Florida Power
CORPORATION

W. P. STEWART, DIRECTOR
POWER PRODUCTION

March 15, 1979

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Dear Sir:

Enclosed are three (3) originals and forty (40) copies of Technical Specification Change Request No. 39 requesting amendment to Appendix A of Operating License No. DPR-72. As part of this request, proposed replacement pages for Appendix A are enclosed.

Technical Specification Change Request No. 39 addresses the Cycle 2 operation of Crystal River Unit 3 at a rated core power level of 2544 MWt which corresponds to the ultimate core power level identified in the CR #3 FSAR. This change request is supported by the Babcock & Wilcox report BAW-1521, Crystal River Unit 3 Cycle 2 Reload Report and the report Environmental Impact Appraisal and Balance of Plant Review for Cycle 2 Reload and Power Level Upgrade, which were submitted to the Commission on February 28, 1979.

Technical Specification Change Request No. 27, submitted on November 27, 1978, concerning Reactor Coolant Pump Power Monitors, our submittal of the above referenced reports on Febrary 28, 1979 and the enclosed Change Request No. 39 are all related to the Cycle 2 operation of CR #3 at 2544 MWt and therefore, the licensing fees submitted on November 29, 1978 and February 28, 1979 should be applied to Change Request No. 39.

7903206446

Also enclosed is one signed copy of Certificate of Service for Change Request No. 39 to the Chief Executive of Citrus County, Florida.

Very truly yours,

FLORIDA POWER CORPORATION

W. P. Stewart

W.P. Stewart

cc: Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Ga 30303

File: 3-0-3-a-3

ECSemhM05
D3

STATE OF FLORIDA

COUNTY OF PINELLAS

W.P. Stewart states that he is the Director, Power Production, 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 15th day of March, 1979.

Charles R Pope

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: July 25, 1980
(Notary 1 D12)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF	)	
	)	DOCKET NO. 50-302
FLORIDA POWER COR ,RATION	)	

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 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
Director, Power Production

SWORN TO AND SUBSCRIBED BEFORE ME THIS 15th DAY OF MARCH, 1979.

Charles L Pope
Notary Public

Notary Public State of Florida at Large
My Commission expires: July 25, 1980

(NOTARIAL SEAL)

(Cert.Serv.D12)

Technical Specification Change Request No. 39 (Appendix A)

Delete and insert the pages in Appendix A of Operating License DPR-72 as indicated below:

DELETE	INSERT
1-1	1-1
2-2	2-2
2-3	2-3
2-5	2-5
2-6	2-6
2-7	2-7
B 2-1	B 2-1
B 2-5	B 2-5
B 2-6	B 2-6
B 2-8	B 2-8
3/4 1-27	3/4 1-27
3/4 1-28	3/4 1-28
3/4 1-29	3/4 1-29
3/4 1-30	3/4 1-30
3/4 1-33	3/4 1-33
3/4 1-34	3/4 1-34
3/4 1-35	3/4 1-35
3/4 1-36	3/4 1-36
3/4 1-37	3/4 1-37
3/4 1-38	3/4 1-38
3/4 1-39	3/4 1-39
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-11	3/4 2-11
3/4 2-13	3/4 2-13
B 3/4 2-2	B 3/4 2-2

Proposed Changes

These changes address the increase in Rated Thermal Power of Crystal River Unit 3 from 2452 MWt to 2544 MWt and the upcoming reactor refueling. This is supported in BAW-1521 "Crystal River Unit 3 - Cycle 2 Reload Report".

These changes include changing: 1) the Rated Thermal Power; 2) the Reactor Core Safety Limits and associated Bases; 3) the Limiting Safety System Settings and associated Bases; 4) the Regulating and Axial Power Shaping Rods insertion limits; 5) the control rod locations and group assignments; 6) the figures that specify the power level cutoff for Xenon Reactivity; 7) the Axial Power Imbalance Envelopes; 8) the Nuclear Heat Flux Hot Channel Factor and associated Bases; 9) the Quadrant Power Tilt Limits; and 10) the DNB Margin Limits.

Pages 2-6 and B 2-8 also contain changes proposed in Technical Specification Change Request No. 27, dated November 29, 1978. Page 3/4 2-11 also contains changes proposed in Technical Specification Change Request No. 29, dated June 6, 1978. Page 3/4 2-13 also contains changes proposed in Technical Specification Change Request No. 33, dated September 22, 1978.

Reason for Proposed Change

Crystal River Unit 3 will operate in Cycle 2 at a Rated Thermal Power of 2544 MWt with 56 fresh fuel assemblies. As stated in BAW-1521 "Crystal River Unit 3 - Cycle 2 Reload Report", some Technical Specifications need to be revised because of this. All of the proposed changes are the result of these modifications.

Safety Analysis of Proposed Change

The licensing considerations for operation of Crystal River Unit 3 with the increased Rated Thermal Power and the fresh fuel has been filed with the Commission in BAW-1521 "Crystal River Unit 3 - Cycle 2 Reload Report". These proposed changes will bring the Technical Specifications into agreement with that filing.

The review of the Technical Specifications based on the analyses presented in that report and the proposed changes ensure that the Final Acceptance Criteria ECCS limits will not be exceeded nor will the thermal design criteria be violated.

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2544 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

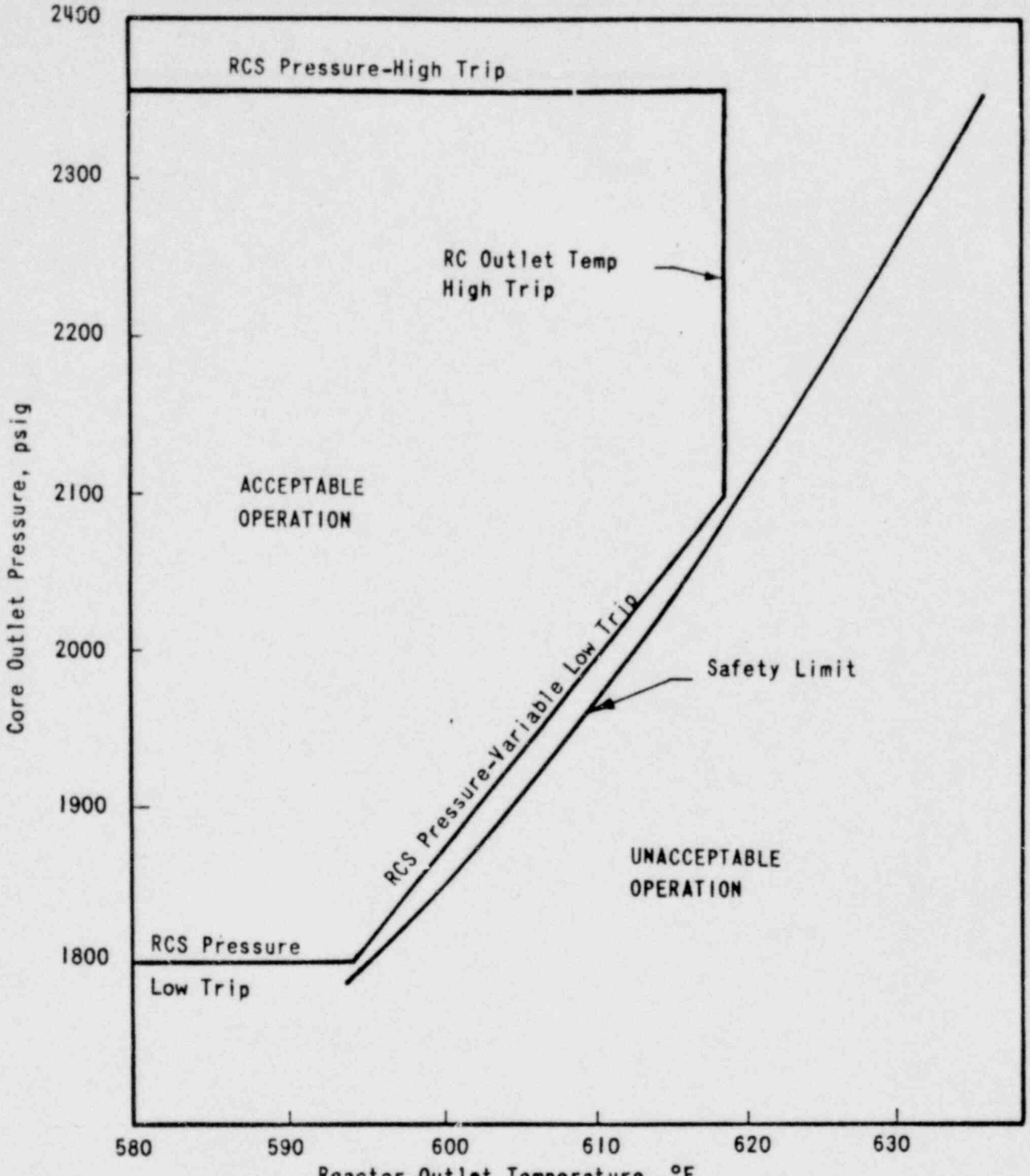
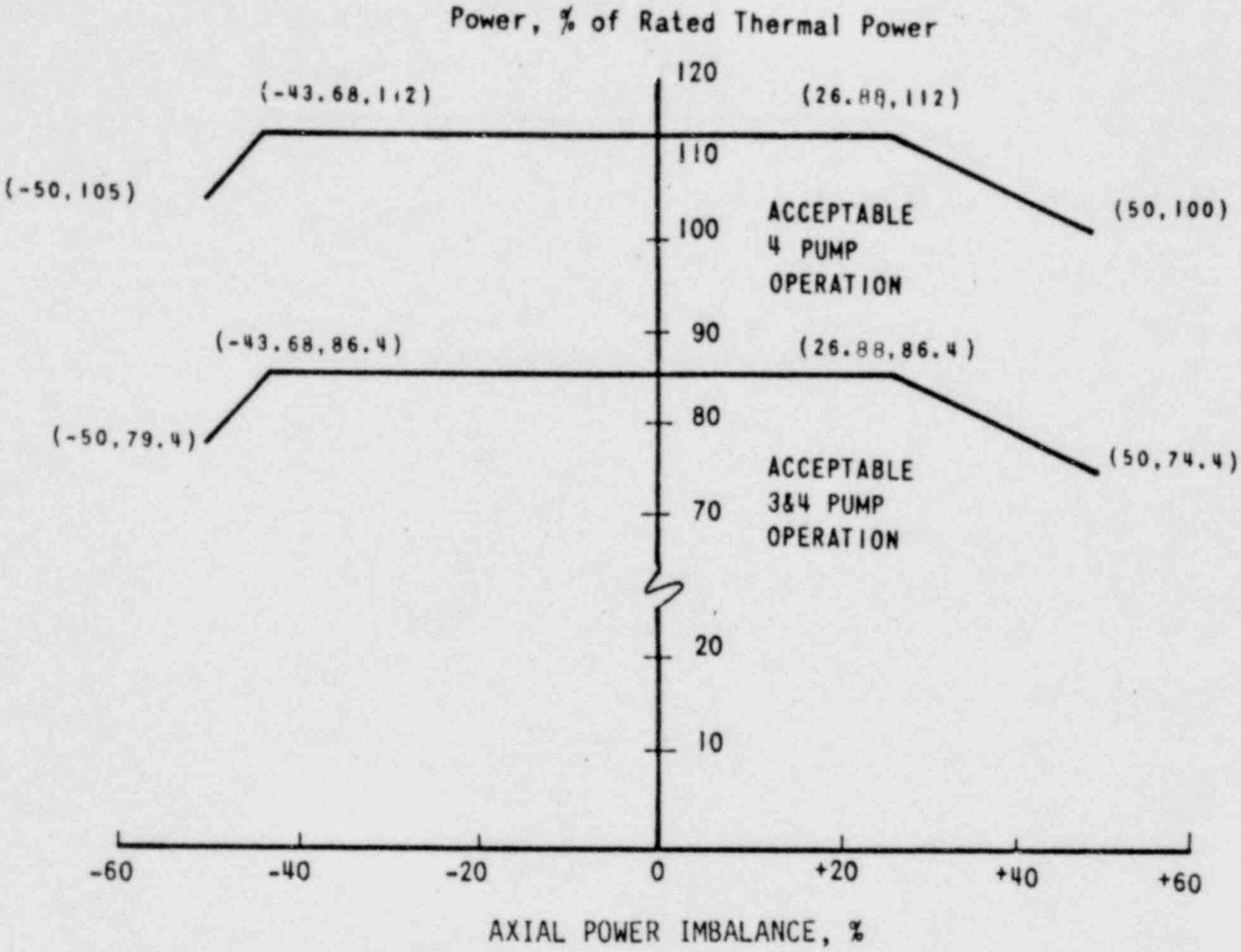


FIGURE 2.1-1

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

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
	$\leq$ 79.9% of RATED THERMAL POWER with three pumps operating	$\leq$ 79.9% 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	$\leq$ 2355 psig	$\leq$ 2355 psig
7. RCS Presssure-Variable Low(1)	$\geq$ (11.80T_{out} °F - 5209.2) psig	$\geq$ (11.80 T_{out} °F - 5209.2) psig

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Nuclear Overpower Based on RCPPMs(1)	$\leq$ 125% of RATED THERMAL POWER with three pumps operating	$\leq$ 125% of RATED THERMAL POWER with three pumps operating
	$\leq$ 0% of RATED THERMAL POWER with with less than three pumps operating	$\leq$ 0.28% of RATED THERMAL POWER with less than three pumps operating
9. Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig

(1)Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of $\leq$ 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

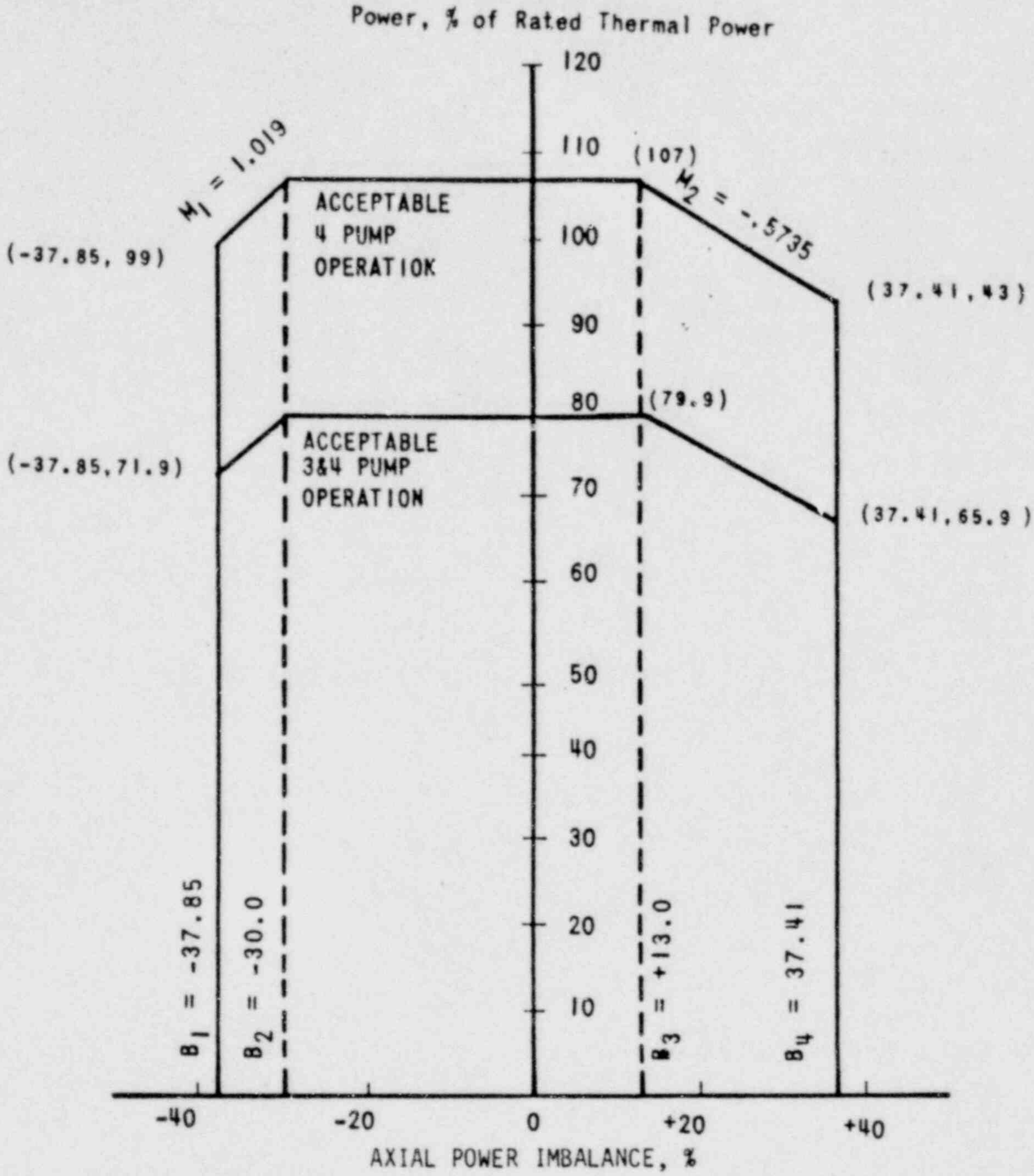


FIGURE 2.2-1

TRIP SETPOINT FOR NUCLEAR OVERPOWER BASED ON RCS FLOW AND AXIAL POWER IMBALANCE

2.1 SAFETY LIMITS

BASES

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 139.6 x 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.

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip $\leq$ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power.

The power level trip setpoint 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 setpoint 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 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is $\geq$ 107.0% and reactor flow rate is 100%, or flow rate is $\leq$ 93.5% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is $\geq$ 79.9% and reactor flow rate is 74.7%, or flow rate is $\leq$ 70.1% and power is 75%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

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 AXILA POWER IMBALANCE reduces the power level trip orduced 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 1.07% 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.

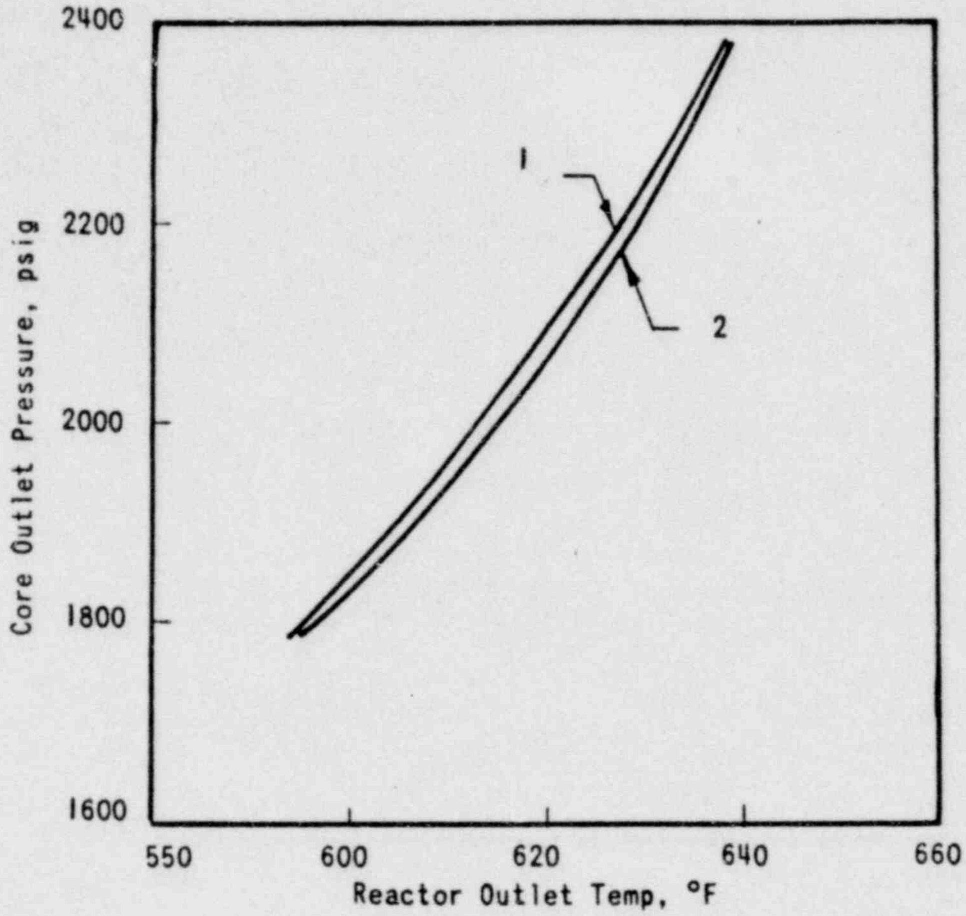
During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure 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 $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure-Low trip.



REACTOR COOLANT FLOW

CURVE	FLOW (lb/hr)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.86×10^6 (106.7%)	113.1%	4 Pumps (DNBR)
2	104.47×10^6 (79.7%)	87.2%	3 Pumps (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM
ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

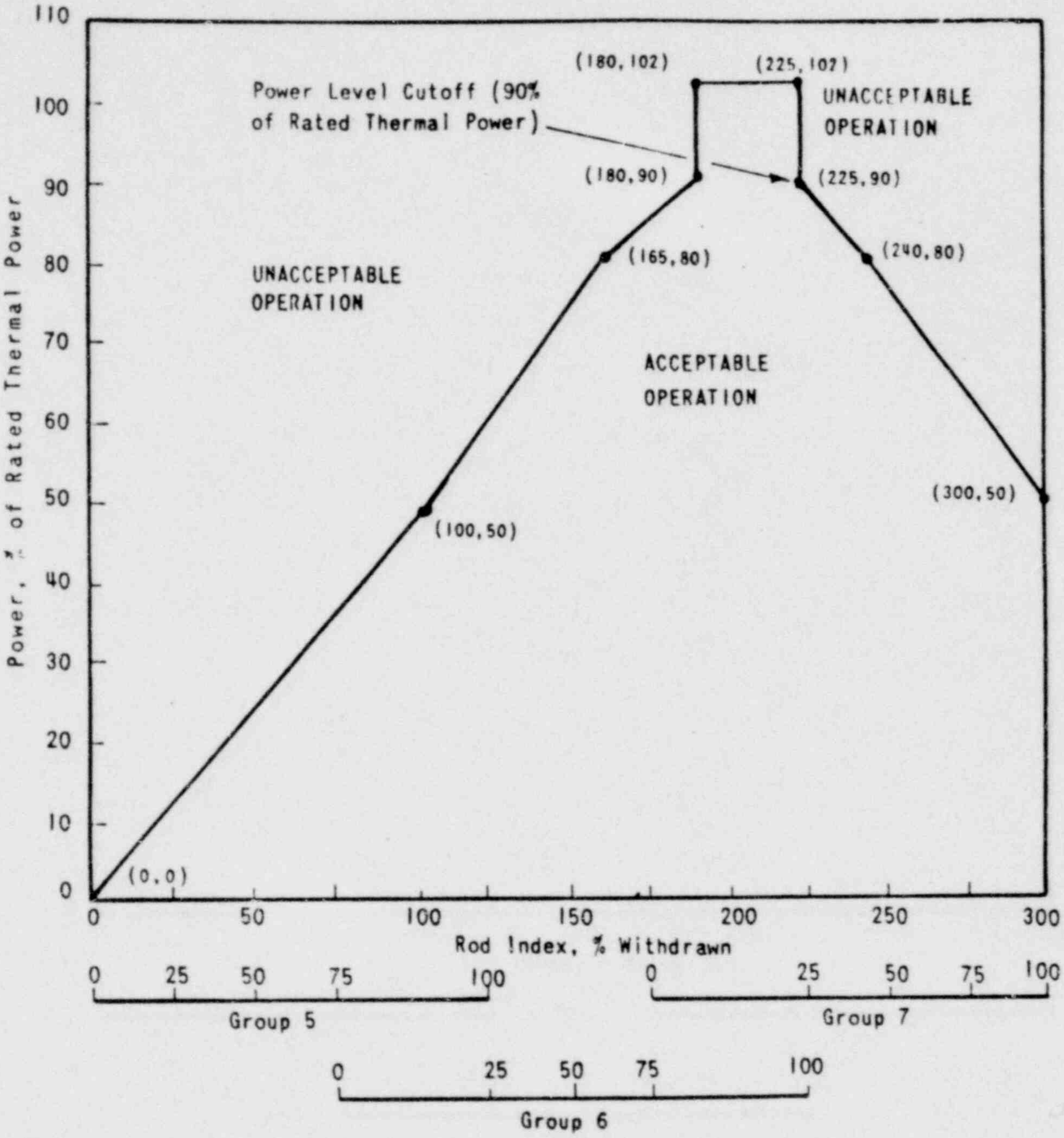


FIGURE 3.1-1

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION FROM 0 EFPD TO 225 ± 10 EFPD

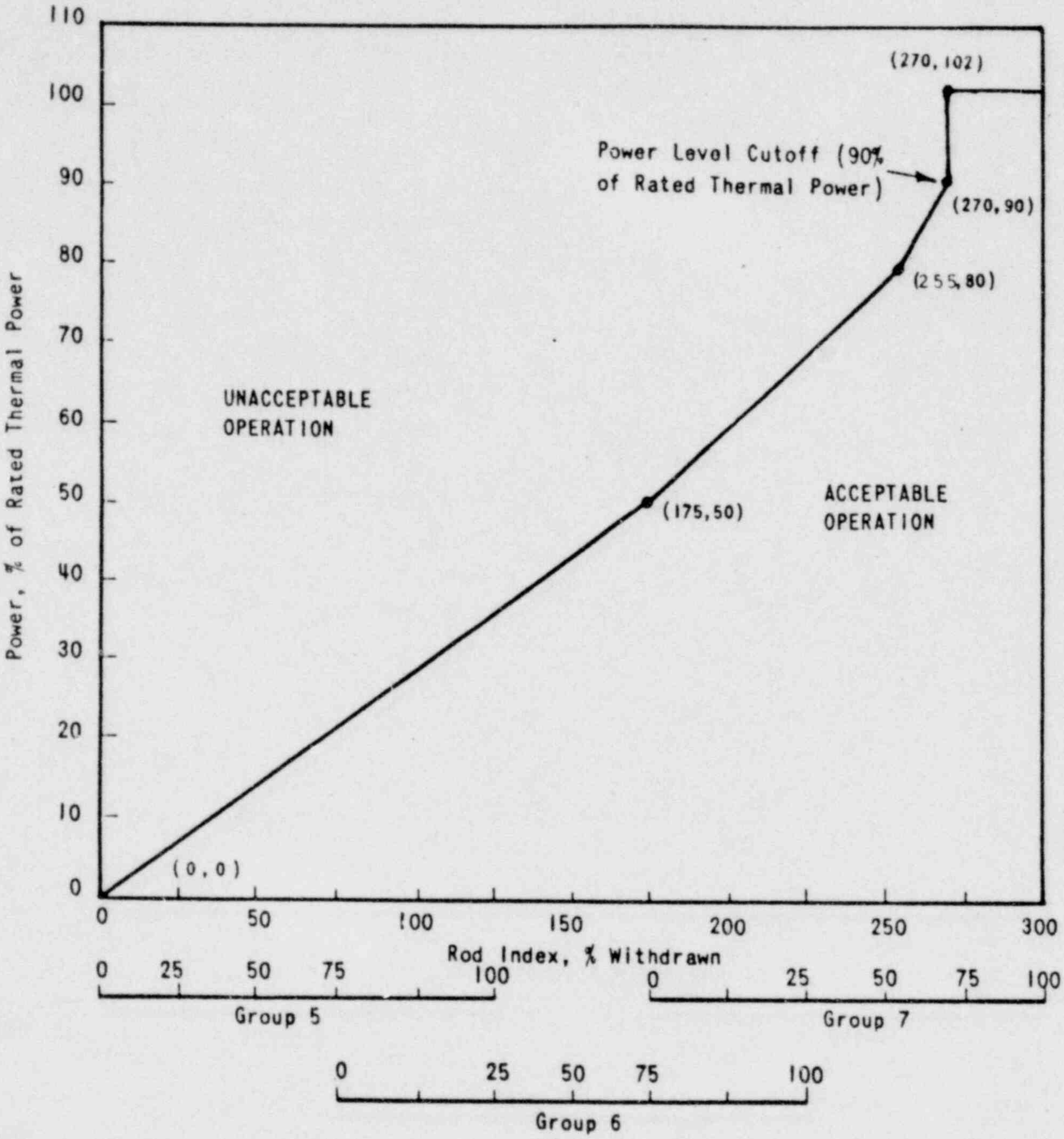


FIGURE 3.1-2

REGULATING ROD GROUP INSERTION LIMITS FOR
4 PUMP OPERATION AFTER 225 ± 10 EFPD

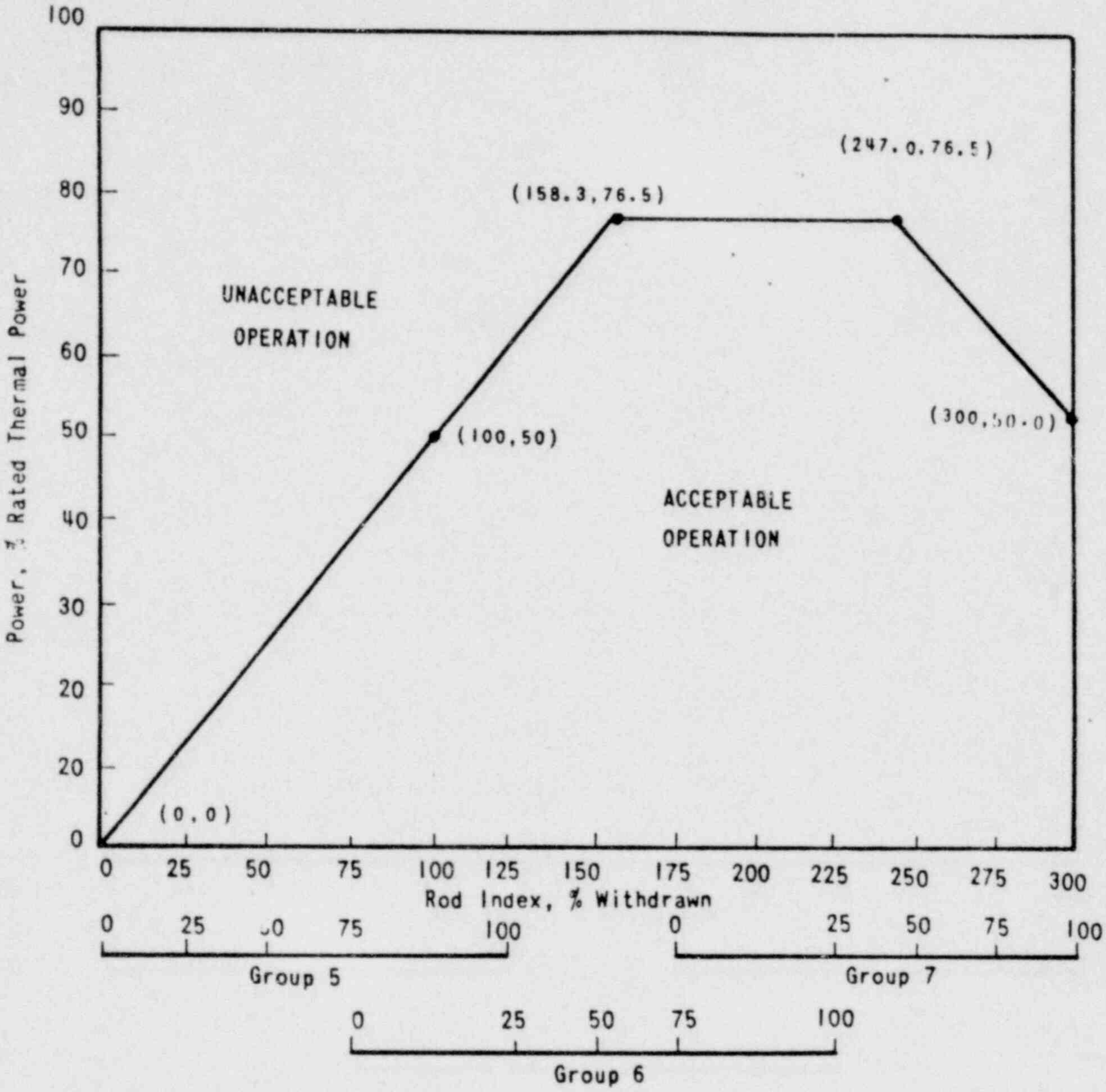


FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION FROM 0 EFPD TO 225 ± 10 EFPD

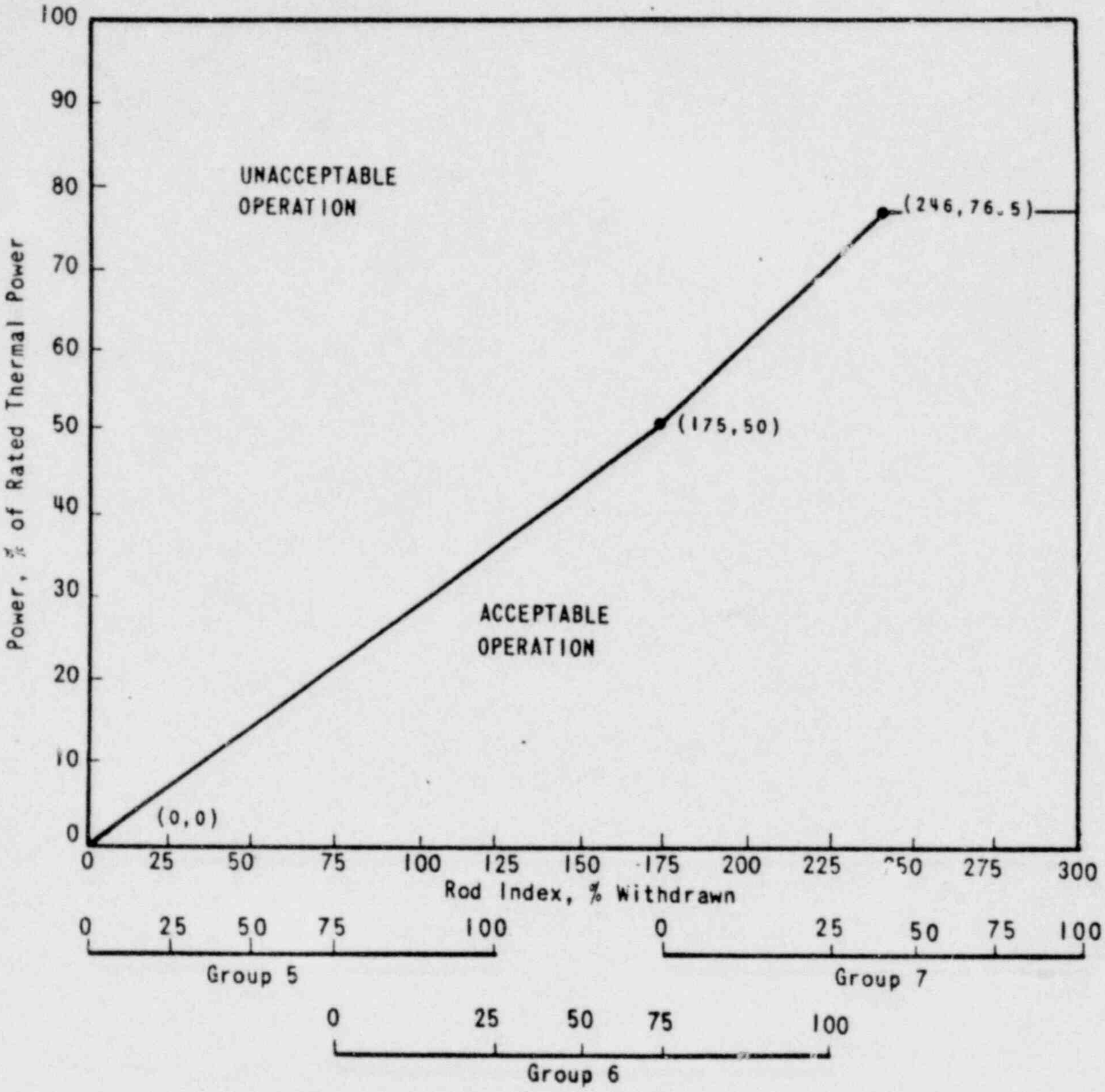


FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR
3 PUMP OPERATION AFTER 225 ± 10 EFPD

REACTIVITY CONTROL SYSTEMS

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:

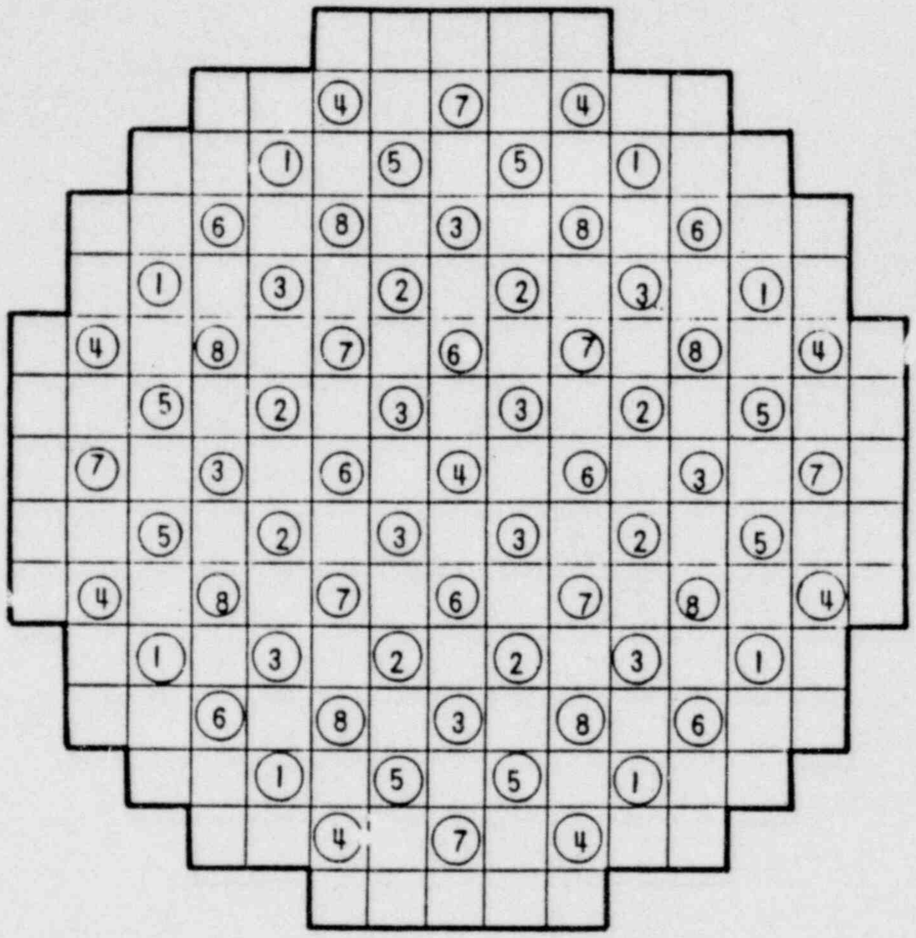
      a) For all control rods, after the control rod drive patchs are locked subsequent to test, reprogramming or maintenance within the panels.

      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.



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

DELETED

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.

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 $k_{eff} \geq 1.0$.

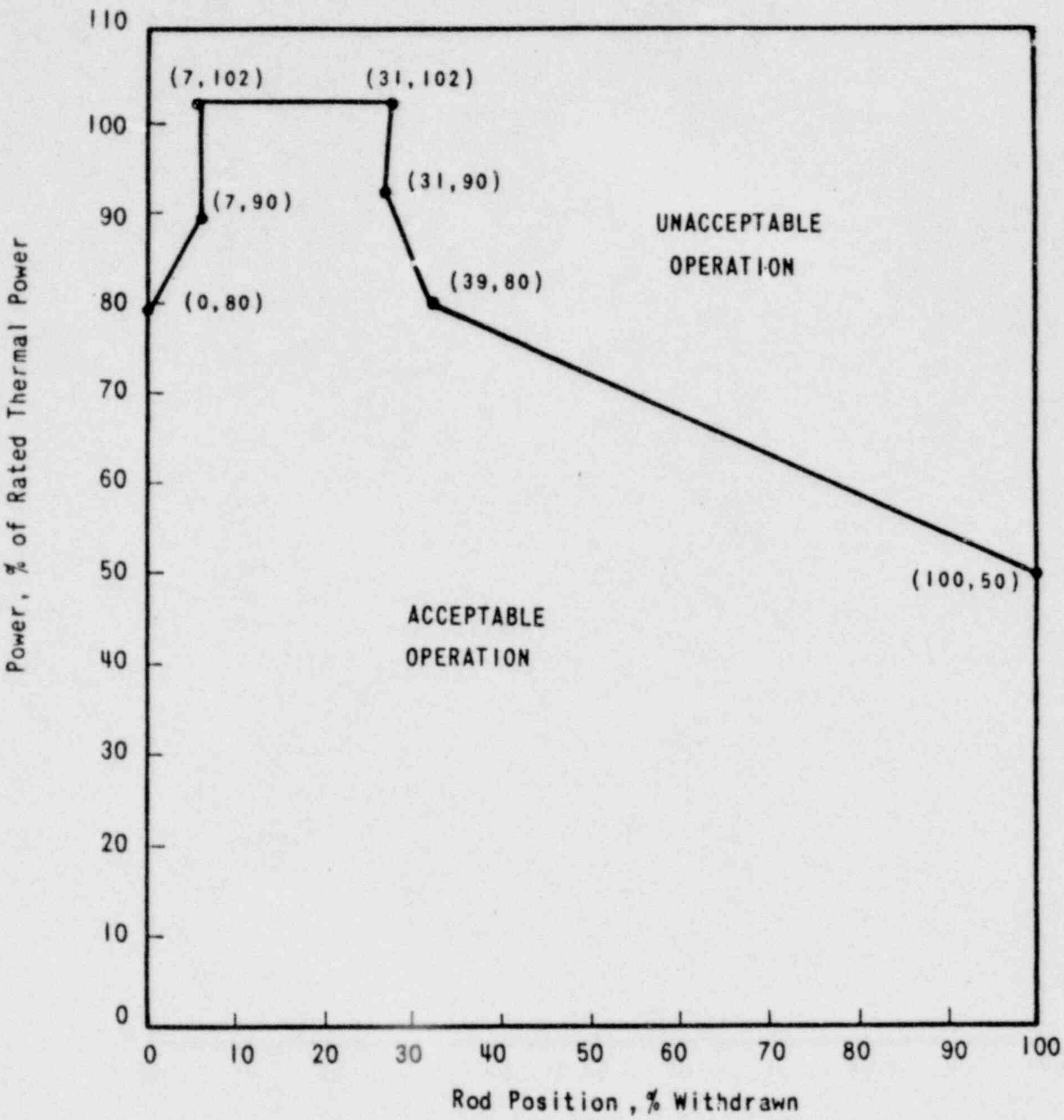


FIGURE 3.1-9

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS
FROM 0 EFPD TO 225 $\pm$ 10 EFPD

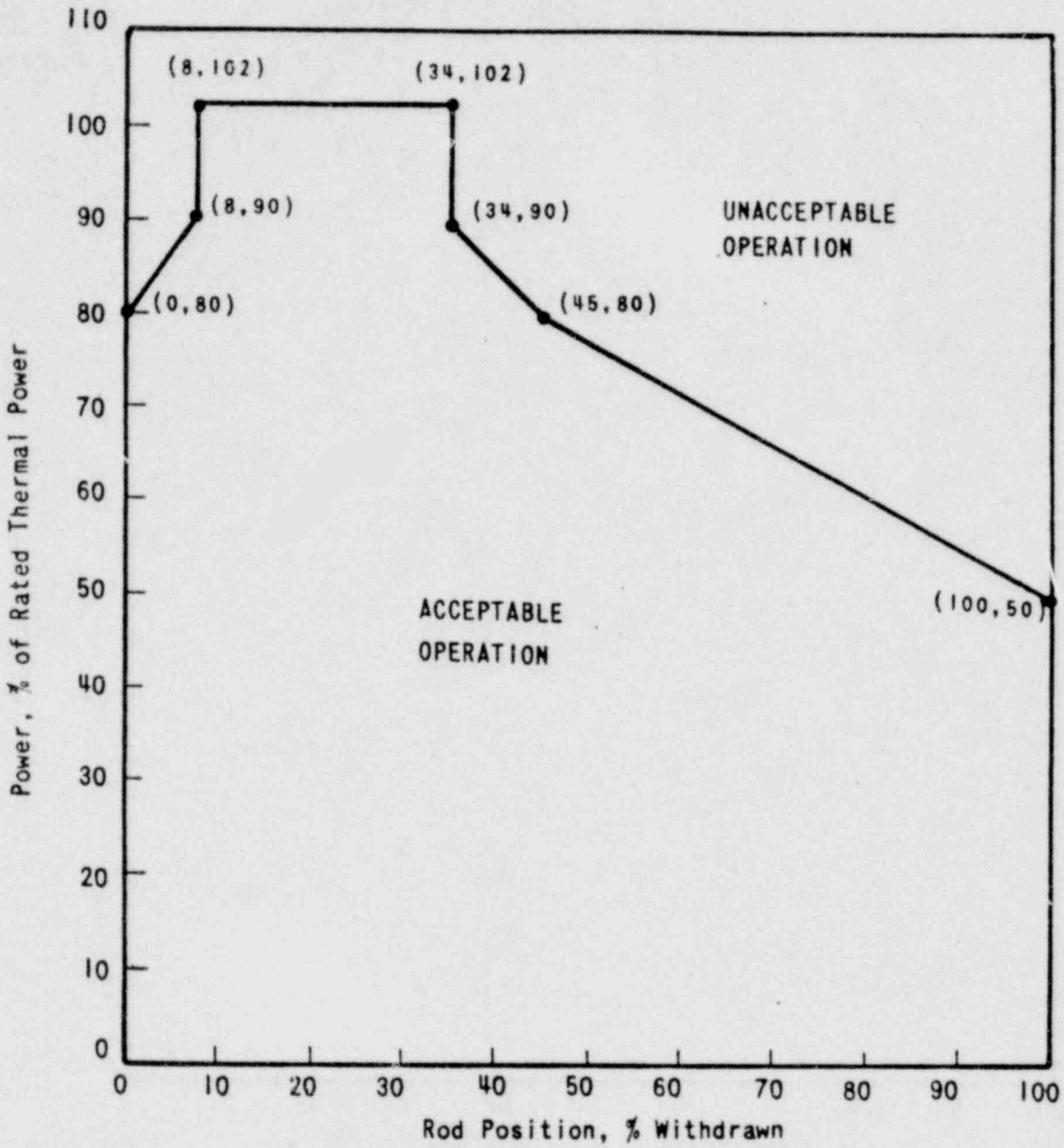


FIGURE 3.1-10

AXIAL POWER SHAPING ROD GROUP
INSERTION LIMITS AFTER 225 ± 10 EFPD

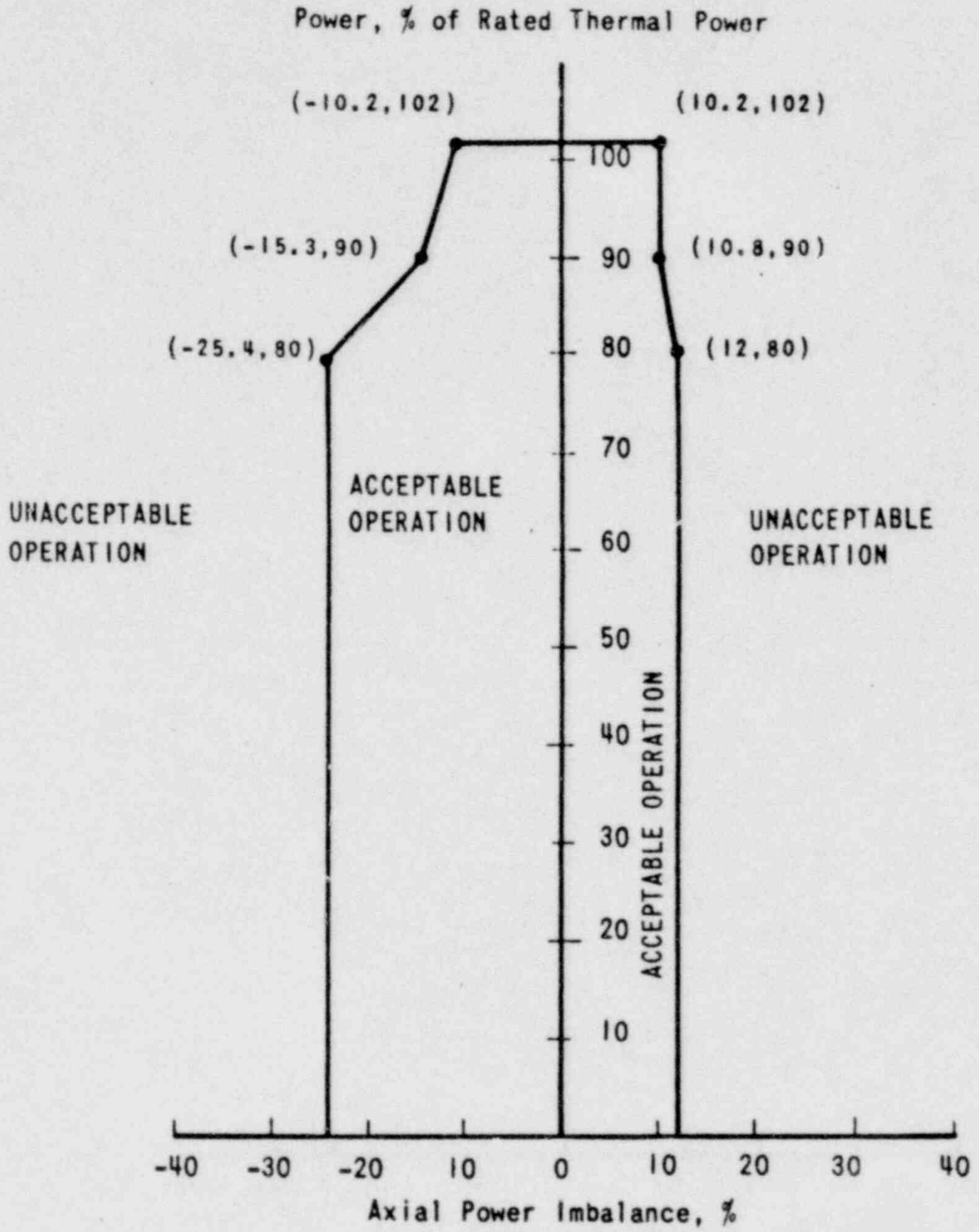


FIGURE 3.2-1

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 EFPD TO 225 ± 10 EFPD

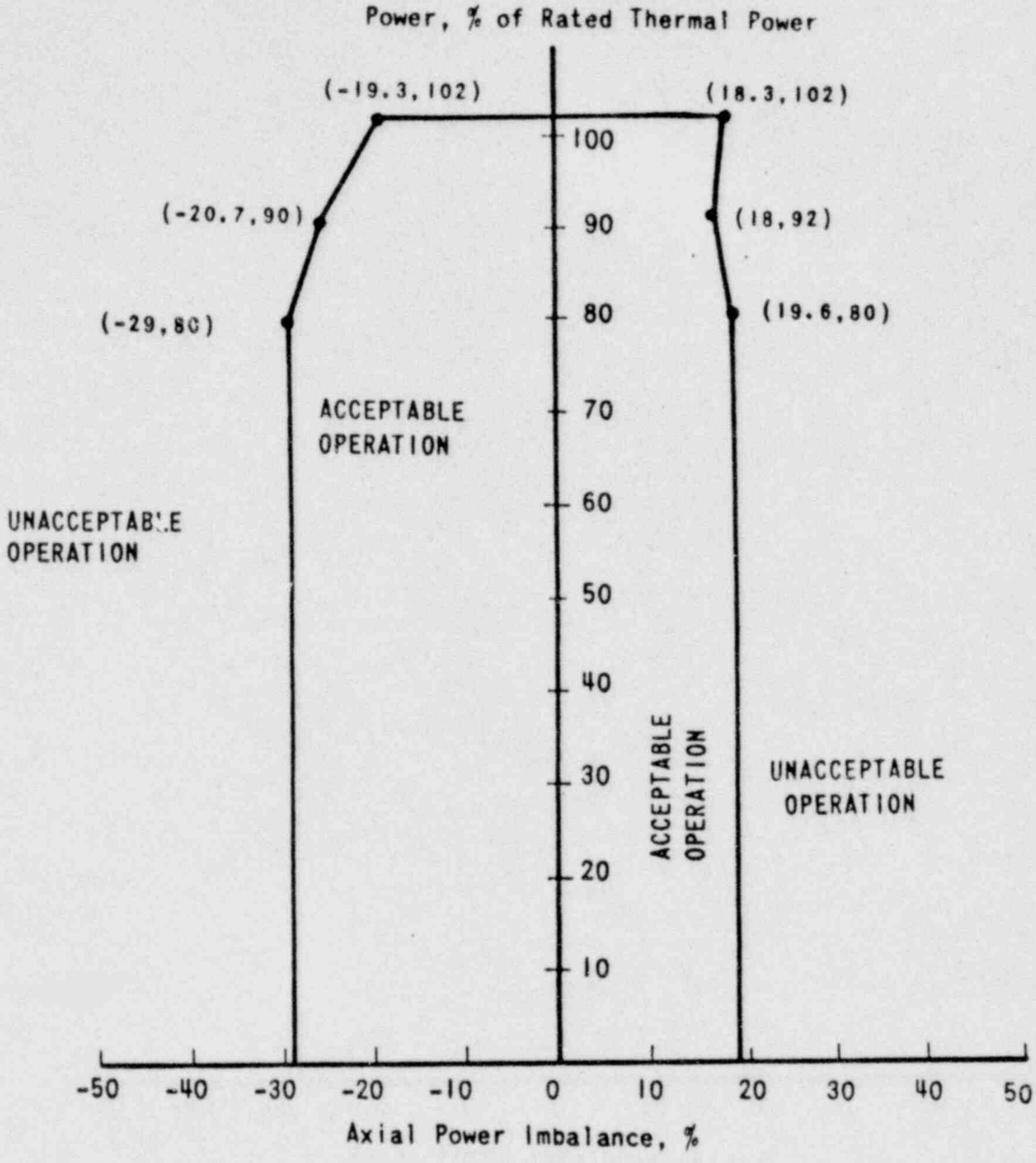


FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 225 $\pm$ 10 EFPD

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:

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	$\geq 139.86 \times 10^6$	$\geq 104.47 \times 10^6$

[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.

$F^N_{\Delta H}$ 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.