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DAVIS-BESSE NUCLEAR POWER STATION

UNIT 1, CYCLE 6 -- RELOAD REPORT

Babcock & Wilcox
a McDermott company

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1. INTRODUCTION AND SUMMARY

This report justifies operation of Davis-Besse Nuclear Power Station Unit 1 at the rated core power of 2772 MWt for cycle 6. 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 6 reactor and fuel parameters related to power capability are summarized in this report and compared to cycle 5. All accidents analyzed in the Davis-Besse Final Safety Analysis Report[1] (FSAR) or the Updated Safety Analysis Report[2] (USAR), as applicable, have been reviewed for cycle 6 operation, and in cases where cycle 6 characteristics were conservative compared to cycle 1, no new analyses were performed.

The Technical Specifications have been reviewed and modified where required for cycle 6 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 6 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 recently completed cycle 5, which achieved criticality on January 15, 1985. Power escalation began on January 19, 1985 and reached 93% of full power (2772 MWt) on January 29, 1985. Normal operation continued until shutdown on June 9, 1985 after a transient. After 18 months, in which numerous operations and plant design enhancements were implemented, cycle 5 again achieved criticality on December 22, 1986. On March 20, 1987 full power was reached.

During cycle 5 operation, no operating anomalies occurred that would adversely affect fuel performance during cycle 6. The cycle 5 nominal design and cycle 6 desired licensed lengths are 400 and 415 effective full power days (EFPD), respectively. Cycle 6 was analyzed out to 405 EFPD and the validity of the Technical Specifications has been verified out to 415 EFPD. Therefore, the resulting Technical Specification Limiting Conditions for Operation accommodate cycle 6 operation through 415 EFPD. The APSRs were pulled at 325 EFPD to increase the lifetime of cycle 5. The APSR pull coupled with a power coastdown resulted in a cycle 5 length of approximately 395 EFPD. The cycle 6 design also includes an APSR pull and power coastdown.

The cycle 6 design minimizes the number of fuel assemblies that are cross core shuffled to reduce the potential for quadrant power tilt amplification. The cycle 6 shuffle pattern is discussed in section 3.

3. GENERAL DESCRIPTION

The Davis-Besse Unit 1 reactor core is described in detail in chapter 4 of the USAR[2] for the unit. The cycle 6 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. All FAs in batches 6, 7, and 8 have a constant nominal fuel loading of 468.25 kg of uranium. The one batch 1A assembly has a fuel loading of 472.24 kg of uranium. The fuel consists of dished-end cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy 4. The undensified nominal active fuel lengths, percent 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 6. One batch 1E assembly, 16 batch 4B assemblies, 8 batch 5A assemblies, and 40 batch 5B assemblies will be discharged at the end of cycle 5. The fuel assemblies in batches 6 and 7 will be shuffled to their cycle 6 locations, with batch 7 on the core periphery. One batch 1A assembly, presently in the spent fuel pool, with an initial uranium-235 enrichment of 1.98 wt % will be reinserted in cycle 6 as the center FA. Batches 6 and 7 have initial enrichments of 2.99 and 3.19 wt %, respectively. The feed batch, consisting of 64 batch 8 assemblies with uranium enrichment of 3.13 wt %, will be inserted in the core interior in a symmetric checkerboard pattern with the batch 6 FAs. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning of cycle (BOC) 6 and its initial enrichment.

Cycle 6 is operated in a feed-and-bleed mode. The core reactivity is controlled by 53 full-length Ag-In-Cd control rod assemblies (CRAs), 64 burnable poison rod assemblies (BPRAs), and soluble boron. In addition to the full-length control rods, eight Inconel-600 axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The cycle 6 locations of the control rods and the group designations are indicated in Figure 3-3. The core locations of the 61 control rods in cycles 5 and 6 are the same, however, the rod group designations differ between cycles. The changes in the rod group designations are to (1) decrease the worth of group 7 to be compatible with the gray APSR imbalance control, and (2) increase the worth of group 4 to facilitate control during physics testing. The cycle 6 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Davis-Besse Cycle 6 Core Loading Diagram

← N

X

A						7 K4	7 K2	7 M6	7 K14	7 K12						
B				7 M4	7 L3	7 N3	8 F	7 O8	8 F	7 N13	7 L13	7 M12				
C			7 M10	8 F	7 K6	8 F	6 A7	8 F	6 A9	8 F	7 K10	8 F	7 L5			
D		7 D11	8 F	6 R10	8 F	6 B4	8 F	6 B11	8 F	6 B12	8 F	6 L1	8 F	7 D5		
E		7 C10	7 F9	8 F	6 B5	8 F	6 B6	8 F	6 B10	8 F	6 E14	8 F	7 F7	7 C6		
F	7 D9	7 C12	8 F	6 D2	8 F	6 H1	8 F	6 A10	8 F	6 A8	8 F	6 D14	8 F	7 C4	7 D7	
G	7 B9	8 F	6 G1	8 F	6 F2	8 F	6 C3	7 M8	6 C13	8 F	6 F14	8 F	6 G15	8 F	7 B7	
W—H	7 L11	7 H13	8 F	6 E2	8 F	6 F1	7 H11	1A F10 cy 1	7 H5	6 L15	8 F	6 M14	8 F	7 H3	7 F5	—Y
K	7 P9	8 F	6 K1	8 F	6 L2	8 F	6 O3	7 E8	6 O13	8 F	6 L14	8 F	6 K15	8 F	7 P7	
L	7 N9	7 O12	8 F	6 N2	8 F	6 R8	8 F	6 R6	8 F	6 H15	8 F	6 N14	8 F	7 O4	7 N7	
M		7 O10	7 L9	8 F	6 M2	8 F	6 P6	8 F	6 P10	8 F	6 P11	8 F	7 L7	7 O6		
N		7 N11	8 F	6 F15	8 F	6 P4	8 F	6 P5	8 F	6 P12	8 F	6 A6	8 F	7 N5		
O			7 F11	8 F	7 G6	8 F	6 R7	8 F	6 R9	8 F	7 G10	8 F	7 E6			
P				7 E4	7 F3	7 D3	8 F	7 C8	8 F	7 D13	7 F13	7 E12				
R						7 G4	7 G2	7 E10	7 G14	7 G12						
								Z								
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

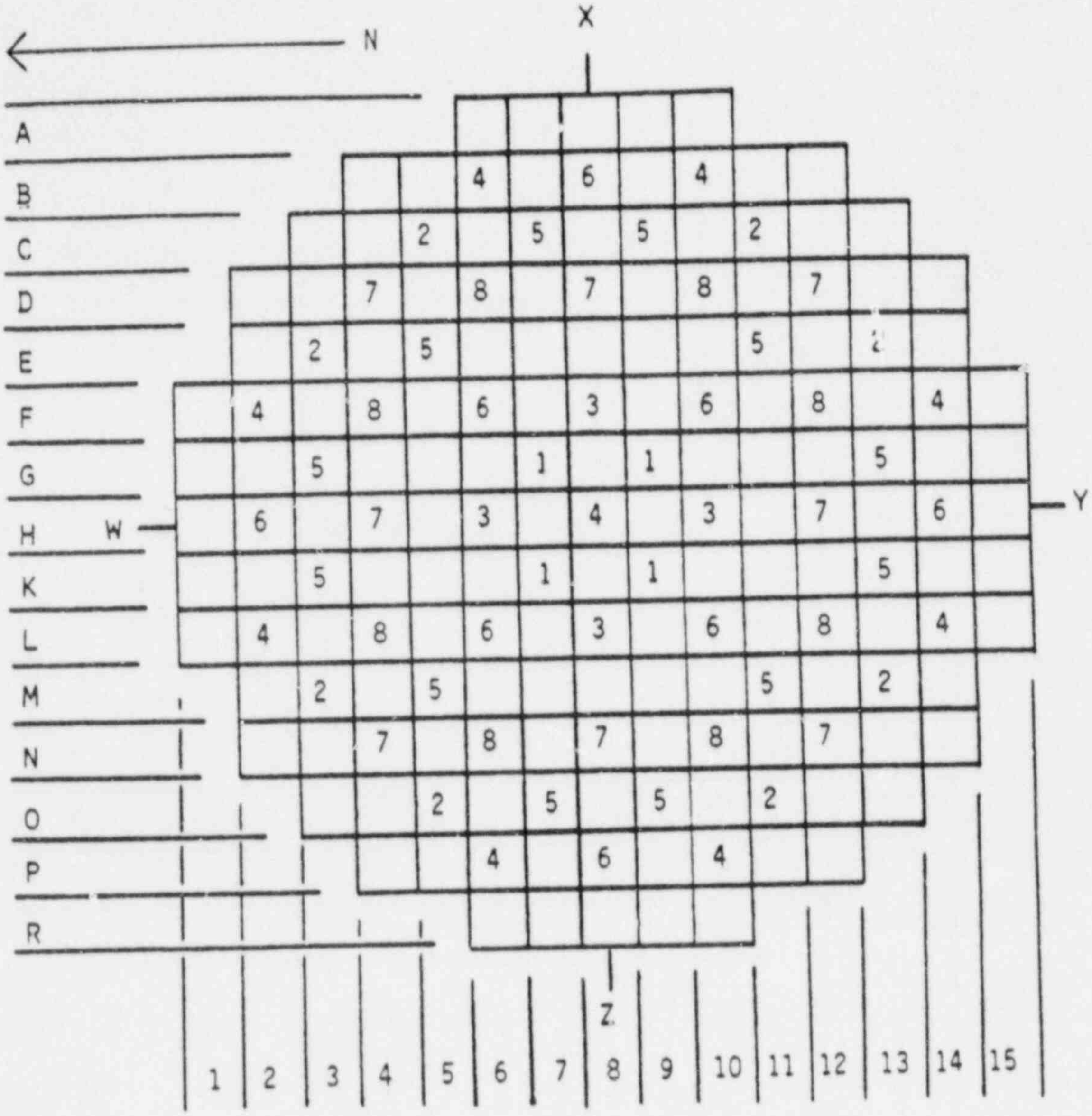
Batch
Cycle 5 Location (F = Fresh Assembly)
Cy 1 = reinserted from cycle 1

Figure 3-2. Enrichment and Burnup Distribution for Davis-Besse 1, Cycle 6

	8	9	10	11	12	13	14	15
H	1.98 12644	3.19 17676	2.99 16085	3.13 0	2.99 17336	3.13 0	3.19 17213	3.19 17475
K	3.19 17673	2.99 15952	3.13 0	2.99 20075	3.13 0	2.99 17166	3.13 0	3.19 15293
L	2.99 16085	3.13 0	2.99 17563	3.13 0	2.99 17841	3.13 0	3.19 13622	3.19 17148
M	3.13 0	2.99 20079	3.13 0	2.99 17337	3.13 0	3.19 17280	3.19 16697	
N	2.99 17336	3.13 0	2.99 17845	3.13 0	2.99 16090	3.13 0	3.19 16972	
O	3.13 0	2.99 17165	3.13 0	3.19 17280	3.13 0	3.19 17425		
P	3.19 17206	3.13 0	3.19 13632	3.19 16691	3.19 16951			
R	3.19 17475	3.19 15288	3.19 17135					

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

Figure 3-3. Control Rod Locations for Davis-Besse 1, Cycle 6



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 6 BPRA Enrichment and Distribution

	8	9	10	11	12	13	14	15
H				1.4		1.4		
K			1.4		1.4		0.2	
L		1.4		1.4		0.5		
M	1.4		1.4		1.1			
N		1.4		1.1		0.2		
O	1.4		0.5		0.2			
P		0.2						
R								

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

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The fuel assembly types (FAs) and pertinent fuel parameters for Davis-Besse 1, cycle 6 are listed in Table 4-1. The Batch 7 and 8 FAs are the Mk-B5 design, while the other batches are the Mk-B4 design. The Mk-B5 FAs are identical in concept to the Mk-B4 with only a change to the upper end fitting design which eliminates retainers for burnable poison rod assembly (BPRA) holddown.

Sixty-four BPRAs are used with the Batch 8 FAs. Also, eight gray APSRs and 53 CRAs are used in cycle 6.

4.2. Fuel Rod and Gray APSR Design

The fuel rod and gray APSR design and mechanical evaluation are discussed below.

4.2.1. Cladding Collapse

A. Fuel Rod

The power histories were reviewed for each fuel assembly in cycle 6. The most limiting power history for each of the four batches of fuel was found. The most limiting assembly is the one with the highest burnup. These four power histories were compared to a generic analysis or to a previous creep collapse analysis. The generic creep ovalization analysis is based on reference 3 and is applicable to the batch 1A, 6, 7, and 8 designs.

The creep collapse analyses predict collapse times longer than 35000 EFPH. The longest incore exposure time for cycle 6 is 25800 EFPH for batch 6 (Table 4-1).

B. Gray APSR

The gray APSRs used in cycle 6 are designed for improved creep life. Cladding thickness and rod ovality, which are the primary factors controlling the time until creep collapse, are improved to extend the life of the gray APSR. The minimum design cladding thickness of the Mark B black APSR is 18 mils, while that of the Mk-B gray APSR is 24 mils. Additionally, the gap width between the end plug and the Inconel-600 absorber material is reduced. Finally, the gap area ovality is controlled to tighter tolerances. The gray APSR is shown in Figure 4-1.

4.2.2. Cladding Stress

A. Fuel Rod

A conservative fuel rod stress analysis envelopes the Davis-Besse Unit 1 cycle 6 stress values. The methods used for the analysis of cycle 6 have been used in the previous cycles.

B. Gray APSR

The gray APSR design has demonstrated the ability to meet specified mechanical design requirements. The APSR cladding stress analysis includes pressure, temperature and ovality effects. The gray APSR has sufficient cladding and weld stress margins.

4.2.3. Cladding Strain

A. Fuel Rod

The fuel design criteria specify a limit of 1.0% on cladding plastic tensile circumferential strain. The pellet design ensures that plastic cladding strain is less than 1% at design local pellet burnup and heat generation rates. The design values are higher than the worst-case values the Davis-Besse Unit 1, cycle 6 fuel is expected to see. The strain analysis is based on the upper tolerance values for the fuel pellet diameter and density, and the lower tolerance for the cladding inside diameter (ID).

B. Gray APSR

The gray APSR strain analysis includes thermal and irradiation swelling effects. The results of this analysis show that no cladding strain is induced due to thermal expansion or irradiation swelling of the Inconel absorber.

4.3. Thermal Design

All fuel in the cycle 6 core is thermally similar. The fresh batch 8 fuel inserted for cycle 6 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The thermal analyses for all fuel were performed with the TACO2 code[4]. Nominal undensified input parameters used in this methodology are provided in Table 4-1. Densification effects are accounted for in the TACO2 code densification model.

Linear heat rate (LHR) to fuel melt capability for all fuel was determined with the TACO2 fuel pin performance code. The analyses performed for cycle 6 demonstrate that 20.5 kw/ft is a conservative limit to preclude centerline fuel melt (CFM) for all fuel batches.

The maximum fuel rod burnup at EOC 6 is predicted to be less than 38300 MWd/mtU. Fuel rod internal pressure has been evaluated with TACO2 for the highest burnup fuel rod and is predicted to be less than the reactor coolant system pressure of 2200 psia at the core outlet.

4.4. Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for batch 8 FAs is identical to that of present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. As of October 31, 1987, the following experience has been accumulated for eight B&W 177 fuel assembly plants using the Mark B fuel assembly:

Reactor	Current Cycle	Max FA burnup, MWd/mtU[(a)] Incore	Discharged	Cumulative net electric output, MWh[(b)]
Oconee 1	10	45,908	50,598	66,183,044
Oconee 2	9	40,580	41,592	60.968,626
Oconee 3	10	33,290	39,701	60,843,663
Three Mile Island	6	26,090	33,444	29,469,976
Arkansas Nuclear One, Unit 1	8	51,540	47,560	51,626,035
Rancho Seco	7	26,242	38,268	39,045,954

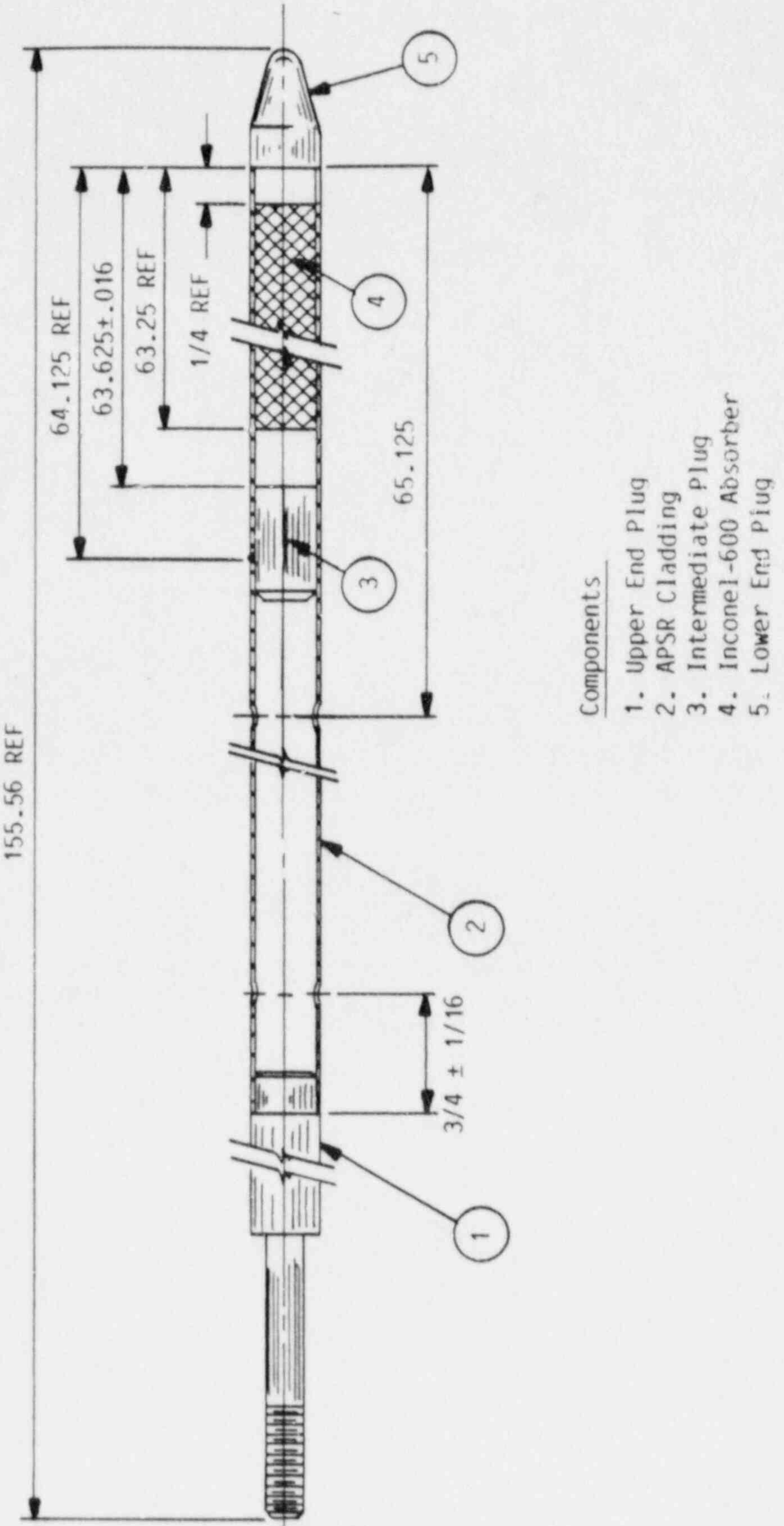
Reactor	Current Cycle	Max FA burnup, MWd/mtU[a] Incore	Discharged	Cumulative net electric output, MWh[b]
Crystal River 3	6	35,350	31,420	38,512,798
Davis-Besse	5	36,960	32,790	25,236,663

[a] As of October 31, 1987.
[b] As of December 31, 1986.

Table 4-1. Fuel Design Parameters

	Batch			
	1A	6	7	8
FA type	Mk-B4	Mk-B4	Mk-B5	Mk-B5
Number of assemblies	1	48	64	64
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377
Flexible spacer type	Spring	Spring	Spring	Spring
Rigid spacer type	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Undensified active fuel length, in.	143.5	143.20	143.20	143.20
Fuel pellet (mean) dia., in.	0.3675	0.3686	0.3686	0.3686
Fuel pellet initial density, % TD mean	96	95	95	95
Initial fuel enrichment, wt % ^{235}U	1.98	2.99	3.19	3.13
Average Burnup BOC, MWd/mtU	12600	17500	16500	0
Exposure Time EOC, (EFPH)	18700	25800	19300	9700
Cladding Collapse Time, (EFPH)	>35000	>35000	>35000	>35000
Nominal Linear Heat Rate at 2772 MWt, kW/ft	6.13	6.14	6.14	6.14
Minimum Linear Heat Rate to Melt, kW/ft	22.0	20.5	20.5	20.5

Figure 4-1. Gray Axial Power Shaping Rod



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 5 and 6 designs. The values for cycle 5 were generated using PDQ07[5] and the values for cycle 6 were calculated with the NOODLE code[6]. The calculational differences resulting from the upgrading of the modeling are negligible. Both PDQ07 and NOODLE have been verified to produce results within the bounds of the same measurement uncertainties (see section 5.2 for a further discussion). Differences in core physics parameters are to be expected between the cycles due to the changes noted below in section 5.2, such as the longer cycle 6 length. Figure 5-1 illustrates a representative relative power distribution for the BOC 6 at full power with equilibrium xenon, all rods out and gray APSRs inserted.

The ejected rod worths in Table 5-1 are the maximum calculated values. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in section 8. The adequacy of the shutdown margin with cycle 6 rod worths is shown in Table 5-2. The following conservatisms were applied for the shutdown margin calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. A maximum flux redistribution penalty.
4. A maximum power deficit with minimum boron.

The maximum flux redistribution was taken into account to ensure that the effects of operational maneuvering transients were included in the shutdown margin analysis.

5.2. Changes in Nuclear Design

Core design changes for cycle 6 include, (1) the increase in cycle lifetime to 405 EFPD, (2) an increase in the BPRA concentrations, (3) the variation in the loading pattern between cycles 5 and 6, (4) the second transition cycle to the LBP, low leakage fuel cycle design, (5) the revised control rod groupings, (6) the removal of the regenerative neutron sources which will result in a sourceless startup, and (7) the replacement of the Silver-Indium-Cadmium APSRs used in all previous cycles with gray APSRs, which have a longer absorber section but use a weaker absorber (Inconel-600). These design differences can explain the differences in the physics parameters between cycles 5 and 6 as shown in Table 5-1. Calculations with the standard three-dimensional model verified that changing to the gray APSR design provides adequate axial power distribution control. As stated in section 5.1, the NOODLE code was used to calculate the physics parameters for cycle 6. The NOODLE modeling of the two-group homogenized fuel assembly is the same as that used in PDQ07. However, the analytical expression NOODLE uses for the spatial flux solution provides more accurate results than the finite difference expression used in PDQ07 when there are few flux solution points per assembly. Reference 6 illustrates the calculational accuracy attainable with NOODLE in comparison to measured results for various physics parameters. PDQ07 results are compared to measured data in reference 7. These comparisons show NOODLE to be as accurate as PDQ07.

Other calculational models and the methods used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle. No significant operational or procedural changes exist with regard to axial or radial power shape, xenon, or tilt control. The stability and control of the core with APSRs withdrawn has been analyzed. The calculated stability index without APSRs is $-0.023\ h^{-1}$, which demonstrates the axial stability of the core.

Table 5-1. Davis-Besse Unit 1, Cycle 6 Physics Parameters

	Cycle 5	Cycle 6
Design cycle length, EFPD	390	405
Cycle burnup, MWd/mtU	13,043	13,545
Average core burnup - EOC, MWd/mtU	22,797	24,335
Initial core loading, mtU	82.9	82.9
Critical boron$^{(a)}$ - BOC, No Xe, ppm		
HZP	1,485	1,451
HFP	1,255	1,285
Critical boron$^{(a)}$ - EOC, Eq. Xe, ppm		
HZP	324	204
HFP	10$^{(b)}$	10$^{(b)}$
Control rod worths - HFP, BOC, % Δk/k		
Group 6	1.13	1.18
Group 7	1.42	1.05
Group 8	0.38	0.21
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.46	1.14
Group 8	NA	NA
Max ejected rod worth - HZP, % Δk/k (location)		
BOC, Groups 5-8 inserted	0.60 (N-12)	0.36 (L-10)
EOC, Groups 5-7 inserted	0.55 (N-12)	0.40 (L-10)
Max stuck rod worth - HZP, % Δk/k (location)		
BOC	0.80 (N-12)	0.66 (M-11)
EOC	0.76 (M-11)	0.79 (M-11)
Power deficit - HZP to HFP, Eq. Xe, % Δk/k		
BOC (4 EFPD)	-1.76	-1.71
EOC	-2.48	-2.51
Doppler coeff - HFP, 10^{-3} % Δk/k/°F		
BOC, No Xe, 1285 ppm,$^{(c)}$ Group 8 inserted	-1.50	-1.55
EOC, Eq. Xe, 0 ppm,$^{(d)}$ Group 8 withdrawn	-1.76	-1.84
Moderator coeff - HFP, 10^{-2} % Δk/k/°F		
BOC, No Xe, 1285 ppm,$^{(c)}$	-0.81	-0.59
EOC, Eq. Xe, 0 ppm$^{(d)}$	-2.86	-2.84
Boron worth - HFP, ppm/% Δk/k		
BOC (1285 ppm)$^{(c)}$	123	124
EOC (0 ppm)$^{(d)}$	106	107
Xenon worth - HFP, % Δk/k		
BOC (4 EFPD)	2.63	2.63
EOC (equilibrium)	2.73	2.78
Effective delayed neutron fraction - HFP		
BOC	0.00631	0.00626
EOC	0.00524	0.00518

Table 5-1 (Continued)

[a] Control rod group 8 is inserted at BOC and withdrawn at EOC.
[b] Power coastdown to EOC at 10 ppm.
[c] Cycle 6 values were calculated at 1285 ppm; cycle 5 values were calculated at 1255 ppm.
[d] Cycle 6 values were calculated at 0 ppm; cycle 5 values were calculated at 10 ppm.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 6

	BOC, % Δk/k	EOC, % Δk/k 325 EFPD Group 8 in	EOC, % Δk/k 405 EFPD Group 8 out
Available Rod Worth			
Total rod worth, HZP	7.38	7.55	7.59
Worth reduction due to burnup of poison material	-0.42	-0.42	-0.42
Maximum stuck rod, HZP	-0.66	-0.74	-0.79
Net worth	6.30	6.39	6.38
Less 10% uncertainty	-0.63	-0.64	-0.64
Total available worth	5.67	5.75	5.74
Required Rod Worth			
Power deficit, HFP to HZP	1.71	2.21	2.51
Max allowable inserted rod worth	0.26	0.38	0.40
Flux redistribution	0.27	0.60	0.63
Total required worth	2.24	3.19	3.54
Shutdown Margin			
Total available minus total required rod worth	3.43	2.56	2.20

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

Figure 5-1. BOC (4 EFPD), Cycle 6 Two-Dimensional Relative Power Distribution - Full Power, Equilibrium Xenon, All Rods Out, APSRs Inserted[a]

	8	9	10	11	12	13	14	15
H	.887	1.111	1.140	1.263	1.158	1.247	1.035	.547
K	1.111	1.101	1.258	1.124	1.274	1.153	1.141	.545
L	1.138	1.256	1.160	1.282	8 1.141	1.286	.913	.394
M	1.260	1.121	1.275	1.172	1.272	1.046	.628	
N	1.155	1.270	8 1.136	1.269	1.054	.933	.405	
O	1.244	1.150	1.282	1.044	.933	.497		
P	1.033	1.139	.910	.627	.405			
R	.546	.544	.393					

x	Inserted Rod Group Number
X.xx	Relative Power Density

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

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 6 operation used the methods and models described in references 2 and 8. The cycle 6 design analysis is the first application of crossflow methodology for the Davis-Besse station. The use of crossflow codes which can predict the flow redistribution effects in an open lattice reactor core, provide significant departure from nucleate boiling ratio (DNBR) margin improvements relative to the traditional closed-channel codes.

The LYNX1[9], LYNX2[10] and LYNXT[11] crossflow codes were used in the cycle 6 design analyses. The LYNX1 and LYNX2 codes were used primarily to benchmark the LYNXT models.

Cycle 5 and 6 thermal-hydraulic design conditions are listed in Table 6-1. The reactor coolant flow, bypass flow and design axial flux shape were revised for the cycle 6 analysis. The cycle 6 reactor coolant flow requirement was reduced approximately 2 percent relative to the cycle 5 original design value. This reduction provides additional operating flexibility. The original design cycle 5 bypass flow of 10.7 percent was determined with no BPRAs. The cycle 6 bypass flow of 8.6 percent is based on 52 BPRAs and produces a conservative bypass flow with respect to the actual core configuration because the number of BPRAs actually in the core is 64. The design axial peak has been increased from 1.50 to 1.65 to provide additional margin for maneuvering analyses and resulting control rod insertion limits. This change was accommodated by the introduction of the crossflow methodology.

Crossflow methodology was used to reanalyze the pressure-temperature limits, the one pump coastdown and corresponding flux/flow limit and the locked rotor transient. All analyses showed a significant increase in DNBR margin relative to the cycle 5 closed-channel analysis. The RPS pressure-temperature trip setpoint has been recalculated based on pressure-temperature limits computed with LYNXT. The flux/flow limit determined with LYNXT indicates that the flux/flow setpoint may be raised from 1.07 to 1.08. Analysis of the locked rotor transient with

LYNXT shows that the minimum DNBR during the transient is greater than the design limit of 1.30[12].

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 5	Cycle 6
Reactor Coolant System		
Rated Thermal Power Level, MWt	2772	2772
Nominal Core Exit Pressure, psia	2200	2200
Minimum Core Exit Pressure, psia	2135	2135
Reactor Coolant Flow, gpm	387,200	380,000
Nominal Vessel Inlet Coolant Temperature,(a) °F	557.7	557.4
Nominal Vessel Outlet Coolant Temperature,(a) °F	606.3	606.6
Bypass Flow, %design	10.7	8.6
Power Distribution		
Design Radial x Local Peaking Factor	1.71	1.71
Design Axial Flux Shape	1.50 cosine with tails	1.65 chopped cosine
Hot Channel Factors		
Enthalpy Rise	1.011	1.011
Heat Flux	1.014	1.014
Flow Area	0.98	0.98
Cold Fuel Stack Height,(b) inches	143.2	143.2
Average Heat Flux, 10^5 Btu/hr-ft^2	1.89(c)	1.86
Maximum Heat Flux, 10^5 Btu/hr-ft^2	4.85(c)	5.25
DNBR Parameters		
Critical Heat Flux Correlation	BAW-2	BAW-2
Minimum DNBR, (at 102% FP)	--	2.07
Minimum DNBR, (at 112 %FP)	1.79	1.79

(a) 100 %FP

(b) Smallest value of all assemblies

(c) Based on densified stack height

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each USAR accident analysis has been examined with respect to changes in the cycle 6 parameters to determine the effects of the cycle 6 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the USAR accident results have been evaluated and are reported in reference 13.

The radiological dose consequences of the FSAR Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source term for DB-1 cycle 6. The dose calculations were performed consistent with the assumptions described in the DB-1 FSAR but used the more conservative source terms (which bound future reload cycles). The results of the dose evaluations showed that offsite radiological doses for each accident were below the respective acceptance criteria values in the current NRC Standard Review Plan (NUREG-0800).

7.2. Accident Evaluation

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

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

A generic loss-of-coolant accident (LOCA) analysis for B&W 177-FA raised-loop nuclear steam systems (NSSs) has been performed using the Final Acceptance Criteria ECCS Evaluation Model.[14] The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used in the LOCA limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in reference 14 provide conservative results for the operation of Davis-Besse Unit 1, cycle 6 fuel. A tabulation showing the bounding values for allowable LOCA peak LHRs for Davis-Besse Unit 1, cycle 6 fuel is provided in Table 7-2.

It is concluded by the examination of cycle 6 core thermal, thermal-hydraulic, and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the ability to safely operate the Davis-Besse Unit 1 plant during cycle 6. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 6 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 6 are bounded by the USAR and/or the fuel densification report.

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

Parameter	FSAR and densif'n report value	Cycle 6 value
BOL[a] Doppler coeff, 10^{-3}, % Δk/k/°F	-1.28	-1.55
EOL(b) Doppler coeff, 10^{-3}, % Δk/k/°F	-1.45[c]	-1.84
BOL moderator coeff, 10^{-2}, % Δk/k/°F	+0.13	-0.59
EOL moderator coeff, 10^{-2}, % Δk/k/°F	-3.0	-2.84
All rod bank worth (HZP), % Δk/k	10.0	7.38
Boron reactivity worth (HFP), ppm/1% Δk/k	100	124
Max ejected rod worth (HFP), % Δk/k	0.65	0.29
Max dropped rod worth (HFP), % Δk/k	0.65	0.20
Initial boron conc (HFP), ppm	1407	1285

(a)BOL denotes beginning of life.

(b)EOL denotes end of life

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

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

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

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 6 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 8-13 are revisions to the previous cycle Technical Specifications. The setpoints shown in Table 8-1 may be changed by pending license amendments, and such changes shall have no effect on the information shown in this report. Based on these Technical Specifications the final acceptance criteria ECCS limits will not be exceeded and the thermal design criteria will not be violated.

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

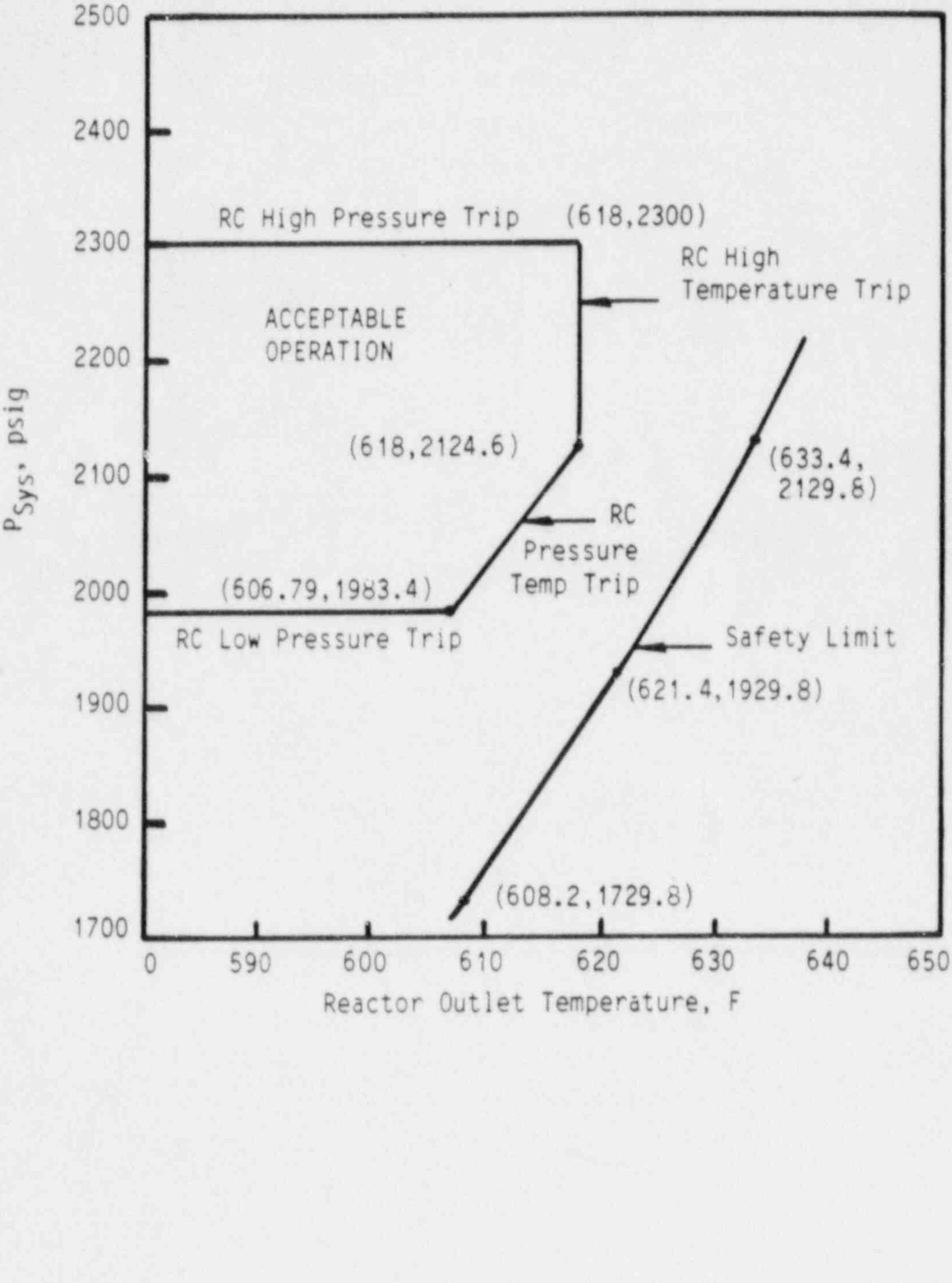
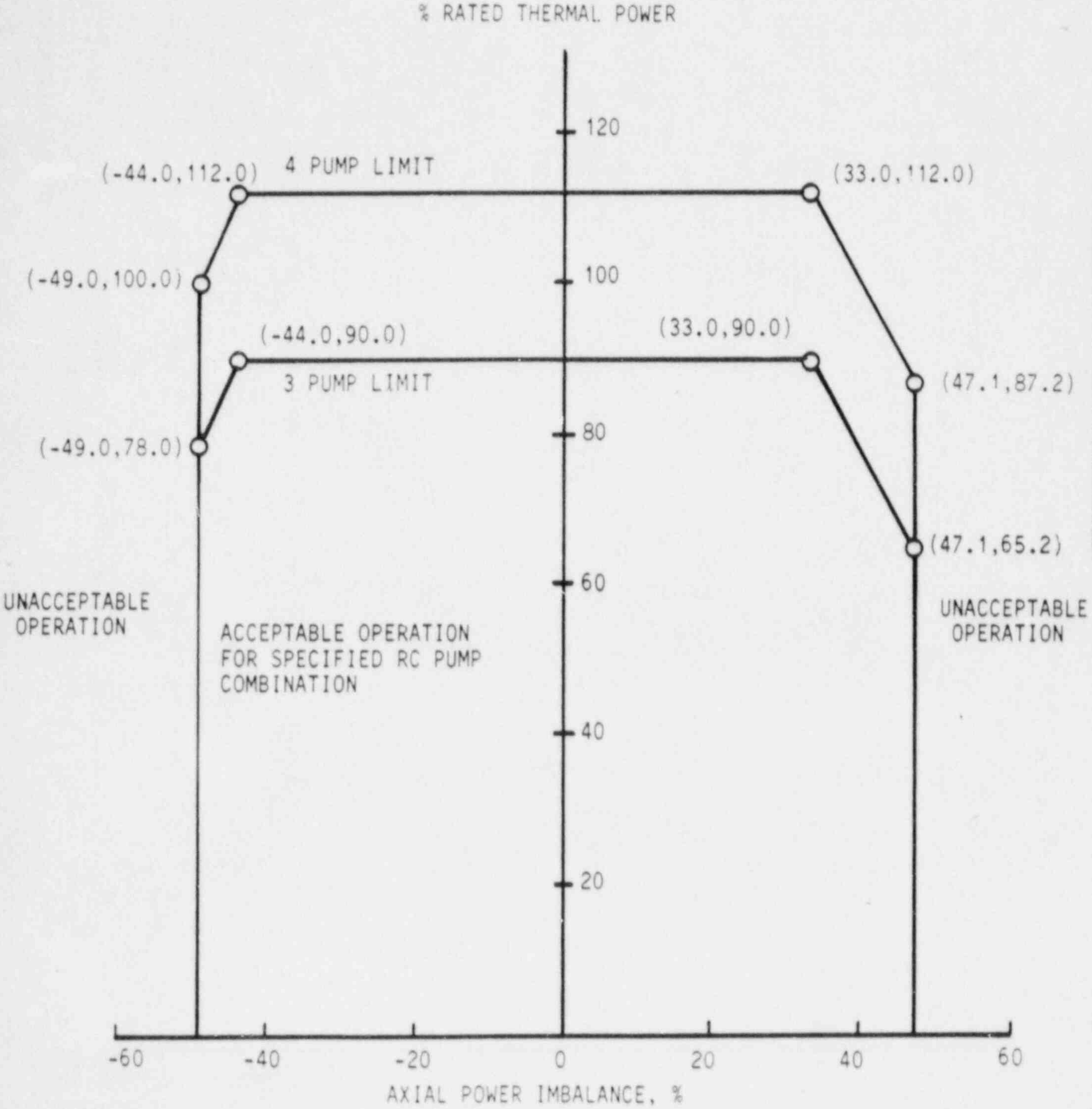


Figure 8-2. Reactor Core Safety Limit
(Tech.Spec. Figure 2.1-2)



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

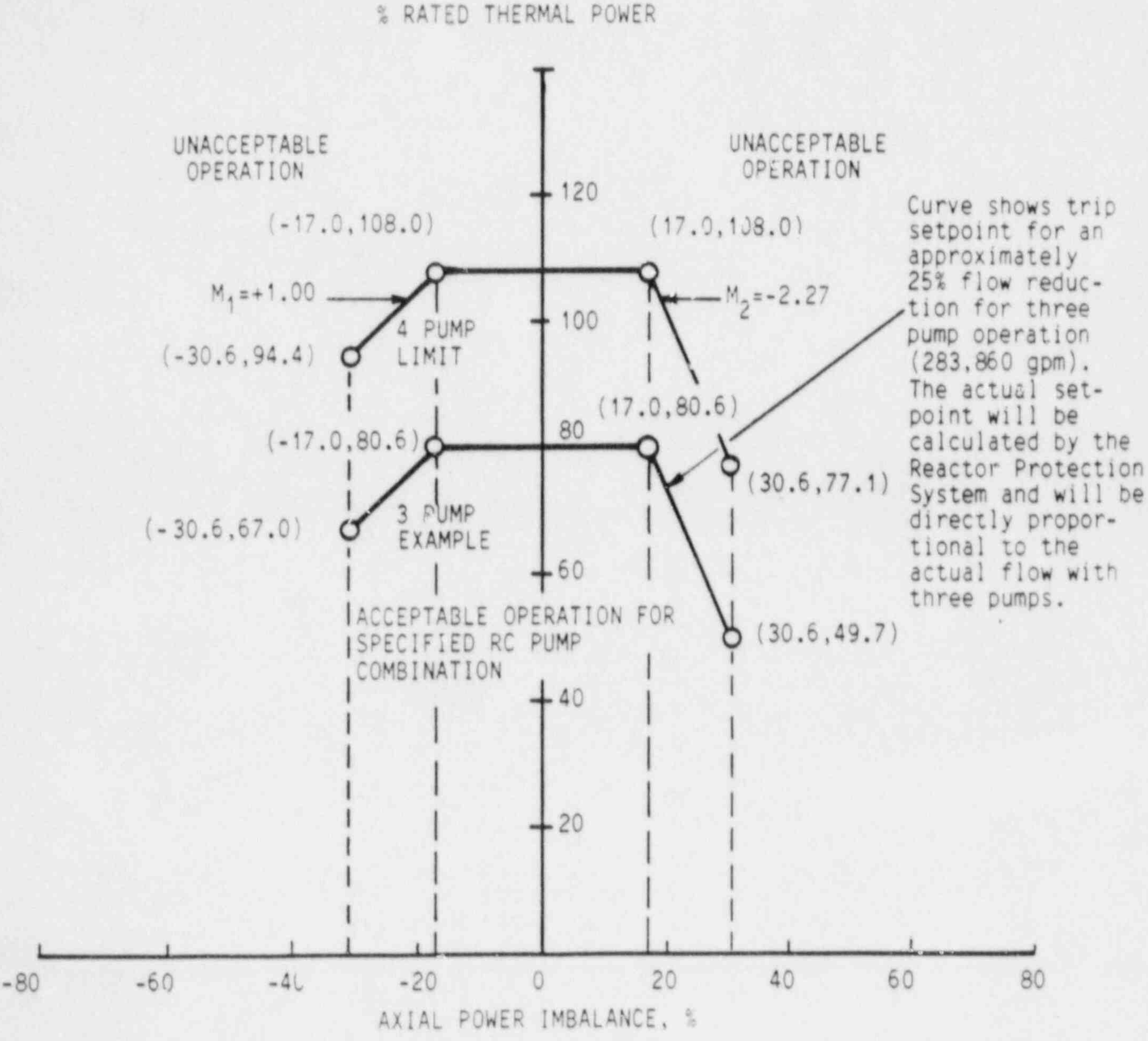
Babcock & Wilcox
a McDermott company

Table 8-1 Reactor Protection System Instrumentation Trip Setpoints

(Tech. Spec. Table 2.2-1)

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable.	Not applicable.
2. High flux	≤104.94% of RATED THERMAL POWER with four pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤80.6% of RATED THERMAL POWER with three pumps operating	≤80.6% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/flow(1)	Four pump trip setpoint not to exceed the limit line of Figure 2.2-1. For three pump operation, see Figure 2.2-1.	Four pump allowable value not to exceed the limit line of Figure 2.2-1#. For three pump operation, see Figure 2.2-1.
5. RC low pressure(1)	≥1983.4 psig	≥1983.4 psig* ≥1983.4 psig**
6. RC high pressure	≤2300 psig	≤2300.0 psig* ≤2300.0 psig**
7. RC pressure-temperature(1)	≥(12.60 T_{out} °F - 5662.2) psig	≥(12.60 T_{out} °F - 5662.2) psig#
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop	≤55.1% of RATED THERMAL POWER with one pump operating in each loop#
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	≤0.0 of RATED THERMAL POWER with no pumps operating or only one pump oprating	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	≤4 psig	≤4 psig#

Figure 8-3. Trip Setpoint for Flux -- Δ Flux/Flow
(Tech. Spec. Figure 2.2-1)



Curve shows trip setpoint for an approximately 25% flow reduction for three pump operation (283,860 gpm). The actual setpoint will be calculated by the Reactor Protection System and will be directly proportional to the actual flow with three pumps.

2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the B&W-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNB of 1.30 is predicted for the maximum possible thermal power of 112% when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM.) This curve is based on the following hot channel factors with potential fuel densification and fuel rod bowing effects:

$$F_Q = 2.83; \quad F^N_{\Delta H} = 1.71; \quad F^N_Z = 1.65$$

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 withdrawal, and form the core DNBR design basis.

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 = 2.83 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 22.0 KW/ft for batch 1F and 20.5 KW/ft for batches 6, 7, and 8.

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 the two curves of Figure 2.1-2 correspond to the analyzed 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 Figures 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.

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than +22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistant with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

RC High Temperature

The RC high temperature trip $\leq 618^{\circ}F$ 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.

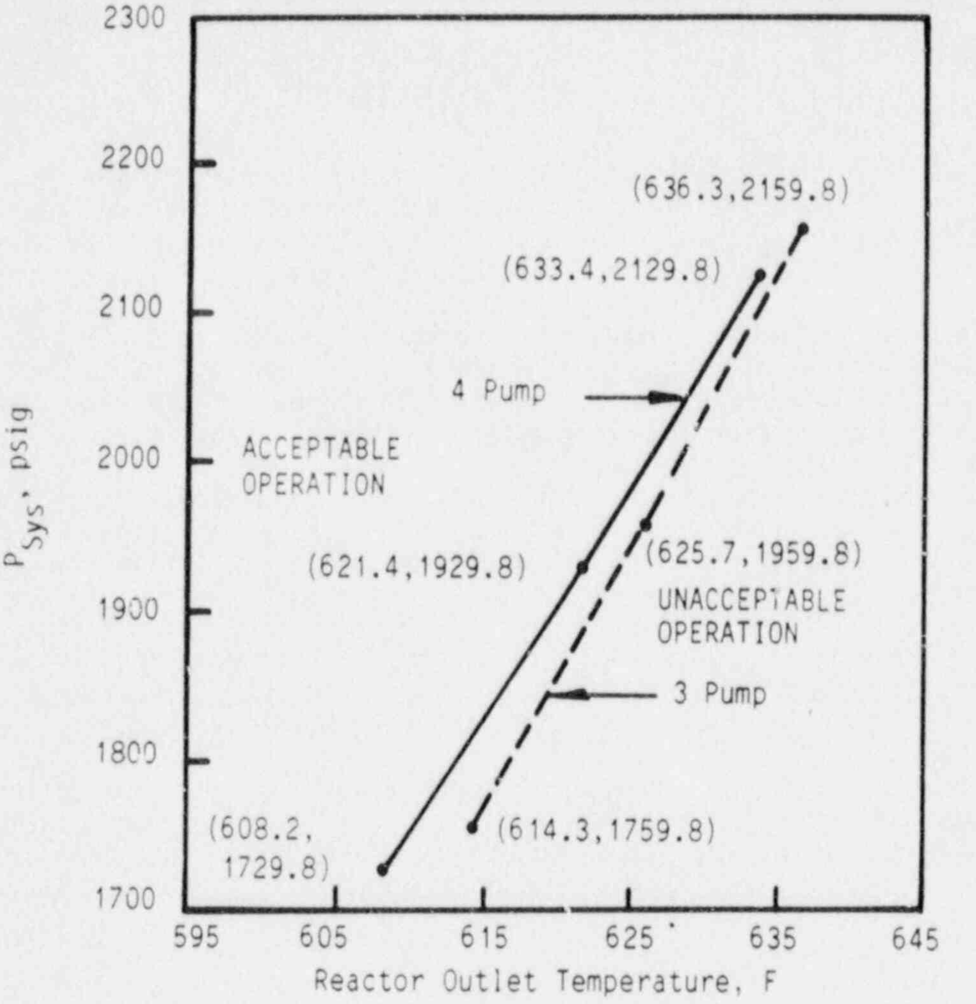
The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection 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 108.0% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 92.59% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.68% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 69.44% of full flow rate and power is 75%. Note that the value of 80.6% in Figure 2.2-1 was truncated from the calculated value of 80.68%.

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. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

Figure 8-4. Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR (Tech. Spec. Bases Figure 2.1)



Pumps	Flow, gpm	Power	Required Measured Flow to ensure Compliance, gpm
4	380,000	112%	389,500
3	283,860	90.5%	290,957

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid addition system with:

1. A minimum available borated water volume of 600 gallons,
2. Between 7875 and 13,125 ppm of boron, and
3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A minimum available borated water volume of 3,000 gallons,
2. A minimum boron concentration of 1800 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATION or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.8 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the available borated water volume of the source, and

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE:

a. The boric acid addition system with:

1. A minimum available borated water volume in accordance with Figure 3.1-1,
2. Between 7875 and 13,125 ppm of boron, and
3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. An available borated water volume of between 482,778 and 550,000 gallons,
2. Between 1800 and 2200 ppm of boron, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: a. With the boric acid addition system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the boric acid addition system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

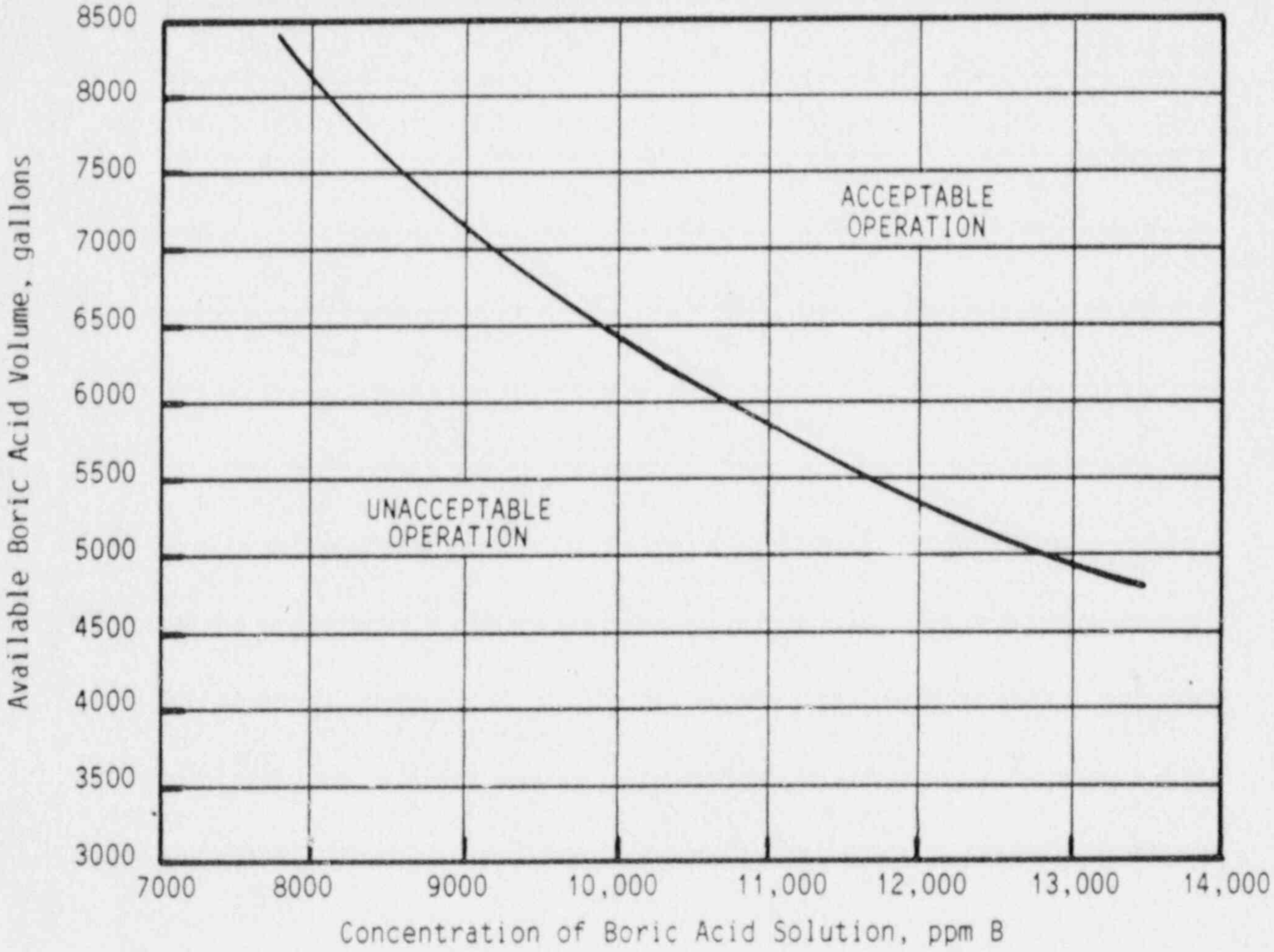
4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,
2. Verifying the available borated water volume of each water source, and
3. Verifying the boric acid addition system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when the outside air temperature is < 35°F.

Figure 8-5. Minimum Boric Acid Tank Available Volume as Function of Stored Boric Acid Concentration-Davis-Besse 1 (Tech. Spec. Figure 3.1-1)



REACTIVITY CONTROL SYSTEMS

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, and -2b, and 3.1-3a, and -3b. A rod group overlap of 25 ±5% shall be maintained between sequential withdrawn groups 5, 6, and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION

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

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

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

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

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

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*See Special Test Exception 3.10.1 and 3.10.2.
#With $k_{eff} \geq 1.0$.

Figure 8-6. Regulating Group Position Limits, 0 to 325±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 6 (Tech. Spec. Figure 3.1-2a)

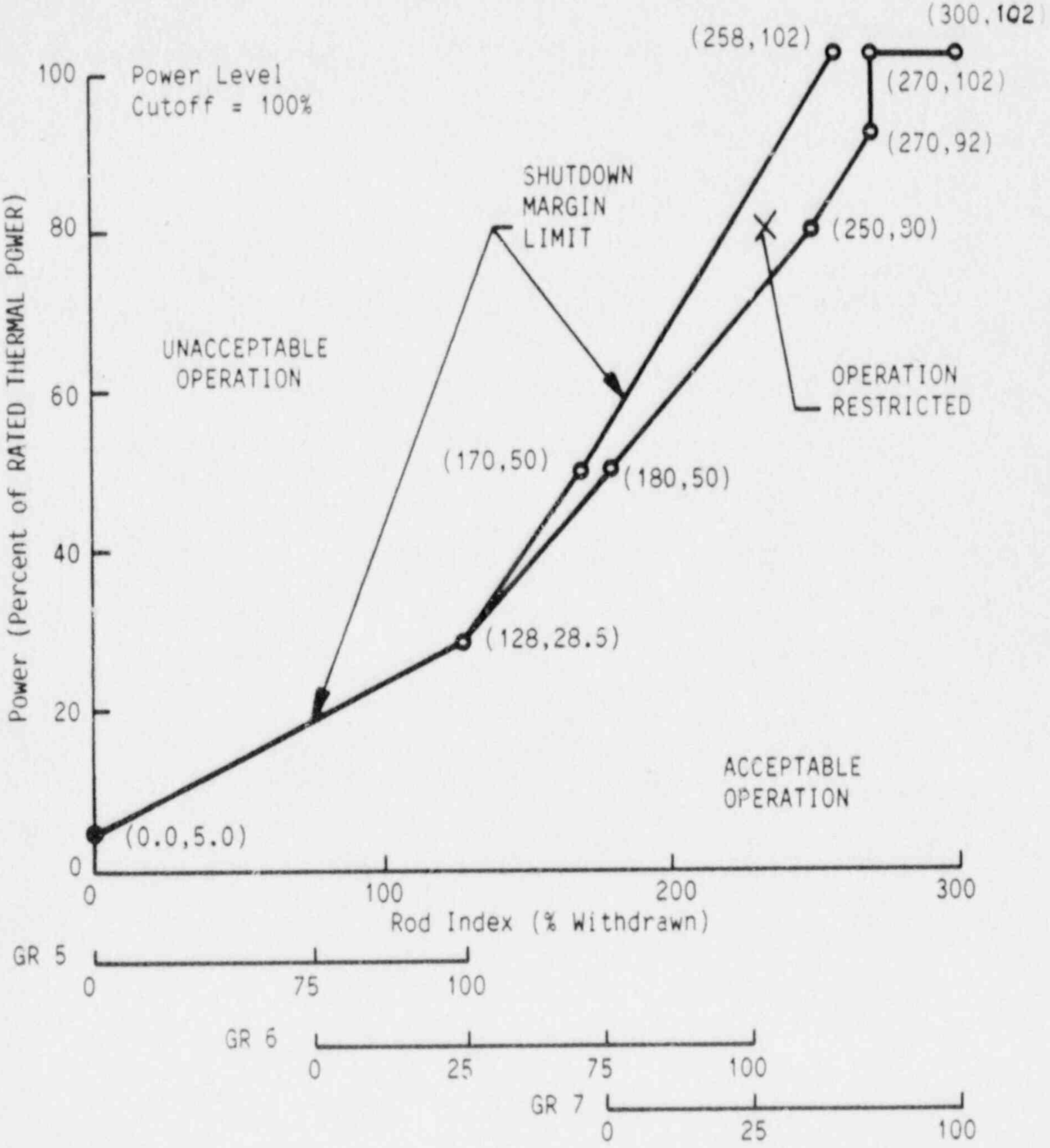


Figure 8-7. Regulating Group Position Limits After 325±10 EFPD, Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6, (Tech. Spec. Figure 3.1-2b)

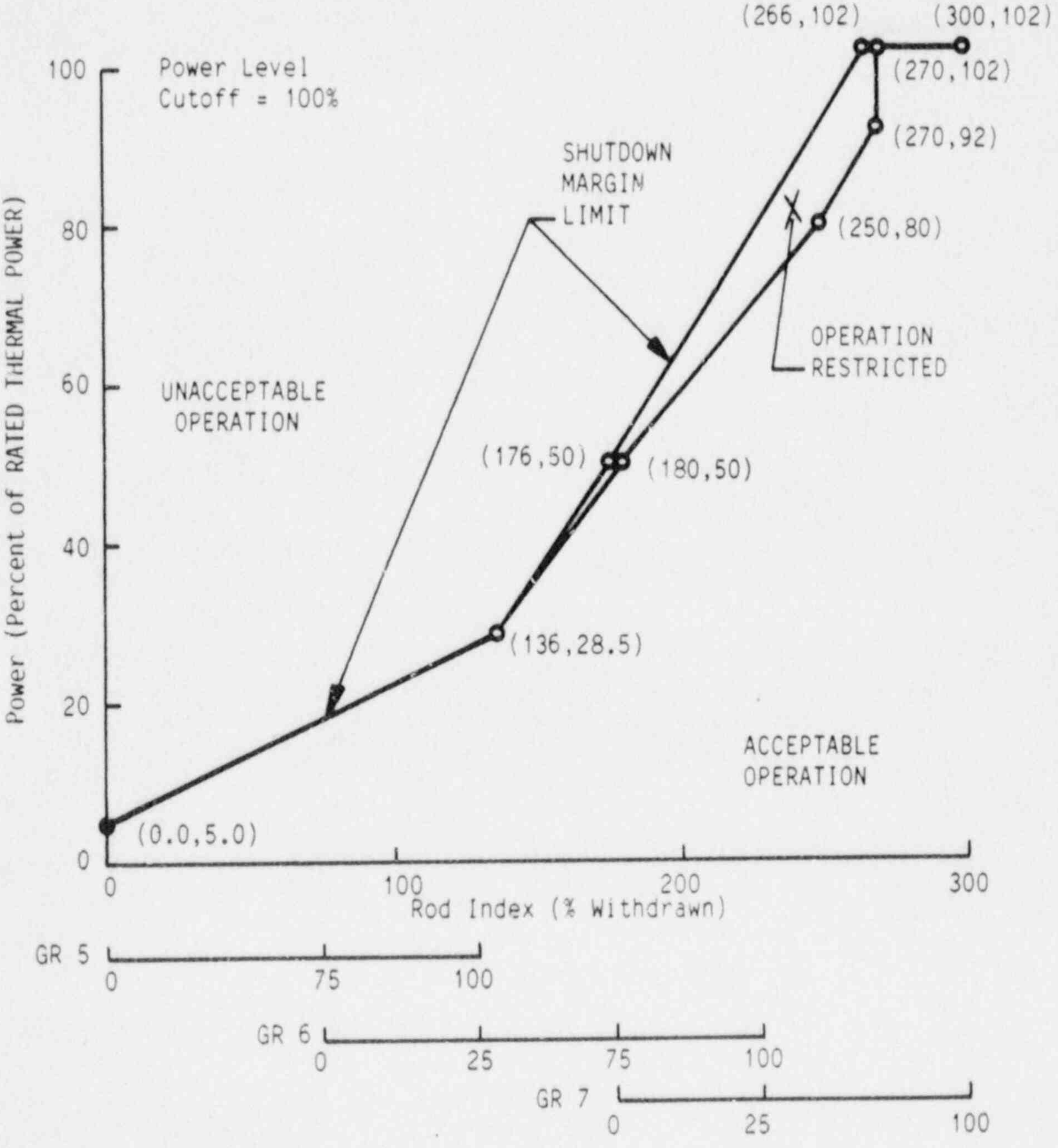


Figure 8-8. Regulating Group Position Limits, 0 to 325±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 6 (Tech. Spec. Figure 3.1-3a)

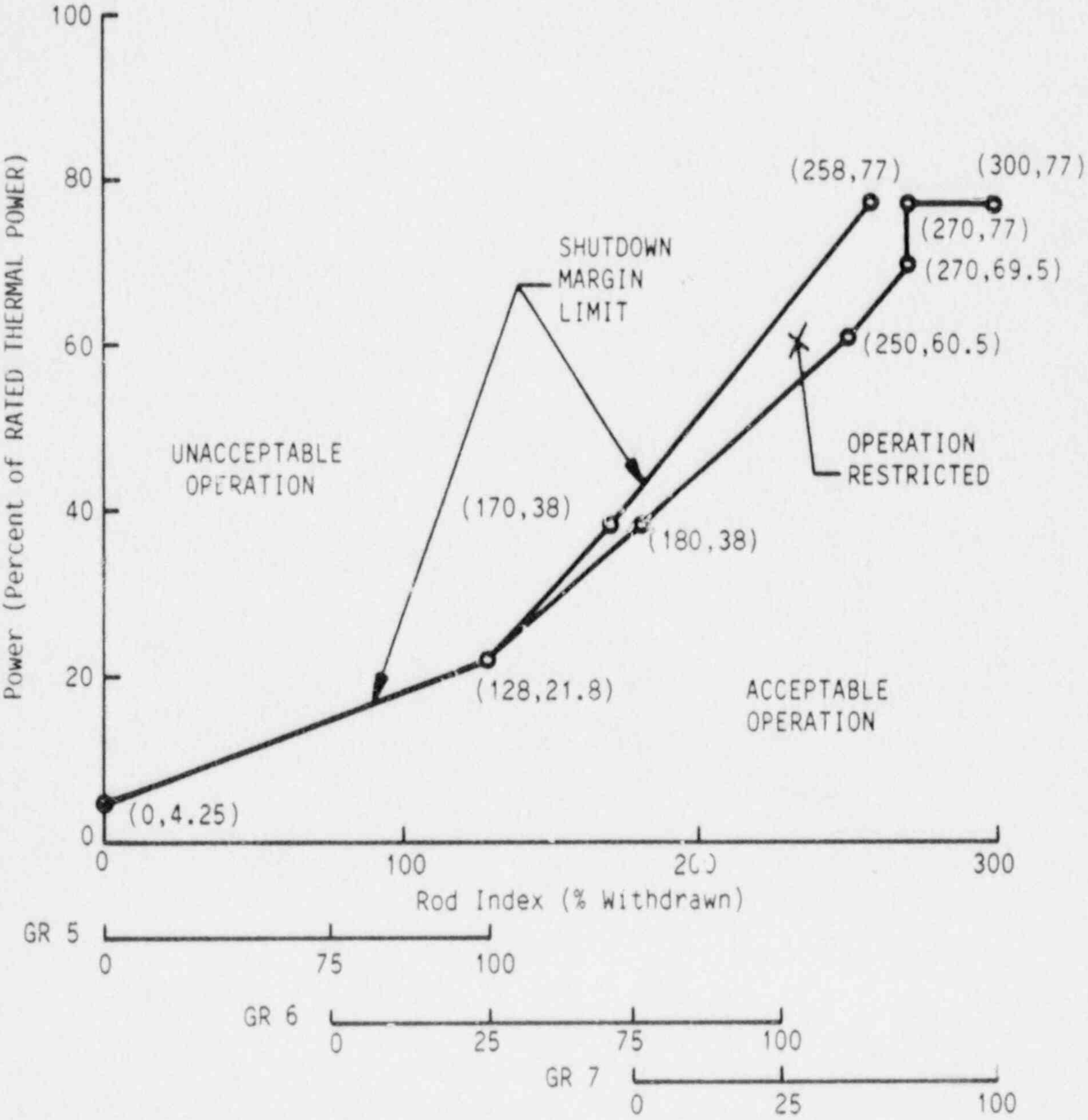


Figure 8-9. Regulating Group Position Limits After 325±10 EFPD, Three RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6 (Tech. Spec. Figure 3.1-3b)

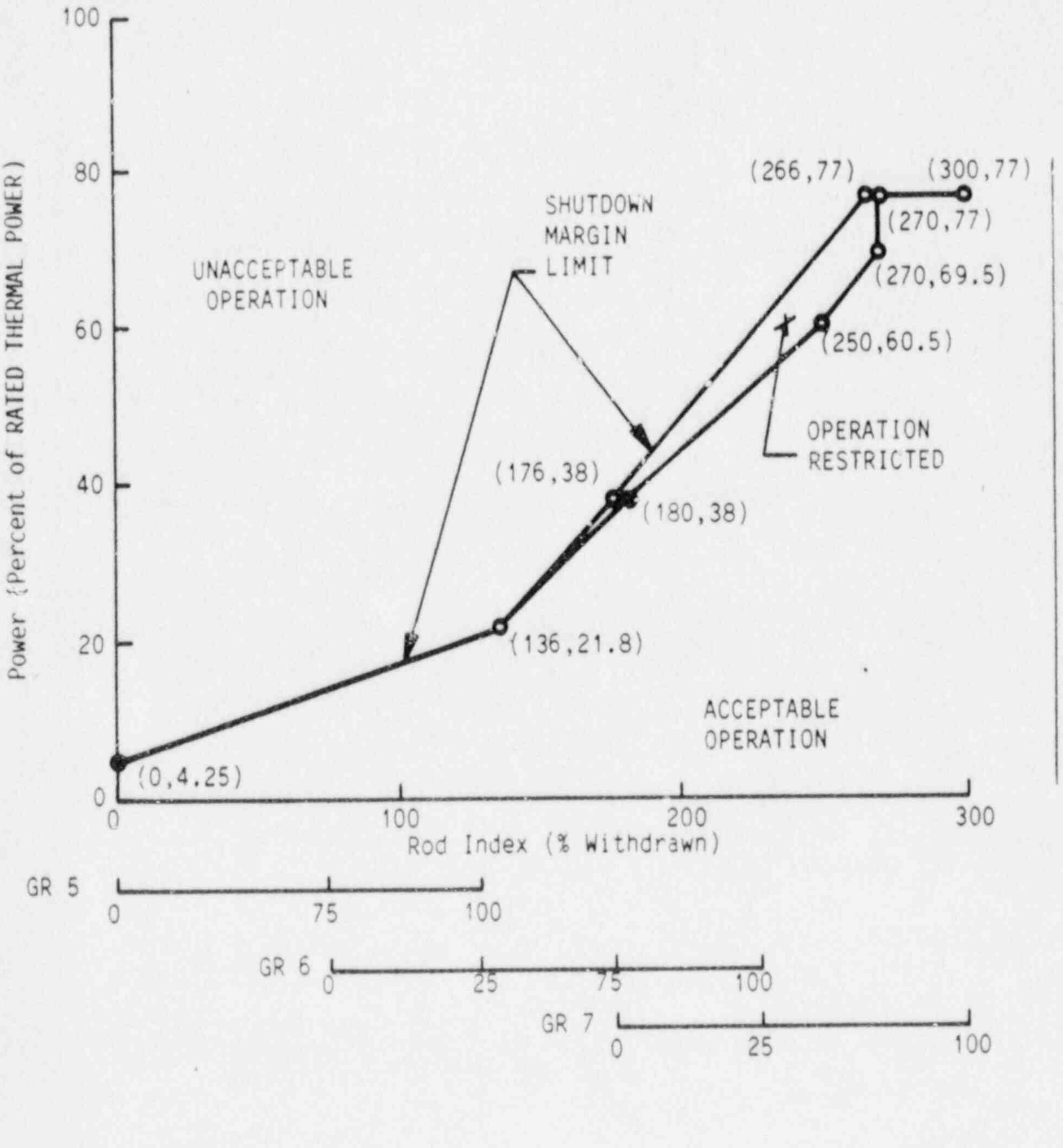
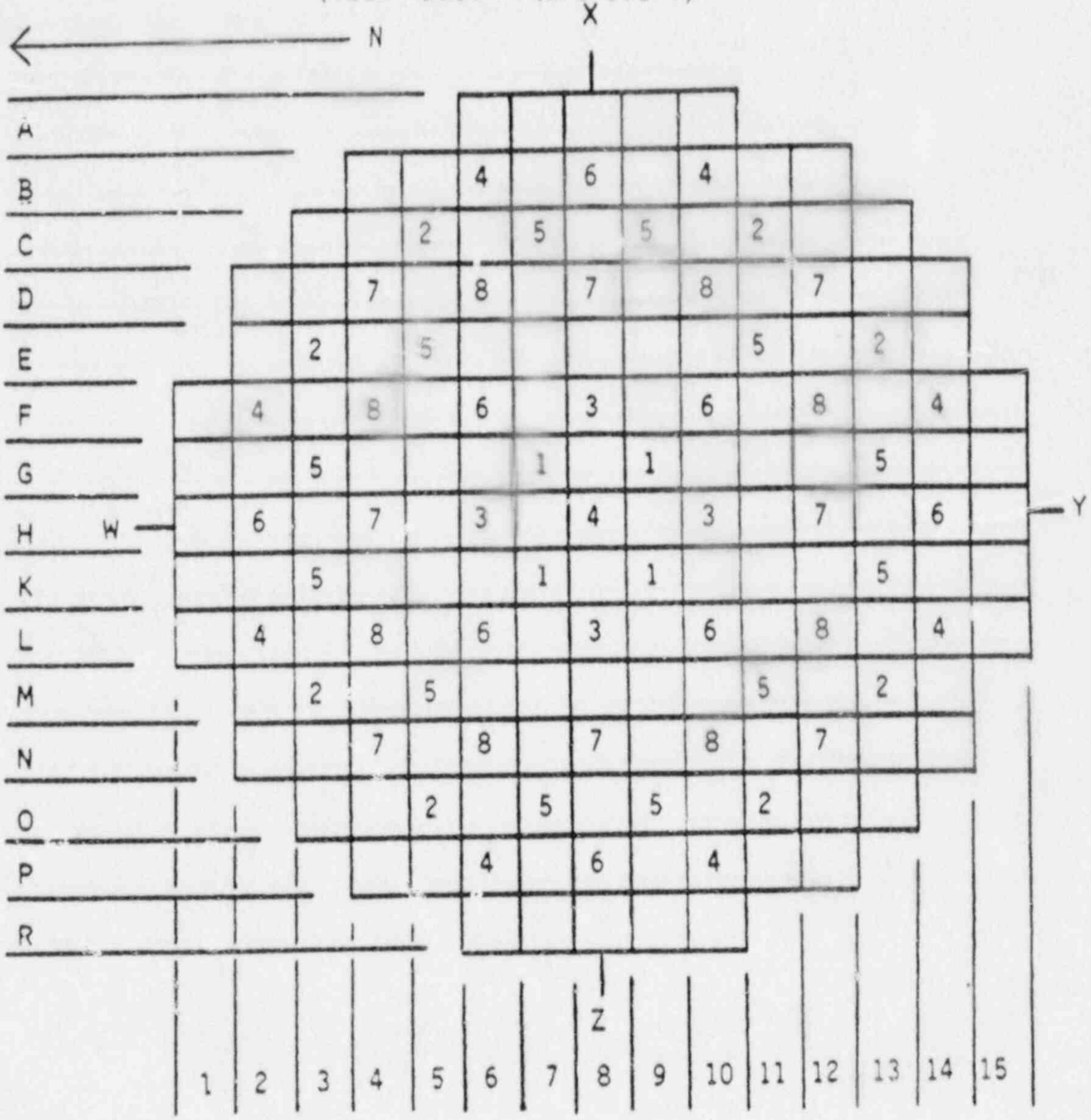


Figure 8-10. Control Rod Core Locations and Group Assignments -- Davis Besse 1, Cycle 6 (Tech Spec. Figure 3.1-4)



	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	

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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 $K_{eff} \geq 1.0$.

Figure 8-11. APSR Position Limits, 0 to 325 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 6 (Tech. Spec. Figure 3.1-5a)

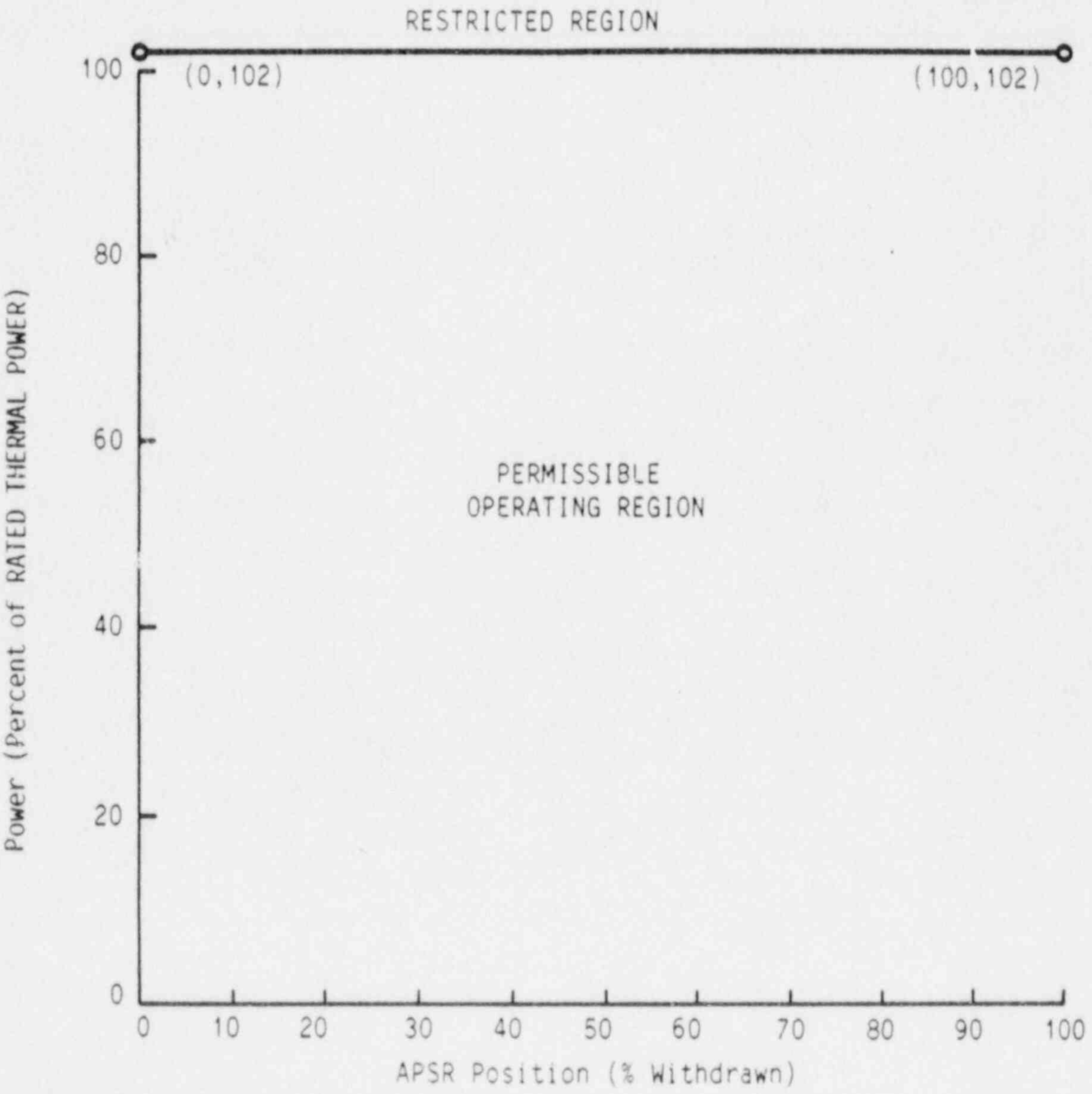


Figure 8-12. APSR Postion Limits After 325 ± 10 EFPD, Three or Four RC Pumps, APSRs Withdrawn -- Davis-Besse 1, Cycle 6 (Tech.Spec. Figure 3.1-5b)

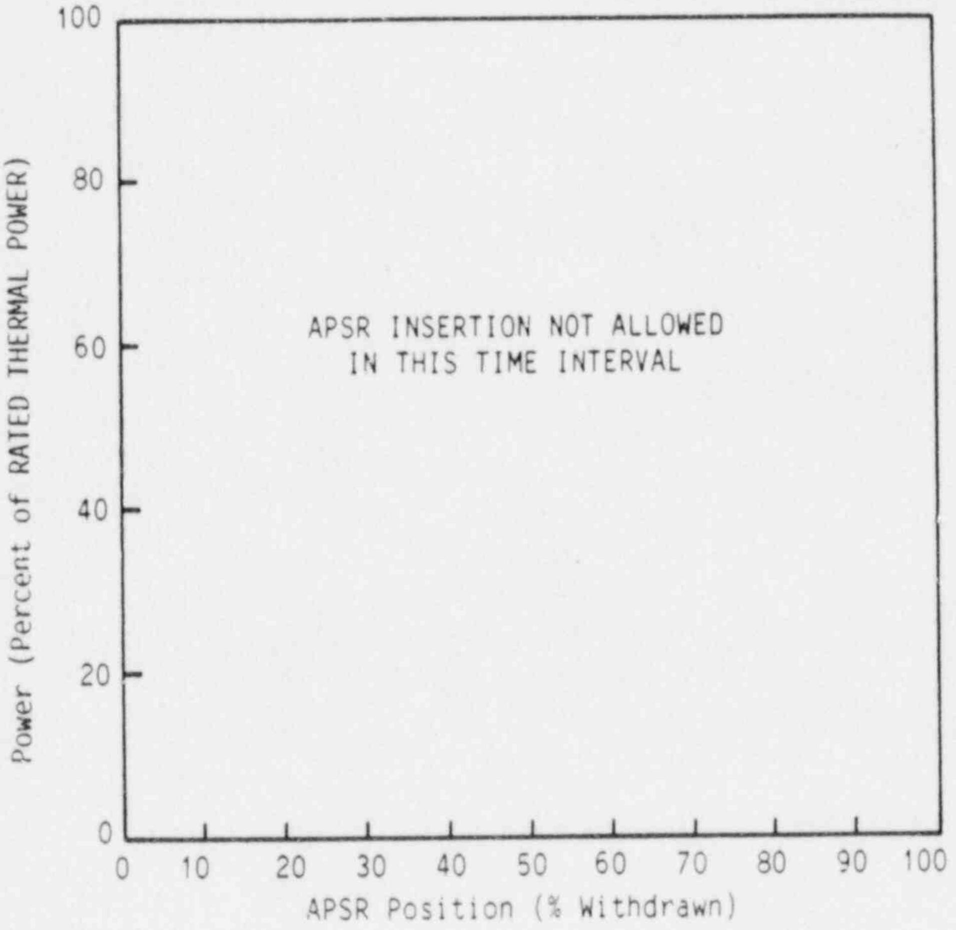
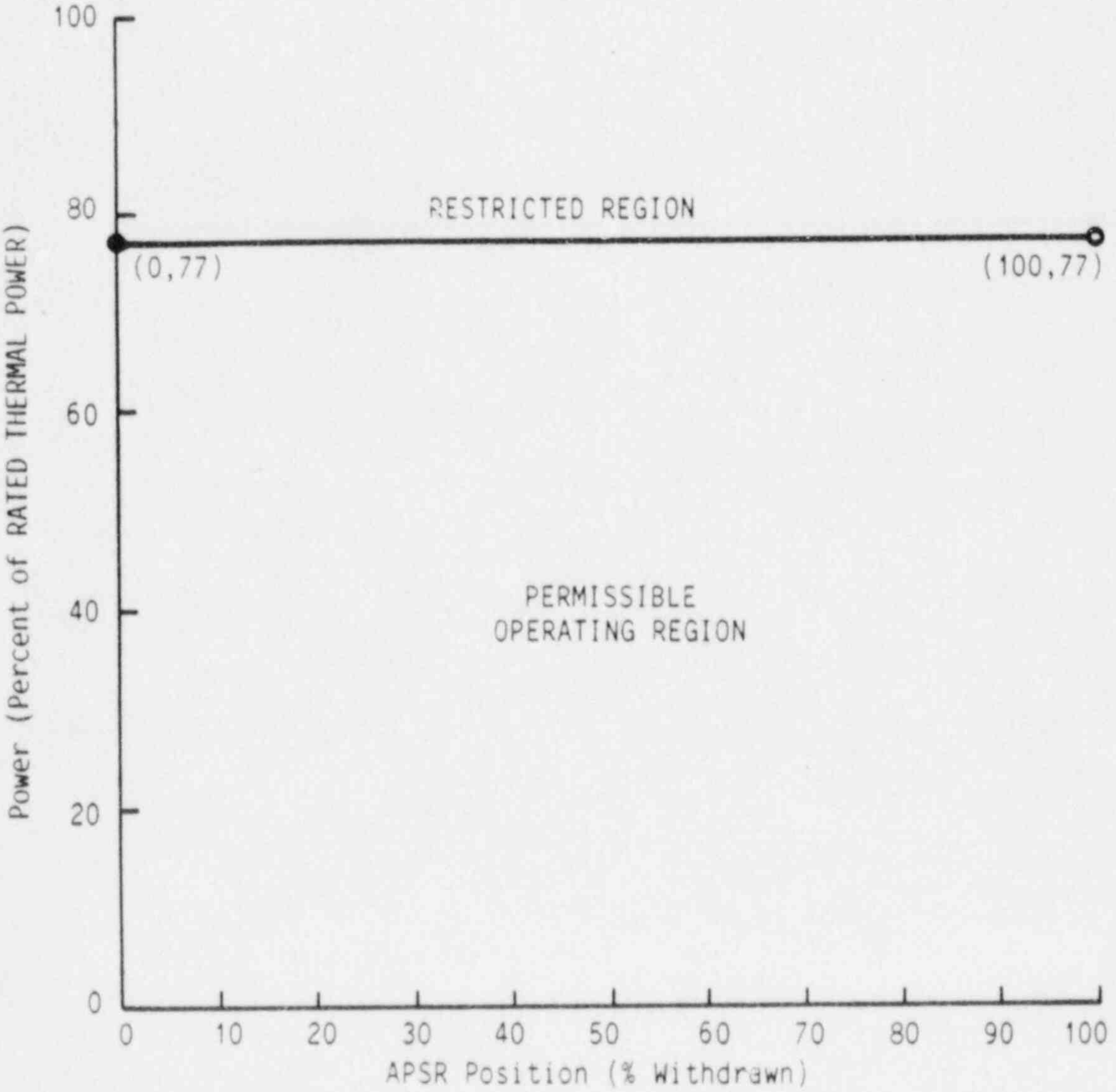


Figure 8-13. APSR Position Limits, 0 to 325±10 EFPD,
Three RC Pumps -- Davis-Besse 1, Cycle 6
(Tech. Spec. Figure 3.1-5c)



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 3.2-1 and 3.2-2.

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.

Figure 8-14. AXIAL POWER IMBALANCE Limits
Four RC Pumps -- Davis-Besse 1, Cycle 6
(Tech. Spec. Figure 3.2-1)

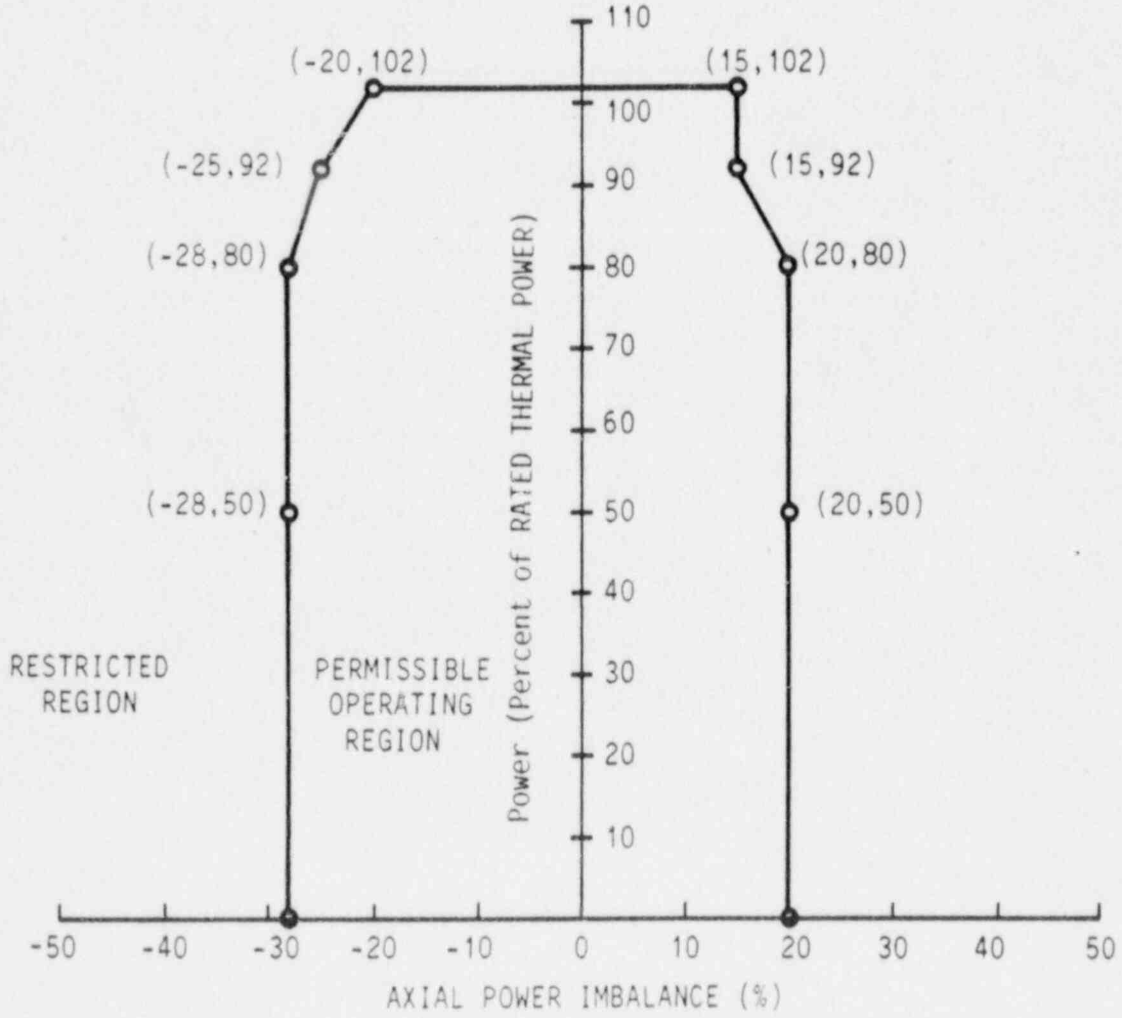
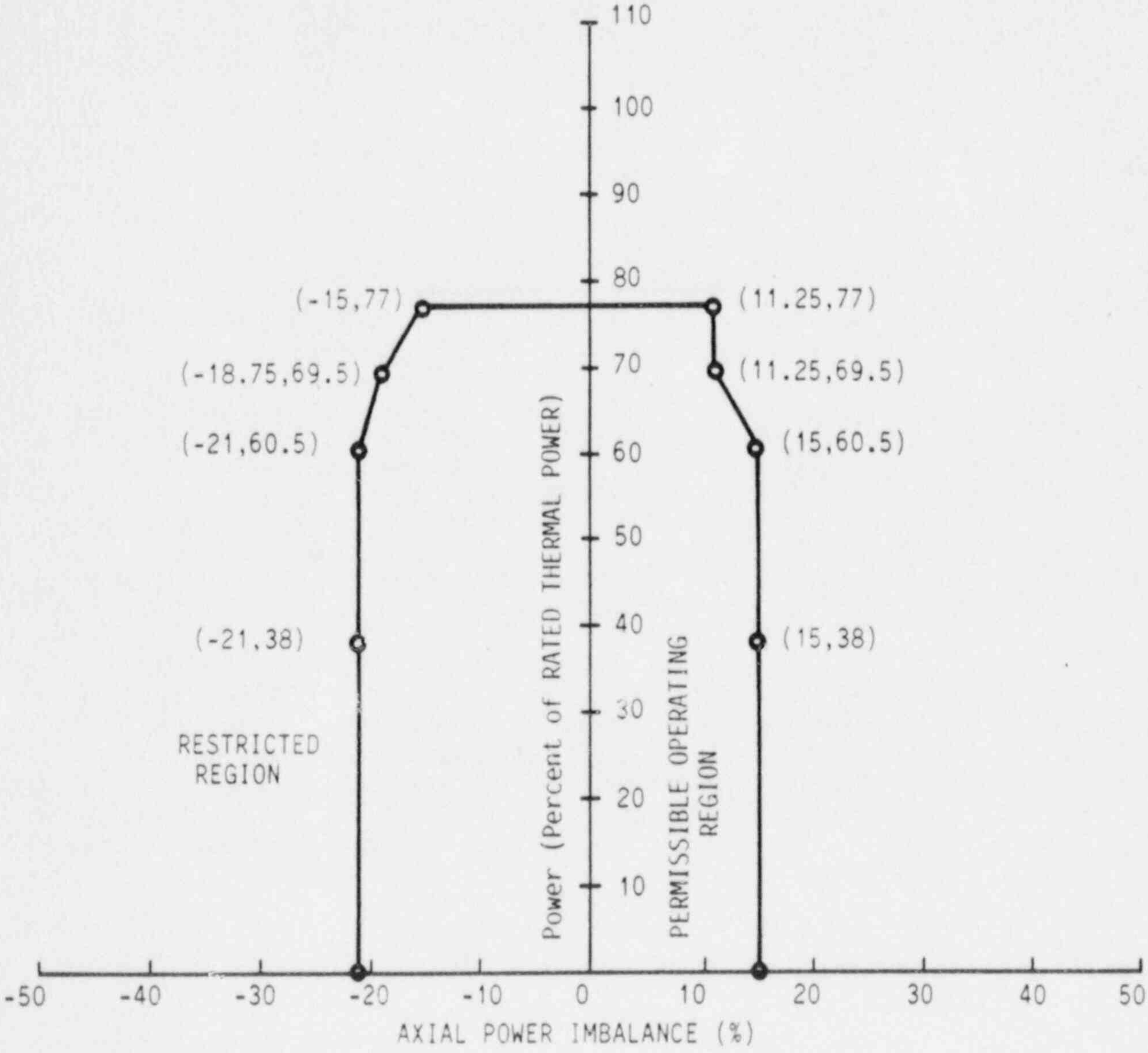


Figure 8-15. AXIAL POWER IMBALANCE Limits, Three RC Pumps -- Davis-Besse 1, Cycle 6 (Tech. Spec. Figure 3.2-2)



QUADRANT POWER TILT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT shall not exceed the Steady State Limit of Table 3.2-1.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER.*

ACTION:

a. With the QUADRANT POWER TILT determined to exceed the Steady State Limit but less than or equal to the Transient Limit of Table 3.2-1.

1. Within 2 hours:

   a) Either reduce the QUADRANT POWER TILT to within its Steady State Limit, or

   b) Reduce THERMAL POWER so as not to exceed THERMAL POWER, including power level cutoff, allowable for the reactor coolant pump combination less at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit and within 4 hours, reduce the High Flux Trip Setpoint and the Flux-Δ Flux-Flow Trip Setpoint at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit.

2. Verify that the QUADRANT POWER TILT is within its Steady State Limit within 24 hours after exceeding the Steady State Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

See Special Test Exception 3.10.1.

b. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table 3.2-1 due to misalignment of either a safety, regulating or axial power shaping rod:

   1. Reduce THERMAL POWER at least 2% for each 1% of indicated QUADRANT POWER TILT in excess of the Steady State Limit within 30 minutes.

   2. Verify that the QUADRANT POWER TILT is within its Transient Limit within 2 hours after exceeding the Transient Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table 3.2-1 due to causes other than the misalignment of either a safety, regulating or axial power shaping rod:

   1. Reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within 2 hours and reduce the High Flux Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

   2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit of Table 3.2-1, reduce THERMAL POWER to ≤ 15% of RATED THERMAL POWER within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT shall be determined to be within the limits at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT alarm is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

Table 8-2. Quadrant Power Tilt Limits
(Tech. Spec. Table 3.2-1)

	Steady state limit for THERMAL POWER < 50%	Steady state limit for THERMAL POWER > 50%	Transient limit	Maximum limit
QUADRANT POWER TILT as measured by:				
Symmetrical incore detector system	6.83	4.12	10.03	20.0
Power range channels	4.05	1.96	6.96	20.0
Minimum incore detector system	2.80	1.90	4.40	20.0

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2.

a. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1

ACTION:

If parameter a or b above exceeds its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

If parameter c exceeds its limit, either:

1. Restore the parameter to within its limit within 2 hours, or
2. Limit THERMAL POWER at least 2% below RATED THERMAL POWER for each 1% parameter c is outside its limit for four pump operation within the next 4 hours, or limit THERMAL POWER at least 2% below 75% of RATED THERMAL POWER for each 1% parameter c is outside its limit for 3 pump operation within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

Table 8-3. DNB Margin
(Tech. Spec. Table 3.2-2)

Parameter	Required Measured Parameters with Four Reactor Coolant Pumps Operating	Required Measured Parameters with Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H °F	≤ 610	≤ 610(1)
Reactor Coolant Pressure, psig(2)	≥ 2062.7	≥ 2058.7(1)
Reactor Coolant Flow Rate, gpm(3)	$\geq 389,500$	$\geq 290,957$

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(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These minimum required measured flows include a flow rate uncertainty of 2.5%, and are based on a minimum of 52 lumped burnable poison rod assemblies in place in the core.

Table 8-4. Reactor Protection System Instrumentation Surveillance Requirements
(Tech. Spec. Table 4.3-1)

Functional Unit		Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	High Flux	S	D(2), and Q(7)	M	1,2
3.	RC High Temperature	S	R	M	1,2
4.	Flux - ΔFlux - Flow	S(4)	M(3) and Q(7,8)	M	1,2
5.	RC Low Pressure	S	R	M	1,2
6.	RC High Pressure	S	R	M	1,2
7.	RC Pressure-Temperature	S	R	M	1,2
8.	High Flux/Number of Reactor Coolant Pumps On	S	R	M	1,2
9.	Containment High Pressure	S	R	M	1,2
10.	Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(5)(1)	1,2 and*
11.	Source Range, Neutron Flux and Rate	S	R(7)	M and S/U(1)(5)	2,3,4 and 5
12.	Control Rod Drive Trip Breakers	N.A.	N.A.	M and S/U(1)	1,2 and*
13.	Reactor Trip Module Logic	N.A.	N.A.	M	1,2 and*
14.	Shutdown Bypass High Pressure	S	R	M	2**,3**,4**,5**

Table 4.3-1 (Continued)

NOTATION

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP] and at steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP} [API_O - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is ≥ 2.5%.

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - Verify at least one decade overlap if not verified in previous 7 days.

(7) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(8) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once per 18 months.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

3/4.4. REACTOR COOLANT SYSTEM

3/4.4.1. COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 80.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced in accordance with Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux
   2. Flux-ΔFlux-Flow

SURVEILLANCE REQUIREMENTS

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

4.4.1.2 The Reactor Protection System trip setpoints for the instrumentation channels specified in the ACTION statement above shall be verified to be in accordance with Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

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

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

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*See Special Test Exception 3.10.3.

BORATED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The borated water storage tank (BWST) shall be operable with:

a. An available borated water volume of between 482,778 and 550,000 gallons,

b. Between 1800 and 2200 ppm of boron, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The BWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the available borated water volume in the tank,

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the water temperature when outside air temperature <35°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the reactor coolant system average temperature less than 525°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2. BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) makeup or DHR pumps, (3) separate flow paths, (4) boric acid pumps, (5) associated heat tracing systems, and (6) an emergency power supply from operable emergency busses.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs from full power equilibrium xenon conditions and requires the equivalent of either 7373 gallons of 8742 ppm borated water from the boric acid storage tanks or 52,726 gallons of 1800 ppm borated water from the borated water storage tank.

The requirement for a minimum available volume of 482,778 gallons of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4: therefore, the larger volume of borated water is specified.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the

3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 70°F. This condition requires either 600 gallons of 7875 ppm borated water from the boric acid storage system or 3,000 gallons of 1800 ppm borated water from the borated water storage tank.

The bottom 4 inches of the borated water storage tank are not available, and the instrumentation is calibrated to reflect the available volume. All boric acid tank volume is available. The limits on contained water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution recirculated within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensures that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The bottom 4 inches of the borated water storage tank are not available, and the instrumentation is calibrated to reflect the available volume. The limits on contained water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution sprayed within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 264°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2500 grams uranium. The initial core loading shall have a maximum enrichment of 3.0 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 53 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The nominal values of absorber material shall be 80 percent Silver, 15 percent Indium and 5 percent Cadmium. All control rods shall be clad with stainless steel tubing. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100% Inconel-600.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

9. STARTUP PROGRAM - PHYSICS TESTING

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

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

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

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

9.1.2. RC Flow

Reactor coolant flow with four RC pumps running will be measured at hot standby conditions. Acceptance criteria require that the measured flow be within allowable limits.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Once initial criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required to achieve the all rods out equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentration must be within ± 100 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal HZP temperature coefficient is measured at approximately the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. The change in reactivity is then calculated by the summation of reactivity associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-2}\%$ $\Delta k/k/^{\circ}F$.

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.9 \times 10^{-2}\%$ $\Delta k/k/^{\circ}F$.

9.2.3. Control Rod Group/Boron Reactivity Worth

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 establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes from 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:

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

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

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

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

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

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

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

The purpose of this test is to evaluate the symmetry of the core at low power during the initial power escalation following a refueling. Symmetry evaluation is based on incore quadrant power tilts during escalation to the intermediate power level. The core symmetry is acceptable if the absolute values of the quadrant power tilts are less than the error adjusted alarm limit.

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

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

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

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

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

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

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

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

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

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

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

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

Items 6 and 7 are established to determine if measured and predicted power distributions are consistent.

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

Imbalances, set up in the core by control rod positioning, are read simultaneously on the incore detectors and excore power range detectors. The excore detector offset versus incore detector offset slope must be greater than 0.96. If this criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required gain.

9.3.4. Temperature Reactivity Coefficient at ∿100% FP

The average reactor coolant temperature is decreased and then increased at constant reactor power. The reactivity associated with each temperature change is obtained from the change in the controlling rod group position. Controlling rod group worth is measured by the fast insert/withdraw method. The temperature reactivity coefficient is calculated from the measured changes in reactivity and temperature. Acceptance criteria state that the moderator temperature coefficient shall be negative.

9.3.5. Power Doppler Reactivity Coefficient at ~100% FP

The power Doppler reactivity coefficient is calculated from data recorded during control rod worth measurements at power using the fast insert/withdraw method.

The fuel Doppler reactivity coefficient is calculated in conjunction with the power Doppler coefficient measurement. The power Doppler coefficient as measured above is multiplied by a precalculated conversion factor to obtain the fuel Doppler coefficient. This measured fuel Doppler coefficient must be more negative than the acceptance criteria limit of $-0.90 \times 10^{-3}\%$ $\Delta k/k/^{\circ}F$.

9.4. Procedure for Use if Acceptance Criteria Not Met

If acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. This evaluation is performed by site test personnel with participation by Babcock & Wilcox technical personnel as required. Further specific actions depend on 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 evaluation shows that plant safety will not be compromised by such escalation.

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