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CRYSTAL RIVER UNIT 3

— Cycle 5 Reload Report —

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of Crystal River Unit 3 (cycle 5) at a rated core power level of 2544 MWt. Included are the required analyses to support cycle 5 operation; these analyses employ analytical techniques and design bases established in reports that have received technical approval by the U.S. Nuclear Regulatory Commission (NRC; see references).

The design for cycle 3 raised the rated thermal power from 2452 to 2544 MWt, which was the ultimate core power level identified in the Crystal River Unit 3 Final Safety Analysis Report (FSAR).[1] The cycle 5 core has been designed with an increased cycle lifetime of 460 effective full power days (EFPD) and the incorporation of burnable poison rod assemblies to aid in reactivity control.

The Technical Specifications have been reviewed, and the modifications for cycle 5 are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for emergency core cooling (ECC), it has been concluded that Crystal River 3 cycle 5 can be safely operated at a core power level of 2544 MWt.

2. OPERATING HISTORY

Cycle 4, the current Crystal River Unit 3 operating cycle, is the reference fuel cycle for the nuclear and thermal-hydraulic analyses performed for cycle 5 operation. Cycle 4 achieved criticality on December 10, 1981, completed power escalation testing on December 19, 1981, and is scheduled for completion in March 1983 after approximately 350 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in cycle 5.

Cycle 5 is scheduled to start operation in July 1983 at a rated power level of 2544 MWt. The design cycle length is 460 EFPD.

3. GENERAL DESCRIPTION

The Crystal River Unit 3 reactor core is described in detail in Chapter 3 of the FSAR for the unit.[1] The cycle 5 core consists of 177 fuel assemblies (FAs), each of which is a 15-by-15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The FAs in batches 5, 6, and 7 have an average nominal fuel loading of 463.6 kg of uranium, whereas the batch 4 assembly maintains an average nominal fuel loading of 468.6 kg of uranium. The cladding is cold-worked Zircaloy-4 with an outside diameter (OD) of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished-end, cylindrical pellets of uranium dioxide (see Table 4-2 for data).

Figure 3-1 is the core loading diagram for cycle 5 of Crystal River 3. The initial enrichments of batches 4D, 5B, 6A, and 6B were 2.64, 2.62, 2.62, and 2.95 wt % ^{235}U, respectively. The design enrichments of fresh batches 7A and 7B are 3.29 and 2.95 wt % ^{235}U, respectively. One batch 4D assembly that was discharged at the end of cycle 3 was re-inserted as the center assembly. Twenty-four batch 5 assemblies and 53 batch 4 assemblies will be discharged at the end of cycle 4. The batch 5B, 6A, and 6B assemblies will be shuffled to new locations with batches 6A and 6B on the periphery. The fresh batch 7 assemblies will be loaded primarily into the core interior in a symmetric checkerboard pattern. Figure 3-2 is an eighth-core map showing the burnup and initial enrichment of each assembly at the beginning of cycle 5.

Cycle 5 will be operated in a feed-and-bleed mode. Core reactivity is controlled by 61 full-length Ag-In-Cd control rod assemblies (CRAs), 56 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length CRAs, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle 5 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The cycle 5 locations and enrichments of the BPRA clusters are shown in Figure 3-4. APSRs will be withdrawn at 399 ± 10 EFPD of operation.

Figure 3-1. Fuel Shuffle for Crystal River 3 Cycle 5

FUEL TRANSFER CANAL →

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

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						6B N03	6A K02	7A F	6A K14	6B N13					
B				6A L05	7A F	6A M02	7A F	5B A10	7A F	6A M14	7A F	6A L11			
C			6B H03	7A F	5B A07	7A F	6A M04	7A F	6A M12	7A F	5B A09	7A F	6B C08		
D		6A E10	7A F	5B A06	7A F	6A K04	7A F	5B C13	7A F	6A K12	7A F	5B F15	7A F	6A E06	
E		7A F	5B G01	7A F	6A C06	7A F	6A K06	7A F	6A K10	7A F	6A F13	7A F	5B G15	7A F	
F	6B C12	6A B11	7A F	6A D09	7A F	5B D02	7B F	5B P06	7B F	5B B12	7A F	6A D07	7A F	6A B05	6B C04
G	6A B09	7A F	6A D11	7A F	6A F09	7B F	5B P12	6A O06	5B N02	7B F	6A F07	7A F	6A D05	7A F	6A B07
H	7A F	5B F01	7A F	5B C03	7A F	5B L14	6A L13	4D CY3 G07	6A F03	5B F02	7A F	5B O13	7A F	5B L15	7A F
K	6A P09	7A F	6A N11	7A F	6A L09	7B F	5B O14	6A C10	5B B04	7B F	6A L07	7A F	6A N05	7A F	6A P07
L	6B O12	6A P11	7A F	6A N09	7A F	5B P04	7B F	5B B10	7B F	5B N14	7A F	6A N07	7A F	6A P05	6B O04
M		7A F	5B K01	7A F	6A L03	7A F	6A G06	7A F	6A G10	7A F	6A O10	7A F	5B K15	7A F	
N		6A M10	7A F	5B L01	7A F	6A G04	7A F	5B O03	7A F	6A G12	7A F	5B R10	7A F	6A M06	
O			6B O08	7A F	5B R07	7A F	6A E04	7A F	6A E12	7A F	5B R09	7A F	6B H13		
P				6A F05	7A F	6A E02	7A F	5B R06	7A F	6A E14	7A F	6A F11			
R						6B O03	6A G02	7A F	6A G14	6B O13					

Legend: Batch Number / Previous Cycle Location

Figure 3-2. Enrichment and BOC Burnup Distribution for Crystal River 3 Cycle 5 Off 350 EFPD Cycle 4

	8	9	10	11	12	13	14	15
H	2.64 17,272	2.62 13,160	2.62 16,584	3.29 0	2.62 13,892	3.29 0	2.62 11,886	3.29 0
K		2.62 14,515	2.95 0	2.62 13,542	3.29 0	2.62 13,331	3.29 0	2.62 10,918
L			2.62 14,557	3.29 0	2.62 13,566	3.29 0	2.62 9,472	2.95 11,397
M				2.62 13,150	3.29 0	2.62 12,634	3.29 0	
N					2.62 11,880	3.29 0	2.62 13,237	
O						2.95 13,383		
P								
R								

X.XX	Enrichment, Initial
XXXXX	Burnup (MWd/mtU), BOC

Figure 3-3. Control Rod Locations and Group Designations for Crystal River 3 Cycle 5

X

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

W — Y

Z

x Group Number

Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	9	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total	69	

Figure 3-4. LBP Enrichment and Distribution for Crystal River 3 Cycle 5

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

X.X	LBP concentration (wt % B_4C in Al_2O_3)

	No. of BPRAs	Concentration, wt % B_4C
	40	1.4
	8	0.5
	8	0.2
Total	56	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Crystal River Unit 3 cycle 5 are listed in Table 4-1. All the FAs are identical in concept and are interchangeable.

Retainer assemblies will be used on 56 assemblies that contain BPRAs and two assemblies that contain regenerative neutron sources. The justification for the design and use of the retainers is described in references 2 and 3.

4.2. Fuel Rod Design

The fuel rod design of batch 7 fuel is identical to that of the batch 6 fuel used in cycle 4 (see Table 4-1). The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

The batch 5 fuel is more limiting than batches 4, 6, and 7 due to previous incore exposure time. The batch 5 assembly power histories were analyzed to determine the most limiting three-cycle power history for creep-collapse. The worst-case power history was then compared against a generic analysis to ensure that creep ovalization will not affect fuel performance during cycle 5. The generic analysis was performed based on reference 4 and is applicable to batch 5 design.

The creep-collapse analyses predict a collapse time greater than 35,000 effective full power hours (EFPH), which is longer than the maximum expected residence time of 26,911 EFPH (Table 4-1).

4.2.2. Cladding Stress

The Crystal River 3 cycle 5 stress parameters are enveloped by a conservative fuel rod stress analysis. The methods used for the analysis of this cycle have been used in previous cycles.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic tensile circumferential strain. The pellet is designed to assure that cladding plastic strain is less than 1% at design local pellet burnup and heat generation rate. The design burnup and heat generation rate are higher than the worst-case values cycle 5 fuel is expected to experience. The strain analysis is also based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance value for the cladding inside diameter (ID).

4.3. Fuel Thermal Design

All fuel in the cycle 5 core is thermally similar. The fresh batch 7 fuel inserted for cycle 5 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The cycle 5 thermal analyses represent a change in the analytical method in that analyses for the fresh batch 7 fuel have been performed with the TACO2[5] code using the methodology described in reference 6. The analysis uses nominal undensified input parameters as provided in Table 4-2. Densification effects are accounted for in the TACO2 code densification model. The TACO2 analyses also apply to the reinserted batch 5B, 6A, and 6B fuel since this fuel is identical in design. Batch 4D fuel continues to be supported by the TAFY3[7] analyses performed for prior cycles.

The results of the thermal design evaluation for cycle 5 core are summarized in Table 4-2. The linear heat rate (LHR) to melt capabilities for the batch 5B, 6A, 6B, 7A, and 7B fuel (95% TD nominal initial density) were determined with the TACO2 fuel pin performance code. Maximum LHR to centerline melt was determined as a function of fuel burnup; the lowest maximum LHR was 20.5 kW/ft for the 95% TD fuel. The 94% TD batch 4D fuel was analyzed with the TAFY3 fuel pin performance code and found to have a maximum LHR to centerline melt of 20.1 kW/ft. The maximum fuel rod burnup at EOC 5 is predicted to be 33,546 MWd/mtU. Fuel rod internal pressure has been evaluated with TAFY3 for the fuel rod of highest burnup and is predicted to be less than the nominal RC system pressure of 2200 psia.

4.4. Operating Experience

Babcock & Wilcox (B&W) operating experience with the Mark B 15-by-15 fuel assembly has verified the adequacy of its design. As of October 31, 1982, the

following experience has been accumulated for the eight operating B&W 177-fuel assembly plants using the Mark B fuel assembly:

Reactor	Current cycle	Max FA burnup[a] Mwd/mtU Incore	Max FA burnup[a] Mwd/mtU Discharged	Cumulative net electrical output,[b] MWh
Oconee 1	7	44,850	40,000	38,723,077
Oconee 2	6	23,750	36,800	34,354,735
Oconee 3	7	20,200	35,450	36,772,920
TMI-1	5	25,000	32,400	23,840,053
ANO-1	5	36,429	33,220	32,834,786
Rancho Seco	5	35,821	37,730	28,636,196
Crystal River 3	4	24,360	29,900	19,803,456
Davis-Besse	3	25,742	25,326	12,021,378

[a]As of October 31, 1982.
[b]As of May 31, 1982.

Table 4-1. Fuel Rod Design Parameters

	Batch					
	4D	5B	6A	6B	7A	7B
Fuel assembly type	Mark B4	Mark B4	Mark B4	Mark B4	Mark B4	Mark B4
Number of assemblies	1	32	56	12	68	8
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377	0.377
Flexible spacer type	Spring	Spring	Spring	Spring	Spring	Spring
Rigid spacer type	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Undensified active fuel length, in.	143.6	141.80	141.80	141.80	141.80	141.80
Fuel pellet (mean specified) diameter, in.	0.3697	0.3686	0.3686	0.3686	0.3686	0.3686
Fuel pellet initial density (mean specified), % TD	94.0	95.0	95.0	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	2.64	2.62	2.62	2.95	3.29	2.95
Estimated residence time, EFPH	22,363	26,911	19,440	19,440	11,040	11,040
Cladding collapse time, EFPH	>35,000	>35,000	>35,000	>35,000	>35,000	>35,000

Table 4-2. Fuel Performance Design Parameters

	Batch					
	4D	5B	6A	6B	7A	7B
No. of assemblies	1	32	56	12	68	8
Initial nominal density, % TD	94[a]	95[b]	95[b]	95[b]	95[b]	95[b]
Pellet diameter, in.	0.3697	0.3686	0.3686	0.3686	0.3686	0.3686
Nominal stack height, in.	143.60	141.80	141.80	141.80	141.80	141.80
Enrichment, wt % ^{235}U	2.64	2.62	2.62	2.95	3.29	2.95
Nominal LHR at 2544 MWt, kW/ft	5.62	5.69	5.69	5.69	5.69	5.69
LHR to centerline fuel melt, kW/ft	20.1	20.5	20.5	20.5	20.5	20.5

Core average LHR = 5.69 kW/ft

Densified Fuel Parameters[a], TAFY3 Analysis Only

Pellet diameter, in.	0.3648
Fuel stack height, in.	141.80
Nominal LHR at 2544 MWt, kW/ft	5.69
Avg fuel temp at nom. LHR (BOL), F	1280

[a] Densification to 96.5% TD assumed.

[b] Densification to 96.3% TD assumed.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 4 and 5; these values were generated using PDQ-7[3-5] for both cycles. The differential cycle burnup will be larger for cycle 5 than for cycle 4 because of the longer cycle 5 lenlength. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 5 at full power with equilibrium xenon and nominal rod positions.

Operational changes as well as differences in cycle length, feed enrichment, BPRA loading, shuffle pattern, and rod group designations for cycle 5 account for differences in the physics parameters from those of cycle 4. The critical boron concentrations for cycle 4 and 5 are given in Table 5-1.

The control rod worths differ between cycles due to changes in radial flux and burnup distributions. 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 maximum stuck rod worth for cycle 5 is greater than that for design cycle 4 at BOC and less at EOC. The adequacy of the shutdown margin with cycle 5 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. Flux redistribution penalty.

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of cycle 5 was analyzed at 399 EFPD and EOC. 399 EFPD is the latest time (±10 EFPD) in core life at which the APSRs are inserted.

5.2. Changes in Nuclear Design

There are no significant core design changes between the reference and reload cycles. The only change is the increase in cycle lifetime to 460 EFPD. The calculational methods and design information used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle. There are two significant operational changes from the reference cycle: the withdrawal of the APSRs at 399 EFPD and a change from rodded to a feed-and-bleed mode of operation. The design cycle length of 460 EFPD is to be achieved by a planned power coastdown from 440 EFPD to EOC. The stability and control of the core in the feed-and-bleed mode with APSRs removed have been analyzed. The calculated stability index without APSRs is -0.0428 h^{-1}, which demonstrates the axial stability of the core. The operational limits and RPS limits (Technical Specification changes) for cycle 5 are presented in section 8.

Table 5-1. Physics Parameters, Crystal River 3, Cycles 4 and 5[a]

	Cycle 4	Cycle 5
Design cycle length, EFPD	350	460
Design cycle burnup, MWd/mtU	10,814	14,260
Design average core burnup — EOC, MWd/mtU	17,737	21,571
Design initial core loading, mtU	82.3	82.1
Critical boron — BOC, ppm (no Xe)		
HZP, group 8 inserted[b]	1,374	1,522
HFP[b,c]	1,090	1,334
Critical boron — EOC, ppm (eq Xe)[d]		
HZP	298	307
HFP	2	(e)
Control rod worths — HFP, BOC, % Δk/k		
Group 6	1.11	1.10
Group 7	0.99	1.47
Group 8	0.41	0.43
Control rod worths — HFP, EOC, % Δk/k[e]		
Group 7	1.23	1.45
Group 8[f]	0.44	0.51
Max ejected rod worth[g] — HZP, % Δk/k		
BOC (N-12)	0.51	0.62
EOC (N-12)	0.54	0.57
Max stuck rod worth — HZP, % Δk/k		
BOC (M13,CY4)(N12,CY5)	1.54	1.72
EOC (H14,CY4)(N12,CY5)	2.14	1.59[h]
Power deficit, HZP to HFP, % Δk/k[b]		
BOC	-1.44	-1.50
EOC	-2.36	-2.33
Doppler coeff, 10^{-5} (Δk/k/°F)		
BOC, 100% power, no Xe	-1.55	-1.47
EOC, 100% power, eq Xe	-1.71	-1.73
Moderator coeff — HFP, 10^{-4} (Δk/k/°F)		
BOC (0 Xe, critical ppm, group 8 inserted)	-0.52	-0.45
EOC (eq Xe, 17 ppm)[d]	-2.89	-2.71
Boron worth — HFP, ppm/% Δk/k		
BOC, 1334 ppmb, bank 8 in.	111	121
EOC, 17 ppmb[d]	99	104
Xenon worth — HFP, % Δk/k		
BOC (4 EFPD)	2.63	2.57
EOC (equilibrium)	2.72	2.69
Effective delayed neutron fraction — HFP		
BOC	0.00628	0.00638
EOC	0.00523	0.00520

(a) Cycle 5 data are for the conditions stated in [illegible] the cycle 4 values given are at the core conditi[illegible] in reference 2.

(b) HZP denotes hot zero power (532F T_{avg}) [illegible] full power (579F T_{avg}).

(c) Group 7, 8 inserted for cycle 4, group 8 inserted for cycle 5.

(d) Group 8 inserted for cycle 4, group 8 out for cycle 5.

(e) EOC ≈ 450 EFPD at 100% FP (17 ppmb) with power coastdown to 460 EFPD for cycle 5.

(f) Bank 8 worth at 399 EFPD, the latest time in core life in which it is inserted.

(g) Ejected rod worth for groups 5 through 8 inserted.

(h) Stuck rod worth at 399 EFPD.

Table 5-2. Shutdown Margin Calculation for Crystal River 3 Cycle 5

	BOC, % $\Delta k/k$	399 EFPD,[a] % $\Delta k/k$	EOC, % $\Delta k/k$
Available Rod Worth			
Total rod worth, HZP[b]	8.75	9.24	9.04
Worth reduction due to burnup of poison material	-0.42	-0.42	-0.42
Maximum stuck rod worth, HZP	-1.72	-1.59	-1.41
Net worth	6.61	7.23	7.21
Less 10% uncertainty	-0.66	-0.72	-0.72
Total available worth	5.95	6.51	6.49
Required Rod Worth			
Power deficit, HFP to HZP	1.50	2.32	2.33
Max allowable inserted rod worth	0.21	0.60	0.60
Flux redistribution	0.52	1.17	1.19
Total required worth	2.23	4.09	4.12
Shutdown Margin			
Total available minus total required	3.72	2.42	2.37

Note: Required shutdown margin is 1.00% $\Delta k/k$.

[a] The latest time in core life at which the APSR bank is in.

[b] HZP: hot zero power, HFP: hot full power.

Figure 5-1. BOC (4EFPD), Cycle 5 Two-Dimensional Relative Power Distribution — HFP, Equilibrium Xenon, Bank 8 Inserted

	8	9	10	11	12	13	14	15
H	0.84	0.95	1.02	1.29	1.11	1.27	1.01	0.81
K		0.97	1.25	1.16	1.26	1.11	1.18	0.56
L			1.10	1.26	8 0.98	1.15	0.86	0.42
M				1.12	1.20	0.97	0.90	
N					1.06	1.15	0.46	
O						0.57		
P								
R								

X	Inserted rod group No.
X.XX	Relative power density

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 5 operation used the methods and models described in references 1, 9, and 10. The incoming batch 7 fuel is hydraulically similar to the fuel remaining in the core from previous cycles. The cycle 4 and 5 maximum design conditions and significant parameters are shown in Table 6-1. The minimum departure from nucleate boiling ratio (DNBR) for cycle 5 is based on 106.5% of reactor coolant design flow, 8.1% maximum core bypass flow, a 1.71 reference design radial × local peaking factor, and includes the effects of incore fuel densification.

A rod bow penalty has been calculated according to the procedure approved in reference 11. The burnup-dependent penalty is calculated using the maximum fuel assembly burnup of the batch that contains the limiting (maximum radial × local peak) fuel assembly. For cycle 5, this burnup is 20,646 MWd/mtU in a batch 7A assembly. The resulting net rod bow penalty after inclusion of the 1% flow area reduction factor credit is 0.0% DNBR reduction.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 4, 2544 MWt	Cycle 5, 2544 MWt
Design power level, MWt[a]	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Reference design radial × local power peaking factor, $F_{\Delta H}$	1.71	1.71
Reference design axial flow shape	1.5 cosine	1.5 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Densified active length, in.[a]	140.2	140.2
Average heat flux at 100% power, Btu/h-ft^2	176×10^3	176×10^3
Maximum heat flux at 100% power, Btu/h-ft^2	452×10^3	452×10^3
CHF correlation	BAW-2	BAW-2
Minimum DNBR, % power	2.05(112)	2.05(112)

[a]Used in analysis.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 5 parameters to determine the effect of the cycle 5 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in reference 9. Since batch 7 reload FAs contain fuel rods whose theoretical density is higher than those considered in the reference 9 report, the conclusions in that reference are still valid with the exception of the four-pump coastdown and the locked-rotor accident. The locked-rotor accident was re-evaluated at 102% of 2568 MWt for cycle 3 operation and remains valid for cycle 5. The cycle 4 four-pump coastdown analysis, performed with an initial power level of 102% of 2544 MWt and a pump monitor delay time of 1.5 seconds, bounds cycle 5 and remains valid.

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: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. The cycle 5 thermal-hydraulic maximum design conditions are compared to the previous cycle 4 values in Table 6-1. These parameters are common to all the accidents considered in this report. A comparison of the key kinetics parameters from the FSAR and cycle 5 is provided in Table 7-1.

Generic LOCA analyses for B&W 177-FA lowered-loop NSSs have been performed using the Final Acceptance Criteria ECCS Evaluation Model. The large-break analysis is presented in a topical report and is further substantiated in a letter report.[12,13] The small-break analysis is also presented in a letter

report.[14] These analyses used the limiting values of key parameters for all plants in the category. Furthermore, the average fuel temperature as a function of LHR and lifetime pin pressure data used in the LOCA limits analysis[12] are conservative compared to those calculated for this reload. Thus, these analyses and LOCA limits provide conservative results for the operation of Crystal River 3 cycle 5.

The proposed Crystal River 3 long-term ECCS modification for small-break LOCAs is presented in reference 15.

The LOCA analysis used a power level of 2772 MWt, which is conservative relative to the 2544 MWt rating. Table 7-2 shows the bounding values for allowable LOCA peak linear heat rates for Crystal River 3. The limits shown in Table 7-2 include the effect of mechanistic fuel densification.

It is concluded from the examination of cycle 5 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the ability of the Crystal River 3 plant to operate safely during cycle 5. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 5 is bounded by previously accepted analyses. The initial conditions for the transients in cycle 5 are bounded by the FSAR with the exception of the four-pump coastdown and locked-rotor accidents, which were re-evaluated for the conditions discussed in section 7.1.

7.3. Dose Consequences of Accidents

The radiological dose consequences of the accidents presented in chapter 14 of the FSAR were re-evaluated for this reload report. This re-evaluation takes into account the irradiation history of the various fuel batches constituting the cycle 5 fuel loading.

Crystal River 3 dose evaluations for cycle 3 have accounted for changes in fuel management procedures which have resulted in increased plutonium fission. Updated isotope half-life and fission yield values have also been previously incorporated. Revised reactor building spray/iodine removal constants were utilized in this reload report.

The steam generator tube rupture accident dose consequence values calculated for this reload report take into consideration the increased amount of steam released to the environment via the main steam relief and atmospheric dump valves which results from the slower depressurization due to reduced heat transfer after a reactor coolant pump trip at actuation of high pressure injection without consideration of iodine spiking (a post-TMI-2 modification).

A comparison of the radiological doses calculated for cycle 5 to those previously reported for cycle 3 as shown in Table 7-3 shows that all cycle 5 dose values are either bounded by the cycle 3 values or are a small fraction of the 10 CFR 100 limits, i.e., below 30 REM to the thyroid and 2.5 REM to the whole body.

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

Parameter	FSAR[1] densification[6] value	Predicted cycle 5 value
BOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.17	-1.47
EOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.30	-1.73
BOL moderator coeff, 10^{-4} $\Delta k/k/°F$	0[(a)]	-0.45
EOL moderator coeff, 10^{-4} $\Delta k/k/°F$	-4.0[(b)]	-2.71
All-rod bank worth at BOL, HZP, % $\Delta k/k$	12.9	8.75
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	121
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.30
Dropped rod worth (HFP), % $\Delta k/k$	0.40	0.20
Initial boron conc. (HFP), ppm	1150	1334

(a) $+0.50 \times 10^{-4}$ $\Delta k/k/°F$ was used for the moderator dilution accident.

(b) -3.0×10^{-4} $\Delta k/k/°F$ was used for the steam line failure and dropped rod accident analyses.

Table 7-2. LOCA Limits for Crystal River 3

	Linear Heat Rate, kW/ft		
Elevation, ft	0-30 days	30-250 days	250 days to EOC
2	13.5	15.0	15.5
4	16.1	16.6	16.6
6	17.5	18.0	18.0
8	17.0	17.0	17.0
10	16.0	16.0	16.0

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

Parameter	FSAR[1] densification[6] value	Predicted cycle 5 value
BOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.17	-1.47
EOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.30	-1.73
BOL moderator coeff, 10^{-4} $\Delta k/k/°F$	$0^{(a)}$	-0.45
EOL moderator coeff, 10^{-4} $\Delta k/k/°F$	$-4.0^{(b)}$	-2.71
All-rod bank worth at BOL, HZP, % $\Delta k/k$	12.9	8.75
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	121
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.30
Dropped rod worth (HFP), % $\Delta k/k$	0.40	0.20
Initial boron conc. (HFP), pp.	1150	1334

$^{(a)}$+0.50 × 10^{-4} $\Delta k/k/°F$ was used for the moderator dilution accident.

$^{(b)}$-3.0 × 10^{-4} $\Delta k/k/°F$ was used for the steam line failure and dropped rod accident analyses.

Table 7-2. LOCA Limits for Crystal River 3

	Linear Heat Rate, kW/ft		
Elevation, ft	0-30 days	30-250 days	250 days to EOC
2	13.5	15.0	15.5
4	16.1	16.6	16.6
6	17.5	18.0	18.0
8	17.0	17.0	17.0
10	16.0	16.0	16.0

Table 7-3. Comparison of Cycle 3 and Cycle 5 Accident Doses

Accident	Cycle 3 dose (Rem)	Cycle 5 dose[c] (Rem)
Fuel handling accident		
Thyroid dose at EAB (2 h)	14.0	2.53
Whole body dose at EAB (2 h)	0.19	0.05
Control rod ejection accident		
Thyroid dose at EAB (2 h)	0.65	0.33
Whole body dose at EAB (2 h)	0.0008	0.0003
Thyroid dose at LPZ (30 day)	0.35	0.17
Whole body dose at LPZ (30 day)	0.0005	0.0002
Steam line break		
Thyroid dose at EAB (2 h)	0.503	0.45
Whole body dose at EAB (2 h)	0.0033	0.0012
Steam generator tube failure		
Thyroid dose at EAB (2 h)	0.0023	1.46
Whole body dose at EAB (2 h)	0.13	0.042
Waste gas tank failure		
Thyroid dose at EAB (2 h)	1.43	1.28
Whole body dose at EAB (2 h)	0.92	0.12
LOCA		
Thyroid dose at EAB (2 h)	2.19	2.13
Whole body dose at EAB (2 h)	0.016	0.008
Thyroid dose at LPZ (30 day)	0.517	0.46
Whole body dose at LPZ (30 day)	0.0081	0.004
MHA		
Thyroid dose at EAB (2 h)	86.8	61.8
Whole body dose at EAB (2 h)	2.28	1.48
Thyroid dose at LPZ (30 day)	18.4	14.8
Whole body dose at LPZ (30 day)	0.44	0.17

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

All Technical Specifications have been reviewed by Florida Power Corporation and B&W and revisions have been made to accommodate cycle 5 operation. Table 8-1 lists the Technical Specification changes and cross-references this reload report numbers with the Technical Specification numbers.

The review of the Technical Specifications based on the analysis presented in this report and the proposed modifications contained in this section, ensure that the Final Acceptance Criteria ECCS limits will not be exceeded nor will the thermal design criteria be violated.

Table 8-1. Technical Specification Changes

Tech Spec No. (figure, table Nos.)	Report page Nos. (figure, table Nos.)	Reason for change
(Figure 2.1-2)	8-3 (Figure 8-1)	Revised for cycle 5 operation.
(Table 2.2-1)	8-4, 8-5 (Table 8-2)	Added nuclear overpower based on RCPPM trip, revised set-points for cycle 5 operation.
(Figure 2.2-1)	8-6 (Figure 8-2)	Revised for cycle 5 operation.
2.2 BASES	8-7 thru 8-10	Revised for cycle 5 operation with RCPPMs.
3/4.1	8-11 thru 8-17	Shutdown margin and borated water volume requirements were revised for cycle 5 operation.
3.1.3.6	8-18	New figure Nos. were added.
	8-19 thru 8-26	Revised regulating rod group limits for three- and four-pump operation for cycle 5.
(Figure 3.1-1)	(Figure 8-3)	
(Figure 3.1-1a)	(Figure 8-4)	
(Figure 3.1-2)	(Figure 8-5)	
(Figure 3.1-2a)	(Figure 8-6)	
(Figure 3.1-3)	(Figure 8-7)	
(Figure 3.1-3a)	(Figure 8-8)	
(Figure 3.1-4)	(Figure 8-9)	
(Figure 3.1-4a)	(Figure 8-10)	
3.1.3.9	8-27	New figure No. was added.
	8-28 thru 8-31	Revised APSR limits for cycle 5 operation.
(Figure 3.1-9)	(Figure 8-11)	
(Figure 3.1-9a)	(Figure 8-12)	
(Figure 3.1-10)	(Figure 8-13)	
(Figure 3.1-10a)	(Figure 8-14)	
3/4.2	8-32	New figure No. was added.
	8-33 thru 8-36	Revised axial power imbalance envelope for cycle 5 operation.
(Figure 3.2-1)	(Figure 8-15)	
(Figure 3.2-1a)	(Figure 8-16)	
(Figure 3.2-2)	(Figure 8-17)	
(Figure 3.2-2a)	(Figure 8-18)	
(Table 3.3-1)	8-37 (Table 8-3)	Added nuclear overpower based on RCPPMs trip.
(Table 3.3-2)	8-38 (Table 8-4)	Added nuclear overpower based on RCPPMs trip.
(Table 4.3-1)	8-39 (Table 8-5)	Added nuclear overpower based on RCPPMs trip.
3/4.4	8-40	Revised for cycle 5 operation with RCPPMs.
3/4.1.2 BASES	8-41	Shutdown margin and borated water volume requirements were revised for cycle 5 operation.

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

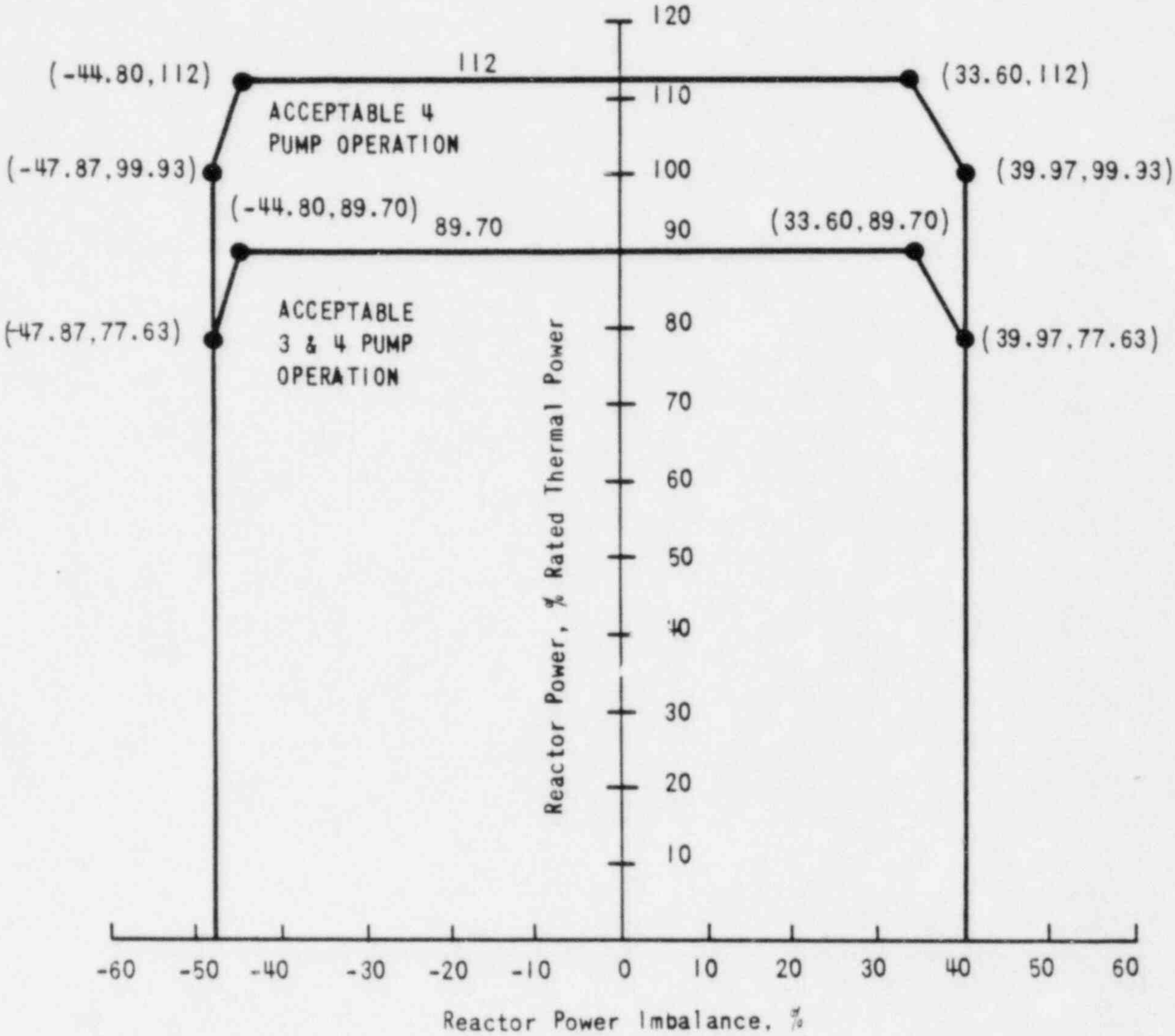


Table 8-2. Reactor Protection System Implementation Trip Setpoints

(Tech Spec Table 2.2-1)

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable	Not applicable
2. Nuclear overpower	≤104.9% of RATED THERMAL POWER with four pumps operating ≤79.92% of RATED THERMAL POWER with three pumps operating[3]	≤104.9% of RATED THERMAL POWER with four pumps operating ≤79.92% of RATED THERMAL POWER with three pumps operating[3]
3. RCS outlet temperature - high	≤618°F	≤618°F
4. Nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE[1]	Trip setpoint not to exceed the limit line of Figure 2.2-1	Allowable values not to exceed the limit line of Figure 2.2-1
5. RCS pressure - low[1]	≥1800 psig	≥1800 psig
6. RCS pressure - high	≤2300 psig	≤2300 psig
7. RCS pressure - variable low[1]	≥(11.59 T_{out} °F - 5037.8) psig	≥(11.59 T_{out} °F - 5037.8) psig
8. Pump status based on reactor coolant pump monitors[1]	More than one pump drawing ≤1075 kw or ≥9000 kw	More than one pump drawing ≤1075 kw or ≥9000 kw

Table 8-2. (Cont'd)

Functional unit	Trip setpoint	Allowable values
9. Reactor containment vessel pressure high	≤4 psig	≤4 psig

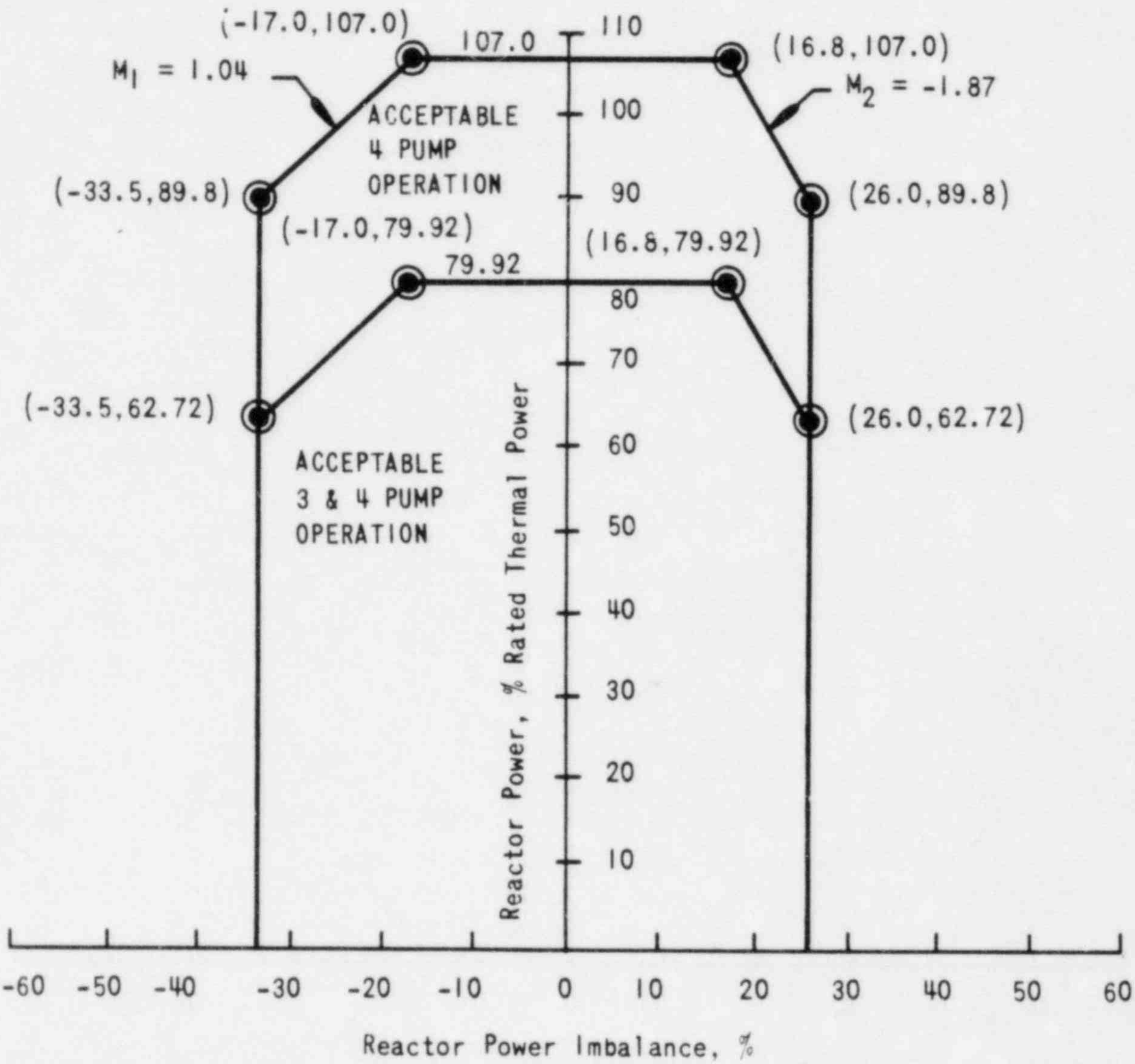
[1]Trip may be manually bypassed when RCS pressure ≤1720 psig by actuating shutdown bypass provided that:

a. The nuclear overpower trip setpoint is ≤5% of RATED THERMAL POWER

b. The shutdown bypass RCS pressure - high trip setpoint of ≤1720 psig is imposed, and

c. The shutdown bypass is removed when RCS pressure >1800 psig.

Figure 8-2. Reactor Trip Setpoints (Tech Spec Figure 2.2-1)



2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

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

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic Reactor Protection System instrumentation channels and provides manual reactor trip capability.

Nuclear Overpower

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

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.9% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip ≤ 618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

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.

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 over-power 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. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is ≥ 107% and reactor flow rate is 100%, or flow rate is ≤ 93.45% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is ≥ 79.92% and reactor flow rate is 74.7%, or flow rate is ≤ 70.09% and power is 75%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.07% for 1% flow reduction.

RCS Pressure - Low, High, and Variable Low

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

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection and is, therefore, set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, (11.59 T_{out} °F - 5037.8) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of (11.59 T_{out} °F - 5077.8) psig.

Reactor Containment Vessel Pressure — High

The Reactor Containment Vessel Pressure-High Trip Setpoint ≤ 4 psig. provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure - Low trip.

Nuclear Overpower Based on RCPPMs

The Reactor Coolant Pump Power Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is ≥ 120% or is ≤ 19.5% of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (typically 7500 kW) to when a pump is operating at full reactor power and is pumping less dense fluid (typically 5500 kW). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000 kW) is based on the nominal operating power for a pump during heatup and the 19.5% trip setpoint (1075 kW) is based on the maximum time within which an RCPPM RPS trip must occur to provide protection for the four pump coastdown. Florida Power has agreed to take credit for the pump overpower trip in order to assure that certain potential faults such as a seismically induced fault high signal will not prevent this instrumentation from providing the protective action (i.e., a trip signal).

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be $\geq$1.0% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN <1.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be $\geq$1.0% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,
   2. Control rod position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2. Each of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the concentrated boric acid storage system via a boric acid pump and makeup or decay heat removal (DHR) pump to the Reactor Coolant System, and

b. A flow path from the borated water storage tank via makeup or DHR pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODES 1, 2, and 3:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path 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 flow path to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

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

MODE 4:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be borated to a SHUTDOWN MARGIN equivalent to 1.0% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.2 At least one makeup pump shall be OPERABLE.

APPLICABILITY: MODE 4*

ACTION:

With no makeup pump OPERABLE, restore at least one makeup pump to OPERABLE status within one hour or be borated to a SHUTDOWN MARGIN equivalent to 1.0% $\Delta k/k$ at 200°F and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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*With RCS pressure $\geq$150 psig.

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODES 1, 2, and 3:

With no boric acid pump OPERABLE, restore at least one boric acid pump 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 at least one boric acid pump to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

MODE 4:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be borated to a SHUTDOWN MARGIN equivalent to 1.0% $\Delta k/k$ at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.7 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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 concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6356 gallons,
2. Between 11,600 and 14,000 ppm of boron, and
3. A minimum solution temperature of 105°F.

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

1. A minimum contained borated water volume of 13,500 gallons.
2. A minimum boron concentration of 2,270 ppm, and
3. A minimum solution temperature of 40°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 contained borated water volume of the tank, and

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6356 gallons,
2. Between 11,600 and 14,000 ppm of boron, and
3. A minimum solution temperature of 105° F.

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

1. A minimum contained borated water volume of 415,200 gallons,
2. Between 2,270 and 2,450 ppm of boron, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

MODES 1, 2, and 3:

a. With the concentrated boric acid storage 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 concentrated boric acid storage system to OPERABLE status within the next 7 days or be in HOT 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 HOT SHUTDOWN within the following 30 hours.

MODE 4:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be

ACTION: (Continued)

borated to a SHUTDOWN MARGIN equivalent to 1.0% $\Delta k/k$ at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 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 contained borated water volume of each water source, and
3. Verifying the concentrated boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when outside air temperature is < 40°F.

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 3.1-1, 3.1-1a, 3.1-2, 3.1-2a, 3.1-3, 3.1-3a, 3.1-4, and 3.1-4a with a rod group overlap of 25 ± 5% between sequential withdrawn groups 5 and 6, and 6 and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits, 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.

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*See Special Test Exceptions 3.10.1 and 3.10.2.

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

Figure 8-3. Regulating Rod Group Insertion Limits for Four Pump Operation From 0 to 30 (+10/-0) EFPD (Tech Spec Figure 3.1-1)

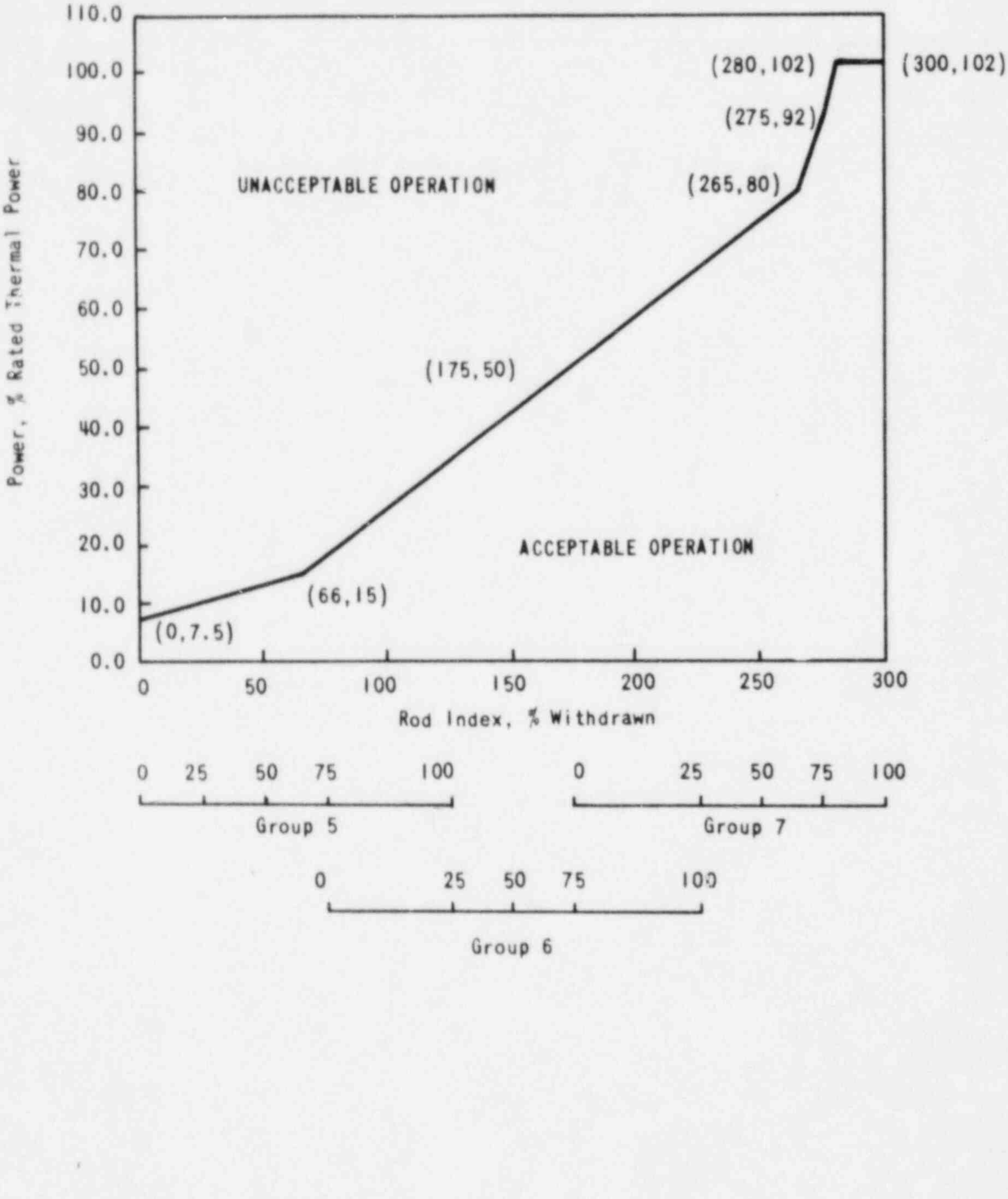


Figure 8-4. Regulating Rod Group Insertion Limits for Four-Pump Operation From 30 (+10/-0) to 250 ±10 EFPD (Tech Spec Figure 3.1-12)

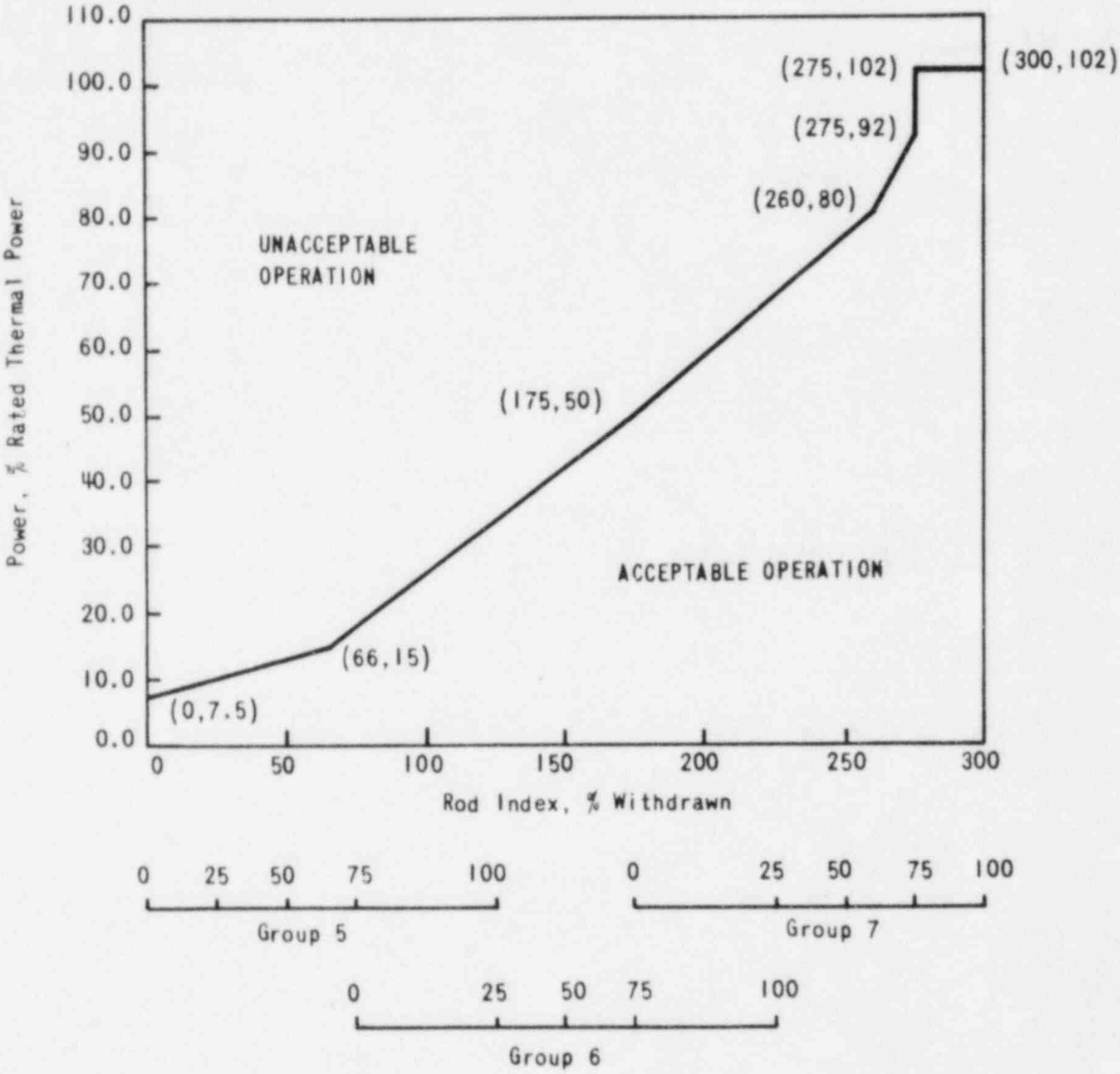


Figure 8-5. Regulating Rod Group Insertion Limits for Four-Pump Operation From 250 ± 10 to 399 ± 10 EFPD (Tech Spec Figure 3.1-2)

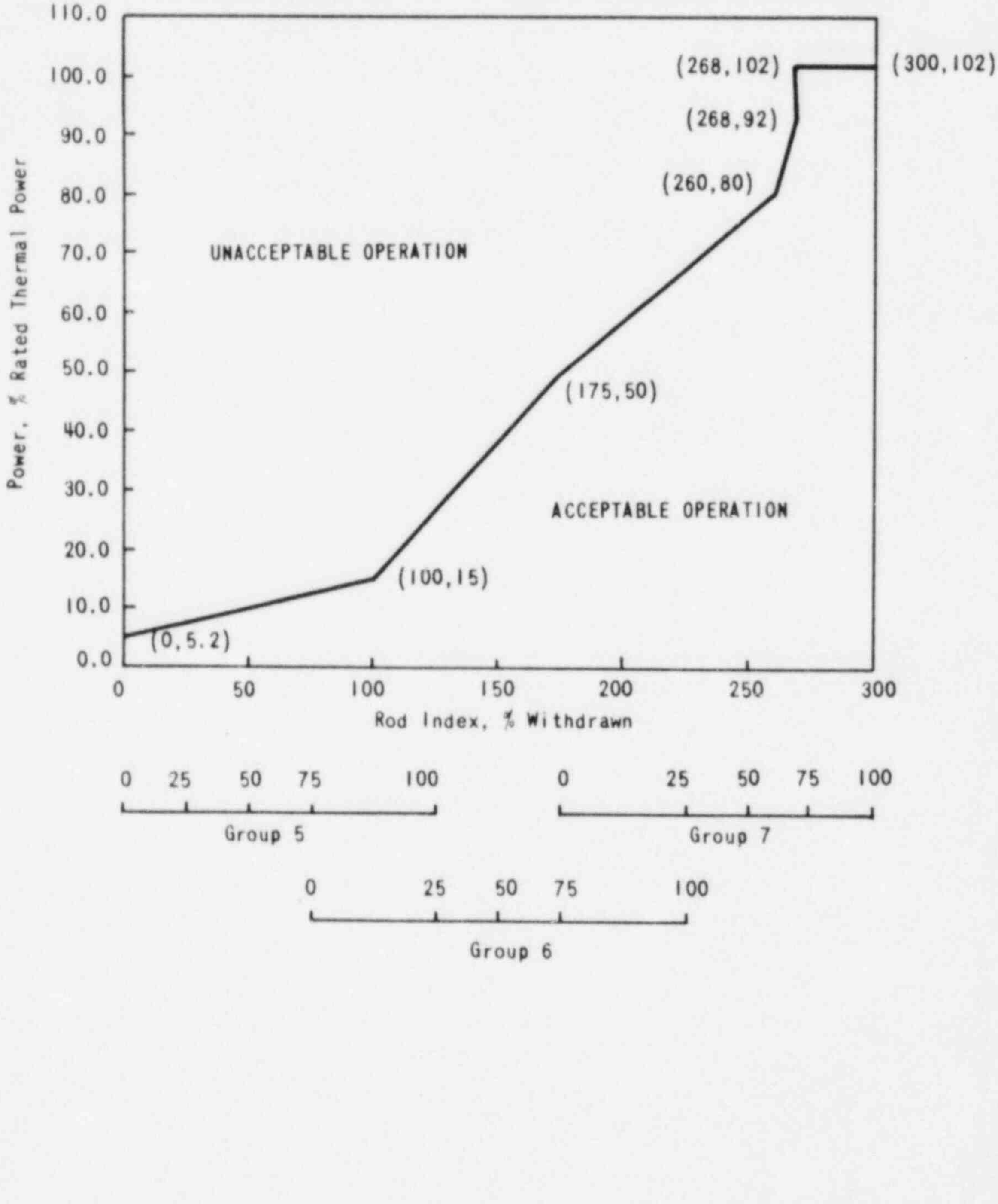


Figure 8-6. Regulating Rod Group Insertion Limits for Four-Pump Operation After 399 ± 10 EFPD (Tech Spec Figure 3.1-2a)

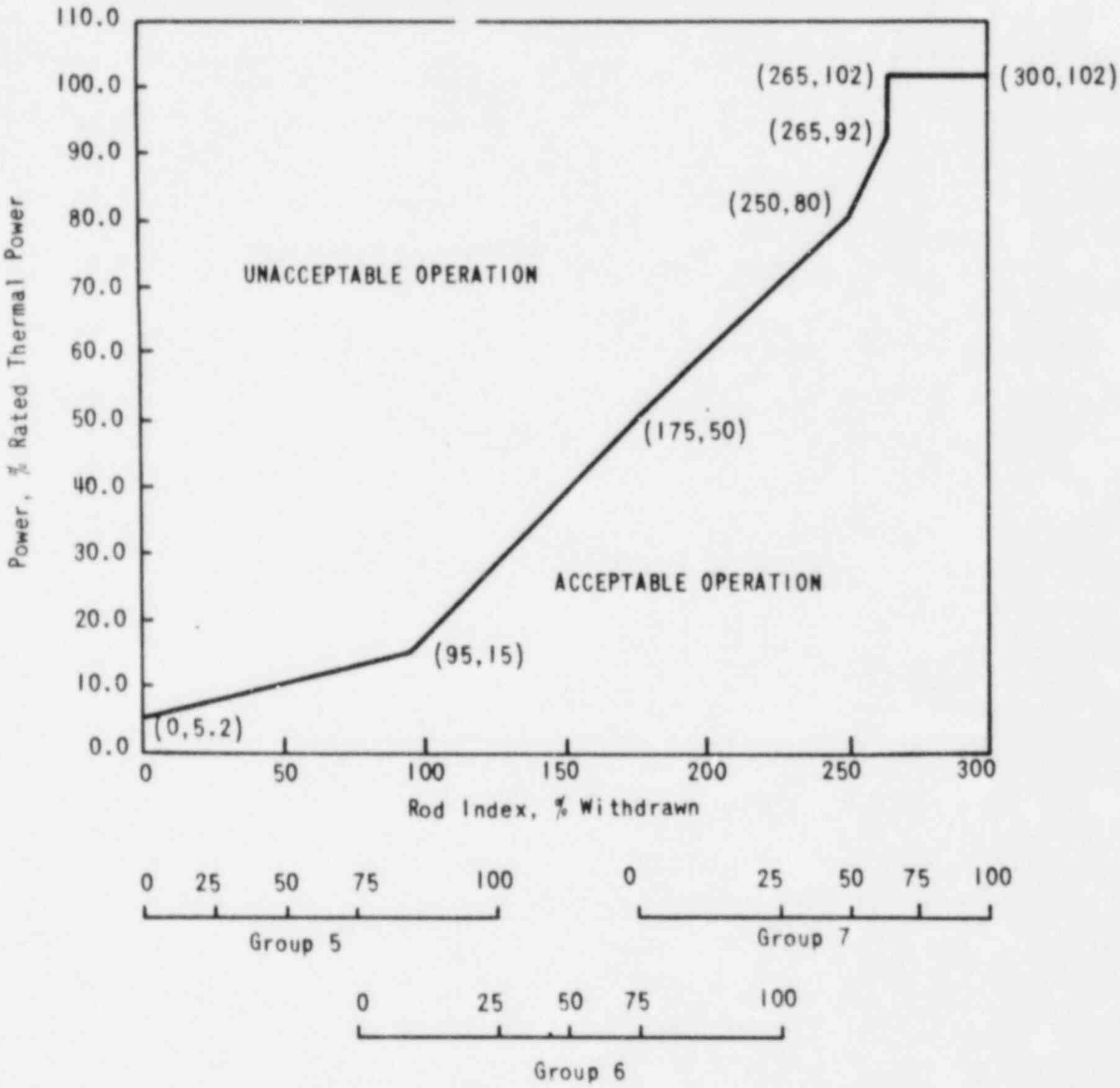


Figure 8-7. Regulating Rod Group Insertion Limits for Three-Pump Operation From 0 to 30 (+10/-0) EFPD (Tech Spec Figure 3.1-3)

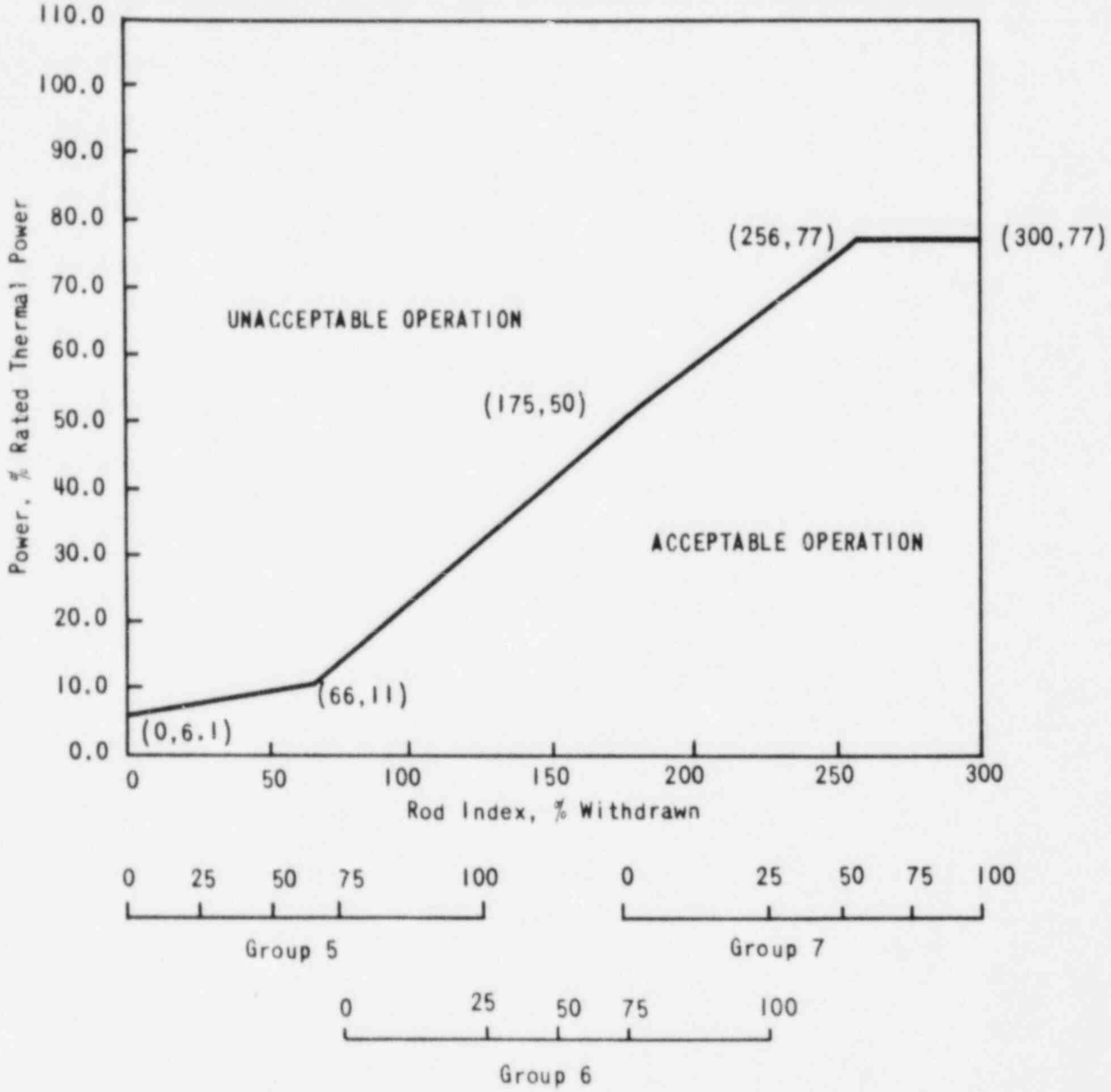


Figure 8-8. Regulating Rod Group Insertion Limits for Three-Pump Operation From 30 (+10/-0) to 250 ± 10 EFPD (Tech Spec Figure 3.1-3a)

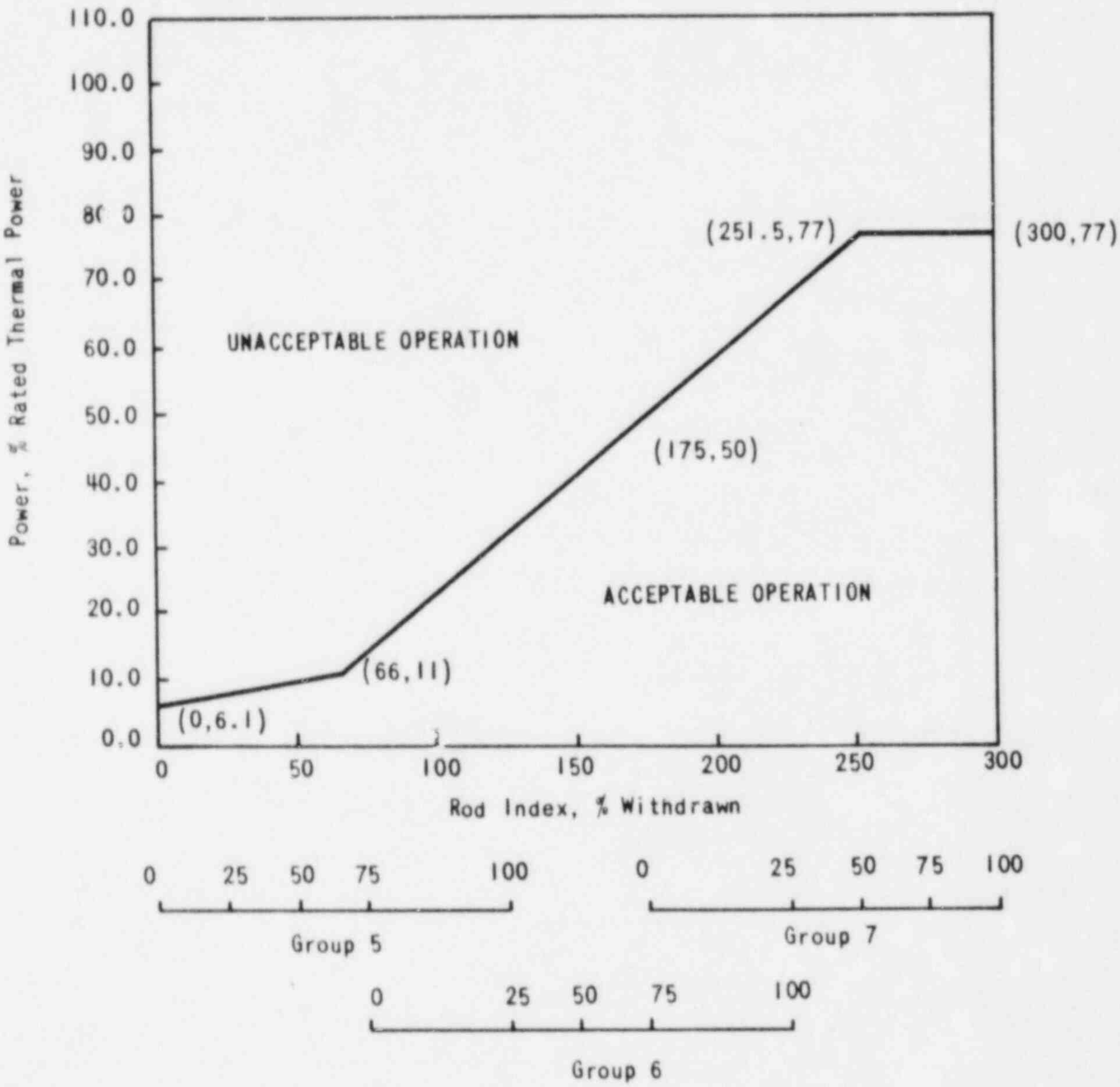


Figure 8-9. Regulating Rod Group Insertion Limits for Three-Pump Operation From 250 ± 10 to 399 ± 10 EFPD (Tech Spec Figure 3.1-4)

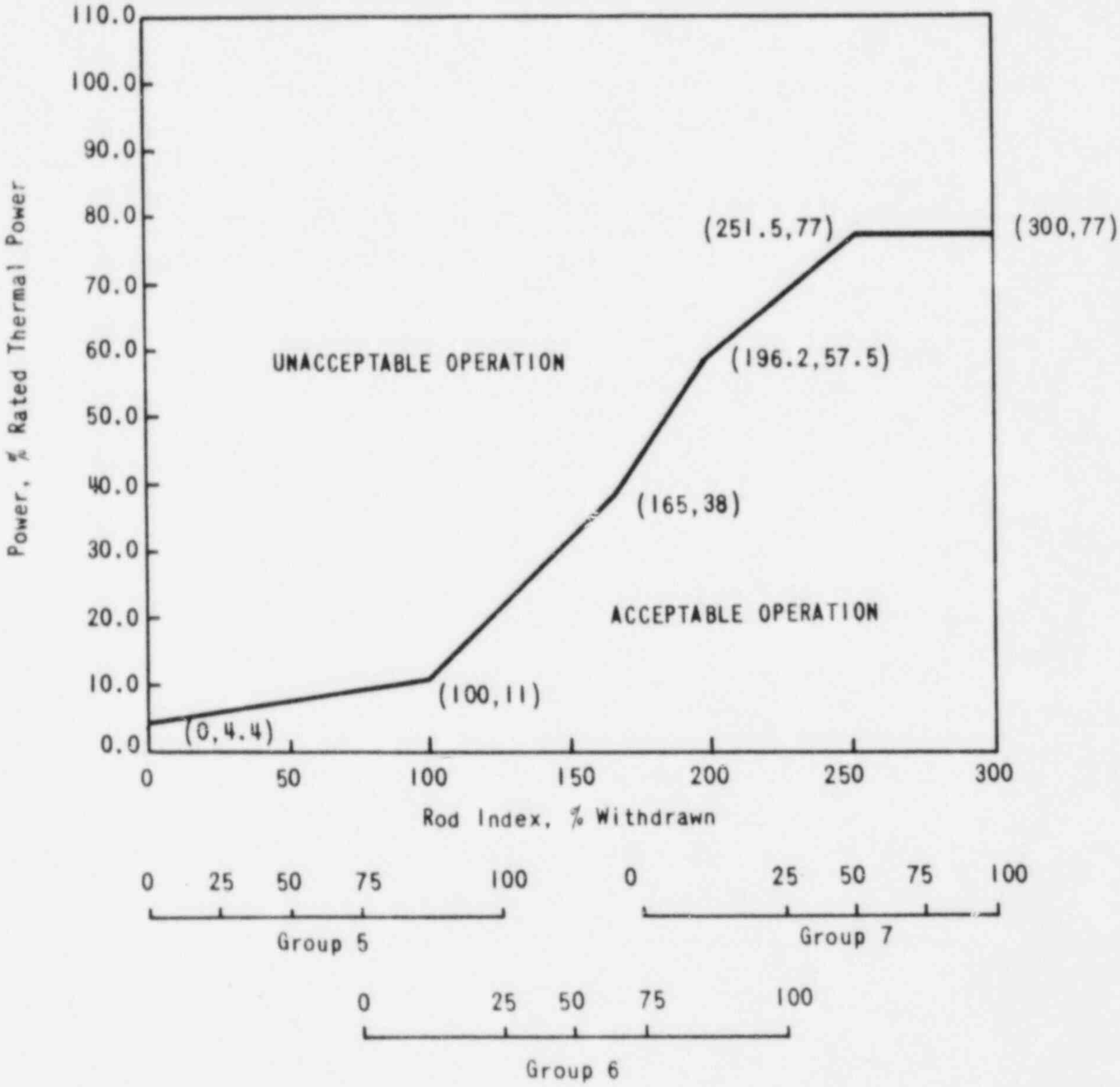
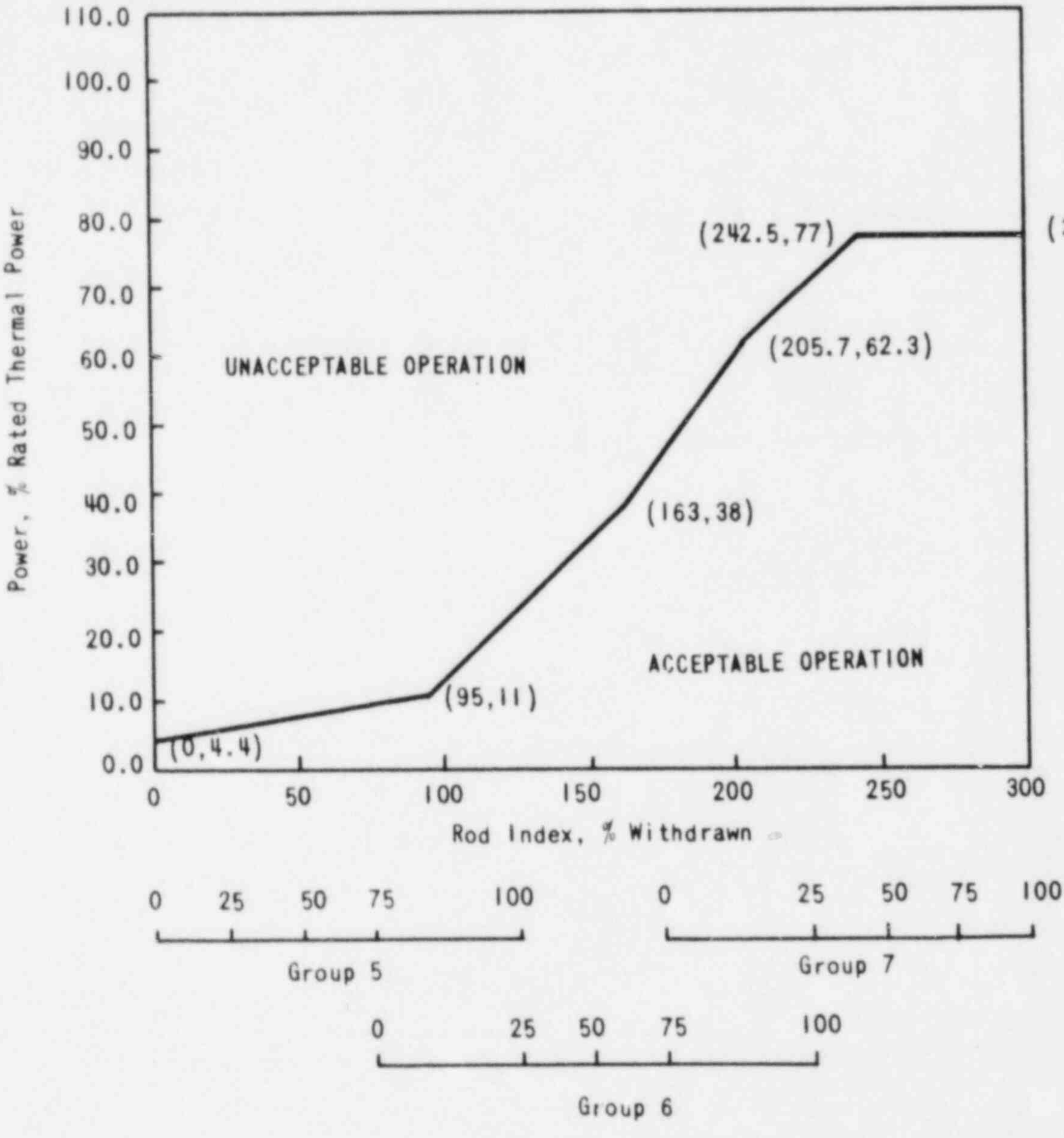


Figure 8-10. Regulating Rod Group Insertion Limits for Three-Pump Operation After 399 ± 10 EFPD (Tech Spec Figure 3.1-4a)



REACTIVITY CONTROL SYSTEMS

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 3.1-9, 3.1.9a, 3.1-10, and 3.1-10a.

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 figure 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.

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

Figure 8-11. APSR Position Limits for 0 to 30 (+10/-0) EFPD
(Tech Spec Figure 3.1-9)

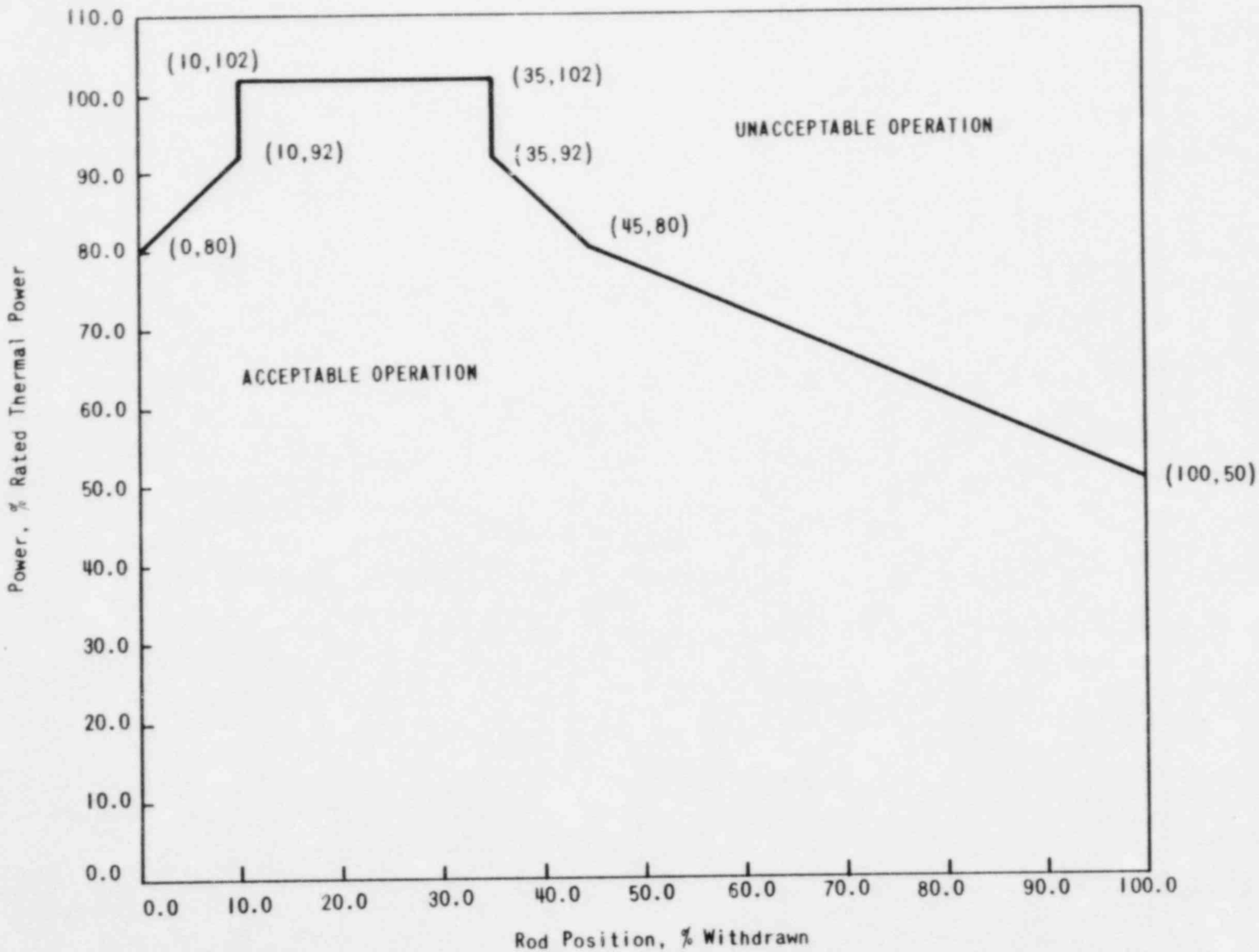


Figure 8-12. APSR Position Limits for 30 (+10/-0) to 250 ± 10 EFPD
(Tech Spec Figure 3.1-9a)

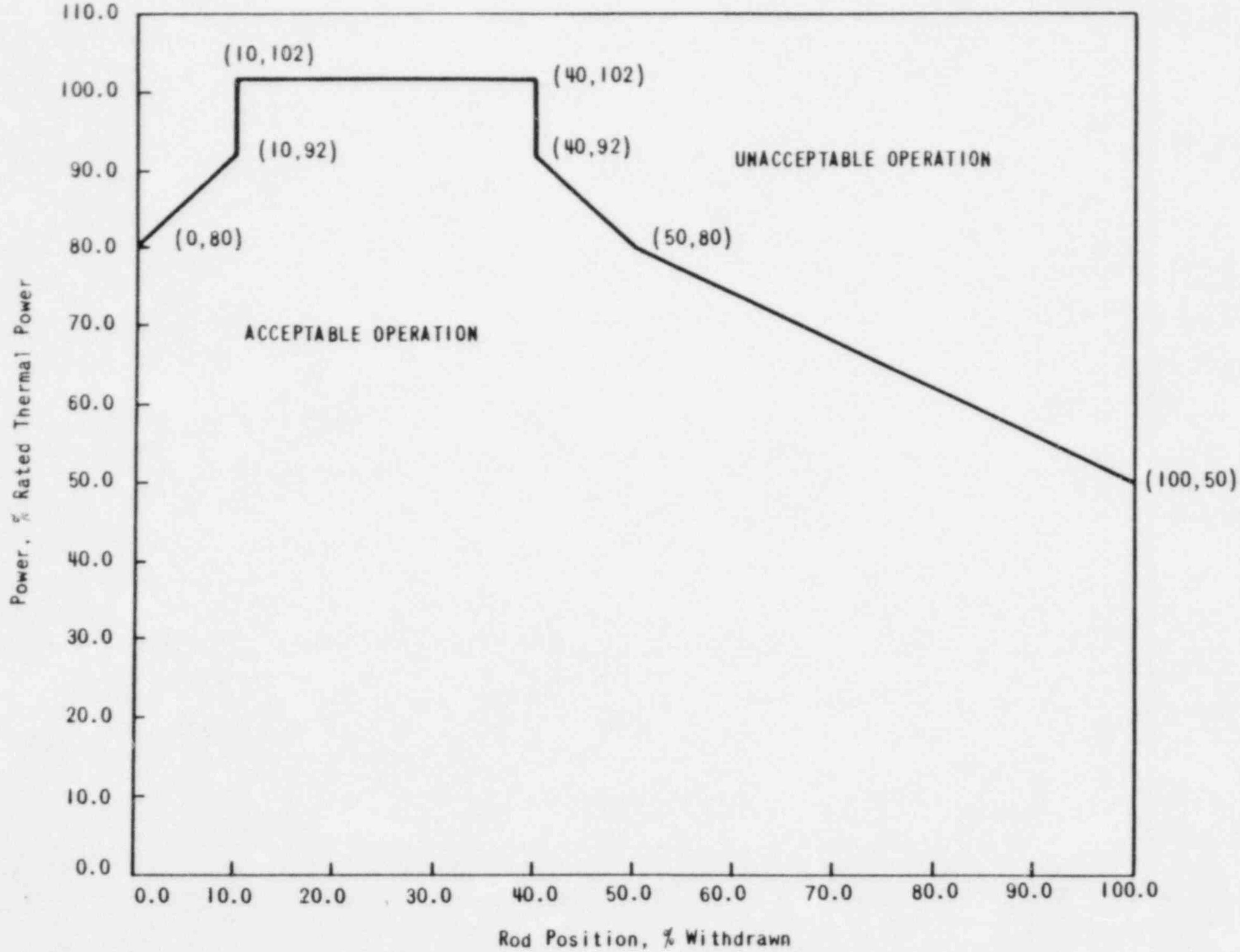


Figure 8-13. APSR Position Limits for 250 ± 10 to 399 ± 10 EFPD
(Tech Spec Figure 3.1-10)

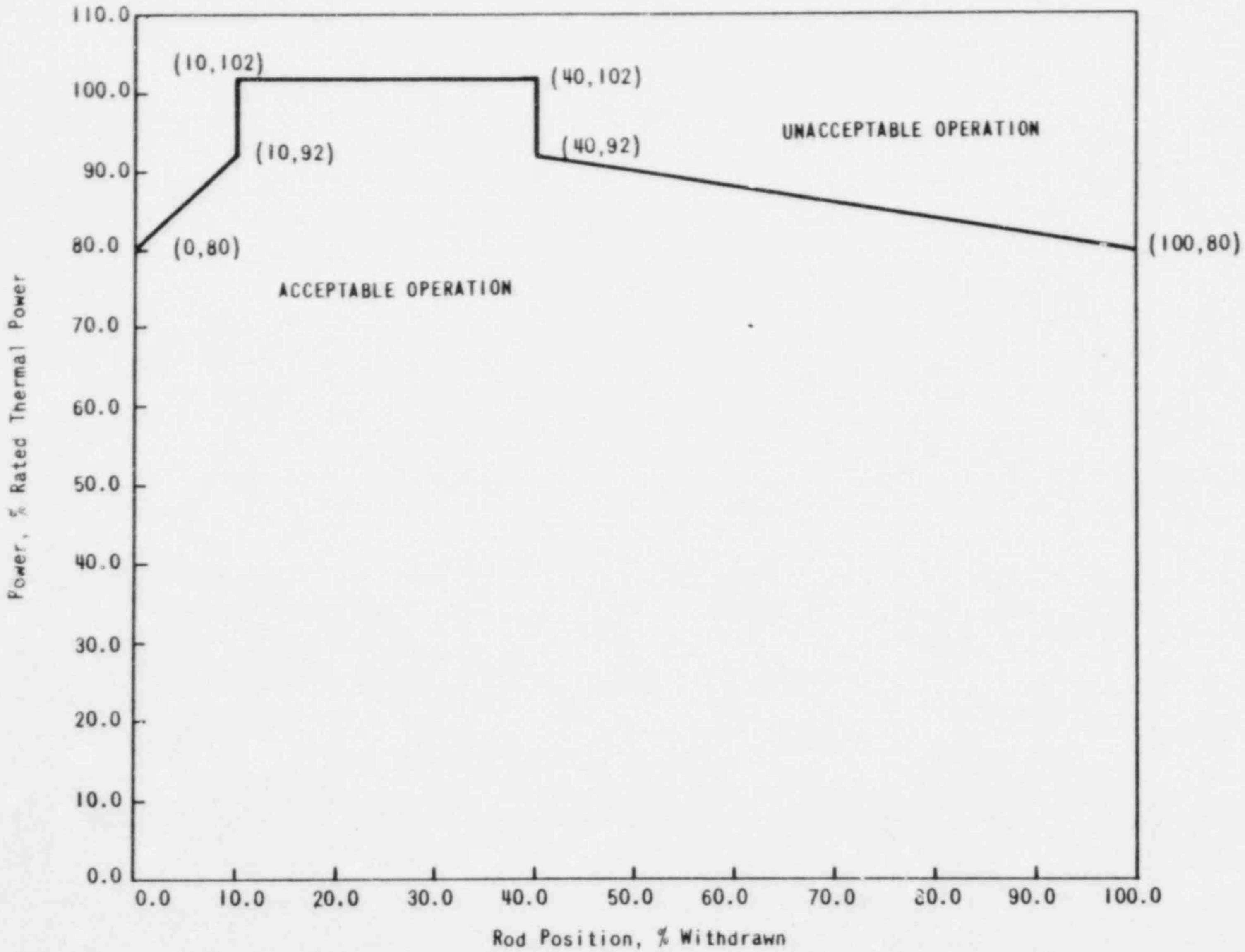
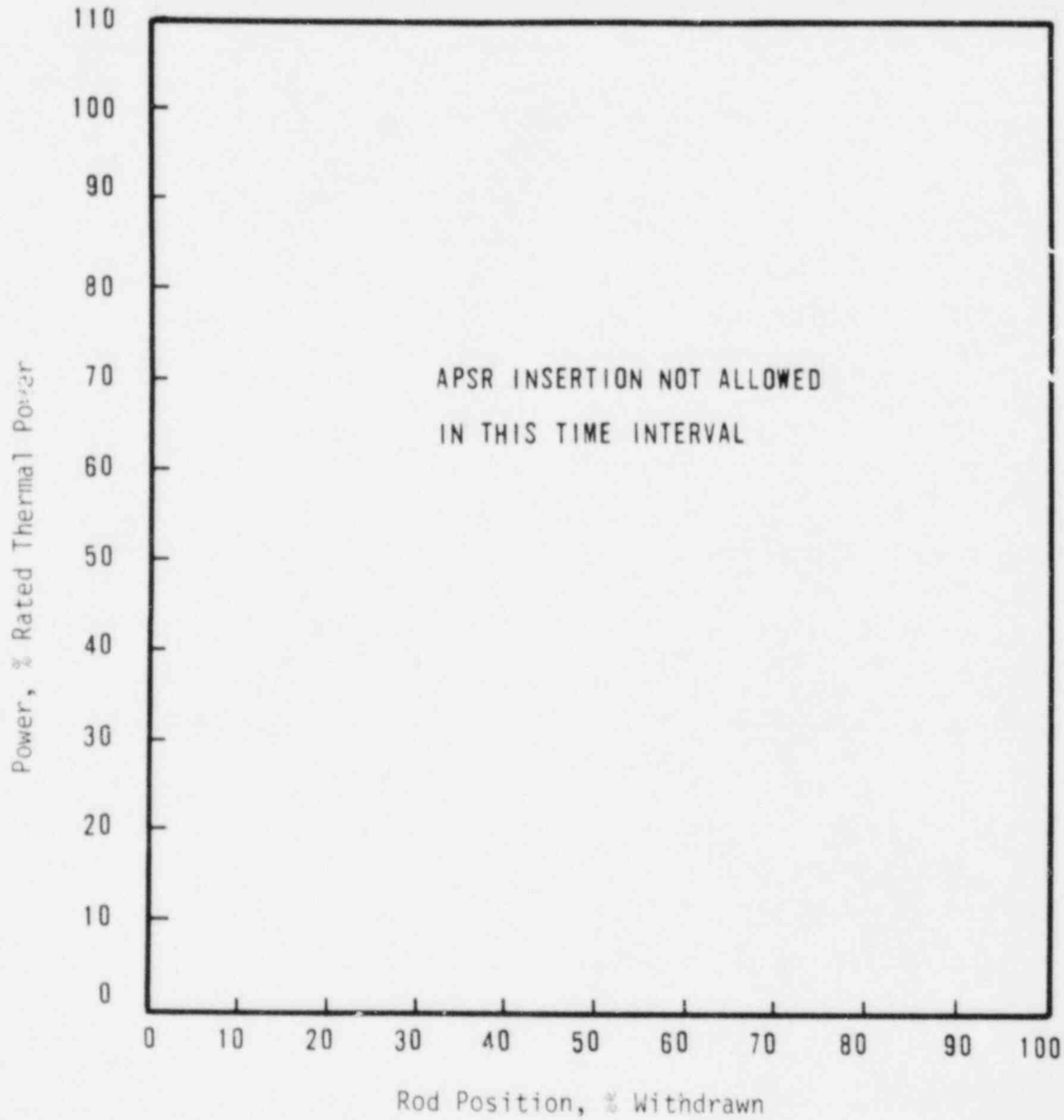


Figure 8-14. APSR Position Limits for Operation After 399 ± 10 EFPD (Tech Spec Figure 3.1-10a)



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, 3.2-1a, 3.2-2, and 3.2-2a.

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 with 15 minutes, or

b. Be in at least HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.1.

Figure 8-15. Axial Power Imbalance Envelope for Operation From 0 to 30 (+10/-0) EFPD (Tech Figure 3.2-1)

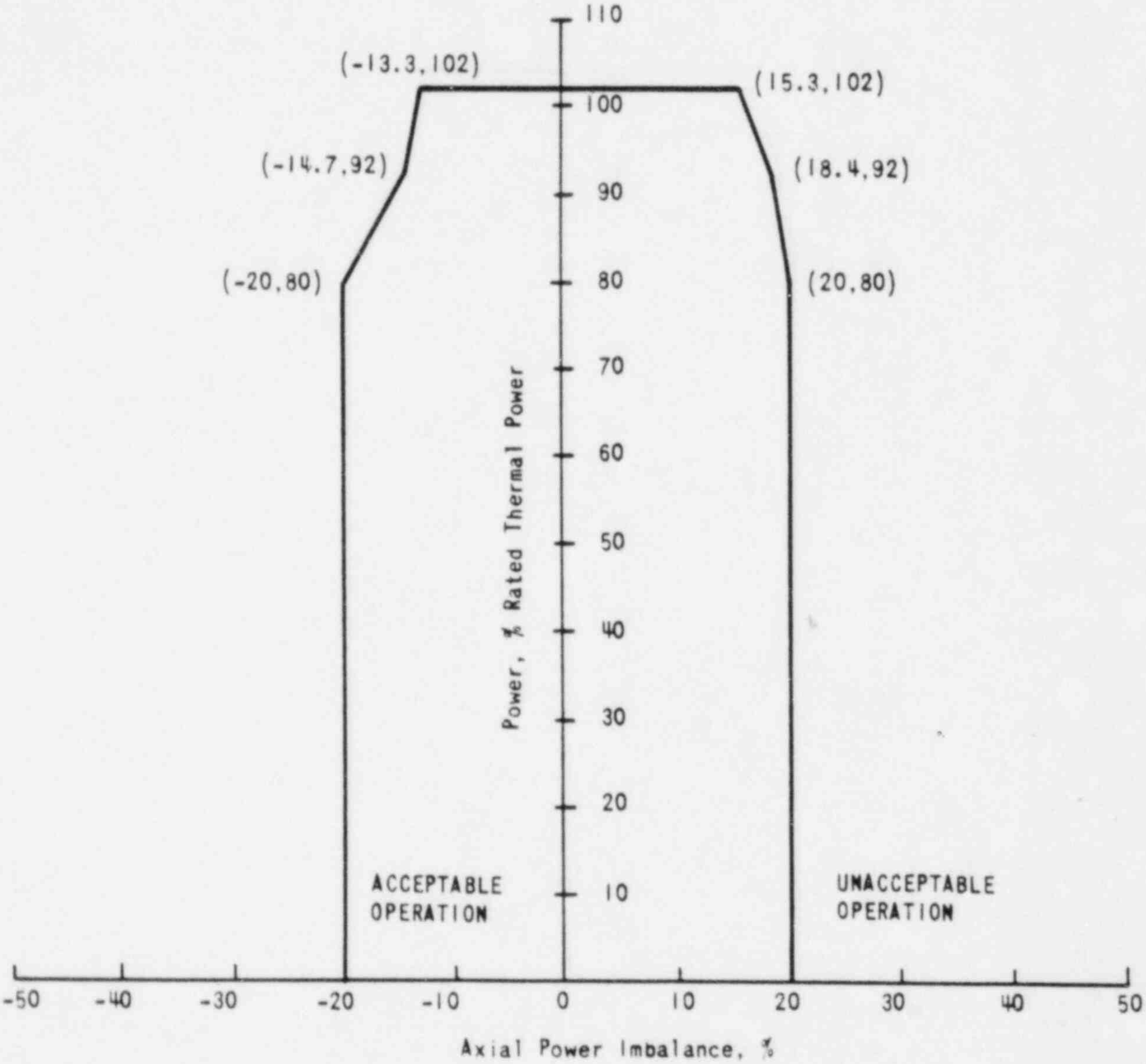


Figure 8-16. Axial Power Imbalance Envelope for Operation From 30 (+10/-0) to 250 ± 10 EFPD (Tech Spec Figure 3.2-1a)

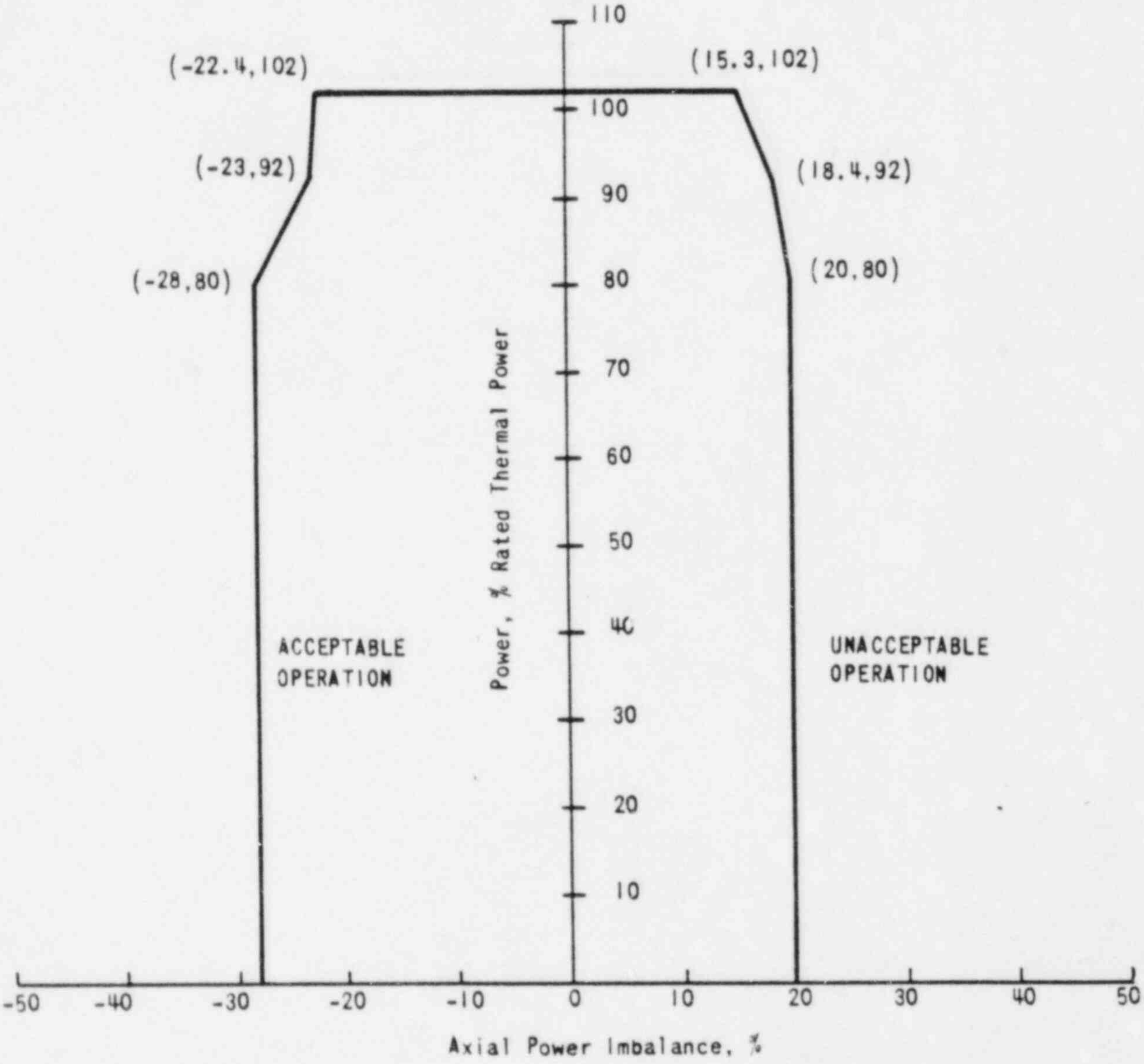


Figure 8-17. Axial Power Imbalance Envelope for Operation From 250 ± 10 to 399 ± 10 EFPD (Tech Spec Figure 3.2-2)

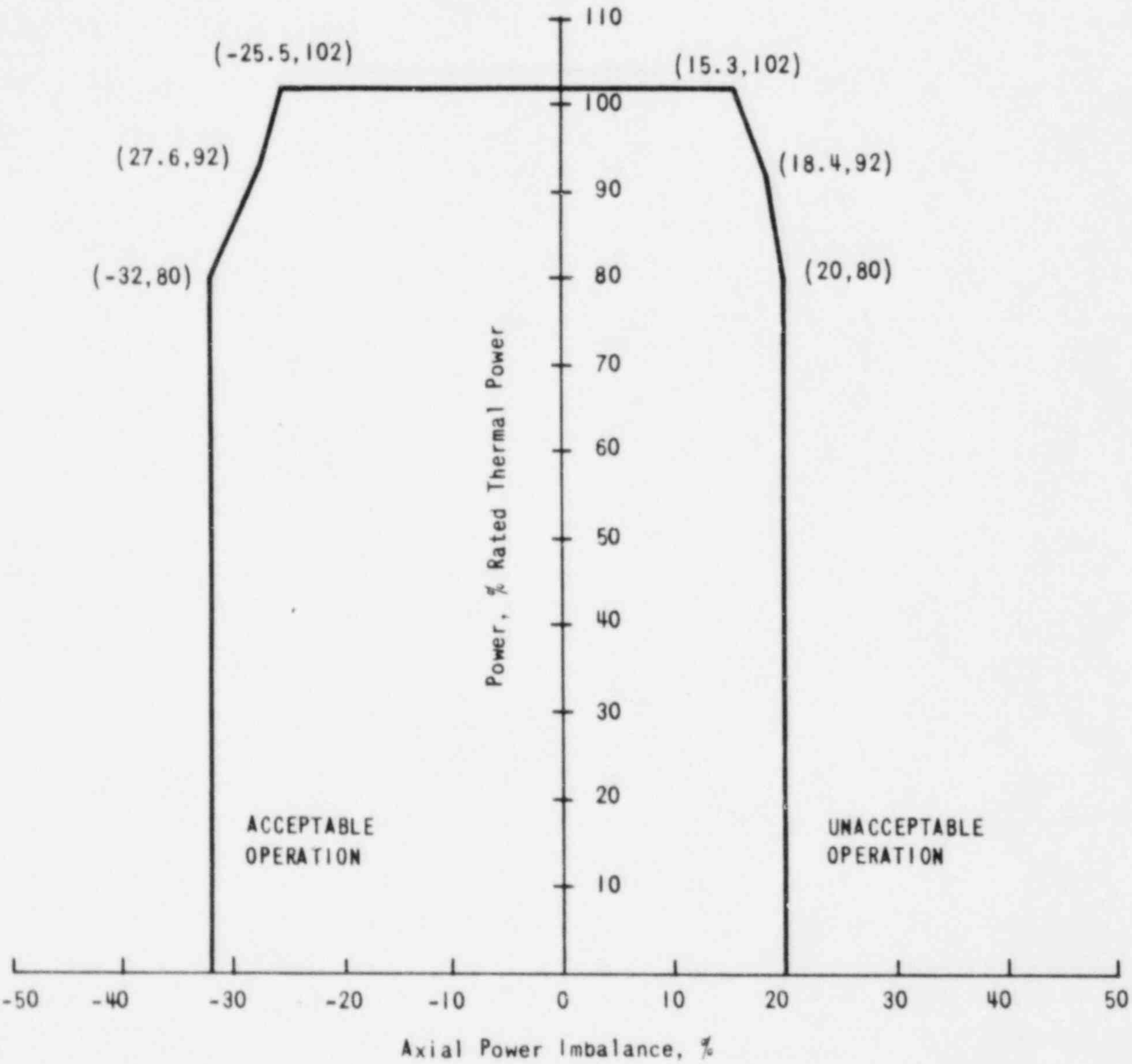


Figure 8-18. Axial Power Imbalance Envelope for Operation After 399 ± 10 EFPD (Tech Spec Figure 3.2-2a)

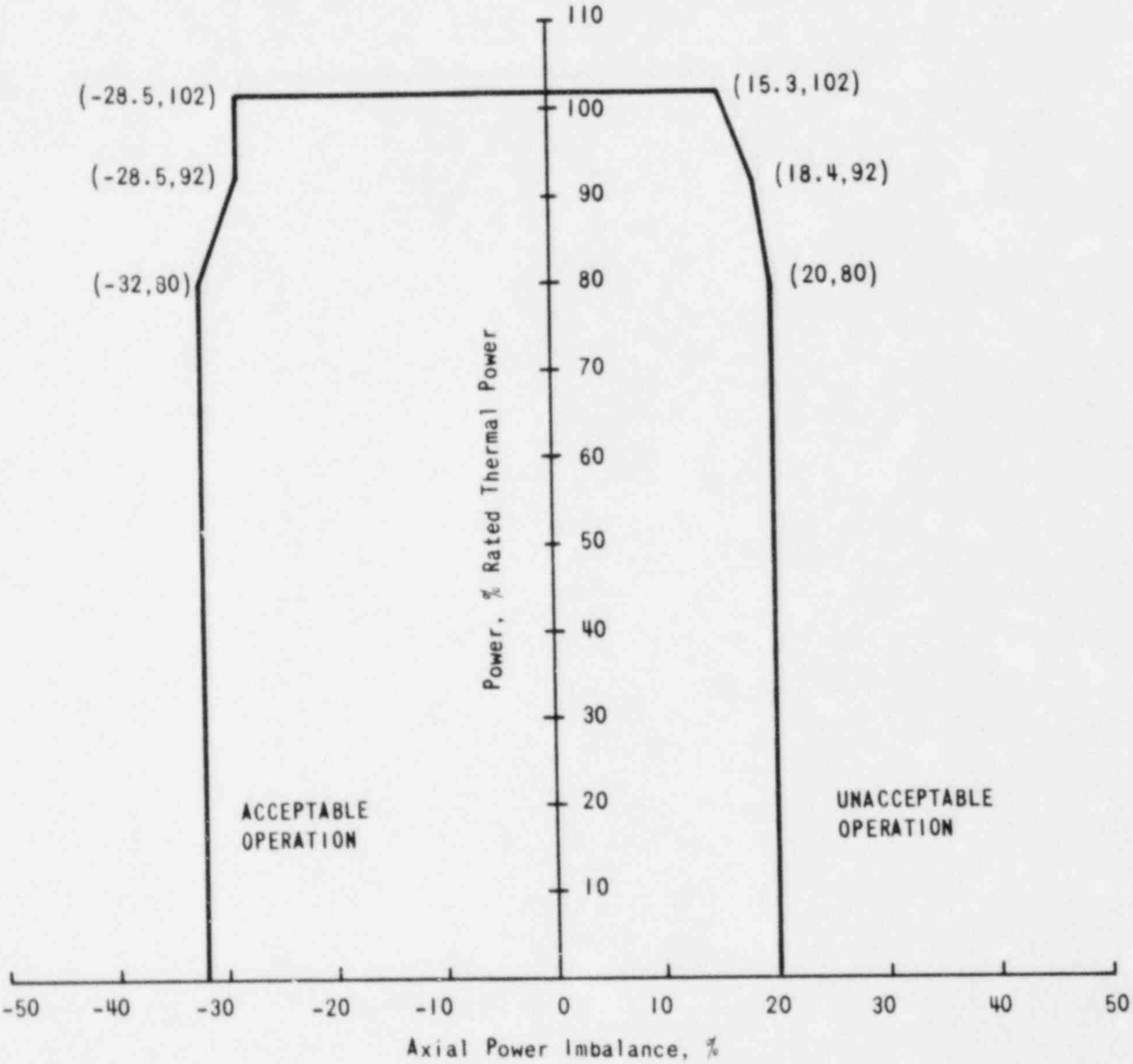


Table 8-3. Reactor Protection System Instrumentation

(Tech Spec Table 3.3-1)

	Functional unit	Total No. of channels	Channels to trip	Minimum channels operable	Applicable modes	Action
1.	Manual reactor trip	1	1	1	1, 2, and *	8
2.	Nuclear overpower	4	2	3	1, 2	2#
3.	RCS outlet temperature - high	4	2	3	1, 2	3#
4.	Nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE	4	2(a)	3	1, 2	2#
5.	RCS pressure - low	4	2(a)	3	1, 2	3#
6.	RCS pressure - high	4	2	3	1, 2	3#
7.	Variable low RCS pressure	4	2(a)	3	1, 2	3#
8.	Reactor containment pressure - high	4	2	3	1, 2	3#
9.	Intermediate range, neutron flux and rate	2	0	2	1, 2, and *	4
10.	Source range, neutron flux and rate					
	a. Startup	2	0	2	2## and *	5
	b. Shutdown	2	0	1	3, 4, and 5	6
11.	Control rod drive trip breakers	2/trip system	1/trip system	1/trip system	1, 2, and *	7#
12.	Reactor trip module	2/trip system	1/trip system	2/trip system	1, 2, and *	7#
13.	Shutdown bypass RCS pressure - high	4	2	3	2**, 3**, 4**, 5**	6#
14.	Nuclear Overpower based on RCPPMs	4	2(a)(b)	3	1, 2	3#

Table 8-4. Reactor Protection System Instrumentation Response Times

(Tech Spec Table 3.3-2)

Functional Unit	Response Times
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	≤0.266 seconds
3. RCS Outlet Temperature - High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	≤1.79 seconds
5. RCS Pressure - Low	≤0.44 seconds
6. RCS Pressure - High	≤0.44 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Pump Status Based on RCPPMs*	≤0.56 seconds
9. Reactor Containment Pressure - High	Not applicable

- - - - -

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

Table 8-5. 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	NA	NA	S/U(1)	NA
2. Nuclear overpower	S	D(2) and Q(7)	M	1, 2
3. RCS outlet temperature - high	S	R	M	1, 2
4. Nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7,8)	M	1, 2
5. RCS pressure - low	S	R	M	1, 2
6. RCS pressure - high	S	R	M	1, 2
7. Variable low RCS pressure	S	R	M	1, 2
8. Reactor containment pressure - high	S	R	M	1, 2
9. Intermediate range, neutron flux and rate	S	R(7)	S/U(1)(5)	1, 2, and *
10. Source range, neutron flux and rate	S	R(7)	S/U(1)(5)	2, 3, 4, and 5
11. Control rod drive trip breaker	NA	NA	M and S/U(1)	1, 2, and *
12. Reactor trip module	NA	NA	M	1, 2, and *
13. Shutdown bypass RCS pressure - high	S	R	M	2**, 3**, 4**, 5**
14. Nuclear overpower based on RCPPMs	S	R	M	1, 2

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 79.92% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

    1. Nuclear Overpower

MODES 3, 4, and 5:

a. Operation may proceed provided at least one reactor coolant loop is operation with an associated reactor coolant pump or decay heat removal pump.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

- - - - -

*See Special Test Exception 3.10.3

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 that (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 either 6356 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 43,478 gallons of 2,270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 415,200 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 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 1.0% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 300 gallons of 11,600 ppm boron from the boric acid storage system or 1,608 gallons of 2,270 ppm boron from the borated water storage tank. To envelop future cycle BWST contained borated water volume requirements, a minimum volume of 13,500 gallons is specified.

9. STARTUP PROGRAM — PHYSICS TESTING

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

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Acceptance criteria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.66 seconds at these conditions.

It should be noted that Safety Analysis calculations are based on a rod drop time of 1.40 seconds from fully withdrawn to two-thirds inserted. Since the most accurate position indication is obtained from the zone reference switch at the 75%-inserted position, this position is used for data gathering instead of the two-thirds inserted position. The Acceptance Criterion of 1.40 seconds corrected to a 75%-inserted position (by rod insertion versus time correlation) is 1.66 seconds.

9.1.2. RC Flow

RC flow with four RC pumps running will be measured at hot zero power, steady-state conditions. The Acceptance Criteria require that the measured flow be within allowable limits.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate. Once criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by

correcting for any rod withdrawal required in achieving equilibrium boron. The Acceptance Criterion placed on critical boron concentration is that the actual boron concentration must be within ±100 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at approximately the all-rods-out configuration and at the hot zero power rod insertion limit. The average coolant temperature is varied by first decreasing and then increasing temperature by 5°F. During the changes in temperature, reactivity feedback is compensated by discrete changes in rod motion; the change in reactivity is then calculated by the summation of the reactivity (obtained from a reactivity calculation on a strip chart recorder) associated with the temperature change. The Acceptance Criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-4}$ $\Delta k/k$°F (predicted value obtained from Physics Test Manual curves).

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurements. After the temperature coefficient has been measured, a predicted value of isothermal fuel Doppler coefficient of reactivity (-2.0×10^{-5} $\Delta k/k/$°F) is subtracted to obtain the moderator coefficient. This value must not be in excess of the Acceptance Criteria limit of $+0.9 \times 10^{-4}$ $(\Delta k/k)/$°F.

9.2.3. Control Rod Group Reactivity Worth

Control bank group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. This method consists of establishing a deboration rate in the RC system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6, and 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on Reactimeter data, and differential rod worths are obtained from the measured reactivity worth versus the change in rod group position. The differential rod worths of each of the controlling groups are then summed to obtain integral rod group worths. The Acceptance Criteria for the control bank group worths are as follows:

1. Individual bank 5, 6, 7 worth:

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

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

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

9.2.4. Ejected Control Rod Reactivity Worth

After CRA groups 7 and 6 have been positioned near the minimum rod insertion limit and CRA group 5 is between 0 and 10% withdrawn, the ejected rod is borated to 100% withdrawn and the worth obtained by adding the incremental changes in reactivity by boration.

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

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

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

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

9.3. Power Escalation Tests

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

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

The following Acceptance Criteria are placed on the 40% FP test:

1. The worst-case maximum LHR must be less than the LOCA limit.

2. The minimum DNBR must be greater than 1.30.

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

4. The value obtained from the extrapolation of the worst-case maximum LHR to the next power plateau overpower trip setpoint must be less than the fuel melt limit, or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.

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

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

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

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

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

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

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

The power distribution tests p rformed at 75 and 100% FP are identical to the 40% FP test except that core equilibrium xenon is established prior to the 75 and 100% FP tests. Accordingly, the 75 and 100% FP measured peak acceptance criteria are as follows:

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

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

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

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

9.3.2. Incore Versus Excore Detector Imbalance Correlation Verification at ~40% FP

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

9.3.3. Temperature Reactivity Coefficient at ~100% FP

The average reactor coolant temperature is decreased and then increased by about 5°F at constant reactor power. The reactivity asociated 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.

9.3.4. Power Doppler Reactivity Coefficient at ~100% FP

Reactor power is decreased and then increased by about 5% FP. The reactivity change is obtained from the change in controlling rod group position. Control rod group worth is measured using the fast insert/withdraw method. Reactivity corrections are made for changes in xenon and reactor coolant temperature that occur during the measurement. The power Doppler reactivity coefficient is calculated from the measured reactivity change, adjusted as stated above, and the measured power change.

The predicted value of the power Doppler reactivity coefficient is given in the Physics Test Manual. Acceptance Criteria state that the measured value shall be more negative than -0.55×10^{-4} $\Delta k/k)/\%$ FP.

9.4. Procedure for Failure to Meet Acceptance Criteria

Florida Power Corporation reviews the results of all startup tests to ensure that all Acceptance Criteria are met. If the review of the test indicates that the results are well within the Acceptance Criteria, no further evaluation is conducted. If the review indicates that the results are approaching or close to the Acceptance Criteria limits, further evaluation of that particular test or other supporting tests is performed to look for trends. This evaluation will determine whether additional support data are required to discover any abnormal conditions. 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 is performed by site test personnel with participation by B&W 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, adding tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until the evaluations show that plant safety will not be compromised by such escalation.

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