

ARKANSAS NUCLEAR ONE, UNIT 1
- Cycle 10 Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the tenth 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 10 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 9 and 10 reactor parameters related to power capability are summarized briefly in section 5 of this report. All of the accidents analyzed in the FSAR¹ have been reviewed for cycle 10 operation. In those cases where cycle 10 characteristics were conservative compared to those analyzed for previous cycles, new accident analyses were not performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 10 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 10 at a rated power level of 2568 Mwt.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Arkansas Nuclear One, Unit 1 is the currently operating cycle 9. Cycle 9 began in December, 1988 and operated at 100% power for approximately 30 effective full power days (EFPD). At that time, backflow through the HPI line led to a shutdown of approximately two months after which the unit was limited to 50% until analyses were available to justify a power increase. Except for one minor shutdown, the unit operated for about 25 EFPD at 50% power and following a (two week) outage, restarted to 80% FP. Operation at 80% lasted about 35 EFPD after which power was limited to 75% due to operation with only three reactor coolant pumps. Three pump operation limited power to 75% for approximately 90 EFPD when a brief outage was used to repair the inoperable RC pump. The remainder of the cycle was to be operated at 80% Full Power.

The cycle 10 design is based on a design cycle 9 length of 420 EFPD. No anomalies occurred during cycle 9 that would adversely affect fuel performance during cycle 10.

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 10 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 in-core 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. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1 for all fuel assemblies.

Figure 3-1 is the fuel shuffle diagram for ANO-1, cycle 10. The initial enrichments of batches 6E, 10B, 11 and 12 are 3.19, 3.35, 3.45 and 3.49 wt % U-235, respectively. The batch 6D assembly, all of batch 9B, and 4 of the twice-burned batch 10 assemblies will be discharged at the end of cycle 9. The center location will contain a batch 6 assembly discharged at the end of cycle 5 (designated 6E). The remaining 60 twice-burned batch 10 assemblies (designated 10B) will be shuffled to new locations, with 12 on the core periphery. The 60 once-burned batch 11 assemblies will be shuffled to new locations, and the 56 fresh batch 12 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 10.

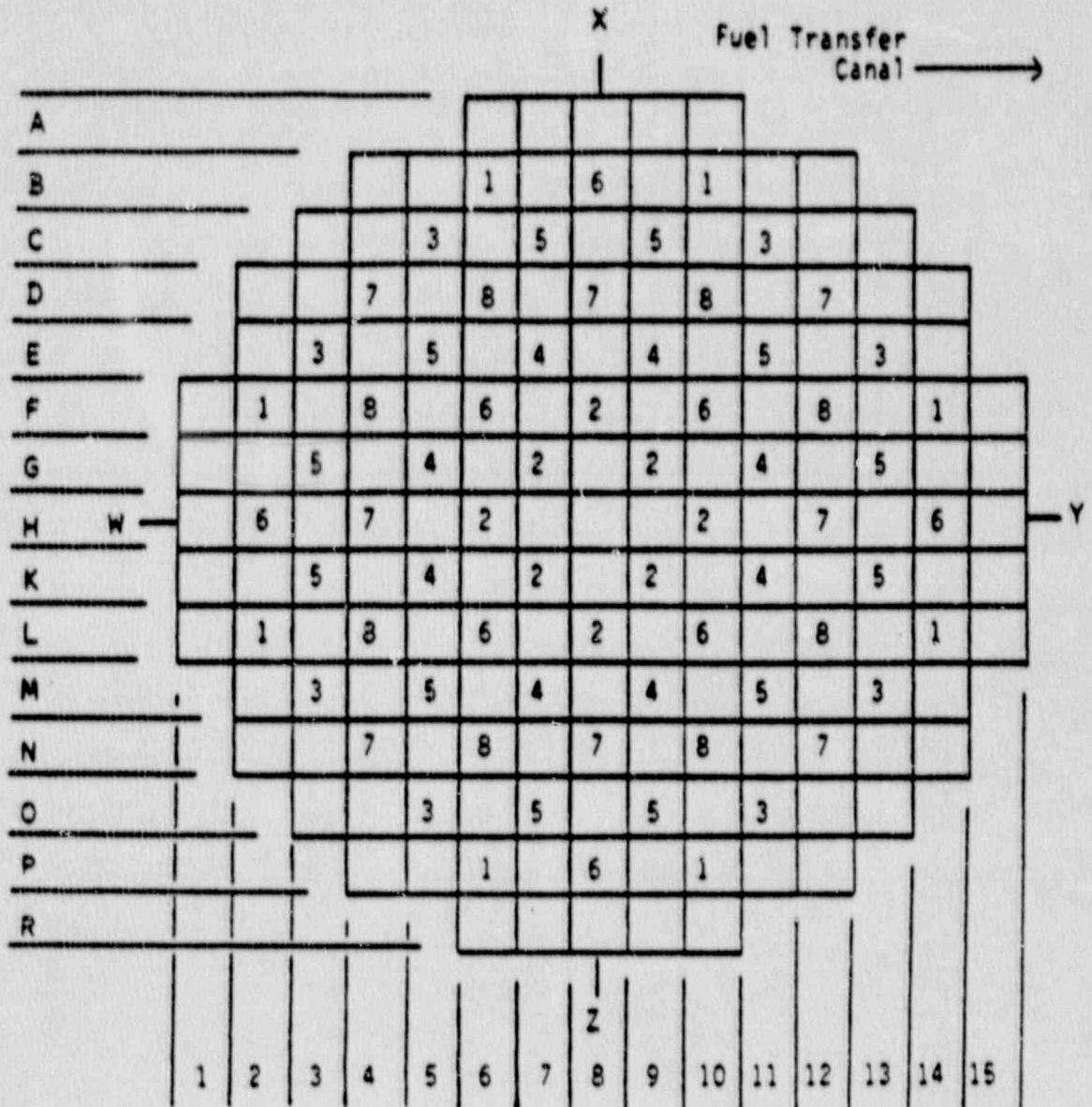
Reactivity is controlled by 60 full-length Ag-In-Cd control rods, 48 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight Inconel axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The cycle 10 locations of the 68 control rods and the group designations are indicated in Figure 3-3. The core locations and the group designations of the 68 control rods for cycle 10 are the same as those of the reference cycle. The locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-2. Enrichment and Burnup Distribution.
ANO-1 Cycle 10 off 420 EFPD Cycle 9

	8	9	10	11	12	13	14	15
H	3.19 20,770	3.45 17,838	3.35 32,917	3.45 13,461	3.35 32,925	3.45 13,460	3.45 17,280	3.45 17,806
K		3.35 22,295	3.49 0	3.35 26,775	3.49 0	3.35 22,788	3.49 0	3.45 17,146
L			3.35 34,293	3.49 0	3.35 25,520	3.49 0	3.45 14,967	3.45 17,274
M				3.35 33,075	3.49 0	3.45 17,588	3.45 17,564	
N					3.35 33,795	3.49 0	3.35 31,648	
O						3.35 22,288		
P								
R								

X.XX	Initial Enrichment,
XX,XXX	wt % U-235 BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations and Group Designations for ANO-1 Cycle 10



X Group Number

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

Figure 3-4. LBP Enrichment and Distribution,
ANO-1 Cycle 10

	8	9	10	11	12	13	14	15
H								
K			0.80		1.10		0.50	
L		0.80		1.10		1.10		
M			1.10		1.10			
N		1.10		1.10				
O			1.10					
P		0.50						
R								

X.XX

LBP Concentration,
wt % B₄C in Al₂O₃)

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for ANO-1, cycle 10 are listed in Table 4-1. Batch 6E and 10B are the Mark-B4 design. Batch 11 is the Mark-B6 design, and Batch 12 is the Mark-B8 design.

Mark-B8 Fuel Assembly Description:

The Mark-B8 fuel assembly is an improved Mark-B6 fuel assembly with design features to allow for high burnup, and to provide protection against debris fretting damage to the fuel rods. To provide for high burnups, the lower end fitting has been shortened by approximately 0.7 inches. The guide and instrument tubes were lengthened by the same amount, and the fuel rod lengthened by .4 inches. This provides additional gas volume and growth room for the fuel rod.

To protect against debris induced fretting failure of the fuel rod the following design changes were made. The lower end plug solid portion was extended in length. The lower spacer grid location was dropped so that the solid end plug extends through the lower spacer grid. The intention of the design is to trap any debris capable of fuel rod fretting below the bottom spacer grid where the solid lower end plug will prevent failure. A comparison of the design features of the various Mark-B type fuel rods is shown in Figure 4-1.

Forty Eight BPRAs will be used with the 56 batch 12 fuel assemblies.

4.2. Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed here.

4.2.1. Cladding Collapse

The operating power history for the most limiting fuel assembly was determined for each of the three previously burned fuel batches. The history for each batch was compared to that used in the generic creep collapse analysis. Batches 10B and 11 slightly exceeded the generic envelope. A new

envelope was formulated and new creep collapse analyses run. Both the new specific analyses and the generic analyses are based on the method of reference 2.

The analysis predicted a creep collapse life longer than 35,000 EFPH. This is longer than the maximum batch residence in cycle 10 which is 29,800 EFPH for batch 10B.

For Batch 12 the creep collapse analysis followed the method from reference 3. The operational conditions and mechanical characteristics of the batch 12 fuel assemblies was compared to an envelope formulated by BWFC (reference 3) and approved by the NRC (reference 4). All values of the Mark-B8 fuel assemblies are encompassed by the corresponding parameters of the limiting envelope. Some as-built data for the Mark-B8 assemblies (presently unavailable) were approximated from as-built values of past BWFC fuel and then compared to the limiting envelope. This is reasonable as the tolerances affecting these as-built values have not changed from past BWFC fuel designs. The creep collapse life of the batch 12 fuel rods based on reference 4 is 55,000 Mwd/mtU. This is greater than the maximum projected end of cycle burnup of 16,061 Mwd/mtU for batch 12.

4.2.2. Cladding Stress

The stress parameters for the fuel rod designs are enveloped by conservative fuel rod stress analyses. The same method was used for analysis of cycle 10 that had been used on the previous cycle. The stress margins are in excess of 11.2%.

4.2.3. Cladding Strain

The fuel rod design criteria specify a limit of 1% on cladding plastic tensile circumferential strain. The fuel pellet is designed to ensure that plastic cladding strain is less than 1% at design local pellet burnup and heat generation rate. The design values are higher than the worst-case values ANO-1 cycle 10 fuel is expected to experience. The strain analysis is conservatively based on enveloping the upper tolerance values for the fuel pellet diameter and density and the lower tolerance limit for cladding inside diameter.

4.3. Thermal Design

All fuel assemblies in the cycle 10 core are thermally similar. The design of the batch 12 Mark B8 assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. The analysis for all fuel was performed with the TACO2 code as described in reference 5. Nominal undensified input parameters used in the thermal analysis are presented in Table 4-2. Densification effects were accounted for in TACO2.

The results of the thermal design evaluation of the cycle 10 core are summarized in Table 4-2. Cycle 10 core protection limits are based on a linear heat rate (LHR) to centerline fuel melt limit of 20.5 kW/ft as determined by the TACO2 code.

The maximum fuel assembly burnup at EOC 10 is predicted to be less than 47,000 Mwd/mtU (batch 10B). 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 12 fuel assemblies is identical to those for previous fuel assemblies because no new materials were introduced in the batch 12 fuel assemblies.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. The accumulated operating experience for eight B&W 177 fuel assembly plants with Mark B fuel is shown in Table 4-3.

Table 4-1. - Fuel Design Parameters and Dimensions

Batch	6E	10B	11	12
Fuel assembly type	MK-B4	MK-B4	MK-B6	MK-B8
Number of assemblies	1	60	60	56
Fuel rod OD nominal, in.	0.430	0.430	0.430	0.430
Fuel Rod ID nominal in.	0.377	0.377	0.377	0.377
Undensified active fuel length, in.	142.25	141.8	141.8	141.8
Fuel pellet OD, (mean), in.	.3695	.3686	.3686	.3686
Fuel pellet initial density, (Nom), % TD	94.0	95.0	95.0	95.0
Initial fuel enrichment wt.% U ₂₃₅	3.19	3.35	3.45	3.49
Average burnup, EOC Mwd/mtU.	20,770	28,337	16,604	0
Maximum assembly burnup, EOC Mwd/mtU.	32,209	46,301	30,735	16,061
Exposure time, EOC EFPH.	28,700	29,800	19,200	9,100
Cladding collapse time, EFPH.	>35,000	>35,000	>35,000	NA
Cladding Collapse Burnup, Mwd/mtU.	NA	NA	NA	55,000

Table 4-2. Fuel Thermal Analysis Parameters

	<u>Batch 6E</u>	<u>Batch 10B</u>	<u>Batch 11</u>	<u>Batch 12</u>
No. of assemblies	1	60	60	56
Initial density, % TD	94	95	95	95
Initial pellet OD, in	0.3695	0.3686	0.3686	0.3686
Initial stack height, in	142.25	141.80	141.80	141.80
Enrichment, wt % U-235	3.19	3.35	3.45	3.49
Nominal linear heat rate at 2568 Mwt, kW/ft(a)	5.73	5.74	5.74	5.74

TACO2-based Predictions

Average fuel temperature

at nominal LHR, F (BOL) 1406 1400 1400 1400

Minimum LHR to melt, kW/ft 20.5 20.5 20.5 20.5

Core average LHR = 5.74 kW/ft

(a) Based on a nominal stack height

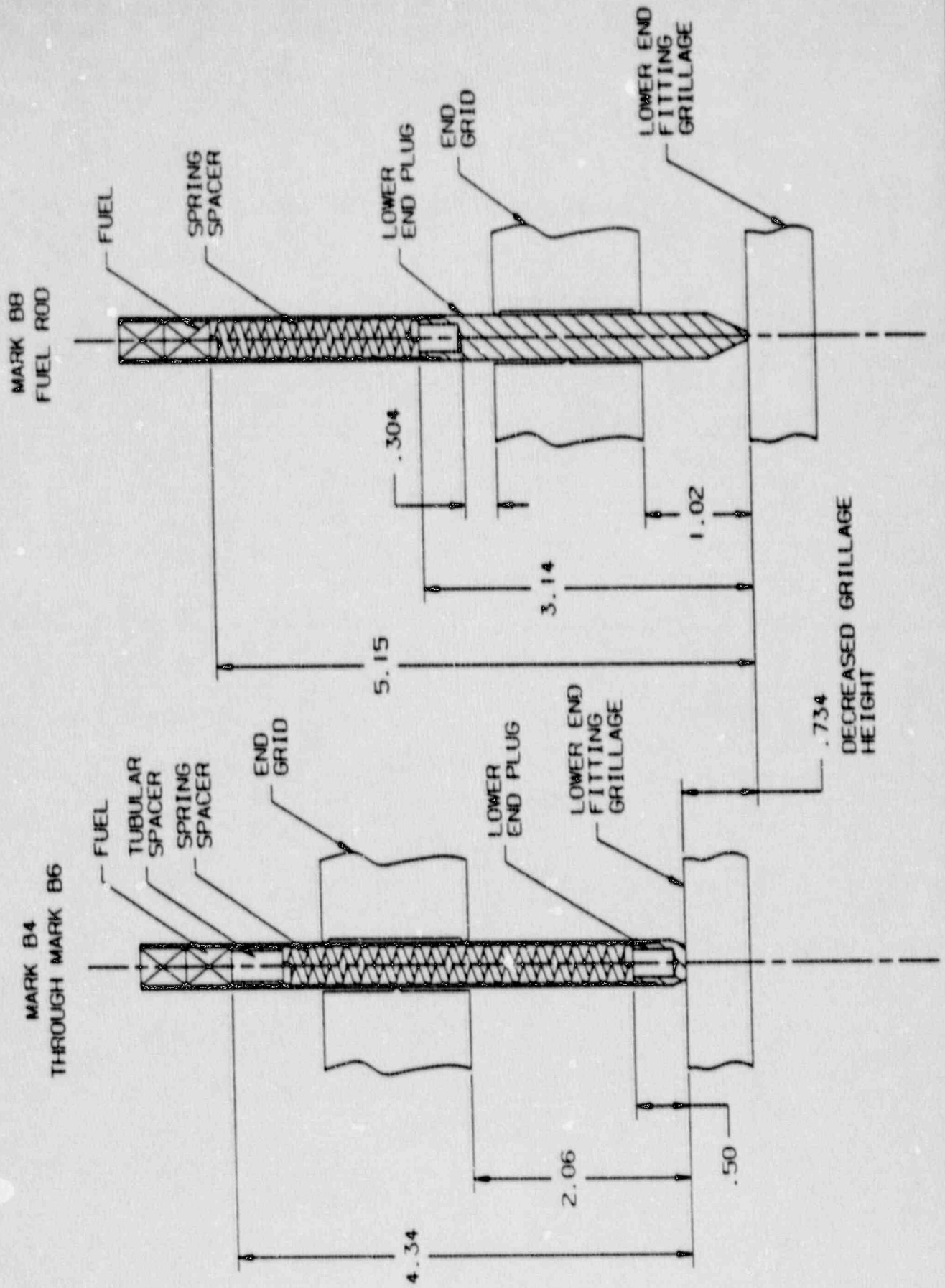
Table 4-3. Operating Experience

<u>Reactor</u>	<u>Current Cycle</u>	<u>Max. FA burnup, Mwd/mtU^(a)</u>		<u>Cumulative electric output, MWh^(b)</u>
		<u>Incore</u>	<u>Discharged</u>	
Oconee 1	12	40,595	58,310	84,346,419
Oconee 2	11	34,646	42,820	78,744,917
Oconee 3	12	35,594	42,740	78,231,767
Three Mile Island	7	33,966	33,863	47,186,490
Arkansas Nuclear One, Unit 1	9	34,972	57,318	63,712,046
Rancho Seco	7	34,123	38,268	43,215,399
Crystal River 3	7	38,793	35,350	50,831,632
Davis-Besse	6	33,690	40,300	38,787,158

(a) As of December 31, 1989.

(b) As of December 31, 1989.

FIGURE 4-1
 MARK B8 FUEL ROD DESIGN FEATURES



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 lists the core physics parameters of design cycles 9 and 10. The values for both cycles were calculated with the NOODLE code⁶. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 10 at full power with equilibrium xenon and nominal rod positions.

The differences in feed enrichment, BFRA loading, and shuffle pattern caused little change in the physics parameters between cycles 9 and 10. 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 worths for cycle 10 are less than for cycle 9 before APSR pull and greater at end of cycle. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 10 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 9 reload report.⁷

5.2. Analytical Input

The cycle 10 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

The only design change for cycle 10 is the shorter design cycle length.

The gray APSRs will be withdrawn from the core near the end of cycle 10 (335 EFPD) where the stability and control of the core in the feed-and-bleed mode with APSRs removed has been analyzed. The calculated stability index at 339

EFPD without APSRs is -0.066 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 9 and 10(a)

	Cycle 9 ^(b)	Cycle 10 ^(c)
Cycle length, EFPD	420	380
Cycle burnup, Mwd/mtU	13,143	11,892
Average core burnup - EOC, Mwd/mtU	27,271	27,244
Initial core loading, mtU	82.1	82.1
Critical boron - BOC, ppm (no Xe)		
HZP ^(d) , group 8 inserted	1552	1548
HFP, group 8 inserted	1379	1373
Critical boron - EOC, ppm (eq Xe)		
HZP, group 8 out	539 ^(e)	274
HFP, group 8 out	0 ^(f)	7
Control rod worths - HFP, BOC, % $\Delta k/k$		
Group 6	1.11	1.08
Group 7	0.98	0.88
Group 8 (maximum)	0.19	0.20
Control rod worths - HFP, EOC, % $\Delta k/k$		
Group 7	1.05	0.96
Max ejected rod worth (L-10) - HZP, % $\Delta k/k$		
BOC, groups 5-8 ins	0.35	0.29
335 EFPD ^(g) , groups 5-8 ins	0.41	0.29
EOC, groups 5-7 ins	0.41	0.33
Max stuck rod worth (M-13) ^(h) - HZP, % $\Delta k/k$		
BOC, groups 1-8 ins	1.49	1.30
335 EFPD ^(g) , groups 1-8 ins	1.47	1.40
EOC, groups 1-7 ins	1.42	1.45
Power deficit, HZP to HFP, % $\Delta k/k$		
BOC	1.60	1.61
EOC	2.35	2.31
Doppler coeff - HFP, 10^{-5} ($\Delta k/k/^\circ F$)		
BOC (no Xe)	-1.59	-1.59
EOC (eq Xe)	-1.86	-1.80

Table 5-1. (Cont'd) (a)

	Cycle 9 ^(b)	Cycle 10 ^(c)
Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^\circ F$)		
BOC (no Xe, crit ppm, group 8 ins)	-0.58	-0.60
EOC (eq Xe, 0 ppm, group 8 out)	-2.82	-2.81
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	130	129
EOC	111	111
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.56	2.55
EOC (equilibrium)	2.71	2.69
Effective delayed neutron fraction - HFP		
BOC	0.0062	0.0061
EOC	0.0052	0.0053

(a) Cycle 10 data are for the conditions stated in this report. The cycle 9 core conditions are identified in reference 7.

(b) Based on 440 EFPD at 2568 Mwt, cycle 8.

(c) Based on 420 EFPD at 2568 Mwt, cycle 9.

(d) HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (581F T_{avg}).

(e) Calculated with no xenon for cycle 9.

(f) At HFP conditions, 0 ppm occurs at 411 EFPD.

(g) Calculated at 360 EFPD for cycle 9.

(h) The maximum worth stuck rod was in location N-12 for cycle 9.

Table 5-2. Shutdown Margin Calculation for ANO-1, Cycle 10

	<u>BOC,</u> <u>% $\Delta k/k$</u>	<u>335 EFPD,</u> <u>% $\Delta k/k$</u>	<u>380 EFPD,</u> <u>% $\Delta k/k$</u>
<u>Available Rod Worth</u>			
Total rod worth, HZP	8.351	8.910	8.855
Worth reduction due to poison material burnup	-0.100	-0.100	-0.100
Maximum stuck rod, HZP	<u>-1.303</u>	<u>-1.398</u>	<u>-1.447</u>
Net Worth	6.948	7.412	7.308
Less 10% uncertainty	<u>-0.695</u>	<u>-0.741</u>	<u>-0.731</u>
Total available worth	6.253	6.671	6.577
<u>Required Rod Worth</u>			
Power deficit, HFP to HZP	1.605	2.239	2.308
Allowable inserted rod worth	0.246	0.370	0.365
Flux redistribution	<u>0.381</u>	<u>0.605</u>	<u>0.561</u>
Total required worth	2.232	3.214	3.234
Shutdown margin (total available worth minus total required worth)	4.021	3.457	3.343

NOTE: The required shutdown margin is 1.00% $\Delta k/k$.

Figure 5-1. ANO-1 Cycle 10, BOC (4 EFPD) Two-Dimensional Relative Power Distribution -- Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.99	1.12	0.95	1.19	0.98	1.25	1.11	0.61
K		1.07	1.29	1.06	1.30	1.14	1.20	0.59
L			0.99	1.29	1.09	1.29	0.96	0.44
M				0.99	1.28	1.11	0.67	
N					0.92	1.11	0.35	
O						0.50		
P								
R								

8	Inserted Rod group No. Relative Power Density
X.XX	

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 10 operation utilized the methods and models described in references 7, 8 and 9 as supplemented by reference 10, which implements the BWC (reference 11) CHF correlation for analysis of the Zircaloy grid fuel assembly. The incoming batch 12 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles.

Cycle 10 is the second cycle in the transition from the Mark-B Inconel-grid fuel design to the Mark-BZ, Zircaloy grid fuel design. The cycle 10 core is comprised of 116 Zircaloy grid fuel assemblies, 48 of which contain BPRAs, and 61 Mark B fuel assemblies. Thirteen (13) of the Mark-B fuel assemblies and 48 of the Zircaloy grid fuel assemblies have open, or unplugged, control rod guide tubes. The core bypass flow, which is dependent on the number of open control rod guide tubes, is 8.5% for this configuration. The Zircaloy grid fuel assemblies exhibit a slightly higher pressure drop than the Mark-B assemblies. This tends to cause some coolant flow diversion from the Zircaloy grid assemblies to the Mark-B fuel and creates the need to consider a "transition core penalty."

The reference analysis for cycle 10 thermal-hydraulic design is the same as that used for cycle 9 and considers a full core of Zircaloy grid fuel assemblies with a bypass flow of 8.8%. A cycle-specific analysis, which modeled the actual cycle 10 core configuration and bypass flow value, has been performed to demonstrate that the reference analysis remains applicable and a transition core penalty is not necessary. Table 6-1 provides a summary of the DNB analysis parameters for cycles 9 and 10.

No rod bow penalty was considered in the cycle 10 analysis as justified by reference 12.

Table 6-1. Maximum Design Conditions, Cycles 9 and 10

	<u>Cycle 9</u>	<u>Cycle 10</u>
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, gpm	374880	374880
Core bypass flow, % (a)	8.8	8.8
DNER modeling	Crossflow	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.55 cosine	1.65 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.97	0.97
Active fuel length, in. (b)	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	BWC	BWC
CHF correlation DNB limit	1.18	1.18
Minimum DNER		
at 112% power	1.77	1.77
at 102% power	2.01	2.01

(a) Used in the analysis.

(b) Cold nominal stack height.

(c) This represents initial condition DNER for accident analyses.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 10 parameters to determine the effect of the cycle 10 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effect of fuel densification on the FSAR accident results has been evaluated and are reported in reference 13. Since batch 12 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference report, the conclusions in that reference are still valid.

The radiological dose consequences of the accidents presented in Chapter 14 of the updated FSAR were re-evaluated for this reload report except for the waste gas tank rupture. The waste gas tank rupture was not reevaluated since Technical Specification 3.25.2.5 controls the maximum tank inventory on the basis of Xe-133 equivalent curie content such that the analysis of the event is not cycle dependant. The evaluation of the remaining events was made in order to incorporate more current plant data as well as the information in the updated FSAR.

All of the cycle 10 accident doses are based on radionuclide sources calculated for the actual cycle 10 core design and irradiation history. Table 7-1 shows a comparison between cycle 9 and cycle 10 doses for the Chapter 14 accidents that result in significant offsite doses. The difference between cycle 9 and cycle 10 LOCA and MHA doses resulted from the adoption of updated iodine species fractions per Reg Guide 1.4 Rev. 2 and inclusion of effects of throttled reactor building spray.

The radiological doses from all of the accidents evaluated with the specific nuclide inventory from cycle 10 are lower than the NRC acceptance criteria of NUREG-0800, and thus are within acceptable limits.

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.

The core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Thermal parameters for fuel batches 6E, 10B, 11, and 12 are given in Table 4-2. The cycle 10 thermal-hydraulic maximum design conditions are compared with the previous cycle 9 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 10 are compared in Table 7-2.

A generic loss-of-coolant accident (LOCA) analysis has been performed for the B&W 177-FA lowered loop nuclear steam supply (NSS) system using the Final Acceptance Criteria Emergency Core Cooling System (ECCS) Evaluation Model (reported in BAW-10103A, Rev. 3¹⁴), updated with an upgraded fuel performance model (reported in BAW-1775¹⁵) and the B&W modified version of FLECSET (reported in BAW-1915PA¹⁶ and BAW-10104PA, Rev. 5¹⁷). These analyses are generic, since the limiting values of key parameters for all the B&W plants in this category were used. Furthermore, the combination of average fuel temperatures as a function of linear heat rate and lifetime pin pressure data used in the generic LOCA linear heat rate limits analysis is conservative compared to those calculated for this reload. Table 7-2 shows the bounding values for maximum allowable LOCA linear heat rate limits for Arkansas Nuclear One - Unit 1 (ANO-1) cycle 10 fuel as a function of burnup. The LOCA linear heat rate limits for beginning of cycle operation include the combined effects of the NUREG-0630 cladding swell and rupture model, use of the B&W CHF correlation, reduced fuel rod prepressure, and implementation of the B&W modified version of FLECSET. In order to improve the calculated peak clad temperature margin to the 10CFR 50.46 limit of 2200°F, at the six foot core elevation, the LOCA linear heat rate limit was reduced to 16.1 kW/ft at the beginning of cycle. The end of cycle LOCA linear heat rate limit was also reduced to 16.1 kW/ft. This change was based on the peak clad temperature behavior as a function of burnup for the ruptured and un-ruptured nodes as shown in BAW-1775.

It is concluded from the examination of cycle 10 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 10. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 10 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 10 are bounded by the FSAR, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of Cycle 9 and Cycle 10 Accident Doses

	Cycle 9 doses, <u>Rem</u>	Cycle 10 doses, <u>Rem</u>
<u>Fuel Handling Accident</u>		
Thyroid dose at EAB (2 h)	1.12	1.13
Whole body dose at EAB (2 h)	0.22	0.22
<u>Steam Line Break</u>		
Thyroid dose at EAB (2 h)	1.82	1.80
Whole body dose at EAB (2 h)	0.01	0.01
<u>Steam Generator Tube Failure</u>		
Thyroid dose at EAB (2 h)	6.53	6.45
Whole body dose at EAB (2 h)	0.56	0.55
<u>Control Rod Ejection Accident</u>		
Thyroid dose at EAB (2 h)	7.02	7.02
Whole body dose at EAB (2 h)	0.006	0.006
Thyroid dose at LPZ (30 d)	5.64	5.64
Whole body dose at LPZ (30 d)	0.005	0.005
<u>Loss of Cooling Accident (a)</u>		
Thyroid dose at EAB (2 h)	4.22	2.32
Whole body dose at EAB (2 h)	0.03	0.03
Thyroid dose at LPZ (30 d)	2.47	1.09
Whole body dose at LPZ (30 d)	0.02	0.02
<u>Maximum Hypothetical Accident (a)</u>		
Thyroid dose at EAB (2 h)	165.1	91.5
Whole body dose at EAB (2 h)	5.03	4.85
Thyroid dose at LPZ (30 d)	87.8	39.4
Whole body dose at LPZ (30 d)	1.78	1.70

(a) The cycle 10 doses were calculated using revised reactor building spray data.

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

<u>Parameter</u>	<u>FSAR and Densification Report Value</u>	<u>WNO-1 Cycle 10</u>
Doppler coeff (BOC), $10^{-4} \Delta k/k/^\circ F$	-0.117	-0.159
Doppler coeff (EOC), $10^{-4} \Delta k/k/^\circ F$	-0.130	-0.180
Moderator coeff (BOC), $10^{-4} \Delta k/k/^\circ F$	0.0(a) (b)	-0.60
Moderator coeff (EOC), $10^{-4} \Delta k/k/^\circ F$	-4.0(c)	-2.81
All-rod group worth (HZP), $\% \Delta k/k$	12.90	8.38
Initial boron concentration (HFP), ppm	1150	1373
Boron reactivity worth (HFP), ppm/ $\% \Delta k/k$	100	129
Max. ejected rod worth (HFP), $\% \Delta k/k$	0.65	0.20
Dropped rod worth (HFP), $\% \Delta k/k$	0.65	≤ 0.20

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

(b) Transient results have been shown to be acceptable with a value of $+0.9 \times 10^{-4} \Delta k/k/^\circ F$.

(c) $-3.0 \times 10^{-4} \Delta k/k/^\circ F$ was used for the steam line failure analysis.

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

<u>Core elevation, ft</u>	<u>Allowable peak LHR, 0-1000 Mwd/mtU, kW/ft</u>	<u>Allowable peak LHR, after 1000 Mwd/mtU, kW/ft</u>
2	14.5	15.5
4	16.1	16.6
6	16.1	16.1
8	17.0	17.0
10	16.0	16.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 10 operation for changes in core reactivity, power peaking and control rod worths. The cycle 10 design analysis basis includes the impact of extended low-power operation at 80% of rated power for cycle 9. The cycle 10 basis also includes a very low leakage fuel cycle design, a mixed Mark B4/Mark B6/Mark B8 fuel assembly core, gray APSRs, and crossflow analysis. The LOCA linear heat rate limits used to develop the Technical Specification Limiting Conditions for Operation include the combined effects of the NUREG-0630 cladding swell and rupture model, use of the BWC CHF correlation, reduced fuel rod pre-pressure, and implementation of the B&W modified version of FLECSET.^{14,15,16,17}

A cycle 10 specific analysis was conducted to generate Technical Specification Limiting Conditions for Operation (rod index, axial power imbalance, and quadrant tilt), based on the methodology described in reference 18. The effects of gray APSR repositioning were included in the analysis. The burnup-dependent allowable LOCA linear heat rate limits used in the analysis are provided in Figure 8-15. The analysis also determined that the cycle 10 Technical Specifications provide protection for the overpower condition that could occur during an overcooling transient because of nuclear instrumentation errors, and verified removal of the power level cutoff hold requirement.

Technical Specification section 3.5.2.4 was revised to accommodate a change in the quadrant tilt setpoint, based on incore detector sensitivity depletion.¹⁹ The full incore quadrant tilt setpoint reported in section 3.5.2.4 is the bounding value, derived from the detector sensitivity depletion at end-of-cycle 10, of the full incore quadrant tilt setpoint values determined for cycle 10. The measurement system-independent rod position and axial power imbalance limits determined by the cycle 10 analysis were error-adjusted to generate alarm setpoints for power operation and are reflected in a Technical Specification revision to sections 3.5.2.5 and 3.5.2.6. The alarm setpoints are provided in Figures 8-3 through 8-14.

Based on the analyses and Technical Specification revisions described 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 the Technical Specifications.

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

- 3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is $\geq 95\%$ of rated thermal power and shall be less positive than $0.9 \times 10^{-4} \Delta k/k/^{\circ}F$ whenever thermal power is $< 95\%$ of rated thermal power and the reactor is not shutdown.
- 3.1.7.2 The MTC shall be determined to be within its limit by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.
- 3.1.7.3 With the MTC outside any one of the above limits, be in at least Hot Standby within 6 hours.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4} \Delta k/k/^{\circ}F$ corrected to 95% of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4} \Delta k/k/^{\circ}F$.

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

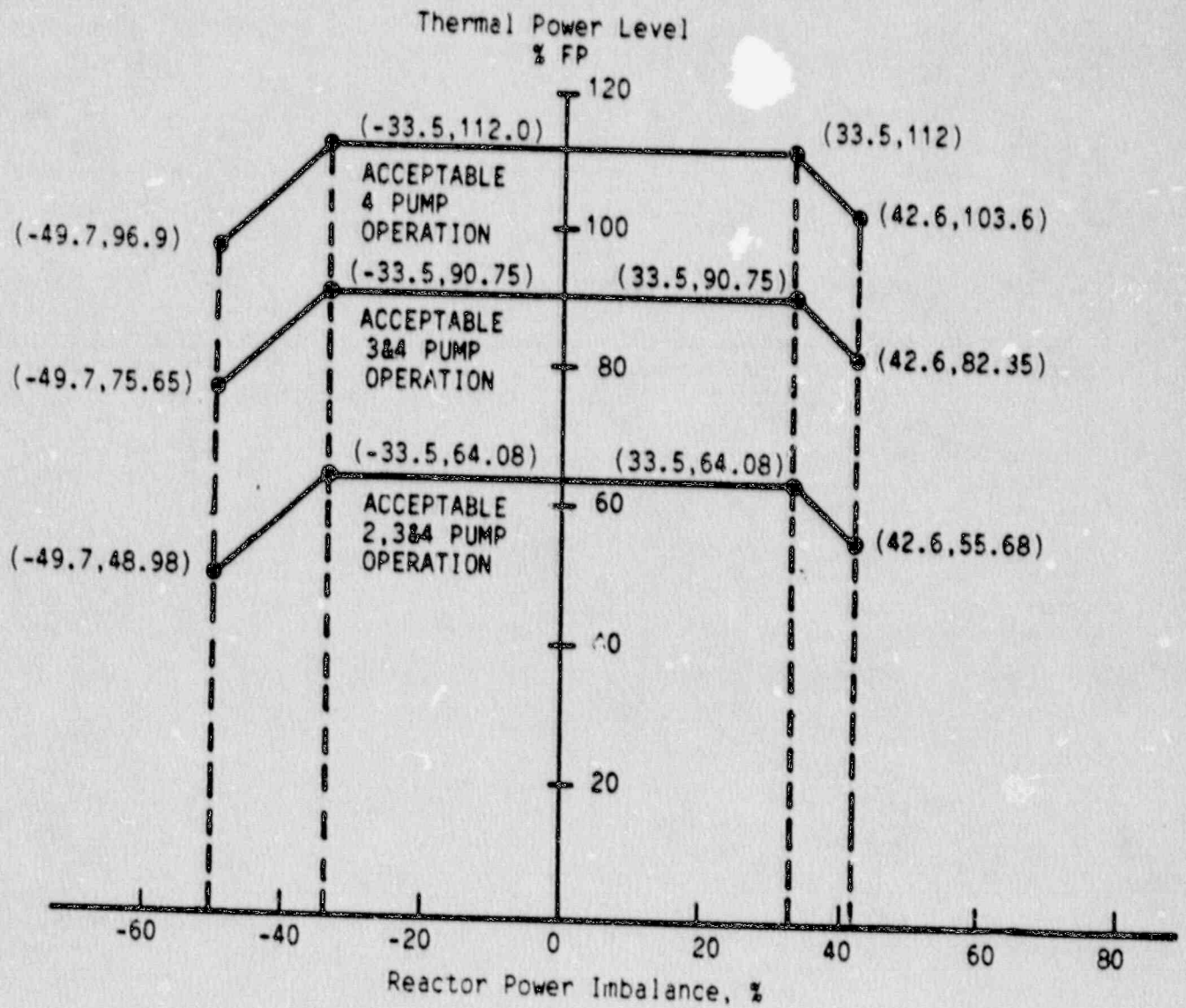
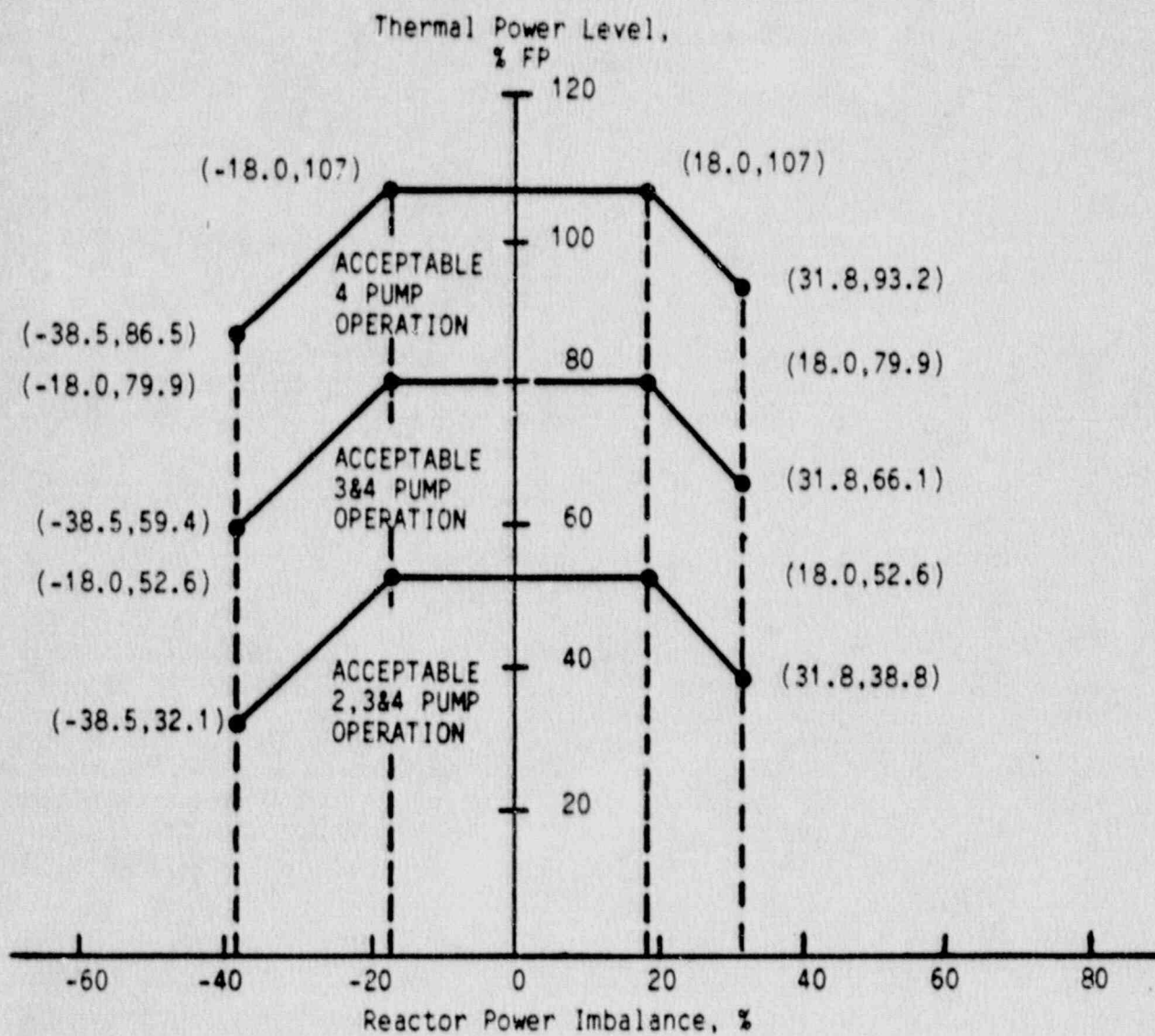


Figure 8-2. Protective System Maximum Allowable Setpoints -- ANO-1
 (Tech Spec Figure 2.3-2)



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 percent 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 4.24%, 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 4.24%.
2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4.24% 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 4.24%.
 - c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 4.24%.
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. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1(A-C), 3.5.2-2(A-C), and 3.5.2-3(A-C) for 4, 3 and 2 pump operation respectively. If the applicable 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.

4. Except for physics tests or exercising axial power shaping rods (APSRs), the following limits apply to APSR position:

Up to 345 EFPD, the APSRs may be positioned as necessary for transient imbalance control, however, the APSRs shall be fully withdrawn by 345 EFPD. After 345 EFPD, the APSRs shall not be reinserted.

With the APSRs inserted after 345 EFPD, corrective measures shall be taken immediately to achieve the full withdrawn position. Acceptable APSR 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 3.5.2-4(A-C). If the imbalance is not within the envelope defined by Figure 3.5.2-4(A-C), 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 3.5.2-4(A-C) is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria. 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

Figure 8-3. Rod Position Setpoints for Four-Pump Operation from 0 to 30 +10/-0 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-1A)

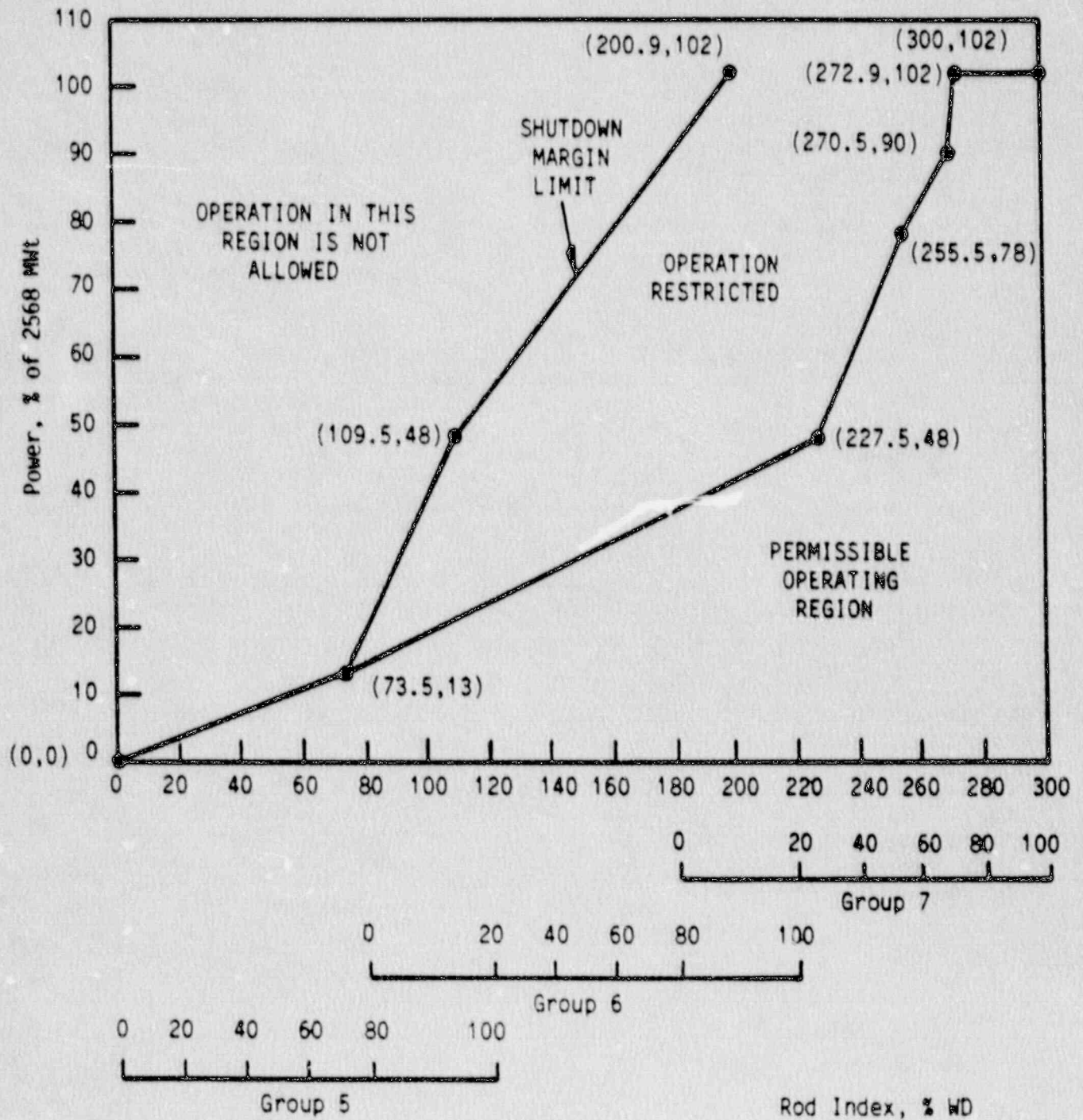


Figure 8-4. Rod Position Setpoints for Four-Pump Operation From 30 +10/-0 to 335 ± 10 EPPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-1B)

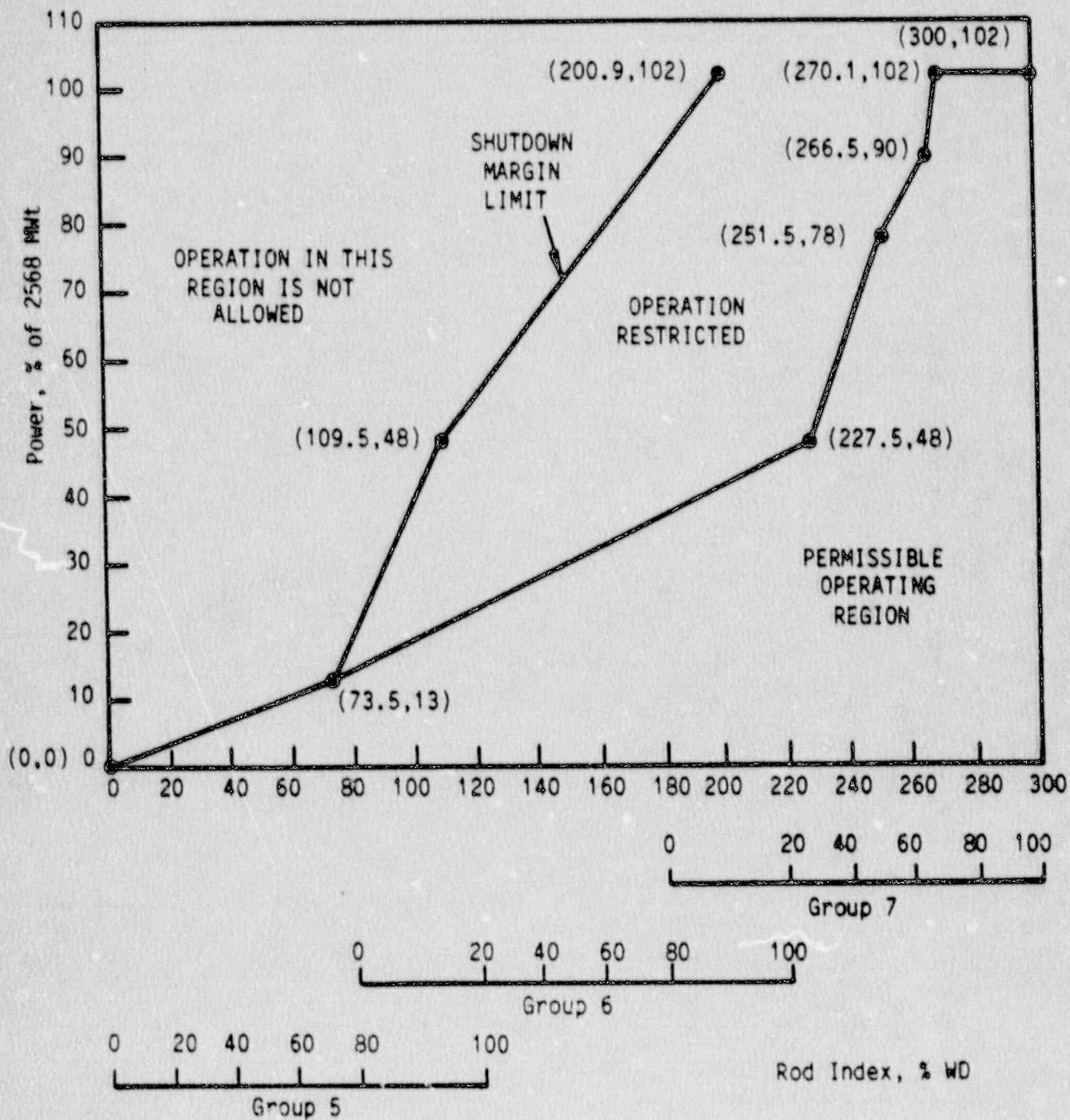


Figure 8-5. Rod Position Setpoints for Four-Pump Operation After 335 ± 10 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-1C)

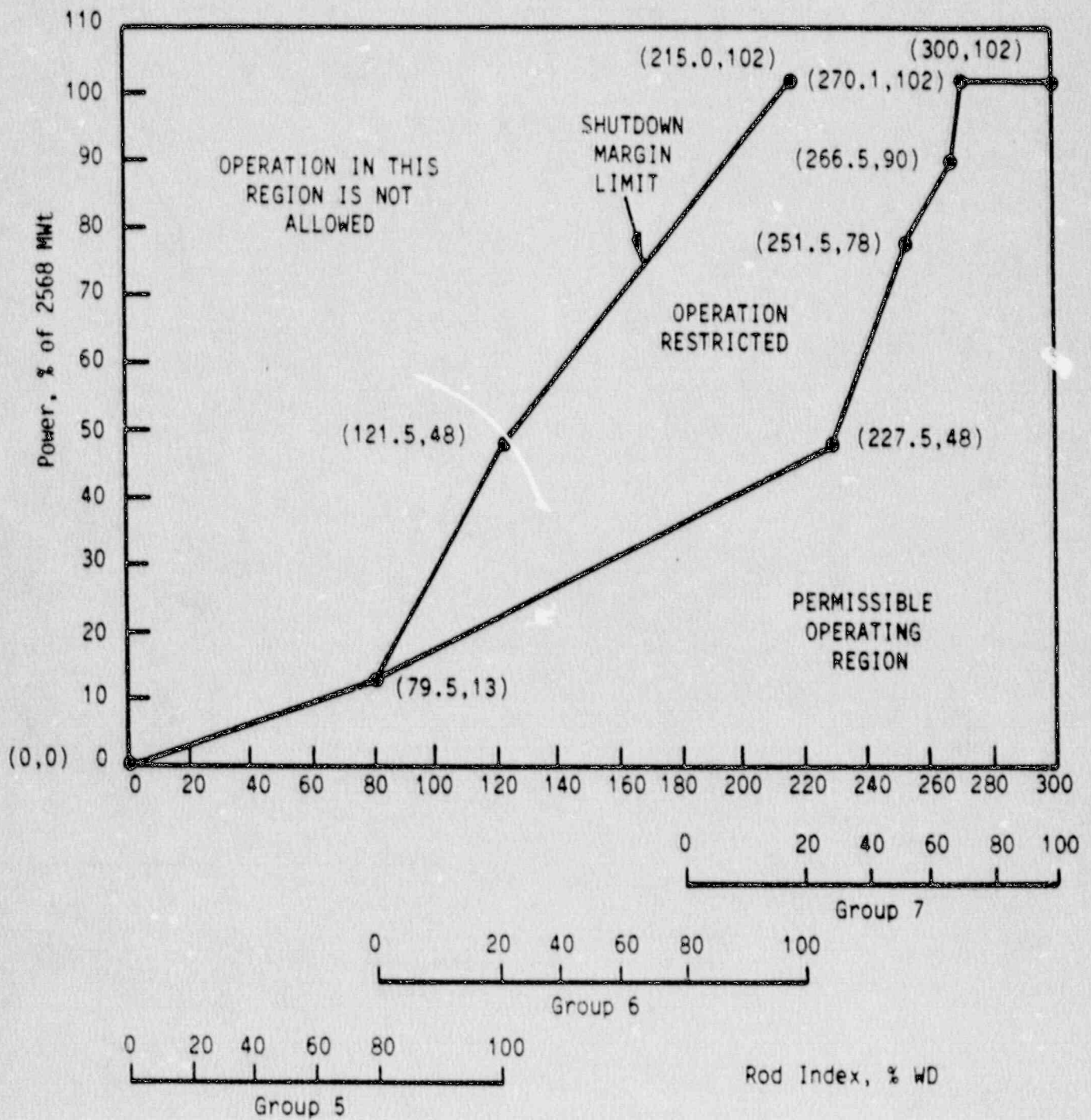


Figure 8-6. Rod Position Setpoints for Three-Pump Operation From 0 to 30 +10/-0 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-2A)

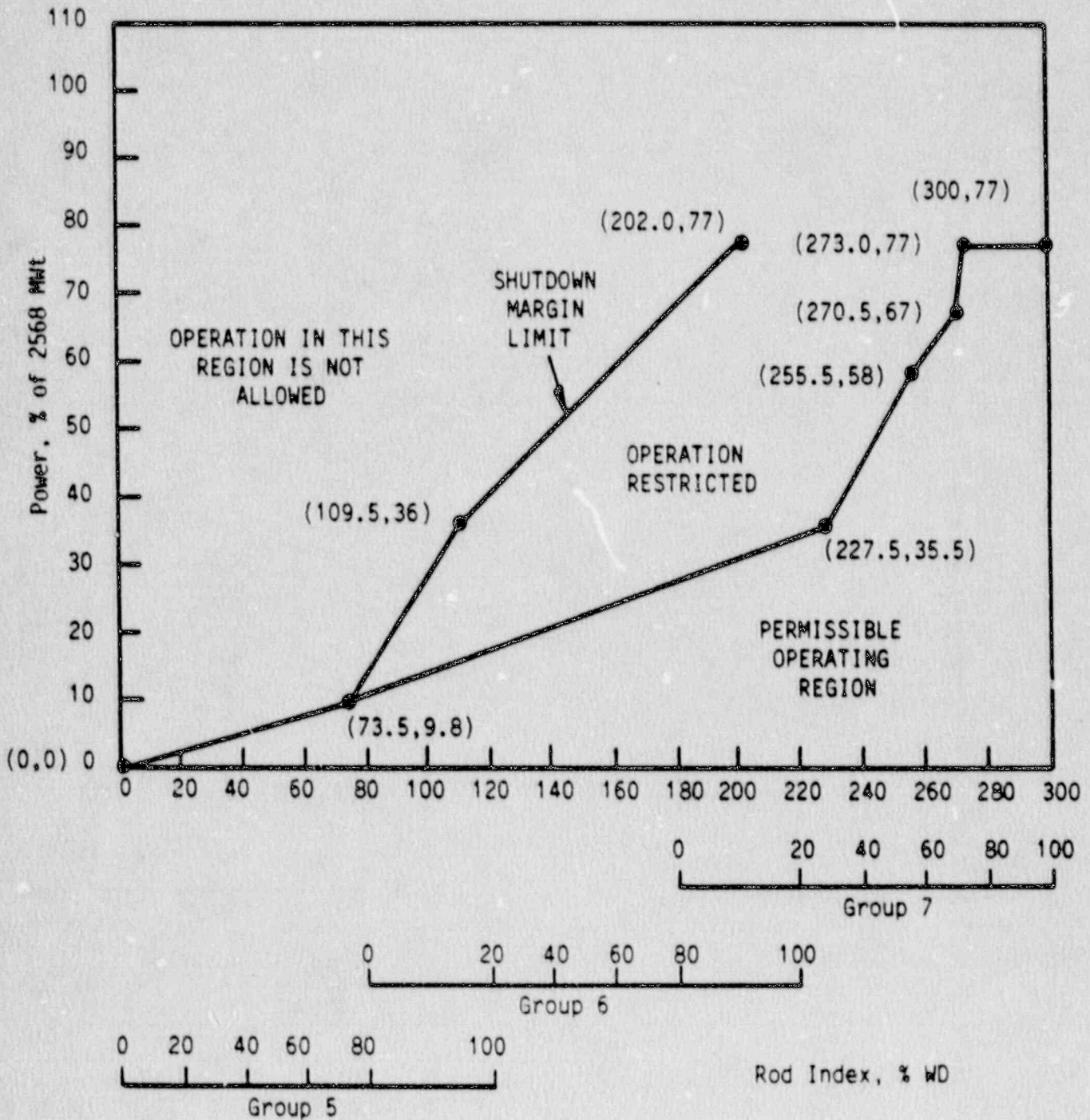


Figure 8-7. Rod Position Setpoints for Three-Pump Operation From 30 +10/-0 to 335 ±10 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-2B)

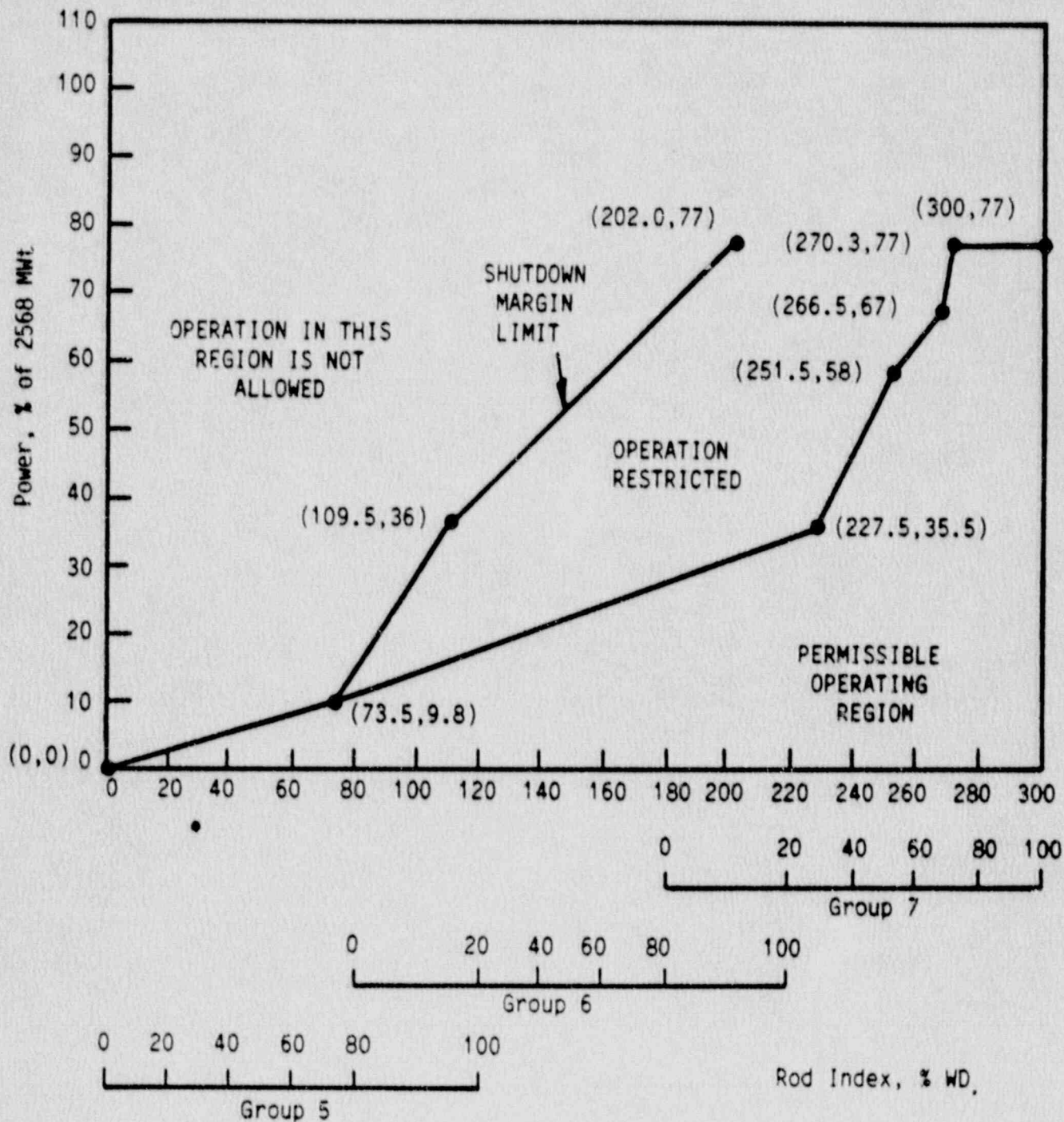


Figure 8-8. Rod Position Setpoints for Three-Pump Operation After 335 ± 10 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-2c)

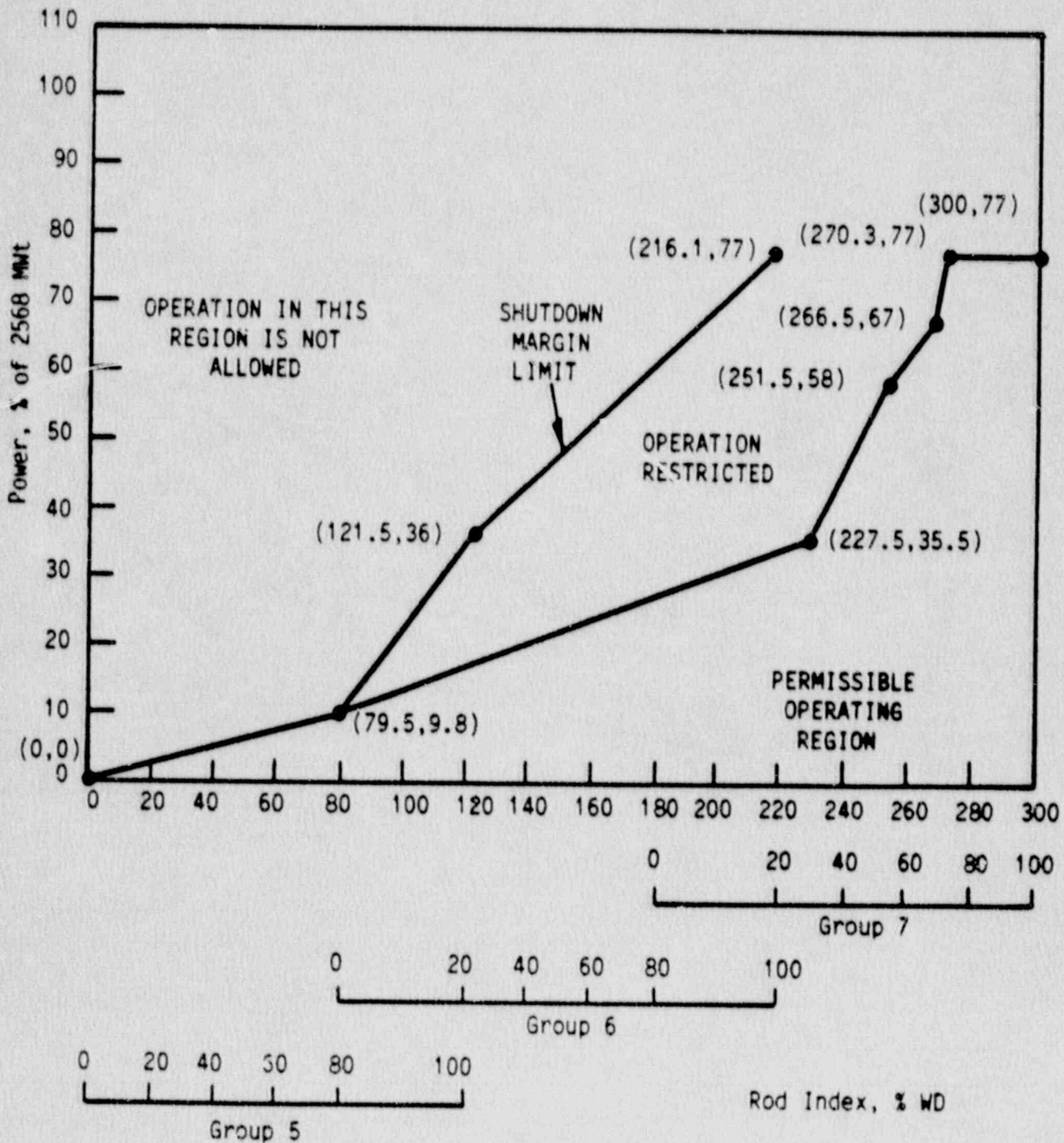


Figure 8-9. Rod Position Setpoints for Two-Pump Operation From 0 to 30 +10/-0 EFPD-- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-3A)

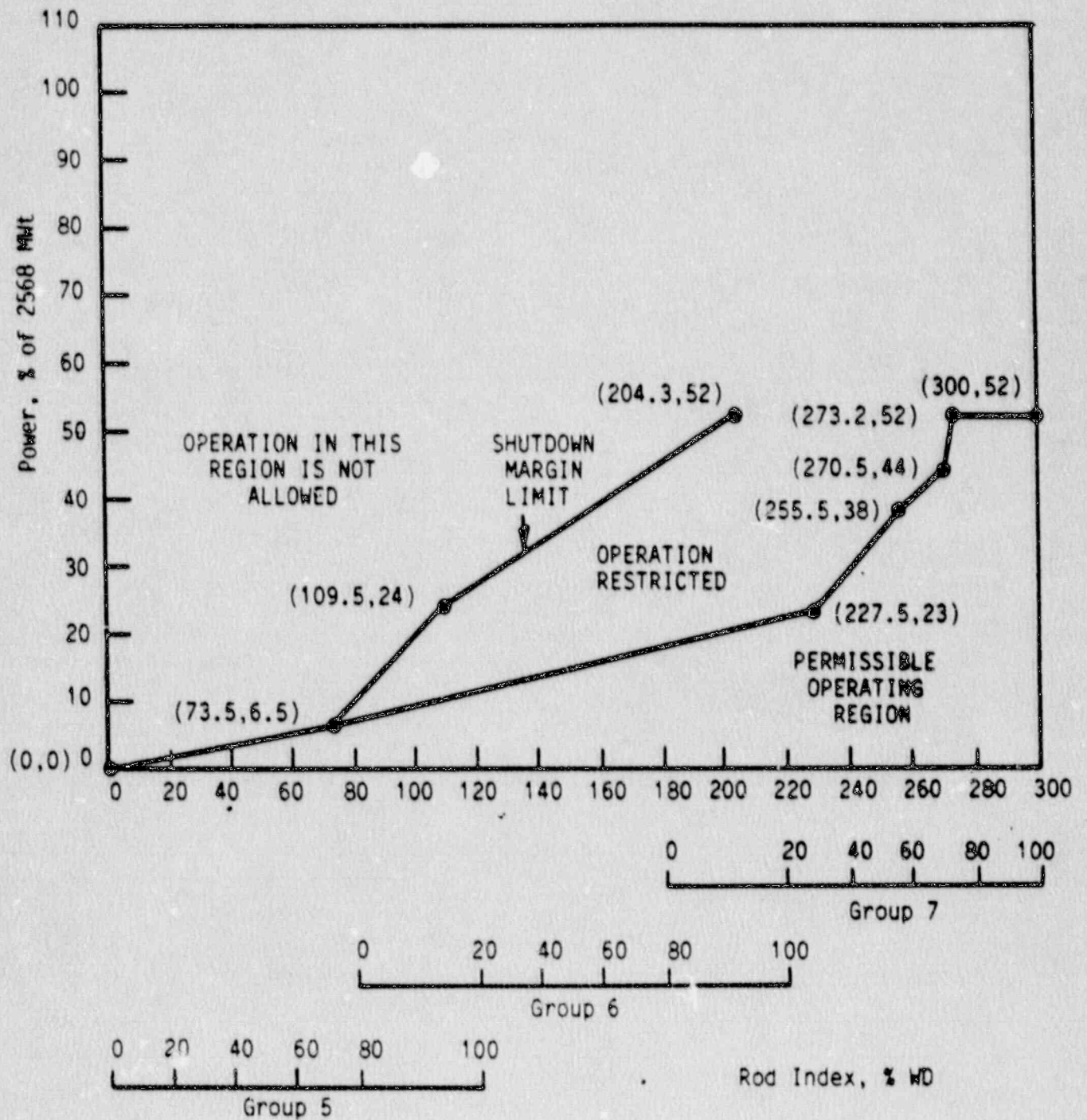


Figure 8-10. Rod Position Setpoints for Two-Pump Operation From $30 \pm 10/-0$ to 335 ± 10 EFPD — ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-3B)

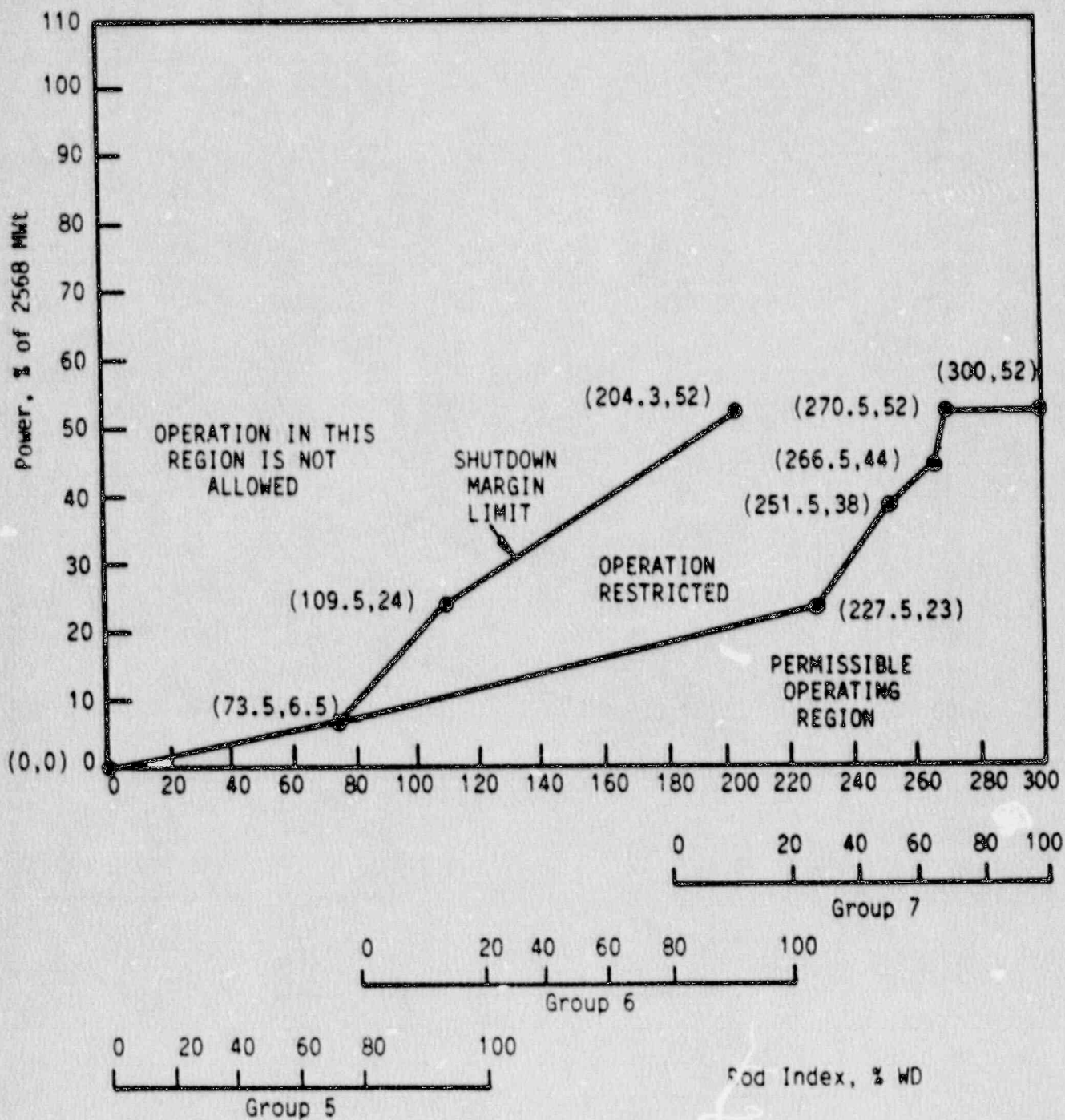


Figure 8-11. Rod Position Setpoints for Two-Pump Operation After 395 ± 10 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-3C)

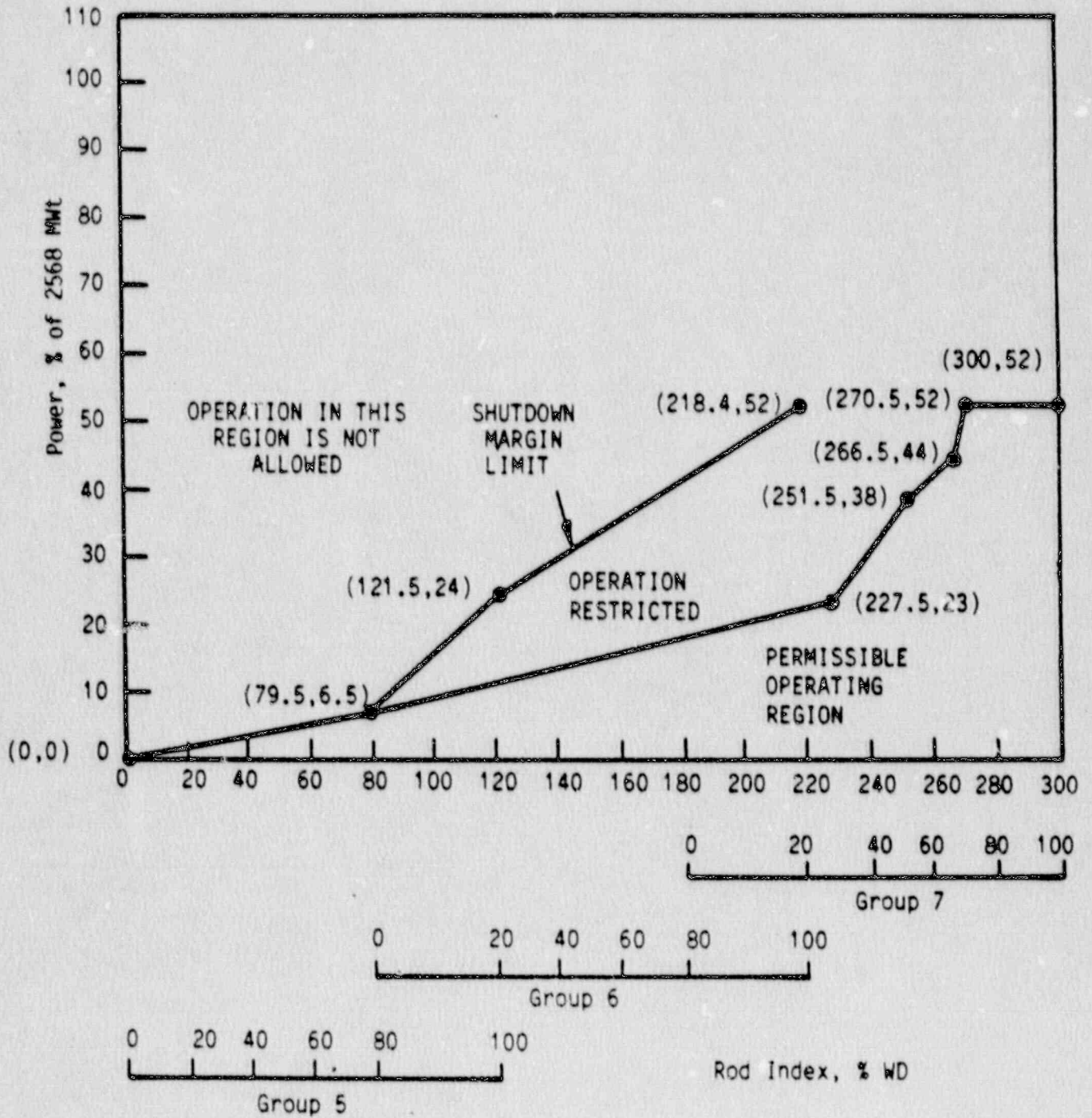


Figure 8-12. Operational Power Imbalance Setpoints for Operation From 0 to 30 +10/-0 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-4A)

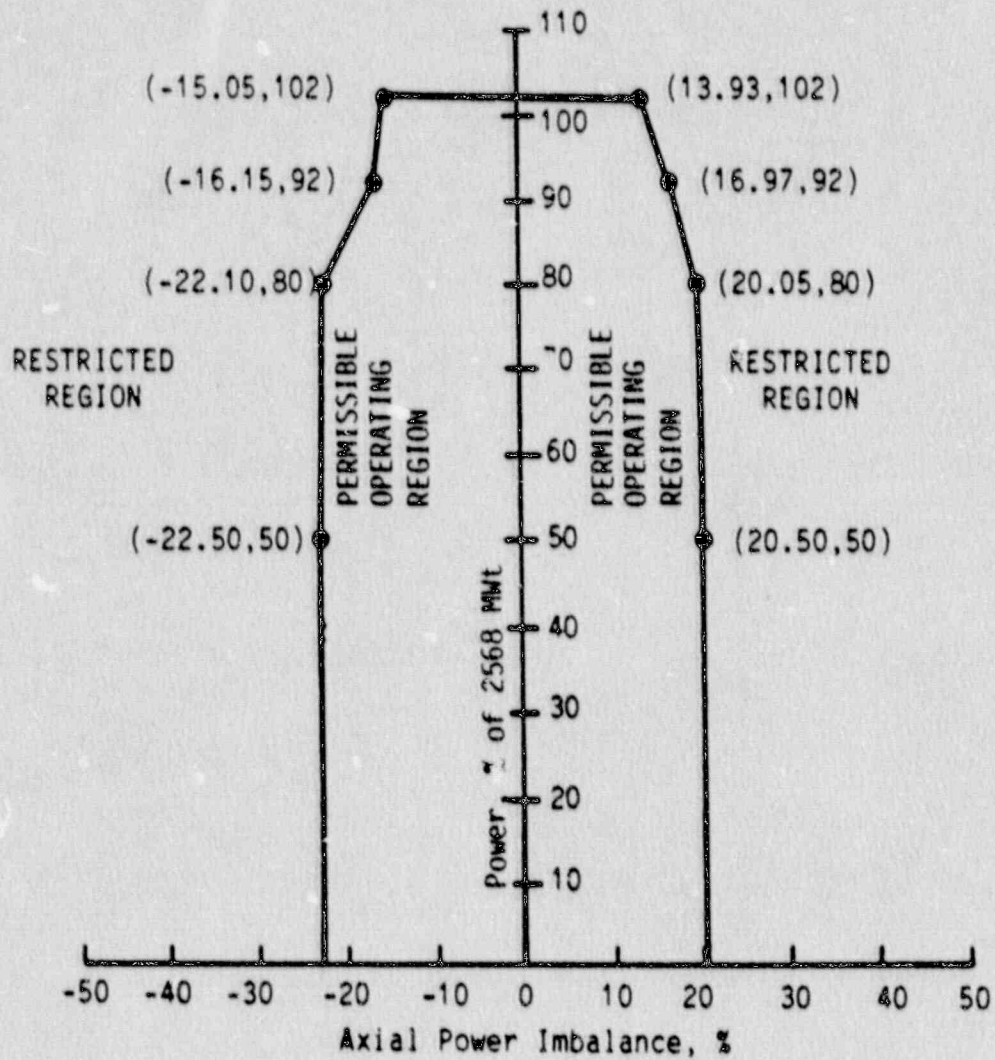


Figure 8-13. Operational Power Imbalance Setpoints for Operation From
 $30 \pm 10/-0$ to 335 ± 10 EFPD -- ANO-1 CYCLE 10
 (Tech Spec Figure 3.5.2-4B)

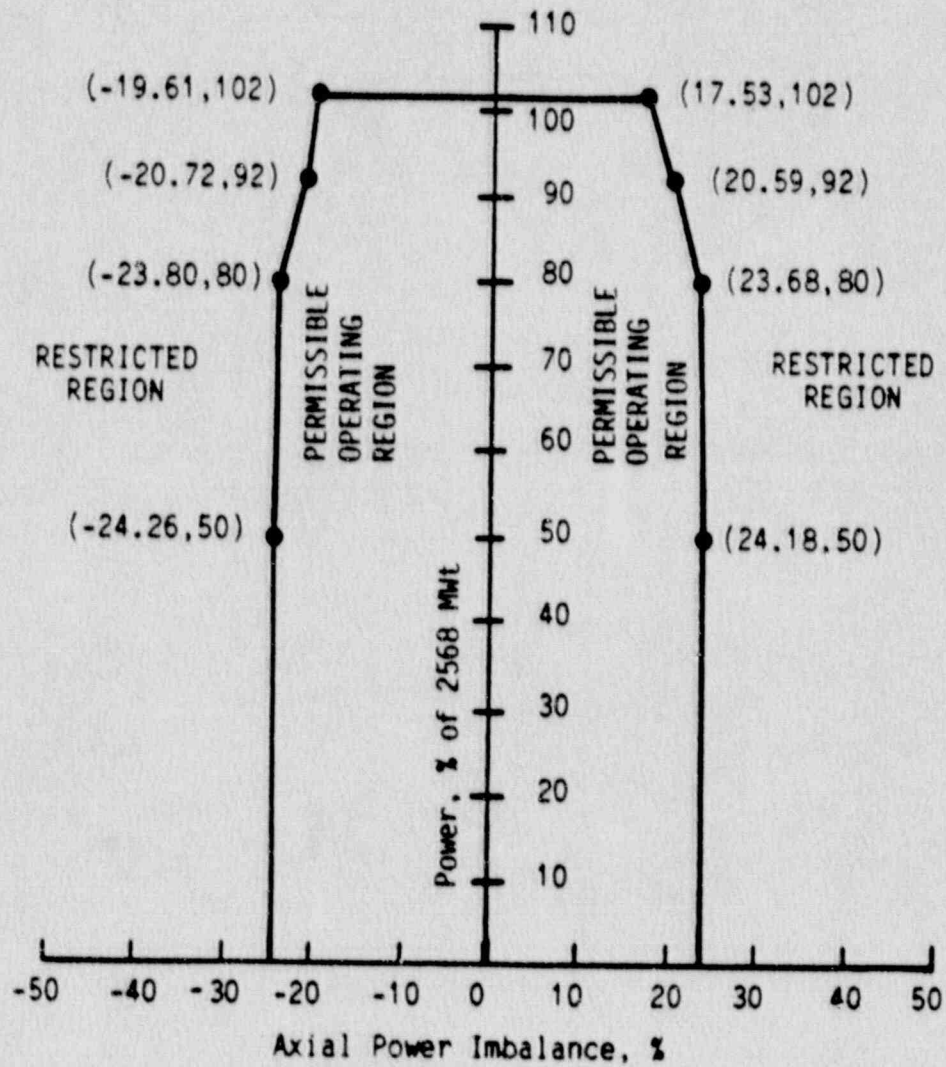


Figure 8-14. Operational Power Imbalance Setpoints for Operation After 335 ±10 EFPD -- ANO-1 Cycle 10 (Tech Spec Figure 3.5.2-4C)

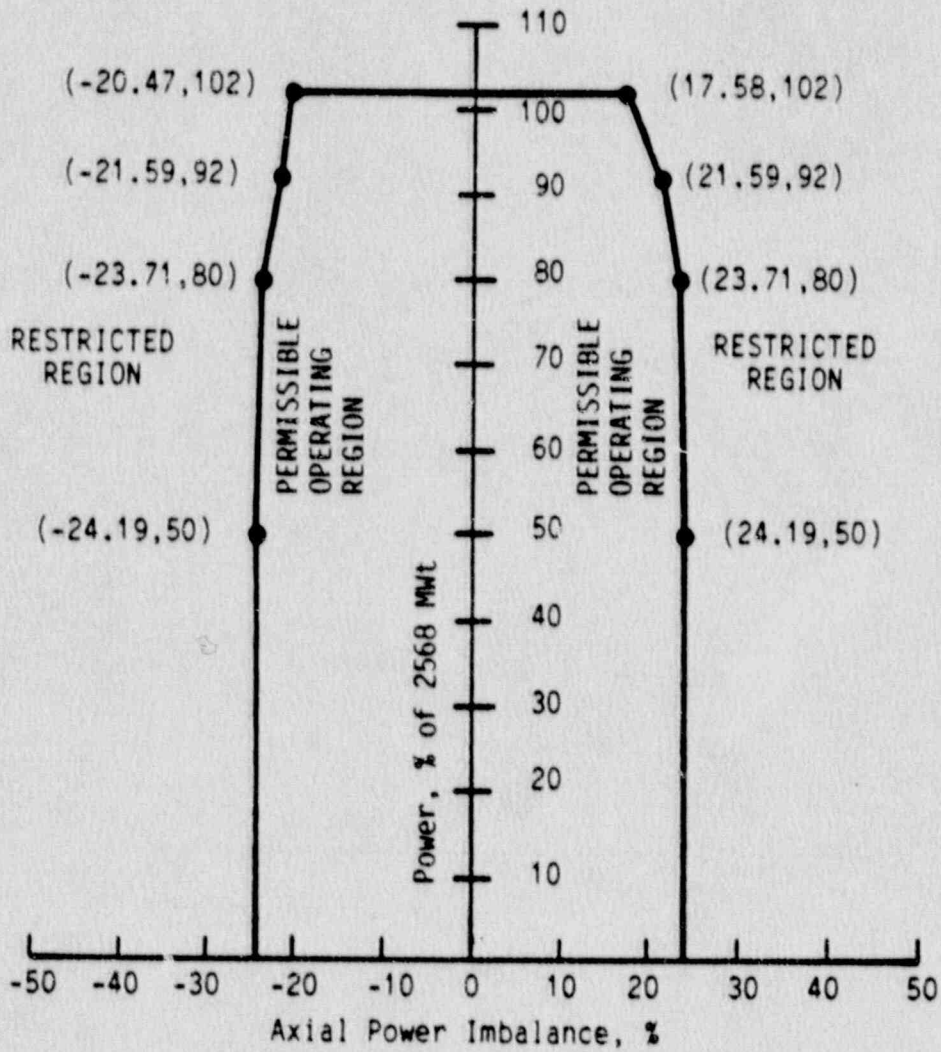
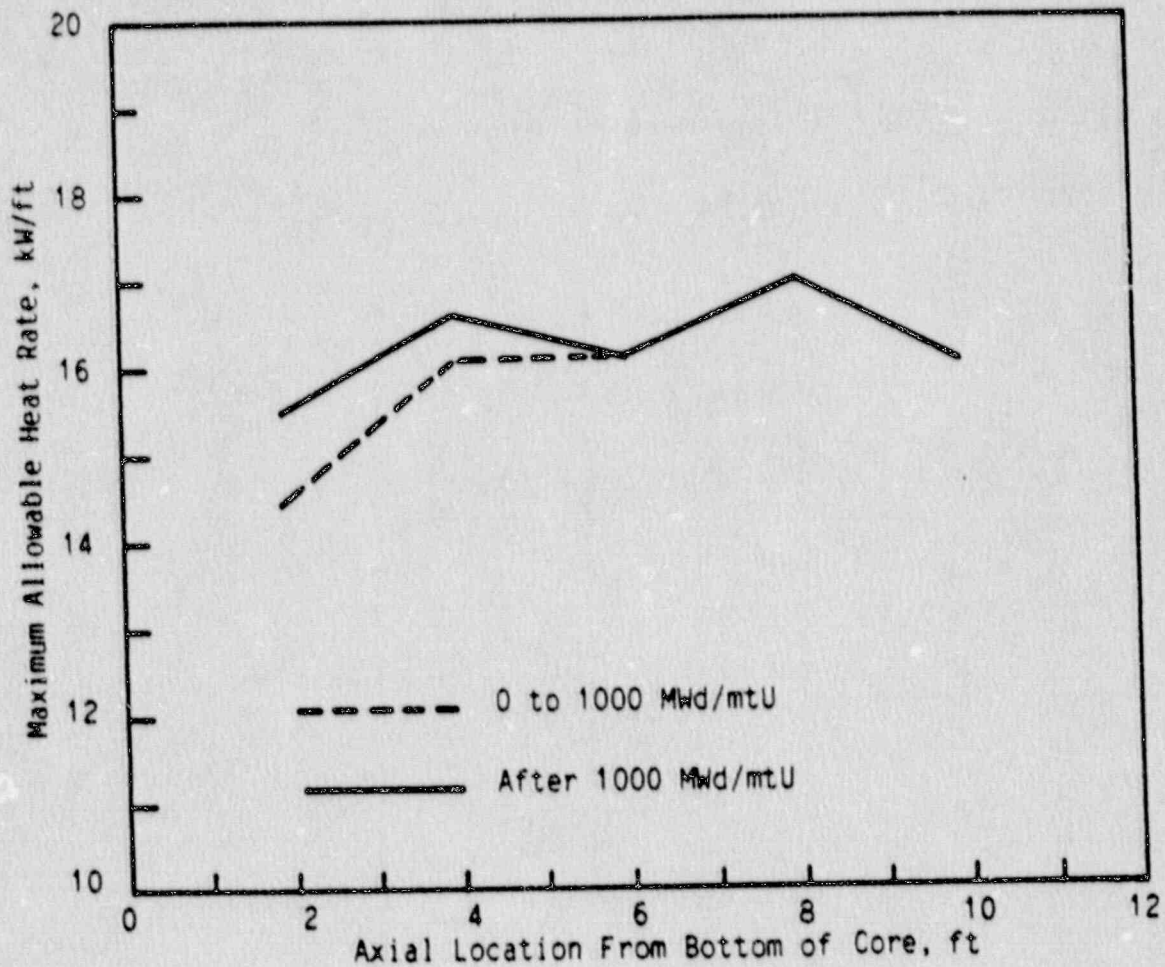


Figure 8-15. LOCA Limited Maximum Allowable Linear Heat Rate -- ANO-1 Cycle 10
(Tech Spec Figure 3.5.2-5)



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.

9.1.2. RC Flow

Reactor coolant flow with four RC pumps running will be measured at hot shutdown conditions. Acceptance criteria require that the measured flow be within allowable limits.

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 ± 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 $\pm 0.4 \times 10^{-4} \Delta k/k/^\circ F$.

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 $+0.9 \times 10^{-4} \Delta k/k/^\circ F$.

9.2.3. Control Rod Group/Boron 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 reactivity 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:

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

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

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

The boron reactivity worth (differential boron worth) is measured by dividing the total inserted rod worth by the boron change made for the rod worth test. The acceptance criterion for measured differential boron worth is as follows:

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

The predicted rod worths and differential boron worth are taken from the PIM.

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

The purpose of this test is to evaluate the symmetry of the core at low power during the initial power escalation following a refueling. Symmetry evaluation is based on incore quadrant power tilts during escalation to the intermediate power level. The core symmetry is acceptable if the absolute values of the quadrant power tilts are less than the error adjusted alarm limit.

9.3.2. Core Power Distribution Verification at Intermediate Power Level (IPL) and 100% FP With Nominal Control Rod Position

Core power distribution tests are performed at the IPL and 100% full power (FP). Equilibrium xenon is established prior to both the IPL and 100% FP tests. The test at the IPL is essentially a check on power distribution in the core to identify any abnormalities before escalating to the 100% FP plateau. Peaking factor criteria are applied to the IPL core power distribution results to determine if additional tests or analyses are required prior to 100% FP operation.

The following acceptance criteria are placed on the IPL and 100% FP tests:

1. The maximum LHR must be less than the LOCA limit.
2. The minimum DNER must be greater than the initial condition DNER limit.
3. The value obtained from extrapolation of the minimum DNER to the next power plateau overpower trip setpoint must be greater than the initial condition DNER limit, 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:

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

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

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

Items 1, 2, and 5 ensure that the safety limits are maintained at the IPL and 100 %FP.

Items 3 and 4 establish the criteria whereby escalation to full power may be accomplished without the potential for exceeding the safety limits at the overpower trip setpoint with regard to DNBR and linear heat rate.

Items 6 and 7 are established to determine if measured and predicted power distributions are consistent.

9.3.3. Incore Vs. Excore Detector Imbalance Correlation Verification at the IPL

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.

9.3.4. 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.

9.3.5. Power Doppler Reactivity Coefficient at ~100% FP

The power Doppler reactivity coefficient is calculated from data recorded during control rod worth measurements at power using the fast insert/withdraw method.

The fuel Doppler reactivity coefficient is calculated in conjunction with the power Doppler coefficient measurement. The power Doppler coefficient as measured above is multiplied by a precalculated conversion factor to obtain the fuel Doppler coefficient. This measured fuel Doppler coefficient must be more negative than the acceptance criteria limit of $-0.90 \times 10^{-5} \Delta k/k/^{\circ}F$.

9.4. Procedure for Use if Acceptance Criteria Not Met

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

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