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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-50

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

   A. The applications for amendment by Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company (the licensees) dated August 8, 1975, as supported by filings dated July 9 and 15, 1975, and October 23, 1975; and January 13, 1976, as amended February 11, 1976, and April 2, 1976, and supported by filings dated January 23, 1976, April 5 and 8, 1976, 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, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. After weighing the environmental aspects involved, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: MAY 18 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise Appendix A as follows:

Remove Pages	Insert Pages
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2-2	2-2
2-3	2-3
2-5	2-5
2-5a	-
2-6	2-6
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2-8	2-8
2-9	2-9
3-1	3-1
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Remove Figures	Insert Figures
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.3-1	2.3-1
2.3-2	2.3-2
3.5-2A - 2F	3.5-2A - 2J

Changes on the revised pages are indicated by marginal lines. Pages 2-8, 3-15, and 3-33 are unchanged and are included for convenience only.

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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 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 x 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 and fuel rod bowing effects;

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

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 and fuel rod bowing;

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.

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

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.

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.

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 - 5143).

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.

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. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0 percent of rated power during reactor shutdown.

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

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation(5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

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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.

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24 hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

a. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2435 PSIG ± 1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24 hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less..

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature.[2, 3]

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.[4] Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.[5] The pressurizer code safety valve lift set point shall be set at 2435 psig ±1 percent allowance for error and each valve shall be capable of relieving 311,700 lb/h of saturated steam at a pressure not greater than three percent above the set pressure.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7

(2) FSAR, Sections 4.2.5.1 and 9.5.2.3

(3) FSAR, Section 4.2.5.4

(4) FSAR, Sections 4.3.10.4 and 4.2.4

(5) FSAR, Section 4.3.7

If reactor coolant leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rate of leakage to which the instrument is sensitive is 0.054 gpm within sixty minutes, assuming the presence of corrosion product activity.

b. The containment radioactive gas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to greater than 10 gpm.

c. A leakage detection system which determines leakage losses from water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

d. Indication of leakage from the above sources shall be cause to require a containment entry and limited inspection at power of the reactor coolant system. Visual inspection means, i.e., looking for steam, floor wetness, or boric acid crystalline formations, will be used. Periodic inspections for indications of leakage within the containment will be conducted to enhance early detection of problems and to assure best on-line reliability.

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. Water inventories around the auxiliary building sump.

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 $\pm$ 2 gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Three Mile Island Operations Group according to routine established in Section 12.1.1 of the FSAR. Under these conditions, an allowable leakage rate of 30 gpm has been established.

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 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 $+0.5 \times 10^{-4}$ ΔK/K/F. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ ΔK/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 exceed 2200 F.

REFERENCES

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

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall not be less than one percent $\Delta K/k$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 and 4.7.2.3 in the safety or regulating rod banks shall not be permitted.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent $\Delta k/k$ hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a one percent $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

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.

3.5.2.4 Quadrant tilt:

a. Except for physics tests if quadrant tilt exceeds 4 percent, power shall be reduced immediately 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.

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.

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)

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.

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:

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% $\Delta k/k$ 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% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% $\Delta k/k$ 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% $\Delta k/k$ ejected rod worth at rated power.

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.

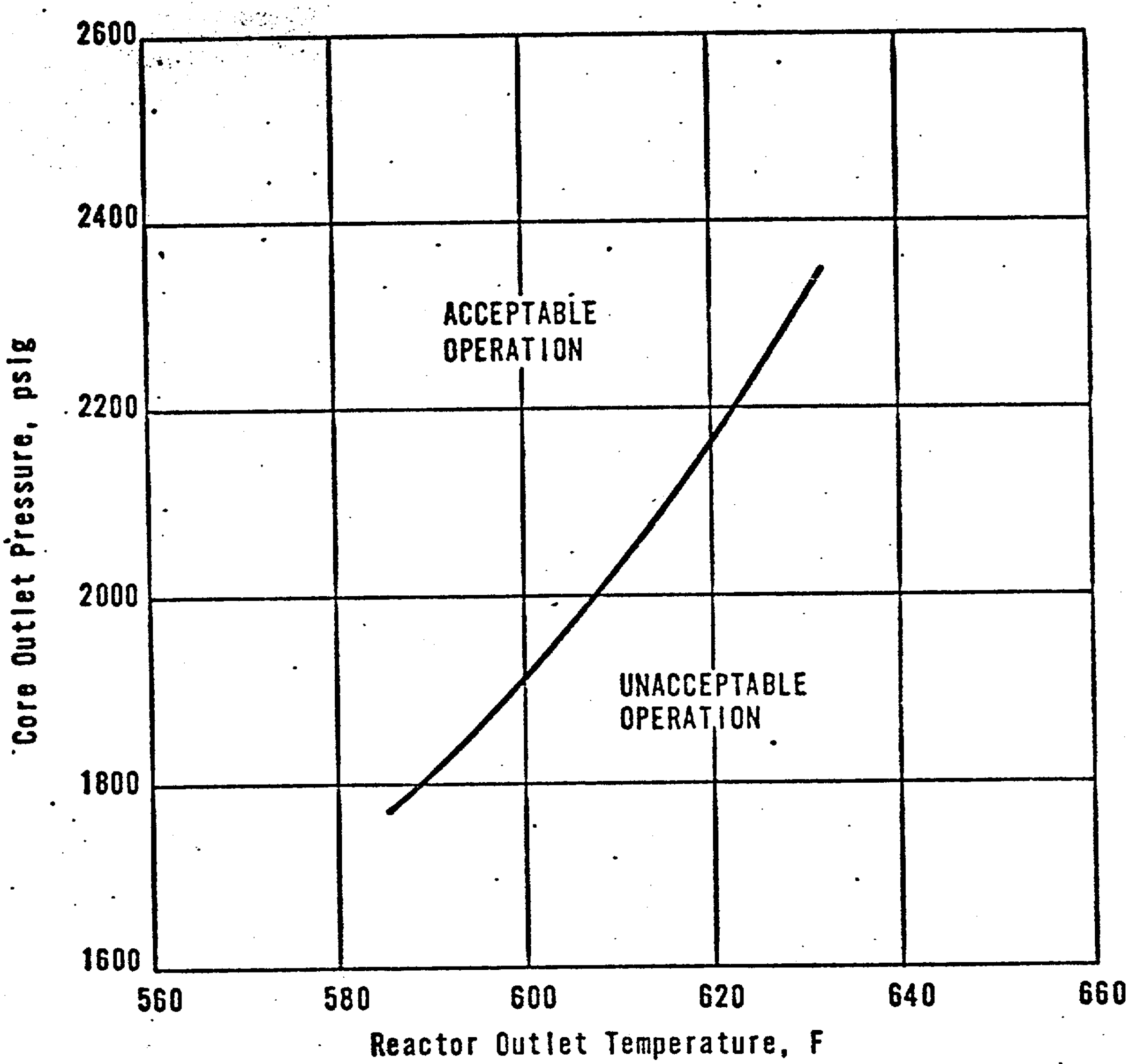
During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

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

REFERENCES

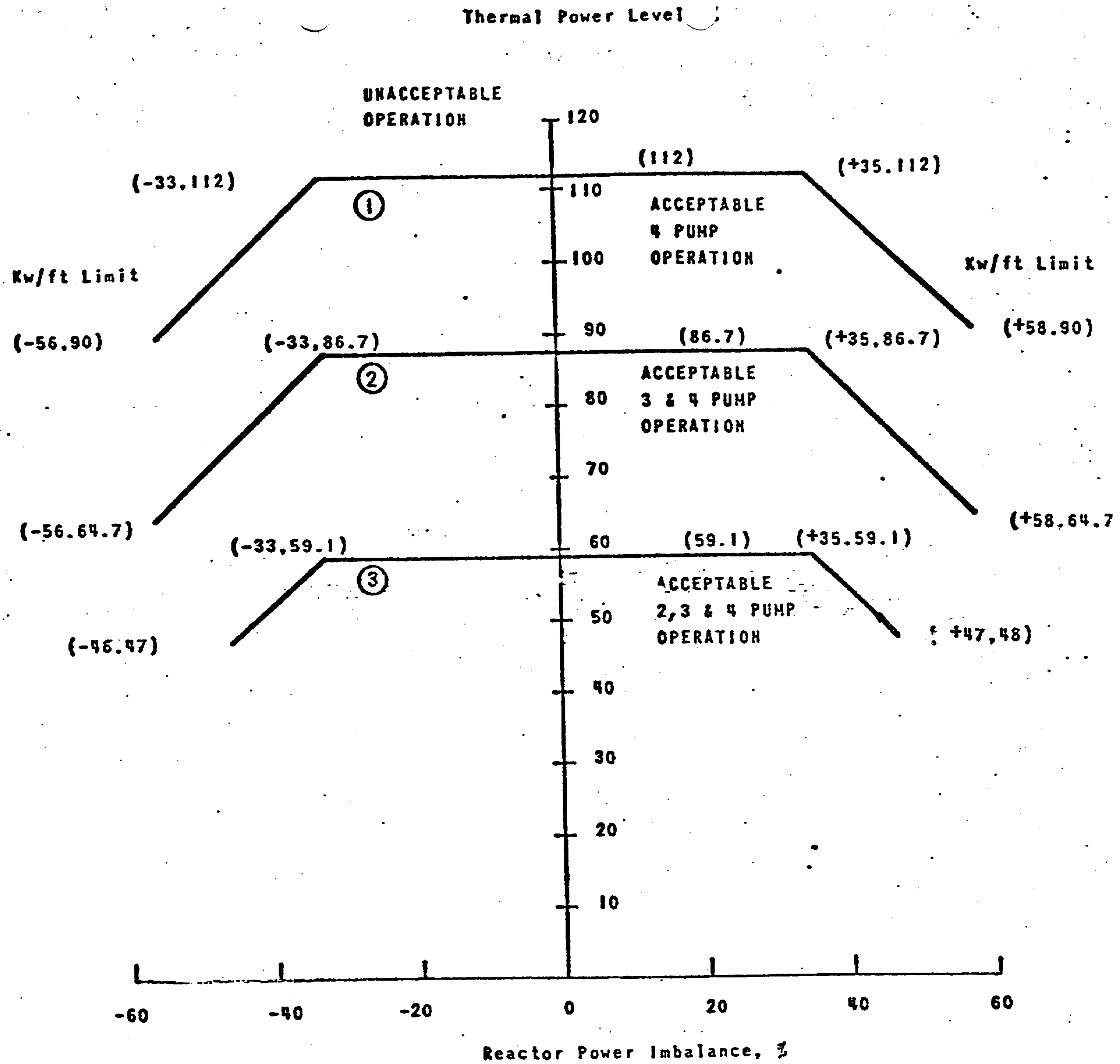
(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2



CORE PROTECTION SAFETY LIMIT

Figure 2.1-1

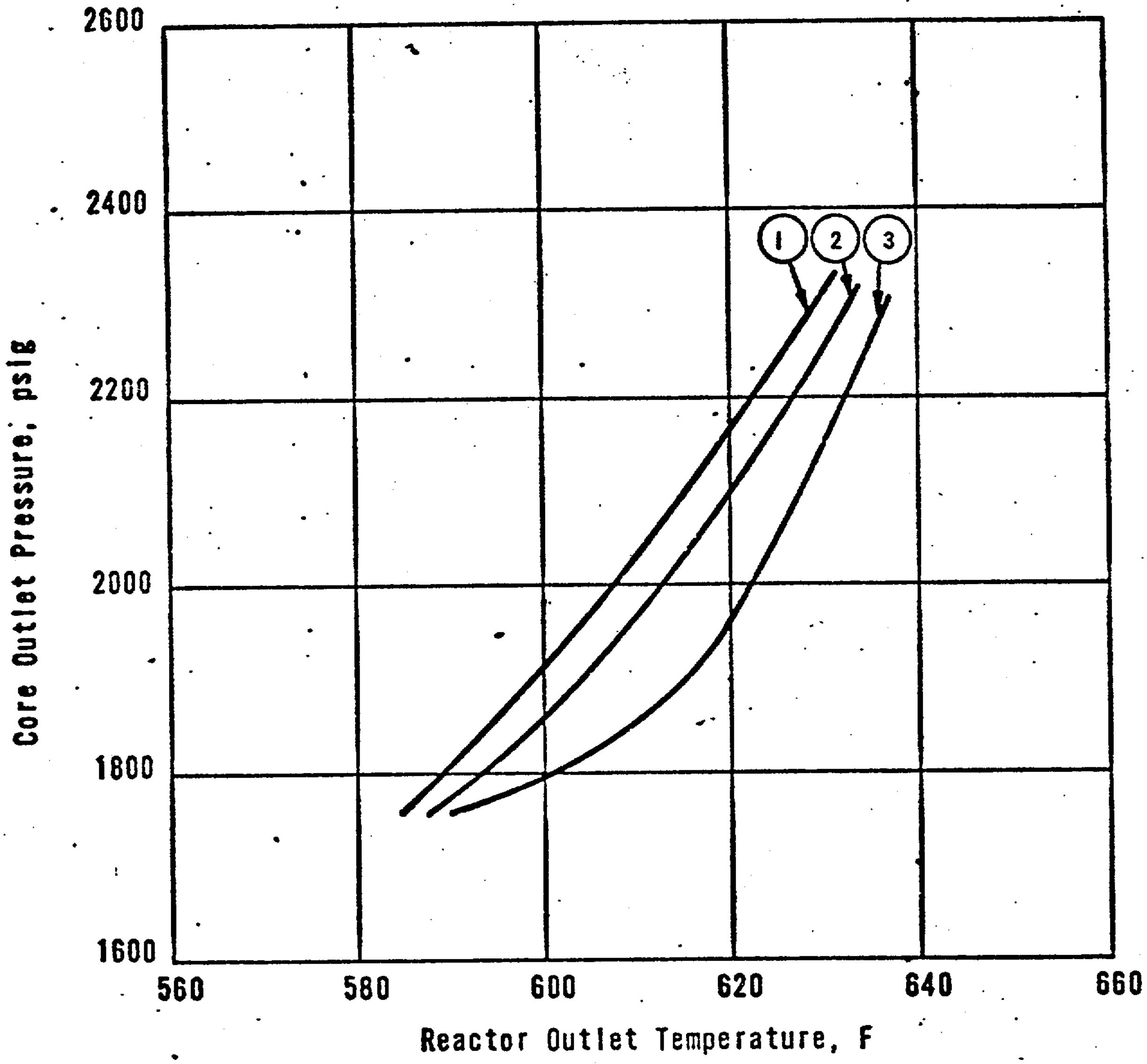


CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

UNIT 1, CYCLE 2

CORE PROTECTION SAFETY LIMITS

Figure 2.1-2

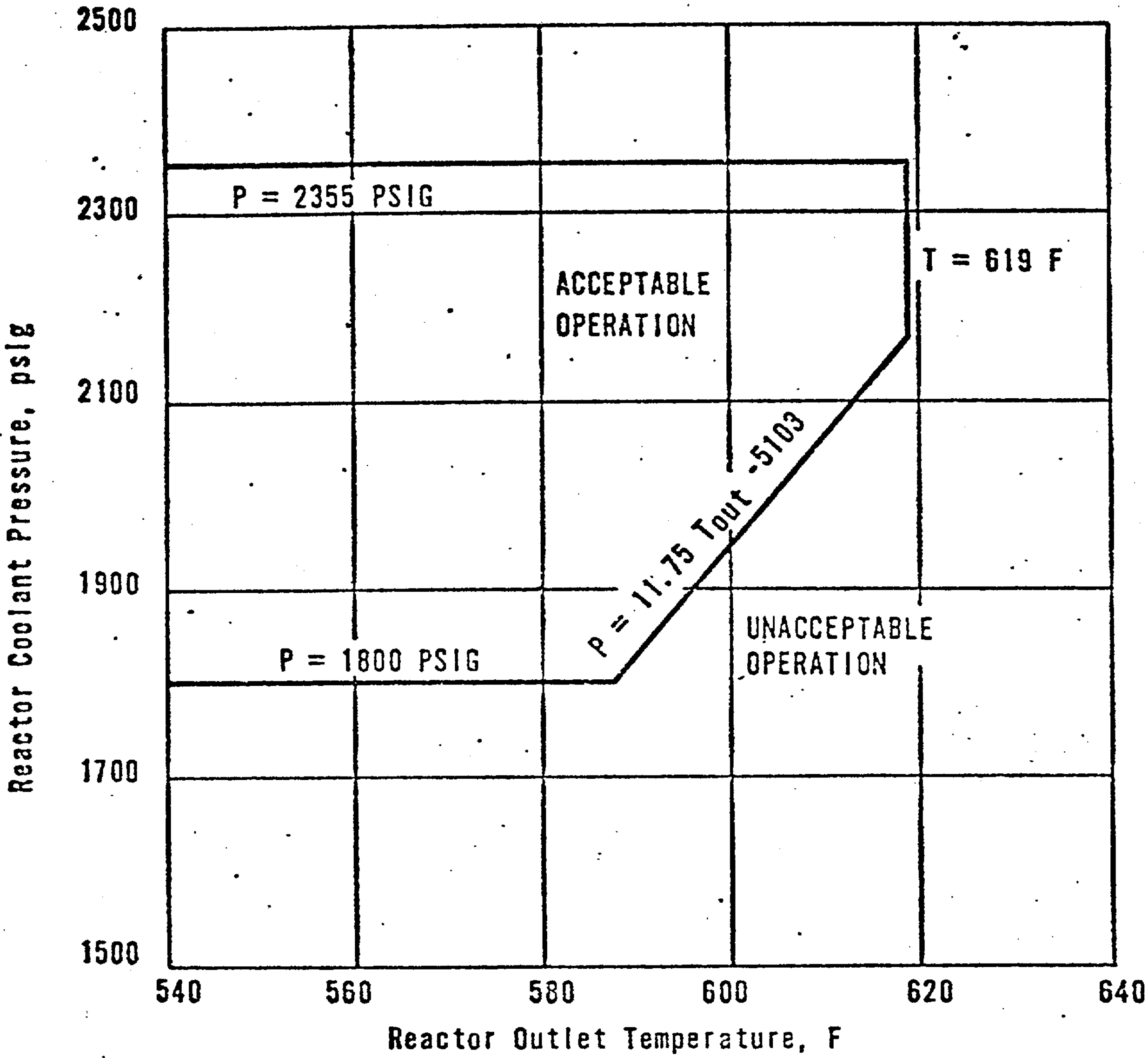


CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	86.7%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.1%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

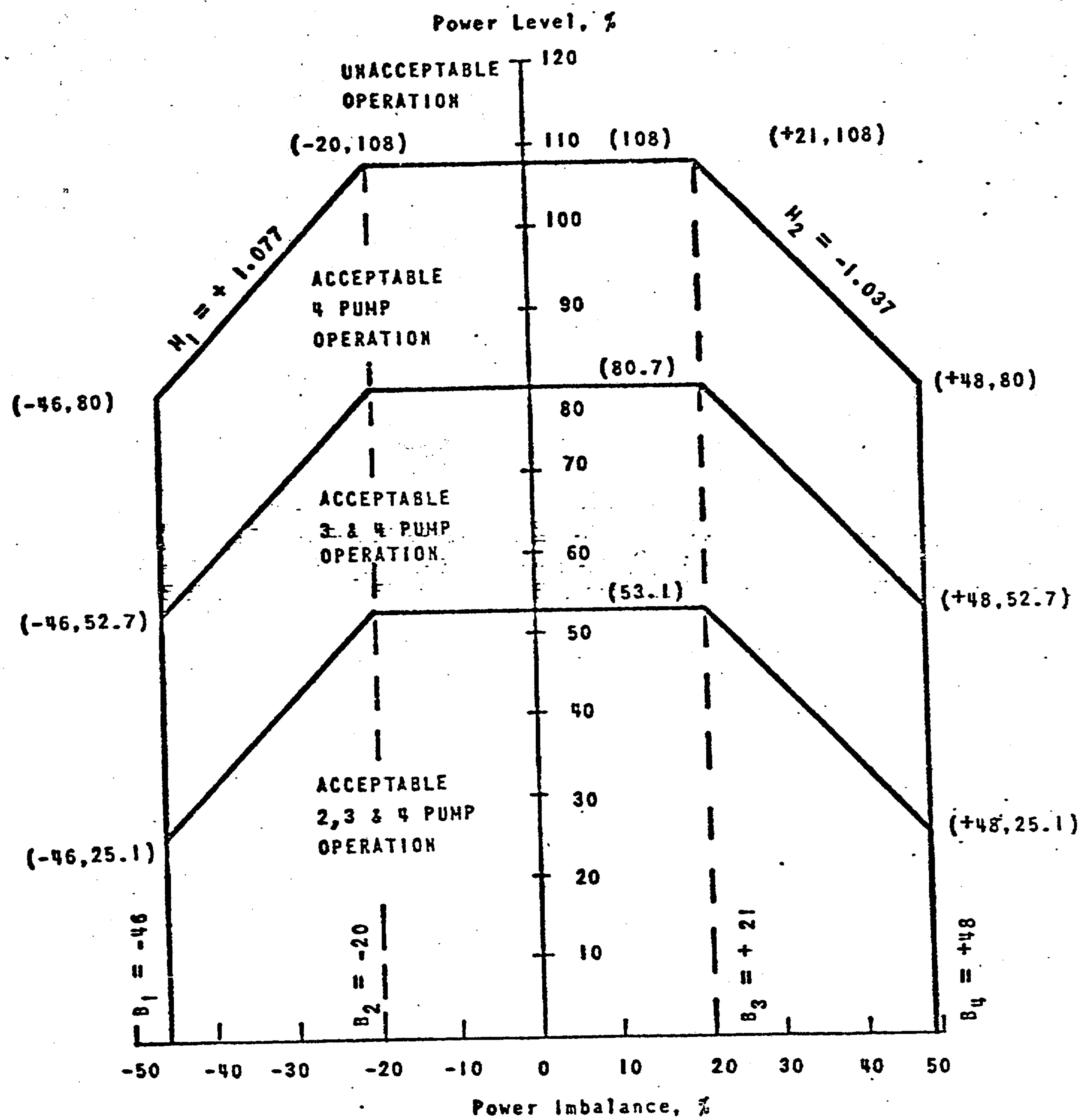
CORE PROTECTION SAFETY

BASES Figure 2.1-3



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1



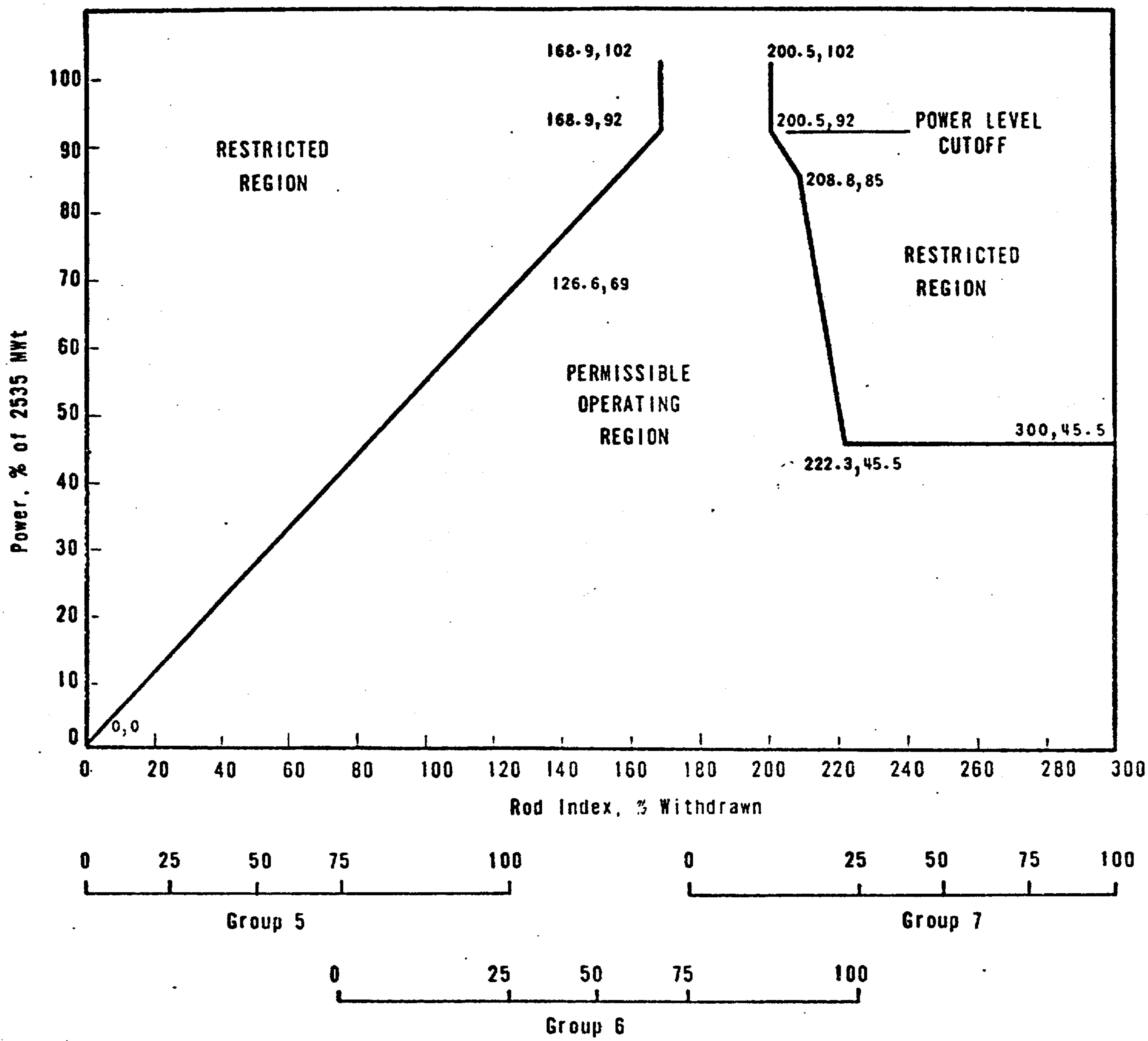
UNIT 1, CYCLE 2

PROTECTION SYSTEM MAXIMUM

ALLOWABLE SET POINTS

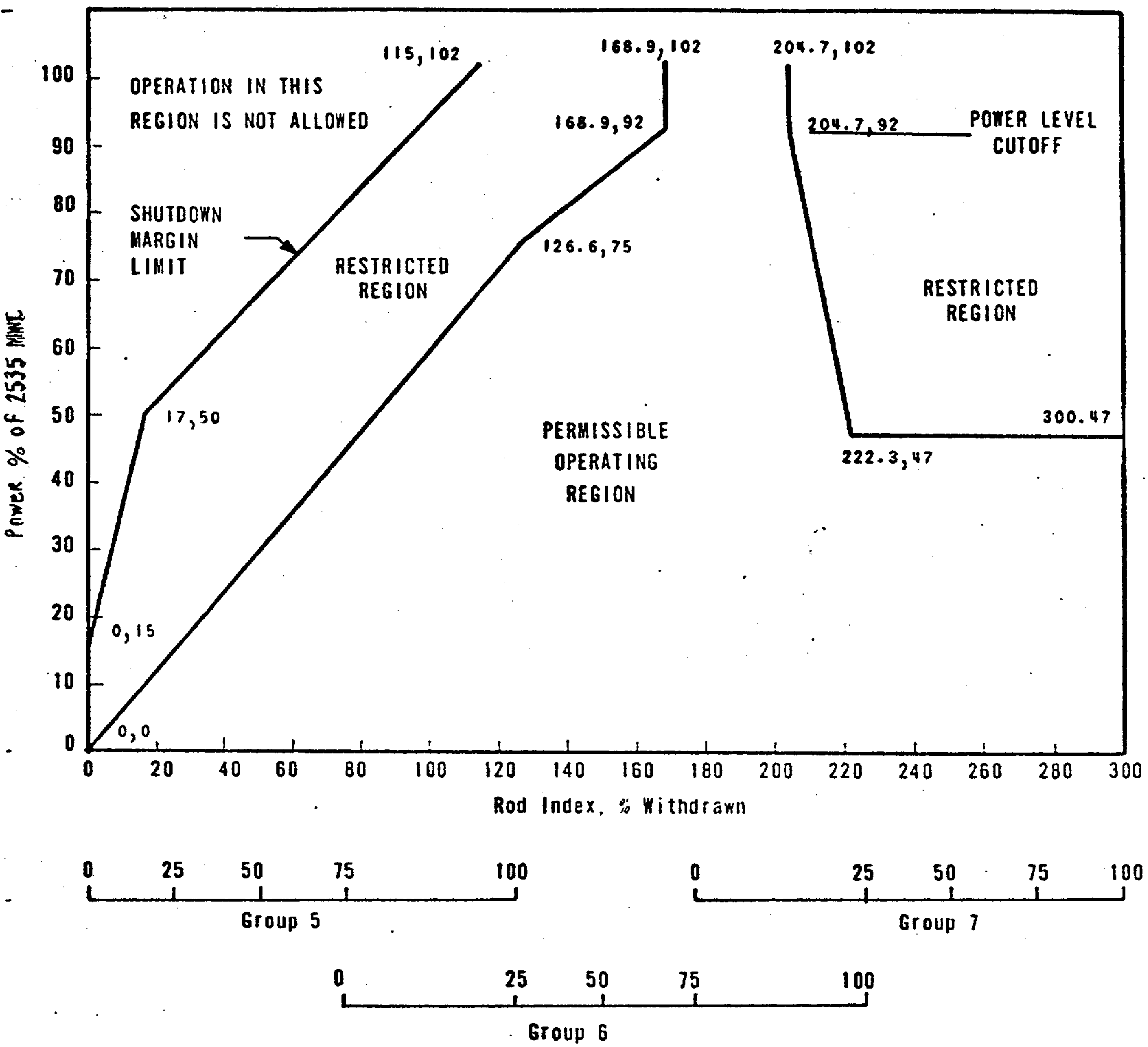
Amendment No. 17

Figure 2.3-2



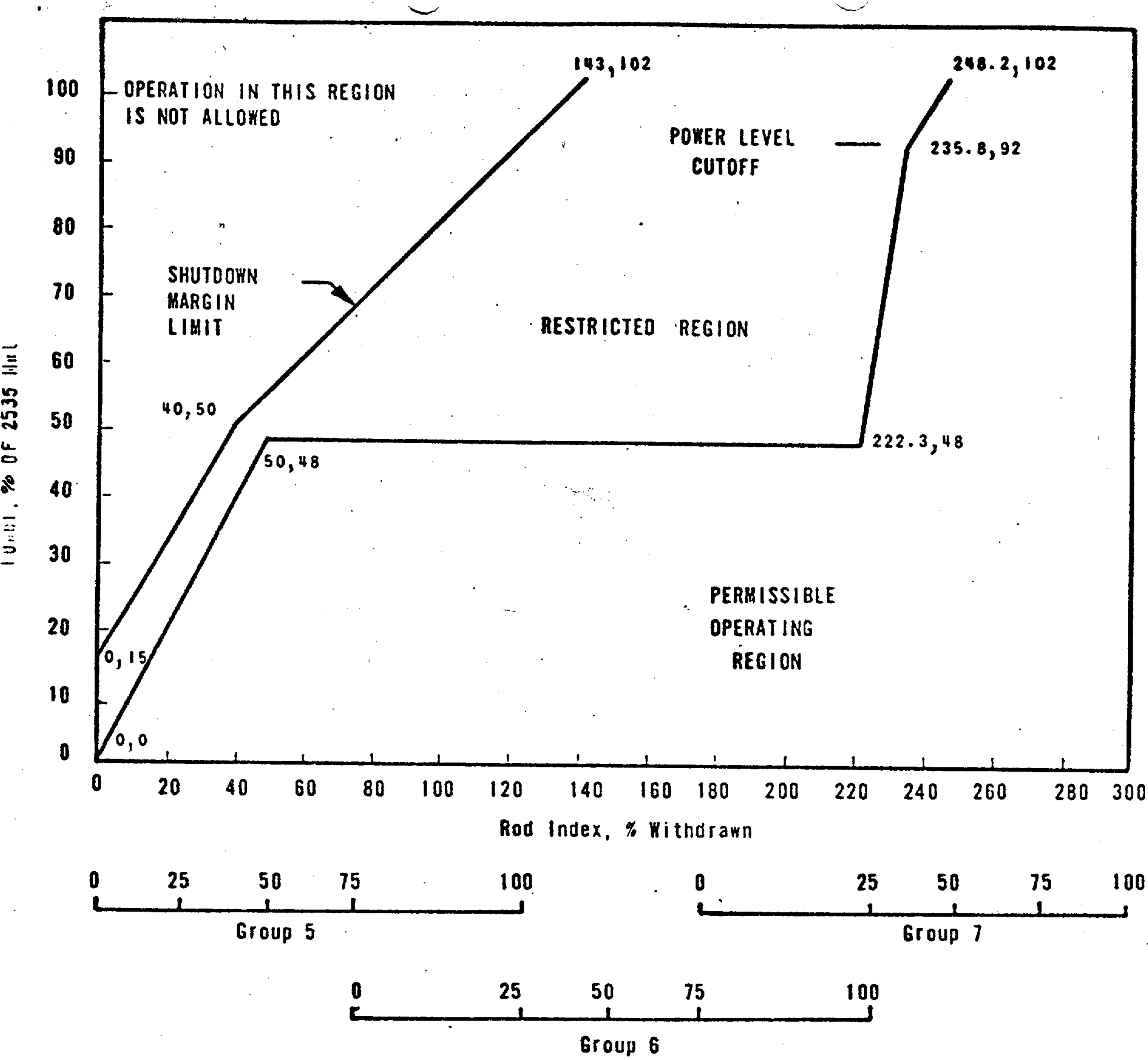
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

Figure 3.5-2A



ROD POSITION LIMITS FOR 4 PUMP
OPERATION APPLICABLE DURING THE
PERIOD FROM 152 ± 10 EFPD:
TO 265 ± 10 EFPD; CYCLE 2

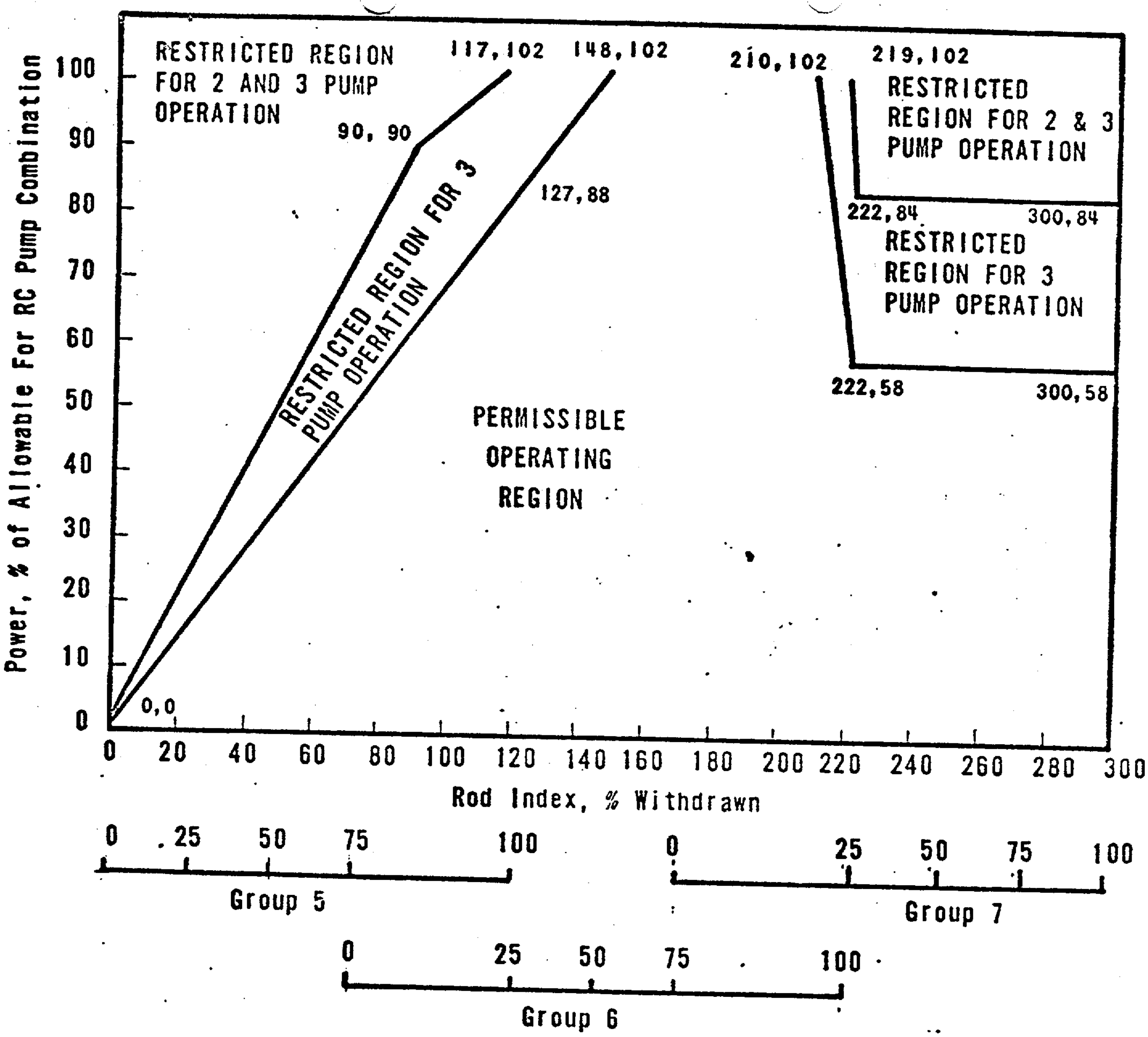
Figure 3.5-2B



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 265 ± 10 EFPD; CYCLE 2

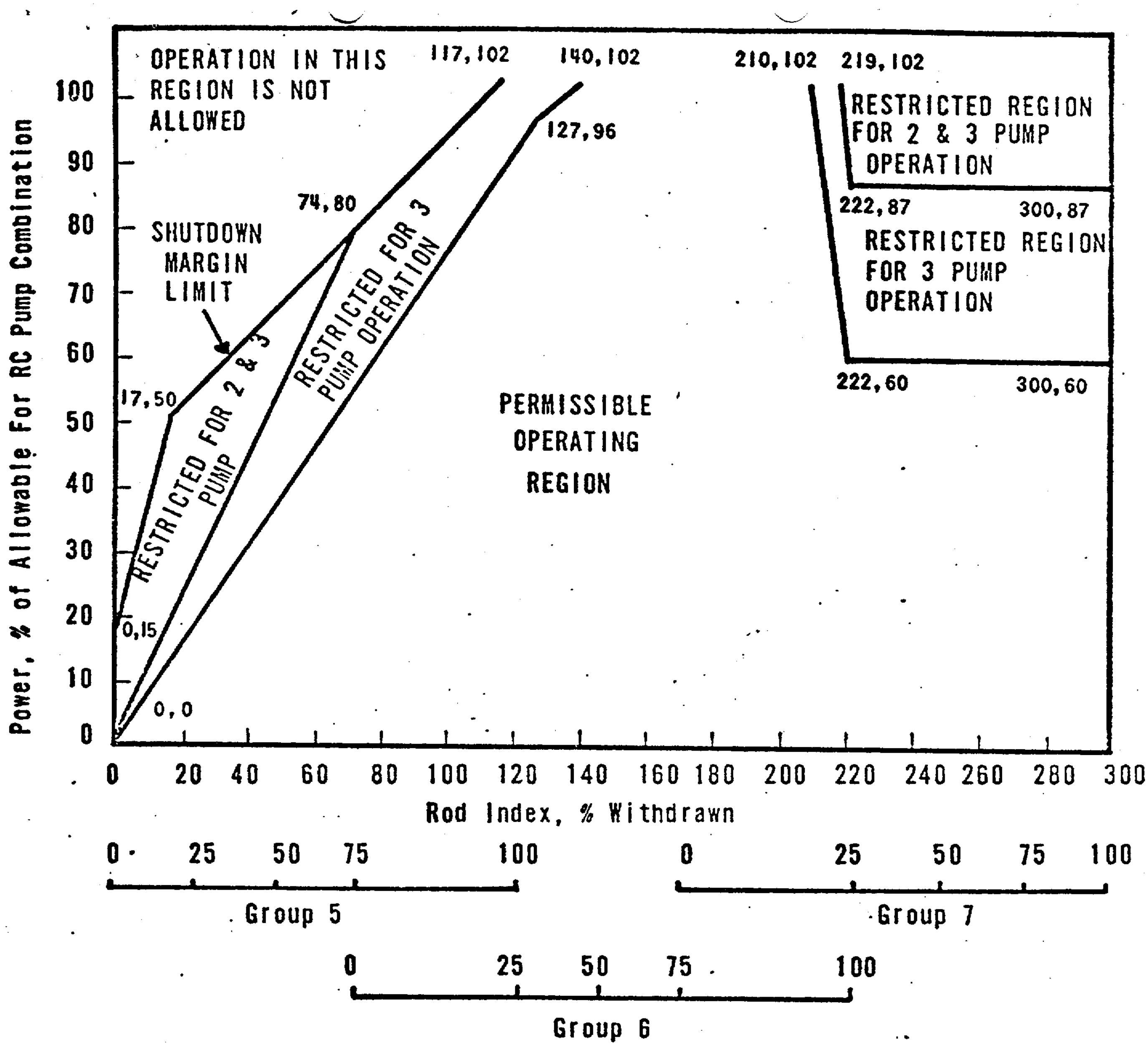
Figure 3.5-2C

Amendment No. ~~10~~, 17



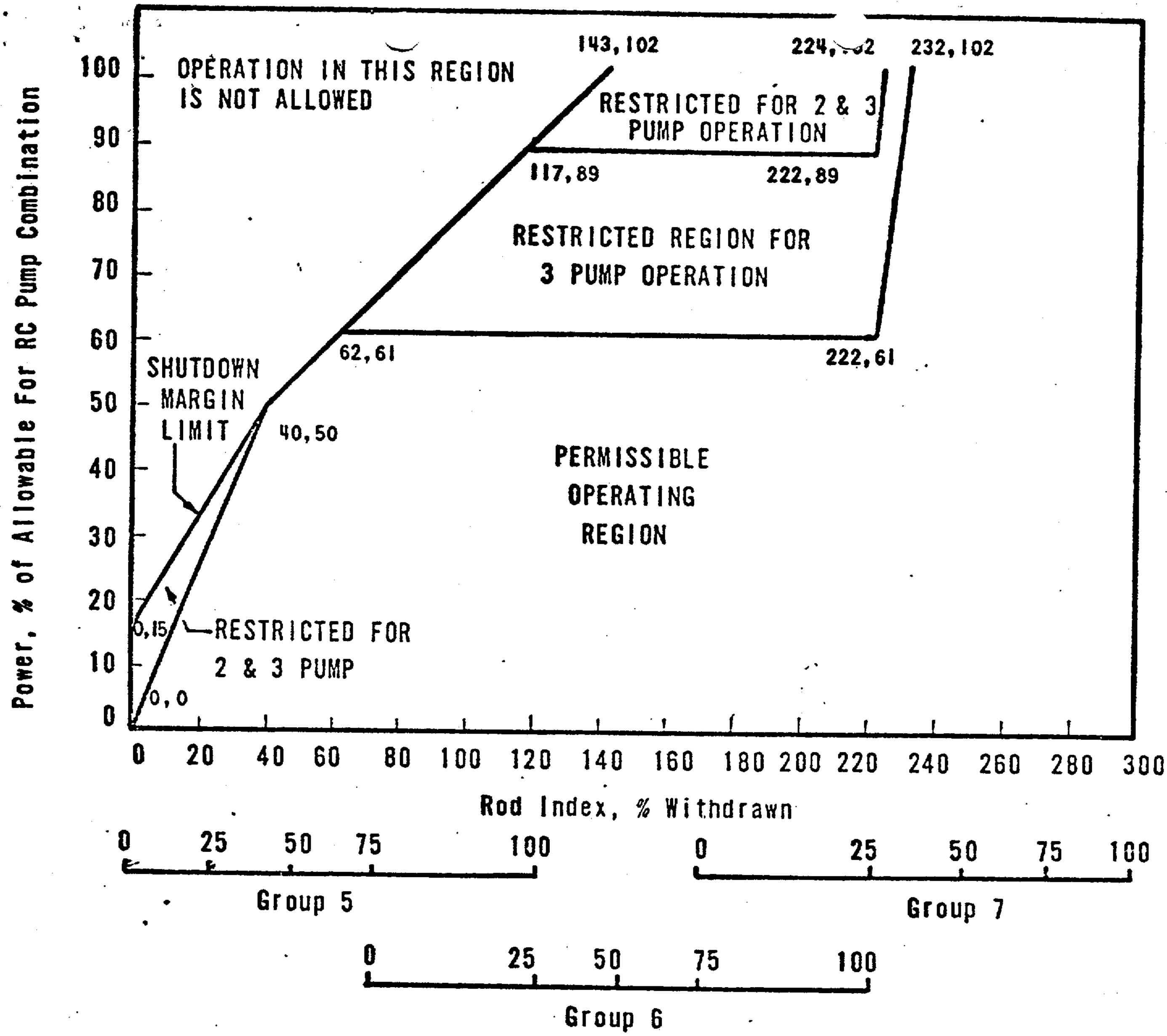
ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

Figure 3.5-2D



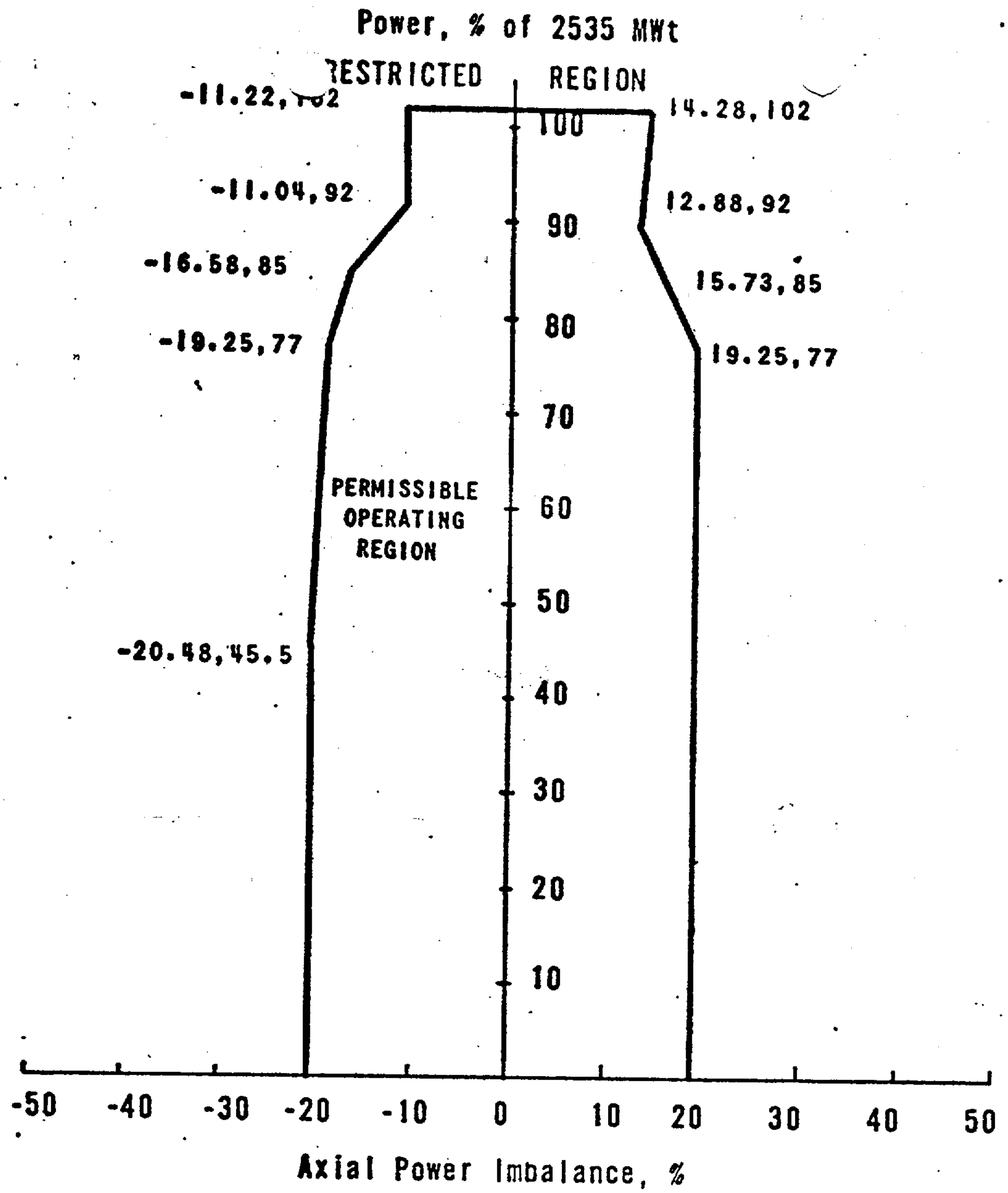
ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 152 ± 10 TO 265 ± 10 EFPD; CYCLE 2

Figure 3.5-2E



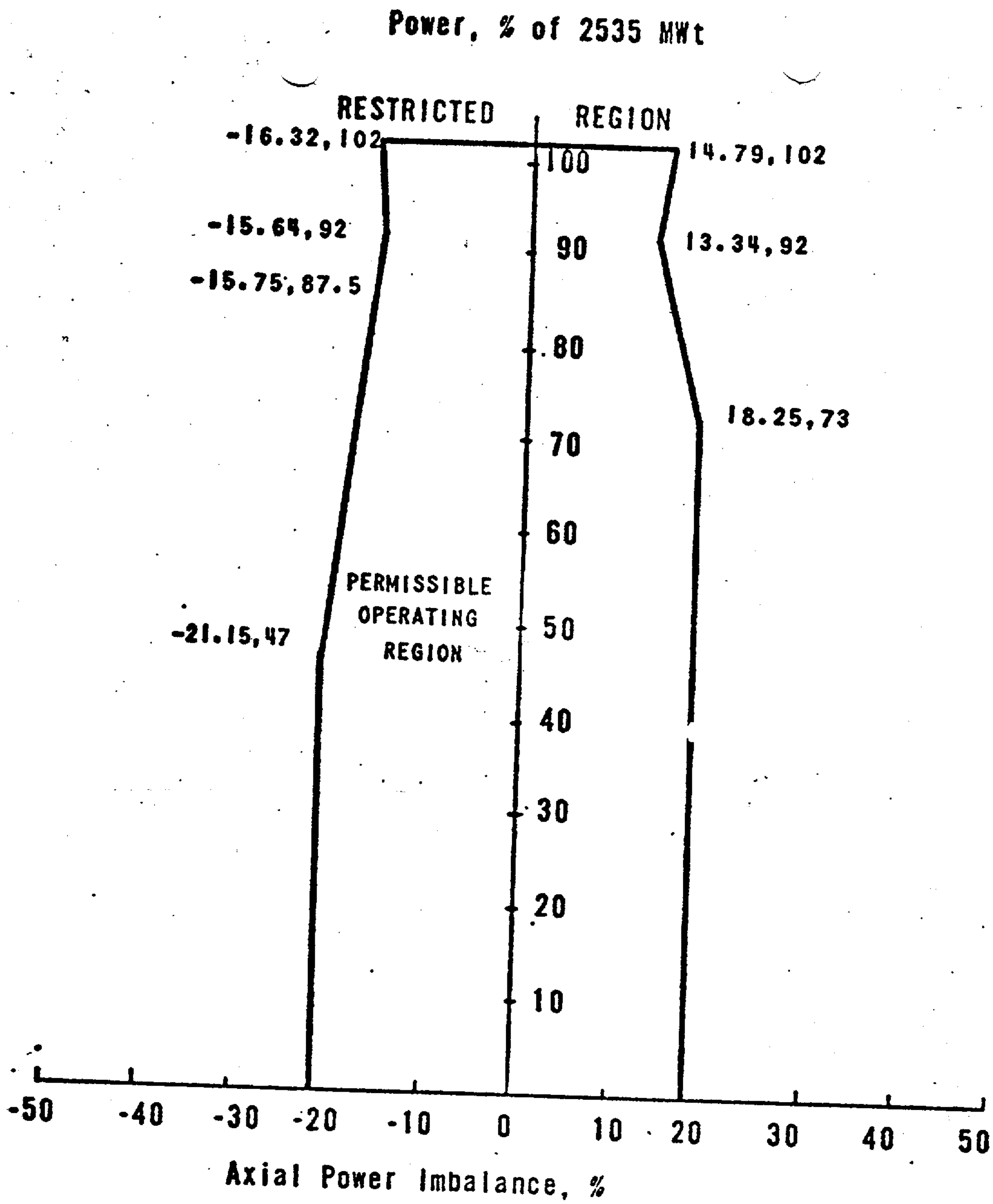
ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 265 ± 10 EFPD; CYCLE 2

Figure 3.5-2F



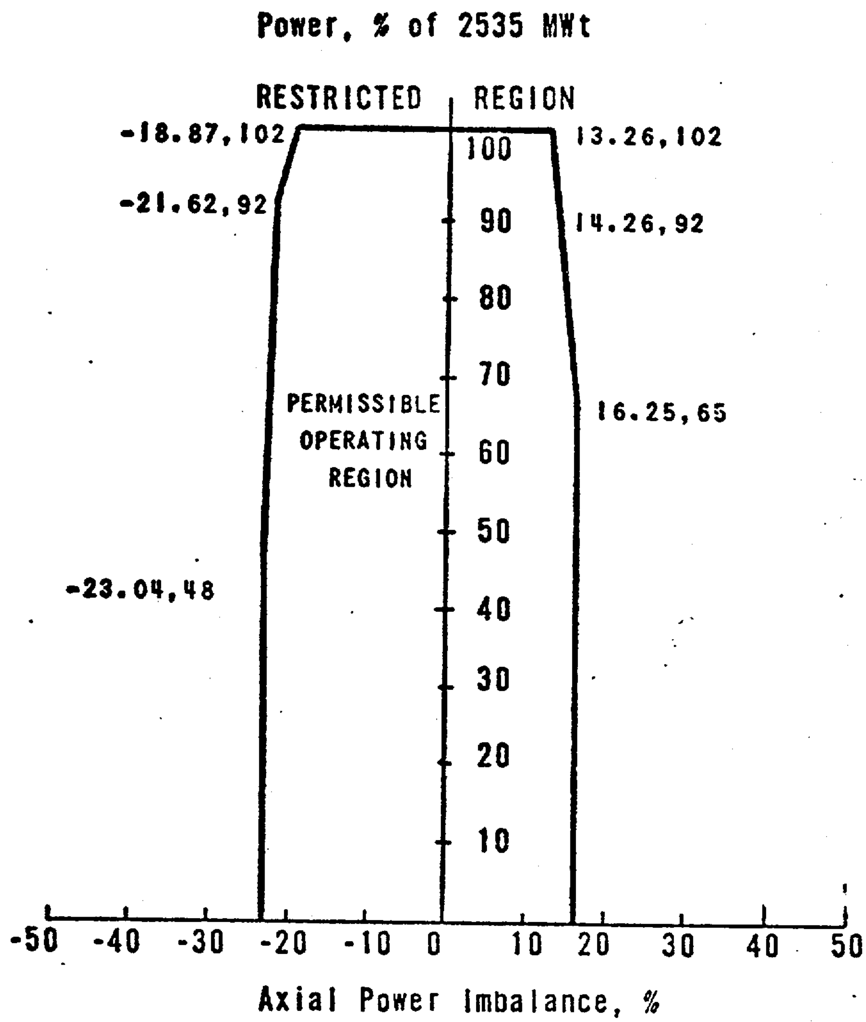
OPERATIONAL POWER IMBALANCE ENVELOPE
APPLICABLE TO OPERATION FROM 0 TO
152 ± 10 EFPD; CYCLE 2

Figure 3.5-2G



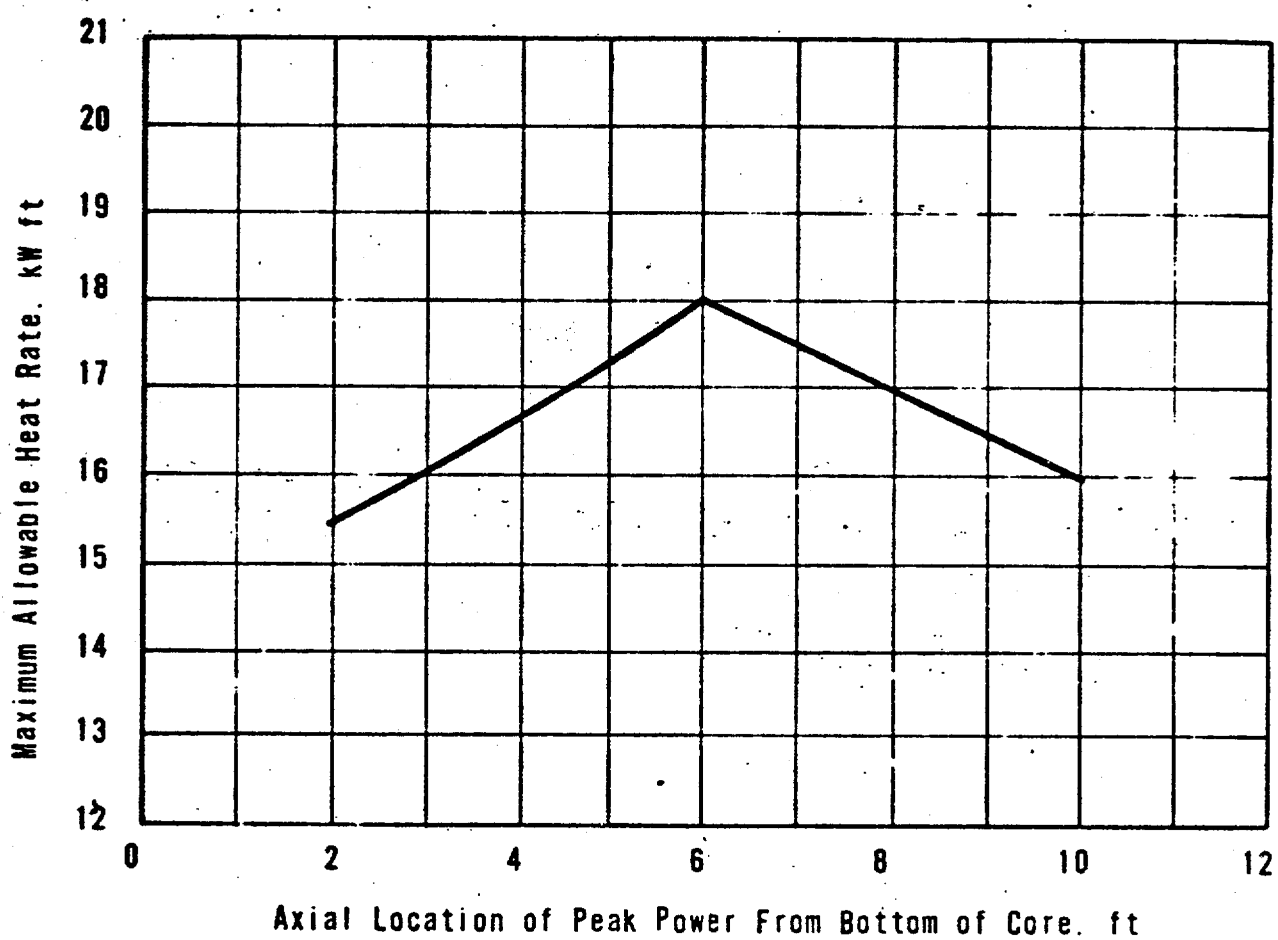
OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION FROM 152 ± 10 TO 265 ± 10 EFPD; CYCLE 2

Figure 3.5-2H



OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION AFTER 265 ± 10 EFPD; CYCLE 2

Figure 3.5-2I



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE

Figure 3.5-2J