UNITED STATES NUCLEAR REGULATORY COMMISSION

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-50

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

   A. The application for amendment by Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company (the licensees) dated April 16, 1975, and supplement dated May 29, 1975, 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 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.c.(2) of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 7."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Richard Dennis, acting

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 7 to the
Technical Specifications

Date of Issuance: JUN 6 1975

JUN 6 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 7 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace pages 3-5 - 3-6, 3-7 - 3-8, 3-35, and 3-35a - 3-36 with the attached revised pages. (No change has been made on pages 3-5 and 3-8.)

Replace figures 3.5-2A - 3.5-2B, 3.5-2C - 3.5-2D and 3.5-2E with the attached revised figures. Add figure 3.5-2F.

The pressure limit li[illegible] on Figure 3.1-1 has been sel[illegible]ted such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

a. A 25 psi error in measured pressure

b. System pressure is measured in either loop

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

For adequate conservatism, in lieu of portions of the Fracture Toughness Testing Requirements of the proposed Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50°F in any one hour has been imposed below 275 F as shown on Figure 3.1-1. 5

The spray temperature difference restriction, based on a stress analysis of the 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) FSAR, Section 4.3.3

(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

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3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525 F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10 F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least one percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

3.1.3.5 Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality with the following exceptions:

(a) Inoperable rod per 3.5.2.2.

(b) Physics testing per 3.1.9.

(c) Shutdown margin may not be reduced below 1% $\Delta k/k$ per 3.5.2.1.

(d) Exercising rods per 4.1.2.

Following safety rod withdrawal, the regulating rods shall be positioned within their position limits as defined by specification 3.5.2.5 prior to deboration. 7

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525 F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature.(2) startup and operation of the reactor when reactor coolant temperature is less that 525 F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT +10 F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal. 7

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.4

Applicability

Applies to the maximum reactor coolant system activity permitted during reactor operation.

Objective

To limit the whole body dose at the site boundary in the event of a double ended rupture of a steam generator tube.

Specification

3.1.4.1 The total activity of the reactor coolant due to nuclides with half lives longer than 30 minutes shall not exceed $130/\bar{E}$ microcuries per ml whenever the reactor is critical. $\bar{E}$ is the average (mean) beta plus the average (mean gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the readionuclids in reactor coolant sampled.

Bases

The above specification is based on limiting the consequences of a postulated accident involving the double-ended rupture of a steam generator tube. The rupture of a steam generator tube enables reactor coolant and its associated activity to enter the secondary system where volatile isotopes could be discharged to the atmosphere through condenser vacuum pumps and through steam safety valves (which may lift momentarily). Since the major portion of the activity entering the secondary system is due to noble gases, the bulk of the activity would be discharged to the atmosphere. The activity release continues until the operator stops the leakage by reducing the reactor coolant system pressure below the set point of the steam safety valves and isolates the faulty steam generator. The operator can identify a faulty steam generator by using the off-gas monitors on the condenser vacuum pump lines; thus he can isolate the faulty steam generator within 34 minutes after the tube break occurred. During that 34 minute period, a maximum of 2763 ft^3 of hot reactor coolant leaked into the secondary system.

The controlling dose for the steam generator tube rupture accident is the whole-body dose resulting from immersion in the cloud of released activity. To ensure that the public is adequately protected, the specific activity of the reactor coolant will be limited to a value which will ensure that the whole-body dose at the site boundary will not exceed 0.5 rem should a steam generator tube rupture accident occur.

Although only volatile isotopes will be released from the secondary system, the following whole-body dose calculation conservatively assumes that all of the radioactivity which enters the secondary system with the reactor coolant is released to the atmosphere. Both the beta and gamma radiation from these isotopes contribute to the whole-body dose. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employs the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud mode is applicable to the beta dose because of the short range of beta radiation in air. It is further assumed that meteorological conditions during the course of the accident correspond to Pasquill Type F and 1 meter per second wind speed, resulting in a χ/Q value of 2×10^{-4} sec/m^3.

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. Except for physics tests or exercising control rods,
the control rod insertion/withdrawal limits are specified
on Figures 3.5-2A (for up to the control rod interchange),
Figure 3.5-2B (from control rod interchange up to 440 full 7
power days of operation), Figure 3.5-2C (for after 440 full power
days of operation) for four pump operation, and Figure 3.5-2D
for three or two pump operation. If the 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 7
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. Corrective measures (reduction of
imbalance by APSR movements and/or reduction in reactor power)
shall be taken to maintain operation within the envelope
defined by Figure 3.5-2E. If the imbalance is not within 7
the envelope defined by Figure 3.5-2E, corrective measures
shall be taken to achieve an acceptable imbalance. If an
acceptable imbalance is not achieved within four hours,
reactor power shall be reduced until imbalance limits are
met.

e. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

3.5.2.7 A power map shall be taken to verify the expected power distribution at periodic intervals of approximately 10 full power days using the incore instrumentation detection system.

<u>Bases</u>

The power-imbalance envelope defined in Figure 3.5-2E is based on LOCA analyses
which have defined the maximum linear heat rate (see Figure 3.5-2F) such that 7
the maximum clad temperature will not exceed the Interim Acceptance Criteria.
Operation outside of the power imbalance envelope alone does not consitute a
situation that would cause the Final Acceptance Criteria to be exceeded should 7
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, and 3.5-2D and if a 4 percent quadrant power tilt exists. Additional conservatism is 7 introducted by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors.

The 30 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	Xenon translent 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 minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming, the highest worth control rod remains in the full out position(1).

Inserted rod groups during power operation will not contain single rod worths greater than 0.65 percent $\Delta k/k$. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident (2). Single inserted control rod worth of 1.0 percent $\Delta k/k$ at beginning of life, hot, zero power would result in lower transient peak thermal power, and therefore, less severe 7
environmental consequences as a 0.65 percent $\Delta k/k$ ejected rod worth at rated power.

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

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6.

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%

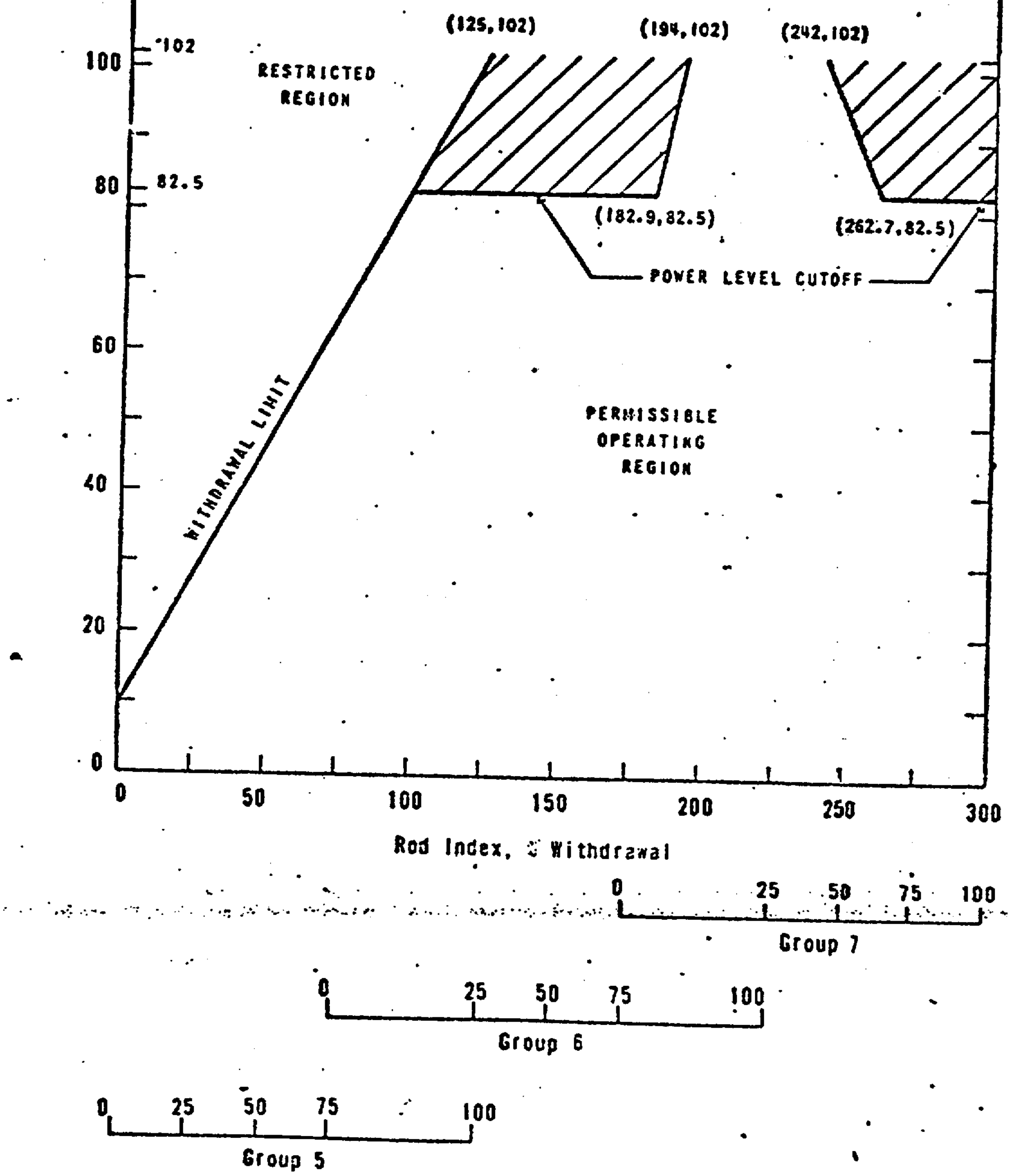
REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

1. Rod index is the percentage sum of the withdrawal of the operating groups.

2. Restrictions on withdrawal (hashed areas) are modified after the control rod interchange (See Figure 3.5-2B)



CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION UNIT 1

Figure 3.5-2A

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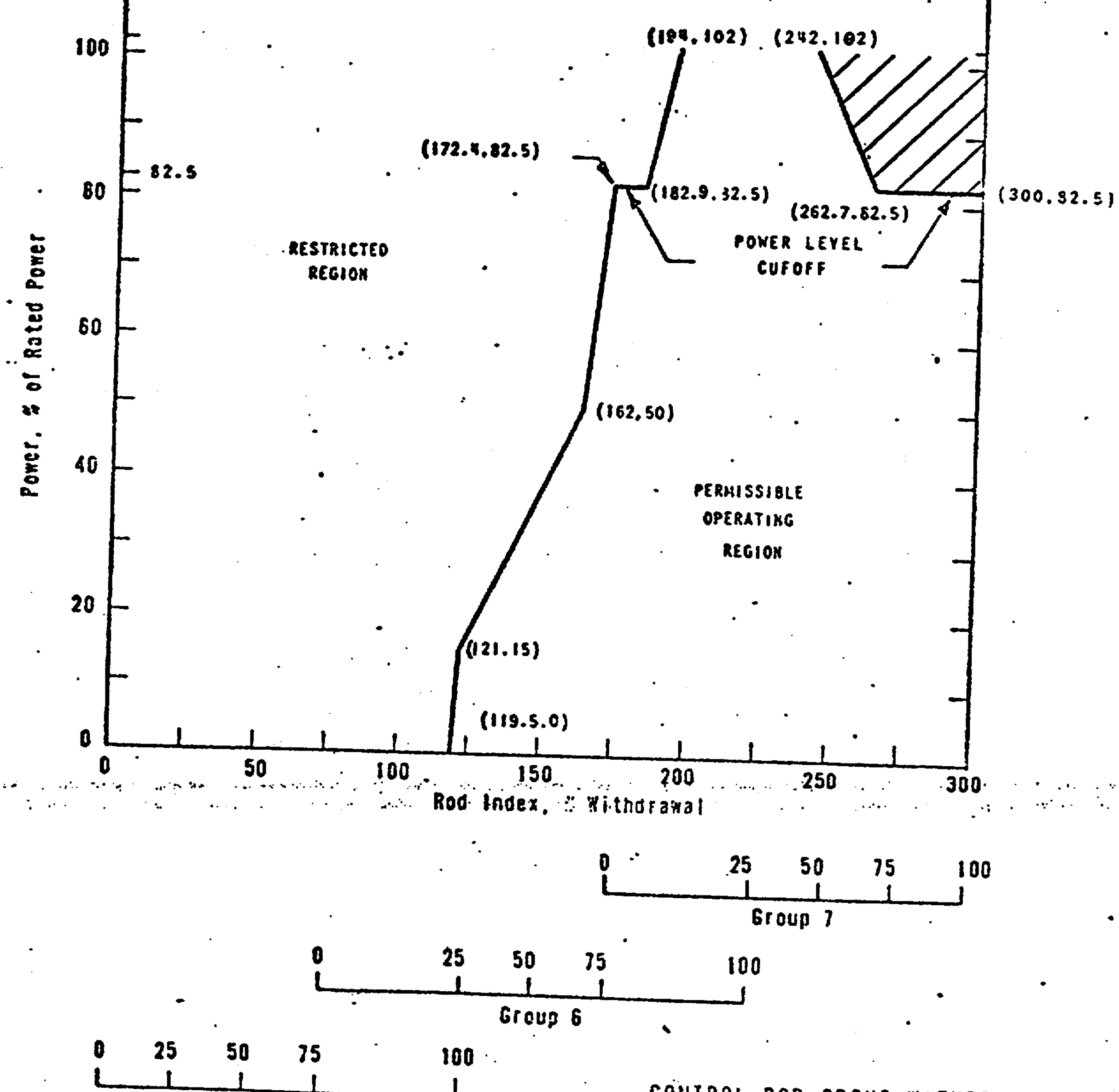
1. Rod index is the percentage sum of the withdrawal of the operating groups.

2. The additional restrictions on withdrawal (hashed areas) are in effect after the control rod interchange. The restrictions on withdrawal are further modified after 440 full power days of operation (See Figure 3.5-2C



CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION UNIT I

Figure 3.5-2B

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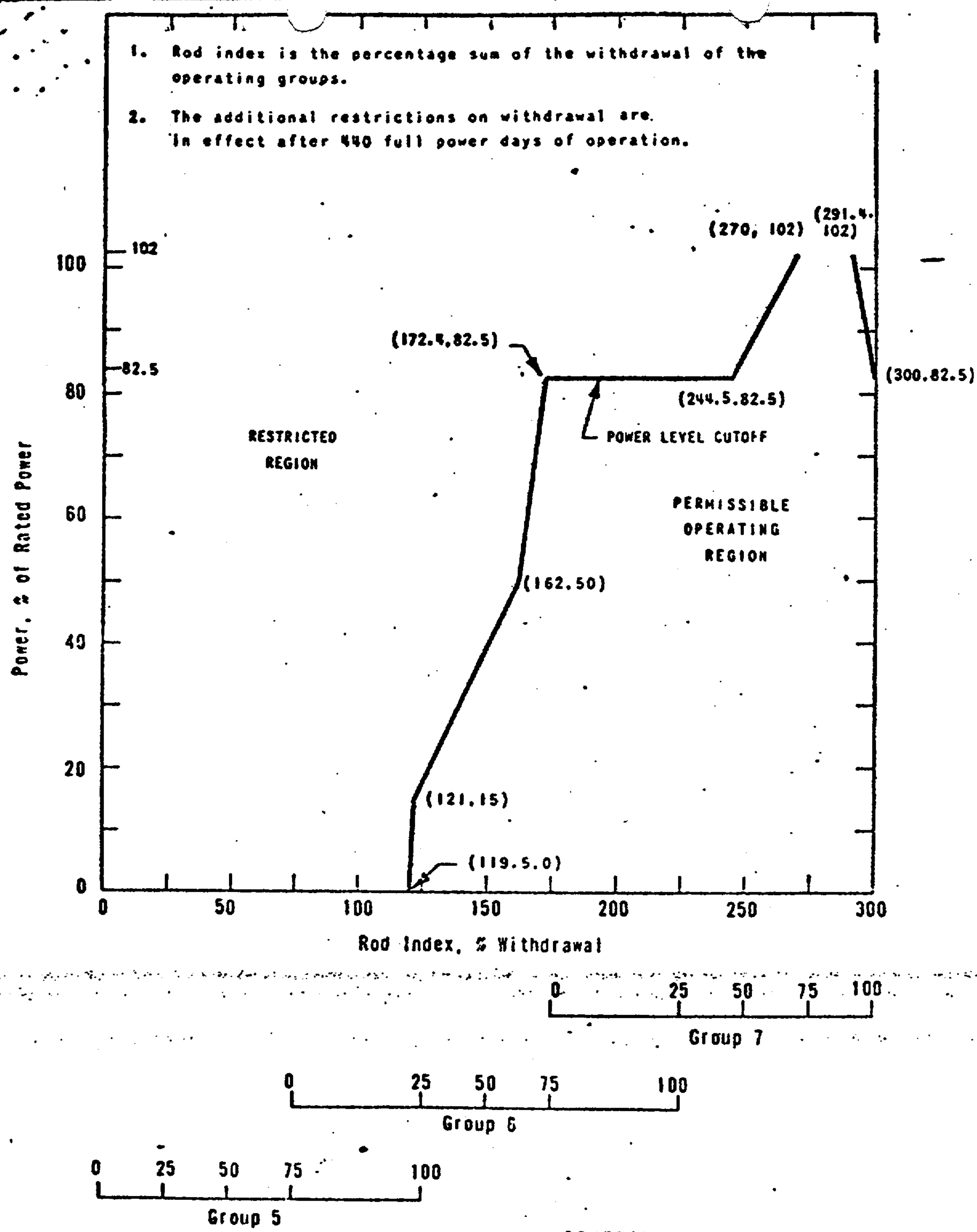
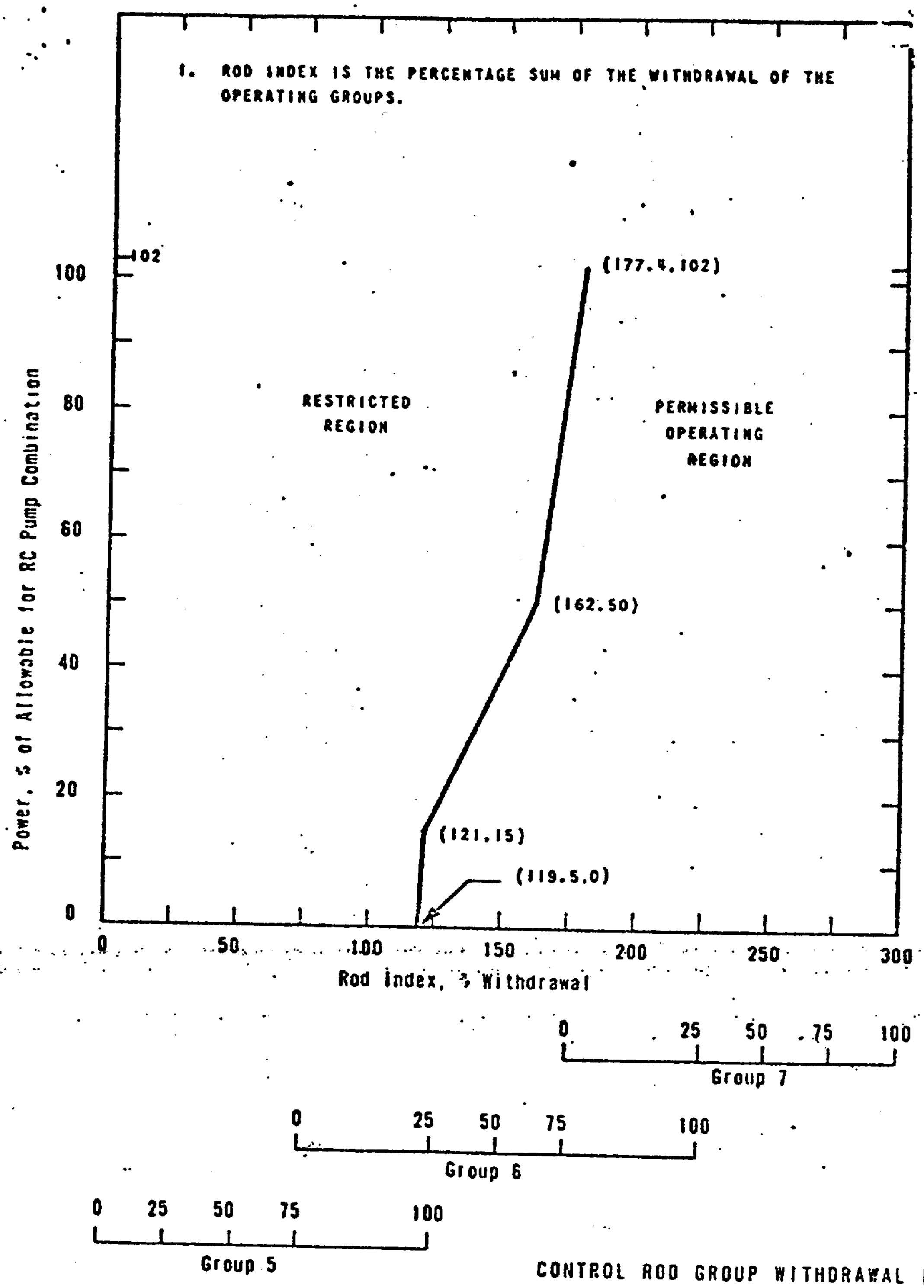


CONTROL ROD GROUP WITHDRAWAL LIMITS
FOR 4 PUMP OPERATION UNIT 1

Figure 3.5-2C



CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 3 AND 2 PUMP OPERATION UNIT I

Figure 3.5-2D

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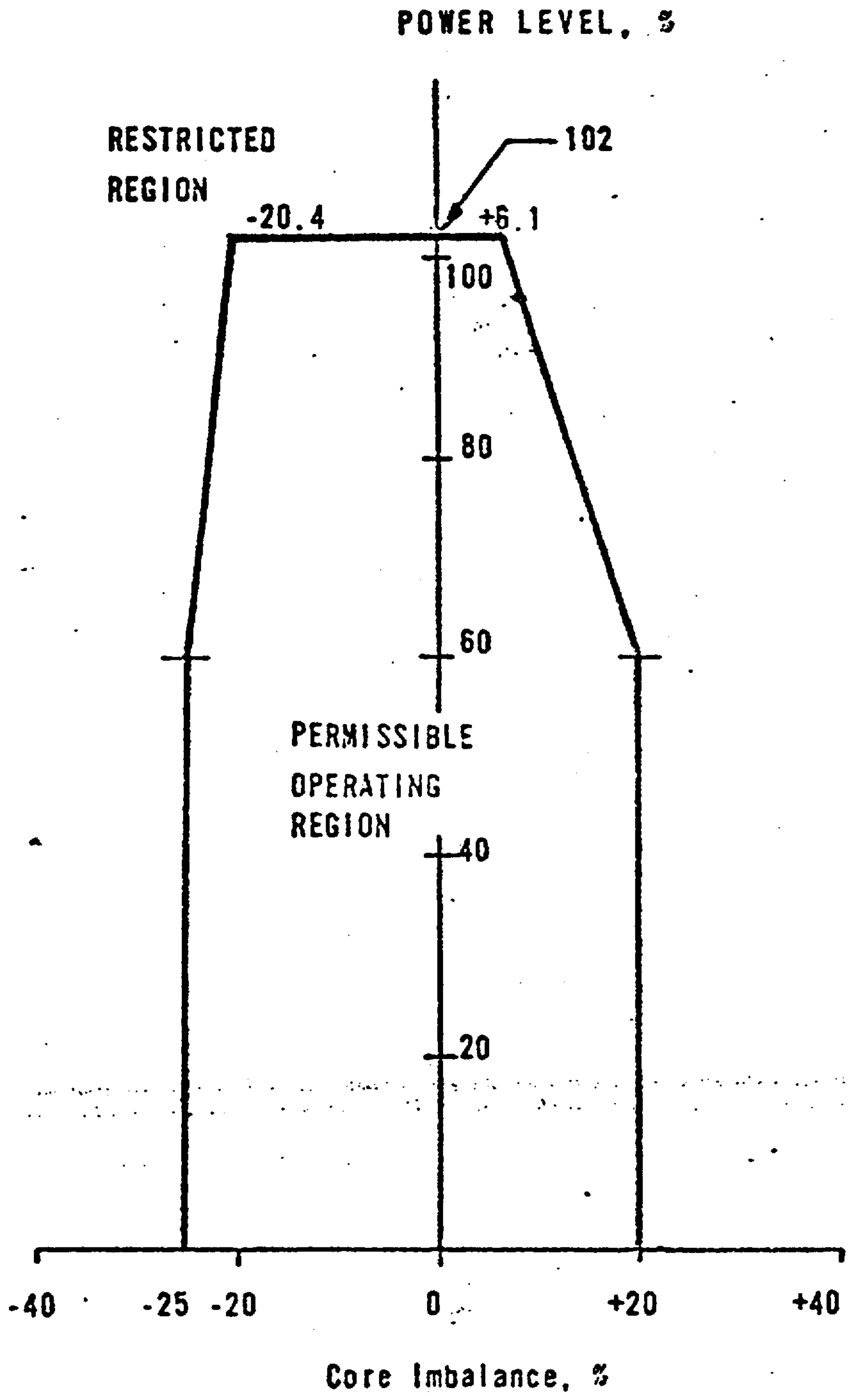
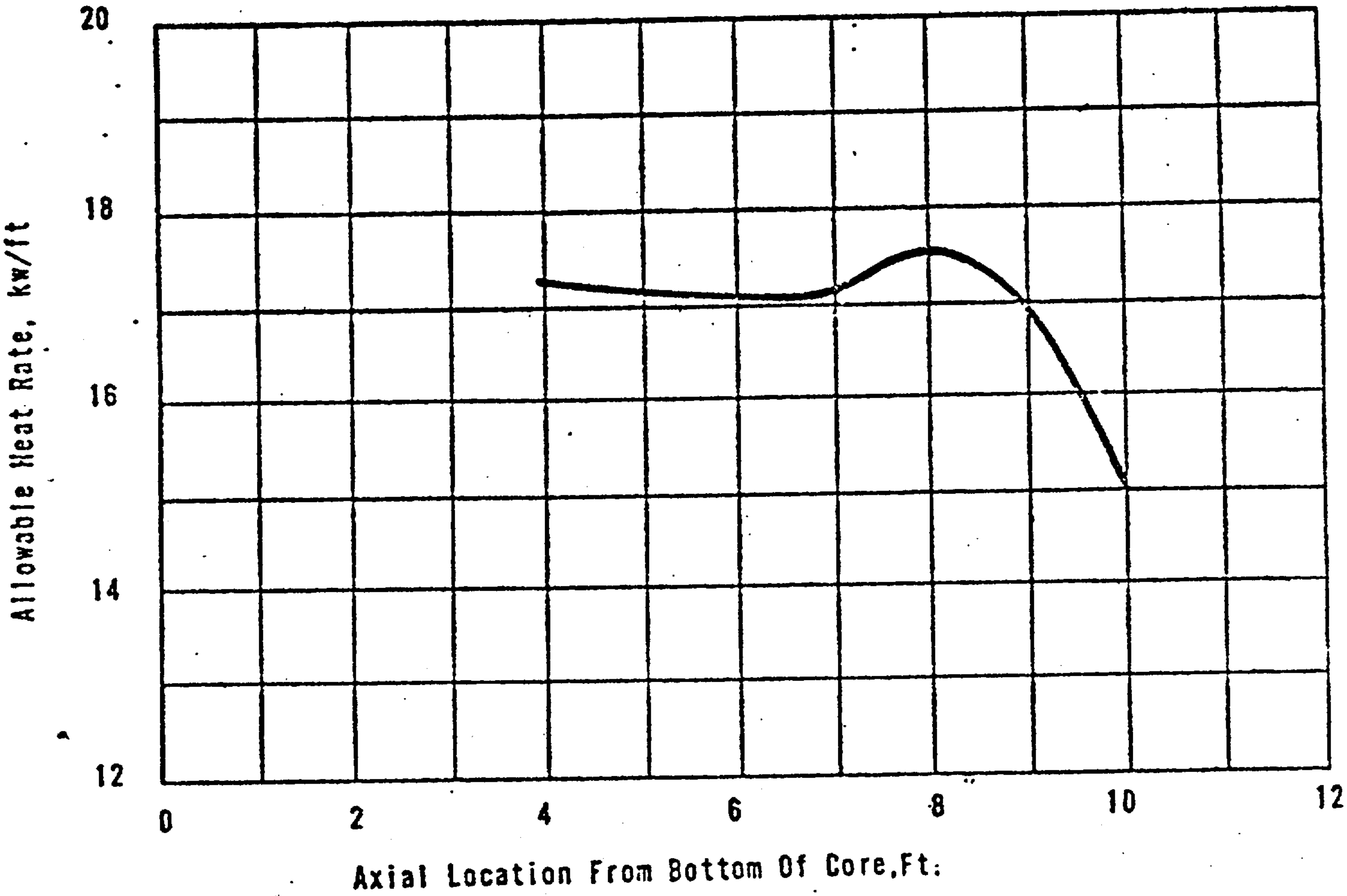


OPERATIONAL POWER IMBALANCE ENVELOPE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2 E



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5–2F