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METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 30

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

Sworn and subscribed to me this ______ day of ______________, 1976

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Notary Public

NOTARY PUBLIC

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-289
OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 30 to Appendix A of the Operating License for Three Mile Island Nuclear Station, Unit 1, dated January 13, 1976 and filed with the U.S. Nuclear Regulatory Commission January 13, 1976, has this 13th day January, 1976, been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania, and of Dauphin County, Pennsylvania, by deposit in the United States Mail, addressed as follows:

Mr. Weldon B. Arehart, Chairman
Board of Supervisors of
Londonderry Township
R.D. #1, Geyers Church Road
Middletown, Pennsylvania 17057

Mr. Charles P. Hoy, Chairman
Board of County Commissioners of
Dauphin County
Dauphin County Courthouse
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

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Metropolitan Edison Co. (Met-Ed)
Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

TECHNICAL SPECIFICATION CHANGE REQUEST NO. (Proposed Change 47)

The licensee requests that the attached changed pages replace pages 2-1, 2-2, 2-3, 2-5, 2-6, 2-7, 2-9, 3-16, 3-34, 3-35, 3-36, figures 2.1-1, 2 & 3, figures 2.3-1 & 2, and figures 3.5-2A thru F of the existing Technical Specifications.

REASON FOR PROPOSED CHANGE

These changes to technical specifications are necessary to ensure safe operation of TMI-1 at rated power of 2535 MWt for the duration of Cycle 2 and are based on a Cycle 1 burnup of 440 $\pm$ 10 EFPD.

Changes to the present technical specifications are necessary as a result of: the effects of introducing 56 fresh batch 4 fuel assemblies combined with relocation of once burned Batch 2 and 3 fuel assemblies; use of the B&W-2 CHF correlation with a 95/95 confidence level, and extended pressure application to 1750 psi; use of a RC flow equal to 106.5% of Cycle 1 design flow; and ECCS Final Acceptance Criteria (FAC).

The 56 batch 4 fuel assemblies are not in general the technical specification limiting assemblies. Their presence combined with relocation of the once burned batch 2 and 3 assemblies produces a redistribution of fuel and assemblies which results in changed core physics and thermal-hydraulic calculations. Further burnup and the cycle 2 locations of the batch 2 and 3 assemblies results in these assemblies being the limiting assemblies thermally and mechanically. Other factors that were considered in the derivation of the Cycle 2 specification limits are the slight differences between the new and once burned fuel assemblies. These minor differences are reduced active length, slightly higher pellet density, and improved flow characteristics for the new assemblies compared to the burned assemblies.

In addition to Fuel changes, the use of the B&W-2 CHF correlation combined with the assumed minimum Flow of 106.5% have had an influence on these proposed specifications. Use of this correlation and flow more realistically predict core performance but still provide conservative technical specification limits.

Met-Ed submitted revised technical specifications based on FAC guidelines in our Technical Specification Change Request 17 (August 8, 1975). Additional ECCS supporting information was provided in our letters of April 19, 1975, July 9, 1975, July 15, 1975, and October 23, 1975. The attached changed pages for TMI-1 Cycle 2 operation include changes that were requested in Change Request 17 which apply to Cycle 2 operation. The number 17 beside the marginal bars indicates those changes that were requested in Change Request 17 and all other marginal bars indicate Cycle 2 changes. All appropriate cycle 2 Technical Specifications were developed based on FAC guidelines.

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SAFETY EVALUATION JUSTIFYING CHANGE

The influence of the minor design changes of the batch 4 fuel assemblies (i.e. increased pellet density and reduced active length) compared to the batch 2 & 3 assemblies have been considered for Cycle 2 operation. The effect on core flow distribution as a result of batch 4 assembly end fittings, and the absence of orifice rods in the core's periphery have also been considered. These proposed specifications were developed conservatively accounting for the above considerations for all applicable transient and steady state conditions.

In general the governing hydraulic, mechanical, thermal and nuclear parameters have been developed using NRC accepted practices, models, and correlations. Note, however, that the applicability of the B&W-2 correlation has been extended downward to 1750 psia. This extention is justified since this correlation produces conservative predictions of data in this range and use of this correlation provides conservative but realistic results.

The FSAR accidents wherein core design or fuel loading are important were considered for Cycle 2 operation. In all instances except for the LOCA, which is based on the FAC guidelines of 10 CFR 50.46 and 10 CFR 50 appendix K, the parameters for the Cycle 2 core are bounded by the FSAR analysis; consequently, the FSAR analyses are valid for Cycle 2. All other FSAR accidents are independent of core physics parameters and therefore also remain valid. The LOCA analysis has been reported in BAW-10103 Rev. 1. With our Change Request 17 and its supporting correspondence we have shown that the TMI-1 case is at least as conservative as the analyses done in BAW-10103 Rev. 1.

Additional and more detailed information supporting the above safety evaluation and providing more detailed design information is provided in the attached Babcock and Wilcox Cycle 2 Reload Report.

Based on the above, it is concluded that this change does not increase the probability of occurrence or the severity of an accident. This change does not create the possibility of occurrence of an accident not previously analyzed. Therefore, this change does not represent undue risk to the health and safety of the public and continued operation of TMI-1 at rated power of 2535 MWt is justified.

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 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 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 (solid line) 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 would 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 the B&W-2 correlation. (1) the B&W-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. 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 1.3. A DNBR of 1.3 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered

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a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is $139.8 \times 10^{+6}$ lbs/h, which is less than the actual flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (2) with potential fuel densification effects;

$$F_q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_z^N = 1.50$$

The 1.5 axial peaking factor associated with the cosine flux shape provides a lesser margin to a DNBR of 1.3 than the 1.7 axial peaking factor associated with a lower core flux distribution. For this reason the cosine flux shape and the associated $F_z = 1.50$ is more limiting and thus the more conservative assumption.

The 1.50 cosine axial flux shape in conjunction with $F\Delta H = 1.78$ define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

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:

a. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than a 1.3 DNBR.

b. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 19.6 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 specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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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 a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent, (3) whichever condition is more restrictive.

Using a local quality limit of 22 percent at the point of minimum DNBR as a basis for curve 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 B&W-2 correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power for three pump operation is 86.7 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.08 = 80.7 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

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 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.3 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

REFERENCES

(1) FSAR, Section 3.2.3.1.1

(2) FSAR, Section 3.2.3.1.1.c

(3) FSAR, Section 3.2.3.1.1.k

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2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION 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 pre-selected 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 set points 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 105.5% of rated power. Adding to this the possible variation in trip set points 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 (1).

a. Overpower trip based on flow and imbalance

The power level trip set point 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 1.3 should a low flow condition exist due to any malfunction.

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The power level trip set point 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 set point 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. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108 percent and reactor flow rate is 100 percent, or flow rate is 92.6 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.7 percent and reactor flow rate is 74.7 percent or flow rate is 69.2 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.9 percent and reactor flow rate is 49.2 percent or flow rate is 45.4 percent and the power level is 49 percent.

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 the 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 and associated reactor power/reactor power-imbalance boundaries by 1.08 percent for a one percent flow reduction.

b. Pump monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below 1.3 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. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. 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.

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The low pressure (1800 psig) and variable low pressure (11.75 Tout - 5103) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (3,4).

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of (11.75 Tout - 5103).

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range.

The calibrated range of the temperature channels of the RPS is 520 to 620 F. The trip setpoint of the channel is 619 F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is $\pm$1F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620F even under worst case conditions. The safety analysis used a high temperature trip set point of 620F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

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.

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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
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2. Nuclear Power based on flow(2) and imbalance, max. of rated power	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	Bypassed
3. Nuclear power based(5) on pump monitors, max. % of rated power	NA	NA	91%	Bypassed
4. High reactor coolant system pressure, psig, max.	2355	2355	2355	1720(4)
5. Low reactor coolant system pressure, psig, min.	1800	1800	1800	Bypassed
6. Variable low reactor coolant system pressure, psig, min.	(11.75 Tout - 5103)(1)	(11.75 Tout - 5103)(1)	(11.75 Tout - 5103)(1)	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown
(4) Automatically set when other segments of the RPS (as specified) are bypassed
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.

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3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to maximum positive moderator temperature coefficient of reactivity at full power conditions.

Objective

To assure that the moderator temperature coefficient stays within the limits calculated for safe operation of the reactor.

Specification

3.1.7.1 The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

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 17
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 $+0.5 \times 10^{-4}$ $\Delta K/K/F$. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta K/K/F$.

The experimental value of the moderator coefficient will be corrected to obtain the hot full power moderator coefficient. The correction factor will be verified during startup testing on earlier B&W reactors.

The Final Acceptance Criteria states that post-LOCA clad temperature will not 17
exceed 2200 F.

REFERENCES

(1) FSAR, Section 14
(2) FSAR, Section 3

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f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2., operation may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "E" above is in groups 5, 6, 7, or 8, the other rods in the group shall be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

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. 17

3.5.2.4 Quadrant tilt:

a. Except for physics tests if quadrant tilt exceeds 4 percent, power shall be reduced immeidately to below the power level cutoff (see Figures 3.5-2A, 3.5-2B and 3.5-2C). Moreover, the power level cutoff value shall be reduced 2 percent for each 1 percent tilt in excess of 4 percent tilt. For less than four pump operation, thermal power shall be reduced 2 percent of the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of 4 percent.

b. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4 percent except for physics tests, or the following adjustments in setpoints and limits shall be made:

1. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt.

2. The control rod group withdrawal limits (Figures 3.5-2A 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E, and 3.5-2F) shall be reduced 2 percent in power for each 1 percent tilt in excess of 4 percent.

3. The operational imbalance limits (Figure 3.5-2G, 3.5-2H and 3.5-2I) shall be reduced 2 percent in power for each 1 percent tilt in excess of 4 percent.

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c. If quadrant tilt is in excess of 25 percent, 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 combinations is restricted as stated in 3.5.2.4.a, above.

d. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated power.

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3.5.2.5 Control rod positions:

a. Operating rod group overlap shall not exceed 25 percent ± 5 percent, between two sequential groups except for physics tests.

b. Except for physics tests or exercising control rods, the control rod insertion/withdrawal limits are specified on Figures 3.5-2A,3.5-2B, and 3.5-2C for four pump operation and Figures 3.5-2D, 3.5-2E, and 3.5-2F, for three or two pump operation. If the 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 four hours.

c. Except for physics tests, power shall not be increased above the power level cutoff (See Figures 3.5-2A, 3.5-2B and 3.5-2C) unless the xenon reactivity is within 10 percent of the equilibrium value foroperation at rated power and asymptotically approaching stability.

d. Core imbalance shall be monitored on a minimum frequency of once every two hours during power operation above 40 percent of rated power. Except for physics tests, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope defined by Figures 3.5-2G, 3.5-2H and 3.5-2I. If the imbalance is not within the envelope defined by Figures 3.5.-2G, 3.5-2H and 3.5-2I corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met. 17

e. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

3.5.2.7 A power map shall be taken to verify the expected power distribution at periodic intervals of approximately 10 full power days using the incore instrumentation detection system.

Bases

The power-imbalance envelope defined in Figures 3.5-2G, 3.5-2H, and 3.5-2I is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2J) such that the maximum clad temperature will not exceed the Final Acceptance Criteria (2200F).Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power imbalance envelope represents the boundary of operation

limited by the Final Acceptance Criteria only if the control rods are at the withdrawal/insertion limits as defined by Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E and 3.5-2F and if a 4 percent quadrant power tilt exists. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

The Rod index versus Allowable Power curves of Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E and 3.5-2F, describe three regions. These three regions are:

1. Permissible operating Region

2. Restricted Regions

3. Prohibited Region (Operation in this region is not allowed) 17

Note: Inadvertent operation within the Restricted Region for a period of 4 hours is not considered a violation of a limiting condition for operation. The limiting criteria within the Restricted Region are potential ejected rod worth and ECCS power peaking and since the probability of these accidents is very low especially in a 4 hour time frame, inadvertant operation within the Restricted Region for a period of 4 hours is allowed.

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The 25±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows: 17

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6 and 7 are overlapped 25 percent. The normal position at power is for groups 6 and 7 to be partially inserted.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% Δk/k at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% Δk/k at beginning of life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% Δk/k ejected rod worth at rated power. 17

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, than manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed at least every two hours until the computer is returned to service.

The quadrant power tilt limits set forth in Specification 3.5.2.- have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6.

During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

Test Power	Trip Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.5%

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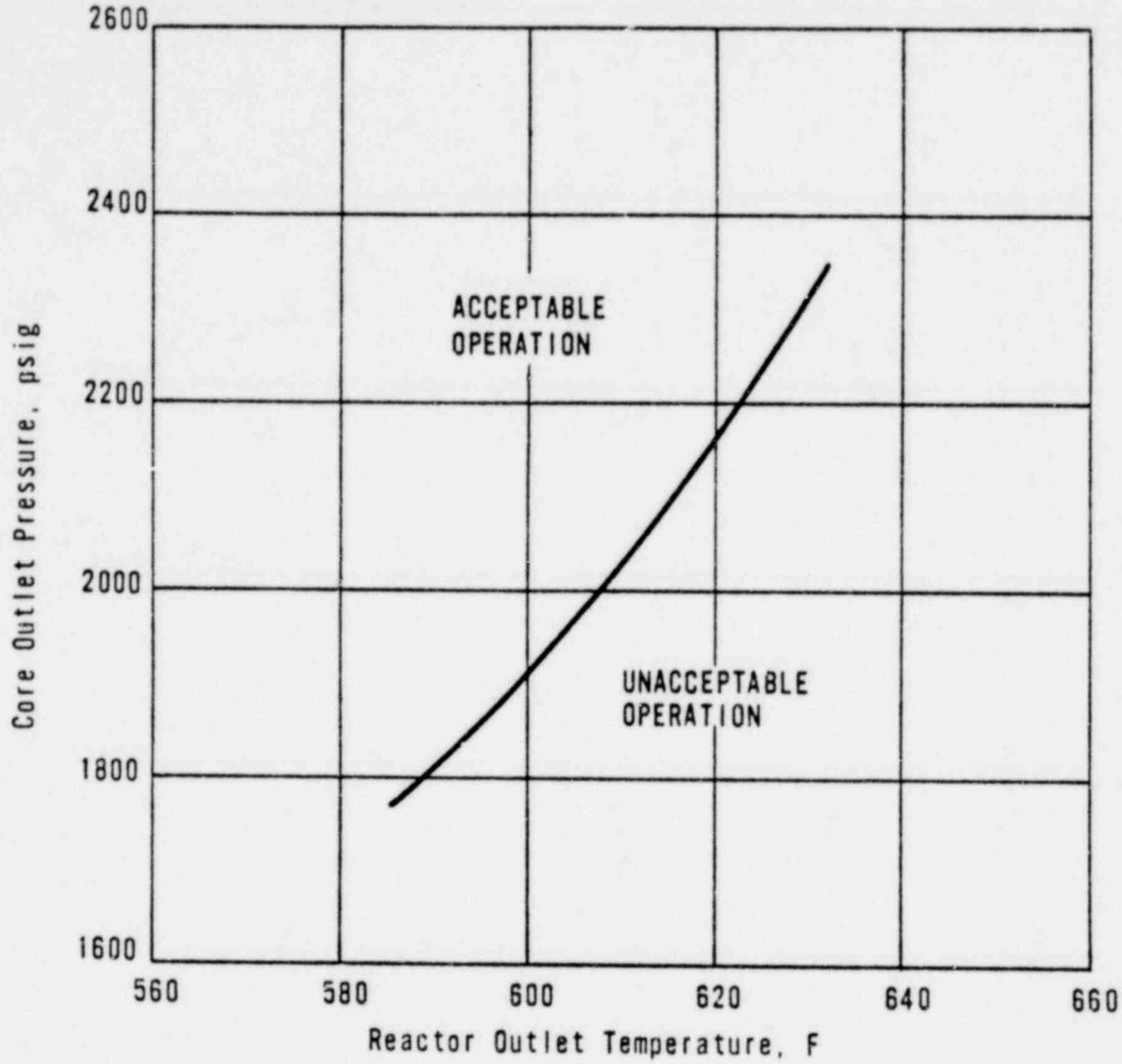
REFERENCES

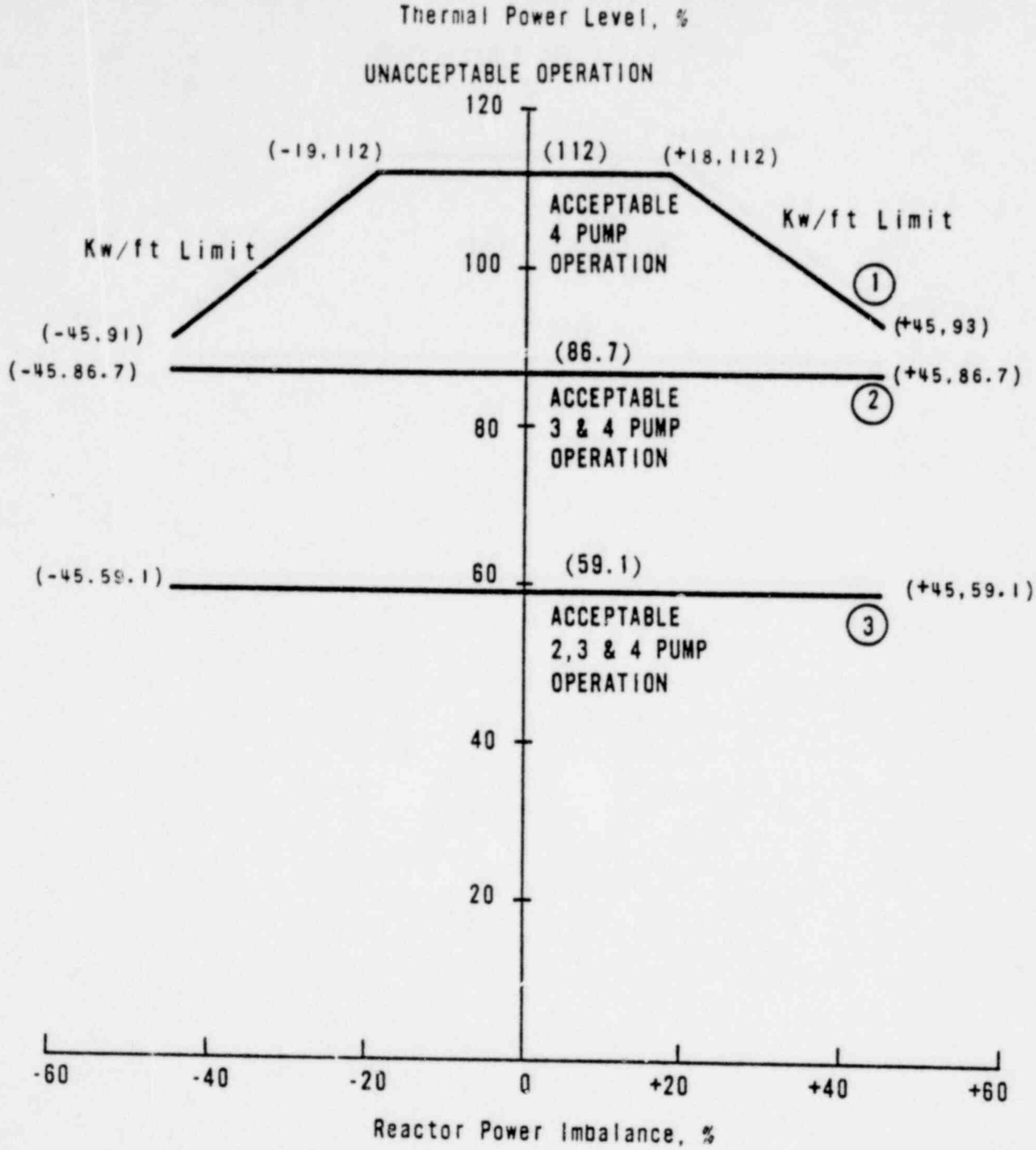
(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2



CORE PROTECTION SAFETY LIMIT

Figure 2.1-1



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

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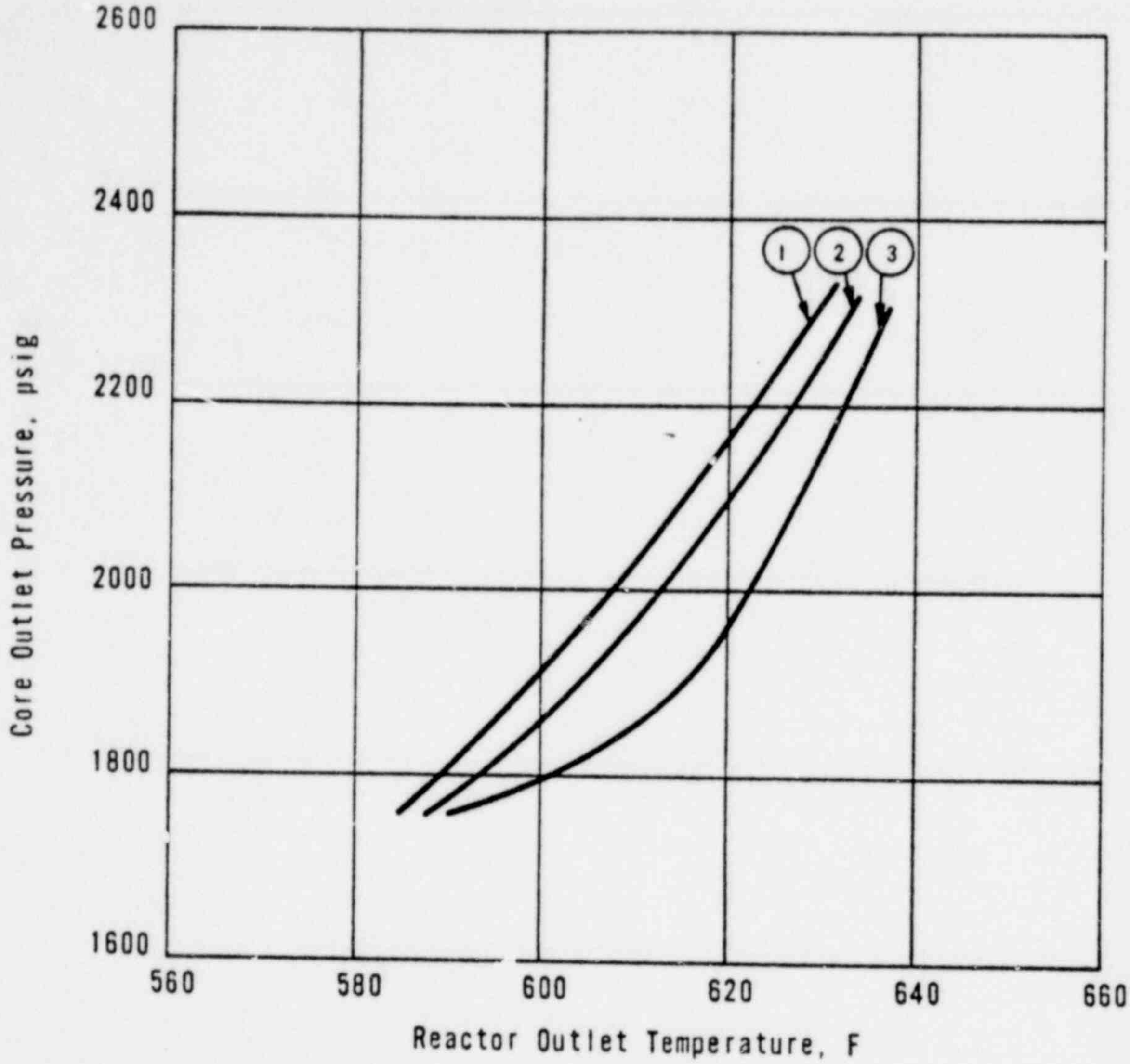
CORE PROTECTION SAFETY LIMITS

Figure 2.1-2



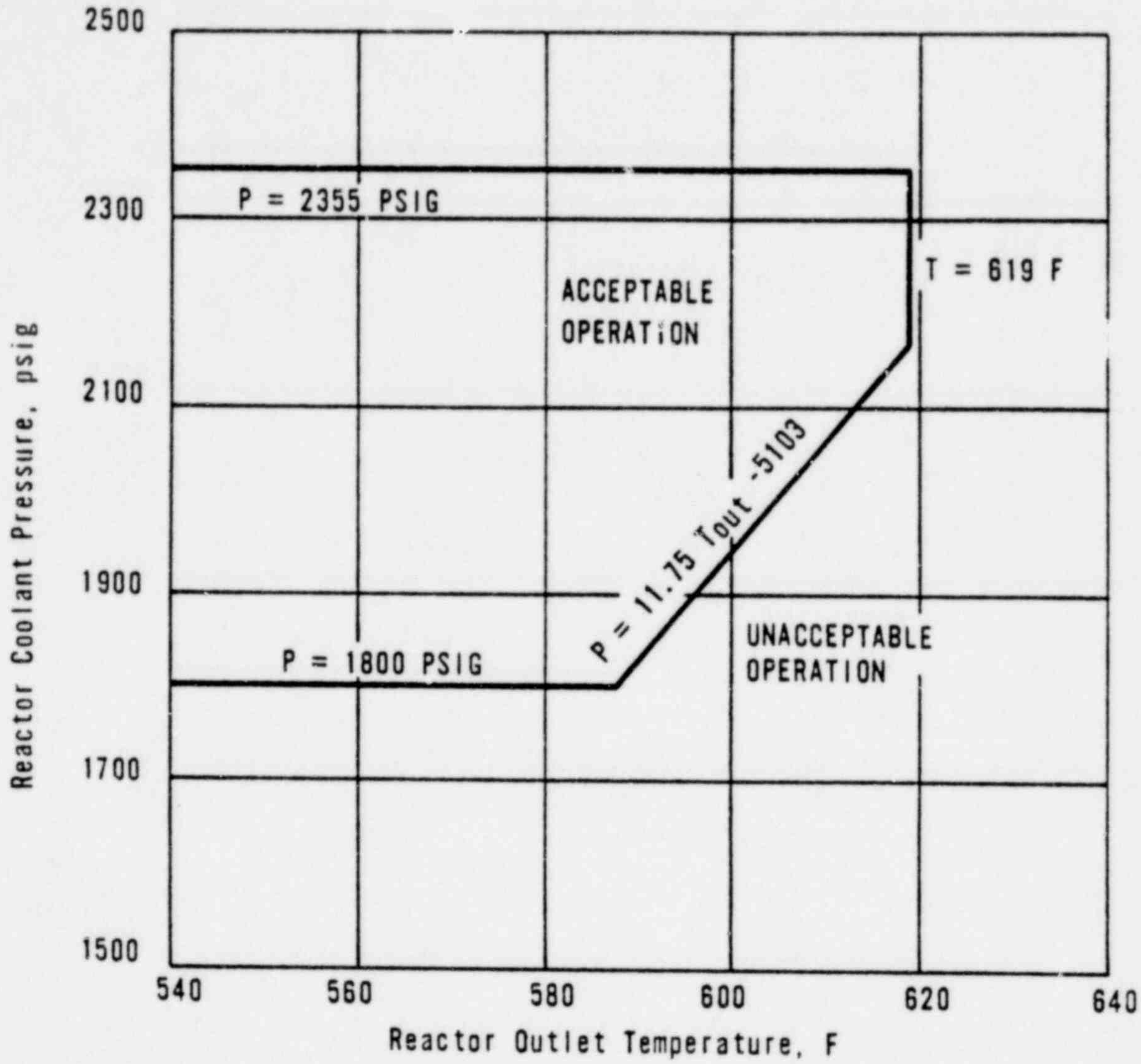
CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	86.7%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.1%	One Pump in Each Loop (Quality Limit)

* 106.5% of Cycle 1 Design Flow.

CORE PROTECTION SAFETY

BASES Figure 2.1-3

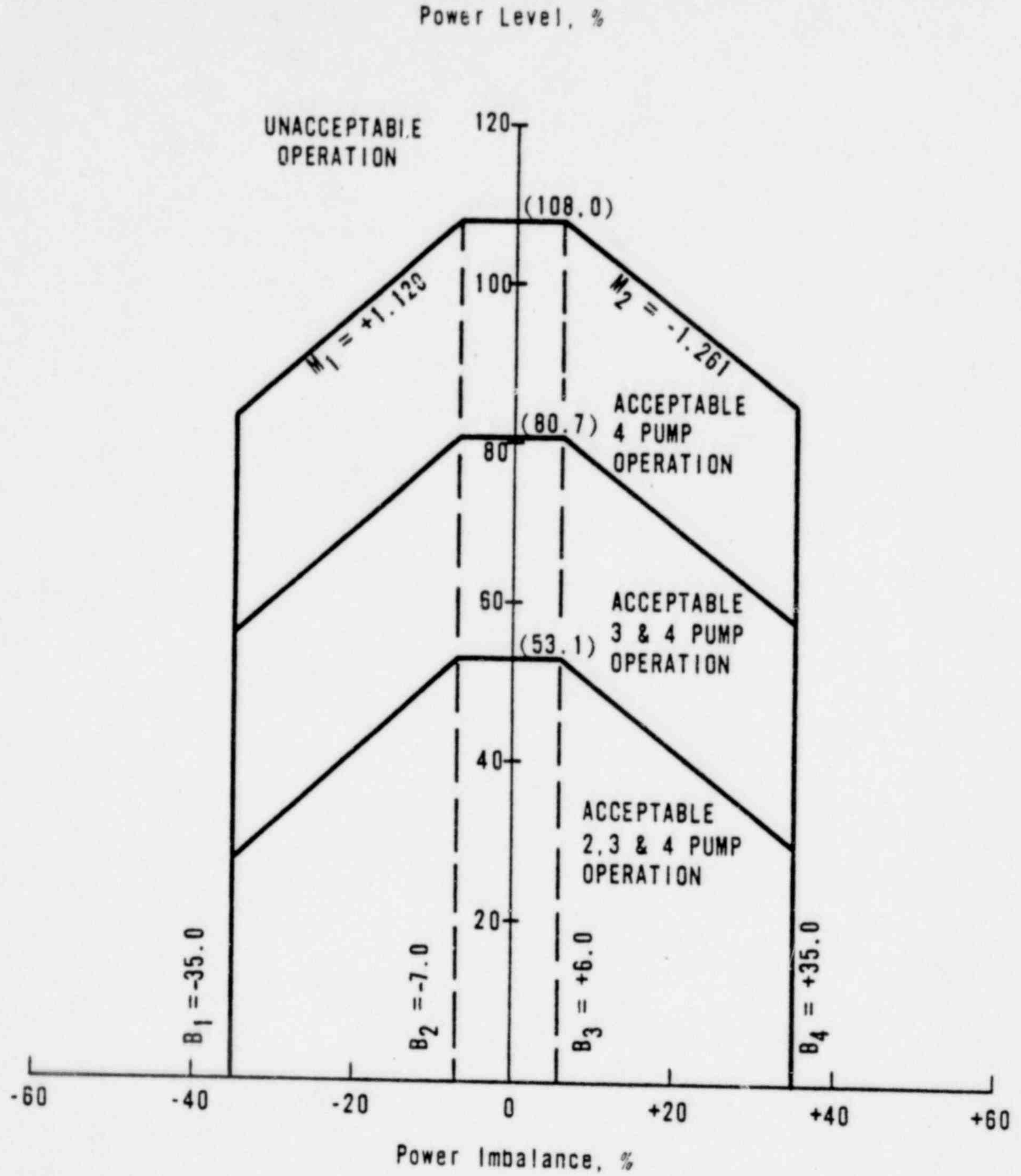
1487 214



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1

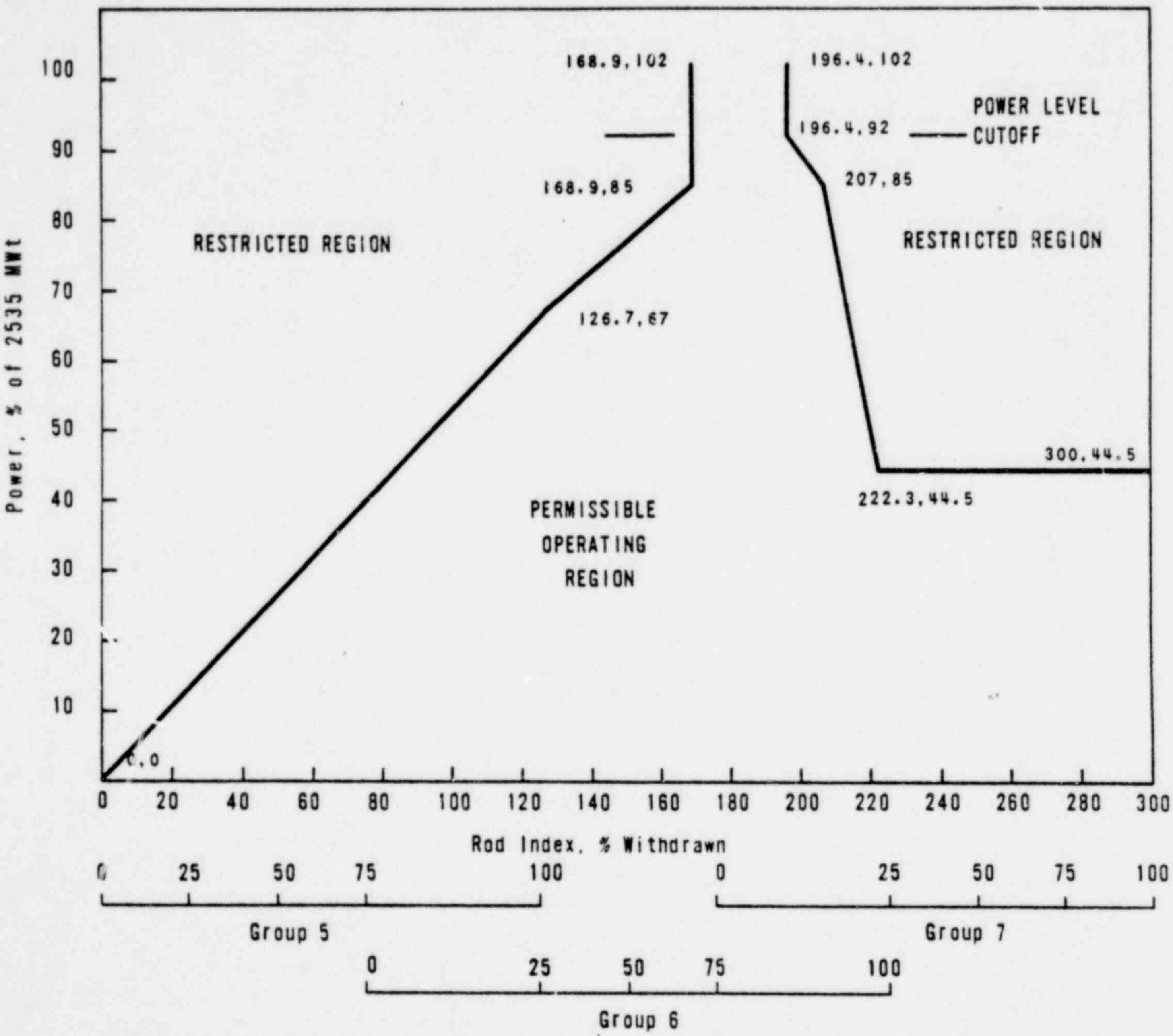
1487 215



PROTECTION SYSTEM MAXIMUM ALLOWABLE SET POINTS

Figure 2.3-2

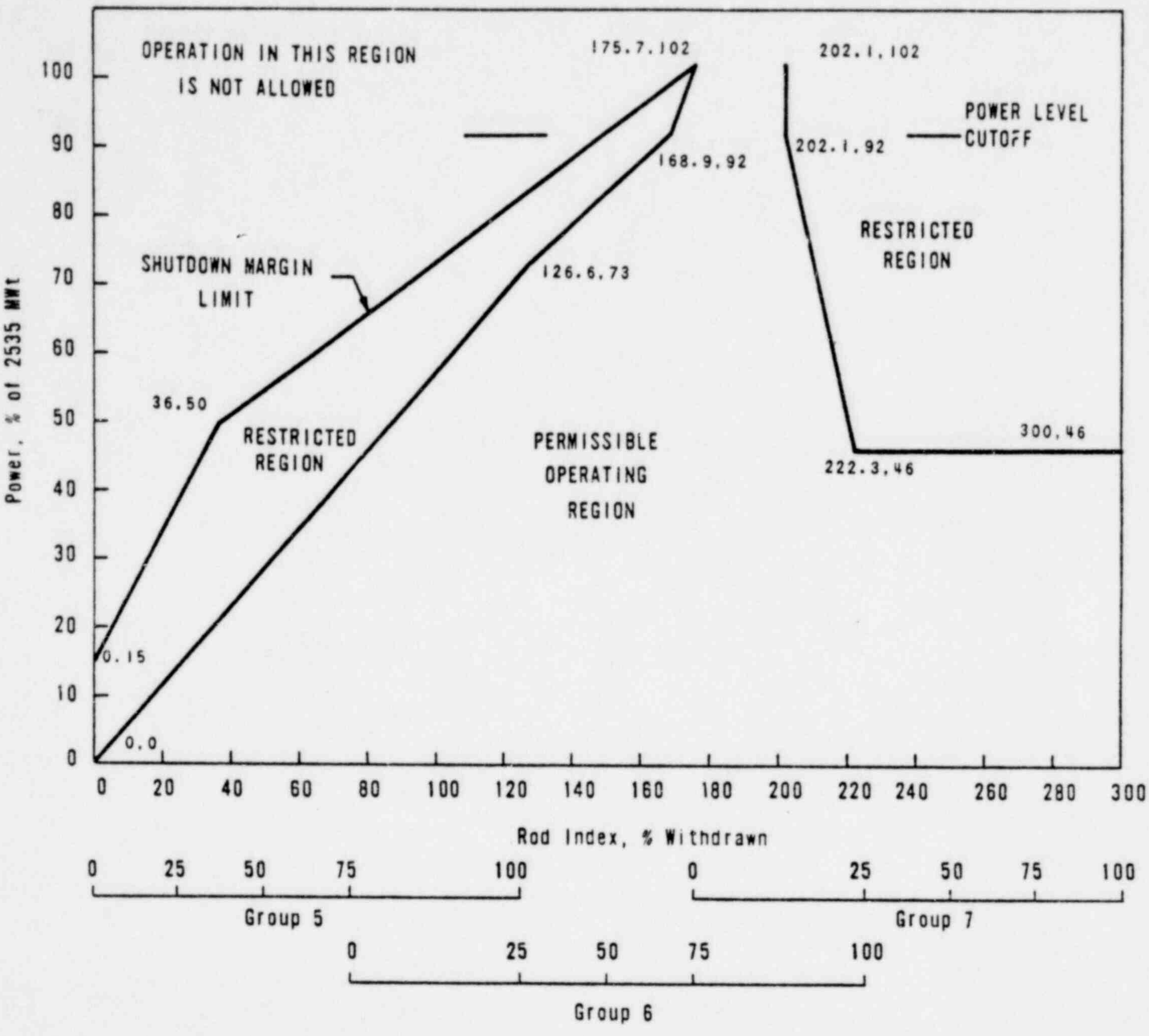
1487 216



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

Figure 3.5-2A

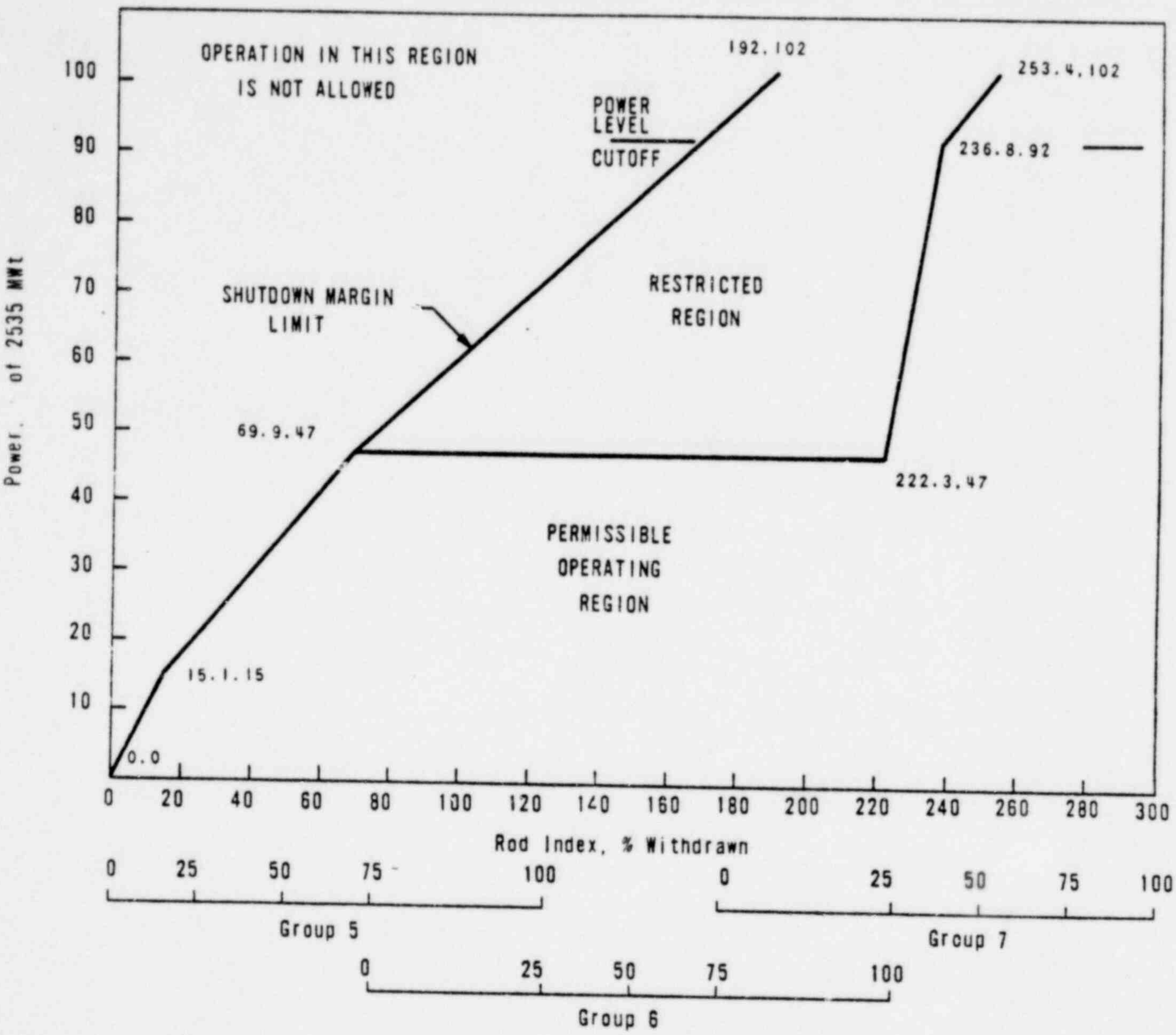
1487 217



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

Figure 3.5-2B

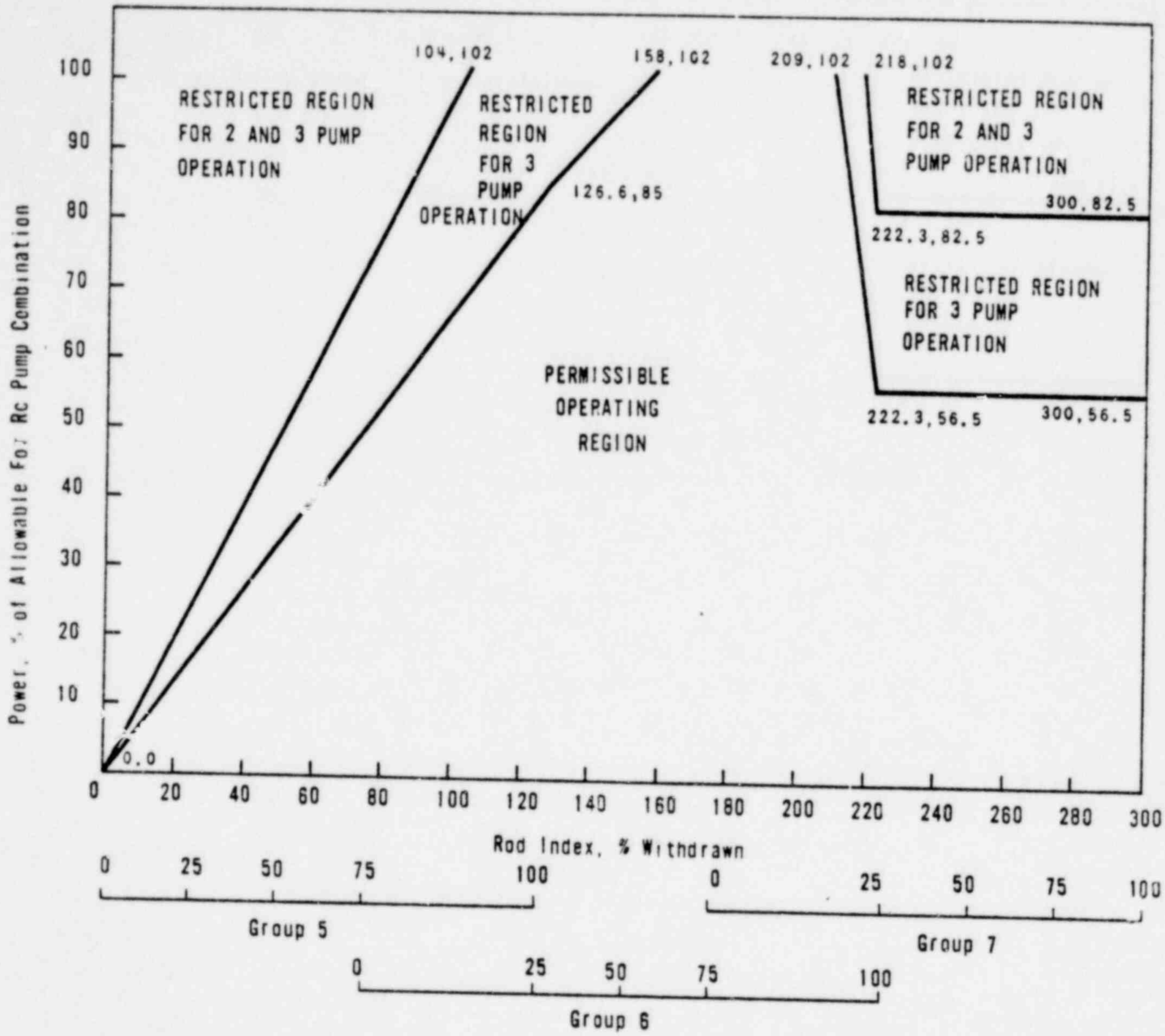
1487 218



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 275 ± 10 EFPD; CYCLE 2

Figure 3.5-2C

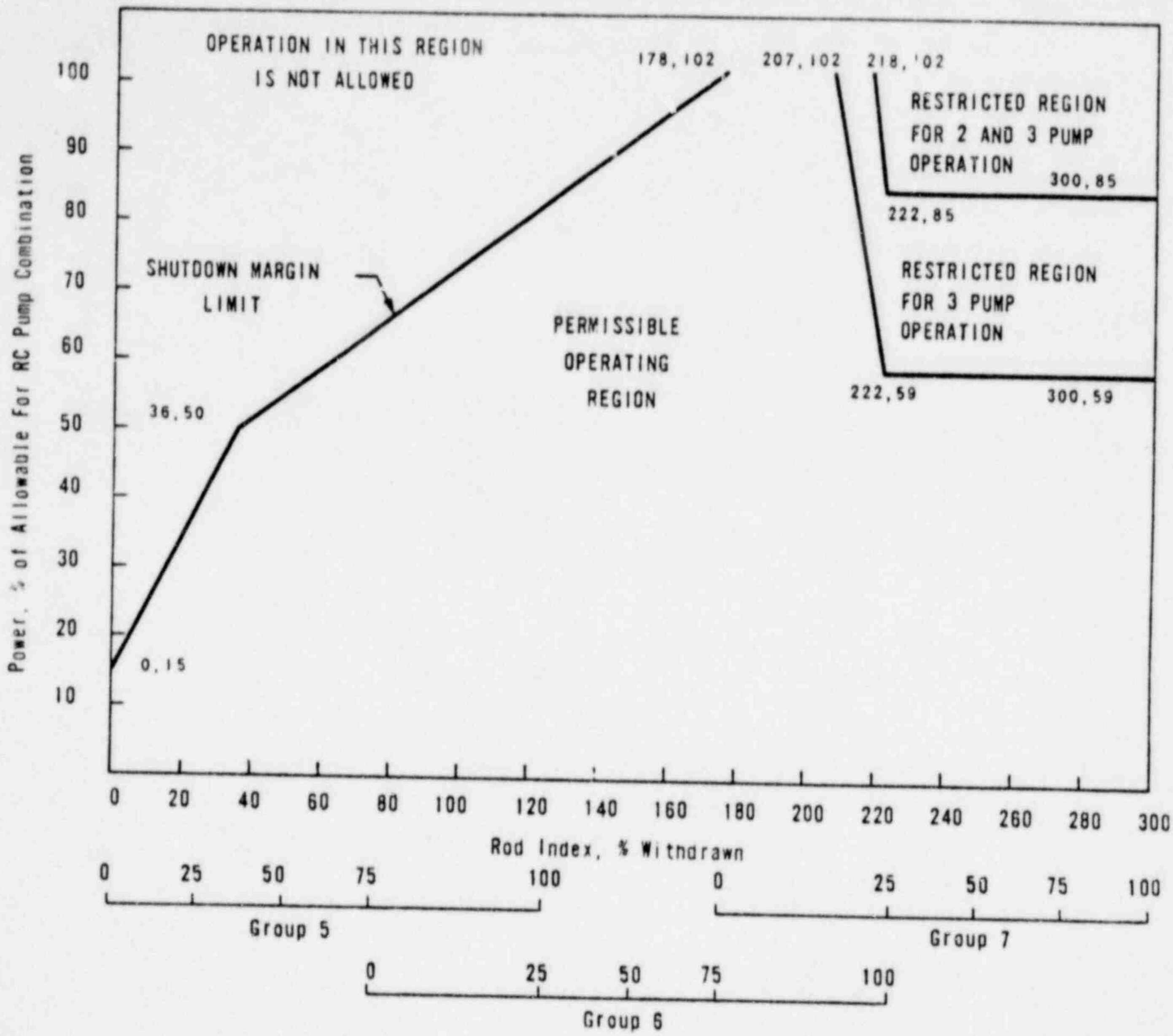
1487 219



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

Figure 3.5-2D

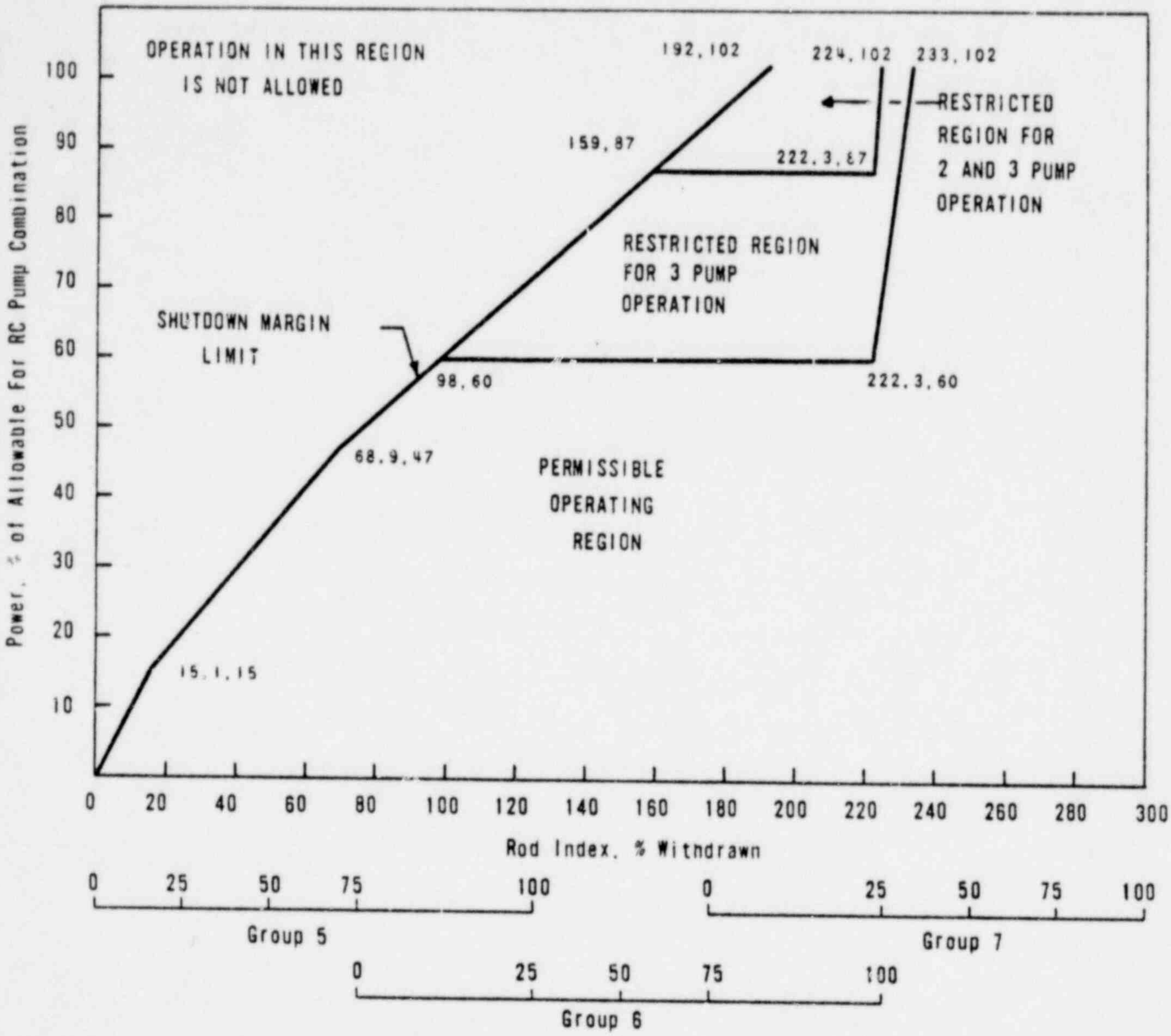
1487 220



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

Figure 3.5-2E

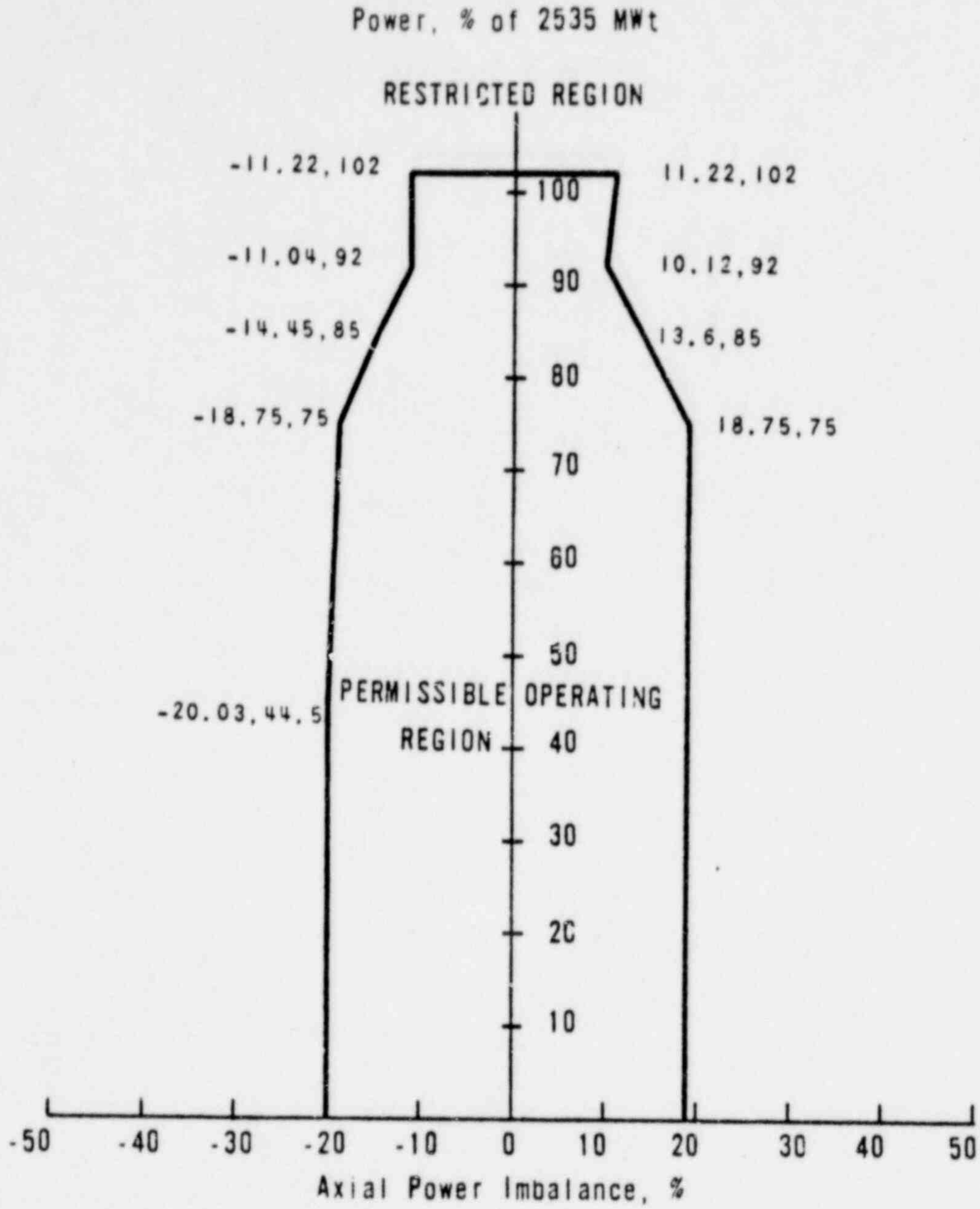
1487 221



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 275 ± 10 EFPD; CYCLE 2

Figure 3.5-2F

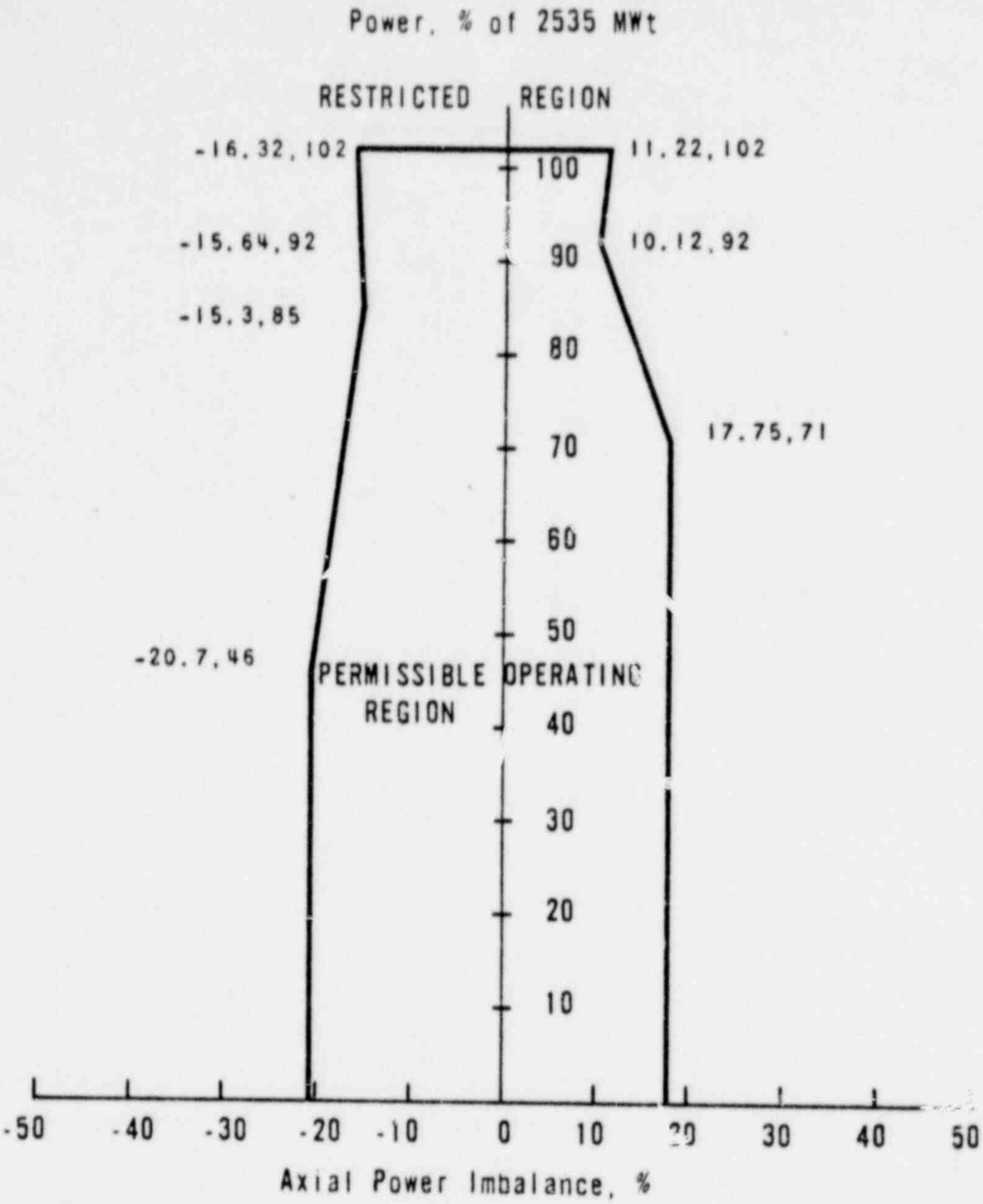
1487 222



OPERATIONAL POWER IMBALANCE ENVELOPE
APPLICABLE TO OPERATION FROM 0 TO 152
± 10 EFPD- CYCLE 2

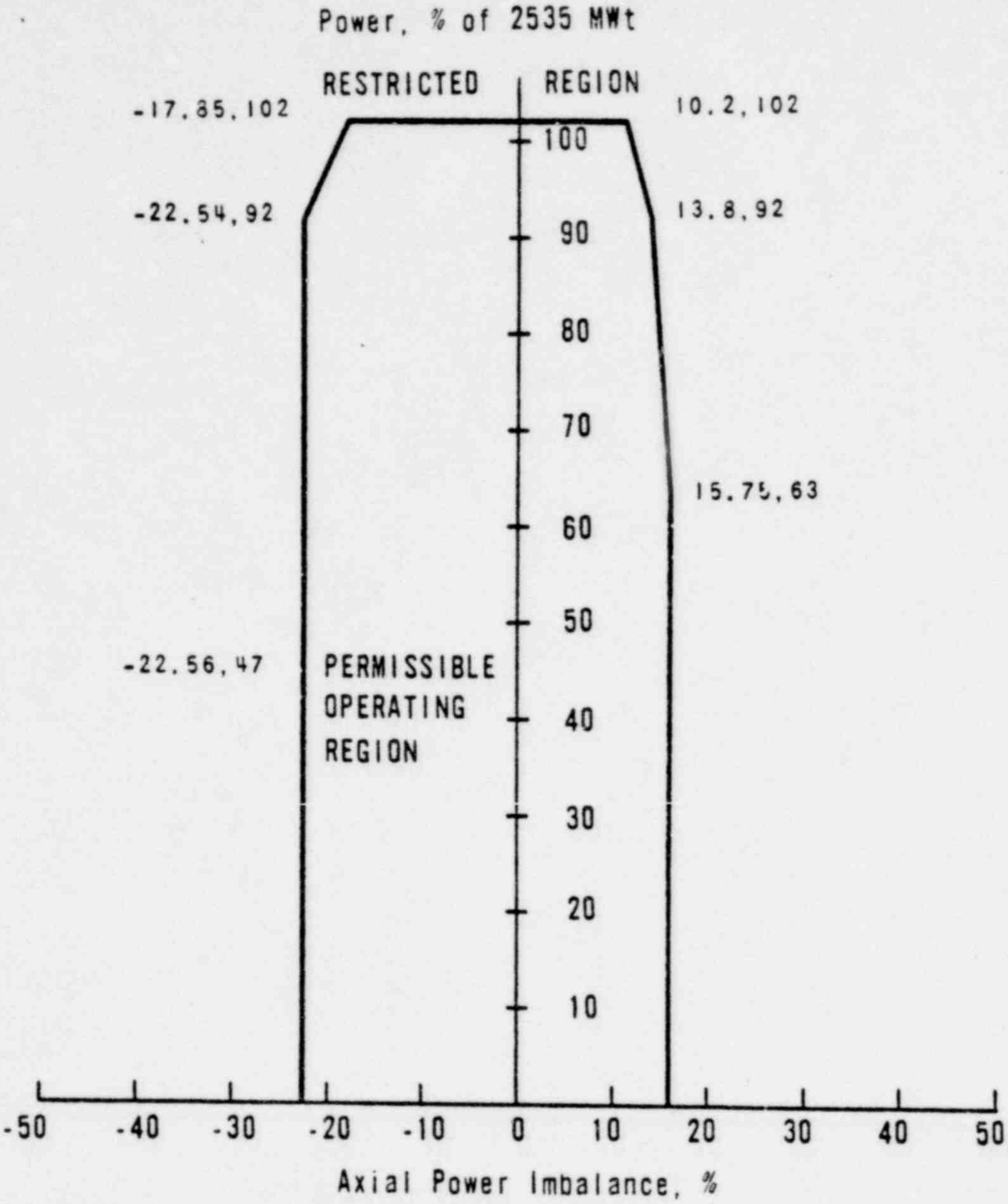
Figure 3.5-2G

1487 223



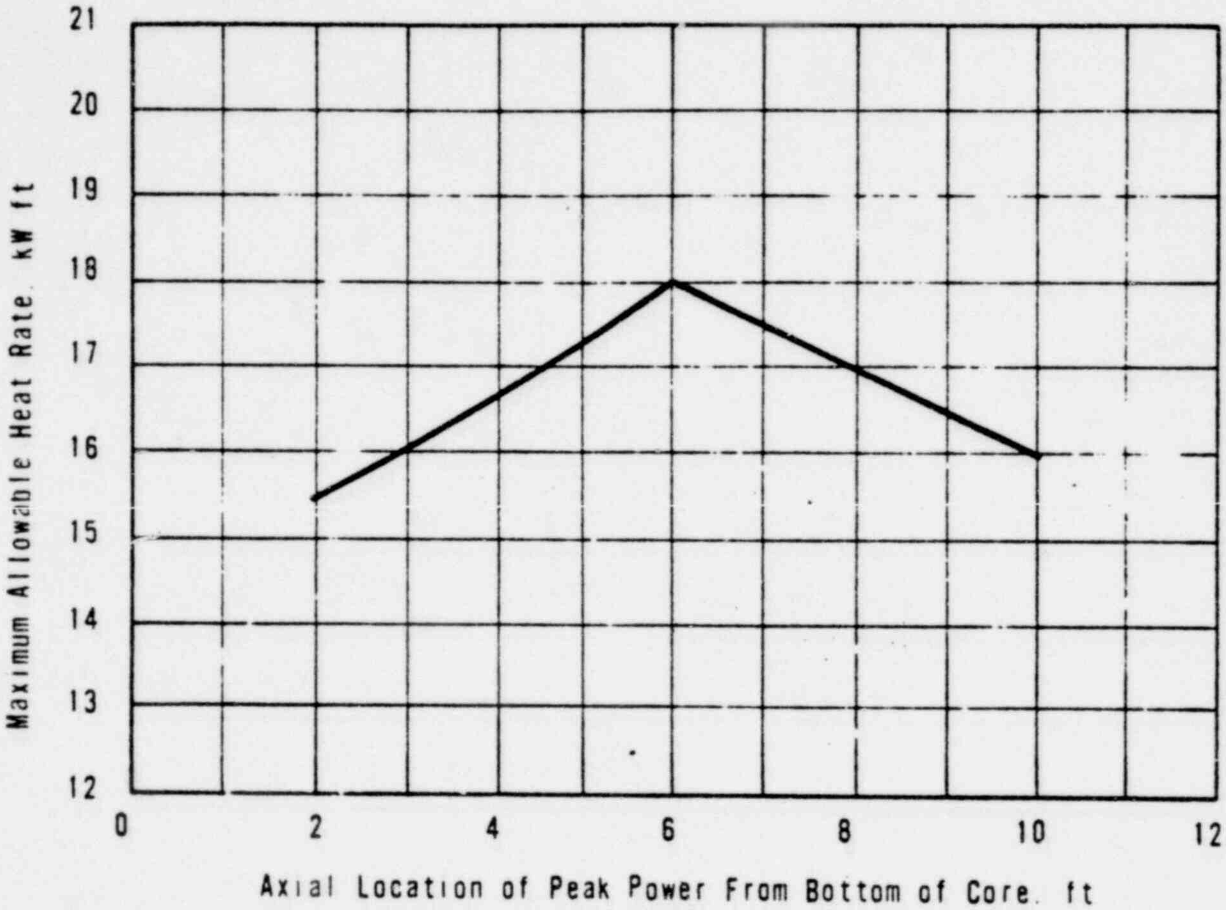
OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

Figure 3.5-2H



OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION AFTER 275 ± 10 EFPD; CYCLE 2

Figure 3.5-21



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
Figure 3.5-2J

1487 226