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ARKANSAS NUCLEAR ONE, UNIT 1

- Cycle 7 Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the seventh cycle of Arkansas Nuclear One, Unit 1 (ANO-1) at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975.

To support cycle 7 operation of ANO-1, this report employs analytical techniques and design bases established in reports that have been submitted to and accepted by the USNRC and its predecessor, the USAEC (see references).

The cycle 6 and 7 reactor parameters related to power capability are summarized briefly in section 5 of this report. All of the accidents analyzed in the FSAR[1] have been reviewed for cycle 7 operation. In those cases where cycle 7 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 7 operation 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 Systems, it has been concluded that ANO-1 can be operated safely for cycle 7 at a rated power level of 2568 MWt.

The cycle 7 core for ANO-1 will contain four twice-burned lead test assemblies (LTAs). These assemblies are part of a Department of Energy Extended Burnup Test Program. The LTA design is described in reference 2.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Arkansas Nuclear One, Unit 1 is the currently operating cycle 6. This cycle 7 design is based on a design cycle 6 length of 400 effective full power days (EFPD).

No anomalies occurred during cycle 6 that would adversely affect fuel performance during cycle 7.

3. GENERAL DESCRIPTION

The ANO-1 reactor core is described in detail in section 3 of the Arkansas Nuclear One, Unit 1, Final Safety Analysis Report (FSAR).

The cycle 7 core contains 177 fuel assemblies, 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 fuel is comprised of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg of uranium, with the exception of four batch 7B LTAs, which have a nominal loading of 440.0 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Tables 4-1 and 4-2 for all fuel assemblies except the LTAs; the corresponding parameters for the LTAs are included in reference 2.

Figure 3-1 is the fuel shuffle diagram for ANO-1, cycle 7. The initial enrichments of batches 7B, 8 and 9 are 2.95, 3.21, and 3.30 wt % ^{235}U, respectively. All the batch 6C assemblies and 31 of the twice-burned batch 7 assemblies will be discharged at the end of cycle 6. The remaining 37 twice-burned batch 7 assemblies (designated batch 7B) will be shuffled to new locations, with 12 on the core peripnery. Sixty of the 72 once-burned batch 8 assemblies will be shuffled to new locations on or near the core periphery. The remaining 12 will surround the center assembly. The 68 fresh batch 9 assemblies will be loaded in a symmetric checkerboard pattern throughout the core. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 7.

Reactivity is controlled by 61 full-length Ag-In-Cd control rods, 64 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle 7 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the total pattern (69 control rods) of cycle 7 are identical to those of the reference cycle indicated in the reload report for ANO-1, cycle 6.[3] There is a minor difference in the group designations between cycle 7 and the reference cycle. The cycle 7 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Core Loading Diagram for ANO-1, Cycle 7

FUEL TRANSFER CANAL →

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						L03 8	M04 8	F 9	M12 8	L13 8					
B				A07 7B	K02 8	M02 8	F 9	M08 8	F 9	M14 8	K14 8	A09 7B			
C			H15 7B	F 9	N03 8	F 9	K04 8	F 9	K12 8	F 9	N13 8	F 9	R08 7B		
D		G01 7B	F 9	L09 8	F 9	A06 7B	F 9	N02 7B	F 9	A10 7B	F 9	K06 8	F 9	G15 7B	
E		B09 8	C12 8	F 9	K10 8	F 9	E07 7B	F 9	E09 7B	F 9	L07 8	F 9	C04 8	B07 8	
F	C10 8	B11 8	F 9	F01 7B	F 9	C03 7B	F 9	M06 8	F 9	C13 7B	F 9	F15 7B	F 9	B05 8	C06 8
G	D11 8	F 9	D09 8	F 9	G05 * 7B	F 9	M10 8	Ø08 8	L05 8	F 9	G11 * 7B	F 9	D07 8	F 9	D05 8
W — H	F 9	H11 8	F 9	P12 7B	F 9	L11 8	H13 8	N14 7B	H03 8	F05 8	F 9	B04 7B	F 9	H05 8	F 9 — Y
K	N11 8	F 9	N09 8	F 9	K05 * 7B	F 9	F11 8	C08 8	E06 8	F 9	K11 * 7B	F 9	N07 8	F 9	N05 8
L	Ø10 8	P11 8	F 9	L01 7B	F 9	Ø03 7B	F 9	E10 8	F 9	Ø13 7B	F 9	L15 7B	F 9	P05 8	Ø06 8
M		P09 8	Ø12 8	F 9	F09 8	F 9	M07 7B	F 9	M09 7B	F 9	G06 8	F 9	Ø04 8	P07 8	
N		K01 7B	F 9	G10 8	F 9	R06 7B	F 9	D14 7B	F 9	R10 7B	F 9	F07 8	F 9	K15 7B	
O			A08 7B	F 9	D03 8	F 9	G04 8	F 9	G12 8	F 9	D13 8	F 9	H01 7B		
P				R07 7B	G02 8	E02 8	F 9	E08 8	F 9	E14 8	G14 8	R09 7B			
R						F03 8	E04 8	F 9	E12 8	F13 8					

Z

Cycle 6 Location
Batch ID

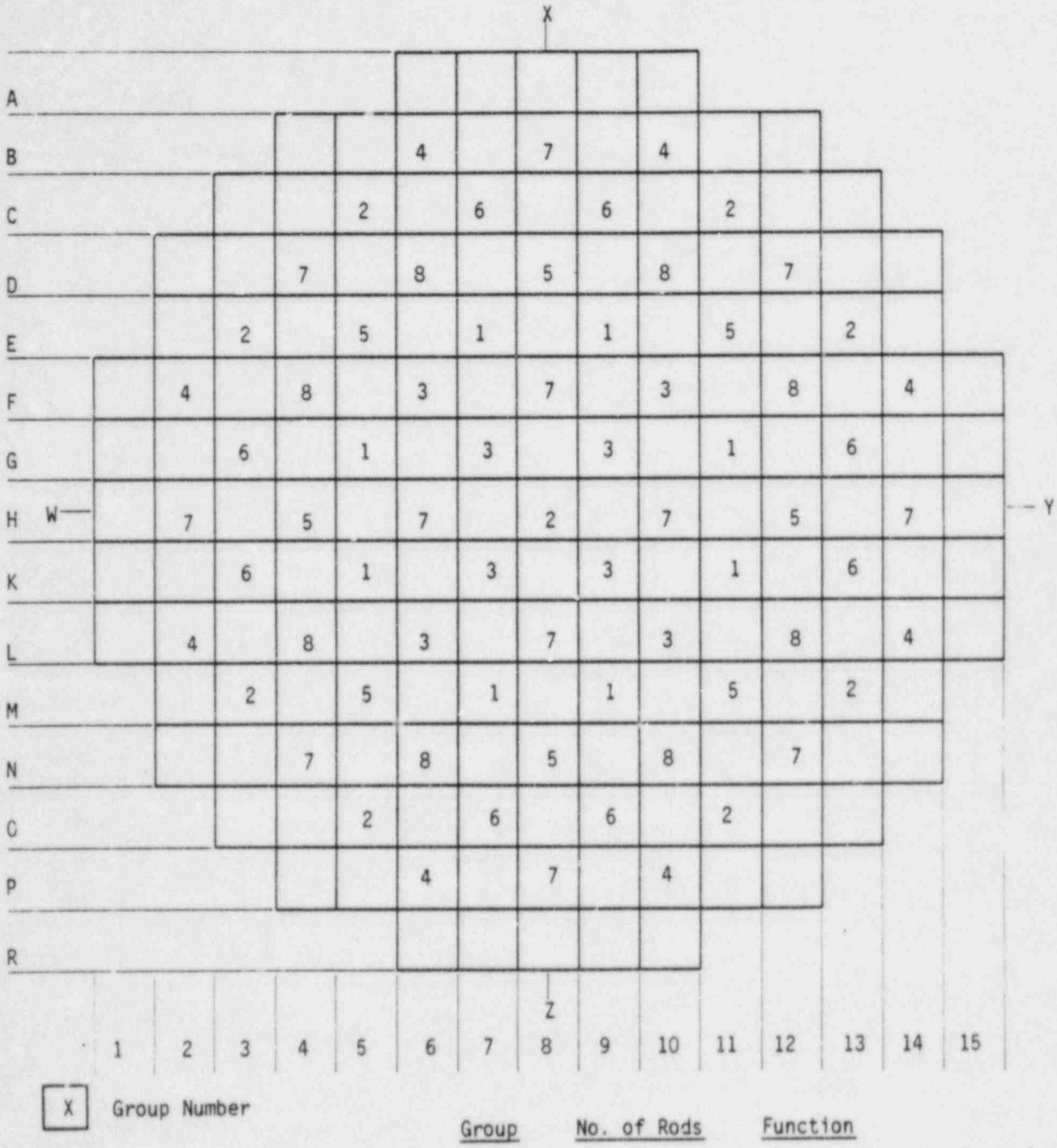
F = Fresh Fuel Assembly
* = Mark BEB LTA

Figure 3-2. Enrichment and Burnup Distribution, ANO-1
Cycle 7 off 400 EFPD Cycle 6

	8	9	10	11	12	13	14	15
H	2.95 22289	3.21 16047	3.21 16564	3.30 0	2.95 22324	3.30 0	3.21 16932	3.30 0
K		3.21 16620	3.30 0	2.95 33065	3.30 0	3.21 16397	3.30 0	3.21 15861
L			2.95 17429	3.30 0	2.95 22291	3.30 0	3.21 10859	3.21 15624
M				3.21 16268	3.30 0	3.21 12456	3.21 13552	
N					3.21 16300	3.30 0	2.95 21495	
O						2.95 23900		
P								
R								

X.XX	Initial Enrichment, wt % ^{235}U
XXXXX	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations and Group Designations for ANO-1, Cycle 7



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

Figure 3-4. LBP Enrichment and Distribution, ANO-1, Cycle 7

	8	9	10	11	12	13	14	15
H				1.4		1.4		
K			1.4		1.1		0.8	
L		1.4		1.4		0.8		
M	1.4		1.4		1.4			
N		1.1		1.4		0.0		
O	1.4		0.8		0.0			
P		0.8						
R								

X.X — LBP Concentration, wt % B_4C in Al_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel parameters for ANO-1 cycle 7 are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Retainer assemblies will be used on two fuel assemblies that contain the regenerative neutron sources, and on sixty-four fuel assemblies that contain BPRAs. Sixty-two of the retainers will be exposed for a fourth cycle of irradiation during cycle 7. This additional cycle or irradiation is justified in reference 4 based on examination of retainers which have undergone three cycles of irradiation. The results of the examination meet criteria developed earlier in terms of wear and holddown force. These criteria ensure that the retainers will perform in a safe and adequate manner in the areas of holddown force, stress, and fatigue during a fourth cycle of in-reactor use. These criteria were developed from analyses similar to those done in the original justification of the design and use of the retainer assemblies in references 5 and 6. Four of the FAs in the highest burnup batch 7B are LTAs. A description and evaluation of the LTAs is found in reference 2.

4.2. Fuel Rod Design

There has been a change in the pellet design for batch 9 fuel rods. The fuel pellet length/diameter (L/D) ratio has been decreased from 1.63 to 1.18. This change in L/D ratio will not adversely affect fuel performance, and at high burnups it is expected to decrease local cladding strains. The results of the mechanical evaluations of the fuel rods are discussed below.

4.2.1. Cladding Collapse

The batch 7 fuel is more limiting than batches 8 and 9 because of its previous incore exposure time. The batch 7 assembly power histories were analyzed to determine the most limiting three-cycle power history for creep collapse.

This worst-case power history was then compared against a generic analysis to ensure that creep-ovalization will not affect fuel performance during ANO-1 cycle 7. The generic analysis was performed based on reference 7 and is applicable for the batch 7 fuel design.

The creep collapse analysis predicts a collapse time greater than 35,000 effective full-power hours (EFPH), which is longer than the maximum expected residence time of 30,394 EFPH (Table 4-1).

4.2.2. Cladding Stress

The ANO-1 stress parameters are enveloped by a conservative fuel rod stress analysis. For design evaluation, the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses must be less than the minimum specified unirradiated yield strength. In all cases, the margin is greater than 30%. The following conservatisms with respect to the ANO-1 fuel were used in the analysis:

1. Low post-densification internal pressure.
2. Low initial pellet density.
3. High system pressure.
4. High thermal gradient across the cladding.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1% on cladding plastic tensile circumferential strain. The pellet is designed to ensure 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 that ANO-1 fuel is expected to see. The strain analysis is also based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance value for the cladding ID.

4.3. Thermal Design

All fuel in the cycle 7 core is thermally similar. The design of the four batch 7B lead test assemblies is such that the thermal performance of this fuel is equivalent to or slightly better than the standard Mark B design used in the remainder of the core. All thermal design analyses for cycle 7 fuel used the TACO2 code, as described in reference 8, for fuel temperature and fuel rod internal pressure prediction.

The results of the thermal design evaluation of the cycle 7 core are summarized in Table 4-2. Cycle 7 core protection limits were based on a linear heat rate (LHR) to centerline fuel melt of 20.5 kW/ft as determined by the TACO2 code. The LHR to melt of the LTA fuel is greater than 20.5 kw/ft. The maximum fuel assembly burnup at EOC 7 is predicted to be less than 44,400 MWd/mtU for the Mark B fuel and less than 45,700 MWd/mtU for the LTA fuel. The fuel rod internal pressures have been evaluated with TACO2 for the highest burnup fuel rods and are predicted to be less than the nominal reactor coolant pressure of 2200 psia.

In the cycle 6 reload report (reference 3), the batch 7 and batch 8 fuel parameters of Table 4-2 were listed in a fashion compatible with the modeling assumptions of the TAFY-3 code (reference 9). In that report the pellet diameter, stack height, and nominal linear heat rate were provided in Table 4-2 based on the assumption of instantaneous fuel densification. The TACO2 code, on the other hand, utilizes a time dependent fuel densification model. With the implementation of the TACO2 code for cycle 7 evaluations, the Table 4-2 parameters are provided based on nominal dimensions.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 9 fuel assemblies is identical to that of the 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 April 30, 1984, 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,[a] MWd/mtU		Cumulative net electric output,[b] MWh
		Incore	Discharged	
Oconee 1	8	34,499	50,598	48,808,138
Oconee 2	7	27,035	36,800	43,444,856
Oconee 3	8	35,123	35,463	45,200,486
Three Mile Island	5	25,200	32,400	23,840,053
Arkansas Nuclear One, Unit 1	6	31,450	36,540	38,872,852
Rancho Seco	6	30,500	38,268	33,923,457
Crystal River 3	5	23,17[illegible]	29,900	27,083,428
Davis-Besse	4	28,5[illegible]	32,790	19,237,628

(a)As of April 30, 1984.

(b)As of January 31, 1984.

4.6. Fuel Assembly Design Changes

A complete list of fuel related design changes are identified in Table 4-3. These changes will not adversely affect fuel performance.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch 7B	Batch 8	Batch 9
Fuel assembly type	Mark B4, Mark BEB	Mark B4	Mark B4
No. of assemblies	33 Mark B, 4 Mark BEB	72	68
Fuel rod OD (nom), in.	0.430	0.430	0.430
Fuel rod ID (nom), in.	0.377	0.377	0.377
Flexible spacers	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4
Undensified active fuel length (nom), in.	141.8	141.8	141.8
Fuel pellet OD (mean specified), in.	0.3686	0.3686	0.3686
Fuel pellet initial density (nom), % TD	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	2.95	3.21	3.30
Average burnup, BOC, MWd/mtU	23,992	14,910	0
Cladding collapse time, EFPH	>35,000	>35,000	>35,000
Estimated residence time, EFPH	30,394	19,680	10,080

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 7B	Batch 8	Batch 9
No. of assemblies	33[a]	72	68
Initial density, % TD	95.0	95.0	95.0
Initial pellet OD, in.	0.3686	0.3686	0.3686
Initial stack height, in.	141.80	141.80	141.80
Nominal linear heat rate at 2568 MWt, kW/ft[b]	5.74	5.74	5.74
TACO2-Based Predictions			
Average fuel temperature at nominal LHR, F	1400	1400	1400
Minimum LHR to melt, kW/ft	20.5	20.5	20.5

[a]Four LTAs were also analyzed; the results of which are reported in reference 2.

[b]Based on a nominal stack height.

Table 4-3

Fuel Assembly Design Changes

	Cycle-6 Part Number	Cycle-7 Part Number	Description of Change
Guide Tube Assy. P/N	510*	1135026-001	Improved manufacturing process; hole removed from lower end plug and holes in lower guide tube wall increased to maintain same internal flow rate.
G/T Lower End Plug P/N	511*	1138974-001	
G/T Upper Nut P/N	103*	1135026-001	
Holddown Spring P/N	563	1135021-001	Improved B10 holddown spring design made of Inconel 718.
Holddown Spider P/N	553	--	
Holddown Spring Ret. Mach. P/N	--	1134885-002	
Fuel Pellet P/N	1004892-001	1134918-001	GE fuel pellets with L/D ratio change from 1.63 to 1.18.
Spacer Sleeve A P/N	517	1135980-001	Part number change only.
" " B P/N	518	1135980-002	Part number change only.
" " C P/N	519	1135980-003	Part number change only.
BPRA Assy. P/N	970	1125783-001	Ball locking feature in coupling spider was eliminated.
BP Rod P/N	641	1125784-001	Short stack LBP configuration.
CRA P/N	600	1142078-001	Longer life CRA; the cladding material changed from stainless steel to Inconel absorber is slightly longer with no change in total poison mass.

*Thirty-two of the cycle 7 fuel assemblies used this type of GT assembly, plug and nut.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 lists the core physics parameters of design cycles 6 and 7. The values for both cycles were generated using PDQ07. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 7 at full power with equilibrium xenon and nominal rod positions.

Differences in cycle length, feed enrichment, BPRA loading, and shuffle pattern make it difficult to compare the physics parameters of cycles 6 and 7. 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 7 is greater than that for the design cycle 6 at BOC and APSR pull, but less at EOC. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 7 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 reference fuel cycle shutdown margin is presented in the ANO-1 cycle 6 reload report.[3]

5.2. Analytical Input

The cycle 7 incore measurement calculation constants to be used for computing core power distributions were prepared in the same manner as those for the reference cycle.

5.3. Changes in Nuclear Design

Core design changes for cycle 7 are the transition to a very low leakage (VLL) design and the use of "short-stack" LBPs. For this transition cycle, twelve twice-burned assemblies are located on the core periphery to reduce fluence levels on the reactor vessel. The LBP used in cycle 7 has a 9-inch shorter poison stack than that used with the standard Mark B design, i.e., 117 versus 126 inches of Al_2O_3-B_4C. The top 9 inches of the poison stack are replaced by a Zircaloy tubular spacer. This LBP design results in only a slight mass reduction versus the standard design, and does not change the dynamic characteristics of the LBP. The "short-stack" design asymmetrically positions the burnable poison stack relative to the fuel column and alters the core axial power shape to create increased "effective maneuvering room" at the beginning of the cycle.

As in cycle 6, the APSRs will be withdrawn near the end of cycle 7.

The calculated stability index at 404 EFPD without APSRs is -0.052 h^{-1} which demonstrates the axial stability of the core. The calculational methods used to obtain the important nuclear design parameters for this cycle were the same as those used for the reference cycle. The operating limits (Technical Specification changes) for the reload cycle are given in section 8.

Table 5-1. Physics Parameters for ANO-1, Cycles 6 and 7[a]

	Cycle 6[b]	Cycle 7[c]
Cycle length, EFPD	387	420
Cycle burnup, MWd/mtU	12,128	13,158
Avg. core burnup, EOC, MWd/mtU	23,009	24,238
Initial core loading, mtU	82.0	82.0
Critical boron - BOC, ppm (No Xe)		
HZP[d], group 8 ins	1463	1578
HFP, group 8 ins	1273	1346
Critical boron - EOC, ppm		
HZP, group 8 out, no Xe	704	696
HFP, group 8 out, eq Xe	95	83
Control rod worths - HFP, BOC, % Δk/k		
Group 6	1.13	1.20
Group 7	1.36	1.65
Group 8	0.42	0.39
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.40	1.53
Max ejected rod worth - HZP, % Δk/k[e]		
BOC (N-12), group 8 ins	0.53	0.69
400 EFPD (N-12), group 8 ins	0.46	0.50
EOC (M-11), group 8 out	0.47	0.52
Max stuck rod worth - HZP, % Δk/k		
BOC (N-12), group 8 ins	1.50	1.71
400 EFPD (H-14), group 8 ins	1.63	1.73
EOC (H-14), group 8 out	1.43	1.29
Power deficit, HFP to HZP, % Δk/k		
BOC	1.68	1.60
EOC	2.38	2.35
Doppler coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (no Xe)	-1.54	-1.53
EOC (eq Xe)	-1.82	-1.80
Moderator coeff - HFP, 10^{-4} (Δk/k-°F)		
BOC, (no Xe, crit ppm, group 8 ins)	-0.84	-0.69
EOC, (eq Xe, 0 ppm, group 8 out)	-2.89	-2.79

Table 5-1. (Cont'd)

	Cycle 6[b]	Cycle 7[c]
Boron worth - HFP, ppm/% Δk/k		
BOC	123	129
EOC	109	109
Xenon worth - HFP, % Δk/k		
BOC (4 EFPD)	2.57	2.55
EOC (equilibrium)	2.69	2.68
Effective delayed neutron fraction - HFP		
BOC	0.0063	0.0063
EOC	0.0053	0.0052

[a]Cycle 7 data are for the conditions stated in this report. The cycle 6 core conditions are identified in reference 3.

[b]Based on 455 EFPD at 2568 MWt, cycle 5; actual cycle length was 446.4 EFPD.

[c]Based on 400 EFPD at 2568 MWt, cycle 6, which is the actual cycle length expected.

[d]HZP denotes hot zero power (532F T_{avg}), HFP denotes hot full power (579F T_{avg}).

[e]Ejected rod worth for groups 5 through 7 inserted, group 8 as stated.

Table 5-2. Shutdown Margin Calculations for ANO-1, Cycle 7

	BOC, % Δk/k	400 EFPD, % Δk/k	420 EFPD, % Δk/k
Available Rod Worth			
Total rod worth, HZP	9.04	9.44	9.14
Worth reduction due to poison material burnup	-0.10	-0.10	-0.10
Maximum stuck rod, HZP	-1.71	-1.73	-1.29
Net worth	7.23	7.61	7.75
Less 10% uncertainty	-0.72	-0.76	-0.78
Total available worth	6.51	6.85	6.97
Required Rod Worth			
Power deficit, HFP to HZP	1.60	2.35	2.35
Allowable inserted rod worth	0.50	0.60	0.65
Flux redistribution	0.75	1.20	1.20
Total required worth	2.85	4.15	4.20
Shutdown margin (total available worth minus total required worth)	3.66	2.70	2.77

Note: The required shutdown margin is 1.00% Δk/k.

Figure 5-1. ANO-1 Cycle 7, BOC (4 EFPD) Two-Dimensional Relative Power Distribution - Full Power Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	1.05	1.24	1.23	1.17	1.01	1.26	1.10	0.83
K		1.26	1.26	0.86	1.19	1.21	1.13	0.60
L			1.09	1.18	8 0.91	1.24	0.94	0.42
M				1.22	1.25	1.14	0.68	
N					1.20	1.06	0.37	
O						0.46		
P								
R								

X X.XX	Inserted Rod Group No. Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The fresh batch 9 fuel is hydraulically and geometrically similar to the previously irradiated batches 7B and 8 fuel. The four batch 7B LTAs have been analyzed to ensure that they are never the limiting assemblies during cycle 7 operation. The results of the thermal-hydraulic analysis for the LTAs are provided in reference 2.

The thermal-hydraulic design evaluation supporting cycle 7 marks the first implementation for ANO-1 of crossflow modeling with the LYNXT codes (references 10-12) for DNB predictions. The crossflow modeling methods and applications are described in reference 13.

A notable difference in the cycle 7 modeling is the use of a 1.71 design radial-local ($F^N_{\Delta H}$) power peak with a 1.65 ($P/\bar{P}$) symmetric chopped cosine design axial flux shape. This is in comparison with the 1.71 radial-local and 1.5 axial flux shape used in cycle 6. The cycle 7 design peaking results in an allowable increase of the total peak from the cycle 6 value of 2.57 to 2.83. The selection of the cycle 7 peaking was based on the desire to increase flexibility in the determination of operating limits (i.e., rod insertion limits). Note that this change in design peaking has no impact on the results of BAW-1829[13] since that report presents the crossflow model development and justification, and not the plant specific analyses. The thermal-hydraulic design conditions for cycles 6 and 7 are summarized in Table 6-1. This table quantifies the DNB improvement for the transition to crossflow modeling with the associated design peaking for cycle 7.

The reactor protection system (RPS) setpoints for the DNB-based variable low pressure trip will remain the same for cycle 7. DNB margin improvement gained with crossflow modeling has resulted in supporting an increase of the flux/flow setpoint up to 1.08 for cycle 7.

Previous fuel cycle evaluations included the calculation of a rod bow penalty for each batch based on the highest fuel burnup in that batch. A rod bow topical report (reference 14), which addresses the mechanisms and resulting local conditions of rod bow, has been submitted to and approved by the NRC. The topical report concludes that rod bow penalty is insignificant and is offset by the reduction in power production capability of the fuel assemblies with irradiation. Therefore, no departure from nucleate boiling ratio (DNBR) reduction due to rod bow need be considered for cycle 7.

Table 6-1. Maximum Design Conditions, Cycles 6 and 7

	Cycle 6	Cycle 7
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Vessel inlet/outlet coolant temp at 100% power, F	555.6/602.4	555.6/602.4
DNBR modeling	Closed-channel	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.5 cosine	1.65 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	140.7[a]	141.8
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	175[a]	174
Max heat flux at 100% power, 10^3 Btu/h-ft^2	450[a]	492
CHF correlation	B&W-2	B&W-2
Minimum DNBR		
At 112% power	2.05	2.08
At 100% power	2.39	2.43

[a]Based on densified length.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 7 parameters to determine the effect of the cycle 7 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 15. Since batch 9 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference 15 report, the conclusions in that reference are still valid.

The radiological dose consequences of the accidents presented in Chapter 14 of the FSAR were re-evaluated for this reload report. The reason for the re-evaluation is that, even though the FSAR dose analyses used a conservative basis for the amount of plutonium fissioning in the core, improvements in fuel management techniques have increased the amount of energy produced by fissioning plutonium. Since plutonium-239 has different fission yields than uranium-235, the mixture of fission product nuclides in the core changes slightly as the plutonium-239 to uranium-235 fission ratio changes, i.e., plutonium fissions produce more of some nuclides and less of other nuclides. Since the radiological doses associated with each accident are impacted to a different extent by each nuclide and by various mitigating factors and plant design features, the radiological consequences of the FSAR accidents were recalculated using the specific parameters applicable to cycle 7. The bases used in the dose calculation are identical to those presented in the FSAR except for the following three differences:

1. The fission yields and half-lives used in the new calculations are based on more current data.
2. Updated (lowered) whole body gamma dose conversion factors.
3. The steam generator tube rupture accident evaluation considers the increased amount of steam released to the environment via the main steam relief and atmospheric dump valves because of the slower depressurization due to the reduced heat transfer rate caused by tripping of the reactor coolant pumps upon actuation of the high pressure injection (a post-TMI-2 modification).

A comparison of the radiological doses presented in the FSAR with those calculated specifically for cycle 7 (Table 7-1) show that some doses are slightly higher and some are slightly lower than the FSAR values. However, with the exception of the maximum hypothetical accident (MHA) all doses are bounded by the values represented in the FSAR or are a small fraction of the 10 CFR 100 limits, i.e., below 30 Rem to the thyroid or 2.5 Rem to the whole body. For the MHA the 2 hour thyroid dose at the exclusion area boundary (EAB) is 157.7 Rem (53% of the 10 CFR 100 limit) and the 30 day thyroid dose at the low population zone (LPZ) is 73.1 Rem (24% of the 10 CFR 100 limit). The small increases in some doses are essentially offset by reductions in other doses. Thus, the radiological impact of accidents during cycle 7 is not significantly different than that described in Chapter 14 of the FSAR.

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. Thermal parameters for fuel batches are 7B, 8, and 9 given in Table 4-2. The cycle 7 thermal-hydraulic maximum design conditions are compared with the previous cycle 6 values in Table 6-1. These parameters are common to all the accidents considered in this report. The key kinetics parameters from the FSAR and cycle 7 are compared in Table 7-2.

The implementation of crossflow modeling (see section 6) for DNB analyses has identified additional DNB margin over that of closed channel modeling used in previous cycle analyses. This additional margin has been incorporated into the DNB-based core protective safety limits for cycle 7. The flux/flow protective system setpoint, which is established by the core DNBR performance during the limiting Condition II transient (two RC pump coastdown), has now increased to 1.08 %FP/%flow for cycle 7 as a result of crossflow modeling. In addition to the gain in the flux/flow setpoint, the minimum DNBR during the limiting transient has increased by over 15 DNB points (where 1 DNB point = 0.01).

A generic LOCA analysis for a B&W 177-FA, lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model (reported in BAW-10103).[16] This analysis is generic since the limiting values of key parameters for all plants in this category were used. Furthermore, the combination of average fuel temperatures as a function of LHR and lifetime pin pressure data used in the BAW-10103 LOCA limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 and substantiated by reference 17 provide conservative results for the operation of the reload cycle. Table 7-3 shows the bounding values for allowable LOCA peak LHRs for ANO-1 cycle 7 fuel. These LHR limits include the effects of NUREG 0630 with offsetting credit taken for FLECSET.

It is concluded from the examination of cycle 7 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the ANO-1 plant's ability to operate safely during cycle 7. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 7 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 7 are bounded by the FSAR, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of FSAR and Cycle 7 Accident Doses

	FSAR doses, Rem	Cycle 7 doses, Rem
Fuel Handling Accident		
Thyroid dose at EAB (2 h)	0.92	1.24
Whole body dose at EAB (2 h)	0.54	0.21
Steam Line Break		
Thyroid dose at EAB (2 h)	1.6	1.71
Whole body dose at EAB (2 h)	--	0.008
Steam Generator Tube Failure		
Thyroid dose at EAB (2 h)	0.0087	6.15
Whole body dose at EAB (2 h)	0.16	0.52
Waste Gas Tank Rupture		
Thyroid dose at EAB (2 h)	0.22	0.054
Whole body dose at EAB (2 h)	--	1.53
Control Rod Ejection Accident		
Thyroid dose at EAB (2 h)	11.4	3.42
Whole body dose at EAB (2 h)	0.014	0.003
Thyroid dose at LPZ (30 d)	8.3	2.55
Whole body dose at LPZ (30 d)	0.0099	0.002
LOCA		
Thyroid dose at EAB (2 h)	3.6	4.10
Whole body dose at EAB (2 h)	0.057	0.026
Thyroid dose at LPZ (30 d)	1.66	1.02
Whole body dose at LPZ (30 d)	0.043	0.008
Maximum Hypothetical Accident		
Thyroid dose at EAB (2 h)	153	157.7
Whole body dose at EAB (2 h)	10	4.73
Thyroid dose at LPZ (30 d)	64.1	73.1
Whole body dose at LPZ (30 d)	3.4	1.54

Table 7-2. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR and densification report value	ANO-1 cycle 7
Doppler coeff (BOC), 10^{-5} $\Delta k/k/°F$	-1.17	-1.53
Doppler coeff (EOC), 10^{-5} $\Delta k/k/°F$	-1.30	-1.80
Moderator coeff (BOC), 10^{-4} $\Delta k/k/°F$	0.0[a]	-0.69
Moderator coeff (EOC), 10^{-4} $\Delta k/k/°F$	-4.0[b]	-2.79
All-rod group worth (HZP), % $\Delta k/k$	12.9	9.04
Initial boron concentration, ppm	1150	1346
Boron reactivity worth (HFP), ppm/% $\Delta k/k$	100	129
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.39
Dropped rod worth (HFP), % $\Delta k/k$	0.65	0.20

[a] $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ was used for the moderator dilution analysis.

[b] -3.0×10^{-4} $\Delta k/k/°F$ was used for the steam line failure analysis.

Table 7-3. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Core elevation, ft	Allowable peak LHR, first 1000 MWd/mtU, kW/ft	Allowable peak LHR, balance of cycle, kW/ft
2	14.0	15.5
4	16.6	16.6
6	17.5	18.0
8	17.0	17.0
10	16.0	16.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 7 operation to account for changes in power peaking and control rod worths. These changes are a result of the very low leakage fuel cycle design and the implementation of crossflow in the analysis. The LOCA limits used to develop the normal operating Technical Specifications include the impact of NUREG 0630 with offsetting credit taken for FLECSET.

Based on the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated. The following pages contain the revisions to previous Technical Specifications.

DNBR of 1.3 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure was actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR greater than 1.3 is predicted. The curve is the most restrictive combination of 3 and 4 pump curves, and is based upon the maximum possible thermal power at 106.5% design flow per applicable pump status. This curve is based on the following nuclear power peaking factors[2] with potential fuel densification effects:

$$F_q^N = 2.83 \qquad F_{\Delta H}^N = 1.71; \qquad F_z^N = 1.65$$

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.83$ or the combination of the radial peak, axial peak, and the position of the axial peak that yields no less than 1.3 DNBR.

2. The combination of radial and axial peaks that prevents central fuel melting at the hot spot. The limit is 20.5 kW/ft.

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 flow rates for curves 1, 2 and 3 of Figure 2.1-3 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR greater than 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. The local quality at the point of minimum DNBR is less than 22 percent.[1]

Using a local quality limit of 22 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow X flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curves 1 and 2 of Figure 2.1-3 are the most restrictive because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curve.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May 1976.

(2) FSAR, Section 3.2.3.1.1.c.

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 trip 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. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107 percent and reactor flow rate is 100 percent or flow rate is 93.5 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80 percent and reactor flow rate is 74.7 percent or flow rate is 70 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52 percent and reactor flow rate is 49.2 percent or flow rate is 45.8 percent and the power level is 49.0 percent.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The 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 reactor power imbalance (power in top half of core minus power in bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip associated with reactor power-to-reactor power imbalance boundaries by 1.07 percent for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high-pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit

Reactor Protection System Trip Setting Limits

(Specifications Table 2.3-1)

	Four RC pumps operating (nominal operating power, 100%)	Three RC pumps operating (nominal operating power, 75%)	One RC pump operating in each loop (nominal operating power, 49%)	Shutdown bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear power based on flow[b] and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
Nuclear power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2300	2300	2300	1720[a]
Low RC system pressure, psig, min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	$11.75\ T_{out} - 5103$[d]	$11.75\ T_{out} - 5103$[d]	$11.75\ T_{out} - 5103$[d]	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a] Automatically set when other segments of the RPS (as specified) are bypassed.

[b] Reactor coolant system flow.

[c] The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d] T_{out} is given in degrees Fahrenheit (F).

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant tilt:

1. Except for physics tests, if quadrant tilt exceeds 3.1% power shall be reduced immediately to below the power level cutoff (92% FP). Moreover, the power level cutoff value shall be reduced 2% for each 1% tilt in excess of 3.1%. For less than 4 pump operation, thermal power shall be reduced 2% of the thermal power allowable for the reactor coolant pump combination for each 1% tilt in excess of 3.1%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 3.1% except for physics tests, or the following adjustments in setpoints and limits shall be made:

    a. The protection system maximum allowable setpoints (Figure 2.3-2) shall be reduced 2% in power for each 1% tilt.

    b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of 3.1%.

    c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 3.1%.

3. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.

4. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

1. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

2. Operating rod group overlap shall be 20% ± 5 between two sequential groups, except for physics tests.

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

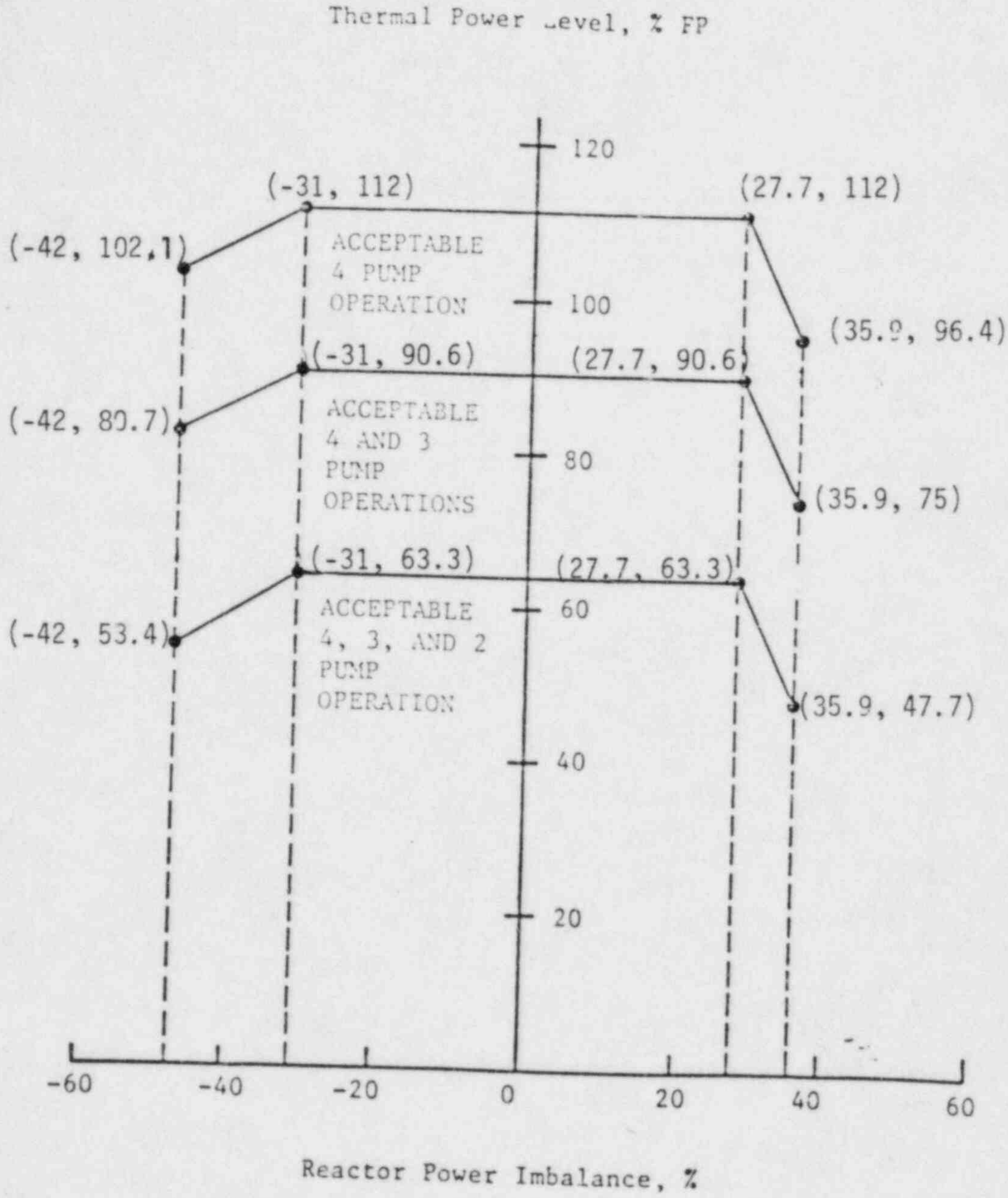
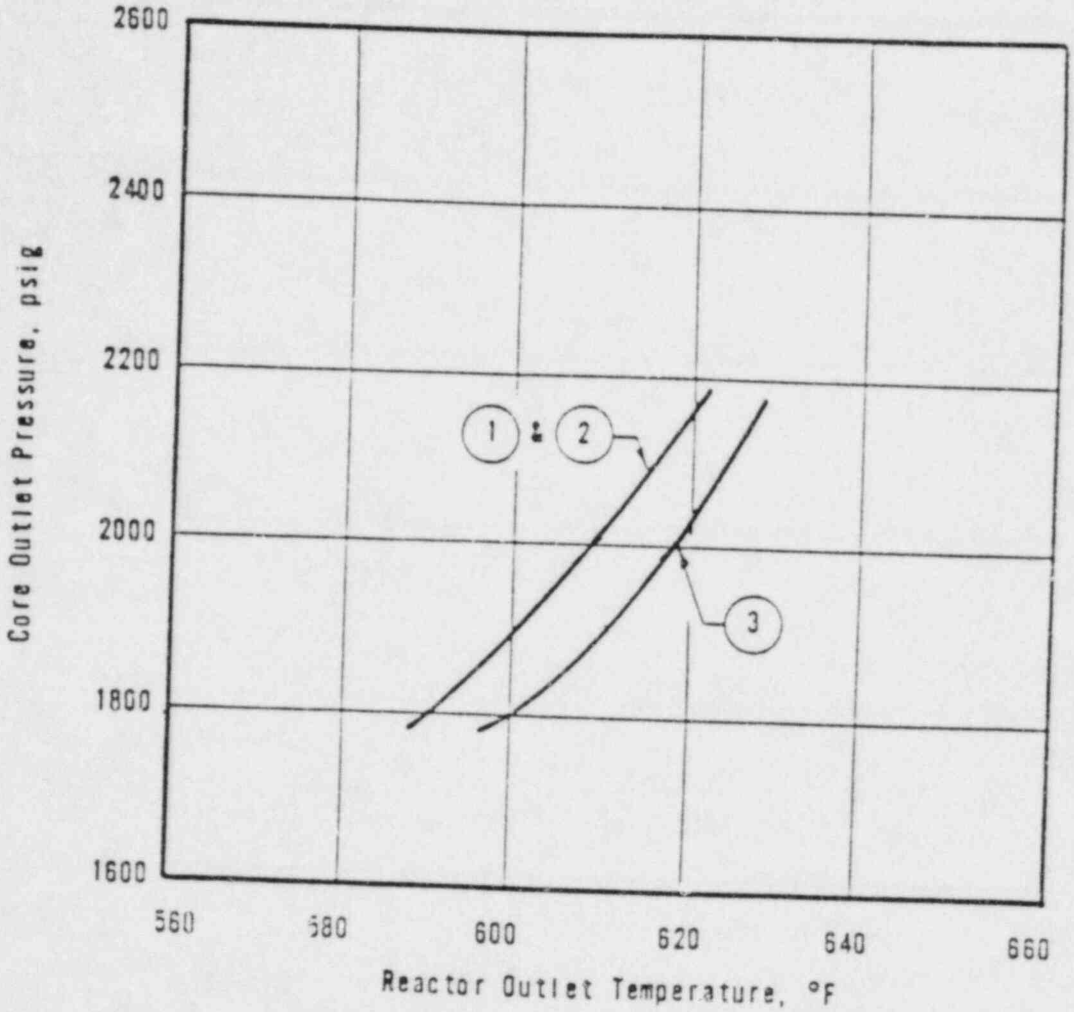


Figure 8-2. Core Protection Safety Limits -- ANO-1, (Tech Spec Figure 2.1-3)



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%)*	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	90.6%	THREE PUMPS (DNBR LIMIT)
3	184,441 (49.2%)	64.1%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

*106.5% OF DESIGN FLOW

Figure 8-3. Protective System Maximum Allowable Setpoints -- ANO-1, (Tech Spec Figure 2.3-2)

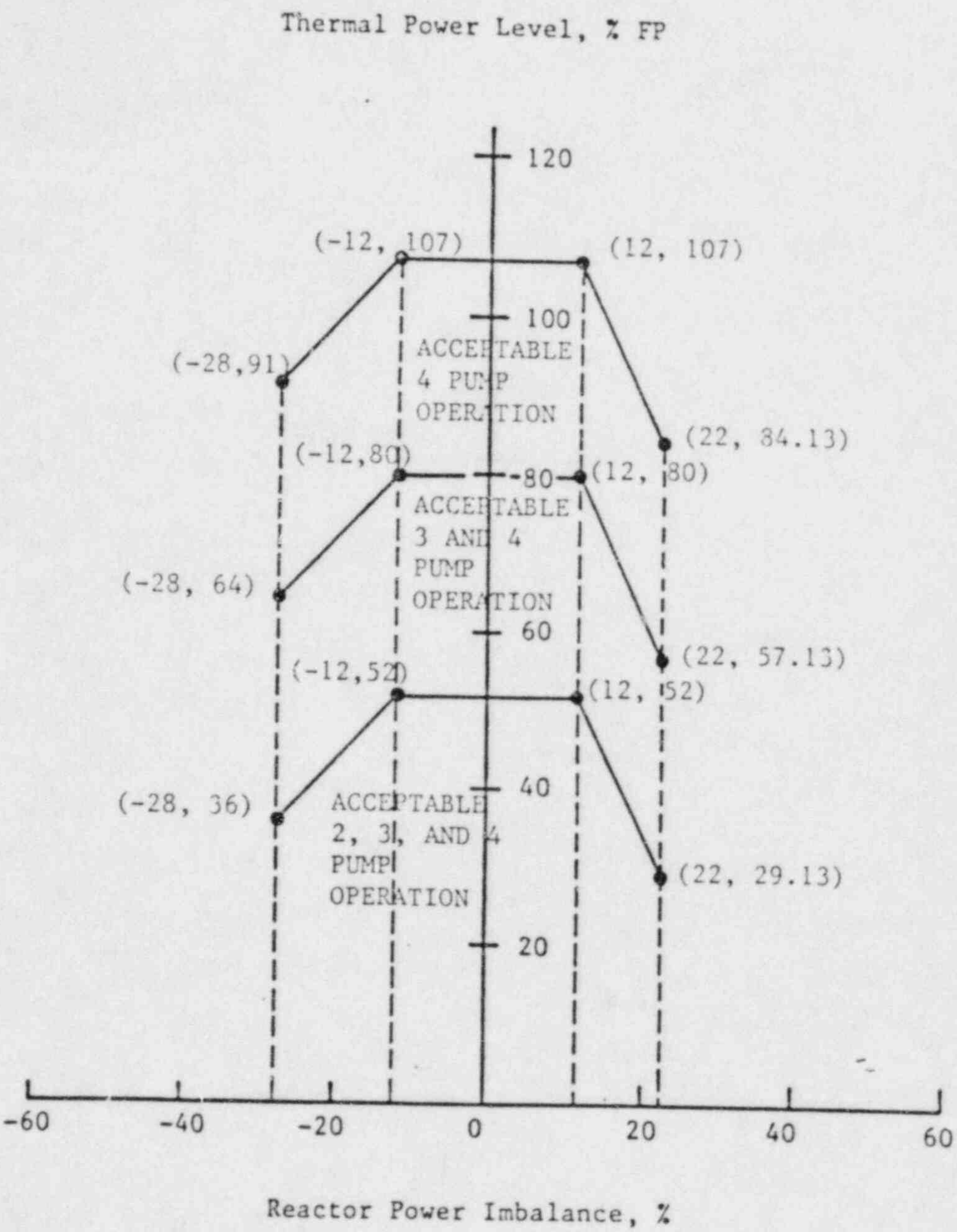
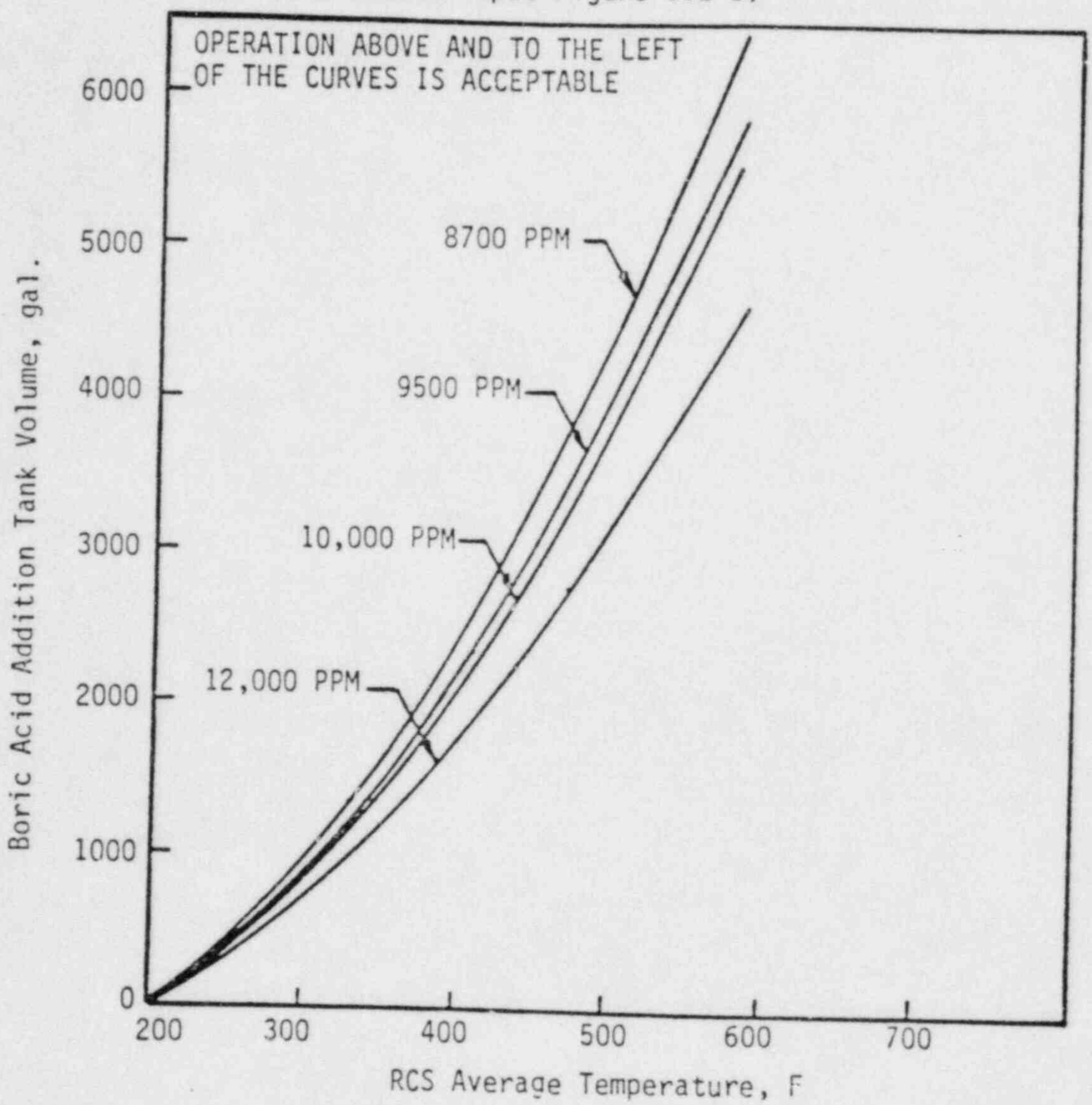


Figure 8-4. Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature -- ANO-1 (Tech Spec Figure 3.2-1)



TEMP. F	REQUIRED VOLUME, GAL.			
	8700PPM	9500PPM	10,000PPM	12,000PPM
579	6436	5863	5554	4589
532	5289	4817	4564	3769
500	4488	4087	3872	3199
400	2434	2218	2101	1737
300	986	898	851	705
200	0	0	0	0

Figure 8-5 Rod Position Limits for 4-Pump Operation
From 0 - EOC ------ ANO-1
(Tech Spec Figure 3.5.2-1)

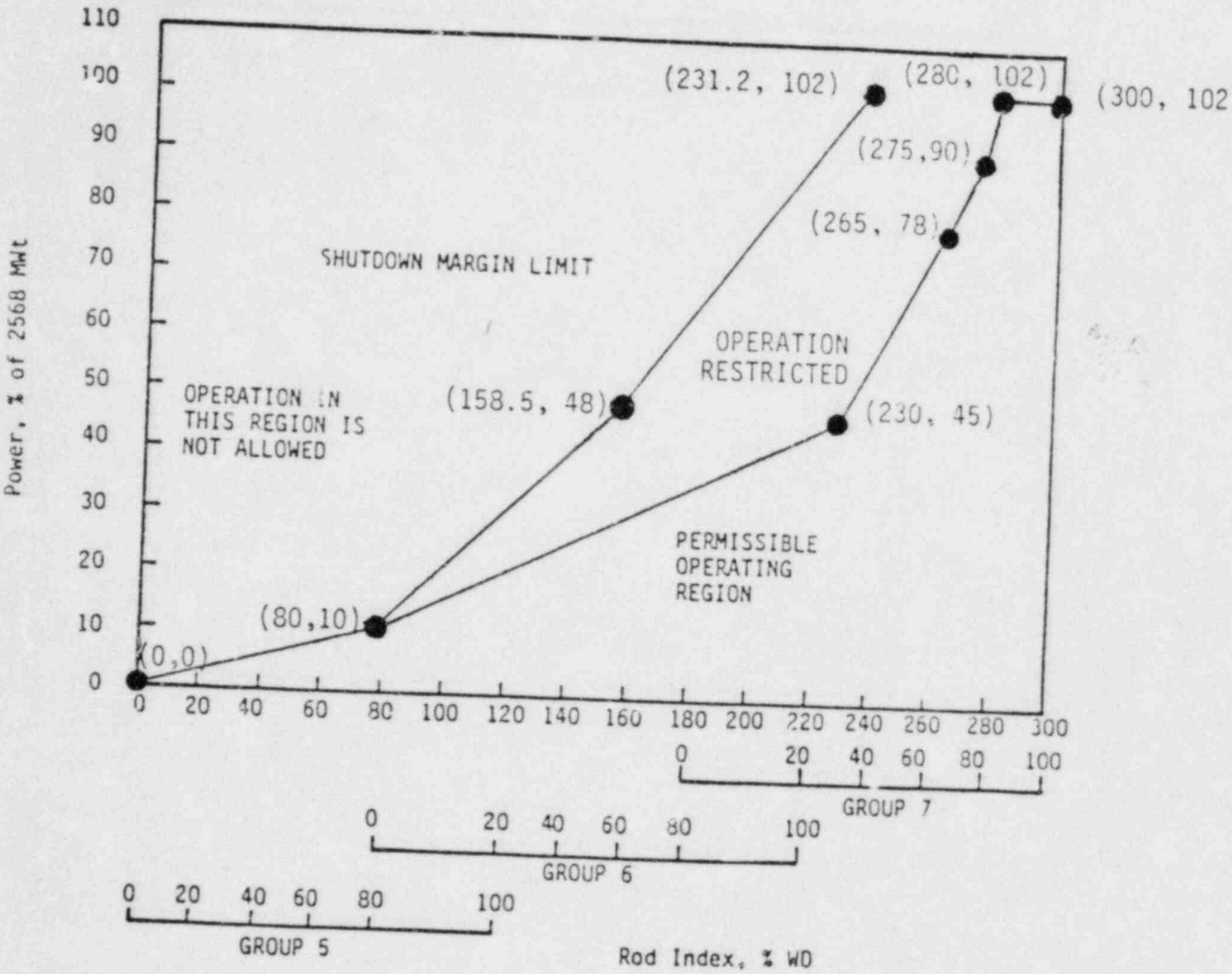


Figure 8-6 Rod Position Limits for 3 -Pump Operation
From 0 - EOC ----- ANO-1
(Tech Spec Figure 3.5.2-2)

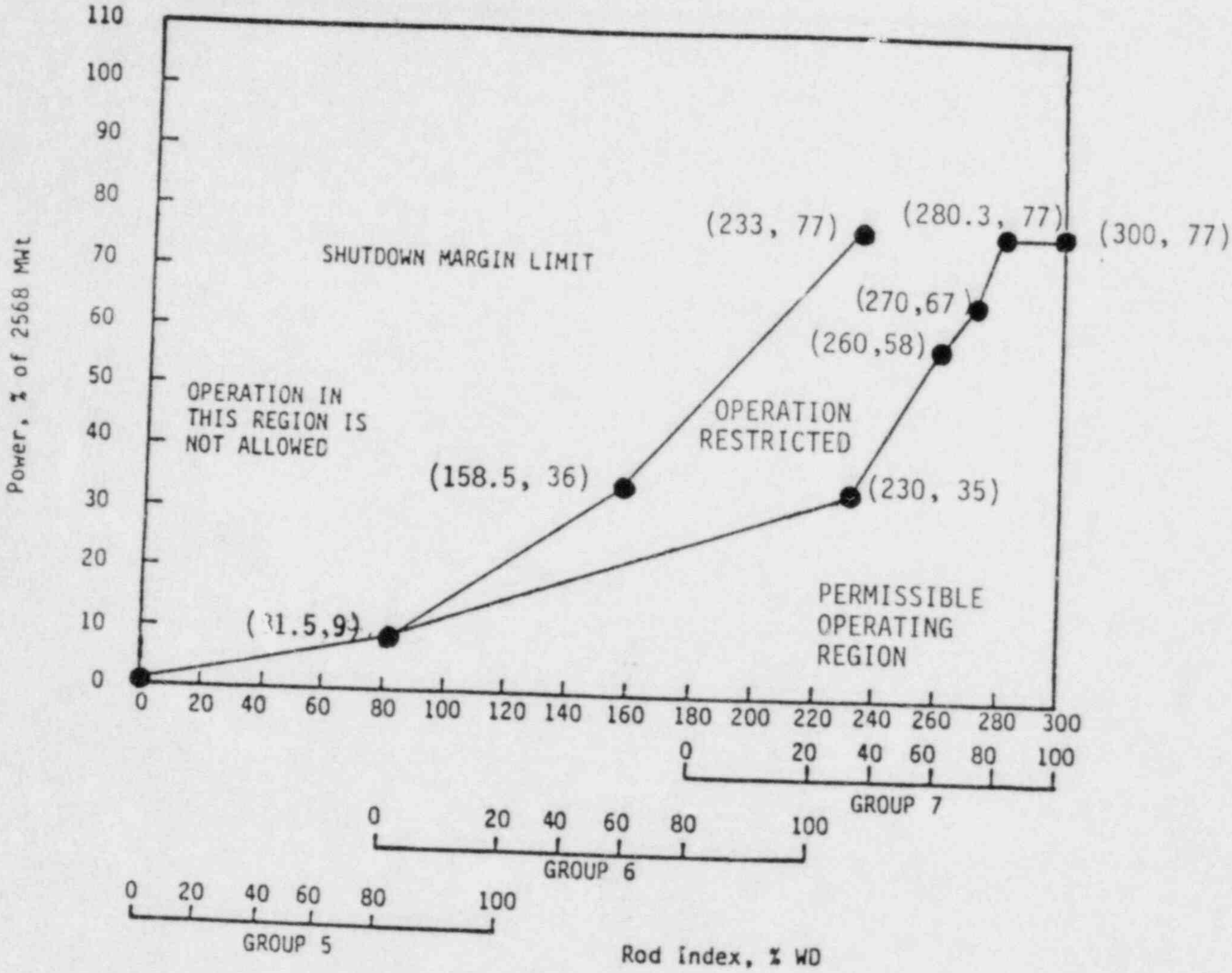


Figure 8-7 Rod Position Limits for 2-Pump Operation From 0 EFPD to EOC -- ANO-1 (Tech Spec Figure 3.5.2-3)

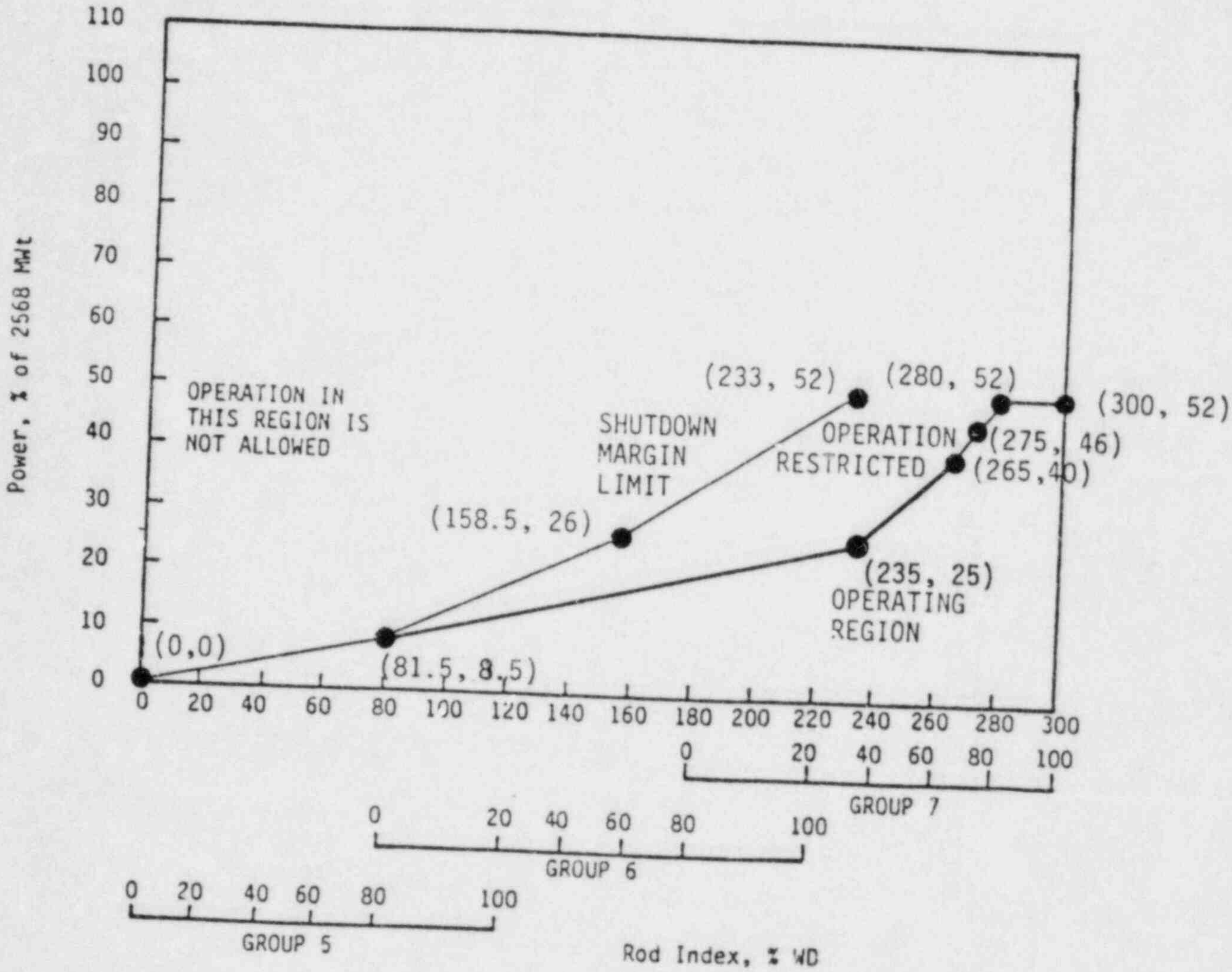


Figure 8-8. Operational Power Imbalance Envelope for Operation From 0 to EOC EFPD -- ANO-1, (Tech Spec Figure 3.5.2-4)

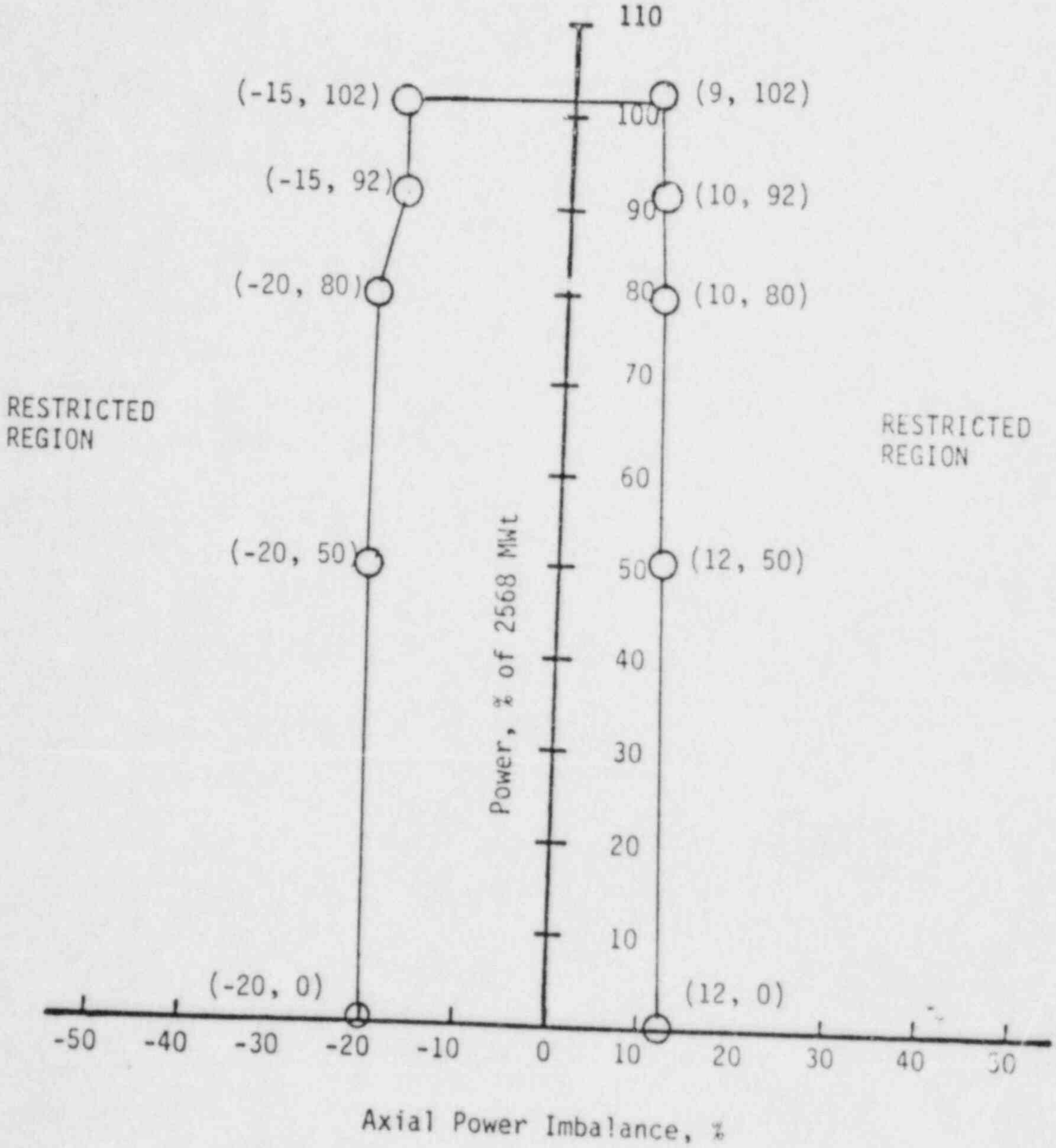


Figure 8-9. APSR Position Limits for Operat... From 0 EFPD to APSR Withdrawal --- ANO-1 (Tech Spec Figure 3.5.2-5A)

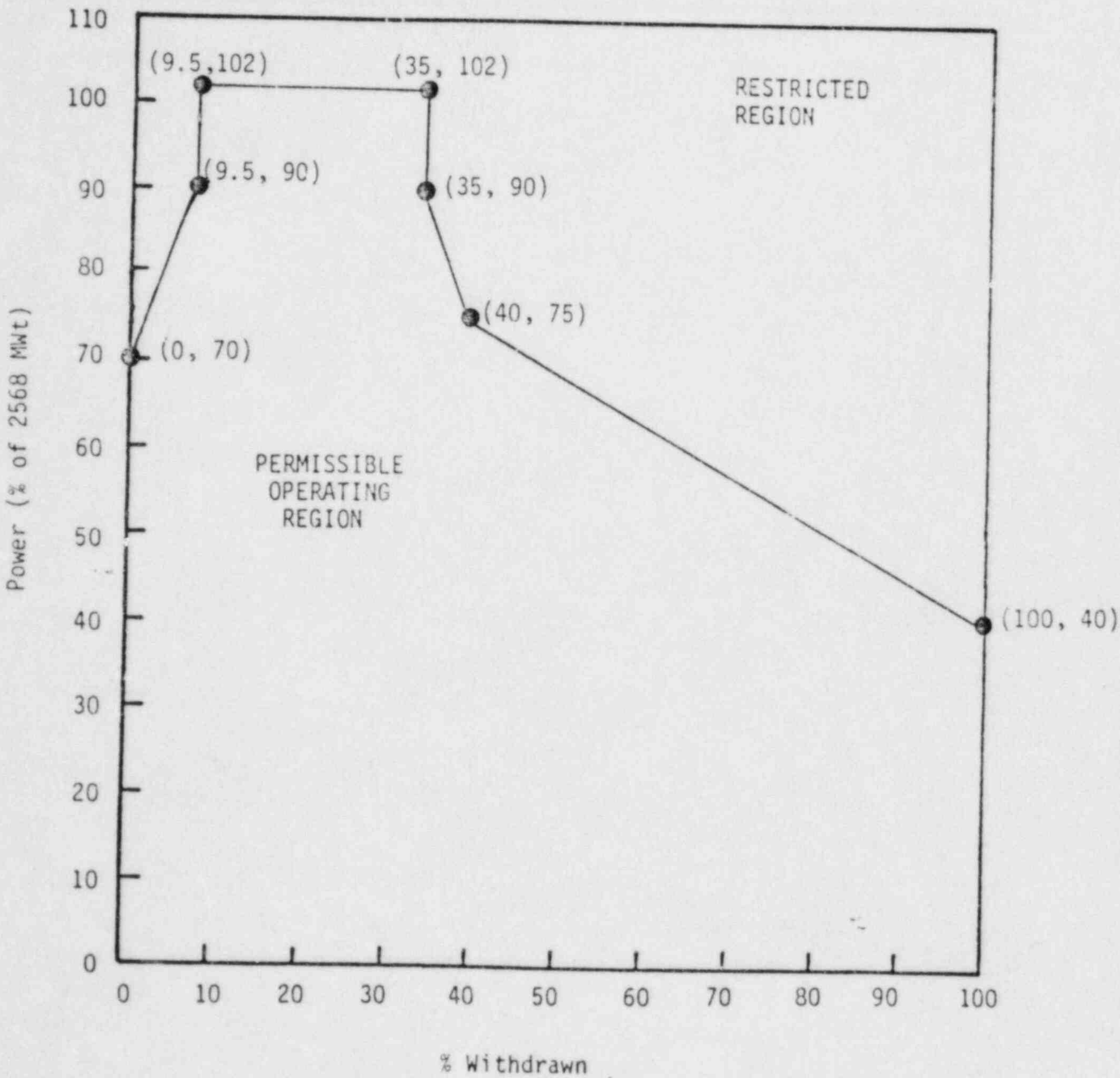


Figure 8-10. APSR Position Limits for Operation After APSR Withdrawal------------ ANO-1 (Tech Spec Figure 3.5.2-5B)

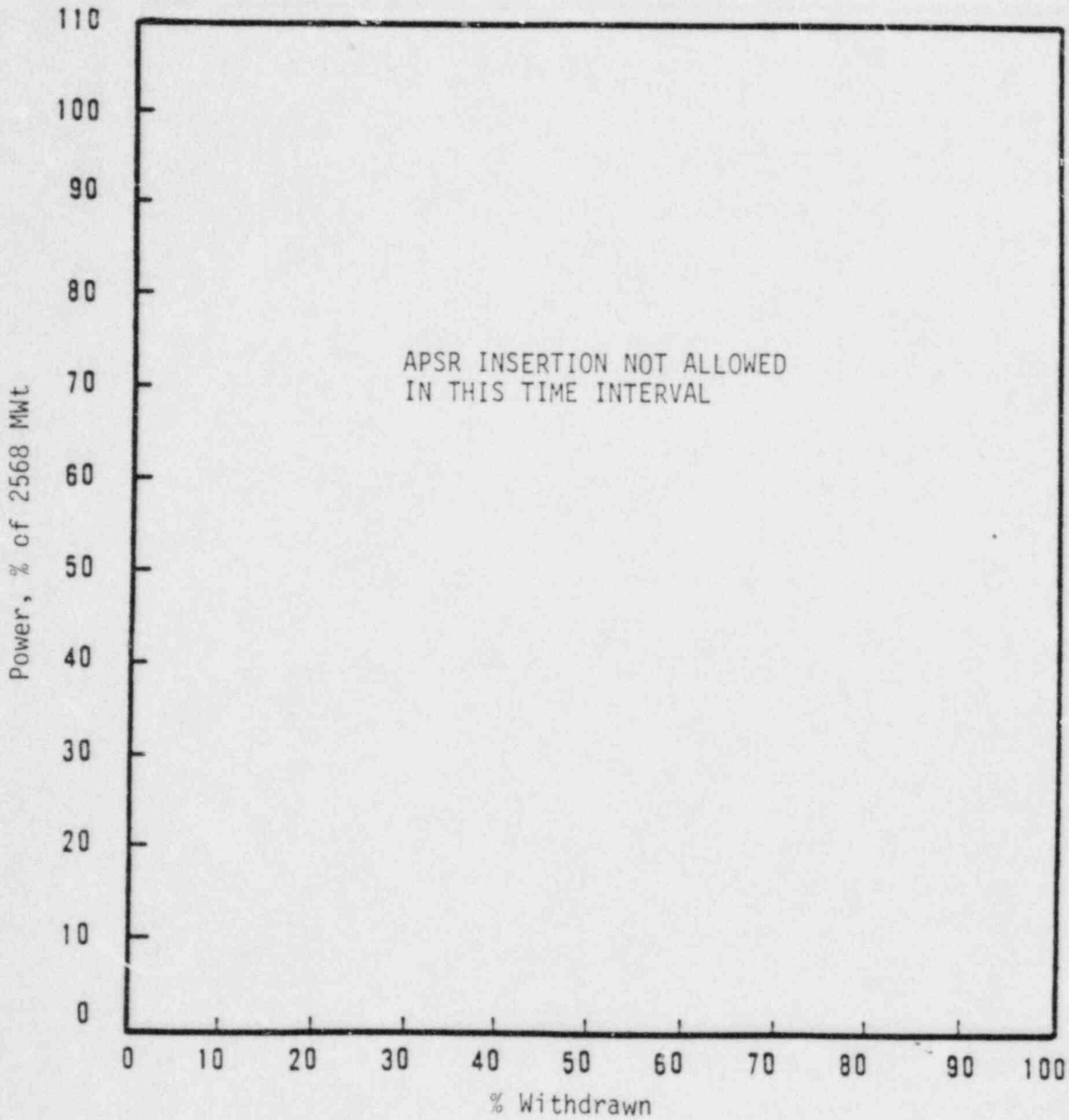
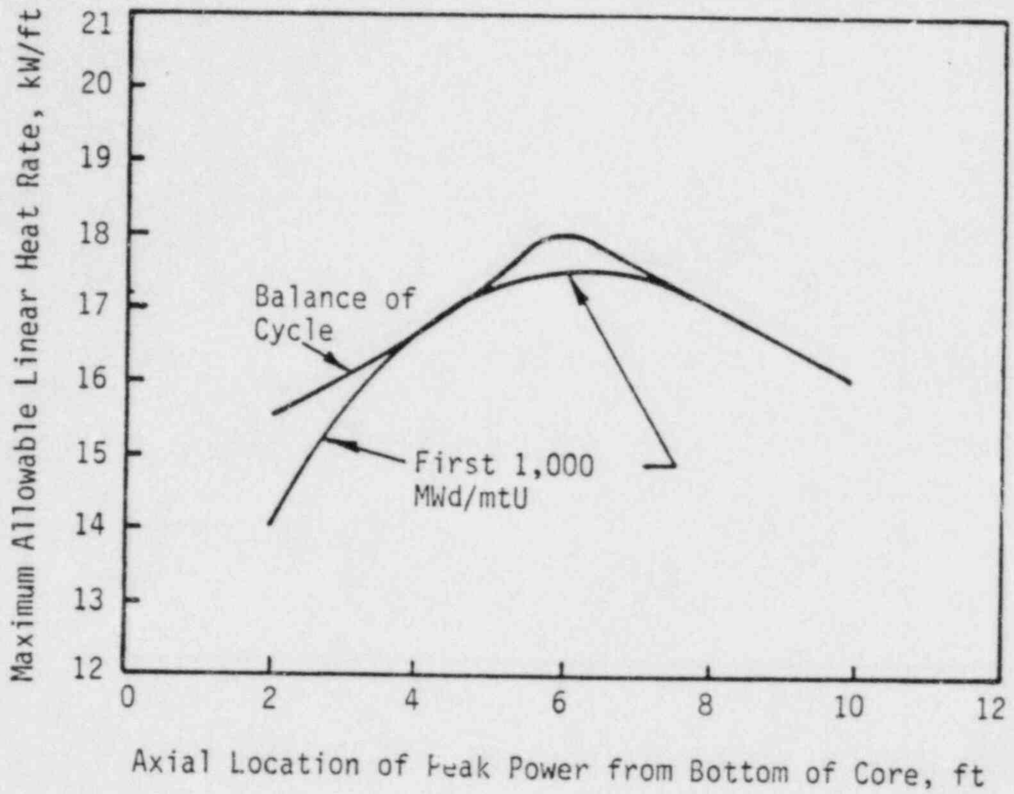


Figure 8-11. LOCA Limited Maximum Allowable Linear Heat Rate (Tech Spec Figure 3.5.2-6)



3. Except for physics tests or exercising control rods, (a) the control rod withdrawal limits are specified on Figures 3.5.2-1, 3.5.2-2 and 3.5.2-3 for 4, 3 and 2 pump operation respectively; and (b) the axial power shaping control rod withdrawal limits are specified on Figures 3.5.2-5A and 3.5.2-5B. If any of these control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours.

4. Except for physics tests, power shall not be increased above the power level cut-off of 92% of the maximum allowable power level unless one of the following conditions is satisfied:

   a. Xenon reactivity is within 10% of the equilibrium value for operation at the maximum allowable power level and asymptotically approaching stability.

   b. Except for xenon free startup, when 3.5.2.5.4a applies, the reactor has operated within a range of 87 to 92% of the maximum allowable power for a period exceeding 2 hours.

3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-4. If the imbalance is not within the envelope defined by Figure 3.5.2-4, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within 4 hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the shift supervisor.

Bases

The power-imbalance envelope defined in Figure 3.5.2-4 is based on (1) LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-6) such that the maximum cladding temperature will not exceed the Final Acceptance Criteria and (2) the Protective System Maximum Allowable Setpoints (Figure 2.3-2). Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications Section 1.6. These limits, in conjunction with the control rod position limits in Specification 3.5.2.5.3, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant tilt and axial imbalance limits in Specification 3.5.2.4 and 3.5.2.6, respectively, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided.

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. Acceptable critieria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.46 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.

9.2. 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 to achieve 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 increasing and then decreasing temperature by 5°F. During the change in temperature, reactivity feedback is compensated by discrete changes in rod motion. The change in reactivity is then calculated by the summation of reactivity (obtained from a reactivity calculator strip chart recorder) associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-4}$ $(\Delta k/k)/°F$ (predicted value obtained from Physics Test Manual curves).

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is added to obtain moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$.

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 technique consists of establishing a deboration rate in the reactor coolant 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:

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

2. Sum of groups 5, 6 and 7:

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

9.2.4. Ejected Control Rod Reactivity Worth

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

After the ejected rod has been borated to 100% withdrawn and equilibrium boron established, the ejected rod is swapped in versus the controling rod group, and the worth is determined by the change in the control rod group position. 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 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 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 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 linear heat rate.

The power distribution tests performed 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 Vs Excore Detector Imbalance Correlation Verification at ~40% FP

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 at least 1.15. If this criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required gain.

9.3.3. Temperature Reactivity Coefficient at ~100% FP

The average reactor coolant temperature is decreased and then increased by about 5°F 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.

9.3.4. 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 Use if Acceptance Criteria Not Met

If the acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. The results of all tests will be reviewed by the plant's nuclear engineering group. If the acceptance criteria of the startup physics tests are not met, an evaluation will be performed by the plant's nuclear engineering group with assistance from general office personnel, Middle South Services, and the fuel vendor, as needed. The results of this evaluation will be presented to the On-site Plant Safety Committee. Resolution will be required prior to power escalation. If a safety question is involved, the Off-site Safety Review Committee would review the situation, and the NRC would be notified if an unreviewed safety question exists.

10. REFERENCES

1. Arkansas Nuclear One, Unit 1--Final Safety Analysis Report, Docket 50-313, Arkansas Power & Light.

2. T. A. Coleman and J. T. Willse, Extended Burnup Lead Test Assembly Irradiation Program, BAW-1626, Babcock & Wilcox, Lynchburg, Virginia, October 1980.

3. Arkansas Nuclear One Unit 1, Cycle 6 Reload Report, BAW-1747, Babcock & Wilcox, Lynchburg, Virginia, November 1982.

4. J. H. Taylor (B&W) to J. F. Stolz (NRC), Letter, "Extension of Retainer Lifetime to Four Cycles," July 24, 1984.

5. BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May 1978.

6. J. H. Taylor (B&W) to S. A. Varga (NRC), Letter, "BPRA Retainer Reinsertion," January 14, 1980.

7. Program to Determine In-Reactor Performance of B&W Fuels -- Cladding Creep Collapse, BAW-10084A, Rev. 2, Babcock & Wilcox, Lynchburg, Virginia, October 1978.

8. Y. H. Hsii, et al., TACO2-Fuel Pin Performance Analysis, BAW-10141P-A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, June 1983.

9. C. D. Morgan and H. S. Kao, TAFY -- Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, Lynchburg, Virginia, May 1972.

10. B. R. Hao and J. M. Alcorn, LYNX1: Reactor Fuel Assembly Thermal Hydraulic Analysis Code, BAW-10129, Babcock & Wilcox, Lynchburg, Virginia, October 1976.

11. LYNX2: Subchannel Thermal-Hydraulic Analysis Program, BAW-10130, Babcock & Wilcox, Lynchburg, Virginia, October 1976.

12. J. H. Jones, et al., LYNXT -- Core Transient Thermal-Hydraulic Program, BAW-10156, Babcock & Wilcox, Lynchburg, Virginia, February 1984.

13. R. L. Harne and J. H. Jones, Thermal-Hydraulic Crossflow Applications, BAW-1829, Babcock & Wilcox, Lynchburg, Virginia, May 1984.

14. Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, May 1983.

15. Arkansas Nuclear One, Unit 1-Fuel Densification Report, BAW-1391, Babcock & Wilcox, Lynchburg, Virginia, June 1973.

16. ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, September 1975.

17. J. H. Taylor (B&W Licensing) to R. L. Baer (Reactor Safety Branch, USNRC), Letter, July 8, 1977.

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