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OCONEE UNIT 2, CYCLE 6

— Reload Report —

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<u>CONTENTS</u>

List of Tables

List of Figures

Figures (Cont'd)

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth cycle of Oconee Nuclear Station, Unit 2, 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 6 operation of Oconee Unit 2, 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 5 and 6 reactor parameters related to power capability is included in section 5 of this report. All of the accidents analyzed in the FSAR[1] have been reviewed for cycle 6 operation. In those cases where cycle 6 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

One Mark BZ demonstration fuel assembly containing Zircaloy intermediate grids is in the core as part of batch 7; reference 2 describes the demonstration assembly. In addition, four burnable poison rod assemblies (BPRAs) will remain in the core for a second cycle to gather burnup data on burnable poison. Neither the Mark BZ demonstration assembly nor the reinserted BPRAs will adversely affect cycle 6 operation.

The Technical Specifications have been reviewed, and the modifications required for cycle 6 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 2 can be operated safely for cycle 6 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses performed for cycle 6 operation is the currently operating cycle 5. Cycle 5 achieved initial criticality on June 21, 1980 and power escalation began on June 24, 1980. The 100% power level of 2568 MWt was reached on July 9, 1980. No operating anomalies occurred during cycle 5 operation that would adversely affect fuel performance during cycle 6.

3. GENERAL DESCRIPTION

The Oconee 2 reactor core is described in detail in Chapter 3 of the FSAR.[1] The cycle 6 core consists of 177 fuel assemblies, all of which have a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. All fuel assemblies in cycle 6 have a constant nominal fuel loading of 463.6 kg of uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are included in Tables 4-1 and 4-2 of this report.

Figure 3-1 is the core loading diagram for Oconee 2, cycle 6. Batches 6B and 7, with initial enrichments of 2.91 and 3.07 wt % ^{235}U, respectively, will be shuffled to new locations. Batch 8, with an initial enrichment of 3.17 wt % ^{235}U, will be loaded in a checkerboard pattern. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 6.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods, 64 fresh BPRAs, and soluble boron shim. In addition to the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided ofr additional control of axial power distribution. The cycle 6 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 6 are identical to those of the reference cycle described in the Oconee 2, cycle 5 reload report.[3] However, the group designations differ between cycle 6 and the reference cycle to minimize power peaking. The cycle 6 locations and enrichments of the BPRA assemblies are shown in Figure 3-4. The four BPRAs in core locations C8, H13, O8, and H3 during cycle 5 remained in their fuel assemblies, which were shuffled to locations R8, H1, A8, and H15, repsectively, for cycle 6. These BPRAs will undergo a second burn during cycle 6.

Figure 3-1. Core Loading Diagram — Oconee 2 Cycle 6

X (top), Z (bottom), W (left), Y (right)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						7 K4	7 L5	7* O8	7 L11	7 K12					
B				7 M4	8	7 L3	8	7 K2	8	7 L13	8	7 M12			
C			7 H15	8	7 N3	8	7 K6	8	7 K10	8	7 N13	8	7 R8		
D		7 D11	8	7 H11	8	6B A7	8	6B N2	8	6B A9	8	7 M8	8	7 D5	
E		8	7 C12	8	6B O13	8	6B L1	8	6B L15	8	6B O3	8	7 C4	8	
F	7 D9	7 C10	8	6B G1	8	6B M14	8	6B P4	8	6B P5	8	6B G15	8	7 C6	7 D7
G	7 E10	8	7 F9	8	6B A10	8	6B P11	7 P7	6B M2	8	6B A6	8	7 F7	8	7 E6
H	7* H13	7 P9	8	6B P12	8	6B N14	7 K14	6B L14	7** G2	6B D2	8	6B B4	8	7 B7	7* H3
K	7 M10	8	7 L9	8	6B R10	8	6B E14	7 B9	6B B5	8	6B R6	8	7 L7	8	7 M6
L	7 N9	7 O10	8	6B K1	8	6B B11	8	6B B12	8	6B E2	8	6B K15	8	7 O6	7 N7
M		8	7 O12	8	6B C13	8	6B F1	8	6B F15	8	6B C3	8	7 O4	8	
N		7 N11	8	7 E8	8	6B R7	8	6B D14	8	6B R9	8	7 H5	8	7 N5	
O			7 A8	8	7 D3	8	7 G6	8	7 G10	8	7 D13	8	7 H1		
P				7 E4	8	7 F3	8	7 G14	8	7 F13	8	7 E12			
R						7 G4	7 F5	7* C8	7 F11	7 G12					

Legend (box): Batch / Previous Core Location

*H1, A8, H15, and R8 contain reinserted BPRAs.

**H9 location contains Mark BZ demonstration assembly (once-burned).

Figure 3-2. Oconee 2 BOC 6 Enrichment and Burnup Distribution After a 390-EFPD Cycle 5

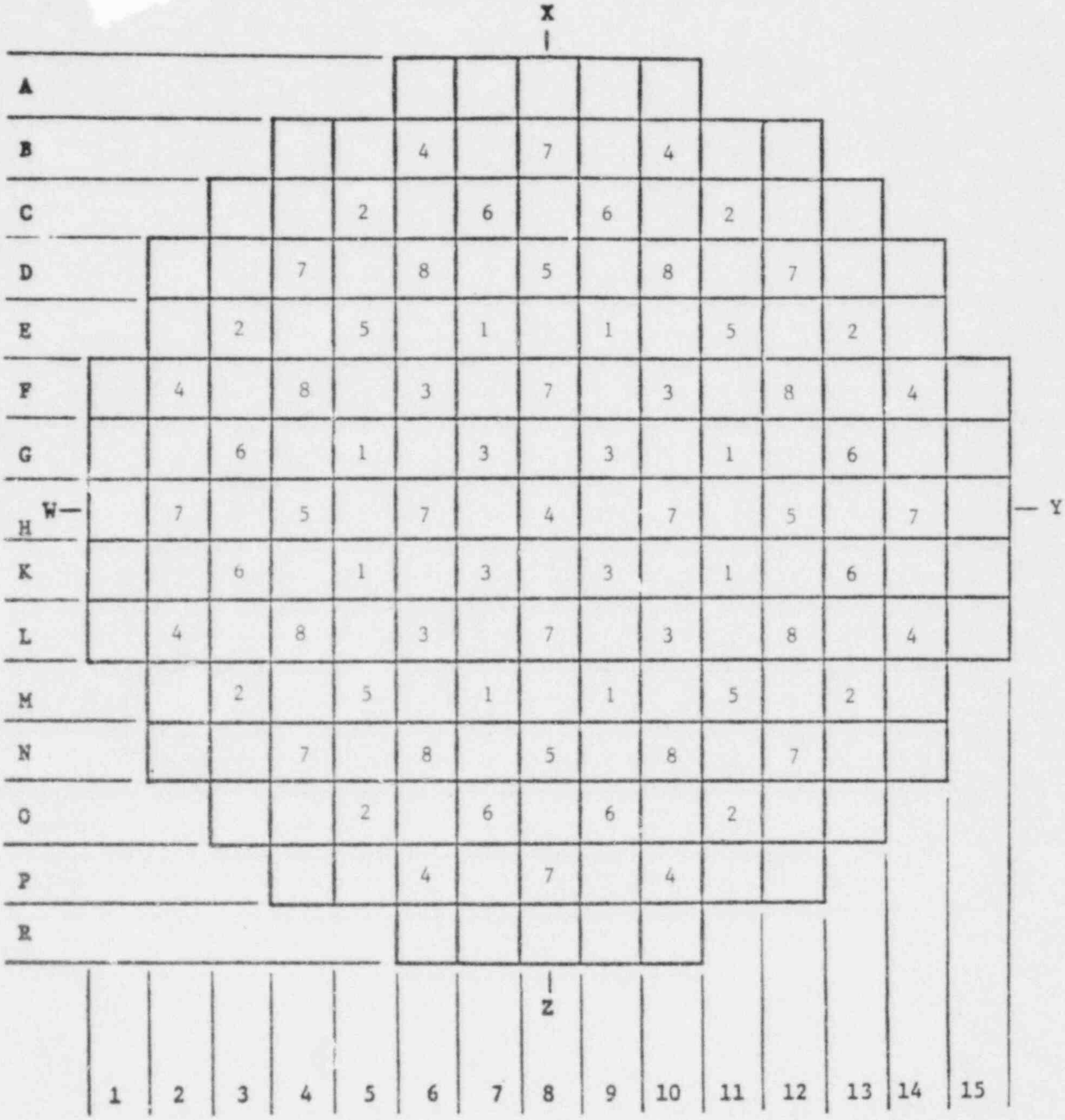
	8	9	10	11	12	13	14	15
H	2.91 20091	3.07* 13823	2.91 18798	3.17 0	2.91 18795	3.17 0	3.07 13827	3.07** 15498
K		2.91 16795	3.17 0	2.91 13861	3.17 0	3.07 16385	3.17 0	3.07 16284
L			2.91 16796	3.17 0	2.91 19317	3.17 0	3.07 14500	3.07 15115
M				2.91 16111	3.17 0	3.07 12826	3.17 0	
N					3.07 15474	3.17 0	3.07 15392	
O						3.07 9608		
P								
R								

*Mark BZ demonstration assembly (once-burned).

**Reinserted BPRA.

XXX	Initial Enrichment wt % ^{235}U
XXXXX	BOC Burnup, MWd/mtU

[illegible]. Oconee 2 Cycle 6 Control Rod Locations and Designations



X — Group Number

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

Figure 3-4. Oconee 2 Cycle 6 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				1.20		1.20		Reins. BPRA*
K			1.20		1.20		0.20	
L		1.20		1.20		0.80		
M	1.20		1.20		1.20			
N		1.20		1.20		0.20		
O	1.20		0.80		0.20			
P		0.20						
R	Reins. BPRA*							

X.XX	LBP Concentration, wt % B_4C in Al_2O_3

*Originally 0.80 wt % B_4C in Al_2O_3 (at BOC 5).

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies (FAs) and pertinent fuel parameters for Oconee 2, cycle 6 are listed in Table 4-1. All Mark B FAs are identical in concept and are mechanically interchangeable.

One reinserted Mark BZ demonstration fuel assembly is included in batch 7. The Mark BZ is a 15 × 15 fuel assembly similar to the Mark B assembly described in reference 1 except that six intermediate spacer grids are of Zircaloy material, and an Inconel 718 spring replaced the Inconel X750 holddown spring. The Mark BZ assembly is described in reference 2, which also states that reactor safety and performance are not adversely affected by the presence of the one demonstration assembly. In addition, batch 7 includes three Mark B4 assemblies containing the Inconel 718 holddown spring.

Retainers will be used on two batch 7 fuel assemblies that contain regenerative neutron sources (RNS), on 64 batch 8 assemblies containing BPRAs, and on the four batch 7 assemlbies with once-burned BPRAs that will be inserted for cycle 6. The justification for the design and use of the retainer is described in references 4 and 5.

4.2. Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1. Cladding Collapse

The fuel of batch 6B is more limiting than batches 7 and 8 due to its previous incore exposure time. The batch 6B assembly power histories were analyzed to determine the most limiting three-cycle power history for creep collapse. This power history was then compared to a generic analysis to ensure that creep-ovalization will not affect fuel pwerformance during Oconee 2 cycle 6. The generic analysis was based on reference 6 and is applicable to the batch 6B design.

The creep-collapse analysis predicts a collapse time longer than 35,000 EFPH, which is longer than the expected residence time of 27,432 EFPH (Table 4-1).

4.2.2. Cladding Stress

The Oconee 2, cycle 6 stress parameters are enveloped by a conservative fuel rod stress analysis. No new method was used for analysis of cycle 6 that had not been used on the previous cycle.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic tensile circumferential strain. The pellet is designed to ensure that plastic cladding strain is less than 1% at design local pellet burnup and heat generation rate. The design values are higher than the worst-case values the Oconee 2 cycle 6 fuel is expected to see. The strain analysis is also based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance for the cladding inside diameter (ID).

4.3. Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 8 fuel inserted for cycle 6 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 and the average fuel temperature for each batch in cycle 6 are shown in Table 4-2. The maximum fuel rod burnup at EOC-6 is predicted to be 37,046 MWd/mtU. Fuel rod internal pressure has been evaluated using TAFY-3[7] for the highest-burnup fuel rod and was predicted to be less than 2200 psia.

4.4. Material Design

The batch 8 FAs are not new in concept, nor do they utilize different component materials. One Mark BZ demonstration assembly, described in section 4.1, will be reinserted during cycle 6. This assembly uses Zircaloy grids and an Inconel 718 spring. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 8 FAs are identical to those of the present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B fuel assembly has verified the adequacy of its design. As of August 31, 1981, the following experience has been accumulated for the eight B&W 177-FA plants using the Mark B fuel assembly:

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU[a]		Cumulative net electrical output, mWh[b]
		Incore	Discharged	
Oconee 1	7	40,000	40,000	36,855,958
Oconee 2	5	34,778	33,780	31,580,263
Oconee 3	6	29,134	32,061	31,113,594
TMI-1	5	32,400	32,300	23,840,053
ANO-1	5	25,000	33,222	27,801,798
Rancho Seco	5	29,493	37,730	25,235,809
Crystal River 3	3	28,892	22,389	15,674,038
Davis Besse 1	2	22,904	13,252	9,907,249

[a] As of August 31, 1981.
[b] As of May 31, 1981.

Table 4-1. Fuel Design Parameters and Dimensions

	Twiced-burned FAs, batch 6B	Once-burned FAs, batch 7	Fresh FAs, batch 8
FA type	Mark B4	Mark B4[(a)]	Mark B4
No. of FAs	37	64/4	72
Fuel rod OD, in.	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4
Undensif. active fuel length (nom), in.	142.25	142.25/ 141.38	141.80
Fuel pellet initial density (nom), % TD	94.0	94.0/ 95.0	95.0
Fuel pellet OD (mean specification), in.	0.3695	0.3695/ 0.3686	0.3686
Initial fuel enrichment, wt % ^{235}U	2.91	3.07	3.17
BOC burnup (avg), MWd/mtU	17,155	14,662	0
Cladding collapse time, EFPH	35,000	35,000	35,000
Estimated residence time (max), EFPH	27,432	28,650	28,800

(a) Batch 7 includes one Mark BZ demonstration assembly containing six Zircaloy intermediate spacer grids and an Inconel 718 holddown spring. Batch 7 also includes three Mark B4 assemblies containing the Inconel 718 holddown spring.

Table 4-2. Fuel Thermal Analysis Parameters — Oconee 2 Cycle 6

	Batch 6B	Batch 7	Batch 8
No. of assemblies	37	64[a]/4[b]	72
Nominal pellet density, % TD	94.0	94.0/95.0	95.0
Pellet diameter, in.	0.3695	0.3695/0.3686	0.3686
Stack height, in.	142.25	142.25/141.38	141.80
Densified Fuel Parameters[c]			
Pellet diameter, in.	0.3646	0.364/0.3649	0.3649
Fuel stack height, in.	140.47	140.47/140.32	140.7
Nominal LHR at 2568 MWt, kW/ft	5.80	5.80	5.79
Avg fuel temp at nominal LHR, F	1320	1320/1310	1310
LHR to ℄ fuel melt, kW/ft	20.15	20.15	20.15

Core average densified LHR is 5.80 kW/ft.

[a]Includes four reinserted LBP assemblies.

[b]Batch 7 includes one Mark BZ demonstration assembly containing six Zircaloy intermediate spacer grids and an Inconel 718 holddown spring. Batch 7 also includes three Mark B4 assemblies containing the Inconel 718 holddown spring

[c]Densification to 96.5% TD assumed.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of design cycles 5 and 6. The values for both cycles were generated using PDQ07.[8-10] The average cycle burnup will be higher in cycle 6 than in the design cycle 5 because of the longer cycle 6 length. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 6 (BOC-6) at full power with equilibrium xenon and normal rod positions.

Both cycle 5 and cycle 6 are feed-and-bleed cycles. The differences between the physics parameters of the two cycles are the result of the longer cycle 6 design life, the different LBP loading, and the different shuffle patterns. The critical boron concentrations for cycle 6 are higher than those for cycle 5 because the additional reactivity necessary for the longer cycle is not completely offset by the burnable poison. The control rod worths differ between cycles because of changes in the radial flux and burnup distributions. This also accounts for differences in ejected and stuck rod worths. 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 position limits presented in section 8. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 6 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

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

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The reference fuel cycle shutdown margin is presented in the Oconee 2, cycle 5 reload report.[3]

5.2. Analytical Input

The cycle 6 incore measurement calculation constants to be used for computing core power distributions were prepared in the same manner as those for the reference cycle.

5.3. Changes in Nuclear Design

There are no significant core design changes between the reference and reload cycles. The same calculational methods and design information were used to obtain the important nuclear design parameters for this cycle.

Table 5-1. Oconee 2, Cycle 5 and 6 Physics Parameters[a]

	Cycle 5[b]	Cycle 6[c]
Cycle length, EFPD	360	400
Cycle burnup, MWd/mtU	11,266	12,518
Average core burnup, EOC, MWd/mtU	20,957	21,736
Initial core loading, mtU	82.1	82.1
Critical boron, BOC (no Xe), ppm[d]		
HZP, group 8 inserted	1459	1552
HFP, group 8 inserted	1263	1344
Critical boron, EOC (eq Xe), ppm		
HZP } group 8 inserted, eq Xe	387	402
HFP } group 8 inserted, eq Xe	83	72
Control rod worth, HFP, BOC, % Δk/k		
Group 7	1.60	1.47
Group 8	0.41	0.39
Control rod worth, HFP, EOC, % Δk/k		
Group 7	1.59	1.53
Group 8	0.49	0.49
Max ejected rod worth, HZP, % Δk/k[e]		
BOC	0.71	0.63
EOC	0.62	0.57
Max stuck rod worth, HZP, % Δk/k		
BOC	1.93	1.78
EOC	1.73	1.80
Power deficit, HZP to HFP, % Δk/k		
BOC	-1.60	-1.56
EOC	-2.27	-2.38
Doppler coeff, 10^{-5} (Δk/k-°F)		
BOC, 100% power, no Xe	-1.57	-1.52
EOC, 100% power, eq Xe	-1.69	-1.77
Moderator coeff, HFP, 10^{-4} (Δk/k-°F)		
BOC (group 8 in, no Xe, 1263 ppm boron)	-0.61	-0.63
EOC (group 8 in, eq Xe, 17 ppm boron)	-2.87	-2.98
Boron worth, HFP, ppm/%(Δk/k)		
BOC (950 ppm boron)	116	124
EOC (17 ppm boron)	106	107
Xenon worth, HFP, % Δk/k		
BOC (4 EFPD)	2.62	2.56
EOC (equilibrium)	2.73	2.70
Eff delayed neutron fraction, HFP		
BOC	0.00613	0.00628
EOC	0.00526	0.00522

(a) Cycle 6 data are for the conditions stated in this report. The cycle 5 core conditions are identified in reference 3.

(b) Cycle 5 data are based on a cycle 4 length of 353 EFPD.

(c) Cycle 6 data are based on a cycle 5 length of 390 EFPD.

(d) HZP denotes hot zero power (532F T_{avg}), HFP denotes hot full power (579F T_{avg}).

(e) Ejected rod worth for groups 5 through 8 inserted.

Table 5-2. Shutdown Margin Calculation — Oconee 2 Cycle 6[a]

	BOC, % Δk/k	EOC[c], % Δk/k
Available Worth		
Total rod worth, HZP[b]	9.04	9.42
Worth reduction due to burnup of poison material	-0.42	-0.42
Maximum stuck rod, HZP	-1.78	-1.80
Net worth	6.84	7.20
Less 10% uncertainty	-0.68	-0.72
Total available worth	6.16	6.48
Required Rod Worth		
Power deficit, HFP to HZP	1.56	2.38
Max allowable inserted rod worth	0.25	0.50
Flux redistribution	0.61	1.20
Total required worth	2.42	4.08
Shutdown Margin		
Total available worth minus total required worth	3.74	2.40

[a] Based on a cycle 5 length of 390 EFPD.

[b] HZP: hot zero power, HFP: hot full power.

[c] 400 EFPD.

Note: Required shutdown margin is 1.00% Δk/k.

Figure 5-1. BOC 6 (4 EFPD) Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, Normal Rod Positions, Group 8 Inserted

	8	9	10	11	12	13	14	15
H	0.945	1.107*	1.055	1.280	1.097	1.253	1.037	0.500
K		1.046	1.250	1.163	1.236	1.174	1.123	0.508
L			1.113	1.224	8 0.926	1.196	0.906	0.398
M				1.082	1.201	1.109	0.899	
N					1.149	1.047	0.487	
O						0.605		
P								
R								

*Mark BZ demonstration assembly (once-burned).

Legend	
X	Inserted rod group No.
XXX	Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 8 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The thermal-hydraulic design evaluation supporting cycle 6 operation utilized the methods and models described in references 1, 3, and 11 except for the core bypass flow.

The maximum core bypass flow due to the insertion of 56 BPRAs in cycle 5 was 8.1%. For cycle 6 operation, 64 fresh and 4 once-burned BPRAs will be inserted, leaving 38 FAs with unplugged guide tubes, resulting in a decrease in maximum core bypass flow to 7.6%. This decreased bypass flow and the consequent increase in core flow establish the cycle 5 analysis as conservative for and applicable to cycle 6 operation. The reinserted BPRAs do not impact the thermal-hydraulic analysis. The cycle 5 and 6 maximum design conditions and significant parameters are shown in Table 6-1.

The Mark BZ low-absorption demonstration assembly will be limited to a 1.54 design peak to ensure that this assembly is not limiting. The 1.71 design radial-local peak remains valid for all other assemblies.

The fuel rod bow penalty for cycle 5 has been calculated with the procedure approved in reference 12. Using this procedure, the penalty is based on the maximum assembly burnup for the fuel batch containing the limiting (highest power) assembly. The calculated penalty is then reduced by 1% to reflect a generic credit, which is available because of the use of a flow area (pitch) reduction factor in the thermal-hydraulic analysis. For cycle 6 operation the net rod bow penalty that results from this calculation is 0.4% DNBR based on the maximum assembly burnup (19, [illegible]0 MWd/mtU) of batch 8 fuel. Additional analyses were performed for the limiting assemblies in fuel batches 6B and 7 (based on steady-state power distributions) to demonstrate that the increase in DNBR associated with the lower peaking of [illegible] assemblies (relative to the limiting batch 8 assembly) more than offsets the increased rod bow DNBR penalties that would be calculated on the basis of maximum assembly burnup values for these batches.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycles 5 and 6
Power level, MWt	2568
System pressure, psia	2200
Reactor coolant flow, % design flow	106.5
Vessel inlet coolant temp., 100% power, F	555.6
Vessel outlet coolant temp., 100% power, F	602.4
Ref design axial flux shape	1.5 cos
Ref design radial-local power peaking power	1.71
Active fuel length, in.	(a)
Average heat flux, 100% power, 10^3 Btu/h-ft^2	176[b]
CHF correlation	BAW-2
Hot channel factors	
Enthalpy rise	1.011
Heat flux	1.014
Flow area	0.98
Minimum DNBR with densification penalty	2.05

[a] See Table 4-2.

[b] Based on densified length of 140.3 inches.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 6 parameters to determine the effect of the cycle 6 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in reference 11. Since batch 8 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 11, the conclusions in that reference are still valid.

7.2. Accident Evaluation

The key parameters that have the greatest effect on the outcome of a transient can typically be classified in three major areas, core thermal parameters, thermal-hydraulic parameters, and kinetics parameters including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters for each batch in cycle 6 are compared in Table 4-2. A comparison of the cycle 6 thermal-hydraulic maximum design conditions to the previous cycle 5 values is presented in Table 6-1. The key kinetics parameters from the FSAR and cycle 6 are compared in Table 7-1.

A generic LOCA analysis has been performed for the B&W 177-FA, lowered-loop NSS using the Final Acceptance Criteria ECCS Evaluation Model; this study is reported in BAW-10103, Rev. 3.[13] The analysis in BAW-10103 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 LHR and the lifetime pin pressure data used in the BAW-10103 LOCA limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 2 cycle 6 fuel, except as noted in the following paragraph.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 2 cycle 6 fuel after 50 EFPD. The LOCA kW/ft limits have been reduced for the first 50 EFPD. The reduction will ensure that conservative limits are maintained while a transition is being made in the fuel performance codes that provide input to the ECCS analysis[14] in order to account for mechanistic fuel densification. The limits for the first 50 EFPD are shown in Table 7-3.

From the examinations of cycle 6 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the ability to operate the Oconee 2 plant safely during cycle 6. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 6 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 6 are bounded by the FSAR[1], the fuel densification report[11], and/or subsequent cycle analyses.

The radiological dose consequences of the accidents presented in Chapter 14 of the FSAR were re-evaluated for this reload. The reason for the re-evaluation is that, even though the FSAR dose analyses used a conservative basis for the amount of plutonium fissioning in the core, improvements in fuel management techniques have increased the amount of energy produced by fissioning plutonium. Since plutonium-239 has different fission yields than uranium-235, the mixture of fission product nuclides in the core changes slightly as the plutonium-239/uranium-235 fission ratio changes, i.e., plutonium fissions produce more of some nuclides and less of others. The general trend is that more plutonium fissions tend to produce slightly higher thyroid doses and slightly lower whole body doses. But, since the radiological doses associated with each accident are impacted to a different extent by each nuclide and by various mitigating factors and plant design features, the radiological consequences of the FSAR accidents were recalculated using the specific cycle 6 parameters in order to obtain an accurate evaluation of the effects of the increase in the amount of plutonium fissioning. The bases used in the dose calculations are identical to those presented in the FSAR except for the following two notable differences:

1. The fission yields and half-lives used in the new calculations are based on more current data.

2. The steam generator tube rupture accident evaluation considers the amount of steam released to the environment via the main steam relief valves because of the slower depressurization due to the reduced heat transfer rate caused by tripping of the reactor coolant pumps upon actuation of the high-pressure injection (a post-TMI-2 modification).

Table 7-4 shows the radiological doses presented in the FSAR and those calculated specifically for cycle 6. As can be seen from the table, some doses are slightly higher and some are slightly lower than the FSAR values; however, all doses are well below the 10 CFR 100 limits of 300 Rem to the thyroid and 25 Rem to the whole body. The small increases in some doses are essentially offset by a reduction in others. Thus, the radiological impact of accidents during cycle 6 would not be significantly different than those described in Chapter 14 of the FSAR.

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

Parameter	FSAR and densification report value	Predicted cycle 6 value
BOL Doppler coeff, 10^{-5} Δk/k-°F	-1.17[a]	-1.52
EOL Doppler coeff, 10^{-5} Δk/k-°F	-1.33	-1.77
BOL moderator coeff, 10^{-4} Δk/k-°F	+0.5[b]	-0.63
EOL moderator coeff, 10^{-4} Δk/k-°F	-3.0	-2.98
All rod bank worth, HZP, % Δk/k	10.0	9.04
Boron reactivity worth (cold), ppm/1%(Δk/k)	75	87
Max ejected rod worth, HFP, % Δk/k	0.65	0.32
Dropped rod worth, HFP, % Δk/k	0.46	0.20
Initial boron conc, HFP, ppm	1400	1344

(a) -1.2×10^{-5} Δk/k-°F was used for steam line failure analysis, and -1.3×10^{-5} Δk/k-°F was used for cold water analysis.

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

Table 7-2. LOCA Limits, Oconee 2 Cycle 6, After 50 EFPD

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

Table 7-3. LOCA Limits, Oconee 2 Cycle 6, 0 to 50 EFPD

Core elevation, ft	Allowable peak LHR, kW/ft
2	14.5
4	16.1
6	17.5
8	17.0
10	16.0

Table 7-4. Comparison of FSAR and Cycle 6 Accident Doses

Accident	FSAR doses, Rems	Cycle 6 doses, Rems
Fuel handling		
Thyroid at EAB	0.43	0.50
Whole body at EAB	0.027	0.028
Steam generator tube failure		
Thyroid at EAB	0.00034	0.310
Whole body at EAB	0.023	0.058
Waste gas tank		
Thyroid at EAB	0.13	0.27
Whole body at EAB	0.19	0.17
Rod ejection accident		
Thyroid at EAB	0.19	0.21
Whole body at EAB	0.001	0.0005
Thyroid at LPZ	0.22	0.23
Whole body at LPZ	(a)	0.0007
Steam line break		
Thyroid at EAB	0.19	0.20
Whole body at EAB	0.002	0.002
LOCA		
Thyroid at EAB	4.6	5.0
Whole body at EAB	0.01	0.01
Thyroid at LPZ	5.0	5.5
Whole body at LPZ	0.014	0.014
MHA		
Thyroid at EAB	186.	193.
Whole body at EAB	1.4	1.4
Thyroid at LPZ	144.	180.
Whole body at LPZ	0.65	0.62

(a) Not reported in the FSAR.

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications for cycle 6 operation have been revised in accordance with the mothods of references 15 through 17 to account for power peaking and control rod worths inherent in an extended, lumped burnable poison cycle.

In addition:

1. A high flux trip setpoint of 104.9% of rated power, and a high RC outlet temperature trip of 618F have been established for cycle 6 operation.
2. The 0-50 EFPD operating limits on rod index, APSR position, and axial power imbalance were established based on the interim LOCA linear heat rate limits, which account for mechanistic fuel densification[14]. After 50 EFPD, the FAC LOCA LHR limits were used.[13]

Based on the Technical Specifications derived from the analysis 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 8-19 are revisions to previous Technical Specification limits.

Figure 8-1. Core Protection Safety Limits for Oconee Unit 2, Cycle 6

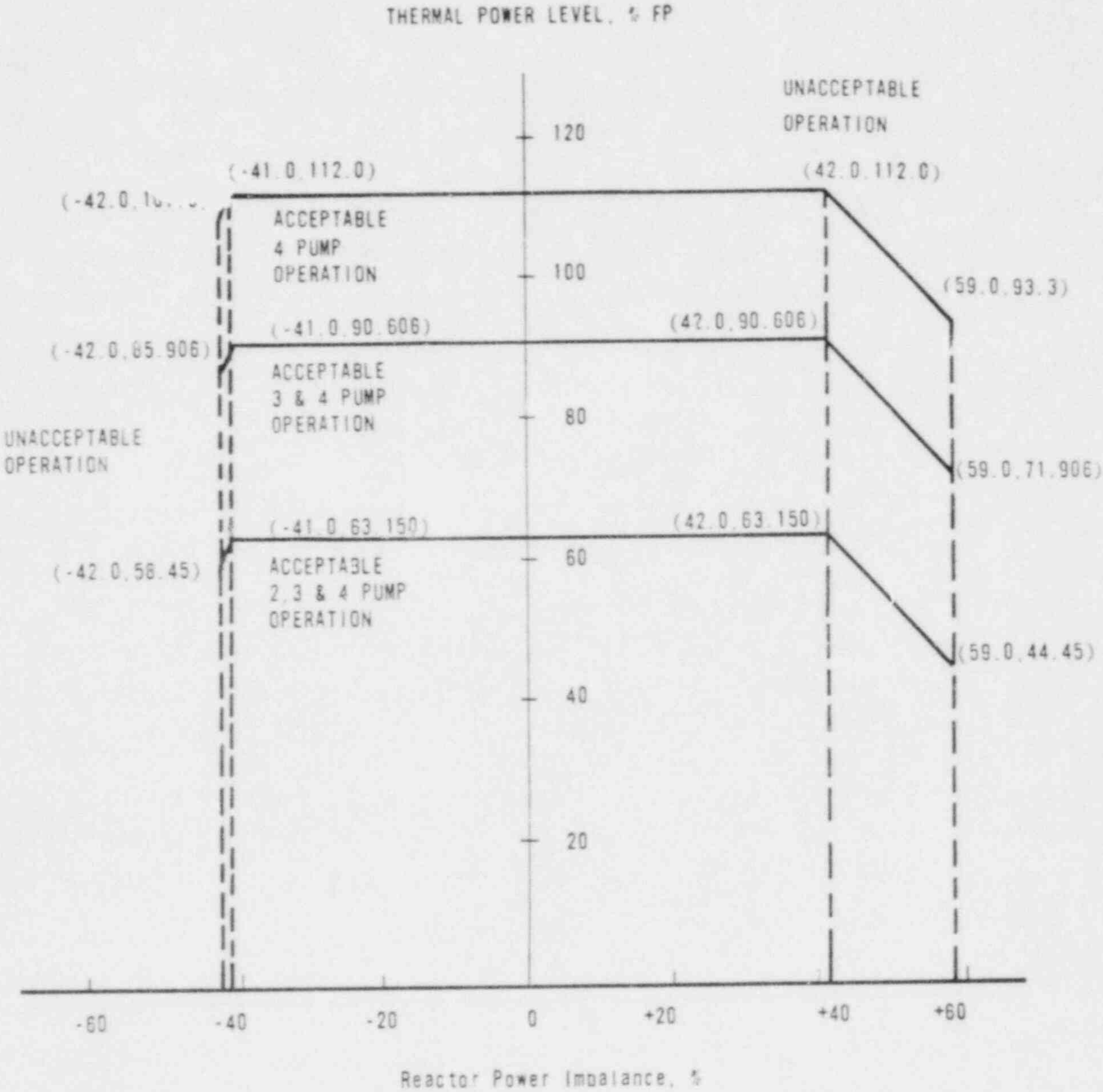
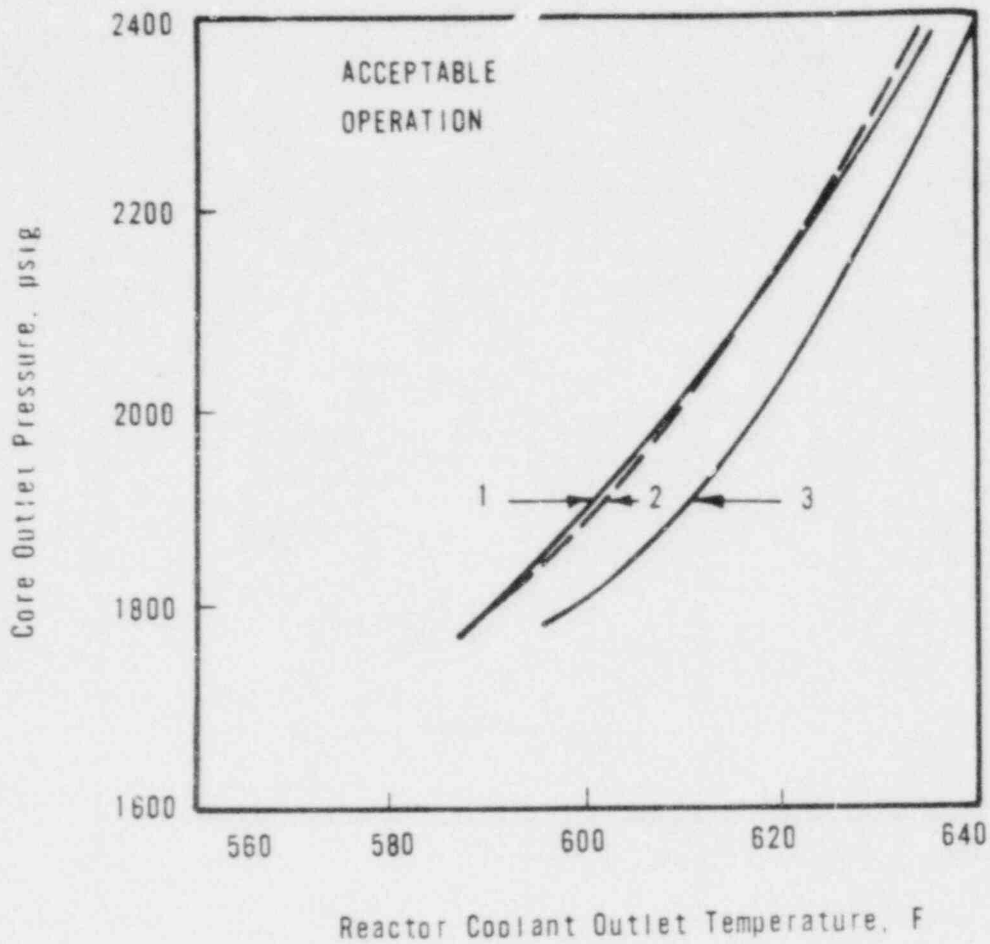


Figure 8-2. Core Protection Safety Limits for Oconee Unit 2, Cycle 6



CURVE	COOLANT FLOW, GPM	POWER, %	PUMPS OPERATING	TYPE OF LIMIT
1	374,880 (100%)	112	4	DNBR
2	280,035 (74.7%)	90.606	3	DNBR
3	183,690 (49.0%)	63.150	1 PER LOOP	QUALITY

Figure 8-3. Maximum Allowable Setpoints for Oconee Unit 2, Cycle 6

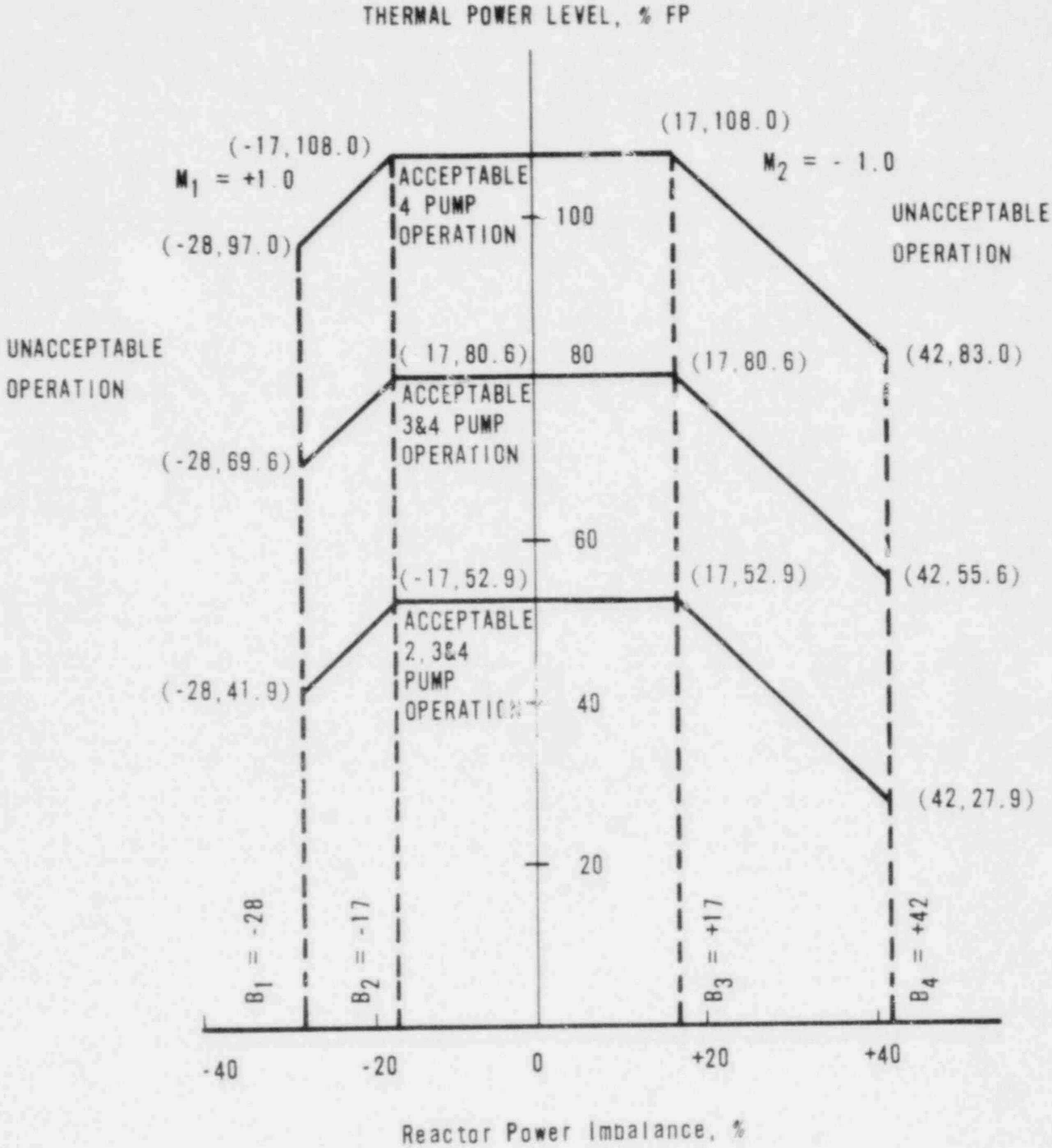


Figure 8-4. Protective System Maximum Allowable Setpoints, Oconee 2 Cycle 6

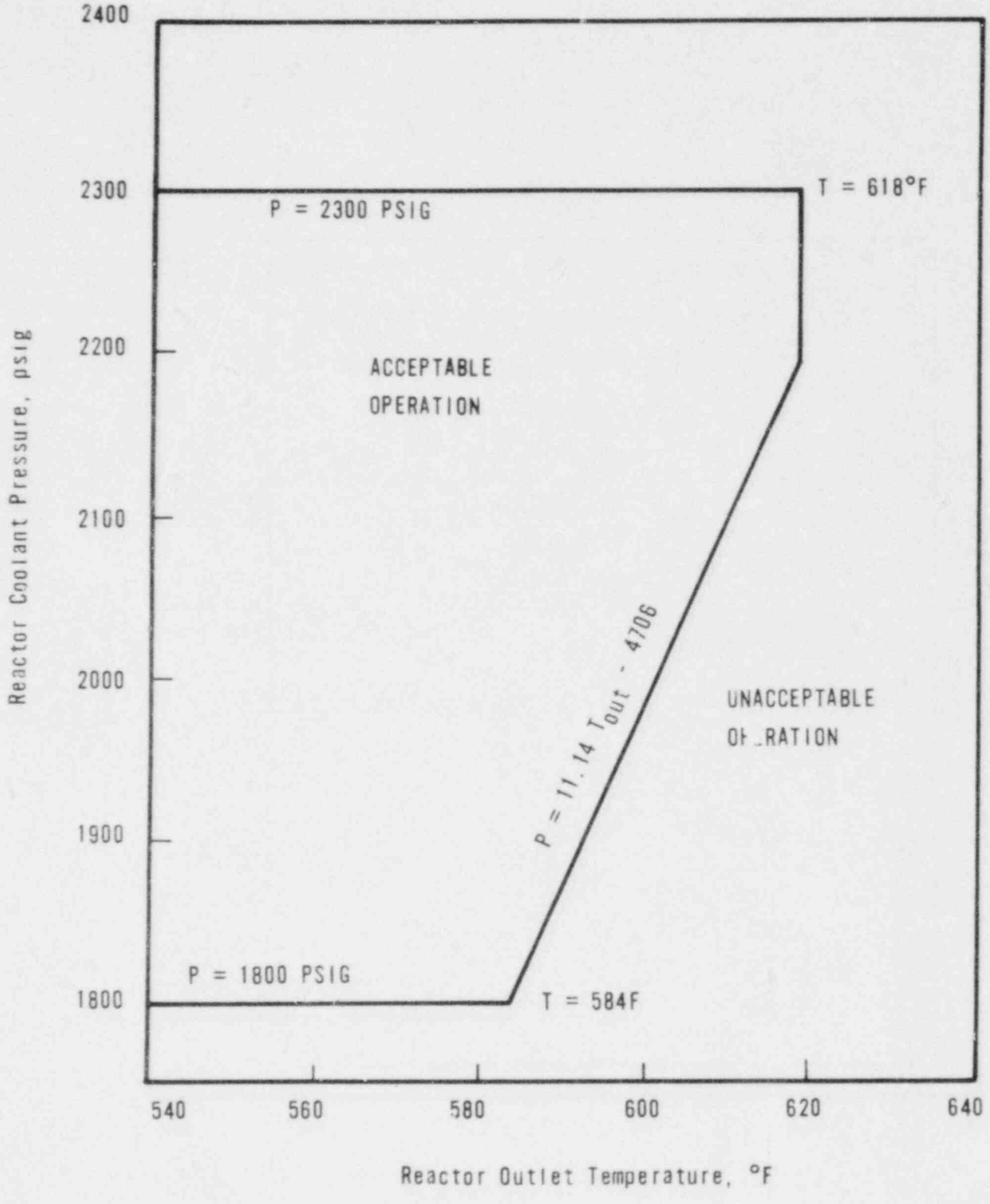


Figure 8-5. Oconee 2 Cycle 6 Rod Position Limits — Four-Pump Operation, 0 to 50 ± 10 EFPD

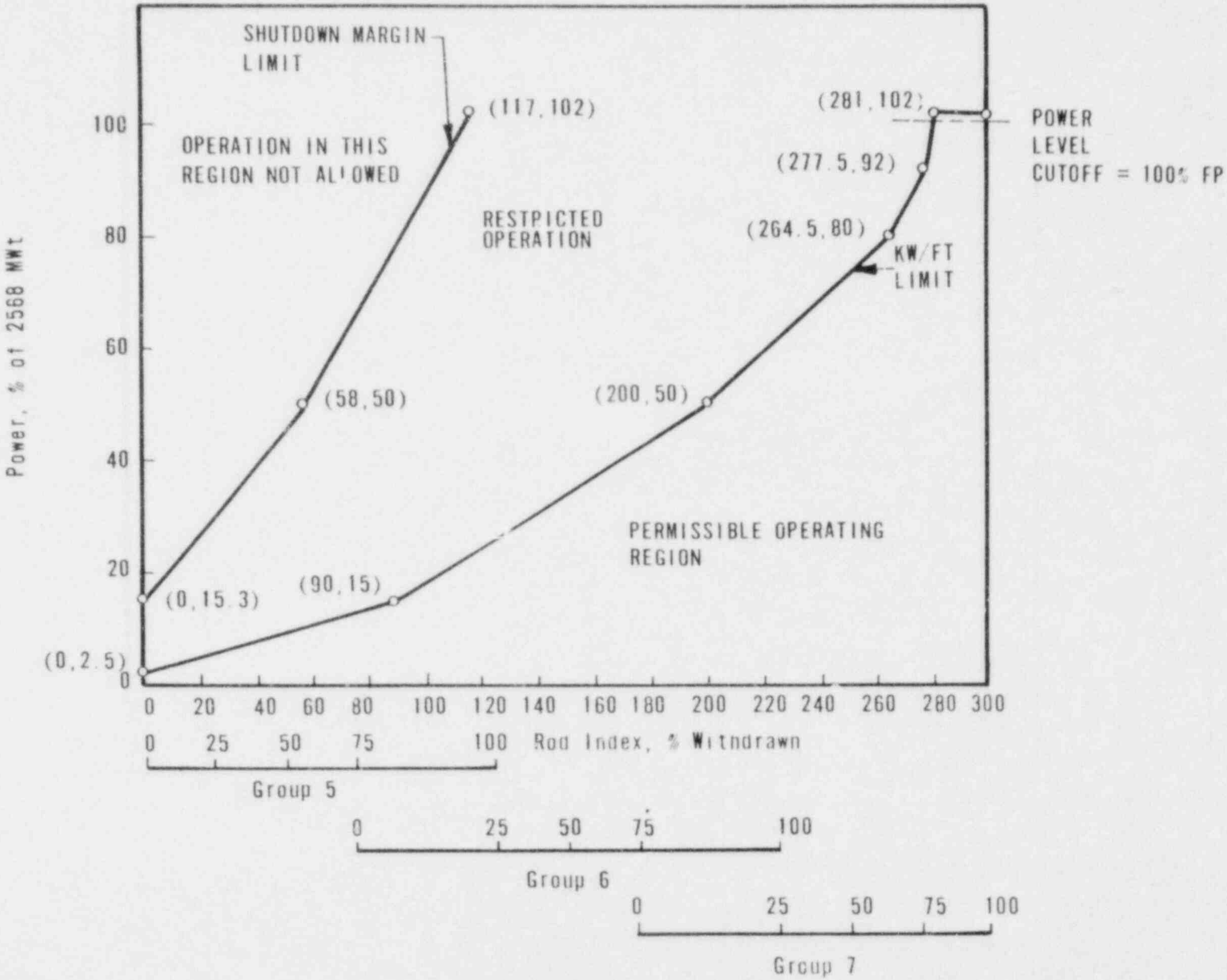


Figure 8-6. Oconee 2 Cycle 6 Rod Position Limits — Four-Pump Operation, 50 ± 10 to 225 ± 10 EFPD

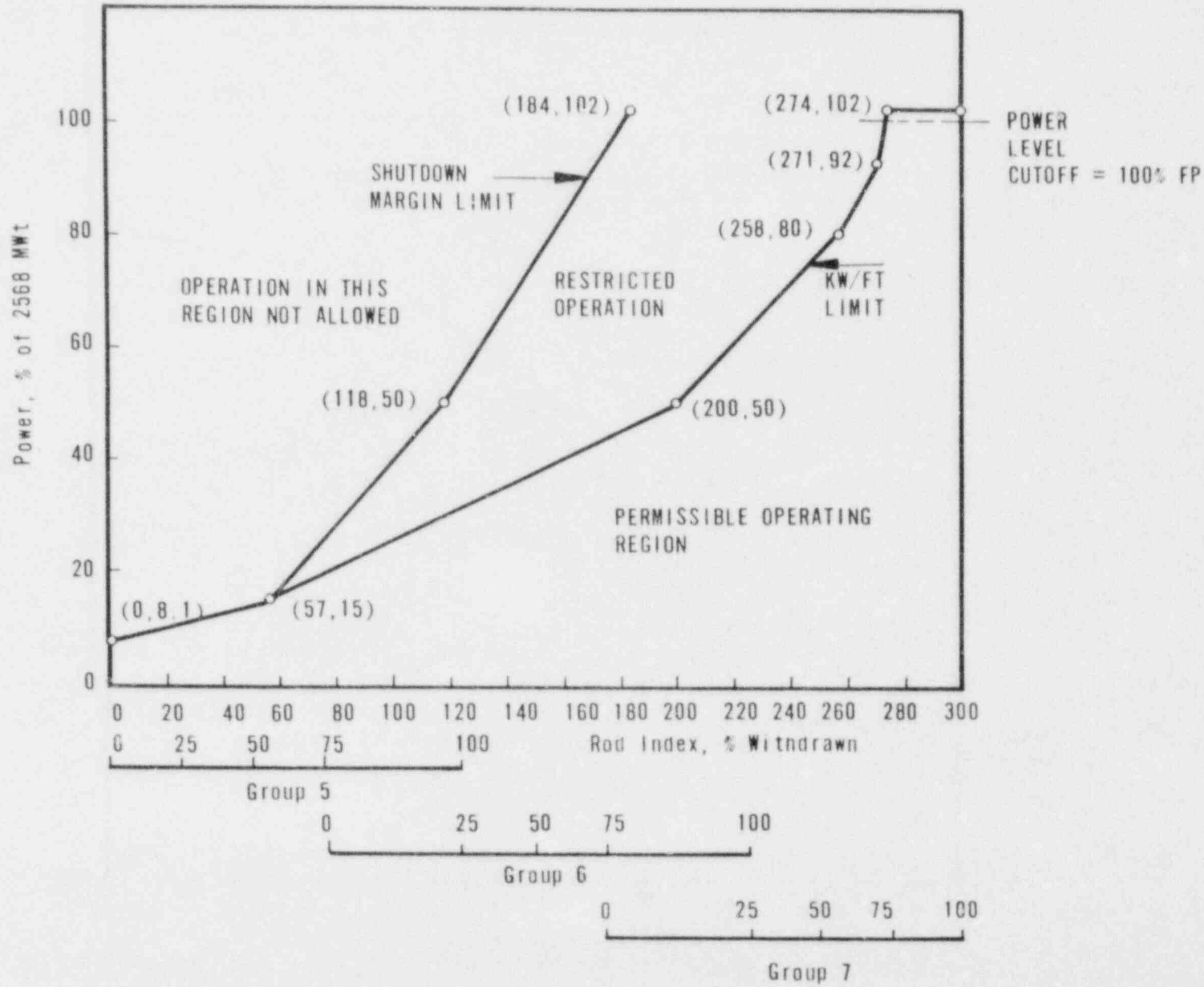


Figure 8-7. Oconee 2 Cycle 6 Rod Position Limits — Four-Pump Operation After 225 ± 10 EFPD

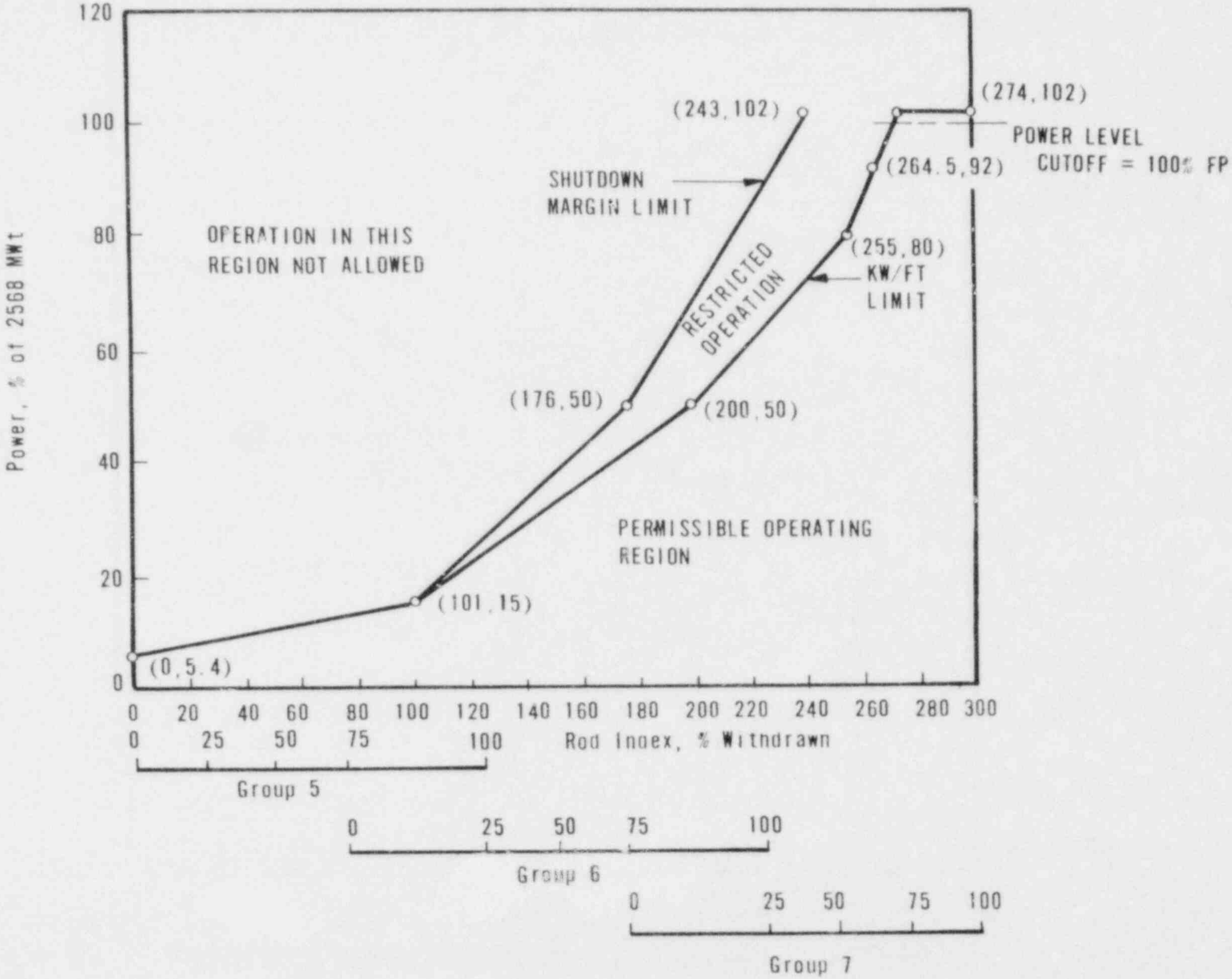


Figure 8-8. Oconee 2 Cycle 6 Rod Position Limits — Three-Pump Operation, 0 to 50 ± 10 EFPD

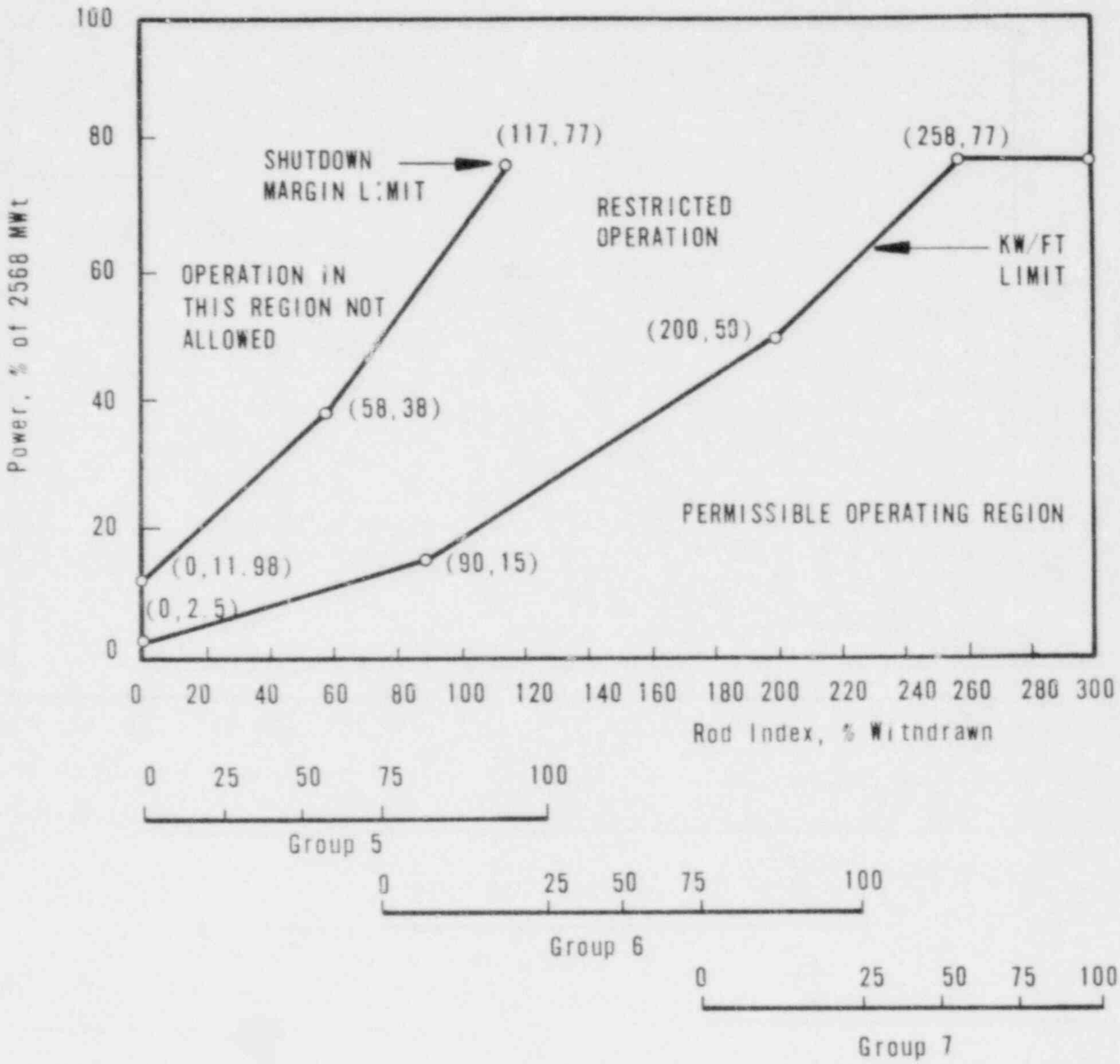


Figure 8-9. Oconee 2 Cycle 6 Rod Position Limits — Three-Pump Operation From 50 ± 10 to 225 ± 10 EFPD

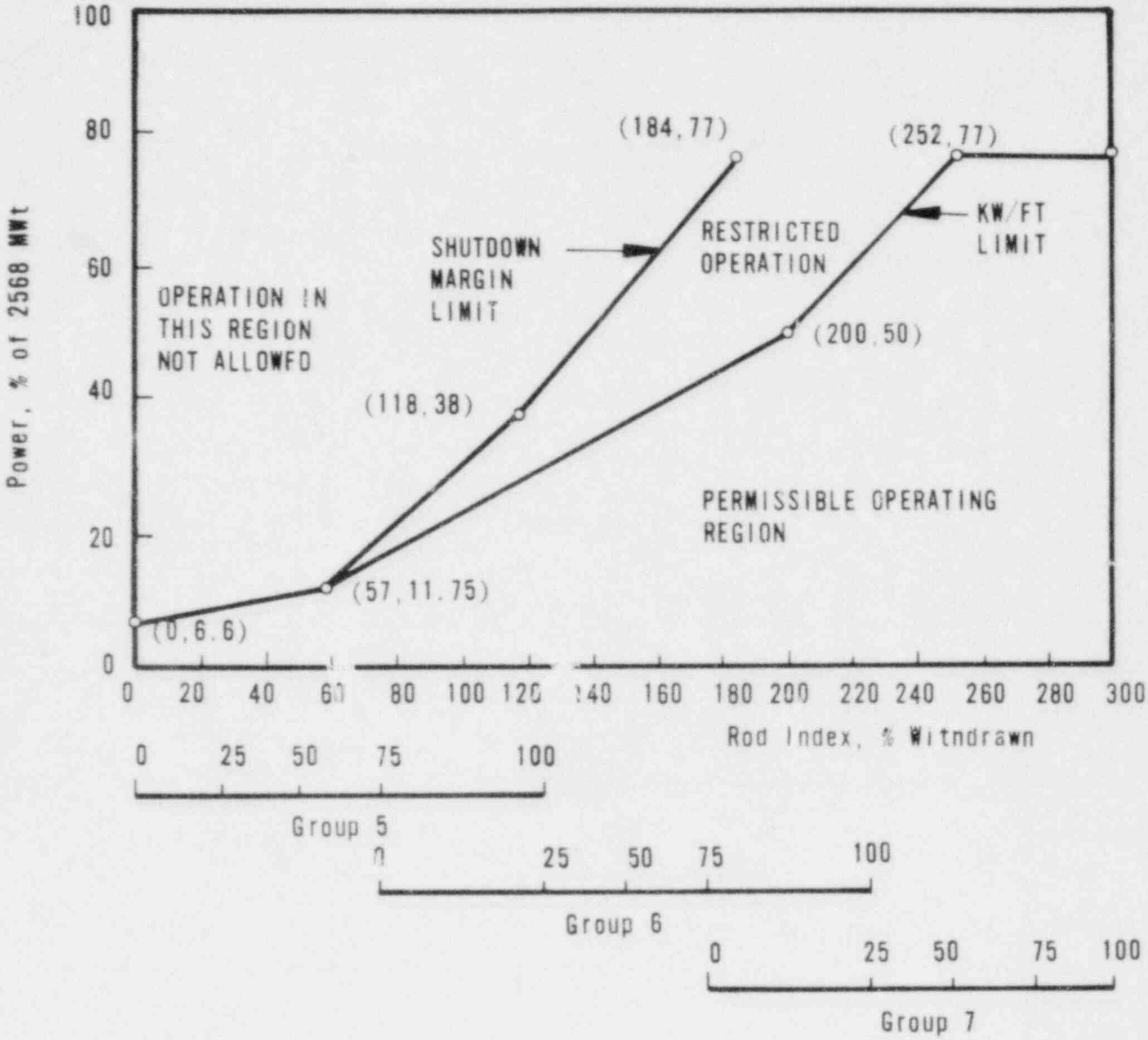


Figure 8-10. Oconee 2 Cycle 6 Rod Position Limits — Three-Pump Operation After 225 ± 10 EFPD

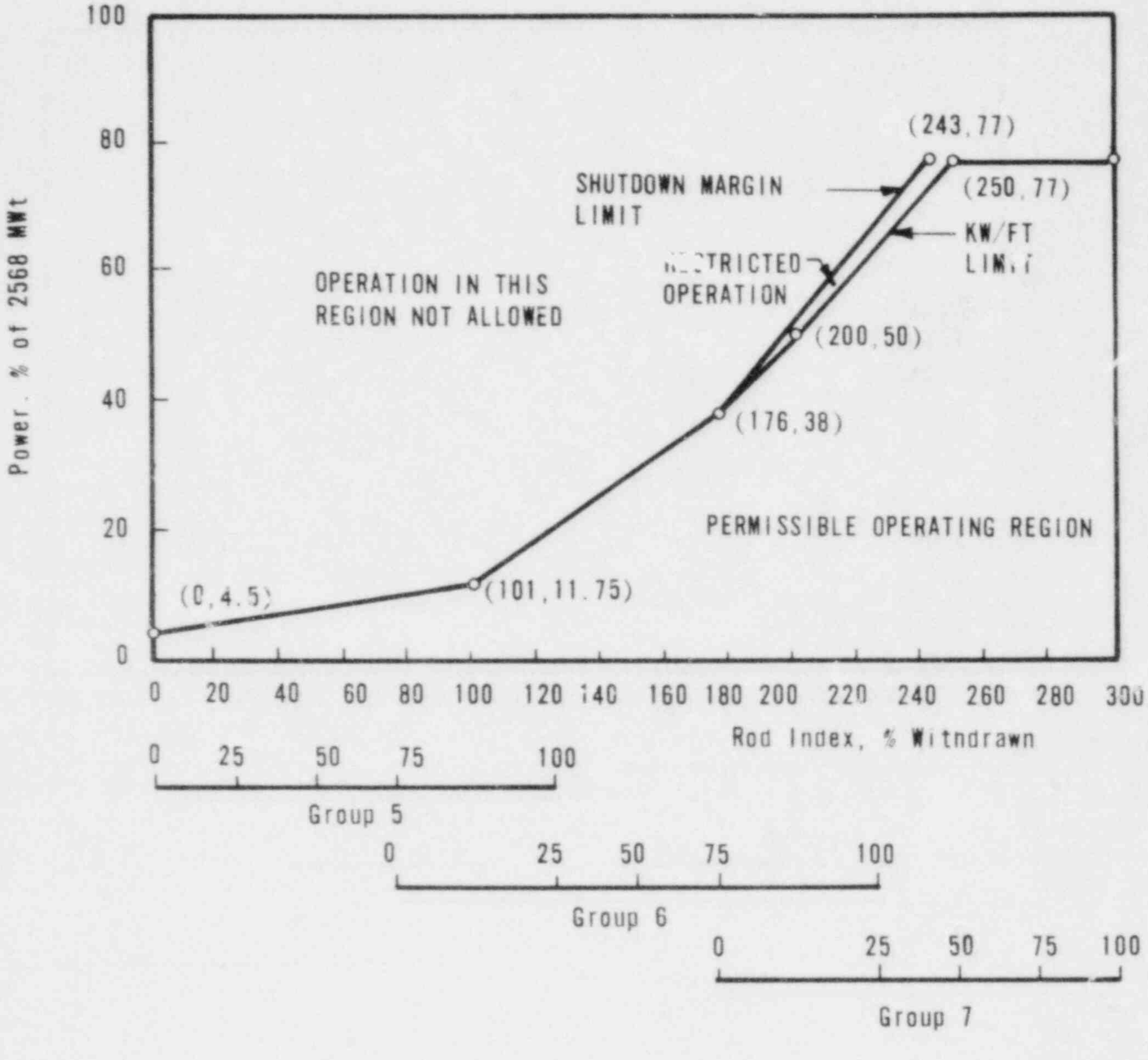


Figure 8-11. Oconee 2 Cycle 6 Rod Position Limits — Two-Pump Operation, 0 to 50 ± 10 EFPD

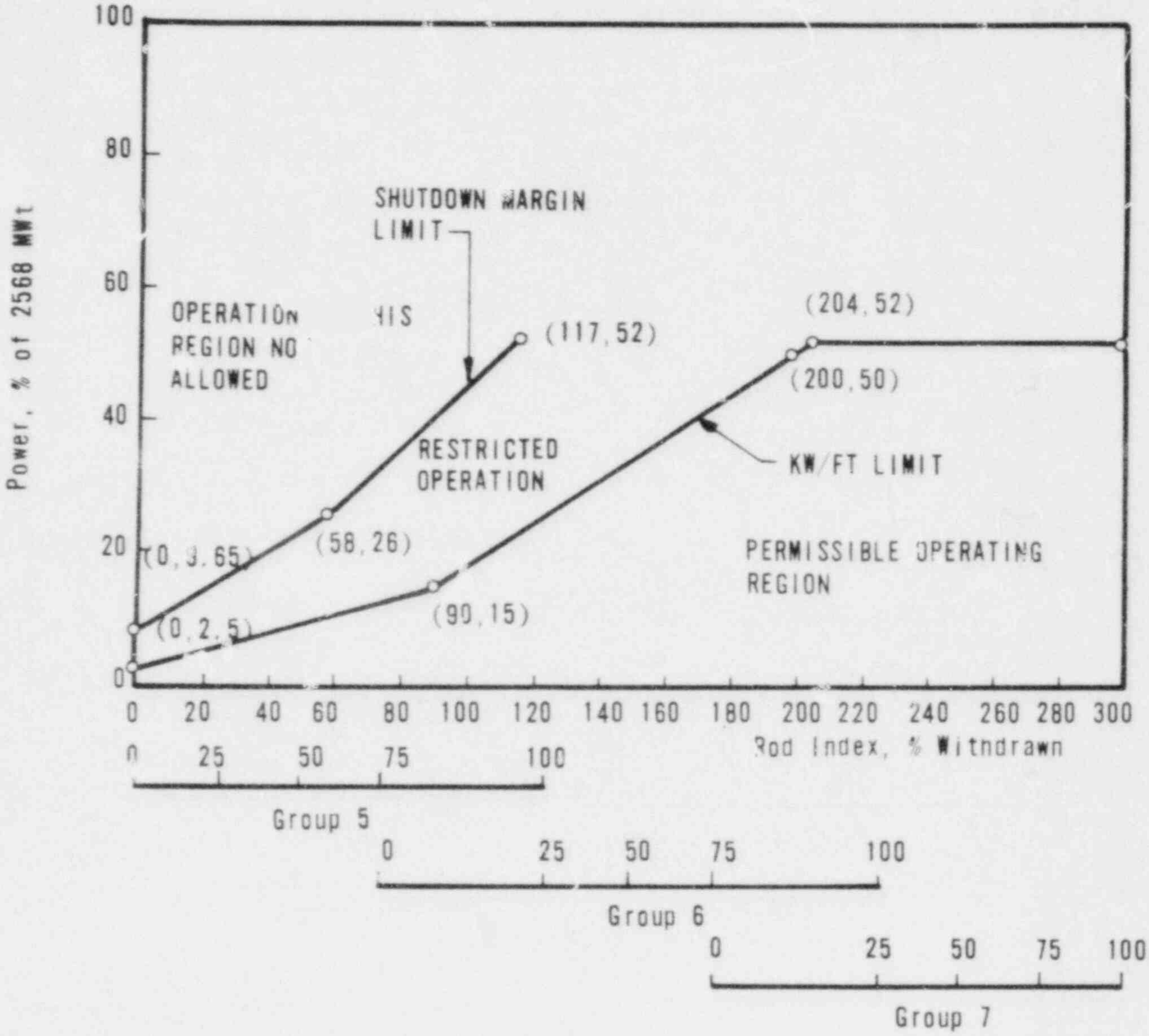


Figure 8-12. Oconee 2 Cycle 6 Rod Position Limits — Two-Pump Operation From 50 ± 10 to 225 ± 10 EFPD

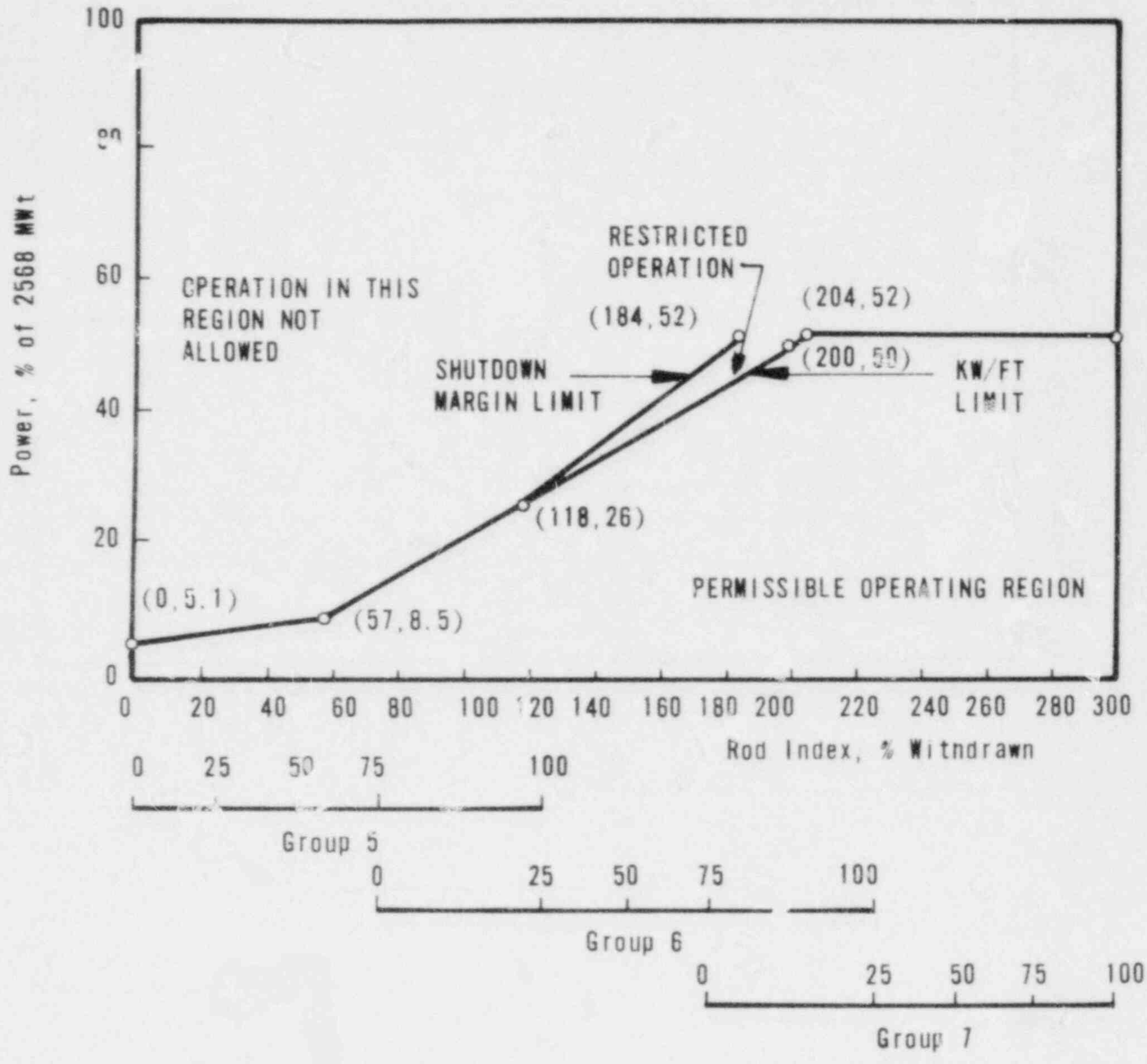


Figure 8-13. Oconee 2 Cycle 6 Rod Position Limits — Two-Pump Operation After 225 ± 10 EFPD

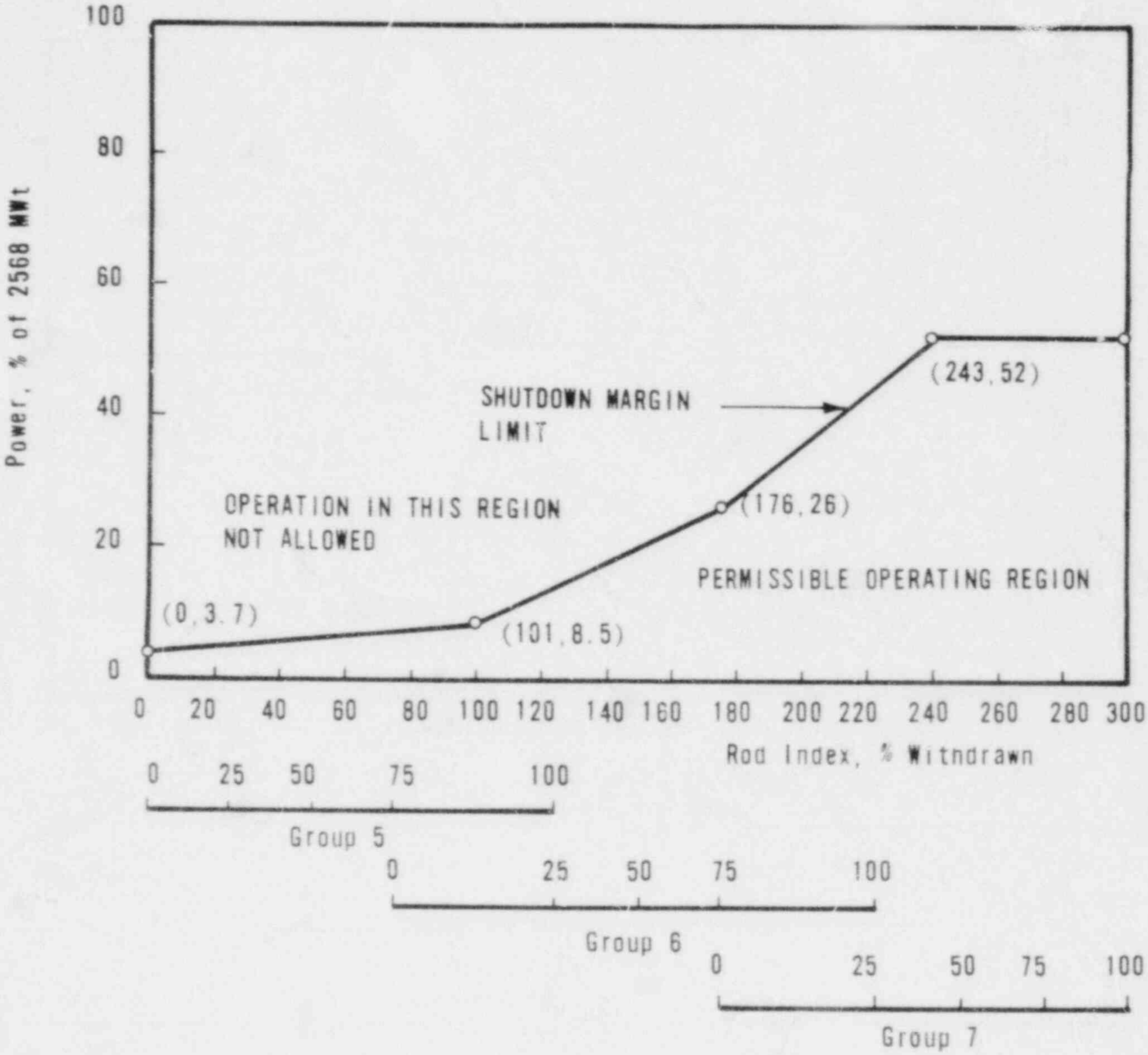


Figure 8-14. Oconee 2 Cycle 6 Operational Power Imbalance Limits, 0 to 50 ± 10 EFPD

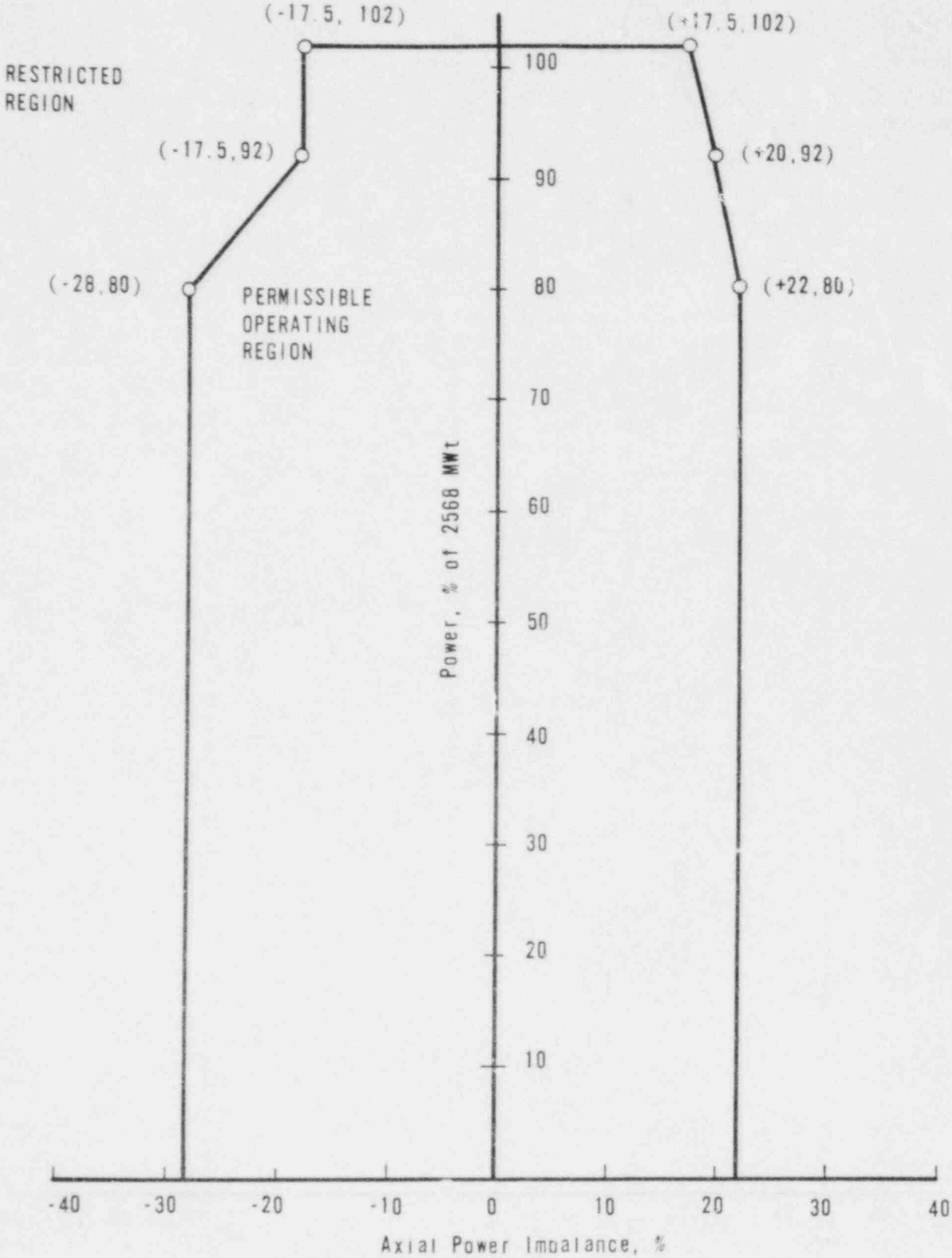


Figure 8-15. Oconee 2 Cycle 6 Operational Power Imbalance Limits, 50 ± 10 to 225 ± 10 EFPD

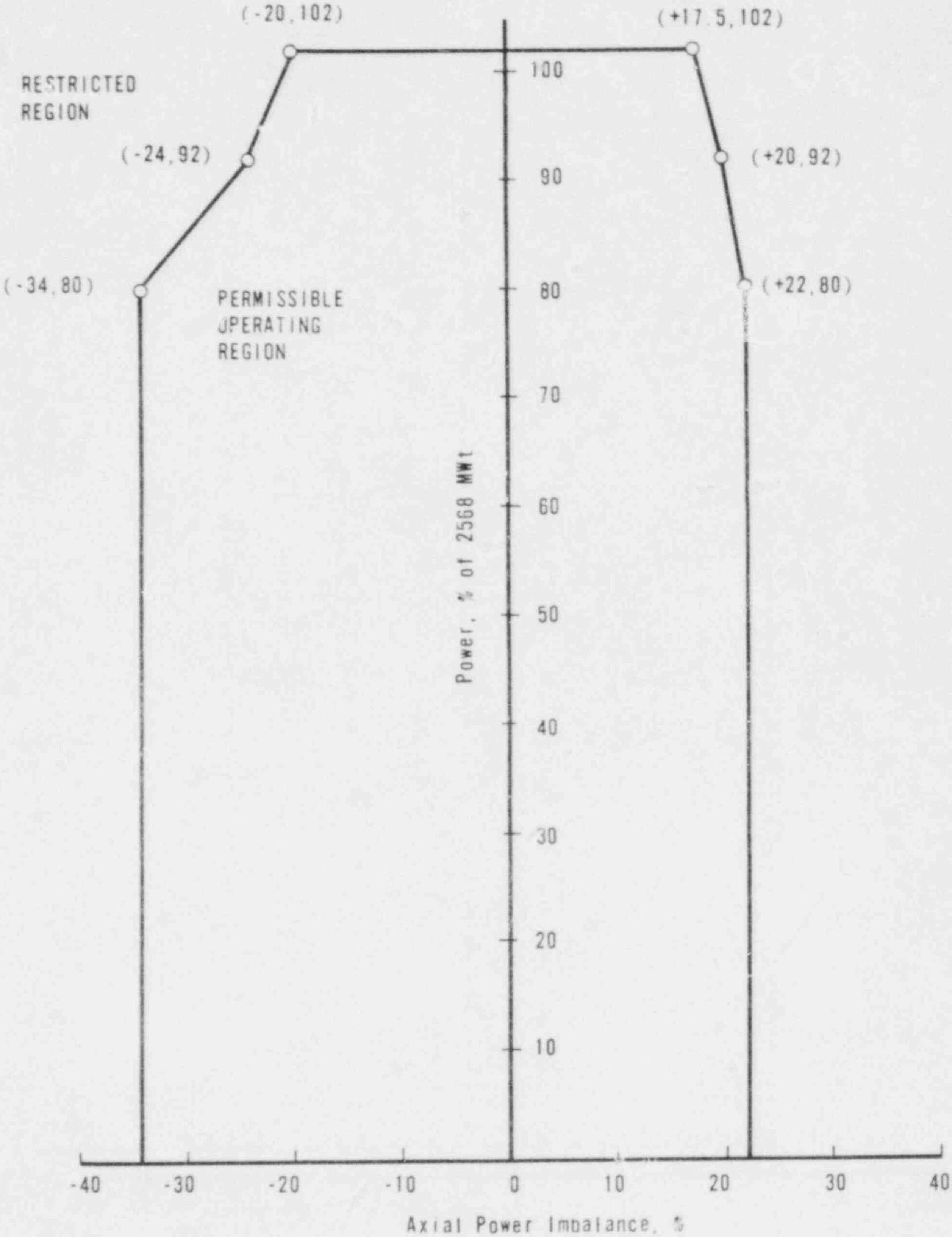


Figure 8-16. Oconee 2 Cycle 6 Operational Power Imbalance Limits After 225 ± 10 EFPD

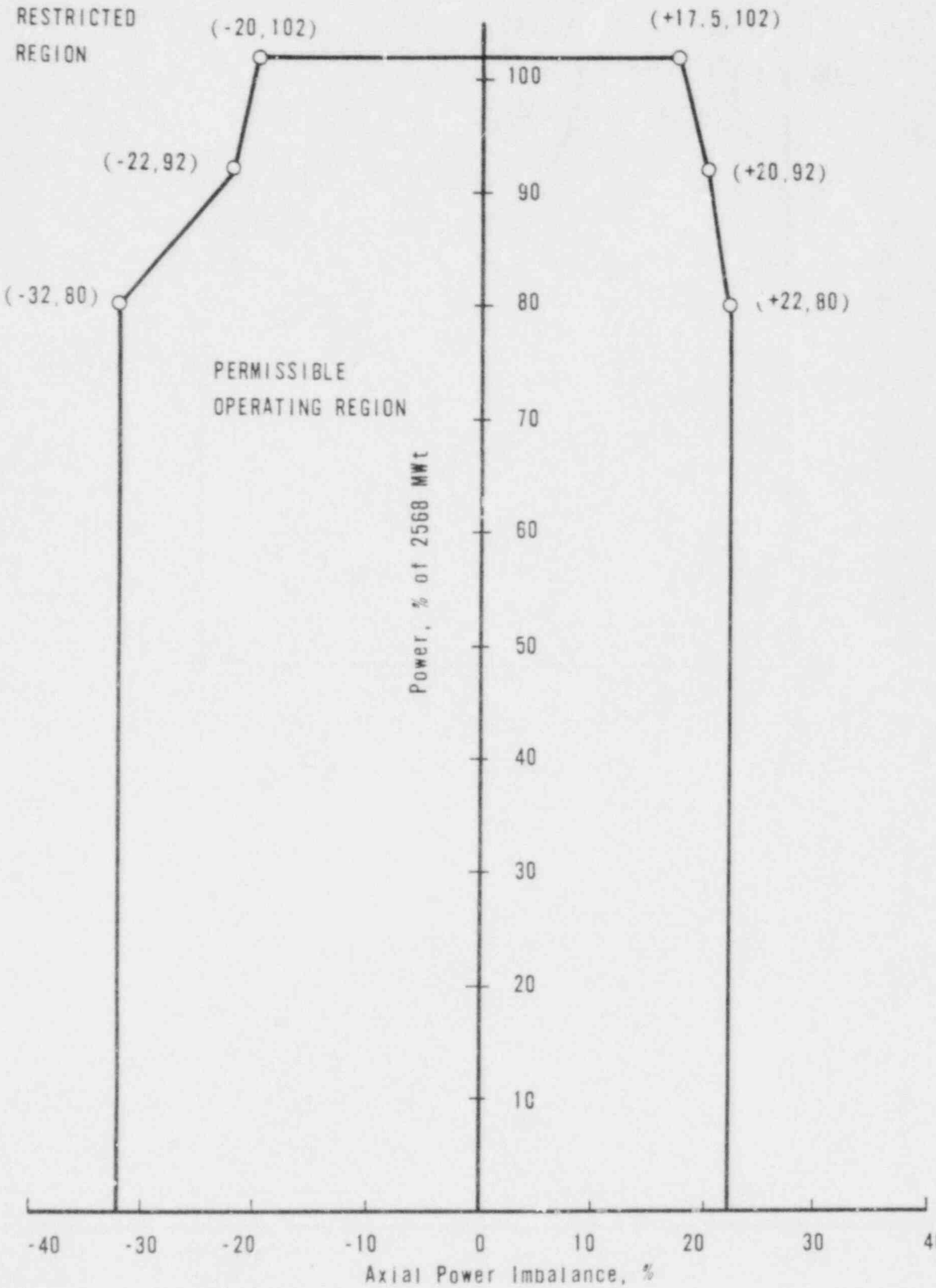


Figure 8-17. Oconee 2 Cycle 6 APSR Position Limits, 0 to 50 ± 10 EFPD

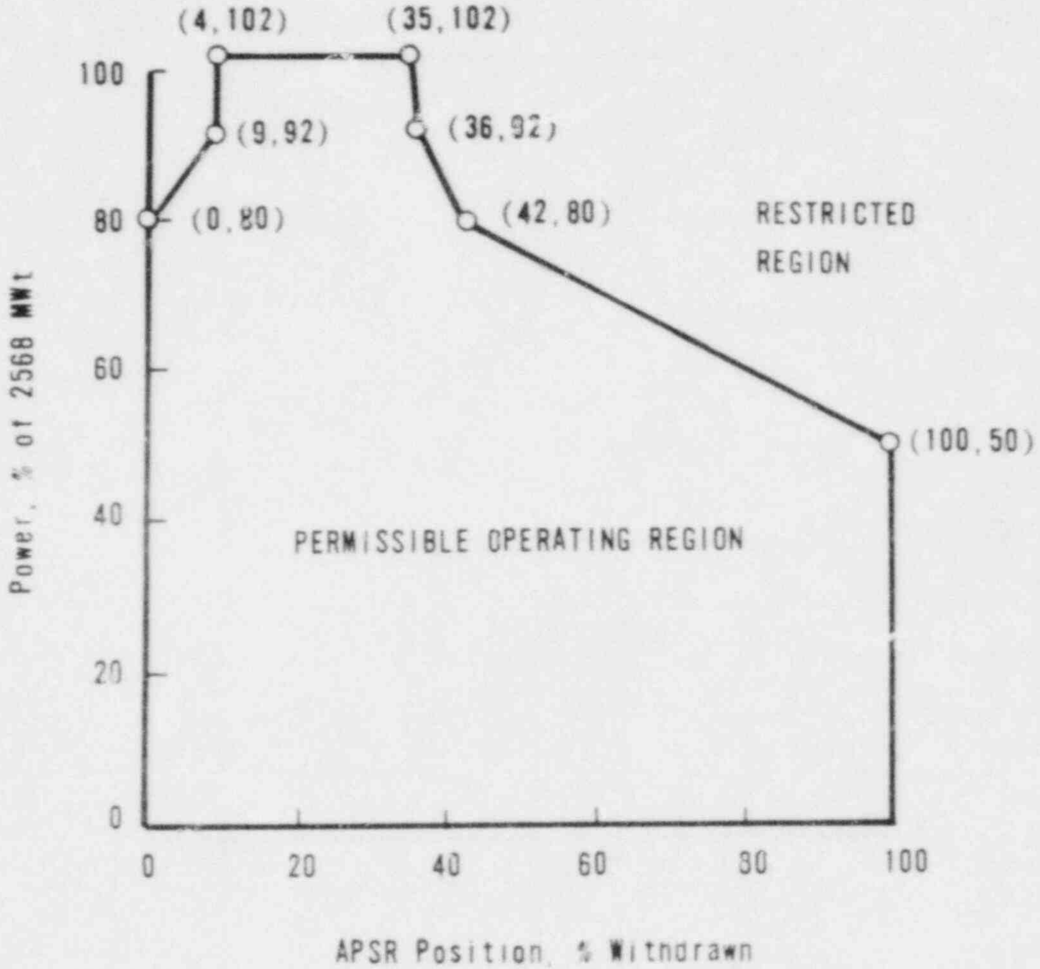


Figure 8-18. Oconee 2 Cycle 6 APSR Position Limits,
50 ± 10 to 225 ± 10 EFPD

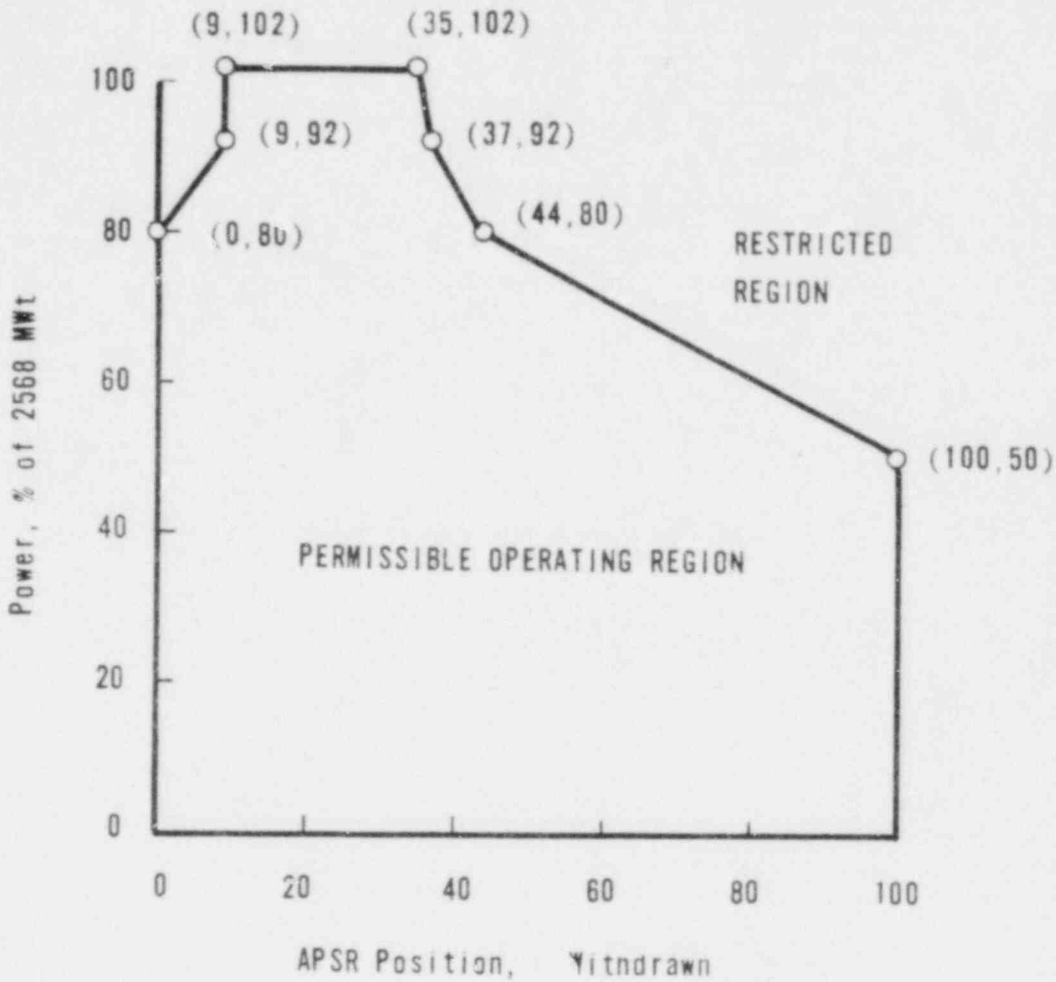
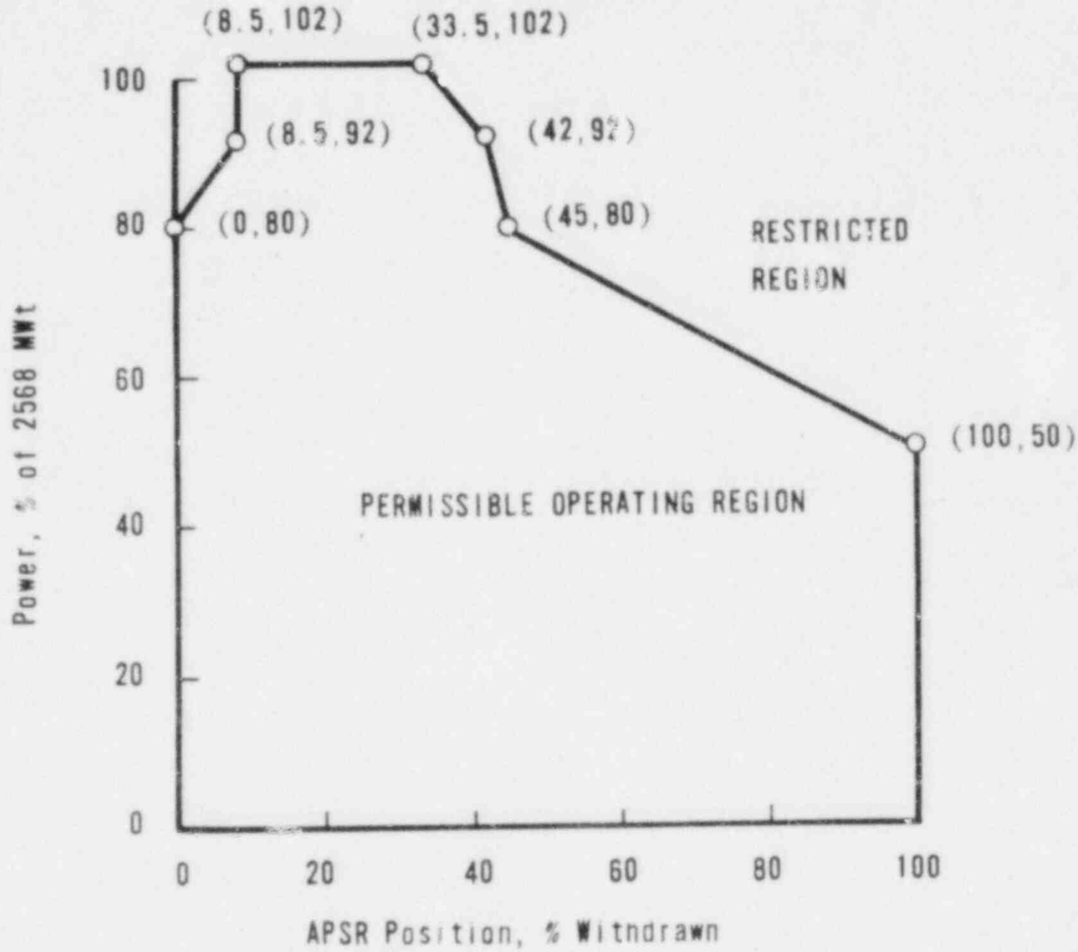


Figure 8-19. Oconee 2 Cycle 6 APSR Position Limits After 225 ± 10 EFPD



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