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December 1992

DAVIS-BESSE NUCLEAR POWER STATION
UNIT 1, CYCLE 9 -- RELOAD REPORT

B&W Fuel Company

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B&W Fuel Company
P.O. Box 10935
Lynchburg, Virginia 24506-0935

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1. INTRODUCTION AND SUMMARY

This report justifies operation of Davis-Besse Nuclear Power Station Unit 1 at the rated core power of 2772 MWt for cycle 9. The required analyses are included as outlined in the Nuclear Regulatory Commission (NRC) document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975. This report utilizes the analytical techniques and design bases that have been submitted to the NRC and approved by that agency.

Cycle 9 reactor and fuel parameters related to power capability are summarized in this report and compared to those for cycle 8. All accidents analyzed in the Davis-Besse Updated Safety Analysis Report[1] (USAR), as applicable, have been reviewed for cycle 9 operation, and in all cases, the initial conditions of the transients in cycle 9 are bounded by previous analyses.

Fuel assembly NJ0542 was reconstituted after cycle 7 and received its second cycle of irradiation during cycle 8. This assembly, with one stainless steel replacement rod, will be used again during cycle 9. The effect of the replacement rod on thermal-hydraulic performance is discussed in section 6.

The Technical Specifications have been reviewed for cycle 9 operation. Based on the reload report analyses performed, taking into account the emergency core cooling system (ECCS) Final Acceptance Criteria and postulated fuel densification effects, it is concluded that Davis-Besse Unit 1, cycle 9 can be operated safely at its licensed core power level of 2772 MWt. The COLR changes for cycle 9 are included in section 8 of this report.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Davis-Besse Unit 1 is the currently operating cycle 8, which achieved criticality on November 5, 1991. Power escalation began on November 7, 1991 and full power (2772 MWt) was attained on November 13, 1991.

During cycle 8 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 9. The nominal length of cycle 9 is 490 effective full power days (EFPD). Cycle 9 was analyzed to 500 EFPD based on cycle 8 operation of 453 + 15/-30 EFPD. The applicability of the cycle 8 reactor protection system (RPS) limits and setpoints to cycle 9 has been verified to 500 EFPD. The cycle 9 operating limits have also been verified to 500 EFPD. The cycle 9 design includes an APSR pull and power coastdown.

The cycle 9 design minimizes the number of fuel assemblies that are cross core shuffled to reduce the potential for quadrant tilt amplification. The cycle 9 shuffle pattern is discussed in section 3.

3. GENERAL DESCRIPTION

The Davis-Besse Unit 1 reactor core is described in detail in chapter 4 of the USAR[1] for the unit. The cycle 9 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array normally containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. All FAs in batches 8, 9, and 10 have a constant nominal fuel loading of 468.25 kg of uranium. The batch 11 FAs have a constant nominal fuel loading of 468.56 kg of uranium. The fuel consists of dished-end cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy 4. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters may be found in Table 4-1 of this report.

Figure 3-1 is the core loading diagram for Davis-Besse Unit 1, cycle 9. Fifty-seven batch 8B assemblies and 8 batch 9B assemblies will be discharged at the end of cycle 8. The remaining batch 9B and batch 10 FAs will be shuffled to their cycle 9 locations, with the core periphery locations occupied by both batch 9B and batch 10 fuel assemblies. One batch 8C assembly, discharged at the end of cycle 7, will be reinserted in cycle 9 as the center FA. Batches 8C, 9B and 10 have initial enrichments of 3.13, 3.38, and 3.69 wt %, respectively. The feed batch, consisting of 64 batch 11 assemblies with uranium enrichment of 3.77 wt %, will be inserted in the core interior in a symmetric checkerboard pattern. This shuffle scheme is a very low leakage (VLL) core loading. The implementation of the VLL reload fuel shuffle scheme for cycle 9 will have a negligible effect on nuclear instrumentation response for all aspects of reactor startup and subsequent power operation. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 9 and its initial enrichment.

Cycle 9 is operated in a feed-and-bleed mode. The core reactivity is controlled by 53 full-length Ag-In-Cd control rod assemblies (CRAs), 56 burnable poison rod assemblies (BPRAs), and soluble boron. Eight of the BPRAs will be reinserted from cycle 7. Eight standard control rods will be replaced in cycle 9 with

extended life control rods that are described in section 4.1. In addition to the full-length control rods, eight Inconel-600 axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The cycle 9 locations of the control rods and the group designations are indicated in Figure 3-3. The core locations and the rod group designations of the 61 control rods in cycles 8 and 9 are the same. The cycle 9 locations and errichments of the BPRAs are shown in Figure 3-4.

Figure 3-1 Davis-Besse Cycle 9 Core Loading Diagram

← North

X (top of column 8); W (left of row H); Y (right of row H); Z (bottom of column 8)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						9B O09	10 F07	9B L02	10 F09	9B O07					
B				9B O11	10 L05	10 E12	11 F	10 M08	11 F	10 E04	10 L11	9B O05			
C			9B H07	11 F	10 G14	11 F	10 F13	11 F	10 F03	11 F	10 G02	11 F	9B G08		
D		9B M13	11 F	10 C04	11 F	10 G12	11 F	9B P11	11 F	10 G04	11 F	10 D13	11 F	9B M03	
E		10 E10	10 P07	11 F	9B M14	11 F	9B A07	11 F	9B A09	11 F	9B P05	11 F	10 P09	10 E06	
F	9B K13	10 N05	11 F	10 N07	11 F	9B R08	11 F	10 O08	11 F	9B H01	11 F	10 N09	11 F	10 N11	9B K03
G	10 G06	11 F	10 O06	11 F	9B G01	11 F	9B F02	10 O04	9B B10	11 F	9B G15	11 F	10 O10	11 F	10 G10
H	9B P10	10 H11	11 F	9B E14	11 F	10 H13	10 N13	8C M14 7	10 D03	10 H03	11 F	9B M02	11 F	10 H05	9B B06
K	10 K06	11 F	10 C06	11 F	9B K01	11 F	9B P06	10 C12	9B L14	11 F	9B K15	11 F	10 C10	11 F	10 K10
L	9B G13	10 D05	11 F	10 D07	11 F	9B H15	11 F	10 C08	11 F	9B A08	11 F	10 D09	11 F	10 D11	9B G03
M		10 M10	10 B07	11 F	9B B11	11 F	9B R07	11 F	9B R09	11 F	9B E02	11 F	10 B09	10 M06	
N		9B E13	11 F	10 N03	11 F	10 K12	11 F	9B B05	11 F	10 K04	11 F	10 O12	11 F	9B E03	
O			9B K08	11 F	10 K14	11 F	10 L13	11 F	10 L03	11 F	10 K02	11 F	9B H09		
P				9B C11	10 F05	10 M12	11 F	10 E08	11 F	10 M04	10 F11	9B C05			
R						9B C09	10 L07	9B F14	10 L09	9B C07					

key:

xxx	xxx - batch no.
yyy	yyy - previous cycle location
zzz	zzz - previous cycle if reinsert

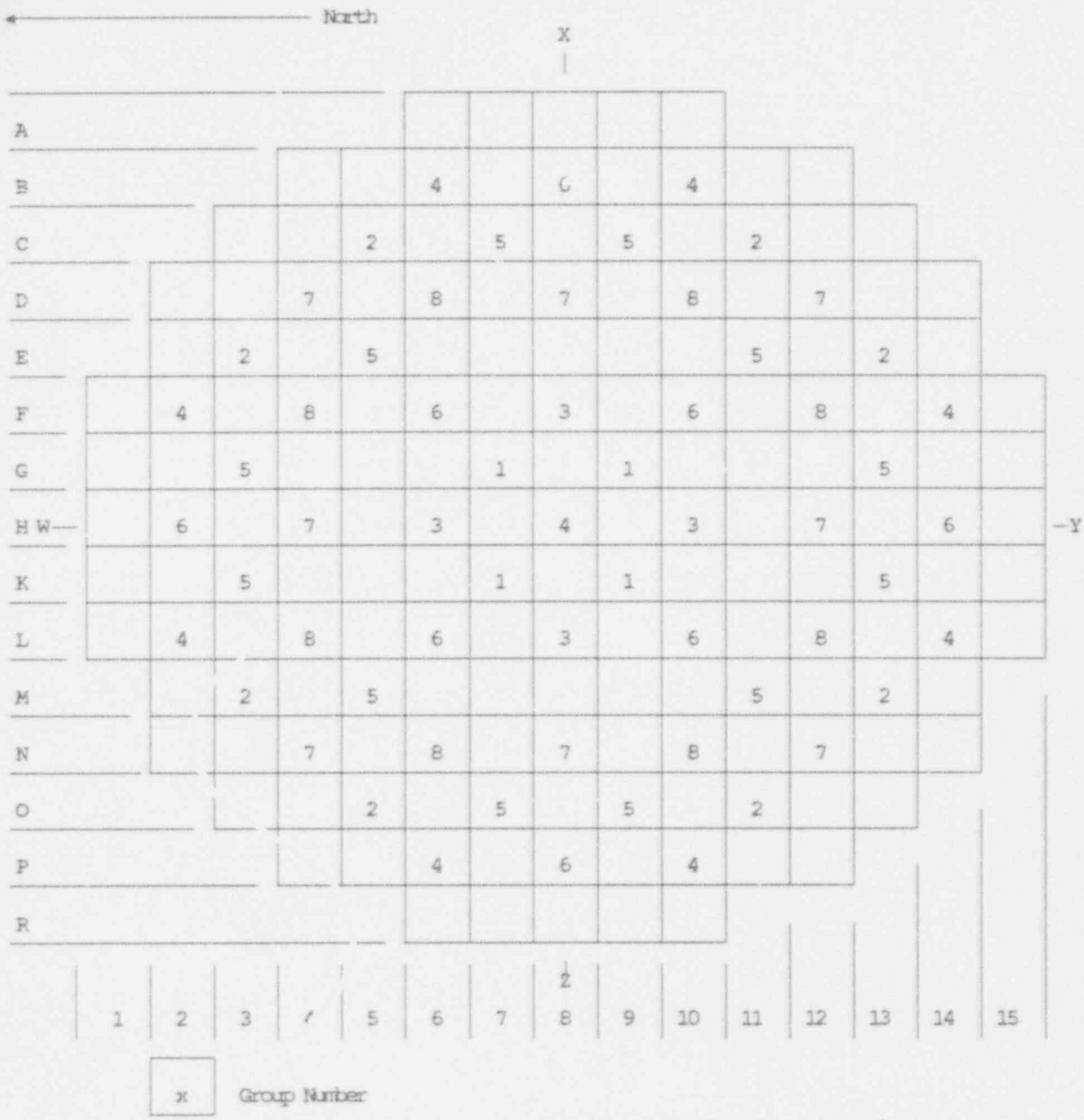
Note: One steel rod in P12.

Figure 3-2 Davis-Besse Cycle 9 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.13 24604	3.69 15594	3.69 20313	3.77 0	3.38 26514	3.77 0	3.69 20666	3.38 28666
K	3.69 15594	3.38 28608	3.77 0	3.38 25892	3.77 0	3.69 19731	3.77 0	3.69 20799
L	3.69 20313	3.77 0	3.38 22330	3.77 0	3.69 20759	3.77 0	3.69 20244	3.38 36204
M	3.77 0	3.38 25927	3.77 0	3.38 26491	3.77 0	3.69 17552	3.69 20926	
N	3.38 26514	3.77 0	3.69 20709	3.77 0	3.69 15611	3.77 0	3.38 33293	
O	3.77 0	3.69 19735	3.77 0	3.69 17601	3.77 0	3.38 30767		
P	3.69 20666	3.77 0	3.69 20207	3.69 20839	3.38 33276			
R	3.38 28666	3.69 20794	3.38 36163					

x.xx yyyyy	Initial Enrichment BOC Burnup MWd/mtU

Figure 3-3 Davis-Besse Cycle 9 Control Rod Locations



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	

Figure 3-4 Davis-Besse Cycle 9 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				1.7		1.7		
K			1.7		2.3		0.0†	
L		1.7		2.3		1.7		
M	1.7		2.3		1.7			
N		2.3		1.7				
O	1.7		1.7					
P		0.0†						
R								

† These BPRAs are reinserted from cycle 7.

x.x	Initial BPRA Concentration, wt% B_4C in Al_2O_3.

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel parameters for Davis-Besse cycle 9 are listed in Table 4-1. Batch 8C is the Mark-B5A design, batches 9B and 10 are the Mark-B8A design, and batch 11 is the Mark-B8B design. Batch 10 fuel incorporates all of the features of the batch 9 fuel but includes a reduction in pre-pressure to increase the similarity in mechanical/thermal performance to that of the Mark-B5A design. Batch 11 fuel, the Mark-B8B design, consists of a Mark-B8 cage with Mark-B9A fuel rods. Compared to the Mark-B8A fuel rods in batches 9 and 10, the Mark-B9A fuel rods have larger OD fuel pellets, shorter fuel stack height, higher uranium loading, higher percent theoretical density, and a lower backfill pressure.

Eight gray APSRs and 53 full length Ag-In-Cd control rods will be used in cycle 9. Forty-eight new Mark-B5 BPRAs will be introduced into the core along with 8 once-burned Mark-B5 BPRAs (burned 405 EFPD in cycle 7) for a total of 56 BPRAs. In terms of creep collapse, stress, strain, and corrosion, the Mark-B5 BPRAs were found to be mechanically adequate for irradiation up to 1000 EFPD.

Eight of the 53 control rod assemblies (CRAs) are of the extended life control rod assembly design (ELCRA). The design differences between the ELCRAs and the CRAs that are replaced are:

Cladding thickness was increased from 18 to 21.5 mils.

Cladding material was changed from 304ss to Inconel 625.

Fill gas pressure was changed from atmospheric to 465 psig.

Ag-In-Cd absorber diameter was sized to increase clad to absorber gap by 2 mils.

Poison length was increased from 134 to 139 inches.

The design changes result in an in-core life of 30 calendar years of 22 EFPY with a limiting thermal fluence of 2.78×10^{22} n/cm^2. The limiting condition is cladding strain.

4.2 Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1 Cladding Collapse

The most limiting power history for each of the four fuel batches was determined. These histories were compared to generic and previous creep collapse analyses based on the methods from references 2 and 3. A previous analysis based on reference 2 was found to be applicable to the batch 8C design for cycle 9 operation. For batches 9B and 10, as in the cycle 8 analysis, the creep collapse analysis followed the methodology from reference 3. The batch 11 creep collapse life was determined using a new analysis based on the method from reference 2. See Table 4-1 for results.

4.2.2 Cladding Stress

The stress parameters for the two fuel rod designs are enveloped by conservative generic fuel rod stress analyses. For design evaluation, certain stress intensity limits for all condition I and II events must be met. Limits are based on ASME criteria. Stress intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities are compared to Sm. Sm is equal to two-thirds of the minimum specified unirradiated yield strength of the material at the operating temperature range (650 deg F). The stress intensity limits are as follows:

Pm	< 1.0 Sm
Pl	< 1.5 Sm
Pm + PB	< 1.5 Sm
Pm + Pb + Q	< 3.0 Sm

Pm: General Primary Membrane Stress Intensity

Pl: Local Primary Membrane Stress Intensity

Pb: Primary Bending Stress Intensity

Q: Secondary Stress Intensity

Stress Intensity calculations combine stresses so that the resulting stress intensity is maximized.

For both fuel rod designs, the margins are in excess of 12.0%. The following conservatisms were used in the stress analyses to ensure that all condition I and

II operating parameters were enveloped:

1. Low post-densification internal pressure, or as-built prepressure.
2. High system pressure.
3. High thermal gradient across the cladding.
4. Minimum specified cladding thickness.

4.2.3 Cladding Strain

The fuel design criteria specify a limit of 1% cladding plastic tensile circumferential strain of the cladding. The fuel pellet is designed to ensure that this strain is less than 1% at the design local pellet burnup and heat generation rate. The design values are higher than the worst case values Davis-Besse Unit 1, cycle 9 fuel is expected to experience. For the batch 8C, Mark-B5A fuel assemblies, a generic strain analysis was reviewed and judged to be conservative based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance limit for the cladding inside diameter. For the Mark-B8A fuel assemblies from batches 9B and 10 and the Mark-B8B fuel assemblies from batch 11, the strain analysis was done utilizing the method of reference 5.

4.3 Thermal Design

All fuel in the cycle 9 core is thermally similar. The design of the batch 11 Mark B8B assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. The analysis for the Mark-B8B fuel was performed with the TACO3 code as described in reference 6. Fuel performance for the fuel remaining or reinserted in the core was evaluated with the TACO2 code as described in reference 5. Nominal undensified input parameters used in the analysis are presented in Table 4-1. Densification effects were accounted for in the TACO2 and TACO3 code densification models.

The results of the thermal design evaluation of the cycle 9 core are summarized in Table 4-1. Cycle 9 core protection limits were based on linear heat rate (LHR) to centerline fuel melt limits determined by the TACO2 and TACO3 codes.

The maximum fuel pin burnup at EOC 9 is predicted to be less than 49,200 MWd/mtU (batch 9B). The fuel rod internal pressures have been evaluated with TACO2 or TACO3 for the highest burnup of each fuel rod type and are predicted to be less

than the nominal reactor coolant pressure of 2200 psia.

4.4 Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interaction for batch 11 fuel assemblies is identical to that of present fuel assemblies.

4.5 Operating Experience

B&W Fuel Company operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. The following experience has been accumulated for eight B&W 177 fuel assembly plants using the Mark B fuel assembly:

Reactor	Current Cycle	Max FA Burnup, MWd/mtU[a] Incore	Discharged	Cumulative net electric output, MWh [b]
Oconee 1	14	35,085	58,310	96,809,743
Oconee 2	12	43,208	42,820	92,442,224
Oconee 3	13	38,217	42,740	91,238,420
TMI-1	9	28,752	36,538	58,157,209
Arkansas Nuclear One, Unit 1	10	42,257	57,318	74,375,248
Rancho Seco	7	[c]	38,268	43,208,092
Crystal River 3	8	35,770	40,600	60,431,740
Davis-Besse	8	30,600	41,820	48,792,488

[a] As of October 31, 1991.

[b] As of December 31, 1991.

[c] Plant Shutdown in June 1989 and core unloaded.

Table 4-1 Fuel Design Parameters

	Batch 8C	Batch 9B	Batch 10	Batch 11
Fuel assembly type	Mark-B5A	Mark-B8A	Mark-B8A	Mark-B8B
No. of assemblies	1	48	64	64
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377
Tubular spacer	Zr-4	NA	NA	NA
Undensified active fuel length, in.	143.2	143.2	143.2	140.6
Pellet OD, in.	0.3686	0.3686	0.3686	0.3700
Fuel pellet initial density, %TD mean	95.0	95.0	95.0	96.0
Initial fuel batch enrichment, w/o U235	3.13	3.38	3.69	3.77
Average burnup BOC, MWd/mtU	24,604	29,511	19,505	0
Cladding collapse burnup, MWd/mtU	>50,175[a]	>55,000[b]	>55,000[b]	>55,000[a]
Maximum assembly burnup, MWd/mtU	38,721	44,411	40,275	22,619
Nominal linear heat rate at 2772 MWt, kW/ft	6.14	6.14	6.14	6.25
Minimum linear heat rate to melt, kW/ft	20.5	20.5	20.5	22.3

[a] Calculated using method from reference 2.
[b] Calculated using method from reference 3.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 8 and 9 designs. The values for cycle 8 were generated with the NOODLE code[7], while the values for cycle 9 were generated with the NEMO code[8]. Differences in core physics parameters are to be expected between the cycles due to the changes in fuel and burnable poison enrichments which create changes in radial flux and burnup distributions. Figure 5-1 illustrates a representative relative power distribution for BOC 9 at full power with equilibrium xenon, all rods out, and gray APSRs inserted.

The ejected rod worths in Table 5-1 are the maximum calculated values. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in section 8. The adequacy of the shutdown margin with cycle 9 rod worths is shown in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. Xenon transient allowance.
4. A maximum power deficit.

The xenon transient allowance was taken into account to ensure that the effects of operational maneuvering transients were included in the shutdown analysis.

5.2. Changes in Nuclear Design

Eight standard control rods will be replaced by extended life control rods. The new control rods consist of the same poison material, but have a smaller poison diameter and a longer poison length. The cladding material of the standard rods is stainless steel 304. Inconel 625 is used to clad the extended life control rods. Although the resulting change in rod worth is small, these changes are

incorporated into the analysis.

As stated in section 5.1, the NEMO code was used to calculate the physics parameters for cycle 9. NEMO is a two group neutronics program employing the nodal expansion method to determine the currents and fluxes at the surface of each node. This nodal expansion method is incorporated in the finite difference method used to solve the multidimensional neutronics problem. These methods treat the spatial dependence of the cross-sections and the flux within a node explicitly. Reference 8 illustrates the calculational accuracy attainable with NEMO in comparison to measured results for various physics parameters. NOODLE results may be found in reference 7. Comparisons of these results show NEMO to be more accurate than NOODLE.

No significant operational or procedural changes exist with regard to axial or radial power shape, xenon, or tilt control. The stability and control of the core with APSRs withdrawn has been analyzed. The calculated stability index without APSRs is $-0.0071h^{-1}$, which demonstrates the axial stability of the core. The operating limits (COLR changes) for the reload cycle are given in section 8.

Table 5-1. Davis-Besse Unit 1, Cycle 9 Physics Parameters

	Cycle 8	Cycle 9[a]
Cycle length, EFPD	479	500[b]
Cycle burnup, MWd/mtU	16,020	16,719
Average core burnup - EOC, MWd/mtU	29,568	31,910
Initial core loading, mtU	82.9	82.9
Critical boron[c] - BOC, No Xe, ppm		
HZP	1,737	1,840
HFP	1,515	1,646
Critical boron[c] - EOC, Eq. Xe, ppm		
HZP	207	239
HFP	10[d]	5[d]
Control rod worths - HFP, BOC, %Δk/k		
Group 6	1.02	1.06
Group 7	1.02	1.06
Group 8	0.15	0.15
Control rod worths - HFP, EOC, %Δk/k		
Group 7	1.13	1.14
Group 8	NA	NA
Max ejected rod worth - HZP, %Δk/k		
BOC, Groups 5-8 inserted (N-12)	0.30	0.55
EOC, Groups 5-7 inserted (L-10,N-12)	0.35	0.49
Max stuck rod worth - HZP, %Δk/k		
BOC (N-12)	0.77	0.72
EOC (M-11,N-12)	0.68	0.66
Power deficit[e] - HZP to HFP, Eq. Xe, %Δk/k		
BOC (4 EFPD)	-1.74	-1.80
EOC	-2.71	-3.37
Doppler coeff - HFP, 10^{-3} %Δk/k/°F		
BOC, No Xe[f] Group 8 inserted	-1.59	-1.59
EOC, Eq. Xe, 0 ppm, Group 8 withdrawn	-1.96	-1.90
Moderator coeff - HFP, 10^{-2} %Δk/k/°F		
BOC, No Xe[f]	-0.66	-0.63
EOC, Eq. Xe, 0 ppm[g]	-3.33	-3.43
Temperature coeff - HZP, 10^{-2} %Δk/k/°F		
EOC, Eq. Xe, Grps 1-7 In, N12 Out, 0 ppm	-2.67	-2.61

Table 5-1. Davis-Besse Unit 1, Cycle 9 Physics Parameters

	Cycle 8	Cycle 9
Boron worth - HFP, ppm/%Δk/k		
BOC [(f)]	138	142
EOC	114	115
Xenon worth - HFP, %Δk/k		
BOC (4 EFPD)	2.63	2.62
EOC (equilibrium)	2.77	2.78
Effective delayed neutron fraction - HFP		
BOC	0.00623	0.00632
EOC	0.00514	0.00526

(a) Based on cycle 7 length of 405.2 EFPD (actual) and cycle 8 length of 453 EFPD.

(b) All end-of-cycle (EOC) values calculated at 500 EFPD; the design cycle 9 length is 490 EFPD.

(c) Control rod group 8 is inserted at BOC and withdrawn at EOC.

(d) Power coastdown to EOC at 10 ppm for cycle 8 and 5 ppm for cycle 9.

(e) Cycle 9 deficits are three-dimensional, while cycle 8 are two-dimensional.

(f) Cycle 9 values were calculated at 1666 ppm; cycle 8 values were calculated at 1549 ppm.

(g) These values were calculated with the control rods at the insertion limit.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 9

	BOC, %Δk/k	EOC, %Δk/k 435 EFPD Group 8 in	EOC, %Δk/k 500 EFPD Group 8 out
Available Rod Worth			
Total rod worth, HZP	6.56	6.82	6.85
Worth reduction due to burnup of poison material	-0.19	-0.19	-0.19
Maximum stuck rod worth, HZP	-0.72	-0.61	-0.66
Net Worth	5.65	6.02	6.00
Less 10% Uncertainty	-0.57	-0.60	-0.60
Total available worth	5.08	5.42	5.40
Required Rod Worth			
Power deficit, HFP to HZP	1.80	3.17	3.37
Xenon transient allowance	0.30	0.30	0.30
Max allowable inserted rod worth	0.28	0.48	0.53
Total required worth	2.38	3.95	4.20
Shutdown Margin			
Total available minus total required	2.70	1.47	1.20

Note: Required shutdown margin is 1.00% Δk/k.

Figure 5-1. Davis-Besse Cycle 9 Relative Power Distribution at BOC (4 EFPD), Full Power, Equilibrium Xenon, Gp 7 90.1 %WD, Gp 8 30.4 %WD

	8	9	10	11	12	13	14	15
H	0.824	1.043	1.152	1.298	7 1.066	1.378	1.065	0.437
K	1.043	0.913	1.251	1.069	1.325	1.275	1.255	0.485
L	1.152	1.249	1.082	1.283	8 1.201	1.279	0.827	0.255
M	1.298	1.064	1.279	1.077	1.345	1.092	0.569	
N	7 1.066	1.316	8 1.200	1.348	7 1.234	1.124	0.307	
O	1.378	1.271	1.278	1.093	1.129	0.421		
P	1.065	1.256	0.828	0.571	0.309			
R	0.437	0.489	0.256					

x	Inserted Rod Group Number
x.xxx	Relative Power Density

6.0 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 9 operation utilized the methods and models described in references 9, 10 and 11 as supplemented by reference 12, which implemented the BWC (Reference 13) CHF correlation for analysis of Zircaloy-grid fuel assemblies. The incoming batch 11 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. Introduction of the batch 9B assembly with a stainless steel rod, the two-cycle burnable poison rod assemblies (BPRAs) and the batch 11 fuel consisting of Mark-B9A fuel rods in Mark-B8A fuel assemblies was determined to have an insignificant impact on fuel rod DNB performance.

With the implementation of the batch 11 Mark-B8B fuel, the cycle 9 core is a full Zircaloy-grid core, except for the one batch 8C Mark-B5A assembly. The batch 11 Mark-B8B fuel contains Mark-B9A fuel rods which have a 140.6 inch stack height. The cycle 9 core contains 56 BPRAs and 60 open fuel assemblies with unplugged control rod guide tubes. The core bypass flow, which is dependent on the number of open control rod guide tubes, is calculated as 8.4% for this configuration. The cycle-specific reference analysis based on the actual cycle 9 core configuration, the 140.6 inch stack height and the 8.4% core bypass flow value, demonstrated that the reference analysis used for cycles 7 and 8 remains bounding for cycle 9 operation. The batch 9B reconstituted fuel assembly contains one stainless steel replacement rod that is surrounded within the fuel rod array by heated fuel rods. The BWC CHF correlation and associated licensing methodologies are approved for this geometry. Calculations show there is no DNB penalty associated with the placement of the stainless steel rod in this configuration. The reconstituted fuel assembly has over 100% DNBR margin relative to the limiting fuel bundle. Therefore, adequate DNBR margin is available to justify operation of the core with the reconstituted fuel assembly. Table 6-1 provides a summary comparison of the DNB analysis parameters for cycles 8 and 9.

Table 6-1. Maximum Design Conditions, Cycles 8 and 9

	Cycle 8	Cycle 9
Design power level, MWt	2772	2772
Nominal core exit pressure, psia	2200	2200
Minimum core exit pressure, psia	2135	2135
Reactor coolant flow, gpm	380,000	380,000
Core bypass flow, %	8.9 [a]	8.4 [b]
DNBR modeling	Crossflow	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.65 chopped cosine	1.65 chopped cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.97	0.97
Active fuel length, in.	143.2	140.6
Avg heat flux at 100% power, 10^5 Btu/h-ft^2	1.86	1.89
Max heat flux at 100% power, 10^5 Btu/h-ft^2	5.25	5.35
CHF correlation	BWC	BWC
CHF correlation DNB limit	1.18	1.18
Minimum DNBR[c]		
at 102% power	1.78	1.78
at 112% power	1.54	1.55

[a] Used in the analysis.

[b] Calculated for the actual cycle 9 core configuration.

[c] Calculated for the instrument guide tube subchannel which is limiting for the Mark-B8A and Mark-B8B fuel assemblies.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each USAR accident analysis has been examined with respect to changes in the cycle 9 parameters to determine the effects of the cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the USAR accident results have been evaluated and are reported in reference 14.

The radiological dose consequences of the USAR Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle-specific source term for Davis-Besse 1 cycle 9. The dose calculations were performed consistent with the assumptions described in the Davis-Besse 1 USAR, but used the more conservative source terms (which bound future reload cycles). The results of the dose evaluations showed that offsite radiological doses for each accident were below the respective acceptance criteria values in the current NRC Standard Review Plan (NUREG-0800).

The effects of inadvertent loading of a fuel assembly into an improper position have been evaluated. This type of misplacement would be detected with the incore detectors during startup tests.

7.2 Accident Evaluation

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: (1) core thermal, (2) thermal-hydraulic, and (3) kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters from each batch in cycle 9 are given in Table 4-1. The cycle 9 thermal-hydraulic maximum design conditions are presented in Table 6-1. A comparison of the key kinetics parameters from the USAR and cycle 9 is provided in Table 7-1.

The EOC moderator temperature coefficient listed in Table 7-1 for cycle 9 is the 3-D, hot full power (HFP) coefficient. An evaluation was performed to verify the acceptability of the more negative cycle 9 moderator temperature coefficient for all USAR accidents excluding steam line breaks. The results of the evaluation were acceptable for all USAR accidents, excluding steam line breaks, for a moderator temperature coefficient as negative as -4.0×10^{-2} %Δk/k/°F.

The steam line break accident was evaluated based on a combined moderator and Doppler temperature coefficient from 532°F to the minimum temperature reached during the event. The combined temperature coefficient used in safety analysis of the steam line break is the sum of the EOC moderator and Doppler coefficients (-3.10×10^{-2} %Δk/k/°F). The combined temperature coefficient for EOC cycle 9 is shown in section 5 as -2.61×10^{-2} %Δk/k/°F. Since the safety analysis value for the combined temperature coefficient is more negative than the cycle 9 value, the steam line break analysis remains bounding for cycle 9.

A generic loss-of-coolant accident (LOCA) analysis has been performed for the 177-FA raised loop nuclear supply system (NSS) using the Final Acceptance Criteria B&W ECCS evaluation model techniques and assumptions, as described in BAW-10104P, Rev. 5[15], updated with upgraded fuel performance models and the B&W modified version of FLECSET (reported in BAW-1915PA[16]). In addition, the BWC CHF correlation was used to determine the time of DNB. The combination of average fuel temperatures as functions of linear heat rate and lifetime pin pressure data used in the generic LOCA LHR limits analysis are conservative compared to those calculated for the Mark-B5A and Mark-B8A fuel in this reload. The average fuel temperatures as functions of linear heat rate and lifetime pin pressure data for the Mark-B8B fuel were calculated using the TACO3 fuel pin performance code. The Mark-B8B fuel data are bounded by the Mark-B8A fuel data. The Mark-B8B LOCA limits analysis verified a value of 18.3 kW/ft at the 10-foot elevation for application between 0 and 1000 MWd/mtU, which is consistent with the Technical Specification F_Q limit of 2.93. The use of the TACO3 fuel pin performance code requires a reduction in the 8-foot linear heat rate to 16.75 kW/ft. At the end of the cycle, the limits for the Mark-B8B fuel must be reduced in order to maintain the internal fuel pin pressure below the TACO3 SER restriction of 2200 psia. A tabulation showing the maximum allowable LOCA linear heat rate limits for Davis-Besse Unit 1, cycle 9 fuel is provided in Table 7-2. It is concluded

by the examination of cycle 9 core thermal, thermal-hydraulic, and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the ability to safely operate the Davis-Besse Unit 1 plant during cycle 9. Considering the previously accepted design basis used in the USAR and subsequent cycles, the transient evaluation of cycle 9 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 9 are bounded by the USAR, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of Key Parameters for Accident Analysis

Parameter	USAR and densif'n report value	Cycle 9 value
BOC[a] Doppler coeff, 10^{-3}, %Δk/k/°F	-1.28	-1.59
EOC[b] Doppler coeff, 10^{-3}, %Δk/k/°F	-1.45[c]	-1.90
BOC moderator coeff, 10^{-2}, %Δk/k/°F	+0.13	-0.63
EOC moderator coeff, 10^{-2}, %Δk/k/°F	-3.0	-3.43[d]
EOC temperature coeff (532 to 510F) 10^{-2}, %Δk/k/°F	-3.10	-2.61
All rod bank worth (HZP), %Δk/k	10.0	6.56
Boron reactivity worth (HFP), ppm/%Δk/k	100	142
Max ejected rod worth (HFP), %Δk/k	0.65	0.26
Max dropped rod worth (HFP), %Δk/k	0.65	≤0.20
Initial boron conc (HFP), ppm	1407	1646

(a) BOC denotes beginning of cycle.
(b) EOC denotes end of cycle.
(c) -1.77×10^{-3} %Δk/k/°F was used for steam line failure analysis.
(d) Moderator coefficient is bounded by generic plant analyses value of -4.00×10^{-2} %Δk/k/°F at HFP.

Table 7-2. Bounding Values for Allowable LOCA Peak Linear F . Rates

Mark-B5A and Mark-B8A Fuel Types

Allowable Peak LHR for Specified Burnup Interval, kW/ft

Core elevation, ft	0-40,000 MWd/mtU kW/ft	after 40,000 MWd/mtU kW/ft
2	16.0	16.0
4	15.75	15.75
6	16.5	18.0
8	17.25	17.25
10	17.0	17.0

Mark-B8B Fuel Type

Allowable Peak LHR for Specified Burnup, kW/ft*

Core Elevation ft	0 MWd/mtU	1000 MWd/mtU	33333 MWd/mtU	34167 MWd/mtU	35000 MWd/mtU	36667 MWd/mtU	37500 MWd/mtU
2	16.0	16.0	16.0	16.0	16.0	16.0	15.75
4	15.75	15.75	15.75	15.75	15.75	15.75	15.75
6	16.5	16.5	16.5	16.5	16.5	16.0	15.75
8	16.75	16.75	16.75	16.75	16.5	16.0	15.75
10	18.3	17.0**	17.0	16.75	16.5	16.0	15.75

* Linear interpolation for Allowable Peak LHR between Specified Burnup points is valid.

** Allowable Peak LHR is reduced from 18.3 kW/ft to 17.0 kW/ft at 1000 MWd/mtU.

8. PROPOSED MODIFICATIONS TO CORE OPERATING LIMITS REPORT

The Core Operating Limits Report (COLR) has been revised for cycle 9 operation to accommodate the influence of the cycle 9 core design on power peaking, reactivity, and control rod worths. Revisions to the cycle-specific parameters were made in accordance with the requirements of NRC Generic Letter 88-16 and Technical Specification 6.9.1.7. The core operating limits were determined from a cycle 9 specific power distribution analysis using NRC approved methodology provided in the reference to Technical Specification 6.9.1.7.

The core operating limits are based on ECCS bounding analyses that were performed to determine the allowable LOCA linear heat rate limits for the B&W 177 fuel assembly raised-loop plant. The analysis for the Mark-B5A and Mark-B8A fuel types incorporated the NUREG-0630 cladding swell and rupture model, TACO2 fuel performance code, the BWC CHF correlation, the B&W modified version of FLECSET reflooding heat transfer coefficient correlation, and the Mark-BZ fuel design. The analysis for the Mark-B8B fuel type incorporated the NUREG-0630 cladding swell and rupture model, TACO3 fuel performance code, the BWC CHF correlation, the B&W modified version of FLECSET reflooding heat transfer coefficient correlation, and the Mark-B9A fuel rod design. Figures 8-1 through 8-20 are revisions to the fuel cycle operating limits contained in the COLR. Table 8-1 presents the quadrant power tilt limits for cycle 9 and Table 8-2 provides the negative moderator temperature coefficient for cycle 9. Based on the analyses and operating limit revisions described in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated.

Figure 8-1

Regulating Group Position Limits, 0 to 75±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 9

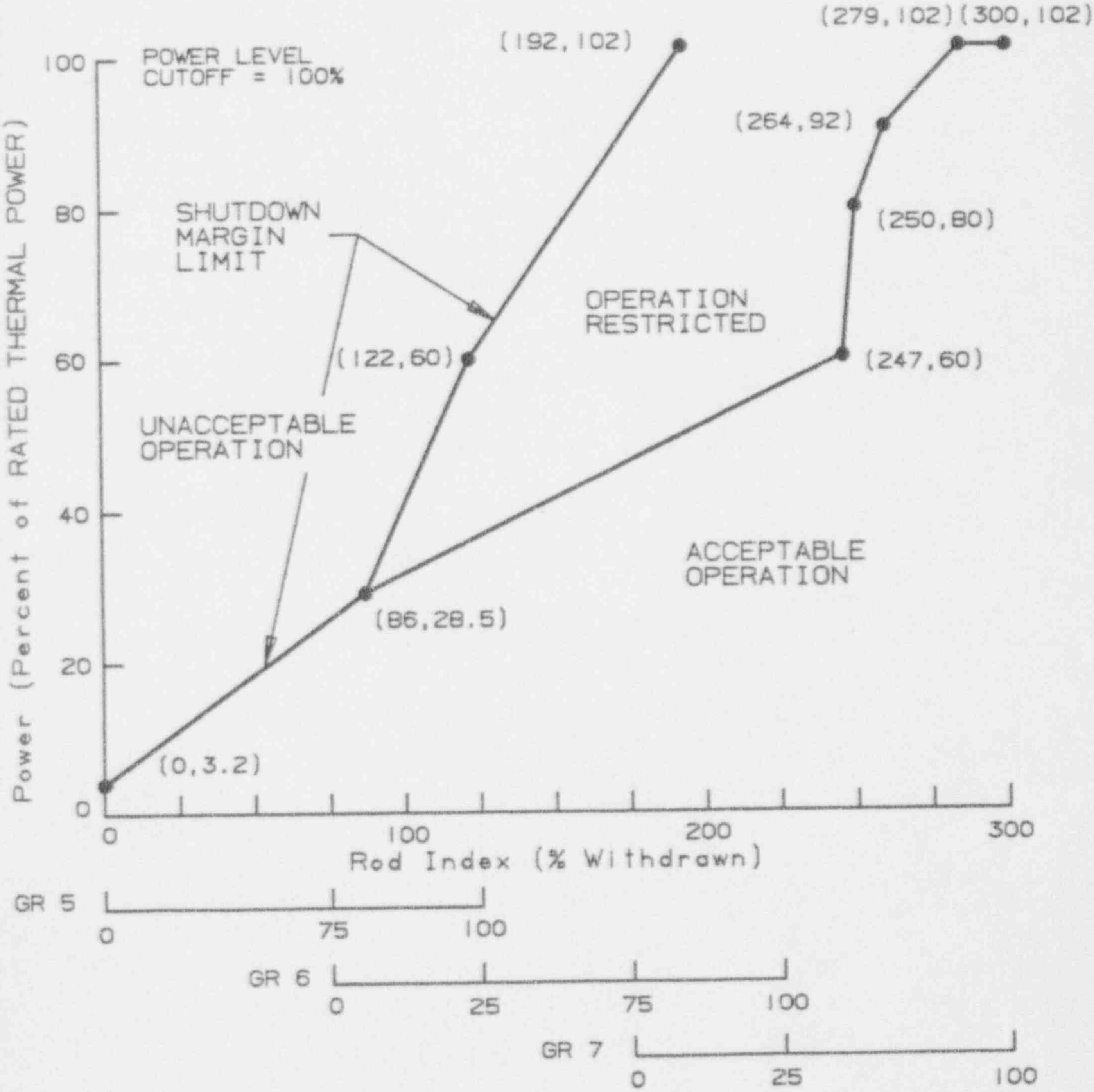


Figure 8-2

Regulating Group Position Limits, 75±10 to 300±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 9

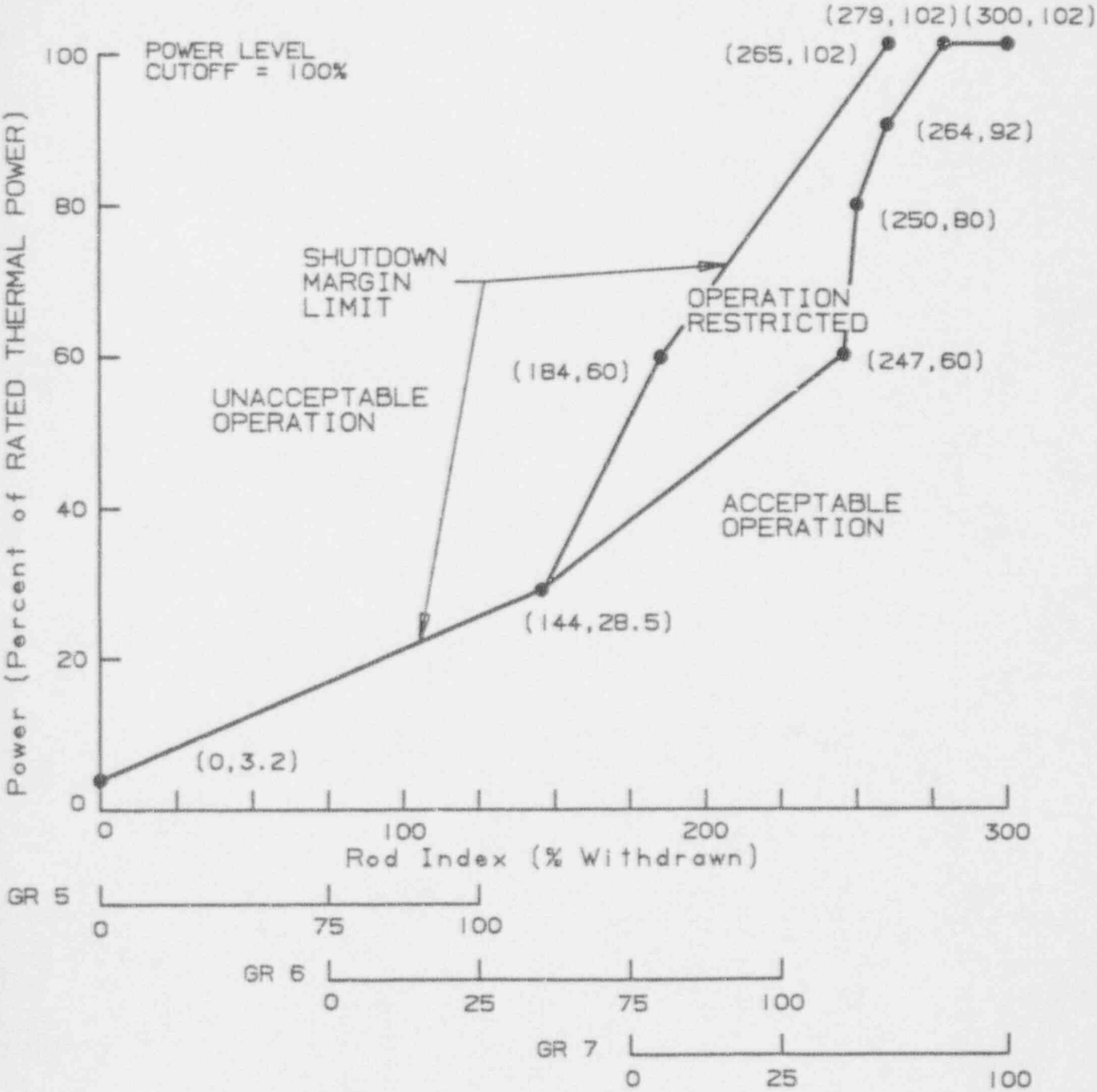


Figure 8-3

Regulating Group Position Limits, 300±10 to 425±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 9

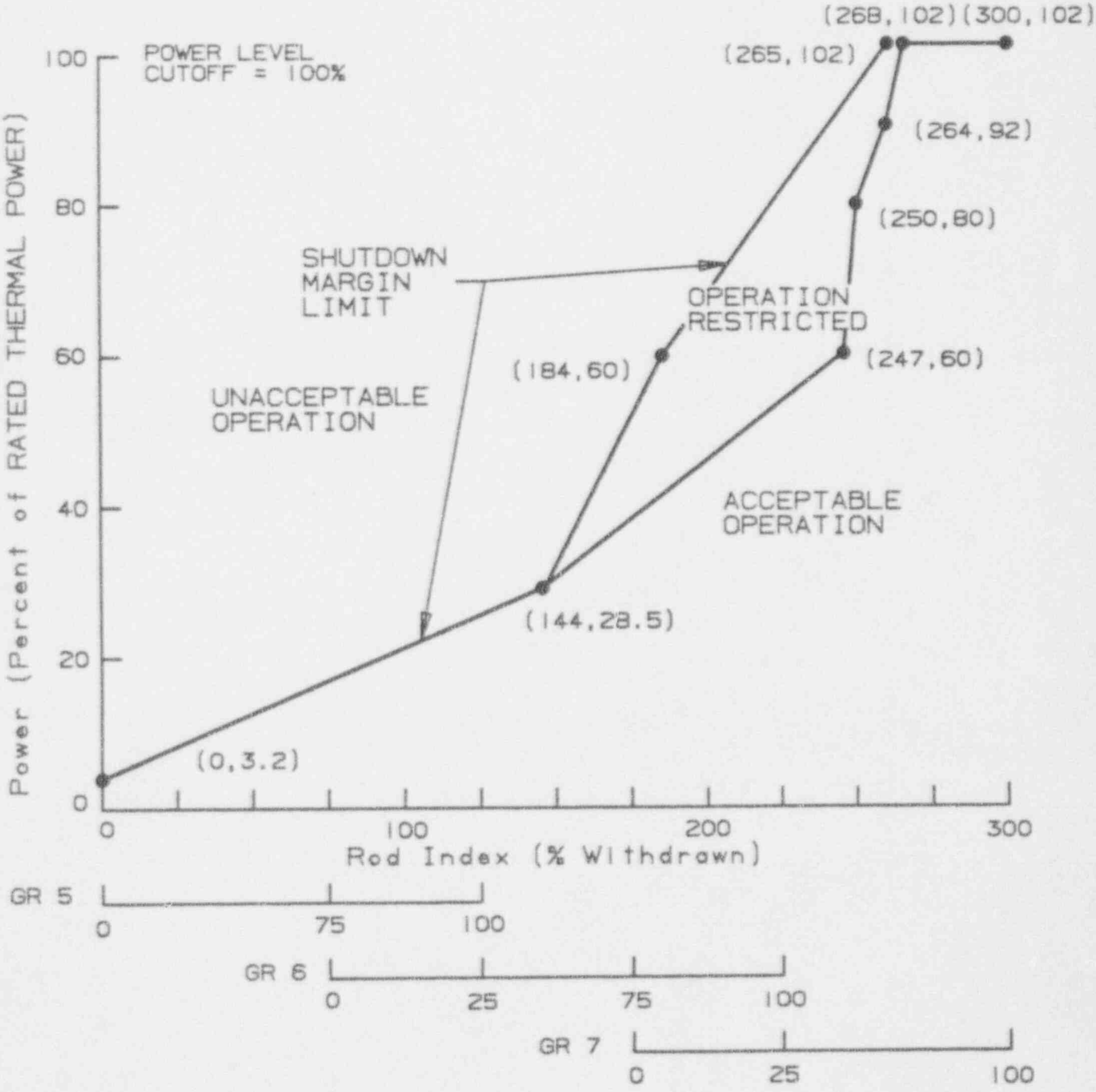


Figure 8-4

Regulating Group Position Limits, After 425±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 9

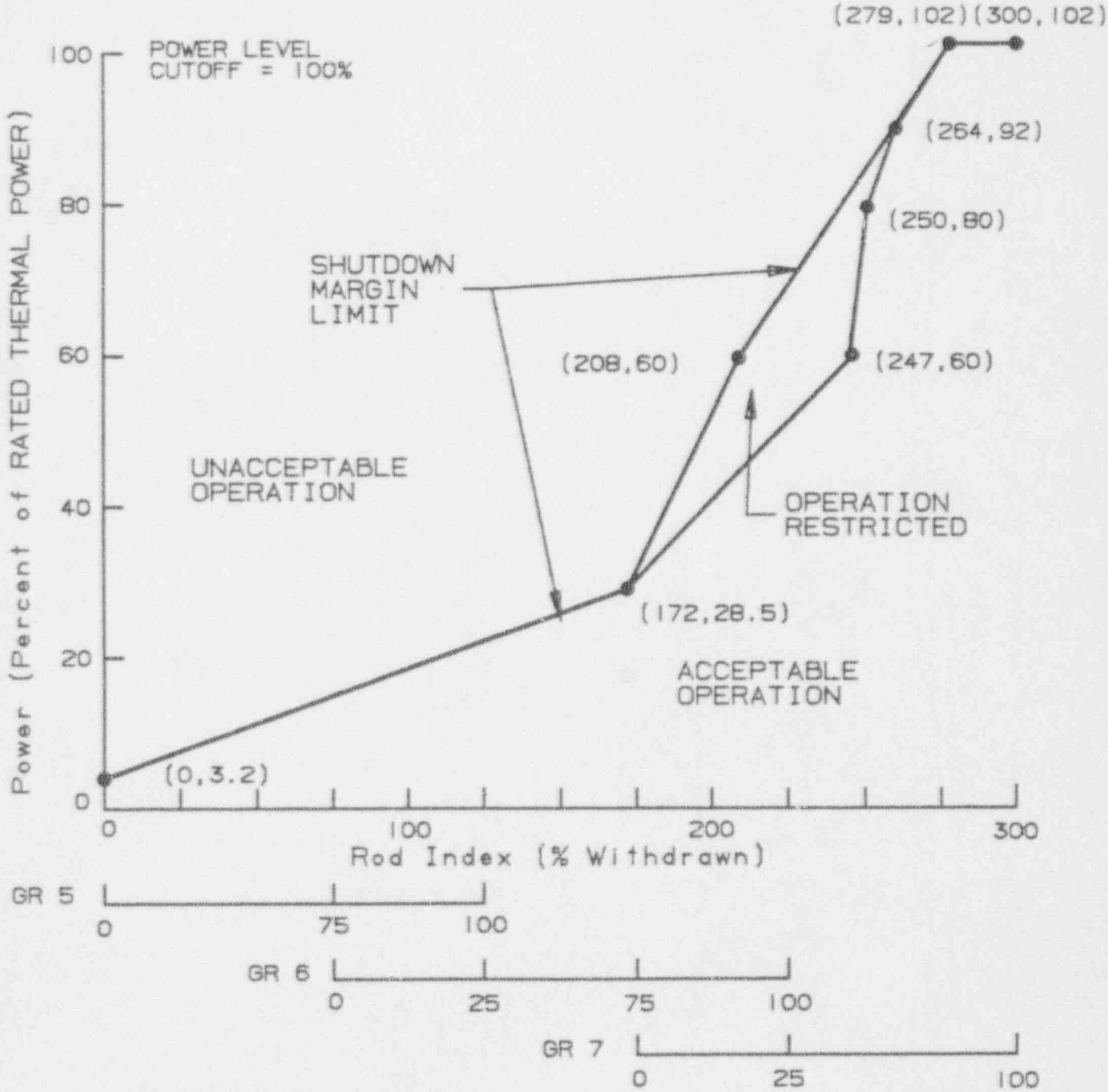


Figure 8-5

Regulating Group Position Limits, 0 to 75±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 9

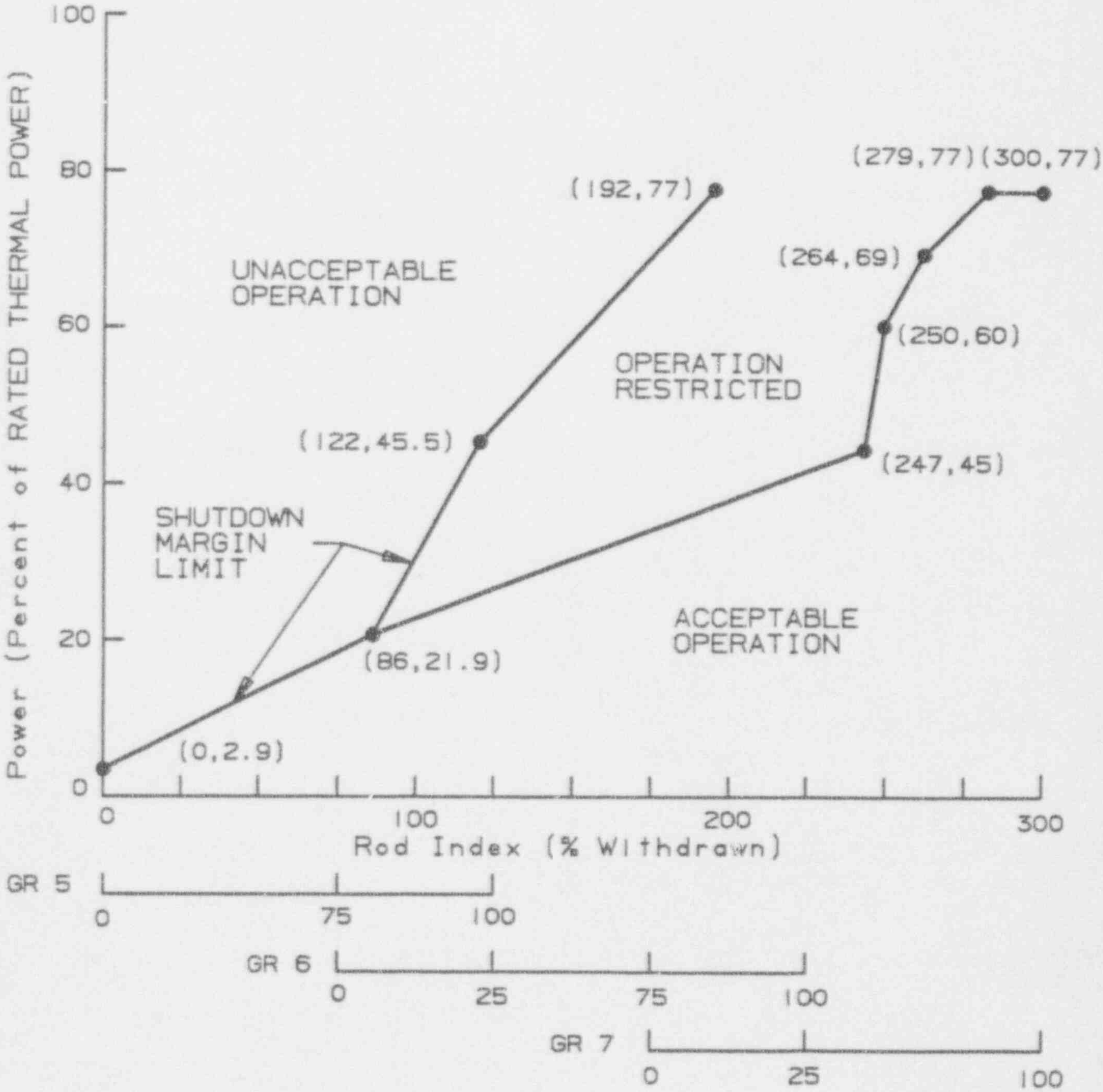


Figure 8-6

Regulating Group Position Limits, 75±10 to 300±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 9

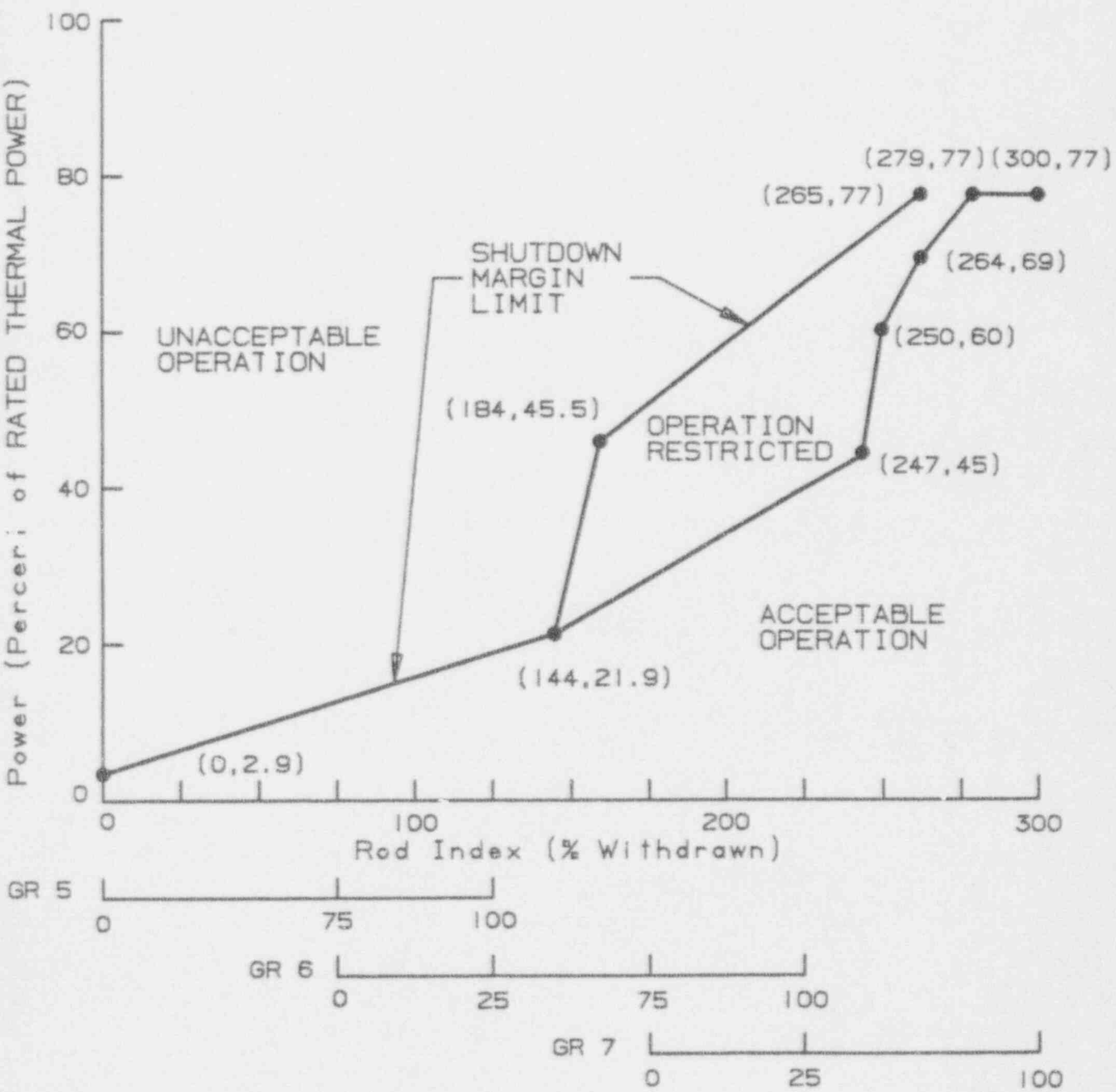


Figure 8-7

Regulating Group Position Limits, 300±10 to 425±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 9

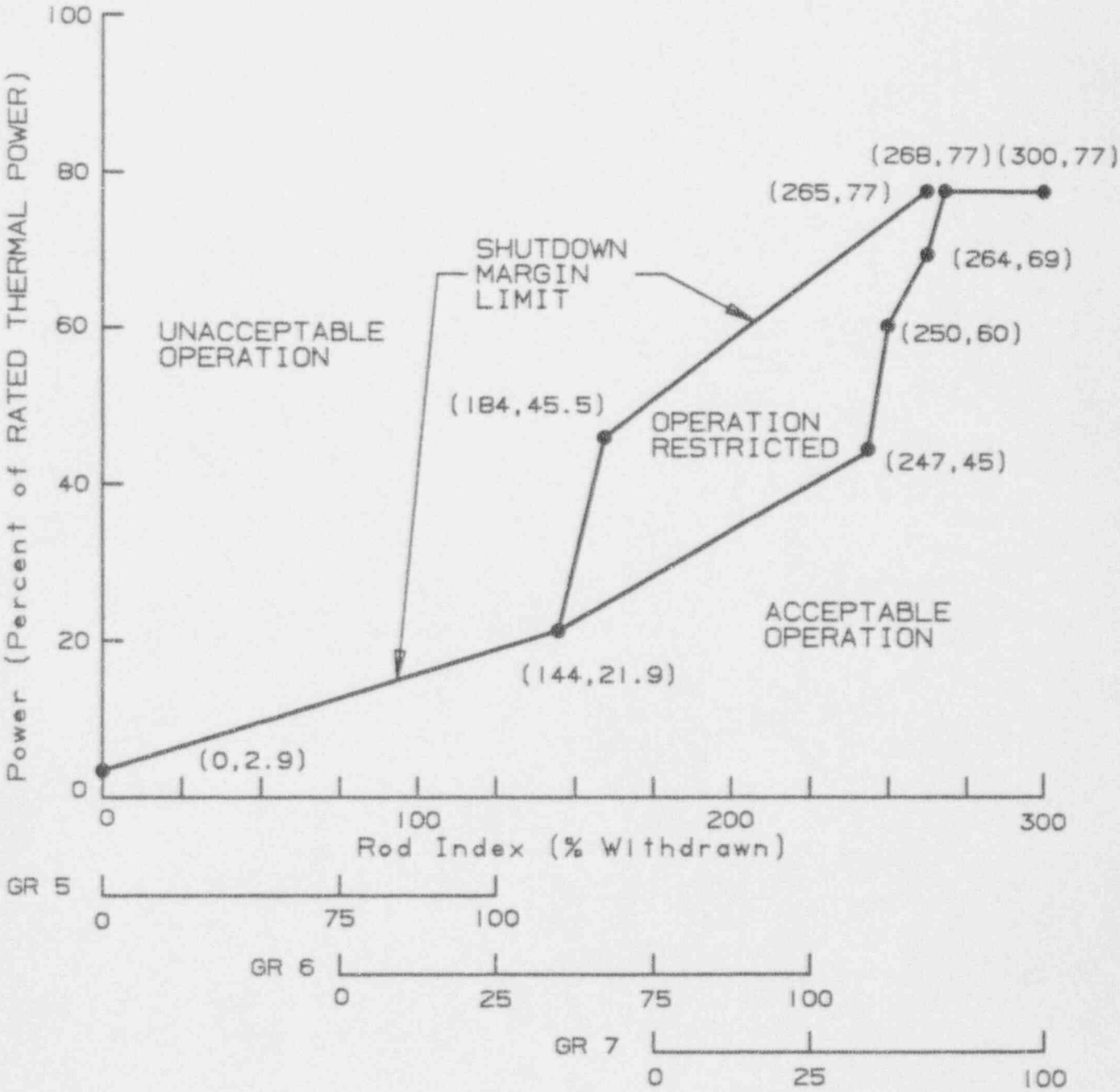


Figure 8-8

Regulating Group Position Limits, After 425±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 9

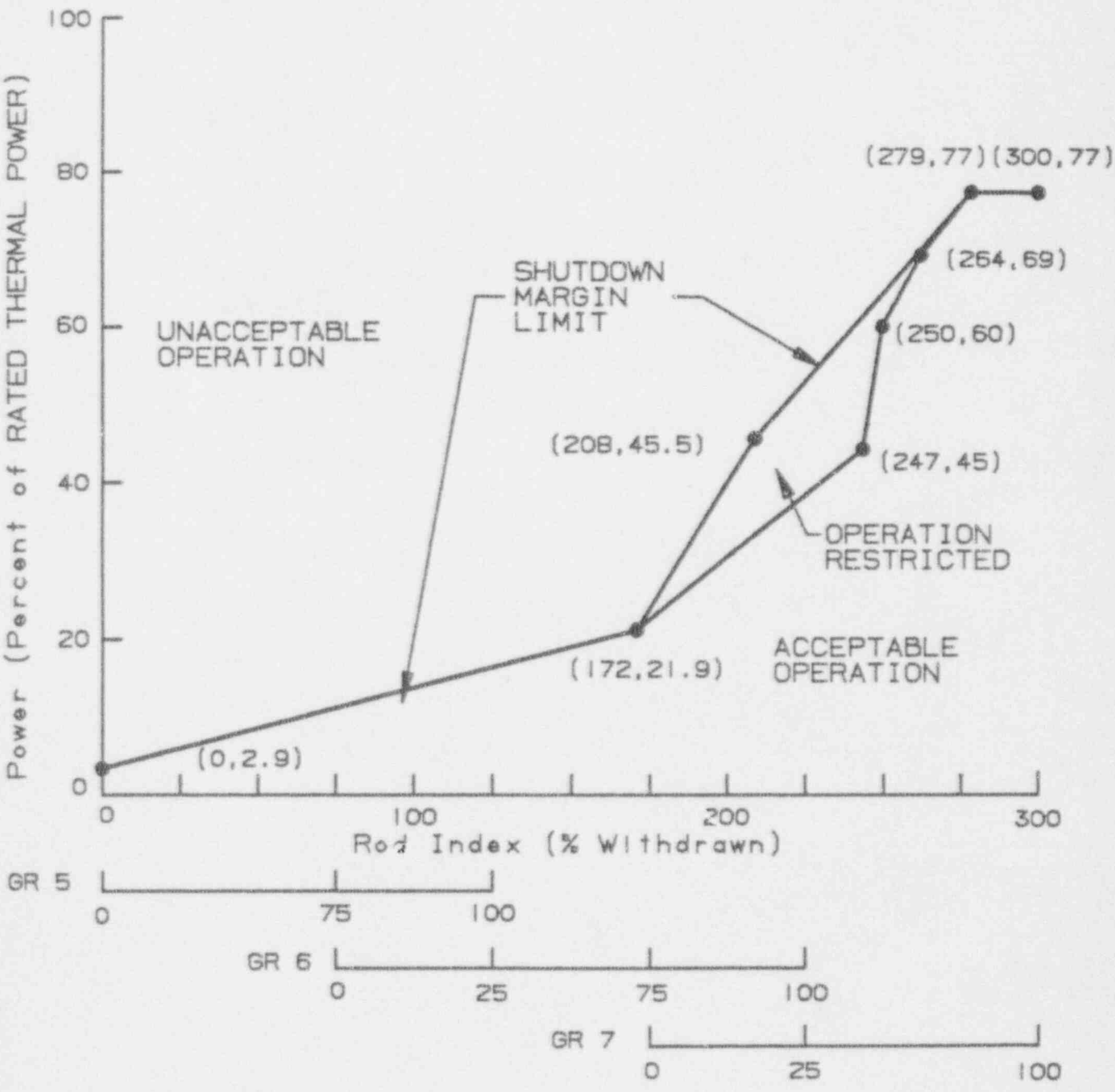
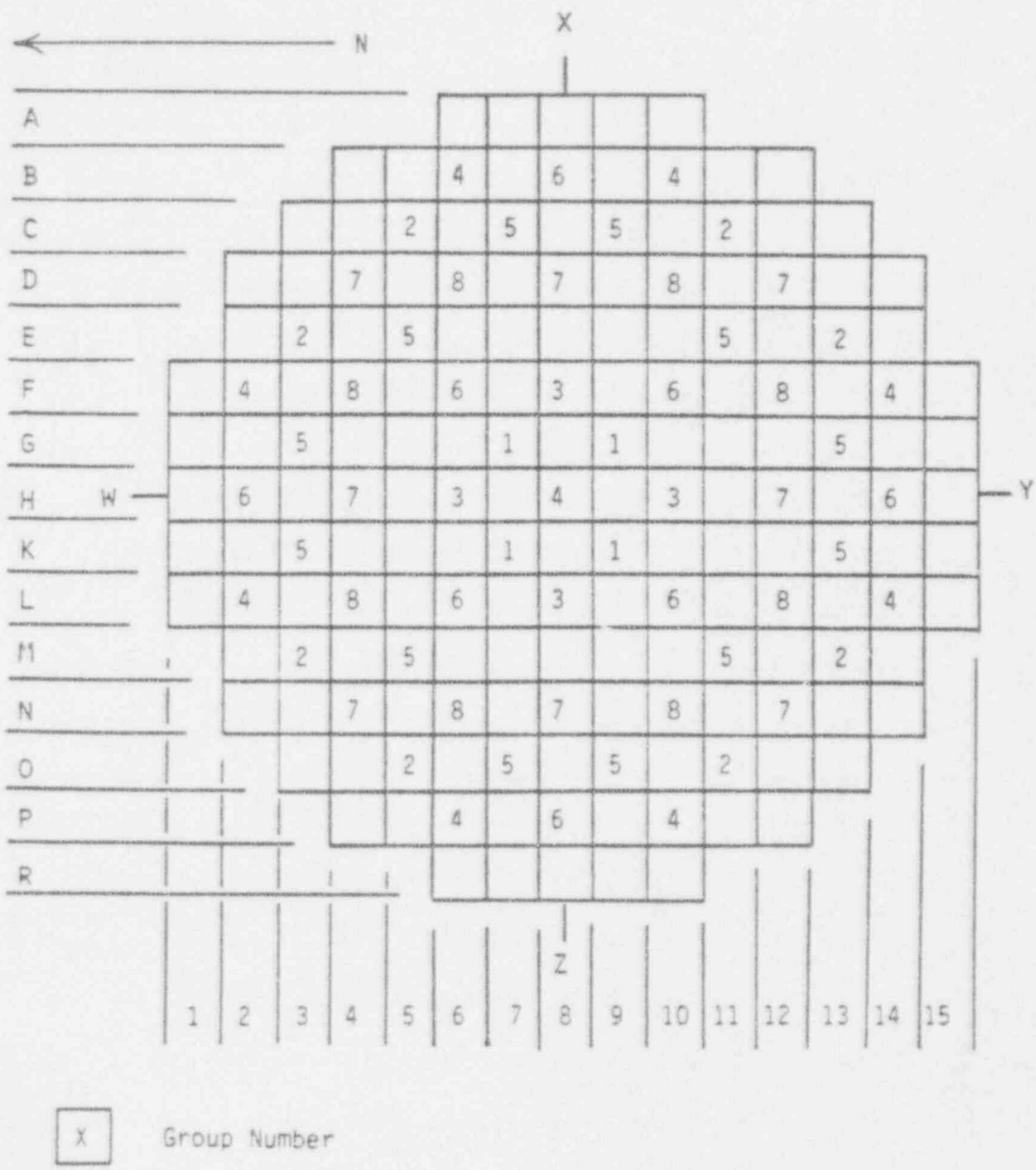


Figure 8-9. Control Rod Locations for Davis-Besse 1, Cycle 9



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	

Figure 8-10

APSR Position Limits, 0 to 425±10 EFPD,
Four RC Pumps -- Davis-Besse 1, Cycle 9

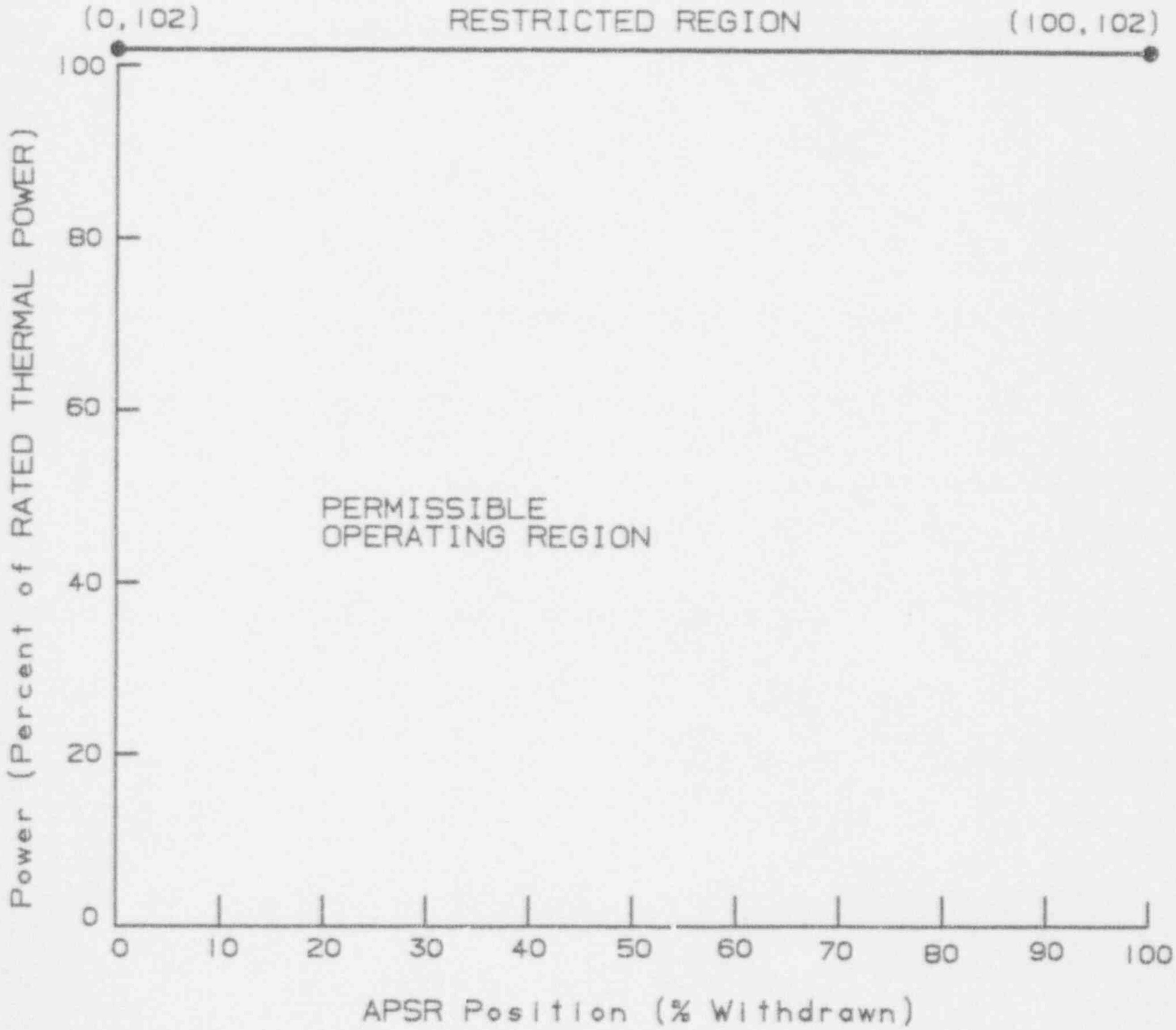


Figure 8-11

APSR Position Limits, After 425±10 EFPD,
Three or Four RC Pumps, APSRs Withdrawn
Davis-Besse 1, Cycle 9

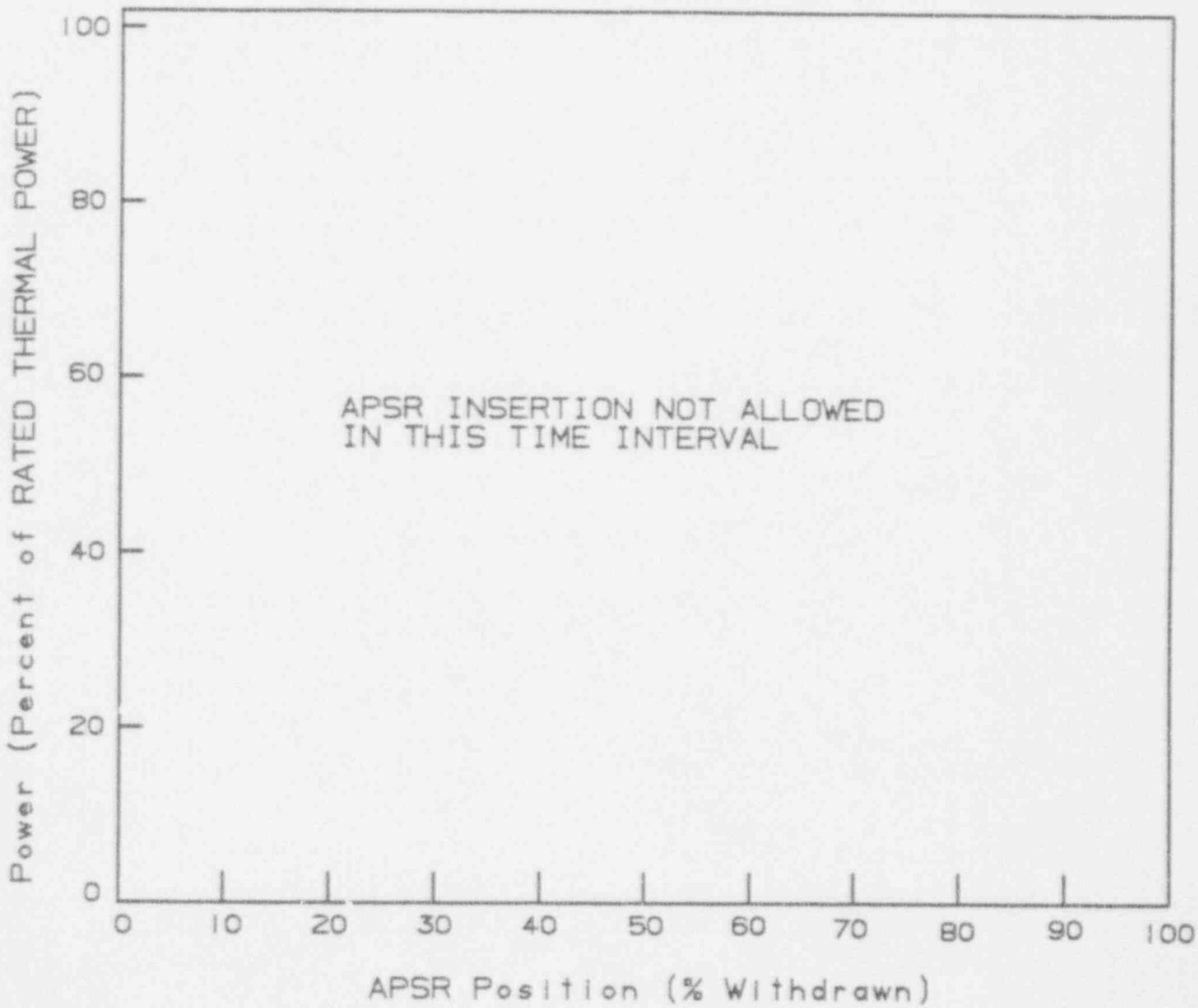


Figure 8-12

APSR Position Limits, 0 to 425±10 EFPD, Three RC Pumps-- Davis-Besse 1, Cycle 9

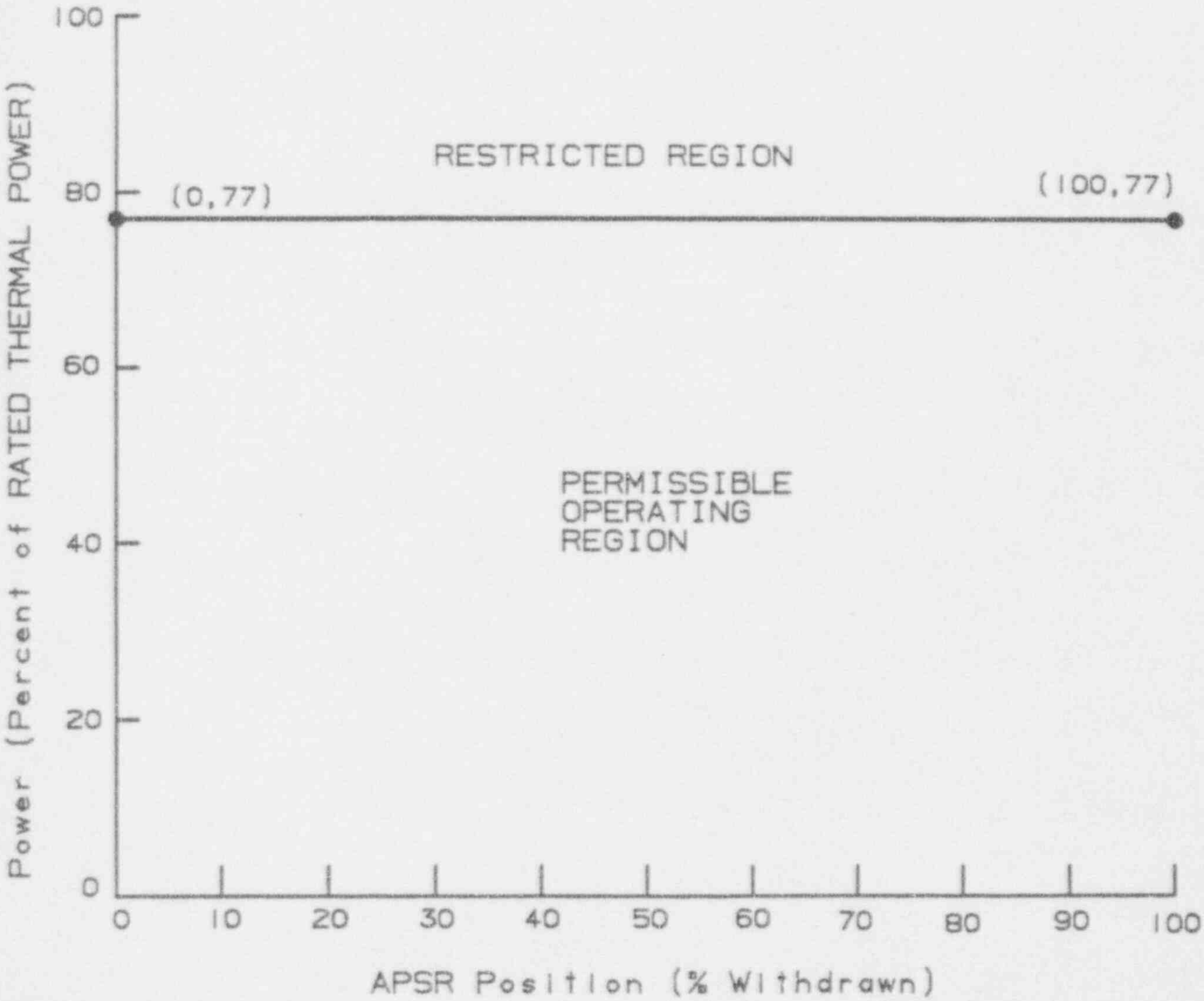


Figure 8-13

AXIAL POWER IMBALANCE Limits, 0 to 75±10 EFPD, Four RC Pumps-- Davis-Besse 1, Cycle 9

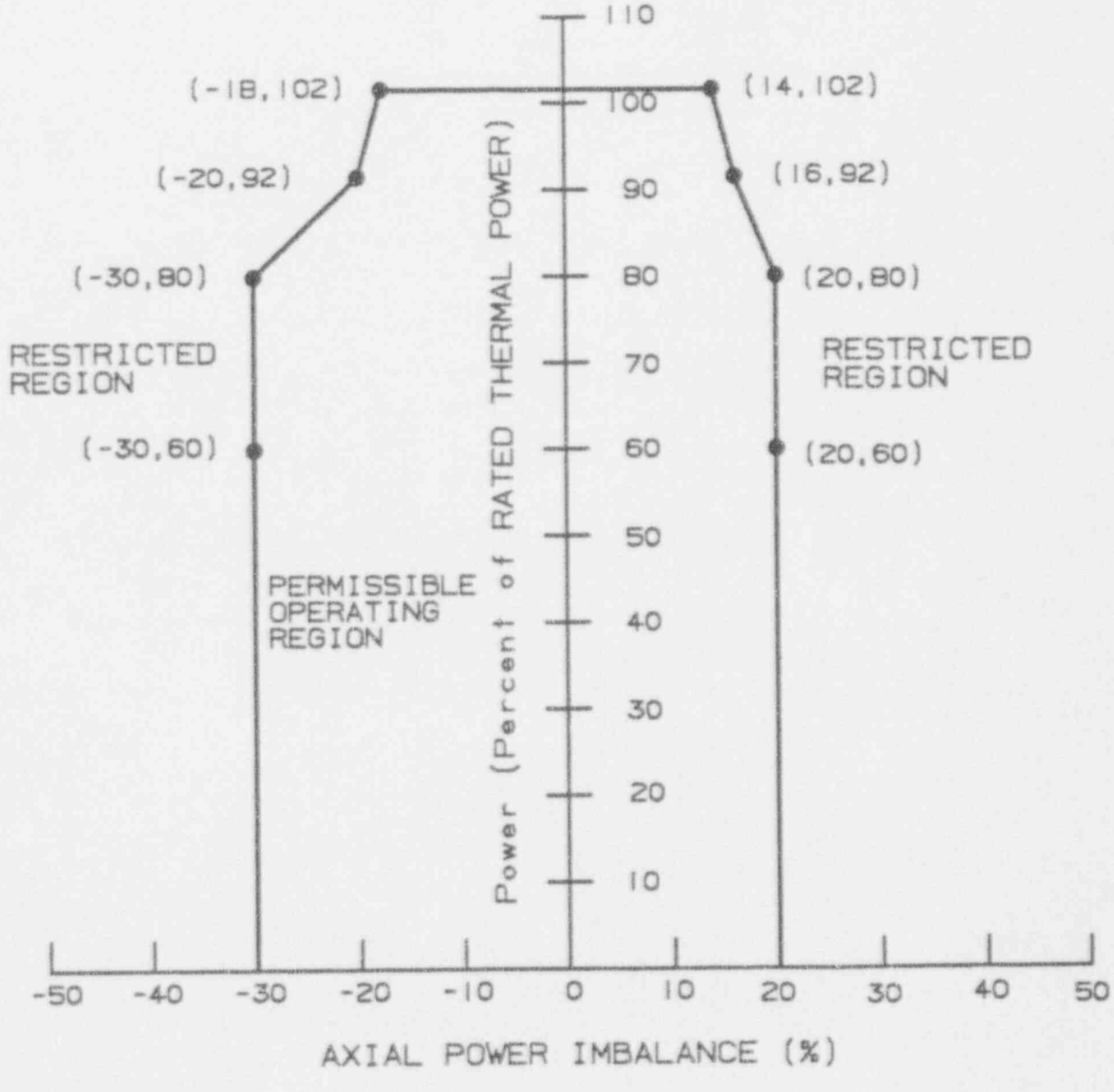


Figure 8-14

AXIAL POWER IMBALANCE Limits, 75±10 TO 300±10 EFPD, Four RC Pumps-- Davis-Besse 1, Cycle 9

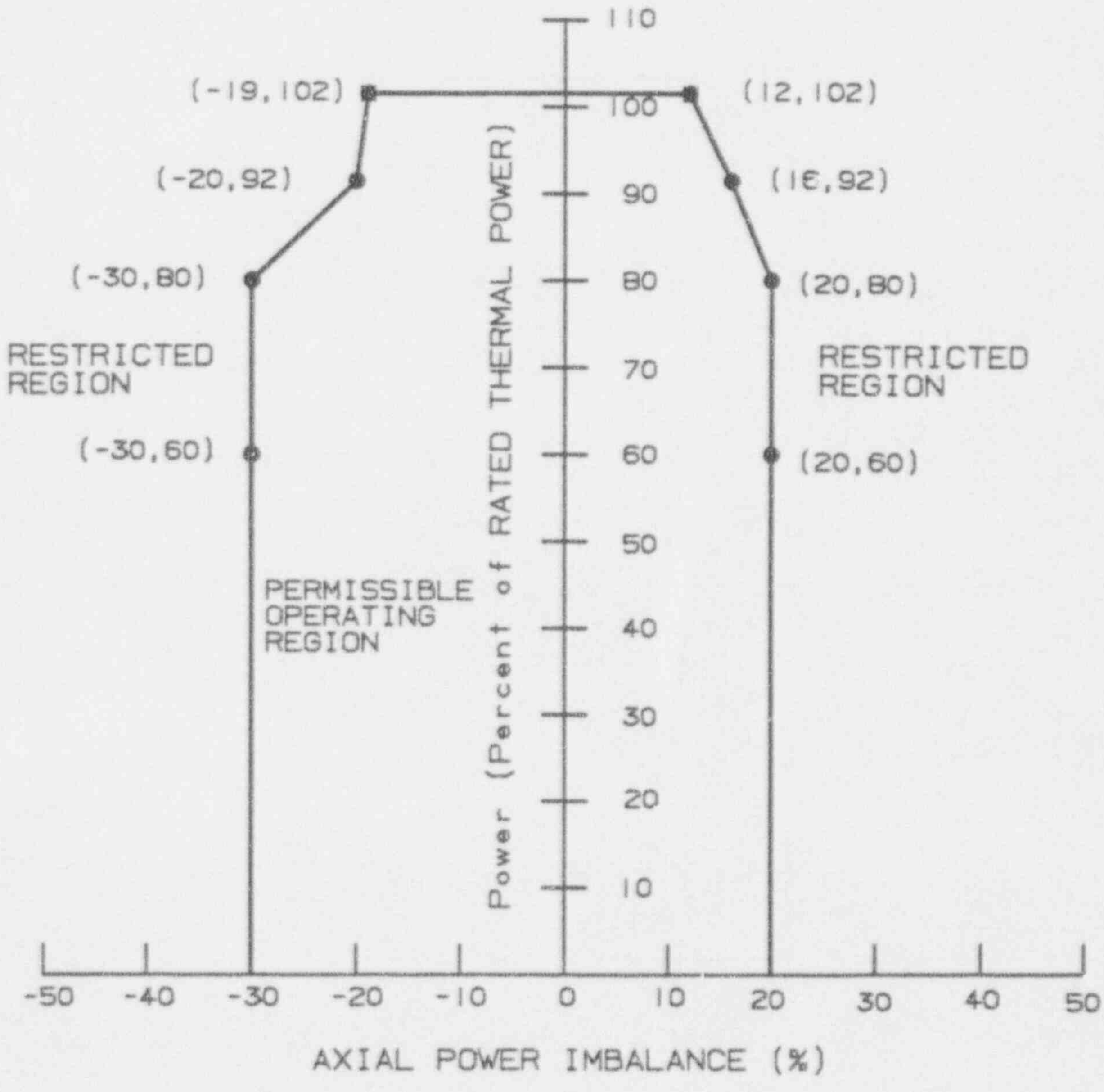


Figure 8-15

AXIAL POWER IMBALANCE Limits, 300±10 to 425±10 EFPD, Four RC Pumps-- Davis-Besse 1, Cycle 9

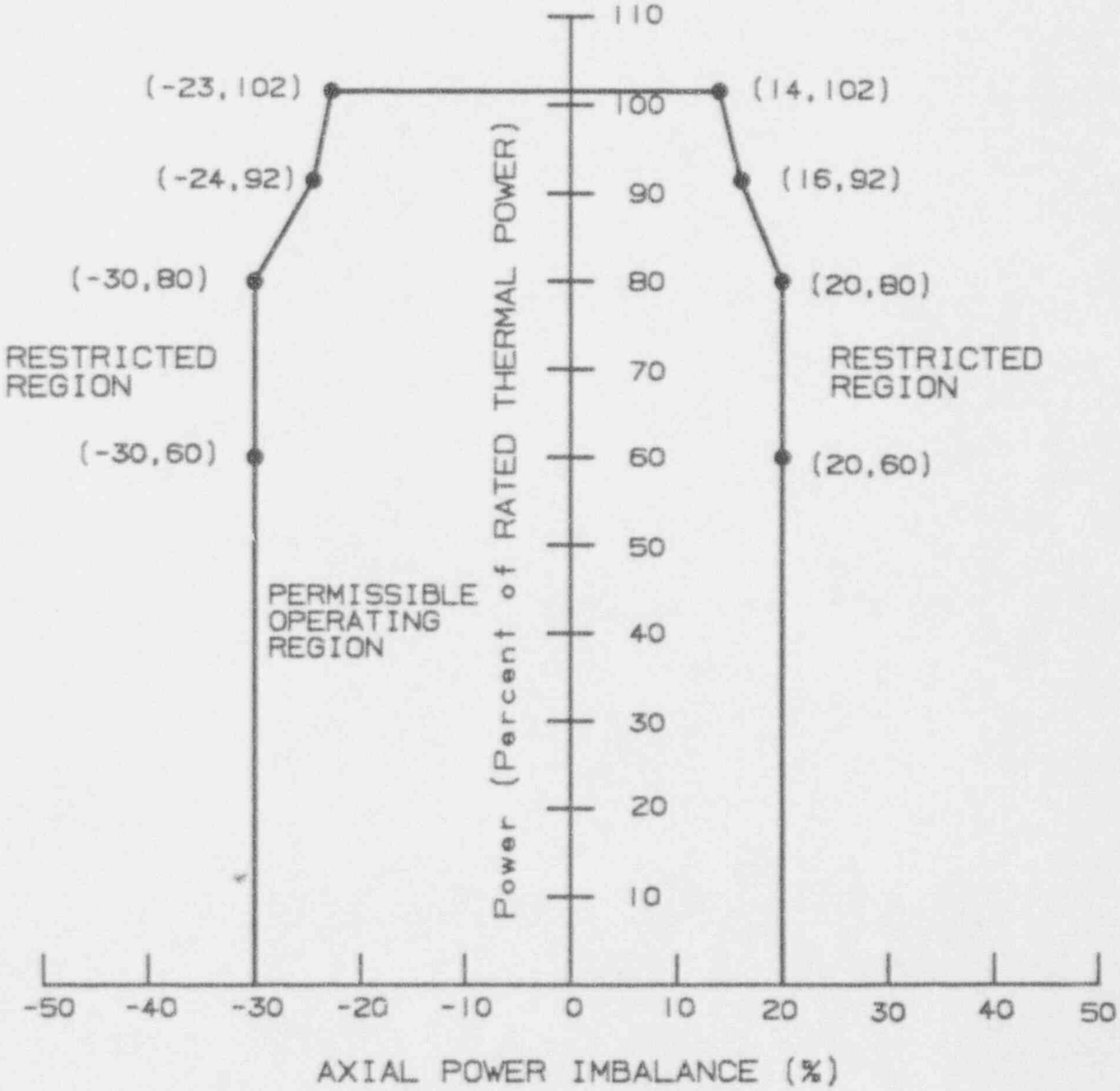


Figure 8-16

AXIAL POWER IMBALANCE Limits, After 425±10 EFPD, Four RC Pumps-- Davis-Besse 1, Cycle 9

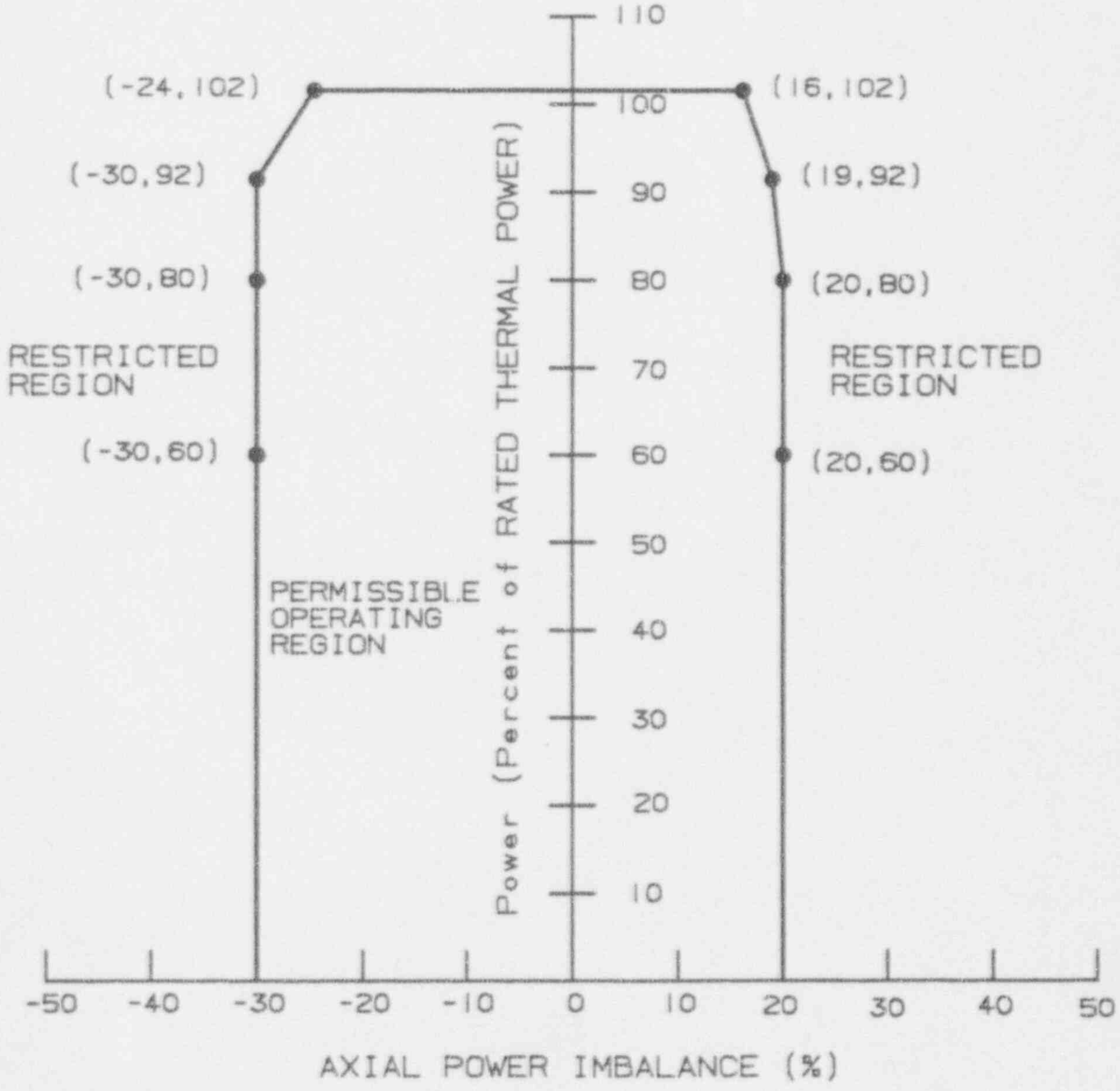


Figure 8-17

AXIAL POWER IMBALANCE Limits, 0 to 75±10 EFPD, Three RC Pumps-- Davis-Besse 1, Cycle 9

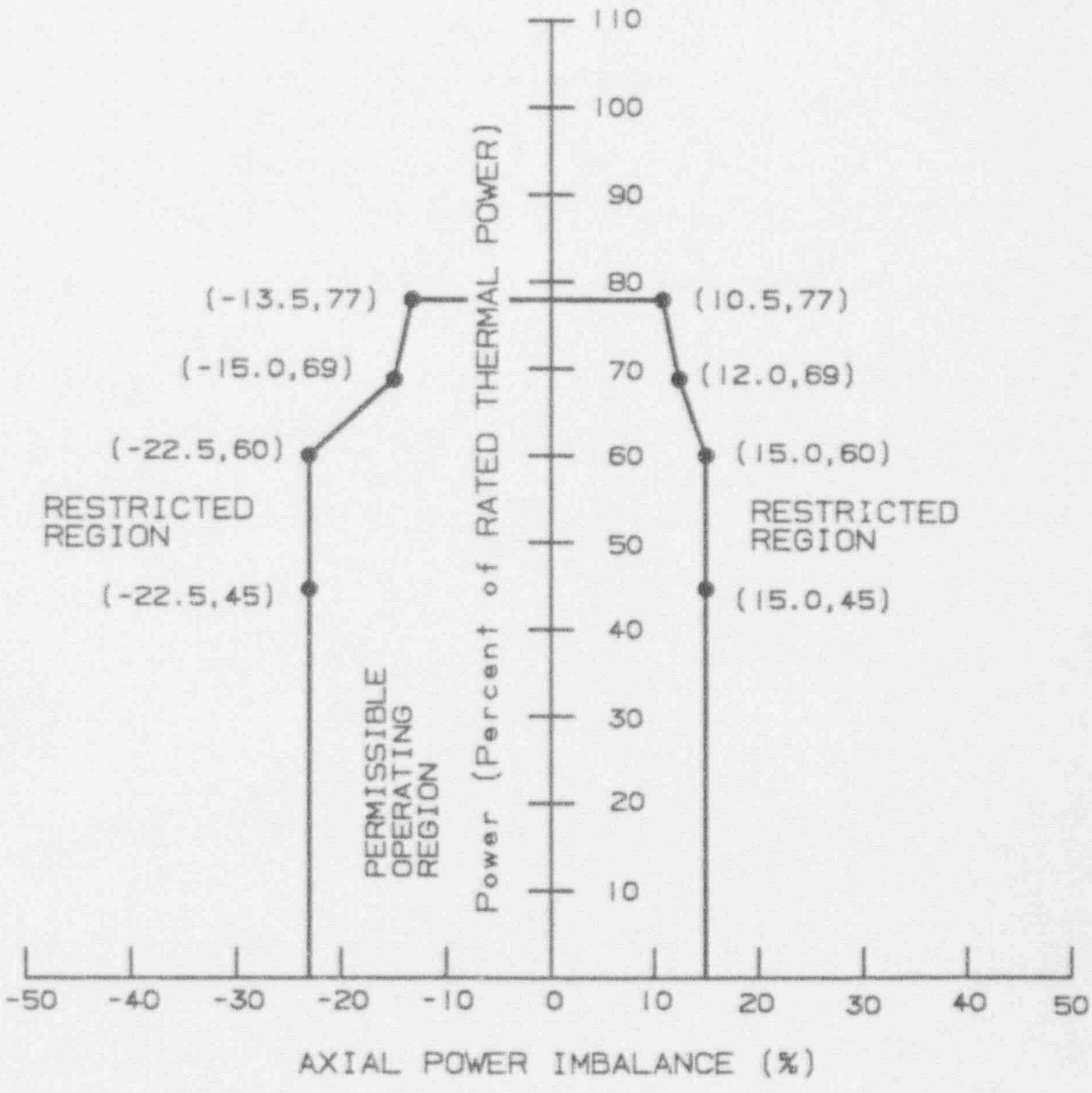


Figure 8-18

AXIAL POWER IMBALANCE Limits, 75±10 to 300±10 EFPD
Three RC Pumps-- Davis-Besse 1, Cycle 9

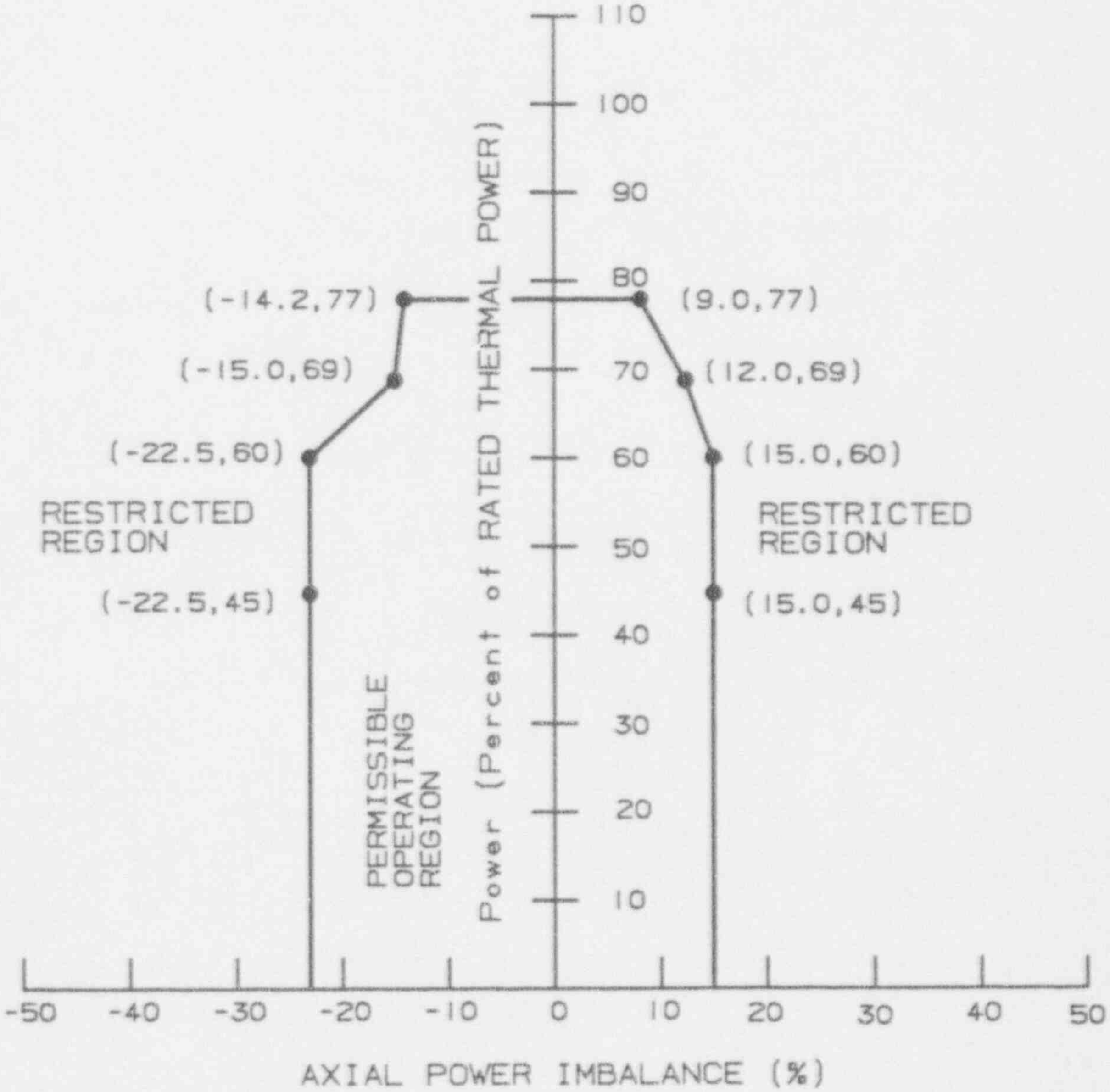


Figure 8-19

AXIAL POWER IMBALANCE Limits, 300±10 to 425±10 EFPD, Three RC Pumps-- Davis-Besse 1, Cycle 9

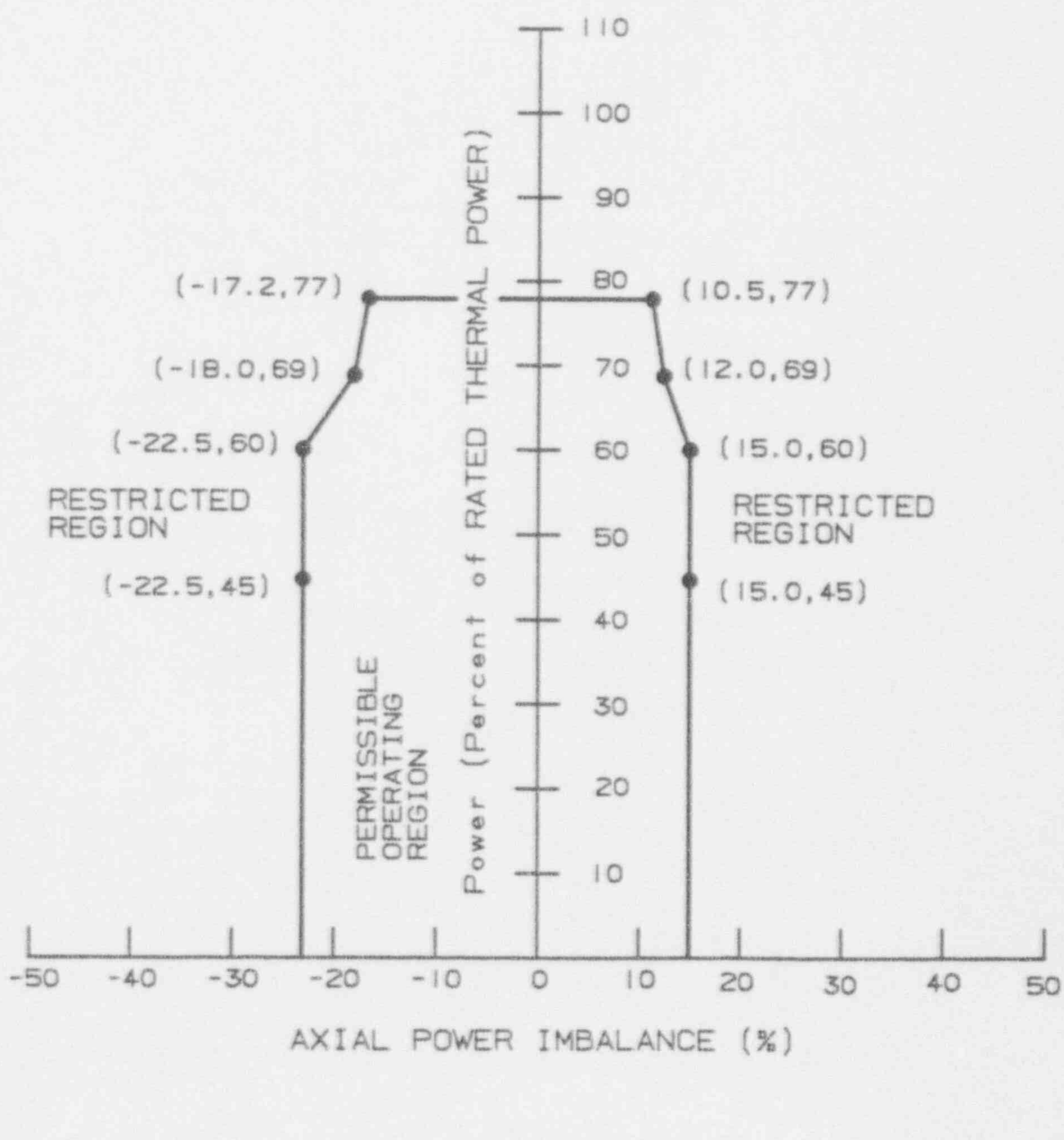


Figure 8-20

AXIAL POWER IMBALANCE Limits, After 425±10 EFPD, Three RC Pumps-- Davis-Besse 1, Cycle 9

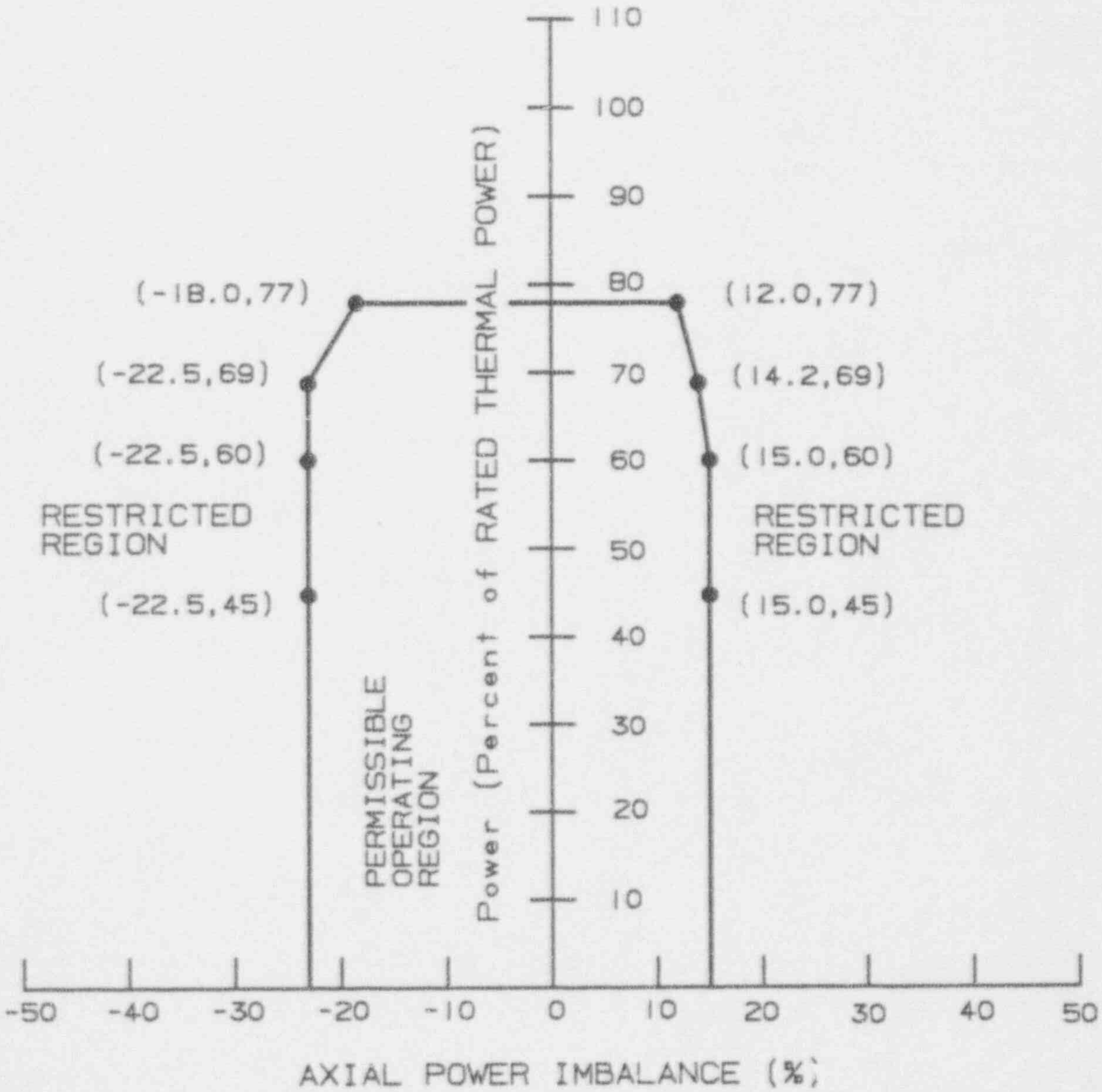


Table 8-1. Quadrant Power Tilt Limits

Quadrant Power Tilt as measured by:	Steady-state Limit for Thermal Power ≤ 60%	Steady-state Limit for Thermal Power > 60%	Transient Limit	Maximum Limit
Symmetrical incore detector system	6.7	3.1	10.03	20.0
Power Range channels	4.0	1.6	6.96	20.0
Minimum incore detector system	2.8*	1.8*	4.40*	20.0

* Assumes detector strings with >60% depletion are excluded from the minimum incore system configuration.

Table 8-2. Negative Moderator Temperature Coefficient Limit

Negative Moderator Temperature Coefficient Limit (at RATED THERMAL POWER)	-3.79×10^{-4} $\Delta K/K/^{\circ}F$

9. STARTUP PROGRAM - PHYSICS TESTING

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide information for continued safe operation of the unit.

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Acceptance criteria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.58 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop from fully withdrawn to two-thirds inserted. Since the most accurate position indication is obtained from the zone reference switch at the 75% inserted position, this position is used instead of the two-thirds inserted position for data gathering.

9.1.2. RC Flow

Reactor coolant flow with four RC pumps running will be measured at hot standby conditions. The measured flow shall be within allowable limits.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Once initial criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required to achieve the all rods out equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentration shall be within ± 50 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal HZP temperature coefficient is measured at approximately the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. The change in reactivity is then calculated by the summation of reactivity associated with the temperature change. The acceptance criterion for the temperature coefficient is that the measured value shall not differ from the predicted value by more than $\pm$ 0.2 x 10^{-2} $\%\Delta k/k/^{o}F$.

The moderator temperature coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator temperature coefficient. This value shall be less than +0.9 x 10^{-2} $\%\Delta k/k/^{o}F$.

9.2.3. Control Rod Group/Boron Reactivity Worth

Individual control rod group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. This technique consists of deborating the reactor coolant system and compensating for the reactivity changes from this deboration by inserting individual control rod groups 7, 6, and 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on reactimeter data, and differential rod worths are obtained from the measured reactivity worth versus the change in rod group position. The differential rod worths of each of the controlling groups are then summed to obtain integral rod group worths. The acceptance criteria for the control rod group worths are as follows:

1. Individual group 5, 6, 7 worth:

$$\left|\frac{\text{predicted value - measured value}}{\text{predicted value}}\right| \times 100\% \text{ shall be } \leq 15\%$$

2. Sums of groups 5, 6, and 7:

$$\left|\frac{\text{predicted value - measured value}}{\text{predicted value}}\right| \times 100\% \text{ shall be } \leq 10\%$$

The boron reactivity worth (differential boron worth) is measured by dividing the

total inserted rod worth by the boron change made for the rod worth test. The acceptance criterion for measured differential boron worth is as follows:

1. $\left|\frac{\text{predicted value - measured value}}{\text{predicted value}}\right| \times 100\%$ shall be $\leq 15\%$

The predicted rod worths and differential boron worth are taken from the PTM.

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

The purpose of this test is to evaluate the symmetry of the core at low power during the initial power escalation following a refueling. Symmetry evaluation is based on incore quadrant power tilts during escalation to the intermediate power level. The absolute values of the quadrant power tilts should be less than the COLR limit.

9.3.2. Core Power Distribution Verification at Intermediate Power Level (IPL) and 100% FP With Nominal Control Rod Position

Core power distribution tests are performed at the IPL and approximately 100% full power (FP). Equilibrium xenon is established prior to both the IPL and 100% FP tests. The test at the IPL is essentially a check of the power distribution in the core to identify any abnormalities before escalating to the 100% FP plateau. Peaking factor criteria are applied to the IPL core power distribution results to determine if additional tests or analyses are required prior to 100% FP operation.

The following acceptance criteria are placed on the IPL and 100% FP tests:

1. The maximum LHR shall be less than the LOCA limit.
2. The minimum DNBR shall be greater than the 102 %FP initial condition DNBR limit (see Table 6-1).
3. The value obtained from extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint shall be greater than the 112 %FP initial condition DNBR limit (see Table 6-1), or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.
4. The value obtained from extrapolation of the worst-case maximum LHR to the next power plateau overpower trip setpoint shall be less than the fuel

melt limit, or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.

5. The quadrant power tilt shall not exceed the limits specified in the COLR.

6. The measured radial (assembly) peaks for fresh fuel locations shall be within the following limits:

$$\frac{\text{predicted value - measured value}}{\text{predicted value}} \times 100\% \text{ more positive than } -3.8\%$$

7. The measured total (segment) peaks for fresh fuel locations shall be within the following limits:

$$\frac{\text{predicted value - measured value}}{\text{predicted value}} \times 100\% \text{ more positive than } -4.8\%$$

The following review criteria also apply to the core power distribution results at the IPL and at 100 %FP:

8. The RMS of the differences between predicted and measured radial (assembly) peaking factors should be less than 0.05.

9. For all other core locations, the (absolute) difference between predicted and measured radial (assembly) peaking factors should be less than 0.10.

Items 1, 2, and 5 ensure that the initial condition limits are maintained at the IPL and 100% FP.

Items 3 and 4 establish the criteria whereby escalation to full power may be accomplished without exceeding the safety limits specified by the safety analysis with regard to DNBR and linear heat rate.

Items 6 and 7 are established to determine if measured and predicted power distributions are within allowable tolerances assumed in the reload analysis.

Items 8 and 9 are review criteria, established to determine if measured and predicted power distributions are consistent.

9.3.3. Incore Vs. Excore Detector Imbalance Correlation Verification at the IPL

Imbalances, set up in the core by control rod positioning, are read simultaneously on the incore detectors and excore power range detectors. The excore detector offset versus incore detector offset slope shall be greater than 0.96 and the y-intercept (excore offset) shall be between -2.5 and 2.5%. If either of these criteria are not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required slope and/or intercept.

9.3.4. Temperature Reactivity Coefficient at ~100% FP

The average reactor coolant temperature is decreased and then increased at constant reactor power. The reactivity associated with each temperature change is obtained from the change in the controlling rod group position. Controlling rod group worth is measured by the fast insert/withdraw method. The temperature reactivity coefficient is calculated from the measured changes in reactivity and temperature. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator temperature coefficient. The measured moderator temperature coefficient shall be negative.

9.3.5. Power Doppler Reactivity Coefficient at ~100% FP

The power Doppler reactivity coefficient is calculated from data recorded during control rod worth measurements at power using the fast insert/withdraw method. The fuel Doppler reactivity coefficient is calculated in conjunction with the power Doppler coefficient measurement. The power Doppler coefficient as measured above is multiplied by a precalculated conversion factor to obtain the fuel Doppler coefficient. This measured fuel Doppler coefficient shall be more negative than the acceptance criteria limit of -0.90×10^{-3} %$\Delta k/k/^{o}F$.

9.3.6. Hot Full Power All Rods Out Critical Boron Concentration

The hot full power (HFP) all rods out critical boron concentration (AROCBC) is determined at ~100 %FP by first recording the RCS boron concentration during equilibrium, steady state conditions. Corrections to the measured RCS boron concentration are made for control rod group insertion and power deficit (if not at 100 %FP) using predicted data for CRG worth, power Doppler coefficient, and

differential boron worth. A correction may also be made to account for the observed difference between the measured and predicted AROCBC at zero power. The acceptance criterion placed on the HFP AROCBC is that the measured AROCBC shall be within ± 50 ppm boron of the predicted value.

9.4. Procedure for Use if Acceptance/Review Criteria Not Met

If an acceptance criterion ("shall" as opposed to "should") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is allowed. This evaluation is performed by site test personnel with participation by B&W Nuclear Technologies technical personnel as required. Further specific actions depend on evaluation results. These actions can include repeating the tests with more detailed test prerequisites and/or steps, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

If a review criterion ("should" as opposed to "shall") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is recommended. This evaluation is similar to that performed to address failure of an acceptance criterion.

10. REFERENCES

1. Davis-Besse Nuclear Power Station No. 1, Updated Safety Analysis Report, Docket No. 50-346.

2. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084P, Rev. 2, Babcock and Wilcox, Lynchburg, VA, October 1978.

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