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 is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.[2]

The low pressure (1800 psig) and variable low pressure ($13.89T_{out}$ -6766) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.[2,3]

To account for the calibration and instrumentation errors, the accident analysis used the safety limit of Figure 2.1-1.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620F.

E. Reactor Building Pressure

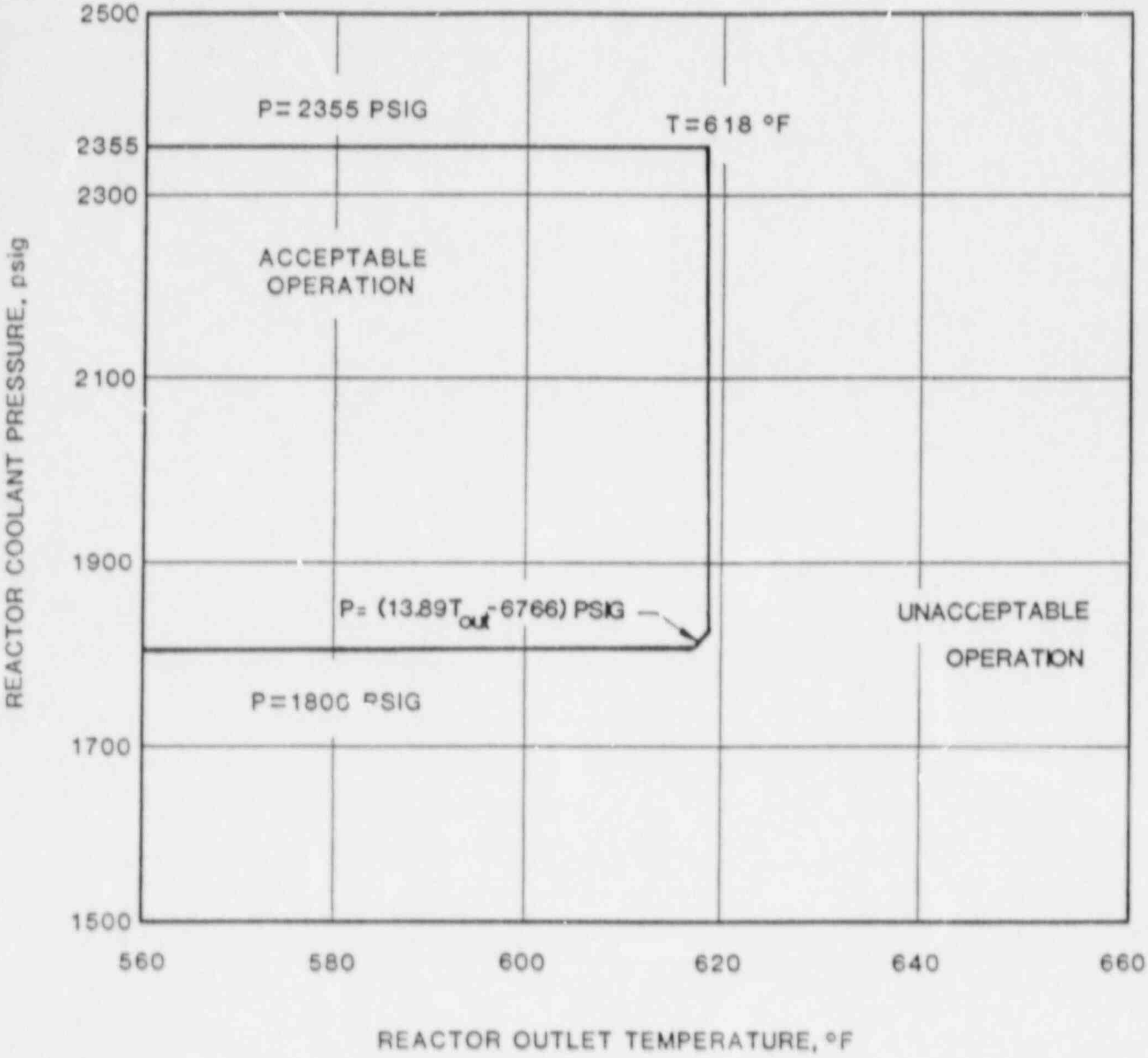
The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip setpoint of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.



PROTECTIVE SYSTEM MAXIMUM

ALLOWABLE SETPOINT

Figure 2.3-1

Table 2.3-1

Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant 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[c], % of rated, max	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig, min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	13.89 T_{out}-6766[d]	13.89 T_{out}-6766[d]	13.89 T_{out}-6766[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).

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ARKANSAS NUCLEAR ONE, UNIT 1

- Cycle 9 Reload Report -

B&W Fuel Company

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ARKANSAS NUCLEAR ONE, UNIT 1

- Cycle 9 Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the ninth 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 9 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 8 and 9 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 9 operation. In those cases where cycle 9 characteristics were conservative compared to those analyzed for previous cycles, new accident analyses were not performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 9 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 9 at a rated power level of 2568 MWt.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Arkansas Nuclear One, Unit 1 is the currently operating cycle 8. This cycle 9 design is based on a design cycle 8 length of 440 effective full power days (EFPD).

The plant was operated at 100% full power for the first 2.5 months of cycle 8. Power was then reduced in order to avoid a summer 1988 refueling. The plant was operated at 65% for 2.5 months, 80% during two summer months and 70% for 2.5 months.

Following a one month maintenance outage the plant was restarted to 80% full power. Continued operation at 80% is planned for the remainder of cycle 8.

No anomalies occurred during cycle 8 that would adversely affect fuel performance during cycle 9.

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).[1]

The cycle 9 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. 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.

Figure 3-1 is the fuel shuffle diagram for ANO-1, cycle 9. The initial enrichments of batches 6D, 9B, 10 and 11 are 3.19, 3.30, 3.35, and 3.45 wt % U-235, respectively. One batch 7D assembly, all of batch 8B, and 16 of the twice-burned batch 9 assemblies will be discharged at the end of cycle 8. The center location will contain a batch 6 assembly discharged at the end of cycle 5 (designated 6D). The remaining 52 twice-burned batch 9 assemblies (designated 9B) will be shuffled to new locations, with 16 on the core periphery. The 64 once-burned batch 10 assemblies will be shuffled to new locations, and the 60 fresh batch 11 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 9.

Reactivity is controlled by 60 full-length Ag-In-Cd control rods, 52 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight Inconel axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The cycle 9 locations of the 68 control rods and the group designations are indicated in Figure 3-3. The core locations of the total pattern (68 control rods) for cycle 9 are the same as those of the reference cycle but the group designations are different. The locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1. Core Loading Diagram for ANO-1 Cycle 9

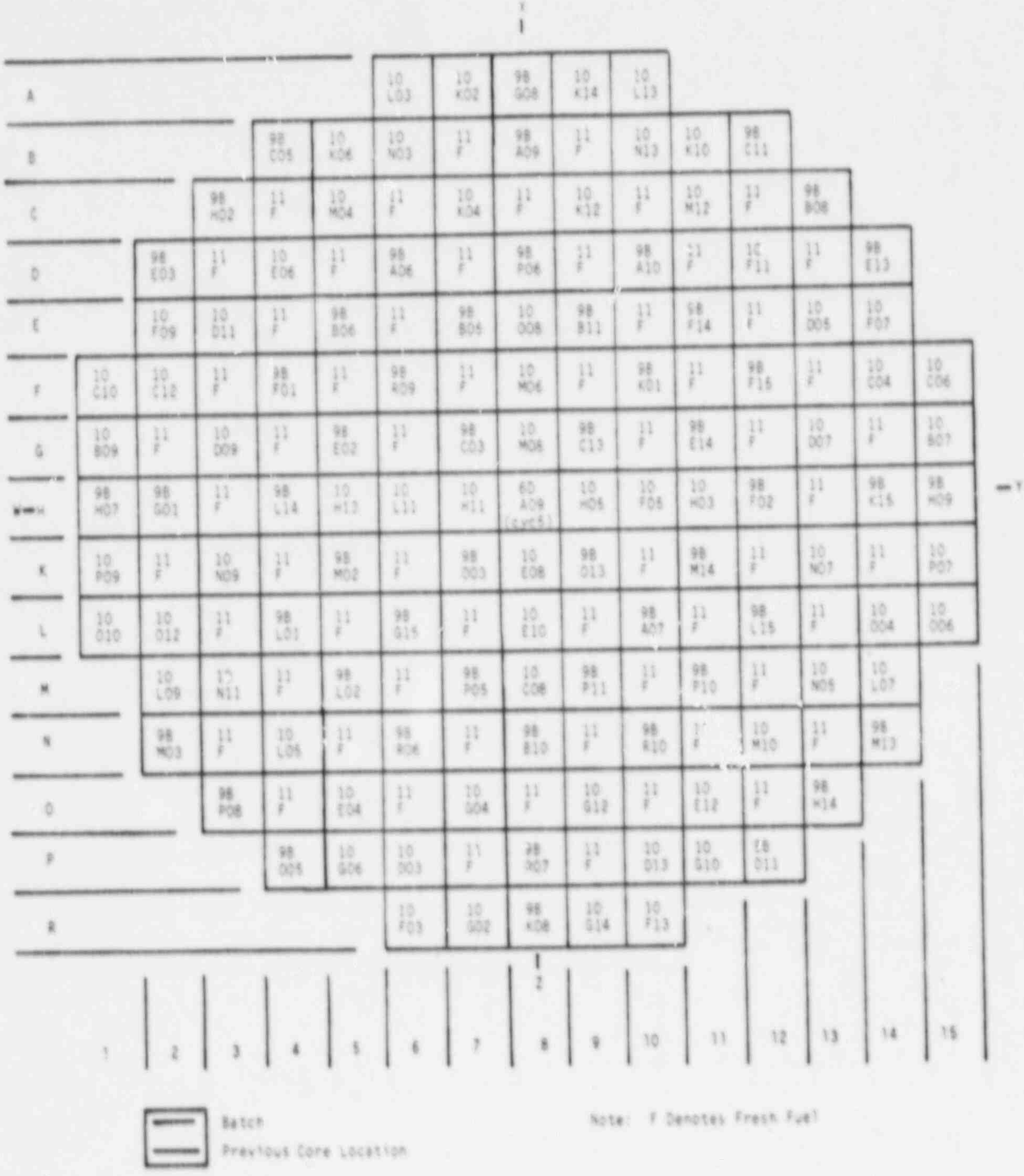
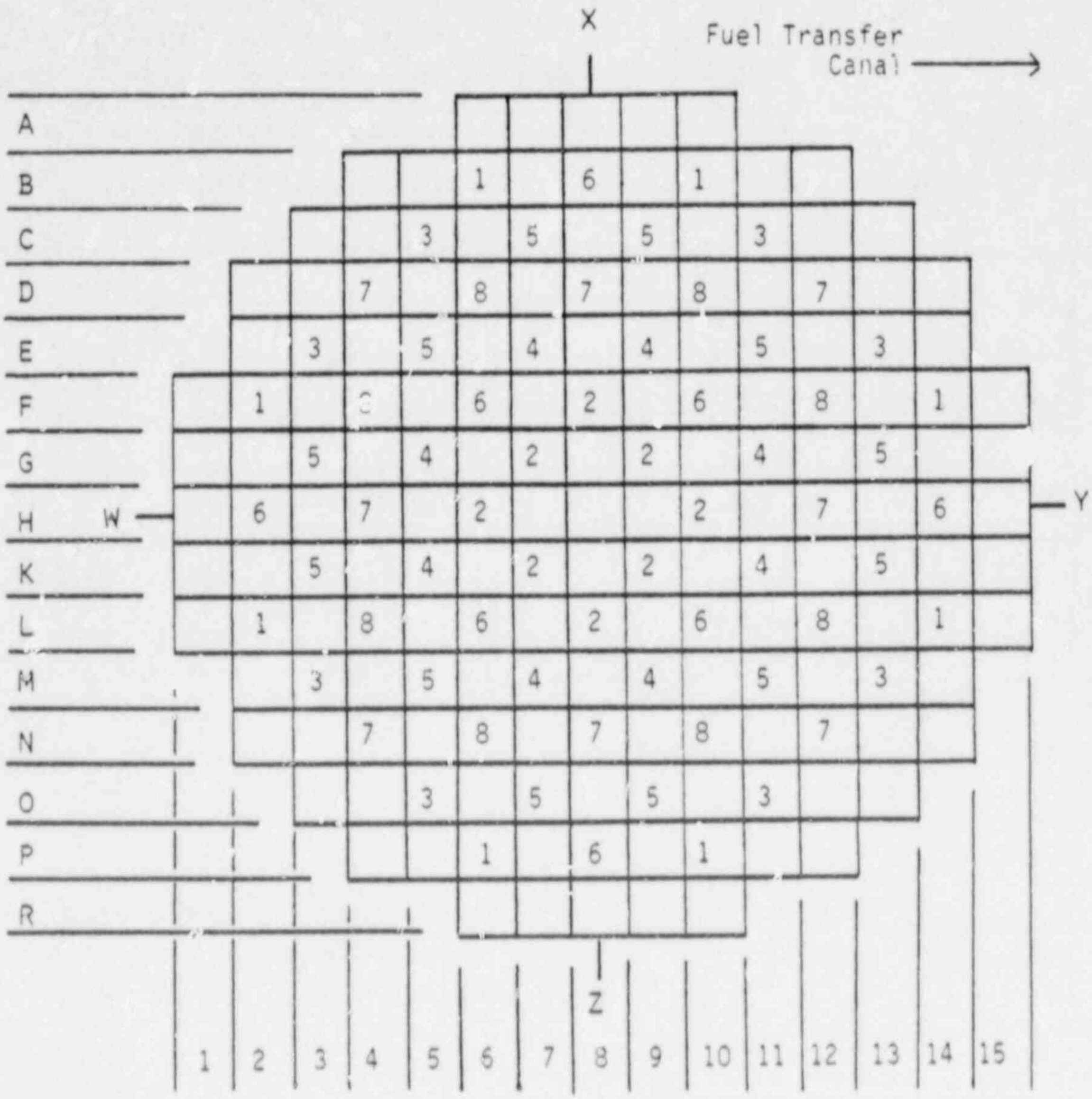


Figure 3-2. Enrichment and Burnup Distribution,
ANO-1 Cycle 9 off 440 EFPD Cycle 8

	8	9	10	11	12	13	14	15
H	3.19 20770	3.35 18734	3.35 18664	3.35 18158	3.30 28714	3.45 0	3.30 20852	3.30 31040
K		3.[illegible] 19390	3.45 0	3.30 26049	3.45 0	3.35 18549	3.45 0	3.35 15225
L			3.30 20853	3.45 0	3.30 23038	3.45 0	3.35 13625	3.35 17387
M				3.30 28713	3.45 0	3.35 18133	3.35 18684	
N					3.35 18625	3.45 0	3.30 31942	
O						3.30 30940		
P								
R								

x.xx xxxxx	Initial Enrichment, wt % U-235 BOC Burnup MWd/mtU

Figure 3-3. Control Rod Locations and Group Designations for ANO-1 Cycle 9



$\boxed{X}$ Group Number

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

Figure 3-4. LBP Enrichment and Distribution,
ANO-1 Cycle 9

	8	9	10	11	12	13	14	15
H						1.355		
K			1.355		1.400		0.200	
L		1.355		1.400		0.800		
M			1.400		1.400			
N		1.400		1.400				
O	1.355		0.800					
P		0.200						
R								

x.xxx	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 design parameters for ANO-1, cycle 9 are listed in Table 4-1. All fuel assemblies are identical in concept and are [illegible] interchangeable. Retainer assemblies will be used on the two fuel assemblies containing the regenerative neutron sources (RNS). The justification for the design and use of the retainers described in references 2 and 3 is applicable to the RNS retainers in cycle 9 of ANO-1.

The batch 11 fuel uses Zircaloy rather than Inconel as the material for the intermediate spacer grids as reported in reference 4. The NRC safety evaluation[5] of that report requires that a licensee who is incorporating that design submit a plant-specific analysis of combined seismic and LOCA loads according to Appendix A of the Standard Review Plan 4.2. The analysis that was presented in reference 4 envelopes the ANO-1 plant design requirements. Therefore, the margin of safety reported for the Mark BZ fuel assembly is applicable to ANO-1.

Batch 11 utilizes the MK-B6 type of assembly. The differences between this assembly and other MK-B types are the method used to retain fixed control components (BPRAs, orifice rod assemblies, and regenerative neutron source components) during reactor operation, Zircaloy spacer grids, and the fact that it is reconstitutable. The removable upper end fitting (Figure 4-1) provides four open slots that align and allow designed movement of the holddown spring and its retainer (Figure 4-2). The fixed control component spider is shown in Figure 4-3. The holddown spring is pre-loaded through a stop pin welded to an ear on each side of the upper end fitting. Incore, as shown in Figure 4-4, the spider feet are captured between the holddown spring retainer and the upper grid pads on the reactor internals. This arrangement retains the fixed control components at all design flow conditions. The removable upper end fitting is identical to the Mark B5 upper end fitting except for the way it is attached to the control rod guide tubes. The Mark B5 upper end fitting has been tested extensively, both in air and in over

1000 hours of simulated reactor environment, to determine analytical input and to assure good incore performance.

The removable upper end-fitting of the reconstitutable fuel assembly is a direct descendent of the Gadolinia Lead Test Assembly (LTA) upper end fitting. The end fitting design was thoroughly analyzed and tested. These results were submitted to the NRC in reference 6. The five Gadolinia LTA's with removable upper end fittings have performed as expected. The last of the LTA's that remains in-core is in its fourth cycle and has achieved a burnup of approximately 53000 MWd/mtU. By the end of this cycle it will have reached a burnup of 59000 Mwd/mTu.

The ability to reconstitute the fuel assembly has no detrimental effect on the assembly in-core performance. This allows selective replacement of damaged fuel rods within an assembly, which has a tremendous cost-saving potential.

4.2 Fuel Rod and Gray APSR Design

The mechanical evaluation of the MK-B fuel rods and the gray APSR's is discussed below.

4.2.1 Cladding Collapse

A. Fuel Rod

Creep collapse analyses were performed for the four different fuel batch power histories. Because of its longer previous incore exposure time, the batch 9B fuel is more limiting than the other batches. The batch 9B assembly power history was analyzed and the most limiting assembly was determined. The power history for the most limiting assembly was used to compare with a conservative generic creep collapse analysis. The collapse time for the most limiting assembly was conservatively determined to be more than 35000 EFPH (effective full power hours), which is greater than the maximum projected residence time (Table 4-1). The creep collapse analysis is based on reference 7.

B. Gray APSR

The gray APSRs used in cycle 9 are designed for improved creep life. Cladding thickness and rod ovality, which are the primary factors controlling the time until creep collapse, are improved to extend the life of the gray

APSR. The minimum design cladding thickness of the Mark B black APSR is 18 mils, while that of the Mk-B gray APSR is 24 mils. Additionally, the gap width between the end plug and the Inconel absorber material is reduced. Finally, the gap area ovality is controlled to tighter tolerances.

The creep collapse analysis of the gray APSR shows that it will not creep collapse during the projected lifetime of the rods. The gray APSR is shown in Figure 4-5.

4.2.2 Cladding Stress

A. Fuel Rod

The ANO-1 cycle 9 stress parameters are enveloped by a conservative fuel rod stress analysis. The same method was used for analysis of cycle 9 that had been used on the previous cycle.

B. Gray APSR

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

4.2.3 Cladding Strain

A. Fuel Rod

The strain analysis is based on the upper tolerance values for the fuel pellet diameter and density and the lower tolerance for the cladding inside diameter. The fuel design criteria specify a 1.0% limit on cladding plastic tensile circumferential strain. The pellet is designed to ensure that plastic cladding strain is less than 1.0% at design local pellet burnup and heat generation rate. The design values are higher than the worst-case values the ANO-1 cycle 9 fuel is expected to see.

B. Gray APSR

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

4.3 Thermal Design

All fuel assemblies in the cycle 9 core are thermally similar. The design of the batch 11 Mark B6 assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core. The analysis for all fuel was performed with the TACO2 code as described in reference 8. Nominal undensified input parameters used in the thermal analysis are presented in Table 4-2. Densification effects were accounted for in TACO2.

The results of the thermal design evaluation of the cycle 9 core are summarized in Table 4-2. Cycle 9 core protection limits are based on a linear heat rate (LHR) to centerline fuel melt limit of 20.5 KW/ft as determined by the TACO2 code.

The maximum fuel assembly burnup at EOC 9 is predicted to be less than 42,800 MWd/mtU (batch 9B). 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.

4.4 Material Design

The chemical compatibility of possible fuel-cladding-coolant-assembly interactions for batch 11 fuel assemblies is identical to those for previous fuel assemblies because no new materials were introduced in the batch 11 fuel assemblies.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. The accumulated operating experience for eight B&W 177 fuel assembly plants with Mark B fuel is shown in Table 4-3.

Table 4-1 - Fuel Design Parameters and Dimensions

	Batch 6D	Batch 9B	Batch 10	Batch 11
Fuel assembly type	MK-B4	MK-B4	MK-B4	MK-B6
Number of assemblies	1	52	64	60
Fuel rod OD nominal, in	0.430	0.430	0.430	0.430
Fuel rod ID nominal, in	0.377	0.377	0.377	0.377
Undensified active fuel length, in	142.25	141.8	141.8	141.8
Fuel pellet OD, (mean), in	0.3695	0.3686	0.3686	0.3686
Fuel pellet initial density, (Nom), % TD	94	95	95	95
Initial fuel enrichment, wt. % U-235	3.19	3.30	3.35	3.45
Average burnup, BOC, MWd/mtU	20800	26400	17300	0
Exposure time, EOC, EFPH	28700	31300	20600	10100
Cladding collapse time, EFPH	>35000	>35000	>35000	>35000

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 6D	Batch 9B	Batch 10	Batch 11
No. of assemblies	1	52	64	60
Initial density, % TD	94	95	95	95
Initial pellet OD, in	0.3695	0.3686	0.3686	0.3686
Initial stack height, in	142.25	141.80	141.80	141.80
Enrichment, wt % U-235	3.19	3.30	3.35	3.45
Nominal linear heat rate at 2568 MWt, kW/ft[(a)]	5.73	5.74	5.74	5.74
TACO2-based Predictions				
Average fuel temperature at nominal LHR, F (BOL)	1406	1400	1400	1400
Minimum LHR to melt, kW/ft	20.5	20.5	20.5	20.5

Core average LHR = 5.74 kW/ft

(a) Based on a nominal stack height

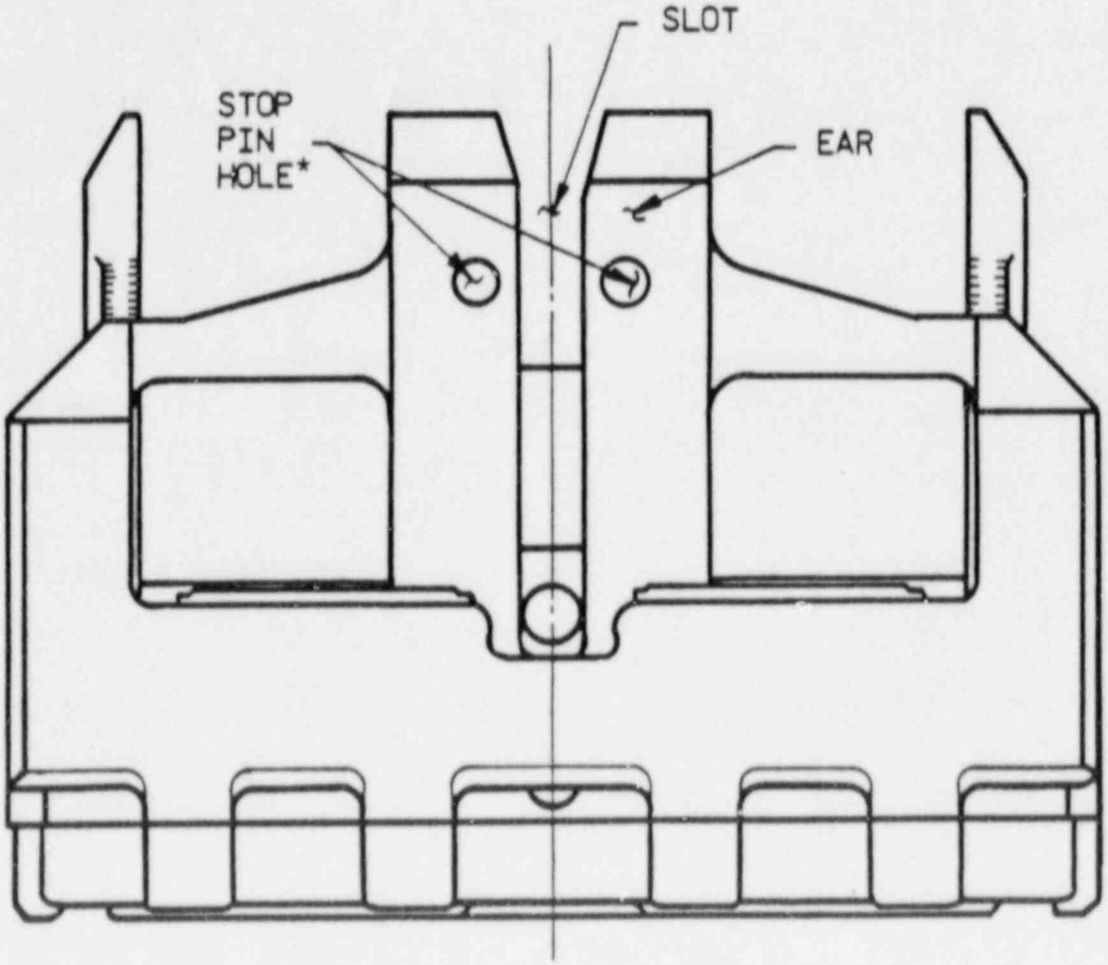
Table 4-3. Operating Experience

Reactor	Current Cycle	Max FA burnup, MWd/mtU[a] Incore	Max FA burnup, MWd/mtU[a] Discharged	Cumulative electric output, MWh[b]
Oconee 1	10	45,908	50,598	66,183,044
Oconee 2	9	40,580	41,592	60,968,626
Oconee 3	10	33,290	39,701	60,843,663
Three Mile Island	6	26,090	33,444	29,469,976
Arkansas Nuclear One, Unit 1	8	51,540	47,560	51,626,035
Rancho Seco	7	26,242	38,268	39,045,954
Crystal River 3	6	35,350	31,420	38,512,798
Davis-Besse	5	36,960	32,790	25,236,663

(a) As of October 31, 1987.

(b) As of December 31, 1986.

Figure 4-1. Removable Upper End Fitting (Side View)



*There are two stop pin holes on each side of the upper end fitting. One contains a stop pin and the other is a spare.

Figure 4-2. Holddown Spring Retainer

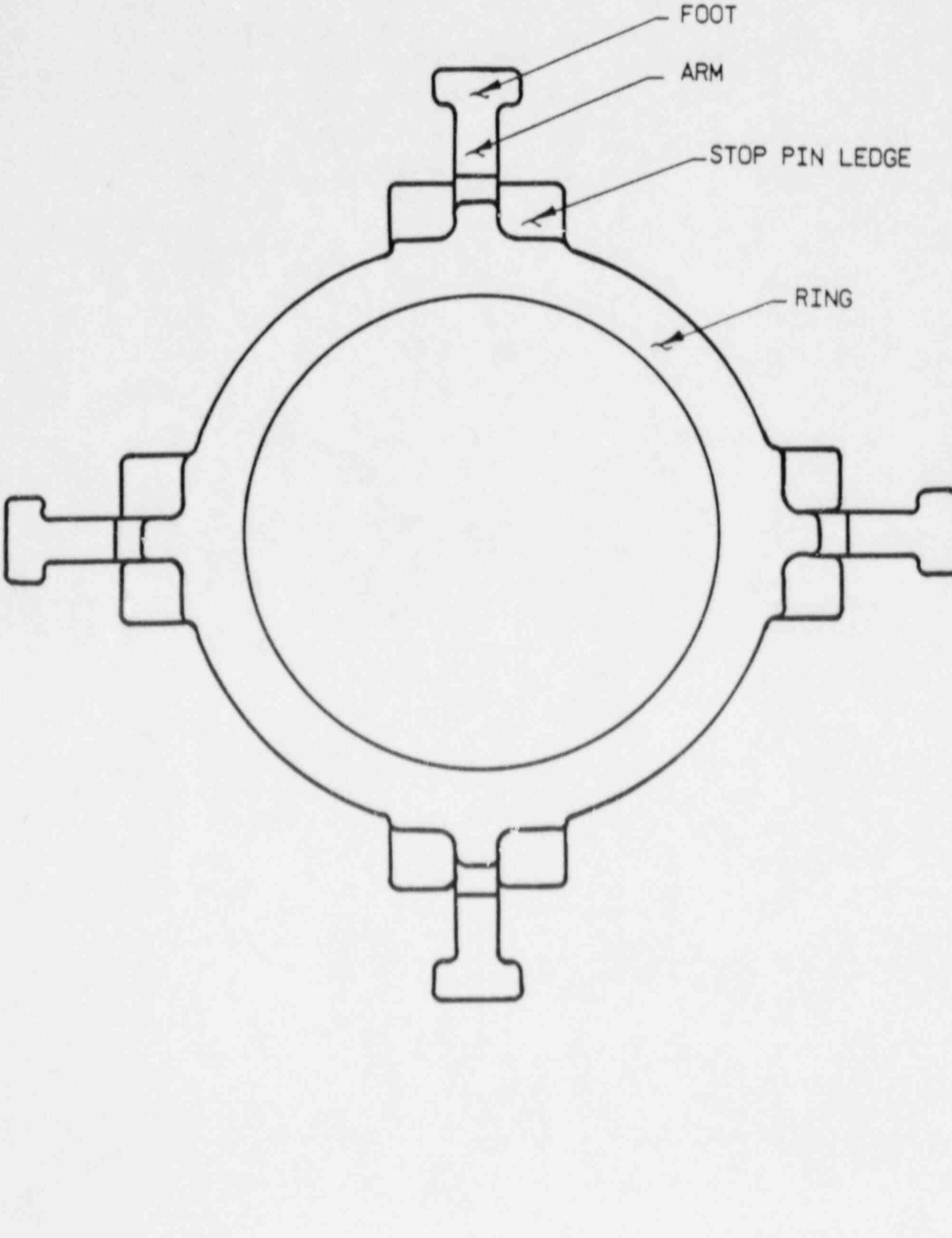


Figure 4-3. BPRA Spider

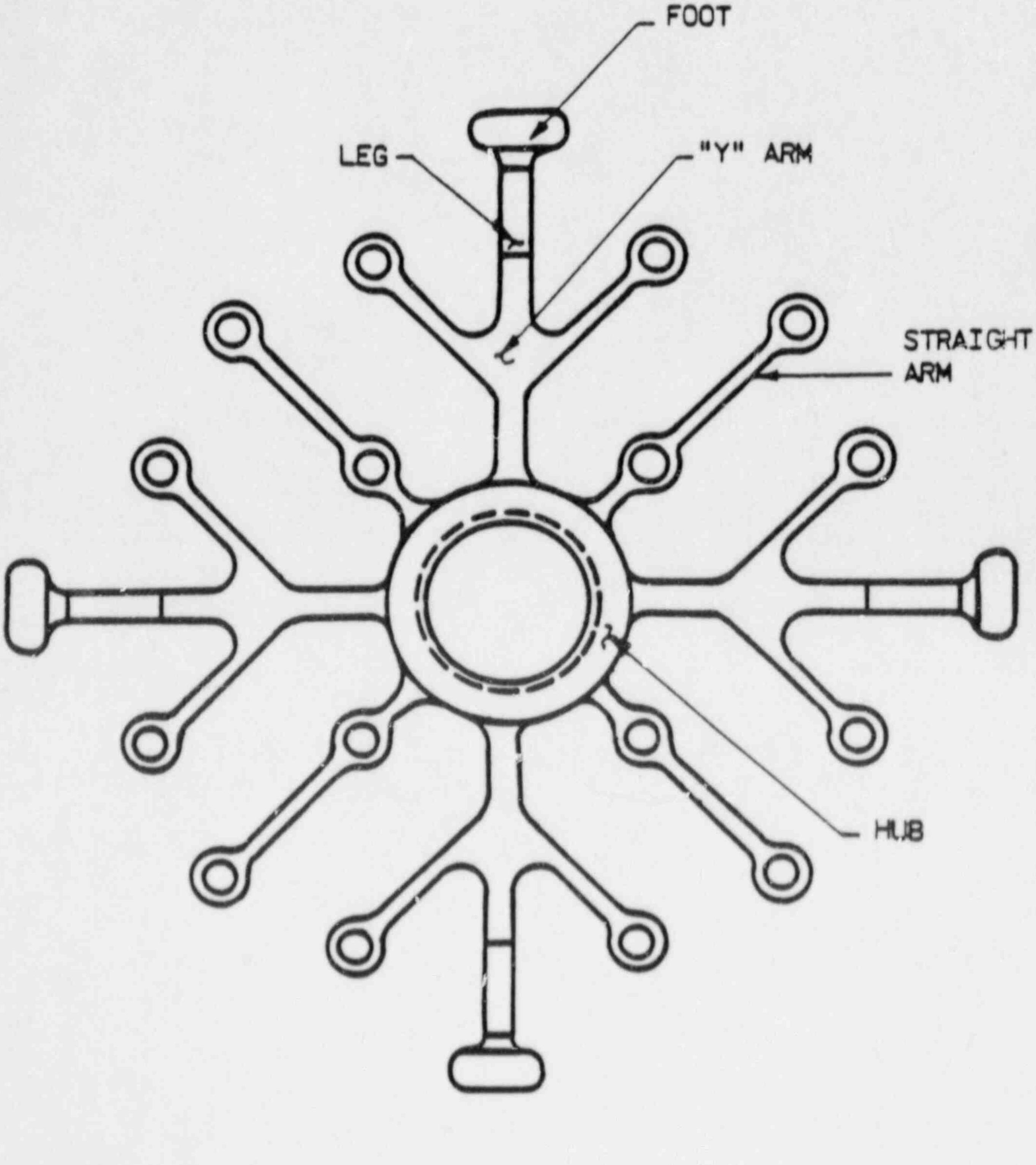


Figure 4-4. BPRA Spider/Upper End Fitting/
Reactor Internals Interaction

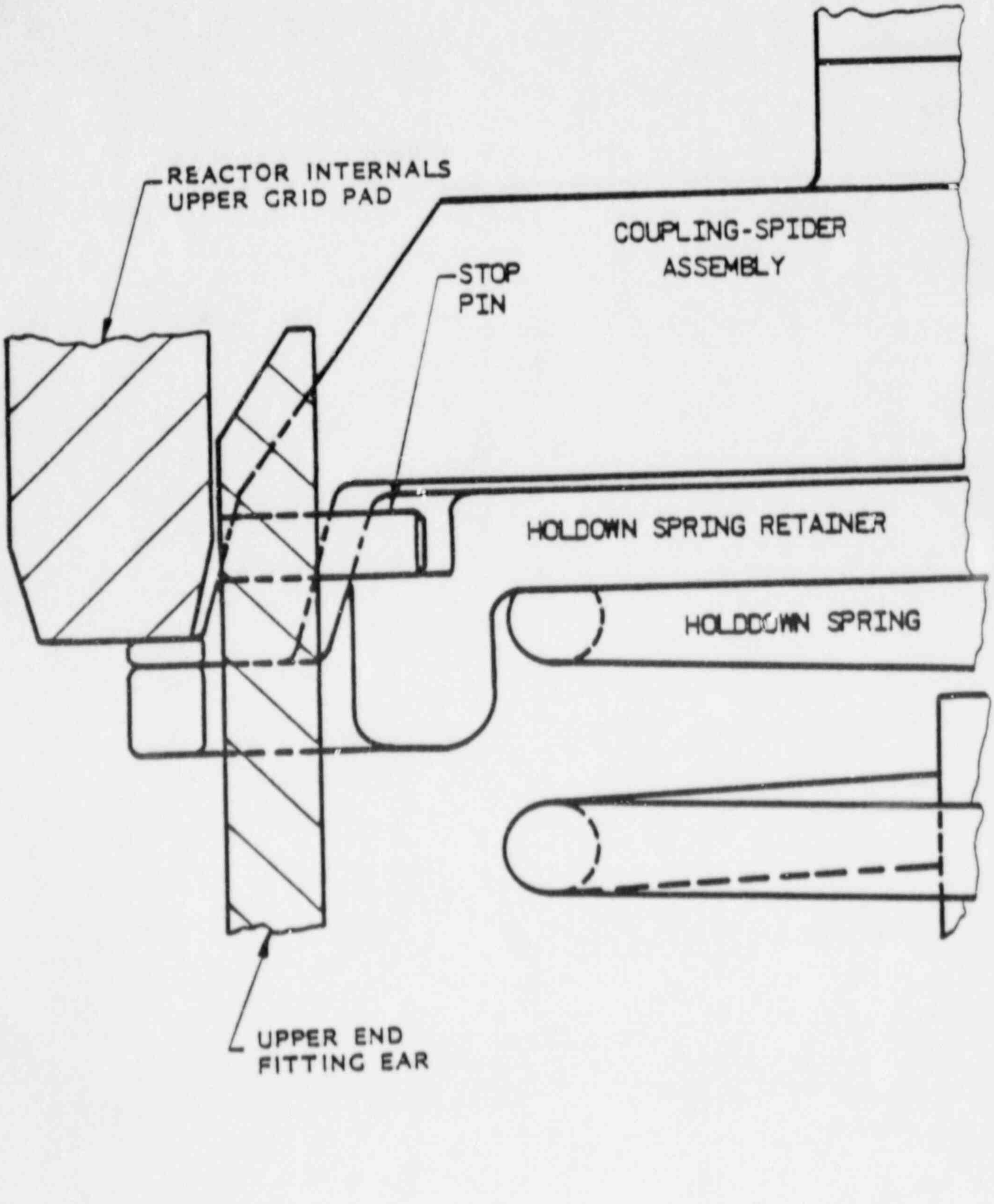
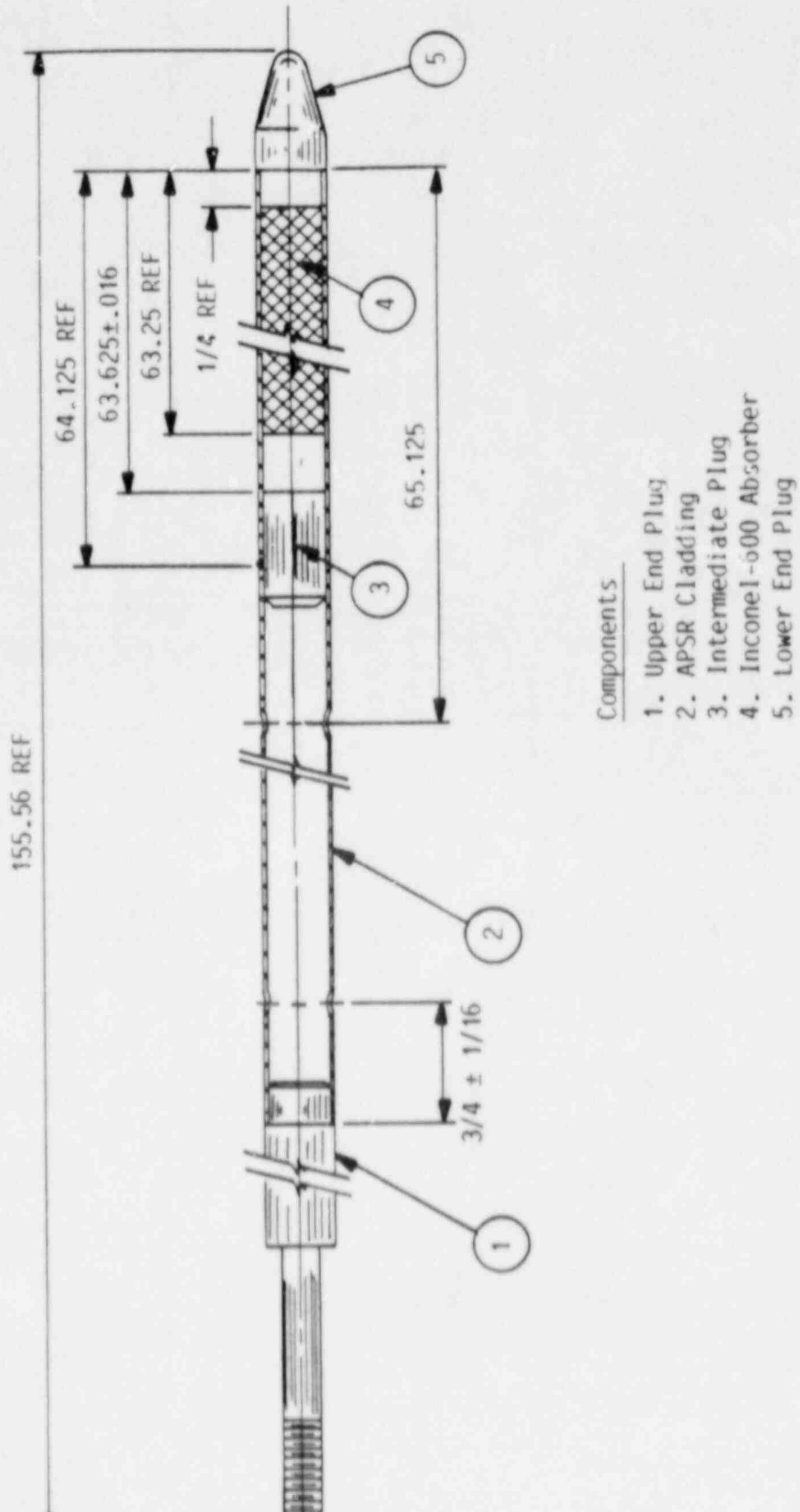


Figure 4-5. Gray Axial Power Shaping Rod



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 lists the core physics parameters of design cycles 8 and 9. The values for both cycles were calculated with the NOODLE code[9]. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 9 at full power with equilibrium xenon and nominal rod positions.

The differences in feed enrichment, BPRA loading, and shuffle pattern caused little change in the physics parameters between cycles 8 and 9. 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 worths for cycle 9 are less than for cycle 8 at all times in cycle. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 9 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 cvcle shutdown margin is presented in the ANO-1 cycle 8 reload report.[10]

5.2. Analytical Input

The cycle 9 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

The core design changes for cycle 9 are the use of gray APSRs and the replacement of the Inconel intermediate spacer grids with Zircaloy spacer grids. Gray APSRs, which are longer and use a weaker absorber (Inconel), replace the silver-indium-cadmium APSRs used in all previous cycles. Calculations with the standard three-dimensional model verified that these APSRs provide adequate axial power distribution control. The substitution of Zircaloy spacer grids reduces the parasitic absorption of neutrons and has a beneficial effect on fuel cycle cost.

The gray APSRs will be withdrawn from the core near the end of cycle 9 (360 EFPD) where the stability and control of the core in the feed-and-bleed mode with APSRs removed has been analyzed. The calculated stability index at 364 EFPD without APSRs is $-0.037\ h^{-1}$ 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 Specifications changes) for the reload cycle are given in section 8.

Table 5-1. Physics Parameters for ANO-1, Cycles 8 and 9[(a)]

	Cycle 8[(b)]	Cycle 9[(c)]
Cycle length, EFPD	420	420
Cycle burnup, MWd/mtU	13,147	13,143
Average core burnup - EOC, MWd/mtU	25,522	27,271
Initial core loading, mtU	82.0	82.1
Critical boron - BOC, ppm (no Xe)		
HZP[(d)], group 8 inserted	1644	1552
HFP, group 8 inserted	1409	1379
Critical boron - EOC, ppm (eq Xe)		
HZP, group 8 inserted	651	539
HFP, group 8 inserted	18	0[(e)]
Control rod worths - HFP, BOC, % Δk/k		
Group 6	1.14	1.11
Group 7	1.49	0.98
Group 8 (maximum)	0.39	0.19
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.52	1.05
Max ejected rod worth - HZP, % Δk/k		
BOC (L-10), groups 5-8 ins	0.55	0.35
360 EFPD (L-10), groups 5-8 ins	0.60	0.41
EOC (L-10), groups 5-7 ins	0.59	0.41
Max stuck rod worth - HZP, % Δk/k		
BOC (N-12), groups 1-8 ins	1.58	1.49
360 EFPD (N-12), groups 1-8 ins	1.86	1.47
EOC (N-12), groups 1-7 ins	1.63	1.42
Power deficit, HZP to HFP, % Δk/k		
BOC	1.56	1.60
EOC	2.34	2.35
Doppler coeff - HFP, 10^{-4} ($\Delta k/k/^{o}F$)		
BOC (no Xe)	-0.154	-0.159
EOC (eq Xe)	-0.184	-0.186

Table 5-1. (Cont'd)[a]

	Cycle 8[b]	Cycle 9[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^{o}F$)		
BOC (no Xe, crit ppm, group 8 ins)	-0.51	-0.58
EOC (eq Xe, 0 ppm, group 8 out)	-2.78	-2.82
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	129	130
EOC	111	111
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.55	2.56
EOC (equilibrium)	2.72	2.71
Effective delayed neutron fraction - HFP		
BOC	0.0062	0.0062
EOC	0.0052	0.0052

(a) Cycle 9 data are for the conditions stated in this report. The cycle 8 core conditions are identified in reference 9.

(b) Based on 425 EFPD at 2568 MWt, cycle 7.

(c) Based on 440 EFPD at 2568 MWt, cycle 8.

(d) HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (581F T_{avg}).

(e) At HFP conditions, 0 ppm occurs at 411 EFPD.

Table 5-2. Shutdown Margin Calculation for ANO-1, Cycle 9

	BOC, % Δk/k	360 EFPD, % Δk/k	420 EFPD, % Δk/k
Available Rod Worth			
Total rod worth, HZP	8.699	9.265	9.222
Worth reduction due to poison material burnup	-0.100	-0.100	-0.100
Maximum stuck rod, HZP	-1.490	-1.471	-1.419
Net Worth	7.109	7.694	7.703
Less 10% uncertainty	-0.711	-0.769	-0.770
Total available worth	6.398	6.925	6.933
Required Rod Worth			
Power deficit, HFP to HZP	1.602	2.291	2.351
Allowable inserted rod worth	0.276	0.422	0.422
Flux redistribution	0.344	0.616	0.573
Total required worth	2.222	3.329	3.346
Shutdown margin (total available worth minus total required worth)	4.176	3.596	3.587

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

Figure 5-1. ANO-1 Cycle 9, BOC (4 EFPD) Two-Dimensional Relative Power Distribution -- Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.98	1.12	1.18	1.13	1.02	1.29	0.94	0.41
K	1.13	1.12	1.29	1.05	1.29	1.24	1.17	0.52
L	1.19	1.31	1.16	1.28	8 1.12	1.31	0.93	0.39
M	1.14	1.06	1.28	1.04	1.28	1.07	0.61	
N	1.02	1.29	8 1.12	1.28	1.16	1.09	0.32	
O	1.29	1.25	1.31	1.07	1.09	0.42		
P	0.94	1.18	0.93	0.61	0.32			
R	0.41	0.53	0.39					

x x.xx	Inserted Rod group No. Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 9 operation utilized the methods and models described in references 10, 11, and 12 as supplemented by reference 4, which implements the BWC (reference 13) CHF correlation for analysis of the Zircaloy grid fuel assembly. The analyses presented in Section 5 of reference 4 demonstrate that changes in the flow parameters resulting from the incorporation of Zircaloy spacer grids do not significantly impact the thermal-hydraulic characteristics of the Zircaloy grid core relative to the Inconel grid core values. Implementation of the Zircaloy grid fuel assemblies into existing reactors, however, is performed on a batch basis, with the transition cycles having both Zircaloy grid and Inconel g..d fuel assemblies.

In a transition core, the Zircaloy grid fuel assemblies, which have a slightly higher pressure drop than the Inconel grid assemblies due to the higher flow resistance of the Zircaloy grids, tend to divert some flow to the Inconel grid fuel. This creates the need to consider a "transition core penalty". The amount of coolant flow reduction in the limiting Zircaloy grid assembly and consequently, the magnitude of the transition penalty, is dependent on the number of Zircaloy grid assemblies (with the smaller number of Zircaloy grid assemblies being more limiting).

Another contributing factor in determining the transition core penalty is the core bypass fraction which is dependent on the number of burnable poison rod assemblies (BPRAs), since these components restrict flow through the control rod guide tubes (CRGTs). For thermal-hydraulic analyses, the most limiting case is that with the higher bypass flow fraction, or smaller number of BPRAs.

The design basis chosen for cycle 9 thermal-hydraulic analyses was a full core of Zircaloy grid assemblies, containing 40 BPRAs, for which the core bypass flow is 8.8%. This design configuration was used to calculate the 1.77 DNBR value shown on Table 6-1. The actual cycle 9 core configuration consists of 60 fresh Zircaloy grid fuel assemblies, of which 52 contain BPRAs

(8.3% core bypass flow). The DNBR for this configuration, using the same core conditions presented in Table 6-1, is 1.80. The full Zircaloy grid core configuration is, therefore, conservative for cycle 9 DNBR analyses and a transition core penalty is not necessary. The reconstitutable upper end fitting (UEF) and the anti-straddle lower end fitting (LEF) were addressed in the evaluation.

The pressure-temperature safety limits have been recalculated using the BWC CHF correlation in the LYNXT[11] crossflow analysis. Table 6-1 provides a summary comparison of the DNB analysis parameters for cycles 8 and 9.

No rod bow penalty has been considered in the cycle 9 analysis based on the justification provided by reference 14. Reference 14 was verified as applicable for Zircaloy grid fuel assemblies in reference 4.

Table 6-1. Maximum Design Conditions, Cycles 8 and 9

	Cycle 8	Cycle 9
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, gpm	374880	374880
Core bypass flow, % [(a)]	8.4	8.8
DNBR modeling	Crossflow	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.65 cosine	1.65 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.97 [(c)]
Active fuel length, in. [(b)]	141.8	141.8
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	174	174
Max heat flux at 100% power, 10^3 Btu/h-ft^2	492	492
CHF correlation	B&W-2	BWC
CHF correlation DNB limit	1.3	1.18
Minimum DNBR		
at 112% power	2.08	1.77 [(c)]
at 102% power[(d)]	2.37	2.01

[(a)]Used in the analysis.

[(b)]Cold nominal stack height.

[(c)]Calculated for the instrument guide tube subchannel which is limiting for the Mark-B6 assembly.

[(d)]This represents initial condition DNBR for accident analyses.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 9 parameters to determine the effect of the cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effect of fuel densification on the FSAR accident results has been evaluated and are reported in reference 15. Since batch 11 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference report, the conclusions in that reference are still valid.

The radiological dose consequences of the accidents presented in Chapter 14 of the updated FSAR were re-evaluated for this reload report except for the waste gas tank rupture. The waste gas tank rupture was not reevaluated since Technical Specification 3.25.2.5 controls the maximum tank inventory on the basis of Xe-133 equivalent curie content such that the analysis of the event is not cycle dependent. The evaluation of the remaining events was made in order to incorporate more current plant data as well as the information in the updated FSAR.

All of the Cycle 9 accident doses are based on radionuclide sources calculated for the actual Cycle 9 core design and irradiation history. In addition, the bases used to analyze some of these accidents were changed to be consistent with the bases in the updated FSAR. The significant differences in the bases for the accident analysis between cycle 8 and cycle 9 are:

- o The atmospheric dispersion factors have been increased slightly.
- o Credit has been taken for the penetration room filter system in calculating the doses associated with the control rod ejection accident. (This makes the control rod ejection accident consistent with the LOCA and MHA.)

- o The iodine removal rate used to calculate the LOCA and MHA doses for Cycle 9 was changed to be consistent with the updated FSAR.

All of the calculated cycle 9 accident doses are below the dose acceptance criteria that are specified in the NRC's Standard Review Plan (NUREG-0800). Table 7-1 shows a comparison between cycle 8 and cycle 9 doses for the Chapter 14 accidents that result in significant offsite doses. With the exception of the maximum hypothetical accident (MHA), all doses are either bounded by the values reported for cycle 8 or are a small fraction of the 10CFR100 limits, i.e., below 30 Rem to the thyroid or 2.5 Rem to the whole body. For the MHA, the doses compare to the criteria as follows:

1. The 2-hour thyroid dose at the exclusion area boundary (EAB) is 165.1 Rem (55% of the NUREG-0800 limit).
2. The 2-hour whole-body dose at the EAB is 5.0 Rem (20% of the NUREG-0800 limit).
3. The 30 day thyroid dose at the low population zone (LPZ) is 87.8 Rem (29% of the NUREG-0800 limit).

The radiological doses from all of the accidents evaluated with the specific nuclide inventory from cycle 9 are lower than the NRC acceptance criteria of NUREG-0800, and thus are within acceptable limits.

7.2. Accident Evaluation

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

The core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Thermal parameters for fuel batches 6D, 9B, 10, and 11 are given in Table 4-2. The cycle 9 thermal-hydraulic maximum design conditions are compared with the previous cycle 8 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 9 are compared in Table 7-2.

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], BAW-10104[17], and BAW-1915P[18]). 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-1915P LOCA limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-1915P 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 9 fuel. These LHR limits include the effects of NUREG 0630, TACO2, and FLECSET.

It is concluded from the examination of cycle 9 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 9. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 9 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 9 are bounded by the FSAR, the fuel densification report, and/or subsequent cycle analyses.

Table 7-1. Comparison of Cycle 8 and Cycle 9 Accident Doses

	Cycle 8 doses, Rem	Cycle 9 doses, Rem
Fuel Handling Accident		
Thyroid dose at EAB (2 h)	1.15	1.12
Whole body dose at EAB (2 h)	0.21	0.22
Steam Line Break		
Thyroid dose at EAB (2 h)	1.71	1.82
Whole body dose at EAB (2 h)	0.008	0.01
Steam Generator Tube Failure		
Thyroid dose at EAB (2 h)	6.14	6.53
Whole body dose at EAB (2 h)	0.52	0.56
Control Rod Ejection Accident		
Thyroid dose at EAB (2 h)	12.2	7.02
Whole body dose at EAB (2 h)	0.008	0.006
Thyroid dose at LPZ (30 d)	9.09	5.64
Whole body dose at LPZ (30 d)	0.005	0.005
LOCA		
Thyroid dose at EAB (2 h)	4.02	4.22
Whole body dose at EAB (2 h)	0.026	0.03
Thyroid dose at LPZ (30 d)	2.05	2.47
Whole body dose at LPZ (30 d)	0.018	0.02
Maximum Hypothetical Accident		
Thyroid dose at EAB (2 h)	157.3	165.1
Whole body dose at EAB (2 h)	4.80	5.03
Thyroid dose at LPZ (30 d)	73.0	87.8
Whole body dose at LPZ (30 d)	1.56	1.78

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

Parameter	FSAR and Densification Report Value	ANO-1 Cycle 9
Doppler coeff (BOC), 10^{-4} $\Delta k/k/^{o}F$	-0.117	-0.159
Doppler coeff (EOC), 10^{-4} $\Delta k/k/^{o}F$	-0.130	-0.186
Moderator coeff (BOC), 10^{-4} $\Delta k/k/^{o}F$	0.0[a]	-0.58
Moderator coeff (EOC), 10^{-4} $\Delta k/k/^{o}F$	-4.0[b]	-2.82
All-rod group worth (HZP), % $\Delta k/k$	12.90	8.70
Initial boron concentration, ppm	1150	1379
Boron reactivity worth (HFP), ppm/% $\Delta k/k$	100	130
Max. ejected rod worth (HFP), % $\Delta k/k$	0.65	0.23
Dropped rod worth (HFP), % $\Delta k/k$	0.65	$\leq$0.20

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

(b) -3.0×10^{-4} $\Delta k/k/^{o}F$ 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, 0-1000 MWd/mtU, kW/ft	Allowable peak LHR, after 1000 MWd/mtU, kW/ft
2	14.0	15.5
4	16.1	16.6
6	16.5	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 9 operation for changes in core reactivity, power peaking, and control rod worths. The Technical Specifications were also revised to describe design features implemented with cycle 9. The cycle 9 design analysis basis includes the impact of extended periods of cycle 8 low-power operation, with cycle 8 power levels ranging between 65% and 100% of rated power. The cycle 9 basis also includes a very low leakage fuel cycle design, a mixed Mark B4/Mark B6 fuel assembly core, gray APSRs, gray APSR withdrawal flexibility, and crossflow analysis. The safety limits in Technical Specification Section 2 (Figures 8-1 through 8-3), have been changed for cycle-specific credits in the fuel cycle design, which allowed for additional operating margin beyond the generic limits used for cycle 8. Error adjusted trip setpoints for the reactor protection system are shown in Figure 8-4. The LOCA linear heat rate limits used to develop the Technical Specification Limiting Conditions for Operation include the impact of NUREG-0630 cladding swell and rupture model, and implement the credit from FLECSET analyses.[18]

A cycle 9 specific analysis was conducted to generate Technical Specification Limiting Conditions for Operation (rod index, axial power imbalance, and quadrant tilt), based on the methodology described in reference 19. The effects of gray APSR repositioning were included in the analysis, as was an APSR withdrawal flexibility window of +50/-10 EFPDs. The burnup-dependent allowable LOCA linear heat rate limits used in the analysis are provided in Figure 8-17. The analysis also determined that the cycle 9 Technical Specifications provide protection for the overpower condition that could occur during an overcooling transient because of nuclear instrumentation errors, and verified removal of the power level cutoff hold requirement.

Technical Specification section 3.5.2.4 was revised to accommodate a change in the quadrant tilt setpoint. The measurement system-independent rod position and axial power imbalance limits determined by the cycle 9 analysis were error-adjusted to generate alarm setpoints for power operation and are reflected in a Technical Specification revision to sections 3.5.2.5 and 3.5.2.6. The error adjusted alarm setpoints are provided in Figures 8-5 through 8-16. Technical Specification section 5.3.1 was revised to include the reconstitutable fuel assembly design and gray axial power shaping rods in the design features.

Based on the analyses and Technical Specification revisions described 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 the Technical Specifications.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the pressure/temperature line the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The B&W-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC).

A DNBR of 1.30 (B&W-2) or 1.18 (BWC) 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 for the allowable reactor coolant pump combination has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (3) with potential fuel densification effects:

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

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 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 position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak 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.

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 the DNBR limit 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 (B&W-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (B&W-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 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 B&W-2 or the BWC correlation continually increases from 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 1.30 (B&W-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (B&W-2) or 26 percent (BWC) for that particular reactor coolant pump situation. Curve 1 of Figure 2.1-3 is 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 curves.

REFERENCES

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

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

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

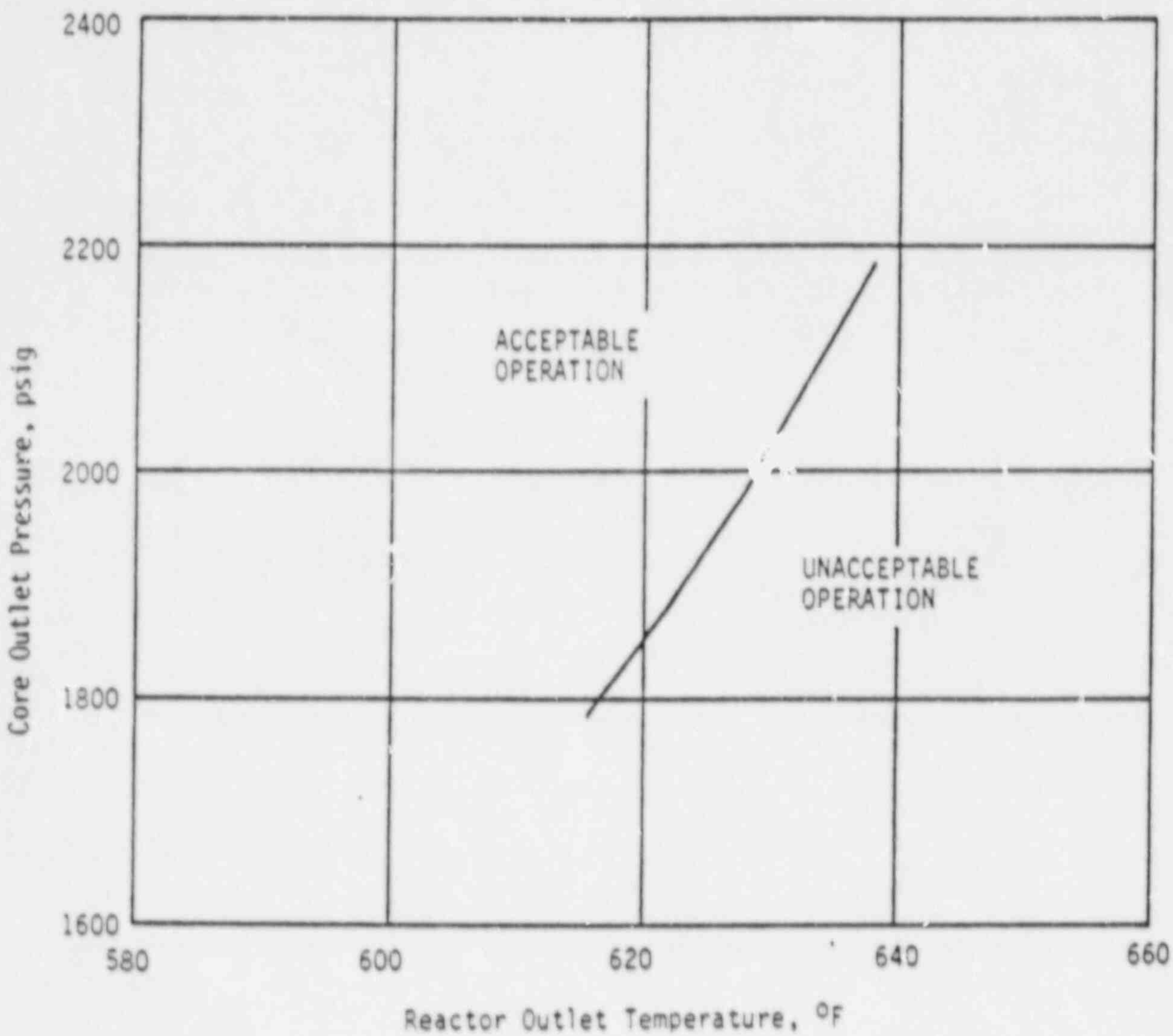


Figure 8-2. Core Protection Safety Limits -- ANO-1
(Tech Spec Figure 2.1-2)

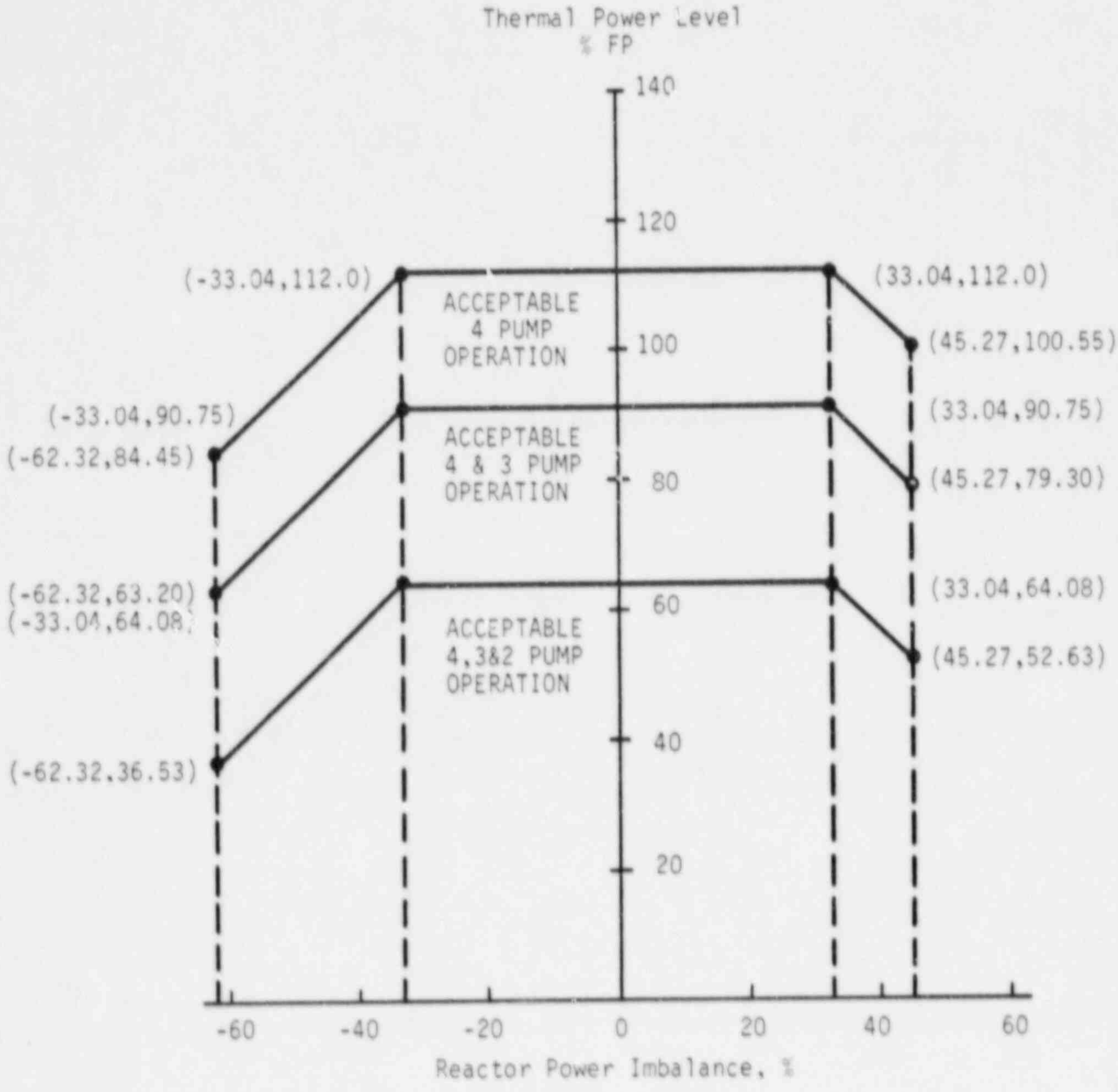
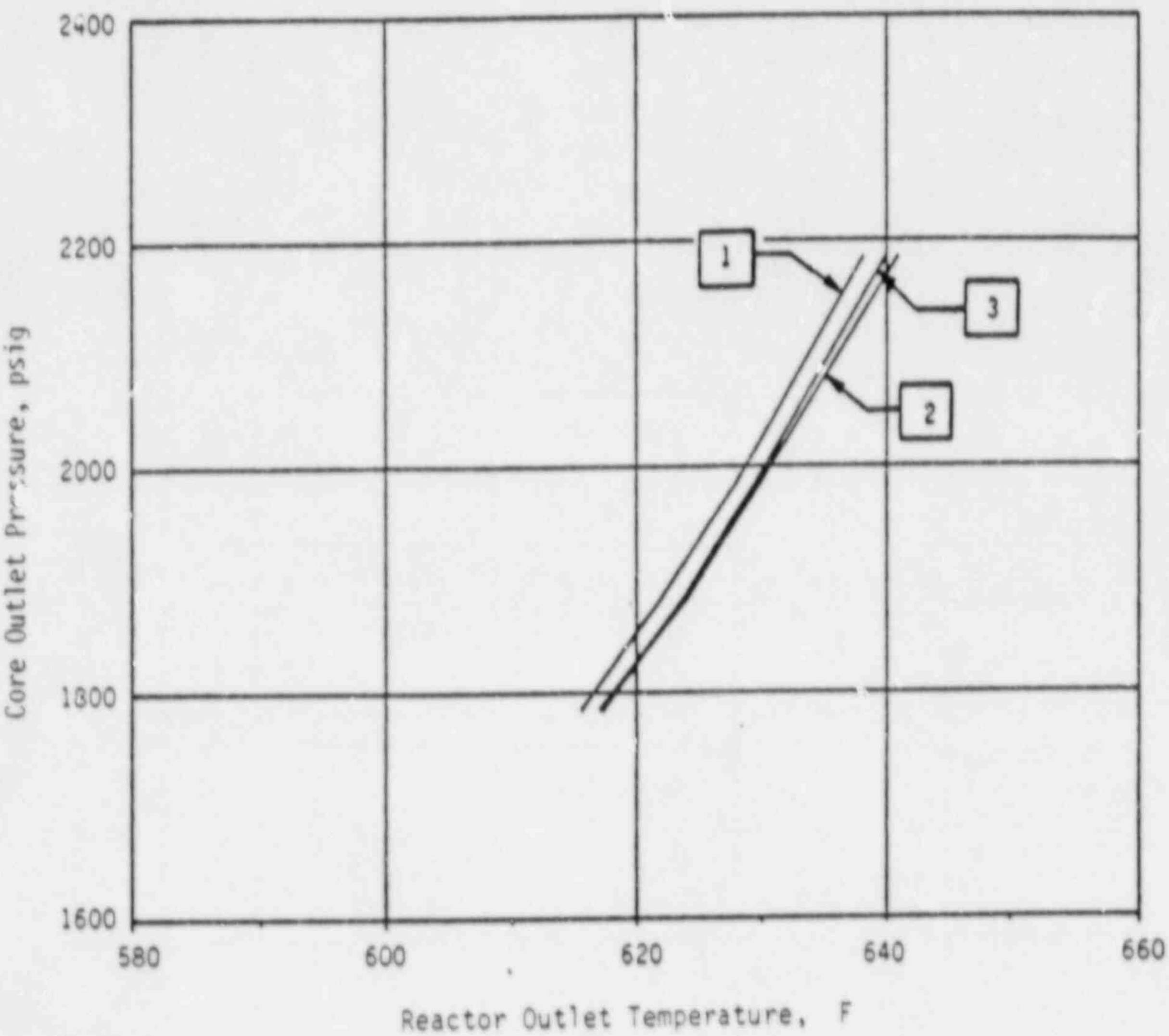


Figure 8-3. Core Protection Safety Limits - ANO-1
(Tech Spec Figure 2.1-3)



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%)*	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	90.8%	THREE PUMPS (QUALITY LIMIT)
3	184,441 (49.2%)	63.7%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

*106.5% OF DESIGN FLOW

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a preselected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip setpoint produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (B&W-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

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 rector 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 is 49.2 percent or flow rate is 45.8 percent and the power level is 49 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 the 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 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 1.30 (B&W-2) or 1.18 (BWC) 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 is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure ($11.75T_{out}$ -5103) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of ($11.75T_{out}$ -5143).

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620F.

E. Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

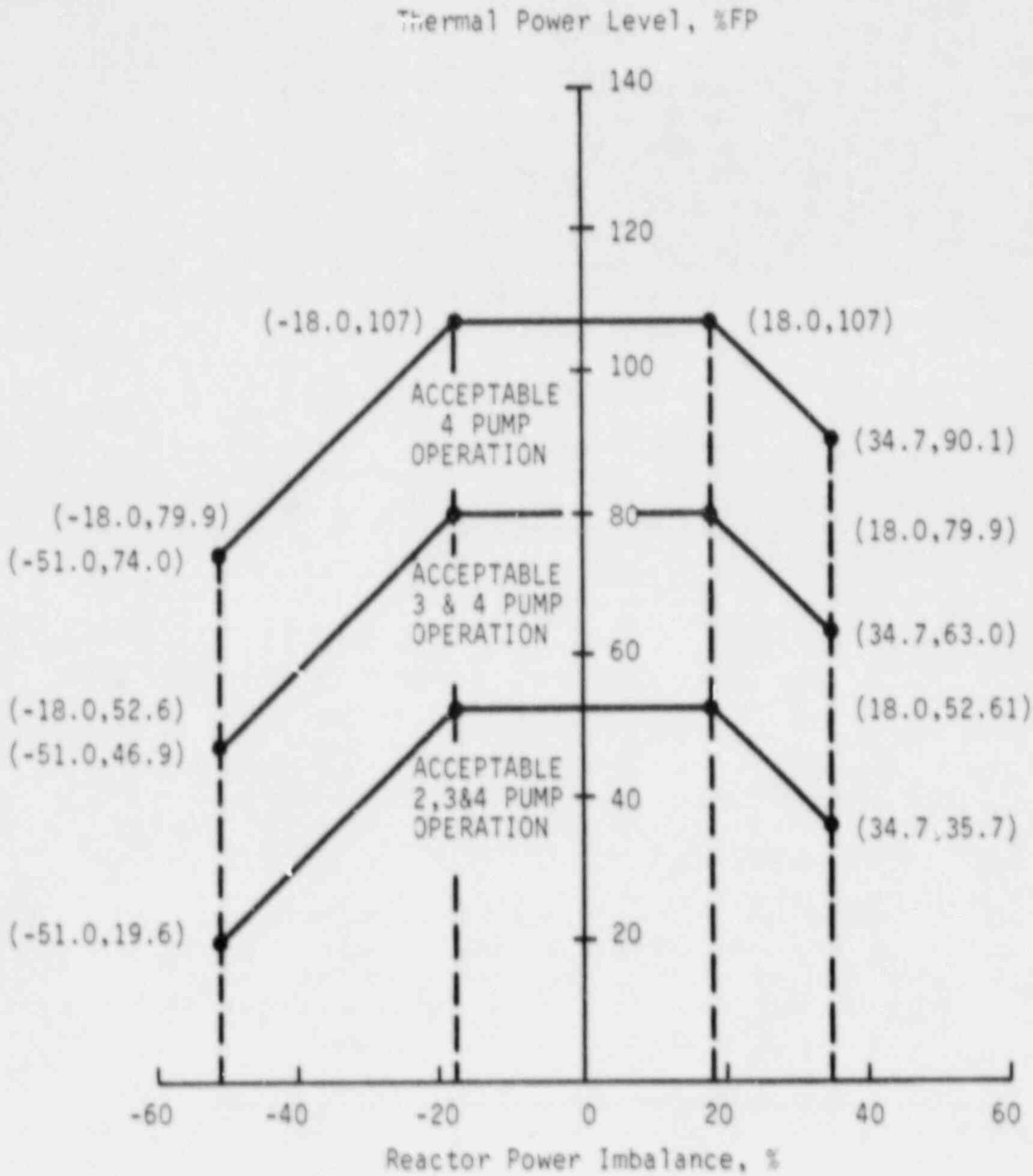
F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip setpoint of $\leq$5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

Figure 8-4. Protective System Maximum Allowable Setpoints -- ANO-1
(Tech Spec Figure 2.3-2)



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 4.12%, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of 4.12%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4.12% 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 4.12%.

   c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 4.12%.

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.

3. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1(A-C), 3.5.2-2(A-C), and 3.5.2-3(A-C) for 4, 3 and 2 pump operation respectively. If the applicable 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 or exercising axial power shaping rods (APSR's), the following limits apply to APSR position:

   Up to 410 EFPD, the APSR's may be positioned as necessary for transient imbalance control, however, the APSRs shall be fully withdrawn by 410 EFPD. After 410 EFPD, the APSR's shall not be reinserted.

   With the APSR's inserted after 410 EFPD, corrective measures shall be taken immediately to achieve the fully withdrawn position. Acceptable APSR positions shall be attained within 4 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(A-C). If the imbalance is not within the envelope defined by Figure 3.5.2-4(A-C), 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 Superintendent.

Bases

The power-imbalance envelope defined in Figure 3.5.2-4(A-C) is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria. 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

Figure 8-5. Rod Position Setpoints for Four-Pump Operation from 0 to 27 +10/-0 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-1A)

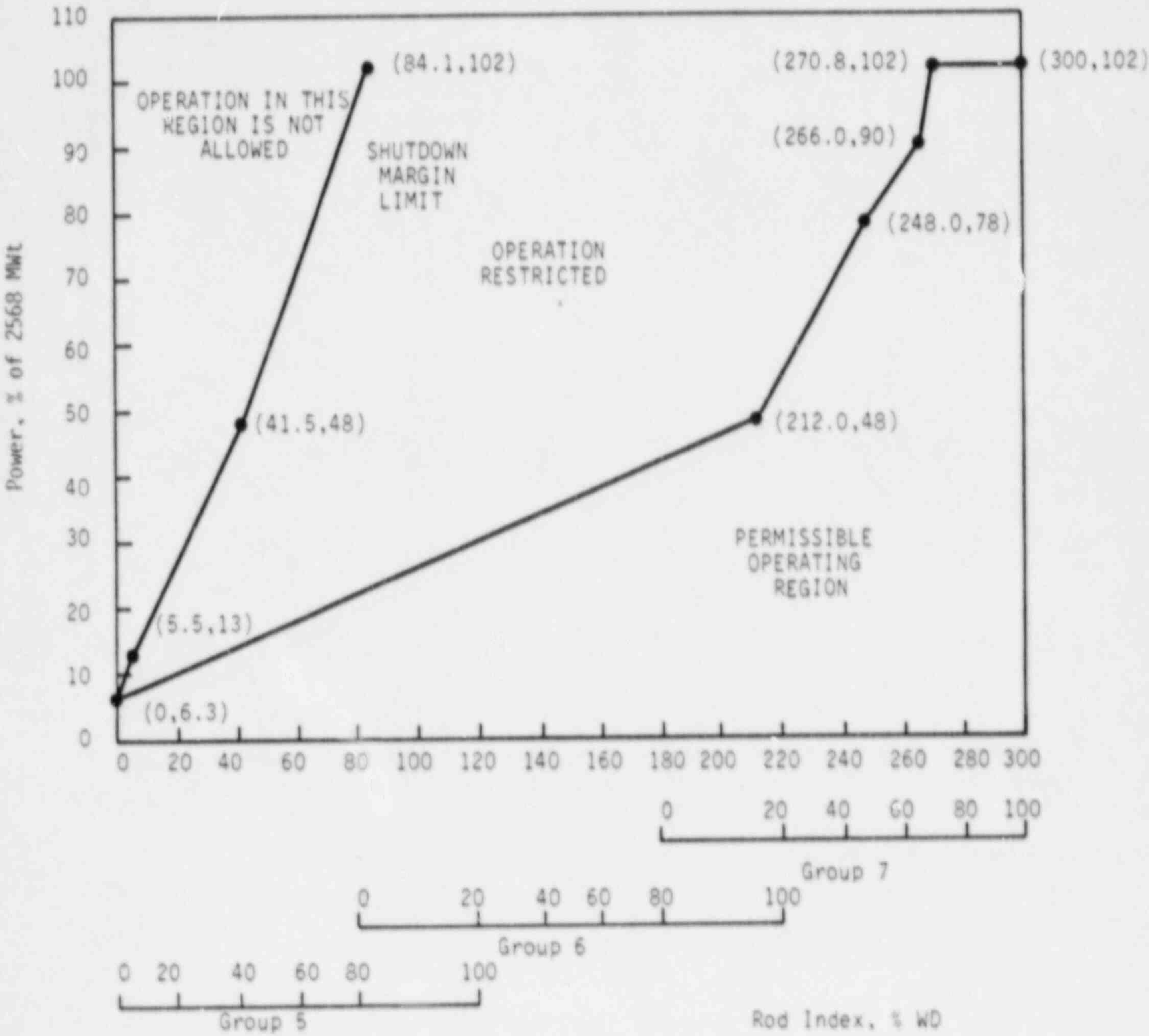


Figure 8-6. Rod Position Setpoints for Four-Pump Operation from 27 +10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-1B)

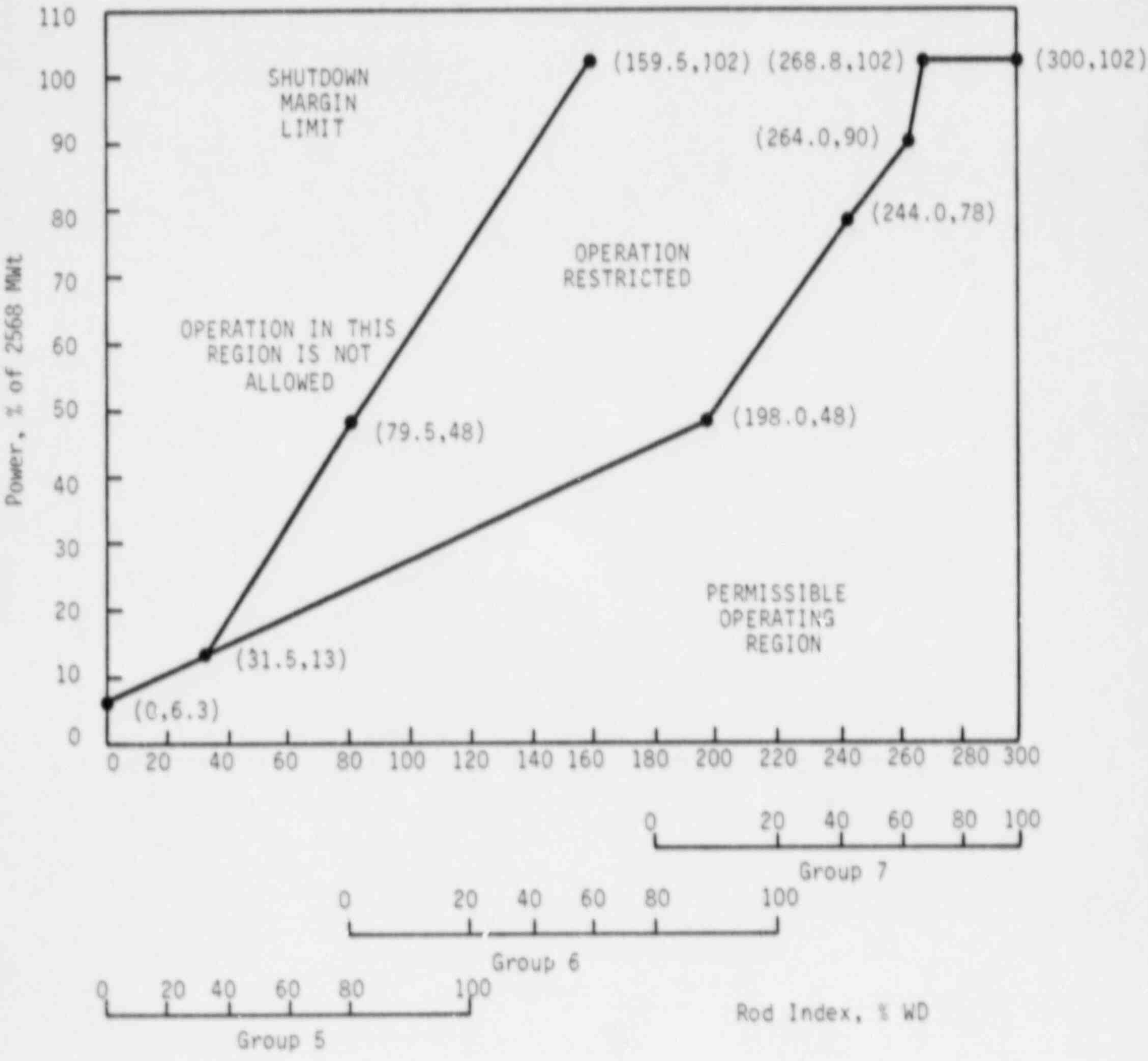


Figure 8-7. Rod Position Setpoints for Four-Pump Operation After 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-1C)

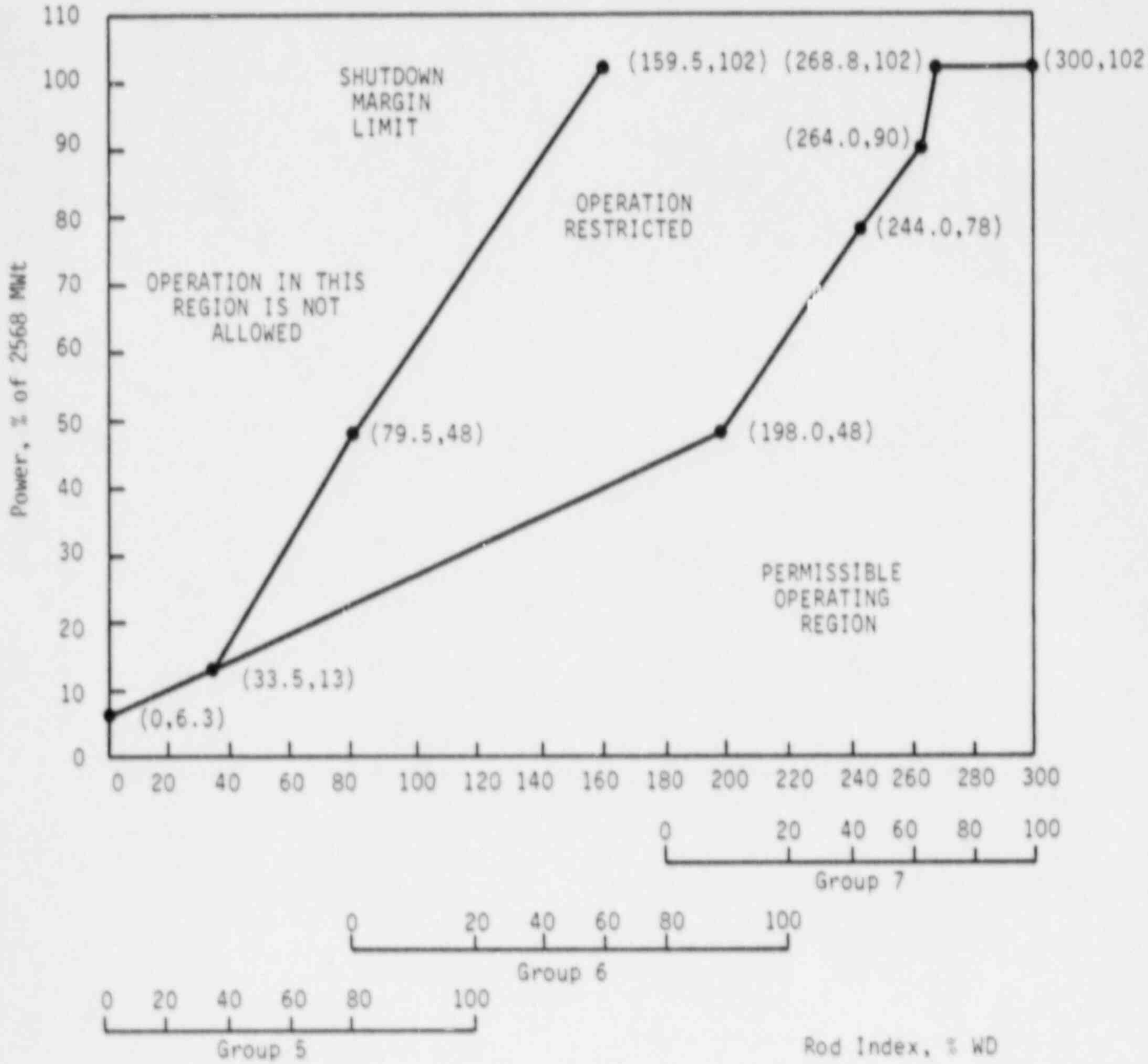


Figure 8-8. Rod Position Setpoints for Three-Pump Operation From 0 to 27 +10/-0 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-2A)

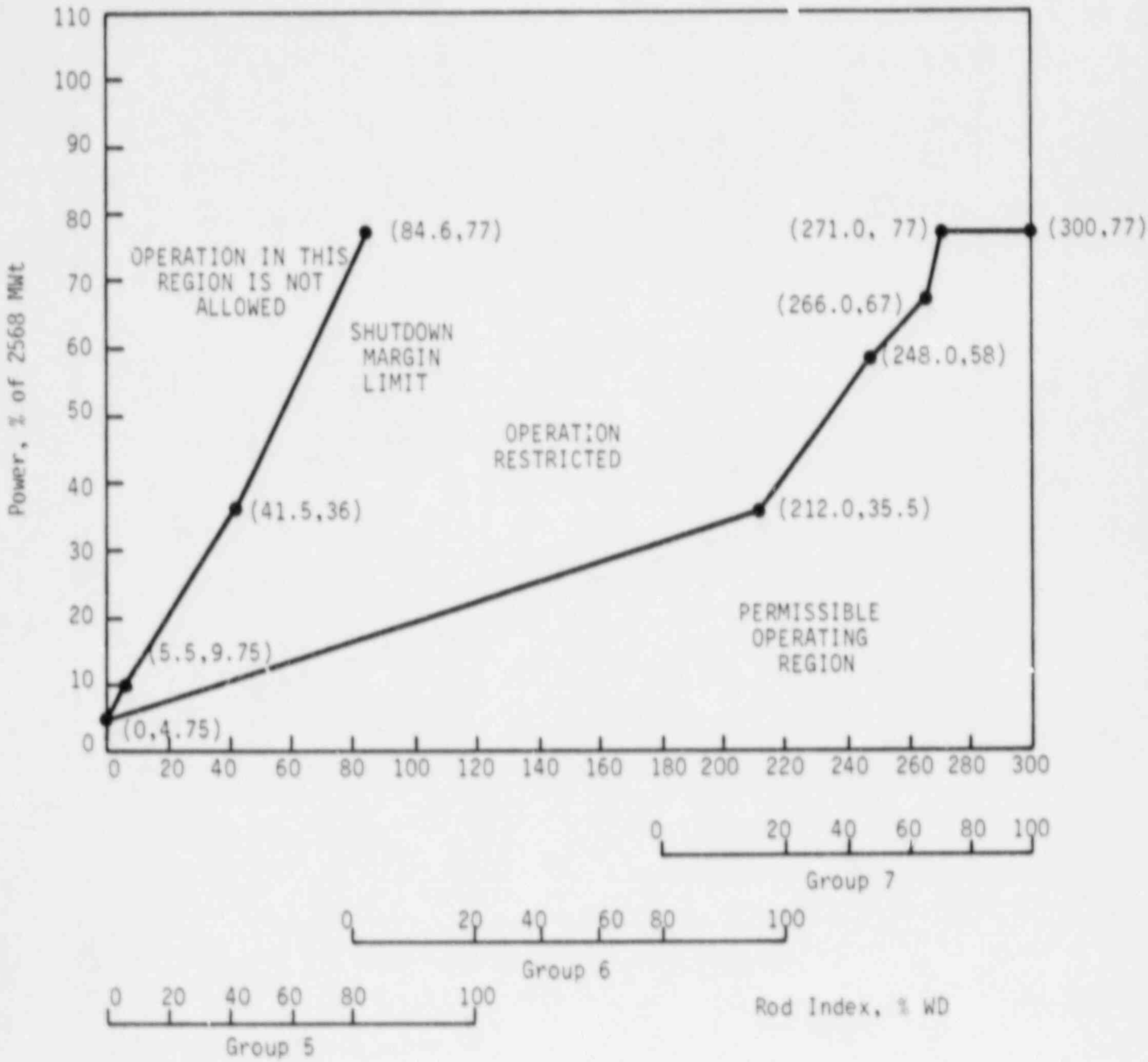


Figure 8-9. Rod Position Setpoints for Three-Pump Operation From 27 +10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-2B)

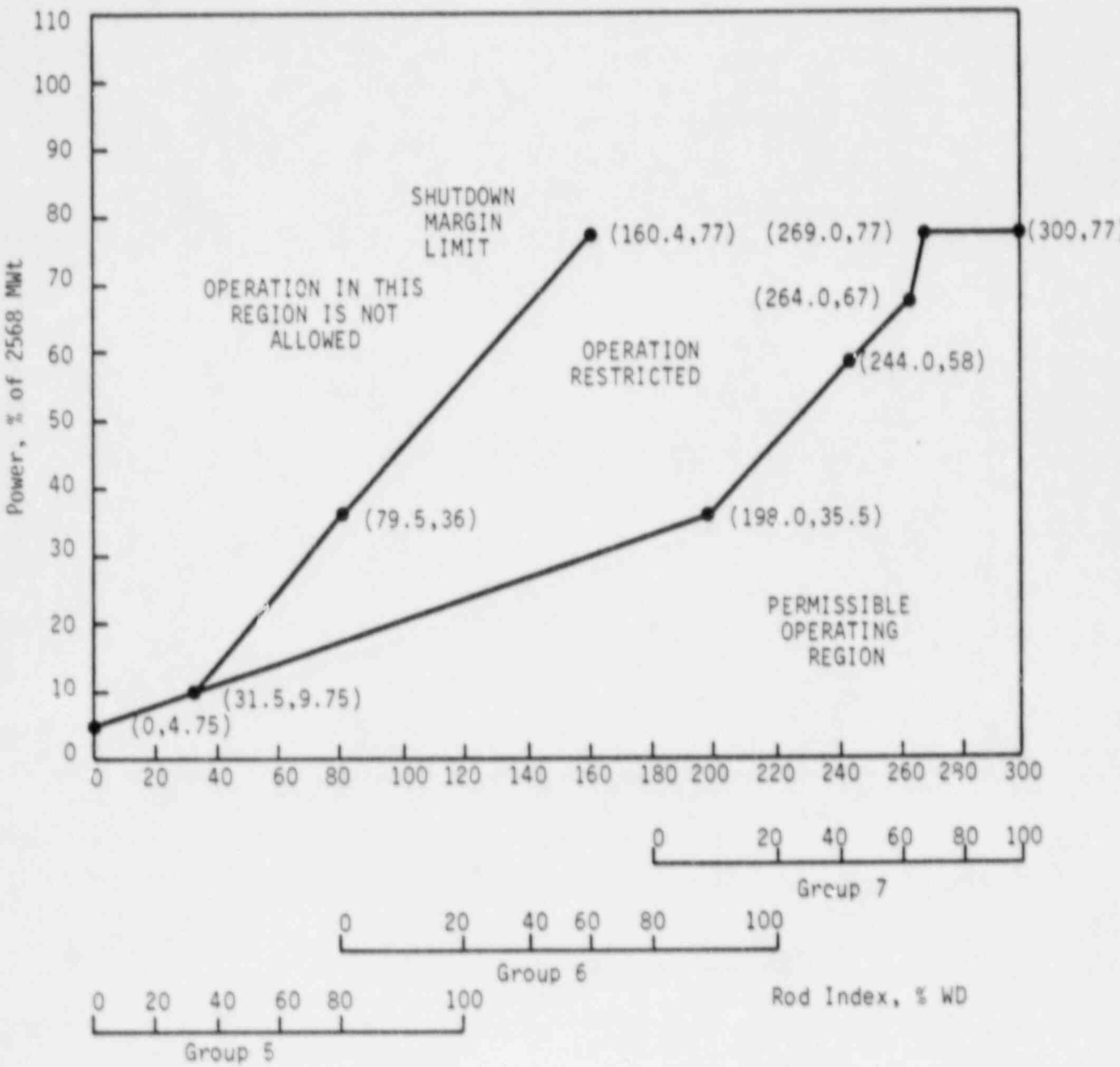


Figure 8-10. Rod Position Setpoints for Three-Pump Operation After 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-2C)

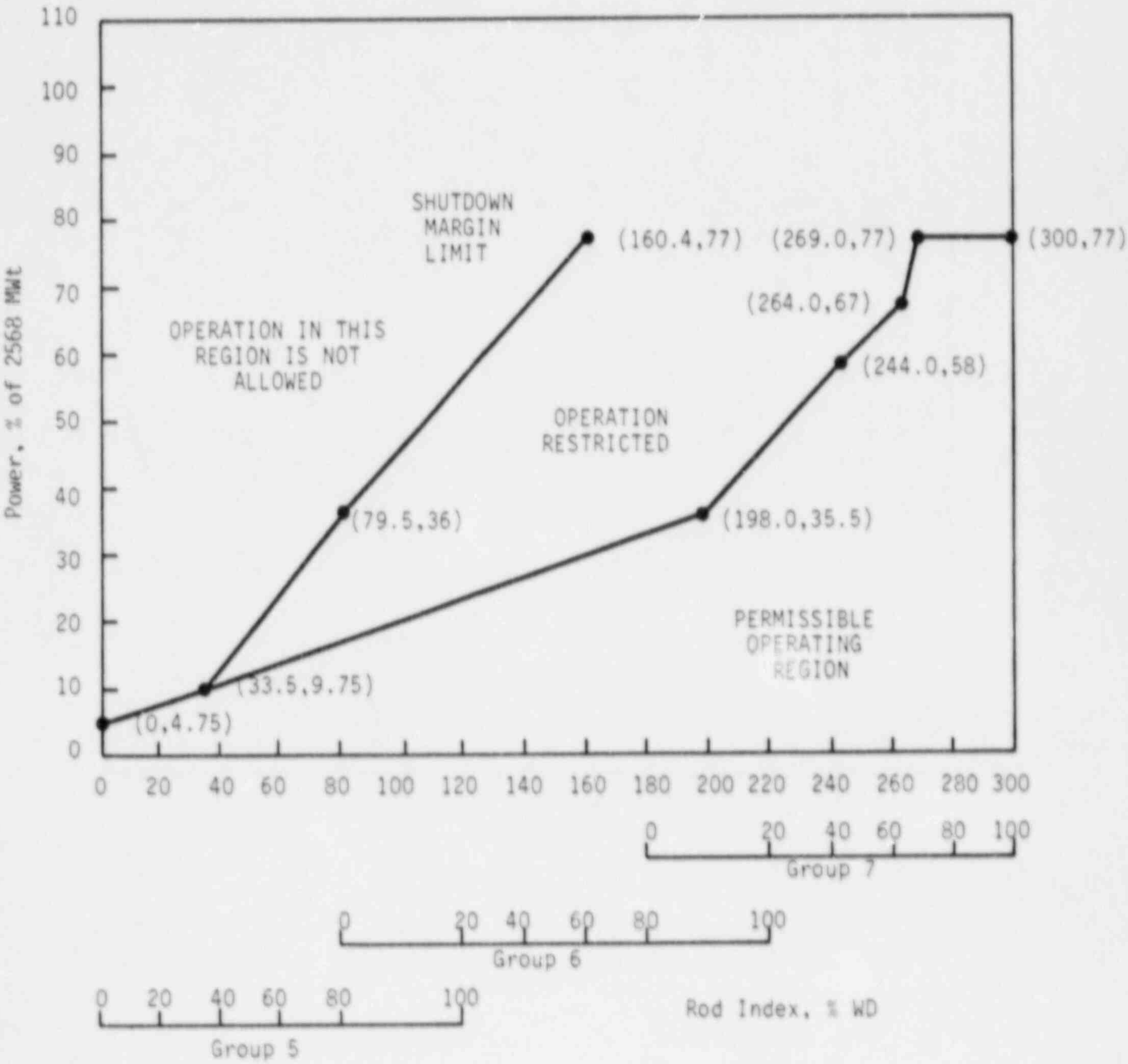


Figure 8-11. Rod Position Setpoints for Two-Pump Operation From 0 to 27 +10/-0 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-3A)

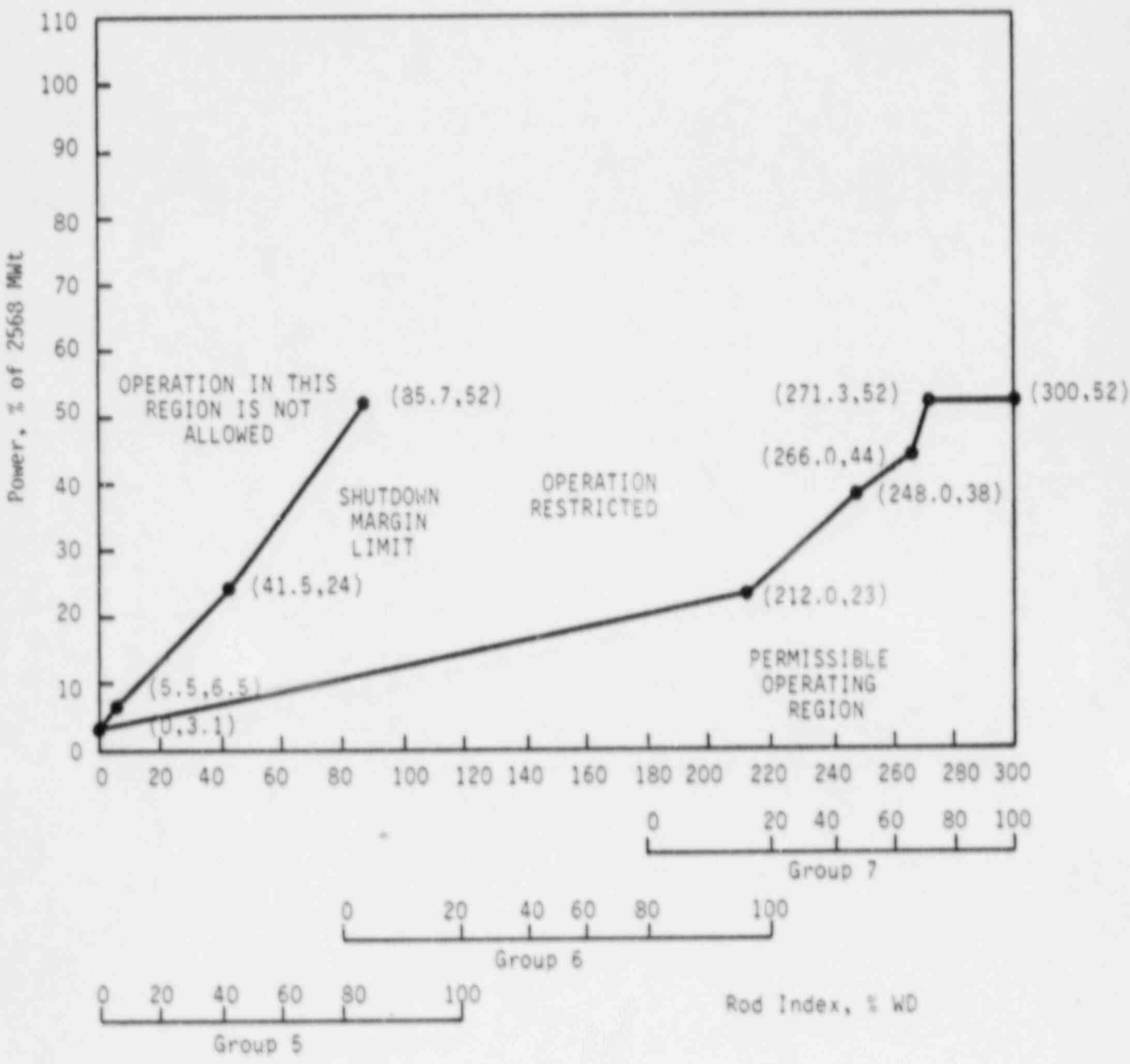


Figure 8-12. Rod Position Setpoints for Two-Pump Operation From 27 +10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-3B)

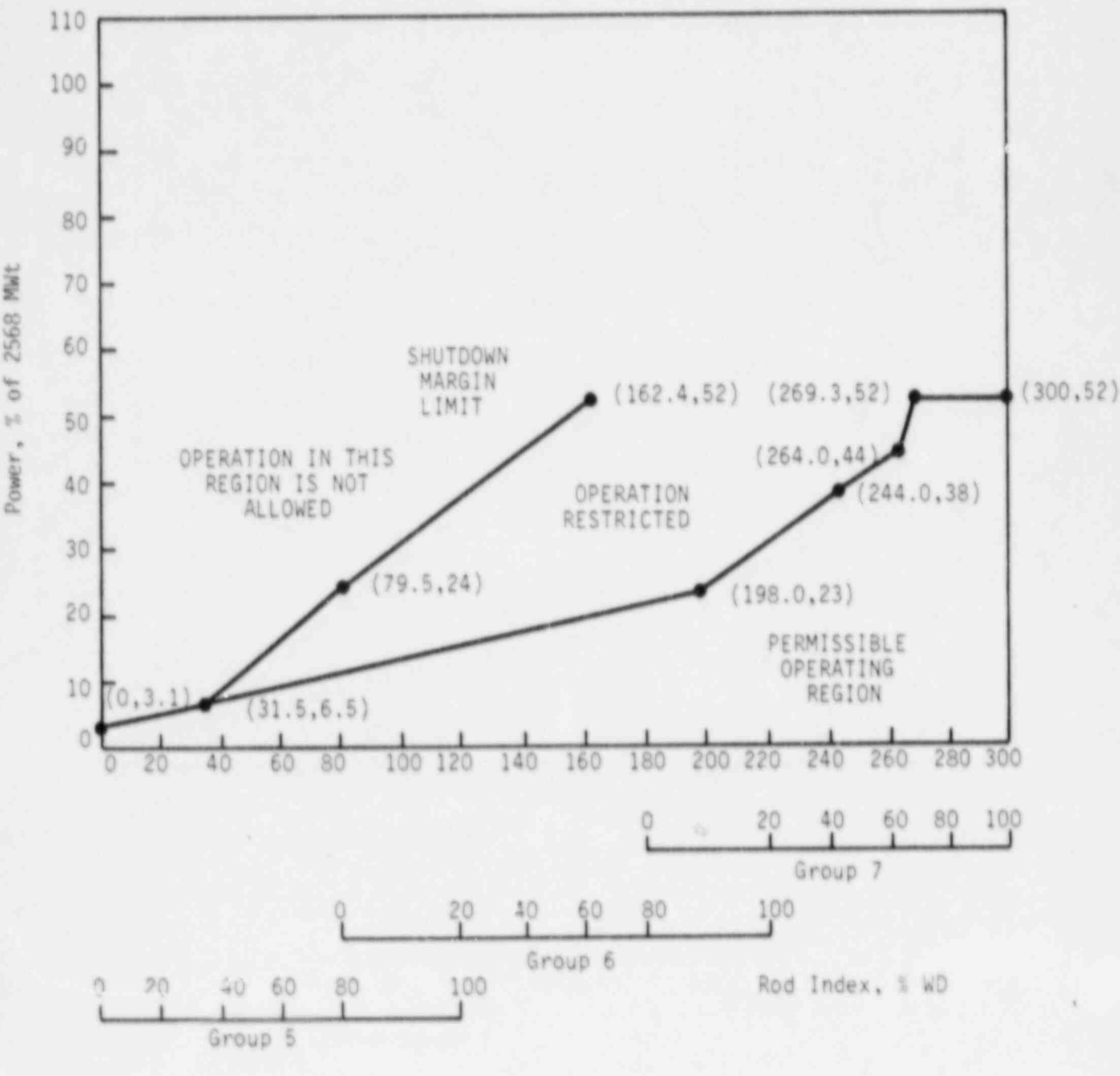


Figure 8-13. Rod Position Setpoints for Two-Pump Operation After 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-3C)

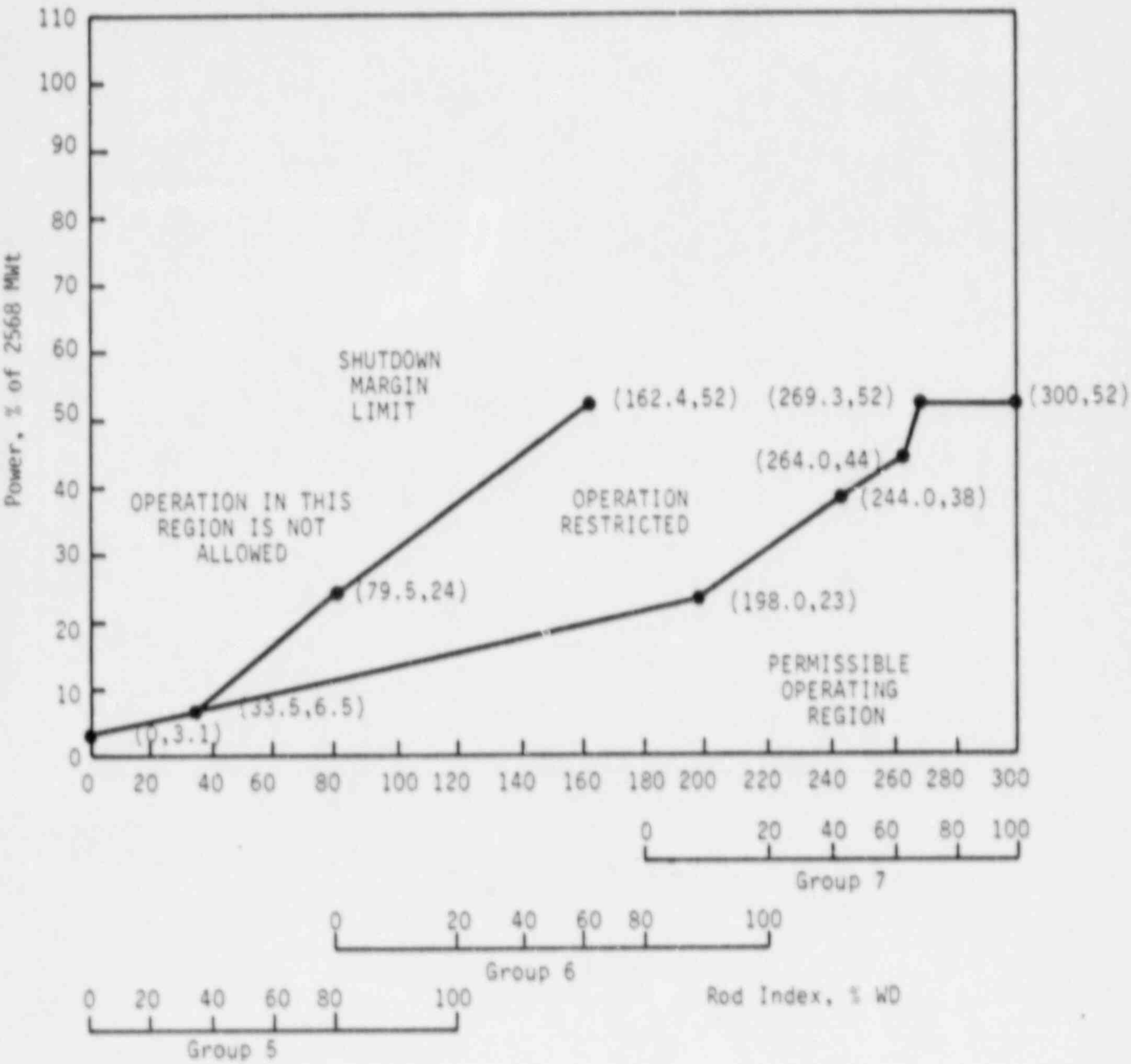


Figure 8-14. Operational Power Imbalance Setpoints for Operation From 0 to 27 +10/-0 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-4A)

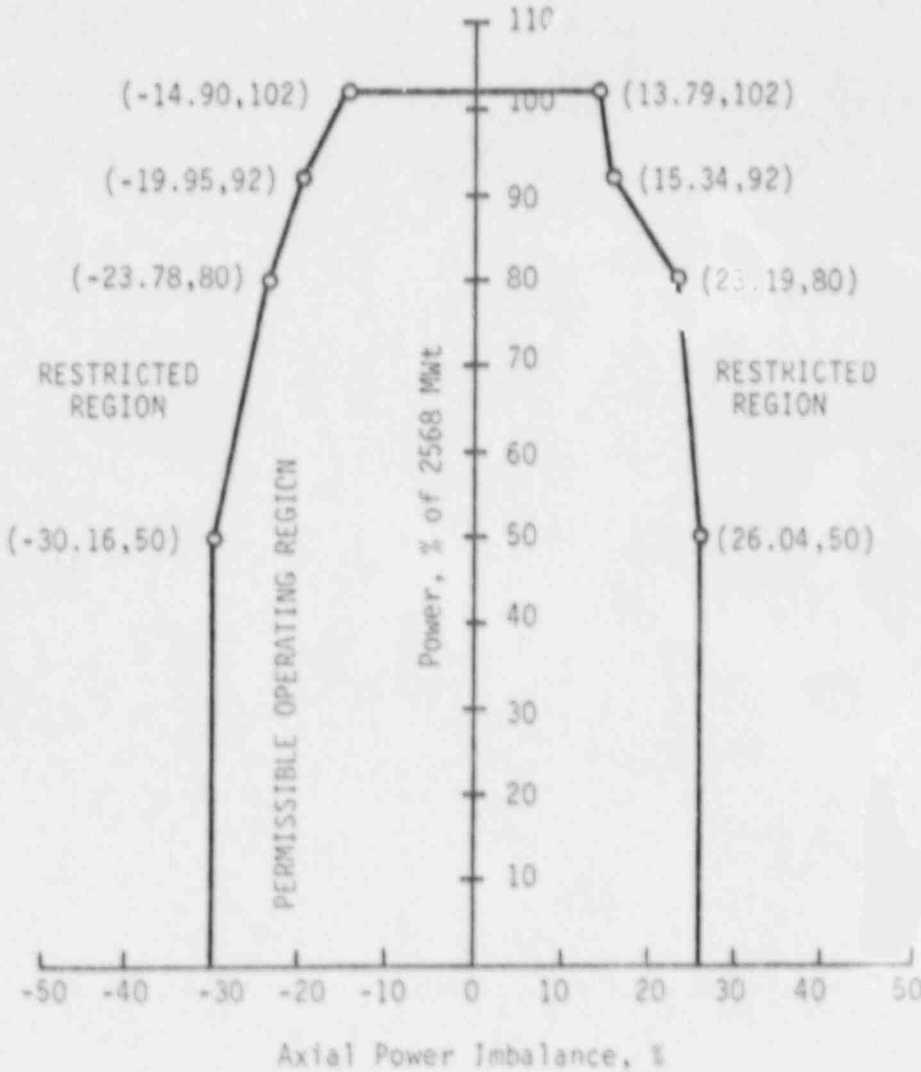


Figure 8-15. Operational Power Imbalance Setpoints for Operation From 27 +10/-0 to 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-4B)

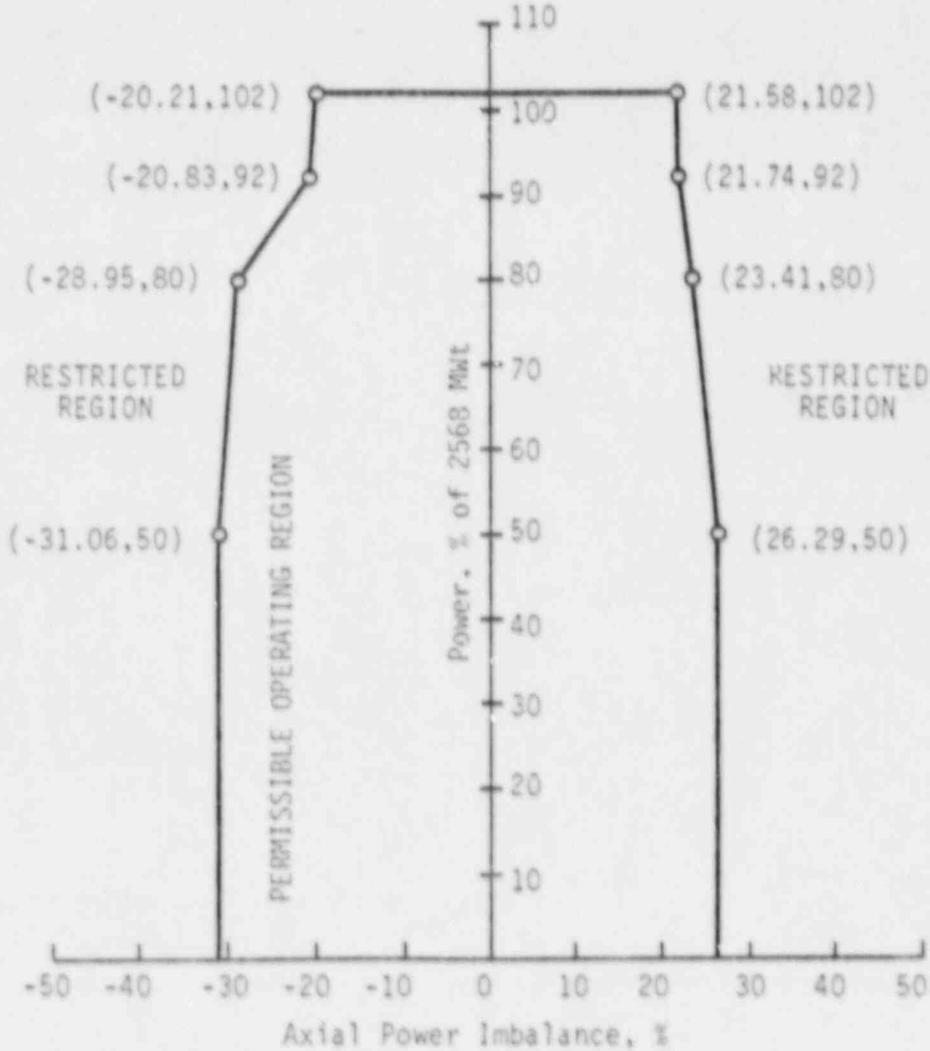


Figure 8-16. Operational Power Imbalance Setpoints for Operation After 360 +50/-10 EFPD -- ANO-1 Cycle 9 (Tech Spec Figure 3.5.2-4C)

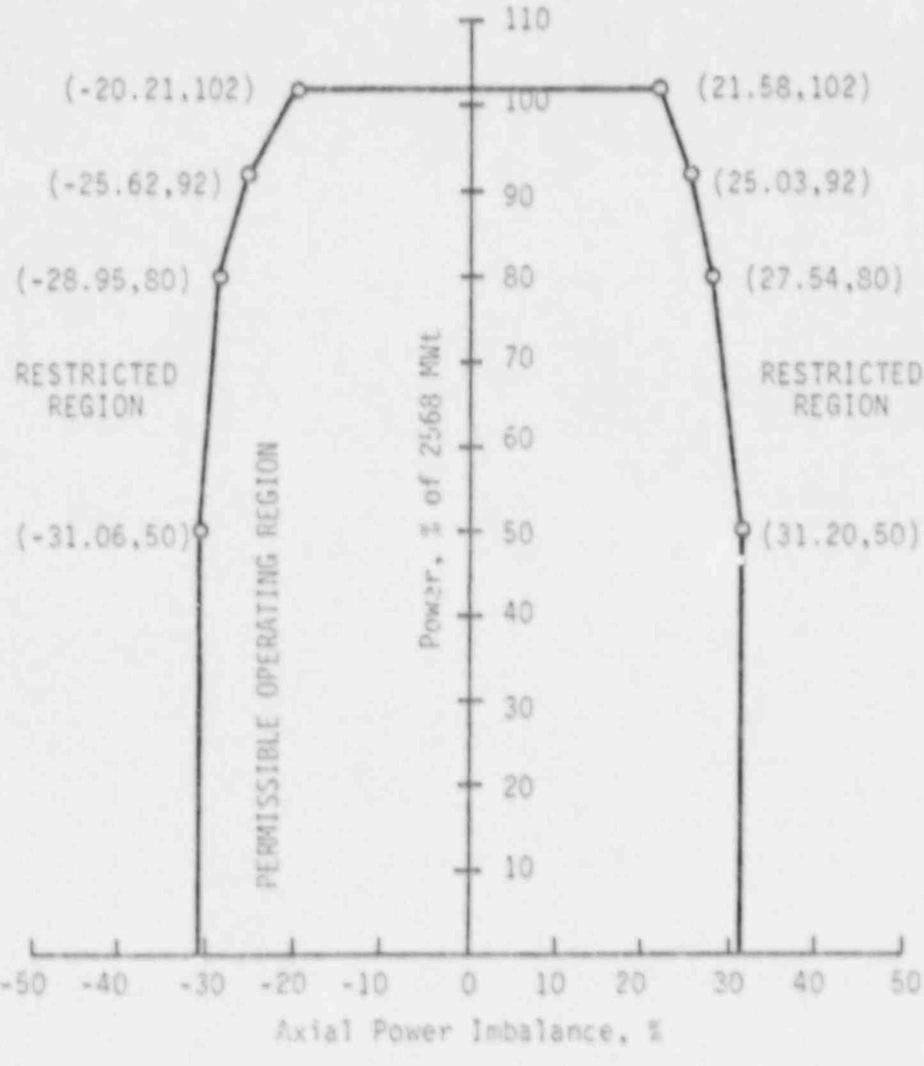
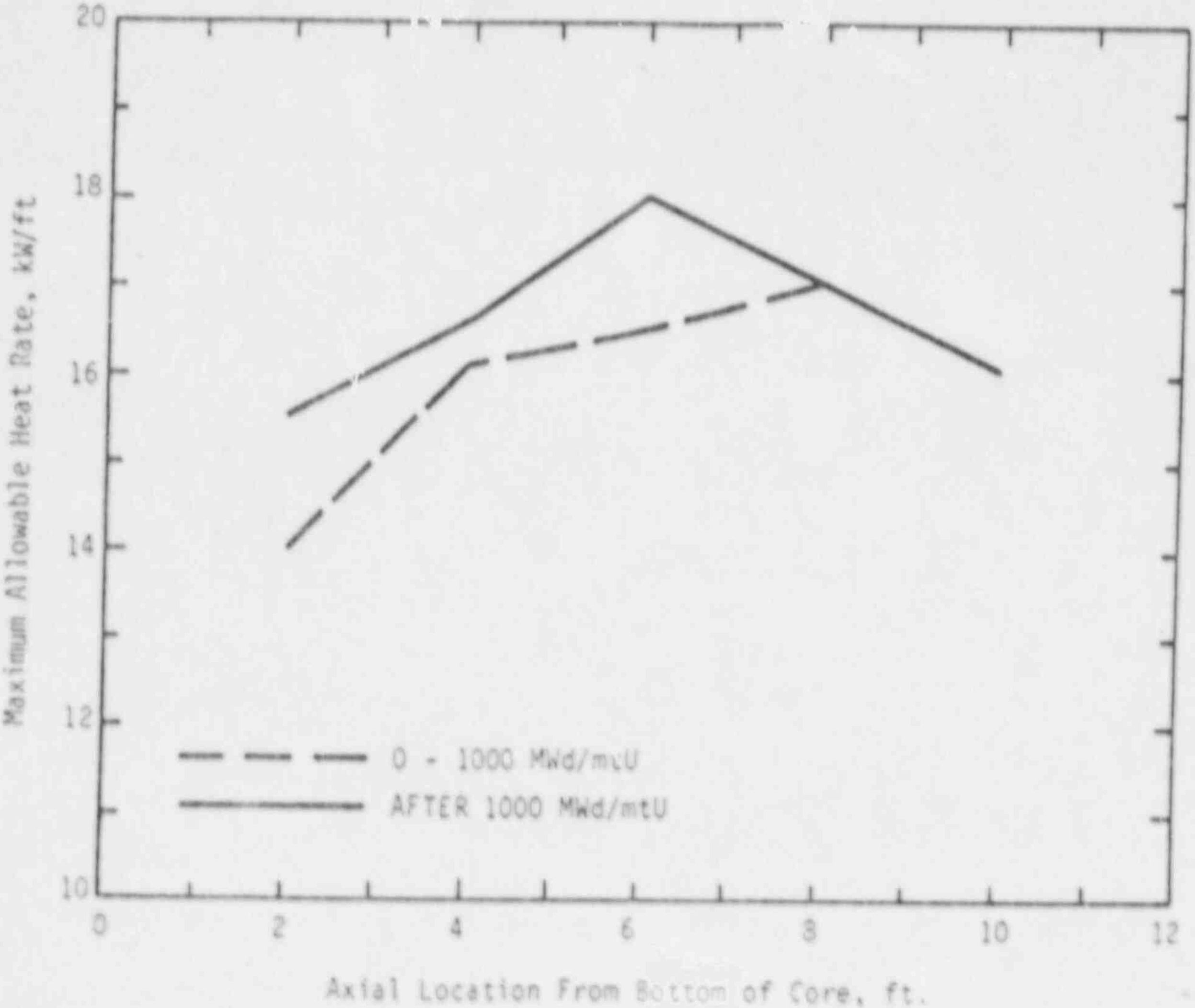


Figure 8-17. LOCA Limited Maximum Allowable Linear Heat Rate
ANO-1 -- Cycle 9 (Tech Spec Figure 3.5.2-5)



5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies. Each fuel assembly is fabricated with 208 fuel rods.[1,2] Starting with Batch 11. a reconstitutable fuel assembly design is implemented. This design allows the replacement of up to 208 fuel rods in the assembly.

5.3.1.2 The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and a height of 144 inches. The active fuel length is approximately 142 inches.[2]

5.3.1.3 The average enrichment of the initial core is a nominal 2.62 weight percent of ^{235}U. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent ^{235}U.

5.3.1.4 There are 60 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3-60. Each full-length CRA contains a 134-inch length of silver-indium-cadmium alloy clad with stainless steel. Each APSRA contains a 63-inch length of Inconel-600 alloy clad with stainless steel.[3]

5.3.1.5 The initial core has 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.

5.3.1.6 Reload fuel assemblies and rods shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 3.5 percent of ^{235}U.

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system is designed and constructed in accordance with code requirements.[4]

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.[5]

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

9. STARTUP PROGRAM - PHYSICS TESTING

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

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

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

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

9.1.2. RC Flow

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

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

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

9.2.2. Temperature Reactivity Coefficient

The isothermal HZP temperature coefficient is measured at approximately the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. The change in reactivity is then calculated by the summation of reactivity (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 $\pm$ 0.4x10^{-4} $\Delta k/k/^{o}F$.

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of the fuel Doppler coefficient of reactivity is added to obtain the moderator coefficient. This value must not be in excess of the acceptance criteria limit of +0.5x10^{-4} $\Delta k/k/^{o}F$.

9.2.3. Control Rod Group/Boron Reactivity Worth

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

1. Individual bank 5, 6, 7 worth:

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

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

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

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

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

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

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

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

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

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

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

1. The worst-case maximum LHR must be less than the LOCA limit.
2. The minimum DNBR must be greater than the initial condition DNBR limit.
3. The value obtained from extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than the initial condition DNBR limit or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.
4. The value obtained from extrapolation of the worst-case maximum LHR to the next power plateau overpower trip setpoint must be less than the fuel melt limit, or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.
5. The quadrant power tilt shall not exceed the limits specified in the Technical Specifications.

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

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

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

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

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

Items 3 and 4 establish the criteria whereby escalation to full power may be accomplished without the potential for exceeding the safety limits at the overpower trip setpoint with regard to DNBR and linear heat rate.

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

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

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

9.3.4. Temperature Reactivity Coefficient at ~100% FP

The average reactor coolant temperature is decreased and then increased 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.

9.3.5. Power Doppler Reactivity Coefficient at ~100% FP

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

The fuel Doppler reactivity coefficient is calculated in conjunction with the power Doppler coefficient measurement. The power Doppler coefficient as measured above is multiplied by a precalculated conversion factor to obtain the fuel Doppler coefficient. This measured fuel Doppler coefficient must be more negative than the acceptance criteria limit of $-0.90 \times 10^{-5}\ \Delta k/k/^{o}F$.

9.4. Procedure for Use if Acceptance Criteria Not Met

If acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. Further specific actions depend on evaluation results. These actions can include repeating the tests with more detailed attention to test prerequisites, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

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