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Docket No. 50-346

Mr. Richard P. Crouse
Vice President - Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

SUBJECT: AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-3; CYCLE 4 OPERATION

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Appendix A Technical Specifications (TSs) in response to your application dated July 5, 1983 (No. 965).

This amendment modifies the TSs to permit operation for Cycle 4. This cycle has a design length of 240 effective full power days. The modified TSs also incorporate revised Reactor Protection System instrumentation trip setpoints and allowable values. In addition, this amendment corrects a typographical error on Bases page B 3/4 1-2.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"
John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 61 to NPF-3
2. Safety Evaluation

cc w/enclosures:
See next page

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No legal objection

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	RIngram	ADe Agazio;cf	JStolz	GLainas	M. Rothschild		
DATE	9/16/83	9/16/83	9/16/83	9/20/83	9/19/83		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Toledo Edison Company

cc w/enclosure(s):

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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. 61
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 July 5, 1983, 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.

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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. 61 , 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

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 61

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 1-36
2-3	3/4 1-37
2-5	3/4 1-38
2-7	3/4 1-39
B2-2	3/4 1-40
B2-4	3/4 1-41
B2-5	3/4 1-42
B2-6	3/4 1-43
3/4 1-26	3/4 2-1
3/4 1-28	3/4 2-2
3/4 1-28a	3/4 2-2a
3/4 1-28b	3/4 2-2b
3/4 1-28c	3/4 2-2c
3/4 1-28d	3/4 2-2d
3/4 1-29	3/4 2-3
3/4 1-29a	3/4 2-3a
3/4 1-29b	3/4 2-3b
3/4 1-29c	3/4 2-3c
3/4 1-29d	3/4 2-3d
3/4 1-31	B3/4 1-2
3/4 1-34	B3/4 2-2
3/4 1-35	

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

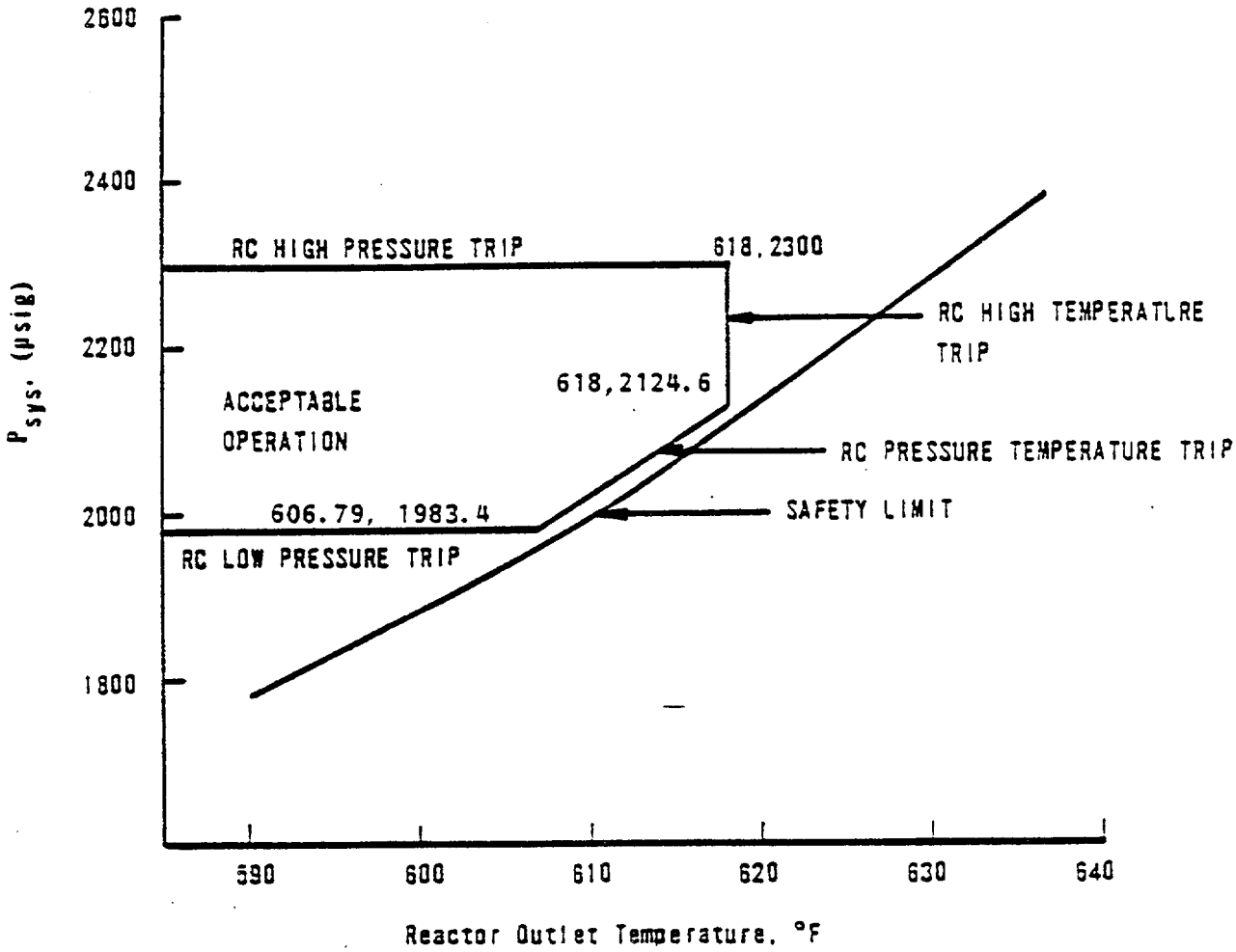
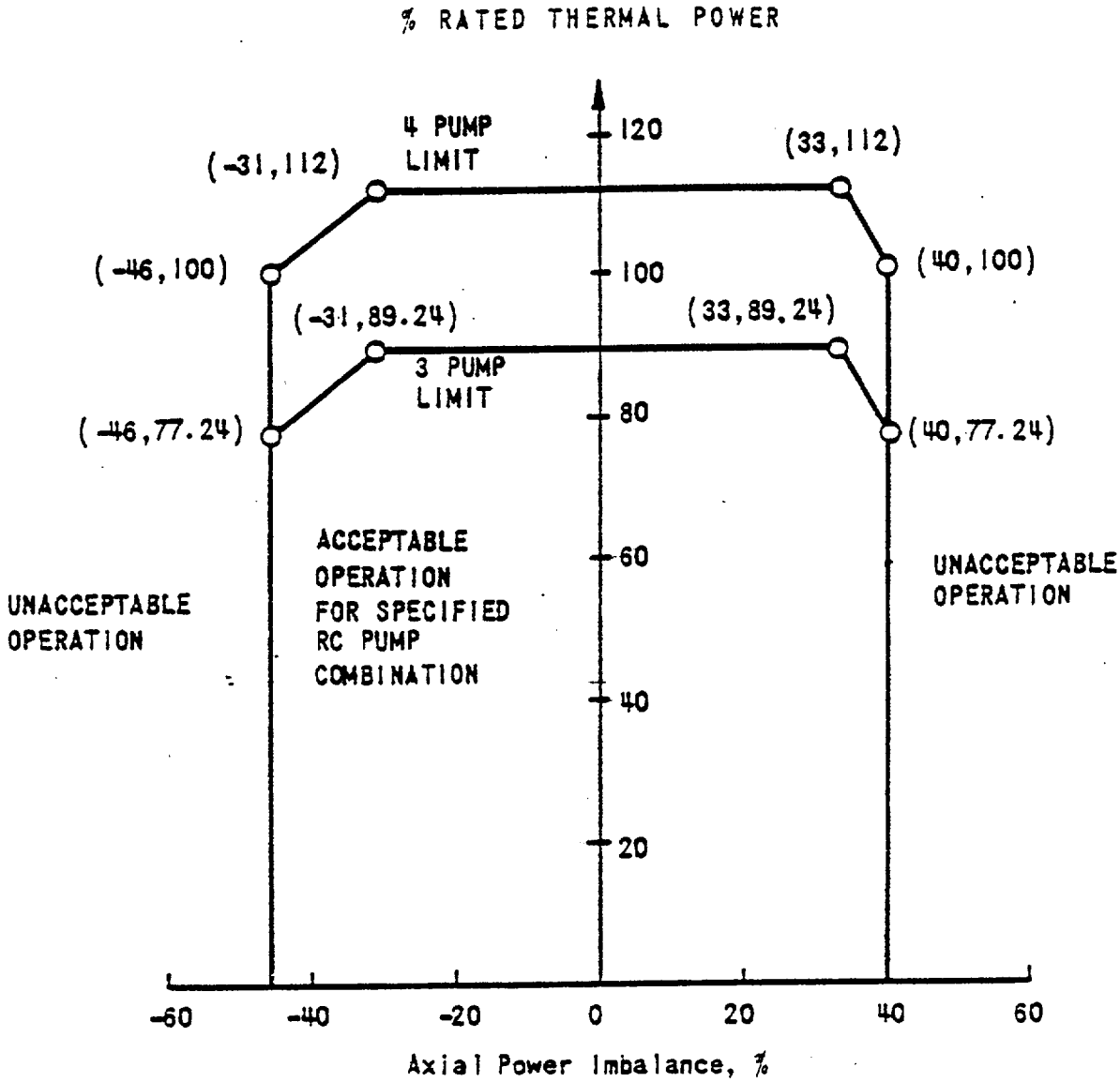


Figure 2.1-2.Reactor Core Safety Limit



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	≤104.94% of RATED THERMAL POWER with four pumps operating ≤79.85% of RATED THERMAL POWER with three pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating# ≤79.85% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/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)	≥(12.60 T_{out} °F - 5662.2) psig	≥(12.60 T_{out} °F - 5662.2) 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 and 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 pumps 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 — ΔFlux/Flow

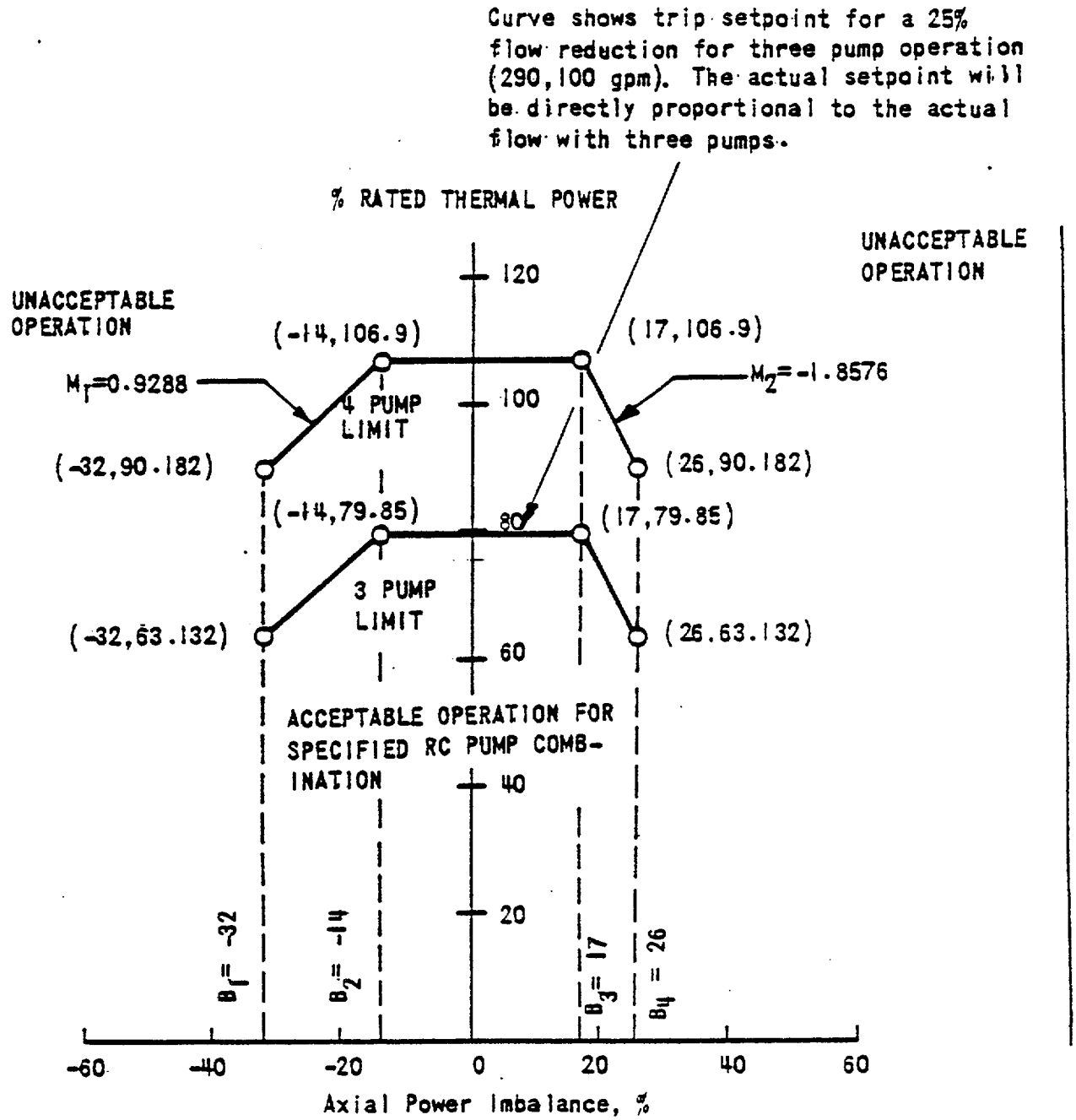


Figure 2.2-2 Allowable Value for Flux-Δ Flux/Flow

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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 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 limits 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 1D, 2B, 4 and 5A and 20.5 kW/ft for batches 5B and 6.

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.

2.2.1. REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation trip setpoints 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 $\leq$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 104.94% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RC High Temperature

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

Flux -- ΔFlux/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 106.9% 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 79.9% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 70.2% of full flow rate and power is 75%.

For safety calculations the 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, $\leq$ 2525 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure , 1983.4 psig, and RC pressure-temperature (12.60 t_{out} - 6662.2) 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.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/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.

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, 3.1-2b, 3.1-2c, 3.1-3a, 3.1-3b and 3.1-3c.

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 24 + 10, -0 EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

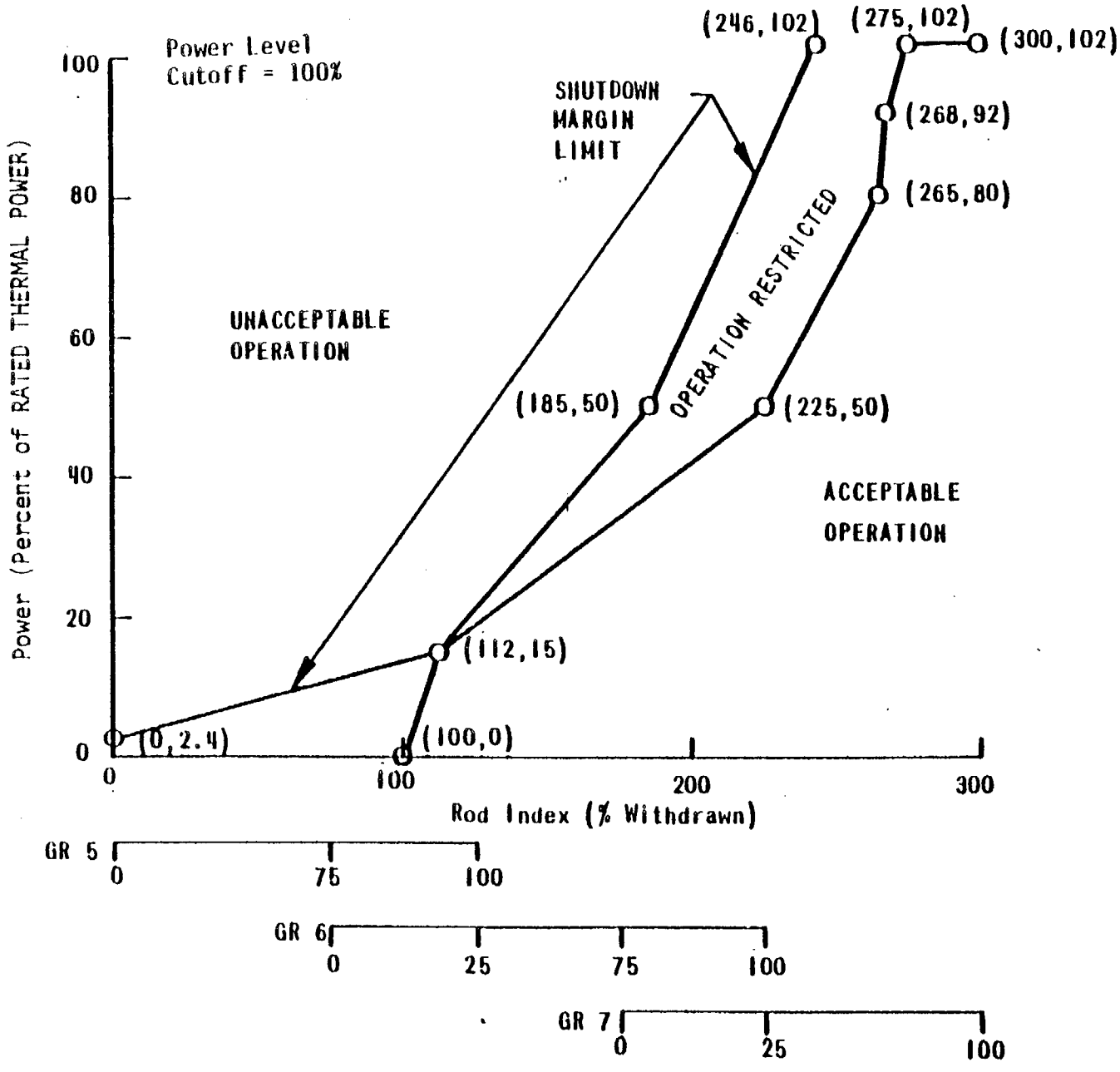

Figure 3.1-2b. Regulating Group Position Limits, 24 + 10, -0 to 150 ± 10 EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

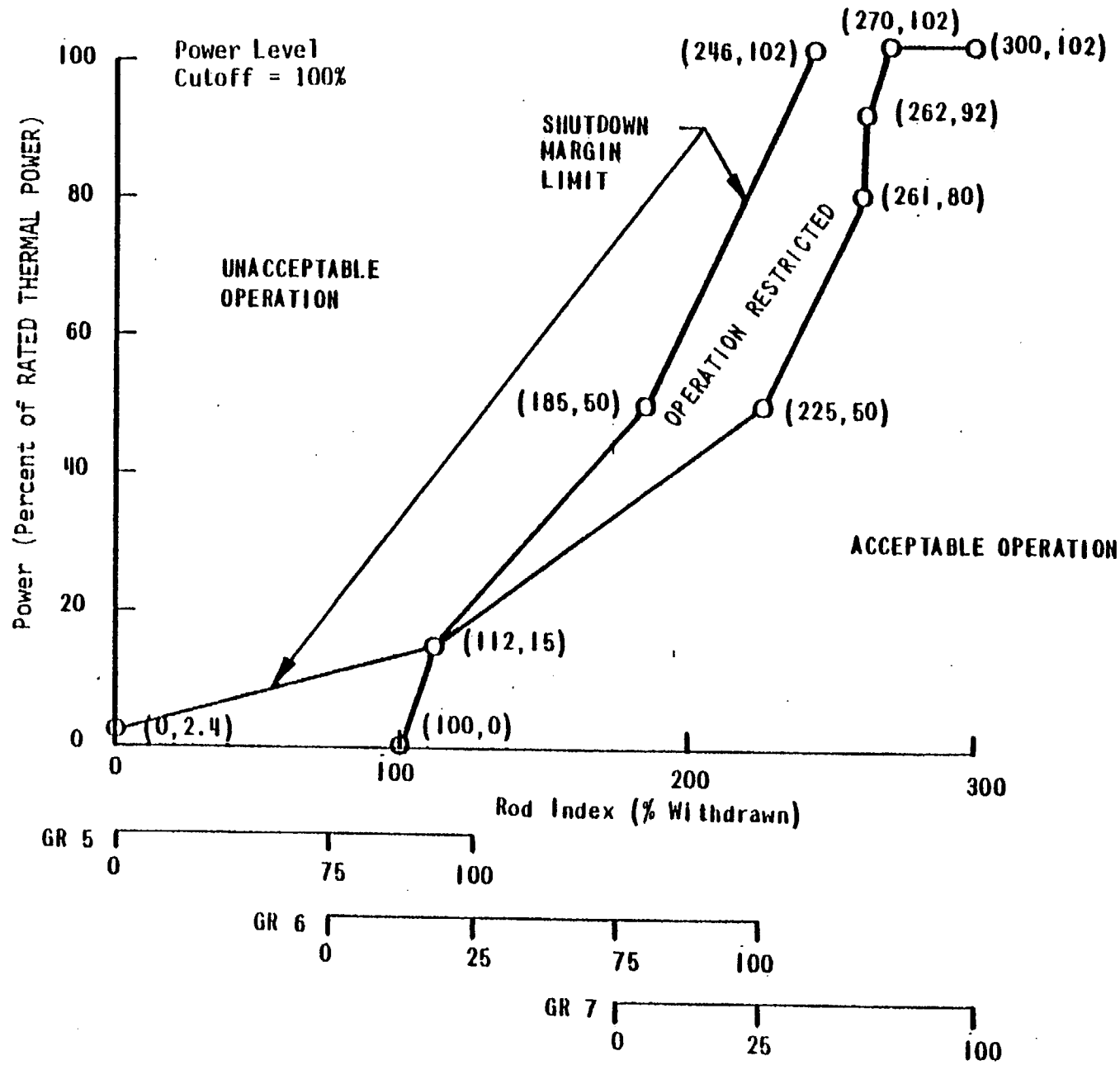

Figure 3.1-2c. Regulating Group Position Limits After 150 ±10 EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

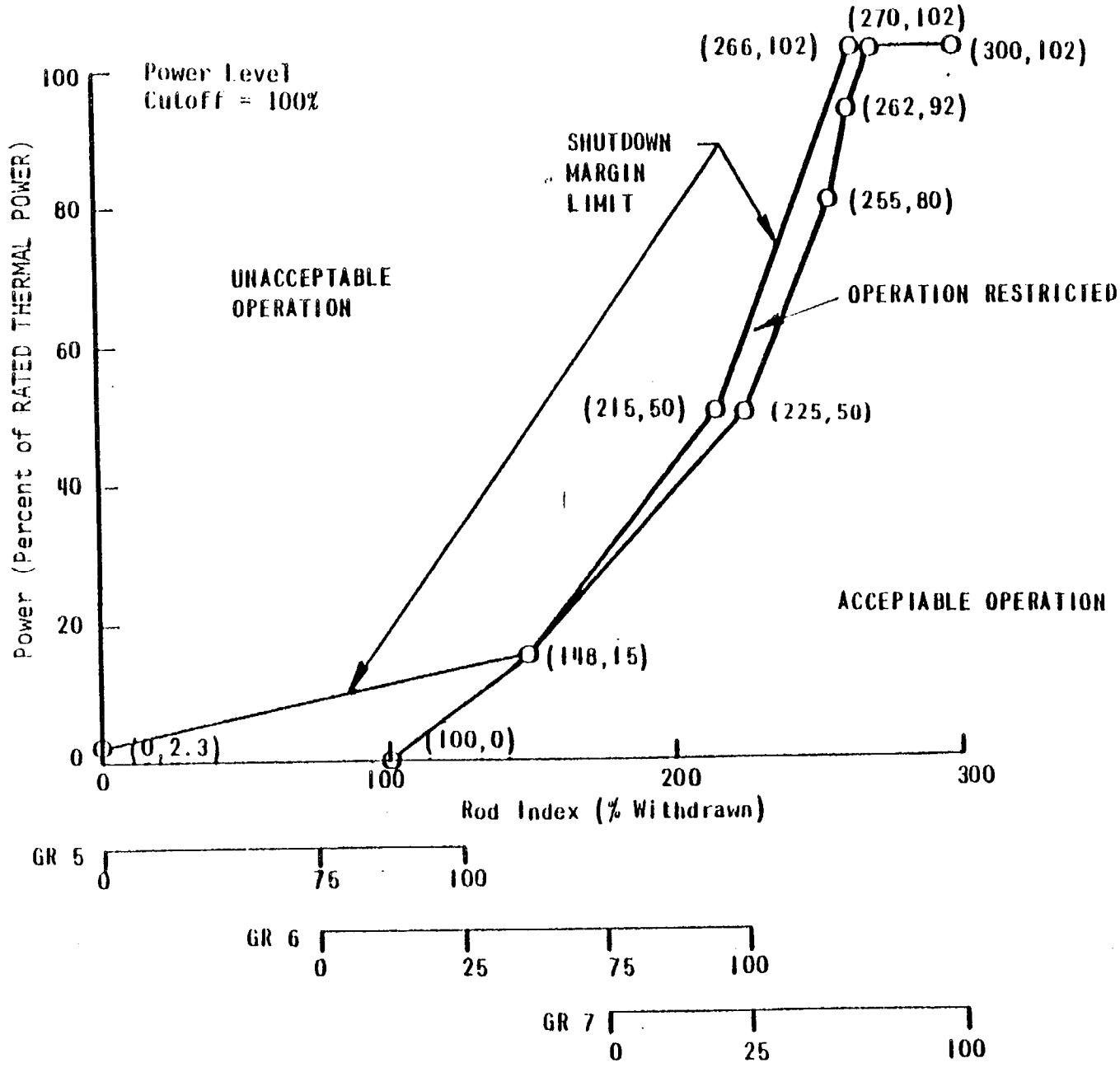

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Figure 3.1-3a. Regulating Group Position Limits, 0 to 24 + 10, -0 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

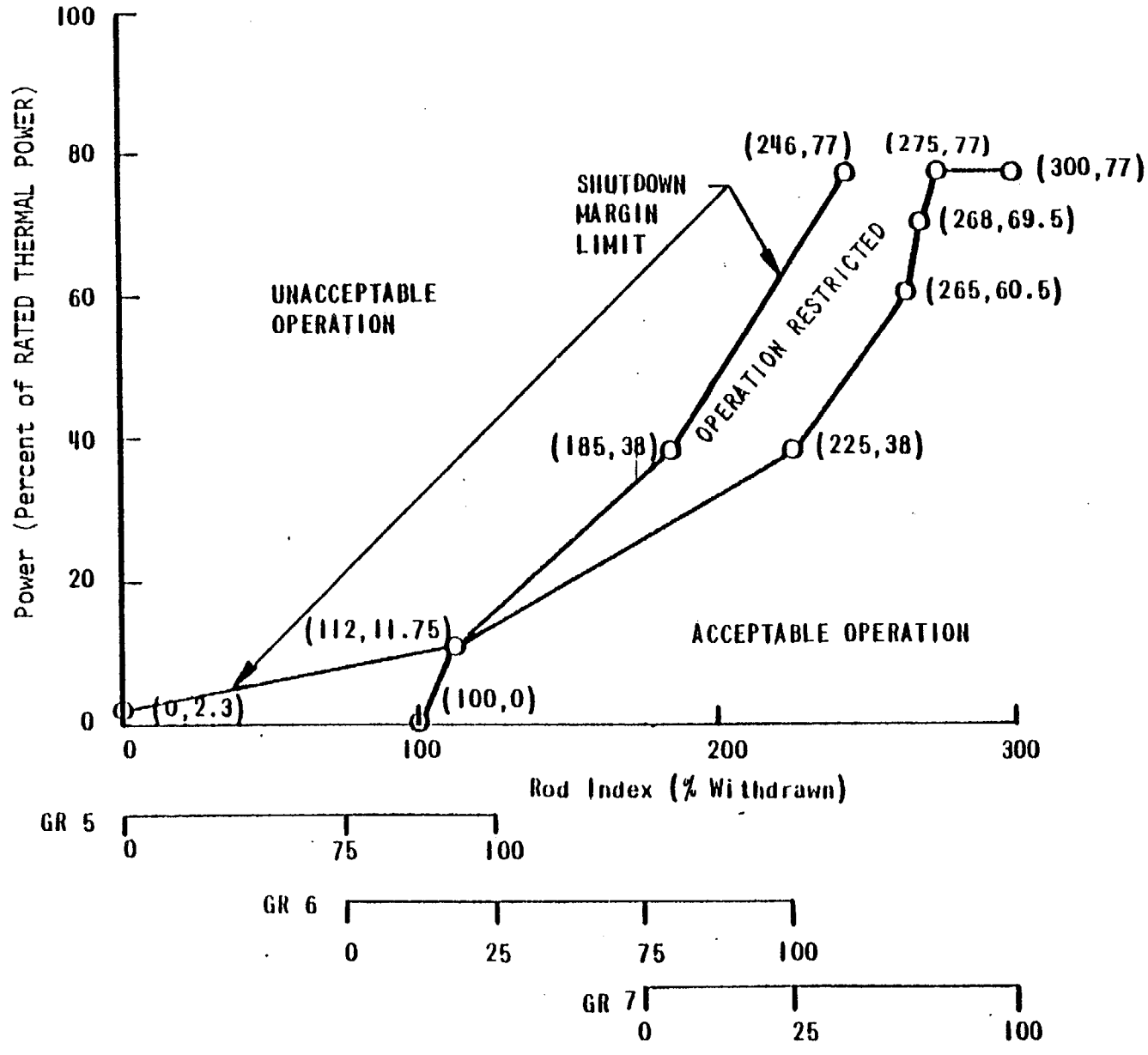


Figure 3.1-3b. Regulating Group Position Limits, 24 + 10, -0 to 150 ± 10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

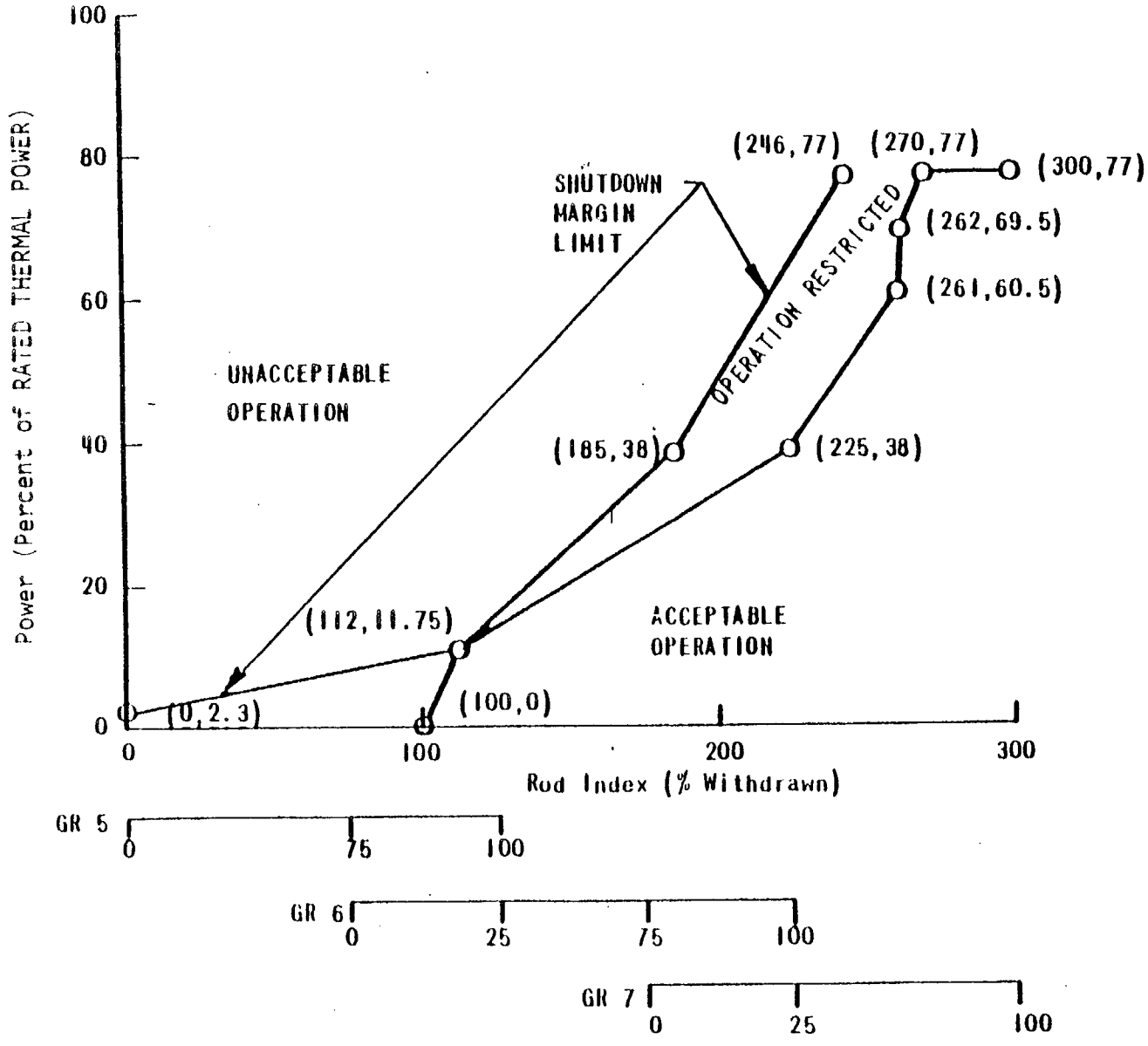
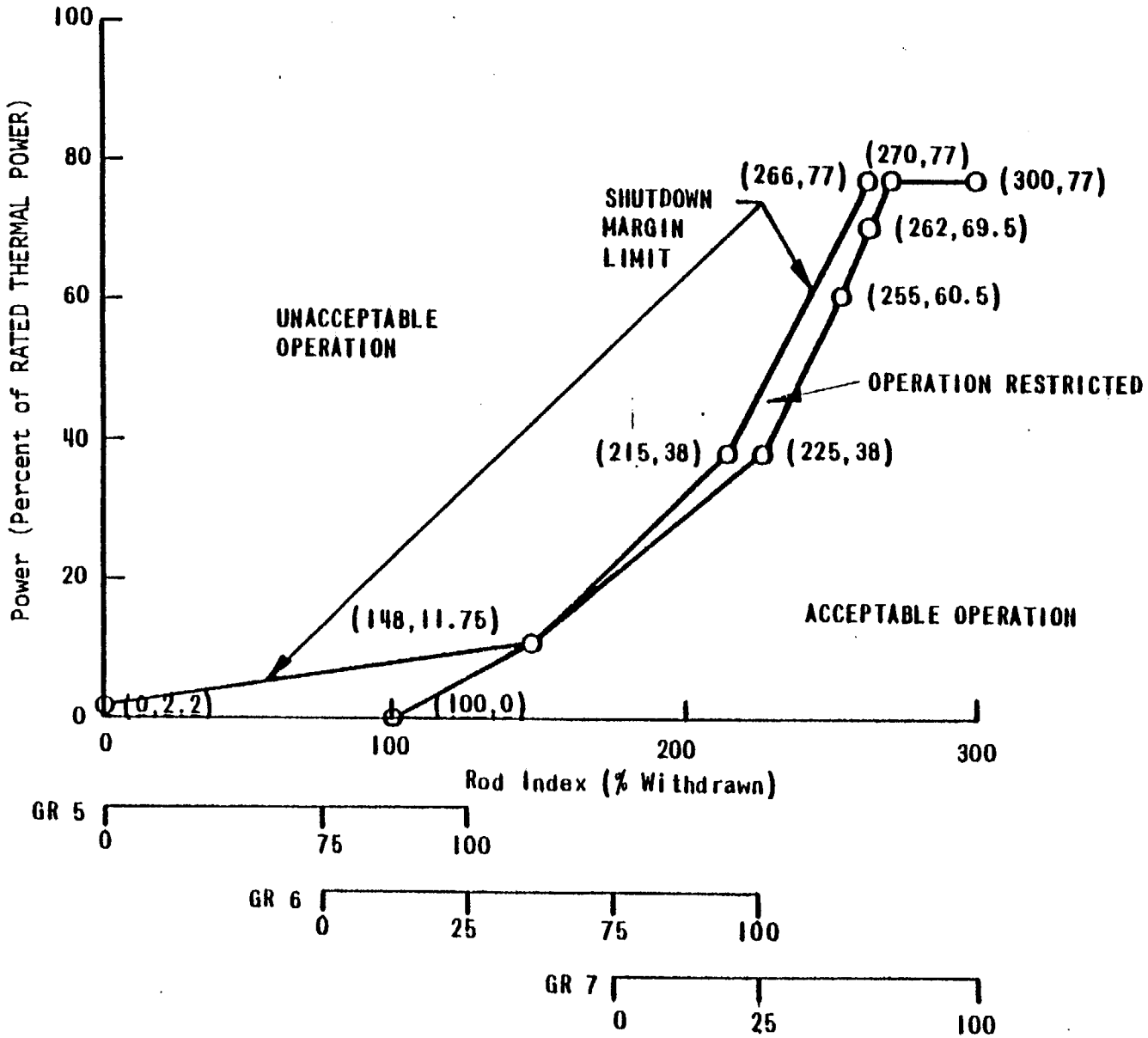


Figure 3.1-3c. Regulating Group Position Limits After 150 ±10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4



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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.

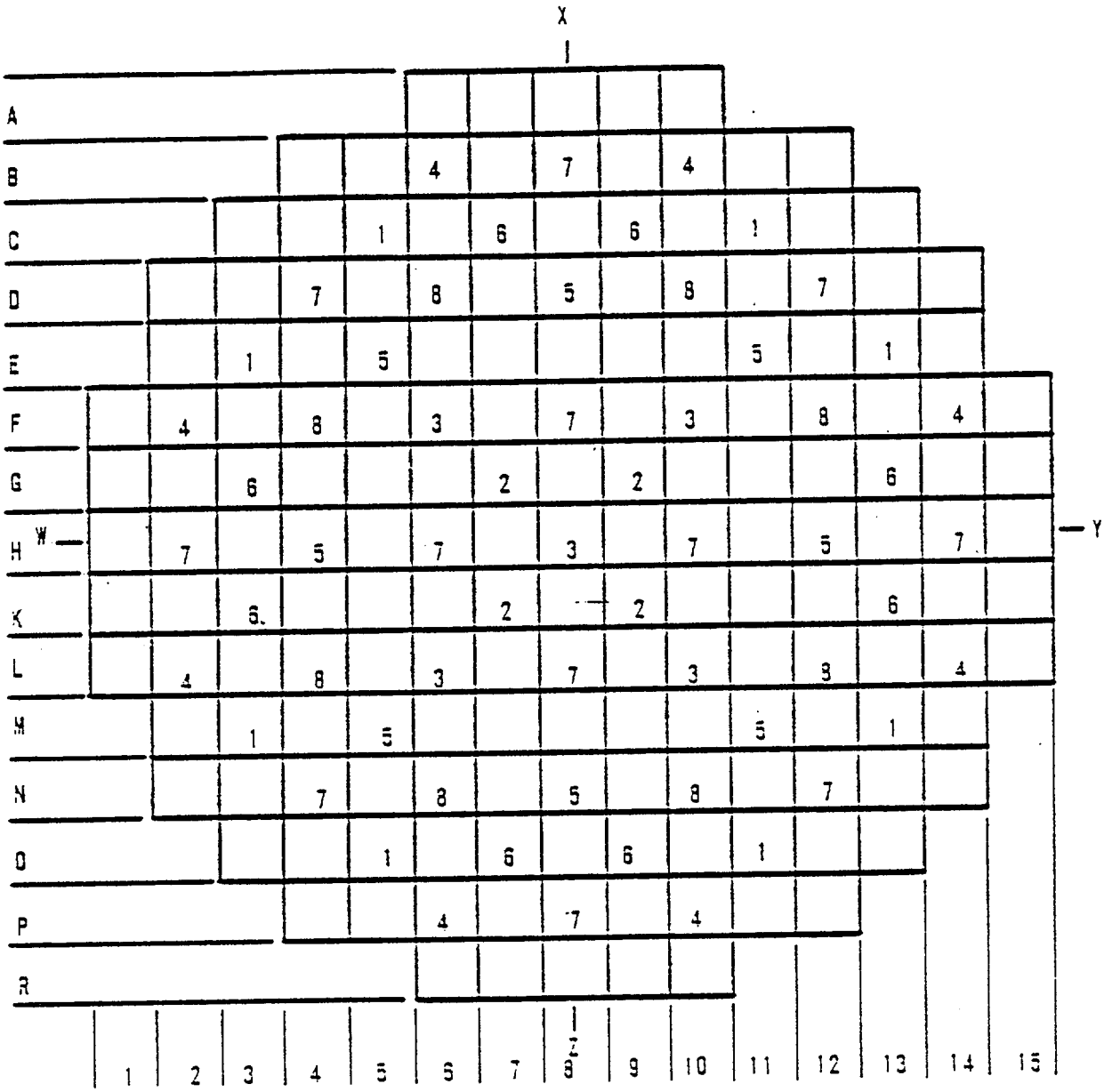
      b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

   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 4



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

X GROUP NUMBER

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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, 3.1-5b, 3.1-5c, 3.1-5d, 3.1-5e and 3.1-5f.

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 24 + 10, -0, EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

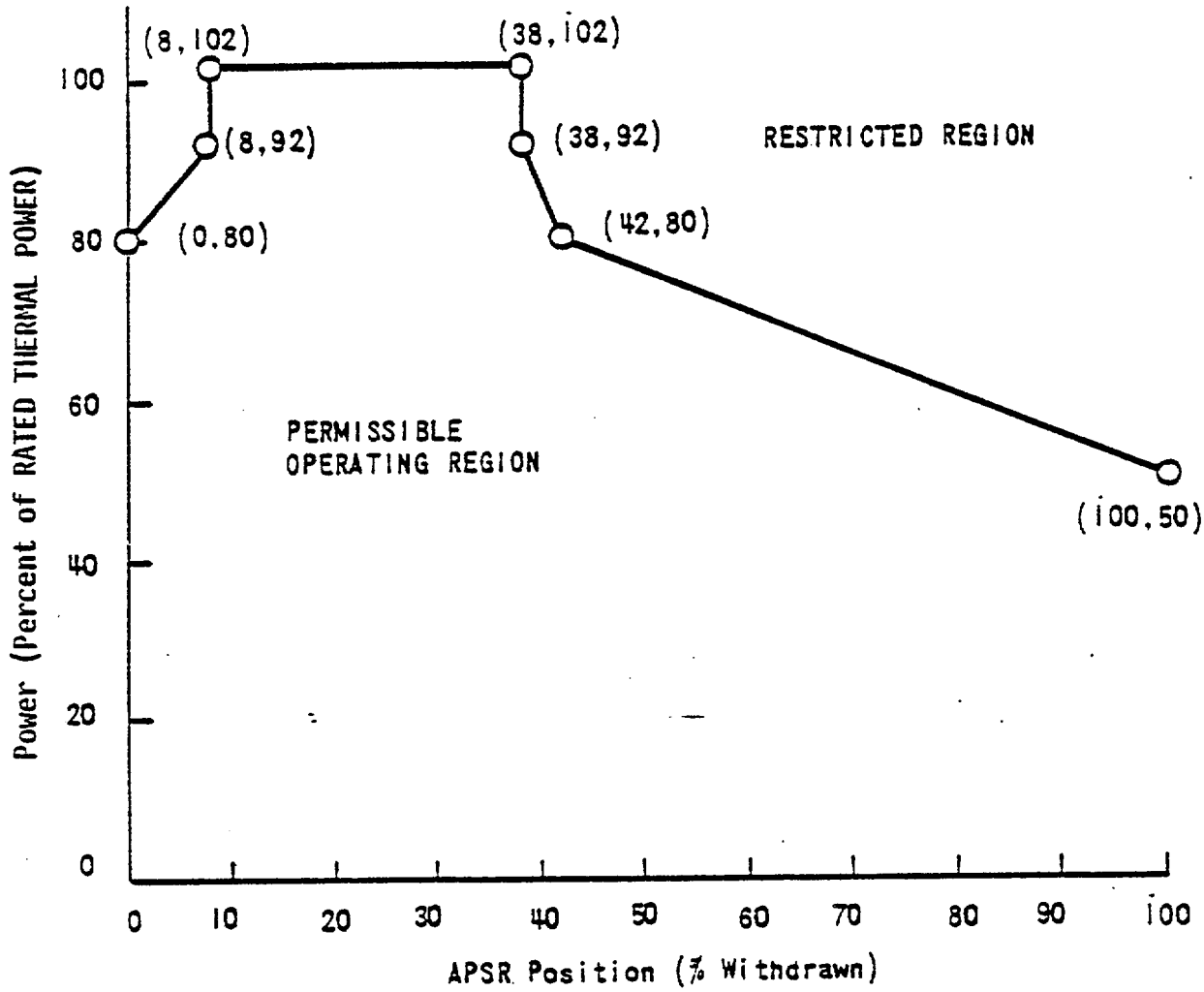


Figure 3.1-5b. APSR Position Limits, 24 + 10, -0 to 150±10 EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

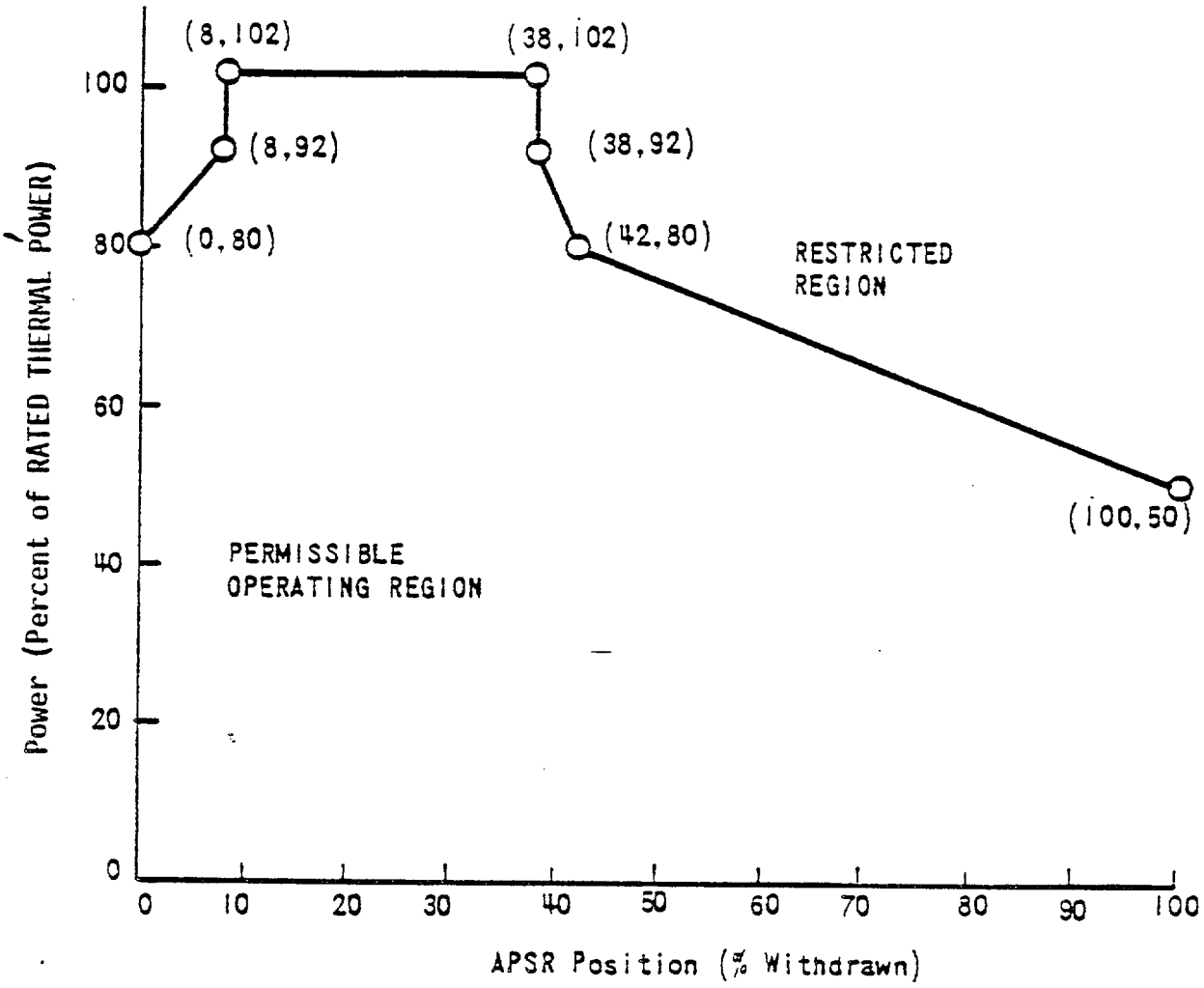


Figure 3.1-5c. APSR Position Limits After 150 ±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4

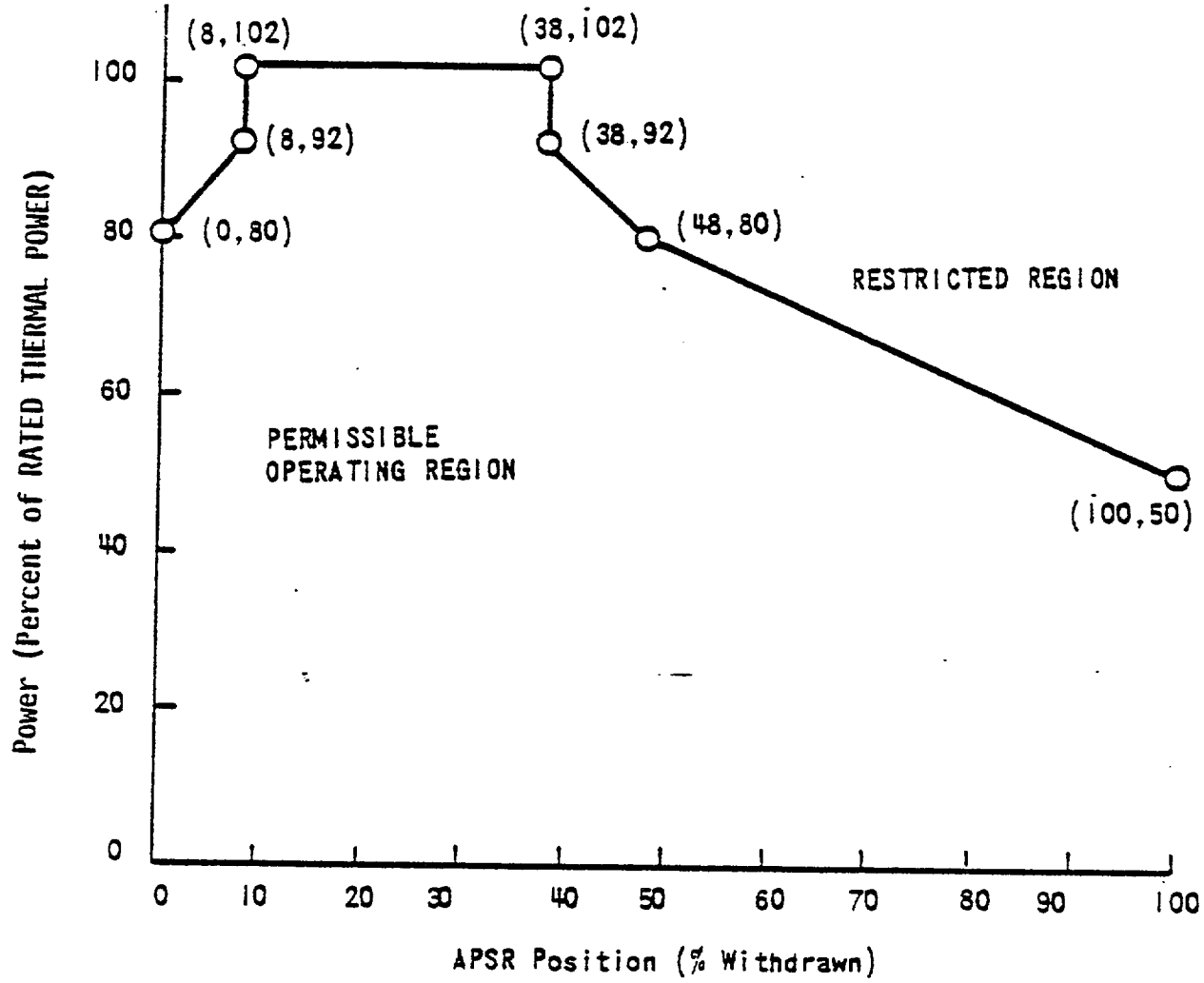


Figure 3.1-5d. APSR Position Limits, 0 to 24 + 10, -0 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

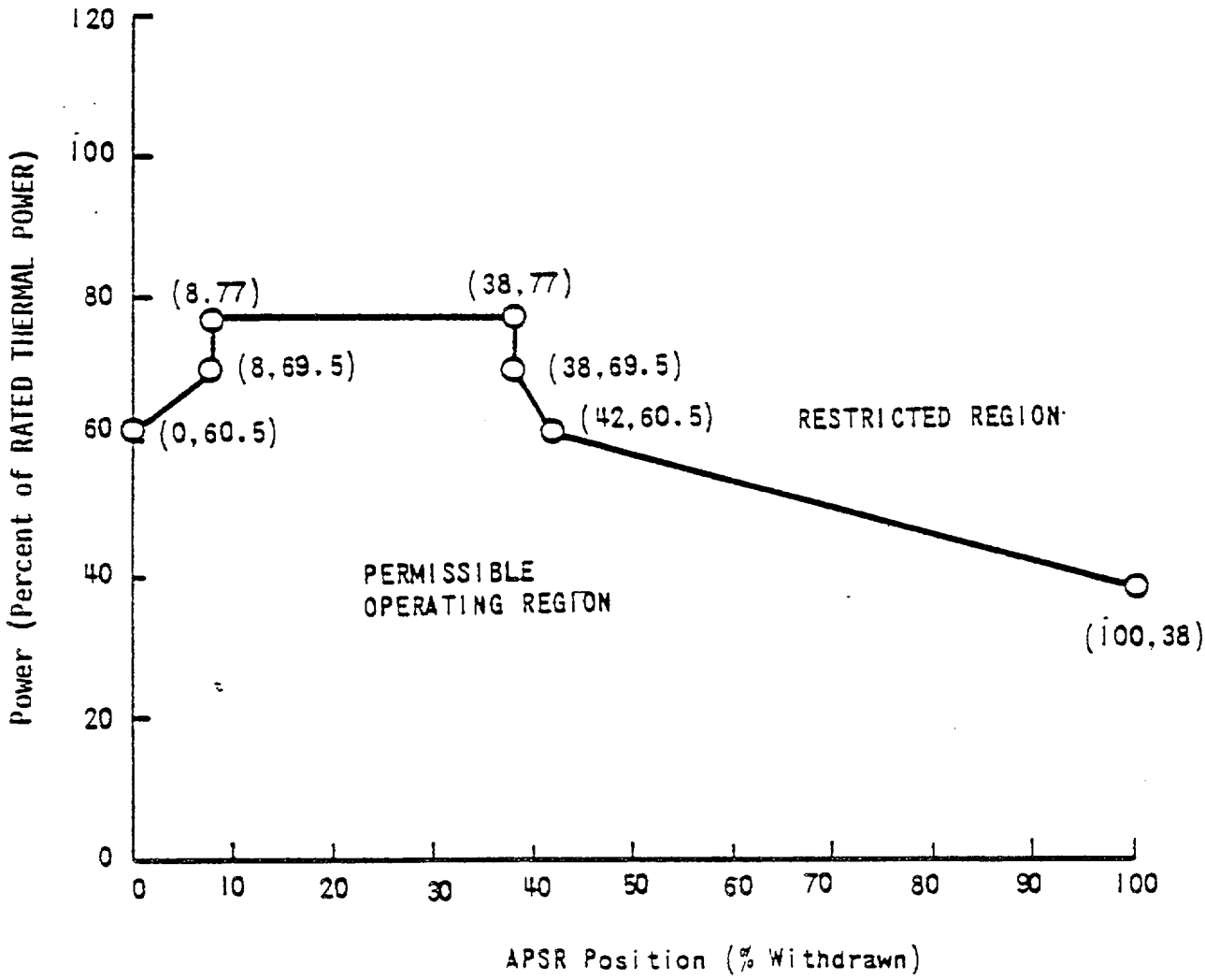


Figure 3.1-5e. APSR Position Limits, 24 + 10, -0 to 150±10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

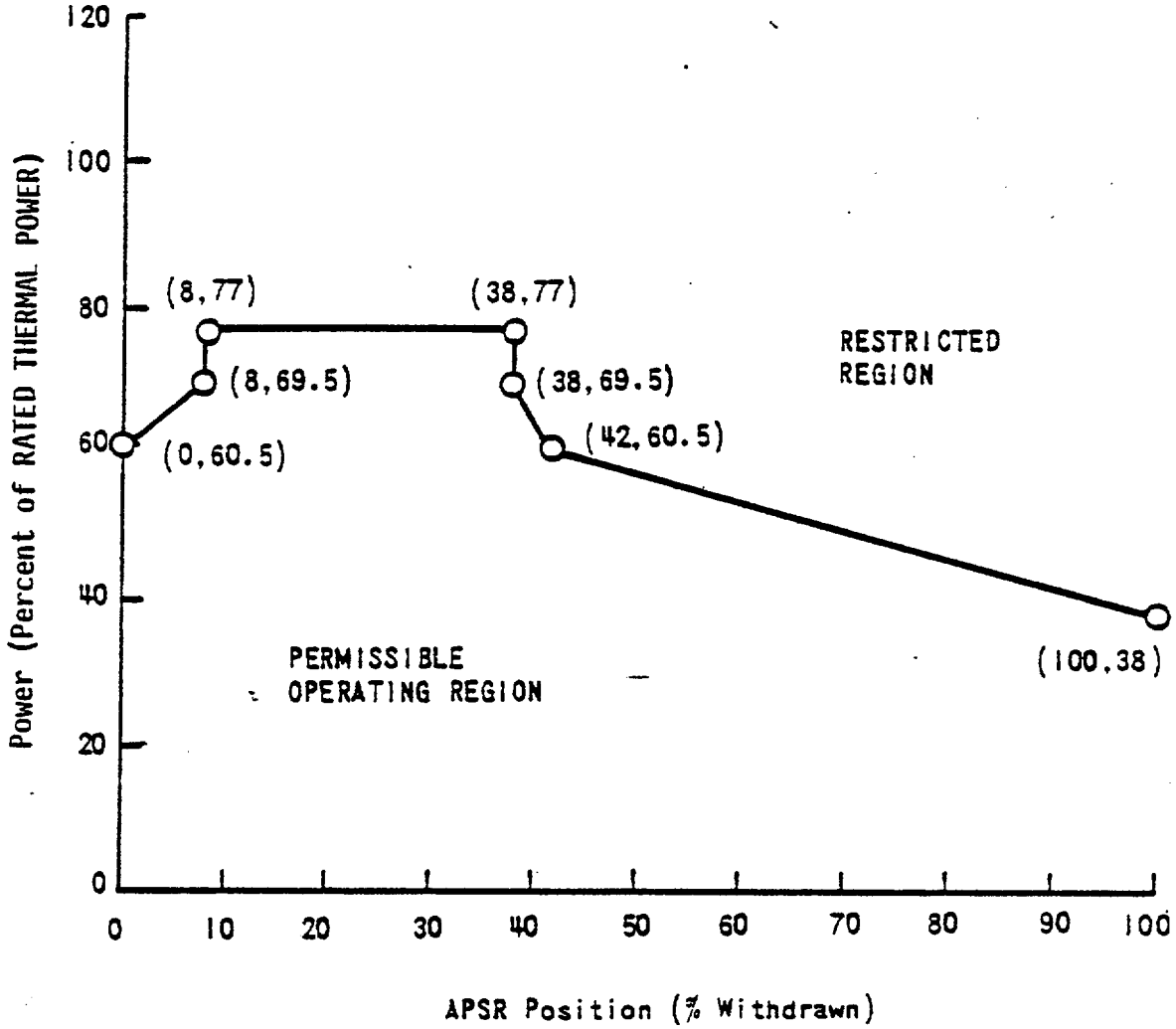
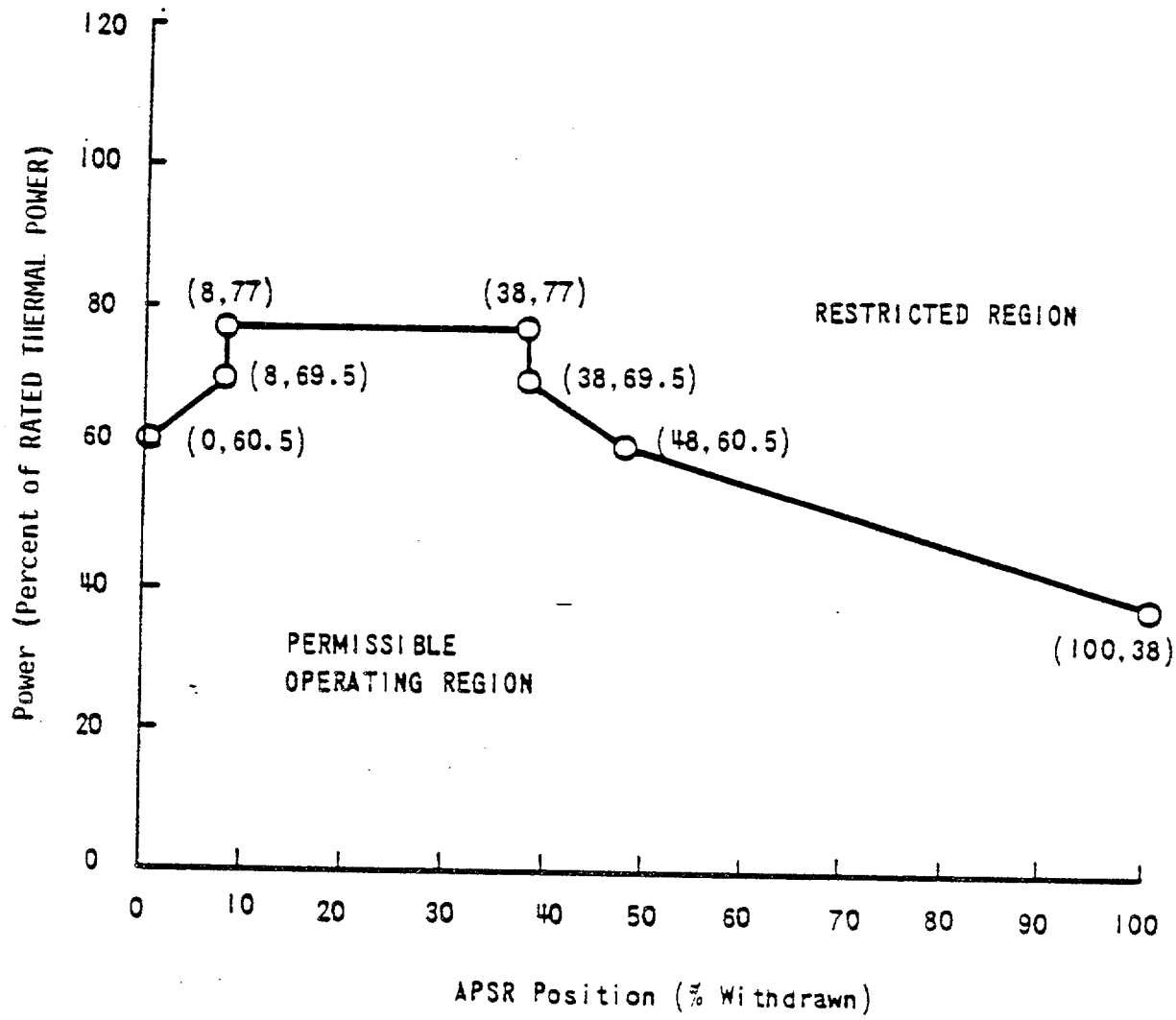


Figure 3.1-5f. APSR Position Limits After 150 ±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4



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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, 3.2-1b, 3.2-1c, 3.2-2a, 3.2-2b and 3.2-2c.

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 24+10, -0 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4

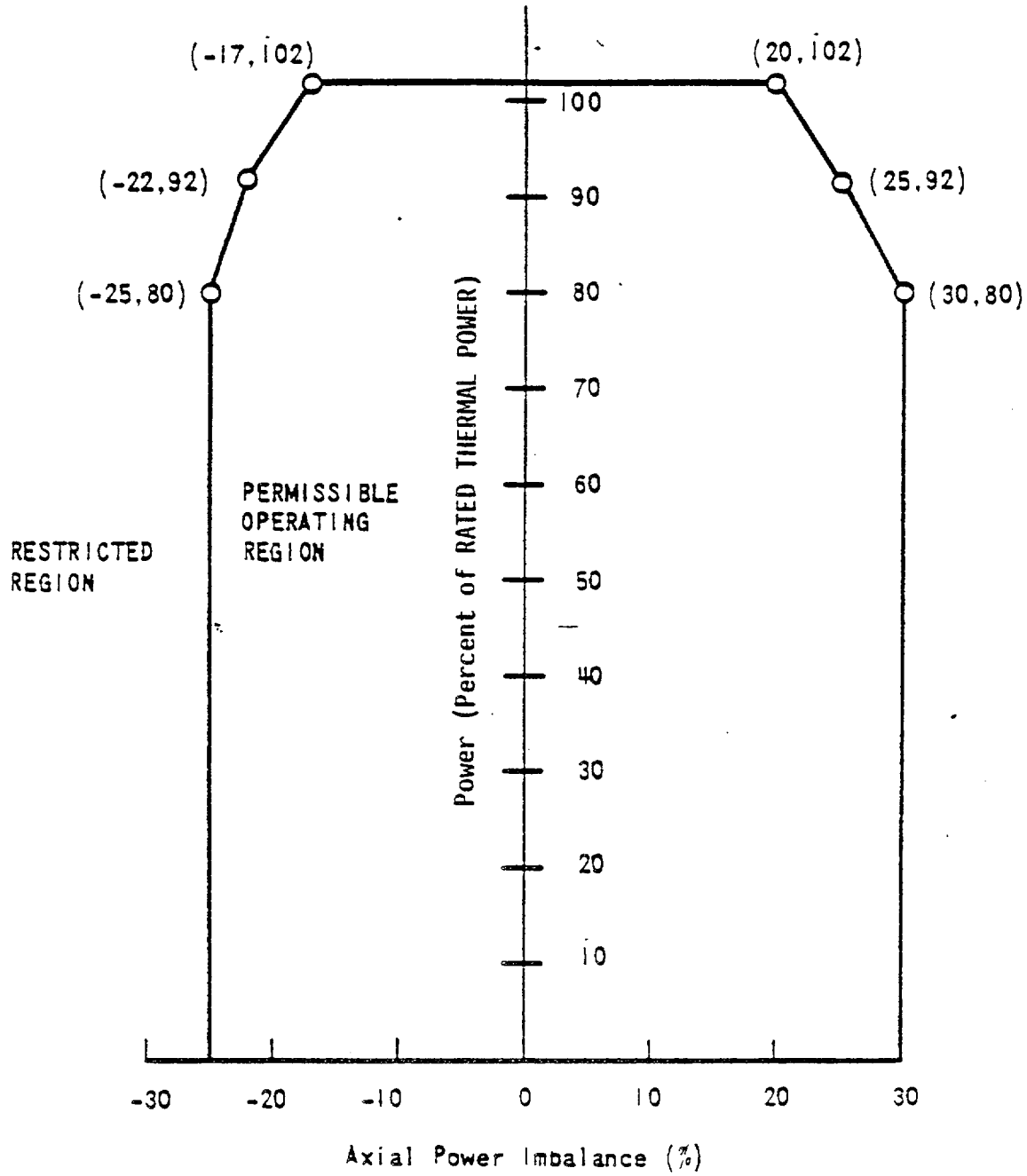

Figure 3.2-1b. Axial Power Imbalance Limits, 24 + 10, -0 to 150 ± 10 EFPD, Four RC Pumps - Davis-Besse 1, Cycle 4

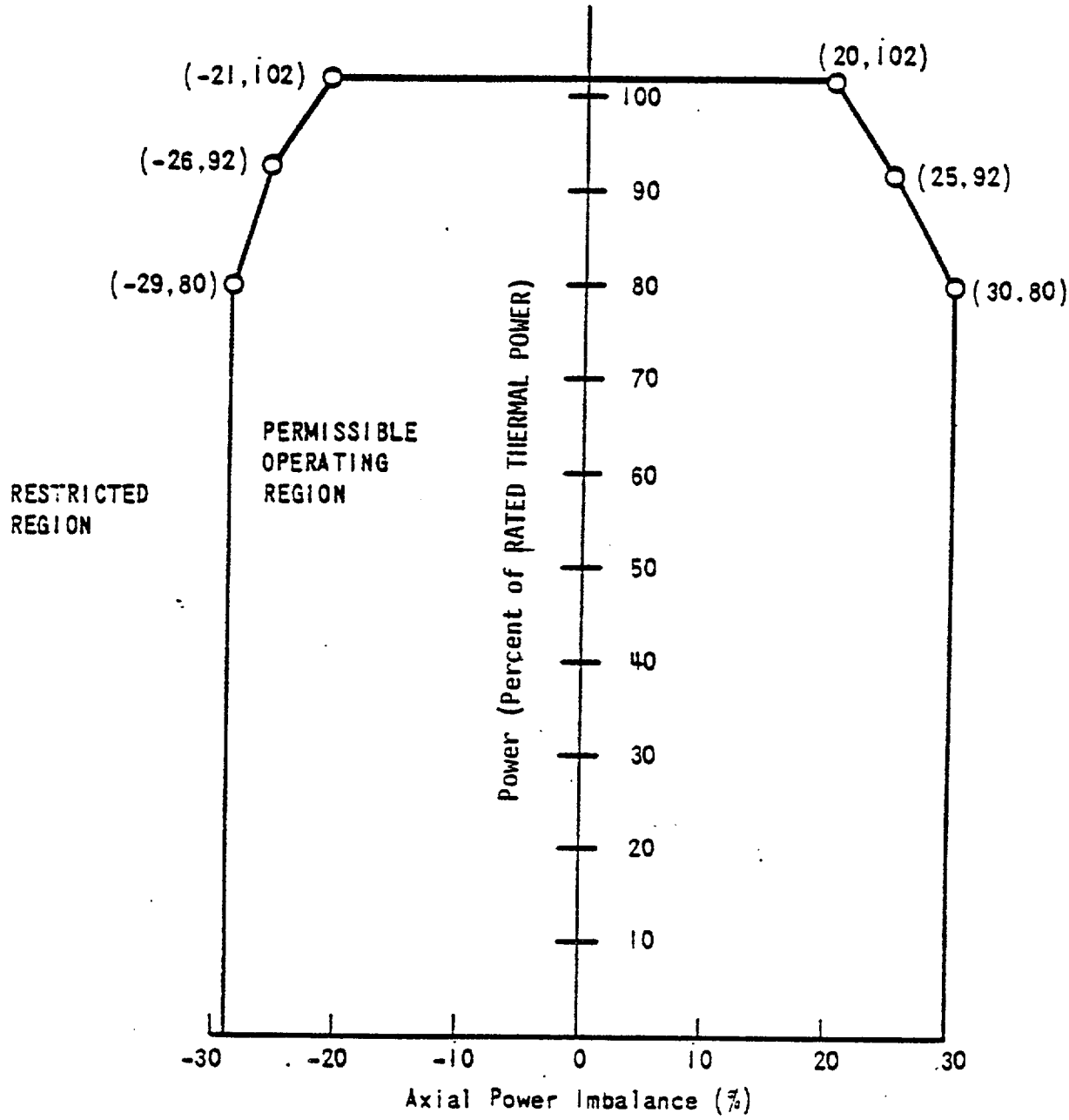

Figure 3.2-1c Axial Power Imbalance Limits After 150 ±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4

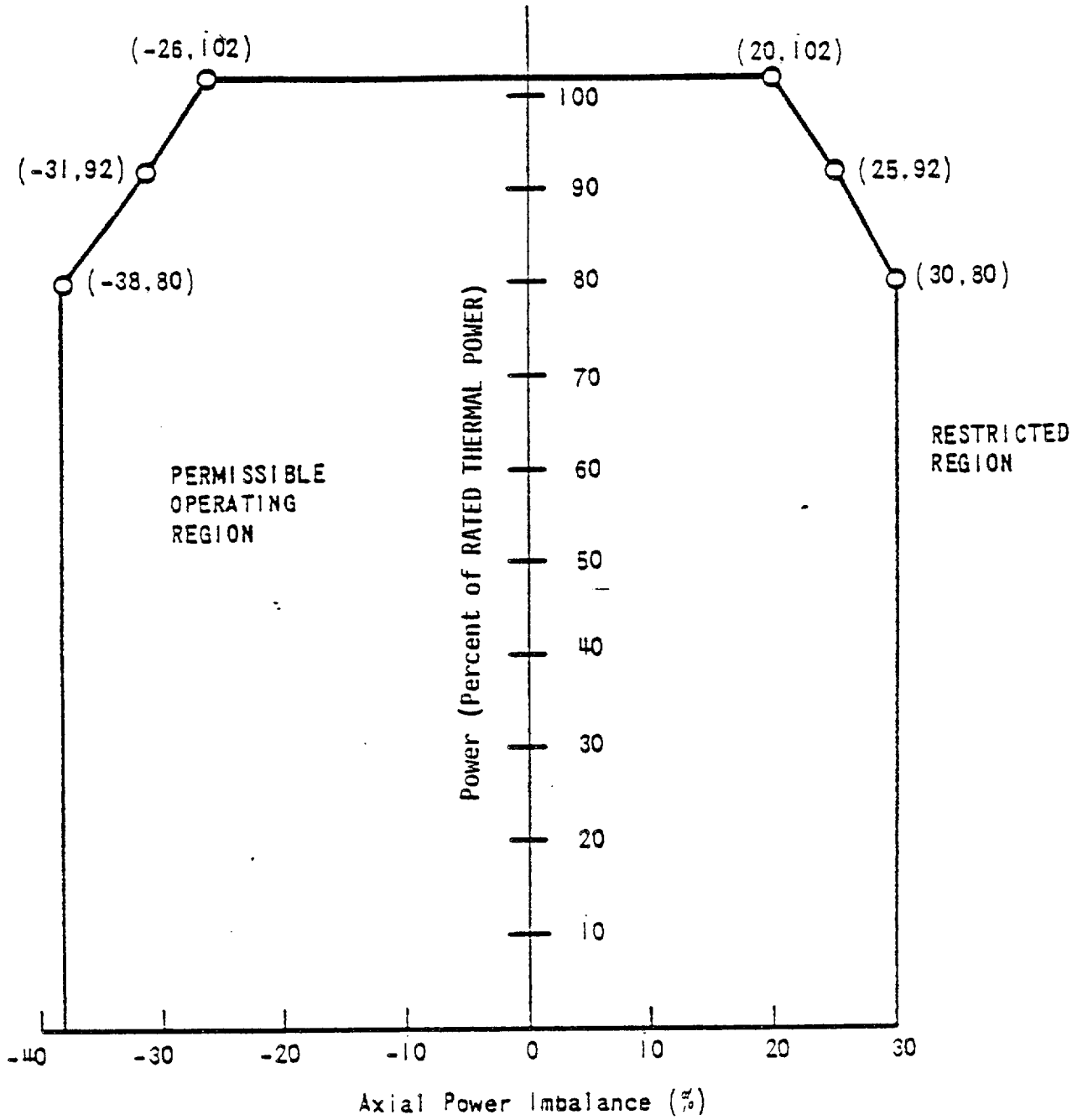

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Figure 3.2-2a. Axial Power Imbalance Limits, 0 to 24 + 10, -0 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

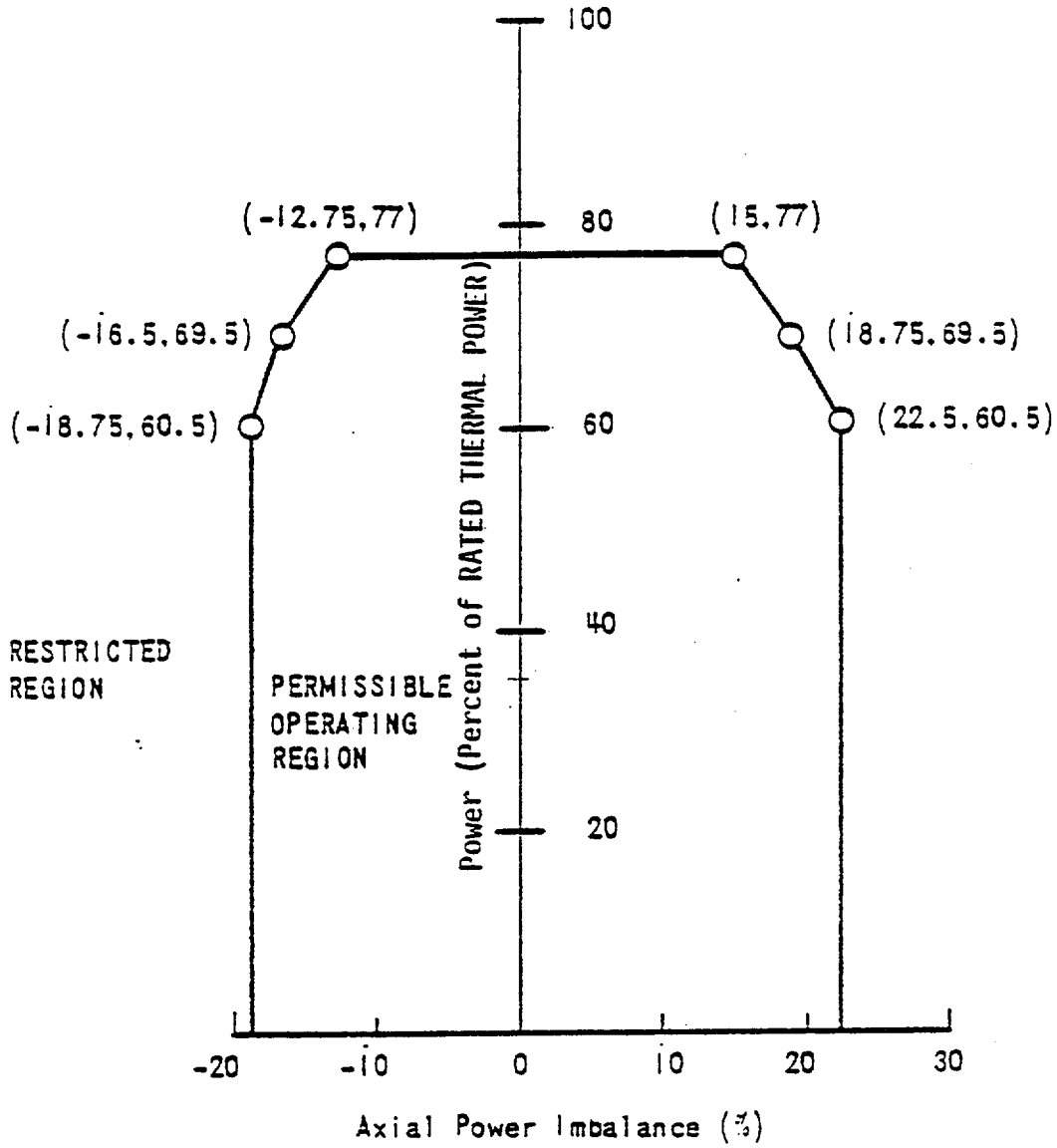


Figure 3.2-2b. Axial Power Imbalance Limits, 24 + 10, -0 to 150 ± 10 EFPD, Three RC Pumps - Davis-Besse 1, Cycle 4

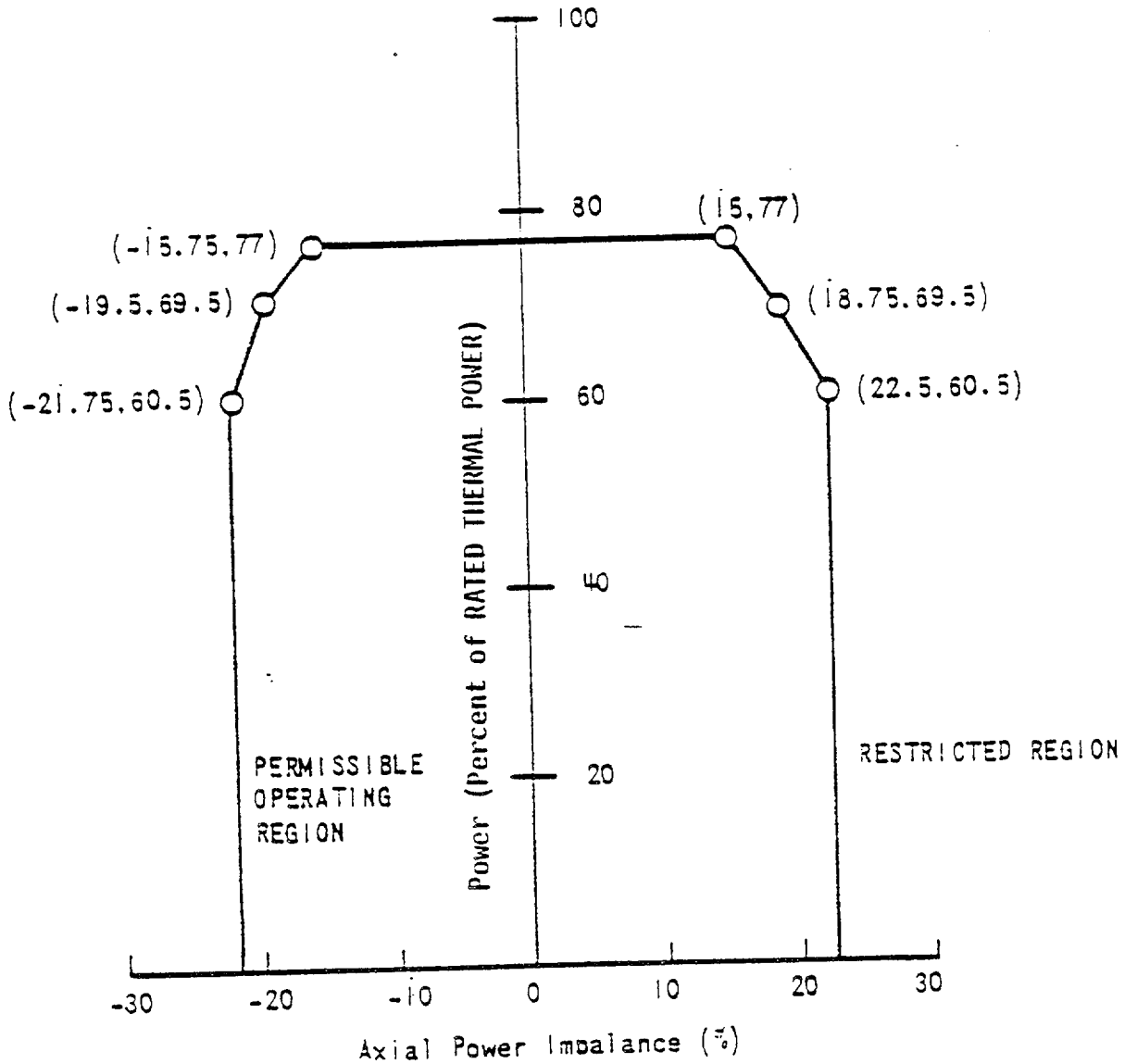
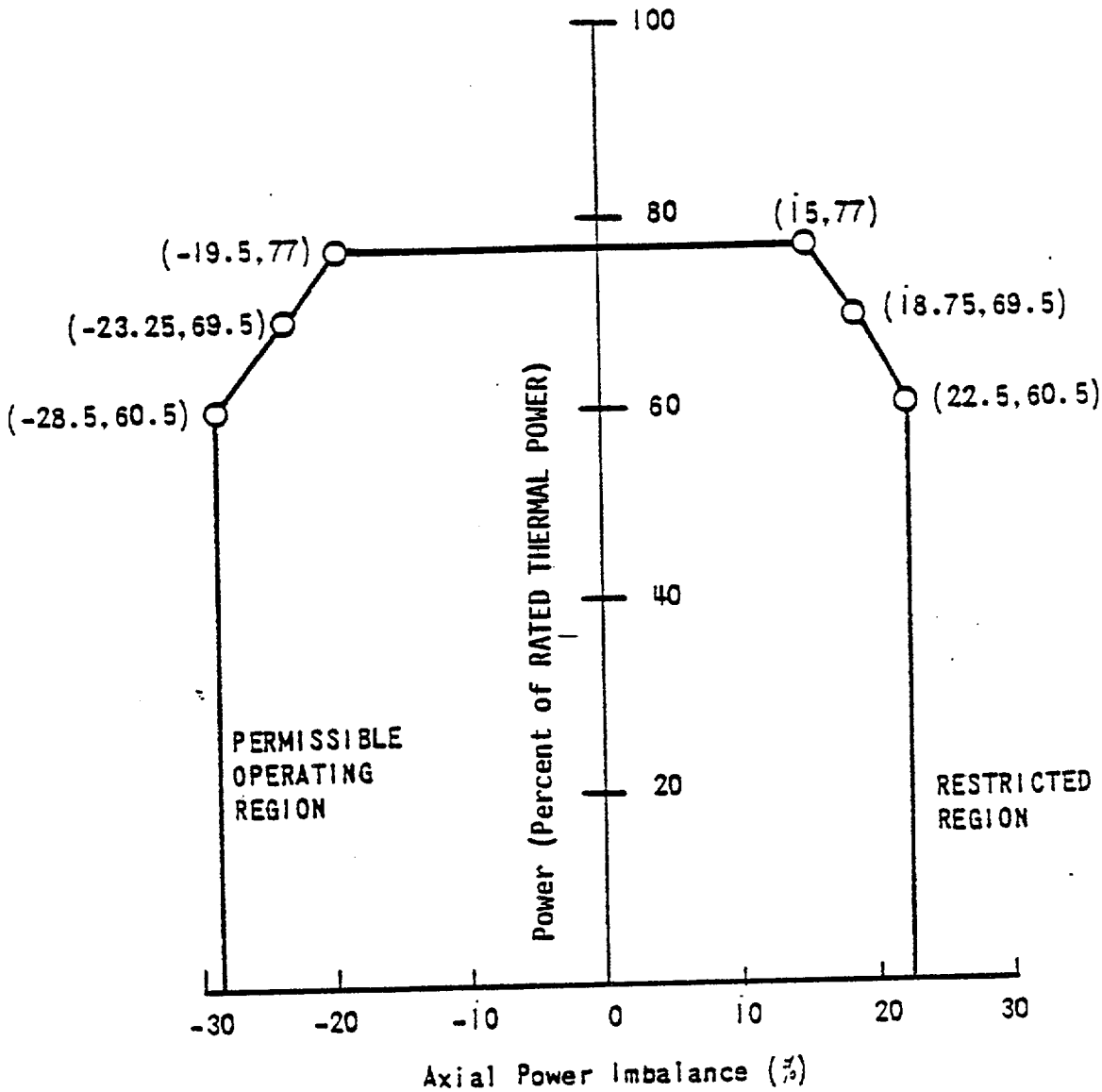


Figure 3.2-2c. Axial Power Imbalance Limits After 150 ±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4



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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% Δk/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 482,778 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

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 tolerance 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:

$$F_Q \leq 2.93; \quad F^N_{\Delta H} \leq 1.71$$

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NPF-3
TOLEDO EDISON COMPANY
AND
CLEVELAND ELECTRIC ILLUMINATING COMPANY
DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1
DOCKET NO. 50-346

1.0 Introduction

By letter dated July 5, 1983 (Ref. 1), Toledo Edison Company (the licensee) made application to modify the Davis-Besse Nuclear Power Station, Unit No. 1, Technical Specifications to permit operation for a fourth cycle. The analysis performed and the resulting modifications to the Technical Specifications are described in the Unit 1, Cycle 4 Reload Report (Ref. 2). The safety analysis for the previous third cycle of operation at Davis-Besse 1 is being used by the licensee for the proposed fourth cycle of operation. Where conditions are identical or limiting in the third cycle analysis, our previous evaluation (Ref. 3) of that cycle continues to apply.

1.1 Description of the Cycle 4 Core

The Davis-Besse Cycle 4 core will consist of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 4 will operate in bleed-and-feed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 53 full length control rod assemblies (CRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. No burnable poison rods will be utilized in the Cycle 4 core.

The length of Cycle 4 is expected to be 240 effective full power days (EFPD) of operation, marginally lower than the 268 EFPD accumulated during Cycle 3. The licensed core full power level remains at 2772 MWt.

2.0 Evaluation of the Cycle 4 Core

2.1 Fuel System Design

The 48 Babcock and Wilcox (B&W) Mark-B4 fuel assemblies loaded as Batch 6 at end of Cycle 3 (EOC 3) are mechanically interchangeable with Batches 1D, 2B, 4, 5A and 5B fuel assemblies previously loaded at Davis-Besse Unit 1. The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for Davis-Besse 1 or were analyzed specifically for Cycle 4 using methods and limits previously reviewed and approved by the NRC. End-of-life fuel rod internal pressures have also been analyzed using previously-approved methods and limits.

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P PDR

The thermal behavior of the fuel in the Cycle 4 core has been analyzed with two B&W fuel thermal performance codes, TAFY-3 (Ref. 4) and TACO-2 (Ref. 5). Although both of these codes have been approved for use in safety analysis, we believe (Ref. 6) that only the newer TACO series of codes are capable of correctly calculating fission gas release (and therefore rod pressure) at high burnups. Babcock & Wilcox has responded (Ref. 7) to this concern with an analytical comparison between the TAFY-3 code and an earlier version of TACO called TACO-1 (Ref. 8). In this response, they have stated that the fuel rod internal pressure predicted by TACO-1 is lower than that predicted by TAFY-3 for fuel rod exposures of up to 42 MWd/kgU. The licensee has stated that the maximum expected exposure of any fuel rod during Cycle 4 is less than this amount. We find this acceptable.

For the Loss of Coolant Accident (LOCA) analysis (Section 7.2 of the Reload Report), the volume-averaged fuel temperature and fuel rod internal pressure were calculated for Cycle 4 as a function of linear heat rating. The licensee has stated that these conditions are bounded by those used in the generic LOCA analysis for Davis-Besse Unit 1.

As mentioned previously, B&W currently has several fuel performance codes which are approved and could be used to calculate LOCA initial conditions. The older TAFY-3 code was used for the generic LOCA analysis cited in the Cycle 4 Reload Report. Information obtained by the NRC staff (Ref. 9) indicates that the TAFY-3 code predictions do not produce higher calculated peak cladding temperatures in the generic LOCA analysis than the newer TACO-1 or TACO-2 codes as suggested by the licensee. The issue involves excessive fuel densification and lowered fuel rod internal gas pressures at beginning of life. Babcock and Wilcox has proposed a method of resolving this issue which has been adopted by Toledo Edison Company (Ref. 10). The method relies on reduced peak linear heat rate (PLHR) limits at low core elevations for the first 24 effective full power days (EFPD) of operation based on comparison of TAFY-3 and TACO-2 calculated LOCA initial conditions. The method is similar to an older TAFY-3/TACO-1 comparison used in the Davis-Besse 1 Cycle 3 safety analysis. However, the resulting PLHR reduction is different for each code.

In addition to the issue of initial fuel temperatures and rod internal pressures used in the LOCA analysis, a second issue involving cladding swelling and rupture models has affected the proposed Cycle 4 operating limits for Davis-Besse 1. In late 1979, the NRC staff reviewed Emergency Core Cooling System (ECCS) fuel cladding models in light of new data. Adequacy of the models then in use was questioned and new models, developed as Appendix K acceptance criteria, were presented in NUREG-0630 (Ref. 11). Each fuel vendor was then asked to show how, in light of the new models, the plants analyzed with their analytical methods continued to meet the applicable LOCA limits. The B&W response (Ref. 12) concluded that the impact of the NRC models was small and did not result in analytical results in excess of the LOCA limits.

A more recent B&W calculation (Ref. 13), however, found that the cladding swelling and rupture models presented by the staff have a non-trivial effect on LOCA peak cladding temperatures in B&W 177 fuel assembly plants. Because this

calculation was applicable to all B&W plants, the licensee was requested (Ref. 14) to provide supplemental calculations for Davis-Besse Unit 1 similar to those provided in Reference 13. The licensee's responses (Refs. 15 to 18) culminated in the supplemental calculation (Ref. 10) cited previously. This calculation, which considers both fuel densification (TAFY-3/TACO-2) and cladding swelling and rupture effects, results in low core elevation PLHR limits which are more restrictive than those which consider only fuel densification. The licensee has proposed (Ref. 2) modification to the Davis-Besse 1 Technical Specifications which account for these reduced PLHR limits.

In general, the supplemental calculation utilizes previously approved methods except for the substitution of the NRC cladding models. However, there are segments of the analysis (e.g. THETA1-B - Ref. 19) that are currently undergoing NRC review. Babcock & Wilcox has also presented results from a calculation using a new FLECSET heat transfer correlation (Refs. 20 and 21). This correlation appears to offset the NUREG-0630 penalties. The licensee has not yet claimed these FLECSET benefits, however, because the benchmarking and other final evaluations of FLECSET have not been completed and provided to the NRC for review.

Considering the above, we conclude that the licensee's proposed Technical Specification changes are both appropriate and necessary. Since these operating limits are more restrictive than those previously used at Davis-Besse Unit 1, since they are only needed for a brief time period, and since potential for unused compensating benefits may exist, we, therefore, conclude that the operating restrictions imposed on an interim basis are acceptable for incorporating the NUREG-0630 penalties until our final evaluation of FLECSET is completed.

2.2 Nuclear Design

To support Cycle 4 operation of Davis-Besse Unit 1, the licensee has provided analyses (Ref. 2) using analytical techniques and design bases established in B&W reports that have been approved by the NRC staff. The validity of the methods also has been reinforced through predictions of a number of cycles for this and other reactors. The licensee has provided a comparison of the core physics parameters (Ref. 2) for Cycles 3 and 4 as calculated with these techniques. We reviewed the characteristics compared to previous cycles, and find them acceptable for use in the Cycle 4 accident and transient analysis, as described in Section 2.4 of this evaluation.

There are no significant core design changes between the reference Cycle 3 and Cycle 4 designs. The Cycle 4 core was shuffled in a manner to minimize the carryover effect on quadrant tilt. The Cycle 4 design cycle length is 240 days, whereas the Cycle 3 design cycle length was 268 days. No significant operational or procedural changes exist for Cycle 4 with regard to axial or radial power shape, xenon, or tilt control.

Due to the differences in design cycle lengths, the critical boron concentrations for Cycle 4 differ from those of Cycle 3. Because of different isotopic distributions, Cycle 4 control rod worths, ejected rod worths, and stuck rod

worths differ from those of Cycle 3. The licensee took into account ejected rod worths and their adherence to shutdown margin requirements in the development of rod position limits for Cycle 4. The licensee presented an analysis of shutdown margin adequacy as a function of predicted control and stuck rod worths. This analysis allowed for a 10 percent uncertainty on net rod worth and for flux redistribution. It shows considerable margin in excess of requirements.

We, therefore, conclude that the licensee has demonstrated adequate provision of shutdown margin for Cycle 4. In addition, control rod worth measurements are made during startup tests. These confirm the adequacy of predicted control rod worths.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic performance for Cycle 4, in which the fresh Batch 6 fuel is hydraulically and geometrically similar to the other fuel in the Cycle 4 core, is identical to that of Cycle 3. The thermal-hydraulic design evaluation supporting Cycle 4 operation is based on the methods and models previously used in Cycle 3 as described in References 22 and 23. The design conditions are given in Table 1 and are identical for Cycles 3 and 4.

A rod bow topical report (Ref. 24) was submitted and approved (Ref. 25) since the last fuel cycle. This report addressed the mechanisms and resulting local conditions of rod bow. The conclusion was that rod bow penalty is insignificant and is offset by the reduction in power production capability of the fuel assemblies with irradiation. Therefore, there is no resulting rod bow penalty for Cycle 4.

The flux/flow trip setpoint for Cycle 4 has been established as 1.069 (Ref. 26) and was 1.070 for Cycle 3. This setpoint and other plant operating limits are based on criteria that meet the design minimum Departure from Nucleate Boiling Ratio (DNBR) limit of 1.30 calculated using the BAW-2 correlation.

The minimum DNBR at 112 percent of full power is 1.79 for Cycle 4 which is the same as for Cycle 3. The NRC staff finds that the thermal-hydraulic design is acceptable since the Cycle 4 and Cycle 3 (previously approved) design conditions are identical and acceptable design methods have been used in the analysis.

2.4 Accident and Transient Analysis

Acceptability of core thermal, thermal-hydraulic, and kinetics parameters, including the reactivity feedback coefficients and control rod worths, was discussed in Sections 2.2 and 2.3. The licensee concluded, by examination of the Cycle 4 values of these parameters with respect to acceptable previous cycle values, that transients and accidents for Cycle 4 are bounded by previously accepted analyses.

A supplemental ECCS calculation (Ref. 10) for Davis-Besse 1 has resulted in reduced PLHR limits at lower core elevations (see Section 2.1). The new LOCA

limits were used in the generation of more stringent control rod insertion and imbalance limit curves for the first 24 EFPD of Cycle 4. These revised curves have been included in the proposed Technical Specifications for Cycle 4.

2.5 Technical Specification Modifications

The pertinent Technical Specifications have been revised for Cycle 4 operation to account for changes in power peaking and control rod worths as discussed in Sections 2.2 and 2.4. We have reviewed these changes as proposed in Reference 2 and find them all acceptable.

2.6 Summary

We conclude from the examination of Cycle 4 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the Davis-Besse Nuclear Power Station's ability to operate safely during Cycle 4.

3.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

The following NRC personnel contributed to this Safety Evaluation: H. Balukjian, M. Dunenfeld, J. Voglewede.

Dated: September 21, 1983

Table 1 Davis-Besse Cycles 3 and 4 Thermal-Hydraulic Design Conditions

Design power level, MWt	2772
System pressure, psia	2200
Reactor coolant flow, gpm	387,200[b]
Reactor coolant flow, % design	110
Vessel inlet/outlet coolant temp., 100% power, F	557.7/606.3
Ref design radial-local power peaking factor	1.71
Ref design axial flux shape	1.5 cosine with tails
Hot channel factors	
Enthalpy rise (F_q)	1.011
Heat flux (F''_q)	1.014
Flow area	0.98
Avg heat flux, 100% power, Btu/h-ft^2	1.89×10^5(a)
Max heat flux, 100% power, Btu/h-ft^2	4.85×10^5(a)
CHF correlation	BAW-2
Minimum DNBR (at 112% power)[b]	1.79

(a) With thermally expanded fuel rod OD of 0.43075 inch.
(b) Telecon, G. Bradley, Toledo Edison, to A. DeAgazio, NRC, September 1, 1983.

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16. R. P. Crouse (Toledo Ed) letter to T. M. Novak (NRC) dated October 19, 1982.

17. R. P. Crouse (Toledo Ed) letter to J. F. Stolz (NRC) dated February 17, 1983.

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