DMB 016

December 13, 1984

Docket No. 50-346

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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. 80 TO FACILITY OPERATING LICENSE NO. NPF-3; CYCLE 5 OPERATION

The Commission has issued the enclosed Amendment No. 80 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 20, 1984 (No. 1062).

This amendment modifies the TSs to permit operation for Cycle 5. This cycle has a design length of approximately 390 effective full power days. The modified TSs incorporate revised reactor protection system instrumentation trip setpoints and allowable values, insertion limits for regulating and axial power shaping rods, and power distribution limits.

A copy of the Safety Evaluation supporting this amendment is enclosed. The Notice of Issuance will be included in the Commission's Monthly Notice in the Federal Register.

Sincerely,

Albert W. De Agazio, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 80
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	ADe Agazio;cf	JStolz		GLainas
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Toledo Edison Company

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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. 80
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 20, 1984, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80 , 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 its date of 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: December 13, 1984

ATTACHMENT TO LICENSE AMENDMENT NO 80

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

Insert Page	Remove Page
2-3	2-3
2-5	2-5
2-7	2-7
B 2-2	B 2-2
B 2-5	B 2-5
3/4 1-26	3/4 1-26
3/4 1-28	3/4 1-28
3/4 1-28a	3/4 1-28a
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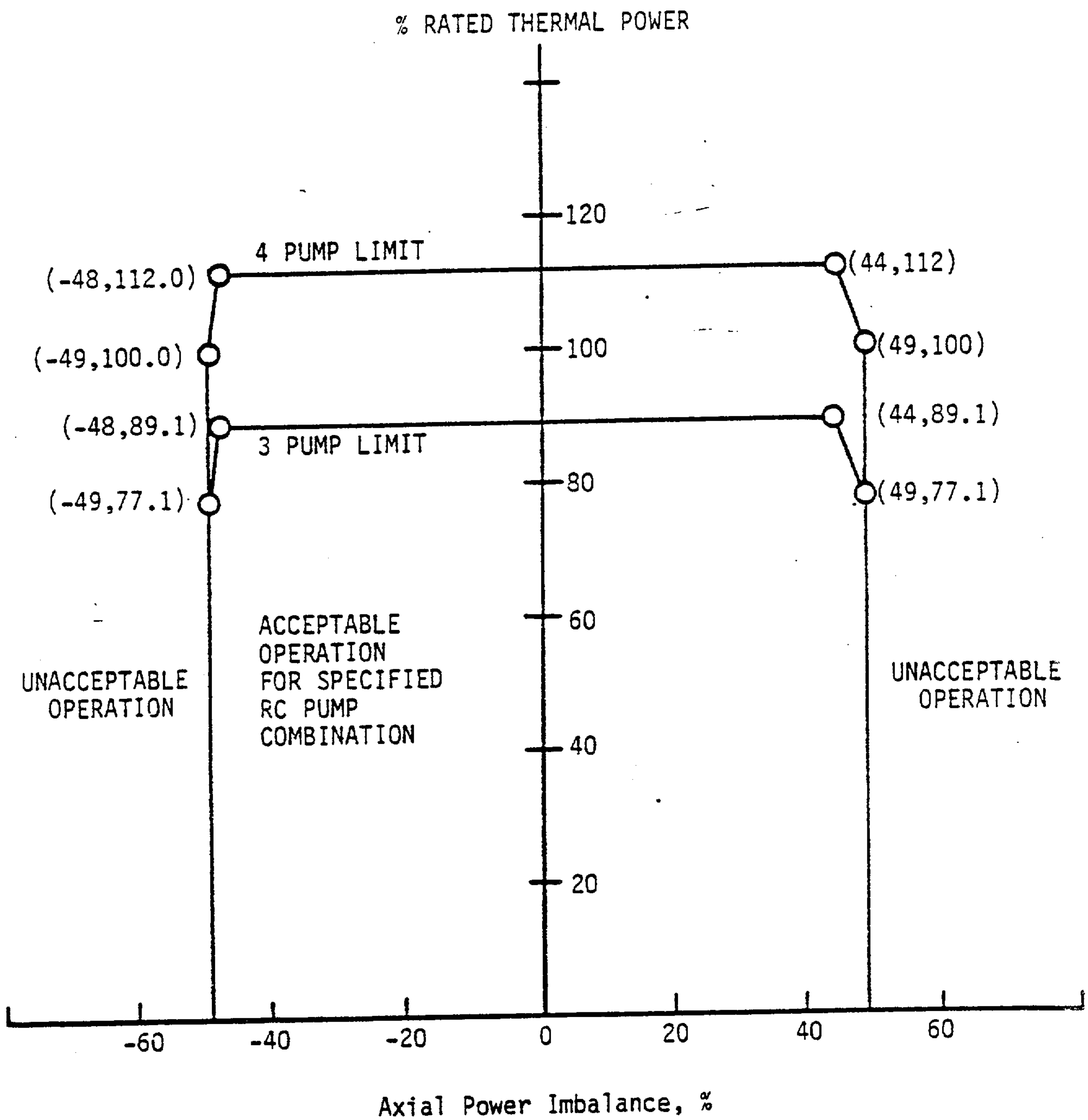
3/4 2-2a	3/4 2-2a
3/4 2-2b	3/4 2-2b
3/4 2-2c	3/4 2-2c
3/4 2-2d	3/4 2-2d
3/4 2-3	3/4 2-3
3/4 2-3a	3/4 2-3a
3/4 2-3b	3/4 2-3b
3/4 2-3c	3/4 2-3c
3/4 2-3d	3/4 2-3d
3/4 2-12	3/4 2-12
3/4 4-1	3/4 4-1

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	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤79.7% of RATED THERMAL POWER with three pumps operating	≤79.7% 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	≤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 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	≤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

Curve shows trip setpoint for a 25% flow reduction for three pump operation (290,100 gpm). The actual setpoint will be directly proportional to the actual flow with three pumps.

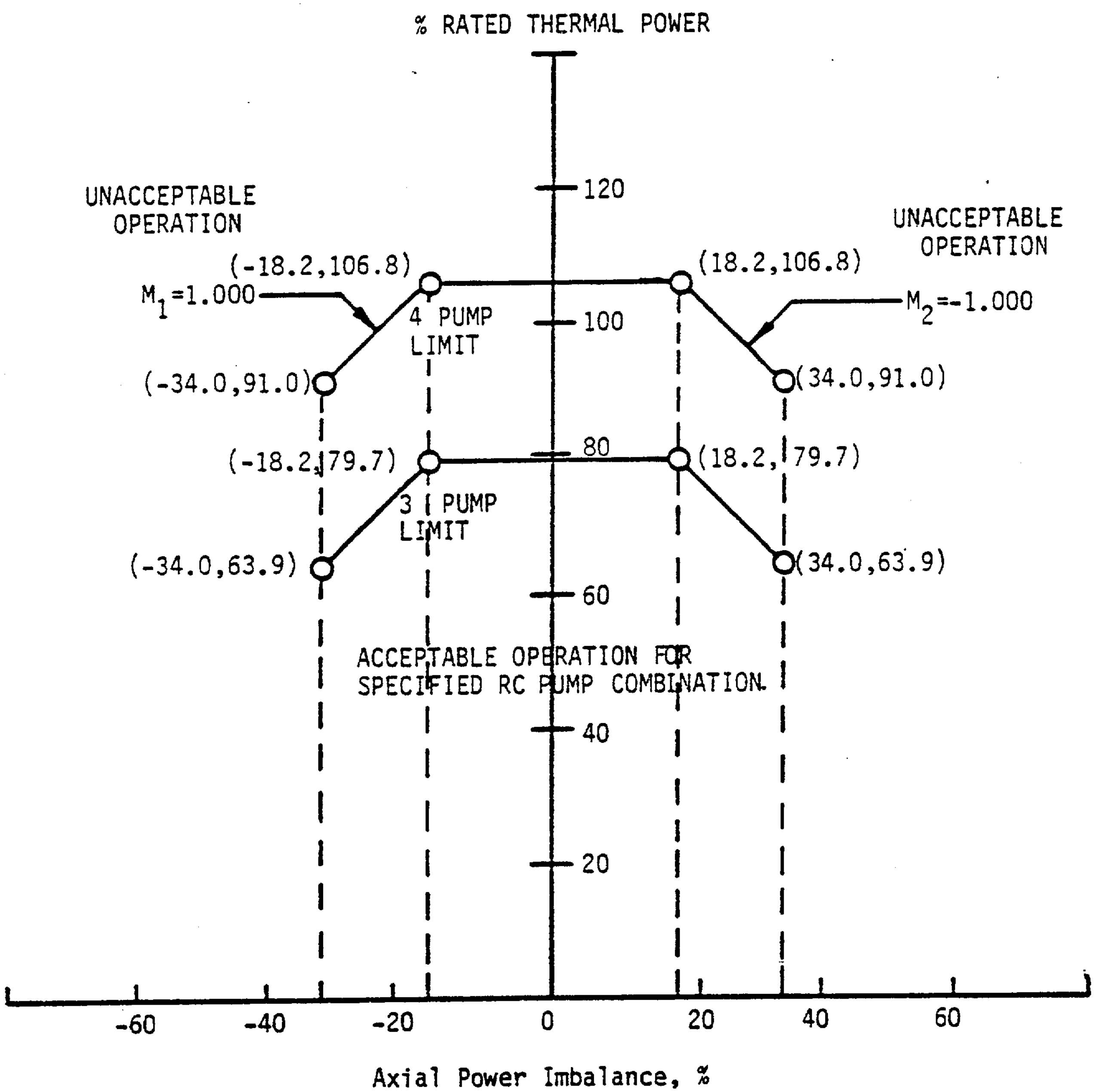


Figure 2.2-2 Allowable Value for Flux-Δ Flux/Flow

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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 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 1E, 4B, and 5A and 20.5 kW/ft for batches 5B, 6, and 7.

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.

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 -- 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 106.8% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 93.63% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.7% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 70.22% 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.

LIMITING SAFETY SYSTEM SETTINGS

BASES

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} - 5662.2) 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, -2b, -2c, and -2d and 3.1-3a, -3b, -3c and -3d. 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 Exception 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 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

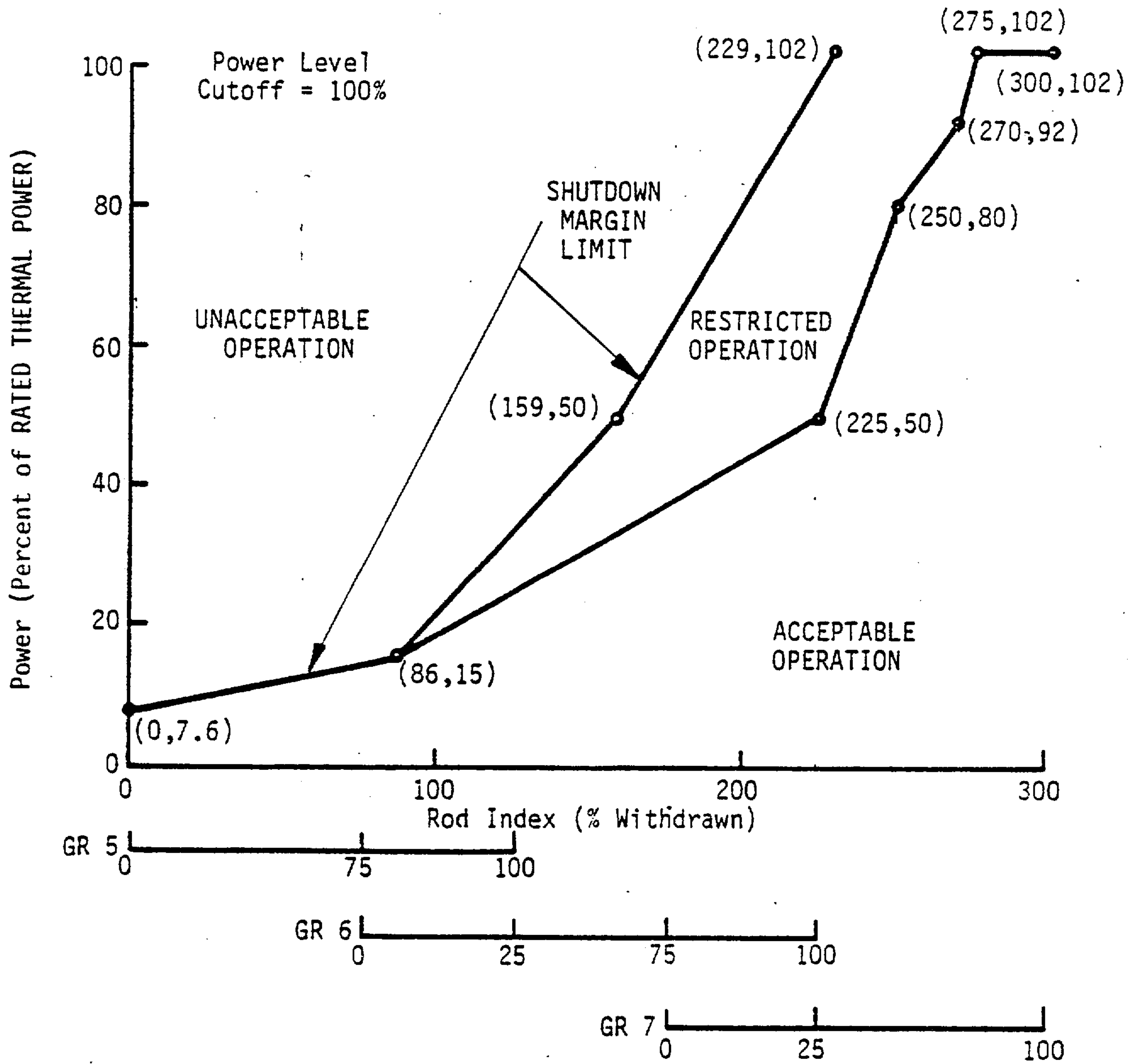

Figure 3.1-2b Regulating Group Position Limits, 25+10/-0 to 200±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

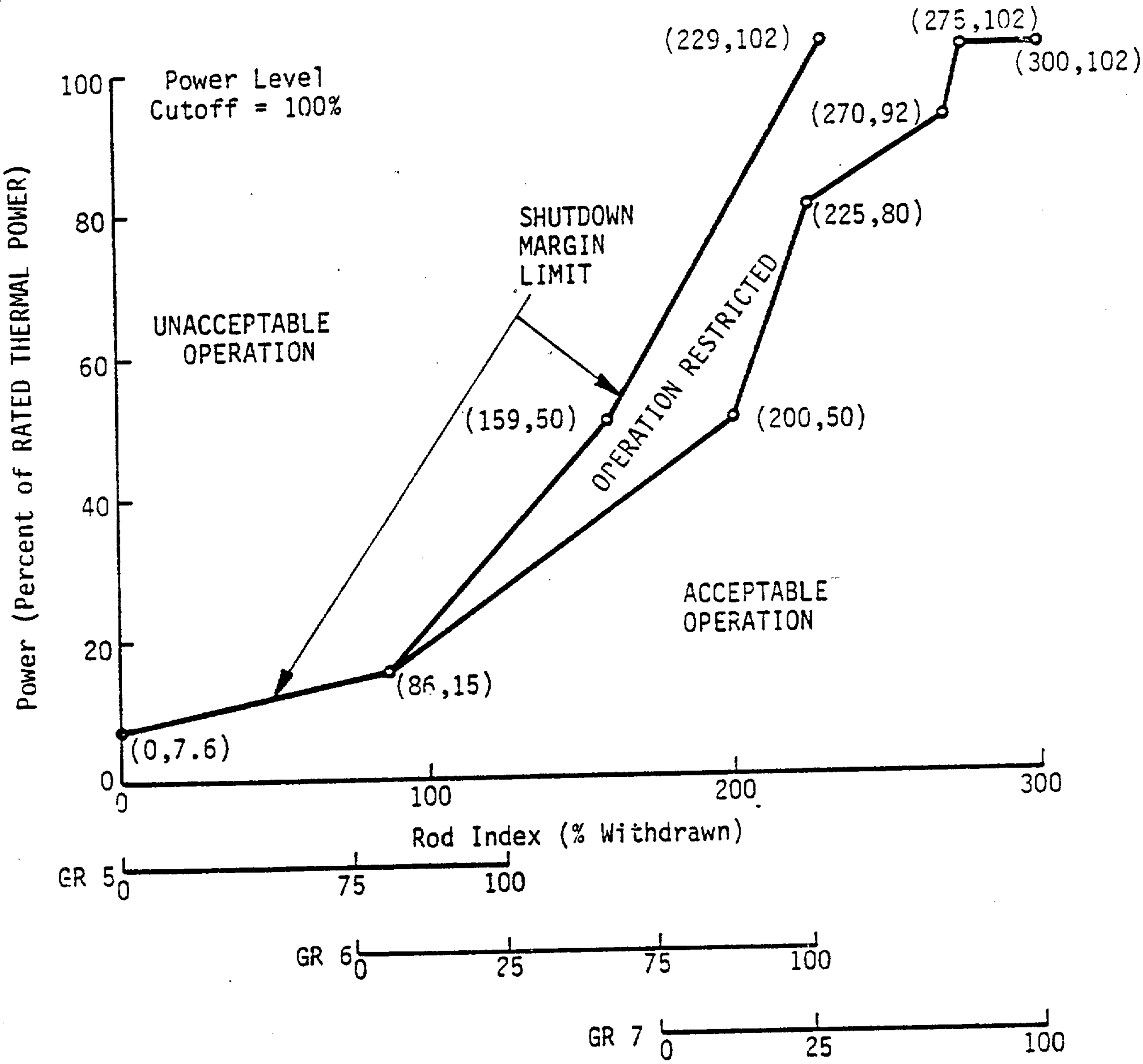

Figure 3.1-2c Regulating Group Position Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

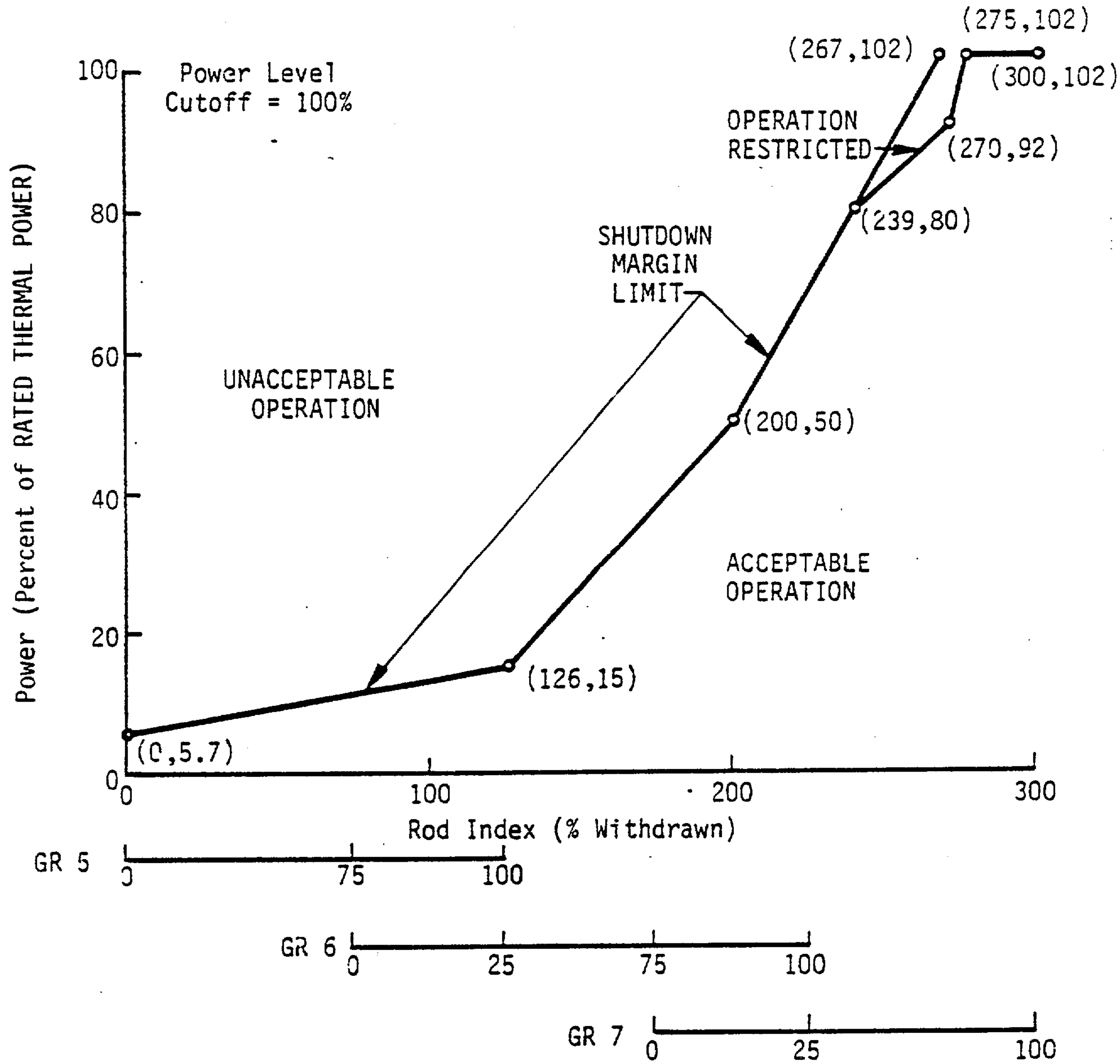

Figure 3.1-2d Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

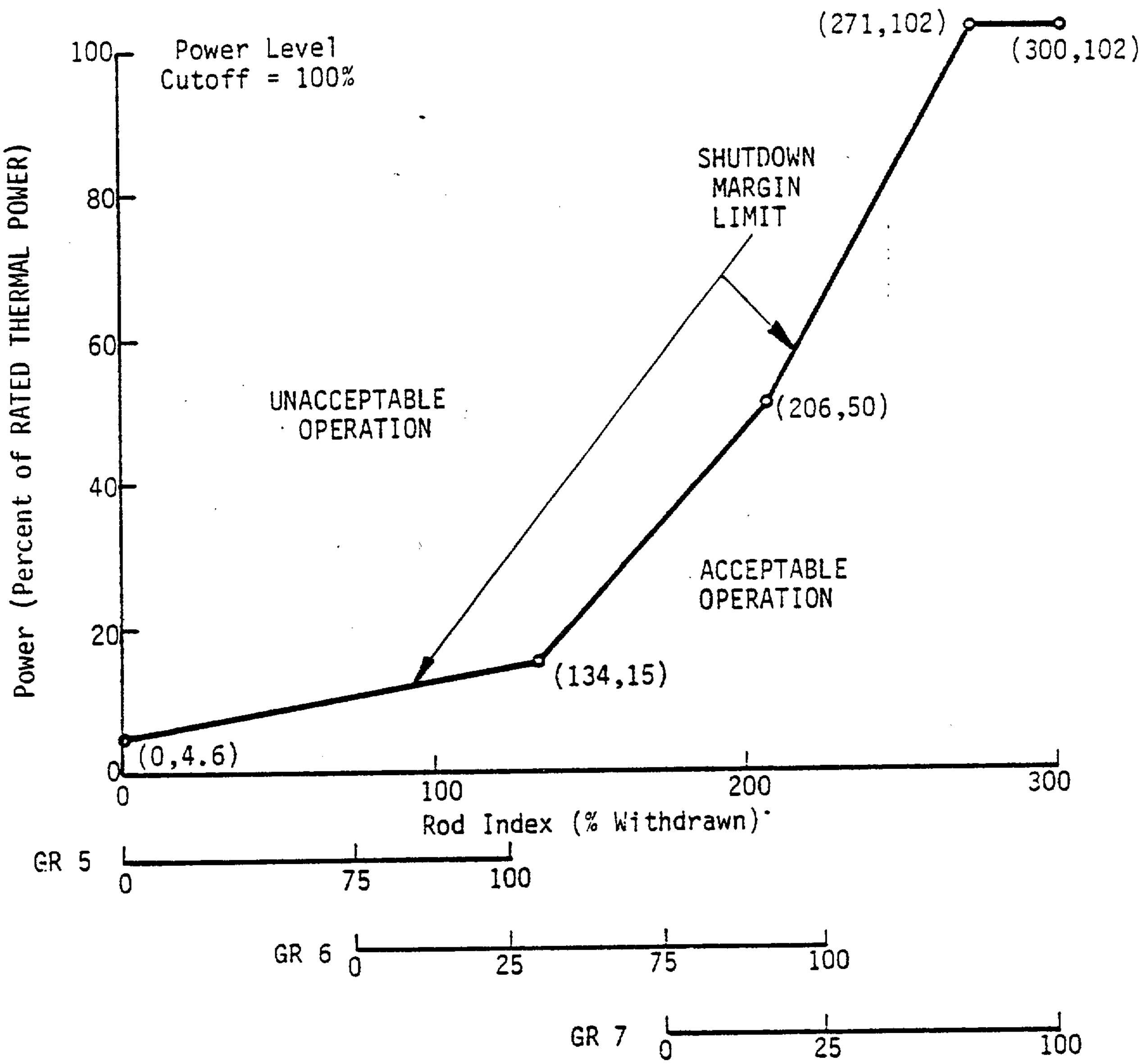

Figure 3.1-2e

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Figure 3.1-3a Regulating Group Position Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

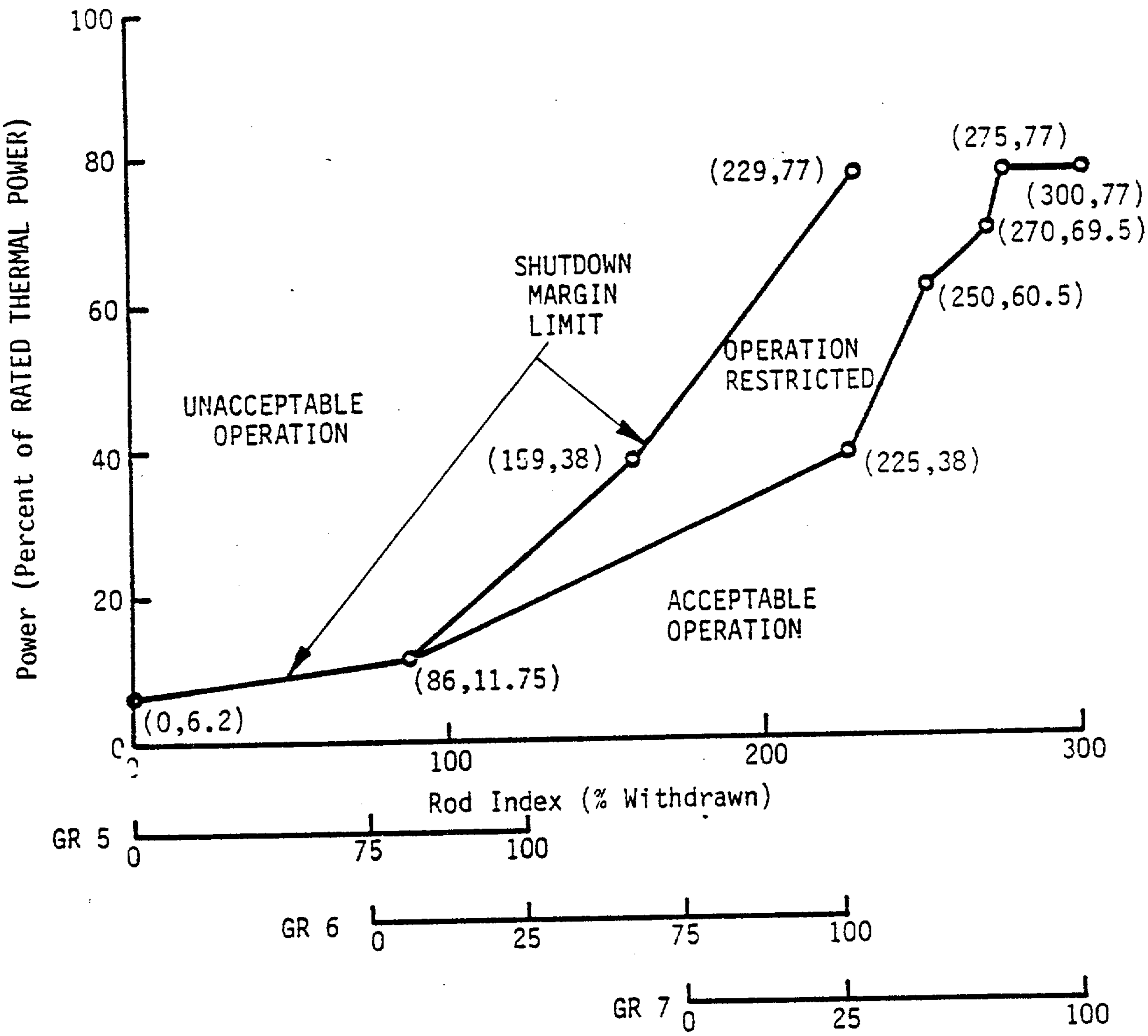

Figure 3.1-3b Regulating Group Position Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

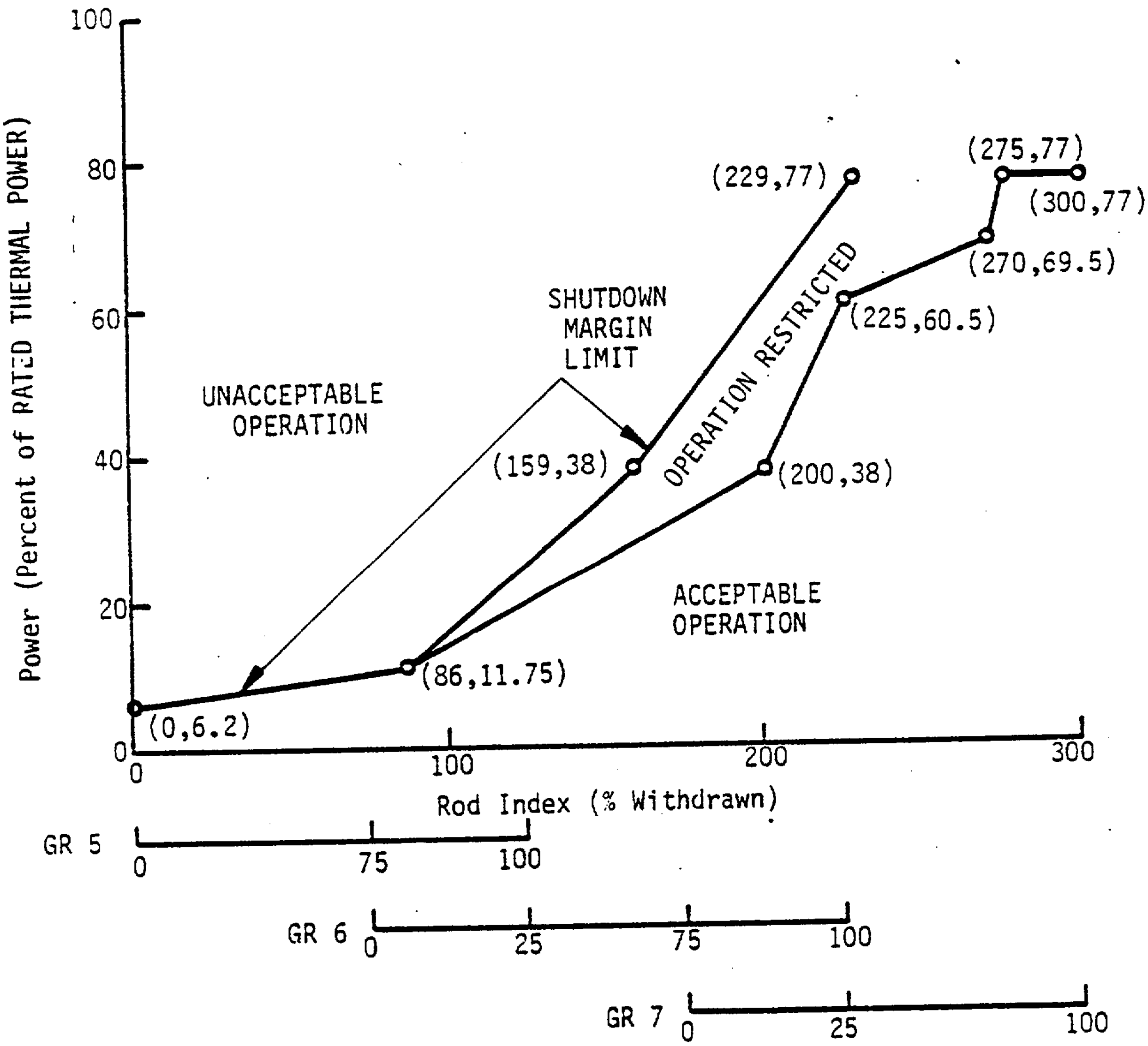

Figure 3.1-3c Regulating Group Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

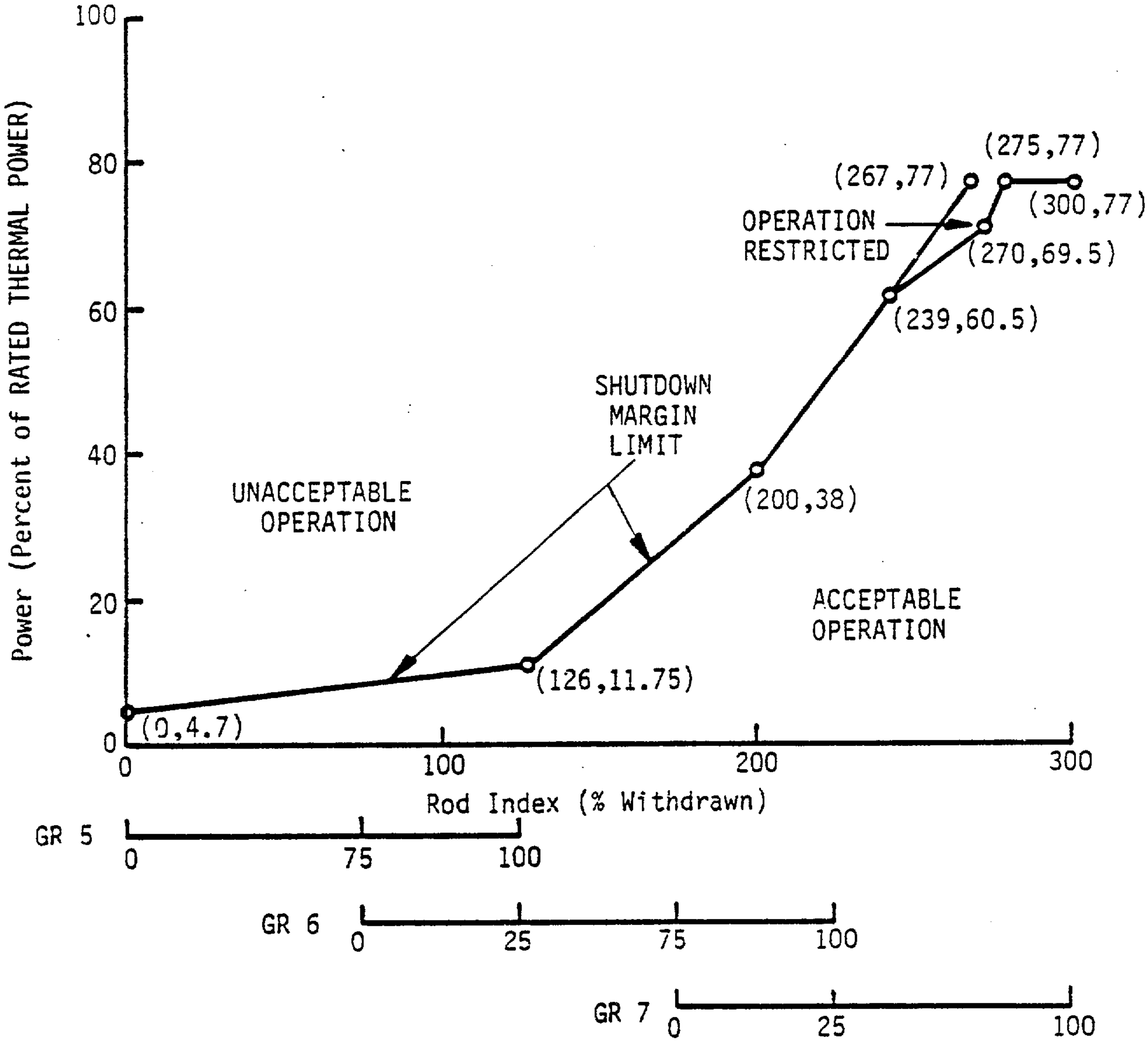

Figure 3.1-3d Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

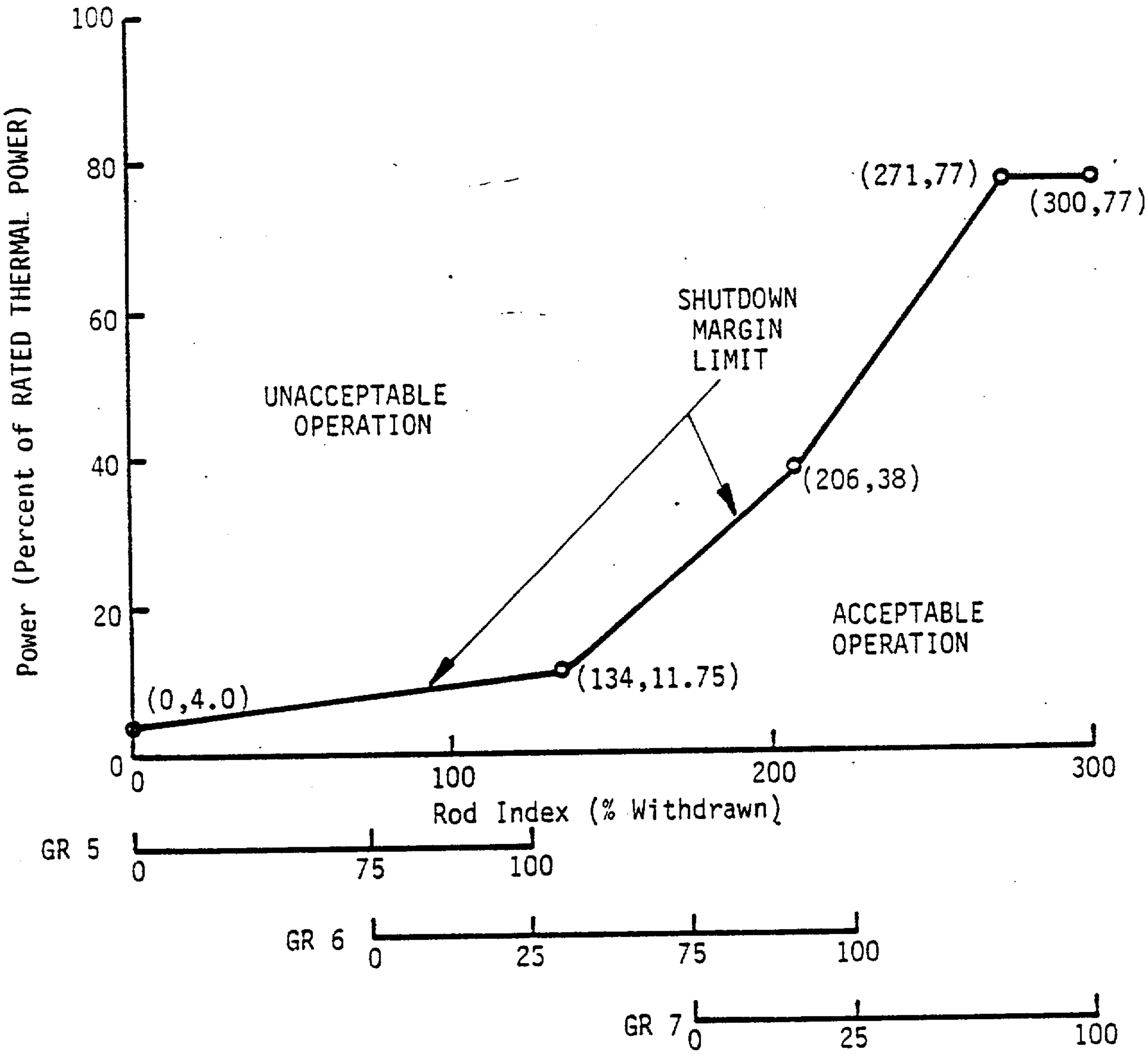

Figure 3.1-3e

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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 5

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																
B						3		7		3						
C					2		6		6		2					
D				7		8		5		8		7				
E			2		5						5		2			
F		3		8		1		7		1		8		3		
G			6				4		4				6			
H W —		7		5		7		1		7		5		7		— Y
K			6				4		4				6			
L		3		8		1		7		1		8		3		
M			2		5						5		2			
N				7		8		5		8		7				
O					2		6		6		2					
P						3		7		3						
R																

Z

$\boxed{X}$ Group Number

Group	No. of rods	Functions
1	5	Safety
2	8	Safety
3	8	Safety
4	4	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, -5d, -5e, -5f, and -5g.

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 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

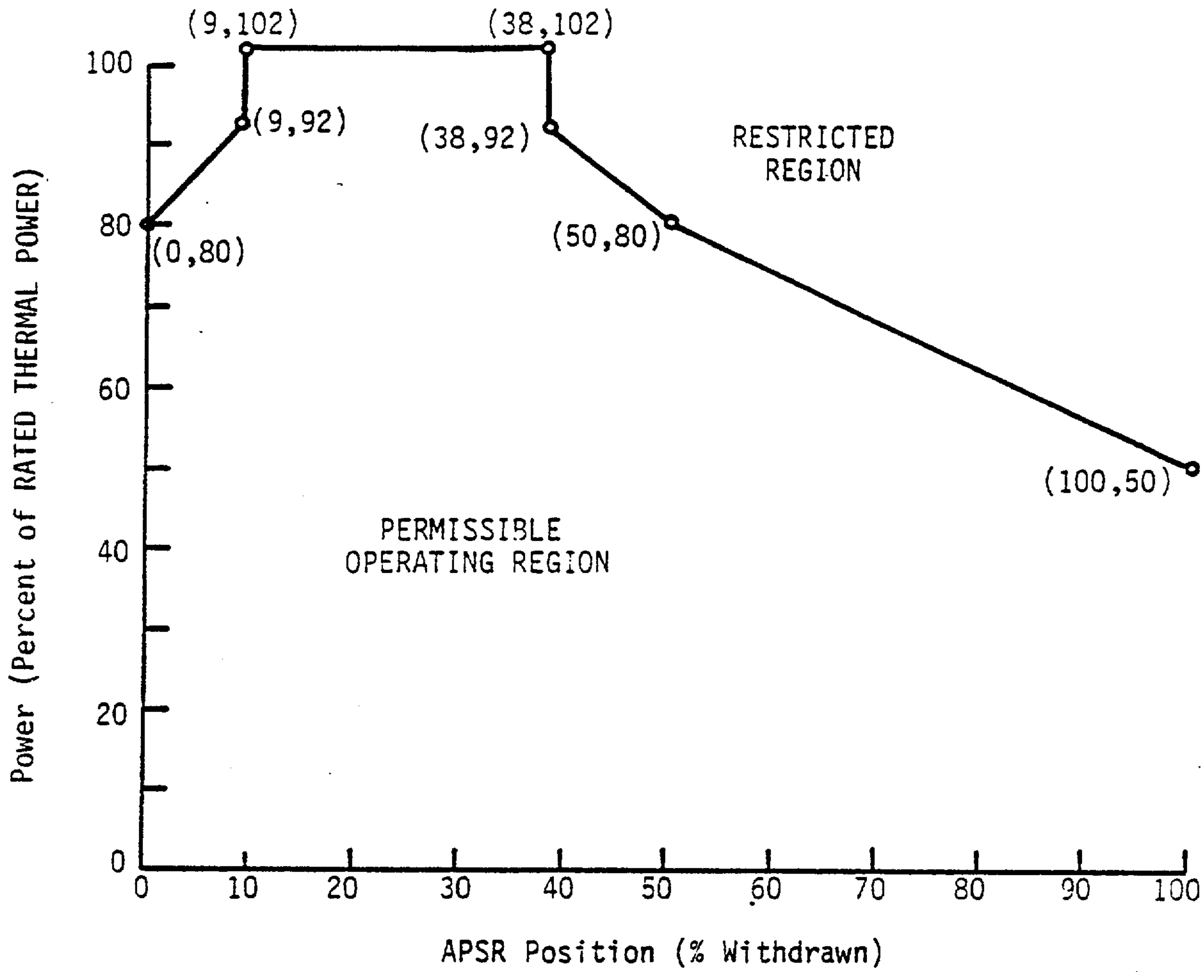

Figure 3.1-5b APSR Position Limits, 25+10/-0 to 200 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

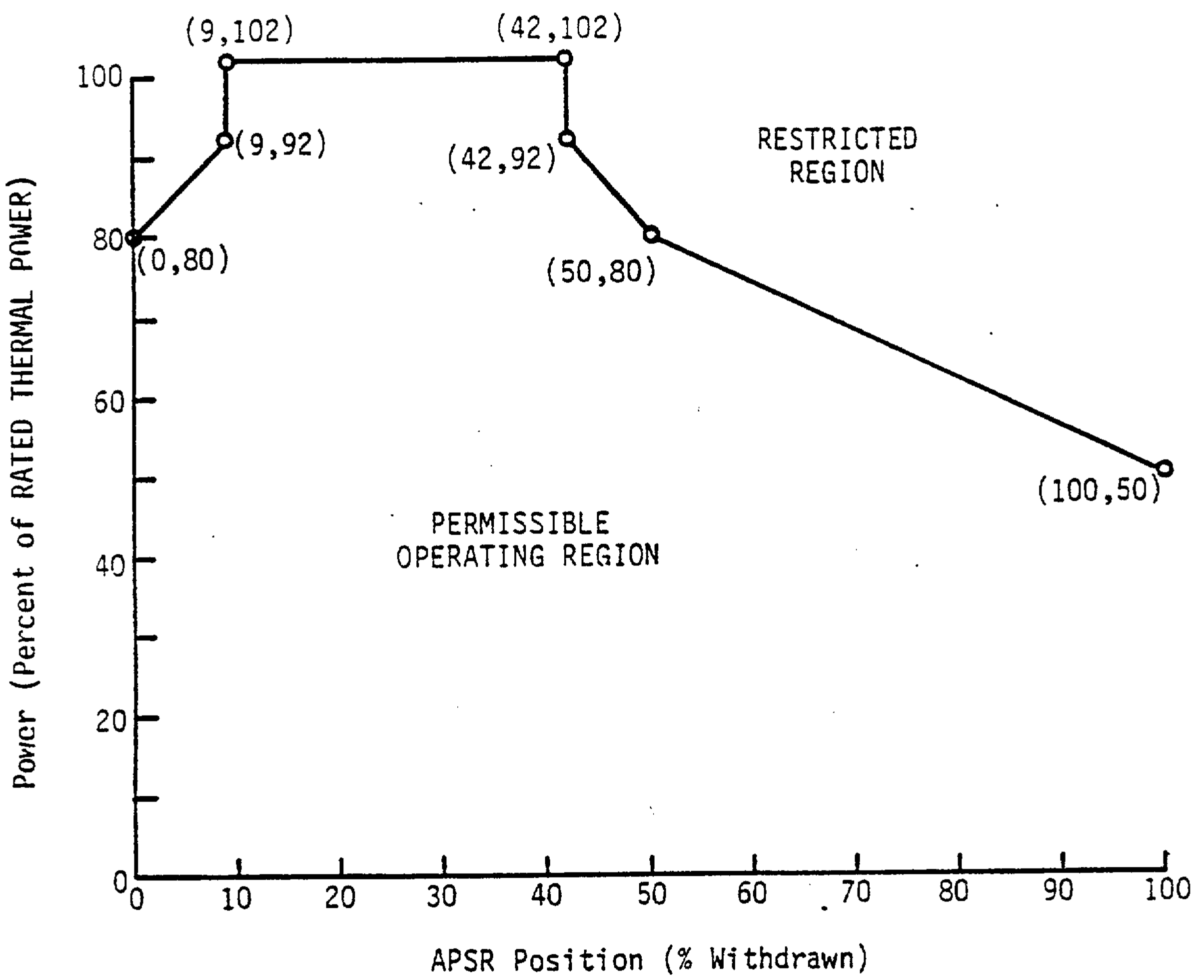

Figure 3.1-5c APSR Position Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

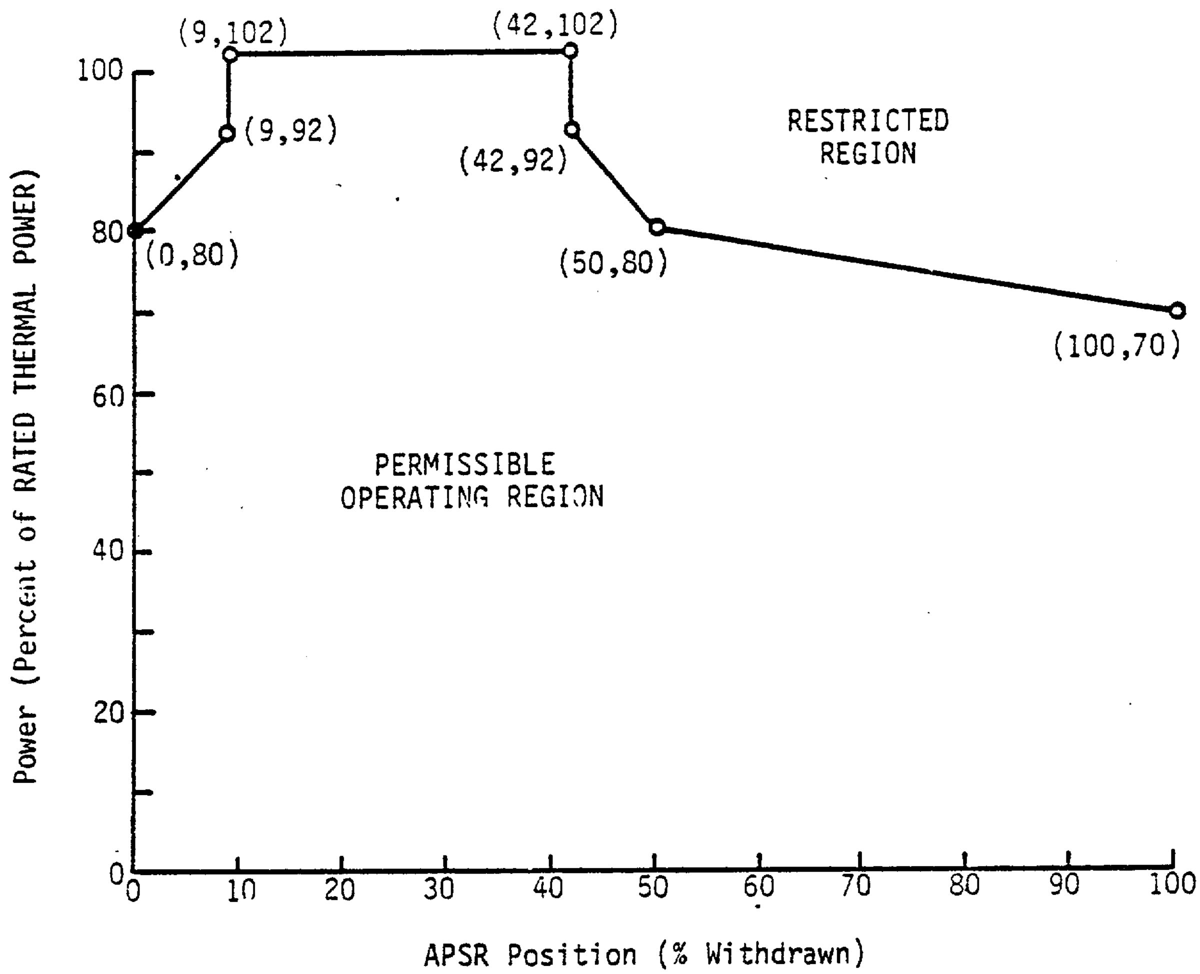

Figure 3.1-5d APSR Position Limits, 330 ±10 to 390 ±10 EFPD, Three or Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

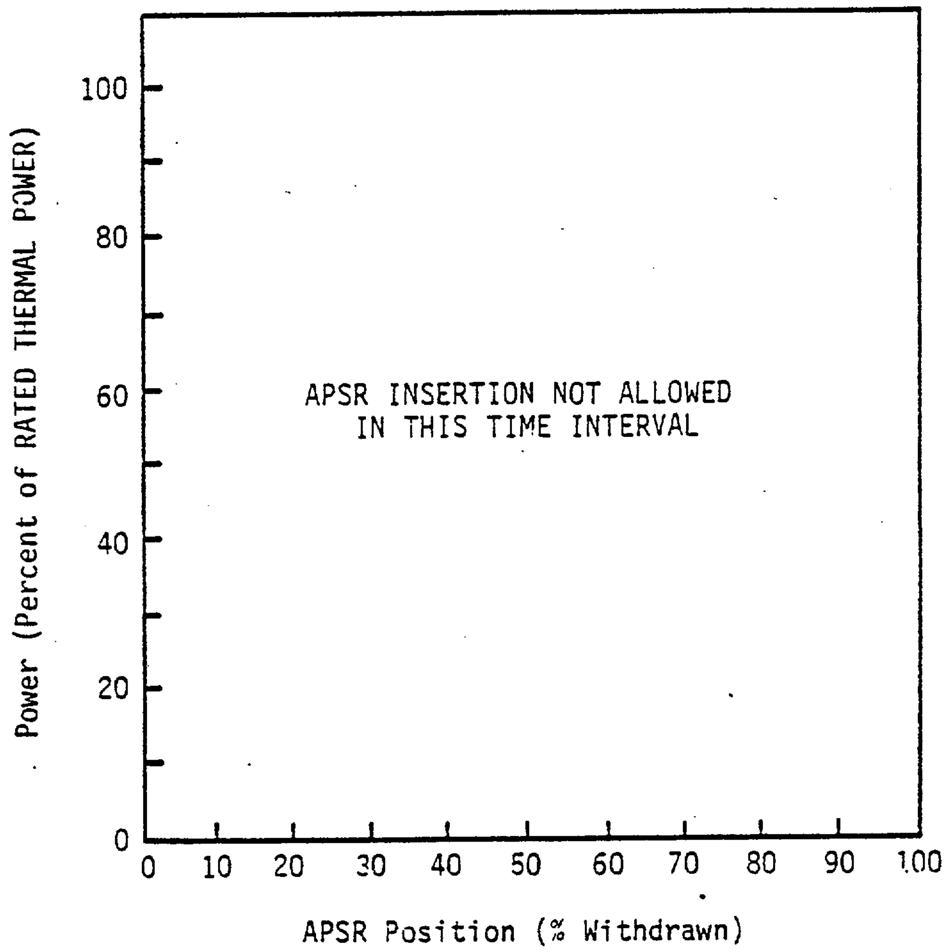

Figure 3.1-5e APSR Position Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

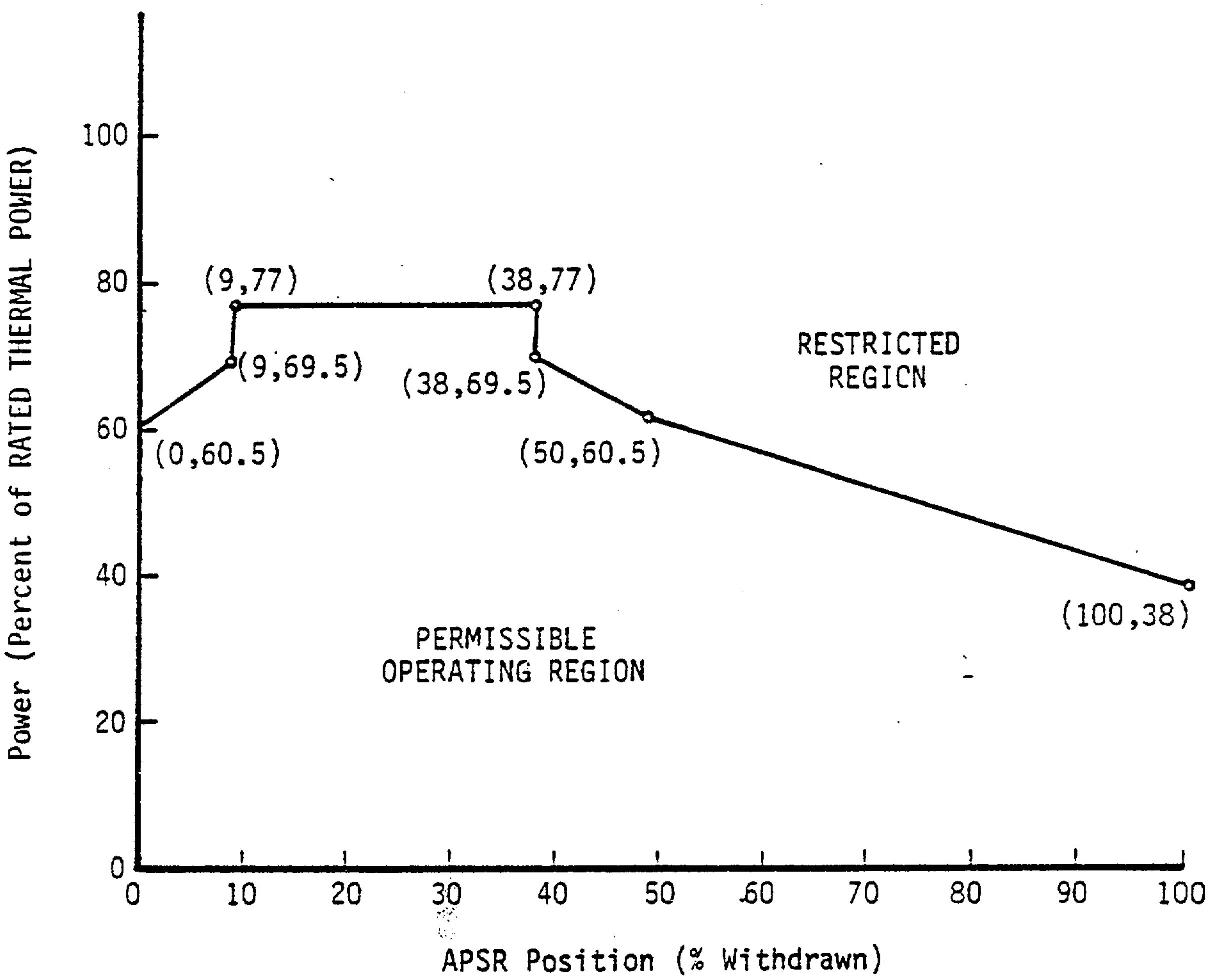

Figure 3.1-5f APSR Position Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

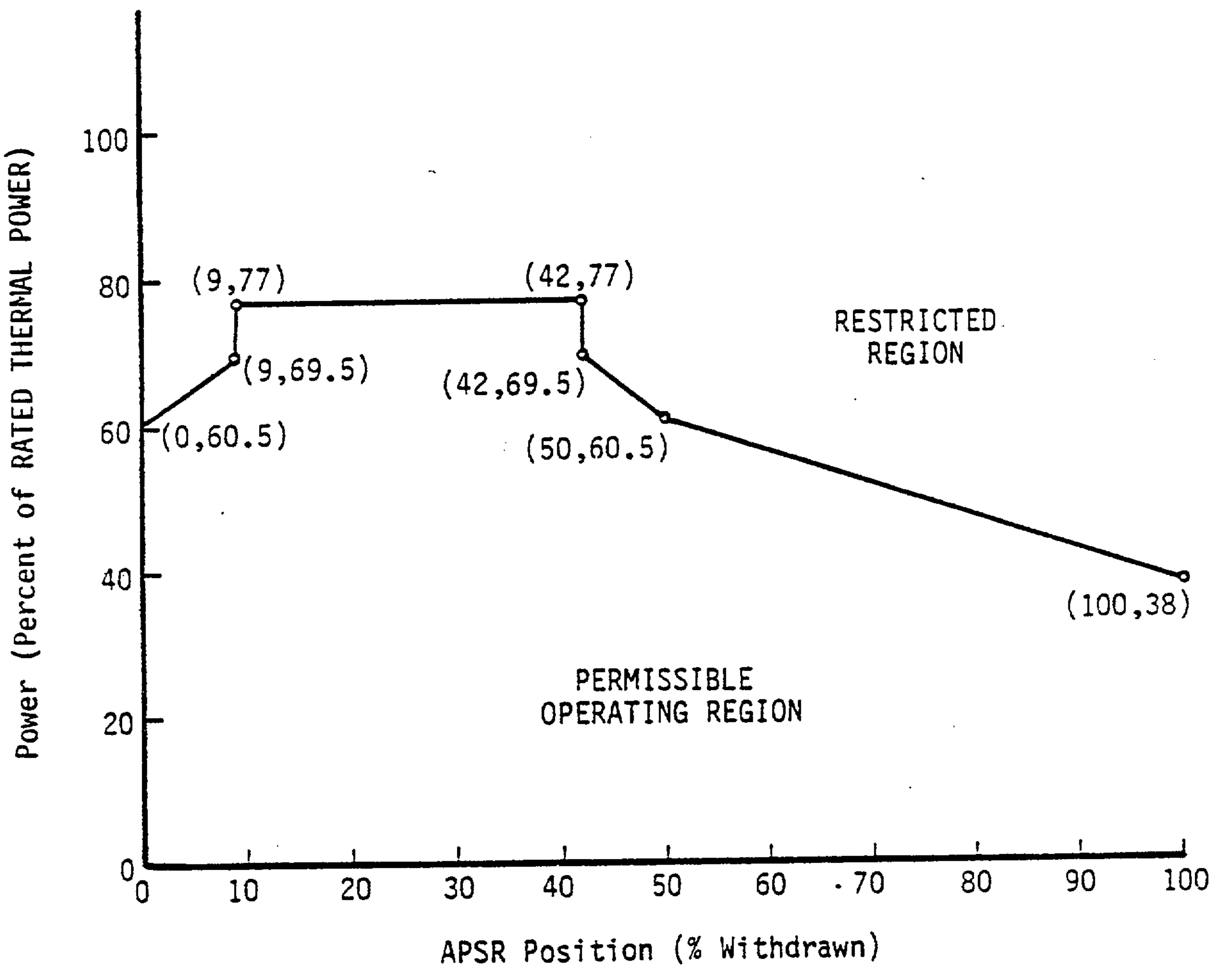

Figure 3.1-5g APSR Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

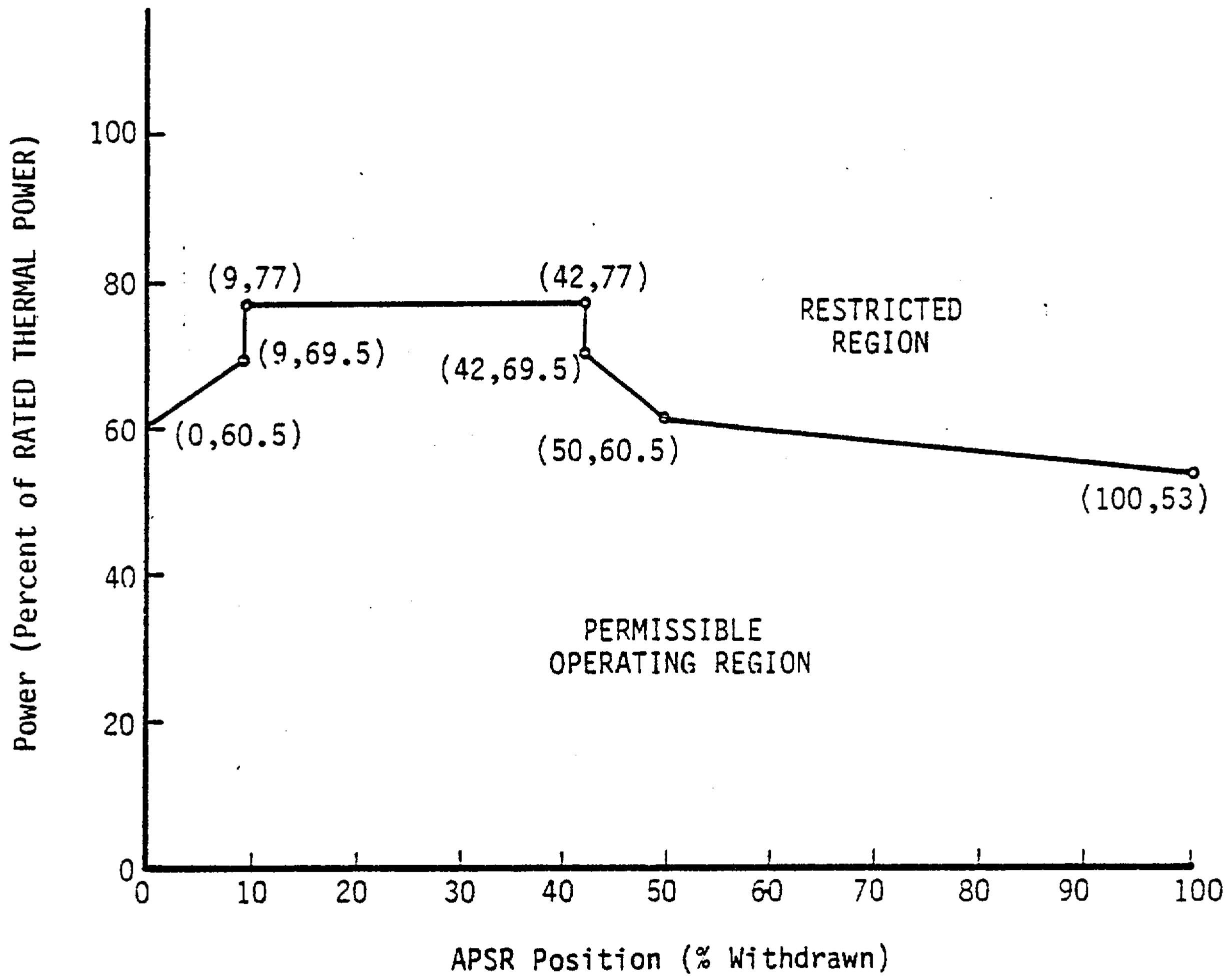

Figure 3.1-5h

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Figure 3.1-5i

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

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 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

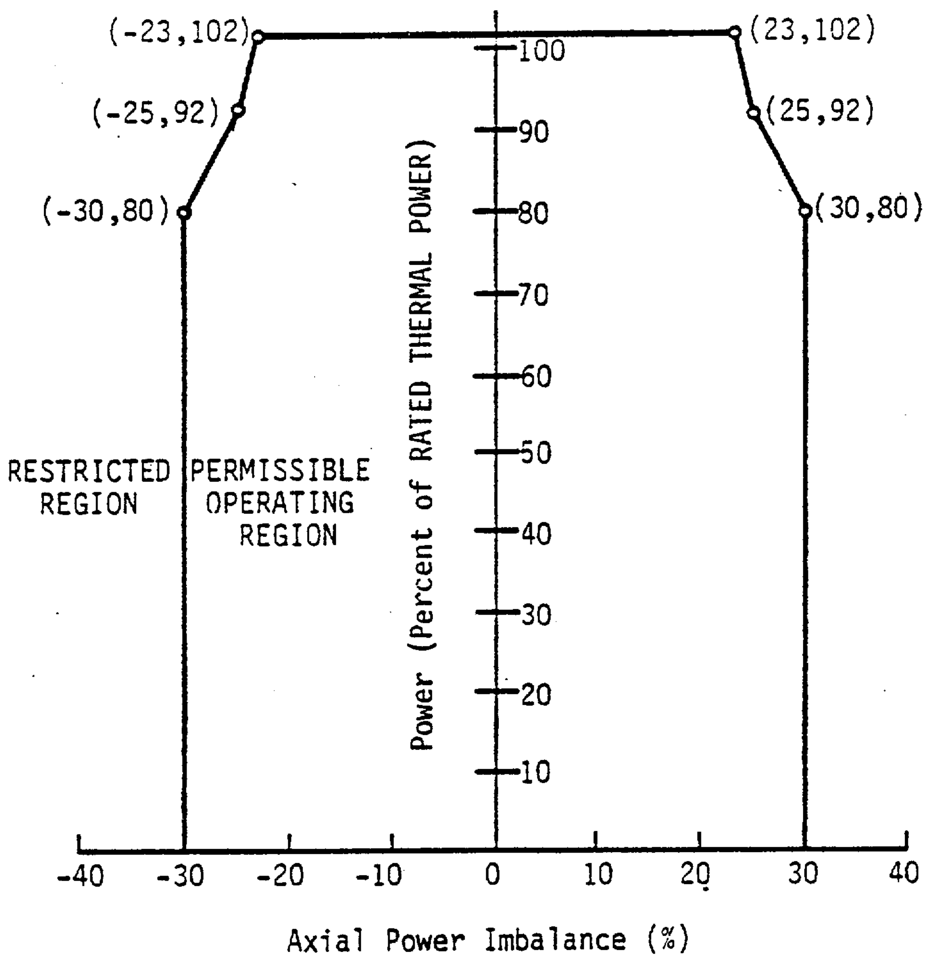

Figure 3.2-1b Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

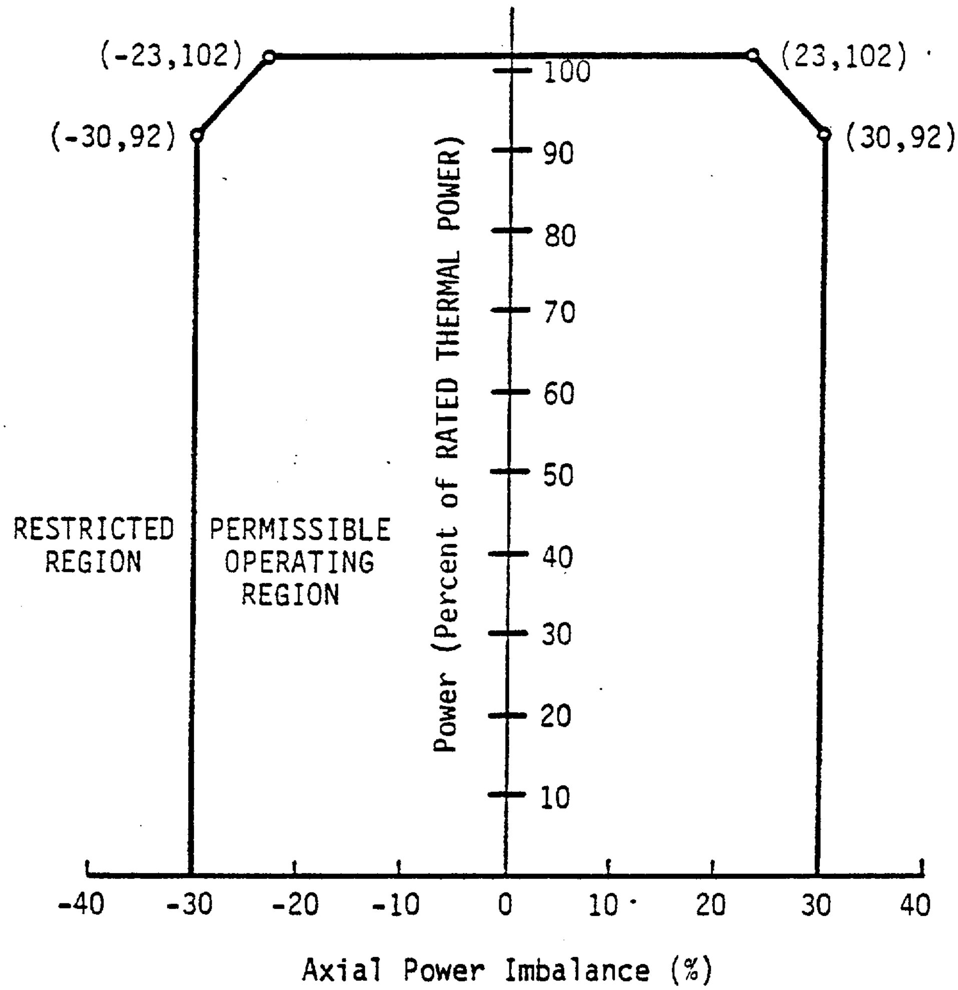

Figure 3.2-1c Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

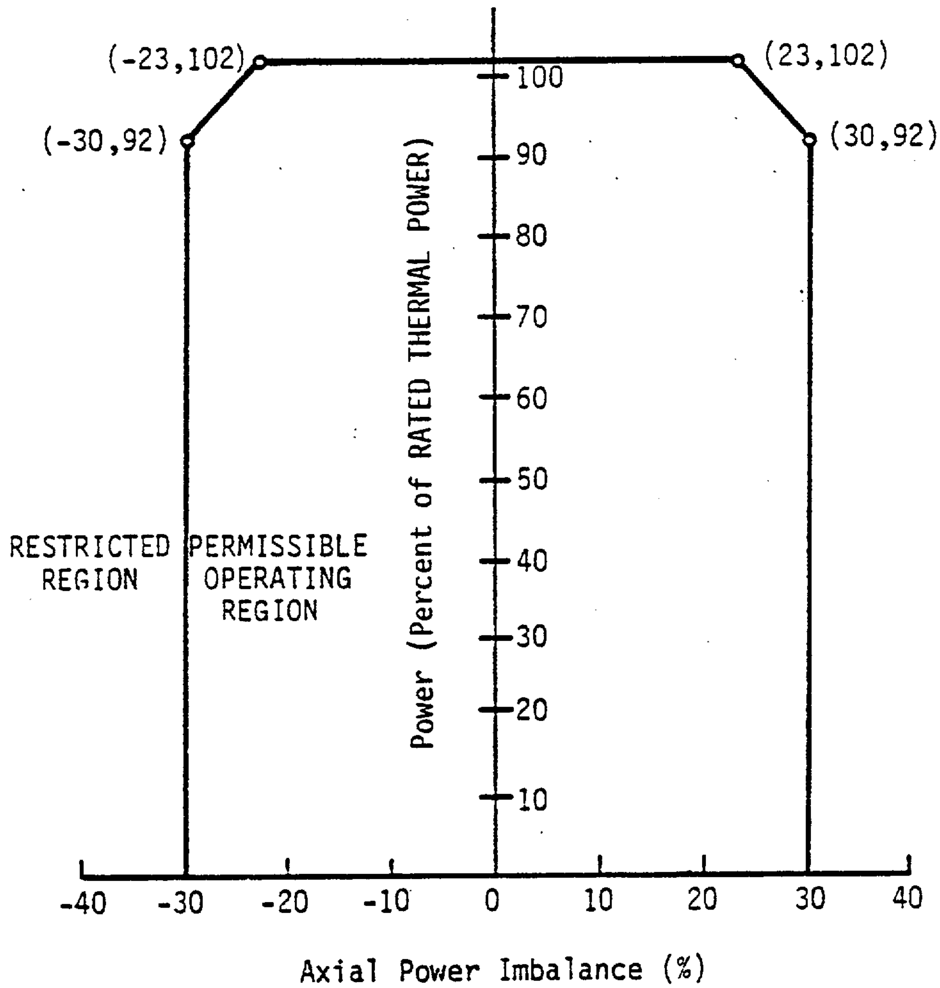

Figure 3.2-1d Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

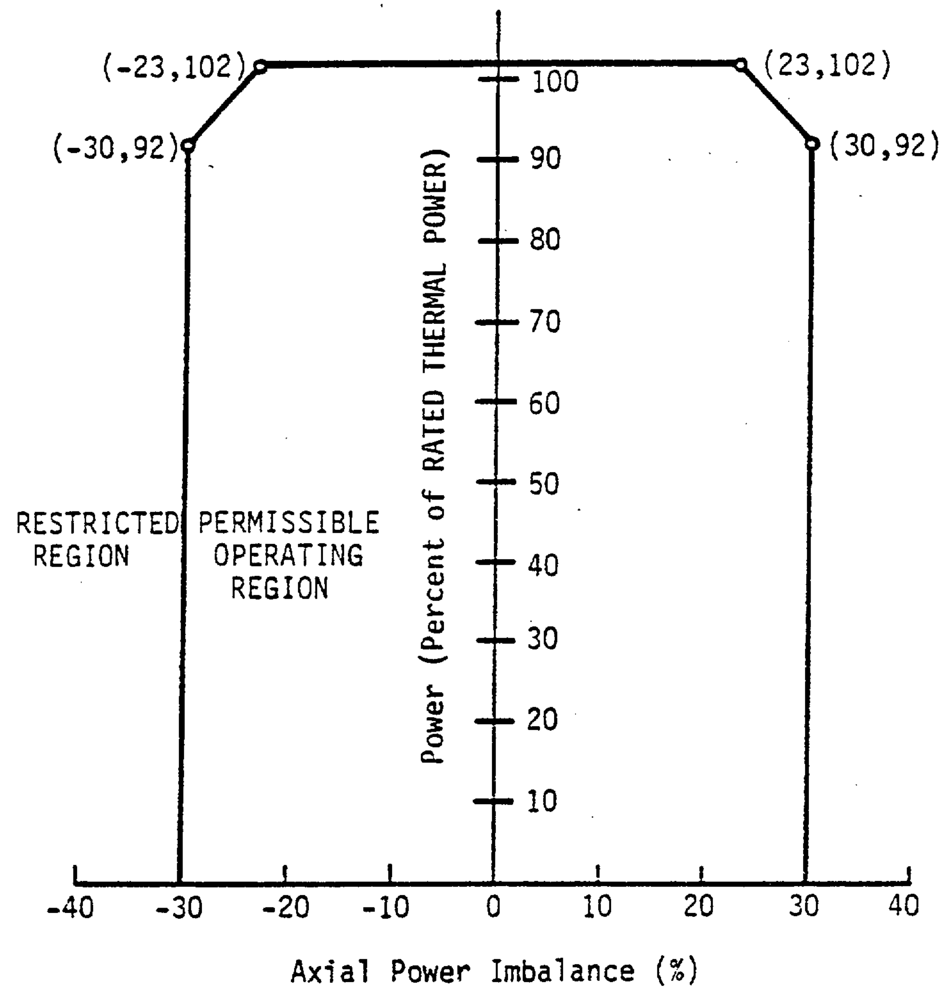

Figure 3.2-1e

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Figure 3.2-2a Axial Power Imbalance Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

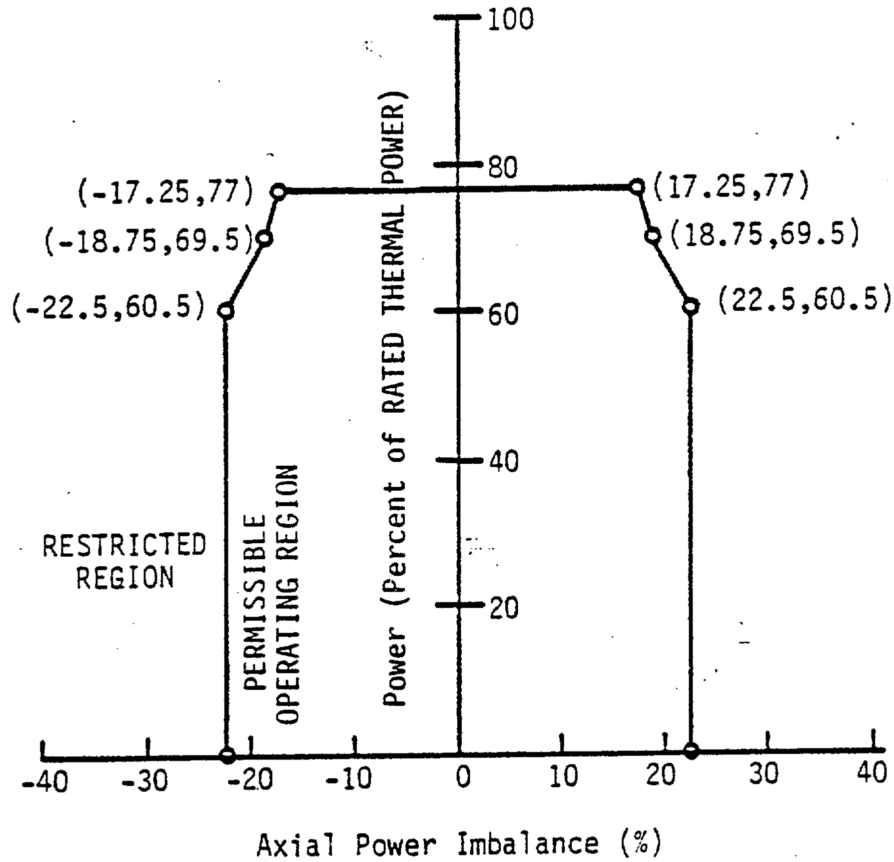

Figure 3.2-2b Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

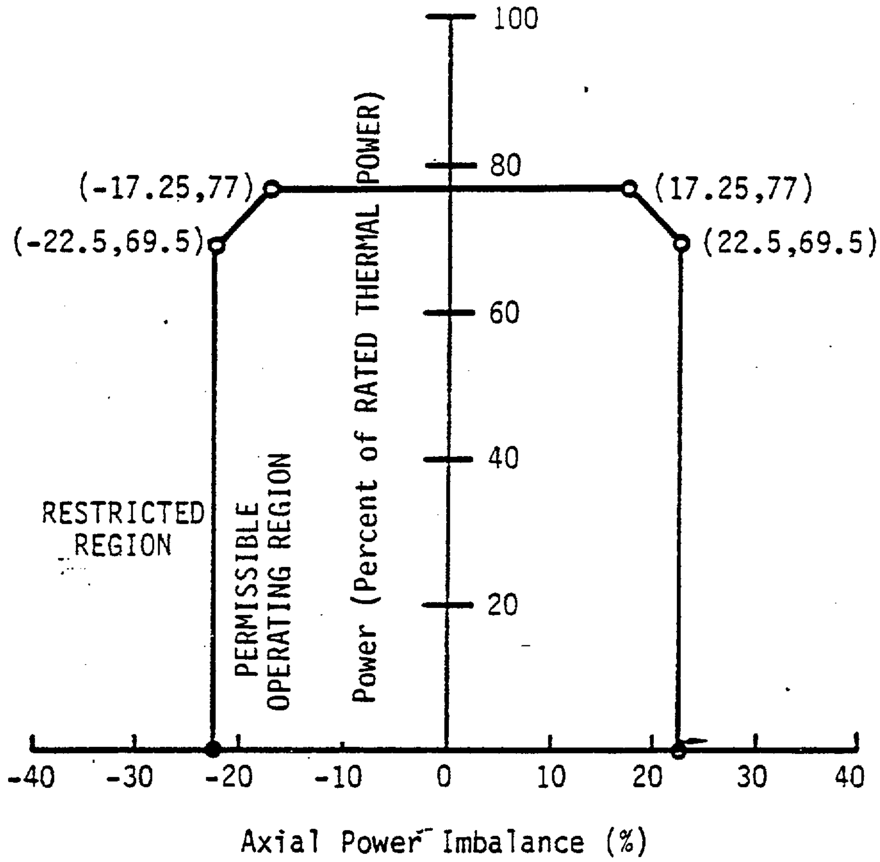

Figure 3.2-2c Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

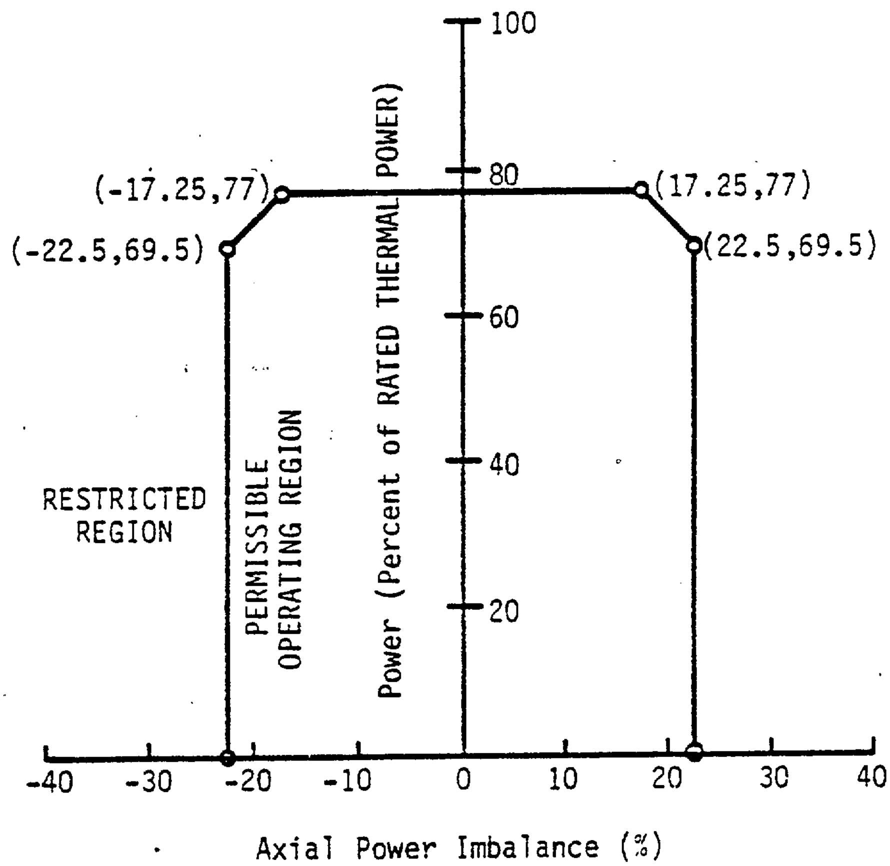

Figure 3.2-2d Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

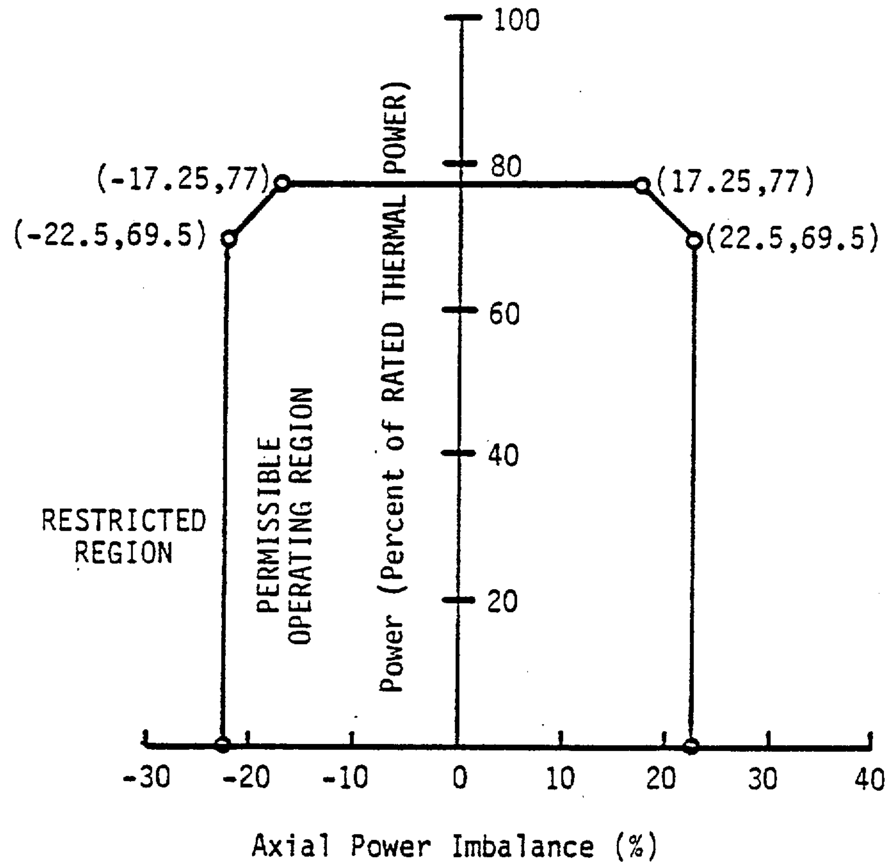

Figure 3.2-2e

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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 ≤ 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, 0-50 ±10 EFPD	3.37	8.52	20.0
Symmetrical incore detector system, after 50 ±10 EFPD	3.02	8.52	20.0
Power range channels	1.96	6.96	20.0
Minimum incore detector system	1.90	4.40	20.0

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 79.7% 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.

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*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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated July 20, 1984 (Ref. 1), Toledo Edison Company made application to modify the Davis-Besse Nuclear Power Station Technical Specifications to permit operation for a fifth cycle. The analysis performed and the resulting modifications to the Technical Specifications are described in the Cycle 5 reload report (Ref. 2). The licensee has used the fourth cycle of operation at Davis-Besse as the reference cycle for the proposed fifth cycle of operation. Where conditions are identical or limiting in the fourth cycle analysis, our previous evaluation (Ref. 3) of that cycle continues to apply.

1.1 Description of the Cycle 5 Core

The Davis-Besse Cycle 5 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. The length of Cycle 5 is expected to be 390 effective full power days (EFPD) of operation. The reference cycle for the nuclear and thermal-hydraulic design of Cycle 5 is Cycle 4 which was scheduled for 280 EFPD. The Cycle 5 design is characterized by only eight fuel assemblies being cross core shuffled so as to minimize any carryover effects from tilts encountered in previous cycles. The licensed power level remains at 2772 MWt.

Cycle 5 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. The Cycle 5 loading will require 64 new fuel assemblies (Batch 7) and the reinsertion of one previously discharged fuel assembly. The 64 new fuel assemblies are fabricated by Babcock and Wilcox (B&W) but contain fuel pellets manufactured by General Electric (GE). Due to the increased length of Cycle 5, additional core reactivity is necessary. This increased reactivity will be controlled in part by 64 burnable poison rod assemblies (BPRAs) located in the fresh fuel. Batch 7 is comprised of the Mark-B5 design which is identical in concept to the Mark-B4 currently used. The only change is to the upper end fitting which has the retention mechanism built in for BPRA holddown. This change is to eliminate the need for retainer assemblies.

2.0 EVALUATION

2.1 Fuel System Design

The 64 BPRA B&W Mark-B5 fuel assemblies loaded as Batch 7 at end of Cycle 4 (EOC 4) are mechanically interchangeable with type Mark-B4 Batches 1E, 4B, 5B and 6 fuel assemblies previously loaded at Davis-Besse. The Mark-B5 upper end fitting provides four open slots that align and guide the movement of the holddown spring, spring retainer, and a new Mark-B5 BPRA spider. The Mark-B5 design also contains a redesigned holddown spring made from Inconel 718 material which provides added margin over the Mark-B4 spring design made from Inconel 750 (Ref. 3). The licensee stated that the Mark-B5 upper end fitting has been tested extensively, both in air and in over 1000 hours of simulated reactor environment, to determine analytical input and to assure good incore performance. The licensee intends to continue visual inspection programs on the new fuel holddown springs (Ref. 4).

The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for Davis-Besse or were analyzed specifically for Cycle 5 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.

The licensee stated (Ref. 4) that there is no change in analysis methodology for fuel rod pin pressure calculations from Cycle 4 to Cycle 5. The licensee further stated that the calculated results show that the fuel rod pressure remains below system pressure at rod exposure up to 45,000 MWd/MTU. We find this acceptable.

For the LOCA analysis (Section 7.2 of Ref. 2) the volume-averaged fuel temperature and fuel rod internal pressure were calculated for Cycle 5 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.

The licensee has stated that the analytical methods which were used and accepted for Cycle 4 reload have also been used to support the proposed amendment. These methods (Ref. 5), including the TACO-2 fuel performance code and the revised cladding models in the Emergency Core Cooling System (ECCS) code package, do not differ from the analytical methods used and accepted for previous cores to demonstrate conformance to acceptance criteria and NRC regulations. The approved TACO-2 code is used to determine the margin for centerline melting and other design calculations for fuel Batches 5B, 6 and 7. The ECCS analysis utilizes the TACO-2 code and incorporates cladding rupture, strain, and flow blockage models based upon data presented in NUREG-0630 (Ref. 6).

2.2 Nuclear Design

To support Cycle 5 operation of Davis-Besse, 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 4 and 5 as calculated with these techniques. We reviewed the characteristics compared to previous cycles, and find them acceptable for use in the Cycle 5 accident and transient analysis, as described in Section 2.4 of this evaluation.

The Cycle 5 design cycle length is 390 days, whereas the Cycle 4 design length was 280 days. The licensee stated that the analytical methods are the same for Cycle 5 as for the reference Cycle 4. The changes in the Cycle 5 physics parameters reflect the change in core loading philosophy. In going to 18-month cycles, the transition to a low leakage core was incorporated. This scheme loads the fresh fuel in a checkerboard pattern with the twice burned fuel in the core interior and loads the once burned fuel on the core periphery. This scheme and the use of the BPRAs produces a flatter radial power distribution causing the changes in reactivity when compared to Cycle 4. No significant operational or procedural changes exist for Cycle 5 with regard to axial or radial power shape, xenon, or tilt control. The Cycle 5 exposure dependent Quadrant Power Tilt limit as presented in Table 8-2 of Ref. 2 was used in the analysis. This shows that the Beginning of Cycle (BOC) steady state Quadrant Power Tilt limit using the incore detector system must be updated at 50±10 EFPD.

Due to the differences in design cycle lengths, the critical boron concentrations for Cycle 5 differ from those of Cycle 4. Because of different isotopic distributions, Cycle 5 control rod worths, ejected rod worths, and stuck rod worths differ from those of Cycle 4. The licensee took into account ejected rod worths and their adherence to shutdown margin requirements in the development of rod position limits for Cycle 5. 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 margin in excess of requirements.

We, therefore, conclude that the licensee has demonstrated adequate provision of shutdown margin for Cycle 5. In addition, control rod worth measurements are made during startup tests. These will confirm the adequacy of predicted control rod worths.

We find the nuclear design of Cycle 5 to be acceptable.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic performance for Cycle 5, in which the fresh Batch 7 fuel is hydraulically and geometrically similar to the other fuel in the Cycle 5 core, is identical to that of Cycle 4. The introduction of the Mark-B5 upper end fitting does not affect either the core flow rate or the thermal-hydraulic performance. The introduction of BPRAs increases the core flow available for heat transfer by reducing the core bypass flow rate from 10.7 to 8.1%. This reduced bypass flow rate conservatively has been neglected for Cycle 5. The thermal-hydraulic design evaluation supporting Cycle 5 operation is based on the methods and models previously used in Cycle 4 as described in References 8 and 9. The design conditions are given in Table 1 and are identical for Cycles 4 and 5.

A rod bow topical report (Ref. 7) was submitted and approved (Ref. 8) before the last fuel cycle. This report addressed the mechanisms and resulting local conditions of the rod bow. The conclusion was that the 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 5.

The flux/flow trip setpoint for Cycle 5 has been established as 1.068 and was 1.069 for Cycle 4. This setpoint and other plant operating limits are based on the design minimum DNBR limit of 1.30 calculated using the BAW-2 correlation. It is noted that the design flow for the reload analysis is 387,200 gpm which is 110% of the design reactor coolant system flow. The latest measured reactor coolant system flow was 404,308 gpm (Ref. 4) which provides a 4.4% margin of flow.

The minimum DNBR at 112 percent of full power is 1.79 for Cycle 5 which is the same as for Cycle 4. We find that the thermal-hydraulic design is acceptable since the Cycle 5 and Cycle 4 (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 5 values of these parameters with respect to acceptable previous cycle values, that transients and accidents for Cycle 5 are bounded by previously accepted analyses.

The licensee stated that each FSAR accident analysis was examined with respect to changes in the Cycle 5 parameters to determine the effects of the Cycle 5 reload and to ensure that thermal performance during anticipated transients is not degraded. A generic loss-of-coolant accident (LOCA) analysis for B&W 177 fuel assembly raised-loop nuclear steam systems (NSSs) was performed by the licensee using the Final Acceptance Criteria

ECCS Evaluation Modes (Ref. 9). The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used in the LOCA limits analysis was found to be conservative compared to those calculated for this reload.

The licensee's accident and transient analysis, reported in Section 7 of Ref. 2, was reviewed and found to have no significant differences from the previously accepted analysis presented for Cycle 4, with the exception of considering the effect of higher burnup on rod internal pressure changes and release of volatile fission products into the pellet-clad gap.

To assess the effect of higher burnup, we evaluated, independently and in accordance with the methodology of Regulatory Guide 1.25, the doses from a postulated fuel handling accident inside containment. Even though the conditions at the end of Cycle 5 will be beyond the bases stated in the Guide, this methodology continues to be conservative if the effect of higher burnup on the rod internal pressure and on the fraction of volatile radioactive fission products in the pellet-clad gap of the highest power assembly is considered appropriately. Ref. 2 shows that the highest power assembly is a freshly exposed Batch 7 assembly. Therefore, the case to be considered is an assembly at about 13,100 MWd/MTU at the highest allowable linear heat generation rate, 18.4 KW/ft. The assumptions used by the NRC staff and the results of the calculation are given in Table 2. The results show that the fuel handling delay to 72 hours from shutdown and site related parameters are adequate to mitigate the consequences of this accident.

The licensee and the NRC staff have considered the factors dependent upon power level (2772 MWt) and burnup (peak assembly discharge exposure of 41,000 MWd/MTU) that impact the radiological consequences of accidents. We find that operation for Cycle 5 with the extended burnup described in the licensee's application is acceptable.

2.5 Technical Specification Modifications

The pertinent Technical Specifications have been revised for Cycle 5 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 acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need 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.

Dated: December 13, 1984

Principal Contributors: H. Balukjian, G. Suh

REFERENCES

1. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated July 20, 1984.

2. "Davis-Besse Nuclear Power Station Unit 1, Cycle 5 - Reload Report," Babcock & Wilcox Company, BAW-1827 (April 1984). Attachment to Reference 1 above.

3. Letter J. F. Stolz (NRC) to R. P. Crouse (Toledo Edison Company), "Amendment No. 61 to Facility Operating License No. NPF-3; Cycle 4 Operation," dated September 21, 1983.

4. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated October 19, 1984.

5. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated May 6, 1983 transmitting "Bounding Analytical Assessment of NUREG-0630 on LOCA and kW/ft Limits," B&W Document No. 77-1142162-00.

6. D. A. Powers and R. O. Meyer, "Cladding Swelling Models for LOCA Analysis," U.S. Nuclear Regulatory Commission Report NUREG-0630 (April 1980).

7. "Fuel Rod Bowing in Babcock and Wilcox Fuel Design," Babcock and Wilcox Company, BAW-10147P (Proprietary) (April 1981).

8. Letter C. O. Thomas (NRC) to J. H. Taylor (B&W) dated February 15, 1983.

9. "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS", BAW 10105, Rev. 1, Babcock and Wilcox Company (July 1975).

TABLE 1

DAVIS-BESSE CYCLES 4 AND 5

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
Average 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. De Agazio, NRC, September 1, 1983.

TABLE 2

CALCULATION OF THE FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Power level	2772 MW_t	
Peaking factor	2.8	
Fuel failures	1 module of 177	
Fractional release of volatiles to environment before containment isolation	20 percent	
Shutdown time	72 hours	
Atmospheric Diffusion and Transport Relative Concentration, X/Q (sec/m^3)		
Exclusion Area Boundary	0-2 hours	2.2×10^{-4}
Low Population Zone	0-8 hours	9.6×10^{-6}
Doses (Rem)	Thyroid	Whole Body
EAB	21	.4
LPZ	0.9	.1