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. 50
License No. DPR-50

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

A. The applications for amendment by Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (the licensees), dated December 28, 1978, as supplemented March 1, 1979, and March 5, 1979, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

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

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

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.c.(2) and 2.c.(4) of Facility Operating License No. DPR-50 are hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (4) The licensee may proceed with and is required to complete the modifications identified in Pargraphs 3.1.1 through 3.1.23 of the NRC's Fire Protection Safety Evaluation (SE) on the facility dated September 19, 1978, and supplements thereto. These modifications shall be completed as specified in Table 3.1 of the SE or supplements thereto. In addition, the licensee shall submit the additional information identified in Table 3.2 of this SE in accordance with the schedule contained herein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 16, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise Appendix A as follows:

Remove Pages	Insert Pages
vi & vii	vi & vii
2-2 & 2-3	2-2 & 2-3
Figures 2.1-1 thru 2.1-3	Figures 2.1-1 thru 2.1-3
2-6	2-6
Figure 2.3-2	Figure 2.3-2
3-19 & 3-20	3-19 & 3-20
3-34 thru 3-36	3-34 thru 3-36
Figures 3.5-2A thru 3.5-2E	Figures 3.5-2A thru 3.5-2E
Figure 3.5-2F	(Deleted)
Figures 3.5-2G & 3.5-2H	Figures 3.5-2G & 3.5-2H
-	3-94
-	4-76a

LIST OF FIGURES

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FIGURES	TITLE

a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is 139.8 x 10^{+6} lbs/h, which is less than the actual flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (2) with potential fuel densification and fuel rod bowing effects;

$$F_q^N = 2.57, \; F_{\Delta H}^N = 1.71; \quad F_z^N = 1.50$$

The 1.5 axial peaking factor associated with the cosine flux shape provides a lesser margin to a DNBR of 1.3 than the 1.7 axial peaking factor associated with a lower core flux distribution. For this reason the cosine flux shape and the associated $F_z^N = 1.50$ is more limiting and thus the more conservative assumption.

The 1.50 cosine axial flux shape in conjunction with $F\Delta H = 1.71$ define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing;

a. The 1.3 DNBR limit produced by a nuclear power peaking factor of F^N=2.57 of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than 1.3 DNBR.

b. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 19.6 kW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-2. 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 87.2 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.

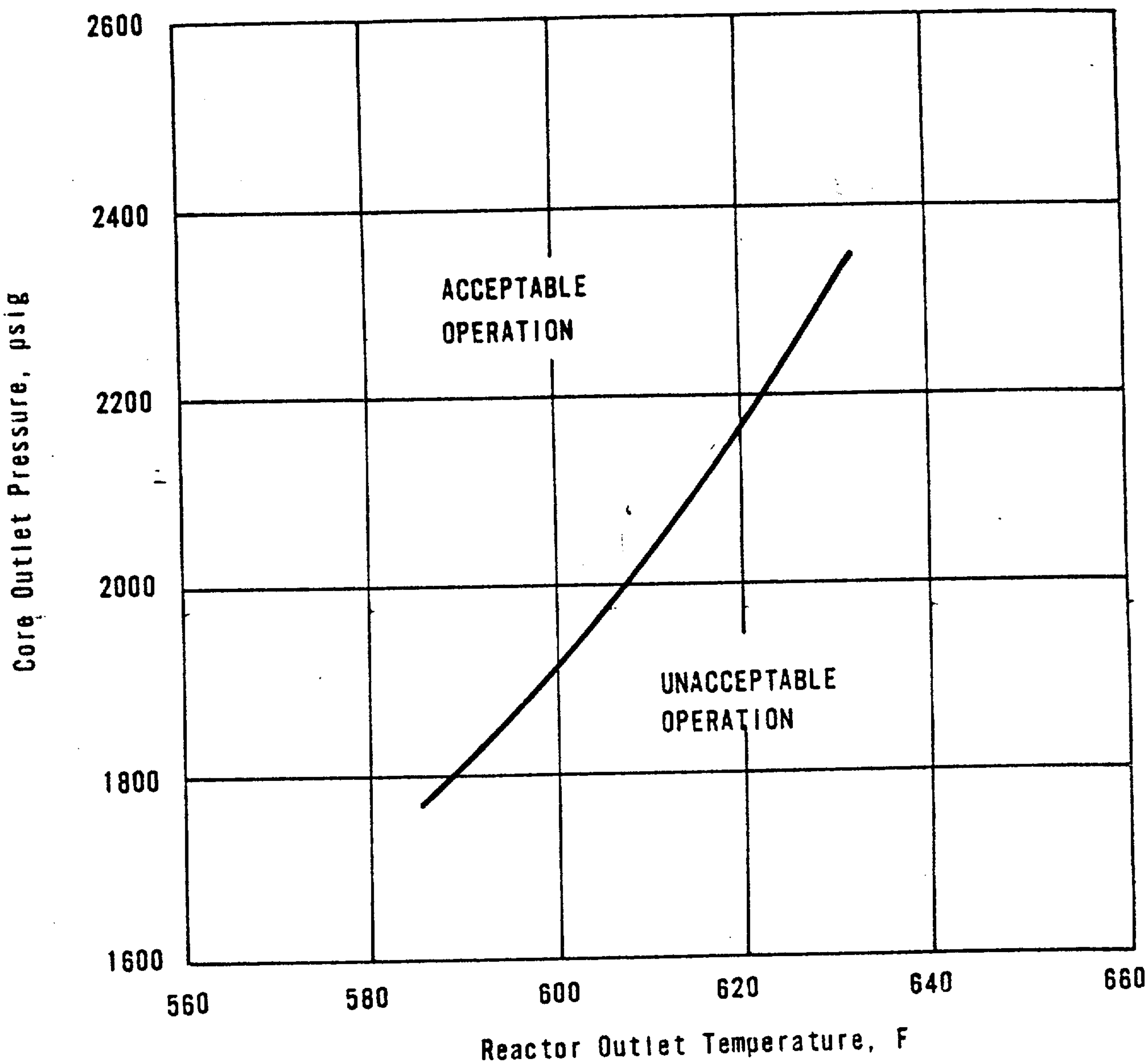
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor 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

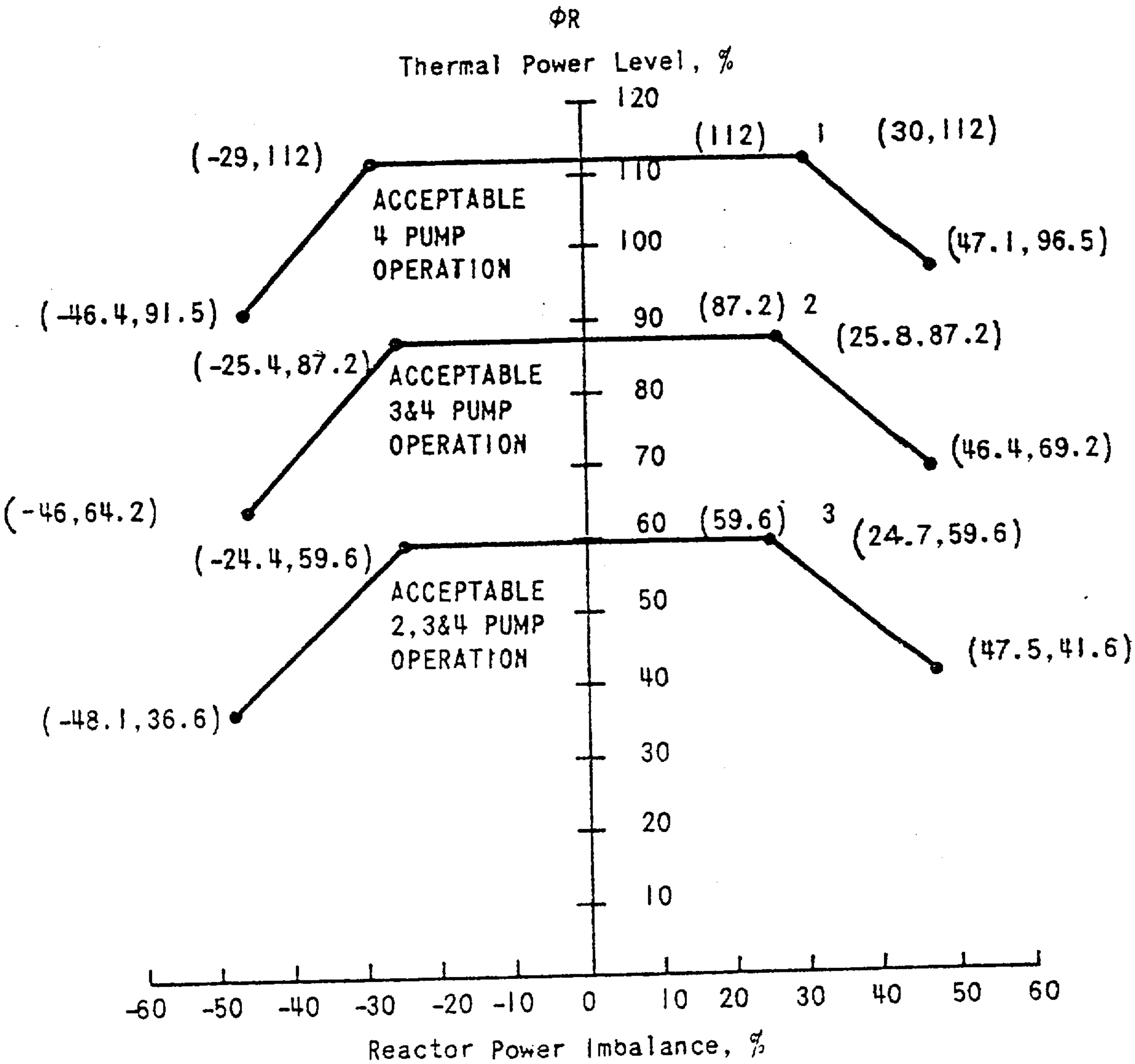
(1) FSAR, Section 3.2.3.1.1

(2) FSAR, Section 3.2.3.1.1.c

(3) FSAR, Section 3.2.3.1.1.k



TMI-1
CORE PROTECTION SAFETY LIMIT
Figure 2.1-1

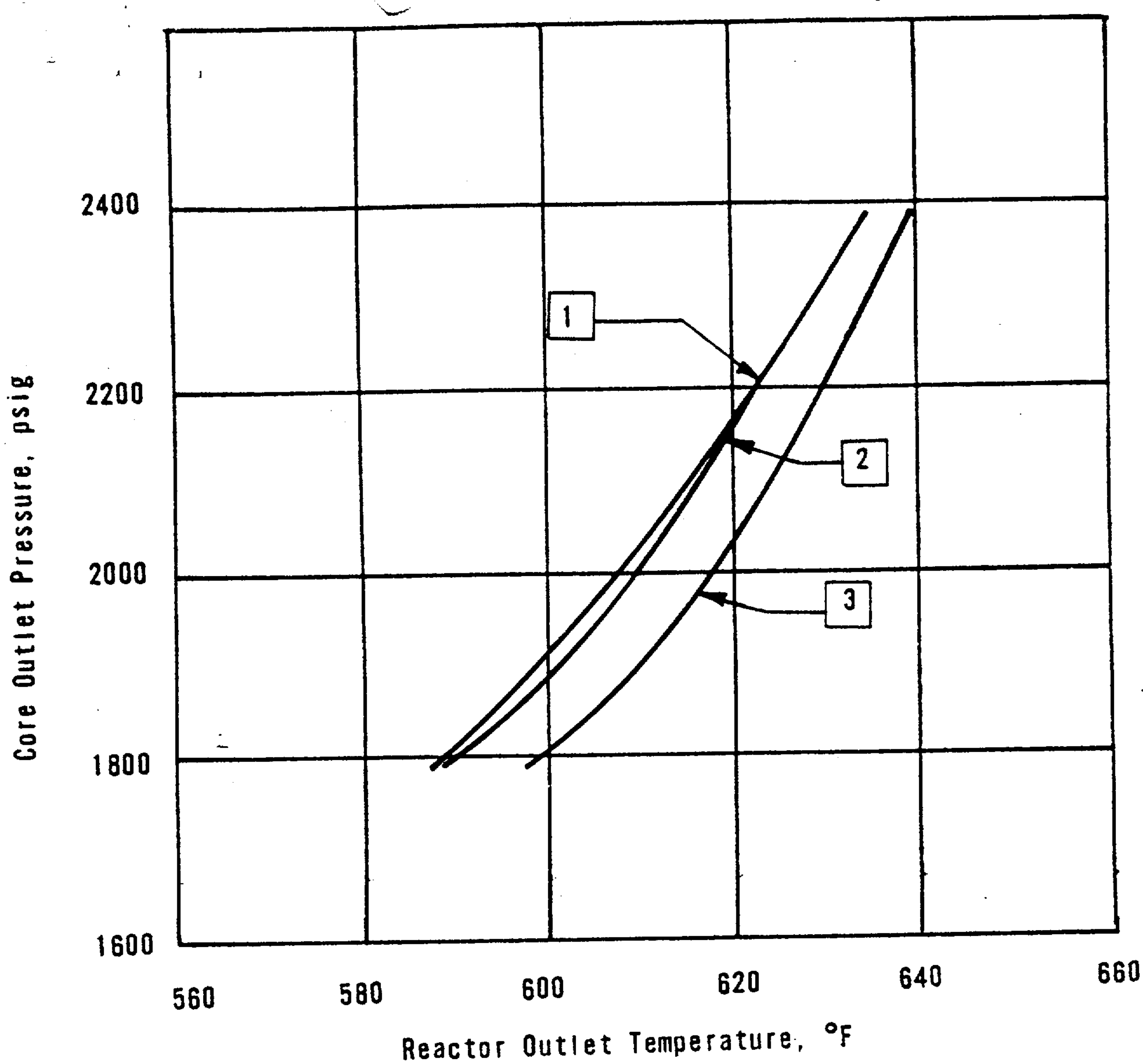


Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

TMI-1

CORE PROTECTION SAFETY LIMITS

Figure 2.1-2



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%)	87.2%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.6%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

TMI-1

CORE PROTECTION SAFETY BASES

Figure 2.1-3

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 92.5 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 69.4 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 53.1 percent and reactor flow rate is 49.2 percent or flow rate is 45.3 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 in such a manner that the reactor protective system receives a conservative indication of the RC flow.

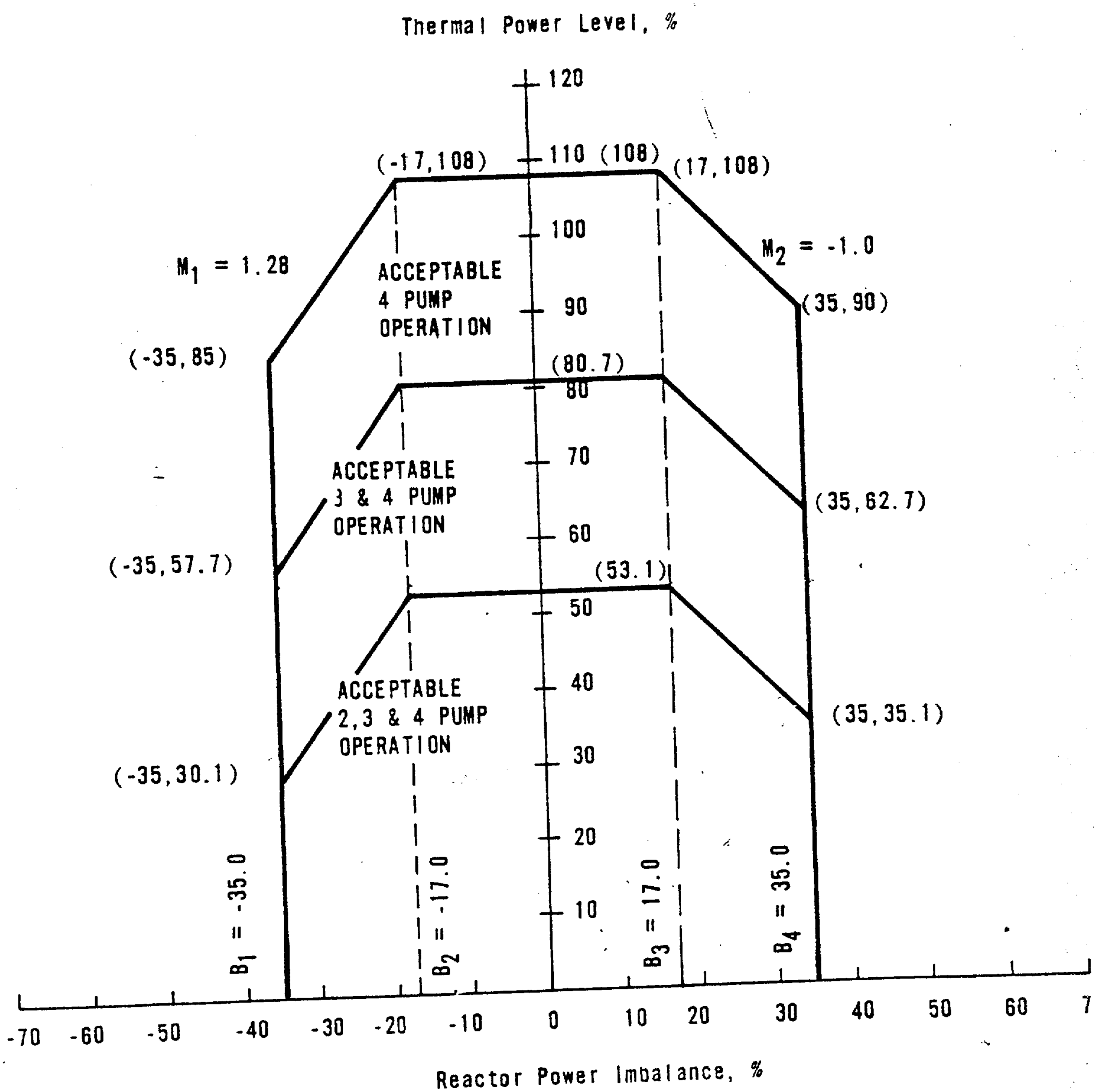
No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage.

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.



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

Figure 2.3-2

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3.2 MAKEUP AND PURIFICATION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the operational status of the makeup and purification and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two makeup and purification pumps are operable except as specified in 3.3.2.

3.2.2 A source of concentrated boric acid solution, in addition to the borated water storage tank, is available and operable. This can be either:

a. The boric acid mix tank containing at least the equivalent of 906 ft^3 of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup and purification system shall also be operable and shall have at least the same temperature requirement as the boric acid mix tank. One associated boric acid pump shall be operable.

b. A reclaimed boric acid storage tank containing at least the equivalent of 906 ft^3 of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup and purification system shall also be operable and shall have at least the same temperature requirement as the reclaimed boric acid tank. One associated reclaimed boric acid pump shall be operable.

Bases

The makeup and purification system and chemical addition systems provide control of the reactor coolant boron concentration. (1) This is normally accomplished by using any of the three makeup and purification pumps in series with a boric acid pump associated with the boric acid mix tank or a reclaimed boric acid pump associated with a reclaimed boric acid storage tank. The alternate method of boration will be the use of the makeup and purification pumps taking suction directly from the borated water storage tank. (2)

The quantity of boric acid in storage from either of the three above mentioned sources is sufficient to borate the reactor coolant system to a one percent subcritical margin in the cold condition at the worst time in core life with a stuck control rod assembly. Minimum volumes (including a 10 percent safety factor) of 906 ft^3 of 8700 ppm boron as concentrated boric acid solution in the boric acid mix tank or in a reclaimed boric acid storage tank or 32,112 gallons of 2270 ppm boron as boric acid solution in the borated water storage tank(3) will each satisfy this requirement. The specification assures that at least two of these supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The primary method of adding boron to the reactor coolant system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using either the 10 gpm boric acid pumps or the 30 gpm reclaimed boric acid pumps. Using only one of the two 10 gpm boric acid pumps, the required volume can be injected in less than 13 hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup and purification pumps. The required 32,112 gallons of boric acid can be injected in less than four hours using only one of the makeup and purification pumps.

Concentration of boron in the boric acid mix tank or a reclaimed boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, the boric acid mix tank is provided with an immersion electric heating element and the reclaimed boric acid tanks are provided with low pressure steam heating jackets to maintain the temperature of their contents well above (10°F or more) the crystallization temperature of the boric acid solution contained in them. Both types of heaters are controlled by temperature sensors immersed in the solution contained in the tanks. Further, all piping, pumps and valves associated with the boric acid mix tank and the reclaimed boric acid storage tanks to transport boric acid solution from them to the makeup and purification system are provided with redundant electrical heat tracing to ensure that the boric acid solution will be maintained 10°F or more above its crystallization temperature. The electrical heat tracing is controlled by the temperature of the external surfaces of the piping systems. Once in the makeup and purification system, the boric acid solution is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

<u>References</u>

(1) FSAR, Sections 9.1 and 9.2

(2) FSAR, Figure 6.2

(3) Technical Specification 3.3

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 may 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 +3.52% 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 +1.90% 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 +1.96% 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, and 3.5-2B). 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.

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) 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 +16.80% determined using the full incore detector system (FIT), or +9.50% determined using the minimum incore detector system (MIT) if the FIT is not available, or +14.20% 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 evey 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 ±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, and 3.5-2B for four pump operation and Figures 3.5-2C and 3.5-2D 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 Figure 3.5-2H. 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 of 92 percent of rated thermal power unless one of the following conditions is satisfied:

    1. Xenon reactivity never deviated more than 10 percent from the equilibrium value for operation at 100 percent of rated thermal power.

    2. Xenon reactivity deviated more than 10 percent and is now within 10 percent of the equilibrium value for operation at 100 percent of rated thermal power and asymptotically approaching stability.

    3. Except for Xenon free startup (when 3.5.2.5.c.2 applies) the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours in the soluble poison control mode.

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 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 at intervals not to exceed 30 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in Figure 3.5-2G.

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 3.5-2G) 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, 3.5-2H, and if quadrant tilt is at the limit. Additional conservatism is introduced 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, and 3.5-2H 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.

The 25±5 percent overl[illegible] between successive control r[illegible] 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 de[illegible]ned 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 sequence beginning with group 1. Groups 5, 6 and 7 are overlapped 25 percent. The normal position at power is for group 7 to be partially inserted.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% $\Delta k/k$ ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, than manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed at least every two hours until the computer is returned to service.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using an actual core tilt of +4.92% which is equivalent to a +3.52% 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	60%
75	85%
>75	105.5%

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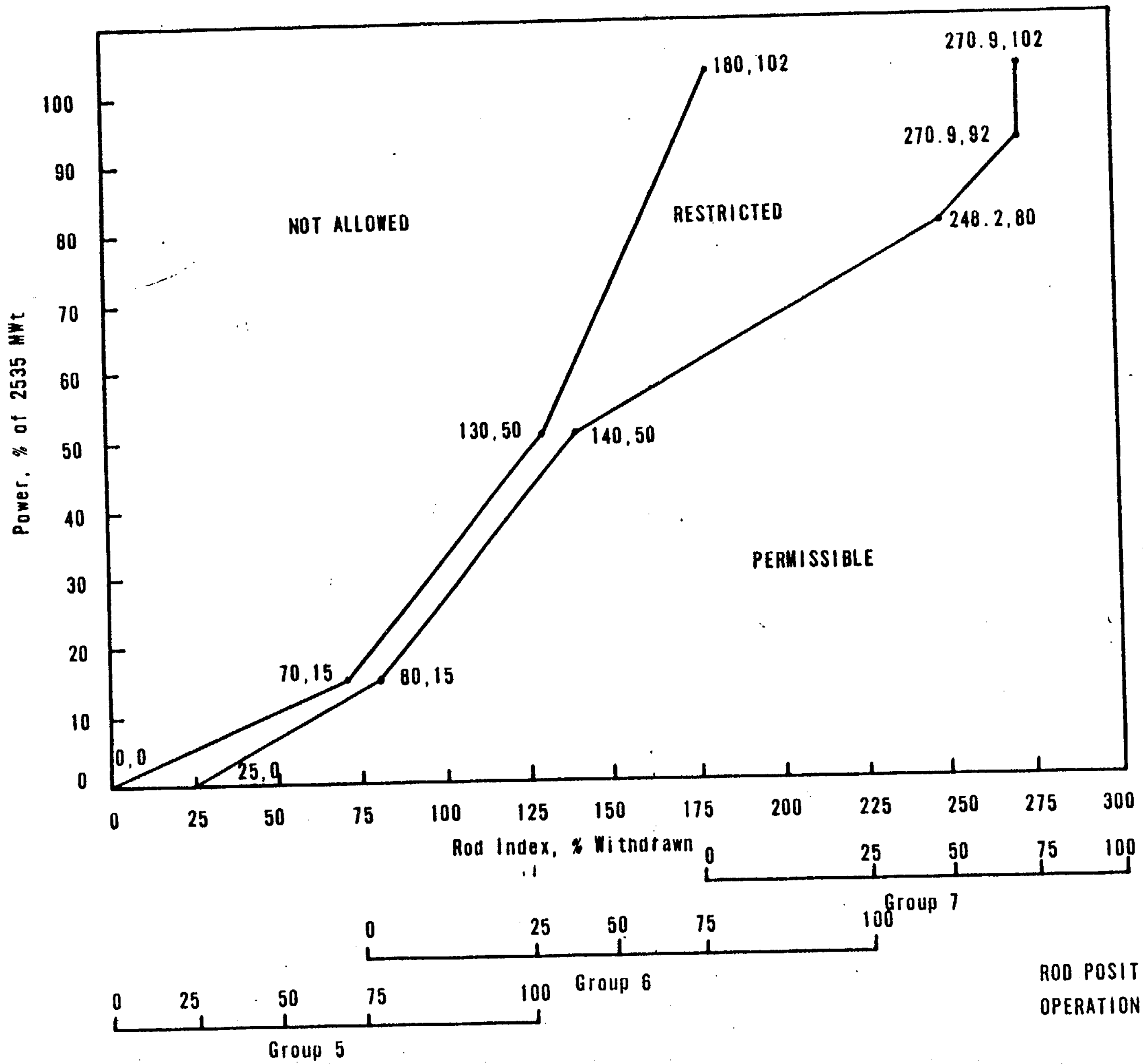

ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 0 TO 125 ± 5 EFPD

TMI-1

Figure 3.5-2A

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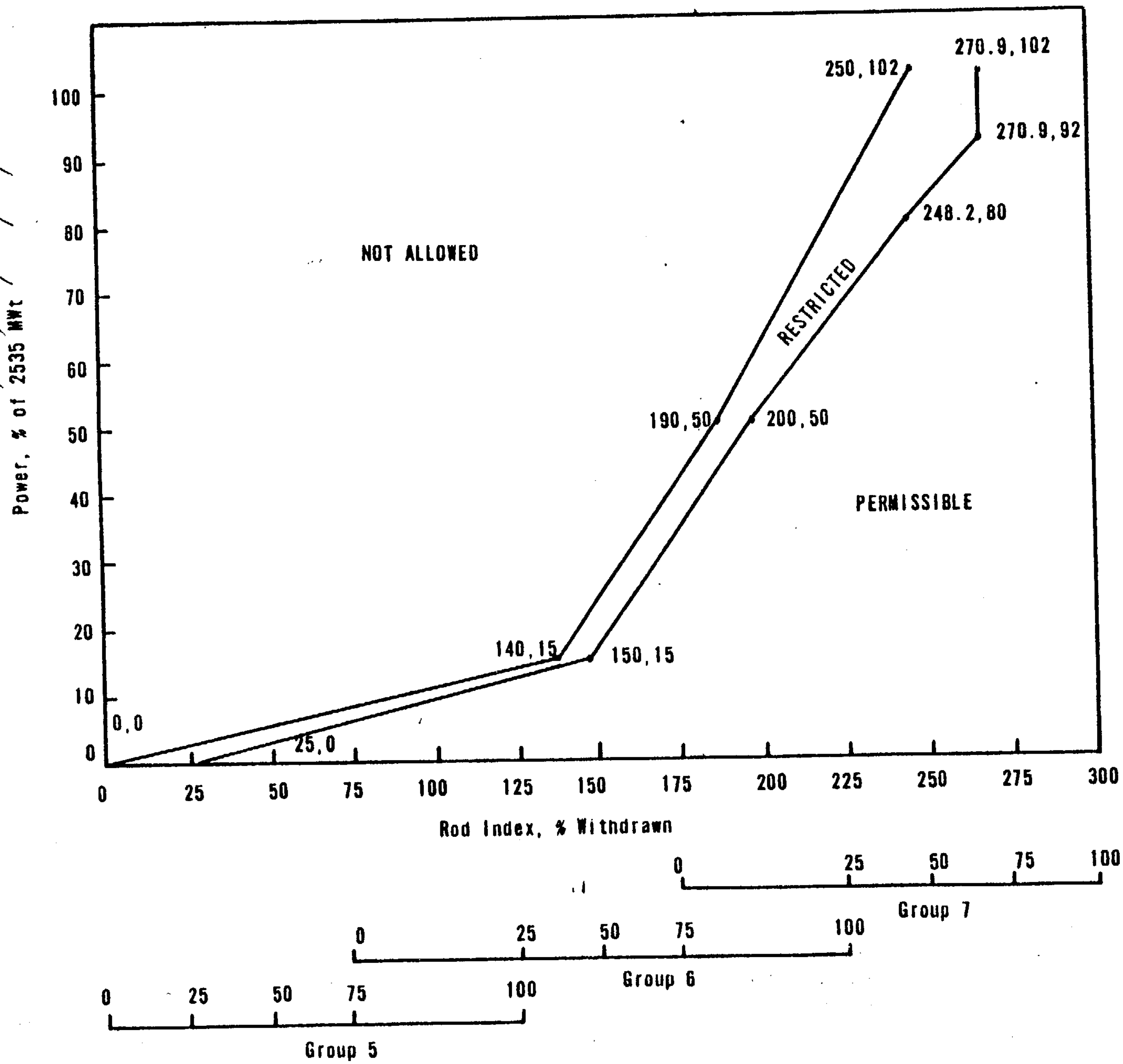
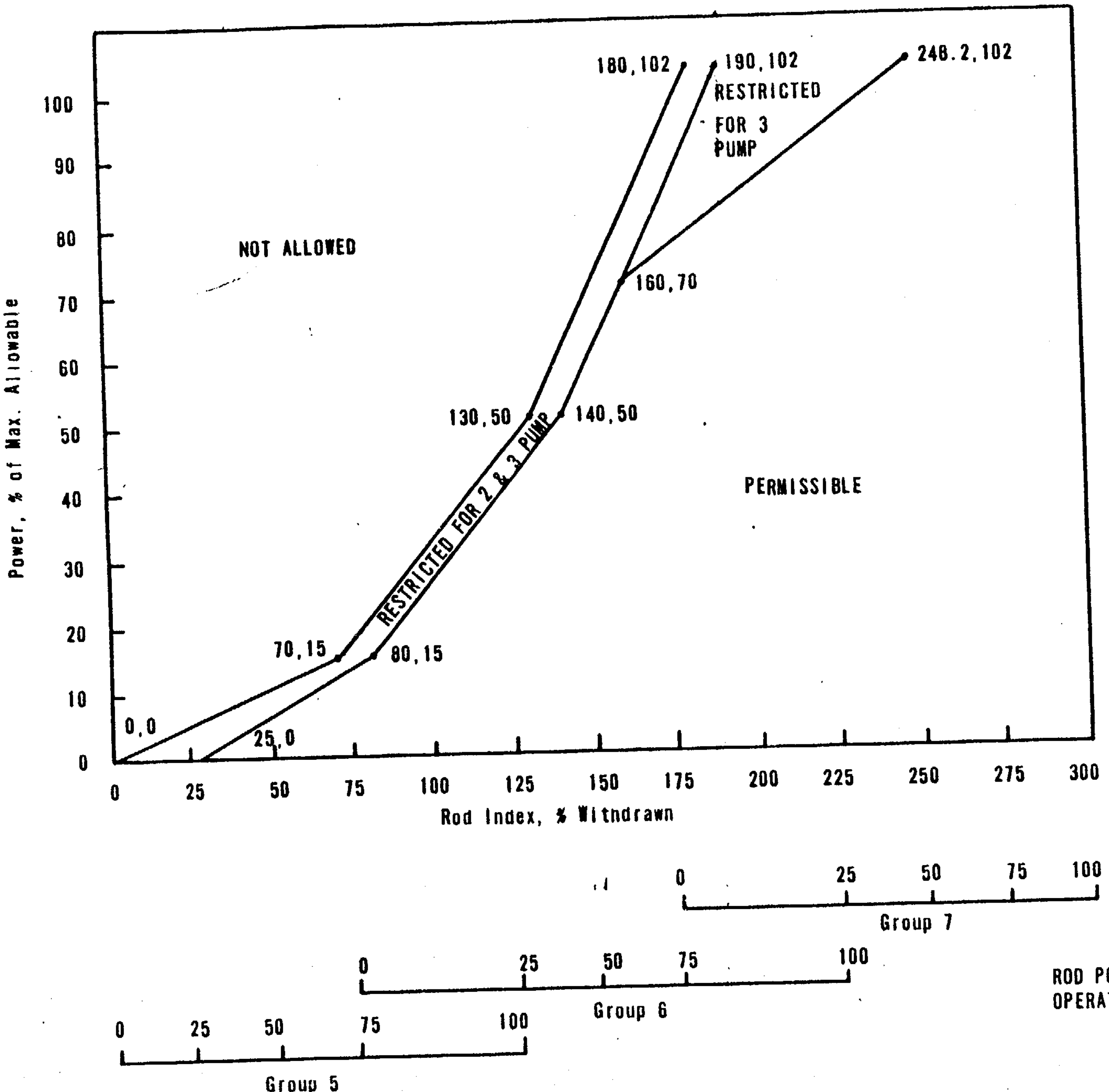


ROD POSITION LIMITS FOR 4 PUMP OPERATION
FROM 125 ± 5 EFPD TO EOC

TMI-1

Figure 3.5-2B



ROD POSITION LIMITS FOR 2 & 3 PUMP
OPERATION FROM 0 TO 125 ± 5 EFPD

TMI-1

Figure 3.5-2C

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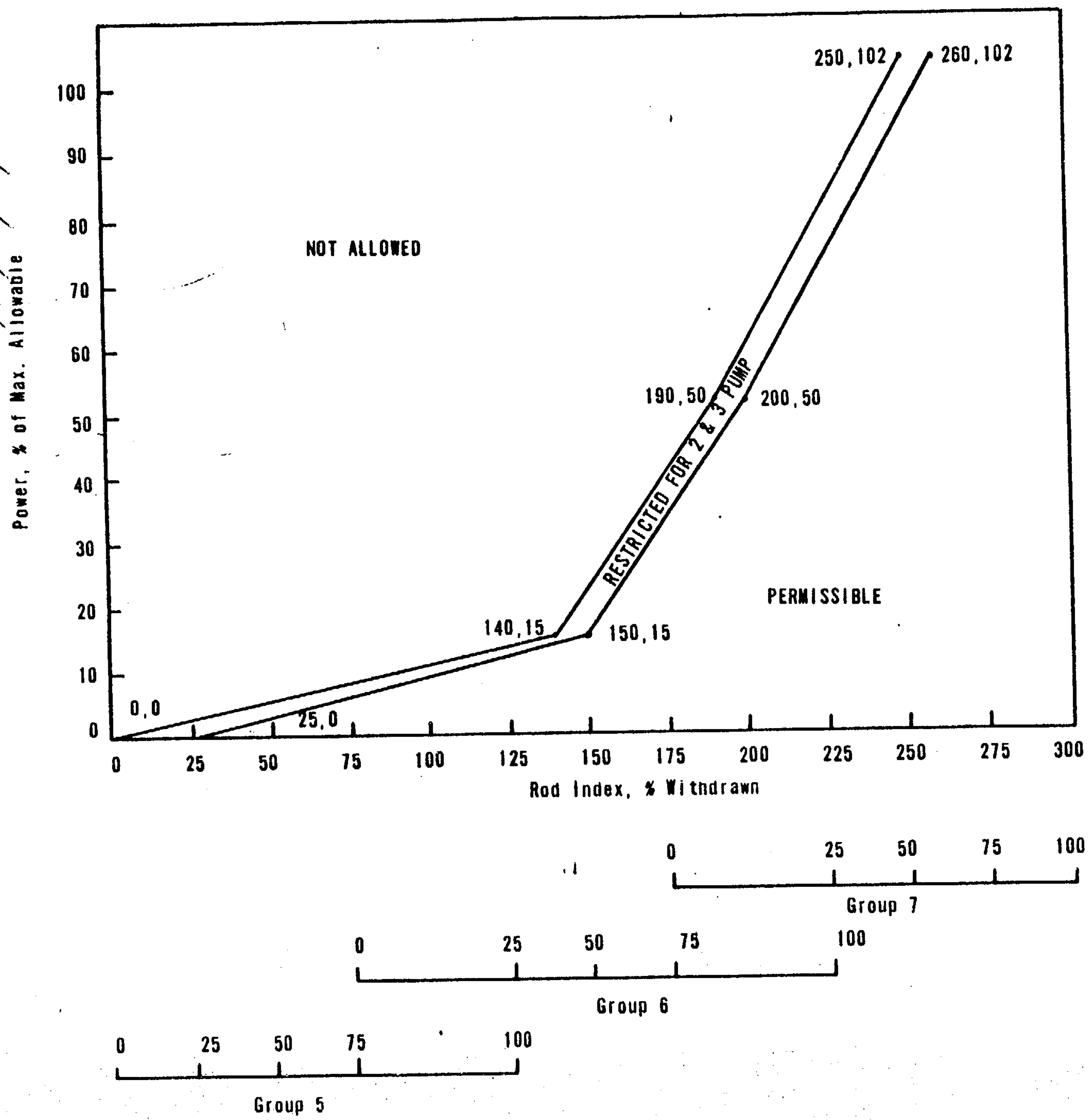
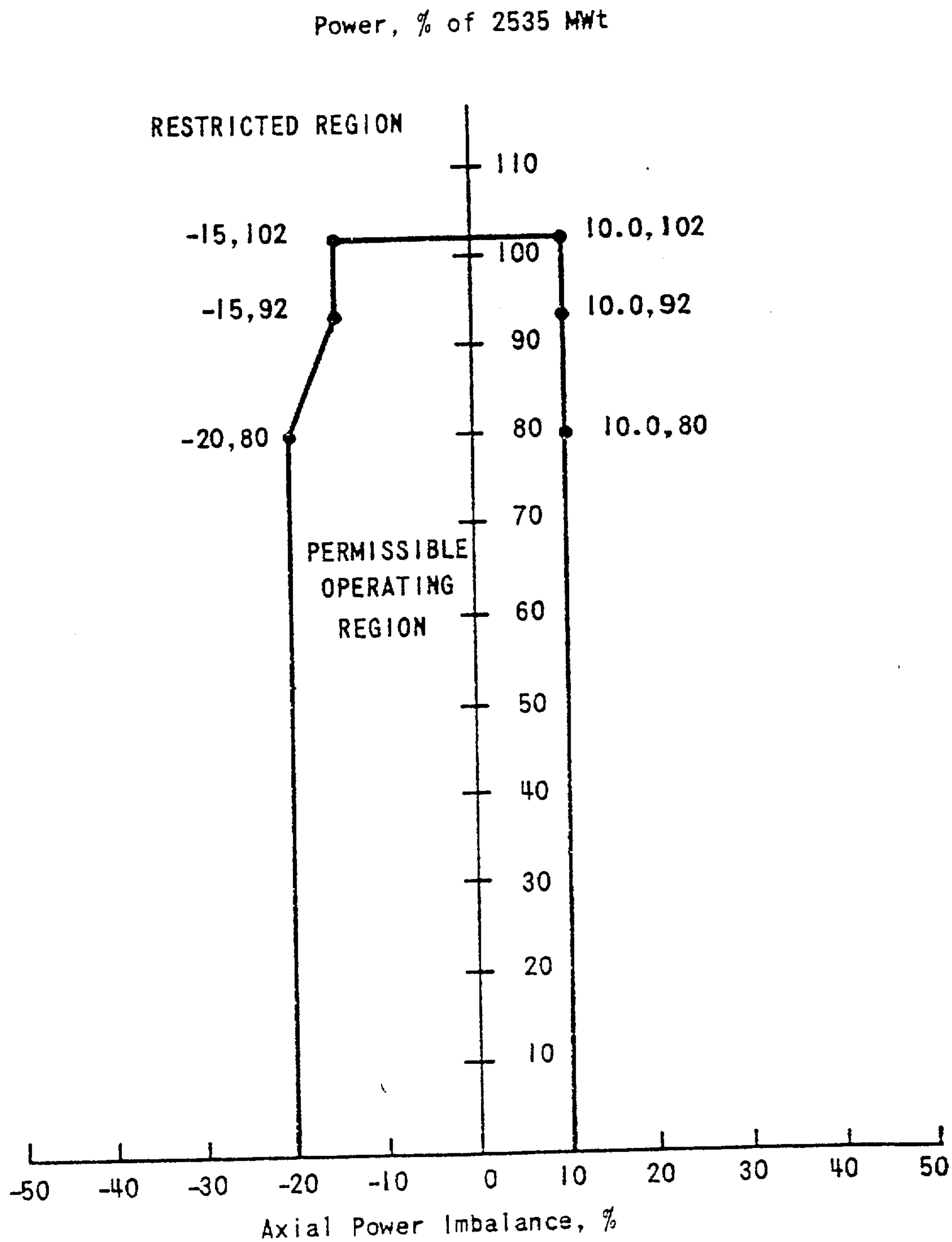


ROD POSITION LIMITS FOR 2 & 3 PUMP
OPERATION FROM 125 $\pm$ 5 EFPD TO EOC
TMI-1
Figure 3.5-2D



POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 0 EFPD TO EOC

Figure 3.5-2E

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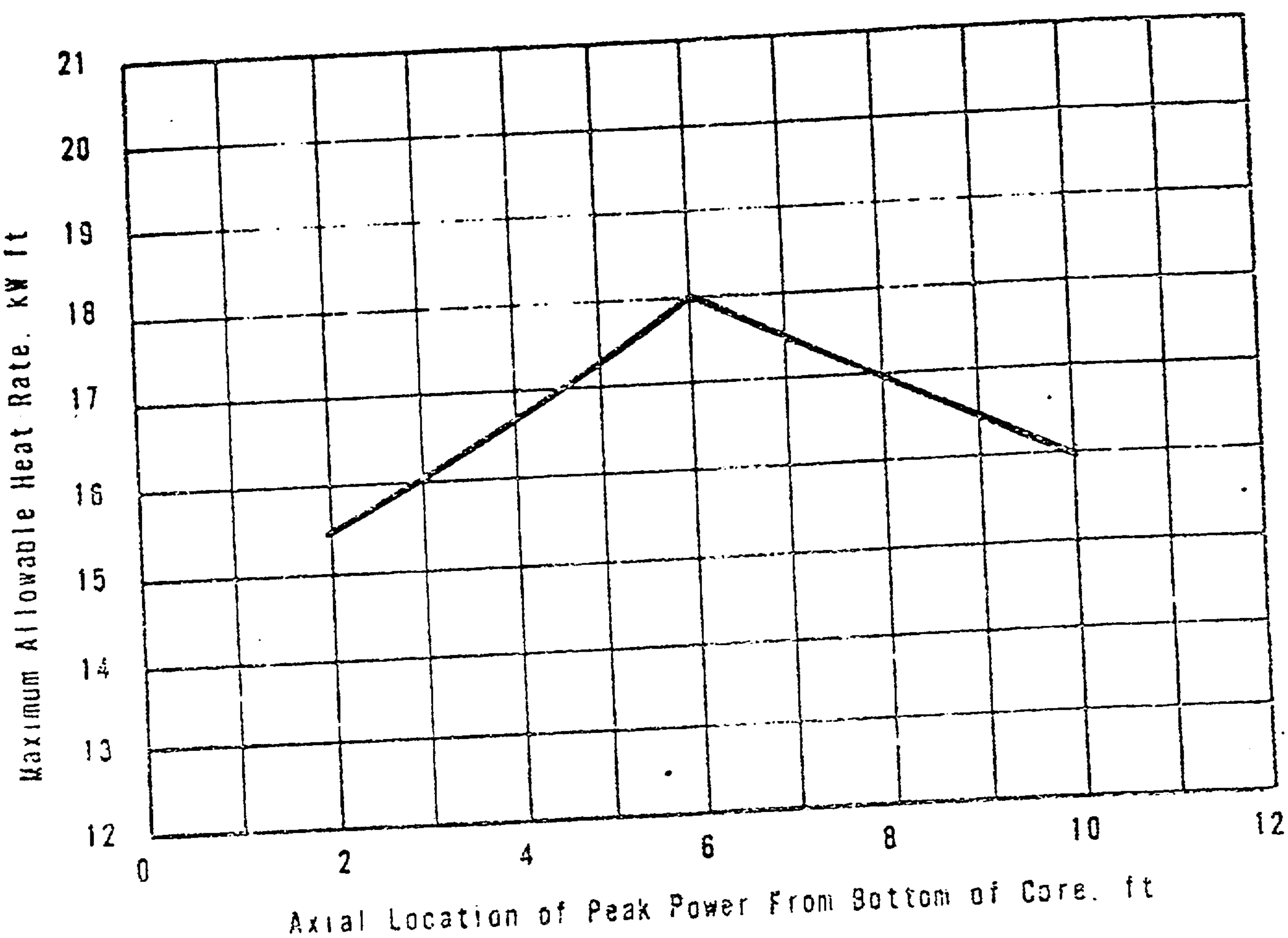
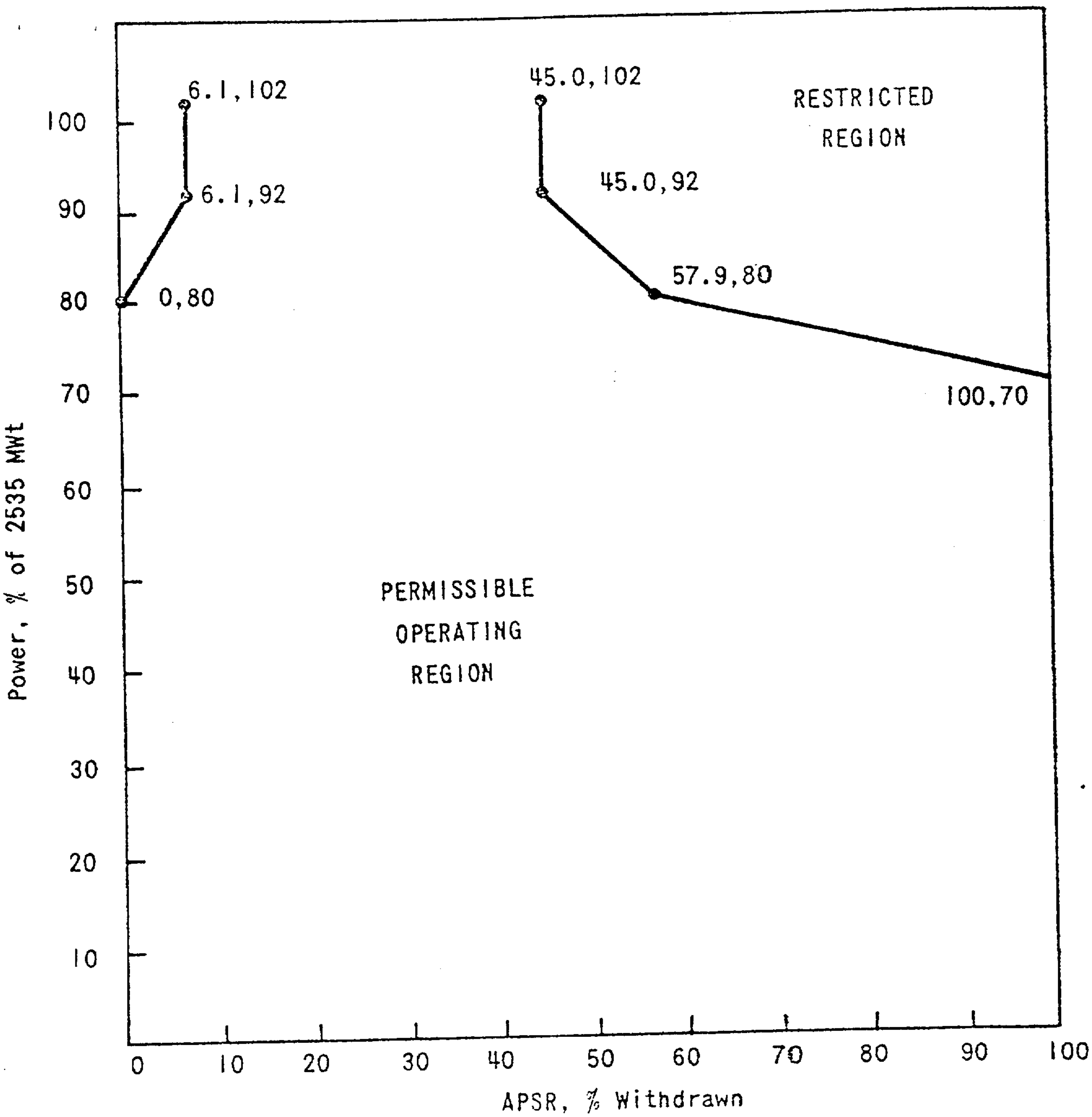
Figure 3.5-2F

Deleted



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE - TMI-1

Figure 3.5-2G



APSR POSITION LIMITS FOR OPERATION

FROM 0 EFPD TO EOC

Figure 3.5-2H

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3.18.7 FIRE BARRIER PENETRATION SEALS

Applicability: At all times when equipment on either side of barrier is required to be operable.*

Objective: To assure the effectiveness of fire barriers.

Specification:

3.18.7.1 All fire barrier penetration seals protecting safety related areas shall be functional.

3.18.7.2 With one or more of the above required fire barrier penetration seals non-functional, establish a continuous fire watch on at least one side of the affected penetration within one hour.

* Subject to installation schedule contained in the Three Mile Island Nuclear Station, Unit No. 1 Fire Protection Safety Evaluation Report dated September 19, 1978 and supplements thereto.

Bases: The functional integrity of the fire barrier penetration seals ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected seal until the seal is restored to functional status.

4.18.7 FIRE BARRIER PENETRATION SEALS

Applicability: Fire barrier penetration seals which protect safety-related areas.

Objective: To assure that the effectiveness of fire barriers protecting safety-related areas is maintained.

Specification:

4.18.7.1 Fire barrier penetration seals shall be verified to be functional by a visual inspection:

a. At least once each refueling interval; and

b. Prior to declaring a fire penetration seal functional following repairs, maintenance or initial installation in accordance with the schedule contained the Three Mile Island Nuclear Station, Unit No. 1, Fire Protection Safety Evaluation Report dated September 19, 1978 and supplements thereto.