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. 45

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 26th day of January, 1977.

L L Sawyer
Notary Public

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

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

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 45 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with the U. S. Nuclear Regulatory Commission and been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania and Dauphin County, Pennsylvania by deposit in the United States mail, addressed as follows:

Mr. Weldon B. Arehart
Board of Supervisors of
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Board of County Commissioners
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Dauphin County Court House
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By [signature]
Vice President-Generation

Dated: January 26, 1977

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Metropolitan Edison Co. (Met-Ed)
Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR50
Docket No. 50-289

Technical Specification Change Request No. 45

The Licensee requests that the attached changed pages replace pages v, vi, vii, 1-5, 2-3, 2-5, 2-6, 2-7, 3-1, 3-2, 3-34, 3-34a, 3-35, 3-35a, 3-36, 4-59 figures 2.1-1, 2.1-2, 2.1-3, 2.3-1, 2.3-2, 3.5-2A through 3.5-2J and table 2.3-1 of the existing technical specifications.

Reason for Proposed Change

These changes are necessary to ensure safe operation of TMI-1 at a rated core power of 2535 MWt for the duration of Cycle 3 and are based on a Cycle 2 burnup of 253 ± 10 EFPD.

Changes to technical specifications are necessary as a result of the changed distribution of fuel by insertion of 48 fresh batch 5 assemblies and 13 once burned batch 1a assemblies, discharge of batch 2 fuel, and relocation of the remaining twice burned batch 3 and once burned batch 4 fuel. Other changes to the technical specification are responsive to recent NRC concerns relative to postulated fuel rod bowing and correcting a previous error in densification penalty as discussed in our Change Request 36 (July 7, 1976). In addition, revised quadrant tilt limits and new limits for the Axial Power Shaping Rods (APSR's) are specified. Furthermore our change request to revise the high pressure trip and pressurizer code safety valve settings (Change Request #39, October 8, 1976 as supplemented by our letter of October 21, 1976) is hereby extended to apply to Cycle 3, therefore approval of these proposed technical specifications effectively fulfills our Change Request #39.

Safety Evaluation Justifying Change

Cycle 3, like the reference Cycle 2, was evaluated using the B&W-2 CHF correlation using a 95/95 confidence level and a RC flow equal to 106.5% of Cycle 1 design flow. The B&W 2 CHF correlation and the 106.5% RC flow value have been reviewed and approved by the NRC. In addition, the assured flow of 106.5% for Cycle 3 is even more conservative than for Cycle 2 in that Cycle 3 incorporates 48 additional Mark B4 assemblies with improved flow characteristics, for a total of 104 of 177 assemblies with improved flow relative to the Mark B design of batches 1, 2 and 3.

The recent NRC concerns relative to rod bowing have been considered by including a DNBR penalty of 11.2% imposed by the NRC. It should be noted that the former error made in the application of densification penalty has also been corrected (reference GQL-0821, June 5, 1976).

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Revised quadrant tilt limits are proposed to clarify how the tilt limits should be applied based on the method used in determining tilt. The revised tilt limits are more restrictive than formerly used resulting in additional operating flexibility for the rod insertion limits while at the same time not reducing the degree of conservatism that would exist had the insertion limits been based on a larger tilt value.

Limits for the APSR's have been established which provide additional control of power peaking through an improved definition of core power distribution. This change provides additional operating flexibility while at the same time not reducing the degree of conservatism relative to the limits that would be appropriate if no APSR limits were specified.

The increases in the high pressure trip and pressurizer code safety valve settings are justified by Change Request #39 (October 8, 1976) as supplemented October 21, 1976. Since the parameters relevant to the supporting analyses of Change Request #39 remain unchanged for Cycle 3, except for the nominal moderator and doppler coefficients used, and since these coefficients are a little different for Cycle 3 and are bounded by the moderator and doppler sensitivity studies, the justification given in Change Request #39 is applicable to Cycle 3.

The FSAR accidents which are dependent on core design or fuel loading have been reviewed for Cycle 3. The results presented in the FSAR bound the Cycle 3 results.

The LOCA analyses were reviewed and approved by the NRC with amendment 17 to the TMI-1 operating license. Since that time an error has been identified in the CRAFT 2 code which comprises part of the ECCS evaluation model required by Appendix K to 10CFR50. The errors to this code have been corrected and selected analysis have been performed to show that the code errors resulted in much less than a 20° F increase in the peak cladding temperatures previously predicted which themselves were well below the 2200° F limit specified by regulations. The NRC evaluated these additional analyses and concluded that no changes in Operating reactor technical specifications were necessary. In spite of the known error in the Appendix K model, the model has been shown to be conservative. Consequently, B&W 10103 Rev. 1 is still considered a valid reference to support approval of these proposed technical specifications while revised analyses are being conducted. Additional and more detailed information supportive of these proposed technical specifications is provided in the attached Three Mile Island Unit 1, Cycle 3, Reload Report.

For all of the attached proposed technical specifications not previously requested the limits have become more conservative or are as conservative as their equivalent for Cycle 2. The above is true since these limits were developed for both Cycle 2 and Cycle 3 to meet the same criter .. The rod insertion and power imbalance limits may be different for various fuel distributions from cycle to cycle, however, the revised limits should not be considered a change that involves significant hazards since the degree of conservatism has not been reduced. The revised high pressure trip and code safety valve settings and the elimination of the reactor vent valve flow penalty (Change Requests 39 and 36 respectively) have been previously noticed to the public. As explained above, no other significant hazards are involved, therefore, public notice should not be required.

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Based on the above, it is concluded that this change does not increase the probability of occurrence or increase the severity of an accident, nor does it create the possibility of occurrence of an accident not previously considered. Therefore, this change does not represent undue risk to the health and safety of the public and continued operation of TMI-1 at a rated core power of 2535 MWt is justified.

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1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left[\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 REACTOR POWER IMBALANCE

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in "b" below.

b. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling or personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required by the "Containment Integrity Check List" attached to the operating procedure "Containment Integrity and Access Limits."

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

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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.

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.

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.

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. Curve 1 is more restrictive than any other reactor coolant pump situation 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) 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. These trip setpoints are setting limits on the setpoint side of the protection system bistable comparators. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors. 39

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

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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.

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 39 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 36 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 the top half of the 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.

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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 has
been established to maintain the system pressure below the safety
limit (2750 psig) for any design transient. Due to calibration and 39
instrument errors, the safety analysis assumed a 30 psi pressure
error in the high reactor coolant system pressure trip setting.

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 - 39
5143) and a low pressure trip value of 1770 psig.

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 ±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 620 F even under worst case conditions. The safety analysis used a high temperature trip set point of 620 F.

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 620 F 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

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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.

TABLE 2.3-1(6)

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)	108 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.	See Table 2.3-2 (7)	See Table 2.3-2 (7)	See Table 2.3-2 (7)	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
(6) Trip settings limits are setting limits on the set point side of the protection system bistable comparators
(7) Settings are specified in Table 2.3-2 corresponding to the pressurizer code safety valve settings.

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TABLE 2.3-2

HIGH PRESSURE REACTOR TRIP AND PRESSURIZER CODE SAFETY VALVES SETTING PAIRS

Alternate Pairs of Settings	High Pressure Reactor Trip Setting	Pressurizer Code Safety Valve Setting	
A 1/	2375 psig	2435 psig	
B 2/	2405 psig	2500 psig	39

Notes:

1/ The pair of settings denoted as A will be utilized until the pressurizer code safety valve setting can be increased from 2435 psig to 2500 psig.

2/ The pair of settings denoted as B will be utilized on a permanent basis following resetting of the pressurizer code safety valves.

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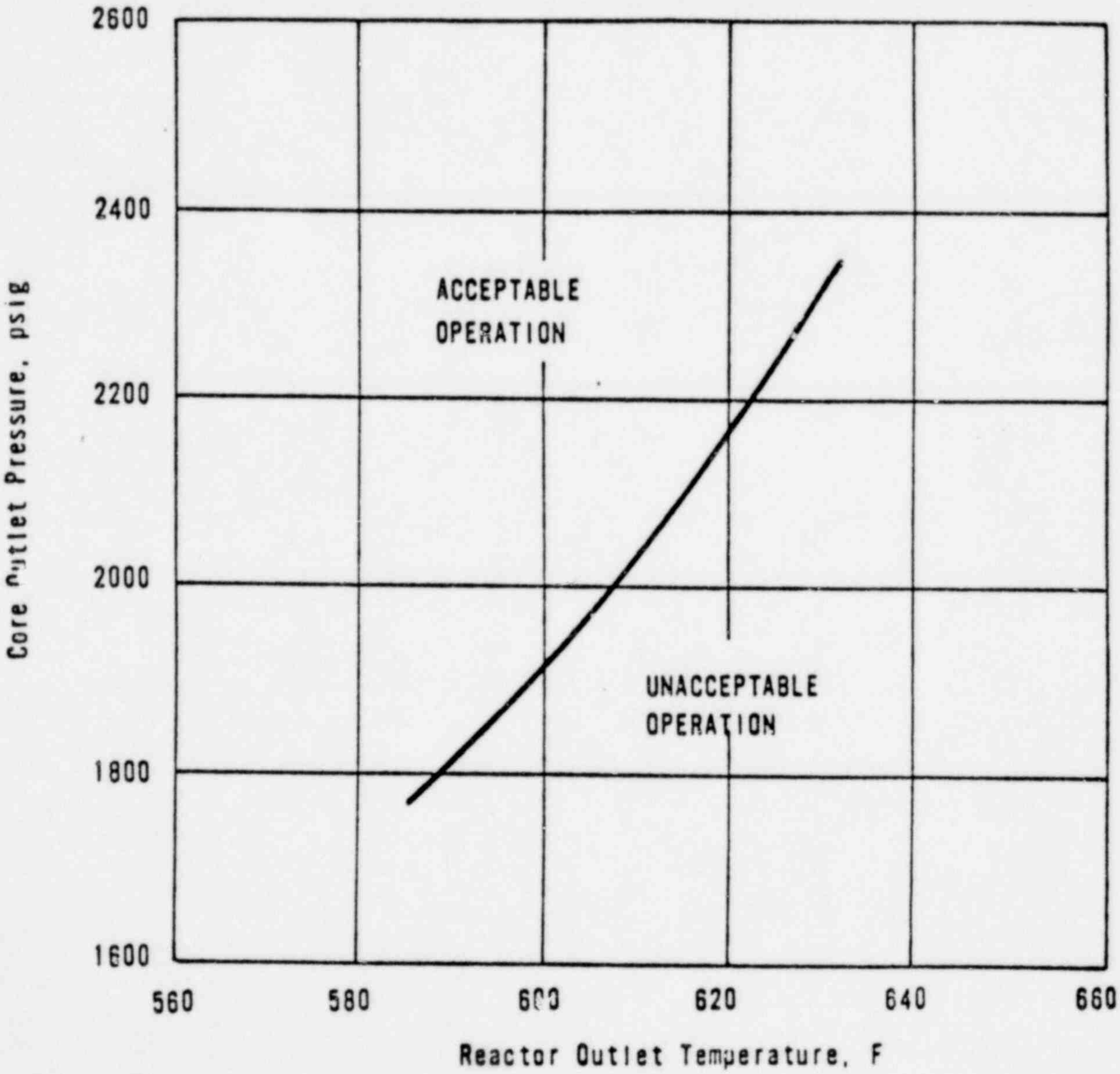

TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY LIMIT
Figure 2.1-1

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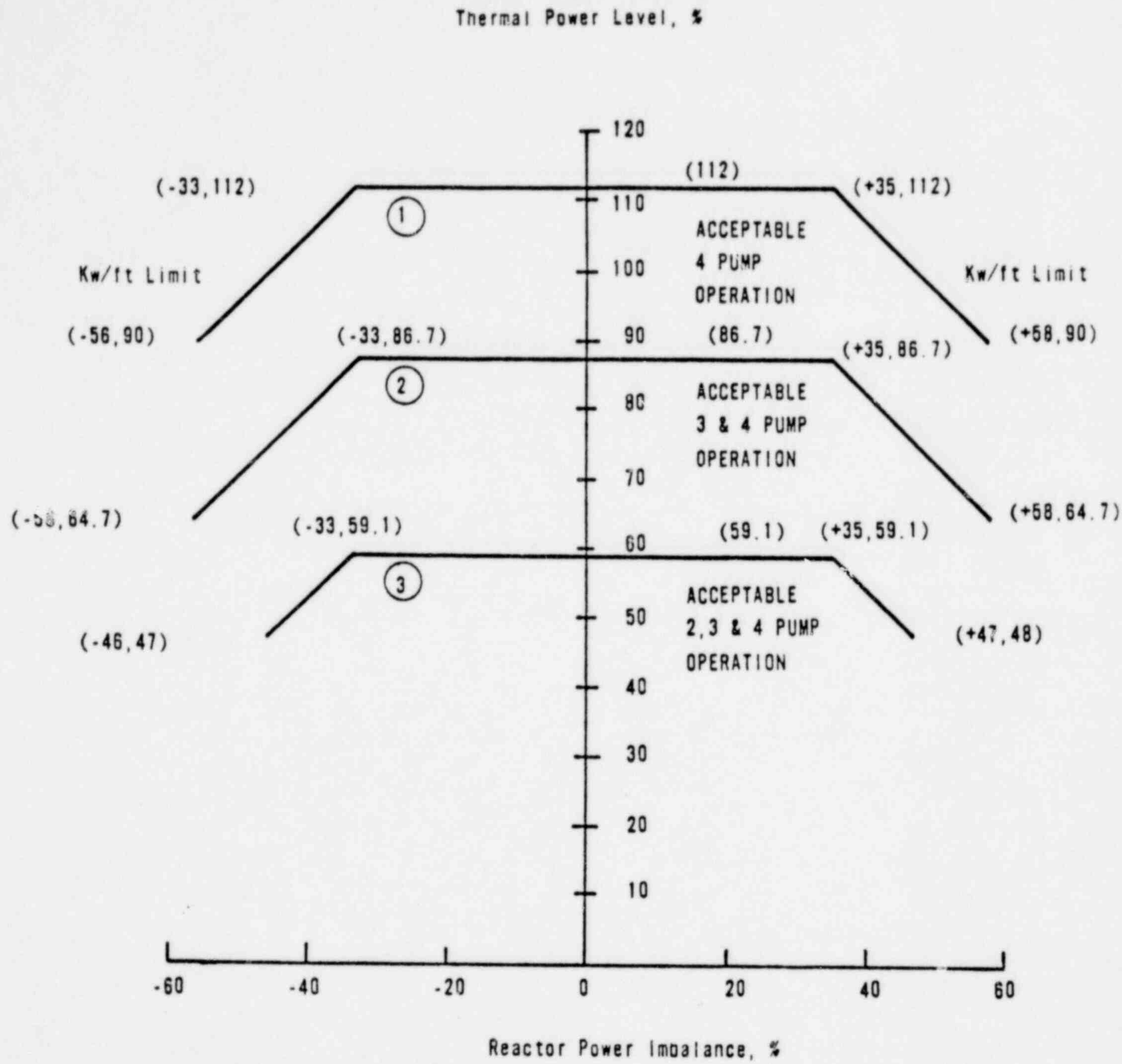

CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY LIMITS

Figure 2.1-2

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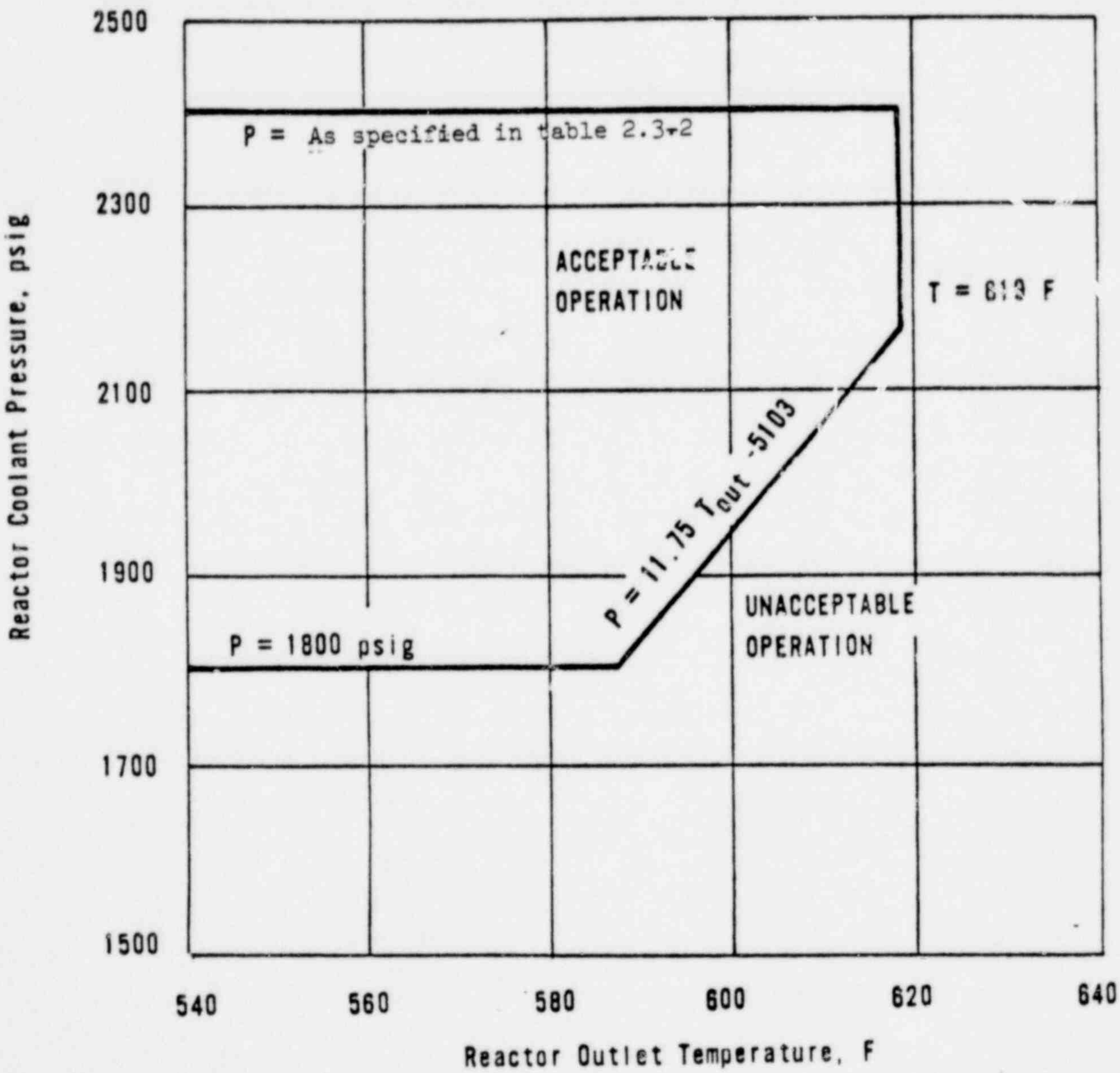

TMI-1, UNIT 1, CYCLE 3
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1

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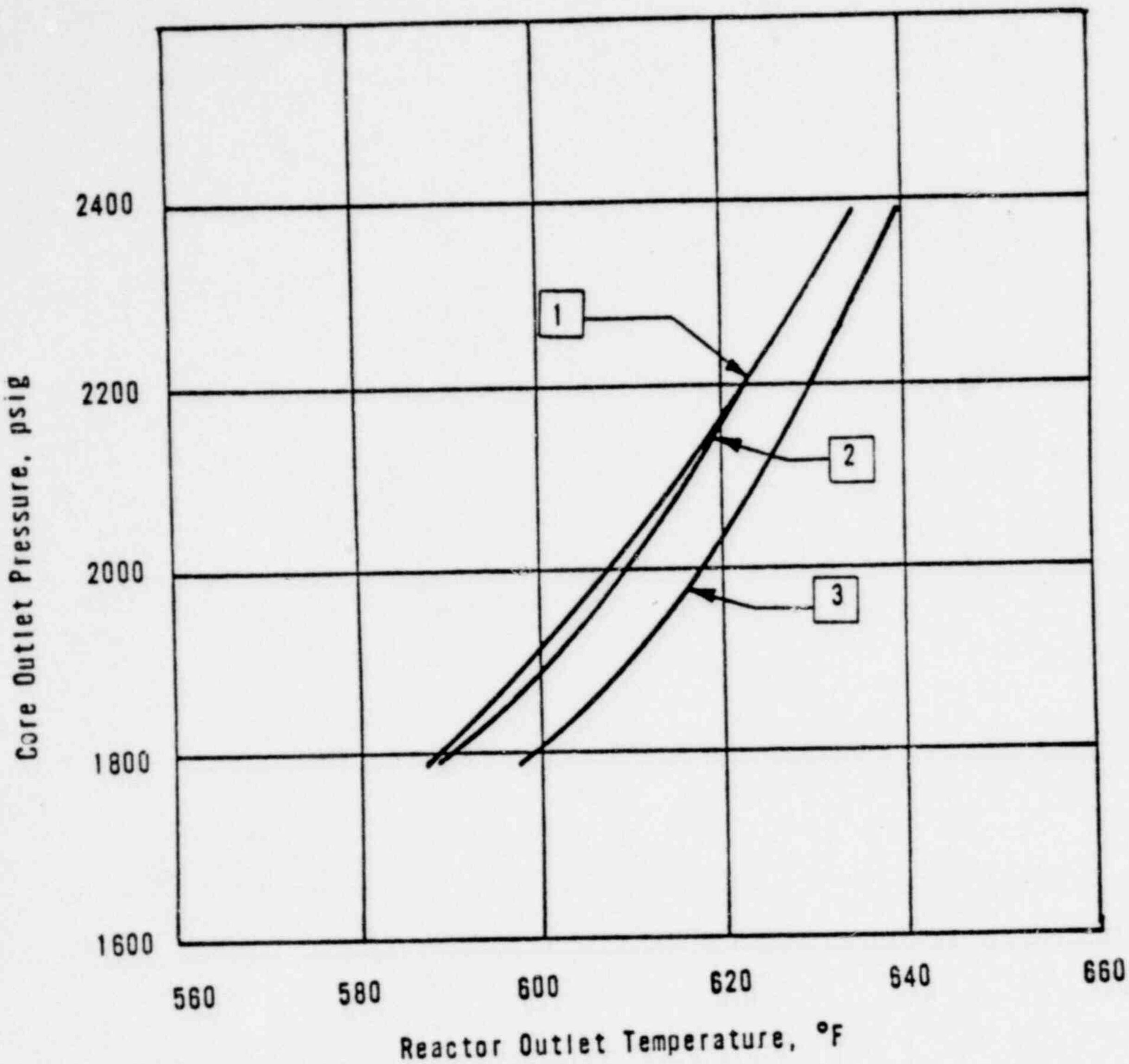

REACTOR COOLANT FLOW

CURVE	(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

TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY
Figure 2.1-3

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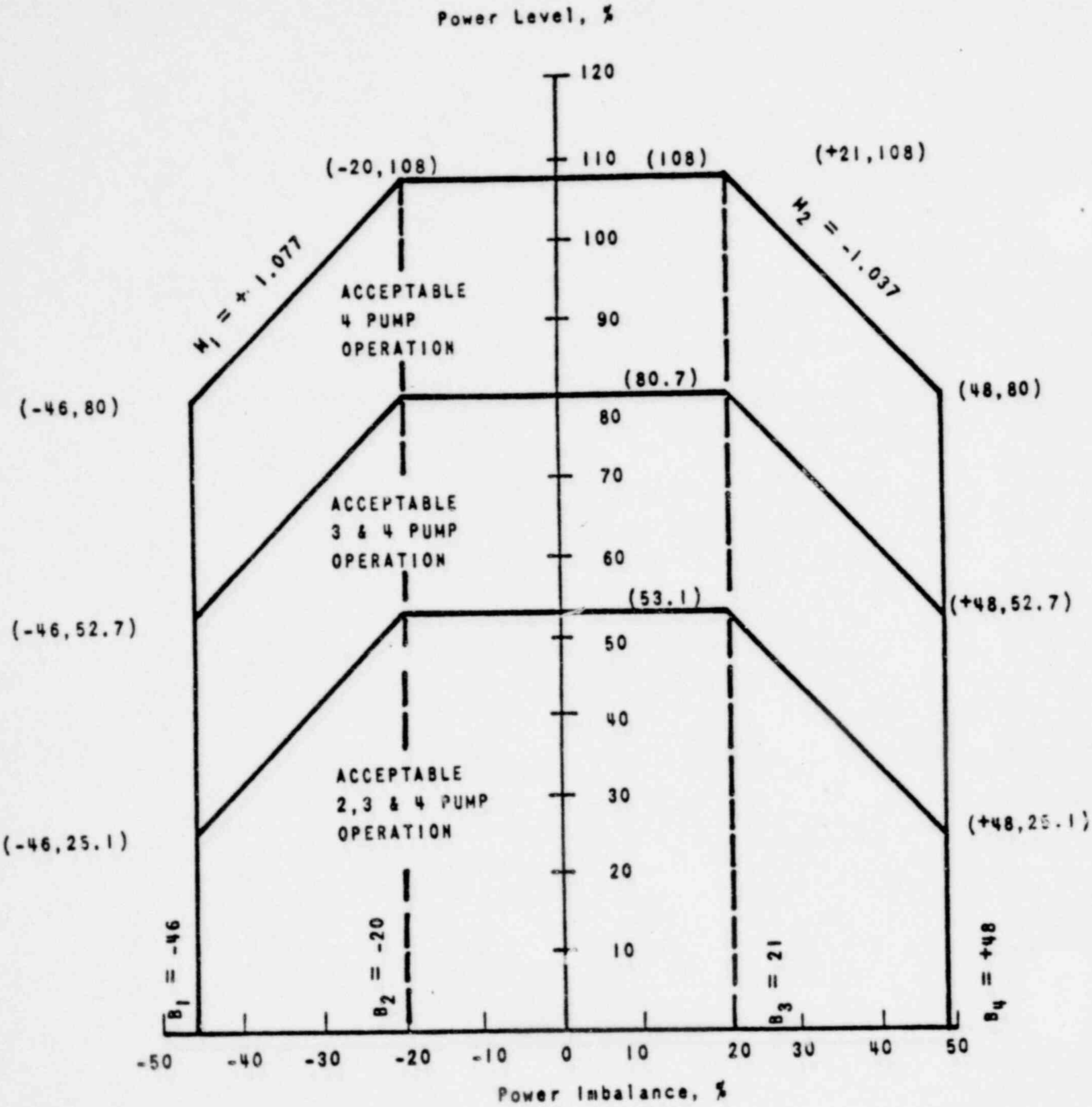

TMI-1, UNIT 1, CYCLE 3
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-2

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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 one of the lift settings specified in Table 2.3-2 ±1% allowance for error. 39

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.

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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 one of the settings specified in Table 2.3-2 ±1 percent allowance for error and each valve shall be capable of relieving 39
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

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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.

3.5.2.4 Quadrant tilt:

a. Except for physics tests the quadrant tilt shall not exceed +2.66% as determined using the full incore detector system.

b. When the full incore detector system is not available and except for physics tests quadrant tilt shall not exceed +1.47% as determined using the minimum incore detector system.

c. When neither incore detector system above is available and except for physics tests quadrant tilt shall not exceed +0.81% as determined using the power range channels displayed on the console for each quadrant (out of core detector system).

d. Except for physics tests if quadrant tilt exceeds the tilt limit 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 the tilt limit. 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 the tilt limit.

e. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit 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.

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

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 the tilt limit.

f. Except for physics or diagnostic testing, if quadrant tilt is in excess of +25.72% determined using the full incore detector system (FIT), or +24.09% determined using the minimum incore detector system (MIT) if the FIT is not vailable, or +21.39% determined using the out of core detector system (OCT) when neither the FIT nor MIT are available, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant tilt is permitted provided that the thermal power allowable is restricted as stated in 3.5.2.4.d above.

g. 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 $\pm$ 5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating 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. Also excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on Figures 3.5-2K, 3.5-2L, and 3.5-2M. If any of these 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 for operation 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 at periodic intervals of 30 full power days using the incore instrumentation detection system, to verify that the power distribution is within the limits shown in Figure 3.5-2 J.

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, 3.5-2F, 3.5-2K, 3.5-2L, and 3.5-2M and if quadrant tilt is at the limit. Additional conservatism is introduc by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors.

e. Postulated fuel rod bow effects

The Rod index versus Allowable Power curves of Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E, 3.5-2F, 3.5-2K, 3.5-2L and 3.5-2M 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 four 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:

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

Control rod groups are withdrawn in sequency 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, then 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 3.5.2.4 have been established within the thermal analysis design base using an actual core tilt of +3.41% which is equivalent to a +2.66% tilt measured with the full incore instrumentation with measurement uncertainties included.

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	50%
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

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4.16 REACTOR INTERNALS VENT VALVES SURVEILLANCE

Applicability

Applies to Reactor Internals Vent Valves.

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

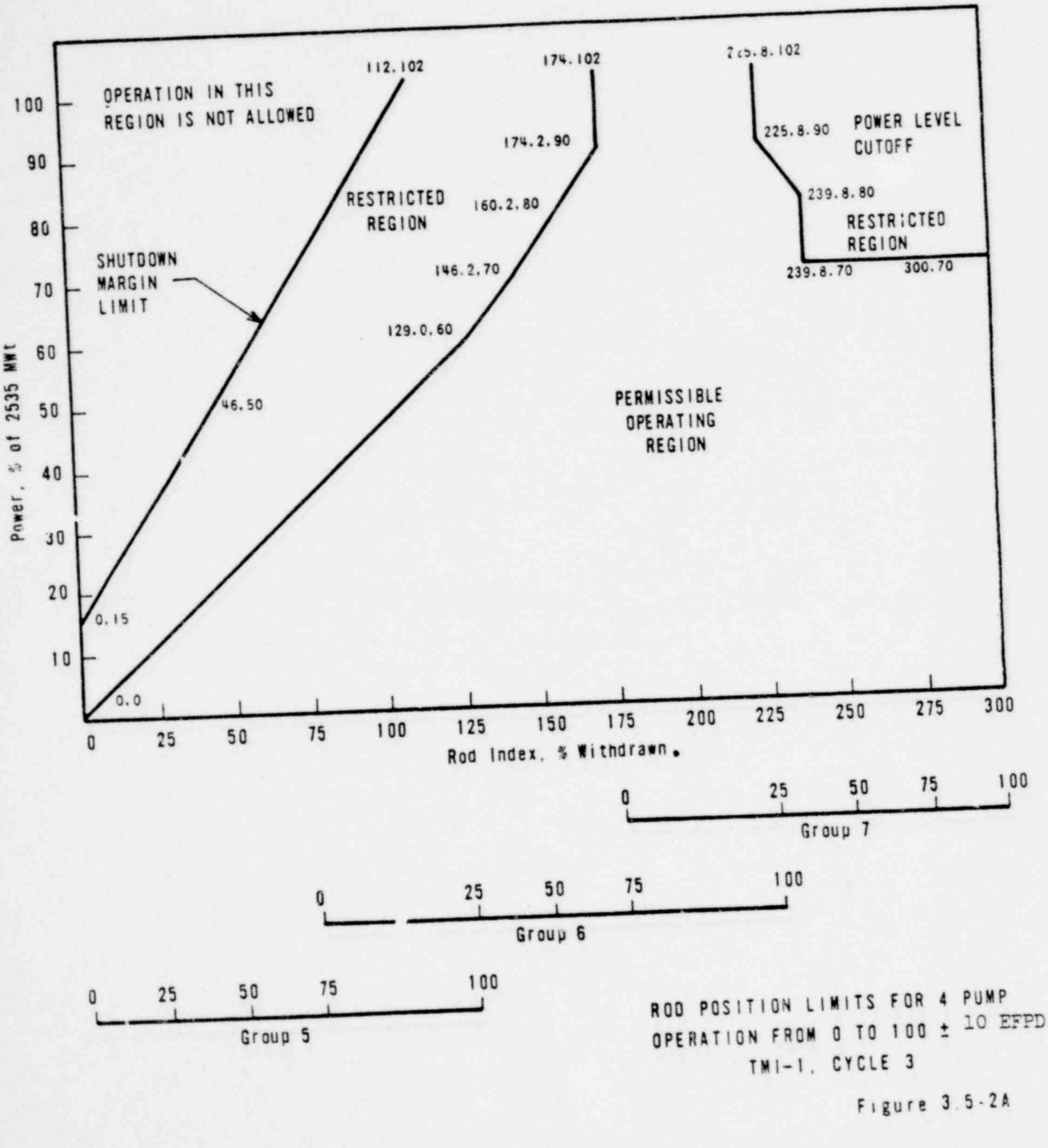
Specification 36

4.16.1 At intervals not exceeding the refueling interval, each reactor internals vent valve will be tested to verify that no valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Bases

Verifying vent valve freedom of movement insures that coolant flow does not bypass the core through reactor internals vent valves during operation and therefore insures the conservatism of Core Protection Safety limits as delineated in figures 2.1-1 and 2.1-3, and the flux/flow trip setpoint.

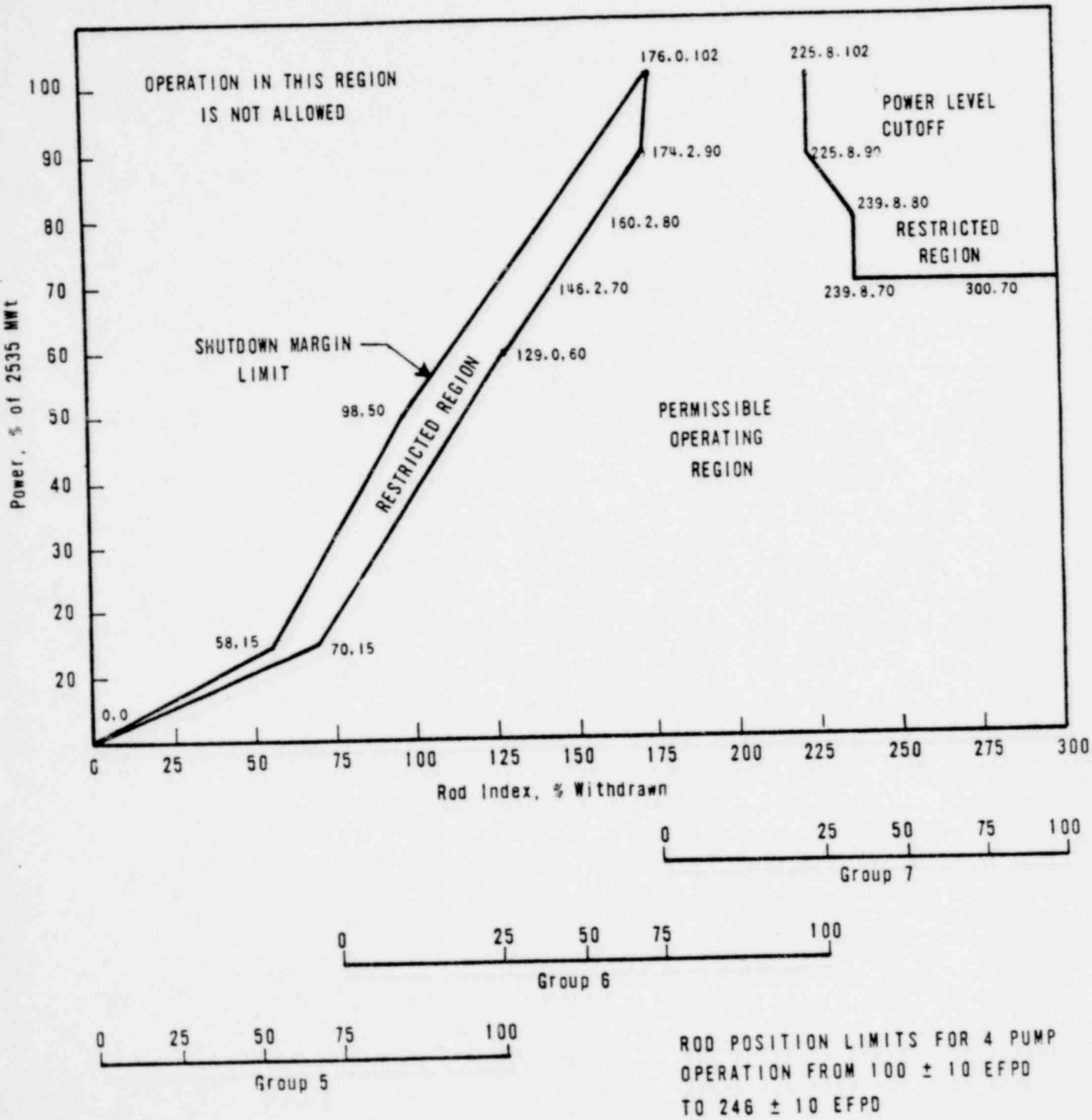
1469 235



ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1, CYCLE 3

Figure 3.5-2A

1469 236

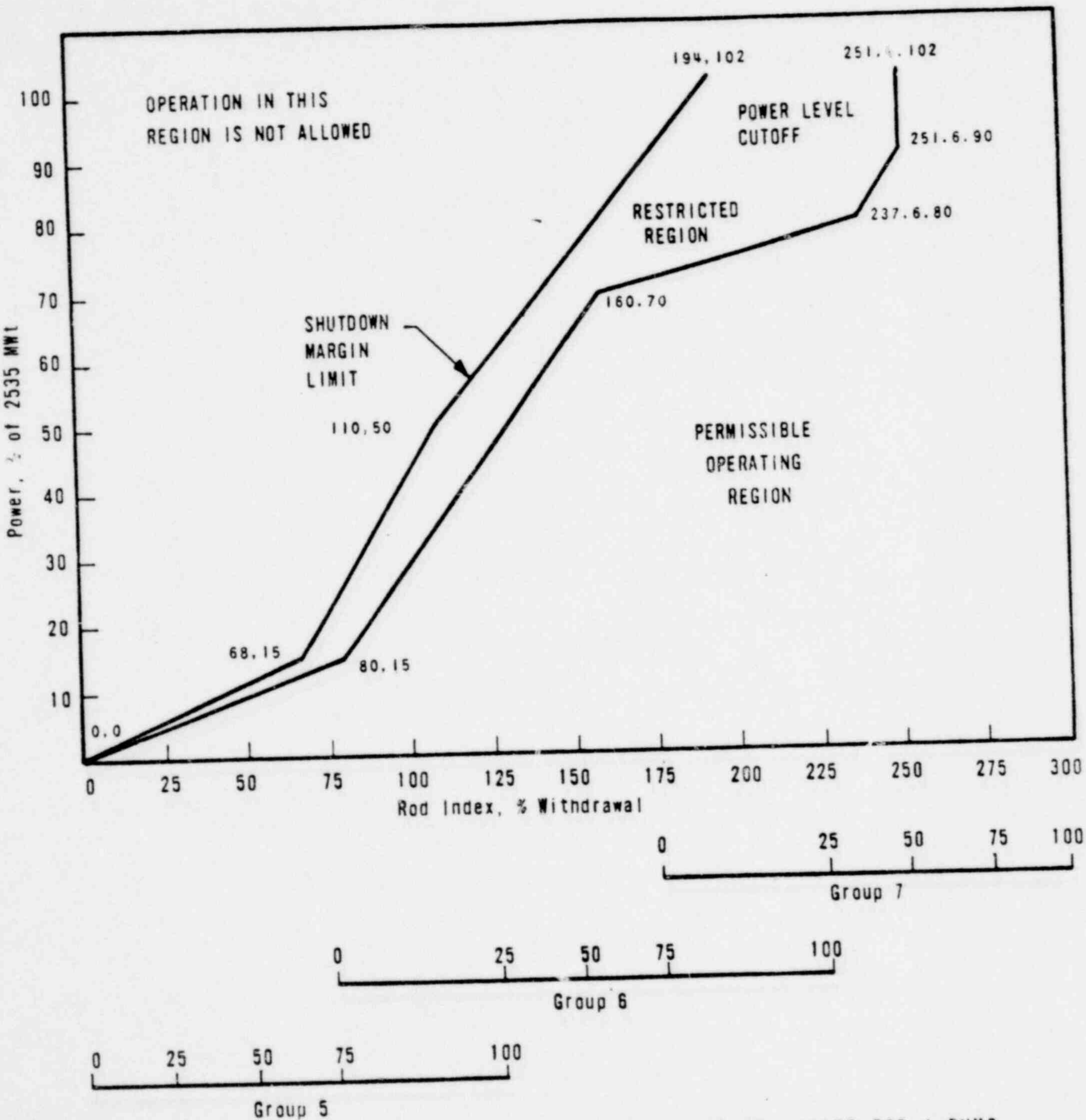


ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 100 ± 10 EFPD TO 246 ± 10 EFPD

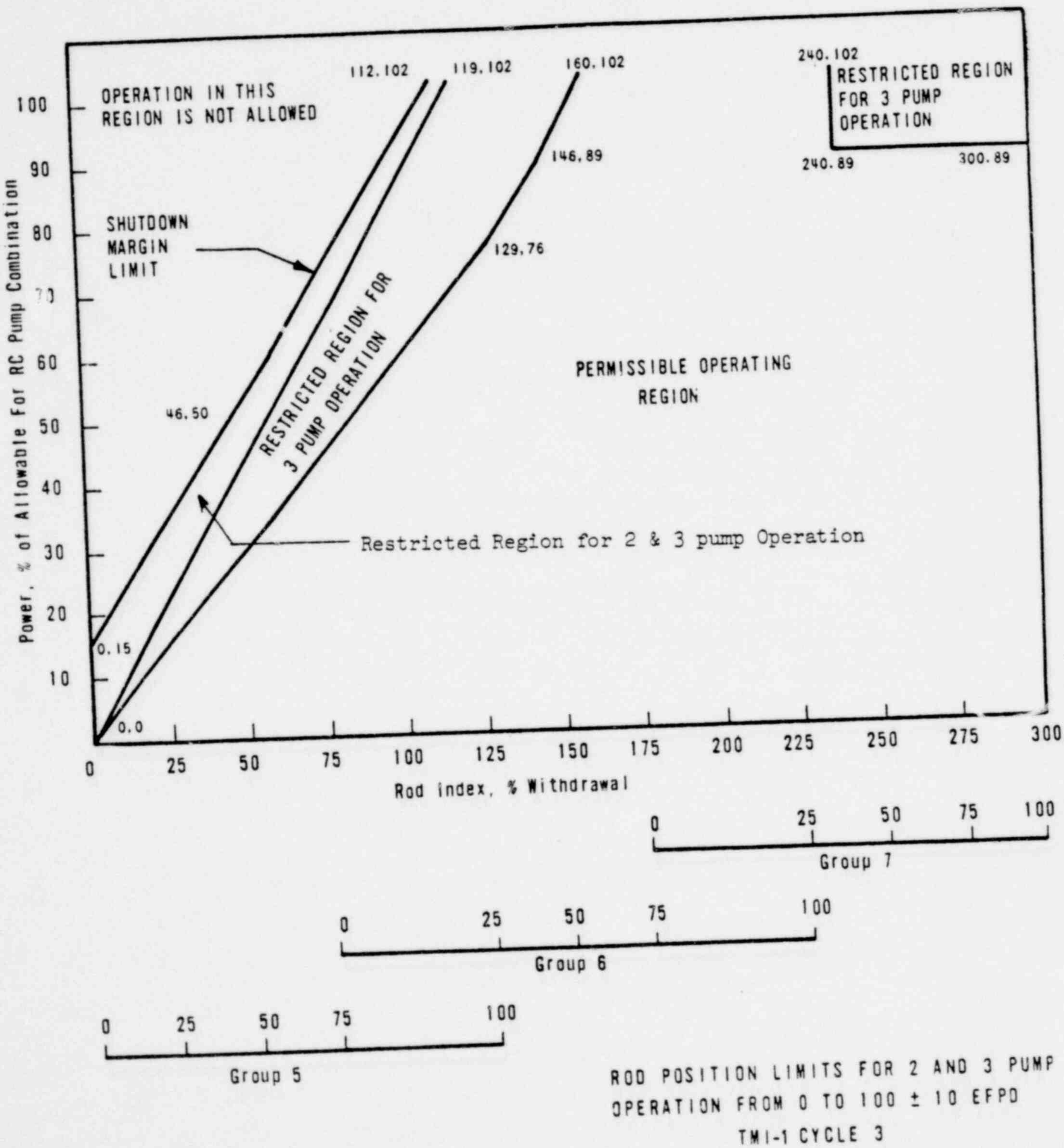
TMI-1, CYCLE 3

Figure 3.5-2B

1469 237



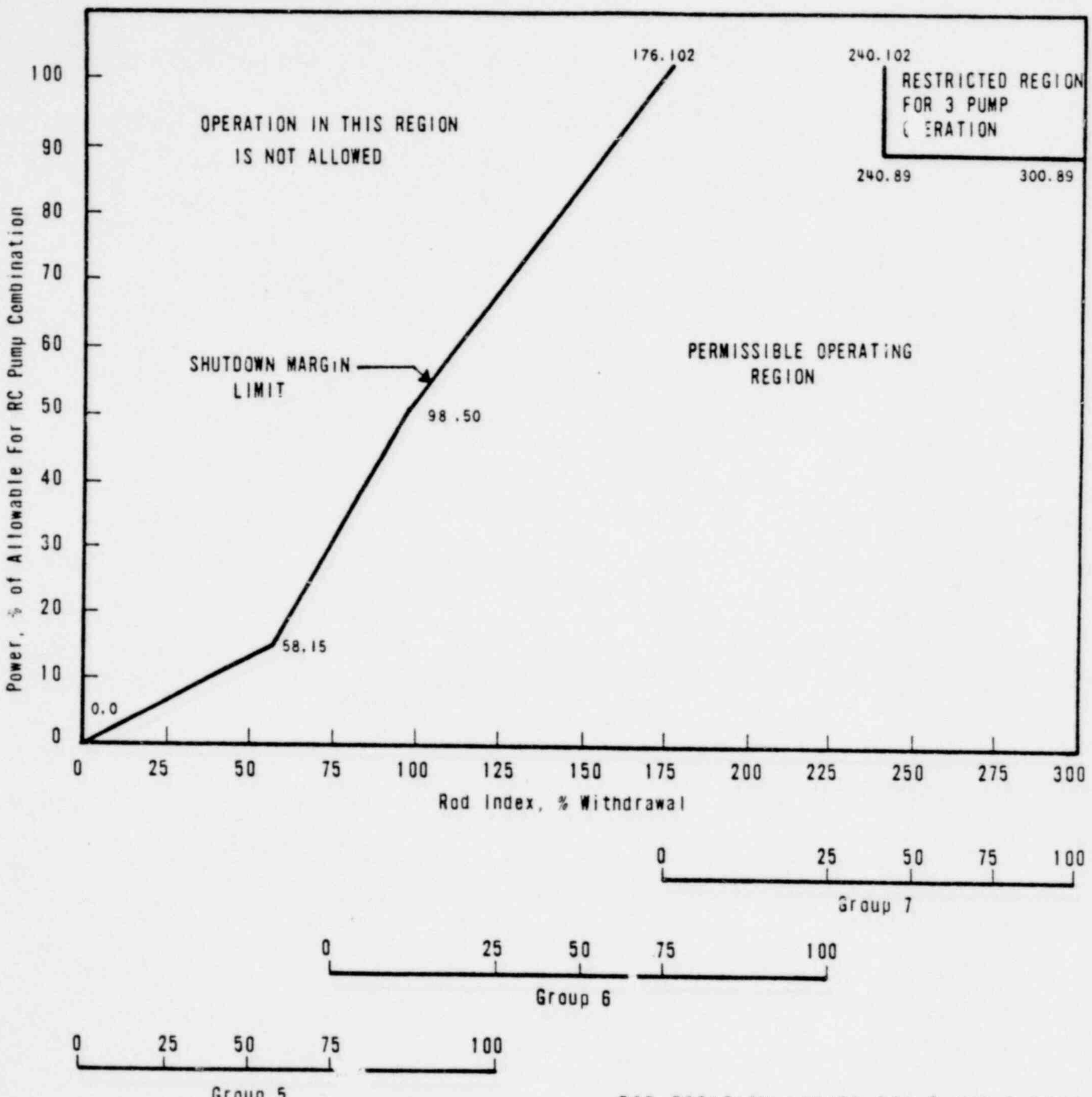
ROD POSITION LIMITS FOR 4 PUMP
OPERATION AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2C



ROD POSITION LIMITS FOR 2 AND 3 PUMP
OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1 CYCLE 3

Figure 3.5-20

1469 239

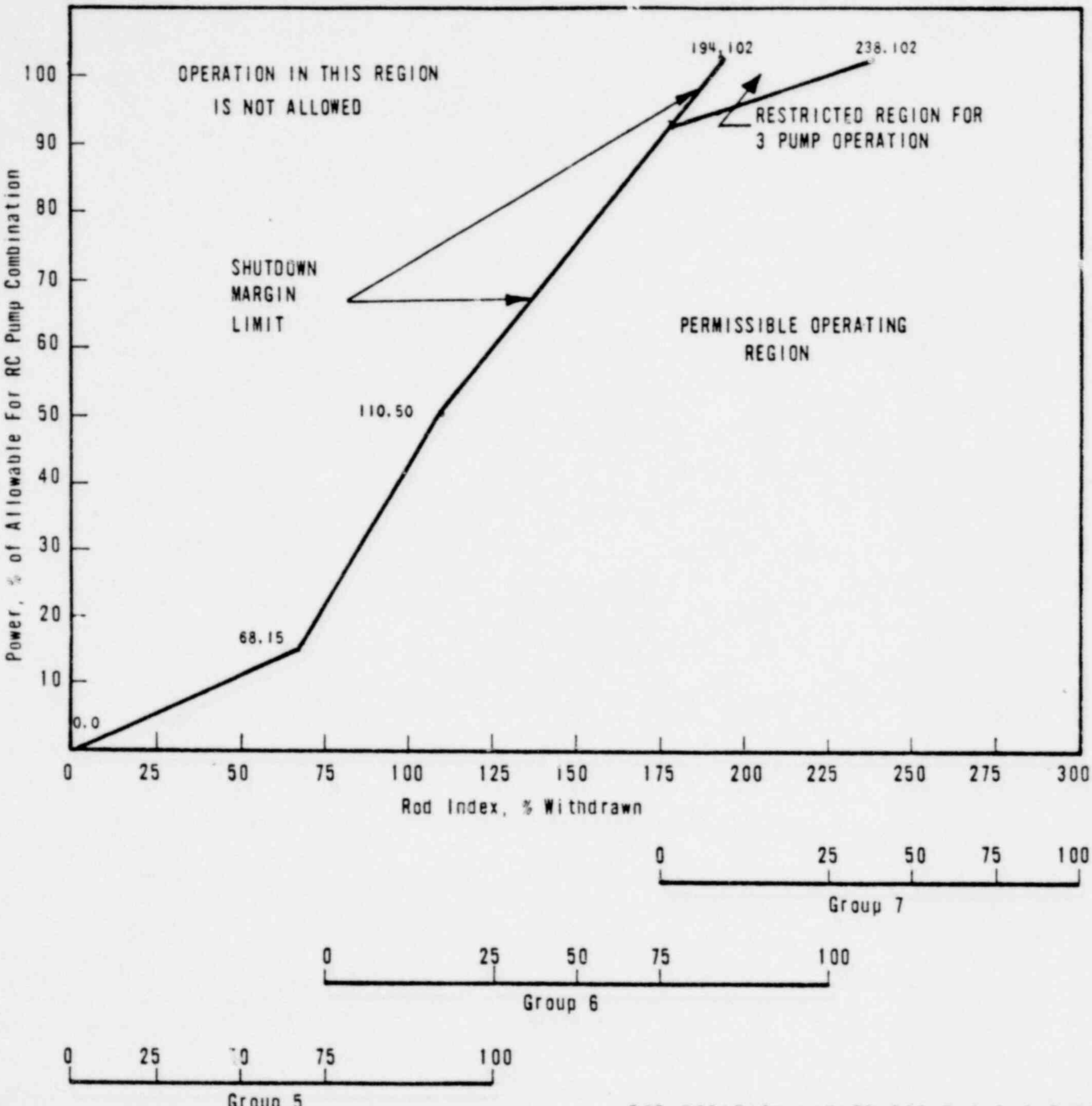


ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION FROM 100 ± 10 EFPD TO 246 ± 10 EFPD

TMI-1, CYCLE 3

Figure 3.5-2E

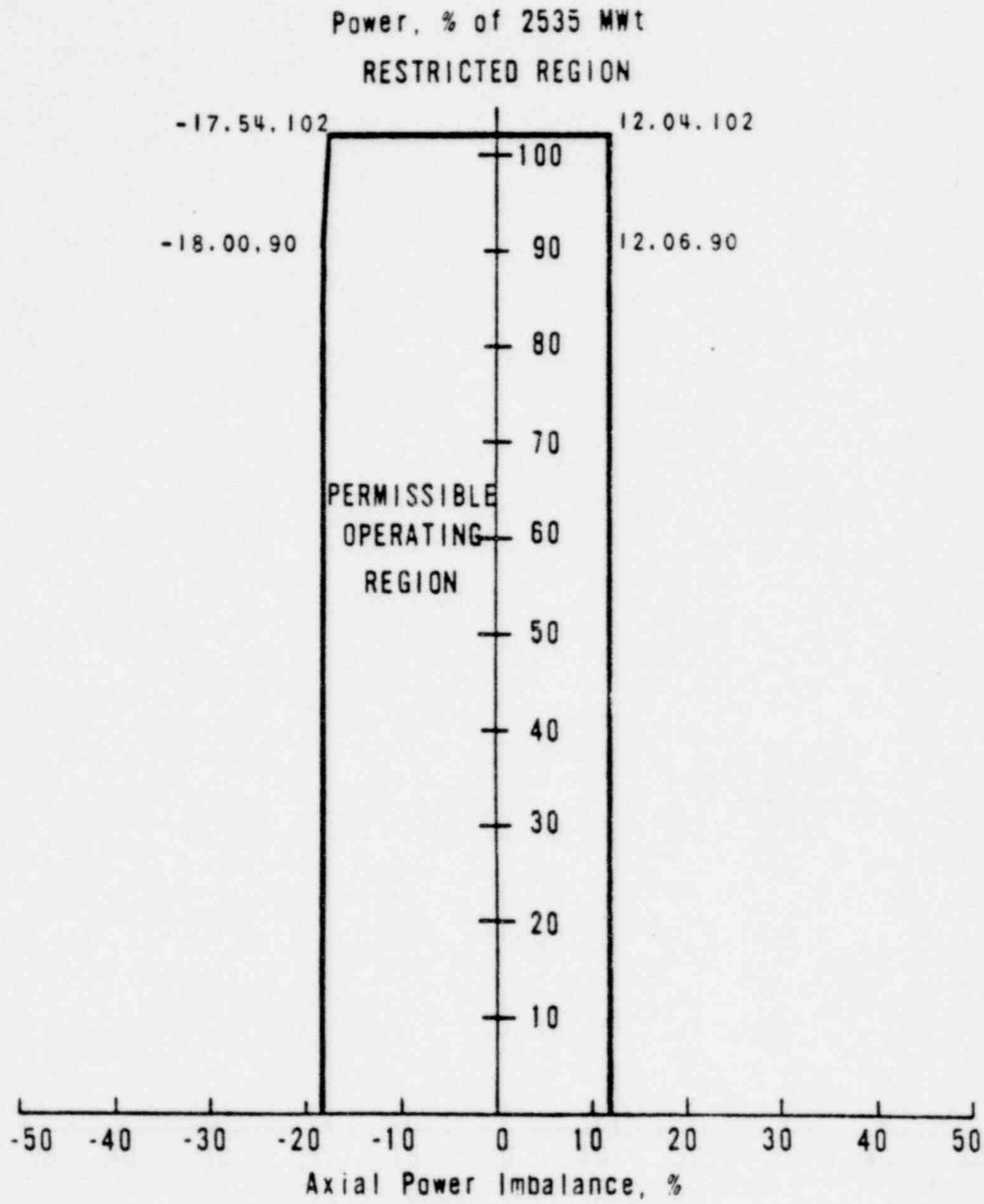
1469 240



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION AFTER 246 ± 10 EFPD
TMI-1, CYCLE 3

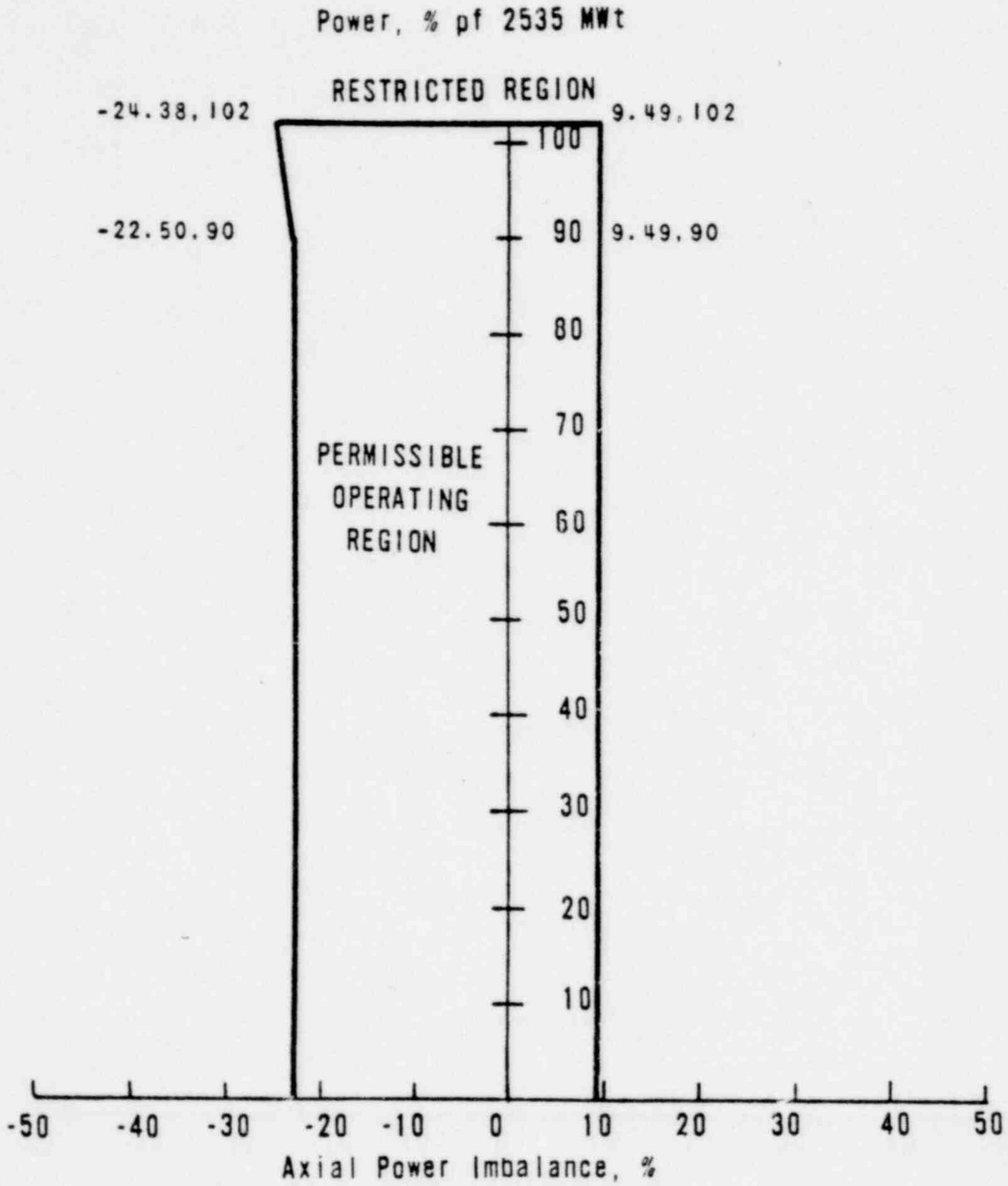
Figure 3.5-2F

1469 241



POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2G

1469 242

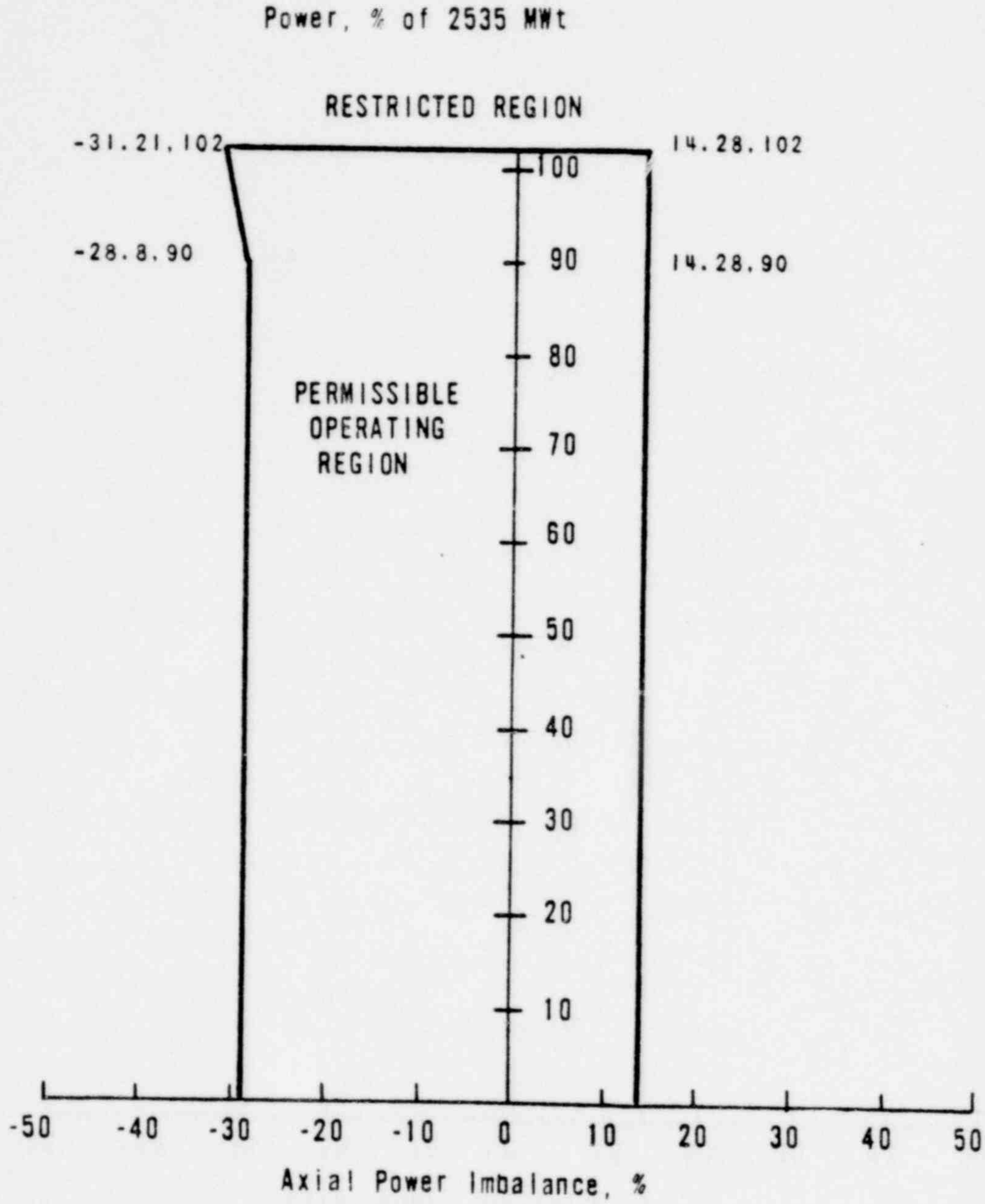


POWER IMBALANCE ENVELOPE FOR OPERATION FROM 100 ± 10 TO 246 ± 10 EFPD

TMI-1 CYCLE 3

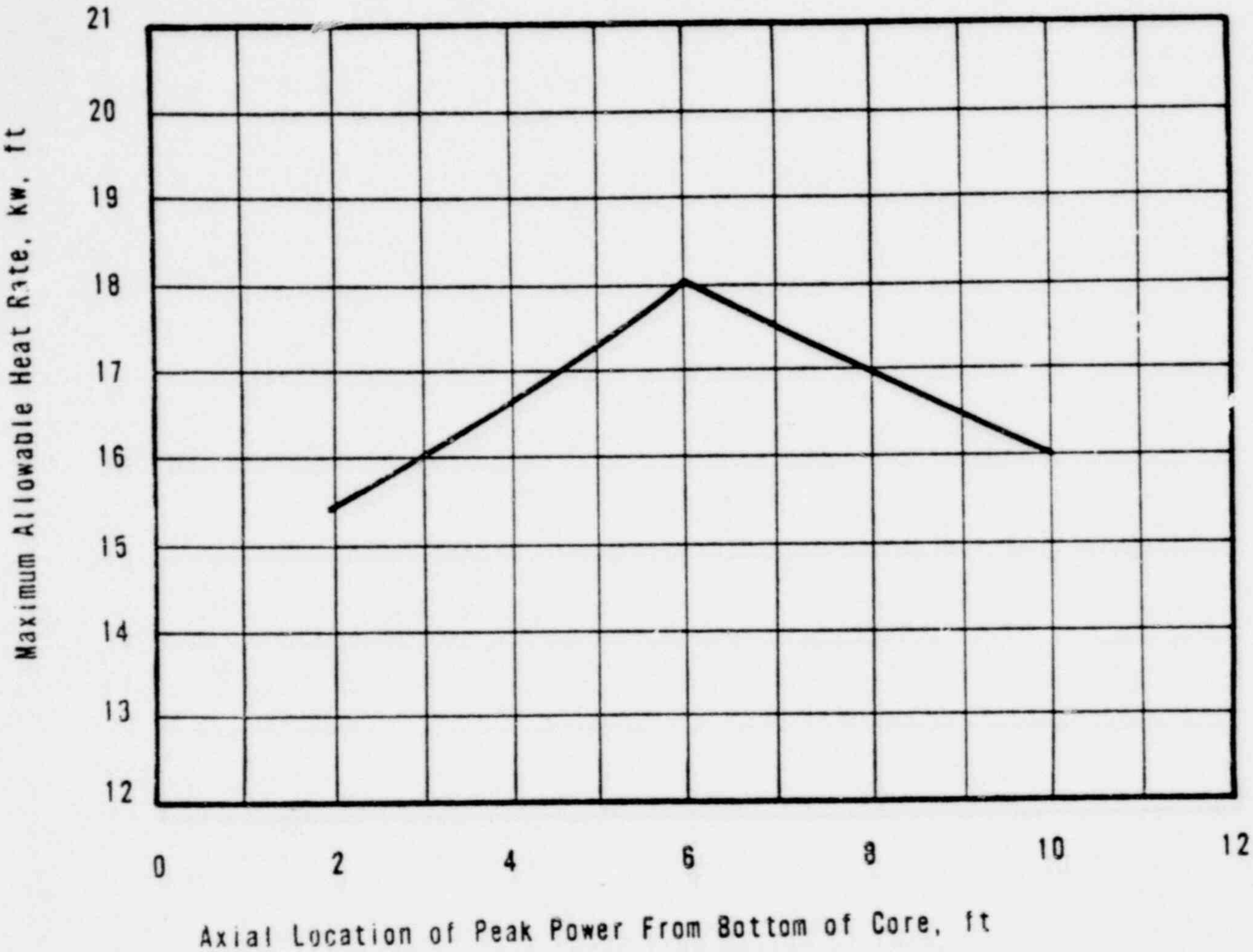
Figure 3.5-2H

1469 243



POWER IMBALANCE ENVELOPE FOR
OPERATION AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2I

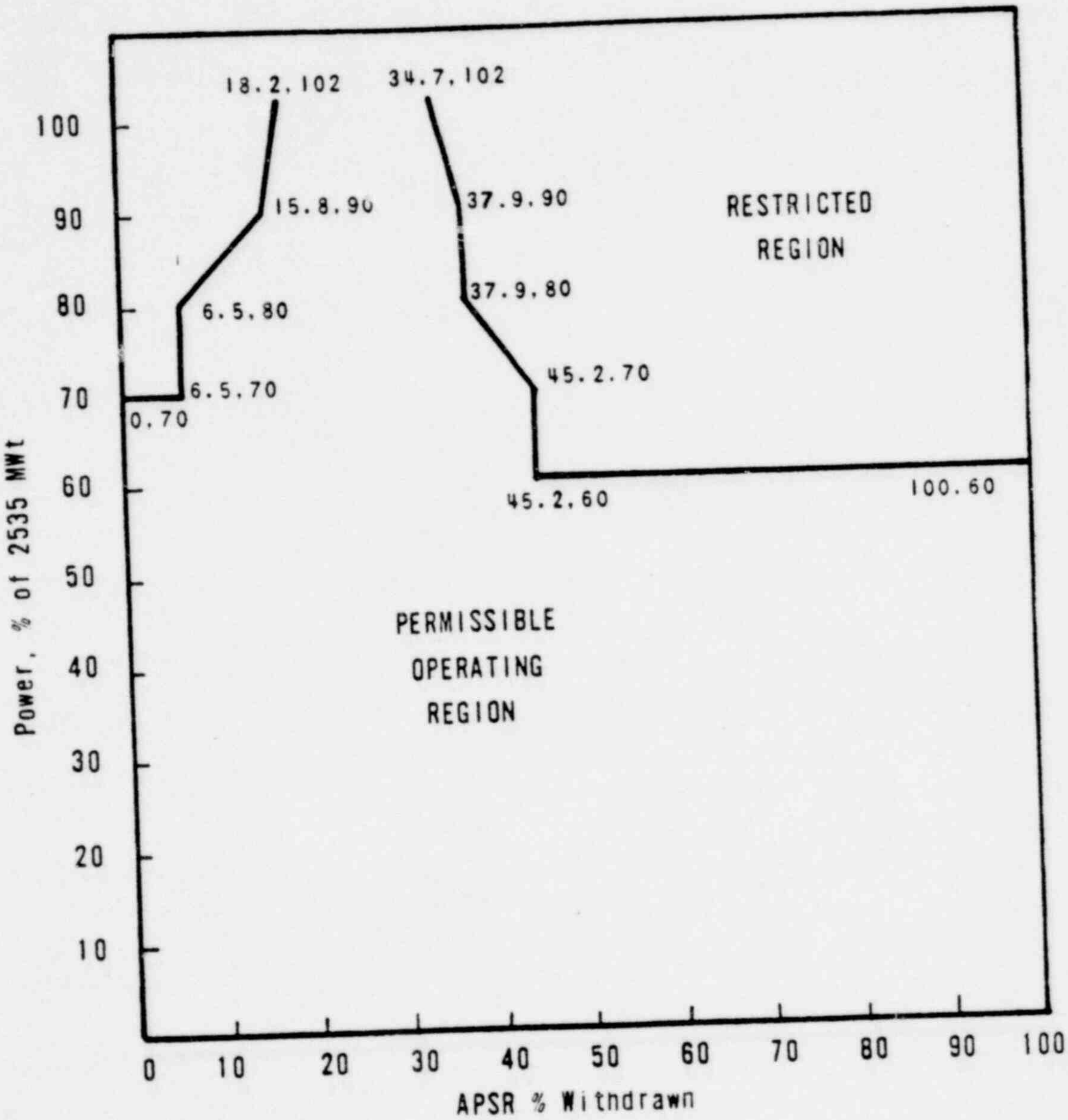
1469 244



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE

Figure 3.5-2J

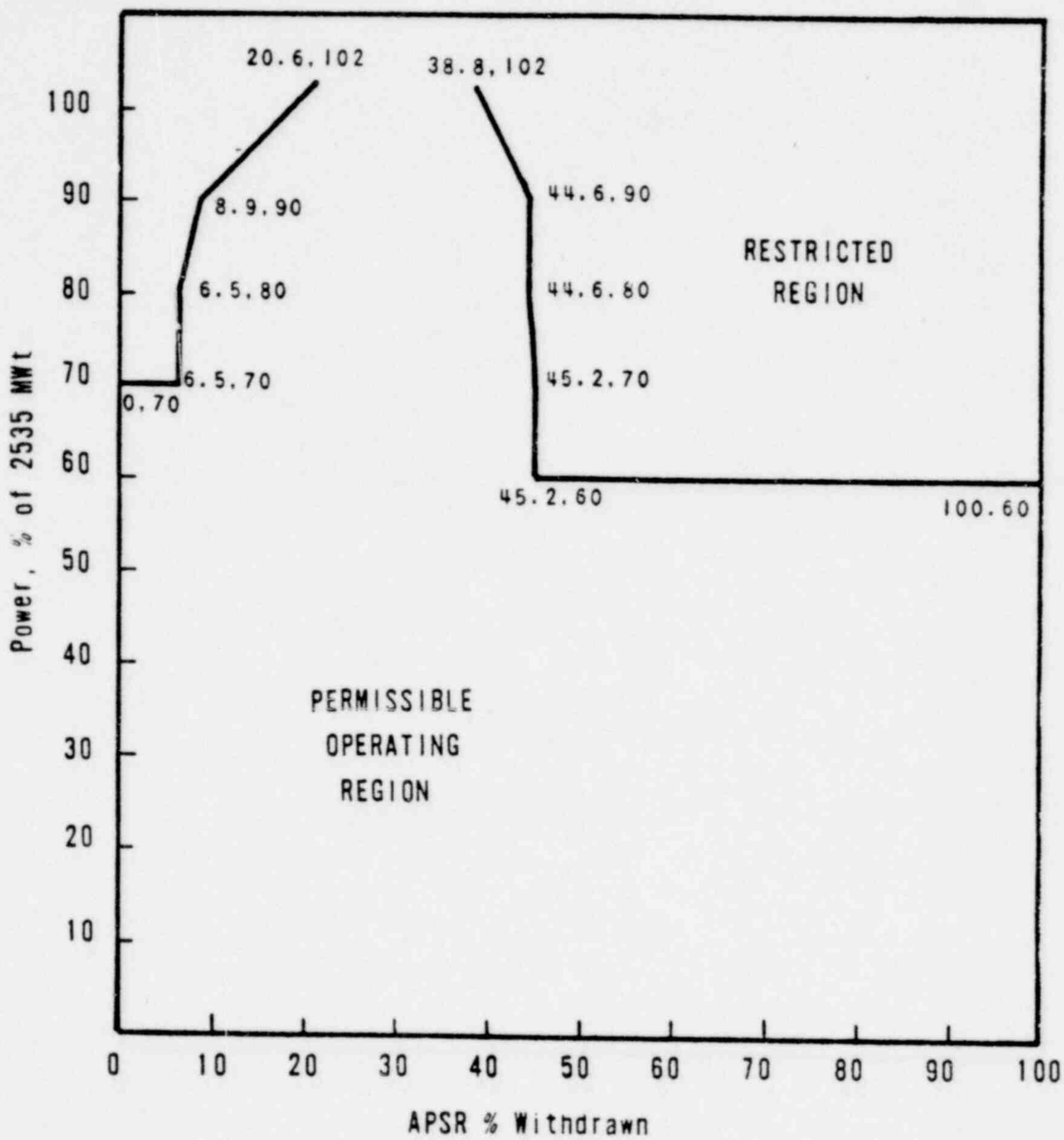
1469 245



APSR POSITION LIMITS FOR
OPERATION FROM 0 TO 100 ±
10 EFPD

TMI-1 CYCLE 3
Figure 3.5-2K

1469 246

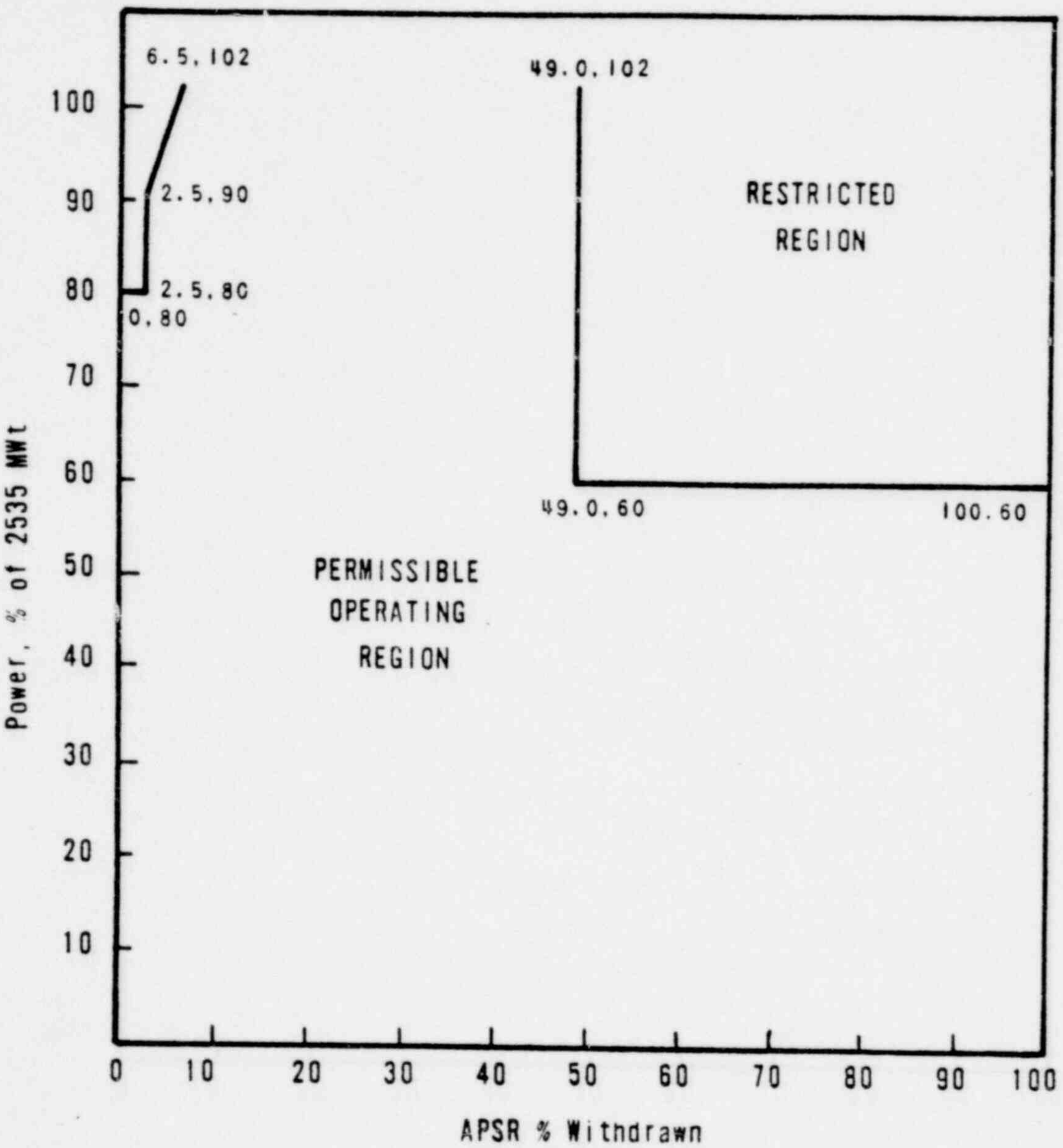


APSR POSITION LIMITS FOR
OPERATION FROM 100 ± 10
TO 246 ± 10 EFPD

TMI-1 CYCLE 3

Figure 3.5-2L

1469 247



APSR POSITION LIMITS FOR OPERATION
AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2M