BAW-1634

August 1980

OCONEE UNIT 3, CYCLE 6

— Reload Report —

BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505

8009030519

Babcock & Wilcox

CONTENTS

List of Tables

List of Figures

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth 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 6 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 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.

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 3 can be operated safely for cycle 6 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 3, cycle 6, is the currently operating cycle 5. Cycle 4 was terminated after 263 EFPD of operation. Cycle 5 achieved initial criticality on October 28, 1979, and power escalation commenced on October 30, 1979. The fuel cycle design length for cycle 6 — 376 EFPD — is based on cycle 5 length of 299 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in cycle 6.

Cycle 6 will operate in a feed-and-bleed mode for its entire design length, as did cycle 5.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 3 of the FSAR.[1] The cycle 6 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 consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Tables 4-1 and 4-2.

Figure 3-1 is the core loading diagram for Oconee 3, cycle 6. Thirty-nine of the batch 5 assemblies will be discharged at the end of cycle 5 along with batches 1D, 4B, and 4C. The remaining 17 batch 5 assemblies, designated "5B," batch 6, and the fresh batch 8 FAs — with initial enrichments of 3.02, 2.97, and 3.07 wt % ^{235}U, respectively — will be loaded into the central portion of the core. Batch 7, with an initial enrichment of 2.80 wt % ^{235}U, will occupy primarily the core periphery as in cycle 5. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 6.

Cycle 6 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 60 burnable poison rod assemblies (BPRAs). 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 6 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the CRAs and APSRAs for cycle 6 are identical to those of the reference cycle; however, the group designations differ between cycle 6 and the reference cycle to minimize power peaking. The cycle 6 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Core Loading Diagram for Oconee 3 Cycle 6

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						R10 7	R9 7	R8 7	R7 7	R6 7					
B				P12 7	8	O12 7	8	P5 7	8	O4 7	8	P4 7			
C			O13 7	8	D6 6	8	K2 5B	8	K14 5B	8	D10 6	8	O3 7		
D		N14 7	8	E5 5B	8	F14 7	8	M8 6	8	F2 7	8	E11 5B	8	N2 7	
E		8	F4 6	8	C7 6	8	K6 6	8	K10 6	8	G13 6	8	F12 6	8	
F	L15 7	N13 7	8	P6 7	8	G3 6	L5 6	N11 5B	L11 6	C9 6	8	P10 7	8	N3 7	L1 7
G	K15 7	8	B9 5B	8	F9 6	E10 6	P11 7	8	M2 7	E6 6	F7 6	8	B7 5B	8	K1 7
H	H15 7	M14 7	8	H11 6	8	E12 5B	8	M12 5B	8	M4 5B	8	H5 6	8	E2 7	H1 7
K	G15 7	8	P9 5B	8	L9 6	M10 6	E14 7	8	B5 7	M6 6	L7 6	8	P7 5B	8	G1 7
L	F15 7	D13 7	8	B6 7	8	O7 6	F5 6	D5 5B	F11 6	K13 6	8	B10 7	8	D3 7	F1 7
M		8	L4 6	8	K3 6	8	G6 6	8	G10 6	8	O9 6	8	L12 6	8	
N		D14 7	8	M5 5B	8	L14 7	8	E8 6	8	L2 7	8	M11 5B	8	D2 7	
O			C13 7	8	N6 6	8	G2 5B	8	G14 5B	8	N10 6	8	C3 7		
P				B12 7	8	C12 7	8	B11 7	8	C4 7	8	B4 7			
R						A10 7	A9 7	A8 7	A7 7	A6 7					

W — Y

Z

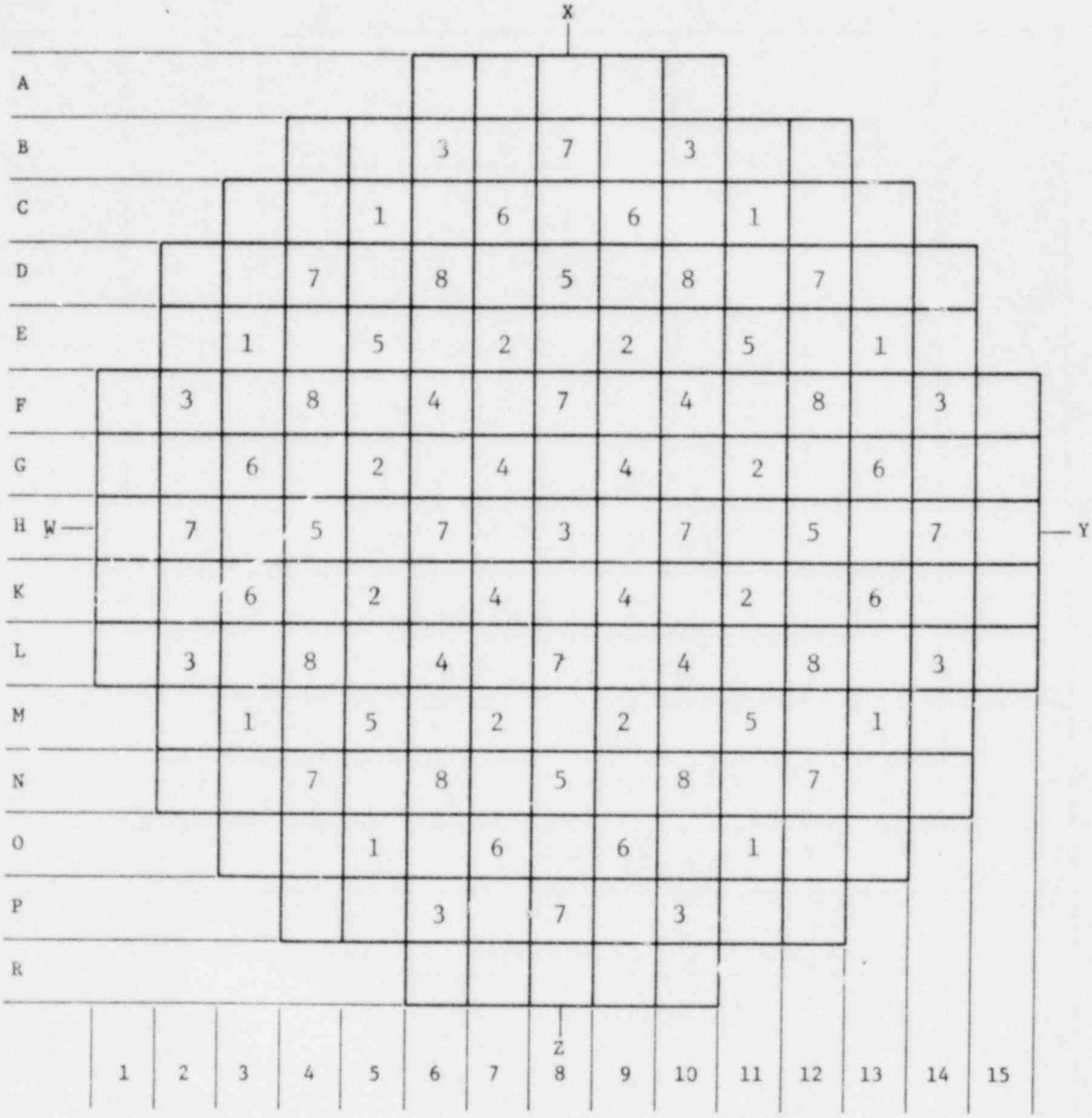
XX	Cycle 5 Location
X	Batch No.

Figure 3-2. Enrichment and Burnup Distribution for Oconee 3, Cycle 6

	8	9	10	11	12	13	14	15
H	3.02 23,433	3.07 0	3.02 23,419	3.07 0	2.97 18,811	3.07 0	2.80 8,410	2.80 7,675
K		2.80 8,407	2.97 20,308	2.97 18,173	3.07 0	3.02 23,215	3.07 0	2.80 7,726
L			2.97 17,748	3.07 0	2.80 10,747	3.07 0	2.80 9,513	2.80 6,438
M				2.97 17,740	3.07 0	2.97 17,250	3.07 0	
N					3.02 21,123	3.07 0	2.80 5,978	
O						2.80 6,772		
P								
R								

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

Figure 3-3. Control Rod Locations for Oconee 3, Cycle 6



| X | ← Group No.

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

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

	8	9	10	11	12	13	14	15
H		1.0		0.5		1.0		
K					1.0		0.5	
L				1.0		0.5		
M					1.0			
N						0.2		
O								
P								
R								

X.X	BPRA Concentration, wt % B_4C in Al_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3, cycle 6, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Retainer assemblies will be used on the 60 fuel assemblies containing BPRAs to provide positive retention during reactor operation. Similar retainer assemblies will be used on the two FAs containing the regenerative neutron sources. The justification for the design and use of the BPRA retainers is described in reference 3, which is also applicable to the RNS retainers of Oconee 3, cycle 6.

Other results presented in the FSAR[1] fuel assembly mechanical discussions and in previous reload reports are applicable to the reload fuel assemblies.

4.2. Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

The fuel of batches 6 and 5B is more limiting than other batches due to its longer previous incore exposure time. The batch 6 abd 5B assembly power histories were analyzed, and the most limiting assembly was used to perform the creep collapse analysis using the CROV computer code and procedures described in topical report BAW-10084, Rev. 2.[2] The collapse time for the most limiting assembly was conservatively determined to be more than 30,000 EFPH, which is greater than the maximum projected residence time of cycle 6 fuel (Table 4-1).

4.2.2. Cladding Stress

The Oconee 3, cycle 6 stress parameters are enveloped by a conservative fuel rod stress analysis. For design evaluation, the primary membrane stress must be less than 2/3 of the minimum specified unirradiated yield strength. In all cases, the margin is in excess of 30%. The following conservatisms with respect to Oconee fuel were used in the analysis:

1. A lower post-densification fuel rod internal pressure.
2. A 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 circumferential plastic strain. The pellet design is such that the plastic cladding strain is less than 1% at 55,000 MWd/mtU. The following cladding strain conservatisms are applicable with respect to the Oconee 3 fuel:

1. The maximum Specification value for the fuel pellet diameter was used.
2. The maximum Specification value for the fuel pellet density was used.
3. The cladding ID used was the lowest permitted Specification tolerance.
4. The maximum expected three-cycle local pellet burnup is less than 55,000 MWd/mtU.

4.3. Thermal Design

All fuel in the cycle 6 core is 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. LHR capabilities are based on centerline fuel melt and were established using the TAFY-3[4] code with consideration for fuel densification. The maximum fuel rod burnup at EOC 6 is predicted to be 37,139 MWd/mtU. Fuel rod internal pressure has been evaluated with TAFY-3 for the rod of highest burnup and is predicted to be less than the nominal RC system pressure of 2200 psia.

4.4. Material Design

The batch 8 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 8 fuel assemblies is identical to those of the present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15 × 15 fuel assembly has verified the adequacy of its design. As of April 30, 1980, the following experience has been accumulated for the eight operating B&W 177-fuel assembly plants using the Mark B fuel assembly:

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU Incore	Discharged	Cumulative net electrical output, MWh
Oconee 1	6	19,600	40,000	29,857,021
Oconee 2	5	23,400	33,700	26,232,944
Oconee 3	5	26,300	29,400	25,980,508
TMI-1	4	32,400	32,200	28,840,053
ANO-1	4	25,100	33,222	23,478,392
Rancho Seco	3	37,729	29,378	20,317,332
Crystal River 3	2	23,194	23,194	11,400,975
Davis Besse 1	1	14,600	--	7,560,018

Table 4-1. Fuel Design Parameters and Dimensions

	Batch No.			
	5B	6	7	8
FA type	Mark B4	Mark B4	Mark B4	Mark B4
No. of FAs	17	36	56	68
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4
Undensif active fuel length, in.	142.23	142.25	142.23	141.8
Fuel pellet OD (mean spec), in.	0.3695	0.3695	0.3695	0.3686
Fuel pellet initial density (mean spec), %TD	94.0	94.0	94.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.02	2.97	2.80	3.07
Est residence time, EFPH	26,338	22,522	26,304	29,232
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>30,000

Table 4-2. Fuel Thermal Analysis Parameters

	Batch No.			
	5B	6	7	8
No. of assemblies	17	36	56	68
Initial density, % TD	94.0	94.0	94.0	95.0
Pellet diameter, in.	0.3695	0.3695	0.3695	0.3686
Stack height, in.	142.2	142.2	142.2	141.8
Densified Fuel Parameters[a]				
Pellet diameter, in.	0.3646	0.3646	0.3646	0.3649
Fuel stack height, in.	140.5	140.5	140.5	140.74
Nominal linear heat rate @ 2568 MWt, kW/ft	5.80	5.80	5.80	5.79
Average fuel temp @ nominal LHR, F	1320	1320	1320	1310
Linear heat rate capability (centerline fuel melt), kW/ft	20.15	20.15	20.15	20.15

Core avg linear heat rate = 5.80 kW/ft.

[a] 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.[5-7] Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The longer cycle 6 will produce a higher cycle burnup than that for the design cycle 5. Figure 5-1 illustrates a representative relative power distribution for the beginning of the sixth cycle at full power with equilibrium xenon and normal rod positions.

The initial BPRA loading, longer design life, and different shuffle pattern for cycle 6 make it difficult to compare the physics parameters with those of cycle 5. The critical boron concentrations for cycle 6 are higher because the additional reactivity necessary for the longer cycle is not completely offset by burnable poison. The control rod worths differ between cycles due to changes in radial flux and burnup distributions, which also accounts for the smaller BOC stuck and ejected rod worths in cycle 6 compared to cycle 5 values. 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 rod 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. 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 reference fuel cycle shutdown margin is presented in the Oconee 3, cycle 5 reload report.[5]

The cycle 6 power deficits, differential boron worths, and effective delayed neutron fractions differ from those of cycle 5 because of the presence of burnable poison and the longer cycle length.

5.2. Analytical Input

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

5.3. Changes in Nuclear Design

There is only one significant core design change between the reference and reload cycles. This change is the increase in cycle lifetime to 376 EFPD and the subsequent incorporation of BPRAs to aid in reactivity control. The calculational methods and design information used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle.

Table 5-1. Oconee 3 Physics Parameters[a]

	Cycle 6[b]	Cycle 5[c]
Cycle length, EFPD	376	292
Cycle burnup, MWd/mtU	11,766	9137
Average core burnup, EOC, MWd/mtU	20,231	18,711
Initial core loading, mtU	82.1	82.1
Critical boron — BOC (no xenon), ppm		
HZP, group 8 inserted	1471	1351
HFP, group 8 inserted	1282	1161
Critical boron — EOC (equil xenon), ppm		
HZP, group 8 inserted	385	339
HFP, group 8 inserted	78	61
Control rod worths — HFP, BOC, % Δk/k		
Group 6	0.98	1.00
Group 7	1.36	1.70
Group 8	0.50	0.49
Control rod worths — HFP, EOC[d], % Δk/k		
Group 7	1.48	1.64
Group 8	0.54	0.51
Max ejected rod worth — HZP, % Δk/k		
BOC, (N12) groups 5-8 inserted	0.38	0.46
EOC, (N12) groups 5-8 inserted	0.51	0.50
Max stuck rod worth — HZP, % Δk/k		
BOC (M13)	1.39	1.81
EOC (M13)	1.52	1.75
Power deficit, HZP to HFP, % Δk/k		
BOC	1.39	1.34
EOC	2.22	2.11
Doppler coeff — BOC, 10^{-5}(Δk/k-°F)		
100% power (no xenon)	-1.49	-1.51
Doppler coeff — EOC, 10^{-5}(Δk/k-°F)		
100% power (equil xenon)	-1.62	-1.57

Table 5-1. (Cont'd)

	Cycle 6[b]	Cycle 5[c]
Moderator coeff — HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (0 xenon, 1282 ppm, group 8 ins.)	-0.65	-0.66
EOC (equil xenon, 17 ppm, group 8 ins.)	-2.82	-2.69
Boron worth — HFP, ppm/% $\Delta k/k$		
BOC (1300 ppm)	116	109
EOC (17 ppm)	102	95
Xenon worth — HFP, % $\Delta k/k$		
BOC (4 days)	2.61	2.65
EOC (equilibrium)	2.74	2.75
Eff delayed neutron fraction — HFP		
BOC	0.00628	0.00585
EOC	0.00526	0.00519

[a] Cycle 6 data are for the conditions stated in this report. The cycle 5 core conditions are identified in reference 5.

[b] Based on a 299-EFPD cycle 5.

[c] Based on 270-EFPD cycle 4.

[d] 292 EFPD in cycle 5, 376 EFPD in cycle 6.

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 6

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.49	8.92
Worth red'n due to poison burnup	-0.29	-0.30
Maximum stuck rod, HZP	-1.39	-1.52
Net worth	6.81	7.10
Less 10% uncertainty	-0.68	-0.71
Total available worth	6.13	6.39
Required Rod Worth		
Power deficit, HFP to HZP	1.39	2.22
Max inserted rod worth	0.42	0.54
Flux redistribution	0.57	1.18
Total required worth	2.38	3.94
Shutdown Margin		
Total avail worth - total req'd worth	3.75	2.45

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

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

	8	9	10	11	12	13	14	15
H	1.084	1.258	0.976	1.266	1.122	1.232	1.103	0.650
K		1.196	0.991	1.079	1.234	1.032	1.123	0.624
L			1.040	1.208	8 1.084	1.242	0.966	0.486
M				1.105	1.185	1.032	0.921	
N					0.970	0.991	0.537	
O						0.583		
P								
R								

0	Inserted Rod Group No.
0.00	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 employed the methods and models described in references 1, 5, and 9.

The maximum core bypass flow for cycle 5 was 10.4% of the total system flow. For cycle 6 operation, 60 BPRAs will be inserted. Retainers will be placed on these assemblies as described in reference 3. Two assemblies contain regenerative neutron sources and retainers. The number of open assemblies is 46, and the maximum core bypass flow is reduced to 8.1%. The cycle 5 and 6 maximum design conditions are summarized in Table 6-1.

A rod bow DNBR penalty has been calculated for cycle 6 operation according to procedures approved by reference 10. The burnup used to calculate the penalty is the highest batch 7 burnup, 23,411 MWd/mtU. The net rod bow penalty is 1.1% after taking credit for the flow area reduction hot channel factor used in all DNBR calculations. However, all plant operating limits based on DNBR criteria include a minimum of 10% DNBR margin from the B&W-2 correlation design limit of 1.30.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 5	Cycle 6
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Core bypass flow, % total flow	10.4	8.1
Vessel inlet/outlet coolant temp at 100% power, F	555.6/602.4	555.6/602.4
Ref design radial-local power peaking factor	1.71	1.71
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.	(a)	(a)
Avg heat flux at 100% power, 10^3 Btu/h-ft^2 [a]	176 [b]	176 [b]
CHF correlation	BAW-2	BAW-2
Min DNBR with densification penalty	1.98	2.05

[a] See Table 4-2.

[b] Heat flux based on densified length of 140.3 in., which is a conservative minimum value.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 5 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 9. Since batch 8 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 9, the conclusions in that reference are still valid.

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. Accident Evaluation

The key parameters that have the greatest effect on determining 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 given in Table 4-2. Table 6-1 compares the cycle 5 and 6 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and cycle 6. Generic LOCA analyses have been performed for the B&W 177-FA lowered-loop NSS using the Final Acceptance Criteria ECCS evaluation model reported in reference 11. These analyses are generic in nature since the limiting values of the key parameters for all plants in this category were used. Furthermore, the combination of the average fuel temperature as a function of linear heat rate and the lifetime pin pressure data used in the LOCA limits analyses[11,12] are conservative compared to those calculated for this reload. Thus, the analyses and the LOCA limits reported in references 11 and 12 provide conservative results for the operation of Oconee 3, cycle 6 fuel. A tabulation showing the bounding values for allowable LOCA peak LHRs for Oconee 3, cycle 6 fuel is provided in Table 7-2.

From the examination 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 safe operation of the Oconee 3 plant 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 and/or the fuel densification report.[9]

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

Parameter	FSAR[1] value	Predicted[c] cycle 6 value
BOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.17	-1.49
EOC Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.33[a]	-1.62
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[b]	-0.65
EOC moderator coeff, 10^{-4} $\Delta k/k/°F$	-3.0	-2.82
All rod bank worth, HZP, % $\Delta k/k$	10.0	8.49
Boron reactivity worth, 70°F, ppm/1% $\Delta k/k$	75	82
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.31
Dropped rod worth, HFP, % $\Delta k/k$	0.46	0.20
Initial boron conc, HFP, ppm	1400	1282

[a] -1.2×10^{-5} $\Delta k/k/F$ was used for steam-line failure analysis.

-1.3×10^{-5} $\Delta k/k/F$ was used for cold water accident (pump start-up).

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

[c] Using reference 6.

Table 7-2. Bounding Values for Allowable LOCA Peak Linear Heat Rates

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

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 6 operation in accordance with the methods of references 13-15 to account for minor changes in power peaking and control rod worths inherent with a transition to 18-month, lumped burnable poison cycles.

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 8-9 are revisions to previous Technical Specification limits.

Figure 8-1. Oconee Unit 3 Core Protection Safety Limits

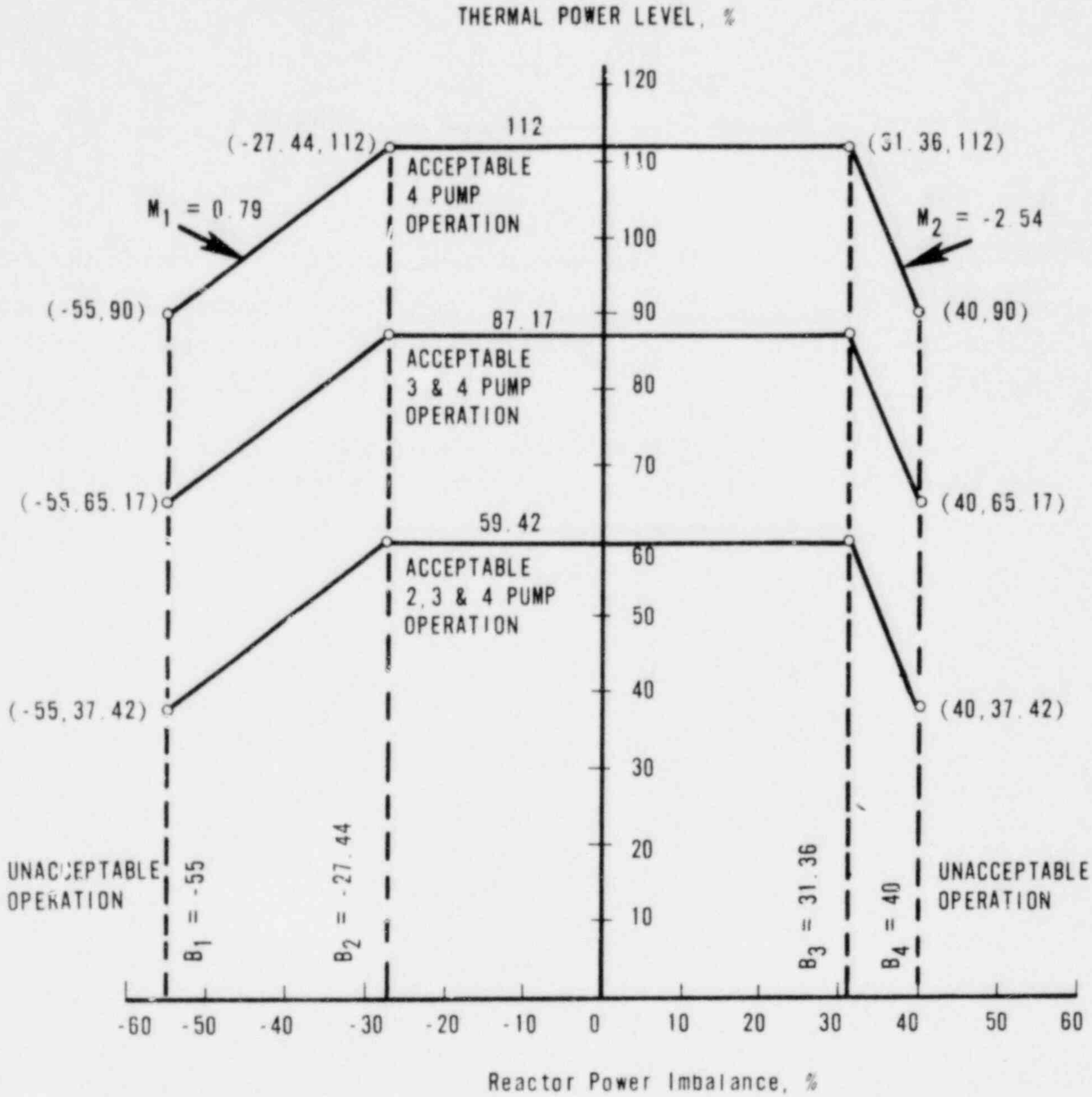


Figure 8-2. Oconee Unit 3 Protective System Maximum Allowable Setpoints

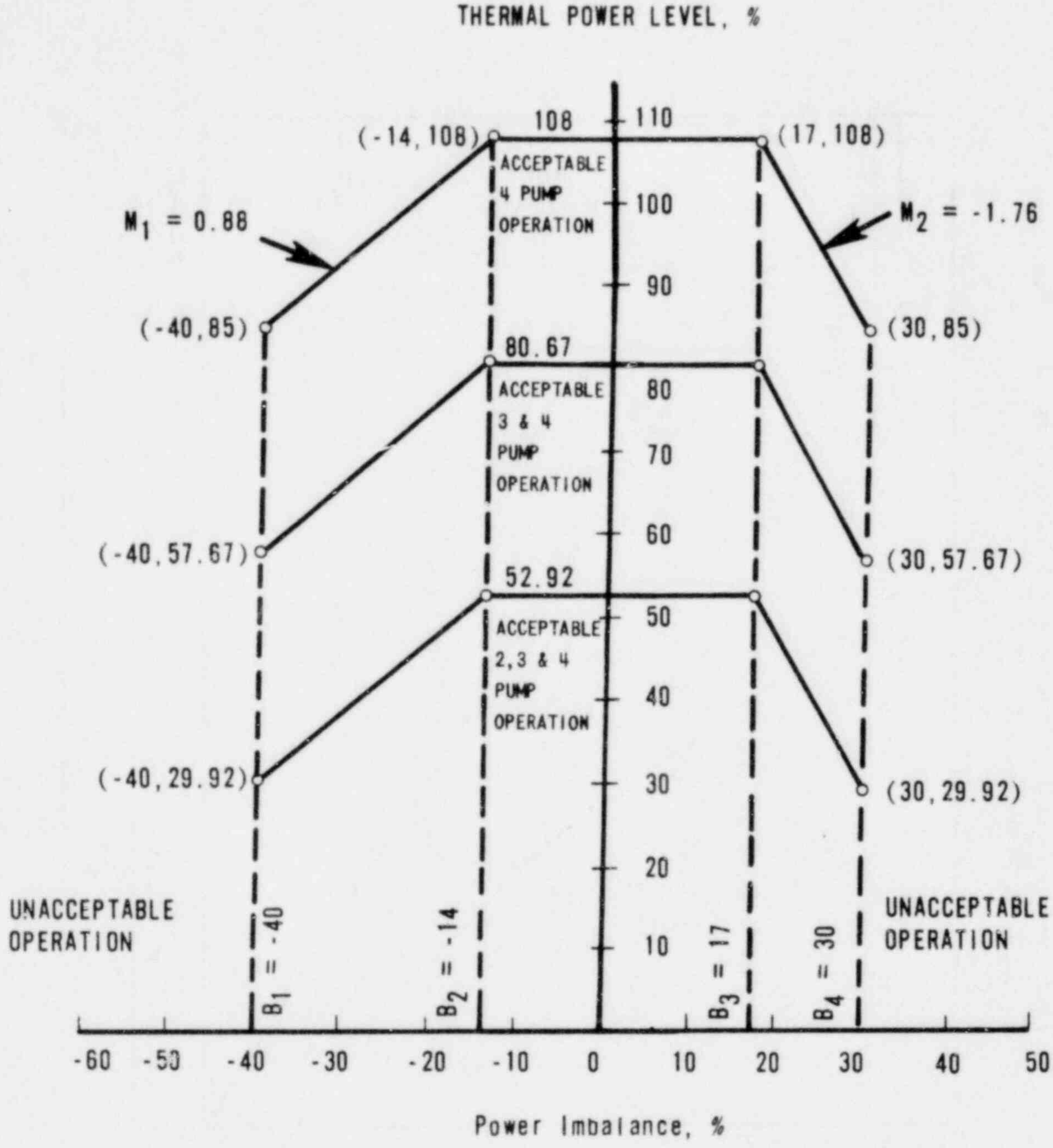


Figure 8-3. Oconee 3 Cycle 6 Rod Position Limits — Four-Pump Operation, 0-200 ± 10 EFPD

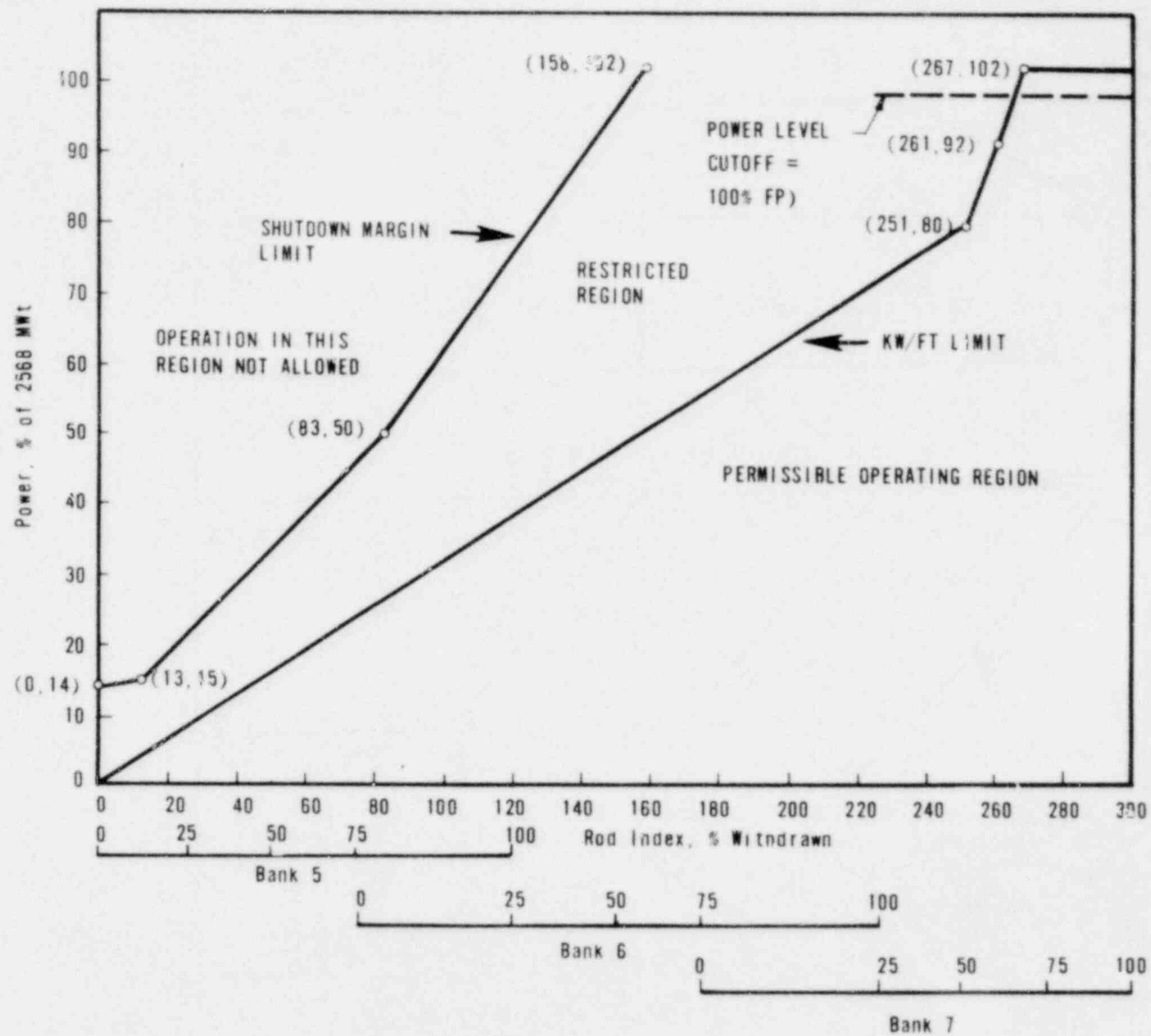
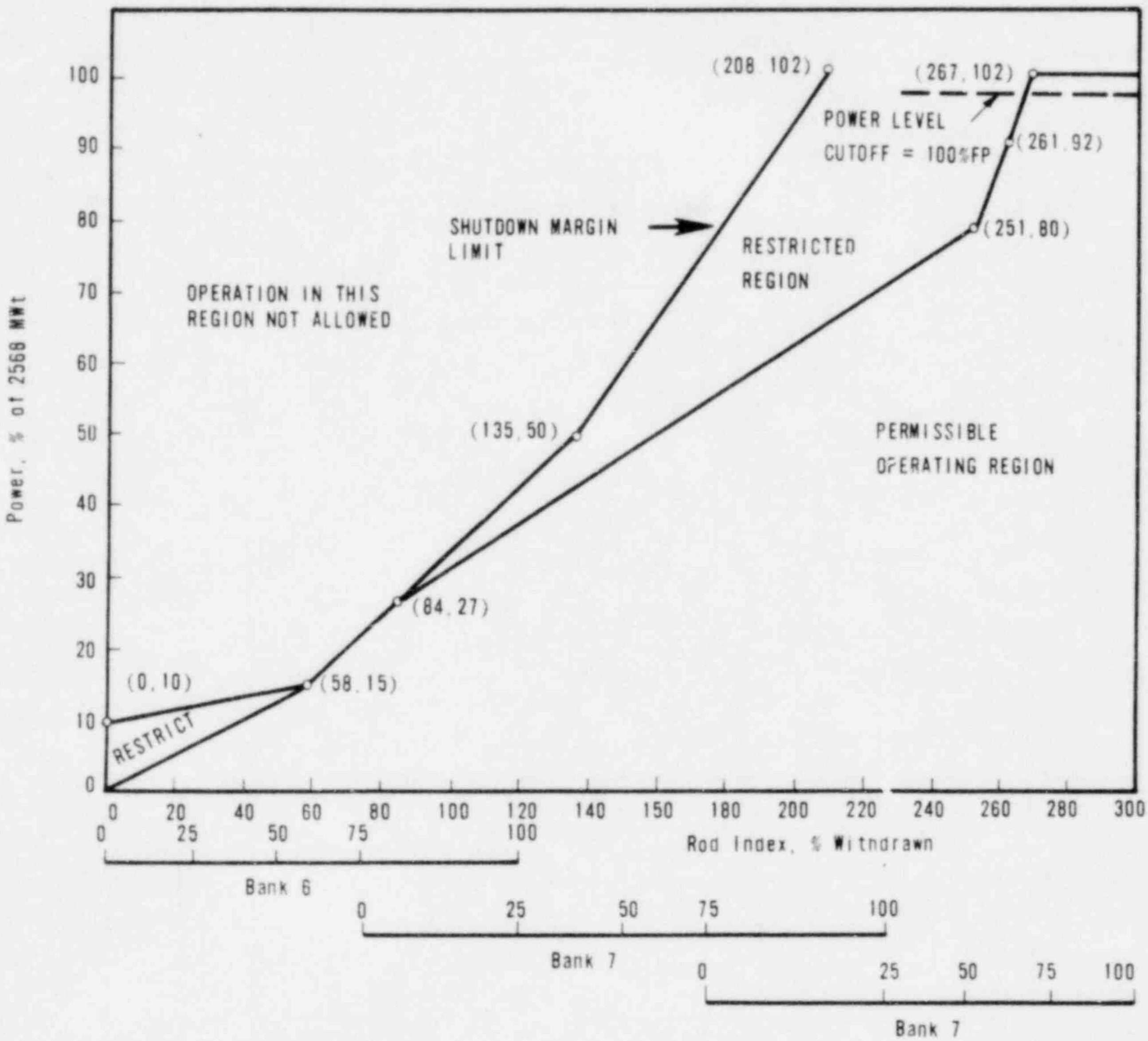


Figure 8-4. Oconee 3 Cycle 6 Rod Position Limits — Four-Pump Operation After 200 ± 10 EFPD



Babcock & Wilcox

Figure 8-5. Oconee 3 Cycle 6 Rod Position Limits — Two- and Three-Pump Operation, 0-200 ± 10 EFPD

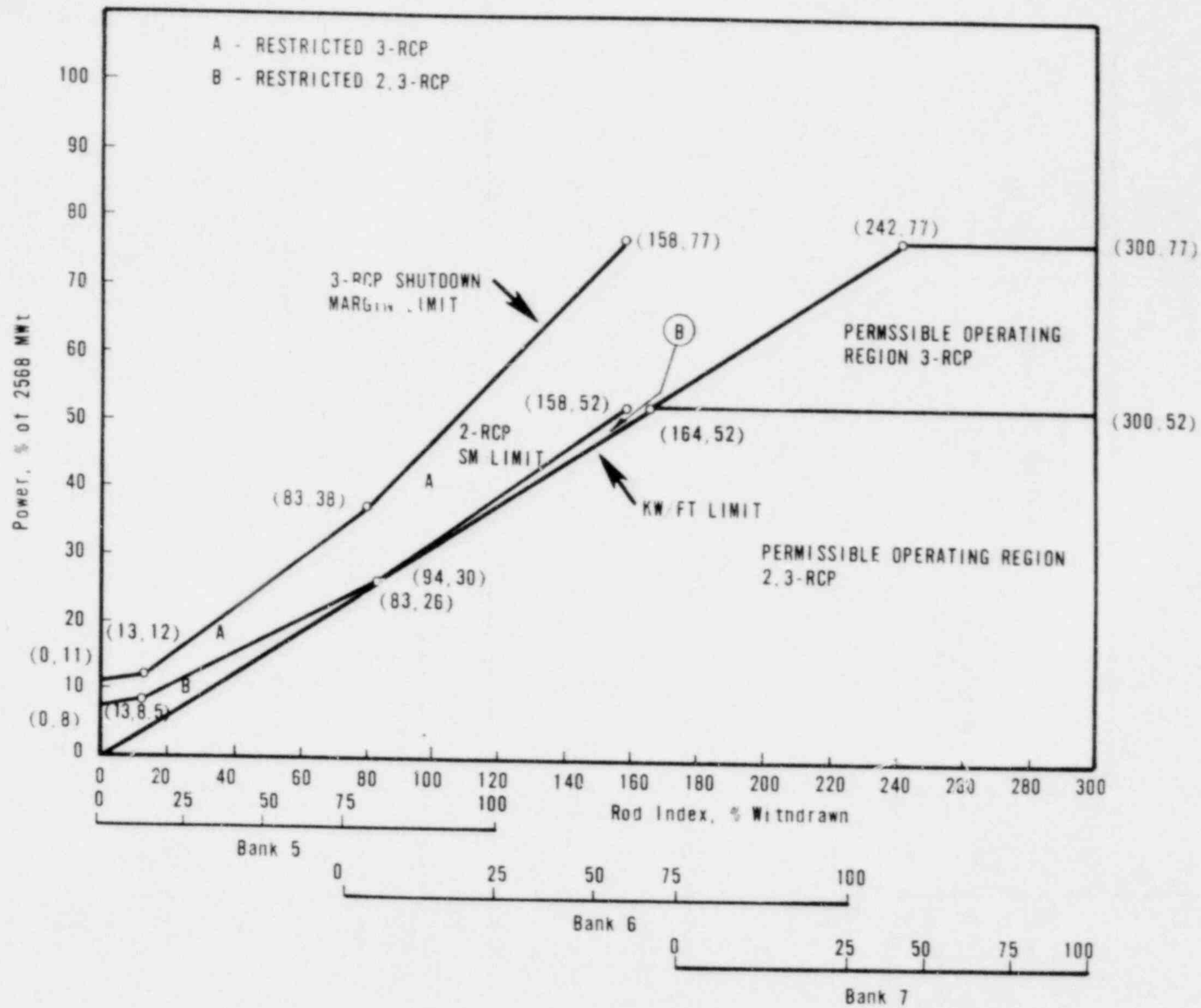


Figure 8-6. Oconee 3 Cycle 6 Rod Position Limits — Two- and Three-Pump Operation After 200 ± 10 EFPD

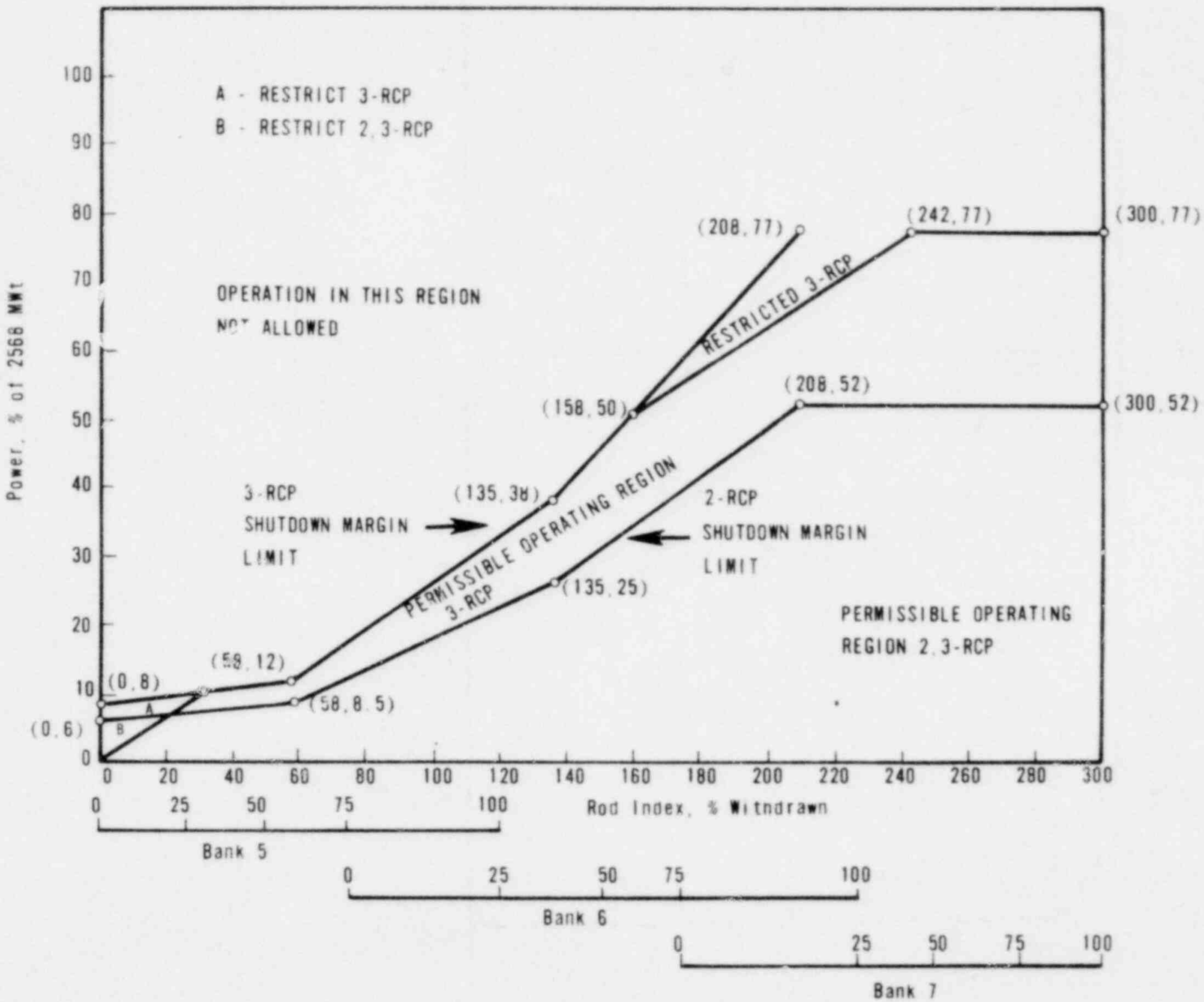


Figure 8-7. Oconee 3 Cycle 6 Operational Power Imbalance Limits

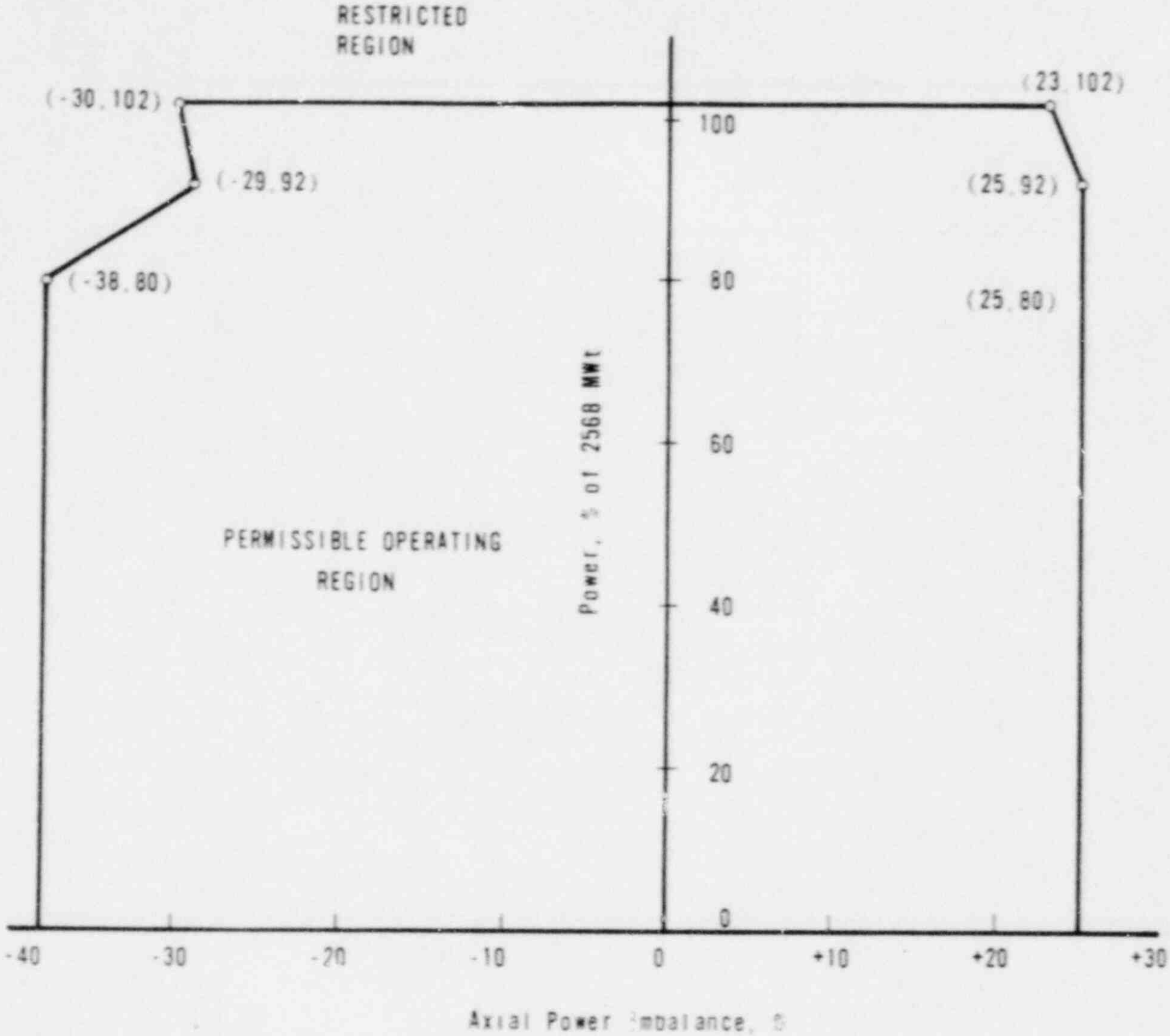


Figure 8-8. Oconee 3 Cycle 6 APSR Position Limits, 0-200 ± 10 EFPD

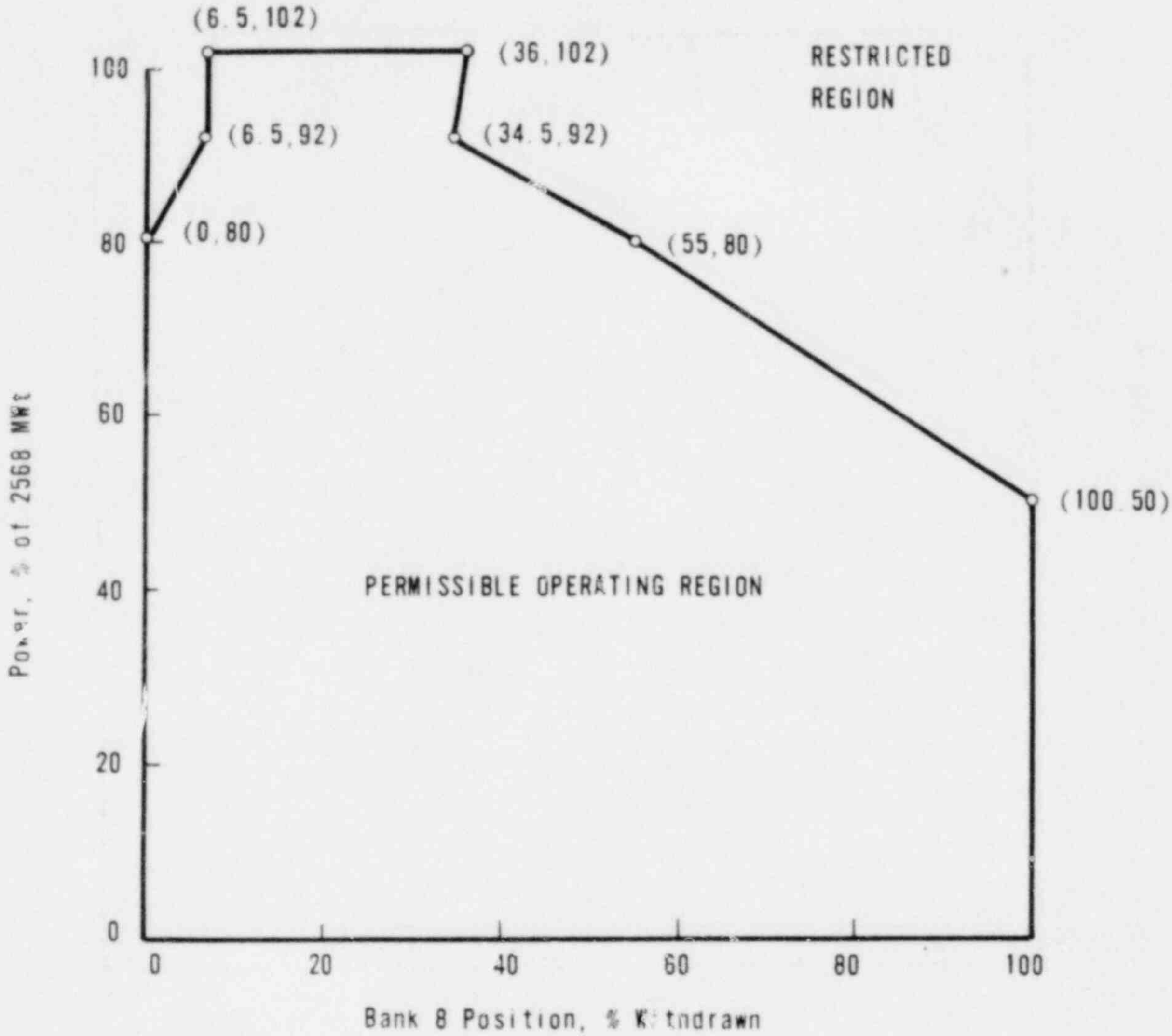
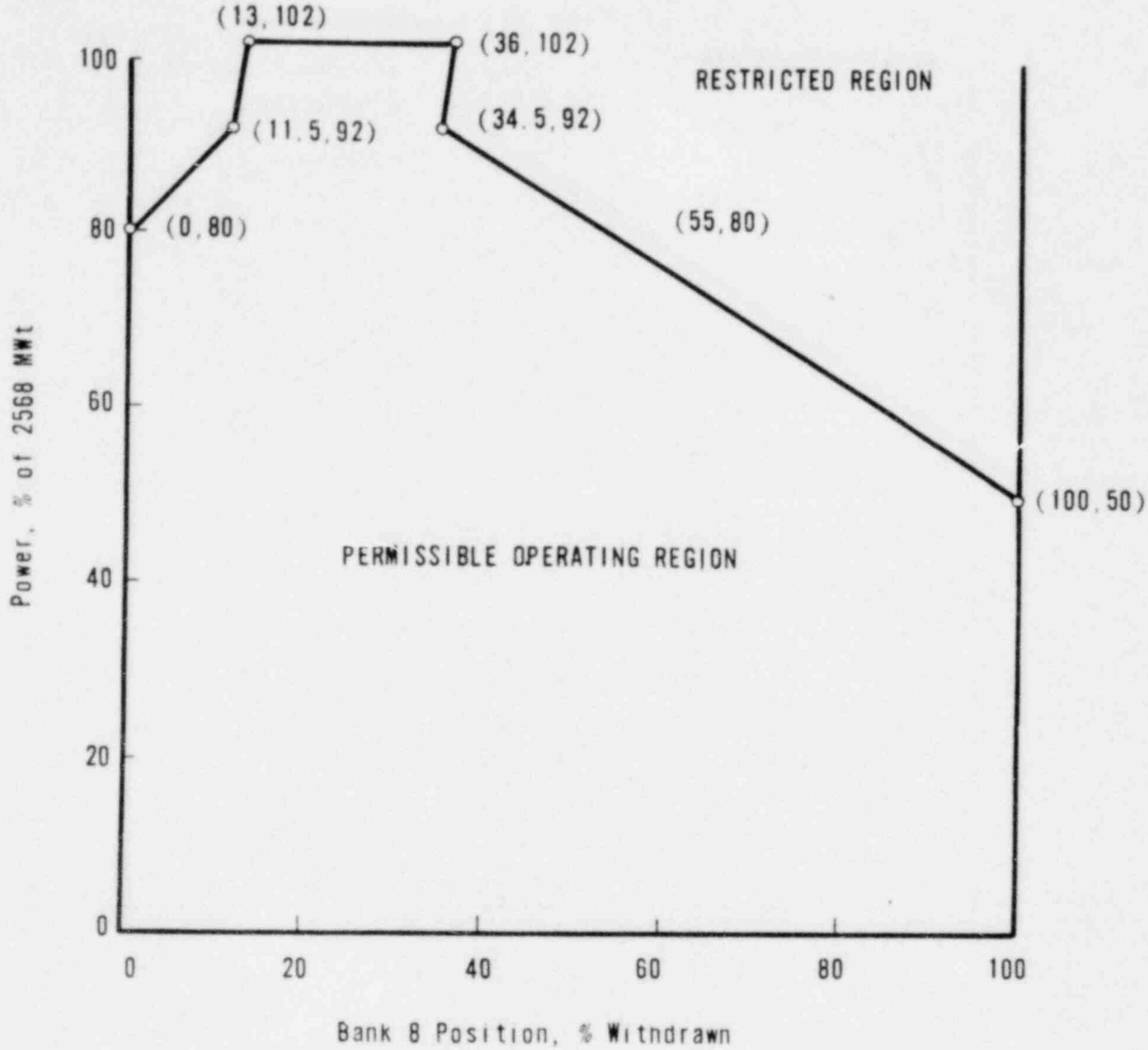


Figure 8-9. Oconee 3 Cycle 6 APSR Position Limits
After 200 ± 20 EFPD



REFERENCES

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

2 A. F. J. Eckert, H. W. Wilson, and K. E. Yoon, Program to Determine In-reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084A, Rev 2, Babcock & Wilcox, January 1979.

3 BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, May 1978.

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

5 Oconee Unit 3, Cycle 5 — Reload Report, BAW-1522, Babcock & Wilcox, March 1979.

6 B&W Version of PDQ07 Code, BAW-10117A, Babcock & Wilcox, January 1977.

7 Core Calculational Techniques and Procedures, BAW-10118, Babcock & Wilcox, October 1977.

8 Assembly Calculations and Fitted Nuclear Data, BAW-10116A, Babcock & Wilcox, May 1977.

9 Oconee 3 Fuel Densification Report, BAW-1399, Babcock & Wilcox, November 1973.

10 L. S. Rubenstein (NRC) to J. H. Taylor (B&W) Letter, "Evaluation of Interim Procedure for Calculating DNBR Reductions Due to Rod Bow," October 18, 1979.

11 ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 2, Babcock & Wilcox, September 1975.

12 J. H. Taylor (B&W) to S. A. Varga (NRC), Letter, July 18, 1978.

13 Power Peaking Nuclear Reliability Factors, BAW-10119, Babcock & Wilcox, January 1977.

[14] Normal Operating Controls, BAW-10122, Babcock & Wilcox, August 1978.

[15] Verification of the Three-Dimensional FLAME Code, BAW-10125A, Babcock & Wilcox, August 1976.