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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-50

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

   A. The applications for amendment by Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company (the licensees) dated October 29, 1976, as supplemented December 29, 1976, and January 20, 1977; January 26, 1977, as supplemented March 31, 1977; and February 23, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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 Appendices A and B, as revised through Amendment No. 29 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Reid acting for

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-50

Docket No. 50-289

The changed areas on the revised pages are shown by marginal lines. Pages 4-27 and 4-28 are unchanged and are included for convenience only.

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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 REACTOR POWER IMBALANCE

Reactor 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" 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 non-automatic 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.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent, (3) whichever condition is more restrictive.

The maximum thermal power for three pump operation is 86.7 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.08 = 80.7 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

Using a local quality limit of 22 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the B&W-2 correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

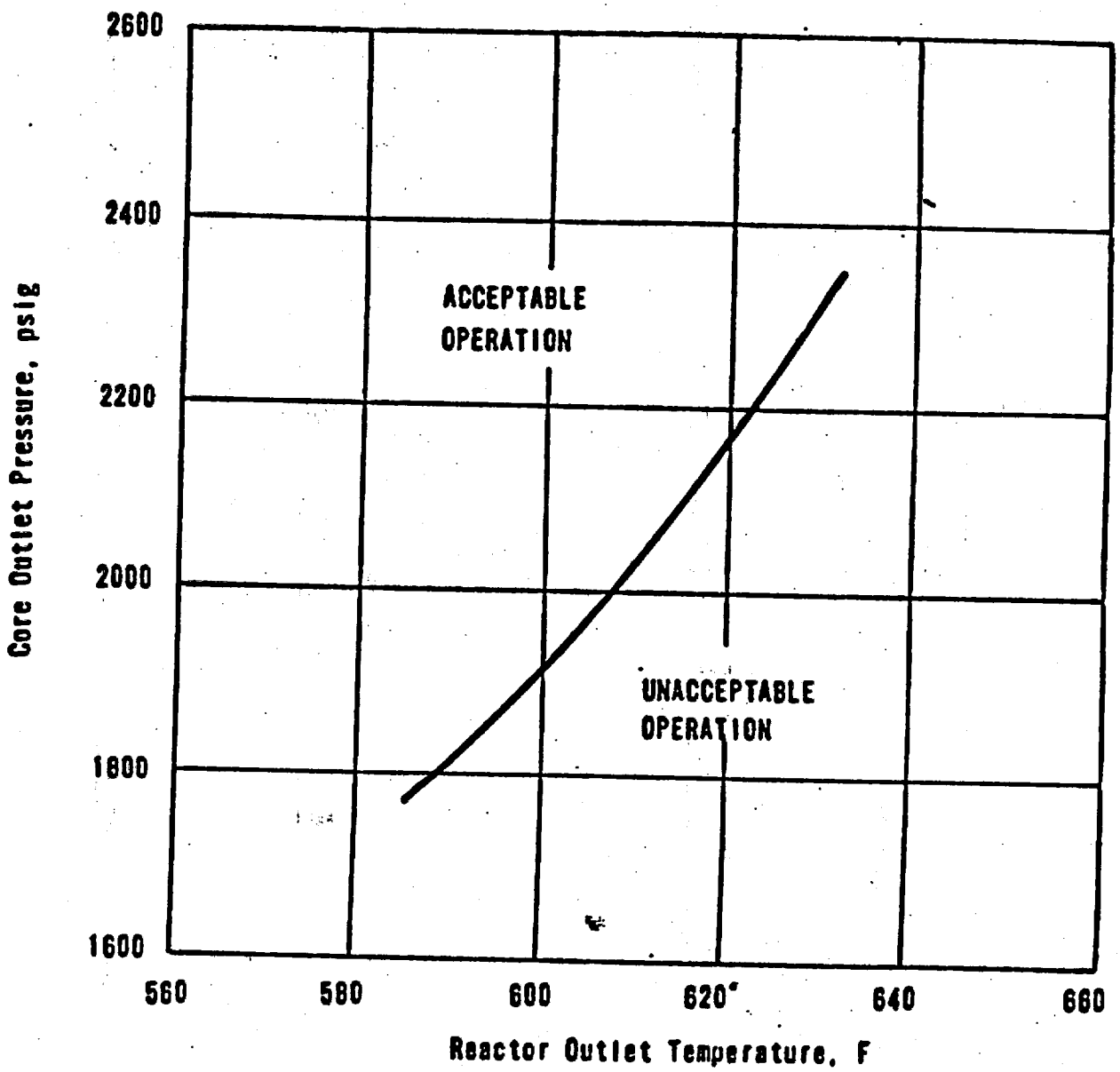
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor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

REFERENCES

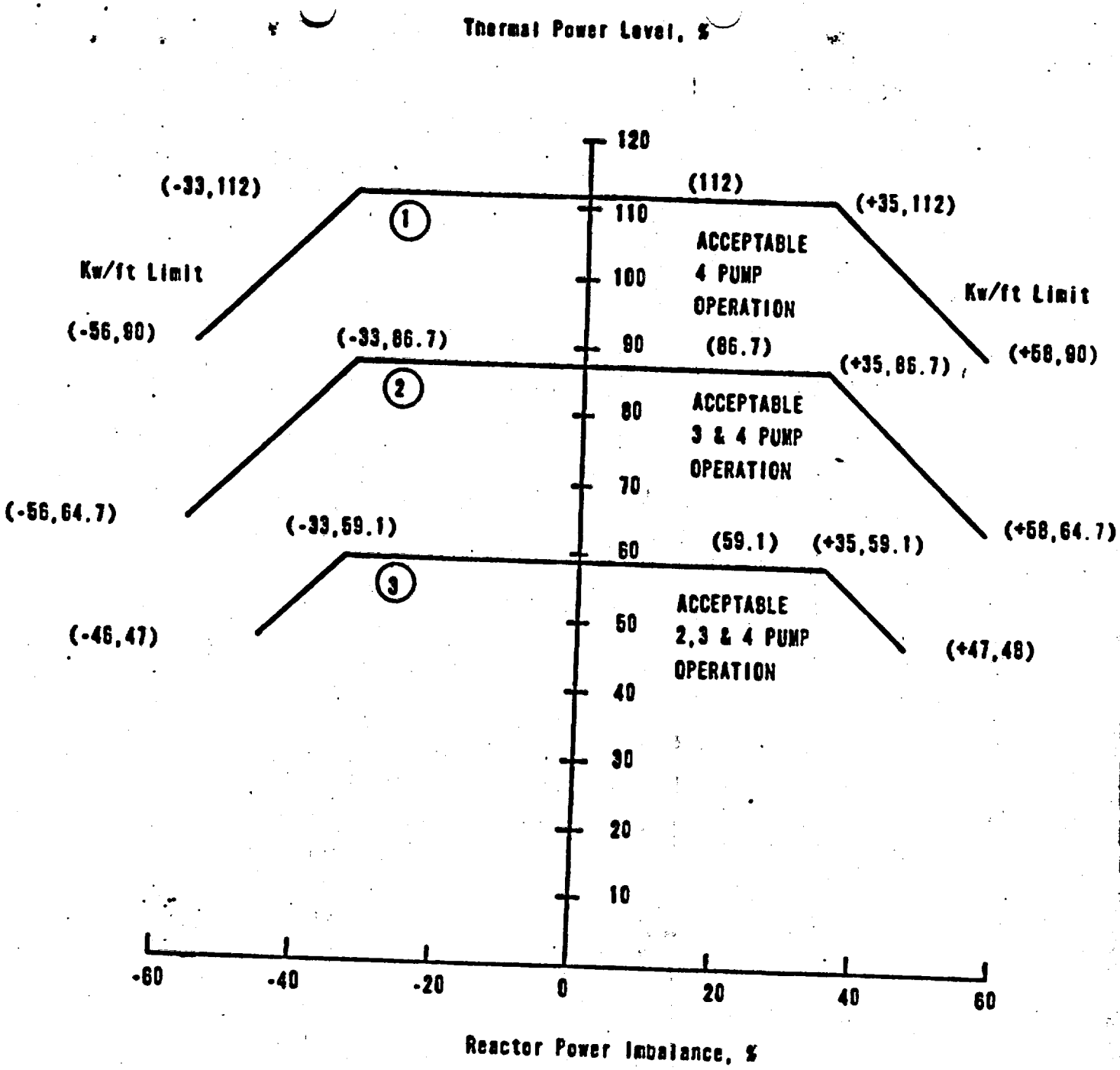
(1) FSAR, Section 3.2.3.1.1

(2) FSAR, Section 3.2.3.1.1.c

(3) FSAR, Section 3.2.3.1.1.k



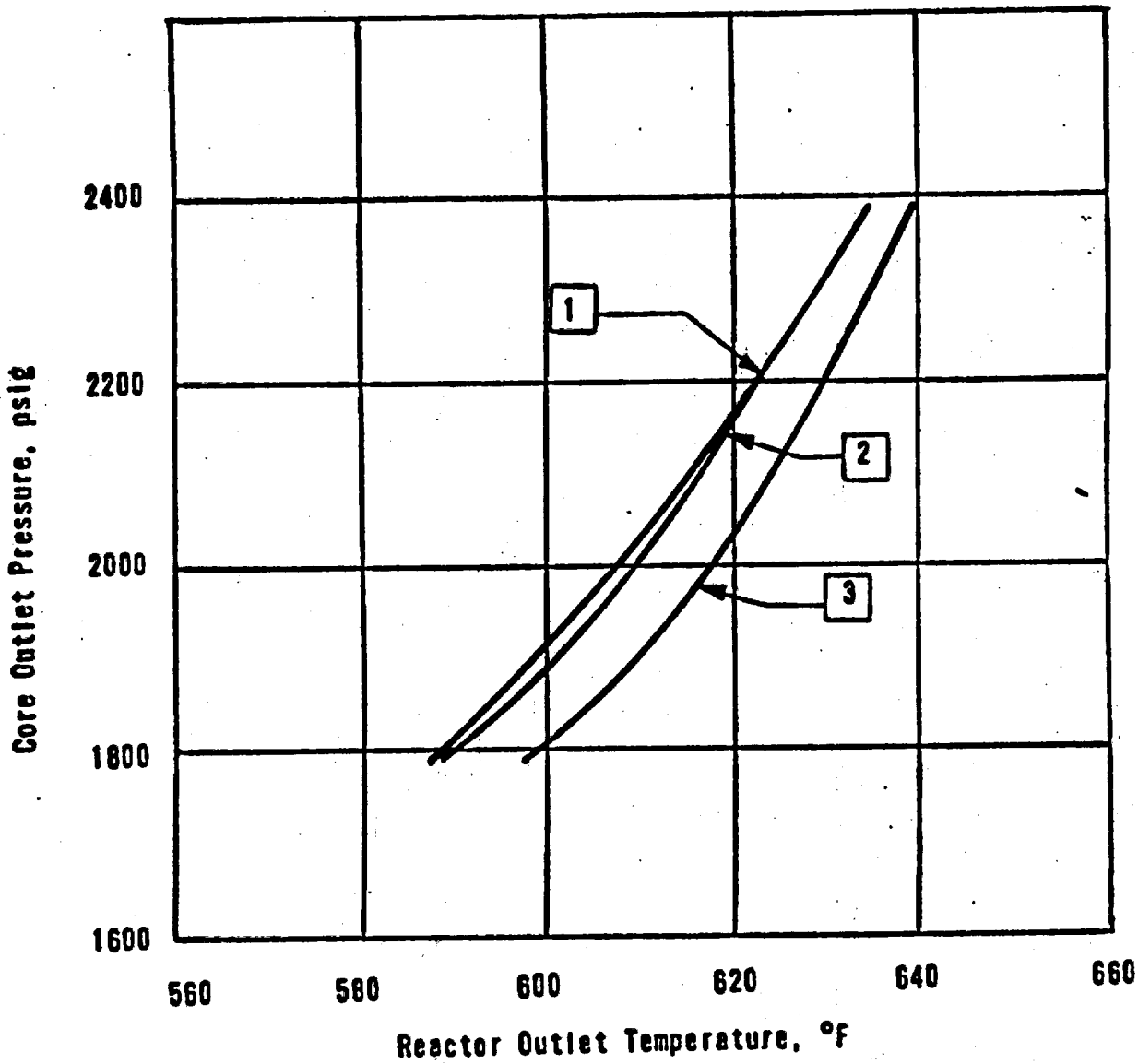
TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY LIMIT
Figure 2.1-1



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY LIMITS

Figure 2.1-2



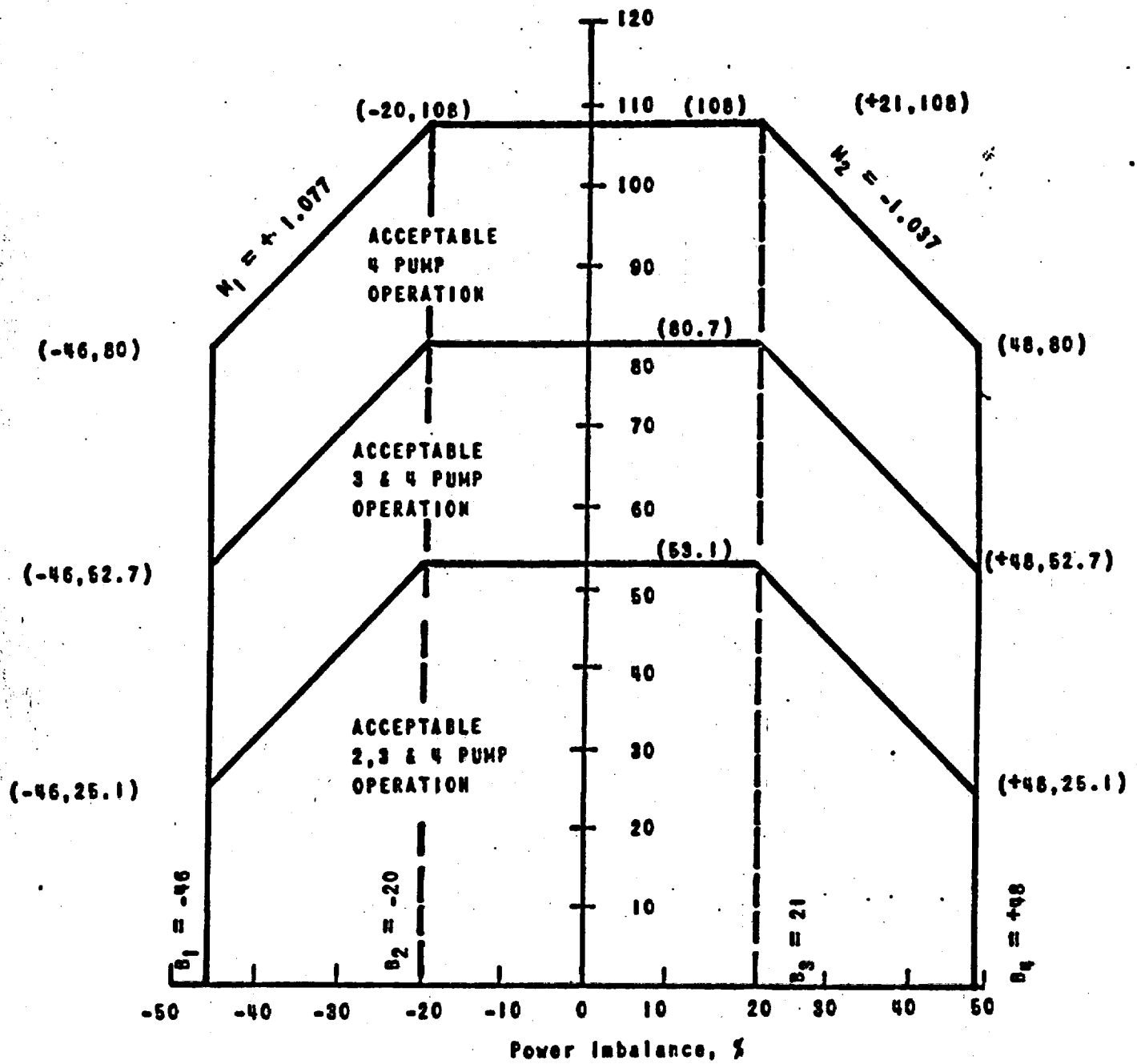
REACTOR COOLANT FLOW

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%)	86.7%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.1%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

TMI-1, UNIT 1, CYCLE 3
CORE PROTECTION SAFETY BASES

Figure 2.1-3



TMI-1, UNIT 1, CYCLE 3
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-2

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objective

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

Specification

3.1.2.1 For operations until five effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and Cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding five effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G, Section V.C.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.(1) These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100° F in any one hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100° F satisfies stress levels for temperatures below the DTT.(3)

The heatup and cooldown rate limits in this specification are not intended to limit instantaneous rates of temperature change, but are intended to limit temperature changes such that there exists no one hour interval in which a temperature change greater than the limit takes place.

The unirradiated reference nil ductility temperature (RT NDT) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT NDT with accumulated nuclear operations. The adjusted reference temperatures have been calculated by adding the predicted radiation-induced RT NDT and the unirradiated RT NDT for each of the reactor coolant beltline materials.

The predicted RT NDT was calculated using the respective neutron fluence after five effective full power years of operation. To reflect the uncertainty in the copper and phosphorus concentration of the beltline welds, the upper limit of Figure 1 of Regulatory Guide 1.99, Revision 1, was utilized for added conservatism. The analysis of the reactor vessel material contained in the first surveillance capsule removed from Three Mile Island Nuclear Station Unit 1 confirmed that the current techniques used for predicting the change in impact properties due to irradiation are conservative.

Analysis of the activation detectors contained in the first surveillance capsule indicates that the average fast flux during Cycle 1 was 1.45×10^{10} n/cm^2-sec maximum at the pressure vessel wall. Extrapolation of the Cycle 1 flux based on predicted fuel reload and burn-up conditions indicates that the maximum average fast neutron (E>1 Mev) flux during six full power years of operation will be 1.68×10^{10} n/cm^2-sec at the reactor vessel wall and 9.33×10^9 n/cm^2-sec at the ¼ T location. The fast neutron exposure during five effective full power years of operation, therefore, is 1.5×10^{18} n/cm^2 at the ¼ T location and 3.7×10^{17} $n cm^2$ at the 3/4 T location.

Based on the predicted RT NDT after five effective full power years of operation, the pressure-temperature limits of Figure 3.1-1 and 3.1-2 have been established in accordance with the requirements of 10 CFR 50, Appendix G. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046. The protection against nonductile failure is assumed by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figures 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12° F error in measured temperature.

c. System pressure is measured in either loop.

d. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) BAW-1439, Analysis of Capsule TMI-IE From Metropolitan Edison Company, Three Mile Island Nuclear Station - Unit #1, Reactor Vessel Materials Surveillance Program.

Assumed RT_{NDT}, °F

Beltline 1/4 T	145
Beltline 3/4 T	82
Closure Head Region	60
Outlet Nozzle	60

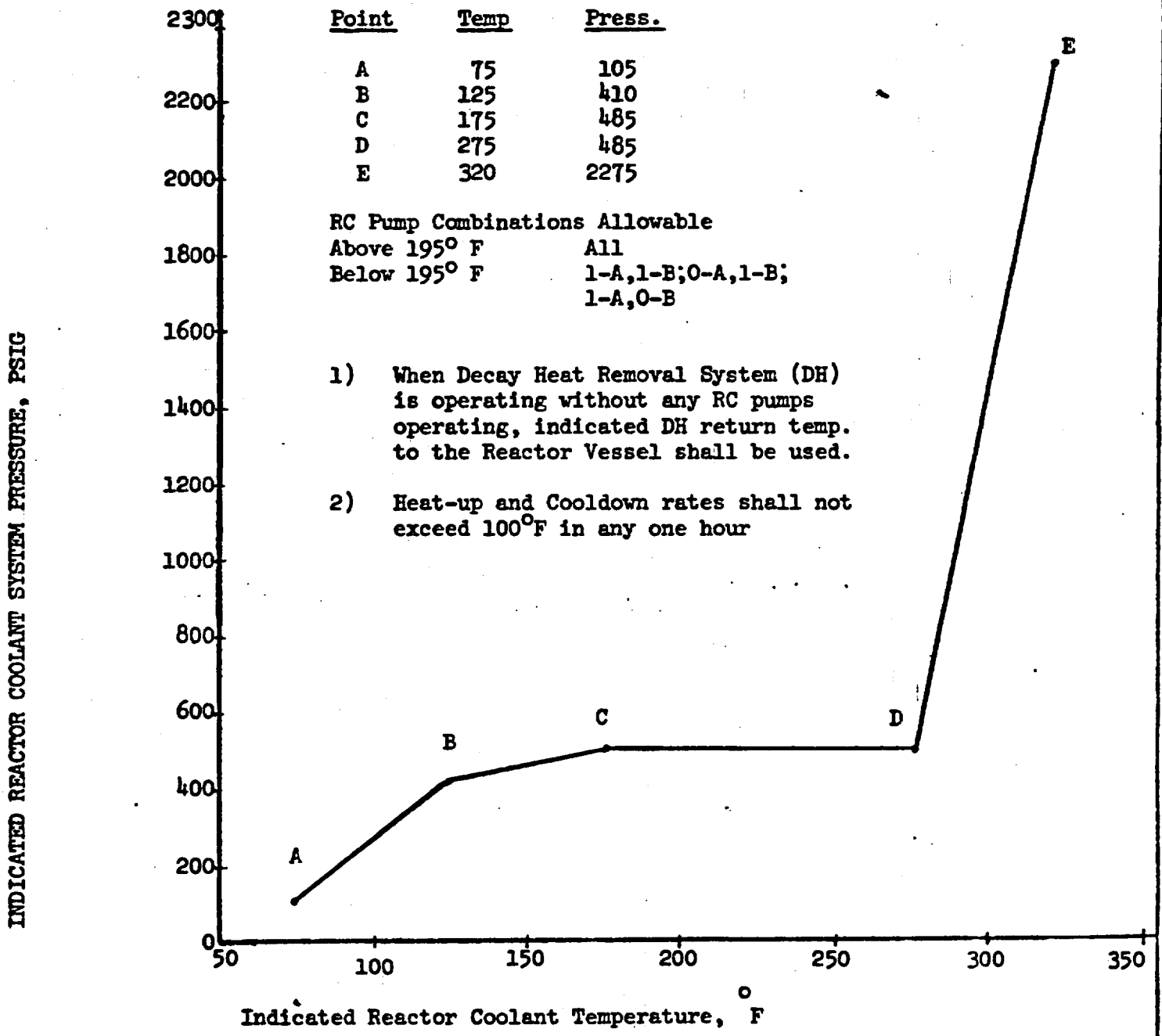
Point	Temp	Press.
A	75	105
B	125	410
C	175	485
D	275	485
E	320	2275

RC Pump Combinations Allowable

Above 195° F	All
Below 195° F	1-A,1-B;0-A,1-B; 1-A,0-B

1) When Decay Heat Removal System (DH) is operating without any RC pumps operating, indicated DH return temp. to the Reactor Vessel shall be used.

2) Heat-up and Cooldown rates shall not exceed 100°F in any one hour



Reactor Coolant System
Heat-up/Cooldown Limitations
(Applicable to 5 EFPY)
Figure 3.1-1

Assumed RT_{NDT}, °F

Beltline Region 1/4 T	145
Beltline Region 3/4 T	82
Closure Head Region	60
Outlet Nozzle	60

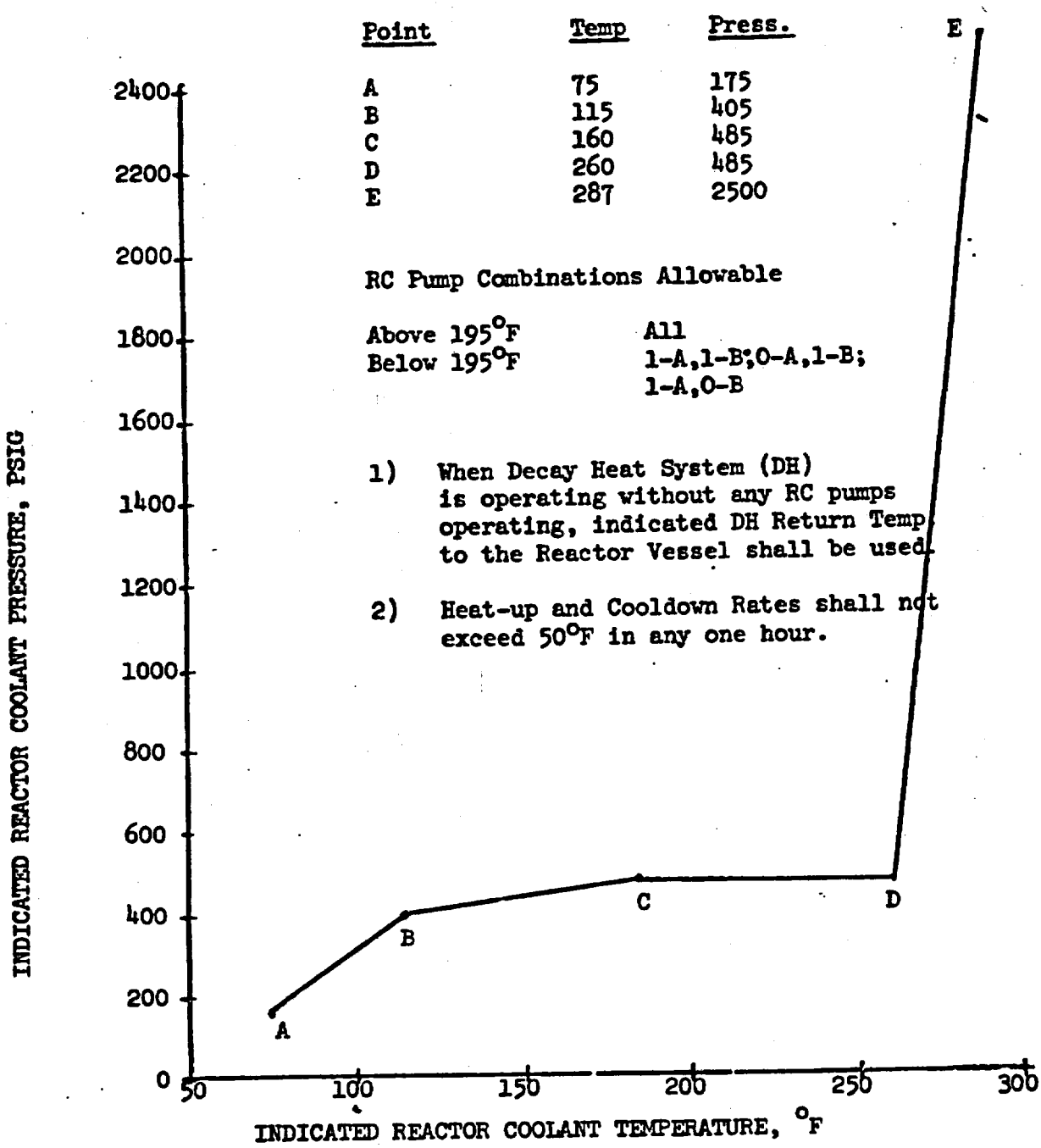
Point	Temp	Press.
A	75	175
B	115	405
C	160	485
D	260	485
E	287	2500

RC Pump Combinations Allowable

Above 195°F	All
Below 195°F	1-A,1-B;0-A,1-B; 1-A,0-B

1) When Decay Heat System (DH) is operating without any RC pumps operating, indicated DH Return Temp to the Reactor Vessel shall be used.

2) Heat-up and Cooldown Rates shall not exceed 50°F in any one hour.



Reactor Coolant System Inservice Leak & Hydrostatic Test Limitations (Applicable to 5 EFPY)

Figure 3.1-2

3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to maximum positive moderator temperature coefficient of reactivity at full power conditions.

Objective

To assure that the moderator temperature coefficient stays within the limits calculated for safe operation of the reactor.

Specification

3.1.7.1 The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

3.1.7.2 The moderator temperature coefficient shall be $\leq + 0.5 \times 10^{-4}$ $\Delta k/k/F$ at power levels $\leq$ 95% of rated power.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.5 \times 10^{-4}$ $\Delta K/K/F$. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta K/K/F$.

The experimental value of the moderator coefficient will be corrected to obtain the hot full power moderator coefficient. The correction factor will be verified during startup testing on earlier B&W reactors.

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

REFERENCES

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

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 shall be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant tilt:

a. Except for physics tests the quadrant tilt shall not exceed +2.66% 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 +1.47% as determined using the minimum incore detector system.

c. When neither incore detector system above is available and except for physics tests quadrant tilt shall not exceed +0.81% as determined using the power range channels displayed on the console for each quadrant (out of core detector system).

d. Except for physics tests if quadrant tilt exceeds the tilt limit power shall be reduced immediately to below the power level cutoff (see Figures 3.5-2A, 3.5-2B and 3.5-2C). Moreover, the power level cutoff value shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent of the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following adjustments in setpoints and limits shall be made:

1. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt.

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

3. The operational imbalance limits (Figure 3.5-2G, 3.5-2H and 3.5-2I) 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 +25.28% determined using the full incore detector system (FIT), or +24.09% determined using the minimum incore detector system (MIT) if the FIT is not available, or +21.39% determined using the out of core detector system (OCT) when neither the FIT nor MIT are available, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant tilt is permitted provided that the thermal power allowable is restricted as stated in 3.5.2.4.d above.

g. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated power.

3.5.2.5 Control Rod Positions:

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

b. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on Figures 3.5-2A, 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 or two pump operation. Also excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on Figures 3.5-2K, 3.5-2L, and 3.5-2M. If any of these control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within four hours.

c. Except for physics tests, power shall not be increased above the power level cutoff (See Figures 3.5-2A, 3.5-2B and 3.5-2C) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

d. Core imbalance shall be monitored on a minimum frequency of once every two hours during power operation above 40 percent of rated power. 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 3.5-2G, 3.5-2H and 3.5-2I. If the imbalance is not within the envelope defined by Figures 3.5-2G, 3.5-2H and 3.5-2I 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 3.5-2J.

Bases

The power-imbalance envelope defined in Figures 3.5-2G, 3.5-2H, and 3.5-2I is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2J) such that the maximum clad temperature will not exceed the Final Acceptance Criteria(2200F). Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the withdrawal/insertion limits as defined by

Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E, 3.5-2F, 3.5-2K, 2.5-2L, and 3.5-2M and if quadrant tilt is at the limit. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors.

e. Postulated fuel rod bow effects

The Rod index versus Allowable Power curves of Figures 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E, 3.5-2F, 3.5-2K, 3.5-2L and 3.5-2M 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 (Xenon transient override)
8	APSR (axial power shaping bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6 and 7 are overlapped 25 percent. The normal position at power is for groups 6 and 7 to be partially inserted.

The 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% Δk/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% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0%Δk/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% Δk/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 +3.41% which is equivalent to a +2.66% tilt measured with the full incore instrumentation with measurement uncertainties included.

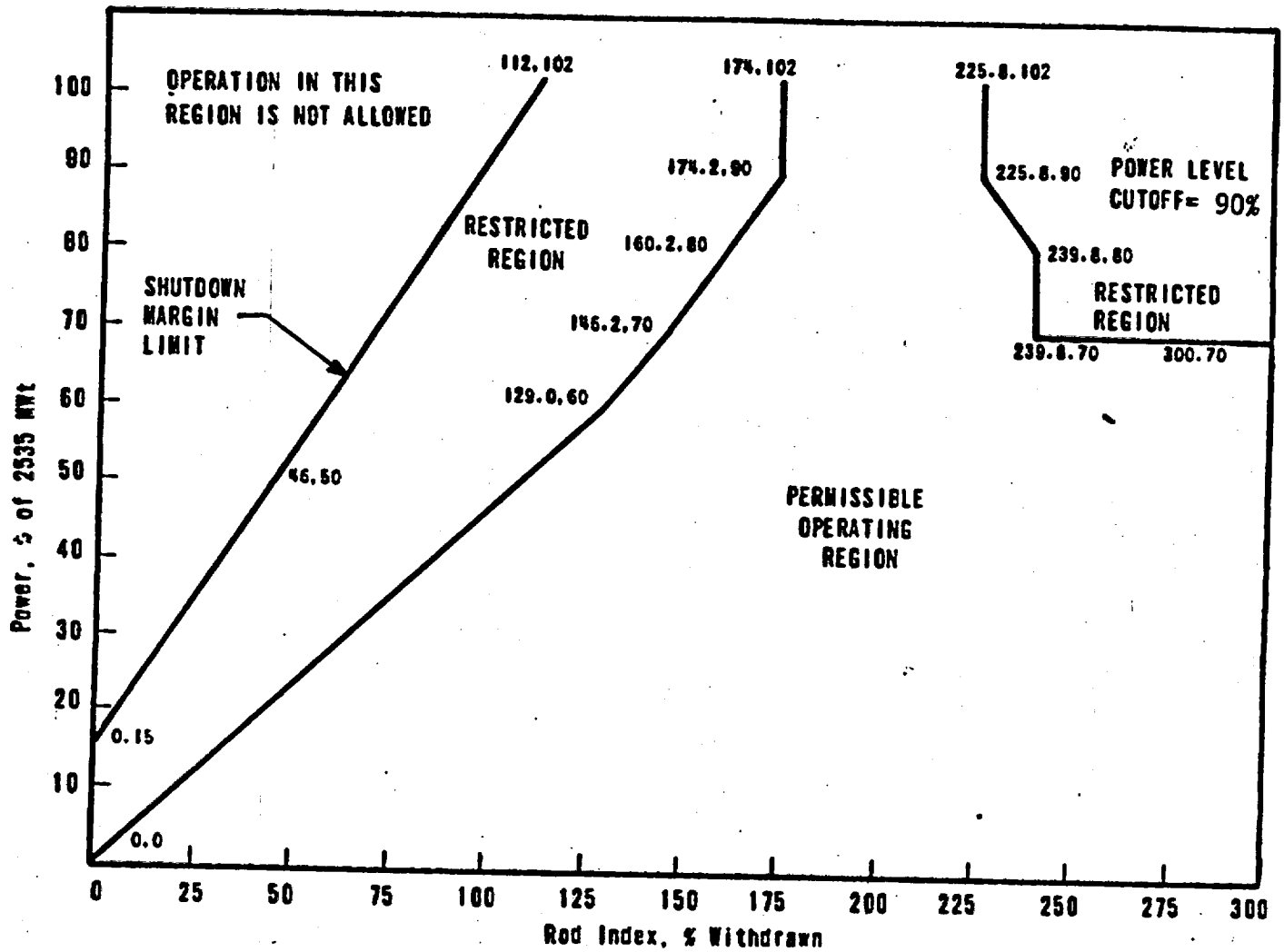
During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

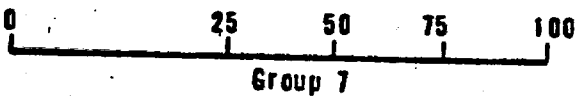
Test Power	Trip Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.5%

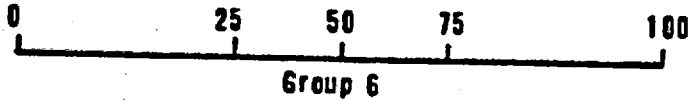
REFERENCES

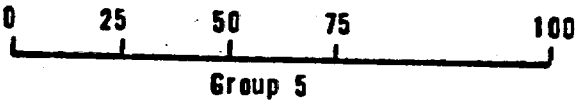
(1) FSAR, SEction 3.2.2.1.2

(2) FSAR, Section 14.2.2.2



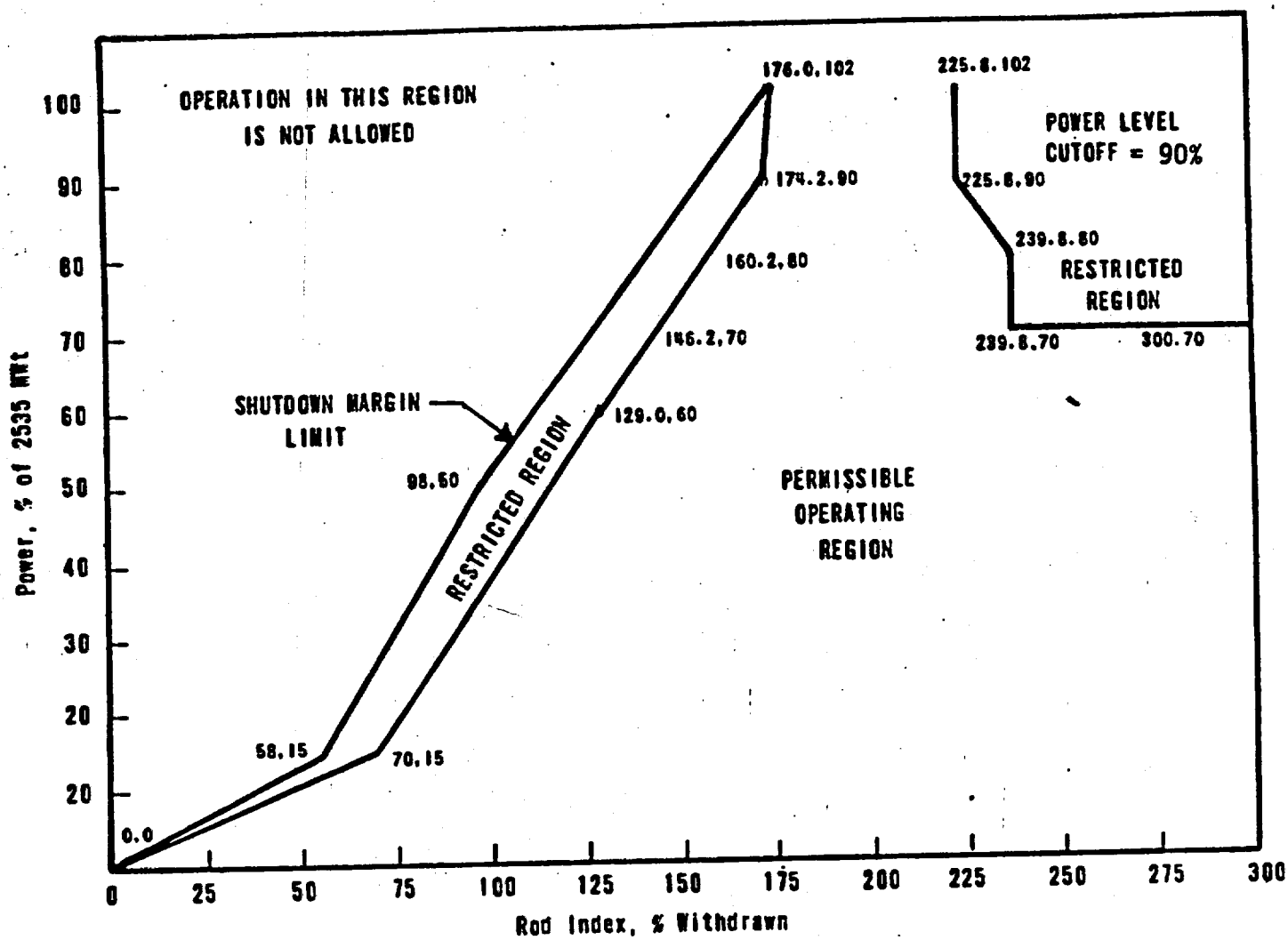


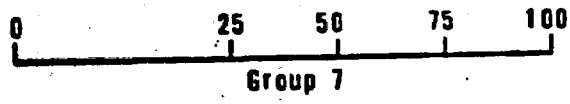


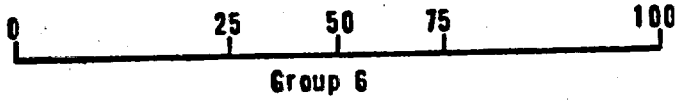


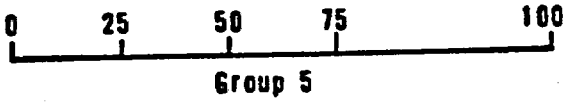
ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1, CYCLE 3

Figure 3.5-2A





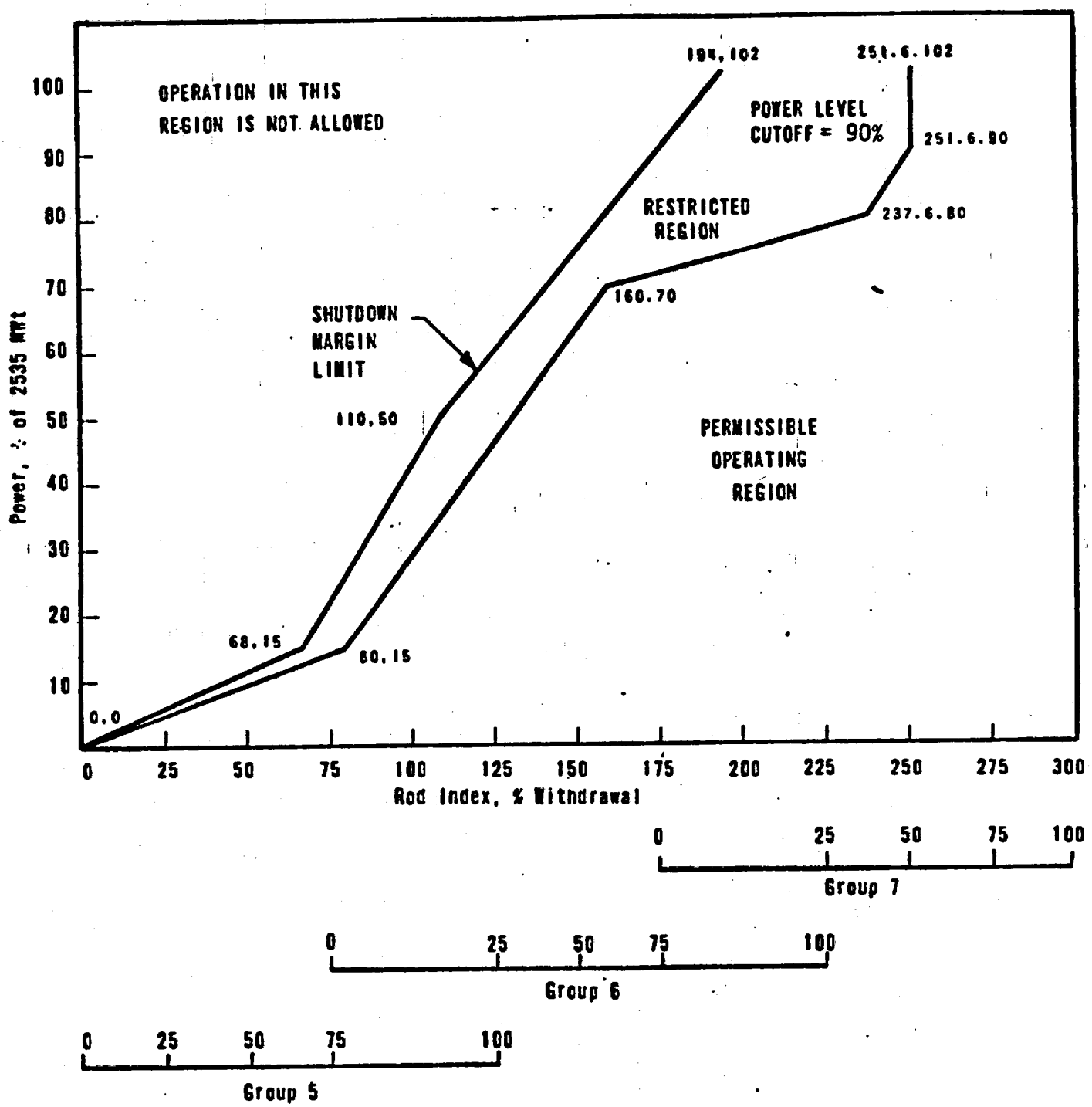




ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 100 ± 10 EFPD TO 246 ± 10 EFPD

TMI-1, CYCLE 3

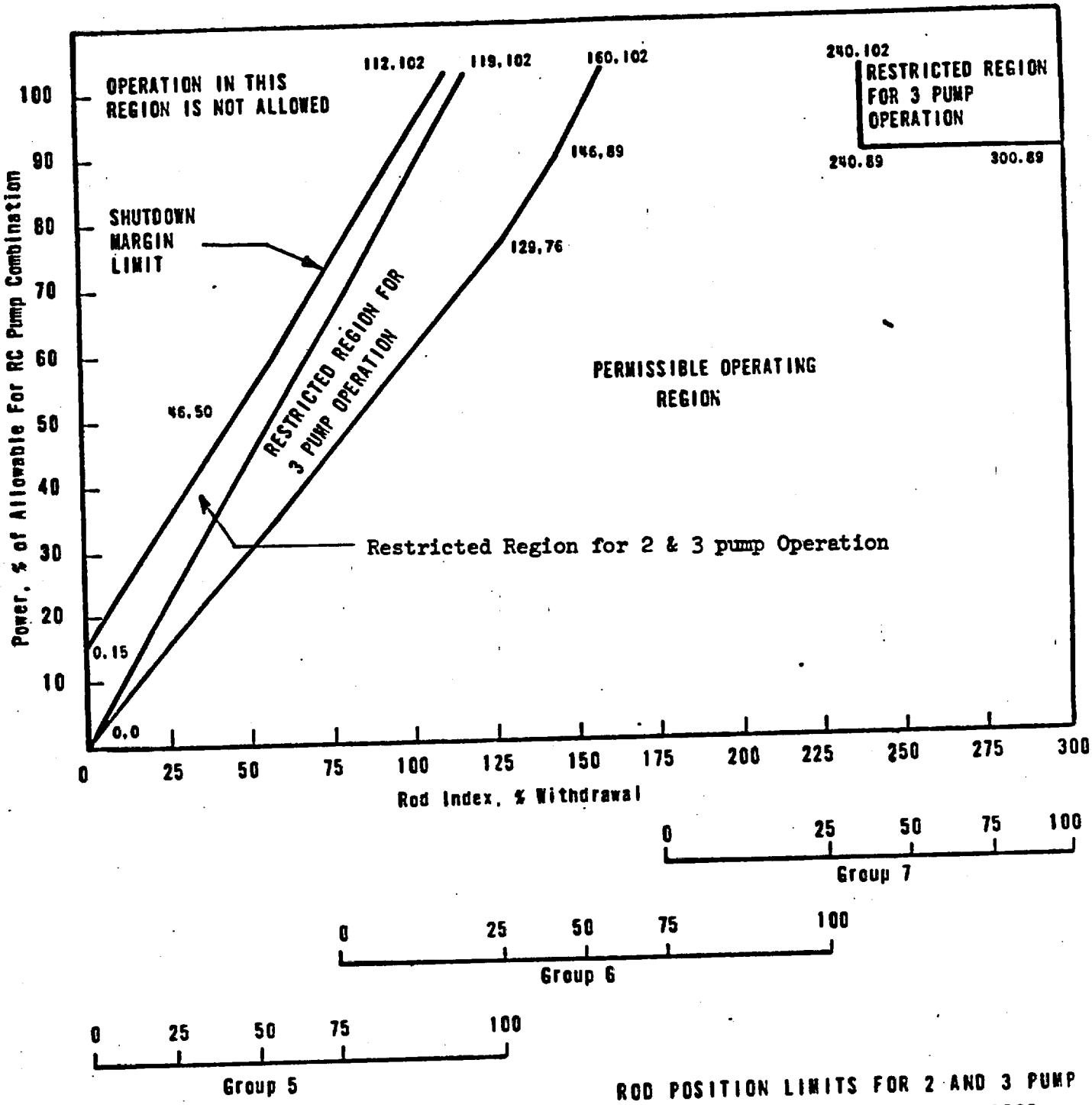
Figure 3.5-2B



ROD POSITION LIMITS FOR 4 PUMP
OPERATION AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3

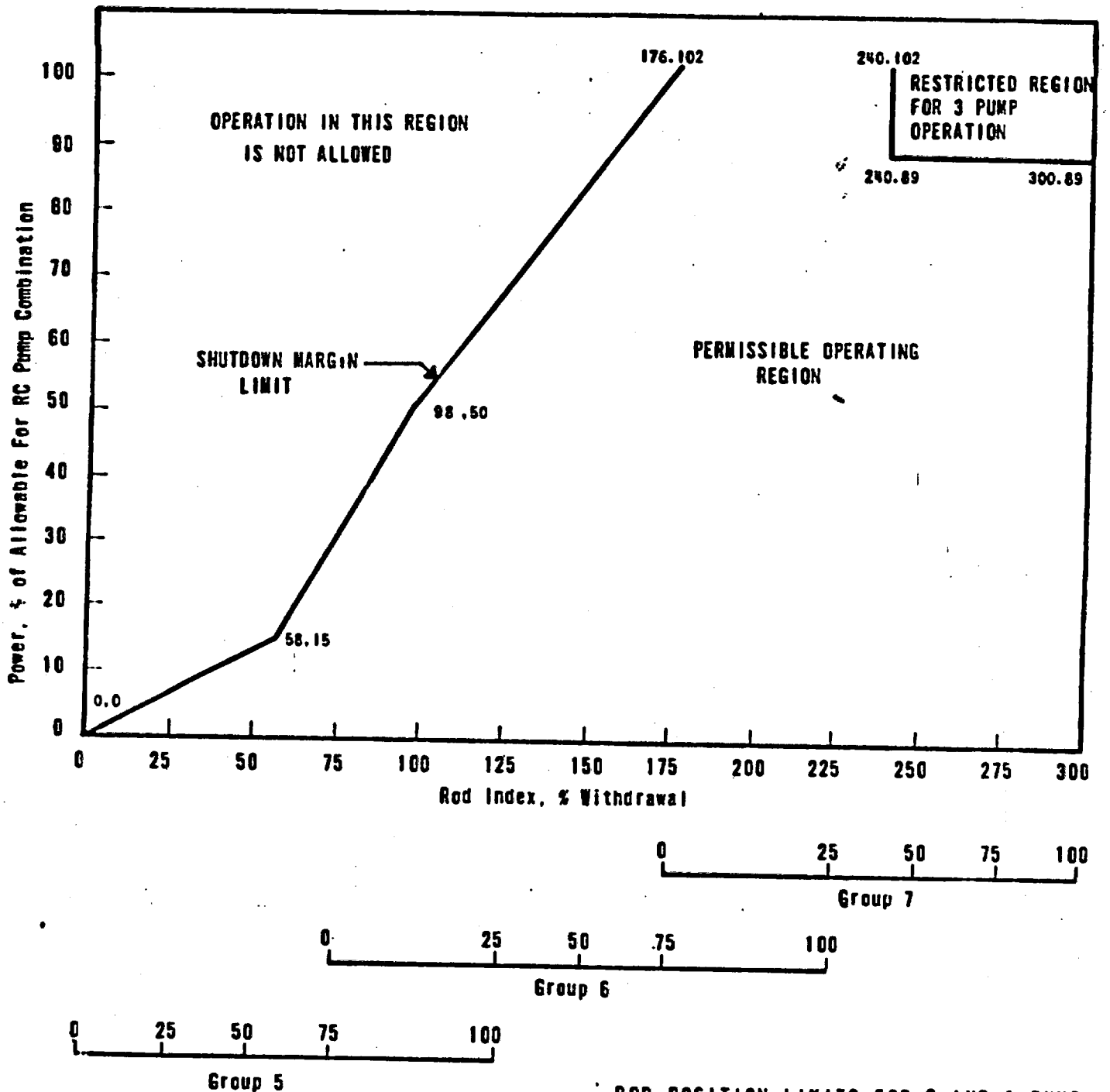
Figure 3.5-2C

Amendment No. ~~16~~, ~~17~~, 29



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1 CYCLE 9

Figure 3.5-2D

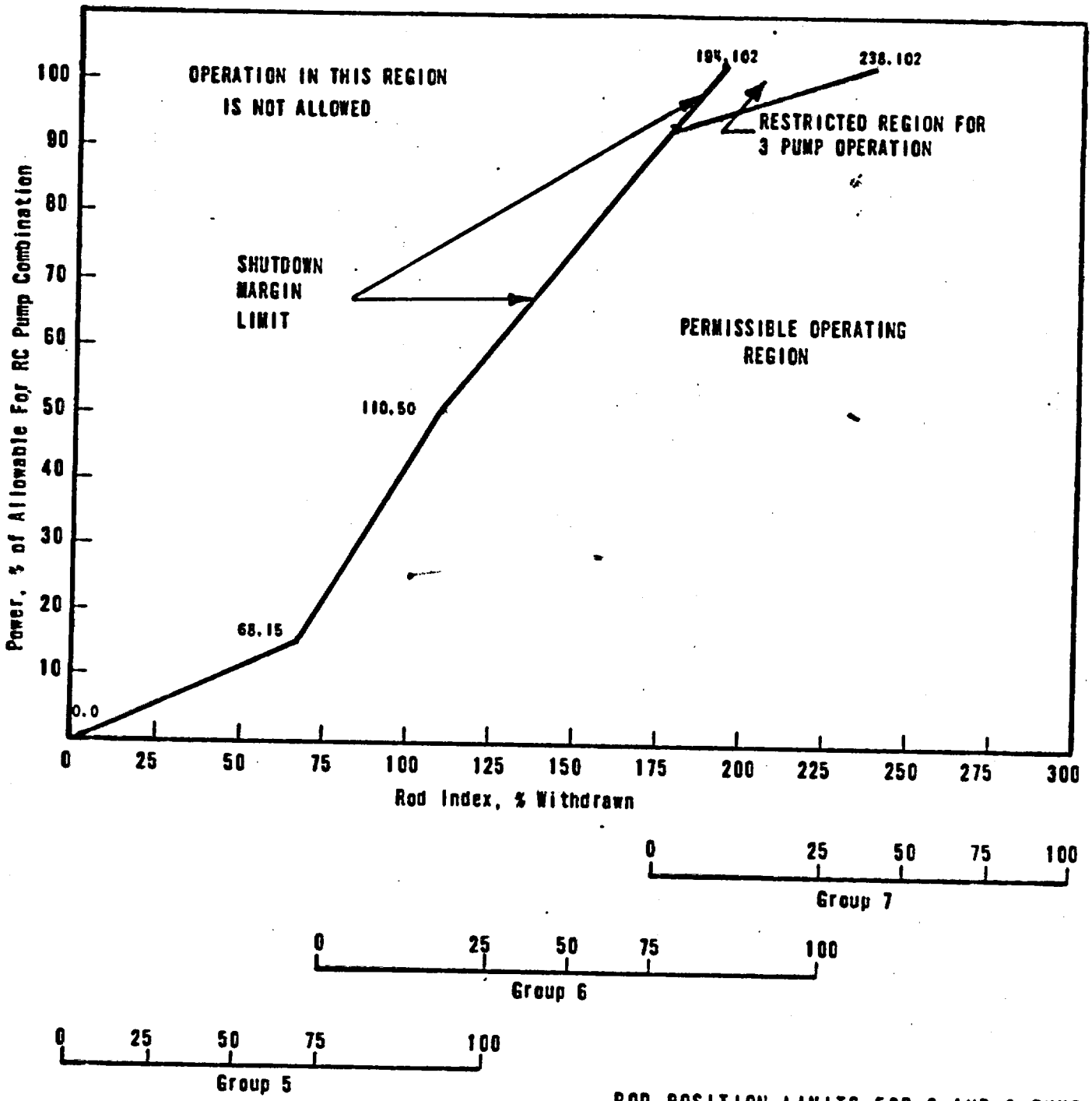


ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION FROM 100 ± 10 EFPD TO 246 ± 10 EFPD

TMI-1, CYCLE 3

Figure 3 5-2E

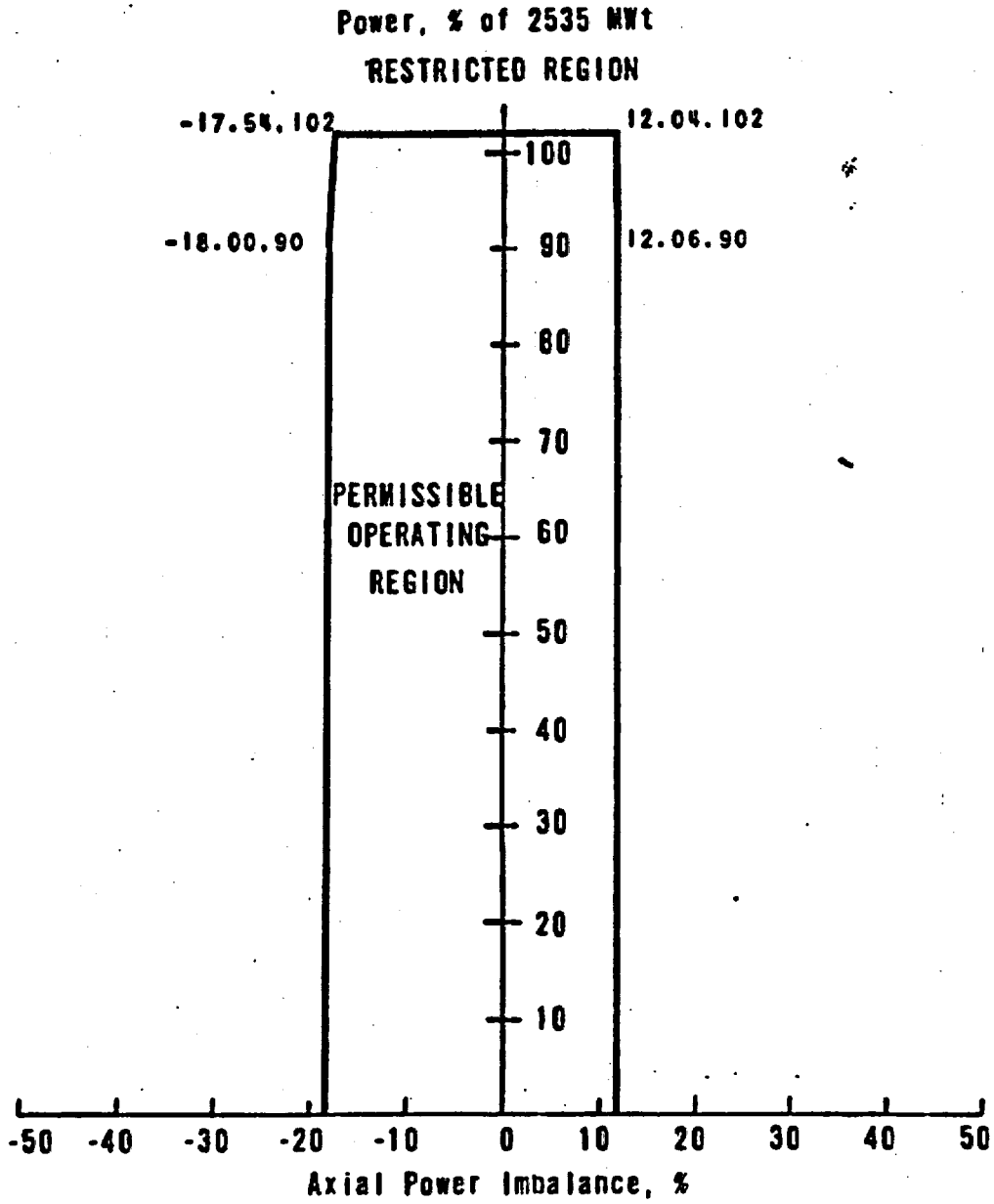
Amendment No. ~~17~~, 29



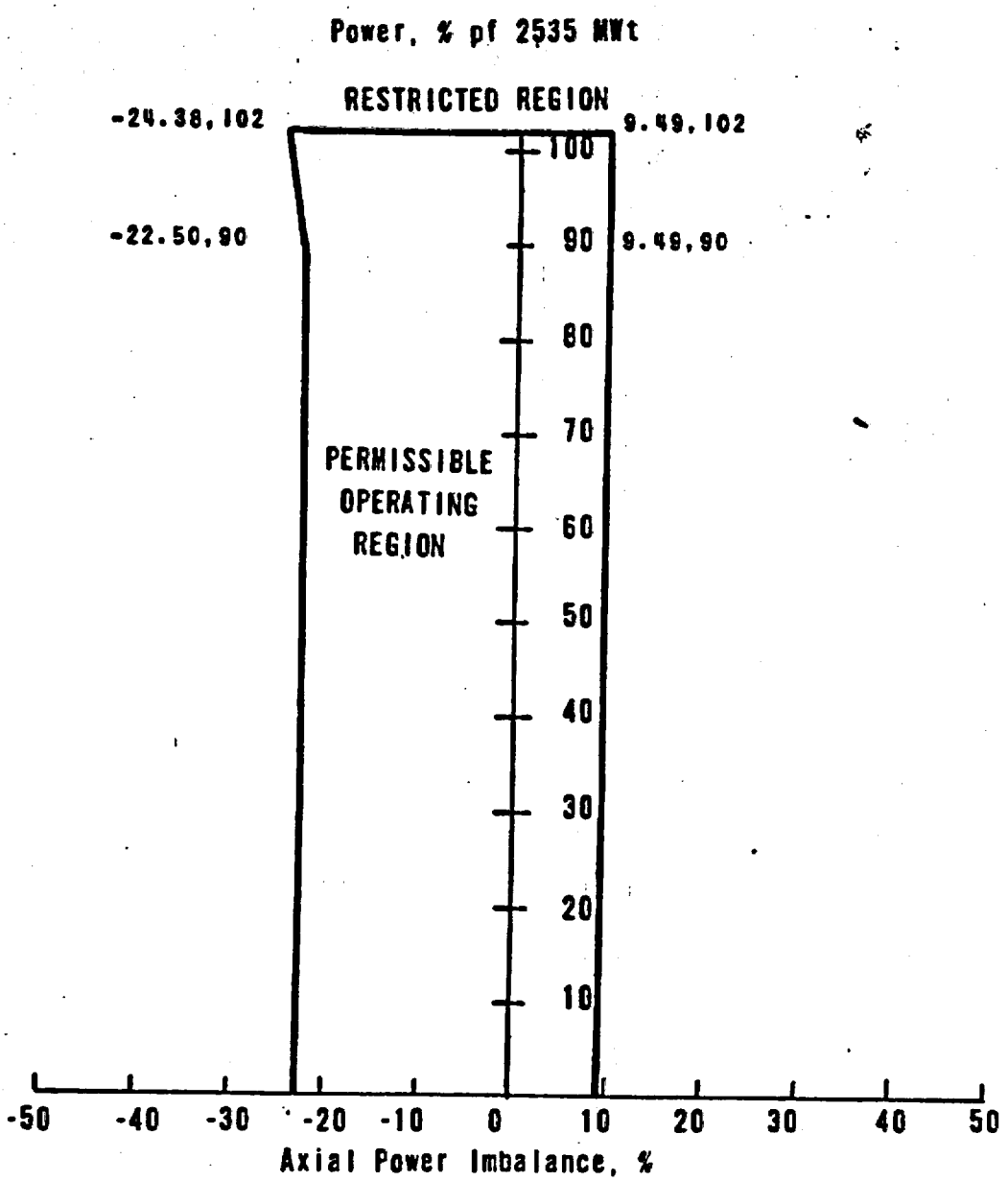
ROD POSITION LIMITS FOR 2 AND 3 PUMP
OPERATION AFTER 246 ± 10 EFPD
TMI-1, CYCLE 3

Figure 3.5-2F

Amendment No. ~~17~~, 29



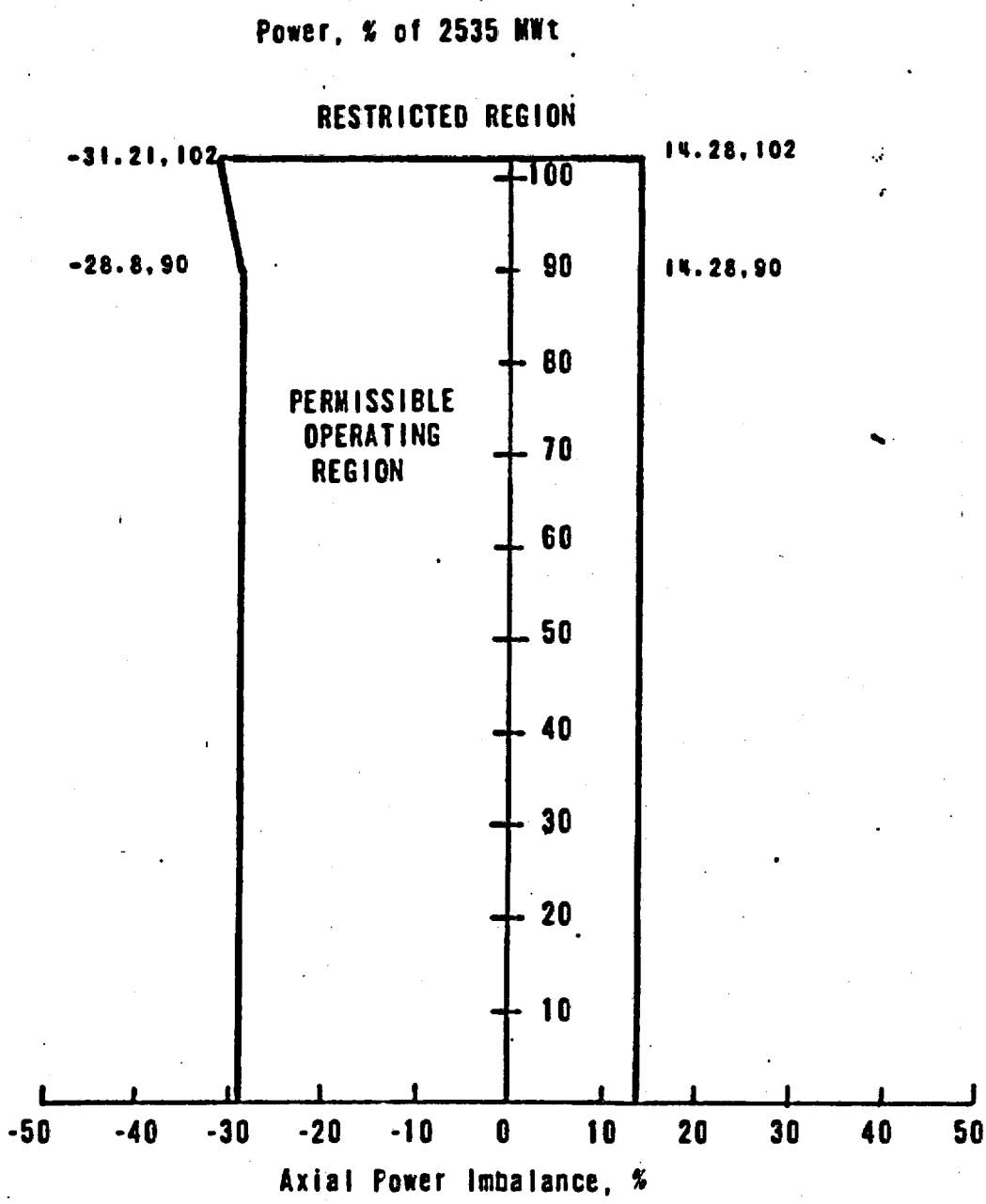
POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 0 TO 100 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2G



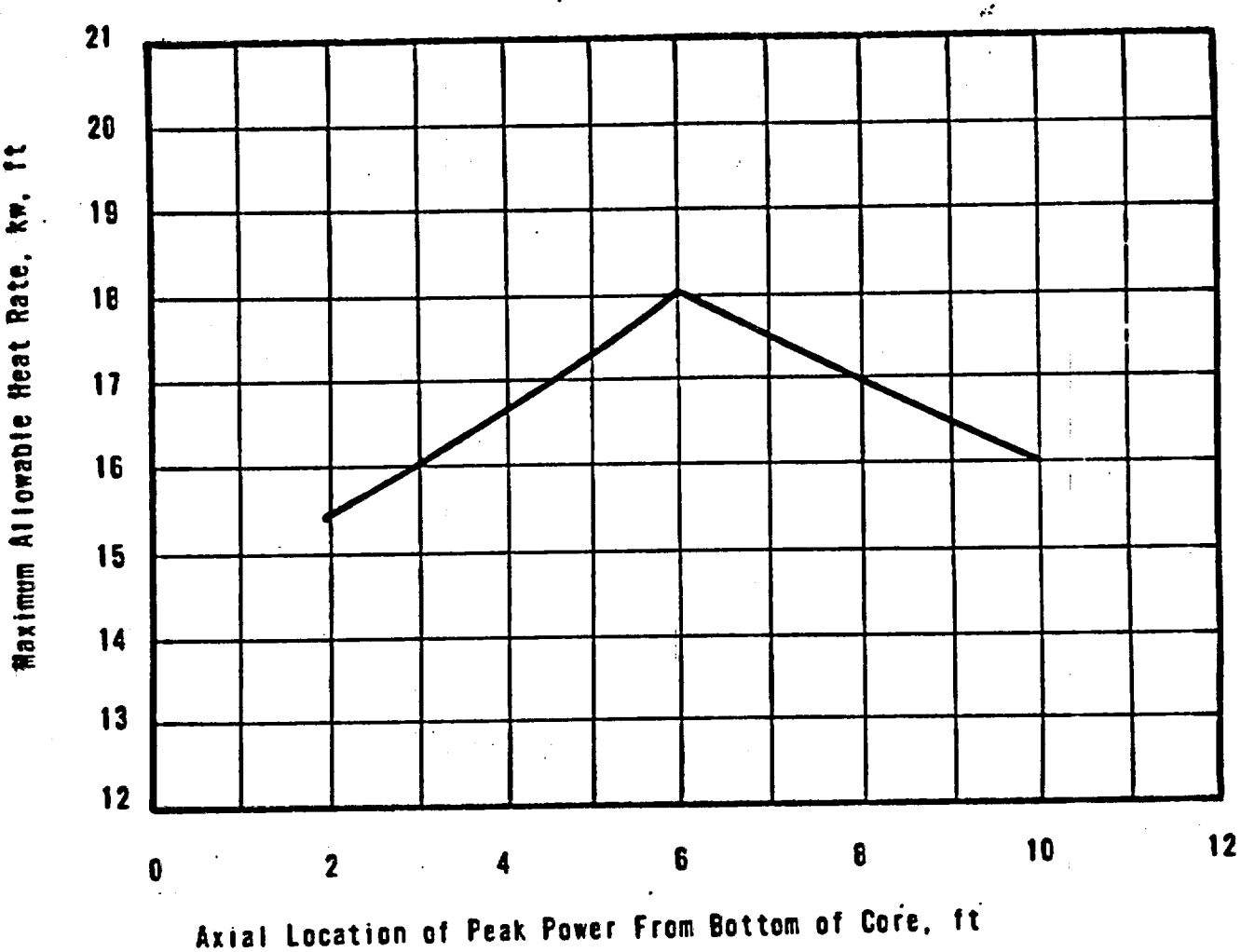
POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 100 ± 10 TO
246 ± 10 EFPD

TMI-1 CYCLE 3

Figure 3.5-2H

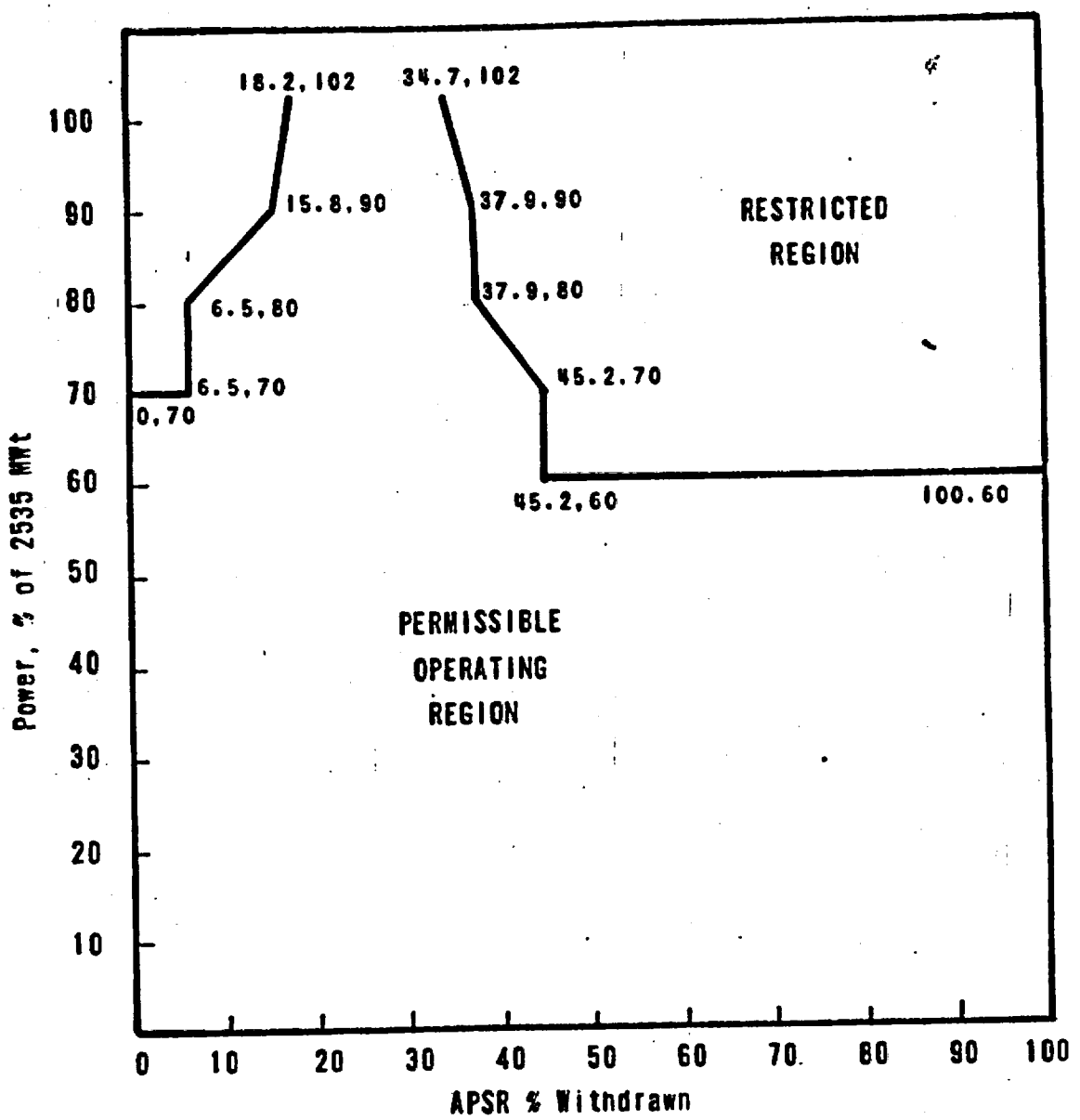


POWER IMBALANCE ENVELOPE FOR
OPERATION AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2I



LOCA LIMITED MAXIMUM ALLOWABLI
LINEAR HEAT RATE

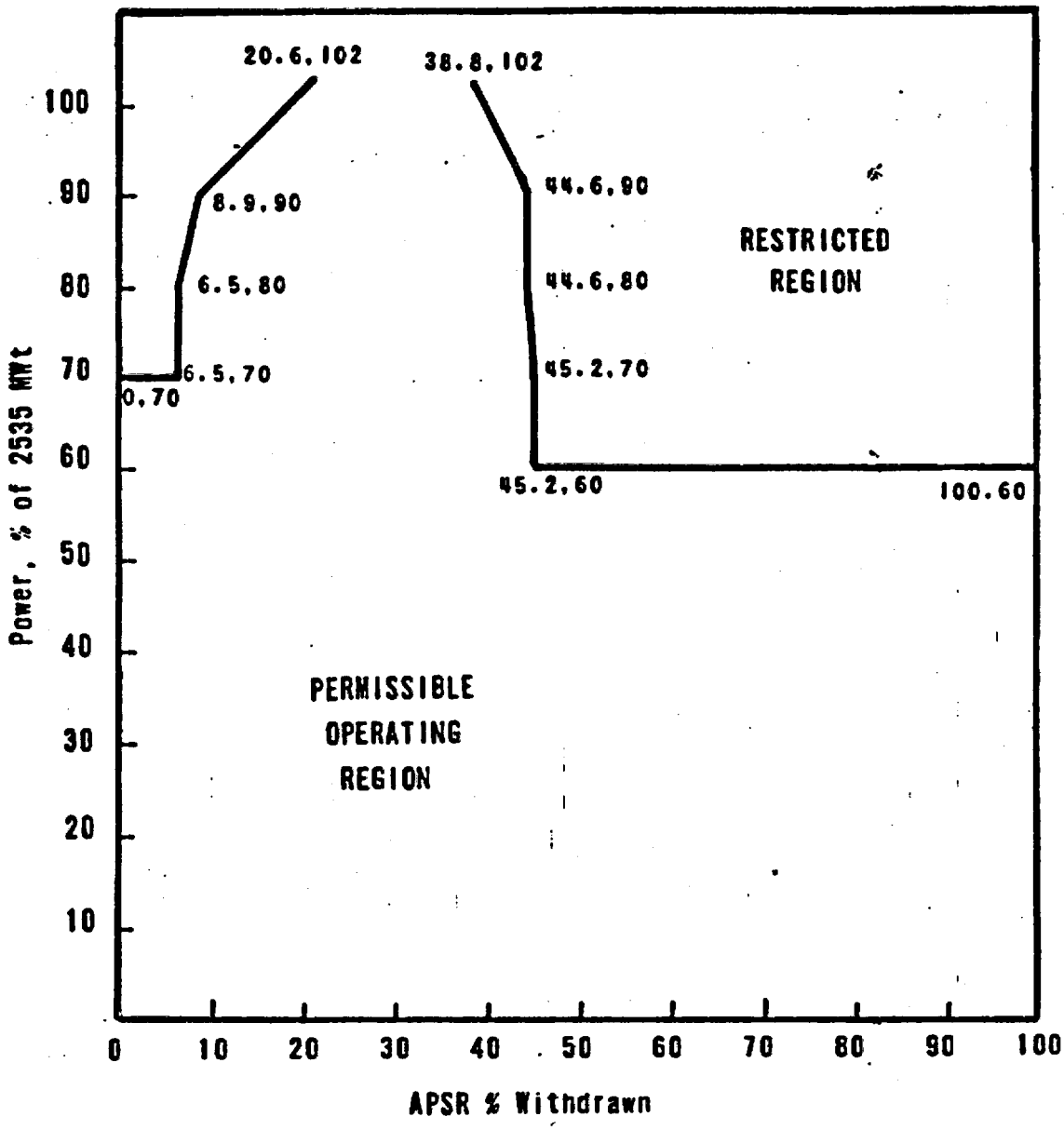
Figure 3.5-2.



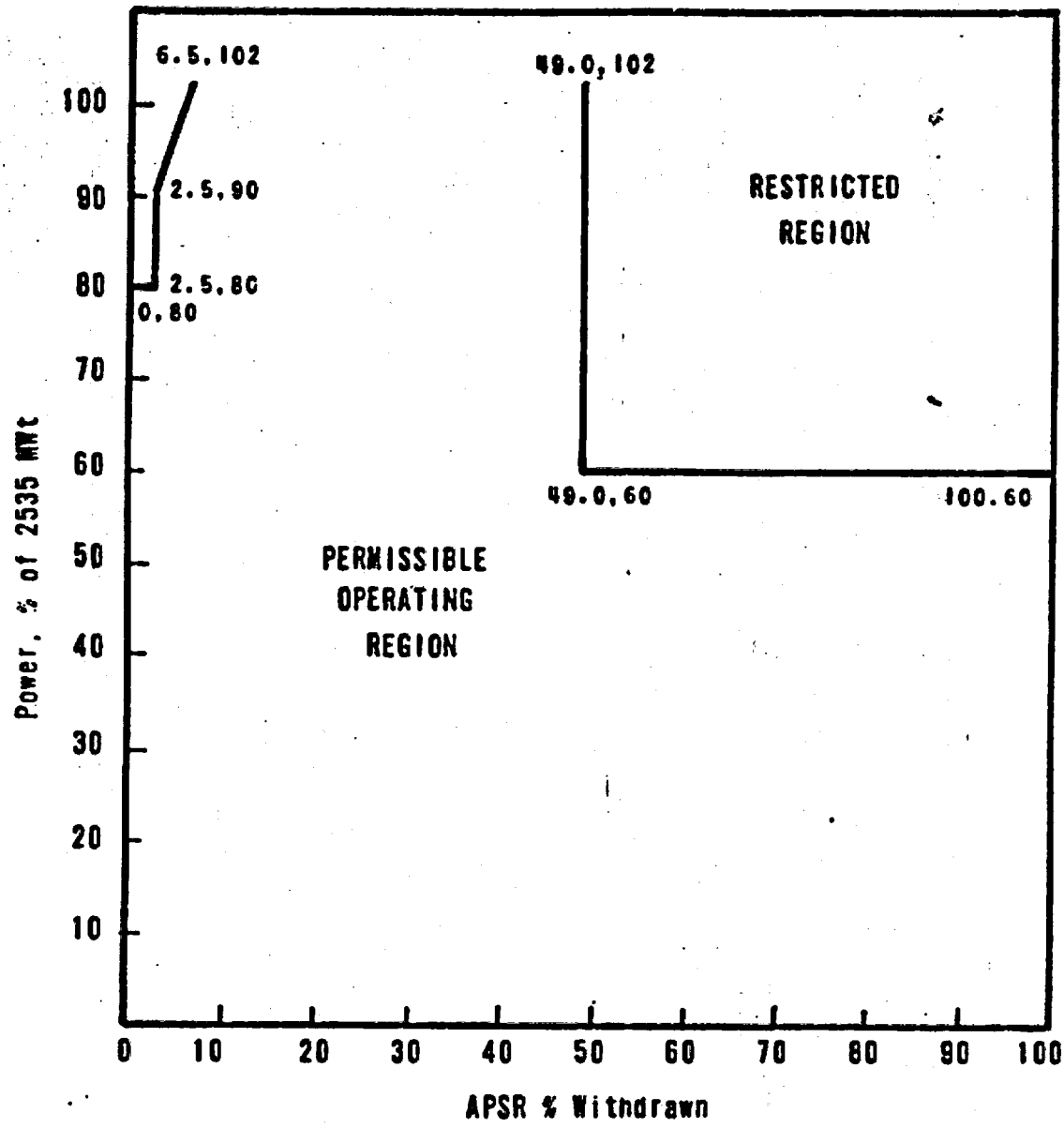
APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 ± 10 EFPD

TMI-1 CYCLE 3

Figure 3.5-2K



APSR POSITION LIMITS FOR
OPERATION FROM 100 ± 10
TO 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2L



APSR POSITION LIMITS FOR OPERATION
AFTER 246 ± 10 EFPD
TMI-1 CYCLE 3
Figure 3.5-2M

4.2 REACTOR COOLANT SYSTEM INSERVICE INSPECTION

Applicability

This technical specification applies to the inservice inspection of the reactor coolant system pressure boundary and portions of other safety oriented system pressure boundaries as shown on Figure 4.2-1.

Objective

The objective of this inservice inspection program is to provide assurance of the continuing integrity of the reactor coolant system while at the same time minimizing radiation exposure to personnel in the performance of inservice inspections.

Specification

4.2.1 The inservice inspection program to be followed is outlined in Table 4.2-1. Except as provided for in this table and as discussed herein, the inservice inspection program is in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, dated January 1, 1970, as modified by the Winter 1970 Addenda. Prior to initial plant operation a preoperational inspection of the plant will be performed of at least the areas listed in the ASME Code; provided accessibility and the necessary inspection techniques are available for each of these areas. The only exception to this will be areas where the necessary base line data is already available and has been obtained by the same techniques as will be used during inservice inspection.

4.2.2 The reactor vessel material surveillance capsules removed from TMI-1 during 1976 shall be inserted, irradiated in and withdrawn from the Three Mile Island Unit No. 2 reactor vessel in accordance with the schedule shown in Table 4.2-2. The licensee shall be responsible for the examination of these specimens and for submission of reports of test results in accordance with 10 CFR 50, Appendix H.

4.2.3 The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within 3-1/3 years, two within 6-2/3 years, and all four by the end of the 10 year inspection interval. However, the U.T. procedure is developmental and will be used only to the extent that it is shown to be meaningful. The extent of coverage will be limited to those areas of the flywheel which are accessible without motor disassembly, i.e., can be reached through the access ports. Also, if radiation levels at the lower access ports are prohibitive, only the upper access ports will be used.

4.2.4 The inspection schedule may be modified to coincide with those refueling or maintenance outages most closely approaching the inspection schedule.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 The inservice inspection shall be reviewed at the end of five years to consider incorporation of new inspection techniques and equipment which have been proven practical, and a possible extension of the program to additional examination areas. The conclusions of this review shall be submitted to the NRC for evaluation.

4.2.7 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of either of the following:

1. Failure of Three Mile Island Unit No. 2 to achieve commercial operation at 100% power by October 1, 1978, or

2. Beginning one year after attainment of commercial operation at 100% power, any time that Three Mile Island Unit No. 2 fails to maintain a cumulative reactor utilization factor of at least 65%.

The report shall provide justification for continued operation of TMI-1 with the reactor vessel surveillance program conducted at Three Mile Island Unit No. 2, or the application for license amendment shall propose an alternative program for conduct of the TMI-1 reactor vessel surveillance program.

For the purpose of this technical specification, the definition of commercial operation is that given in Regulatory Guide 1.16, Revision 4. The definition of cumulative reactor utilization factor is:

Cumulative reactor utilization factor = (Cumulative megawatt hours (thermal) since attainment of commercial operation at 100% power x (100)) divided by (licensed power (MWt) x (Cumulative hours since attainment of commercial operation at 100% power)).

4.2.8 In addition to the reports required by Specification 4.2.7, a report shall be submitted to the NRC prior to September 1, 1982, which summarizes the first five years of operating experience with the TMI-1 integrated surveillance program performed at TMI-2. If, at the time of submission of this report, it is desired to continue the surveillance program at TMI-2, such continuation shall be justified on the basis of the attained operating experience.

Inspection Bases

a. The nuclear plant was designed prior to the issuance of Section XI of the ASME Code, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems dated January 1, 1970. However, sufficient accessibility was included in the design to perform most inspections discussed in the code. The proposed inspection program follows the code except that inspections are focused on areas which engineering analysis has indicated are subject to the more critical stress, radiation, or transient conditions. The areas selected for inspection on this basis are listed in Table 4.2-1. These areas are exposed to the more severe conditions (which are still well within code limits) in the reactor coolant system. Therefore, they are expected to indicate potential problems before significant flaws develop in the selected areas or in other areas. It is considered that the focused approach specified herein will result in a meaningful inspection program in that it will provide assurance of continuing plant integrity.

In those areas where inspection methods are developmental, such as for remote inspection of the reactor vessel welds, reactor vessel nozzle inside radii and welds, and ultrasonic inspection of pressurizer support bracket welds, the inspection methods will be developed and tested to the extent practicable during preoperational inspections. (Development of inspection techniques will not be attempted on radioactive equipment unless necessary to explore a specific problem.) A preoperational inspection is planned of areas listed in the ASME Code which are within the inservice inspection boundaries and which are accessible for inspection. However, as discussed above, in areas where inspection methods are developmental, the inspections will only be performed to the extent practicable. Once an inspection method is selected for a particular inspection (e.g., U.T. for most volumetric inspections), it is intended that all subsequent inservice inspections be performed using the identical method and on the same component parts wherever practicable.

In addition to the above inspection, if any of the components within the inservice inspection boundary are disassembled for maintenance, the accessible parts will be given a normal visual examination as part of the routine plant maintenance operations.

b. Because of damage to the surveillance capsule holder tubes originally installed in TMI-1, irradiation of the TMI-1 capsules will be conducted in TMI-2 pursuant to 10 CFR 50, Appendix H, Section II.C.4. Because of the similarity of TMI-1 and TMI-2, irradiation in TMI-2 will be substantially equivalent to irradiation in TMI-1, and appropriate adjustments and margins can be imposed in applying the irradiation data to account for such differences as do exist.

The withdrawal schedule has been formulated to optimize the availability of irradiation data from the capsules of both Units 1 and 2.

Because the irradiation program is dependent upon the successful operation and a reasonable utilization of TMI-2, reporting requirements are included to permit reevaluation of the program if TMI-2 does not achieve full power operation in a reasonable period of time or suffers extended outages after the first year of operation.

c. The reactor coolant pump motor flywheel ultrasonic test procedure is being developed to detect flaws of a small enough size to provide assurance of continued integrity based upon a conservative fracture mechanic's evaluation.

REFERENCE

(1) FSAR, Section 4.4

(2) BAW-10100A, February 1975

Notes To Table

1. Inspections of the reactor vessel shell welds and other welds in the reactor vessel annulus, will be attempted using remote U.T. inspection equipment from the vessel OD and will only be inspected to the extent practicable. Further, reactor vessel head penetrations and reactor vessel head and flange welds may be inspected using remote, automated U.T. equipment. It is not certain that in all cases the remote U.T. equipment can practicably be made to provide meaningful results, and some adjustment to the proposed inspections may be necessary after completion of preservice development.

2. The most probable defects in studs or bolts are cracks starting from the roots of threads or in the fillet radii at the ends of the shanks. These may be difficult to detect using U.T. Likewise, in nuts, the most probable defects may not be revealed by U.T. Therefore, surface inspections ("Magnaglo") of studs, bolts and nuts may be performed in lieu of U.T. if joints are disassembled for other reasons. Ligaments and threads in base material will be visually inspected when joints are disassembled for other reasons.

3. In the event that it is demonstrated that ultrasonic inspection from the outside of a vessel can provide a meaningful inspection of interior cladding, then U.T. may be substituted for surface or visual inspection of cladding.

TABLE 4.2-2

SURVEILLANCE CAPSULE INSERTION & WITHDRAWAL SCHEDULE AT TMI-2

Capsule Designation	Schedule	
	Insertion	Withdrawal
TMI-1A	TMI-2 Start-up	End of 3rd Cycle
TMI-1B	End of 1st Cycle	End of 6th Cycle
TMI-1C	End of 3rd Cycle	End of 11th Cycle
TMI-1D	End of 6th Cycle	End of 15th Cycle
TMI-1E	Removed end of 1st Cycle of TMI-1	
TMI-1F	End of 10th Cycle	End of 24th Cycle

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2285 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs for modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section IX, IS-400.

For normal opening, the integrity of the Reactor Coolant System, in terms of strength, is unchanged. If the system does not leak at 2285 psig (operating pressure +100 psi; ±50 psi is normal system pressure fluctuation), it will be leak tight during normal operation.(1)

REFERENCE

(1) FSAR, Section 4