



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

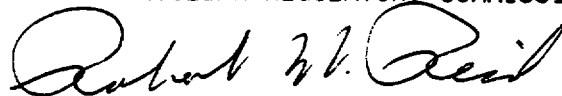
Amendment No. 45
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company (the licensees), dated June 23, 1978, as supplemented August 7, 1978, complies 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 application, 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. 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, Facility Operating License No. DPR-50 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:
 - A. Revise paragraph 2.c.(2) to read as follows:
 - (2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. Metropolitan Edison Company shall operate the facility in accordance with the Technical Specifications.
 - B. Delete paragraph 2.c.(3) which was added by Amendment No. 40, dated May 19, 1978.
3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO.45

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise the Appendix A Technical Specifications as follows:

Remove Pages

vi and vii

Figure 2.1-2

2-4

2-7 thru 2-9

Figures 2.3-1 and

2.3-2

3-16

3-34a

Figure 3.5-2H

Insert Pages

vi and vii

Figure 2.1-2

2-4

2-7 thru 2-9

Figures 2.3-1 and

2.3-2

3-16

3-34a

Figures 3.5-2B, 3.5-2D, 3.5-2F

and 3.5-2H

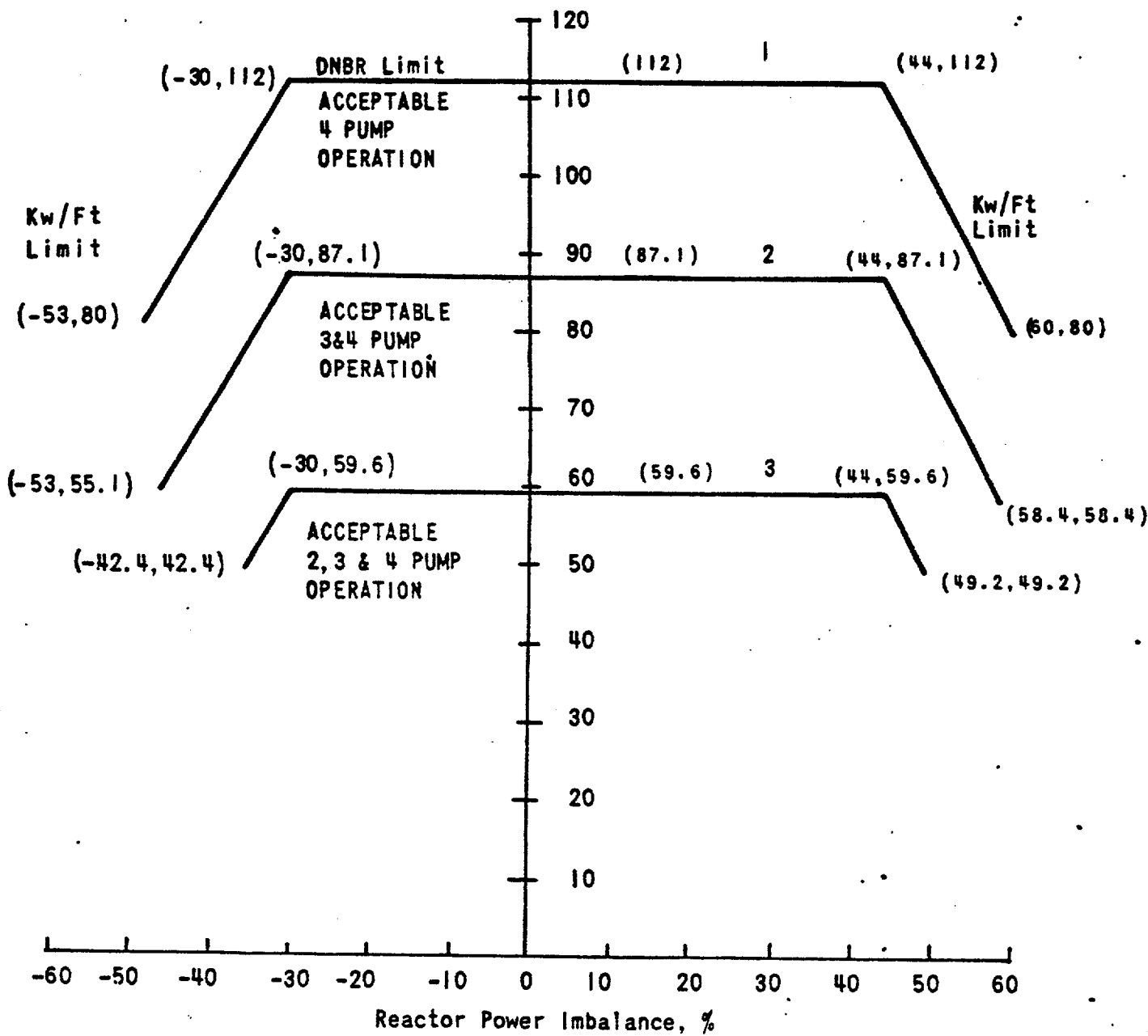
The changed areas on the revised pages are shown by marginal lines.

LIST OF FIGURES

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2.1-1	Core Protection Safety Limit
2.1-2	Core Protection Safety Limits
2.1-3	Core Protection Safety Basis
2.3-1	Protection System Maximum Allowable Set Points
2.3-2	Protection System Maximum Allowable Set Points
3.1-1	Reactor Coolant System Heatup Limitations
3.1-2	Reactor Coolant System Cooldown Limitations
3.1-3	Limiting Pressure Vs. Temperature Curve for 100 STD cc/Liter H ₂ O
3.5-1	Incore Instrumentation Specification Axial Imbalance Indication
3.5-2	Incore Instrumentation Specification Radial Flux Tilt Indication
3.5-2A	Rod Position Limits for 4 Pump Operation Appli- cable During the Period from 0 to 125 ± EFPD; Cycle 4
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3.5-2C	Rod Position Limits for 2 and 3 Pump Operation Applicable During the Period from 0 to 125 ± 5 EFPD; Cycle 4
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3.5-2G	LOCA Limited Maximum Allowable Linear Heat Rate
3.5-2H	APSR Position Limits for Operation from 0 to 265 ± 15 EFPD; Cycle 4
3.5-2I	Deleted
3.5-2J	Deleted
3.5-2K	Deleted
3.5-2L	Deleted
3.5-2M	Deleted
3.5-2N	Deleted
3.5-3	Incore Instrumentation Specification
4.2-1	Equipment and Piping Requiring Inservice Inspection in Accordance with Section XI of the ASME Code
4.4-1	Ring Girder Surveillance
4.4-2	Ring Girder Surveillance Crack Pattern Chart
4.4-3	Ring Girder Surveillance Crack Pattern Chart
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4.4-5	Ring Girder Surveillance Crack Pattern Chart
6-1	Organization Chart

Thermal Power Level, %



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

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 atmosphere. 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. (2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 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 4-1

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.⁽⁶⁾ Due to calibration and instrument errors, the safety analysis assumed a 45 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 - 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 temperature 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 11F. 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 safe 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

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 ≤ 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 ≤ 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
- (6) Technical Specification Change Request No. 31, January 16, 1976, and Technical Specification Change Request No. 84, June 23, 1978.

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 - 29%)	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.	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)

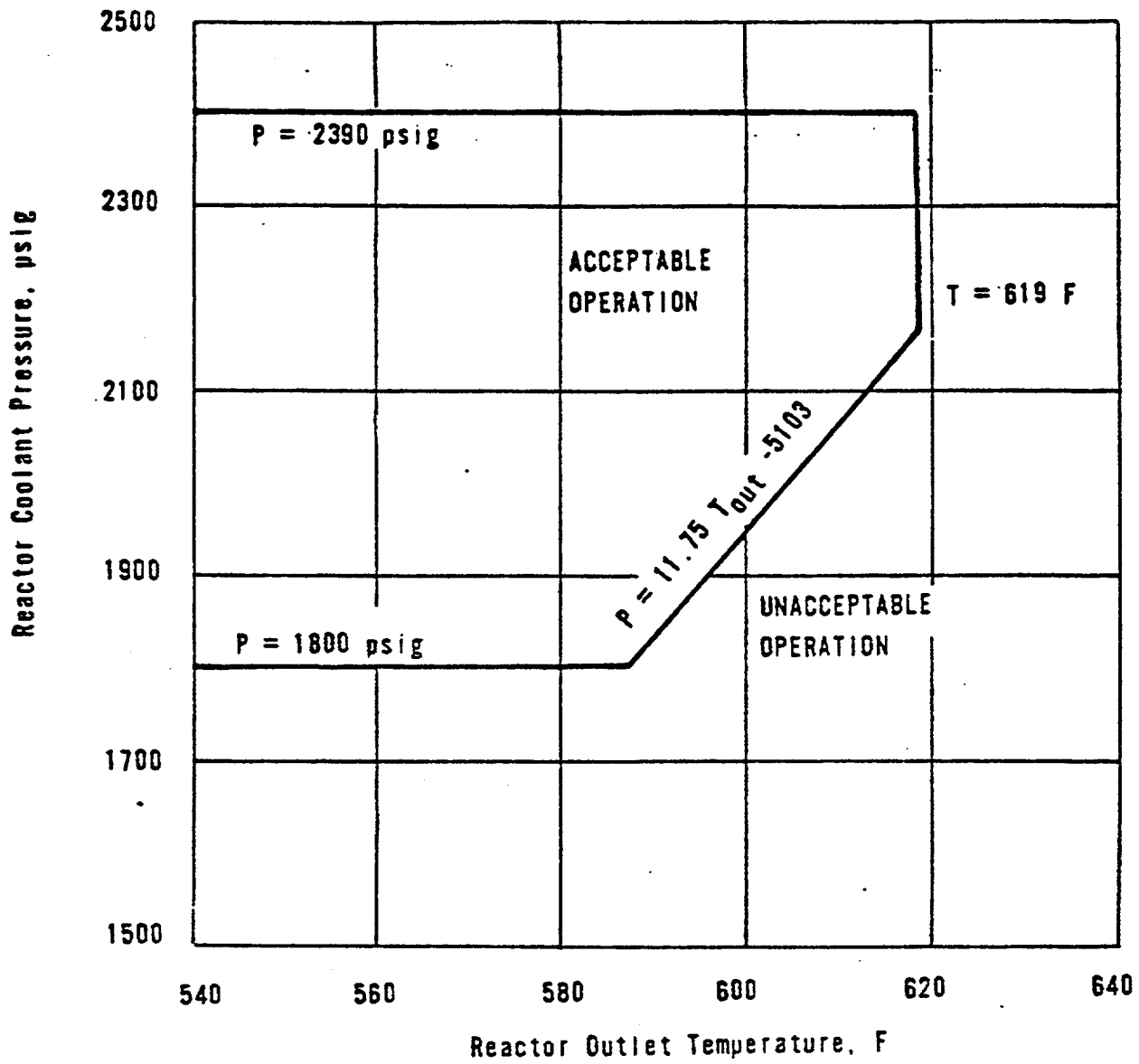
(2) Reactor coolant system flow, %

(3) Administratively controlled reduction set only during reactor shutdown

(4) Automatically set when other segments of the RPS (as specified) are bypassed

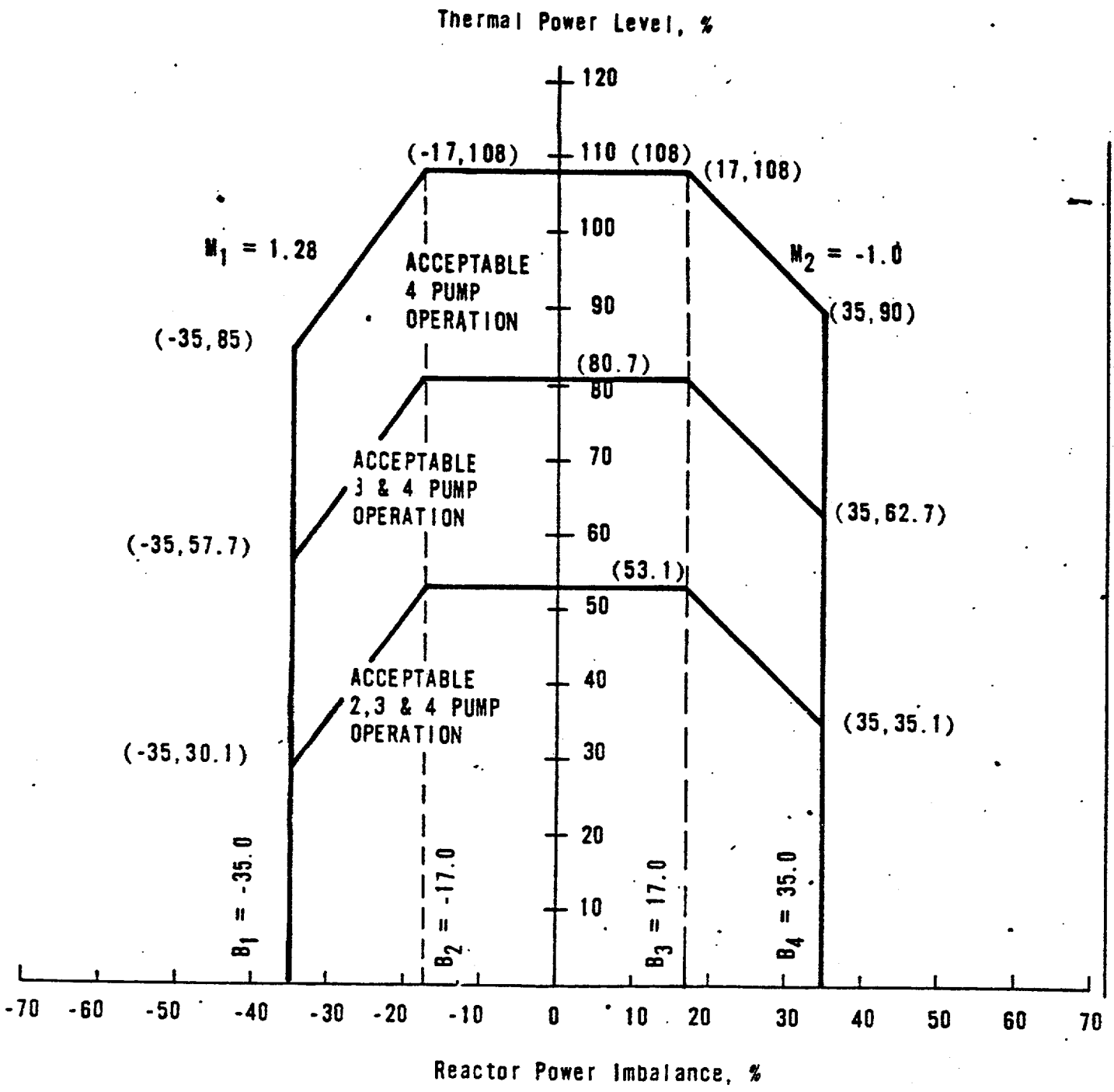
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.

(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable comparators



TMI-1
 PROTECTION SYSTEM MAXIMUM
 ALLOWABLE SET POINTS

Figure 2.3-1



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

Figure 2.3-2

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.

3.1.7.2 The moderator temperature coefficient shall be $\leq + 0.5 \times 10^{-4}$ $\Delta k/k/F$ at power levels $\leq 95\%$ 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 $+0.5 \times 10^{-4} \Delta K/K/F$. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4} \Delta K/K/F$.

A non-positive moderator coefficient at power levels above 95% of rated power is also required to prevent overpressurization of the reactor coolant system in the event of a feedwater line break (see Specification 2.3.1, Basis C, Reactor Coolant System Pressure).

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.

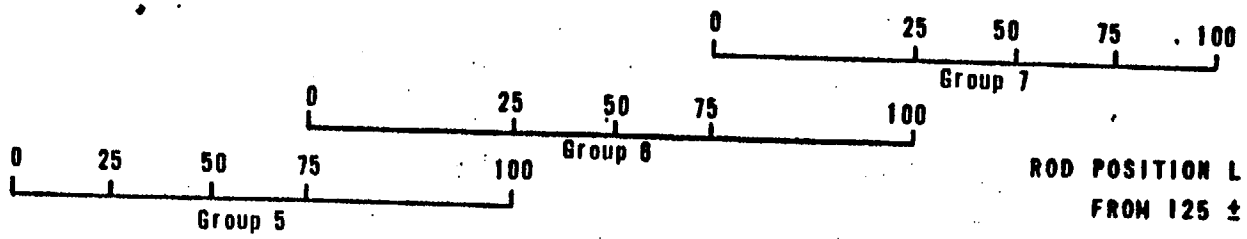
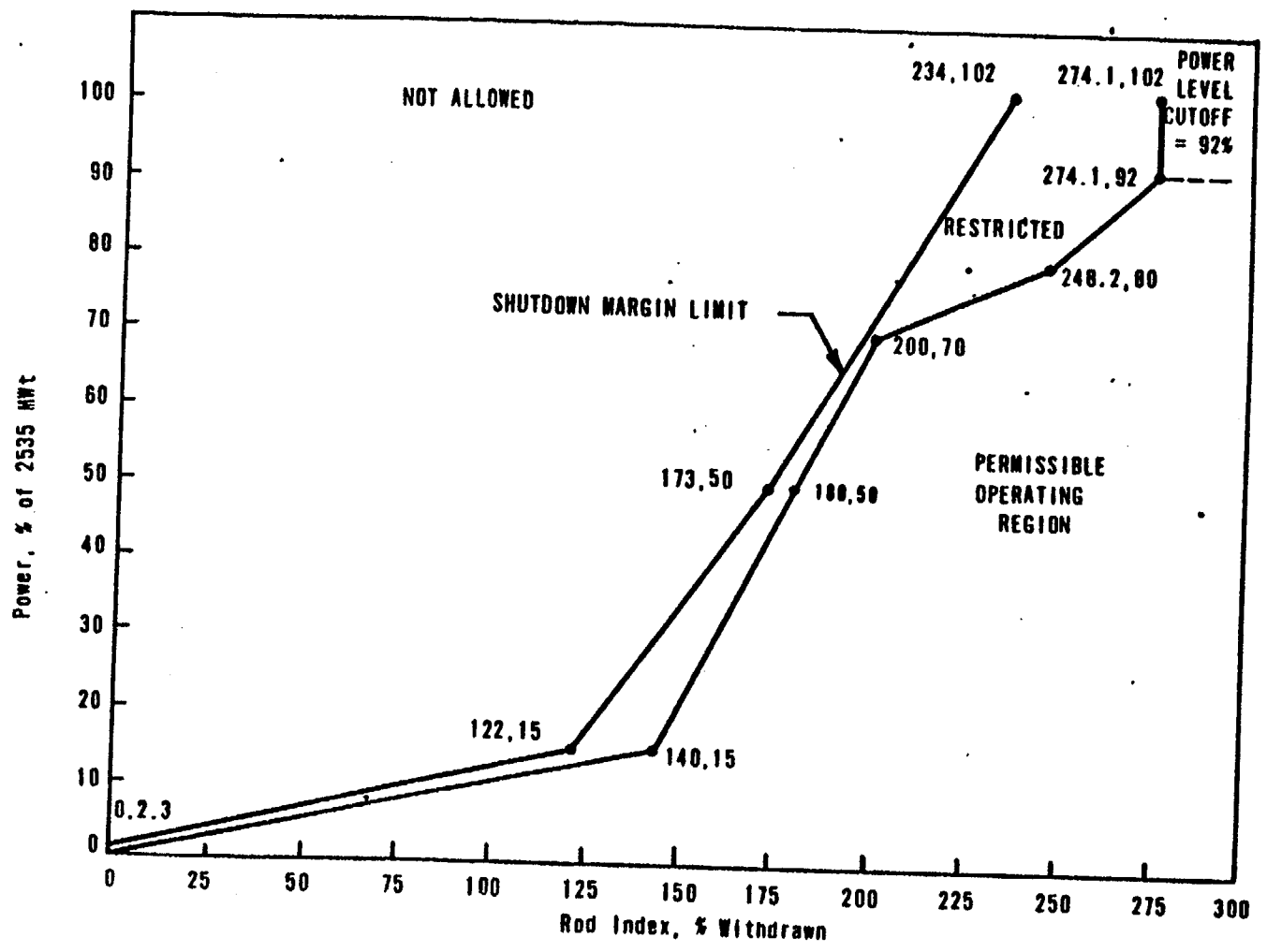
The Final Acceptance Criteria states that post-LOCA clad temperature will not exceed 2200 F.

REFERENCES

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

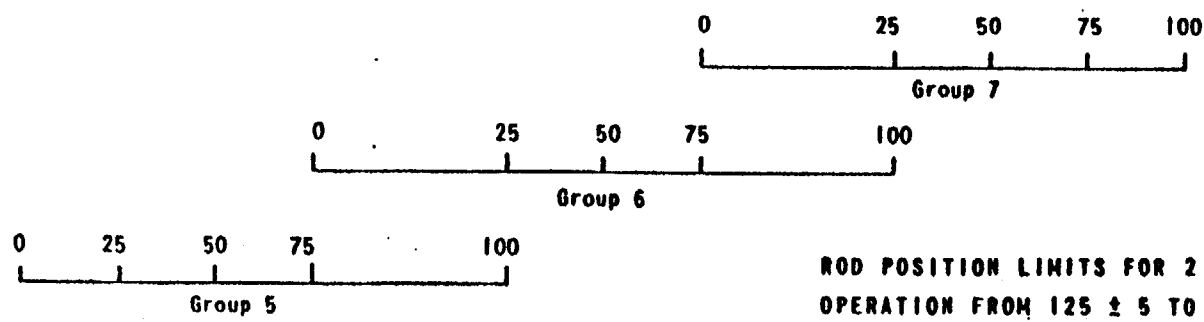
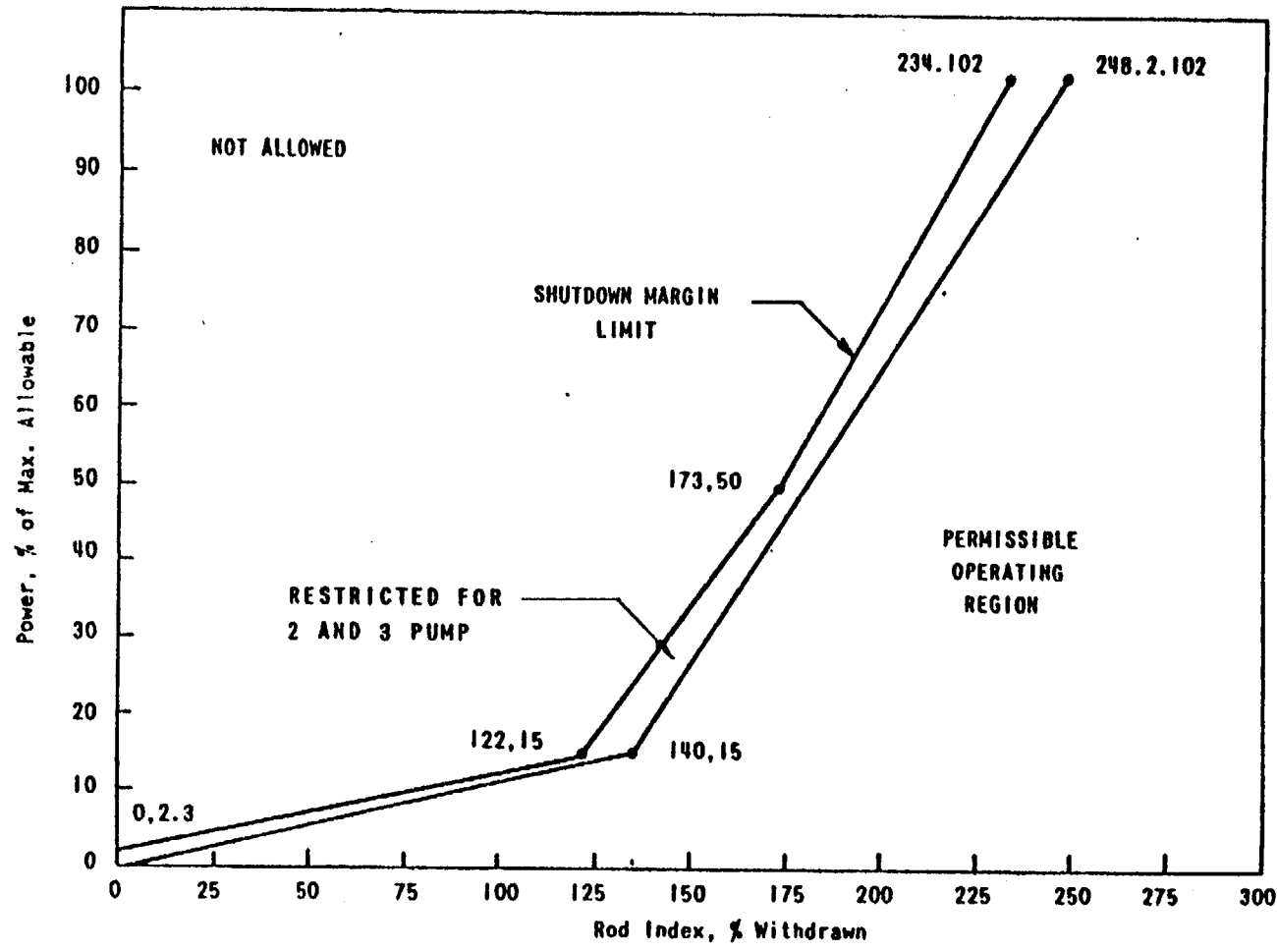
2. The control rod group withdrawal limits (Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, and 3.5-2H, shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.
3. The operational imbalance limits (Figures 3.5-2E, and 3.5-2F) shall be reduced 2 percent in power for each 1 percent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

Amendment No. 10, 11, 35, 36, 45



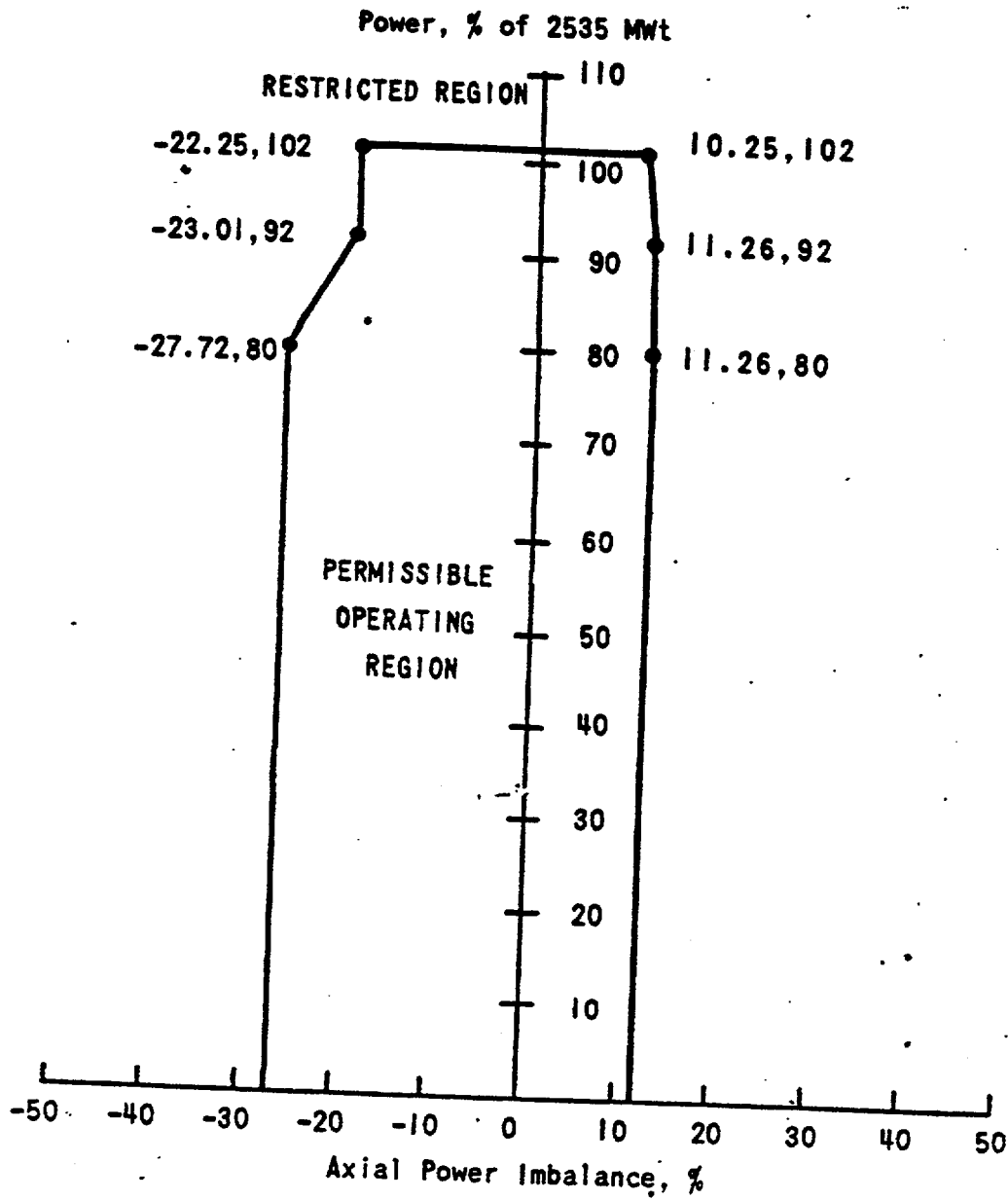
ROD POSITION LIMITS FOR 4 PUMP OPERATION
FROM 125 ± 5 EFPD TO 265 ± 15 EFPD
THI-1, Cycle 4
Figure 3.5-2B

Amendment No. ~~11~~, ~~25~~, ~~35~~, ~~45~~



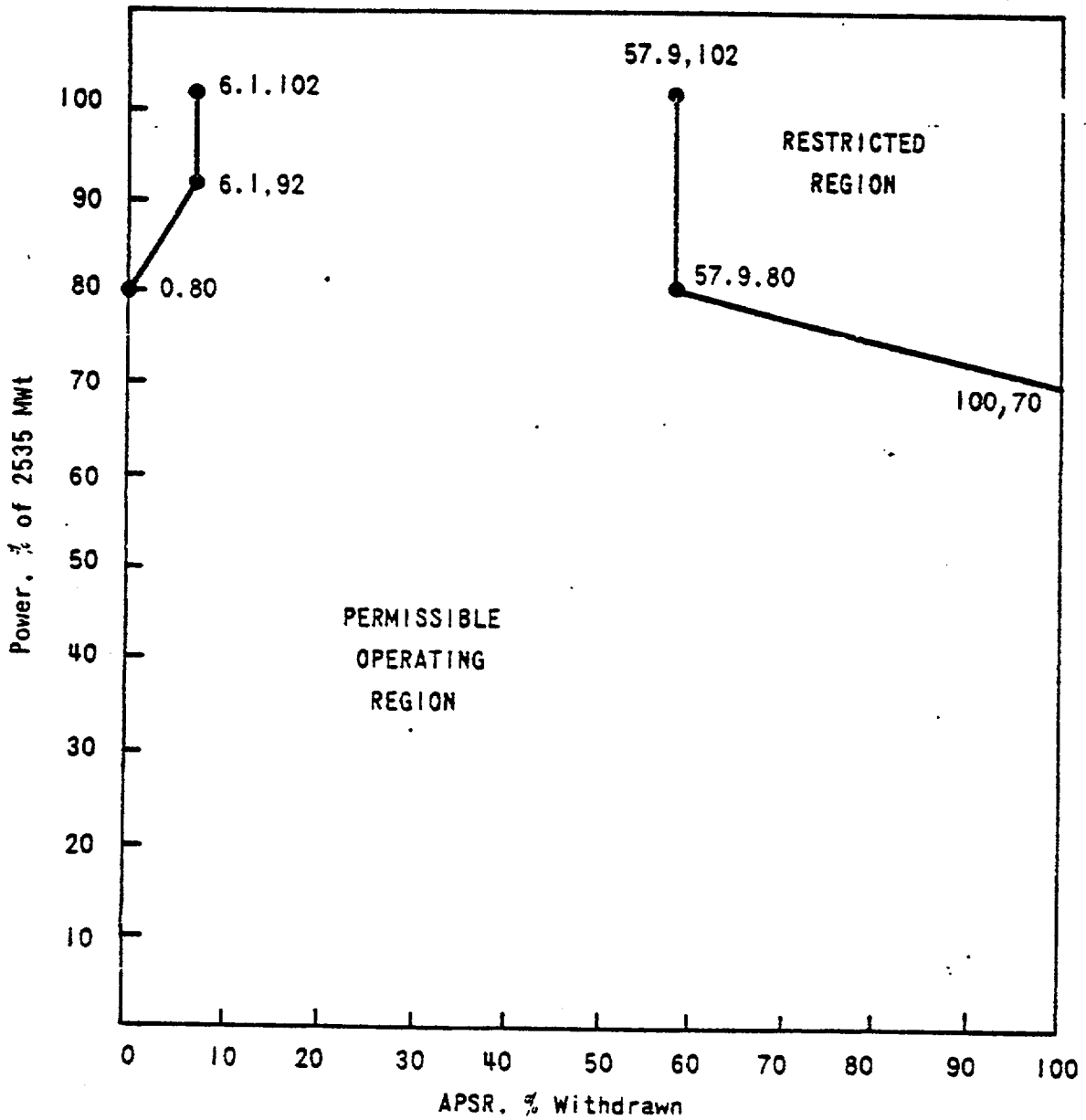
ROD POSITION LIMITS FOR 2 & 3 PUMP
OPERATION FROM 125 ± 5 TO 265 ± 15 EFPO
TMI-1, Cycle 4

Figure 3.5-20



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

Figure 3.5-2F



APSR POSITION LIMITS FOR OPERATION
 FROM 0 TO 265 ± 15 EFPD
 TMI-1

Figure 3.5-2H