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ARKANSAS NUCLEAR ONE, UNIT 1

— Cycle 6 Reload Report —

Babcock & Wilcox
a McDermott company

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth cycle of Arkansas Nuclear One, Unit 1 (ANO-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 ANO-1, this report employs analytical techniques and design bases established in reports that have been submitted to and accepted by the USNRC and its predecessor, the USAEC (see references).

The cycle 5 and 6 reactor parameters related to power capability are summarized briefly 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 ANO-1 can be operated safely for cycle 6 at a rated power level of 2568 MWt.

The cycle 6 core for ANO-1 will contain four once-burned lead test assemblies (LTAs). These assemblies are part of a Department of Energy Extended Burnup Test program. The LTA design is described in reference 2.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Arkansas Nuclear One, Unit 1 is the currently operating cycle 5. This cycle 6 design is based on a nominal cycle 5 length of 455 effective full power days (EFPD).

No anomalies occurred during cycle 5 that would adversely affect fuel performance during cycle 6.

3. GENERAL DESCRIPTION

The ANO-1 reactor core is described in detail in section 3 of the Arkansas Nuclear Station, Unit 1, Final Safety Analysis Report (FSAR).

The cycle 6 core contains 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 comprises 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, with the exception of four batch 7 lead test assemblies (LTAs), which have a nominal loading of 440.0 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 for all fuel assemblies except the LTAs; the corresponding parameters for the LTAs are included in reference 2.

Figure 3-1 is the fuel shuffle diagram for ANO-1, cycle 6. The initial enrichments of batches 6C, 7, and 8 are 3.19, 2.95, and 3.21 wt % ^{235}U, respectively. All the batch 5B assemblies and 23 of the twice-burned batch 6 assemblies will be discharged at the end of cycle 5. The remaining 37 twice-burned batch 6 assemblies (designated batch 6C) will be shuffled to the core interior. Most of the once-burned batch 7 assemblies will be shuffled to locations on or near the core periphery. The 72 fresh batch 8 assemblies will be loaded in a symmetric checkerboard pattern throughout the core. 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, 64 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 of the total pattern (69 control rods) for cycle 6 are identical to those of the reference cycle indicated in the reload report for ANO-1, cycle 5.[3] The group designations, however, differ between cycle 6 and the reference cycle in order 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 ANO-1 Cycle 6

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A						7 K6	7 K2	7 O8	7 K14	7 K10						
B				7 L3	8 F	7 N3	8 F	7 M8	8 F	7 N13	8 F	7 L13				
C			7 H15	8 F	7 L5	8 F	7 K4	8 F	7 K12	8 F	7 L11	8 F	7 R8			
D		7 C10	8 F	6C F1	8 F	6C B6	8 F	6C M2	8 F	6C B10	8 F	6C A10	8 F	7 C6		
E		8 F	7 E10	8 F	6C A6	8 F	7 (P) N11	8 F	7 (P) N5	8 F	6C F15	8 F	7 E6	8 F		
F	7 F9	7 C12	8 F	6C F2	8 F	6C A7	8 F	6C P5	8 F	6C G15	8 F	6C F14	8 F	7 C4	7 F7	
G	7 B9	8 F	7 D9	8 F	7 (LTA) M12	8 F	6C O13	6C B12	6C O3	8 F	7 (LTA) M4	8 F	7 D7	8 F	7 B7	
W — H	7 H13	7 H11	8 F	6C P11	8 F	6C M14	6C D2	6C N2	6C N14	6C E2	8 F	6C B5	8 F	7 H5	7 H3	— Y
K	7 P9	8 F	7 N9	8 F	7 (LTA) E12	8 F	6C C13	6C P12	6C C3	8 F	7 (LTA) E4	8 F	7 N7	8 F	7 P7	
L	7 L9	7 O12	8 F	6C L2	8 F	6C K1	8 F	6C B11	8 F	6C R9	8 F	6C L14	8 F	7 O4	7 L7	
M		8 F	7 M10	8 F	6C L1	8 F	7 (P) D11	8 F	7 (P) D5	8 F	6C R10	8 F	7 M6	8 F		
N		7 O10	8 F	6C R6	8 F	6C P6	8 F	6C E14	8 F	6C P10	8 F	6C L15	8 F	7 O6		
O			7 A8	8 F	7 F5	8 F	7 G4	8 F	7 G12	8 F	7 F11	8 F	7 H1			
P				7 F3	8 F	7 D3	8 F	7 E8	8 F	7 D13	8 F	7 F13				
R						7 G6	7 G2	7 C8	7 G14	7 G10						

Z

LEGEND

Batch

Cycle 6 Location

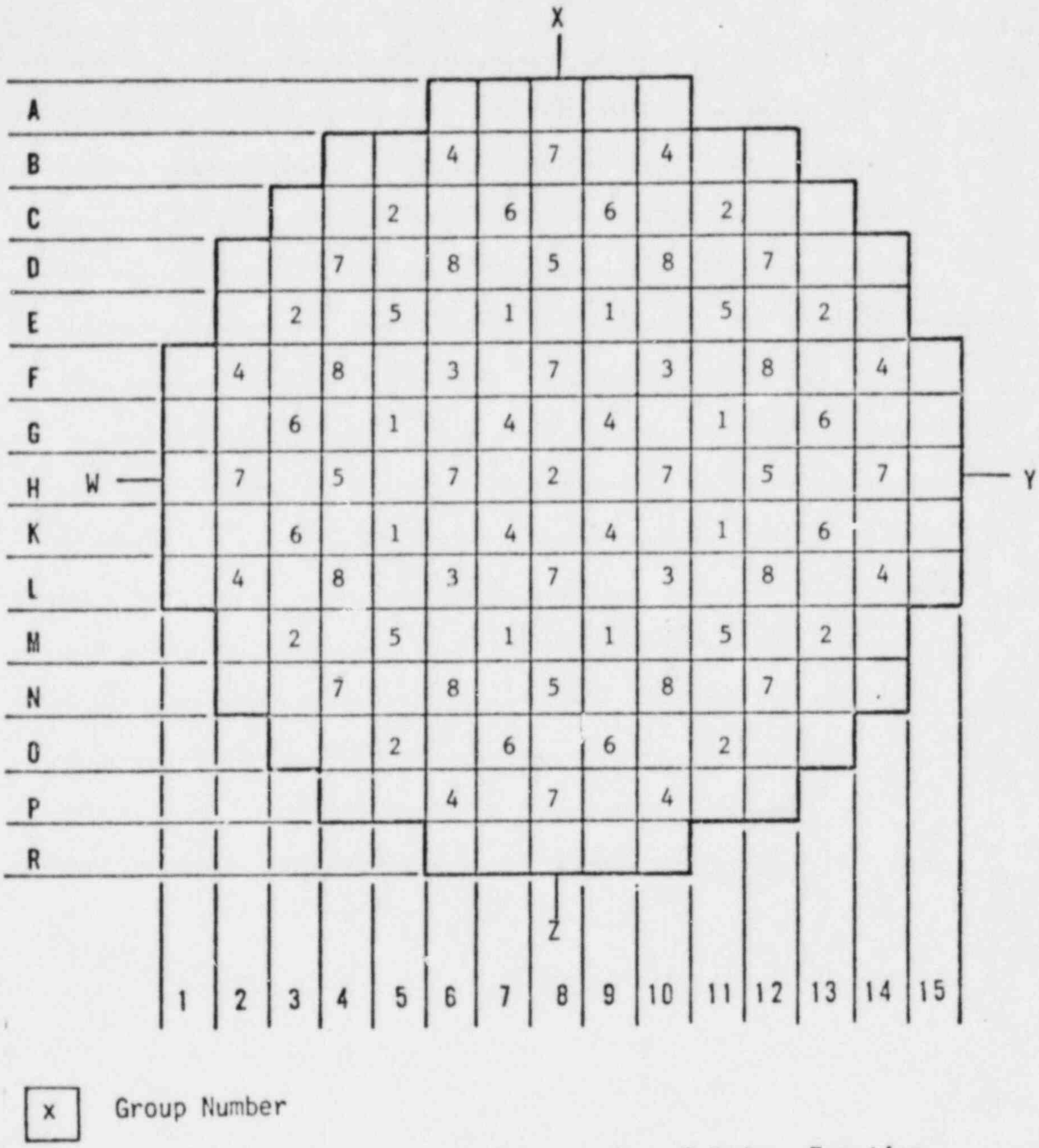
F = Fresh Fuel Assembly
LTA = Lead Test Assemblies
P = Precharacterized Standard Mark B Assembly

Figure 3-2. Enrichment and Burnup Distribution, ANO-1 Cycle 6 Off 455 EFPD Cycle 5

	8	9	10	11	12	13	14	15
H	3.19 20051	3.19 20070	3.19 22678	3.21 0	3.19 22664	3.21 0	2.95 18373	2.95 18118
K		3.19 21951	3.21 0	2.95 17715	3.21 0	2.95 18302	3.21 0	2.95 15576
L			3.19 20941	3.21 0	3.19 23462	3.21 0	2.95 13108	2.95 17818
M				3.19 19880	3.21 0	2.95 17831	3.21 0	
N					3.19 19952	3.21 0	2.95 16854	
O						2.95 10521		
P								
R								

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

Figure 3-3. Control Locations and Group
Designations for ANO-1 Cycle 6



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

Figure 3-4. LBP Enrichment and Distribution, ANO-1 Cycle 6

	8	9	10	11	12	13	14	15
H				1.4		1.1		
K			1.4		1.4		0.2	
L		1.4		1.4		0.5		
M	1.4		1.4		1.1			
N		1.4		1.1		0.2		
O	1.1		0.5		0.2			
P		0.2						
R								

X.X	LBP Concentration, wt % B_4C in Al_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The type of fuel assemblies and pertinent fuel design parameters for ANO-1 cycle 6 are listed in Table 4-1. All fuel assemblies listed are identical in concept and are mechanically interchangeable. All results, references, and identified conservatisms presented in section 4.1 of the cycle 5 reload report are applicable to Mark B4 assemblies. In addition to the assemblies listed, four lead test assemblies (LTAs) are in their second cycle of irradiation in cycle 6. One standard Mark B fuel assembly contains annealed guide tubes to compare with the LTAs. The analysis and justification for the LTAs and annealed guide tubes are reported in reference 2.

Retainer assemblies will be used on the fuel assemblies that contain BPRAs to provide positive retention during reactor operation. This will be the third cycle of operation for the retainer assemblies. The justification for the design and use of the retainers is described in reference 4, and is applicable to ANO-1, cycle 6. Similar retainer assemblies will be used on the two fuel assemblies containing the regenerative neutron sources.

4.2. Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

The batch 6 fuel is more limiting than batches 7 and 8 because of its previous incore exposure time. The batch 6 assembly power histories were analyzed to determine the most limiting three-cycle power history for creep collapse.

This worst-case power history was then compared against a generic analysis to ensure that creep-ovalization will not affect fuel performance during ANO-1 cycle 6. The generic analysis was performed based on reference 5 and is applicable for the batch 6 fuel design.

The creep collapse analysis predicts a collapse time greater than 35,000 effective full-power hours (EFPH), which is longer than the maximum expected residence time of 28,992 EFPH (Table 4-1).

4.2.2. Cladding Stress

The ANO-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 must be less than the minimum specified unirradiated yield strength. In all cases, the margin is greater than 30%. The following conservatisms with respect to the ANO-1 fuel were used in the analysis:

1. Low post-densification internal pressure.
2. Low initial pellet density.
3. High system pressure.
4. High thermal gradient across the cladding.

4.2.3. Cladding Strain

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

4.3. Thermal Design

All fuel in the cycle 6 core is thermally similar. The design of the four batch 7 lead test assemblies is such that the thermal performance of this fuel is equivalent to or slightly better than the standard Mark B design used in the remainder of the fuel. The thermal design analysis of the LTAs using the TACO-2[6] code is described in reference 2.

The results of the thermal design evaluation of the cycle 6 core are summarized in Table 4-2. Cycle 6 core protection limits were based on a linear heat rate (LHR) to centerline fuel melt of 20.15 kW/ft as determined by the TAFY-3 code[7], with no credit taken for the increased LHR capability of the LTA fuel. The maximum fuel rod burnup at EOC 6 is predicted to be less than 42,000 MWd/mtU.

Fuel rod internal pressure has been evaluated with TAFY-3 for the highest burnup fuel rod and is predicted to be less than the nominal reactor coolant system pressure of 2200 psia.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 8 fuel assemblies is identical to that 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 May 31, 1982, 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	Max FA burnup[(a)], MWd/mtU		Cumulative net electrical output,[(b)] MWh
		Incore	Discharged	
Oconee 1	7	41,600	40,000	37,367,569
Oconee 2	6	20,565	36,800	34,229,828
Oconee 3	7	19,650	35,072	35,984,594
TMI-1	5	25,000	32,400	23,840,053
ANO-1	5	32,484	33,220	31,843,210
Rancho Seco	5	33,280	37,730	27,752,205
Crystal River 3	4	20,700	29,900	18,359,090
Davis-Besse 1	3	24,617	25,326	11,473,226

[(a)] As of May 31, 1982.
[(b)] As of January 31, 1982.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch 6	Batch 7	Batch 8
Fuel assembly type	Mark B4	Mark B4, Mark BEB	Mark B4
No. of assemblies	37	64 Mark B, 4 Mark BEB	72
Fuel rod OD (nom), in.	0.430	0.430	0.430
Fuel rod ID (nom), in.	0.377	0.377	0.377
Flexible spacers	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4
Undensified active fuel length (nom), in.	142.25	141.8	141.8
Fuel pellet OD (mean specified), in.	0.3695	0.3686	0.3686
Fuel pellet initial density (nom), % TD	94.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.19	2.95	3.21
Average burnup, BOC, MWd/mtU	21,630	16,554	0
Cladding collapse time, EFPH	>35,000	>35,000	>35,000
Estimated residence time, EFPH (max)	28,344	28,992	26,616

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 6C	Batch 7	Batch 8
No. of assemblies	37	64[a]	72
Initial density, % TD	94.0	95.0	95.0
Pellet diameter, in.	0.3695	0.3686	0.3686
Stack height, in.	142.25	141.80	141.80
Densified Fuel Parameters			
Pellet diameter, in.	0.3646	0.3649	0.3649
Fuel stack height, in.	140.5	140.74	140.74
Nominal linear heat rate at 2568 MWt, kW/ft	5.80	5.79	5.79
Avg fuel temperature at nominal LHR, F	1320	1310	1310
LHR capability, kW/ft[b]	20.15	20.15	20.15

Nominal core avg LHR = 5.79 kW/ft at 2568 MWt.

[a] Four LTAs were also analyzed; the results are reported in reference 2.

[b] Centerline fuel melt based on fuel specification values.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 lists the core physics parameters of design cycles 5 and 6. The values for both cycles were generated using PDQ07. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 6 at full power with equilibrium xenon and nominal rod positions.

Operational changes as well as differences in cycle length, feed enrichment, BPRA loading, and shuffle pattern make it difficult to compare the physics parameters of cycles 5 and 6. 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. The maximum stuck rod worth for cycle 6 is less than that for the design cycle 5 at BOC and EOC. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 6 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. 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 ANO-1 cycle 5 reload report.

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.

5.2. Changes in Nuclear Design

There are no significant core design changes between the reference and reload cycles. 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. There is one significant operational change from the reference cycle: the withdrawal of the APSRs during the last 37 EFPD of cycle 6. The calculated stability index without APSRs is $-0.031\ h^{-1}$, which demonstrates the axial stability of the core. The operating limits (Technical Specification changes) for the reload cycle are given in section 8.

Table 5-1. Physics Parameters for ANO-1, Cycles 5 and 6[(a)]

	Cycle 5[(b)]	Cycle 6[(c)]
Cycle length, EFPD	455	387
Cycle burnup, MWd/mtU	14,259	12,128
Avg core burnup, EOC, MWd/mtU	23,18[illegible]	23,009
Initial core loading, mtU	82.0	82.0
Critical boron — BOC, ppm (no Xe)		
HZP[(d)], group 8 ins	1538	1463
HFP, group 8 ins	1370	1273
Control rod worths — HFP, BOC, % Δk/k		
Group 6	1.26	1.13
Group 7	1.47	1.36
Group 8	0.46	0.42
Control rod worths — HFP, EOC, % Δk/k		
Group 7	1.59	1.40
Max ejected rod worth — HZP, % Δk/k[(e)]		
BOC (N-12), group 8 ins	0.53	0.53
350 EFPD (N-12), group 8 ins	--	0.46
EOC (M-11), group 8 out	0.58	0.47
Max stuck rod worth — HZP, % Δk/k		
BOC (N-12)	1.57	1.50
350 EFPD (N-12), group 8 ins	--	1.63
EOC (N-12), group 8 out	1.74	1.43
Critical boron — EOC, ppm		
HZP, group 8 out, no Xe	487	704
HFP, group 8 out, eq Xe	17	95
Power deficit, HZP to HFP, % Δk/k		
BOC	1.33	1.68
EOC	2.39	2.38

Table 5-1. (Cont'd)

	Cycle 5	Cycle 6
Doppler coeff — HFP, 10^{-5} ($\Delta k/k/F$)		
BOC (no Xe)	-1.52	-1.54
EOC (eq Xe)	-1.82	-1.82
Moderator coeff — HFP, 10^{-4} ($\Delta k/k/F$)		
BOC, (no Xe, crit ppm, group 8 ins)	-0.49	-0.84
EOC, (eq Xe, 0 ppm, group 8 out)	-3.00	-2.89
Boron worth — HFP, ppm/% $\Delta k/k$		
BOC	122	123
EOC	103	109
Xenon worth — HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.58	2.57
EOC (equilibrium)	2.70	2.69
Effective delayed neutron fraction — HFP		
BOC	0.0063	0.0063
EOC	0.0052	0.0053

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

[b] Based on 329 EFPD at 2568 MWt, cycle 4.

[c] Based on 455 EFPD at 2568 MWt, cycle 5.

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

[e] Ejected rod worth for groups 5 through 7 inserted, group 8 as stated.

Table 5-2. Shutdown Margin Calculations for ANO-1, Cycle 6

	BOC, % Δk/k	350 EFPD, % Δk/k	387 EFPD, % Δk/k
Available Rod Worth			
Total rod worth, HZP	8.80	9.27	9.01
Worth reduction due to poison material burnup	-0.42	-0.42	-0.42
Maximum stuck rod, HZP	-1.50	-1.63	-1.43
Net Worth	6.88	7.22	7.16
Less 10% uncertainty	-0.69	-0.72	-0.72
Total available worth	6.19	6.50	6.44
Required Rod Worth			
Power deficit, HFP to HZP	1.97	2.35	2.38
Allowable inserted rod worth	0.41	0.50	0.55
Flux redistribution	0.48	0.93	0.95
Total required worth	2.86	3.78	3.88
Shutdown margin (total available worth minus total required worth)	3.33	2.72	2.56

Note: The required shutdown margin is 1.00% Δk/k.

Figure 5-1. ANO-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	0.90	0.96	1.04	1.27	1.14	1.28	0.96	0.49
K		0.98	1.21	1.16	1.25	1.16	1.14	0.50
L			1.12	1.24	8 0.98	1.28	0.94	0.39
M				1.15	1.24	1.06	0.93	
N					1.08	1.04	0.47	
O						0.58		
P								
R								

X	Inserted Rod Group No.
XXX	Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The fresh batch 8 fuel is hydraulically and geometrically similar to the previously irradiated batch 6C and 7 fuel. The four batch 7 (LTAs) have been analyzed to ensure that they are never the limiting assemblies during cycle 6 operation. The results of the thermal-hydraulic analysis for the LTAs are provided in reference 2.

The thermal-hydraulic design evaluation supporting cycle 6 operation is based on methods and models described in references 1, 3, and 4. The cycle 6 thermal-hydraulic design is identical to cycle 5. The thermal hydraulic design conditions for cycles 5 and 6 are summarized in Table 6-1.

The rod bow penalty for cycle 6 is based on a procedure approved by reference 10. A rod bow penalty was calculated for each fuel batch based on the highest assembly burnup in that batch. The penalty was then applied to the minimum DNBR determined for each batch. Results are summarized in Table 6-2. The fresh batch 8 fuel is the most restrictive both before and after the rod bow penalties are applied. Therefore, the cycle 6 rod bow penalty is based on the highest predicted batch 8 assembly burnup of 16,883 MWd/mtU. The resulting rod bow penalty is 0.1% when 1% credit is taken for use of the flow area reduction hot channel factor in DNBR calculations.

Table 6-1. Maximum Design Conditions, Cycles 5 and 6

	Cycle 5	Cycle 6
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Vessel inlet/outlet coolant temp at 100% power, F	555.6/602.4	555.6/602.4
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.5 cosine	1.5 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	140.2	140.2
Avg heat flux at 100% power, 10^3 Btu/h-ft^2 [(a)]	175	175
Max heat flux at 100% power, 10^3 Btu/h-ft^2 [(b)]	449	449
CHF correlation	BAW-2	BAW-2
Minimum DNBR		
At 112% power	2.05	2.05
At 108% power	2.18	2.18
At 100% power	2.39	2.39

[(a)] Heat flux was based on densified length (in the hottest core location).

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

Table 6-2. DNBR Rod Bow Penalty, ANO-1 Cycle 6

Batch	Max BU, MWd/mtU	Rod bow penalty, %	Max pin peak	DNBR BAW-2	DNBR less rod bow penalty[a]
6C	36,818	5.3	1.27	3.38	3.23
7	33,181	4.3	1.25	3.45	3.34
8	16,883	1.1	1.46	2.77	2.77

[a] Includes 1% credit for flow area reduction hot channel factor.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

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

The radiological dose consequences of the accidents presented in Chapter 14 of the FSAR were re-evaluated for this reload report. The reason for the re-evaluation is that, even though the FSAR dose analyses used a conservative basis for the amount of plutonium fissioning in the core, improvements in fuel management techniques have increased the amount of energy produced by fissioning plutonium. Since ^{239}Pu has different fission yields than ^{235}U, the mixture of fission product nuclides in the core changes slightly as the ^{239}Pu to ^{235}U fission ratio changes; i.e., plutonium fissions produce more of some nuclides and less of others. Since the radiological doses associated with each accident are impacted to a different extent by each nuclide and by various mitigating factors and plant design features, the radiological consequences of the FSAR accidents were recalculated using the specific parameters applicable to cycle 6. The bases used in the dose calculation are identical to those presented in the FSAR except for the following two differences.

1. The fission yields and half-lives used in the new calculations are based on more current data.
2. The steam generator tube rupture accident evaluation considers the increased amount of steam released to the environment via the main steam relief and atmospheric dump valves because of the slower depressurization due to the reduced heat transfer rate caused by tripping of the reactor

coolant pumps upon actuation of the high-pressure injection (a post-TMI-2 modification).

A comparison of the radiological doses presented in the FSAR to those calculated specifically for cycle 6 (Table 7-1) show that some doses are slightly higher and some are slightly lower than the FSAR values. However, except for MHA, all doses are either bounded by the values presented in the FSAR or are a small fraction of the 10 CFR 100 limits; i.e., below 30 Rem to the thyroid or 2.5 Rem to the whole body. For the MHA, the 2-hour EAB, 30-day LPZ thyroid doses are 157.3 and 72.9 Rem, respectively (52 and 24% of 10 CFR 100 limits, respectively), while the 2-hour EAB, 30-day LPZ whole body doses are 7.1 and 2.4 Rem, respectively (28 and 10% of 10 CFR 100 limits, respectively). The small increases in some doses are essentially offset by reductions in other doses. Thus, the radiological impact of accidents during cycle 6 are not significantly different than those described in Chapter 14 of the FSAR.

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. The thermal parameters for Batches 6B, 7, and 8 are given in Table 4-2. The cycle 6 thermal-hydraulic maximum design conditions are compared to the previous cycle 5 values in Table 6-1. These parameters are common to all the accidents considered in this report. The key kinetics parameters from the FSAR and cycle 6 are compared in Table 7-2.

A generic LOCA analysis for a B&W 177-FA, lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model (reported in BAW-10103).[11] This analysis is generic since the limiting values of key parameters for all plants in this category were used. Furthermore, the combination of average fuel temperatures as a function of LHR and 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 and substantiated by reference 12 provide conservative results for the operation of the reload cycle. Table 7-3 shows the bounding values for allowable LOCA peak LHRs for ANO-1 cycle 6 fuel. The basis for two sets of LOCA limits is provided in reference 13.

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 ANO-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, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of FSAR and Cycle 6 Doses

Accidents	FSAR dose, Rem	Cycle 6 dose, Rem
Fuel Handling		
2-hour thyroid (EAB)	0.920	1.060
2-hour whole body (EAB)	0.540	0.5888
30-day thyroid (LPZ)	(a)	0.1582
30-day whole body (LPZ)	(a)	0.08787
Rod Ejection		
2-hour thyroid (EAB)	11.40	12.13
2-hour whole body (EAB)	0.014	0.0132
30-day thyroid (LPZ)	8.3	9.037
30-day whole body (LPZ)	0.0099	0.009724
Steam Line Break		
2-hour thyroid (EAB)	1.6	1.712
2-hour whole body (EAB)	(a)	0.01328
30-day thyroid (LPZ)	(a)	0.2555
30-day whole body (LPZ)	(a)	0.001981
Steam Generator Tube Failure		
2-hour thyroid (EAB)	0.0087	6.139
2-hour whole body (EAB)	0.16	1.13
30-day thyroid (LPZ)	(a)	0.9162
30-day whole body (LPZ)	(a)	0.1686
Waste Gas Tank Release		
2-hour thyroid (EAB)	0.22	0.05409
2-hour whole body (EAB)	(a)	3.366
30-day thyroid (LPZ)	4.5	0.008072
30-day thyroid (LPZ)	(a)	0.5023
LOCA		
2-hour thyroid (EAB)	3.6	3.992
2-hour whole body (EAB)	0.057	0.05873
30-day thyroid (LPZ)	1.66	2.038
30-day whole body (LPZ)	0.043	0.0446
Maximum Hypothetical Accident		
2-hour thyroid (EAB)	153.0	157.3
2-hour whole body (EAB)	10.0	7.099
30-day thyroid (LPZ)	64.1	72.91
30-day whole body (LPZ)	3.4	2.439

[a] Not calculated in FSAR.

Table 7-2. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR and densification report value	ANO-1 cycle 6
Doppler coeff (BOC), 10^{-5} $\Delta k/k/°F$	-1.17	-1.54
Doppler coeff (EOC), 10^{-5} $\Delta k/k/°F$	-1.30	-1.82
Moderator coeff (BOC), 10^{-4} $\Delta k/k/°F$	0.0[a]	-0.84
Moderator coeff (EOC), 10^{-4} $\Delta k/k/°F$	-4.0[b]	-2.89
All-rod group worth (HZP), % $\Delta k/k$	12.9	8.80
Initial boron concentration, ppm	1150	1273
Boron reactivity worth (HFP), ppm/% $\Delta k/k$	100	123
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.27
Dropped rod worth (HFP), % $\Delta k/k$	0.65	0.20

[a] $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ was used for the moderator dilution analysis.
[b] -3.0×10^{-4} $\Delta k/k/°F$ was used for the steam line failure analysis.

Table 7-3. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Core elevation, ft	Allowable peak LHR, first 50 EFPD, kW/ft	Allowable peak LHR, balance of cycle, kW/ft
2	14.5	15.5
4	16.1	16.6
6	17.5	18.0
8	17.0	17.0
10	16.0	16.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 6 operation to account for changes in power peaking and control rod worths inherent with the transition to 18-month LBP fuel 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. The following pages contain the revisions to previous Technical Specifications.

Figure 8-1. Core Protection Safety Limits
(Tech Spec Figure 2.1-2)

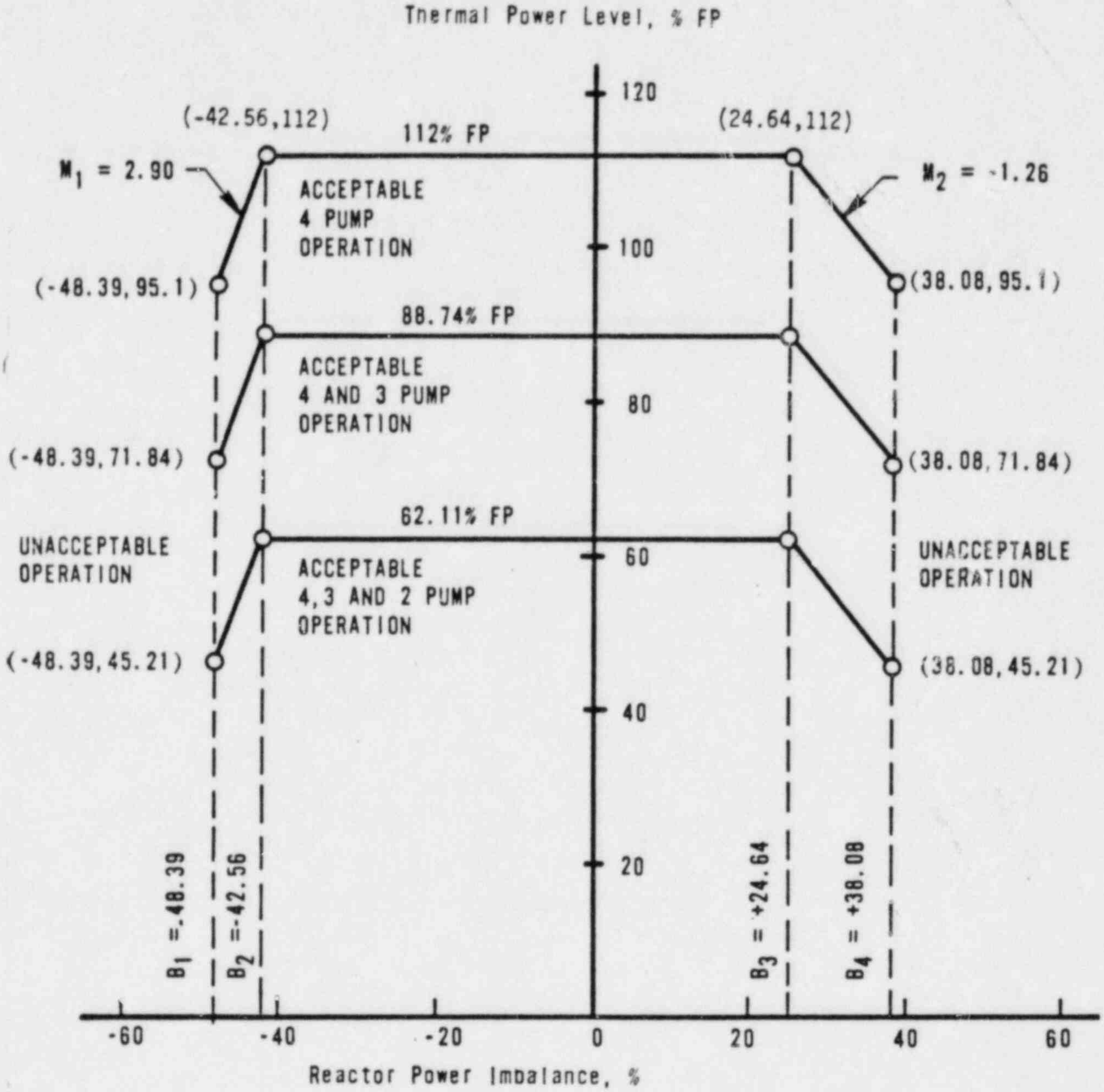


Figure 8-2. Protective System Maximum Allowable Setpoints
(Tech Spec Figure 2.3-2)

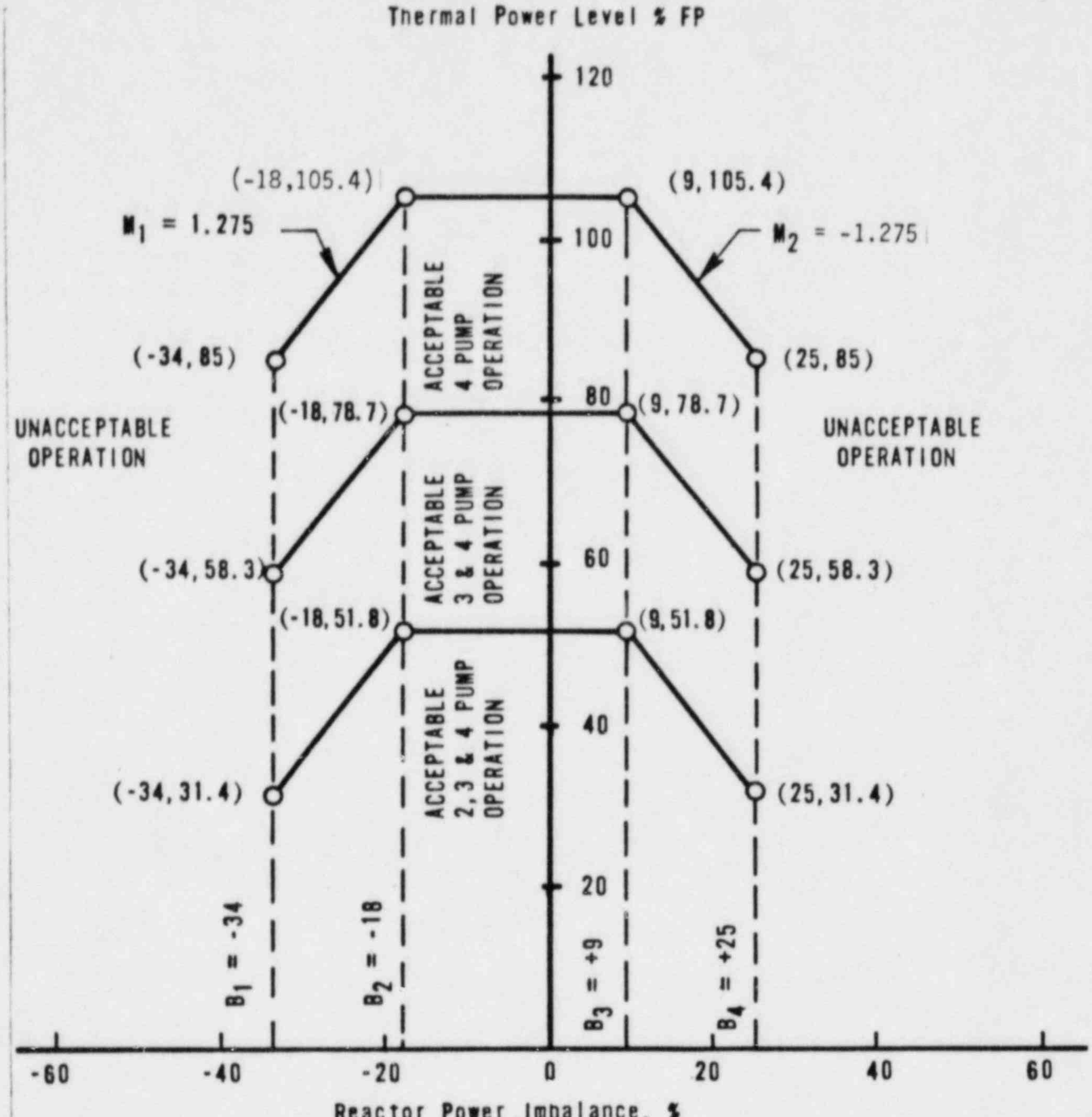
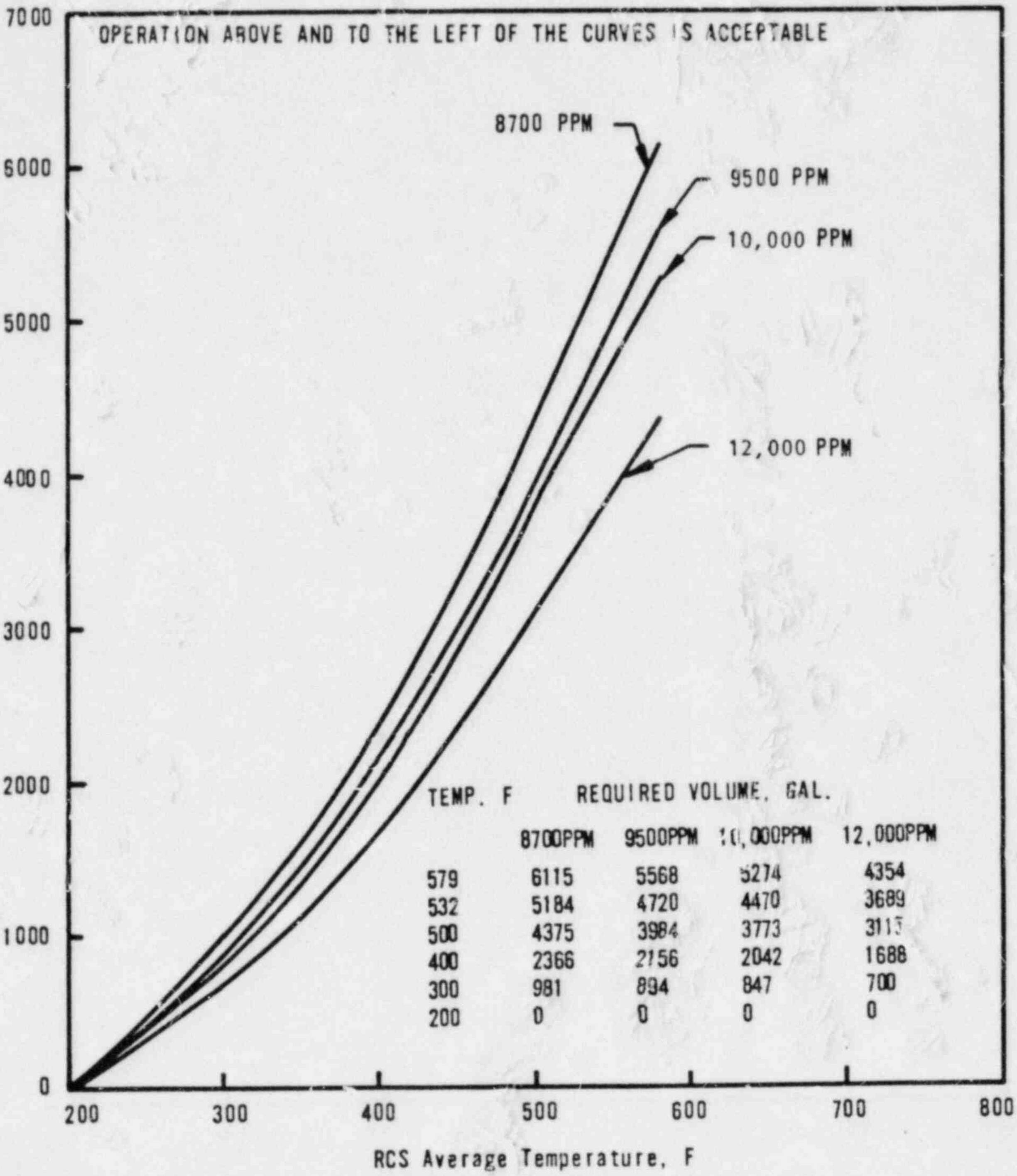


Figure 8-3. Boric Acid Addition Tank Volume and Concentration Requirements Vs RCS Average Temperature (Tech Spec Figure 3.2-1)



TEMP. F	REQUIRED VOLUME, GAL.			
	8700PPM	9500PPM	10,000PPM	12,000PPM
579	6115	5568	5274	4354
532	5184	4720	4470	3689
500	4375	3984	3773	3113
400	2366	2156	2042	1688
300	981	894	847	700
200	0	0	0	0

Figure 8-4. Rod Position Limits for Four-Pump Operation
From 0 to 60 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-1A)

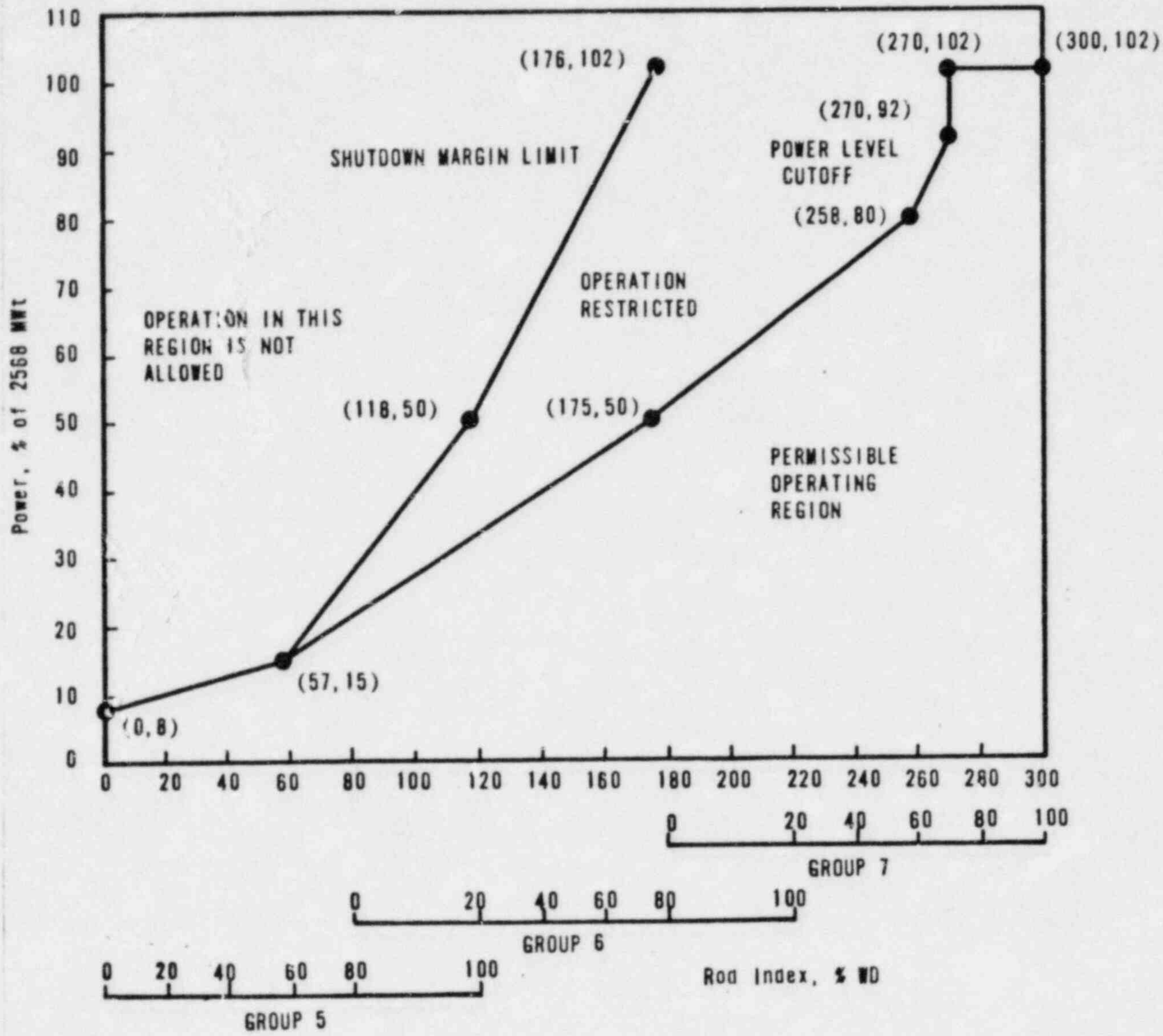


Figure 8-5. Rod Position Limits for Four-Pump Operation From 50 to 200 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-1B)

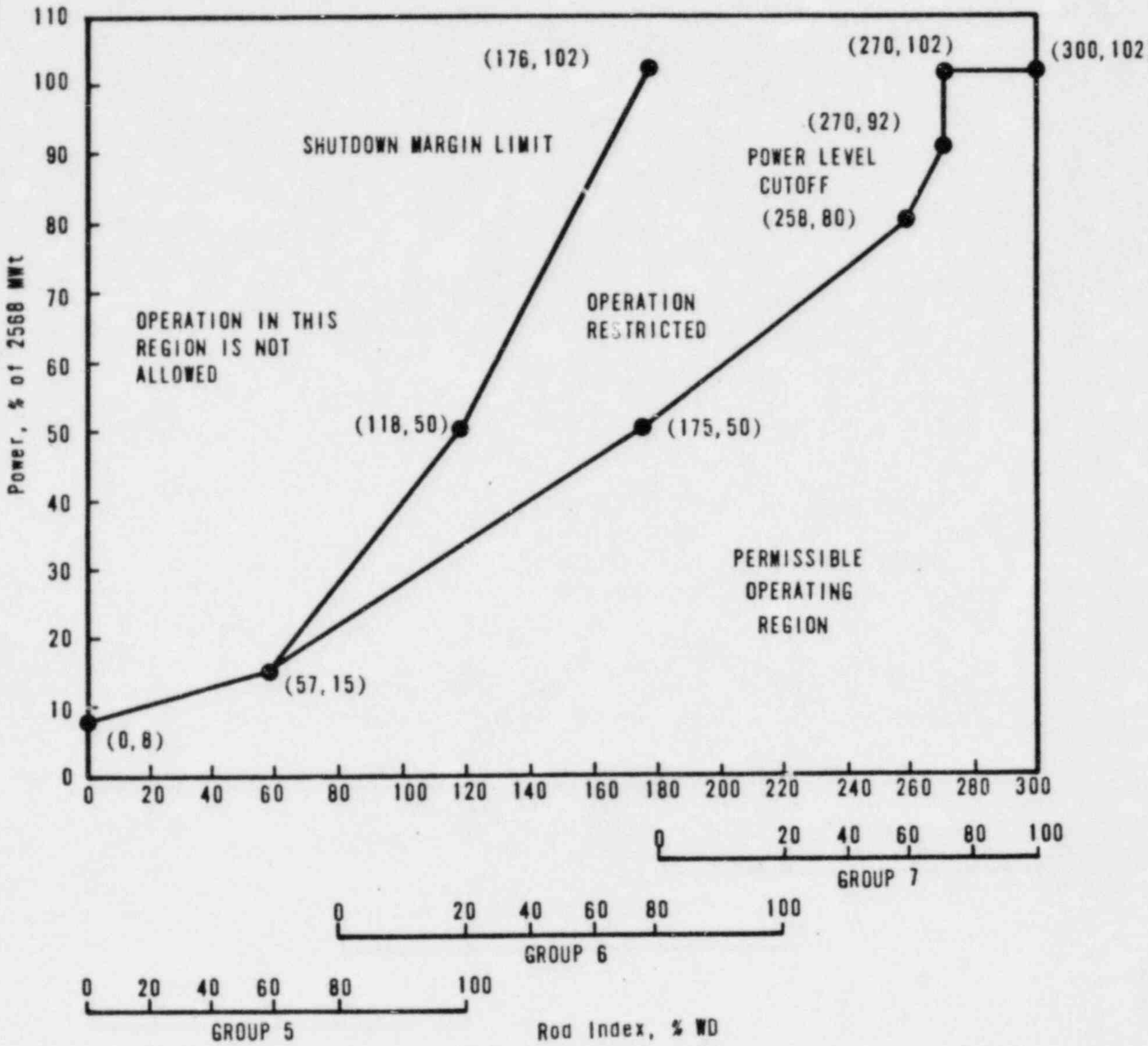


Figure 8-6. Rod Position Limits for Four-Pump Operation From 200 ± 10 to 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-1C)

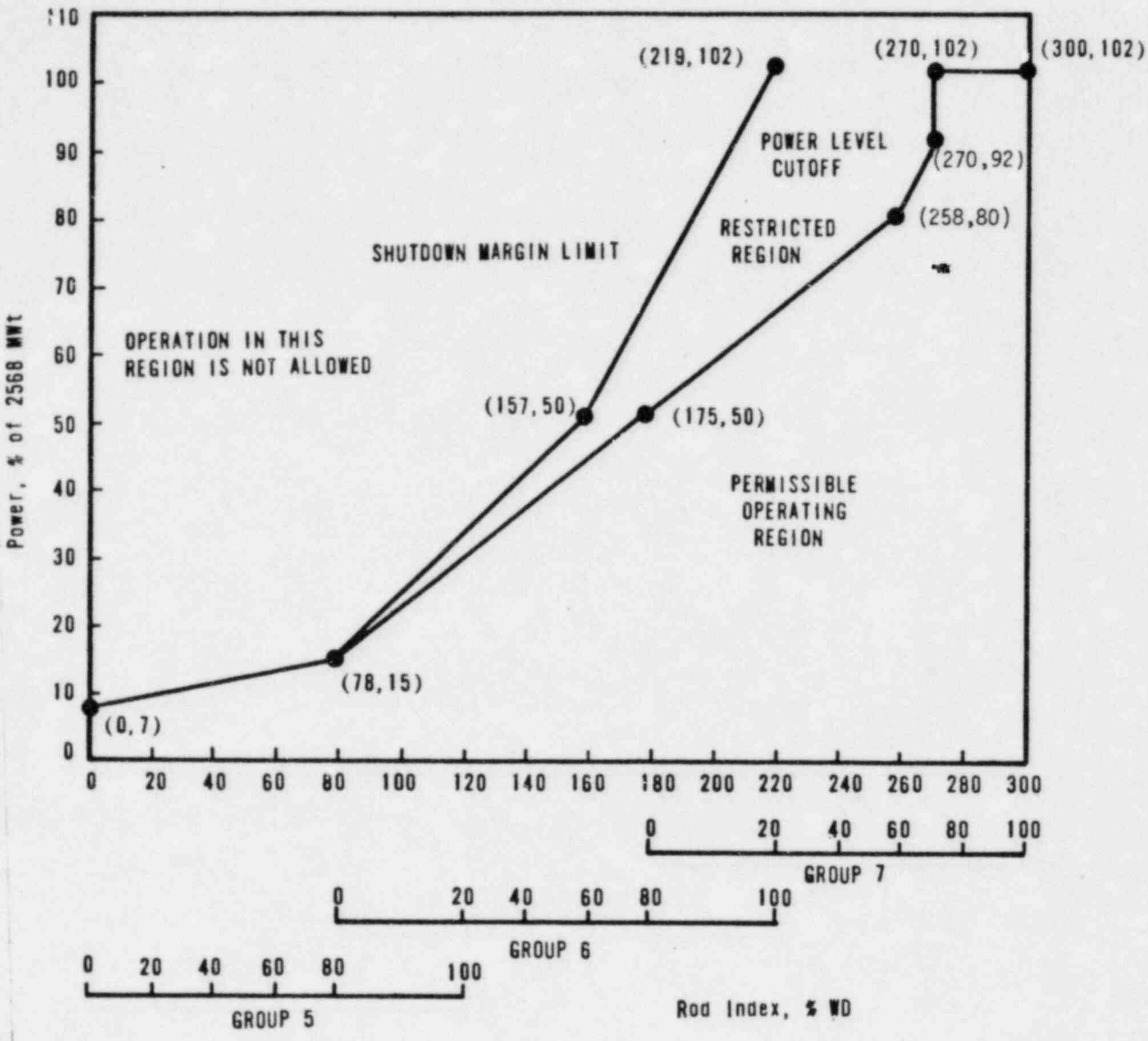


Figure 8-7. Rod Position Limits for Four-Pump Operation
After 350 ± 10 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-1D)

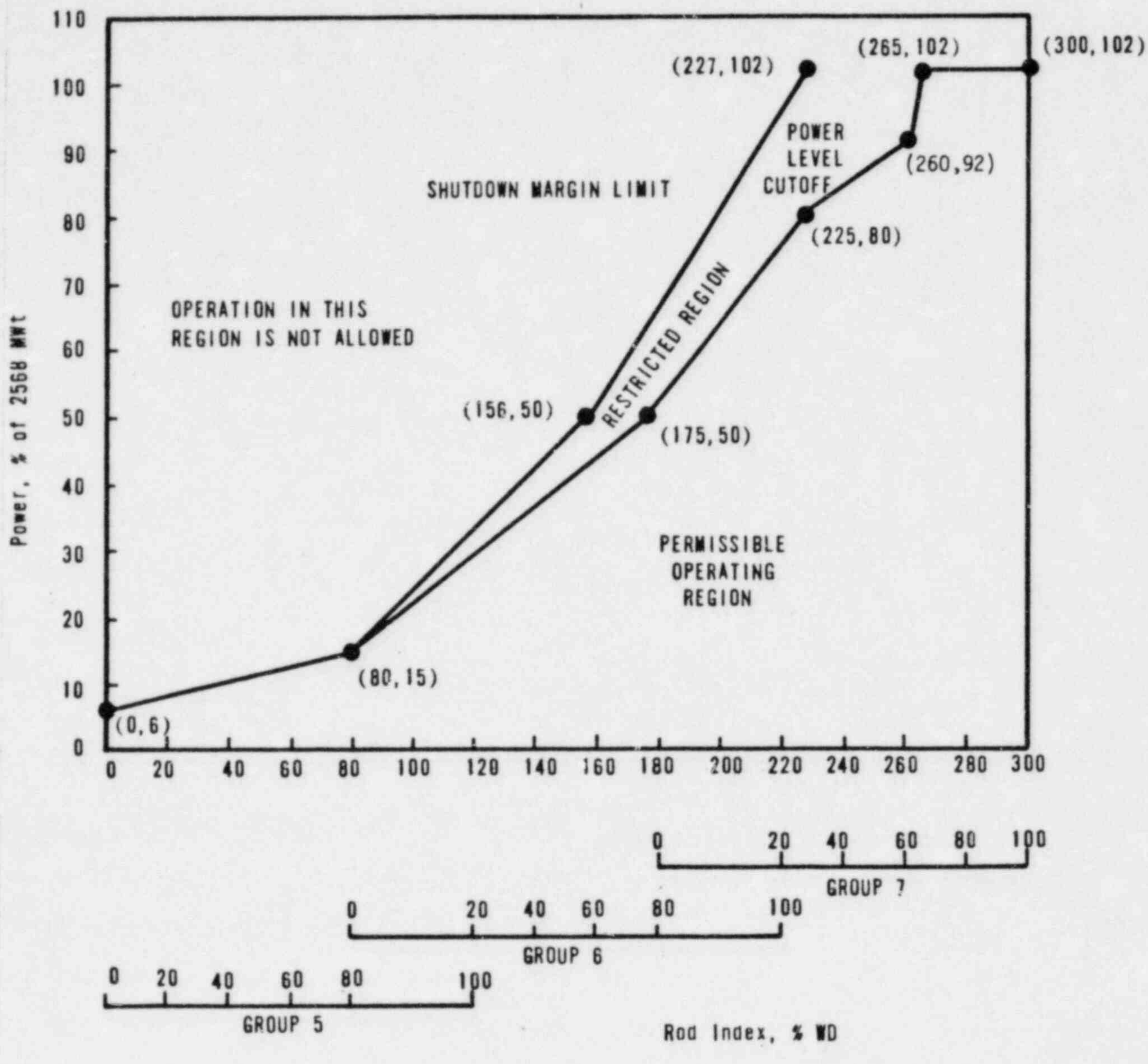


Figure 8-8. Rod Position Limits for Three-Pump Operation From 0 to 60 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-2A)

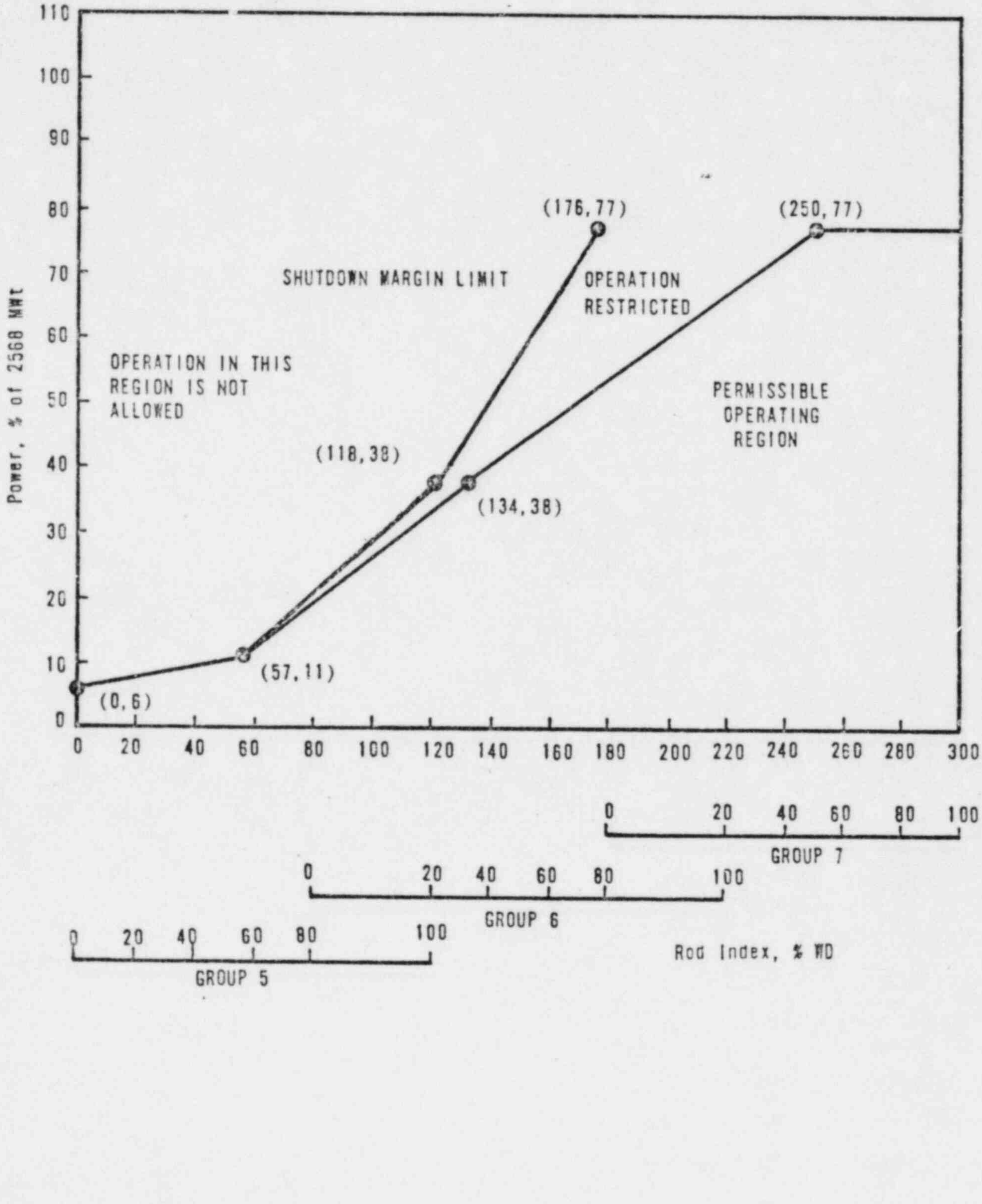


Figure 8-9. Rod Position Limits for Three-Pump Operation
From 50 to 200 ± 10 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-2B)

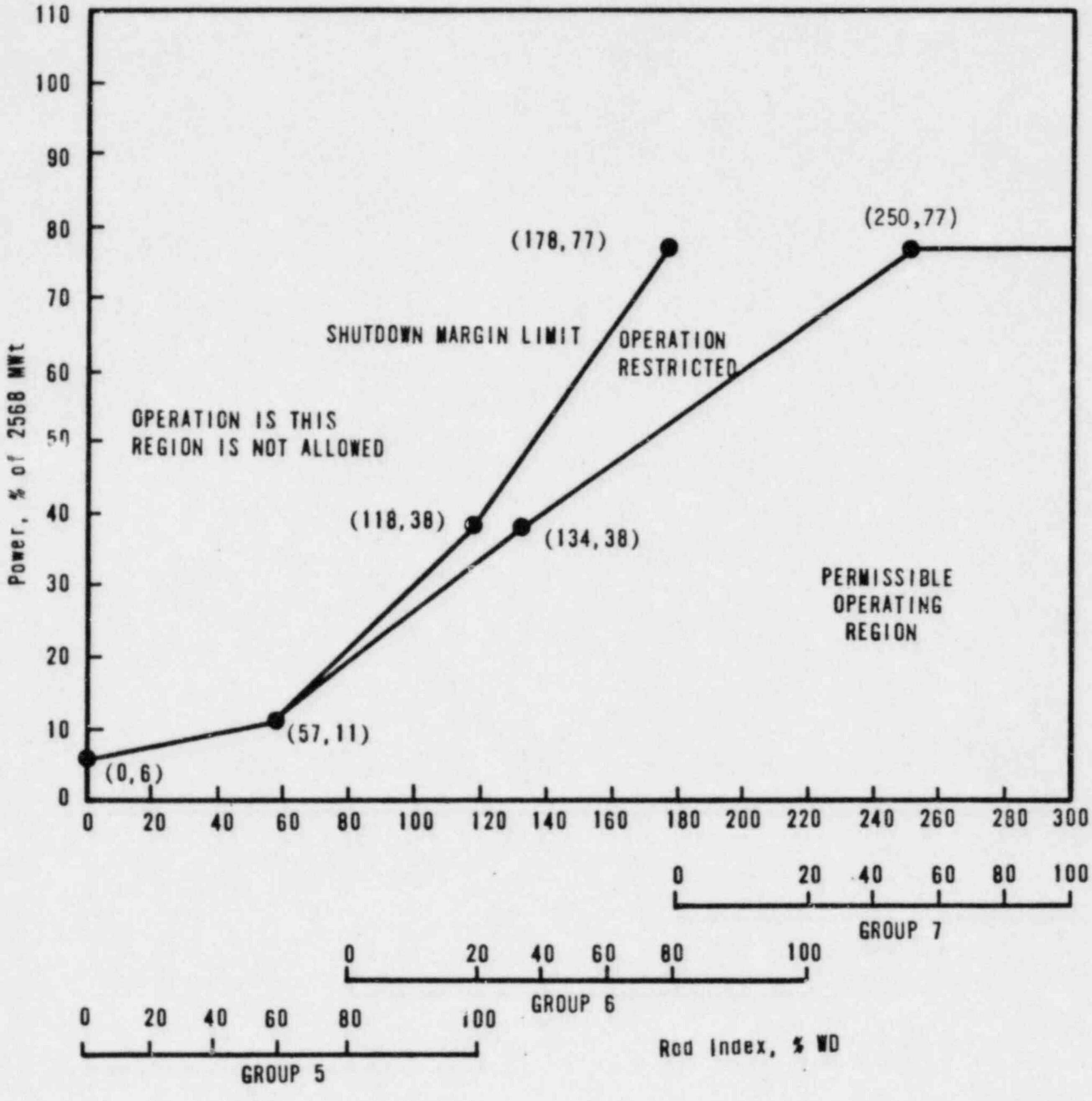


Figure 8-10. Rod Position Limits for Three-Pump Operation From 200 ± 10 to 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-2C)

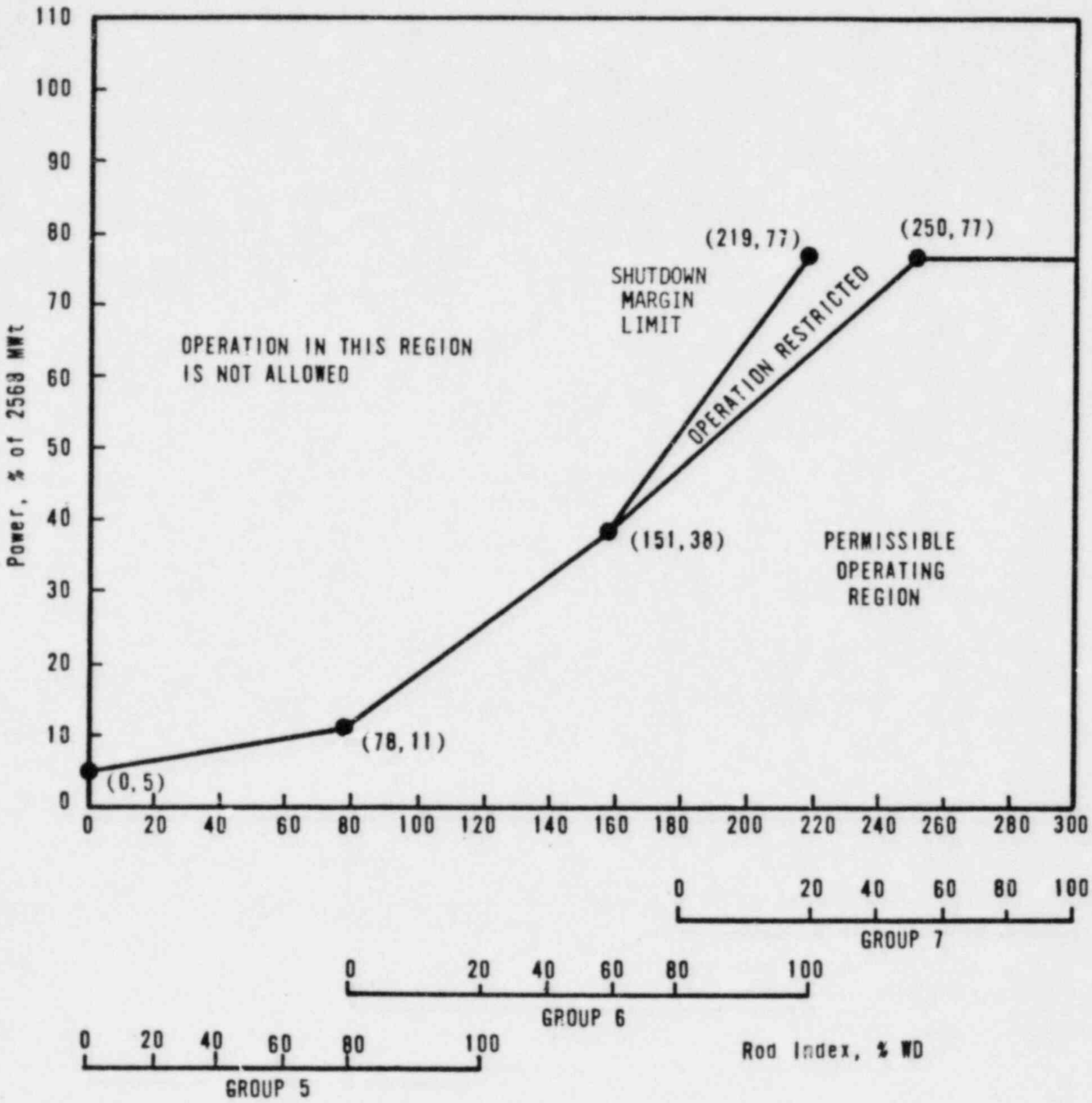


Figure 8-11. Rod Position Limits for Three-Pump Operation
After 350 ± 10 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-2D)

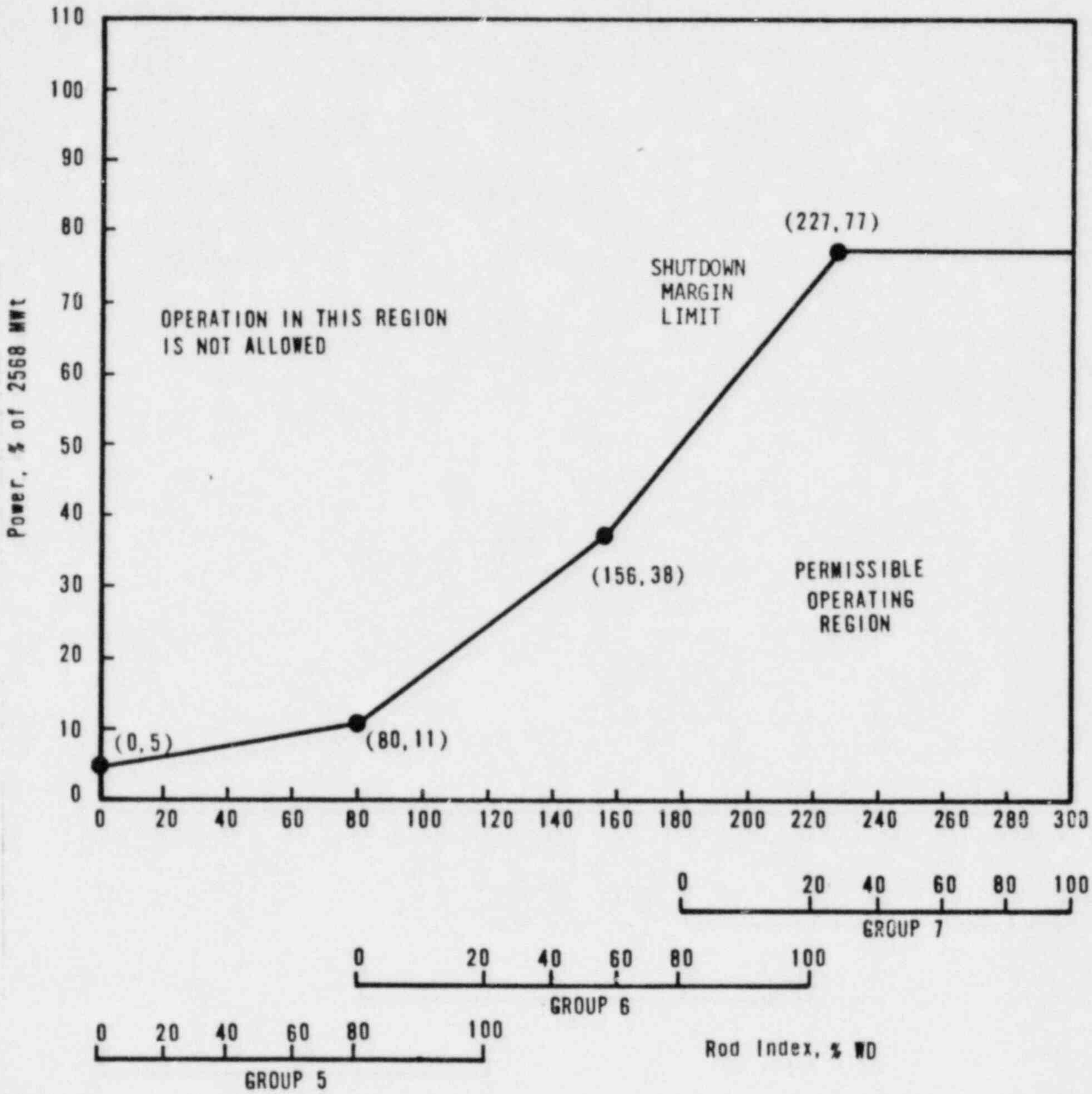


Figure 8- :od Position Limits for Two-Pump Operation From
0 to 60 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-2E)

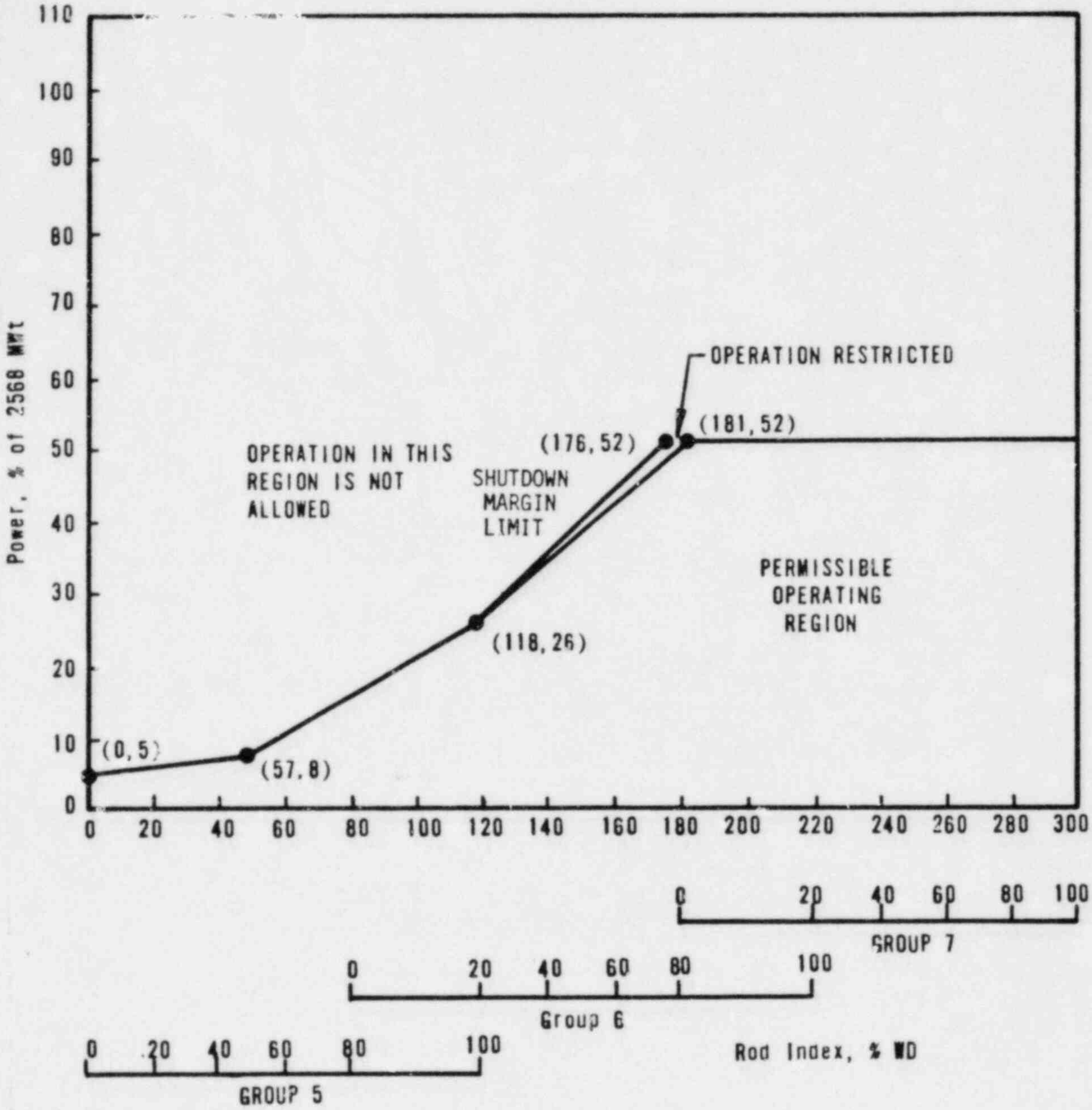


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Figure 8-13. Rod Position Limits for Two-Pump Operation From 50 to 200 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-2F)

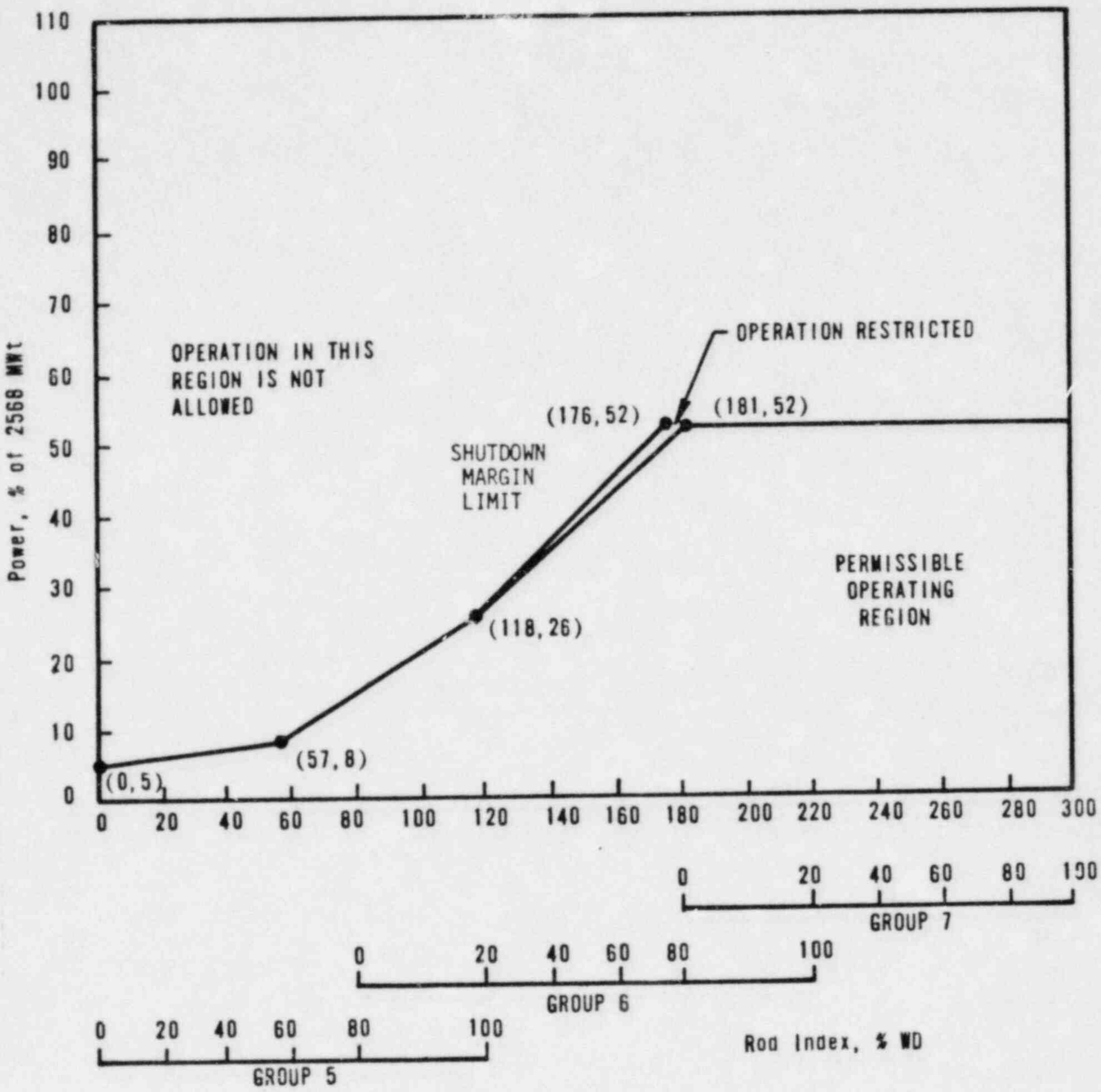


Fig.re 8-14. Rod Position Limits for Two-Pump Operation From 200 ± 10 to 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-2G)

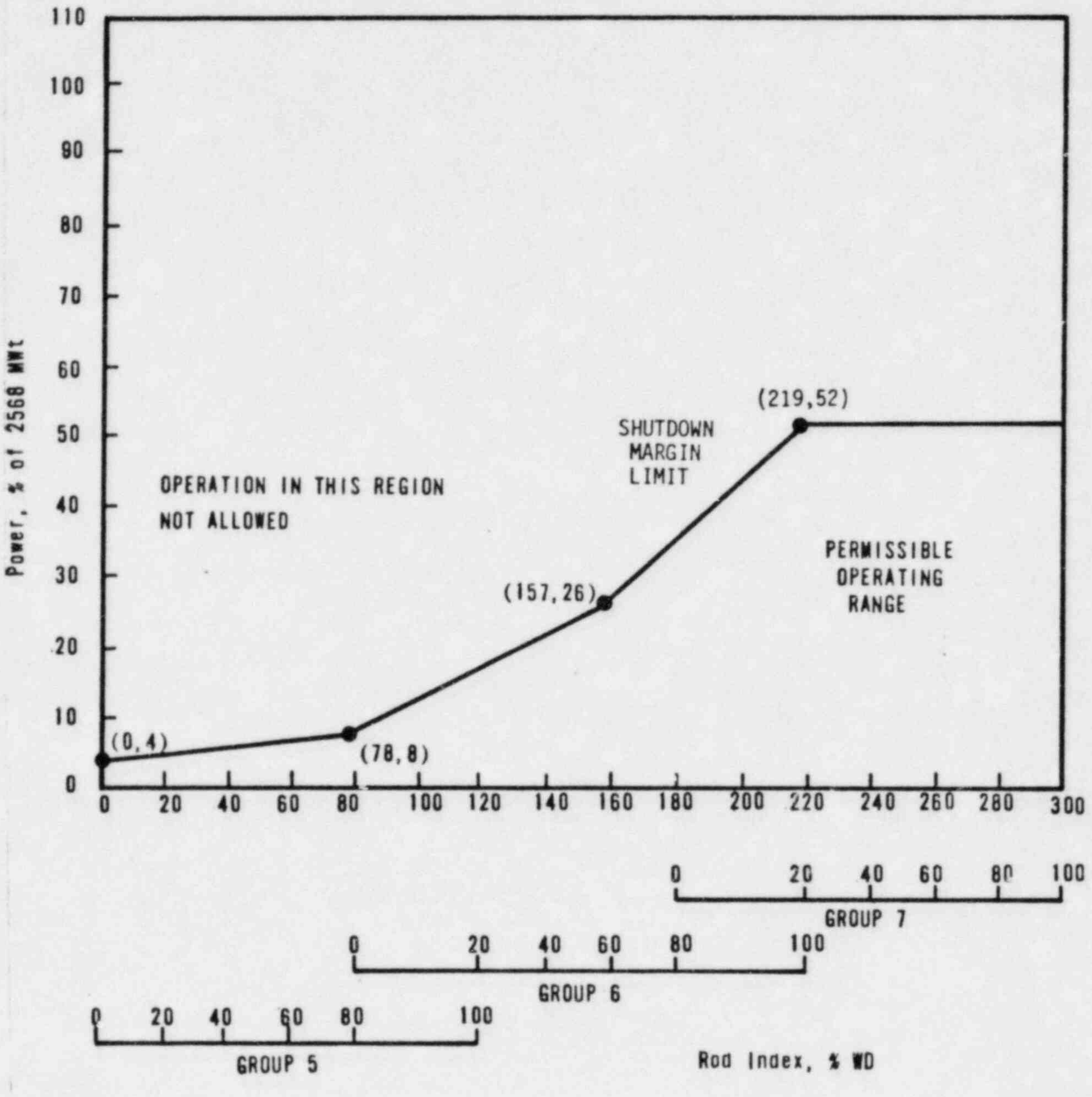


Figure 8-15. Rod Position Limits for Two-Pump Operation
After 350 ± 10 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-2H)

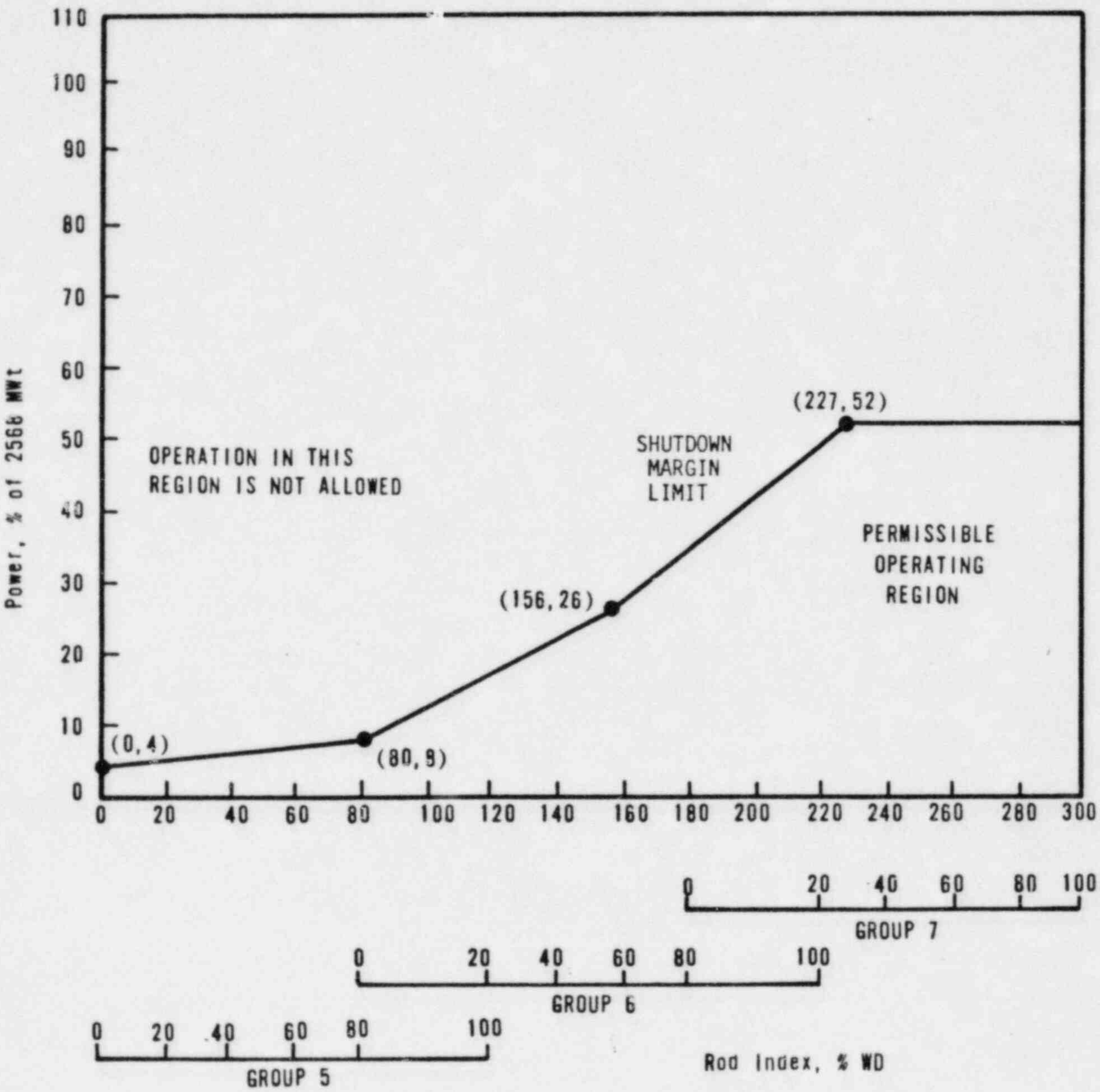


Figure 8-17. Operational Power Imbalance Envelope for Operation From 50 to 200 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-3B)

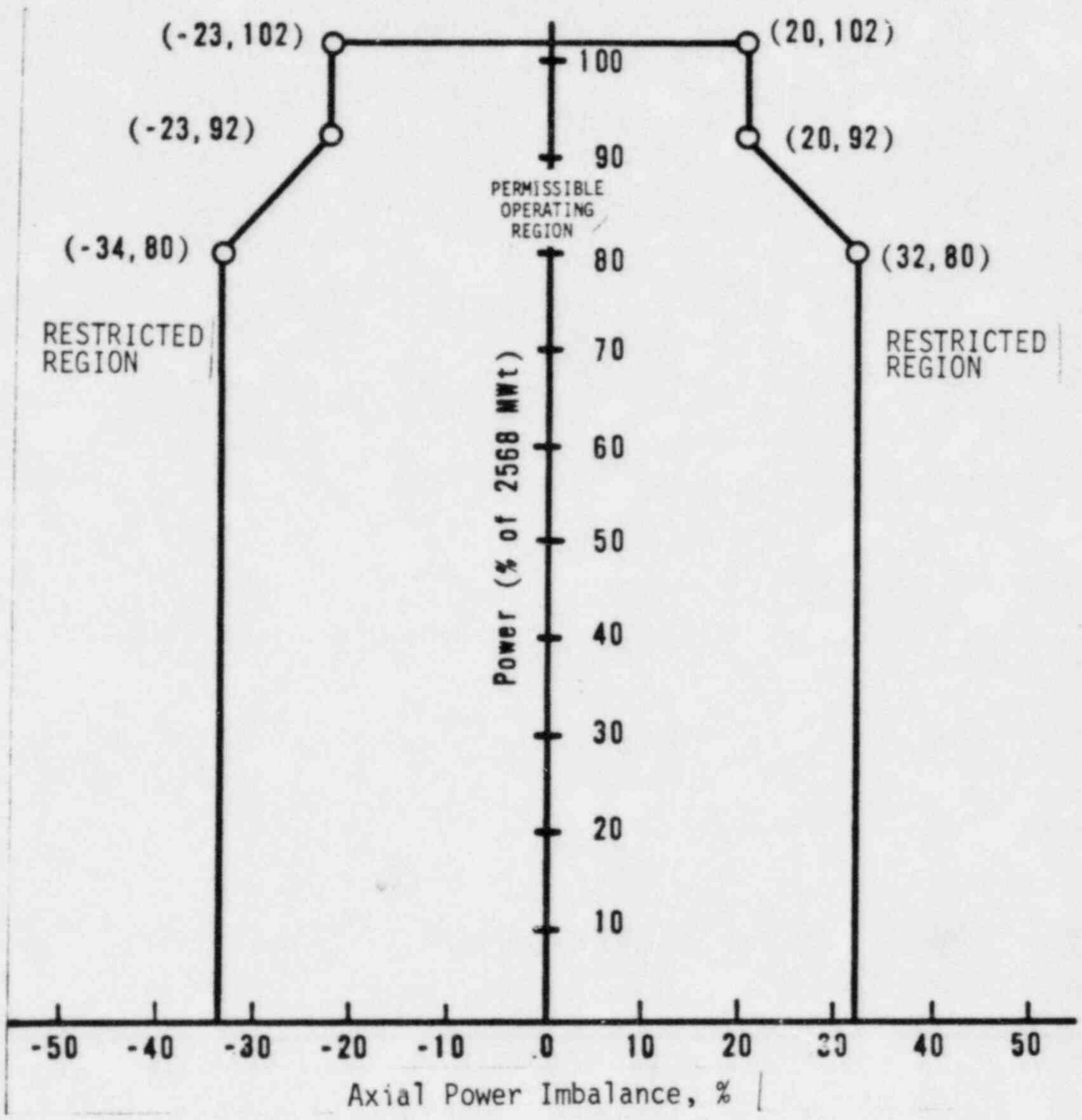


Figure 8-16. Operational Power Imbalance Envelope for Operation From 0 to 60 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-3A)

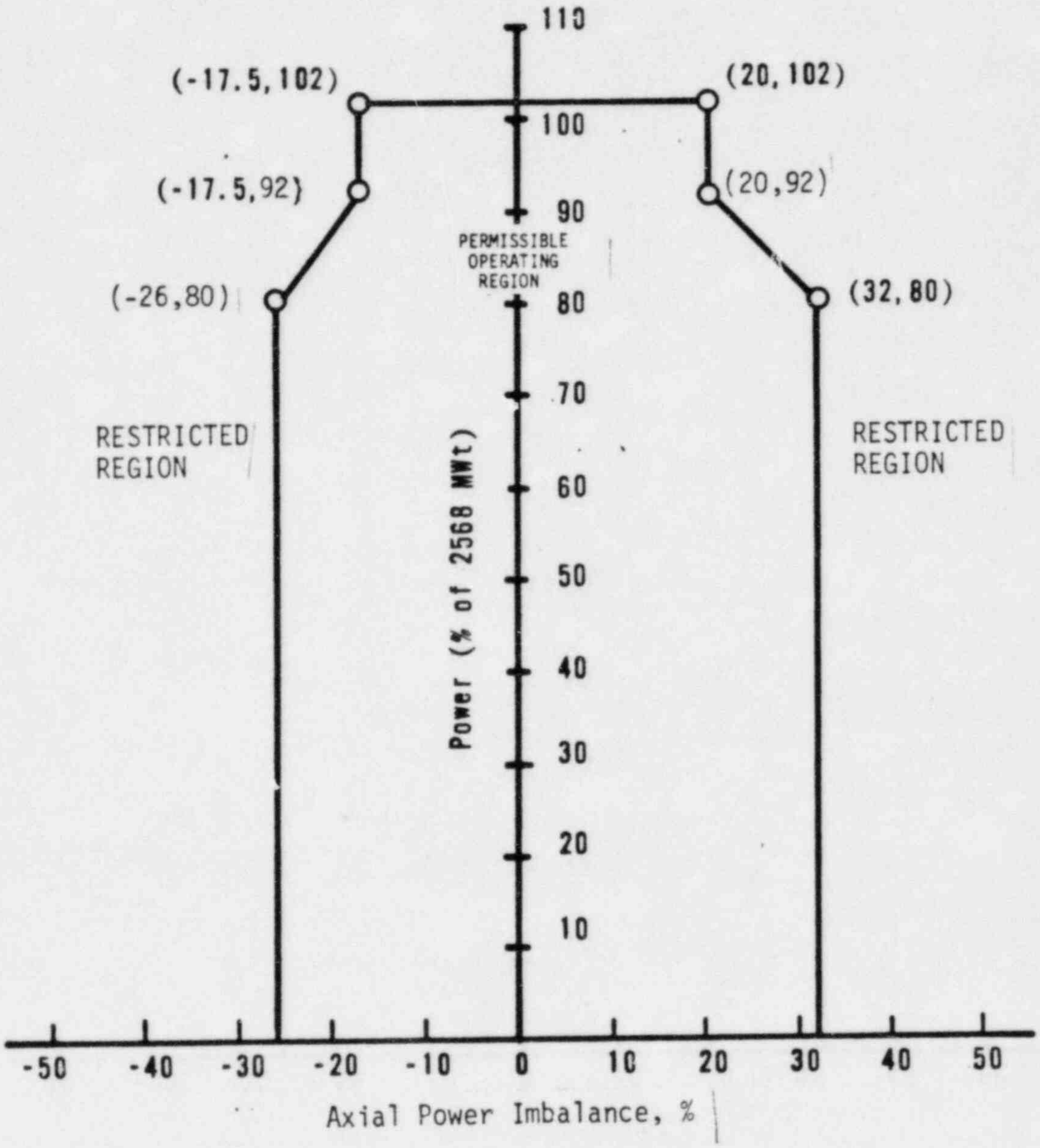


Figure 8-18. Operational Power Imbalance Envelope for Operation From 200 ± 10 to 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-3C)

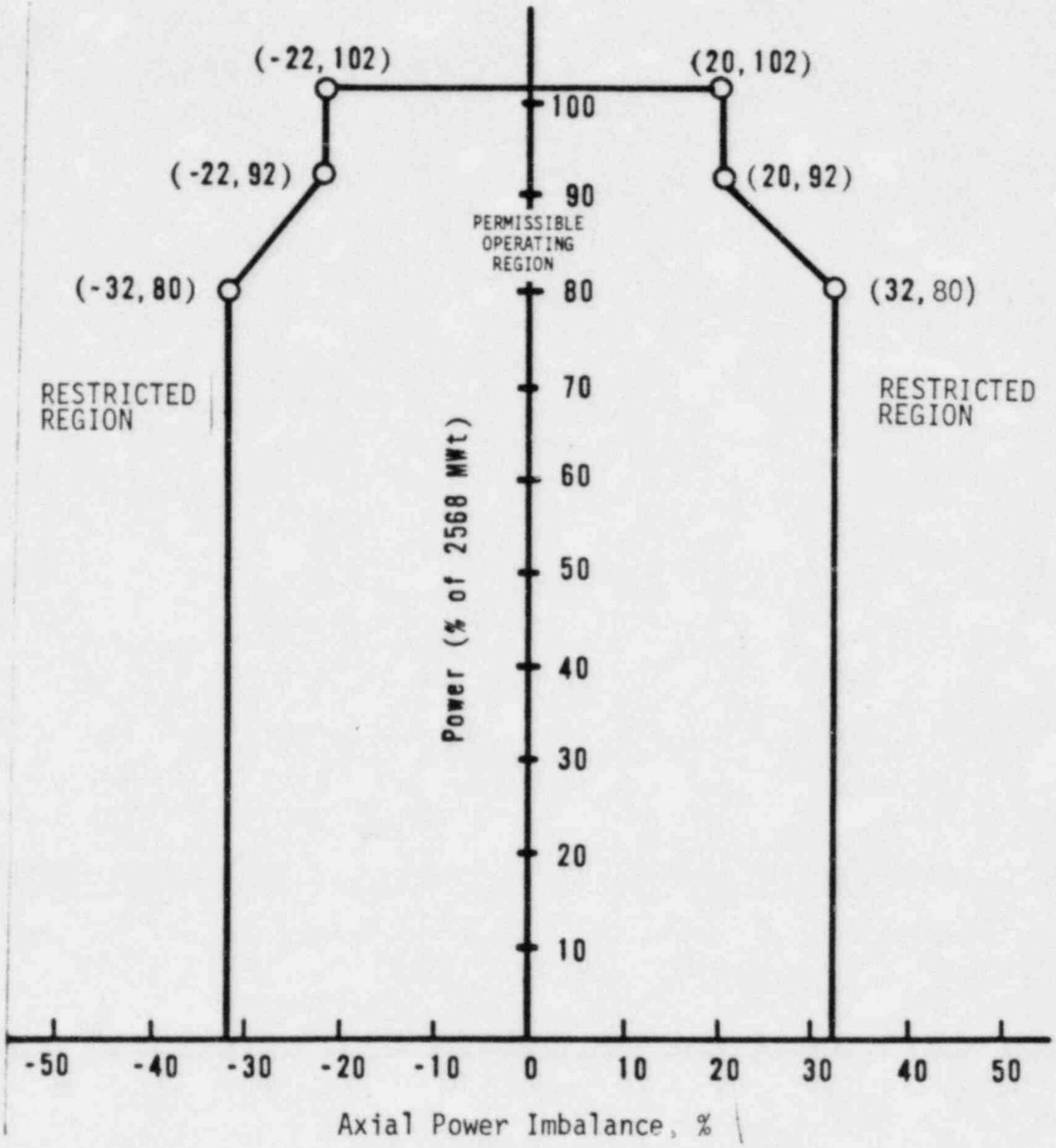


Figure 8-19. Operational Power Imbalance Envelope for Operation
After 350 ± 10 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-3D)

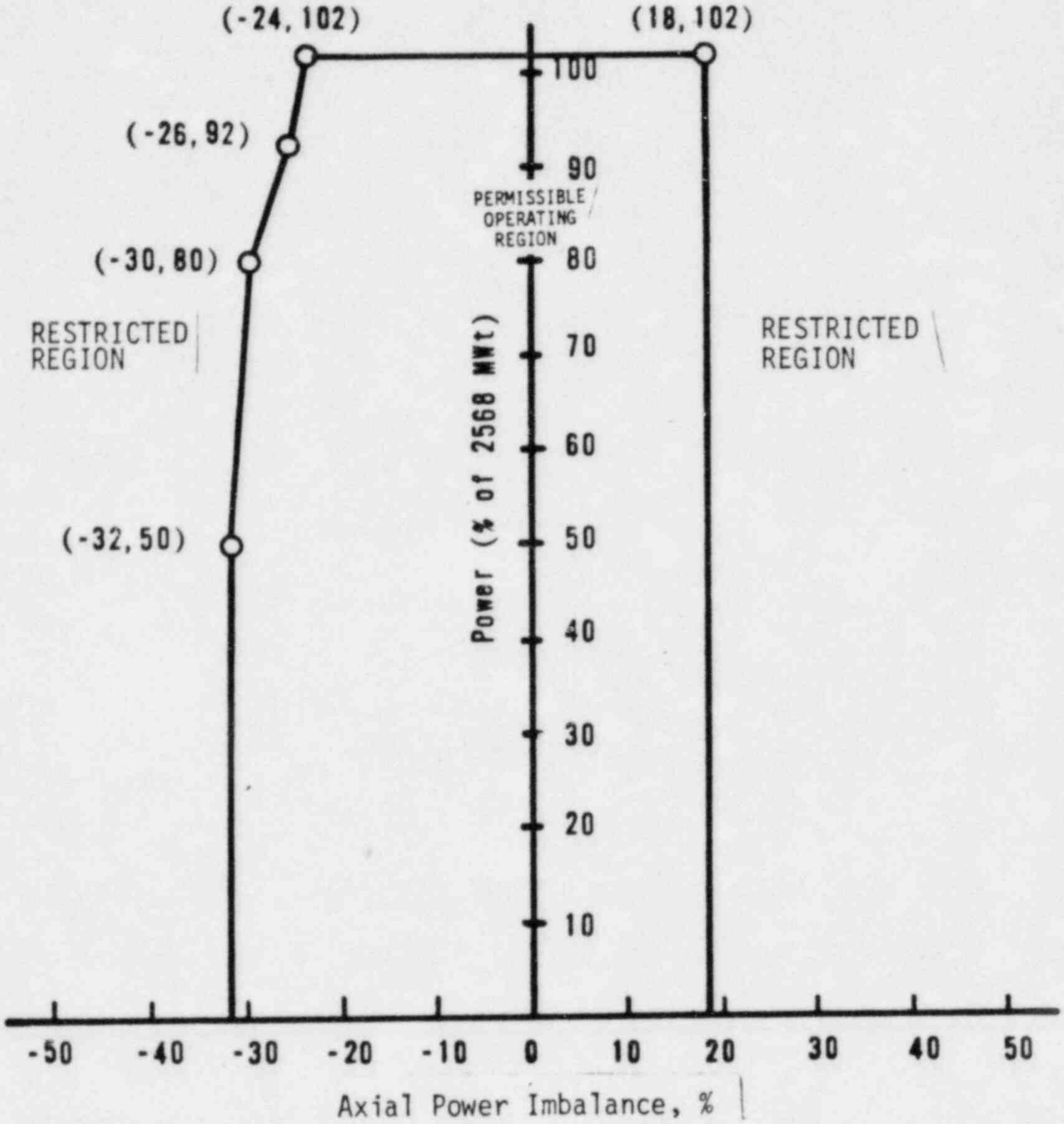


Figure 8-20. APSR Position Limits for Operation From
0 to 60 EFPD — ANO-1, Cycle 6
(Tech Spec Figure 3.5.2-4A)

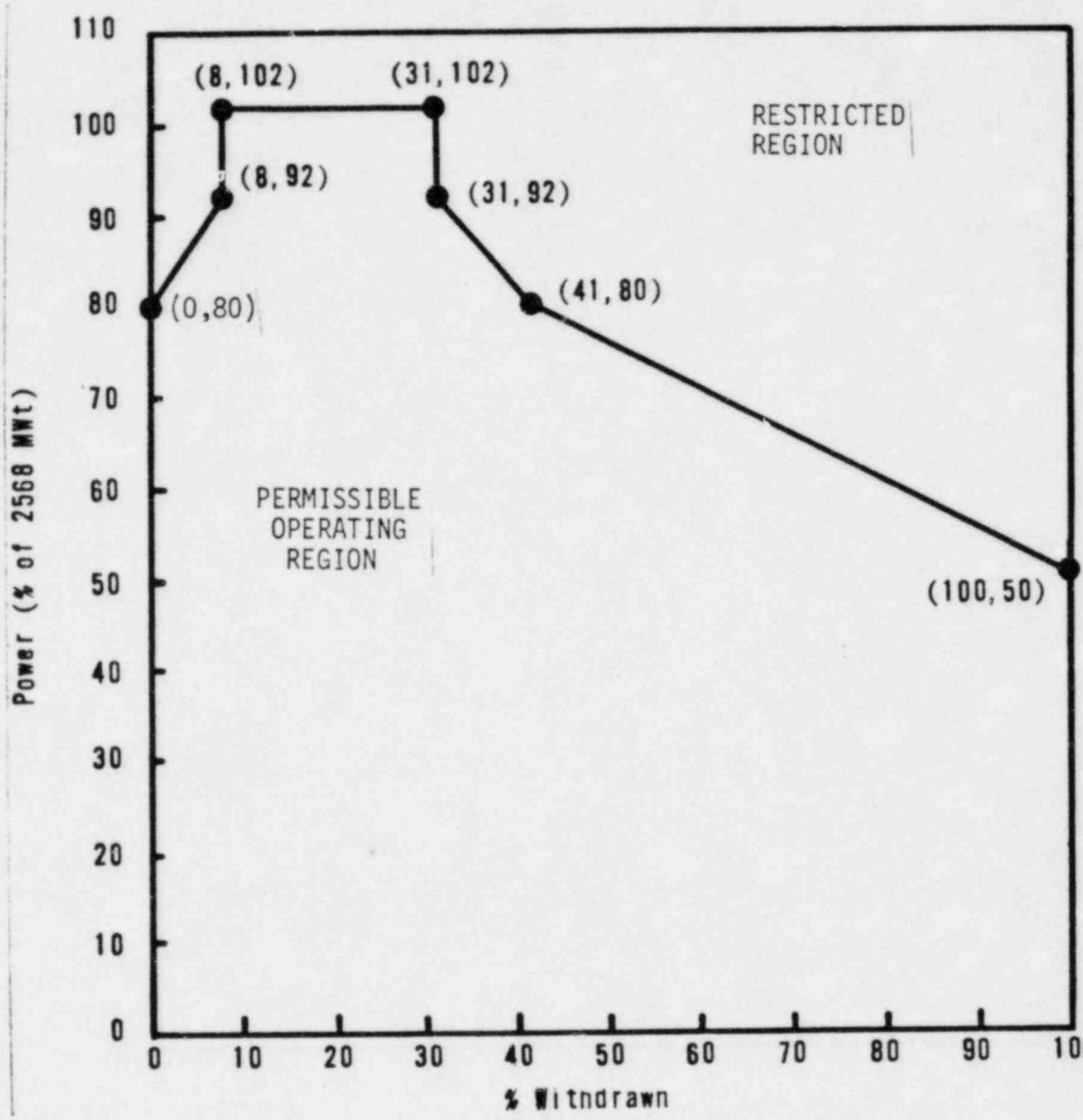


Figure 8-21. APSR Position Limits for Operation From 50 to 200 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-4B)

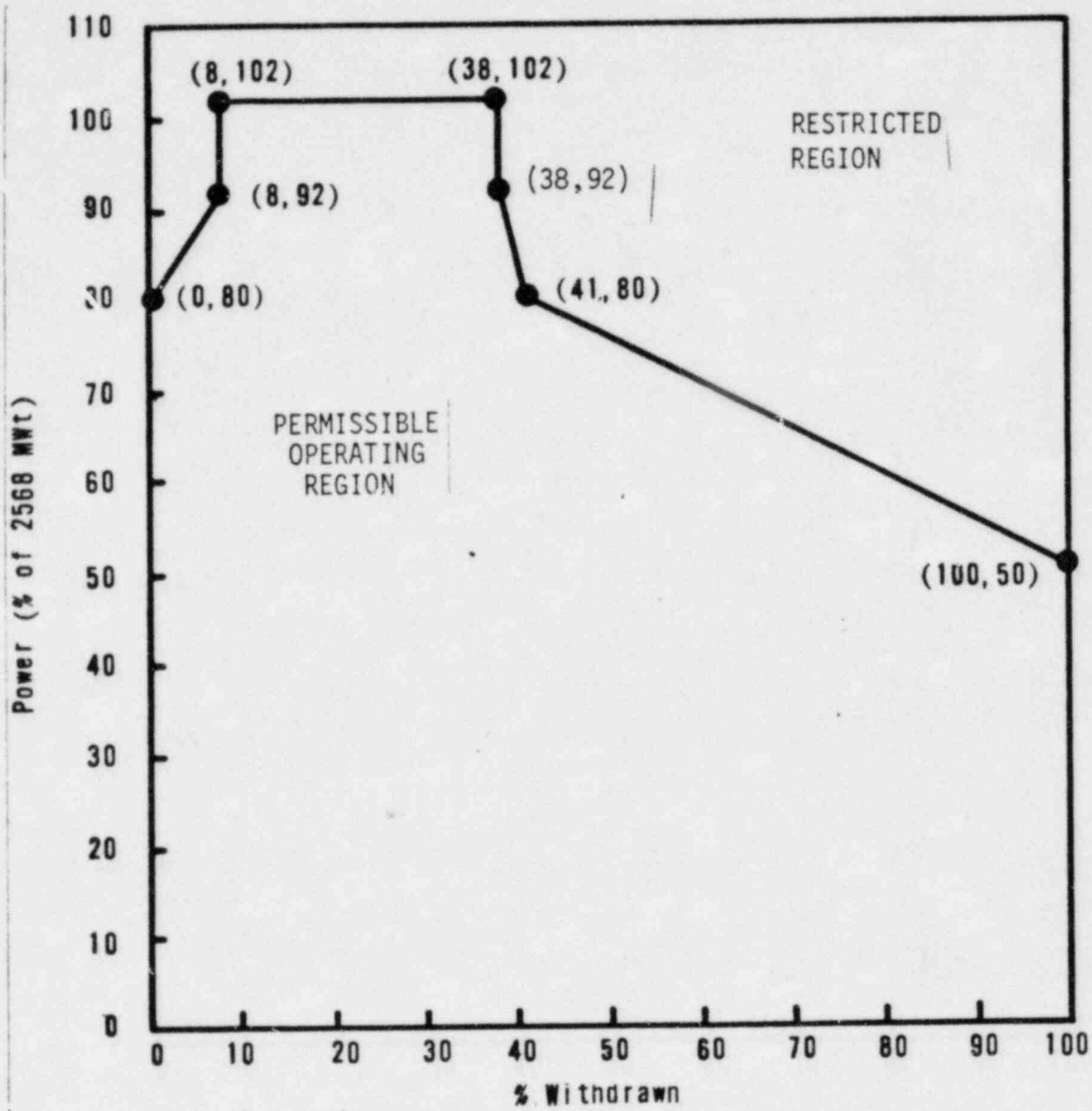


Figure 8-22. APSR Position Limits for Operation From 200 ± 10 to 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-4C)

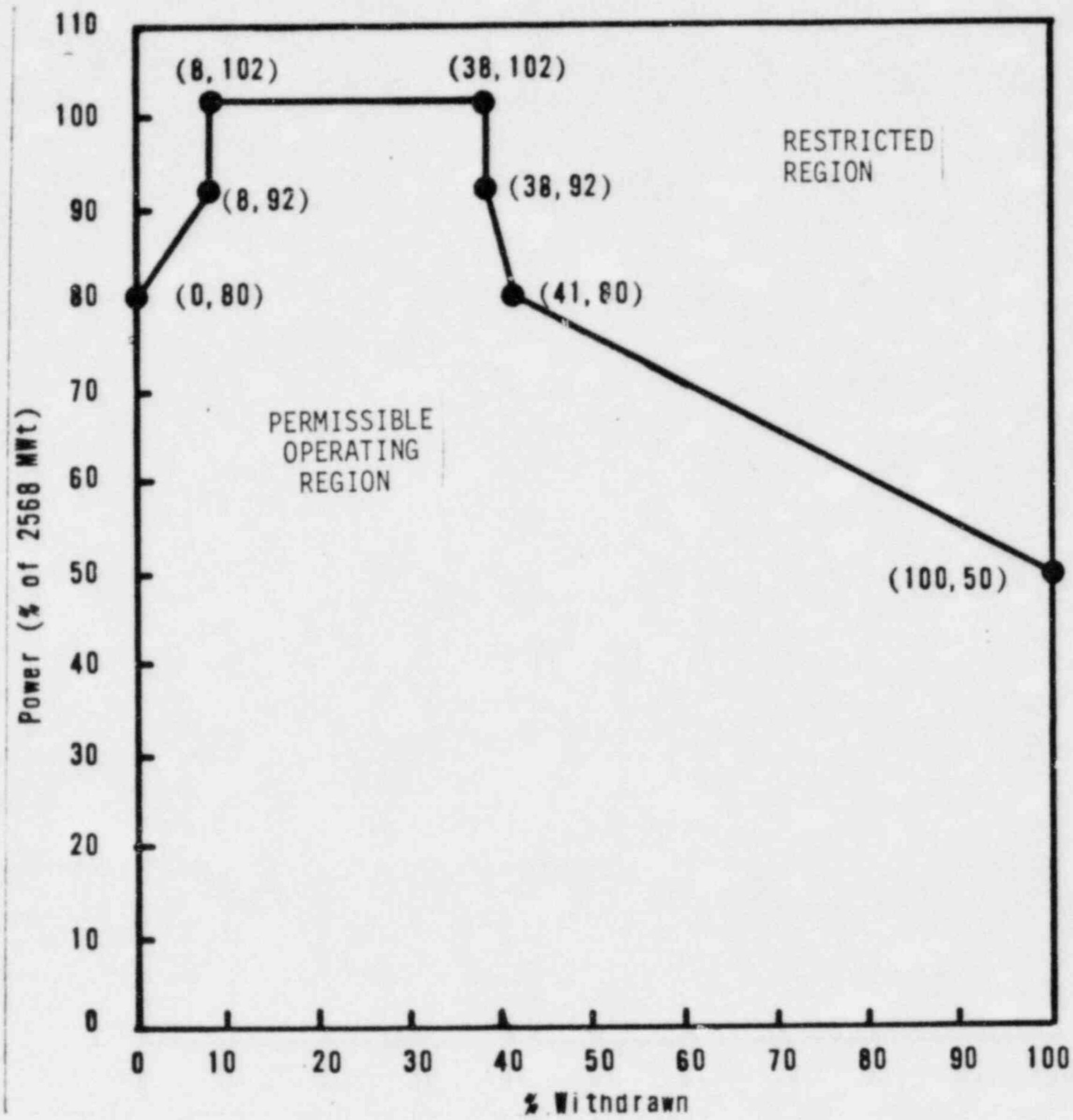
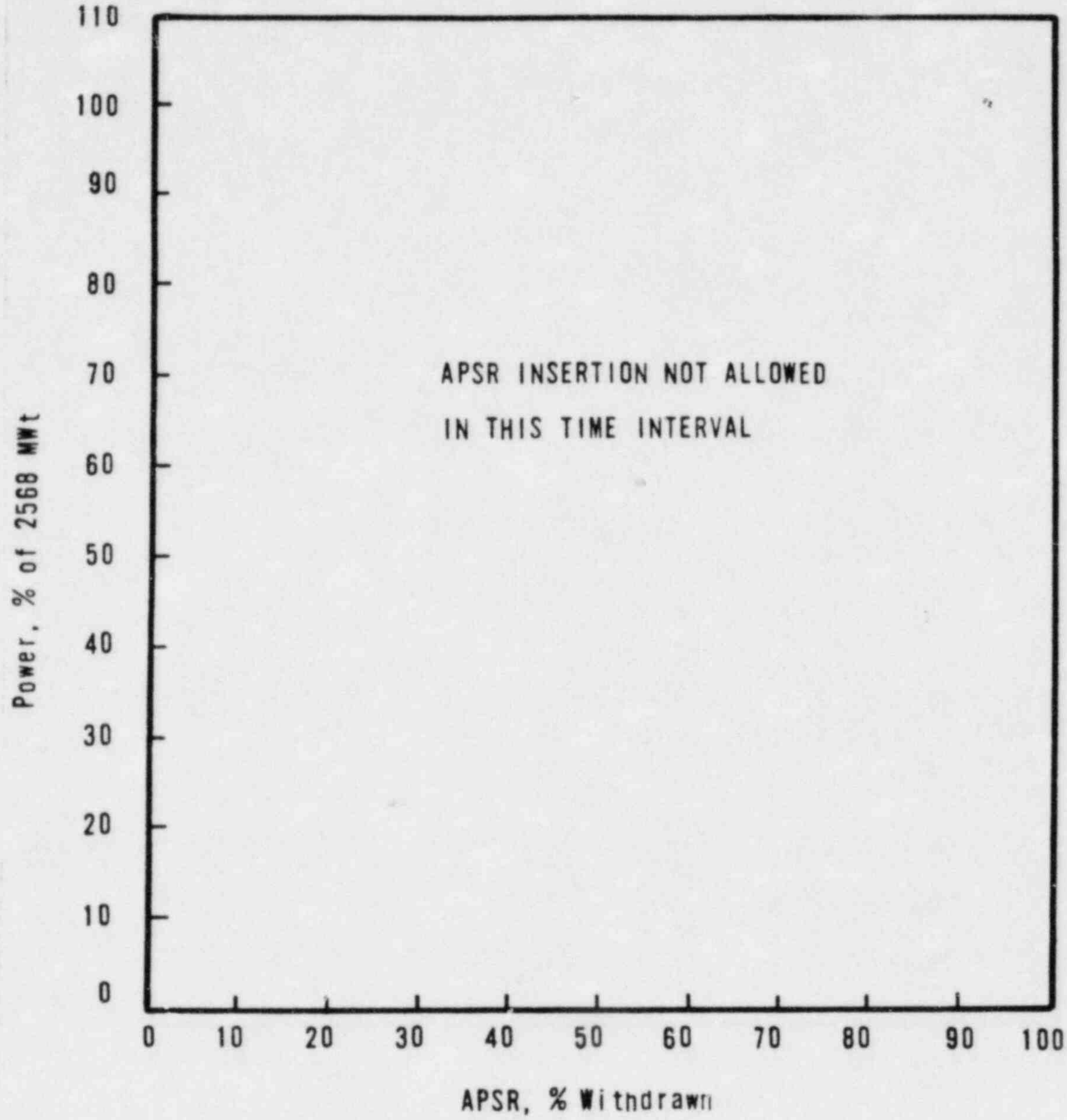


Figure 8-23. APSR Position Limits for Operation After 350 ± 10 EFPD — ANO-1, Cycle 6 (Tech Spec Figure 3.5.2-4D)



Using a local quality limit of 22% at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR, as calculated by the BAW-2 correlation, continually increases from point of minimum DNBR so that the exit DNBR is always higher and is a function of the pressure.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup-dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. Pressure-temperature operating limits are currently based on an original method[3] of calculating rod bowing penalties that are more conservative than those that would be obtained with new approved procedures[4]. For the current cycle of operation, this subrogation results in a DNBR margin of 10%, which is partially used to offset the reduction in DNBR due to fuel rod bowing. The flux-flow limit is calculated with the DNBR margin necessary to offset the rod bow penalty calculated with the new procedures.

The maximum thermal power for three-pump operation is 88.74% due to a power level trip produced by the flux-flow ratio (74.7% flow × 1.054 = 78.73% power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22% for that particular reactor coolant pump situation. Curves 1 and 2 of Figure 2.1-3 are the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curve.

References

1 Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, Babcock & Wilcox, Lynchburg, Virginia, May 1976.

2 FSAR, Section 3.2.3.1.1.c.

[3] D. F. Ross and D. G. Eisenhut (NRC) to D. B. Vassallo and K. R. Goller (NRC), Memorandum, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," December 8, 1976.

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

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 1.46 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop 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.

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 to achieve equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentraiton 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 increasing and then decreasing temperature by 5°F. During the change in temperature, reactivity feedback is compensated by discrete changes in rod motion. The change in reactivity is then calculated by the summation of reactivity (obtained from a reactivity calculator 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 $+0.9 \times 10^{-4}$ $(\Delta k/k)/°F$.

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. This technique 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 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 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 CRA groups 7, 6, and 5 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 swapped in versus the controlling rod group, and the worth is determined by the change in the control rod group position. 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 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 LHR must be less than the LOCA limit.

2. The minimum DNBR must be greater than 1.30.

3. The value obtained from 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 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:

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

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

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

Items 1, 2, 5, 6, and 7 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:

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

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

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

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

Imbalances, set up in the core by control rod positioning, are read simultaneously on the incore detectors and excore power range detectors. The excore detector offset versus incore detector offset slope must be at least 1.15. If this 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 if Acceptance Criteria Not Met

If the acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. The results of all tests will be reviewed by the plant's nuclear engineering group. If the acceptance criteria of the startup physics tests are not met, an evaluation will be performed by the plant's nuclear engineering group with assistance from general

Office personnel, Middle South Services, and the fuel vendor, as needed. The results of this evaluation will be presented to the On-site Plant Safety Committee. Resolution will be required prior to power escalation. If a safety question is involved, the Off-site Safety Review Committee would review the situation, and the NRC would be notified if an unreviewed safety question exists.

REFERENCES

[1] Arkansas Nuclear One, Unit 1 — Final Safety Analysis Report, Docket 50-313, Arkansas Power & Light.

[2] T. A. Coleman and J. T. Willse, Extended Burnup Lead Test Assembly Irradiation Program, BAW-1626, Babcock & Wilcox, October 1980.

[3] Arkansas Nuclear One, Unit 1 — Cycle 5 Reload Report, BAW-1658, Rev. 2, Babcock & Wilcox, May 1982.

[4] J. H. Taylor to S. A. Varga, Letter, "BPRA Retainer Reinsertion," January 14, 1980.

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

[6] Y. H. Hsii, et al., TACO-2 — Fuel Pin Performance Analysis, BAW-10141P, Babcock & Wilcox, January 1979.

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

[8] Arkansas Nuclear One, Unit 1 — Fuel Densification Report, BAW-1391, Babcock & Wilcox, June 1973.

[9] D. F. Ross and D. G. Eisenhut (NRC), Memorandum to D. S. Vassallo and K. R. Goller (NRC) on "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," December 8, 1976.

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

[11] ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 1, Babcock & Wilcox, September 1975.

[12] J. H. Taylor (B&W Licensing) to R. L. Baer (Reactor Safety Branch, USNRC), Letter, July 8, 1977.

[13] J. H. Taylor (B&W Licensing) to L. S. Rubenstein (USNRC), Letter, September 5, 1980.

Note: All Babcock & Wilcox reports mentioned are from NPGD, Lynchburg, Virginia.