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October 1979

OCONEE UNIT 2, CYCLE 5

— Reload Report —

Babcock & Wilcox

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the fifth 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 5 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 4 and 5 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 5 operation. In those cases where cycle 5 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 5 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 5 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 5 operation is the currently operating cycle 4. Cycle 4 achieved initial criticality on December 27, 1978 and power escalation began on December 29, 1978. The 100% power level of 2568 MWt was reached on January 9, 1979. No operating anomalies occurred during cycle 4 operation that would adversely affect fuel performance during cycle 5.

3. GENERAL DESCRIPTION

The Oconee 2 reactor core is described in detail in Chapter 3 of the FSAR.[1] The cycle 5 core consists of 177 fuel assemblies, 175 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. The other two fuel assemblies in cycle 5 are demonstration 17 by 17 Mark-CR assemblies. All fuel assemblies in cycle 5 except the 17 by 17 demonstration assemblies 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 5. Batches 5A, 5C, and 6 with initial enrichments of 3.03, 2.53, and 2.91 wt % ^{235}U, respectively, will be shuffled to new locations. Batch 7, with an initial enrichment of 3.07 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 5.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods, 56 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 5 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 5 are identical to those of the reference cycle described in the Oconee 2, cycle 4 reload report.[2] However, the group designations differ between cycle 5 and the reference cycle to minimize power peaking. The cycle 5 locations and enrichments of the BPRA assemblies are shown in Figure 3-4.

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

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Figure 3-1. Core Loading Diagram — Oconee 2, Cycle 5

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						6 R10	6 P11	7	6 P5	6 R6					
B				6 P10	6 P12	6 K1	7	5A E11	7	6 K15	6 P4	6 P6			
C			6 O13	7	5A N3	7	5A L4	7	5A L12	7	5A N13	7	6 O3		
D		6 L14	7	6 H12	7	5A M3	7	5A E10	7	M13	7	6 N8	7	6 L2	
E		6 N14	5A C12	7	6 K9	7	5A K3	7	5A K13	7	6 K7	7	5A C4	6 N2	
F	6 L15	6 A9	7	5A C11	7	5A G5	7	5A K5	7	5A E9	7	5A C5	7	6 A7	6 L1
G	6 M14	7	5A D10	7	5A C9	7	6 H15	(MKCR) 5A L5	6 R8	7	5A C7	7	5A D6	7	6 M2
H	7	5A E5	7	5A F5	7	5A M9	5A M10	5C H10	5A E6	5A E7	7	5A L11	7	5A M11	7
K	6 E14	7	5A N10	7	5A O9	7	6 A8	(MKCR) 5A F11	6 H1	7	5A O7	7	5A N6	7	6 E2
L	6 F15	6 R9	7	5A O11	7	5A M7	7	5A G11	7	5A K11	7	5A O5	7	6 R7	6 F1
M		6 D14	5A O12	7	6 G9	7	5A G3	7	5A G13	7	6 G7	7	5A O4	6 D2	
N		6 F14	7	6 D8	7	5A E3	7	5A M6	7	5A E13	7	6 H4	7	6 F2	
O			6 C13	7	5A D3	7	5A F4	7	5A F12	7	5A D13	7	6 C3		
P				6 B10	6 B12	6 G1	7	5A M5	7	6 G15	6 B4	6 B6			
R						6 A10	6 B11	7	6 B5	6 A6					

XX	Batch
XXX	Cycle 4 location

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Figure 3-2. Oconee 2, BOC 5 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	2.53 21356	3.03 20245	3.03 21342	3.07 0	3.03 20233	3.07 0	3.03 18671	3.07 0
K		2.91 6679	3.07 0	3.03 20905	3.07 0	3.03 17403	3.07 0	2.91 10125
L			3.03 21340	3.07 0	3.03 19063	3.07 0	2.91 7232	2.91 6717
M				2.91 12647	3.07 0	3.03 17076	2.91 7103	
N					2.91 12517	3.07 0	2.91 11322	
O						2.91 7528		
P								
R								

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

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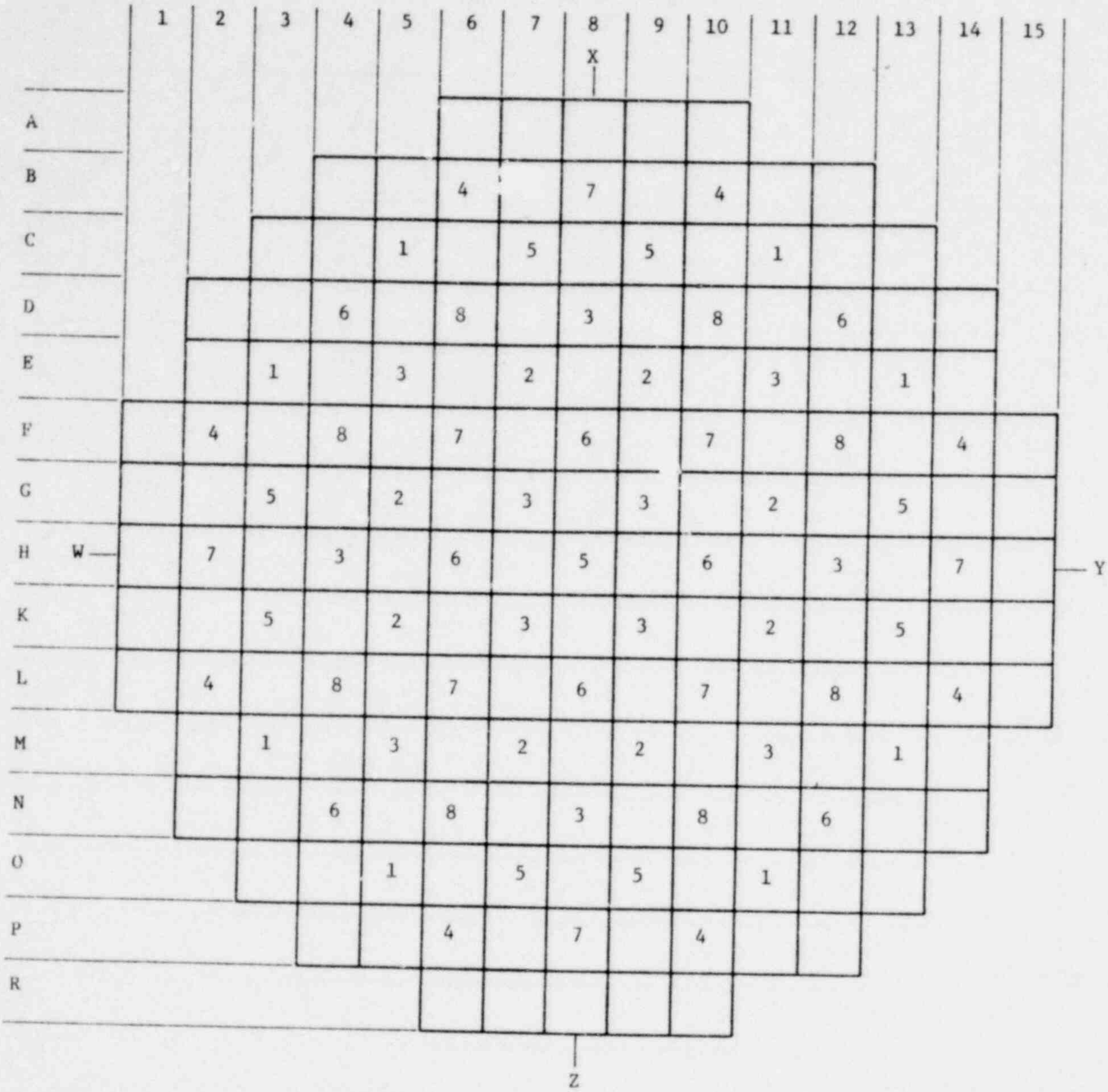
Figure 3-3. Oconee 2 Cycle 5 Control Rod Locations



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

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Figure 3-4. Oconee 2, Cycle 5 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				1.4		0.8		
K			0.5		1.4			
L				0.5		0.8		
M					0.8			
N						0.2		
O								
P								
R								

X.X	BPRA concentration, wt % B_4C in Al_2O_3

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4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies (FAs) and pertinent fuel design parameters for Oconee 2, cycle 5 are listed in Table 4-1. All Mark-B FAs are identical in concept and are mechanically interchangeable. The Mark-CR demonstration assemblies of batch 5 are mechanically identical in function to the Mark-C assemblies of batch 4 described in reference 3.

Retainer assemblies will be used on the 56 FAs containing BPRAs to provide positive retention during reactor operation. Similar retainer assemblies will be used on the two FAs containing the regenerative neutron sources (RNS). The justification for the design and use of the BPRA retainers is described in reference 4, which is also applicable to the retainer assemblies used in Oconee 2, cycle 5.

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

4.2. Fuel Rod Design

The fuel pellet end configuration has changed from a spherical dish for batches 1 through 6 to a truncated cone dish for batch 7. This minor design changes facilitates manufacturing while maintaining the same end void volume. 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 5 is more limiting than other batches due to its longer previous incore exposure time. The batch 5 assembly power histories were analyzed and the most limiting Mark B and Mark C assemblies were used to perform the creep collapse analyses using the CROV computer code and procedures described in reference 5. The collapse times for the most limiting assemblies were both conservatively determined to be more than 30,000 effective full-power

hours (EFPH), which is greater than their maximum projected residence time (Table 4-1).

4.2.2. Cladding Stress

The Oconee 2 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 2 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 2 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 inside diameter (ID).

4.3. Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 7 fuel inserted for cycle 5 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 5 are shown in Table 4-2. LHR capabilities are based on centerline fuel melt and were established using the TAFY3 code[6] with fuel densification to 96.5% of theoretical density (TD).

4.4. Material Design

The batch 7 FAs 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 7 FAs 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-FA plants using the Mark-B fuel assembly:

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU[a] Incore	Discharged	Cumulative net electrical output, mWh[b]
Oconee 1	5	35,300	31,100	25,423,997
Oconee 2	4	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.

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Table 4-1. Fuel Design Parameters and Dimensions

	Twice-burned FAs		Once-burned FAs,	Fresh FAs,
	Batch 5A/5C	Mark CR demonstration	batch 6	batch 7
FA type	Mark-B4	17 × 17 array	Mark-B4	Mark-B4
No. of FAs	50/1	2	56	68
Fuel rod OD, in.	0.430	0.379	0.430	0.430
Fuel rod ID, in.	0.377	0.332	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 (nom), in.	142.25	143.0	142.25	142.25
Fuel pellet initial density (nom), % TD	94.0	94.0	94.0	94.0
Fuel pellet OD (mean specification), in.	0.3695	0.324	0.3695	0.3695
Initial fuel enrichment, wt % ^{235}U	3.03/2.53	3.03	2.91	3.07
BOC burnup (avg), MWd/mtU	19,286	21,356	8883	0
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>32,000
Estimated residence time (max), EFPH	23,962	23,962	27,360	30,336

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Table 4-2. Fuel Thermal Analysis Parameters — Oconee 2, Cycle 5

	Batch 5A/5C[b]	Batch 6	Batch 7
No. of assemblies	52/1	56	68
Nominal pellet density, % TD	94.0	94.0	94.0
Pellet diameter, in.	0.3695	0.3695	0.3695
Stack height, in.	142.25	142.25	142.25
Densified Fuel Parameters[a]			
Pellet diameter, in.	0.3646	0.3646	0.3646
Fuel stack height, in.	140.47	140.47	140.47
Nominal LHR at 2568 MWt, kW/ft	5.80	5.80	5.80
Avg fuel temp. at nominal LHR, F	1320	1320	1320
LHR to ℄ fuel melt, kW/ft	20.15	20.15	20.15

[a] Densification to 96.5% TD assumed.

[b] Batch 5A includes two twice-burned Mark-CR demonstration assemblies.

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5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of design cycles 4 and 5. The values for both cycles were generated using PDQ07.[7-9] The average cycle burnup will be higher in cycle 5 than in the design cycle 4 because of the longer cycle 5 length. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle (BOC) 5 at full power with equilibrium xenon and normal rod positions.

Both cycles 4 and 5 are rodded cycles with a group 7 pull near the end of cycle. The differences between the physics parameters of the two cycles are the result of the longer cycle 5 design life, the initial BPRA loading, and the different shuffle patterns. The critical boron concentrations for cycle 5 are higher than those for cycle 4 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 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 5 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.

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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 4 reload report.[2]

5.2. Analytical Input

The cycle 5 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 design length to 392 EFPD and the incorporation of BPRA clusters to aid in reactivity control. The same calculational methods and design information were used to obtain the important nuclear design parameters for this cycle as for the reference cycle.

The nuclear characteristics of the two batch 5A Mark-CR (17 by 17) FAs are nearly identical to the Mark B (15 by 15) assemblies that make up the balance of batch 5A. Therefore, the presence of the 17 by 17 demonstration FAs will not discernably affect overall core reactivity coefficients or performance. However, since the two Mark-CR assemblies are demonstration assemblies, standard practice dictates their placement in non-limiting core locations during cycle 5.

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Table 5-1. Oconee 2, Cycle 4 and 5 Physics Parameters[a]

	Cycle 4[b]	Cycle 5[c]
Cycle length, EFPD	292	392
Cycle burnup, MWd/mtU	9138	12268
Average core burnup, EOC, MWd/mtU	18,341	20,865
Initial core loading, mtU	82.0	82.1
Critical boron, BOC (no Xe), ppm[d]		
HZP, group 8 inserted	1340	1546
HZP, groups 7 and 8 inserted	1271	1436
HFP, groups 7 and 8 inserted	1078	1244
Critical boron, EOC (eq Xe), ppm		
HZP } group 8 inserted, eq Xe	324	408
HFP } group 8 inserted, eq Xe	55	83
Control rod worth, HFP, BOC, % Δk/k		
Group 6	0.87	1.04
Group 7	0.74	1.05
Group 8	0.47	0.39
Control rod worth, HFP (290 EFPD), % Δk/k		
Group 7	1.03	1.21
Group 8	0.48	0.45
Max ejected rod worth, HZP, % Δk/k[e]		
BOC	0.38	0.60
290 EFPD	0.41	0.53
Max stuck rod worth, HZP, % Δk/k		
BOC	2.22	1.73
290 EFPD	2.09	2.29
Power deficit, HZP to HFP, % Δk/k		
BOC	-1.52	-1.51
290 EFPD	-1.99	-2.15
Doppler coeff, 10^{-5} (Δk/k-°F)		
BOC, 100% power, no Xe	-1.47	-1.52
Doppler coeff, 10^{-5} (Δk/k-°F)		
EOC, 100% power, eq Xe	-1.59	-1.70
Moderator coeff, HFP, 10^{-4} (Δk/k-°F)		
BOC (group 8 in, no Xe, 1363 ppm boron)	-0.63	-0.54
EOC (group 8 in, eq Xe, 17 ppm boron)	-2.58	-2.87
Boron worth, HFP, ppm/%(Δk/k)		
BOC (1150 ppm boron)	107	117
EOC (17 ppm boron)	96	103
Xenon worth, HFP, % Δk/k		
BOC (4 EFPD)	2.65	2.60
EOC (equilibrium)	2.75	2.73
Eff delayed neutron fraction, HFP		
BOC	0.00591	0.00618
EOC	0.00517	0.00517

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(a) Cycle 5 data are for the conditions stated in this report. The cycle 4 core conditions are identified in reference 2.

(b) Based on a cycle 3 length of 300 EFPD.

(c) Cycle 5 data are based on a cycle 4 length of 297 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.

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Table 5-2. Shutdown Margin Calculation — Oconee 2, Cycle 5

	BOC, % Δk/k	EOC, % Δk/k[b]
Available Worth		
Total rod worth, HZP[a]	8.86	9.19
Worth reduction due to burnup of poison material	-0.37	-0.37
Maximum stuck rod, HZP	-1.73	-2.29
Net worth	6.76	6.53
Less 10% uncertainty	-0.68	-0.65
Total available worth	6.08	5.88
Required Rod Worth		
Power deficit, HFP to HZP	1.51	2.15
Max allowable inserted rod worth	1.20	1.48
Flux redistribution	0.47	0.90
Total required worth	3.18	4.53
Shutdown Margin		
Total available worth minus total required worth	2.90	1.35

[a] HZP denotes hot zero power, HFP denotes hot full power.

[b] For shutdown margin calculations, this is defined as approximately 290 EFPD — the latest time in core life at which the transient bank is nearly fully in.

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

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

	8	9	10	11	12	13	14	15
H	0.883	1.0515	1.074	1.189	1.072	1.154	7 0.5995	0.677
K		1.285	1.240	1.053	1.174	1.1165	1.2025	0.6095
L			7 0.688	1.220	8 1.0055	1.2185	1.0335	0.5075
M				1.260	1.3085	1.050	0.7725	
N					1.226	1.0625	0.483	
O						0.639		
P								
R								

X	Inserted rod group No.
X.XX	Relative power density

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6. THERMAL HYDRAULIC DESIGN

The incoming batch 7 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The thermal-hydraulic design evaluation supporting cycle 5 operation utilized the methods and models described in references 1, 2, and 10 except for the core bypass flow and the inclusion of retainers to provide positive holddown of burnable poison rod assemblies (BPRAs).

The maximum core bypass flow due to the removal of all orifice rod assemblies (ORAs) in cycle 4 was 10.4%. For cycle 5 operation, 56 BPRAs will be inserted, leaving 50 vacant assemblies (FAs), 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 4. Reactor core safety limits have been re-evaluated based on the insertion of these BPRAs with retainers and increased core flow. The cycle 4 and 5 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 two Mark-CR demonstration assemblies will be limited to a design peak of 1.39 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 11, 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[12,17] for use until the topical report is completed and reviewed.

The rod bow penalty applicable to cycle 5 was calculated using the interim rod bow penalty evaluation procedure. The burnup is based on the maximum FA burnup of the batch that contains the FA with the maximum radial × local

peak. For cycle 5 this burnup is 15,812 MWd/mtU, a batch 7 fuel assembly. The calculated penalty using this procedure is less than 0.9%. Utilizing the 1% DNB credit for the flow area reduction factor, the actual penalty applied to the DNB calculations is zero.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 4[2]	Cycle 5
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 power	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 inches.

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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 5 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 7 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 10, 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 5 are compared in Table 4-2. A comparison of the cycle 5 thermal-hydraulic maximum design conditions to the previous cycle 4 values is presented in Table 6-1. The key kinetics parameters from the FSAR and cycle 5 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 reported in reference 13. This analysis is 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 analysis[13] is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in reference 13 provided conservative results for the operation of Oconee 2, cycle 5 fuel. Table 7-2 shows the bounding values for allowable LOCA peak linear heat rates for Oconee 2, cycle 5 fuel.

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From examinations of cycle 5 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 5. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 5 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 5 are bounded by the FSAR[1], 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 5 value
BOL Doppler coeff, 10^{-5} $\Delta k/k$-°F	-1.17[a]	-1.52
EOL Doppler coeff, 10^{-5} $\Delta k/k$-°F	-1.33	-1.70
BOL moderator coeff, 10^{-4} $\Delta k/k$-°F	+0.5[b]	-0.54
EOL moderator coeff, 10^{-4} $\Delta k/k$-°F	-3.0	-2.87
All rod bank worth, HZP, % $\Delta k/k$	10.0	8.86
Boron reactivity worth @ 70F, ppm/1% ($\Delta k/k$)	75	82 @ 68F
Max ejected rod worth, HFP, % $\Delta k/k$	0.65	0.55
Dropped rod worth, HFP, % $\Delta k/k$	0.46	0.20
Initial boron conc, HFP, ppm	1400	1244

[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 analysis.

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

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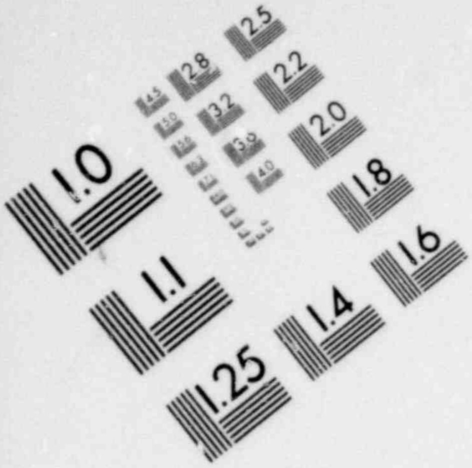
IMAGE EVALUATION
TEST TARGET (MT-3)

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



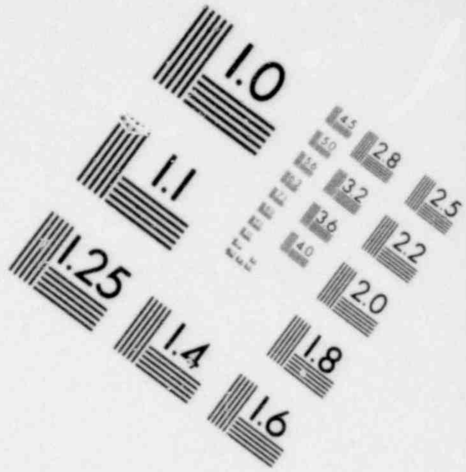
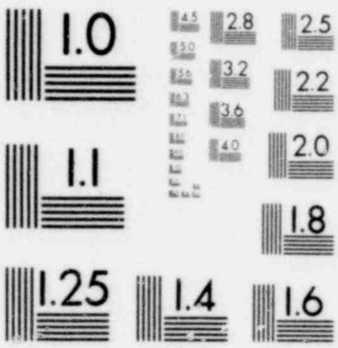
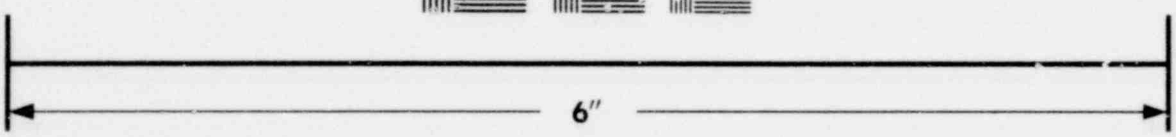
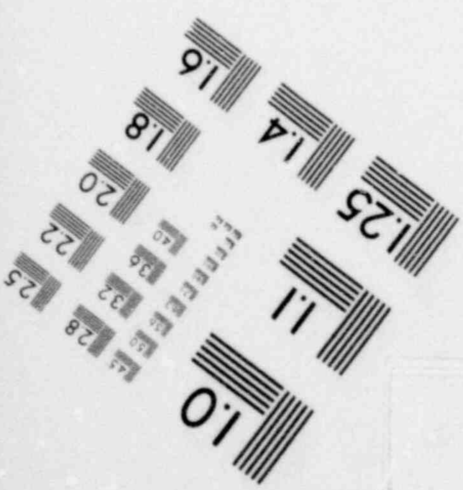


IMAGE EVALUATION TEST TARGET (MT-3)







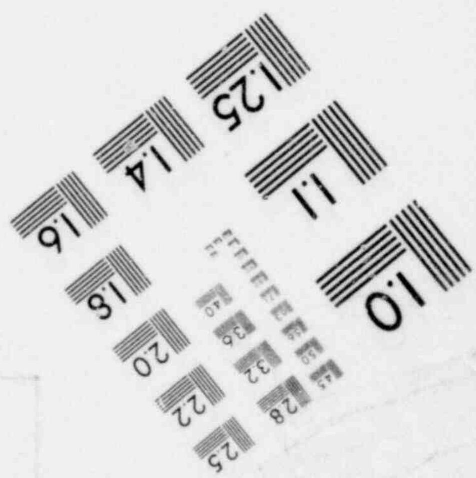


Table 7-2. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Core elevation, ft	Allowable peak LHR, 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 5 operation in accordance with the methods of references 14, 15, and 16 to account for changes in power peaking and control rod worths inherent with a transition to 18-month, lumped burnable poison cycles. In addition:

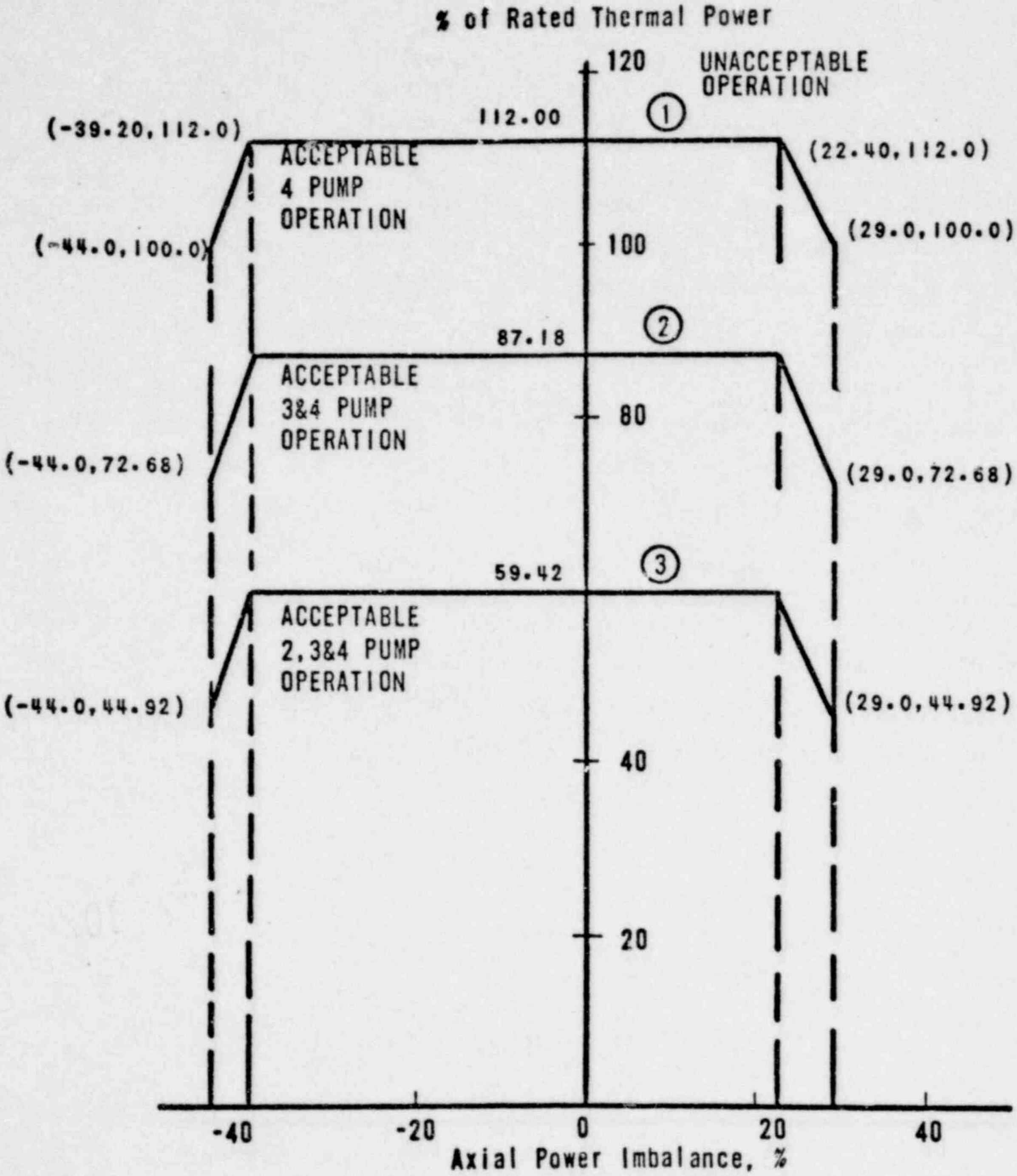
1. The behavior of power peaking with transient xenon has been reviewed for cycle 5. This review has confirmed that the effect of transient xenon on power peaking is conservatively accounted for by the xenon penalty factor of 8%.

2. A flux/flow trip setpoint of 1.08 is established for cycle 5 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.

1387 002

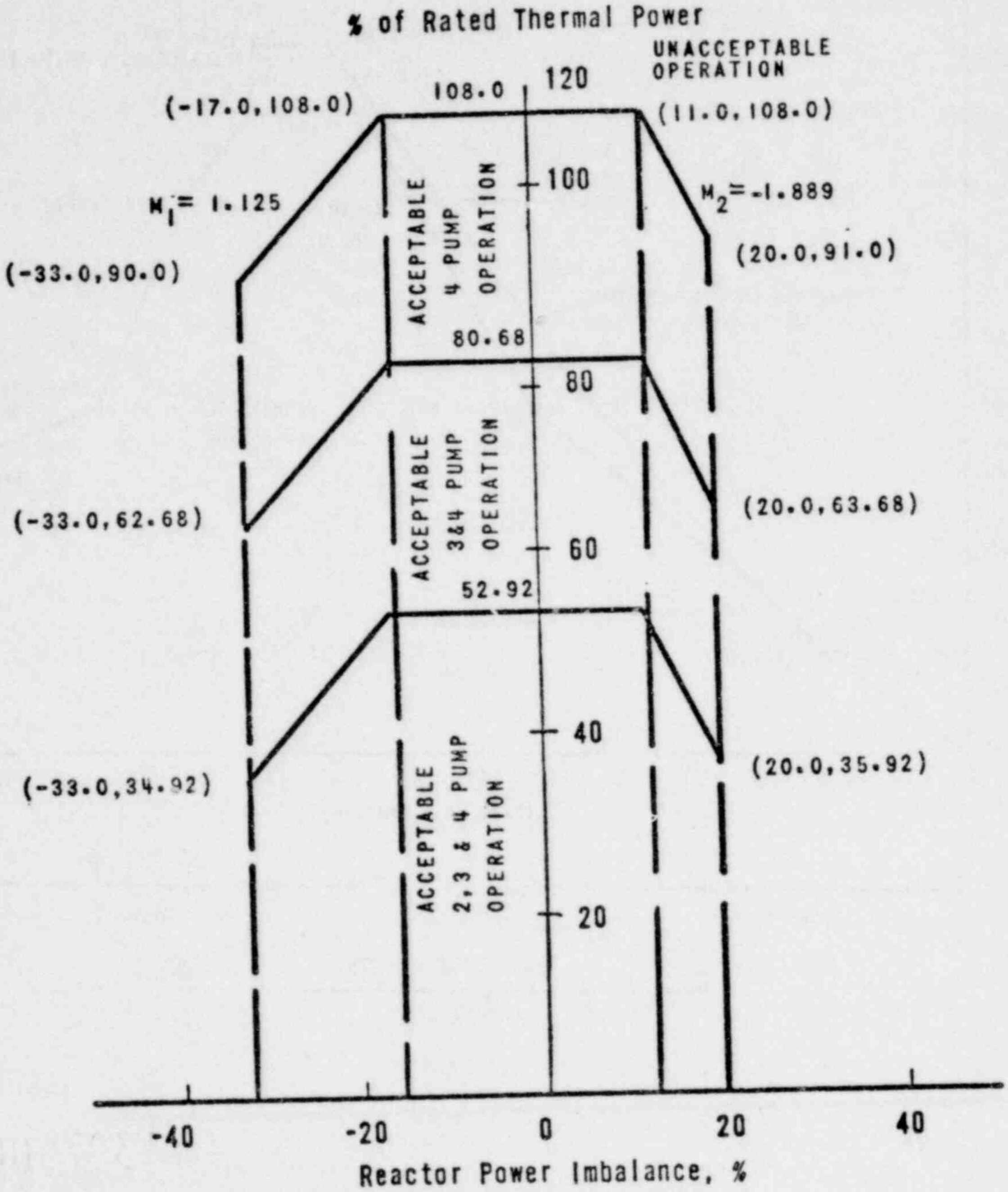
Figure 8-1. Core Protection Safety Limits,
Oconee Unit 2



CURVE	REACTOR COOLANT FLOW (GPM)
1	374,880
2	280,035
3	183,690

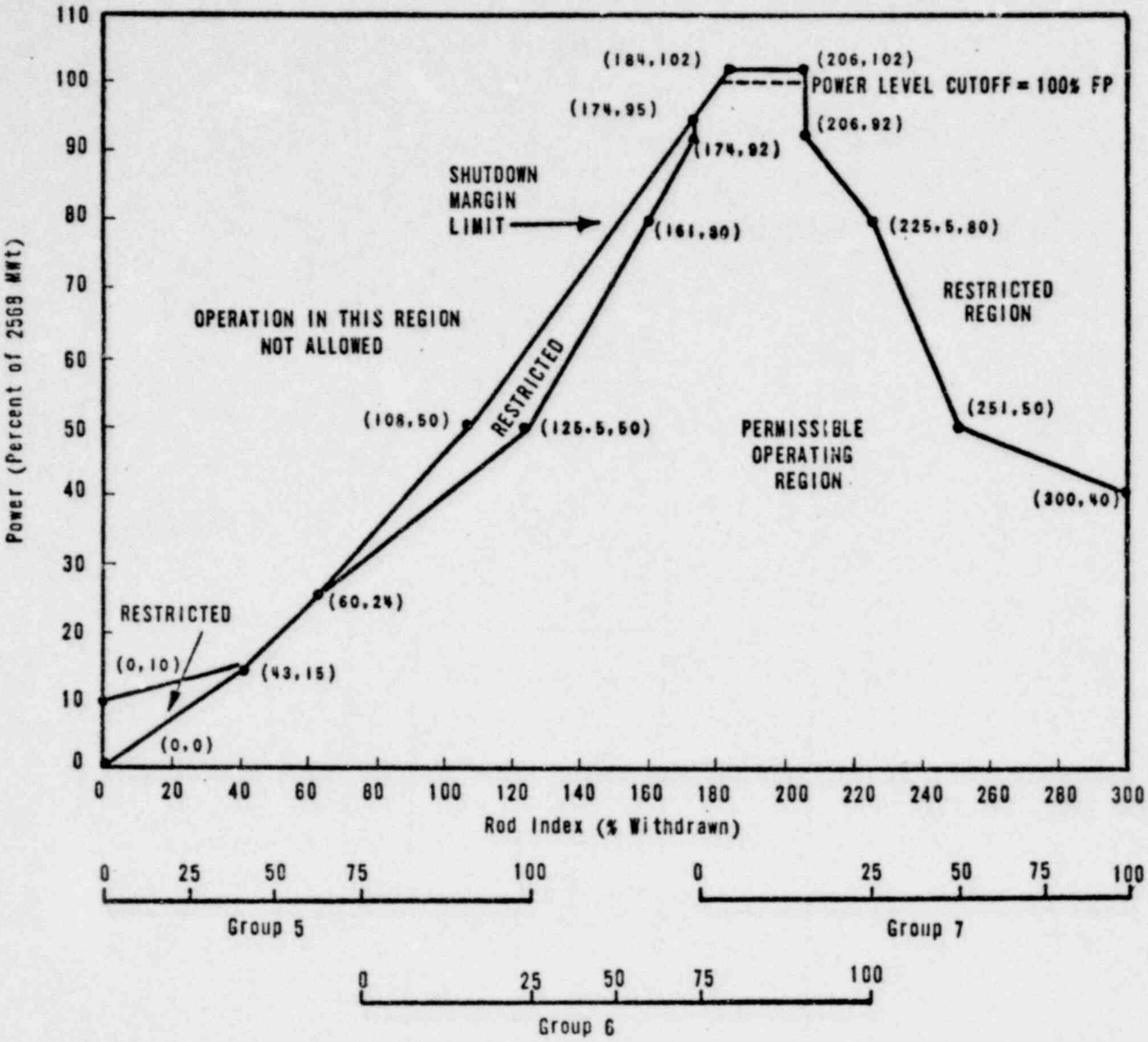
1327 003

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



1397 004

Figure 8-3. Rod Position Limits for Four-Pump Operation — Oconee 2, Cycle 5 (From 0 to 290 ± 10 EFPD)



1397 005

Figure 8-4. Rod Position Limits for Four-Pump Operation — Oconee 2, Cycle 5 (After 290 ± 10 EFPD)

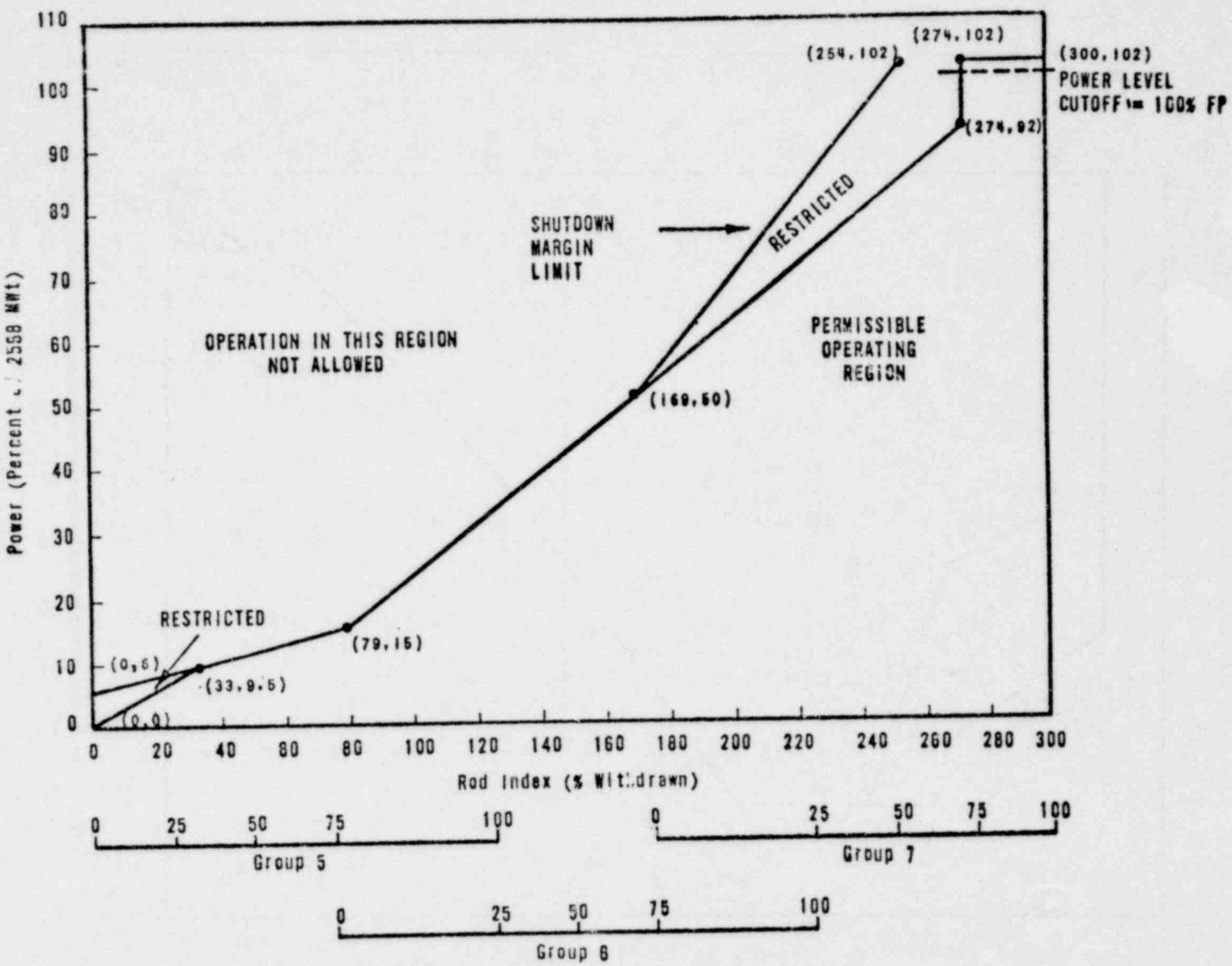


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Figure 8-5. Rod Position Limits for Two- and Three-RC Pump Operation — Oconee 2, Cycle 5 (0 to 290 ± 10 EFPD)

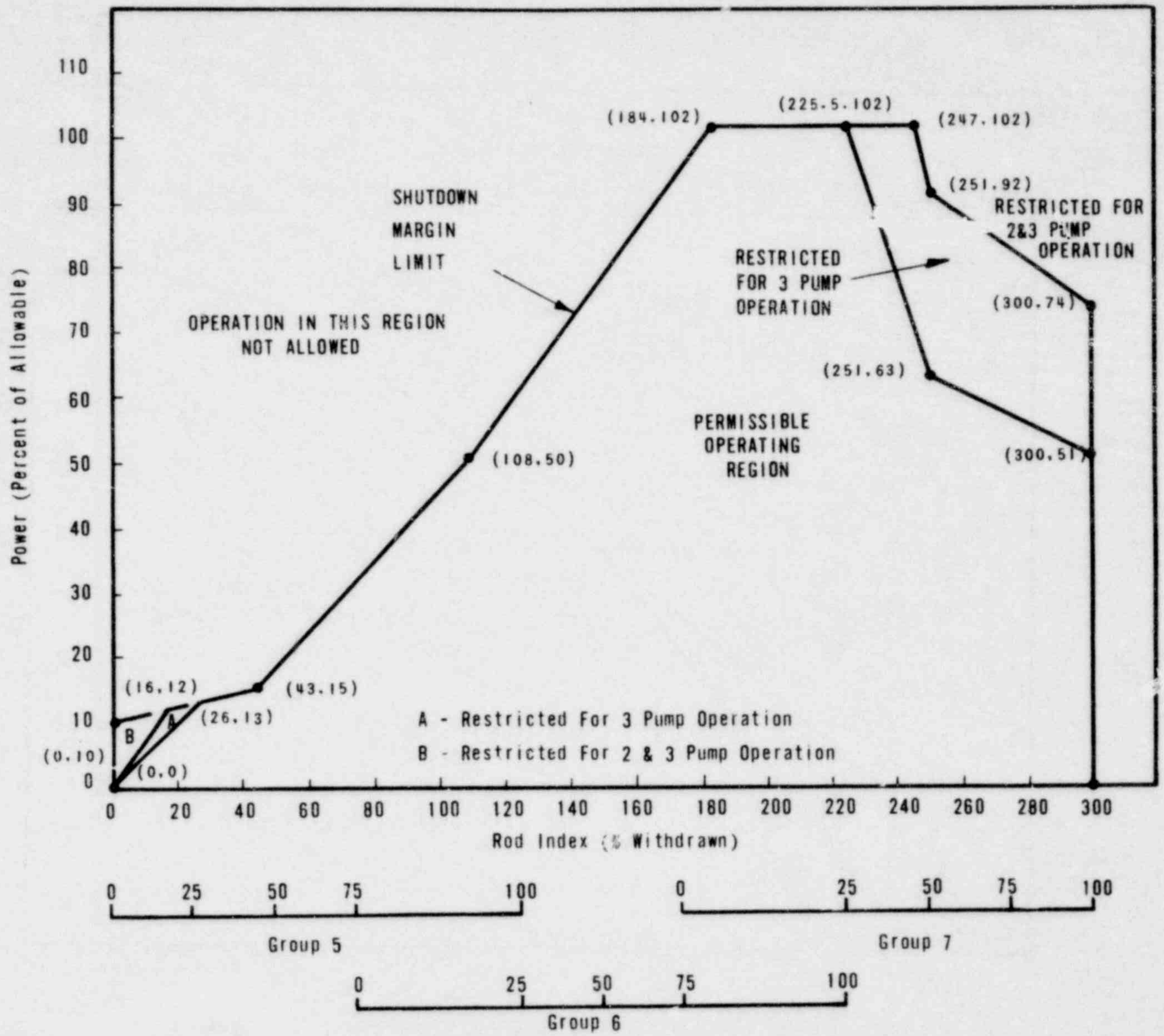

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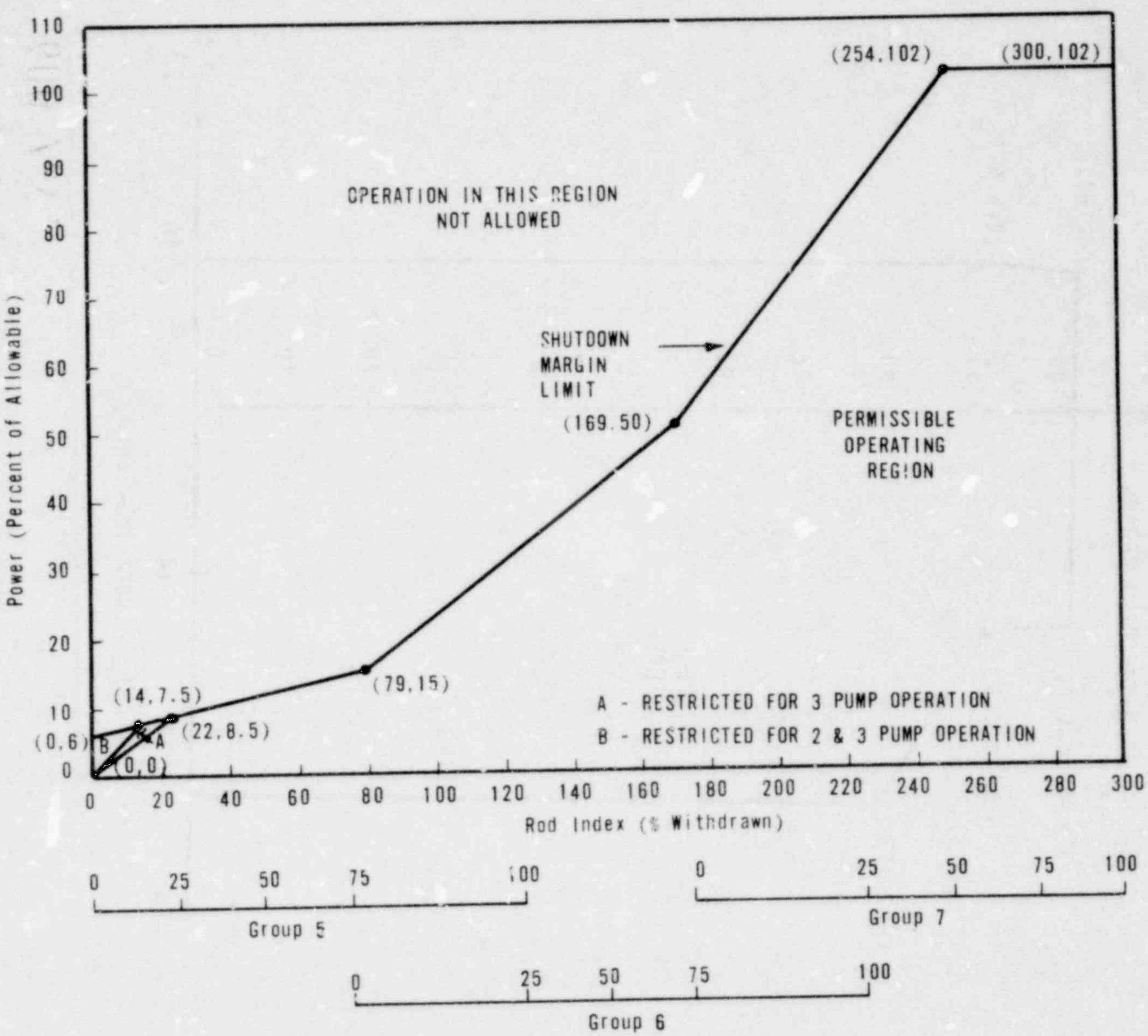
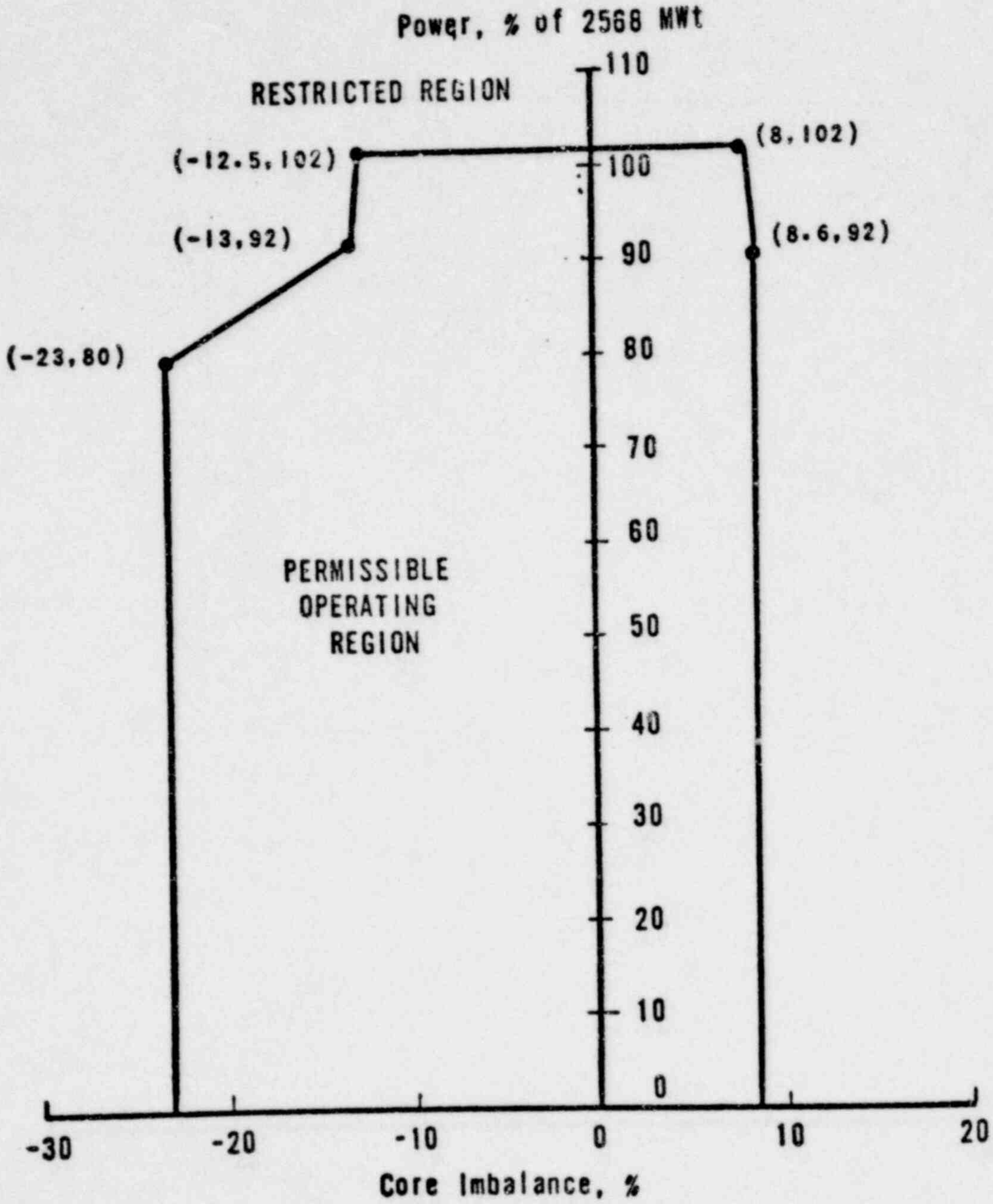
Figure 8-6. Rod Position Limits for Two- and Three-RC Pump Operation — Oconee 2, Cycle 5 (After 290 ± 10 EFPD)



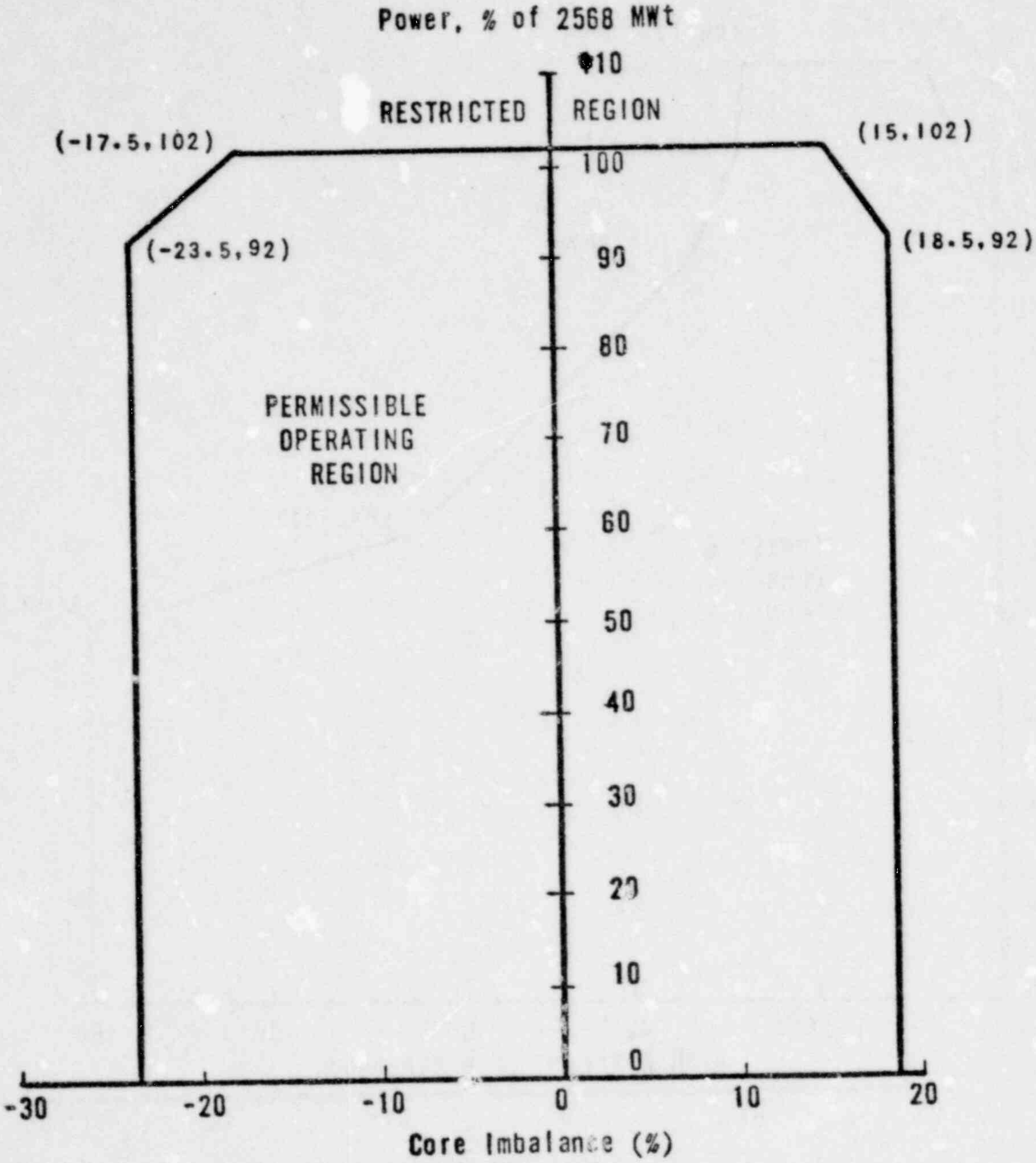
1397 008

Figure 8-7. Operational Power Imbalance Limits — Oconee 2, Cycle 5 (From 0 to 290 ± 10 EFPD)



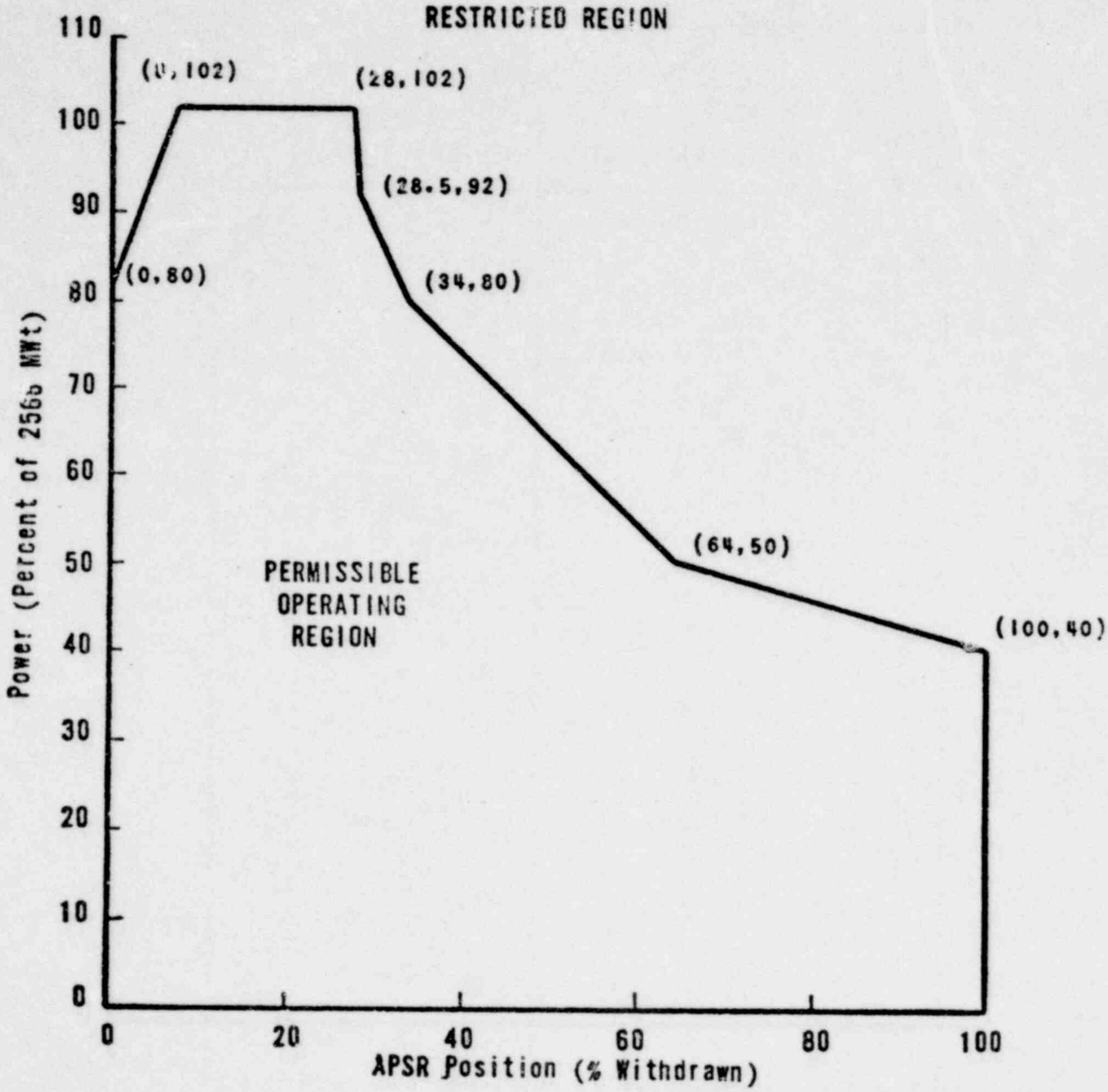
1397 009

Figure 8-8. Operational Power Imbalance Limits — Oconee 2, Cycle 5 (After 290 ± 10 EFPD)



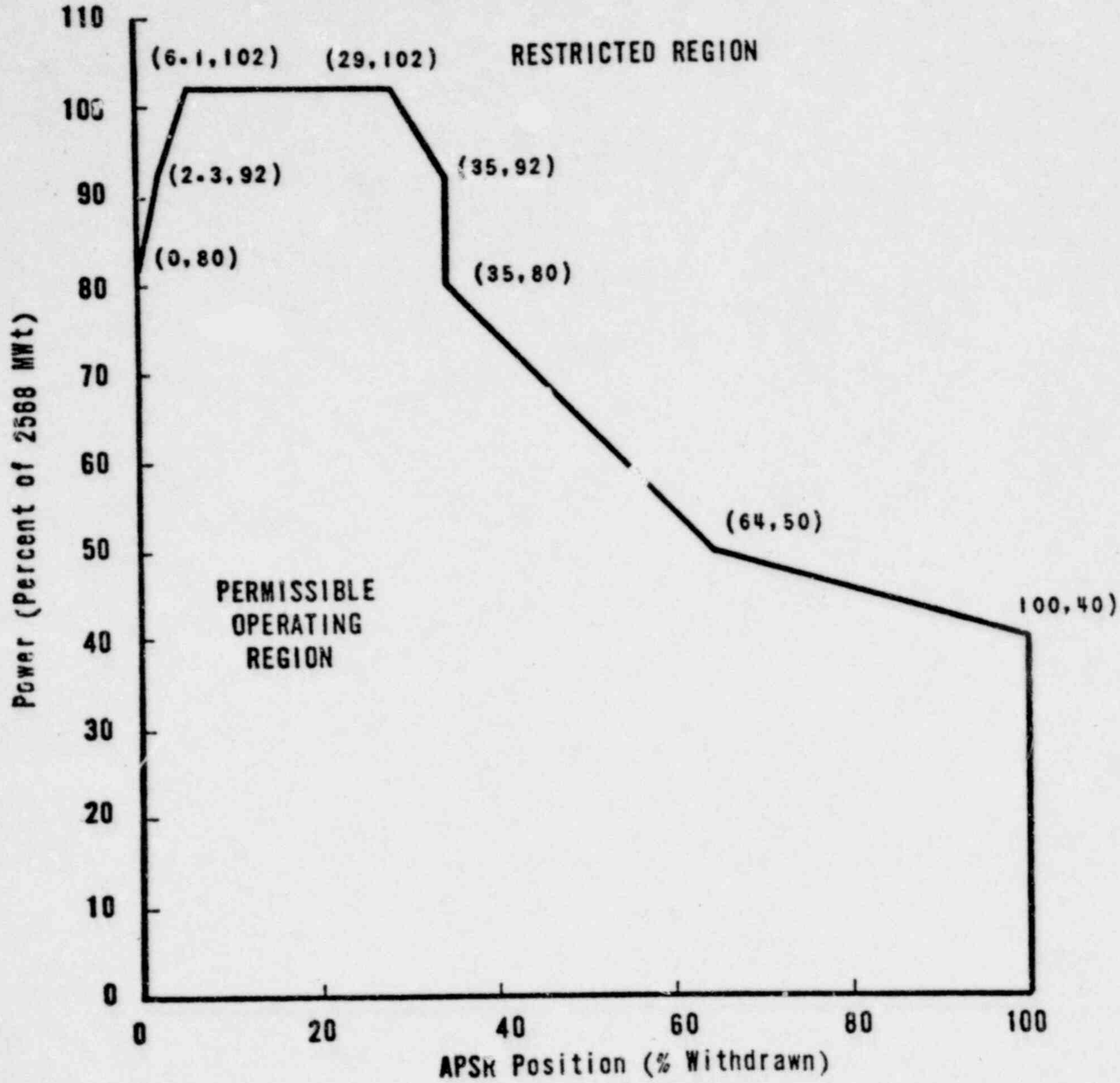
1397 010

Figure 8-9. APSR Position Limits — Oconee 2, Cycle 5
(From 0 to 290 ± 10 EFPD)



1397 011

Figure 8-10. APSR Position Limits — Oconee 2, Cycle 5
(After 290 ± 10 EFPD)



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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 — Control Rod Trip

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 HZP 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 HZP conditions using the boron/rod swap method. The boron/rod swap method consists of establishing a deboration rate in the RC system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6, and 5 in 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 groups 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. $\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 20$

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

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 FP 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.

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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 LHR 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 LHR.

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:

1397 016

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 $\sim$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 versus incore detector offset slope must be at least 1.15. If the excore detector offset versus 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 $\sim$100% FP

The average RC 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 $\sim$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 RC 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.

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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.

1387 018

REFERENCES

[1] Oconee Nuclear Station, Units 1, 2, and 3 — Final Safety Analysis Reports, Docket Nos. 50-269, 50-270, and 50-287, Duke Power Company.

[2] Oconee Unit 2, Cycle 4 Reload Report, BAW-1491, Babcock & Wilcox, Lynchburg, Virginia, August 1978.

[3] Irradiation of Two 17 × 17 Demonstration Assemblies in Oconee 2, Cycle 2 — Reload Report, BAW-1424, Babcock & Wilcox, Lynchburg, Virginia, January 1976.

[4] BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May 1978.

[5] Program to Determine In-Reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084, Rev 2, Babcock & Wilcox, Lynchburg, Virginia, October 1978.

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

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[9] Assembly Calculations and Fitted Nuclear Data, BAW-10116A, Babcock & Wilcox, Lynchburg, Virginia, May 1977.

[10] Oconee 2 Fuel Densification Report, BAW-1395, Babcock & Wilcox, Lynchburg, Virginia, June 1973.

[11] D. B. Vassallo (USNRC) to J. H. Taylor (B&W), Letter (with enclosure), "Calculation of the Effect of Fuel Rod Bowing on the Critical Heat Flux for Pressurized Water Reactors," June 12, 1978 and as revised September 15, 1978.

1397 019

12 J. H. Taylor (B&W) to D. B. Vassallo (USNRC), Letter (with enclosure), "Determination of the Fuel Rod Bow DNB Penalty," December 13, 1978.

13 ECCS Analysis if B&W's 177-FA Lowered-Loop NSS, BAW-10103A, Rev 2, Babcock & Wilcox, Lynchburg, Virginia, September 1977.

14 Power Peaking Nuclear Reliability Factors, BAW-10119, Babcock & Wilcox, Lynchburg, Virginia, January 1977.

15 Normal Operating Controls, BAW-10122, Babcock & Wilcox, Lynchburg, Virginia, August 1978.

16 Verification of the Three-Dimensional FLAME Code, BAW-10125A, Babcock & Wilcox, Lynchburg, Virginia, August 1976.

17 J. H. Taylor (B&W) to S. A. Varga (USNRC), Letter, "Determination of CHF Penalty at 55% Closure," June 22, 1979.

1327 020