November 8, 1988

Docket No. 50-313

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

Dear Mr. Campbell:

SUBJECT: ISSUANCE OF AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. 69056)

The Commission has issued the enclosed Amendment No. 113 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 in response to your applications dated July 20 and August 31, 1988.

The amendment permits operation of ANO-1 for Cycle 9 and modifies the variable low pressure reactor trip setpoint.

A copy of our related Safety Evaluation and the Notice of issuance are also enclosed.

Sincerely,
/s/
C. Craig Harbuck, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.113 to DPR-51
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	BGrimes	PNoonan (3)	ACRS (10)
NRC PDR	TBarnhart (4)	CHarbuck	GPA/PA
Local PDR	Wanda Jones	JCalvo	ARM/LFMB
PD4 Reading	EButcher	OGC-Rockville	DHagan
EJordan	Plant File		

LTR NAME: ANO1 AMEND 10/12

*See previous concurrences:

PD4/LA*	PD4/PM*	OGC-Rockville*	PD4/D PWO for J.C.
PNoonan	CHarbuck:sr		JCalvo
10/24/88	10/25/88	10/26/88	11/8/88

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[illegible] w/ language changes

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

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cc:

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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. 113
License No. DPR-51

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

   A. The applications for amendment by Arkansas Power and Light Company (the licensee) dated July 20 and August 31, 1988, comply 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 applications, as amended, 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.

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 Appendices A and B, as revised through Amendment No.113 , 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

Paul W O'Connor

for Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 8, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 113

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
7	7
8	8
9	9
9a	9a
9b	9b
9c	9c
11	11
12	12
13	13
14a	14a
14b	14b
15	15
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
48o	-
48p	-
48q	-
48r	-
48s	-
48t	-
48u	-
48v	-
114	114
115	115

LIST OF FIGURES

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the pressure/temperature line the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (3) with potential fuel densification effects:

$$F_q^N = 2.83; \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.65.$$

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.83$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 20.5 kW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for curves 1, 2, and 3 of Figure 2.1-3 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

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

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

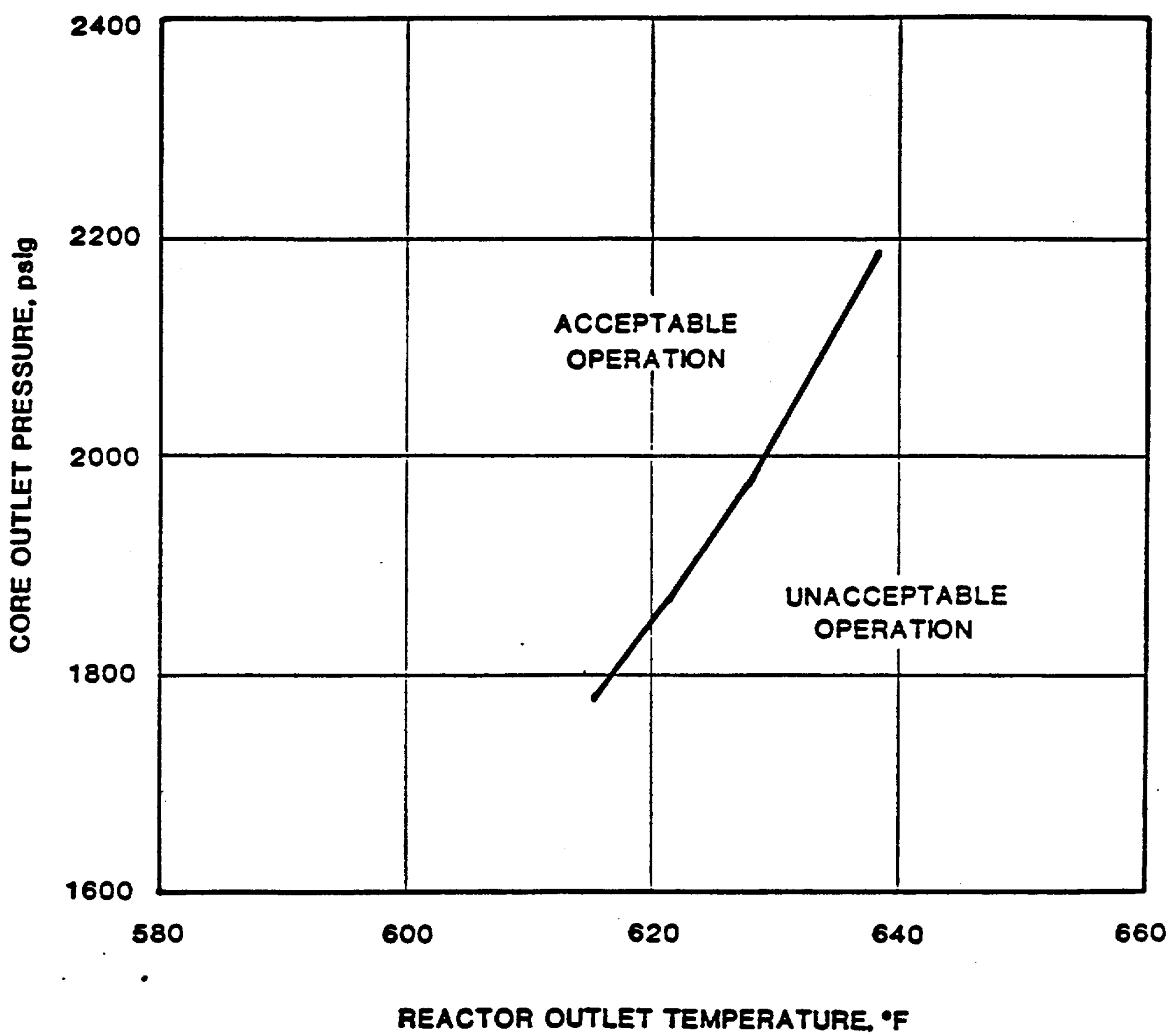
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump situation. Curve 1 of Figure 2.1-3 is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

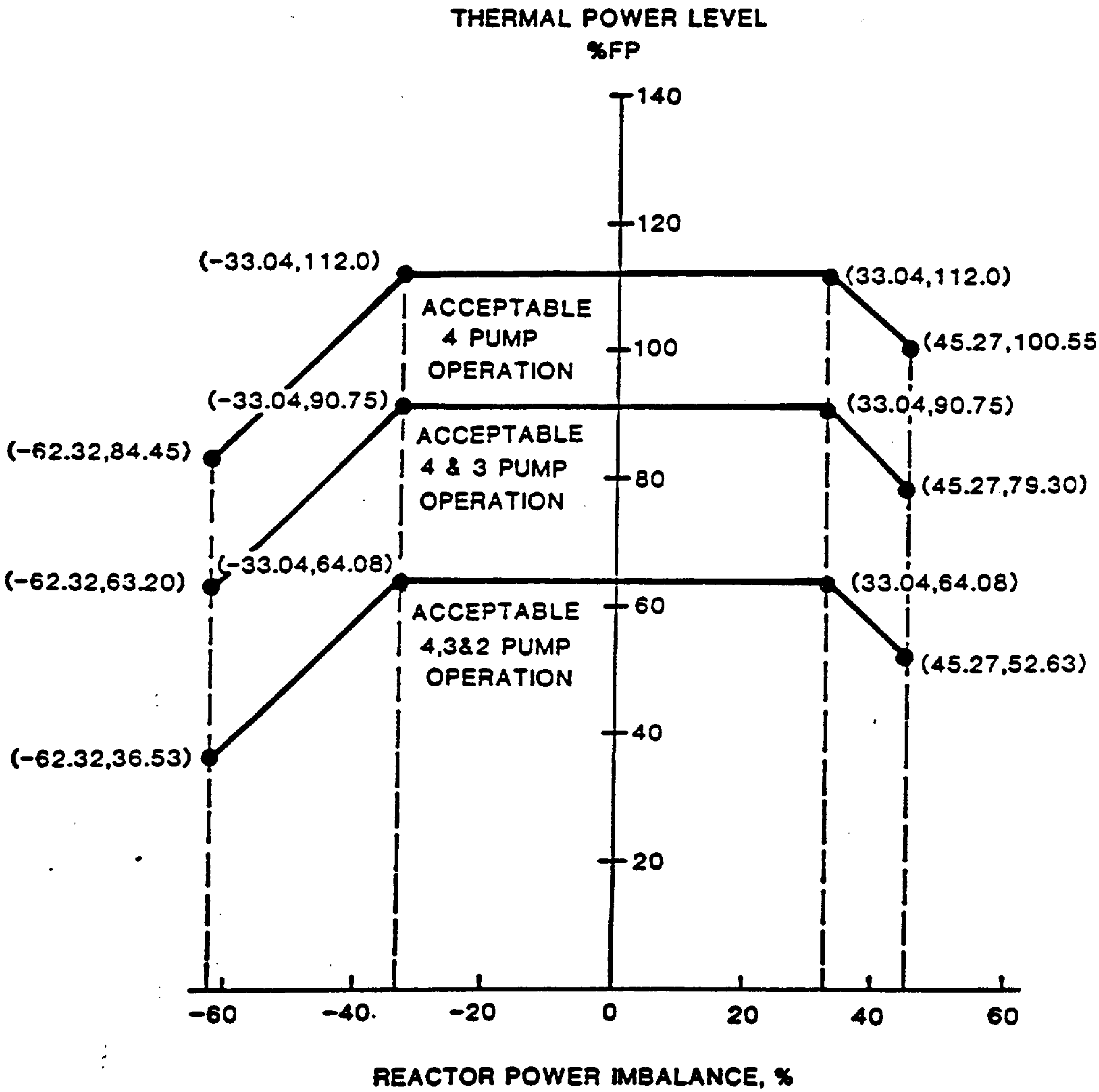
(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

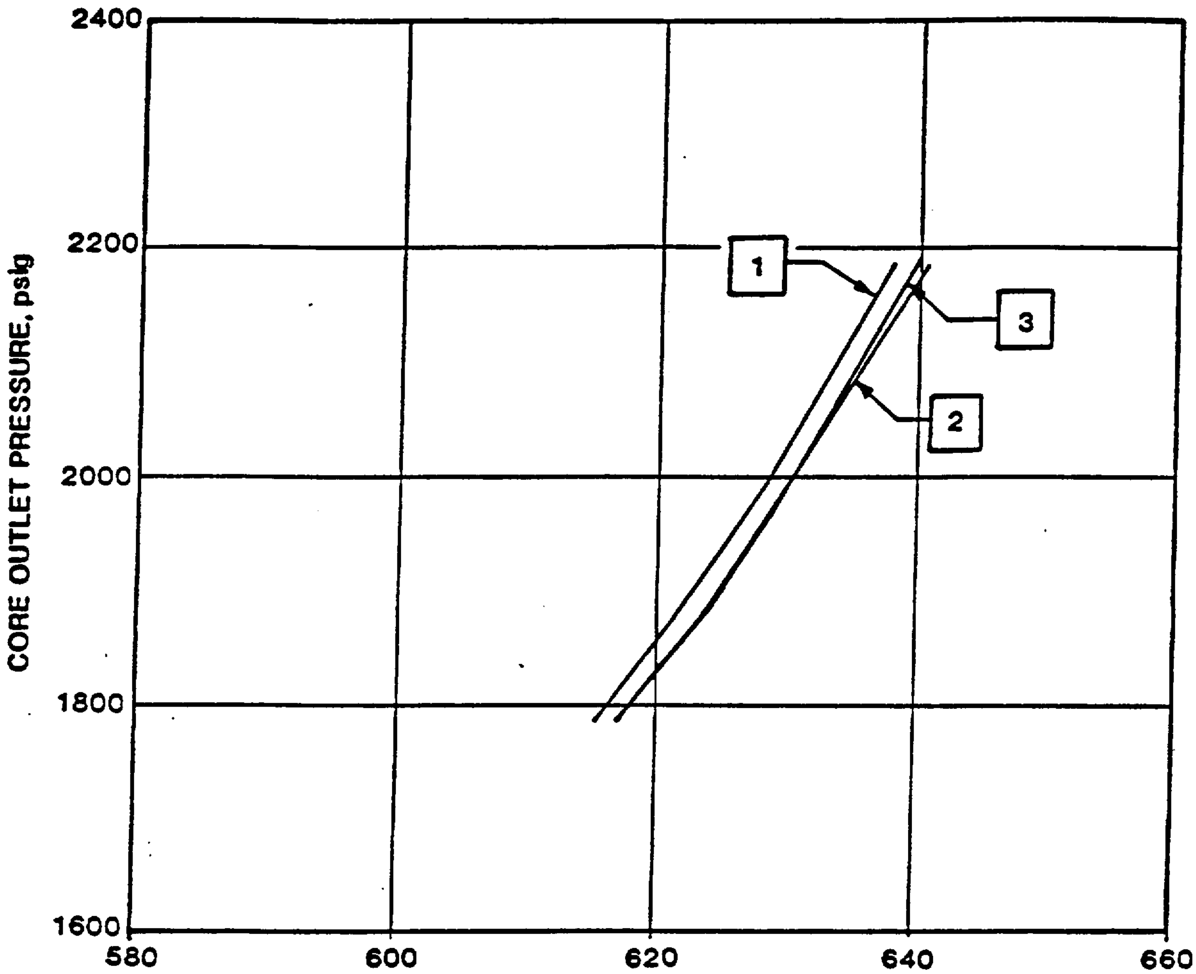


CORE PROTECTION SAFETY LIMIT FIGURE NO. 2.1-1

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



Core Protection Safety Limits - ANO-1
Figure 2.1-3



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%) *	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	90.8%	THREE PUMPS (QUALITY LIMIT)
3	184,441 (49.2%)	63.7%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

* 106.5% OF DESIGN FLOW

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and figure 2.3-2.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a preselected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip setpoint produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by 1.07 percent for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure ($13.89T_{out}$ -6766) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the safety limit of Figure 2.1-1.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620F.

E. Reactor Building Pressure

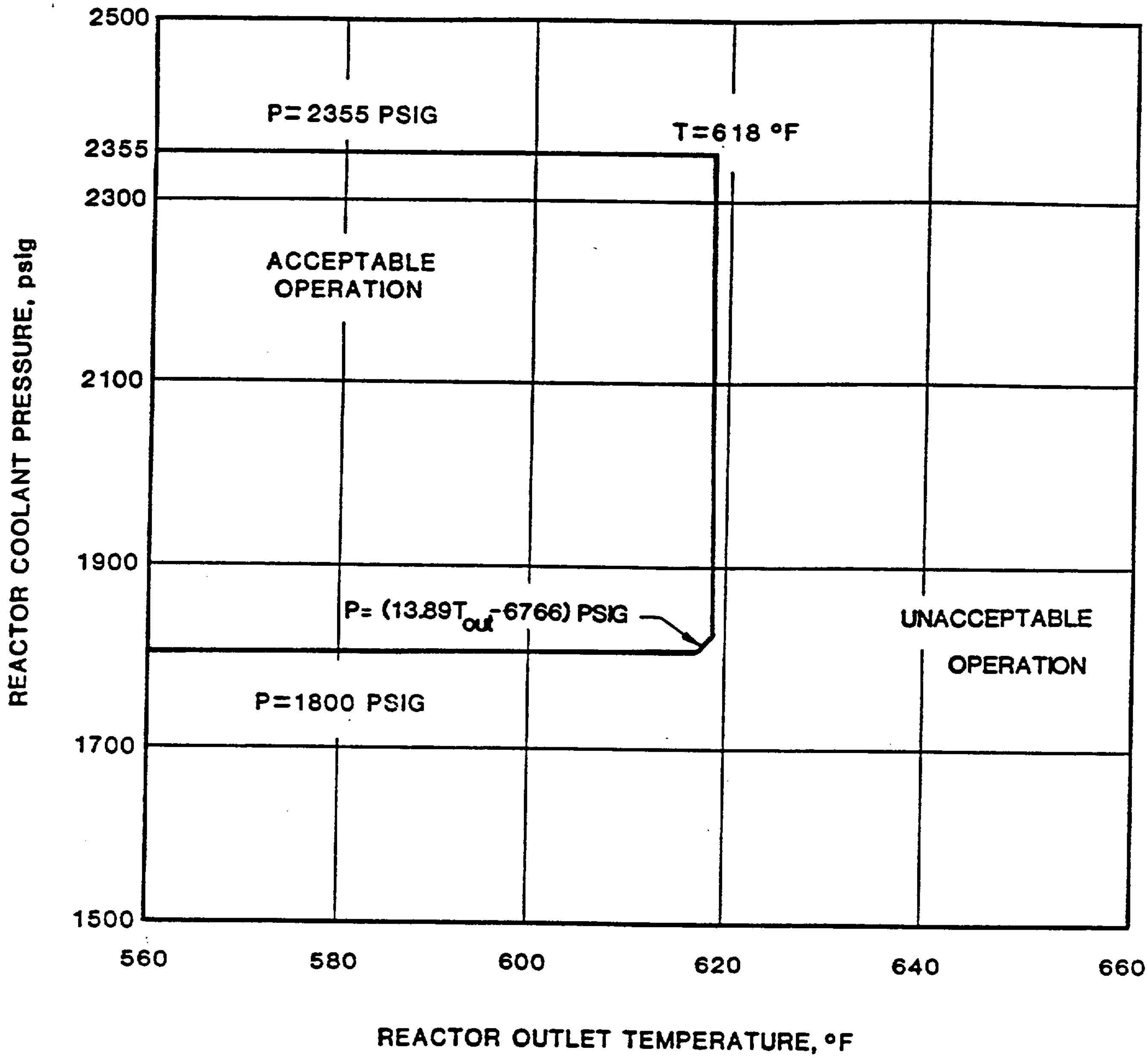
The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip setpoint of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.



PROTECTIVE SYSTEM MAXIMUM

ALLOWABLE SETPOINT

Figure 2.3-1

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

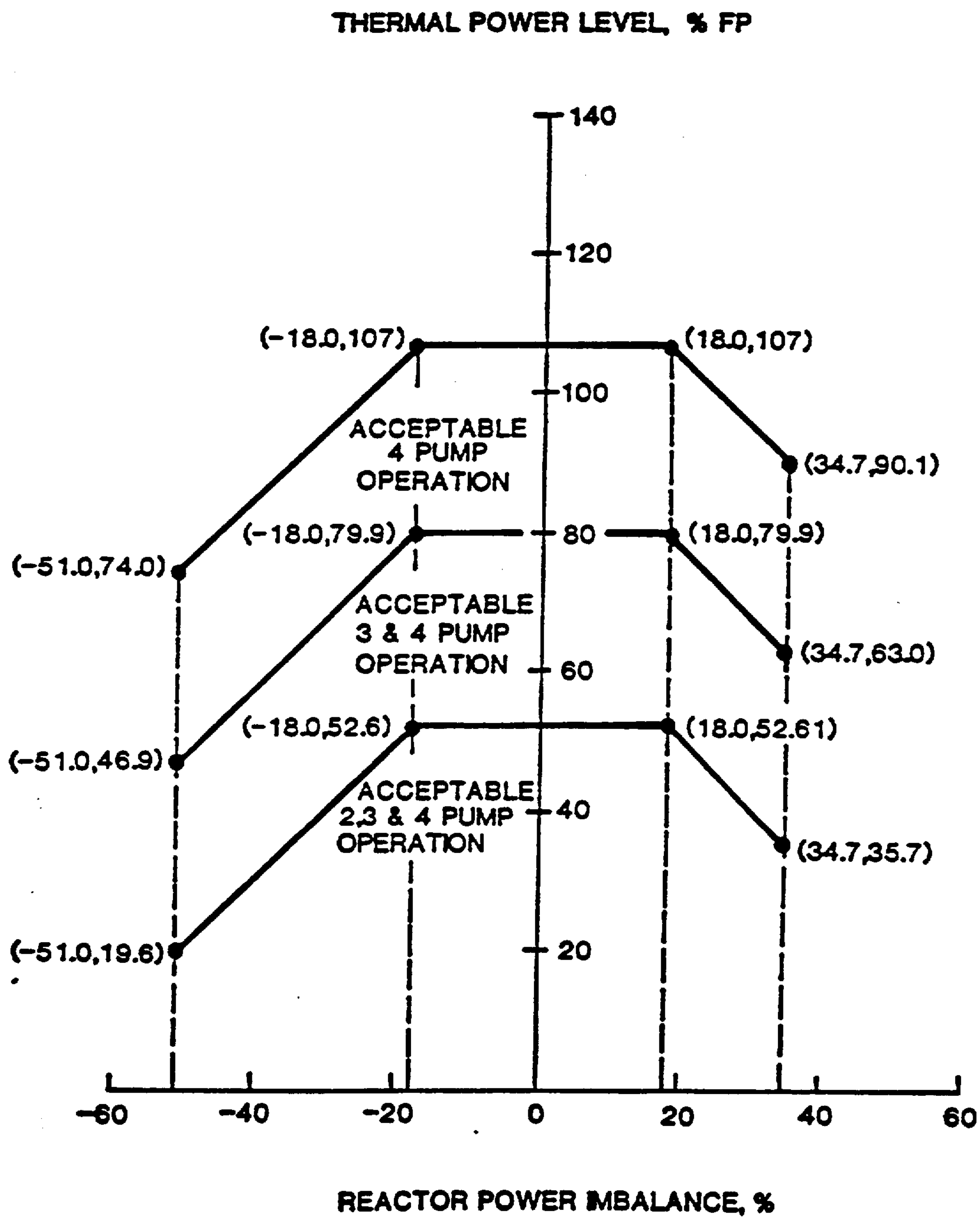


Table 2.3-1
Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
Nuclear Power based on pump monitors[c], % of rated, max	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig, min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	13.89 T_{out}-6766[d]	13.89 T_{out}-6766[d]	13.89 T_{out}-6766[d]	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a] Automatically set when other segments of the RPS (as specified) are bypassed.

[b] Reactor coolant system flow.

[c] The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d] T_{out} is given in degrees Fahrenheit (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 4.12%, 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 4.12%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4.12% 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 4.12%.

   c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 4.12%.

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 410 EFPD, the APSRs may be positioned as necessary for transient imbalance control, however, the APSRs shall be fully withdrawn by 410 EFPD. After 410 EFPD, the APSRs shall not be reinserted.

   With the APSRs inserted after 410 EFPD, corrective measures shall be taken immediately to achieve the full withdrawn position. Acceptable APSR positions shall be attained within 4 hours.

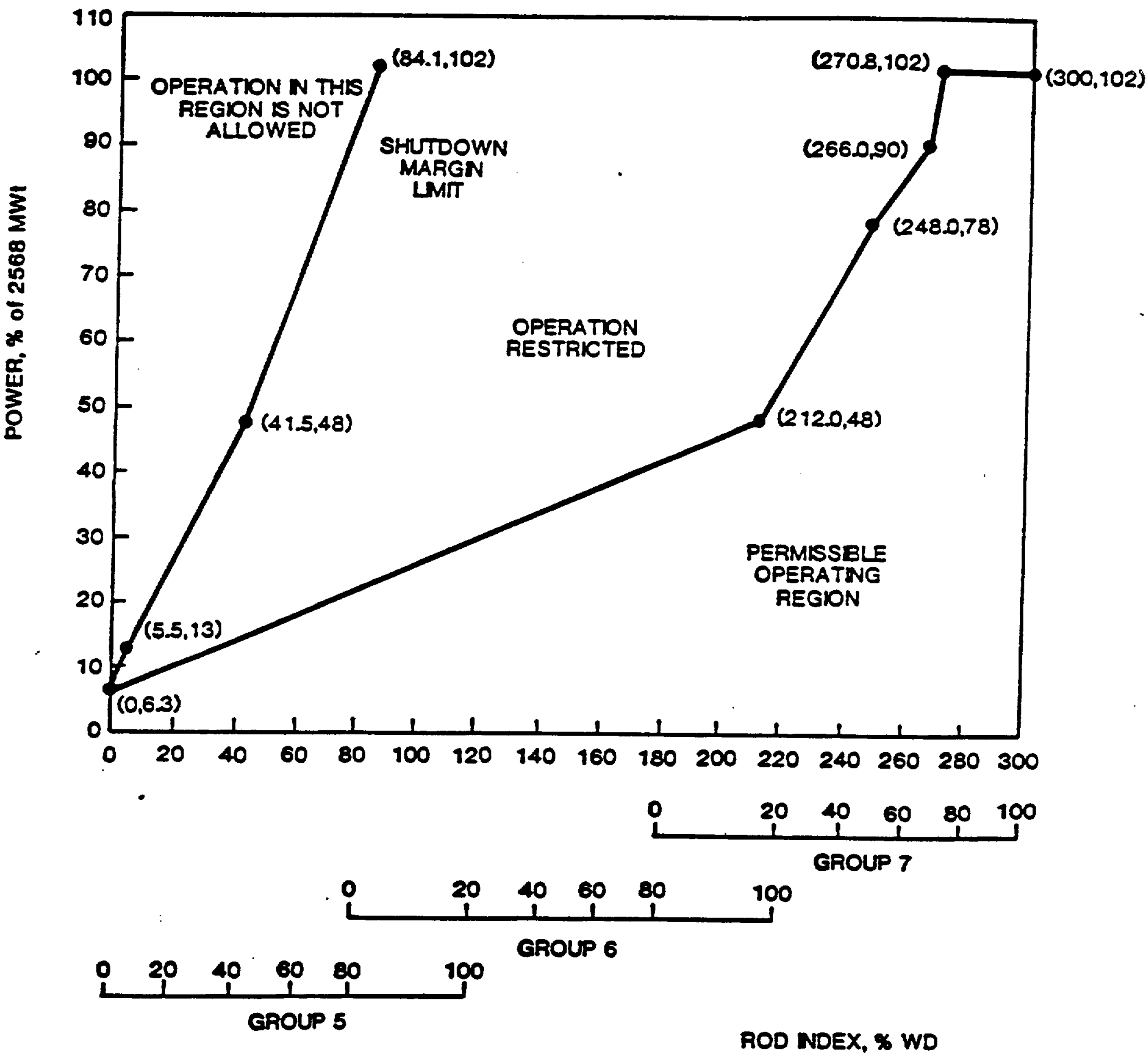
3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-4(A-C). If the imbalance is not within the envelope defined by Figure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

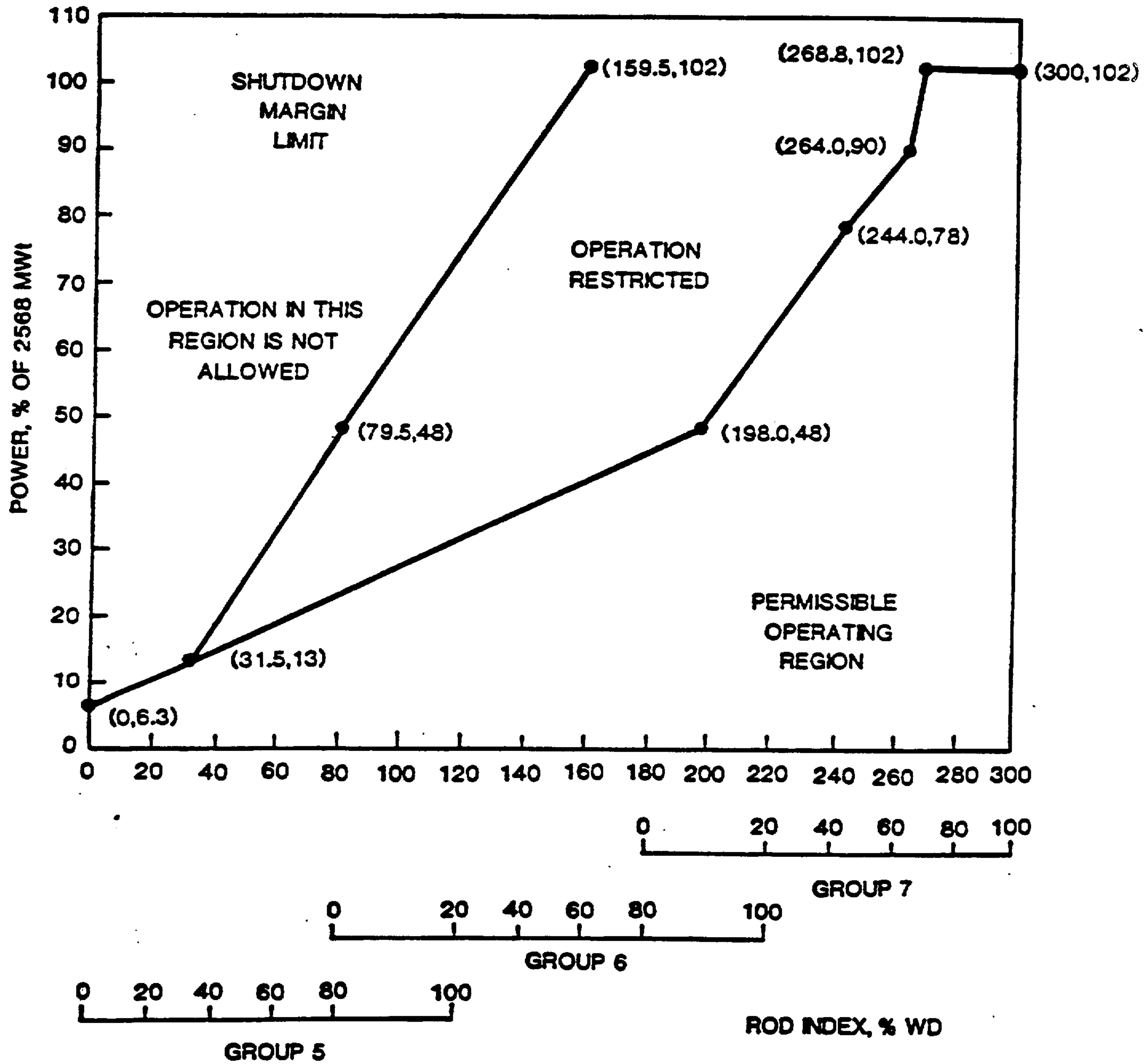
Bases

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

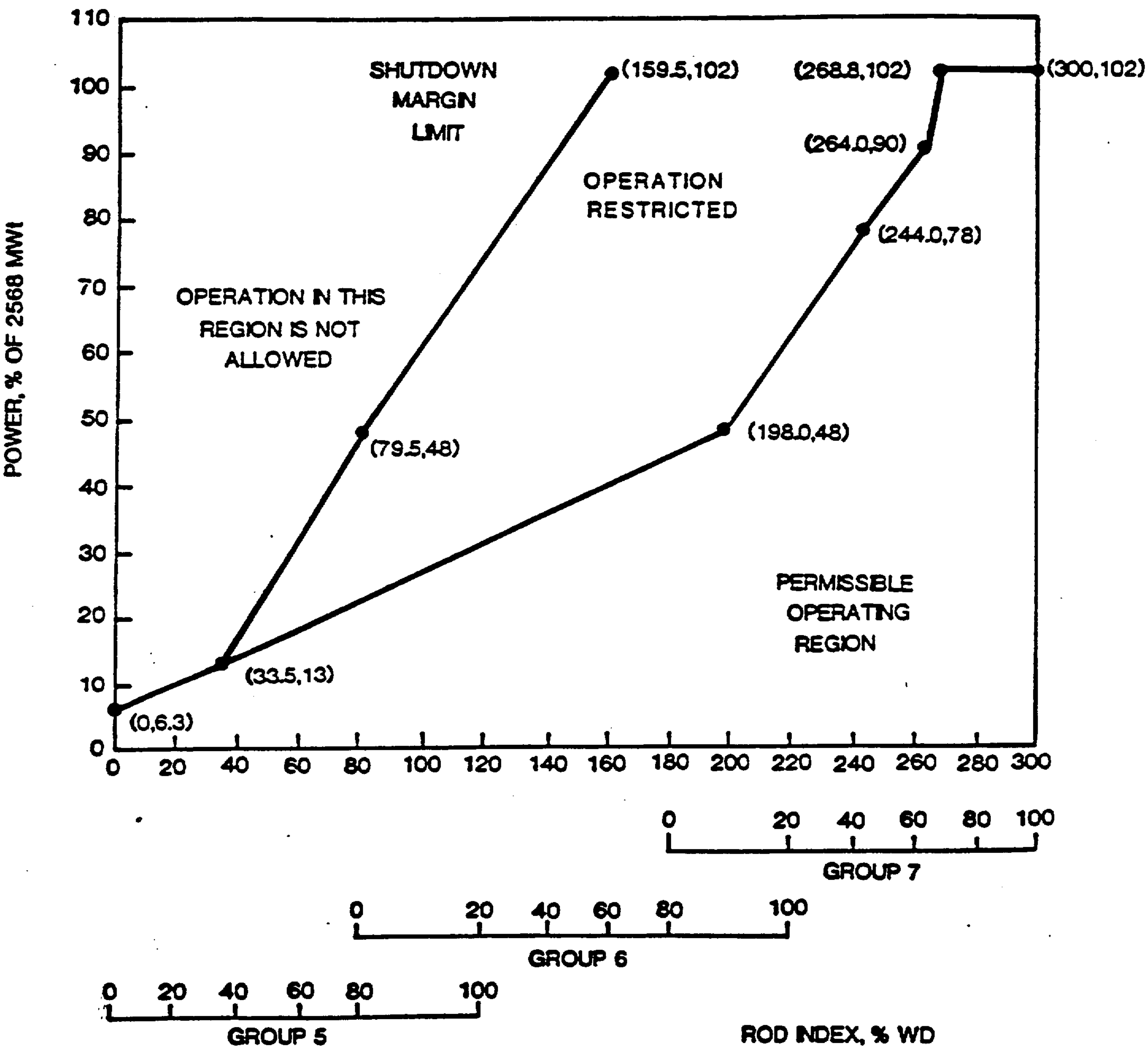
Rod Position Setpoints for 4-Pump Operation
From 0 to 27+10/-0 EFPD ANO-1 Cycle 9
Figure 3.5.2-1A



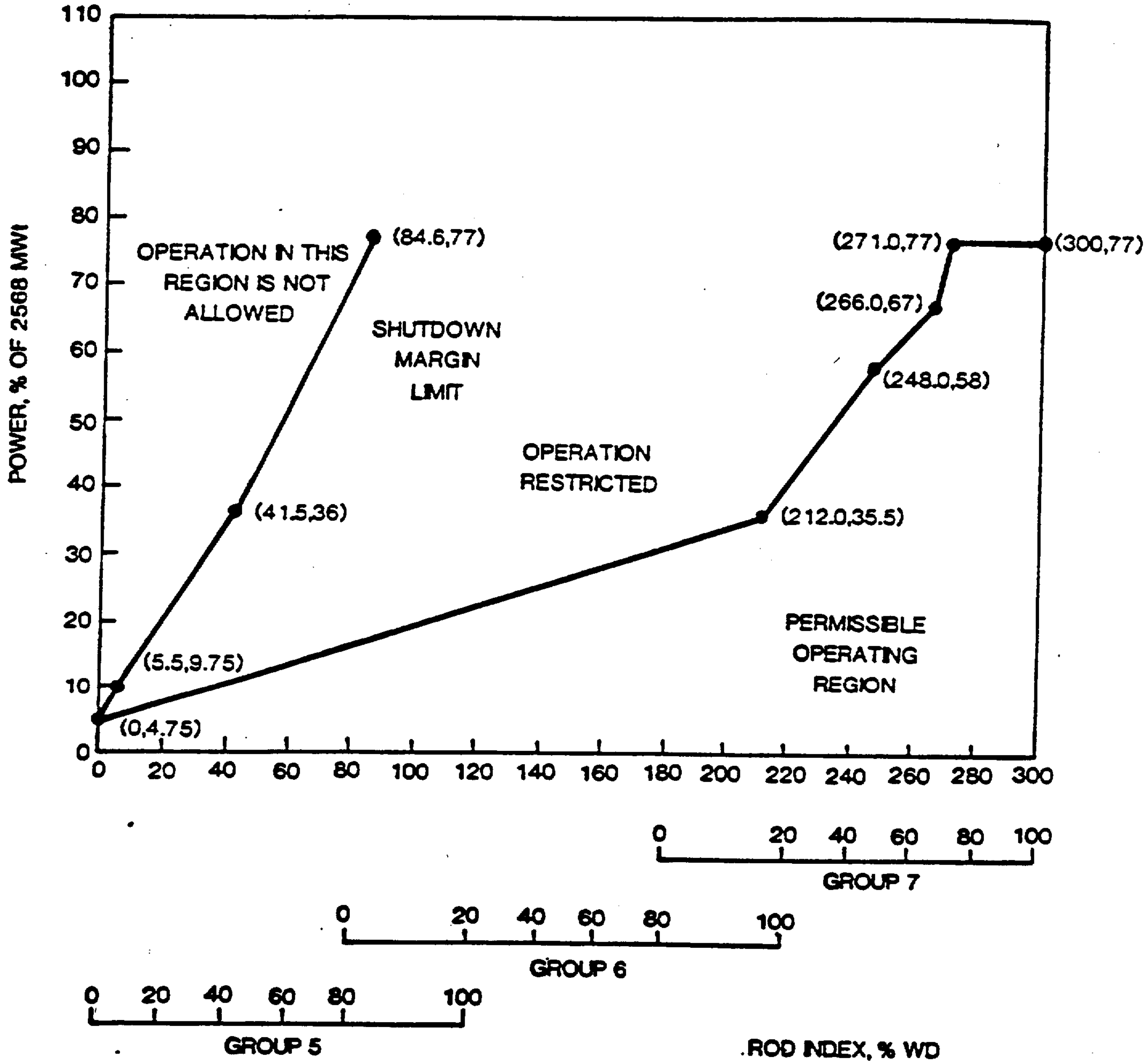
Rod Position Setpoints for 4-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD ANO-1 Cycle 9
Figure 3.5.2-1B



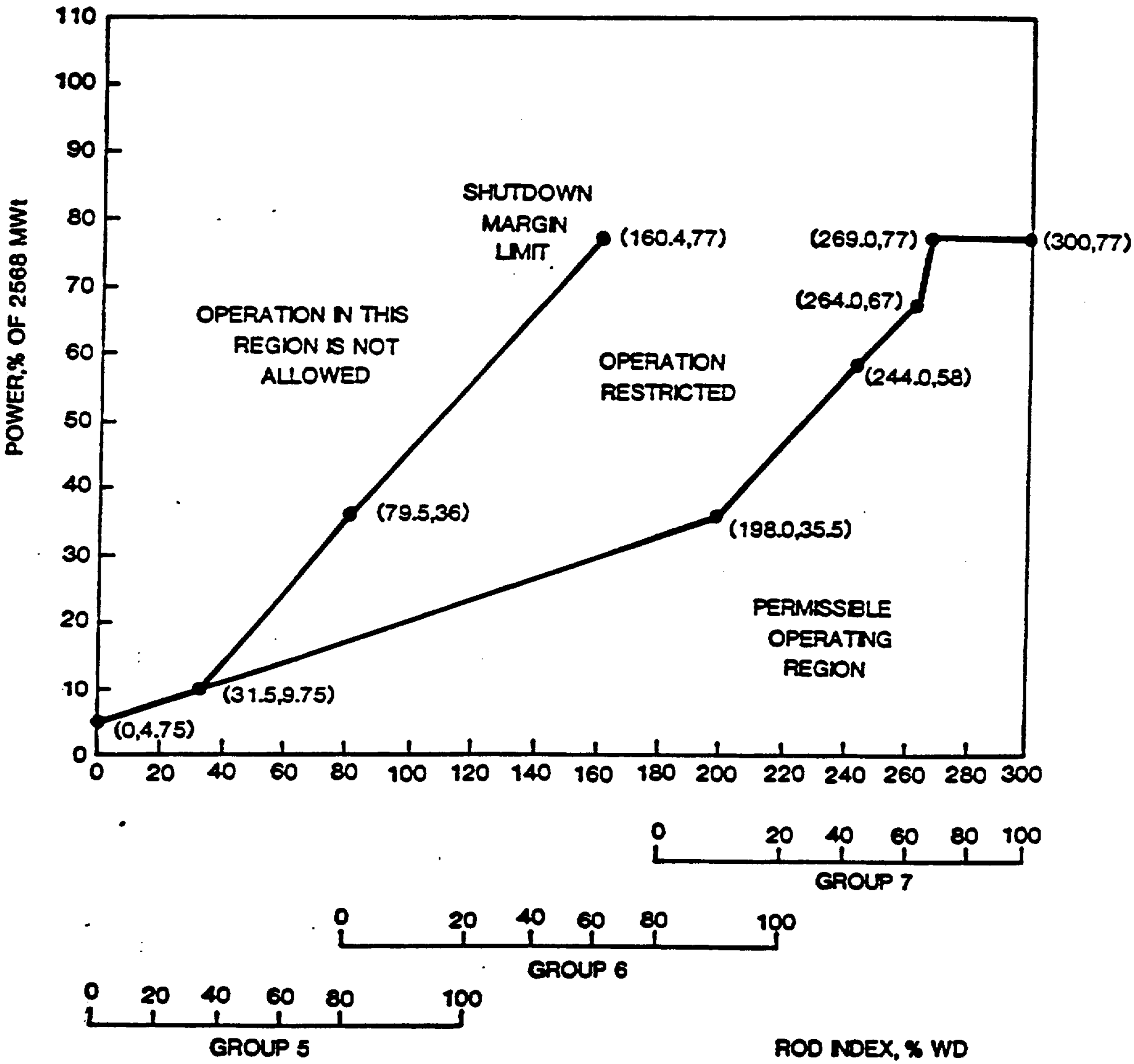
Rod Position Setpoints for 4 - Pump Operation
After 360 +50/-10 EFPD ANO-1 Cycle 9
Figure 3.5.2-1C



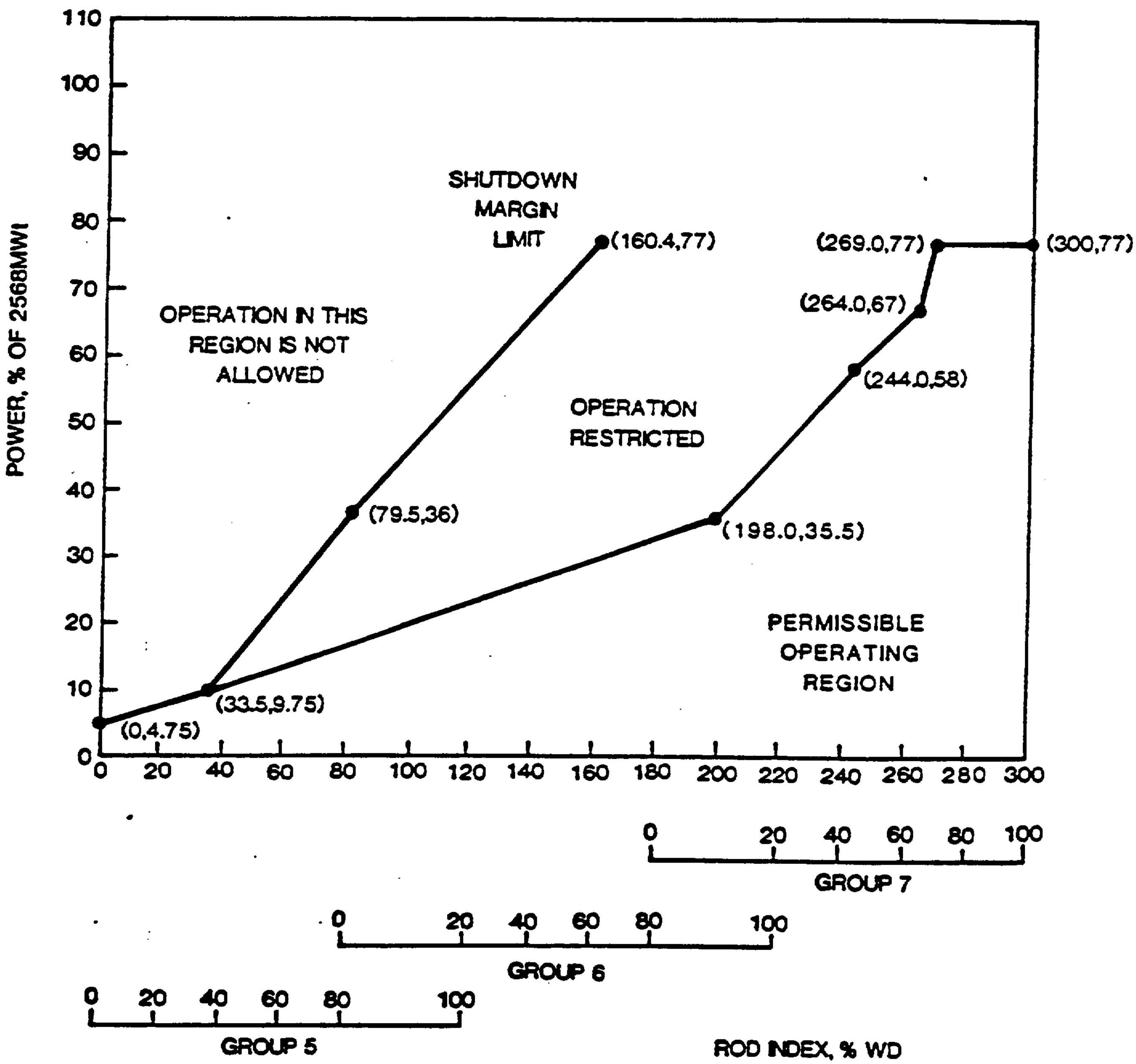
Rod Position Setpoints for 3-Pump Operation
From 0 to 27+10/-0 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2A



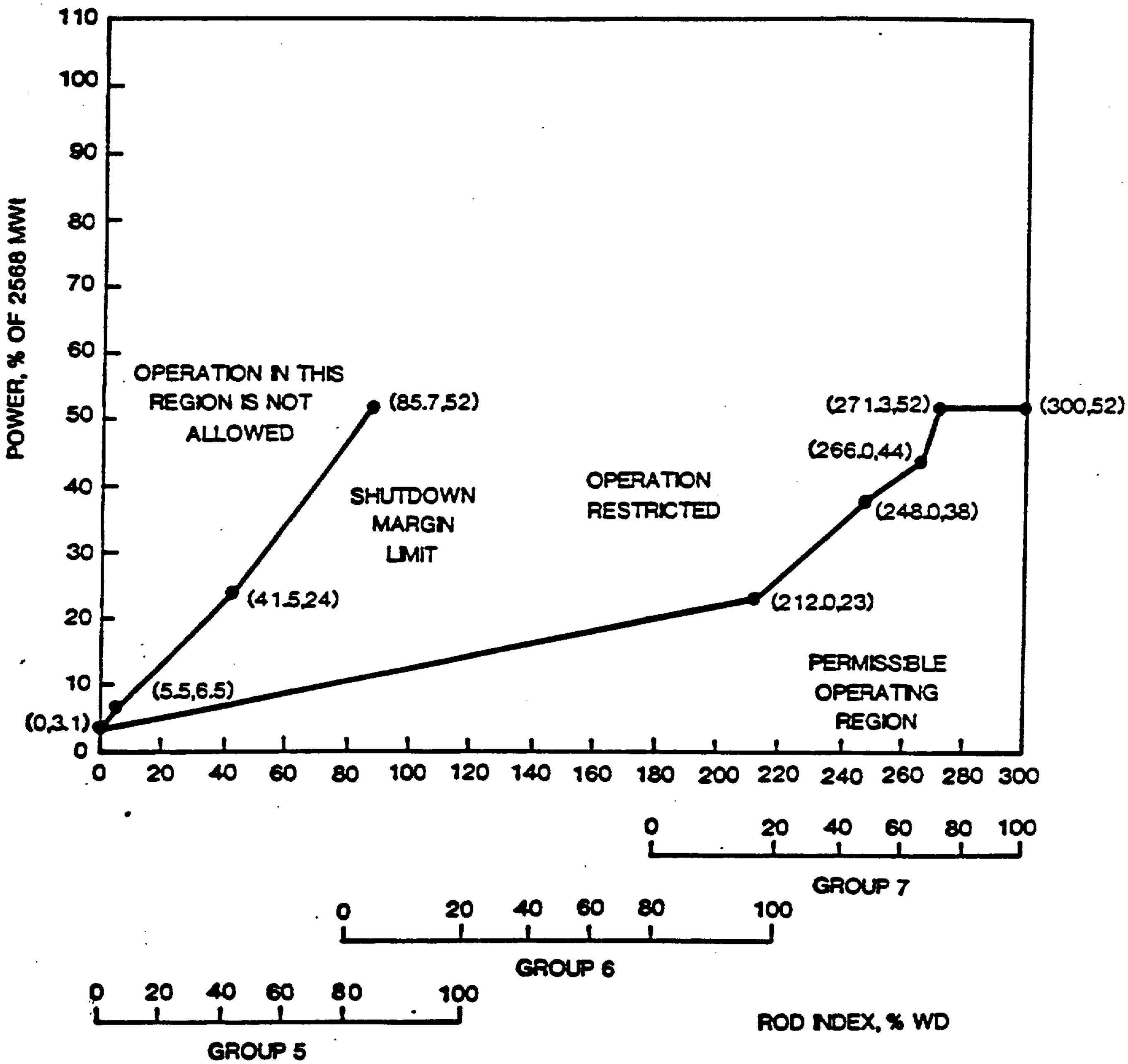
Rod Position Setpoints for 3-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2B



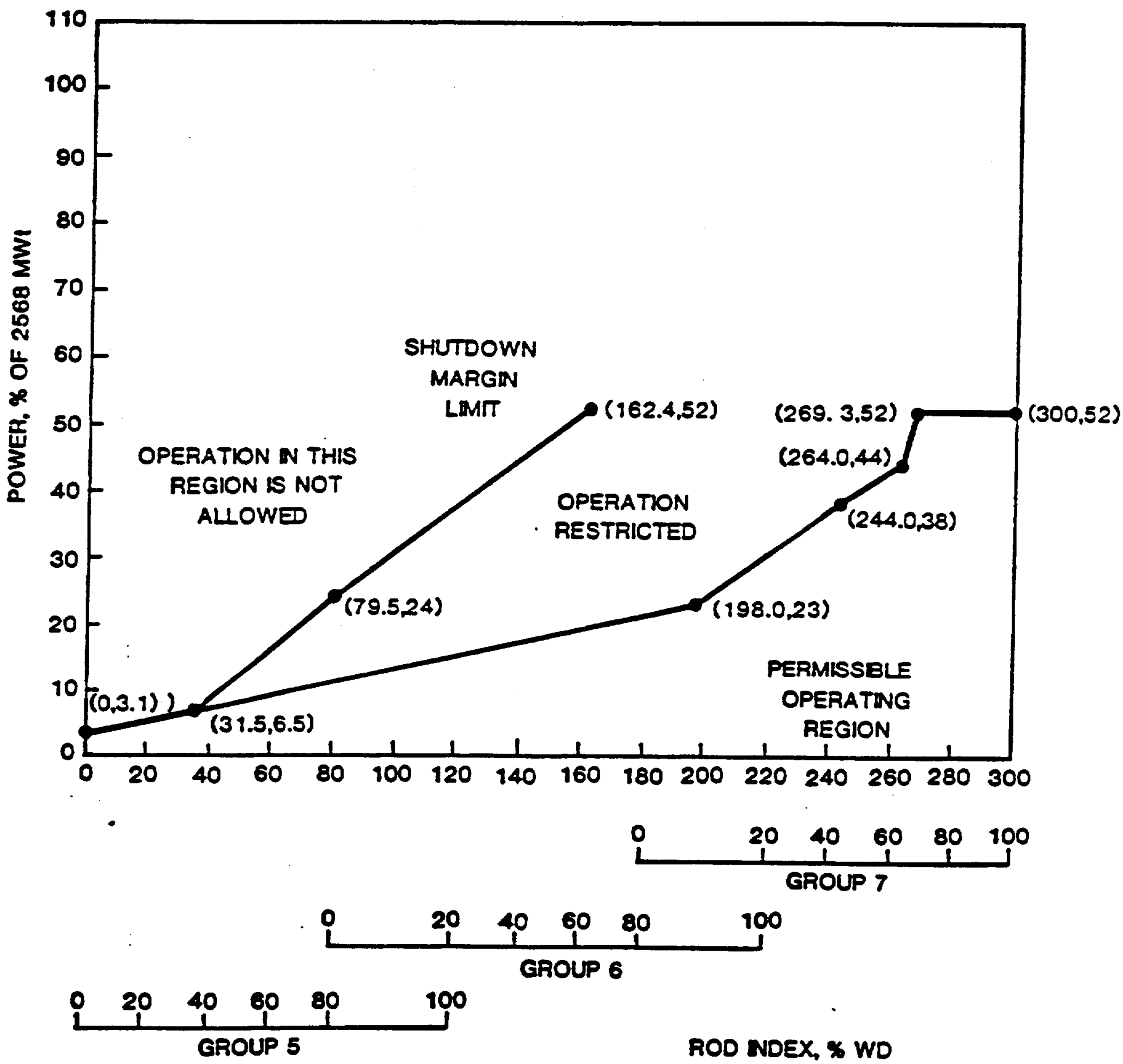
Rod Position Setpoints for 3-Pump Operation
After 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2C



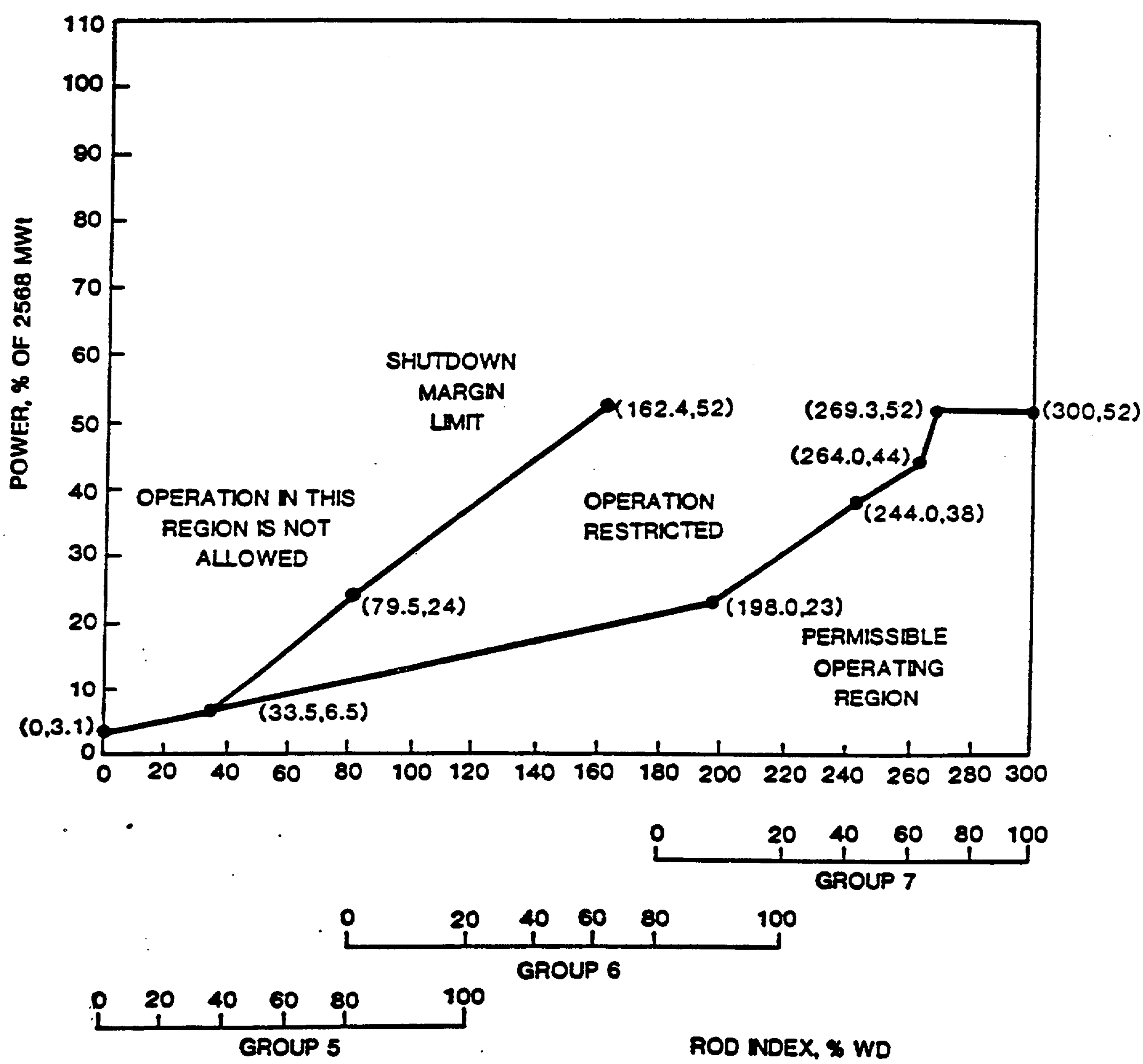
Rod Position Setpoints for 2-Pump Operation
From 0 to 27+10/-0 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3A



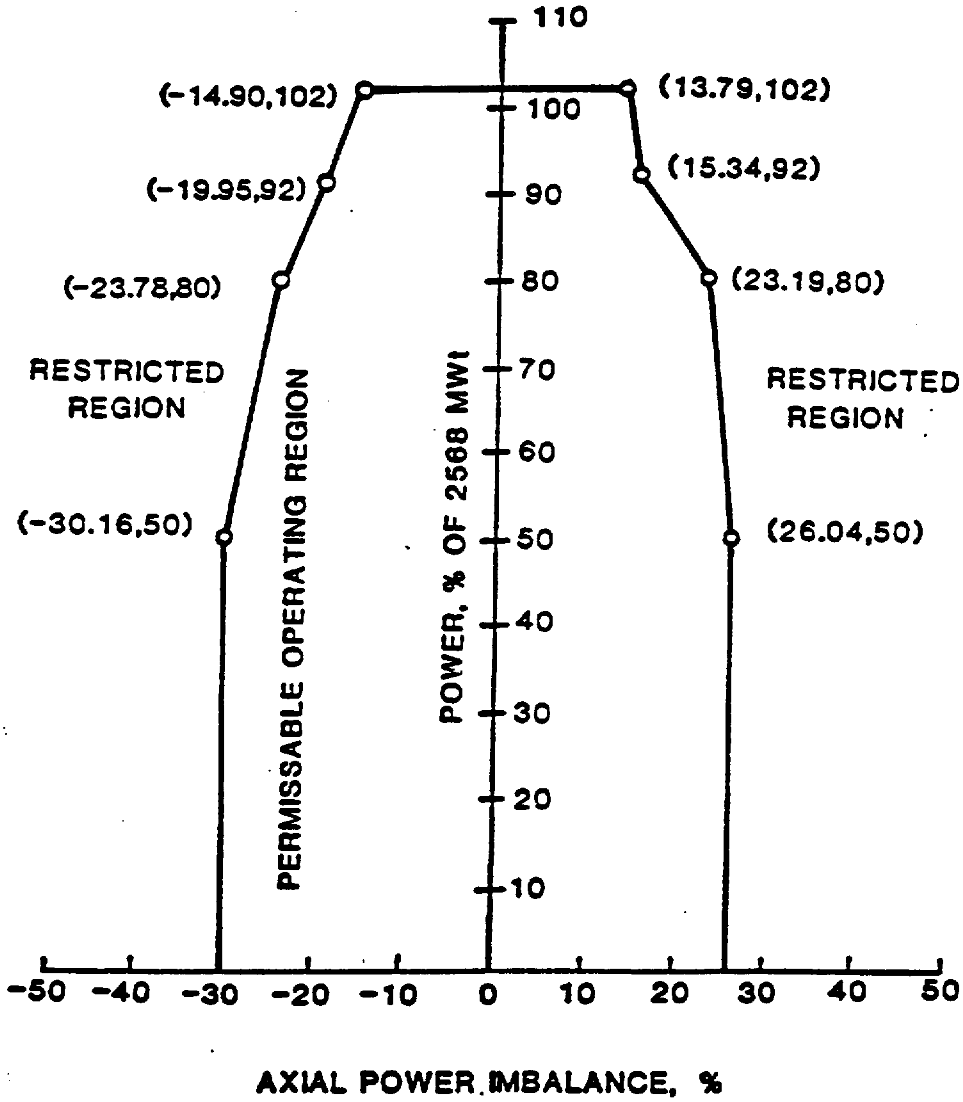
Rod Position Setpoints for 2-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3B



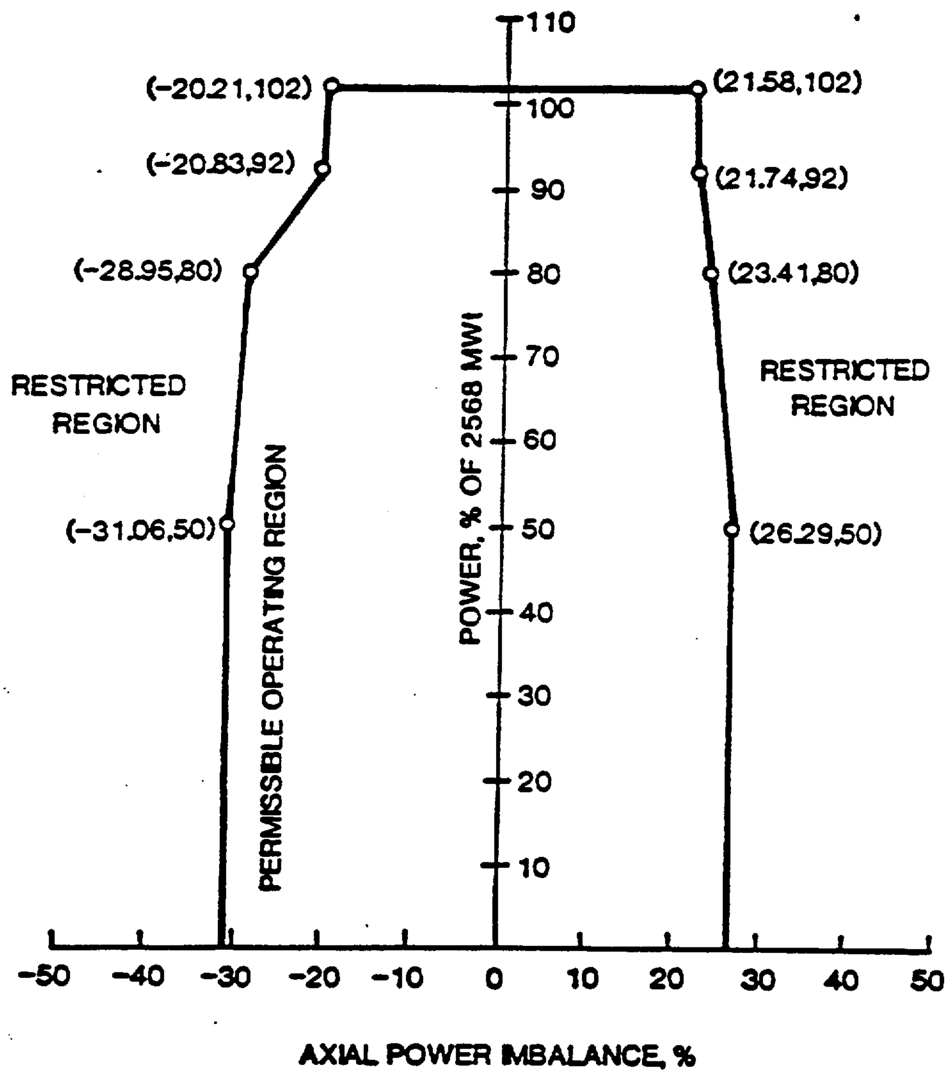
Rod Position Setpoints for 2-Pump Operation
After 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3C



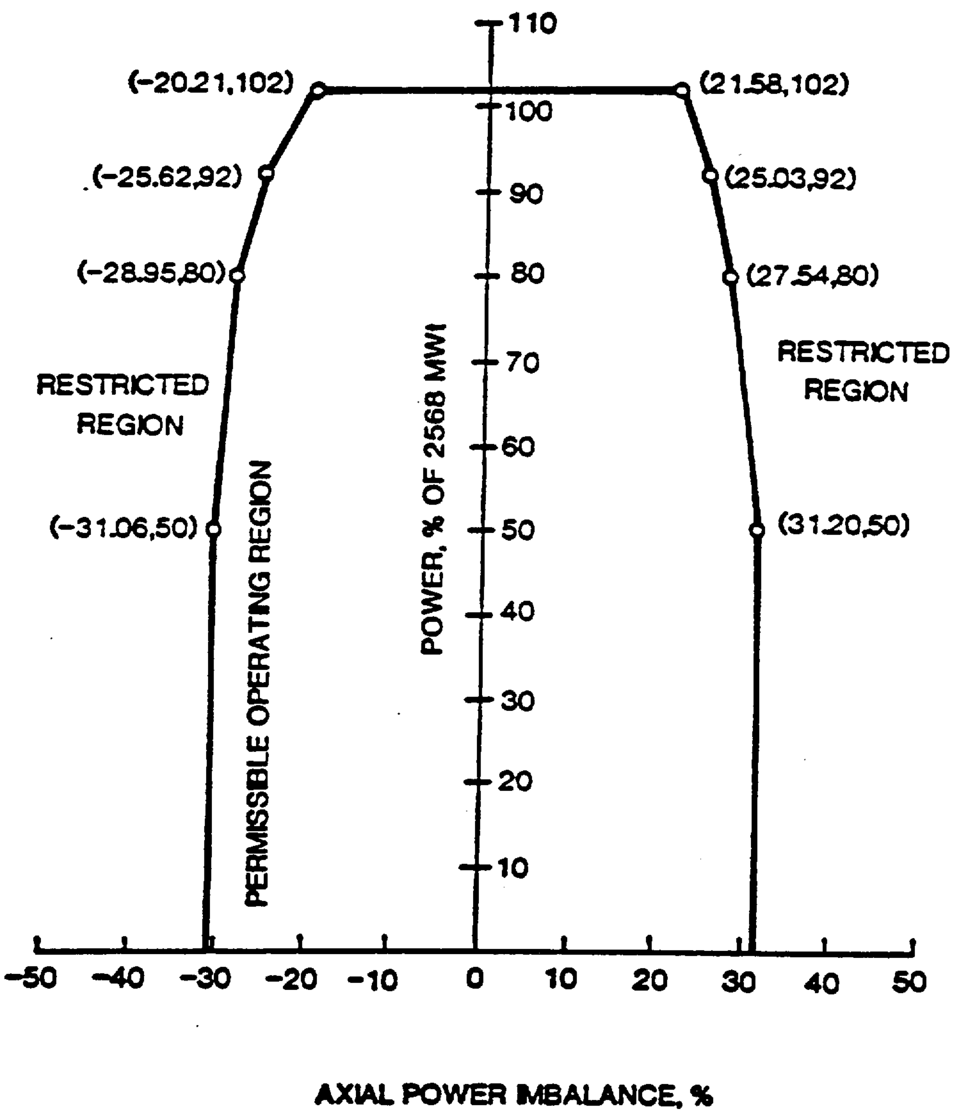
Operational Power Imbalance Setpoints for Operation
From 0 to 27+10/-0 EFPD -- ANO-1, Cycle 9
Figure 3.5.2-4A



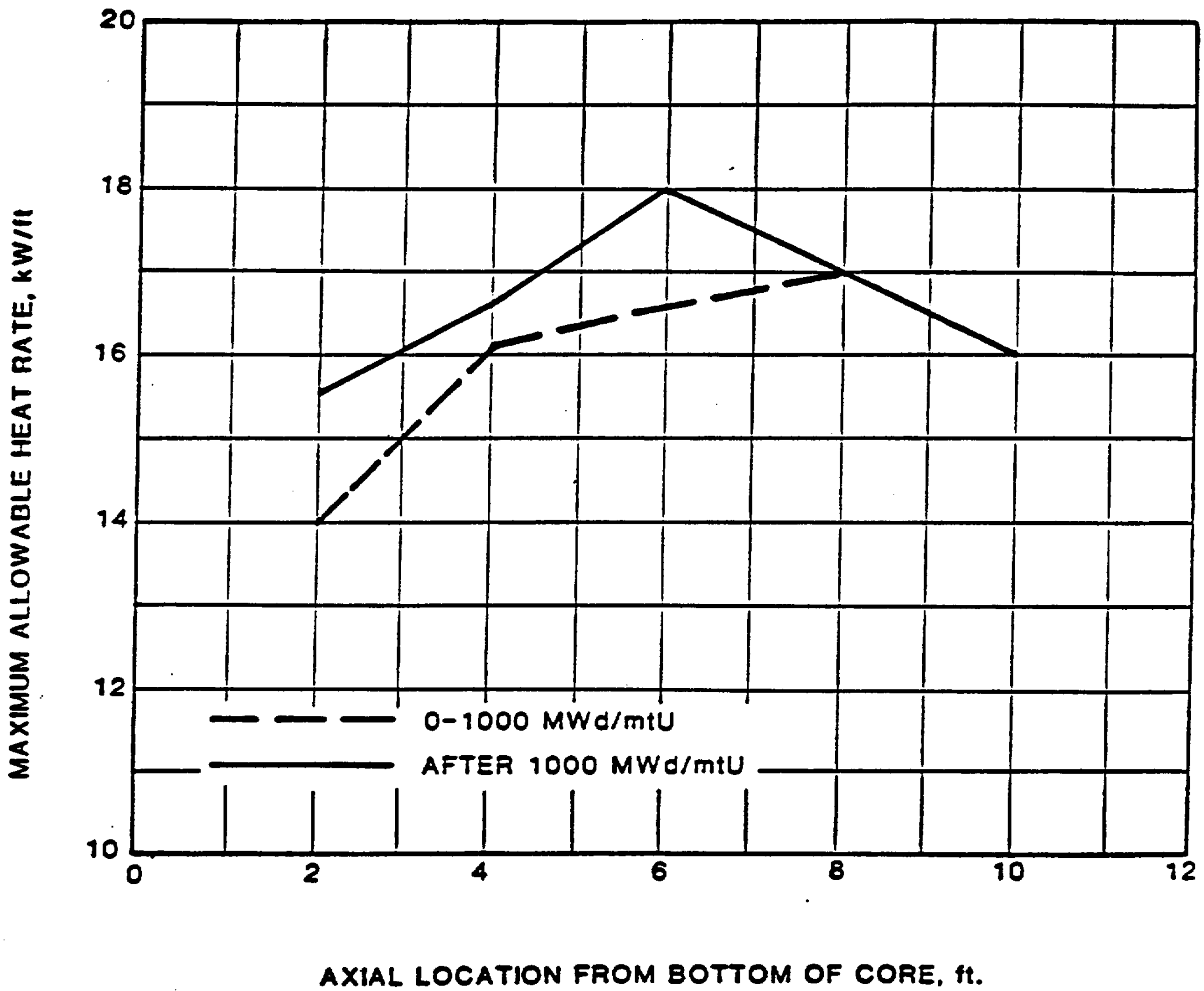
Operational Power Imbalance Setpoints for Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1, Cycle 9
Figure 3.5.2-4B



Operational Power Imbalance Setpoints for Operation
After 360 +50/-10 EFPD -- ANO-1, Cycle 9
Figure 3.5.2-4C



LOCA Limited Maximum Allowable
Linear Heat Rate
Figure 3.5.2-5



5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies. Each fuel assembly is fabricated with 208 fuel rods. ([1,2]) Starting with Batch 11, a reconstitutable fuel assembly design is implemented. This design allows the replacement of up to 208 fuel rods in the assembly.

5.3.1.2 The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches. The active fuel length is approximately 142 inches.([2])

5.3.1.3 The average enrichment of the initial core is a nominal 2.62 weight percent of ^{235}U. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent ^{235}U.

5.3.1.4 There are 60 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3-60. The full-length CRA contain a 134-inch length of silver-indium-cadminum alloy clad with stainless steel. Each APSRA contains a 63-inch length of Inconel-600 alloy.([3])

5.3.1.5 The initial core has 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.

5.3.1.6 Reload fuel assemblies and rods shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 3.5 percent of ^{235}U.

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system is designed and constructed in accordance with code requirements.([4])

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.([5])

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

REFERENCES:

(1) FSAR, Section 3.2.1

(2) FSAR, Section 3.2.2

(3) FSAR, Section 3.2.4.2

(4) FSAR, Section 4.1.3

(5) FSAR, Section 4.1.2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113 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

In a letter dated July 20, 1988, Arkansas Power and Light Company made application to modify the Technical Specifications for Arkansas Nuclear One, Unit 1 (ANO-1) to permit operation for Cycle 9. An associated additional application was submitted by letter dated August 31, 1988 to modify the variable low pressure trip setpoint. The safety analyses performed are described in the Cycle 9 reload report. The reference cycle for this reload is Cycle 8. All accidents analyzed in the Final Safety Analysis Report (FSAR) have been reviewed for Cycle 9 operation.

1.1 Description of the Cycle 9 Core

The ANO-1 Cycle 9 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes and one incore instrument guide tube. Reactivity is controlled by 60 full-lenghth Ag-In-Cd control rods, 52 burnable poison rod assemblies (BPRAs) and soluble boron shim. Eight Inconel-600 axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution.

1.2 Significant Areas of Review for this Reload

For the most part, Cycle 9 of ANO-1 will be identical in operation to Cycle 8 and most Technical Specification changes such as reactor core safety limit trip setpoints, rod insertion limits and imbalance limits are the result of changes resulting from insertion of new fuel, cycle lifetime and time of APSR(s) withdrawal which are often made on B&W reactors. Significant changes are replacing the Ag-In-Cd with Inconel for neutron absorption in the APSRs, using a low leakage fuel cycle design and using a mixed core fuel assembly with Inconel and Zircaloy spacer grids. These changes have been made previously on B&W reactors and are evaluated in the fuel design and nuclear performance section of this Safety Evaluation.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

2.1 Fuel Assembly Mechanical Design

The feed batch, batch 11, consists of 60 assemblies of the MK-B6 type with uranium enrichment of 3.45%. Cycle 9 will consist of 1 batch 6D, 52 Batch 9B, 64 batch 10 and 60 batch 11 assemblies. The differences between MK-B6 and other MK-B types are the zircaloy spacer grids, the fact that it is reconstitutable, and the method used to retain fixed control components during reactor operation. This type of fuel has been previously used on other B&W reactors. All fuel assemblies are identical in concept and are mechanically interchangeable.

2.2. Fuel Rod Design

The cladding stress, strain and collapse analyses methods used for the Cycle 9 fuel rod design are the same ones used for previous cycles. We find that no further review in these areas is necessary.

2.3 Fuel Thermal Design

All fuel assemblies in the Cycle 9 core are thermally similar. The design of the batch 11 MarkB6 assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. Cycle 9 core protection limits are based on a linear heat rate (LHR) to centerline fuel melt limit of 20.5 kw/ft. Maximum fuel assembly burnup at EOC9 is predicted to be less than 42,800 MWd/MtU. The fuel rod internal pressures have been evaluated for the highest burnup fuel rods and predicted to be less than the nominal reactor pressure of 2200 psia. The fuel thermal design analyses was performed using TACO2 which was previously reviewed and approved by the staff (Ref. 1).

2.4 Gray APSR Design

The gray APSR design was analyzed for cladding stress due to pressure, temperature and ovality. It was found that the gray APSR had sufficient cladding and weld stress margins. The gray APSR was also analyzed for cladding strain due to thermal and irradiation swelling. The results of B&W analysis showed that no cladding strain is enduced due to thermal expansion or irradiation swelling of the inconel absorbers.

The staff has reviewed the mechanical design of the gray APSR's and has previously found it acceptable (Ref. 2). This design has been used in several B&W reactors.

3.0 EVALUATION OF THE NUCLEAR DESIGN

The core design changes for Cycle 9 are the use of gray APSRs and the replacement of the Inconel intermediate spacer grids with Zircaloy spacer grids. The gray APSRs, are longer and use a weaker absorber. Calculations with the standard three-dimensional model verified that these APSRs provide adequate axial power distribution control. The substitution of Zircaloy spacer grids reduces the parasite absorption of neutrons. The calculational methods used to obtain the important nuclear design parameters for this cycle were the same ones as used for the reference cycle.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

The design basis chosen for Cycle 9 thermal-hydraulic analyses was a full core of Zircaloy grid assemblies, containing 40 BPRAs, for which the core bypass flow is 8.8%. The actual Cycle 9 core contains 52 BPRAs with core bypass flow of 8.3%. The pressure-temperature safety limits have been recalculated using the BWR CHF correlation in the LYNXT crossflow analysis. Based on the similarity with Cycle 8 and the use of approved models and methods, we conclude that the thermal-hydraulic design of Cycle 9 is acceptable.

5.0 EVALUATION OF TRANSIENT AND ACCIDENT ANALYSIS

Each FSAR accident analysis has been examined with respect to changes in Cycle 9 parameters to determine the effect of the Cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The radiological dose consequences of the accidents presented in Chapter 14 of the updated FSAR were reevaluated, except for waste gas tank rupture, which is not cycle dependent. All of the calculated Cycle 9 accident doses were below the dose acceptance criteria specified in the NRC Standard Review Plan.

The key parameters that have the greatest effect on determining the outcome of a transient are core thermal properties, thermal-hydraulic parameters and kinetics parameters. Comparisons of these parameters with those from previous cycles and the FSAR values showed that the Cycle 9 parameters are within the bounds of those used for previous analyses. Thus, Cycle 9 is bounded by the previous analyses.

6.0 TECHNICAL SPECIFICATION CHANGES

The Technical Specifications have been revised for Cycle 9 operation for changes in core reactivity, power peaking and control rod worths. The Cycle 9 core also includes very low leakage fuel cycle design, a mixed Mark B4/Mark B6 fuel assembly core, gray APSRs, gray APSR withdrawal flexibility and crossflow analysis. These contribute to the number of Techncial Specification changes needed.

Figures 2.1-1, 2.1-2, 2.1-3 and the bases to TS 2.1
Figures 2.3-1, 2.3-2, Table 2.3-1 and the bases to TS 2.3

These sections are modified to reflect the mixed core of fuel assemblies with Inconel and Zircaloy spacer grids, and to use cycle specific credits. Also included is a revised variable low pressure trip setpoint.

Section 3.5.2.4, 3.5.2.5.3, 3.5.2.5.4
Figures 3.5.2-1, 3.5.2-2 and 3.5.2-3, 3.5.2-4

These sections are modified to reflect new quadrant power tilt limits, changes to control rod insection limits, operational limits on the gray axial power shaping rods (APSRs), and new operational power imbalance limit.

Section 5.3.1

This section has been modified to reflect the reconstitutable fuel assembly design and the gray axial power shaping rods.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.22 and 51.35, an environmental assessment and finding of no signifiant impact was published in the Federal Register on November 4, 1988 (53 FR 44684).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design and the transient and accident analyses information presented in the Arkansas Nuclear One Unit 1 Cycle 9 reload submittals. Based on these considerations, the staff has concluded 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 8, 1988

Principal Contributor: M. Chatterton

REFERENCES

1. Y. Hsii, et. al., "TACO2-Fuel Pen Performance Analyses," Babcock & Wilcox Company Report BAW-10141P-A, Rev. 1, dated June 1983.

2. L. S. Rubenstein (NRC) to G. C. Lainas (NRC) Safety Evaluation of the Crystal River Unit 3 Cycle 6 Reload, dated June 18, 1985.

UNITED STATES NUCLEAR REGULATORY COMMISSION

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-313

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.113 to Facility Operating License No. DPR-51, to Arkansas Power and Light Company, which revised the Technical Specifications for operation of the Arkansas Nuclear One, Unit No. 1, located in Pope County, Arkansas. The amendment was effective as of the date of its issuance.

The amendment approved changes necessary to permit operation of ANO-1 for Cycle 9. The significant changes for Cycle 9 are the replacement of "black" axial power shaping rods (APSRs) with gray APSRs, use of a mixed core of fuel assemblies with Inconel and Zircaloy spacer grids and use of a low leakage fuel cycle design. A change in the variable low pressure trip setpoint, which is based on the Cycle 9 reload analysis, is also included in this amendment.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on September 16, 1988 (53 FR 36141). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributed to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated February 1973.

For further details with respect to this action, see (1) the applications for amendment dated July 20 and August 31, 1988, (2) Amendment No.113 to Facility Operating License No. DPR-51, and (3) the Environmental Assessment and Finding of No Significant Impact (53 FR 44684). All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555, and at the Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commisssion, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 8th day of November, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

Paul W. O'Connor, Acting Project Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation