Enclosure 2

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DAVIS-BESSE NUCLEAR POWER STATION
UNIT 1, CYCLE 11 -- RELOAD REPORT
AND ADDENDUM

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1. INTRODUCTION AND SUMMARY

This report justifies operation of Davis-Besse Nuclear Power Station Unit 1 at the rated core power of 2772 MWt for cycle 11. The required analyses are included as outlined in the Nuclear Regulatory Commission (NRC) document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975. This report utilizes the analytical techniques and design bases that have been submitted to the NRC and approved by that agency.

Cycle 11 reactor and fuel parameters related to power capability are summarized in this report and compared to those for cycle 10. All accidents analyzed in the Davis-Besse Updated Safety Analysis Report[1] (USAR), as applicable, have been reviewed for cycle 11 operation, and in all cases, the initial conditions of the transients in cycle 11 are bounded by previous analyses.

The cycle 11 design includes implementation of an end-of-cycle (EOC) HFP extension maneuver which includes a moderator temperature (T_{avg}) reduction of up to 7°F (actual). The effects of the EOC T_{avg} reduction on the RCS structural, RCS operation, core mechanical (fuel), radiological dose consequences, nuclear (design-peaking), and thermal-hydraulic parameters as well as any potential effects and/or consequences on LOCA and non-LOCA safety analyses have been evaluated and were found to be acceptable. The analyses also verified that the operational maneuver at EOC is bounded by the safety analyses assumptions and will be accomodated by the core protective and operating limits.

Fuel assembly NJ06KZ, which will be reinserted from cycle 9, was reconstituted and now contains one stainless steel replacement rod. Assembly NJ06KZ will receive its second cycle of irradiation during cycle 11. The effect of the replacement rod on core performance is discussed in the applicable sections.

The Technical Specifications have been reviewed for cycle 11 operation. Based on the reload report analyses performed, taking into account the emergency core cooling system (ECCS) Final Acceptance Criteria and postulated fuel densification effects, it is concluded that Davis-Besse Unit 1, cycle 11 can be operated safely at its licensed core power level of 2772 MWt. The Core Operating Limits Report (COLR) changes for cycle 11 are included in section 8 of this report.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Davis-Besse Unit 1 is the currently operating cycle 10[2], which achieved criticality on November 14, 1994. Power escalation began on November 15, 1994 and full power (2772 MWt) was attained on November 18, 1994.

During cycle 10 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 11. Cycle 11 was analyzed to 675 effective full power days (EFPD) based on cycle 10 operation of 488±15 EFPD with an APSR pull, end-of-cycle (EOC) T_{avg} reduction, CRG 7 withdrawal to 97%wd, and power coastdown. The cycle 11 design includes an APSR pull, EOC T_{avg} reduction, CRG 7 withdrawal to 97%wd, and power coastdown.

3. GENERAL DESCRIPTION

The cycle 11 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array normally containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The FA in batch 9E has a constant nominal fuel loading of 468.25 kg of uranium. The batch 11 through 13 FAs have a constant nominal fuel loading of 468.56 kg of uranium. The fuel consists of dished-end cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters may be found in Table 4-1 of this report.

Figure 3-1 is the core loading diagram for Davis-Besse Unit 1, cycle 11. Forty-eight batch 10B assemblies, 24 batch 11B assemblies, 4 batch 12A assemblies, 4 batch 9A assemblies, and 1 batch 9D assembly will be discharged at the end of cycle 10. The remaining batch 11C and batch 12B FAs will be shuffled to their cycle 11 locations, with the core periphery locations occupied by batch 11C and batch 13A fuel assemblies. One batch 9E assembly, discharged at the end of cycle 8, will be reinserted in cycle 11 as the center FA. Four batch 11A assemblies, discharged at the end of cycle 9, will be reinserted in cycle 11. One of these batch 11A assemblies, designated NJ06KZ, has been reconstituted with a stainless steel replacement rod. This assembly will be loaded at core location D08, as shown in Figure 3-1.

Batches 9E, 11A, and 11C have initial enrichments of 3.38, 3.77, and 3.77 wt %, respectively. The radially zone loaded batch 12B has an average enrichment of 4.06 wt %. The feed batch, consisting of 12 batch 13A and 64 batch 13B assemblies with uranium enrichment of 4.46 wt %, will be loaded in a symmetric checkerboard pattern throughout the core. The two batches differ in fuel pin prepressure. The cycle 11 shuffle scheme is a very low leakage (VLL) core loading. The VLL reload fuel shuffle scheme for cycle 11 will have a negligible effect on Nuclear Instrumentation response for all aspects of reactor startup and subsequent power operation. The cycle 11 design minimizes the number of fuel assemblies that are cross core shuffled to reduce the potential for quadrant tilt

amplification. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 11 and its initial enrichment.

Cycle 11 is operated in a feed-and-bleed mode. The core reactivity is controlled by 53 full-length Ag-In-Cd control rod assemblies, 52 burnable poison rod assemblies (BPRAs), and soluble boron. In addition to the full-length control rods, eight Inconel-600 axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The cycle 11 locations of the control rods and the group designations are indicated in Figure 3-3. The core locations and the rod group designations of the 61 control rods in cycles 10 and 11 are the same. The cycle 11 locations and concentrations of the BPRAs are shown in Figure 3-4.

Figure 3-1. Davis-Besse Cycle 11 Core Loading Diagram

← North

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						11C G13	11C M03	11C G09	11C M13	11C G03					
B				11C L05	13A F	12B P09	13B F	11C P10	13B F	12B P07	13A F	11C L11			
C			11C L14	13B F	12B M02	13B F	12B D13	13B F	12B D03	13B F	12B M14	13B F	11C P06		
D		11C E10	13B F	12B L09	13B F	12B F13	13B F	11A N09 9	13B F	12B F03	13B F	12B K06	13B F	11C E06	
E		13A F	12B B11	13B F	12B M04	13B F	12B K04	13A F	12B K12	13B F	12B D05	13B F	12B B05	13A F	
F	11C C07	12B K14	13B F	12B O06	13B F	12B O08	13B F	12B K10	13B F	12B H03	13B F	12B O10	13B F	12B K02	11C O09
G	11C C11	13B F	12B O04	13B F	12B D09	13B F	12B M12	13B F	12B N05	13B F	12B D07	13B F	12B O12	13B F	11C C05
H W—	11C G07	11C F14	13B F	11A G12 9	13A F	12B F09	13B F	9E N04 8	13B F	12B L07	13A F	11A K04 9	13B F	11C L02	11C K09
K	11C O11	13B F	12B C04	13B F	12B N09	13B F	12B D11	13B F	12B E04	13B F	12B N07	13B F	12B C12	13B F	11C O05
L	11C C07	12B G14	13B F	12B C06	13B F	12B H13	13B F	12B G06	13B F	12B C08	13B F	12B C10	13B F	12B G02	11C C09
M		13A F	12B P11	13B F	12B N11	13B F	12B G04	13A F	12B G12	13B F	12B E12	13B F	12B P05	13A F	
N		11C M10	13B F	12B G10	13B F	12B L13	13B F	11A D07 9	13B F	12B L03	13B F	12B F07	13B F	11C M06	
O			11C B10	13B F	12B E02	13B F	12B N13	13B F	12B N03	13B F	12B E14	13B F	11C F02		
P				11C F05	13A F	12B B09	13B F	11C B06	13B F	12B B07	13A F	11C F11			
R						11C K13	11C E03	11C K07	11C E13	11C K03					

Z

key:

xxx
yyy
zzz

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

Note: One steel rod in D08.

Figure 3-2. Davis-Besse Cycle 11 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.38 34540	4.46 0	4.06 21255	4.46 0	3.77 22855	4.46 0	3.77 35320	3.77 39581
K	4.46 0	4.06 22184	4.46 0	4.06 22976	4.46 0	4.06 17575	4.46 0	3.77 36960
L	4.06 21255	4.46 0	4.06 22251	4.46 0	4.06 21600	4.46 0	4.06 18791	3.77 41688
M	4.46 0	4.06 22978	4.46 0	4.06 22193	4.46 0	4.06 14784	4.46 0	
N	3.77 22855	4.46 0	4.06 21586	4.46 0	4.06 21206	4.46 0	3.77 36112	
O	4.46 0	4.06 17644	4.46 0	4.06 14808	4.46 0	3.77 35334		
P	3.77 35320	4.46 0	4.06 18718	4.46 0	3.77 36147			
R	3.77 39581	3.77 37006	3.77 41690					

x.xx yyyyy	Initial Enrichment BOC Burnup MWd/mtU

Figure 3-3. Davis-Besse Cycle 11 Control Rod Locations

← North

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

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

x Group Number

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

Figure 3-4. Davis-Besse Cycle 11 BPRA Concentration and Distribution

	8	9	10	11	12	13	14	15
H		3.5		3.5		1.4		
K	3.5		3.5		3.5			
L		3.5		3.5		2.3		
M	3.5		3.5		3.0			
N		3.5		3.0				
O	1.4		2.3					
P								
R								

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

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel parameters for Davis-Besse cycle 11 are listed in Table 4-1. Batch 9E is the Mark-B8A design, batch 11A and 11C are the Mark-B8B design, batch 12B is the Mark-B10AZL design, and batch 13 is the Mark-B10A design. Batch 13A and 13B fuel consists of a Mark-B10 cage with Mark-B9A fuel rods. The batch 13 fuel is identical to the batch 12 fuel, except for the enrichment in the fuel rods and the rod prepressure. All batch 12 assemblies utilize two different enrichments of uranium -- 184 fuel rods use an enrichment of 4.09 wt%, while the remaining 24 use 3.79 wt%; these are called zone-loaded fuel assemblies. The zone loading of the fuel rods in these assemblies is illustrated in Figure 3-2 of reference 2. In the new batch 13 assemblies, all fuel rods contain the same enrichment -- 4.46 wt%. Also, the batch 13A fuel has the same rod prepressure as batch 12, and the batch 13B fuel has a lower prepressure.

Eight gray APSRAs and 53 Ag-In-Cd CRAs will be used in cycle 11. Fifty-two fresh BPRAs will be used, with varying concentrations of Al_2O_3-B_4C. No BPRAs will be reinserted from a previous cycle.

4.2 Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1 Cladding Collapse

The computer code TACO3 (reference 3) is used to provide conservative values of cladding temperature and pin pressure to the computer code CROV (reference 4), which determines whether or not cladding collapse is predicted during the cycle. One new TACO3 run was required for cycle 11's batch 13B; all other batches were able to use those TACO3 runs performed for Davis-Besse cycle 10. New CROV runs were made for cycle 11 for batches 11, 12, 13A, and 13B.

Mark-B8A Fuel Rods (Batch 9)

The most limiting power history for batch 9 was determined. This history was shown to be enveloped by a power history used in the cycle 10 Mark-B8A fuel rod TACO3 run. Therefore, the results of the cycle 10 cladding collapse analysis still apply. No creep collapse is predicted to occur through a burnup of at least 60 GWd/mtU, which exceeds the cycle 11 incore life of these fuel rods.

Mark-B9A Fuel Rods (Batches 11, 12, and 13A)

The most limiting power histories for batches 11, 12, and 13 were determined. The batch 11 and 12 histories were shown to be enveloped by a power history used in the cycle 10 Mark-B9A fuel rod TACO3 run. The batch 13A power history slightly exceeded this cycle 10 TACO3 history. New CROV runs were made for each of the three batches. A new version of CROV that more realistically models the fuel rod than the version used in the previous cycle extended the lifetime of the batch 11 and 12 rods, in terms of creep collapse, to at least 60 GWd/mtU. For batch 13A, flux and temperature levels input to CROV were increased in order to account for the slight increase in the power history. Worst-case dimensions were used, as well. No creep collapse is predicted to occur through a burnup of at least 60 GWd/mtU, which exceeds the cycle 11 incore life of these fuel rods.

Mark-B9A Fuel Rods (Batch 13B)

A new TACO3 run and a new CROV run were performed for this batch, due to its lower rod prepressure. An enveloping power history and prepressure were used in the analysis. It was shown that cladding creep collapse is not predicted in these fuel rods until after 60 GWd/mtU burnup.

4.2.2 Cladding Stress

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

temperature (650 F). The stress intensity limits are as follows:

Primary general membrane stress intensities (P_m) shall not exceed S_m.

Local primary membrane stress intensities (P_l) shall not exceed 1.5 S_m. These include the contact stresses from the spacer grid stop and the fuel rod.

Primary membrane + bending stress intensities ($P_l + P_b$) shall not exceed 1.5 S_m.

Primary membrane + bending + secondary stress intensities ($P_l + P_b + Q$) shall not exceed 3.0 S_m.

where

P_m	=	General primary membrane stress intensity
P_l	=	Local primary membrane stress intensity
P_b	=	Primary bending stress intensity
Q	=	Secondary stress intensity

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

For both the Mark-B8A and Mark-B9A fuel rod designs, the margins are in excess of 12%. The following conservatisms were used in the stress analyses to ensure that all Condition I and II operating parameters were enveloped:

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

4.2.3 Cladding Strain

The fuel design criteria specify a limit of 1% cladding plastic tensile circumferential strain of the cladding. The fuel pellet is designed to ensure that this strain is less than 1% at the design local pellet burnup and heat generation rate. The design values are higher than the worst-case values that Davis-Besse Unit 1 cycle 11 fuel is expected to experience.

4.3 Thermal Design

All fuel rods in the cycle 11 core are thermally similar. The design of the fresh batches 13A and 13B Mark-B10A assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. The analyses for the Mark-B10A fuel and for fuel batches 12B, 11A, and 11C were performed with the TACO3 code as described in reference 3. Fuel performance for fuel batch 9E was evaluated with the TACO3 and the TACO2 code (as described in reference 5). The nominal undensified input parameters used in these thermal analyses are presented in Table 4-1. Densification effects were accounted for in the TACO2 and TACO3 code models.

The results of the thermal design evaluation of the cycle 11 core are summarized in Table 4-1. Cycle 11 core protection limits were based on linear heat rate (LHR) to centerline fuel melt limits determined by the TACO2 and TACO3 codes.

The maximum fuel pin burnup at EOC-11 is predicted to be 58,423 MWd/mtU (batch 9E). The fuel rod internal pressures have been evaluated with TACO3 for the highest burnup of each fuel rod type. The batch 13 fuel internal pressure was predicted to be less than the nominal reactor coolant pressure of 2200 psia. The internal pressures for the remaining batches (9E, 11A, 11C, and 12B) exceed the 2200 psia nominal reactor coolant pressure and were justified with the approved fuel rod gas pressure criterion described in reference 6.

4.4 Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interaction for batch 13 fuel assemblies is identical to that of present fuel assemblies.

4.5 Operating Experience

Framatome Cogema Fuels operating experience with the Mark-B 15x15 assembly has verified the adequacy of its design. Mark-B fuel assemblies have operated successfully in 100 fuel cycles at 8 nuclear power plant facilities.

Table 4-1 Fuel Design Parameters

	Batch 9E	Batch 11A	Batch 11C	Batch 12B	Batch 13A/13B
Fuel assembly type	Mark-B8A	Mark-B8B	Mark-B8B	Mark-B10AZL	Mark-B10A
No. of assemblies	1	4	36	60	12/64
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377
Undensified active fuel length, in.	143.2	140.6	140.6	140.6	140.6
Pellet OD, in.	.3686	.3700	.3700	.3700	.3700
Fuel pellet initial density, %TD mean	95.0	96.0	96.0	96.0	96.0
Initial fuel batch enrichment, wt% ^{235}U	3.38	3.77	3.77	24 rods at 3.79 and 184 rods at 4.09	4.46
Average burnup BOC, MWd/mtU	34,540	22,855	37,760	20,037	0
Cladding collapse burnup, MWd/mtU[a]	>60,000	>60,000	>60,000	>60,000	>60,000
Maximum pin burnup, MWd/mtU	58,423	49,935	54,565	52,070	34,072
Nom. linear heat rate at 2772 MWt, kW/ft	6.14	6.25	6.25	6.25	6.25
Maximum linear heat rate to melt, kW/ft	20.5	22.3	22.3	22.3	22.3

[a] Calculated using method from reference 4.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 10 and 11 designs. The values for cycles 10 and 11 were generated with the NEMO code[7]. Differences in core physics parameters are to be expected between the cycles due to the changes in fuel and burnable poison concentrations which create changes in flux and burnup distributions. Figure 5-1 illustrates a representative relative power distribution for BOC-11 at full power with equilibrium xenon, group 7 inserted to nominal HFP position, and gray APSRs partially inserted.

The ejected rod worths in Table 5-1 are the maximum calculated values. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in section 8. The adequacy of the shutdown margin with cycle 11 rod worths is shown in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 6% uncertainty on net rod worth[8].
3. Xenon transient allowance.
4. A maximum power deficit.

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

5.2. Changes in Nuclear Design

The design changes for cycle 11 include increased enrichment, larger feed batch size and a longer cycle length. These changes are incorporated in the physics model. The impact of the stainless steel rod was also evaluated and determined not to significantly impact core reactivity, stuck rod worth, or ejected rod worth.

Thirty-three standard control rods and twenty extended life control rods (ELCRAs) will continue to be used in cycle 11. The impact of their continued use on boron concentrations and shutdown margin has been evaluated. Conservatism in the xenon transient allowance is sufficient to compensate for the additional poison material depletion that resulted from using the 33 standard control rods in cycle 11 instead of replacing them with ELCRAs. The cladding material of the standard rods is stainless steel 304; Inconel 625 is used to clad the ELCRAs.

No significant operational or procedural changes exist with regard to axial or radial power shape, xenon, or tilt control. The stability and control of the core with APSRs withdrawn has been analyzed. The operating limits (COLR changes) for the reload cycle are given in section 8.

Table 5-1. Davis-Besse Unit 1, Cycle 11 Physics Parameters

	Cycle 10	Cycle 11[a]
Cycle length, EFPD	520	675
Cycle burnup, MWd/mtU	17,384	22,561
Average core burnup - 675 EFPD[b], MWd/mtU	34,491	37,745
Initial core loading, mtU	82.9	82.9
Critical boron[c] - 0 EFPD, No Xe, ppm		
HZP	1,973	2,439
HFP	1,754	2,225
Critical boron[c] - 675 EFPD[b], 100%FP Xe, ppm		
HZP	196	258
HFP	5[d]	5[d]
Control rod worths - HFP, 4 EFPD, %Δk/k		
Group 6	0.87	0.82
Group 7	1.12	0.95
Group 8	0.14	0.12
Control rod worths - HFP, 675 EFPD[b], %Δk/k		
Group 7	1.15	1.05
Group 8	NA	NA
Max ejected rod worth - HZP, %Δk/k		
0 EFPD, Groups 5-8 inserted[e] (N-12)	0.76	0.28
675 EFPD[b], Groups 5-7 inserted (L-10)	0.56	0.30
Max stuck rod worth - HZP, %Δk/k		
0 EFPD (N-12, Cy10; M-11, Cy11)	0.66	0.46
675 EFPD[b] (M-13, Cy10; M-11, Cy11)	0.69	0.60
Power deficit[f] - HZP to HFP, 100%FP Xe, %Δk/k		
4 EFPD	-1.86	-1.68
675 EFPD[b]	-3.16	-3.29
Doppler coeff[f] - HFP, 10^{-3} %Δk/k/°F		
0 EFPD, No Xe[g]	-1.55	-1.51
675 EFPD[b], 100%FP Xe, 0 ppm	-1.70	-1.82
Moderator coeff[f] - HFP, 10^{-2} %Δk/k/°F		
0 EFPD, No Xe[g]	-0.60	-0.19
675 EFPD[b], 100%FP Xe, 0 ppm[h]	-3.38	-3.51
Temperature coeff[f] - HZP, 10^{-2} %Δk/k/°F		
675 EFPD[b], 100%FP Xe, Grps 1-7 In, M11 out[i], 0 ppm	-2.70	-2.63

Table 5-1. Davis-Besse Unit 1, Cycle 11 Physics Parameters

	Cycle 10	Cycle 11[a]
Boron worth[f] - HFP, ppm/%Δk/k		
0 EFPD	150	166
675 EFPD[b], 100%FP Xe	118	126
Xenon worth[f] - HFP, %Δk/k		
4 EFPD	2.60	2.51
675 EFPD[b], 100%FP Xe	2.77	2.72
Effective delayed neutron fraction[f] - HFP		
4 EFPD	0.00628	0.00648
675 EFPD[b], 100%FP Xe	0.00527	0.00526

(a) Based on cycle 9 length of 500.8 EFPD (actual) and cycle 10 length of 488 EFPD.

(b) Calculated at 520 EFPD for cycle 10.

(c) Control rod group 8 is inserted for calculation at 0 EFPD and withdrawn for calculation at 675 EFPD.

(d) Power coastdown to 520 EFPD at 5 ppm for cycle 10 and to 675 EFPD at 5 ppm for cycle 11.

(e) Cycle 10 value calculated with rod group 5 at 15%WD and groups 6-7 inserted.

(f) All calculations done with control rod groups 1-7 at 100%WD and control rod group 8 at nominal HFP position, unless otherwise noted.

(g) Cycle 11 values were calculated at 2339 ppm (includes allowances for B10 atom fraction variation and reactivity anomalies); cycle 10 values were calculated at 1831 ppm.

(h) These values were calculated with the control rods at rod index 260%WD.

(i) Cycle 10 value calculated with M13 out.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 11

	BOC	EOC, %Δk/k	
		620 EFPD	675 EFPD
	%Δk/k	Group 8 in	Group 8 out
Available Rod Worth			
Total rod worth, HZP	6.05	6.72	6.73
Worth reduction due to burnup of poison material	-0.00	-0.03	-0.03
Maximum stuck rod worth, HZP	-0.46	-0.57	-0.60
Net Worth	5.59	6.12	6.10
Less 6% Uncertainty	-0.34	-0.37	-0.37
Total available worth	5.25	5.75	5.73
Required Rod Worth			
Power deficit, HFP to HZP	1.68	3.20	3.29
Xenon transient allowance	0.30	0.30	0.30
Max allowable inserted rod worth	0.31	0.50	0.52
Total required worth	2.29	4.00	4.11
Shutdown Margin			
Total available minus total required	2.96	1.75	1.62

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

Figure 5-1. Davis-Besse Cycle 11 Relative Power Distribution at BOC (4 EFPD), Full Power, Equilibrium Xenon, Rods at Nominal HFP Positions

	8	9	10	11	12	13	14	15
H	0.807	1.218	1.203	1.307	7 1.124	1.296	0.663	0.241
K	1.218	1.158	1.316	1.191	1.297	1.188	1.089	0.270
L	1.203	1.315	1.211	1.321	8 1.158	1.255	0.809	0.208
M	1.307	1.190	1.319	1.212	1.321	1.160	0.872	
N	7 1.124	1.296	8 1.156	1.320	7 1.138	1.153	0.344	
O	1.296	1.183	1.253	1.158	1.153	0.403		
P	0.663	1.081	0.808	0.871	0.344			
R	0.241	0.268	0.208					

x	Inserted Rod Group Number
x.xxx	Relative Power Density

6.0 THERMAL-HYDRAULIC DESIGN

The reference core analysis for cycle 11, which was originally performed for the cycle 10 core, corresponds to a full core of the Mark-B10A fuel assembly design. The approved methods in reference 9 and statistical core design (SCD) methodology[10] were used in the analysis.

The Mark-B10A fuel design has control rod guide tubes designed to minimize core bypass flow, thus producing a reduced core bypass flow fraction, compared to earlier fuel designs. The resulting core bypass flow for the reference core configuration is 5.3% of the reactor coolant system flow rate.

The cycle 11 core comprises 136 Mark-B10A fuel assemblies (batches 12 and 13) and 41 Mark-B8A and B8B assemblies (batches 9E, 11A, and 11C), with a core bypass flow of 6.52%. The effect of this increased bypass flow, relative to the reference analysis, is offset by retained DNB margin.

One batch 11A fuel assembly contains one stainless steel replacement rod. This has been evaluated and found acceptable in accordance with reference 11.

Table 6-1 summarizes DNB analysis parameters for cycles 10 and 11.

Table 6-1. Maximum Design Conditions, Cycles 10 and 11

	Cycle 10	Cycle 11
Design power level, MWt	2772	2772
Nominal core exit pressure, psia	2200	2200
Minimum core exit pressure, psia	2135	2135
Reactor coolant flow, gpm	380,000	380,000
Core bypass flow, %	5.3 [a]	5.3 [a]
DNBR modeling	SCD	SCD
Reference design (radial x local) power peaking factor	1.795	1.795
Reference design axial flux shape	1.65 chopped cosine	1.65 chopped cosine
Hot channel factors		
Enthalpy rise	1.011	1.015 [b]
Heat flux	N/A [c]	N/A [c]
Flow area	0.97	0.97
Active fuel length, in.	140.6	140.6
Avg heat flux at 100% power, 10^5 Btu/h-ft^2	1.89	1.89
Max heat flux at 100% power, 10^5 Btu/h-ft^2	5.60	5.60
CHF correlation	BWC	BWC
Thermal design DNB limit	1.40	1.40
Minimum DNBR		
at 102% power	2.02	2.02
at 112% power	1.79	1.79

[a] Used in the analysis. Actual cycle 11 value is 6.52%.

[b] The statistical core design methodology allows for a higher enthalpy rise hot channel factor.

[c] The hot channel factor for heat flux is not applied in DNB calculations.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each USAR accident analysis has been examined with respect to changes in the cycle 11 parameters to determine the effects of the cycle 11 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the USAR accident results have been evaluated and are reported in reference 12.

The radiological dose consequences of the USAR chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source terms for Davis-Besse cycle 11. The dose calculations were performed consistent with the assumptions described in the Davis-Besse USAR but used the more-conservative source terms (which bound future reload cycles). The results of the dose evaluations showed that offsite radiological doses for each accident were below the respective acceptance criteria values in the current NRC Standard Review Plan (NUREG-0800).

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

7.2 Accident Evaluation

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas. These areas are: (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 11 are given in Table 4-1. The cycle 11 thermal-hydraulic maximum design conditions are presented in Table 6-1. A comparison of the key kinetics parameters from the USAR and cycle 11 is provided in Table 7-1.

The EOC moderator temperature coefficient listed in Table 7-1 for cycle 11 is the 3-D, hot full power (HFP) temperature coefficient. An evaluation was performed to verify the acceptability of the more negative cycle 11 moderator temperature coefficient for all USAR accidents excluding steam line breaks. The results of the evaluation were acceptable for all USAR accidents, excluding steam line breaks, for a moderator temperature coefficient as negative as -4.0×10^{-2} %Δk/k/°F.

The steam line break accident was evaluated based on the total reactivity change from 532°F to a minimum temperature of 510°F. The temperature coefficient used in safety analysis of the steam line break is -3.10×10^{-2} %Δk/k/°F. This value is based on the sum of the moderator density, control rod worth degradation and Doppler reactivity, over the temperature range from 532°F to 510°F. The combined EOC temperature coefficient for cycle 11 is shown in section 5, and in Table 7-1, as -2.63×10^{-2} %Δk/k/°F. Since the safety analysis value for the EOC temperature coefficient is more negative than the cycle 11 value, the steam line break analysis remains bounding for cycle 11.

Generic loss-of-coolant accident (LOCA) analyses for the B&W 177-FA raised-loop nuclear steam system (NSS) have been performed to calculate allowable LOCA linear heat rate (LHR) limits that are applicable to the Mark-B8A, Mark-B8B, Mark-B10A, and Mark-B10AZL fuel types. The final acceptance criteria B&W ECCS evaluation model techniques and assumptions, as described in BAW-10104P, Rev. 5[13], were used in the analyses. The application of the evaluation model[14] included the effects of the NUREG-0630 fuel pin rupture curves, FLECSET reflooding heat transfer coefficient calculations, and the BWC CHF correlation.

Table 7-2 shows the maximum allowable LOCA linear heat rate limits for the different types of fuel in the Davis-Besse Unit 1 cycle 11 core as functions of burnup. For batch 9E, linear interpolation between the elevation-specific linear heat rate limits at 24,500 MWd/mtU and the linear heat rate limit of 10.5 kW/ft at 60,000 MWd/mtU was justified for cycle 11.

For batches 11A, 11C, and 12B, the cycle-specific fuel temperature and internal pin pressure as functions of burnup for cycle 11 were found to be bounded by the fuel performance data assumed in the Mark-B10AZL LOCA analyses, which utilized

TACO3 fuel performance data. The LOCA analyses assumed a different prepressure than that used in batches 11A, 11C, and 12B, but the increased prepressure was evaluated to be acceptable. The batch 11A/11C fuel is Mark-B8B, but the Mark-B10AZL LOCA limits are applicable to this batch as well. At high fuel burnups, the limits for batches 11A, 11C, and 12B are reduced in order to maintain the internal fuel pin pressure less than or equal to the limit based on the NRC-approved fuel rod gas pressure criterion[6].

The maximum allowable LOCA linear heat rate limits for the fresh batch 13 (13A/13B) Mark-B10A fuel are provided from BOC to the approximate EOC burnup of 35,000 MWd/mtU. The fuel temperatures and pin pressures at this burnup were shown to be less than those used in the applicable Mark-B10AZL LOCA analyses. The fuel temperatures and pin pressures evaluated include the effects of the split batch 13 initial prepressure values.

The impact of one stainless steel rod in batch 11A assembly NJ06KZ (cycle 11 location D-08) on power peaking was evaluated based on full-core power distributions. An evaluation of the peaking changes in both assembly NJ06KZ and adjacent fuel assemblies was performed to verify that the fuel rod peaking increases meet the criteria established in reference 11 for minimum DNBR prediction and LOCA analysis initial conditions. Based on the results of the peaking increases for affected fuel rods, no additional DNBR or ECCS analyses were required.

It is concluded by the examination of cycle 11 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 plant during cycle 11. Considering the previously accepted design basis used in the USAR and subsequent cycles, the transient evaluation of cycle 11 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 11 are bounded by the USAR and/or subsequent cycle analyses.

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

Parameter	USAR value	Cycle 11 value
BOL[a] Doppler coeff, 10^{-3} %Δk/k/°F	-1.28	-1.51
EOL[b] Doppler coeff, 10^{-3} %Δk/k/°F	-1.45[c]	-1.82
BOL moderator coeff, 10^{-2} %Δk/k/°F	+0.13	-0.19
EOL moderator coeff, 10^{-2} %Δk/k/°F	-4.0	-3.51
EOL temperature coeff (532 to 510°F) 10^{-2} %Δk/k/°F	-3.10	-2.63
All rod bank worth (HZP), %Δk/k	10.0	6.05
Boron reactivity worth (HFP), ppm/%Δk/k	100	169
Max ejected rod worth (HFP), %Δk/k	0.65	<0.65[d]
Max dropped rod worth (HFP), %Δk/k	0.65	<0.20
Initial boron conc (HFP), ppm	1407	2339[e]

(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.

(d) Calculational uncertainty (15%) is applied to the limit in the design analysis when determining cycle-specific regulating group position limits.

(e) Includes allowances for B^{10} atom variations and reactivity anomalies.

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

Mark-B8A Fuel Type

	Allowable Peak LHR for Specified Burnup, kW/ft	
Core Elevation, ft	Less Than 24,500 MWd/mtU	60,000 MWd/mtU
0	12.8	8.4
0.375	12.8	8.4
1	16.0	10.5
2	16.0	10.5
4	15.75	10.5
6	16.5	10.5
8	17.25	10.5
10	17.0	10.5
11	17.0	10.5
11.625	13.6	8.4
12	13.6	8.4

Mark-B8B and Mark-B10AZL Fuel Type

	Allowable Peak LHR for Specified Burnup, kW/ft					
Core Elevation, ft	Less Than 40,000 MWd/mtU	47,016 MWd/mtU	47,557 MWd/mtU	49,180 MWd/mtU	55,000 MWd/mtU	60,000 MWd/mtU
0	12.9	12.5	12.5	12.4	10.2	10.0
0.375	12.9	12.5	12.5	12.4	10.2	10.0
1	16.2	15.6	15.6	15.5	12.8	12.5
2	16.2	15.6	15.6	15.5	12.8	12.5
4	16.2	15.6	15.6	15.5	12.8	12.5
6	16.2	15.6	15.6	15.5	12.8	12.5
8	16.8	16.2	16.2	15.5	12.8	12.5
10	17.0	16.4	16.2	15.5	12.8	12.5
11	17.0	16.4	16.2	15.5	12.8	12.5
11.625	13.6	13.1	12.9	12.4	10.2	10.0
12	13.6	13.1	12.9	12.4	10.2	10.0

Mark-B10A Fuel Type

	Allowable Peak LHR for Specified Burnup, kW/ft	
Core Elevation, ft	Less Than 20,000 MWd/mtU	35,000 MWd/mtU
0	13.6	12.9
0.375	13.6	12.9
1	17.0	16.2
2	17.0	16.2
4	17.8	16.2
6	17.5	16.2
8	18.0	16.8
10	18.3	17.0
11	18.3	17.0
11.625	14.6	13.6
12	14.6	13.6

Linear interpolation between burnup points to calculate the Allowable LHR is allowed. The specified burnup ranges are those required to encompass cycle 11 operation.

8. PROPOSED MODIFICATIONS TO CORE OPERATING LIMITS REPORT

The Core Operating Limits Report (COLR) has been revised for cycle 11 operation to accommodate the influence of the cycle 11 core design on power peaking, reactivity, and control rod worths. Revisions to the cycle-specific parameters were made in accordance with the requirements of NRC Generic Letter 88-16 and Technical Specification 6.9.1.7. The core protective and operating limits were determined from a cycle 11 specific power distribution analysis using NRC approved methodology provided in the references of Technical Specification 6.9.1.7.

A cycle 11 specific analysis was conducted to generate the axial power imbalance protective limits, corresponding trip setpoints, and the Limiting Conditions for Operation (rod index, axial power imbalance, and quadrant tilt), based on the NRC-approved methodology described in reference 9. The analysis incorporates DNB peaking limits based on the allowable increase in design (radial x local) peaking provided by the statistical core design methodology described in reference 10. The effects of gray APSR repositioning were included explicitly in the analysis. The analysis also determined that the cycle 11 core operating limits provide protection for the overpower condition that could occur during an overcooling transient because of nuclear instrumentation errors. The single stainless steel rod in assembly NJ06KZ was determined to have an insignificant impact on the cycle 11 core operating limits.

The capability to perform the end of cycle (EOC) hot full power maneuver is included in the rod index and axial power imbalance limits and setpoints. The maneuver consists of an APSR withdrawal starting at 610 $\pm$ 10 EFPD and a T_{avg} reduction of up to 7°F (actual) to extend HFP operation. The xenon stability index after APSR withdrawal was determined to be -0.0338 h^{-1}, which demonstrates the axial stability of the core during operation with the APSRs fully withdrawn. The analysis verified that the operational maneuver at EOC is bounded by the safety analyses assumptions and will be accommodated by the core protective and operating limits.

The maximum allowable LOCA linear heat rate limits used in the analysis are based on the ECCS analysis described in section 7.2. Table 8-4 provides the burnup-

and elevation-dependent linear heat rate limits for each incore segment. They are the basis of the F_Q power peaking surveillance limits required by Technical Specification 3/4.2.2.

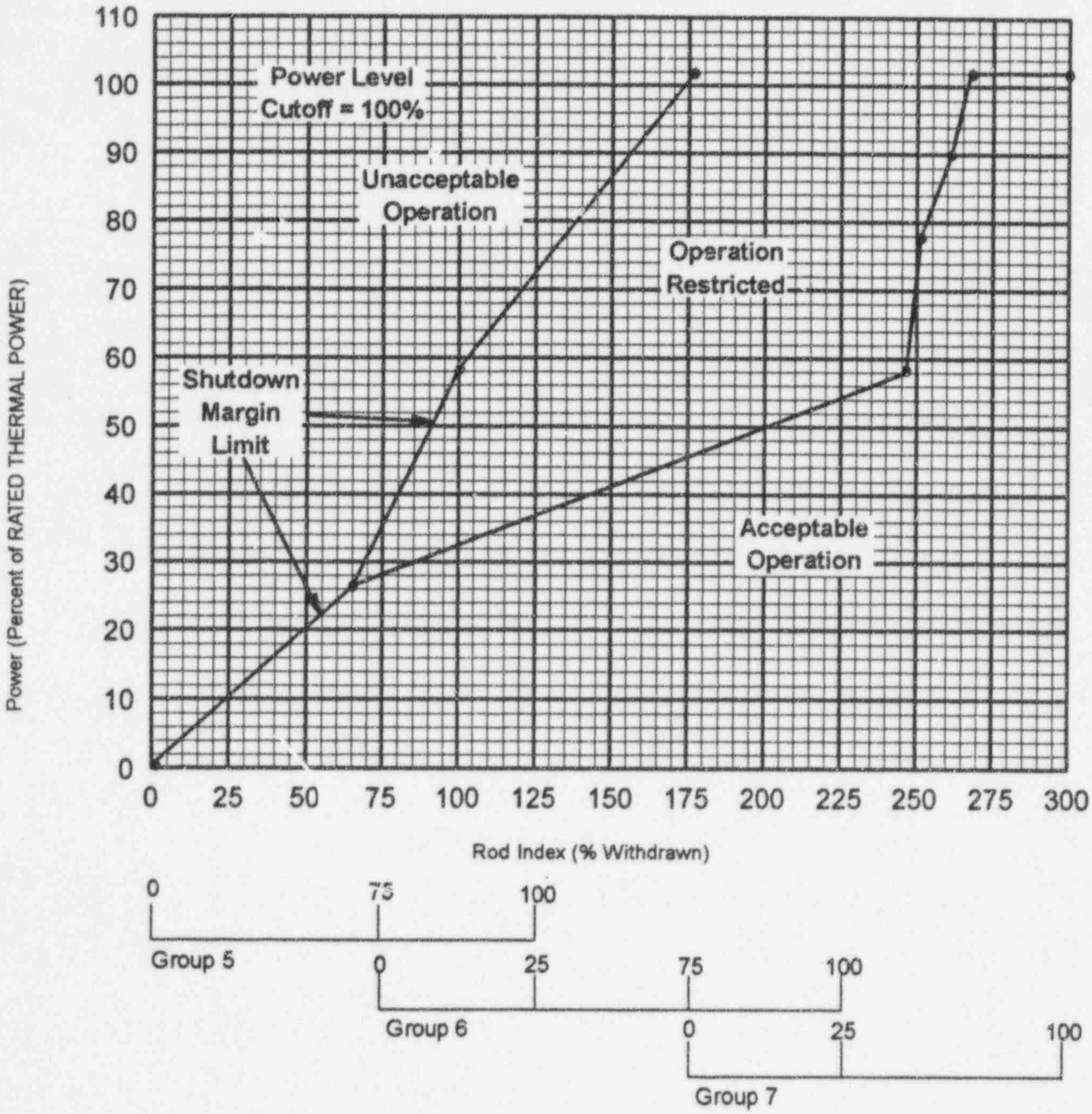
The measurement system-independent rod position and axial power imbalance limits determined by the cycle 11 analysis were error adjusted to generate alarm setpoints for power operation. Figures 8-1 through 8-6 and Figures 8-8 through 8-14 are revisions to the operating limits contained in the COLR and have been adjusted for instrument error to provide alarm setpoints. Figure 8-7 provides the control rod core locations and group assignments for cycle 11. Figures 8-15 and 8-16 are the core protective limits and RPS imbalance trip setpoints defined for cycle 11 based on revised 18 month RCS flow string error calculations. All other cycle 10 RPS setpoints (reference 2) were determined to be valid for cycle 11. Figure 8-17 provides the allowable radial peaking factors to be used in the calculation of the $F^N_{\Delta H}$ limits. They are the basis of the $F^N_{\Delta H}$ power peaking surveillance limits required by Technical Specification 3/4.2.3. The 3-RCP axial power imbalance alarm setpoints provided in Figures 8-12 through 8-14 are based on the 4-RCP, full-power LOCA LHR limits, which were verified to be applicable for 3-RCP operation. Table 8-1 presents the quadrant power tilt setpoints for cycle 11, Table 8-2 provides the negative moderator temperature coefficient limit for cycle 11, and Table 8-3 provides minimum linear heat rate to melt (kW/ft) limits. Table 8-4 provides the F_Q limits and Table 8-5 provides the $F^N_{\Delta H}$ limits. These limits are preserved by the rod index and axial power imbalance operating limits required by Technical Specification 3.1.3.6 and 3.2.1. The F_Q limits reflect the two different active fuel lengths and respective allowable linear heat rate limits as functions of incore segment (core elevation) and burnup. The $F^N_{\Delta H}$ relationship defined in Table 8-5 ensures acceptable DNBR performance using Statistical Core Design methodology in the event of the limiting Condition I and II transient. The family of curves in Figure 8-17 preserves the initial condition DNBR limit in the form of equivalent allowable initial condition peaking. Allowable $F^N_{\Delta H}$ values can be determined based on particular axial peaks at a given axial elevation for either three or four RC pump operation.

Based on the analyses and operating limit revisions described in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated.

Figure 8-1

Figure Regulating Group Position Alarm Setpoints
0 to 300±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



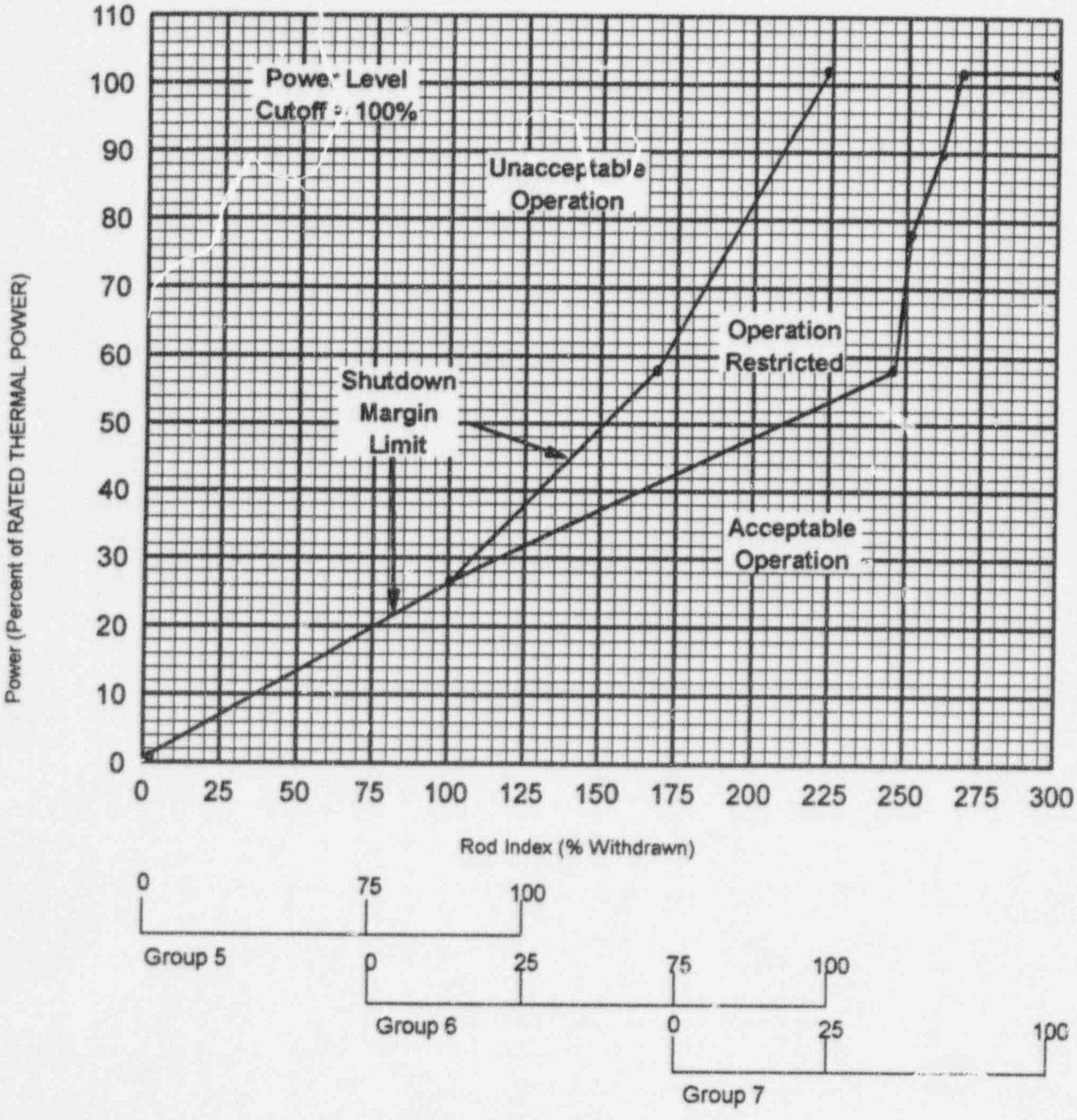
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-2

Figure Regulating Group Position Alarm Setpoints 300±10 to 610±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



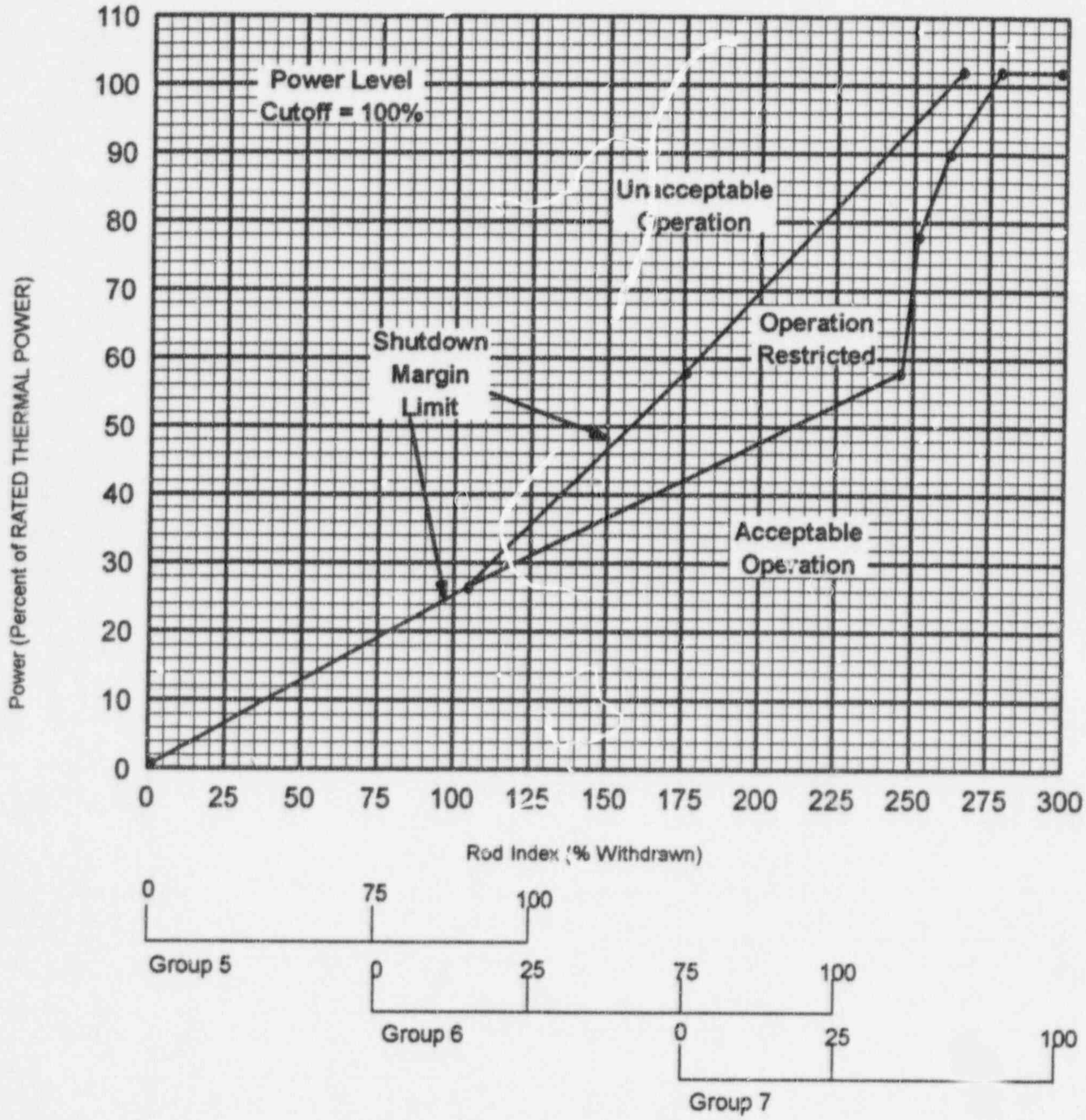
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-3

Figure Regulating Group Position Alarm Setpoints
After 810±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



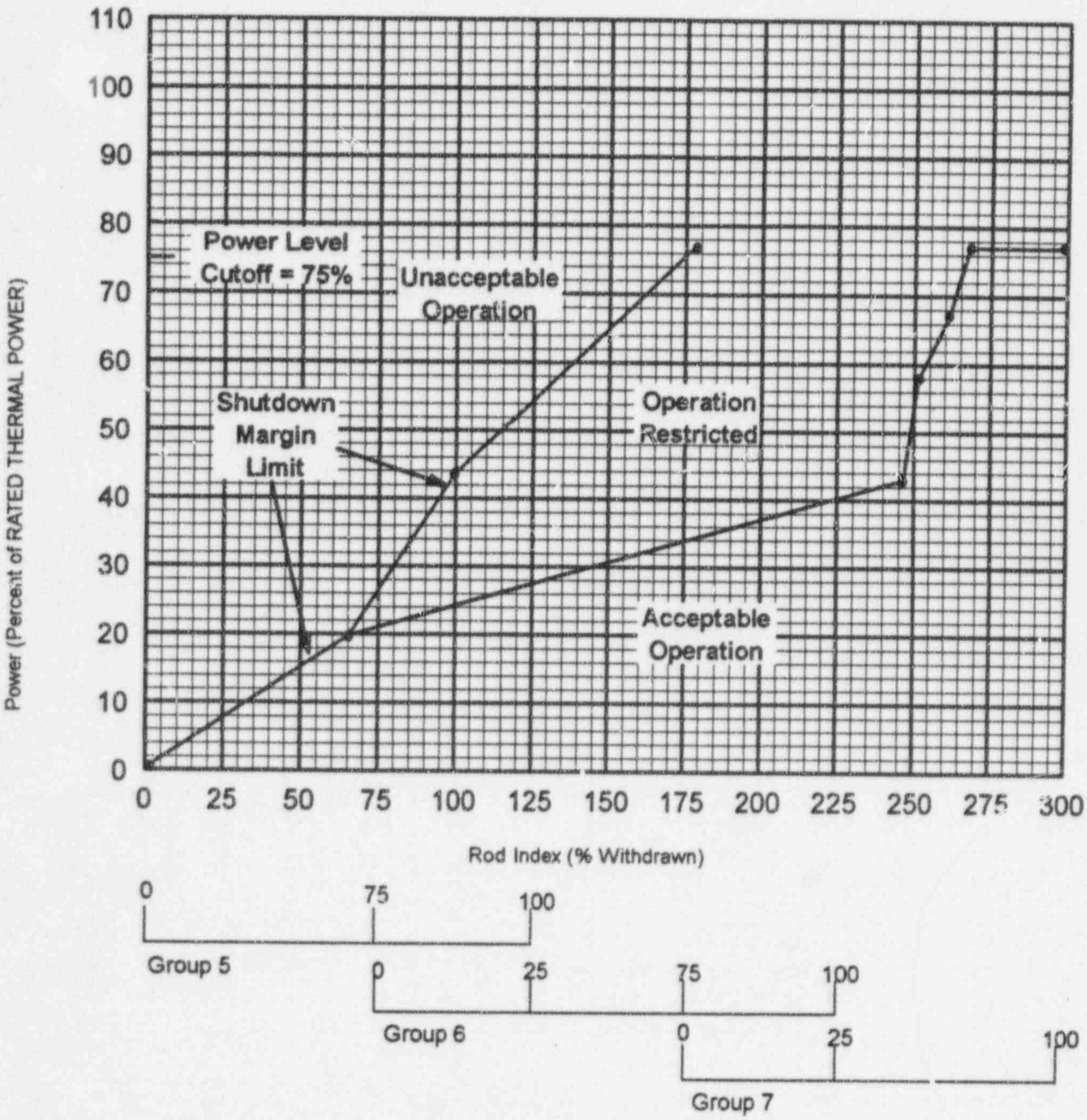
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-4

Figure — Regulating Group Position Alarm Setpoints
0 to 300±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



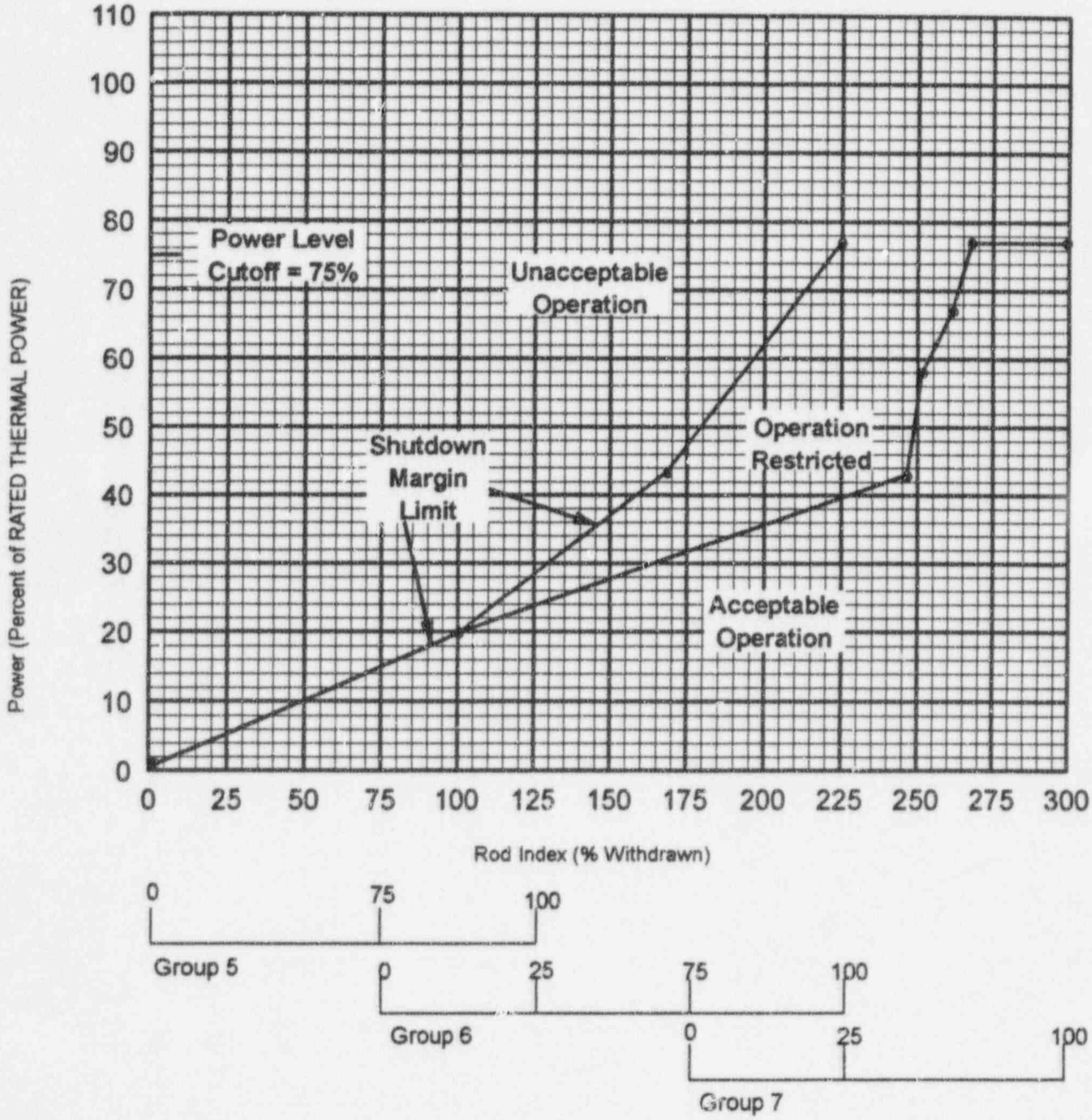
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-5

Figure Regulating Group Position Alarm Setpoints
300±10 to 610±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



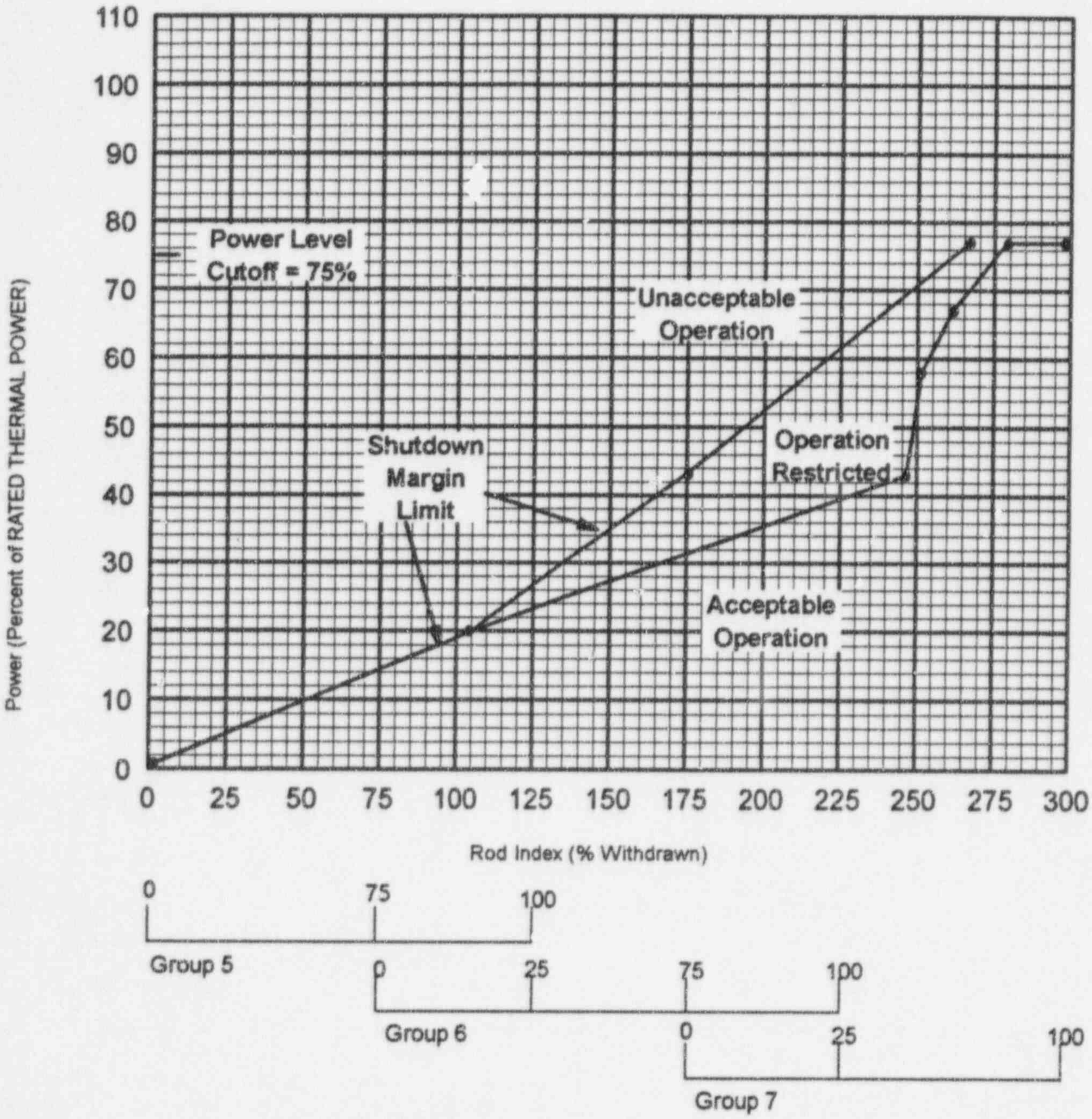
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-6

Figure Regulating Group Position Alarm Setpoints After 610±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



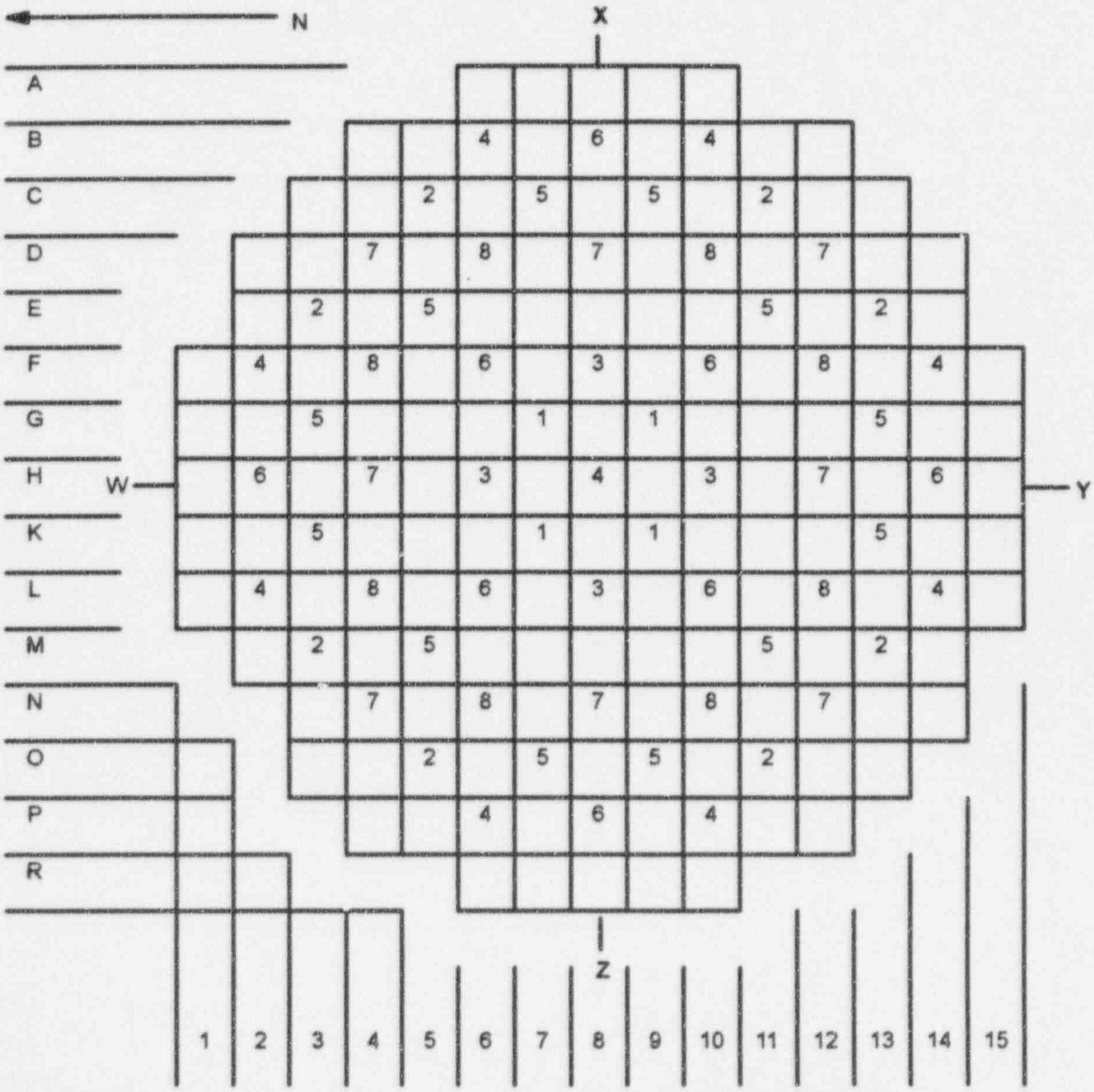
Note 1: A Rod Group overlap of 25±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these setpoints.

Figure 8-7

Figure Control Rod Core Locations
and Group Assignments
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specifications 3.1.3.7



X Group Number

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

Figure 8-8

Figure APSR Position Alarm Setpoints

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull - 0 EFPD to 610 +/- 10 EFPD,
<u>Three or Four RC pumps operation</u>*

Lower Setpoint: 0 %WD

Upper Setpoint: 100 %WD

After APSR Pull - 610 +/- 10 EFPD to End-of-Cycle
<u>Three or Four RC pumps operation</u>*

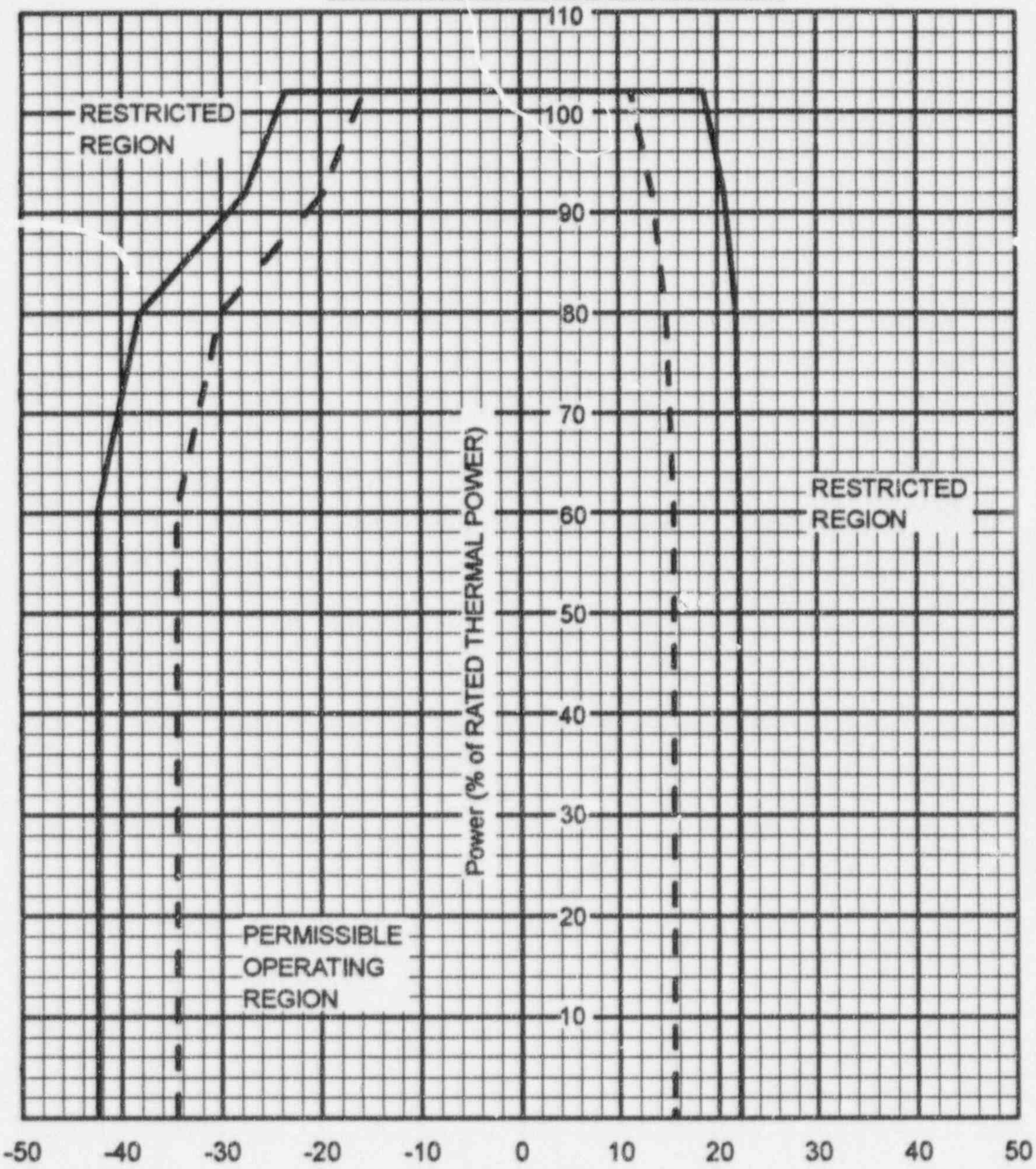
Insertion Prohibited per Tech Spec 3.1.3.9

* Power restricted to 77% for 3 pump operation

Figure 8-9

Figure AXIAL POWER IMBALANCE Alarm Setpoints
0 to 300±10 EFPD, Four RC Pumps –
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1



LEGEND

FULL INCORE

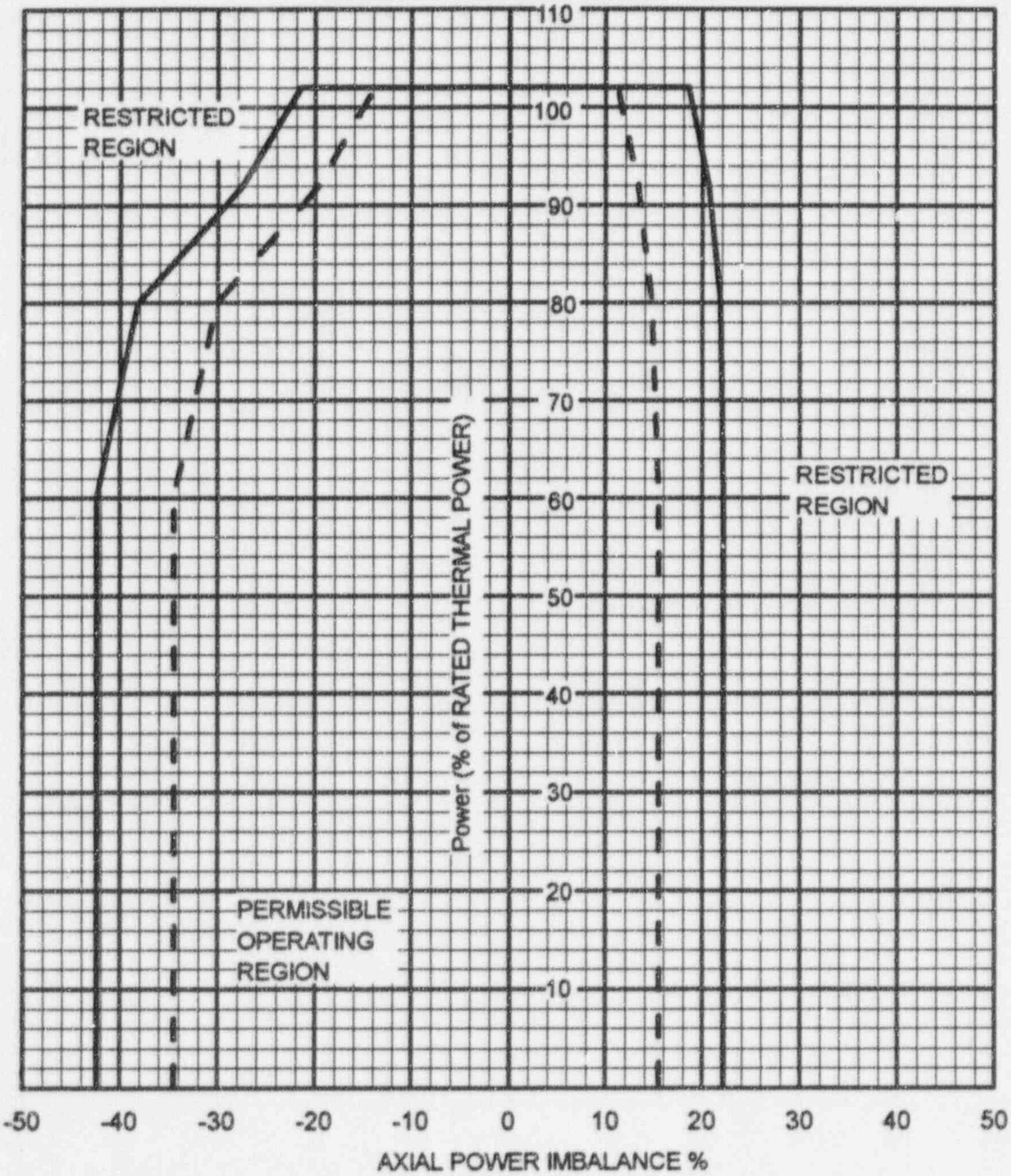
EXCORE

Note 1: Instrument error is accounted for in these setpoints

Figure 8-10

Figure AXIAL POWER IMBALANCE Alarm Setpoints
300±10 to 610±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1



LEGEND

FULL INCORE

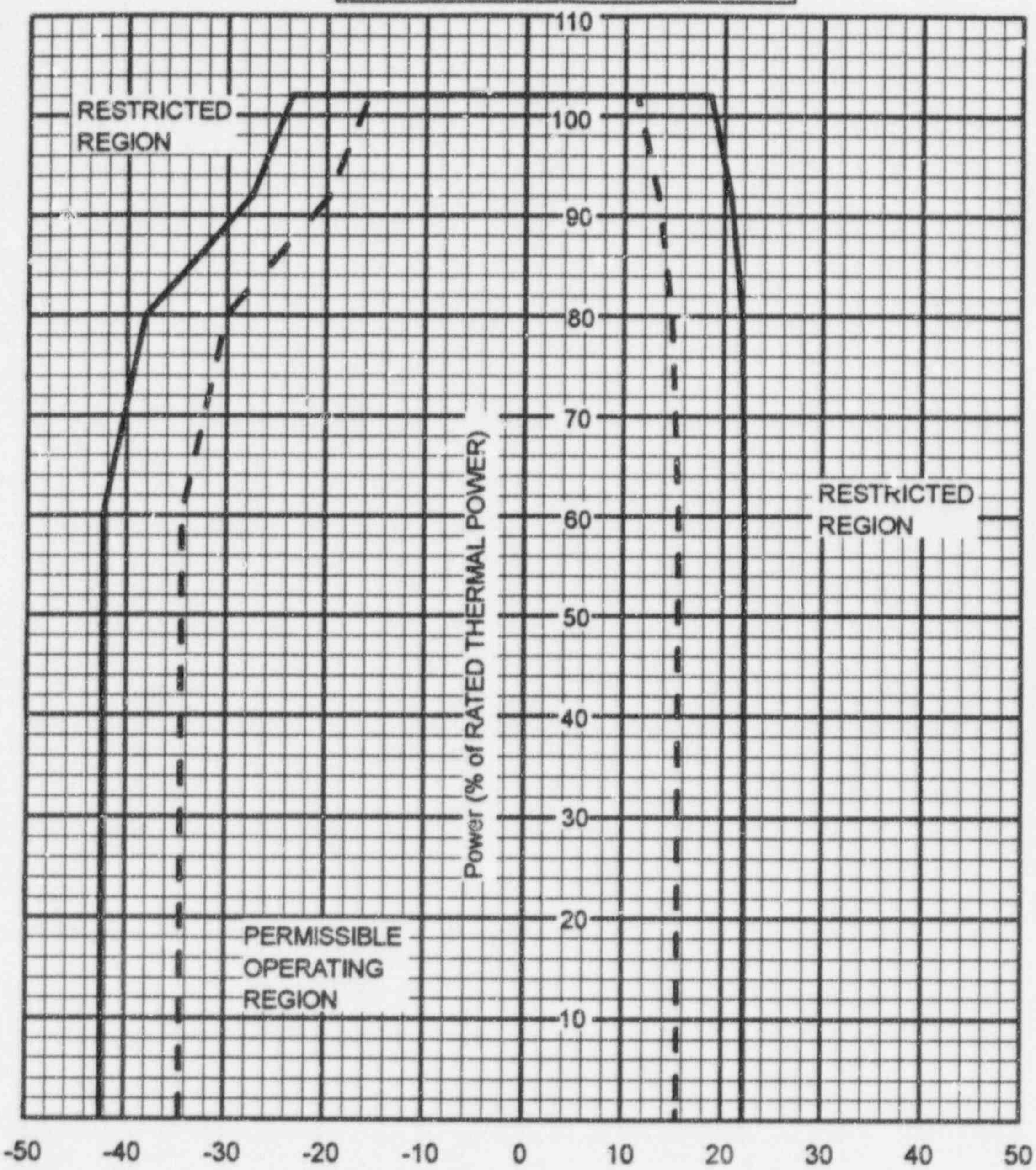
EXCORE

Note 1: Instrument error is accounted for in these setpoints

Figure 8-11

Figure AXIAL POWER IMBALANCE Alarm Setpoints
After 610±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1



LEGEND

FULL INCORE

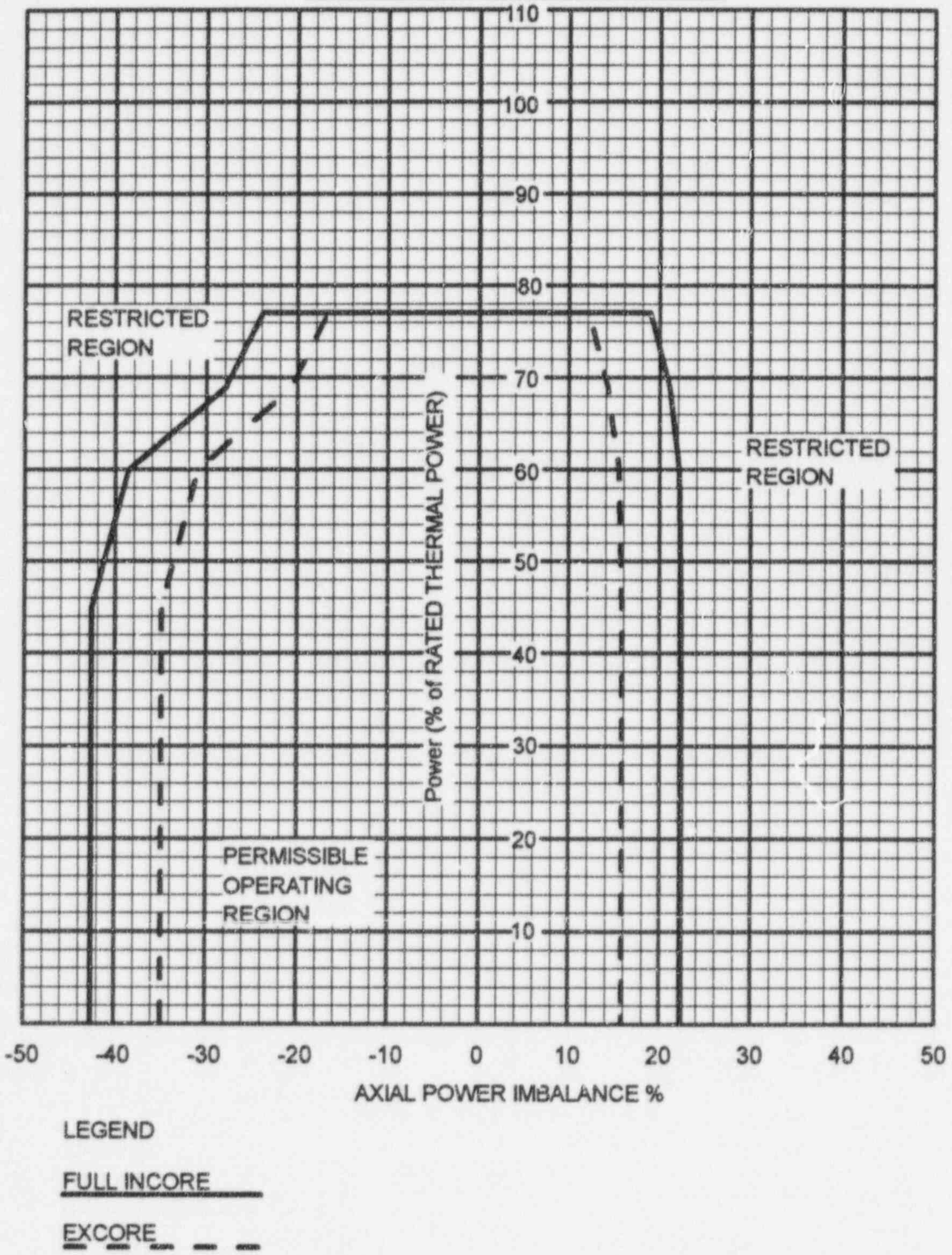
EXCORE

Note 1: Instrument error is accounted for in these setpoints

Figure 8-12

Figure AXIAL POWER IMBALANCE Alarm Setpoints
0 to 300±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1

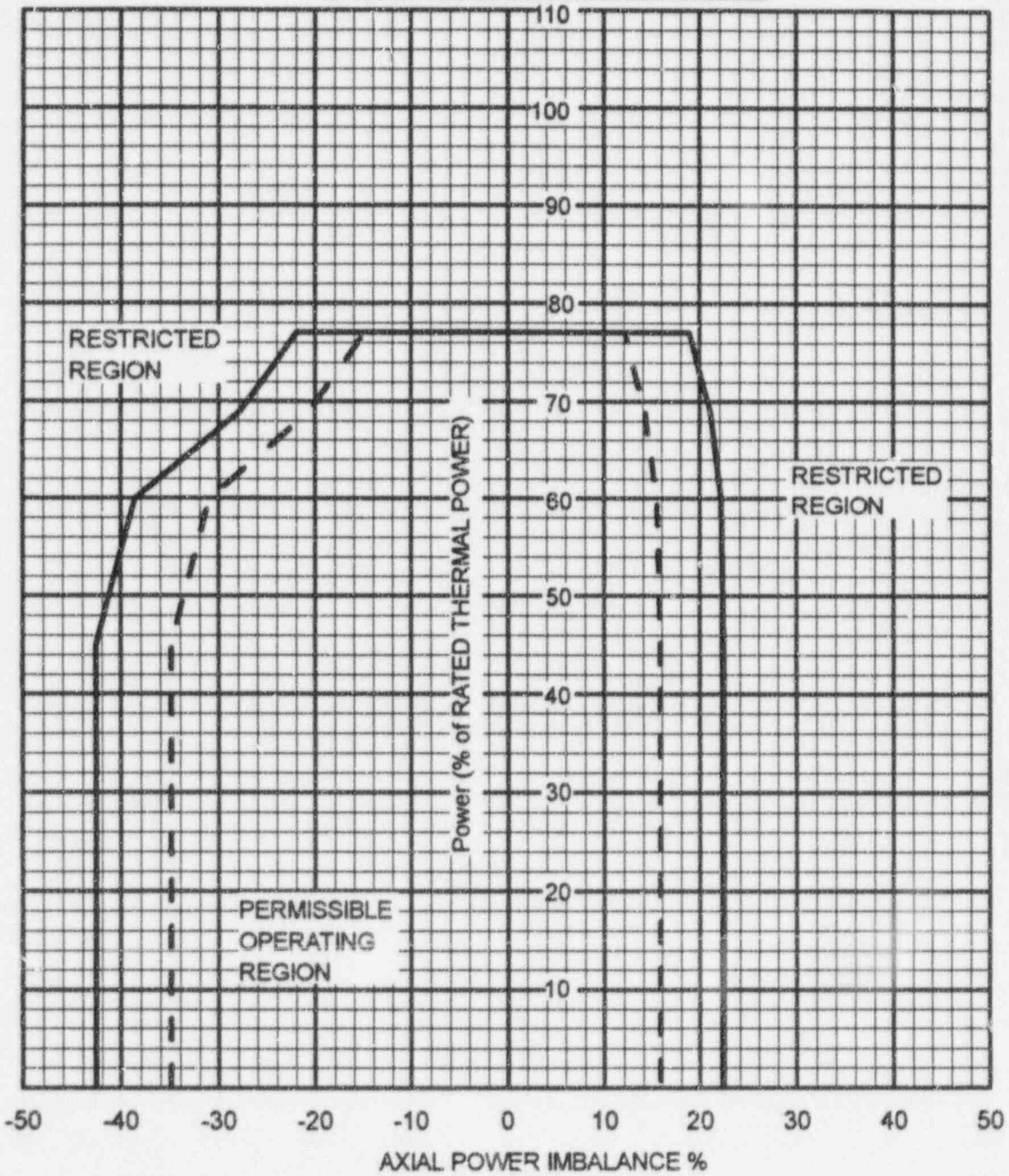


Note 1: Instrument error is accounted for in these setpoints

Figure 8-13

Figure AXIAL POWER IMBALANCE Alarm Setpoints 300±10 to 610±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1



LEGEND

FULL INCORE ______

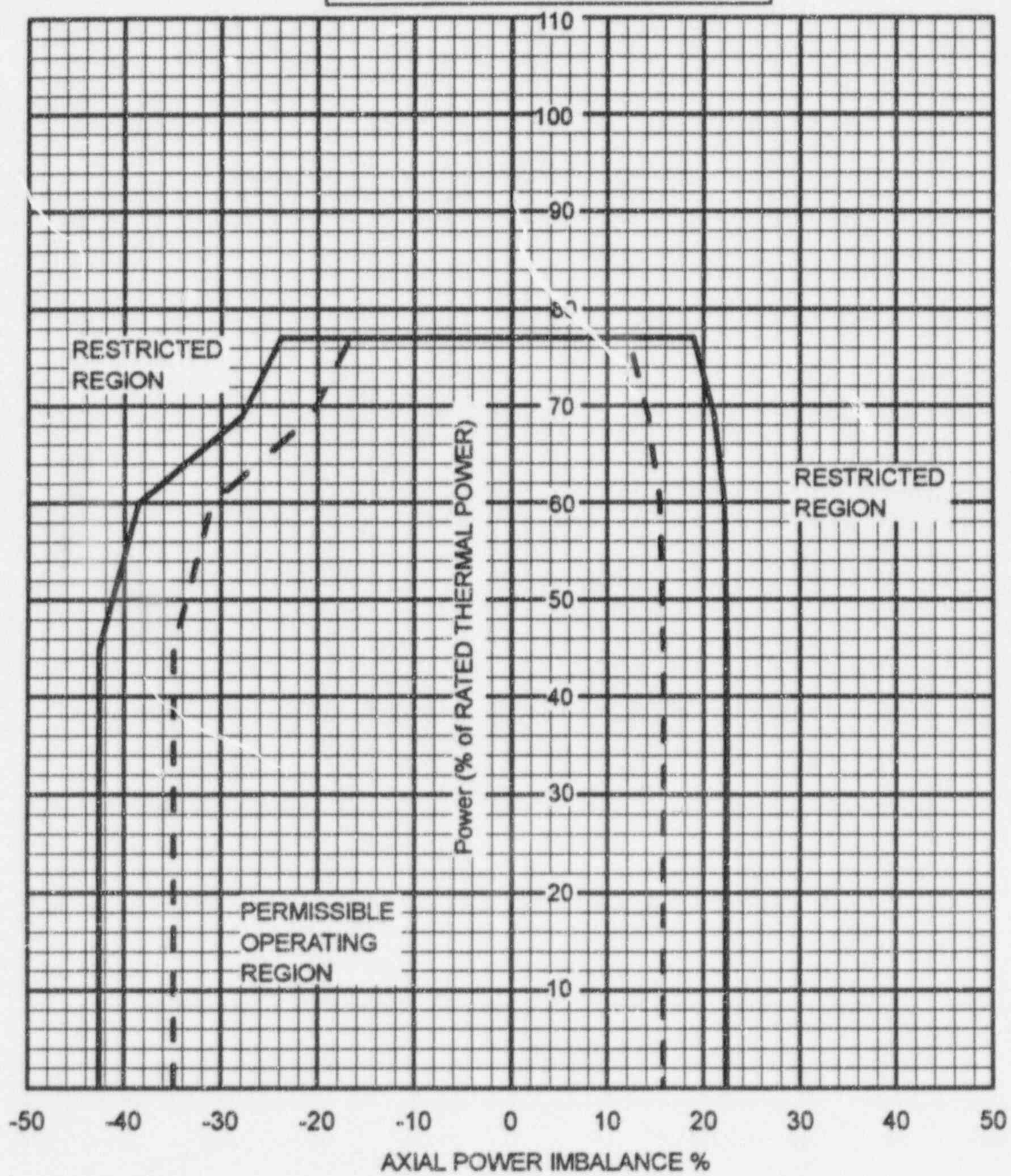
EXCORE — — — —

Note 1: Instrument error is accounted for in these setpoints

Figure 8-14

Figure AXIAL POWER IMBALANCE Alarm Setpoints
After 610±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 11

This Figure is referred to by Technical Specification 3.2.1



LEGEND

FULL INCORE ___

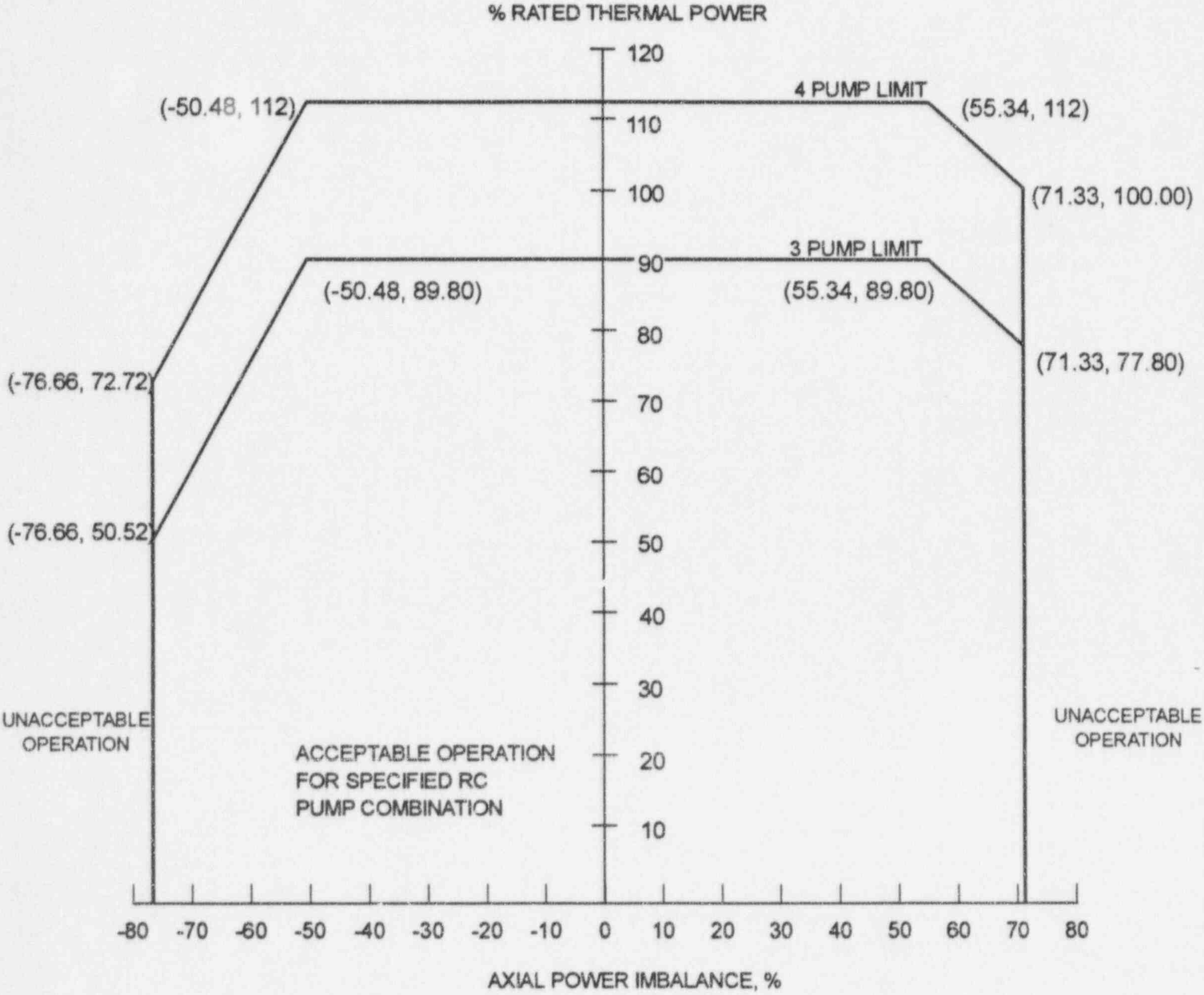
EXCORE _ _ _

Note 1: Instrument error is accounted for in these setpoints

Figure 8-15

Figure AXIAL POWER IMBALANCE Protective Limits

This Figure is referred to by Technical Specification 2.1.2



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

Figure 8-16

Figure Flux--ΔFlux/Flow (or Power/Imbalance/Flow) Trip Setpoints

This Figure is referred to by Technical Specification 2.2.1

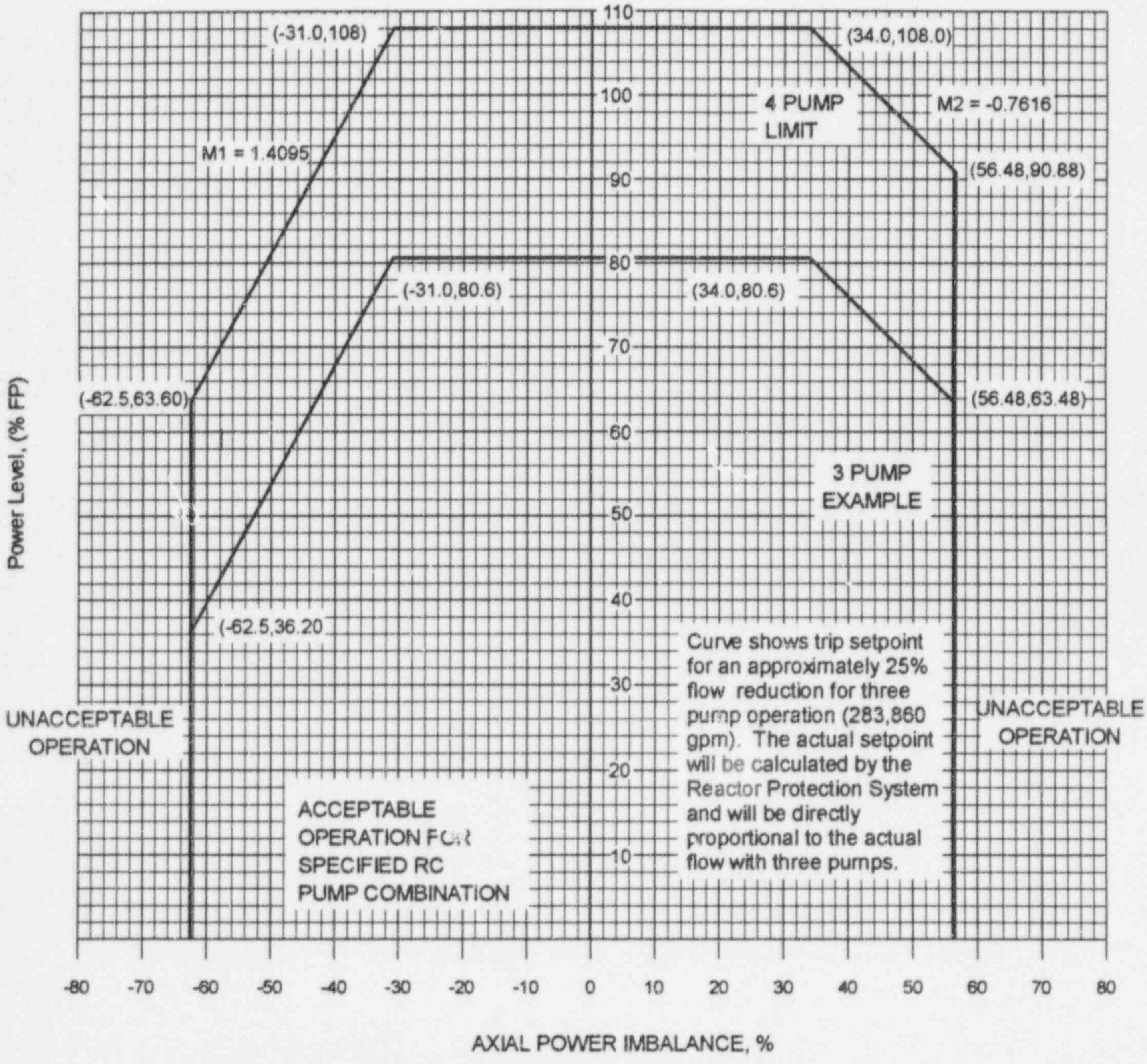


Table 8-1

Table QUADRANT POWER TILT Limits

This Table is referred to by Technical Specification 3.2.4

QUADRANT POWER TILT as measured by:	Steady-state Limit for THERMAL POWER ≤ 60% (%)	Steady-state Limit for THERMAL POWER > 60% (%)	Transient Limit (%)	Maximum Limit (%)
Symmetrical Incore detector system	6.8	4.1	10.03	20.0

Table 8-2

Table Negative Moderator Temperature Coefficient Limit

This Table is referred to by Technical Specification 3.1.1.3c

Negative Moderator Temperature Coefficient Limit (at RATED THERMAL POWER)	$-4x10^{-4}\Delta k/k/F$

Table 8-3

Table Power to Melt Limits

This Table is referred to by Technical Specification Bases B2.1

	Batch 9E	Batch 11C/11A	Batch 12B	Batch 13A/13B
Fuel Assembly Type	Mark-B8A	Mark-B8B	Mark-B10AZL	Mark-B10A
Minimum linear heat rate to melt, kW/ft	20.5	22.3	22.3	22.3

Table 8-4

Table Nuclear Heat Flux Hot Channel Factor - F_Q

This Table is referred to by Technical Specification 3.2.2

Heat Flux Hot Channel Factor F_Q

F_Q shall be limited by the following relationships:

$F_Q \leq LHR^{ALLOW}(Bu)/[\ LHR^{AVG} * P\ ]$ (for $P \leq 1.0$)

$LHR^{ALLOW}(Bu)$: See the Tables below

LHR^{AVG} = 6.139 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.253 kW/ft for Mark-B8B fuel

LHR^{AVG} = 6.253 kW/ft for Mark-B10AZL fuel

LHR^{AVG} = 6.253 kW/ft for Mark-B10A fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

Batch 9E (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Axial Segment	24,500 MWd/mtU	60,000 MWd/mtU
1	12.8	8.4
2	15.2	9.9
3	15.7	10.5
4	15.8	10.5
5	16.4	10.5
6	17.0	10.5
7	16.2	10.0
8	13.6	8.4

Batch 11A (Mark-B8B) LHR^{ALLOW} kW/ft[a]

Axial Segment	Less than 40,000 MWd/mtU	47,016 MWd/mtU	47,557 MWd/mtU	49,180 MWd/mtU	55,000 MWd/mtU	60,000 MWd/mtU
1	12.9	12.4	12.4	12.4	10.2	10.0
2	15.3	14.7	14.7	14.6	12.1	11.8
3	16.2	15.6	15.6	15.5	12.8	12.5
4	16.2	15.6	15.6	15.5	12.8	12.5
5	16.2	15.6	15.6	15.5	12.8	12.5
6	16.6	16.0	16.0	15.5	12.8	12.5
7	16.2	15.6	15.4	14.7	12.2	11.9
8	13.6	13.1	12.9	12.4	10.2	10.0

Table 8-4 (Con't)

Table Nuclear Heat Flux Hot Channel Factor - F_Q

Batch 11C (Mark-B8B) LHR^{ALLOW} kW/ft[a]

Axial Segment	Less than 40,000 MWd/mtU	47,016 MWd/mtU	47,557 MWd/mtU	49,180 MWd/mtU	55,000 MWd/mtU	60,000 MWd/mtU
1	12.9	12.4	12.4	12.4	10.2	10.0
2	15.3	14.7	14.7	14.6	12.1	11.8
3	16.2	15.6	15.6	15.5	12.8	12.5
4	16.2	15.6	15.6	15.5	12.8	12.5
5	16.2	15.6	15.6	15.5	12.8	12.5
6	16.6	16.0	16.0	15.5	12.8	12.5
7	16.2	15.6	15.4	14.7	12.2	11.9
8	13.6	13.1	12.9	12.4	10.2	10.0

Batch 12B (Mark-B10AZL) LHR^{ALLOW} kW/ft[a]

Axial Segment	Less than 40,000 MWd/mtU	47,016 MWd/mtU	47,557 MWd/mtU	49,180 MWd/mtU	55,000 MWd/mtU	60,000 MWd/mtU
1	12.9	12.4	12.4	12.4	10.2	10.0
2	15.3	14.7	14.7	14.6	12.1	11.8
3	16.2	15.6	15.6	15.5	12.8	12.5
4	16.2	15.6	15.6	15.5	12.8	12.5
5	16.2	15.6	15.6	15.5	12.8	12.5
6	16.6	16.0	16.0	15.5	12.8	12.5
7	16.2	15.6	15.4	14.7	12.2	11.9
8	13.6	13.1	12.9	12.4	10.2	10.0

Batches 13 A & B (Mark-B10A) LHR^{ALLOW} kW/ft[a]

Axial Segment	Less than 20,000 MWd/mtU	35,000 MWd/mtU
1	13.6	12.9
2	16.0	15.3
3	17.2	16.2
4	17.5	16.2
5	17.5	16.2
6	17.8	16.6
7	17.4	16.2
8	14.6	13.6

(a) Linear interpolation for allowable linear heat rate between specified burnup points is valid for these tables.

Table 8-5

<u>Table Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$</u>

This Table is referred to by Technical Specification 3.2.3

<u>Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$</u>

$F^N_{\Delta H} \leq$ ARP $[1 + 0.3(1 - P/P_m)]$

ARP = Allowable Radial Peak, see Figure

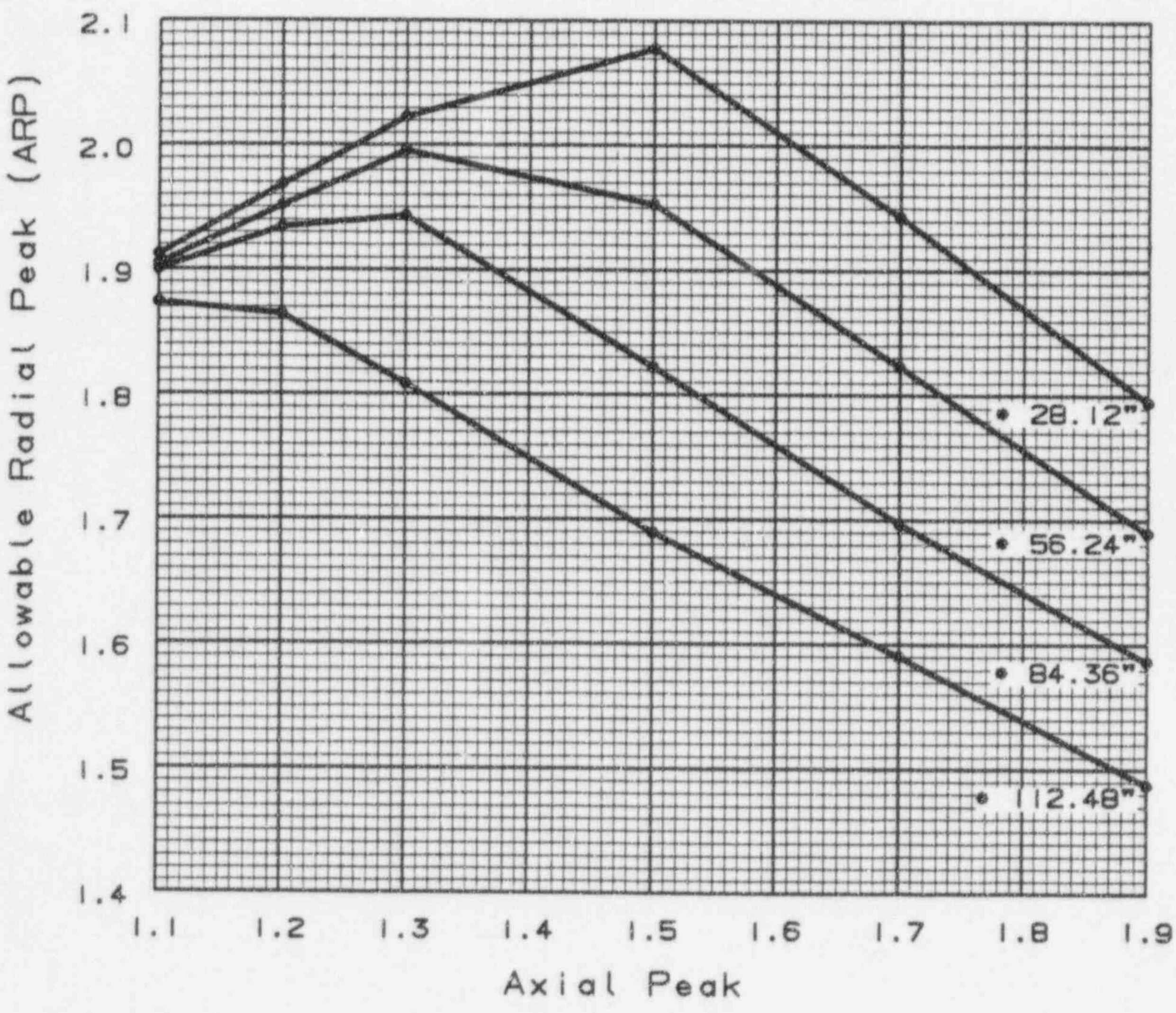
P = THERMAL POWER/RATED THERMAL POWER and $P \leq 1.0$

P_m = 1.0 for 4-RCP operation

P_m = 0.75 for 3-RCP operation

Figure 8-17

Figure Allowable Radial Peak for $F^N_{\Delta H}$



* Based on an active core height of 140.6 inches. Linear interpolation and extrapolation above 112.48 inches are acceptable. For axial heights <28.12 inches, the value at 28.12 inches will be used.

9. STARTUP PROGRAM - PHYSICS TESTING

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

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

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

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

9.1.2. RC Flow

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

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

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

9.2.2. Temperature Reactivity Coefficient

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

The moderator temperature coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator temperature coefficient (MTC). This value shall be less than +0.9 x 10^{-2} %Δk/k/°F. The MTC is also extrapolated to full power conditions.

9.2.3. Control Rod Group/Boron Reactivity Worth

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

1. Individual group 5, 6, 7 worth:

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

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

$$\left|\frac{\text{predicted value - measured value}}{\text{predicted value}}\right| \times 100\% \text{ shall be } \leq 6\%$$

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

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

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

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

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

9.3.2. Core Power Distribution Verification at Intermediate Power Level (IPL) and ~100% FP

Core power distribution tests are performed at the IPL and approximately 100% full power (FP). Equilibrium xenon is established prior to the ~100% FP test. The test at the IPL (40-80 %FP) is essentially a check of the power distribution in the core to identify any abnormalities before escalating to the ~100% FP plateau. Peaking factor criteria are applied to the IPL core power distribution results to determine if additional tests or analyses are required prior to ~100% FP operation.

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

1. The maximum F_Q values shall not exceed the limits specified in the COLR.

2. The maximum $F^N_{\Delta H}$ value shall not exceed the limits specified in the COLR.

3. The measured radial (assembly) peaks for each 1/8 core fresh fuel location shall be within the following limits:

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

4. The measured total (segment) peaks for each 1/8 core fresh fuel location shall be within the following limits:

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

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

5. The 1/8 core RMS of the differences between predicted and measured radial (assembly) peaking factors should be less than 0.05.

6. For all 1/8 core locations, the (absolute) difference between predicted and measured radial (assembly) peaking factors should be less than 0.10.

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

Items 3 and 4 are established to determine if measured and predicted power distributions are within allowable tolerances assumed in the reload analysis.

Items 5 and 6 are review criteria, established to determine if measured and predicted power distributions are consistent.

9.3.3. Incore vs. Excore Detector Imbalance Correlation Verification

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

9.3.4. Hot Full Power All Rods Out Critical Boron Concentration

The hot full power (HFP) all rods out critical boron concentration (AROCBC) is determined at ~100% FP by first recording the RCS boron concentration during equilibrium, steady state conditions. Corrections to the measured RCS boron concentration are made for control rod group insertion and power deficit (if not at 100%FP) using predicted data for CRG worth, power Doppler coefficient, and differential boron worth. A correction may also be made to account for the observed difference between the measured and predicted AROCBC at zero power. The review criterion placed on the HFP AROCBC is that the measured AROCBC should be within ± 50 ppm boron of the predicted value.

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

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

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

10. REFERENCES

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

2. Davis-Besse Nuclear Power Station Unit 1, Cycle 10 Reload Report, BAW-2223 Revision 1, B&W Fuel Company, Lynchburg, Virginia, dated October 1994.

3. TACO3: Fuel Pin Thermal Analysis Computer Code, BAW-10162P-A, Babcock & Wilcox, Lynchburg, Virginia, dated November 1989.

4. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084P-A, Rev. 3, Babcock and Wilcox, Lynchburg, Virginia, dated July 1995.

5. TACO2: Fuel Performance Analysis, BAW-10141P-A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, dated June 1983.

6. Fuel Rod Gas Pressure Criterion (FRGPC), BAW-10183P-A, B&W Fuel Company, Lynchburg, Virginia, dated July 1995.

7. NEMO- Nodal Expansion Method Optimized, BAW-10180-A, Rev. 1, B&W Fuel Company, Lynchburg, Virginia, dated March 1993.

8. Letter, Robert Jones (NRC) to J. H. Taylor (FIT), Subject: Acceptance of Revised Measurement Uncertainty for Control Rod Worth Calculations, dated January 26, 1996.

9. Safety Criteria and Methodology for Acceptable Cycle Reload Analysis, BAW-10179P-A, B&W Fuel Company, Lynchburg, Virginia, dated August 1993.

10. Statistical Core Design for B&W-Designed 177-FA Plants, BAW-10187P-A, B&W Fuel Company, Lynchburg, Virginia, dated March 1994.

11. Evaluation of Replacement Rods in BWFC Assemblies, BAW-2149-A, B&W Fuel Company, Lynchburg, Virginia, dated September 1993.

12. Davis-Besse Unit 1 Fuel Densification Report, BAW-1401, Babcock & Wilcox, Lynchburg, Virginia, dated April 1975.

13. B&W's ECCS Evaluation Model, BAW-10104P, Rev. 5, Babcock & Wilcox, Lynchburg, Virginia, dated April 1986.

14. ECCS Evaluation of B&W's 177-FA Raised-Loop NSS, BAW-10105, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, dated July 1975.

Davis-Besse Nuclear Power Station Unit 1
Cycle 11 Reload Report Addendum

Introduction

At the completion of cycle 10 operation, each fuel assembly was ultrasonically tested for leaking fuel rods. As a result of this complete core inspection, three fuel assemblies were identified as having one leaking fuel rod each. A reconstitution campaign was initiated to replace the identified failed fuel rods with stainless steel rods. Two assemblies, a twice-burned (NJ06K6), and a once-burned (NJ07KR), received one stainless steel rod each to replace the leaking fuel rods. However, in the remaining fuel assembly (NJ07L0), the leaking fuel rod separated during reconstitution and the assembly was discharged along with three symmetric assemblies. A redesign was performed for cycle 11 using a replacement set of four fuel assemblies that were similar in burnup to NJ07L0. This addendum summarizes the analyses performed to verify operation of the redesigned core. The analyses performed for the original design are presented in BAW-2271, Rev. 1.

Fuel assembly NJ06KZ, which will be reinserted from cycle 9, was reconstituted prior to the cycle 10 outage and contains one stainless steel replacement rod. Assembly NJ06KZ will receive its second cycle of irradiation during cycle 11.

Davis-Besse Unit 1 completed cycle 10 operation on April 8, 1996 with an EOC burnup of 501.3 EFPD; this is within the allowable shutdown window of 488±15 EFPD. All evaluations in this addendum are still based on the original cycle 10 design length of 488 EFPD.

Fuel Cycle Design Analysis

Figure 3-1a is the revised core loading diagram for Davis-Besse Unit 1 cycle 11. The only changes to the core loading diagram were for locations G07, G09, K07, and K09. Four batch 12 assemblies were substituted for NJ07L0 and three symmetrically located assemblies. The burnup of the substituted assemblies is 647 MWd/mtU greater than those they replaced. The core locations and group designations of the control rods and the BPRA locations, and their concentrations, are the same as in BAW-2271, Rev. 1.

The three reconstituted assemblies, designated NJ06KZ, NJ06K6, and NJ07KR, will be loaded into core locations D08, K01, and G11, respectively, as shown in Figure 3-1a.

Figure 3-2a is a quarter-core map showing each assembly's burnup at the BOC-11 and its initial enrichment. Figure 5-1a illustrates a representative relative power distribution for BOC-11 (4 EFPD) at full power with equilibrium xenon, group 7 inserted to nominal HFP position, and gray APSRs partially inserted. The redesigned cycle 11 core was analyzed with the NEMO code, based on a 488±15 EFPD cycle 10 length.

The impacts of the redesign on power distribution and reactivity were small relative to the original design. The primary changes observed in the core power distribution and reactivity were:

1) assembly powers were reduced by 2% in the central nine assemblies. Core power shifted slightly toward the outside of the core. However, there was less than a 0.2% increase in the core peak pin power at any time in cycle 11.

2) The critical boron concentration was reduced by 3 ppm at BOC. The HFP lifetime was reduced by about 0.5 EFPD due to this design change.

Each of the reconstituted assemblies contains one stainless steel replacement rod. The impact of the stainless steel rods on core power distribution was conservatively estimated. A 3% increase in peak pin power for assemblies containing stainless steel rods and a 0.5% increase in peak pin power for the remaining assemblies bound the effects of the stainless steel rods. These estimates are conservative based on a comparison of the Davis-Besse Unit 1 cycle 11 redesign with previous core designs that resulted in more limiting configurations of reconstituted fuel assemblies. The criteria related to peaking, DNBR performance, and ECCS performance specified in BAW-2149-A were not exceeded with the introduction of the reconstituted fuel.

Mechanical Design

The redesign of cycle 11 resulted in small changes to the end of cycle maximum pin burnups. Those changes are summarized in Table 4-1a and were found acceptable for fuel rod performance analysis. The effects of the stainless steel replacement pins were also considered and found acceptable. None of the cycle 11 fuel assembly burnups exceeded the fuel rod cladding collapse burnup of 60,000 MWd/mtU. Therefore, the mechanical analysis results reported in BAW-2271, Rev.1 remain valid for cycle 11.

Thermal Design

The results of the thermal design evaluation summarized in BAW-2271, Rev. 1 remain valid and applicable for the redesigned core with the exception of the predicted maximum pin burnup for each fuel batch. Maximum pin burnups are provided in Table 4-1a.

The fuel rod internal pressures have been evaluated with TACO3 for the burnup of each fuel rod type. The batch 13 fuel internal pressure was predicted to be less than the nominal reactor coolant pressure of 2200 psia. The internal pressures for the remaining batches (9E, 11A, 11C, and 12B) exceed 2200 psia and were justified with the fuel rod gas pressure criterion methodology described in BAW-10183P-A.

Thermal-Hydraulic Design

The thermal-hydraulic design evaluation summarized in section 6 of BAW-2271, Rev.1 was not impacted by the cycle 11 redesign. The thermal-hydraulic impact of the three stainless steel replacement rods was conservatively evaluated for cycle 11 following the guidelines established in BAW-2149-A. The anticipated local peaking change as a result of the replacement rods was determined to be bounded by the local peaking changes predicted for a more limiting situation (TMI-1 cycle 11) where up to 10 replacement rods were used in a single fuel assembly. The analyses that justify the TMI-1 configuration are presented in BAW-2250, ADDENDUM 1. The DNBR impact for the TMI-1 cycle 11 arrangement was found to be acceptable, therefore, the less limiting arrangement of replacement rods in Davis-Besse 1 cycle 11 was also deemed acceptable. The impact of the replacement rods on the local flow redistribution was also conservatively determined and evaluated for its impact on the LBLOCA analyses as discussed in the following section.

ECCS Analysis

The thermal-hydraulic evaluation concluded that the original thermal-hydraulic analyses remain valid for the redesigned core. Based on those results, the Mark-B8A, Mark-B8B, Mark-B10AZL, and Mark-B10A LOCA LHR limits established for the original cycle 11 core remain applicable to the redesigned core.

The local flow redistribution impact of the three stainless steel replacement rods in Davis-Besse 1 cycle 11 was bounded by the results of the TMI-1 cycle 11

evaluation where up to 10 stainless steel replacement rods were introduced into a single fuel assembly. The presence of up to 10 stainless steel replacement rods in a single fuel assembly can reduce the flowrate in adjacent fuel assemblies up to 0.5%. For LBLOCA analyses, a reduction in the initial steady-state flow distribution of less than 0.5 percent represents a negligible change that would not affect the results predicted during a LOCA transient. Therefore, the three replacement stainless steel rods in the redesigned cycle 11 core do not adversely affect the allowable LOCA LHR limits.

Nuclear Analysis

The nuclear characteristics in the original cycle 11 licensing analysis were evaluated to determine the impact of the redesign described above, including the effects of the stainless steel rods. This redesign introduces small perturbations in the core design. A limited number of physics parameters were recalculated to show that the impacts of the redesign on these physics parameters were small. These parameters represent those which are likely to be sensitive to this redesign or change in the direction where the redesign value is more limiting. Table 5-1a provides a summary of the changes in the physics parameters.

This redesign did not cause large changes in the nuclear parameters, as seen in Table 5-1a. For instance, the HFP critical boron concentration decreased by approximately 3 ppm throughout the cycle relative to the base design. A slight decrease in the critical boron concentration was expected because the redesign resulted in the substitution of four assemblies with the same enrichment but higher burnup (i.e., less core excess reactivity).

The assembly substitutions also produce a higher core average burnup, which leads to a decreased (more negative) moderator temperature coefficient (MTC). For example, the EOC HFP MTC decreased by 0.01 x 10^{-4} $\Delta k/k/^{\circ}F$. At BOC, the MTC also decreased, but the magnitude was much smaller. Since the changes in the Doppler temperature coefficient were insignificant, it follows that the temperature coefficients would exhibit changes similar to those seen for the MTCs. The more negative EOC coefficients, including those for checking MSLB, were evaluated in the accident analysis.

The HFP MTC COLR limit remained unchanged. This limit is the more limiting of -4.0 x 10^{-4} $\Delta k/k/^{\circ}F$ and the transformed MSLB limit. The transformation of the MSLB temperature coefficient limit resulted in a value that was more positive than that calculated for the original design; however, neither transformed value was more limiting than the -4.0 x 10^{-4} $\Delta k/k/^{\circ}F$ limit. Therefore, the MTC COLR limit did not change for the redesign.

The variations in rod worths with this redesign were also minor. For example, the HZP total rod worth at BOC decreased by 0.01 %$\Delta k/k$. This reduction in total worth was the largest decrease observed for the cycle. At BOC, the HZP ejected rod worth increased from 0.28 to 0.30 %$\Delta k/k$. The redesign perturbation that caused this increased worth, however, was quickly dampened with power level; the BOC ejected rod worth at HFP only increased by 0.003 %$\Delta k/k$ (Table 5-1a indicates the difference is 0.01 %$\Delta k/k$ because of less significant digits). With burnup, the differences in ejected rod worth between the two designs were negligible. The redesign did not significantly alter stuck rod worth as seen in Table 5-1a.

The core response to changes in the temperature, power level, and flux distribution as measured by the power deficit (HFP to HZP) did not change significantly with the redesign. At both BOC and MOC, the power deficit remained unchanged, while the power deficit at EOC decreased approximately 0.02 %$\Delta k/k$. An illustration of the shutdown margin for the redesign is shown in Table 5-2a. Comparisons of this table to Table 5-2 in BAW-2271, Rev. 1 show the net impact on shutdown margin due to the changes in the deficits as well as the rod worths.

In particular, the shutdown margin decreased at BOC by approximately 0.01 %Δk/k. With burnup, however, the redesign eventually shows additional shutdown margin. The impacts on rod insertion limits due to the decrease in shutdown margin and the increase in the HZP BOC ejected rod worth were evaluated in the maneuvering analysis discussed below.

Maneuvering Analysis

A power distribution evaluation of the redesigned cycle 11 core was performed to verify that the core protective and operating limits for the original core design are applicable. The evaluation was performed in accordance with the approved methods described in BAW-10179P-A. Margins to the F_q and $F^N_{\Delta H}$ peaking limits were determined to be bounded by limits set for the original analysis. The reactivity parameters pertaining to determination of rod index limits based on shutdown margin and ejected rod worths were examined using inputs for the redesigned core. No changes to the reactivity based rod index limits were required. The three stainless steel rods in the core were determined to have an insignificant impact on the core power distribution for the resigned core. Based on the results of the evaluation, it is concluded that the core protective limits and operating limits provided for the original cycle 11 design are valid and applicable for operation with the cycle 11 redesign.

Accident Analysis

The reactor core kinetics and other fuel design parameters used in the USAR safety analyses were compared with the associated parameters for the Davis-Besse Unit 1 cycle 11 redesign. Since the licensing parameters for the redesign are similar to the original design, the USAR analyses remain bounding for the redesigned core.

Trip Setpoint and Protective Limit Verification

The RPS power/imbalance/flow trip setpoints were evaluated to verify their continuing applicability to the cycle 11 redesign. The flux-to-flow limit used in the original cycle 11 RPS trip setpoint preparation did not change due to the cycle 11 redesign. In addition, the RPS offset protective limits, the detector uncertainty, and the error adjustment factors used in the original cycle 11 analysis were not adversely affected by the cycle 11 redesign. Therefore, the power/imbalance/flow trip setpoints and Axial Power Imbalance Protective Limits determined during the original cycle 11 analysis remain valid.

Control Analysis

Since the soluble boron shutdown requirements for the redesigned cycle 11 are less than those of the original design, the results of the original analysis remain valid and are conservative for cycle 11.

Radiological Evaluation

The original cycle 11 offsite dose consequences have been examined relative to the cycle 11 redesign. The dose consequences as determined for the original cycle 11 design remain valid.

Figure 3-1a. Davis-Besse Cycle 11 Core Loading Diagram

← North

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						11C G13	11C M03	11C G09	11C M13	11C G03					
B				11C L05	13A F	12B P09	13B F	11C P10	13B F	12B P07	13A F	11C L11			
C			11C L14	13B F	12B M02	13B F	12B D13	13B F	12B D03	13B F	12B M14	13B F	11C P06		
D		11C E10	13B F	12B L09	13B F	12B F13	13B F	11A N09 9	13B F	12B F03	13B F	12B K06	13B F	11C E06	
E		13A F	12B B11	13B F	12B M04	13B F	12B K04	13A F	12B K12	13B F	12B D05	13B F	12B B05	13A F	
F	11C O07	12B K14	13B F	12B O06	13B F	12B O08	13B F	12B K10	13B F	12B H03	13B F	12B O10	13B F	12B K02	11C O09
G	11C C11	13B F	12B O04	13B F	12B D09	13B F	12B H11	13B F	12B M08	13B F	12B D07	13B F	12B O12	13B F	11C O05
H W—	11C G07	11C F14	13B F	11A G12 9	13A F	12B F09	13B F	9F N04 8	13B F	12B L07	13A F	11A K04 9	13B F	11C L02	11C K09
K	11C O11	13B F	12B C04	13B F	12B N09	13B F	12B E08	13B F	12B H05	13B F	12B N07	13B F	12B C12	13B F	11C C05
L	11C C07	12B G14	13B F	12B C06	13B F	12B H13	13B F	12B G06	13B F	12B C08	13B F	12B C10	13B F	12B G02	11C C09
M		13A F	12B P11	13B F	12B N11	13B F	12B G04	13A F	12B G12	13B F	12B E12	13B F	12B P05	13A F	
N		11C M10	13B F	12B G10	13B F	12B L13	13B F	11A D07 9	13B F	12B L03	13B F	12B F07	13B F	11C M06	
O			11C B10	13B F	12B E02	13B F	12B N13	13B F	12B N03	13B F	12B E14	13B F	11C F02		
P				11C F05	13A F	12B B09	13B F	11C B06	13B F	12B B07	13A F	11C F11			
R						11C K13	11C E03	11C K07	11C E13	11C K03					

Z

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

Note: One stainless steel replacement rod in D08, K01, and G11.

Figure 3-2a. Davis-Besse Cycle 11 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.38 34540	4.46 0	4.06 21255	4.46 0	3.77 22855	4.46 0	3.77 35320	3.77 39581
K	4.46 0	4.06 22831	4.46 0	4.06 22976	4.46 0	4.06 17575	4.46 0	3.77 36960
L	4.06 21255	4.46 0	4.06 22251	4.46 0	4.06 21600	4.46 0	4.06 18791	3.77 41686
M	4.46 0	4.06 22978	4.46 0	4.06 22193	4.46 0	4.06 14784	4.46 0	
N	3.77 22855	4.46 0	4.06 21586	4.46 0	4.06 21206	4.46 0	3.77 36112	
O	4.46 0	4.06 17644	4.46 0	4.06 14808	4.46 0	3.77 35334		
P	3.77 35320	4.46 0	4.06 18718	4.46 0	3.77 36147			
R	3.77 39581	3.77 37006	3.77 41690					

x.xx yyyyy	Initial Enrichment BOC Burnup MWd/mtU

Table 4-1a Fuel Design Parameters and Dimensions

Batch	9E	11A	11C	12B	13A/13B
Maximum pin burnup, MWd/mtU	58,423	49,956	54,589	52,054	34,056

Table 5-1a Comparison of Selected Davis-Besse Cycle 11 Physics Parameters

	Original[a] Cycle 11	Redesign[a] Cycle 11
Cycle length, EFPD	675	675
Cycle burnup, MWd/mtU	22,561	22,561
Average core burnup -- 675 EFPD, MWd/mtU	37,745	37,759
Critical boron -- 0 EFPD, (no Xe), ppm HFP, group 8 inserted	2225	2222
Max ejected rod worth -- %Δk/k		
BOC, HZP	0.28	0.30
BOC, HFP	0.20	0.21
EOC, HFP	0.22	0.22
Max stuck rod worth -- HZP, %Δk/k		
BOC	0.46	0.46
EOC	0.60	0.60
Power deficit, HZP to HFP, %Δk/k		
BOC	-1.68	-1.68
EOC	-3.29	-3.27
Doppler coeff -- HFP, 10^{-3} (%Δk/k/°F)		
BOC(no Xe, critical boron, group 8 inserted)	-1.51	-1.51
EOC(eq Xe, 0 ppm, group 8 inserted)	-1.82	-1.83
Moderator coeff -- HFP, 10^{-2} (%Δk/k/°F)		
BOC(no Xe, critical boron, group 8 inserted)	-0.19	-0.19
EOC(eq Xe, 0 ppm, group 8 inserted, group 7 at the rod insertion limit)	-3.51	-3.52
Temperature coeff -- 532°F to 510°F, 10^{-2} (%Δk/k/°F)		
EOC (eq Xe, 0 ppm, ARI-M11)	-2.63	-2.65

[a] Based on cycle 9 length of 500.8 EFPD (actual) and cycle 10 length of 488 EFPD.

Table 5-2a Shutdown Margin Illustration for Davis-Besse Cycle 11 Redesign

	BOC %Δk/k	EOC[a] %Δk/k
Available Rod Worth		
Total rod worth, HZP	6.04	6.73
Worth reduction due to burnup of poison material	0.00	-0.03
Maximum stuck rod, HZP	-0.46	-0.60
Net worth	5.58	6.10
Less 6% uncertainty	-0.34	-0.37
Total available worth	5.24	5.73
Required Rod Worth		
Power deficit, HFP to HZP	1.68	3.27
Xenon transient allowance	0.30	0.30
Max allowable inserted rod worth	0.31	0.52
Total required worth	2.29	4.09
Shutdown Margin		
Total available minus total required	2.95	1.64

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

[a] 675 EFPD.

Figure 5-1a. Davis-Besse Cycle 11 Relative Power Distribution at BOC (4 EFPD), Full Power, Equilibrium Xenon, Rods at Nominal HFP Positions

	8	9	10	11	12	13	14	15
H	0.789	1.195	1.191	1.303	7 1.125	1.300	0.666	0.242
K	1.195	1.127	1.305	1.188	1.299	1.192	1.093	0.271
L	1.191	1.303	1.205	1.321	8 1.160	1.260	0.813	0.209
M	1.303	1.187	1.318	1.214	1.326	1.165	0.876	
N	7 1.125	1.298	8 1.159	1.325	7 1.143	1.159	0.346	
O	1.300	1.188	1.257	1.163	1.158	0.405		
P	0.666	1.086	0.811	0.875	0.346			
R	0.242	0.269	0.209					

x	Inserted Rod Group Number
x.xxx	Relative Power Density

REFERENCES

Davis-Besse Nuclear Power Station Unit 1, Cycle 11 Reload Report, BAW-2271, Revision 1, Framatome Cogema Fuels, Lynchburg, Virginia, dated April 1996.

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Safety Criteria and Methodology for Acceptable Cycle Reload Analysis, BAW-10179P-A, Rev. 1, Framatome Cogema Fuels, Lynchburg, Virginia, dated February 1996.