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. 9

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.

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

Sworn and subscribed to me this 15th day of April, 1975

Richard I Ruth
Notary Public

My Commission Expires September 13, 1976

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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. 9 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, dated April 15, 1975, and filed with the U. S. Nuclear Regulatory Commission on April 15th, 1975, has this 15th day of April 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, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, Pennsylvania 17057

Mr. Charles P. Hoy, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Court House
P.O. Box 1295
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

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THREE MILE ISLAND NUCLEAR STATION UNIT I
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289

TECHNICAL SPECIFICATION CHANGE REQUEST NO. 9

It is requested that the present TMI-1 Technical Specification pages 3-6, 3-7, 3-35, 3-35a, 3-36, and figures 3.5-2A, 2B, 2C, 2D and 2E be replaced with the respectively designated pages of the attached Appendix 1; and that a new figure 3.5-2F (previously designated figure 3.5-2E, and included in Appendix 1), be added to the TMI-1 Technical Specifications. Further, it should be noted that this Technical Specification Change Request, if approved, will serve to: (a) revise the present Power vs. Rod Withdrawal limits of the TMI-1 Technical Specifications Figures3.5-2A, 2B, and 2C, and (b) revise the related Technical Specification text so as to be consistent with the revised figures.*

*Note: Technical Specification Change Request No. 4, previously submitted by the Licensee, requested that changes be made to some of the same enclosed pages, and these previously requested changes are noted by "C. R. 4".

Reason for Technical Specification Change Request No. 9

The present Power vs. Rod Withdrawal limits as provided by the present Technical Specification figures 3.5-2A, 2B, and 2C, are not adequate to ensure not exceeding the Technical Specification single rod worth limits of TMI-1 Technical Specification 3.5.2.3., for plant operations subsequent to the next control rod interchange (note: refer to Licensee Non Routine 30 Day Report 75-04, attached as Appendix II, for additional background in this subject area). The reason for this Change Request, therefore, is to obtain Power vs. Rod Withdrawal limits which are adequate to ensure not exceeding the rod worth limits of Specification 3.5.2.3 for plant operations in the time period from the first control rod interchange until completion of the first refueling.

Safety Analysis Justifying Change Request No. 9

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It is the Licensee's position that the limits provided by the enclosed, revised Power vs. Rod Withdrawal Figures are adequate to ensure not exceeding the single rod worth limits of Specification 3.5.2.3, for plant operations subsequent to the control rod interchange until completion of the first refueling, in that

a. the reason for the present figures being adequate only until sometime between the control rod interchange and the first refueling is not because of incorrect models or analyses having been utilized to derive the figures, but because of the analyses not having been conducted at the worst core burn-up condition (i.e. that burn-up condition for which single rod worths are highest), and

b. the revised figures were derived by conducting the same analyses with the same correct, previously approved models for the worst core burn-up condition.

Further, it should be noted that actual core rod worth measurements will be obtained, in accordance with the March 31, 1975, request of the Commiss[illegible]n's Mr. George Lear, to provide further verification that the revised li[illegible]ts are in f[illegible]equate.

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TMI-I TECHNICAL SPECIFICATION

CHANGE REQUEST NO. 9

CHANGE PAGES

Appendix (1)

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3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from r[illegible]ctivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525 F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10 F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least one percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

3.1.3.5 Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality with the following exceptions:

(a) Inoperable rod per 3.5.2.2.

(b) Physics testing per 3.1.9.

(c) Shutdown margin may not be reduc d below 1% $\Delta Ak/k$ per 3.5.2.1.

(d) Exercising rods per -.1.2.

Following safety rod withdrawal, the regulating rods shall be positioned within their position limits as defined by specification 3.5.2.5 prior to deboration.

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Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525 F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less that 525 F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT +10 F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limit ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.-

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3.5.2.5 Control rod positions:

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

b. Except for physics tests or exercising control rods, the control rod insertion/withdrawal limits are specified on Figures 3.5-2A (for up to the control rod interchange), Figure 3.5-2B (from control rod interchange up to 440 full power days of operation), Figure 3.5-2C (for after 440 full power days of operation) for four pump operation, and Figure 3.5-2D 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 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. 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 Figure 3.5-2E. If the imbalance is not within the envelope defined by Figure 3.5-2E, 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 Figure 3.5-2E is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2F) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. C.R.4
Operation outside of the power imbalance envelope alone does not consitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. C.R.4
The power imbalance envelope represents the boundary of operation

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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, and 3.5-2D and if a 4 percent quadrant power tilt exists. Additional conservatism is introducted by application of: C.R.4

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors.

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The 30 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 minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming, the highest worth control rod remains in the full out position(1).

Inserted rod groups during power operation will not contain single rod worths greater than 0.65 percent $\Delta k/k$. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident(2). Single inserted control rod worth of 1.0 percent $\Delta k/k$ at beginning of life, hot, zero power would result in lower transient peak thermal power, and therefore, less severe environmental consequences as a 0.65 percent $\Delta k/k$ 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 3.5.2.4 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:

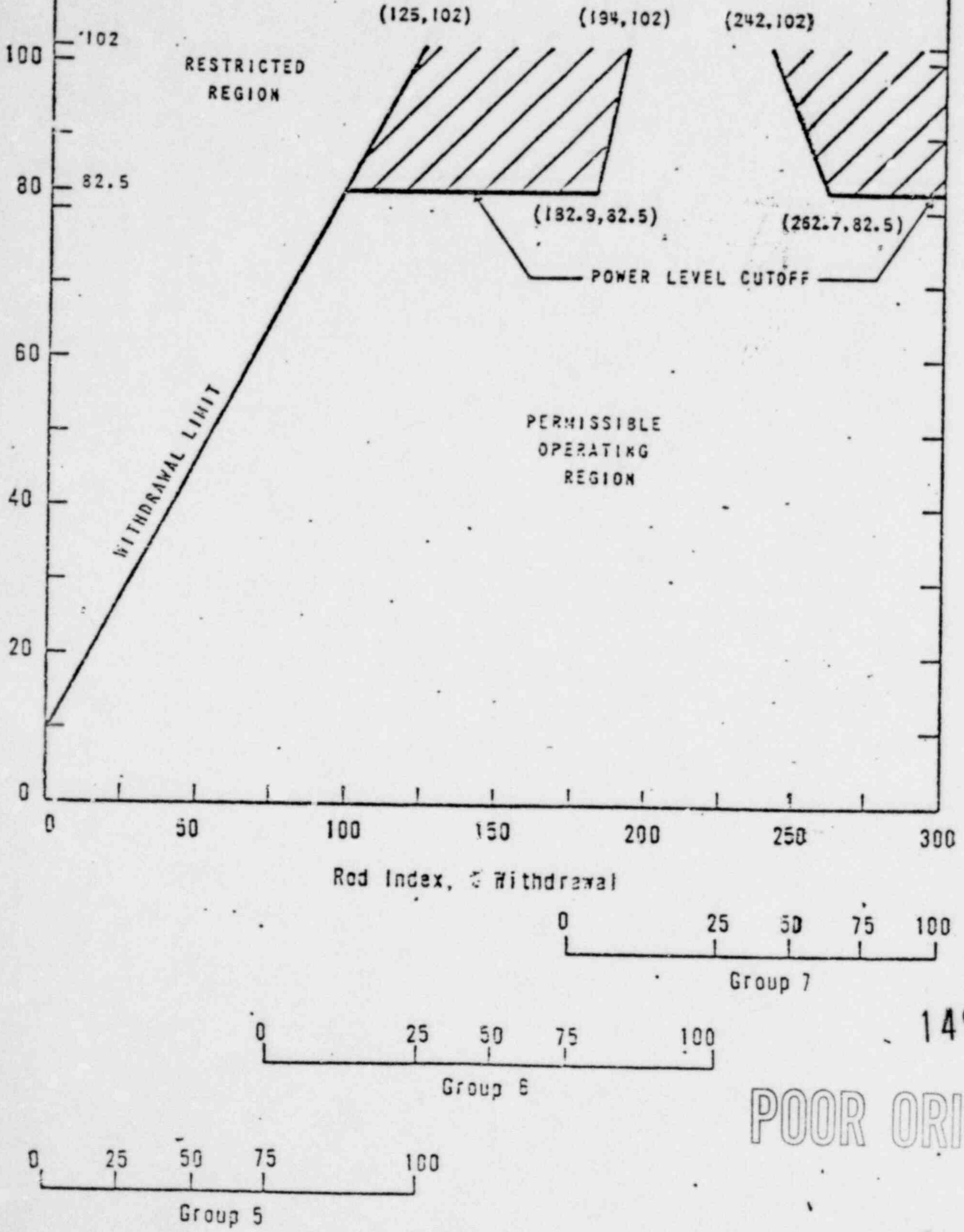
Test Power	Trip Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.5%

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

1. Rod index is the percentage [illegible] of the withdrawal of the operating groups.
2. Restrictions on withdrawal [illegible] hashed areas) are modified after th[illegible] [illegible]trol rod interchange (See Figure 3. [illegible] 23)



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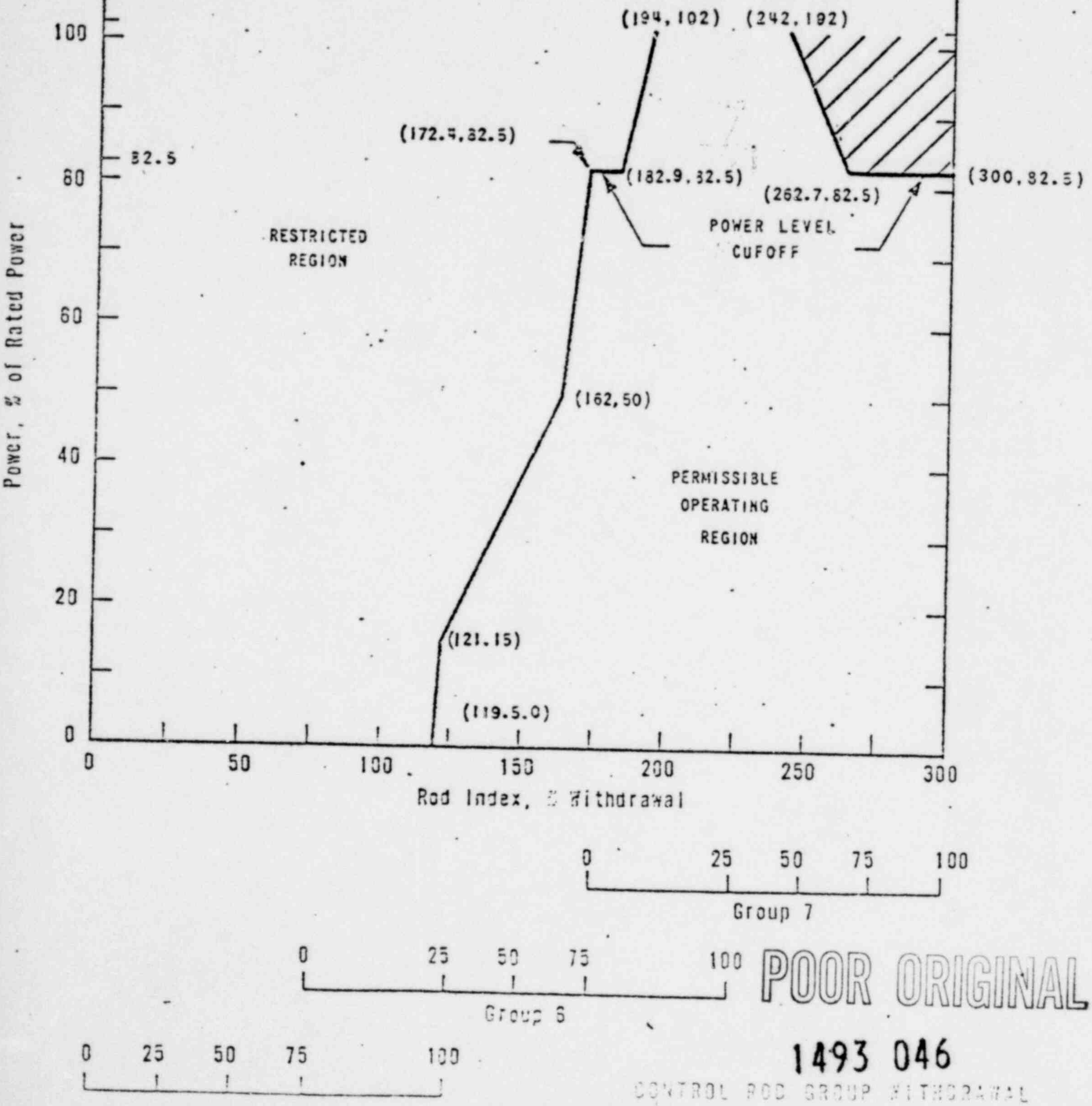
CONTROL ROD GROUP WITHDRAWAL LIMITS
FOR 4 PUMP OPERATION UNIT 1
Figure 3.5-2A

1. Rod index is the percentage sum of the withdrawal of the operating groups.

2. The additional restrictions on withdrawal (hashed areas) are in effect after the control rod interchange. The restrictions on withdrawal are further modified after 440 full power days of operation (See Figure 3.5-2C



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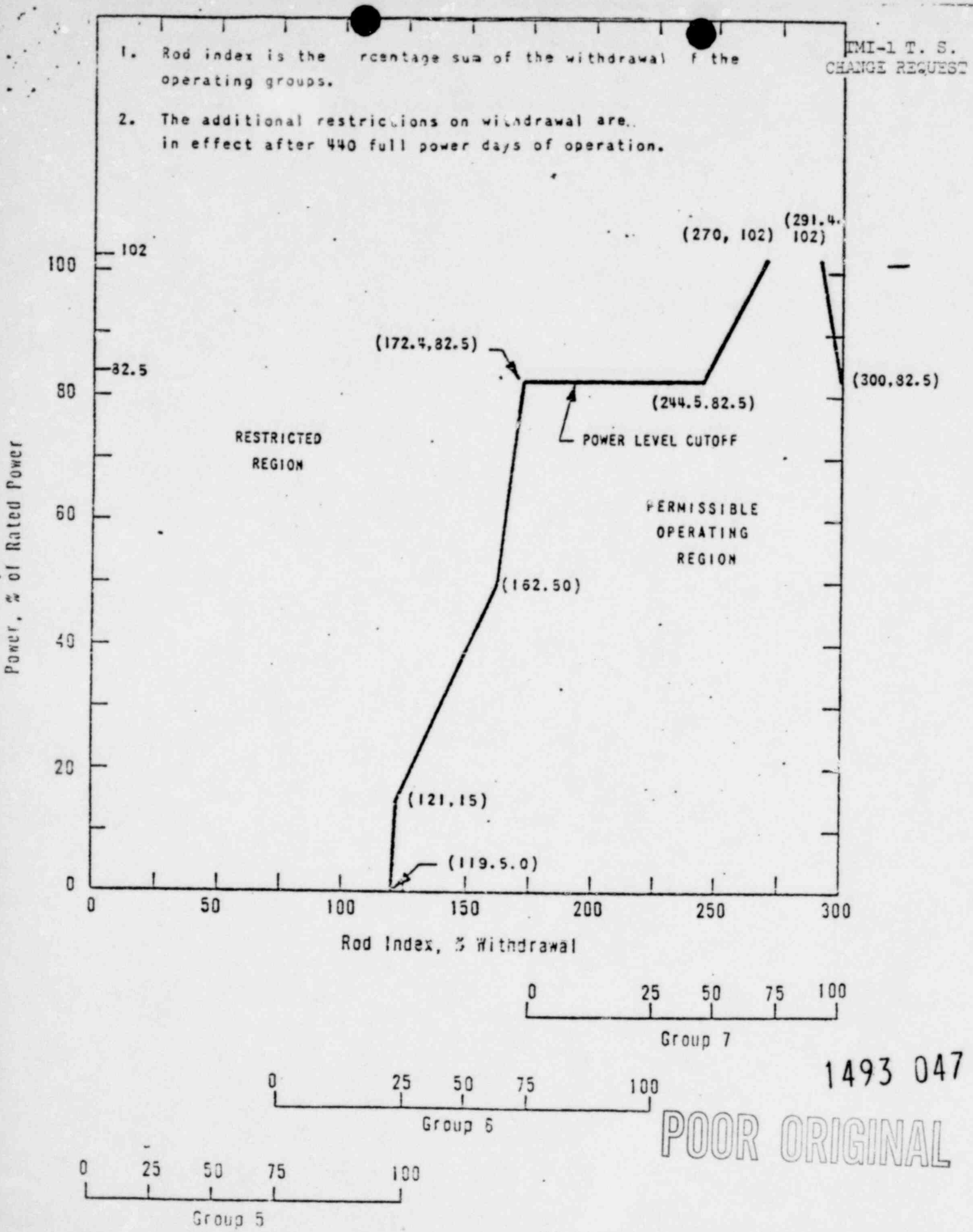
CONTROL ROD GROUP WITHDRAWAL
LIMITS FOR 4 PUMP OPERATION UNIT I

Figure 3.5-2B



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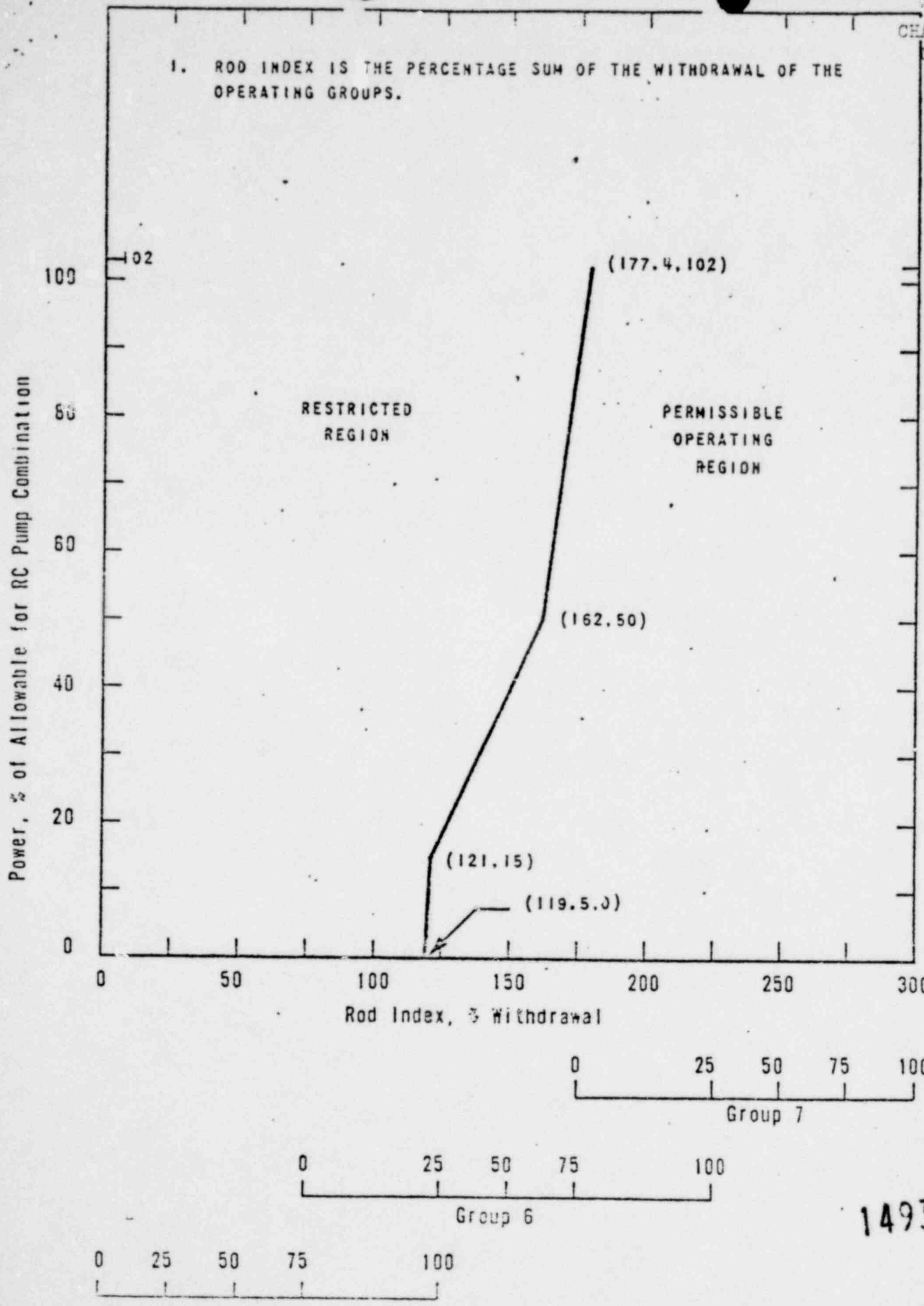
CONTROL ROD GROUP WITHDRAWAL LIMITS
FOR 4 PUMP OPERATION UNIT I

Figure 3.5-2C



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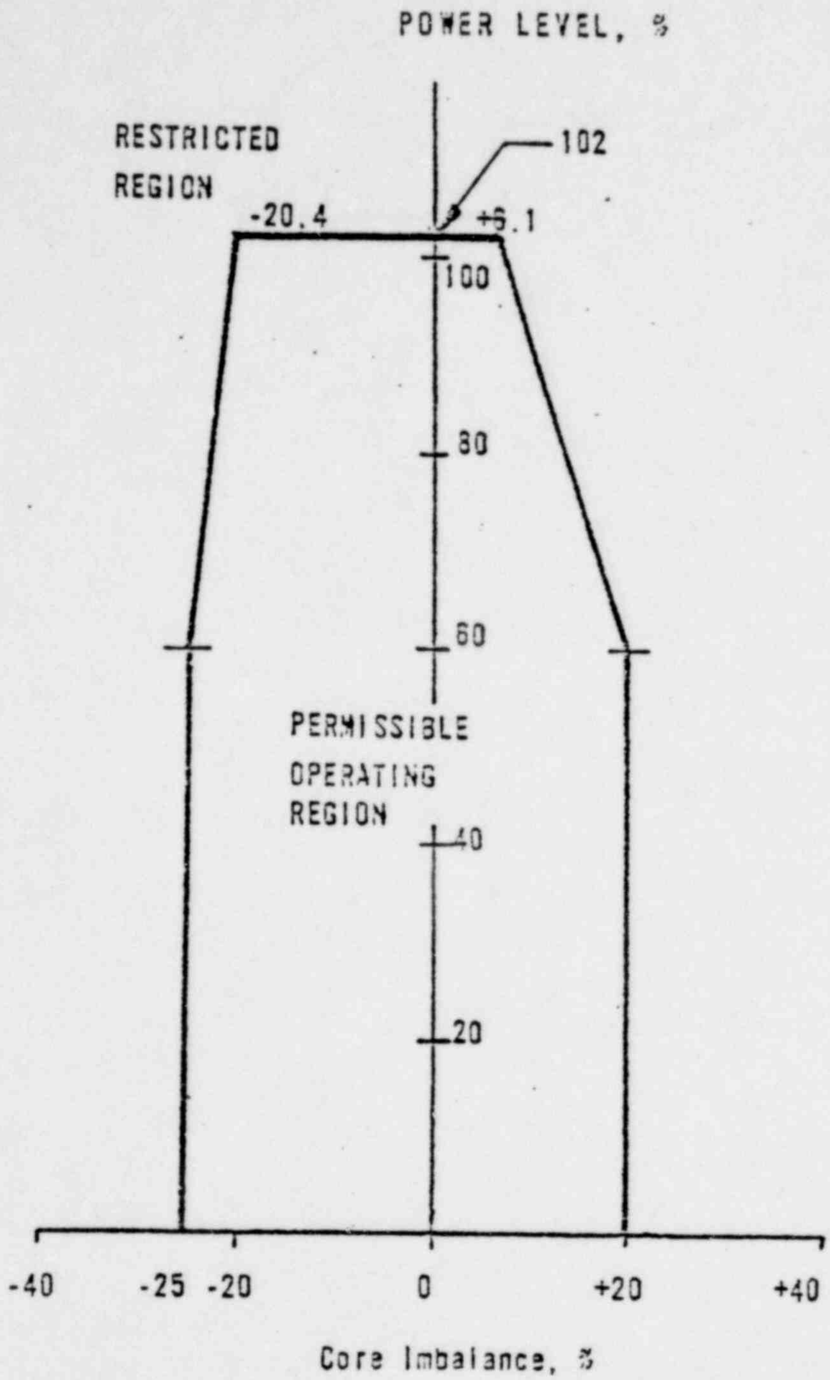
CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 3 AND 2 PUMP OPERATION UNIT I

Figure 3.5-2D



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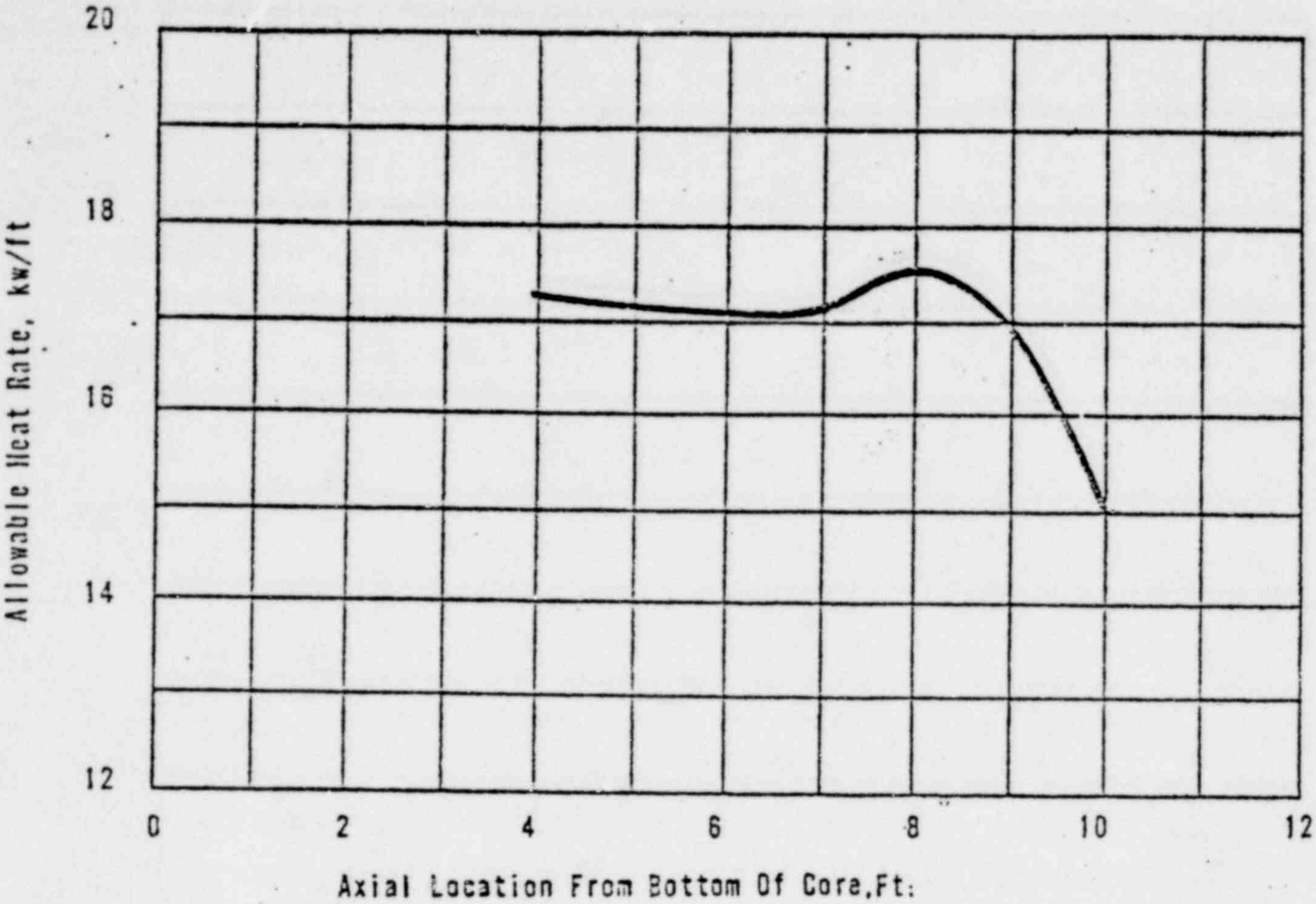
OPERATIONAL POWER IMBALANCE ENVELOPE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2 E



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LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2F

TMI-1 TECHNICAL SPECIFICATION

CHANGE REQUEST No. 9

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Appendix (2)