March 20, 1987

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letters dated November 3, 1986 (Technical Specification Change Request (TSCR) 149) and July 16, 1986 (TSCR 159).

This amendment revises the TMI-1 TSs to support the core reload for Cycle 6 operation. Additionally, the amendment changes the order of preference of instrumentation used to monitor reactor power quadrant tilt.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

John O. Thoma, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 126 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

PBD-6	PBD-6	PBD-6	PBD-6	PBD-6	OGC
RIngram	JThoma:eh	TRoss	RWeller	JStolz	SH Lewis
3/5/87	3/5/87	3/5/87	3/6/87	3/6/87	3/11/87

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Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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Atomic Safety & Licensing Appeal
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WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. DPR-50

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

   A. The applications for amendment by GPU Nuclear Corporation, et al. (the licensees) dated July 16, 1986, and November 3, 1986, 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.

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

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 126, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1987

ATTACHMENT TO LICENSE AMENDMENT NO.126

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
vii	vii
viii	viii
2-2	2-2
2-3	2-3
2-5	2-5
2-6	2-6
2-7	2-7
2-9	2-9
3-34	3-34
3-34a	3-34a
3-35	3-35
3-35a	3-35a
3-36	3-36
3-36a	3-36a
5-3	5-3*
5-4	5-4
Figures:	
2.1-2	2.1-2
2.1-3	2.1-3
2.3-1	2.3-1
2.3-2	2.3-2
3.5-2A	3.5-2A
3.5-2B	3.5-2B
3.5-2C	3.5-2C
3.5-2D	3.5-2D
3.5-2E	3.5-2E
3.5-2F	3.5-2F
3.5-2G	3.5-2G
3.5-2H	3.5-2H
3.5-2I	3.5-2I
---	3.5-2J
---	3.5-2K
----	3.5-2L

*Overleaf page provided for document completeness

LIST OF FIGURES

Figure	Title
2.1-1	TMI-1 Core Protection Safety Limit
2.1-2	TMI-1 Core Protection Safety Limits
2.1-3	TMI-1 Core Protection Safety Bases
2.3-1	TMI-1 Protection System Maximum Allowable Set Points
2.3-2	Protection System Maximum Allowable Set Points for Reactor Power Imbalance, TMI-1
3.1-1	Reactor Coolant System Heatup/Cooldown Limitations (Applicable to 5 EFPY)
3.1-2	Reactor Coolant System, Inservice Leak and Hydrostatic Test Limitations (Applicable to 5 EFPY)
3.1-3	Limiting Pressure vs. Temperature Curve for 100 STD cc/Liter H_2O
3.5-2A	Rod Position Limits for 4 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2B	Rod Position Limits for 4 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2C	Rod Position Limits for 4 Pump Operation after 250±10 EFPD, TMI-1
3.5-2D	Rod Position Limits for 3 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2E	Rod Position Limits for 3 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2F	Rod Position Limits for 3 Pump Operation after 250±10 EFPD, TMI-1
3.5-2G	Rod Position Limits for 2 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2H	Rod Position Limits for 2 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2I	Rod Position Limits for 2 Pump Operation after 250±10 EFPD, TMI-1
3.5-2J	Power Imbalance Envelope for Operation from 0 to 30+10/-0 EFPD, TMI-1

LIST OF FIGURES

Figure	Title
3.5-2K	Power Imbalance Envelope for Operation after 30+10/-0 EFPD, TMI-1
3.5-2L	LOCA Limited Maximum Allowable Linear Heat Rate
3.5-1	Incore Instrumentation Specification Axial Imbalance Indication, TMI-1
3.5-2	Incore Instrumentation Specification Radial Flux Tilt Indication, TMI-1
3.5-3	Incore Instrumentation Specification
3.11-1	Transfer Path to and from Cask Loading Pit
4.17-1	Snubber Functional Test - Sample Plan 2
5-1	Extended Plot Plan TMI
5-2	Site Topography 5 Mile Radius
5-3	Site Boundary for Gaseous Effluents
5-4	Site Boundary for Liquid Effluents
6-1	GPU Nuclear Corporation Organization Chart
6-2	TMI-1 Onsite Organization

Amendment Nos. 72, ~~77~~, 126

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 DNBR of 1.3 or greater 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;

$$F_q^N = 2.82, \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.65$$

The 1.65 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 = 1.65 is more limiting and thus the more conservative assumption.

The 1.65 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_q^N = 2.82 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 20.50 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 DNBR of 1.3 or greater 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 89.3 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.03 = 80.6 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 the 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 3.2.3.1.1.3

(3) FSAR, Section 3.2.3.1.1.11

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.

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 operations with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.1% 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

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 92.5 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.6 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.

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 setpoint. 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 high pressure trip setpoint was subsequently lowered from 2390 psig to 2300 psig. The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

The low pressure (1800 psig) and variable low pressure ($11.75\ T_{OUT}$-5103) trip setpoint were initially 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). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. Figure 2.3-1 shows the high pressure, low pressure, and variable low pressure trips.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618.8F) 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 618.8F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1.2°F. 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.

Table 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS(6)

	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	105.1	105.1	105.1	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	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (5) on pump monitors, Max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2300	2300	2300	1720(4)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	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.	618.8	618.8	618.8	618.8
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 setting limits are setting limits on the setpoint side of the protection system bistable connectors.

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 is limited by the restriction 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 +4.12% 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 +1.96% as determined using the power range channels displayed on the console for each quadrant (out of core detection system).

c. When neither detector system above is available and, except for physics tests, quadrant tilt shall not exceed +1.90% as determined using the minimum incore detector system.

d. Except for physics tests if quadrant tilt exceeds the tilt limit power 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, in excess of the tilt limit.

2. The control rod group withdrawal limits (Figures 3.5-2A to 3.5-2I) shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

3. The operational imbalance limits (Figures 3.5-2J and 3.5-2K) 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 +14.2% determined using the out of core detector system (OCT) if the FIT is not available, or +9.5% using the minimum incore detector system (MIT) when neither the FIT nor OCT 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. ~~29~~, ~~38~~, ~~39~~, ~~40~~, ~~45~~,
~~50~~, ~~120~~, 126

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 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 pump operation. Two pump operation is specified on Figures 3.5-2G, 3.5-2H, and 3.5-2I. 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. Deleted

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 3.5-2J and 3.5-2K. If the imbalance is not within the envelopes defined by Figures 3.5-2J or 3.5-2K at the appropriate time in cycle, 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 3.5-2L.

Bases

The power-imbalance envelope defined in Figures 3.5-2J and 3.5-2K is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2L) such that the maximum clad temperature will not exceed the Final Acceptance Criteria (2200°F). 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, 3.5-2G, 3.5-2H, 3.5-2I, and if quadrant tilt is at the limit. The effects of the gray APSRs are also included. 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

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

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, 3.5-2G, 3.5-2H, and 3.5-2I 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

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 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, 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 +4.92% which is equivalent to a +4.12% tilt measured with the full incore instrumentation with statistically combined measurement uncertainties included. The maximum allowable quadrant power tilt setpoint of +16.8% tilt measured with the full incore detector system represents a +20% actual core tilt and includes bounding measurement uncertainty allowances.

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	Test Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.1%

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in Section 14, with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of energy sources and heat sinks.

5.2.2 REACTOR BUILDING ISOLATION SYSTEM

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves.[2]

REFERENCES

(1) FSAR Section 5.1

(2) FSAR Section 5.3.1

5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

5.3.1 REACTOR CORE

5.3.1.1 The reactor core contains approximately 93.1 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies. Each fuel assembly contains 208 fuel rods.(1)(2)

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 142.25 inches.(2)

5.3.1.3 The average initial enrichment of the current core for Unit 1 is a nominal 2.86 weight percent of U^{235}. The highest enrichment is less than 3.5 weight percent U^{235}.

5.3.1.4 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3.2-1. The full-length CRA contain a 134 inch length of silver-indium-cadmium alloy clad with stainless steel.(3) The gray APSRA contain a 63 inch length of Inconel.

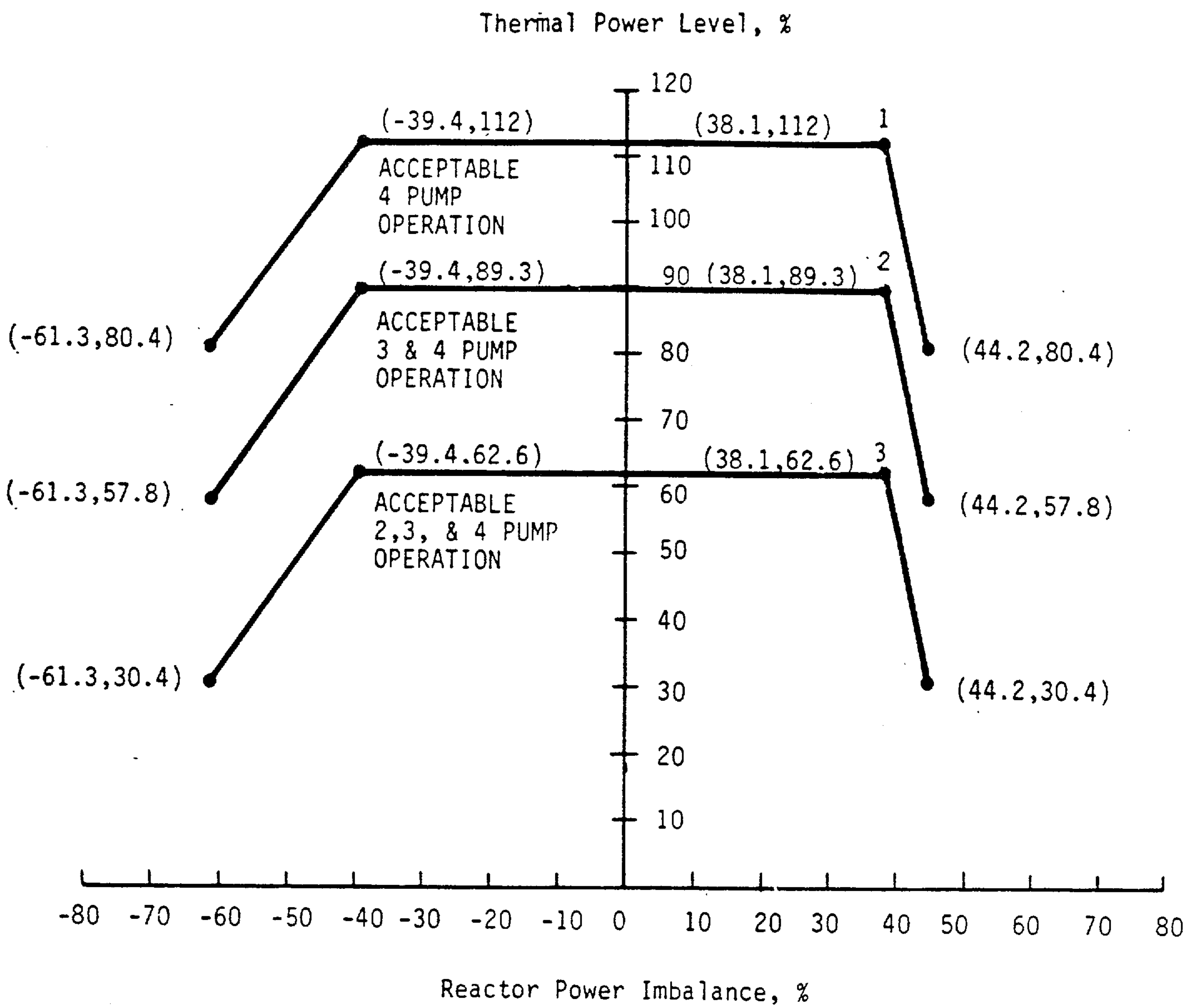
5.3.1.5 The core will have 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding will be zircaloy-4 filled with alumina-boron.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation described in FSAR and shall not exceed an enrichment of 3.5 percent of U^{235}.

5.3.2 REACTOR COOLANT SYSTEM

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements.(4)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670 F.(5)

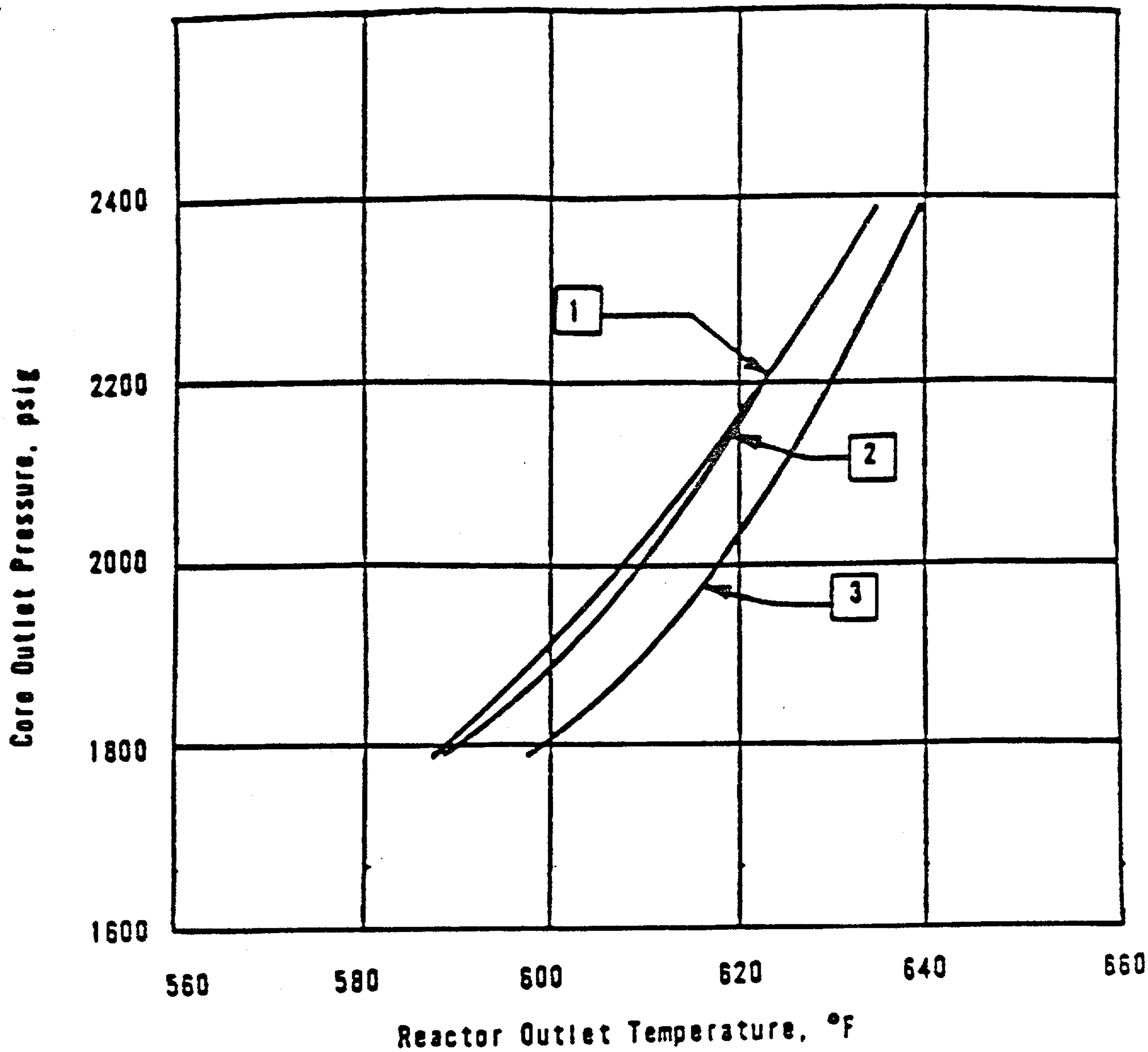


Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

TMI-1

CORE PROTECTION SAFETY LIMITS

Figure 2.1-2



REACTOR COOLANT FLOW

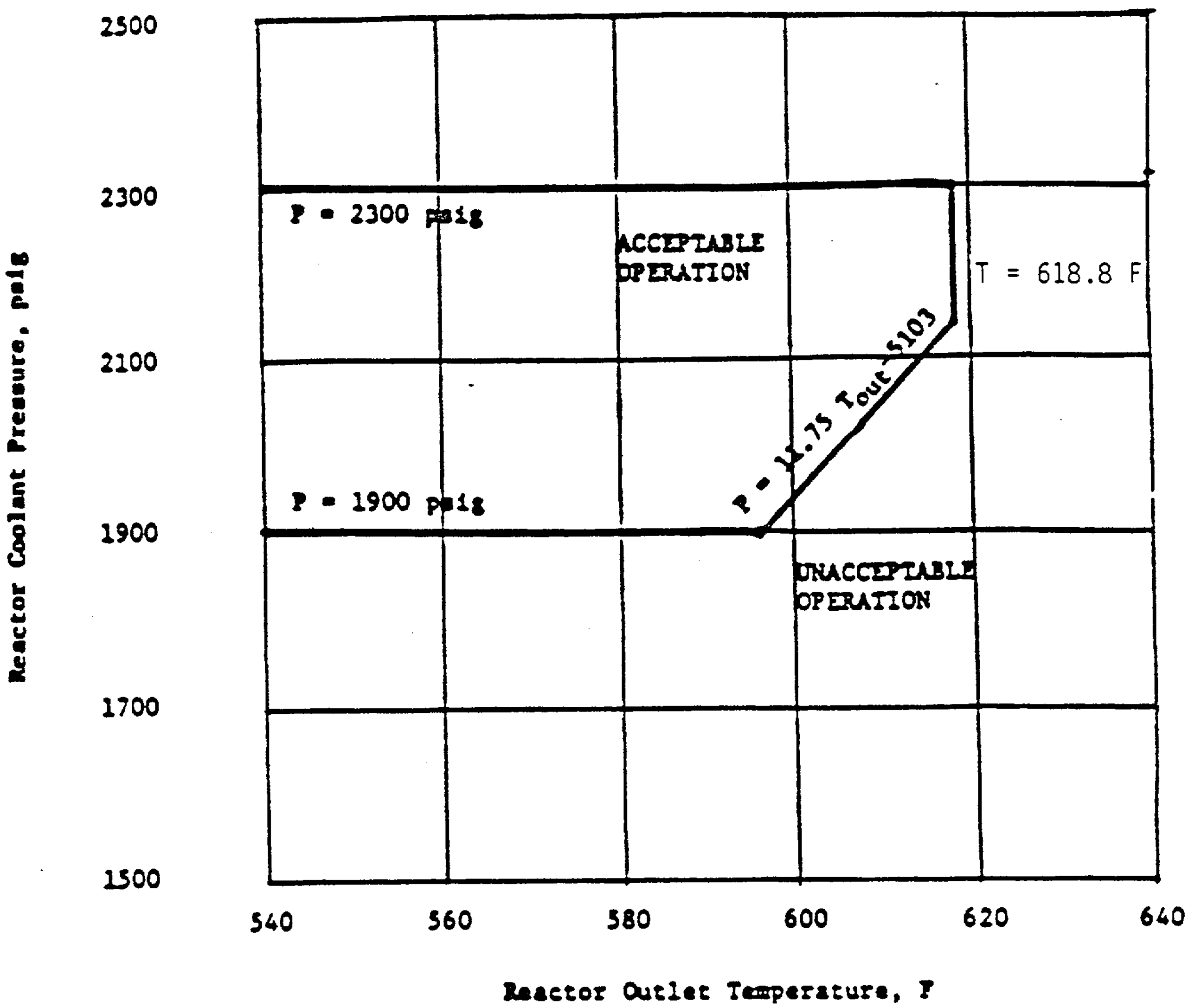
CURVE	(LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	89.3%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	62.6%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

TMI-1

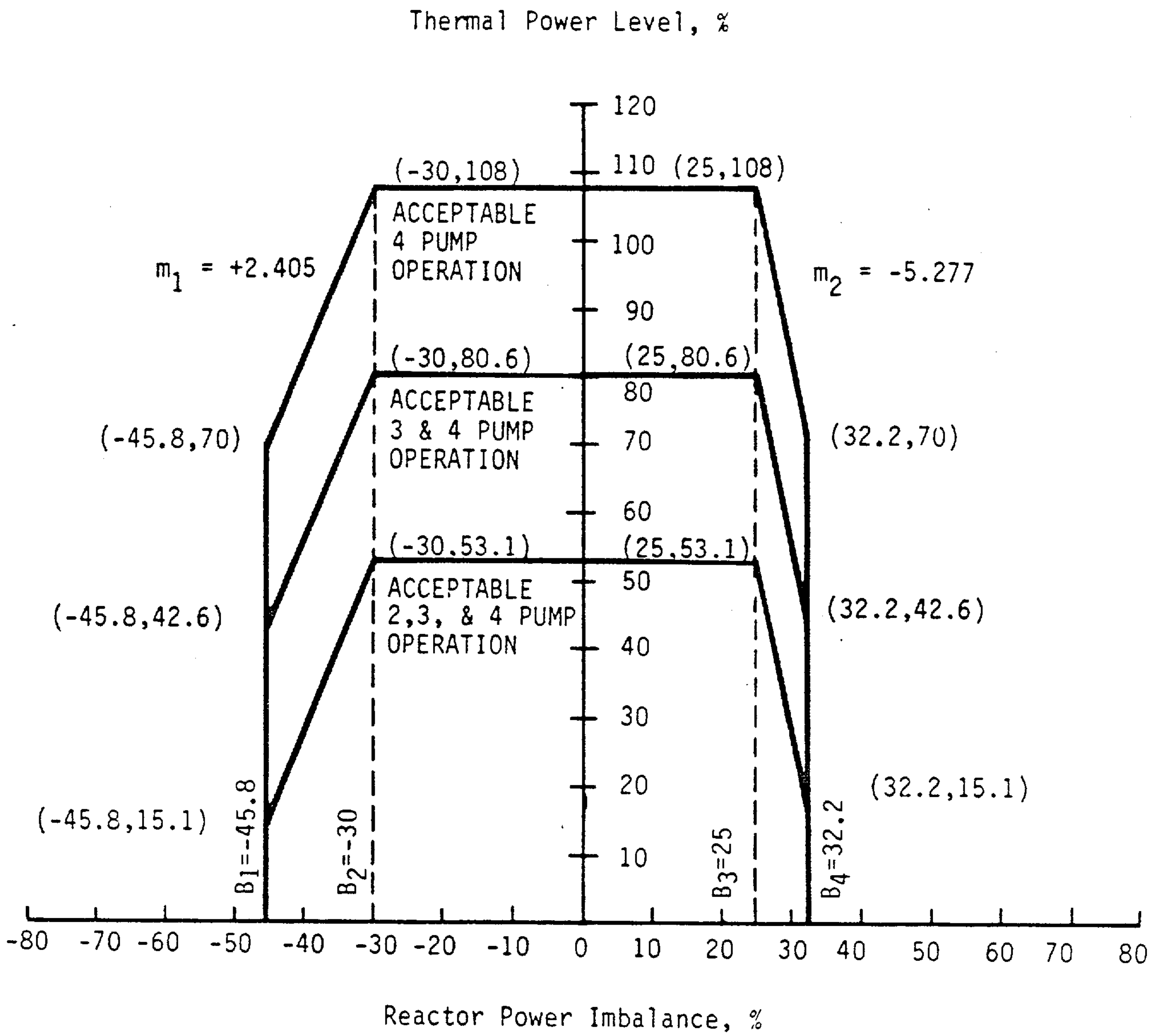
CORE PROTECTION SAFETY BASES

Figure 2.1-3



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

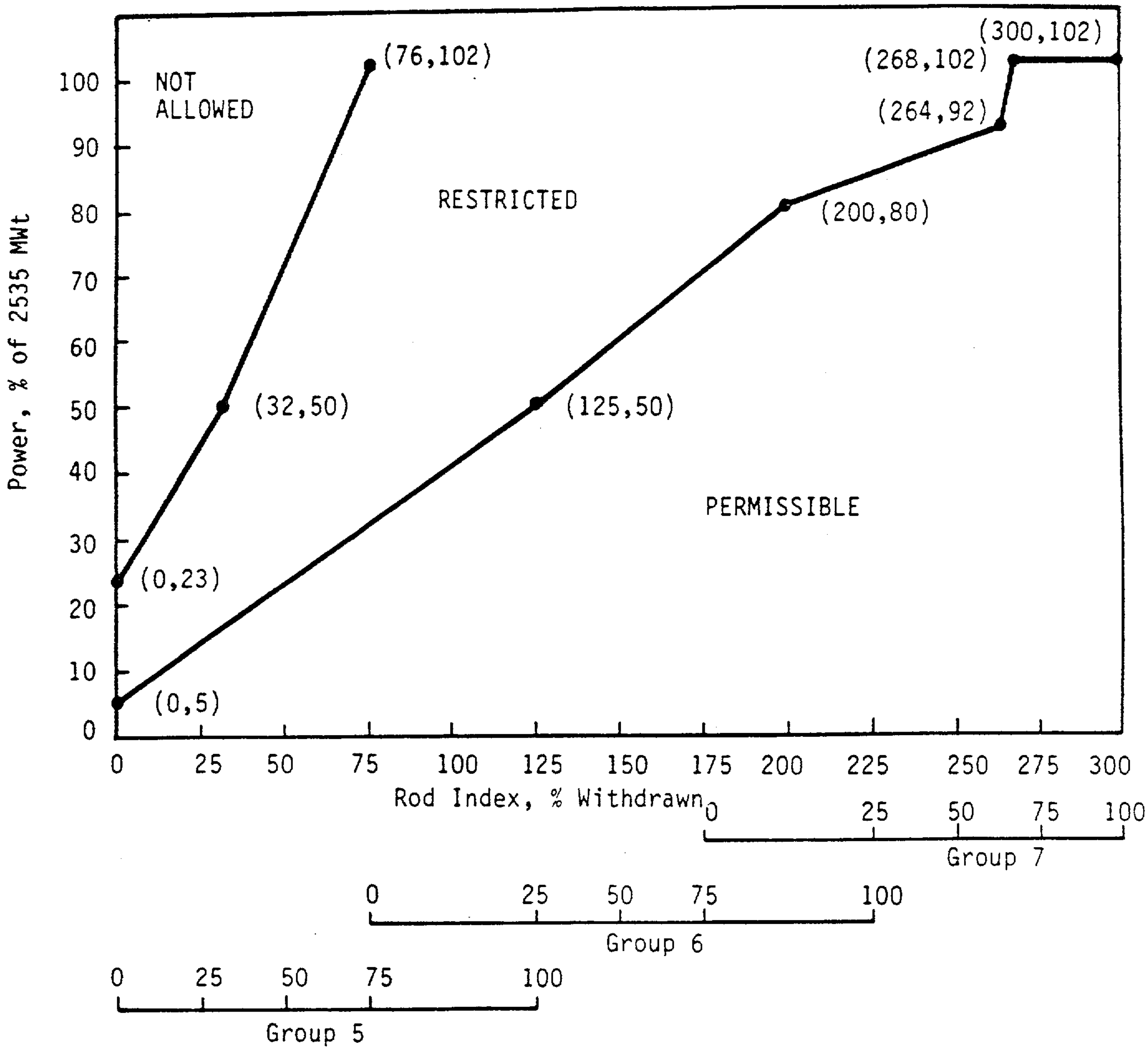
Figure 2.3-1



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS FOR
REACTOR POWER IMBALANCE

TMI-1

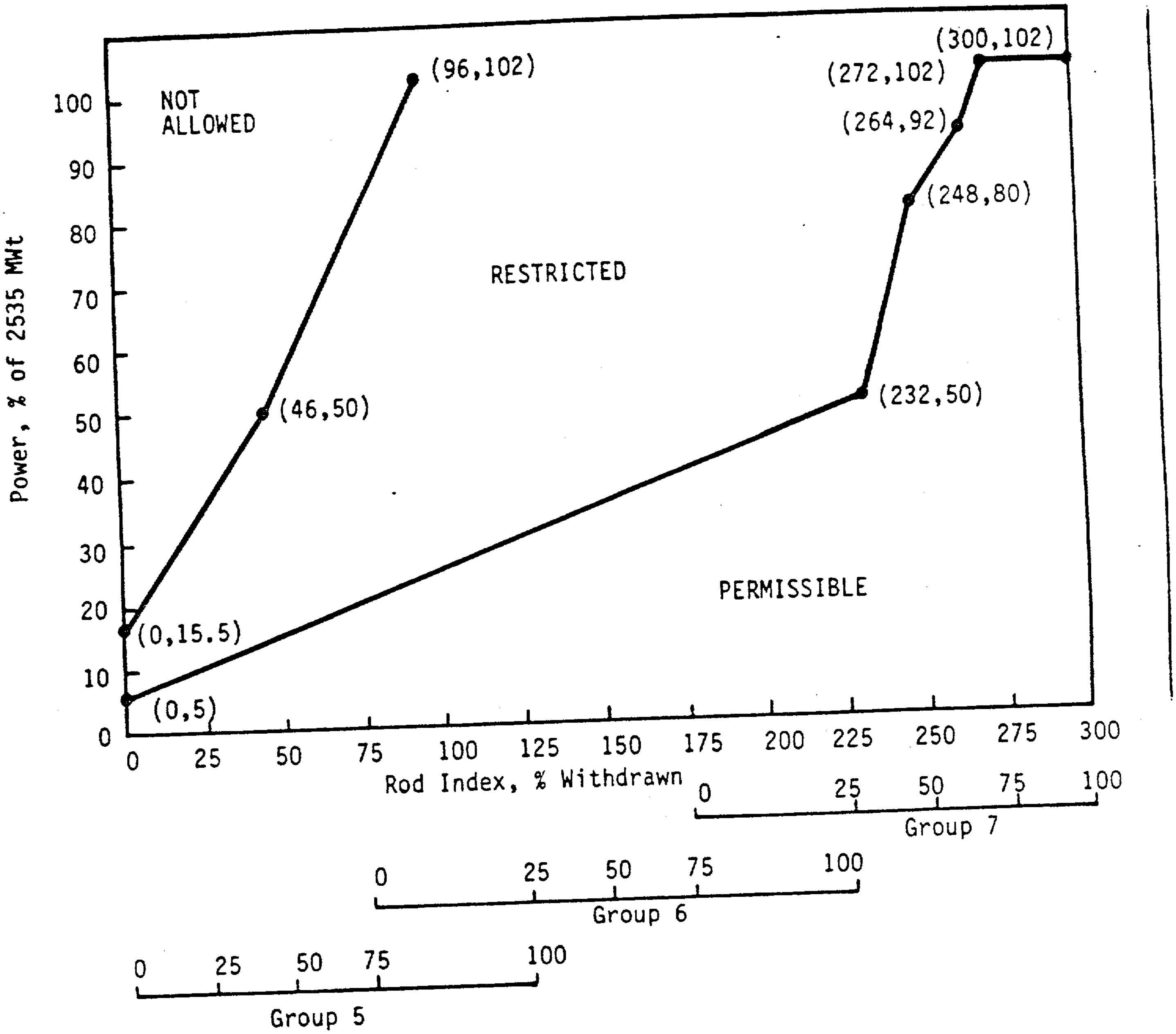
Figure 2.3-2



ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
0 to 30+10/-0 EFPD

TMI-1

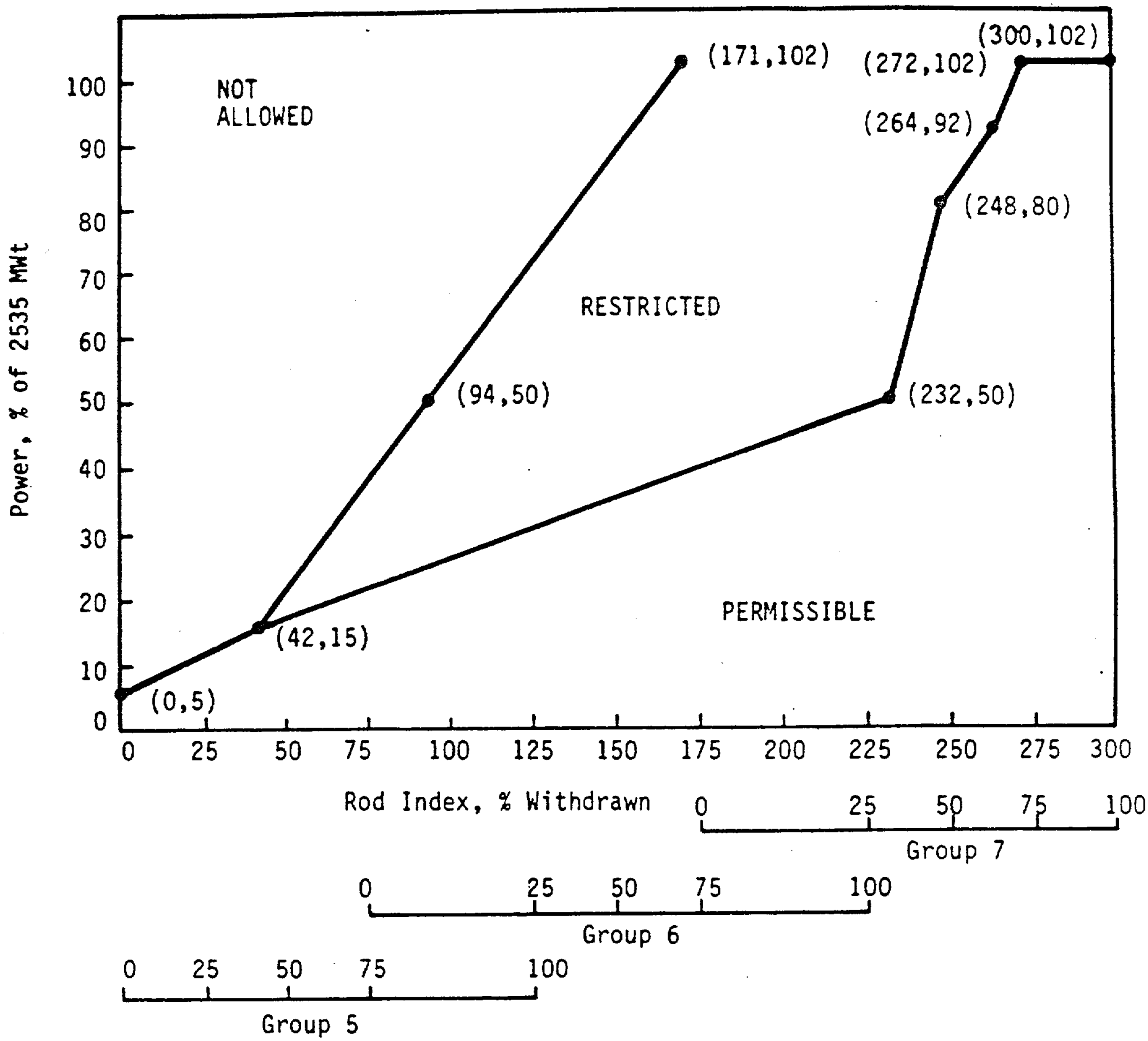
Figure 3.5-2A



ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
30+10/-0 TO 250±10 EFPD

TMI-1

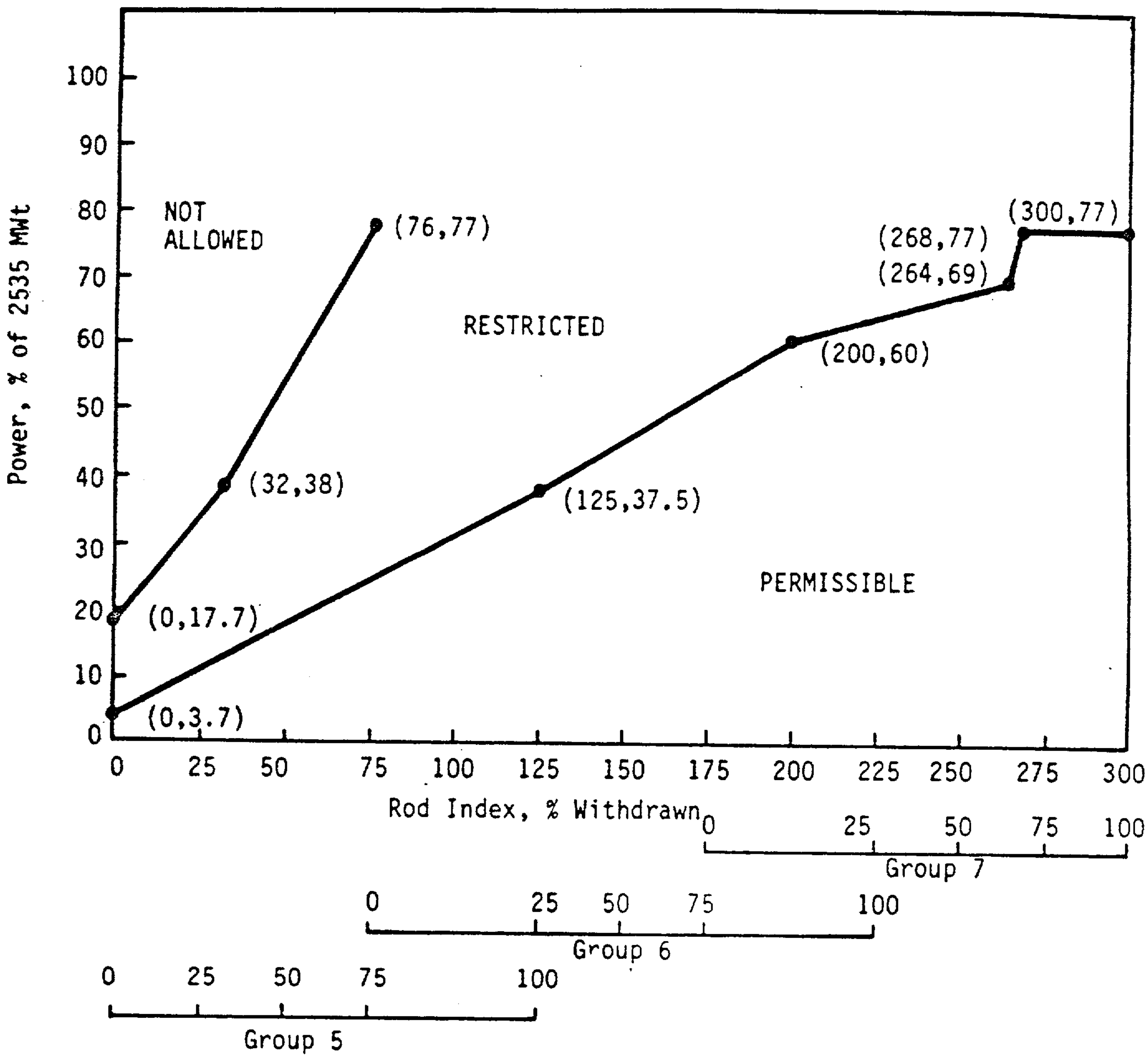
Figure 3.5-2B



ROD POSITION LIMITS FOR
4 PUMP OPERATION AFTER
250±10 EFPD

TMI-1

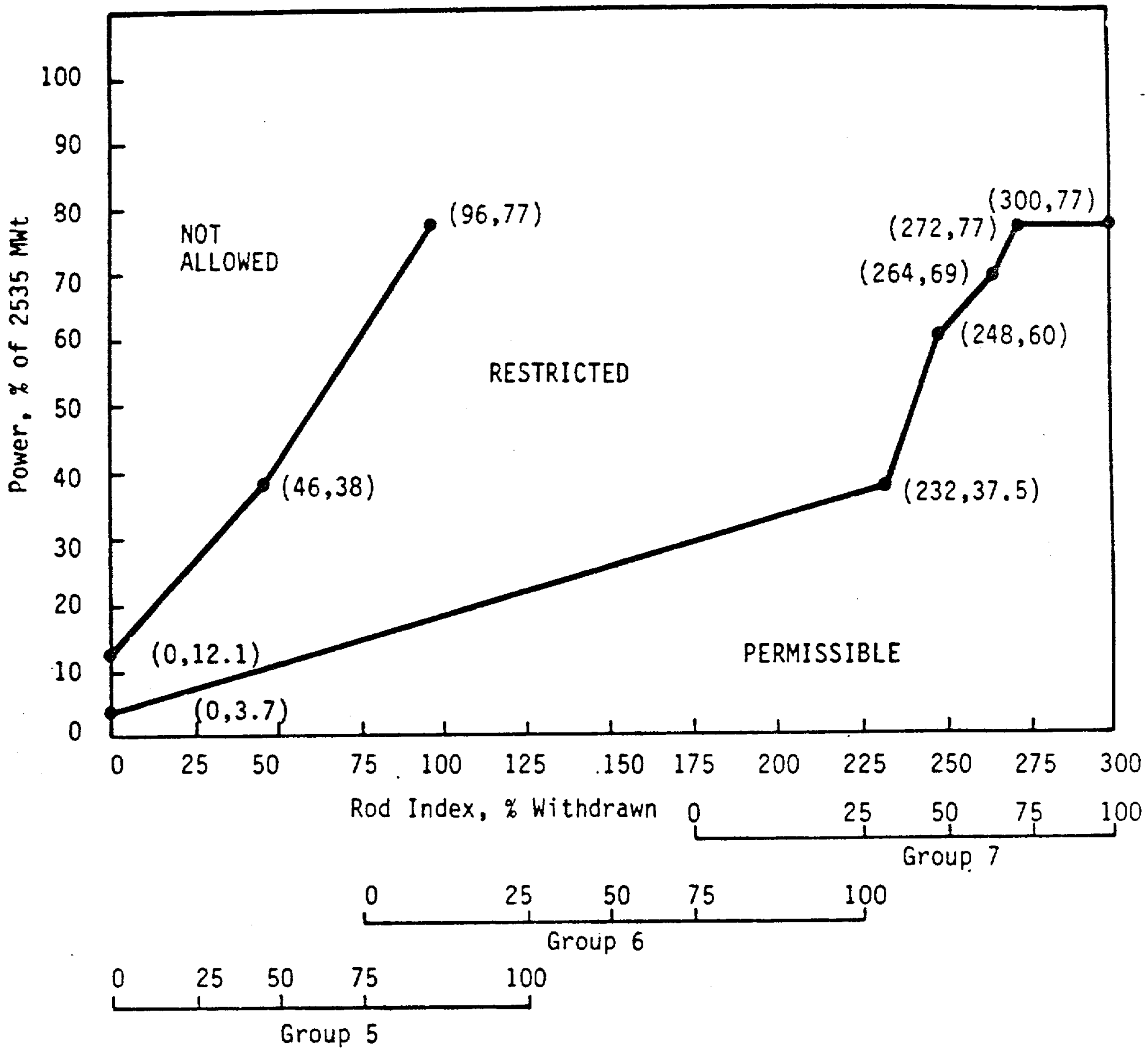
Figure 3.5-2C



ROD POSITION LIMITS FOR
3 PUMP OPERATION FROM
0 TO 30+10/-0 EFPD

TMI-1

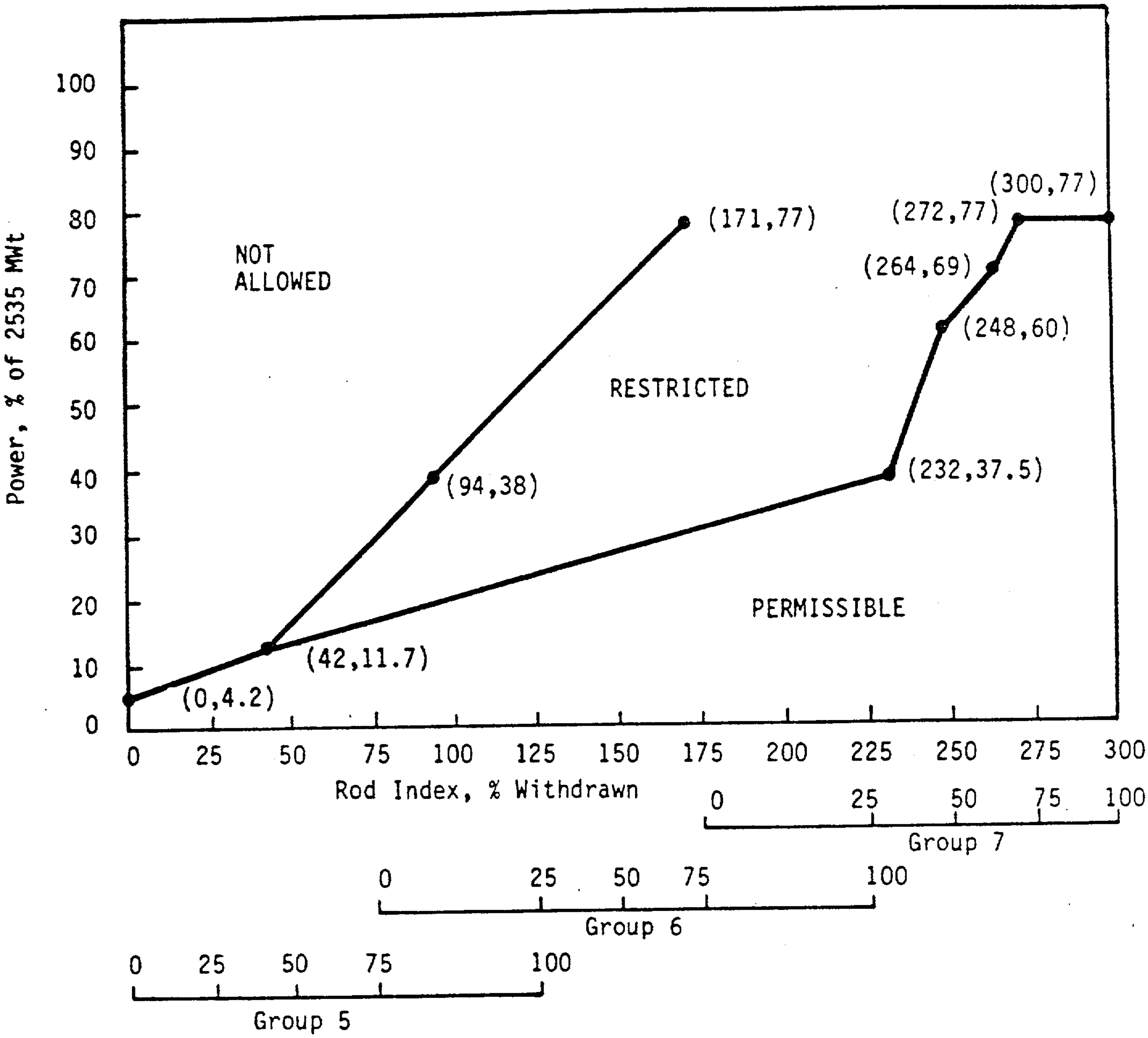
Figure 3.5-2D



ROD POSITION LIMITS FOR
3 PUMP OPERATION FROM
30+10/-0 TO 250±10 EFPD

TMI-1

Figure 3.5-2E

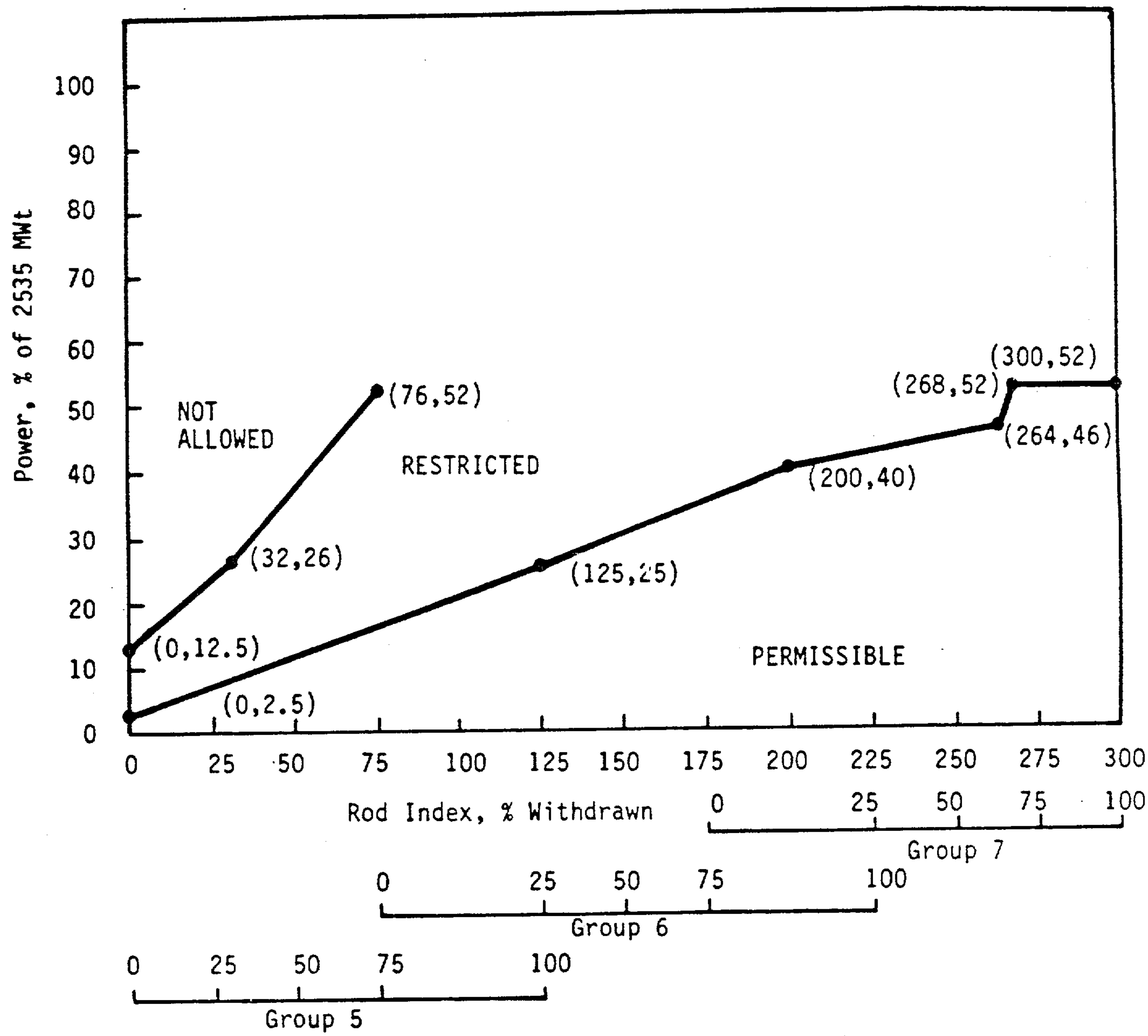


ROD POSITION LIMITS FOR
3 PUMP OPERATION AFTER
250±10 EFPD

TMI-1

Figure 3.5-2F

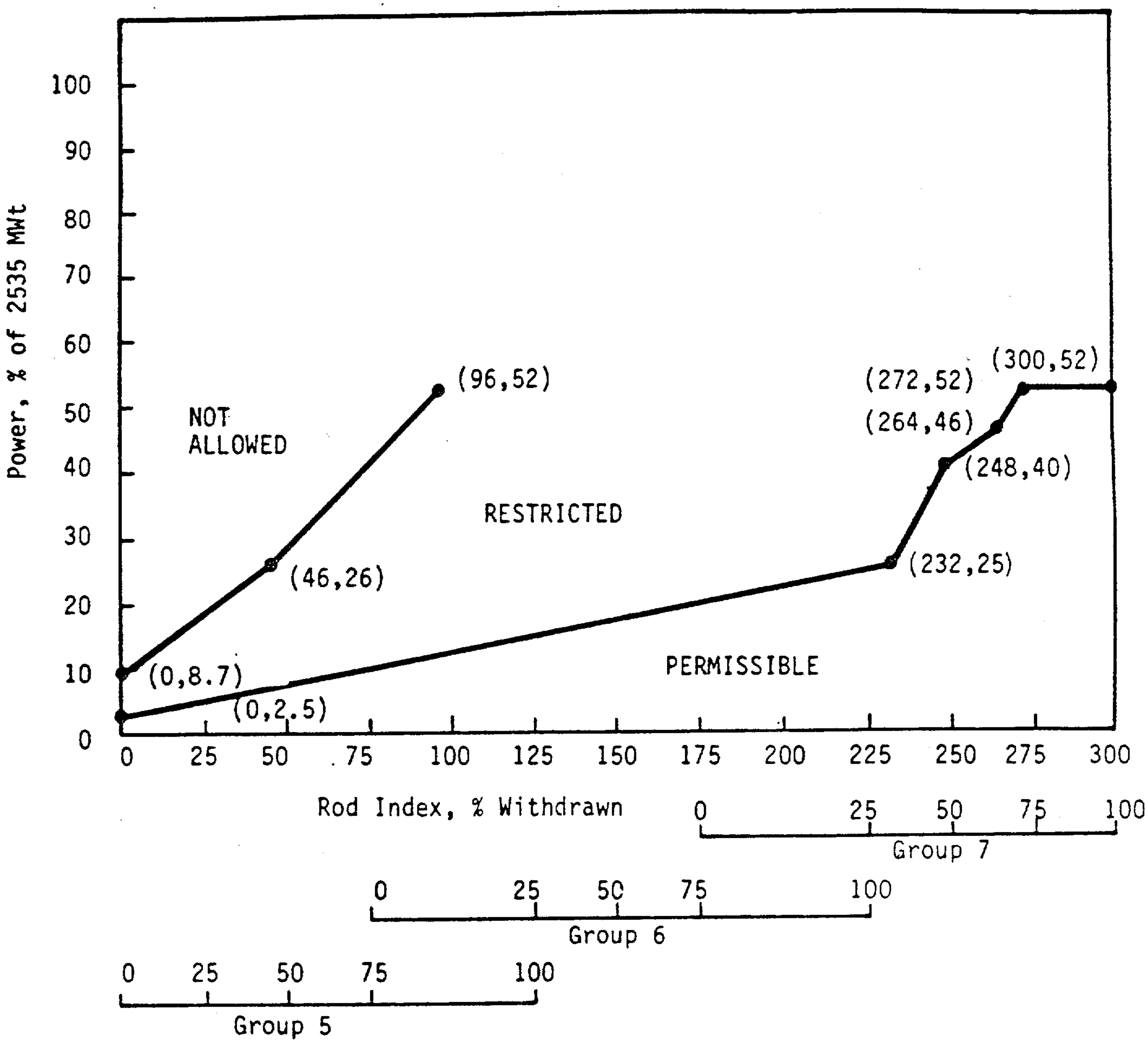
Amendment No. ~~17~~, ~~29~~, ~~39~~, ~~45~~, ~~50~~, ~~120~~, 126



ROD POSITION LIMITS FOR
2 PUMP OPERATION FROM
0 TO 30+10/-0 EFPD

TMI-1

Figure 3.5-2G

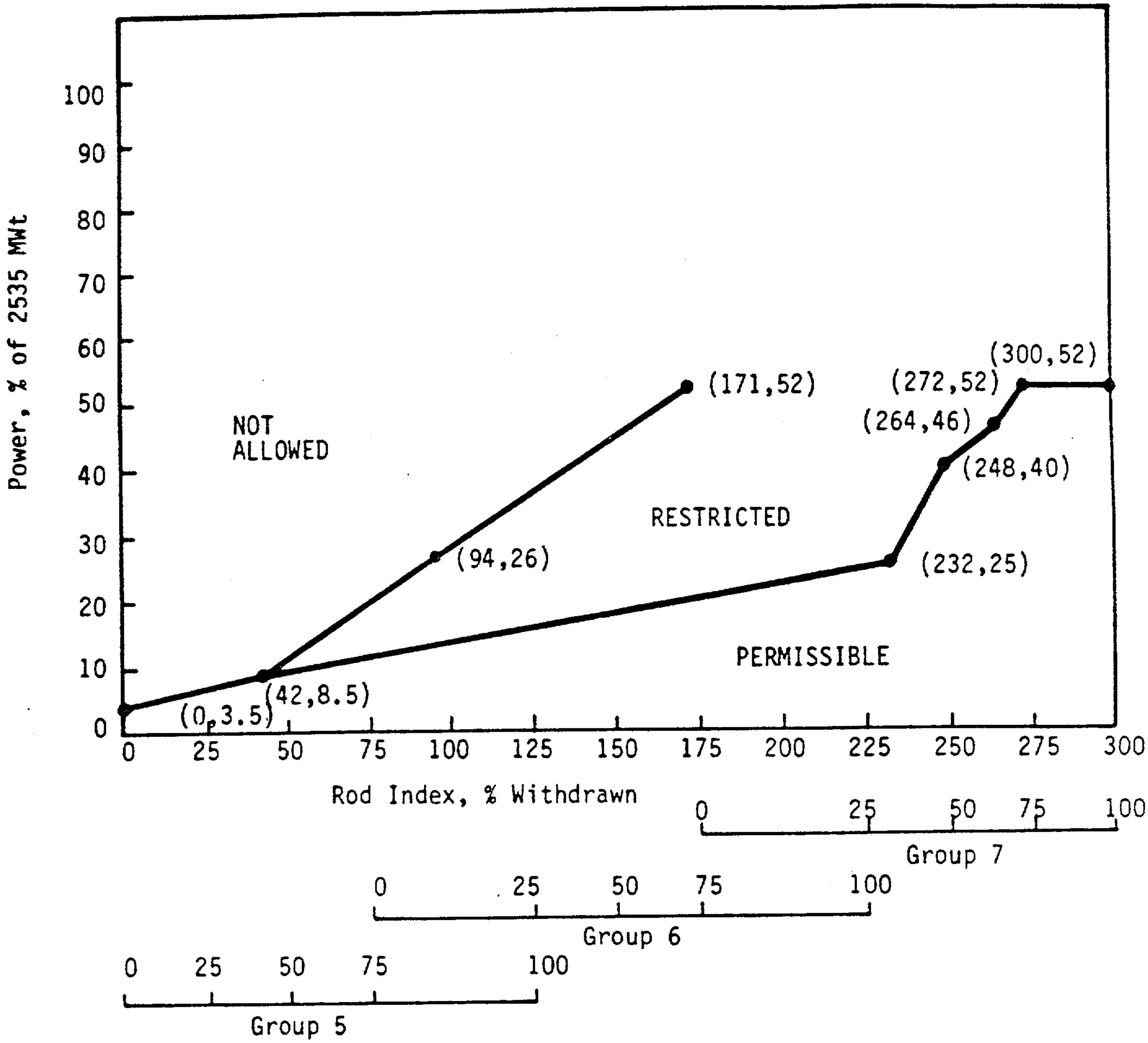


ROD POSITION LIMITS FOR
2 PUMP OPERATION FROM
30+10/-0 TO 250±10 EFPD

TMI-1

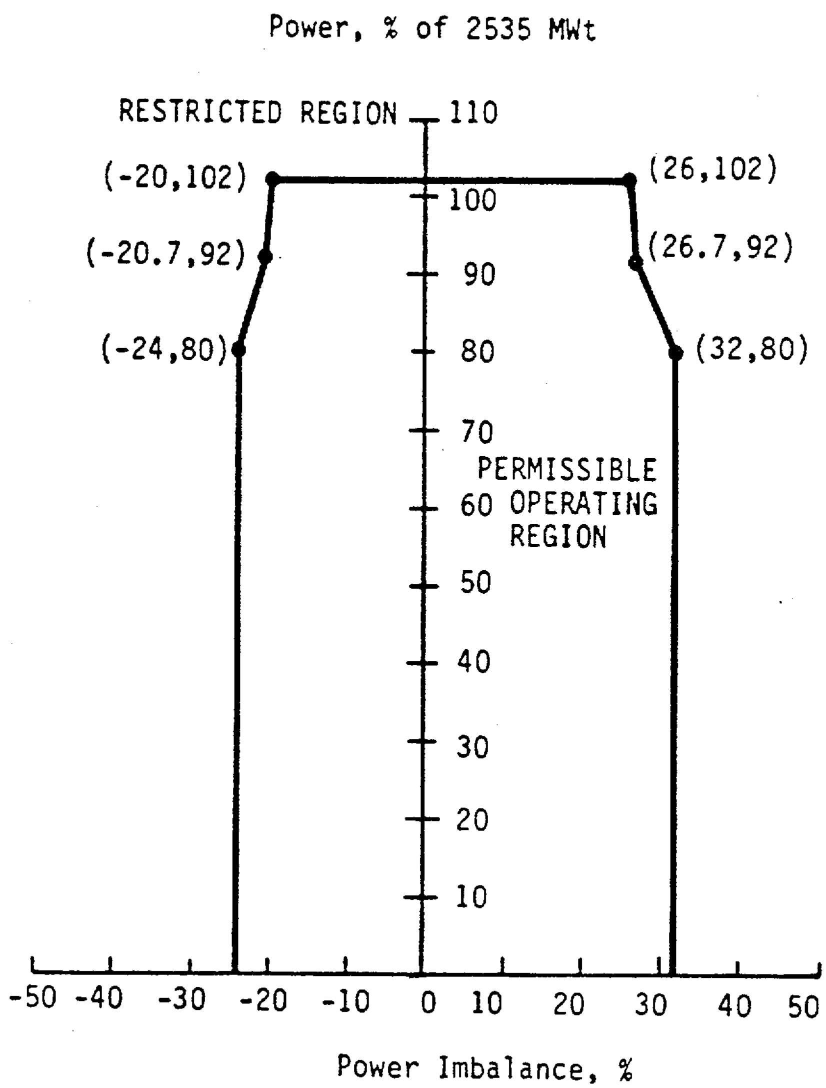
Figure 3.5-2H

Amendment No. ~~29~~, ~~39~~, ~~40~~, ~~43~~, ~~50~~, ~~120~~, 126



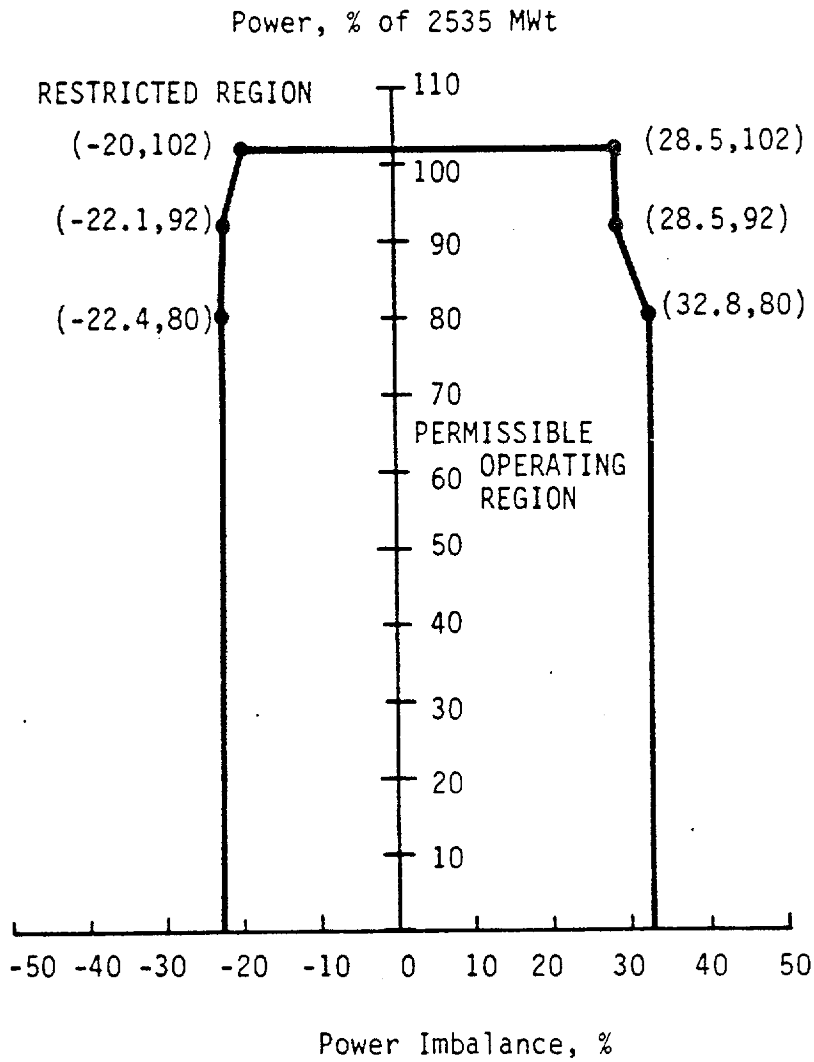
ROD POSITION LIMITS FOR
2 PUMP OPERATION AFTER
250±10 EFPD

TMI-1



POWER IMBALANCE
ENVELOPE FOR OPERATION
FROM 0 TO 30+10/-0 EFPD

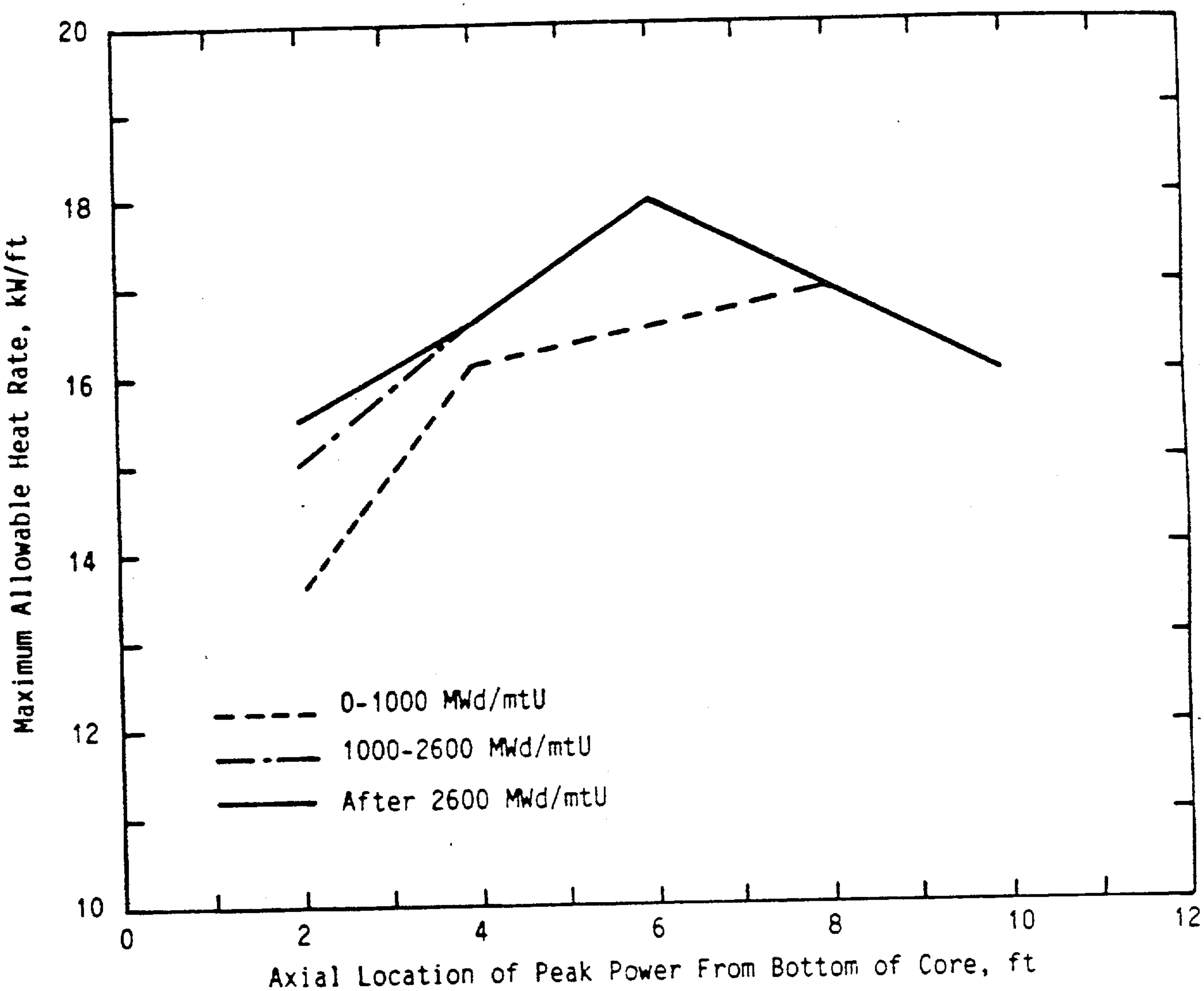
TMI-1



POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 30+10/-0
EFPD

TMI-1

Figure 3.5-2K



LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE

TMI-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated November 3, 1986 (Ref. 1), the GPU Nuclear Corporation (GPU or the licensee) made application to amend the Technical Specifications of the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The proposed changes would modify the Technical Specifications to permit operation for a sixth cycle (Cycle 6). The safety analyses performed and the resulting modifications to the TMI-1 Technical Specifications are described in the Cycle 6 reload report (Ref. 2).

The safety analysis for the previous fifth cycle of operation at TMI-1 is being used by the licensee as the reference cycle for the proposed sixth cycle of operation. Cycle 5 operated with no anomalies that would affect Cycle 6. Where conditions are identical or limiting in the fifth cycle safety analysis, our previous Safety Evaluation (Ref. 3) continues to apply.

By letter dated July 16, 1986, the licensee applied to amend the Technical Specifications for TMI-1. The proposed amendment (Technical Specification Change Request No. 159) would allow withdrawing of the axial power shaping rods under end of Cycle 5 conditions and would change the order of preference for instruments used to measure quadrant power tilt. Withdrawal of the axial power shaping rods was approved in Amendment No. 120 issued September 2, 1986.

Our evaluation of the safety analysis for the TMI-1 Cycle 6 reload and changing the order of preference for instruments used to monitor quadrant tilt follows.

2.0 DESCRIPTION OF THE CYCLE 6 CORE

The TMI-1 core consists of 177 fuel assemblies, each of which is a 15 x 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel management scheme is basically a low-leakage design with loading pattern and enrichments chosen to provide a Cycle 6 length of 425 ± 15 effective full power days (EFPDs). The loading pattern consists of 49 Batch 6 fuel assemblies shuffled to new locations with 41 fuel assemblies distributed in a checkerboard pattern in the interior of the core and 8 fuel assemblies placed on and near the core periphery on the core flats, 52 Batch 7 fuel assemblies shuffled to new locations on and near the core periphery, and 76 fresh fuel assemblies distributed in a checkerboard pattern in the interior of the core. Sixty-four of the fresh fuel assemblies comprise Batch 8A, while 12 comprise Batch 8B. The Batch 6 fuel assemblies are characterized as being twice-burned, and the Batch 7 fuel assemblies are once-burned. The initial fuel enrichment for Batches 6, 7 and 8B is 2.85 weight percent (w/o) uranium-235, while the initial fuel enrichment of Batch 8A is 2.95 w/o uranium-235.

Reactivity control for Cycle 6 will be provided by 61 full-length silver-indium-cadmium control rods, 68 burnable poison rod assemblies (BPRAs) containing varying amounts of B_4C admixed with Al_2O_3, 8 Inconel axial power shaping rods (APSRs), and soluble boron in the coolant. The APSRs have been changed for Cycle 6 to Inconel (a "gray" absorber in contrast to the previous APSR absorber material) and provide for control of the axial power distribution. All of the core locations, except the peripheral core locations, will contain either a control rod (in the once- and twice-burned fuel) or a BPRA (in the fresh fuel).

The licensed core power level is 2535 MWt. The safety analysis provided in the reload report (Ref. 2) demonstrates the safe operation of TMI-1 throughout Cycle 6 at full power. The following sections describe our evaluation of the safety analysis.

3.0 EVALUATION OF THE FUEL SYSTEM DESIGN

3.1 Fuel Assembly Mechanical Design and Gray APSR Design

The 76 Babcock & Wilcox (B&W) Mark B4 15 x 15 fuel assemblies to be loaded as Batches 8A and 8B for Cycle 6 operation are mechanically interchangeable with Batches 6 and 7 fuel assemblies previously loaded at TMI-1. The cladding stress, strain, and collapse analyses are bounded by conditions previously analyzed for TMI-1 or were analyzed specifically for Cycle 6 using methods and limits previously reviewed and approved by the NRC. The licensee similarly evaluated the gray APSRs that are to be inserted in Cycle 6 and established that the gray APSR design met applicable limits on cladding stress, strain, and collapse.

3.2 Fuel Rod Design

All batches in the TMI-1 Cycle 6 core utilize the same B&W Mark B4 fuel design, and the Batches 8A and 8B fuel parameters are virtually identical to the previously loaded Batches 6 and 7 except that the enrichment of Batch 8A fuel has been increased from 2.85 to 2.95 w/o uranium-235.

Resinter tests were performed on all fuel pellet lots comprising Batches 8A and 8B fuel assemblies. The tests were based on a modified sampling plan and are described in References 4 and 5. The resinter tests confirm the conservatism of densification characteristics assumed in the TACO2 (Ref. 6) analyses. The licensee states that the results of the TMI-1 Fuel Densification Report (Ref. 7) remain bounding for all Cycle 6 fuel since those analyses were based on a lower initial pellet density and an assumption of densification to 96.5% of theoretical density. We have reviewed the information presented in References 4 and 5 on the resintering tests based on a modified sampling plan and find it acceptable.

3.3 Fuel Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 8) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. Our current criterion is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. GPU has stated that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 6. This analysis is based on the use of the approved B&W TACO2 code (Ref. 6). The staff concludes that the fuel rod internal pressure limit has been acceptably considered for Cycle 6 operation.

3.4 Fuel Thermal Design

There are no major changes between the thermal design of the new Batches 8A and 8B fuel and previous batches that will be reinserted in the Cycle 6 core. The thermal design analyses were performed with the approved TACO2 code (Ref. 6). The Cycle 6 core protection limits are based on a linear heat generation rate (LHGR) to centerline fuel melt of 20.5 kW/ft, which is applicable to all Cycle 6 fuel batches. The results of the thermal design evaluation show no significant differences between the new Batches 8A and 8B fuel and the previous Batches 6 and 7 fuel. We have reviewed the fuel thermal design parameters for normal operation and find them acceptable.

3.5 Loss of Coolant Accident (LOCA) Initial Conditions

In addition to the steady-state conditions, the average fuel temperature as a function of LHGR and lifetime fuel pin pressure data used in the LOCA analysis (see Section 7.2 of Reference 2) are also calculated with the TACO2 code (Ref. 6). The reload report (Ref. 2) states that the fuel temperature and fuel pin pressure data used in the generic LOCA analysis (Ref. 9) are conservative compared to those calculated for TMI-1 Cycle 6. The bounding values of the allowable LOCA LHGRs (see Table 7.2 of Reference 2) include the effects of NUREG-0630 (Ref. 10) regarding fuel cladding swelling and rupture behavior during LOCA.

3.6 Conclusion On Cycle 6 Fuel System Design

We have reviewed the fuel system design and analysis for Cycle 6 operation and find it acceptable, as discussed above.

4.0 EVALUATION OF THE NUCLEAR DESIGN

To support Cycle 6 operation of TMI-1, the licensee has provided analyses using analytical methods and design bases established in licensing topical reports that have been approved by the NRC. The licensee has provided a comparison of the core physics parameters for Cycles 5 and 6 as calculated with these approved methods. The parameters for Cycle 5 were generated using PDQ07 (Ref. 11), while the parameters for Cycle 6 were generated using the NOODLE code (Ref. 12). The two codes give comparable results when compared to measured data. There are differences in the neutronic parameters compared between Cycles 5 and 6. These differences can be attributed to differences in cycle lengths, BPRA loading, and fuel loading pattern. All of the transients and accidents analyzed in the Final Safety Analysis Report (FSAR) were reviewed for Cycle 6 operation. The Cycle 6 parameters were conservative when compared to analyses accepted for previous cycles, and no new transient and accident analyses are included in the reload report (Ref. 2).

The control rod worths and shutdown margin requirements at beginning-of-cycle (BOC) and at the most limiting time at end-of-cycle (EOC) for the Cycle 6 nuclear design are presented in Table 5-2 of Reference 2. At EOC 6, the reactivity worth with all control rods inserted, assuming that the highest worth control rod is stuck out of the core, is 6.48%. This reactivity worth also assumed a reduction in worth of 10% for uncertainty and a reduction in worth due to control rod absorber burnup. The reactivity worth required for shutdown, including the contribution required to accommodate the reactivity effects of the steamline break event at EOC 6, is 4.85%. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break event at the worst time in core life allowing for the most reactive control rod stuck in the full withdrawn position and allowing for calculational uncertainties and control absorber burnup. We have reviewed the calculated control rod worths and uncertainties in these worths based upon comparisons of calculations with experiments in other B&W reports. On the basis of this review, we conclude that GPU's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod is stuck in the full withdrawn position.

We conclude that the licensee's predicted neutronic parameters are acceptable because they were obtained using approved methods, the validity of which has been demonstrated through many cycles of predictions, including startup tests, for this and other reactors. As a result of this review of the neutronic parameters compared to previous cycles, we concur with the licensee's conclusions regarding the Cycle 6 transient and accident analyses.

The licensee has made a number of changes in the nuclear design of Cycle 6. These changes are: (1) the increase in cycle lifetime to 425 EFPDs with the incorporation of BPRAs to aid in reactivity control, (2) the use of gray APSRs, (3) the use of the NOODLE code to calculate the physics parameters for Cycle 6, and (4) the removal of the power level hold requirements for xenon in Technical Specifications 3.5.2.4.d and 3.5.2.5.c. The effects of the change in the cycle lifetime, of the use of BPRAs, and of the use of gray APSRs have all been taken into account in the nuclear design. In particular, the licensee verified that the gray APSRs provide adequate axial power distribution control. The NOODLE code has been reviewed and approved by the NRC staff (Ref. 13). An extensive analysis has been performed by B&W for the licensee (Ref. 14) to justify removal of the power level cut-off requirements. This power level cut-off had been utilized to accommodate transient xenon effects on power peaking factors before ascending to 100% power. The analysis showed that the 5% total xenon factor applied in the computation of LOCA margin provides conservative operating limits. The 2.5% radial xenon factor applied in the evaluation of initial condition departure from nucleate boiling (DNB) margin was also shown to be conservative. We conclude that these changes in the Cycle 6 nuclear design are acceptable since the nuclear design and resulting Technical Specifications for Cycle 6 include the effects of the changes calculated with approved methods.

5.0 EVALUATION OF THE THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design of Cycle 6 is nearly identical to that of Cycle 5 as shown in the comparison of maximum design conditions in Table 6-1 of Reference 2. The thermal-hydraulic design evaluation utilizes, for the first time, the LYNX series of codes (Refs. 15, 16 and 17) for crossflow modeling for DNB predictions. These LYNX codes have been reviewed and approved by the NRC staff. The application of cross-flow modeling for reload cores is described in Reference 18. The fresh Batches 8A and 8B fuel assemblies are hydraulically and geometrically similar to irradiated Batches 6 and 7 fuel assemblies. No departure from nucleate boiling ratio (DNBR) penalty is required since the approved rod bow topical report (Ref. 19) shows that the reduction in power production capability more than offsets any rod bow effects as burnup increases. The bypass flow for Cycle 6 decreased to 7.6% from 10.4% for Cycle 5 because of the insertion of the 68 BPRAs. The thermal-hydraulic analysis used, however, a conservative value of 8.4% for the bypass flow. Based on the thermal-hydraulic similarities of Cycle 6 with Cycle 5 and the use of approved models and methods, we conclude that the thermal-hydraulic design of Cycle 6 is acceptable.

6.0 EVALUATION OF TRANSIENT AND ACCIDENT ANALYSES

The licensee has examined each FSAR transient and accident analysis with respect to changes in the Cycle 6 parameters to ensure that the calculated consequences still meet applicable criteria. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, the thermal-hydraulic parameters, and the physics static and kinetic parameters. Fuel thermal analysis values are listed in Table 4-1 of Reference 2 for all fuel batches in Cycle 6. Table 6-1 of Reference 2 compares the thermal-hydraulic parameters for Cycles 5 and 6. These parameters are either the same for both cycles or exhibit differences due to, for example, modeling changes. The physics parameters are provided in Table 5-1 of Reference 2. A comparison of key kinetics parameters from the FSAR and the densification report with predicted Cycle 6 values is provided in Table 7-1 of Reference 2. These data indicate that the FSAR data are bounding for most of the parameters. For those parameters not bounded by the FSAR data, the licensee states that their effect on affected transients or accidents will produce less severe consequences than in previous bounding analyses. The effects of fuel densification on the FSAR accident analyses have also been evaluated.

A generic LOCA analysis for the B&W 177-fuel assembly, lowered loop plant design has been performed using the Final Acceptance Criteria (FAC) emergency core cooling system (ECCS) evaluation model (Ref. 9 as updated by Refs. 20 and 21). That analysis used the limiting values of the key parameters for all plants in the 177-fuel assembly, lowered loop class and is, therefore, bounding for TMI-1 Cycle 6 plant operation. Table 7-2 of Reference 2 presents the limiting values of the allowable LOCA LHGRs for TMI-1 Cycle 6 fuel.

The radiological dose consequences of the accidents presented in the FSAR have been reevaluated for Cycle 6. The reason for the reevaluation is the increased amount of energy produced by fissioning plutonium caused by the extended cycle fuel management strategy. The bases used in the radiological dose evaluation are the same as in the FSAR except for two factors: (1) the fission yields and half-lives used in the Cycle 6 evaluation are based on more current data, and (2) the steam generator tube rupture (SGTR) accident considers the increased amount of steam released to the environment because of a post-TMI modification. The radiological doses are still a small fraction (10%) of the 10 CFR Part 100 limits and are consistent with those of the reference cycle.

We conclude from the review of Cycle 6 core thermal and kinetic parameters, with respect to previous cycle values and with respect to the FSAR values, that this reload core will not adversely affect TMI-1 plant's ability to operate safely during Cycle 6.

7.0 TECHNICAL SPECIFICATIONS

As indicated in our evaluation of the nuclear design, provided in Section 4, the operating characteristics of Cycle 6 were calculated with well-established, approved methods. The proposed Technical Specifications are the result of the cycle-specific analyses for power peaking, control rod worths, and quadrant tilt allowance. The analyses performed include the implementation of a low leakage fuel shuffle pattern, the implementation of a crossflow thermal-hydraulic analysis, and the removal of the power level hold requirements for transient xenon. The removal of the power level cut-off to accommodate transient xenon effects is discussed in Section 4. We conclude that the Technical Specification changes proposed by the licensee in Reference 1 and repeated in Section 8 of the Cycle 6 reload report (Ref. 2) are acceptable. The proposed Technical Specification changes are as follows:

1. Pages vii and viii are changed to accommodate new Cycle 6 figures. Since these changes are administrative, they are acceptable.

2. Basis page 2-2

   The description of the curve on Figure 2.1-1 is changed to indicate that the curve no longer represents minimum DNBR conditions. This change is acceptable since it more accurately describes the curve presented on the figure.

   The axial peaking factor is changed to 1.65 (and concomitantly the total peaking factor is changed to 2.82). These changes are acceptable since they are reflected in the Cycle 6 analysis and, in particular, the implementation of the crossflow model in the DNBR analysis.

   The centerline fuel melt LHGR is changed to 20.5 kW/ft. This change is acceptable since it is reflected in the Cycle 6 analysis and, in particular, the change to the TACO2 fuel analysis methodology.

3. Basis page 2-3

   The description of the curves on Figure 2.1-3 is changed to indicate that the curves no longer represent minimum DNBR conditions. This change is acceptable since it more accurately describes the curves presented on the figure.

The thermal powers for three pump operation are truncated from two decimal to one decimal figure. This change is acceptable since it reflects more realistically the accuracy in measured thermal powers.

References to FSAR sections are changed. These changes are administrative since they reflect the updated FSAR.

4. Figure 2.1-2, Core Protection Safety Limits

   The curves on the figure present the power-imbalance envelopes, for various pump operation, that meet the centerline fuel melt and DNBR criteria. These curves are derived from Figure 2.3-2 by removal of instrumentation error and calculational uncertainties in flux/flow determinations. The curves reflect widening of the power-imbalance envelopes as well as increased Reactor Protection System instrument errors. Figure 2.1-2 is acceptable since it reflects Cycle 6 analysis by being derived from Figure 2.3-2.

5. Figure 2.1-3, Core Protection Safety Bases

   The curves on this figure have been revised to reflect small changes in the 2 and 3 pump allowable power levels. This change is acceptable since it reflects the revised Reactor Protection System instrument errors.

6. Page 2-5, Basis For Technical Specification 2.3.1

   The nuclear overpower at which a reactor trip occurs has been revised slightly. This change is acceptable since it reflects the revised Reactor Protection System instrument errors.

7. Page 2-6, Basis For Technical Specification 2.3.1

   The power level at which trip occurs for three pump operation has been revised. This change is acceptable since it merely reflects the truncation of a number to 80.6 instead of 80.7.

8. Page 2-7, Basis For Technical Specification 2.3.1

   The high reactor coolant outlet temperature trip setting limit has been changed to 618.8°F from 619°F. This change is acceptable since it reflects the revised Reactor Protection System instrument error, which in this case is 1.2°F instead of the previous 1°F.

9. Page 2-9, Table 2.3-1, Reactor Protection System Trip Setting Limits

   This table is changed to reflect the overpower and high temperature trip setpoint changes discussed in Items 6 and 8 above.

These changes are acceptable for the reasons stated in Items 6 and 8 above.

10. Figure 2.3-1, Reactor Protection System Maximum Allowable Setpoints

This figure has been revised to reflect the change in the high reactor coolant outlet temperature trip setting discussed in Item 8 above and is, therefore, acceptable.

11. Figure 2.3-2, Reactor Protection System Maximum Allowable Setpoints for Power Imbalance

The figure is based on the Cycle 6 analyses discussed in previous sections of this Safety Evaluation. A flux/flow setpoint has been maintained to provide additional margin for the Cycle 6 pump coastdown analysis. Since the power/imbalance limits on the figure conservatively bound the thermal limits, they are acceptable.

12. Technical Specification 3.5.2.4, Quadrant Tilt

Technical Specification 3.5.2.4.a has been changed to reflect an adjustment in the incore detector uncertainty factor caused by depletion. Therefore, the change in the allowable quadrant tilt to +4.12% from +3.52% is acceptable.

Technical Specification 3.5.2.4.d has been changed to delete reference to the power level cut-off for transient xenon. This is acceptable as discussed in Section 4 above.

Technical Specification 3.5.2.4.e.1 has been changed for consistency with Technical Specifications 3.5.2.4.e.2 and 3.5.2.4.e.3. This change is acceptable since it is administrative in nature.

13. Page 3-34a

Technical Specifications 3.5.2.4.e.2 and 3.5.2.4.e.3 are being changed to incorporate new control rod group limits and power imbalance limits derived from the Cycle 6 safety analysis. These changes are, therefore, acceptable.

Technical Specification 3.5.2.4.f is changed to revise the order of preference for selecting the system that shall be used to determine quadrant power tilt. The purpose of the change is to insure that the most accurate system available is used to determine quadrant tilt. It does not change any setpoint, required system accuracy, or surveillance interval. Therefore, we conclude that the change is acceptable.

14. Page 3-35

Technical Specification 3.5.2.5.6 has been changed to delete reference to the APSPs. The Cycle 5 figures for APSR position limits are also deleted. These changes are acceptable since the Cycle 6 safety analysis concludes, and we concur, that no position limits are required for the Cycle 6 gray APSRs.

Technical Specification 3.5.2.5.c has been deleted since, as discussed in Section 4 above, no power level cut-off for transient xenon is required for Cycle 6.

Technical Specification 3.5.2.5.d specifies the new power imbalance limits for Cycle 6 with renumbered figures. This is acceptable since renumbering the figures is administrative and the figures are based on the Cycle 6 safety analysis.

Technical Specification 3.5.2.7 reflects a new figure number. This change is acceptable since it is administrative.

15. Page 3-35a, Bases 3.5.2

A sentence has been added to the bases to indicate that the effect of the gray APSRs has been included in the derivation of the power imbalance limits. This change is acceptable since APSR position limits are no longer specified for Cycle 6.

Item f has been included in Bases 3.5.2 to indicate that the loss of coolant flow transient is limiting for portions of Cycle 6 rather than LOCA. This is acceptable since it reflects the Cycle 6 safety analysis.

Bases 3.5.2 also includes changes to various figure numbers. These changes are acceptable since they are administrative.

16. Page 3-36, Bases 3.5.2

A sentence has been added to explain the 16.8% quadrant tilt value used in Technical Specification 3.5.2.4. This change is acceptable since it provides a clarification of the quadrant tilt Technical Specification.

17. Page 3-36a, Bases 3.5.2

The trip setpoint for Test Power greater than 75% power has been changed to 105.1%. This is acceptable since it is based on revised Reactor Protection System errors.

Other changes on the page are administrative and, therefore, acceptable.

18. Technical Specification 3.5.2.5, Control Rod Positions

New control rod insertion limits (Figure 3.5-2A through Figure 3.5-2I) are provided for 4, 3 and 2 pump operation, as well as a function of burnup interval. These Figures are acceptable since they are based on the Cycle 6 safety analysis discussed previously.

19. Technical Specification 3.5.2, Reactor Power Imbalance

New reactor power imbalance limits (Figure 3.5-2J and Figure 3.5-2K) are provided for two different burnup intervals. These Figures are acceptable since they are based on the Cycle 6 safety analysis discussed previously.

20. Figure 3.5-2L, LOCA Limited Maximum Allowable Linear Heat Generation Rate

For Cycle 6, the LOCA kW/ft limits have been changed due to revisions in the LOCA analysis since Cycle 5. LOCA limits are provided for three burnup intervals. These changes to the LOCA limits are acceptable since they are based on approved changes to the LOCA analysis.

21. Technical Specification 5.3.1, Reactor Core

Changes are made to this Technical Specification to more accurately reflect the nominal design features of the core including the active fuel assembly length, the average enrichment of the current core, and the use of gray APSRs. These changes are acceptable since they provide for an accurate description of the reactor core.

8.0 STARTUP TESTING

We have reviewed the startup physics testing program for TMI-1 Cycle 6 presented in Reference 2. We conclude that this program is acceptable since it will provide confirmation that the as-loaded core conforms to the Cycle 6 nuclear design and since the data required by the Technical Specifications will be satisfied.

9.0 SUMMARY

We have reviewed the fuel system design, nuclear design, thermal-hydraulic design, and the transient and accident analysis information presented in the TMI-1 Cycle 6 reload submittal. We have concluded that the proposed reload and associated modified Technical Specifications are acceptable. Additionally, we conclude that changing the order of preference of instruments used to monitor quadrant power tilt is acceptable.

10.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

11.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 20, 1987

Principal Contributor:
D. Fieno

12.0 REFERENCES

1. H. D. Hukill (GPU) letter (5211-86-2182) to J. F. Stolz (NRC) on Technical Specification Change Request No. 149, dated November 3, 1986.

2. "Three Mile Island Unit 1 - Cycle 6 Reload Report," BAW-1977, Babcock & Wilcox Company Report, October 1986 (transmitted with Reference 1 above).

3. Letter from R. W. Reid (NRC) to J. G. Herbein (Metropolitan Edison Company), dated March 16, 1979.

4. H. D. Hukill (GPU) letter (5211-86-2103) to J. F. Stolz (NRC) on Fuel Resintering, dated June 9, 1986.

5. H. D. Hukill (GPU) letter (5211-86-2176) to J. F. Stolz (NRC) on Fuel Resintering, dated October 13, 1986.

6. Y. Hsii, et al., "TACO2 - Fuel Pin Performance Analysis," Babcock & Wilcox Company Report BAW-10141P-A, Rev. 1, June 1983.

7. "TMI-1 Fuel Densification Report," Babcock & Wilcox Company Report BAW-1389, June 1973.

8. Standard Review Plan, Section 4.2, Rev. 1, "Fuel System Design. U. S. Nuclear Regulatory Commission Report NUREG-0800, July 1981.

9. "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, Rev. 3, Babcock & Wilcox, Lynchburg, Virginia, July 1977.

10. D. A. Powers and R. O. Meyer, "Cladding Swelling Models for LOCA Analysis," U. S. Nuclear Regulatory Commission Report NUREG-0630, April 1980.

11. "Babcock & Wilcox Version of PDQ - User's Manual," Hassan, H. H., et al., BAW-10117P-A, Babcock & Wilcox, January 1977.

12. "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," Mays, C. W., et al., BAW-10152A, Babcock & Wilcox, June 1985.

13. NRC memorandum from L. S. Rubenstein to D. M. Crutchfield on review of the NOODLE code, April 10, 1985.

14. "TMI- Power Level Cutoff Removal Analysis," T. N. Ake, Babcock & Wilcox Company, September 1986.

15. "LYNX1: Reactor Fuel Assembly Thermal Hydraulic Analysis Code," BAW-10129A, Babcock & Wilcox, July 1985.

16. "LYNX2: Subchannel Thermal Hydraulic Analysis Code," BAW-10130A, Babcock & Wilcox, July 1985.

17. "LYNXT: Core Transient Thermal Hydraulic Program," BAW-10156, Babcock & Wilcox, February 1984.

18. "Thermal Hydraulic Crossflow Applications," BAW-1829, Babcock & Wilcox, May 1984.

19. "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 1983.

20. "TACO2 Loss-of-Coolant Accident Limit Analysis for 177-FA Lowered Loop Plants," BAW-1775, Rev. 0, Babcock & Wilcox, February 1983.

21. "Bounding Analytical Assessment of NUREG-0630 Models on LOCA kW/ft Limits with Use of FLECSET," BAW-1915P, Babcock & Wilcox, May 1986.