BAW-1918

August 1986

ARKANSAS NUCLEAR ONE, UNIT 1

- Cycle 8 Reload Report -

BABCOCK & WILCOX
Nuclear Power Division
P. O. Box 10935
Lynchburg, Virginia 24506-0935

Babcock & Wilcox
a McDermott company

CONTENTS

Contents (Cont'd)

List of Tables

List of Figures

List of Figures (Cont'd)

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the eighth 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 8 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 7 and 8 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 8 operation. In those cases where cycle 8 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 8 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 8 at a rated power level of 2568 MWt.

The cycle 8 core for ANO-1 will contain one thrice-burned lead test assembly (LTA). This assembly is 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 7. This cycle 8 design is based on a design cycle 7 length of 425 effective full power days (EFPD).

No anomalies occurred during cycle 7 that would adversely affect fuel performance during cycle 8.

3. GENERAL DESCRIPTION

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

The cycle 8 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 is comprised of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg of uranium, with the exception of one batch 7D LTA, which has 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 LTA; the corresponding parameters for the LTA are included in reference 2.

Figure 3-1 is the fuel shuffle diagram for ANO-1, cycle 8. The initial enrichments of batches 7D, 8B, 9 and 10 are 2.95, 3.21, 3.30, and 3.35 wt% U-235, respectively. All but one of the batch 7B assemblies and 28 of the twice-burned batch 8 assemblies will be discharged at the end of cycle 7. The center location will contain the remaining batch 7 assembly (designated 7D), and the remaining 44 batch 8 assemblies (designated 8B) will be shuffled to new locations, with 12 on the core periphery. Sixty of the 68 once-burned batch 9 assemblies will be shuffled to new locations, primarily on or near the core periphery. The remaining 8 will surround the center assembly. The 64 fresh batch 10 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.

Reactivity is controlled by 60 full-length Ag-In-Cd control rods, 64 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle 8 locations of the 68 control rods and the group designations are indicated in Figure 3-3. The core locations and group designations of the total pattern (68 control rods) for cycle 8 are the same as those of the reference cycle[3] (69 control rods) except for the center location. The cycle 8 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Core Loading Diagram for ANO-1 Cycle 8

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						9 L05	9 N03	8B G08	9 N13	9 L11					
B				8B P10	9 M04	9 L03	10 F	9 E08	10 F	9 L13	9 M12	8B P06			
C			9 H15	10 F	9 K06	10 F	8B A06	10 F	8B A10	10 F	9 K10	10 F	9 R08		
D		8B L14	10 F	9 K04	10 F	8B B05	10 F	9 D09	10 F	8B B11	10 F	9 D07	10 F	8B L02	
E		9 D11	9 F09	10 F	9 H03	10 F	8B C05	10 F	8B C11	10 F	9 C08	10 F	9 F07	9 D05	
F	9 E10	9 C10	10 F	8B E02	10 F	8B P09	10 F	8B R07	10 F	8B K01	10 F	8B E14	10 F	9 C06	9 E06
G	9 C12	10 F	8B F01	10 F	8B E03	10 F	9 P09	9 B09	9 K02	10 F	8B E13	10 F	8B F15	10 F	9 C04
W—H	8B H07	9 H05	10 F	9 G04	10 F	8B K15	9 G02	7D K05	9 K14	8B G01	10 F	9 K12	10 F	9 H11	8B H09
K	9 Ø12	10 F	8B L01	10 F	8B M03	10 F	9 G14	9 P07	9 B07	10 F	8B M13	10 F	8B L15	10 F	9 Ø04
L	9 M10	9 Ø10	10 F	8B M02	10 F	8B G15	10 F	8B A09	10 F	8B A07	10 F	8B M14	10 F	9 Ø06	9 M06
M		9 N11	9 L09	10 F	9 Ø08	10 F	8B Ø05	10 F	8B Ø11	10 F	9 H13	10 F	9 L07	9 N05	
N		8B F14	10 F	9 N09	10 F	8B P05	10 F	9 N07	10 F	8B P11	10 F	9 G12	10 F	8B F02	
O			9 A08	10 F	9 G06	10 F	8B R06	10 F	8B R10	10 F	9 G10	10 F	9 H01		
P				8B B10	9 E04	9 F03	10 F	9 M08	10 F	9 F13	9 E12	8B B06			
R						9 F05	9 D03	8B K08	9 D13	9 F11					

— Y

Z

BATCH

Previous Core Location

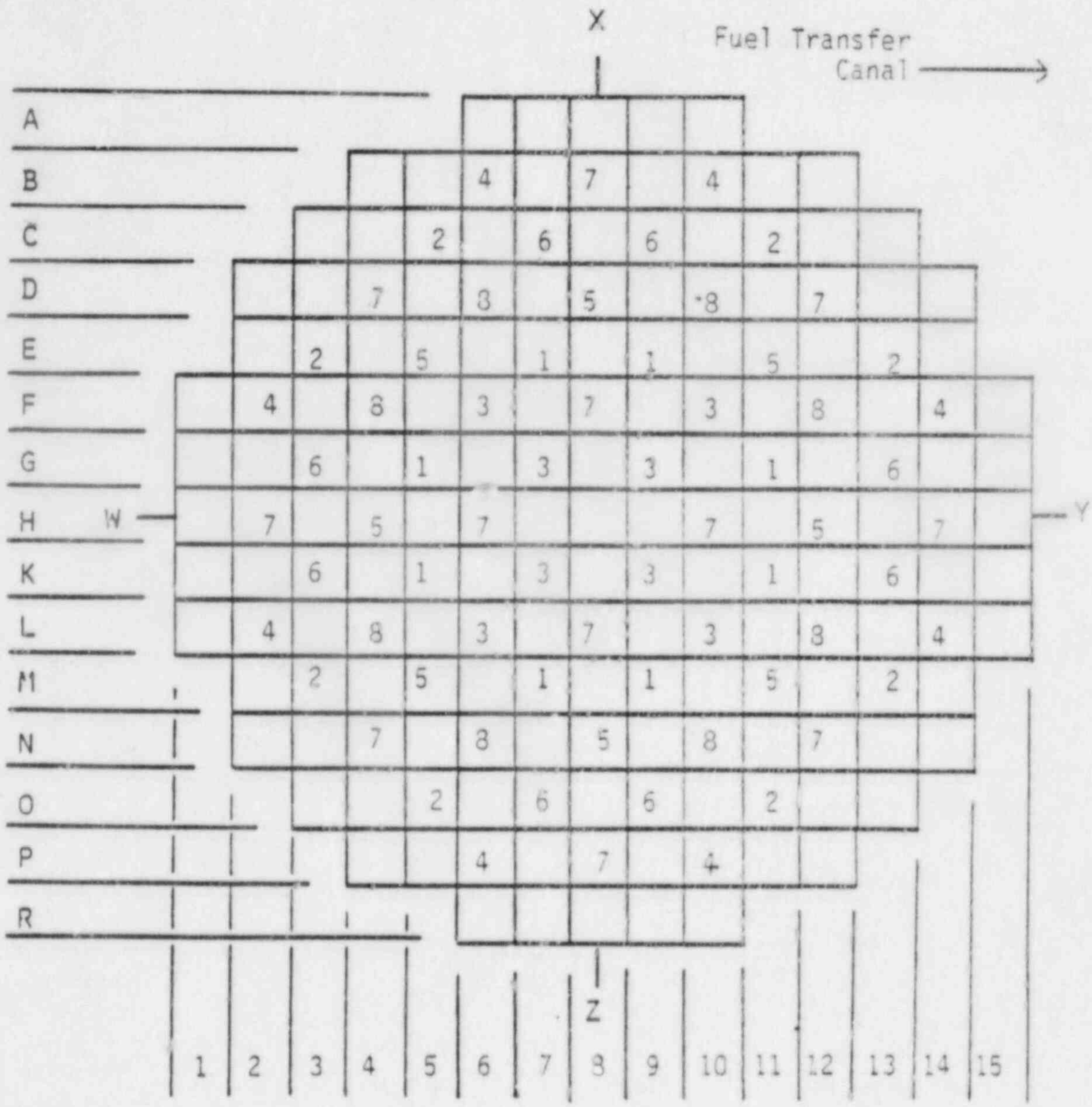
Note: F Denotes Fresh Fuel

Figure 3-2. Enrichment and Burnup Distribution, ANO-1 Cycle 8 off 425 EFPD Cycle 7

	8	9	10	11	12	13	14	15
H	2.95 45792	3.30 15232	3.21 23972	3.35 0	3.30 16877	3.35 0	3.30 16806	3.21 31130
K		3.30 15233	3.35 0	3.21 26611	3.35 0	3.21 21394	3.35 0	3.30 12866
L			3.21 23982	3.35 0	3.21 22404	3.35 0	3.30 16528	3.30 16833
M				3.30 17455	3.35 0	3.30 17330	3.30 16865	
N					3.30 16905	3.35 0	3.21 23087	
O						3.30 10891		
P								
R								

x.xx	Initial Enrichment, wt % U-235
xxxxx	BOC Burnup MWd/mtU

Figure 3-3. Control Rod Locations and Group Designations for ANO-1 Cycle 8



X Group Number

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

Figure 3-4. LBP Enrichment and Distribution, ANO-1 Cycle 8

	8	9	10	11	12	13	14	15
H				1.35		1.35		
K			1.21		1.18		0.20	
L		1.21		1.24		0.20		
M	1.35		1.24		1.24			
N		1.18		1.24		0.20		
O	1.35		0.20		0.20			
P		0.20						
R								

x.xx — LBP Concentration, wt % B_4C in Al_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel parameters for ANO-1 cycle 8 are discussed below and listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Retainer assemblies will be used on two fuel assemblies that contain the regenerative neutron sources, and on sixty-four fuel assemblies that contain BPRAs. Justification of the design and use of retainer assemblies is given in references 4 and 5. The batch 7D fuel assembly has the highest burnup in the core.

The sixty four batch 10 fuel assemblies incorporate the design features of anti-straddle lower end fittings and annealed guide tubes. The anti-straddle end-fitting prevents mis-positioning fuel assemblies during refueling operations and the annealed guide tubes reduce incore irradiation fuel assembly growth which allows for higher burnup capability.

The MK-BEB fuel assembly differs from the MK-B design in that it permits easy removal of a limited number of fuel rods. In addition, windows are cut into the upper grid skirt to permit easy observation of fuel rod growth.

4.2. Fuel Rod Design

The MK-BEB fuel rod design differs from the MK-B fuel rod in several areas. The MK-BEB fuel rod cladding is thicker with a lower prepressure to achieve better high burnup performance. Annular pellets, which are expected to improve high burnup performance are contained in some of the MK-BEB rods. The pin pre-pressure in the batch 10 fuel rods has been reduced 50 psi to improve fuel performance. The reduced pre-pressure has been considered in all mechanical and thermal analysis. The results of the mechanical evaluations of the fuel rods are discussed below.

4.2.1. Cladding Collapse

The batch 7D fuel assembly is more limiting than batches 8, 9, and 10 because of its previous incore exposure time. The batch 7D power history was analyzed to ensure that creep ovalization will not affect the fuel performance during cycle 8. The creep collapse analysis is based on reference 6.

The creep collapse analysis predicts a collapse time greater than 45000 effective full power hours. (EFPH), which is longer than the maximum expected residence time of 41000 EFPH (Table 4-1).

4.2.2. Cladding Stress

The ANO-1 cycle 8 stress parameters are enveloped by conservative fuel rod stress analysis. The same method was used for analysis of cycle 8 that had been used on the previous cycle.

4.2.3. Cladding Strain

The fuel design criteria specify a 1.0% limit on cladding plastic tensile circumferential strain. The pellet is designed to ensure that plastic cladding strain is less than 1.0% at design local pellet burnup and heat generation rate. The design values are higher than the worst-case values the ANO-1 cycle 8 fuel is expected to see. The strain analysis is also based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance for the cladding inside diameter.

4.3. Thermal Design

All fuel in the cycle 8 core is thermally similar. The design of the batch 7D lead test assembly 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 core. All thermal design analyses for cycle 8 fuel used the TACO2 code, as described in Reference 7, for fuel temperature and fuel rod internal pressure prediction.

The results of the thermal design evaluation of the cycle 8 core are summarized in Table 4-2. Cycle 8 core protection limits were based on a linear heat (LHR) to centerline fuel melt of 20.5 kw/ft as determined by the TACO2 code. The LHR to melt of the LTA fuel is greater than 20.5 kw/ft. The maximum fuel assembly burnup at EOC 8 is predicted to be less than

41,600 MWd/mtU for the Mark-B fuel and less than 56,300 MWd/mtU for the LTA fuel. The fuel rod internal pressures have been evaluated with TACO2 for the highest burnup fuel rods and are predicted to be less than the nominal reactor coolant pressure of 2200 psia.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for Batch 10 fuel assemblies is identical to that of the previous fuel batches.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. As of April 30, 1986, the following experience has been accumulated for eight B&W 177 fuel assembly plants using the Mark B fuel assembly:

Reactor	Current cycle	Max FA burnup,[(a)] MWd/mtU Incore	Max FA burnup,[(a)] MWd/mtU Discharged	Cumulative net electric output,[(b)] MWh
Oconee 1	10	33,710	50,598	62,028,968
Oconee 2	8	38,100	37,326	55,785,115
Oconee 3	9	37,714	39,229	55,385,714
Three Mile Island	5	28,440	32,400	25,105,483
Arkansas Nuclear One, Unit 1	7	41,960	36,820	48,299,124
Rancho Seco	7	26,100	38,268	39,078,111
Crystal River 3	6	24,970	31,420	35,863,252
Davis-Besse	5	31,020	32,790	25,233,177

[(a)] As of April 30, 1986.

[(b)] As of January 31, 1986.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch 7D	Batch 8B	Batch 9	Batch 10
Fuel assembly type	MK BEB	MK B4	MK B4	MK B4
Number of assemblies	1	44	68	64
Fuel rod OD (nom) in	0.430	0.430	0.430	0.430
Fuel rod ID (nom) in	0.371	0.377	0.377	0.377
Undensified active fuel length in	138.25	141.8	141.8	141.8
Fuel pellet OD (mean) in	0.3635	0.3686	0.3686	0.3686
Fuel pellet initial density (nom) % TD	95	95	95	95
Initial fuel enrichment wt. % ^{235}U	2.95	3.21	3.30	3.35
Average burnup, BOC, MWd/mtU	45800	24200	15900	0
Cladding collapse time, EFPH	>45000	>35000	>35000	>35000
Estimated residence exposure time, EFPH EOC	41000	30000	20000	10000

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 7D	Batch 8B	Batch 9	Batch 10
No. of assemblies	1[(a)]	44	68	64
Initial density, % TD	95.0	95.0	95.0	95.0
Initial pellet OD, in	0.3635	0.3686	0.3686	0.3686
Initial stack height, in	138.25	141.80	141.80	141.80
Enrichment, % U-235	2.95	3.21	3.30	3.35
Nominal linear heat rate at 2568 MWt, kW/ft[(b)]	5.89	5.74	5.74	5.74
TACO2-based Predictions				
Average fuel temperature at nominal LHR, F	<1400	1400	1400	1400
Minimum LHR to melt, kW/ft	21.1	20.5	20.5	20.5

Core average LHR = 5.74 kW/ft

[(a)] LTA analysis results are reported in Reference 2.

[(b)] Based on a nominal stack height.

Babcock & Wilcox
a McDermott company

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 lists the core physics parameters of design cycles 7 and 8. The values for cycle 7 were generated using PDQ07[8] and the values for cycle 8 were calculated with the NOODLE code.[9] Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 8 at full power with equilibrium xenon and nominal rod positions.

Differences in feed enrichment, BPRA loading, and shuffle pattern make it difficult to compare the physics parameters of cycles 7 and 8. 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 8 is less than that for the design cycle 7 at BOC, but greater at APSR pull and EOC. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 8 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.

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 7 reload report.[3]

5.2. Analytical Input

The cycle 8 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

Core design changes for cycle 8 include the removal of the center CRA and a change in the LBP length. The center CRA will be replaced with a stand pipe and blind flange. Removal of the center CRA will have a negligible effect on the nuclear parameters for cycle 8. The LBP used in cycle 8 has a 4.5-inch longer poison stack than that used in cycle 7, i.e., 121.5 versus 117 inches of Al_2O_3-B_4C. The top 4.5 inches of the poison stack are replaced by a Zircaloy tubular spacer. This LBP design asymmetrically positions the burnable poison stack relative to the fuel column and alters the core axial power shape to create increased "effective maneuvering room" at the beginning of the cycle.

As stated in section 5.1, the NOODLE code was used to calculate the physics parameters for cycle 8. The NOODLE modeling of the two-group homogenized fuel assembly is the same as that used in PDQ07. However, the analytical expression NOODLE uses for the spatial flux solution provides more accurate results than the finite difference expression used in PDQ07 when there are few flux solution points per assembly. Reference 9 illustrates the calculational accuracy attainable with NOODLE in comparison to measured results for various physics parameters. PDQ07 results are compared to measured data in references 10 and 11. These comparisons show NOODLE to be as accurate as PDQ07.

As in cycle 7, the APSRs will be withdrawn near the end of cycle 8 (380 EFPD). The calculated stability index at 384 EFPD without APSRs is -0.022 h^{-1} which demonstrates the axial stability of the core. The calculational methods used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle. The operating limits (Technical Specifications changes) for the reload cycle are given in section 8.

Table 5-1. Physics Parameters for ANO-1, Cycles 7 and 8[a]

	Cycle 7[b]	Cycle 8[c]
Cycle length, EFPD	420	420
Cycle burnup, MWd/mtU	13,158	13,147
Avg. core burnup, EOC, MWd/mtU	24,238	25,522
Initial core loading, mtU	82.0	82.0
Critical boron - BOC, ppm (No Xe)		
HZP,[d] group 8 ins	1578	1644
HFP, group 8 ins	1346	1409
Critical boron - EOC, ppm		
HZP, group 8 out, no Xe	696	651
HFP, group 8 out, eq Xe	83	18
Control rod worths - HFP, BOC, % k/k		
Group 6	1.20	1.14
Group 7	1.65	1.49
Group 8	0.39	0.39
Control rod worths - HFP, EOC, % k/k		
Group 7	1.53	1.52
Max ejected rod worth - HZP, % k/k[e]		
BOC (N-12), group 8 ins	0.69	0.55
380 EFPD (N-12), group 8 ins	0.50	0.60
EOC (N-12), group 8 out	0.52	0.59
Max stuck rod worth - HZP, % k/k		
BOC (N-12), group 8 ins	1.71	1.58
380 EFPD (N-12), group 8 ins	1.73	1.86
EOC (N-12), group 8 out	1.29	1.63
Power deficit, HFP to HZP, % k/k		
BOC	1.60	1.56
EOC	2.35	2.34

	Cycle 7 (b)	Cycle 8 (c)
Doppler coeff - HFP, 10^{-5} ($\Delta k/k/^{\circ}F$)		
BOC (no Xe)	-1.53	-1.54
EOC (eq Xe)	-1.80	-1.84
Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^{\circ}F$)		
BOC, (no Xe, crit ppm, group 8 ins)	-0.69	-0.51
EOC, (eq Xe, 0 ppm, group 8 out)	-2.79	-2.78
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	129	129
EOC	109	111
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.55	2.55
EOC (equilibrium)	2.68	2.72
Effective delayed neutron fraction - HFP		
BOC	0.0063	0.0062
EOC	0.0052	0.0052

(a) Cycle 8 data are for the conditions stated in this report. The Cycle 7 core conditions are identified in Reference 3.

(b) Based on 400 EFPD at 2568 MWt, Cycle 6.

(c) Based on 425 EFPD at 2568 MWt, Cycle 7.

(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 8

	BOC % Δk/k	380 EFPD % Δk/k	420 EFPD % Δk/k
Available Rod Worth			
Total rod worth, HZP	8.85	9.38	9.15
Worth reduction due to poison material burnup	-0.10	-0.10	-0.10
Maximum stuck rod, HZP	-1.58	-1.86	-1.63
Net worth	7.17	7.42	7.42
Less 10% uncertainty	-0.72	-0.74	-0.74
Total available worth	6.45	6.68	6.68
Required Rod Worth			
Power deficit, HFP to HZP	1.57	2.30	2.34
Allowable inserted rod worth	.50	.60	.65
Flux redistribution	.84	1.20	1.20
Total required worth	2.91	4.10	4.19
Shutdown margin (total available worth minus total required worth)	3.54	2.58	2.49

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

Figure 5-1. ANO-1 Cycle 8, BOC (4 EFPD) Two-Dimensional Relative Power Distribution -- Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.70	1.17	1.08	1.27	1.28	1.29	1.02	0.42
K	1.16	1.24	1.29	1.05	1.28	1.12	1.15	0.54
L	1.08	1.29	1.10	1.27	8 0.99	1.31	0.88	0.39
M	1.27	1.04	1.26	1.26	1.28	1.06	0.62	
N	1.27	1.28	8 0.99	1.28	1.18	1.01	0.36	
O	1.28	1.12	1.31	1.06	1.02	0.59		
P	1.02	1.15	0.88	0.62	0.37			
R	0.42	0.54	0.39					

x x.xx	Inserted Rod group No. Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The fresh batch 10 fuel is hydraulically and geometrically similar to the previously irradiated batches 8B and 9 fuel. The modified Mark B lower end fitting (LEF) was found to have an insignificant impact on thermal-hydraulic results. The batch 7D LTA has been analyzed to ensure that it is never the limiting assembly during cycle 8 operation. The results of the thermal-hydraulic analysis for the LTA are provided in reference 2.

The thermal-hydraulic design evaluation supporting cycle 8 operation is based on methods and models described in references 12, 13, 14, and 15. The cycle 8 thermal-hydraulic design is identical to cycle 7. The thermal-hydraulic design conditions for cycles 7 and 8 are summarized in Table 6-1.

The reactor protection system (RPS) setpoints for the DNB-based variable low pressure trip will remain the same for cycle 8. The 1.08 flux/flow setpoint remains applicable for cycle 8.

A rod bow topical report (reference 16), which addresses the mechanisms and resulting conditions of rod bow, has been submitted to and approved by the NRC. The topical report concludes that rod bow penalty is insignificant and is offset by the reduction in power production capability of the fuel assemblies with irradiation. Therefore, no departure from nucleate boiling ratio (DNBR) reduction due to rod bow need be considered for cycle 8.

Table 6-1. Maximum Design Conditions, Cycles 7 and 8

	Cycle 7	Cycle 8
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
DNBR modeling	Crossflow	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.65 cosine	1.65 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.	141.8	141.8
Avg. heat flux at 100% power, 10^3 Btu/h-ft^2	174	174
Max. heat flux at 100% power, 10^3 Btu/h-ft^2	492	492
CHF correlation	B&W-2	B&W-2
Minimum DNBR		
at 112% power	2.08	2.08
at 100% power	2.43	2.43

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 8 parameters to determine the effect of the cycle 8 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effect of fuel densification on the FSAR accident results have been evaluated and are reported in reference 17. Since batch 10 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference 17 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 plutonium-239 has different fission yields than uranium-235, the mixture of fission product nuclides in the core changes slightly as the plutonium-239 to uranium-235 fission ratio changes, i.e., plutonium fissions produce more of some nuclides and less of other nuclides. 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 8. The bases used in the dose calculation are identical to those presented in the FSAR except for the following three differences:

1. The fission yields and half-lives used in the new calculations are based on more current data.
2. Updated (lowered) whole body gamma dose conversion factors.

3. 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 with those calculated specifically for cycle 8 (Table 7-1) show that some doses are slightly higher and some are slightly lower than the FSAR values. However, with the exception of the maximum hypothetical accident (MHA) all doses are bounded by the values represented 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 thyroid dose at the exclusion area boundary (EAB) is 157.3 Rem (53% of the 10 CFR 100 limit) and the 30 day thyroid dose at the low population zone (LPZ) is 73.0 Rem (24% of the 10 CFR 100 limit). Thus, the radiological impact of accidents during cycle 8 is not significantly different than that 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. Thermal parameters for fuel batches 7D, 8, 9 and 10 are given in Table 4-2. The cycle 8 thermal-hydraulic maximum design conditions are compared with the previous cycle 7 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 8 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).[18] 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 19 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 8 fuel. These LHR limits include the effects of NUREG 0630.

It is concluded from the examination of cycle 8 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 8. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 8 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 8 are bounded by the FSAR, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of FSAR and Cycle 8 Accident Doses

	FSAR doses, Rem	Cycle 8 doses, Rem
Fuel Handling Accident		
Thyroid dose at EAB (2 h)	0.92	1.15
Whole body dose at EAB (2 h)	0.54	0.21
Steam Line Break		
Thyroid dose at EAB (2 h)	1.6	1.71
Whole body dose at EAB (2 h)	-	0.008
Steam Generator Tube Failure		
Thyroid dose at EAB (2 h)	0.0087	6.14
Whole body dose at EAB (2 h)	0.16	0.52
Waste Gas Tank Rupture		
Thyroid dose at EAB (2 h)	0.22	0.054
Whole body dose at EAB (2 h)	-	1.53
Control Rod Ejection Accident		
Thyroid dose at EAB (2 h)	11.4	12.2
Whole body dose at EAB (2 h)	0.014	0.008
Thyroid dose at LPZ (30 d)	8.3	9.09
Whole body dose at LPZ (30 d)	0.0099	0.005
LOCA		
Thyroid dose at EAB (2 h)	3.6	4.02
Whole body dose at EAB (2 h)	0.057	0.026
Thyroid dose at LPZ (30 d)	1.66	2.05
Whole body dose at LPZ (30 d)	0.043	0.018
Maximum Hypothetical Accident		
Thyroid dose at EAB (2 h)	153	157.3
Whole body dose at EAB (2 h)	10	4.80
Thyroid dose at LPZ (30 d)	64.1	73.0
Whole body dose at LPZ (30 d)	3.4	1.56

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

Parameter	FSAR and densification report value	ANO-1 cycle 8
Doppler coeff (BOC), 10^{-5} $\Delta k/k/^{\circ}F$	-1.17	-1.54
Doppler coeff (EOC), 10^{-5} $\Delta k/k/^{\circ}F$	-1.30	-1.84
Moderator coeff (BOC), 10^{-4} $\Delta k/k/^{\circ}F$	0.0[a]	-0.51
Moderator coeff (EOC), 10^{-4} $\Delta k/k/^{\circ}F$	-4.0[b]	-2.78
All-rod group worth (HZP), % $\Delta k/k$	12.9	8.85
Initial boron concentration, ppm	1150	1409
Boron reactivity worth (HFP), ppm/% $\Delta k/k$	100	129
Max. ejected rod worth (HFP), % $\Delta k/k$	0.65	0.34
Dropped rod worth (HFP), % $\Delta k/k$	0.65	$\leq$0.20

(a) $+0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ was used for the moderator dilution analysis.

(b) -3.0×10^{-4} $\Delta k/k/^{\circ}F$ 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, 0-1000 MWd/mtU, kW/ft	Allowable peak LHR, 1000-2600 MWd/mtU, kW/ft	Allowable peak LHR after 2600 MWd/mtU, kW/ft
2	13.5	15.0	15.5
4	16.1	16.6	16.6
6	16.5	18.0	18.0
8	17.0	17.0	17.0
10	16.0	16.0	16.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 8 operation to account for changes in power peaking and control rod worths. As in cycle 7, a very low leakage fuel cycle and crossflow analysis were implemented in the fuel cycle design. The LOCA linear heat rate limits used to develop the Technical Specification Limiting Conditions for Operation include the impact of NUREG-0630 cladding swell and rupture model. In addition, an analysis was conducted to verify removal of the power level cutoff hold requirements of Technical Specification sections 3.5.2.4 and 3.5.2.5.

A cycle 8 specific analysis was conducted to generate Technical Specification Limiting Conditions for Operation (rod index, APSR position, axial imbalance, quadrant tilt). The analysis generated measurement-independent LCO limits which were then error-adjusted to give alarm setpoints for power operation. The Technical Specification LCO figures are presented as alarm setpoint figures. The fuel cycle design allows for Axial Power Shaping Rod (APSR) withdrawal at 380 $\pm$ 10 EFPD, and is reflected in the LCO figures. Figure 3.5.2-4 is also provided, which illustrates the burnup-dependent allowable LOCA linear heat rate limits used in the analysis. The analysis also verified the 3.1% quadrant tilt setpoints referenced in Technical Specification 3.5.2.4.

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.

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation above 60% of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.5.2.3. The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4. Quadrant tilt:

1. Except for physics tests, if quadrant tilt exceeds 3.1%, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of 3.1%.
2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 3.1% except for physics tests, or the following adjustments in setpoints and limits shall be made:
   a. The protection system maximum allowable setpoints (Figure 2.3-2) shall be reduced 2% in power for each 1% tilt.
   b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of 3.1%.
   c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 3.1%.
3. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.

4. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5. Control rod positions:

1. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

2. Operating rod group overlap shall be 20% ± 5 between two sequential groups, except for physics tests.

3. Except for physics tests or exercising control rods, (a) the control rod withdrawal limits are specified on Figures 3.5.2-1, 3.5.2-2A and 3.5.2-2B for 4, 3 and 2 pump operation respectively; and (b) the axial power shaping control rod withdrawal limits are specified on Figures 3.5.2-4A and 3.5.2-4B. If any of these control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours.

3.5.2.6. Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance in not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within 4 hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7. The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Superintendent.

Bases

The power-imbalance envelope defined in Figure 3.5.2-3 is based on (1) LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria and (2) the Protective System Maximum Allowable

Setpoints (Figure 2.3-2). Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while

4.7. REACTOR CONTROL ROD SYSTEM TESTS

4.7.1. Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1. The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2. If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3. If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14.

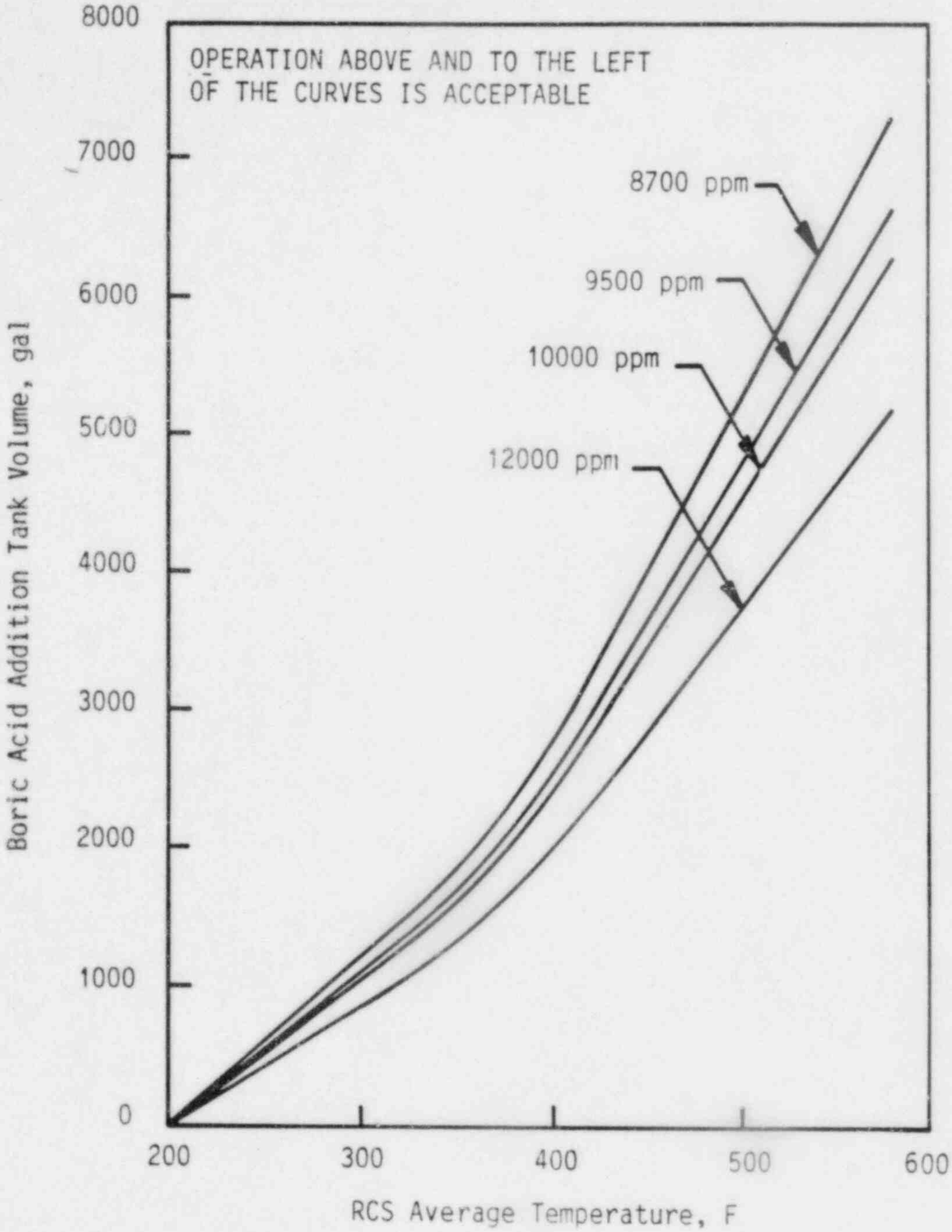
Each control rod drive mechanism shall be exercised by a movement of approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanism in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCES

(1) FSAR, Section 14

Figure 8-1. Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature -- ANO-1, Cycle 8



Temp., F	Required Volume, gal			
	8700 ppm	9500 ppm	10000 ppm	12000 ppm
579	7308	6657	6306	5200
532	6126	5580	5289	4355
500	5273	4802	4548	3749
400	2790	2543	2409	1984
300	1234	1129	1070	877
200	0	0	0	0

Figure 8-2. Rod Position Setpoints for 4-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8

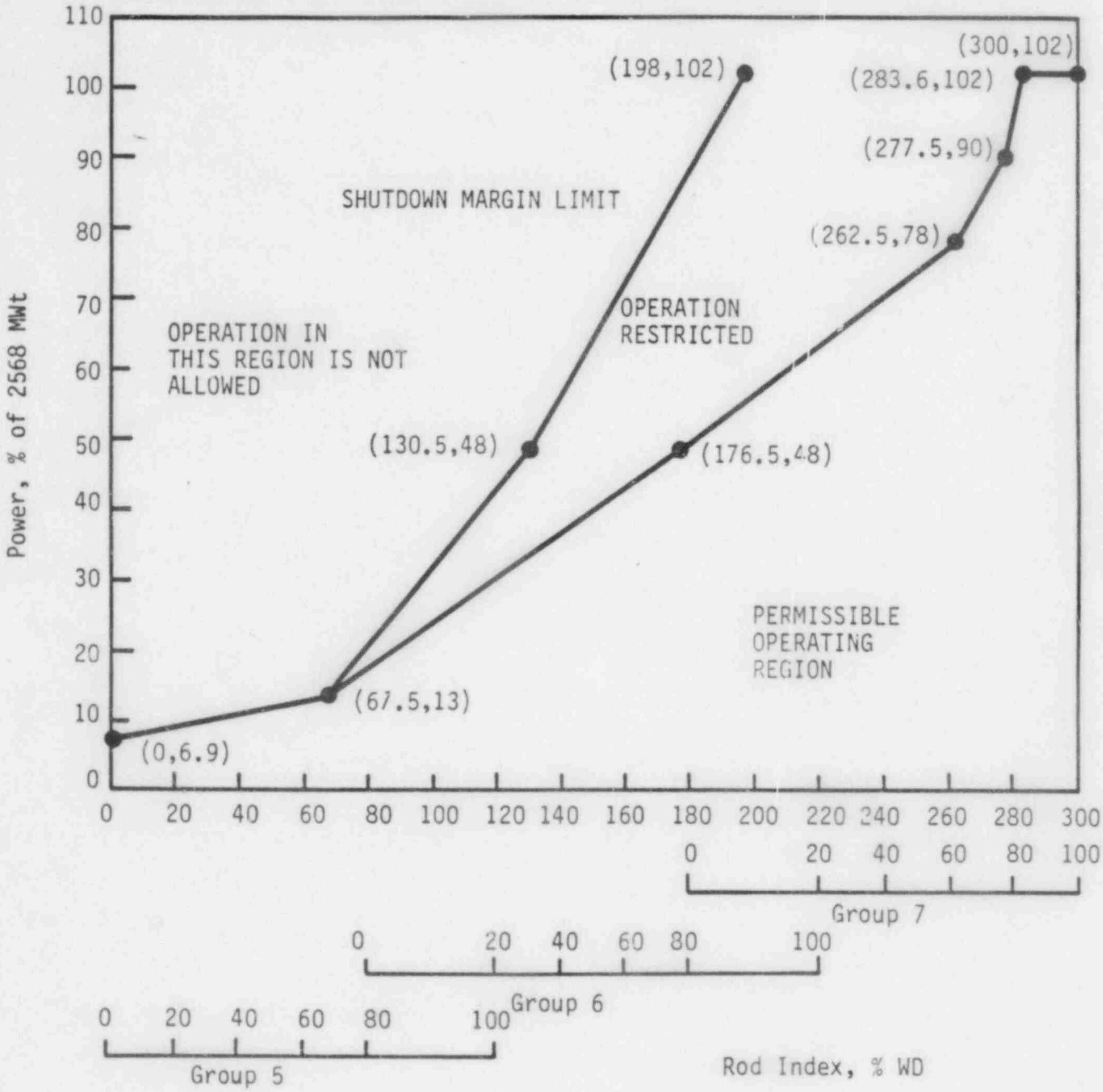


Figure 8-3. Rod Position Setpoints for 4-Pump Operation
From 25+10/-0 to 200± EFPD -- ANO-1 Cycle 8

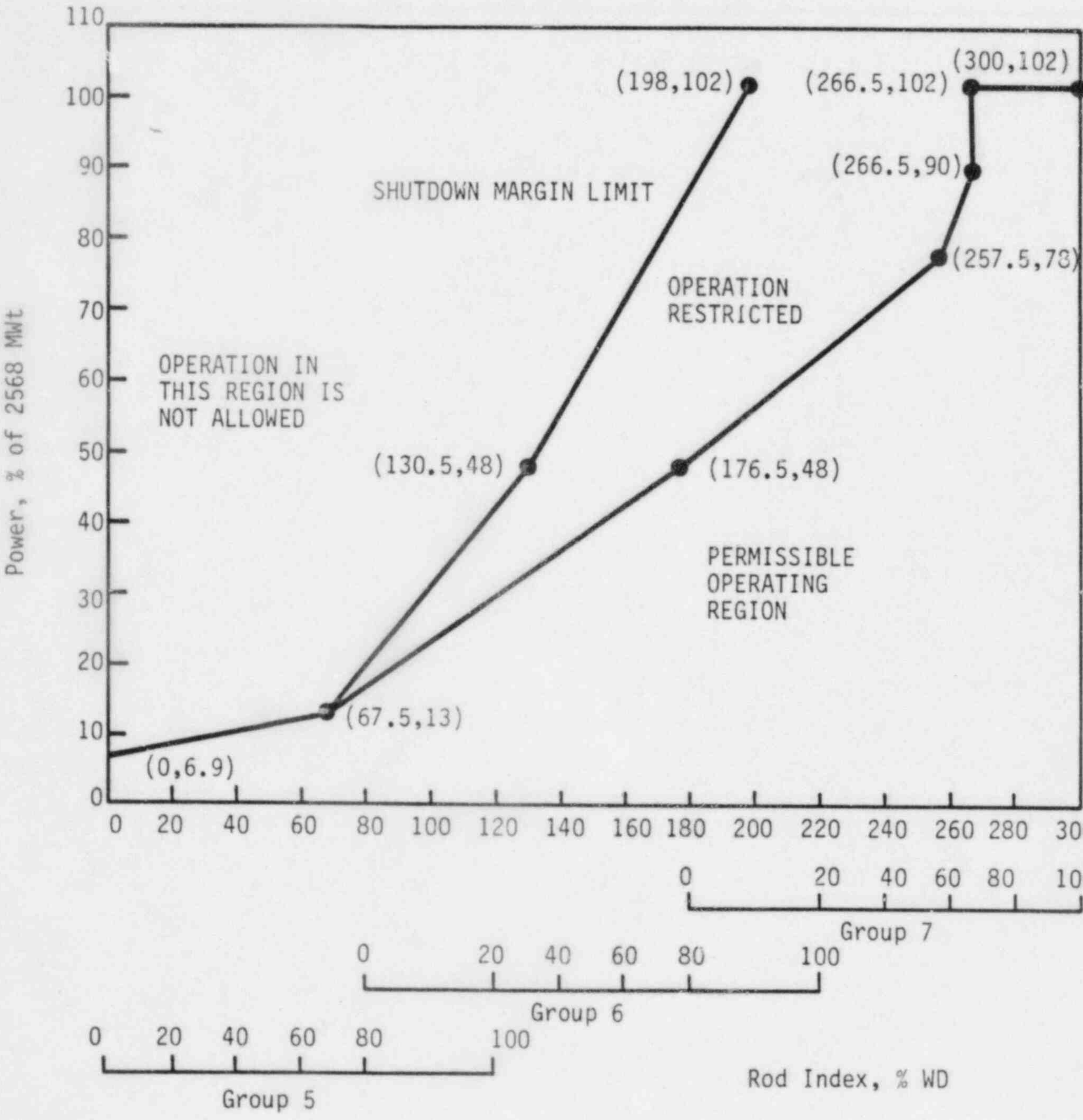


Figure 8-4. Rod Position Setpoints for 4-Pump Operation From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8

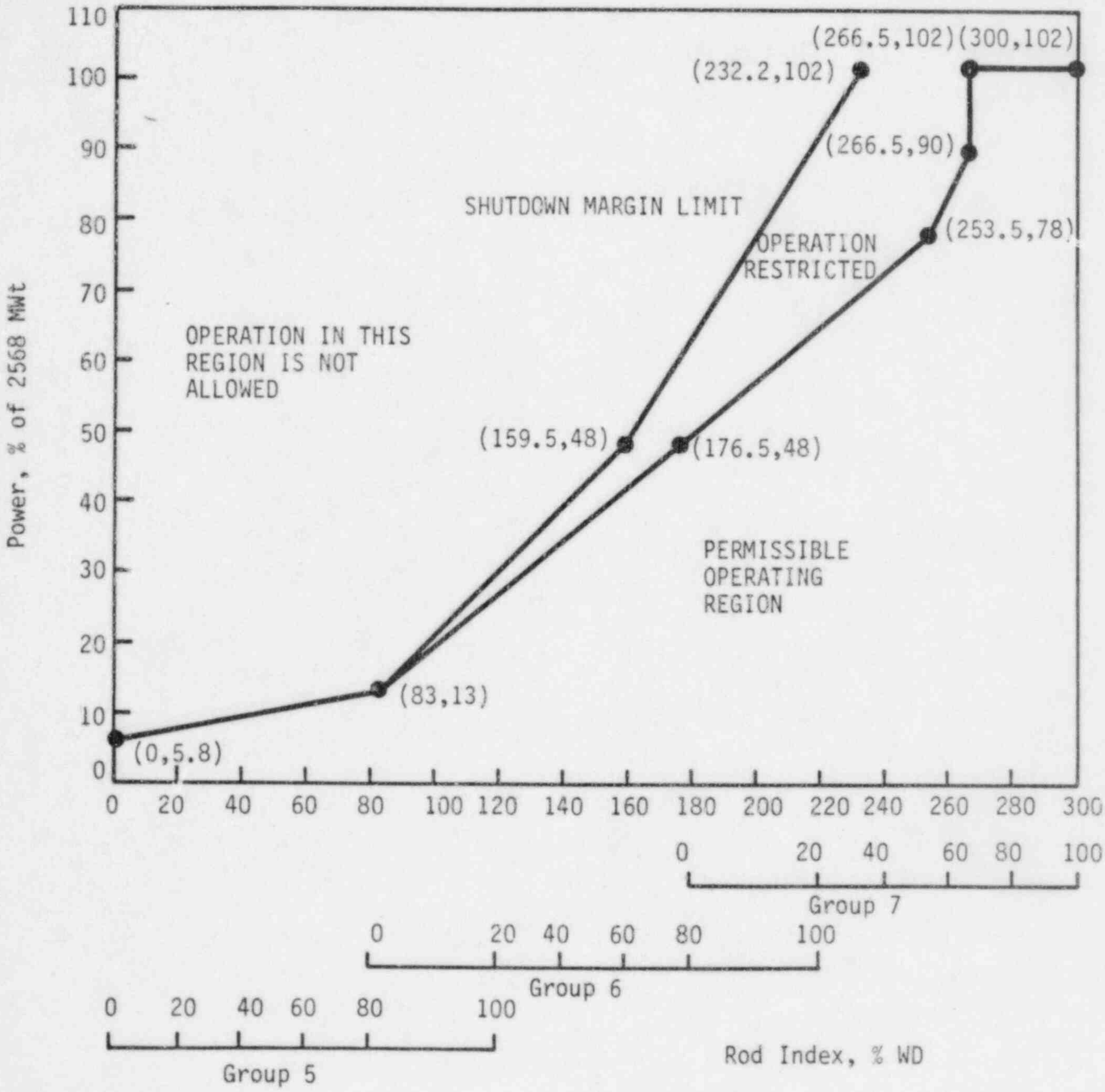


Figure 8-5. Rod Position Setpoints for 4-Pump Operation After 380±10 EFPD -- ANO-1 Cycle 8

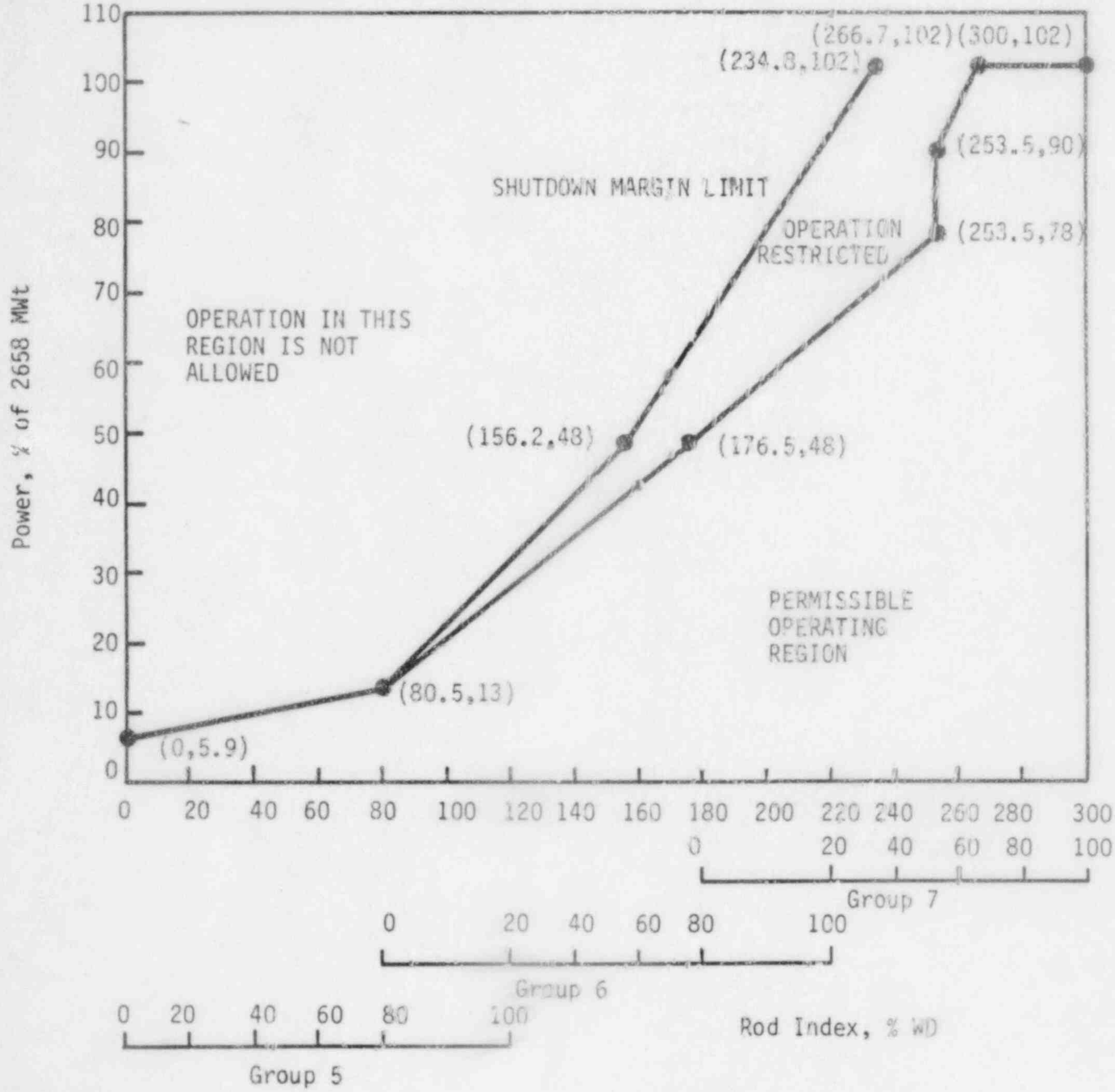


Figure 8-6. Rod Position Setpoints for 3-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8

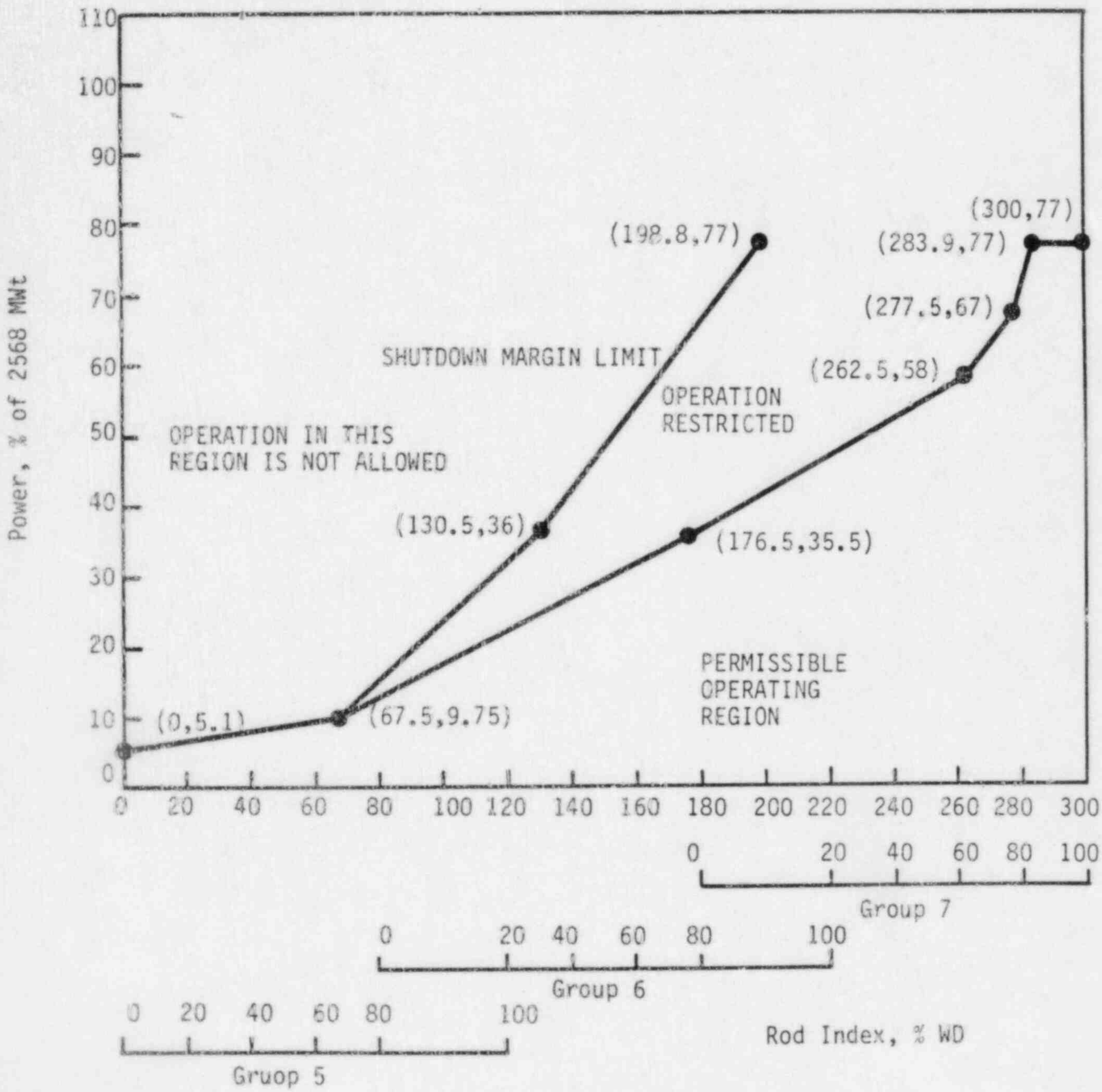


Figure 8-7. Rod Position Setpoints for 3-Pump Operation From 25+10/-0 to 200±10 EFPD -- ANO-1 Cycle 8

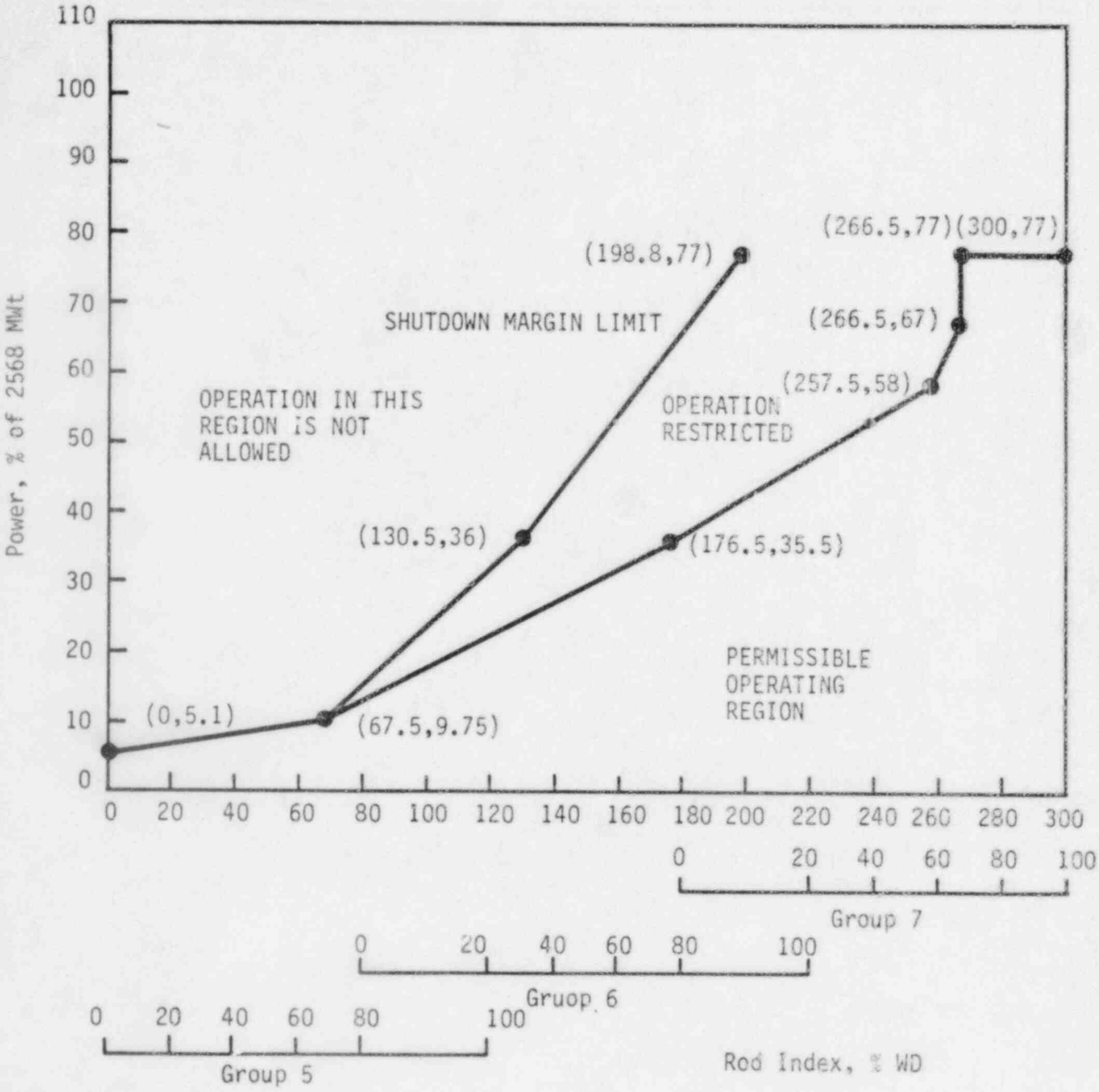


Figure 8-8. Rod Position Setpoints for 3-Pump Operation
From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8

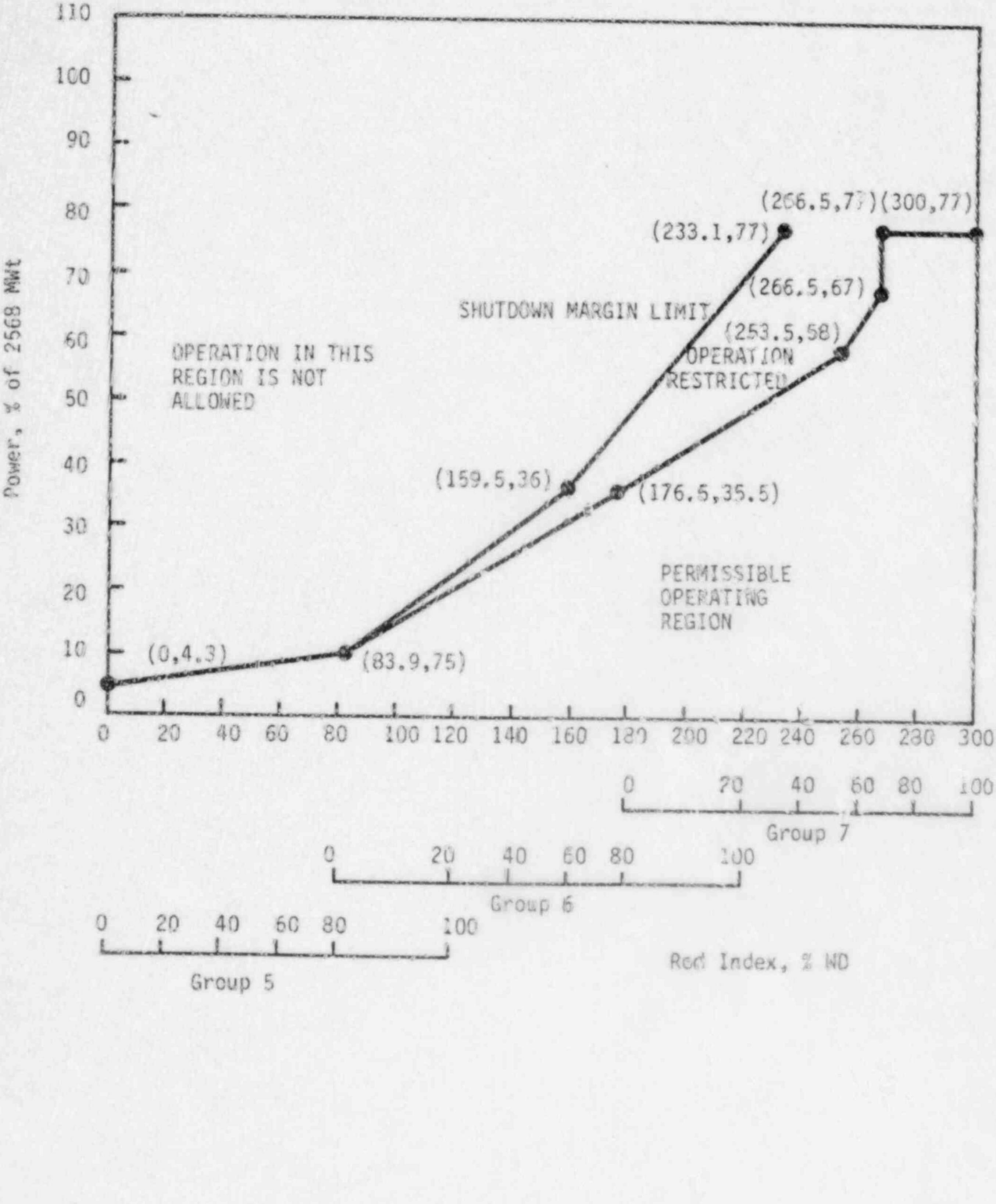


Figure 8-9. Rod Position Setpoints for 3-Pump Operation After 380±10 EFPD -- ANO-1 Cycle 8

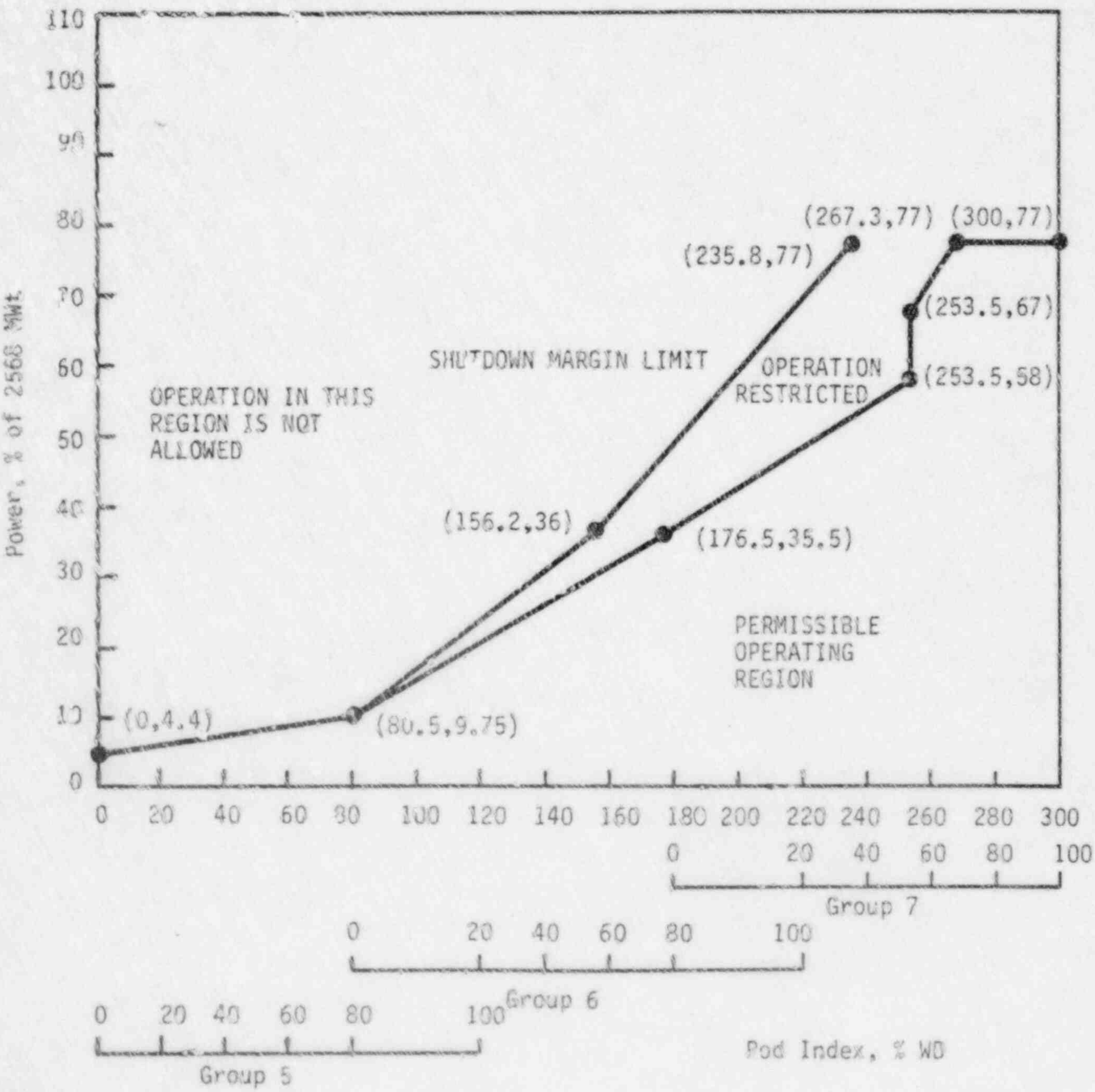


Figure 8-10. Rod Position Setpoints for 2-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8

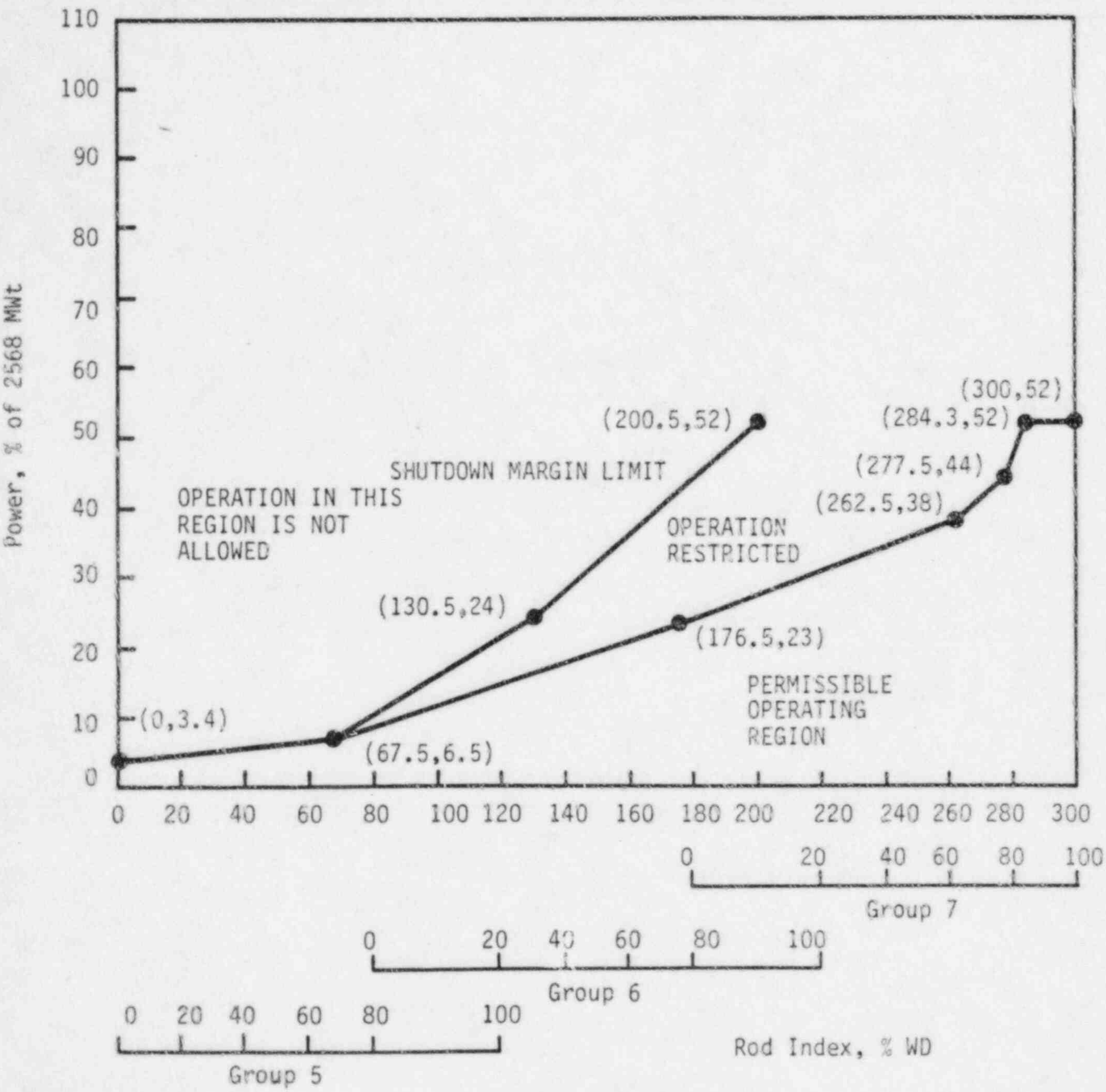


Figure 8-11. Rod Position Setpoints for 2-Pump Operation From 25+10/-0 to 200±10 EFPD -- ANO-1 Cycle 8

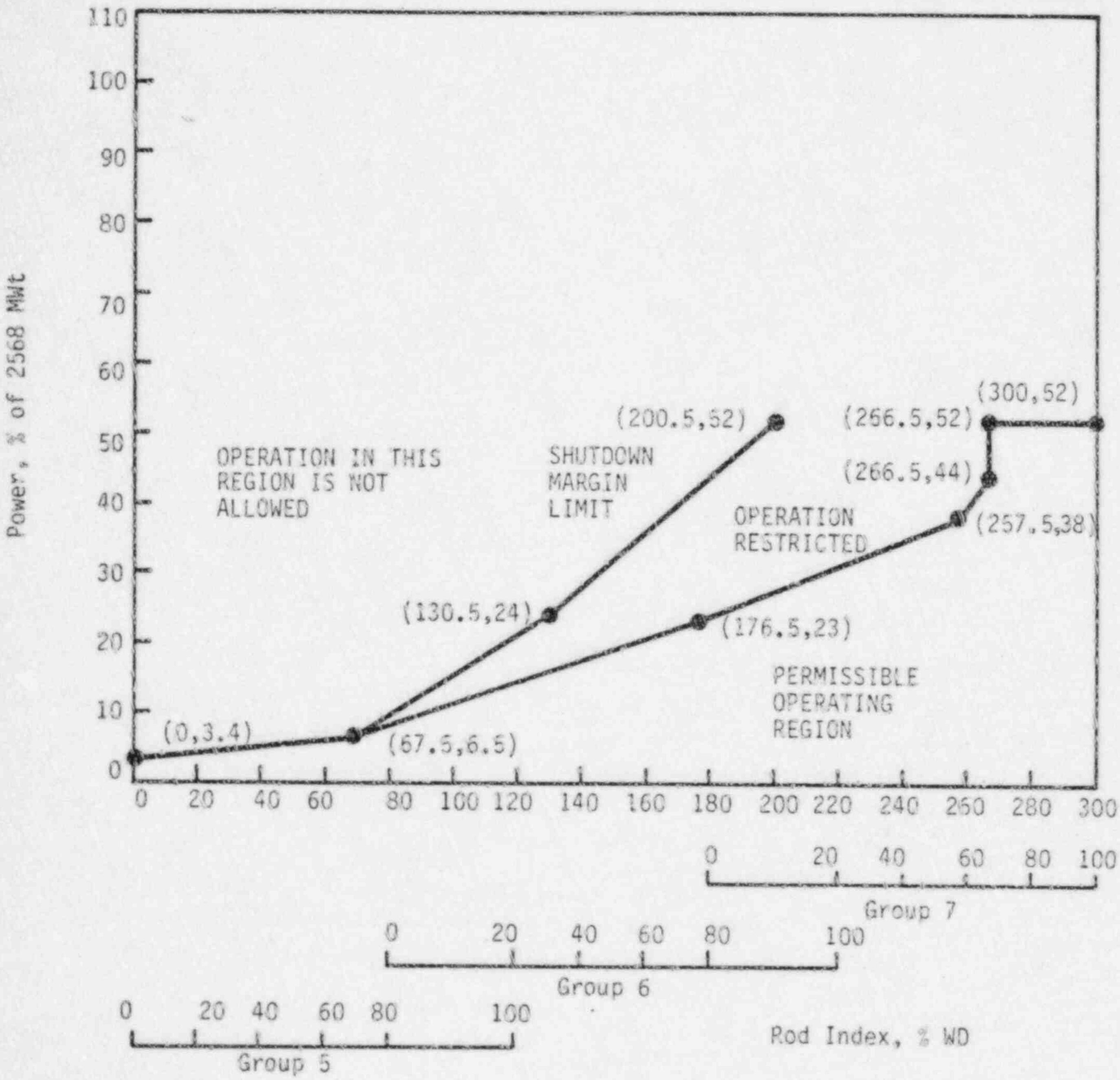


Figure 8-12. Rod Position Setpoints for 2-Pump Operation From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8

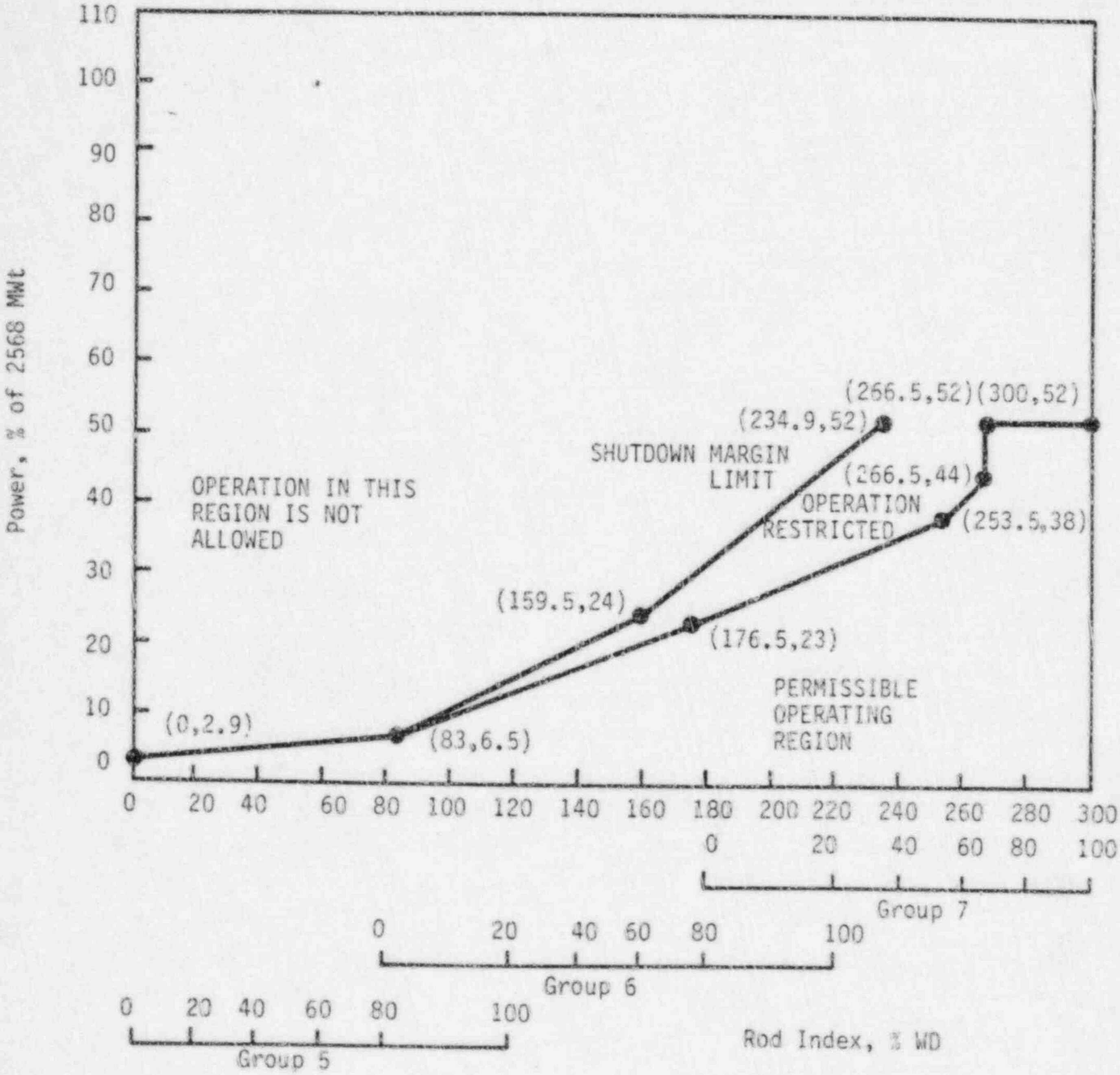


Figure 8-13. Rod Position Setpoints for 2-Pump Operation After 380±10 EFPD -- ANO-1 Cycle 8

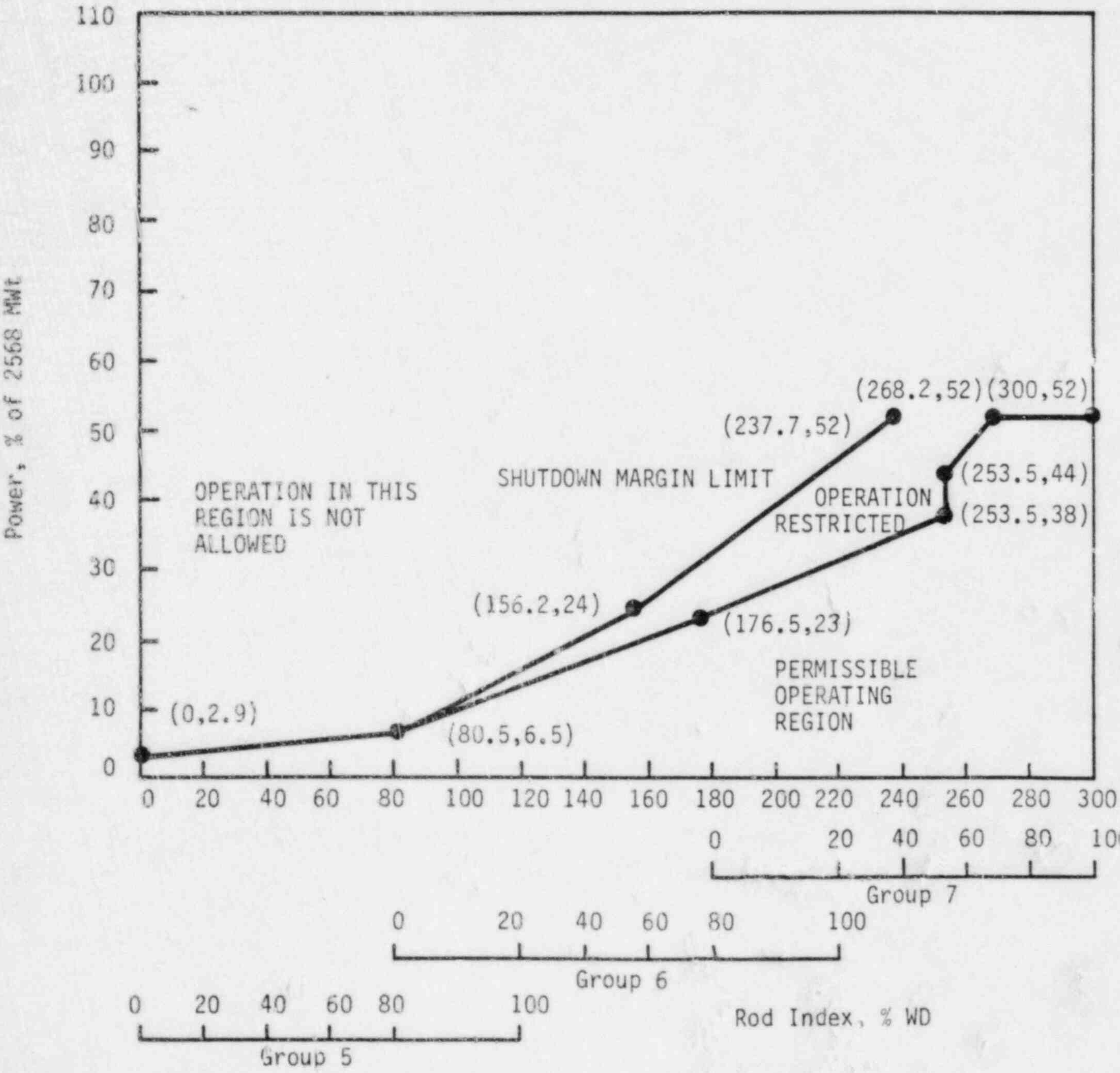


Figure 8-14. Operational Power Imbalance Setpoints for Operation From 0 to 25+10/-0 EFPD -- ANO-1, Cycle 8

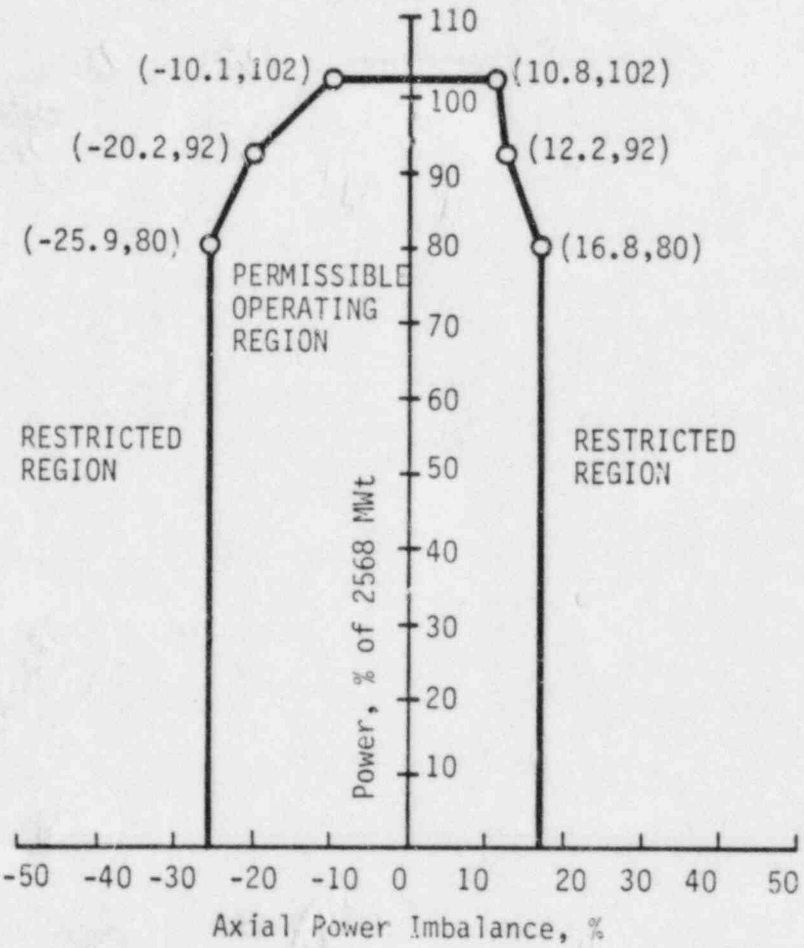


Figure 8-15. Operational Power Imbalance Setpoints for Operation From 25+10/-0 to 200±10 EFPD -- ANO-1, Cycle 8

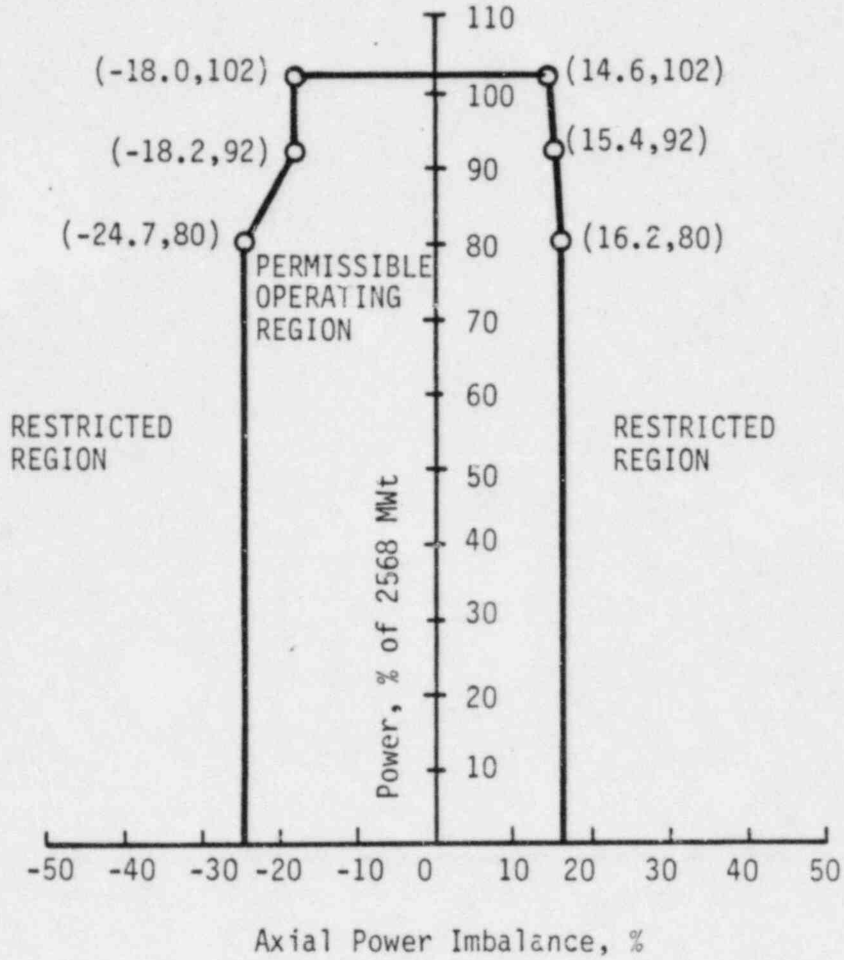


Figure 8-16. Operational Power Imbalance Setpoints for Operation From 200±10 to 380±10 EFPD -- ANO-1, Cycle 8

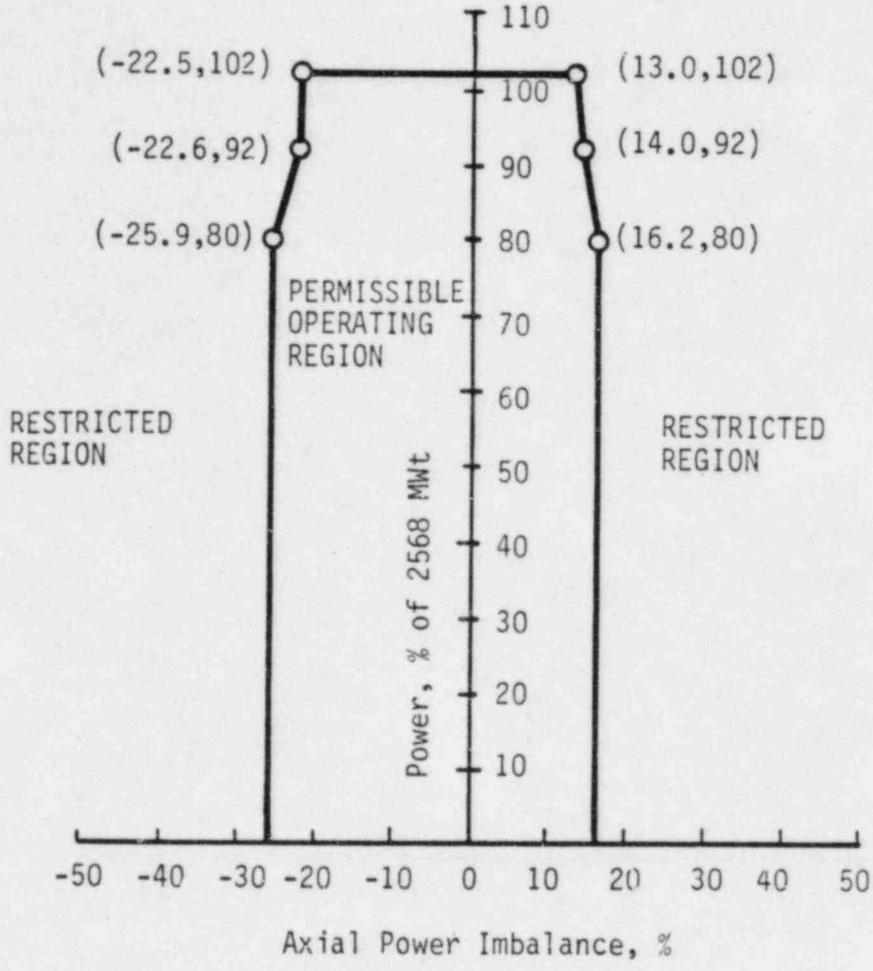


Figure 8-17. Operational Power Imbalance Setpoints for Operation After 380±10 EFPD -- ANO-1, Cycle 8

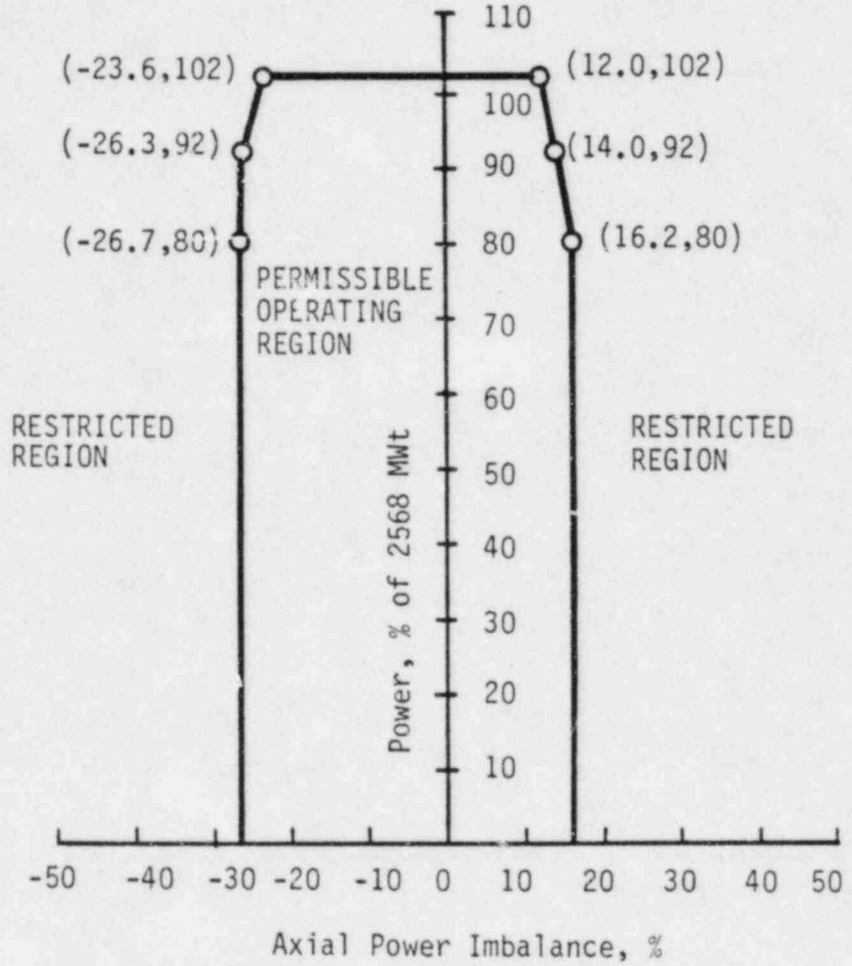


Figure 8-18. LOCA Limited Maximum Allowable Linear Heat Rate

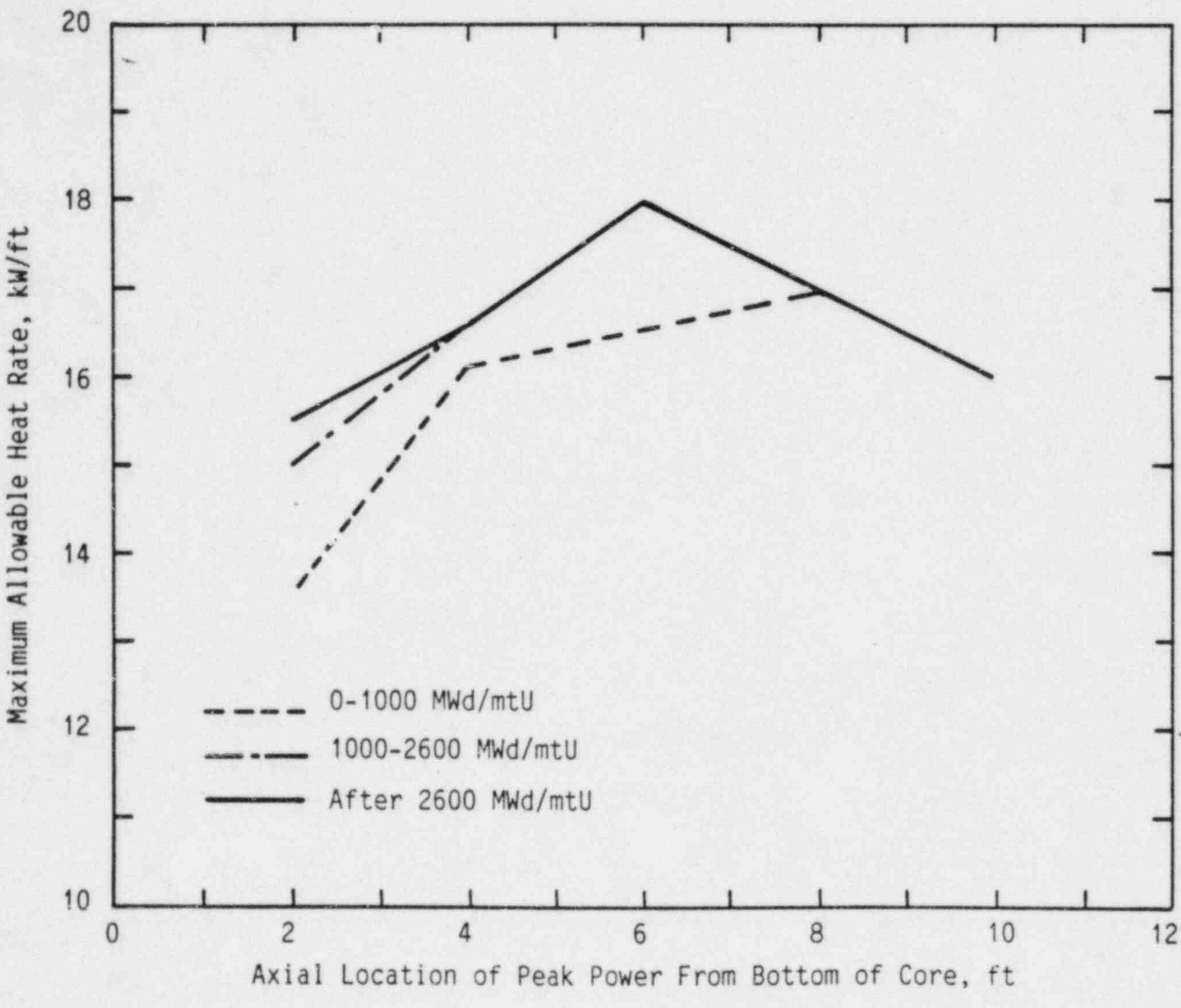


Figure 8-19. APSR Position Setpoints for Operation From 0 to 25+10/-0 EFPD -- ANO-1, Cycle 8

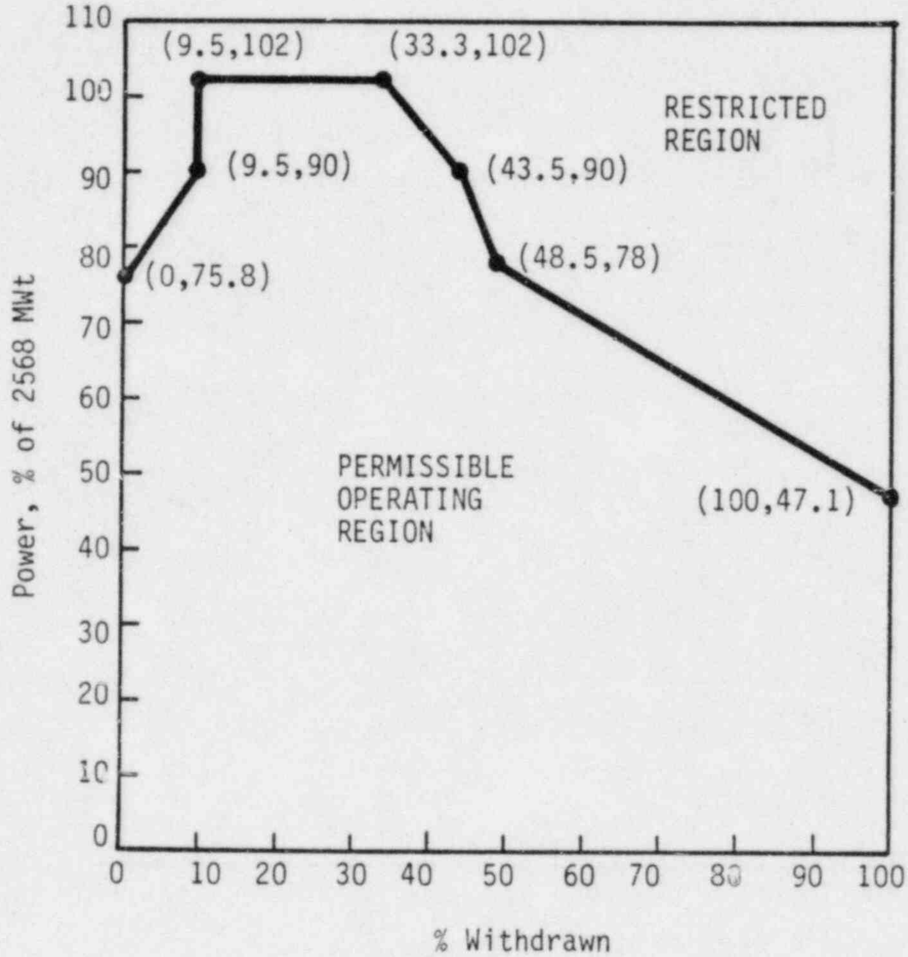
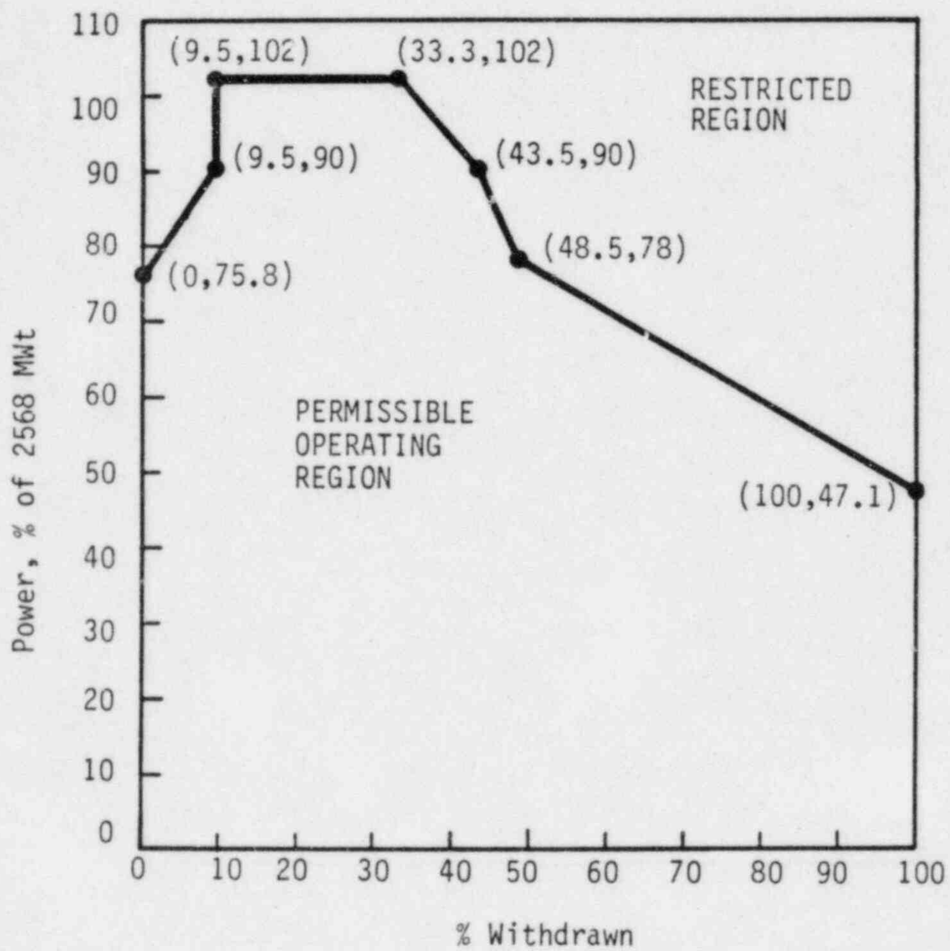


Figure 8-20. APSR Position Setpoints for Operation
From 25+10/-0 to 200±10 EFPD -- ANO-1, Cycle 8



Babcock & Wilcox
a McDermott company

Figure 8-21. APSR Position Setpoints For Operation From 200±10 to 380±10 EFPD -- ANO-1, Cycle 8

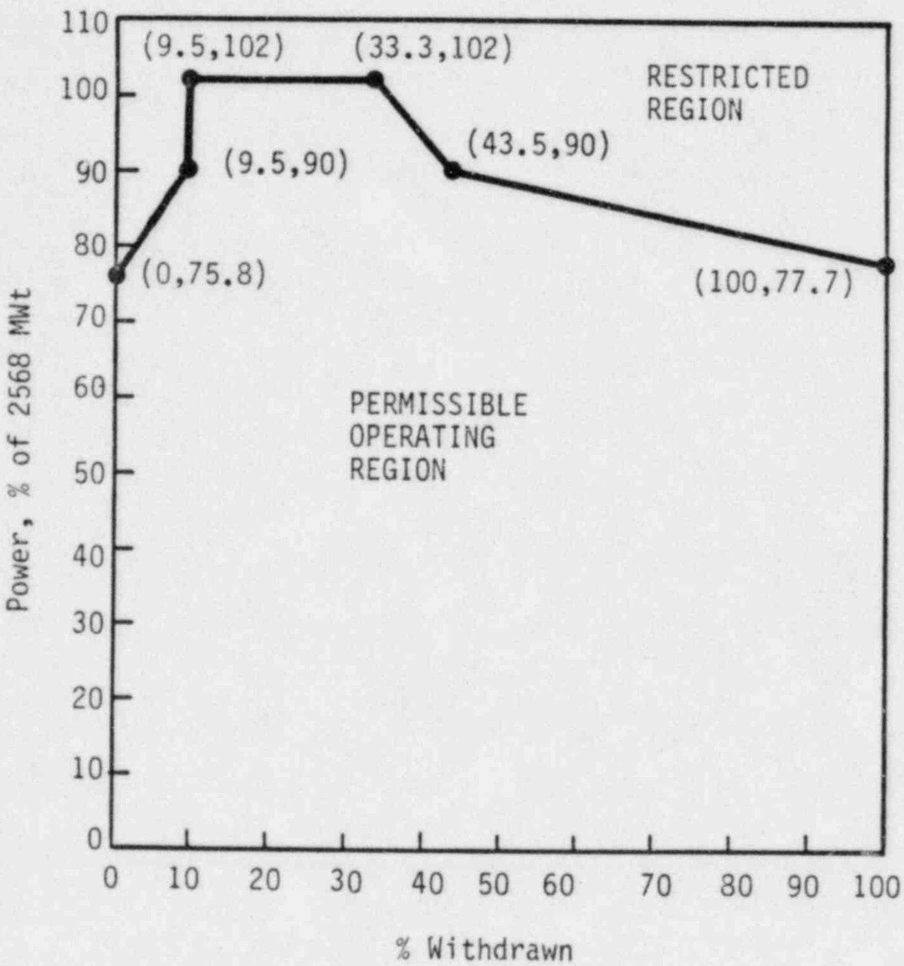
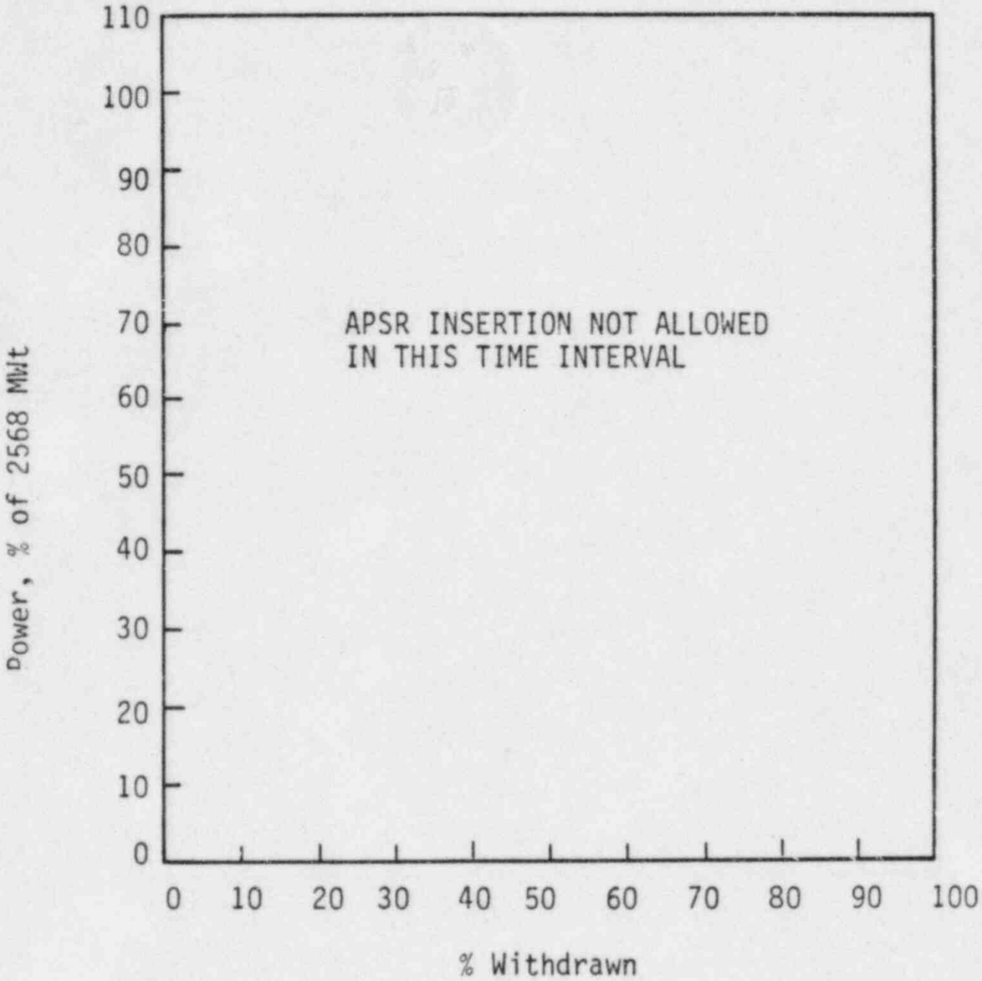


Figure 8-22. APSR Position Setpoints for Operation After 380 ± 10 EFPD -- ANO-1, Cycle 8



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 information 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. Acceptance 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 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

Once initial 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 concentration must be within $\pm$ 100 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal HZP temperature coefficient is measured at approximately the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. 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 $\pm$ 0.4×10^{-4} $\Delta k/k/^{\circ}F$.

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 the moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$.

9.2.3. Control Rod Group Reactivity Worth

Control rod 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 from 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:

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

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

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

9.3. Power Escalation Tests

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

Core power distribution tests are performed at 40 and 100% full power (FP). The test at 40% FP is essentially a check on power distribution in the core to identify any abnormalties before escalating to the 100% FP plateau. Peaking factor criteria are applied to the 40% FP core power distribution results to determine if 75% FP tests are required prior to 100% FP operation. If these criteria are met, the 75% FP tests are not required.

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

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:

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ more positive than } -8$$

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

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ more positive than } -12$$

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

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

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ more positive than } -5$$

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

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ more positive than } -7.5$$

9.3.2. Incore Vs. Excore Detector Imbalance Correlation Verification

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 greater than 0.96. 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 5F 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 measured fuel Doppler coefficient must be more negative than the acceptance criteria limit of -0.90×10^{-5} $\Delta k/k/^{\circ}F$.

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.

10. 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, Lynchburg, Virginia, October 1980.

3. Arkansas Nuclear One Unit 1, Cycle 7 Reload Report, BAW-1840, Babcock & Wilcox, Lynchburg, Virginia, August 1984.

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

5. J. H. Taylor (B&W) to S. A. Varga (NRC), Letter, "BPRA Retainer Reinsertion," January 14, 1980.

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

7. Y. H. Hsii, et al., TACO2-Fuel Pin Performance Analysis, BAW-10141P-A, Rev.1, Babcock & Wilcox, Lynchburg, Virginia, June 1983.

8. Babcock & Wilcox Version of PDQ User's Manual, BAW-10117P-A, Babcock & Wilcox, Lynchburg, Virginia, January 1977.

9. NOODLE -- A Multi-Dimensional Two-Group Reactor Simulator, BAW-10152, Babcock & Wilcox, Lynchburg, Virginia, September 1984.

10. Comparison of Core Physics Calculations with Measurements, BAW-10120, Babcock & Wilcox, Lynchburg, Virginia, June 1978.

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

12. B. R. Hao and J. M. Alcorn, LYNX1: Reactor Fuel Assembly Thermal Hydraulic Analysis Code, BAW-10129-A, Babcock & Wilcox, Lyncuburg, Virginia, July 1985.

13. LYNX2: Subchannel Thermal-Hydraulic Anlaysis Program, BAW-10130-A, Babcock & Wilcox, Lynchburg, Virginia, July 1985.

14. J. H. Jones, et al., LYNXT -- Core Transient Thermal-Hydraulic Program, BAW-10156, Babcock & Wilcox, Lynchburg, Virginia, February 1984.

15. R. L. Harne and J. H. Jones, Thermal-Hydraulic Crossflow Applications, BAW-1829, Babcock & Wilcox, Lynchburg, Virginia, May 1984.

16. Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, May 1983.

17. Arkansas Nuclear One, Unit 1-Fuel Densification Report, BAW-1391, Babcock & Wilcox, Lynchburg, Virginia, June 1973.

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

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