July 18, 1988

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
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Dear Mr. Hukill:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 67808)

The Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letter dated April 5, 1988.

The amendment revises the TMI-1 Technical Specifications for Cycle 7 of operation. Your April 18, 1988 letter requesting an amendment to increase the licensed rated power from 2535 MWt to 2568 MWt for TMI-1 will be the subject of a separate amendment.

A copy of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Ronald W. Hernan

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 142 to DPR-50
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFC	:LA:PDI-4	:PM:PDI-4	:PD:PDI-4	:OCG *	:	:	:
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METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & 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. 142
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated April 5, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 142, 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
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 18, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 142

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
i	i
vi	vi
vii	vii
viii	viii
1-5	1-5
2-1	2-1
2-2	2-2
2-3	2-3
Figure 2.1-1	Figure 2.1-1
Figure 2.1-2	Figure 2.1-2
Figure 2.1-3	Figure 2.1-3
2-5	2-5
2-6	2-6
2-7	2-7
2-8	2-8
2-9	2-9
-	2-10 (Table 2.3-1)
Figure 2.3-1	Figure 2.3-1
Figure 2.3-2	Figure 2.3-2
3-34	3-34
3-34a	3-34a
3-35	3-35
3-35a, 3-35b	3-35a
3-36	3-36
3-36a	3-36a
Figure 3.5-2A thru Figure 3.5-2L	Figure 3.5-2A thru Figure 3.5-2M
5-4	5-4

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1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left[\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in "b" or "f" below.

b. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling or personnel passage through these hatches.

c. All nonautomatic containment isolation valves and blind flanges are closed as required by the "Containment Integrity Check List" attached to the operating procedure "Containment Integrity and Access Limits".

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

f. One door of the personnel hatch or emergency hatch may be open for up to 24 hours for maintenance, repair or modification provided the other door of the hatch is maintained closed and has been leak tested and found to meet the local leak rate criteria for door seals within 24 hours prior to the maintenance, repair or modification.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/axial-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The B&W-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). A DNBR of 1.30 (B&W-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered 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 curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (3):

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

The 1.65 cosine axial flux shape in conjunction with $F^N \Delta H = 1.71$ 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 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 the DNBR limit.

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 axial 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 the DNBR limit 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, (B&W-2)(4), or 26 percent (BWC)(2) 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 per cent flow x 1.08 = 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 (B&W-2), or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 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 or BWC 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 1.30 (B&W-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (B&W-2), or 26 percent (BWC) 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.

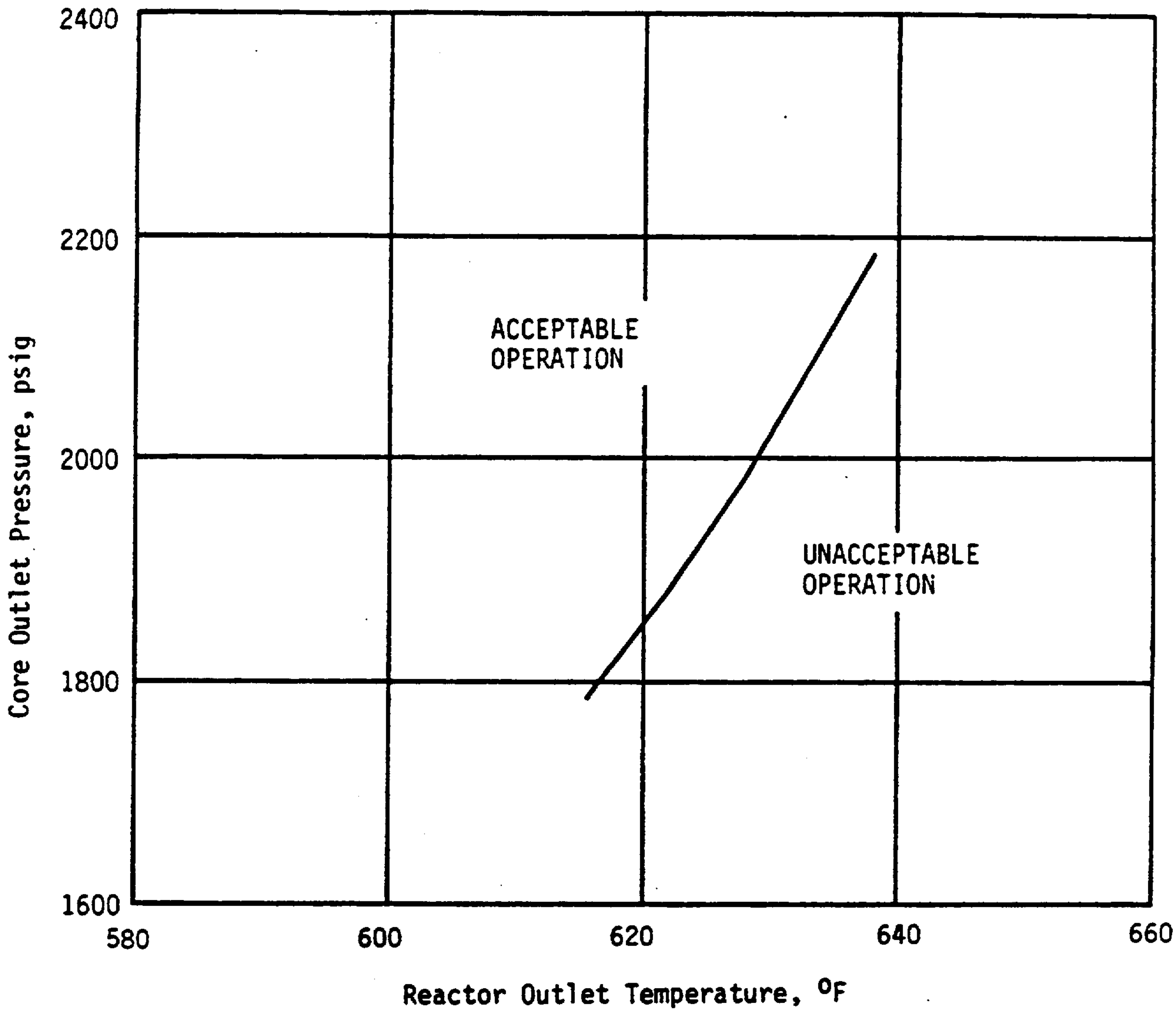
REFERENCES

(1) FSAR, Section 3.2.3.1.1

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, Babcock & Wilcox, Lynchburg, Virginia, April 1985

(3) FSAR, Section 3.2.3.1.1.3

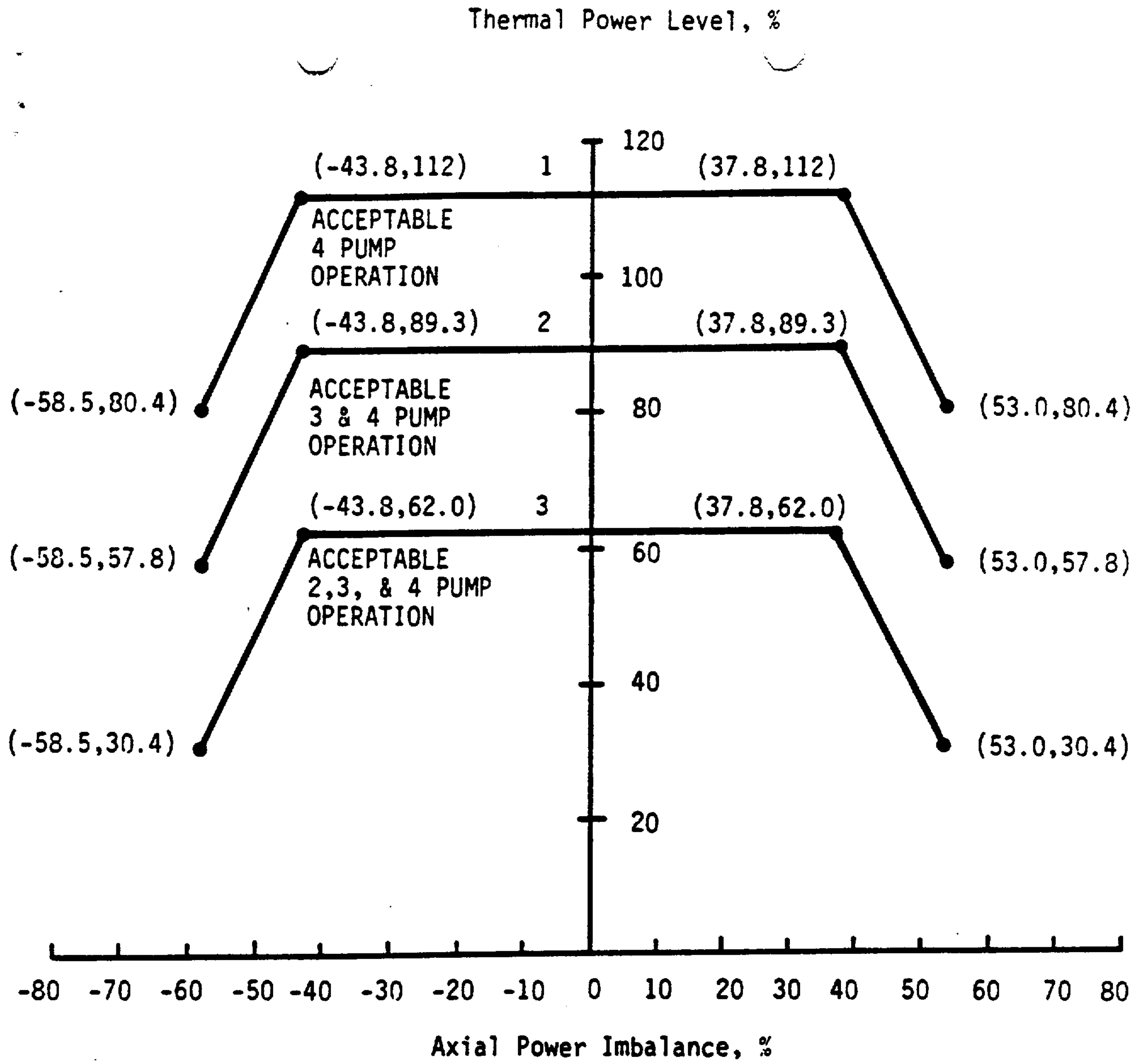
(4) FSAR, Section 3.2.3.1.1.11



CORE PROTECTION SAFETY LIMIT

TMI-1

Figure 2.1-1

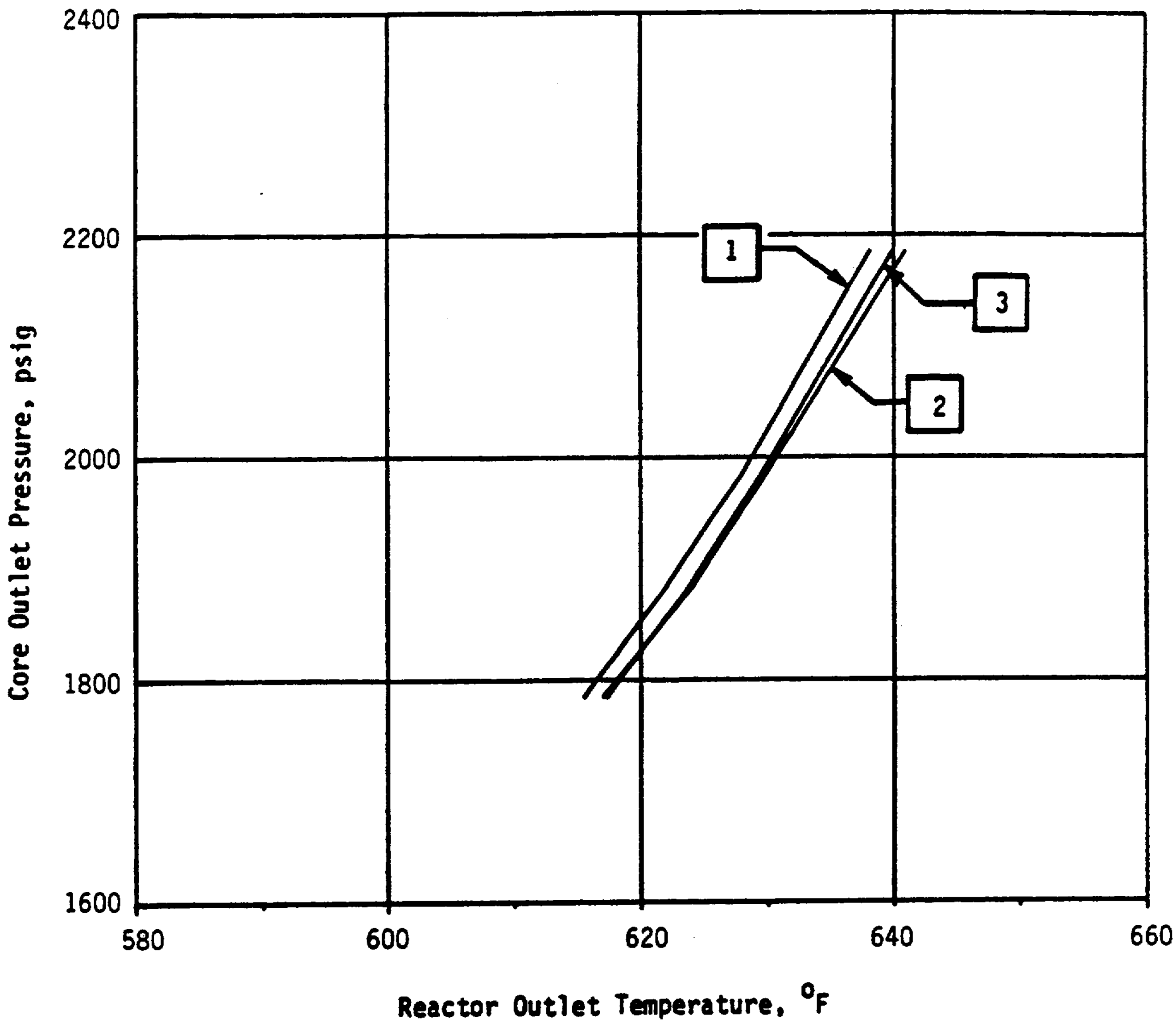


Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

CORE PROTECTION SAFETY LIMITS

TMI-1

Figure 2.1-2



Curve	Reactor Coolant Flow (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%)	89.4%	Three Pumps (Quality Limit)
3	68.8×10^6 (49.2%)	62.0%	One Pump in Each Loop (Qualitv Limit)

*106.5% of Cycle 1 Design Flow

CORE PROTECTION SAFETY BASES

TMI-1

Figure 2.1-3

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, axial 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 1.30 (B&W-2) or 1.18 (BWC) 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 axial 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/axial 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 1.30 (B&W-2) or 1.18 (BWC) 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 setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained 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.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) 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.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated overpressurization transients (8). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

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 and 7). 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. Application of the B&W

crossflow model resulted in safety limits (see Figures 2.1-1 and 2.1-3) outside the acceptable operating region formed by the low pressure, high pressure, and high temperature trip setpoints (see Figure 2.3-1) which justifies the removal of the variable low pressure trip.

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.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) FSAR, Section 14.1.2.3

(2 FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.9

(5) FSAR, Section 14.1.2.6

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

(7) "ECCS Analysis of B&W's 177-FA Lowered Loop NNS," BAW-10103-A, Rev. 3, Babcock and Wilcox, Lynchburg, Virginia, July 1977.

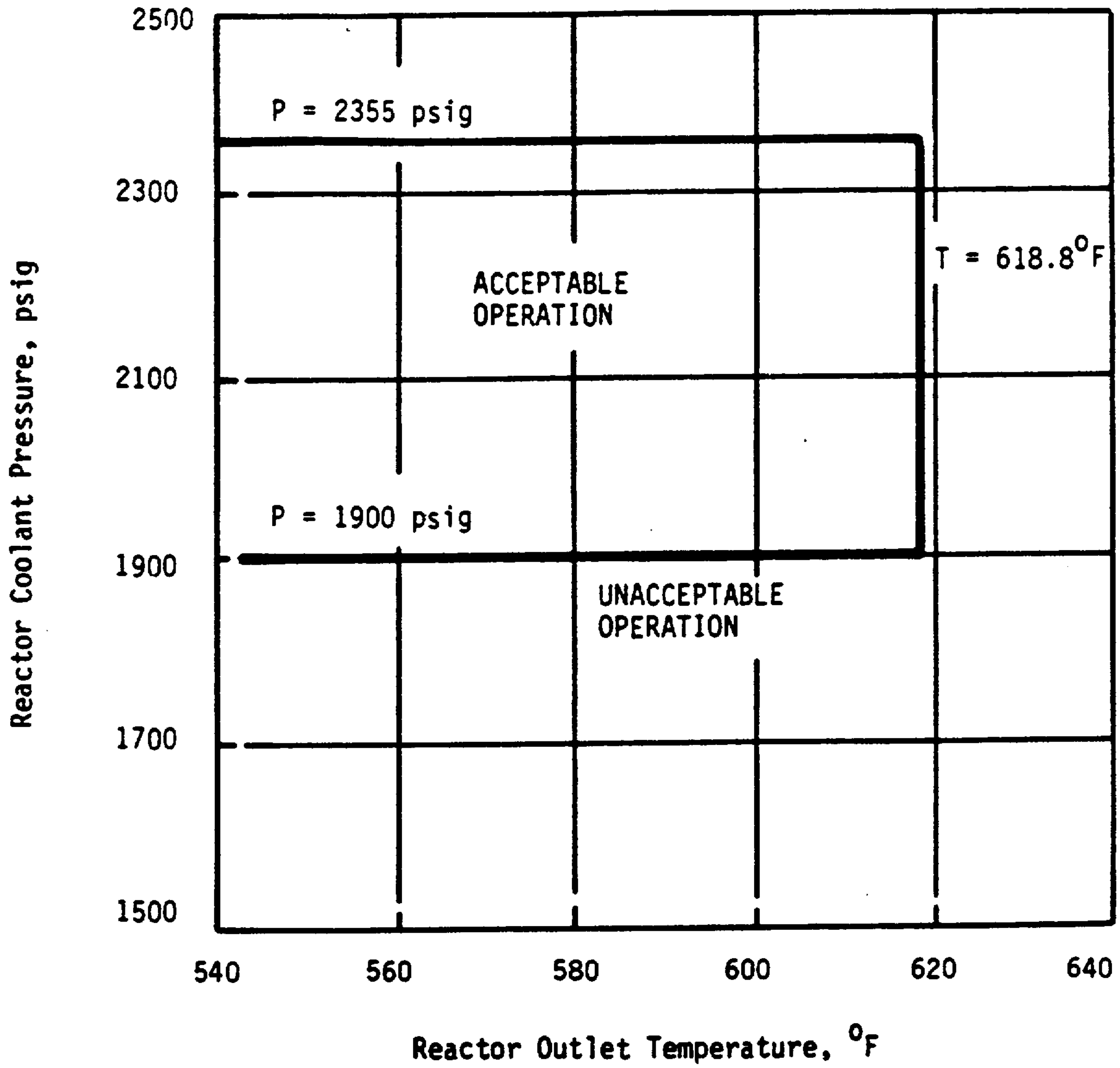
(8) "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, Rev. 0, Babcock and Wilcox, September 1985.

Table 2.3-1

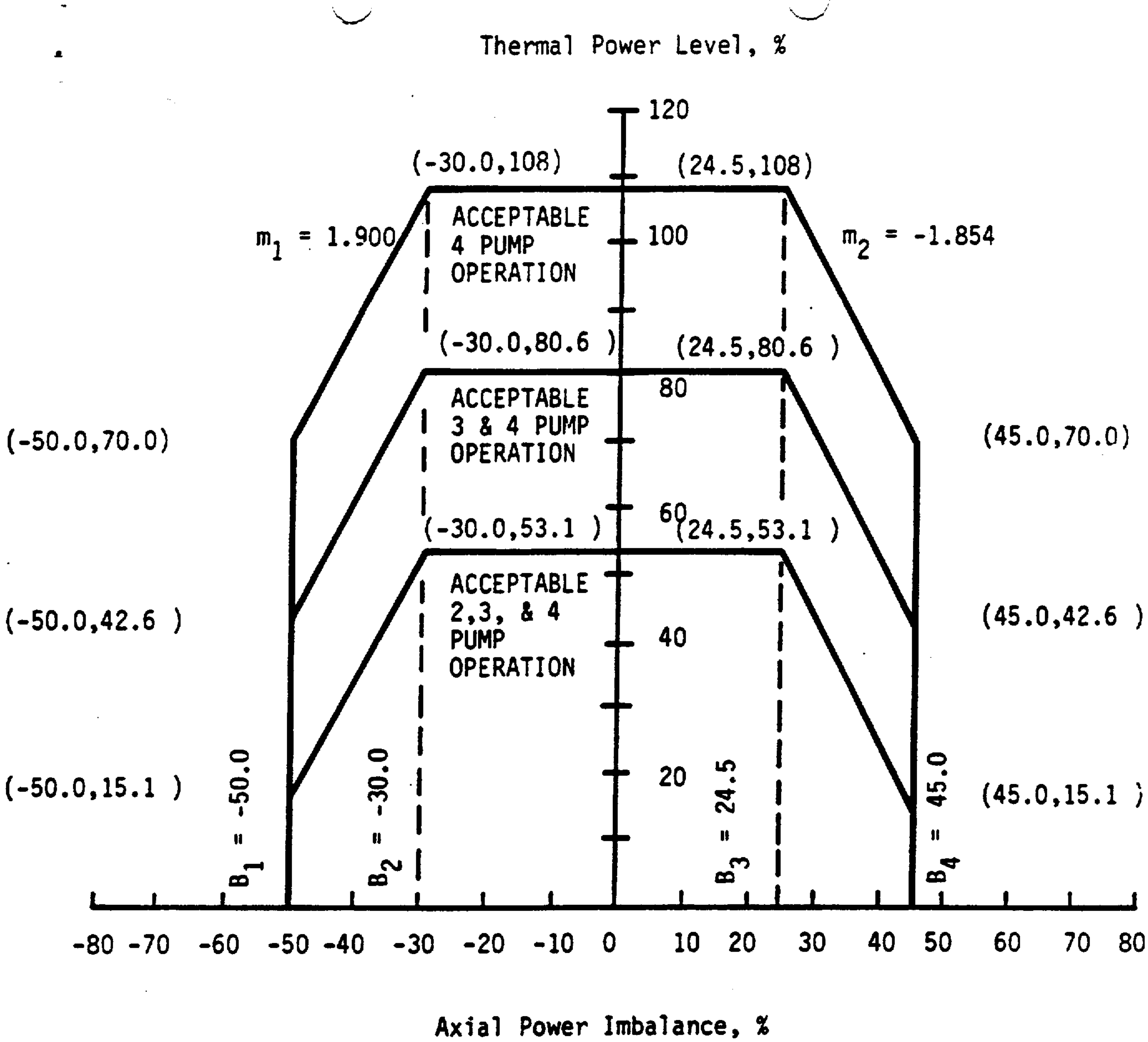
REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS (5)

	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(2)
2. Nuclear power based on flow (1) 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 (4) on pump monitors, max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Reactor coolant temp. F., max.	618.8	618.8	618.8	618.8
7. High Reactor Building pressure, psig max.	4	4	4	4

(1) Reactor coolant system flow, %.
(2) Administratively controlled reduction set only during reactor shutdown.
(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) 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.
(5) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors.



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS FOR
AXIAL POWER IMBALANCE

TMI-1

Figure 2.3-2

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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 the values in Table 3.5-1A 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 the values in Table 3.5-1A 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 the values in Table 3.5-1A as determined using the minimum incore detector system.

d. Except for physics tests if quadrant tilt exceeds the tilt limit, allowable 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, or when thermal power is equal to or less than 50% full power with four reactor coolant pumps running, set the nuclear overpower trip setpoint equal to or less than 60% full power.

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, 3.5-2K, and 3.5-2L) 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.

Table 3.5-1A - Quadrant Tilt Limits

Quadrant Tilt as Indicated By:	Tilt Limit (indicated power ≤ 50%)	Tilt Limit (indicated power > 50%)
Full incore detector system	6.83%	4.12%
Power range channels	4.05%	1.96%
Minimum incore detector system	2.80%	1.90%

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. Axial power 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 envelopes defined by Figures 3.5-2J, 3.5-2K, and 3.5-2L. If the imbalance is not within the envelopes defined by Figures 3.5-2J, 3.5-2K, or 3.5-2L 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 3.5-2M.

Bases

The axial power imbalance envelopes defined in Figures 3.5-2J, 3.5-2K, and 3.5-2L are based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2M) such that the maximum clad temperature will not exceed the Final Acceptance Criteria (2200°F). Operation outside of the axial power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The axial 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 axial power imbalance envelopes given in Figures 3.5-2J, 3.5-2K, and 3.5-2L have been error adjusted for observability and measurement uncertainties. Therefore, the limits specified in these figures are the maximum axial power imbalance alarm setpoints for power operation.

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 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 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
8	APSR (axial power shaping rod 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 rod position limits given in 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 have been error adjusted for observability and measurement uncertainties. Therefore, the limits specified in these figures are the maximum rod position alarm setpoints for operation.

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 for thermal power greater than 50% 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 quadrant power tilt limits for thermal power less than or equal to 50% set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using an actual core tilt of +7.50% which is equivalent to a +6.83% 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.

Reduction of the nuclear overpower trip setpoint to 60% full power when thermal power is equal to or less than 50% full power maintains both core protection and an operability margin at reduced power similar to that at full power.

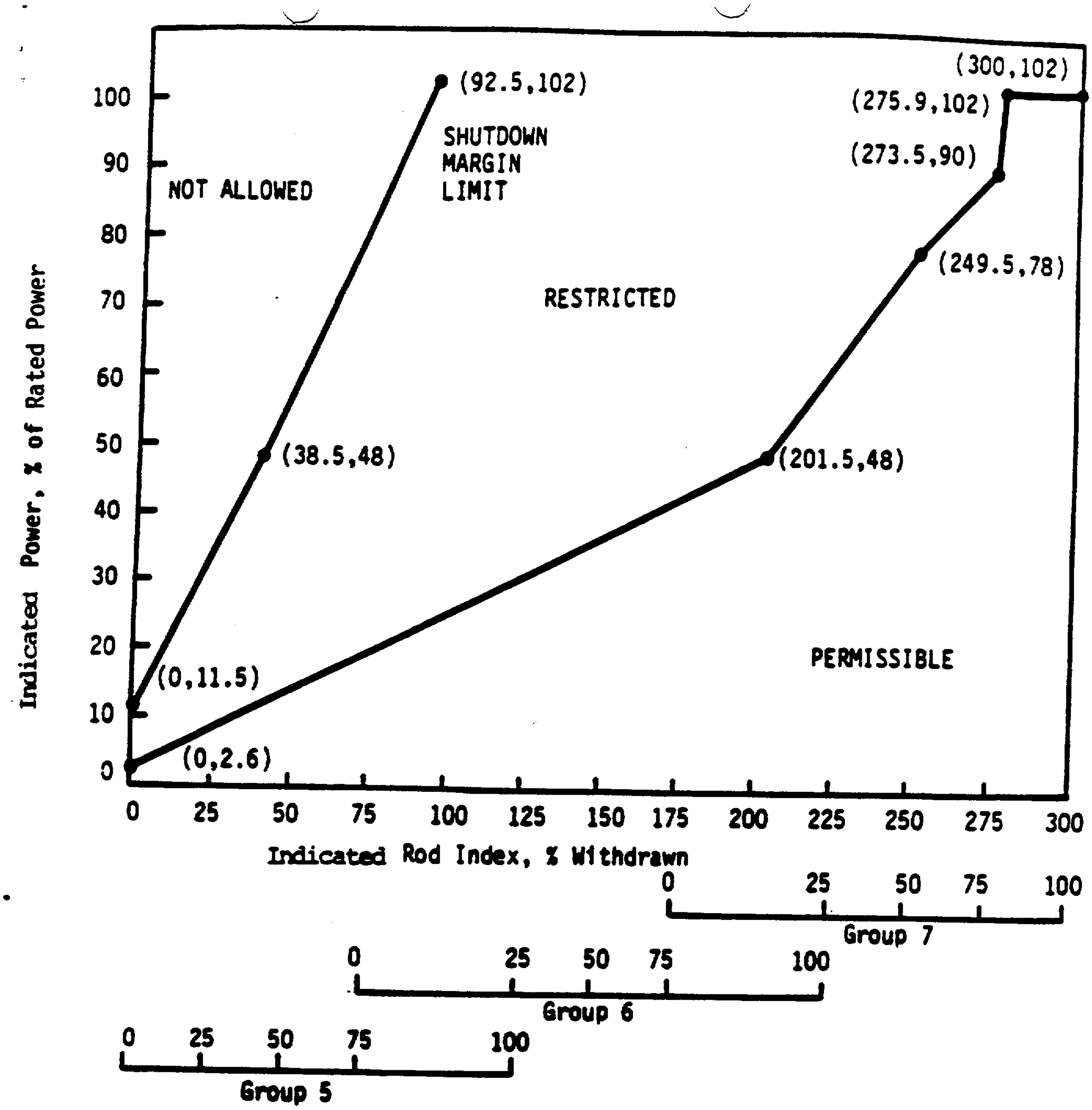
During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

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

REFERENCES

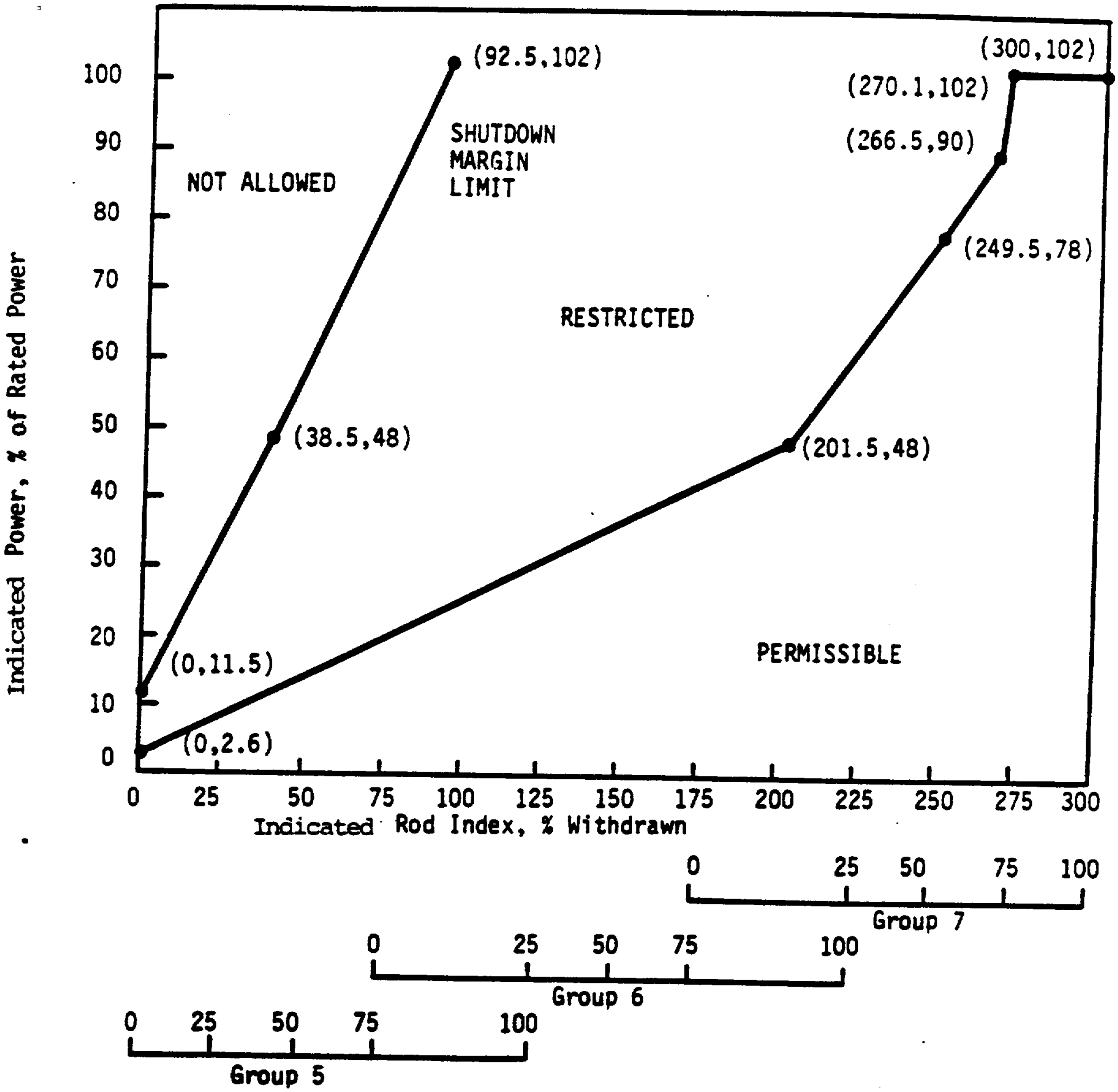
(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2



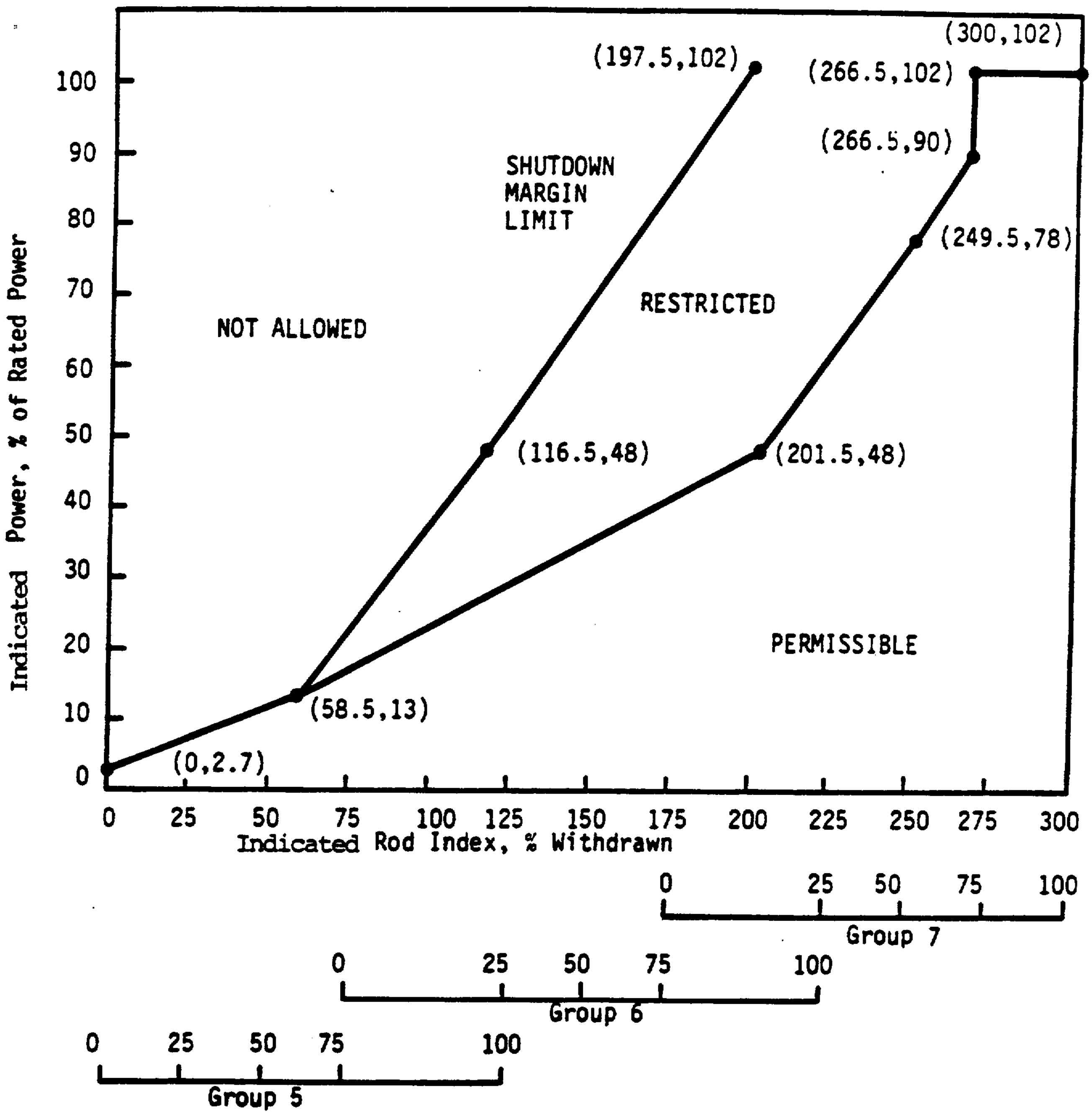
ROD POSITION SETPOINTS FOR
4 PUMP OPERATION FROM
0 TO 40 +10/-0 EFPD

TMI-1



ROD POSITION SETPOINTS FOR
4 PUMP OPERATION FROM 40
+10/-0 TO 100 +10/-0 EFPD

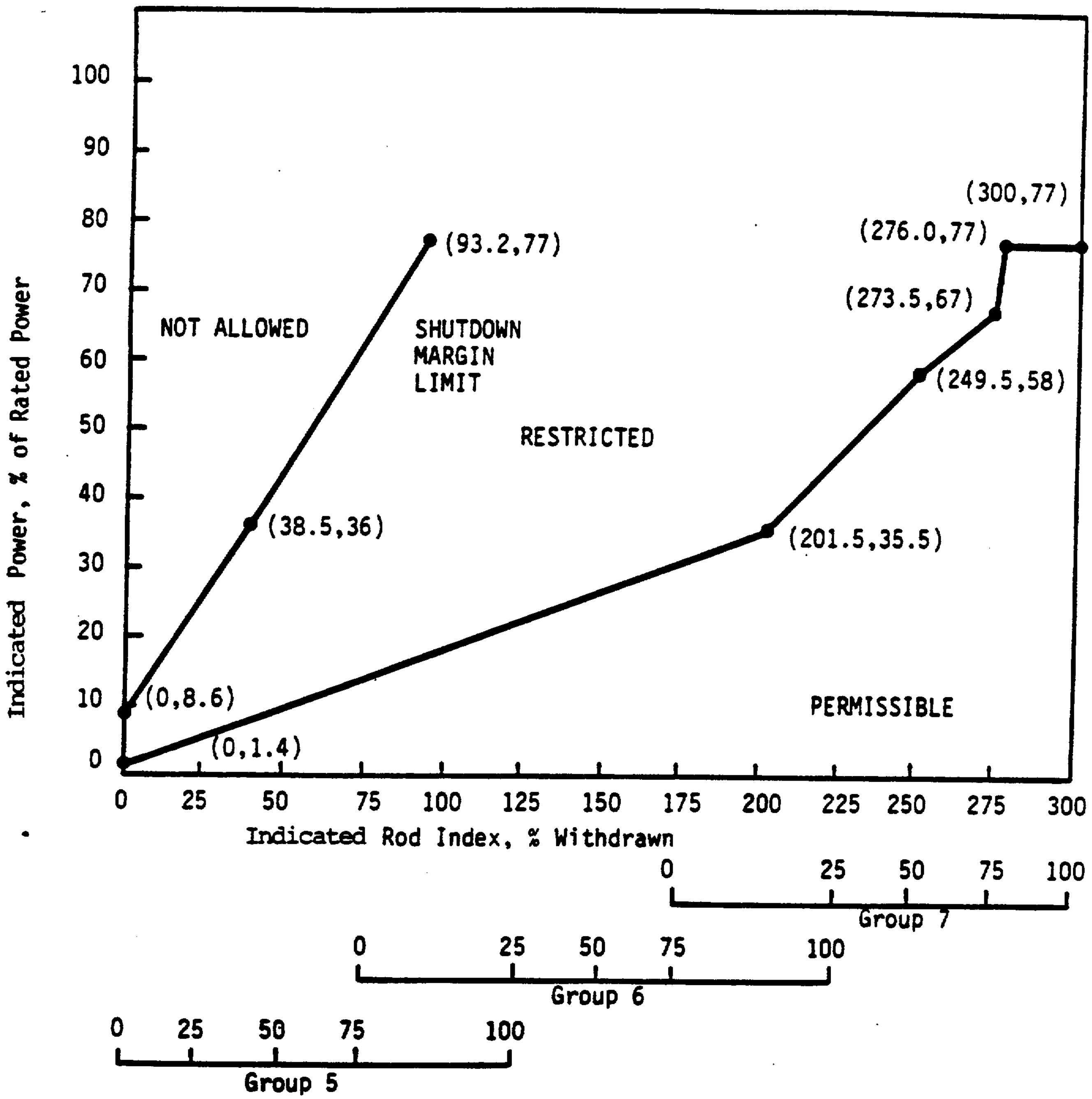
TMI-1



ROD POSITION SETPOINTS FOR
4 PUMP OPERATION AFTER 100
+10/-0 EFPD

TMI-1

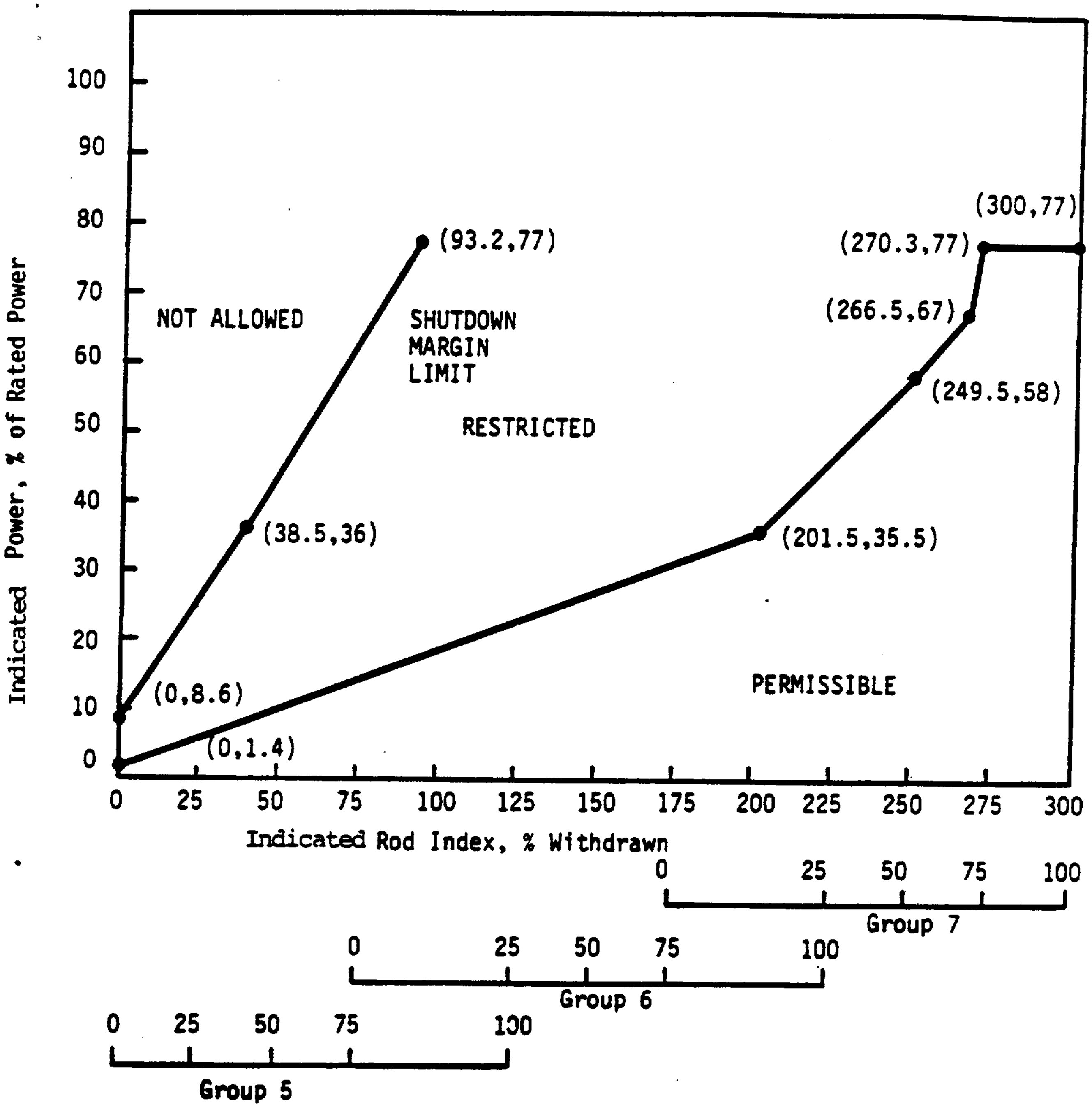
Figure 3.5-2C



ROD POSITION SETPOINTS FOR
3 PUMP OPERATION FROM 0 TO
40 +10/-0 EFPD

TMI-1

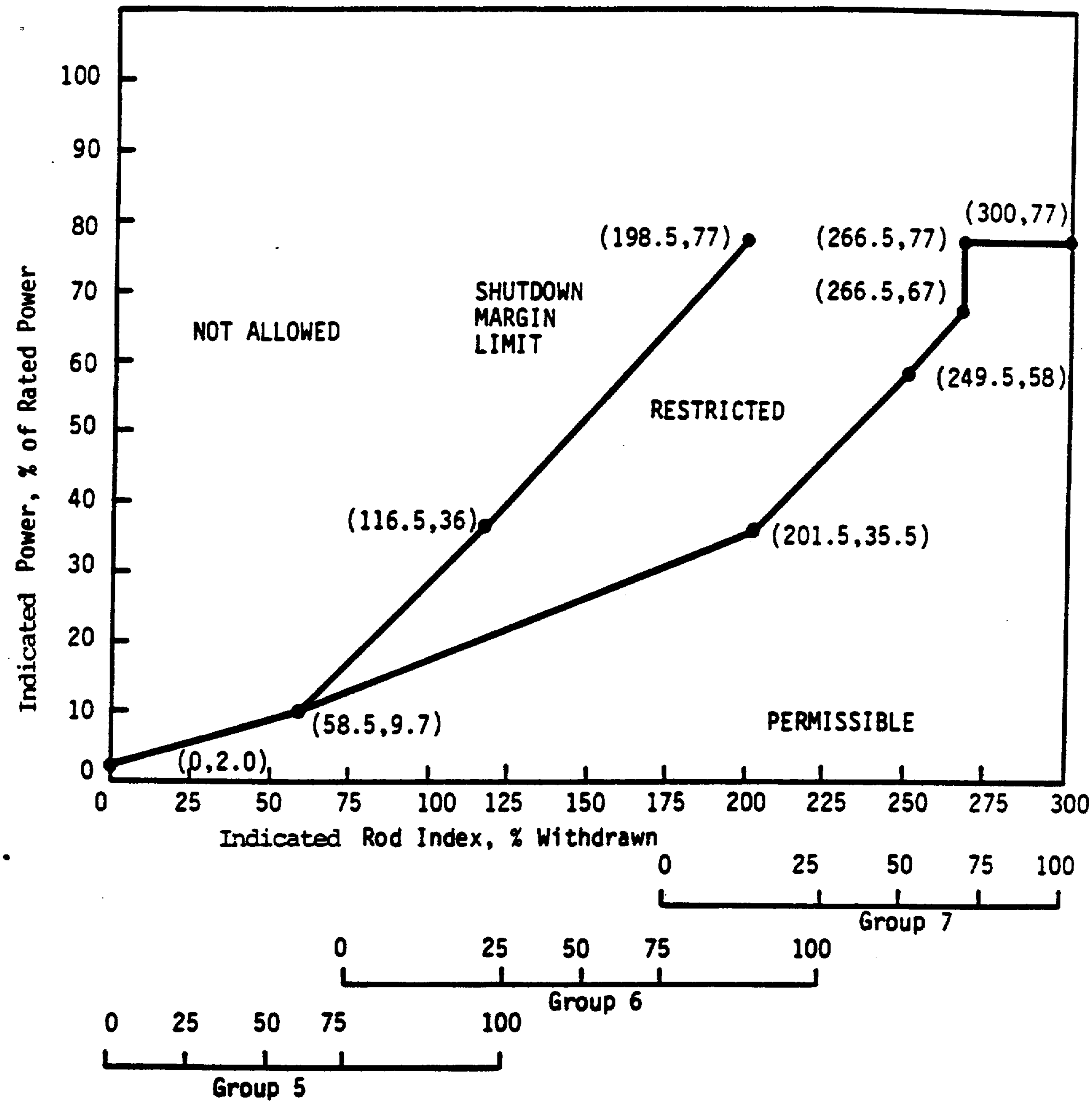
Figure 3.5-2D



ROD POSITION SETPOINTS FOR
3 PUMP OPERATION FROM 40
+10/-0 TO 100 +10/-0 EFPD

TMI-1

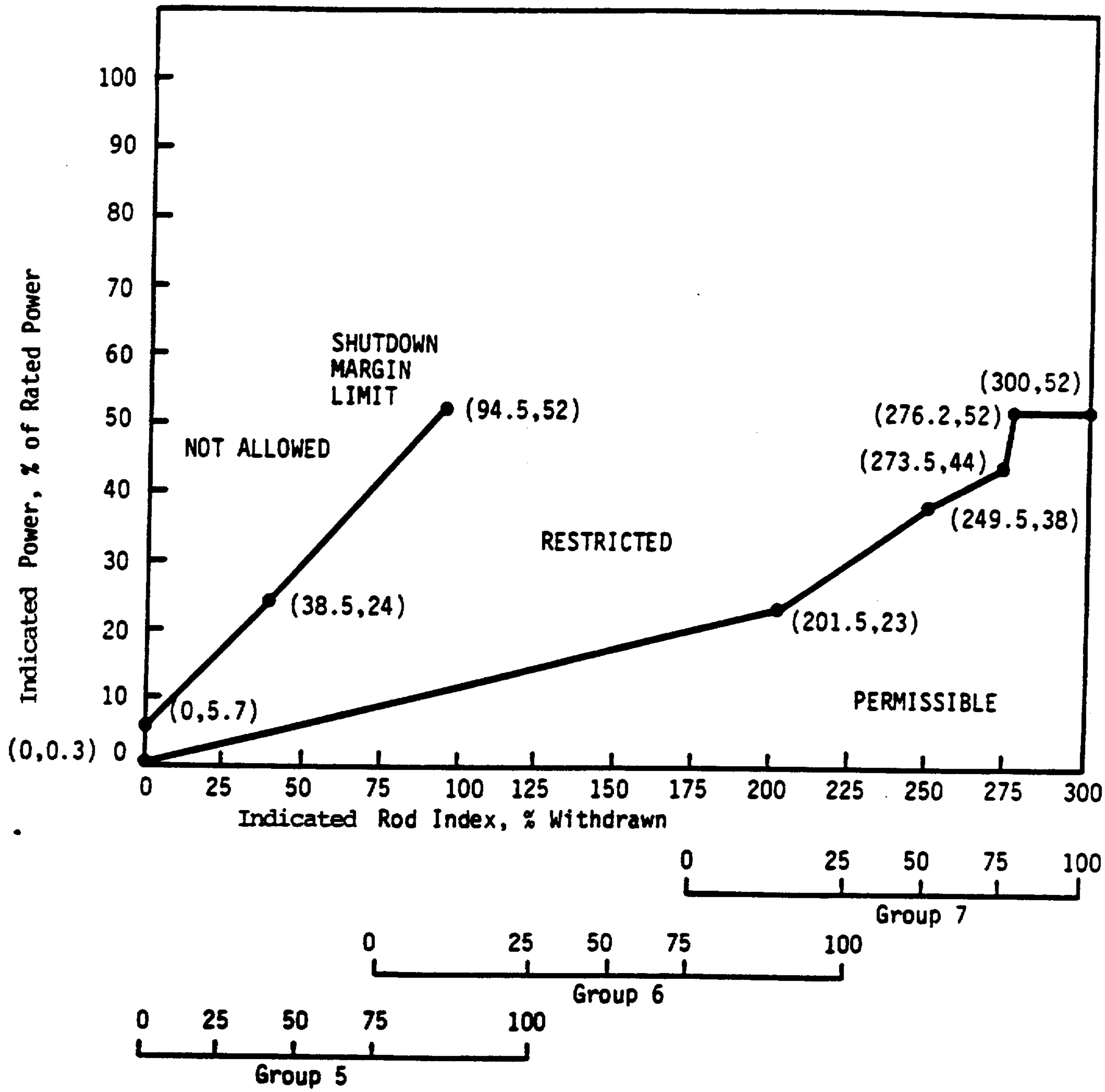
Figure 3.5-2E



ROD POSITION SETPOINTS FOR
3 PUMP OPERATION AFTER 100
+10/-0 EFPD

TMI-1

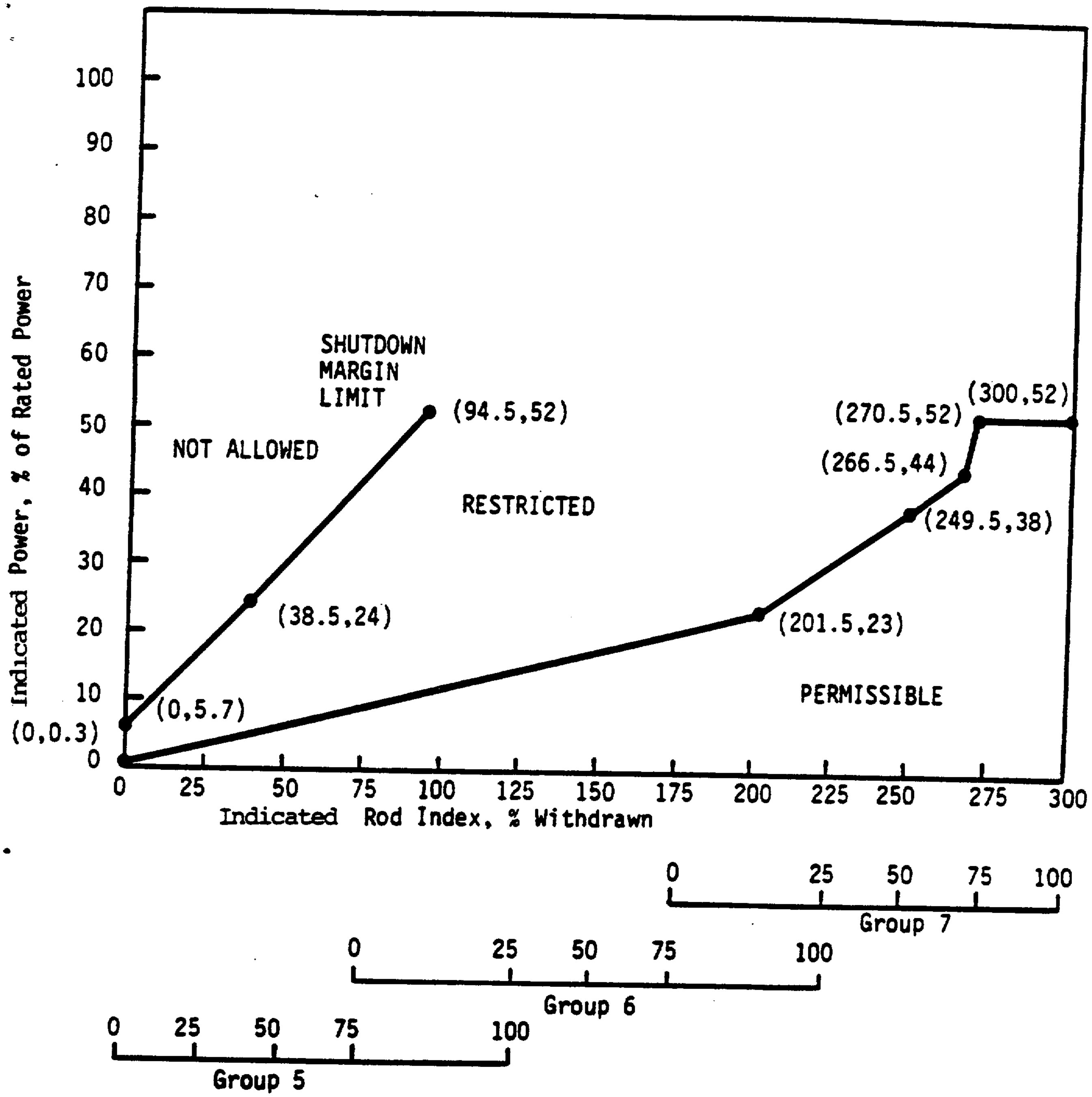
Figure 3.5-2F



ROD POSITION SETPOINTS FOR
2 PUMP OPERATION FROM 0
TO 40 +10/-0 EFPD

TMI-1

Figure 3.5-2G

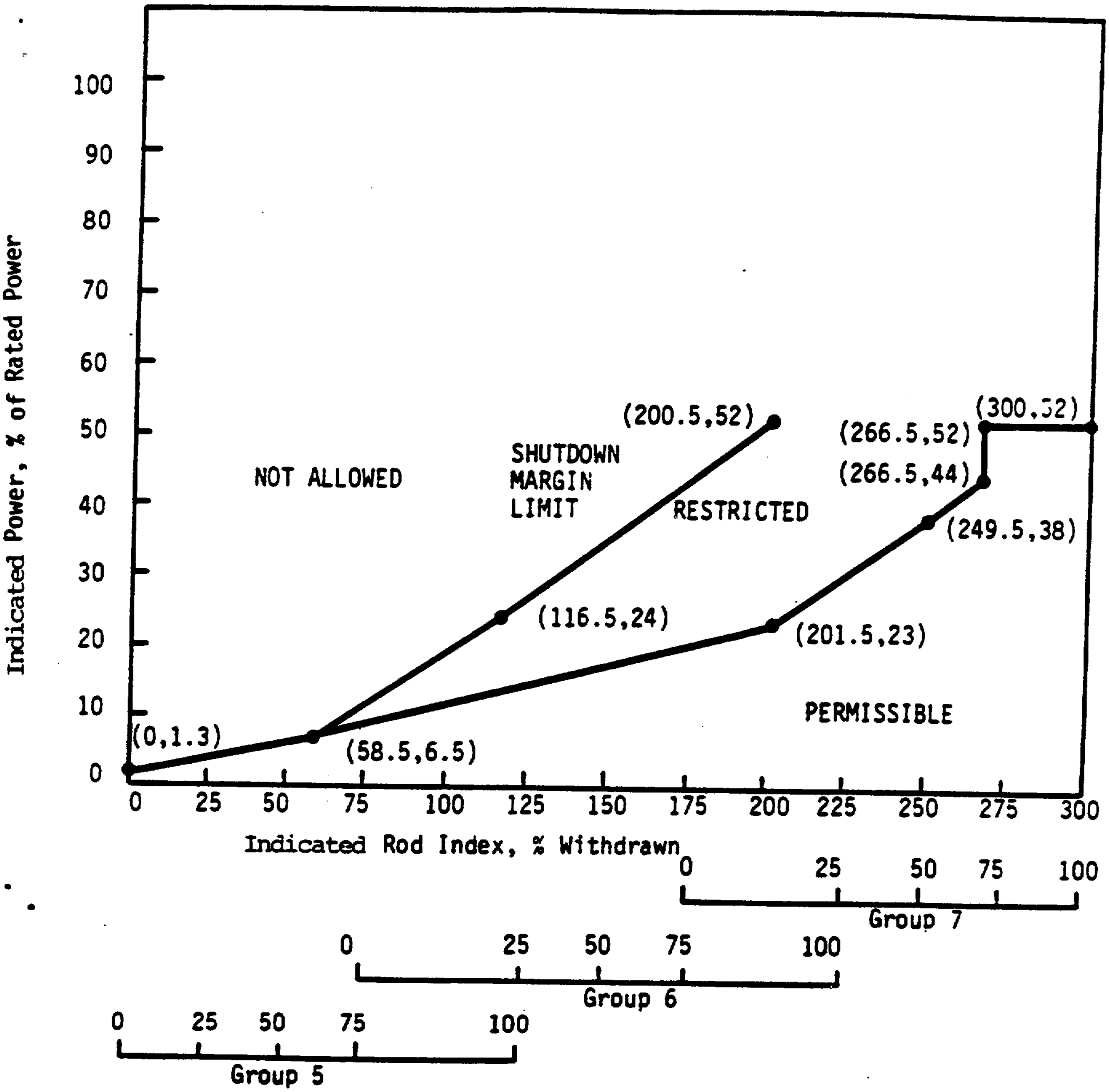


ROD POSITION SETPOINTS FOR
2 PUMP OPERATION FROM 40
+10/-0 TO 100 +10/-0 EFPD

TMI-1

Amendment No. ~~29~~, ~~39~~, ~~40~~, ~~45~~, ~~50~~, ~~120~~, ~~126~~, 142

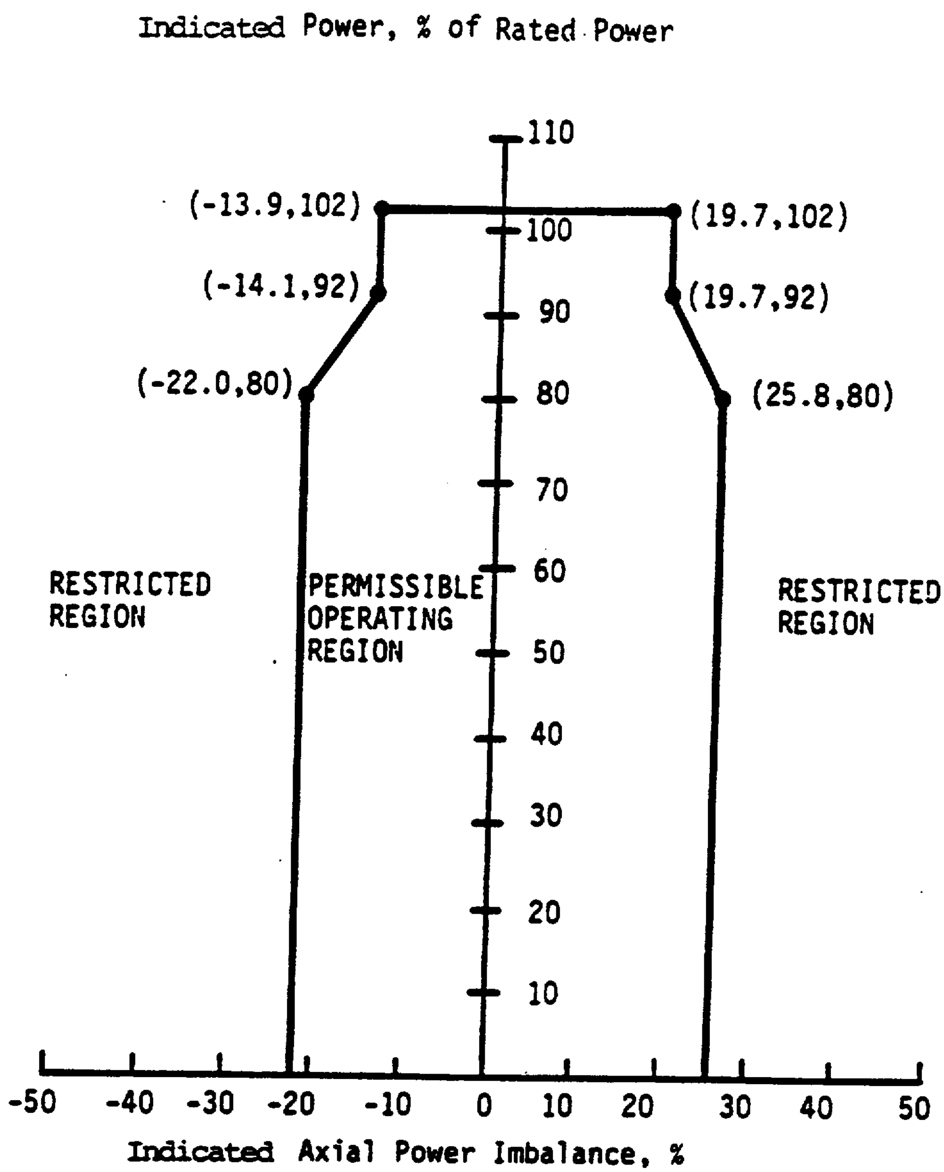
Figure 3.5-2H



ROD POSITION SETPOINTS FOR
2 PUMP OPERATION AFTER
100 +10/-0 EFPD

TMI-1

Figure 3.5-2I

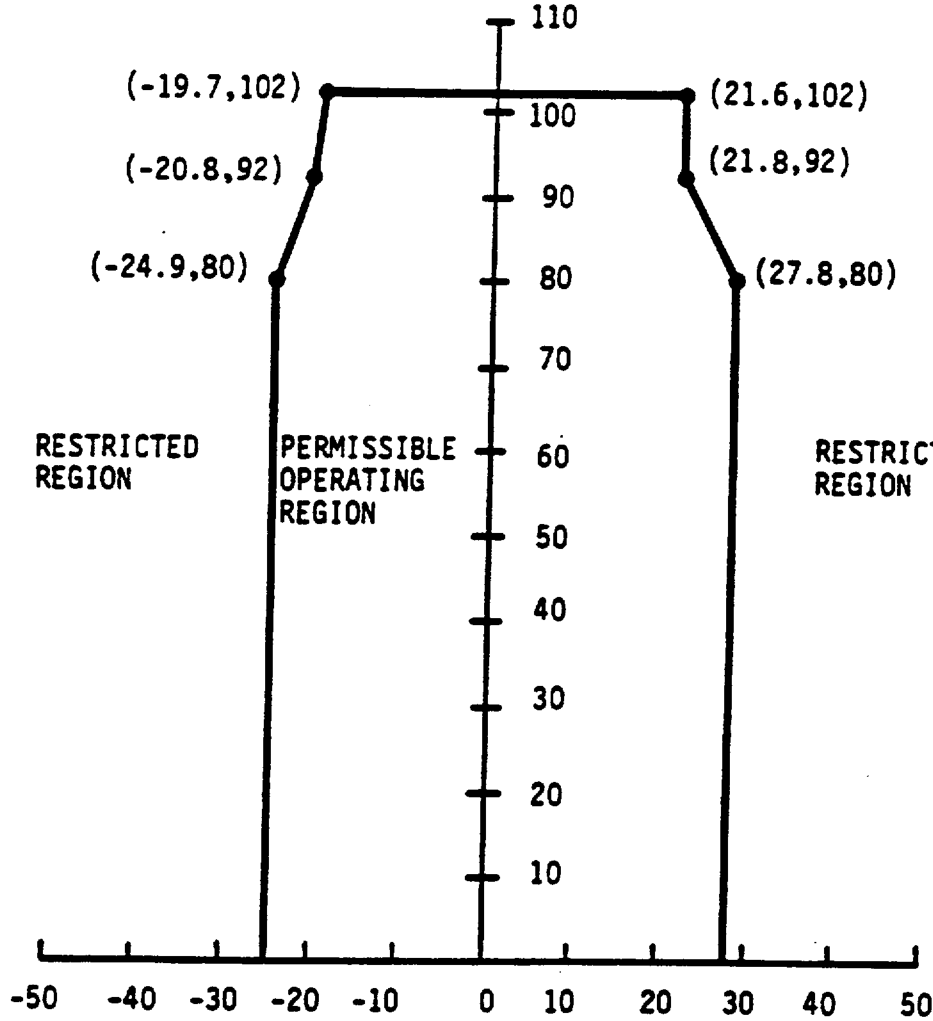


AXIAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 40 +10/-0 EFPD

TMI-1

Figure 3.5-2J

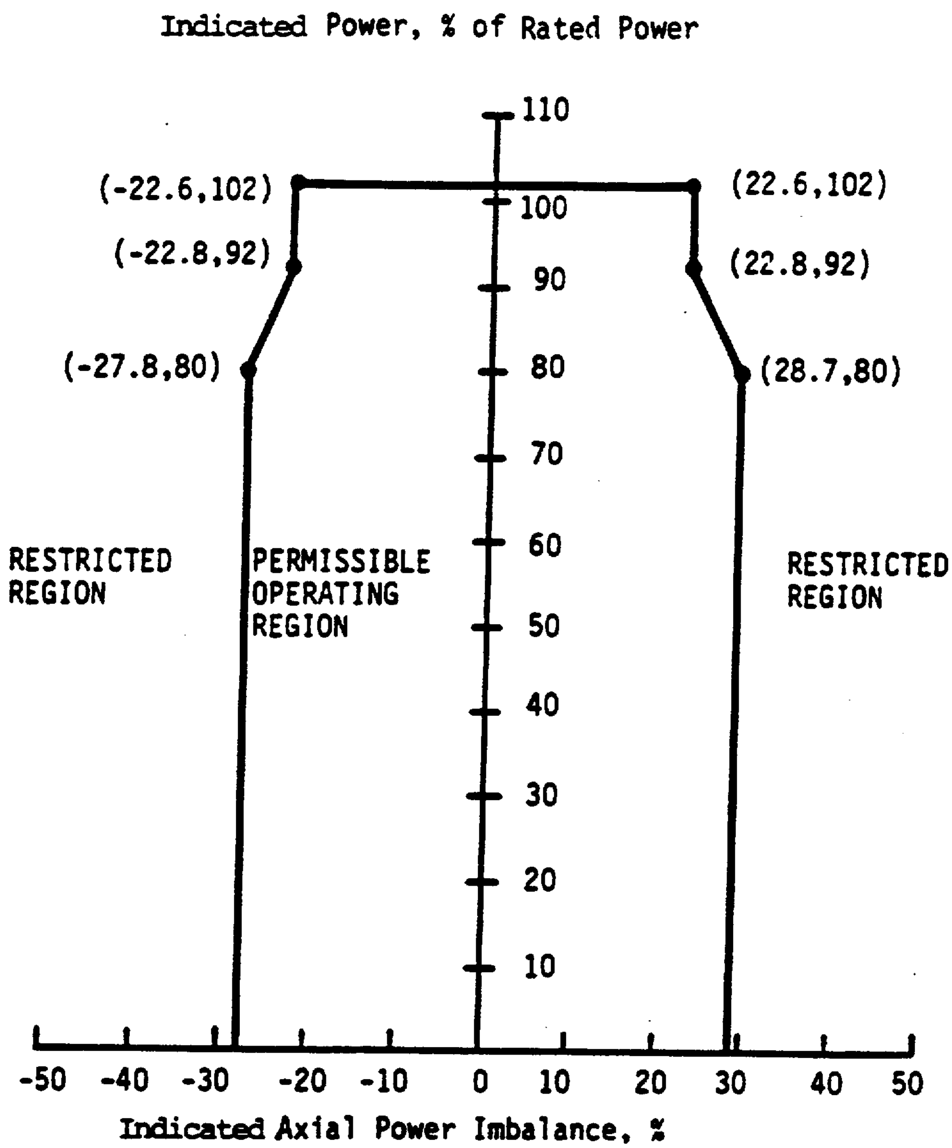
Indicated Power, % of Rated Power



Indicated Axial Power Imbalance, %

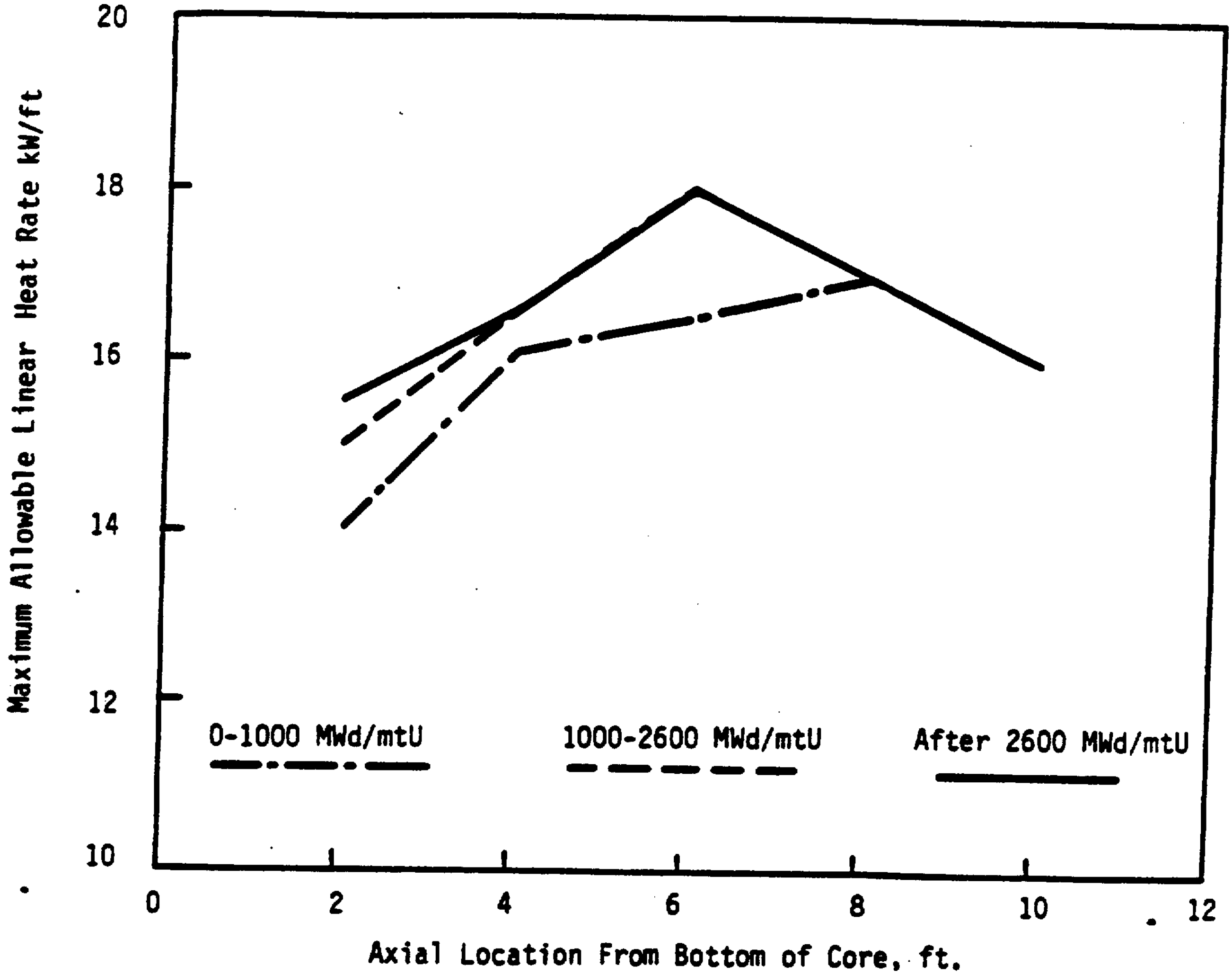
AXIAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
40 +10/-0 TO 100 +10/-0 EFPD

TMI-1



AXIAL POWER IMBALANCE
ENVELOPE FOR OPERATION AFTER
100 +10/-0 EFPD

TMI-1



LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE

TMI-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 inches.[2]

5.3.1.3 The average initial enrichment of the current core for Unit 1 is a nominal 3.02 weight percent of U^{235}. The highest enrichment is less than 3.7 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 4.3 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]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & 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 April 5, 1988 (Ref. 1), GPU Nuclear Corporation (GPUN) submitted an application to reload Unit No. 1 of the Three Mile Island (TMI) Nuclear Generating Station and operate it for a seventh cycle. To support the application, GPUN submitted report BAW-2015 (Ref. 2) entitled "Three Mile Island Unit 1 Cycle 7 Reload Report" and proposed changes to the Unit 1 Technical Specifications.

The Cycle 7 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rods, and one incore instrument guide tube. Cycle 7 is to have an operating length of approximately 445 effective full power days (EFPD). Cycle 7 will be operated in a rods out, feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length silver-indium-cadmium (Ag-In-Cd) control rods and 68 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution.

Although the licensed core full power level is 2535 megawatts-thermal (MWt), the Cycle 7 analyses were performed at a core power level of 2568 MWt. By letter dated April 18, 1988 (Ref. 16), GPUN submitted a request for an increase in the licensed rated power from 2535 MWt to 2568 MWt for TMI-1. This is also evaluated in part, herein, and will be the subject of a separate amendment and safety evaluation.

2.0 EVALUATION

2.1 EVALUATION OF FUEL SYSTEM DESIGN

Cycle 7 will contain 36 fresh (unirradiated) Mark B4 fuel assemblies with a U-235 enrichment of 2.85 weight percent (Batch 9A), four fresh Mark B4 assemblies with a 2.95 weight percent U-235 enrichment (Batch 9B) and 36 fresh Mark B4Z fuel assemblies with a 3.63 weight percent U-235 enrichment (Batch 9C). The remainder of the core will contain 12 Mark B4 once-burned Batch 8A assemblies, 64 once-burned Batch 8B assemblies and 25 twice-burned Batch 7 assemblies. All of these fuel assemblies are mechanically interchangeable. The Batch 9C Mark BZ assembly design is similar to the Mark B4 fuel assembly except that the six intermediate Inconel spacer grids have been replaced with zircaloy grids.

Although the Mark BZ fuel design (Ref. 3) has been reviewed and approved by the NRC (Ref. 4), the NRC safety evaluation states that a licensee incorporating this design is required to submit a plant-specific analysis of combined seismic and loss of coolant accident (LOCA) loads according to Appendix A of Standard Review Plan 4.2 (Ref. 5). The licensee has verified that the analysis that was presented in the Rancho Seco Cycle 7 reload report (Ref. 3) envelopes the TMI-1 plant design requirements and, therefore, the margin of safety reported for the Mark BZ fuel is applicable to TMI-1. Therefore, the staff concludes that the Mark BZ assemblies satisfy the above mentioned NRC requirement for Cycle 7.

The pin prepressure in some of the Batch 9 fuel assemblies has been lowered by 50 psi in order to provide a higher burnup limit for pin pressure but may be limiting in terms of cladding collapse. The licensee has stated that the cladding collapse time for the most limiting Cycle 7 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 7 assembly. The methods and procedures used for the analyses (Ref. 6) have been previously reviewed and approved by the staff. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 7 operation.

All other fuel rod thermal and mechanical analyses were also performed with previously approved methodology and the results were within the design criteria, including capability to centerline melt and internal pin pressure.

Based on the fact that approved methods have been used and fuel design criteria are all met, the staff finds the fuel design for Cycle 7 acceptable.

2.2 EVALUATION OF NUCLEAR DESIGN

The nuclear design parameters characterizing the TMI-1 Cycle 7 core have been computed by methods previously used and approved for Babcock and Wilcox (B&W) reactors (Ref. 7). Comparisons have been made between the parameters for Cycle 6 and Cycle 7. Core design changes including a core power level increase to 2568 MWt, an increase in cycle length to 445 ± 15 EFPD, as well as U-235 enrichment and shuffle pattern differences between cycles account for the differences in control rod worths, critical boron concentrations, Doppler coefficients, and moderator temperature coefficients (MTCs). The low neutron leakage Cycle 7 design is consistent with the GPUN reactor vessel fluence reduction efforts for TMI-1 as described in their response on the Pressurized Thermal Shock Rule 10 CFR 50.61 (Ref. 8).

The fresh Batch 9C Mark BZ fuel will have an initial enrichment of 3.63 weight percent U-235. The staff finds this acceptable since the TMI-1 spent fuel pool has been designed to store fuel with a maximum enrichment of 4.3 weight percent U-235.

Shutdown margin calculations for Cycle 7 include the effects of poison material depletion, a 10% calculational uncertainty, allowance for rod bite, the power deficit in going from hot full power (HFP) to hot zero power (HZP), and neutron flux redistribution as well as a maximum worth stuck rod. Beginning of cycle (BOC) and end of cycle (EOC) shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. Shutdown margins at BOC and EOC are 4.2% delta k/k and 3.0% delta k/k, respectively, compared to the minimum required value of 1.0% delta k/k.

Based on its review, the staff concludes that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of TMI-1 Cycle 7 is acceptable.

2.3 EVALUATION OF THERMAL HYDRAULIC DESIGN

Although a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions (Ref. 4), incompatibility in the hydraulic characteristics has an effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies co-exist in the core. Since the Mark BZ assemblies have a higher hydraulic resistance due to the BPRA retainers and the zircaloy intermediate spacer grids, some of the coolant flow is diverted from the Mark BZ fuel to the lower-powered Mark B fuel. The fact that the Mark BZ assemblies have less flow in a mixed core results in lower maximum allowable power peaking and a lower enthalpy rise factor required in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. The licensee, therefore, performed a bounding thermal-hydraulic design analysis in which a full Mark BZ core and a core bypass flow of 8.8% were assumed. The DNB results were compared to an analysis using the actual mixed core configuration and bypass flow (7.6%) and found to be bounding. Therefore, a transition core penalty due to the introduction of Mark BZ assemblies is not required for Cycle 7.

For Cycle 7, the BWC critical heat flux correlation (Ref. 9) was used for analysis of the Mark BZ fuel assembly instead of the B&W-2 correlation used in Cycle 6. The BWC correlation has been reviewed and approved by the staff and has been found to be applicable to the Mark BZ design.

Based on the fact that the licensee's thermal-hydraulic analyses were performed using approved analytical methods and correlations and resulted in acceptable performance, the staff finds the thermal-hydraulic design of Cycle 7 acceptable.

2.4 ACCIDENT AND TRANSIENT ANALYSIS

The important physics, thermal-hydraulic, and kinetics parameters for Cycle 7 have been compared to the values used in the FSAR (Ref. 10), fuel densification report (Ref. 11), reference cycle and/or the generic LOCA analyses (Refs. 12, 13, & 14). Although some Cycle 7 values are not bounded by those previously used, the licensee has determined that the initial conditions defined by these parameters would produce less severe transients than the initial conditions assumed in the reference analyses and, therefore, no reanalysis was necessary.

The consequences of certain transients and accidents are not affected by physics, thermal-hydraulic, or kinetics parameters but rather by radiological considerations due to core isotopic inventory changes. Although the radionuclide inventory generated at a bounding power level of 2568 MWt was found to be only slightly greater than that obtained at a power level of 2535 MWt (Cycle 6), the licensee conservatively assumed a 10% increase in the Cycle 7 core fission product inventory in reevaluating the most adversely affected events. All of the resulting Cycle 7 accident doses were well below the dose acceptance criteria based on 10 CFR 100.

The important cycle specific parameters for Cycle 7 have also been compared to the limiting values used in the generic LOCA analyses and have been found to be bounded. Therefore, adherence to the linear heat rate (LHR) limits for Cycle 7 given in Table 7-2 of the Reload Report assures that the emergency core cooling system (ECCS) Final Acceptance Criteria will be met.

Based on the safety analysis review, the staff finds that the consequences of transients and accidents during Cycle 7 meet all safety criteria and are acceptable.

2.5 TECHNICAL SPECIFICATION CHANGES

The TMI-1 Cycle 7 Technical Specifications have been modified to support a longer fuel cycle length (445 EFPD) as well as various operational and design changes. These include changes in power peaking and control rod worths and the removal of the variable low pressure trip as well as incorporation of a low leakage fuel design, mixed Mark B/Mark BZ fuel, and a power level upgrade from 2535 MWt to 2568 MWt.

Changes were made to the following Technical Specification items:

(a) core protection safety limit pressure/temperature curves;
(b) core protection safety limit axial power imbalance limits;
(c) protection system maximum allowable setpoints;
(d) power level dependent quadrant tilt setpoints;
(e) overpower trip setpoint at 50% power or less;
(f) rod position setpoints;
(g) axial power imbalance envelope for operation;
(h) LOCA limited maximum allowable LHR;
(i) maximum allowable enrichment of Cycle 7 and future reload fuel;
(j) BWC correlation with DNBR limit of 1.18 for Mark BZ fuel.

In addition, various administrative and editorial changes were made.

The staff has reviewed the proposed changes (Ref. 15) and finds them acceptable because they have been derived from analyses performed using approved methods and have been appropriately considered in the Cycle 7 safety analyses.

2.6 RATED POWER UPGRADE

As shown above, the staff has found the proposed Cycle 7 reload and the associated modified Technical Specifications acceptable. The Cycle 7 core characteristics and Technical Specification limits were developed for a full power level of 2568 MWt or higher and, therefore, the proposed power upgrade does not change the original design conditions. In addition, the staff concludes that the power upgrade effect on reactor vessel accumulated fluence is acceptable.

The staff has reviewed the high pressure injection (HPI) flow split of 64% to the core and 36% out a cold leg discharge break which was justified in the TMI-1 Restart Report based on a rated power of 2535 MWt. Although the B&W generic small break LOCA analysis, which was performed at a rated power of 2772 MWt, used an HPI flow split of 70% - 30%, the 64% - 36% flow split was reevaluated for the requested increased rated power of 2568 MWt. Based on this reevaluation, which demonstrated that the TMI-1 HPI system will deliver as much water to the core as the generic LOCA analysis assumed during the time period of concern, the staff concludes that TMI-1 has sufficient HPI capacity at a rated power of 2568 MWt.

TMI-1 has an estimated natural circulation cooldown time of 22 hours (at 10°F/hr). Since the condensate-grade feedwater supply has sufficient inventory to support a cooldown time in excess of 100 hours, the staff concludes that this large margin assures that a natural circulation cooldown will not be affected by the proposed small increase in rated power.

The design basis safety analyses of flooding from plant sources assumed a flow rate greater than that expected to support operation at 2568 MWt. Since the flood level is limited by the amount of water available to be pumped into the building, and the upgraded power level will not change the available water inventory, the staff concludes that the maximum FSAR predicted flood level will not change due to the proposed power uprate.

The proposed upgraded power level will not cause a change in either the primary system or secondary system available water inventory. Since the flood level is limited by the amount of primary/secondary water available to be pumped into the building, the staff concludes that the maximum predicted flood levels from either a primary or secondary break will not change due to the upgraded power level.

Based on the Cycle 7 reload evaluation and the design basis safety analyses evaluations discussed above, the staff concludes that the proposed power uprate does not change the original design conditions and that all existing reactor design and safety criteria are preserved at the upgraded power level of 2568 MWt. Further evaluation of this power uprate will be contained in a separate safety evaluation to be issued in support of an amendment approving the power upgrade.

2.7 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, thermal-hydraulic, and accident information presented in the TMI-1 Cycle 7 reload report and finds the proposed reload and the associated modified Technical Specifications acceptable. Based on this evaluation and the separate safety evaluation supporting the amendment approving the power upgrade, the staff also finds that Cycle 7 can be operated at a rated core power of either 2568 MWt or at the existing rated power level of 2535 MWt without exceeding the established safety criteria.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact relating to the proposed license amendment was published in the Federal Register on July 18, 1988 (53 FR 27092).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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: July 18, 1988

Principal Contributor: Lawrence I. Kopp

REFERENCES

1. Letter from H. D. Hukill (GPUN) to USNRC, C311-88-2033, Technical Specification Change Request No. 182, Cycle 7 Reload, April 5, 1988.

2. "Three Mile Island Unit 1, Cycle 7 Reload Report," BAW-2015, March 1988.

3. "Rancho Seco Cycle 7 Reload Report, Volume 1, Mark BZ Fuel Assembly Design Report," BAW-1781P, April 1983.

4. Letter from J. F. Stolz (NRC) to Sacramento Municipal Utility District (SMUD), "Rancho Seco Nuclear Generating Station, Evaluation of Mark BZ Fuel Assembly Design," November 16, 1984.

5. "Standard Review Plan," NUREG-0800, Revision 2, July 1981.

6. "Program to Determine In-Reactor Performance of B&W Fuels, Cladding Creep Collapse," BAW-10084A, Revision 2, October 1978.

7. "NOODLE-A Multi-Dimensional Two-Group Reactor Simulator," BAW-10152-A, June 1985.

8. Letter from GPUN to USNRC, 5211-86-2007, January 23, 1986.

9. "BWC Correlation of Critical Heat Flux," BAW-10143P-A, April 1985.

10. "Three Mile Island Nuclear Station, Unit 1, Final Safety Analysis Report," USNRC Docket No. 50-289.

11. "Three Mile Island Unit 1 Fuel Densification Report," BAW-1389, June 1973.

12. "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103-A, Rev. 3, July 1977.

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

14. "Bounding Analytical Assessment of NUREG-0630 Models on LOCA kW/ft Limits With Use of FLECSET," BAW-1915P, May 1986.

15. "Technical Specification Change Request No. 182, Cycle 7 Reload," Attachment to Reference 1.

16. Letter from H. D. Hukill (GPUN) to USNRC, C311-88-2036, Technical Specification Change Request No. 184, Rated Power Upgrade, April 18, 1988.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION, ET AL.

DOCKET NO. 50-289

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 142 to Facility Operating License No. DPR-50, issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of the Three Mile Island Nuclear Station, Unit 1 located in Dauphin County, Pennsylvania.

The amendment modified the Technical Specifications to support core reload for Cycle 7 of operation. The core design changes for Cycle 7 include a slight increase in core lifetime from approximately 425 effective full power days (EFPD) to approximately 445 EFPD. The fresh fuel has a slightly higher U-235 enrichment than previous fuel.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on April 25, 1988 (53 FR 13456). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated April 5, 1988, (2) Amendment No. 142 to License No. DPR-50, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street NW, Washington, D.C. 20555, and at the Local Public Document Room, Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Rockville, Maryland this 18 th day of July, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronald W. Hernan

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Division of Reactor Projects I/II
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