October 29, 1990

Docket No. 50-313

Mr. Neil S. Carns
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. 77288)

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 8, 1990.

The amendment revises portions of TS 2.1 (Safety Limits, Reactor Core), 2.3 (Limiting Safety System Settings, Protective Instrumentation), 3.1.7 (Moderator Temperature Coefficient of Reactivity), and 3.5.2 (Control Rod Group and Power Distribution Limits). The changes support the reload for Cycle 10.

The staff notes that your application committed to submit to the NRC the results of a reanalysis of the steam line break accident to assure that the end-of-cycle moderator temperature coefficient limit is adequate and will not be violated during Cycle 10 operation. Your application committed to submit these results early in Cycle 10. The staff finds this commitment to be acceptable.

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

Sincerely,

ORIGINAL SIGNED BY:

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 137 to DPR-51
2. Safety Evaluation

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OFC	:PD4-1/LA	:PD4-1/PM	:OGC	:PD4-1/D (A)	:	:	:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Neil S. Carns
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated August 8, 1990, 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 license 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.

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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:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 137 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Director
Project Directorate IV-1
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Revise 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 PAGES	INSERT PAGES
iv	iv
v	v
9b	9b
14b	14b
30	30
47	47
48	48
48b	48b
48c	48c
48d	48d
48e	48e
48f	48f
48g	48g
48h	48h
48i	48i
48j	48j
48k	48k
48l	48l
48m	48m
48n	48n

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2.1-2	CORE PROTECTION SAFETY LIMITS	9b
2.1-3	CORE PROTECTION SAFETY LIMITS	9c
2.3-1	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINT	14a
2.3-2	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS	14b
3.1.2-1	REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITATIONS	20a
3.1.2-2	REACTOR COOLANT SYSTEM NORMAL OPERATION-HEATUP LIMITATIONS	20b
3.1.2-3	REACTOR COOLANT SYSTEM, NORMAL OPERATION COOLDOWN LIMITATIONS	20c
3.1.9-1	LIMITING PRESSURE VS. TEMPERATURE FOR CONTROL ROD DRIVE OPERATION WITH 100 STD CC/LITER H_2O	33
3.2-1	BORIC ACID ADDITION TANK VOLUME AND CONCENTRATION VS. RCS AVERAGE TEMPERATURE	35a
3.5.2-1A	ROD POSITION SETPOINTS FOR FOUR-PUMP OPERATION FROM 0 to 30 +10/-0 EFPD - ANO-1 CYCLE 10	48b
3.5.2-1B	ROD POSITION SETPOINTS FOR FOUR-PUMP OPERATION FROM 30 +10/-0 TO 335 +10/-10 EFPD - ANO-1 CYCLE 10	48c
3.5.2-1C	ROD POSITION SETPOINTS FOR FOUR-PUMP OPERATION AFTER 335 +10/-10 EFPD - ANO-1 CYCLE 10	48d
3.5.2-2A	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION FROM 0 TO 30 +10/-0 EFPD - ANO-1 CYCLE 10	48e
3.5.2-2B	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION FROM 30 +10/-0 TO 335 +10/-10 EFPD - ANO-1 CYCLE 10	48f
3.5.2-2C	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION AFTER 335 +10/-10 EFPD - ANO-1 CYCLE 10	48g

3.5.2-3A	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION FROM 0 TO 30 +10/-0 EFPD - ANO-1 CYCLE 10	48h
3.5.2-3B	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION FROM 30 +10/-0 TO 335 +10/-10 EFPD - ANO-1 CYCLE 10	48i
3.5.2-3C	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION AFTER 335 +10/-10 EFPD - ANO-1 CYCLE 10	48j
3.5.2-4A	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION FROM 0 TO 30 +10/-0 EFPD - ANO-1 CYCLE 10	48k
3.5.2-4B	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION FROM 30 +10/-0 TO 335 +10/-10 EFPD - ANO-1 CYCLE 10	48l
3.5.2-4C	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION AFTER 335 +10/-10 EFPD - ANO-1 - CYCLE 10	48m
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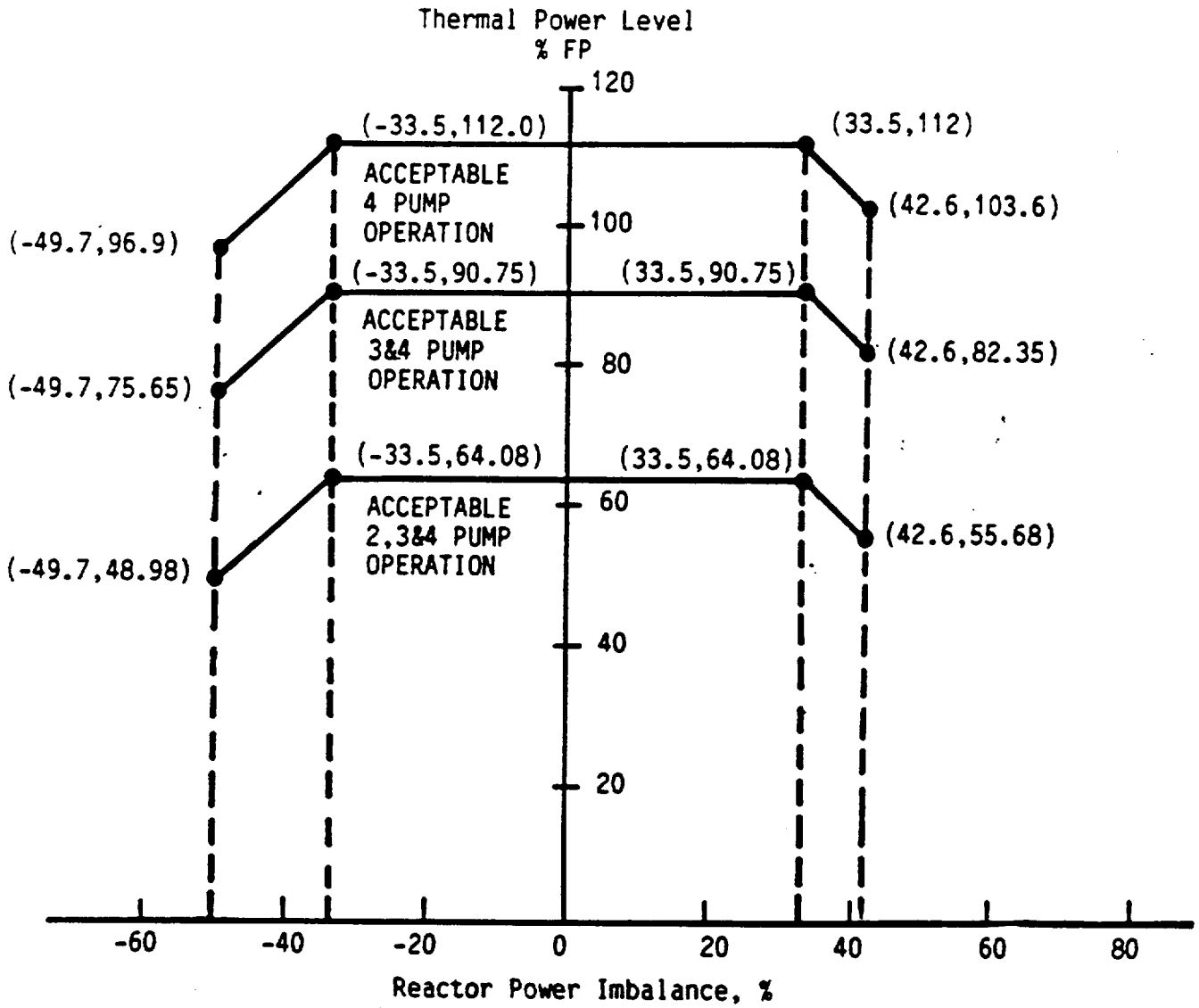
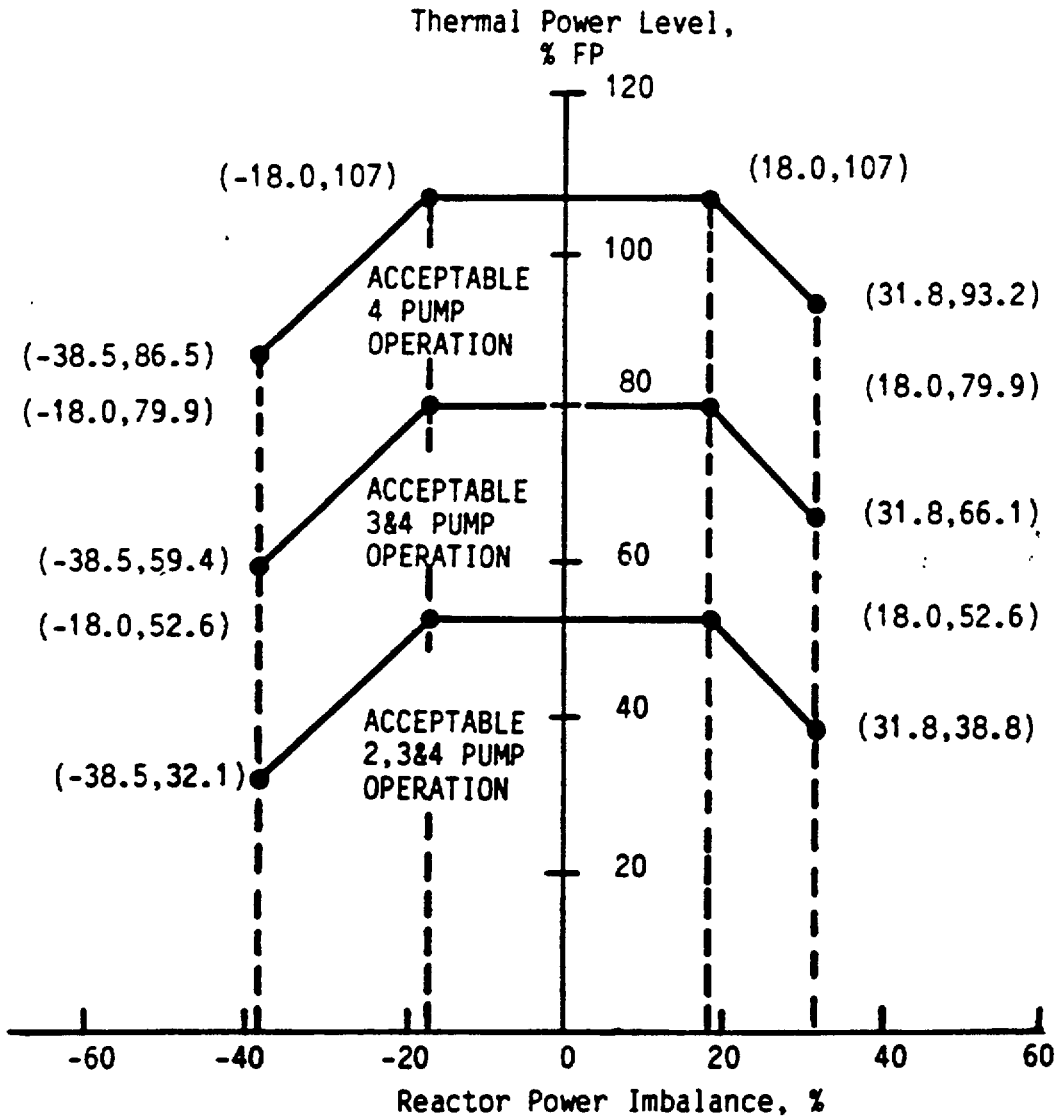
3.24-1	HYDROGEN LIMITS FOR ANO-1 WASTE GAS SYSTEM	110bc
4.4.2-1	NORMALIZED LIFTOFF FORCE - HOOP TENDONS	85b
4.4.2-2	NORMALIZED LIFTOFF FORCE - DOME TENDONS	85c
4.4.2-3	NORMALIZED LIFTOFF FORCE - VERTICAL TENDONS	85d

Core Protection Safety Limits - ANO-1
Figure 2.1-2



Protective System Maximum Allowable Setpoints
ANO-1, Figure 2.3-2



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 ≥95% of rated thermal power and shall be less positive than 0.9×10^{-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 limits 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. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k/^{\circ}F$.

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 in excess of 4.24%.

   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.

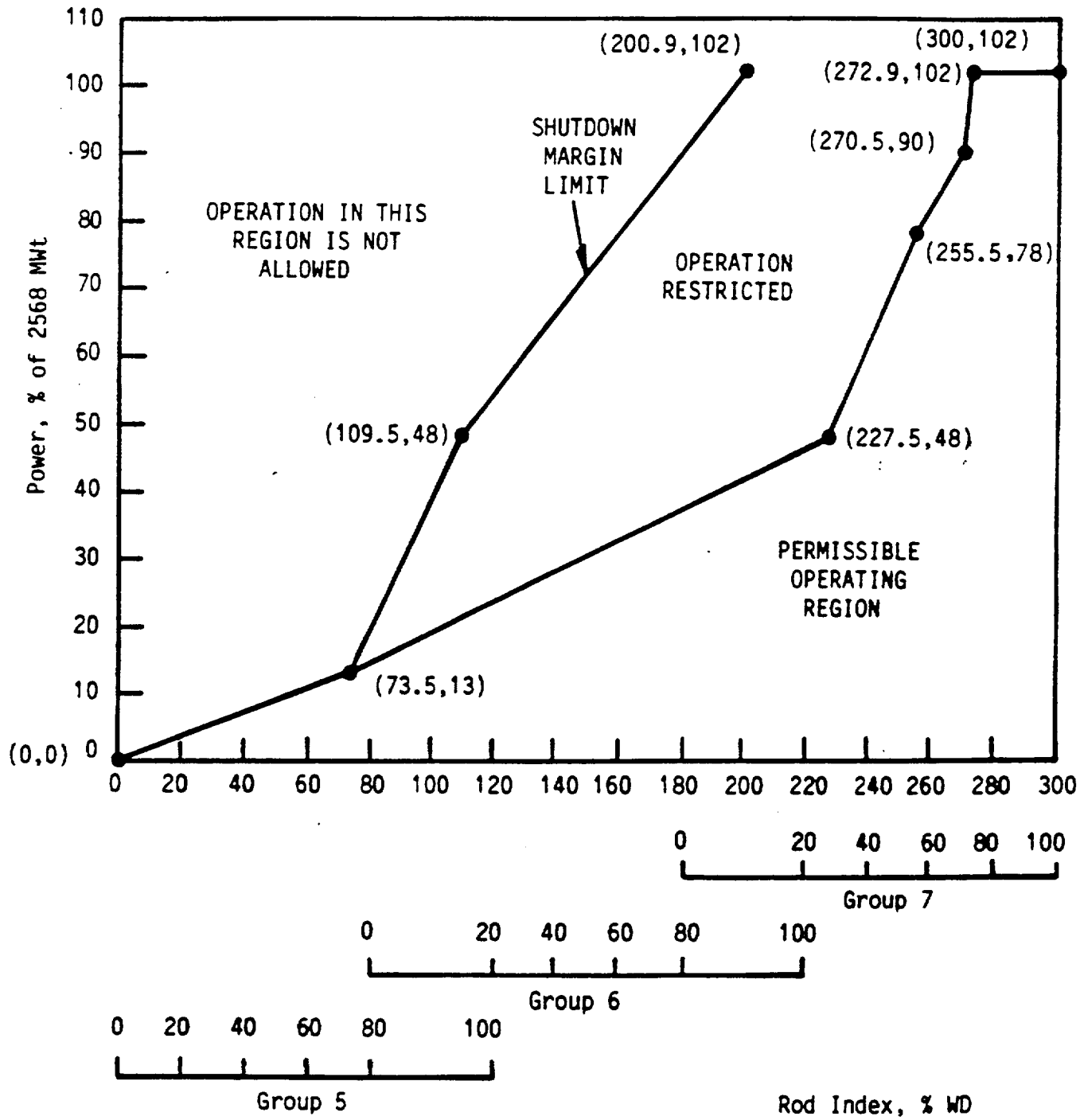
3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-4(A-C). If the imbalance is not within the envelope defined by Figure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

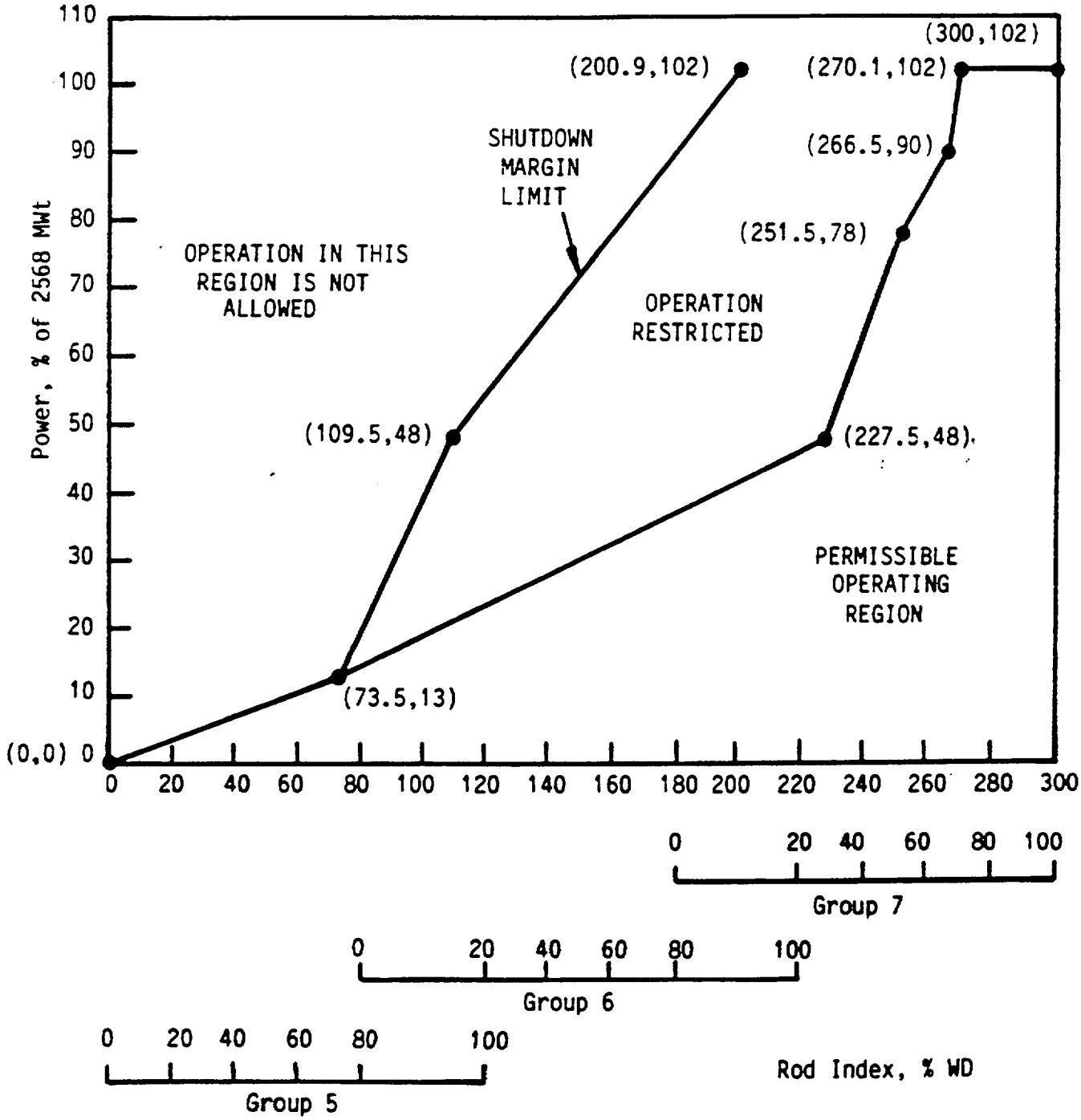
The power-imbalance envelope defined in Figure 3.5.2-4(A-C) is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the 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

Rod Position Setpoints for 4-Pump Operation
From 0 to 30 +10/-0 EFPD ANO-1 Cycle 10
Figure 3.5.2-1A

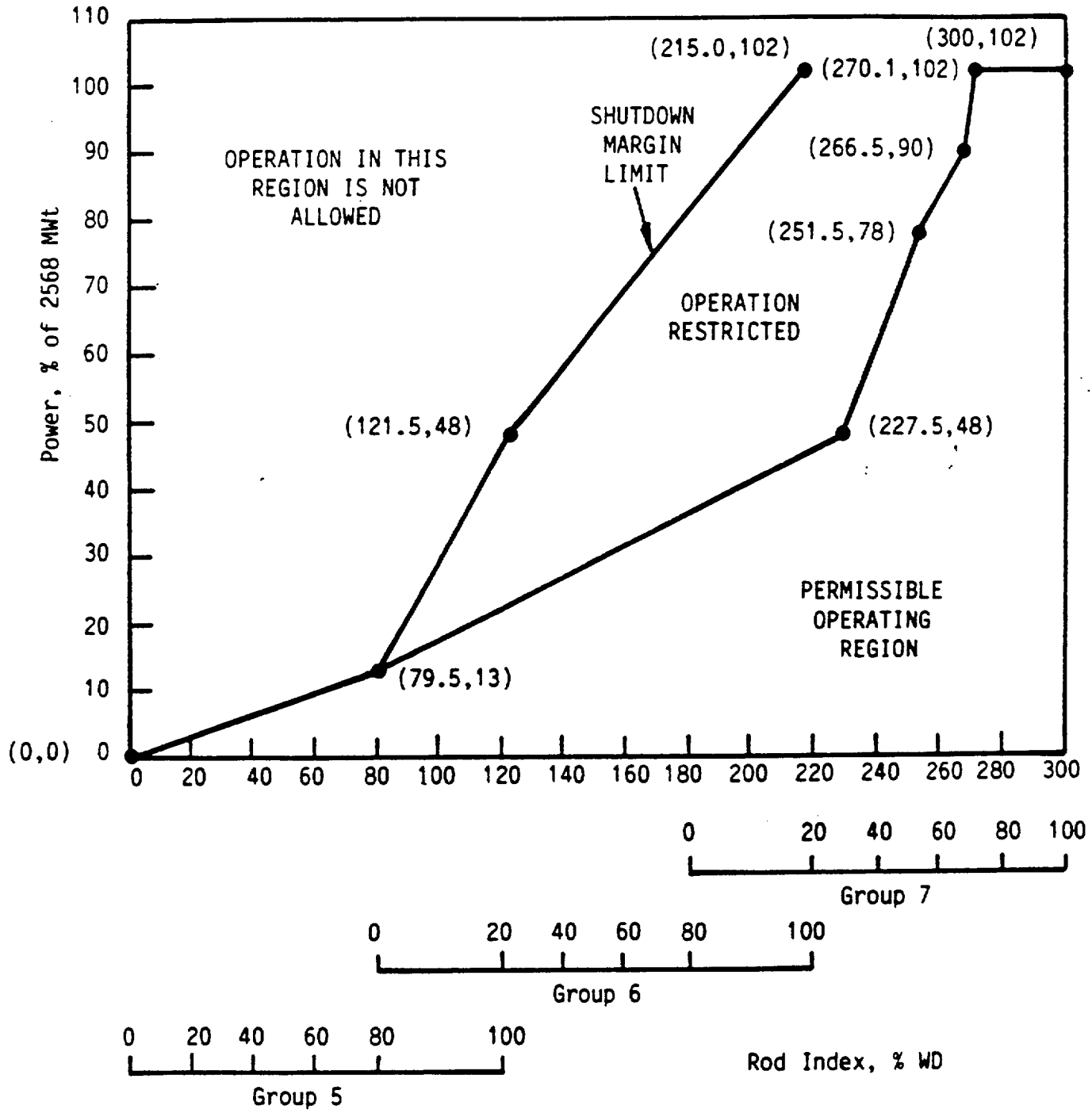


Amendment No. 6, 21, 41, 43, 52, 71, 92, 105, 48b
113, 137,

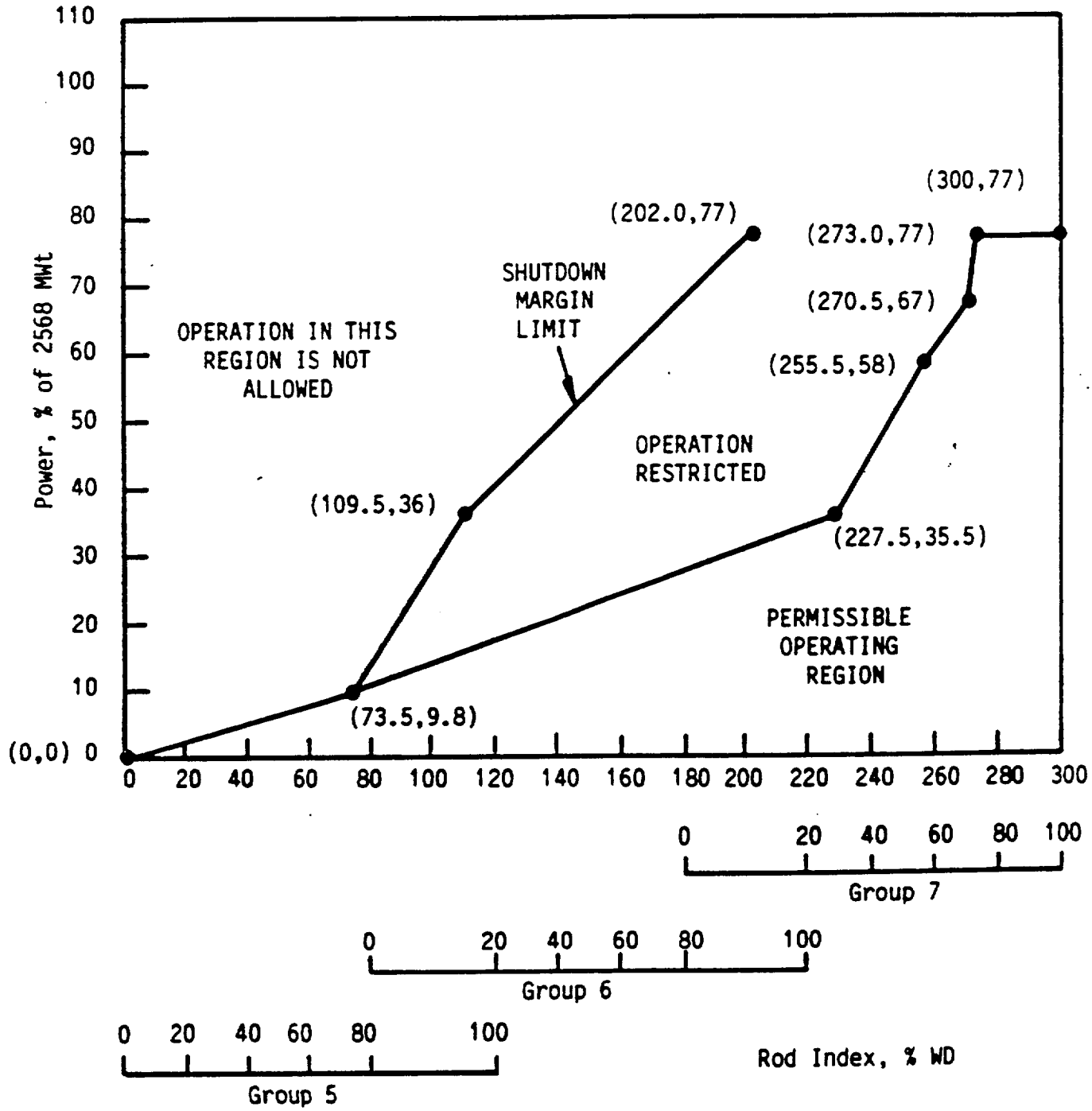
Rod Position Setpoints for 4-Pump Operation
From 30 +10/-0 to 335 +10/-10 EFPD ANO-1 Cycle 10
Figure 3.5.2-1B



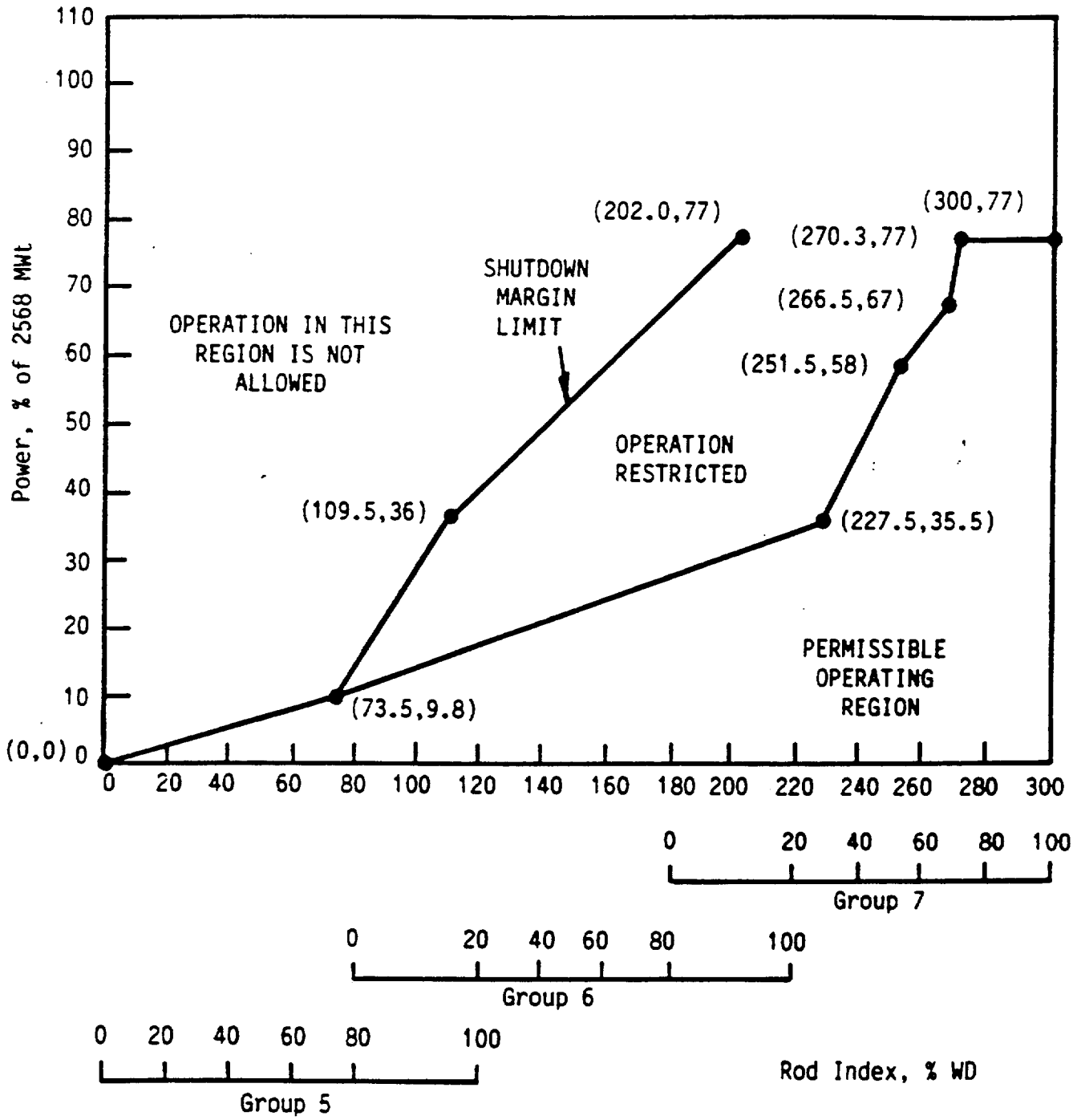
Rod Position Setpoints for 4 - Pump Operation
After 335 +10/-10 EFPD ANO-1 Cycle 10
Figure 3.5.2-1C



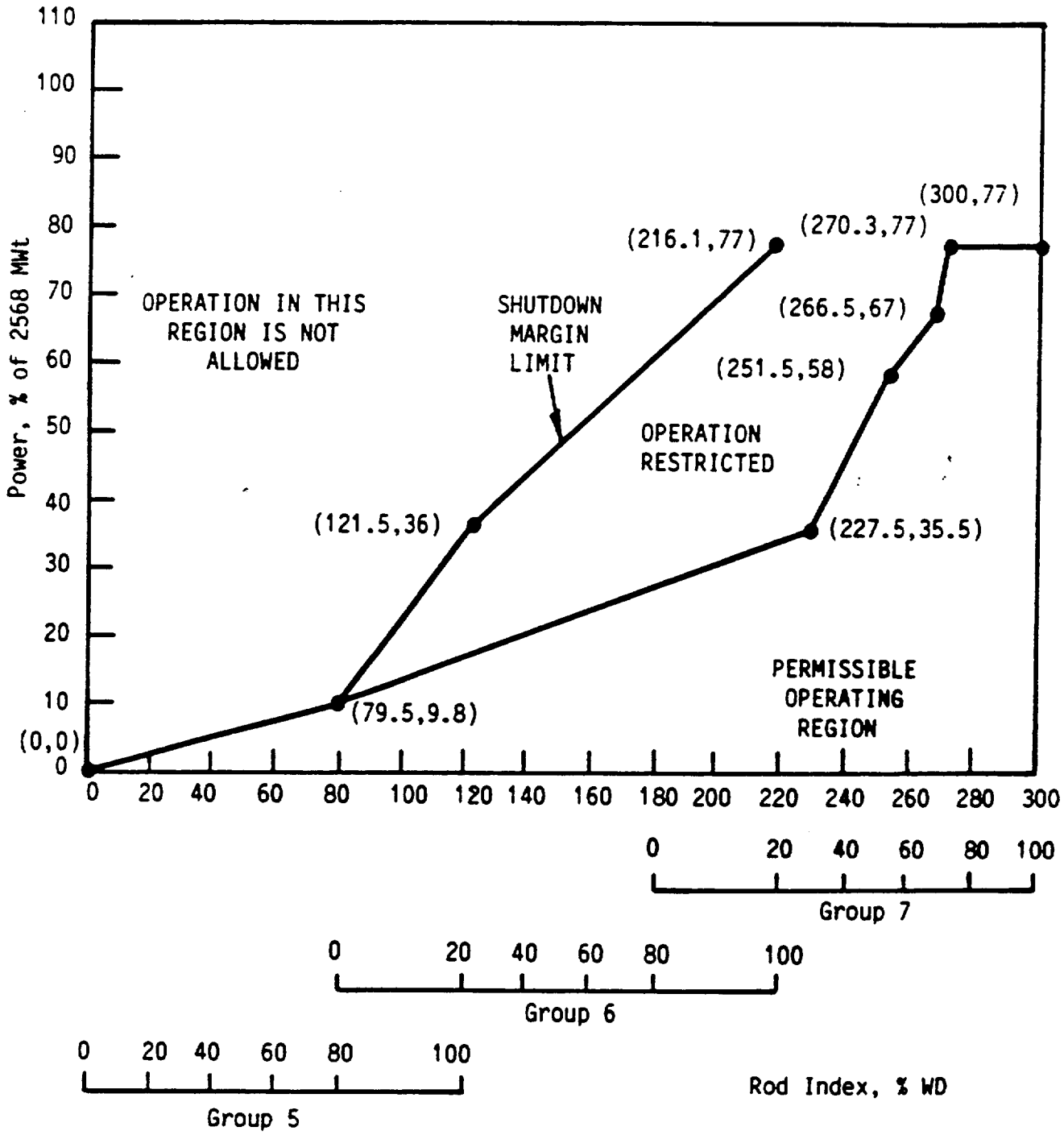
Rod Position Setpoints for 3-Pump Operation
From 0 to 30 +10/-0 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-2A



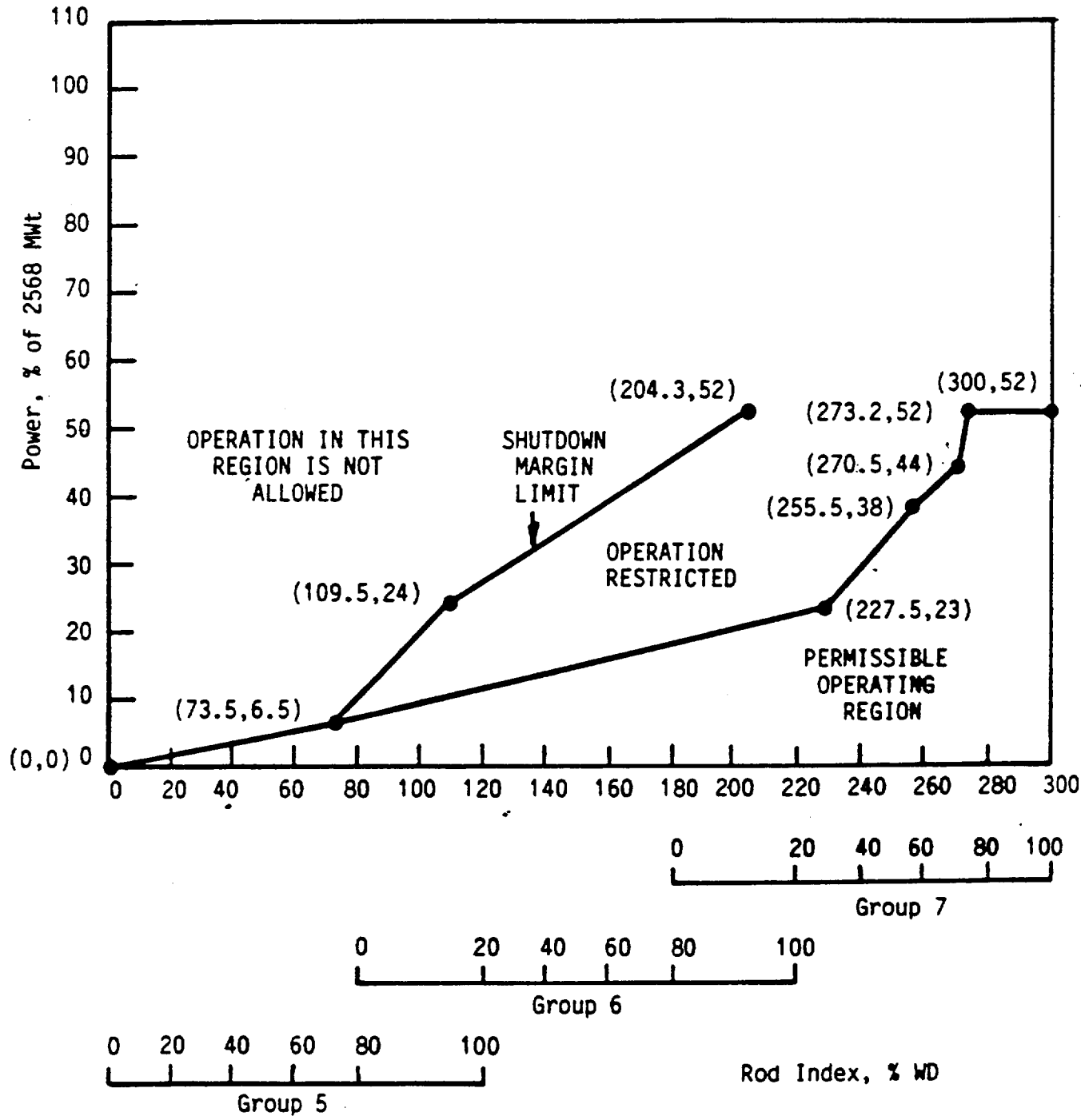
Rod Position Setpoints for 3-Pump Operation
From 30 +10/-0 to 335 +10/-10 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-2B



Rod Position Setpoints for 3-Pump Operation
After 335 +10/-10 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-2C

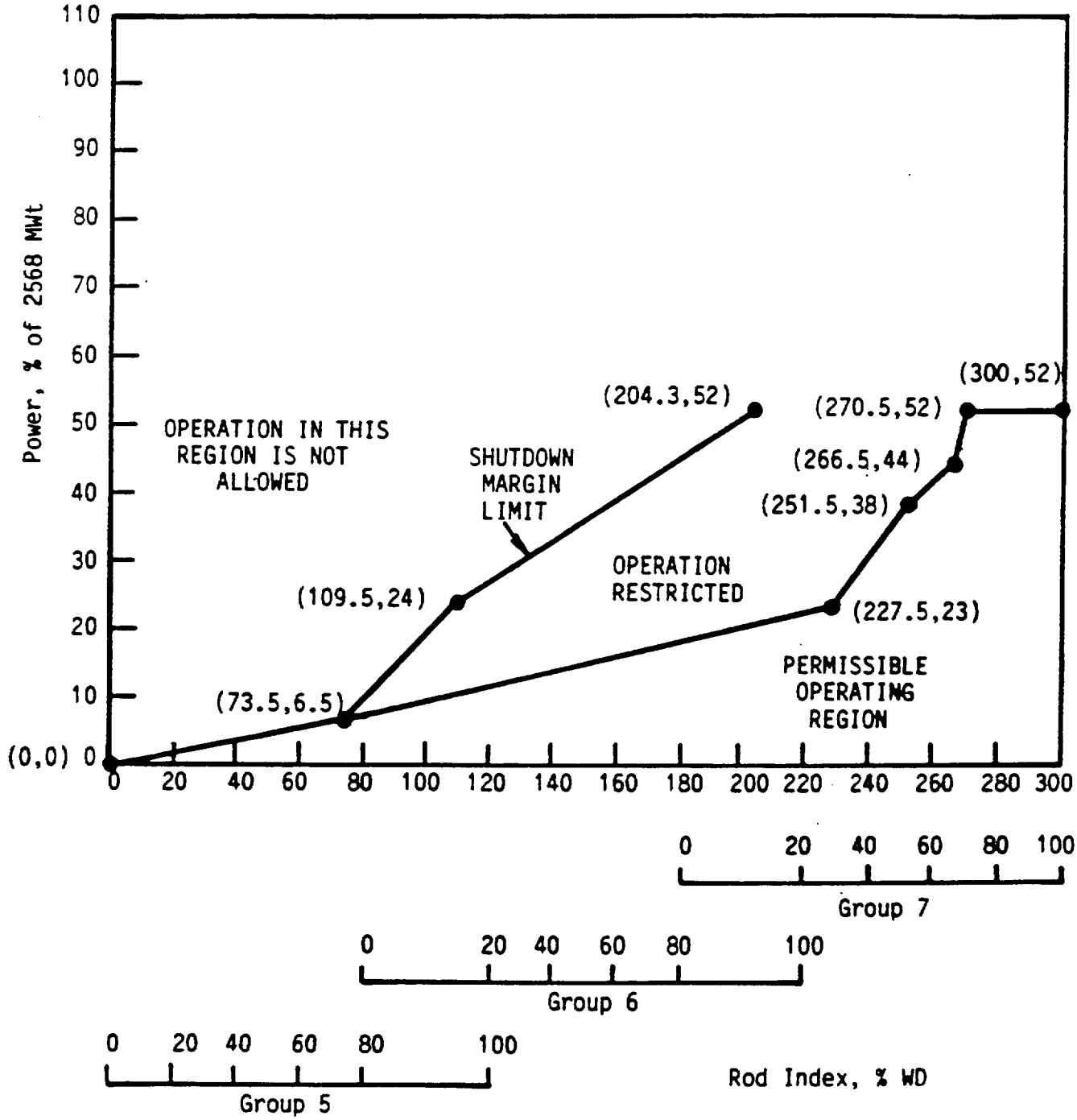


Rod Position Setpoints for 2-Pump Operation
From 0 to 30 +10/-0 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-3A

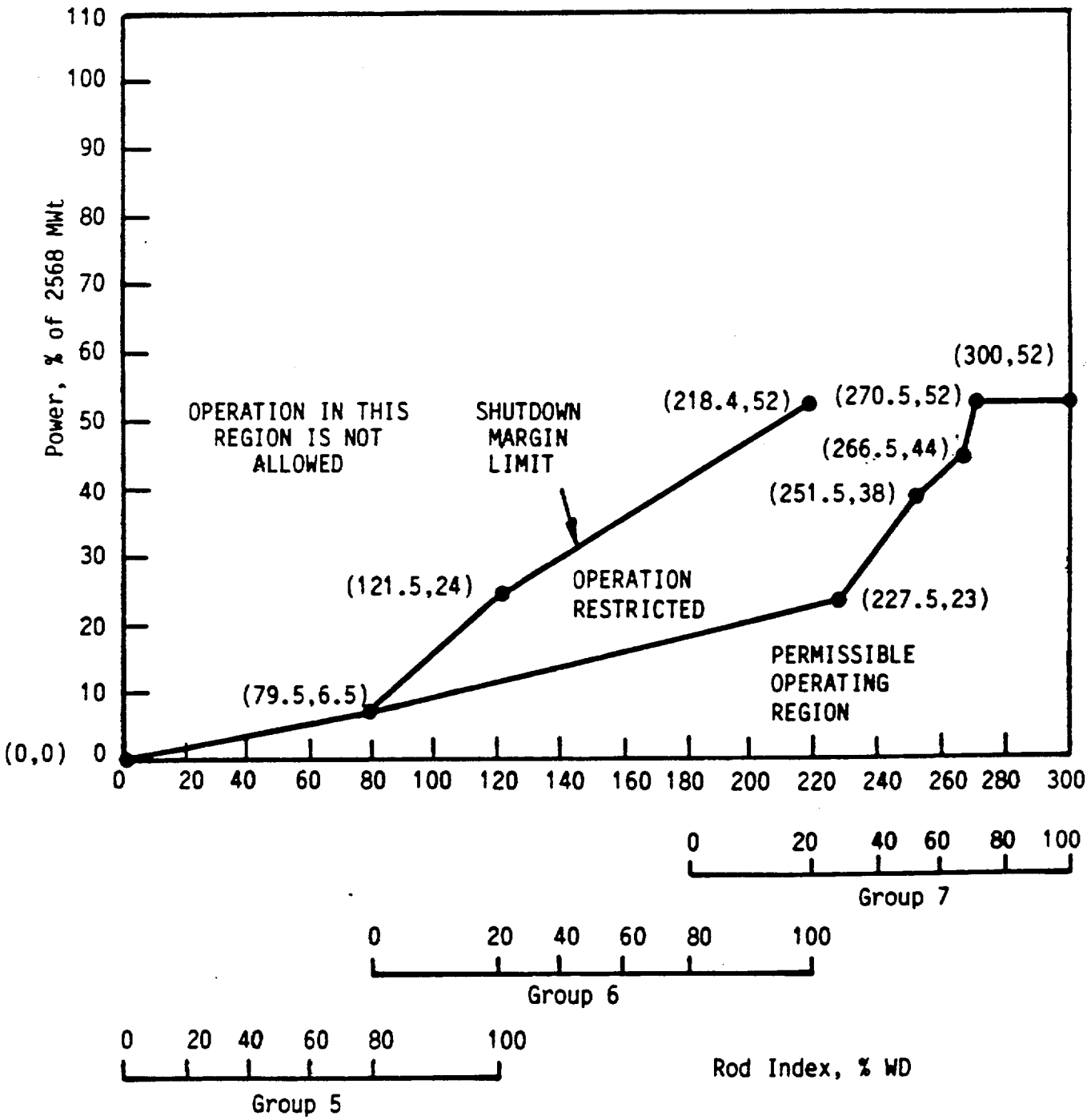


Amendment No. 21, 31, 43, 52, 71, 92, 105, 113, 137, 48h

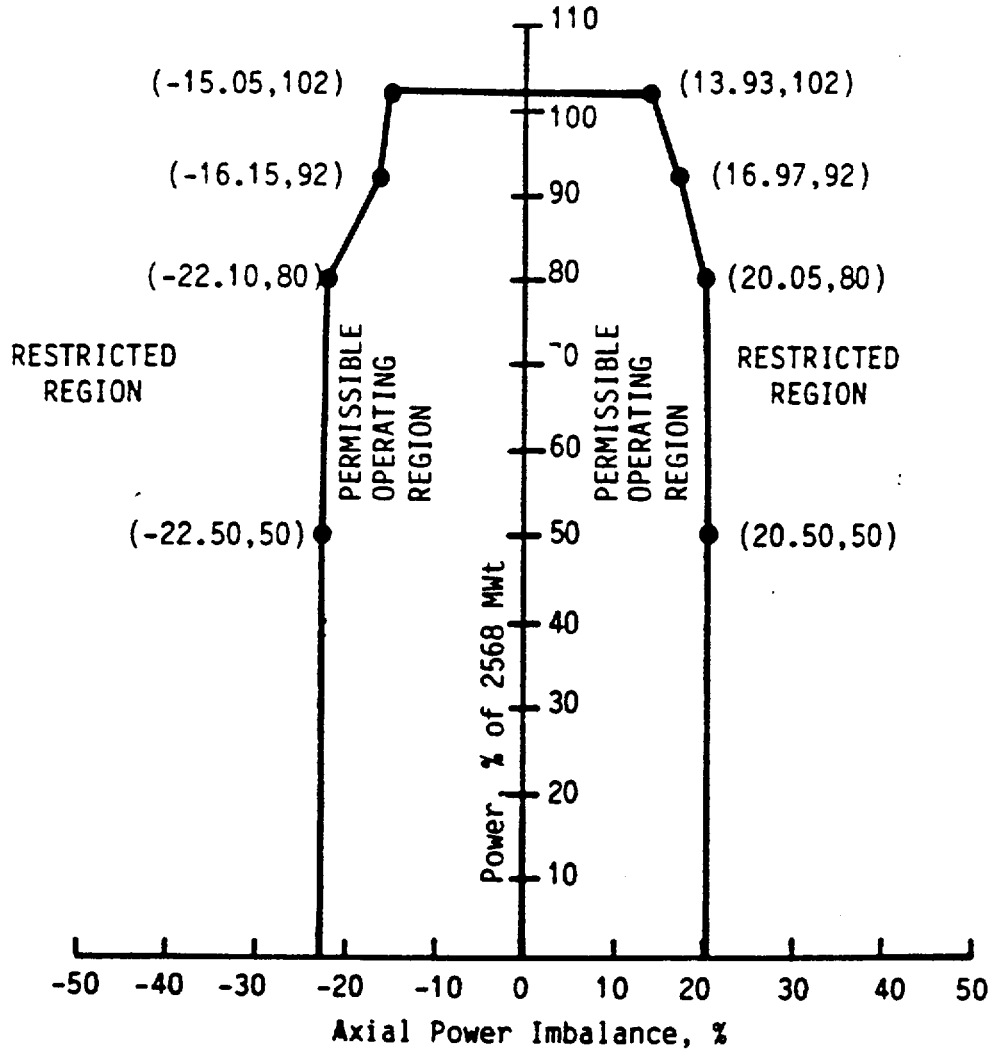
Rod Position Setpoints for 2-Pump Operation
From 30 +10/-0 to 335 +10/-10 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-3B



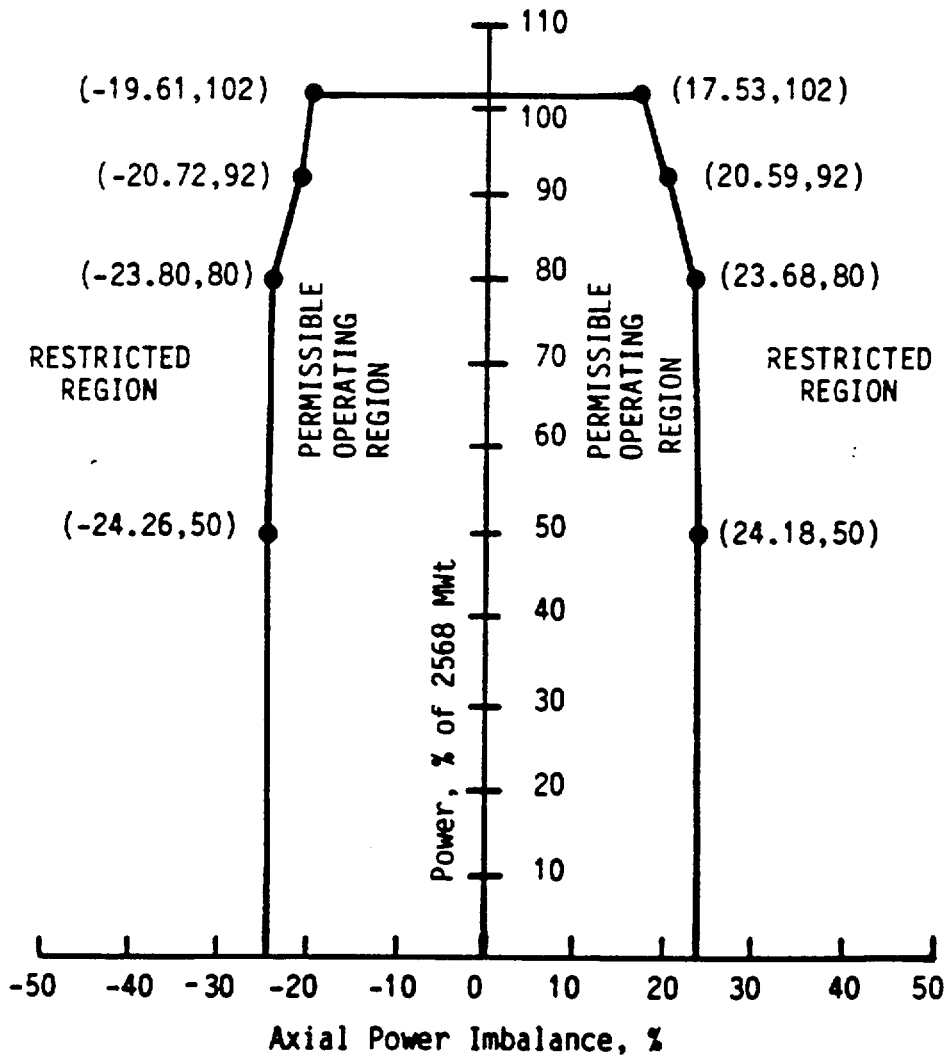
Rod Position Setpoints for 2-Pump Operation
After 335 +10/-10 EFPD -- ANO-1 Cycle 10
Figure 3.5.2-3C



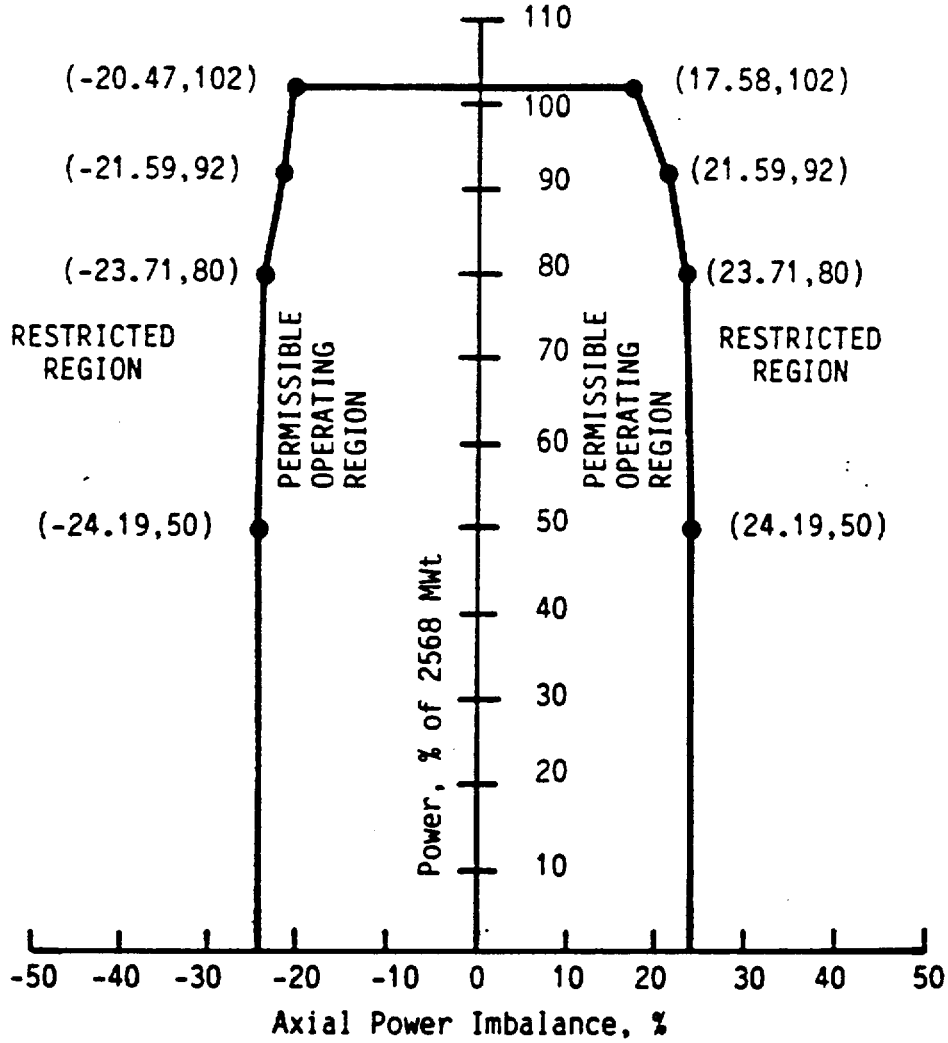
Operational Power Imbalance Setpoints for Operation
From 0 to 30 +10/-0 EFPD -- ANO-1, Cycle 10
Figure 3.5.2-4A



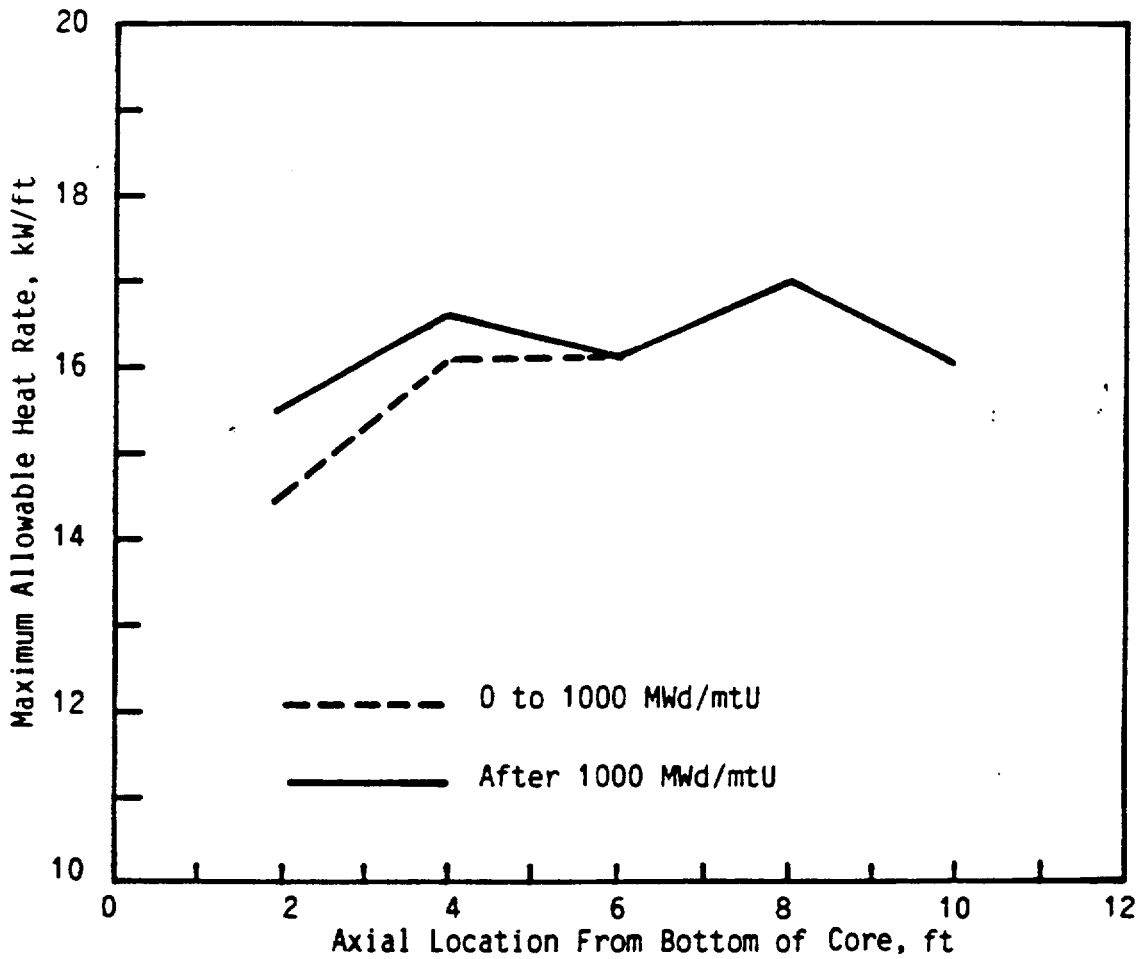
Operational Power Imbalance Setpoints for Operation
From 30 +10/-0 to 335 +10/-10 EFPD -- ANO-1, Cycle 10
Figure 3.5.2-4B



Operational Power Imbalance Setpoints for Operation
After 335 +10/-10 EFPD -- ANO-1, Cycle 10
Figure 3.5.2-4C



LOCA Limited Maximum Allowable
Linear Heat Rate -- ANO-1 Cycle 10
Figure 3.5.2-5



Amendment No. 43, 52, 71, 92, 103, 48n
113, 137,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137

TO FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated August 8, 1990, Entergy Operations, Inc., the licensee, submitted proposed Technical Specification (TS) changes for Cycle 10 operation of Arkansas Nuclear One, Unit 1 (ANO-1). A Cycle 10 Reload Report, BAW-2114, dated June 1990, was also submitted to support Cycle 10 operation and TS changes.

2.0 EVALUATION

2.1 Fuel Design

Cycle 10 will contain 1 Batch 6E (Mark-B4), 60 Batch 10B (Mark-B4), 60 Batch 11 (Mark-B6), and 56 Batch 12 (Mark-B8) assemblies. All of these fuel assemblies are mechanically interchangeable. The Mark-B8 fuel incorporates slightly longer fuel rods and a shorter lower end fitting. The longer fuel rods have increased plenum volume allowing for higher fuel burnup. The lower end plug solid portion was also extended in length and the lower spacer grid location was dropped causing the solid end plug to extend through the lower spacer grid. The intention of this change is to trap any debris capable of fuel rod fretting below the bottom spacer grid where the solid lower end plug would prevent failure.

Because of the previous incore exposure of Batch 10B fuel, it is the most limiting in terms of cladding creep collapse. The licensee has stated that the cladding collapse time for the most limiting Cycle 10 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 10 assembly. The methods and procedures used for the analyses have been previously reviewed and approved by the staff. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 10 operation.

The cladding stress and strain analyses for the Cycle 10 fuel designs were calculated using methods and limits previously reviewed and approved by the NRC. The staff concludes that the analysis of cladding stress and strain has been appropriately considered for Cycle 10 operation and is acceptable.

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The thermal behavior of all fuel in the Cycle 10 core is virtually identical. The thermal analysis was performed with the approved TACO2 code and the Cycle 10 core protection limits were based on the calculated linear heat rate (LHR) to centerline fuel melt limit of 20.5 kw/ft. This limiting value is satisfactorily incorporated into the TS for Cycle 10 through the operating limits on rod index and axial power imbalance.

Standard Review Plan (SRP) 4.2, Section II.A.1(f), contains the requirement that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on TACO2 analyses, the licensee has stated that the internal pressure in the highest burnup rod of each fuel type will not reach the nominal reactor coolant system (RCS) pressure of 2200 psia. The staff finds this acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 10 operation.

Based on its review, the staff concludes that approved methods have been used, that the fuel design parameters meet applicable criteria and that the fuel design for ANO-1 Cycle 10 is acceptable.

2.2 Nuclear Design

The nuclear design parameters characterizing the ANO-1 Cycle 10 core have been computed by methods previously used and approved for Babcock and Wilcox (B&W) reactors. Comparisons between the parameters for Cycle 9 and Cycle 10 have shown little change. Shutdown margin calculations for Cycle 10 include the effects of poison material depletion, a 10% calculational uncertainty, allowance for rod bite, and neutron flux redistribution as well as a maximum worth stuck rod. Beginning-of-cycle (BOC) and end-of-cycle (EOC) shutdown margins show that adequate reactivity worth exists above the total required worth during the cycle. Shutdown margins at BOC and EOC are 4.02% delta k/k and 3.34% delta k/k, respectively, compared to the minimum required value of 1.0% delta k/k.

Based on its review, the staff concludes that approved methods have been used, that the nuclear design parameters meet applicable criteria, and that the nuclear design of ANO-1 Cycle 10 is acceptable.

2.3 Thermal-Hydraulic Design

Cycle 10 is the second cycle in which a mixture of Mark B and Mark BZ fuel assemblies coexist in the core. Although a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions, incompatibility in the hydraulic characteristics has an effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies coexist in the core. Since the Mark BZ assemblies have a higher hydraulic resistance due to the burnable poison rod assembly (BPRA) retainers and the Zircaloy intermediate spacer grids, some of the coolant flow is diverted from the Mark BZ fuel to the lower power Mark B fuel. The fact that the Mark BZ assemblies have less flow

in a mixed core results in lower maximum allowable power peaking and a lower enthalpy rise factor required in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. A cycle-specific analysis which modeled the actual Cycle 10 core configuration and bypass flow value has shown that the Cycle 9 thermal-hydraulic reference analysis remains applicable for Cycle 10 operation.

2.4 Accident and Transient Analysis

The important physics, thermal-hydraulic, and kinetics parameters for Cycle 10 have been compared to the values used in the previous cycle, the FSAR, and/or the fuel densification report. The licensee has shown that the Cycle 10 values are bounded by those previously used and, therefore, the transient and accident evaluation of Cycle 10 are considered to be bounded by previously accepted analyses.

Babcock and Wilcox (B&W) has performed a generic loss-of-coolant accident (LOCA) analysis for the B&W 177FA lowered-loop nuclear steam supply system (NSSS) using the final acceptance criteria emergency core cooling system (ECCS) evaluation model, updated with an upgraded fuel performance model and the B&W modified version of FLECSET. The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used is conservative relative to those calculated for this cycle. Two sets of bounding values for allowable LOCA peak LHRs for Cycle 10 are given as a function of core height. These limits apply during the periods from 0 to 1000 MWD/MTU and for the balance of the cycle. The B&W analysis has been approved by the NRC and the LHR limits are satisfactorily incorporated into the Technical Specifications for Cycle 10.

The most negative moderator temperature coefficient (MTC) occurs at EOC and is based on the steam line break accident. Because of longer fuel cycles, some B&W plants are approaching their predicted EOC limit on MTC. Evaluations made by B&W have revealed a small nonconservative (positive) bias in the MTC calculational methodology which has not been incorporated into the Cycle 10 reload analyses. To account for this in Cycle 9, Entergy Operations has administratively limited operations to a minimum soluble boron concentration sufficient to conservatively ensure that MTC will not exceed the limiting value used in the safety analyses. However, the steam line break event is being reanalyzed for Cycle 10 using the latest approved methodology. The results of similar analyses performed for other plants using the newer methodology have shown a significant relaxation in the EOC MTC limit, compared to the FSAR results, which more than compensated for the MTC calculational bias mentioned above. The results of this reanalysis and its effect on the EOC MTC limits will be submitted to the NRC well before the Cycle 10 EOC MTC limit is approached. Based on this, initial Cycle 10 operation with the current negative MTC limit is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

ANO-1 Technical Specifications have been modified for Cycle 10 operation to reflect changes in core reactivity, power peaking, and control rod worths. The specific changes are evaluated below.

(1) The Core Protection Safety Limits specified in Figure 2.1-2 are modified.

The changes reflect the Cycle 10 core design which has been reviewed and approved by the staff in this safety evaluation report.

(2) The Protective System Maximum Allowable Setpoints specified in Figure 2.3-2 are modified.

The changes reflect the Cycle 10 core design which has been reviewed and approved by the staff in this safety evaluation report.

(3) The Moderator Temperature Coefficient (MTC) specified in TS 3.1.7 would be increased from $+0.5 \times 10^{-4}$ delta k/k/°F to $+0.9 \times 10^{-4}$ delta k/k/°F at power levels less than or equal to 95 percent of rated thermal power (RTP). The MTC would still be required to be nonpositive at power levels above 95 percent of RTP.

The most limiting accident adversely affected by the positive MTC is the startup accident in which an uncontrolled addition of reactivity is caused by a rapid withdrawal of control rods from a subcritical or low power condition. A positive MTC would yield the maximum peak heat flux. The accident analyses presented in the FSAR included $+0.9 \times 10^{-4}$ delta k/k/°F in a sensitivity study which examined the effects of variations in the MTC for all pertinent accidents including the startup accident. The resulting peak thermal power and reactor coolant system peak pressure remained well below the acceptance criteria. In addition, no adverse impact is expected on the hydraulic and neutronic stability of ANO-1 due to the larger positive MTC. The hydraulic and neutronic stability of ANO-1 is not expected to be different from the other B&W reactors such as Crystal River 3, Oconee 1, 2, and 3, Rancho Seco, Davis-Besse, and Three Mile Island 1, which are already licensed for an MTC limit of $+0.9 \times 10^{-4}$ delta k/k/°F from 0 to 95 percent of RTP. The staff concludes that the proposed increase in the MTC limit for power levels less than 95 percent of rated power has been adequately considered and is bounded by the existing FSAR safety analyses and is, therefore, acceptable.

(4) TS Section 3.5.2.4 would be modified to reflect a quadrant power tilt limit of 4.24%.

The quadrant power tilt revision is based on incore detector sensitivity depletion. The tilt specified is the bounding value, derived from the detector sensitivity depletion at EOC 10, of the full incore quadrant tilt setpoint values determined for Cycle 10. The change is, therefore, acceptable.

The staff notes, however, that there are no required completion times specified for the power reduction if the quadrant tilt exceeds 4.24% (TS 3.5.2.4.1) or for the shutdown requirement if the quadrant tilt exceeds 25% (TS 3.5.2.4.3). We recommend that 2 hours and 6 hours, respectively, be inserted to conform to present accepted practice.

(5) TS Section 3.5.2.5.4 and associated Figures 3.5.2-1 (A-C), 3.5.2-2 (A-C), and 3.5.2-3 (A-C) would be modified to accommodate Cycle 10 changes to the control rod insertion limits and operational limits on the gray axial power shaping rods (APSRs).

The revised rod insertion limits provide assurance of achieving hot shutdown by reactor trip at any time during Cycle 10 and ensure that power peaking criteria are not exceeded. These limits also preclude insertion of rod groups which could result in any single rod worth greater than the safety analysis assumption for the rod ejection accident. The gray APSR position limits have been derived using worst case conditions and time of core life such that core power peaking limits are not violated. These revised limits were derived with NRC approved methods and computer codes and are, therefore, acceptable.

(6) TS Figures 3.5.2-4 (A-C) power imbalance limits would be modified.

The power imbalance envelope defined in the revised Figures is based on LOCA analyses which have defined the maximum linear heat rate (Figure 3.5.2-5) to assure that the maximum cladding temperature will not exceed the final acceptance criteria in 10 CFR 50, Appendix K, assuming worst case power distributions. These revised setpoints have been derived using NRC approved methodology and computer codes, taking into account all perceived uncertainties, worst case conditions and core burnup. The proposed changes are, therefore, acceptable.

(7) TS Figure 3.5.2-5 would be modified to provide provide new loss-of-coolant accident (LOCA) linear heat rate (LHR) limits for Cycle 10.

The revised LHRs are based on a worst case range of power distributions, power shapes and power peaking factors within the core during Cycle 10 which result in the most severe calculated consequences for the spectrum of postulated accidents including LOCA. These revised LHRs continue to provide assurance that the fuel rod cladding temperature remains below the final acceptance criteria of 10 CFR 50, Appendix K. The revised values have been derived using NRC approved ECCS evaluation codes and methodology and are, therefore, acceptable.

4.0 SUMMARY

The NRC staff has reviewed the proposed ANO-1 TS changes submitted by letter dated August 8, 1990, from Entergy Operations, Inc., to support operation for a 10th cycle and finds them acceptable.

The results of a reanalysis of the steam line break accident will be submitted to the NRC early in Cycle 10 to assure that the EOC MTC limit is adequate and will not be violated during Cycle 10 operation.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has 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 exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 the amendment.

6.0 CONCLUSION

The staff has 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 29, 1990

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