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DAVIS-BESSE NUCLEAR POWER STATION
UNIT 1, CYCLE 4 - RELOAD REPORT

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CONTENTS

CONTENTS (Cont'd)

List of Tables

List of Figures

Figures (Cont'd)

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 4. 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 4 reactor and fuel parameters related to power capability are summarized in this report and compared to cycle 3. All accidents analyzed in the Davis-Besse Final Safety Analysis Report (FSAR) have been reviewed for cycle 4 operation, and in cases where cycle 4 characteristics were conservative compared to cycle 1, no new analyses were performed.

Retainers[1] 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 4.

The Technical Specifications have been reviewed and modified where required for cycle 4 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 4 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 3, which achieved criticality on August 29, 1982. Power escalation began on September 1, 1982 and full power (2772 MWt) was reached on October 29, 1982. During cycle 3 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 4. The duration of cycle 3 and the planned duration of cycle 4 are 268 and 240 effective full power days (EFPD) respectively.

A quadrant power tilt that was larger than that experienced in previous cycles was measured at the beginning of cycle 3 in quadrant WX. To reduce the potential for tilt amplification, the cycle 4 design minimizes the number of assemblies that are cross-core shuffled. The cycle 4 shuffle pattern is discussed in section 3.

The axial power shaping rods (APSR) were pulled at 200 EFPD to increase the lifetime of cycle 3. The APSR pull coupled with a power coastdown will result in a cycle 3 length of approximately 268 EFPD.

3. GENERAL DESCRIPTION

The Davis-Besse Unit 1 reactor core is described in detail in chapter 4 of the FSAR[2] for the unit. The cycle 4 core consists of 177-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, and 6 have a constant nominal fuel loading of 468.25 kg of uranium. Batches 1D and 2B have 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.

Section 2 addresses the tilt amplification that occurred in cycle 3. Reference 3 provides guidelines for a fuel shuffle method that reduces the number of assemblies that are cross-core shuffled. The cycle 4 design expanded upon this method so that only eight fuel assemblies are cross-core shuffled. This will minimize any carryover effects from tilts in previous cycles.

Figure 3-1 is the core loading diagram for Davis-Besse Unit 1, cycle 4. Twenty-five batch 1C assemblies and 60 batch 3 assemblies will be discharged at the end of cycle 3. The batch 4 and 5 assemblies will be shuffled to their cycle 4 locations. Batches 4 and 5A have an initial uranium-235 enrichment of 3.04 wt %. Batch 5B has an initial enrichment of 2.99 wt %. Seventeen batch 1D assemblies with an initial enrichment of 1.98 wt % and 20 batch 2B assemblies with an initial enichment of 2.63 wt % will be reinserted in cycle 4. A feed batch consisting of 48 batch 6 assemblies with uranium enrichment of 2.99 wt % will be inserted in cycle 4 and occupy the periphery of the core. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 4 and its initial enrichment.

Cycle 4 is operated in a feed-and-bleed mode. The core reactivity control is supplied mainly by soluble boron and supplemented by 53 full-length Ag-In-Cd control rod assemblies (CRAs). 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 4 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 4 are identical to those of reference cycle 3.

Figure 3-1. Davis-Besse Cycle 4 Full Core Loading Diagram

FUEL TRANSFER CANAL →

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

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						6	6	6	6	6					
B				6	6	6	2B L4 Cy2	5B R8	2B L12 Cy2	6	6	6			
C			6	5B M2	4 K3	1D D8 Cy1	5B N3	4 F10	5B N13	1D F8 Cy1	4 K13	5B M14	6		
D		6	5B B11	5B A7	4 L5	2B C6 Cy2	4 D5	5B D8	4 D11	2B C10 Cy2	4 L11	5B G15	5B B5	6	
E		6	4 C9	4 E10	5B G1	4 B6	5A L1	4 L8	5A L15	4 B10	5B A9	4 E6	4 C7	6	
F	6	6	1D H6 Cy1	2B F3 Cy2	4 F2	4 G9	1D E5 Cy1	5B B12	1D G9 Cy1	4 K9	4 F14	2B F13 Cy2	1D H12 Cy1	6	6
G	6	2B D10 Cy2	5B C12	4 E4	5A A10	1D G7 Cy1	5B B4	2B C11 Cy2	5B D14	1D E11 Cy1	5A A6	4 E12	5B C4	2B D6 Cy2	6
H	6	5B H15	4 F6	5B H4	4 H10	5B D2	2B E3 Cy2	1D L6 Cy1	2B M13 Cy2	5B N14	4 H6	5B H12	4 L10	5B H1	6
K	6	2B N10 Cy2	5B O12	4 M4	5A R10	1D M5 Cy1	5B N2	2B O5 Cy2	5B P12	1D K9 Cy1	5A R6	4 M12	5B O4	2B N6 Cy2	6
L	6	6	1D H4 Cy1	2B L3 Cy2	4 L2	4 G7	1D K7 Cy1	5B P4	1D M11 Cy1	4 K7	4 L14	2B L13 Cy2	1D H10 Cy1	6	6
M		6	4 O9	4 M10	5B R7	4 P6	5A F1	4 F8	5A F15	4 P10	5B K15	4 M6	4 O7	6	
N		6	5B P11	5B K1	4 F5	2B O6 Cy2	4 N5	5B N8	4 N11	2B O10 Cy2	4 F11	5B R9	5B P5	6	
O			6	5B E2	4 G3	1D L8 Cy1	5B D3	4 L6	5B D13	1D N8 Cy1	4 G13	5B E14	6		
P				6	6	6	2B F4 Cy2	5B A8	2B F12 Cy2	6	6	6			
R						6	6	6	6	6					

Batch
Location in previous cycle.
NOTE: Cy1 = reinserted from cycle 1
Cy2 = reinserted from cycle 2

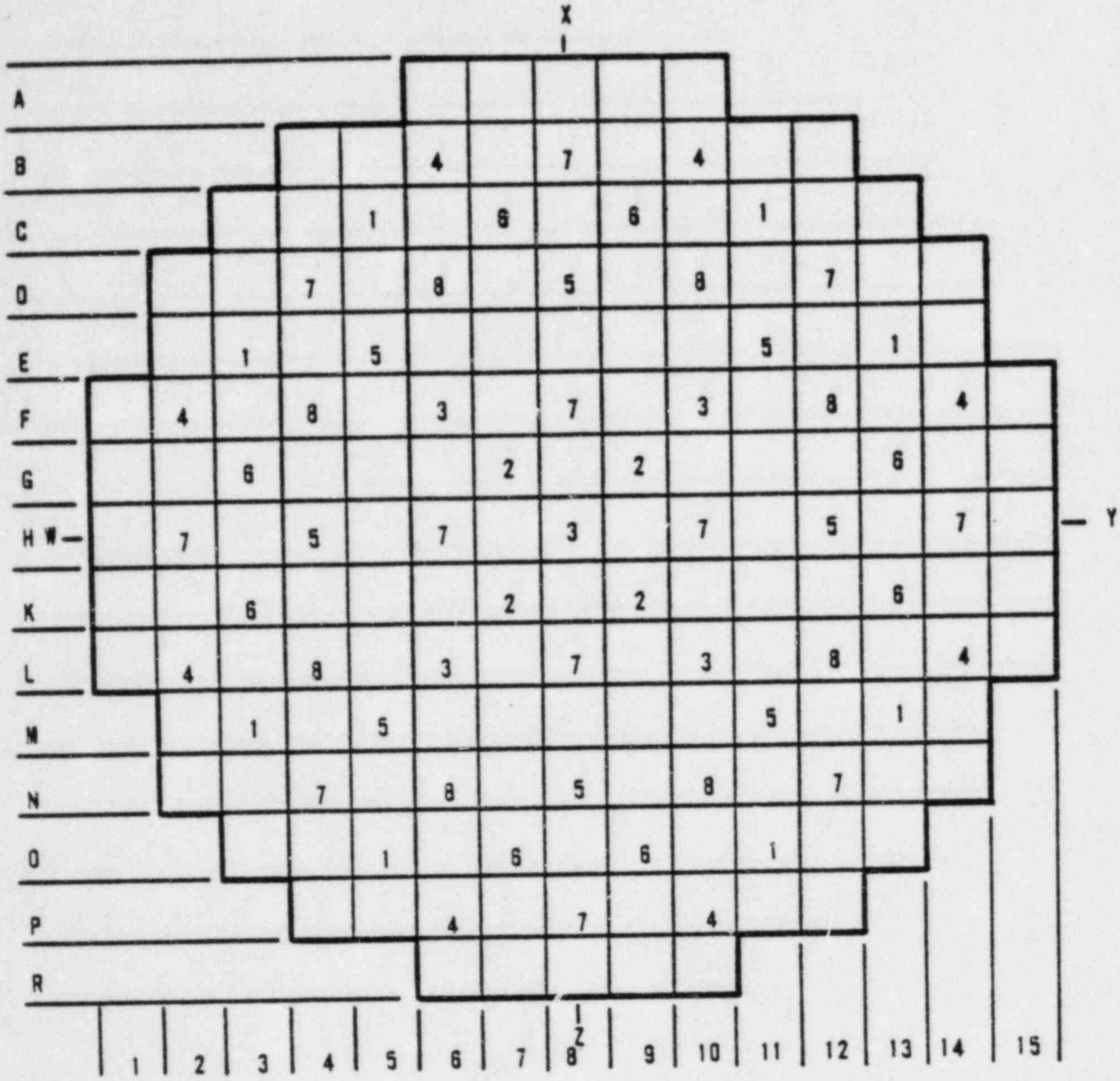
Figure 3-2. Enrichment and Burnup Distribution for Davis-Besse Unit 1, Cycle 4

	8	9	10	11	12	13	14	15
H	1.98 12,644	2.63 24,006	2.99 6,602	3.04 19,798	2.99 11,579	3.04 19,764	2.99 8,340	2.99 0
K	2.63 24,006	2.99 6,586	1.98 13,254	3.04 6,533	3.04 19,646	2.99 10,184	2.63 23,008	2.99 0
L	2.99 6,602	1.98 12,983	3.04 19,670	3.04 16,248	2.63 23,499	1.98 13,416	2.99 0	2.99 0
M	3.04 19,828	3.04 6,527	3.04 16,237	2.99 8,012	3.04 17,950	3.04 21,080	2.99 0	
N	2.99 11,568	3.04 19,617	2.63 23,501	3.04 17,930	2.99 8,007	2.99 8,906	2.99 0	
O	3.04 19,764	2.99 10,141	1.98 13,560	3.04 21,087	2.99 8,889	2.99 0		
P	2.99 8,335	2.63 23,007	2.99 0	2.99 0	2.99 0			
R	2.99 0	2.99 0	2.99 0					

x.xx — Initial enrichment

xx,xxx — BOC burnup, MWt

Figure 3-3. Control Rod Locations for Davis-Besse
Unit 1, Cycle 4



GROUP	NO. OF RODS	FUNCTIONS
1	8	SAFETY
2	4	SAFETY
3	5	SAFETY
4	8	SAFETY
5	8	CONTROL
6	8	CONTROL
7	12	CONTROL
8	8	APSRs
	TOTAL # 61	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of FAs and pertinent fuel parameters for Davis-Besse Unit 1, cycle 4 are listed in Table 4-1. All Mark-B (Mk-B) FAs are identical in concept and are mechanically interchangeable. Retainer assemblies will be used on two FAs that contain the regenerative neutron sources. The justification for the design and use of retainer assemblies is described in references 1 and 4.

4.2. Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1. Cladding Collapse

Due to its previous incore exposure time, the fuel of batch 2B is more limiting than batches 1D, 4, 5A, 5B, and 6. The batch 2B assembly power histories were analyzed to determine the most limiting three-cycle power history for creep collapse. This power history was compared to a generic analysis to ensure that creep ovalization will not affect the fuel performance during Davis-Besse Unit 1, cycle 4. The generic analysis was based on reference 5 and is applicable to the batch 2B design.

The creep collapse analysis (Table 4-1) predicts a collapse time longer than 35,000 effective full power hours (EFPH), which is longer than the expected residence time of 21,840 EFPH.

4.2.2. Cladding Stress

The Davis-Besse Unit 1, cycle 4 stress parameters are enveloped by a conservative fuel rod stress analysis. The methods used for the analysis of cycle 4 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 4 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 4 core is thermally similar. The cycle 4 thermal analyses represent a change in analytical method. The analyses for the incoming batch 6 fuel have been performed with the TACO2[6] code using the analysis methodology described in reference 7. This methodology uses nominal undensified input parameters provided in Table 4-2. Densification effects are accounted for in the TACO2 densification model. The TACO2 analyses also apply to the batch 5B fuel since this fuel is identical in design to the batch 6 fuel. Reinserted FAs from batches 1D, 2B, 4, and 5A were evaluated using TAFY3[8] analyses performed for prior cycles.

The thermal design evaluation for the cycle 4 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 and 6 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 4 is less than 42,000 MWd/mtU.

4.4. Material Compatibility

The compatibility of all possible fuel cladding - coolant assembly interactions for batch 6 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 February 28, 1983, 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 Incore	Discharged	Cumulative net electric output,[b] MWh
Oconee 1	7	48,010	40,000	41,241,515
Oconee 2	6	27,240	36,800	36,475,957
Oconee 3	7	22,975	35,450	37,022,597
Three Mile Island 1	5	25,000	32,400	23,840,053
Arkansas Nuclear Unit 1	6	23,160	36,540	34,949,454
Rancho Seco	5	37,883	37,730	29,933,402
Crystal River 3	4	28,110	29,900	22,081,044
Davis-Besse	3	28,820	25,326	12,898,260

(a)As of February 28, 1983.

(b)As of October 31, 1982.

Table 4-1. Fuel Design Parameters

	Batch					
	1D	2B	4	5A	5B	6
FA type	Mk-B4A	Mk-B4A	Mk-B4A	Mk-B4A	Mk-B4A	Mk-B4A
Number of assemblies	17	20	44	8	40	48
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	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Undensified active fuel length, in.	143.5	143.5	143.44	143.44	143.20	143.20
Fuel pellet (mean) dia., in.	0.3675	0.3675	0.3697	0.3697	0.3686	0.3686
Fuel pellet initial density, % TD mean	96	96	94	94	95	95
Initial fuel enrichment, wt % ^{235}U	1.98	2.63	3.04	3.04	2.99	2.99
Estimated residence time, EFPH	14,736	21,840	19,104	12,000	12,000	5,760
Cladding collapse time, EFPH	>35,000	>35,000	>35,000	>35,000	>35,000	>35,000

Table 4-2. Fuel Thermal Analysis Parameters

	Batch			
	1D/2B	4/5A	5B	6
Number of assemblies	17/20	44/8	40	48
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	1.98/2.63	3.04	2.99	2.99
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.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycle 3 with those of cycle 4. These values were generated using PDQ07[9-11] for both cycles. Since the core has not yet reached an equilibrium cycle, differences between the cycles in core physics parameters are to be expected. Figure 5-1 illustrates a representative relative power distribution for the BOC at full power (FP) with equilibrium xenon and group 8 inserted.

Due to the difference in design cycle lengths, the critical boron concentrations for cycle 4 differ from those of reference cycle 3. Because of different isotopic distributions, cycle 4 control rod worths, ejected rod worths, and stuck rod worths differ from those of cycle 3. 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 4 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 4 moderator coefficients and the power deficits from hot zero power (HZP) to hot full power (HFP) are similar to those for cycle 3. The differential boron and xenon worths are also similar in both cycles. The effective delayed neutron fraction for cycle 4 show a decrease with burnup (also shown in reference cycle 3).

5.2. Changes in Nuclear Design

There are no significant core design changes between the reference cycle and the cycle 4 designs, although the cycle 4 core was shuffled in a manner to minimize the carryover effect on quadrant tilt. 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.

Table 5-1. Davis-Besse Unit 1, Cycle 4 Physics Parameters

	Cycle 3	Cycle 4
Cycle length, EFPD[(a)]	268	240
Cycle burnup, MWd/mtU	8,929	8,014
Average core burnup - EOC[(b)], MWd/mtU	20,191	18,924
Initial core loading, mtU	83.2	83.0
Critical boron - BOC, No Xe, ppm		
HZP[(c)] } Group 8 inserted	1,231	1,250
HFP[(c)] } Group 8 inserted	1,015	1,042
Critical boron - EOC, Eq. Xe, ppm		
HZP[(d)]	284	337
HFP[(d)]	10	42
Control rod worths - HFP, BOC, % Δk/k		
Group 6	0.93	1.02
Group 7	1.52	1.73
Group 8	0.31	0.27
Control rod worths - HFP, EOC % Δk/k		
Group 7	1.53	1.74
Group 8	NA	0.35
Max ejected rod worth - HZP, % Δk/k (location)		
BOC } Groups 5-8 inserted	0.78 (N-12)	0.85[(e)] (N-12)
EOC[(d)] } Groups 5-7 inserted	0.72 (N-12)	0.85 (N-12)
Max stuck rod worth - HZP, % Δk/k (location)		
BOC	1.44 (N-12)	1.70 (L-14)
EOC	1.25 (L-14)	1.54 (L-14)
Power deficit, HZP to HFP, Eq. Xe, % Δk/k		
BOC (4 EFPD)	-1.79	-1.77
EOC	-2.34	-2.36
Doppler coeff - HFP, 10^{-5} Δk/k/°F		
BOC, No Xe, 1042 ppm, Group 8 inserted	-1.46	-1.47
EOC, Eq. Xe, 10 ppm, Group 8 inserted	-1.63	-1.58
Moderator coeff - HFP, 10^{-4} Δk/k/°F		
BOC, No Xe, 1042 ppm, Group 8 inserted	-1.13	-1.00
EOC, Eq Xe, 10 ppm, Group 8 inserted	-2.89	-2.87
Boron worth - HFP, ppm/% Δk/k		
BOC (1042 ppmb)	110	108
EOC (10 ppm)	97	97
Xenon worth - HFP, % Δk/k		
BOC (4 EFPD)	2.67	2.67
EOC (equilibrium)	2.73	2.74

Table 5-1. (Cont'd)

	Cycle 3	Cycle 4
Effective delayed neutron fraction - HFP		
BOC	0.00595	0.00598
EOC	0.00530	0.00538

[(a)]EFPD denotes effective full power days.

[(b)]EOC denotes end of cycle.

[(c)]HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (584F core T_{avg}).

[(d)]Group 8 is withdrawn at EOC 3 and inserted at EOC 4.

[(e)]Ejected rod worth at the rod insertion limit.

Table 5-2. Shutdown Margin Calculation for Davis-Besse Unit 1, Cycle 4

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP[a]	7.69	7.89
Worth reduction due to burnup of poison material	-0.16	-0.19
Maximum stuck rod, HZP	-1.70	-1.54
Net worth	5.83	6.16
Less 10% uncertainty	-0.58	-0.62
Total available worth	5.25	5.54
Required Rod Worth		
Power deficit, HFP[a] to HZP	1.77	2.36
Max allowable inserted rod worth	0.51	0.64
Flux redistribution	0.73	1.15
Total required worth	3.01	4.15
Shutdown Margin		
Total available minus total required	2.24	1.39

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

[a]HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (584F core T_{avg}).

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

	8	9	10	11	12	13	14	15
H	0.816	0.885	1.293	1.142	1.179	1.043	1.155	0.986
K	0.889	1.206	0.982	1.286	1.000	1.108	0.909	0.937
L	1.298	0.988	1.034	1.045	8 0.725	0.829	1.240	0.774
M	1.143	1.285	1.040	1.155	0.967	0.909	1.022	
N	1.180	1.001	8 0.724	0.967	1.184	1.029	0.714	
O	1.004	1.109	0.829	0.911	1.035	0.811		
P	1.156	0.910	1.241	1.024	0.715			
R	0.986	0.937	0.775					

X Inserted rod group number

X.XXX Relative power density

(a)Calculated results from two-dimensional pin-by-pin PDQ07[9].

6. THERMAL-HYDRAULIC DESIGN

The fresh batch 6 fuel is hydraulically and geometrically similar to the other fuel loaded into the cycle 4 core. The thermal-hydraulic design evaluation supporting cycle 4 operation is based on the methods and models described in references 13 and 14. The cycle 4 thermal-hydraulic design is identical to that of cycle 3. The thermal-hydraulic design conditions for cycles 3 and 4 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[15], 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 4.

Table 6-1. Davis-Besse Cycles 3 and 4 Thermal-Hydraulic Design Conditions

Design power level, MWt	2772
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[2] accident analysis has been examined with respect to changes in the cycle 4 parameters to determine the effects of the cycle 4 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 13.

The radiological dose consequences of the FSAR chapter 15 accidents based on cycle 4 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.

7.2. 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 4 are given in Table 4-2. A comparison of the cycle 4 thermal-hydraulic maximum design conditions to the previous cycle values is presented in Table 6-1. A comparison of the key kinetics parameters from the FSAR and cycle 4 is provided in Table 7-1.

A generic loss-of-coolant accident (LOCA) analysis for B&W 177-FA raised-loop nuclear steam systems (NNSs) has been performed using the Final Acceptance Criteria ECCS Evaluation Model.[16] 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 16 provide conservative results for the operation of Davis-Besse Unit 1, cycle 4 fuel. A tabulation showing the bounding values for allowable LOCA peak LHRs for Davis-Besse Unit 1, cycle 4 fuel are provided in Table 7-2.

It is concluded by the examination of cycle 4 core thermal, thermal-hydraulic, and kinetics properties, with repsect 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 4. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 4 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 4 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 4 value
BOL[a] Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.28	-1.47
EOL[b] Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.45[c]	-1.58
BOL moderator coeff, 10^{-4}, $\Delta k/k/°F$	+0.13	-1.00
EOL moderator coeff, 10^{-4}, $\Delta k/k/°F$	-3.0	-2.87
All rod bank worth (HZP), % $\Delta k/k$	10.0	7.69
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	108
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.46
Max dropped rod worth (HFP), % $\Delta k/k$	0.65	0.20
Initial boron conc (HFP), ppm	1407	1042

(a)BOL denotes beginning of life.

(b)EOL denotes end of life.

(c)-1.77×10^{-5} $\Delta k/k/°F$ 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 24 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 MODIFICATION TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 4 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 8-20 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	$\leq$104.94% of RATED THERMAL POWER with four pumps operating	$\leq$104.94% of RATED THERMAL POWER with four pumps operating#
	$\leq$79.85% of RATED THERMAL POWER with three pumps operating	$\leq$79.85% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	$\leq$618°F	$\leq$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)	$\geq$1983.4 psig	$\geq$1983.4 psig* $\geq$1983.4 psig**
6. RC high pressure	$\leq$2300 psig	$\leq$2300.0 psig* $\leq$2300.0 psig**
7. RC pressure-temperature(1)	$\geq$(12.60 T_{out} °F - 5662.2) psig	$\geq$(12.60 T_{out} °F - 5662.2) psig#
8. High flux/number of RC pumps on(1)	$\leq$55.1% of RATED THERMAL POWER with one pump operating in each loop	$\leq$55.1% of RATED THERMAL POWER with one pump operating in each loop#
	$\leq$0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	$\leq$0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	$\leq$0.0 of RATED THERMAL POWER with no pumps operating or only one pump operating	$\leq$0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	$\leq$4 psig	$\leq$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 1D, 2B, 4 and 5A and 20.5 kW/ft for batches 5B and 6.

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.

2.2. LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1. REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation trip setpoints specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The trip setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The high flux trip setpoint of $\leq$5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.94% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RC High Temperature

The RC high temperature trip $\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 106.9% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 93.5% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.9% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 70.2% 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.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux trip setpoint. The trip setpoint for RC high pressure, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2435 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure , 1983.4 psig, and RC pressure-temperature (12.60 t_{out} - 5662.2) psig, trip setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

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, 3.1-2b, 3.1-2c, 3.1-3a, 3.1-3b and 3.1-3c.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

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

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

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

---

*See Special Test Exceptions 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, 3.1-5b, 3.1-5c, 3.1-5d, 3.1-5e and 3.1-5f.

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 $k_{eff} \geq 1.0$.

3/4.2. POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

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

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.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 2.93; \quad F^N_{\Delta H} \leq 1.71$$

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than ±6.5% (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

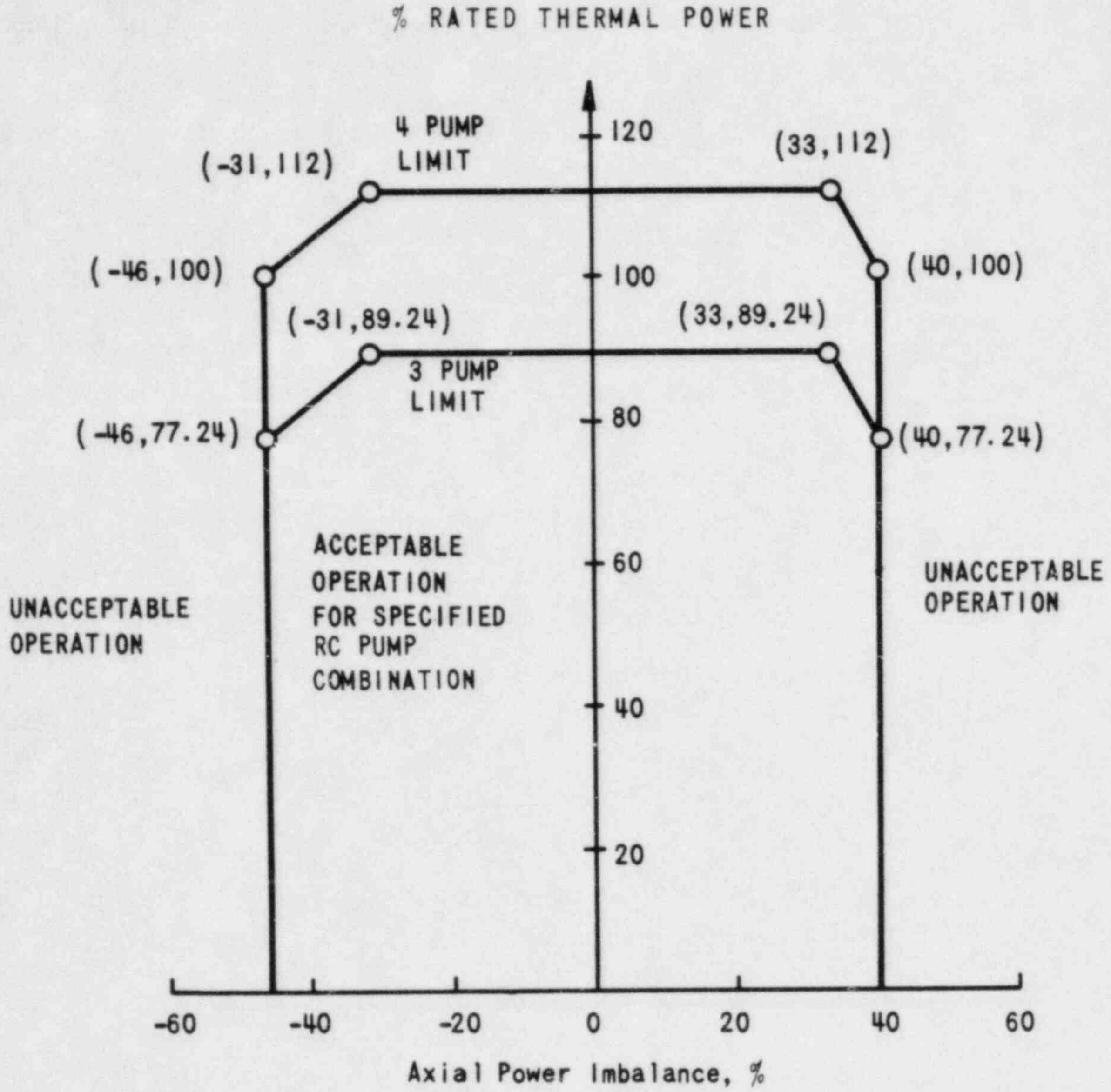
3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between the limits specified in Specification 3.2.1.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

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



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)

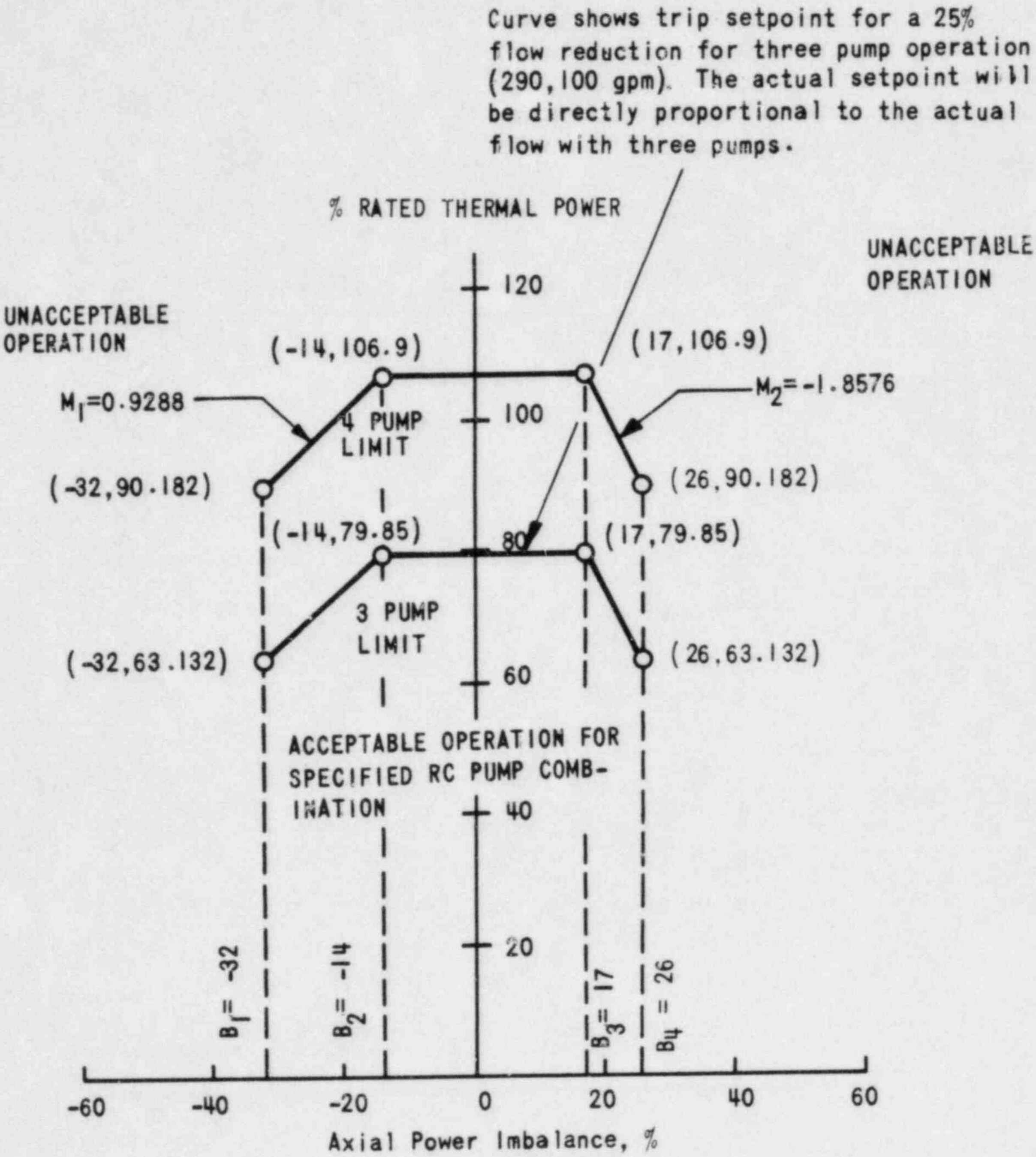


Figure 8-2a. Reactor Core Safety Limit
(Tech Spec Figure 2.1-1)

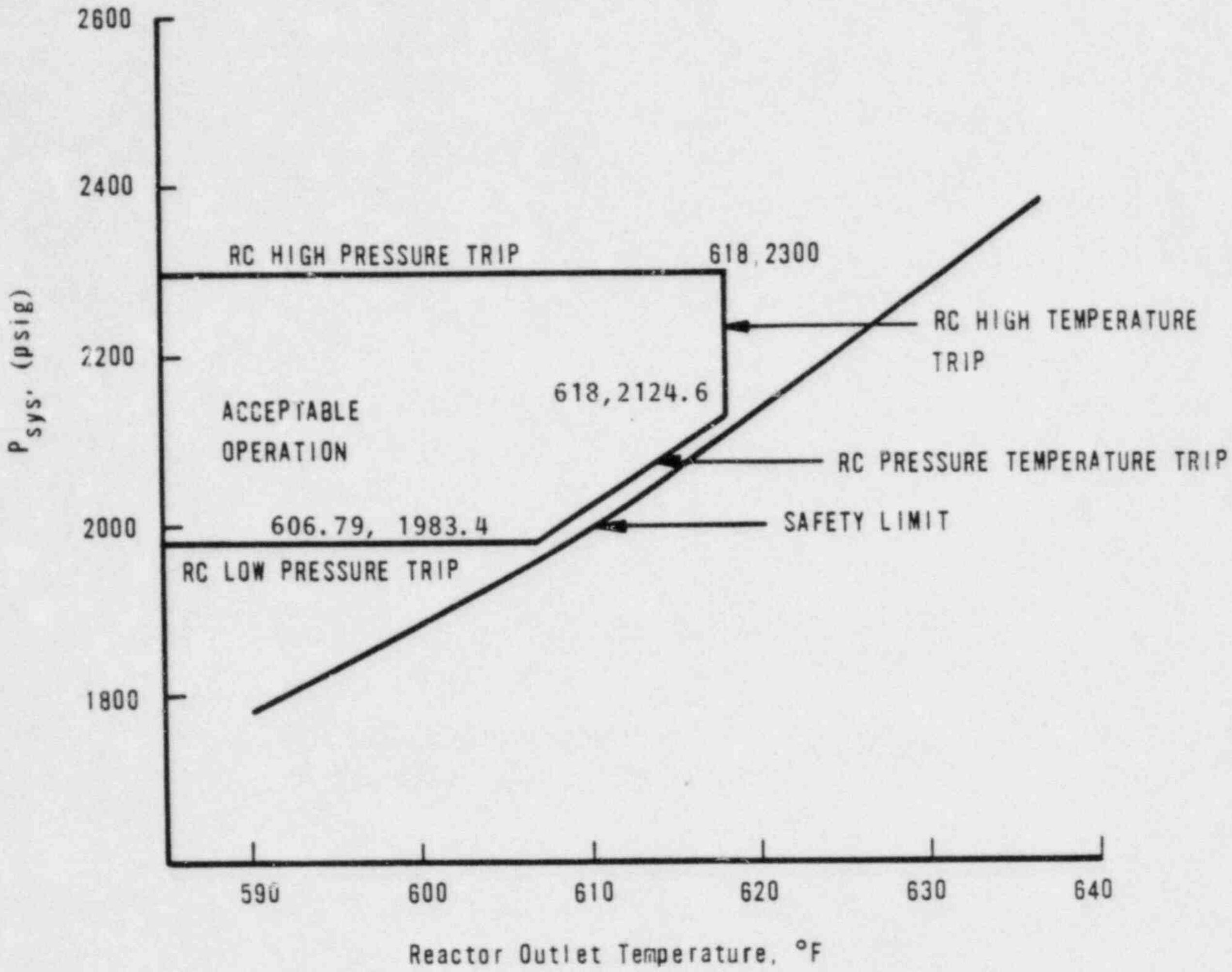


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Figure 8-3. Regulating Group Position Limits, 0 to 24+10/-0 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-2a)

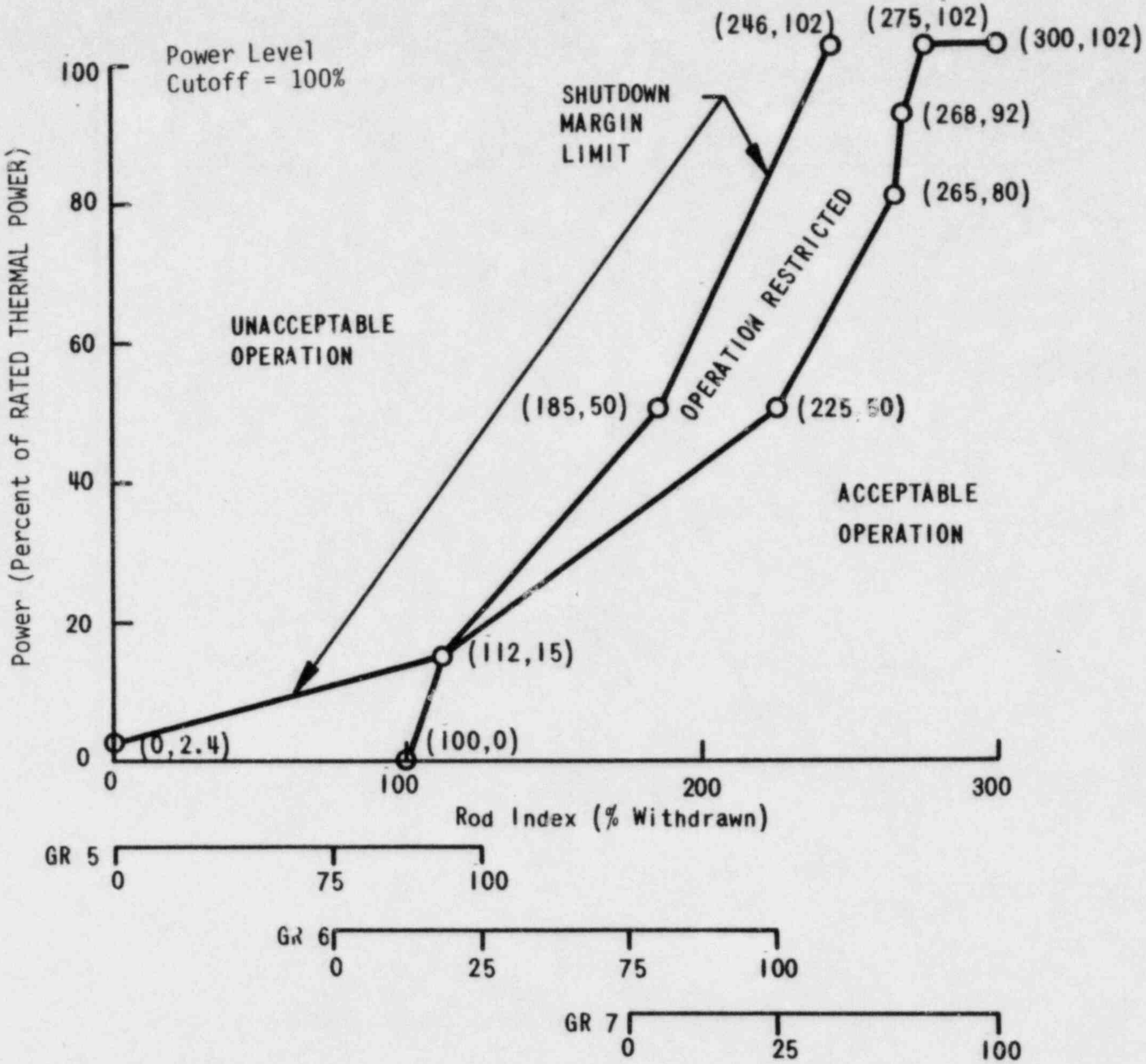


Figure 8-4. Regulating Group Position Limits, 24+10/-0 to 150±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-2b)

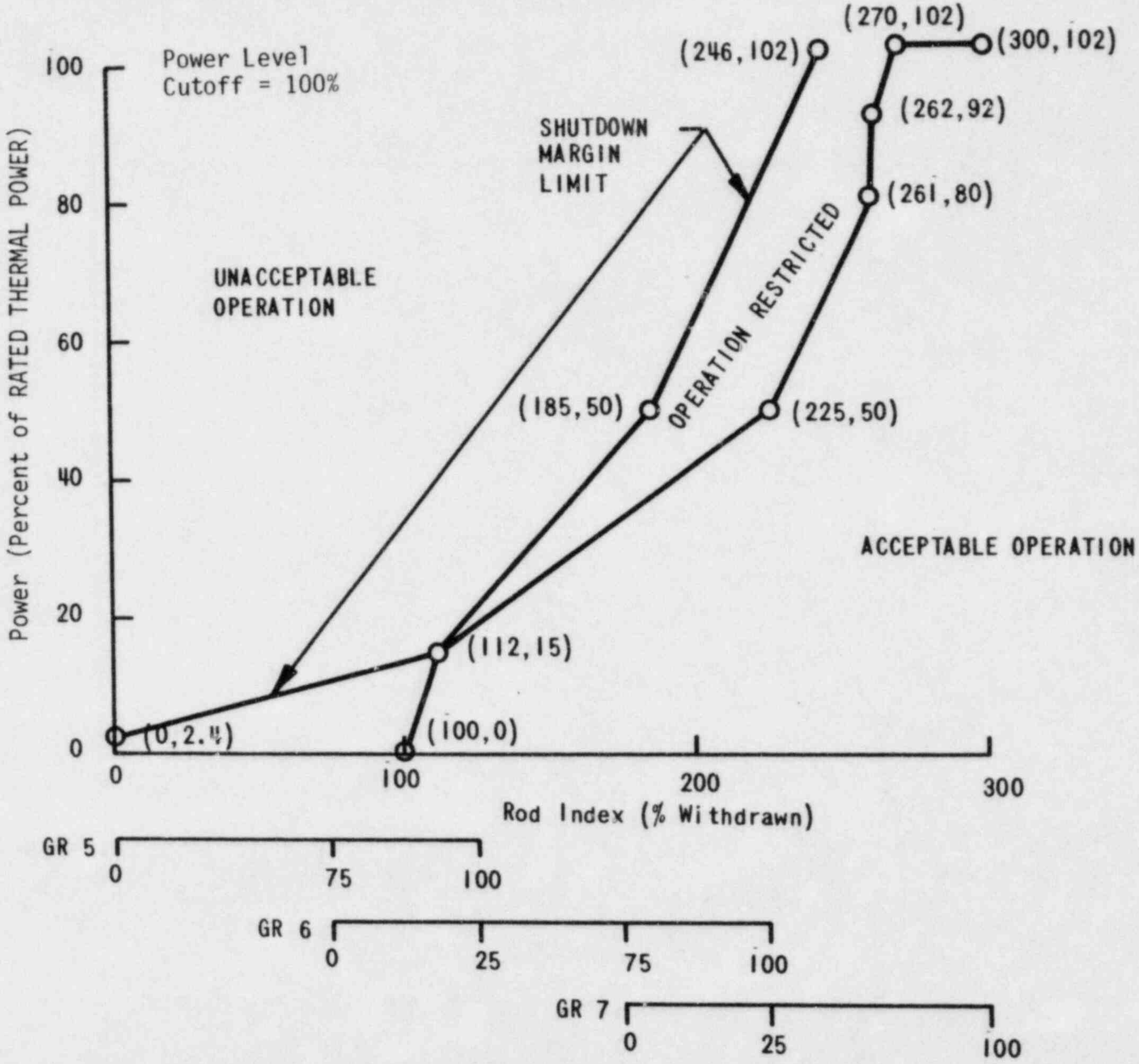


Figure 8-5. Regulating Group Position Limits After 150 ±10 EFPD,
Four RC Pumps — Davis-Besse 1, Cycle 4
(Tech. Spec. Figure 3.1-2c)

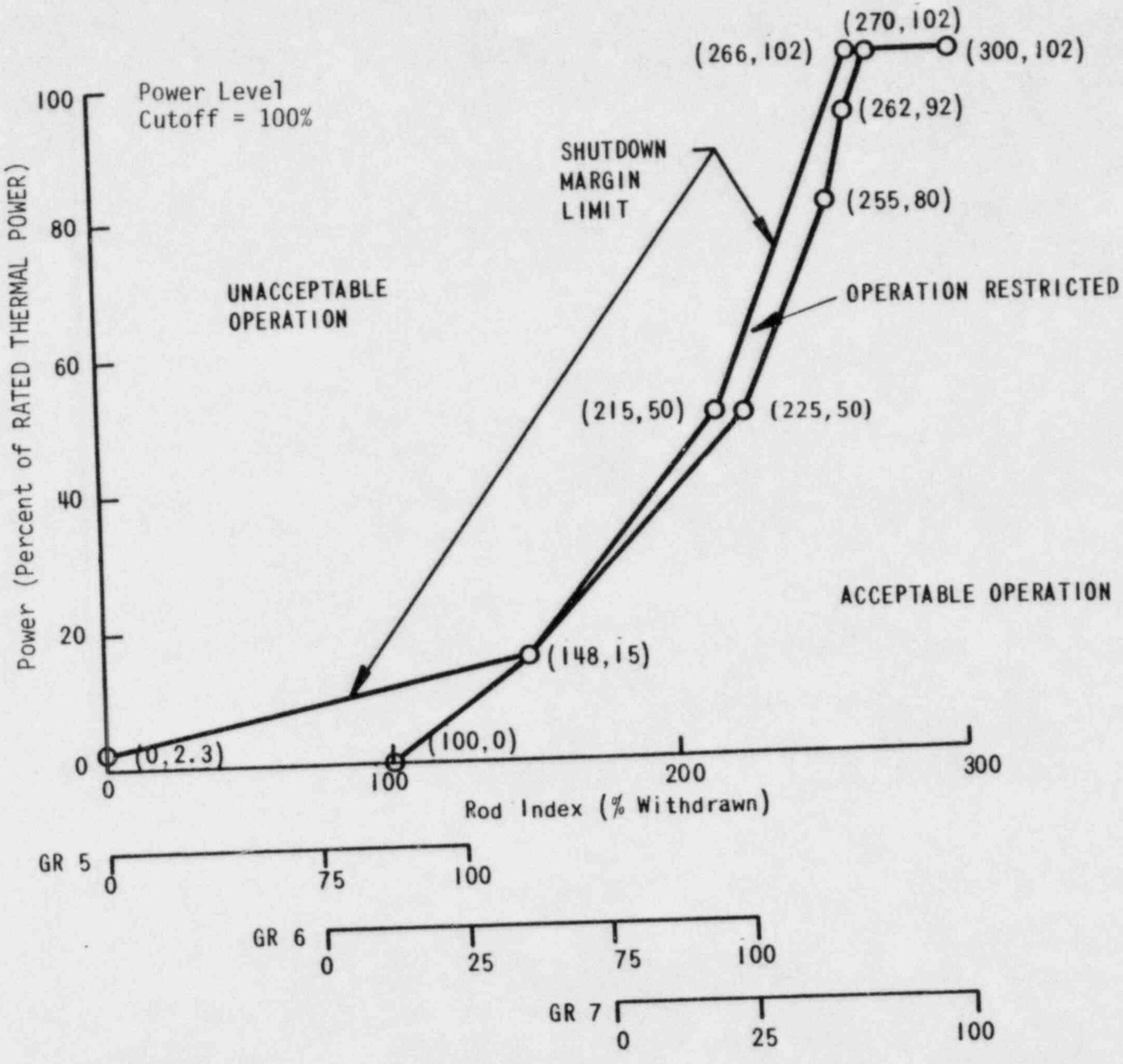


Figure 8-6. Regulating Group Position Limits, 0 to 24+10/-0 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-3a)

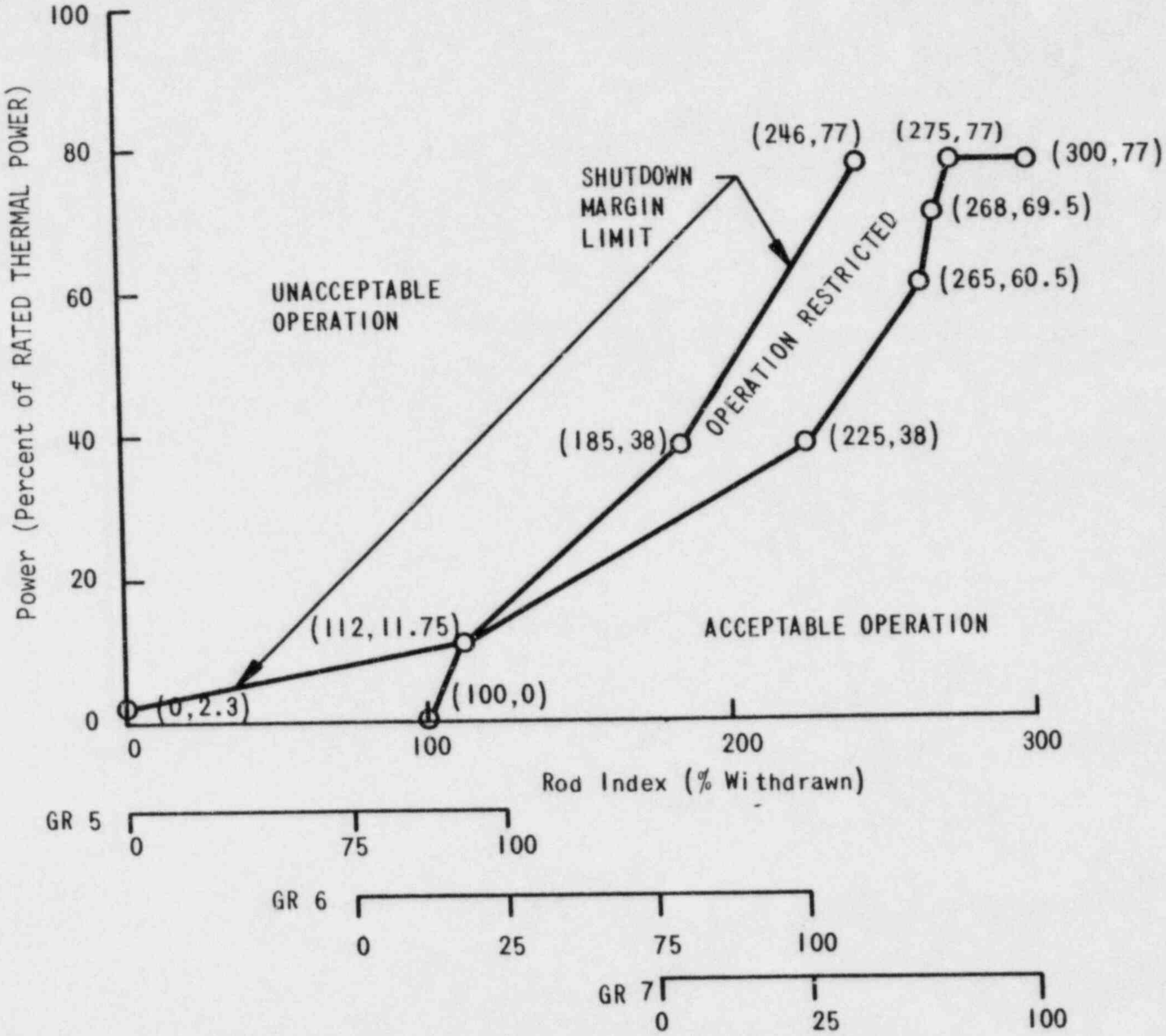


Figure 8-7. Regulating Group Position Limits, 24+10/-0 to 150±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-3b)

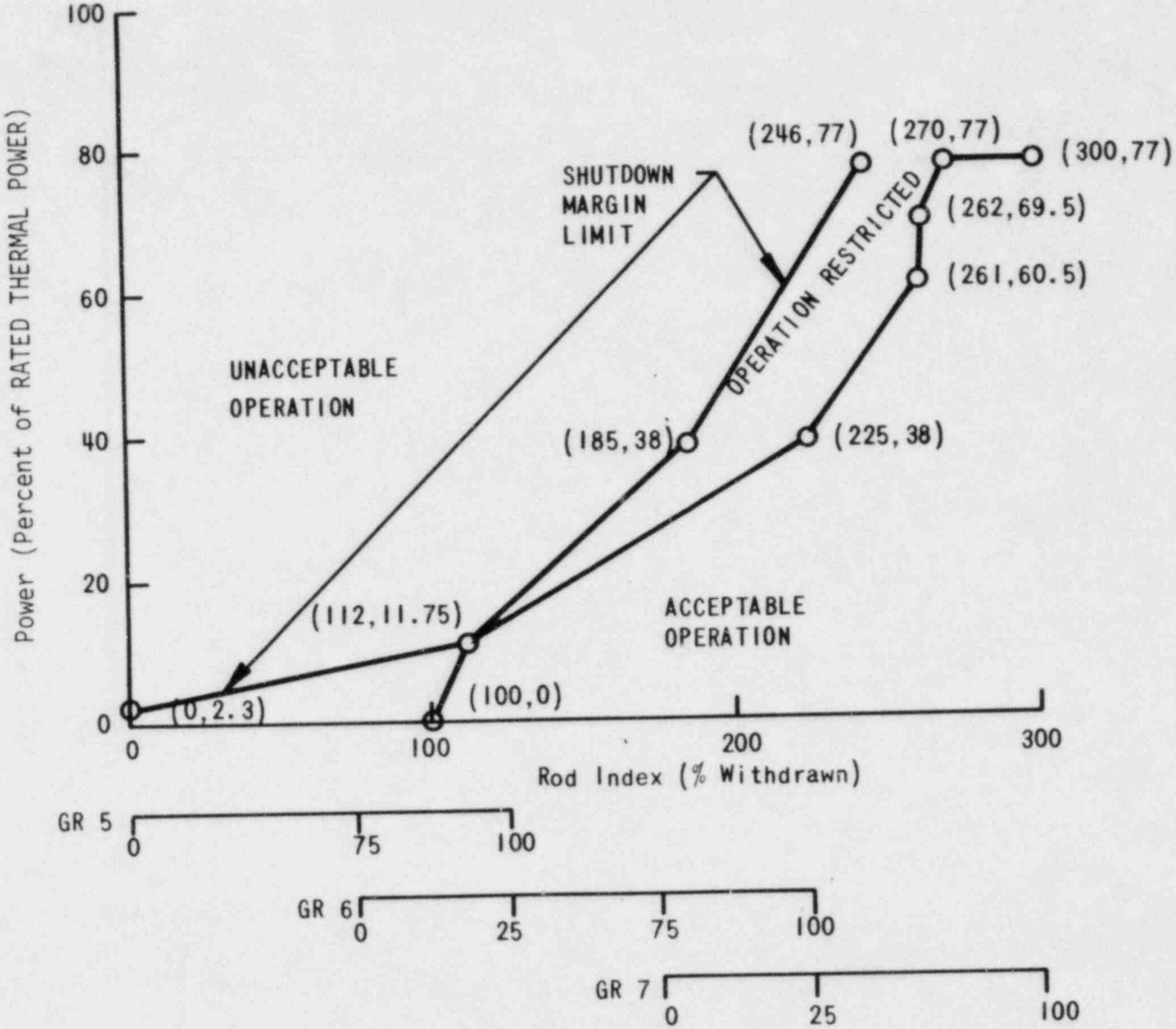


Figure 8-8. Regulating Group Position Limits After 150 ±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-3c)

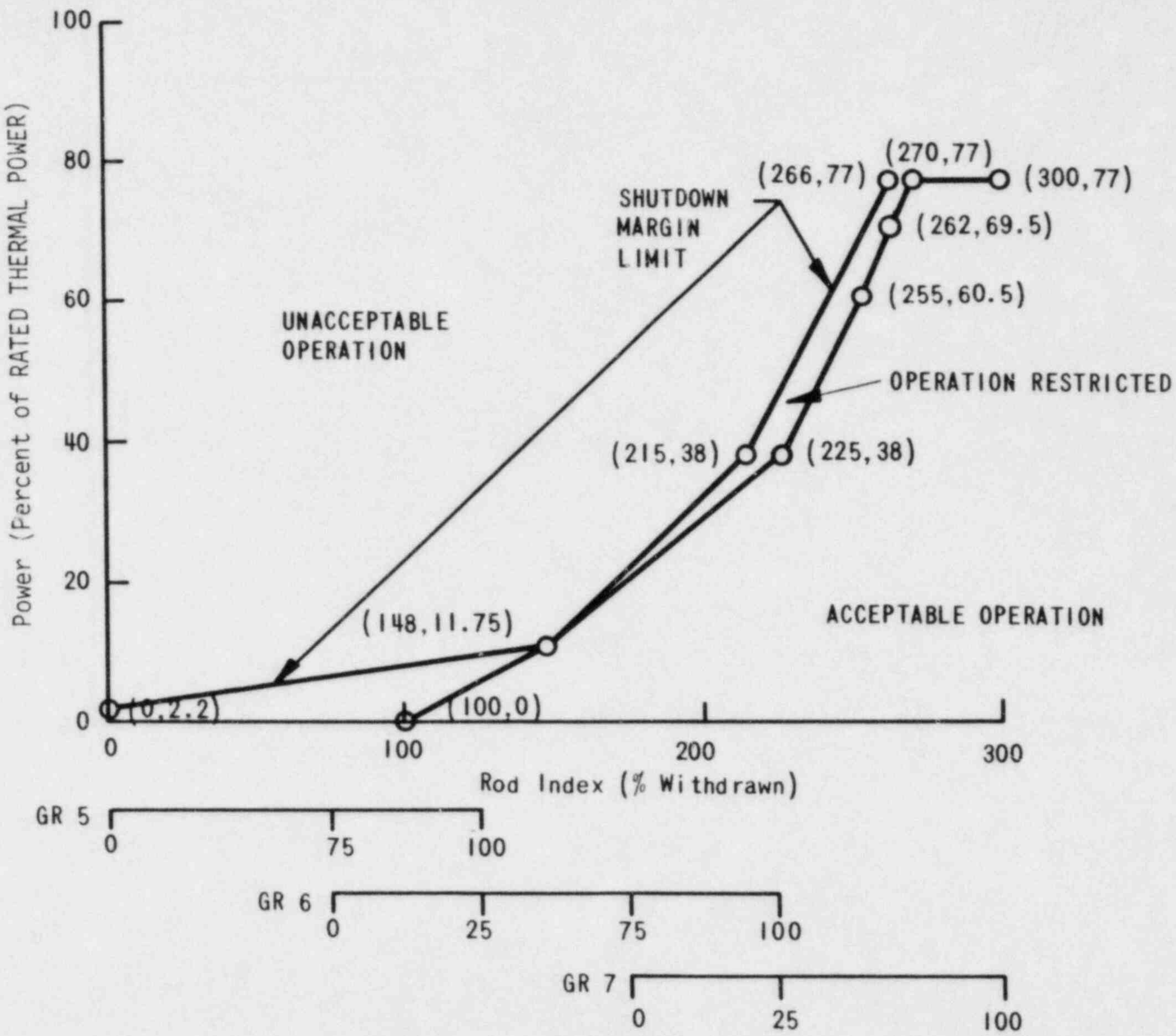


Figure 8-9. APSR Position Limits, 0 to 24+10/-0 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-5a)

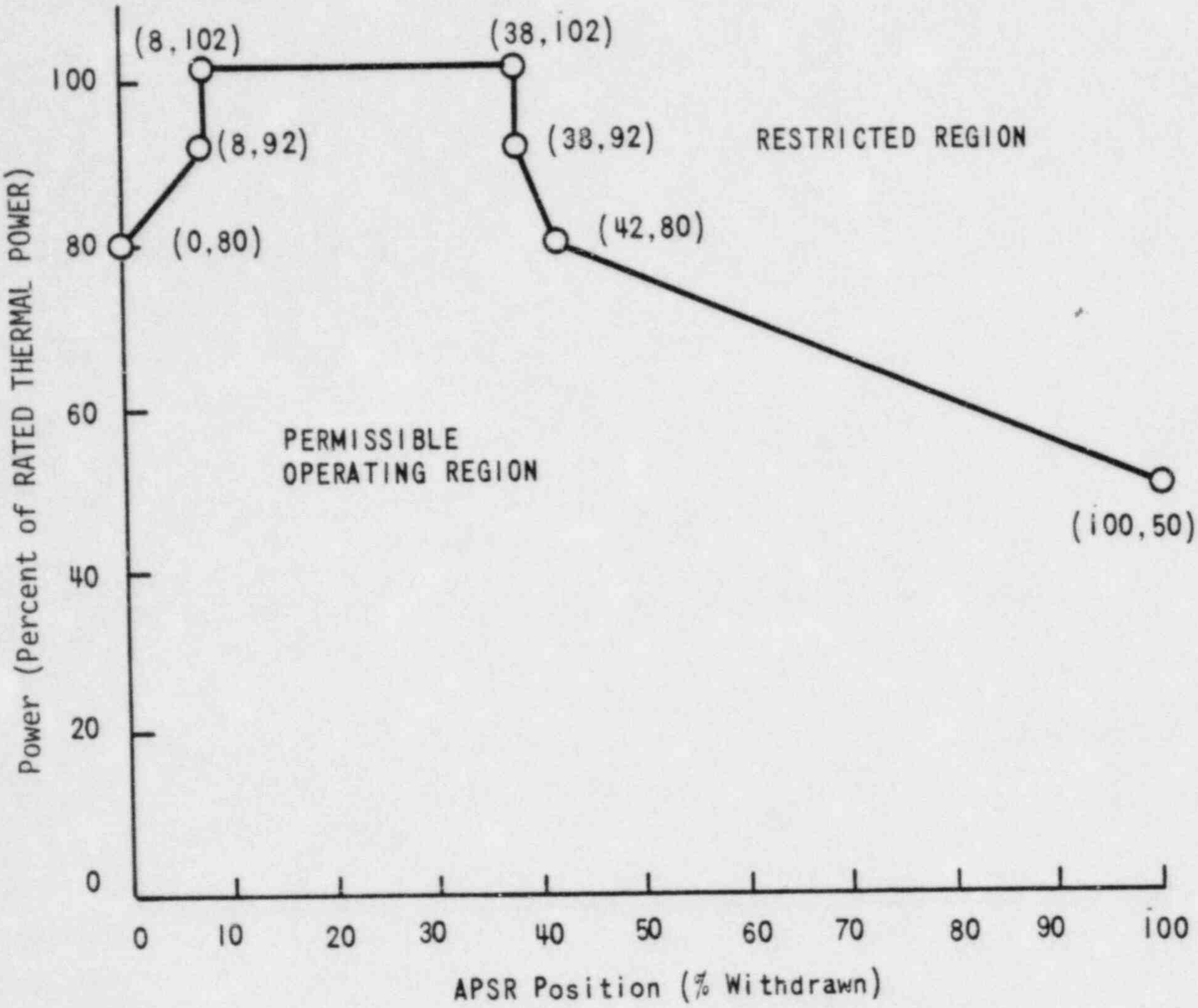


Figure 8-10. APSR Position Limits, 24+10/-0 to 150±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-5b)

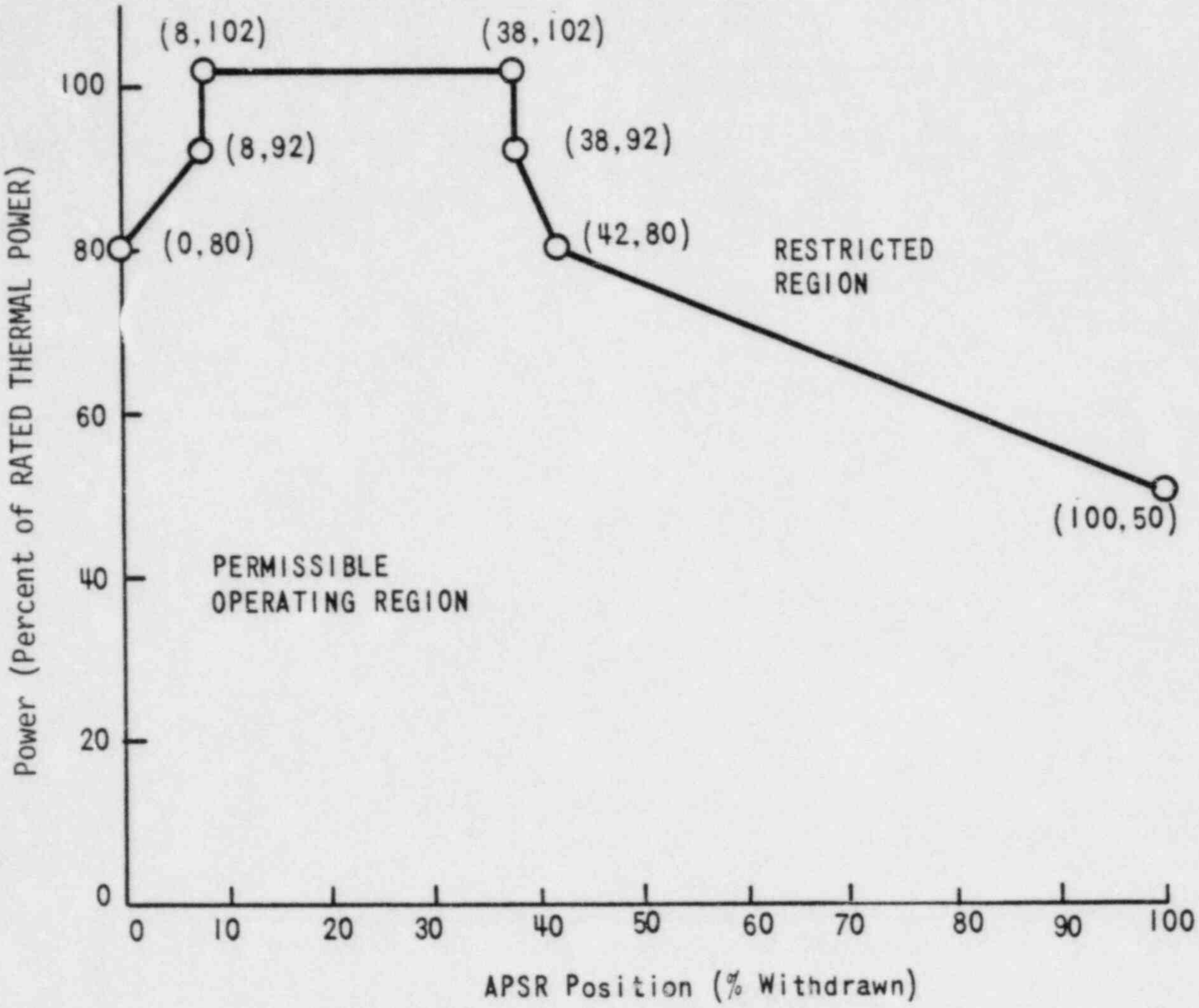


Figure 8-11. APSR Position Limits After 150 ±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-5c)

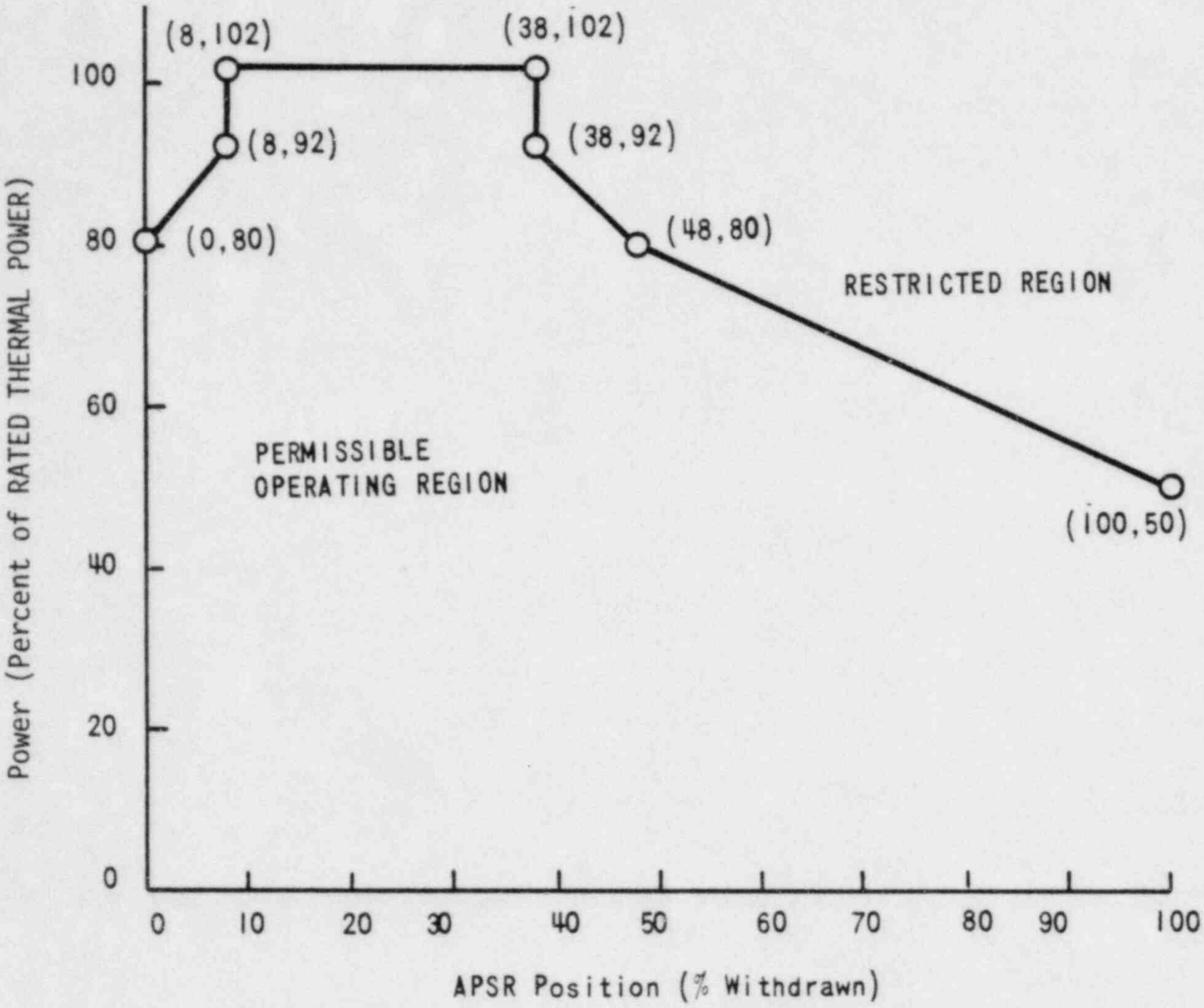


Figure 8-12. APSR Position Limits, 0 to 24+10/-0 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-5d)

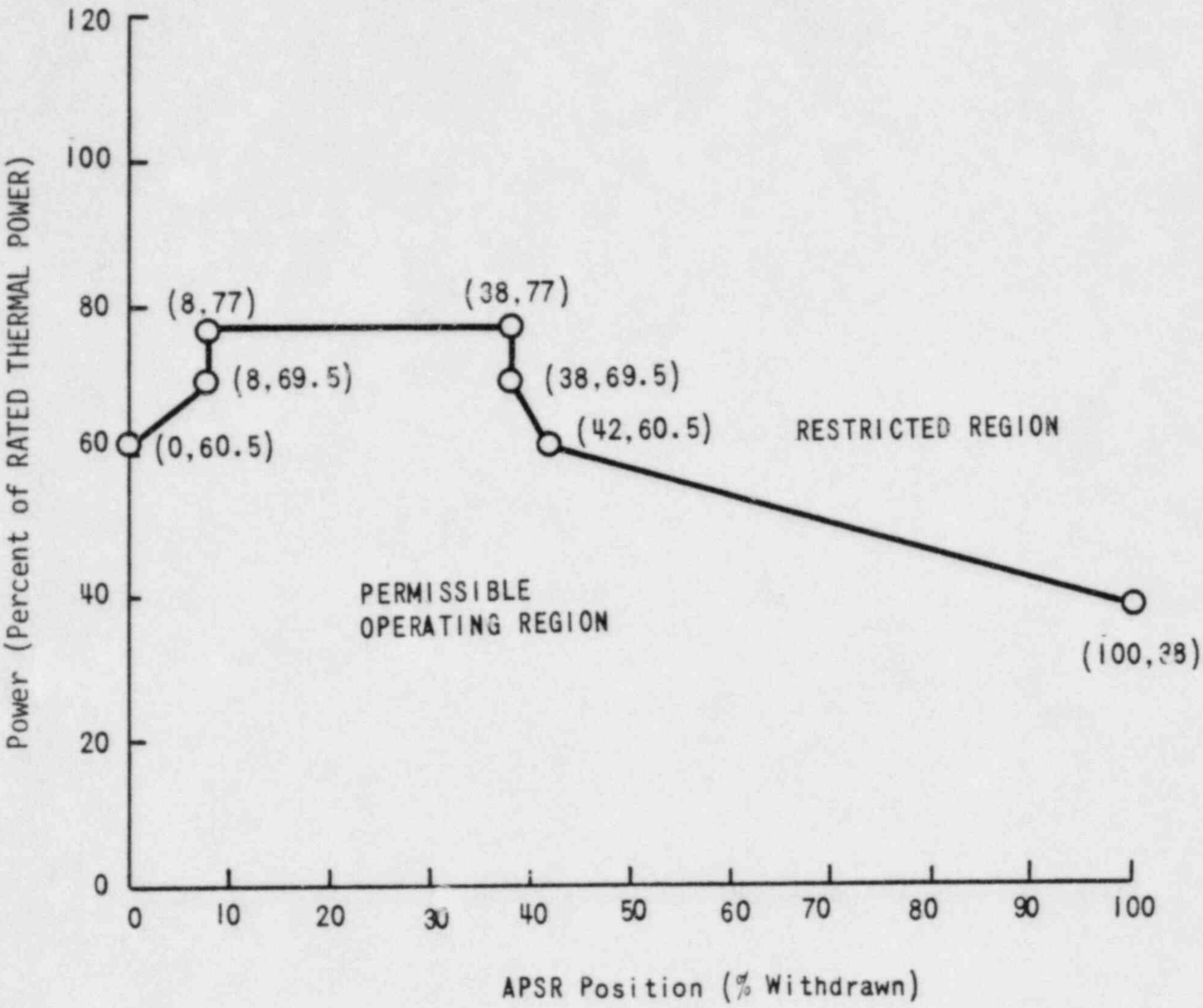


Figure 8-13. APSR Position Limits, 24+10/-0 to 150±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.1-5e)

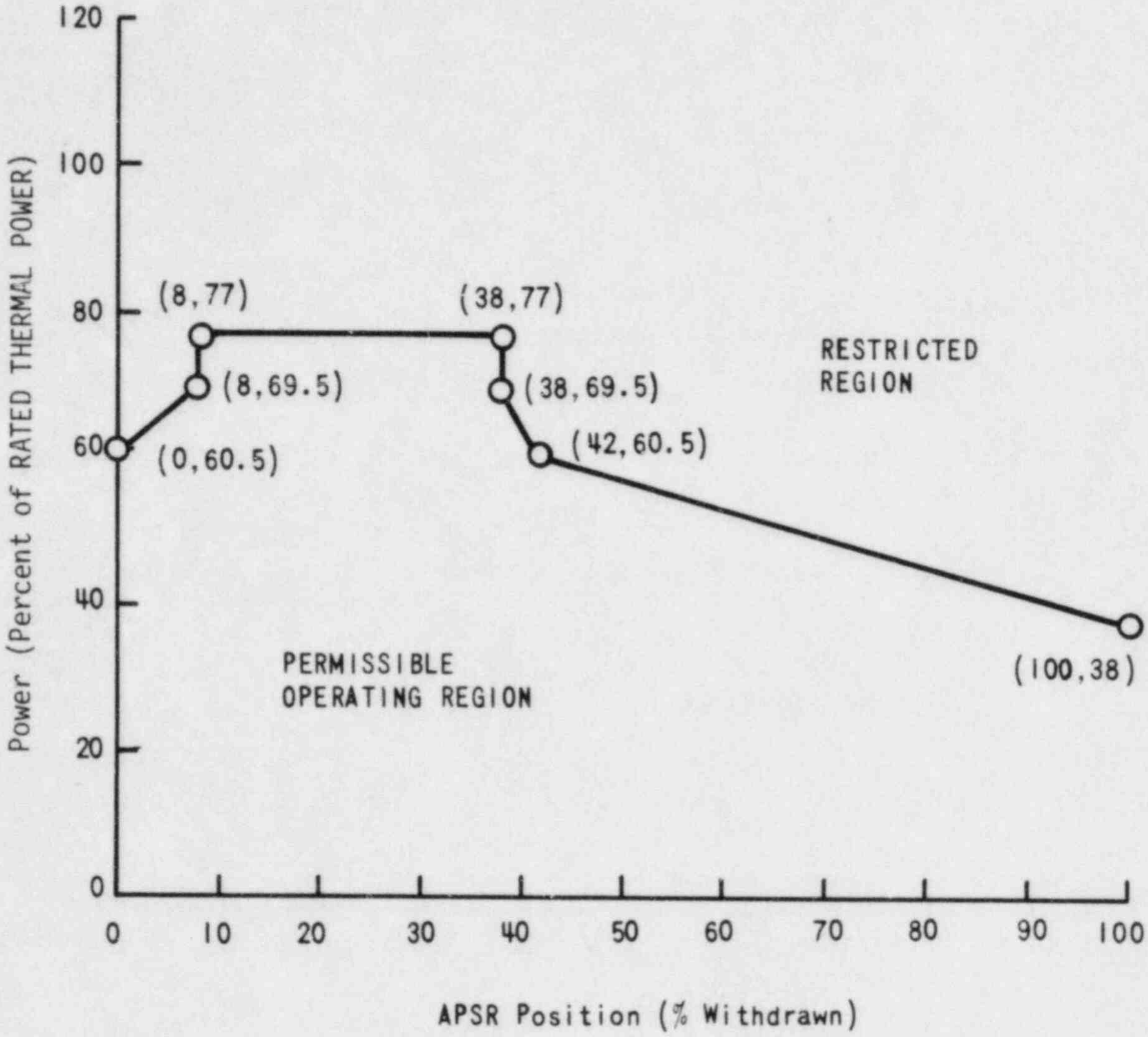


Figure 8-14. APSR Position Limits After 150 ±10 EFPD,
Three RC Pumps — Davis-Besse 1, Cycle 4
(Tech. Spec. Figure 3.1-5f)

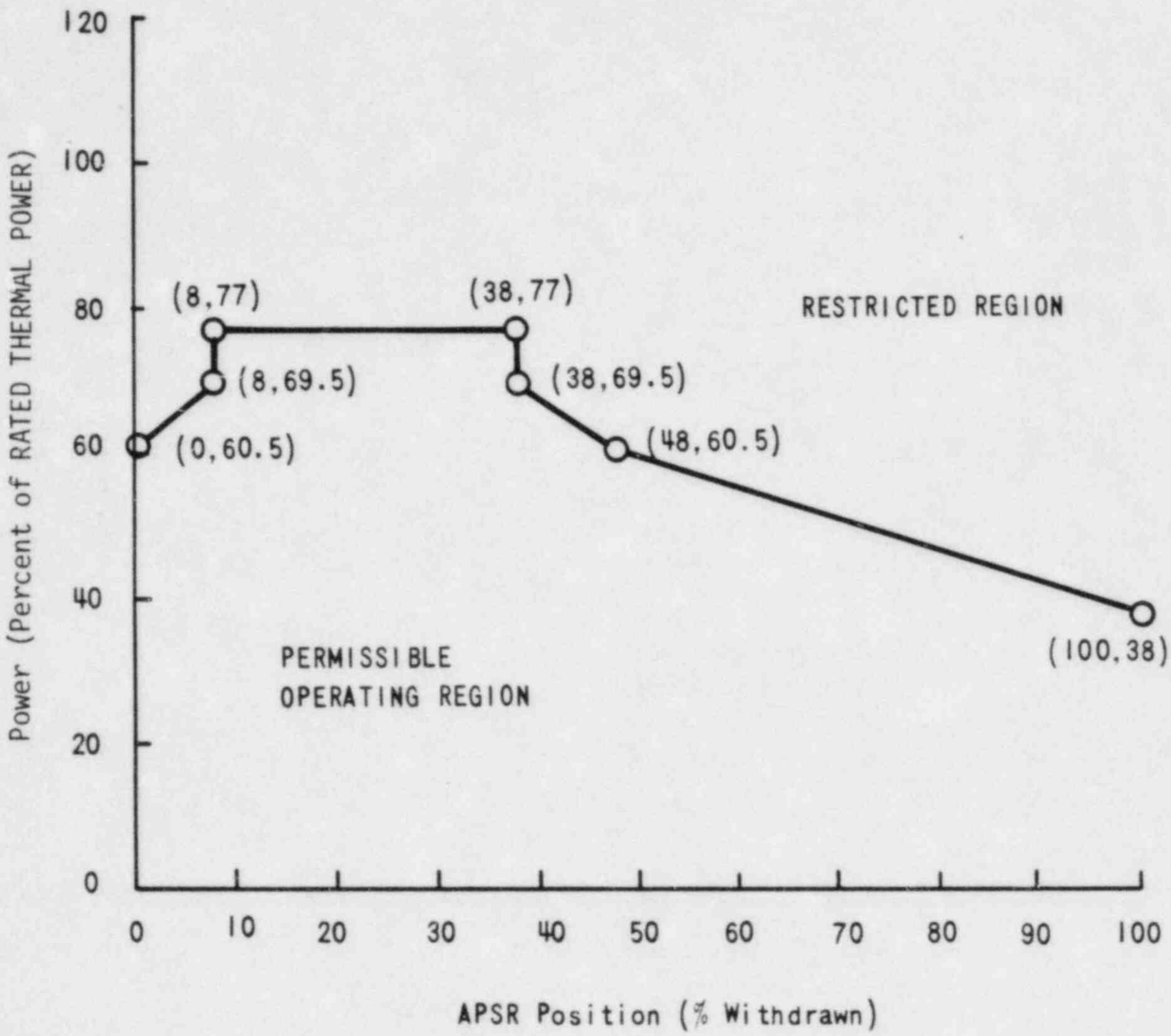


Figure 8-15. Axial Power Imbalance Limits, 0 to 24+10/-0 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-1a)

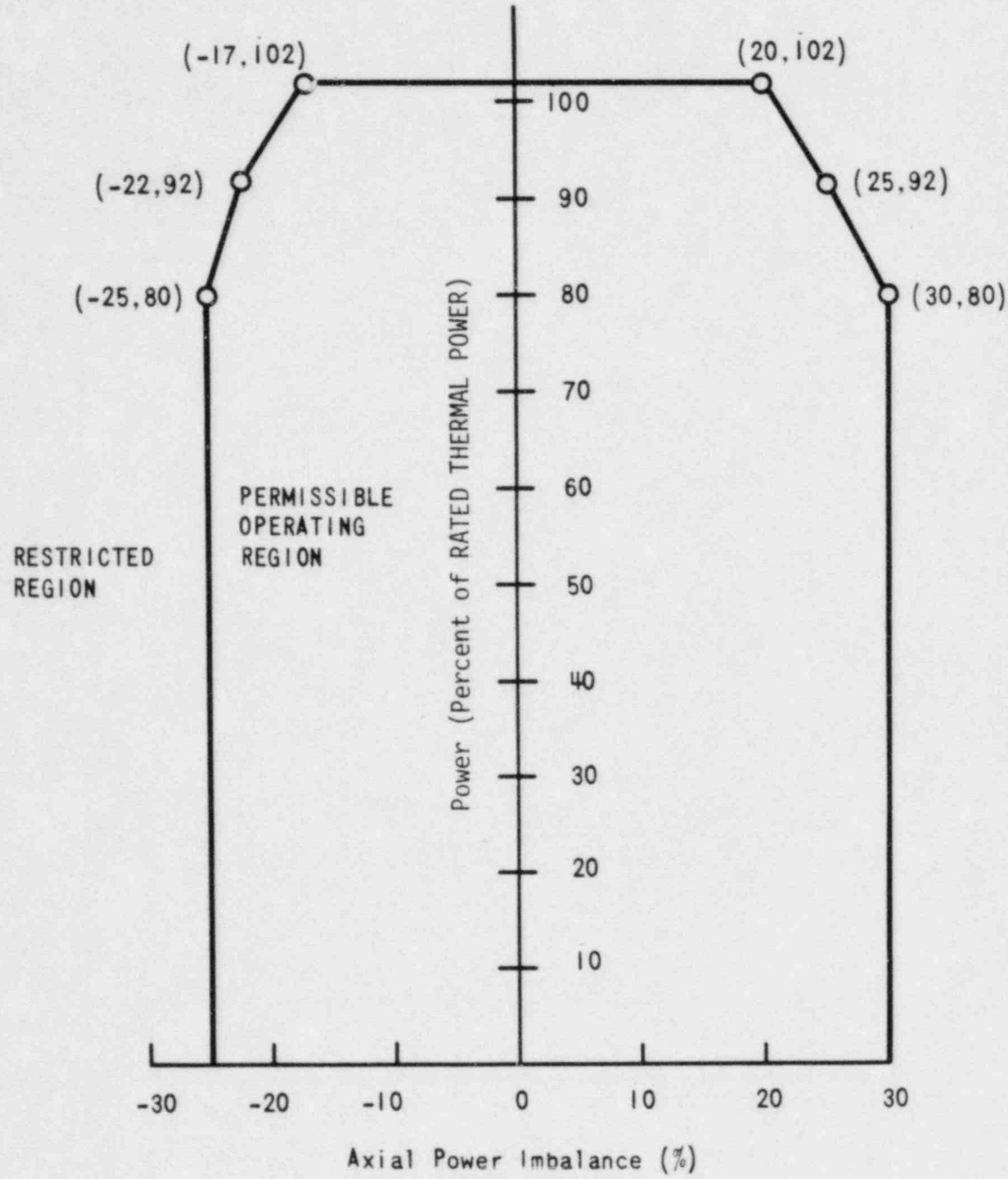


Figure 8-16. Axial Power Imbalance Limits, 24+10/-0 to 150±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-1b)

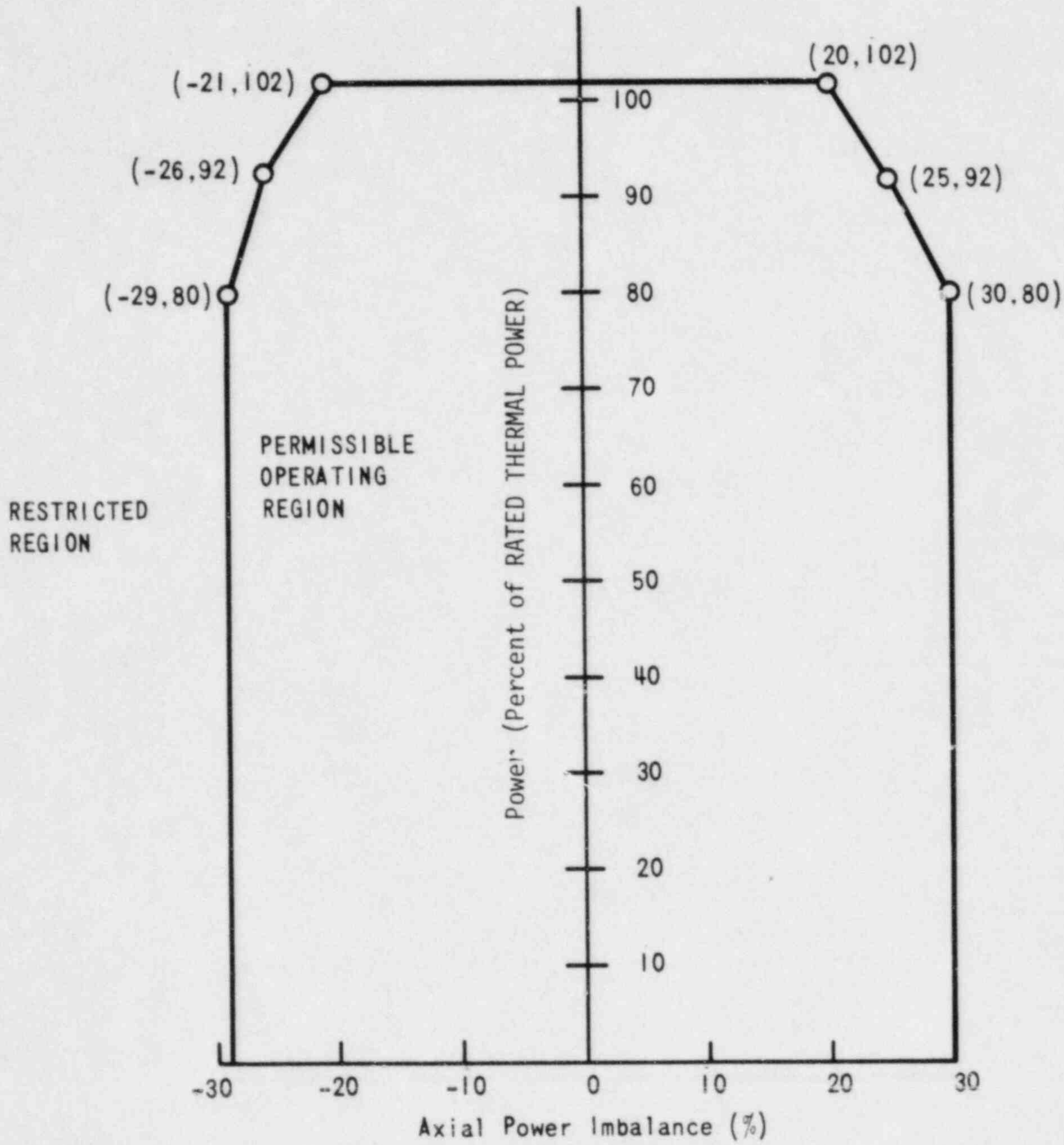


Figure 8-17. Axial Power Imbalance Limits After 150 ±10 EFPD, Four RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-1c)

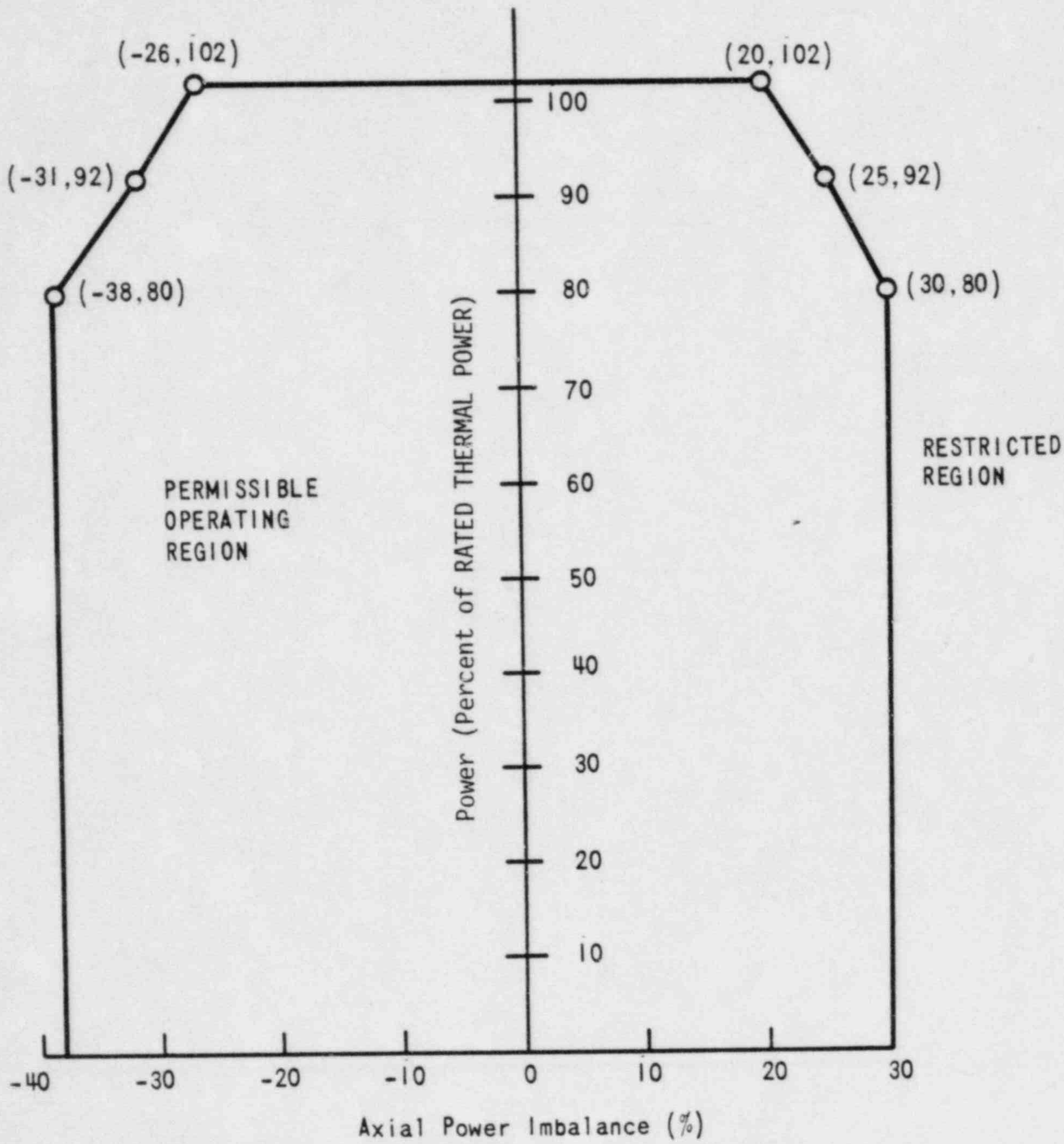


Figure 8-18. Axial Power Imbalance Limits, 0 to 24+10/-0 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-2a)

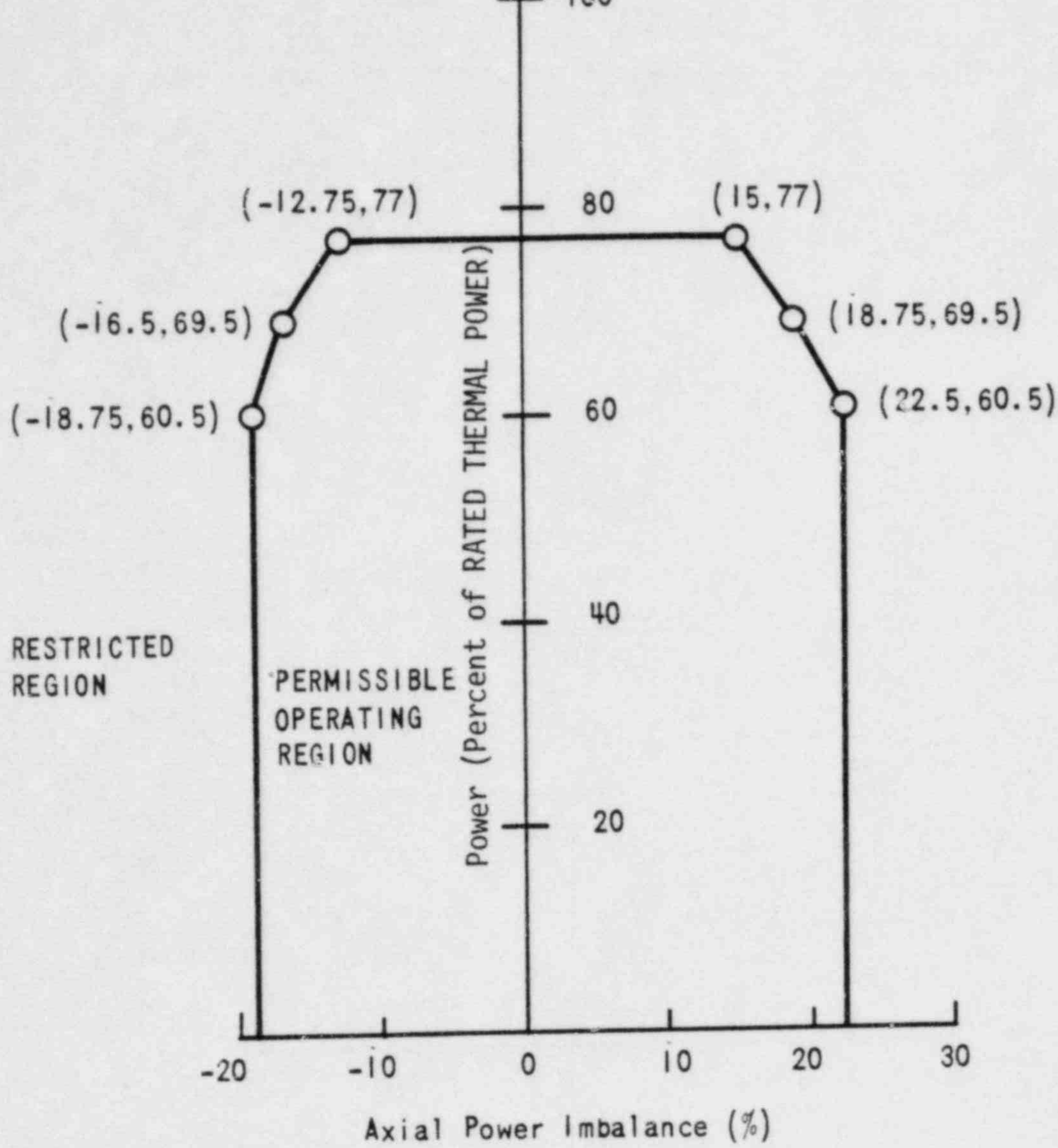


Figure 8-19. Axial Power Imbalance Limits, 24+10/-0 to 150±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-2b)

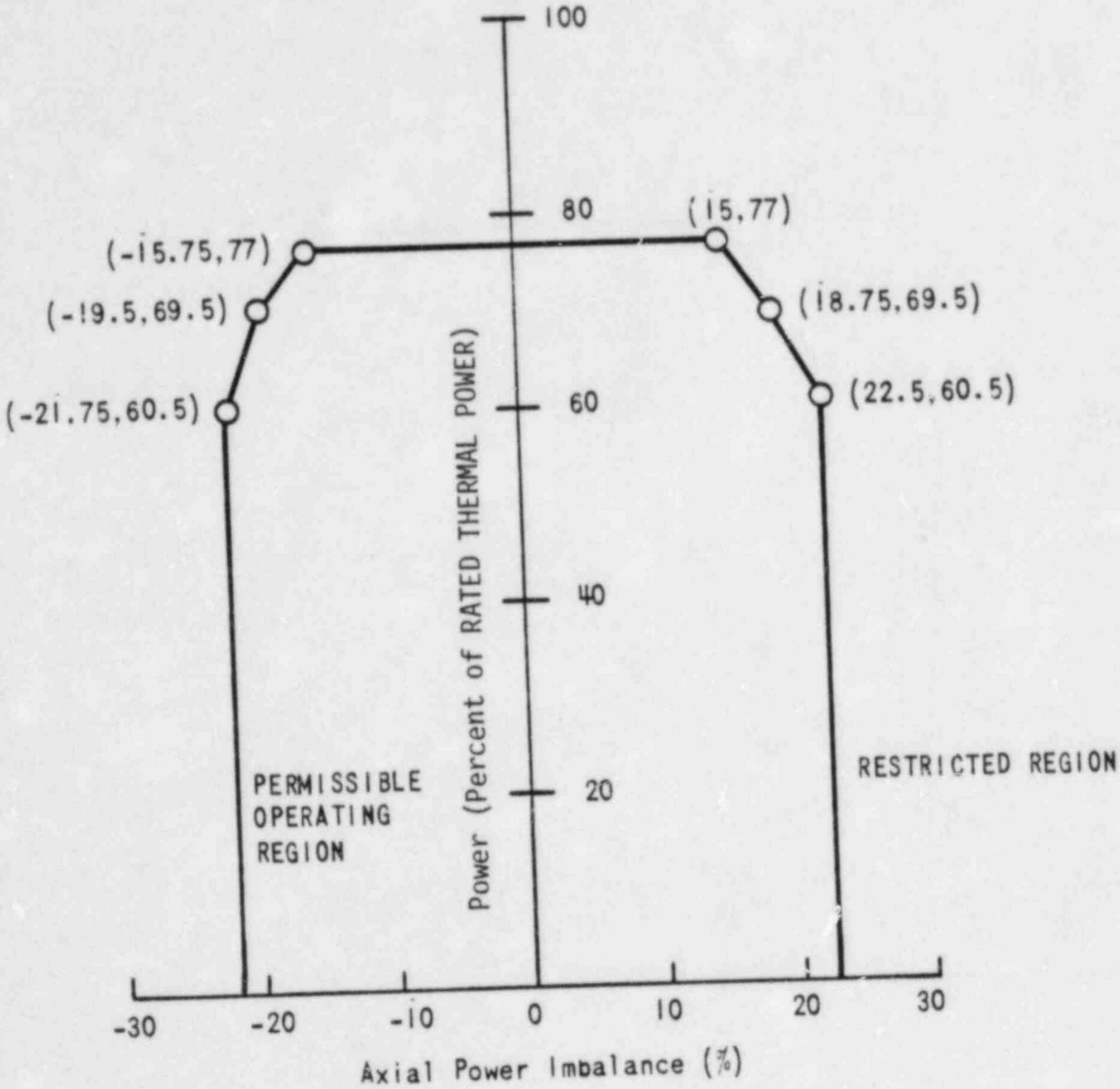


Figure 8-20. Axial Power Imbalance Limits After 150 ±10 EFPD, Three RC Pumps — Davis-Besse 1, Cycle 4 (Tech. Spec. Figure 3.2-2c)

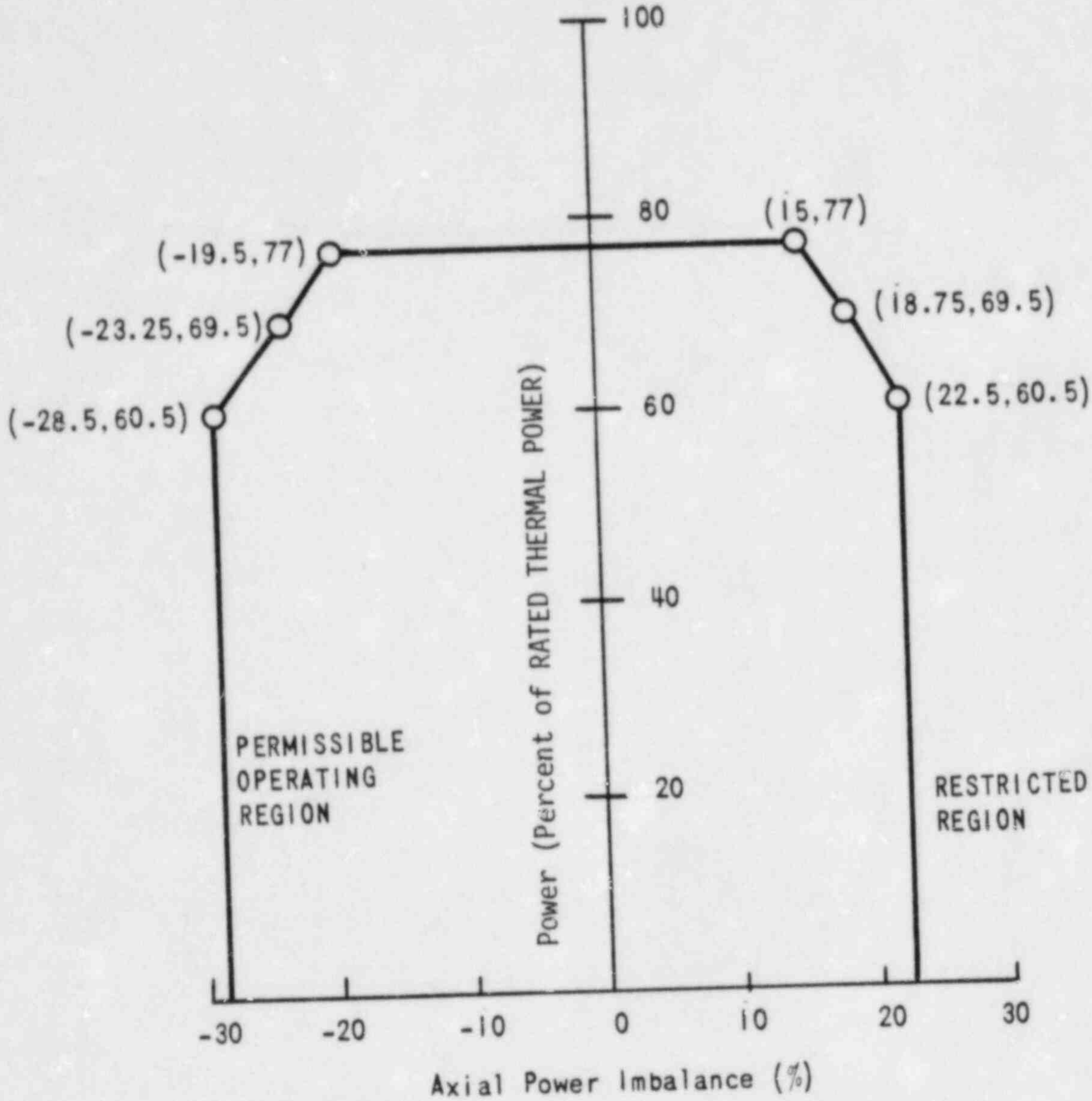
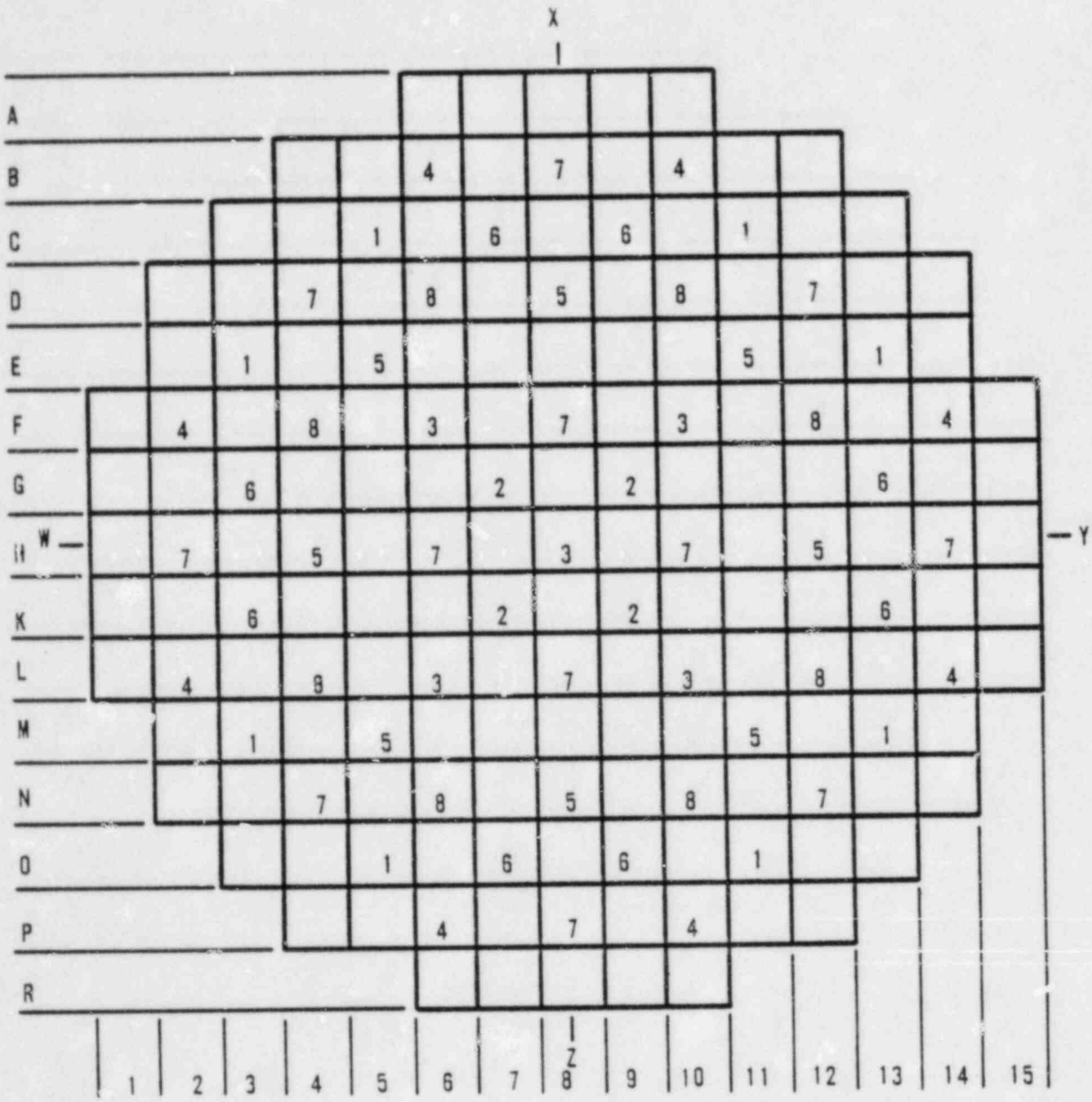


Figure 8-21. Control Rod Core Locations and Group Assignments — Davis Besse 1, Cycle 4

(Tech Spec Figure 3.1-4)



X GROUP NUMBER

GROUP	NO. OF RODS	FUNCTIONS
1	8	SAFETY
2	4	SAFETY
3	5	SAFETY
4	8	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 first increasing and then decreasing the temperature 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 $\pm 0.4 \times 10^{-4}$ $(\Delta k/k)/°F$ (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 $+0.9 \times 10^{-4}$ $(\Delta k/k)/°F$.

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 ∿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 ∿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 ∿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/withdraw 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^{-4} $(\Delta k/k)/\%$ 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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