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April 1984

DAVIS-BESSE NUCLEAR POWER STATION

UNIT 1, CYCLE 5 — RELOAD REPORT

Babcock & Wilcox
a McDermott company

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1. INTRODUCTION AND SUMMARY

This report justifies operation of the Davis-Besse Nuclear Power Station Unit 1 at the rated core power of 2772 MWt for cycle 5. 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 documented in several reports that have been submitted to the NRC and approved by that agency.

Cycle 5 reactor and fuel parameters related to power capability are summarized in this report and compared to cycle 4. All accidents analyzed in the Davis-Besse Final Safety Analysis Report[1] (FSAR) have been reviewed for cycle 5 operation, and in cases where cycle 5 characteristics were conservative compared to cycle 1, no new analyses were performed.

Retainers[2] and neutron sources will remain in the core. The effects on continued operation without orifice rod assemblies (ORAs) and with the retainers have been accounted for in the analysis performed for cycle 5.

The Technical Specifications have been reviewed and modified where required for cycle 5 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 5 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 the currently operating cycle 4, which achieved criticality on September 27, 1983. Power escalation began on September 29, 1983 and full power (2772 MWt) was reached on November 2, 1983. During cycle 4 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 5. The scheduled durations of cycles 4 and 5 are 280 and 390 effective full power days (EFPD), respectively.

The APSRs were pulled at 200 EFPD to increase the lifetime of cycle 4. The APSR pull coupled with a power coastdown resulted in a potential cycle 4 length of approximately 280 EFPD. The cycle 5 design also includes an APSR pull and power coastdown.

The cycle 5 design minimizes the number of fuel assemblies that are cross core shuffled to reduce the potential for quadrant tilt amplification. The cycle 5 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 FSAR[1] for the unit. The cycle 5 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. All FAs in batches 4, 5, 6, and 7 have a constant nominal fuel loading of 468.25 kg of uranium. The batch 1E assembly has a fuel loading of 472.24 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 Tables 4-1 and 4-2 of this report.

Figure 3-1 is the core loading diagram for Davis-Besse Unit 1, cycle 5. Seventeen batch 1D assemblies, 20 batch 2B assemblies, and 28 batch 4A assemblies will be discharged at the end of cycle 4. The fuel assemblies in batches 4B, 5A, 5B, and 6 will be shuffled to their cycle 5 locations. Batches 4B and 5A have an initial uranium-235 enrichment of 3.04 wt %. Batches 5B and 6 have an initial enrichment of 2.99 wt %. One batch 1E assembly with an initial enrichment of 1.98 wt % will be reinserted in cycle 5. A feed batch consisting of 64 batch 7 assemblies with uranium enrichment of 3.19 wt % will be inserted in the core interior. Batch 6 will occupy the core periphery. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 5 and its initial enrichment.

Cycle 5 is operated in a feed-and-bleed mode. The core reactivity is controlled by 53 full-length Ag-In-Cd control rod assemblies (CRAs), 64 burnable poison rod assemblies (BPRAs), and soluble boron. In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle

5 locations of the 61 control rods and the group designations are indicated in Figure 3-3. The core locations of 61 control rods for cycle 5 are identical to those of reference cycle 4. However, the cycle 5 rod group designations differ from the cycle 4 designations. The cycle 5 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Davis-Besse Cycle 5 Full Core Loading Diagram

FUEL TRANSFER CANAL ⟶

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						6 M 2	6 K 1	6 R 8	6 K15	6 M14					
B				6 L 2	6 L 1	6 N 2	7 F	5B C 9	7 F	6 N14	6 L15	6 L14			
C			6 O13	7 F	5A E 7	7 F	5B C 4	7 F	5B C12	7 F	5A E 9	7 F	6 O 3		
D		6 B10	7 F	5B G 7	7 F	4B M 4	7 F	5B B 8	7 F	4B M12	7 F	5B G 9	7 F	6 B 6	
E		6 A10	5A G 5	7 F	5B E 5	7 F	4B L 5	7 F	4B L11	7 F	5B E11	7 F	5A G11	6 A 6	
F	6 B11	6 B12	7 F	4B D11	7 F	5B C 7	7 F	5B D 8	7 F	5B G13	7 F	4B [illegible] 5	7 F	6 B 4	6 B 5
G	6 A 9	7 F	B5 D 3	7 F	4B E10	7 F	5B D 4	5B L 8	5B D12	7 F	4B E 6	7 F	5B D13	7 F	6 A 7
W-H	6 H15	5B G 3	7 F	5B H 2	7 F	5B H 4	5B H10	1E F6 Cy1	5B H 6	5B H12	7 F	5B H14	7 F	5B K13	6 H 1
K	6 R 9	7 F	5B N 3	7 F	4B M10	7 F	5B N 4	5B F 8	5B N12	7 F	4B M 6	7 F	5B N13	7 F	6 R 7
L	6 P11	6 P12	7 F	4B N11	7 F	5B K 3	7 F	5B N 8	7 F	5B O 9	7 F	4B N 5	7 F	6 P 4	6 P 5
M		6 R10	5A K 5	7 F	5B M 5	7 F	4B F 5	7 F	4B F11	7 F	5B M11	7 F	5A K11	6 R 6	
N		6 P10	7 F	5B K 7	7 F	4B E 4	7 F	5B P 8	7 F	4B E12	7 F	5B K 9	7 F	6 P 6	
O			6 C13	7 F	5A M 7	7 F	5B O 4	7 F	5B O12	7 F	5A M 9	7 F	6 C 3		
P				6 F 2	6 F 1	6 D 2	7 F	5B O 7	7 F	6 D14	6 F15	6 F14			
R						6 E 2	6 G 1	6 A 8	6 G15	6 E14					

X (top, column 8) — Y (right, row W-H) — Z (bottom, column 8)

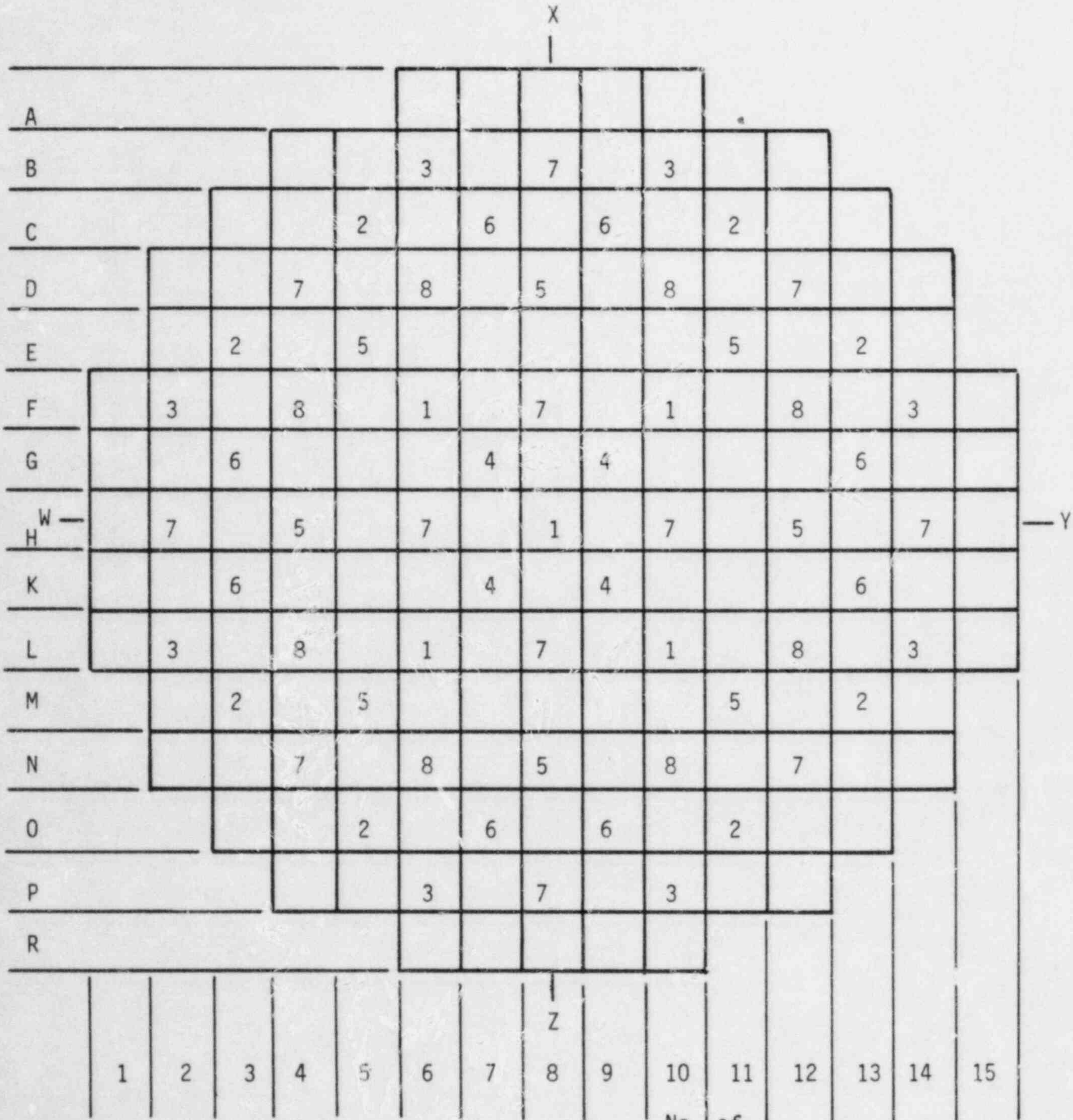
Legend (box):
Batch
Cycle 4 Location
Cy 1 = reinserted from cycle 1

Figure 3-2. Enrichment and Burnup Distribution for Davis-Besse 1, Cycle 5

	8	9	10	11	12	13	14	15
H	1.98	2.99	2.99	3.19	2.99	3.19	2.99	2.99
	12644	17710	22063	0	18631	0	20249	8600
K	2.99	2.99	3.19	3.04	3.19	2.99	3.19	2.99
	17738	18518	0	25725	0	18235	0	8250
L	2.99	3.19	2.99	3.19	3.04	3.19	2.99	2.99
	22055	0	20211	0	26978	0	6450	8993
M	3.19	3.04	3.19	2.99	3.19	3.04	2.99	
	0	25676	0	18367	0	17637	6904	
N	2.99	3.19	3.04	3.19	2.99	3.19	2.99	
	18631	0	26956	0	17133	0	10719	
O	3.19	2.99	3.19	3.04	3.19	2.99		
	0	18252	0	17622	0	7216		
P	2.99	3.19	2.99	2.99	2.99			
	20249	0	6460	6905	10720			
R	2.99	2.99	2.99					
	8602	8251	9000					

x.xx	Initial Enrichment
xx,xxx	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Davis-Besse 1, Cycle 5



X Group Number

Group	No. of rods	Functions
1	5	Safety
2	8	Safety
3	8	Safety
4	4	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total#	61	

Figure 3-4. Davis-Besse Cycle 5 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				0.8		1.1		
K			0.8		1.1		0.5	
L		0.8		0.8		0.5		
M	0.8		0.8		0.8			
N		1.1		0.8		0.2		
O	1.1		0.5		0.2			
P		0.5						
R								

X.X	BPRA Concentration, wt% B_4C in Al_2O_3.

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies (FAs) and pertinent fuel parameters for Davis-Besse 1, cycle 5 are listed in Table 4-1. The Batch 7 FAs are the MK-B5 design, while the other reinserted FAs are the Mk-B4 design. The Mk-B5 FAs are identical in concept to the Mk-B4 with only a change to the upper end fitting design which eliminates retainers for burnable poison rod assembly (BPRA) holddown. Retainer assemblies will be used on two FAs (Mk-B4 design) that contain the regenerative neutron sources. The justification for the design and use of the retainer assemblies is described in references 2 and 3.

Sixty-four BPRAs will be used with the sixty-four Batch 7 FAs. The retention mechanism for the Mk-B5 design BPRAs is built into each assembly and retainer assemblies are not required.

The Mk-B5 upper end fitting (Figure 4-1) provides four open slots that align and guide the movement of the holddown spring, spring retainer (Figure 4-2), and the new Mk-B5 BPRA spider (Figure 4-3). The holddown spring is preloaded through stop pins welded to ears on each side of the upper end fitting. Incore, as shown in Figure 4-4, the BPRA spider feet are captured between the holddown spring retainer and the upper grid pads on the reactor internals. This arrangement retains the fixed control components at all design flow conditions. The Mk-B5 design also contains a redesigned holddown spring made from Inconel 718 material which provides added margin over the Mk-B4 spring design. The Mk-B5 upper end fitting has been tested extensively, both in air and in over 1000 hours of simulated reactor environment, to determine analytical input and to assure good incore performance.

4.2. Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1. Cladding Collapse

The power histories were reviewed for each fuel assembly in cycle 5. For each of the six batches the most limiting assembly power history was determined. The most limiting assembly is the one with the highest burnup. These six power histories were compared to a generic analysis, or were used as input to a creep collapse analysis to ensure that creep ovalization will not affect the fuel performance during Davis-Besse 1 cycle 5. Both specific and generic creep ovalization analyses are based on reference 4 and are applicable to the designs for batches 1E, 4B, 5A, 5B, 6, and 7.

Creep collapse analyses predict collapse times longer than 35,000 EFPH. The longest incore exposure time for cycle 5 is 29,249 EFPH for batch 4B (Table 4-1).

4.2.2. Cladding Stress

The Davis-Besse Unit 1, cycle 5 stress parameters are enveloped by a conservative fuel rod stress analysis. The methods used for the analysis of cycle 5 have been used in the previous cycles.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic tensile circumferential strain. The 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 the Davis-Besse Unit 1, cycle 5 fuel is expected to see. The strain analysis is also based on the upper tolerance values for the fuel pellet diameter and density, and the lower tolerance for the cladding inside diameter (ID).

4.3. Thermal Design

All fuel in the cycle 5 core is thermally similar. The analyses for the incoming batch 7 fuel and for the reinserted fuel from batches 5B and 6 have been performed with the TACO2[5] code using the analysis methodology described in reference 6. This methodology uses nominal undensified input parameters provided in Table 4-2. Densification effects are accounted for in the TACO2 densification model. Reinserted fuel from batches 1E, 4B, and 5A was evaluated using TAFY3[7] analyses performed for prior cycles.

The thermal design evaluation for the cycle 5 core is summarized in Table 4-2. Linear heat rate (LHR) capabilities are based on centerline fuel melt (CFM) with core protection limits based on a 20.4 kW/ft LHR to CFM. The TACO2 analyses performed for batches 5B, 6, and 7 demonstrate that 20.5 kW/ft is the CFM limit for this fuel. Using TAFY3, the fuel internal pressure has been evaluated for the highest burnup fuel rod and is predicted to be less than the nominal reactor coolant system pressure of 2200 psia. The maximum burnup of any fuel rod during cycle 5 is less than 45,000 MWd/mtU.

4.4. Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for batch 7 FAs is identical to that of present fuel.

4.5. Operating Experience

Operating experience with the Mark-B 15x15 FA has verified the adequacy of its design. As of January 15, 1984, the following experience has been accumulated for eight Babcock & Wilcox (B&W) 177-FA plants using the Mark-B FA:

Reactor	Current cycle	Max FA burnup,[a] MWd/mtU		Cumulative net electric output,[b] MWh
		Incore	Discharged	
Oconee 1	8	31,629	50,598	47,528,595
Oconee 2	7	24,221	36,800	42,405,541
Oconee 3	7	33,943	35,463	43,931,451
Three Mile Island 1	5	25,000	32,400	23,840,053
Arkansas Nuclear Unit 1	6	28,914	36,540	37,642,004
Rancho Seco	6	28,195	38,268	32,926,908
Crystal River 3	5	20,350	29,900	25,926,908
Davis-Besse	4	25,364	32,790	18,515,778

[a] As of January 15, 1984.

[b] As of November 30, 1983.

Table 4-1. Fuel Design Parameters

	Batch					
	1E	4B	5A	5B	6	7
FA type	Mk-B4	Mk-B4	Mk-B4	Mk-B4	Mk-B4	Mk-B5
Number of assemblies	1	16	8	40	48	64
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377	0.377
Flexible spacer type	Spring	Spring	Spring	Spring	Spring	Spring
Rigid spacer type	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Undensified active fuel length, in.	143.5	143.44	143.44	143.20	143.20	143.20
Fuel pellet (mean) dia., in.	0.3675	0.3697	0.3697	0.3686	0.3686	0.3686
Fuel pellet initial density, % TD mean	96	94	94	95	95	95
Initial fuel enrichment, wt % ^{235}U	1.98	3.04	3.04	2.99	2.99	3.19
Estimated residence time, EFPH	18,336	29,249	22,145	22,145	15,600	9,360
Cladding collapse time, EFPH	>35,000	>35,000	>35,000	>35,000	>35,000	>35,000

Table 4-2. Fuel Thermal Analysis Parameters

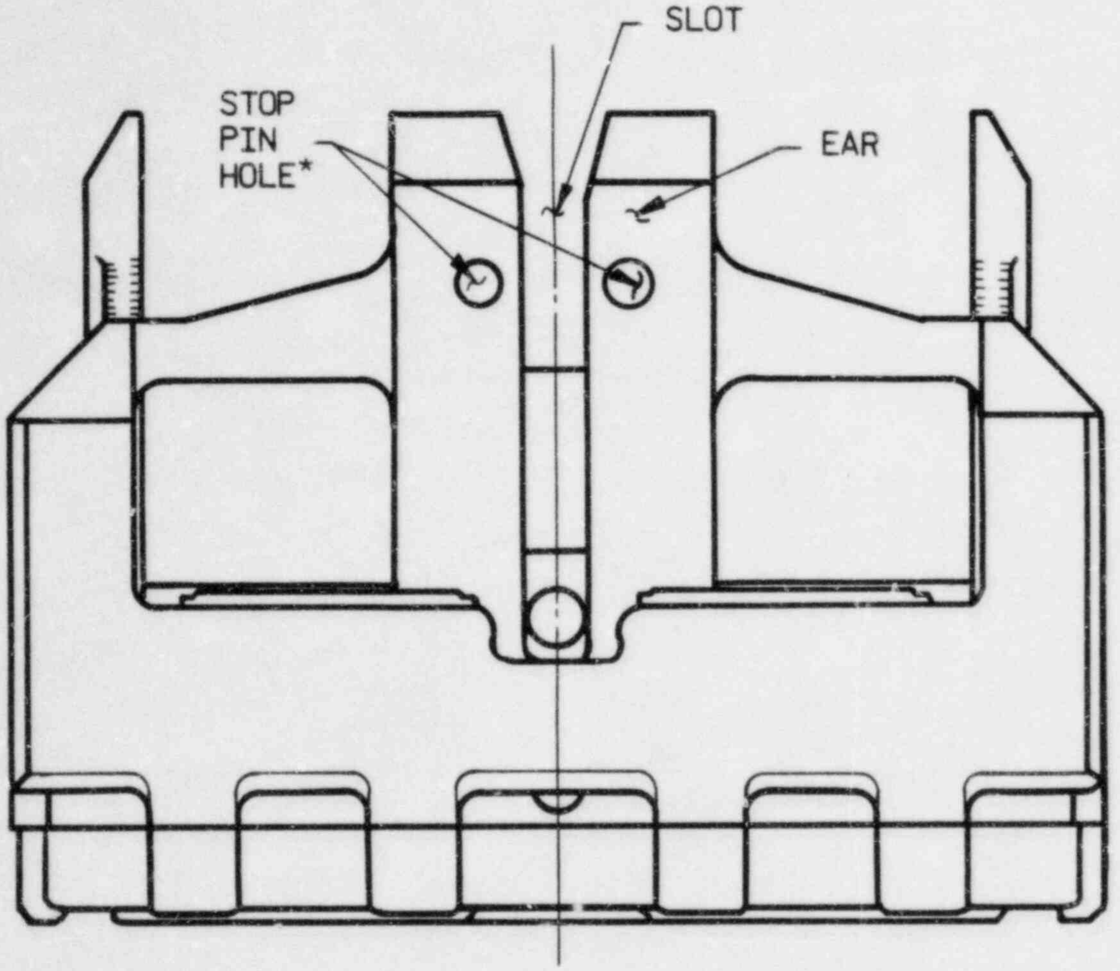
	Batch			
	1E	4B/5A	5B/6	7
Number of assemblies	1	16/8	40/48	64
Initial density, % TD	96	94	95	95
Pellet diameter, in.	0.3675	0.3697	0.3686	0.3686
Nominal stack height, in.	143.5	143.44	143.2	143.2
Enrichment, wt % ^{235}U	1.98	3.04	2.99	3.19
LHR capability, kW/ft to CFM	20.4	20.4	20.5	20.5
Densified fuel parameters[(a)] TAFY3 Code Analysis Only				
Pellet diameter, in.	0.3651	0.3648	0.3649[(b)]	0.3649[(b)]
Fuel stack height, in.	143.14	141.65	142.13	142.13
Average fuel temperature, °F	1340	1355	1464[(c)]	1464[(c)]
Nominal LHR, kW/ft at 2772 MWt	6.14	6.21	6.19	6.19

[(a)]Densification to 96.5% TD assumed for TAFY3 analysis.

[(b)]This data is provided for comparative purposes only and does not represent parameter values used in TACO2 analyses.

[(c)]BOL, TACO2 code.

Figure 4-1. Mark B5 Upper End Fitting (Side View)



*There are two stop pin holes on each side of the upper end fitting. One contains a stop pin and the other is a spare.

Figure 4-2. Holddown Spring Retainer

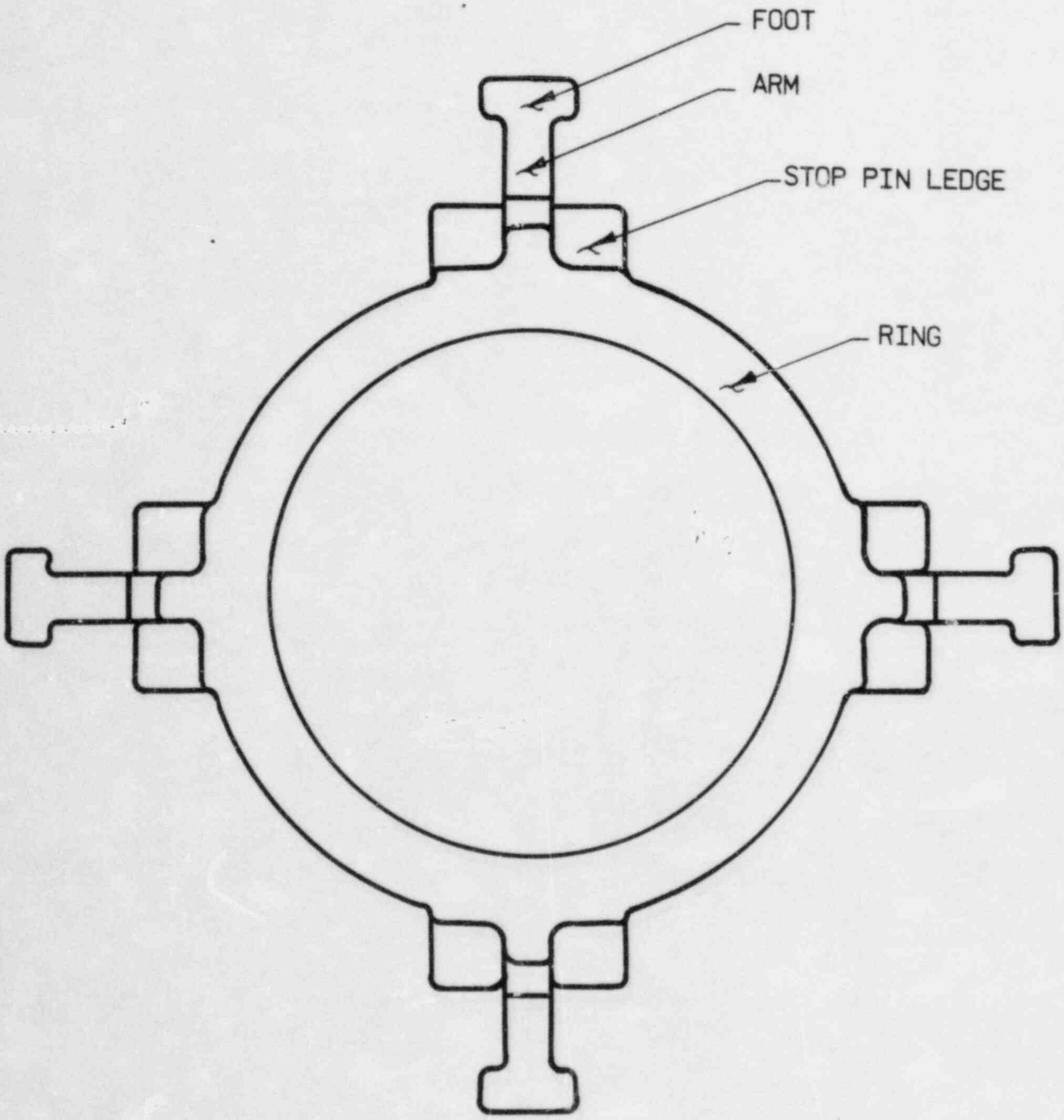


Figure 4-3. Mark B5 BPRA Spider

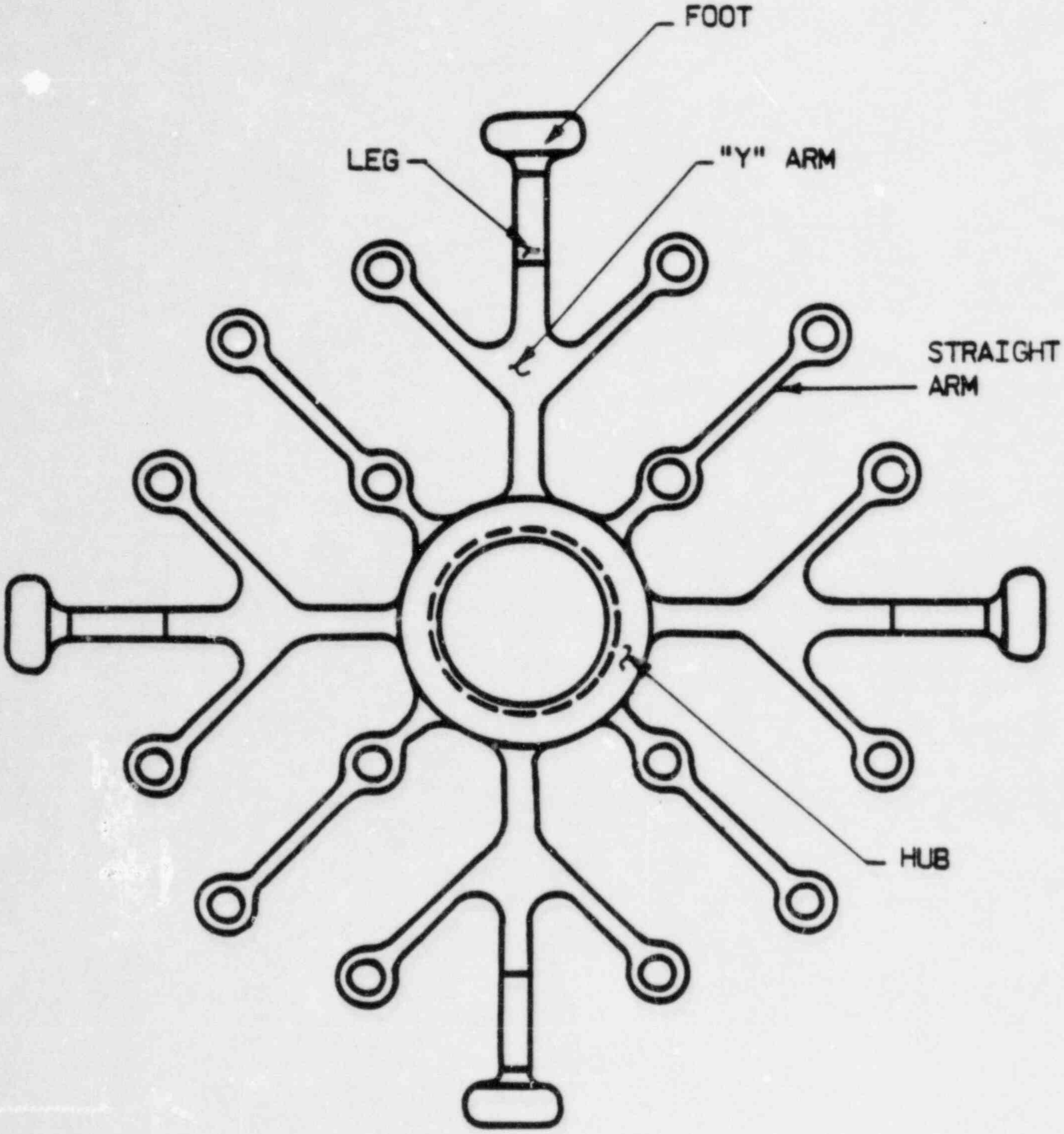
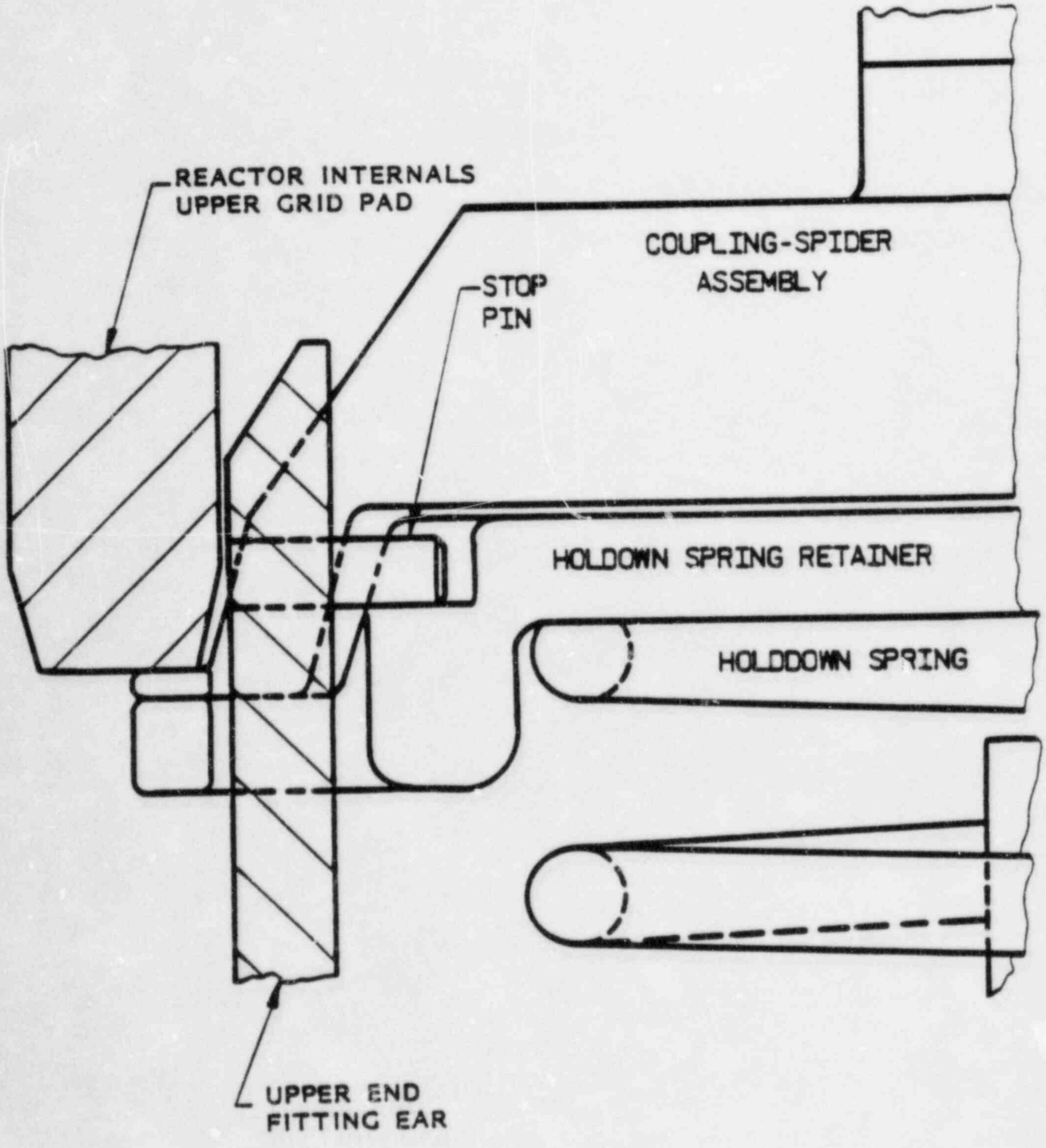


Figure 4-4. Mark B5 BPRA Spider/Upper End Fitting/
Reactor Internals Interaction



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycle 4 with those of cycle 5. These values were generated using PDQ07[8-10] for both cycles. Differences in core physics parameters are to be expected between the cycles due to the initial BPRA loading, the longer cycle 5 length, and the different shuffle pattern for cycle 5. Figure 5-1 illustrates a representative relative power distribution for BOC-5 at full power (FP) with equilibrium xenon and group 8 inserted.

Because of different isotopic distributions, cycle 5 control rod worths, ejected rod worths, and stuck rod worths differ from those of cycle 4. 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 5 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. Flux redistribution penalty.

Flux redistribution was taken into account since the shutdown analysis was calculated using a two-dimensional model. The cycle 5 moderator coefficients and the power deficits from hot zero power (HZP) to hot full power (HFP) are similar to those for cycle 4. The differential boron and xenon worths are also similar in both cycles. The effective delayed neutron fraction for cycle 5 shows a decrease with burnup (also shown in reference cycle 4).

5.2. Changes in Nuclear Design

There is only one significant core design change between the reference cycle and the cycle 5 designs. The change is the increase in cycle lifetime to 390 EFPD and the accompanying use of BPRAs to aid in reactivity control. The same calculational methods and design information were used to obtain the important nuclear design parameters. 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 have been analyzed. The calculated stability index without APSRs is $-0.044h^{-1}$, which demonstrates the axial stability of the core.

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

	Cycle 4	Cycle 5
Licensed cycle length, EFPD	280	390
Cycle burnup, MWd/mtU	9,350	13,043
Average core burnup - EOC, MWd/mtU	20,259	22,797
Initial core loading, mtU	83.0	82.9
Critical boron - BOC, No Xe, ppm		
HZP (Group 8 inserted)	1,250	1,485
HFP (Group 8 inserted)	1,042	1,255
Critical boron - EOC, Eq. Xe, ppm		
HZP (Group 8 withdrawn)	267	324
HFP (Group 8 withdrawn)	10[a]	10[a]
Control rod worths - HFP, BOC, % Δk/k		
Group 6	1.02	1.13
Group 7	1.73	1.42
Group 8	0.27	0.38
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.68	1.46
Group 8	NA	NA
Max ejected rod worth - HZP, % Δk/k (location)		
BOC Groups 5-8 inserted	0.85[b] (N-12)	0.60 (N-12)
EOC Groups 5-7 inserted, Group 8 withdrawn	0.85[b] (N-12)	0.55 (N-12)
Max stuck rod worth - HZP, % Δk/k (location)		
BOC	1.70 (L-14)	0.80 (N-12)
EOC	1.46 (N-12)	0.76 (M-11)
Power deficit-HZP to HFP, Eq. Xe, % Δk/k		
BOC (4 EFPD)	-1.77	-1.76
EOC	-2.33	-2.48
Doppler coeff - HFP, 10^{-3} % Δk/k/°F		
BOC, No Xe, 1255 ppm,[c] Group 8 inserted	-1.47	-1.50
EOC, Eq. Xe, 10 ppm, Group 8 withdrawn	-1.63	-1.76
Moderator coeff - HFP, 10^{-2} % Δk/k/°F		
BOC, No Xe, 1255 ppm,[c] Group 8 inserted	-1.00	-0.81
EOC, Eq Xe, 10 ppm, Group 8 withdrawn	-2.76	-2.86
Boron worth - HFP, ppm/% Δk/k		
BOC (1255 ppm)[c]	108	123
EOC (10 ppm)	96	106

Table 5-1. (Cont'd)

	Cycle 4	Cycle 5
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.67	2.63
EOC (equilibrium)	2.74	2.73
Effective delayed neutron fraction - HFP		
BOC	0.00598	0.00631
EOC	0.00530	0.00524

[a]Power coastdown to EOC at 10 ppmb.

[b]Ejected rod worth at the rod insertion limit.

[c]Cycle 5 values were calculated at 1255 ppm. Cycle 4 values were calculated at 1042 ppm.

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

	BOC, % Δk/k	EOC, % Δk/k 330 EFPD Bank 8 in	EOC, % Δk/k 390 EFPD Bank 8 out
Available Rod Worth			
Total rod worth, HZP	7.31	7.57	7.44
Worth reduction due to burnup of poison material	-0.17	-0.19	-0.18
Maximum stuck rod, HZP	-0.80	-0.74	-0.76
Net worth	6.34	6.64	6.50
Less 10% uncertainty	-0.63	-0.66	-0.65
Total available worth	5.71	5.98	5.85
Required Rod Worth			
Power deficit, HFP to HZP	1.76	2.52	2.48
Max allowable inserted rod worth	0.42	0.49	0.52
Flux redistribution	0.65	1.18	1.20
Total required worth	2.83	4.19	4.20
Shutdown Margin			
Total available minus total required	2.88	1.79	1.65

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

Figure 5-1. BOC (4 EFPD), Cycle 5 Two-Dimensional Relative Power Distribution - Full Power, Equilibrium Xenon, APSRs Inserted[a]

	8	9	10	11	12	13	14	15
H	.794	.935	1.009	1.266	1.122	1.244	.940	.639
K	.938	1.006	1.249	1.004	1.212	1.111	1.155	.643
L	1.011	1.249	1.107	1.250	8 .876	1.263	1.061	.502
M	1.265	1.002	1.245	1.127	1.253	1.029	.791	
N	1.120	1.210	8 .874	1.250	1.097	1.063	.505	
O	1.243	1.109	1.261	1.029	1.064	.648		
P	.939	1.154	1.060	.790	.505			
R	.638	.642	.501					

X	Inserted Rod Group Number
X.XX	Relative Power Density

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

6. THERMAL-HYDRAULIC DESIGN

The fresh batch 7 fuel is hydraulically and geometrically similar to the other fuel loaded into the cycle 5 core. The introduction of the Mk-B5 upper end fitting does not affect either the core flow rate or the thermal-hydraulic performance. The introduction of BPRAs increases the core flow available for heat transfer by reducing the core bypass flow rate from 10.7 to 8.1%. This reduced bypass flow rate has been conservatively neglected for cycle 5. Therefore, the cycle 5 thermal-hydraulic design is identical to that of cycle 4. The thermal-hydraulic evaluation supporting cycle 5 operation is based on the methods and models described in references 11 and 12. The thermal-hydraulic design conditions for cycles 4 and 5 are summarized in Table 6-1.

Previous fuel cycle evaluations included the calculation of a rod bow penalty for each fuel batch based on the highest fuel rod burnup in that batch. A rod bow topical report[13], which addresses the mechanisms and resulting local conditions of rod bow, has been submitted to and approved by the NRC. The topical report concludes that rod bow penalty is insignificant and is offset by the reduction in power production capability of the FAs with irradiation. Therefore, no departure from nucleate boiling ratio (DNBR) reduction due to fuel rod bow need be considered for cycle 5.

Table 6-1. Davis-Besse Cycles 4 and 5 Thermal-Hydraulic Design Conditions

Design power level, MWt	2772
Nominal system pressure, psia	2200
Reactor coolant flow, % design	110
Vessel inlet/outlet coolant temp., 100% power, F	557.7/606.3
Ref design radial-local power peaking factor	1.71
Ref design axial flux shape	1.5 cosine with tails
Hot channel factors	
Enthalpy rise (F_q)	1.011
Heat flux (F_q'')	1.014
Flow area	0.98
Active fuel length	See Table 4-2
Avg heat flux, 100% power, Btu/h-ft^2	1.89×10^5[a]
Max heat flux, 100% power, Btu/h-ft^2	4.85×10^5[a]
Critical heat flux (CHF) correlation	BAW-2
Minimum DNBR, (% power)	1.79 (112%)

[a]With thermally expanded fuel rod OD of 0.43075 inch.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in the cycle 5 parameters to determine the effects of the cycle 5 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in reference 11.

The radiological dose consequences of the FSAR chapter 15 accidents based on cycle 5 iodine and noble gas inventories have been evaluated. These doses are either bounded by the FSAR values or are a small fraction of the 10 CFR 100 limits.

7 . 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 5 are given in Table 4-2. The cycle 4 and cycle 5 thermal-hydraulic maximum design conditions are presented in Table 6-1. A comparison of the key kinetics parameters from the FSAR and cycle 5 is provided in Table 7-1.

A generic loss-of-coolant accident (LOCA) analysis for B&W 177-FA raised-loop nuclear steam systems (NSSs) has been performed using the Final Acceptance Criteria ECCS Evaluation Model.[14] The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used in the LOCA limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits

reported in reference 14 provide conservative results for the operation of Davis-Besse Unit 1, cycle 5 fuel. A tabulation showing the bounding values for allowable LOCA peak LHRs for Davis-Besse Unit 1, cycle 5 fuel are provided in Table 7-2.

It is concluded by the examination of cycle 5 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 5. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 5 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 5 are bounded by the FSAR and/or the fuel densification report.

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

Parameter	FSAR and densif'n report value	Cycle 5 value
BOL[a] Doppler coeff, 10^{-3}, % Δk/k/°F	-1.28	-1.50
EOL[b] Doppler coeff, 10^{-3}, % Δk/k/°F	-1.45[c]	-1.76
BOL moderator coeff, 10^{-2}, % Δk/k/°F	+0.13	-0.81
EOL moderator coeff, 10^{-2}, % Δk/k/°F	-3.0	-2.86
All rod bank worth (HZP), % Δk/k	10.0	7.31
Boron reactivity worth (HFP), ppm/1% Δk/k	100	123
Max ejected rod worth (HFP), % Δk/k	0.65	0.32
Max dropped rod worth (HFP), % Δk/k	0.65	0.20
Initial boron conc (HFP), ppm	1407	1255

(a)BOL denotes beginning of life.

(b)EOL denotes end of life.

(c)-1.77×10^{-3} % Δk/k/°F was used for steam line failure analysis.

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

Core elevation, ft	Allowable peak LHR, first 25 EFPD, kW/ft	Allowable peak LHR, balance of cycle, kW/ft
2	15.5	16.5
4	16.8	17.2
6	18.0	18.4
8	17.5	17.5
10	17.0	17.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 5 operation to account for changes in power peaking and control rod worths. The effects of NUREG-0630 have been incorporated into the operating limits. Figures 8-1 through 8-26 are revisions to the previous cycle Technical Specifications. Based on these Technical Specifications the final acceptance criteria ECCS limits will not be exceeded and the thermal design criteria will not be violated.

Table 8-1. Reactor Protection System Instrumentation Trip Setpoints

Table 2.2-1

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable.	Not applicable.
2. High flux	≤104.94% of RATED THERMAL POWER with four pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤79.7% of RATED THERMAL POWER with three pumps operating	≤79.7% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/flow(1)	Trip setpoint not to exceed the limit line of Figure 2.2-1	Allowable values not to exceed the limit line of Figure 2.2-1#
5. RC low pressure(1)	≥1983.4 psig	≥1983.4 psig* ≥1983.4 psig**
6. RC high pressure	≤2300 psig	≤2300.0 psig* ≤2300.0 psig**
7. RC pressure-temperature(1)	≥(12.60 T_{out} °F - 5662.2) psig	≥(12.60 T_{out} °F - 5662.2) psig#
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop	≤55.1% of RATED THERMAL POWER with one pump operating in each loop#
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	≤0.0 of RATED THERMAL POWER with no pumps operating or only one pump operating	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	≤4 psig	≤4 psig#

The reactor trip envelope appears to approach the safety limits more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow.

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of F_Q = 2.56 or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits are 20.4 kW/ft for batches 1E, 4B, and 5A and 20.5 kW/ft for batches 5B, 6, and 7.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to +22%, whichever condition is more restrictive. These curves include the potential effects of fuel rod bow and fuel densification.

The DNBR as calculated by the B&W-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of +22% is justified on the basis of experimental data.

RC High Temperature

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

Flux -- ΔFlux/Flow

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Examples of typical power level and low flow rate combinations for the pump situations of Table 2.2-1 that would result in a trip are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 106.8% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 93.63% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.7% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 70.22% of full flow rate and power is 75%.

For safety calculations the instrumentation errors for the power level were used. Full flow rate in the above two examples is defined as the flow calculated by the heat balance at 100% power.

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2a, -2b, -2c, and -2d and 3.1-3a, -3b, -3c and -3d. A rod group overlap of 25 ±5% shall be maintained between sequential withdrawn groups 5, 6 and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION

With the regulating rod groups inserted beyond the above insertion limits (in a region other than acceptable operation), or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

NOTE: If in unacceptable region, also see Section 3/4.1.1.1.

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*See Special Test Exception 3.10.1 and 3.10.2.

#With $k_{eff} \geq 1.0$.

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 The axial power shaping rod group shall be limited in physical insertion as shown on Figures 3.1-5a, -5b, -5c, -5d, -5e, -5f, and -5g.

APPLICABILITY: MODES 1 and 2*.

ACTION

With the axial power shaping rod group outside the above insertion limits, either:

a. Restore the axial power shaping rod group to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.9 The position of the axial power shaping rod group shall be determined to be within the insertion limits at least once every 12 hours except when the axial power shaping rod insertion limit alarm is inoperable, then verify the group to be within the insertion limit at least once every 4 hours.

*With $K_{eff} \geq 1.0$.

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1a, -1b, -1c, and -1d and 3.2-2a, -2b, -2c and -2d.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

b. Within one hour reduce power until imbalance limits are met or to 40% of RATED THERMAL POWER or less.

SURVEILLANCE REQUIREMENTS

4.2.1. The AXIAL POWER IMBALANCE shall be determined to be within limits at least once every 12 hours when above 40% of RATED THERMAL POWER except when the AXIAL POWER IMBALANCE alarm is inoperable, then calculate the AXIAL POWER IMBALANCE at least once per hour.

---

*See Special Test exception 3.10.1.

Table 8-2. Quadrant Power Tilt Limits
(Tech. Spec. Table 3.2.2)

	Steady state limit	Transient limit	Maximum limit
Measurement independent QUADRANT POWER TILT	4.92	11.07	20.0
QUADRANT POWER TILT as measured by:	--	--	--
Symmetrical incore detector system, 0-50 ±10 EFPD	3.37	8.52	20.0
Symmetrical incore detector system, after 50 ±10 EFPD	3.02	8.52	20.0
Power range channels	1.96	6.96	20.0
Minimum incore detector system	1.90	4.40	20.0

3/4 2-12

3/4.4. REACTOR COOLANT SYSTEM

3/4.4.1. COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 79.7% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux
   2. Flux-ΔFlux-Flow

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

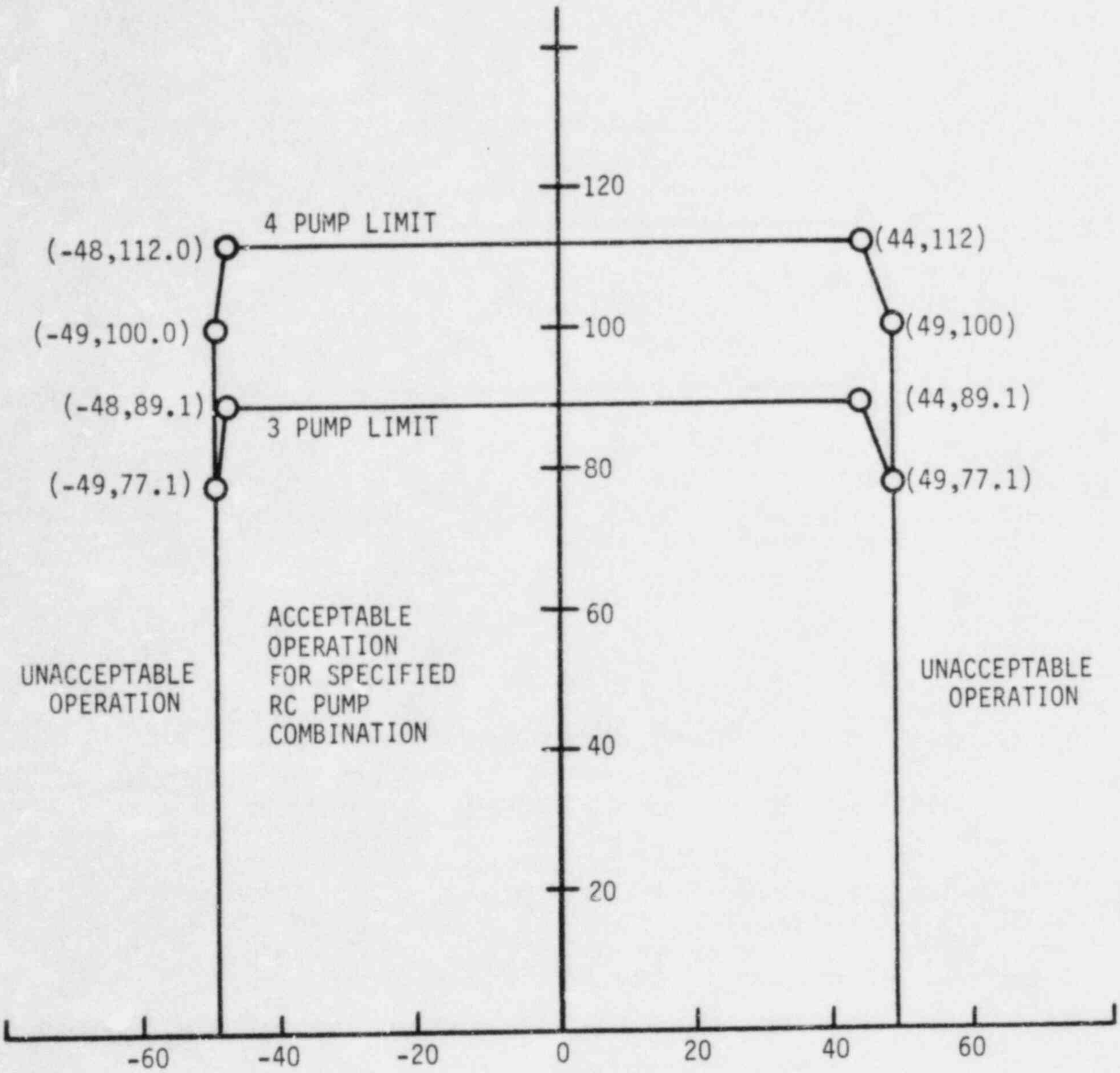
4.4.1.2 The reactor protective instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

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

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

*See Special Test Exception 3.10.3.

Figure 8-1. Reactor Core Safety Limit
(Tech. Spec. Figure 2.1-2)



Axial Power Imbalance, %

PUMPS OPERATING	REACTOR COOLANT FLOW, GPM
4	387,200
3	290,100

Figure 8-2. Trip Setpoint for Flux -- ΔFlux/Flow
(Tech. Spec. Figure 2.2-1)

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

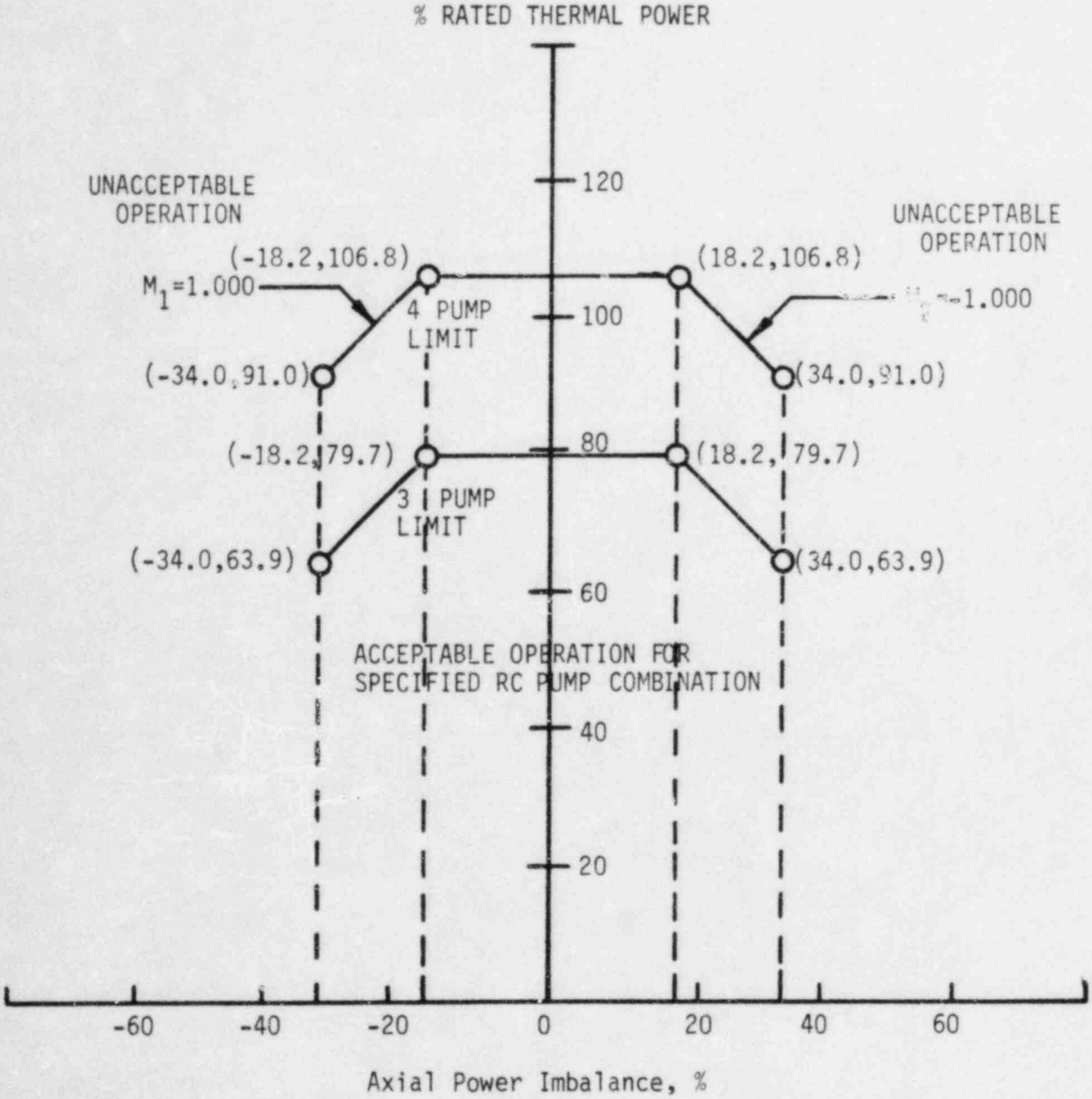


Figure 8-3. Regulating Group Position Limits, 0 to 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-2a)

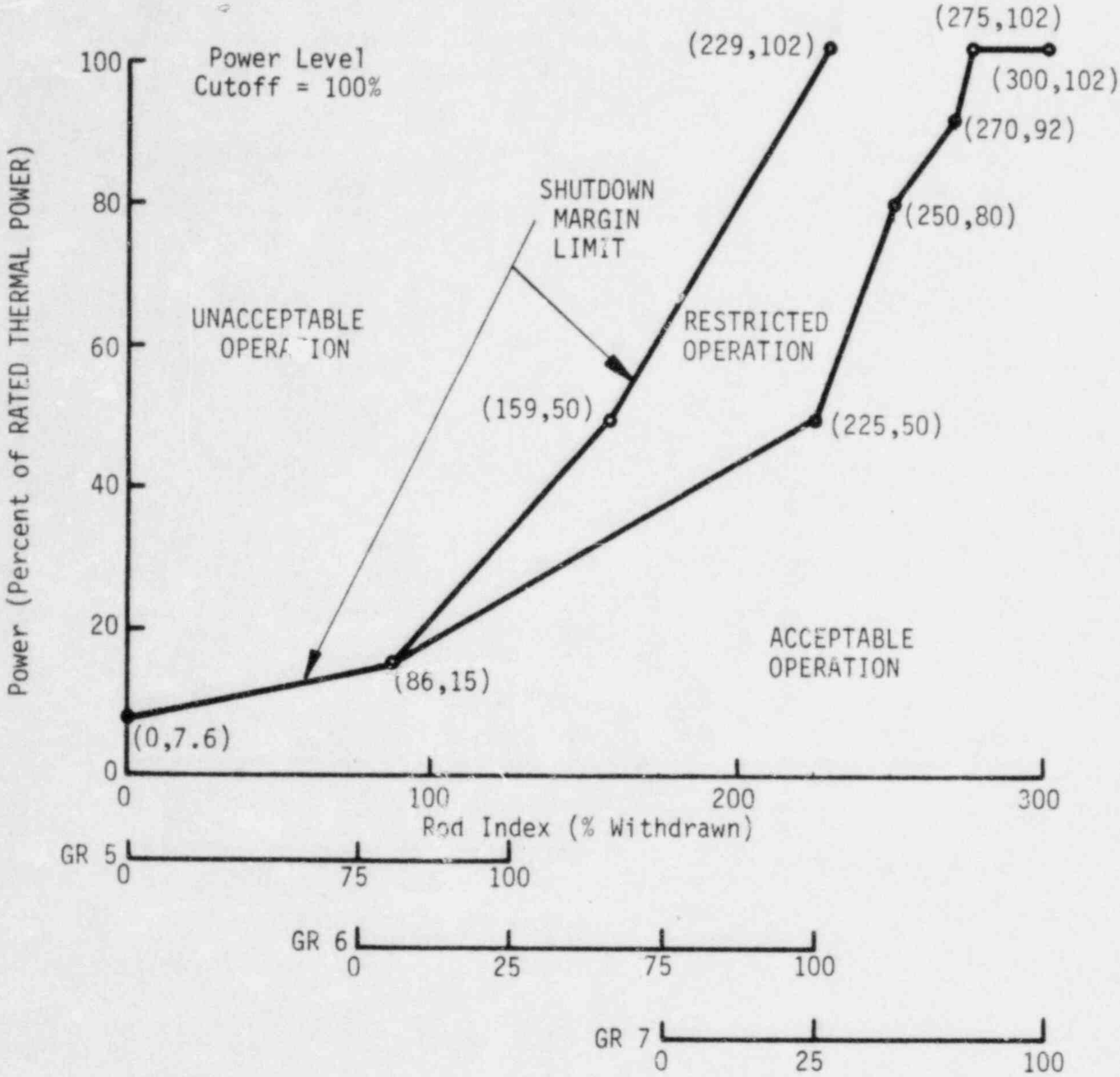


Figure 8-4. Regulating Group Position Limits, 25+10/-0 to 200±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-2b)

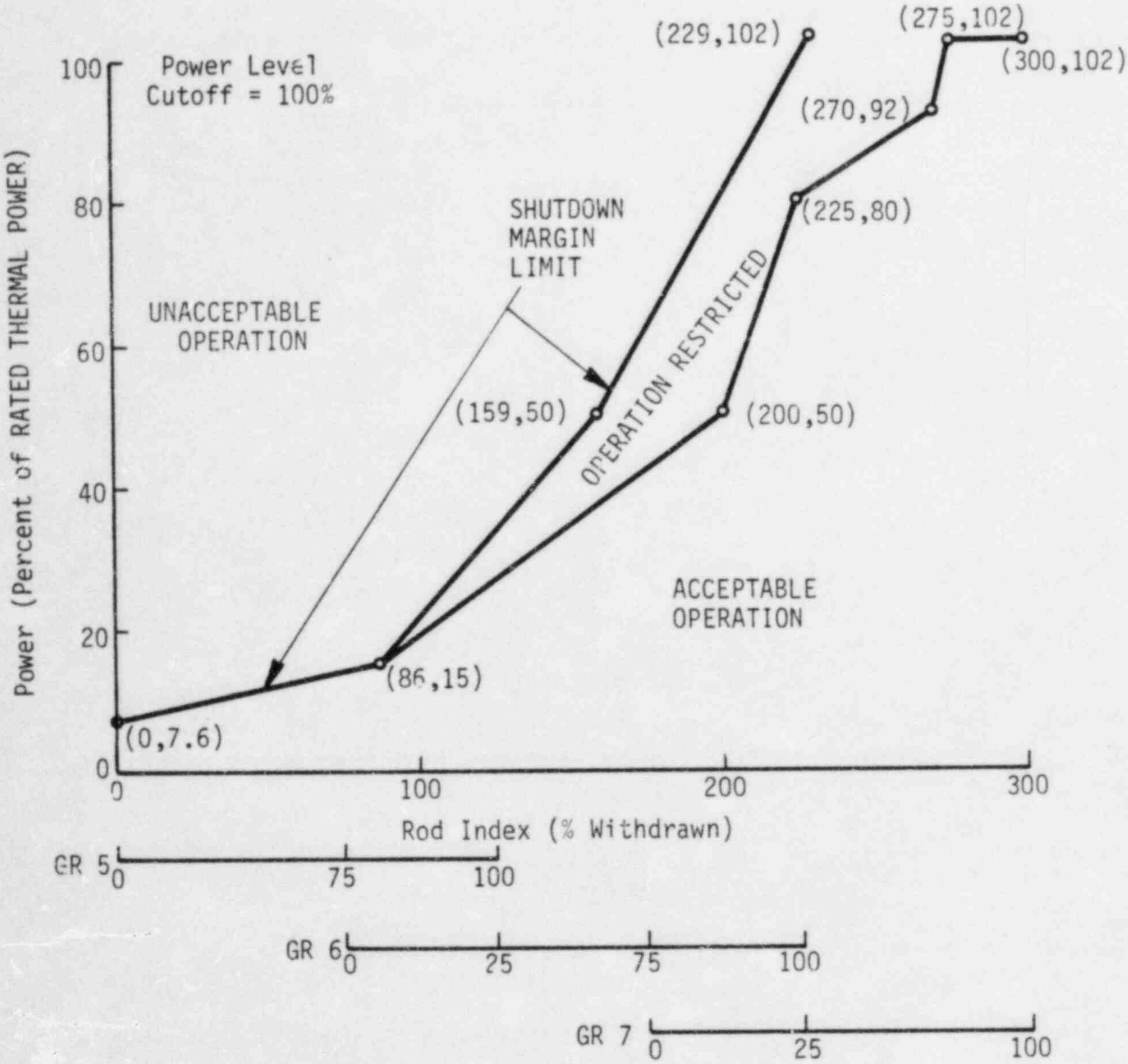


Figure 8-5. Regulating Group Position Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-2c)

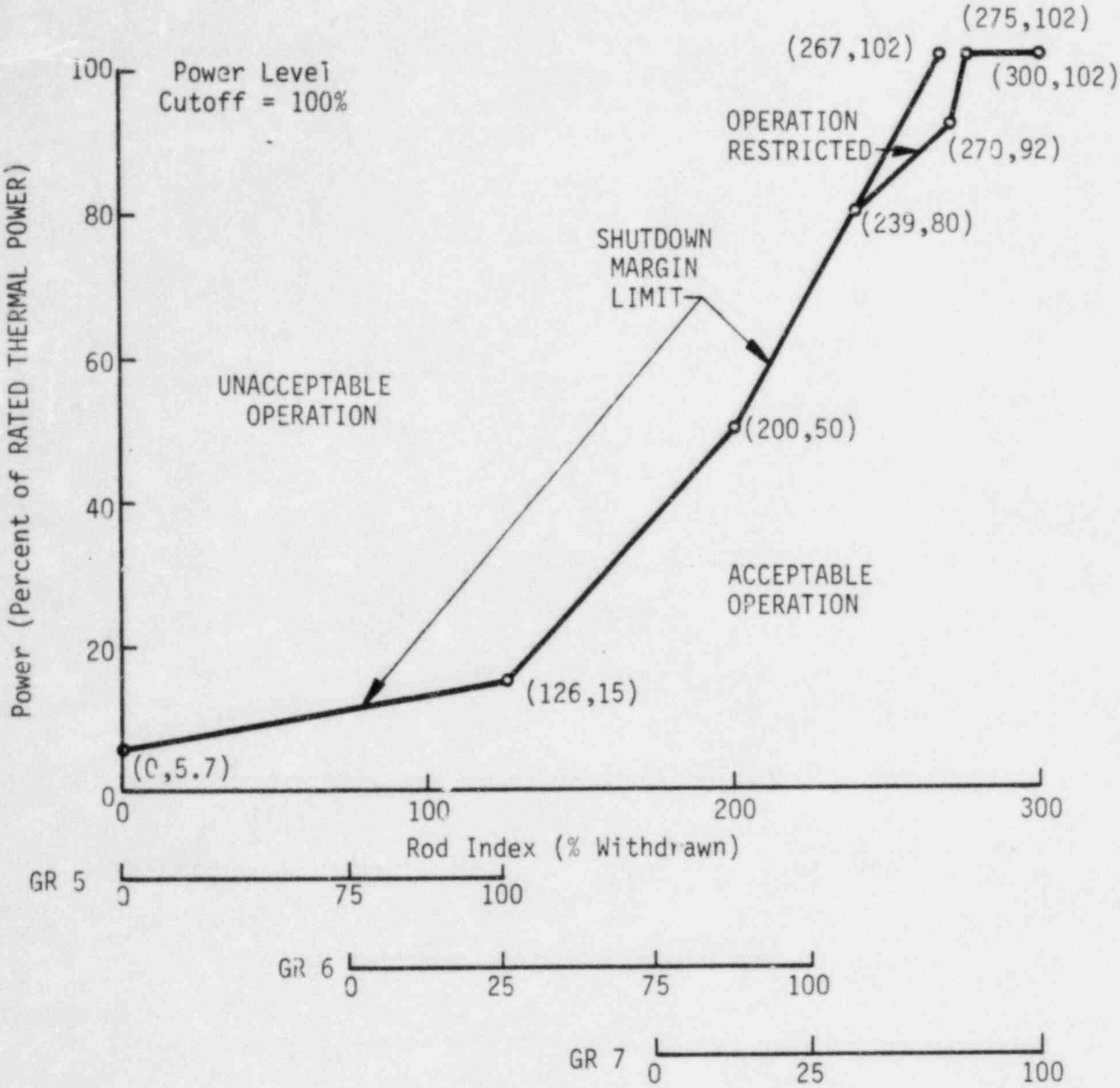


Figure 8-6. Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-2d)

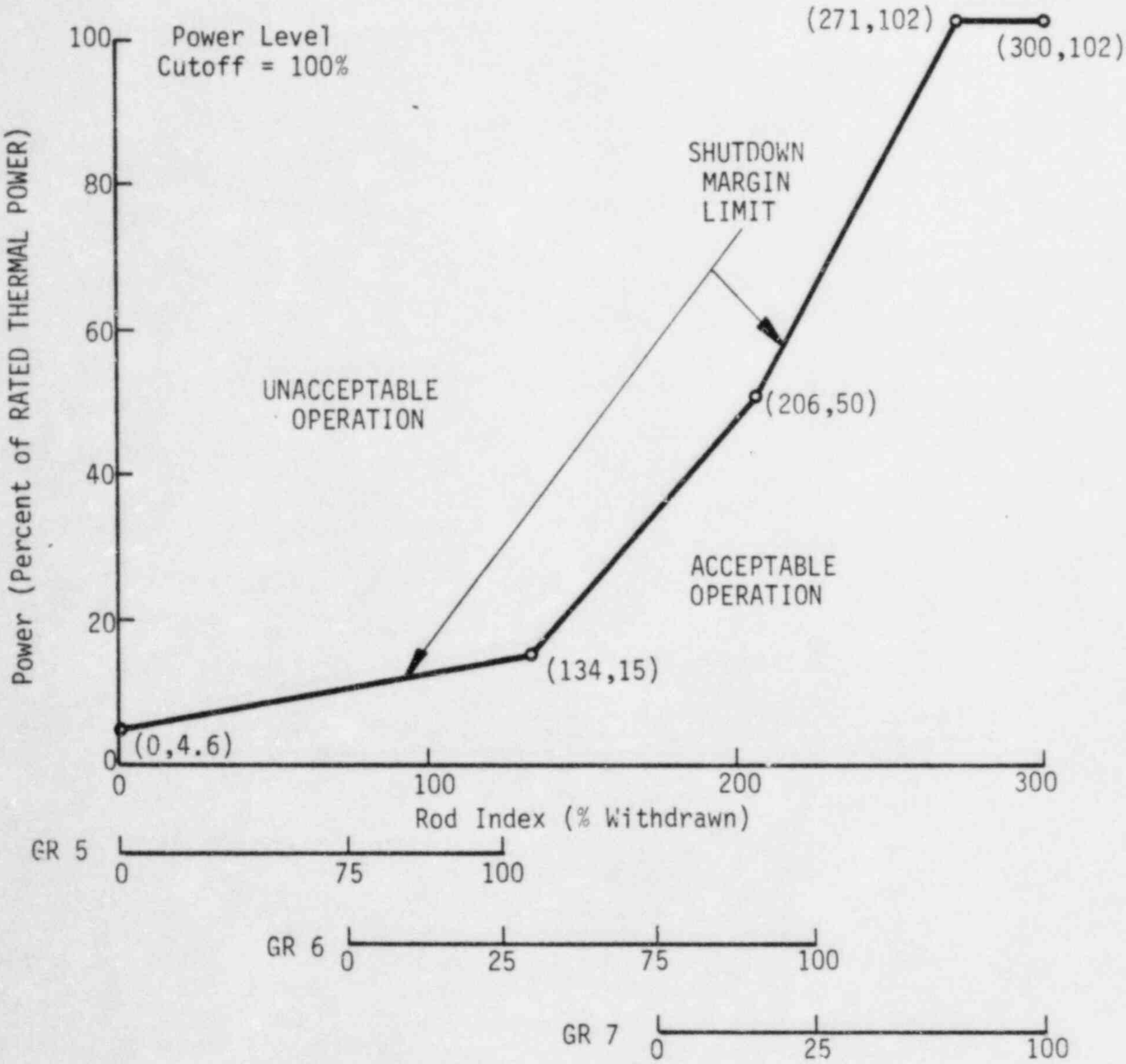


Figure 8-7. Regulating Group Position Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-3a)

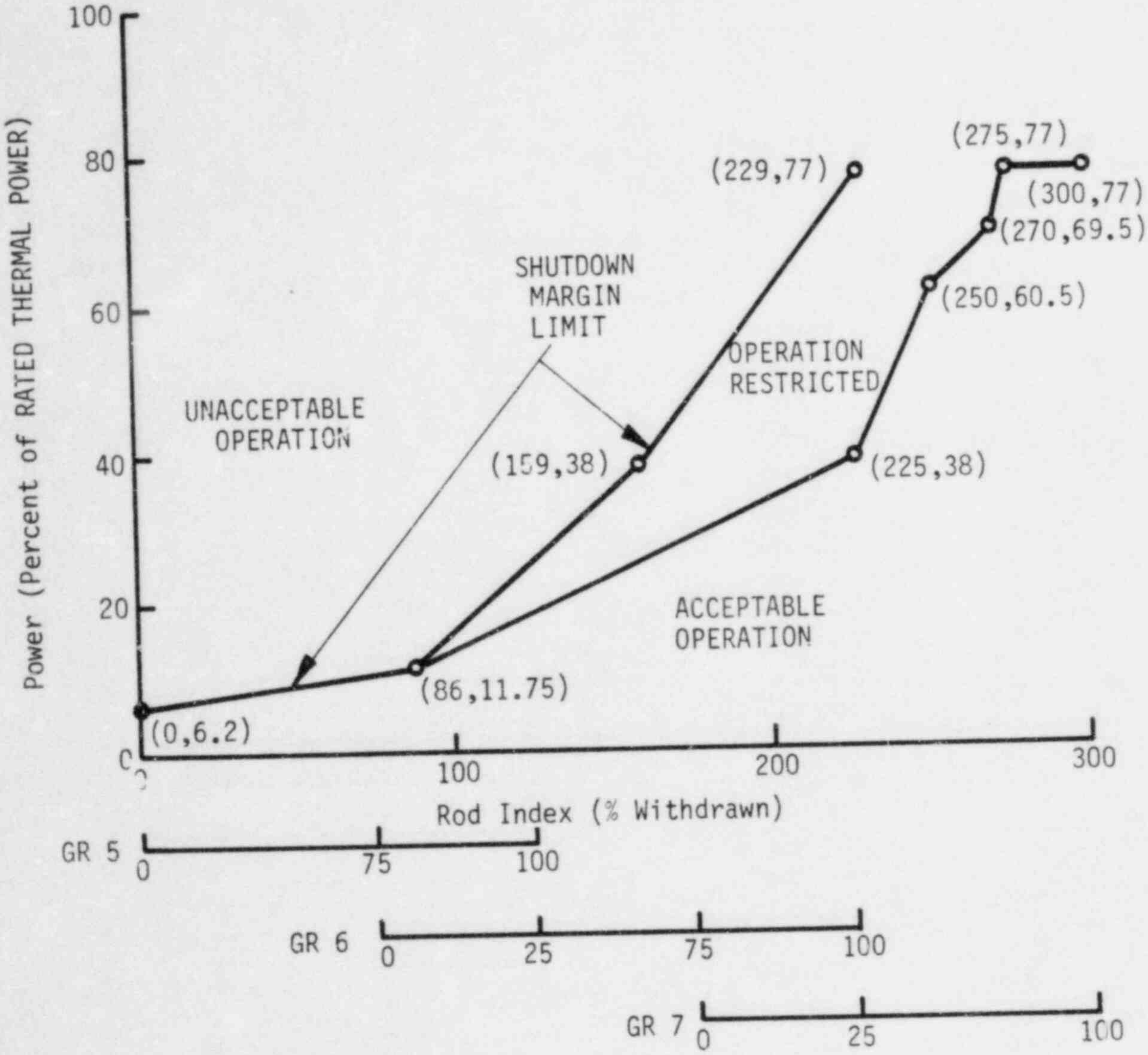


Figure 8-8. Regulating Group Position Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-3b)

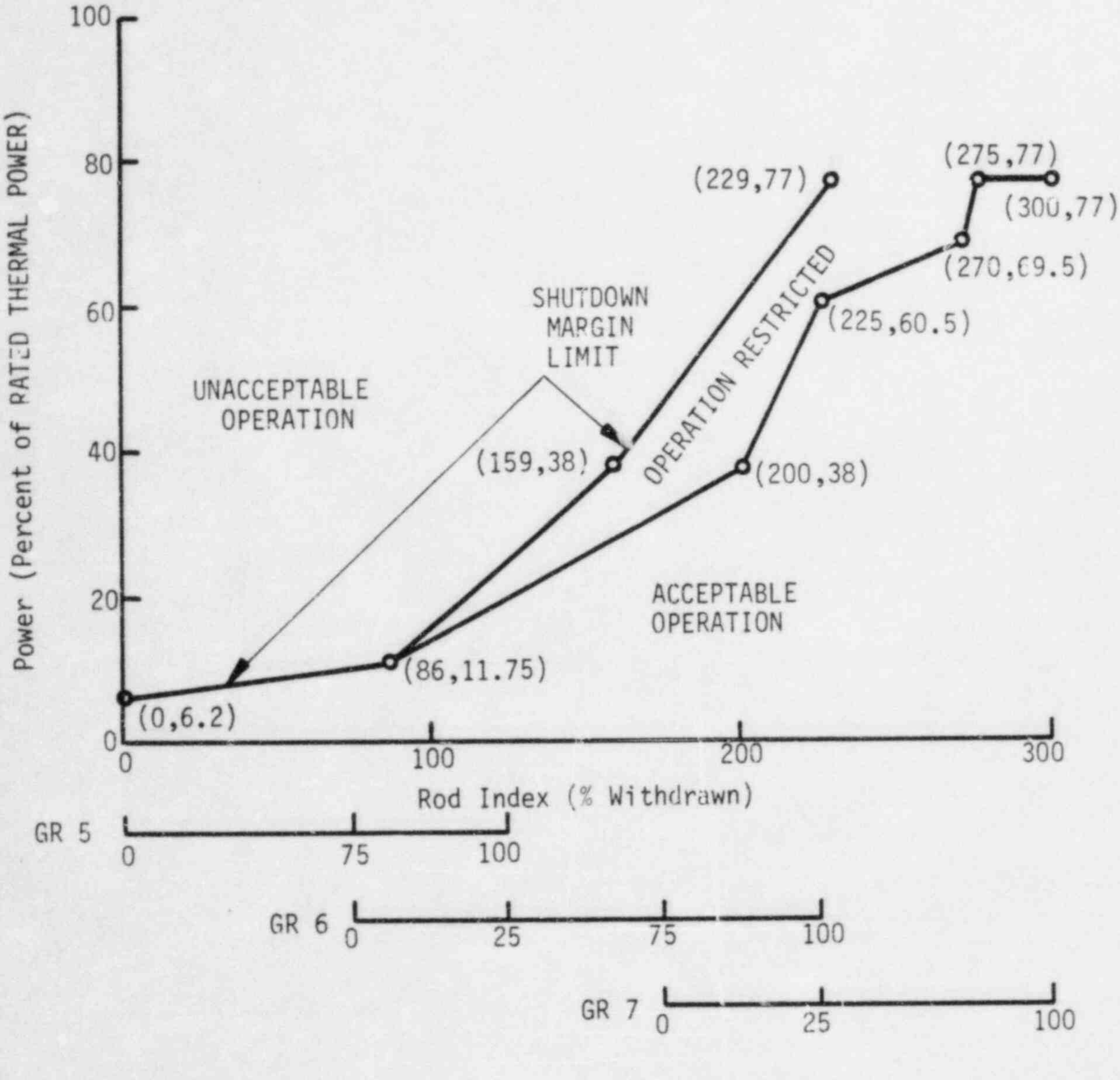


Figure 8-9. Regulating Group Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-3c)

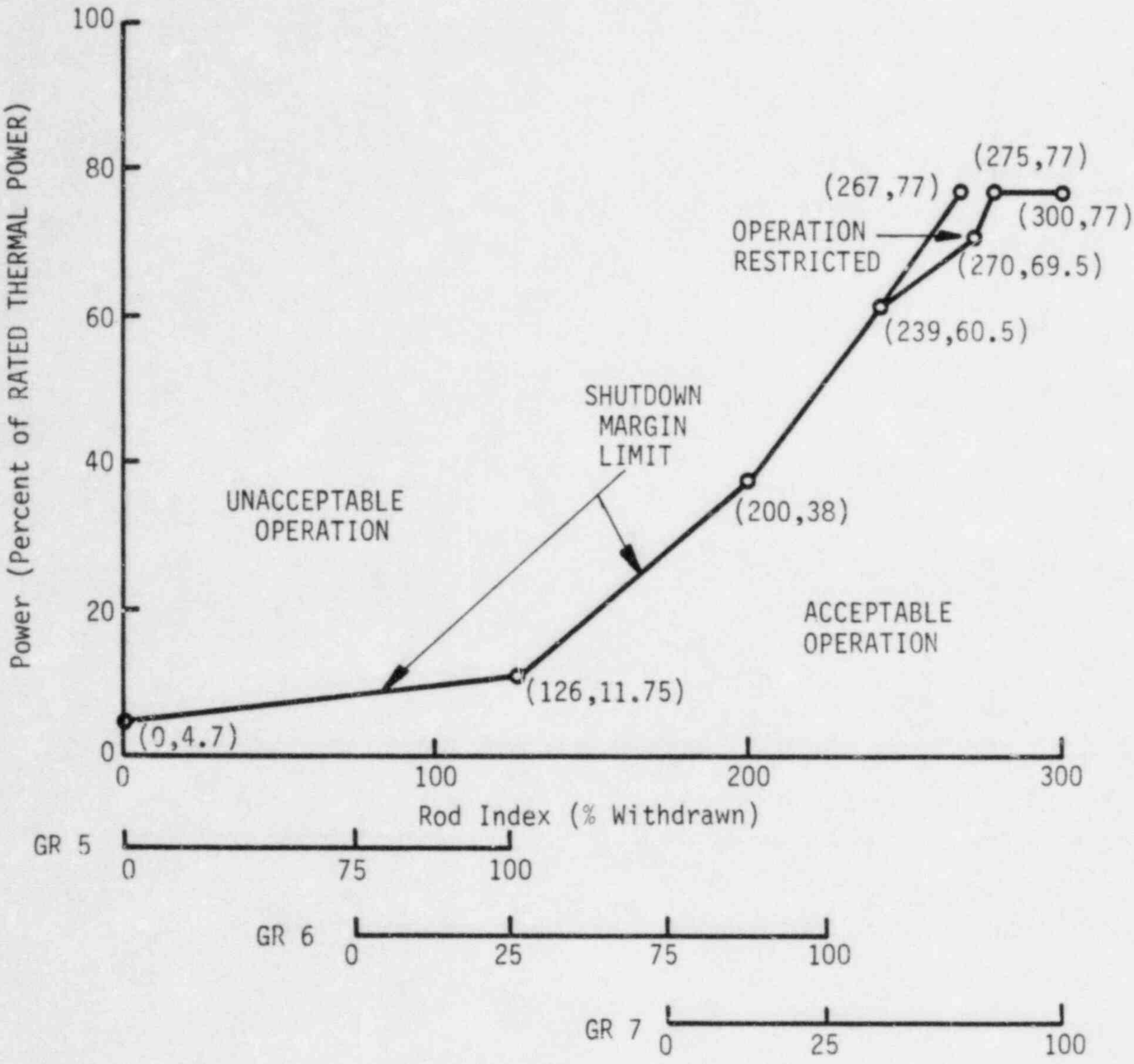


Figure 8-10. Regulating Group Position Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-3d)

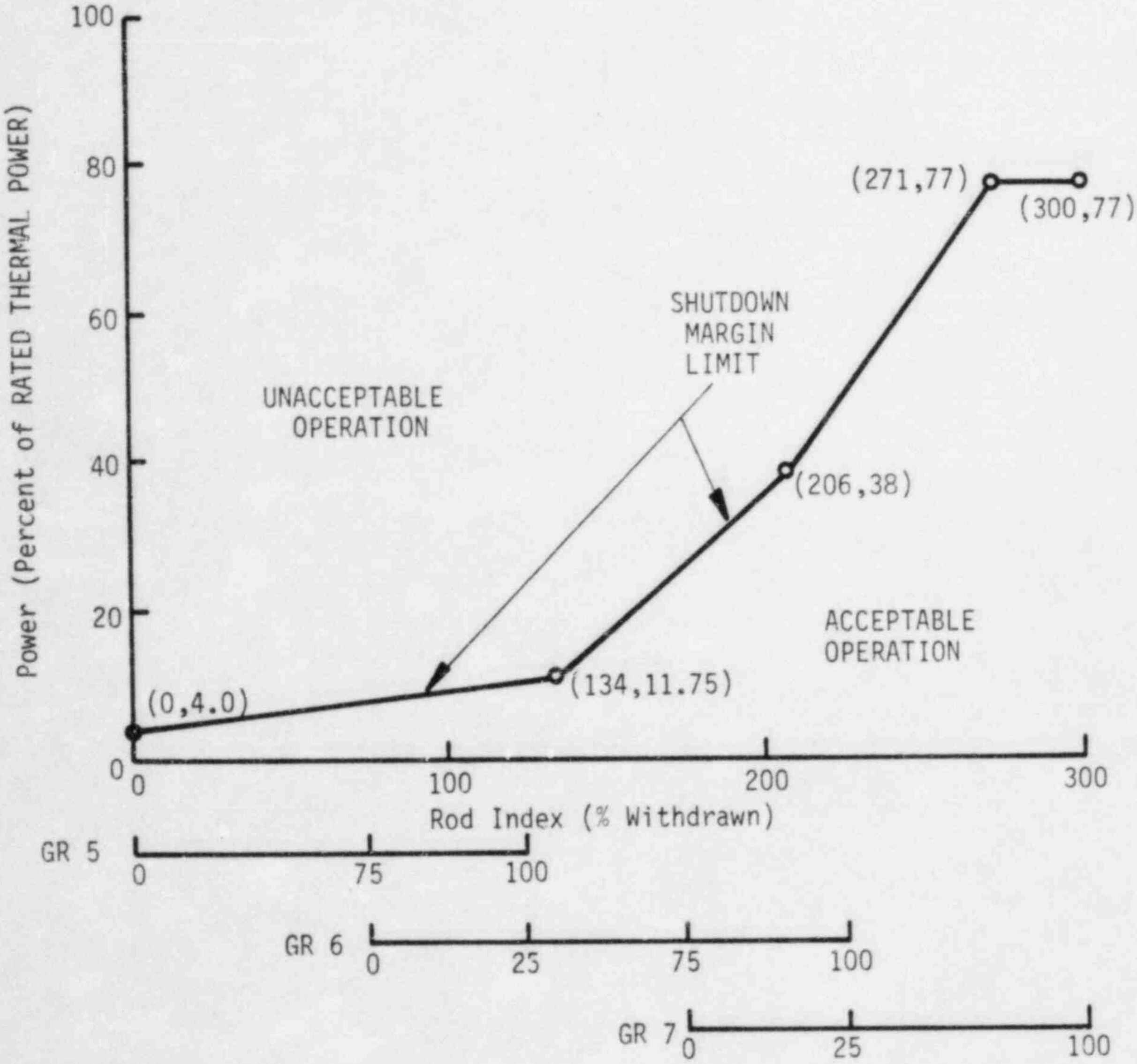


Figure 8-11. APSR Position Limits, 0 to 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5a)

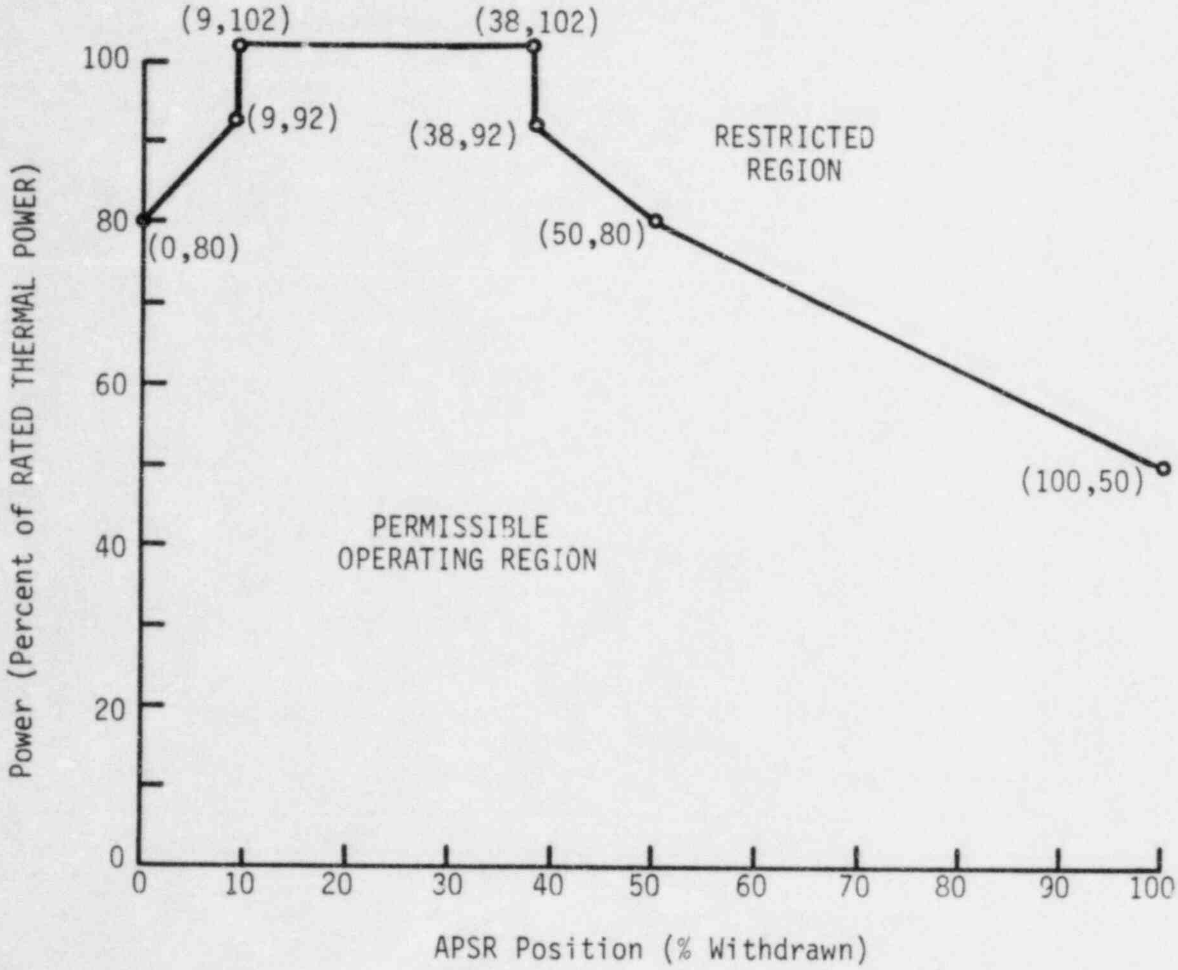


Figure 8-12. APSR Position Limits, 25+10/-0 to 200 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5b)

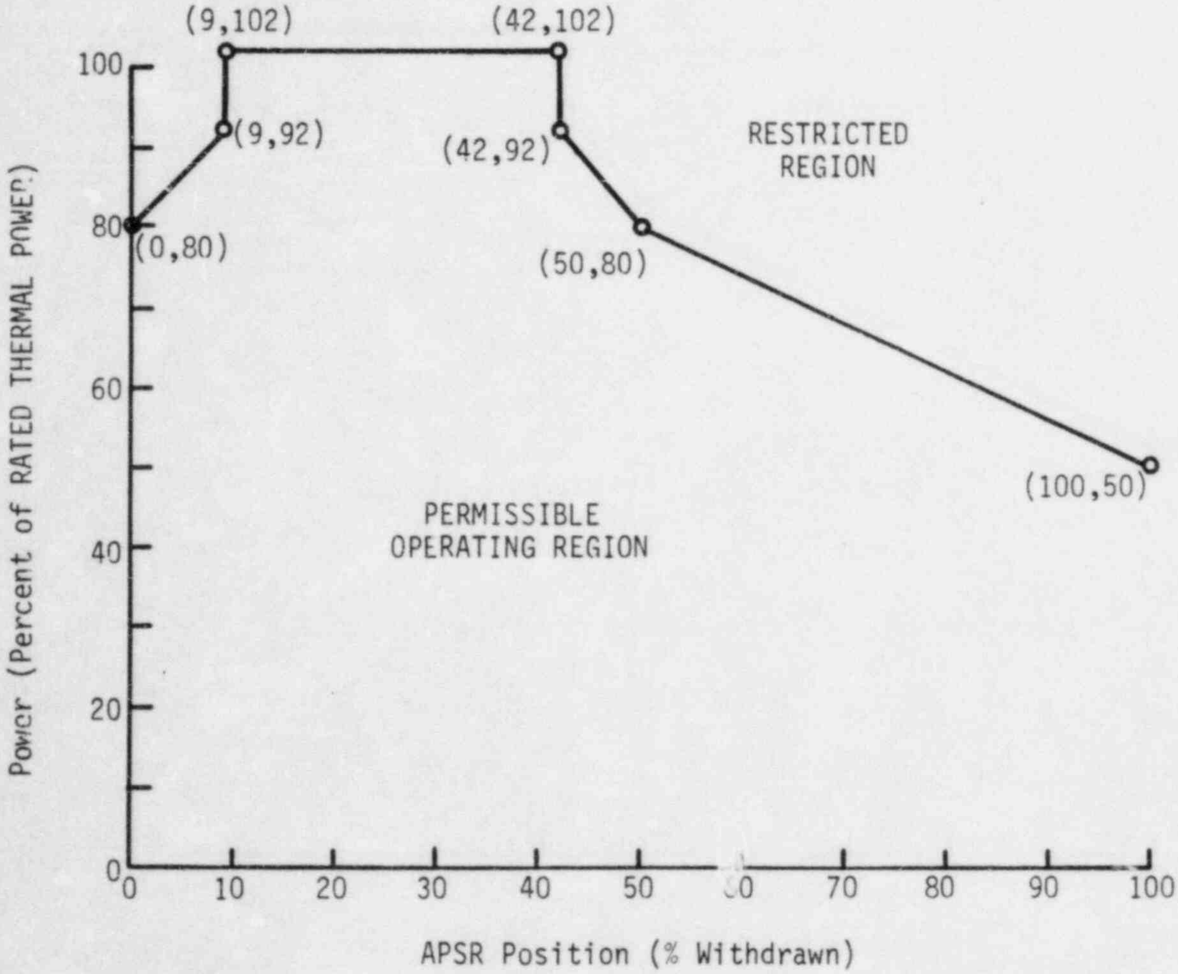


Figure 8-13. APSR Position Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5c)

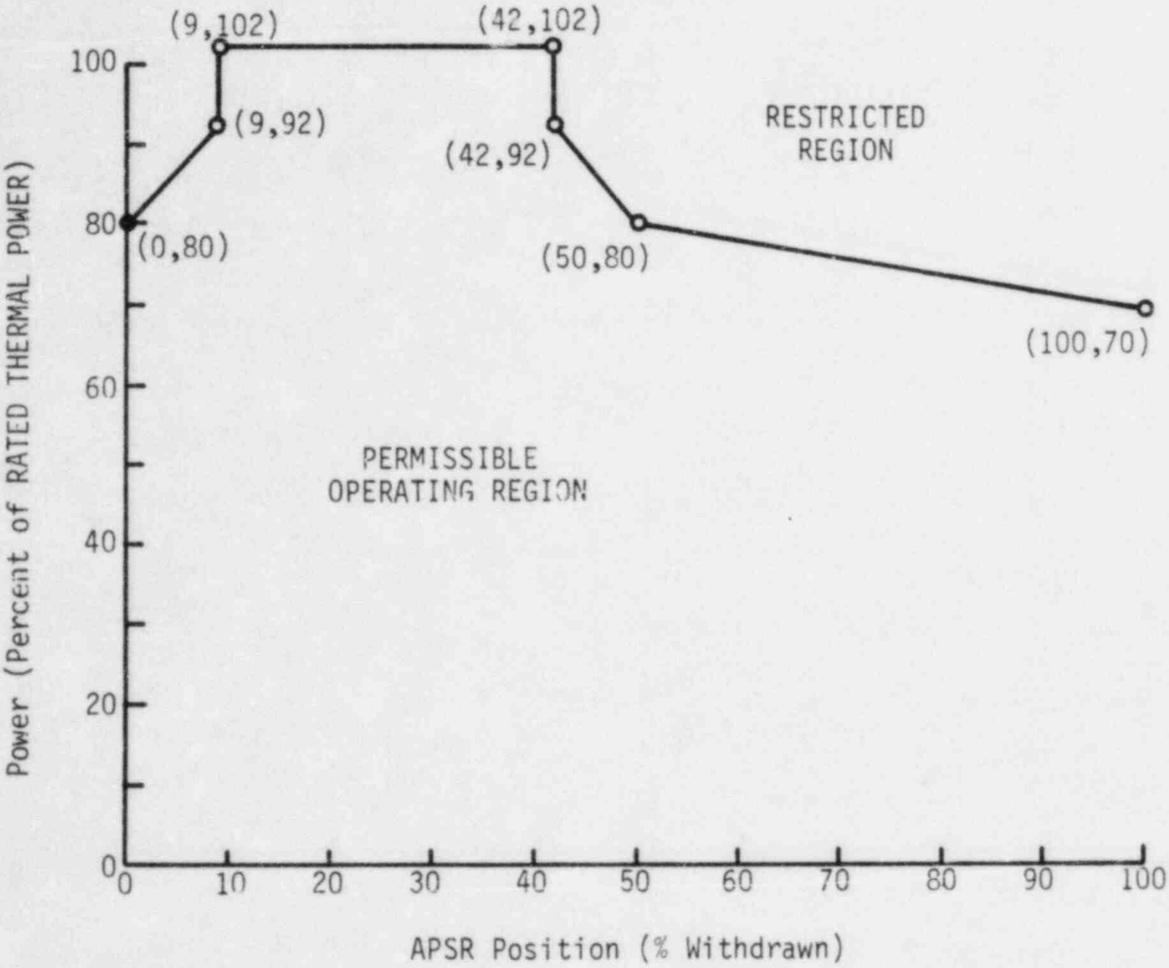


Figure 8-14. APSR Position Limits, 330 ±10 to 390 ±10 EFPD, Three or Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5d)

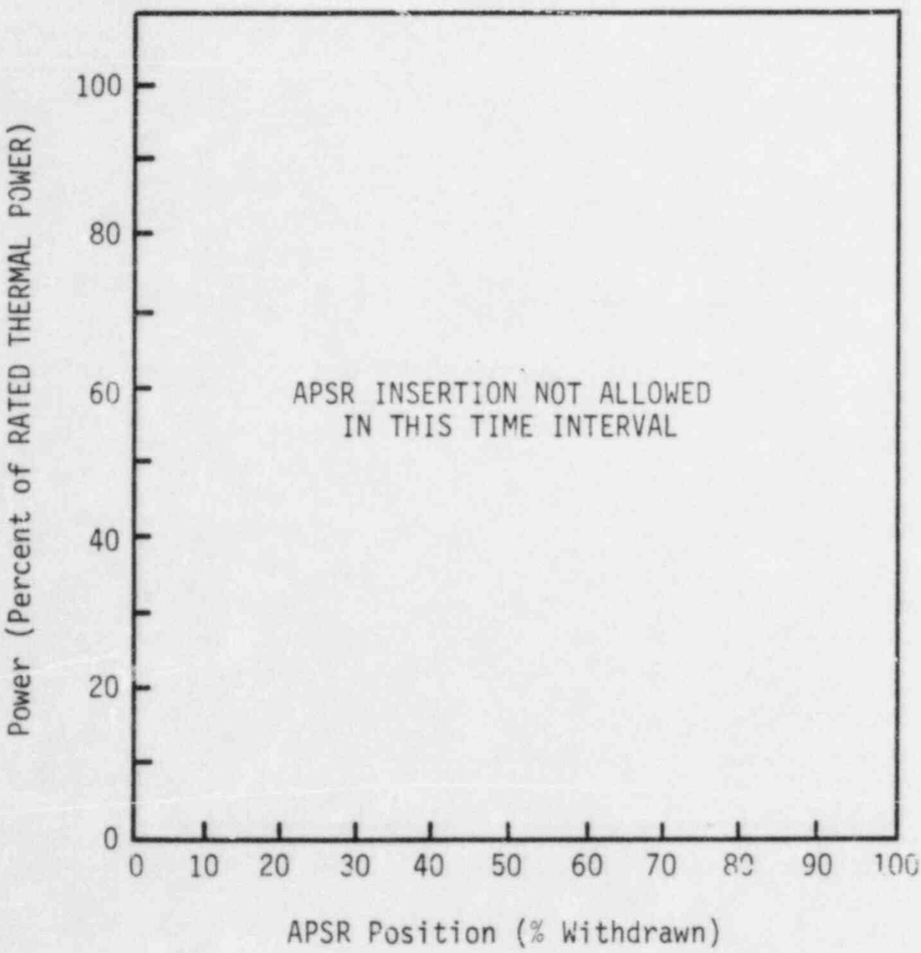


Figure 8-15. APSR Position Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5e)

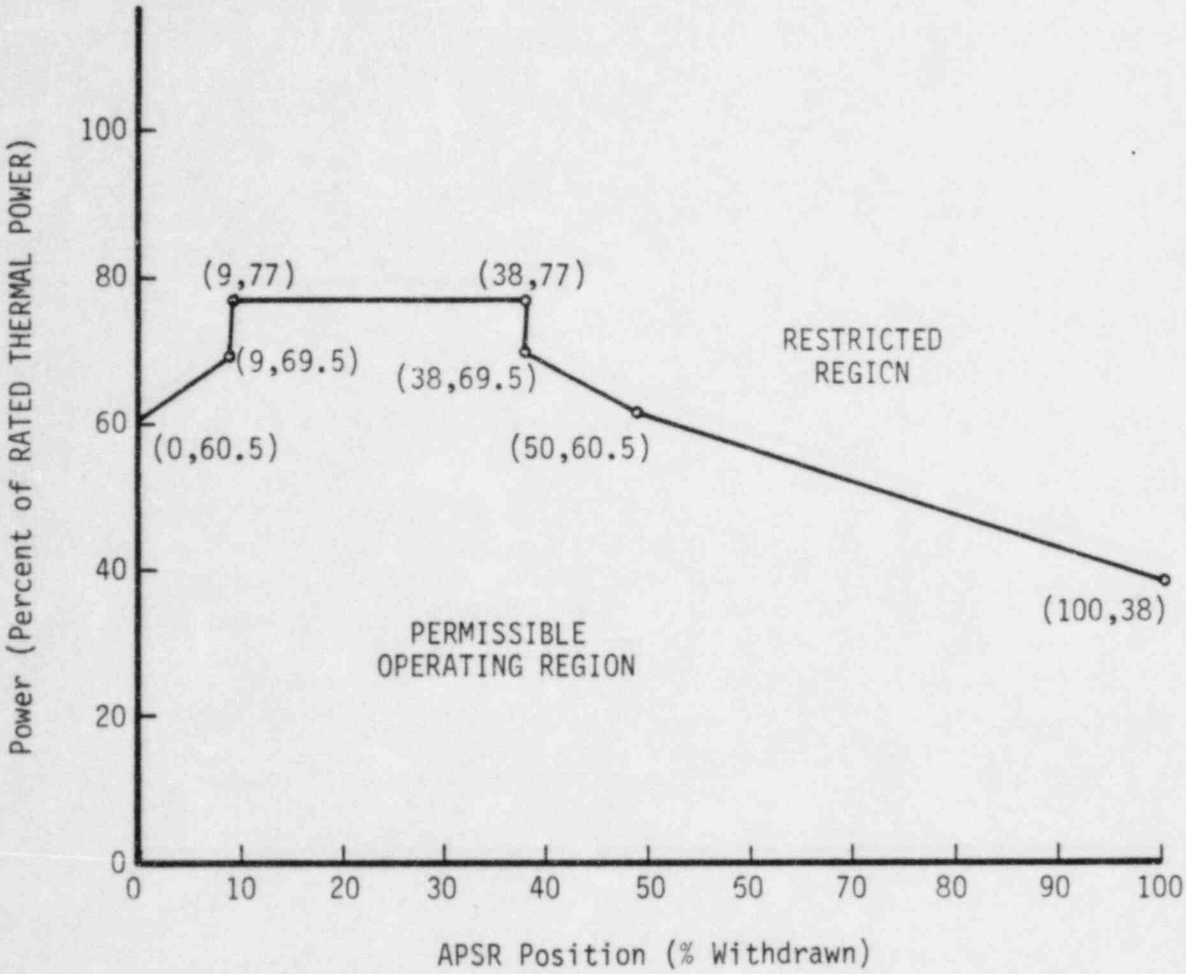


Figure 8-16. APSR Position Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5f)

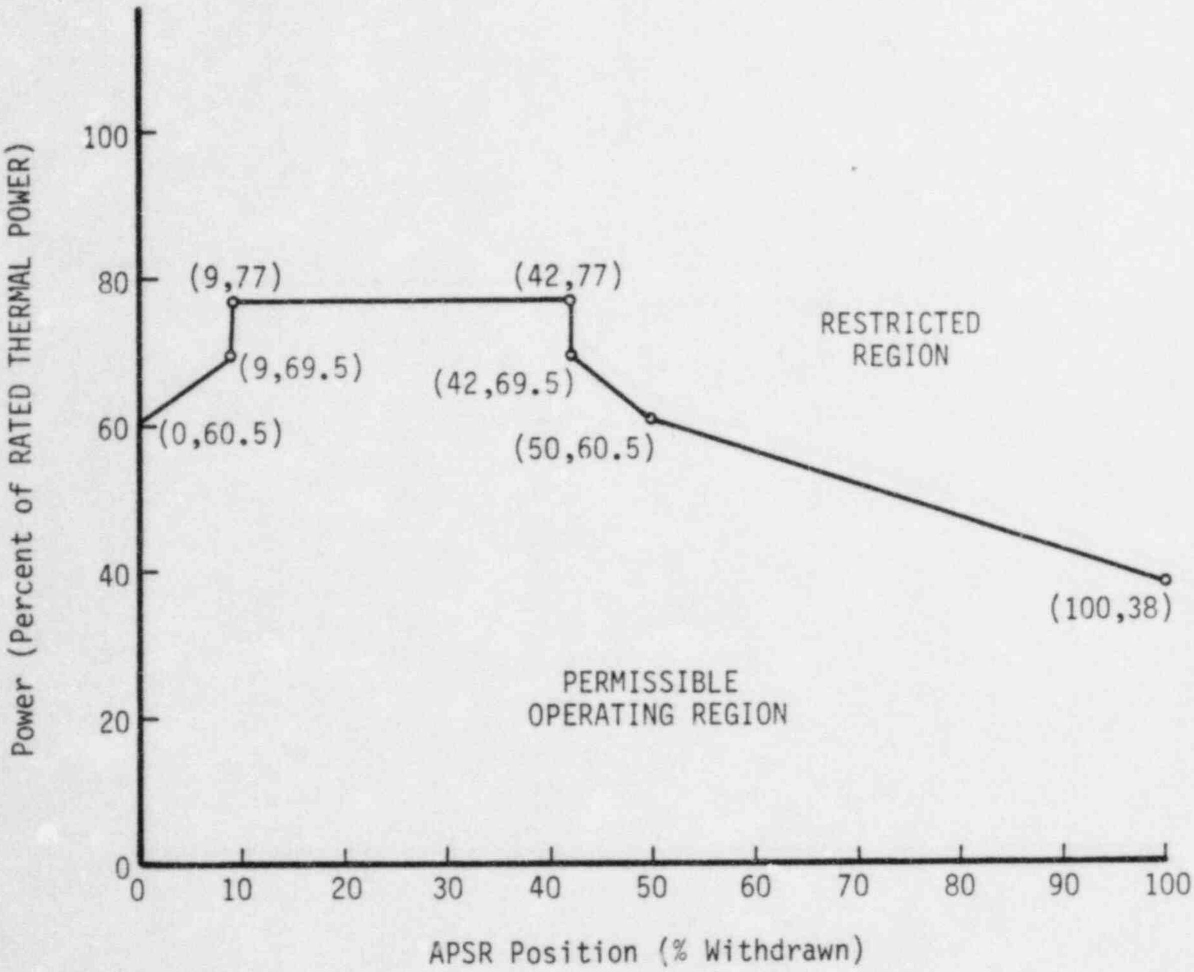


Figure 8-17. APSR Position Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-5g)

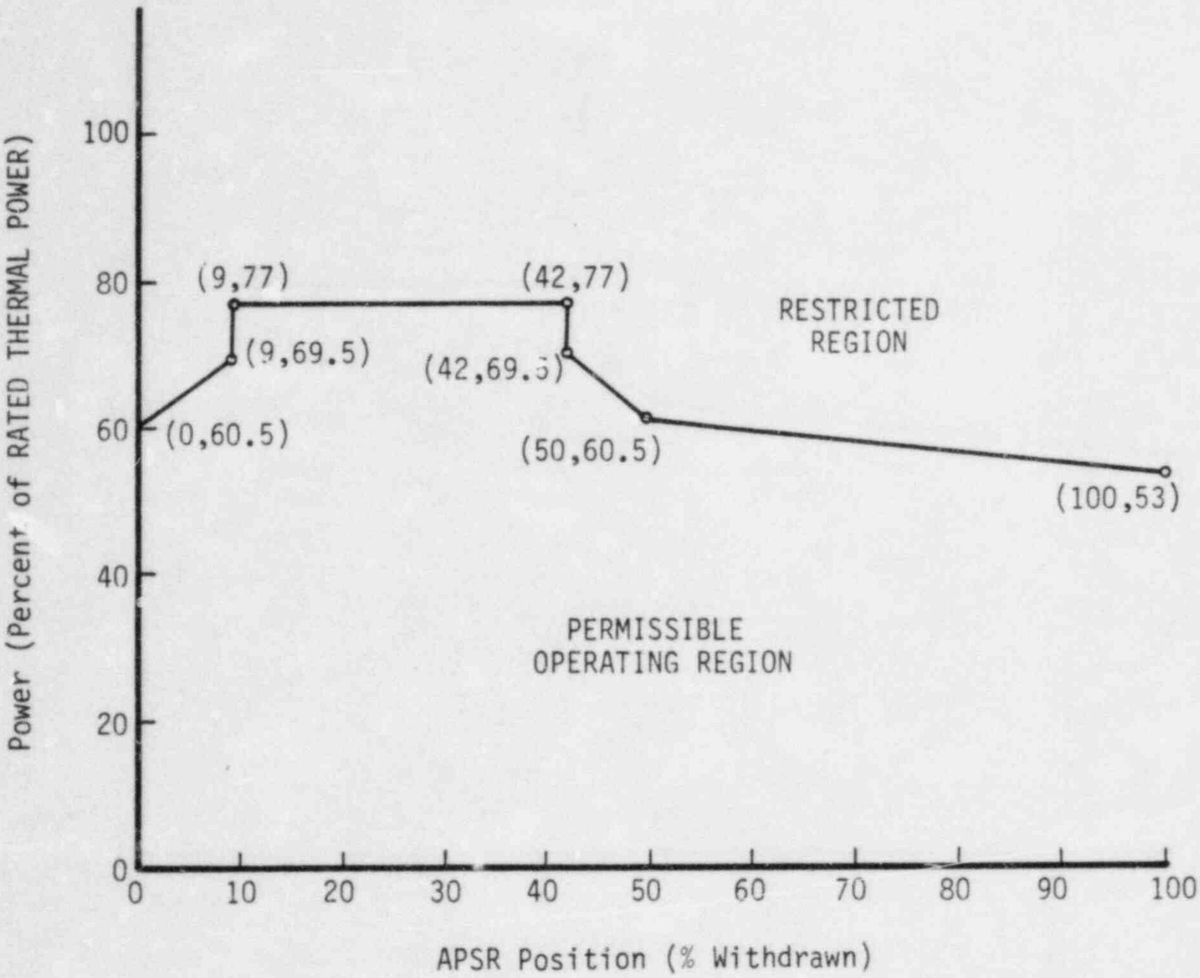


Figure 8-18. Axial Power Imbalance Limits, 0 to 25+10/-0 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-1a)

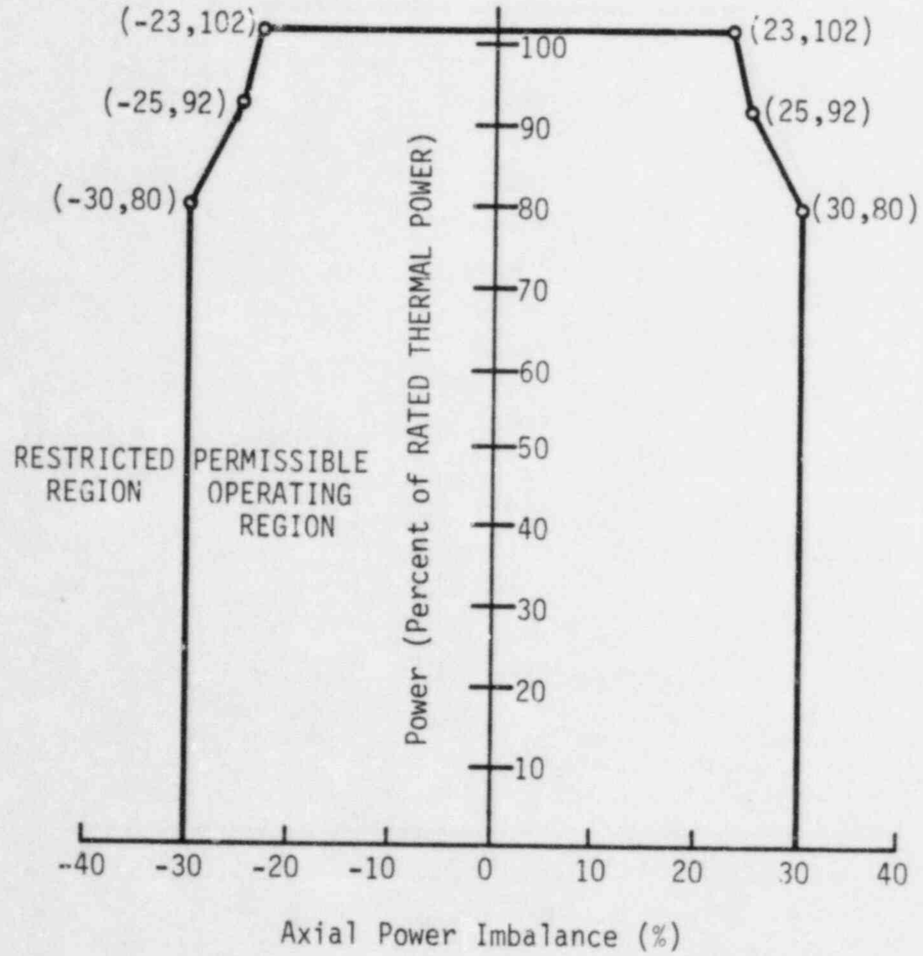


Figure 8-19. Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-1b)

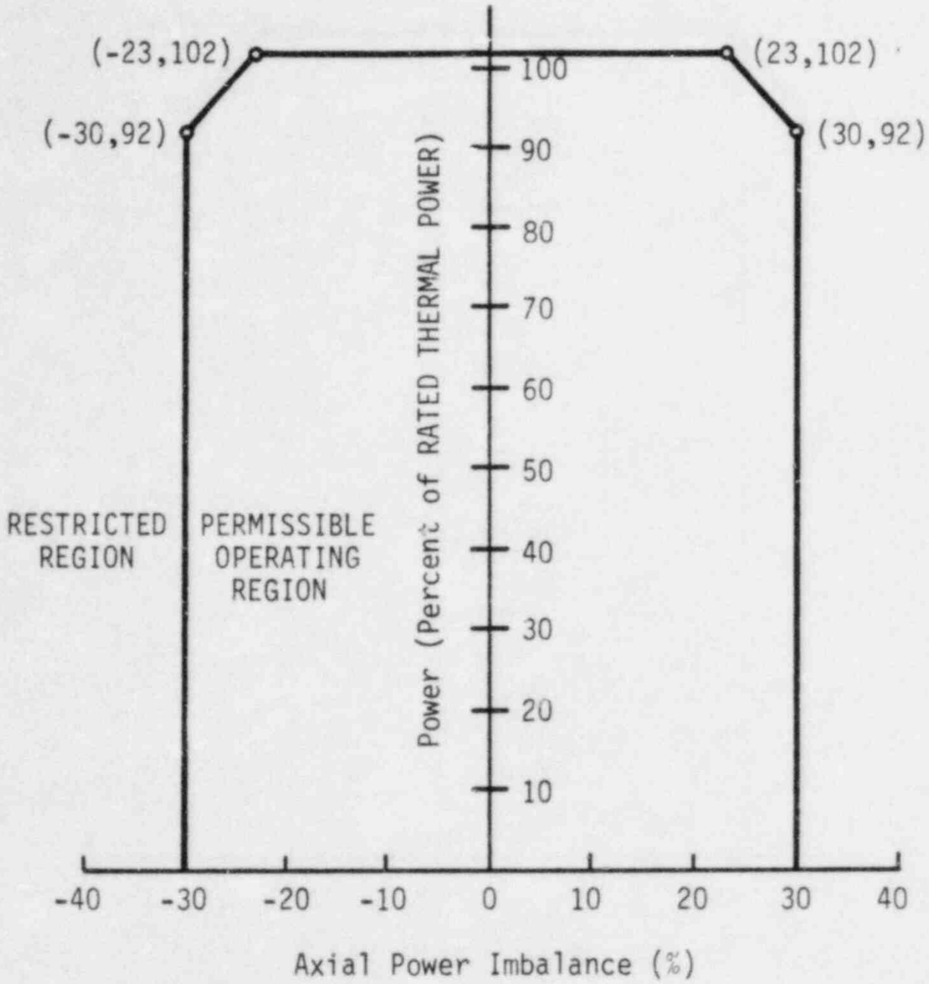


Figure 8-20. Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-1c)

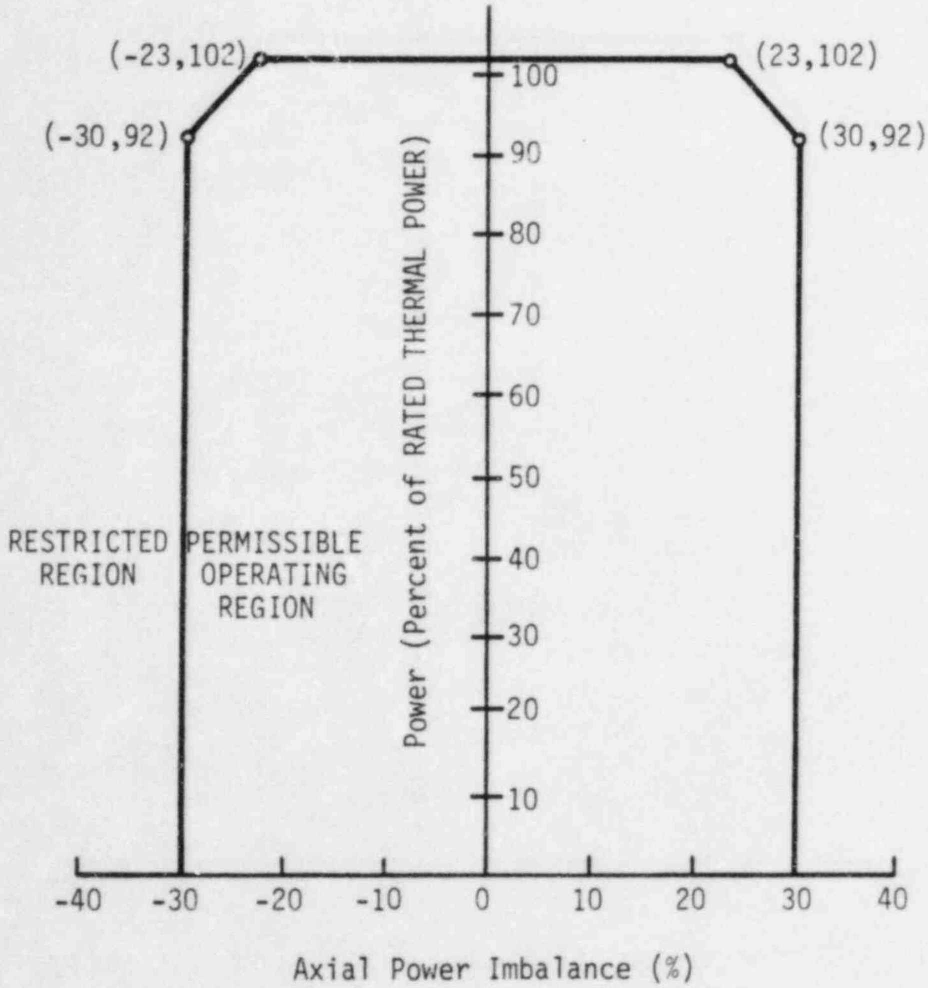


Figure 8-21. Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-1d)

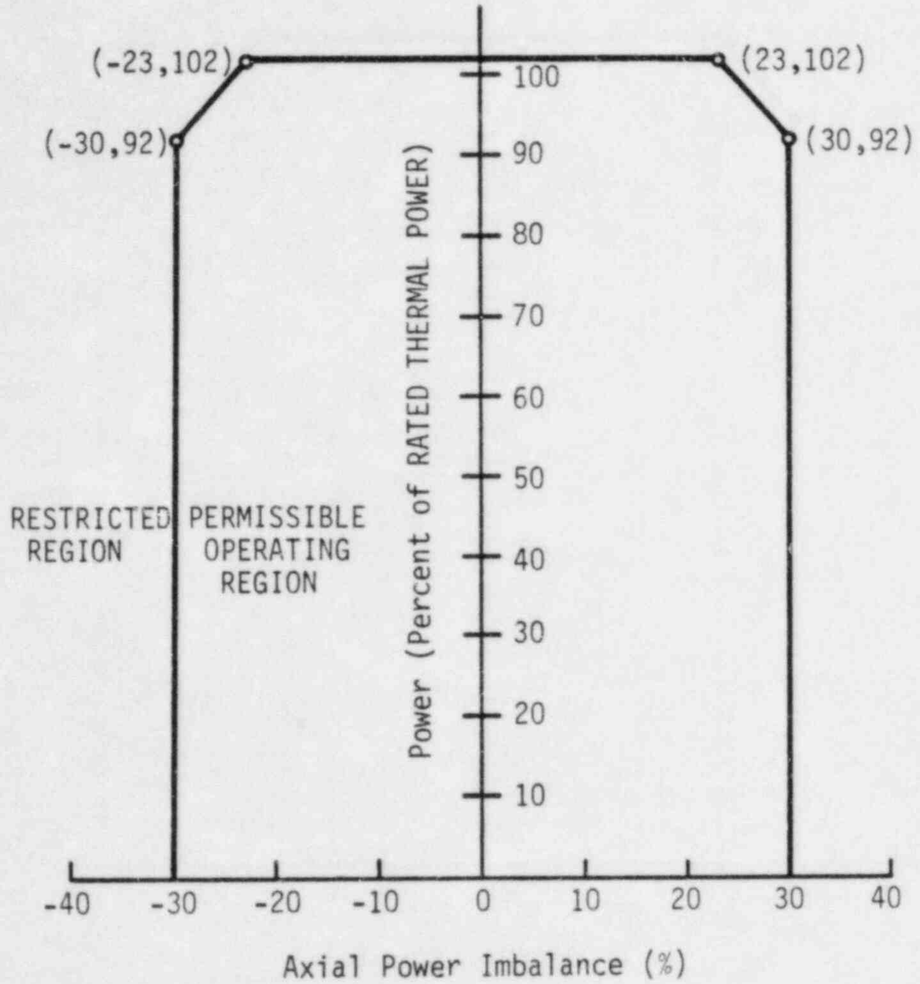


Figure 8-22. Axial Power Imbalance Limits, 0 to 25+10/-0 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-2a)

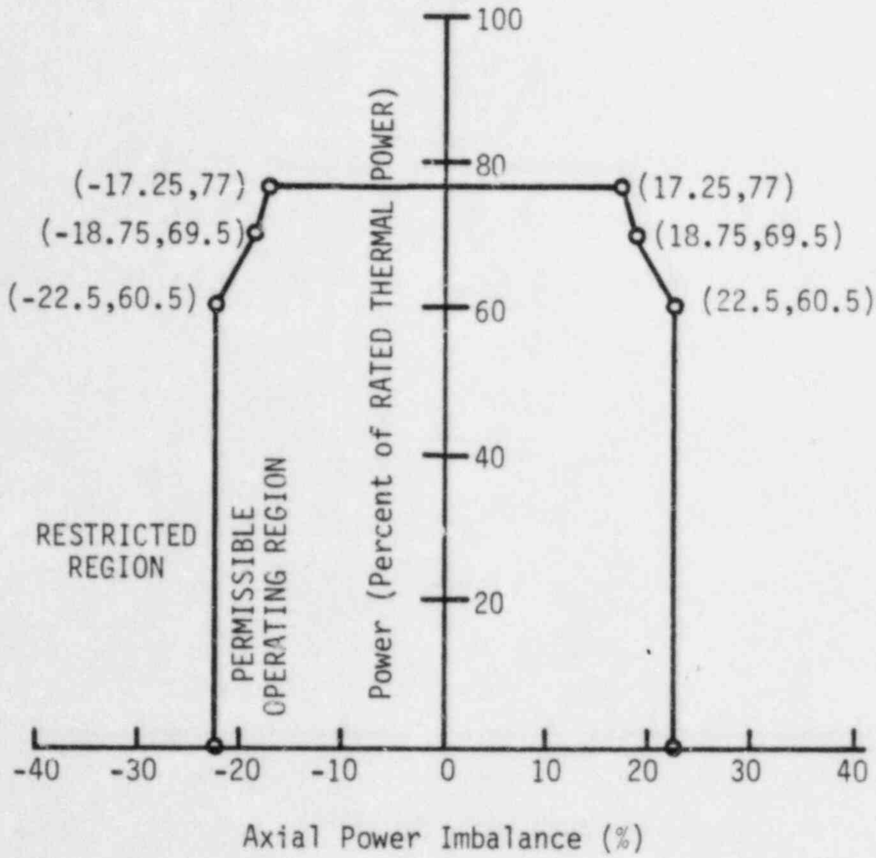


Figure 8-23. Axial Power Imbalance Limits, 25+10/-0 to 200 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-2b)

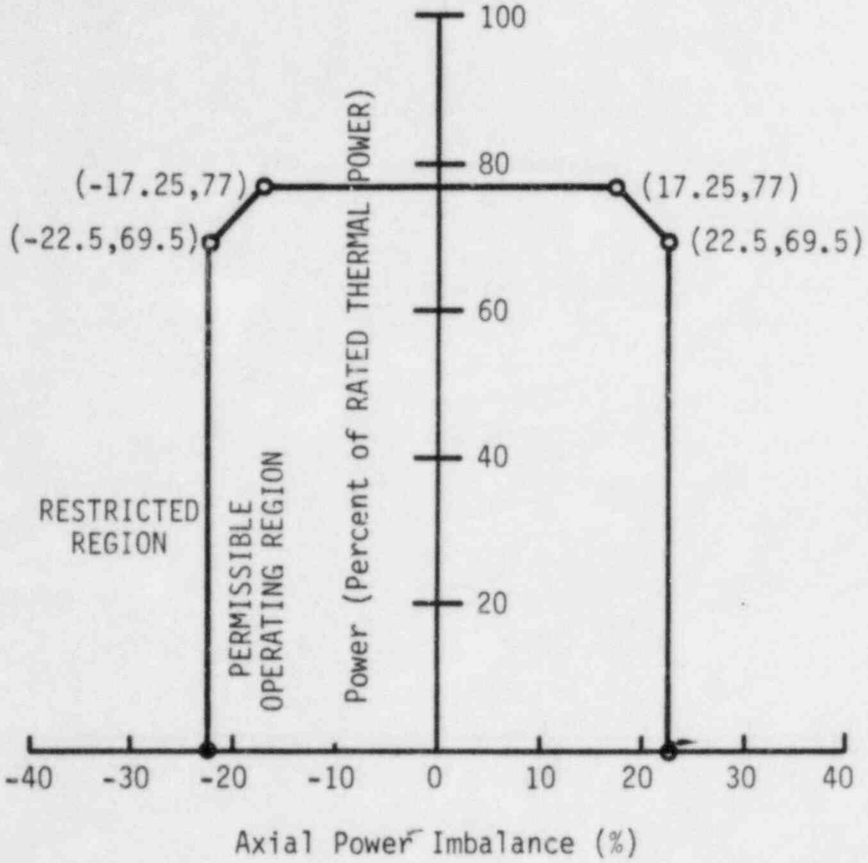


Figure 8-24. Axial Power Imbalance Limits, 200 ±10 to 330 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-2c)

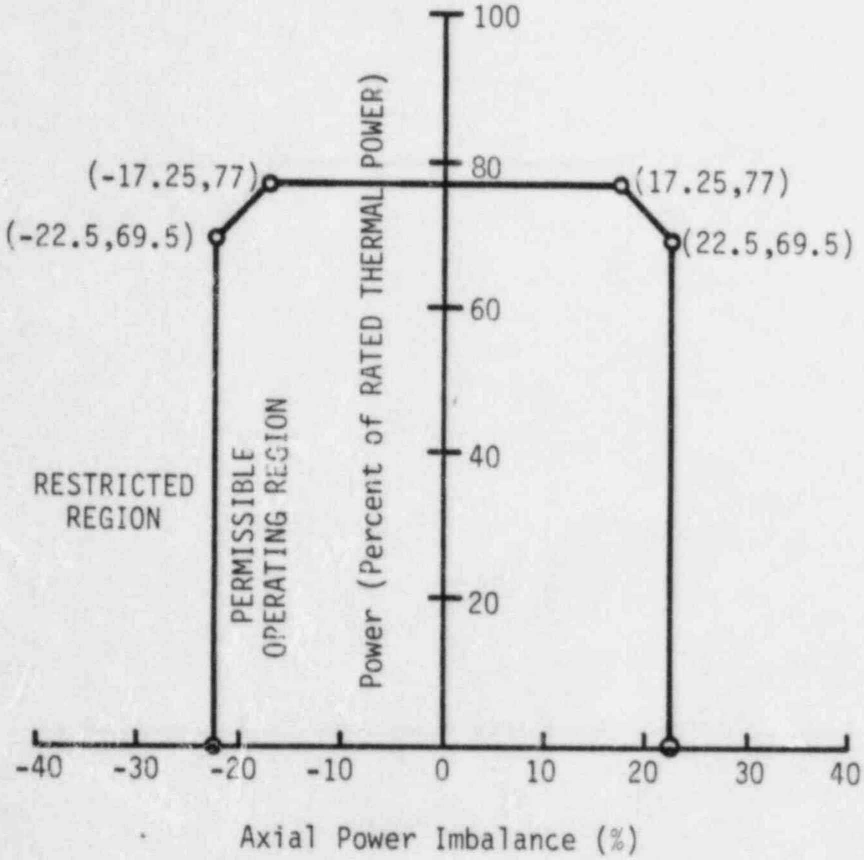


Figure 8-25. Axial Power Imbalance Limits, 330 ±10 to 390 ±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.2-2d)

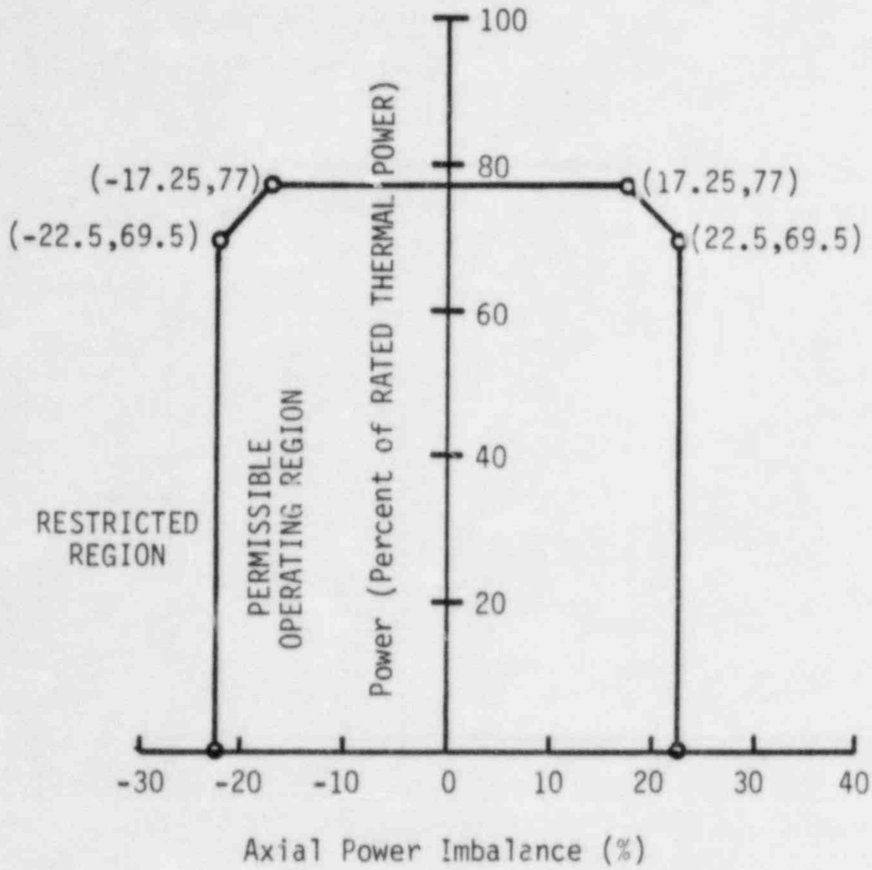
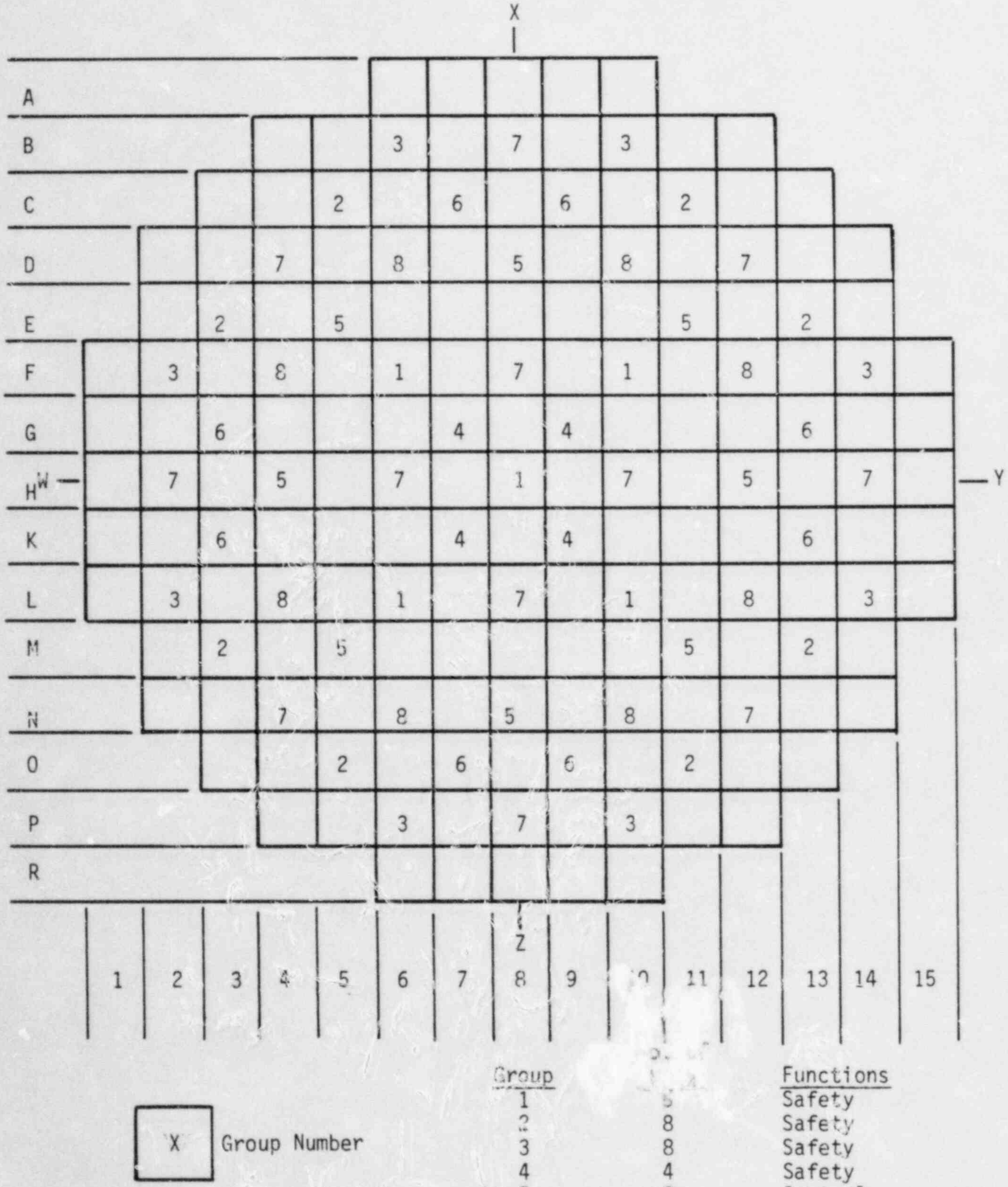


Figure 8-26. Control Rod Core Locations and Group Assignments -- Davis-Besse 1, Cycle 5 (Tech. Spec. Figure 3.1-4)



Group	[illegible]	Functions
1	5	Safety
2	8	Safety
3	8	Safety
4	4	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total #	61	

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 confirmation 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 1.66 seconds at the conditions stated above.

It should be noted that safety analysis calculations are based on a rod drop time of 1.40 seconds 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 for data gathering instead of the two-thirds inserted position. The acceptance criterion of 1.40 seconds corrected to a 75% inserted position (by rod insertion versus time correlation) is 1.66 seconds.

9.1.2. Reactor Coolant Flow

Reactor coolant (RC) flow with four reactor coolant pumps (RCPs) running will be measured at HZP steady-state conditions. Acceptance criteria require that the measured flow be within allowable limits.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate. Once 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 in achieving 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 temperature coefficient is measured at approximately the all-rods-out configuration and at the HZP rod insertion limit. The average coolant temperature is varied by ±5°F. During the change in temperature, reactivity feedback is compensated by a discrete change in rod motion; the change is then calculated by the summation of reactivity (obtained from a reactivity calculation on a strip chart recorder) associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-2}$ $(\% \Delta k/k)/°F$ (predicted value obtained from Physics Test Manual curves).

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 the fuel Doppler coefficient of reactivity is added to obtain the moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.9 \times 10^{-2}$ $(\% \Delta k/k)/°F$.

9.2.3. Control Rod Group Reactivity Worth

Control bank group reactivity worths (groups 5, 6, and 7) are measured at HZP conditions using the boron/rod swap method. The boron/rod swap method consists of establishing a deboration rate in the RC system and compensating for the reactivity changes of this deboration by inserting 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:

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

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

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

9.2.4. Ejected Control Rod Reactivity Worth

After CRA groups 7, 6, and 5 have been positioned near the minimum rod insertion limit, the ejected rod is borated to 100% withdrawn and the worth obtained by adding the incremental changes in reactivity by boration.

After the ejected rod has been borated to 100% withdrawn and equilibrium boron established, the ejected rod is then swapped with the controlling rod group and the worth determined by the change in the previously calibrated controlling rod group position. The boron and rod swap values are averaged and error-adjusted to determine ejected rod worth. Acceptance criteria for the ejected rod worth test are as follows:

1. $\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 20.$

2. Measured value (error adjusted) $\leq$ 1.0% $\Delta k/k$.

The predicted ejected rod worth is given in the Physics Test Manual.

9.3. Power Escalation Tests

9.3.1. Core Power Distribution Verification at ~40, ~75, and ~100% FP With Nominal Control Rod Position

Core power distribution tests are performed at ~40, ~75, and ~100 FP. The test at 40% FP is essentially a check on power distribution in the core to identify any abnormalities before escalating to the 75% FP plateau. Rod index is established at a nominal FP rod configuration at which the core power distribution was calculated. APSR position is established to provide

a core power imbalance corresponding to the imbalance at which the core power distribution calculations were performed.

The following acceptance criteria are placed on the 40% FP test:

1. The worst-case maximum linear heat rate must be less than the LOCA limit.

2. The minimum DNBR must be greater than 1.30.

3. The value obtained from the extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than 1.30 or the extrapolated value of imbalance must fall outside the reactor protector system (RPS) power/imbalance/flow trip envelope.

4. The value obtained from the 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 Technical Specifications.

6. The highest measured and predicted radial peaks shall be within the following limits:

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 8.$$

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

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 12.$$

Items 1, 2, 5, 6, and 7 above are established to verify core nuclear and thermal calculational models, thereby verifying the acceptability of data from these models for input to safety evaluations.

Items 3 and 4 establish the criteria whereby escalation to the next power plateau may be accomplished without exceeding the safety limits specified by the safety analysis with regard to DNBR and LHR.

The power distribution tests performed at 75 and 100% FP are identical to the 40% FP test except that core equilibrium xenon is established before

the 75 and 100% FP tests. Accordingly, the 75 and 100% FP measured peak acceptance criteria are as follows:

1. The highest measured and predicted radial peaks shall be within the following limits:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 5.$$

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

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 7.5.$$

9.3.2. Incore Versus Excore Detector Imbalance Correlation Verification at $\sim$40% FP

Imbalances are set up in the core by control rod positioning. Various imbalances are read simultaneously on the incore detectors and excore power range detectors. The excore versus incore detector offset slopes must be at least 1.15. If the excore versus incore detector offset slope criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required gain.

9.3.3. Temperature Reactivity Coefficient at $\sim$100% FP

The average RC temperature is decreased and then increased by about 5F 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 not be positive above 95% FP.

9.3.4. Power Doppler Reactivity Coefficient at $\sim$100% FP

Reactor power is decreased and then increased by about 5% FP. The reactivity change is obtained from the change in controlling rod group

position. Control rod group worth is measured using the fast insert/with-draw method. Reactivity corrections are made for changes in xenon and RC temperature that occur during the measurement. The power Doppler reactivity coefficient is calculated from the measured reactivity change, which is adjusted as stated above, and the measured power change.

The predicted value of the power Doppler reactivity coefficient is given in the Physics Test Manual. Acceptance criteria state that the measured value shall be more negative than -0.55×10^{-2} (% $\Delta k/k$)/% FP.

9.4. Procedure for Use When Acceptance Criteria Are Not Met

If acceptance criteria for any test are not met, an evaluation is performed with participation by B&W technical personnel as required. Further specific actions depend on the evaluation results. These actions can include repeating the tests with more detailed attention to test prerequisites, 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 the evaluation shows that plant safety will not be compromised by such escalation.

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