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

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-3

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

   A. The application for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated March 5, 1982, as revised and supplemented March 23, 1982, June 1, 1982, and June 21, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. NPF-3 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   Revise paragraph 2.C.(2) to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The Toledo Edison Company 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

(for) John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 28, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages	
2-2	3/4 2-1
2-3	3/4 2-2
2-5	3/4 2-2a
2-6	3/4 2-2b
2-7	3/4 2-2c (new page)
2-8	3/4 2-2d (" ")
B2-2	3/4 2-3
B2-3	3/4 2-3a
B2-4	3/4 2-3b
B2-5	3/4 2-3c (new page)
B2-6	3/4 2-3d (" ")
B2-8	3/4 2-5
3/4 1-4	3/4 2-12
3/4 1-26	3/4 3-6
3/4 1-28	3/4 4-1
3/4 1-28a	B3/4 1-2
3/4 1-28b	B3/4 1-4
3/4 1-28c (new page)	B3/4 2-1
3/4 1-28d (" ")	B3/4 2-3
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3/4 1-29d (new page)	
3/4 1-31	
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3/4 1-36	
3/4 1-37	
3/4 1-38	
3/4 1-39	
3/4 1-40 (new page)	
3/4 1-41 (" ")	
3/4 1-42 (" ")	
3/4 1-43 (" ")	

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

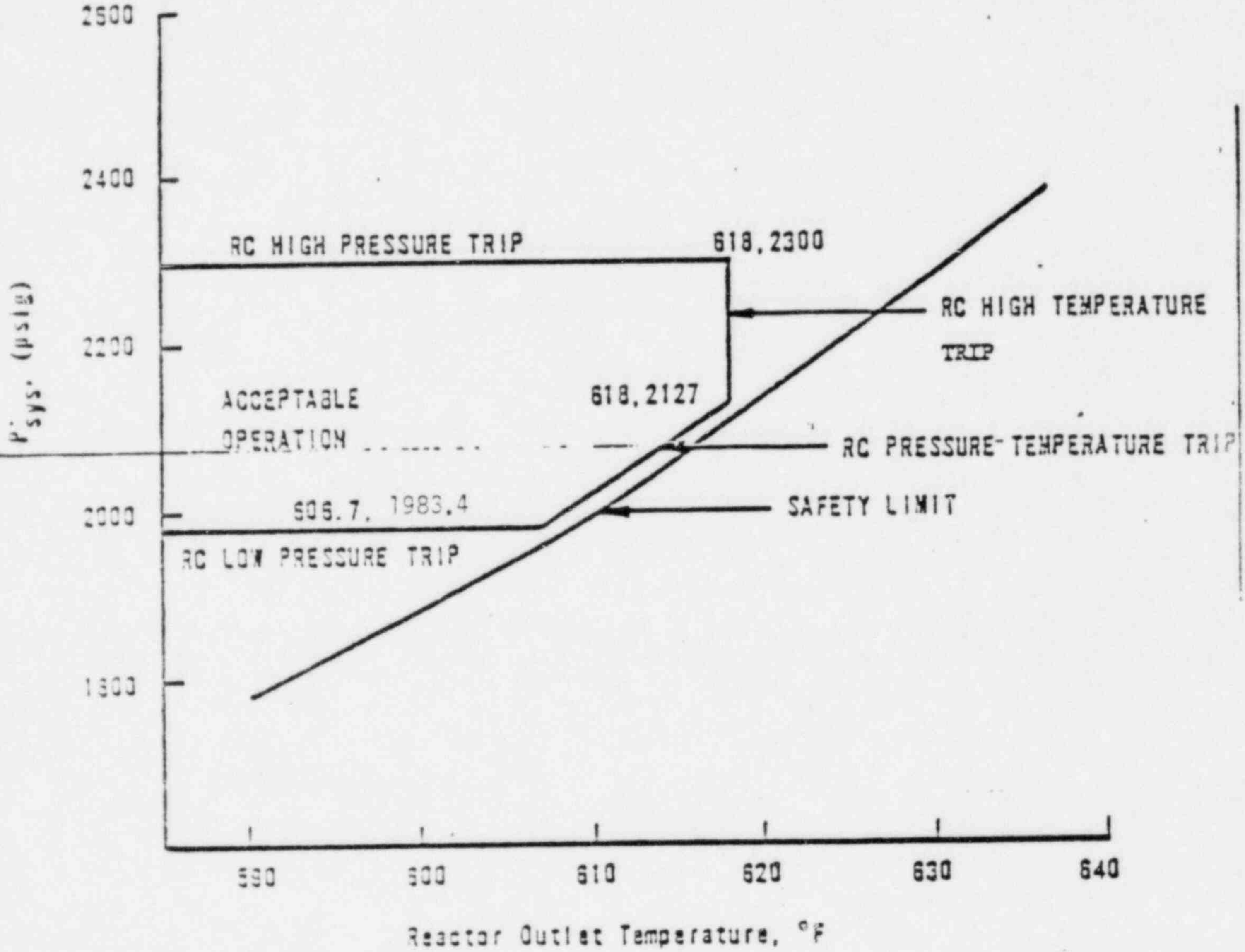
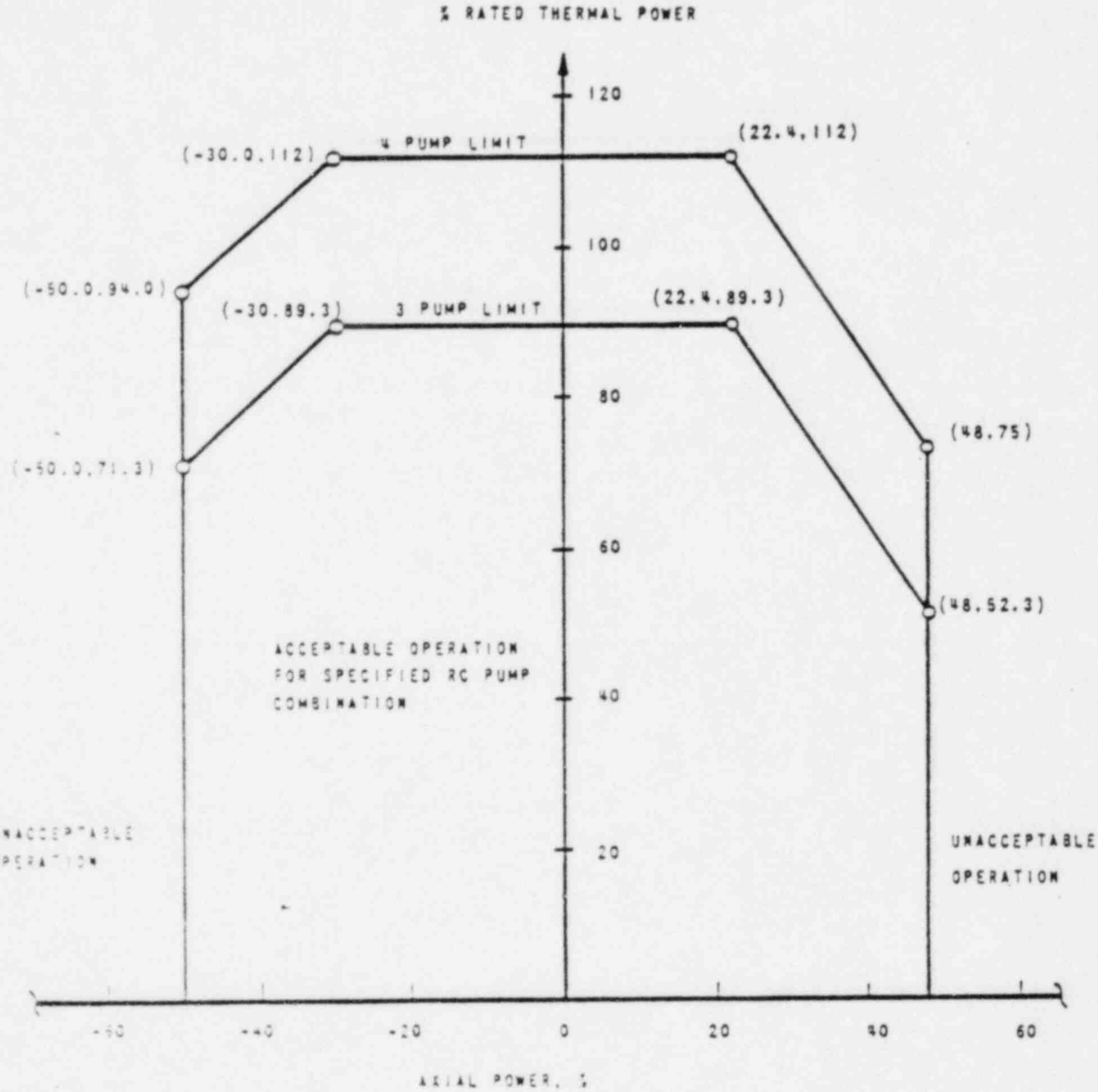


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT

Figure 2.1-2 - Reactor Core Safety Limit



PUMPS OPERATING	REACTOR COOLANT FLOW, (GPM)
4	387,200
3	290,100

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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. High flux	≤105.1% of RATED THERMAL POWER with four pumps operating ≤79.6% of RATED THERMAL POWER with three pumps operating	≤105.1% of RATED THERMAL POWER with four pumps operating# ≤79.6% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux – Δ flux/flow(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. RC low pressure(1)	≥ 1983.4 psig	≥1983.4 psig* ≥1983.4 psig**
6. RC high pressure	≤2300 psig	≤2300.0 psig* ≤2300.0 psig**
7. RC pressure-temperature(1)	$\geq(12.60\ T_{out}\ °F - 5662)$ psig	$\geq(12.60\ T_{out}\ °F - 5662)$ psig#
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop ≤0.0% of RATED THERMAL POWER with two pumps operating in one loop an no pumps operating in the other loop ≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤55.1% of RATED THERMAL POWER with one pump operating in each loop# ≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop# ≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	≤4 psig	≤4 psig#

Table 2.2-1. (Cont'd)

(1) Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

**Allowable value for CHANNEL CALIBRATION.

#Allowable value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

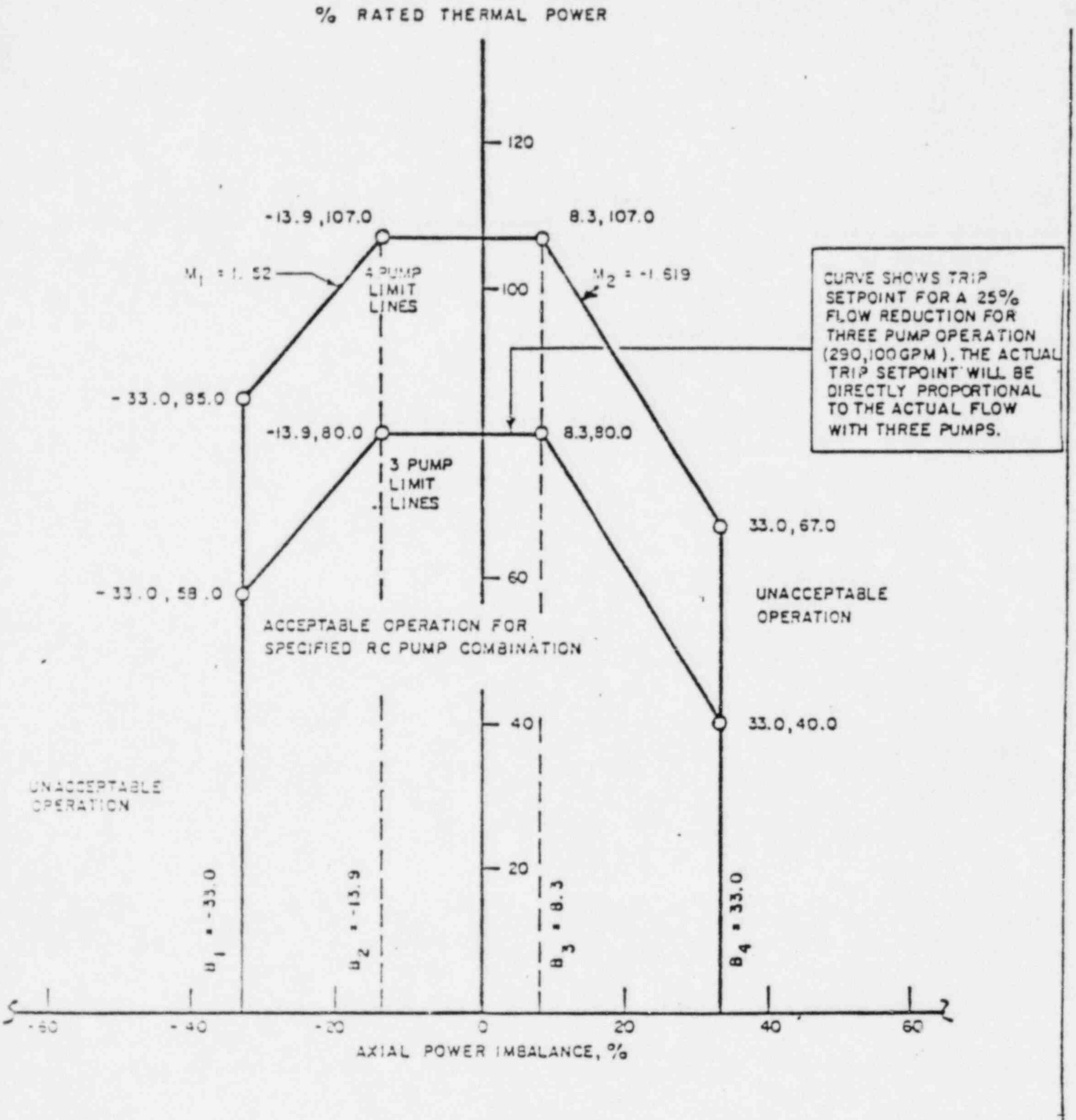


FIGURE 2.2-1 TRIP SETPOINT FOR FLUX - Δ FLUX/FLOW

Figure 2.2-2 Allowable Value for Flux-Δ Flux/Flow

DELETED

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 regime 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 B&W-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 387, 200 GPM, which is 110% of design flow rate for four operating reactor coolant pumps. This curve is based on the following hot channel factors with potential fuel densification and fuel rod bowing effects:

$$F_Q = 2.56; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = 1.50$$

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 reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of F_Q = 2.56 or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits are 20.4 kW/ft for batches 1C, 4, and 55 batch 3 assemblies; 20.35 for the five remaining batch 3 assemblies; and 20.3 for batch 5.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 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%, whichever condition is more restrictive. These curves include the potential effects of fuel rod bow and fuel densification.

The DNBR as calculated by the B&W-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of +22% is justified on the basis of experimental data.

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than +22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for three pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the three pump curve will be above and to the left of the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2. LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1. REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation trip setpoint specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The trip setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The high flux trip setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.1% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RC High Temperature

The RC high temperature trip $\leq$618 F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux — Δ Flux/Flow

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 where protection is not provided by the high flux/number of reactor coolant pumps on trips.

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. Examples of typical power level and low flow rate combinations for the pump situations of Table 2.2-1 that would result in a trip are as follows:

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

2. Trip would occur when three reactor coolant pumps are operating if power is 80.0% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 70.0% of full flow rate and power is 75%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used. Full flow rate in the above two examples is defined as the flow calculated by the heat balance at 100% power.

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 a flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced.

RC Pressure — Low, High, and Pressure Temperature

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 RC high pressure setpoint is reached before the high flux trip setpoint. The trip setpoint for RC high pressure, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2435 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1283.4 psig, and RC pressure-temperature ($12.60\ T_{out} - 5662$) 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.

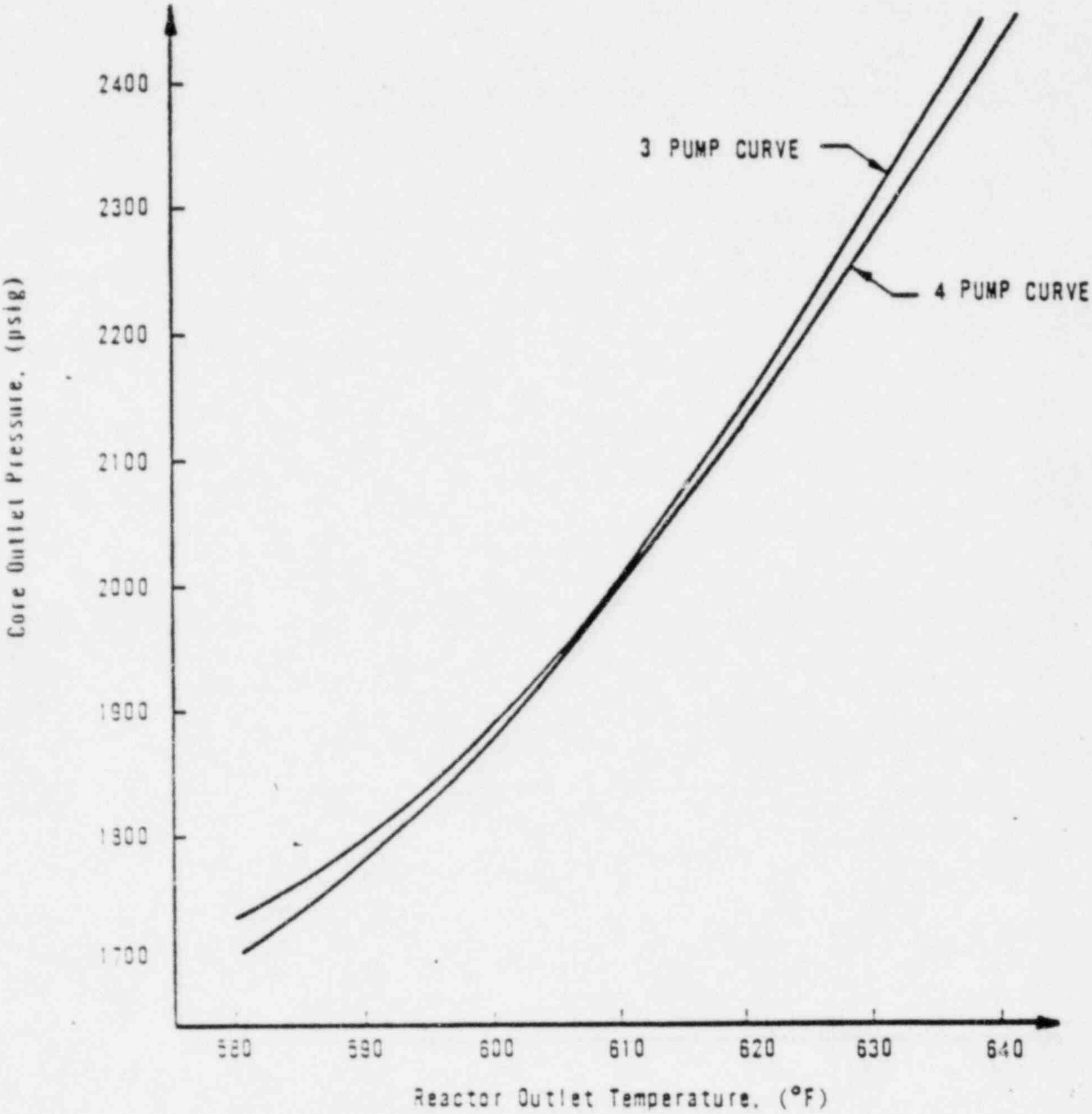
High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux − Δ flux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

Containment High Pressure

The Containment High Pressure 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 RC Low Pressure trip.

Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR



PUMPS	FLOW (GPM)	POWER
4	387,200	112%
3	290,100	89.3%

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.2 The flow rate of reactor coolant through the Reactor Coolant System shall be ≥ 2800 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All MODES.

ACTION:

With the flow rate of reactor coolant through the Reactor Coolant System < 2800 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The flow rate of reactor coolant through the Reactor Coolant System shall be determined to be ≥ 2800 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one DHR pump is in operation and supplying ≥ 2800 gpm to the Reactor Coolant System.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.9×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is < 95% of RATED THERMAL POWER,

b. Less positive than 0.0×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is ≥ 95% of RATED THERMAL POWER, and

c. Equal to or less negative than -3.0×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside any of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 days after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

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

#See Special Test Exception 3.10.2.

REACTIVITY CONTROL SYSTEMS

SAFETY ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All safety rods shall be fully withdrawn.

APPLICABILITY: 1* and 2*#.

ACTION:

With a maximum of one safety rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each safety rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any regulating rod during an approach to reactor criticality.

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2a, -2b, and -2c and 3.1-3a, -3b, and -3c for the first 200 ±10 EFPD of operation. If the axial power shaping rods are completely withdrawn at 200 ±10 EFPD for extension of cycle length, then the regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2e and 3.1-3e for the remainder of the cycle. However, if the axial power shaping rods are not completely withdrawn at 200 ±10 EFPD, then the regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2d and 3.1-3d for the remainder of the cycle. A rod group overlap of 25 ±5% shall be maintained between sequential withdrawn groups 5, 6 and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits (in a region other than acceptable operation), or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups 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 figures within 2 hours, or

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

NOTE: If in unacceptable region, also see Section 3/4.1.1.1.

*See Special Test Exceptions 3.10.1 and 3.10.2.

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

REGULATING ROD INSERTION LIMITS

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating group shall be determined to be within the insertion, sequence and overlap limits at least once every 12 hours except when:

a. The regulating rod insertion limit alarm is inoperable, then verify the groups to be within the insertion limits at least once per 4 hours;

b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.

Figure 3.1-2a Regulating Group Position Limits, 0 to 60 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

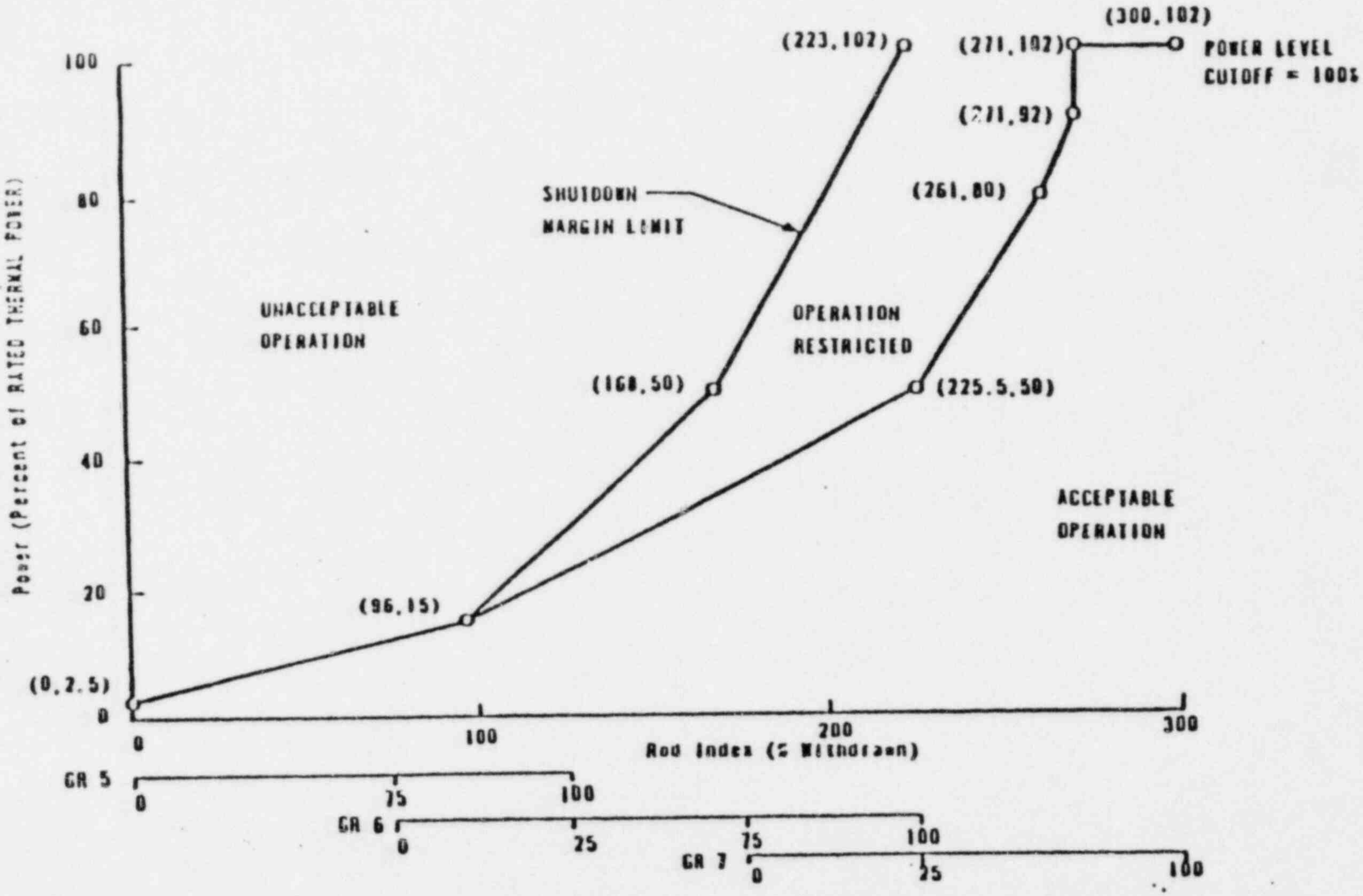


Figure 3.1-2b Regulating Group Position Limits, 50 to 150 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

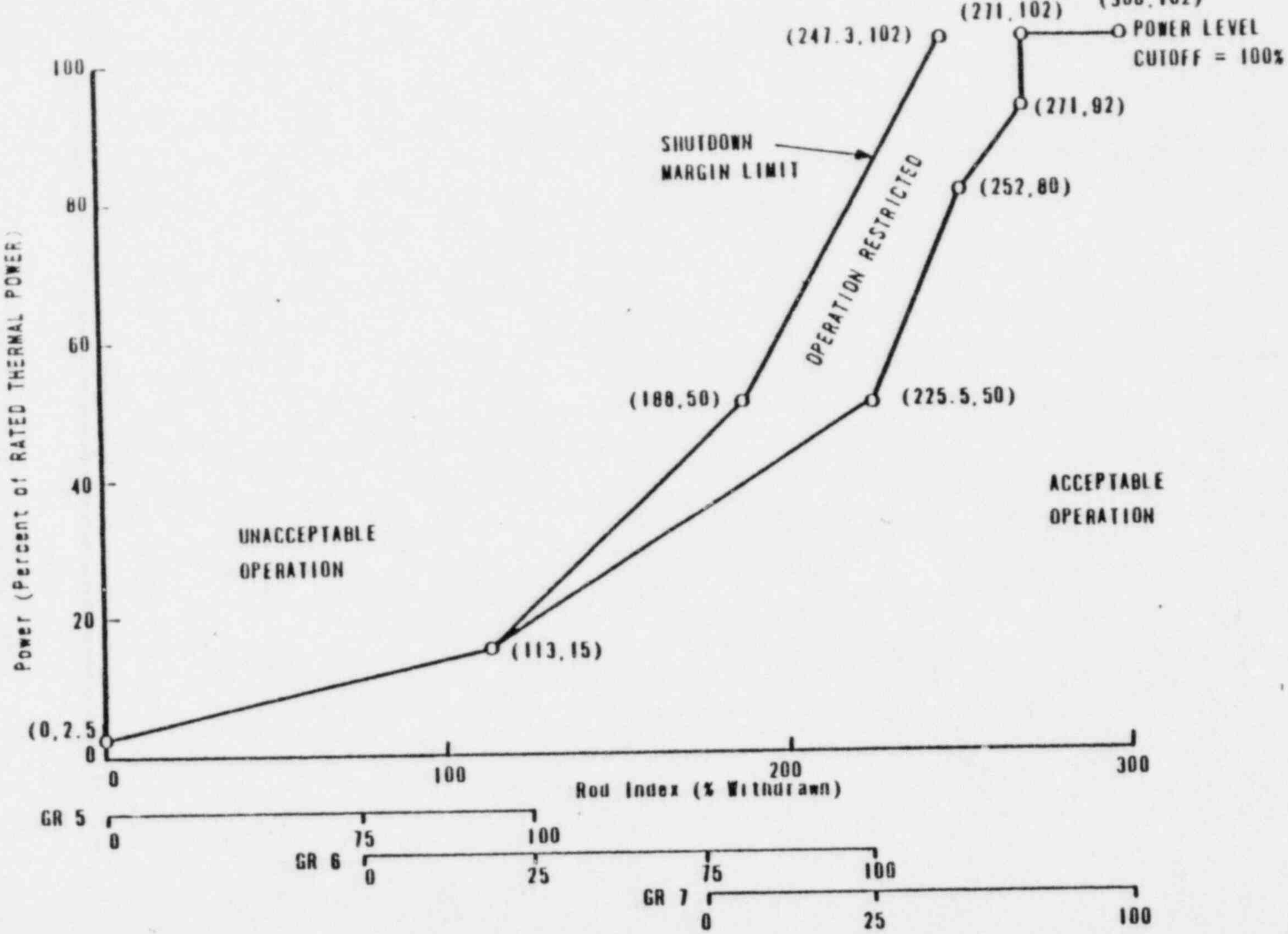


Figure 3.1-2c Regulating Group Position Limits, 150 ± 10 to 200 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

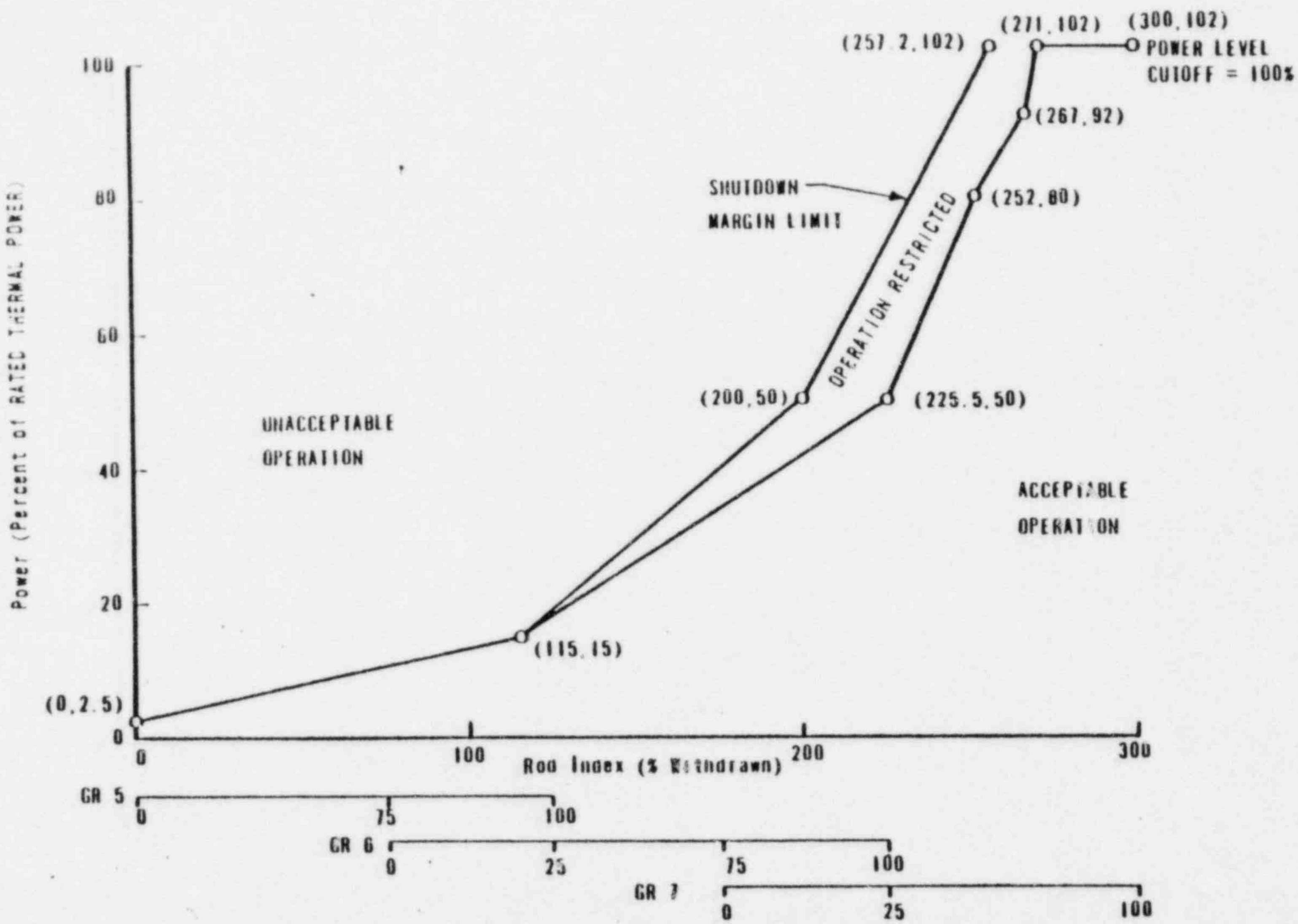


Figure 3.1-2d Regulating Group Position Limits, 200 ± 10 to 230 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

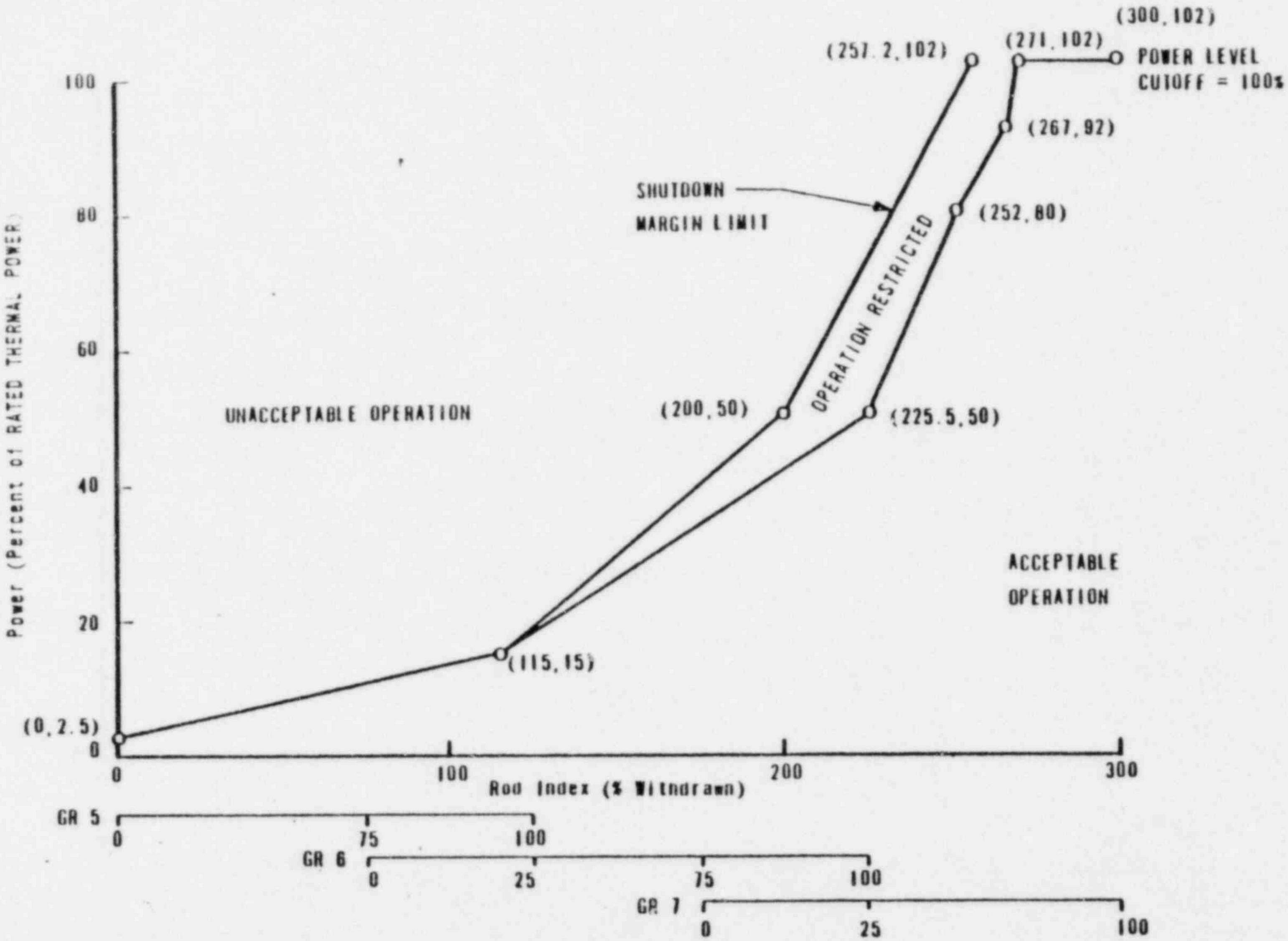


Figure 3.1-2e Regulating Group Position Limits, 200 ± 10 to 268 ± 10 EFPD, Four RC Pumps, APSRs Withdrawn — Davis-Besse 1, Cycle 3

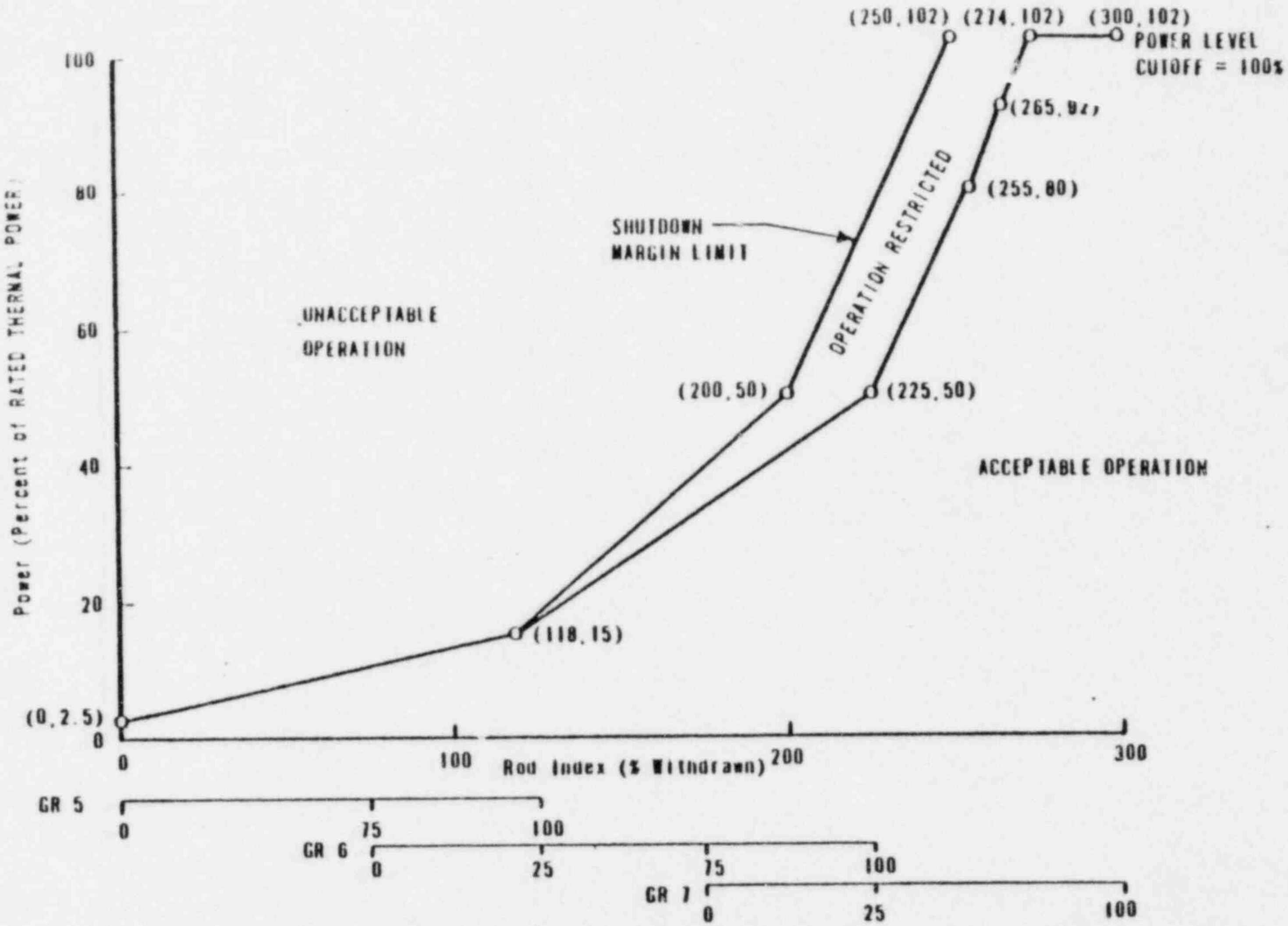


Figure 3.1-3a Regulating Group Position Limits, 0 to 60 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

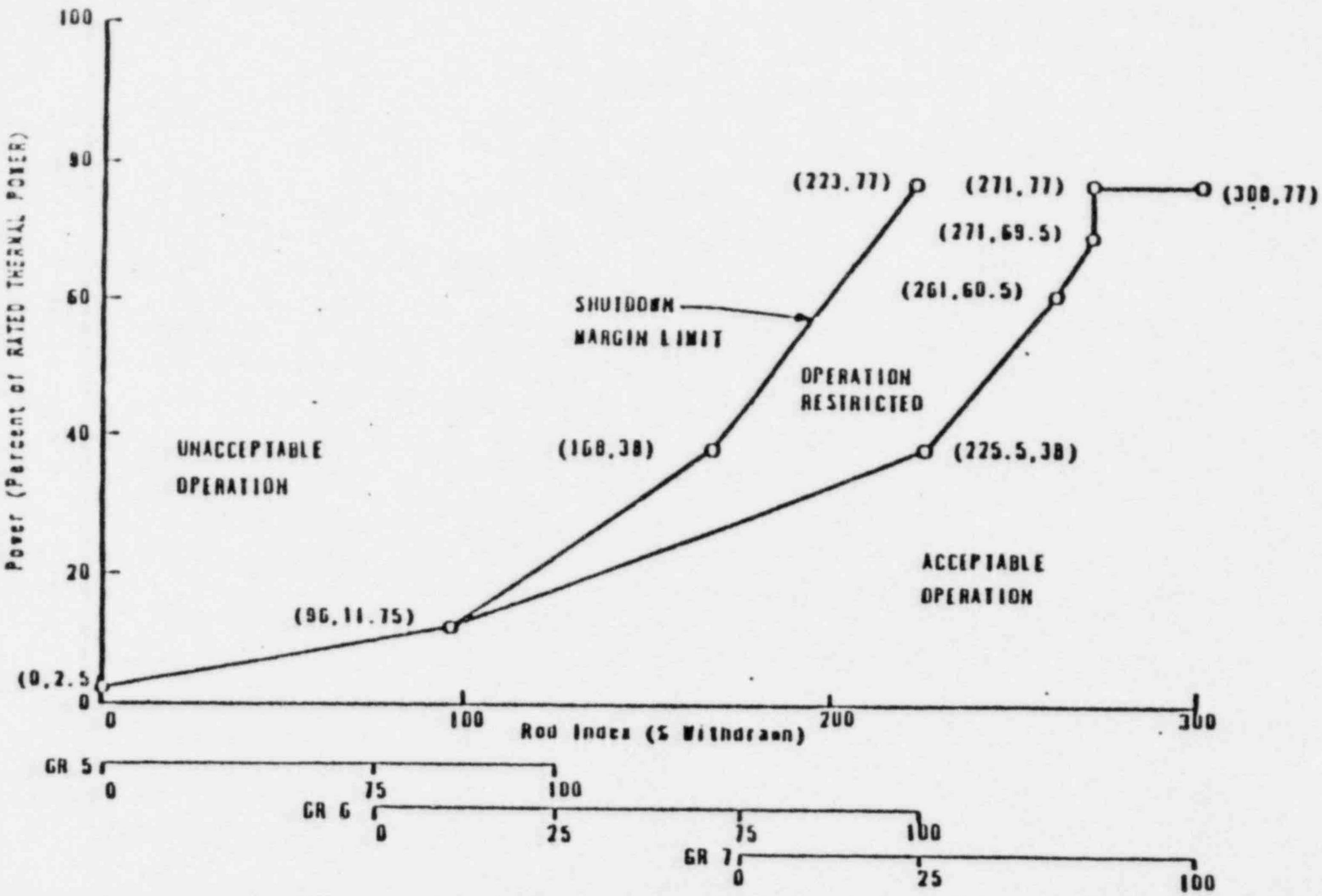


Figure 3.1-3b   Regulating Group Position Limits, 50 to 150 ± 10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 3

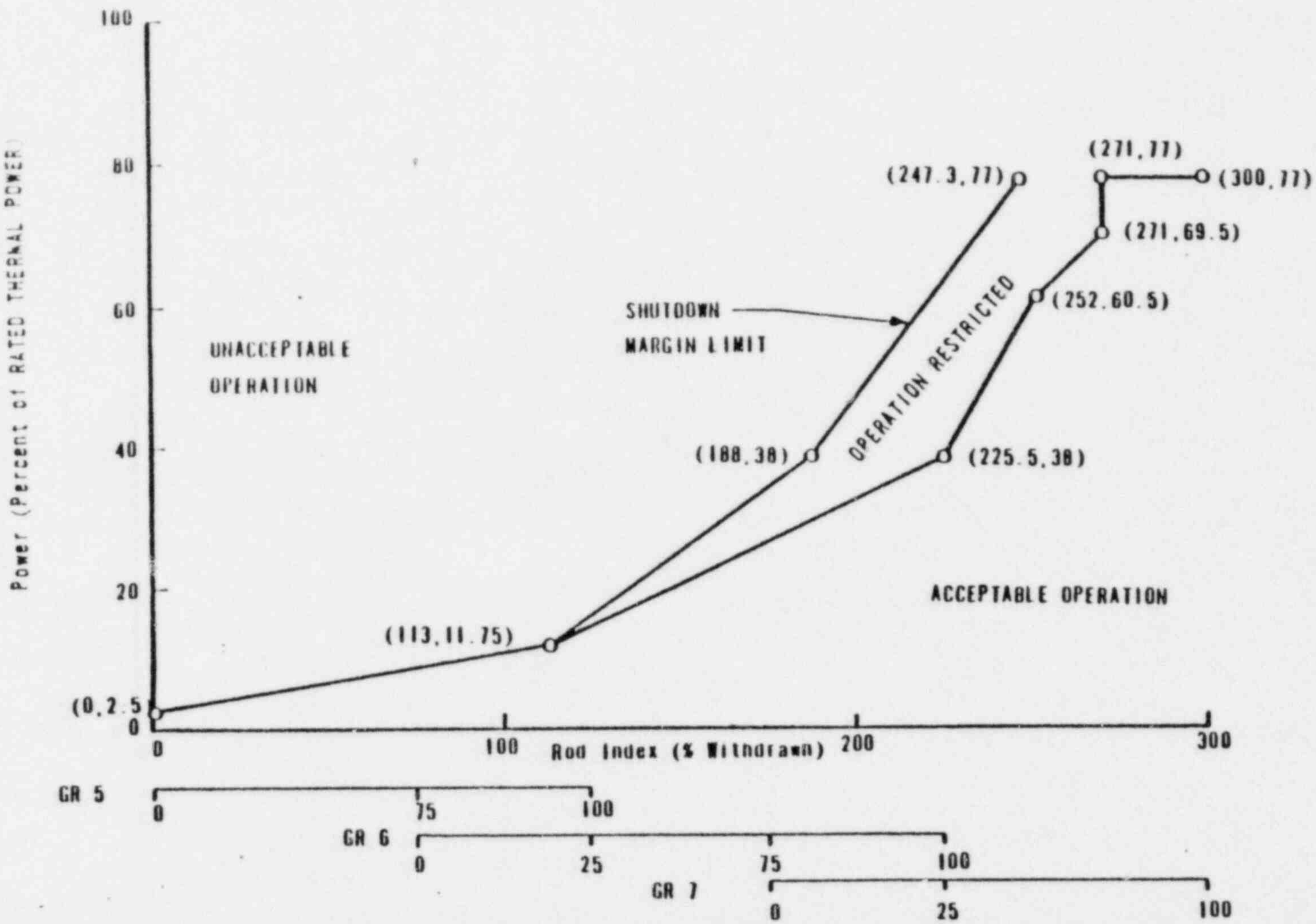


Figure 3.1-3c Regulating Group Position Limits, 150 ± 10 to 200 ± 10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 3

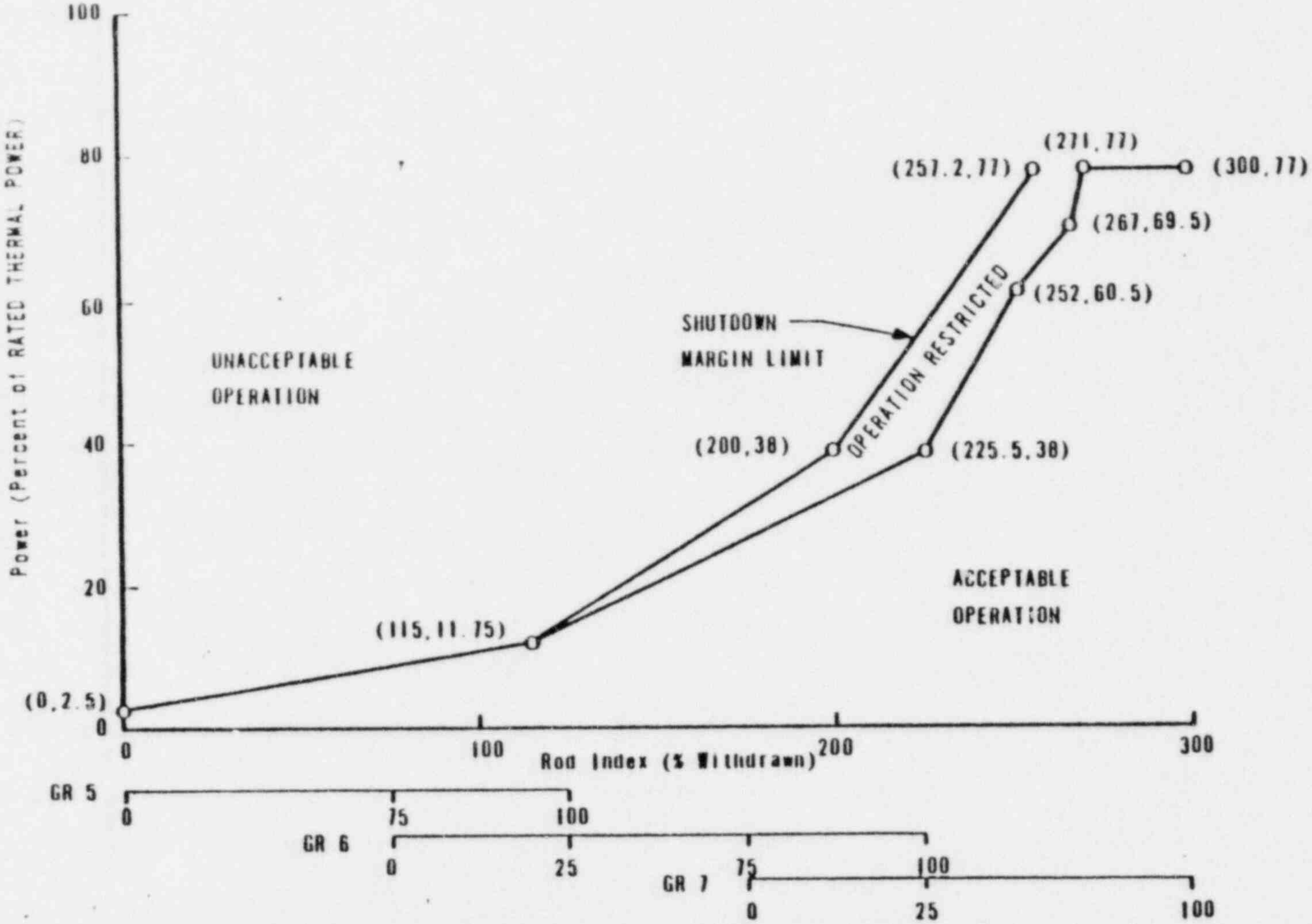


Figure 3.1-3d Regulating Group Position Limits, 200 ± 10 to 230 ± 10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 3

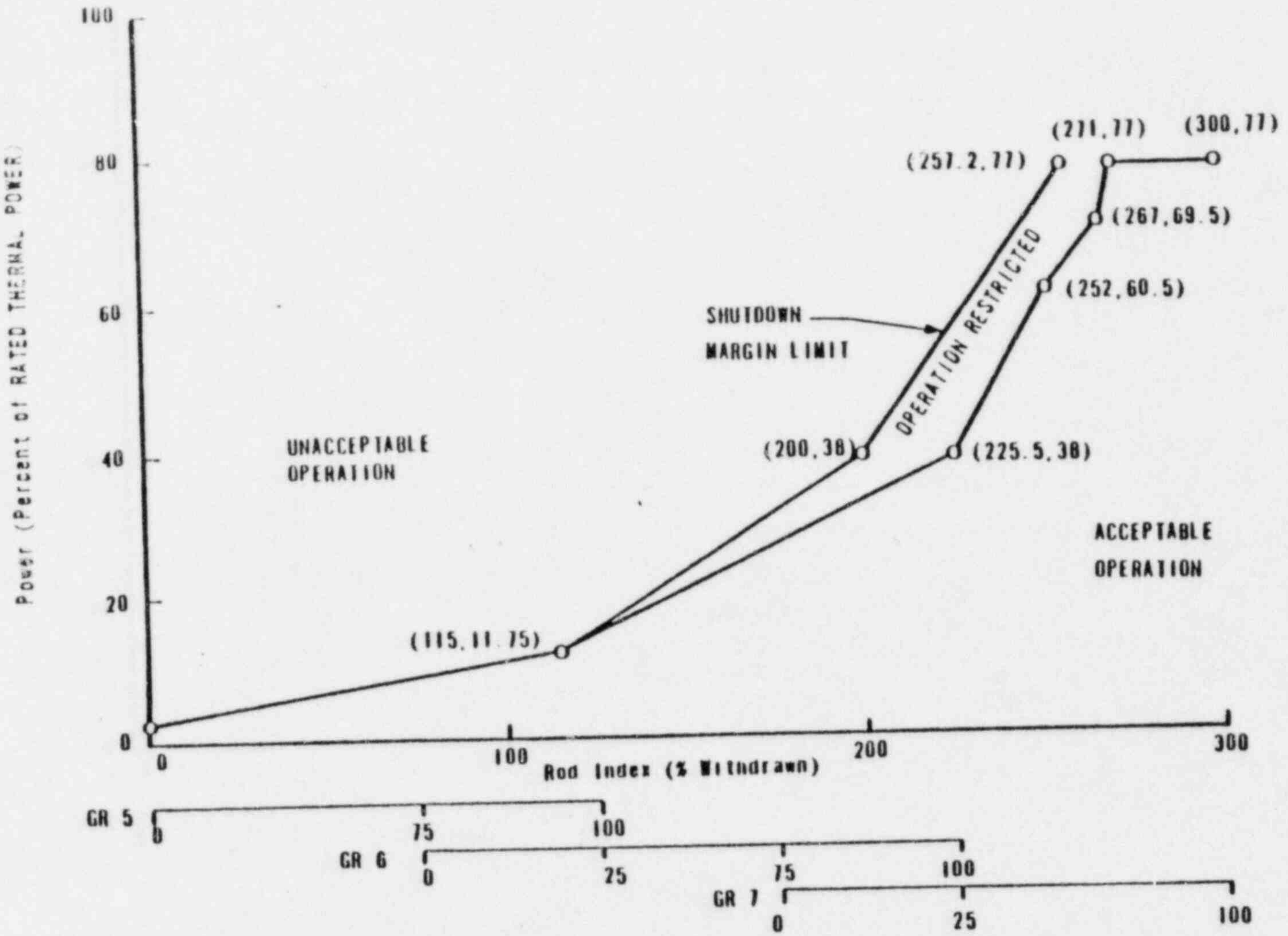
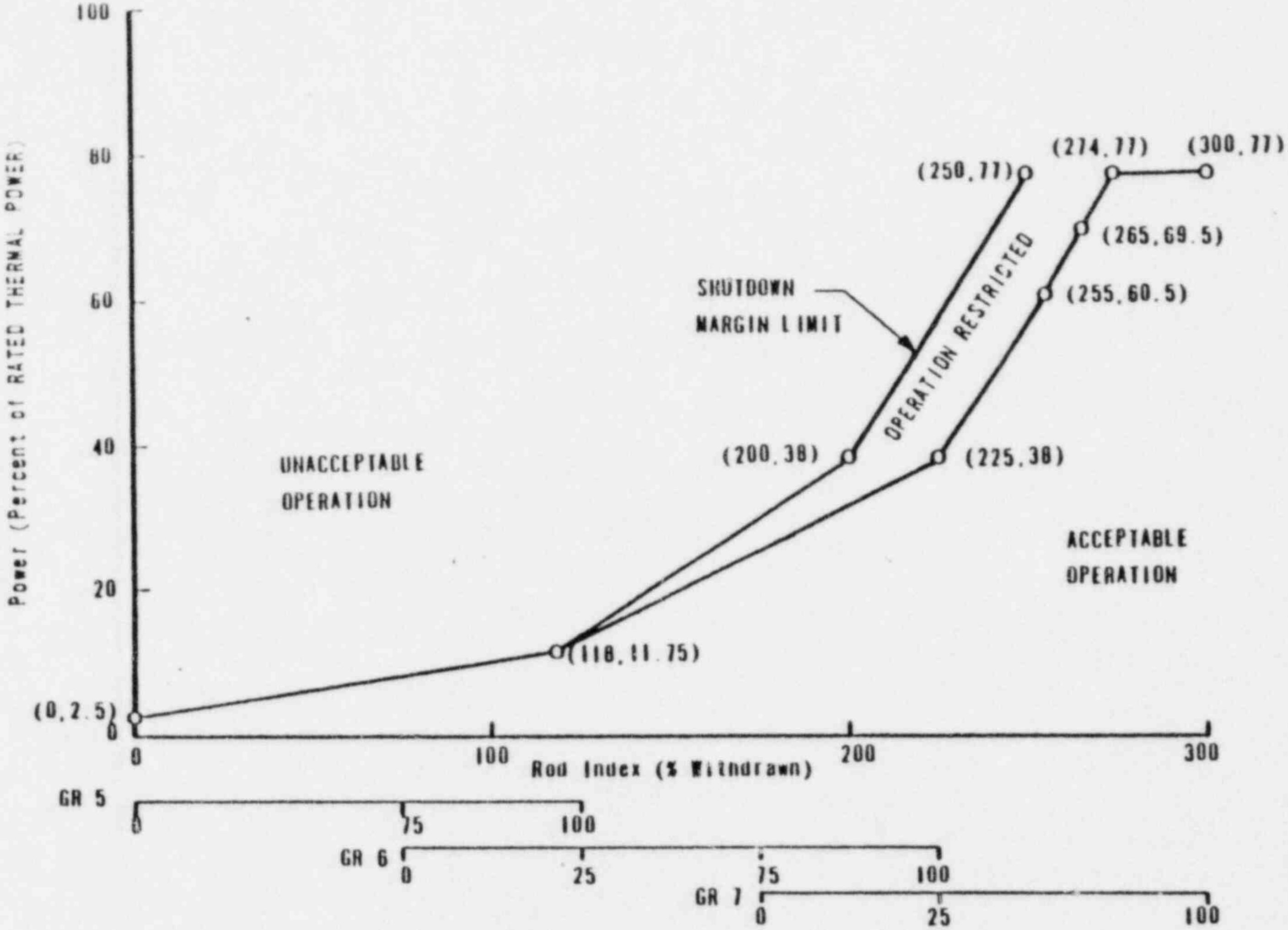


Figure 3.1-3e Regulating Group Position Limits, 200 ± 10 to 268 ± 10 EFPD
Three RC Pumps, APSRs Withdrawn — Davis-Besse 1, Cycle 3



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 3.1-4.

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.

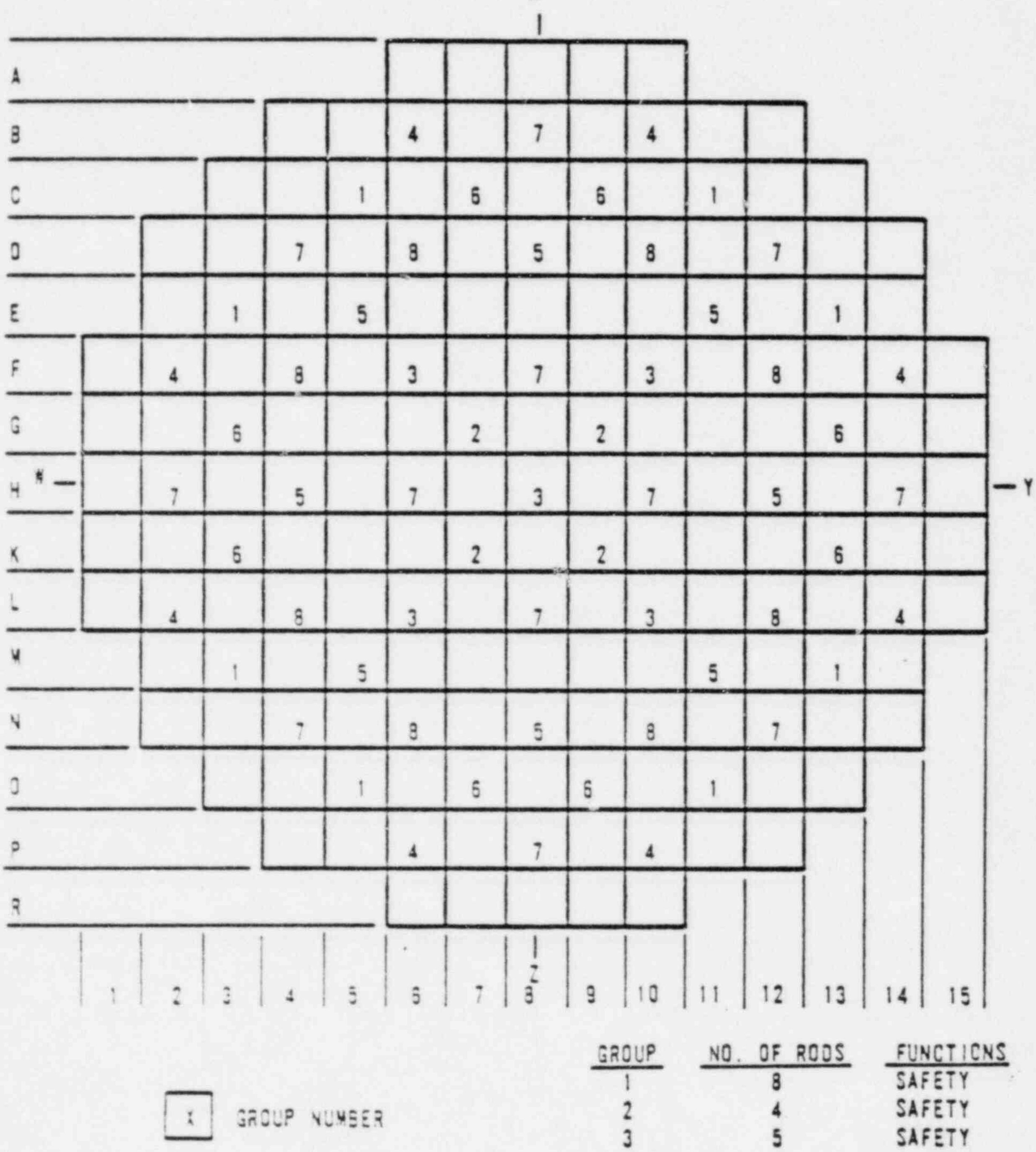
   b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

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.

Figure 3.1-4 Control Rod Core Locations and Group Assignments — Davis-Besse 1, Cycle 3



GROUP	NO. OF RODS	FUNCTIONS
1	8	SAFETY
2	4	SAFETY
3	5	SAFETY
4	8	SAFETY
5	8	CONTROL
6	8	CONTROL
7	12	CONTROL
8	8	APSRs
TOTAL #	61	

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XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in Figure 3.1-2 unless one of the following conditions is satisfied:

a. Xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability, or

b. THERMAL POWER has been within a range of 87 to 92 percent of RATED THERMAL POWER for a period exceeding 2 hours in the soluble poison control mode, excluding xenon free start-ups.

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 or it shall be determined that the THERMAL POWER has been in the range of 87 to 92% of RATED THERMAL POWER for $\geq$ 2 hours, 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 3.1-5a, -5b, -5c, -5f, -5g and -5h for the first 200 ±10 EFPD of operation. If this rod group is completely withdrawn at 200 ±10 EFPD for extension of cycle length, it shall not be reinserted in the core for remainder of the cycle and the limits of Figure 3.1-5e shall be applicable. However, if the rod group is not completely withdrawn at 200 ±10 EFPD, the group shall be limited in physical insertion as shown on Figures 3.1-5d and -5i for the remainder of the cycle.

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 figures 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 except when the axial power shaping rod insertion limit alarm is inoperable, then verify the group to be within the insertion limit at least once every 4 hours.

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

Figure 3.1-5a APSR Position Limits, 0 to 60 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

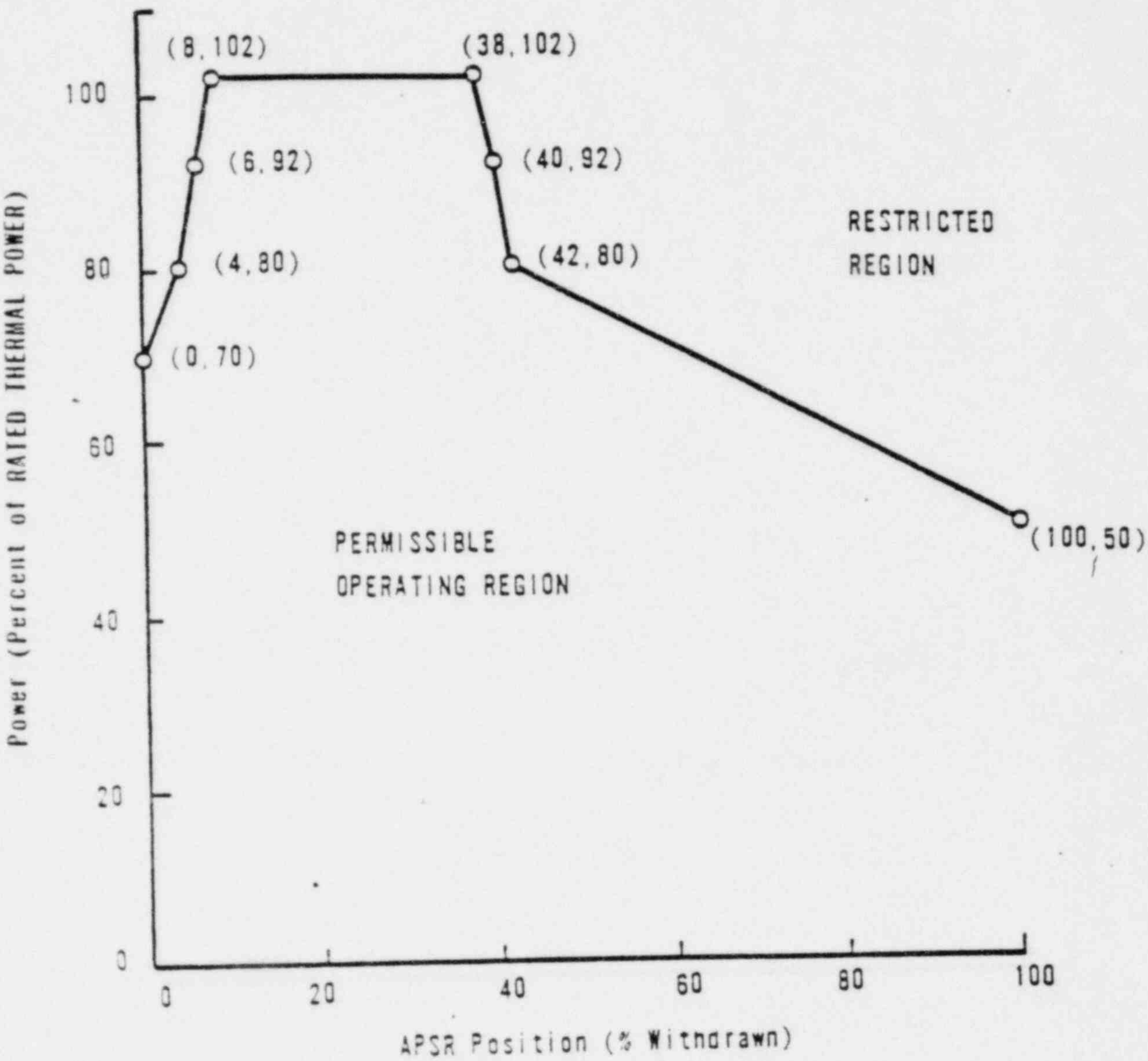


Figure 3.1-5b APSR Position Limits, 50 to 150 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

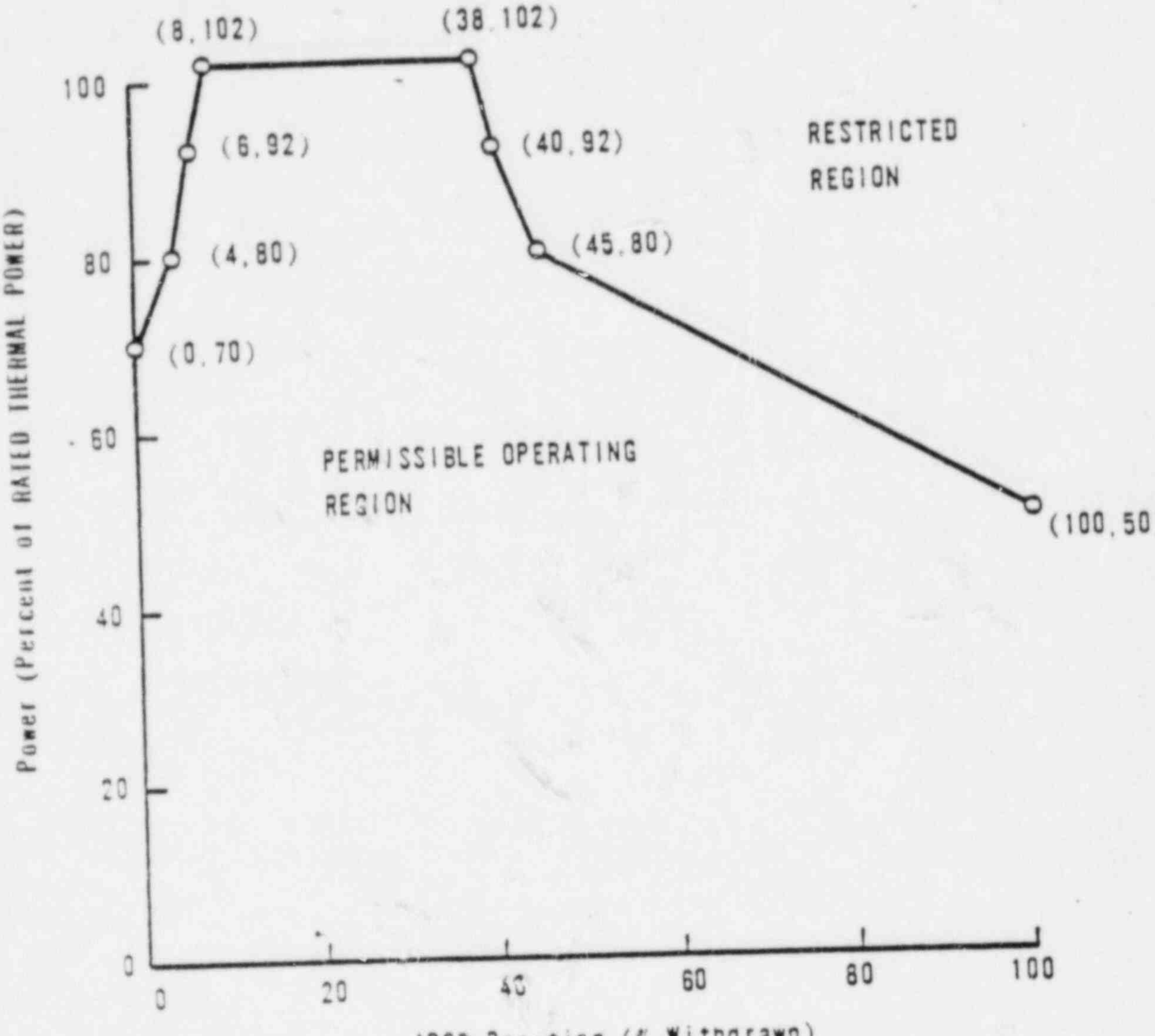


Figure 3.1-5c APSR Position Limits, 150 ± 10 to 200 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

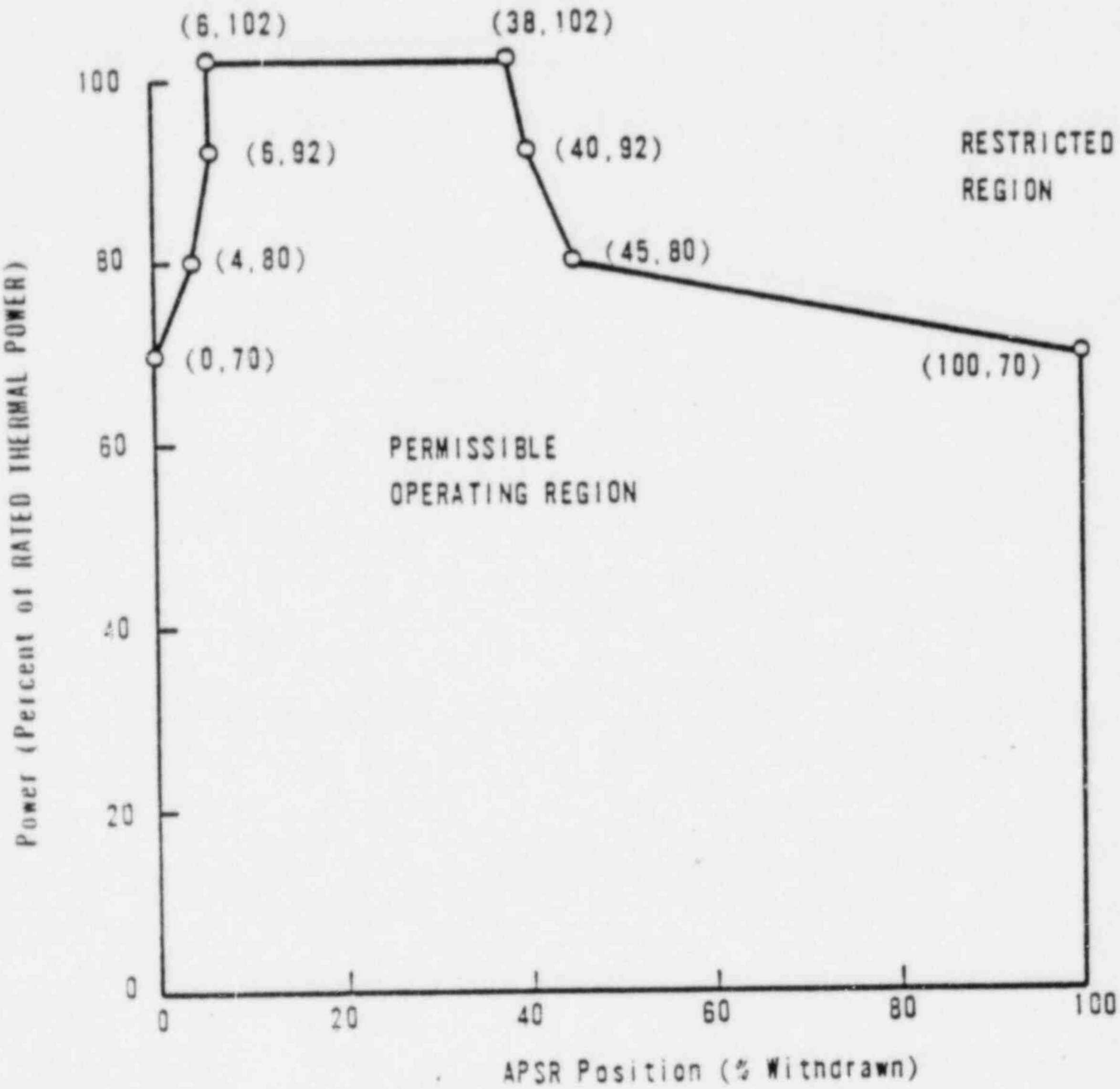


Figure 3.1-5d APSR Position Limits, 200 ± 10 to 230 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

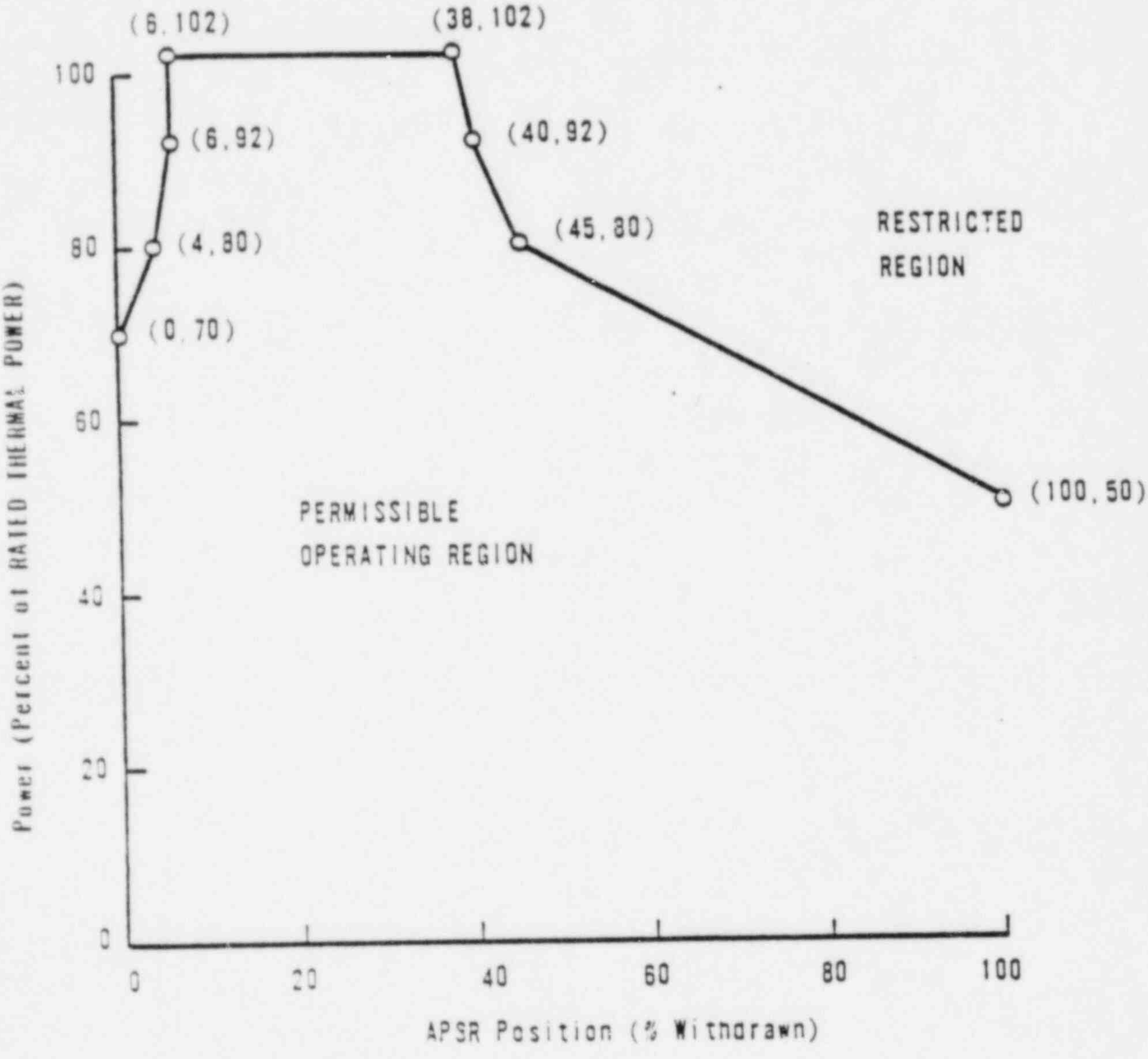


Figure 3.1-5e APSR Position Limits, 200 ± 10 to 268 ± 10 EFPD, Three or Four RC Pumps, APSRs Withdrawn — Davis-Besse 1, Cycle 3

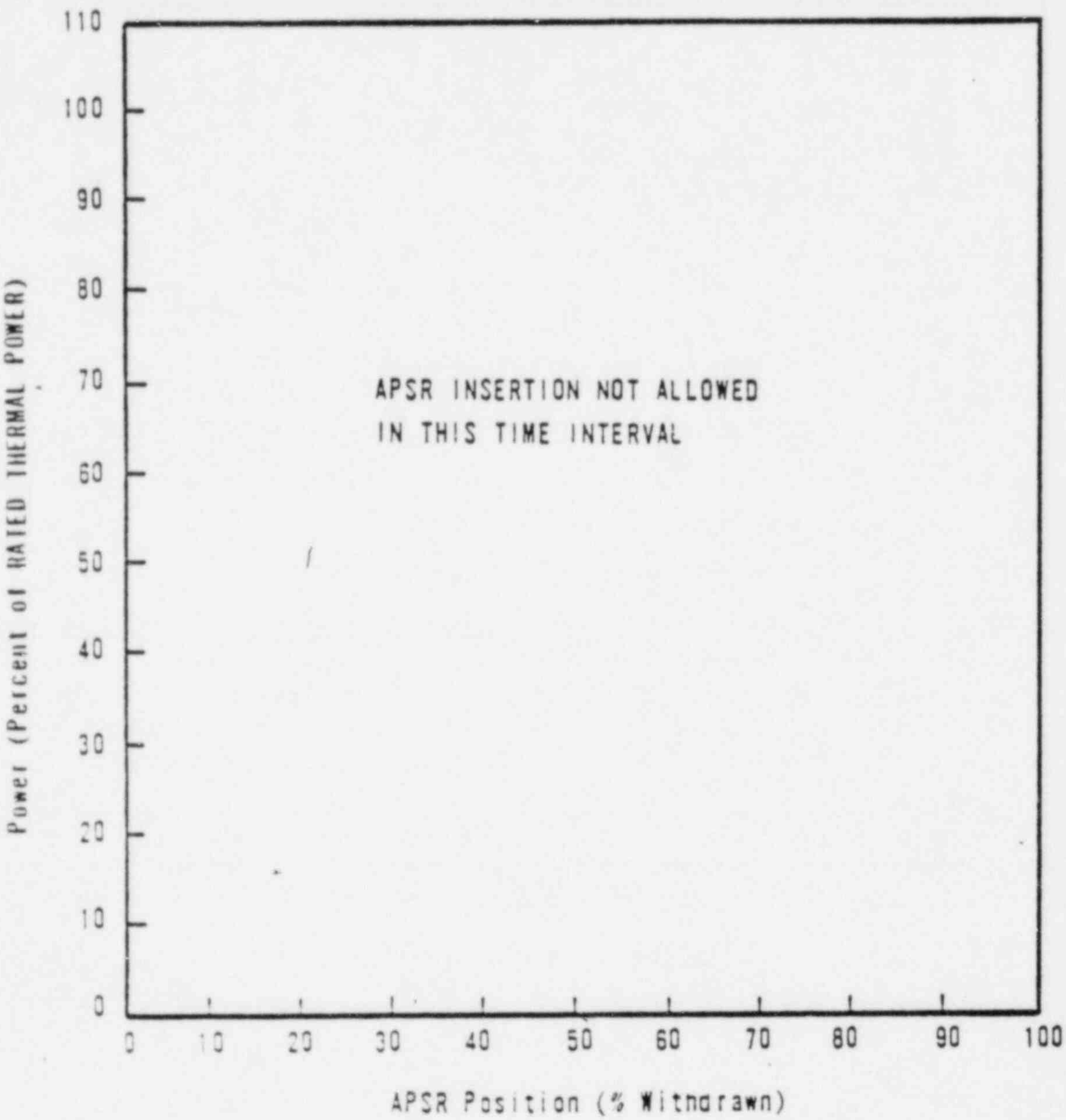


Figure 3.1-5f APSR Position Limits, 0 to 60 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

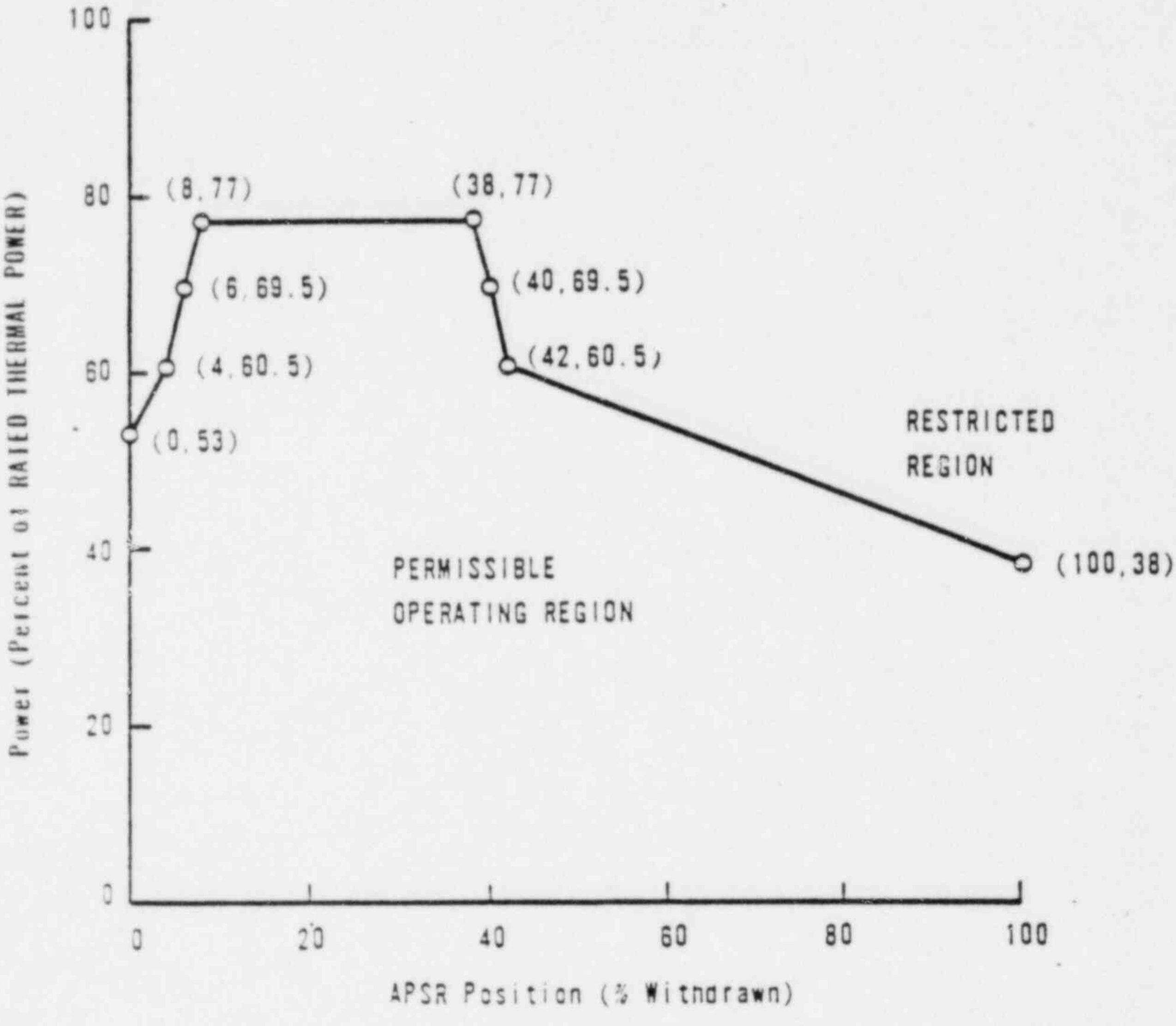


Figure 3.1-5g APSR Position Limits, 50 to 150 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

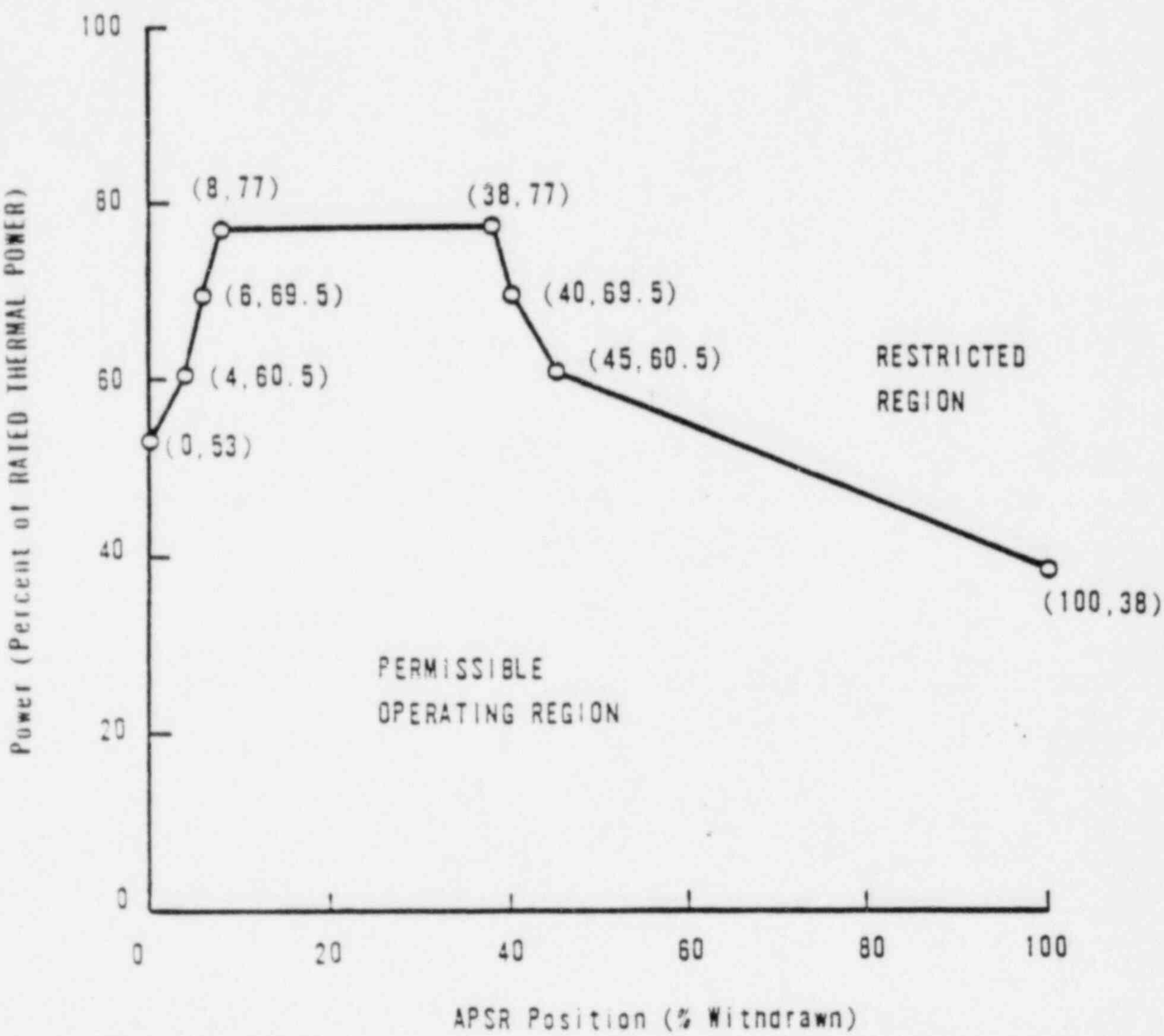


Figure 3.1-5h APSR Position Limits, 150 ± 10 to 200 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

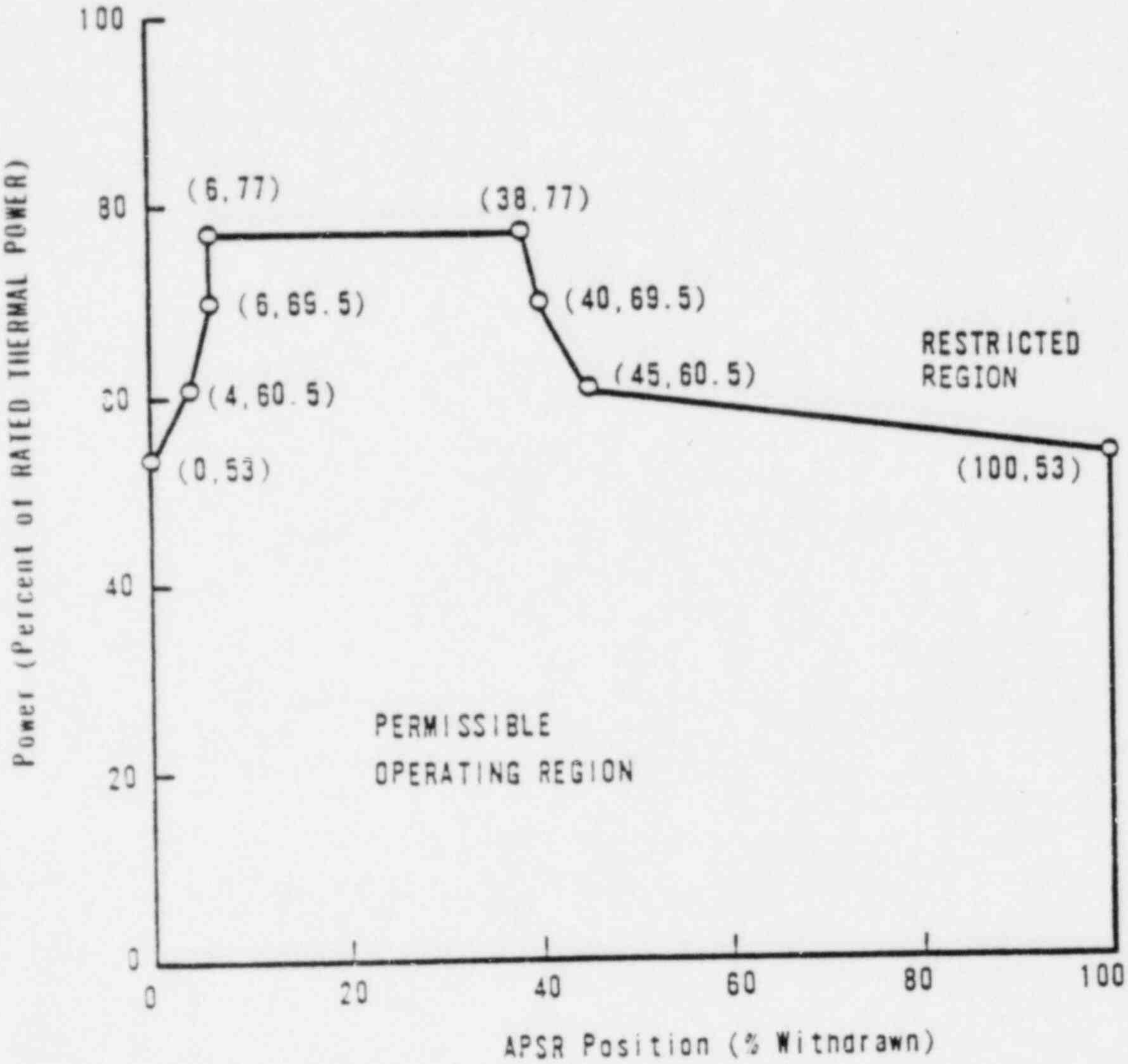
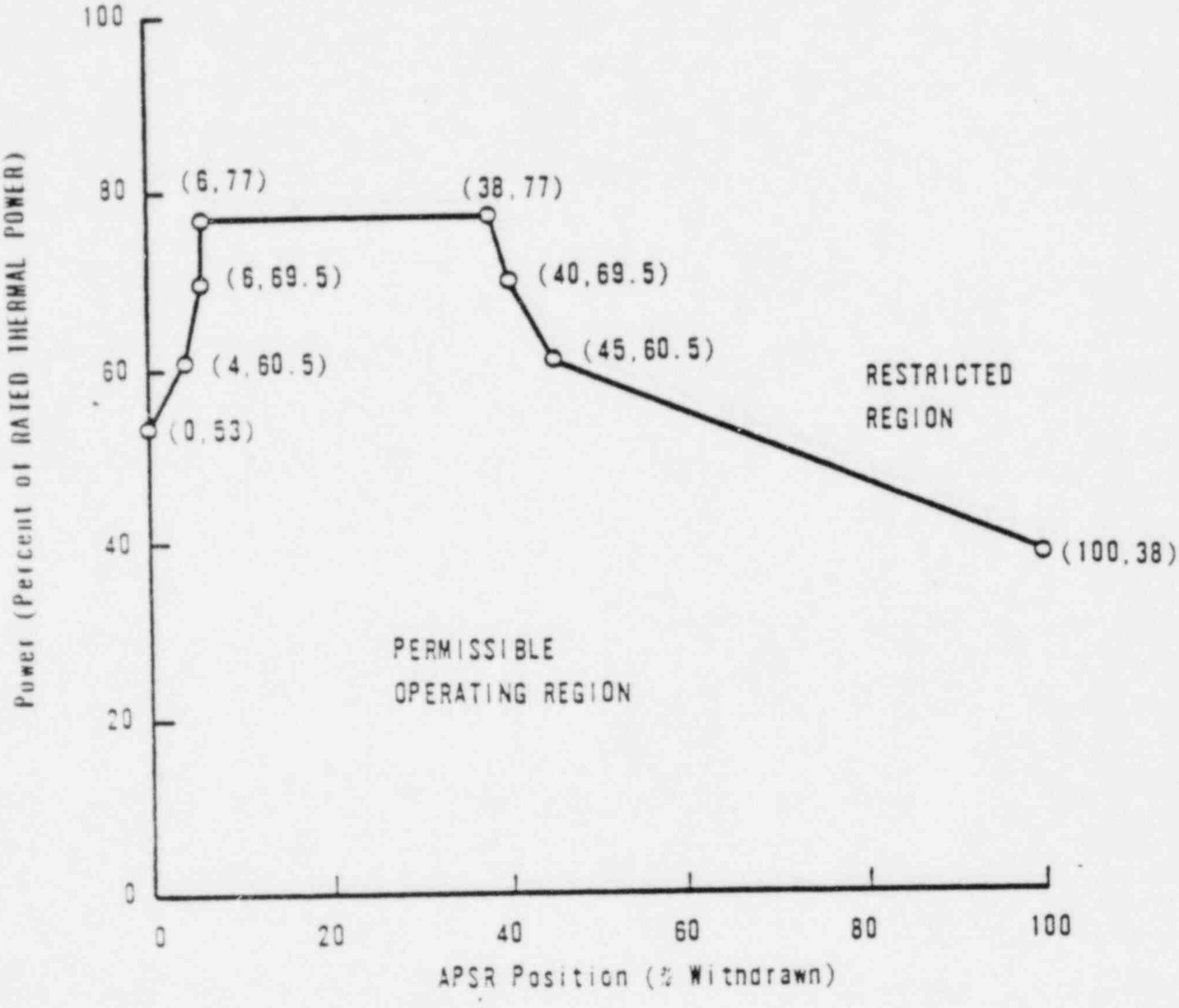


Figure 3.1-5i APSR Position Limits, 200 ± 10 to 230 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3



3/4.2. POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1a, -1b and -1c and 3.2-2a, -2b and -2c for the first 200 ±10 EFPD of operation. If the axial power shaping rods are completely withdrawn at 200 ±10 EFPD for extension of cycle length, then the AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1e and 3.2-2e for the remainder of the cycle. However, if the axial power shaping rods are not completely withdrawn at 200 ±10 EFPD, then the AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1d and 3.2-2d for the remainder of the cycle.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

b. Within one hour reduce power until imbalance limits are met or to 40% of RATED THERMAL POWER or less.

SURVEILLANCE REQUIREMENTS

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within limits at least once every 12 hours when above 40% of RATED THERMAL POWER except when the AXIAL POWER IMBALANCE alarm is inoperable, then calculate the AXIAL POWER IMBALANCE at least once per hour.

*See Special Test Exception 3.10.1.

Figure 3.2-1a Axial Power Imbalance Limits, 0 to 60 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

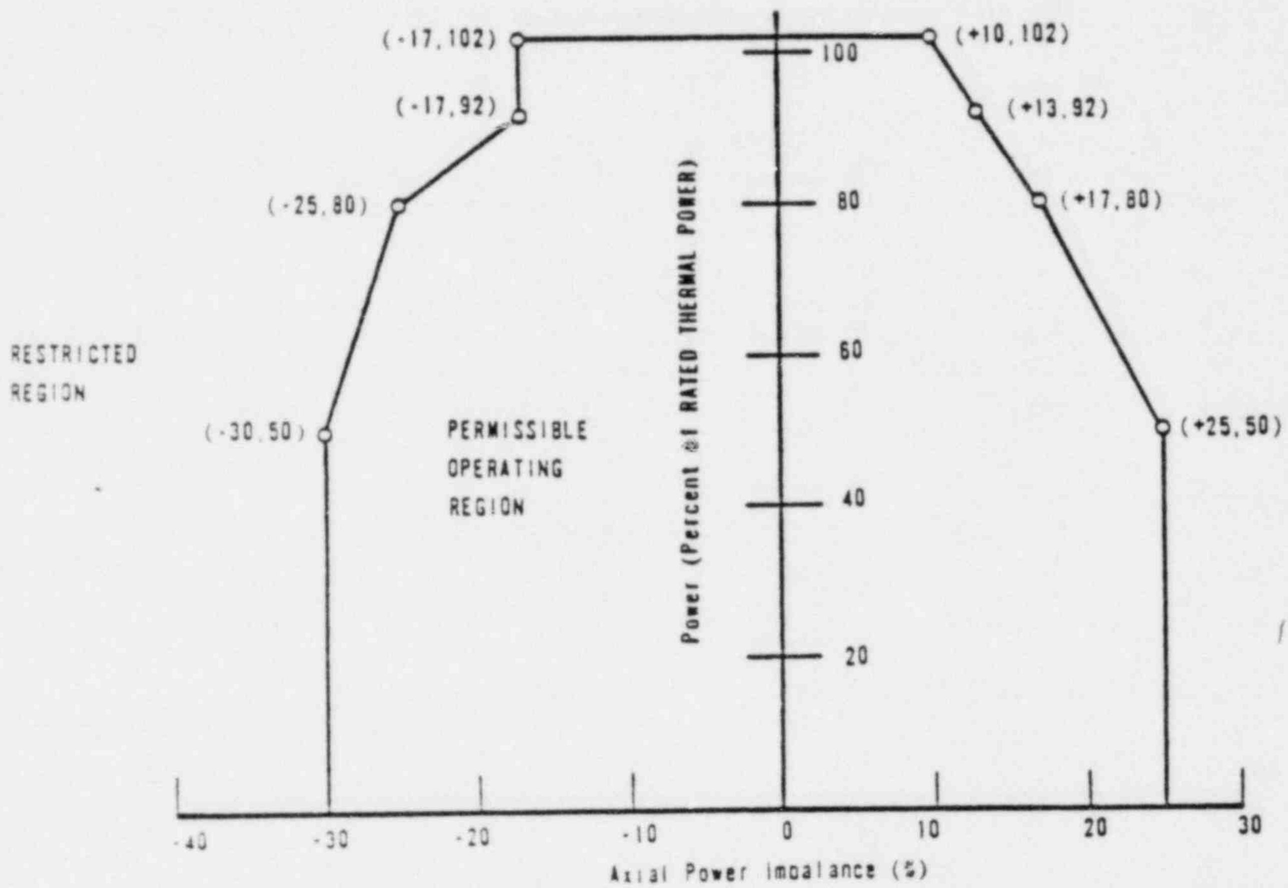


Figure 3.2-1b Axial Power Imbalance Limits, 50 to 150 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

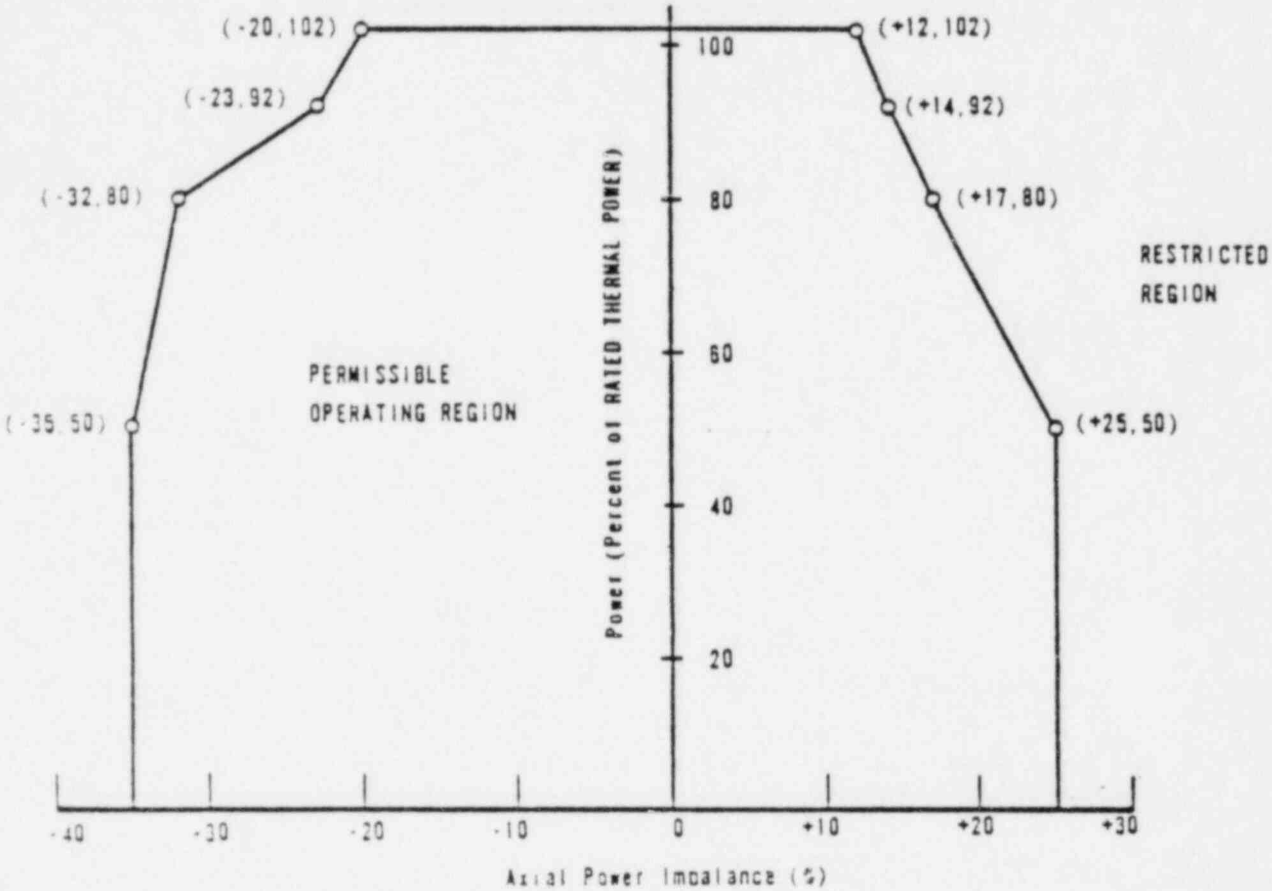


Figure 3.2-1c Axial Power Imbalance Limits, 150 ± 10 to 200 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

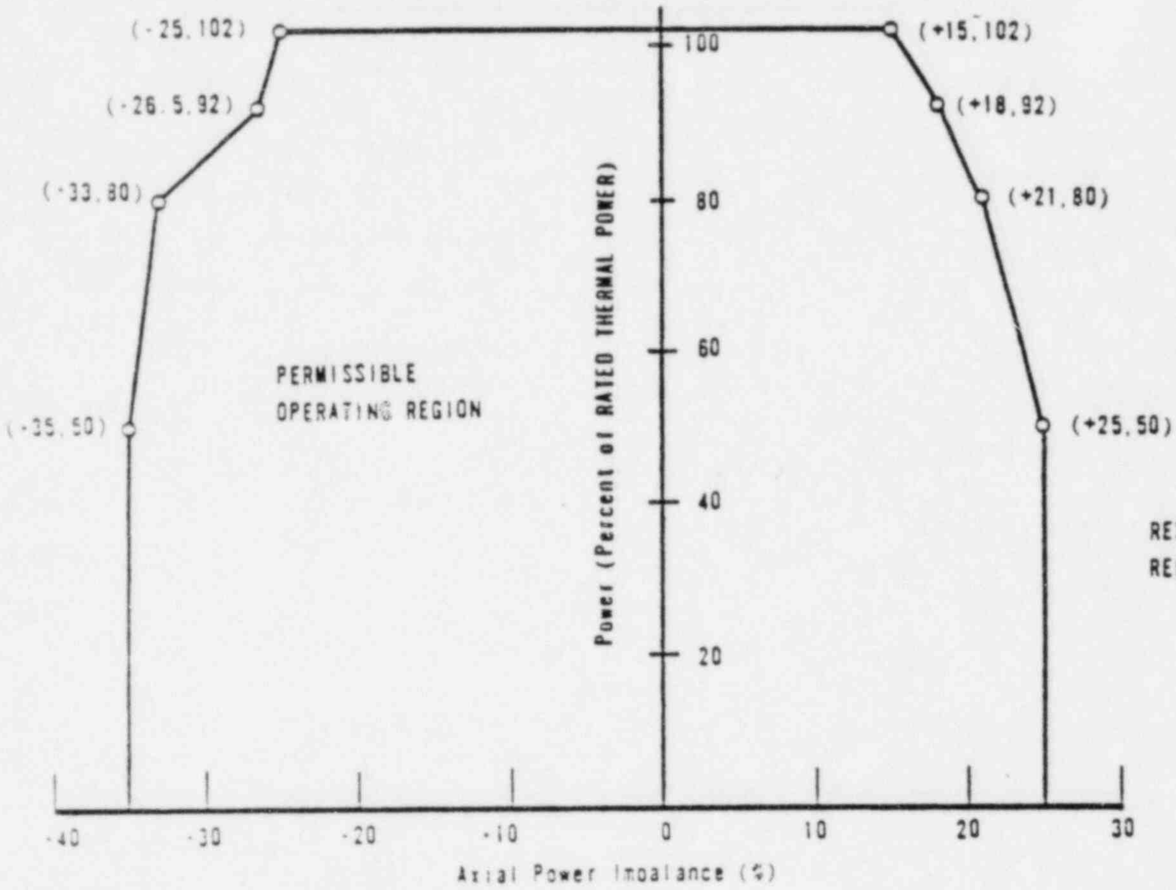


Figure 3.2-1d Axial Power Imbalance Limits, 200 ± 10 to 230 ± 10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 3

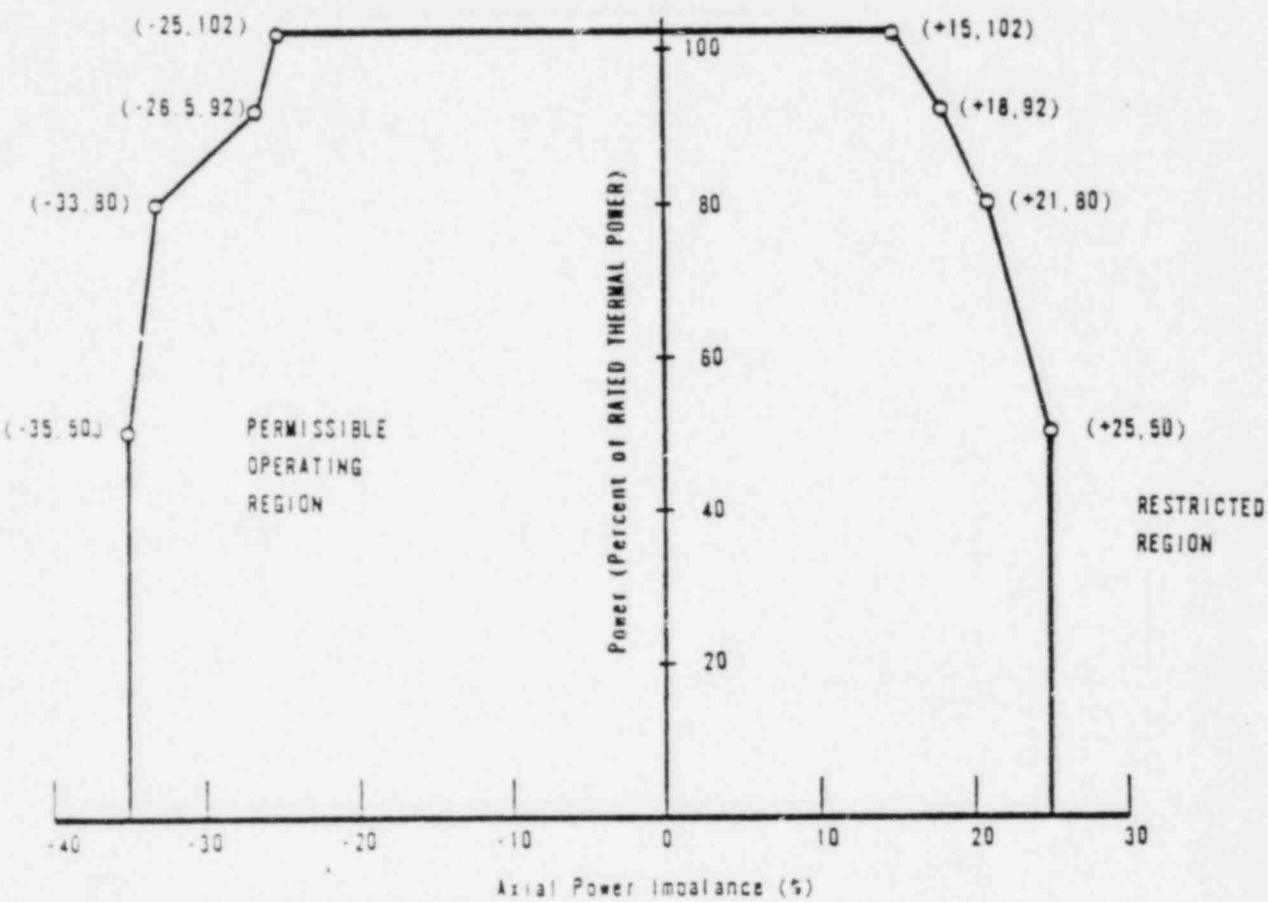


Figure 3.2-1e Axial Power Imbalance Limits, 200 ± 10 to 268 ± 10 EFPD, Four RC Pumps, APSRs Withdrawn — Davis-Besse 1, Cycle 3

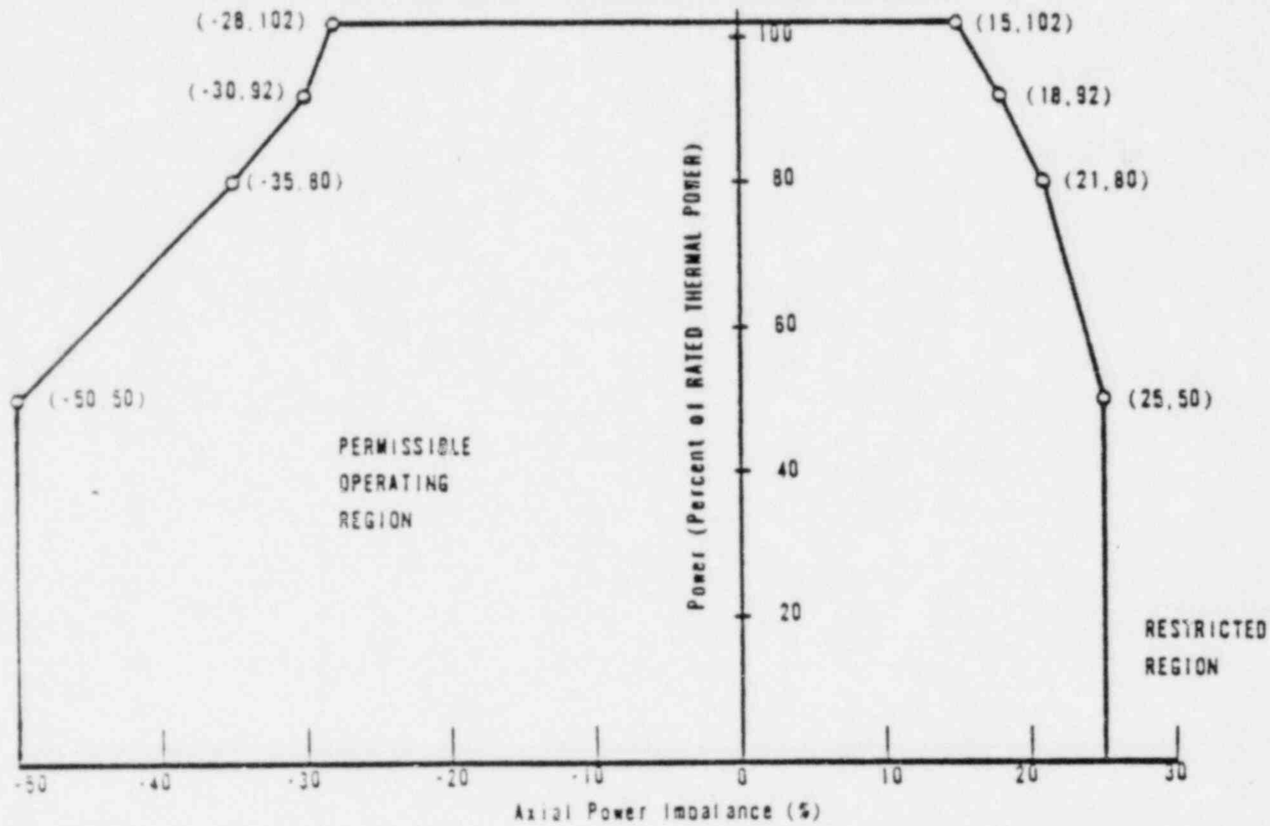


Figure 3.2-2a Axial Power Imbalance Limits, 0 to 60 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

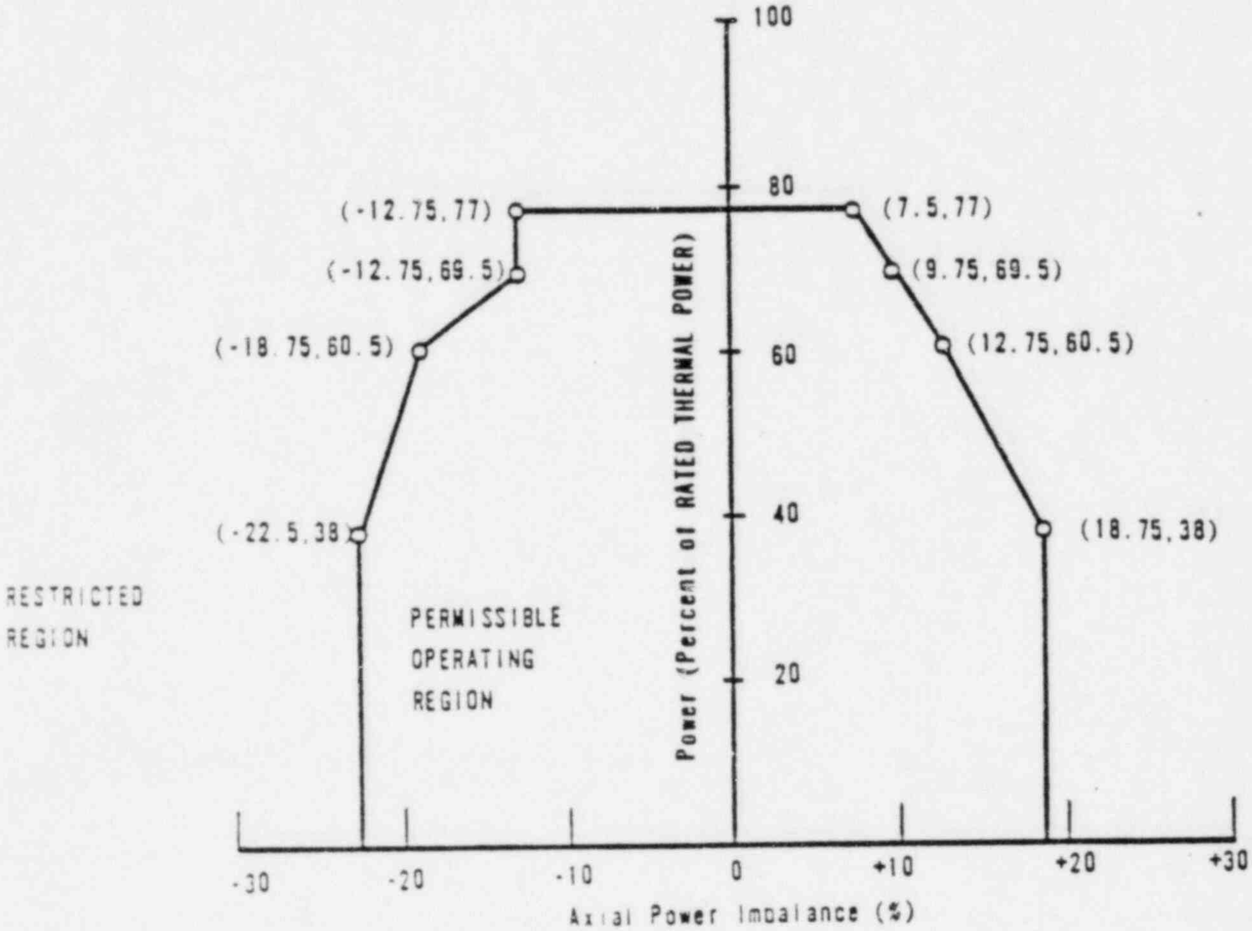


Figure 3.2-2b Axial Power Imbalance Limits, 50 to 150 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

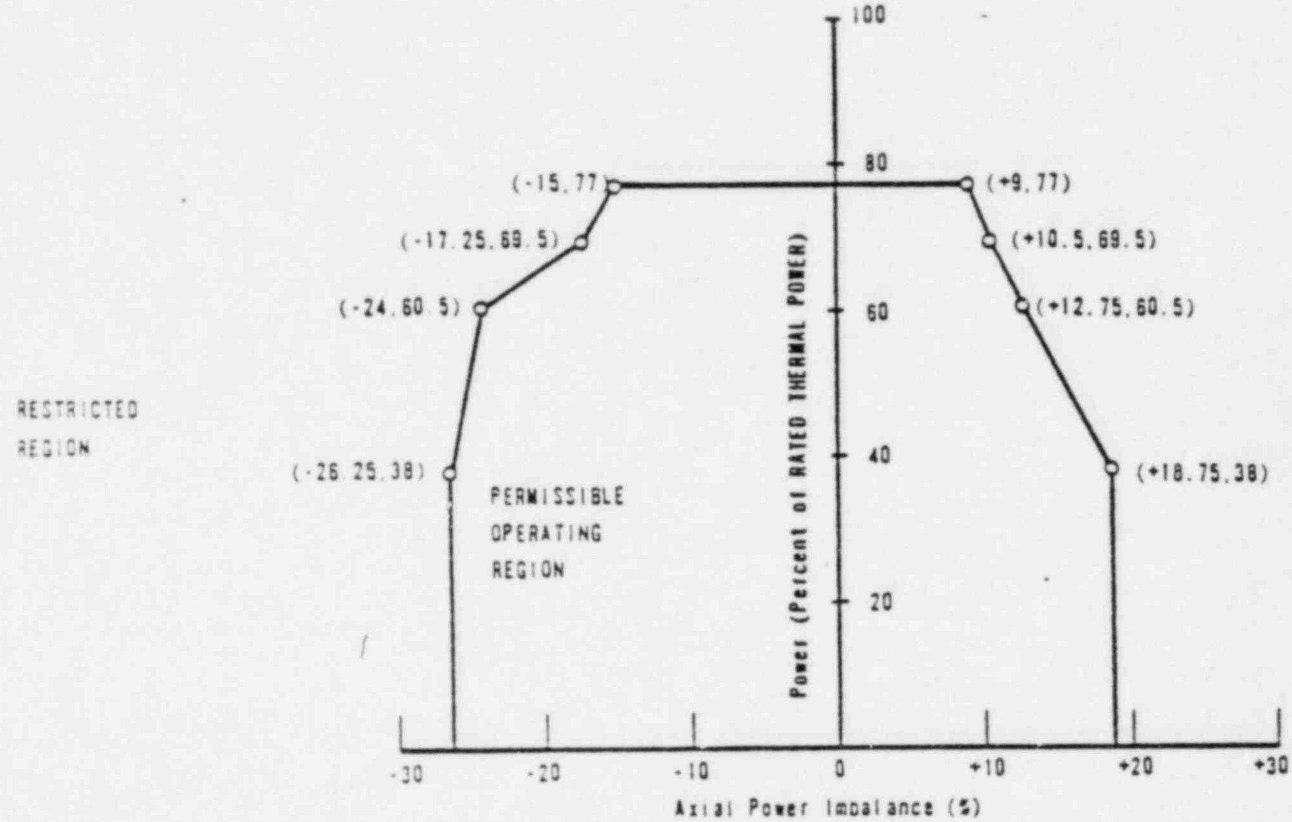


Figure 3.2-2c Axial Power Imbalance Limits, 150 ± 10 to 200 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

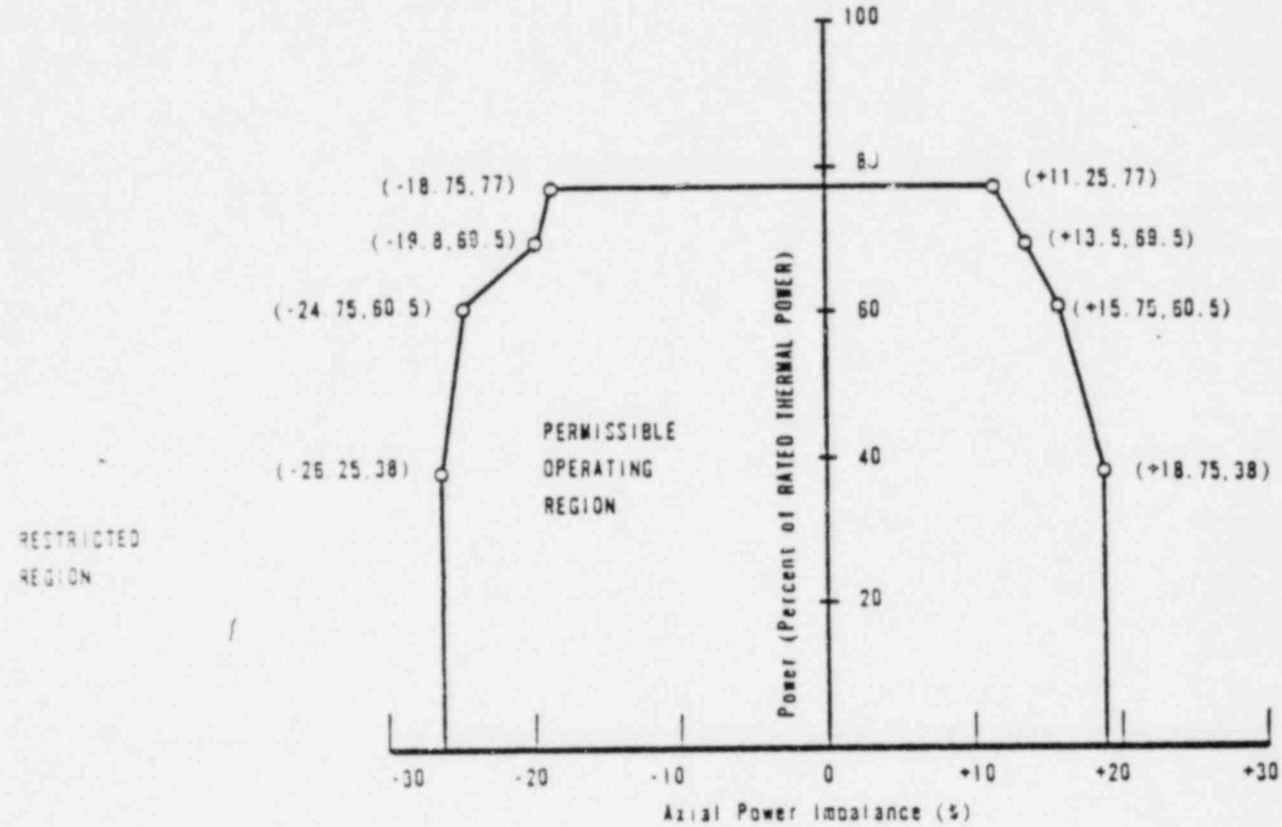


Figure 3.2-2d Axial Power Imbalance Limits, 200 ± 10 to 230 ± 10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 3

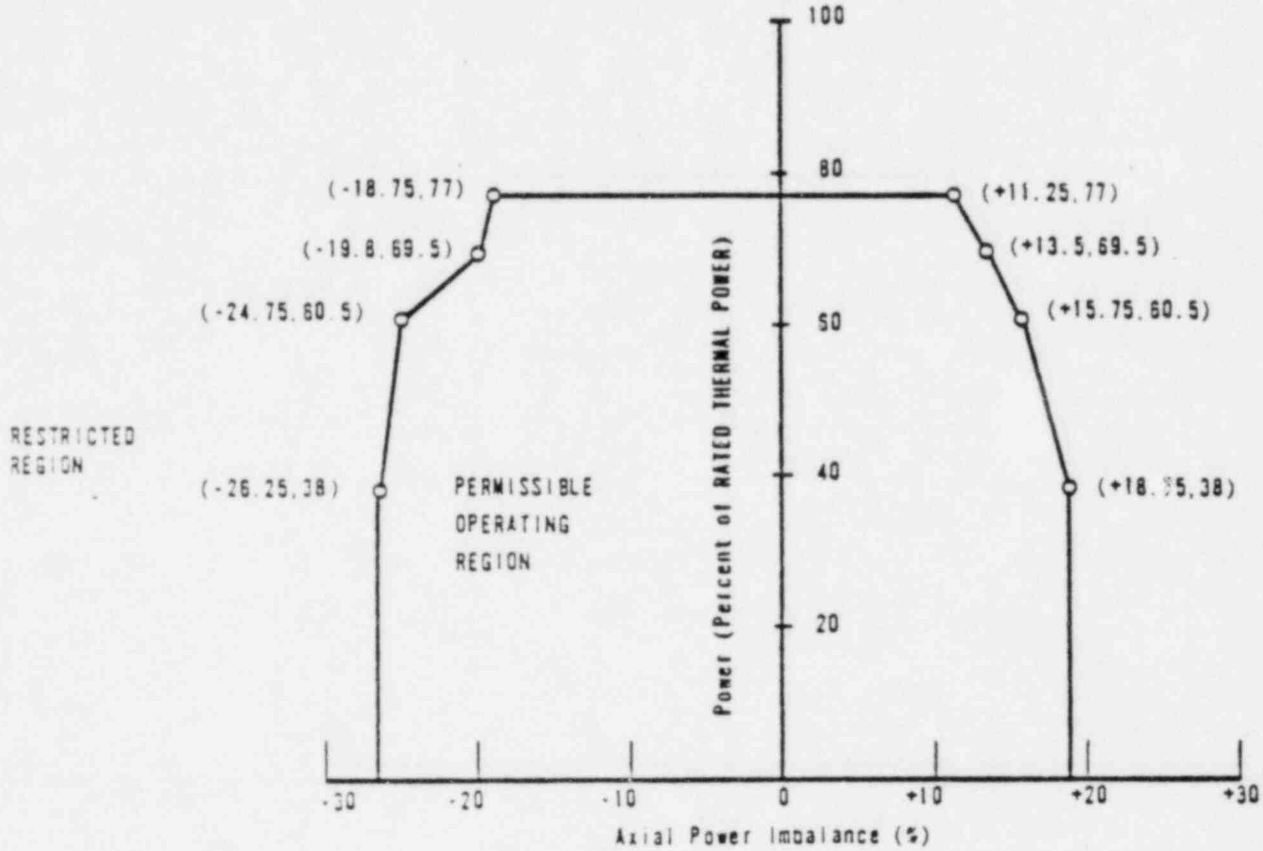
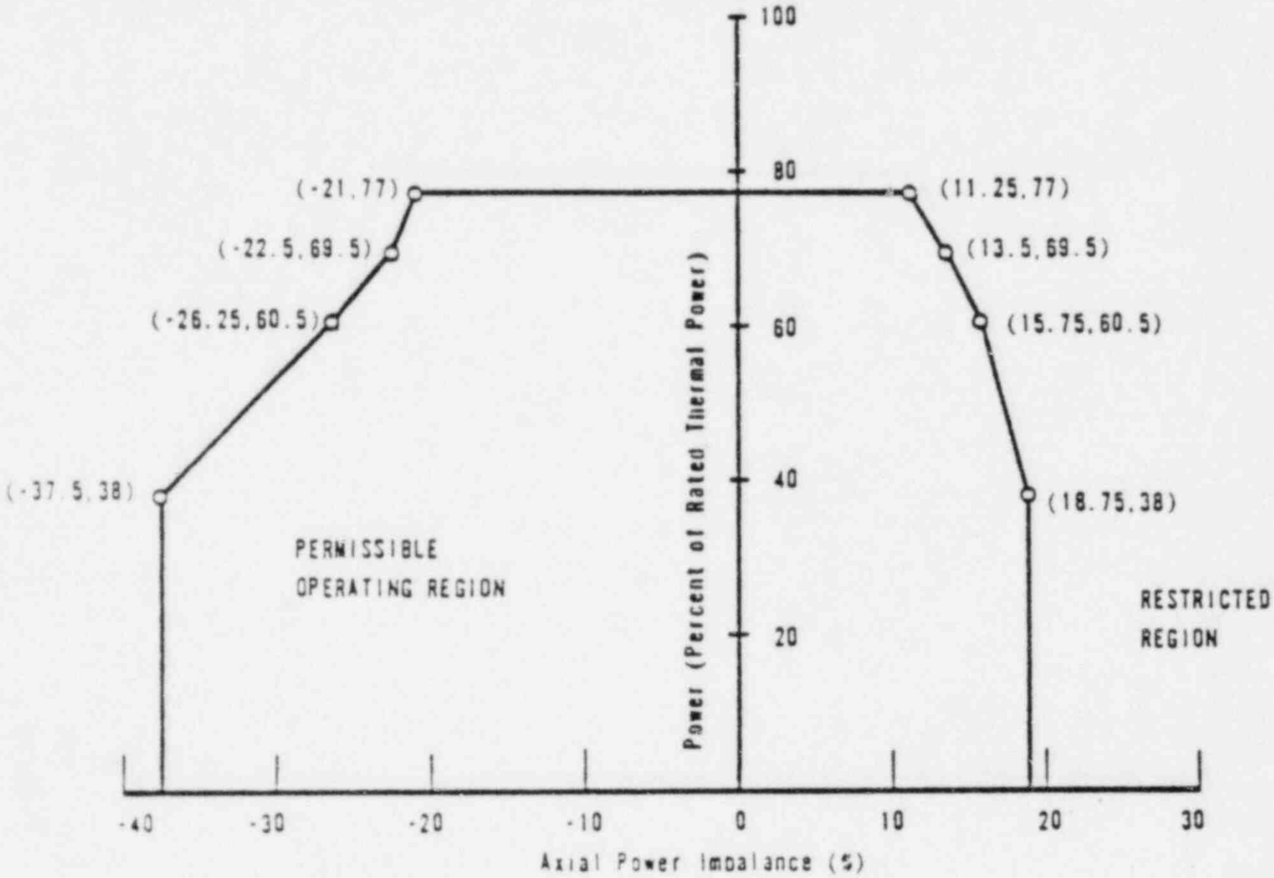


Figure 3.2-2e Axial Power Imbalance Limits, 200 ± 10 to 268 ± 10 EFPD, Three RC Pumps, APSRs Withdrawn — Davis-Besse 1, Cycle 3



NUCLEAR HEAT FLUX HOT CHANNEL FACTOR — F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{2.93}{P}$$

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 high flux trip setpoint and flux-Δ flux-flow trip setpoint within 4 hours.

b. Demonstrate through incore 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 incore 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 incore detectors to obtain a power distribution map:

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to initial operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured F_Q of 4.2.2.1 above, shall be increased by 1.4% to account for manufacturing tolerances and further increased by 7.5% to account for measurement uncertainty.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit of Table 3.2-2, reduce THERMAL POWER to $\leq$ 15% of RATED THERMAL POWER within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT shall be determined to be within the limits at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT alarm is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

Table 3.2-2. Quadrant Power Tilt Limits

	Steady state limit	Transient limit	Maximum limit
Measurement Independent QUADRANT POWER TILT	4.92	11.07	20.0
QUADRANT POWER TILT as measured by:			
Symmetrical Incore Detector System	3.03	8.53	20.0
Power Range Channels	1.96	6.96	20.0
Minimum Incore Detector System	1.90	4.40	20.0

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. $\leq 10^{-10}$ amps on the Intermediate Range (IR) instrumentation, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10^{-10} amps on the IR instrumentation.

b. $> 10^{-10}$ amps on the IR instrumentation, operation may continue.

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARCIN requirements of Specification 3.1.1.1 within one hour and at least once per 12 hours thereafter.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. Within 1 hour:

1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperative channel.

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1, and the inoperable channel above may be bypassed for to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIMES** (seconds)
1. Manual Reactor Trip		Not Applicable
2. High Flux*		≤ 0.266
3. RC High Temperature		Not Applicable
4. Flux - Δ Flux - Flow*	- Variable Flow	≤ 1.77
	- Constant Flow	≤ 0.266
5. RC Low Pressure		≤ 0.341
6. RC High Pressure		≤ 0.341
7. RC Pressure - Temperature - Constant Temperature		Not Applicable
8. High Flux/Number of Reactor Coolant Pumps On*		≤ 0.451***
9. Containment High Pressure		Not Applicable

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

** Including sensor (except as noted), RPS instrument delay and the breaker delay.

*** A 0.24 sec delay time has been assumed for pump contact monitor.

3/4.4. REACTOR COOLANT SYSTEM

3/4.4.1. COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 79.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

    1. High Flux
    2. Flux-ΔFlux-Flow

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2 The reactor protective instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

*See Special Test Exception 3.10.3.

SHUTDOWN AND HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator,

2. Reactor Coolant Loop 2 and its associated steam generator,

3. Decay Heat Removal Loop 1,*

4. Decay Heat Removal Loop 2.*

b. At least one of the above coolant loops shall be in operation.**

c. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operator, or manual valves DH-21 and DH-23 are opened.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

APPLICABILITY: MODES 3, 4 and 5

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible, or be in COLD SHUTDOWN within 20 hours.

b. With none of the above required coolant loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*The normal or emergency power source may be inoperable in MODE 5.

**The decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature. The SHUTDOWN MARGIN required is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2800 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2800 GPM will circulate an equivalent Reactor Coolant System volume of 12,110 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the reactor coolant system average temperature less than 525°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2. BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) makeup or DHR pumps, (3) separate flow paths, (4) boric acid pumps, (5) associated heat tracing systems, and (6) an emergency power supply from operable emergency busses.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs from full power equilibrium xenon conditions and requires the equivalent of either 7373 gallons of 8742 ppm borated water from the boric acid storage tanks or 52,726 gallons of 1800 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 434,650 gallons of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4; therefore, the larger volume of borated water is specified.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the

3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 8603 gallons of 8742 ppm borated water from the boric acid storage system or 28,200 gallons of 1800 ppm borated water from the borated water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution recirculated within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.3. MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. Measurement with $T_{avg} \geq 525°F$ and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

Technical Specification 3.1.3.8 provides the ability to prevent excessive power peaking by transient xenon at RATED THERMAL POWER. Operating restrictions resulting from transient xenon power peaking, including xenon-free startup, are inherently included in the limits of Sections 3.1.3.6 (Regulating Rod Insertion Limits), 3.1.3.9 (Axial Power Shaping Rod Insertion Limits), and 3.2.1 (Axial Power Imbalance) for transient peaking behavior bounded by the following factors. For the period of cycle operation where regulating rod groups 6 and 7 are allowed to be inserted at RATED THERMAL POWER, an 8% peaking increase is applied at or above 92% FP. An 18% increase is applied below 92% FP. For operation where only regulating rod group 7 is allowed to be inserted at RATED THERMAL POWER, a 5% peaking increase is applied at or above 92% FP and a 13% increase is applied below 92% FP.

If these values, checked every cycle, conservatively bound the peaking effects of all transient xenon, then the need for any hold at a power level cutoff below RATED THERMAL POWER is precluded. If not, either the power level at which the requirements of Section 3.1.3.8 must be satisfied or the above-listed factors will be suitably adjusted to preserve the LOCA linear heat rate limits.

The limitation on axial power shaping rod insertion is necessary to ensure that power peaking limits are not exceeded.

3/4.2. POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (normal operation) and II (incidents of moderate frequency) events by: (a) maintaining the minimum DNBR in the core ≥ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density ≤ 18.4 kW/ft during normal operation, and (c) maintaining the peak power density less than the limits given in the bases to specification 2.1 during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power imbalance envelope defined in Figures 3.2-1 and 3.2-2 and the insertion limit curves, Figures 3.1-2 and 3.1-3 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. 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 insertion limits, as defined by Figures 3.1-2 and 3.1-3 and if the steady-state limit QUADRANT POWER TILT exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing olerance factors.

e. Potential fuel rod bow effects.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear heat flux hot channel factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F^N_{\Delta H}$ 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:

$$F_Q \leq 2.94; \quad F^N_{\Delta H} \leq 1.71$$

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 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 between the limits specified in Specification 3.2.1.

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.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding the values given in the bases to specification 2.1 locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, AXIAL POWER IMBALANCE pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the reactor protection system.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

3/4.2.5. DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate using delta P instrumentation is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.