress calculated at various times in life. In no case does stress exceed yield.
2. Cumulative fatigue damage after three cycles <0.9.

3.5.3. Fuel Pellet Irradiation Swelling

Results

1. Circumferential plastic strain is less than 1% at EOL.

Table 3.5-1. Cladding Circumferential Stress

Case	P_{ext}, psia	P_{int}, psia	T_{clad}, F	Densified σ_{total}, psi	Yield strength, psi	Ultimate strength, psi
Beginning of life — preoperational hot standby — 0% power	2200	460	532	-22,500	48,000	57,000
	2500	460	532	-27,600	48,000	57,000
Beginning of life — void section of cladding — 100% power	2200	580	650	-20,800	45,000	50,000
	2500	580	650	-25,700	45,000	50,000
Beginning of life — void section of cladding — 114% power	2200	600	650	-20,500	45,000	50,000
	2500	600	650	-25,400	45,000	50,000
Beginning of life — fueled section of cladding — 100% power	2200	580	723	-24,900	42,000	44,000
	2500	580	723	-30,000	42,000	44,000
Beginning of life — fueled section of cladding — 114% power	2200	600	733	-25,100	41,500	43,500
	2500	600	733	-30,200	41,500	43,500
End of life — hot standby — 0% power	2200	460	532	-22,500	48,000	57,000
	2500	460	532	-27,600	48,000	57,000
End of life — fueled section of cladding — 100% power	2200	580	704	-23,800	43,000	46,000
	2500	580	704	-28,900	43,000	46,000
End of life — fueled section of cladding — 114% power	2200	600	711	-23,900	43,000	46,000
	2500	600	711	-28,900	43,000	46,000
End of life — Immediately after shutdown	2200	460	535	-22,800	48,000	57,000
	2500	460	535	-27,800	48,000	57,000
End of life, cladding temp of 425F	1725	400	425	-16,300	50,000	62,500

APPENDIX

Design Parameters for Oconee Unit 3

1. Core Operating Conditions

a. The reactor vessel inlet temperatures are as follows:

	Nominal, F	Maximum, F
100% power	554*	556
114% power	550.6	552.6

b. The nominal outlet pressure is 2200* psia, and the minimum outlet pressure is 2135 psia.

2. Core Design

2.1. Fuel Assembly Information

a. There are 177* fuel assemblies in the core.

b. There are 208* fuel rods per assembly with an outside diameter of 0.430* inch and an inside diameter of 0.377* inch.

c. There are 16* control rod guide tubes per assembly with dimensions of 0.530* inch OD × 0.016* inch wall thickness and one instrument tube per assembly with dimensions of 0.493* inch OD × 0.441* inch ID.

d. The fuel rod pitch is 0.568* inch.

2.2. Fuel

a. The undensified active fuel length is 144* inches.

b. The active length of the fuel with densification is 139.94 inches.

c. The cladding is Zircaloy-4 (cold worked) with a thickness of 0.0265 inch.

d. The undensified pellet is 0.370* inch diameter and 0.700 inch long.

e. Unit 3 core 1 is 92.5%* of theoretical density (specified).

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*Values given in the Oconee 3 FSAR.

3. Power Distribution

a. The design core radial power map is shown in Figure A-1.

b. The maximum fuel assembly local rod power peaking distribution is shown in Figure A-2.

c. The percentage of power generated in the fuel is 97.3%*.

d. The percentage of power generated in non-fuel regions is 2.7%*.

4. Fluid Flow

a. Coolant Flows and Mass Velocities:

	Vent valves closed	One vent valve open
Total reactor vessel coolant flow, 10^6 lbm/h	131.32*	132.60
Effective core coolant flow, 10^6 lbm/h	124.23*	118.52
Average mass velocity at core inlet, 10^6 lbm/h-ft^2	2.53	2.41
Inlet mass velocity to hot assembly, 10^6 lbm/h-ft^2	2.235	2.13

b. The core flow area (effective for heat transfer) is 49.19* ft^2.

5. Hot Channel Factors

a. The hot channel factor on average pin power (F_q) is 1.011.* It is applied on the enthalpy rise for the entire channel. The hot channel factor on local surface heat flux (F_q'') is 1.014.* This value is applied locally on the calculated local surface heat flux.

b. Flow area is reduced in the hot channel by a flow area reduction factor (F_A) of 0.98.* This value is applied over the entire length of the channel.

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*Values given in the Oconee 3 FSAR.

c. Flow is reduced in the hot bundle by a flow maldistribution factor, which is 95%* of the nominal isothermal bundle flow.

d. The energy mixing coefficient (α) is 0.02.*

6. Core Peaking Conditions

The 1.5 cosine, symmetrical axial power shape of the reference design was used as a base case to determine whether other axial power shapes in any way magnified the variation in DNBR. A 1.78 radial-local nuclear peaking factor ($F\Delta h$) associated with a 1.5 cosine axial flux shape establishes the maximum design condition resulting in the 1.71 DNBR at 114% of 2568 MWt.

The results indicate that outlet peaks with the spike show an overall larger degradation in DNBR than does the densified 1.5 cosine axial power shape and its associated power spike. B&W utilized a conservative 1.83 ($P/\overline{P}$) outlet axial power shape in conjunction with a 1.49 ($P/\overline{P}$) radial-local peak to maintain the reference design DNBR of 1.55 at 114% of 2568 MWt.

This set of peaking conditions maximizes the DNBR penalty associated with fuel densification and prevents the need to re-evaluate all DNBR data for the power/imbalance/flow trip system. The penalty determined in this manner was used to modify the power/imbalance/flow system as indicated in section 3.3.4. The 1.83 ($P/\overline{P}$) outlet axial power shape shown in Figure A-4 is precluded during normal operation as described in the technical specifications and as such is not a design criterion.

The 1.5 axial power shape, in conjunction with a 1.783 radial shape peaking combination, is used for transient and accident analyses. This particular shape results in a more conservative DNBR than any other shape existing during normal operation. This shape is shown in Figure A-3.

For LOCA analysis, the design basis axial power shape was a 1.816 peaking at a distance of 1.0 feet below the core midplane. This shape and peak, in conjunction with the calculated radial factor, are most conservative for the LOCA peak cladding temperature analysis and

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*Values given in the Oconee 3 FSAR.

could occur momentarily during the period of xenon undershoot following a design basis (100-30-100%) transient. The peaking factor and the associated radial factor are within the DNBR limiting criteria statement given in the previous paragraph. The reason is that in LOCA analysis, the important parameter is peak cladding temperature, whereas for DNBR protection, the important parameter is not only heat flux and flux shape, but also the integration of heat input up the channel and the resultant enthalpy rise.

The non-densified DNBR at design overpower is 1.55. With densification and the spike utilizing the 1.83 axial power shape, the DNBR is 1.48. The reduction in overpower limit given in section 3.3.4 increased the 1.48 DNBR to the design value of 1.55.

7. Heat Flux Conditions

The following data are based on the peaking conditions above so that a meaningful comparison between non-densified and densified fuel can be made.

7.1. Non-Densified Conditions

a. The heat transfer surface area per fuel pin is 1.3509 ft^2.

b. The average heat flux ($\bar{q}_s''$)* is 171,470 Btu/h-ft^2.

c. The maximum heat flux at minimum DNBR is 457,774 Btu/h-ft^2.

[q_s'' (MDNBR) = $\bar{q}_s''$ × 1.55 × 1.49 × 1.14 × 1.014].

Axial ($P/\bar{P}$) at MDNBR = 1.55.

($P/\bar{P}$) radial-local = 1.49.

Max overpower = 114% of 2568 MWt*.

Hot channel factor on local surface heat flux = 1.014*.

d. The average power density in the core is 83.38 kW/liter, and the average linear heat rate is 5.66 kW/ft.

e. The maximum surface temperature at the exterior of the cladding at 100% power is 650F for a pressure of 2135 psia.

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*Values given in the Oconee 3 FSAR.

7.2. Densified Conditions

a. The heat transfer surface area per fuel pin is 1.3128 ft^2.

b. The average heat flux is 176,446 Btu/h-ft^2.

c. The maximum heat flux at minimum DNBR is 483,213 Btu/h-ft^2:

$[q''_s \text{(MDNBR)} = \bar{q}''_s \times 1.59 \times 1.47 \times 1.14 \times 1.014]$.

Axial $(P/\bar{P})$ at MDNBR with power spike = 1.59.

$(P/\bar{P})$ radial local = 1.49.

Max overpower = 114% of 2568 MWt*.

Hot channel factor on local surface heat flux = 1.014*.

d. Average volumetric power density in the core is 83.38 kW/liter, and the average linear heat rate is 5.82 kW/ft. This assumes that all fuel pins have the densified active length, which is conservative.

e. The maximum surface temperature at the exterior of the cladding at 100% power is 650F for a pressure of 2135 psia.

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*Values given in the Oconee 3 FSAR.

Figure A-1. Design Radial Power Distribution

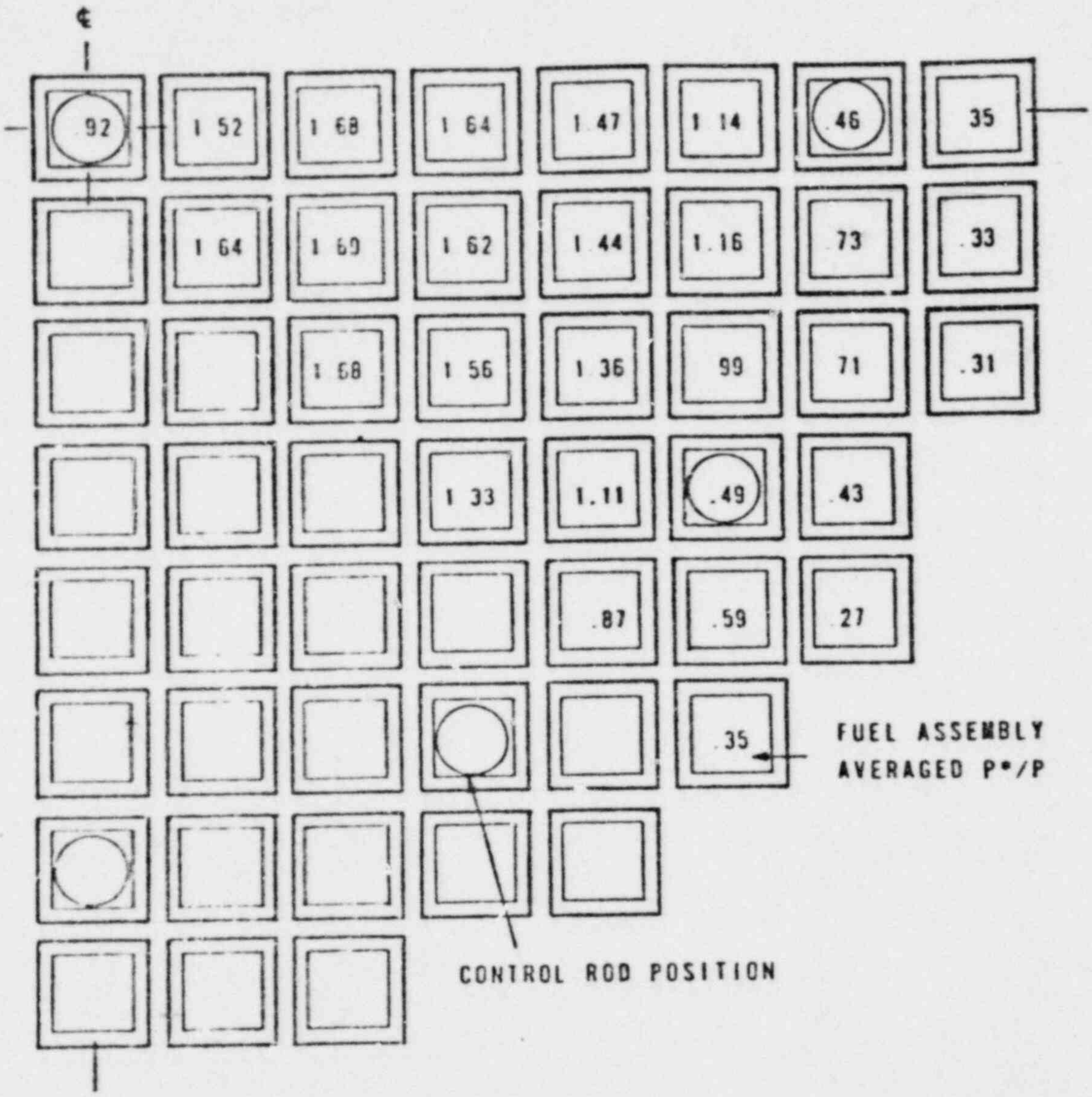
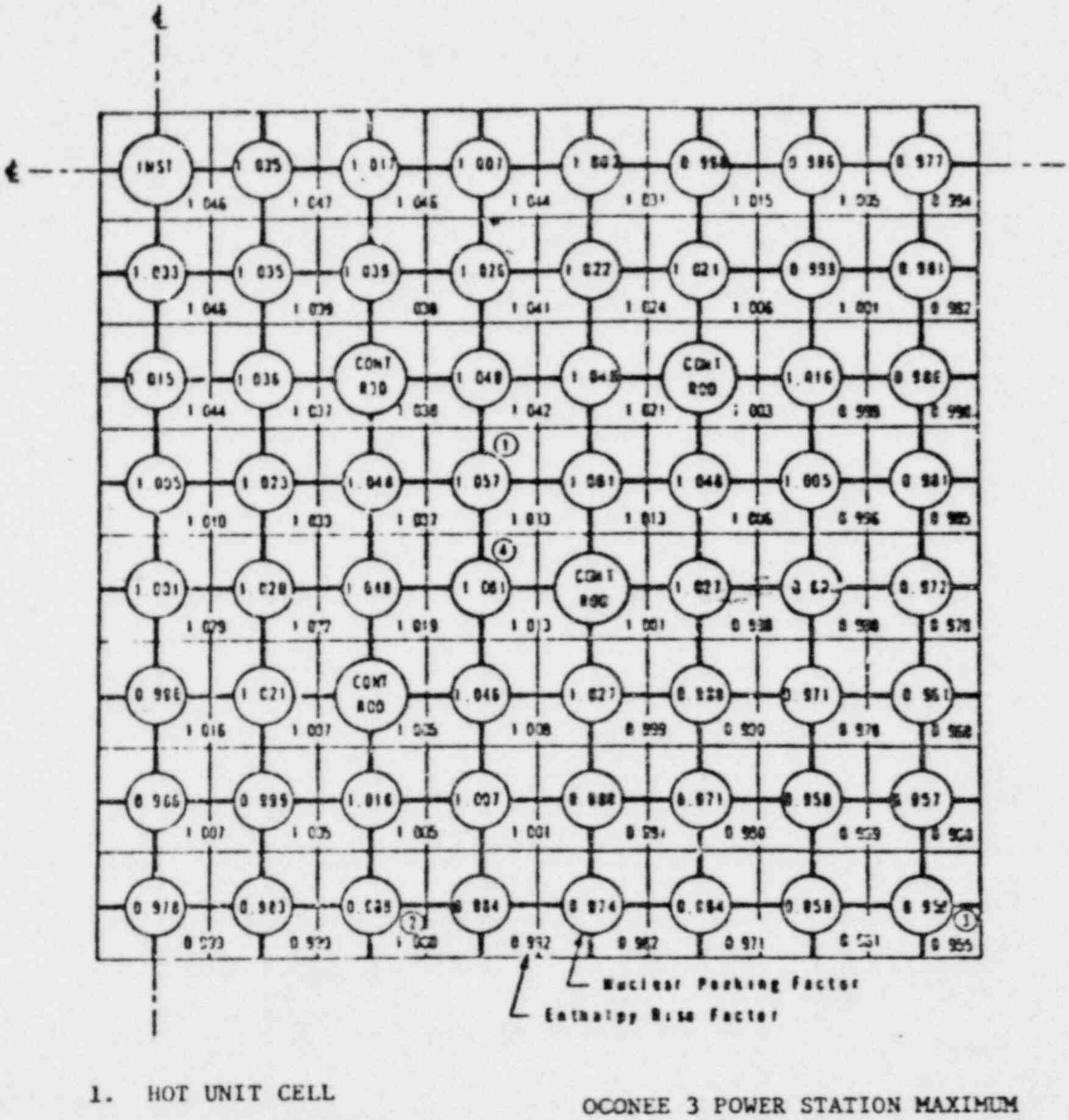


Figure A-2. Maximum Fuel Rod Power Peaks



1. HOT UNIT CELL
2. HOT WALL CELL
3. HOT CORNER CELL
4. HOT CONTROL ROD CELL

OCONEE 3 POWER STATION MAXIMUM FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS

Figure A-3. Effects of Fuel Densification on 1.5 Cosine Reference Design Axial Flux Shape

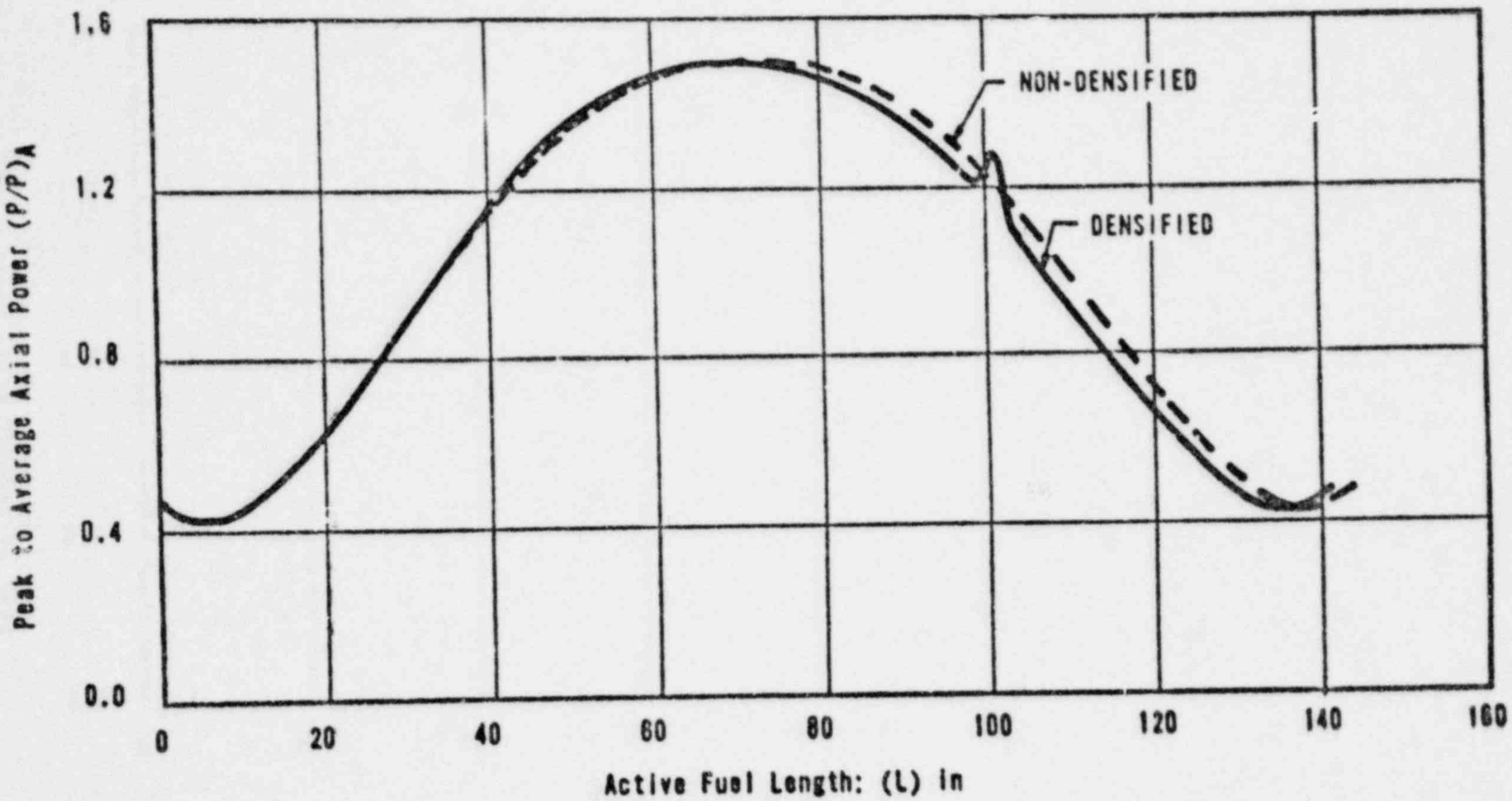
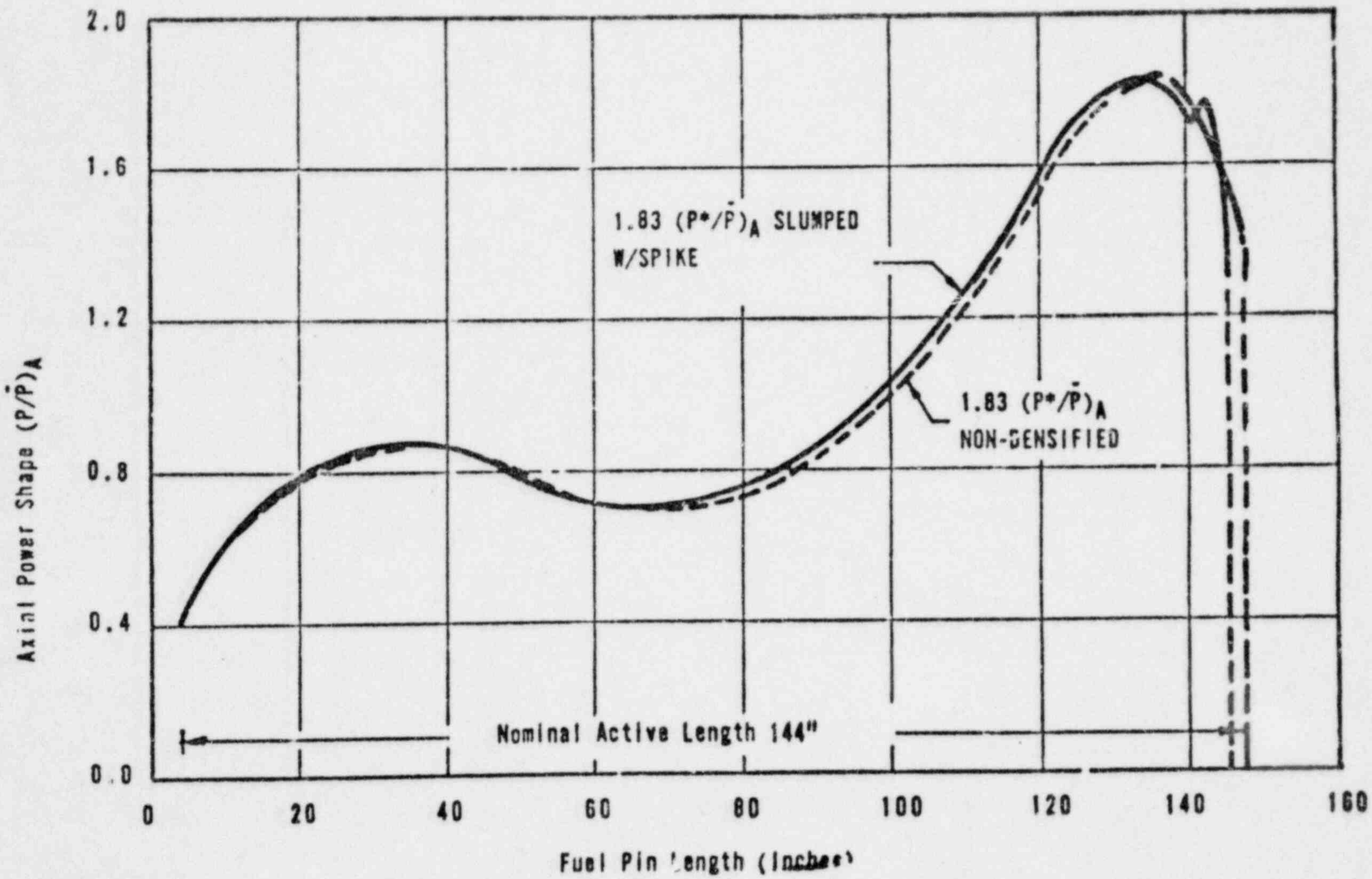


Figure A-4. Effects of Densification on 1.833 $(P/\bar{P})_A$ Axial Flux Shape



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