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OCONEE 1, CYCLE 2

- Reload Report -

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OCONEE 1, CYCLE 2

— Reload Report —

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1. INTRODUCTION AND SUMMARY

This report describes the basis to justify operation of the Oconee Nuclear Station, Unit 1, Cycle 2 at the rated core power of 2568 MWt. Included herein are the required analyses including fuel densification effects and revised ECCS (FAC) criteria supporting the necessary revision to the Technical Specifications associated with cycle 2 operation.

In July 1973, Babcock & Wilcox filed non proprietary topical report BAW-10055 (Rev. 1), "Fuel Densification Report," which describes the methods used in analyzing fuel densification effects,[1] and BAW-1388 (Rev. 1), Oconee 1 "Fuel Densification Report." BAW-1388 utilized the methods described in BAW-10055 and supported the operation of the cycle 1 of Oconee 1 at the rated core power of 2568 MWt. An additional report, BAW-10079, "Operational Parameters for B&W Rodded Plants," filed in October 1973, set forth the core operating parameters for B&W rodded plants and outlined the analysis used to determine plant operating restrictions owing to postulated effects of fuel densification.

The calculational methods and procedures used in the Arkansas and SMUD "Classification and Selective Loading" letter reports (December 1973) were used to determine the as-built fuel linear heat rate capabilities and as a guideline for fuel placement.

This report employs the analytical techniques and design bases established in the reports mentioned above to support cycle 2 operation of Oconee 1 at 2568 MWt.

A brief summary of cycle 1 and 2 reactor parameters that are related to power capability (similar to those in the FSAR) is included in this report. In those cases where cycle 2 characteristics proved to be conservative with respect to those analyzed for cycle 1 operation, no new analysis was conducted. In several instances, credit for new technology and operating experience, where applicable, have been employed to provide increased design margin.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria, it has been concluded that Oconee 1, cycle 2 can be safely operated in the proposed manner at the rated core power level of 2568 MWt.

2. CORE DESIGN

2.1. Introduction

Oconee 1 achieved initial criticality on April 19, 1973, and power generation commenced on May 4, 1973. The 100% power level of 2568 MWt was reached on November 8, 1973. Control rod interchanges were performed at 92 and 196 effective full-power days (EFPD). The design fuel cycle of 310 EFPD is scheduled for completion in the latter part of October 1974.

Operation of cycle 2 is scheduled to begin in early December 1974. The design cycle length is 290 EFPD with one control rod interchange to be performed at approximately 50 EFPD.

2.2. Description

The Oconee 1 reactor core is described in detail in section 3 of the Oconee Nuclear Station, Final Safety Analysis Report.

The cycle 2 core consists of 177 Fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel rods have an undensified nominal active length of 144 inches. The cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished end, cylindrical pellets of uranium dioxide which are 0.700 inch in length and 0.370 inch in diameter (see Table 2-1 for additional data).

Figure 2-1 is the core loading diagram for Oconee 1, cycle 2. The initial enrichments of batches 2 and 3 were 2.10 and 2.15 wt % ^{235}U respectively. Batches 4A and 4B are enriched to 2.60 and 3.20 wt % ^{235}U, respectively. All of the batch 1 assemblies and 20 of the batch 2 assemblies will be discharged at the end of cycle 1. The remaining batch 2 and 3 assemblies will be shuffled to new locations. The batch 4A assemblies are inserted in the center core location and on the major axes. The batch 4B assemblies will occupy the periphery and one location on each of the major axes and diagonals.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods and soluble boron shim. In addition to the full-length control rods, eight partial length control rods are provided for additional control of axial power distribution. The locations of the 69 control rods are indicated on Figure 2-2.

The system pressure is 2200 psia, and the undensified nominal heat rate is 5.656 kW/ft at the rated core power of 2568 MWt. These values are the same as for cycle 1 operation.

2.3. Core Physics

Table 2-2 compares the core physics parameters of cycles 1 and 2. The values for both cycles were generated using PDQ07. The cycle 1 values also reflect application of data from the Oconee 1 Startup Report. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles.

The shorter cycle 2 will produce a smaller cycle differential burnup than that for the cycle 1, although the lower batch 4 loading (mtU) will result in a slightly higher burnup per EFPD of operation (MWd/mtU-EFPD). The accumulated average core burnup will be higher in the cycle 2 than for cycle 1 because of the presence of the once-burned batch 2 and 3 fuel. Figure 2-3 illustrates a representative relative power distribution for the beginning of the second cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 2 are lower in all cases than for the cycle 1. The control rod worths for hot full power (due to changes in radial flux distribution and isotopics) are somewhat less than those for cycle 1 (at BOL), although they are sufficient to maintain the required shutdown margin, as indicated in Table 2-3. The stuck rod and ejected rod worths are slightly higher than for cycle 1. No adverse safety implications are associated with these higher worths since the previous safety analysis assumed an ejected rod worth well in excess of the values calculated for cycle 2; the adequacy of the shutdown margin with cycle 2 stuck rod worths is demonstrated in Table 2-3.

The cycle 2 power deficits from hot zero power to hot full power are higher than those for cycle 1 due to a more negative moderator coefficient in cycle 2. The power Doppler coefficients in both cycles are nearly equal.

The differential boron worths and total xenon worths for cycle 2 are lower than for cycle 1 due to depletion of the fuel and the associated buildup of fission products.

2.4. Core Loading — Batch 4 Fuel

The batch 4 fuel assemblies will be loaded as shown in Figure 2-1. As-built data have been used to ensure eighth-core symmetry in ^{235}U loading. Also, fuel assemblies with higher ^{235}U loadings will be placed in locations of low power density to minimize power peaking.

As specified in the Nuclear Analysis section of this report, a 20.15 kW/ft fuel melt limit has been employed in calculating the reactor protection system (RPS) setpoints. This value is the same as that used in the cycle 1 analysis. Based on as-built data, all batch 4 assemblies meet or exceed this criterion except for three assemblies that have been assigned a maximum linear power rating of 20.02 kW/ft. These assemblies will be placed in core locations that will have maximum heat rates less than 15.3 kW/ft in cycle 2 and less than 19.8 kW/ft in cycle 3 to maintain adequate fuel melt margins. These values have been calculated conservatively with respect to the calculational method used in the Arkansas and SMUD "Classification and Selective Loading Reports"*.

In addition, assembly 1D61 will be placed in core location D-14 in conjunction with B&W's continuing program to evaluate fuel performance. Contained in one fuel rod of assembly 1D61 are three ceramic spacers which simulate fuel densification gaps. The proposal to insert this special assembly into Oconee Unit 1 has been described in a letter (6/18/74) to Angelo Giambusso, USAEC.

Table 2-1. Cycle 2 Fuel Data

	Batch			
	2	3	4A	4B
Enrichment, wt% ^{235}U	2.10	2.15	2.60	3.20
Nominal geom. density, % theor	93.5	93.5	95.5	95.5
No. of assemblies	36	80	5	56
Burnup at BOC 2, MWd/mtU	11,695	7,100	0	0

Table 2-2. Oconee 1 Cycle 2 Physics Parameters

	Cycle 2	[illegible]
Cycle length, EFPD	[illegible]	[illegible]
Cycle burnup, MWd/mtU	[illegible]	[illegible]
Average core burnup – EOC, MWd/mtU	[illegible]	[illegible]
Initial core loading, mtU	82.6	[illegible]
Critical boron – BOC, ppm		
HZP – all rods out	[illegible]	[illegible]
HZP – banks 7 and 8 inserted	1,154	[illegible]
HFP – banks 7 and 8 inserted	1,028	[illegible]
Critical boron – EOC, ppm		
HZP – all rods out	[illegible]	[illegible]
HFP – bank 8 (37.5% wd, equil Xe)	75	[illegible]
Control rod worths – HFP, BOC, %Δk/k		
Banks 1-7 (bank 8, 37.5% wd)	[illegible]	[illegible]
Bank 6	1.12	1.17
Bank 7	[illegible]	[illegible]
Bank 8 (37.5% wd)	[illegible]	0.35
Control rod worths – HFP, EOC, %Δk/k		
Banks 1-7	[illegible]	[illegible]
Bank 7	[illegible]	[illegible]
Bank 8 (37.5% wd)	0.41	[illegible]
Ejected rod worth – HZP, %Δk/k		
EOC	[illegible]	[illegible]
BOC	0.25	[illegible]
Stuck rod worth – HZP, %Δk/k		
BOC	2.55	[illegible]
EOC	1.56	1.69
Power deficit, HZP to HFP, %Δk/k		
BOC	-1.34	[illegible]
EOC	-1.99	[illegible]
Power Doppler coeff – BOC, 10^{-2}(%Δk/k-% power)		
100% power (0 Xe)	-0.94	-0.94
95% power (0 Xe)	-1.01	-1.00
75% power (0 Xe)	-1.03	[illegible]
40% power (0 Xe)	-1.03	[illegible]
Power Doppler coeff – EOC, 10^{-2}(%Δk/k-% power)		
95% power (equil Xe)	-1.15	-1.14
Moderator coeff – HFP, 10^{-2} (%Δk/k-°F)		
BOC (0 Xe, 1000 ppm)	-0.79	-0.12
EOC (equil Xe, 17 ppm)	-2.35	-2.27
Boron worth – HFP, ppm/%Δk/k		
BOC (1000 ppm)	0.97	0.84
EOC (17 ppm)	0.91	0.82
Xenon worth – HFP, %Δk/k		
BOC (4 days)	2.64	2.77
EOC (equilibrium)	2.69	2.74

Table 2-3. Shutdown Margin Calculation — Oconee 1, Cycle 2

	BOC, HZP	EOC, HZP*
1. Available Rod Worth		
Total rod worth, HFP**	10.30	11.60
Worth reductions		
Poison material burnup	10.16	10.64
HFP to HZP	8.53	8.94
Stuck rod worth (HZP)	2.55	1.66
Net	5.98	6.98
10% uncertainty	0.60	0.70
a. Total available rod worth	5.38	6.28
2. Required Rod Worth		
Power deficit, HFP to HZP	1.34	1.99
Inserted rod worth, HZP	1.05	1.89
Flux redistribution	0.40	1.00
a. Total required worth	2.79	4.88
Shutdown Margin (1a - 2a)	2.59	1.40

*For shutdown margin calculations this is defined as ~265 FPD, the time at which the transient control rod (7) begins to move out of the core.

**HFP denotes Hot, Full Power
HZP denotes Hot, Zero Power

Figure 2-1. Cycle 2 - Core Loading Diagram

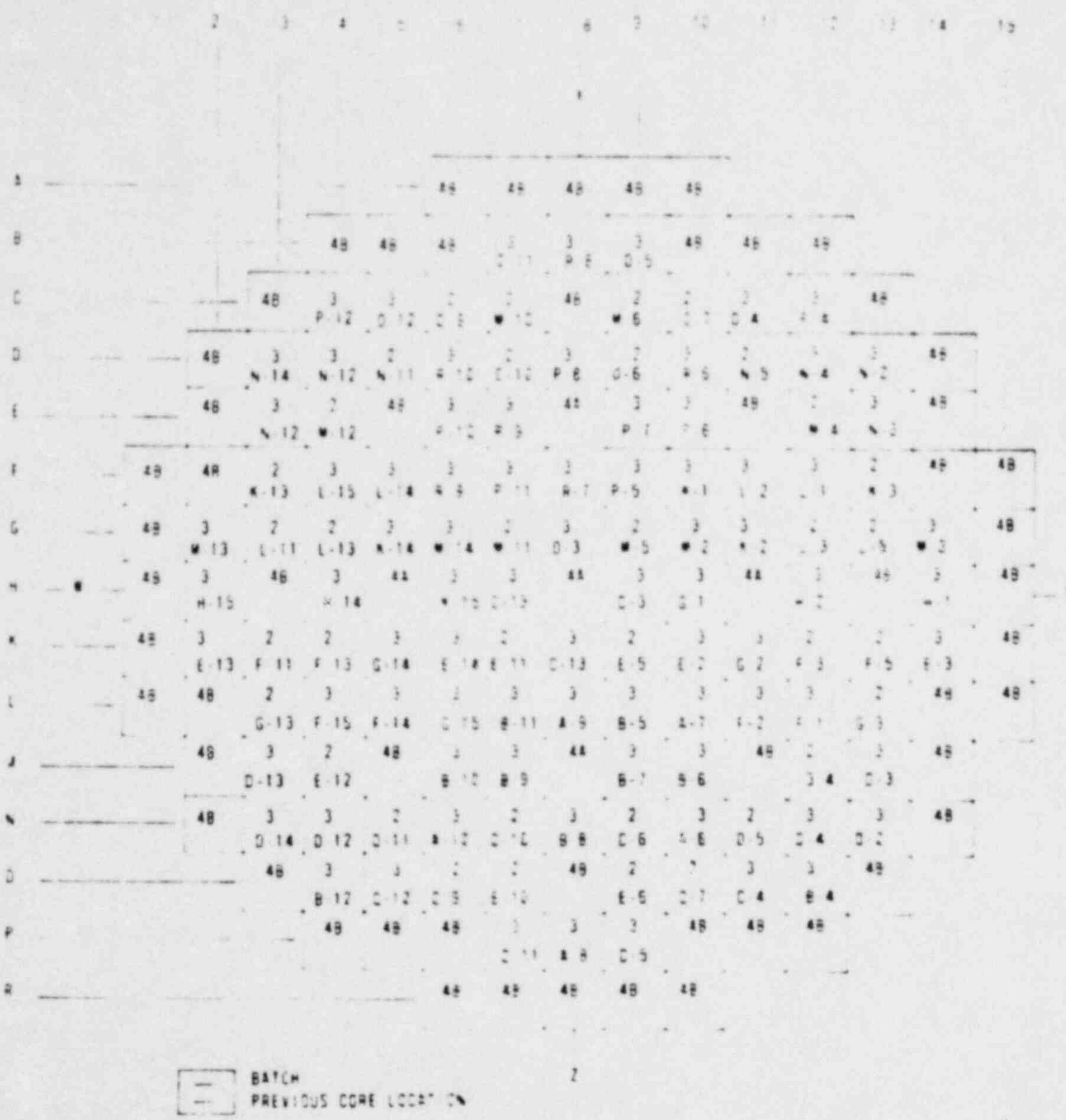
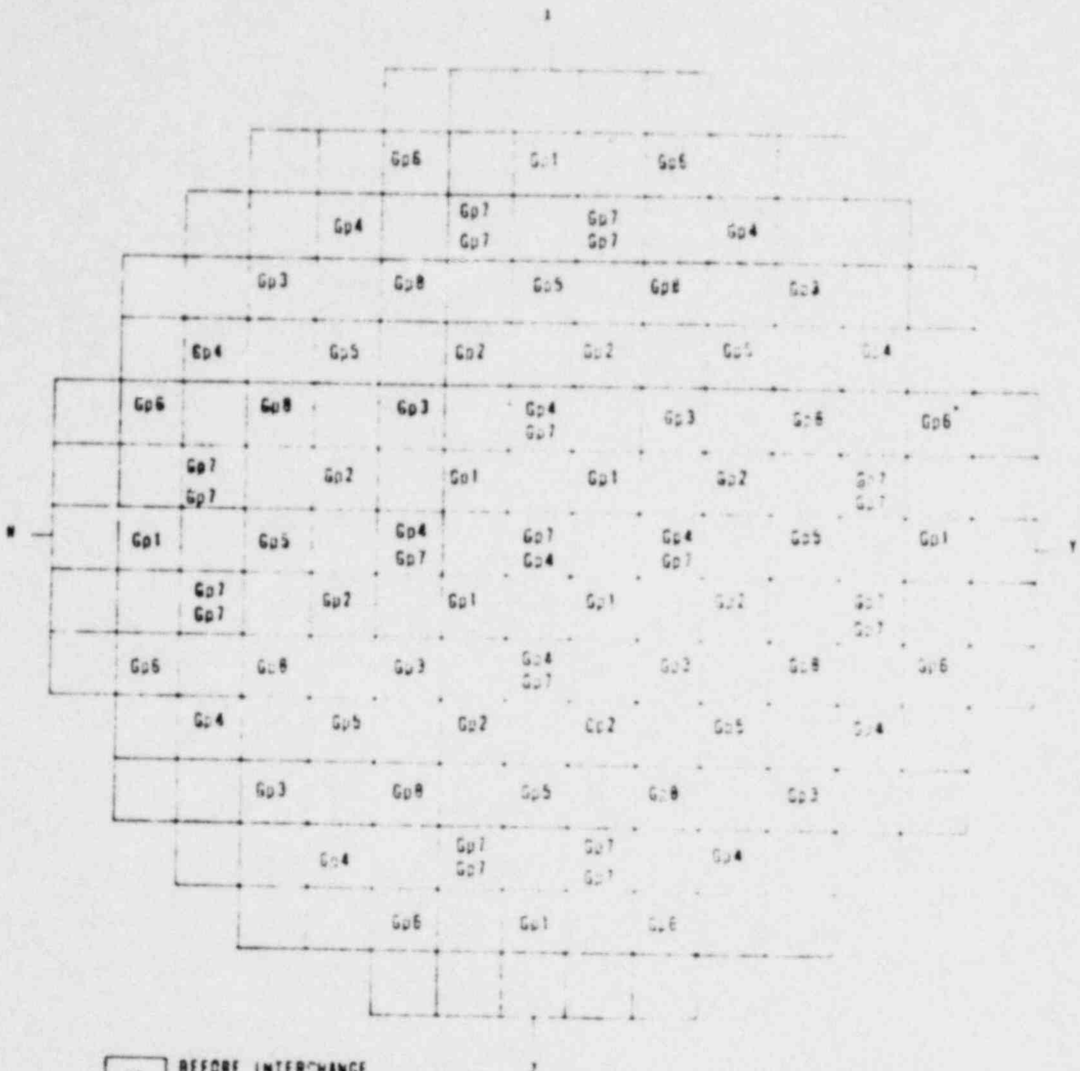


Figure 2-2. Cycle 2 - Control Rod Locations



BEFORE INTERCHANGE
AFTER INTERCHANGE

GROUP	BEFORE INTERCHANGE	AFTER INTERCHANGE
1	0	8
2	8	8
3	6	8
4	12	5
5	8	8
6	8	8
7	9	12
8	8	8
TOTAL	61	61

Figure 2-3. BOC, Cycle 2 Two-Dimensional Relative Power Distribution - Full Power, Equilibrium Xenon, Normal Rod Positions (Groups 7 and 8 Inserted)

7 .88	.93	1.05	1.19	.83	1.05	.96	1.16
.93	.96	1.00	.94	.75	7 .63	.93	1.14
1.05	1.00	1.02	.99	8 .85	.84	1.40	1.04
1.19	.94	.99	1.47	1.00	1.02	1.30	
.83	.75	8 .85	1.00	1.03	.94	.93	
1.05	[illegible] .63	.84	1.02	.94	.95		
.96	.93	1.40	1.30	.93			
1.16	1.11	1.04					

3. PERFORMANCE ANALYSIS

3.1. Thermal-Hydraulic Analysis

3.1.1. Design Limitations

DNBR – Design limitations for cycle 2 operation were 1.55 (BAW-2) minimum DNBR at reference design conditions with vent valves closed and 1.32 (BAW-2) minimum DNBR on the P-T envelope with one vent valve open.

Central Fuel Melt – Limitations on the linear heat rate were established utilizing full fuel densification penalties. This results in a minimum linear heat rate capability of 20.15 kW/ft for unrestricted loading.

Pressure-Temperature (P-T) Envelope – Modifications to the P-T envelope were required to meet the design DNBR limitations for the BAW-2 correlation under the assumed worst-case densified conditions.

3.1.2. Power Spike Model

The power spike model utilized in this analysis is identical to that presented in BAW-[illegible] except for two modifications. The modifications have been applied to F_g and F_k. These probabilities have been changed to reflect additional data from operating reactors that support a somewhat different approach and yield less severe penalties due to power spikes. F_g was changed from 1.0 to 0.5. F_k was changed from a Guassian distribution to a linear distribution, which reflects a decreasing frequency with increasing gap size.

The maximum gap size versus axial position is shown in Figure 3-1, and the power spike factor versus axial position is shown in Figure 3-2. These figures also show the initial and final theoretical densities (TDI, TDF) used in the calculations.

3.1.3. DNBR Analysis

The DNBR analysis utilized the B&W-2 CHF correlation and the actual operating primary system flow. The B&W-2 CHF correlation is a more realistic prediction of the "burnout" phenomena and has been reviewed and approved for use with the Mark-B fuel assembly design (BAW-10000[6]). The actual primary system flow (107.6% of design flow) has been verified as noted in reference 5. Utilization of the B&W-2 CHF correlation and the actual primary system flow provide a more accurate prediction of DNBR margin in the core.

In addition to the maximum design conditions considered in the FSAR, the effects of fuel densification on DNBR were taken into account. As input to the DNBR analysis for the batch 4 fuel, the minimum lot average density and the densified as-built stack height were used. Using this input and the corresponding power spike, the most limiting DNBR conditions were calculated.

The axial flux shape that gave the maximum DNBR change from the original design value was an outlet peak with a core offset of +11.8%.[1,2] The spike magnitude and the maximum gap size used in the analysis are 1.07 and 1.96 inches, respectively. These two effects results in -5.4 and -3.0% changes in minimum hot channel DNBR and peaking margin, respectively.

In addition, all fuel assembly water channel spacings were measured and each assembly was evaluated as to its DNBR capability. Any penalties incurred were accounted for in the thermal, nuclear, and safety analyses.

3.1.4. Fuel Temperatures

The minimum capability with respect to linear heat rate of the fuel is 20.15 kW/ft. The basis for the analysis utilized is given in BAW-10055[1] and BAW-10044[8] with the following additional modifications:

1. In the equiaxed zone, 3% porosity is assumed but is not used in the calculation. That is, the input value for fuel density is used and therefore no credit is taken in the calculation for increased thermal conductivity of UO_2 for the higher fuel density.

2. The option in the code for no restructuring of fuel has been used in the analysis presented here in accordance with AEC's interim evaluation of TAFY.[5]

3. The calculated gap conductance is reduced by 25% by the code, also in accordance with AEC's interim evaluation of TAFY.[5]

All fuel lots were inspected for average and LTL density and diameter values. Each lot was then evaluated as to its limiting linear heat rate in accordance with reference 4. As a result, three assemblies will be selectively loaded as described in section 2.4.

3.1.5. 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. In addition, the power spike is assumed to be located at the hot channel position that minimizes the DNBR. The resultant 5.4% DNBR loss, or 3.0% reduction in power peaking margin, will be compensated by changes in the Technical Specifications, so that the plant can function at rated power without violating the initial design criteria for DNBR and/or fuel melting.

Table 3-1 compares thermal-hydraulic operating conditions for cycles 2 and 1.

3.2. Nuclear Analysis

The RPS power/imbalance limits (DNBR and centerline fuel melt protection) and the operational limits (administrative LOCA kW/ft controls) have been established for cycle 2 operation according to the methods and procedures described in BAW-10079[3] and BAW-1388[2]. Following is a summary of cycle 2 design parameters utilized in the analysis:

Parameter	Cycle 2 value
Fuel melt limit, kW/ft	20.15
DNB peaking margin penalty due to densification, %	-3.0
Overpower, % of 2568 MWt	112
Densified nominal heat rate at 100% power, kW/ft	5.80
Power spike factor	Figure 3-1
Nuclear power peaking uncertainty	1.075
LOCA limit, kW/ft	Figure 3-8

The power peaks resulting from cycle 2 operation were examined. The plant can operate at rated power without exceeding DNBR, fuel melt, and ECCS criteria by adhering to the limits specified in Figures 3-3 through 3-7.

3.3. Safety Analysis

3.3.1. General Safety Analysis

The significant effects of fuel densification were identified, and the effects on the safety analysis were reported in BAW-1388[2]. This detailed analysis showed that no safety margins that previously existed were jeopardized by the postulated effects of densification.

It was further established that the limiting transients were the rod ejection accident and the loss of coolant flow. Table 2-2 shows that the ejected rod worth for cycle 2 (0.32%) will be much less than the rod worth used in BAW-1388(0.50%). In addition, the moderator and Doppler coefficients of reactivity are more favorable than those used in the previous analysis. Therefore, it can be concluded that the rod ejection accident will result in conditions no more severe than previously reported. The loss-of-coolant-flow type accidents (LOCAs) will be less severe than previously reported since the initial DNBR will be higher.

As shown in Table 3-1, the initial DNBR at the overpower of 114% of rated power for cycle 2, batch 4, is much higher using the measured flow of 107.6% and the BAW-2 correlation.[5,6] Thus, the transient results for cycle 2 fuel will be less severe than or equal to the results reported previously.

The peaking values are consistent with the discussion presented in section 3 of BAW-1388[2].

3.3.2. LOCA Analysis

A generic LOCA analysis for B&W 177-fuel assembly nuclear steam systems with lowered steam generators has been performed using the Final Acceptance Criteria ECCS Evaluation Model and is reported in BAW-10091[7]. That analysis is generic in nature since the limiting values of key parameters for all plants in this category were used. Thus, the analysis provides conservative results for operation of Oconee 1.

Portions of the LOCA analysis have been repeated for Oconee 1 using specific parameters associated with the Oconee 1 plant and cycle 2 fuel. The purpose of this re-analysis was to reduce some of the over-conservatisms from the generic analysis. Only the 2- and 4-foot elevations relative to the bottom of the core were reexamined.

At the 2-foot elevation, an explicit evaluation of the LOCA limiting peak linear heat rate to maintain the peak cladding temperature below 2200F for fuel batches 2, 3, and 4 (which constitute cycle 2 fuel) was performed. The results show that the limits are 16.0, 16.0, and 17.0 kW/ft for batches 2, 3, and 4, respectively.

At the 4-foot elevation, fuel batches 2 and 3 were evaluated for a peak linear heat rate of 17.5 kW/ft; the results show that the peak cladding temperature is less than 2200F. Batch 4 was not examined at 4 feet since the stored energy in the fuel for this batch is less than that for batches 2 and 3 and therefore would result in a lower peak cladding temperature than for batches 2 and 3 at the same linear heat rate.

Figure 3.3-8 shows bounding curves for allowable LOCA peak linear heat rates for cycle 2 fuel, i.e., fuel batches 2 and 3 and the reload batch, 4.

3.4. Mechanical Analysis

Cladding strain due to fuel pellet irradiation growth was calculated according to the procedures outlined in B&W topical report, BAW-10055[1]. Results for the batch 2 and 3 assemblies remaining in the core for the cycle 2 have been reported in BAW-138-[-]. These results indicated that cladding strain after three cycles was less than the 1% strain limit. Analyses of the batch 4 fuel through three cycles of operation resulted in a cladding strain of 0.52%, which is well within the 1% strain limit. Input to the batch 4 analysis included applicable as-built data on cladding and fuel pellet dimensions and a peak pellet burnup of 42,181 MWd/mtU. Conservatisms in this analysis included a minimum pellet-to-cladding gap, a maximum pellet density greater than 96.5%, and pellet thermal expansion equal to that at the fuel-melt power limit.

Predictions of cladding collapse into an axial gap were made using the CROV computer code described in the B&W topical report BAW-10084.[1-] The analysis assumed that densification could occur in two ways — at the beginning of life or over a 2400-hour period. The second assumption produced higher cladding temperatures over the 2400-hour period and the most conservative results. Other conservatisms used in the analysis included minimum cladding wall thickness, maximum initial ovality, a decrease in the prepressure level at BOL, no fission gas release, and severe burnup and radial peak histories over each of the cycles. The predicted values for time to collapse as a result of this analysis for batches 2 and 3 were reported in the letter report, "Oconee 1 Cladding Collapse," August 1974. The most conservative results (densification occurring in 2400 hours) indicated a time to collapse of 25,650 hours as opposed to a three-cycle residence time of 21,500 hours. The results for batch 4 predict that collapse will occur at 29,990 EFPH of operation. This corresponds with a projected burnup of 37,800 MWd/mtU, whereas the expected three-cycle burnup and exposure time for batch 4 is 27,751 MWd/mtU and 21,500 EFPH.

Table 3-1. Operating Conditions

	Cycle 1	Cycle 2
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % FSAR design flow	100.0	107.6
Ref design radial-local power peaking factor	1.78	1.78
Ref design axial flux shape	1.5 cosine	1.5 cosine
CHF correlation	W-3	B&W-2
Minimum DNBR (max design conditions, 114% power, no densification effects	1.55	2.05
Mechanical hot channel factor on enthalpy rise (F_q)	1.011	1.011
Mechanical hot channel factor on local surface heat flux (F_q'')	1.014	1.014
Densification penalty, %		
DNBR margin	-4.5	-5.4
Power peaking margin	-2.6	-3.0
Power spike at outlet	12.5	7.1
Linear heat rate to central fuel melting (Class I fuel, AEC restrictions), kW/ft	20.15	20.15

Figure 3-1. Maximum Gap Size Vs Axial Position

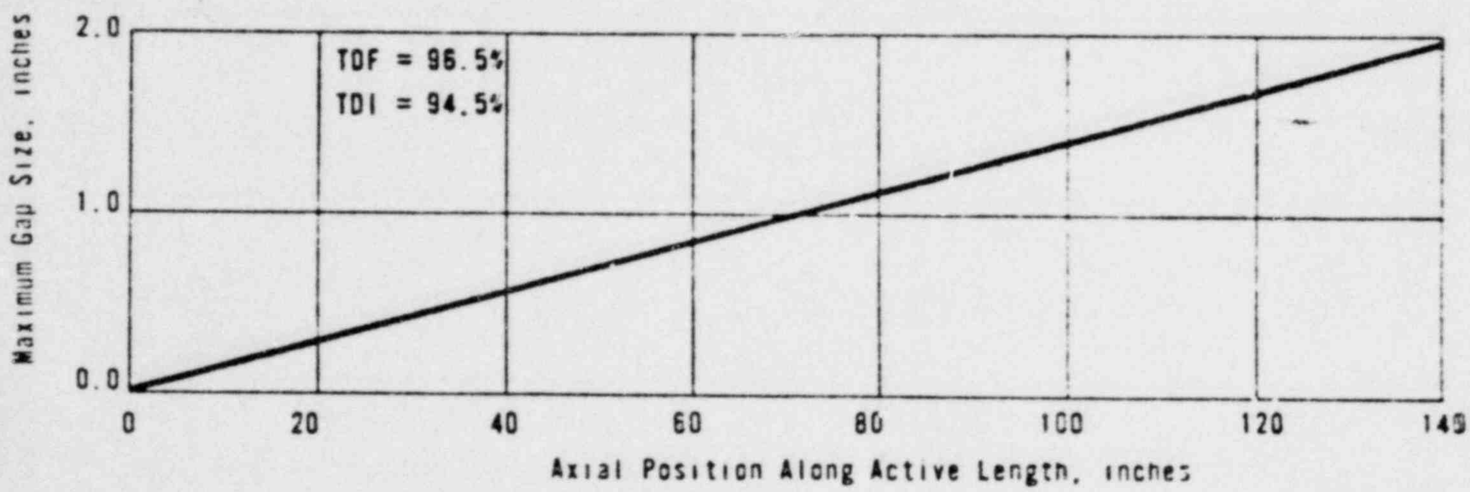


Figure 3-2. Power Spike Factor Vs Axial Position

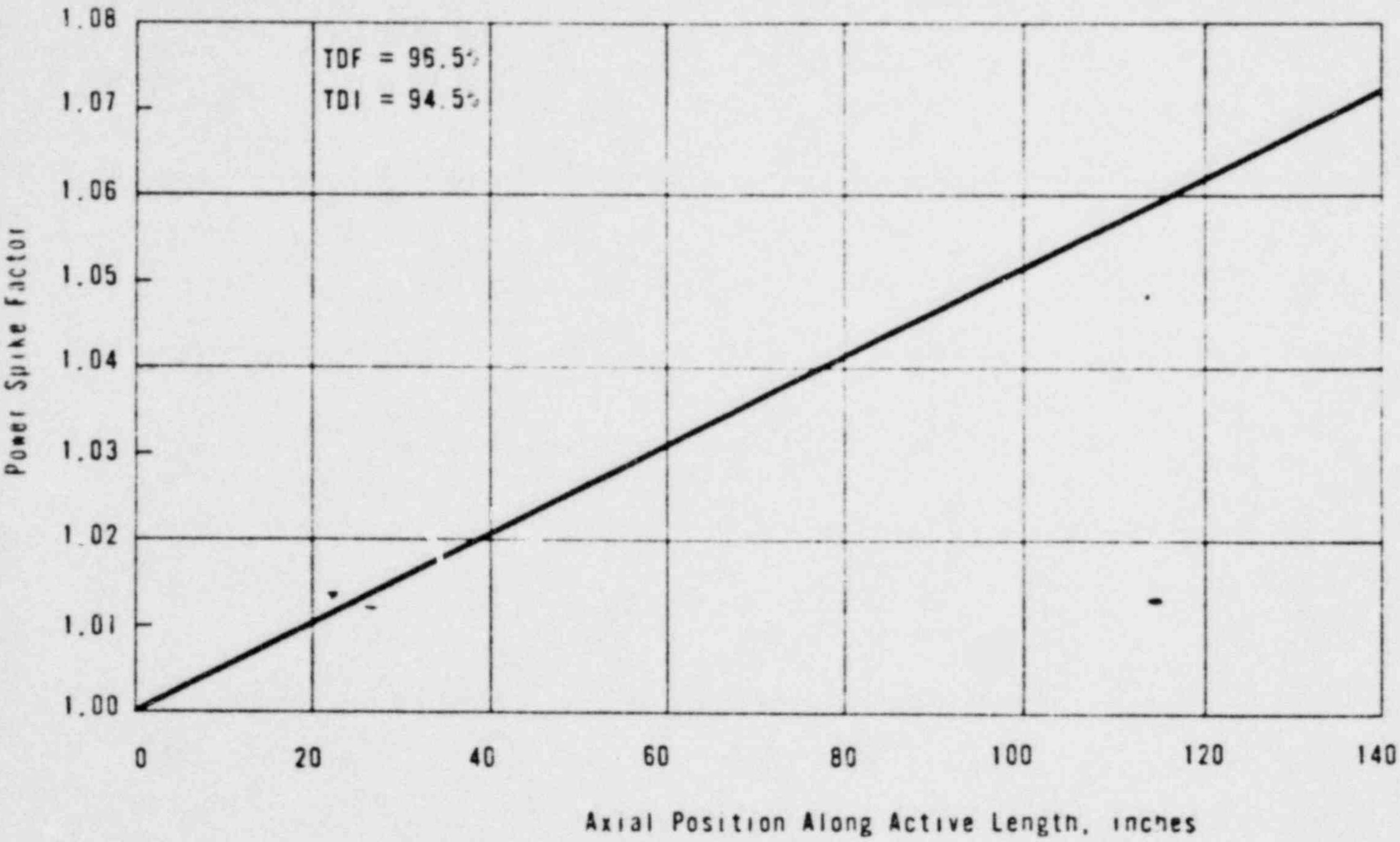
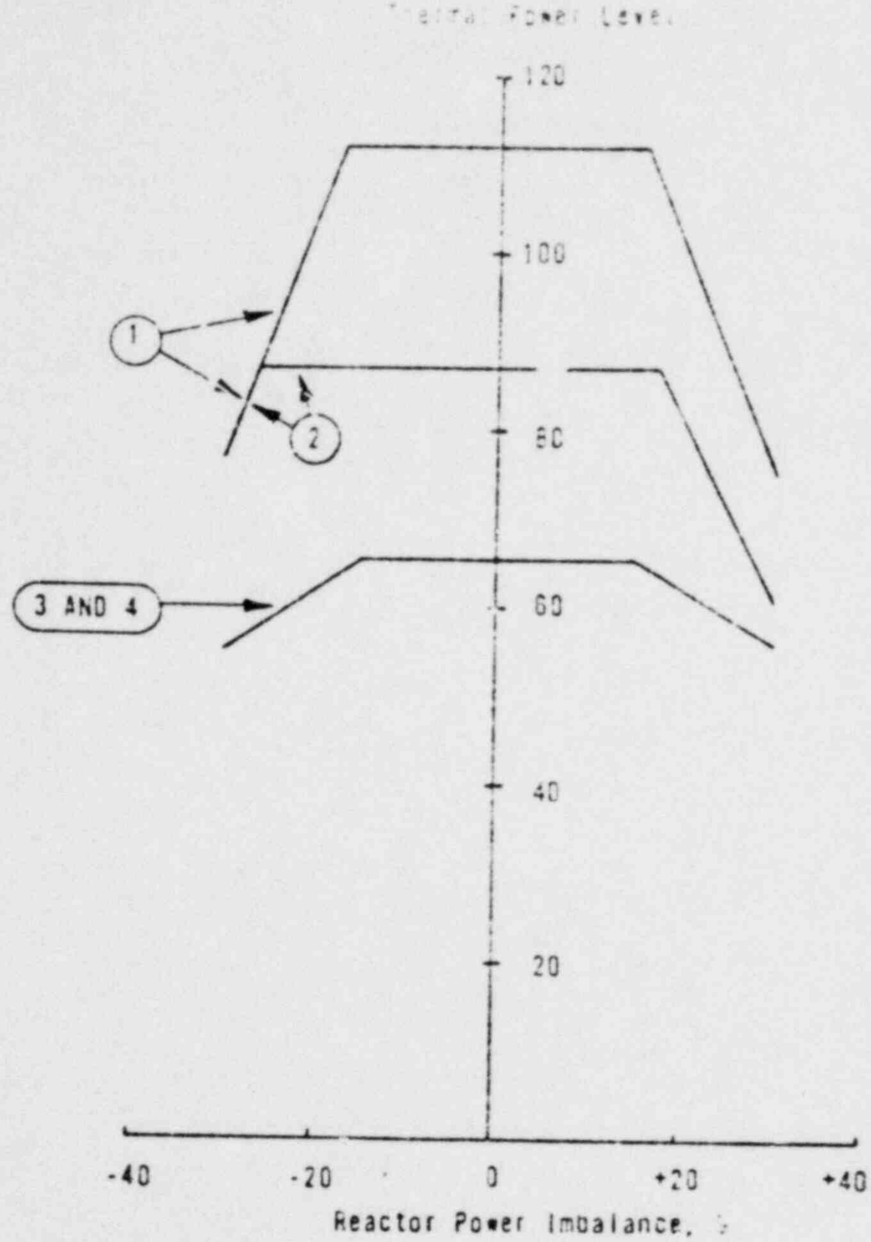


Figure 3-[illegible]. Core Protection Safety Limits



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

Figure 3-4. Operational [illegible] Imbalance Envelope

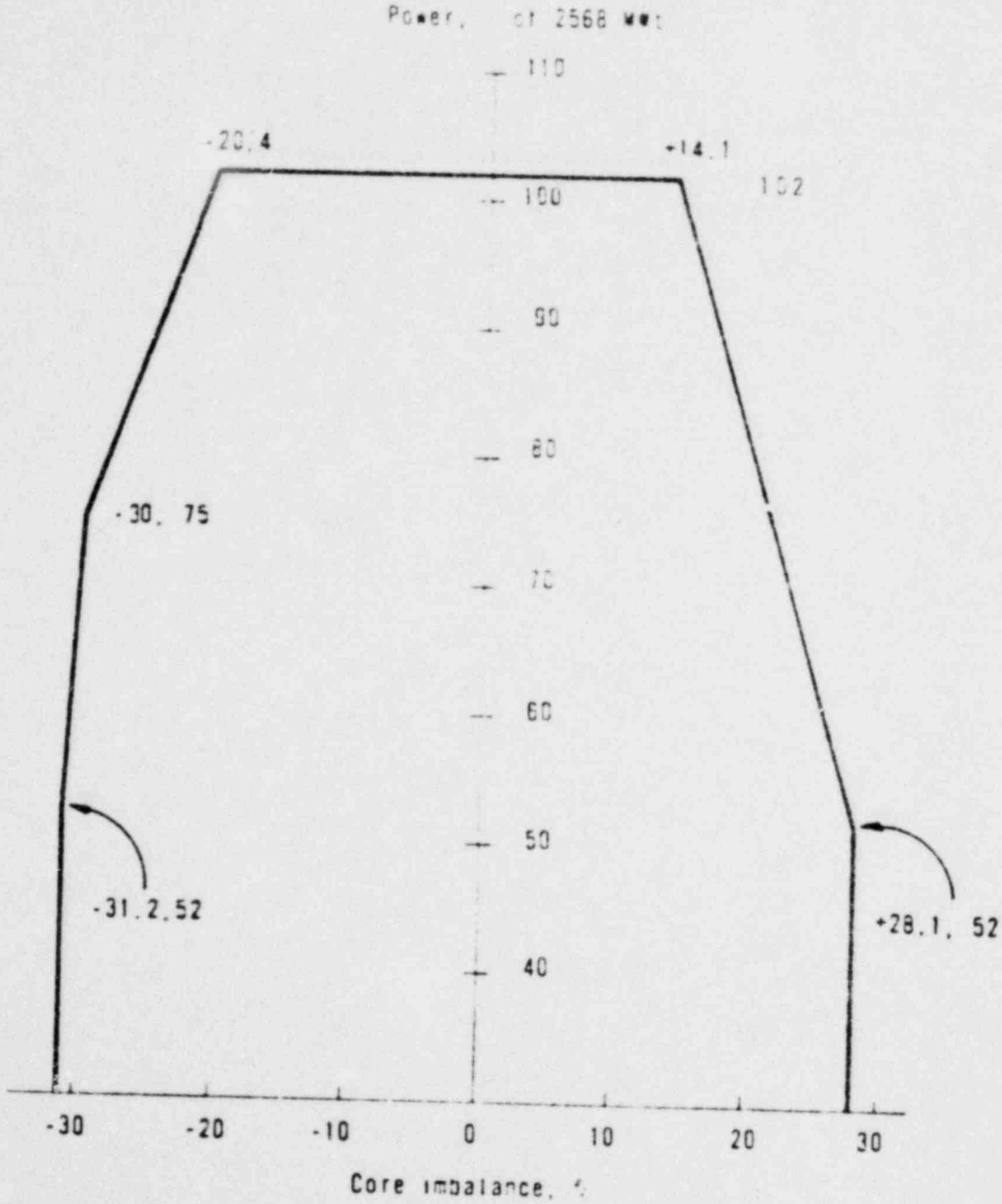


Figure 3-5. Control Rod [illegible] Operation

1. Rod index is the percentage sum of the withdrawn [illegible] groups.
2. The withdrawal limits are [illegible] operation.

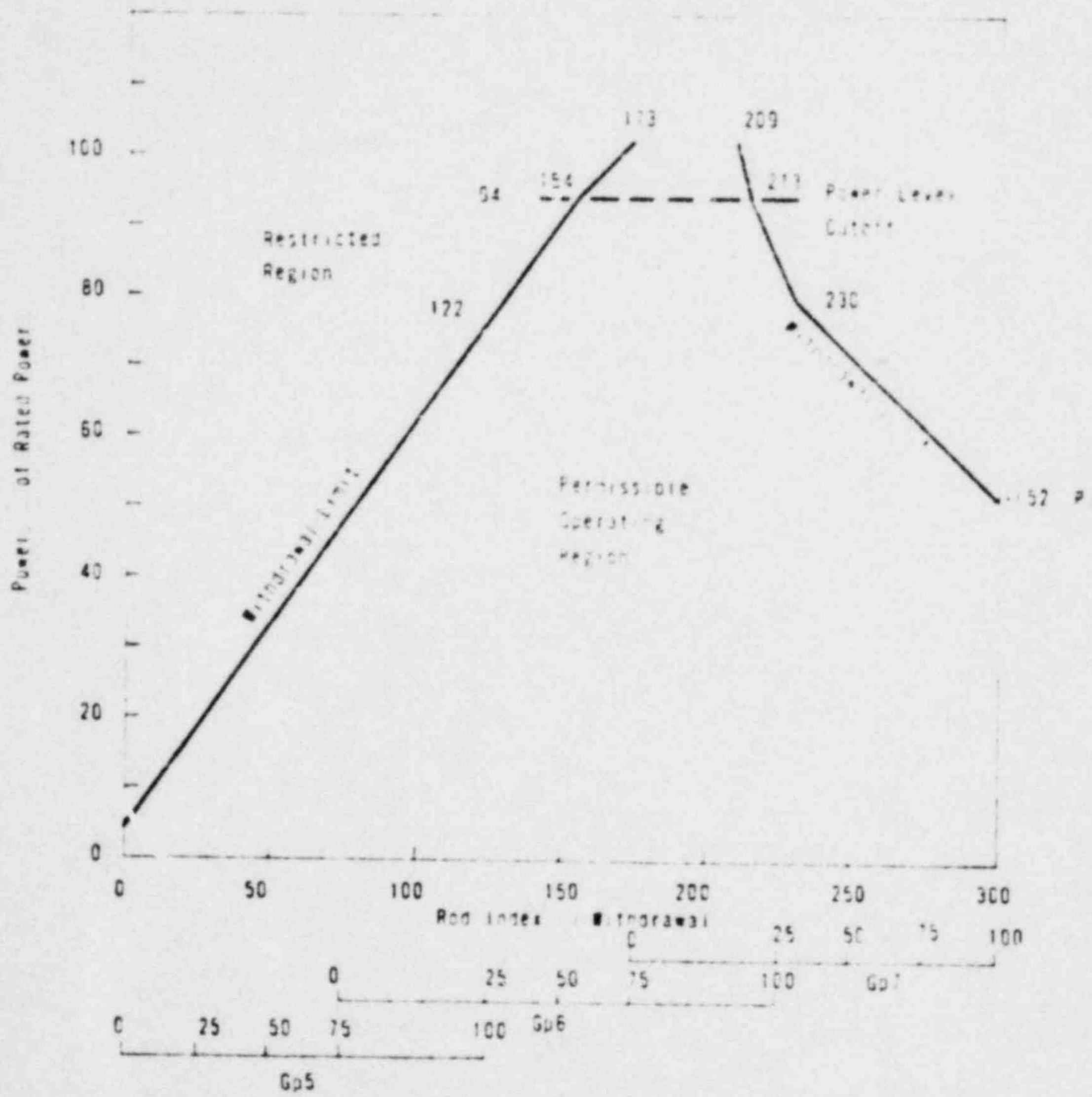


Figure 3-6. Control Rod Group Withdrawal Limits for Four-Pump Operation

1. Rod Index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are in effect after 250 ± 5 [illegible] days of operation. The applicable power level cutoff is 100 [illegible]

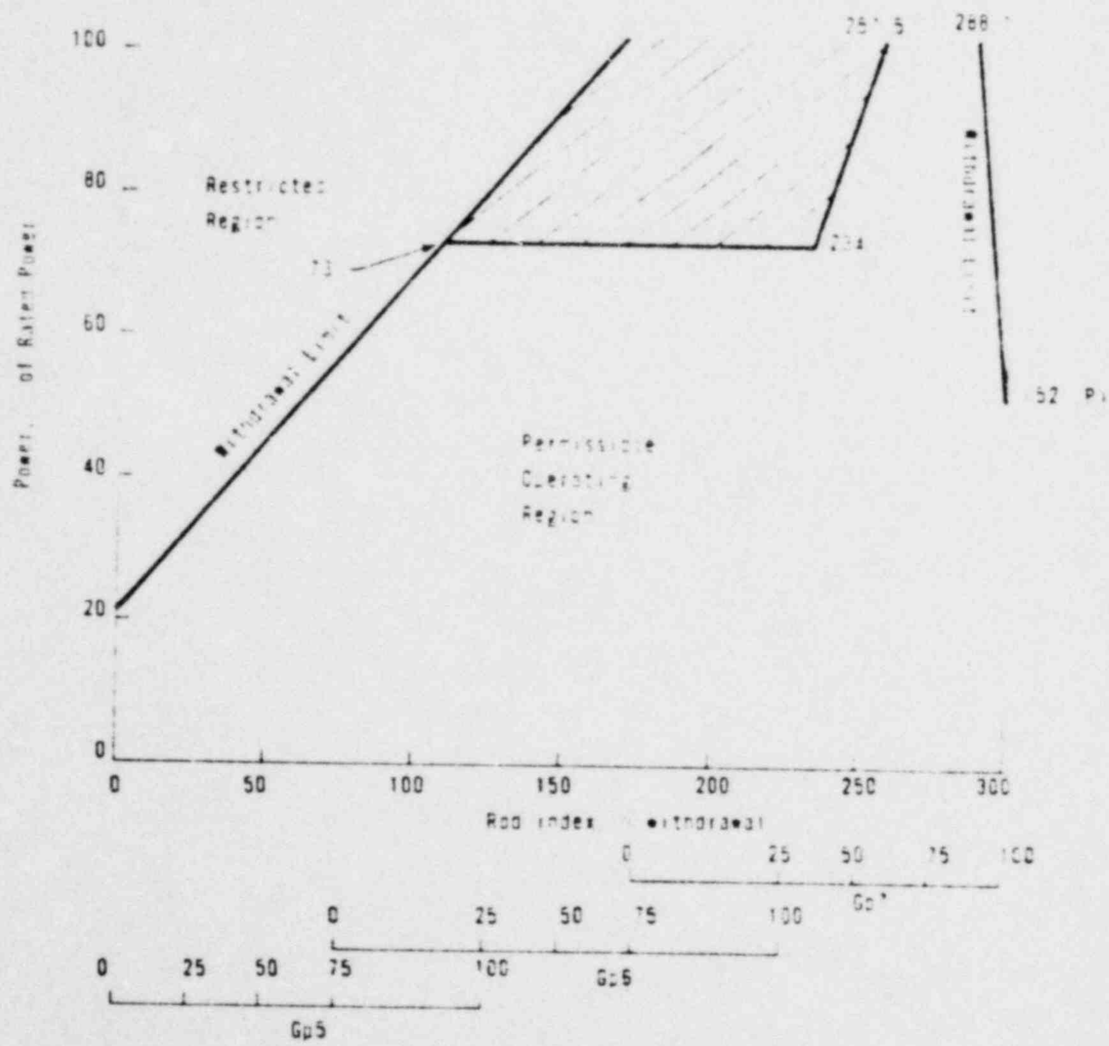


Figure 3-7. Control Rod Group [illegible]

1. Rod index is the percentage sum of the withdrawal of the operating groups.

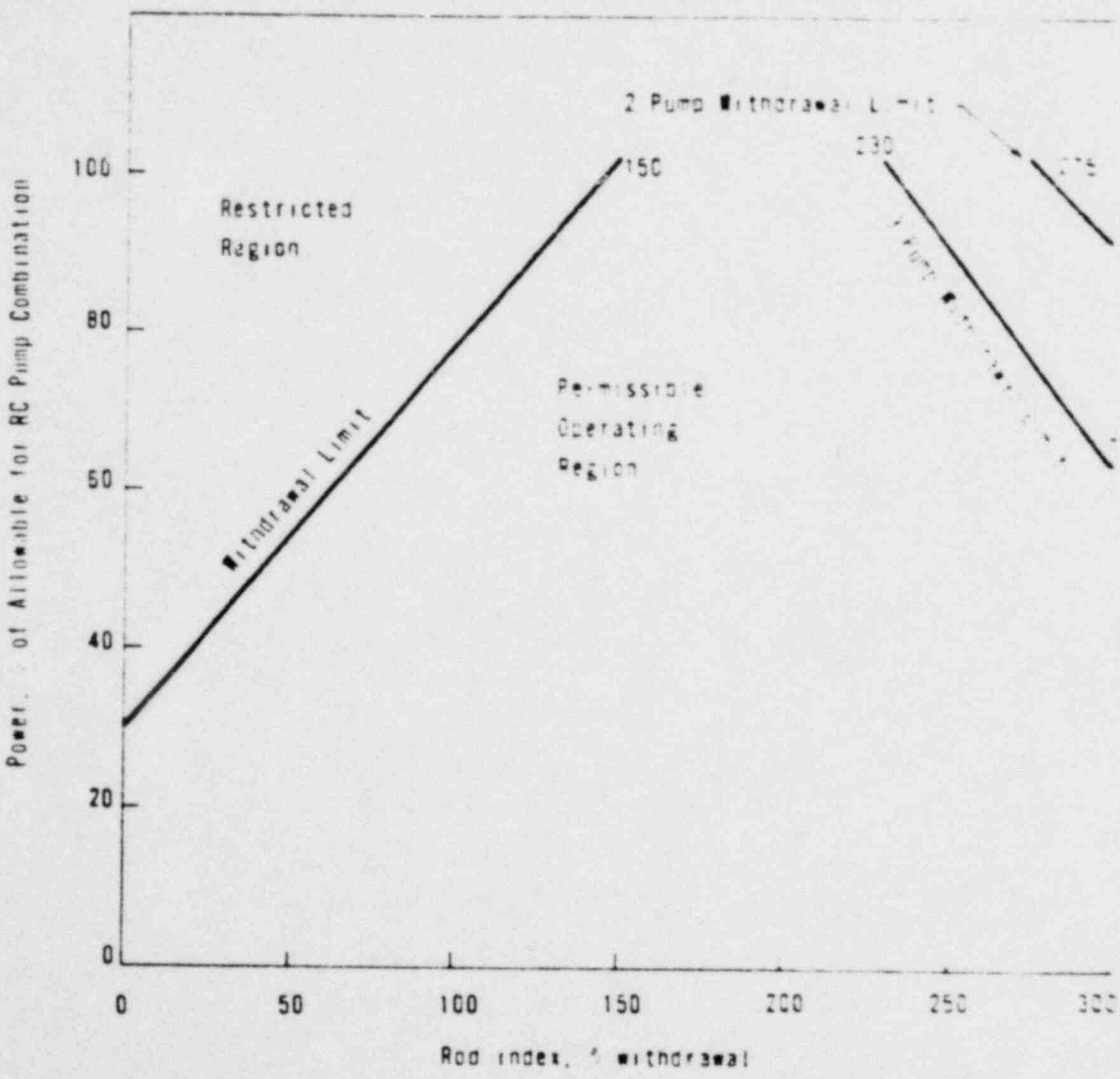
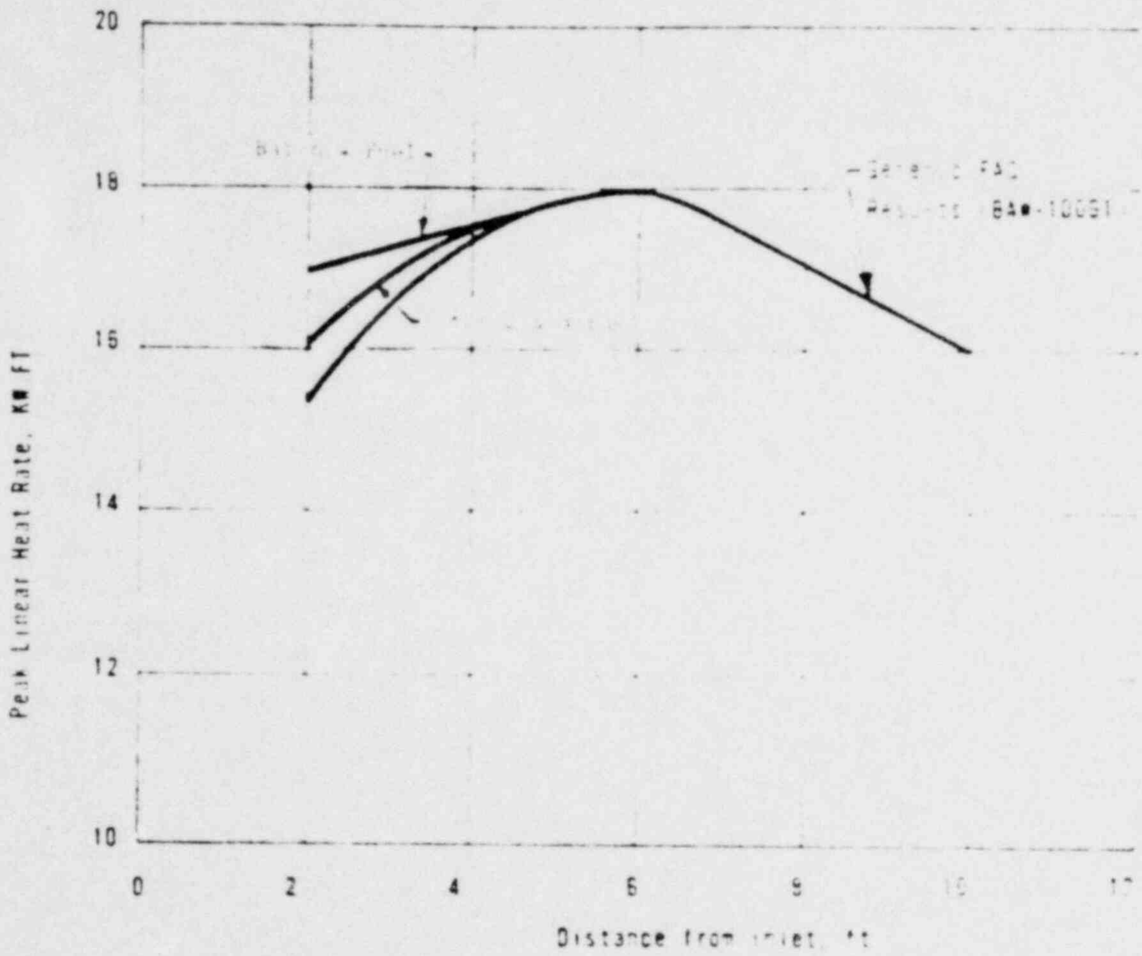


Figure [illegible]. Allowable [illegible]



4. CONCLUSIONS

The analysis documented in this report was performed using penalties for fuel densification and methods and procedures that have been accepted by the USAEC. Cycle 2 operation at rated power has been shown to be consistent with the thermal design criteria and LOCA kW/ft limits. An analysis of cladding creep-collapse performance in accordance with reference 8 has shown that no cladding collapse will occur during three cycles of operation.

Based upon the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, and the thermal design criteria will not be violated.

5. REFERENCES

1. Fuel Densification Report, BAW-10055, Rev. 1, Babcock & Wilcox, July 1973.

2. Oconee 1 Fuel Densification Report, BAW-1387, Rev. 1, Babcock & Wilcox, July 1973.

3. Operational Parameters for B&W Rodded Plants, BAW-10079, Babcock & Wilcox, October 1973.

4. Letter Report: "Classification and Selective Loading of Fuel for Rancho Seco," Babcock & Wilcox, December 1973, (Proprietary).

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