PDR 016

November 24, 1986

Docket No. 50-313

DISTRIBUTION
Docket File
NRC & LPDRs
GEdison
PBD-6
FMiraglia
OGC-MNBB 9604
EButcher
LHarmon
ACRS-10
CMiles
D. Fieno
BGrimes
WJones
WRegan
RDiggs
RIngram
GVissing
Gray File
EJordan
TBarnhart-4
JPartlow
NThompson
JRamsey

Mr. Gene Campbell
Vice President, Nuclear
Operations
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Campbell:

The Commission has issued the enclosed Amendment No.105 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 10, 1986, as supplemented by letter dated September 19, 1986, and revised by letter dated November 7, 1986.

The amendment modifies the TSs to permit operation of ANO-1 for an eighth cycle (Cycle 8).

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Guy S. Vissing, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 105 to DPR-51
2. Safety Evaluation

cc w/enclosures:
See next page

PBD-6	PBD-6	PBD-6	PBD-6	PBD-6	PBD-6	OGC
RIngram	GVissing:eh	JRamsey	CMcCracken	GEdison	JPStolz	
11/18/86	11/19/86	11/18/86	11/19/86	11/19/86	11/19/86	11/21/86

No legal objection

8612030642 861124
PDR ADOCK 05000313
P PDR

Mr. G. Campbell
Arkansas Power & Light Company

Arkansas Nuclear One, Unit 1

cc:
Mr. J. Ted Enos, Manager
Nuclear Engineering and Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James M. Levine, Director
Site Nuclear Operations
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Mr. Nicholas S. Reynolds
Bishop, Liberman, Cook, Purcell & Reynolds
1200 Seventeenth Street, N.W.
Washington, D.C. 20036

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 2090
Russellville, Arkansas 72801

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Frank Wilson, Director
Division of Environmental Health
Protection
Department of Health
Arkansas Department of Health
4815 West Markham Street
Little Rock, Arkansas 72201

Honorable William Abernathy
County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated September 10, 1986, as supplemented September 19, 1986, and revised November 7, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 105, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 24, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. DPR-51

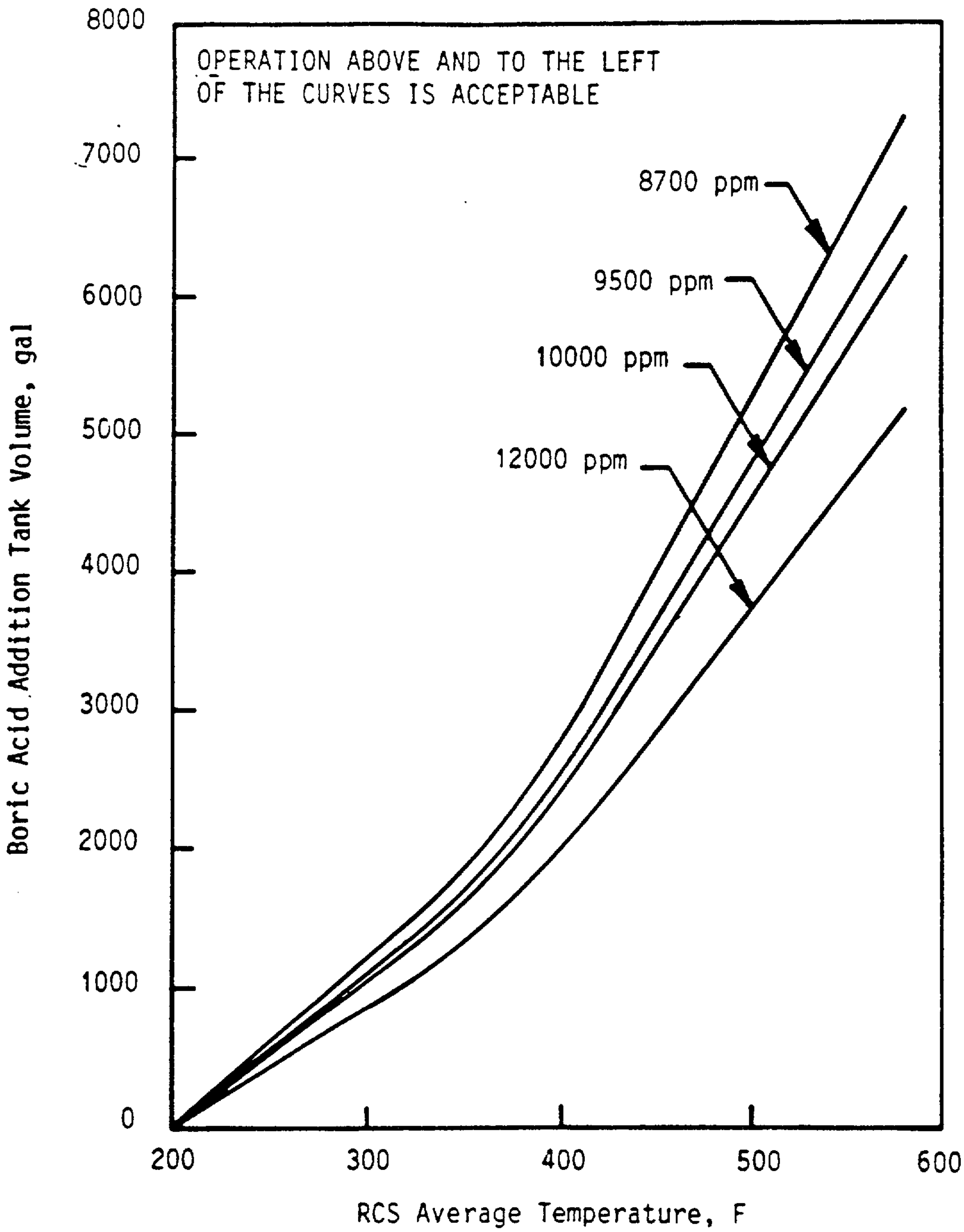
DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iv	iv
v	v
35a	35a
47	47
48	48
48b	48b
48c	48c
48d	48d
48e	48e
48f	48f
48g	48g
48h	48h
--	48i
--	48j
--	48k
--	48l
--	48m
--	48n
--	48o
--	48p
--	48q
--	48r
--	48s
--	48t
--	48u
--	48v
102	102

LIST OF FIGURES

Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature -- ANO-1, Cycle 8
Figure 3.2-1



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

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 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. Except for physics tests or exercising control rods, (a) the control rod withdrawal limits are specified on Figures 3.5.2-1(A-D), 3.5.2-2(A-D), and 3.5.2-3(A-D) for 4, 3 and 2 pump operation respectively; and (b) the axial power shaping control rod withdrawal limits are specified on Figure 3.5.2-6(A-D). 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.

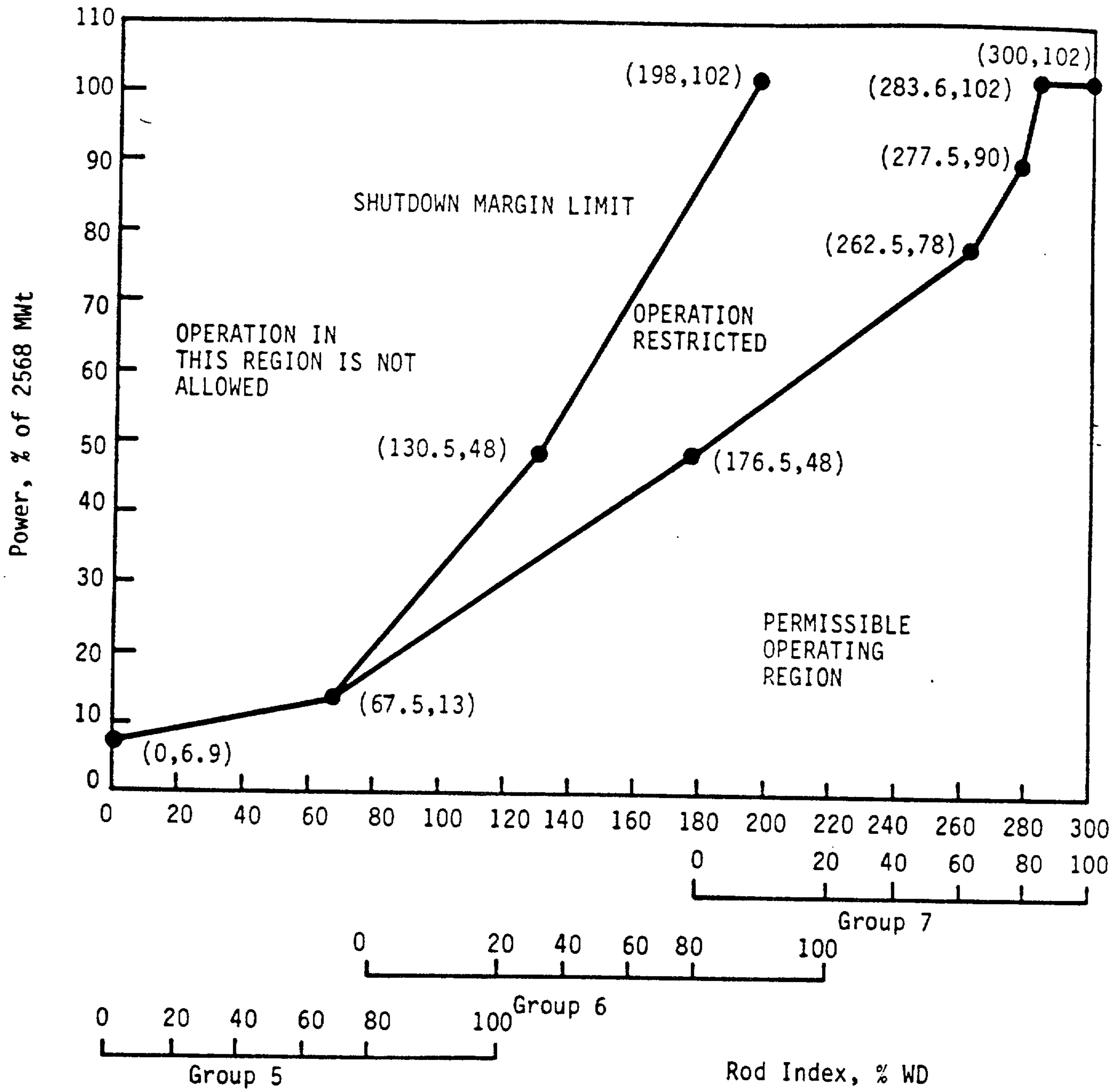
3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-4(A-D). If the imbalance is not within the envelope defined by Figure 3.5.2-4(A-D), 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

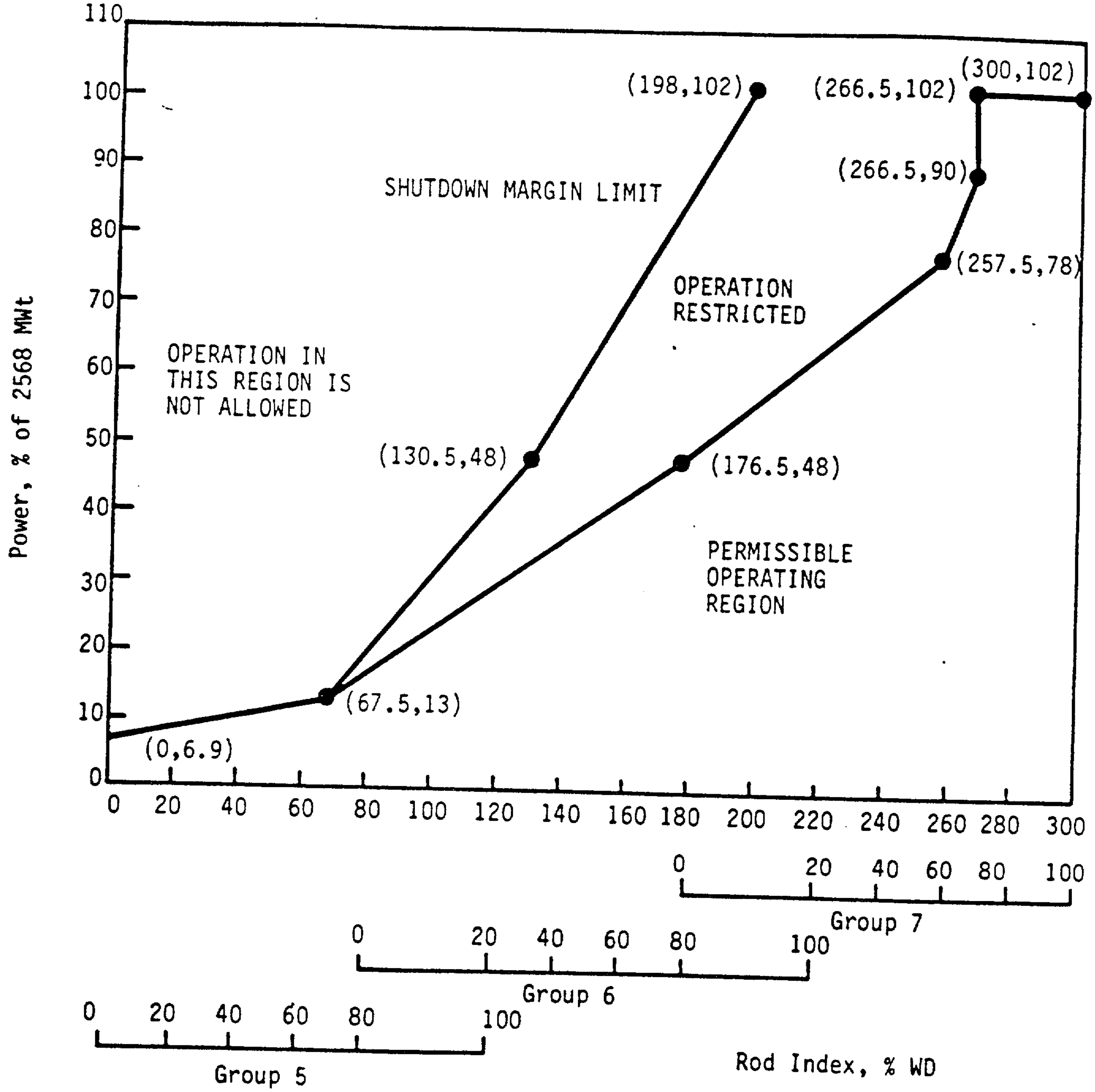
The power-imbalance envelope defined in Figure 3.5.2-4(A-D) is based on (1) LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria and (2) the Protective System Maximum Allowable Setpoints (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

Rod Position Setpoints for 4-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8
Figure 3.5.2-1A

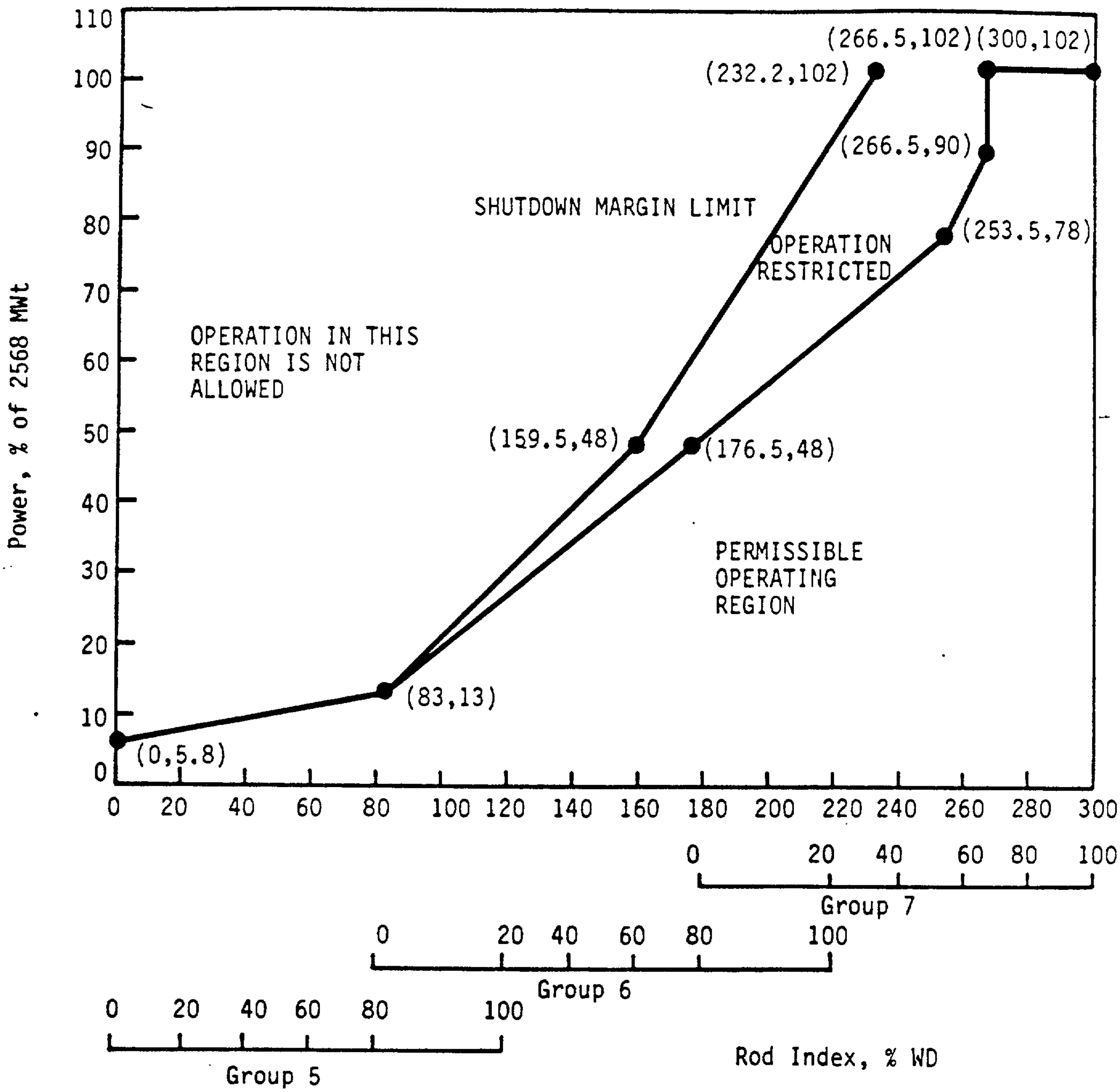


Rod Position Setpoints for 4-Pump Operation
From 25+10/-0 to 200± 10 EFPD ANO-1 Cycle 8

Figure 3.5.2-1B

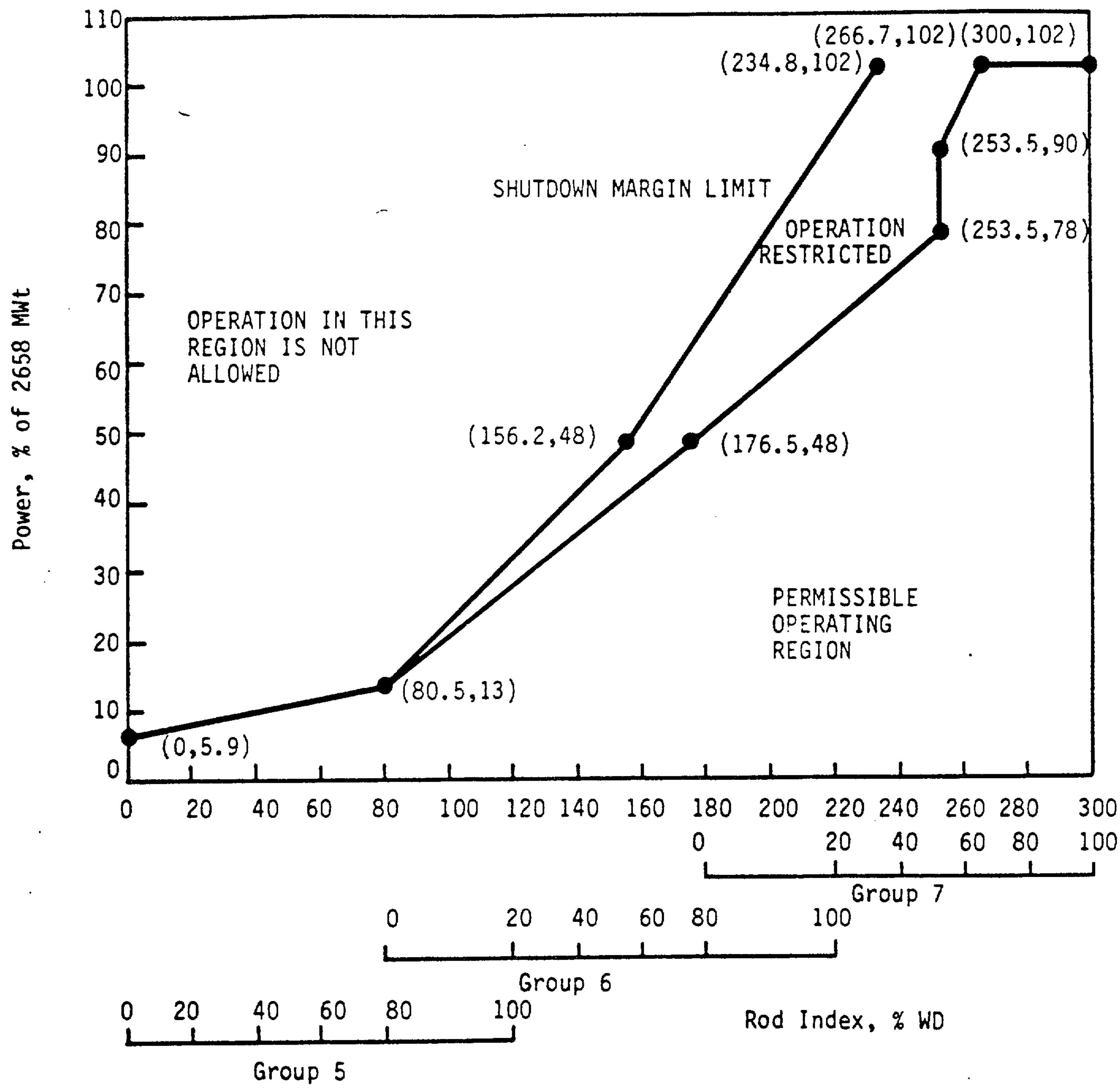


Rod Position Setpoints for 4-Pump Operation
From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8
Figure 3.5.2-1C



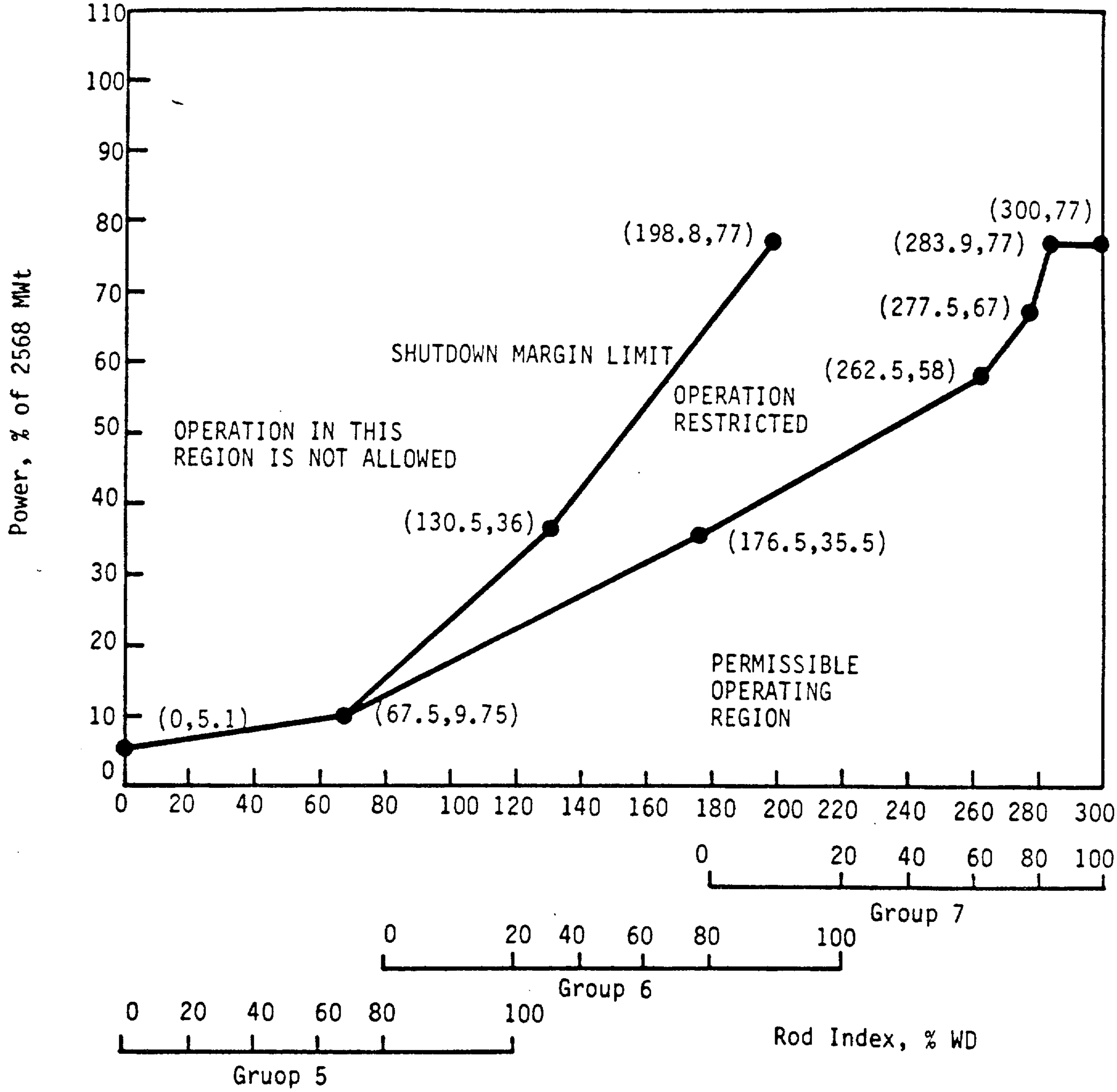
Rod Position Setpoints for 4-Pump Operation
After 380±10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-1D



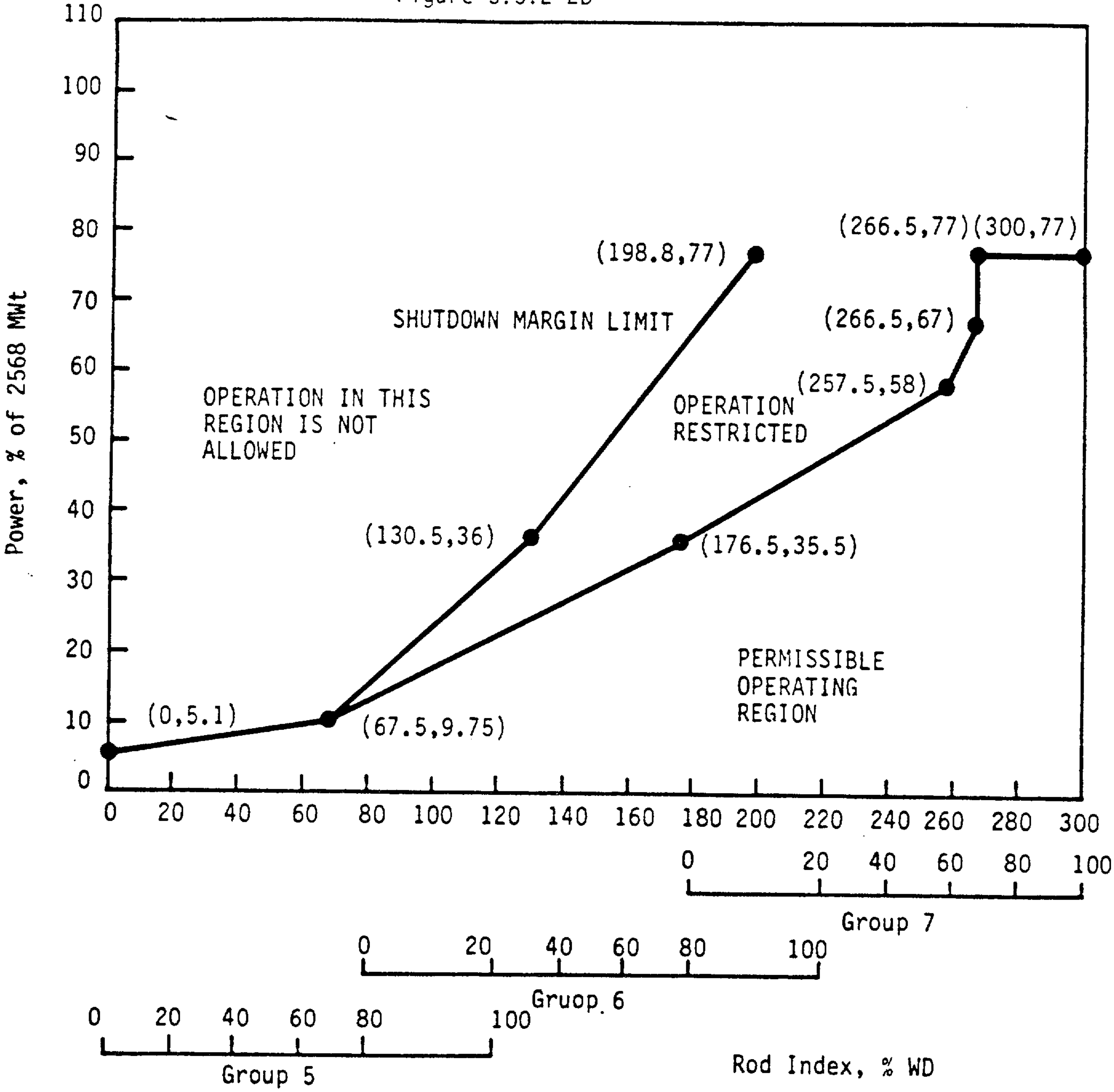
Rod Position Setpoints for 3-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-2A



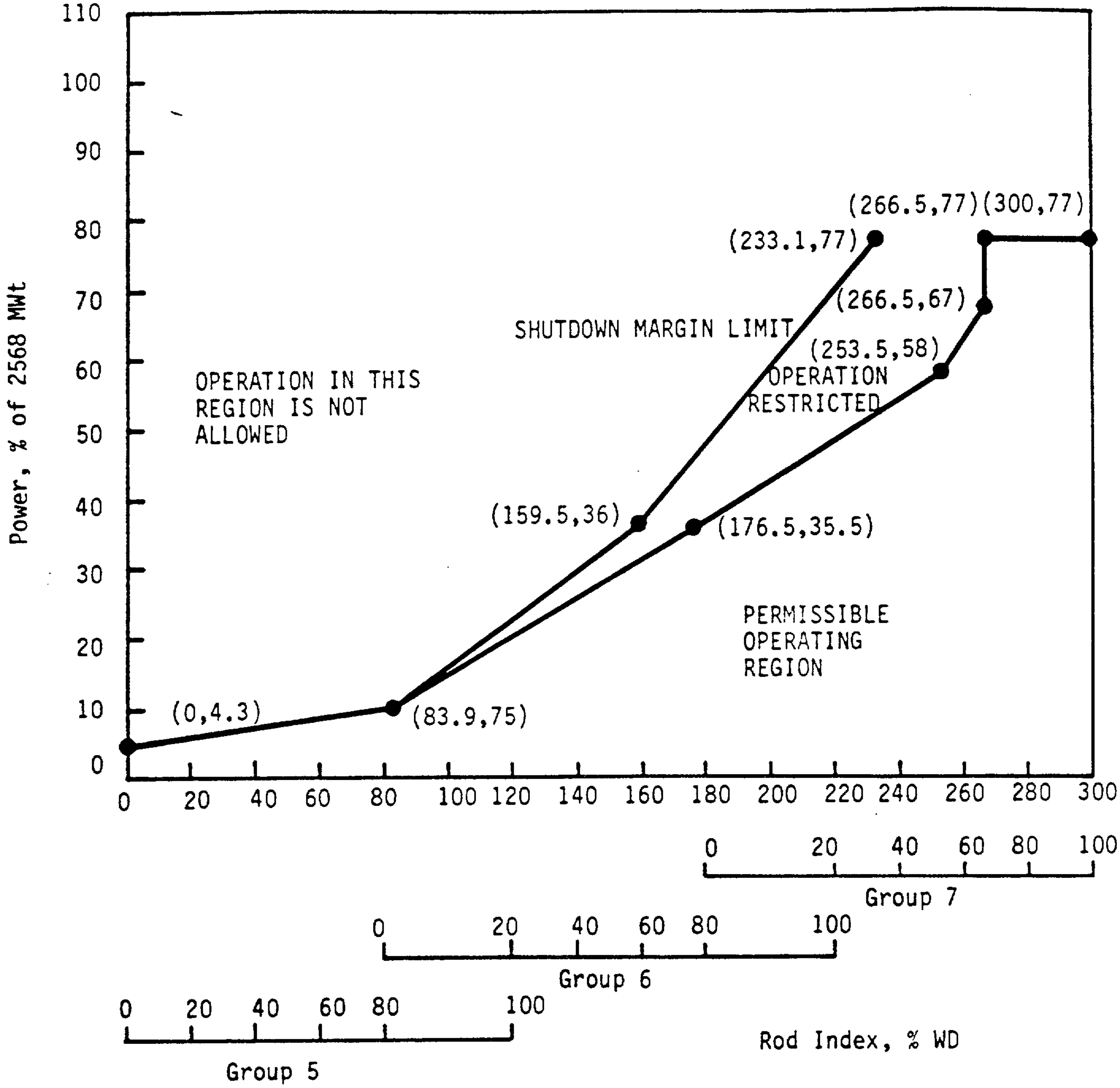
Rod Position Setpoints for 3-Pump Operation
From 25+10/-0 to 200±10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-2B



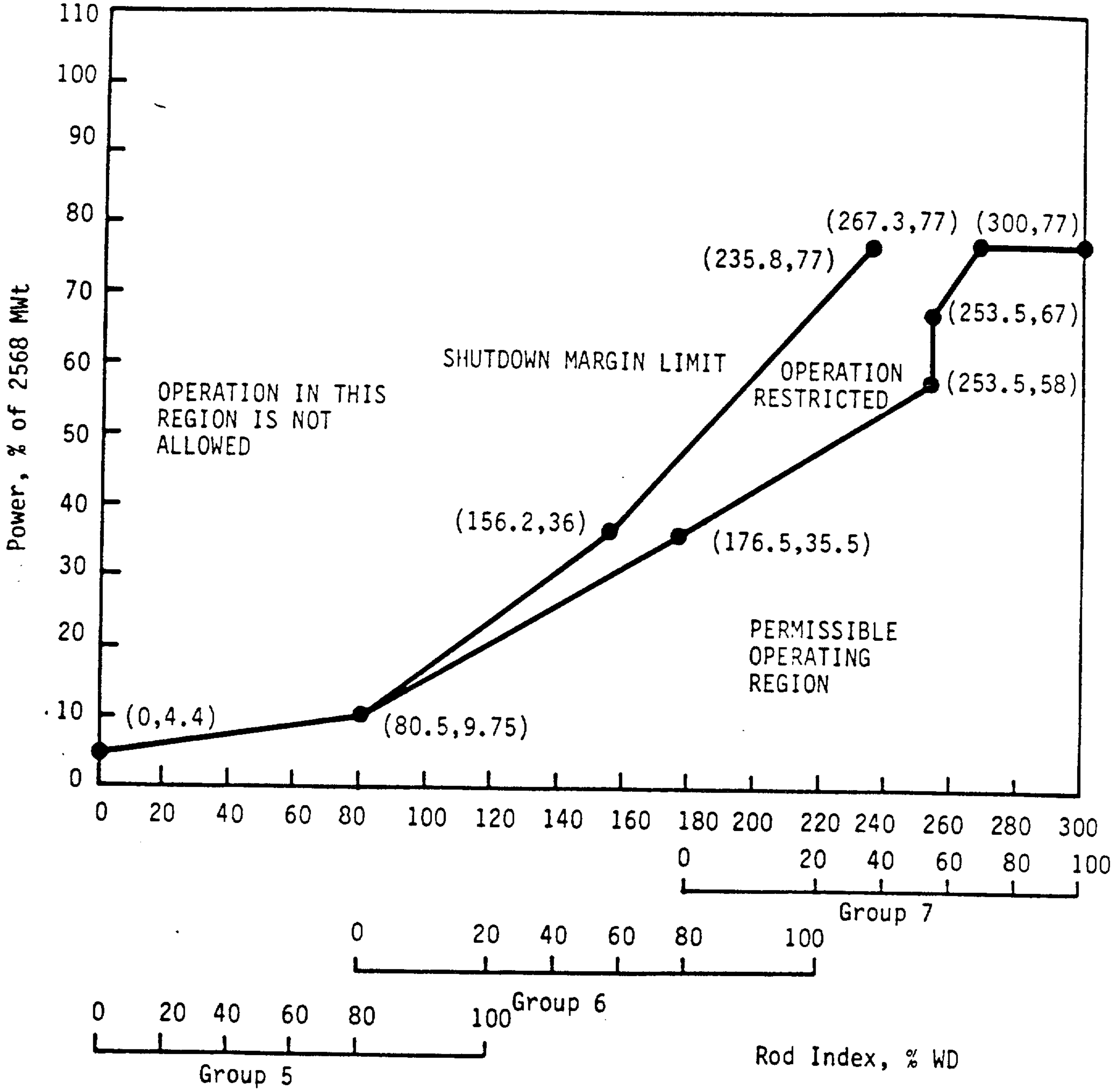
Rod Position Setpoints for 3-Pump Operation
From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-2C



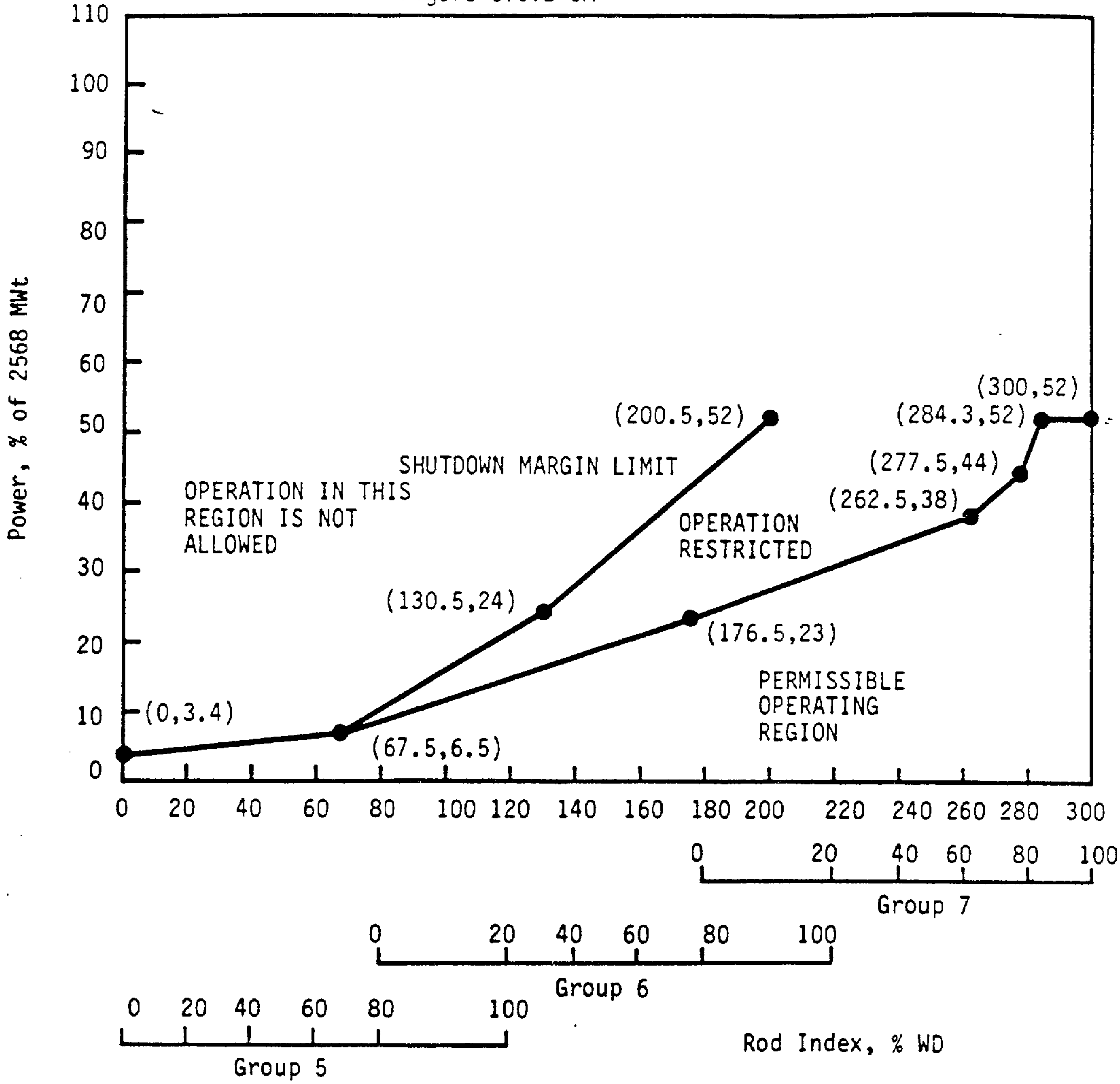
Rod Position Setpoints for 3-Pump Operation
After 380$\pm$10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-2D



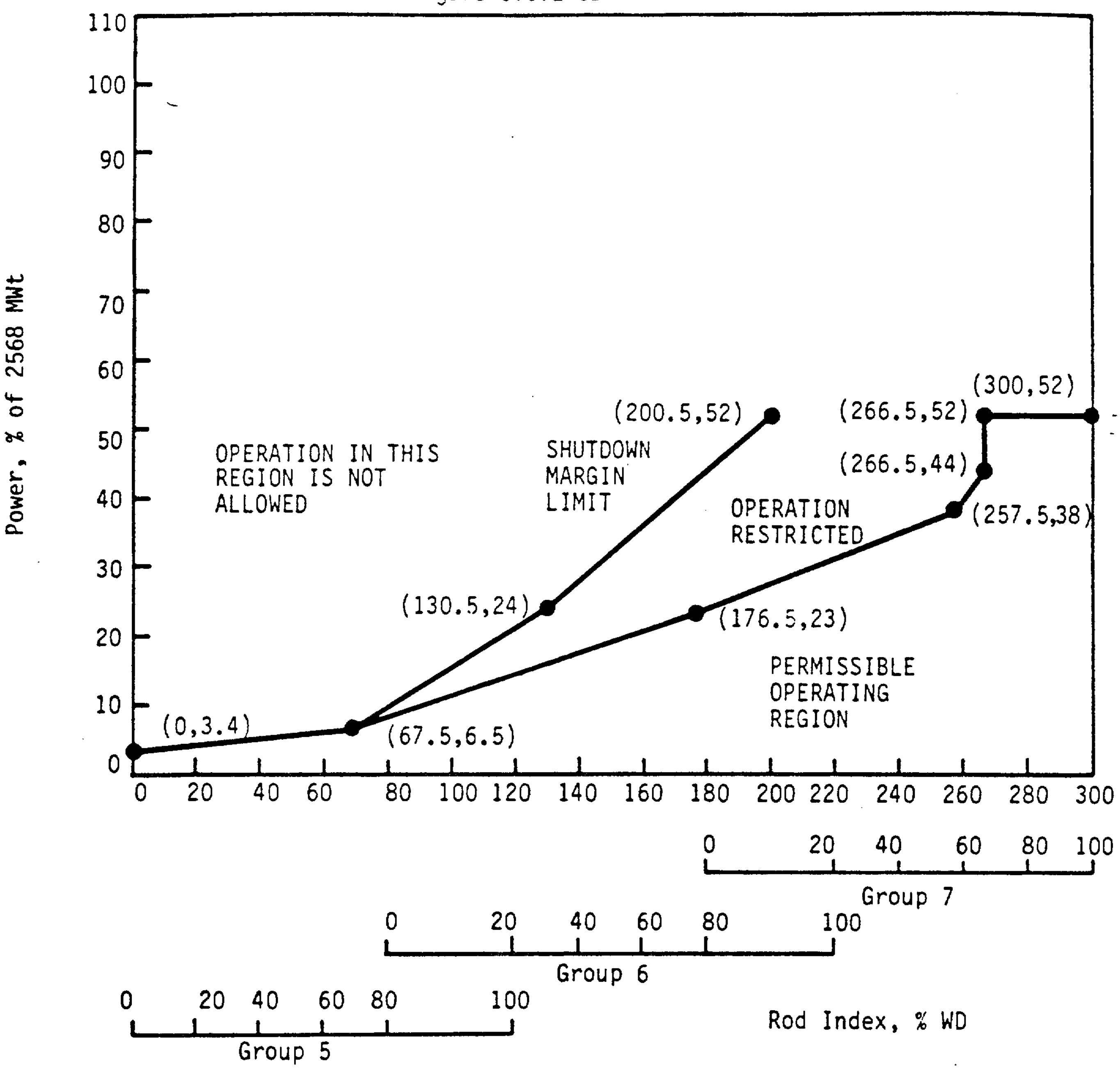
Rod Position Setpoints for 2-Pump Operation
From 0 to 25+10/-0 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-3A



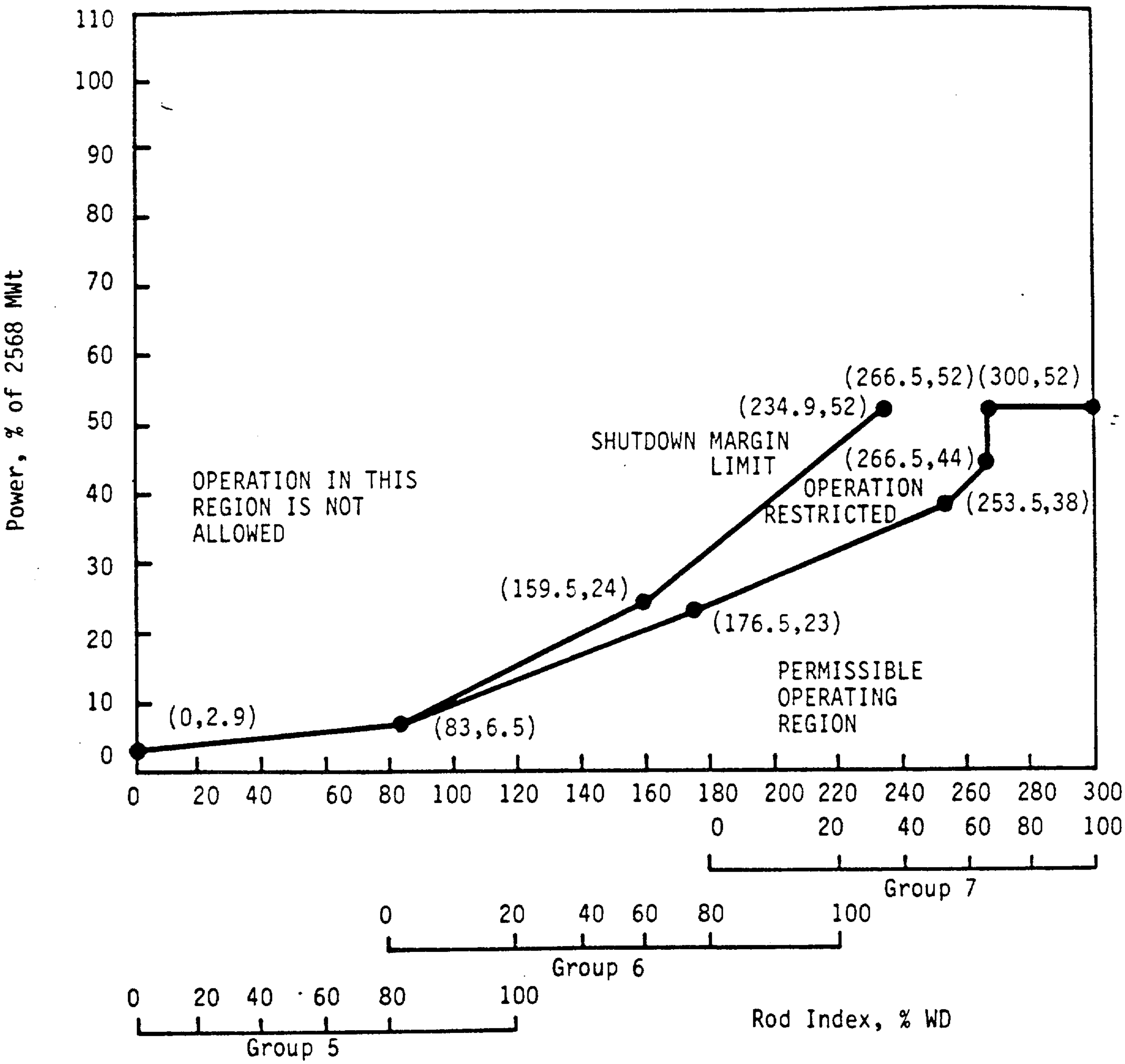
Rod Position Setpoints for 2-Pump Operation
From 25+10/-0 to 200±10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-3B



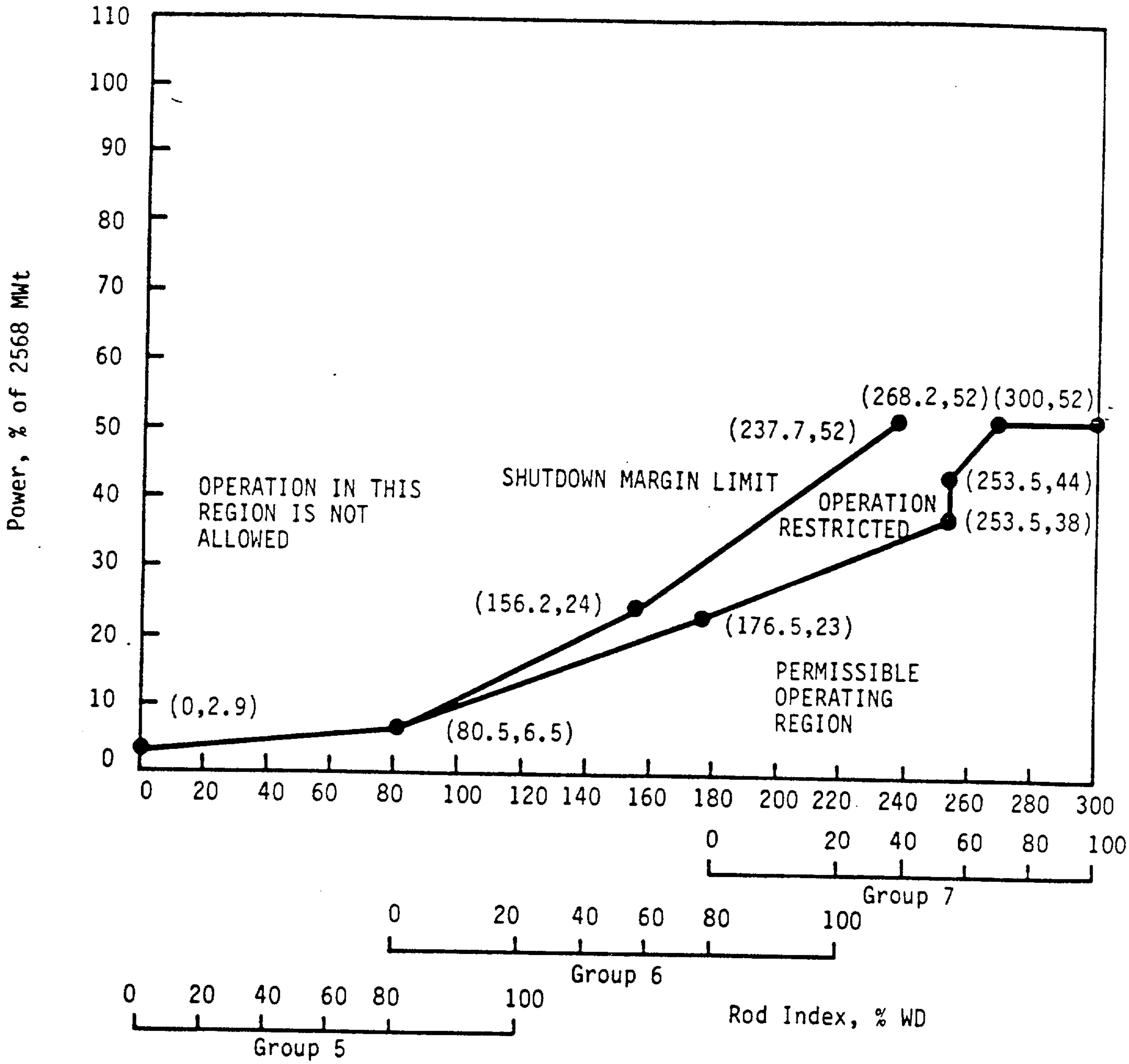
Rod Position Setpoints for 2-Pump Operation
From 200±10 to 380±10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-3C



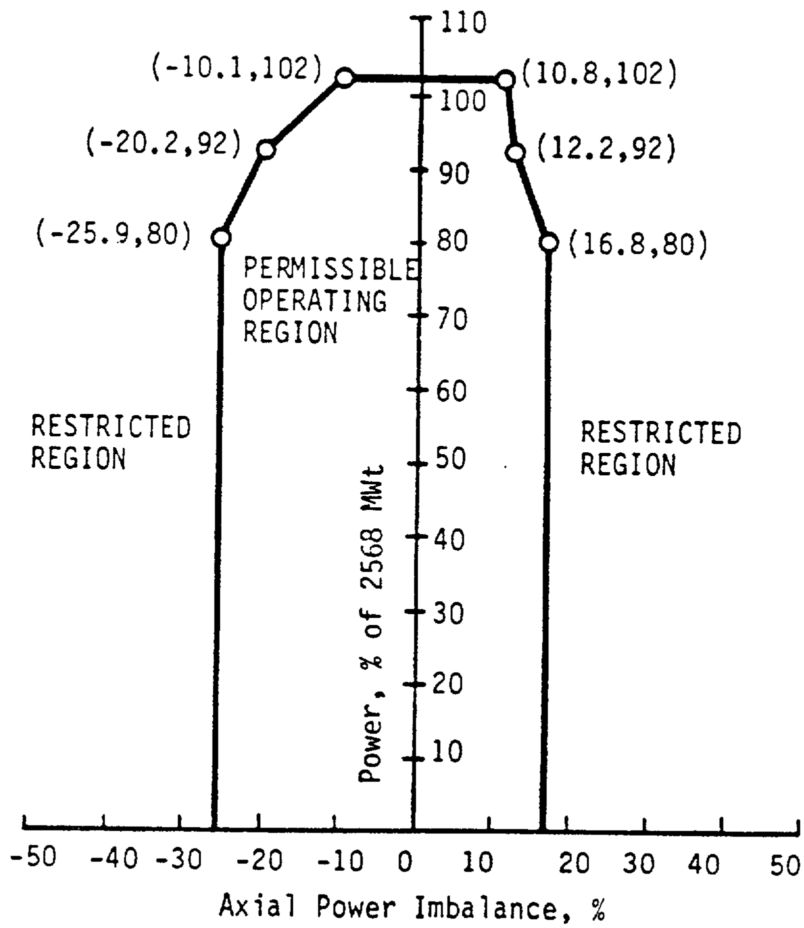
Rod Position Setpoints for 2-Pump Operation
After 380$\pm$10 EFPD -- ANO-1 Cycle 8

Figure 3.5.2-3D



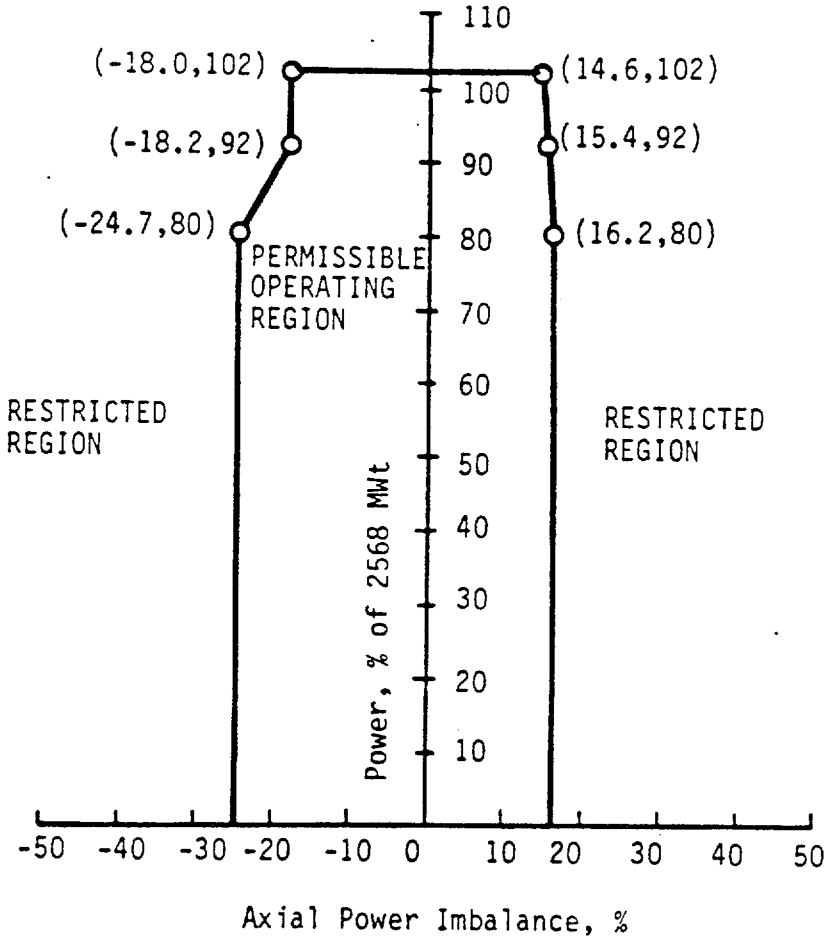
Operational Power Imbalance Setpoints for Operation
From 0 to 25+10/-0 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-4A



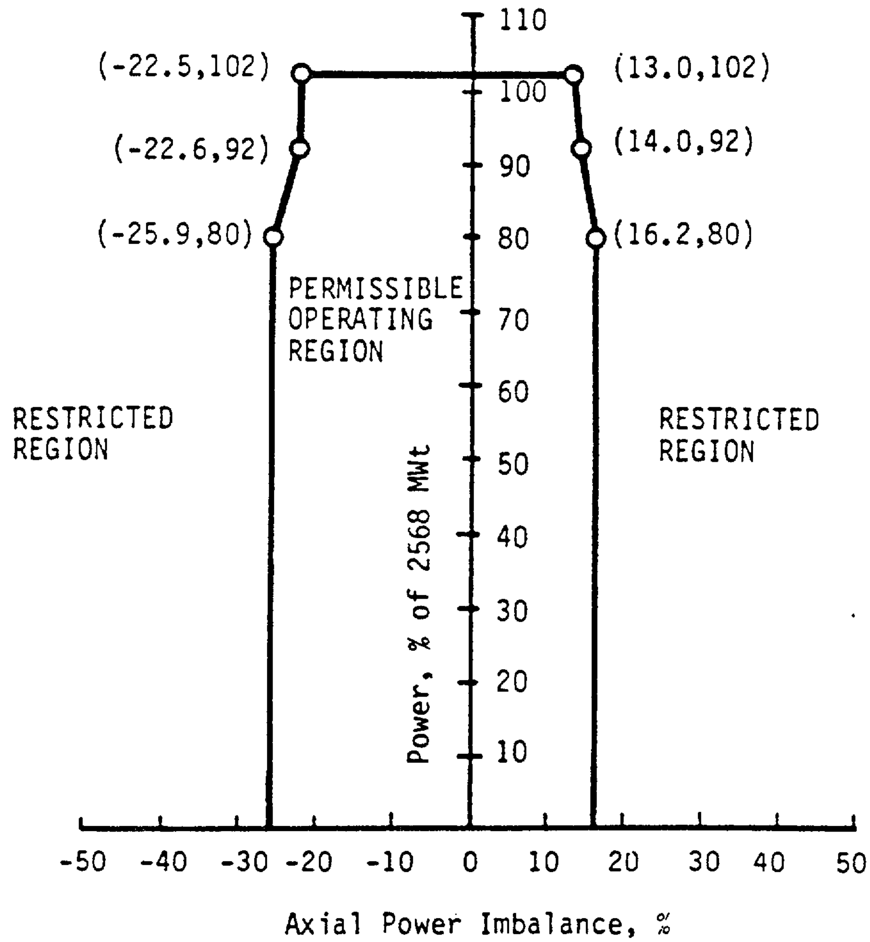
Operational Power Imbalance Setpoints for Operation From 25+10/-0 to 200±10 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-4B



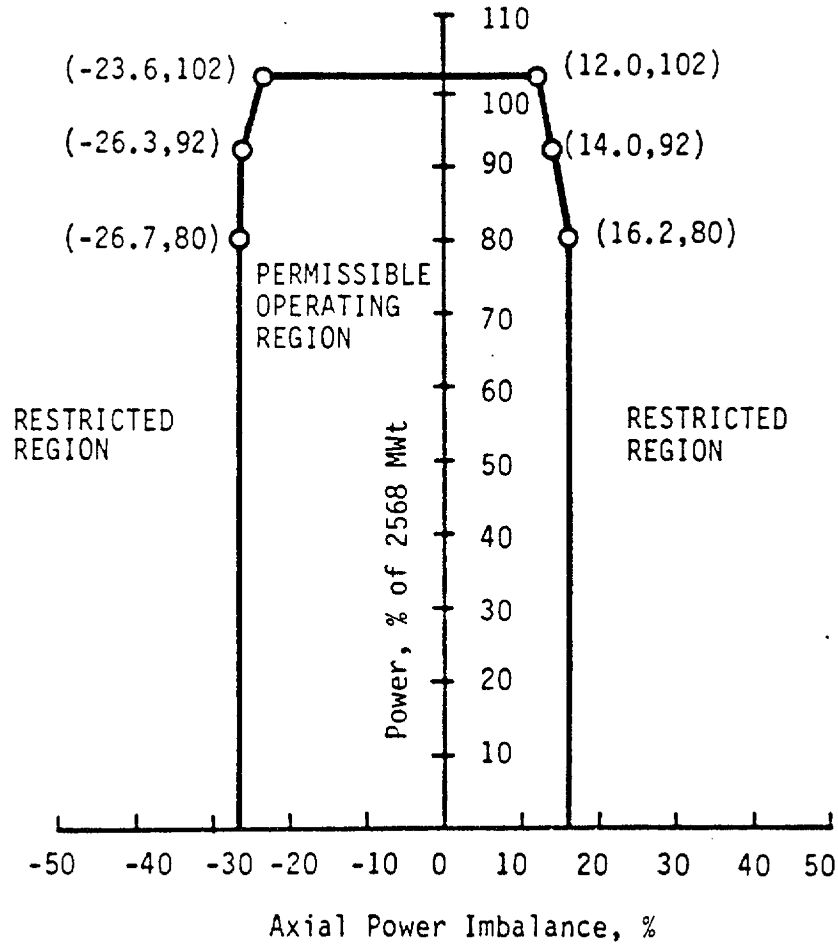
Operational Power Imbalance Setpoints for Operation
From 200±10 to 380±10 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-4C



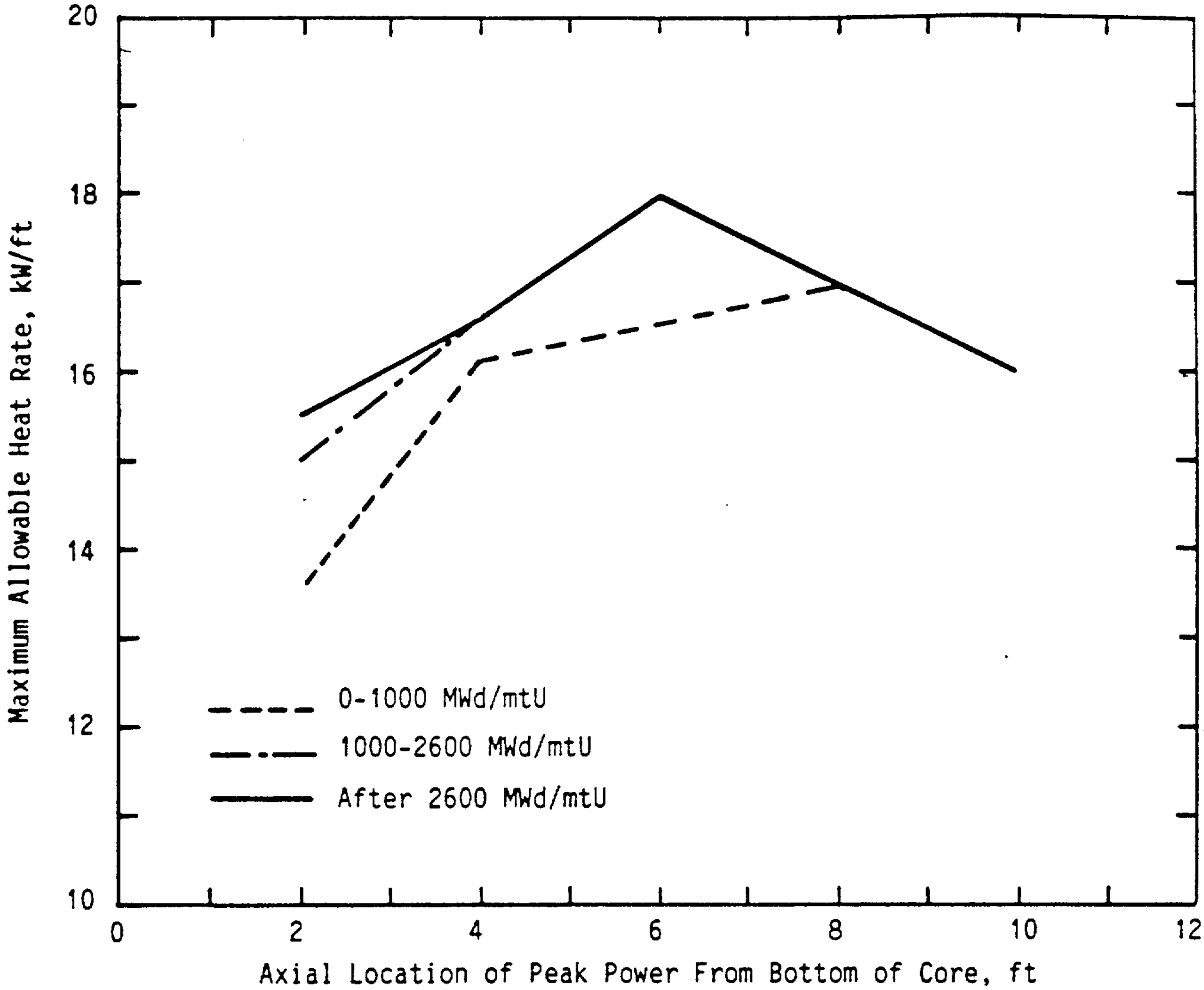
Operational Power Imbalance Setpoints for Operation After 380±10 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-4D



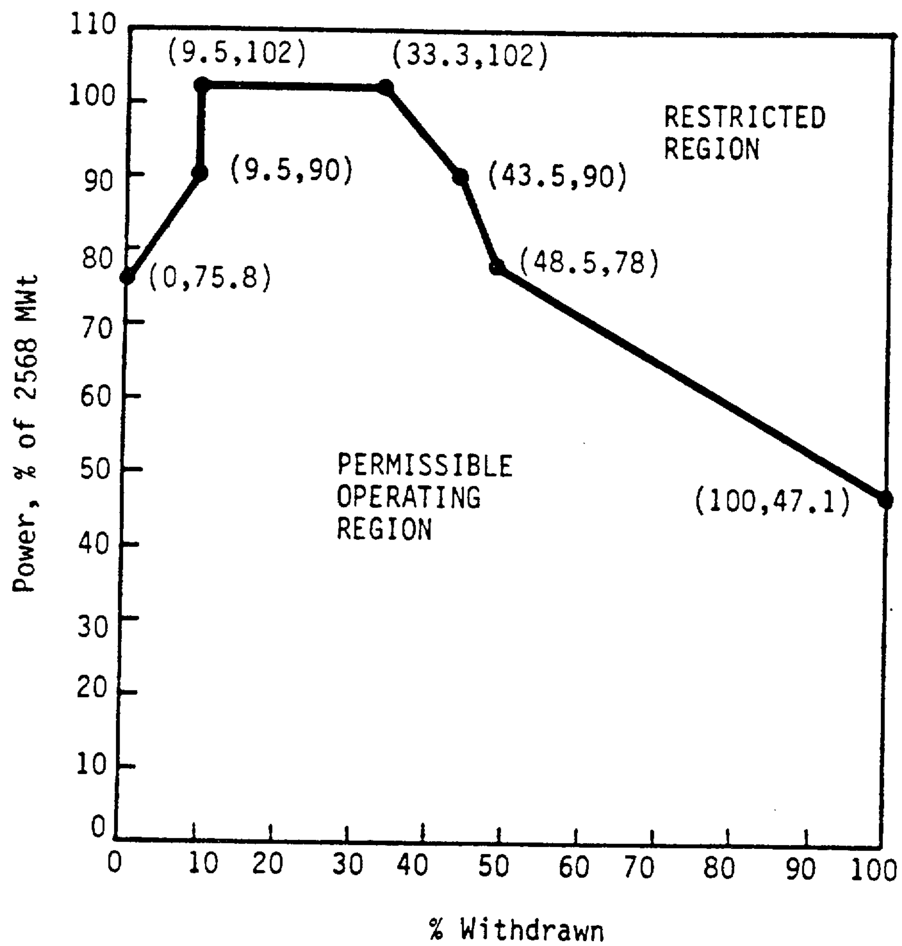
LOCA Limited Maximum Allowable
Linear Heat Rate

Figure 3.5.2-5



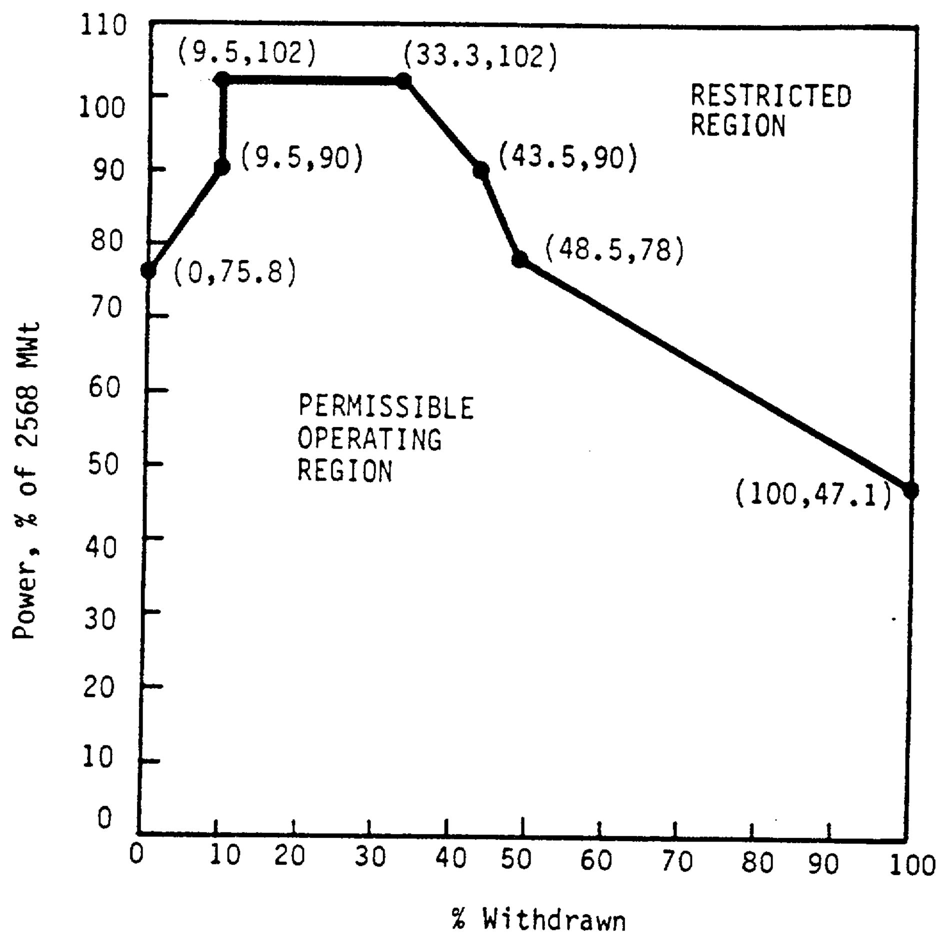
APSR Position Setpoints for Operation
From 0 to 25+10/-0 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-6A



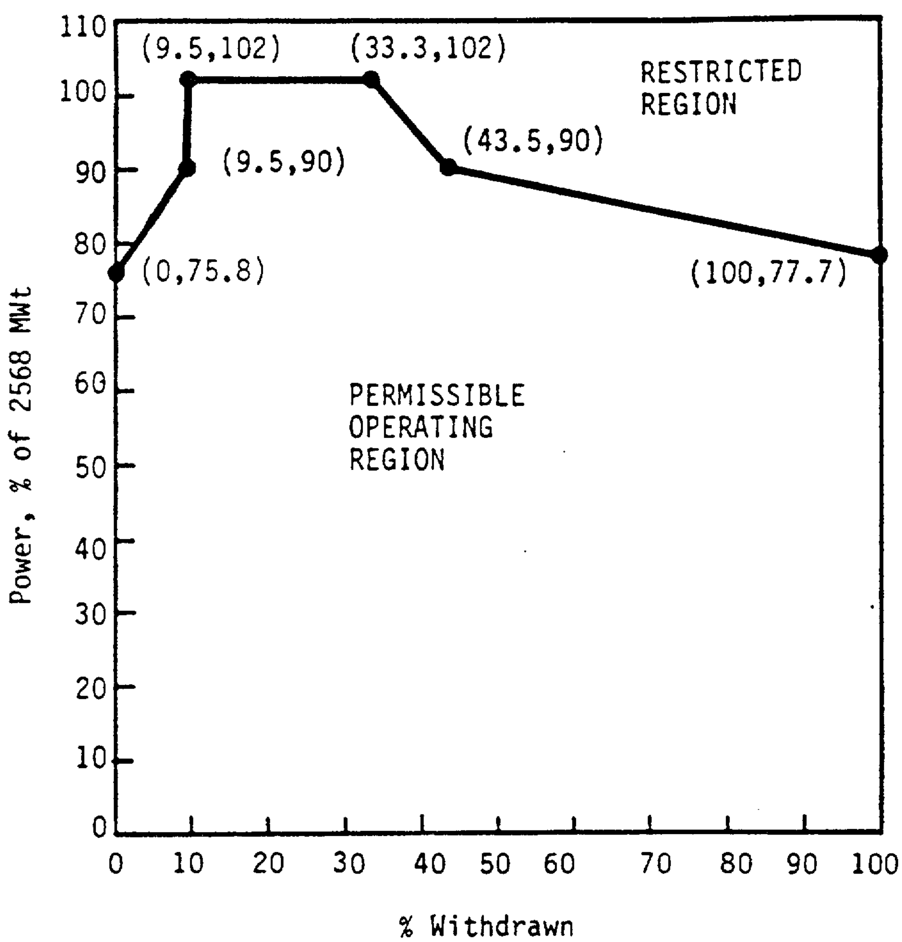
APSR Position Setpoints for Operation
From 25+10/-0 to 200$\pm$10 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-6B



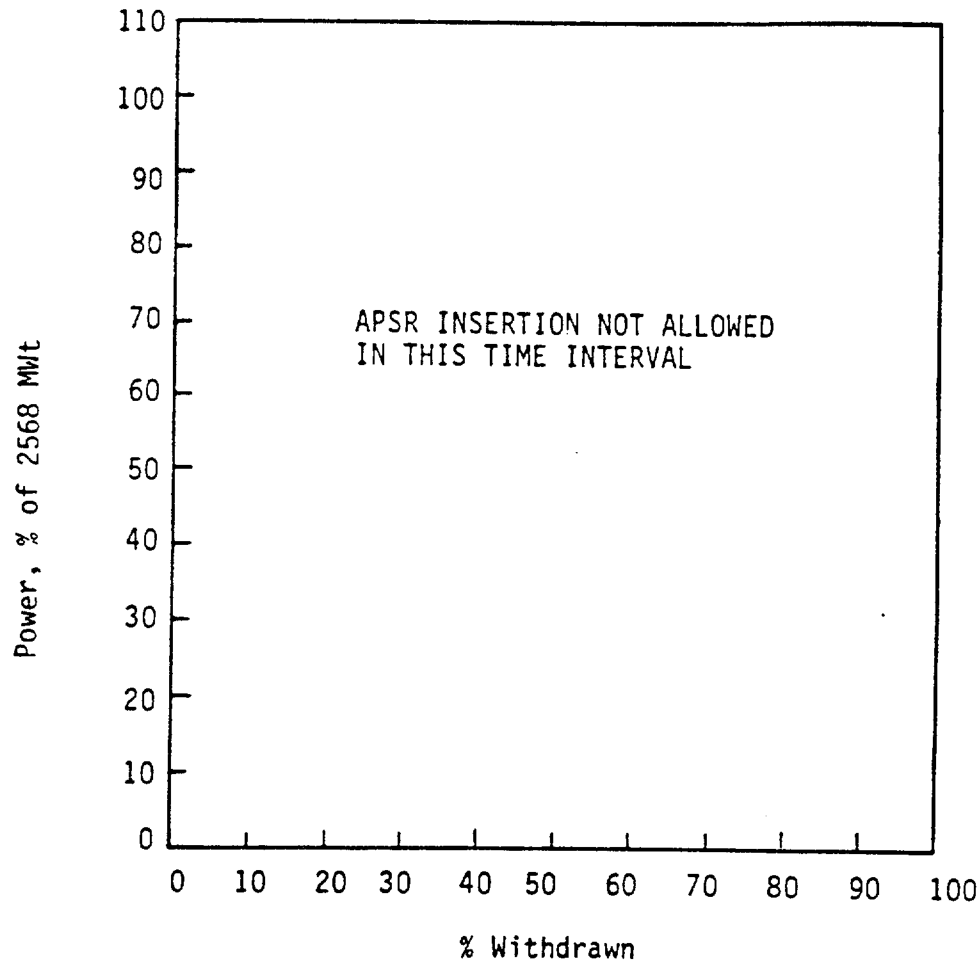
APSR Position Setpoints for Operation
From 200±10 to 380±10 EFPD -- ANO-1, Cycle 8

Figure 3.5.2-6C



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

Figure 3.5.2-6D



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 1.20 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 mechanisms 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

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 1.20 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 mechanisms 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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated September 10, 1986 (Ref. 1), with supporting data provided by letter dated September 19, 1986 (Ref. 14), and a revision provided by letter dated November 7, 1986 (Ref. 15), Arkansas Power and Light Company (AP&L or the licensee) requested amendment to the Technical Specifications appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The proposed amendment would modify the Technical Specifications to permit operation for an eighth cycle (Cycle 8). The safety analyses performed and the resulting modifications for ANO-1 are described in the Cycle 8 Reload Report (Ref. 2).

The safety analysis for the previous seventh cycle of operation at ANO-1 is being used by the licensee as the reference cycle for the proposed eighth cycle of operation. Cycle 7 operated with no anomalies that would adversely affect Cycle 8. Where conditions are identical or limiting in the seventh cycle safety analysis, our previous evaluation (Ref. 3) continues to apply.

Our evaluation of the safety analysis for the ANO-1 Cycle 8 reload follows.

2.0 EVALUATION

2.1 Description of the Cycle 8 Core

The ANO-1 core consists of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel management scheme is basically a low-leakage design with loading pattern and enrichments chosen to provide a Cycle 8 length of 420 effective full power days (EFPDs). The loading pattern consists of one batch 7 (redesignated batch 7D) lead test assembly (LTA) located at the center of the core; 44 batch 8 (redesignated batch 8B) assemblies will be shuffled to new locations, with 12 on the core periphery; 60 of the batch 9 assemblies will be shuffled to locations at or near the core periphery, with 8 batch 9 assemblies surrounding the center location; 64 fresh batch 10 assemblies will be loaded in a symmetric checkerboard pattern throughout the core. The batch 8B assemblies are characterized as being twice-burned assemblies while the batch 9 assemblies are once-burned. The batch 7D assembly is a thrice-burned assembly. The fuel enrichments for batches 7D, 8B, 9, and 10 are 2.95, 3.21, 3.30, and 3.35 weight percent uranium-235, respectively.

Reactivity control for Cycle 8 will be provided by 60 full-length silver-indium-cadmium control rods, 64 burnable poison rod assemblies (BPRA) containing varying amounts of B_4C admixed with Al_2O_3, and soluble boron in the primary coolant. Cycle 8 will not contain a centrally located control rod. The core will contain eight axial power shaping rods (APSRs) for additional control of the axial power distribution. Except for the five centrally located fuel assemblies and those fuel assemblies located on the core periphery, each fuel assembly will contain either a control rod or a BPRA. Cycle 8 will operate at full power such that only regulating control rod Bank 7 is partially inserted and such that the Bank 8 APSRs are within the range of 9.5 to 33.3 percent withdrawn for most of Cycle 8. After 380 EFPDs, the APSRs will be completely withdrawn from the reactor core.

The licensed core full power level is 2568 MW_t. The safety analysis provided in the reload report (Ref. 2) demonstrates the safe operation of ANO-1 throughout Cycle 8 at full power. The following sections describe our evaluation of the safety analysis.

2.2 Evaluation of the Fuel System Design

2.2.1 Fuel Assembly Mechanical Design

The 64 Babcock and Wilcox (B&W) Mark B4 15x15 fuel assemblies to be loaded as batch 10 fuel for Cycle 8 operation are mechanically interchangeable with batches 7D, 8B, and 9 fuel assemblies previously loaded at ANO-1. The batch 10 fuel assemblies incorporate the design features of anti-straddle lower end fittings and annealed guide tubes. The anti-straddle lower end fitting prevents mispositioning a fuel assembly in the lower grid during fuel assembly movement. The annealed guide tubes reduce incore irradiation fuel assembly growth which permits higher burnup capability. The Mark MK-BEB fuel assembly loaded as batch 7D differs from the Mark B4 assemblies in that some fuel rods can be easily removed and windows are cut in the upper grid skirt to permit observation of fuel rod growth.

2.2.2 Fuel Rod Design

Batches 8B, 9, and 10 in the ANO-1 Cycle 8 core utilize the same B&W Mark B4 fuel design. The Batch 10 fuel parameters are identical to the previously loaded batches 8B and 9 except for enrichment, which has been increased to 3.35 weight percent uranium-235.

There has been a change in the fuel rod pre-pressurization in that the batch 10 fuel rods have a decrease in the fuel rod pre-pressure of 50 psi. The licensee states that this change will improve fuel performance and has been included in all mechanical and thermal analyses.

The one fuel assembly in batch 7D is an extended burnup LTA, which is scheduled for its fourth cycle of burnup in Cycle 8. This assembly, which is described in Reference 4, is similar in design to the Mark B4 assemblies except for changes to the fuel rod and fuel assembly structure to extend its burnup capability. We previously concluded (Ref. 5) that the irradiation of the LTA in ANO-1 is acceptable.

The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for ANO-1 or were analyzed specifically for Cycle 8 using methods and limits previously reviewed and approved by the NRC.

2.2.3 Fuel Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 6) addresses a number of acceptance criteria used to establish the design bases to evaluate the fuel system. Among those which may affect the operation of a fuel rod is the internal pressure limit. The NRC staff's current criterion is that fuel rod internal gas pressure should remain below nominal system pressure during normal operation unless otherwise justified. AP&L states that fuel rod internal pressure will not exceed nominal system pressure during normal operation of Cycle 8. This is based on analyses performed with the approved B&W TACO2 code (Ref. 7). We conclude that the rod internal pressure limit has been acceptably considered for Cycle 8 operation.

2.2.4 Fuel Thermal Design

There are no major changes between the thermal design of the new batch 10 fuel and previous batches that will be reinserted in the Cycle 8 core. The licensee presented results of the thermal design evaluation of the Cycle 8 core. These are based on analyses performed with the approved TACO2 code (Ref. 7). The Cycle 8 core protection limits are based on a linear heat generation rate (LHGR) to centerline fuel melt of 20.5 kW/ft, which is applicable to fuel batches 8B, 9 and 10. The LHGR limit for the one batch 7D fuel assembly is greater than 20.5 kW/ft. The results of the thermal design evaluation show no difference between the new batch 10 fuel and the previous batches 8 and 9 fuel. We have reviewed the fuel thermal design parameters for normal operation and find them acceptable.

2.2.4.1 Loss of Coolant Accident (LOCA) Initial Conditions

In addition to the steady-state conditions, the average fuel temperature as a function of LHGR and lifetime fuel pin pressure data used in the LOCA analysis (see Section 7.2 of Ref. 2) are also calculated with the TACO2 code (Ref. 7). The reload report (Ref. 2) states that the fuel temperature and pin pressure data used in the generic LOCA analysis (Ref. 8) are conservative compared to those calculated for ANO-1 Cycle 8. The bounding values of the allowable LOCA LHGRs (see Table 7.3 of Ref. 2) include the effects of NUREG-0630 regarding fuel cladding swelling and rupture behavior during LOCA.

2.2.5 Conclusion on Cycle 8 Fuel System Design

We have reviewed the fuel system design and analysis for ANO-1 Cycle 8 operation and find it acceptable, as discussed above.

2.3 Evaluation of the Nuclear Design

To support Cycle 8 operation of ANO-1, the licensee has provided analyses using analytical methods and design bases established in licensing topical reports that have been approved by the NRC. The licensee has provided a comparison of the core physics parameters for Cycles 7 and 8 as calculated with these approved methods. The parameters for Cycle 7 were generated using PDQ07 (Ref. 9) while the parameters for Cycle 8 were generated using the NOODLE code (Ref. 10). The two codes give comparable results when compared to measured data. There are slight differences in the parameters compared between Cycles 7 and 8. These differences can be attributed to differences in new fuel assembly enrichment, BPRA loading, and shuffle pattern. All of the accidents analyzed in the Final Safety Analysis Report (FSAR) were reviewed for Cycle 8 operation. The Cycle 8 parameters were conservative when compared to analyses accepted for previous cycles and no new accident analyses are included in the reload report (Ref. 2).

We conclude that the licensee's predicted nuclear parameters are acceptable because they were obtained using approved methods, the validity of which has been reinforced through a number of cycles of predictions, including startup tests, for this and other reactors. As a result of this review of the nuclear parameters compared to previous cycles, we concur with the licensee's conclusions regarding Cycle 8 accident analysis. The licensee plans to withdraw the APSRs near the end of Cycle 8, at 380 EFPDs. The calculated stability index is -0.022 per hour at 384 EFPDs, which ensures the axial stability of the core to axial xenon transients.

The licensee has made a number of changes in the nuclear design of Cycle 8. These changes are (1) the center control rod has been removed, (2) the lumped burnable poison (LBP) has a 4.5 inch longer poison stack than was used for Cycle 7, that is, 121.5 versus 117 inches of $B_4C-Al_2O_3$, (3) the NOODLE code was used to calculate the physics parameters for Cycle 8, and (4) the power level hold requirements of Technical Specifications 3.5.2.4 and 3.5.2.5 have been removed. The removal of the center control rod has been taken into account in the nuclear design and, according to the licensee, had a negligible effect on the Cycle 8 nuclear parameters. The LBP design alters the axial power shape and increases operating flexibility at the beginning of the cycle. The NOODLE code has been reviewed and approved by the staff (Ref. 11). An extensive analysis has been performed by B&W for the licensee (Ref. 13) to justify removal of the power level cut-off requirements. This power level cutoff had been utilized to accommodate transient xenon effects on power peaking factors before ascending to 100% power. The analysis showed that the 5 percent total xenon factor applied in the computation of LOCA margin provides conservative operating limits. The 2.5 percent radial xenon factor applied in the evaluation of initial condition departure from nucleate boiling (DNB) margin was also shown to be conservative. We conclude that these changes in the Cycle 8 nuclear design are acceptable since the nuclear design and resulting Technical Specifications for Cycle 8 include the effects of the changes calculated with approved methods.

2.4 Evaluation of the Thermal-Hydraulic Design

The thermal-hydraulic design of Cycle 8 is identical to that of Cycle 7 as shown in the comparison of maximum design conditions in Table 6-1 of Reference 2. The same methods and models approved for use in Cycle 7 are used for Cycle 8. The fresh batch 10 fuel assemblies are hydraulically and geometrically similar to irradiated batches 8B and 9 fuel assemblies. The modified lower end fitting of the batch 10 fuel has, according to the licensee, negligible impact on the thermal-hydraulic design. The one batch 7D LTA is never the limiting assembly during Cycle 8 operation. No departure from nuclear boiling ratio (DNBR) penalty is required since the approved rod bow topical report (Ref. 12) shows that the reduction in power production capability more than offsets any rod bow effects as burnup increases. Based on the similarities of Cycle 8 with Cycle 7 and the use of approved methods and models, we conclude that the thermal-hydraulic design of Cycle 8 is acceptable.

2.5 Evaluation of the Accident and Transient Analyses

The licensee has examined each FSAR transient and accident analysis with respect to changes in the Cycle 8 parameters to ensure that the calculated consequences still meet applicable criteria. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, the thermal-hydraulic parameters, and the physics static and kinetic parameters. Fuel thermal analysis values are listed in Table 4-2 of Reference 2 for all fuel batches in Cycle 8. Table 6-1 of Reference 2 compares the thermal-hydraulic parameters for Cycles 7 and 8. These parameters are the same for both cycles. The physics parameters are provided in Table 5-1 of Reference 2. A comparison of key kinetic parameters from the FSAR and for Cycle 8 is provided in Table 7-2 of Reference 2. These changes indicate no significant changes or changes in the conservative direction for all parameters except for the hot-zero power all rod group worth. The value for Cycle 8 is somewhat less than the value in the FSAR analysis. However, the licensee has demonstrated ample shutdown margin for Cycle 8. The effects of fuel densification on the FSAR accident analyses have also been evaluated.

A generic LOCA analysis for the B&W 177-fuel assembly, lowered loop plant design has been performed using the Final Acceptance Criteria (FAC) emergency core cooling system (ECCS) evaluation model (Ref. 8). That analysis used the limiting values of the key parameters for all plants in the 177-FA lowered-loop category and is, therefore, bounding for the ANO-1 Cycle 8 operation.

The radiological dose consequences of the accidents presented in the FSAR have been reevaluated for Cycle 8. The reason for the reevaluation is the increased amount of energy produced by fissioning plutonium caused by the extended cycle fuel management strategy. The bases used in the radiological dose evaluation are the same as in the FSAR except for three factors: (1) the fission yield and half-lives used in the Cycle 8 evaluation are based on current data, (2) whole body gamma dose conversion factors are based on updated (lowered) factors, and (3) the steam generator tube rupture accident (SGTR) evaluation considers the increased amount of steam released to the environment because of a post-TMI modification. All radiological doses are bounded by the values presented in the FSAR or are a small fraction (10%) of the 10 CFR Part 100 limits except for the maximum hypothetical accident (MHA) which meets 10 CFR Part 100 limits.

We conclude from the examination of Cycle 8 core thermal and kinetic parameters, with respect to previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the ANO-1 plant's ability to operate safety during Cycle 8.

2.6 Technical Specifications

As indicated in our evaluation of the nuclear design, provided in Section 2.3, the operating characteristics of Cycle 8 were calculated with well-established, approved methods. The proposed Technical Specifications are the result of the cycle-specific analyses for power peaking, control rod worths, and quadrant tilt allowance. The removal of the power level cut-off to accommodate transient xenon effects was discussed in Section 2.3. We conclude that the Technical Specification changes proposed by the licensee in Reference 1 and repeated in Section 8 of the Cycle 8

Reload Report (Ref. 2) are acceptable. The proposed Technical Specification changes are as follows:

1. A new Figure 3.2-1 will be provided giving the boric acid addition tank volume and concentration as a function of reactor coolant system temperature to accommodate Cycle 8 shutdown margin requirements.

2. TS 3.5.2.4.1 Quadrant Tilt
   The power level cutoff requirement has been deleted.

3. TS 3.5.2.5.3 Control Rod Positions
   New control rod insertion limits are provided for 4, 3 and 2 pump operation, as well as a function of burnup interval. APSR limits are also provided as a function of burnup interval.

4. TS 3.5.2.5.4 Control Rod Positions
   The power level cutoff requirements have been deleted.

5. TS 3.5.2.6 Reactor Power Imbalance
   Power-imbalance curves for Cycle 8 as a function of burnup interval are added.

6. TS 3.5
   The basis has been modified to include the Cycle 8 power-imbalance curve and the maximum allowed linear heat rate as a function of burnup interval that meets the FAC on ECCS.

7. TS 4.7.1
   This TS has been changed to reflect the removal of a rod bow penalty on DNBR margin. However, since the rod bow penalty is no longer required based on an approved B&W rod bow topical report (Ref. 12), the original Cycle 1 control rod insertion time is proposed by the licensee. We conclude that this change is acceptable for the reason cited above.

2.7 Startup Testing

We have reviewed the startup physics testing program for ANO-1 Cycle 8 presented in Reference 2. We conclude that this program is acceptable since it will provide confirmation that measurements for the as-loaded core conform to the Cycle 8 nuclear design and since the data required by the Technical Specifications will be satisfied.

2.8 Conclusions

We have reviewed the fuel system design, nuclear design, thermal-hydraulic design, and the transient and accident analysis information presented in the ANO-1 Cycle 8 Reload Report. We conclude that the proposed reload and associated modified Technical Specifications are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 24, 1986

Principal Contributors: D. Fieno, G. Vissing

REFERENCES

1. T. Gene Campbell (AP&L) letter to J. F. Stolz (NRC) on "Cycle 8 Reload Report and Proposed Technical Specification Change Request," dated September 10, 1986.

2. "Arkansas Nuclear One, Unit 1 - Cycle 8 Reload Report," BAW-1918, Babcock & Wilcox Company Report, August 1986 (transmitted with Reference 1 above).

3. Guy S. Vissing (NRC) letter to John M. Griffin (AP&L) transmitting Amendment No. 92 to Facility Operating License No. DPR-51, December 20, 1984.

4. D. C. Trimble (AP&L) letter to R. W. Reid (NRC) dated November 6, 1980, transmitting BAW-1626.

5. Robert W. Reid (NRC) letter to William Cavanaugh III (AP&L) on Amendment No. 52, Cycle 5, dated March 9, 1981.

6. Standard Review Plan, Section 4.2, Rev. 1, "Fuel System Design," U. S. Nuclear Regulatory Commission Report, NUREG-0800, July 1981.

7. "TACO2 - Fuel Pin Performance Analysis," Y. Hsii, et al., BAW-10141P-A, Rev. 1, June 1983.

8. "ECCS Analysis of B&W's 177-FA Lowered-Loop NSSS, W. L. Bloomfield, et al., BAW-10103, Revision 1, September 1975.

9. "Babcock & Wilcox Version of PDQ-User's Manual," Hassan, H. H., et al., BAW-10117P-A, Babcock & Wilcox, January 1977.

10. "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," Mays, C. W., et al., BAW-10152, Babcock & Wilcox, September 1984.

11. NRC memorandum from L. S. Rubenstein to D. M. Crutchfield on review of the NOODLE code, April 10, 1985.

12. "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," BAW-10147P-A, Rev. 1, Babcock & Wilcox Company, May 1983.

13. "ANO-1 Power Level Cutoff Removal Analysis," P. L. Holman and B. J. Delano Babcock & Wilcox Company report, May 1986 (transmitted by letter from J. Ted Enos (AP&L) to John F. Stolz (NRC), September 19, 1986).

14. J. Ted Enos (AP&L) letter to J. F. Stolz (NRC) on "ANO-1 Cycle 8 Reload Report," dated September 19, 1986.

15. J. Ted Enos (AP&L) letter to J. F. Stolz (NRC) on "ANO-1 Cycle 8 Reload Report (BAW-1918)," dated November 7, 1986.