ENCLOSURE II

BAW-2137, Rev. 1
October 1991

DAVIS-BESSE NUCLEAR POWER STATION
UNIT 1, CYCLE 8 -- RELOAD REPORT

B&W Fuel Company

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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 8. 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.

Toledo Edison has an objective of operating with zero fuel defects. To accomplish this objective, all once- and twice-burned fuel assemblies were ultrasonically tested for leaking rods at the end of cycle 7. As a result of this inspection, five fuel assemblies were determined to have one questionable or defective fuel rod. Three of the fuel assemblies were in batch 8 and two were in batch 9. The batch 8 assemblies did not have reconstitutable mechanical features, and were discharged (in order to prevent leaking fuel from being used in cycle 8) along with symmetrically located assemblies. The batch 9 assemblies have the reconstitutable mechanical features that provide easy repair capability. However, one of the batch 9 assemblies had one fuel rod that was severely damaged. That assembly was discharged along with three other symmetrically located batch 9 assemblies. The remaining batch 9 assembly was repaired, as described in section 4.1, with a stainless steel rod and reused in cycle 8.

The changes described above resulted in a revision to the core loading that was presented and analyzed in the original cycle 8 reload report.[1] The balance of this report provides the description and analysis of the revised cycle 8 design.

Cycle 8 reactor and fuel parameters related to power capability are summarized in this report and compared to those for cycle 7. All accidents analyzed in the Davis-Besse Updated Safety Analysis Report[2] (USAR), as applicable, have been reviewed for cycle 8 operation, and in cases where cycle 8 characteristics were conservative when compared to previous values, new analyses were not performed. In all cases, the cycle 8 parameters are bounded.

NRC Generic Letter 88-16 allows the placement of numeric values of certain cycle specific parameters in a Core Operating Limits Report (COLR). On September 20, 1990 Toledo Edison issued a revision to the plant Technical Specifications which implemented the COLR concept for Davis-Besse 1, cycle 7. The COLR changes for cycle 8 are included in section 8 of this report.

The Technical Specifications have been reviewed for cycle 8 operation. Based on the 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 8 can be operated safely at its licensed core power level of 2772 MWt.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Davis-Besse Unit 1 is cycle 7, which achieved criticality on July 1, 1990. Power escalation began on July 3, 1990 and full power (2772 MWt) was attained on July 10, 1990. Cycle 7 was shutdown for refueling after 405.4 effective full power days (EFPD) of operation.

During cycle 7 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 8. The scheduled duration of cycle 8 is 469 EFPD. Cycle 8 was analyzed to 479 EFPD and the applicability of the cycle 7 reactor protection system (RPS) limits and setpoints to cycle 8 has been verified to 479 EFPD. The cycle 8 operating limits have also been verified to 479 EFPD. The cycle 8 design includes an APSR pull and power coastdown.

The cycle 8 design minimizes the number of fuel assemblies that are cross core shuffled to reduce the potential for quadrant tilt amplification. The cycle 8 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[2] for the unit. The cycle 8 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 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 8. Fifty-three batch 7B assemblies, 7 batch 8A assemblies, and 4 batch 9A assemblies were discharged at the end of cycle 7. The fuel assemblies in batches 8B and 9B were shuffled to their cycle 8 locations, with the core periphery locations occupied by both batch 8B and 9B fuel assemblies. Batches 8 and 9 had initial enrichments of 3.13 and 3.38 wt %, respectively. The feed batch, consisting of 64 batch 10 assemblies with uranium enrichment of 3.69 wt %, was inserted in the core interior in a symmetric checkerboard pattern with the batch 8B and 9B FAs. The shuffle scheme is a partial very low leakage core loading. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 8 and its initial enrichment.

Cycle 8 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 were reinserted from cycle 7. 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 8 locations of the control rods and the group designations are indicated in Figure 3-3. The core locations and the rod group

group designations of the 61 control rods in cycles 7 and 8 are the same. The cycle 8 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Davis-Besse Cycle 8 Core Loading Diagram

← North

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						8B P10	9B K4	9B N3	9B K12	8B M8					
B				8B C5	9B L3	9B K2	10 F	8B P11	10 F	9B K14	9B L13	8B C11			
C			8B H14	10 F	9B K6	10 F	9B L5	10 F	9B L11	10 F	9B K10	10 F	8B P8		
D		8B E3	10 F	9B E4	10 F	8B A7	10 F	8B A8	10 F	8B A9	10 F	9B D11	10 F	8B E13	
E		9B C10	9B F9	10 F	8B D2	10 F	8B L14	10 F	8B G8	10 F	8B B12	10 F	9B F7	9B C6	
F	8B H11	9B B9	10 F	8B G1	10 F	8B B4	10 F	9B O8	10 F	8B D14	10 F	8B G15	10 F	9B B7	8B L2
G	9B D9	10 F	9B E10	10 F	8B H7	10 F	8B A6	9B O4	8B F15	10 F	8B P6	10 F	9B E6	10 F	9B D7
H	9B O12	8B E14	10 F	8B H1	10 F	9B H13	9B N13	8B P5	9B D3	9B H3	10 F	8B H15	10 F	8B M2	9B C4
K	9B N9	10 F	9B M10	10 F	8B B10	10 F	8B L1	9B C12	8B K10	10 F	8B H9	10 F	9B M6	10 F	9B N7
L	8B F14	9B P9	10 F	8B K1	10 F	8B N2	10 F	9B C8	10 F	8B P12	10 F	8B K15	10 F	9B P7	8B H5
M		9B O10	9B L9	10 F	8B P4	10 F	8B K8	10 F	8B F2	10 F	8B N14	10 F	9B L7	9B O6	
N		8B M3	10 F	9B N5	10 F	8B R7	10 F	[illegible]B K8	10 F	8B R9	10 F	9B M12	10 F	8B M13	
O			8B B8	10 F	9B G6	10 F	9B F5	10 F	9B F11	10 F	9B G10	10 F	8B H2		
P				8B O5	9B F3	9B G2	10 F	8B B5	10 F	9B G14	9B F13	8B O11			
R						8B E8	9B G4	9B D13	9B G12	8B B6					

Key:

xxx yyy

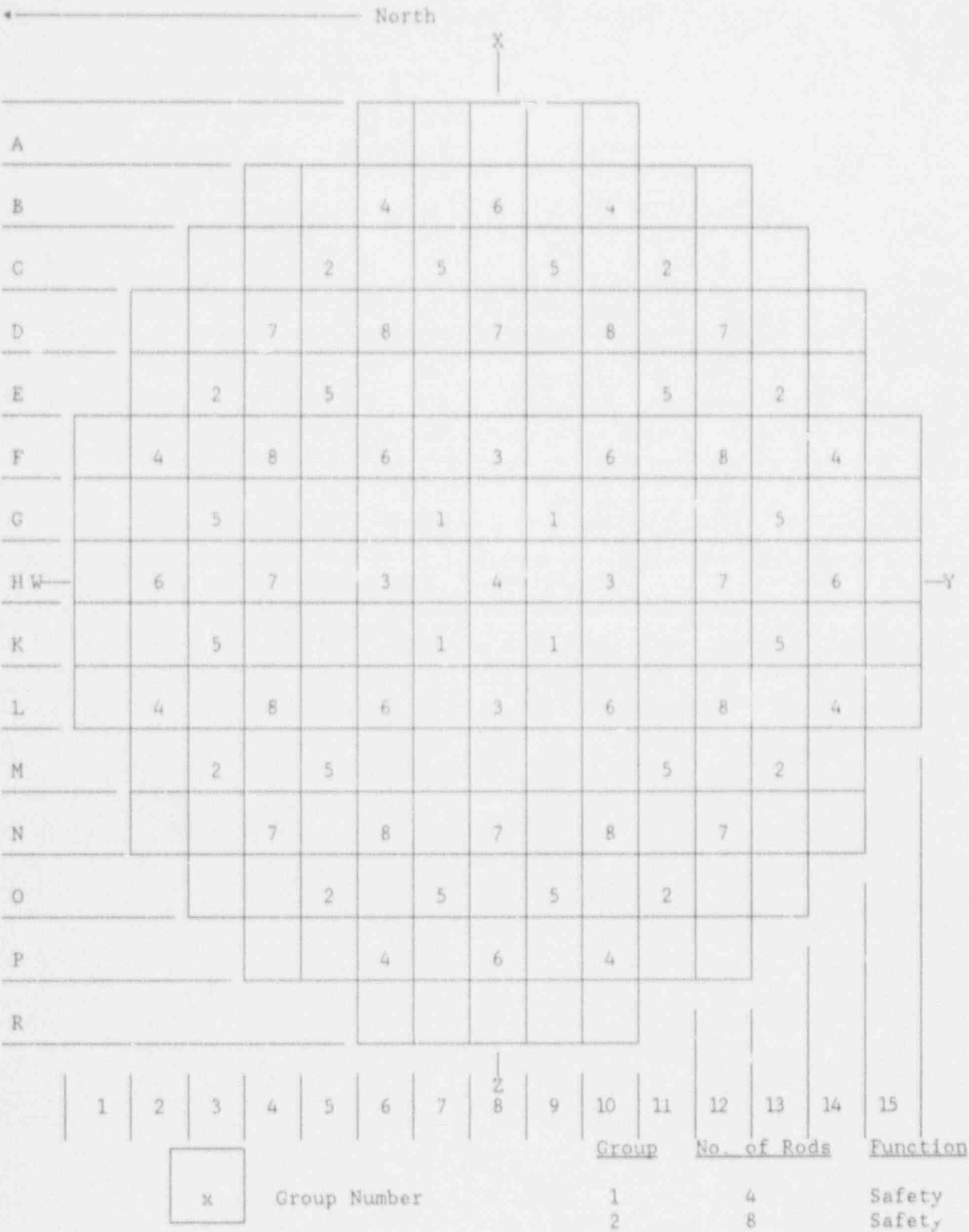
xxx - Batch ID
yyy - Location in Previous Cycle

Figure 3-2. Davis-Besse Cycle 8 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.13 24534	3.38 13334	3.38 17635	3.69 0	3.13 23757	3.69 0	3.13 24529	3.38 13362
K	3.38 13334	3.13 22398	3.69 0	3.13 26977	3.69 0	3.38 17902	3.69 0	3.38 17633
L	3.38 17636	3.69 0	3.13 22588	3.69 0	3.13 24106	3.69 0	3.38 15233	3.13 27961
M	3.69 0	3.13 27735	3.69 0	3.13 22590	3.69 0	3.38 17661	3.38 16812	
N	3.13 23753	3.69 0	3.13 24092	3.69 0	3.38 17573	3.69 0	3.13 28050	
O	3.69 0	3.38 17859	3.69 0	3.38 17645	3.69 0	3.13 30466		
P	3.13 24529	3.69 0	3.38 15240	3.38 16816	3.13 28063			
R	3.38 13362	3.38 17624	3.13 27722					

x.xx yyyyy	Initial Enrichment BOC Burnup MWD/MTU

Figure 3-3. Control Rod Locations for Davis-Besse 1, Cycle 8



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 8 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				2.0		2.0		
K			1.7		1.7		0.2†	
L		1.7		1.4		1.1		
M	2.0		1.4		1.7			
N		1.7		1.7				
O	2.0		1.1					
P		0.2†						
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 8 are listed in Table 4-1. Batch 8B is the Mark-B5A design, while batches 9 and 10 are of the Mark-B8A design. Batch 10 fuel incorporates all of the features of the batch 9 fuel but includes a slight reduction in pin prepressure to increase the similarity in mechanical and thermal performance to that of the Mark-B5A design.

Eight gray APSRs and 53 full length Ag-In-Cd control rods will be used in cycle 8. Forty-eight new Mark-B5 BPRAs were introduced into the core along with 8 once-burned Mark-B5 BPRAs 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 1,000 EFPD.

One fuel assembly (NJ0542) was reconstituted. This consisted of removing the upper end fitting and removing one leaking fuel rod which had been identified by UT. The leaking fuel rod was replaced by another fuel rod from within that assembly. That rod in turn was replaced by a SS304 solid rod or pin. Then the upper end fitting was reattached to the fuel assembly.

When a fuel assembly is reconstituted, failed rods are replaced with solid stainless steel (SS304) rods. The steel rods are designed and analyzed to ensure that there is no adverse impact on fuel assembly performance. The dummy steel rod design is determined such that:

a) The steel rod engages all spacer grid stops under all conditions,
b) Clearance is maintained between the dummy steel rod and the top grillage under reactor temperature and irradiation conditions.
c) Thermal expansion of the steel rod will not cause any set of the spring stops on the spacer grids, and

d) The difference in mass between the steel rod and the fuel pins will not affect fuel assembly lift.

The effect of differential temperature expansion between the stainless steel pin and the Zircaloy guide tubes was analyzed over the operating and shutdown temperature range. Additionally, the irradiation growth of the guide tubes was analyzed with respect to the stainless steel rod.

The thermal expansion of the stainless steel pin in the radial direction will be about three times more than a Zircaloy-4 clad fuel pin. With thermal expansion the stainless steel pin will compress the spring stops on the spacer grids about 0.002 inches more than the fuel rods. That compression is within the elastic range of the spring stops and will not cause any set of the spring stops.

The mass of the steel rod is slightly less than that of a fuel rod. The difference is about one lb_m. For one dummy rod the effect on fuel assembly lift and response is insignificant.

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 three fuel batches was determined. These histories were compared to generic and previous creep collapse analyses. Both the generic and the previous analyses are based on the method from reference 3 and are applicable to the batch 8B design for cycle 8 operation. The analyses predicted a creep collapse life in excess of 35,000 effective full power hours (EFPH). This is longer than the maximum batch 8B residence of 29,983 EFPH.

For batches 9 and 10, the creep collapse analysis followed the method from reference 4. The operational conditions and mechanical characteristics of the batch 9 and 10 fuel assemblies were compared to an envelope formulated by BWFC and approved by the NRC[5]. All values for the Mark-B8A fuel assemblies are encompassed by the corresponding parameters of the limiting envelope. The as-built data for the batch 10 fuel assemblies was approximated from partial batch 9 as-built data for Mark-B8A assemblies and then compared to the limiting envelope. This is reasonable as the tolerances affecting these as-built values have not changed from previous batch fuel designs. The creep collapse life of the batch 9 and 10 fuel rods based on reference 5 is 55,000 MWd/mtU.

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:

$$P_m < 1.0\ S_m$$
$$P_l < 1.5\ S_m$$
$$P_m + P_b < 1.5\ S_m$$
$$P_m + P_b + Q < 3.0\ S_m$$

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 18.7%. 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 1.0% on cladding plastic tensile circumferential strain. The fuel pellet is designed to ensure that plastic cladding strain is less than 1% at design local pellet burnup and heat generation rate. The design values are higher than the worst case values Davis-Besse Unit 1, cycle 8 fuel is expected to experience. For the batch 8B, Mark-B5A fuel assemblies, a generic strain analysis was reviewed conservatively based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance limit for cladding inside diameter. For the Mark-B8A fuel assemblies from batches 9 and 10, a strain analysis was done utilizing the method of reference 6.

4.3 Thermal Design

All fuel in the cycle 8 core is thermally similar. The design of the batch 10 Mark-B8A assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. The Mark-B8A fuel rod design includes a grippable upper end cap and a bullet nose lower end cap. These fuel rod design features have no effect on thermal performance. One batch 9B fuel assembly was reconstituted; this has no effect on its thermal performance. Fuel thermal analyses were performed with the TACO2[6] fuel pin performance code. Nominal undensified input parameters used in the analysis are presented in Table 4-1. Densification effects were accounted for in the TACO2 code densification model.

The results of the thermal design evaluation of the cycle 8 core are summarized in Table 4-1. Linear heat rate (LHR) to fuel melt capability for all fuel was determined with TACO2. The analyses performed for cycle 8 demonstrate that 20.5 kW/ft is a conservative limit to preclude centerline fuel melt (CFM) for all fuel batches.

The maximum fuel pin burnup at EOC 8 is predicted to be less than 46,840 MWd/mtU (batch 8B). Fuel rod internal pressure has been evaluated with TACO2 for the highest burnup fuel rod and is predicted to be less than the 2200 psia reactor coolant pressure at the core outlet.

4.4 Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for batch 10 fuel assemblies is identical to that of present fuel assemblies.

4.5 Operating Experience

Babcock & Wilcox 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	Max FA Burnup, MWd/mtU[a] Discharged	Cumulative net electric output, MWh[b]
Oconee 1	13	39,438	58,310	85,555,367
Oconee 2	12	36,393	42,820	79,956,383
Oconee 3	12	45,795	39,701	79,420,910
Three Mile Island	8	30,366	33,975	47,252,376
Arkansas Nuclear One, Unit 1	10	34,421	57,318	64,615,141
Rancho Seco	7	[c]	38,268	43,208,092
Crystal River 3	8	33,857	40,600	51,714,542
Davis-Besse	7	36,220	40,300	39,295,898

[a] As of February 28, 1991.

[b] As of February 28, 1990.

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

Table 4-1. Fuel Design Parameters

	Batch 8B	Batch 9B[a]	Batch 10
Fuel assembly type	Mark-B5A	Mark-B8A	Mark-B8A
No. of assemblies	57	56	64
Fuel rod OD, in.	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377
Tubular spacer	Zr-4	NA	NA
Undensified active fuel length, in.	143.2	143.2	143.2
Fuel pellet (mean) diameter, in.	0.3686	0.3686	0.3686
Fuel pellet initial density, %TD mean	95.0	95.0	95.0
Initial fuel batch enrichment w/o U235	3.13	3.38	3.69
Average burnup BOC, MWd/mtU	25,766	16,595	0
Exposure Time, EFPH	29,983	20,856	11,256
Cladding collapse time, EFPH	>35,000	NA	NA
Maximum assembly burnup, MWd/mtU	44,250	36,530	21,530
Cladding Collapse Burnup, MWd/mtU	NA	55,000	55,000
Nominal linear heat rate at 2772 MWt, kW/ft	6.14	6.14	6.14
Minimum linear heat rate to melt, kW/ft	20.5	20.5	20.5

[a] Includes one reconstituted assembly (NJ0542) which has a SS304 solid pin in place of a fuel rod.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 7 and 8 designs. The values for cycles 7 and 8 were both generated with the NOODLE[7] code. | 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 8 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 8 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. A maximum flux redistribution penalty.
4. A maximum power deficit with minimum boron.

The maximum flux redistribution 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

There are no significant core design changes for cycle 8. The calculational models and the methods used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle, except the calculations performed at end of cycle (EOC) for the Doppler and moderator coefficients as discussed in the footnotes to Table 5-1. An additional parameter, temperature coefficient at steam line break conditions, is included for cycle 8 (see footnote (f) of Table 5-1). The purpose of this new parameter is discussed in section 7. 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 $-0.0439h^{-1}$, 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 8 Physics Parameters

	Cycle 7	Cycle 8[a]
Cycle length, EFPD	415	479[b]
Cycle burnup, MWd/mtU	13,880	16,020
Average core burnup - EOC, MWd/mtU	26,806	29,568
Initial core loading, mtU	82.9	82.9
Critical boron[c] - BOC, No Xe, ppm		
HZP	1,506	1,737
HFP	1,323	1,515
Critical boron[c] - EOC, Eq. Xe, ppm		
HZP	200	207
HFP	10[d]	10[d]
Control rod worths - HFP, BOC, %Δk/k		
Group 6	1.14	1.02
Group 7	1.03	1.02
Group 8	0.19	0.15
Control rod worths - HFP, EOC, %Δk/k		
Group 7	1.09	1.13
Group 8	NA	NA
Max ejected rod worth - HZP, %Δk/k		
BOC, Groups 5-8 inserted (L-10,N-12)	0.32	0.30
EOC, Groups 5-7 inserted (L-10,N-12)	0.38	0.35
Max stuck rod worth - HZP, %Δk/k		
BOC (M-13,N-12)	0.60	0.77
EOC (M-11)	0.81	0.68
Power deficit - HZP to HFP, Eq. Xe, %Δk/k		
BOC (4 EFPD)	-1.71	-1.74
EOC	-2.55	-2.71
Doppler coeff - HFP, 10^{-3} %Δk/k/°F		
BOC, No Xe[e] Group 8 inserted	-1.58	-1.[illegible]
EOC, Eq. Xe, 0 ppm, Group 8 withdrawn	-1.86	-1.96
Moderator coeff - HFP, 10^{-2} %Δk/k/°F		
BOC, No Xe[e]	-0.70	-0.66
EOC, Eq. Xe, 0 ppm[f]	-2.93	-3.33

Table 5-1. Davis-Besse Unit 1, Cycle 8 Physics Parameters (Continued)

	Cycle 7	Cycle 8
Boron worth - HFP, ppm/%Δk/k		
BOC [e]	129	138
EOC	110	114
Xenon worth - HFP, %Δk/k		
BOC (4 EFPD)	2.61	2.63
EOC (equilibrium)	2.78	2.77
Effective delayed neutron fraction - HFP		
BOC	0.00618	0.00623
EOC	0.00520	0.00514

[a] Based on cycle 6 length of 380.3 EFPD (actual) and cycle 7 length of 400 EFPD; however, all boron concentrations have been adjusted to the actual cycle length of 405.4 EFPD.

[b] All end-of-cycle values calculated at 479 EFPD; the design cycle 8 length is 469 EFPD.

[c] Control rod group 8 is inserted at BOC and withdrawn at EOC.

[d] Power coastdown to EOC at 10 ppm.

[e] Cycle 8 values were calculated at 1549 ppm; cycle 7 values were calculated at 1323 ppm.

[f] The EOC 8 value was calculated with the control rods at the insertion limit. The temperature coefficient calculated at the steam line break conditions is -2.67×10^{-2} %Δk/k/°F and is discussed in section 7.0.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 8

	BOC[a] %Δk/k	410 EFPD[a] %Δk/k	479 EFPD[a] %Δk/k
Available Rod Worth			
Total rod worth, HZP	6.92	7.01	7.05
Worth reduction due to burnup of poison material	-0.16	-0.18	-0.19
Maximum stuck rod, HZP	-0.77	-0.68	-0.68
Net worth	5.99	6.15	6.18
Less 10% uncertainty	-0.60	-0.62	-0.62
Total available worth	5.39	5.53	5.56
Required Rod Worth			
Power deficit, HFP to HZP	1.74	2.66	2.71
Max allowable inserted rod worth	0.33	0.51	0.50
Flux redistribution	0.53	1.02	0.96
Total required worth	2.60	4.19	4.17
Shutdown Margin			
Total available minus total required rod worth	2.79	1.34	1.39

(a) Group 8 is at the nominal position at BOC and 410 EFPD and is out at 479 EFPD.

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

Figure 5-1. Davis-Besse Cycle 8 Relative Power Distribution at BOC (4 EFPD), Full Power, Equilibrium Xenon, All Rods Out, APSRs Inserted[a]

	8	9	10	11	12	13	14	15
H	0.966	1.180	1.204	1.255	1.063	1.304	0.904	0.567
K	1.184	1.048	1.280	1.000	1.294	1.244	1.250	0.524
L	1.209	1.283	1.066	1.288	8 1.041	1.292	0.881	0.298
M	1.259	0.998	1.285	1.083	1.299	1.061	0.612	
N	1.066	1.295	8 1.040	1.297	1.182	1.140	0.332	
O	1.309	1.249	1.295	1.062	1.142	0.428		
P	0.909	1.257	0.887	0.614	0.333			
R	0.571	0.530	0.307					

x	Inserted Rod Group Number
x.xxx	Relative Power Density

[a] Calculated results from the two dimensional pin-by-pin PDQ07.

6.0 THERMAL-HYDRAULIC DESIGN

The Mark-B8A fuel assemblies inserted for cycle 8 have Zircaloy intermediate spacer grids. Zircaloy intermediate spacer grids were first inserted on a full batch basis in cycle 7 of Davis-Besse Unit 1. The thermal-hydraulic design evaluation supporting cycle 8 operation utilized the same methods and models as cycle 7 (described in references 8, 9 and 10 as supplemented by reference 11) which implemented the BWC (reference 12) CHF correlation for analysis of Zircaloy-grid fuel assemblies. The analyses presented in section 5 of reference 11 demonstrated that changes in the flow parameters resulting from the incorporation of Zircaloy spacer grids do not significantly impact the thermal-hydraulic characteristics of a Zircaloy grid core relative to the standard Inconel-grid (Mark-B) core. Implementation of the Zircaloy-grid fuel assemblies into existing reactors, however, is performed on a batch basis, with the transition cycles having both Zircaloy grid and standard Mark-B fuel assemblies.

The Mark-B8A fuel assembly has a slightly higher pressure drop than the standard Mark-B assembly due to the higher flow resistance of the Zircaloy spacer grids. The presence of Mark-B8A fuel assemblies in a mixed core, will, therefore, tend to divert some flow from the more restrictive Mark-B8A assemblies to the Mark-B fuel. As a result, the Mark-B8A fuel assemblies in a mixed core will experience slightly less coolant flow than in a homogeneous Mark-B8A core. This reduced flow results in a reduced thermal margin for the Mark-B8A assemblies relative to a full Mark-B8A core. The amount of coolant flow reduction is dependent on the number of Mark-B8A assemblies (with the smaller number of Mark-B8A assemblies being more limiting). A "transition core penalty" is, therefore, considered for a mixed Zircaloy-grid and standard Mark-B core. For cycle 8 of Davis-Besse Unit 1, this transition penalty is offset by the consideration of a core bypass flow fraction in the thermal-hydraulic model that is higher than the actual value.

The Mark-B8A fuel rod design includes a grippable upper end cap and a bullet nose lower end cap. These fuel rod design features have an insignificant effect on thermal-hydraulic performance.

The 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 during normal operation. If the core should approach the DNB-based core safety limits, the reconstituted fuel assembly would still retain over 40% 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.

B&W-designed reactors, including Davis-Besse Unit 1, currently operate without orifice rod assemblies in the control rod guide tubes (CRGTs). The core bypass fraction is dependent on the number of unplugged guide tubes, which is in turn dependent on the number of burnable poison rods (BPRAs) and control rod assemblies (CRAs), since these components restrict flow through the (CRGTs). For thermal-hydraulic analysis, the most limiting case is that with the higher bypass flow fraction, or smaller number of BPRAs.

The design basis chosen for cycle 8 thermal-hydraulic analyses was a full Zircaloy grid core, containing 37 BPRAs, for which the core bypass flow is 8.9%. This design configuration was used to calculate the 1.54 DNBR (112% FP) shown on Table 6-1 for cycle 8. The actual cycle 8 core configuration consists of 64 fresh Mark-B8A fuel assemblies, 56 second cycle Mark-B8A fuel assemblies, 57 standard Mark-B fuel assemblies and 56 BPRAs. The core bypass flow for the cycle 8 configuration is 8.5%. The DNBR for this configuration, using the same core conditions presented in Table 6-1, is greater than 1.56. A comparison of the DNBRs for the design and the actual core configuration shows that the design configuration is conservative for cycle 8 DNBR analyses. Therefore, a transition core penalty is not necessary. Table 6-1 provides a summary comparison of the DNB analysis parameters for cycles 7 and 8.

Table 6-1. Maximum Design Conditions, Cycles 7 and 8

	Cycle 7	Cycle 8
Rated thermal 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, % [a]	8.9	8.9
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	143.2
Avg heat flux at 100% power, 10^5 Btu/h-ft^2	1.86	1.86
Max heat flux at 100% power, 10^5 Btu/h-ft^2	5.25	5.25
CHF correlation	BWC	BWC
CHF correlation DNB limit	1.18	1.18
Minimum DNBR		
at 102% power	1.78	1.78 [b]
at 112% power	1.54	1.54 [b]

[a] Used in the analysis.

[b] Calculated for the instrument guide tube subchannel which is limiting for the Mark-B8A 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 8 parameters to determine the effects of the cycle 8 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 13.

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 Unit 1 cycle 8. The dose calculations were performed consistent with the assumptions described in the Davis-Besse Unit 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).

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 8 are given in Table 4-1. The cycle 8 thermal-hydraulic maximum design conditions are presented in Table 6-1. A comparison of the key kinetics parameters from the USAR and cycle 8 is provided in Table 7-1.

The EOC moderator temperature coefficient listed in Table 7-1 for cycle 8 is the 3-D, hot full power (HFP) temperature coefficient. An evaluation was performed to verify the acceptability of the more negative cycle 8 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 8 is shown in section 5, Table 5-1, footnote (f) as -2.67×10^{-2} %Δk/k/°F. Since the safety analysis value for the combined temperature coefficient is more negative than the cycle 8 value, the steam line break analysis remains bounding for cycle 8.

A generic loss-of-coolant accident (LOCA) analysis for B&W 177-FA raised-loop nuclear steam system (NSS) has been performed. The final acceptance criteria B&W ECCS evaluation model techniques and assumptions, as described in BAW-10104P, Rev. 5[14], were used in this analysis. The application of the evaluation model[15] includes the impacts of the NUREG-0630 fuel pin rupture curves and FLECSET reflooding heat transfer coefficient calculations. 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 (LHR) and the lifetime pin pressure data for the Mark-B8A fuel used in the LOCA LHR limits analyses is bounded by those calculated for the B&W 177-FA raised loop plant for previous reload evaluations[15]. A tabulation showing the allowable LOCA LHRs for Davis-Besse Unit 1, cycle 8 fuel is provided in Table 7-2.

It is concluded by the examination of cycle 8 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 8. Considering the previously accepted design basis used in the USAR and subsequent cycles, the transient evaluation of cycle 8 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 8 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 8 value
BOL [a] Doppler coeff, 10^{-5}, %Δk/k/°F	-1.28	-1.59
EOL [b] Doppler coeff, 10^{-5}, %Δk/k/°F	-1.45 [c]	-1.96
BOL moderator coeff, 10^{-2}, %Δk/k/°F	+0.13	-0.66
EOL moderator coeff, 10^{-2}, %Δk/k/°F	-3.0	-3.33 [d]
EOL temperature coeff, (532F to 510F), 10^{-2}, %Δk/k/°F	-3.10	-2.67
All rod bank worth (HZP), %Δk/k	10.0	6.92
Boron reactivity worth (HFP), ppm/1%Δk/k	100	138
Max ejected rod worth (HFP), %Δk/k	0.65	0.17
Max dropped rod worth (HFP), %Δk/k	0.65	≤0.20
Initial boron conc (HFP), ppm	1407	1515

[a] BOL denotes beginning of life.
[b] EOL denotes end of life
[c] -1.77×10^{-5} %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 Heat Rates

Core elevation, ft	Allowable peak LHR, 0-40,000 MWd/mtU kW/ft	Allowable peak LHR, 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

8. PROPOSED MODIFICATIONS TO CORE OPERATING LIMITS REPORT

The Core Operating Limits Report (COLR) has been revised for cycle 8 operation to accommodate the influence of the cycle 8 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 8 specific power distribution analysis using NRC approved methodology provided in the references to Technical Specification 6.9.1.7.

The core operating limits are based on an ECCS bounding analysis that was performed to determine the allowable LOCA linear heat rate limits for the B&W 177 fuel assembly raised-loop plant. The analysis incorporated the NUREG-0630 cladding swell and rupture model, TACO2 code, the BWC CHF correlation, and the B&W modified version of FLECSET reflooding heat transfer coefficient correlation. Figures 8-1 through 8-16 are revisions to the previous cycle operating limits contained in the COLR. Table 8-1 presents the quadrant power tilt limits for cycle 8 and Table 8-2 provides the negative moderator temperature coefficient for cycle 8. Based on the analyses and operating limit revisions described in this report, the Final Acceptance ECCS limits will not be exceeded, nor will the thermal design criteria be violated.

Figure 8-1. Regulating Group Position Limits 0 to 200±10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

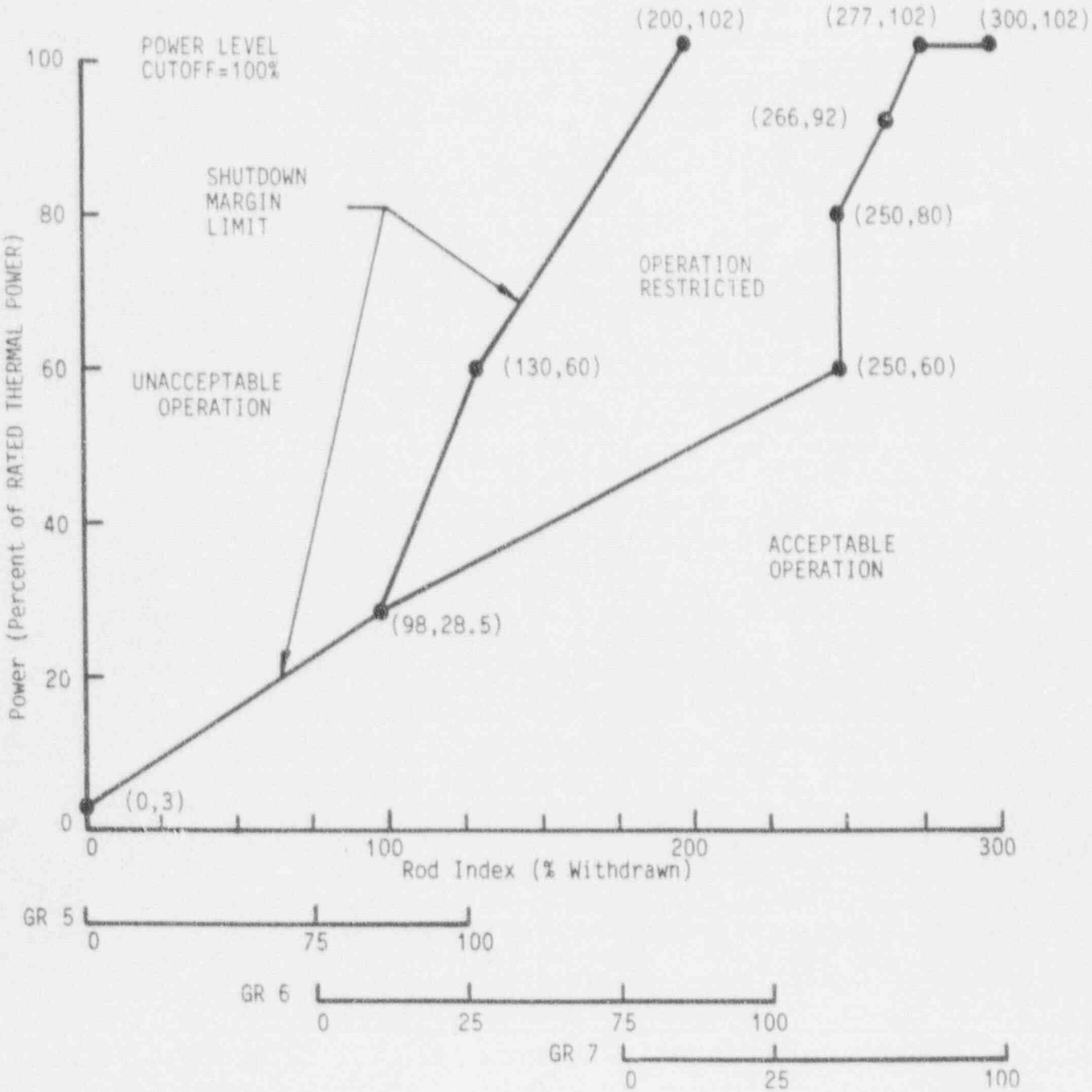


Figure 8-2. Regulating Group Position Limits, 200±10 to 400±10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

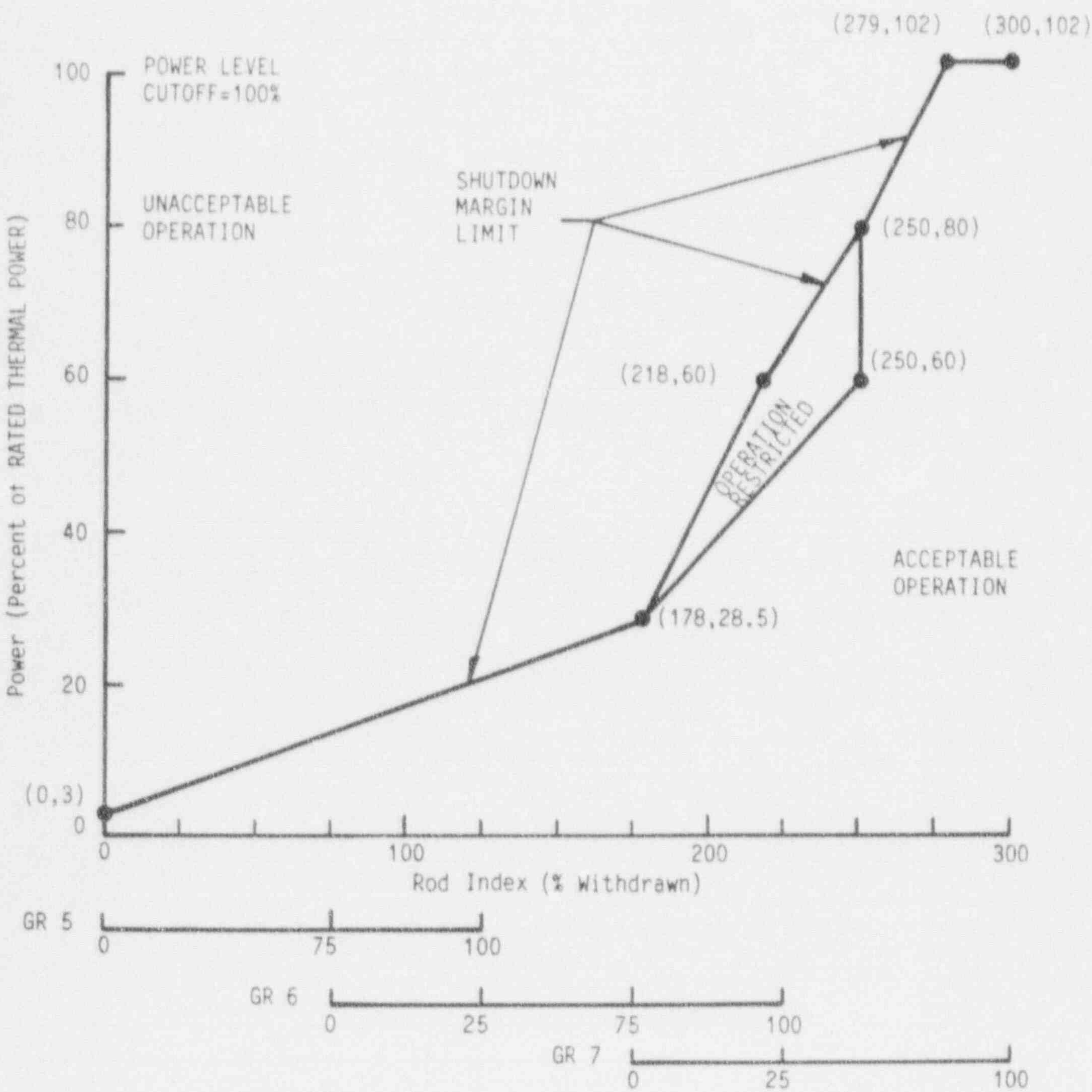


Figure 8-3. Regulating Group Position Limits After 400±10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

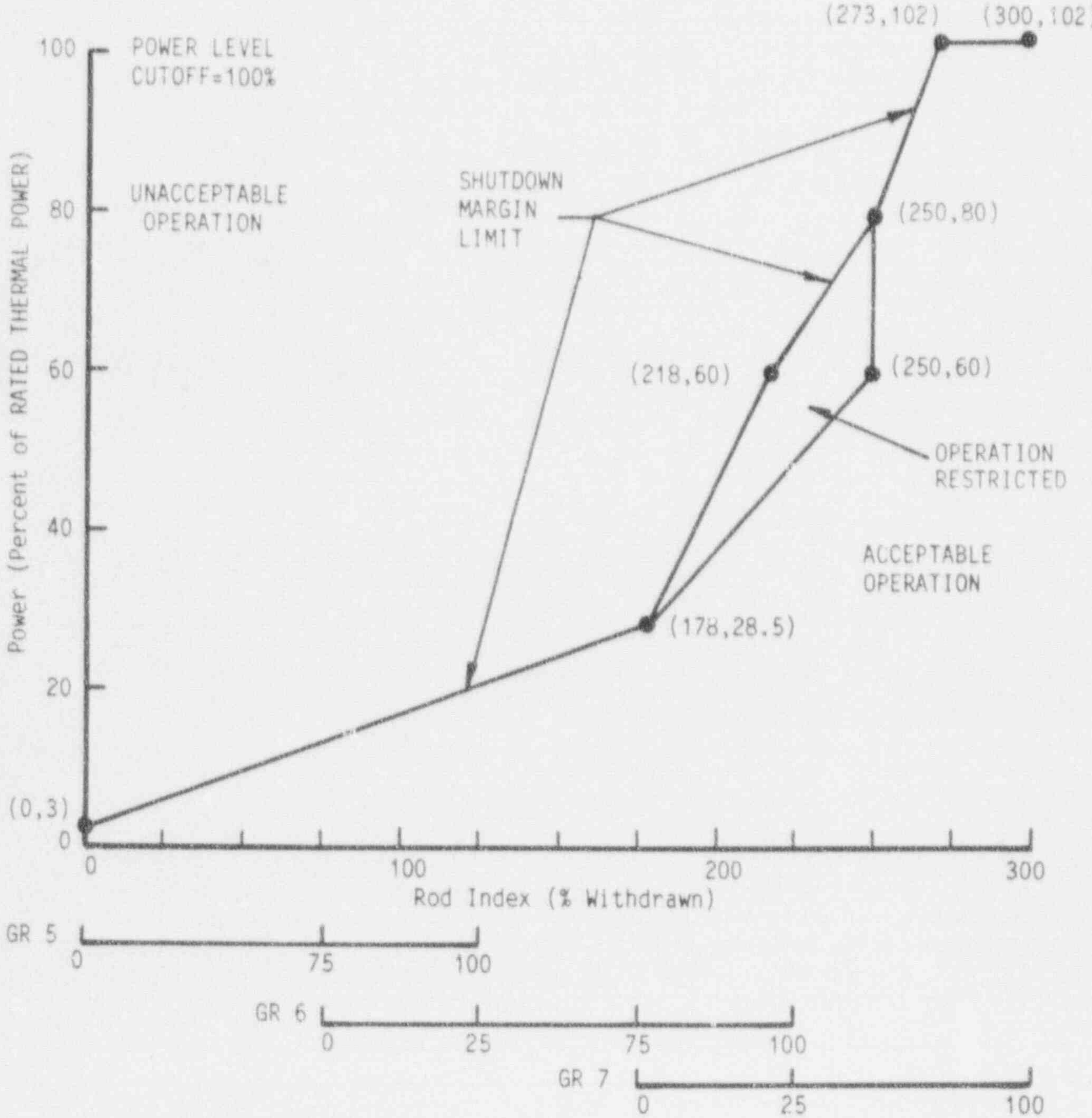


Figure 8-4. Regulating Group Position Limits,0 to 200±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 8

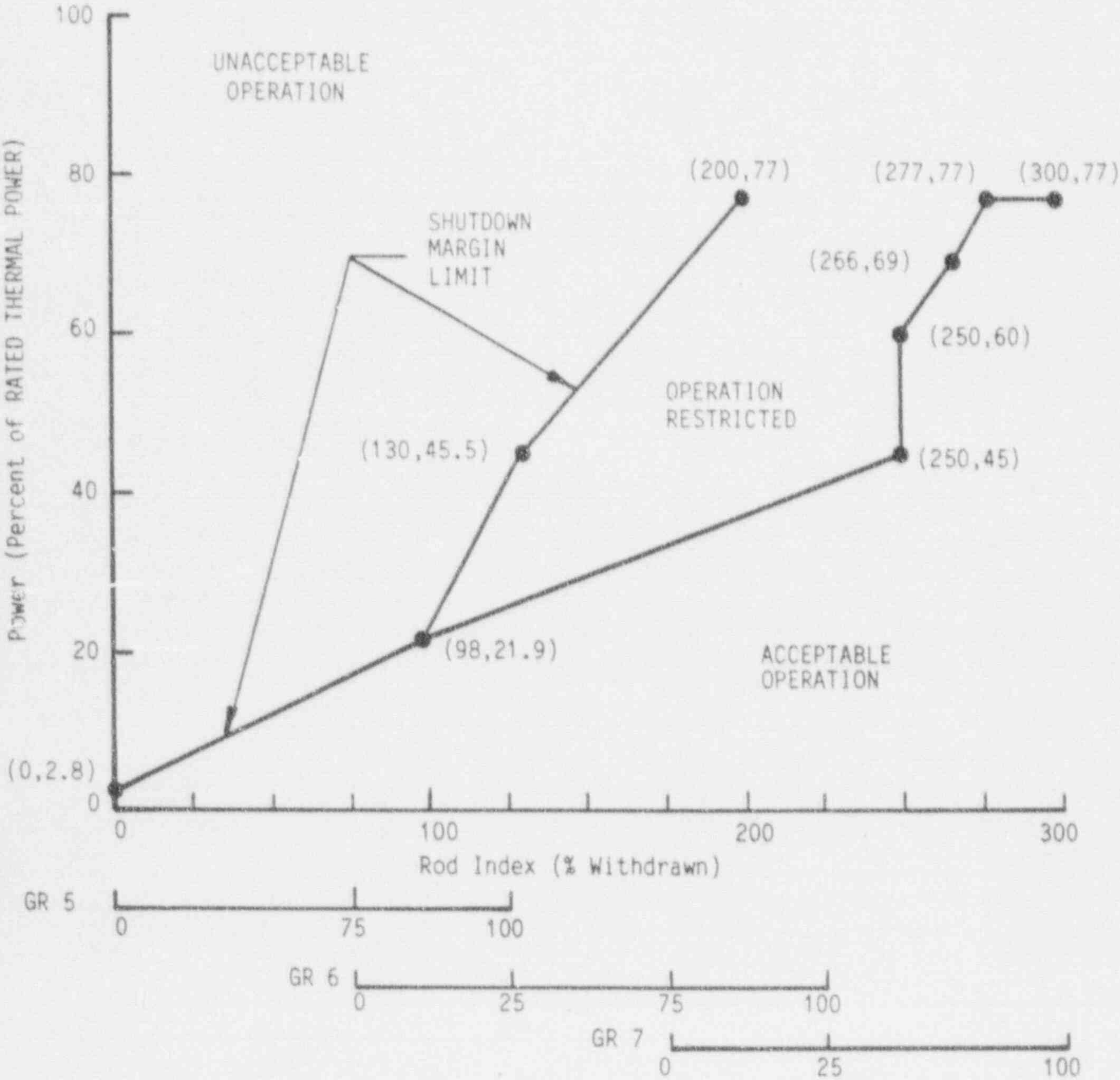
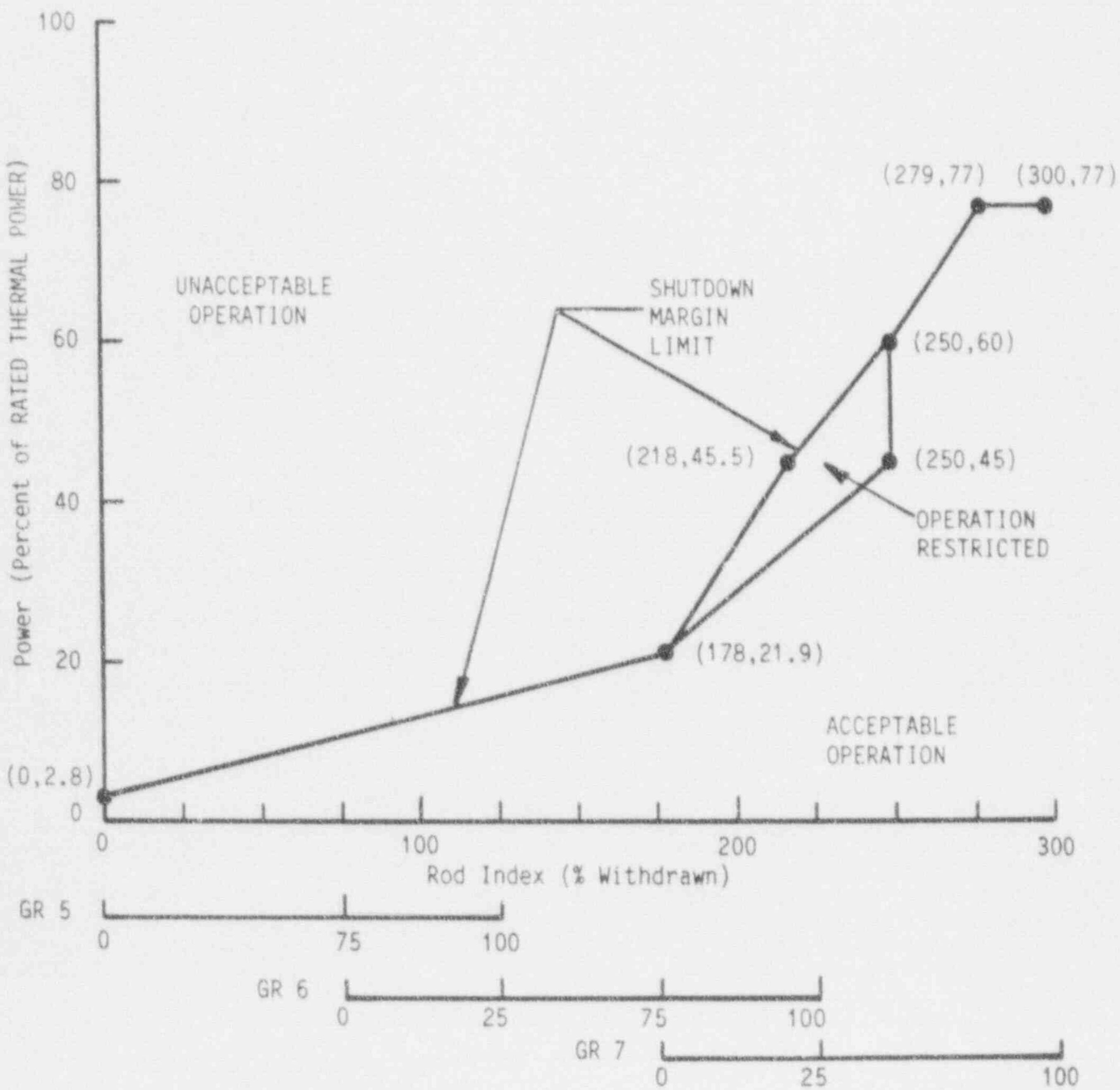


Figure 8-5. Regulating Group Position Limits, 200±10 to 400±10 EFPD
Three RC pumps -- Davis-Besse 1, Cycle 8



B&W Fuel Company

Figure 8-6. Regulating Group Position Limits, After 400±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 8

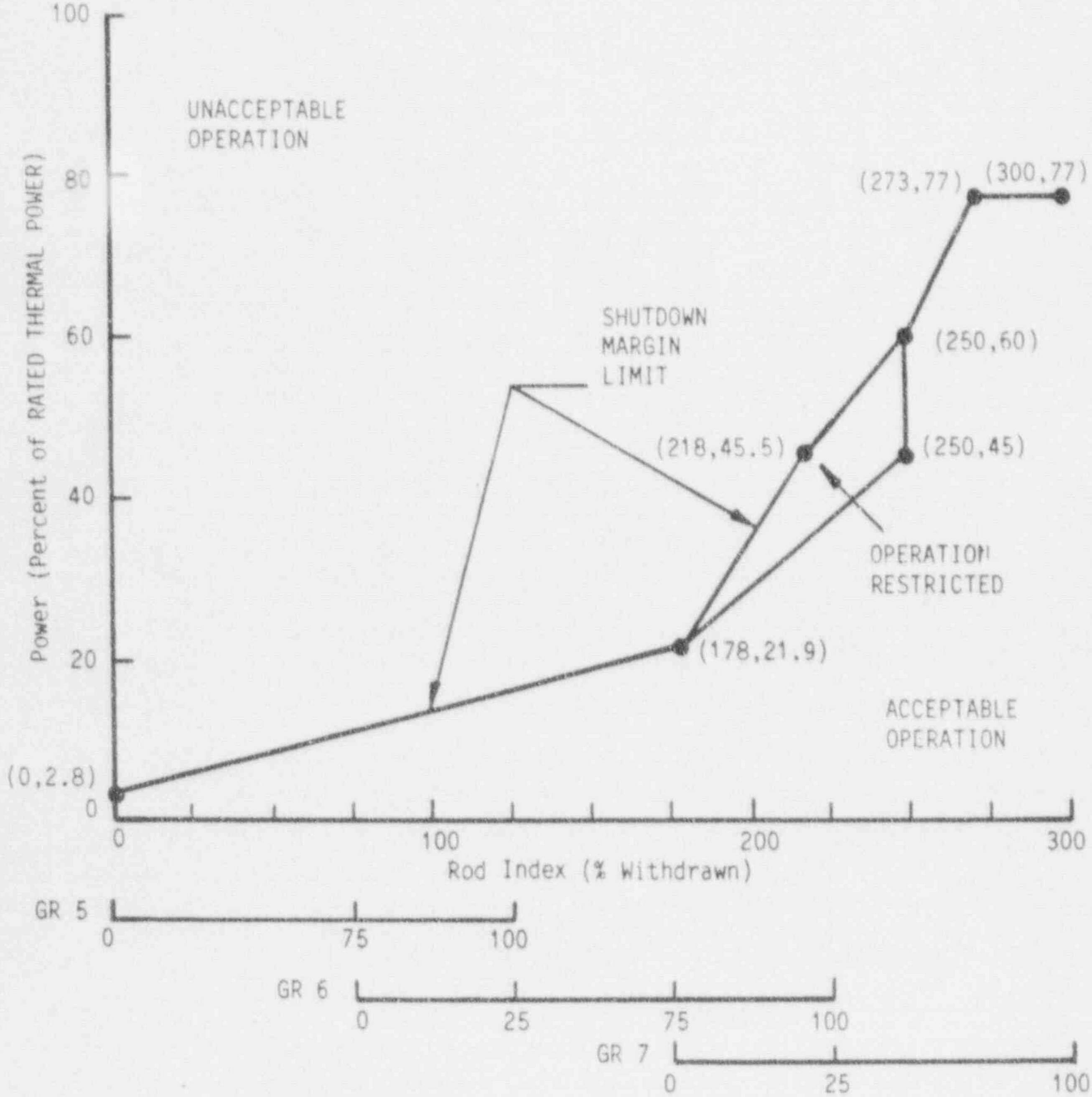
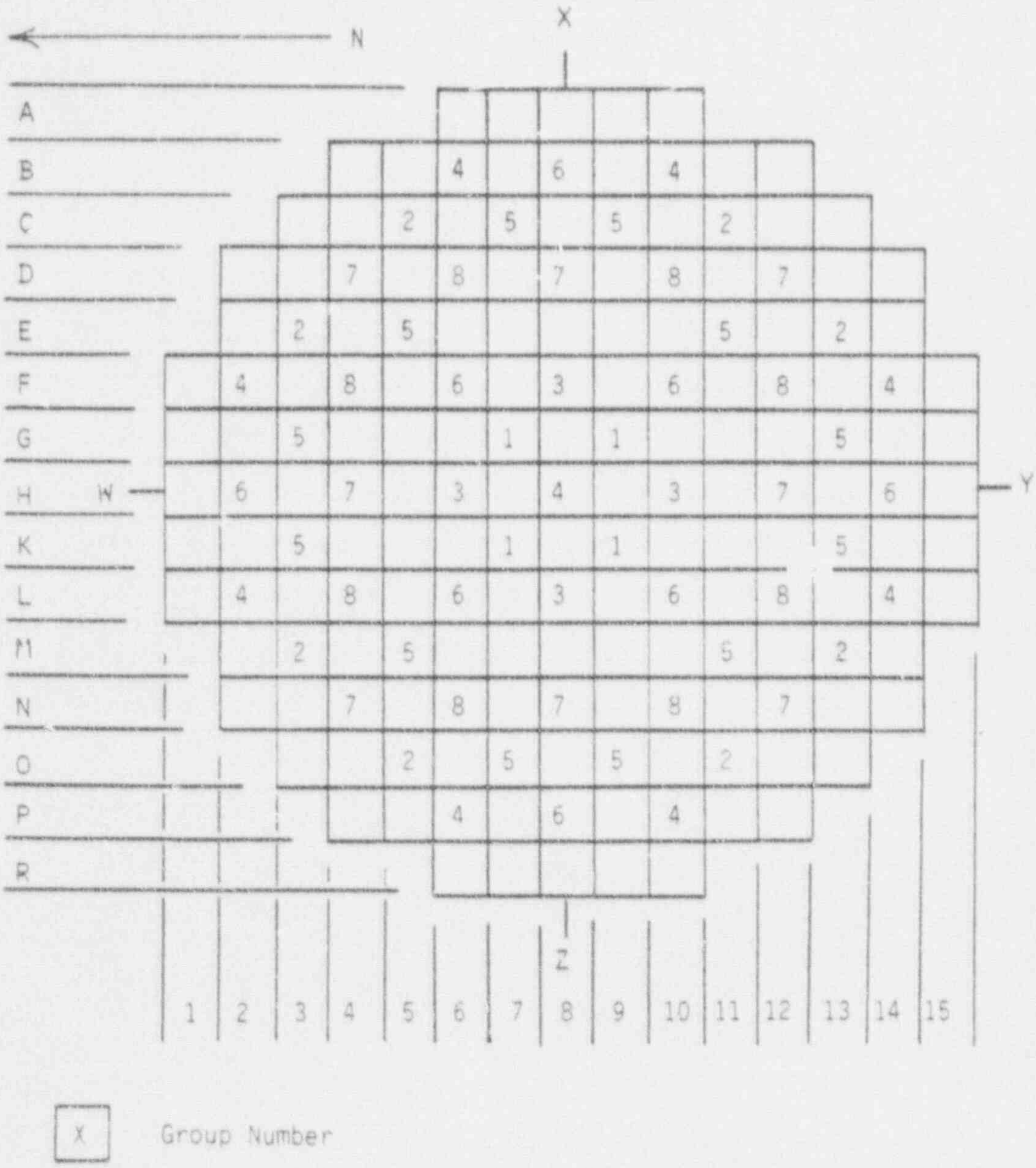


Figure 8-7. Control Rod Locations for Davis-Besse 1, Cycle 8



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 8-8. APSR Position Limits, 0 to 400 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 8

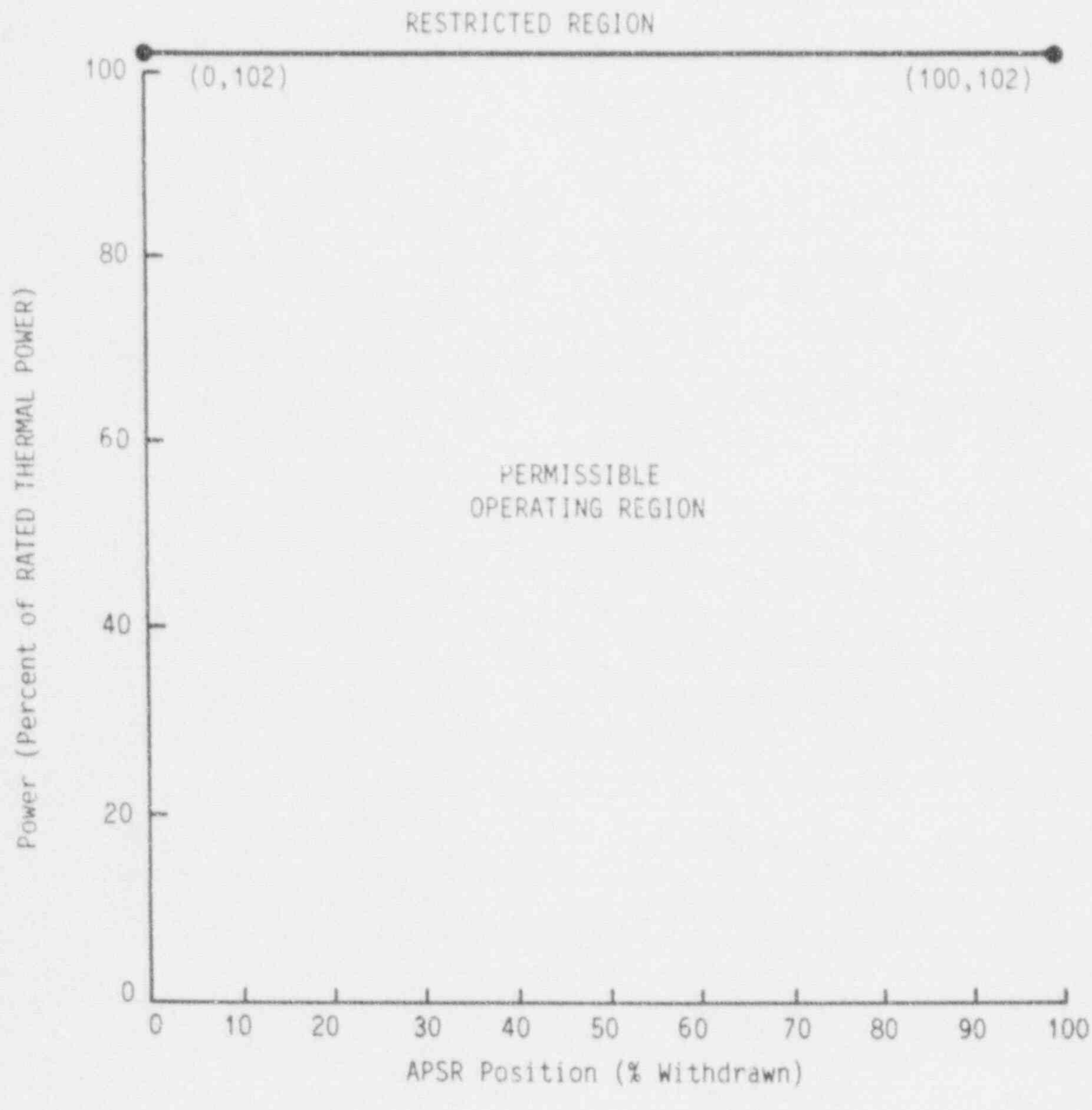


Figure 8-9. APSR Postion Limits After 400 ± 10 EFPD, Three or Four RC Pumps. APSRs Withdrawn -- Davis-Besse 1, Cycle 8

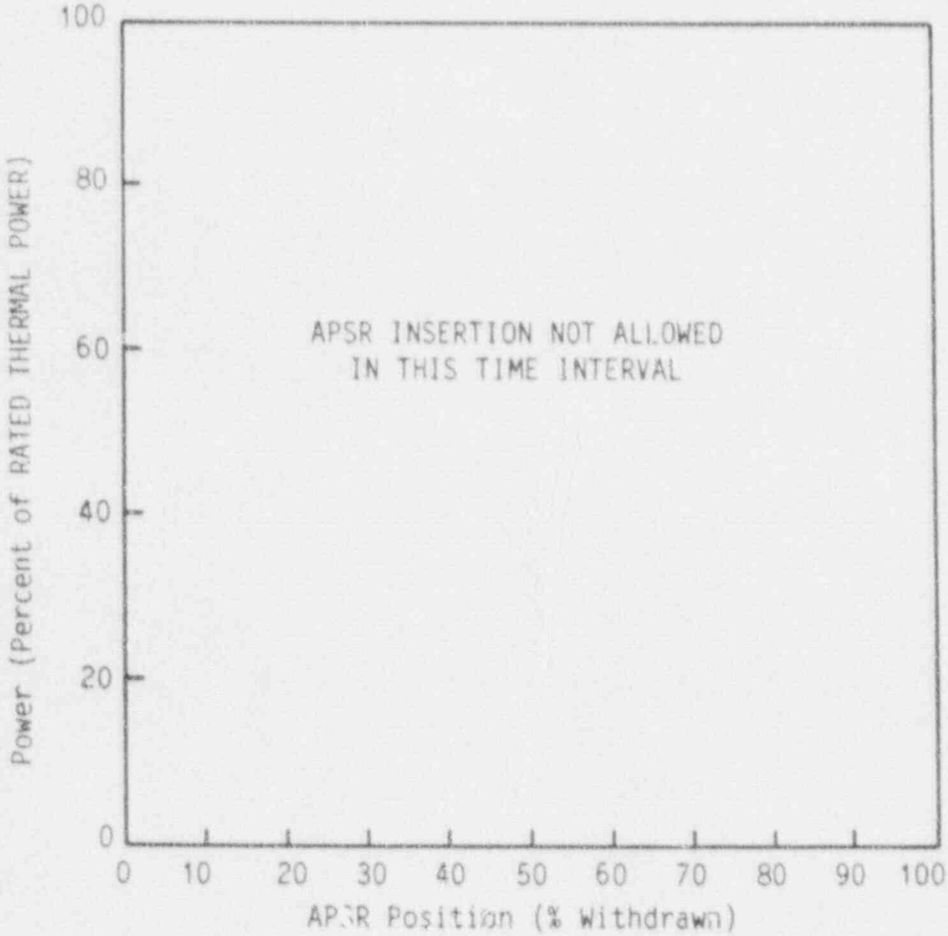


Figure 8-10. APSR Position Limits, 0 to 400±10 EFPD,
Three RC Pumps -- Davis-Besse 1, Cycle 8

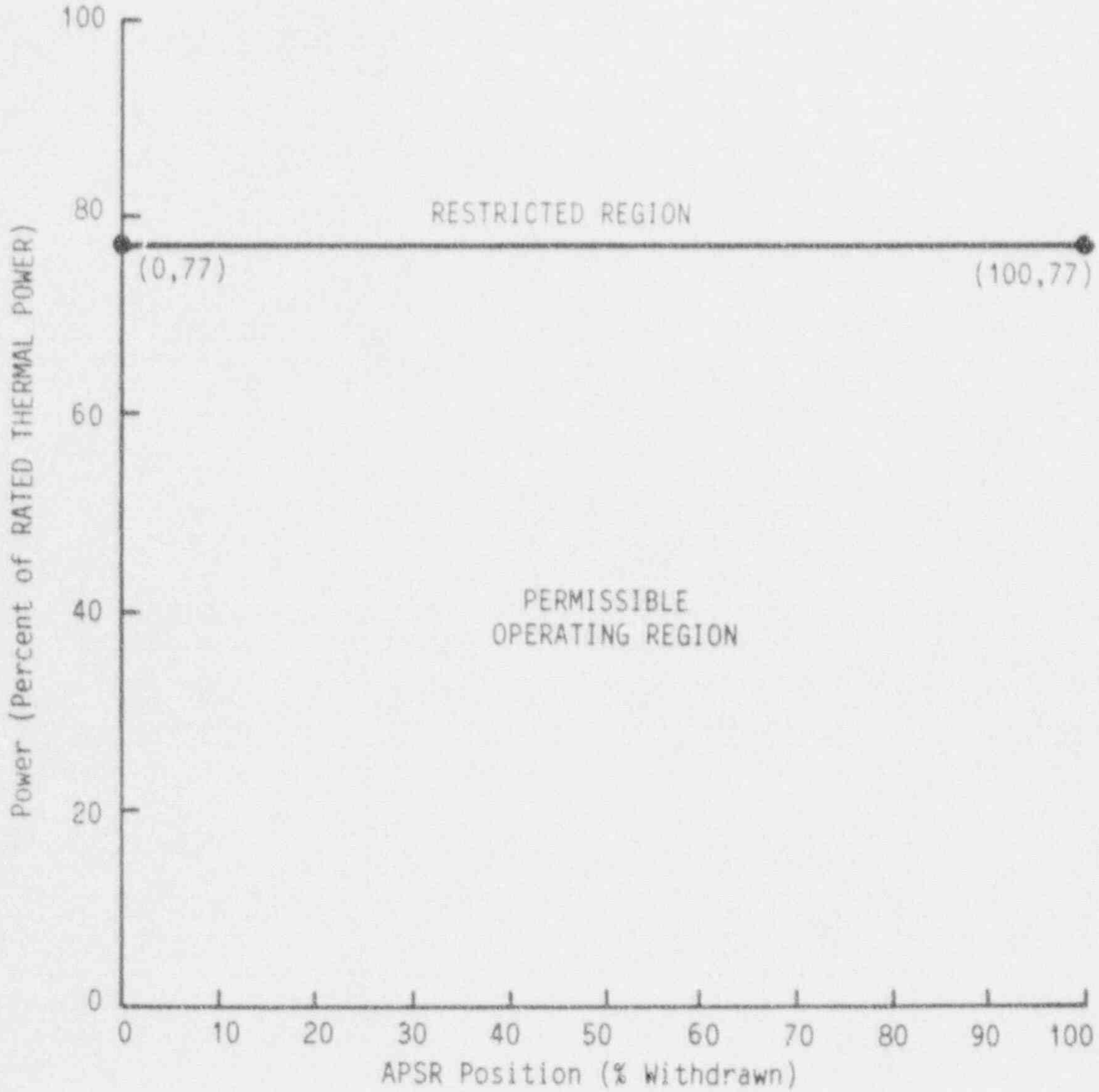


Figure 8-11. AXIAL POWER IMBALANCE Limits, 0 to 200 ± 10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

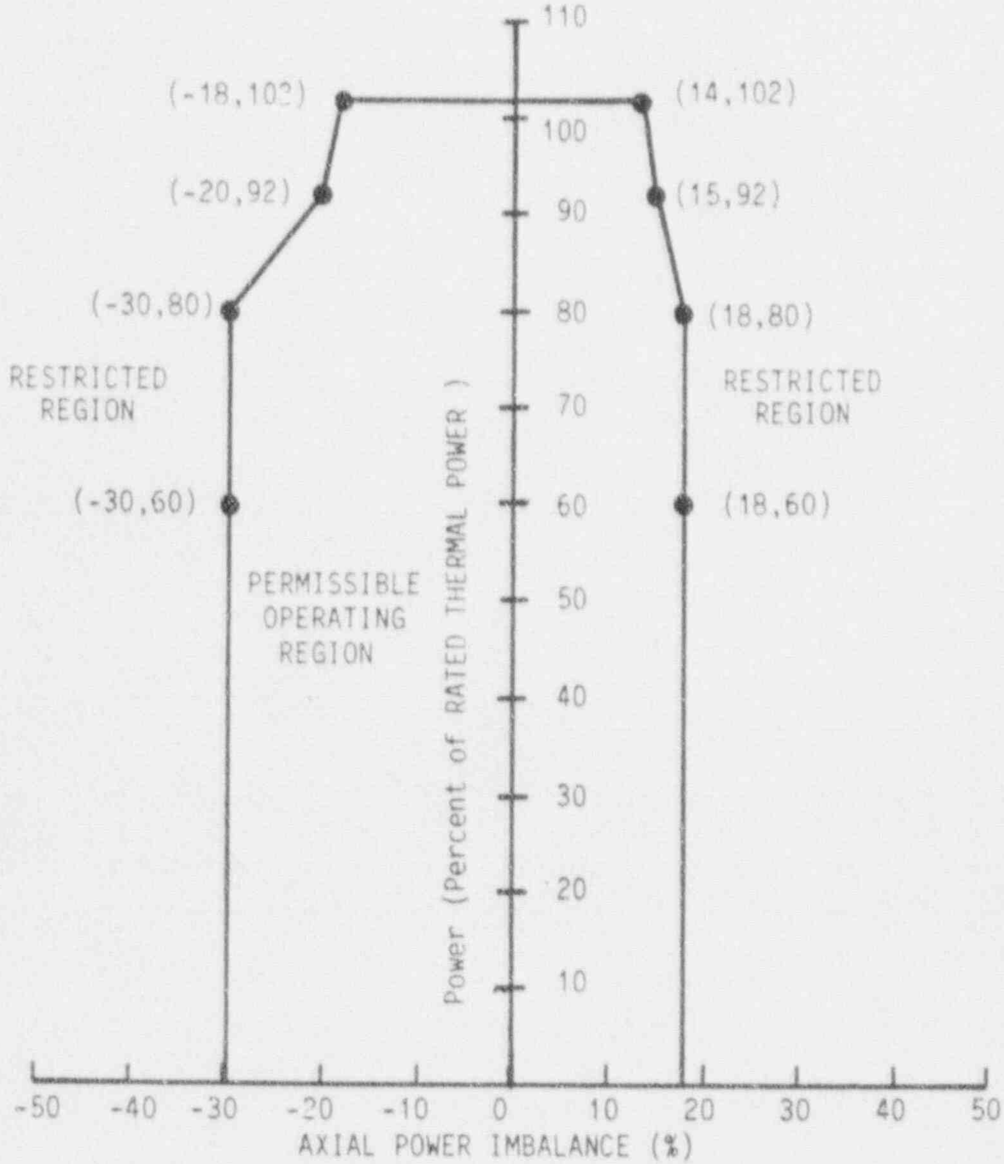


Figure 8-12. AXIAL POWER IMBALANCE Limits, 200±10 to 400±10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

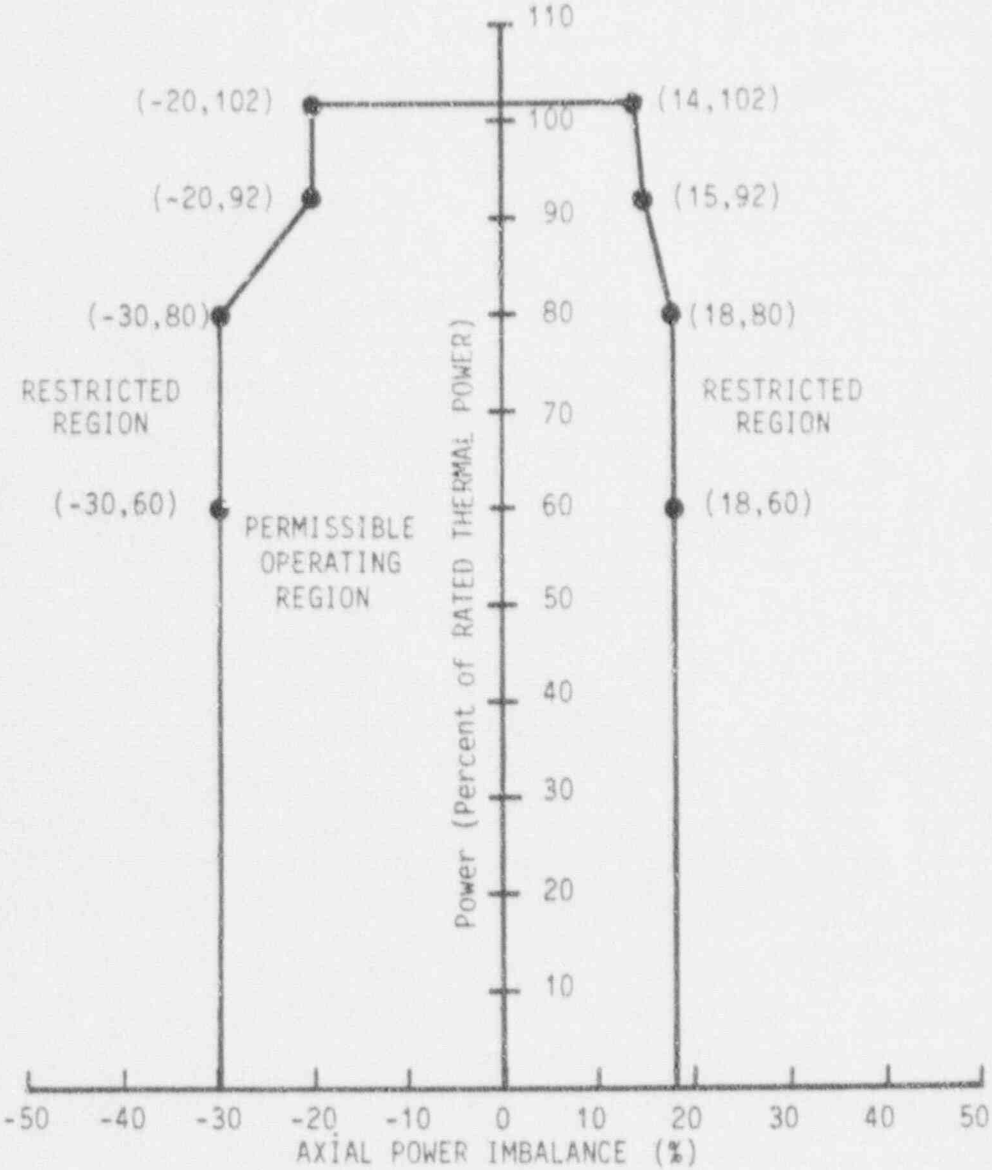


Figure 8-13. AXIAL POWER IMBALANCE Limits, After 400±10 EFPD
Four RC Pumps -- Davis-Besse 1, Cycle 8

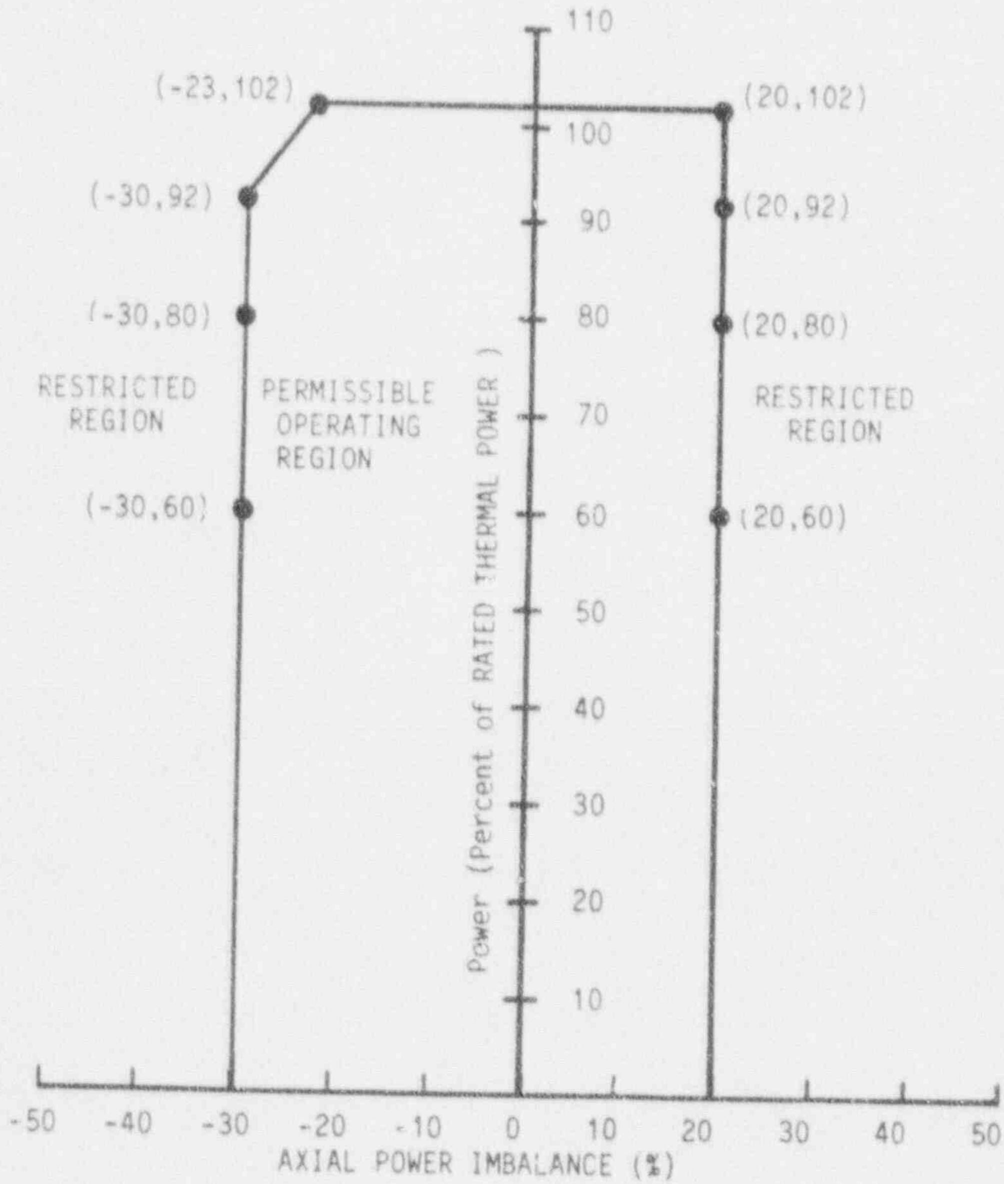


Figure 8-14. AXIAL POWER IMBALANCE Limits, 0 to 200±10 EFPD
Three RC Pumps -- Davis-Besse 1, Cycle 8

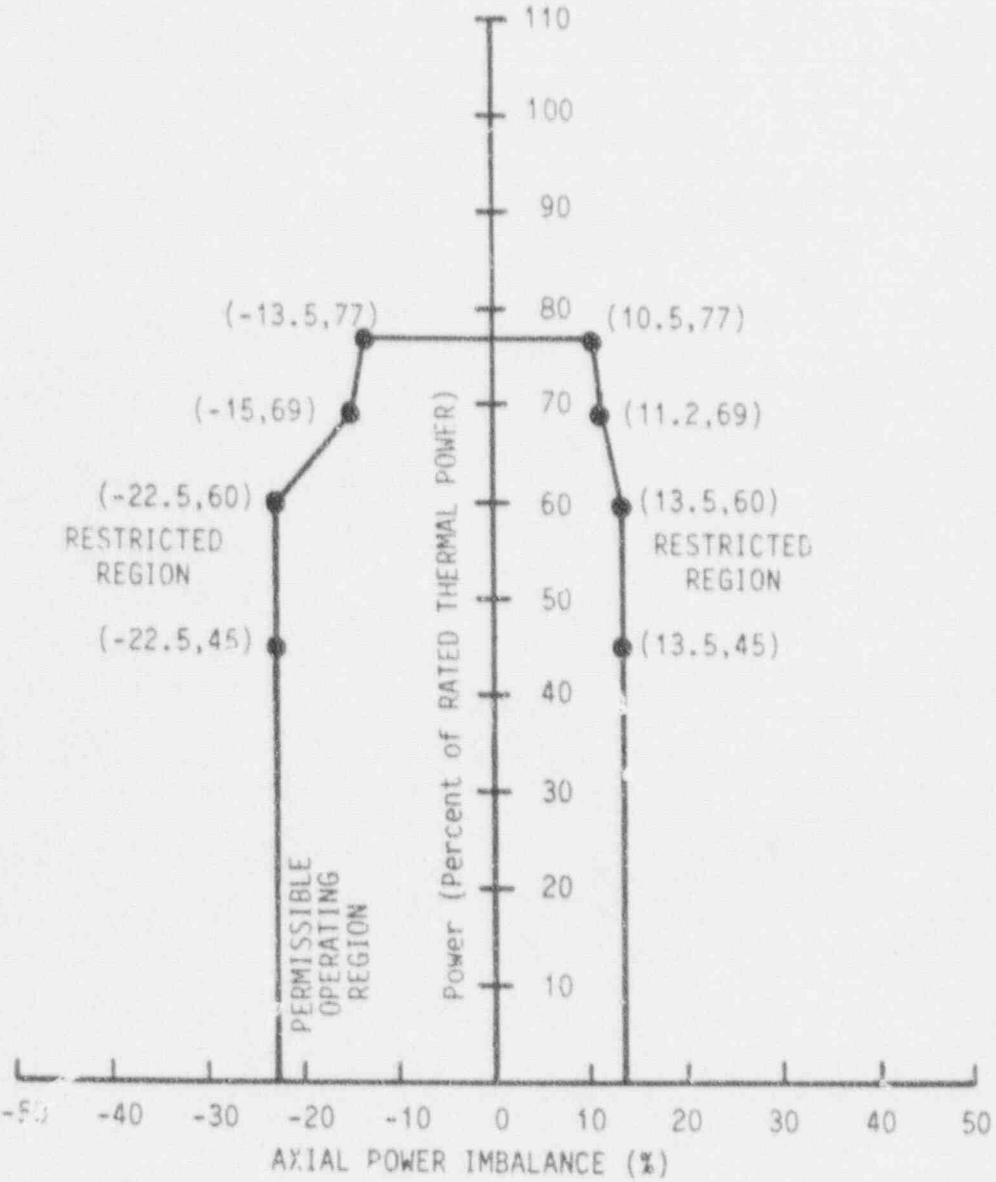


Figure 8-15. AXIAL POWER IMBALANCE Limits, 200±10 to 400±10 EFPD
Three RC Pumps -- Davis-Besse 1, Cycle 8

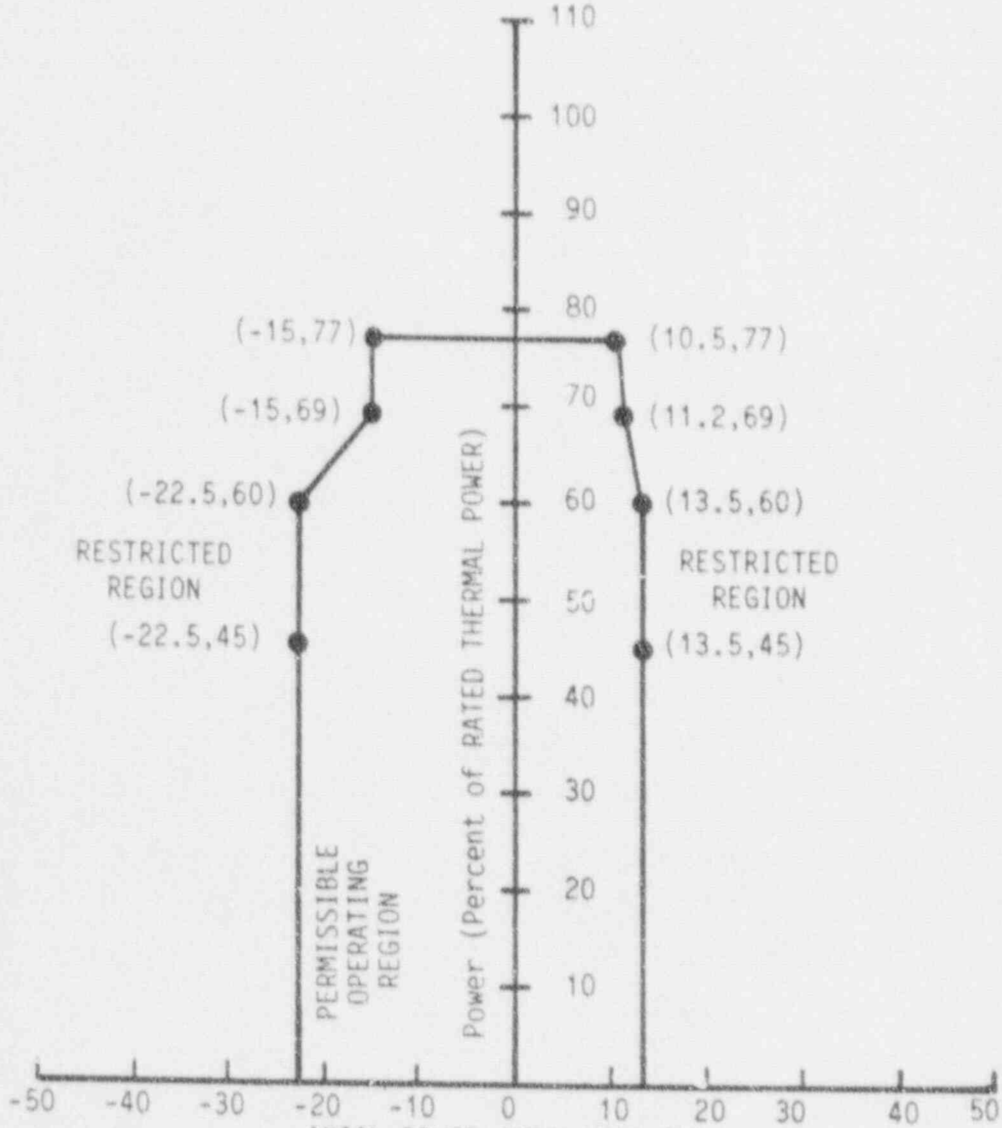


Figure 8-16. AXIAL POWER IMBALANCE Limits, After 400±10 EFPD
Three RC Pumps -- Davis-Besse 1, Cycle 8

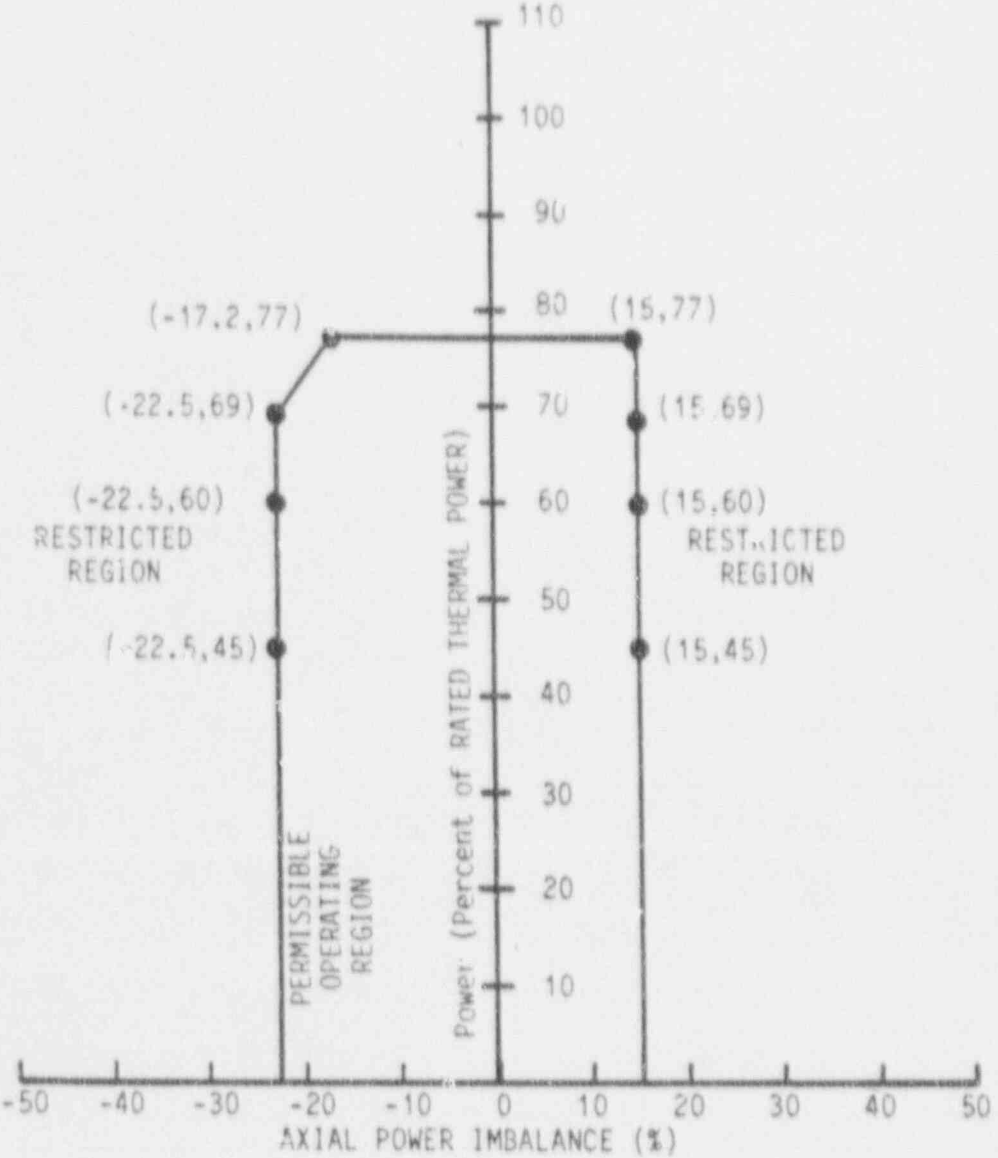


Table 8-1. Quadrant Power Tilt Limits

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	6.83	4.11	10.03	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*

*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)	-3.73×10^{-4} Δk/k/°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 con l rod drop times are recorded for all control rods at hot full-flow condi..ons before zero power physics testing begins. Acceptance criteria state that the rod drop time from fully withdrawn to 75%-inserted shall be less than 1.50 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. Acceptance criteria require that the measured flow 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 must be within ± 100 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. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm$ 0.4x10^{-2} %Δk/k/°F.

The moderator 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 coefficient. This value must not be in excess of the acceptance criteria limit of +0.9x10^{-2} %Δk/k/°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 bank group worths are as follows:

1. Individual bank 5, 6, 7 worth:

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

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

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

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. $\left|\dfrac{\text{predicted value - measured value}}{\text{measured value}}\right| \times 100 \leq 15$

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 core symmetry is acceptable if the absolute values of the quadrant power tilts are less than the error adjusted alarm 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 must be less than the LOCA limit.
2. The minimum DNBR must be greater than the initial condition DNBR limit.
3. The value obtained from extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than the calculated 112% FP DNBR value, 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 must 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 highest measured and predicted radial peaks shall be within the following limits:

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ must be more positive than } -5$$

7. The highest measured and predicted total peaks shall be within the following limits:

$$\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \text{ must be more positive than } -7.5$$

Items 1, 2, and 5 ensure that the initial condition LOCA, initial condition DNBR, and quadrant power tilt limits respectively 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 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 must be greater than 0.96. If this criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required slope.

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. Acceptance criteria state that the 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 must be more negative than the acceptance criteria limit of $-0.90 \times 10^{-3}\% \ \Delta k/k/{}^{\circ}F$.

9.4. Procedure for Use if Acceptance Criteria Not Met

If acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. 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.

10. REFERENCES

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