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:

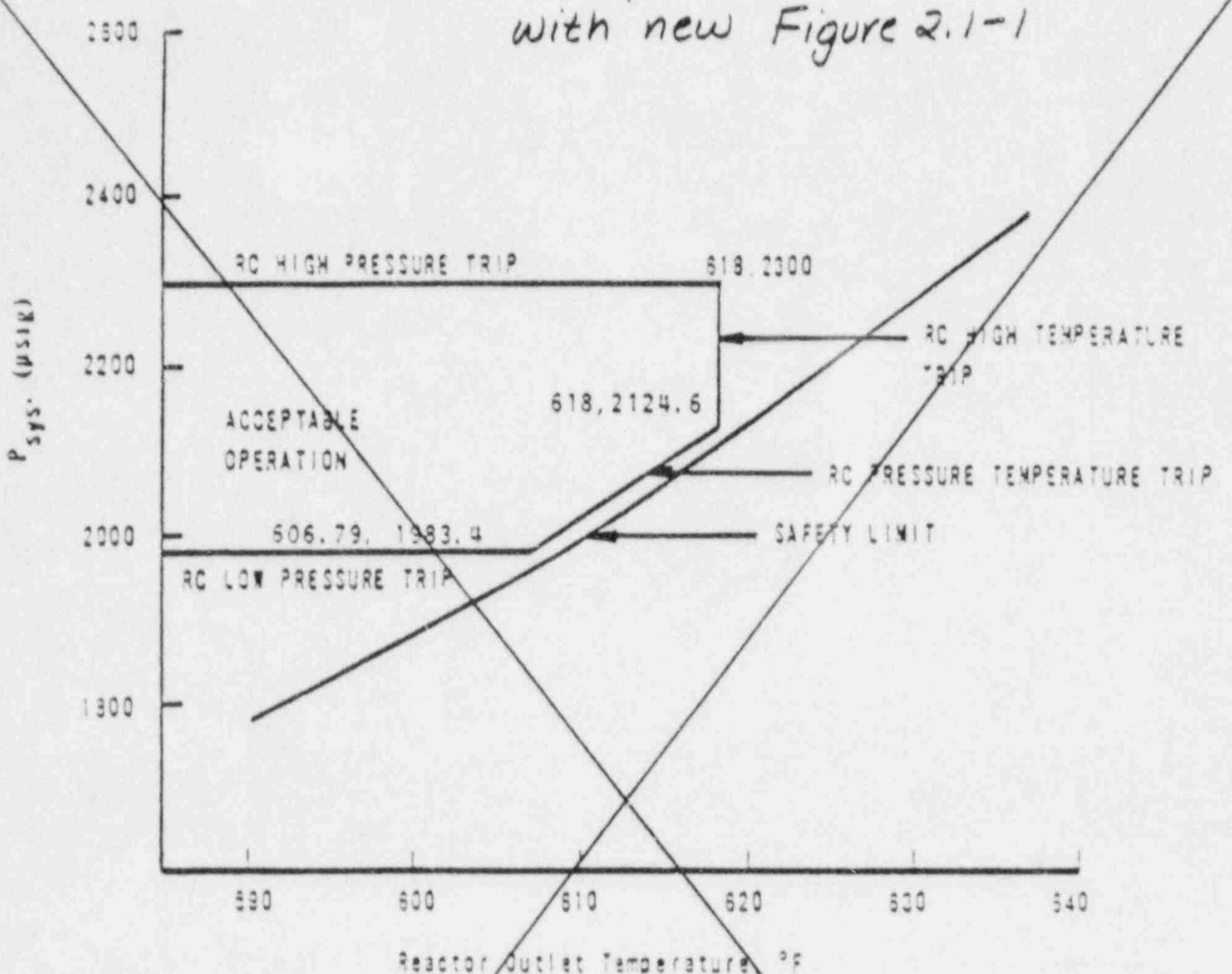
MODES 1 and 2 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

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.

8805250325 880518
PDR ADOCK 05000346
P DCD

Figure 2.1-1. Reactor Core Safety Limit

Superseded with new Figure 2.1-1



ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 1464 Date 2-1-88

Figure 2.1-1 Reactor Core Safety Limit

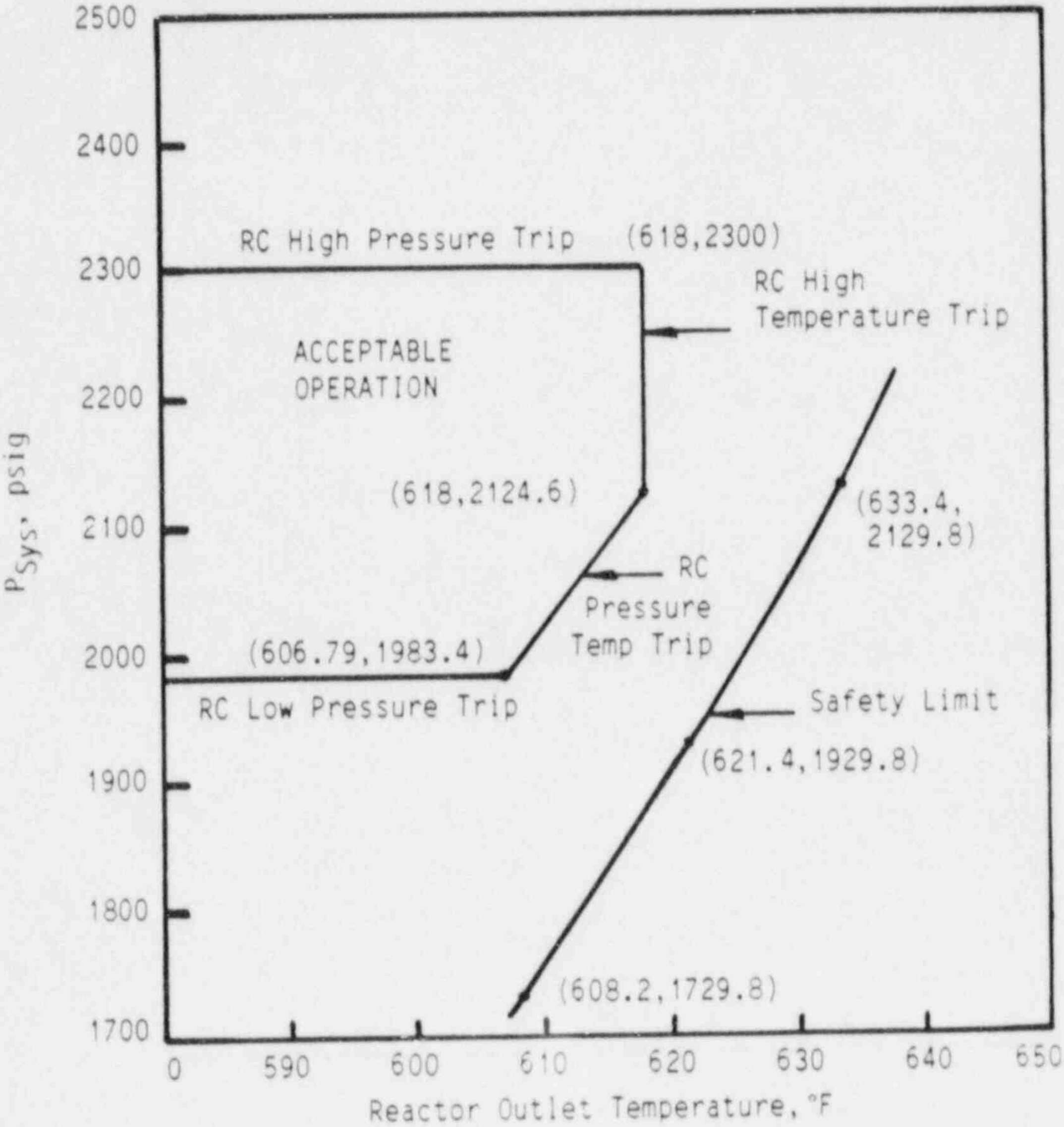
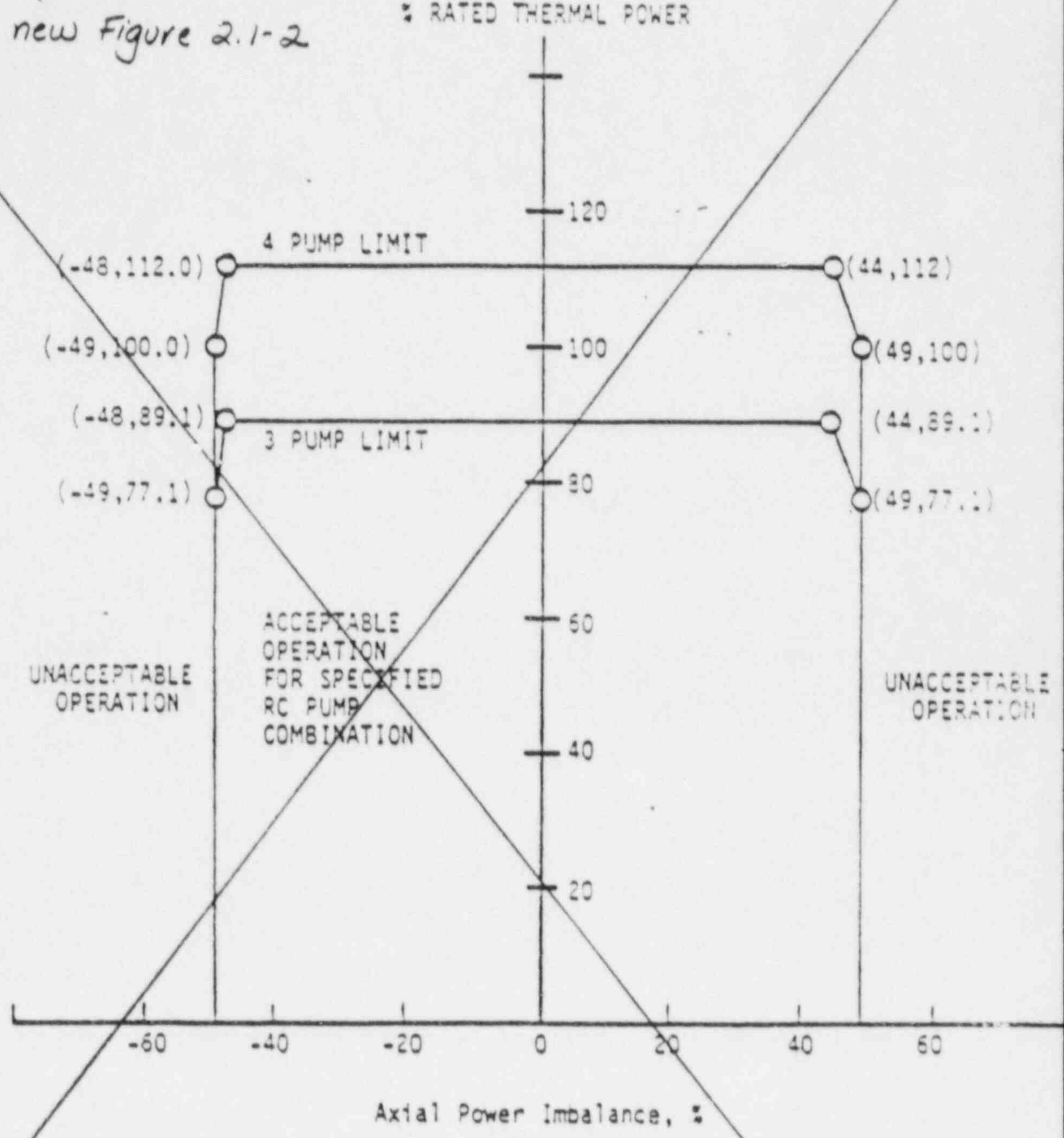


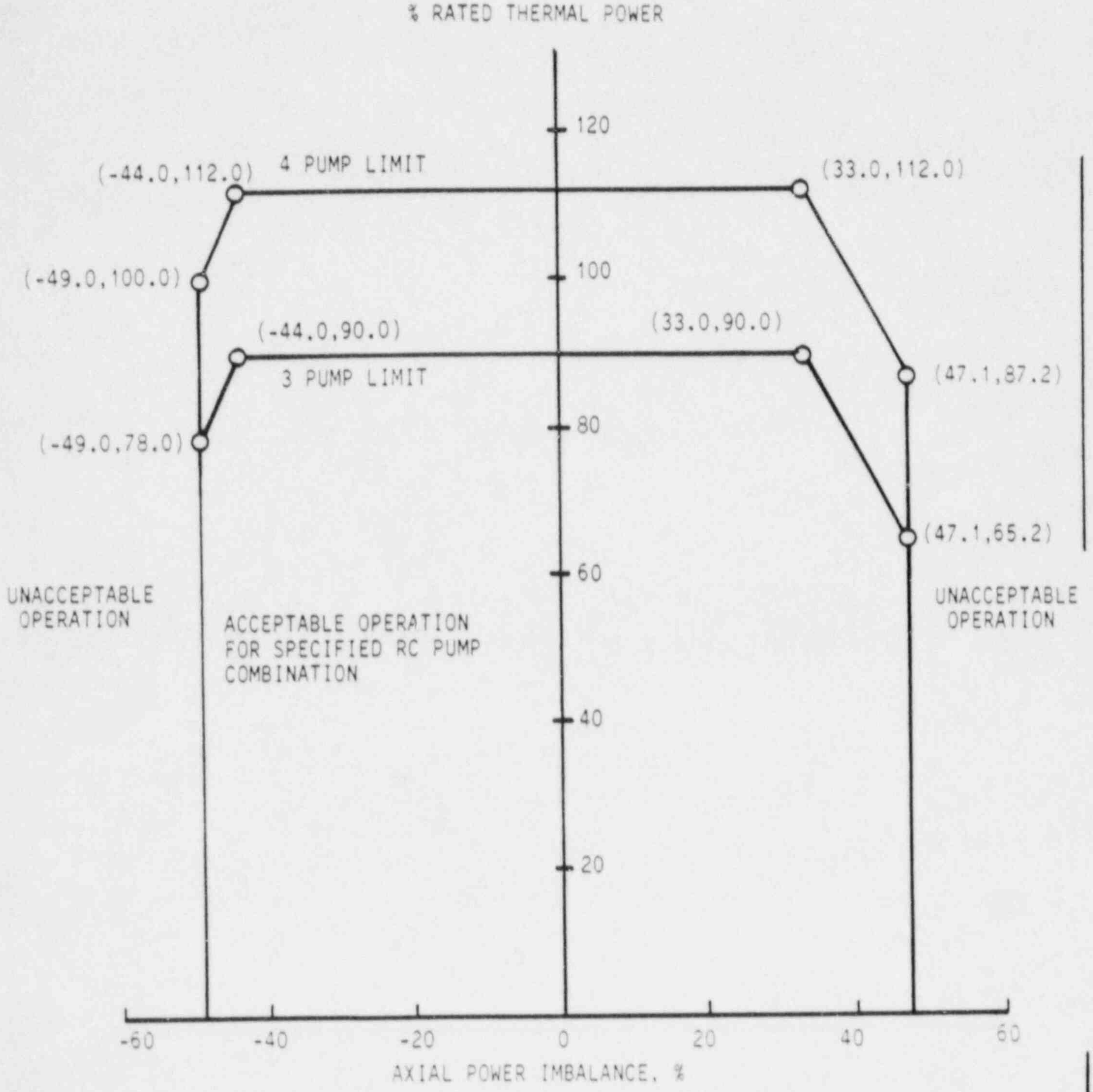
Figure 2.1-2 Reactor Core Safety Limit

Superseded with new Figure 2.1-2



PUMPS OPERATING	REACTOR COOLANT FLOW, GPM
4	380,160
3	283,980

Figure 2.1-2 Reactor Core Safety Limit



Pumps Operating	Reactor Coolant Flow, gpm	Required Measured Flow to Ensure Compliance, gpm
4	380,000	389,500
3	283,860	290,957

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.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 1464 Date 2-1-88

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#
	≤80.6% ~~≤79.7%~~ of RATED THERMAL POWER with three pumps operating	≤80.6% ~~≤79.7%~~ of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/flow(1)	Four pump Trip setpoint not to exceed the limit line of Figure 2.2-1. For three pump operation, see Figure 2.2-1.	Four pump Allowable values not to exceed the limit line of Figure 2.2-1#. For three pump operation, see 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.

Superseded with new Figure 2.2-1

Figure 2.2-1 Trip Setpoint for Flux -- ΔFlux/Flow

Curve shows trip setpoint for an approximately 25% flow reduction for three pump operation (283,980 gpm). The actual setpoint will be directly proportional to the actual flow with three pumps.

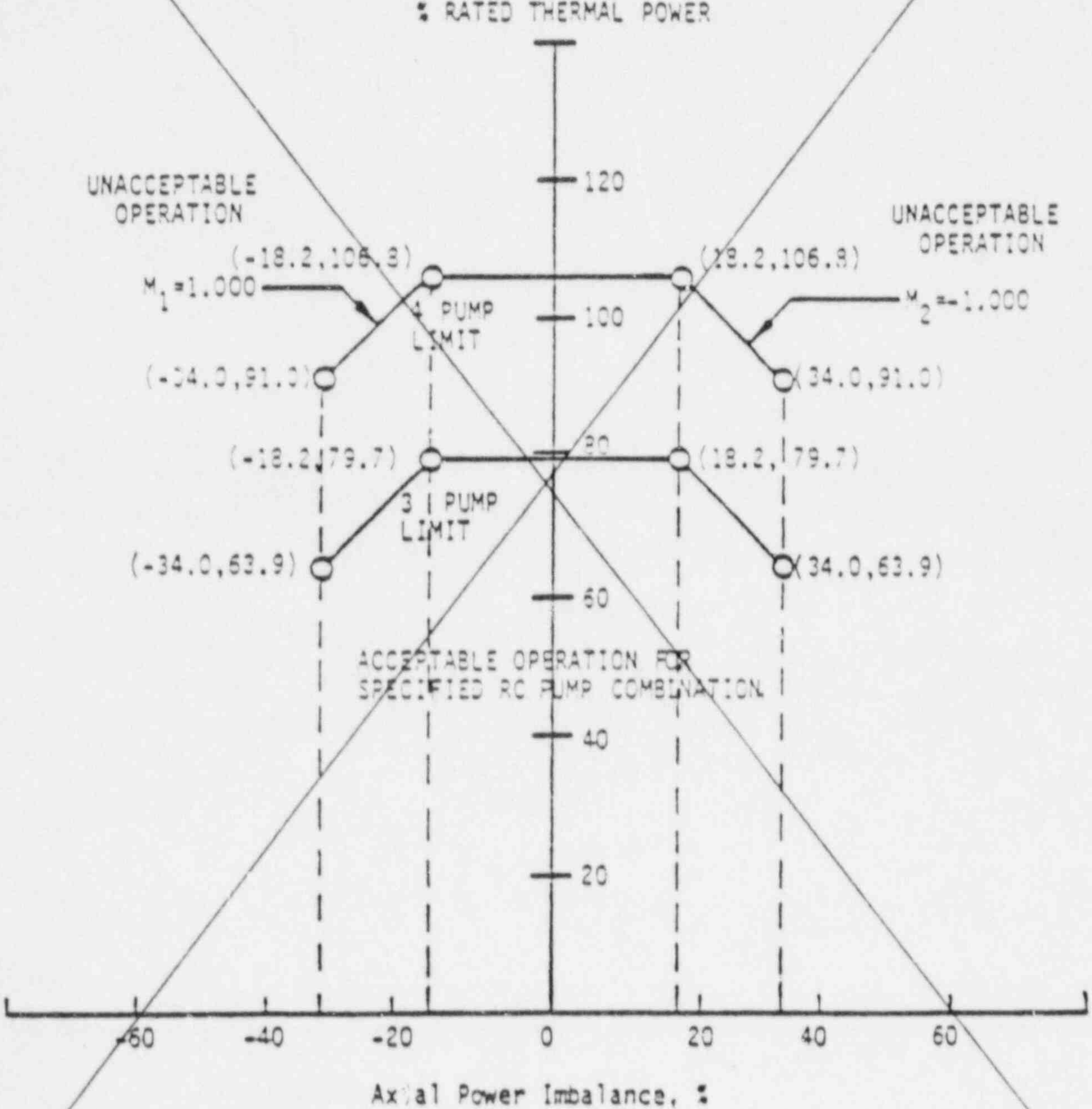
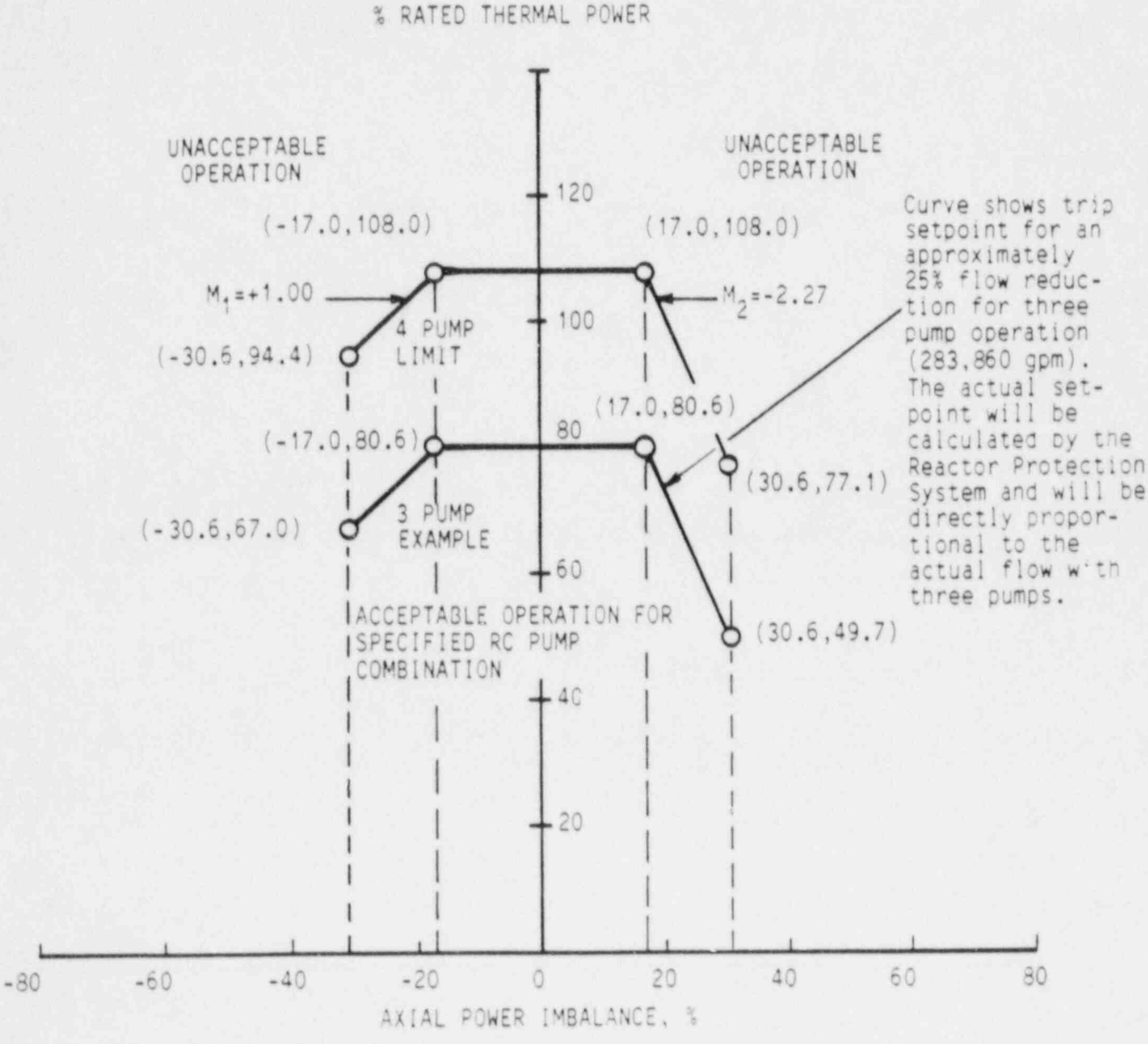


Figure 2.2-1 Trip Setpoint for Flux -- ΔFlux/Flow



2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the B&W-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is ~~380,160~~ 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required flow is 389,500 GPM.) This curve is based on the following hot channel factors with potential fuel densification and fuel rod bowing effects:

$$F_Q = \sout{2.55}\ 2.83; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = \sout{1.50}\ 1.65$$

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 = 2.83 ~~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.~~ 20.0 kW/ft for batch 1F and 20.5 kW/ft for batches 6, 7 and 8.

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 the two curves ~~1 and 2~~ of Figure 2.1-2 correspond to the ~~expected~~ analyzed 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.~~ less restrictive than 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:

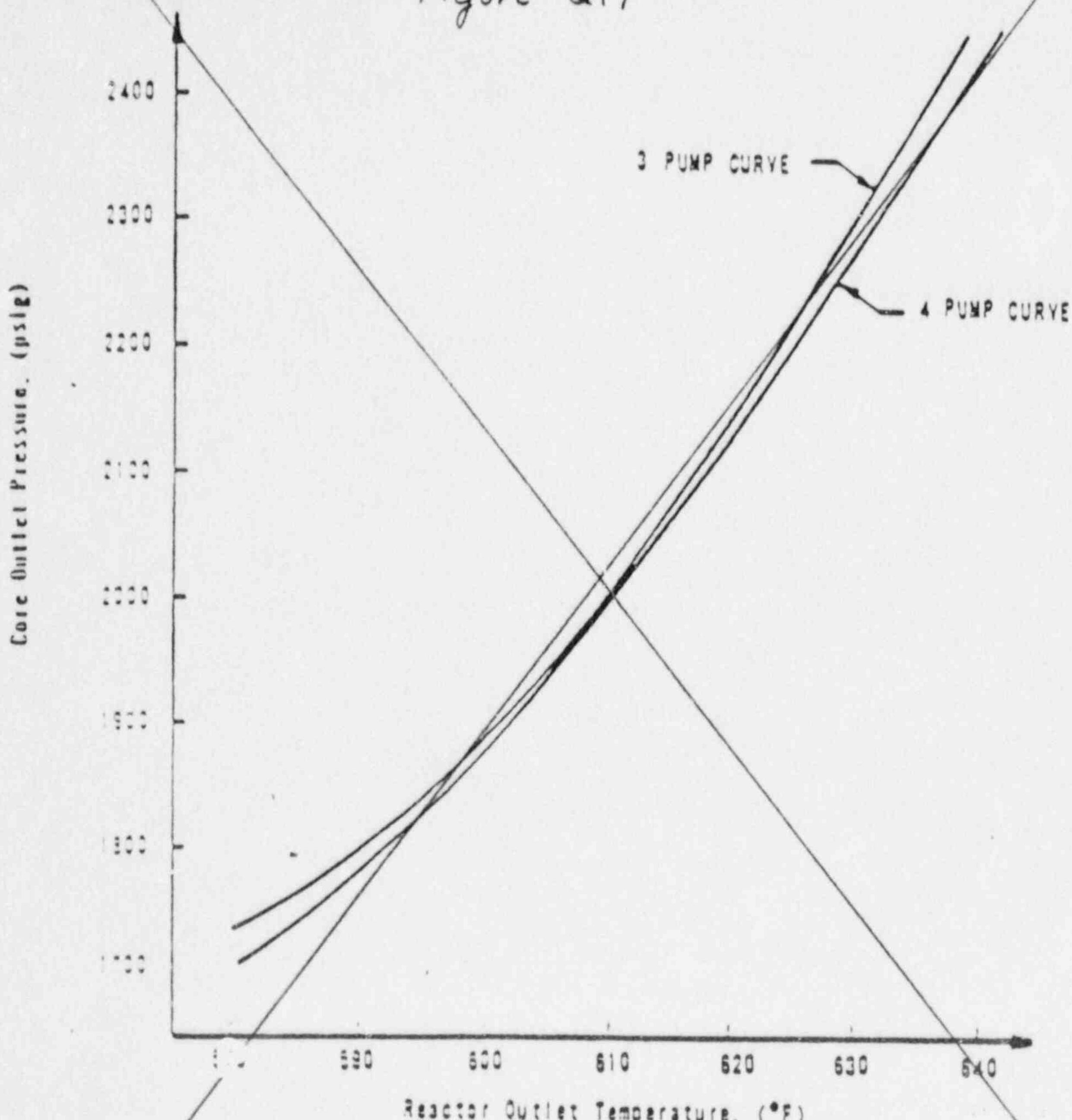
1. Trip would occur when four reactor coolant pumps are operating if power is ~~106.8%~~ 108.0% and reactor coolant flow rate is 100% of full flow rate, or flow rate is ~~93.63%~~ 92.59% of full flow rate and power level is 100%.
2. Trip would occur when three reactor coolant pumps are operating if power is ~~79.7%~~ 80.68% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is ~~70.22%~~ 69.44% 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. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

Note that the value of 80.6% in Figure 2.2-1 was truncated from the calculated value of 80.68%

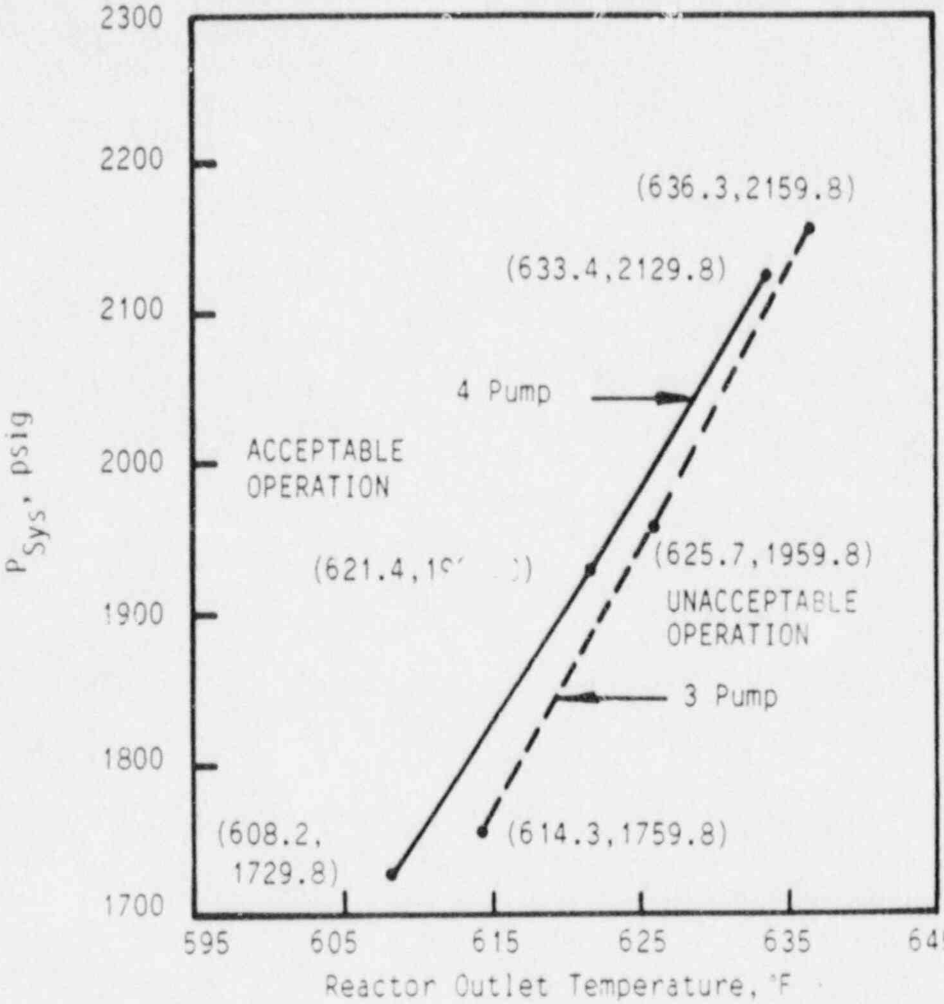
Bases-Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR

Superseded with new Bases Figure 2.1



PUMPS	FLOW (GPM)	POWER
4	380,150	112%
3	283,980	89.1%

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



Pumps	Flow, gpm	Power	Required Measured Flow to ensure Compliance, gpm
4	380,000	112%	389,500
3	283,860	90.5%	290,957

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid addition system with:

1. A minimum ~~contained~~ available borated water volume ~~in accordance with Figure 3.1-1,~~ of 600 gallons,
2. Between 7875 and 13,125 ppm of boron, and
3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A minimum ~~contained~~ available borated water volume of ~~70,700~~ 3,000 gallons,
2. A minimum boron concentration of 1800 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATION or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.8 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the ~~contained~~ available borated water volume of the source, and

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE:

a. The boric acid addition system with:

1. A minimum ~~contained~~ available borated water volume in accordance with Figure 3.1-1,

2. Between 7875 and 13,125 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. An ~~contained~~ available borated water volume of between 482,778 and 550,000 gallons,

2. Between 1800 and 2200 ppm of boron, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid addition system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the boric acid addition system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the ~~contained~~ available borated water volume of each water source, and

3. Verifying the boric acid addition system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when the outside air temperature is < 35°F.

Superseded with new Figure 3.1-1

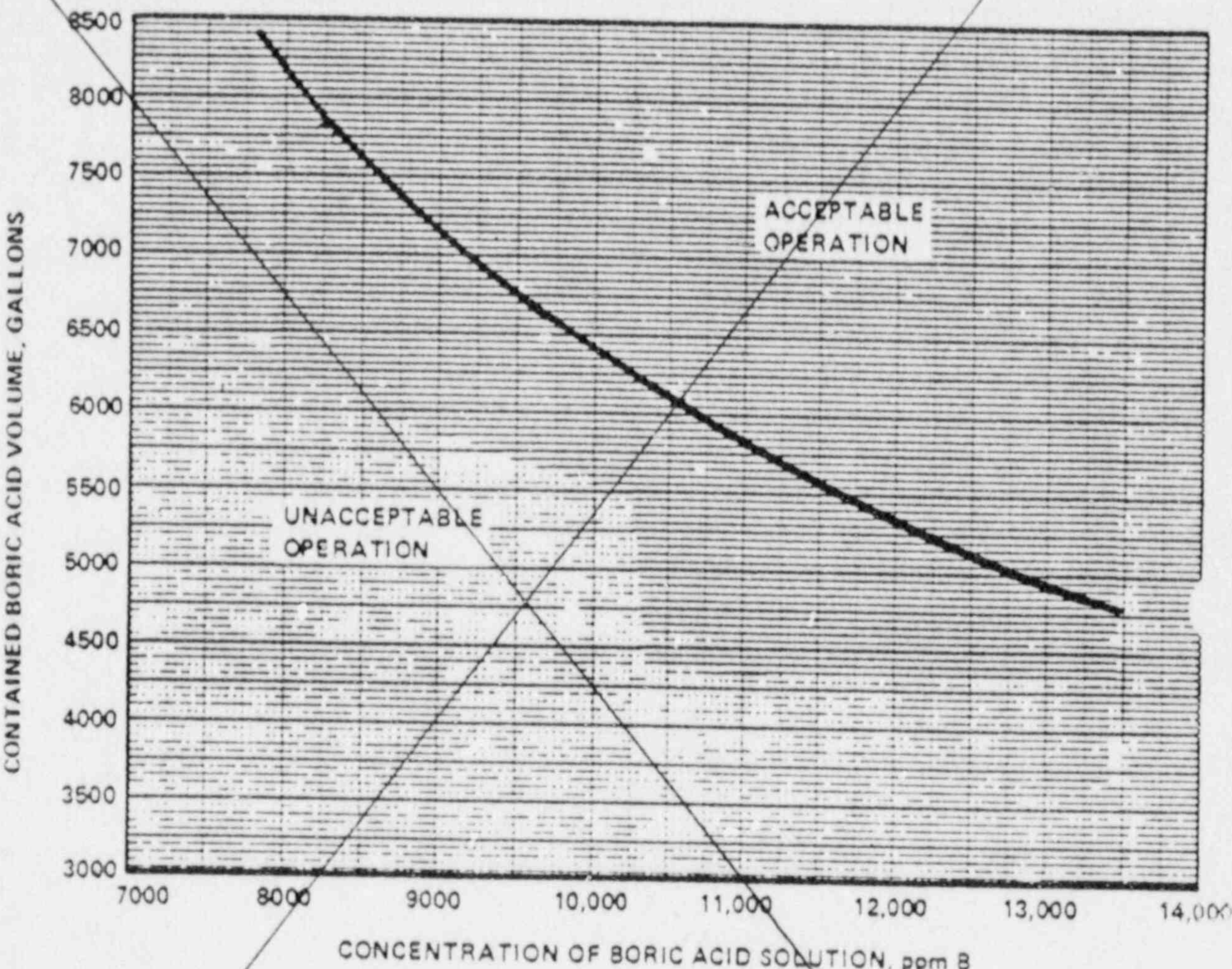
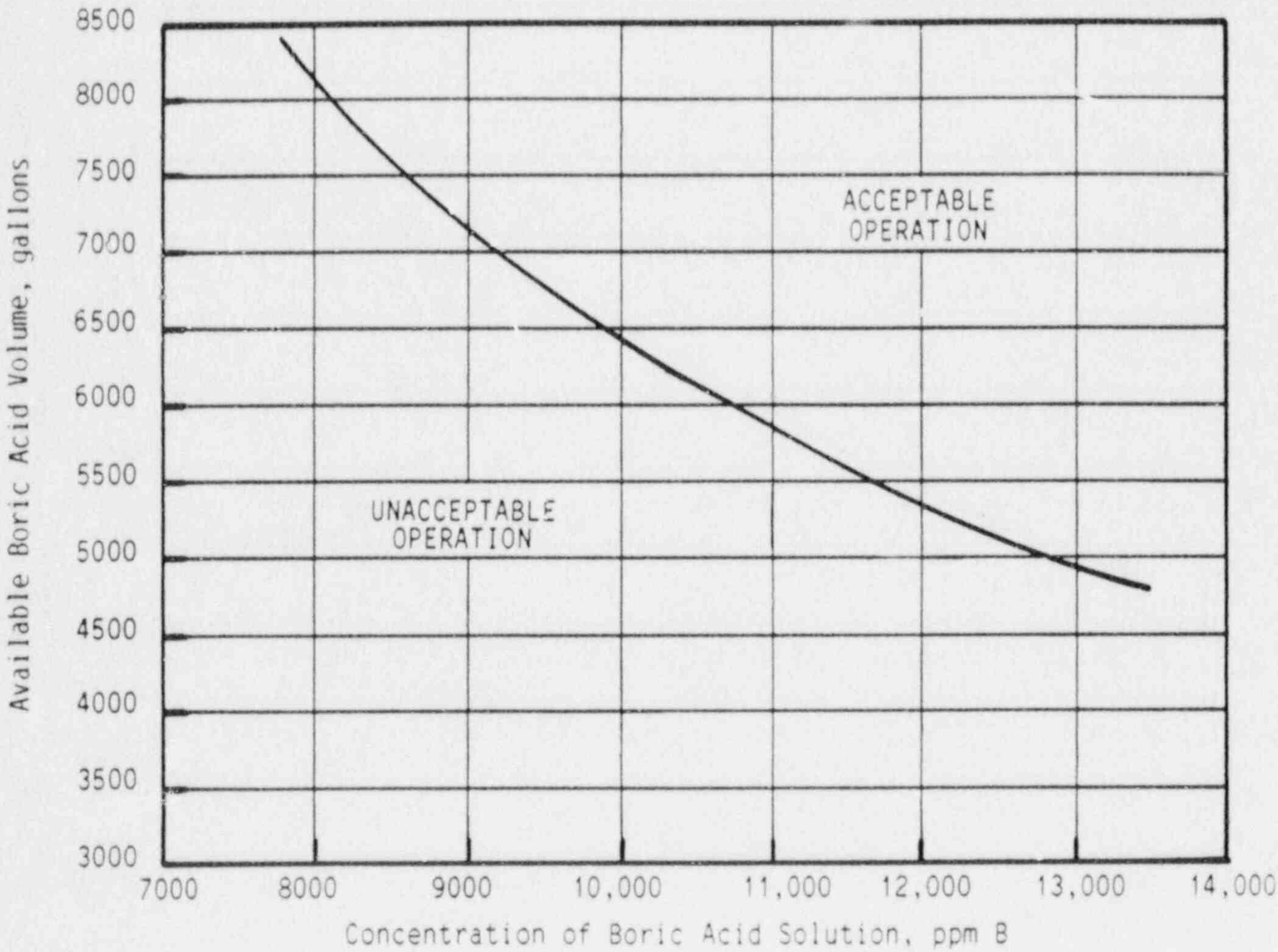


Figure 3.1-1 Minimum Boric Acid Tank Contained Volume as Function of Stored Boric Acid Concentration-Davis-Besse 1, Cycle 1

Figure 3.1-1 Minimum Boric Acid Tank Available Volume as Function of Stored Boric Acid Concentration -- Davis-Besse 1



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.~~ and -2b, 3.1-3a, and -3b. 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 $k_{eff} \geq 1.0$.

Superseded with new Figure 3.1-2a

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

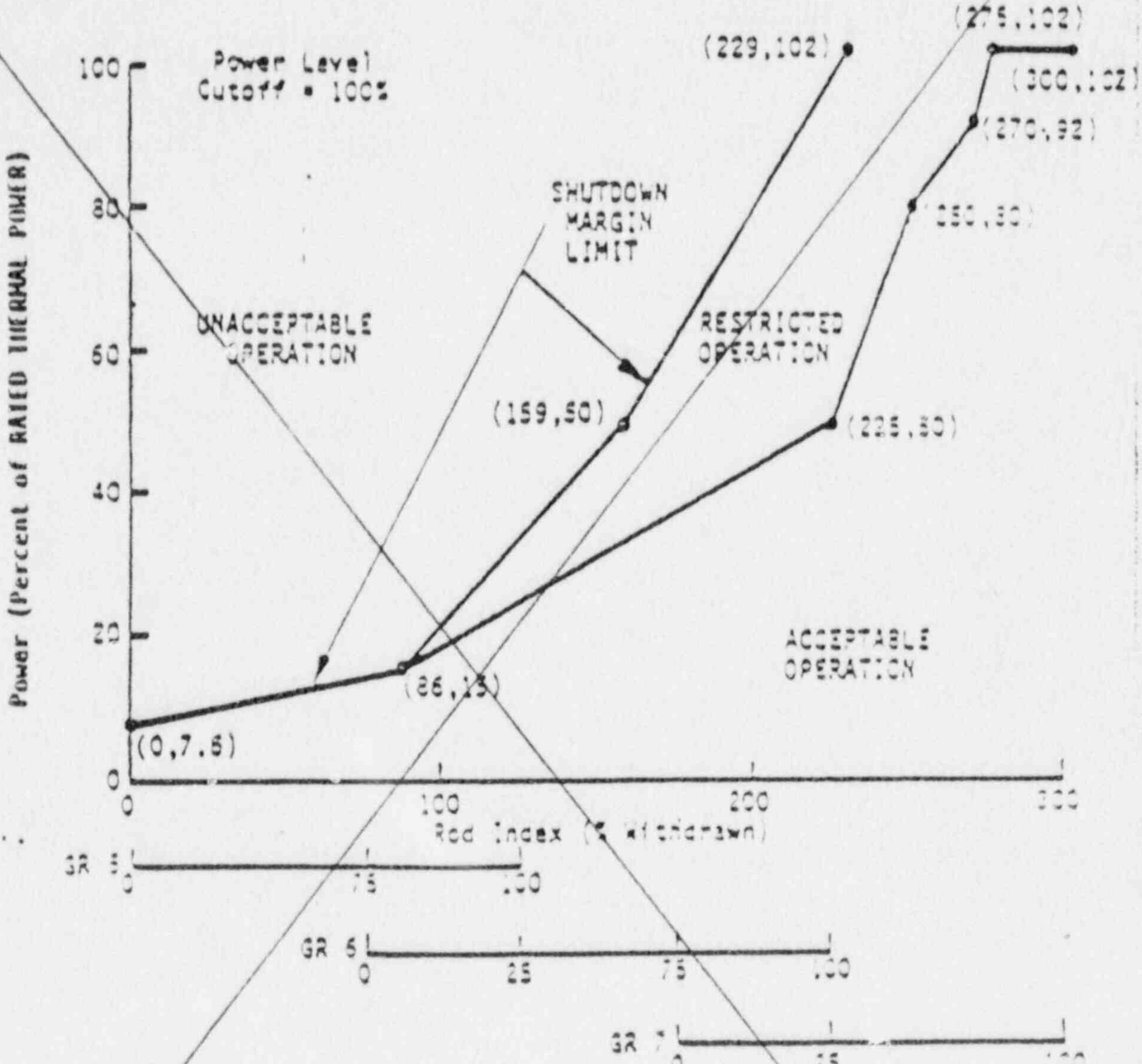
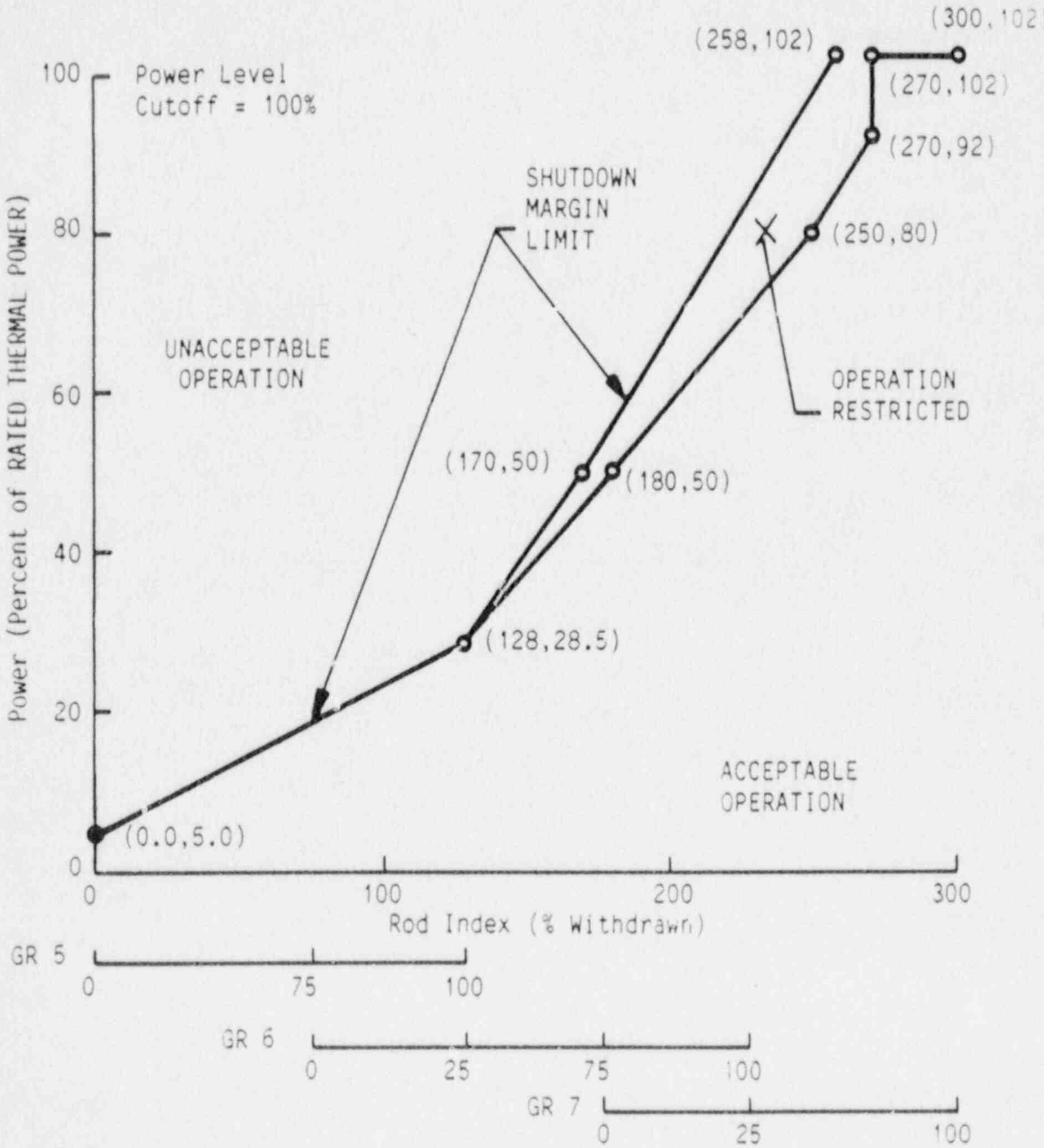


Figure 3.1-2a Regulating Group Position Limits, 0 to 325± 10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 6



Superseded with new Figure 3.1-2b

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

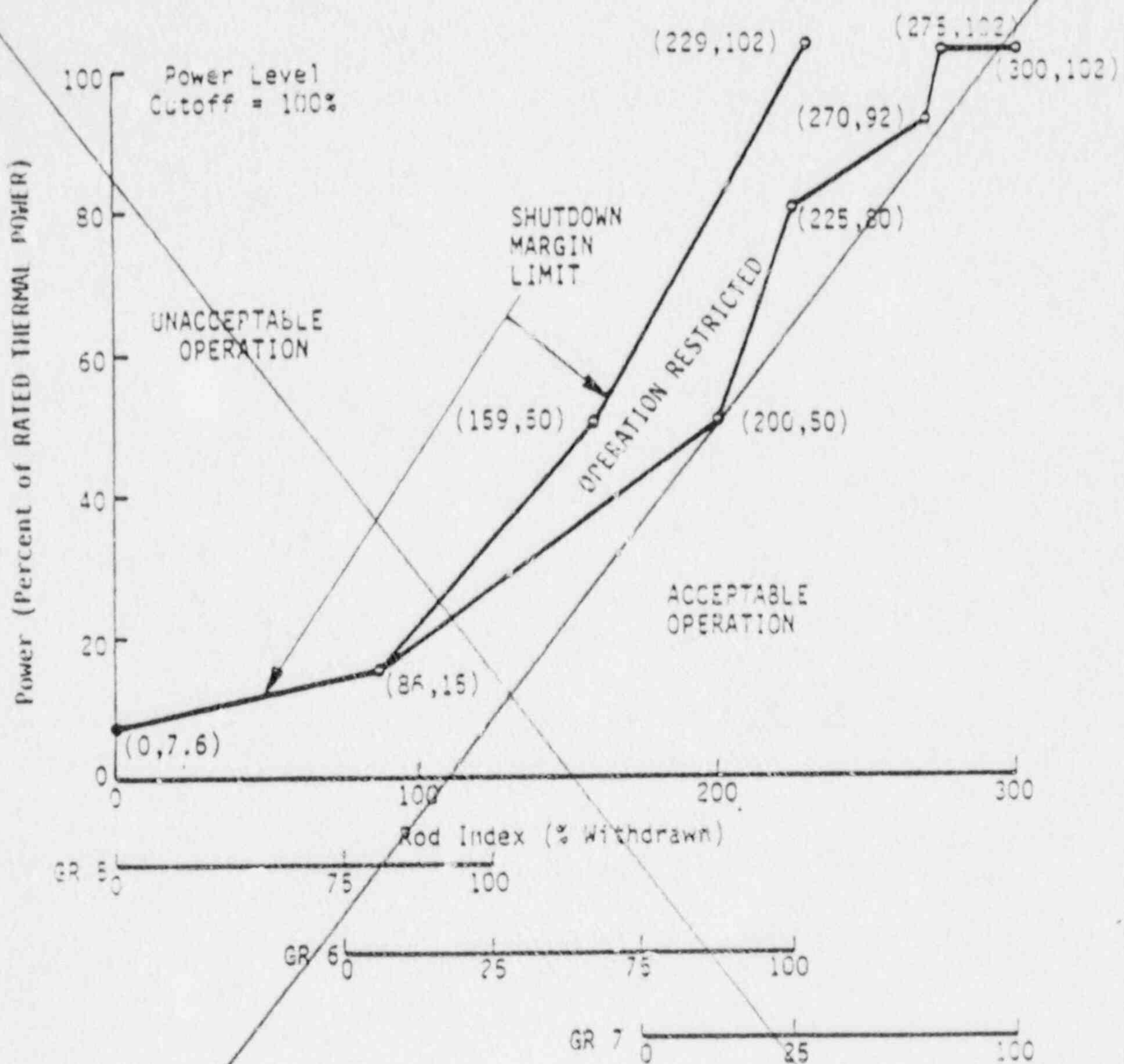


Figure 3.1-2b Regulating Group Postion Limits After 325± 10 EFPD, Four RC pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6

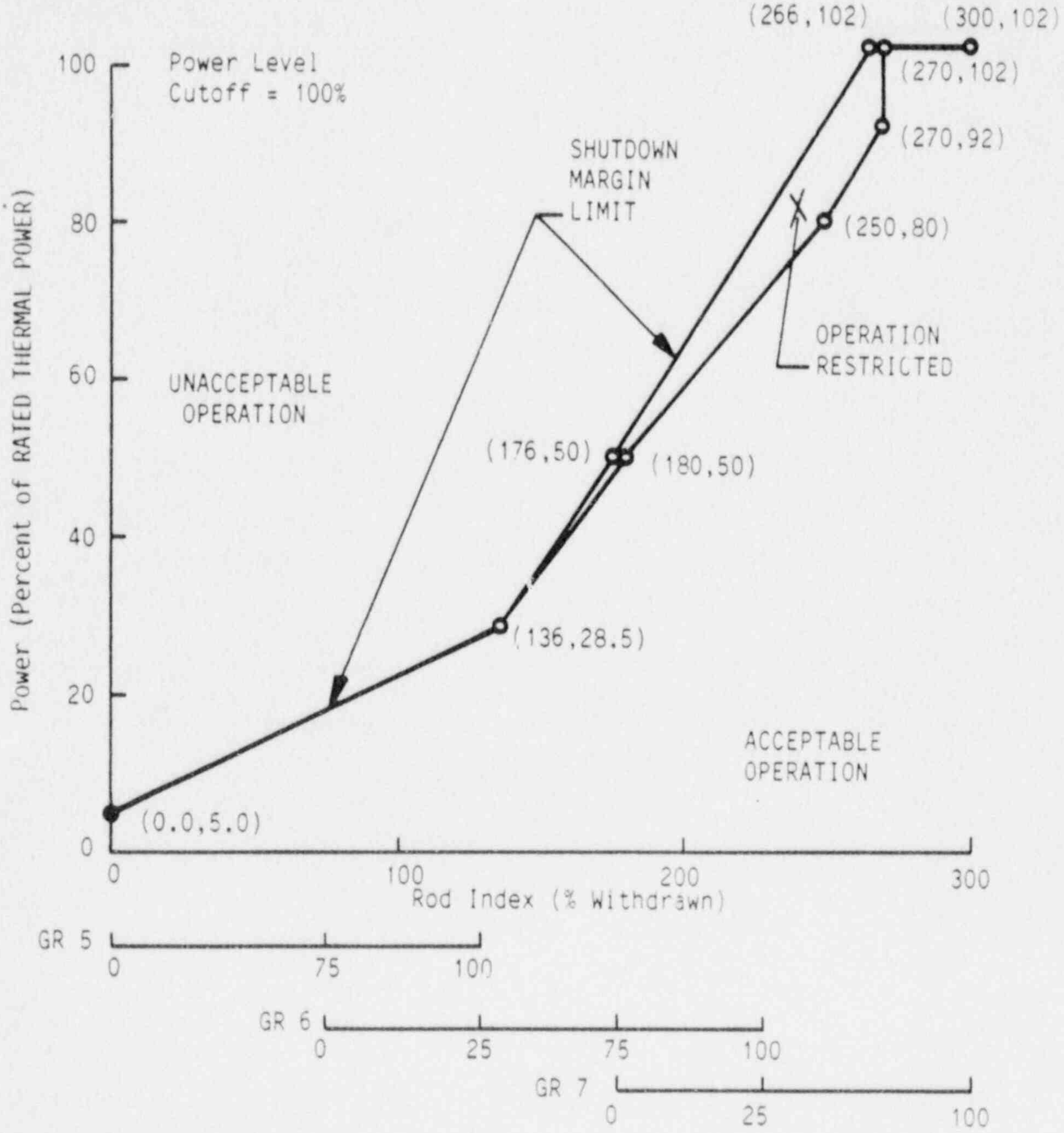


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Figure 3.1-2c Regulating Group Position Limits, 200 ±10 to 300 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

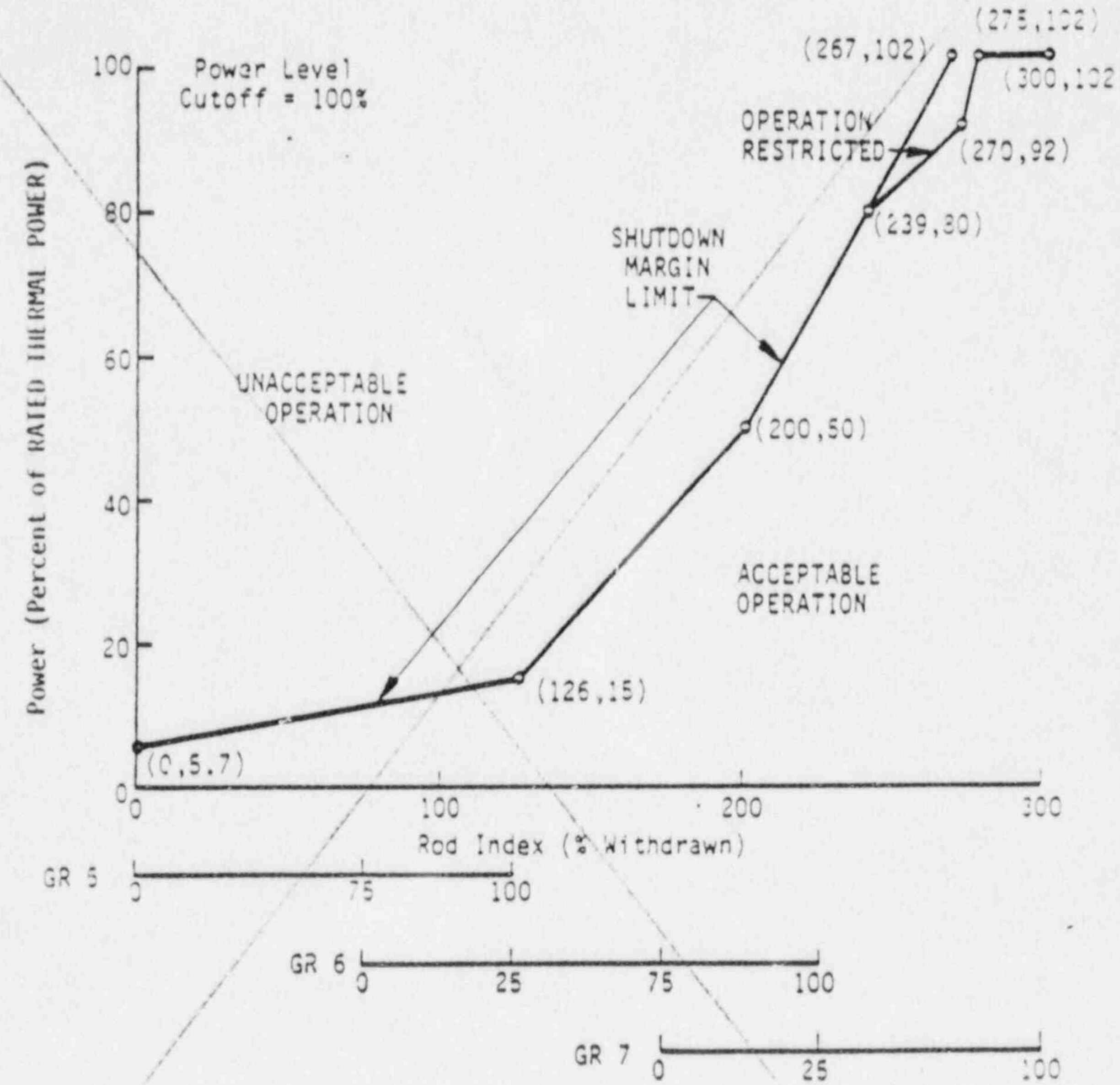

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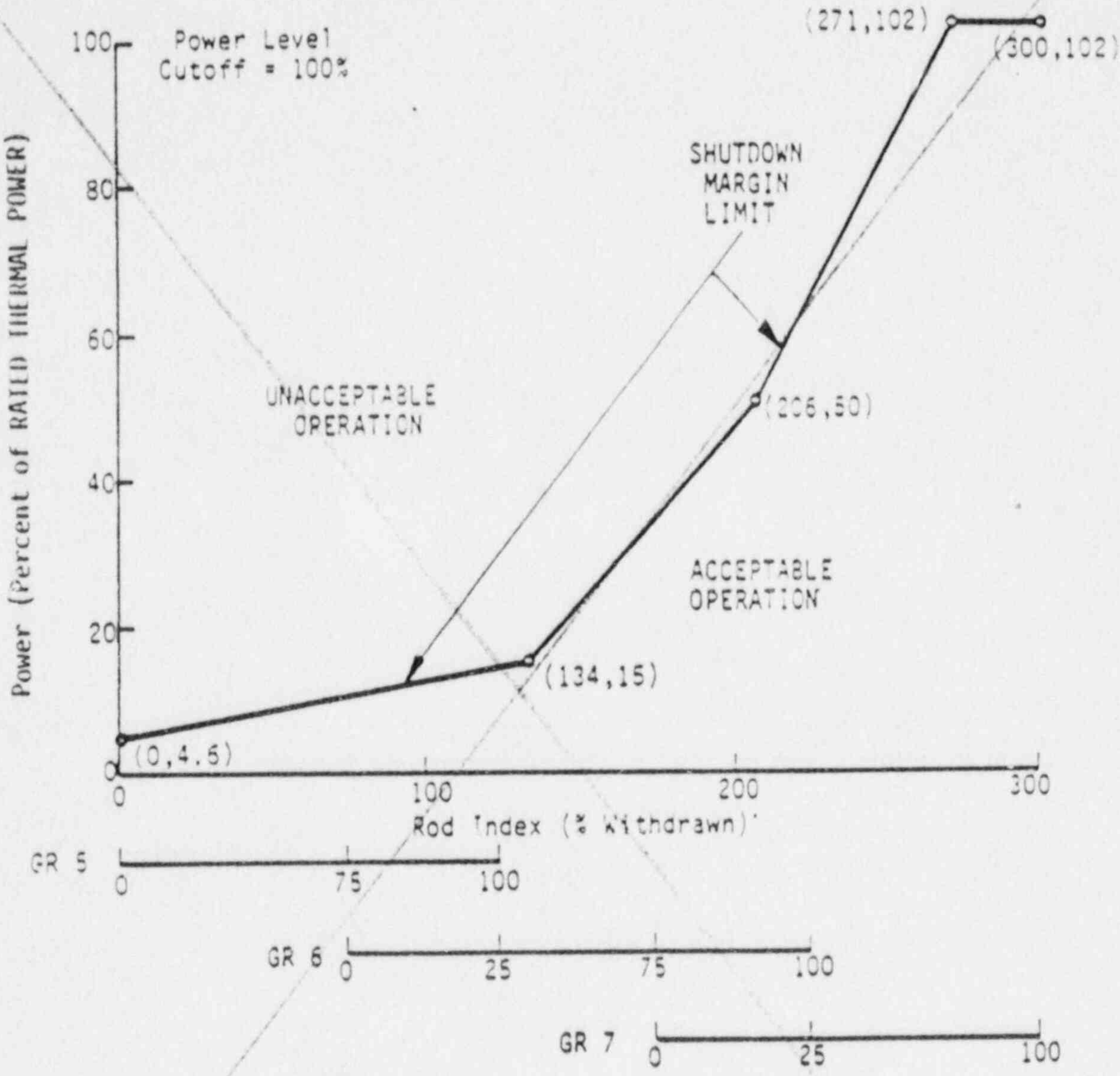
Figure 3.1-2d Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6



Superseded with new Figure 3.1-3a

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

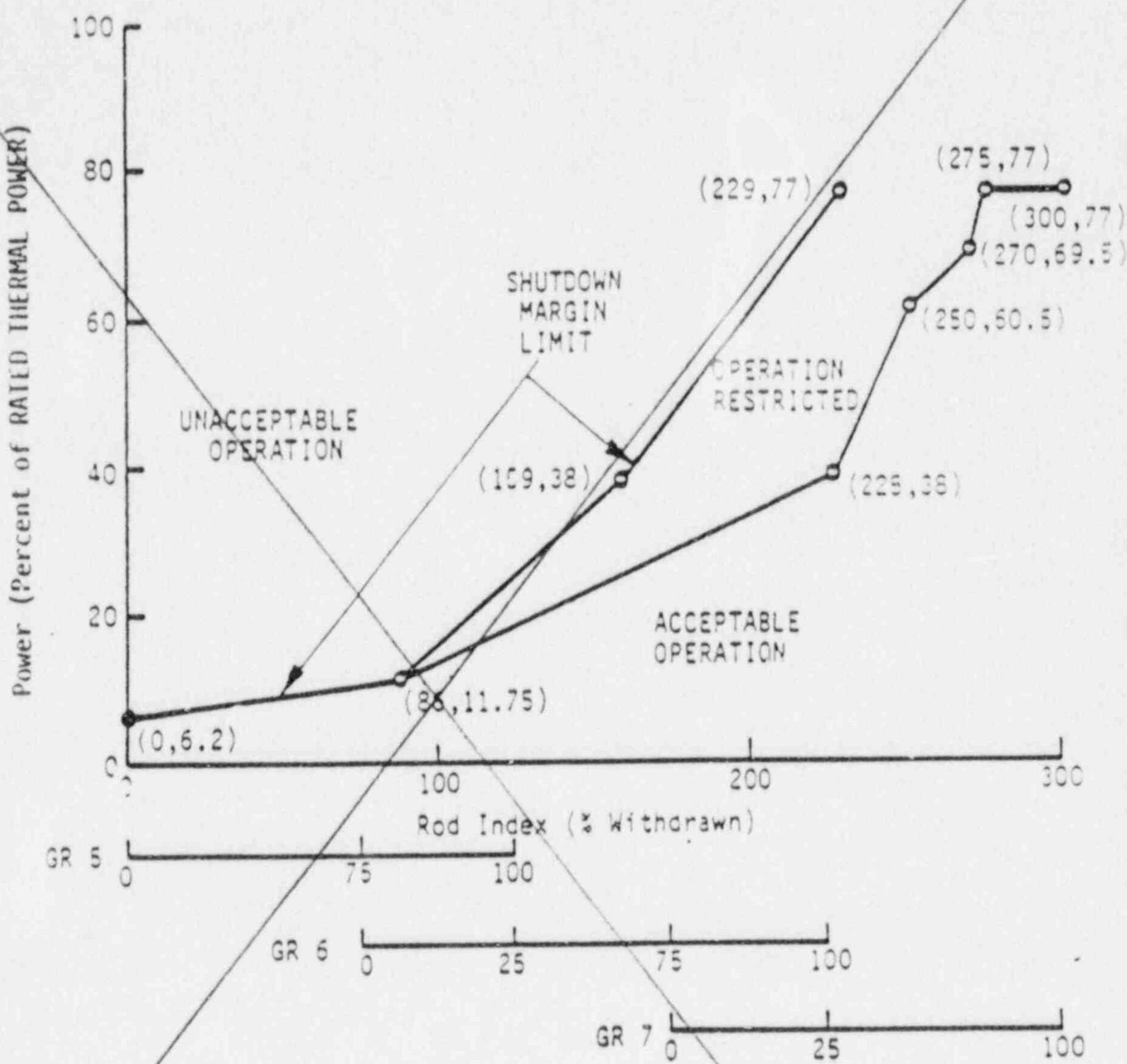
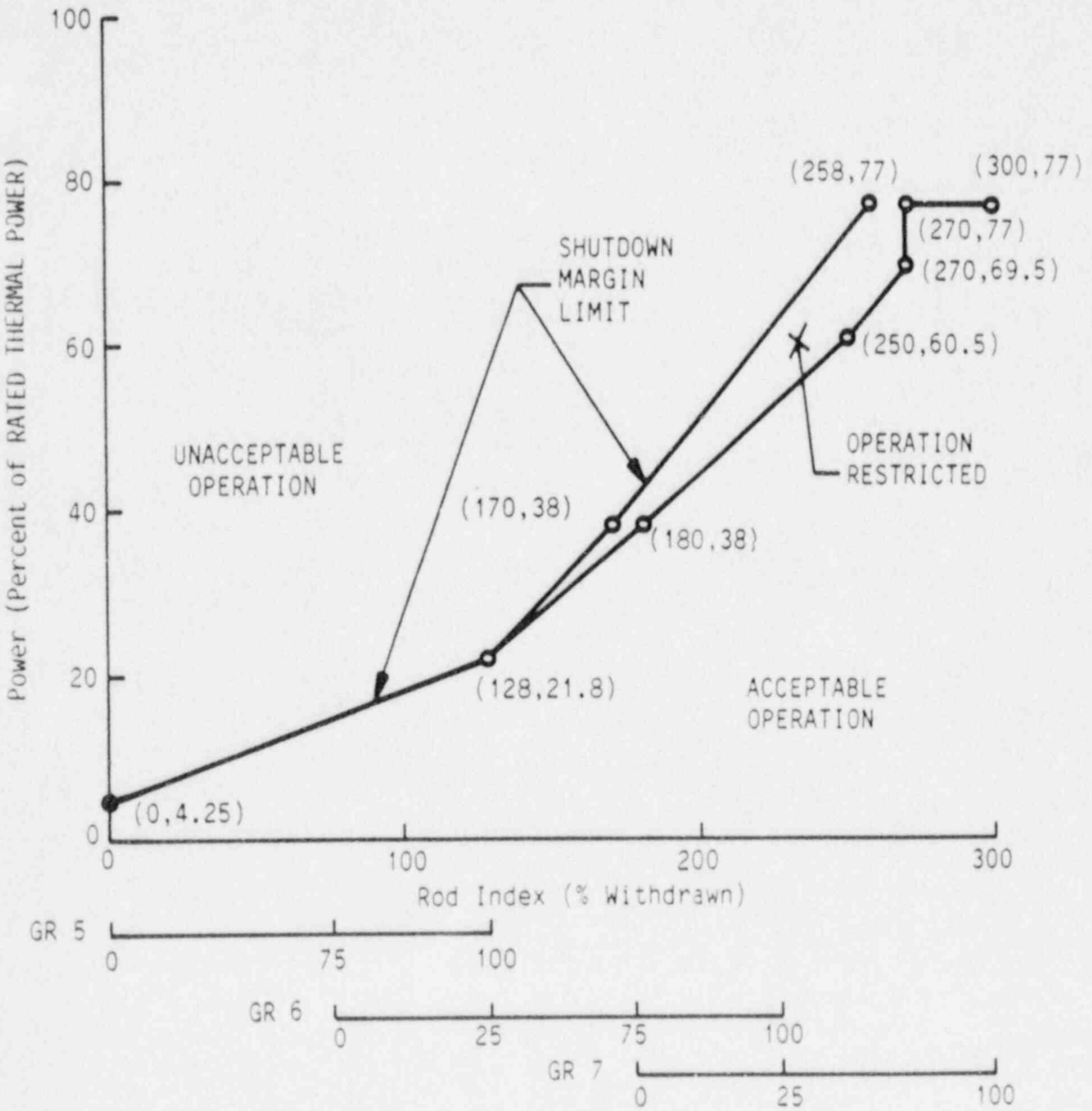


Figure 3.1-3a Regulating Group Position Limits, 0 to 325± 10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 6



Superseded with new Figure 3.1-3b

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

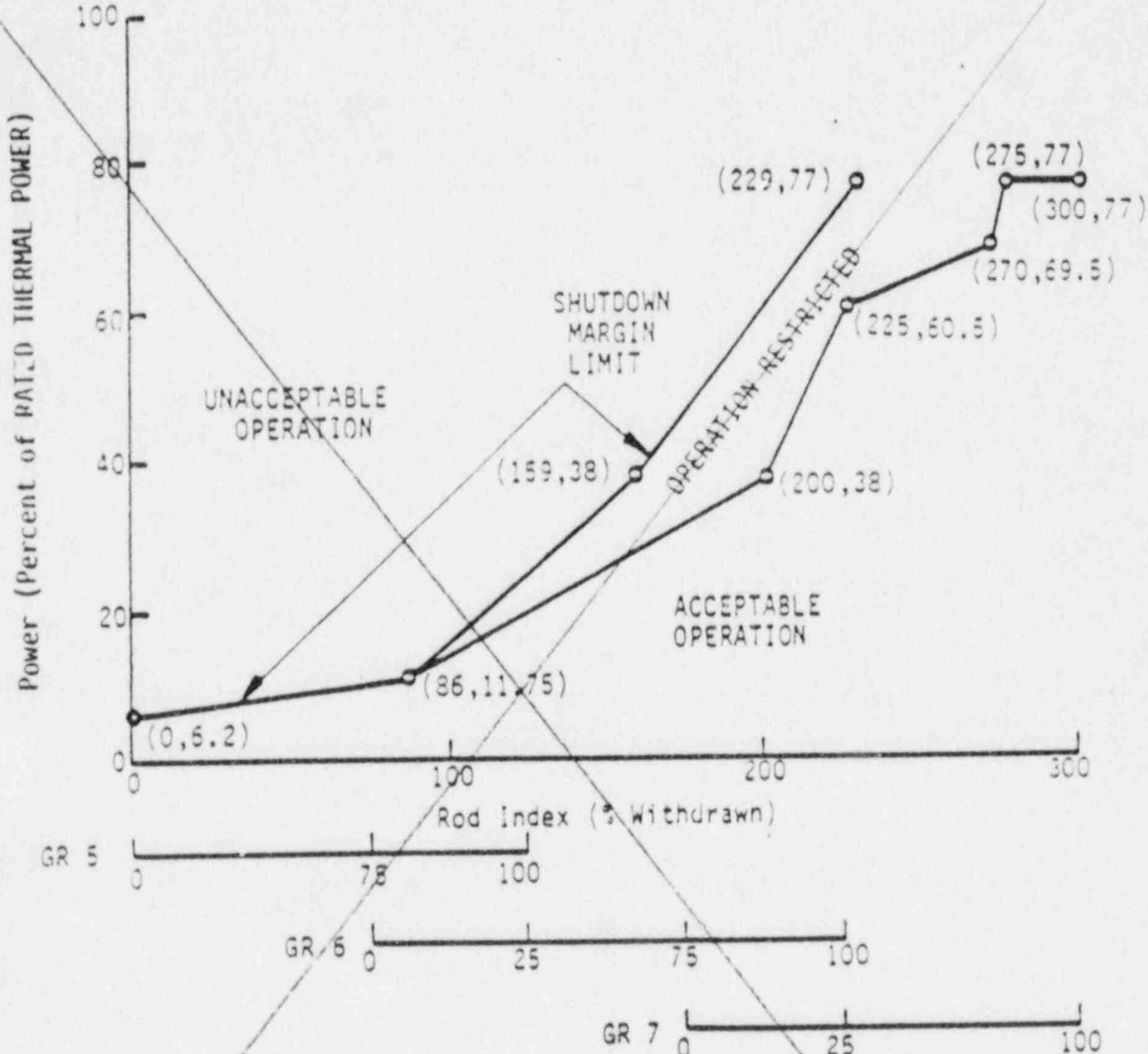


Figure 3.1-3b Regulating Group Position Limits After 325± 10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6

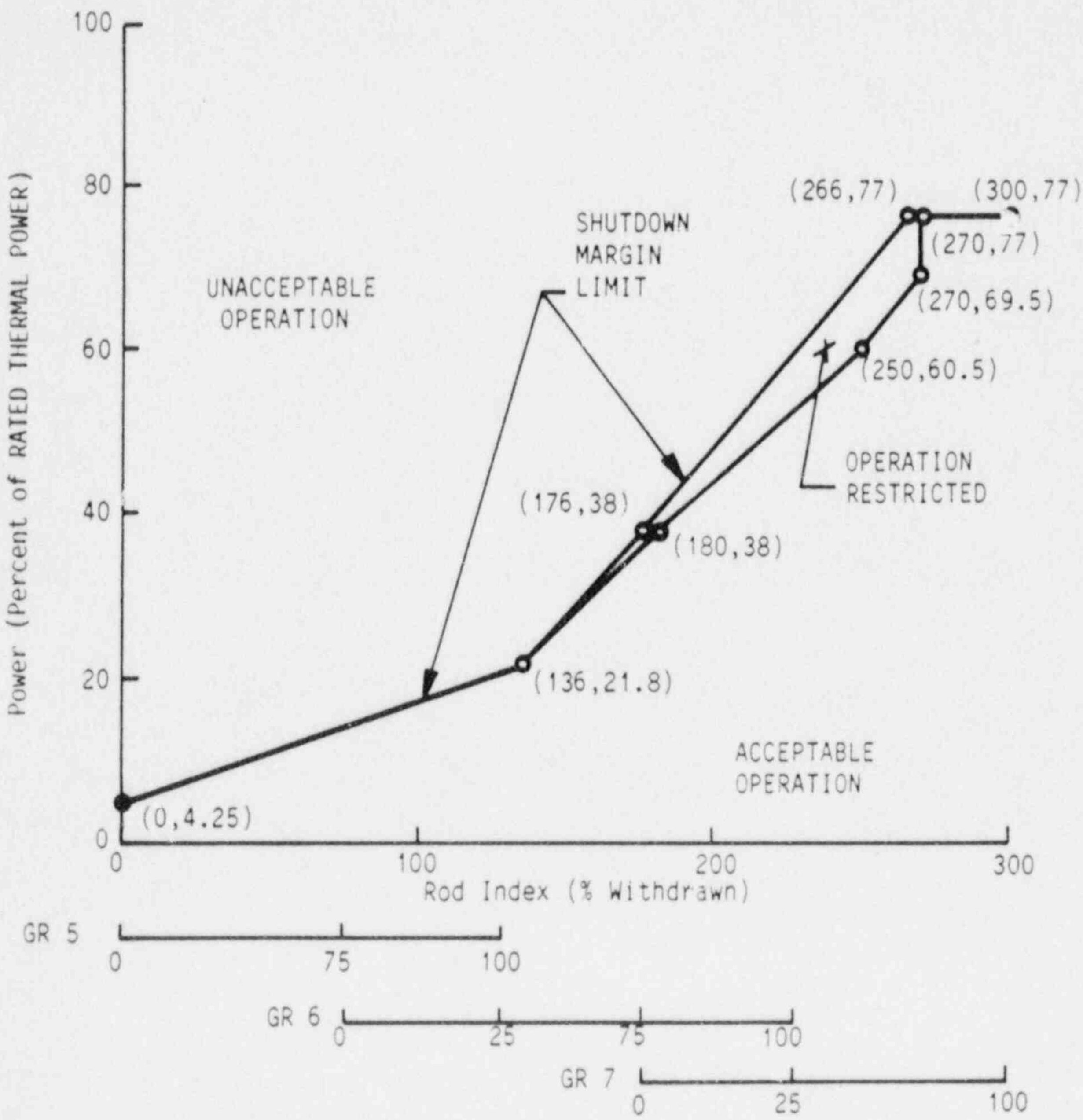


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Figure 3.1-3c Regulating Group Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

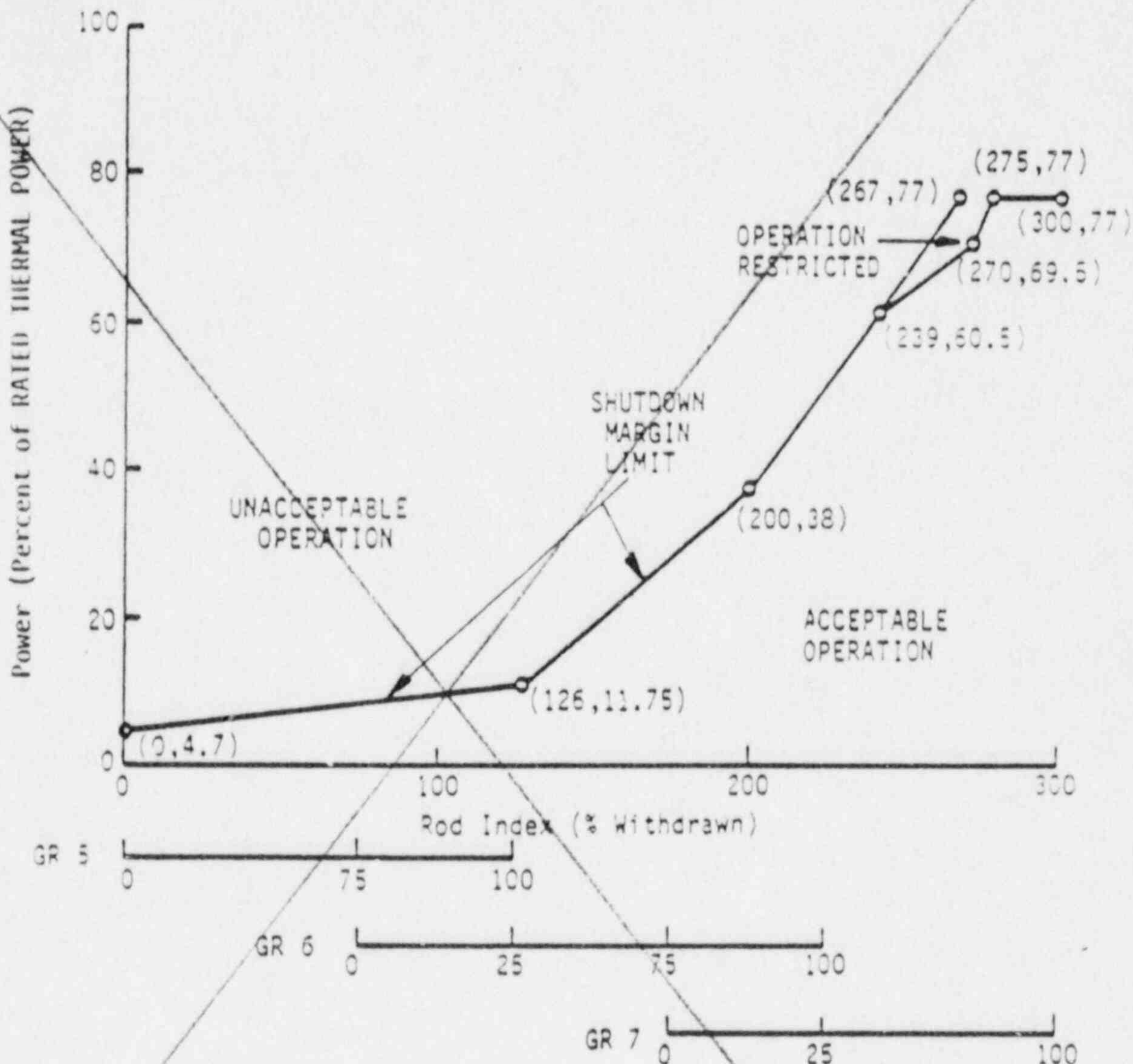

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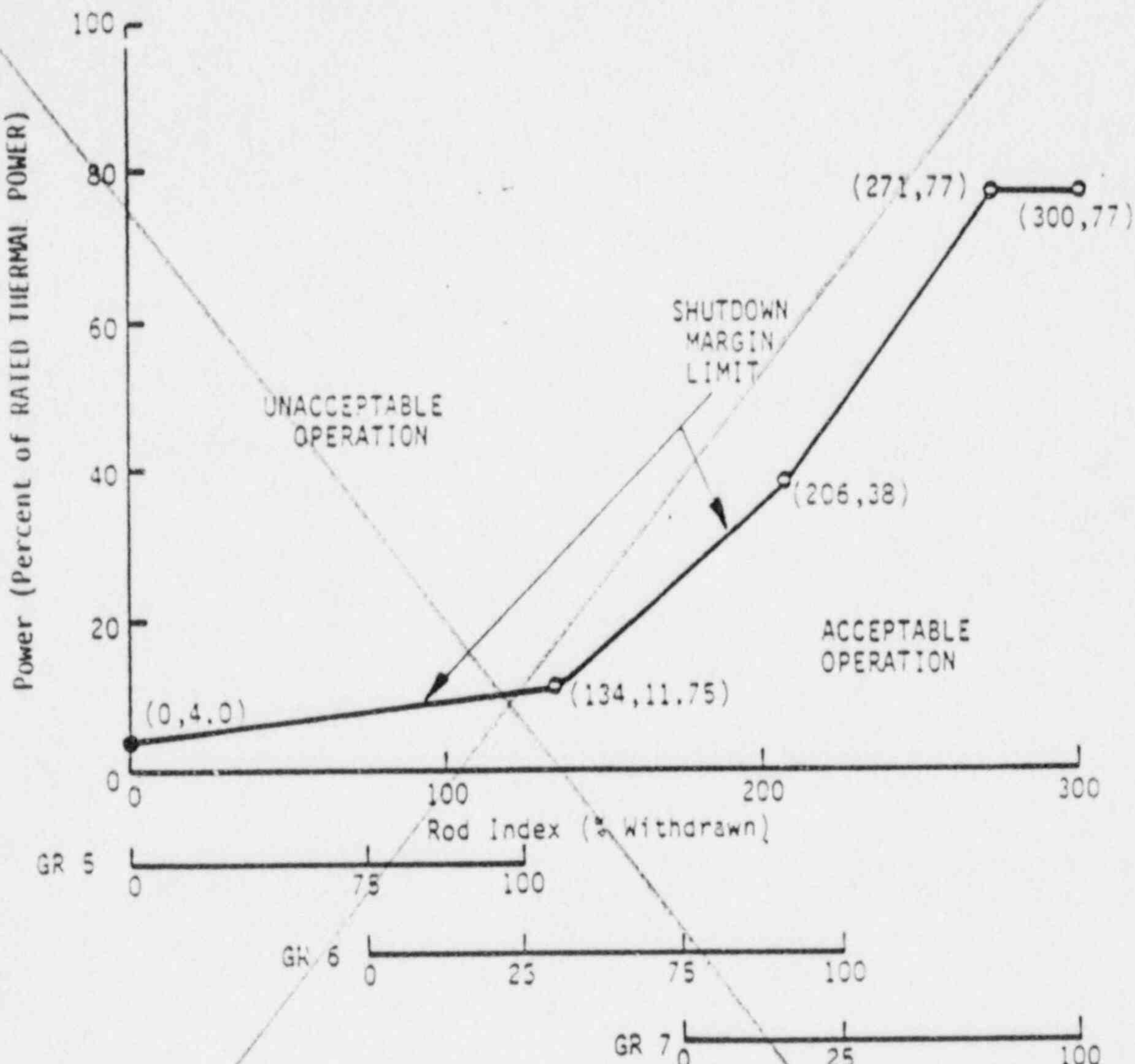
Figure 3.1-3d Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5



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

Superseded with new Figure 3.1-4

X (top) / Z (bottom); W (left) / Y (right)

	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		7		5		7		1		7		5		7	
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															

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	

Figure 3.1-4 Control Rod Core Locations and Group Assignments -- Davis-Besse 1, Cycle 6

← N

X (top), Y (right), Z (bottom), W (left)

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

[X] Group Number

Group	No. of Rods	Function
1	4	Safety
2	8	Safety
3	4	Safety
4	9	Safety
5	12	Control
6	8	Control
7	8	Control
8	8	APSRs
Total	61	

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, and -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 $K_{eff} \geq 1.0$.

Superseded with new Figure 3.1-5a

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

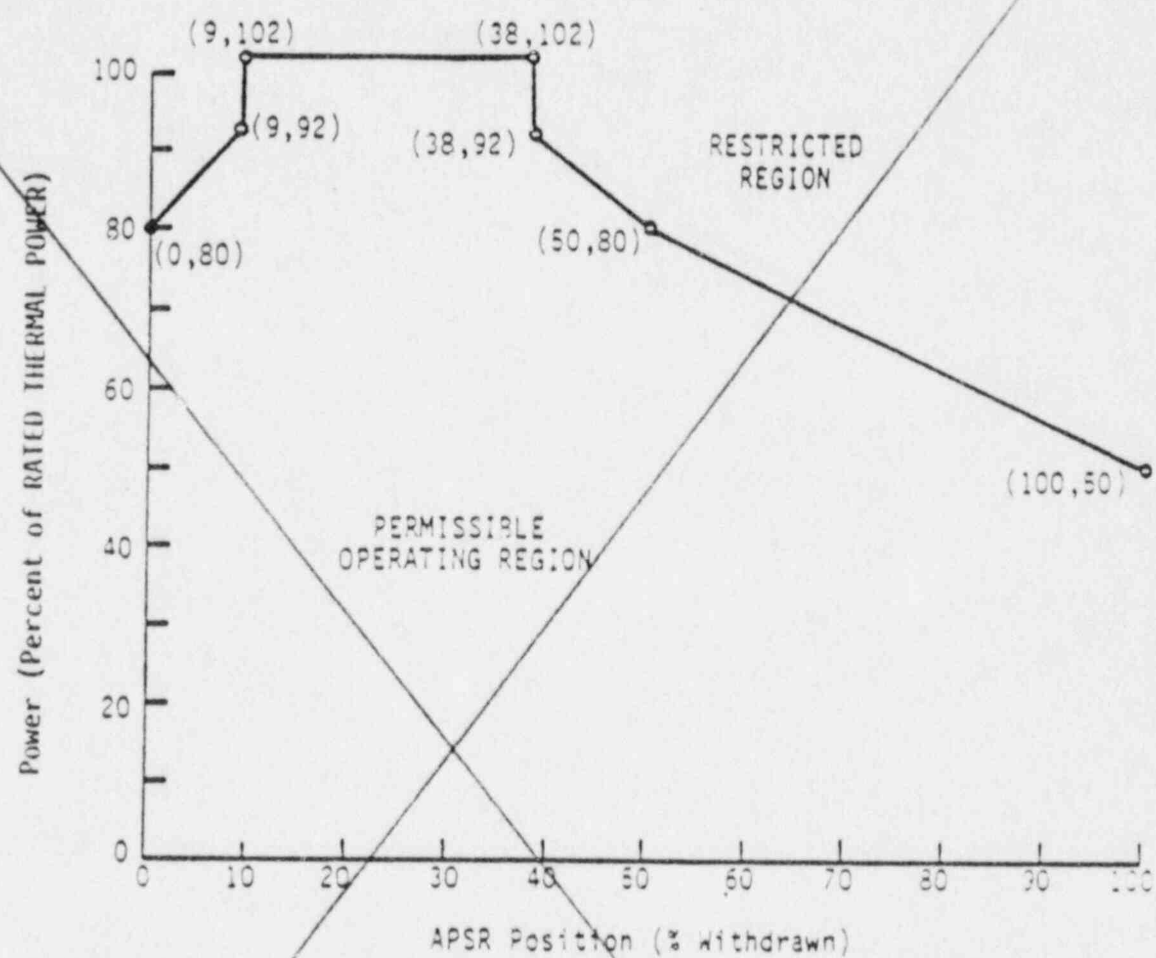
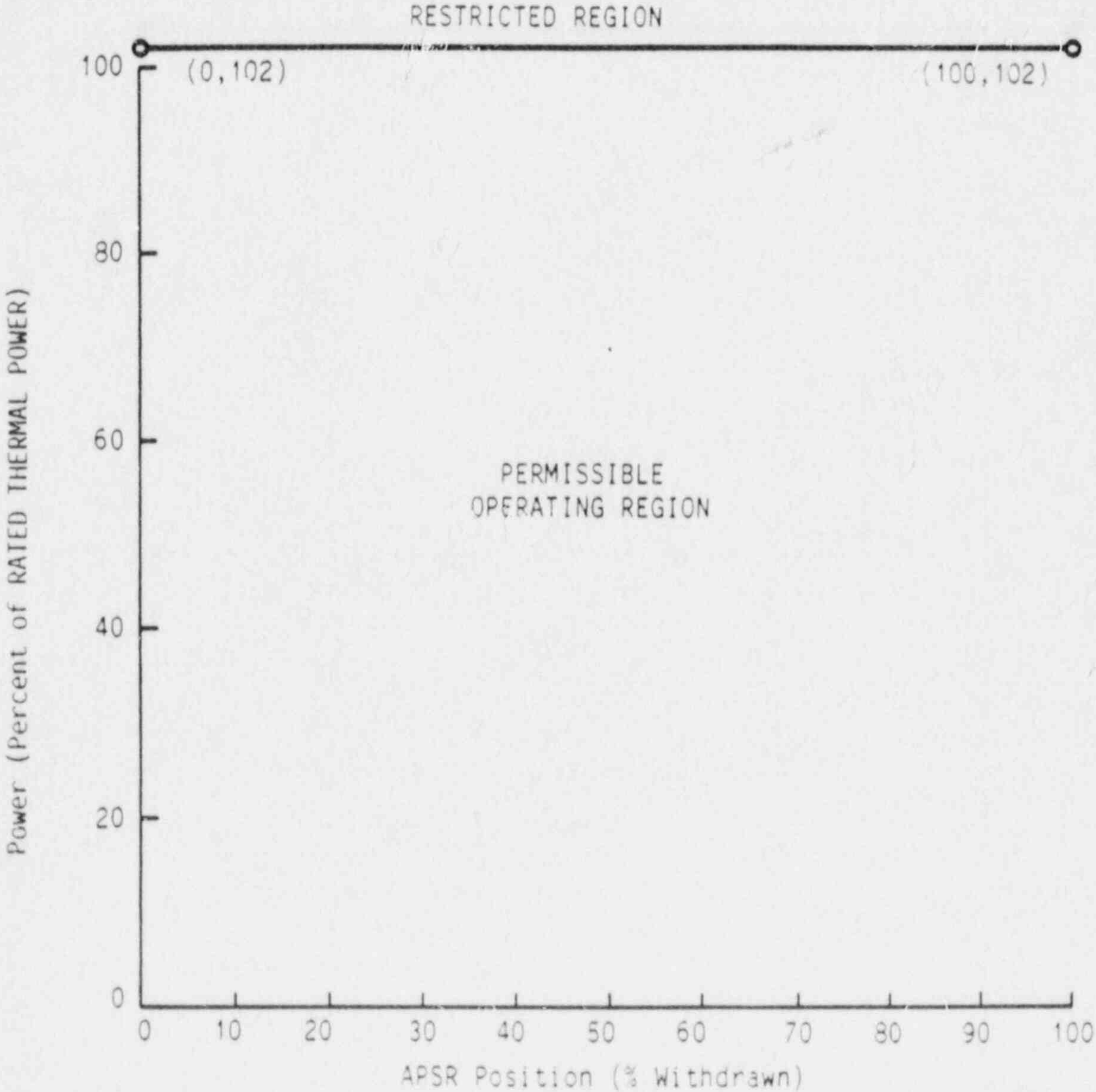


Figure 3.1-5a APSR Position Limits, 0 to 325± 10 EFPD,
Four RC Pumps -- Davis-Besse 1, Cycle 6



Superseded with new Figure 3.1-5b

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

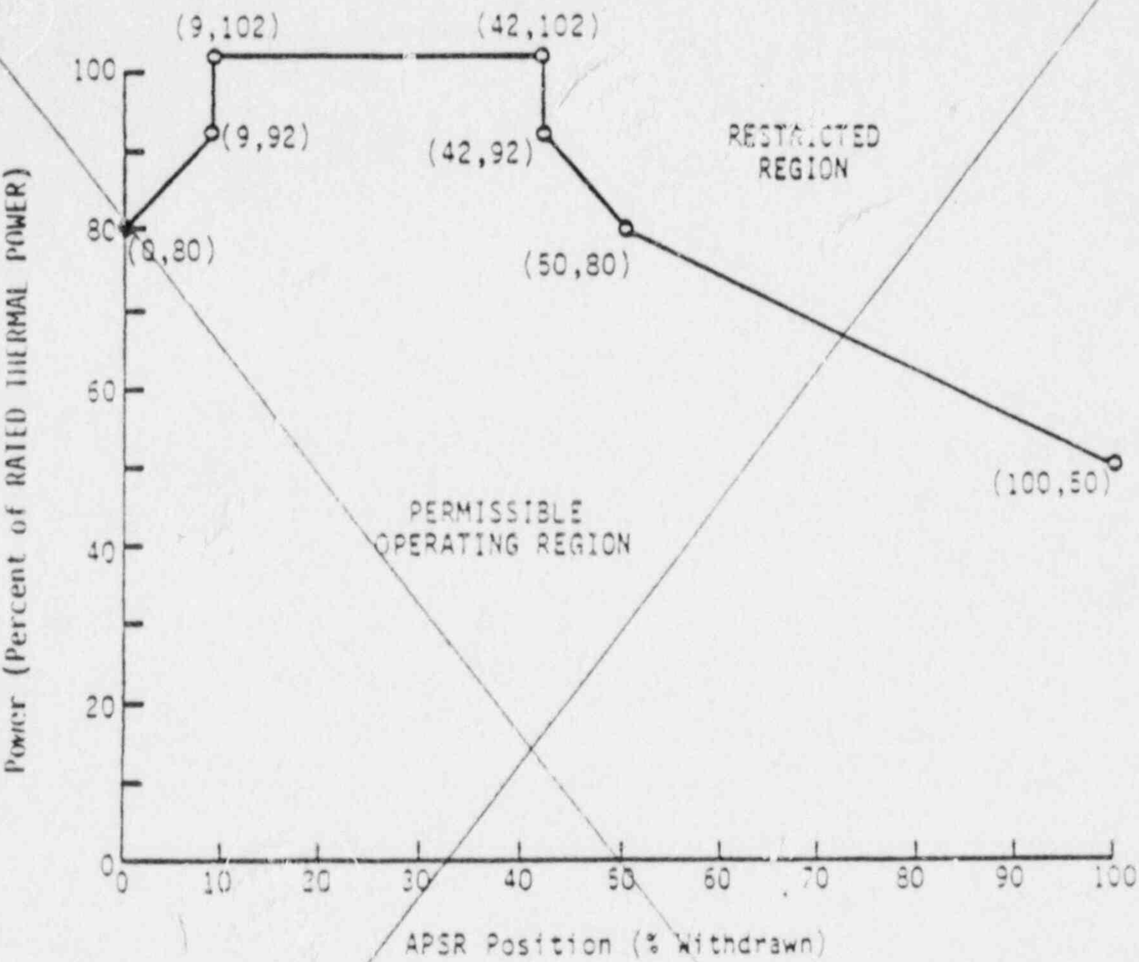
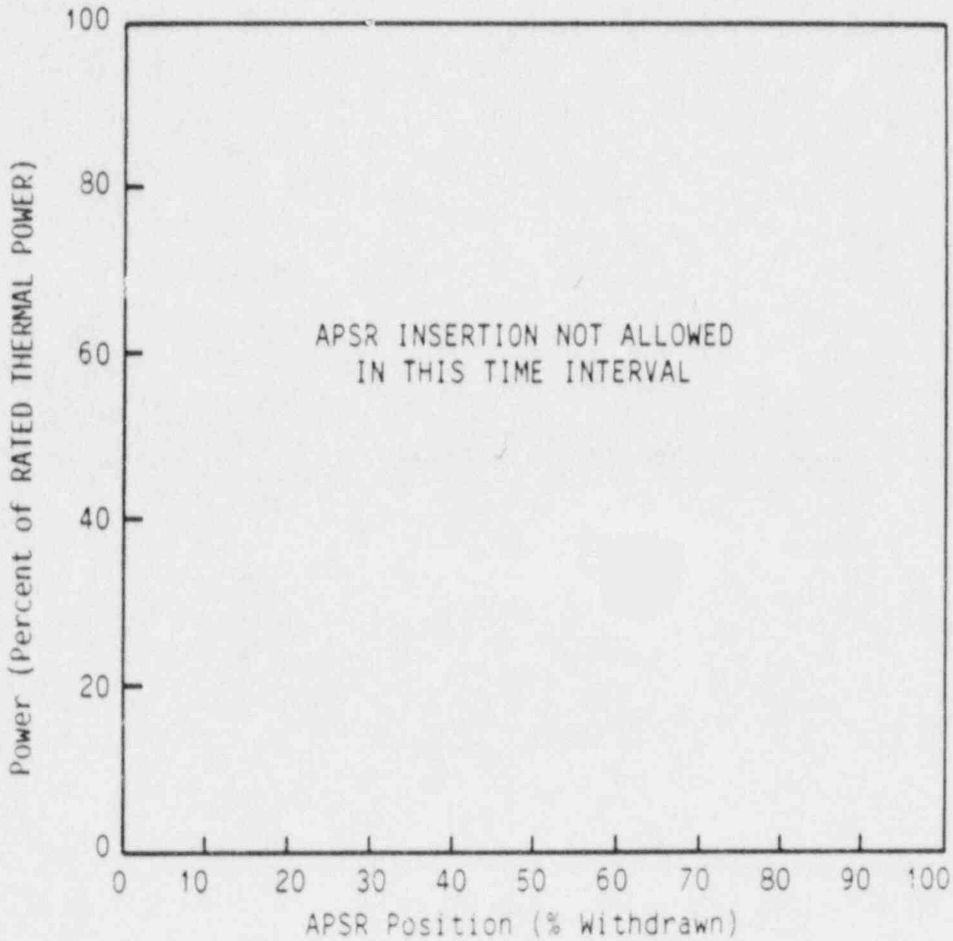


Figure 3.1-5b APSR Position Limits After 325± 10 EFPD, Three or Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6



Superseded with new Figure 3.1-5c

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

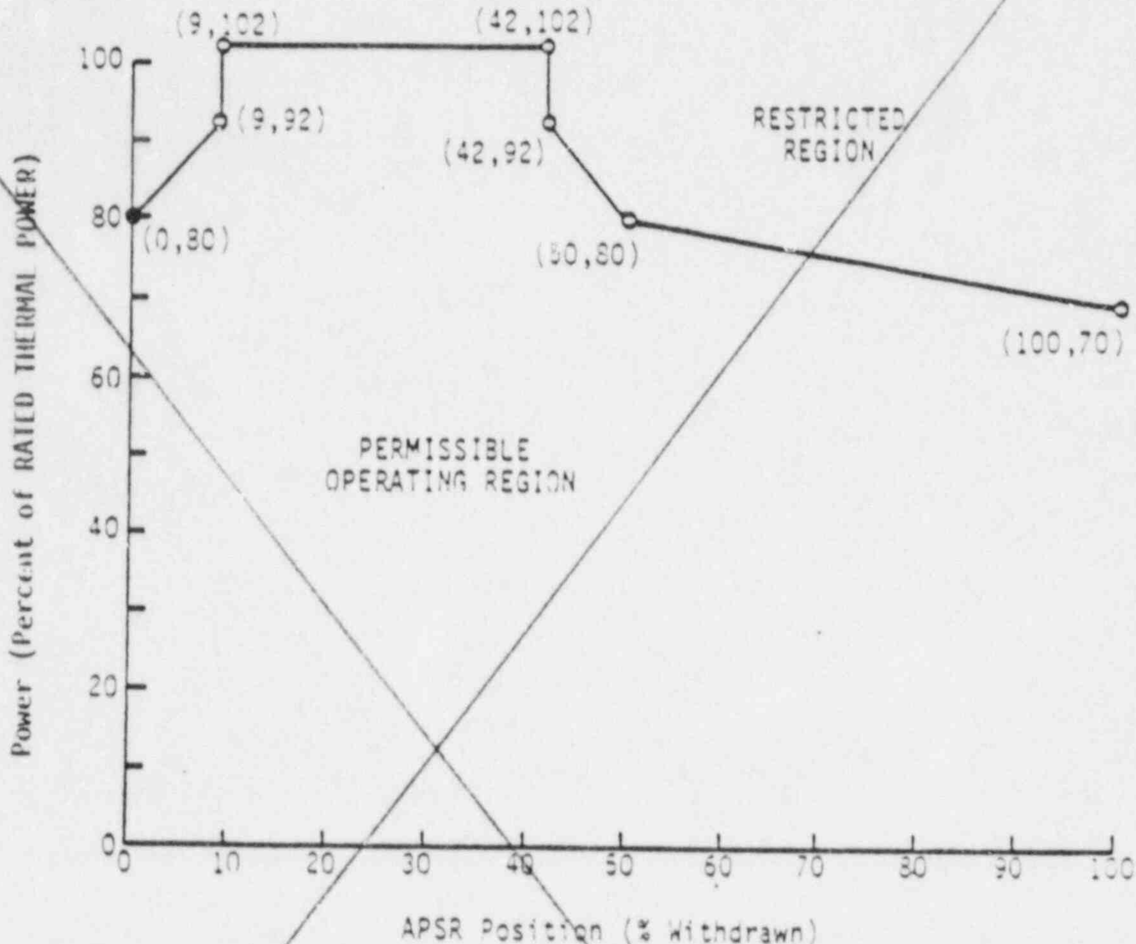


Figure 3.1-5c APSR Position Limits, 0 to 325± 10 EFPD,
Three RC Pumps -- Davis-Besse 1, Cycle 6

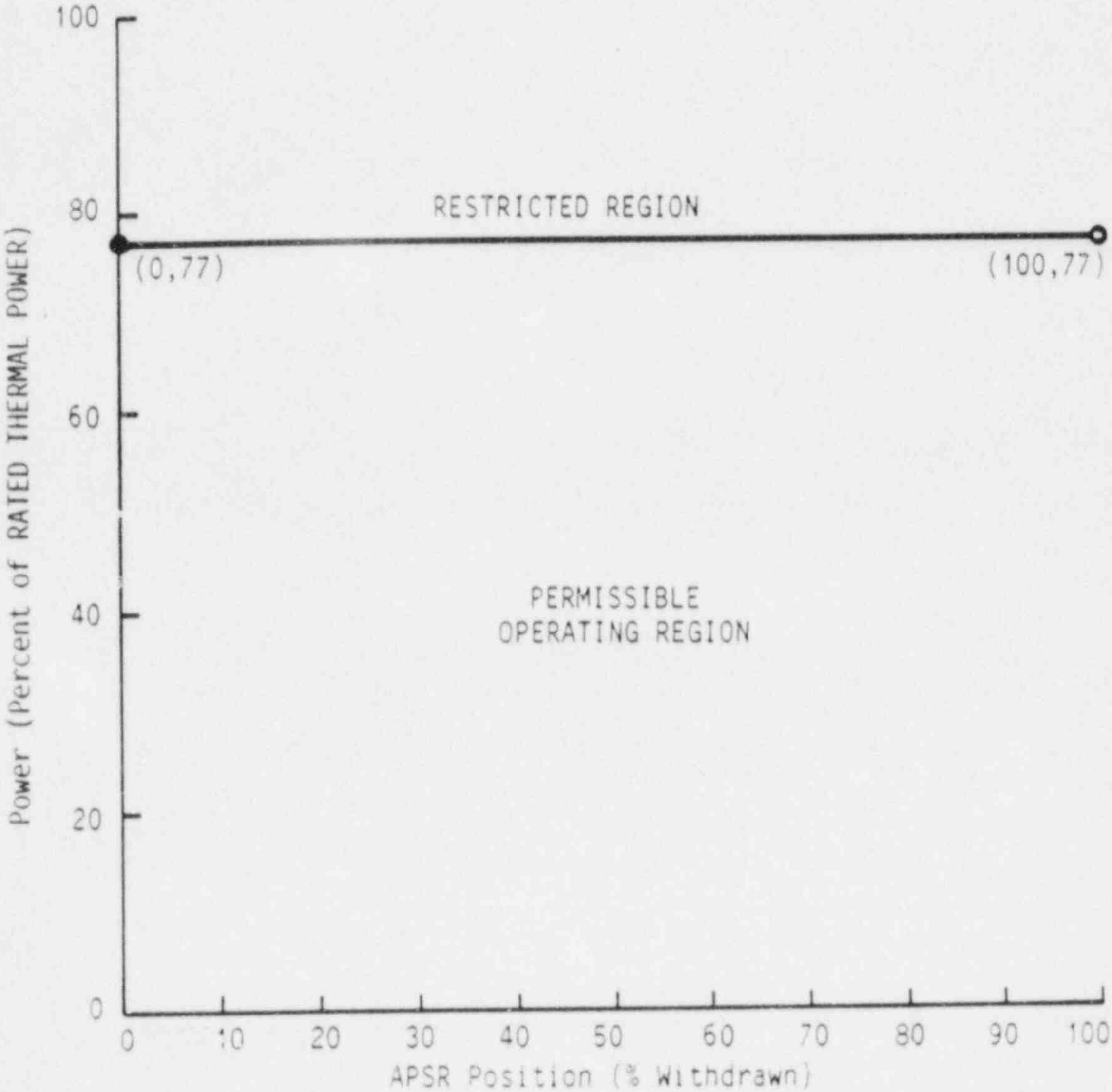


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Figure 3.1-5d APSR Position Limits, 330 ±10 to 390 ±10 EFPD, Three or Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5

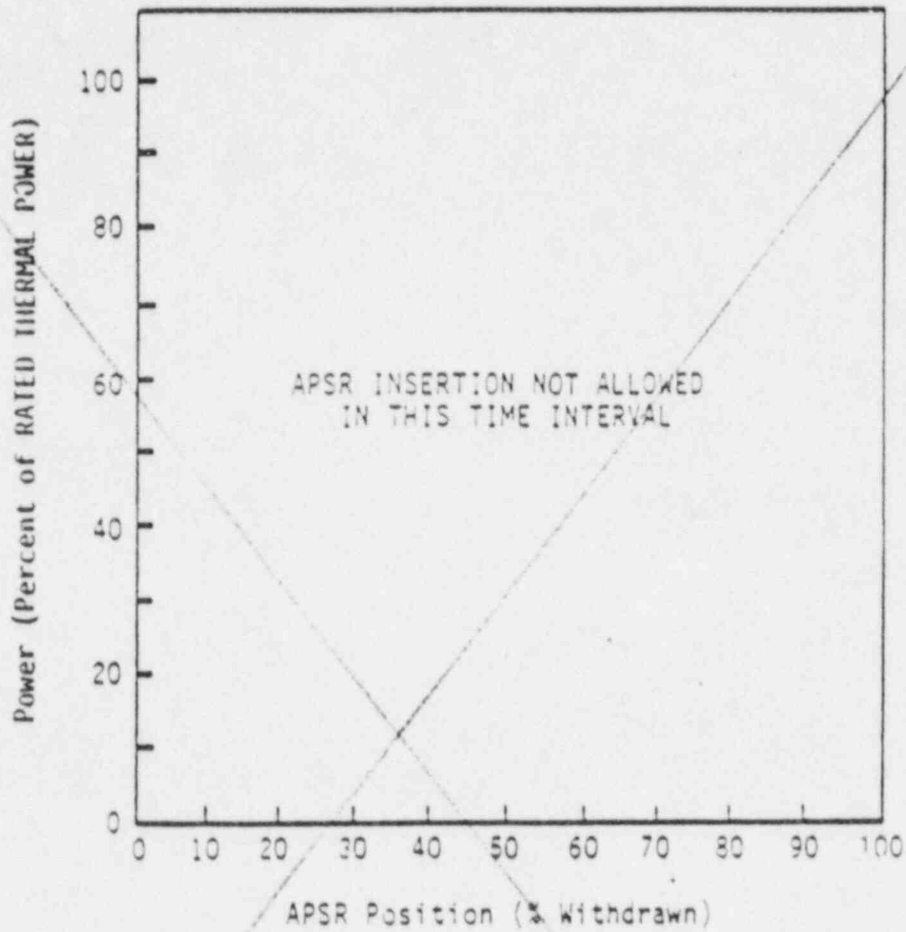

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Figure 3.1-5e APSR Position Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

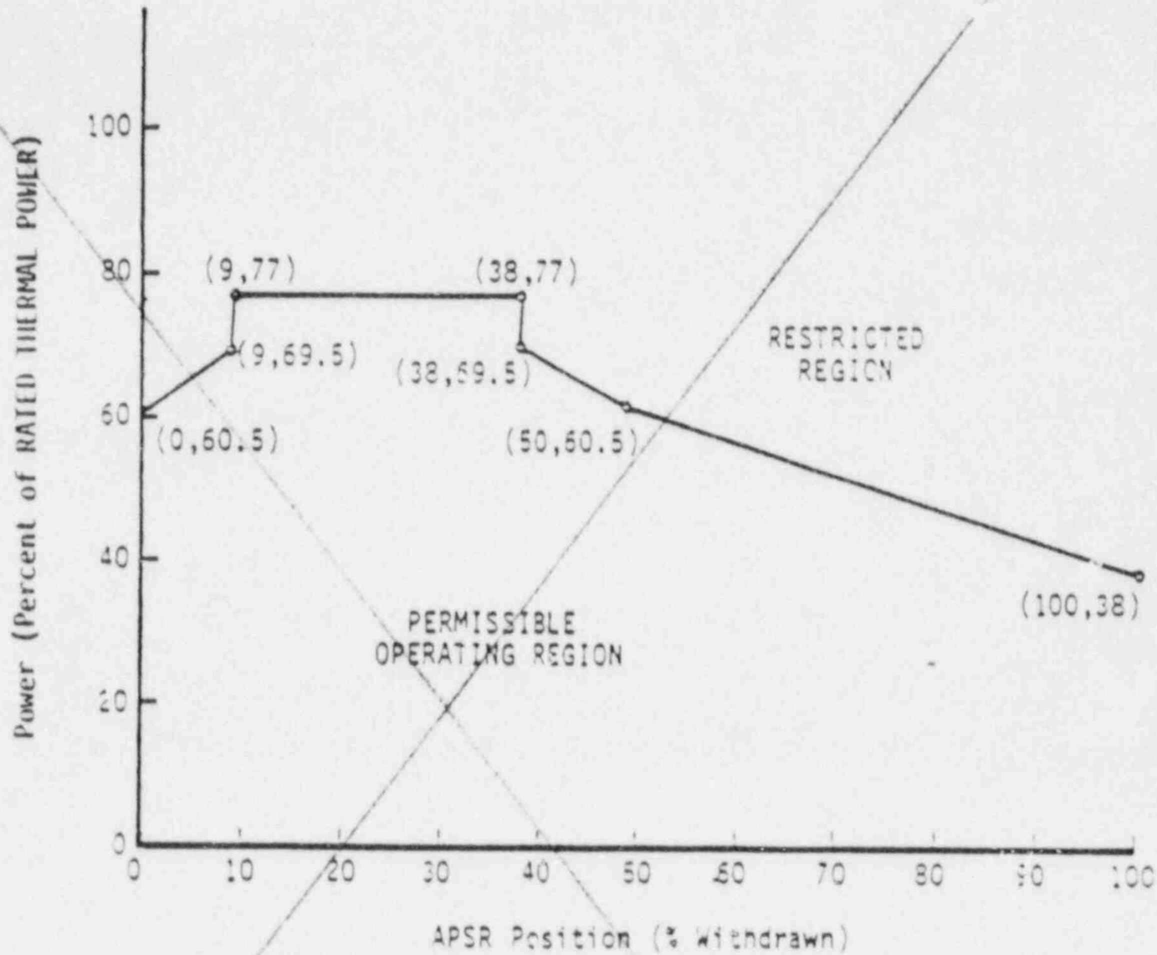

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Figure 3.1-5f APSR Position Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

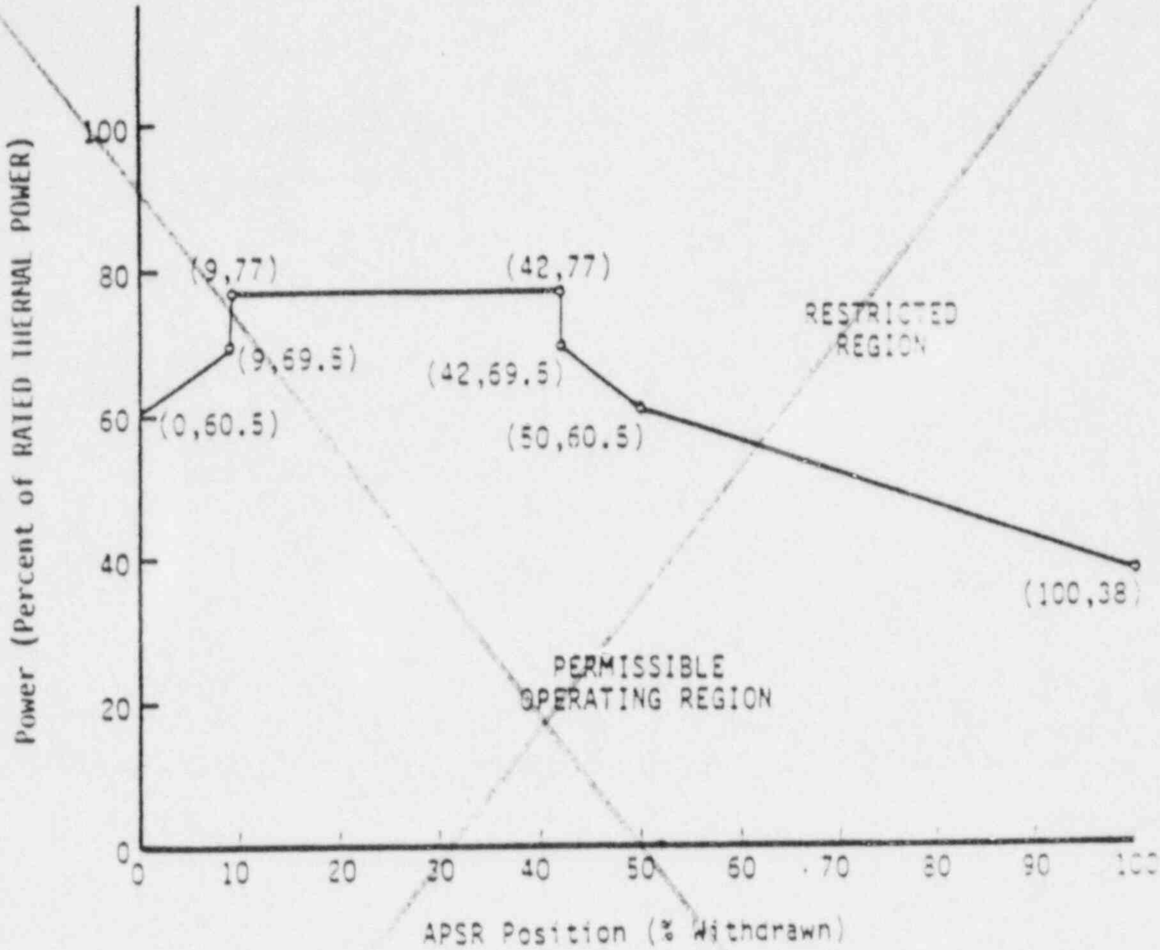

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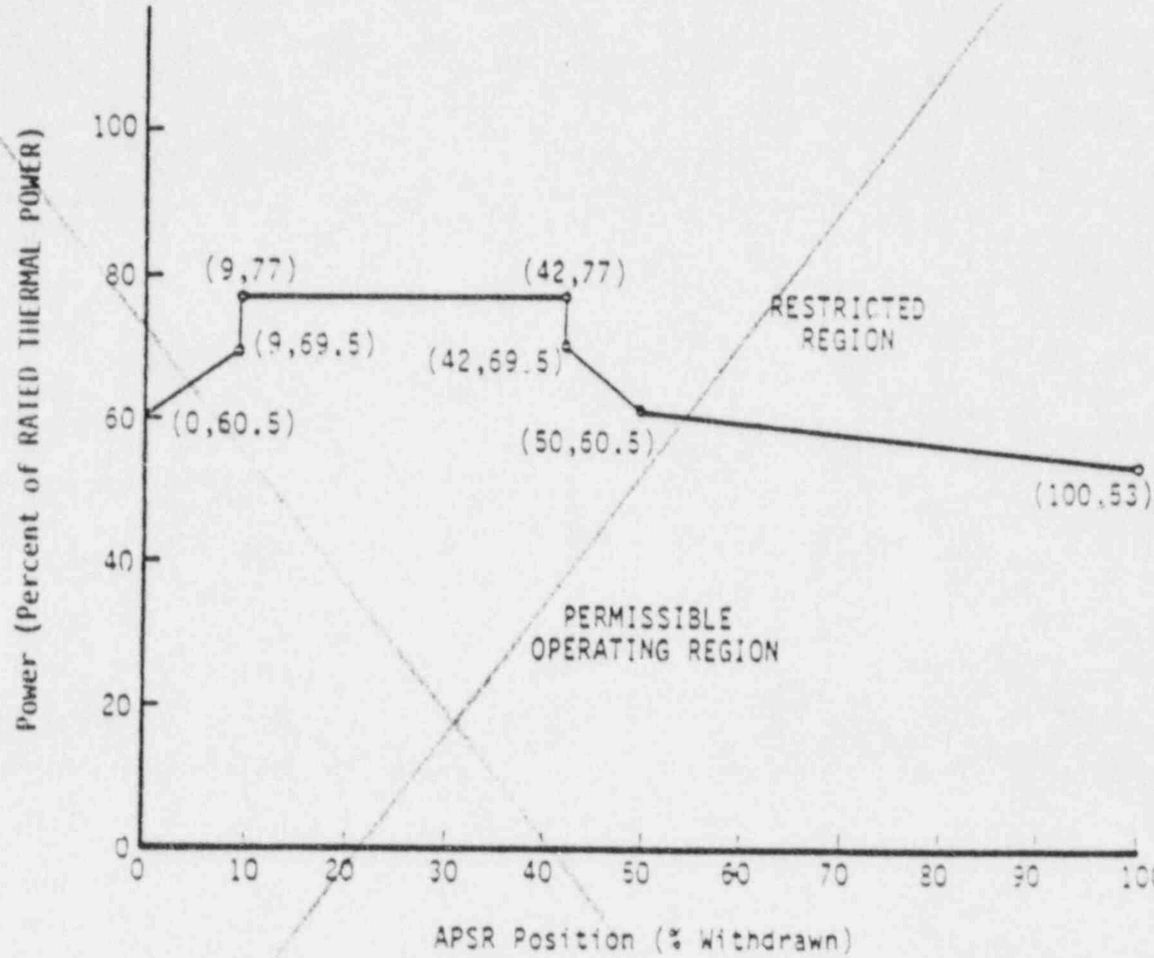
Figure 3.1-5g APSR Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5



ADDITIONAL CHANGES PREVIOUSLY
PROPOSED BY LETTER
Serial No. 1407 Date 11-2-87

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.~~ 3.2-1 and 3.2-2.

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.

Superseded with new Figure 3.2-1

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

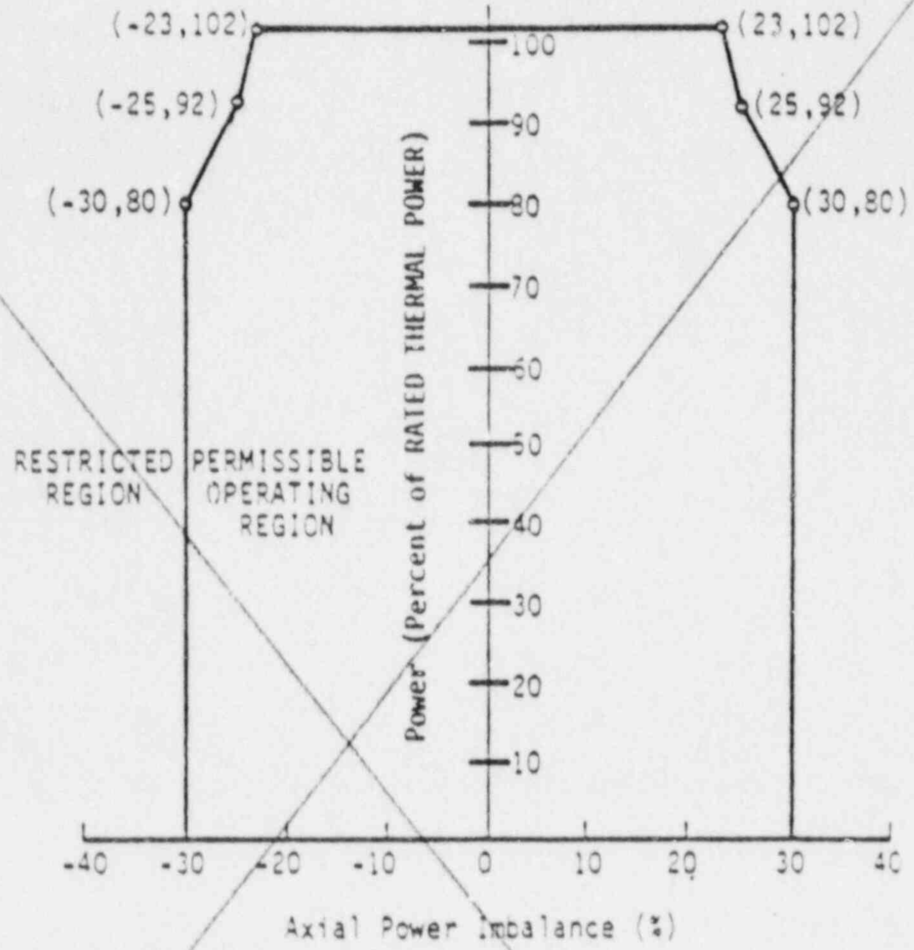


Figure 3.2-1 AXIAL POWER IMBALANCE Limits, Four RC Pumps -- Davis-Besse 1, Cycle 6

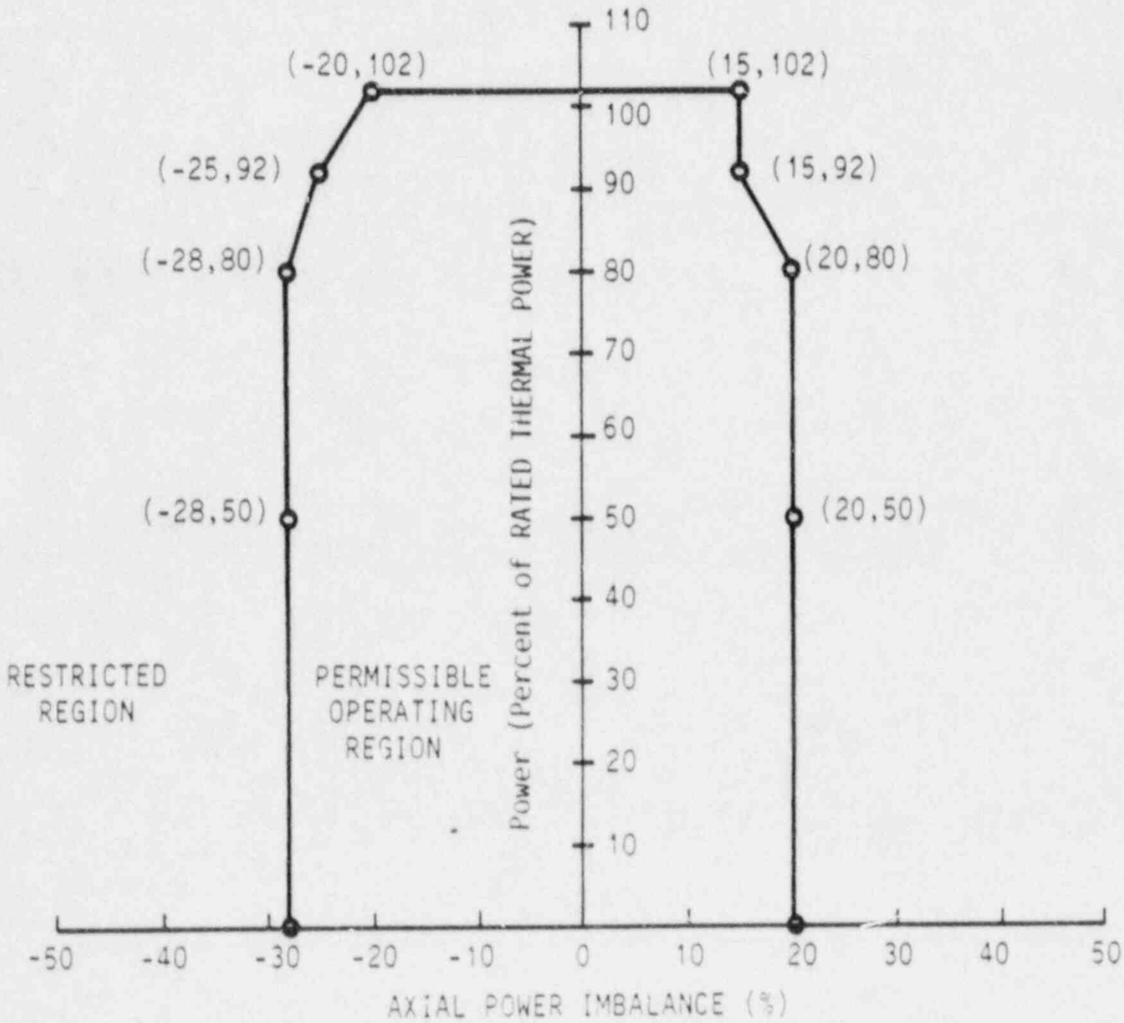


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Figure 3.2-1b Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

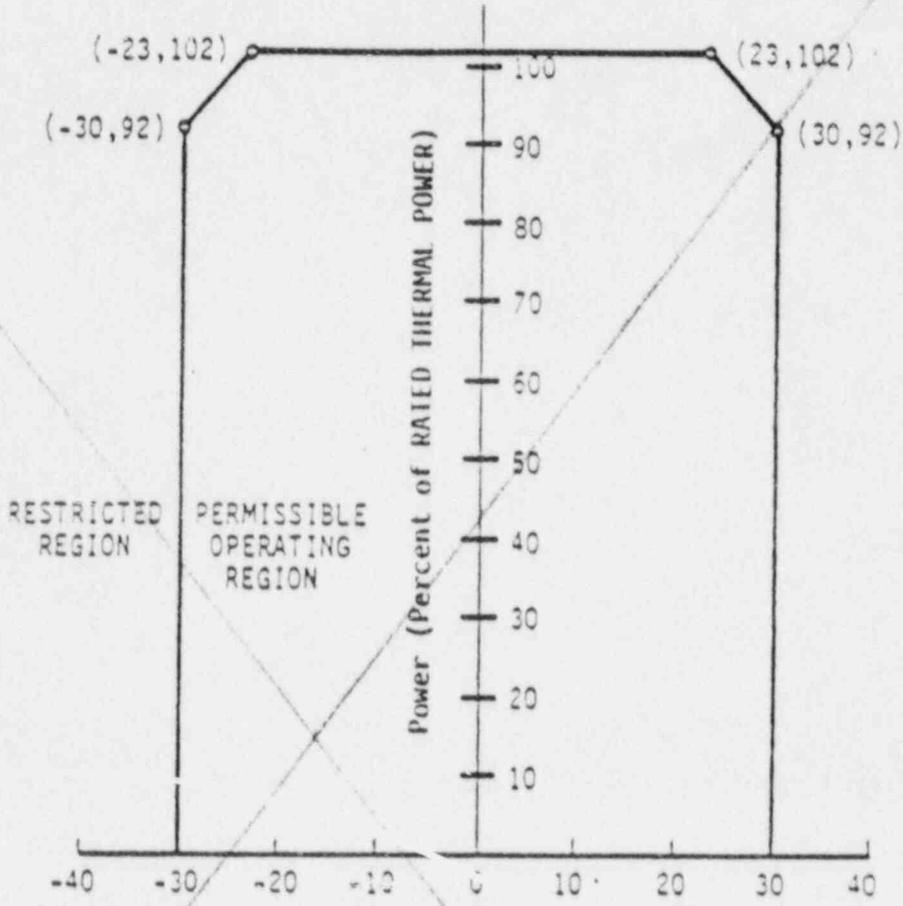

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Figure 3.2-1c Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5

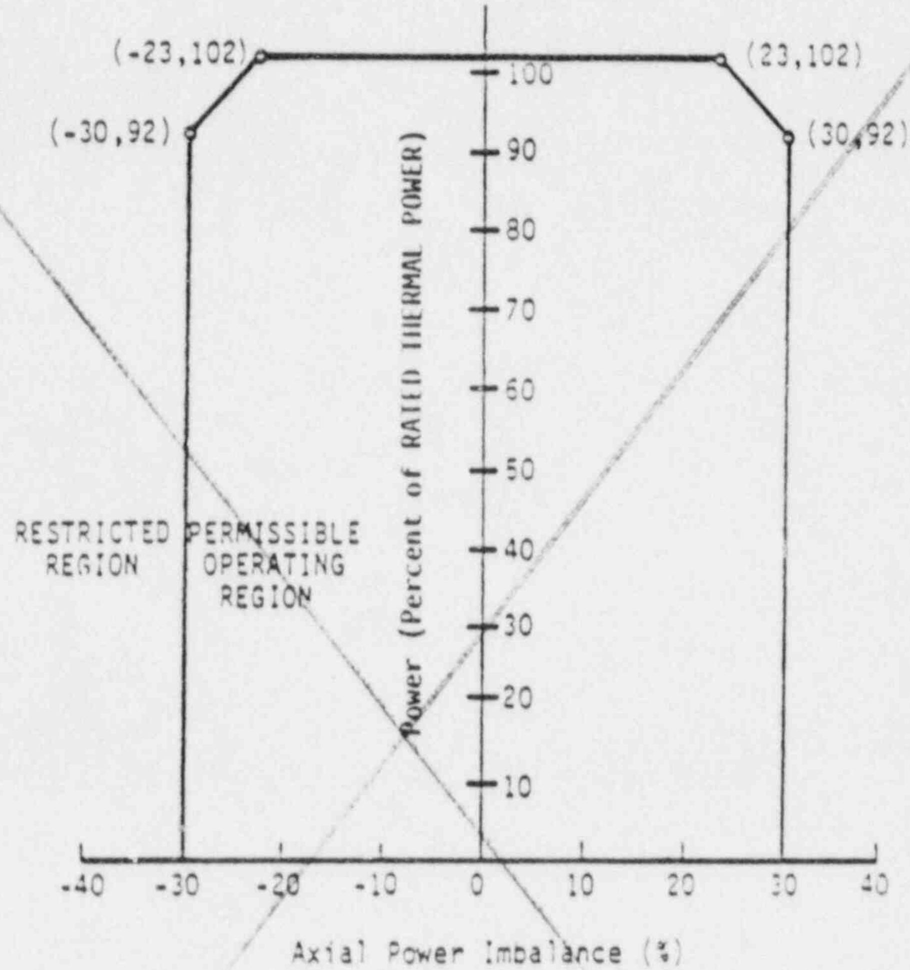

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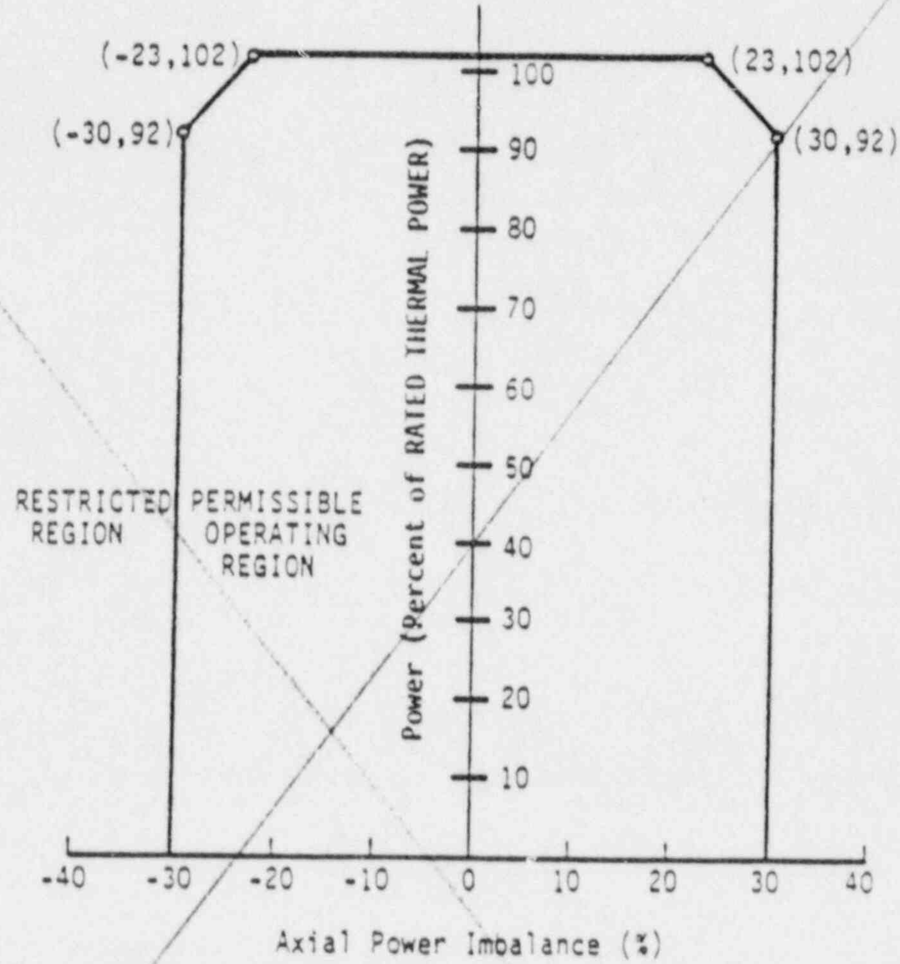
Figure 3.2-1d Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5



Superseded with new Figure 3.2-2

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

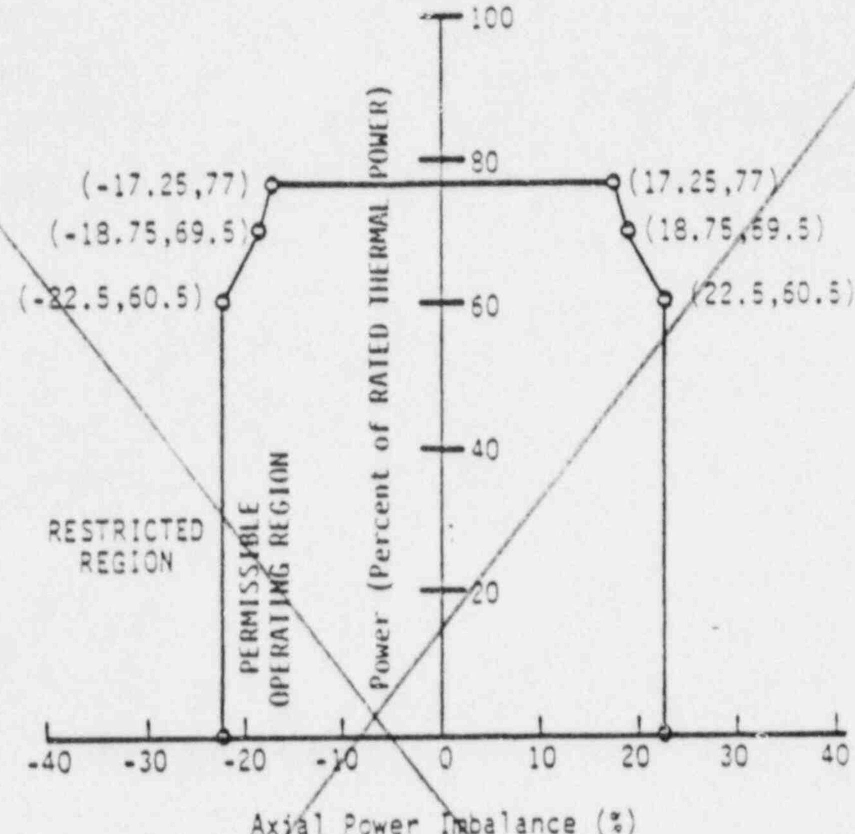


Figure 3.2-2 AXIAL POWER IMBALANCE Limits, Three RC Pumps -- Davis-Besse 1, Cycle 6

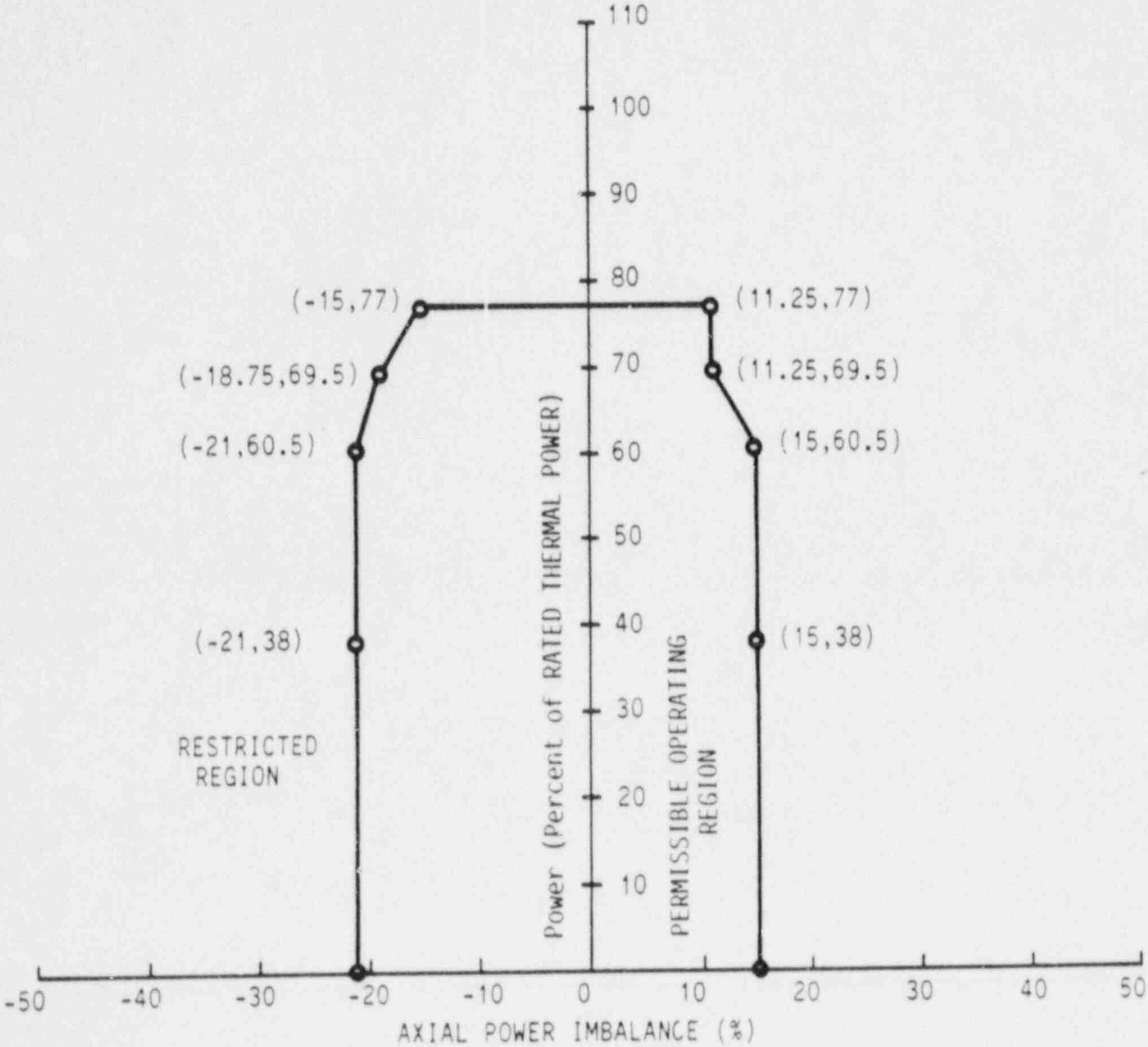


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Figure 3.2-2b Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

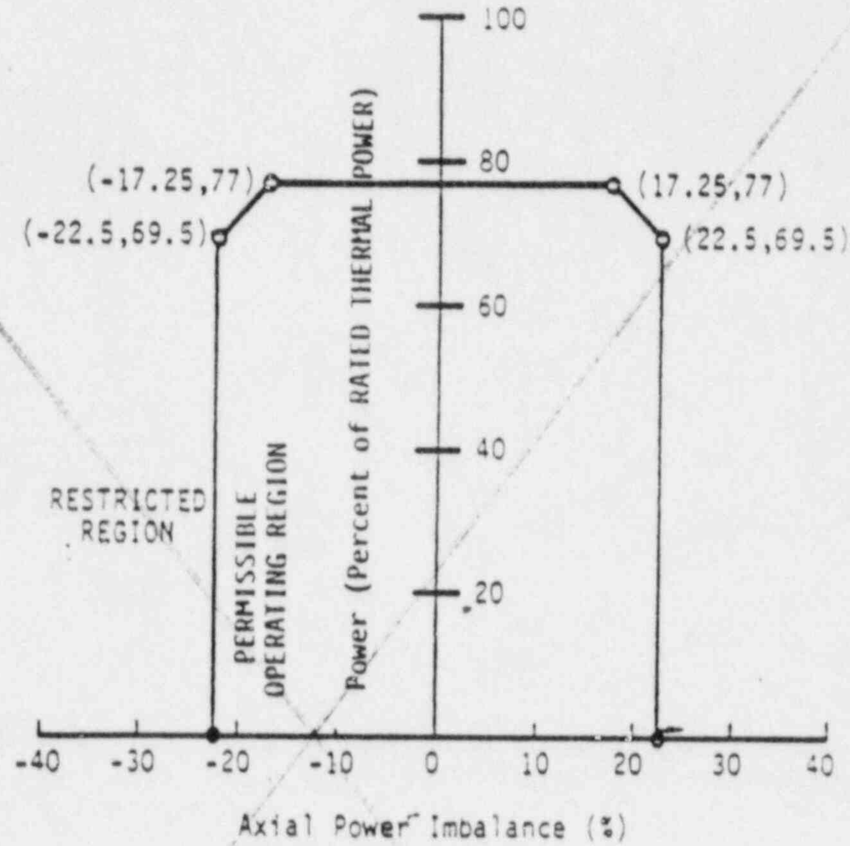

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Figure 3.2-2c Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5

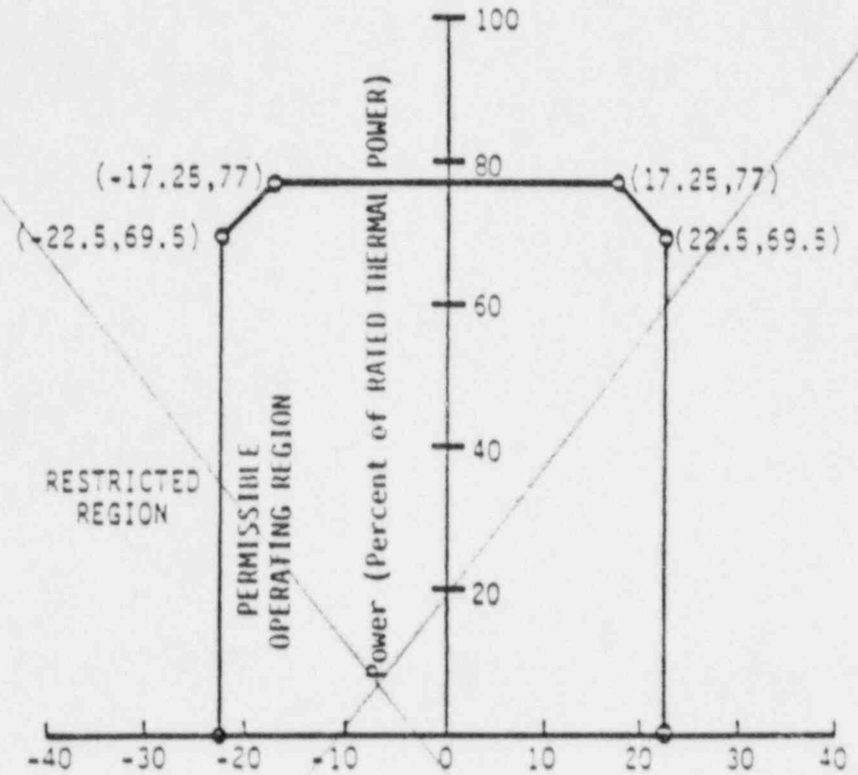

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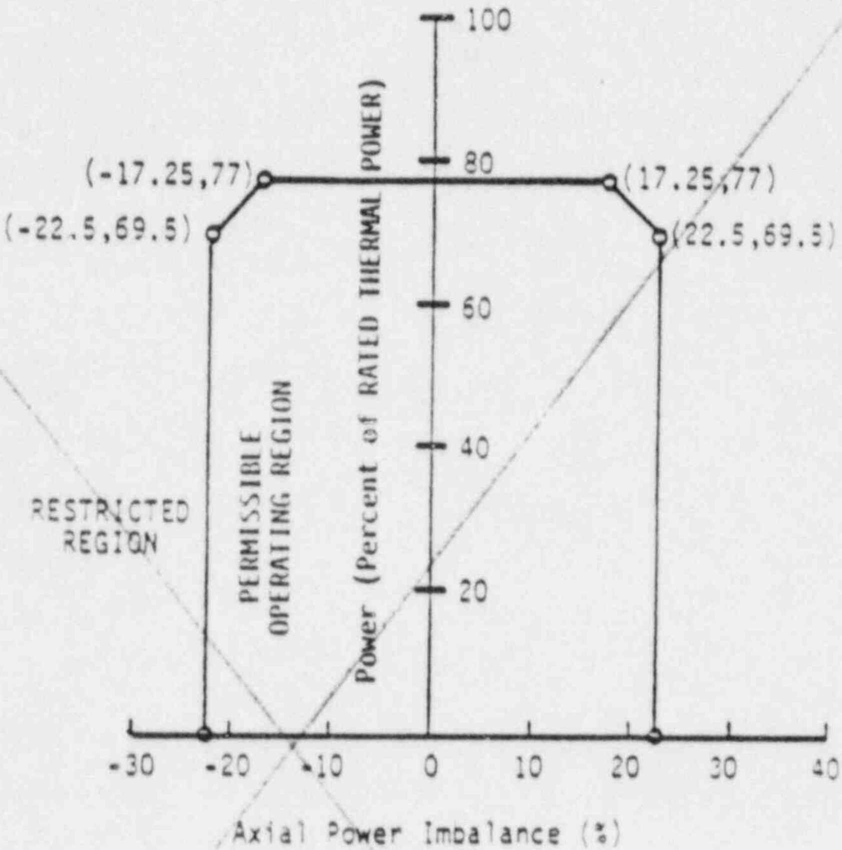
Figure 3.2-2d Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5



QUADRANT POWER TILT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT shall not exceed the Steady State Limit of Table ~~3.2-2.~~ 3.2-1.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER.*

ACTION:

a. With the QUADRANT POWER TILT determined to exceed the Steady State Limit but less than or equal to the Transient Limit of Table ~~3.2-2.~~ 3.2-1.

1. Within 2 hours:

a) Either reduce the QUADRANT POWER TILT to within its Steady State Limit, or

b) Reduce THERMAL POWER so as not to exceed THERMAL POWER, including power level cutoff, allowable for the reactor coolant pump combination less at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit and within 4 hours, reduce the High Flux Trip Setpoint and the Flux-Δ Flux-Flow Trip Setpoint at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit.

2. Verify that the QUADRANT POWER TILT is within its Steady State Limit within 24 hours after exceeding the Steady State Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.1.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 1407 Date 11-2-87

LIMITING CONDITION FOR OPERATION (Continued)

b. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table ~~3.2-2~~ 3.2-1, due to misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER at least 2% for each 1% of indicated QUADRANT POWER TILT in excess of the Steady State Limit within 30 minutes.

2. Verify that the QUADRANT POWER TILT is within its Transient Limit within 2 hours after exceeding the Transient Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table ~~3.2-2~~ 3.2-1, due to causes other than the misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit of Table ~~3.2-2~~ 3.2-1, 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.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 1407 Date 1-2-87

Table ~~3.2-2~~ 3.2-1 Quadrant Power Tilt Limits

	Steady State limit for THERMAL POWER ≤ 50%	Steady state limit (for THERMAL POWER > 50%)	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 ~~, 0-60 ±10 EFPD~~	6.83	4.12 ~~3.37~~	10.03 ~~8.52~~	20.0
~~Symmetrical incore detector system, after 60 ±10 EFPD~~		~~3.02~~	~~8.52~~	~~20.0~~
Power range channels	4.05	1.96	6.96	20.0
Minimum incore detector system	2.80	1.90	4.40	20.0

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table ~~3.2-1~~ 3.2-2.

a. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1

ACTION:

If parameter a or b above exceeds its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

If parameter c exceeds its limit, either:

1. Restore the parameter to within its limit within 2 hours, or
2. Limit THERMAL POWER at least 2% below RATED THERMAL POWER for each 1% parameter c is outside its limit for four pump operation within the next 4 hours, or limit THERMAL POWER at least 2% below 75% of RATED THERMAL POWER for each 1% parameter c is outside its limit for 3 pump operation within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table ~~3.2-1~~ 3.2-2 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE ~~3.2-1~~ 3.2-2

DNB MARGIN

Parameter	Required Measured Parameters with Four Reactor Coolant Pumps Operating	~~LIMITS~~ Required Measured Parameters with Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H °F	$\leq$ 610	$\leq$ 610 (1)
Reactor Coolant Pressure, psig. (2)	$\geq$ 2062.7	$\geq$ 2058.7 (1)
Reactor Coolant Flow Rate, gpm (3)	$\geq$ ~~389,664~~ 389,500	$\geq$ ~~291,080~~ 290,957

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

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

(3) These minimum required measured flows include a flow rate uncertainty of 2.5%, and are based on a minimum of ~~64~~ 52 lumped burnable poison rod assemblies in place in the core.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each Reactor Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

BLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(7)	M	1, 2
3. RC High Temperature	S	R	M	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(7,8)	M	1, 2
5. RC Low Pressure	S	R	M	1, 2
6. RC High Pressure	S	R	M	1, 2
7. RC Pressure-Temperature	S	R	M	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	R	M	1, 2
9. Containment High Pressure	S	R	M	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(5)(1)	1, 2 and*
11. Source Range, Neutron Flux and Rate	S	R(7)	M and S/U(1)(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	M(9) and S/U(1)(9)	1, 2 and*
13. Reactor Trip Module Logic	N.A.	N.A.	M	1, 2 and*
14. Shutdown Bypass High Pressure	S	R	M	2**,3**,4**,5**
15. SCR Relays	N.A.	N.A.	R	1,2 and *

TABLE 4.3-1 (Continued)

NOTATION

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above ~~30%~~ 50% of RATED THERMAL POWER [RTP] and at steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_o] to incore measured AXIAL POWER IMBALANCE [API_I], as ~~Recalibrate if:~~ follows:

$$\frac{RTP}{TP}\ [API_o - API_I]\ \text{~~≥ 3.5%~~} = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is ≥ 2.5%.

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - Verify at least one decade overlap if not verified in previous 7 days.

~~(6) - Each train tested every other month.~~

(7) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(8) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once per 18 months.

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

ADDITIONAL CHANGES PREVIOUSLY
PROPOSED BY LETTER
Serial No. 1407 Date 11-2-87

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%~~ *80.6%* of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced ~~to the values specified in~~ *in accordance with* 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~~ *Reactor Protection System trip setpoints for the* instrumentation channels specified in the applicable ACTION statement above shall be verified to ~~have had their trip setpoints changed to the values specified in~~ *be in accordance with* Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

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

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

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*See Special Test Exception 3.10.3.

EMERGENCY CORE COOLING SYSTEMS

BORATED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The borated water storage tank (BWST) shall be OPERABLE with:

a. An ~~contained~~ available borated water volume of between 482,778 and 550,000 gallons,

b. Between 1800 and 2200 ppm of boron, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The BWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the ~~contained~~ available borated water volume in the tank,

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the water temperature when outside air temperature <35°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

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

3/4.1.2 BORATION SYSTEMS

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

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

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

The requirement~~s~~ for a minimum ~~contained~~ available 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

3/4.1.2 BORATION SYSTEMS (Continued)

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

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to ~~140°F~~ 70°F. This condition requires either ~~8603~~ 600 gallons of ~~[illegible]~~ 7875 ppm borated water from the boric acid storage system or ~~[illegible],200~~ 3,000 gallons of 1800 ppm borated water from the borated water storage tank.

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

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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

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

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

The bottom 4 inches of the borated water storage tank are not available, and the instrumentation is calibrated to reflect the available volume. All boric acid tank volume is available.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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

The bottom 4 inches of the borated water storage tank are not available, and the instrumentation is calibrated to reflect the available volume.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 264°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2500 grams uranium. The initial core loading shall have a maximum enrichment of 3.0 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 53 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. ~~The APSR's shall contain a nominal 36 inches of absorber material at their lower ends.~~ The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100 percent Inconel-600.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.