Florida Power
CORPORATION

June 25, 1979
File: 3-0-3-a-3

Director
Office of Nuclear Reactor Regulation
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 39

Dear Sir:

On May 25, 1979, Florida Power Corporation submitted revisions to the Cycle 2 reload Report which was filed on February 28, 1979 because FPC decided to license Crystal River Unit #3 at its present licensed power level of 2452 MWt instead of the Cycle 2 proposed power level of 2544 MWt.

This letter is to advise you that the Reactor Coolant Power Pump Monitors will not be installed prior to the initial Cycle 2 startup. As a result of this, the Technical Specifications submitted in support of Cycle 2 licensing, as amended on May 25, 1979, have been reviewed. The revisions to the proposed Cycle 2 Technical Specification, as submitted in TSCRN39 (February 28, 1979), have been revised, deleting reference to the Reactor Coolant Power Pump Monitors and those specifications resulting from their use, deleting the power upgrade level of 2544 MWt, and including the lowered RCS Pressure High setpoint as required by IE Bulletin 79-05B. The proposed Technical Specification pages attached are the actual changes from our present Technical Specifications as a result of the work actually performed during the refueling outage.

If you or members of your staff require any further discussion of this submittal, please contact us as soon as possible.

Sincerely,

FLORIDA POWER CORPORATION

W. P. Stewart

W. P. Stewart
Manager, Nuclear Operations

325 045

TSCRN39(WPShewR02)D27
Attachment

7907100393

STATE OF FLORIDA

COUNTY OF PINELLAS

W. P. Stewart states that he is the Manager, Nuclear Operations, of Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

W. P. Stewart

W. P. Stewart

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 25th day of June, 1979

Sarah L. Crockett

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: August 24, 1979

(CrockettNotary 2 D12)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF FLORIDA POWER CORPORATION	)))	DOCKET NO. 50-302

CERTIFICATE OF SERVICE

W. P. Stewart deposes and says that the following has been served on the Chief Executive of Citrus County, Florida by deposit in the United States mail, addressed as follows:

Chairman, Board of County
Commissioners of Citrus County
Citrus County Courthouse
Inverness, Florida 32650

An original copy of revisions to Technical Specification Change Request No. 39 requesting amendment to Appendix A of Operating License No. DPR-72.

FLORIDA POWER CORPORATION

W.P. Stewart
W. P. Stewart
Manager, Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME THIS 25th DAY OF JUNE, 1979.

Sarah F. Crockett
Notary Public

Notary Public State of Florida at Large
My Commission expires: August 24, 1979

(NOTARIAL SEAL)

(Cert.Serv.D12)

325 047

Revisions to Technical Specification Change Request No. 39 for (1) 2452 MWt versus 2544 MWt; (2) deletion of the Reactor Coolant Power Pump Monitors; and (3) Item 5 of IE Bulletin 79-05B.

Page 1-1: In Definition 1.3, RATED THERMAL POWER should be defined as 2452 MWt (as it is in the present Technical Specification) instead of 2544 MWt.

Page 2-2: In Figure 2.1-1, the RCS Pressure-High Trip should be redrawn as 2300 psig instead of 2355 psig.

Page 2-3: In Figure 2.1-2, the limit line for acceptable 3 & 4 pump operation should be lowered by 2.0% of RATED THERMAL POWER.

Page 2-5: In Table 2.2-1, in Item 2, the Trip Setpoint and Allowable Value with three pumps operating should be changed to $\leq$78% of RATED THERMAL POWER (as it is in the present Technical Specification) instead of $\leq$79.9%.

In Table 2.2-1, in Item 6, the Trip Setpoint and Allowable Value should be changed to $\leq$2300 psig instead of $\leq$2355 psig.

Page 2-6: In Table 2.2-1, delete Item 8 referring to the Reactor Coolant Pump Power Monitors and revise Item 9 to Item 8 (as it is in the present Technical Specifications).

Page 2-7: In Figure 2.2-1, change the Trip Setpoint envelopes for acceptable 4 and acceptable 3 & 4 pump operation to the envelopes that are in the present Technical Specifications.

Page B2-5: In the Bases for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance, change Items 1 and 2 of the second paragraph to the Items 1 and 2 that are in the present Technical Specifications.

Page B2-6: In the first paragraph the flux-to-flow ratio should be changed to 1.043% (as it is in the present Technical Specifications) instead of 1.07%.

In the Bases for RCS Pressure-Low, High and Variable Low, in the second paragraph, change the trip setpoint for RCS Pressure-High to 2300 psig instead of 2355 psig.

Page B2-8: For Curves 1 and 2, change the Power (% RTP) to 117.3% and 90.5%, respectively, instead of 113.1% and 87.2%, respectively.

Page 3/4 1-27: In Figure 3.1-1, change the Power Level Cutoff to 92% of Rated Thermal Power instead of 90%.

Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 1-28: In Figure 3.1-2, change the Power Level Cutoff to 92% of Rated Thermal Power instead of 90%.

Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 1-29: Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 1-30: Change the length of mid-Cyucle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 1-38: Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 1-39: Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 2-2: Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

Page 3/4 2-3: In Figure 3.2-2 the point defined as (18, 92) should be changed to (18, 90).

Change the length of mid-Cycle 2 to 233±10 EFPD instead of 225±10 EFPD.

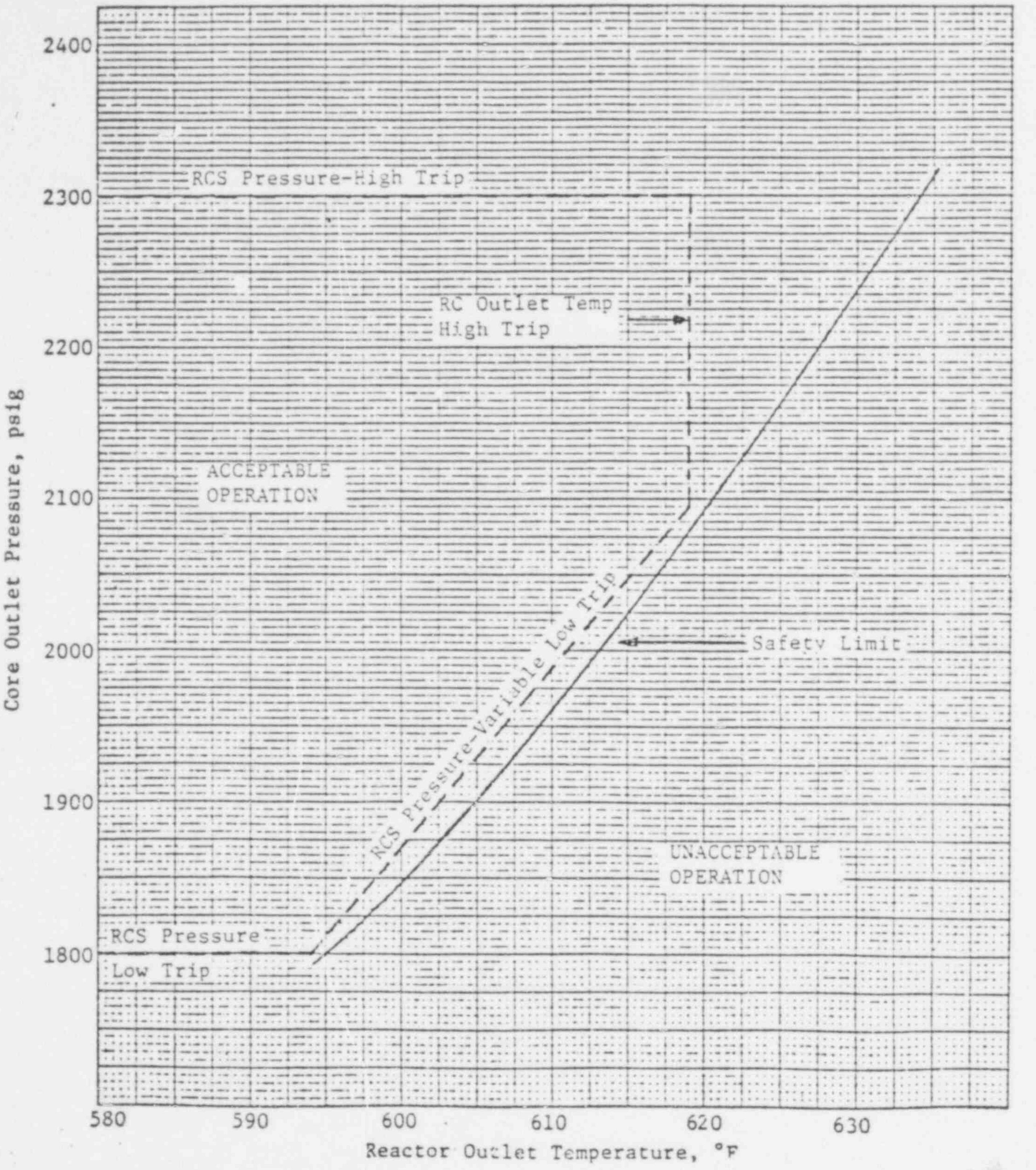
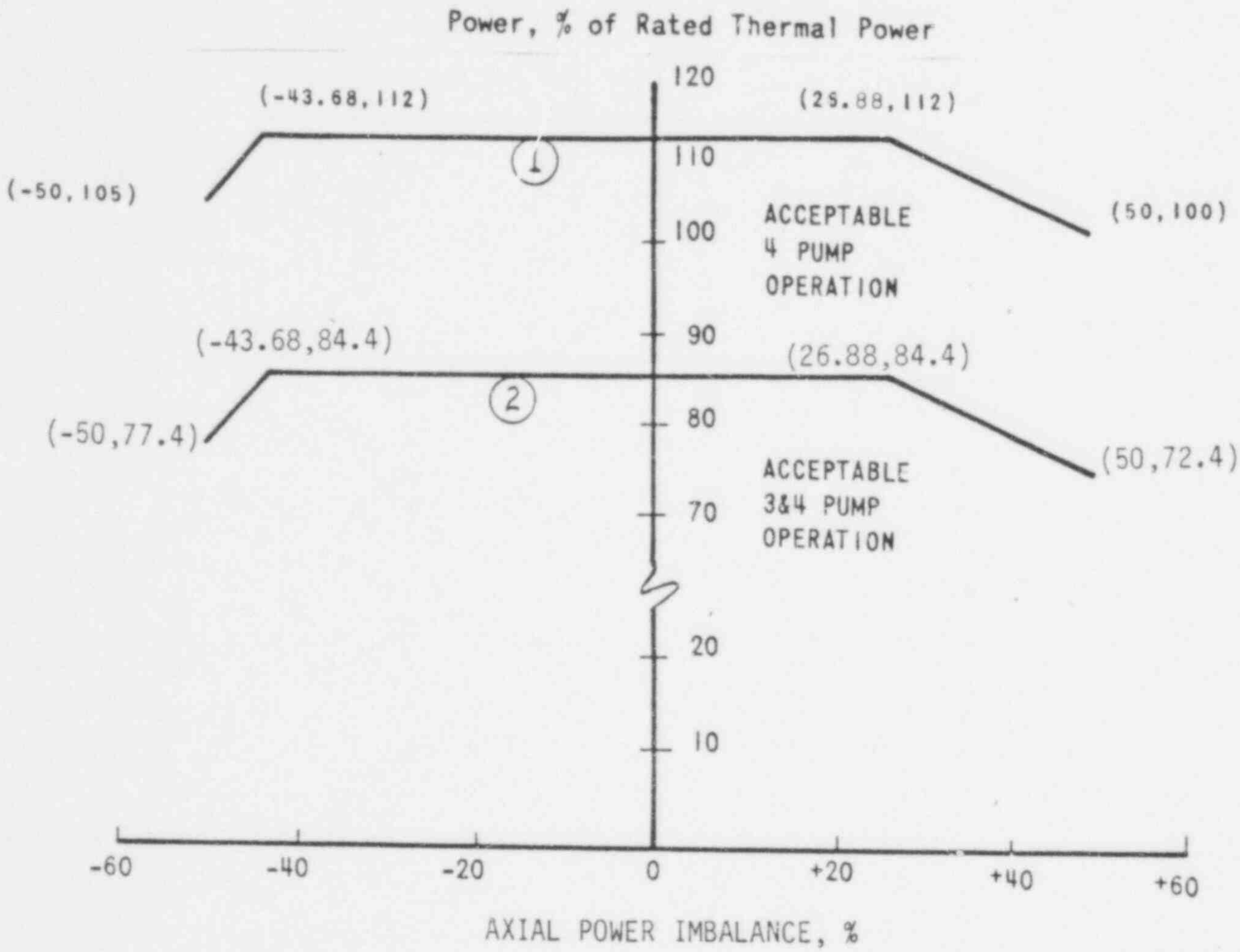


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.86×10^6
2	104.47×10^6

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT

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TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$105.5% of RATED THERMAL POWER with four pumps operating	$\leq$105.5% of RATED THERMAL POWER with four pumps operating
	$\leq$78% of RATED THERMAL POWER with three pumps operating	$\leq$78% of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature-High	$\leq$619°F	$\leq$619°F
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1	Allowable Values not to exceed the limit line of Figure 2.2-1
5. RCS Pressure-Low(1)	$\geq$1800 psig	$\geq$1800 psig
6. RCS Pressure-High	$\leq$2300 psig	$\leq$2300 psig
7. RCS Presssure-Variable Low(1)	$\geq$(11.80T_{out} °F -5209.2) psig	$\geq$(11.80 T_{out} °F -5209.2) psig

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2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling rgime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the BAW-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local 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. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power, 112%, when the reactor coolant flow is 139.86×10^6 lbs/hr, which is 106.5% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F^N_Q = 2.57; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = 1.50$$

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

The AXIAL 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 reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.043% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurized code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Presssure-Variable Low, (11.80 T_{out}°F-5209.2) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of (11.80 T_{out}°F-5249.2) psig.

Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint ≤4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containmer vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure-Low trip.

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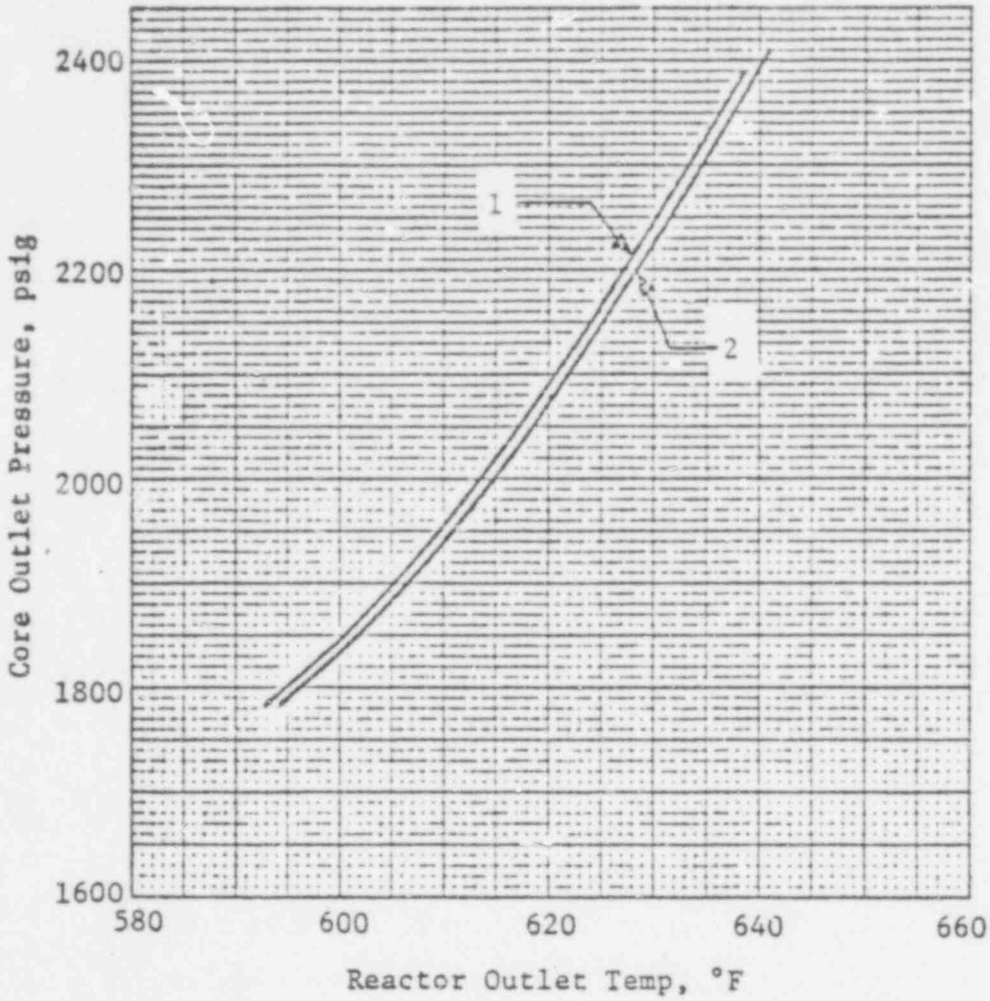

POOR ORIGINAL

REACTOR COOLANT FLOW

CURVE	FLOW (lb/hr)	POWER(%RTP)	PUMPS OPERATING (TYPE OF LIMIT)
1	139.86×10^6 (106.7%)	117.3%	4 Pumps (DNBR)
2	104.47×10^6 (79.7%)	90.5%	3 Pumps (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

325 055

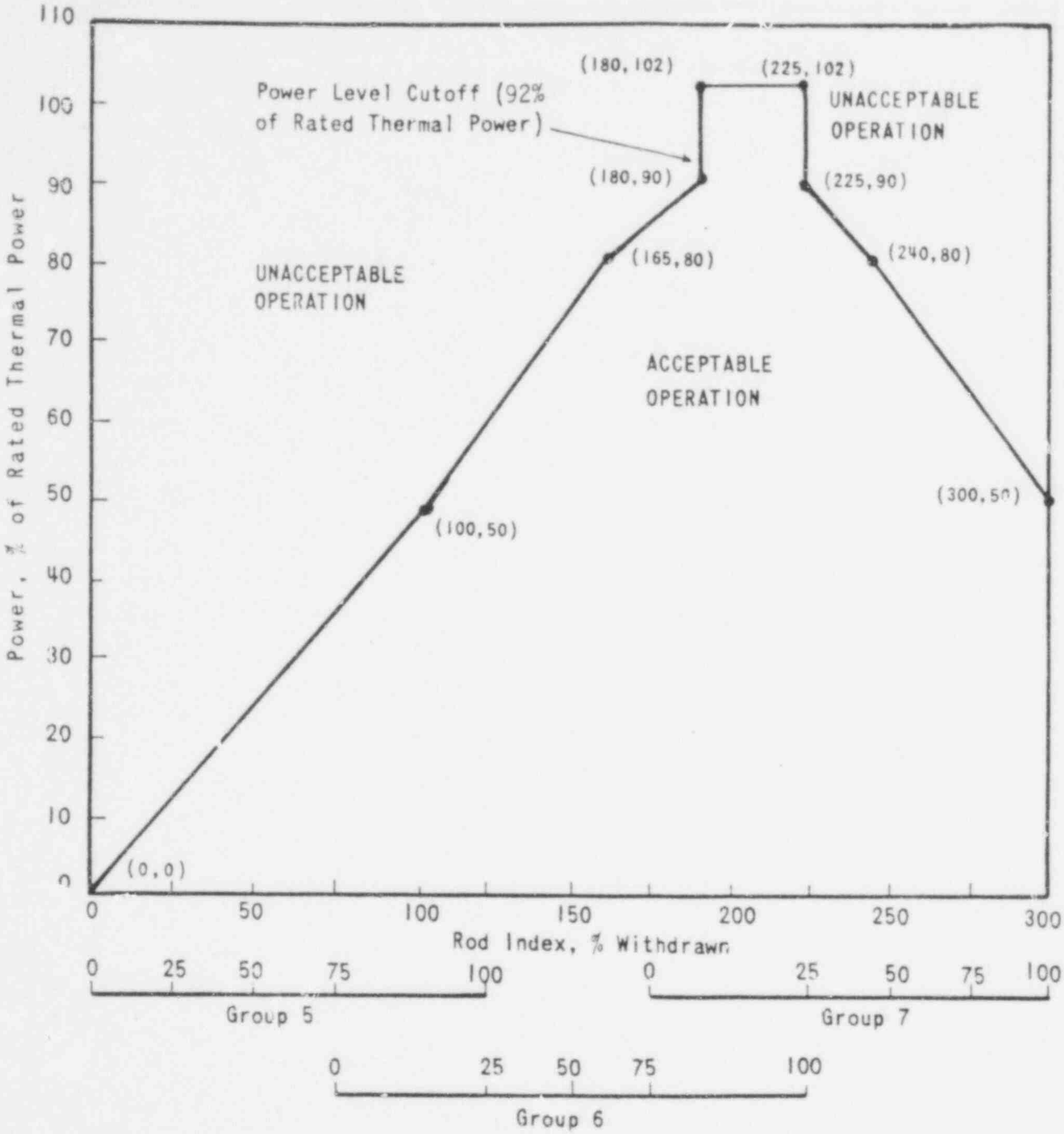

FIGURE 3.1-1

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION FROM 0 EFPD TO 233 $\pm$ 10 EFPD

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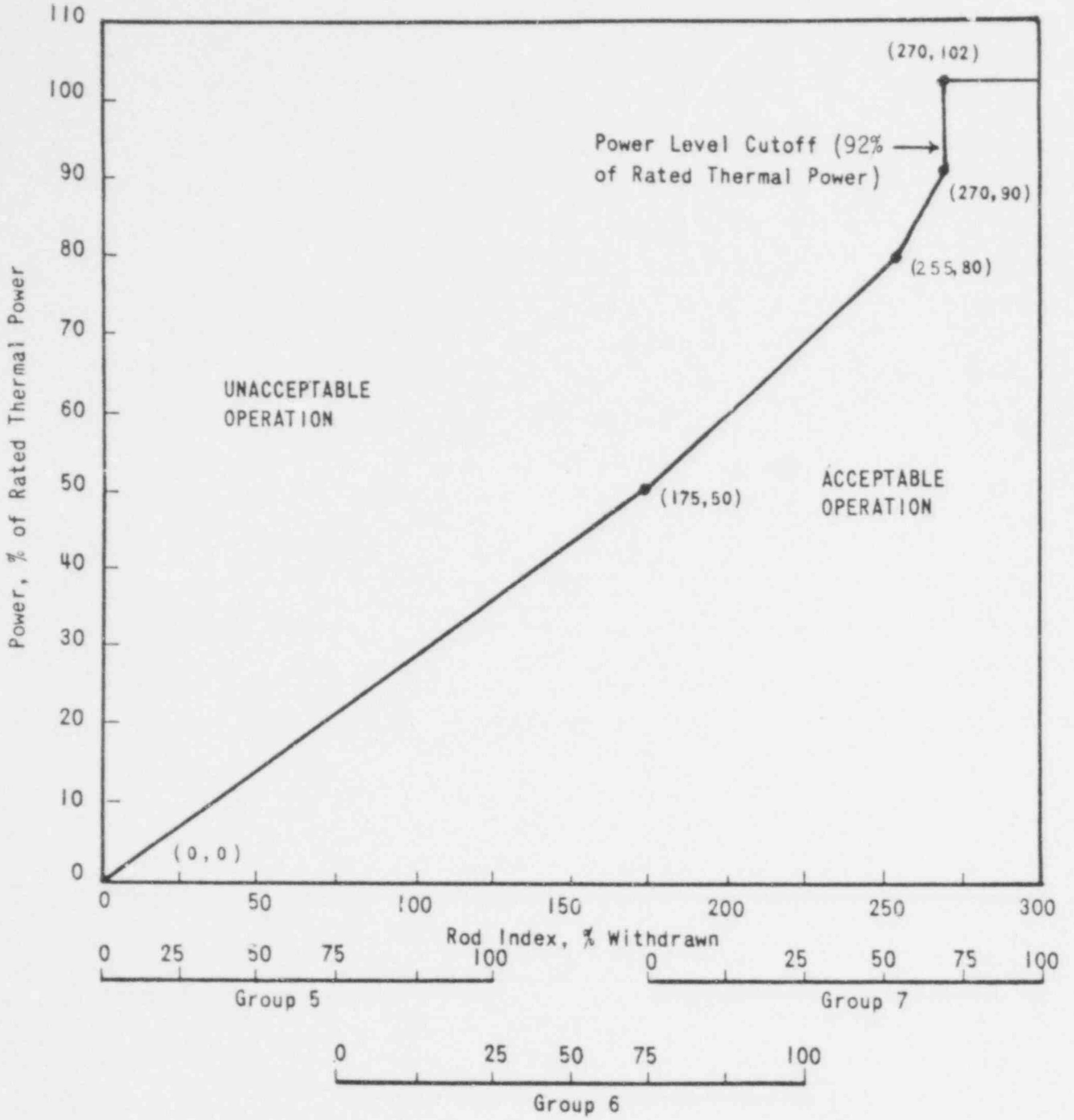

FIGURE 3.1-2

REGULATING ROD GRO INSERTION LIMITS FOR
4 PUMP OPERATION ... TER 233 ± 10 EFPD

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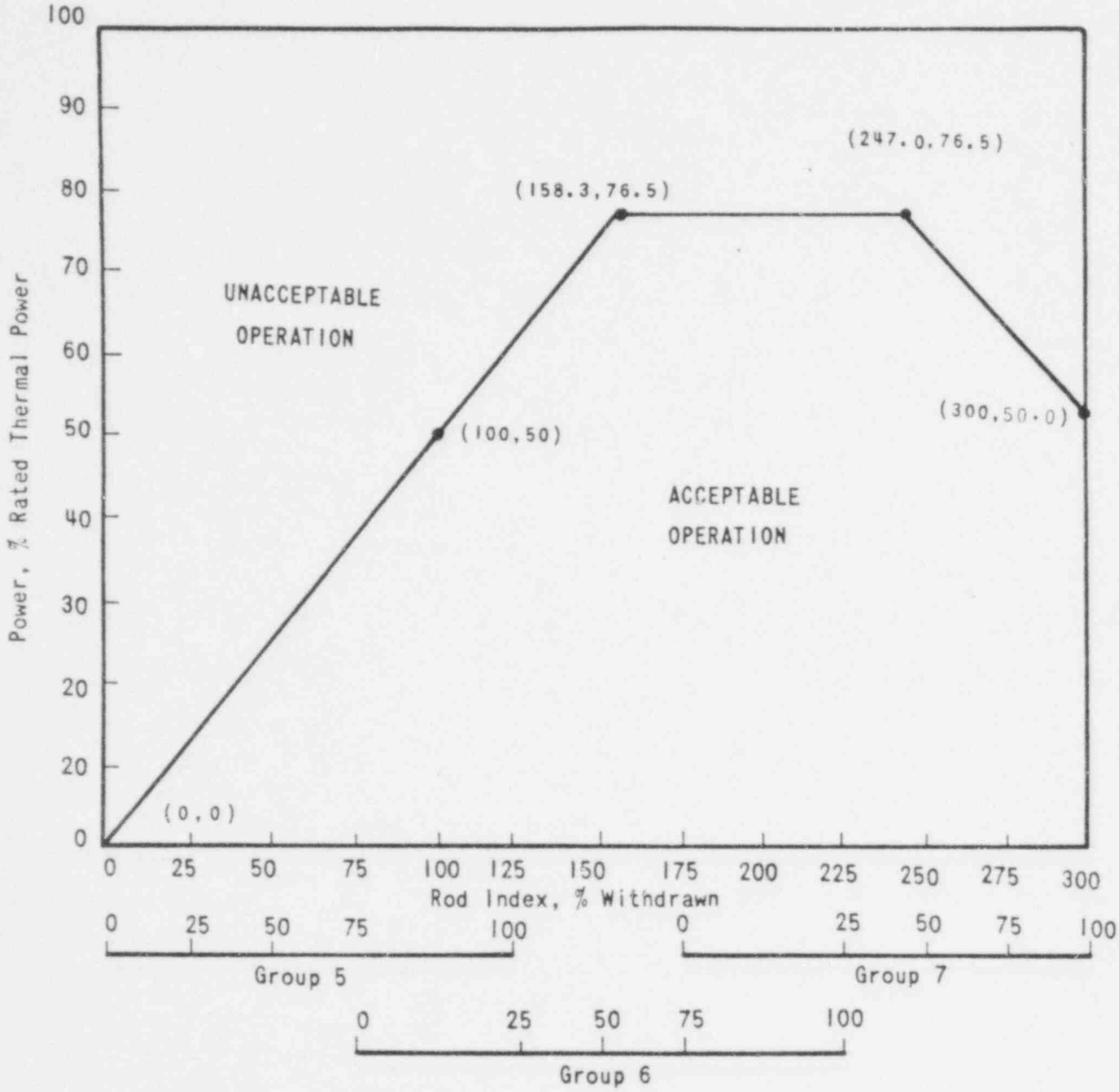

FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION FROM 0 EFPD TO 233 ± 10 EFPD

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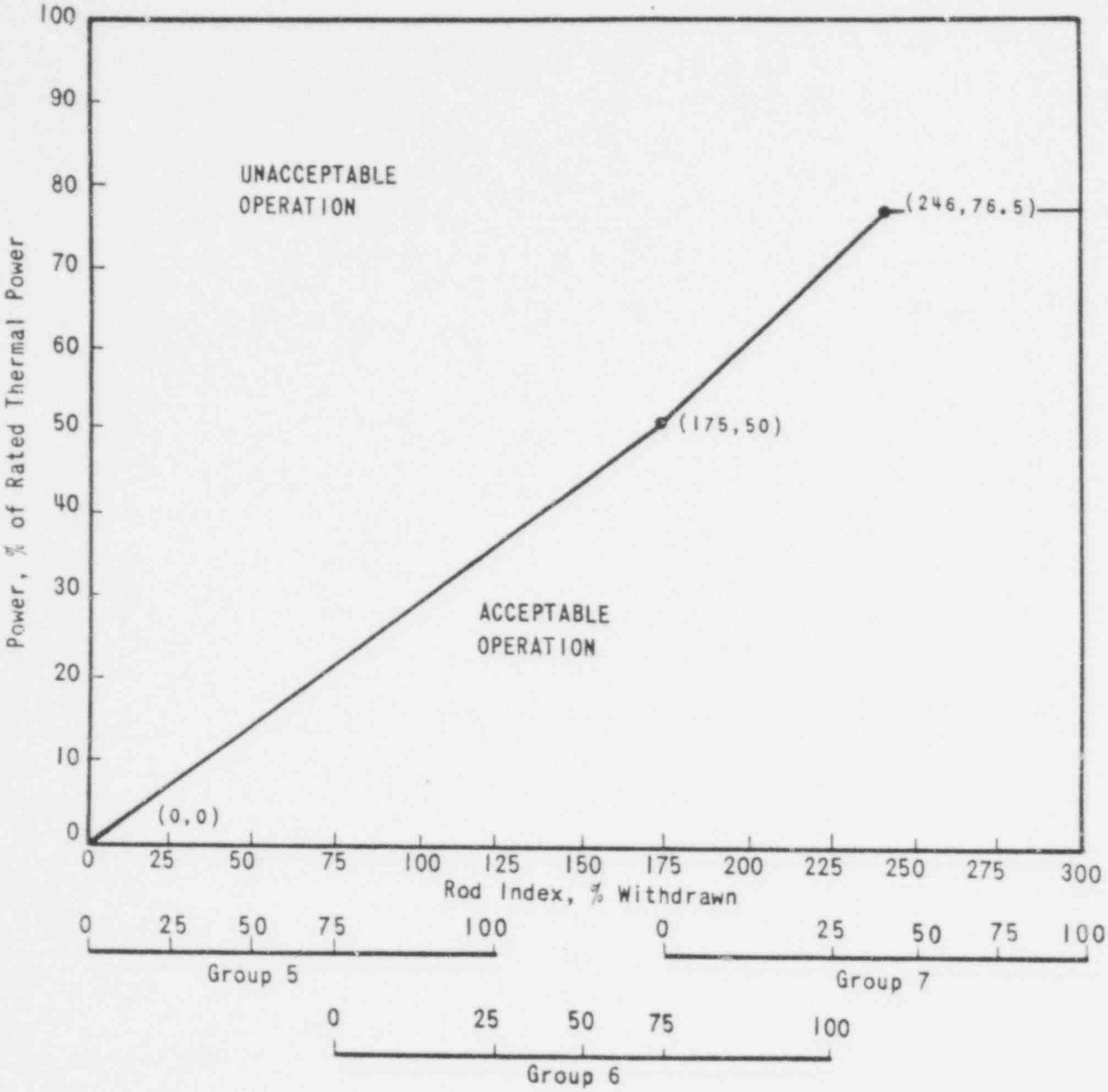

FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR
3 PUMP OPERATION AFTER 233 ± 10 EFPD

325 059

ROD PROGRAM

LIMITING CONDITION FOR OPERATION

3.1.3.7 Each control rod (safety, regulating and APSR) shall be programmed to operate in the core position and rod group specified in Figure 3.1-7.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With any control rod not programmed to operate as specified above, be in HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.3.7

a. Each control rod shall be demonstrated to be programmed to operate in the specified core position and rod group by:

   1. Selection and actuation from the control room and verification of movement of the proper rod as indicated by both the absolute and relative position indicators:

      a) For all control rods, after the control rod drive patches are locked subsequent to test, reprogramming or maintenance within the panels.

      b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

   2. Verifying that each cable that has been disconnected has been properly matched and reconnected to the specified control rod drive.

b. At least once each 7 days, verify that the control rod drive patch panels are locked.

*See Special Test Exceptions 3.10.1 and 3.10.2.

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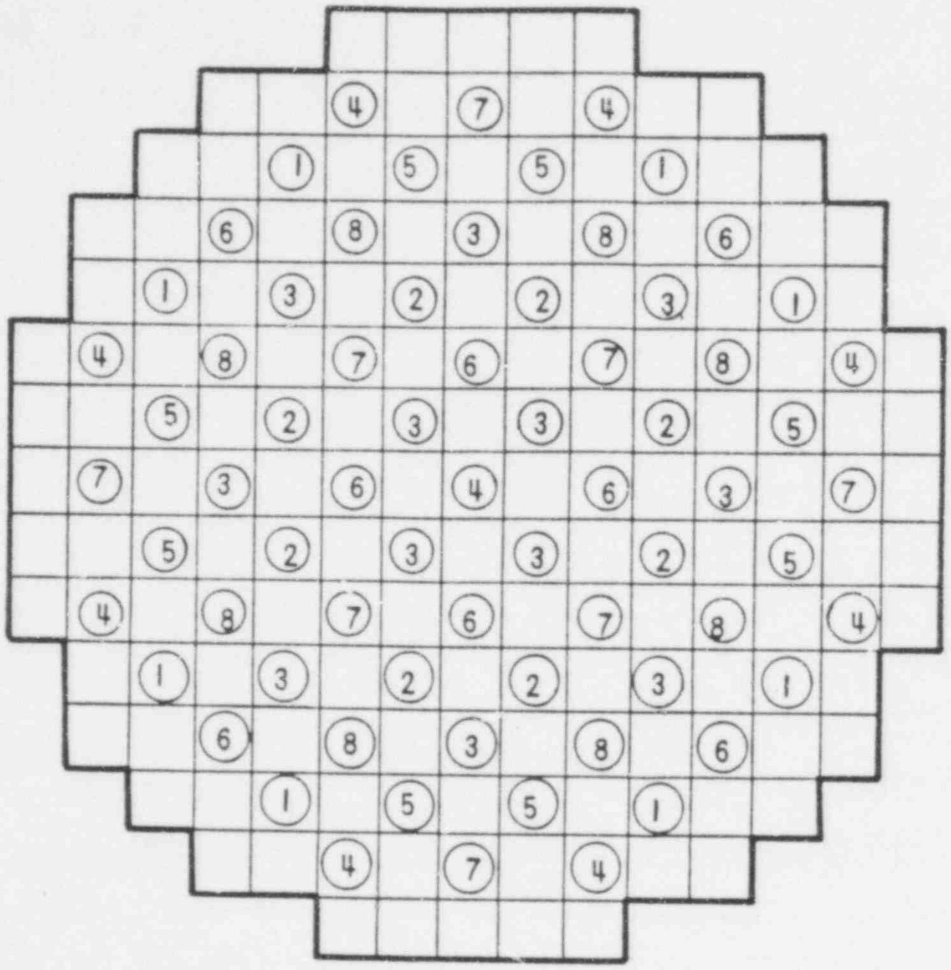

GROUP	NUMBER OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	12	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
TOTAL	69	

FIGURE 3.1-7

CONTROL ROD LOCATIONS AND GROUP ASSIGNMENTS

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DELETED

325 062

XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in Figures 3.1-1 and 3.1-2 unless xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability.

APPLICABILITY: MODE 1.

ACTION:

With the requirements of the above specification not satisfied, reduce THERMAL POWER to less than or equal to the power level cutoff within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.3.8 Xenon reactivity shall be determined to be within 10% of the equilibrium value for RATED THERMAL POWER and to be approaching stability prior to increasing THERMAL POWER above the power level cutoff.

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AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 The axial power shaping rod group shall be limited in physical insertion as shown on Figures 3.1-9 and 3.1-10.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the axial power shaping rod group outside the above insertion limits, either:

a. Restore the axial power shaping rod group to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figure within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.9 The position of the axial power shaping rod group shall be determined to be within the insertion limits at least once every 12 hours.

*With $k_{eff} \geq 1.0$.

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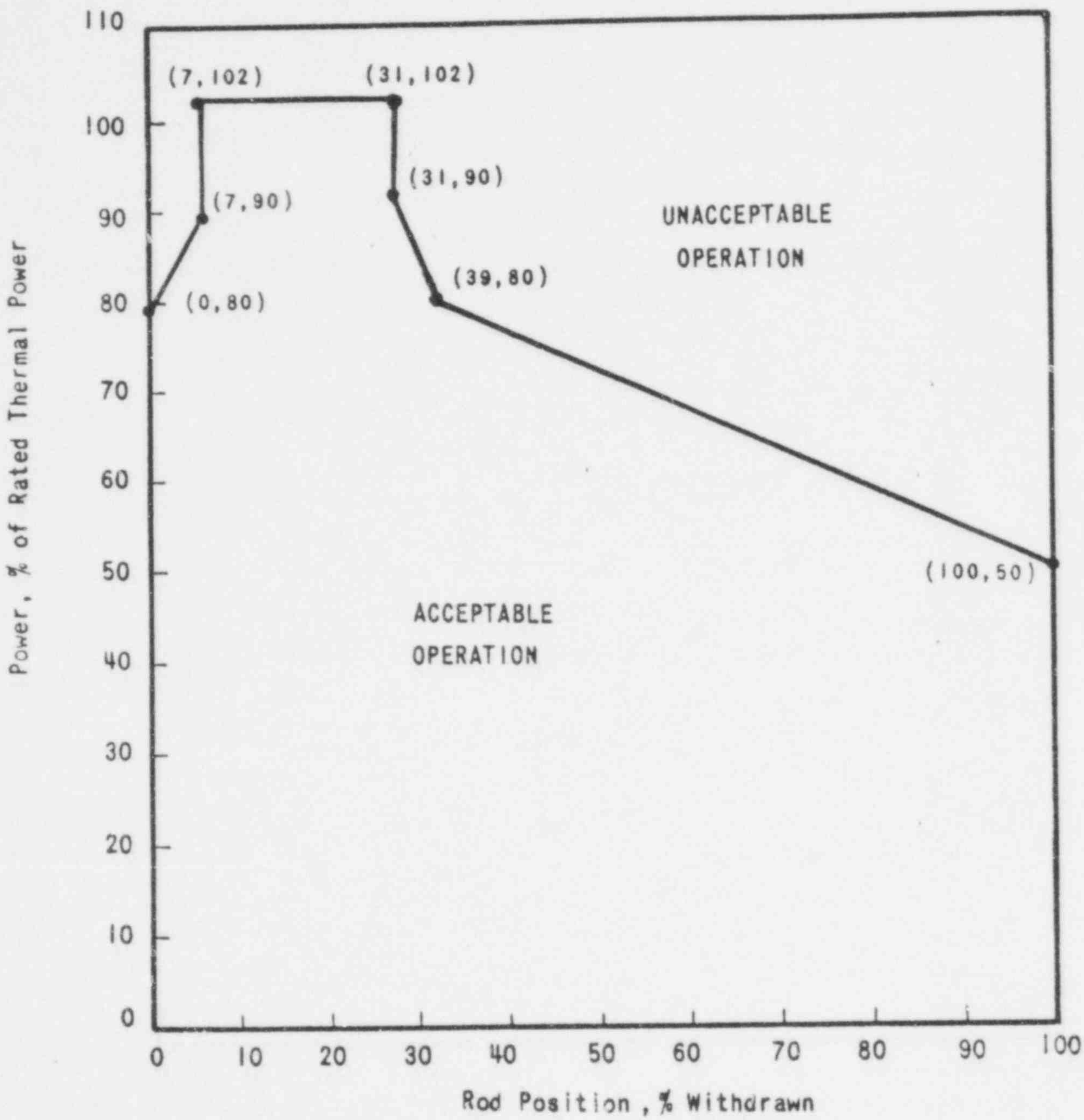

FIGURE 3.1-9

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS
FROM 0 EFPD TO 233 ± 10 EFPD

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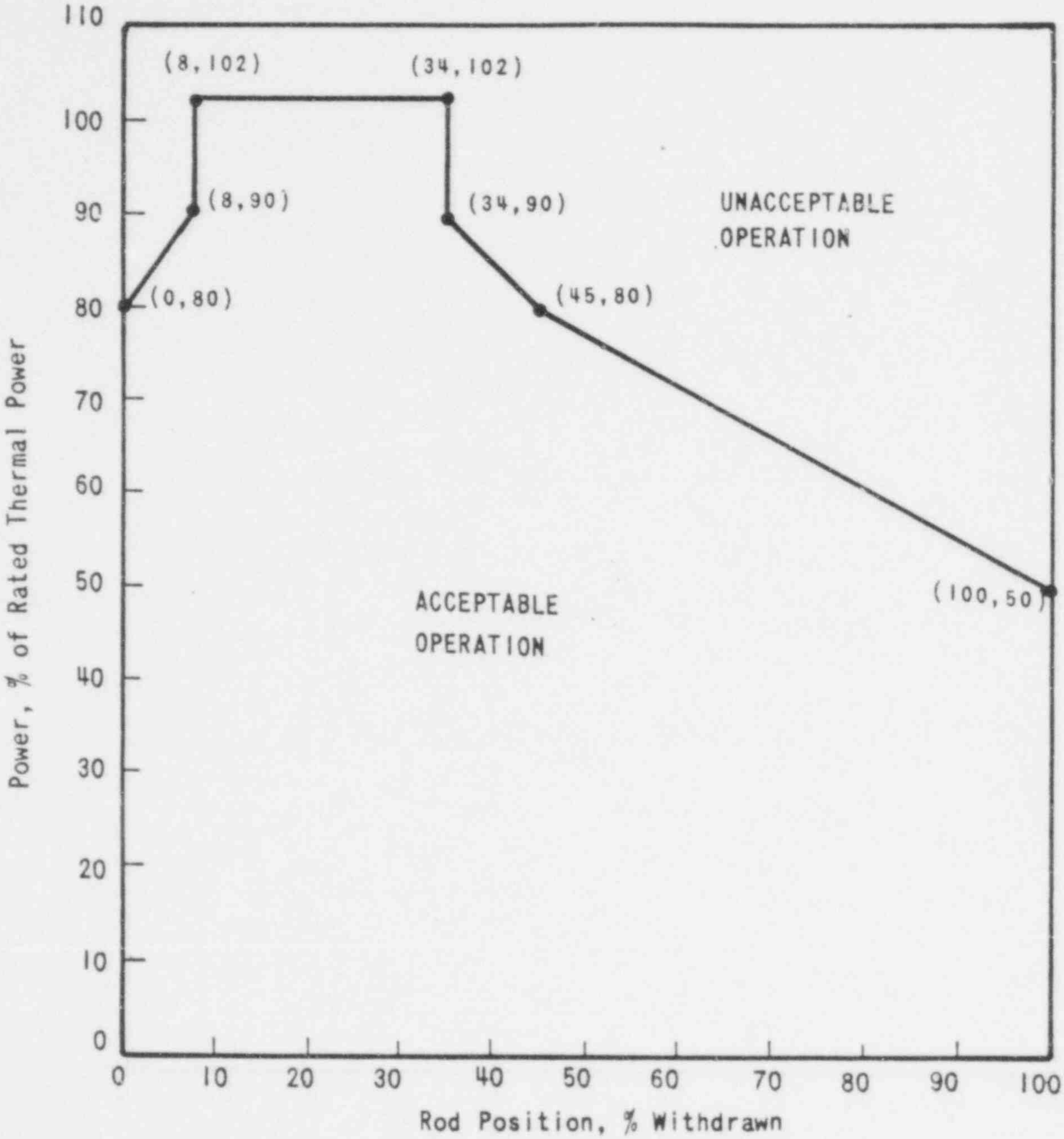

FIGURE 3.1-10

AXIAL POWER SHAPING ROD GROUP
INSERTION LIMITS AFTER 233 ± 10 EFPD

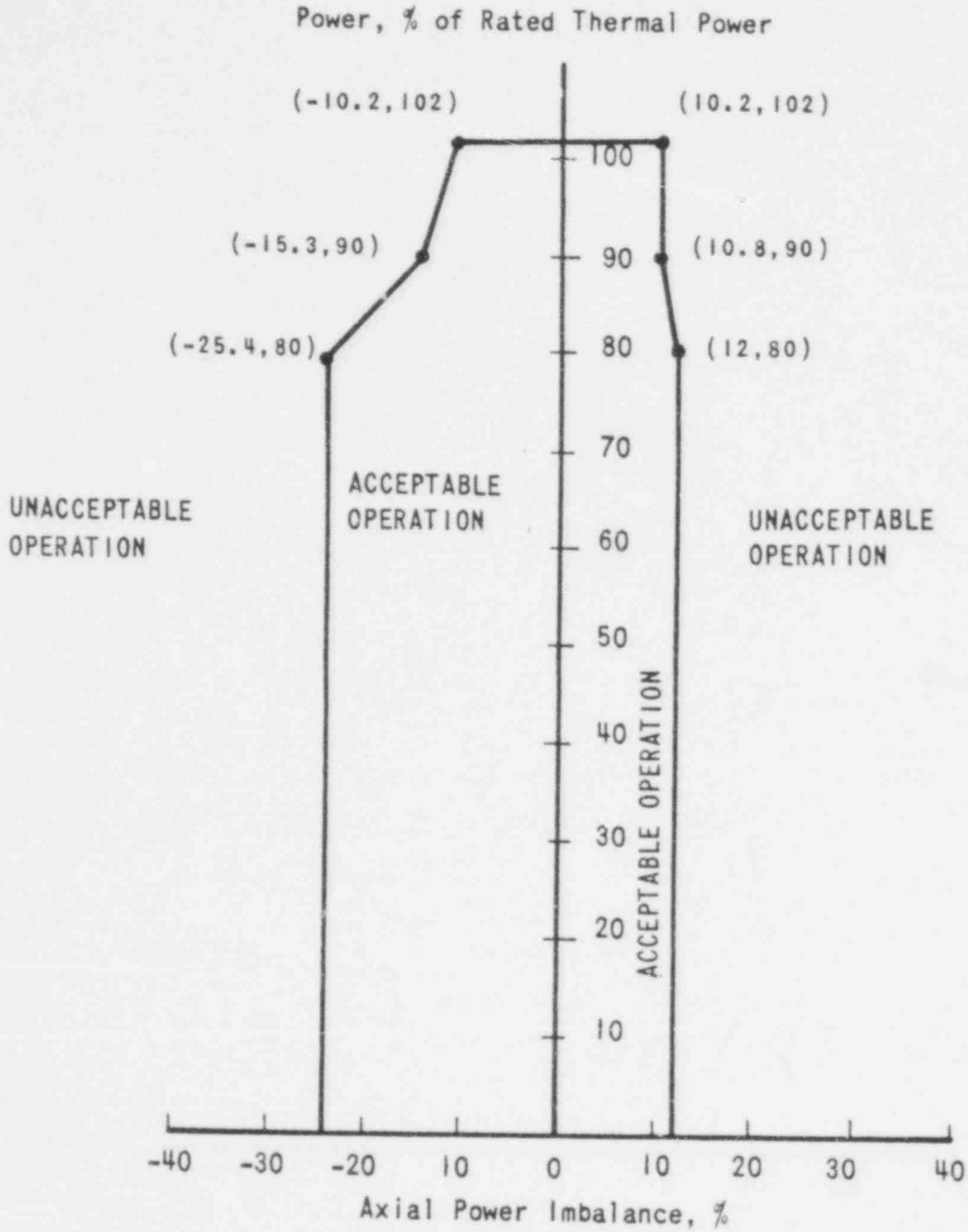

FIGURE 3.2-1

AXIAL POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 0 EFPD TC 233 $\pm$ 10 EFPD

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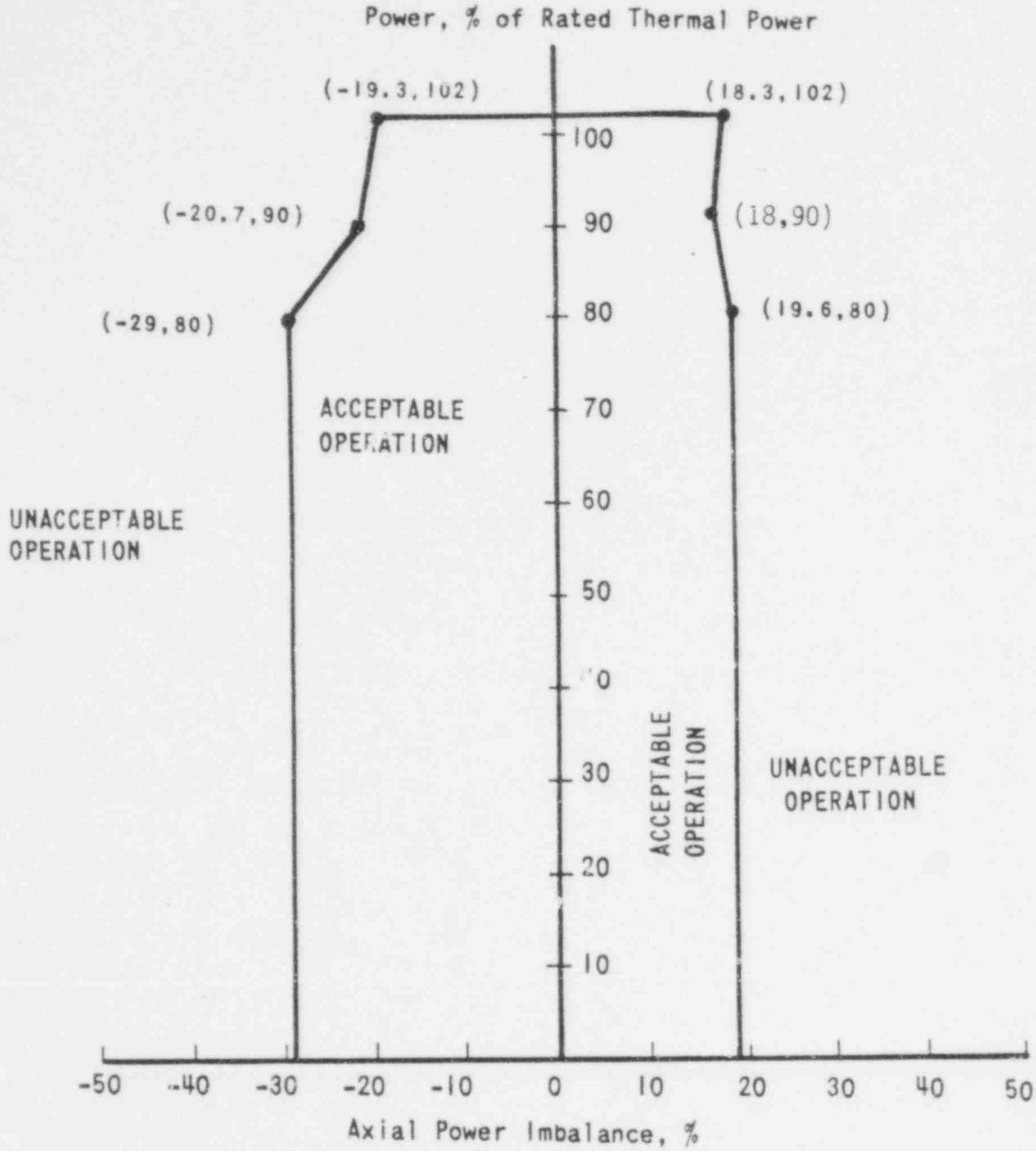

FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR
OPERATION AFTER 233 ± 10 EFPD

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NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{3.08}{P}$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ and $P \leq 1.0$.

APPLICABILITY: MODE 1.

ACTION:

With F_Q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the Nuclear Overpower Trip Setpoint and Nuclear Overpower based on RCS Flow and AXIAL POWER IMBALANCE Trip Setpoint within 4 hours.

b. Demonstrate through in core mapping that F_Q is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that F_Q is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_Q shall be determined to be within its limit by using the in-core detectors to obtain a power distribution map:

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TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	3.46	8.96	20.0
Power Range Channels	1.96	6.96	20.0
Minimum Incore Detector System	1.90	4.40	20.0

TABLE 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature, T_H°F	$\leq$ 604.6	$\leq$ 604.6(1)
Reactor Coolant Pressure, psig.(2)	$\geq$ 2061.6	$\geq$ 2057.2(1)
Reactor Coolant Flow Rate, lb/hr	$\geq 139.86\times10^6$	$\geq 104.47\times10^6$

(1)Appplicable to the loop with 2 Reactor Coolant Pumps Operating.

(2)Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase greater than 10% of RATED THERMAL POWER.

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$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 3.08; \quad F^N_{\Delta H} \leq 1.71$$

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than ± 6.5% (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 and the axial power shaping rod insertion limits of Specification 3.1.3.9 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained within the limits of Figures 3.2-1 and 3.2-2.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

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