PROPOSED TECHNICAL SPECIFICATION CHANGES

LIST OF FIGURES

Number	Title	Page
2.1-1	CORE PROTECTION SAFETY LIMITS	9a
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 27+10/-0 EFPD - ANO-1 CYCLE 9	48b
3.5.2-1B	ROD POSITION SETPOINTS FOR FOUR-PUMP OPERATION FROM 27+10/-0 TO 360+50/-10 EFPD - ANO-1 CYCLE 9	48c
3.5.2-1C	ROD POSITION SETPOINTS FOR FOUR-PUMP OPERATION AFTER 360+50/-10 EFPD - ANO-1 CYCLE 9	48d
3.5.2-2A	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION FROM 0 TO 27+10/-0 EFPD - ANO-1 CYCLE 9	48e
3.5.2-2B	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION FROM 27+10/-0 TO 360+50/-10 EFPD - ANO-1 CYCLE 9	48f
3.5.2-2C	ROD POSITION SETPOINTS FOR THREE-PUMP OPERATION AFTER 360+50/-10 EFPD - ANO-1	48g

3.5.2-3A	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION FROM 0 TO 27+10/-0 EFPD - ANO-1 CYCLE 9	48h
3.5.2-3B	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION FROM 27+10/-0 TO 360+50/-10 EFPD - ANO-1 CYCLE 9	48i
3.5.2-3C	ROD POSITION SETPOINTS FOR TWO-PUMP OPERATION AFTER 360+50/-10 EFPD - ANO-1 CYCLE 9	48j
3.5.2-4A	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION FROM 0 TO 27+10/-0 EFPD - ANO-1 CYCLE 9	48k
3.5.2-4B	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION FROM 27+10/-0 TO 360+50/-10 EFPD - ANO-1 CYCLE 9	48l
3.5.2-4C	OPERATIONAL POWER IMBALANCE SETPOINTS FOR OPERATION AFTER 360+50/-10 EFPD - ANO-1 - CYCLE 9	48m
3.5.2-5	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	48n
3.5.4-1	INCORE INSTRUMENTATION SPECIFICATION AXIAL IMBALANCE INDICATION	53a
3.5.4-2	INCORE INSTRUMENTATION SPECIFICATION RADIAL FLUX TILT INDICATION	53b
3.5.4-3	INCORE INSTRUMENTATION SPECIFICATION	53c
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

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.

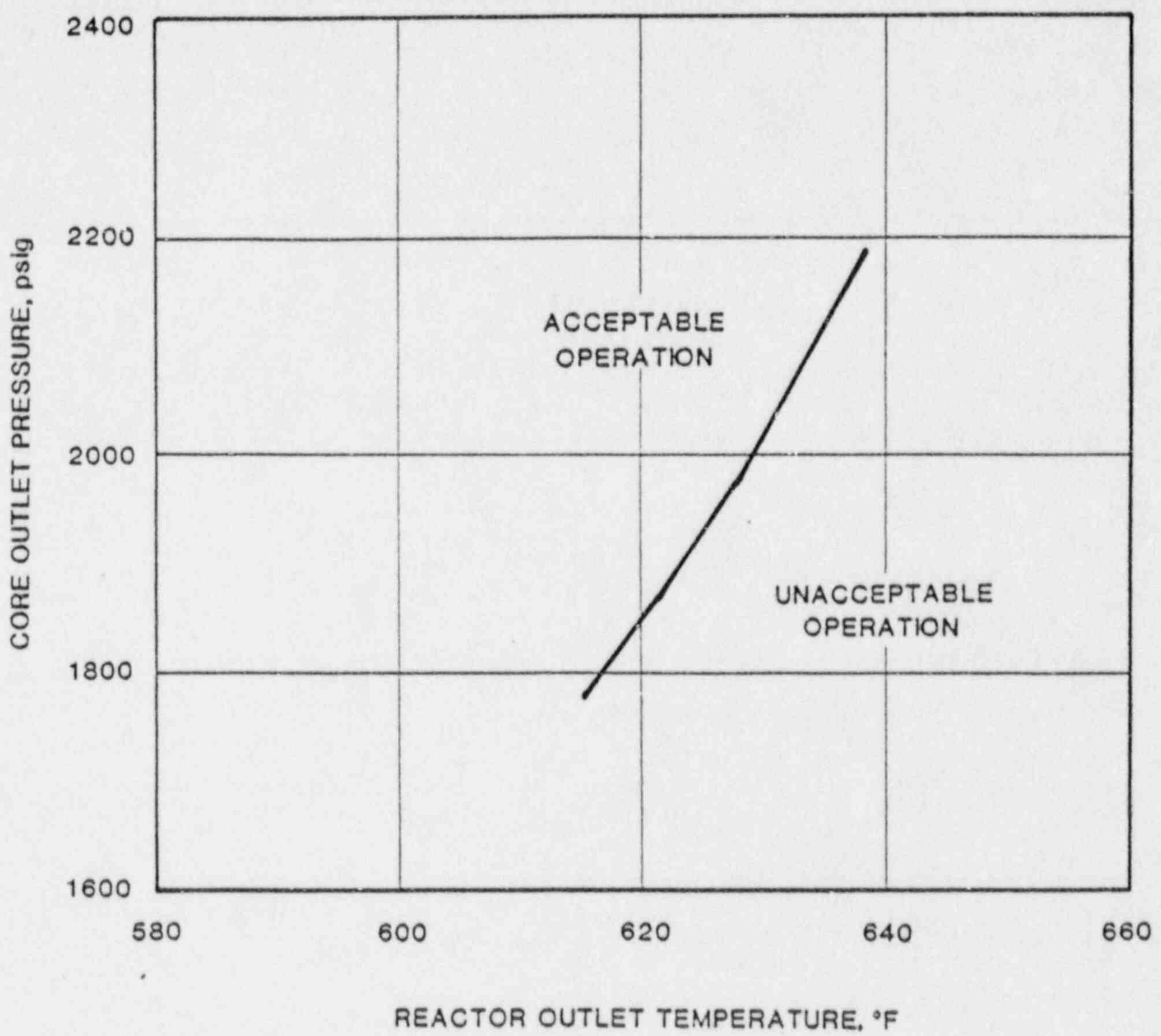
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump situation.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

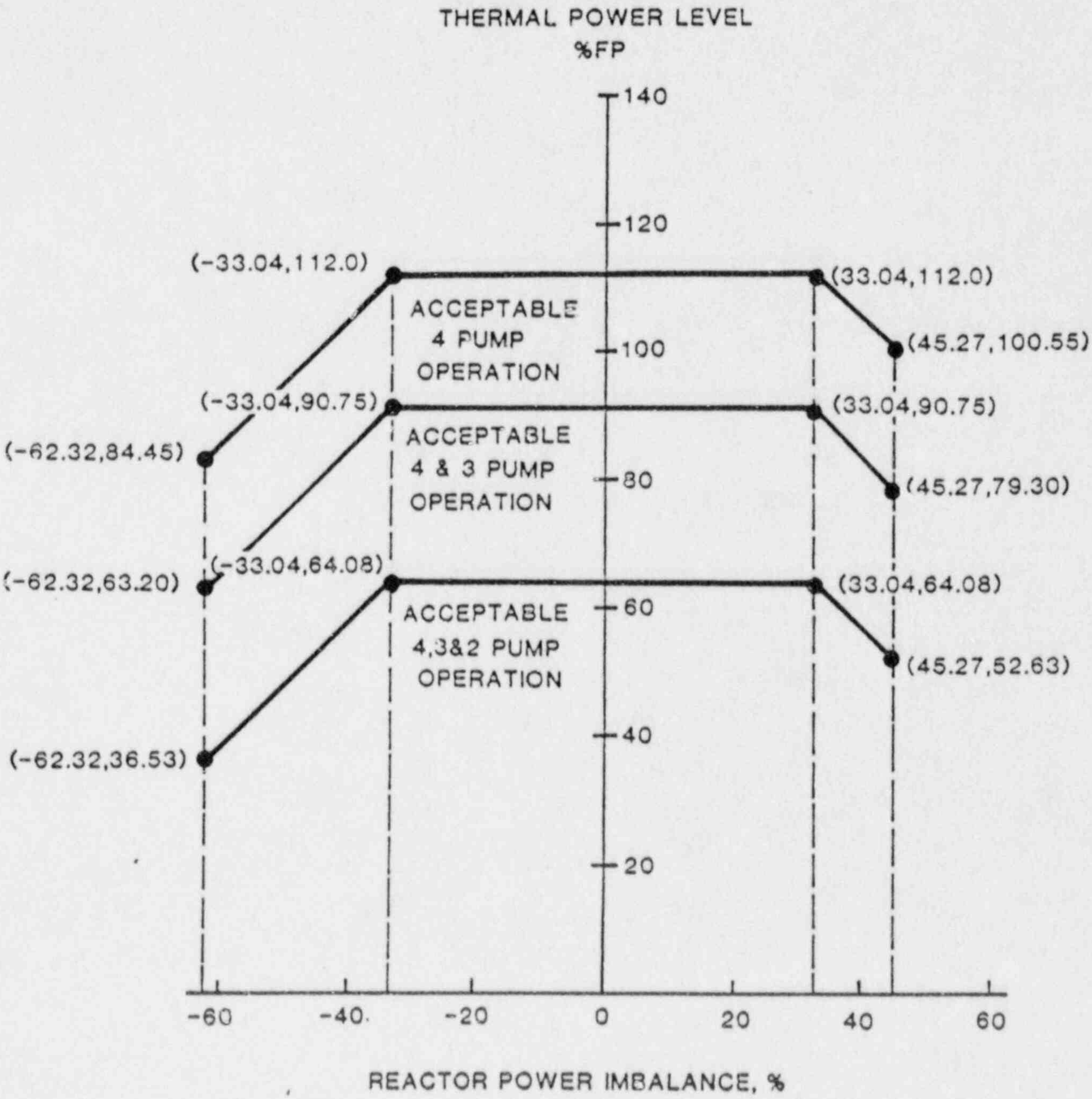
(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

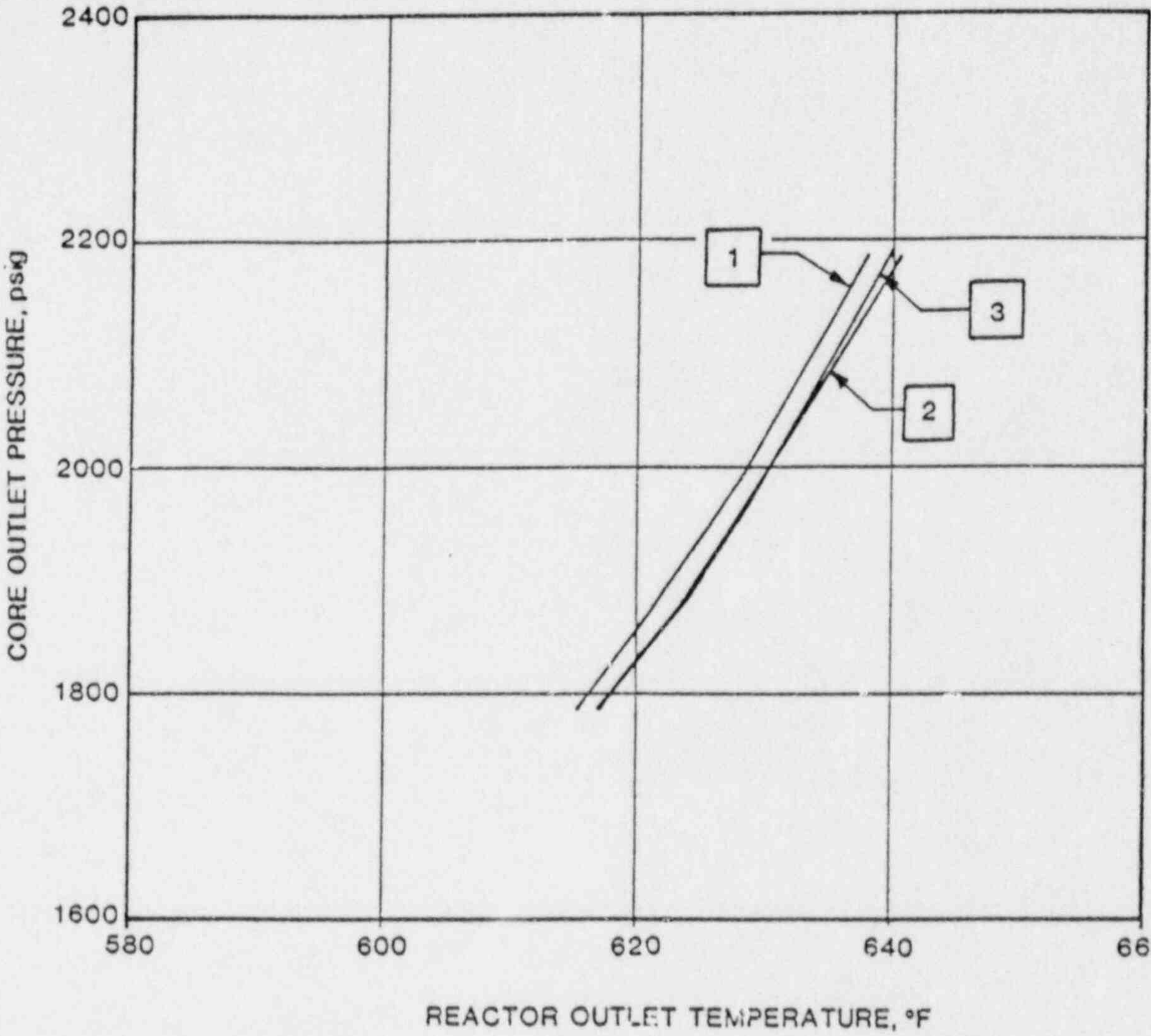


CORE PROTECTION SAFETY LIMIT FIGURE NO. 2.1-1

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 LIM[illegible])
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 ($11.75T_{out}$ -5103) 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]

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of ($11.75t_{out}$ -5143).

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.

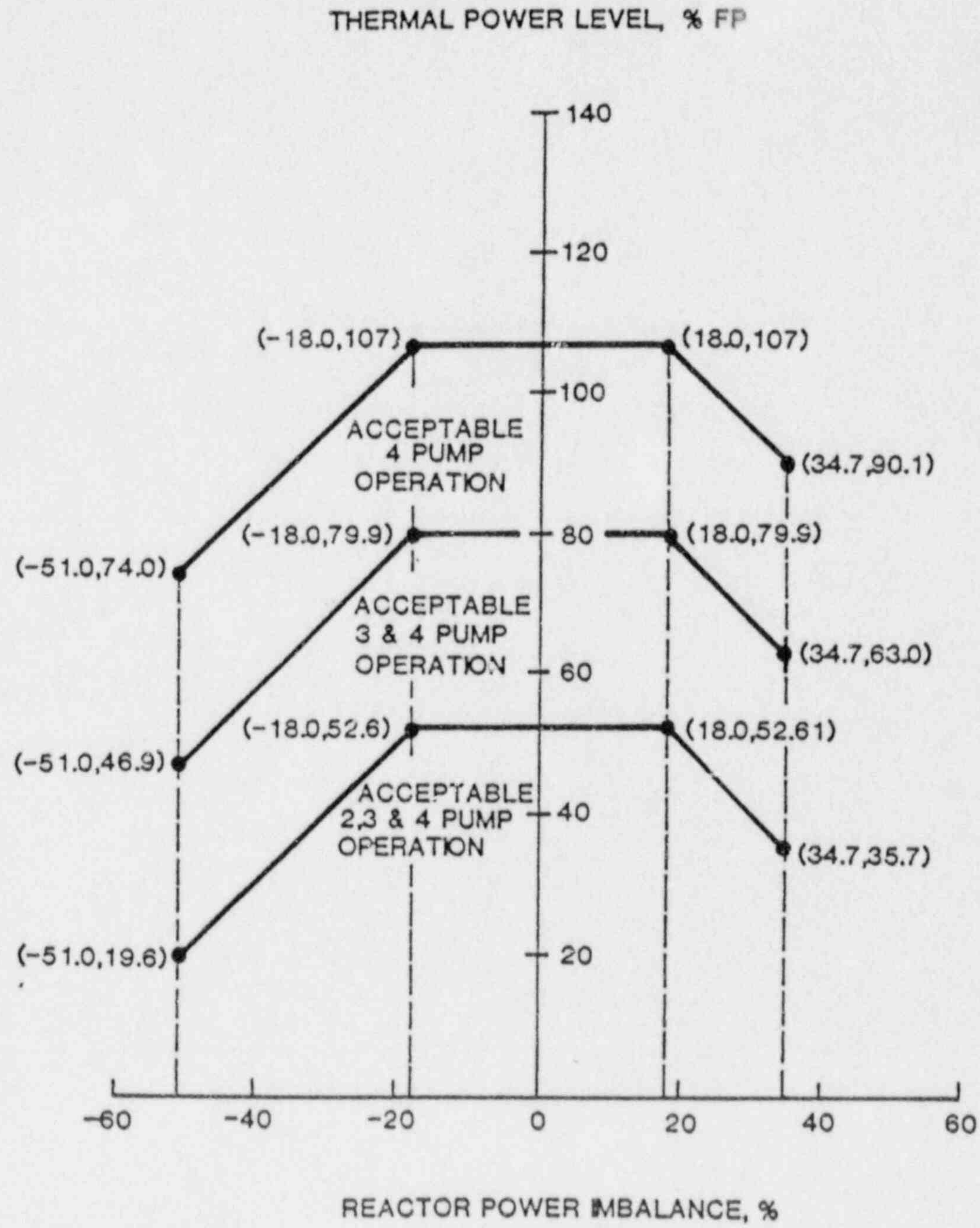
F. Shutdown Bypass

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 Setpoints
ANO-1, Figure 2.3-2



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

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

3.5.2.4 Quadrant tilt:

1. Except for physics tests, if quadrant tilt exceeds 4.12%, 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.

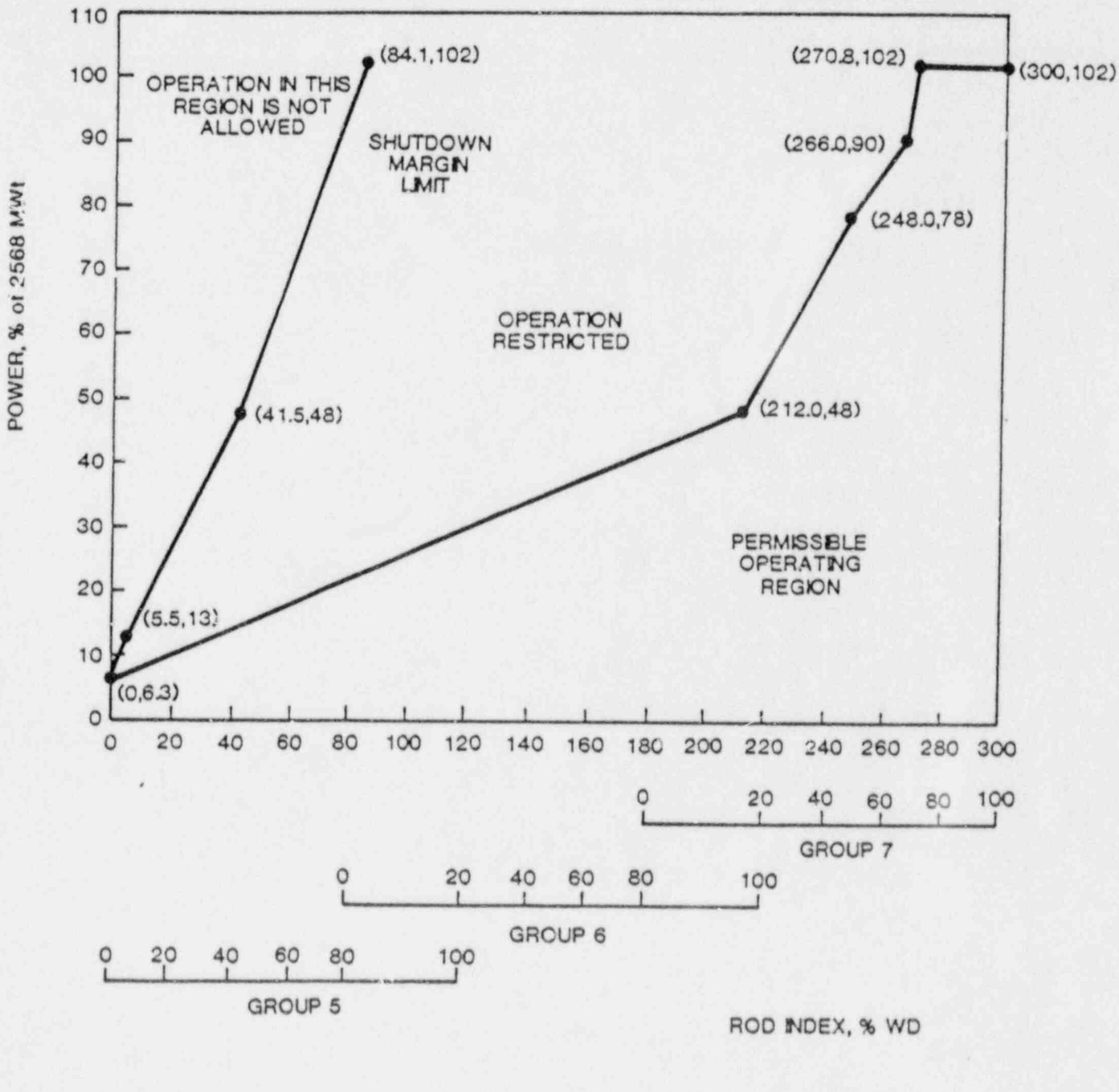
3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-4(A-C). If the imbalance is not within the envelope defined by Figure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

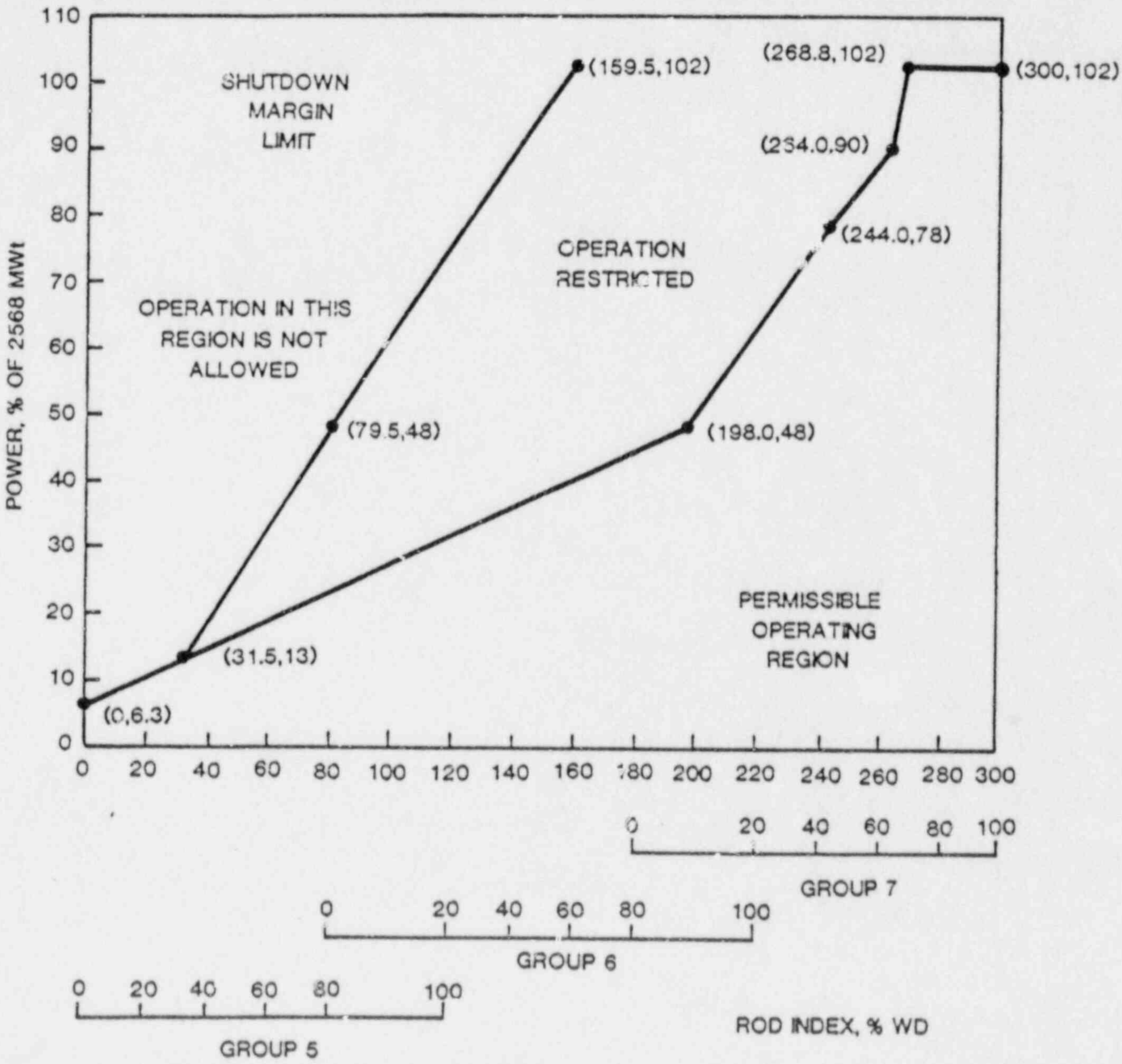
Bases

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

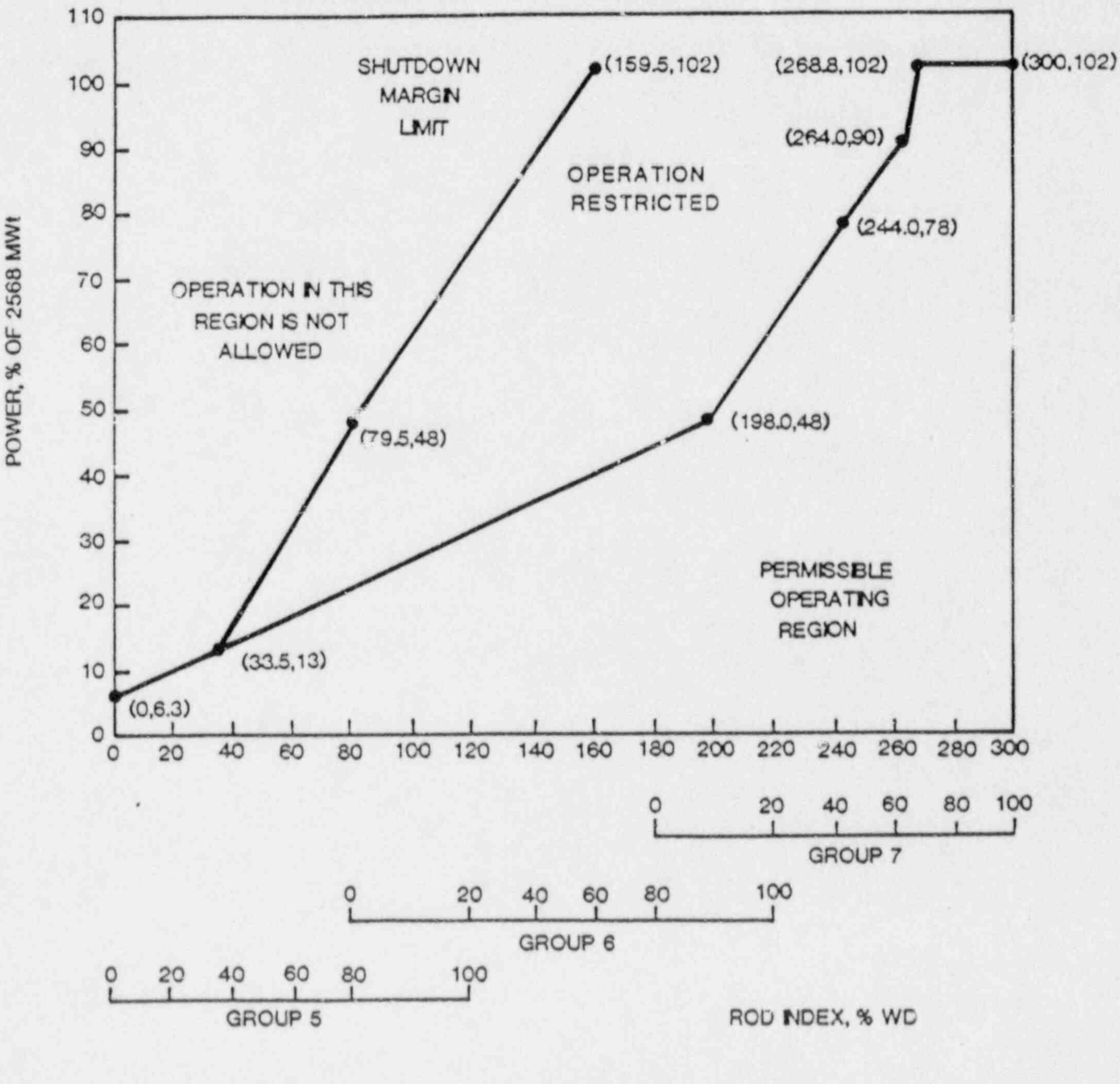
Rod Position Setpoints for 4-Pump Operation
From 0 to 27+10/-0 EFPD ANO-1 Cycle 9
Figure 3.5.2-1A



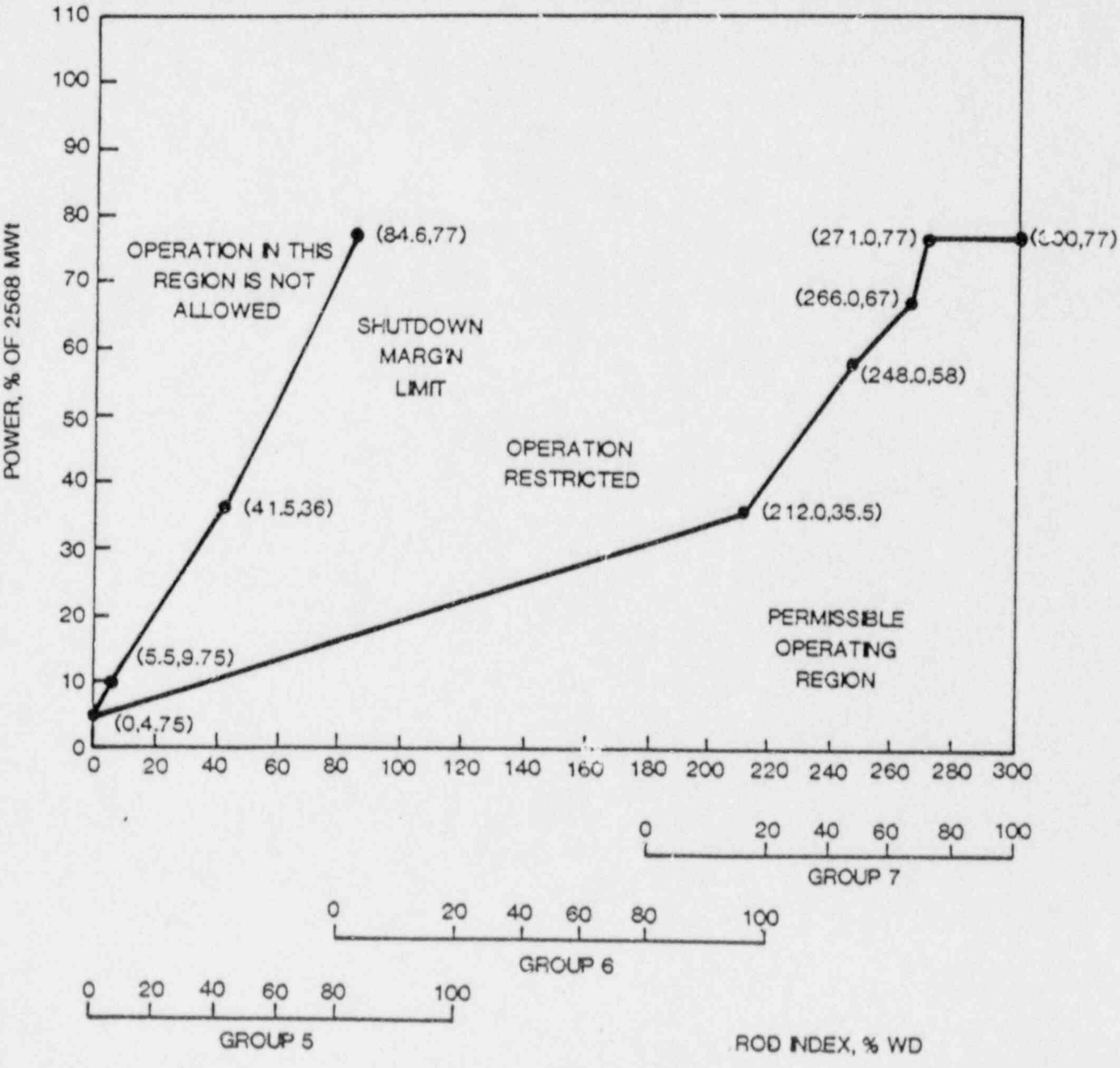
Rod Position Setpoints for 4-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD ANO-1 Cycle 9
Figure 3.5.2-1B



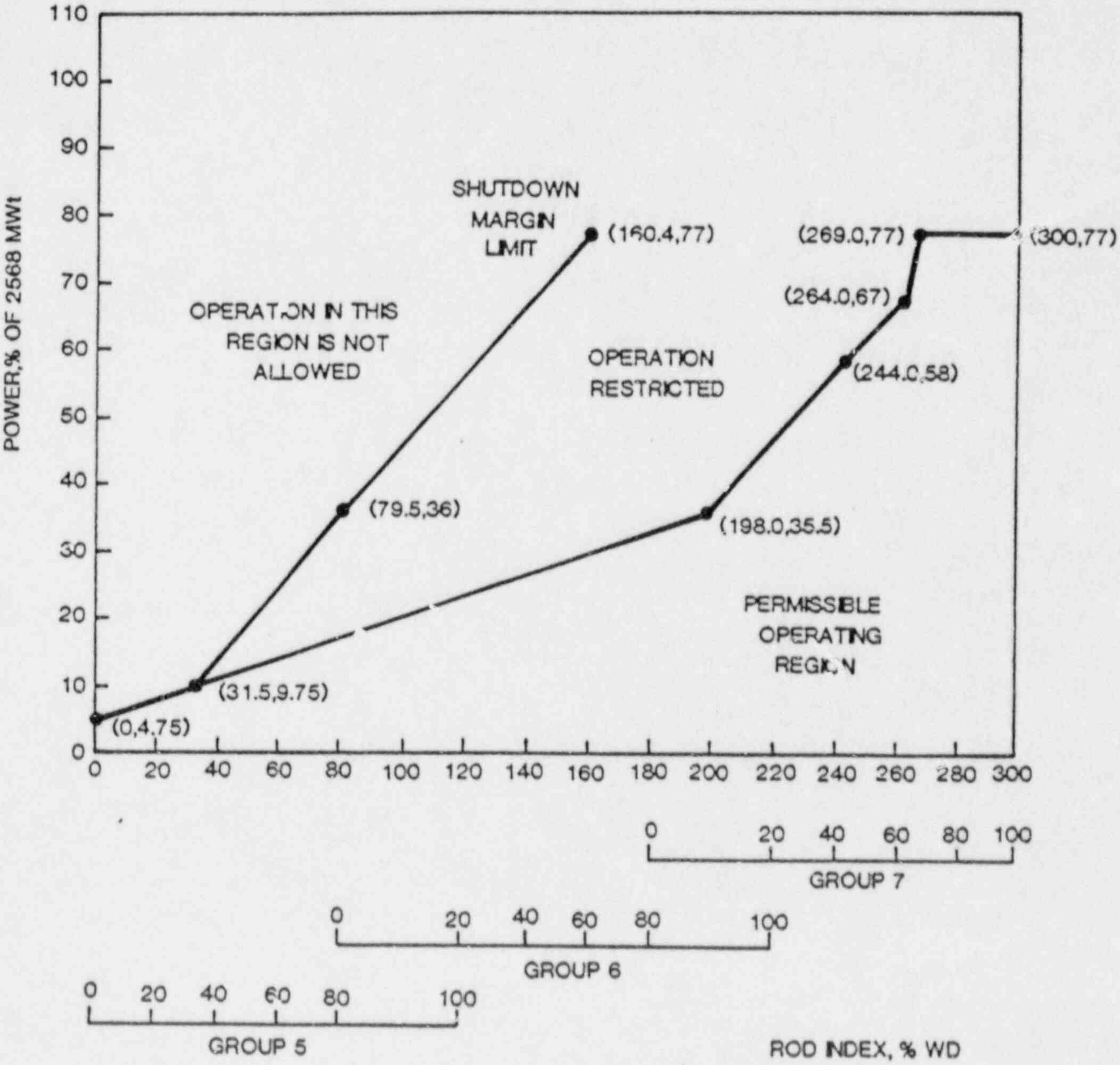
Rod Position Setpoints for 4 - Pump Operation
After 360 +50/-10 EFPD ANO-1 Cycle 9
Figure 3.5.2-1C



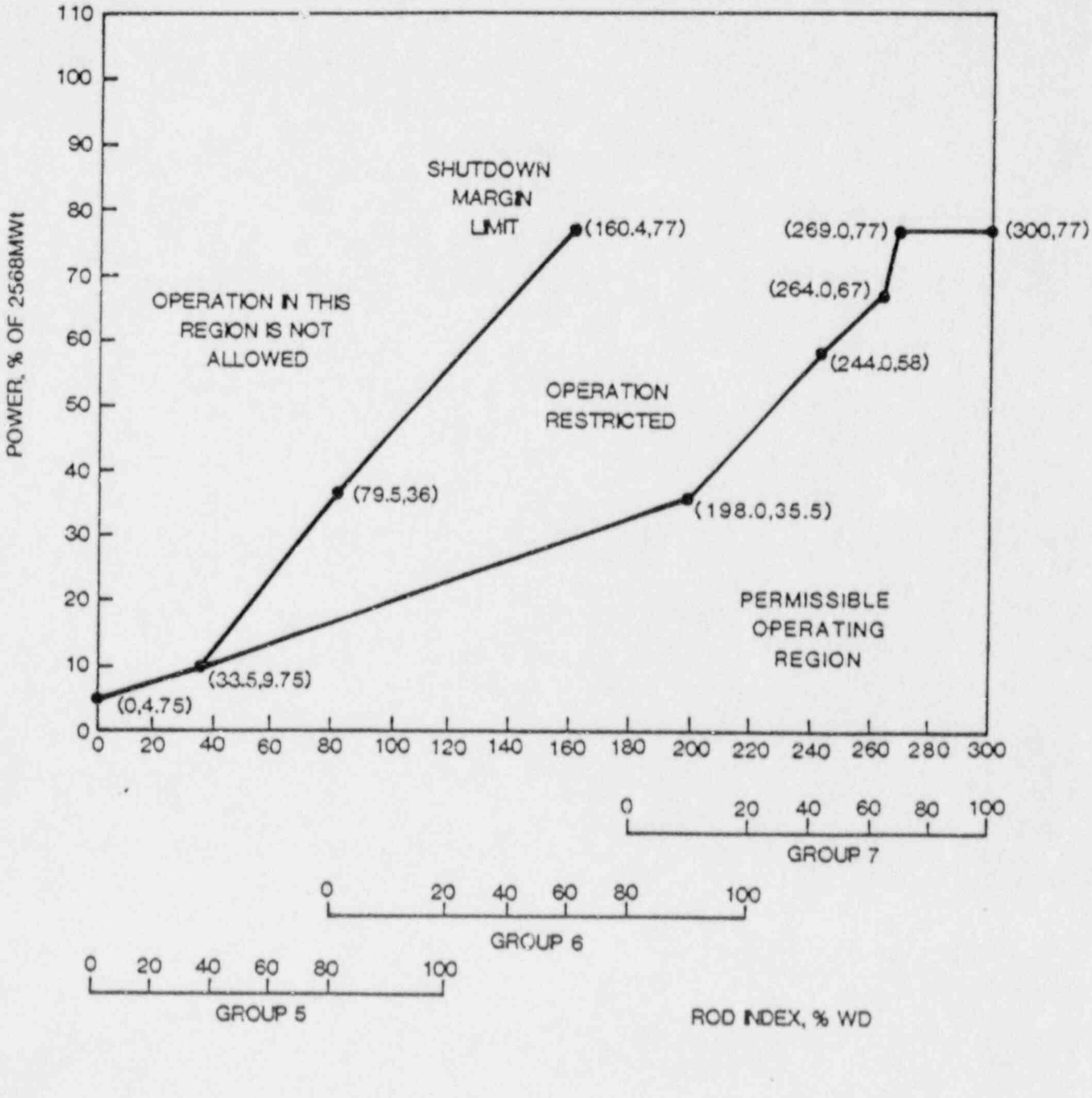
Rod Position Setpoints for 3-Pump Operation
From 0 to 27+10/-0 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2A



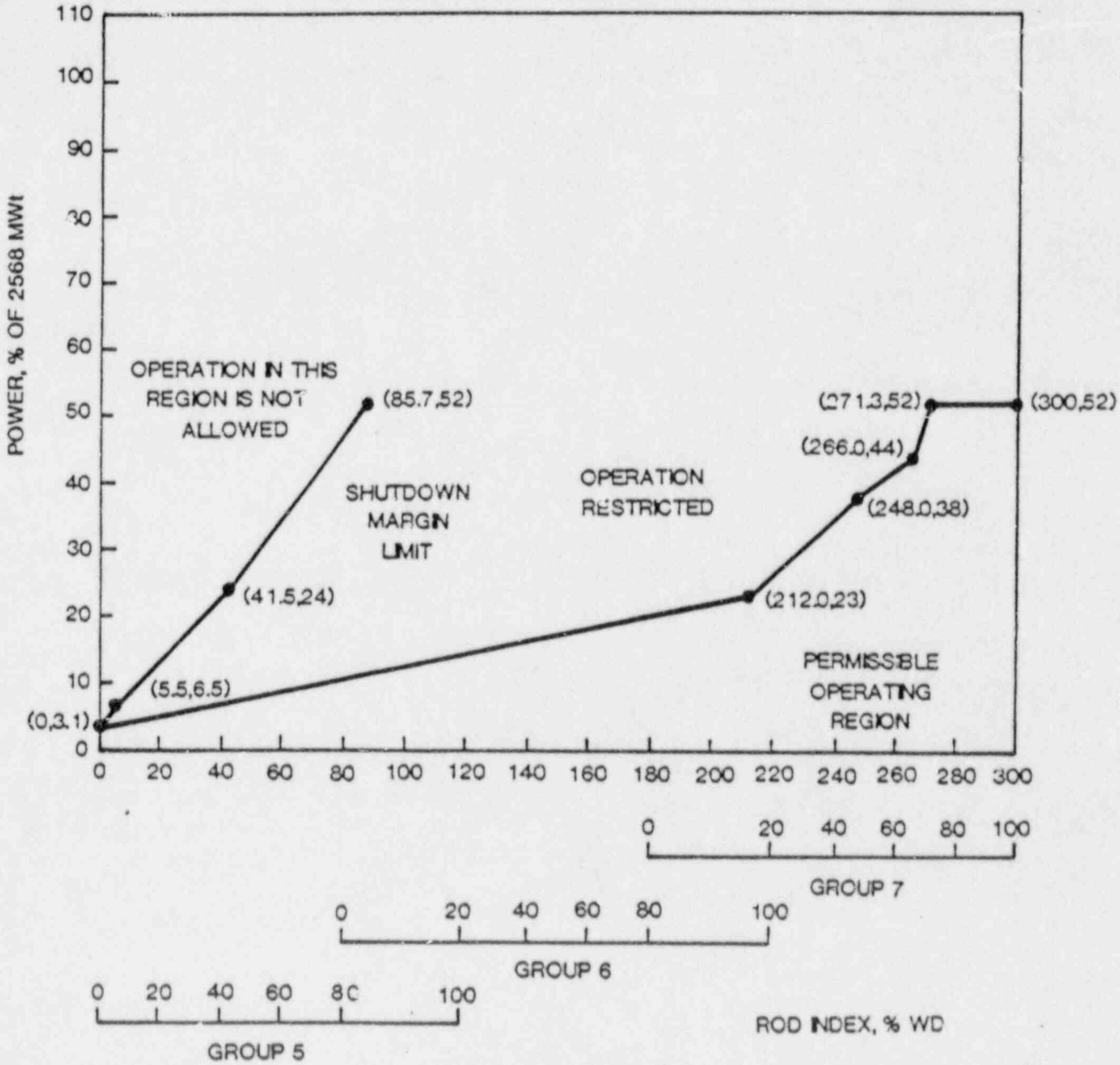
Rod Position Setpoints for 3-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2B



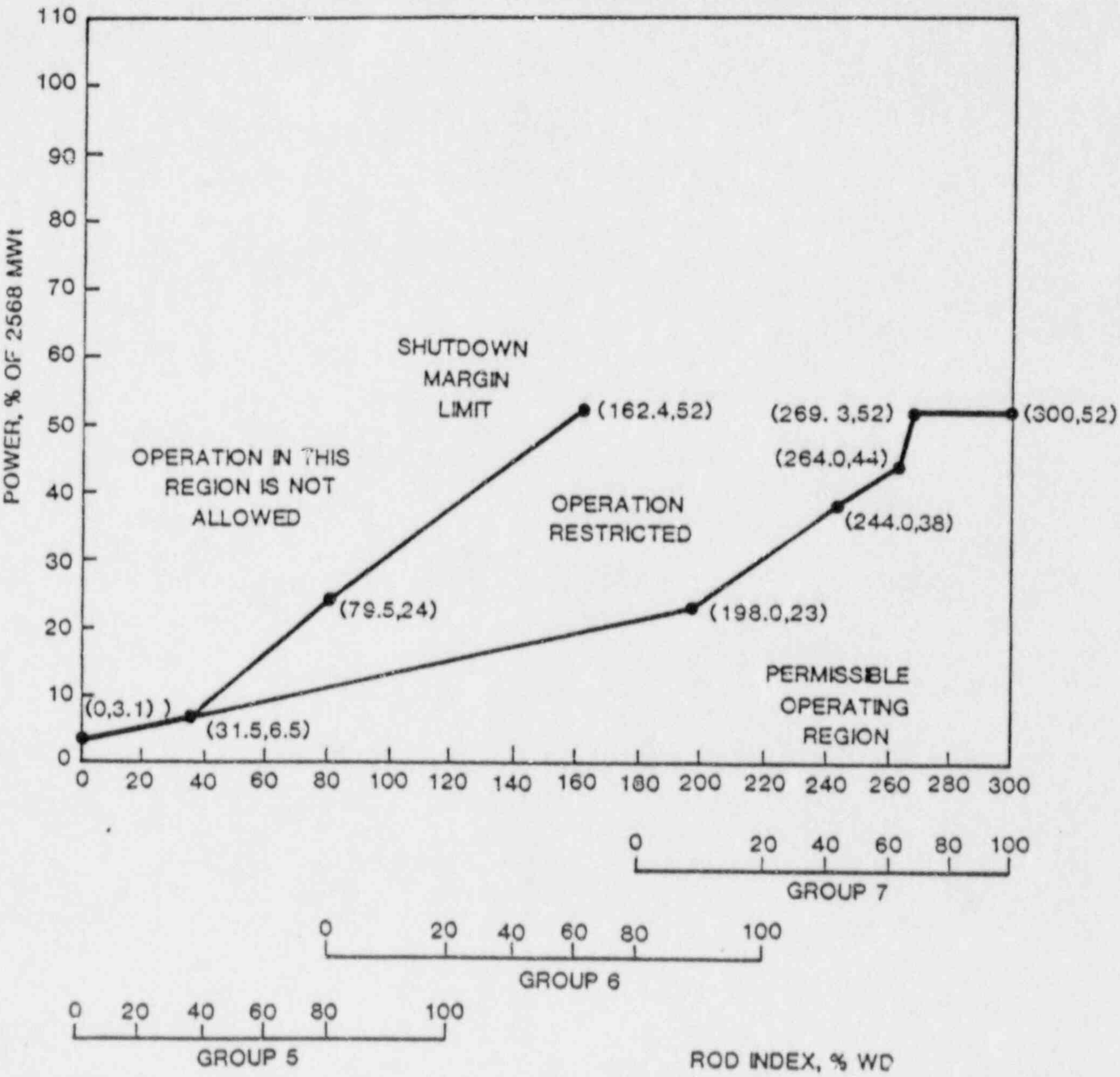
Rod Position Setpoints for 3-Pump Operation
After 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-2C



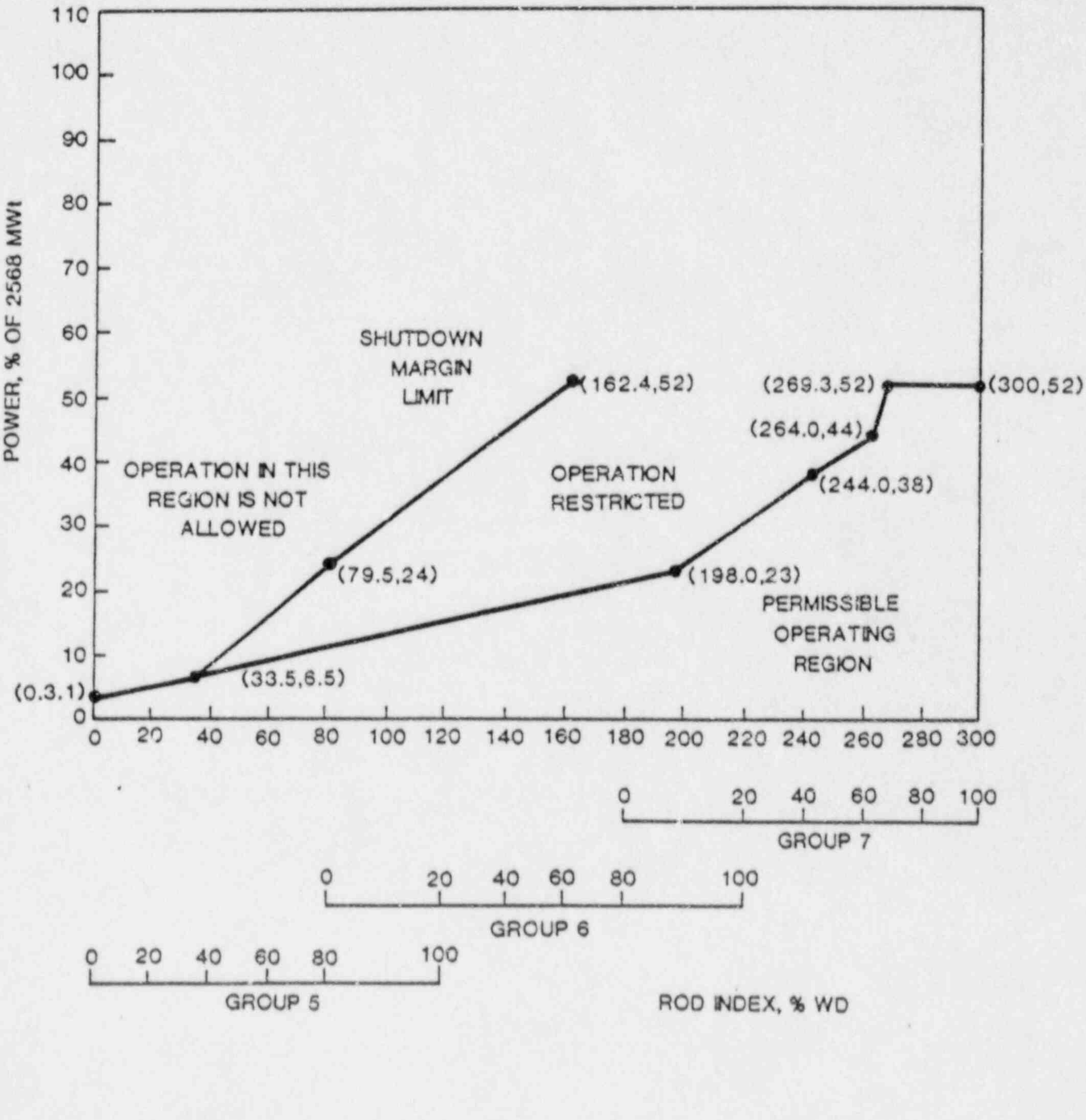
Rod Position Setpoints for 2-Pump Operation
From 0 to 27+10/-0 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3A



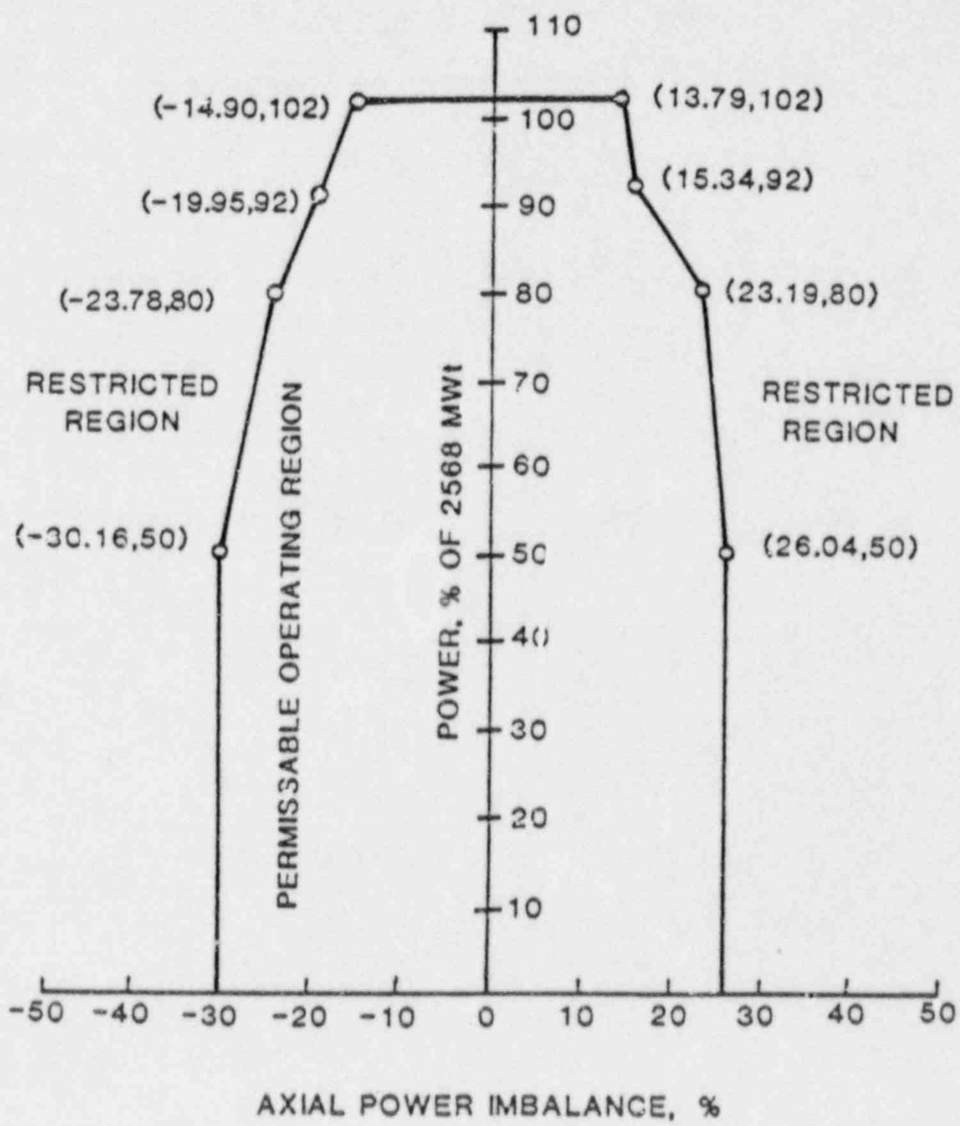
Rod Position Setpoints for 2-Pump Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3B



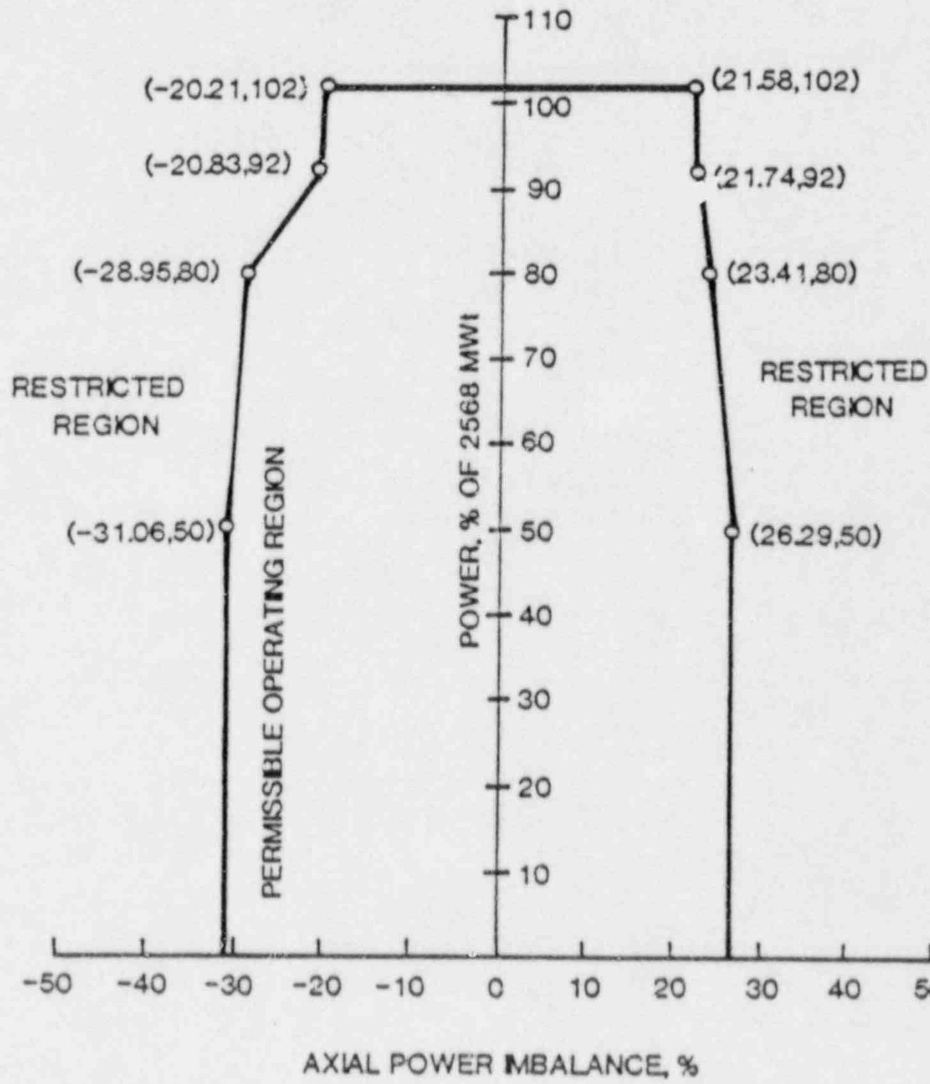
Rod Position Setpoints for 2-Pump Operation
After 360 +50/-10 EFPD -- ANO-1 Cycle 9
Figure 3.5.2-3C



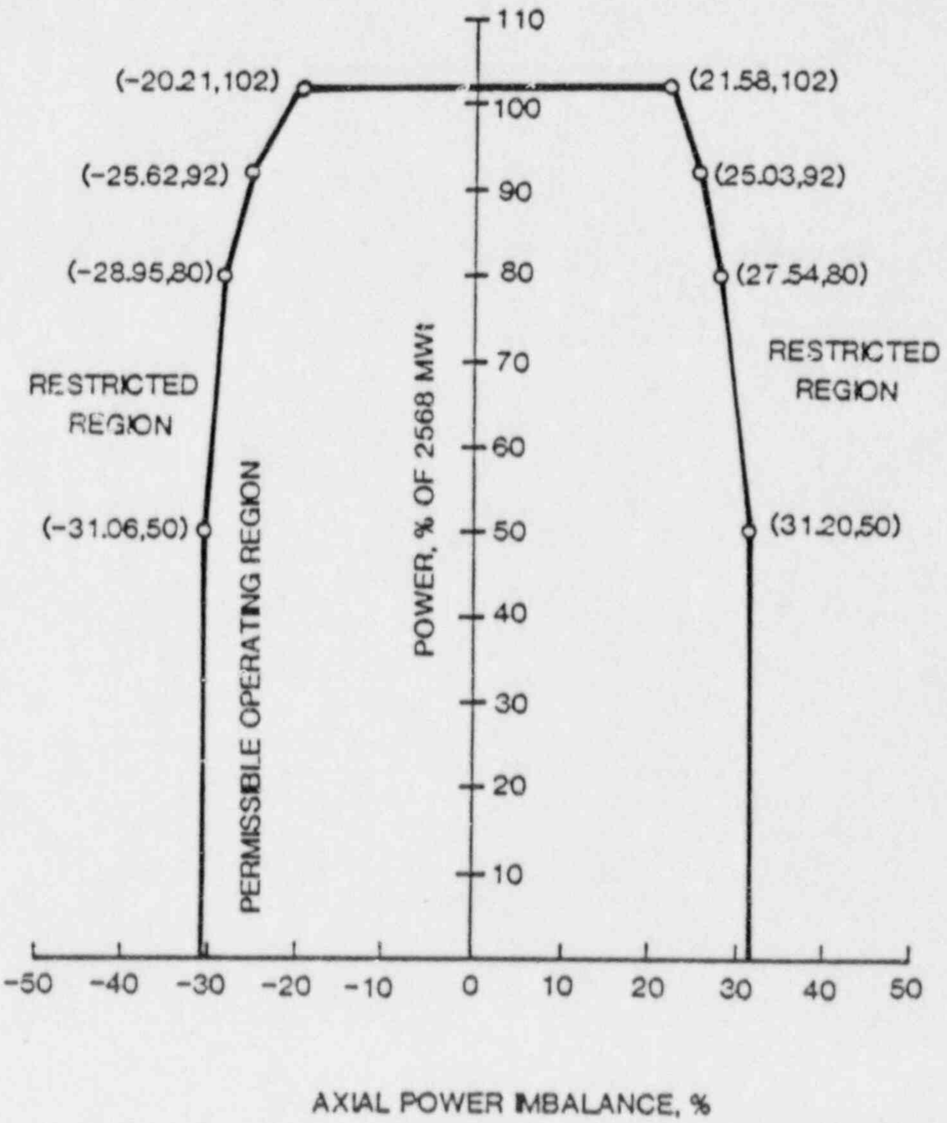
Operational Power Imbalance Setpoints for Operation
From 0 to 27+10/-0 EFPD -- ANO-1, Cycle 9
Figure 3.5.2-4A



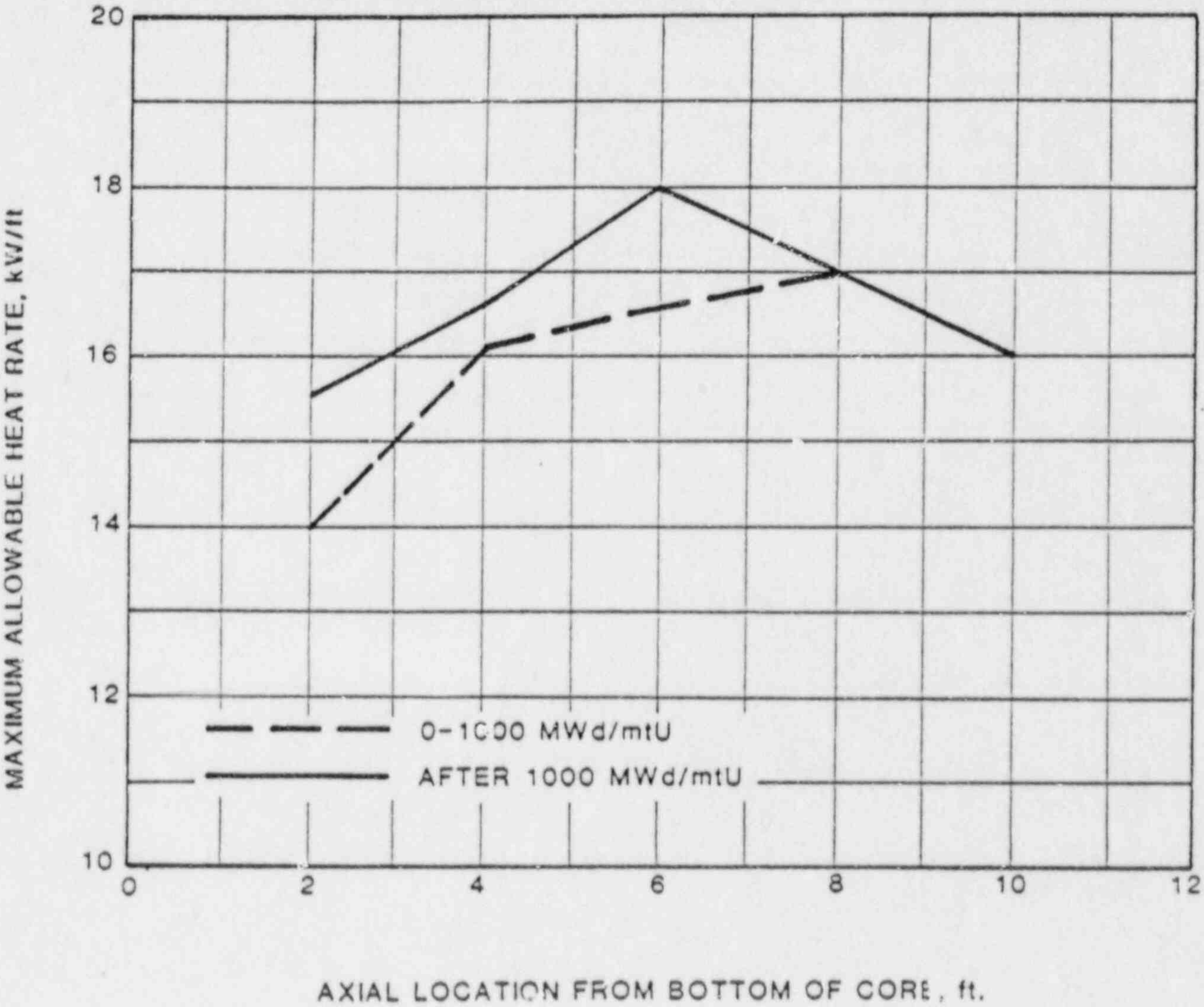
Operational Power Imbalance Setpoints for Operation
From 27+10/-0 to 360 +50/-10 EFPD -- ANO-1, Cycle 9
Figure 3.5.2-4B



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