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OCONEE UNIT 1, CYCLE 6

— Reload Report —

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth cycle of Oconee Nuclear Station, Unit 1, 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 1, 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[2] 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.

Four Mark-B2 demonstration fuel assemblies containing Zircaloy intermediate grids will be inserted in the core as part of the fresh batch 8; a description of these fuel assemblies is contained in reference 1. These assemblies will not adversely affect cycle 6 operation.

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 1 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 of Oconee 1, cycle 6 is the currently operating cycle 5. The cycle 6 design length of 372 EFPD is based on a planned cycle 5 length of 320 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in cycle 6.

3. GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design basis are described in detail in section 3 of the Final Safety Analysis Report[2] for Oconee Nuclear Station, Unit 1. The cycle 6 core contains 177 fuel assemblies, each of which is a 15 by 15 array of 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 of 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 1, cycle 6. Three of the batch 6 assemblies will be discharged at the end of cycle 5 along with batches 4D and 5. The remaining 53 batch 6 assemblies, designated 6B, and the fresh batches 8A and 8B — with initial enrichments of 2.79, 2.97, and 3.07 wt % ^{235}U, respectively — will be loaded into the central portion of the core. The batch 7 fuel, with an initial enrichment of 3.02 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.

Reactivity is controlled by 61 full-length Ag-In-Cd control rods, 60 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the 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 and group designations are identical to those of the reference cycle. The cycle 6 locations and enrichments of the BPRAs are shown in Figure 3-4.

The nominal system pressure is 2200 psia and the core average densified nominal heat rate is 5.80 kW/ft at the rated core power of 2568 MWt.

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

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						7 R10	7 R9	7 R8	7 R7	7 R6					
B				7 P12	8B	7 P11	8B	7 N8	8B	7 P5	8B	7 P4			
C			7 O13	8B	6B D5	8B	6B N3	8B	6B N13	8B	6B D11	8B	7 O3		
D		7 N14	8B	6B G3	8B	7 F14	8A	6B B8	8A	7 F2	8B	6B C9	8B	7 N2	
E		8B	6B E4	8B	6B D4	8A	6B K5	8A	6B K11	8A	6B D12	8B	6B E12	8B	
F	7 L15	7 M14	8B	7 P6	8A	6B E6	6B L4	6B E10	6B L12	6B F11	8A	7 P10	8B	7 M2	7 L1
G	7 K15	8B	6B C12	8A	6B E9	6B D10	7 K9	8B *	7 K7	6B D6	6B E7	8A	6B C4	8B	7 K1
H	7 H15	7 H12	8B	6B H2	8A	6B F5	8B *	6B O9	8B *	6B L11	8A	6B H14	8B	7 H4	7 H1
K	7 G15	8B	6B O12	8A	6B M9	6B N10	7 G9	8B *	7 G7	6B N6	6B M7	8A	6B O4	8B	7 G1
L	7 F15	7 E14	8B	7 B6	8A	6B L5	6B F4	6B M6	6B F12	6B M10	8A	7 B10	8B	7 E2	7 F1
M		8B	6B M4	8B	6B N4	8A	6B G5	8A	6B G11	8A	6B N12	8B	6B M12	8B	
N		7 D14	8B	6B O7	8B	7 L14	8A	6B P8	8A	7 L2	8B	6B K13	8B	7 D2	
O			7 C13	8B	6B N5	8B	6B D3	8B	6B D13	8B	6B N11	8B	7 C3		
P				7 B12	8B	7 B11	8B	7 D8	8B	7 B5	8B	7 B4			
R						7 A10	7 A9	7 A8	7 A7	7 A6					

*Mark-BZ demonstration assemblies.

X	Batch
YY	Cycle 5 Location

Figure 3-2. Enrichment and Burnup Distribution for Oconee 1, Cycle 6

	8	9	10	11	12	13	14	15
H	2.79 19,847	3.07 Mark-BZ 0	2.79 18,000	2.97 0	2.79 17,231	3.07 0	3.02 13,294	3.02 9,195
K		3.02 13,099	2.79 18,706	2.79 17,059	2.97 0	2.79 16,964	3.07 0	3.02 8,819
L			2.79 17,996	2.97 0	3.02 11,705	3.07 0	3.02 9,545	3.02 7,247
M				2.79 17,878	3.07 0	2.79 17,106	3.07 0	
N					2.79 19,847	3.07 0	3.02 6,675	
O						3.02 7,485		
P								
R								

x.xx	Initial Enrichment
xx,xxx	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Oconee 1, Cycle 6

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						3		5		3					
C					1		7		7		1				
D				6		8		4		8		6			
E			1		5		2		2		5		1		
F		3		8		7		6		7		8		3	
G			7		2		4		4		2		7		
H		5		4		6		3		6		4		5	
K			7		2		4		4		2		7		
L		3		8		7		6		7		8		3	
M			1		5		2		2		5		1		
N				6		8		4		8		6			
O					1		7		7		1				
P						3		5		3					
R															

W — Y; X — Z

x Group Number

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 1, Cycle 6

	8	9	10	11	12	13	14	15
H		1.2 Mark-BZ		0.8		1.2		
K					1.2		0.5	
L				0.8		0.8		
M					0.8			
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 1, cycle 6 are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable.

Four Mark-BZ demonstration fuel assemblies are included in batch 8B. The Mark-BZ is a 15 by 15 fuel assembly similar to the Mark-B assembly described in reference 2, except that the six intermediate spacer grids will be of Zircaloy material and a slightly redesigned holddown spring is incorporated. The Mark-BZ assembly is described in reference 1, which also states that reactor safety and performance are not adversely affected by the presence of the four demonstration assemblies.

Retainer assemblies will be used on the two fresh batch 8 fuel assemblies that contain regenerative neutron source (RNS) assemblies and on the 60 assemblies that contain BPRAs. The justification for the design and use of the retainers is described in reference 3.

4.2. Fuel Rod Design

The fuel pellet end configuration has changed from a spherical dish for batches 1 through 8A to a truncated cone dish for batch 8B. The new design reduces pellet end laminations during manufacturing. Fuel performance will not be adversely affected by this change. The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

The fuel of batch 6 is more limiting than other batches due to its longer previous incore exposure time. The batch 6 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 reference 4. The

collapse time for the most limiting assembly was conservatively determined to be more than 35,000 effective full-power hours (EFPH), which is greater than the maximum projected residence time of cycle 6 fuel (Table 4-1).

4.2.2. Cladding Stress

The Oconee 1 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 (primary and secondary) must be less than the minimum specified unirradiated yield strength. In all cases, the margin is in excess of 30%. With respect to Oconee 1 fuel, the following conservatisms were used in the analysis:

1. A lower post-densification 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 established for plastic cladding strain of less than 1% at maximum design local pellet burnup (55,000 MWd/mtU) and heat generation rate (20.15 kW/ft) values that are higher than the values the Oconee 1 fuel is expected to see. 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 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) capacity and the average fuel temperature for each batch in cycle 6 are shown in Table 4-2. Batch 8 contains 48 fuel assemblies with a 95% nominal density fuel pellet design and 20 fuel assemblies with 94% nominal density fuel pellets. LHR capabilities are based on centerline fuel melt and were established using the TAFY-3 code[5] with consideration for fuel densification.

4.4. Material Design

The batch 8 fuel assemblies are not new in concept, nor do they utilize different component materials, except for the Zircaloy grids of the four Mark-BZ assemblies described in section 4.1. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 8 fuel assemblies are identical to those of the present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark-B, 15 by 15 fuel assembly has verified the adequacy of its design. As of March 31, 1979, the following experience has been accumulated for the nine operating B&W 177-fuel assembly plants using the Mark-B fuel assembly:

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU[a] Incore	Maximum assembly burnup, MWd/mtU[a] Discharged	Cumulative net electrical output, mWh[b]
Oconee 1	5	35,300	31,100	25,423,997
Oconee 2	3	22,400	33,700	20,311,630
Oconee 3	4	24,100	29,400	22,410,960
TMI-1	4	32,400	32,200	23,880,710
TMI-2	1	4,300	--	786,294
ANO-1	3	33,240	28,300	19,739,776
Rancho Seco	3	31,800	29,378	15,514,225
Crystal River 3	1	16,300	--	7,643,351
Davis Besse 1	1	8,900	--	3,750,428

[a] As of March 31, 1979.
[b] As of February 28, 1979.

Table 4-1. Fuel Design Parameters and Dimensions

	Twice-burned FAs, batch 6B	Once-burned FAs, batch 7	Fresh FAs, batch 8A	Fresh FAs, batch 8B
FA type	Mark-B4	Mark-B4	Mark-B4	Mark-B4/Mark-BZ
No. of FAs	53	56	20	44/4
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
Undensified active fuel length (nom), % TD	142.25	142.25	142.25	141.38
Fuel pellet initial density (nom), % TD	94.0	94.0	94.0	95.0
Fuel pellet OD (mean specification), in.	0.3695	0.3695	0.3695	0.3686
Initial fuel enrichment, wt % ^{235}U	2.79	3.02	2.97	3.07
BOC burnup (avg), MWd/mtU	17,781	9,361	0	0
Cladding collapse time, EFPH	>35,000	>35,000	>35,000	>35,000
Estimated residence time (max), EFPH	22,990	27,432	29,856	29,856

Table 4-2. Fuel Thermal Analysis Parameters — Oconee 1, Cycle 6

	Batch 6	Batch 7	Batch 8A	Batch 8B[a]
No. of assemblies	53	56	20	48
Nominal pellet density, % TD	94.0	94.0	94.0	95.0
Pellet diameter, in.	0.3695	0.3695	0.3695	0.3686
Stack height, in.	142.25	142.25	142.25	141.38
Densified Fuel Parameters				
Pellet diameter, in.	0.3646	0.3646	0.3646	0.3649
Fuel stack height, in.	140.47	140.47	140.47	140.32
Nominal LHR at 2568 MWt, kW/ft	5.80	5.80	5.80	5.80
Avg fuel temp at nominal LHR, F	1320	1320	1320	1320
LHR to $\mathbb{C}_L$ fuel melt, kW/ft	20.15	20.15	20.05	20.15

[a] Includes four Mark-BZ demonstration assemblies.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of design cycle 6 with those of the reference cycle 5. The values for both cycles were generated using PDQ07.[6-8] 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 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 the burnable poison. The control rod worths differ between cycles due to changes in radial flux and burnup distributions. This also accounts for the smaller ejected and stuck 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 1, cycle 5 reload report.[9]

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

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 is only one significant core design change between the reference and reload cycles. This change is the increase in cycle lifetime to 372 EFPD and the 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 1 Physics Parameters[a]

	Cycle 5[b]	Cycle 6[c]
Cycle length, EFPD	320	372
Cycle burnup, MWd/mtU	10,014	11,641
Average core burnup, EOC, MWd/mtU	19,055	19,927
Initial core loading, mtU	82.1	82.1
Critical boron, BOC (no Xe), ppm		
HZP, group 8 37.5% wd[d]	1426	1448
HFP, group 8 inserted	1242	1253
Critical broon, EOC (eq Xe), PPM		
HZP, group 8 37.5% wd	338	377
HFP, group 8 37.5% wd	43	83
Control rod worths, HFP, BOC, % Δk/k		
Group 6	1.19	1.04
Group 7	1.44	1.51
Group 8 37.5% wd	0.42	0.50
Control rod worths, HFP, EOC, % Δk/k		
Group 7	1.52	1.60
Group 8 37.5% wd	0.48	0.54
Max ejected rod worth, HZP, % Δk/k[e]		
BOC (N-12)	0.57	0.42
EOC (N-12)	0.70	0.44
Max stuck rod worth, HZP, % Δk/k		
BOC (N-12)	2.17	1.24
EOC (N-12)	2.01	1.44
Power deficit, HZP to HFP, % Δk/k		
BOC	1.31	1.56
EOC	2.11	2.48
Doppler coeff, 10^{-5} (Δk/k-°F)		
BOC, 100% power, no Xe	-1.45	-1.48
EOC, 100% power, eq Xe	-1.61	-1.61
Moderator coeff, HFP, 10^{-4} (Δk/k-°F)		
BOC (0 Xe, crit ppm, gp 8 ins)	-0.48	-0.72
EOC (eq Xe, 17 ppm, gp 8 ins)	-2.63	-2.85
Boron worth, HFP, ppm/% Δk/k		
BOC (1150 ppm)	108	116
EOC (17 ppm)	97	102
Xenon worth, HFP, % Δk/k		
BOC (4 EFPD)	2.62	2.60
EOC (equilibrium)	2.74	2.73
Eff delayed neutron fraction, HFP		
BOC	0.00595	0.00612
EOC	0.00520	0.00516

[a] Cycle 6 data are for the conditions stated in this report. The cycle 5 core conditions are identified in reference 9.

[b] Based on 250 EFPD at 2568 MWt, cycle 4.

[c] Cycle 6 data are based on a cycle 5 length of 320 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 for Oconee 1, Cycle 6

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.32	8.87
Worth reduction due to burnup of poison material	-0.42	-0.42
Maximum stuch rod, HZP	-1.24	-1.44
Net worth	6.66	7.01
Less 10% uncertainty	0.67	0.70
Total available worth	5.99	6.31
Required Rod Worth		
Power deficit, HFP to HZP	1.56	2.48
Max allowable inserted rod worth	0.65	0.70
Flux redistribution	0.40	0.84
Total required worth	2.61	4.02
Shutdown margin (total available worth minus total required worth)	3.38	2.29

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

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

	8	9	10	11	12	13	14	15
H	0.995	1.132 Mark-BZ	0.949	1.154	1.077	1.231	1.108	0.696
K		1.075	0.932	1.014	1.182	1.100	1.175	0.683
L			0.962	1.175	8 1.109	1.237	1.052	0.542
M				1.062	1.211	1.005	0.957	
N					0.973	1.023	0.580	
O						0.631		
P								
R								

x	Inserted Rod Group No.
x.xxx	Relative Power Density

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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 2, 9, and 10 except for the core bypass flow, the inclusion of retainers to provide positive holddown of BPRAs, and the insertion of four low absorption grid (LAG) demonstration assemblies which contain six Zircaloy intermediate spacer grids.

The maximum core bypass flow due to the removal of all ORAs in cycle 5 was 10.4%. For cycle 6 operation 60 BPRAs will be inserted, leaving 46 vacant fuel assemblies, resulting in a decrease in calculated maximum core bypass flow to 8.1%. The BPRA retainers introduce a small DNBR penalty as discussed in reference 11. Reactor core safety limits have been re-evaluated based on the insertion of these BPRAs with retainers and increased core flow. The cycle 5 and 6 maximum design conditions and significant parameters are shown in Table 6-1. The increase in core flow more than compensates for the decrease in DNBR due to the BPRA retainers. The four Mark-BZ assemblies will be limited to a design peak of 1.650 to ensure that they are never the limiting assemblies, while the 1.71 design radial-local peak remains valid for all other assemblies.

For cycle 6 operation a flux/flow trip setpoint of 1.08 is established. This setpoint and other plant operation limits based on minimum DNBR criteria contain a DNBR margin of 10.2% from the design minimum DNBR limit of 1.30.

In response to reference 12, B&W has committed to prepare a topical report addressing the potential for, and effects of, fuel rod bow. In addition, B&W has submitted an interim rod bow penalty evaluation procedure[13] for use until the topical report is completed and reviewed. As shown in reference 13, there is no DNBR penalty due to fuel rod bow for fuel burnup to approximately 21,300 MWd/mtU when this interim procedure is used. For Oconee 1, cycle 6 the limiting (highest power) fuel assembly is always in fuel burned less than 21,300 MWd/mtU and no DNBR rod bow penalty is required.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 5[9]	Cycle 6
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Vessel inlet coolant temp, 100% power, F	555.6	555.6
Vessel outlet coolant temp, 100% power, F	602.4	602.4
Ref design axial flux shape	1.5 cos	1.5 cos
Ref design radial-local power peaking factor	1.71	1.71
Active fuel length, in.	(a)	(a)
Average heat flux, 100% power, 10^3 Btu/h-ft^2	176[b]	176[b]
CHF correlation	BAW-2	BAW-2
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Minimum DNBR with densification penalty	1.98	2.05

[a] See Table 4-2.

[b] Based on densified length of 140.3 in.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[2] 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 10. Since batch 8 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference 10 report, the conclusions in that reference are still valid.

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.

Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Fuel thermal analysis values for each batch 6 are compared in Table 4-2. The cycle 6 thermal-hydraulic maximum design conditions are compared to the previous cycle 5 values[9] in Table 6-1. These parameters are common to all the accidents considered in this report. A comparison of the key kinetics parameters from the FSAR and cycle 6 is provided in Table 7-1.

A generic LOCA analysis for the B&W 177-FA, lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103, Rev 1.[14] 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 1, cycle 6 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 1, cycle 6 fuel.

It is concluded from the examination of cycle 6 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the Oconee 1 plant's ability to operate 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 for the transients in cycle 6 are bounded by the FSAR[2], the fuel densification report[10], and/or subsequent cycle analyses.

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

Parameter	FSAR and densification report value	Predicted cycle 6 value
Doppler coeff, Δk/k/°F		
BOC	-1.17×10^{-5}	-1.48×10^{-5}
EOC	-1.33×10^{-5}	-1.61×10^{-5}
Moderator coeff, Δk/k/°F		
BOC	$+0.5 \times 10^{-4}$	-0.72×10^{-4}
EOC	-3.0×10^{-4}	-2.85×10^{-4}
All-rod group worth, HZP % Δk/k	10	8.32
Initial boron conc'n, HFP, ppm	1400	1253
Boron reactivity worth at 70F, ppm/1% Δk/k	75	81
Max ejected rod worth, HFP, % Δk/k	0.65	0.27
Dropped rod worth (HFP), % Δk/k	0.46	0.20

Table 7-2. LOCA Limits, Oconee 1, Cycle 6

Elevation, ft	LHR limits, 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 15, 16, and 17 to account for changes in power peaking and control rod worths inherent with a transition to 18-month, lumped burnable poison cycles. In addition:

1. The behavior of power peaking with transient xenon has been reviewed for cycle 6. This review has confirmed that the effect of transient xenon on power peaking is conservatively accounted for by the xenon penalty factor of 5%.

2. A flux/flow trip setpoint of 1.08 is established for cycle 6 operation.

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 8-10 are revisions to previous Technical Specifications limits.

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

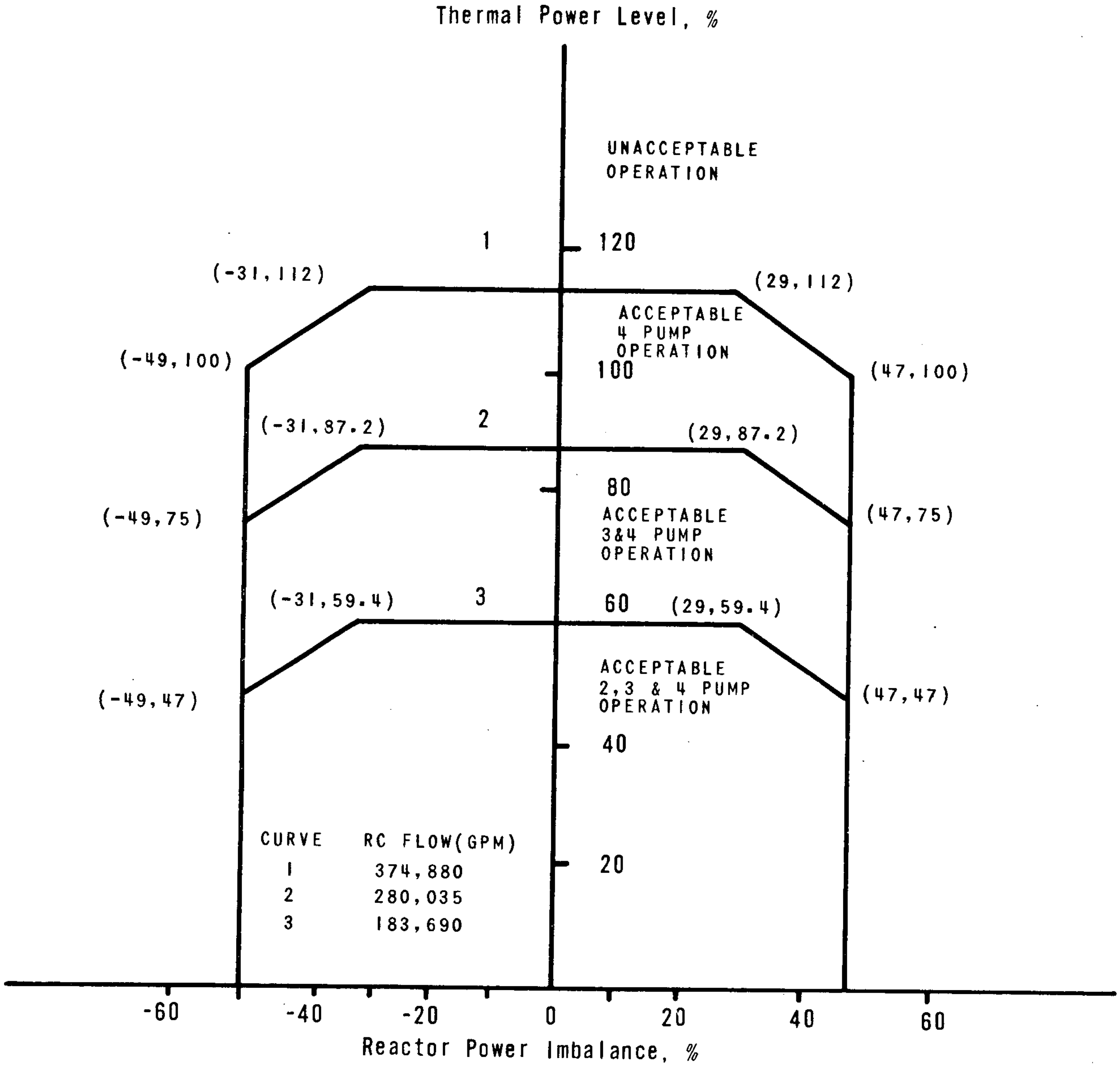


Figure 8-2. Protective System Maximum Allowable Setpoints, Oconee Unit 1

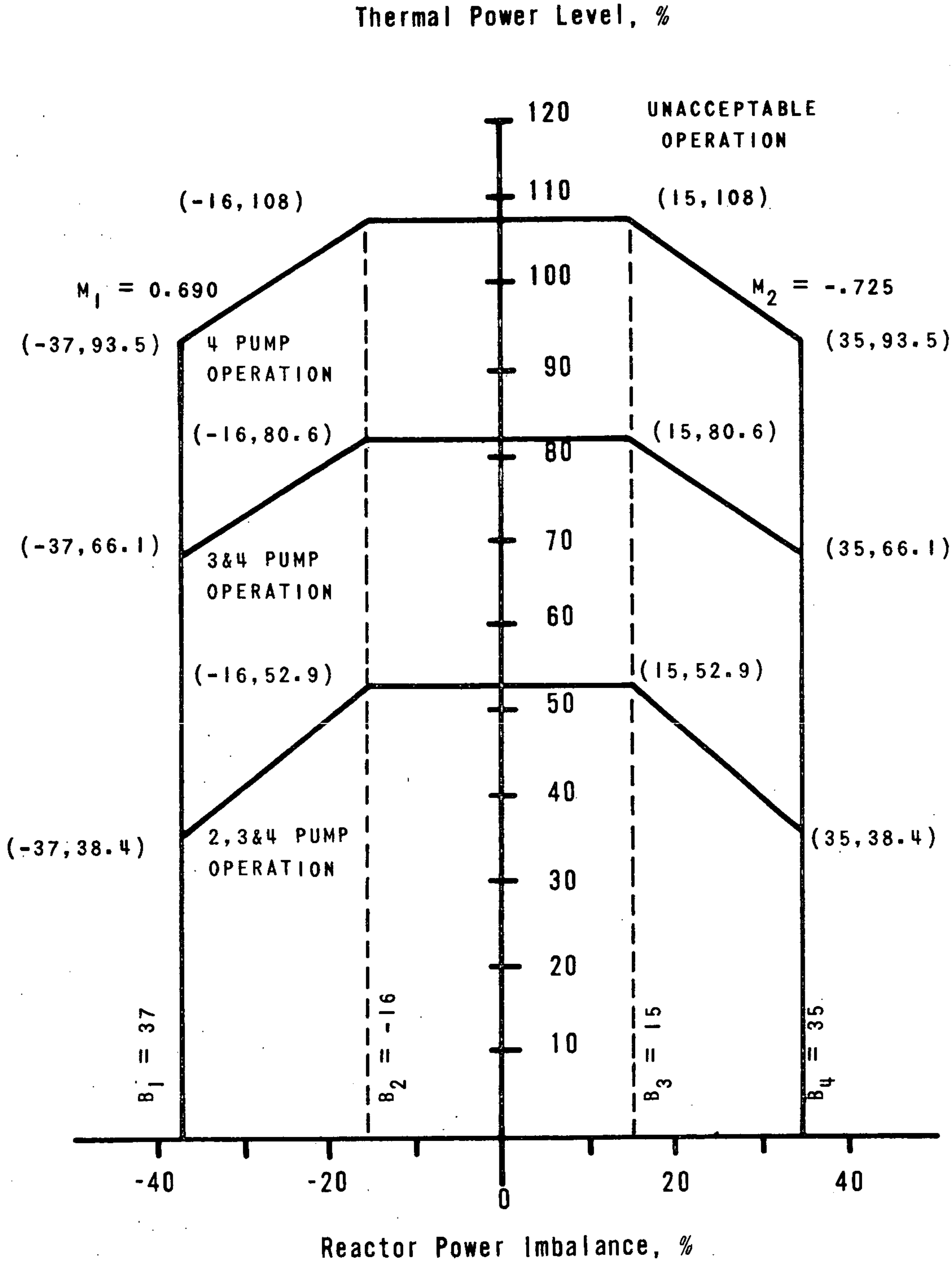


Figure 8-3. Rod Position Limits for Four-Pump Operation, Oconee Unit 1 (0 to 200 ± 10 EFPD)

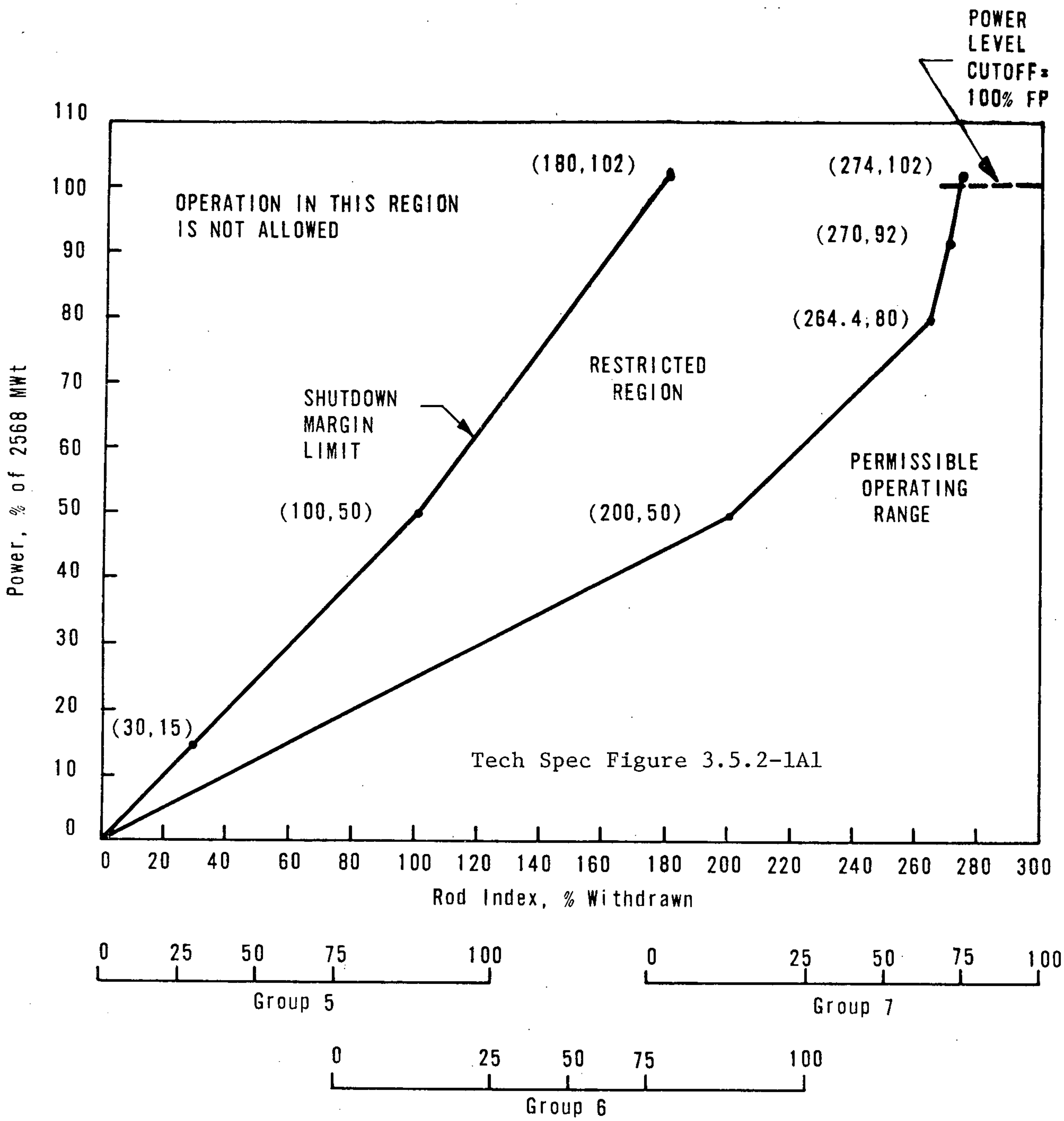


Figure 8-4. Rod Position Limits for Four-Pump Operation, Oconee Unit 1 (After 200 ± 10 EFPD)

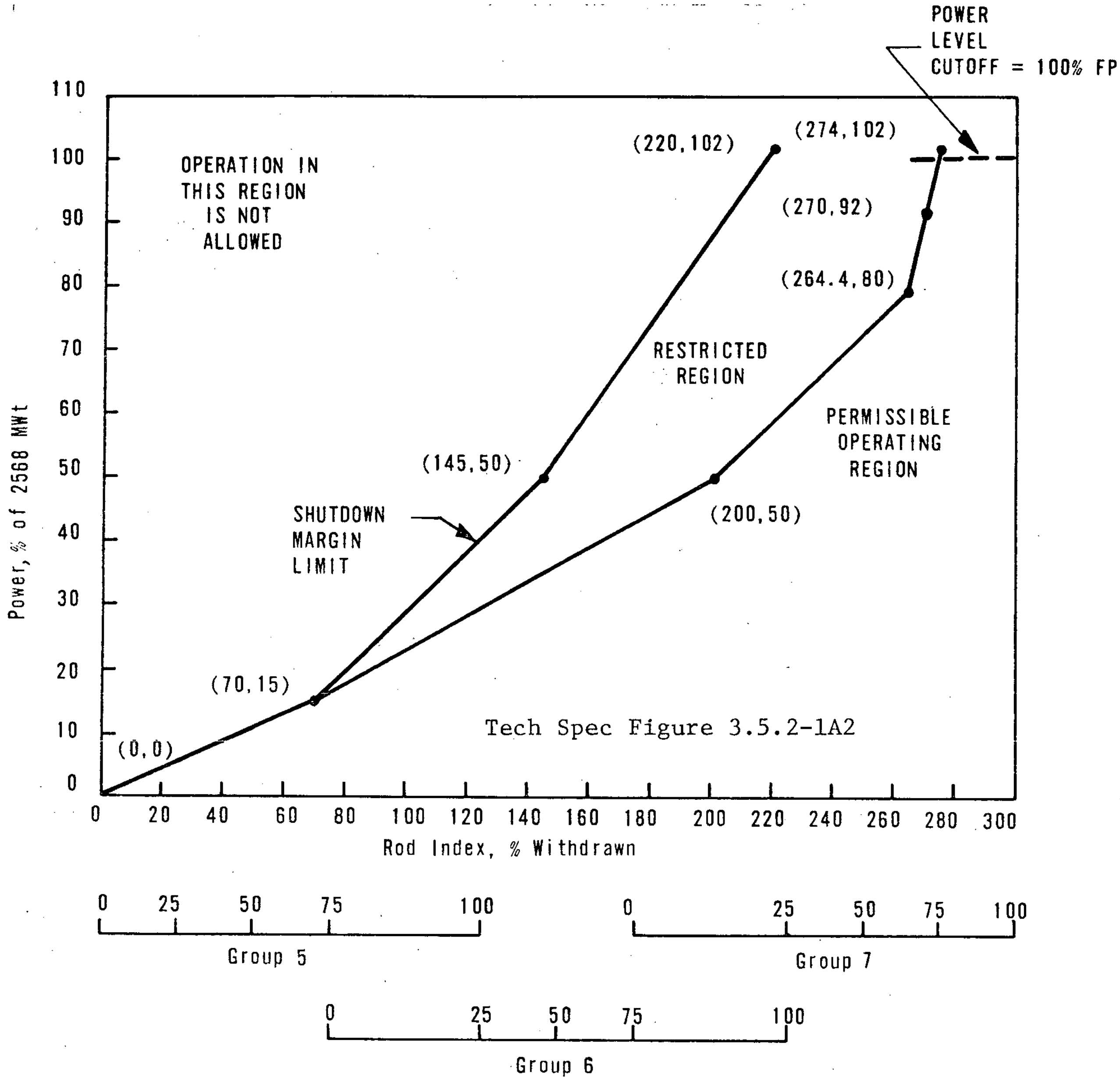


Figure 8-5. Rod Position Limits for Two- and Three-Pump Operation, Oconee Unit 1 (0 to 200 ± 10 EFPD)

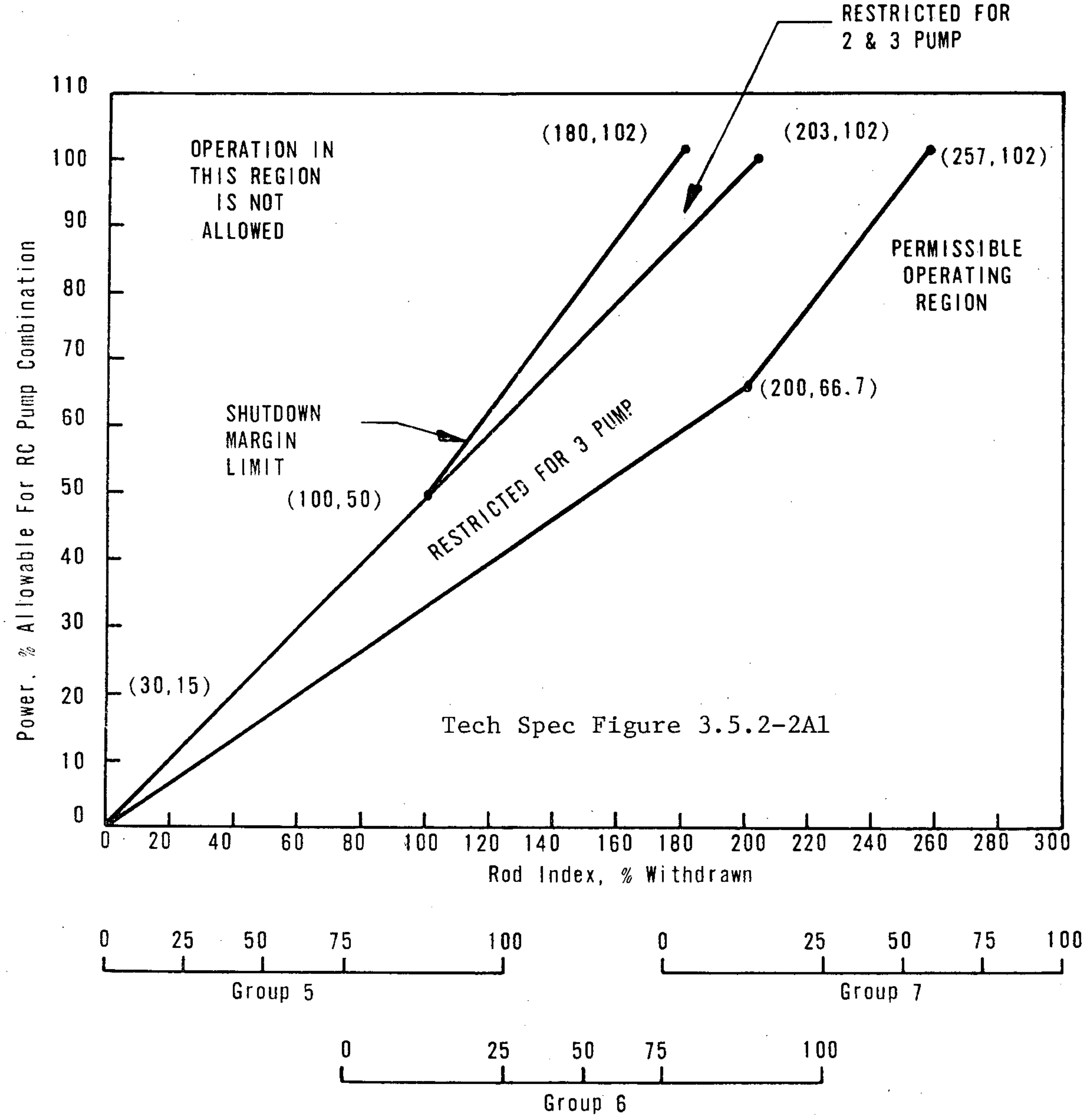


Figure 8-6. Rod Position Limits for Two- and Three-Pump Operation, Oconee Unit 1 (After 200 ± 10 EFPD)

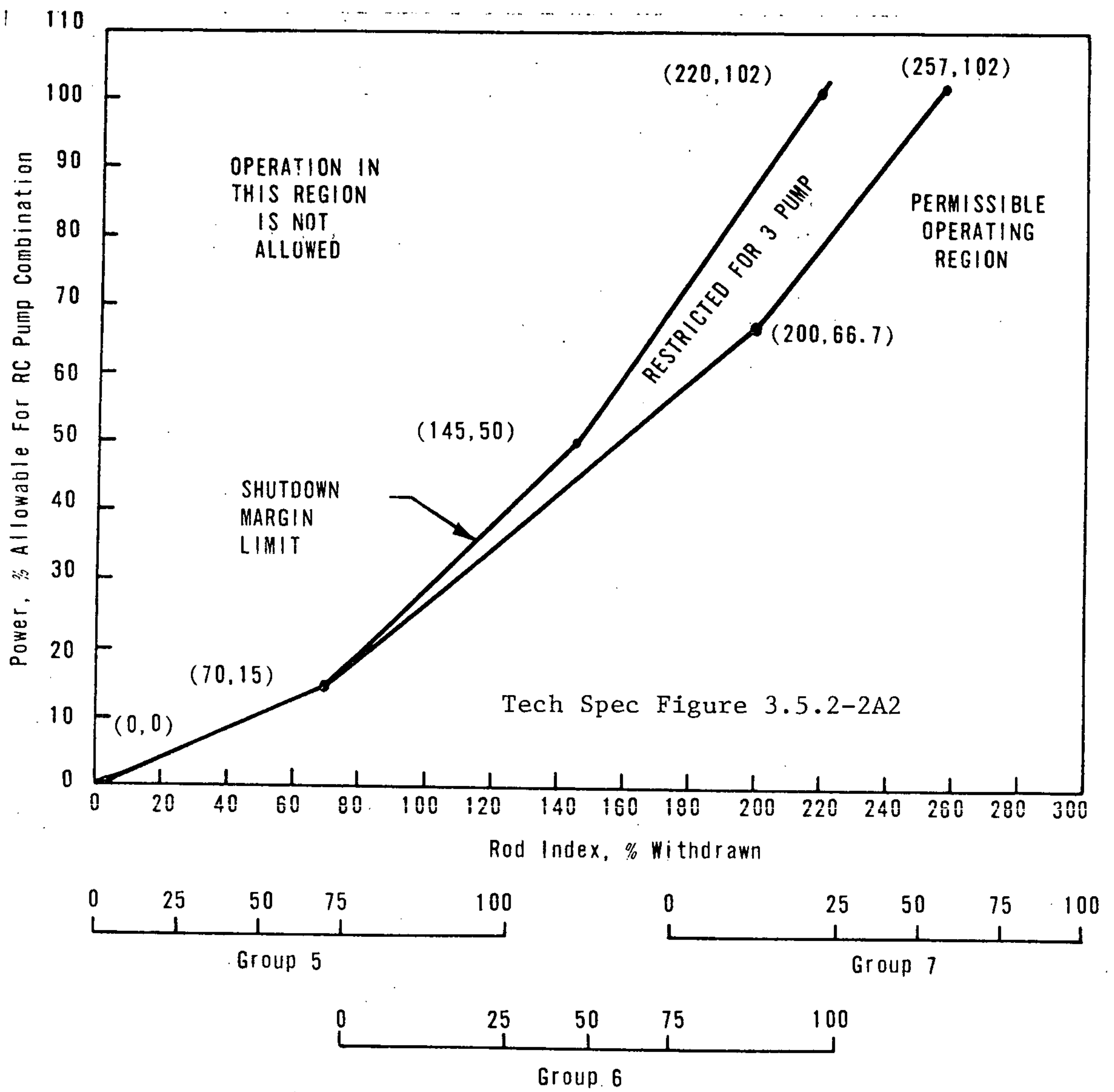
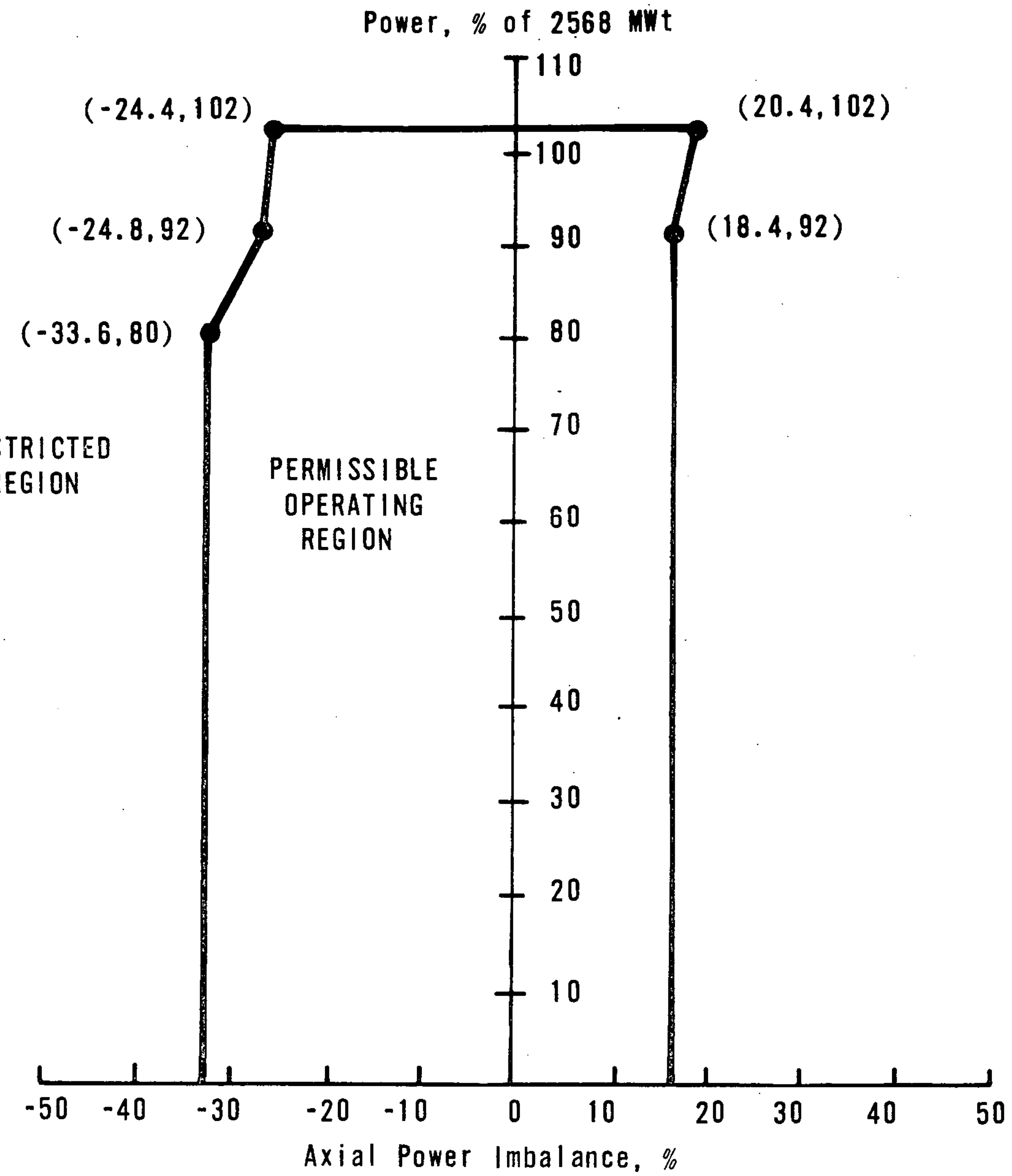
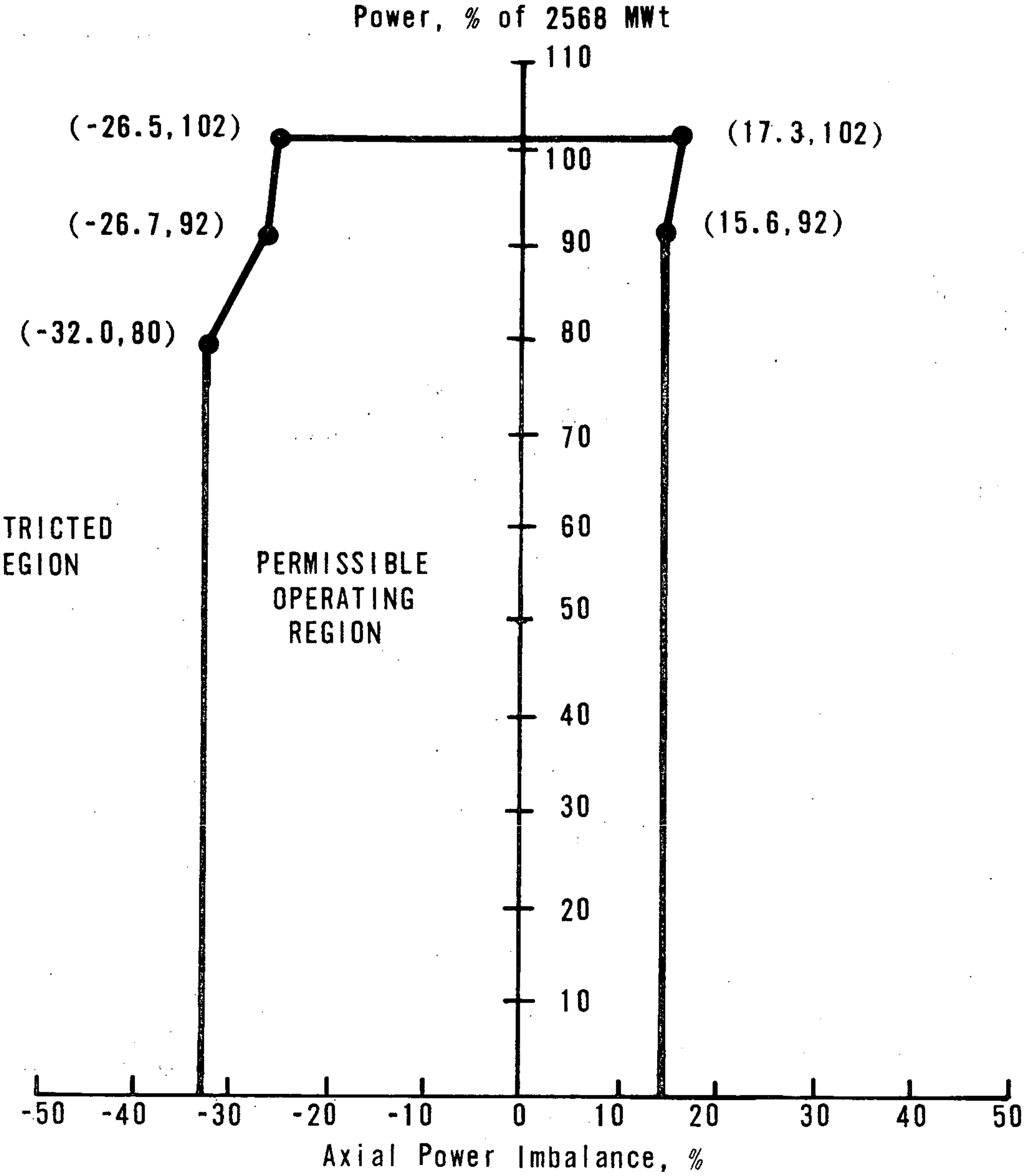


Figure 8-7. Power Imbalance Limits, Oconee Unit 1 (0 to 200 ± 10 EFPD)



Tech Spec Figure 3.5.2-3A1

Figure 8-8. Power Imbalance Limits, Oconee Unit 1 (After 200 ± 10 EFPD)



Tech Spec Figure 3.5.2-3A2

Figure 8-9. APSR Position Limits, Oconee Unit 1
(From 0 to 200 ± 10 EFPD)

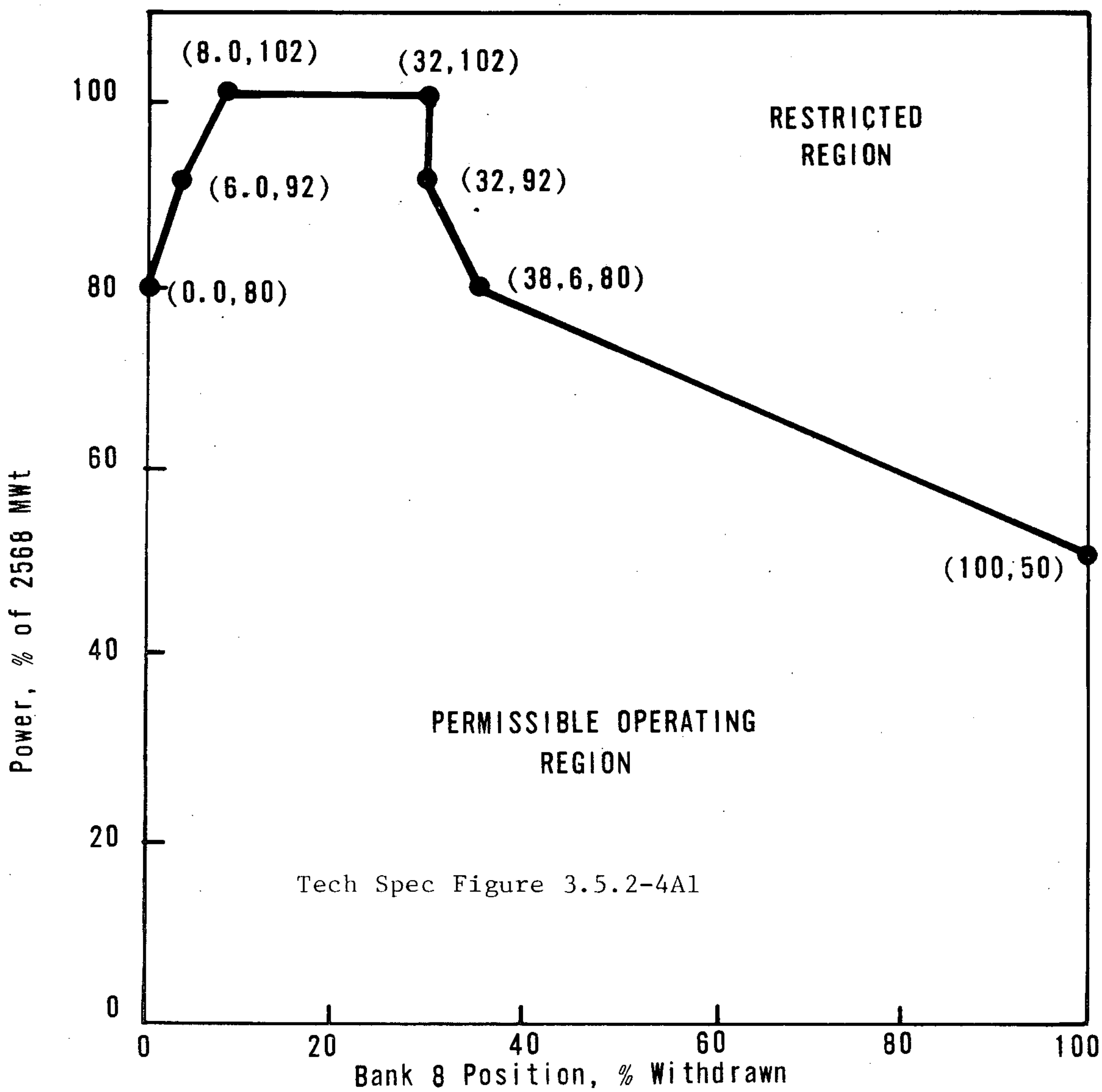
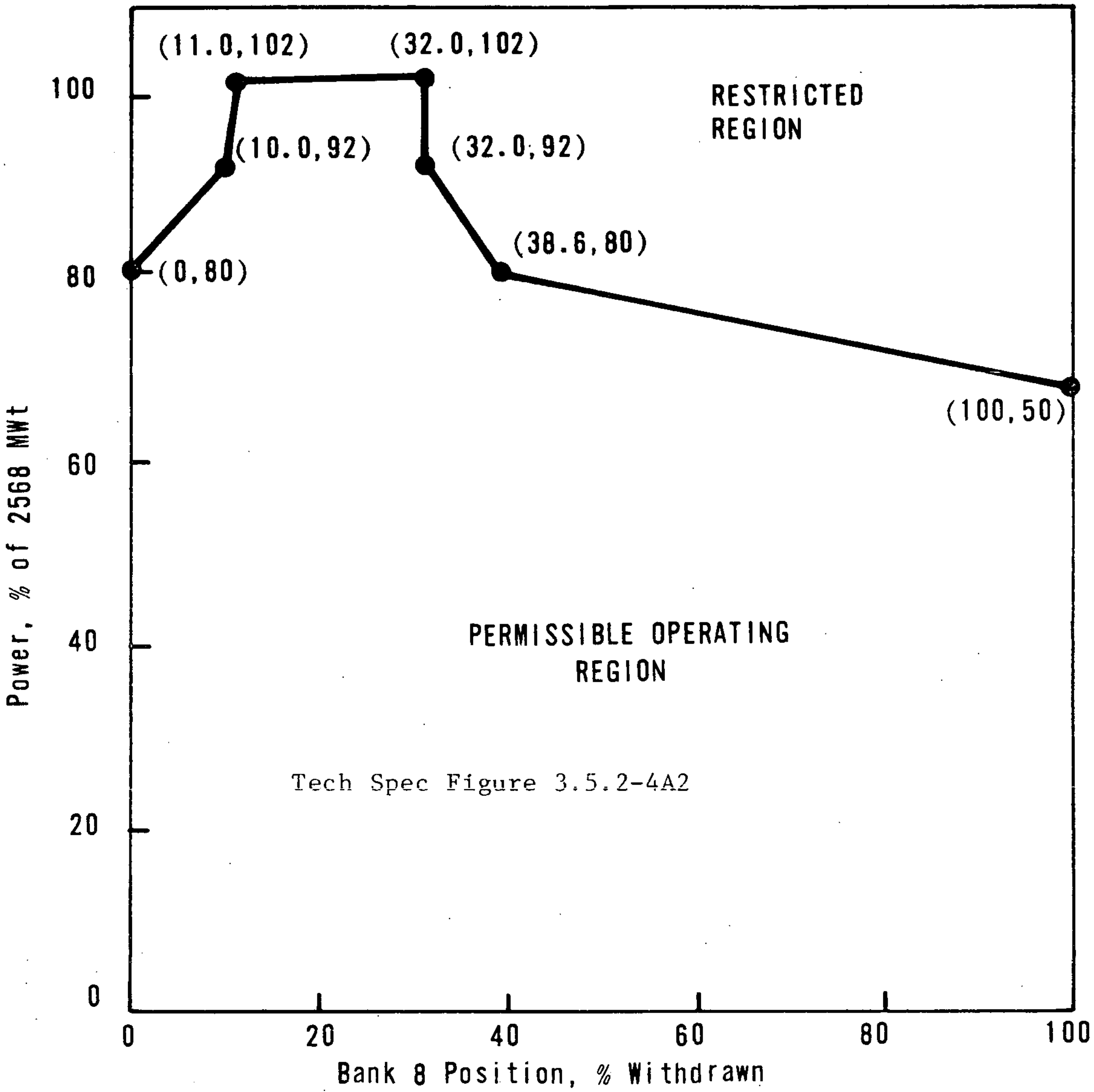


Figure 8-10. APSR Position Limits, Oconee Unit 1
(After 200 ± 10 EFPD)



9. STARTUP PROGRAM — PHYSICS TESTING

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide confirmation for continued safe operation of the unit.

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Acceptable criteria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.66 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop time of 1.40 seconds from fully withdrawn to two-thirds inserted. Since the most accurate position indication is obtained from the zone reference switch at the 75%-inserted position, this position is used instead of the two-thirds inserted position for data gathering. The acceptance criterion of 1.40 seconds corrected to a 75%-inserted position (by rod insertion versus time correlation) is 1.66 seconds.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate. Once criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required in achieving equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentration must be within ±100 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at approximately the all-rods-out configuration and at the hot zero power rod insertion limit. The average coolant temperature is varied by first decreasing then increasing temperature by 5°F. During the change in temperature, reactivity feedback is compensated by discrete change in rod motion, the change in reactivity is then calculated by the summation of reactivity (obtained from reactivity calculation on a strip chart recorder) associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-4}$ $(\Delta k/k)/°F$ (predicted value obtained from Physics Test Manual curves).

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is added to obtain moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$.

9.2.3. Control Rod Group Reactivity Worth

Control bank group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. The boron/rod swap method consists of establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6, and 5 incremental steps. The reactivity changes that occur during these measurements are calculated based on reactimeter data, and differential rod worths are obtained from the measured reactivity worth versus the change in rod group position. The differential rod worths of each of the controlling groups are then summed to obtain integral rod group worths. The acceptance criteria for the control bank group worths are as follows:

1. Individual bank 5, 6, 7 worth:

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 15$$

2. Sum of groups 5, 6, and 7:

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 10$$

9.2.4. Ejected Control Rod Reactivity Worth

After the CRA groups have been positioned near the minimum rod insertion limit, the ejected rod is borated to 100% withdrawn and the worth obtained by adding the incremental changes in reactivity by boration.

After the ejected rod has been borated to 100% withdrawn and equilibrium boron established, the ejected rod is then swapped in versus the controlling rod group and the worth determined by the change in the previously calibrated controlling rod group position. The boron swap and rod swap values are averaged and error-adjusted to determine ejected rod worth. Acceptance criteria for the ejected rod worth test are as follows:

1. $\left|\dfrac{\text{predicted value} - \text{measured value}}{\text{measured value}} \times 100\right| \le 20$

2. Measured value (error-adjusted) $\le$ 1.0% $\Delta k/k$

The predicted ejected rod worth is given in the Physics Test Manual.

9.3. Power Escalation Tests

9.3.1. Core Power Distribution Verification at ~40, 75, and 100% FP With Nominal Control Rod Position

Core power distribution tests are performed at 40, 75, and 100% full power (FP). The test at 40% FP is essentially a check on power distribution in the core to identify any abnormalities before escalating to the 75% FP plateau. Rod index is established at a nominal full power rod configuration at which the core power distribution was calculated. APSR position is established to provide a core power imbalance corresponding to the imbalance at which the core power distribution calculations were performed.

The following acceptance criteria are placed on the 40% FP test:

1. The worst-case maximum linear heat rate must be less than the LOCA limit.

2. The minimum DNBR must be greater than 1.30.

3. The value obtained from the extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than 1.30 or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.

4. The value obtained from the extrapolation of the worst-case maximum linear heat rate to the next power plateau overpower trip setpoint must be less than the fuel melt limit or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.

5. The quadrant power tilt shall not exceed the limits specified in the Technical Specifications.

6. The highest measured and predicted radial peaks shall be within the following limits:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 8$$

7. The highest measured and predicted total peaks shall be within the following limits:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 12$$

Items 1, 2, 5, 6, and 7 above are established to verify core nuclear and thermal calculational models, thereby verifying the acceptability of data from these models for input to safety evaluations.

Items 3 and 4 establish the criteria whereby escalation to the next power plateau may be accomplished without exceeding the safety limits specified by the safety analysis with regard to DNBR and linear heat rate.

The power distribution tests performed at 75 and 100% FP are identical to the 40% FP test except that core equilibrium xenon is established prior to the 75 and 100% FP tests. Accordingly, the 75 and 100% FP measured peak acceptance criteria are as follows:

1. The highest measured and predicted radial peaks shall be within the following limits:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 5$$

2. The highest measured and predicted total peaks shall be within the following limits:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 7.5$$

9.3.2. Incore Vs Excore Detector Imbalance Correlation Verification at ∿40% FP

Imbalances are set up in the core by control rod positioning. Imbalances are read simultaneously on the incore detectors and excore power range detectors for various imbalances. The excore detector offset Vs incore detector offset slope must be at least 1.15. If the excore detector offset Vs incore detector offset slope criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required gain.

9.3.3. Temperature Reactivity Coefficient at ∿100% FP

The average reactor coolant temperature is decreased and then increased by about 5°F at constant reactor power. The reactivity associated with each temperature change is obtained from the change in the controlling rod group position. Controlling rod group worth is measured by the fast insert/withdraw method. The temperature reactivity coefficient is calculated from the measured changes in reactivity and temperature.

Acceptance criteria state that the moderator temperature coefficient shall be negative.

9.3.4. Power Doppler Reactivity Coefficient at ∿100% FP

Reactor power is decreased and then increased by about 5% FP. The reactivity change is obtained from the change in controlling rod group position. Control rod group worth is measured using the fast insert/withdraw method. Reactivity corrections are made for changes in xenon and reactor coolant temperature that occur during the measurement. The power Doppler reactivity coefficient is calculated from the measured reactivity change, adjusted as stated above, and the measured power change.

The predicted value of the power Doppler reactivity coefficient is given in the Physics Test Manual. Acceptance criteria state that the measured value shall be more negative than -0.55×10^{-4} $(\Delta k/k)/\%$ FP.

9.4. Procedure for Use When Acceptance Criteria Are Not Met

If acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. This evaluation is performed by site test personnel with participation by Babcock & Wilcox technical personnel as

required. Further specific actions depend on evaluation results. These actions can include repeating the tests with more detailed attention to test prerequisites, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

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