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. 84

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 J G Herbein
Vice President

Sworn and subscribed to me this 23rd day of June, 1978.

George J Troffer
Notary Public

GEORGE J. TROFFER
Notary Public, Reading, Berks Co.
My Commission Expires Jan. 25, 1982

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Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Technical Specification Change Request No. 84

The Licensee requests that the following changes be made to the TMI-1 License No. DPR-50, Appendix A, Technical Specifications:

1) The attached pages 2-4, 2-9 and 3-34a replace the corresponding pages of the existing Technical Specifications.

2) The attached figures 2.1-2, 2.3-1 and 2.3-2 replace corresponding figures of the existing Technical Specifications.

3) The attached figures 3.5-2B, 3.5-2D and 3.5-2F be inserted where there are pages reserved for the figures in the existing Technical Specifications.

Reasons for Change

Technical Specification Change Request No. 70 (January 9, 1978) was submitted based on a Cycle 3 burnup of 270 ± 10 EFPD. Amendment A to Technical Specification Change Request No. 70 (April 03, 1978) was submitted as a conservative set of Technical Specifications based on a 315 EFPD Cycle 3 and only specified operating parameters to 125 ± 5 EFPD of Cycle 4 operation. This Change Request includes the changes resulting from the actual 287.1 EFPD Cycle 3 and refinement of the conservative Technical Specification submitted as Amendment A to Technical Specification Change Request No. 70, to extend Cycle 4 operation to 265 ± 15 EFPD, as well as those changes to operating limits necessary to accommodate the discrepancies between measured and predicted radial and total peaking noted during Cycle 4 startup.

In Metropolitan Edison Company letter (GQL 0743) of April 20, 1978, a commitment was made to reanalyze the reactor coolant system peak pressure following a feedwater line break and to submit the results with the refined cycle 4 Technical Specifications. Therefore, this submittal also includes the revised reactor coolant system high pressure trip setpoint based on the reanalysis performed by Babcock & Wilcox. The revised setpoint is acceptable for cycle 4 and subsequent cycle operation.

Safety Evaluation Justifying Change

The Proposed Technical Specifications and the Cycle 4 Reload Report (Revised June 1978) are structured relative to the current Technical Specifications and operating history to date. The revisions made to the Cycle 4 Reload Report (Revision 1, January 1978) include the changes based on, 1) Amendment No. 39, April 27, 1978, (in response to Tech Spec Change Request No. 70, Amendment A, April 3, 1978); and 2) Amendment No. 40, May 19, 1978 (in response to Tech Spec

Change Request Nos. 79 and 80); 3) the reevaluation of Cycle 4 based on the actual EOC-3 burnup of 287.1 EFPD to determine the key parameters for a Cycle 4 burnup from 0 to 280 EFPDs; 4) the generic study over the small break LOCA spectrum; and 5) the high pressure trip setpoint reanalysis.

Centerline fuel melt margins and steady-state DNB margins to limiting linear heat rates were reviewed for impact on the Core Protection Safety Limits and Setpoints for Cycle 4. Centerline fuel melt margins versus reactor power imbalance are more restrictive than the current TMI-1 Technical Specifications due to the power distribution anomoly observed in the Cycle 4 Startup Tests, and as a result, the calculational nuclear uncertainty factors were increased to 11% for the radial peak and 13.50% for the total peak. Figure 2.1-2, "Core Protection Safety Limits, TMI-1" indicates a negative imbalance more restrictive than the current negative imbalance safety limits. Further refinement in the analysis to take into account the calculation/measurement adjustment, resulted in changes to Figure 2.3-2. Figure 2.3-2 and Figure 2.1-2, which provides the basis data for Figure 2.3-2, have been refined to account for the extension of Cycle 3 to 287.1 EFPDs and to incorporate the increased uncertainty factors. Both Figures are valid for Cycle 4 (0 to 280 EFPDs).

The DNB dependent points of the pressure temperature limit curves (variable low pressure trip) and the flux/flow trip setpoints are based upon design peaking rather than calculated peaking. The Cycle 4 margin between calculated nuclear pin peaks and the reference design peaks used for thermal-hydraulic analysis was found to increase due to the Cycle 3 extension. Therefore, all Technical Specification limits based on design peaking remain conservative for Cycle 4.

The thermal-hydraulic rod bow penalty and fuel temperature/pin pressure analyses were reviewed for applicability during Cycle 4 operation. Both the maximum assembly burnup (31094 MWD/MTU) and the hot assembly burnup (16,000 MWD/MTU) at EOC-4 (280 EFPDs) were below the assembly burnup assumed for the rod bow penalty (33000 MWD/MTU). Also, the calculated maximum pin burnup (35447 MWD/MTU) and the pin power history were bounded by the values used for the fuel temperature/pin pressure analysis.

The mechanical performance of the fuel was found to be acceptable relative to cladding stress, strain and creep collapse for Cycle 4. Therefore, all Technical Specification limits based on fuel integrity are valid for Cycle 4.

The two-hour thyroid dose was reanalyzed using the actual EOC-3 burnup of 287.1 EFPDs and a Cycle 4 of 280 EFPDs. The Cycle 4 doses remain only a very small fraction of 10 CFR 100 limits, and are acceptable for Cycle 4 operation.

The accident analyses were reviewed based on the refined Cycle 4 values, and the Cycle 4 parameters were compared to the FSAR/densification report analysis values (See Table 1). Table 1, in conjunction with Table 2, illustrates that for all accidents considered in the FSAR, the initial conditions defined by Cycle 4 parameters produce less severe transients than the initial conditions assumed in the FSAR analysis. Because the Cycle 4 transients are bounded by previously accepted analyses, no reanalyses of these events have been performed.

Section 3.5.2, "Control Rod Group and Power Distribution Limits, was reviewed for changes in shutdown margin, ejected rod worth and peaking margin to LOCA kw/ft limits for Cycle 4. The current Technical Specification Figures 3.5-2A, 3.5-2C and 3.5-2E are more restrictive than the like figures developed from the most recent analysis. Therefore, in order to expedite the licensing review process, the current, more restrictive, limits for Figures 3.5-2A, 3.5-2C and 3.5-2E have been retained for the first 130 EFPDs of Cycle 4.

The remaining Cycle 4 period (130 to 280 EFPDs) with respect to rod position limits and the LOCA dependent power imbalance envelope, will be governed by the attached Figures 3.5-2B, 3.5-2D and 3.5-2F. In general, the rod position limits have a slightly more restrictive shutdown margin limit curve and permissible operating region than the figures previously submitted. The high power operating region of Figures 3.5-2B and 3.5-2D has been retained by imposing a more restrictive power imbalance envelope (Figure 3.5-2F), and APSR position limit curve (Figure 3.5-2H, Amendment No. 40, May 19, 1978). Revised calculations indicate that Cycle 4's minimum shutdown margin at 280 EFPDs was 2.19% Δk/k, well above the 1% Δk/k requirements. The maximum ejected rod worth was 0.35% Δk/k at Hot Zero Power and 0.25% Δk/k at Hot Full Power, well below the limits of 1.0% Δk/k and 0.65%Δk/k, respectively.

Figure 3.5-2D for 2 and 3 pump operation was developed by power scaling the limiting boundary for the "Permissible Region" of Figure 3.5-2B in the recognition of the ordinate of Figure 3.5-2D. The shutdown margin limit of Figure 3.5-2B was not scaled, adding additional conservatism to that limit in Figure 3.5-2D.

Based on the above safety evaluation review, it can be concluded that the revised Technical Specification changes to those previously submitted for TMI-1's Cycle 4 support a full power Cycle 4 operation for 0 to 280 EFPDs without endangering the health and safety of the public.

Pages 2-4 and 2-9 and Figure 2.3-1, have been changed to reduce the RC System high pressure trip setpoint to 2390 psig so that in the event of a feedwater line break, the peak RCS pressure will not exceed 2750 psig. Attached to this safety evaluation are the results of the reanalysis which justify the 2390 psig setpoint for Cycle 4 and subsequent cycles. TMI-1 reduced the setpoint to 2390 psig prior to obtaining greater than 40% FP during Cycle 4 startup. This issue does not involve a safety concern and is resolved with this Technical Specification submittal.

Page 3-34a is being submitted to correct a typographical error in the quadrant tilt value as determined by the out-of-core detector system. The value should be +22.96% rather than the current Tech. Spec. value of +22.92%.

These Technical Specification changes discussed above and the Cycle 4 Reload Report (Revised June 1978) support a full power 280 EFPD Cycle 4, and do not create a threat to the health and safety of the public.

Classification of Amendment (10 CFR 170.22)

This amendment to authorize operation in Cycle 4 to 265 ± 15 EFPD is the result

of refinements to the original Cycle 4 Technical Specification Submittal (Technical Specification Change Request No. 70, January 9, 1978, as amended April 3, 1978) and do not involve a significant hazards consideration. Therefore, the Licensee has determined that this is a Class III amendment.

The reactor coolant system high pressure trip setpoint submittal included herewith is a refinement of the analysis submitted on April 17, 1978 (GQL 0669), and is in response to our commitment of April 20, 1978 (GQL 0743). The License has determined that the resulting changes to the Technical Specifications are not subject to the scheduled amendment fees.

Therefore, the proper remittance for this Change Request is $4,000.00.

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TABLE I

COMPARISON OF KEY ACCIDENT AND TRANSIENT PARAMETERS FOR CYCLE 4*

Parameter	FSAR and densif'n report value	Predicted Value
Doppler coeff (BOC), $\Delta k/k/^{\circ}F$	-1.17×10^{-5}	-1.48×10^{-5}
Doppler coeff (EOC), $\Delta k/k/^{\circ}F$	-1.33×10^{-5}	-1.60×10^{-5}
Moderator coeff (BOC), $\Delta k/k/^{\circ}F$	$+0.5 \times 10^{-4}$	-0.71×10^{-4}
Moderator coeff (EOC), $\Delta k/k/^{\circ}F$	-3.0×10^{-4}	-2.53×10^{-4}
All rod group worth, % $\Delta k/k$	10.0	8.62
Initial boron conc. (HFP) ppm	1200	1045
Boron reactivity worth ($70^{\circ}F$), ppm/1% $\Delta k/k$	75	73
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.25
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20

* Based on a 287 EFPD Cycle 3 and 280 EFPD Cycle 4.

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TABLE 2

COMPARISON OF CYCLE 4* PARAMETERS TO THE FSAR

Transient	Key Parameters[1]	Safety Margin Increases with	Cycle 4 margin greater than FSAR margin?
Moderator dilution accident	Initial boron concentration (BOC).	lower concentration	yes
	Boron reactivity worth	larger value	no[2]
	Moderator coefficient (BOC)	more negative value	yes
Cold Water Accident	Moderator coefficient (EOC)	less negative	yes
	Doppler coefficient (EOC)	more negative	yes
Loss of Coolant Flow	Doppler coefficient (BOC)	more negative	yes
	Moderator coefficient (BOC)	more negative	yes
Steam Line Failure	Moderator coefficient (EOC)	less negative	yes
Rod Ejection Accident	Doppler coefficient (BOC)	more negative	yes
	Moderator coefficient (BOC)	more negative	yes
	Ejected rod worth	smaller worth	yes

* Based on a 287 EFPD Cycle 3 and 280 EFPD Cycle 4

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Transient	Key Parameters[1]	Safety Margin Increases with	Cycle 4 Margin greater than FSAR Margin?
Dropped Control Rod	Dropped Rod Worth	Smaller worth	Yes
	Moderator Coefficient (EOC)	Less Negative	Yes
	Doppler Coefficient (EOC)	Less Negative	No [3]
Rod Withdrawal Accident	Doppler Coefficient (BOC)	More Negative	Yes
	Moderator Coefficient (BOC)	More Negative	Yes

NOTES:

(1) Certain key parameters do not normally vary from cycle to cycle. These have been excluded in this table: RC pump flow; RC pump flow characteristics, and design radial-local and axial peaking factors.

(2) The quotient of initial boron concentration and boron worth yields a lower reactivity addition (higher safety margin) than the FSAR reference analysis.

(3) Although a more negative EOC Doppler is not conservative for the dropped rod accident, the transient results are still conservative with respect to the FSAR analysis because of the smaller Cycle 4 dropped rod worth and less negative moderator coefficient.

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May 1, 1978

PEAK RC PRESSURE FOLLOWING FEEDWATER LINE BREAK
TMI-1 Cycle 4

The Peak Reactor Coolant System Pressure following a Feedwater Line Break has been recalculated for Cycle 4 and subsequent cycles based on the following assumptions.

1. Safety valve relief rate - 156#/s @ 2500 psig (combined relief rate)
2. Rosemont pressure transmitter error of 45 psi in a degraded environment
3. Trip string delay time of 450 ms
4. High pressure trip setpoint 2405 (curve 1); 2395 (curve 2); and 2390 (curve 3).

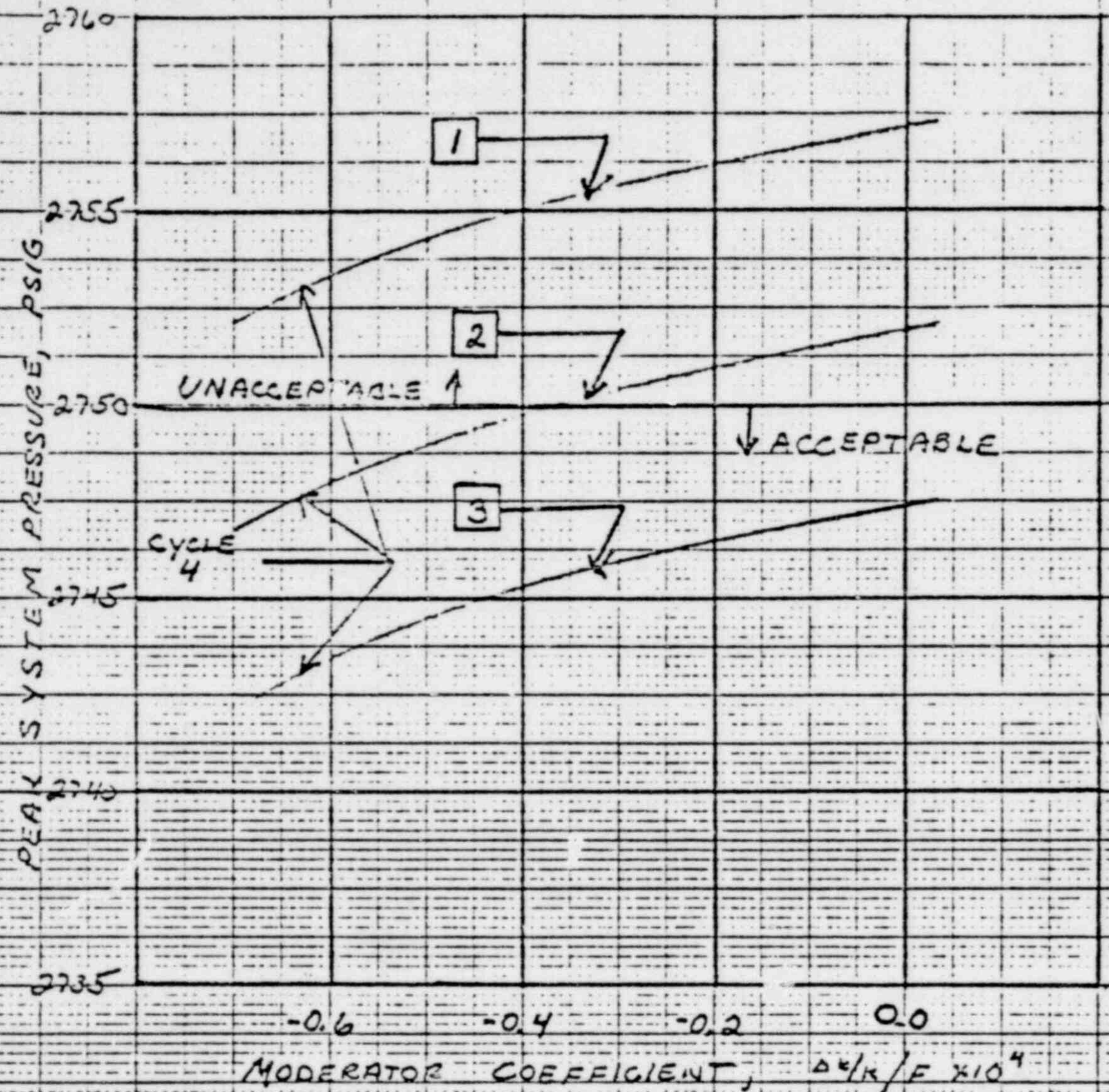
The results in the attached figure show that unacceptable peak system pressures result with the use of the 2405 psig trip setpoint for Cycle 4 and subsequent cycles. This conclusion supercedes that submitted to the NRC April 20, 1978, GQL 0743, where too great a safety valve relief flowrate was used.

To produce acceptable peak system pressures following a feedwater line break for Cycle 4, it will be necessary to lower the high RC pressure trip setpoint to 2395 psig, keeping the safety valve setpoint at 2500 psig. This change will result in acceptable peak pressures for a reasonable range of moderator reactivity coefficients. The attached figure (curve 3) indicates that the peak system pressure is less than 2750 psig for moderator coefficients more negative than -0.35×10^{-4} ΔK/K/F. This acceptable range includes the present Cycle 4 moderator coefficient of -0.63×10^{-4} ΔK/K/F.

The analysis was performed also for a 2390 psig setpoint for reactor trip. Peak system pressure remains below the 2750 psig limit for all negative moderator coefficients. It can be concluded that a high pressure trip setpoint of 2390 psig would suffice for Cycle 4 and all subsequent cycles since a Tech Spec exists to assure that the moderator coefficient will always be negative.

The analysis performed is insensitive to Doppler coefficient since the fuel temperature changes (increases) by only a few degrees prior to development of the peak system pressure. This analysis is based on a Doppler coefficient of -1.49×10^{-4} ΔK/K/F which should be conservative with respect to subsequent cycles.

The results of this analysis indicate that a setpoint of 2395 psig is necessary to give acceptable results for Cycle 4. A setpoint of 2390 psi will produce acceptable results for Cycle 4 and subsequent cycles. The 239 psig high RC pressure trip setpoint has been incorporated in the TMI-1 cycle 4 refined Tech Specs in order to have a fixed setpoint for subsequent cycles.



PEAK SYSTEM PRESSURE FOLLOWING
A FEED WATER LINE BREAK VS.
MODERATOR COEFFICIENT

SV RELIEF RATE: 156 #/S @ 2500 PSIG
PRESSURE TRANS. ERROR: 45 PSI
HIGH PRESSURE TRIP:

1 - 2405 3 - 2390
2 - 2395

DELAY 450 MS.

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atomsphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established. (2) The maximum settings for the reactor high pressure trip (2390 psig) and the pressurizer code safety valves (2500 psig) (3) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2255 psig. (4)

References

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 41

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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)	108 times flow minue 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.	2390	2390	2390	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 setpoint side of the protection system bistable comparators.

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

3. The operational imbalance limits (Figure 3.5-2E, and 3.5-2F) 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 +26.75% determined using the full incore detector system (FIT), or +15.21% determined using the minimum incore detector system (MIT) if the FIT is not available, or +22.96% 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. Diagnositc 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.

POOR ORIGINAL

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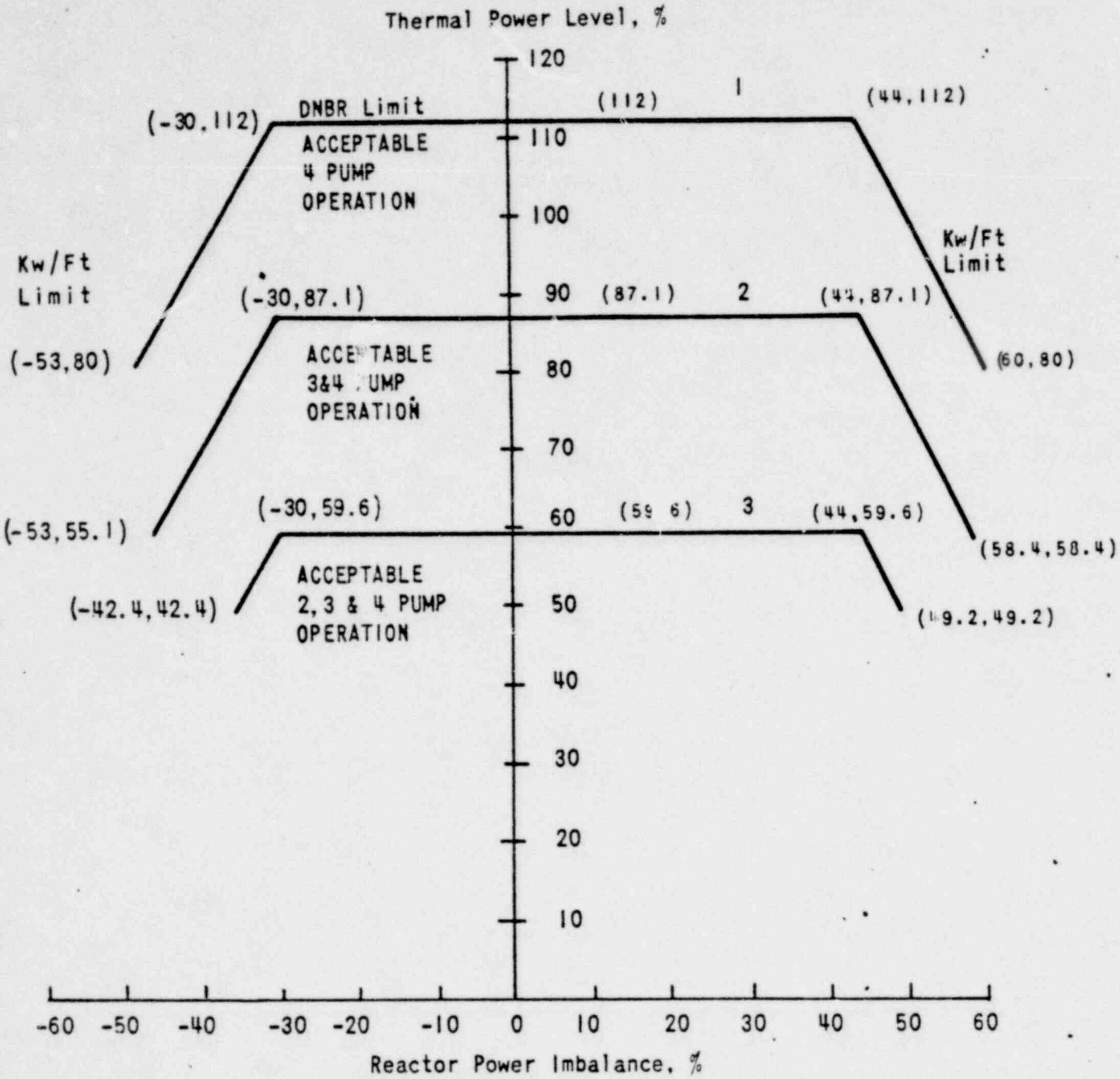

Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

CORE PROTECTION SAFETY LIMITS

TMI-1, Cycle 4

Figure 2.1-2

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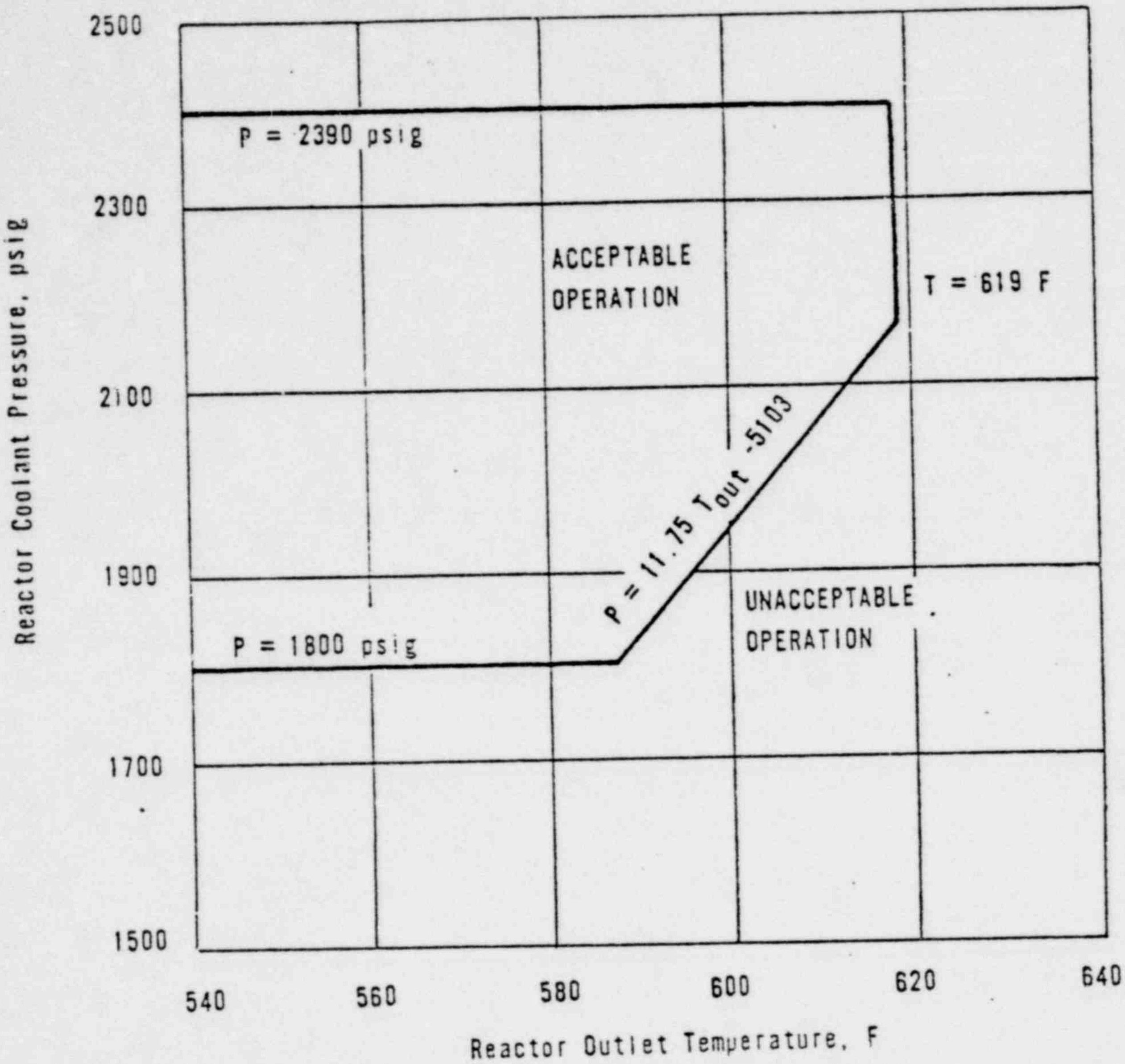

TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1

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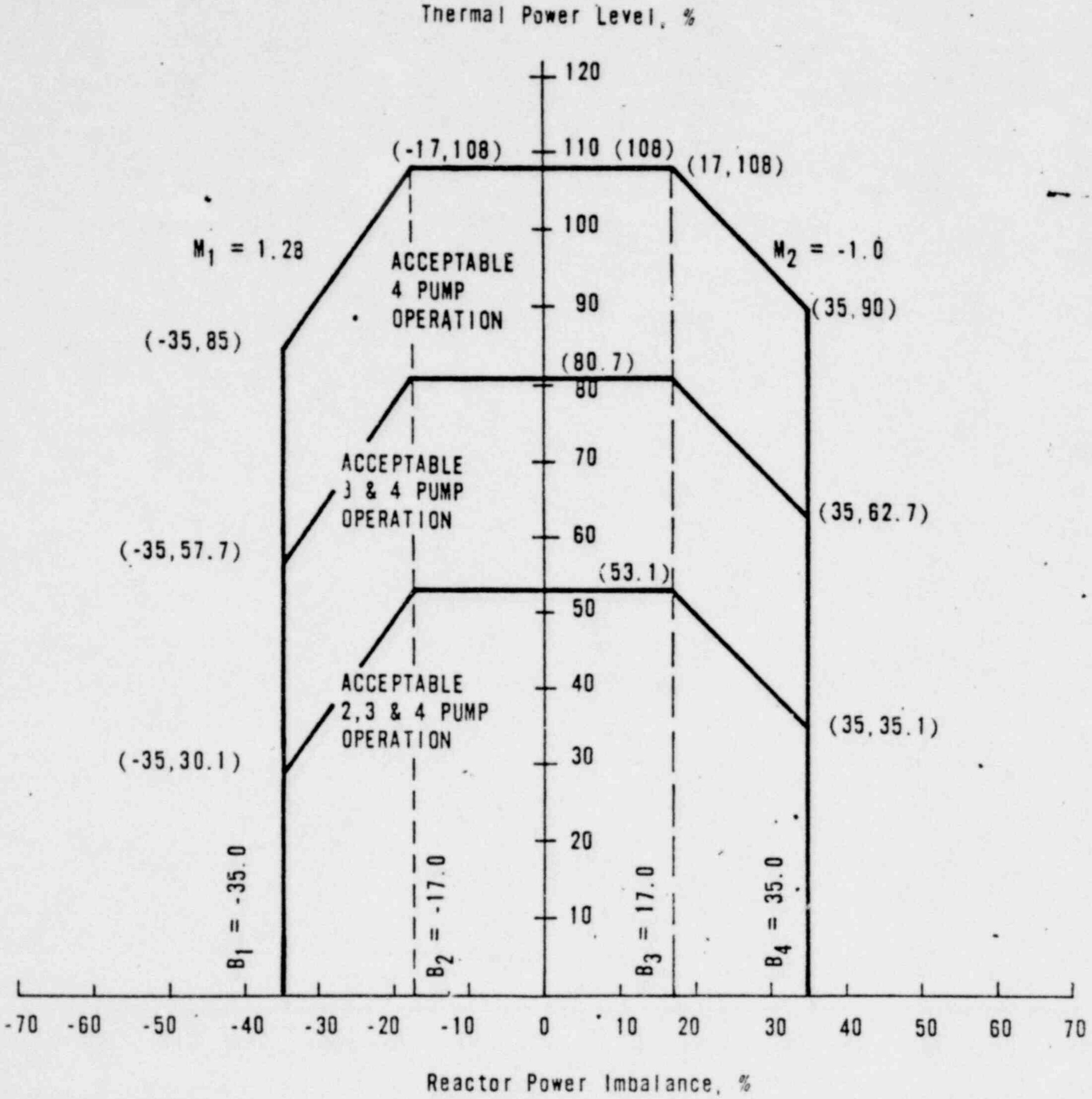

PROTECTION SYSTEM MAXIMUM ALLOWABLE
SETPOINTS FOR REACTOR POWER IMBALANCE
TMI-1, Cycle 4

Figure 2.3-2

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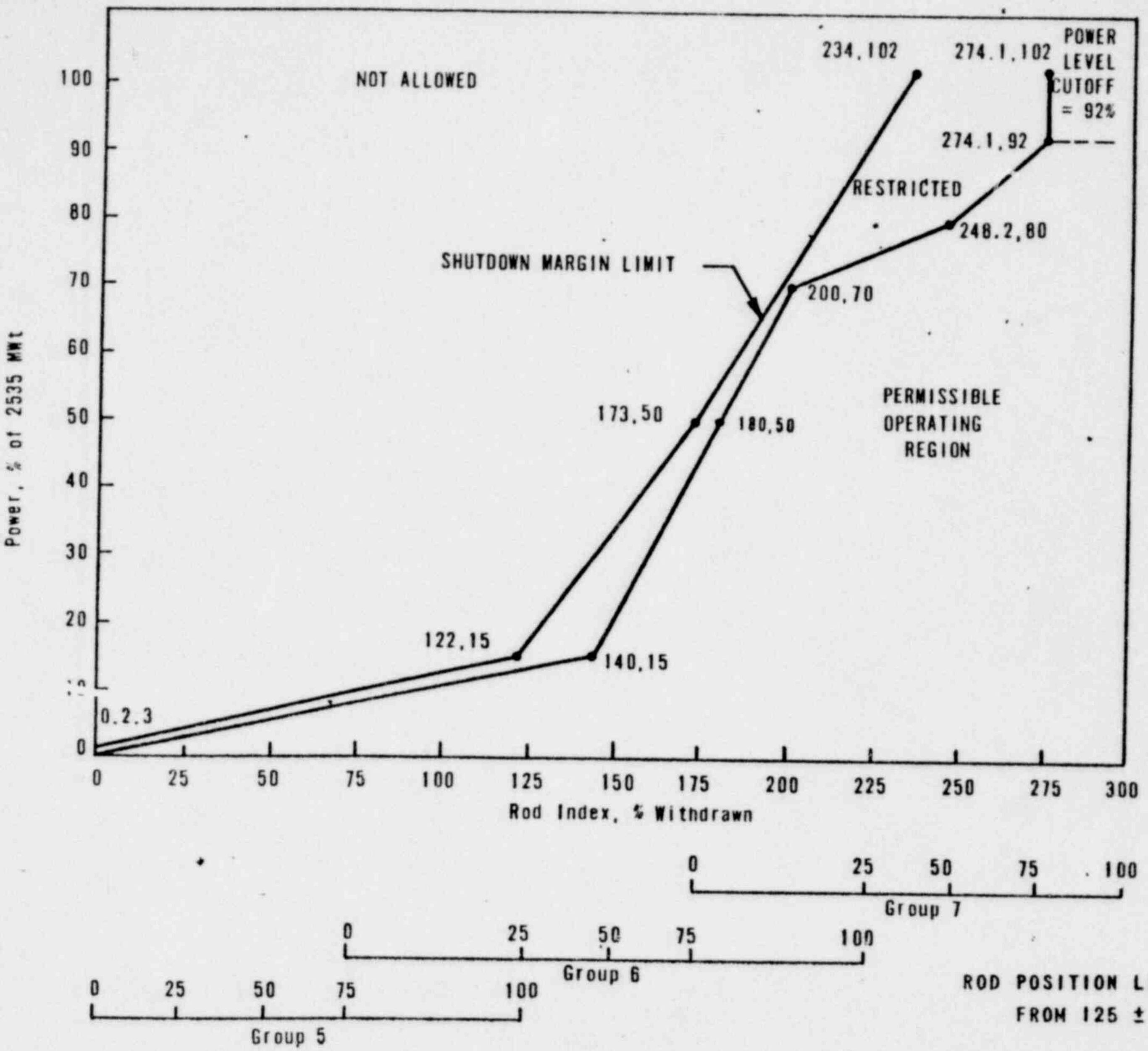

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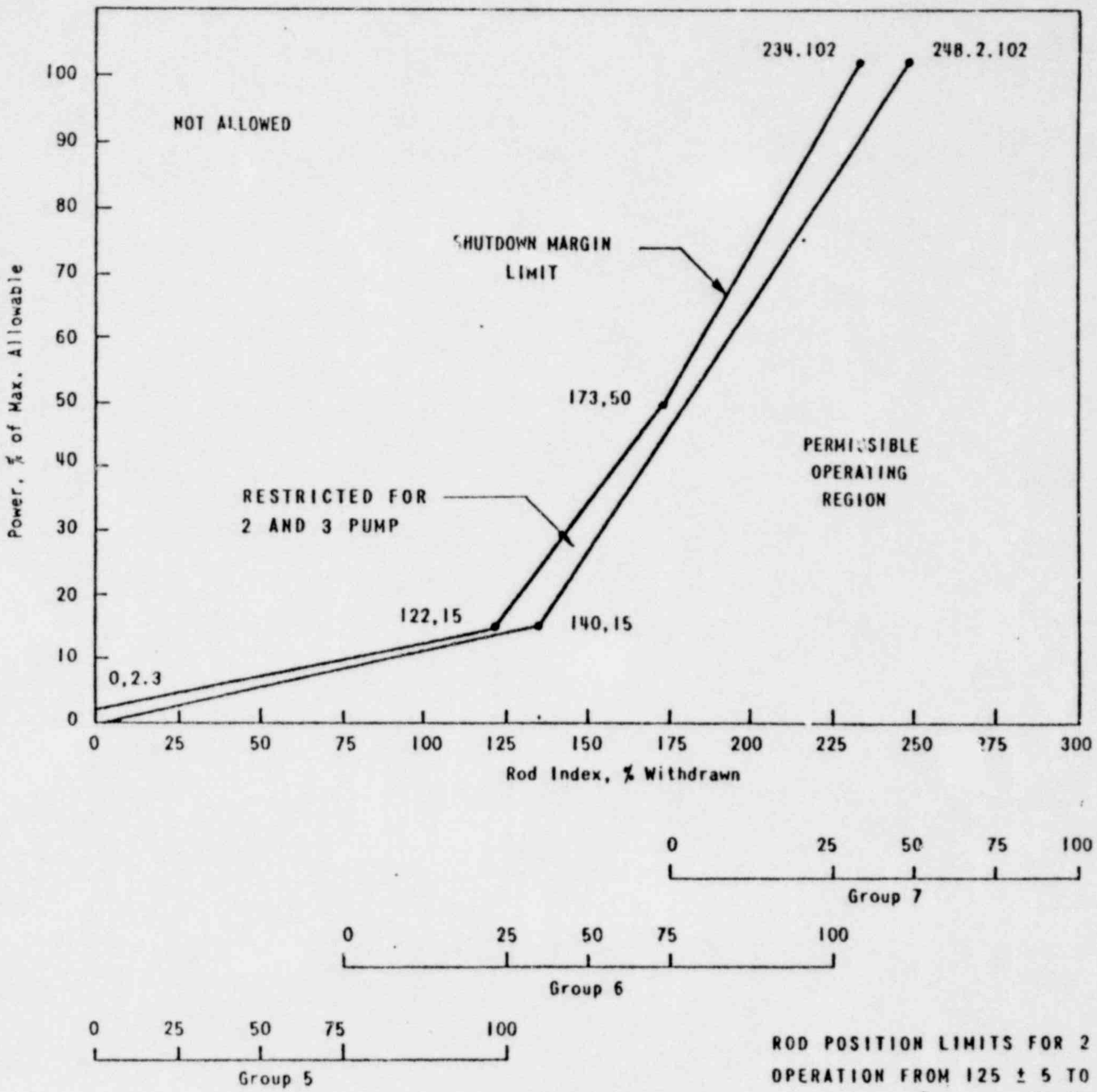
ROD POSITION LIMITS FOR 4 PUMP OPERATION
FROM 125 ± 5 EFPD TO 265 ± 15 EFPD
TMI-1, Cycle 4
Figure 3.5-2B



ROD POSITION LIMITS FOR 2 & 3 PUMP
OPERATION FROM 125 ± 5 TO 265 ± 15 EFPD
TMI-1, Cycle 4

Figure 3.5-2D

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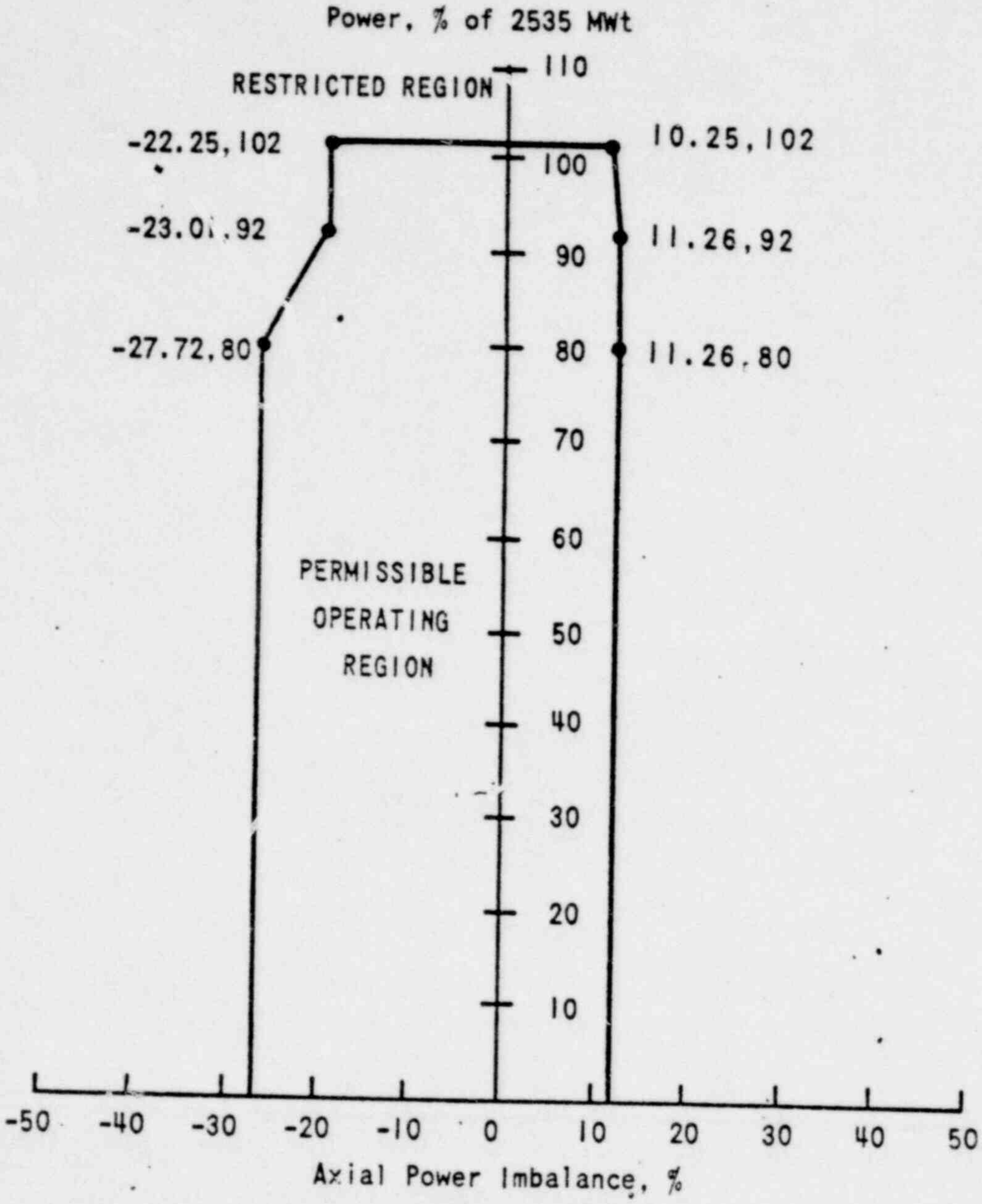

POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 125 ± 5 TO 265 ± 15 EFPD
TMI-1, Cycle 4

Figure 3.5-2F

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