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RETURN TO REGULATORY CENTRAL FILES
ROOM 016

OCONEE UNIT 3, CYCLE 3

- Reload Report -

RETURN TO REGULATORY CENTRAL FILES
ROOM 016

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the third cycle of Oconee Nuclear Station, Unit 3 at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975.

To support cycle 3 operation of Oconee Unit 3, this report employs analytical techniques and design bases established in reports that were previously submitted and accepted by the USNRC and its predecessor (see references).

A brief summary of cycle 2 and 3 reactor parameters related to power capability is included in section 5 of this report. All of the accidents analyzed in the FSAR have been reviewed for cycle 3 operation. In those cases where cycle 3 characteristics proved to be conservative with respect to those analyzed for cycle 2, no new analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 3 operation are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems, it has been concluded that Oconee Unit 3, Cycle 3, can be safely operated at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses of the Oconee Nuclear Station, Unit 3, is the currently operating cycle 2. Cycle 1 was terminated after 478 EFPD of operation. Cycle 2 achieved initial criticality on November 7, 1976, and power escalation commenced on November 10, 1976. The 100% power level of 2568 MWt was reached on November 21, 1976. The fuel cycle design length is 282 EFPD. No operating anomalies occurred during cycle 2 operation that would adversely affect fuel performance in cycle 3.

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3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core is described in detail in Chapter 3 of the Unit 3 FSAR.[1] The cycle 3 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 rod 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 0.370 inch in diameter. (See Table 4-1 for additional data.) All fuel assemblies in cycle 3 maintain a constant nominal fuel loading of 463.6 kg of uranium. The undensified nominal active fuel lengths and theoretical densities vary between batches, however, and these values are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 3, cycle 3. All of the batch 2 assemblies will be discharged at the end of cycle 2. Five once-burned batch 1 assemblies, with an initial enrichment of 2.01 wt % ^{235}U, will be reloaded into the central portion of the core. Batches 3, 4, and 4A — with initial enrichments of 3.00, 2.53, and 2.64 wt % ^{235}U, respectively — will be shuffled to new locations. Batch 5, with an initial enrichment of 3.02 wt % ^{235}U, will occupy primarily the core periphery and eight interior locations. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 3.

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 axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 3 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the total pattern (69 control rods) for cycle 3 are identical to those of the reference cycle indicated in Chapter 3 of the FSAR.[1] However, the group designations differ between cycle 3 and the reference cycle to minimize power peaking. Neither control rod interchange nor burnable poison rods are necessary for cycle 3.

Figure 3-1. Core Loading Diagram for Oconee 3, Cycle 3

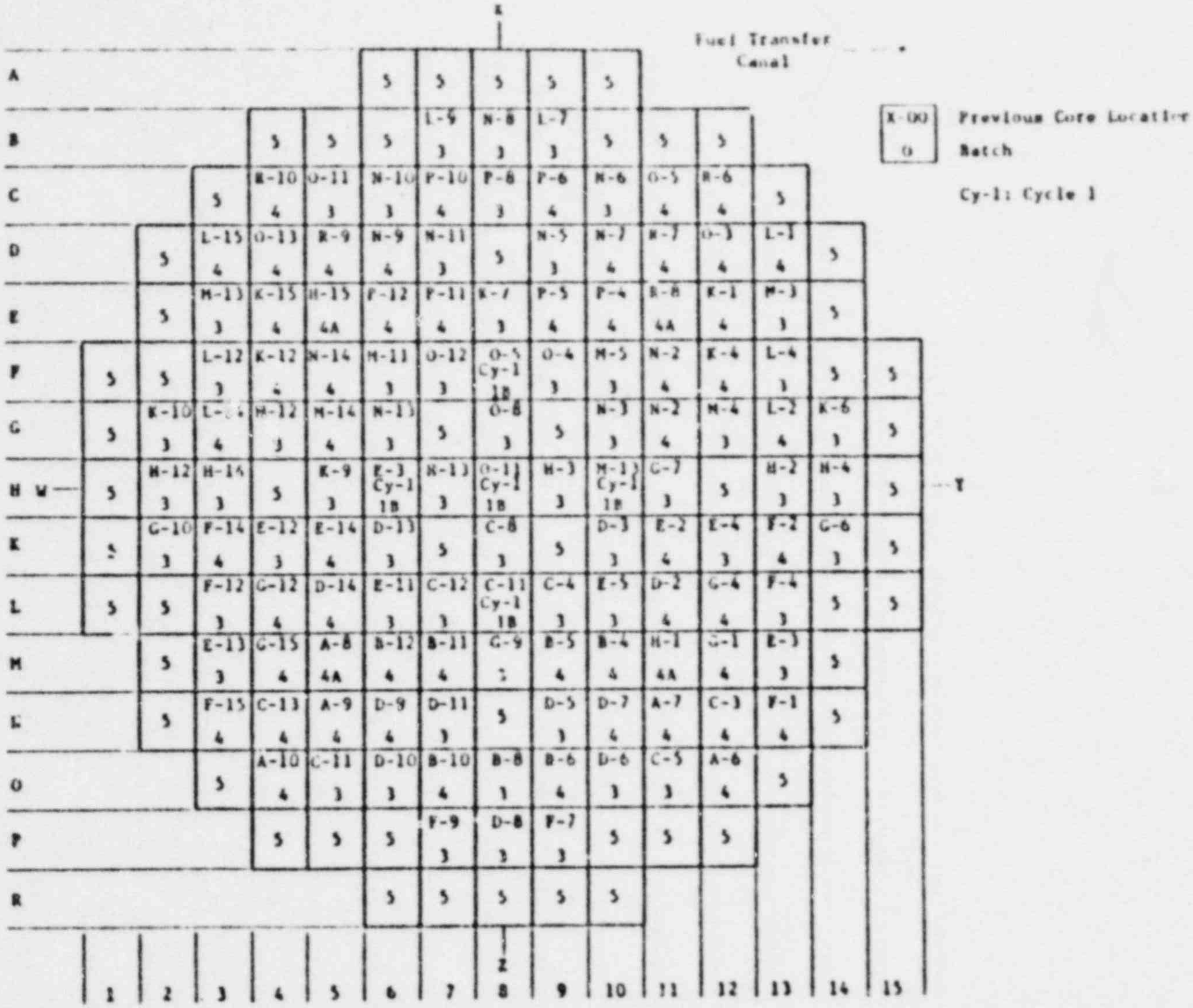
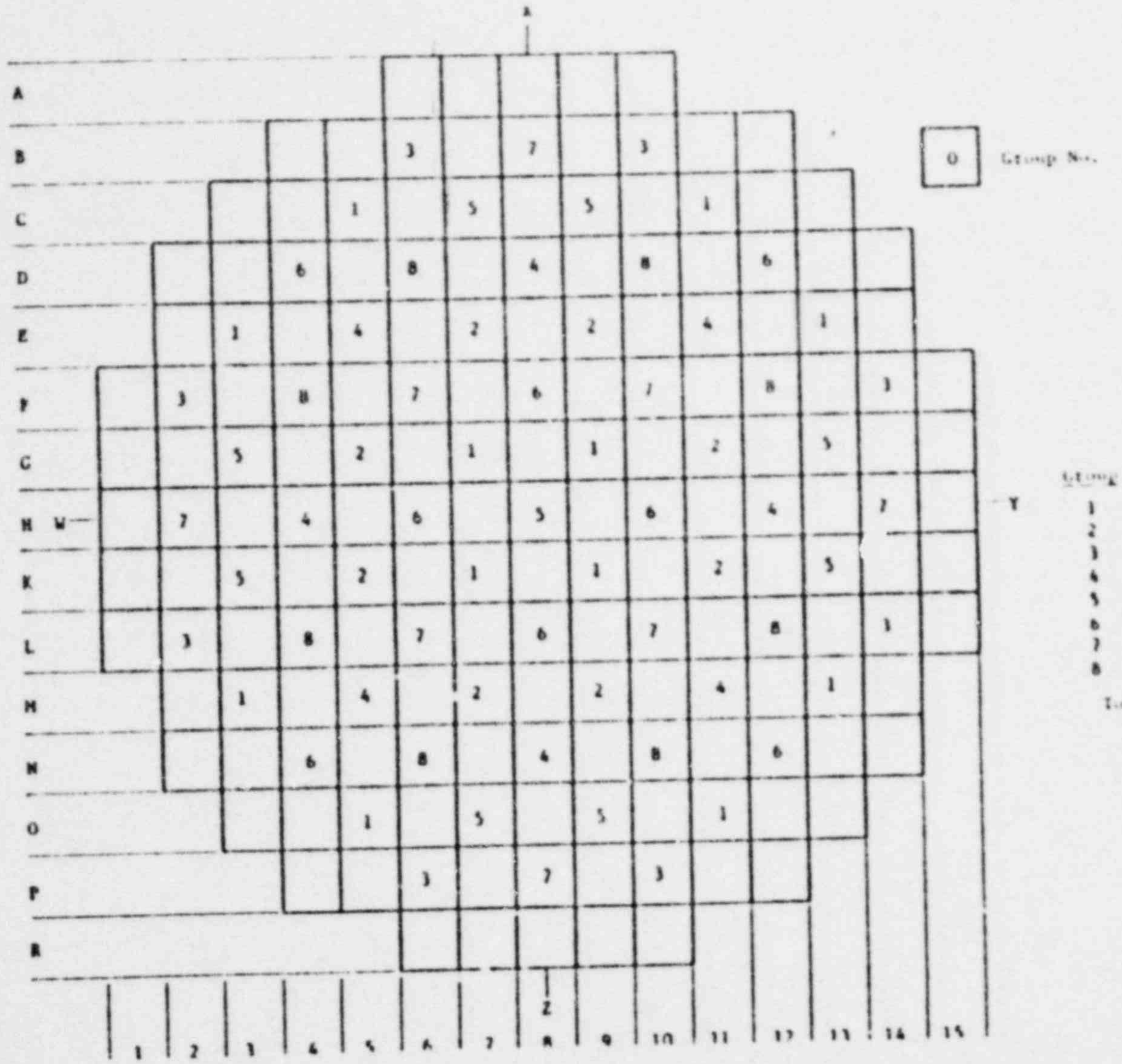


Figure 3-2. Enrichment and Burnup Distribution for
Oconee 3, Cycle 3

	8	9	10	11	12	13	14	15
H	2.01 14,215	3.00 24,660	2.01 14,215	3.00 23,483	3.02 0	3.00 20,623	3.00 23,480	3.02 0
K		3.02 0	3.00 18,391	2.53 8,718	3.00 27,527	2.53 9,557	3.00 24,151	3.02 0
L			3.00 20,426	2.53 6,194	2.53 11,043	3.00 21,958	3.02 0	3.02 0
M				2.64 7,043	2.53 6,684	3.00 19,880	3.02 0	
N					2.53 6,694	2.53 5,683	3.02 0	
O						3.02 0		
P								
R								

0.00	Initial Enrichment
00,000	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Oconee 3, Cycle 3



Group	No. of Rods	Function
1	12	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	9	Control
6	8	Control
7	8	Control
8	8	APSRs
Total	69	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters and dimensions for Oconee 3, cycle 3 are listed in Table 4-1. The fresh fuel assemblies (batch 5) incorporate minor design modifications to the spacer grid corner cells to reduce spacer grid interaction during handling. In addition, improved test methods (dynamic impact testing) show that the spacer grids have a higher seismic capability and thus an increased safety margin over the values reported in reference 3.

All other results, references, and identified conservatisms presented in the previous Oconee 3 reload report[2] (section 4.1) are applicable to the cycle 3 reload core.

4.2. Fuel Rod Design

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories. Batches 3 and 4 were analyzed using as-built data. The batch 3 fuel is more limiting for cladding collapse due to its previous incore exposure time.

The assembly power history for the most limiting assembly was used to calculate the fast neutron flux level for the energy range above 1 MeV. The collapse time for the most limiting assembly was conservatively determined to be more than 30,000 EFPH (effective full-power hours), which is longer than the maximum three-cycle design lives (Table 4-1). The creep collapse analyses were performed based on the conditions set forth in references 2 and 4.

4.2.2. Cladding Stress

The Oconee 3 stress parameters are enveloped by a conservative fuel rod stress analysis. For design evaluation, the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses must be less than the minimum specified unirradiated yield strength. In all cases, the margin is in excess of 30%. The following conservatisms with respect to Oconee 3 fuel were used in the analysis:

1. A lower post-densification internal pressure.
2. ' lower initial pellet density.
3. A higher system pressure.
4. A higher thermal gradient across the cladding.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic circumferential strain. The pellet design is established for plastic cladding strain of less than 1% at values of maximum design local pellet burnup and heat generation rate, which are considerably higher than the values the Oconee 3 fuel is expected to see. This will result in an even greater margin than the analysis demonstrated. The strain analysis is also based on the maximum Specification value for the fuel pellet diameter and density and the lowest permitted Specification tolerance for the cladding ID.

4.3. Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 5 fuel inserted for cycle 3 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The design minimum linear heat rate (LHR) capability is 20.15 kW/ft, as shown in Table 4-2. LHR capabilities are based on centerline fuel melt and were established using the TAFY-3 code[5] with fuel densification to 96.5% of theoretical density.

4.3.1. Power Spike Model (Densification)

The power spike model used for cycle 3 analysis is the same as that used for cycle 2.[2] Figures 4-1 and 4-2 show the maximum gap size and power spike factor, respectively, versus axial position. The power spike factor and gap size were based on unirradiated batch 4 and 5 fuel (94.0% TD) with an assumed enrichment of 3.0 wt % ^{235}U. These values are conservatively high for batch 1 and 3 fuel.

4.3.2. Fuel Temperature Analysis

Thermal analysis of the fuel rods assumed in-reactor densification to 96.5% theoretical density. The analytical methods utilized are the same as those documented in references 2 and 6 for cycle 2. The average fuel temperatures shown in Table 4-2 are taken from the analyses used to define the LHR capability for the fuel.[2,6] These analyses were based on the lower tolerance limit of the specification fuel density and assumed isotropic diametral shrinkage and anisotropic axial shrinkage (consistent with reference 7) resulting from fuel densification.

4.4. Material Design

The batch 5 fuel assemblies are not new in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant assembly interactions for the batch 5 fuel assemblies are identical to those of the present fuel.

4.5. Operating Experience

B&W's operating experience with the Mark B, 15 by 15 fuel assembly design has verified the adequacy of this design. As of April 30, 1977, the following operating experience has been accumulated for the seven B&W 177-fuel assembly plants using the Mark B fuel assembly:

Reactor	Current cycle	Max assembly burnup, MWd/mtU	Cumulative net electrical output, MWh
Oconee 1	3	25,400	16,742,549
Oconee 2	2	25,900	12,919,680
Oconee 3	2	23,400	12,130,627
TMI-1	2	26,200	13,306,085
Arkansas One	2	20,700	9,826,476
Rancho Seco	1	15,400	6,040,979
Crystal River 3	1	1,000	575,364

Table 4-1. Fuel Design Parameters and Dimensions

	Twice-burned FAs, batch 3	Once-burned FAs			Fresh FAs, batch 5
		Batch 1	Batch 6	Batch 4A	
FA type	Mark B3	Mark B3	Mark B4	Mark B4	Mark B4
No. of FAs	60	5	52	4	56
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377
Flex. spacers, type	Spring	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Undensif active fuel length (nominal), in.	142.0	142.0	142.23	142.23	142.25
Fuel pellet initial density (nom), % TD	95.5[(a)]	95.5[(a)]	94.0	94.0	94.0
Fuel pellet OD (mean specif), in.	0.3680[(a)]	0.3680[(a)]	0.3695	0.3695	0.3695
Initial fuel enrich, wt % ^{235}U	3.00	2.01	2.53	2.64	3.02
BOC burnup (avg), MWd/mtU	21,766	14,320	7,881	7,043	0
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>30,000	>30,000
Design life, EFPH	24,888	18,120	20,928	20,928	>21,144

[(a)] Nominal values after resintering.

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 1	Batch 3	Batch 4	Batch 5
No. of assemblies	5	60	56	56
Initial density, % TD	95.3	95.5[a]	94.0	94.0
Pellet diameter, in.	0.3682	0.3680[a]	0.3695	0.3695
Stack height, in.	141.0	141.0[a]	142.2	142.2
Densified Fuel Parameters[b]				
Pellet diameter, in.	0.3649	0.3649	0.3646	0.3646
Fuel stack height, in.	140.2	140.3	140.5	140.5
Nominal LHR at 2568 MWt, kW/ft	5.80	5.80	5.80	5.80
Avg fuel temp at nominal LHR, F	1310	1305	1320	1320
LHR capability (centerline fuel melt), kW/ft	20.15	20.15	20.15	20.15

[a] Nominal values after resintering.

[b] Densification to 96. [illegible] TD assumed.

Figure 4-1. Maximum Gap Size Vs Axial Position -- Oconee 3, Cycle 3

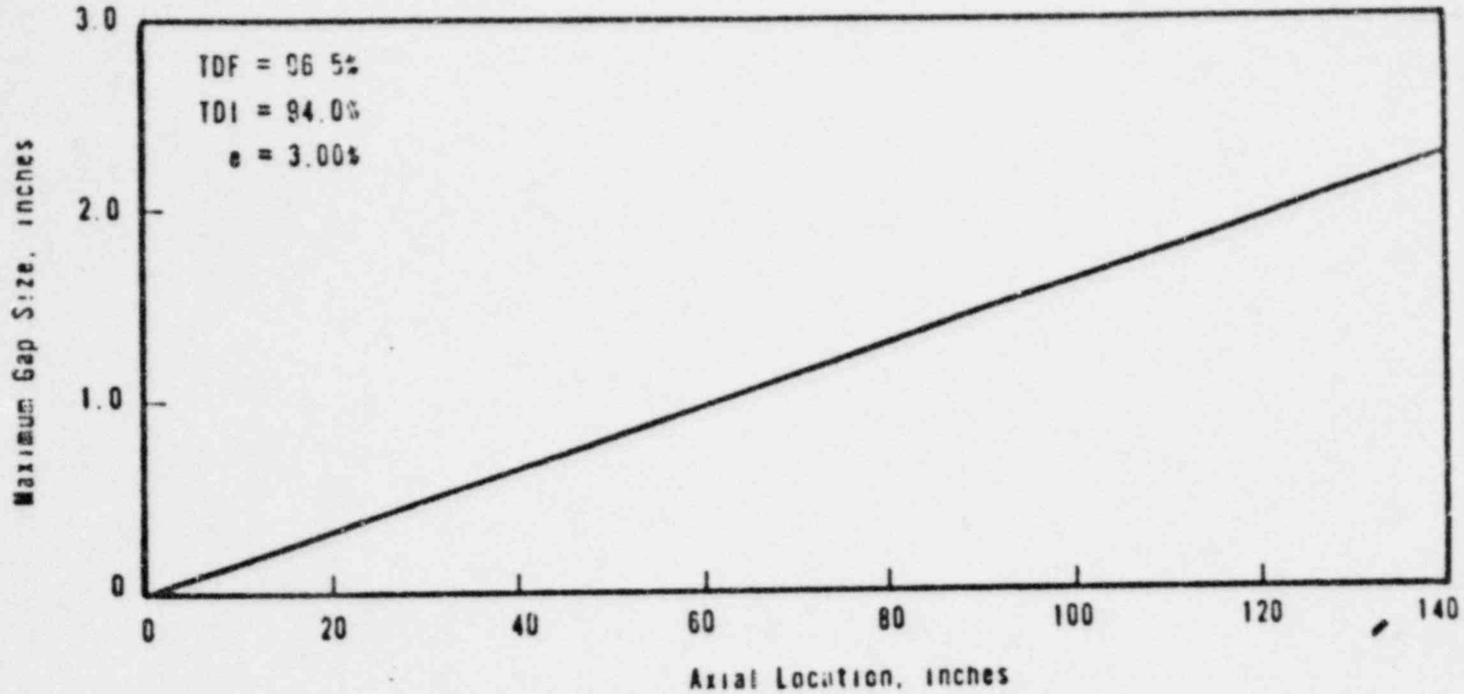
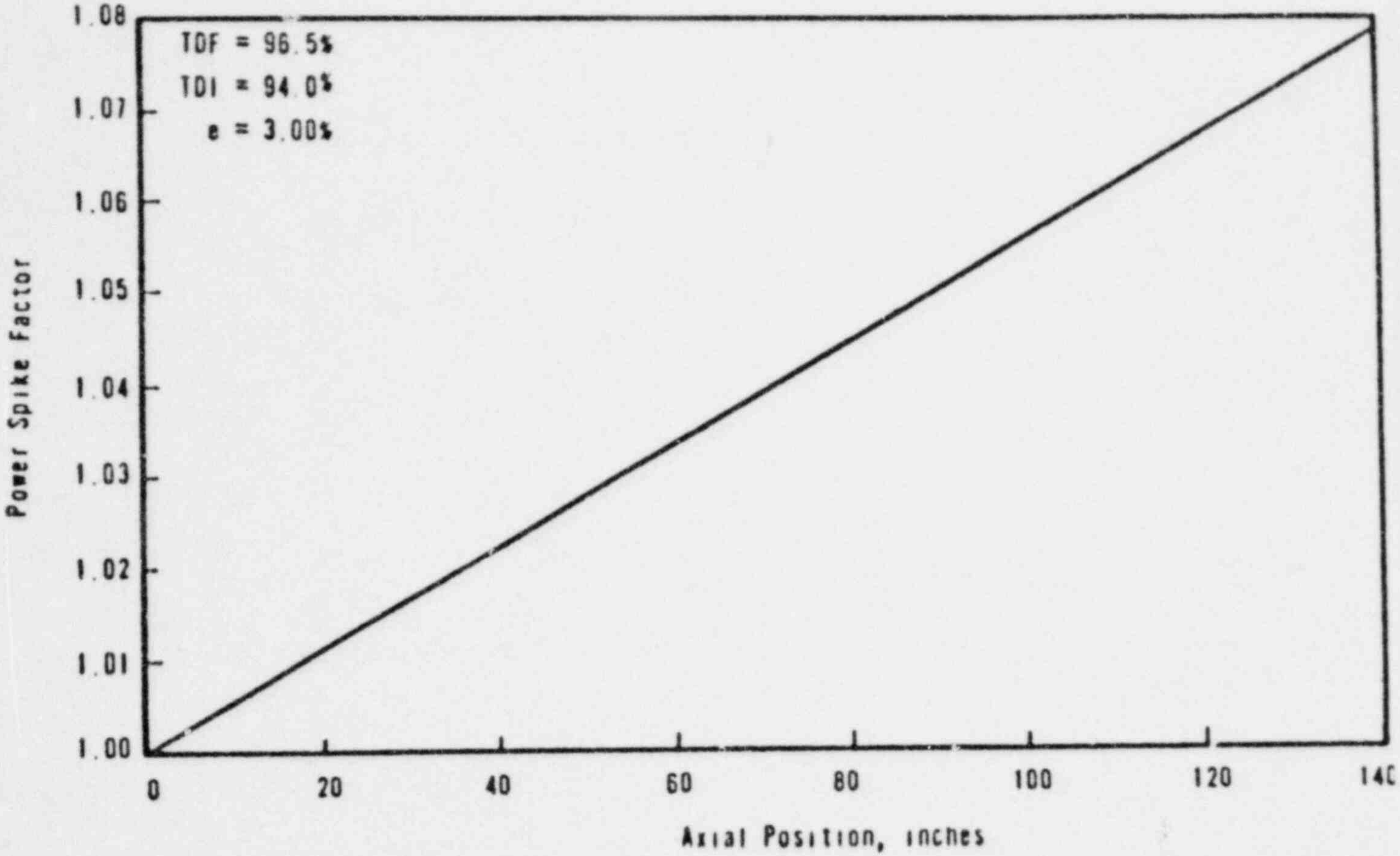


Figure 4-2. Power Spike Factor Vs Axial Position — Oconee 3, Cycle [illegible]



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5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 2 and 3; the values for both cycles were generated using PDQ07. 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 cycle 3. The accumulated average core burnup will be higher in cycle 3 than in cycle 2 because of the presence of the once-burned batch 1, 3, 4, and 4A fuel. Figure 5-1 illustrates a representative relative power distribution for the beginning of the third cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 3 are higher than for cycle 2 because of a higher feed enrichment, different radial power distribution, etc. As indicated in Table 5-2, the control rod worths are sufficient to maintain the required shutdown margin. However, due to changes in isotopics and the radial flux distribution, the BOC hot, full-power control rod worths are generally less than those for cycle 2. The cycle 3 ejected rod worths are lower than those in cycle 2 for the same number of regulating banks inserted. It is difficult to compare values betw en cycles or between rod patterns since neither the rod patterns from which the CRA is assumed to be ejected nor the isotopic distributions are identical. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod insertion limits presented in section 8. The maximum stuck rod worths for cycle 3 are less than those in cycle 2. The adequacy of the shutdown margin with cycle 3 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. Flux redistribution penalty.

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of cycle 3 is analyzed at approximately 235 EFPD. This is the latest time (± 10 days) in core life at which the transient bank is nearly fully inserted. After 235 EFPD, the transient bank will be almost fully withdrawn, thus increasing the available shutdown margin. The reference fuel cycle shutdown margin is presented in reference 2, Table 5-1.

The cycle 3 power deficits from hot zero power to hot full power are similar to but slightly higher than those for cycle 2. Doppler coefficients, moderator coefficients, and xenon worths are similar for the two cycles. The differential boron worths for cycle 3 are lower than for cycle 2 due to depletion of the fuel and the associated buildup of fission products. The effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2. Analytical Input

The cycle 3 incore measurement calculation constants used to compute core power distributions were prepared in the same manner as for the reference cycle.

5.3. Changes in Nuclear Design

The same calculational methods and design information were used to obtain the important nuclear design parameters for cycles 2 and 3. In addition, there are no significant operational procedure changes from the reference cycle with regard to axial or radial power shape control, xenon control, or tilt control. The operational limits (Technical Specification changes) for the reload cycle are shown in section 8.

A fuel melt limit of 20.15 kW/ft has been employed in calculating the reactor protection system setpoints and is the same as in cycles 1 and 2. The batch 5 fuel assemblies will be loaded as shown in Figure 3-1. Two batch 5 assemblies have been assigned a maximum linear power rating of 19.74 kW/ft based on as-built data. These assemblies will be placed in non-limiting locations during their entire core residence. For cycle 3, investigation has determined that if these assemblies are placed in locations M-14 and E-2, they will not experience linear power rates higher than 19.15 kW/ft. Thus, as shown in Table 5-3 for various times during the nominal fuel cycle, the margin to fuel melt will always be greater in these assemblies than in the most limiting assembly in the core.

Table 5-1. Oconee 3, Cycle 2 and 3 Physics Parameters

	Cycle 2	Cycle 3
Cycle length, EFPD	265	277
Cycle burnup, MWd/mtU	8293	8668
Average core burnup, EOC, MWd/mtU	18,160	18,921
Initial core loading, mtU	82.1	82.1
Critical boron — BOC, ppm (no Xe)		
HZP, group 8 inserted	1251[a]	1261
HZP, groups 7 and 8 inserted	1108	1188
HFP, groups 7 and 8 inserted	931	1000
Critical boron — EOC, ppm (eq Xe)		
HZP, group 8 37.5% wd, eq Xe	328[a]	288
HFP, group 8 37.5% wd, eq Xe	29	35
Control rod worths — HFP, BOC, % $\Delta k/k$		
Group 6	1.18	1.08
Group 7	0.97	0.77
Group 8 37.5% wd	0.54	0.40
Control rod worths — HFP, 235 EFPD, % $\Delta k/k$		
Group 7	1.29[a]	1.05
Group 8 37.5% wd	0.49[a]	0.44
Max ejected rod worth — HZP, % $\Delta k/k$		
BOC, groups 5-8 inserted	0.66	0.73
235 EFPD, groups 5-8 inserted	0.60[a]	0.61
Max stuck rod worth — HZP, % $\Delta k/k$		
BOC	2.30	2.54
235 EFPD	2.18[a]	2.24
Power deficit, HZP to HFP, % $\Delta k/k$		
BOC	1.65	1.55
235 EFPD	2.[illegible][a]	1.98
Doppler coeff — BOC, 10^{-5} ($\Delta k/k/°F$)		
100% power (0 Xe)	-1.54	-1.43
Doppler coeff — EOC, 10^{-5} ($\Delta k/k/°F$)		
100% power (eq Xe)	-1.54	-1.56
Moderator coeff — HFP, 10^{-4} ($\Delta k/k/°F$)		
BOC (0 Xe, 1150 ppm, group 8 inserted)	-1.06[a]	-0.53
EOC (eq Xe, 17 ppm, group 8 inserted)	-2.39	-2.55
Boron worth — HFP, ppm/% $\Delta k/k$		
BOC (1050 ppm)	107[a]	105
EOC (17 ppm)	101	95

Table 5-1. (Cont'd)

	Cycle 2	Cycle 3
Xenon worth — HFP, % Δk/k		
BOC (4 days)	2.64	2.66
EOC (equilibrium)	2.68	2.75
Effective delayed neutron fraction — HFP		
BOC	0.00585	0.00594
EOC	0.00520	0.00522

(a) For conditions applicable to these values, refer to BAW-1432.[2]

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 3

	BOC, % Δk/k	EOC, % Δk/k (a)
Available Rod Worth		
Total rod worth, HZP	8.74	8.70
Worth red'n due to burnup of poison	-0.24	-0.31
Maximum stuck rod, HZP	-2.54	-2.24
Net worth	5.96	6.15
Less 10% uncertainty	-0.60	-0.62
Total available worth	5.36	5.53
Required Rod Worth		
Power deficit, HFP to HZP	1.55	1.98
Max allowable inserted rod worth	1.06	1.31
Flux redistribution	0.45	0.77
Total required worth	3.06	4.06
Shutdown Margin		
Total avail worth - total req'd worth	2.30	1.47

Required shutdown margin = 1.00% Δk/k

(a) For shutdown margin calculations, this is defined as about 235 EFPD, the latest time in core life at which the transient bank is nearly fully inserted.

Table 5-3. Comparison of Fuel Melt Margins for Selectively Loaded Assemblies and Limiting Assembly in Core

	Selectively loaded		Most limiting	
EFPD	Location	Margin, %	Location	Margin, %
4	M14	32.23	L14	29.47
100	M14	38.23	L14	35.23
200	M14	41.93	K9	38.14
277	M14	48.69	K9	38.31

Figure 5-1. BOC (4 EFPD) Cycle 3 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, Normal Rod Positions, Groups 7 and 8 Inserted

	8	9	10	11	12	13	14	15
H	0.892	1.004	0.958	0.989	1.407	0.867	7 0.439	0.675
K	1.004	1.380	0.967	1.054	0.980	0.955	0.770	0.767
L	0.859	0.967	7 0.624	1.111	8 0.974	0.982	1.268	0.751
M	0.991	1.056	1.1[illegible]	1.337	1.301	1.085	1.139	
N	1.409	0.982	0.978	1.307	1.361	1.147	0.813	
O	0.868	0.957	0.984	1.087	1.148	0.920		
P	7 0.440	0.771	1.269	1.140	0.814			
R	0.676	0.768	0.751					

0 0.000	Inserted Rod Group No. Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

6.1. Evaluation

The thermal-hydraulic design evaluation in support of cycle 3 operation utilized the methods and models described in references 1, 2, and 6. Cycle 3 analyses have been based on 106.5% of the (first core) design reactor coolant (RC) system flow rate. Cycle 2 analyses[2] used 107.6% of design flow based on a measured flow value of 110.0%. The reduced flow rate has been selected for cycle 3 analyses to provide consistency with Oconee units 1 and 2.[8,9] The decreases in RC flow used for these analyses are changes in calculational parameters only and do not represent changes in operation of the plant.

The core configuration for cycle 3 differs slightly from that of cycle 2 in that the batch 2 fuel removed at the end of cycle 2 is the Mark B3 fuel assembly design, and the fresh batch 5 fuel inserted for cycle 3 is the Mark B4 assembly design. Mark B4 assemblies differ from the Mark B3 primarily in the design of the end fitting, which results in a slight reduction in flow resistance for the B4 design. No credit was taken in the analyses for the increased flow to the Mark B4 assemblies, located in the hottest core locations, as a result of slight changes in the core flow distribution or for the increase in system flow resulting from the reduction in total core pressure drop.

6.2. DNBR Analysis

The BAW-2 CHF correlation was used for thermal-hydraulic analysis of cycle 3. This correlation, which has been reviewed and approved for use with the Mark B fuel assembly design,[12] has been used previously for licensing of cycle 2 of the Oconee 3 core.[2]

The effect of fuel densification on minimum DNBR is primarily a result of the reduction in active fuel length, which increases the average heat flux. The cycle 3 DNBR analysis was based on a cold densified active length of 140.2 inches, a value selected to apply generically to a number of B&W plants. This is a conservative method of applying the densification effect since all the fuel assemblies in cycle 3 have longer densified lengths (Table 4-2) and because no credit is taken for axial thermal expansion of the fuel column. This analysis differs from that of cycle 2 in two respects: First, the effect of the densification power spike is no longer considered for DNBR analysis based on information presented in references 11, 12, and 13. Second, the densified active length is incorporated directly into the DNBR analysis, resulting in a calculated minimum DNBR of 1.90% at 112% power (Table 6-1). The cycle 2 analysis had been based on a 144-inch active length with the effect of a reduced active length and the densification power spike calculated separately.

The potential effect of fuel rod bow on DNBR can be considered by incorporating suitable margins into DNB-limited core safety limits and RPS setpoints. The maximum rod bow magnitude would be calculated from the equation $\sigma_b = 11.5 + 0.069 \cdot \sqrt{BU}$, where σ_b is the rod bow magnitude (in mils) and BU is the burnup (in MWd/mtU). The resultant DNBR penalty based on the maximum predicted assembly burnup at the end of cycle 3 is approximately 6.0%. However, since NRC review of this bow model had not been completed before the design of this reload core, the maximum rod bow magnitude was calculated using the NRC interim model, $\Delta C/C_o = 0.065 + 0.001449 \sqrt{BU}$, where ΔC is the rod bow magnitude (in mils) and C_o is the initial gap. The resultant DNBR penalty, based on the maximum predicted assembly burnup at EOC 3, is 11.2%.

A thermal margin credit equivalent to 1% DNBR is available as a result of the flow area (pitch) reduction factor included in all the thermal-hydraulic analyses to partially offset the projected fuel rod bow penalty. For the flux/flow trip setpoint analysis, an additional thermal margin credit equivalent to 2% excess flow has been applied. The NRC Staff has accepted, on a plant-specific basis, the use of thermal margin credits resulting from RC system flow rates in excess of that assumed for safety analyses.[9] The 2% flow credit is claimed on the basis that 106.5% of design RC flow was used for safety analysis and an RC flow of 110% of design has been proven in the plant. For those analyses performed for previous cycles that are applicable to cycle 3 or future cycles, credit will be taken (as appropriate) for the removal of the densification power spike penalty. A more specific discussion of thermal margin credits is provided in sections 6.3 and 6.4.

6.3. Pressure-Temperature Limit Analysis

The pressure-temperature limit curves shown in Figure 3-3 provide the basis for the variable low-pressure trip setpoint. The curves shown for four- and three-pump operation each represent a locus of points for which the calculated minimum DNBR is equal to 1.30 (BAW-2) plus the margin required to offset an 11.2% DNBR reduction due to rod bow. The specific credits used in this analysis to account for rod bow are as follows:

		% DNBR credit
Credit for rod bow penalty already included in analysis	=	10.2
Credit for flow area reduction factor in analysis	=	1.0
Credit for plant excess flow (3.5% available)	=	None claimed
Total		11.2

6.4. Flux/Flow Trip Setpoint Analysis

The flux/flow trip setpoint was determined by analyzing an assumed two-pump coastdown starting from an initial indicated power level of 102% plus flux measurement and heat balance errors (equal to 108% full power in core). The analytical method was the same as that used for licensing of cycle 2[2] with the following exceptions: (1) The initial system flow on which this analysis is based was reduced from 107.6% of the design flow rate to 106.5%. (2) The densification power spike penalty was deleted from the analysis. (3) Suitable margin was included for an 11.2% DNBR reduction due to rod bow. The specific credits used in this analysis to account for rod bow are as follows:

		% DNBR credit
Credit for rod bow penalty already included in analysis	=	5.8
Credit for flow area reduction factor in analysis	=	1.0
Credit for 2% excess RC flow (3.5% available)	=	4.4
Total		11.2

Table 6-1. Cycle 2 and 3 Maximum Design Conditions

	Cycle 2	Cycle 3
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	107.6	106.5
Vessel inlet/outlet coolant temp at 100% power, F	555.9/602.2	555.6/602.4
Ref design radial-local power peaking factor	1.78	1.78
Ref design axial flux shape	1.5 cosine	1.5 cosine
Hot channel factors: Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	140.2	140.2
Avg heat flux at 100% power, Btu/h-ft^2 [(a)]	175,640	175,427
Max heat flux at 100% power, Btu/h-ft^2 [(b)]	468,959	468,391
CHF correlation	BAW-2	BAW-2
Min DNBR (% power)[(c)]	1.86 (112)	1.90 (112)

[(a)] Cycle 2 heat flux was based on batch 3 densified length. Cycle 3 uses batch 4 and 5 densified length (located in hottest core location).

[(b)] Based on average heat flux with reference peaking.

[(c)] Cycle 2 DNBR included effects of densification power spike; cycle 3 does not.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 3 parameters to determine the effects of the cycle 3 reload and to ensure that thermal performance is not degraded during hypothetical transients.

The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. Cycle 1 values (FSAR values) of core thermal parameters are compared with those used in the cycle 3 analysis in Table 6-1. These parameters are common to all of the accident analyses presented herein. For each accident of the FSAR, a discussion and the key parameters are provided. A comparison of the key parameters (see Table 7-1) from the FSAR and the present cycle 3 is provided with the accident discussion to show that the initial conditions of the transient are bounded by the FSAR analysis.

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in BAW-1399.[6] Since cycle 3 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in reference 6, the conclusions derived in that reference are still valid.

Calculational techniques and methods for cycle 3 analyses remain consistent with those used for the FSAR. Additional DNBR margin is shown for cycle 3 because the B&W-2 CHF correlation was used instead of the W-3.

No new dose calculations were performed for the reload report. The dose considerations in the FSAR were based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

7.2. Rod Withdrawal Accidents

This accident is defined as an uncontrolled reactivity addition to the core due to withdrawal of control rods during startup conditions or from rated power conditions. Both types of incidents were analyzed in the FSAR.¹ The important parameters during a rod withdrawal accident are Doppler coefficient, moderator temperature coefficient, and the rate at which reactivity is added to the core. Only high-pressure and high-flux trips are accounted for in the FSAR analysis, which ignores multiple alarms, interlocks, and trips that normally preclude this type of incident. For positive reactivity additions indicative of these events, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were -1.17×10^{-5} ($\Delta k/k/°F$) for the Doppler coefficient, 0.5×10^{-4} $\Delta k/k$ for the moderator temperature coefficient and rod group worths up to and including a 10% $\Delta k/k$ rod bank worth. Comparable cycle 3 parametric values are -1.43×10^{-5} ($\Delta k/k/°F$) for the Doppler coefficient, -0.53×10^{-4} ($\Delta k/k/°F$) for the moderator temperature coefficient, and a maximum rod bank worth of 8.74% $\Delta k/k$. Therefore, cycle 3 parameters are bounded by design values assumed for the FSAR analysis. Thus, for the rod withdrawal transients, the consequences will be no more severe than those presented in the FSAR. For the rod withdrawal from rated power, the transient consequences are also less severe than those presented in the densification report.[6]

7.3. Moderator Dilution Accident

Boron in the form of boric acid is utilized to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup and transient xenon effects with dilution water supplied by the makeup and purification system. The moderator dilution transients considered are the pumping of water with zero boron concentration from the makeup tank to the RCS under conditions of full-power operation, hot shutdown, and refueling. The key parameters in this analysis are the initial boron concentration, boron reactivity worth, and moderator temperature coefficient for power cases.

For positive reactivity additions of this type, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were 1400 ppm for the initial boron concentration, 75 ppm/1% ($\Delta k/k$) boron reactivity worth and $+0.94 \times 10^{-4}$ $\Delta k/k/°F$ for the moderator temperature coefficient.

Comparable cycle 3 values are 1000 ppm for the initial boron concentration, 60 ppm/1% ($\Delta k/k$) boron reactivity worth and -0.53×10^{-4} ($\Delta k/k$)/°F for the moderator temperature coefficient. The FSAR shows that the core and RCS are adequately protected during this event. Sufficient time for operator action to terminate this transient is also shown in the FSAR, even with maximum dilution and minimum shutdown margin. The predicted cycle 3 parametric values of importance to the moderator dilution transient are bounded by the FSAR design values; thus, the analysis in the FSAR is valid.

7.4. Cold Water (Pump Startup) Accident

There are no check or isolation valves in the reactor coolant piping; therefore, the classic cold water accident is not possible. However, when the reactor is operated with one or more pumps not running, and then these are turned on, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, then reactivity will be added to the core and a power rise will occur.

Protective interlocks and procedures prevent starting idle pumps if the reactor power is above 22%. However, these restrictions were ignored, and two-pump startup from 50% power was analyzed as the most severe transient.

To maximize reactivity addition, the FSAR analysis assumed the most negative moderator temperature coefficient of -3.0×10^{-4} $(\Delta k/k)/°F$ and the least negative Doppler coefficient of -1.30×10^{-5} $\Delta k/k$. The corresponding most negative moderator temperature coefficient and least negative Doppler coefficient predicted for cycle 2 are -2.55×10^{-4} and -1.56×10^{-5} $(\Delta k/k)/°F$, respectively. Since the predicted cycle 2 moderator temperature coefficient is less negative and the Doppler coefficient is more negative than the values used in the FSAR, the transient results would be less severe than those reported in the FSAR.

7.5. Loss of Coolant Flow

The reactor coolant flow rate decreases if one or more of the reactor coolant pumps fail. A pumping failure can be caused by mechanical failure or loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect the operation of others. With the reactor at power, the effect of loss of coolant flow is a rapid increase in coolant temperature due to the reduction of heat removal capability. This increase could result in DNB if corrective action were not taken immediately. The key parameters for four-pump coastdown or a locked-rotor incident are the flow rate, flow coastdown characteristics, Doppler coefficient, moderator temperature coefficient, and hot channel DNB peaking factors. The most conservative initial conditions were assumed for the densification report[5]: FSAR values of flow and coastdown, -1.17×10^{-5} $(\Delta k/k)/°F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator temperature coefficient, with densified fuel power spike and peaking. The results showed that the DNBR remained above 1.3 (W-3) for the four-pump coastdown, and the fuel cladding temperature remained below criteria limits for the locked-rotor transient.

The predicted parametric values for cycle 3 are $-1.[illegible]3 \times 10^{-[illegible]}$ $(\Delta k/k)/°F$ Doppler coefficient, -0.53×10^{-4} $(\Delta k/k)/°F$ moderator temperature coefficient, and peaking factors as shown in Table 6-1. Since the predicted cycle 3 values are bounded by those used in the densification report, the results of that analysis represent the most severe consequences from a loss-of-flow incident.

7.6. Stuck-Out, Stuck-In, or Dropped Control Rod

If a control rod were dropped into the core while it was operating, a rapid decrease in neutron power would occur, accompanied by a decrease in the core average coolant temperature. The power distribution might be distorted due to a new control rod pattern, under which conditions a return to full power might lead to localized power densities and heat fluxes in excess of design limitations.

The key parameters for this transient are moderator temperature coefficient, dropped rod worth, and local peaking factors. The FSAR analysis was based on 0.46 and 0.36% $\Delta k/k$ rod worths with a moderator temperature coefficient of -3.0×10^{-4} $(\Delta k/k)/°F$. For cycle 3, the maximum worth rod at power is 0.20% $\Delta k/k$ and a moderator temperature coefficient of -2.55×10^{-4} $(\Delta k/k)/°F$. Since the predicted rod worth is less positive and the moderator temperature coefficient is more positive, the consequences of this transient are less severe than the results presented in the FSAR.

7.7. Loss of Electric Power

Two types of power losses were considered in the FSAR: (1) a loss-of-load condition caused by separation of the unit from the transmission system and (2) a hypothetical condition resulting in a complete loss of all system and unit power except that from the unit batteries.

The FSAR analysis evaluated the loss of load with and without turbine runback. When there is no runback, a reactor trip occurs on high reactor coolant pressure or temperature. This case results in a non-limiting accident. The largest offsite dose occurs for the second case, i.e., loss of all electrical power except unit batteries, and assuming operation with failed fuel and steam generator tube leakage. These results are independent of core loading; therefore, the results of the FSAR are applicable for any reload.

7.8. Steam Line Failure

A steam line failure is defined as a rupture of any of the steam lines from the steam generators. Upon initiation of the rupture, both steam generators start to blow down, causing a sudden decrease in the primary system temperature, pressure, and pressurizer level. The temperature reduction leads to positive reactivity insertion, and the reactor trips on high flux or low RC pressure. The FSAR has identified a double-ended rupture of the steam line between the steam generator and steam stop valve as the worst-case situation at end-of-life conditions.

The key parameter for the core response is the moderator temperature coefficient, which was assumed in the FSAR to be -3.0×10^{-4} $(\Delta k/k)/°F$. The cycle 3 predicted value of moderator temperature coefficient is -2.55×10^{-4} $(\Delta k/k)/°F$. This value is bounded by those used in the FSAR analysis; hence, the results in the FSAR represent the worst situation.

7.9. Steam Generator Tube Failure

A rupture or leak in a steam generator tube allows reactor coolant and associated activity to pass to the secondary system. The FSAR analysis is based on complete severance of a steam generator tube. The primary concern for this incident is the potential radiological release, which is independent of core loading. Hence, the FSAR results are applicable to this reload.

7.10. Fuel Handling Accident

The mechanical damage accident is considered the maximum potential source of activity release during fuel handling activities. The primary concern is radiological releases that are independent of core loading; therefore, the FSAR results are applicable to all reloads.

7.11. Rod Ejection Accident

For reactivity to be added to the core more rapidly than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the control rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. This incident represents the most rapid reactivity insertion that can be reasonably postulated. The values used in the FSAR and densification report at BOL conditions, -1.17×10^{-5} $(\Delta k/k)/°F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator temperature coefficient, and an ejected rod worth of 0.65% $\Delta k/k$ represent the maximum possible transient. The corresponding cycle 3 parametric values of -1.43×10^{-5} $(\Delta k/k)/°F$ Doppler, -0.53×10^{-4} $(\Delta k/k)/°F$ moderator temperature coefficient (both more negative than those used in reference 5), and a maximum predicted ejected rod worth of 0.44 $\Delta k/k$ ensure that the results will be less severe than those presented in the FSAR[1] and the densification report[5].

7.12. Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur since it would require a multitude of failures in the engineered safeguards. The hypothetical accident is based solely on a gross release of radioactivity to the reactor building. The consequences of this accident are independent of core loading; hence, the results reported in the FSAR are applicable for all reloads.

7.13. Waste Gas Tank Rupture

The waste gas tank was assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel. Rupture of the tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. The consequences of this incident are independent of core loading; therefore, the results reported in the FSAR are applicable to any reload.

7.14. LOCA Analysis

A generic LOCA analysis has been performed for the B&W 177-FA, lowered-loop NSS using the Final Acceptance Criteria ECCS evaluation model.[14] The analysis is generic since the limiting values of key parameters for all plants in this category were used. Furthermore, the combination of average fuel temperature as a function of linear heat rate and the lifetime pin pressure data used in the reference 14 LOCA limits analysis are conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in reference 14 provide conservative results for the operation of Oconee 3, cycle 3 fuel. The following tabulation shows the bounding values for allowable LOCA peak LHRs for Oconee 3, cycle 3 fuel.

Core elevation, ft	Allowable peak linear heat rate, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

Table 7-1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR, densified value	Predicted cycle 3 value
BOL Doppler coeff, 10^{-5} $(\Delta k/k)/°F$	-1.17[a]	-1.43
EOL Doppler coeff, 10^{-5} $(\Delta k/k)/°F$	-1.33	-1.56
BOL moderator coeff, 10^{-4} $(\Delta k/k)/°F$	+0.5[b]	-0.[illegible]
EOL moderator coeff, 10^{-4} $(\Delta k/k)/°F$	-3.0	-2.[illegible]
All rod bank worth (HZP), % $\Delta k/k$	10.0	8.[illegible]
Initial boron conc (HFP), ppm	1400	10[illegible]
Boron reactivity worth (70F), ppm/1% $\Delta k/k$	75	[illegible]0
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.44
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20

[a] $(-1.2 \times 10^{-5}\ \Delta k/k/F)$ was used for steam line failure analysis; $(-1.3 \times 10^{-5}\ \Delta k/k/F)$ was used for cold water analysis.

[b] $(+0.94 \times 10^{-4}\ \Delta k/k/F)$ was used for the moderator dilution accident.

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8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 3 operation. Changes were the results of the following:

1. Specifying APSR position limits in addition to the usual regulating control rod and imbalance limits for ECCS. The APSR position limits will provide additional control of power peaking and assurance that LOCA kW/ft limits are not exceeded.

2. Using 106.5% of design flow rather than 107.6% as discussed in section 6.1.

3. The FLAME computer code used in setting the Technical Specification limits.[15,16]

4. The Technical Specification limits based on DNBR and LHR criteria include appropriate allowances for projected fuel rod bow penalties, i.e., potential reduction in DNBR and increase in power peaks. A statistical combination of the nuclear uncertainty factor, engineering hot channel factor, and rod bow peaking penalty was used in evaluating LHR criteria, as approved in reference 17.

5. Per reference 18, the power spike penalty due to fuel densification was not used in setting the DNBR- and ECCS-dependent Technical Specification limits.

Based on the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated. Figures 8-1 through 8-14 illustrate revisions to previous Technical Specification limits; Figures 8-15 through 8-17 illustrate limits not previously included in the Technical Specifications.

Figure 8-1. Core Protection Safety Limits — Oconee 3, Cycle 3

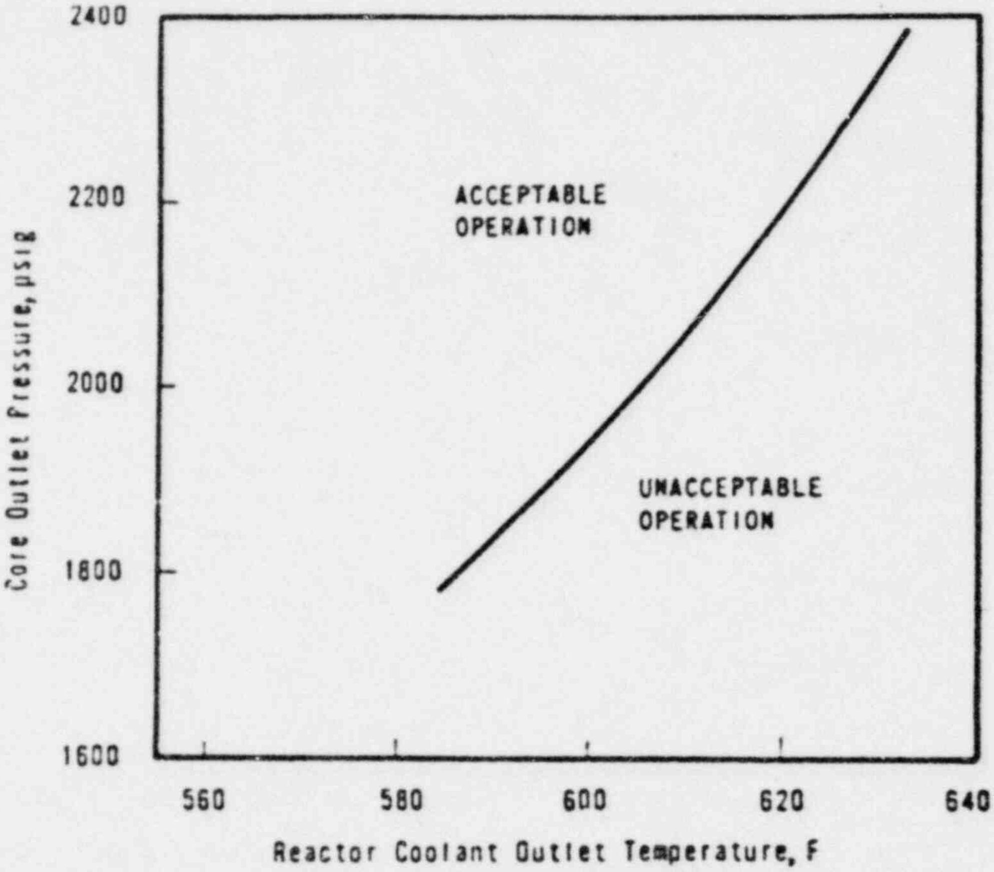
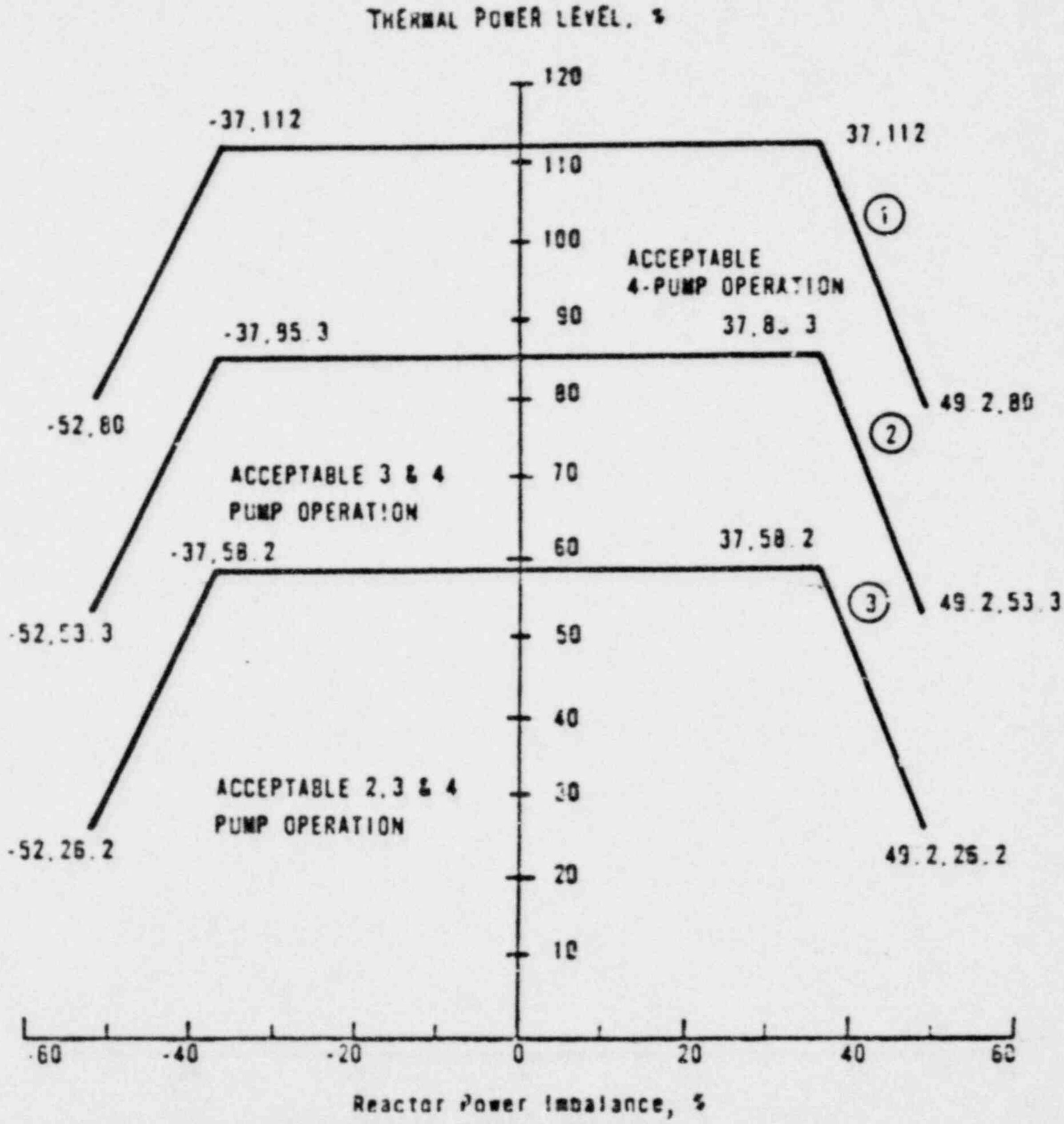


Figure 3-2. Core Protection Safety Limits — Oconee 3, Cycle 3



Curve	Reactor Coolant Flow, gpm
1	374,880 (100%)*
2	280,035 (74.7%)
3	183,690 (49.0%)

*106.5% of first-core design flow.

Figure 8-5. Protective System Maximum Allowable Setpoints — Oconee 3, Cycle 3

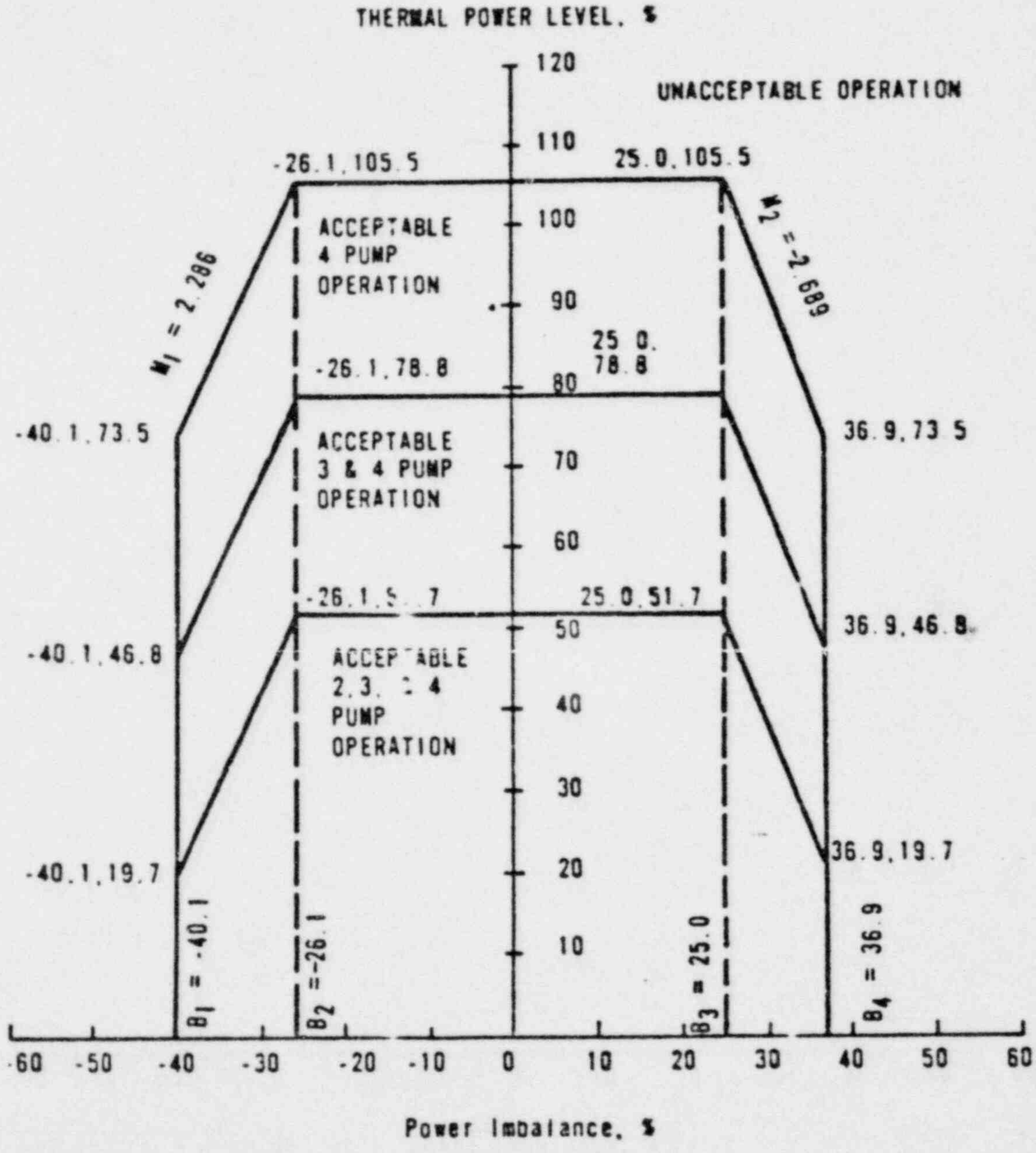


Figure 8-6. Rod Position Limits for Four-Pump Operation From 0 to 100 ± 10 EFPD — Oconee 3, Cycle 3

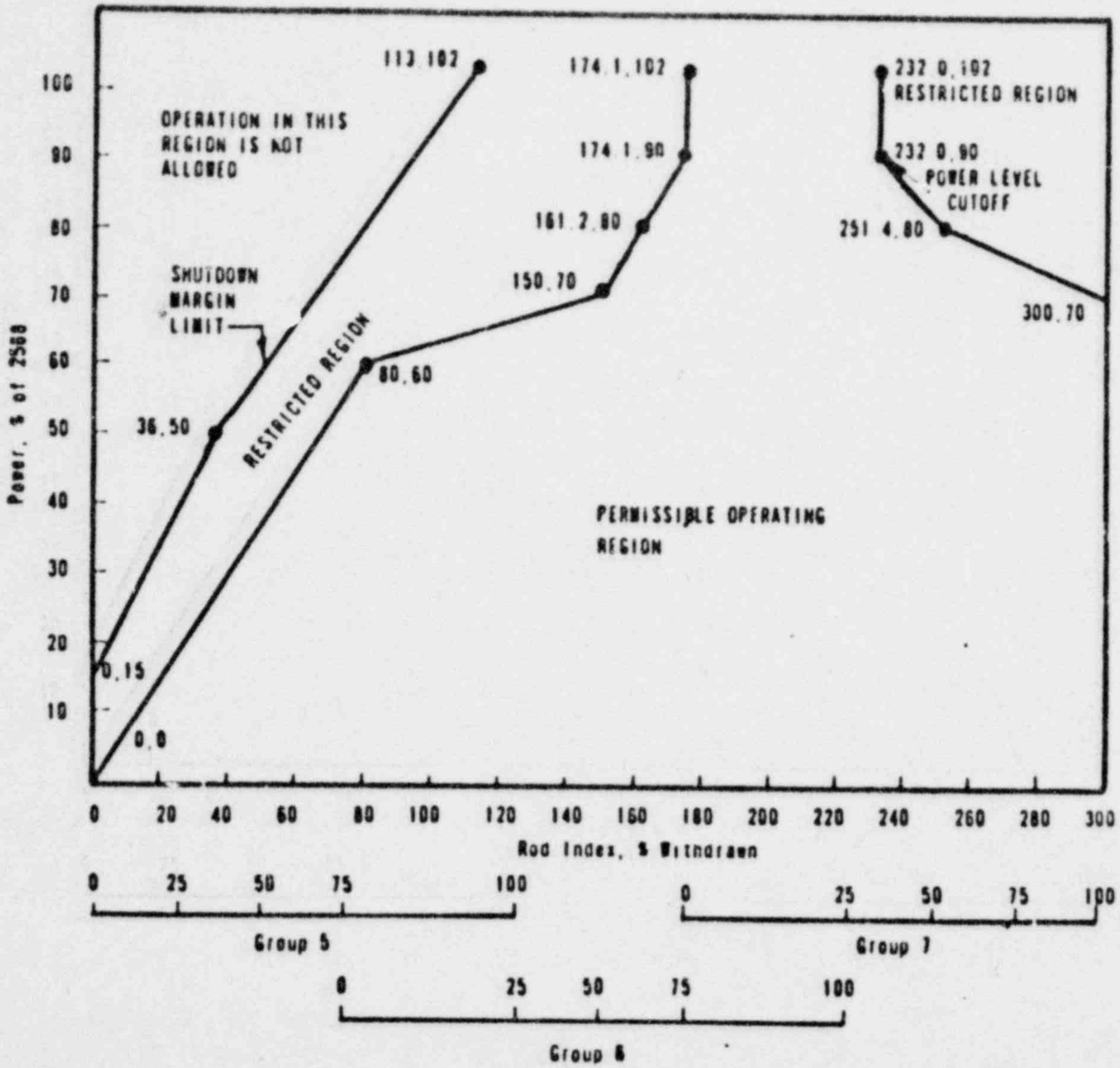


Figure 8-7. Rod Position Limits for Four-Pump Operation From 100 ± 10 to 235 ± 10 EFPD - Oconee 3, Cycle 3

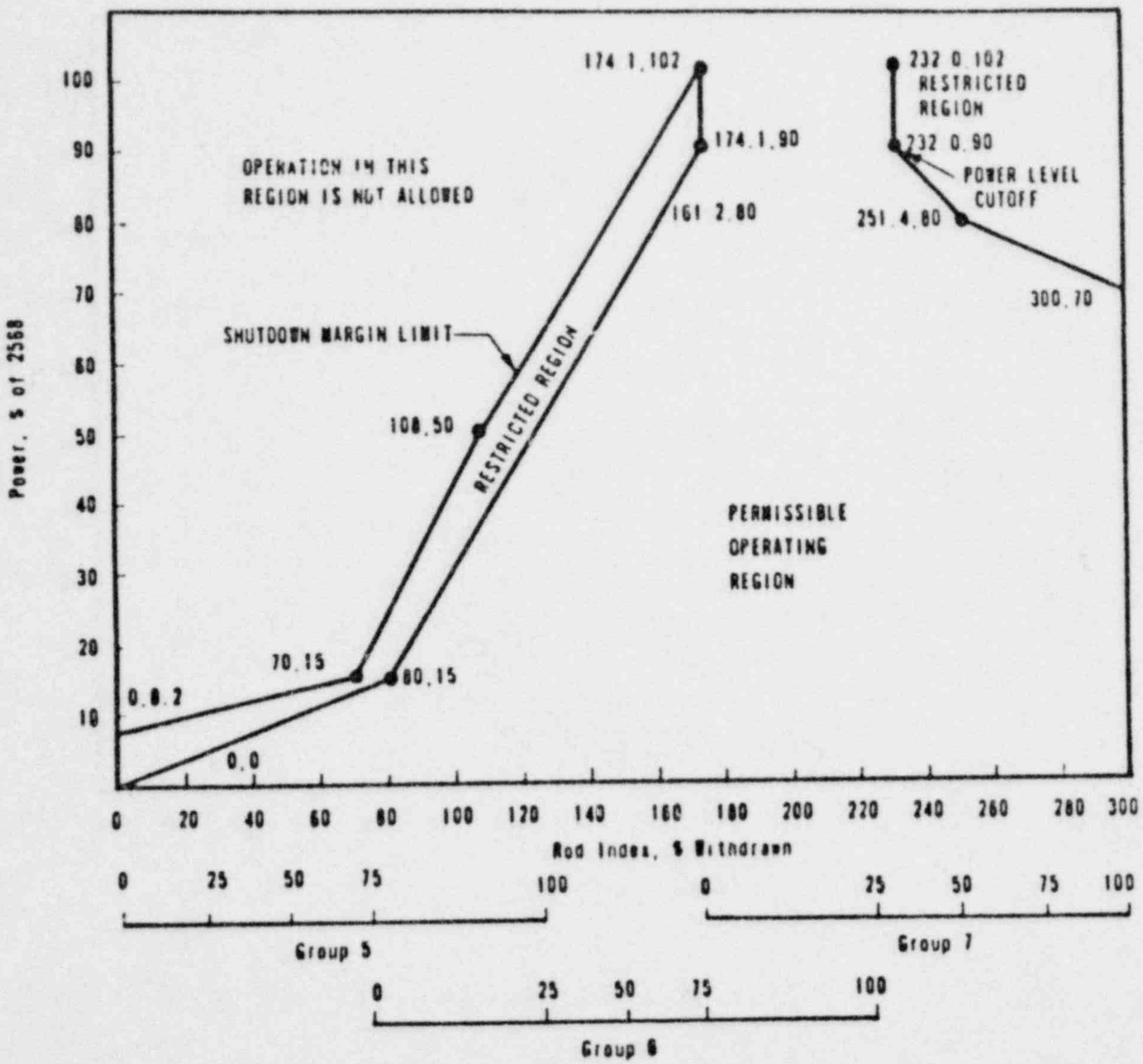


Figure 8-8. Rod Position Limits for Four-Pump Operation After 235 ± 10 EFPD – Oconee 3, Cycle 3

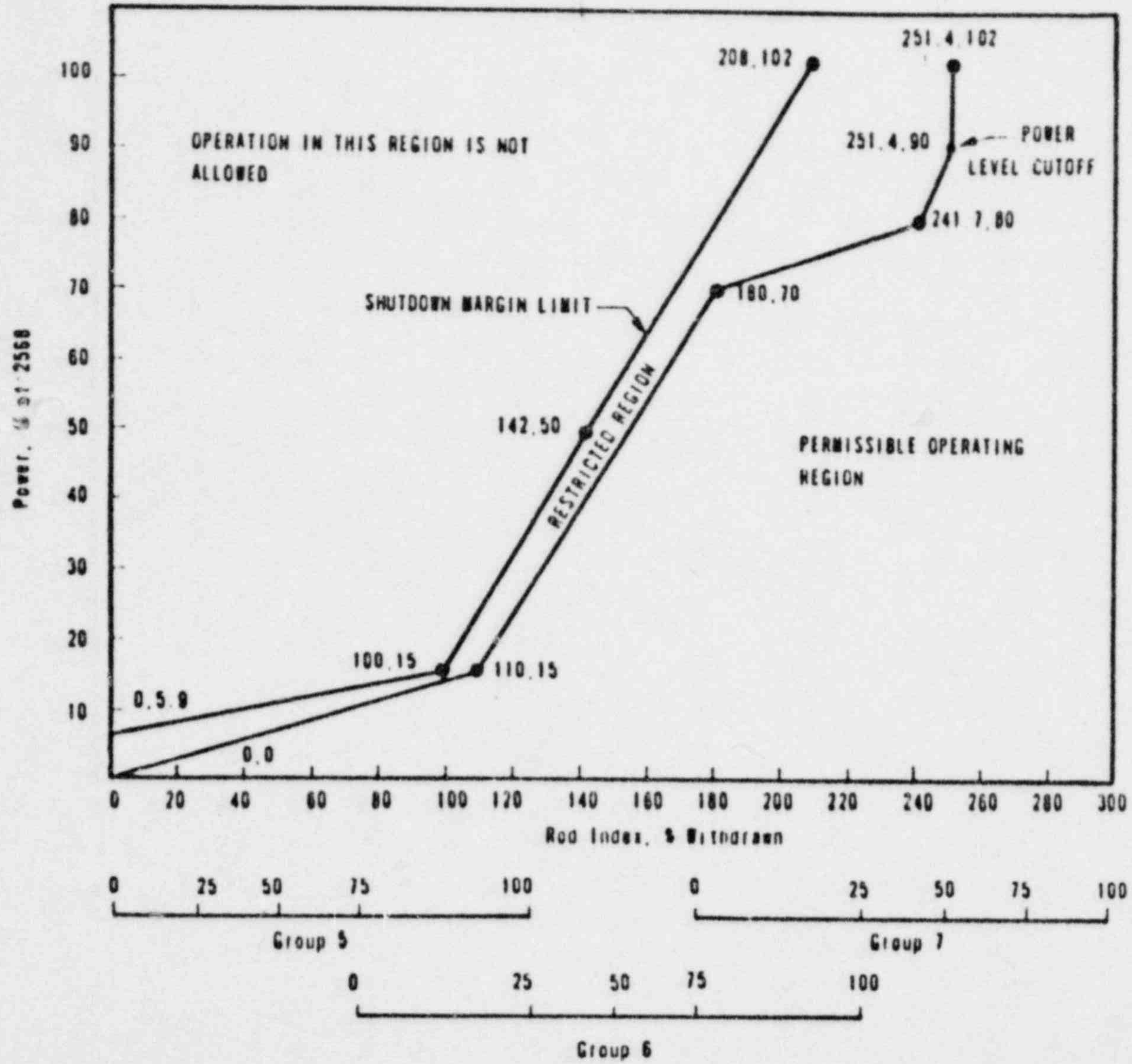


Figure 8-9. Rod Position Limits for Two- and Three-Pump Operation From 0 to 100 ± 10 EFPD — Oconee 3, Cycle 3

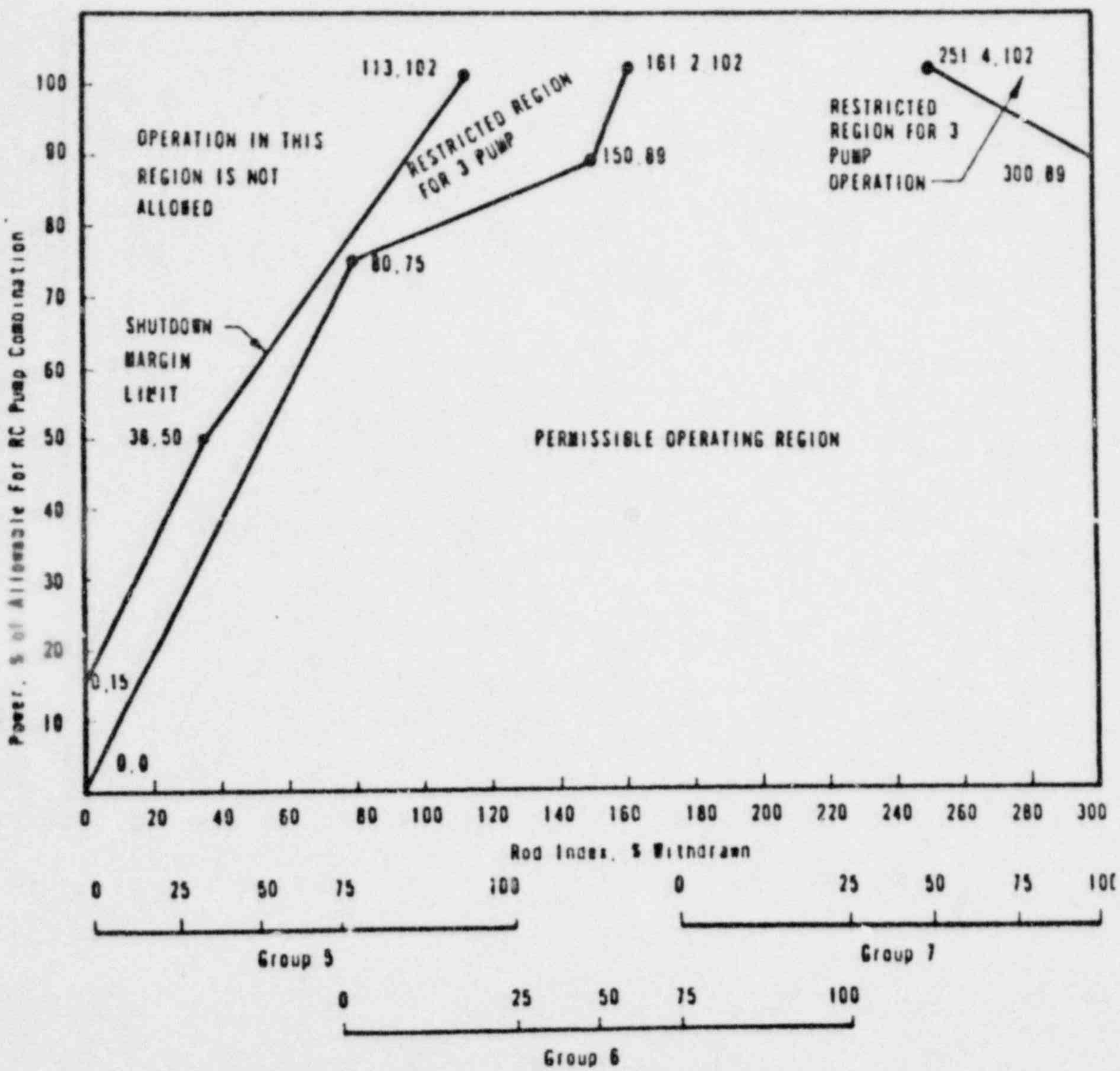


Figure 8-10. Rod Position Limits for Two- and Three-Pump Operation From 100 ± 10 to 235 ± 10 EFPD — Oconee 3, Cycle 3

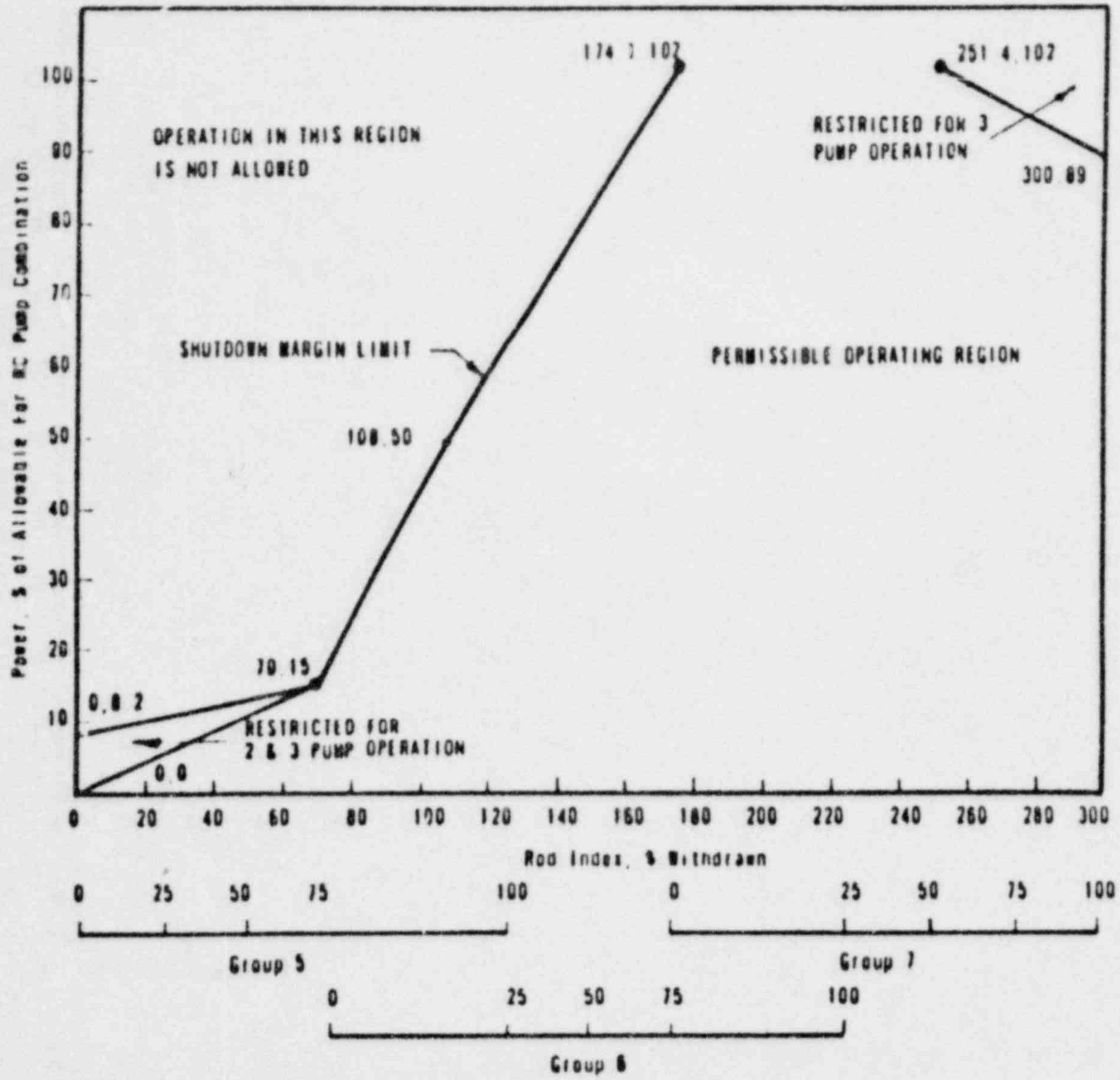


Figure 8-11. Rod Position Limits for Two- and Three-Pump Operation After 235 ± 10 EFPD — Oconee 3, Cycle 3

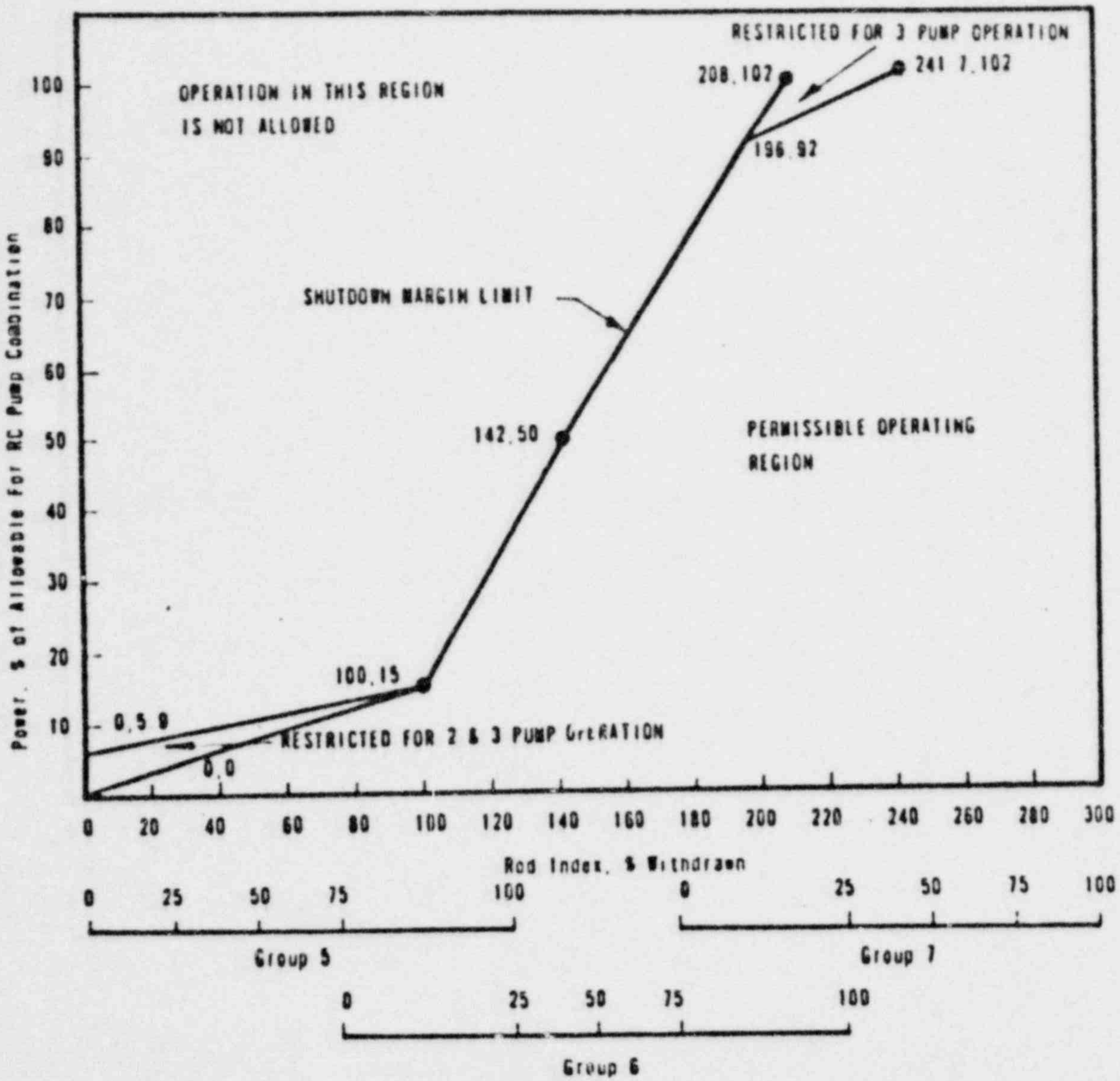


Figure 8-12. Operational Power Imbalance Envelope for Operation From 0 to 100 ± 10 EFPD — Oconee 3, Cycle 3

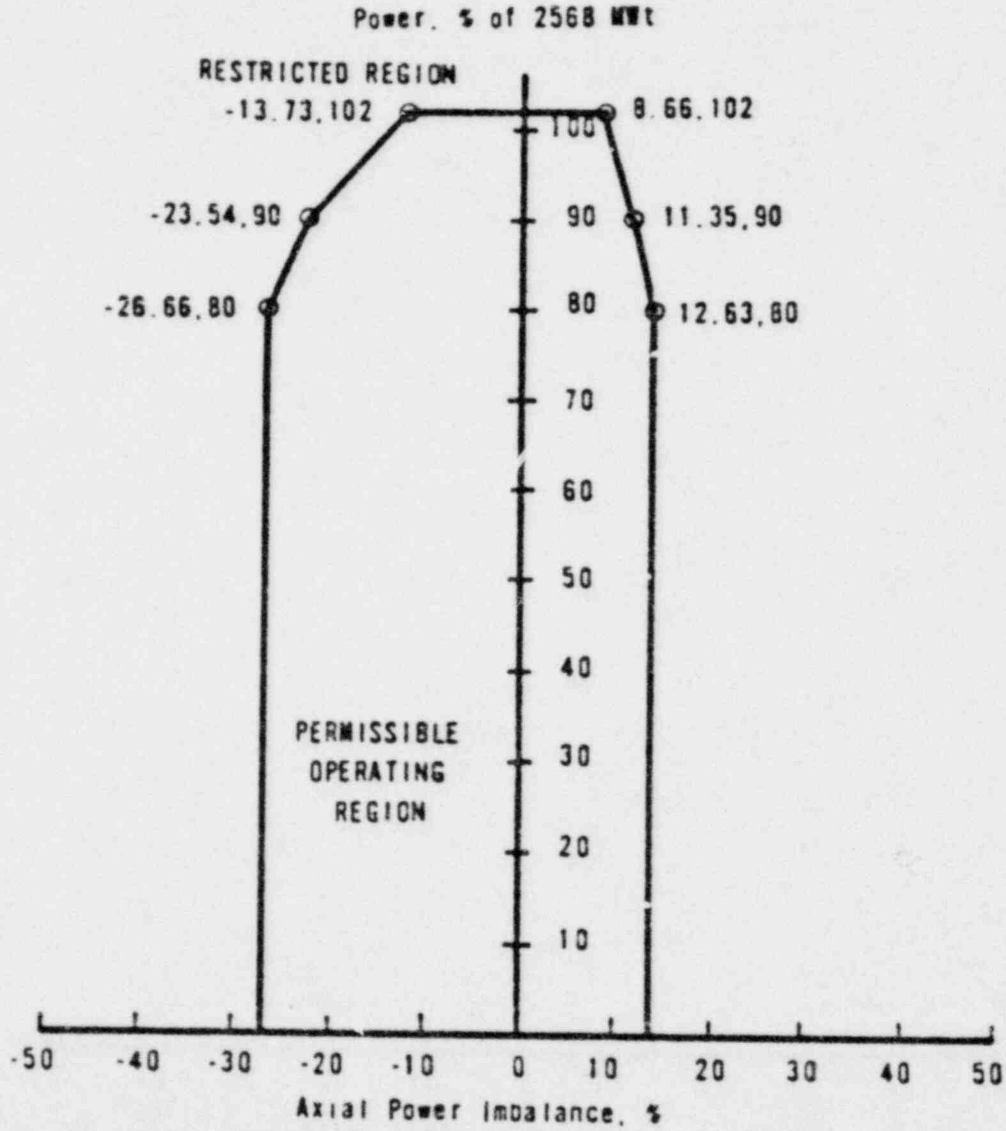


Figure 8-13. Operational Power Imbalance Envelope for Operation From 100 ± 10 to 235 ± 10 EFPD — Oconee 3, Cycle 3

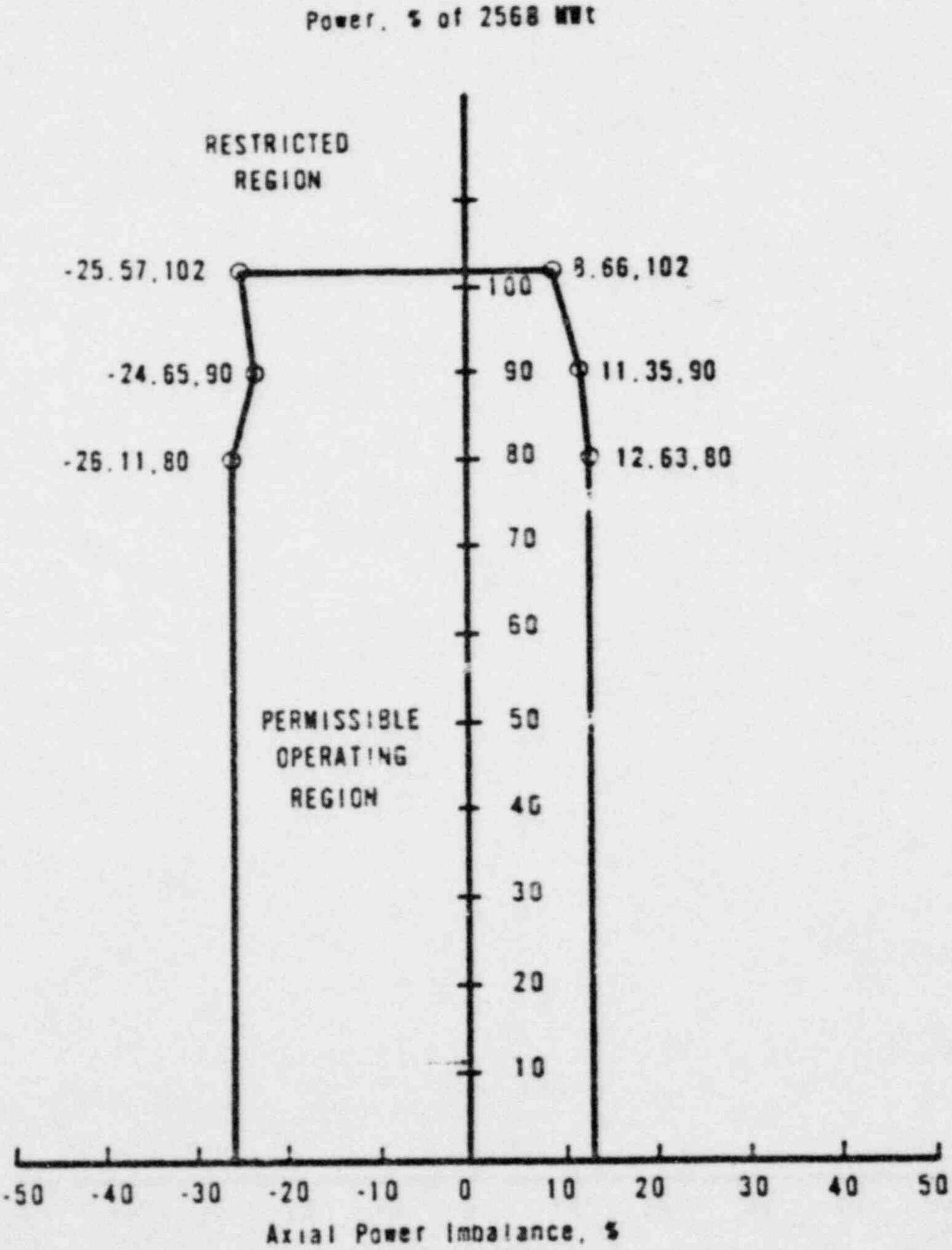


Figure 8-14. Operational Power Imbalance Envelope for Operation After 235 ± 10 EFPD — Oconee 3, Cycle 3

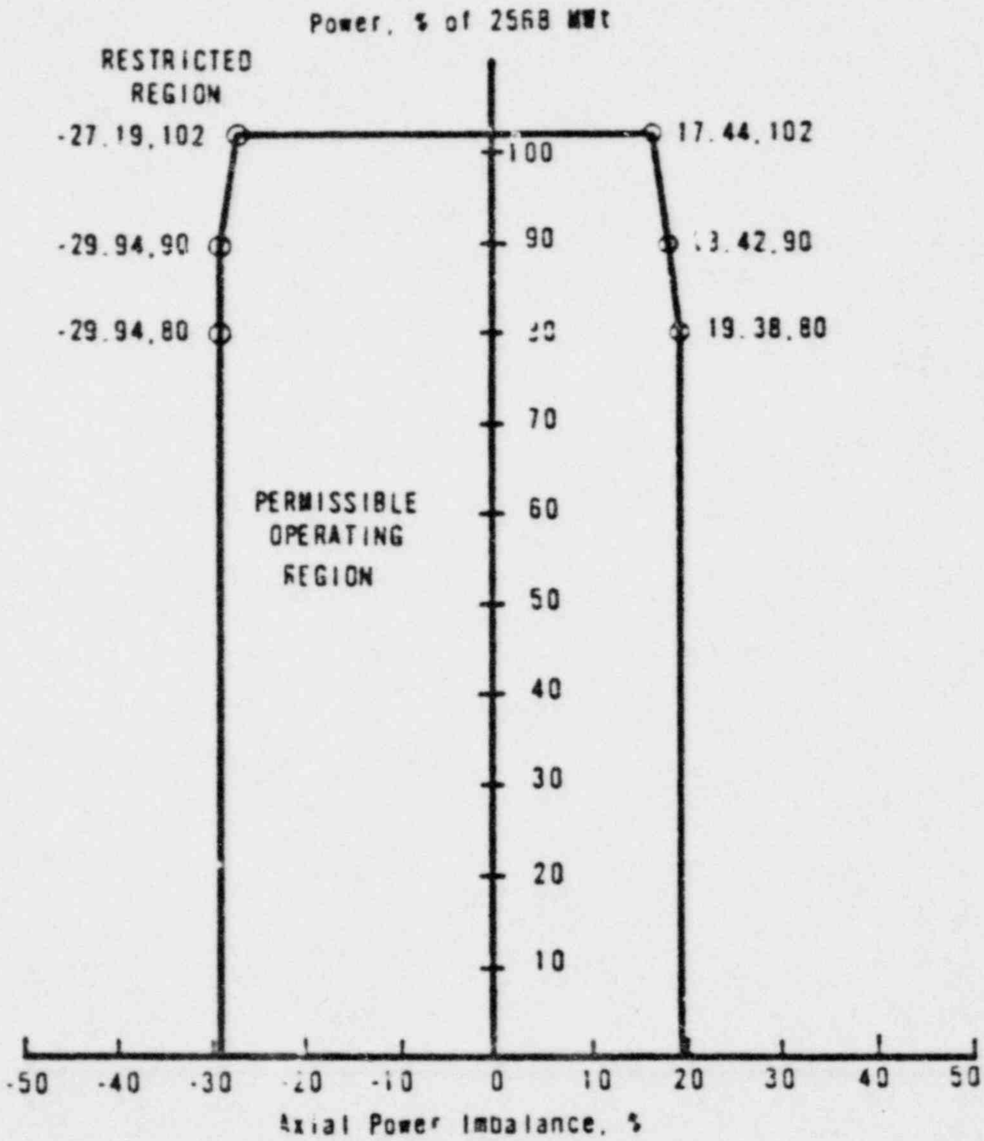


Figure 8-15. APSR Position Limits for Operation From 0 to 100 ± 10 EFPD — Oconee 3, Cycle 3

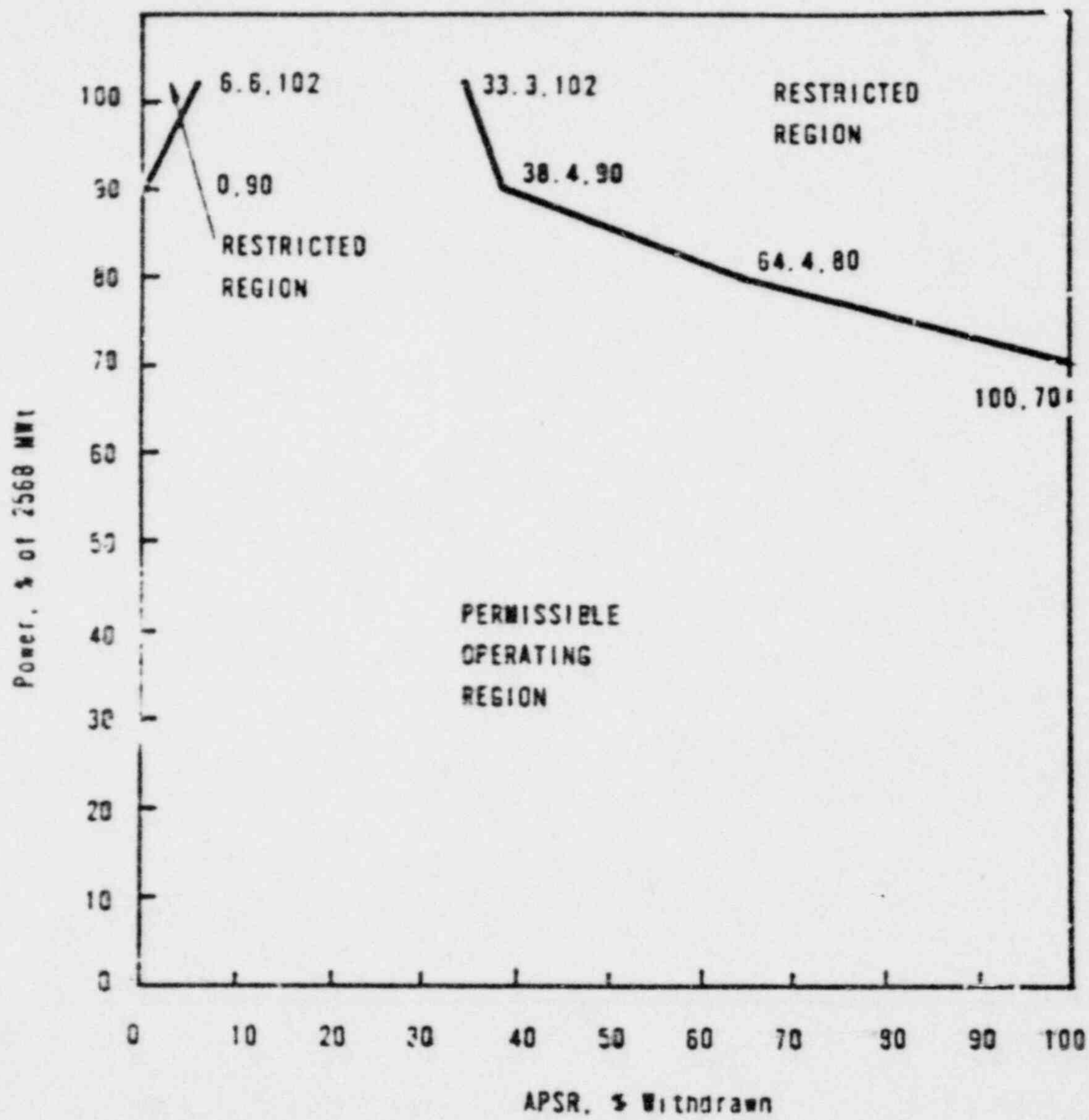


Figure 8-16. APSR Position Limits for Operation From 100 ± 10 to 235 ± 10 EFPD — Oconee 3, Cycle 3

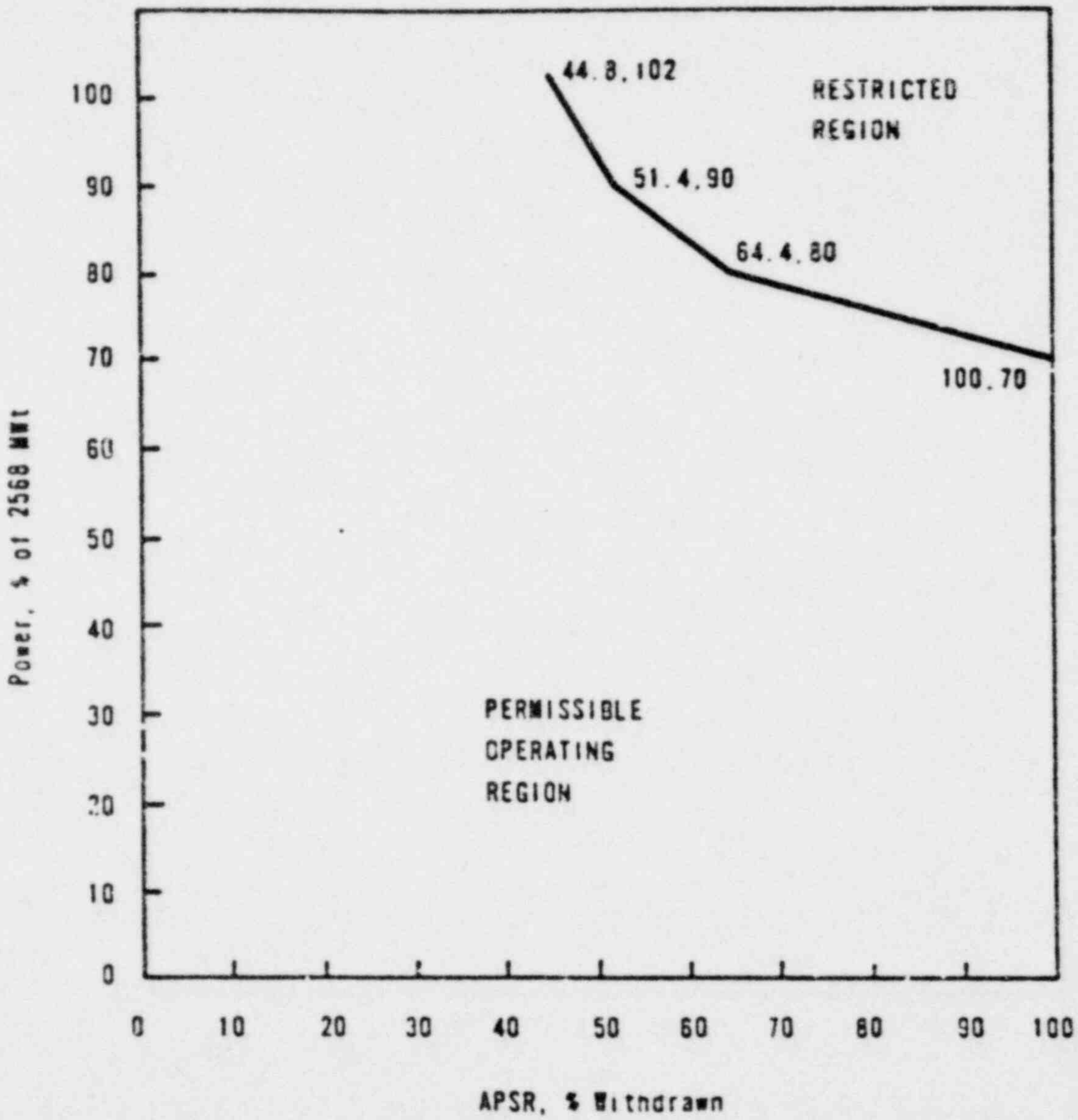
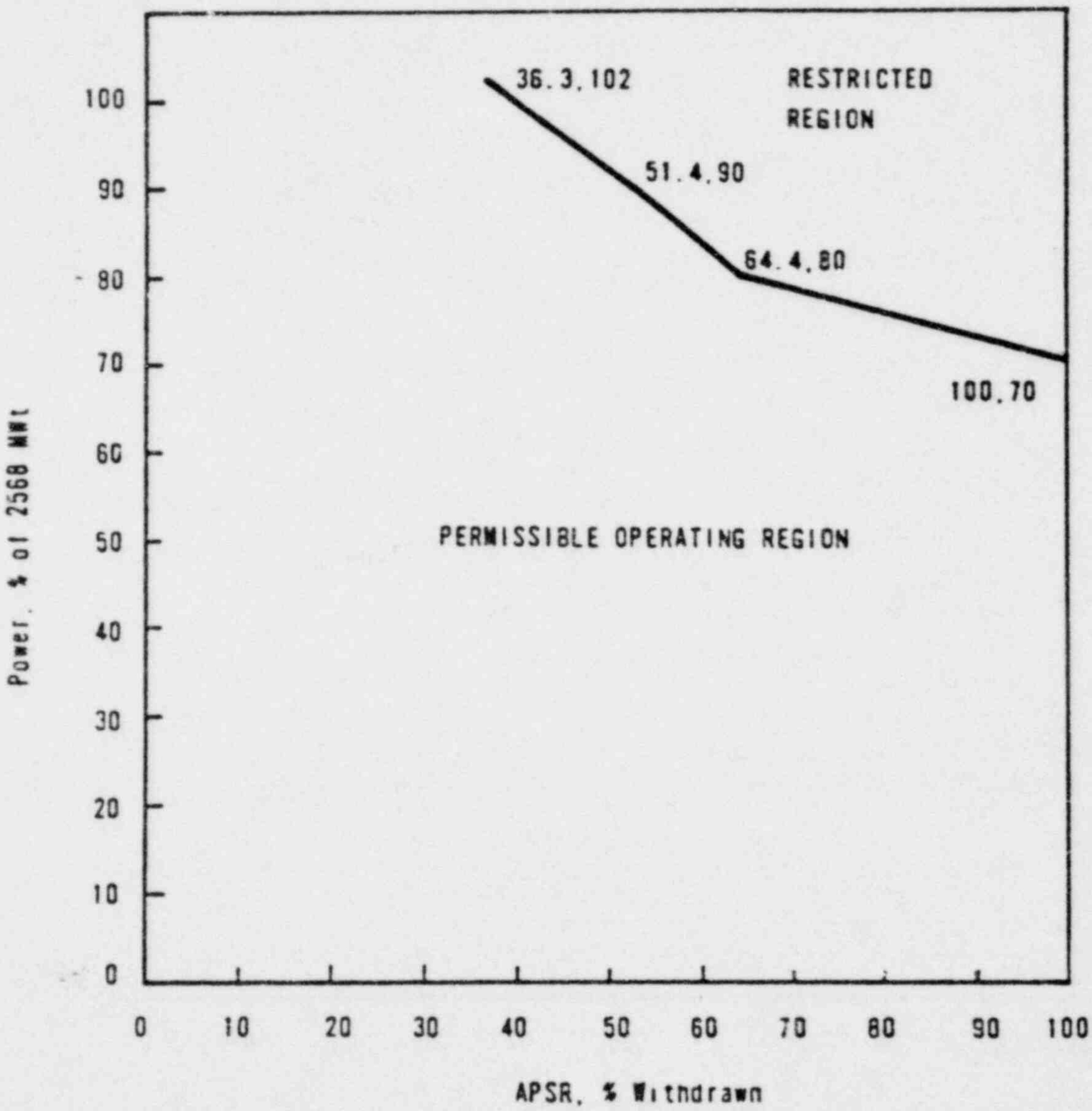


Figure 8-17. APSR Position Limits for Operation After 235 ± 10 EFPD — Oconee 3, Cycle 3



9. STARTUP PROGRAM — PHYSICS TESTING

The planned startup testing associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide the necessary data for continued safe plant operation.

Pre-Critical Tests

1. Control rod drop time

Zero Power Tests

1. Critical boron concentration
2. Temperature reactivity coefficient
3. Control rod group worth
4. Ejected rod worth

Power Tests

1. Core power distribution verification at approximately 40, 75, and 100% FP, normal control rod group configuration.
2. Incore/out-of-core detector imbalance correlation verification at approximately 75% FP.
3. Power Doppler reactivity coefficient at approximately 100% FP.
4. Temperature reactivity coefficient at approximately 100% FP.

REFERENCES

[1] Oconee Nuclear Station, Units 1, 2, and 3, Final Safety Analysis Report, Docket Nos. 50-269, 50-270, and 50-287.

[2] Oconee Unit 3, Cycle 2 Reload Report, BAW-1432, Babcock & Wilcox, June 1976.

[3] Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation, BAW-10035, Babcock & Wilcox, June 1970.

[4] Program to Determine In-Reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, November 1976.

[5] C. D. Morgan and H. S. Kao, TAFY — Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, May 1972.

[6] Oconee 3 Fuel Densification Report, BAW-1399, Babcock & Wilcox, November 1973.

[7] B. J. Buescher and J. W. Pegram, Babcock & Wilcox Model for Predicting In-Reactor Densification, BAW-10083P, Rev. 1, Babcock & Wilcox, November 1976.

[8] Oconee Unit 1, Cycle 4 Reload Report, BAW-1447, Babcock & Wilcox, March 1977.

[9] Oconee Unit 2, Cycle 3 Reload Report, BAW-1452, Babcock & Wilcox, April 1977.

[10] Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, Babcock & Wilcox, June 1976.

[11] K. W. Hill, et al., "Effects on Critical Heat Flux of Local Heat Flux Spike or Local Flow Blockage in PWR Rod Bundles," 74-WA/HT-54, ASME Winter Annual Meeting, New York, November 1974.

[12] CHF — Critical Heat Flux Correlation for CE FA With Standard Spacer Grid — Part 2, "Nonuniform Axial Power Distribution," CENPD-207, Combustion Engineering, June 1976.

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[13] "Core Physics Methods Data Used as Input to LOCA Analysis," XN-75-4?, August 1975 and letter, D. A. Bixel, Consumers Power, to R. A. Purple, April 5, 1976.

[14] ECCS Analysis of B&W's 177-Fuel Assembly, Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

[15] FLAME — Three-Dimensional Noded Code for Calculating Reactivity and Pow. Distributions, BAW-10124A, Babcock & Wilcox, Lynchburg, Virginia, August 1976.

[16] C. W. Mays, Verification of Three-Dimensional FLAME Code, BAW-10125, Babcock & Wilcox, Lynchburg, Virginia, August 1976.

[17] S. A. Varga (NRC) to J. H. Taylor (B&W), Letter, "Comments on B&W's Submittal on Combination of Peaking Factors," May 13, 1977.

[18] K. E. Suhrke (B&W) to S. A. Varga (NRC), Letter, "Densification Power Spike," December 6, 1976.

SUPPLEMENT TO BAW-1447
OCONEE 1, CYCLE 4
RELOAD REPORT

SEPTEMBER, 1977

Introduction and Summary

This report supplements the Oconee 1, Cycle 4 Reload Report (BAW-1447, March 1977) to account for a modified Cycle 4 core loading. The modified core loading consists of the loading of four once burned Batch 2 fuel assemblies at core locations D-4, D-12, N-4 and N-12, originally intended to be loaded with four Batch 4 fuel assemblies. The replacement of the four Batch 4 fuel assemblies with the four Batch 2 assemblies was necessitated because of mechanical damage to one Batch 4 assembly while it was examined in the spent fuel pool. The modified core loading does not affect the results of the core safety analysis or limiting conditions of operation, as documented in BAW-1447.

The following paragraphs describe the four replacement fuel assemblies and provide evaluations of the impact of the modified core loading upon the previous analyses.

Nuclear Design

The modified core loading consists of replacing 4 twice burned, 3.2 wt % batch 4 fuel assemblies with 4 once burned, 2.1 wt % batch 2 fuel assemblies. The batch 2 assemblies were selected to match the reactivity of the replaced batch 4 assemblies as closely as possible. BOC radial power distribution results for the revised core loading show that all fuel assembly powers are within 1% of the original Cycle 4 core loading, except for the replacement location N-12, which has about 3.5% less power in the revised core loading. A Cycle 4 FDQ depletion analysis was performed which showed that these differences become progressively smaller as the cycle is depleted. Figure 1 shows the revised core loading diagram for Oconee 1 Cycle 4 and Figure 2 is an eighth-core map showing the burnup and enrichment distribution at the beginning of Cycle 4 with the revised core loading.

Mechanical Design

The replacement fuel assemblies have been evaluated for mechanical design adequacy and found to meet the criteria for allowable cladding strain and irradiation swelling specified in the Reload Report. The creep collapse time was determined to be > 30,000 EFPH, which is greater than the maximum design life of 14,232 EFPH. Pertinent fuel design parameters for the 4 Batch 2 fuel assemblies are given in Table 1.

Thermal-Hydraulic Design

The revised core loading has been evaluated for thermal-hydraulic design considerations and found not to affect the design presented in the Reload Report. The Mark B-2 replacement assemblies have a higher resistance to flow than the Mark B-3 assemblies being replaced. However, the hot assembly during Cycle 4 is either a Mark B-3 or B-4 assembly that is not in a core location where the replacement assemblies will be placed. Since the insertion of the higher resistance Mark B-2 assemblies will cause an increase in flow in the hot assembly, the thermal-hydraulic design presented in Section 6 of the Reload Report is conservative.

Safety Analysis

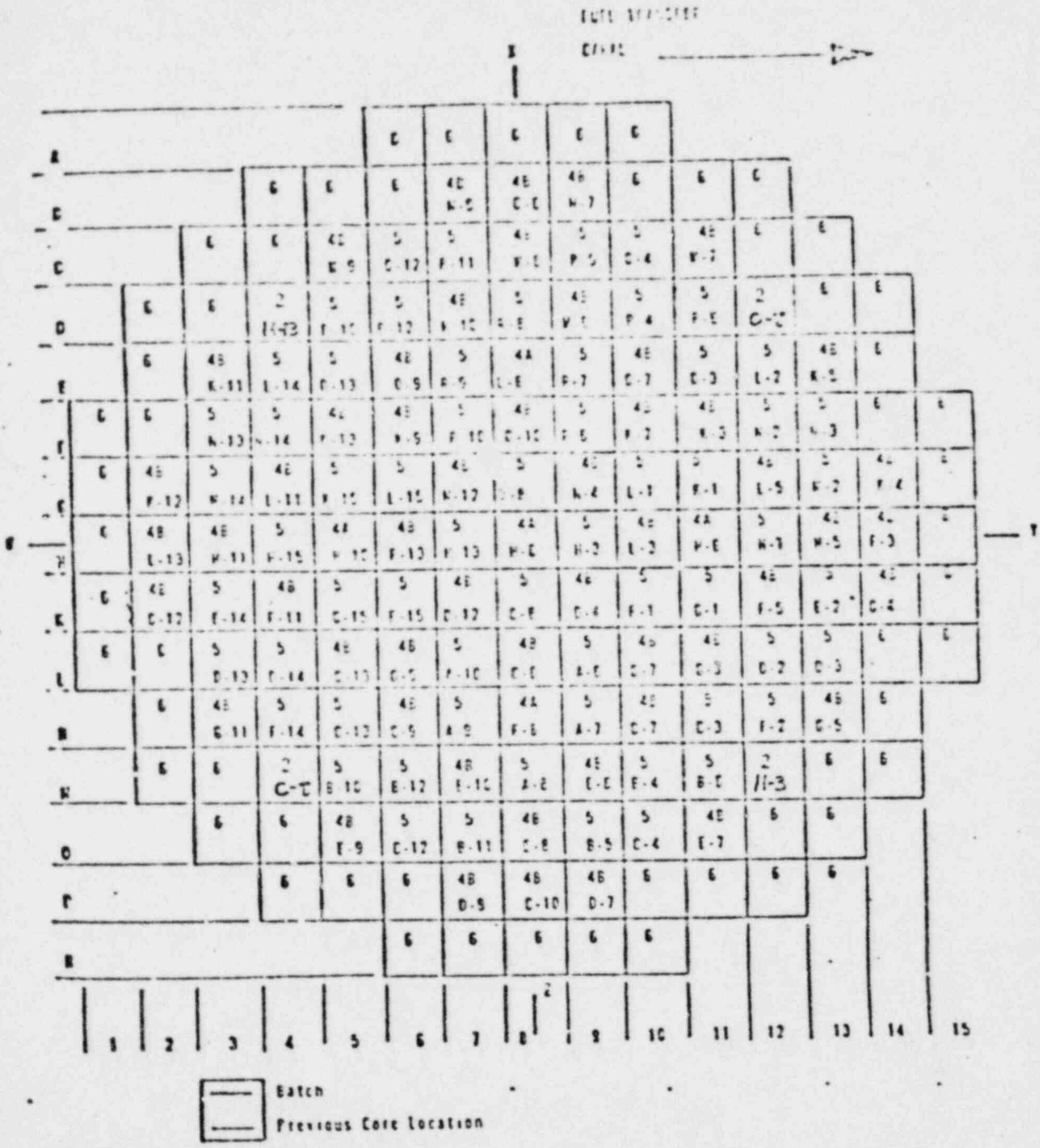
The system parameters important to safety analysis are not affected by the revised core loading, and the safety evaluation presented in the Reload Report therefore remains valid.

Technical Specifications

The power peaking limits and the ejected and shutdown rod insertion limits have been verified and found to be less limiting than the limits calculated previously. Based on these results, the Technical Specifications previously submitted are valid for the modified loading and do not require revision.

Table 1. Fuel Design Parameters for Oconee 1 Cycle 4 Batch 2 Fuel Assemblies

Fuel Assembly Type	Mark-B2
No. of F.A.'s	4
Initial Fuel Enrichment, wt. % ^{235}U	2.10
Initial Fuel Density, % TD	93.5
Batch Burnup, BOC, MWD/MTU	12,630
Cladding Collapse Time, EFPH	> 30,000
Fuel Pellet O.D., in.	0.3700
Undens. Active Fuel Length, in.	144
Flexible Spacers, Type	Corrugated
Solid Spacer Material	ZrO_2



9/8/77

	8	9	10	11	12	13	14	15
H	2.00 22042	2.75 11755	3.20 10734	2.00 21003	2.75 7370	3.20 21432	3.20 10734	2.795 0
K		3.20 15540	2.75 6411	2.75 7323	3.20 15705	2.75 6503	3.20 21003	2.795 0
L			3.20 10004	3.20 20070	2.75 6303	2.75 9503	2.795 0	2.795 0
M				2.75 7117	2.75 10514	3.20 23533	2.795 0	
N					2.10 12467	2.795 0	2.795 0	
O						2.795 0		
P								
R								

X.XX	Initial Enrichment
XXXX	BOC Burnup (MWD/MTU)

9/18/77

END

MICROPHOTOGRAPHER D.J.

DATE 1-6-78

MICROFILM SECTION

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