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OCONEE UNIT 2, CYCLE 2

— Reload Report —

Revision 1

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Babcock & Wilcox

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OCONEE UNIT 2, CYCLE 2

– Reload Report –

Revision 1

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the second cycle of Oconee Nuclear Station, Unit 2, at the rated core power of 2568 MWt. Included are the required analyses outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," dated June 1975. To support cycle 2 operation of Oconee Unit 2, this report employs analytical techniques and design bases established in reports that have been submitted and accepted by the USNRC (see references).

Cycle 1 and 2 reactor parameters related to power capability are summarized briefly in section 5. All the accidents analyzed in the FSAR have been reviewed for cycle 2 operation. In cases where cycle 2 characteristics proved to be conservative with respect to those analyzed for cycle 1 operation, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 2 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 Oconee 2, cycle 2, can be operated safely at the rated core level of 2568 MWt.

2. OPERATING HISTORY

Unit 2 of the Oconee Nuclear Station achieved initial criticality on November 11, 1973, and power escalation commenced on December 1, 1973. The 100% power level of 2568 MWt was reached on June 19, 1974. A control rod interchange was performed at 248 effective full-power days (EFPD). The fuel cycle was terminated on April 7, 1976, after 440 EFPD. The first cycle involved no operating anomalies that would adversely affe fuel performance during the second cycle. 1

Operation of cycle 2 is scheduled to begin in early June 1976. The design cycle length is 306 EFPD, and no control rod interchanges are planned. 1

3. GENERAL DESCRIPTION

The Oconee Unit 2 reactor core is described in detail in Chapter 3 of the Unit 2 FSAR.[1] The cycle 2 core consists of 177 fuel assemblies, 175 of which are 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel rod cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished-end, cylindrical pellets of uranium dioxide which are 0.700 inch in length and 0.370 inch in diameter. (See Tables 4-1 and 4-2 for additional data.) The other two fuel assemblies in cycle 2 are demonstration 17 by 17 Mark C fuel assemblies.[2] All fuel assemblies in cycle 2 except the 17 by 17 demonstration assemblies maintain a constant nominal fuel loading of 463.6 kg of uranium. The undensified nominal active fuel lengths and theoretical densities vary between batches, however, and these values are given in Tables 4-1 and 4-2.

Figure 3-1 is the core loading diagram for Oconee Unit 2, cycle 2. The initial enrichments of batches 2 and 3 were 2.75 and 3.05 wt % uranium-235, respectively. Batch 4 is enriched to 2.64 wt % uranium-235. All the batch 1 assemblies will be discharged at the end of cycle 1. The batch 2 and 3 assemblies will be shuffled to new locations. The batch 4 assemblies will occupy primarily the periphery of the core and eight locations in its interior. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 2.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods and soluble boron shim. In addition to the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 2 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) for cycle 2 are identical to those of the reference cycle indicated in Chapter 3 of the FSAR.[1] However, the group designations differ between cycle 2 and the reference cycle to minimize power peaking. Neither control rod interchange nor burnable poison rods are necessary for cycle 2.

The nominal system pressure is 2200 psia, and the core average densified nominal heat rate is 5.78 kW/ft at the rated core power of [illegible] MWt. This heat rate is slightly higher than that of cycle 1 because of the shorter stack height of batch 4.

Figure 3-1. Oconee 2, Cycle 2 - Core Loading Diagram

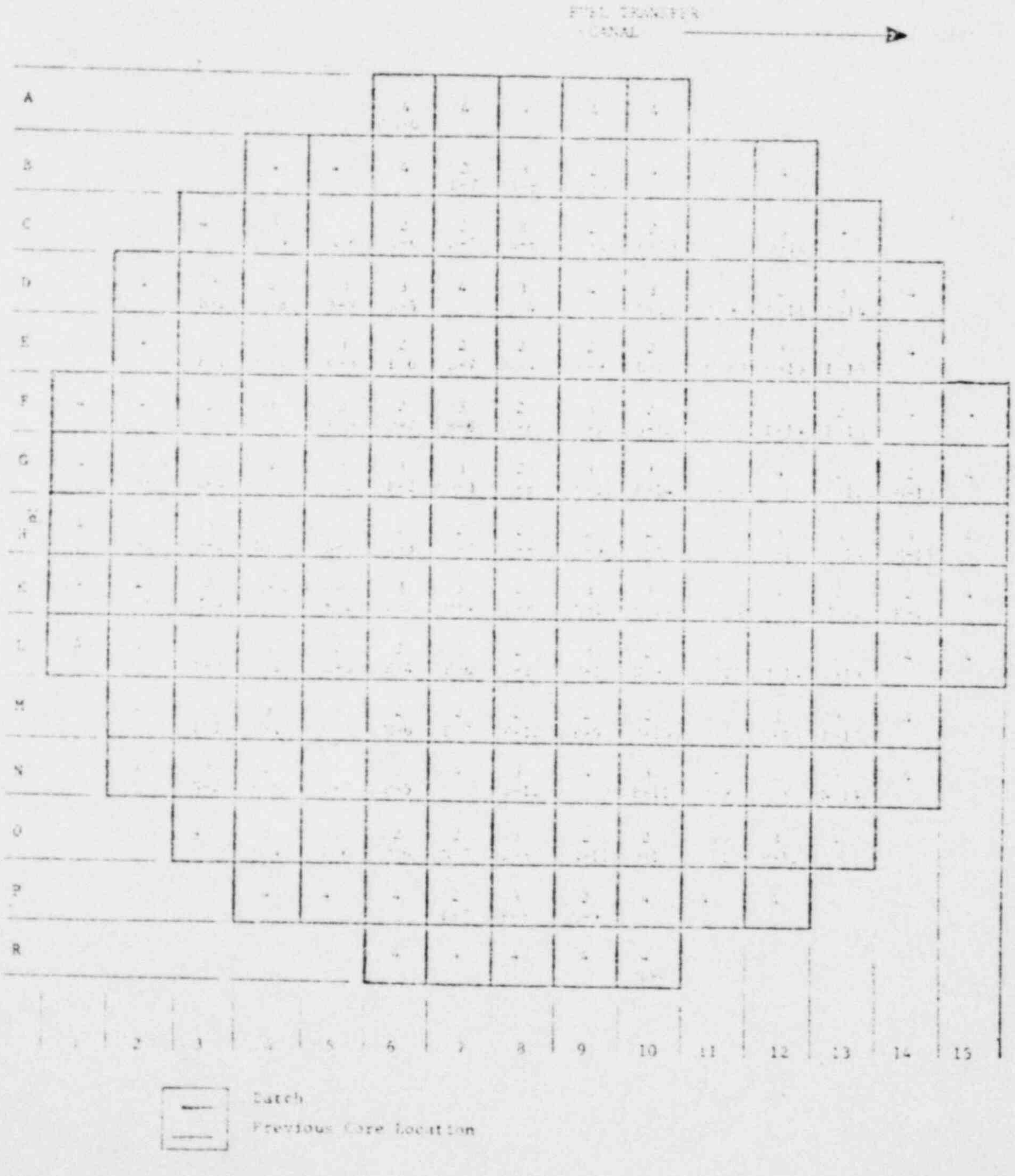


Figure 3-2. Oconee 2 Enrichment and Burnup Distribution for Cycle 2

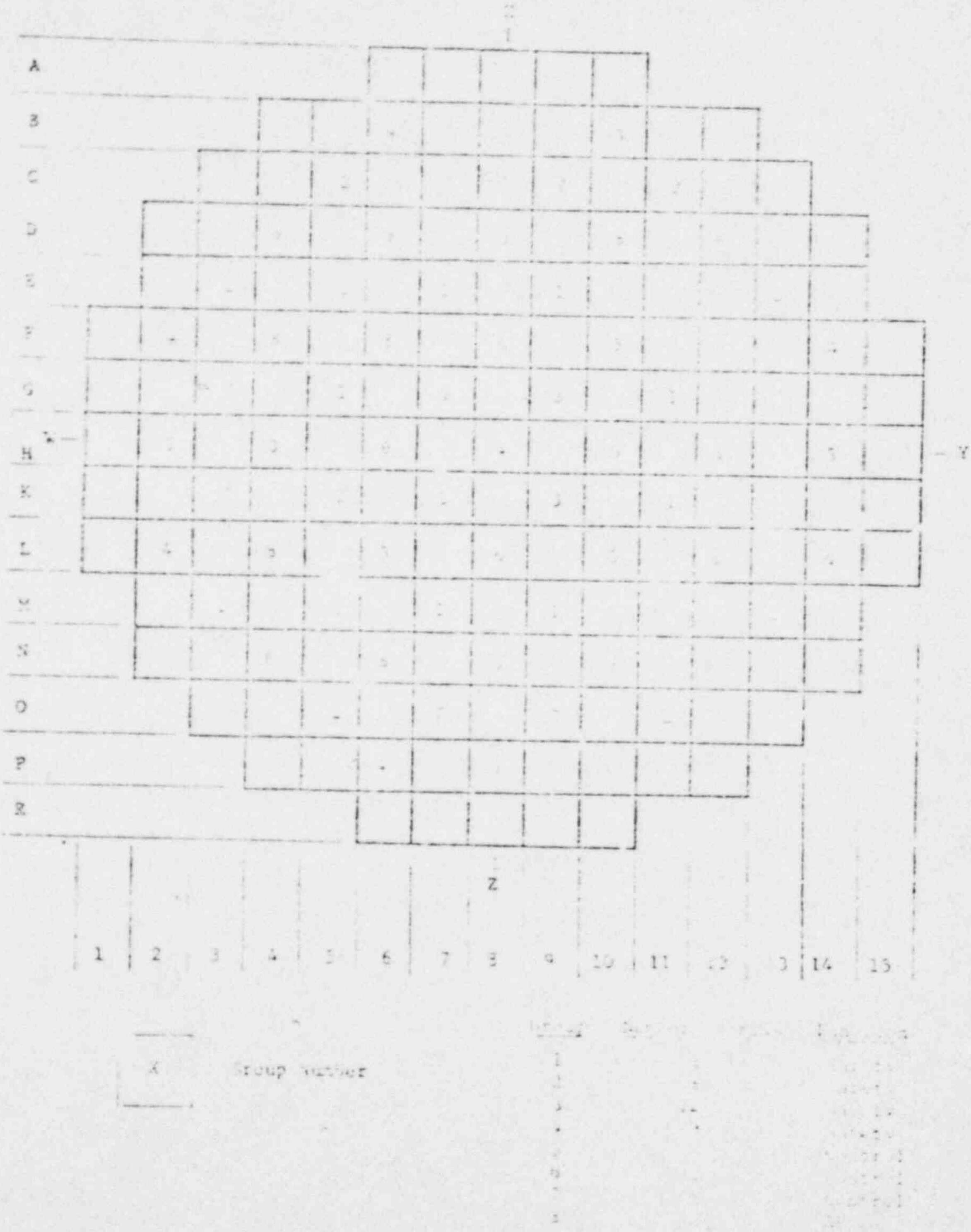
	8	9	10	11	12	13	14	15
H	2.75 11,956	2.75 16,711	2.75 16,282	2.75 16,941	3.05 11,660	3.05 14,01[illegible]	3.05 10,694	2.64 0
K		3.05 11,660	3.05 12,001	2.75 13,494	2.64 0	2.75 14,851	2.75 18,137	2.64 0
L			2.75 16,941	2.75 17,203	3.05 11,233	2.75 14,681	2.64 0	2.64 0
M				3.05 8318	3.05 10,267	3.05 7886	2.64 0	
N					2.75 18,206	3.05 7825	2.64 0	
O						2.64 0		
P								
R								

xxx xxxx

xxx Initial Enrichment

xxxx BOC Burnup, MWd/mtU

Figure 3-1. [illegible] Control [illegible] Locations



4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

Pertinent fuel design parameters are listed in Table 4-1. All fuel assemblies are mechanically interchangeable, but the Mark C demonstration assemblies cannot be located in a rodded location. The 54 new Mark B-4 fuel assemblies incorporate minor modifications to the end fittings, primarily to reduce fuel assembly pressure drop and increase holddown margin. Two of the 54 Mark B-4 reload fuel assemblies were especially constructed to examine the effect on rod bow of positioning the fuel rods 0.6 inch above the fuel assembly bottom grillage. This repositioning of the fuel with respect to the lower end fitting affects no mechanical operating parameters with the possible exception of rod bow, which may be decreased. In addition, repositioning the fuel within the two fuel assemblies will not decrease fuel assembly integrity during accident conditions. Standard Mark B fuel rods are used, and the form and function of these two fuel assemblies remain identical to the standard Mark B-4 fuel assemblies. The two demonstration 17 by 17 Mark C fuel assemblies are described in reference 2.

All other results presented in the FSAR fuel assembly mechanical discussion are applicable to the reload fuel assemblies.

4.2. Fuel Rod Design

Pertinent fuel rod dimensions for residual and new fuel are listed in Table 4-2. The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories for Oconee 2. Table 4-3 is a summary of the batch 2, 3, and 4 fuel rod designs. The fuel assembly power histories were analyzed, and the most limiting histories were determined. Specific assembly power histories were used in the analyses of batches 2 and 3. Batch 4 was analyzed using a conservative power history envelope. Actual operating history was used when available. This included the initial power operation at 40 and 80% core power. The predicted assembly power history for the most limiting assembly was used to determine the most limiting collapse time, as described in BAW-10084P-A.[2]

The 2000-hour densification assumption described in reference 3 was used in the analysis since it represents the most severe condition. The conservatisms in the analytical procedure are summarized below.

1. The CROV computer code was used to predict the time to collapse; CROV conservatively predicts collapse times.[3]
2. No credit is taken for fission gas release. Therefore, the net differential pressures used in the analysis are conservatively high.
3. The cladding thickness used was the lower tolerance limit (LTL) of the as-built measurements. The initial ovality of the cladding used was the upper tolerance limit (UTL) of the as-built measurements. These values were taken from a statistical sampling of the cladding.
4. A conservative power history envelope was used in the batch 4 fuel analysis.

The most limiting assembly was found to have a collapse time greater than the maximum projected three-cycle lifetime of 24,912 effective full-power hours (see Table 4-1). This analysis was performed using the assumptions on densification described in reference 3. 1

4.2.2. Cladding Stress

Since the batch 2 and 3 fuel is the most limiting from a cladding stress point of view due to the lower prepressurization and low density, the calculations performed in the Oconee 2 Fuel Densification Report, BAW-1395,[4] represented the most limiting case for Oconee 2, cycle 2.

4.2.3. Fuel Pellet Irradiation Swelling

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is such that the plastic cladding strain is less than 1% at 55,000 MWd/mtU. The following conservatisms were used in this analysis:

1. The maximum specification value for the fuel pellet diameter was used.
2. The maximum specification value for the fuel pellet density was used.
3. The cladding ID used was the lowest permitted specification tolerance.
4. The maximum expected three-cycle local pellet burnup is less than 55,000 MWd/mtU.

4.3. Thermal Design

The core loading for cycle 2 operation is shown in Figure 3-1. There are 56 fresh (batch 4) fuel assemblies and 121 once-burned (batches 2 and 3) assemblies. Two of the fresh assemblies are Mark C, 17 by 17 demonstration fuel: these were not considered in the thermal design analysis since the core configuration would be more limiting if they were not present. The thermal design of the two Mark C demonstration assemblies is described in a separate report.[2] The two Mark B-4 lifted rod demonstration fuel assemblies are thermally identical to the other 52 Mark B-4 assemblies and do not require separate analyses. The batch 4 fuel has a higher initial theoretical density (TD) and a correspondingly higher linear heat rate capability (20.15 Vs 19.8 kW/ft) than the batch 2 and 3 fuel. These linear heat rate limitations were established using the TAFY-3 code[5] with full fuel densification penalties.

4.3.1. Power Spike Model

The power spike model used in this analysis is identical to that presented in BAW-10055[6] except for two modifications; the modifications have been applied to F_g and F_k as described in reference 7. These probabilities have been changed to reflect additional data from operating reactors that support a somewhat different approach and yield less severe penalties due to power spikes. F_g was changed from 1.0 to 0.5. F_k was changed from a Gaussian to a linear distribution, which reflects a decreasing frequency with increasing gap size.

The maximum gap size versus axial position is shown in Figure 4-1, and the power spike factor versus axial position is shown in Figure 4-2. The calculated power spike and gap size were based on an initial theoretical density of 92.5% and an enrichment of 3.0 wt % uranium-235. The corresponding values for the batch 4 Mark B-4 and Mark C demonstration fuel would be smaller because of the increased density and lower enrichment of this fuel.

4.3.2. Fuel Temperature Analysis

Thermal analysis of the fuel rods assumed in-reactor fuel densification to 96.5% TD. The basis for the analysis is given in references 5 and 6, with the following modifications:

1. The code option for no restructuring of fuel has been used in this analysis in accordance with the NRC's interim evaluation of TAFY.

2. The calculated gap conductance was reduced by 25% in accordance with the NRC's interim evaluation of TAFY.

During cycle 2 operation the highest relative assembly power levels occur in batch 3 fuel (see Figures 3-1 and 5-1). The fuel temperature analysis for this fuel documented in the Oconee 2 Fuel Densification Report[4] is applicable for cycle 2 and is based on limiting BOC conditions (zero burnup), as shown in Table 4-4. Although batch 4 fuel has a reduced active fuel length and a correspondingly higher average linear heat rate, the maximum predicted centerline temperature of this fuel is lower than that of batch 3 fuel, even with the same peaking factors applied. This is due to the higher initial density of the batch 4 fuel.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 4 fuel assemblies is identical to that of the present fuel.

4.5. Operating Experience

The Mark B-4 assemblies and Mark C demonstration assemblies do not constitute a departure from past design philosophy, the adequacy of which has been demonstrated in the operation of six 177-fuel assembly plants.

Table 4-1. Fuel Design Parameters

	Residual fuel assemblies		New fuel assemblies batch 4		
	Batch 2	Batch 3			
Fuel assembly type	Mark B-3	Mark B-3	Mark B-4	Mark-C	
No. of assemblies	61	60	54	2	
Initial fuel enrich, wt % ^{235}U	2.75	3.05	2.64	2.64	
Initial fuel density, % TD	92.5	92.5	93.5	94.0	
Initial fill gas pressure, psig	(a)	(a)	(a)	(a)	
Batch burnup, BOC, MWd/mtU	16,135	10,318	0	0	1
Cladding collapse time, EFPH	>25,000	>28,000[(b)]	>25,000	>25,000	
Design life, EFPH	17,904	24,912	21,360	21,360	1

(a) Proprietary.

(b) Batch 3 is most limiting.

Table 4-2. Fuel Rod Dimensions

	Residual fuel, batches 2, 3	New fuel, batch 4	
		Mark B-4	Mark C demo
Fuel rod OD, in.	0.430	0.430	0.379
Fuel rod ID, in.	0.377	0.377	0.332
Fuel pellet OD, in.	0.370	0.370	0.324
Fuel pellet density, % TD	92.5	93.5	94.0
Undensified active fuel length, in.	144.0	142.6	143.0
Type of flexible spacers	Corrugated	Spring	Spring
Solid spacer material	ZrO_2	Zr-4	Zr-4

Table 4-3. Input Summary for Cladding Creep Collapse Calculations

	Batches 2, 3	Batch 4	Mark C demo fuel assemblies
Pellet OD (mean specified), in.	0.3700	0.3700	0.3240
Pellet density (mean specified), % TD	92.5	93.5	94.0
Densified pellet OD, in.	0.3650	0.3663	0.3213
Cladding ID (mean specified), in.	0.377	0.377	0.332
Cladding ovality (CTL), in.	*	*	*
Cladding thickness (LTL), in.	*	*	*
Prepressure (minimum specified), psia	*	*	*
Post-densification prepressure (cold), psia	*	*	*
Reactor system pressure, psia	2200	2200	2200
Stack height (undensified), in.	144.0	142.6	143.0

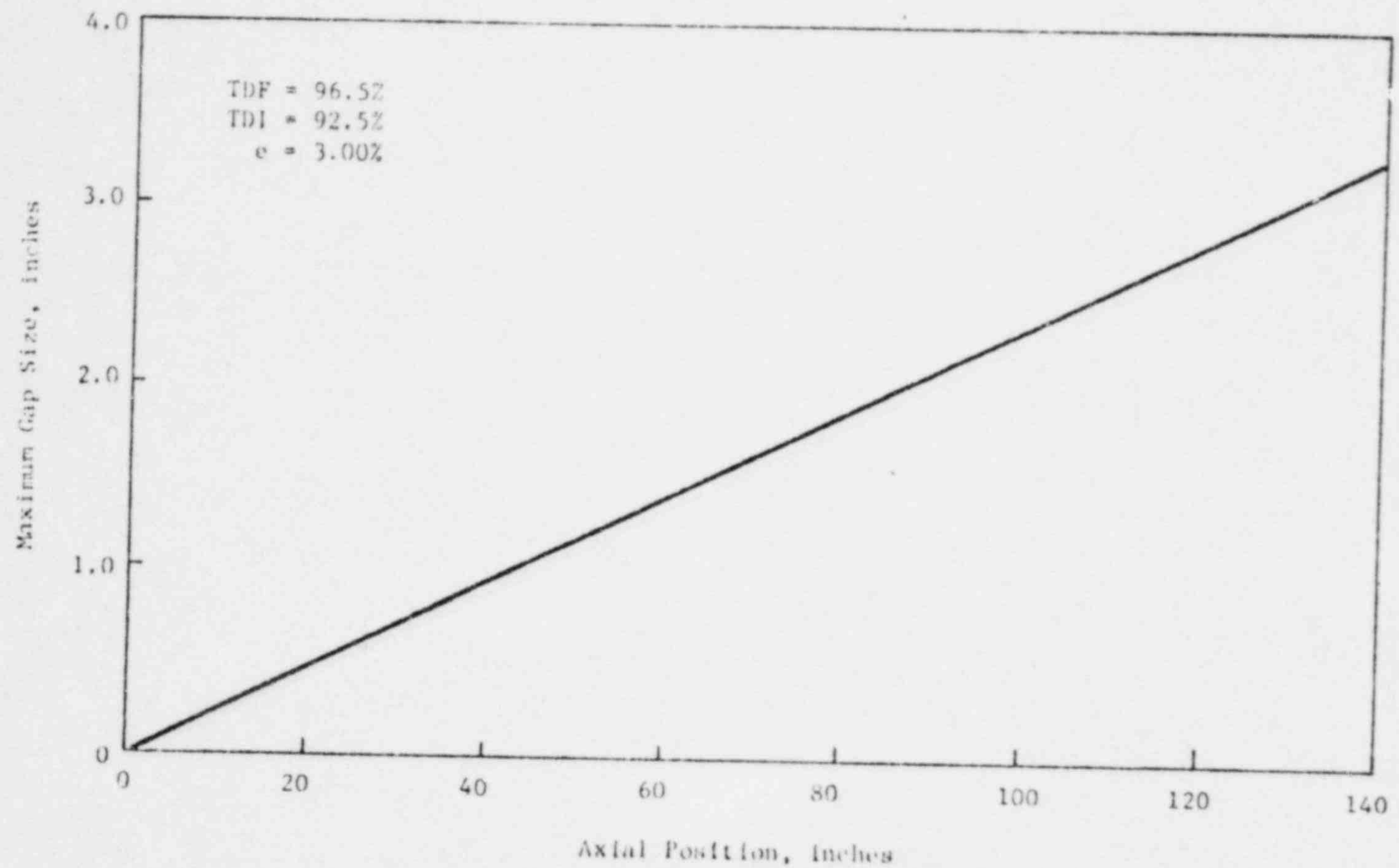
*Proprietary.

Table 4-4. Fuel Thermal Analysis Parameters

	Batches 2, 3	Batch 4
Nominal linear heat rate, kW/ft	5.77	5.80
Densified active fuel length, in.	141.1	140.5
Linear heat rate (LHR) to central fuel melt, kW/ft	19.8	20.15
Hot channel factor on LHR	1.014	1.014
Initial TD %	92.5	93.5
initial fuel pellet OD, in.	0.370	0.370
Avg fuel temp @ nominal LHR, F	1335	1311
Fuel centerline melting temp @ BOL[a], F	5080	5080

[a] At zero burnup.

Figure 4-1. Maximum Gap Size Vs Axial Position - Oconee 2, Cycle 2



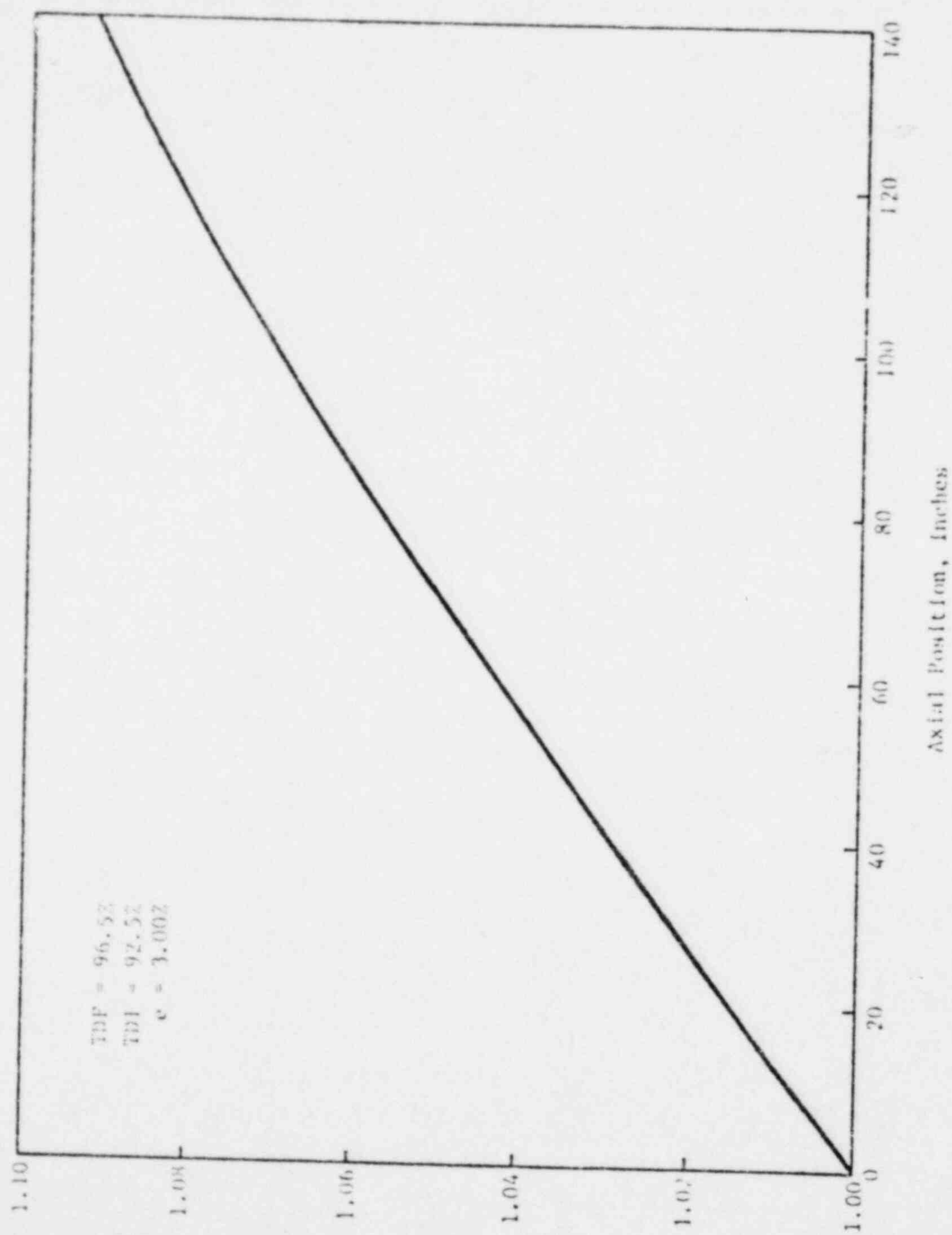


Figure 4-2. Power Spike Factor Vs Axial Position — Group 2, Cycle 2

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 1 and 2. The values for both cycles were generated using PDQ07 Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles.

The shorter cycle 2 will produce a smaller cycle differential burnup than that for cycle 1. The accumulated average core burnup will be higher in cycle 2 than in cycle 1 because of the presence of the once-burned batch 2 and 3 fuel. Figure 5-1 illustrates a representative relative power distribution for the beginning of the second cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 2 are lower in all cases than for cycle 1. As indicated in Table 5-2, the control rod worths are sufficient to maintain the required shutdown margin. However, due to changes in isotopics and the radial flux distribution, the BOC hot, full-power control rod worths are generally less than those for cycle 1. The cycle 2 ejected rod worths for the same number of regulating banks inserted are lower than those in cycle 1. It is difficult to compare values between cycles or between rod patterns since neither the rod patterns from which the CRA is assumed to be ejected nor the isotopic distributions are identical. 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 insertion limits presented in section 8. The maximum stuck rod worths for cycle 2 are less than those in cycle 1. The adequacy of the shutdown margin with cycle 2 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

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

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of cycle 2 is analyzed at approximately 267 EFPD. This is the latest time (± 10 days) in core life at which the transient bank is nearly fully inserted. After 267 EFPD, the transient bank will be almost fully withdrawn, thus increasing the available shutdown margin. The reference fuel cycle shutdown margin is presented in the Oconee 1, 2, and 3 FSAR, Table 3-5.

The cycle 2 power deficits from hot zero power to hot full power are similar to but slightly higher than those for cycle 1. Doppler coefficients, moderator coefficients, and xenon worths are similar for the two cycles. The differential boron worths for cycle 2 are lower than those for cycle 1 due to depletion of the fuel and the associated buildup of fission products. The effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2. Analytical Input

The cycle 2 incore measurement calculation constants used for computing core power distributions were prepared in the same manner as for the reference cycle.

5.3. Changes in Nuclear Design

There are four demonstration fuel assemblies in batch 4. The two 17 by 17 Mark C demonstration fuel assemblies have been shown to have no significant effect on the nuclear design of cycle 2.[2]

The potential impact of raised fuel rods (0.6 inch above the lower end fitting grillage) on the nuclear design and safety analysis of Oconee 2, cycle 2 has been reviewed. Since only two fuel assemblies with raised fuel rods are being inserted into the core as part of batch 4, the impact on the overall nuclear parameters is negligible, and no additional analysis is required. Furthermore, no additional restrictions on the placement of these assemblies in the core are necessary.

The same calculational methods and design information were used to obtain the important nuclear design parameters for cycles 1 and 2. In addition, there are no significant operational procedure changes from the reference cycle with regard to axial or radial power shape control, xenon control, or tilt control. The operational limits (Technical Specifications changes) for the reload cycle are shown in section 8.

Table 5-1. Oconee 2 Cycle 1 and 2 Physics Parameters

	Cycle 1[d]	Cycle 2[e]
Cycle length, EFPD	460	306
Cycle burnup, MWd/mtU	14,396	9582
Average core burnup — EOC, MWd/mtU	14,396	18,606
Initial core loading, mtU	82.1	82.1
Critical boron — BOC, ppm (no Xe)		
HZP[a], all rods out	1634	1445
HZP, groups 7 and 8 inserted	1494	1330
HFP, groups 7 and 8 inserted	1382	1140
Critical boron — EOC, ppm (eq Xe)		
HZP, all rods out	480	434
HFP, group 8 (37.5% wd, eq Xe)	180	87
Control rod worths — HFP[a], BOC, %Δk/k		
Group 6	1.58	1.20
Group 7	0.99	0.96
Group 8 (37.5% wd)	0.44	0.54
Control rod worths — HFP, EOC, %Δk/k		
Group 7	1.37	1.33
Group 8 (37.5% wd)	0.26	0.51
Max ejected rod worth — HZP, %Δk/k		
BOC	0.48[c]	0.59[b]
EOC	0.72[c]	0.58[b]
Max stuck rod worth — HZP, %Δk/k		
BOC	4.27	2.16
EOC	2.69	2.22
Power deficit, HZP to HFP, %Δk/k		
BOC	1.32	1.65
EOC	2.10	2.49
Doppler coeff — BOC, 10^{-5} (Δk/k/°F)		
100% power (0 Xe)	-1.51	-1.51
Doppler coeff — EOC, 10^{-5} (Δk/k/°F)		
100% power (eq Xe)	-1.67	-1.55
Moderator coeff — HFP, 10^{-4} (Δk/k/°F)		
BOC (0 Xe, 1000 ppm, groups 7, 8 ins)	-0.23	-1.03
EOC (eq Xe, 17 ppm, group 8 ins)	-2.70	-2.60
Boron worth — HFP, ppm/Δk/k		
BOC (1000 ppm)	98	109
EOC (17 ppm)	95	101

Table 5.1-1. (Cont'd)

	Cycle 1[d]	Cycle 2[e]	
Xenon worth — HFP, %Δk/k			1
BOC (4 days)	2.71	2.60	
EOC (equilibrium)	2.65	2.66	
Effective delayed neutron fraction — HFP			
BOC	0.00690	0.00577	
EOC	0.00514	0.00516	

[a] HZP: hot zero power; HFP: hot full power.

[b] Ejected rod value for groups 5, 6, 7, and 8 inserted.

[c] Ejected rod value for groups 6, 7, and 8 inserted.

[d] For cycle 1 length of 460 EFPD.

[e] Based on cycle 1 length of 440 EFPD. 1

Table 5-2. Shutdown Margin Calculation — Oconee 2, Cycle 2

Available Rod Worth	BOC, % Δk/k	EOC[a], % Δk/k
Total rod worth, HZP[b]	9.77	9.80
Worth reduction due to burnup of poison material	-0.19	-0.30
Maximum stuck rod, HZP	-2.16	-2.22
Net worth	7.42	7.28
Less 10% uncertainty	-0.74	-0.73
Total available worth	6.68	6.55
Required Rod Worth		
Power deficit, HFP to HZP	1.65	2.49
Max allowable inserted rod worth	1.07	1.27
Flux redistribution	0.40	1.00
Total required worth	3.12	4.76
Shutdown Margin		
Total avail. worth - total req. worth	3.56	1.79

1

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

(a) For shutdown margin calculations, this is defined as approximately 257 EFPD, the latest time in core life at which the transient bank is nearly full-in. 1

(b) HZP: hot zero power; HFP: hot full power.

Figure 5-1. BOC (4 EFPD), Cycle 2 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, Normal Rod Positions (Groups 7 and 8 Inserted)

	8	9	10	11	12	13	14	15
H	1.40	1.27	1.22	1.17	1.21	0.85	0.77	0.60
K	1.27	1.41	1.37	1.23	1.23	7 0.57	0.69	0.59
L	1.22	1.37	1.18	1.13	8 1.12	0.93	0.95	0.52
M	1.17	1.23	1.13	1.35	1.29	1.24	0.92	
N	1.21	1.23	8 .12	1.29	1.10	1.09	0.66*	
O	0.85	7 0.57	0.95	1.24	1.09	0.72		
P	0.77	0.69	0.95	0.92	0.66			
R	0.60	0.59	0.52					

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

1

6. THERMAL-HYDRAULIC DESIGN

6.1. Thermal-Hydraulic Design Calculations

Thermal-hydraulic design calculations for support of cycle 2 operation utilized the analytical methods documented in references 1 and 4. These calculations were made to account for the introduction of the Mark B-4 assemblies in batch 4, to consider the minimum actual reactor coolant system flow rate as measured during first cycle operation, and to incorporate the BAW-2 CHF correlation in place of the previously used W-3 correlation. The BAW-2 CHF correlation was used in licensing Oconee 1, cycle 2.

6.1.1. Introduction of Mark B-4 Assemblies

As discussed in section 4.1, the Mark B-4 assemblies differ from the Mark B-3 assemblies primarily in the end fittings. This difference causes a slight reduction in the flow resistance of the B-4 assemblies. Since the B-4 assemblies are loaded primarily on the periphery of the core, the hottest (highest power) assembly is a B-3 (see Figure [illegible] and 5-1). In order to conservatively account for the introduction of the two Mark C assemblies, the thermal-hydraulic model utilized a [illegible] B-4, 1[illegible]1 B-3 configuration and retained the B-3 assemblies in the hottest core locations. This assumption increases the conservatism of the cycle 2 design by reducing the calculated hot assembly flow rate. For the two Mark B-4 [illegible]-rod demonstration assemblies, the assembly form loss will be less than the Mark B-3 fuel. Hence, the B-3 assembly will still have the highest hydraulic losses and will be DNB-flow-limited.

6.1.2. Increased RC System Flow

Reactor coolant flow data obtained during cycle 1 operation verified that the system flow was greater than the design flow. The measured flow was 111.5% of the design flow. For the cycle 2 thermal-hydraulic design analysis, the increase in system flow was conservatively chosen to be 107.6% of design.

6.1.3. B&W-2 DNB Correlation

The B&W-2 DNB correlation, a realistic prediction of the burnout phenomena,[illegible] has been reviewed and approved for use with the Mark B fuel assembly design. In the application of this correlation to the Oconee [illegible] cycle 2 core, two modifications which have also been applied to the TMI-1, cycle 2, and Oconee 1, cycle 3 cores, have been instituted:

1. The limiting design DNBR of 1.30 was used, representing a 95% confidence level for 95% population protection. The limiting DNBR of 1.32, which had been used for this correlation in previous design analyses, represented a 99% confidence level for a 95% protection. This change is consistent with industry practice and the statistical standards associated with limiting design DNBR values accepted by the NRC staff and the ACRS.
2. The pressure range applicable to the correlation has been extended downward from 2000 to 1750 psia. This revision is based on a review of rod bundle CHF data taken at pressures below 2000 psia, which shows that the B&W-2 correlation conservatively predicts the data in this range.

The use of this correlation in conjunction with increased system flow for the cycle 2 analysis indicates that the margin to DNB is higher than had been predicted for first core operation, as shown in Table 6-1.

6.2. DNBR Analysis

In addition to the items discussed above, the maximum design conditions considered in the FSAR and generic fuel assembly geometry based on Mark B as-built data were taken into account. This resulted in a minimum DNBR of 1.9[illegible] at 112% power for undensified fuel.

The effects of densification can be divided into two categories: (1) the reduced stack height and (2) the power spike resulting from densification-induced gaps in the fuel column. The active length was calculated to be 141.1 inches, a reduction from the undensified length of 144.0 inches. These densified and undensified lengths are based on fuel batches 2 and 3, the limiting batches in cycle 2, as discussed in section 4.3.1.

The axial flux shape that produced the maximum change in DNBR from the original design value was an outlet peak with a core offset of +11.[illegible]%. The spike magnitude and the maximum gap size are discussed in section 4.3, and the values used in the analysis are 1.087 and 3.10 inches, respectively. The results of the two effects are -1.88 and -1.06% change* in the minimum hot channel DNBR and peaking margin, respectively. The changes in these margins are summarized in Table 6-1, which includes comparisons of other pertinent cycle 1 and 2 data.

The DNBR analysis has been based on a core configuration consisting of 177 Mark B fuel assemblies. The incorporation of two Mark C demonstration fuel assemblies in batch 4 in place of two Mark B assemblies, results in an increase in the overall margin to DNB, as discussed in reference 2.

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*Change from undensified values. These are actually an improvement over cycle 1 densified conditions.

Table 6-1. Cycle 1 and 2 Maximum Design Conditions

	Cycle 1[a]	Cycle 2
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	[illegible]	107.6
Vessel inlet coolant temperature, 100% power, F	554.0	555.9
Vessel outlet coolant temperature, 100% power, F	602.8	602.2
Ref. design radial-local power peaking factor	1.78	1.78
Ref. design axial flux shape	1.5 cosine	1.5 cosine
Active fuel length, in.	144 (undens.)	141.1 (dens.)
Avg heat flux, 100% power, Btu/h-ft²	171,470	174,995
Max heat flux, 100% power, Btu/h-ft² (for DNBR calc)	457,825	467,236
CHF correlation	W-3	B&W-2
Minimum DNBR (max design conditions, no densif penalties)	1.5[illegible] ([illegible] power)	1.98 (112% power)
Hot channel factors		
Enthalpy rise	1.022	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Densification effects		
Change in DNBR margin, %	-6.08[b]	-1.88
Change in power peaking margin, %	-2.[illegible]2[b]	-1.06

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 2 parameters to determine the effects of the cycle 2 reload and to ensure that thermal performance is not degraded during hypothetical transients.

The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. Cycle 1 values (FSAR values) of core thermal parameters are compared with those used in the cycle 2 analysis in Table 6-1. These parameters are common to all of the accident analyses presented herein. For each accident of the FSAR, a discussion and the key parameters are provided. A comparison of the key parameters (see Table 7-1) from the FSAR and the present cycle 2 is provided with the accident discussion to show that the initial conditions of the transient are bounded by the FSAR analysis.

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in BAW-1395.[4] Since cycle 2 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in reference 4, the conclusions derived in that reference are still valid.

Calculational techniques and methods for cycle 2 analyses remain consistent with those used for the FSAR. Additional DNBR margin is shown for cycle 2 because the BAW-2 CHF correlation was used instead of the W-3.

No new dose calculations were performed for the reload report. The dose considerations in the FSAR were based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

7.2. Rod Withdrawal Accidents

This accident is defined as an uncontrolled reactivity addition to the core due to withdrawal of control rods during startup conditions or from rated power conditions. Both types of incidents were analyzed in the FSAR.[1]

The important parameters during a rod withdrawal accident are Doppler coefficient, moderator temperature coefficient, and the rate at which reactivity is added to the core. Only high-pressure and high-flux trips are accounted for in the FSAR analysis, which ignores multiple alarms, interlocks, and trips that normally preclude this type of incident. For positive reactivity additions indicative of these events, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were $-1.17 \cdot 10^{-5}$ ($\Delta k/k/°F$) for the Doppler coefficient, $0.5 \cdot 10^{-4}$ $\Delta k/k$ for the moderator temperature coefficient and rod group worths up to and including a 10% $\Delta k/k$ rod bank worth. Comparable cycle 2 parametric values are $-1.31 \cdot 10^{-5}$ ($\Delta k/k/°F$) for the Doppler coefficient, $-1.03 \cdot 10^{-4}$ ($\Delta k/k/°F$) for the moderator temperature coefficient, and a maximum rod bank worth of 9.8% $\Delta k/k$. Therefore, cycle 2 parameters are bounded by design values assumed for the FSAR analysis. Thus, for the rod withdrawal transients, the consequences will be no more severe than those presented in the FSAR. For the rod withdrawal from rated power, the transient consequences are also less severe than those presented in the densification report.[4] 1

7.3. Moderator Dilution Accident

Boron in the form of boric acid is utilized to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup and transient xenon effects with dilution water supplied by the makeup and purification system. The moderator dilution transients considered are the pumping of water with zero boron concentration from the makeup tank to the RCS under conditions of full-power operation, hot shutdown, and refueling. The key parameters in this analysis are the initial boron concentration, boron reactivity worth, and moderator temperature coefficient for power cases.

For positive reactivity addition of this type, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were 1400 ppm for the initial boron concentration, 75 ppm/1% ($\Delta k/k$) boron reactivity worth and $+0.94 \cdot 10^{-4}$ $\Delta k/k/°F$ for the moderator temperature coefficient.

Comparable cycle 2 values are 1140 ppm for the initial boron concentration, 84 ppm/1% ($\Delta k/k$) boron reactivity worth and $-1.03 \cdot 10^{-4}$ ($\Delta k/k$)/°F for the 1 moderator temperature coefficient. The FSAR shows that the core and RCS are adequately protected during this event. Sufficient time for operator action to terminate this transient is also shown in the FSAR, even with maximum dilution and minimum shutdown margin. The predicted cycle 2 parametric values of importance to the moderator dilution transient are bounded by the FSAR design values; thus, the analysis in the FSAR is valid.

7.4. Cold Water (Pump Startup) Accident

There are no check or isolation valves in the reactor coolant piping; therefore, the classic cold water accident is not possible. However, when the reactor is operated with one or more pumps not running, and then these are turned on, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, then reactivity will be added to the core and a power rise will occur.

Protective interlocks and procedures prevent starting idle pumps if the reactor power is above 22%. However, these restrictions were ignored, and two-pump startup from 50% power was analyzed as the most severe transient.

To maximize reactivity addition, the FSAR analysis assumed the most negative moderator temperature coefficient of -3.0×10^{-4} $(\Delta k/k)/°F$ and least negative Doppler coefficient of -1.30×10^{-5} $\Delta k/k/°F$. The corresponding most negative moderator temperature coefficient and least negative Doppler coefficient predicted for cycle 2 are -2.60×10^{-4} and -1.31×10^{-5} $(\Delta k/k)/°F$, respectively. 1 Since the predicted cycle 2 moderator temperature coefficient is less negative and the Doppler coefficient is more negative than the values used in the FSAR, the transient results would be less severe than those reported in the FSAR.

7.5. Loss of Coolant Flow

The reactor coolant flow rate decreases if one or more of the reactor coolant pumps fail. A pumping failure can be caused by mechanical failures or a loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect the operation of others. With the reactor at power, the effect of loss of coolant flow is a rapid increase in coolant temperature due to the reduction of heat removal capability. This increase could result in DNB if corrective action were not taken immediately. The key parameters for four-pump coastdown or a locked-rotor incident are the flow rate, flow coastdown characteristics, Doppler coefficient, moderator temperature coefficient, and hot channel DNB peaking factors. The most conservative initial conditions were assumed for the densification report: FSAR values of flow and coastdown, -1.17×10^{-5} $(\Delta k/k)/°F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator temperature coefficient, with densified fuel power spike and peaking. The results showed that the DNBR remained above 1.3 (W-3) for the four-pump coastdown, and the fuel cladding temperature remained below criteria limits for the locked-rotor transient.

The predicted parametric values for cycle 2 are -1.31×10^{-5} $(\Delta k/k)/°F$ Doppler coefficient, -1.03×10^{-4} $(\Delta k/k)/°F$ moderator temperature coefficient, and peaking factors as shown in Table 6-1. Since the predicted cycle 2 values are bounded by those used in the densification report, the results of that analysis represent the most severe consequences from a loss of flow incident. 1

7.6. Stuck-Out, Stuck-In, or Dropped Control Rod

If a control rod were dropped into the core while it was operating, a rapid decrease in neutron power would occur, accompanied by a decrease in the core average coolant temperature. The power distribution might be distorted due to a new control rod pattern, under which conditions a return to full power might lead to localized power densities and heat fluxes in excess of design limitations.

The key parameters for this transient are moderator temperature coefficient, dropped rod worth, and local peaking factors. The FSAR analysis was based on 0.46 and 0.36% $\Delta k/k$ rod worths with a moderator temperature coefficient of -3.0×10^{-4} $(\Delta k/k)/°F$. For cycle 2, the maximum worth rod at power is 0.20% $\Delta k/k$ and a moderator temperature coefficient of -2.6[illegible] $\times 10^{-4}$ $(\Delta k/k)/°F$. Since 1
the predicted rod worth is less positive and the moderator temperature coefficient is more positive, the consequences of this transient are less severe than the results presented in the FSAR.

7.7. Loss of Electric Power

Two types of power losses were considered in the FSAR: (1) a loss of load condition caused by separation of the unit from the transmission system and (2) a hypothetical condition resulting in a complete loss of all system and unit power except that from the unit batteries.

The FSAR analysis evaluated the loss of load with and without turbine runback. When there is no runback, a reactor trip occurs on high reactor coolant pressure or temperature. This case results in a non-limiting accident. The largest offsite dose occurs for the second case, i.e., loss of all electrical power except unit batteries, and assuming operation with failed fuel and steam generator tube leakage. These results are independent of core loading; therefore, the results of the FSAR are applicable for any reload.

7.8. Steam Line Failure

A steam line failure is defined as a rupture of any of the steam lines from the steam generators. Upon initiation of the rupture, both steam generators start to blow down, causing a sudden decrease in the primary system temperature, pressure, and pressurizer level. The temperature reduction leads to positive reactivity insertion, and the reactor trips on high flux or low RC pressure. The FSAR has identified a double-ended rupture of the steam line between the steam generator and steam stop valve as the worst-case situation, at end-of-life conditions.

The key parameter for the core response is the moderator temperature coefficient, which was assumed in the FSAR to be -3.0×10^{-4} $(\Delta k/k)/°F$. The cycle 2 predicted value of moderator temperature coefficient is -2.60×10^{-4} $(\Delta k/k)/°F$. This value is bounded by those used in the FSAR analysis; hence, the results in the FSAR represent the worst situation. 1

7.9. Steam Generator Tube Failure

A rupture or leak in a steam generator tube allows reactor coolant and associated activity to pass to the secondary system. The FSAR analysis is based on complete severance of a steam generator tube. The primary concern for this incident is the potential radiological release, which is independent of core loading. Hence, the FSAR results are applicable to this reload.

7.10. Fuel Handling Accident

The mechanical damage accident is considered the maximum potential source of activity release during fuel handling activities. The primary concern is radiological releases that are independent of core loading; therefore, the FSAR results are applicable to all reloads.

7.11. Rod Ejection Accident

For reactivity to be added to the core more rapidly than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the control

The tabulation below shows the bounding values for allowable LOCA peak linear heat rates for Oconee 2, cycle 2 fuel.

Core elevation, ft	Allowable peak linear heat rate, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

The Mark C 17 by 17 demonstration assembly will be located on the periphery of the core. Because of its location, the maximum linear heat rate within the assembly will be approximately 10 kW/ft. Operation at this low linear heat rate will prevent cladding rupture during blowdown should a LOCA occur. Since rupture during blowdown causes the highest peak cladding temperature, the consequences of the LOCA should be less severe for the Mark C 17 by 17 demonstration fuel. In addition, the low linear heat rate provides substantial margin relative to the LOCA limits calculated in BAW-10103.[11] Therefore, compliance with the acceptance criteria of 10 CFR 50.46 is ensured.

rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. This incident represents the most rapid reactivity insertion that can be reasonably postulated. The values used in the FSAR and densification report at BOL conditions, -1.17×10^{-5} $(\Delta k/k)/°F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator temperature coefficient, and an ejected rod worth of 0.65% $\Delta k/k$, represent the maximum possible transient. The corresponding cycle 2 parametric values of -1.51×10^{-5} $(\Delta k/k)/°F$ Doppler, -1.03×10^{-4} $(\Delta k/k)/°F$ moderator temperature coefficient (both more negative than those used in reference 4), and a maximum predicted ejected rod worth of 0.18% $\Delta k/k$ ensure that the results will be less severe than those presented in the FSAR[1] and the densification report.[4] 1

7.12. Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur since it would require a multitude of failures in the engineered safeguards. The hypothetical accident is based solely on a gross release of radioactivity to the reactor building. The consequences of this accident are independent of core loading; hence, the results reported in the FSAR are applicable for all reloads.

7.13. Waste Gas Tank Rupture

The waste gas tank was assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel. Rupture of the tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. The consequences of this incident are independent of core loading; therefore, the results reported in the FSAR are applicable to any reload.

7.14. LOCA Analysis

A generic LOCA analysis for B&W's 177-FA, lowered-loop NSS (category I plant) has been performed using the Final Acceptance Criteria ECCS Evaluation Model.[10] That analysis is generic since limiting values of key parameters for all plants in this category were used. The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the BAW-10103[10] LOCA limits analysis are conservative compared to those calculated for this reload. Therefore, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of the Oconee 2, cycle 2 fuel.

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

Parameter	FSAR, densif value	Predicted cycle 2 value	
BOL Doppler coeff, 10^{-5} $(\Delta k/k)/°F$	$-1.17^{(a)}$	-1.51	
EOL Doppler coeff, 10^{-5} $(\Delta k/k)/°F$	-1.33	-1.55	
BOL moderator coeff, 10^{-4} $(\Delta k/k)/°F$	$+0.5^{(b)}$	-1.03	
EOL moderator coeff, 10^{-4} $(\Delta k/k)/°F$	-3.0	-2.60	1
All rod bank worth (HZP), % $\Delta k/k$	10.0	9.8	
Initial boron conc (HFP), ppm	1400	1140	
Boron reactivity worth (70F), ppm/1% $\Delta k/k$	75	84	
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.18	
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20	

(a) $(-1.2 \times 10^{-5}\ \Delta k/k/F)$ was used for steam line failure analysis. $(-1.3 \times 10^{-5}\ \Delta k/k/F)$ was used for cold water analysis.

(b) $(+0.94 \times 10^{-4}\ \Delta k/k/F)$ was used for the moderator dilution accident.

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 2 operation. Changes were the results of the following:

1. Using the B&W-2 CHF correlation rather than W-3, as discussed in section 6.1.

2. Using a 95/95 confidence level rather than 99/95, as discussed in section 6.1.

3. Using 107.6% of design flow rather than 100%, as discussed in section 6.1.

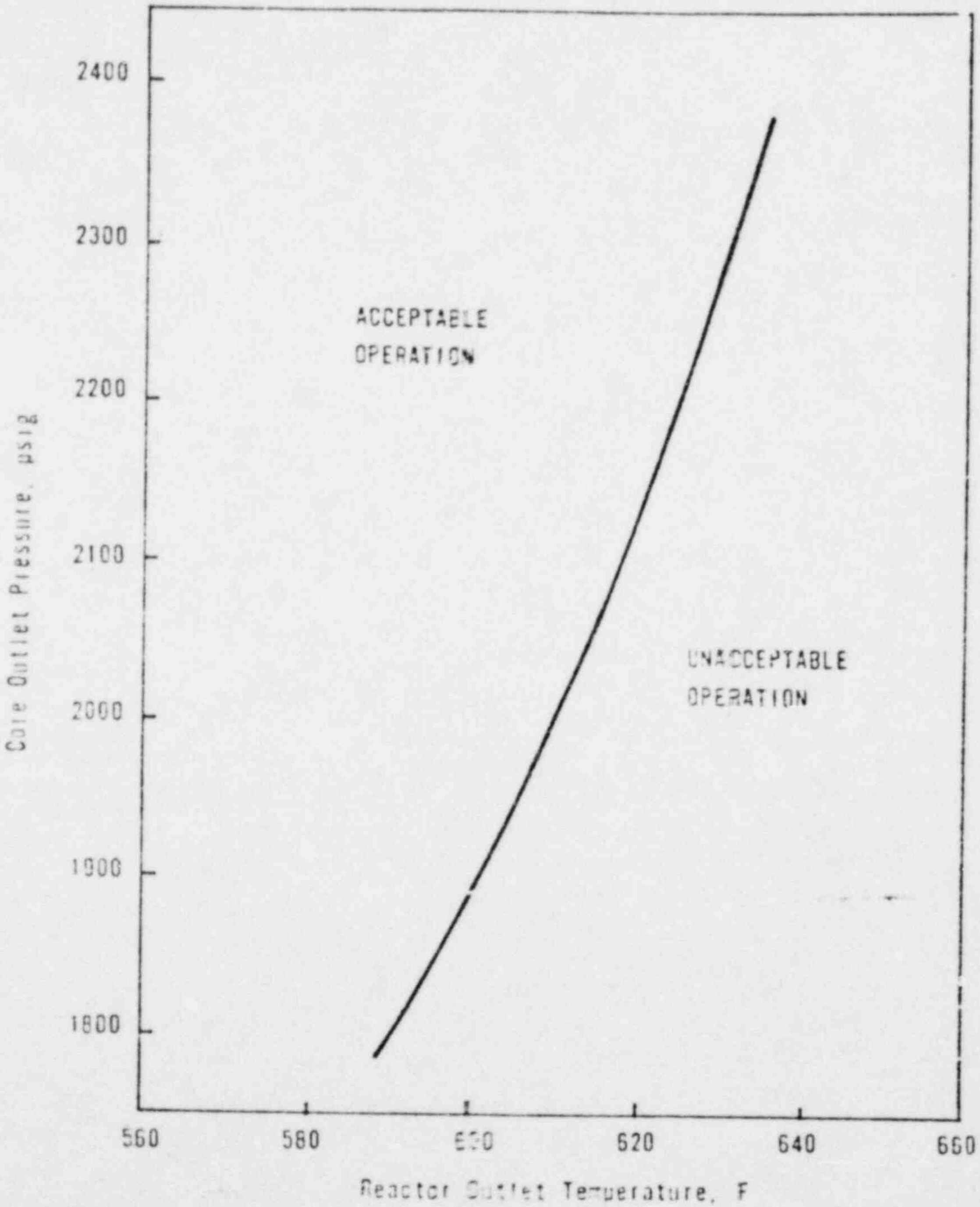
4. Using the Final Acceptance Criteria LOCA Analysis for restricting peaks during operation, as discussed in section 7.14.

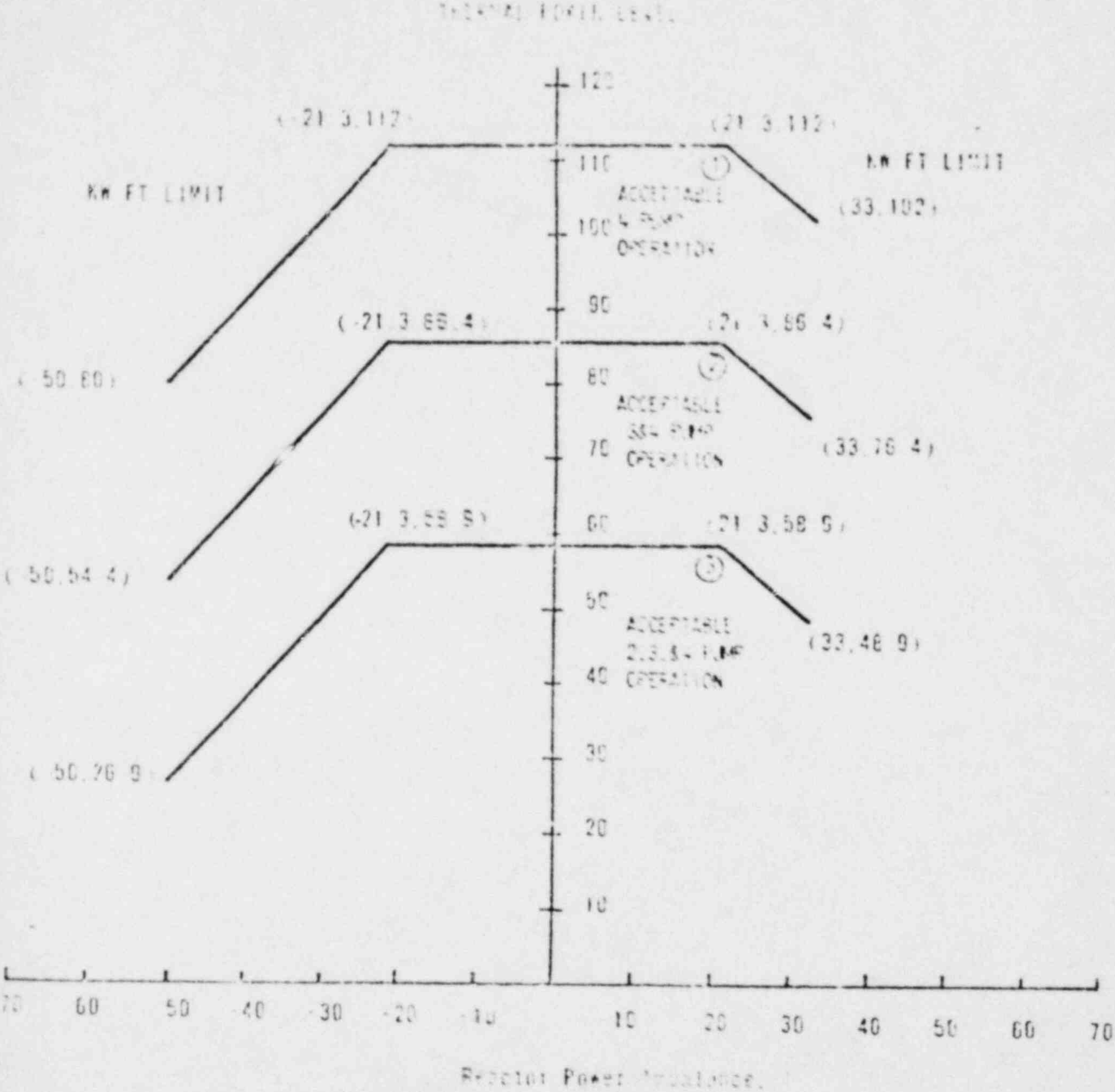
5. Revising the assumptions on which the flux flow RPS setpoint is based. This setpoint now accounts for signal noise on the basis of data accumulated from operating B&W reactors.

6. An analysis incorporating the effects of fuel rod bow on core parameters.

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. Figures 8-1 through 8-14 illustrate revisions to previous Technical Specification safety limits.

Figure 8-1. Oconee 2, Cycle 2 — Core Protection Safety Limits





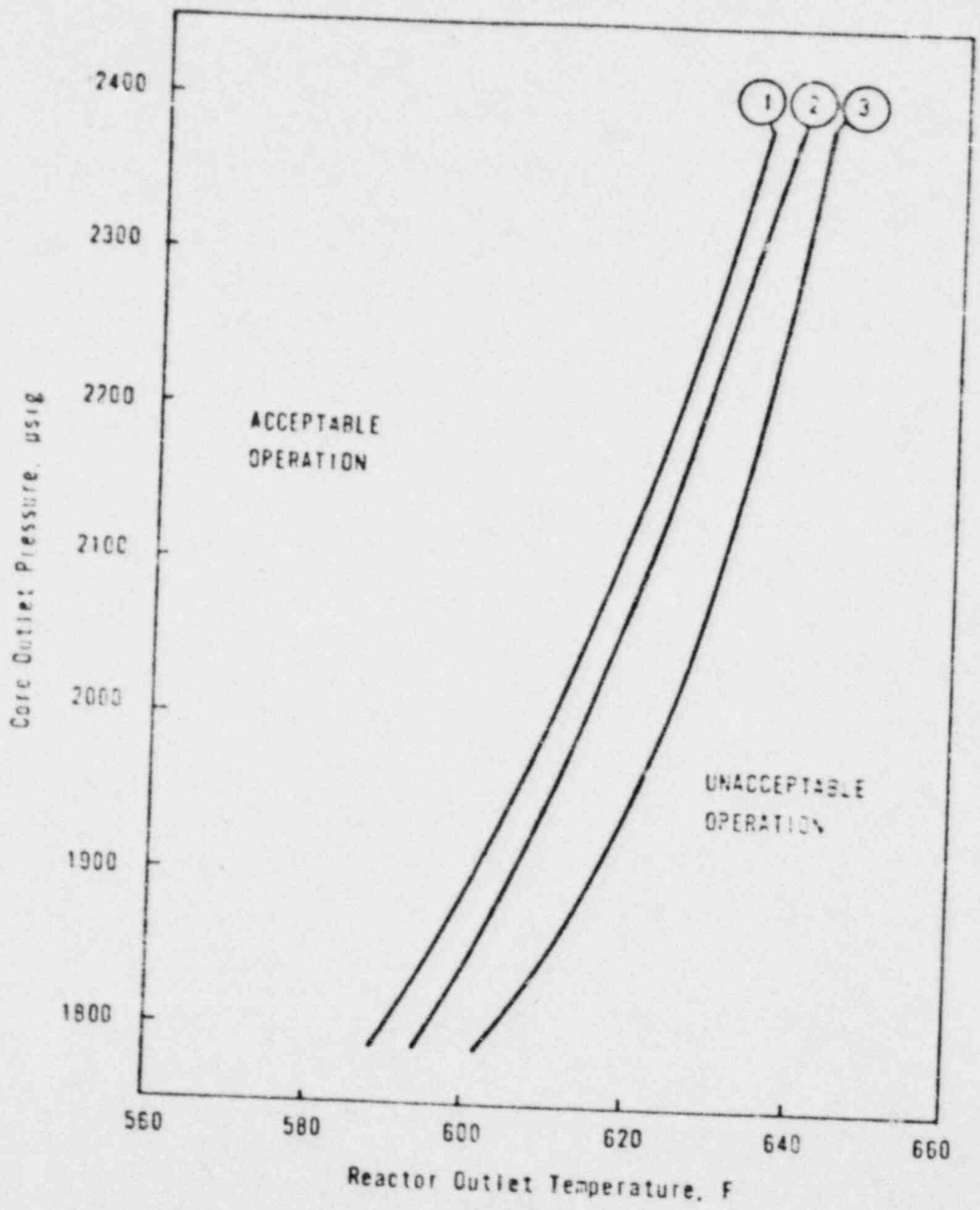
CURVE	REACTOR COOLANT FLOW (LB/HR)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6

Unit 2, Cycle 2

CORE PROTECTION SAFETY LIMITS

Figure 8-2

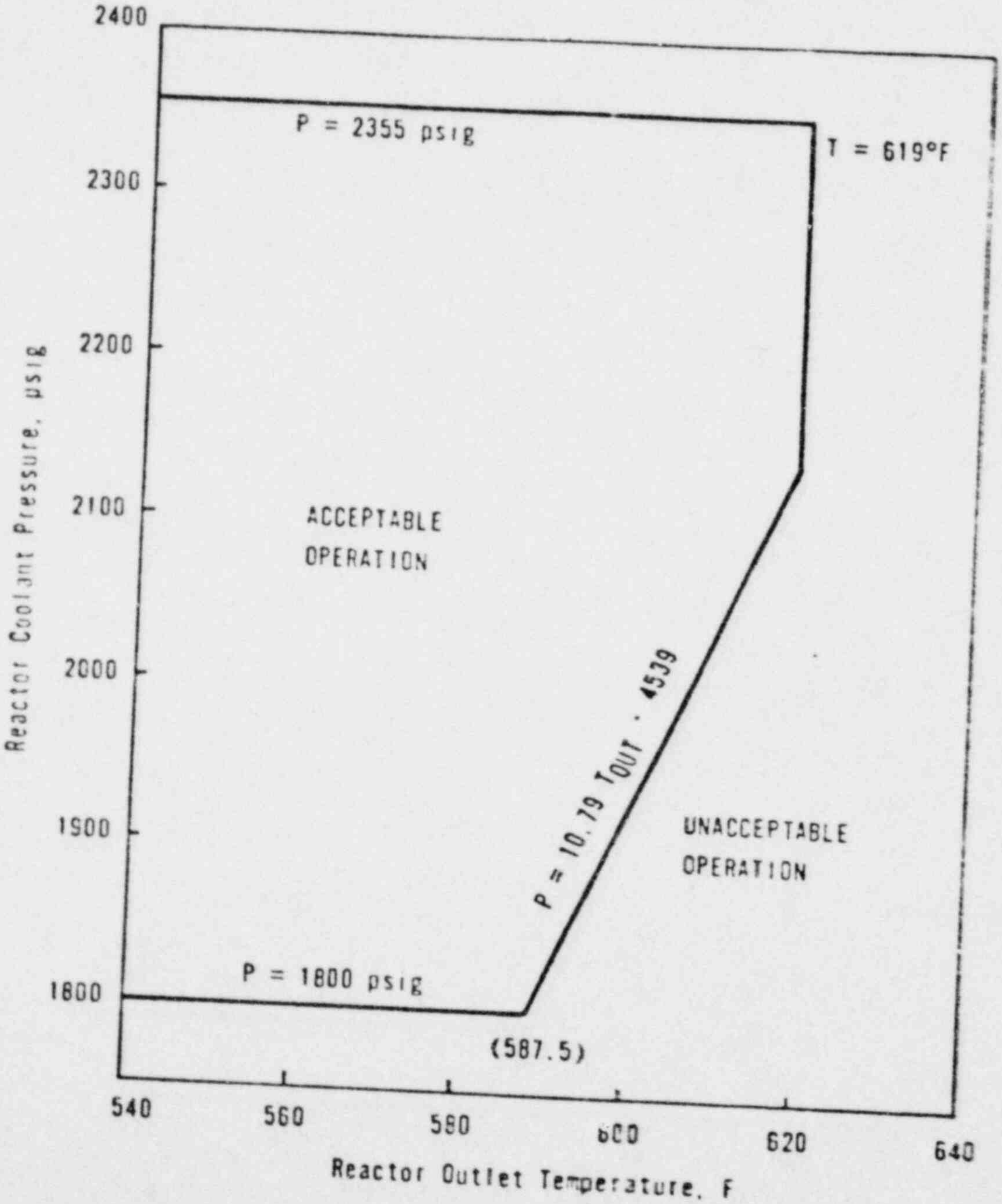
Figure 8-3. Oconee 2, Cycle 2 — Core Protection Safety Limits

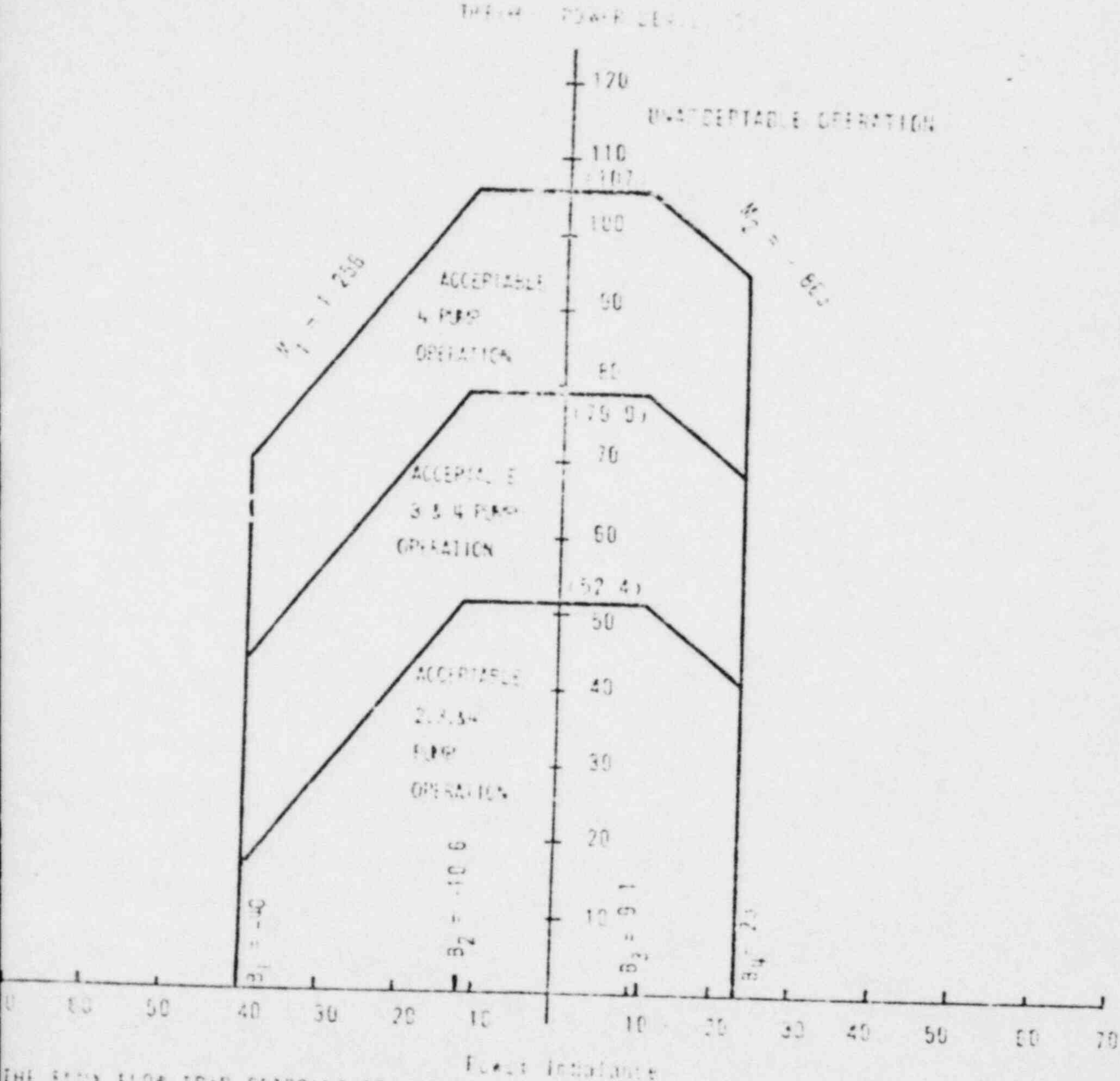


CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	141.3×10^6 (100%)*	112%	FOUR PUMP (DNBR LIMITED)
2	105.6×10^6 (74.7%)	86.4%	THREE PUMP (DNBR LIMITED)
3	69.3×10^6 (49%)	58.9%	ONE PUMP IN EACH LOOP (QUALITY LIMITED)

* 107.6% OF CYCLE 1 DESIGN FLOW

Figure 8-4. Oconee 2, Cycle 2 — Protective System Maximum Allowable Setpoints





THE [illegible] FLOW TRIP SETPOINT FOR
2 0 PUMP OPERATION MUST BE SET AT
0 96[illegible]

Unit 2 Cycle 2

PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS

Figure 8-5

Figure 5-6. [illegible] 2, Cycle [illegible] Rod Position Limits for Four-Pump Operation From 0 to 150 [illegible]

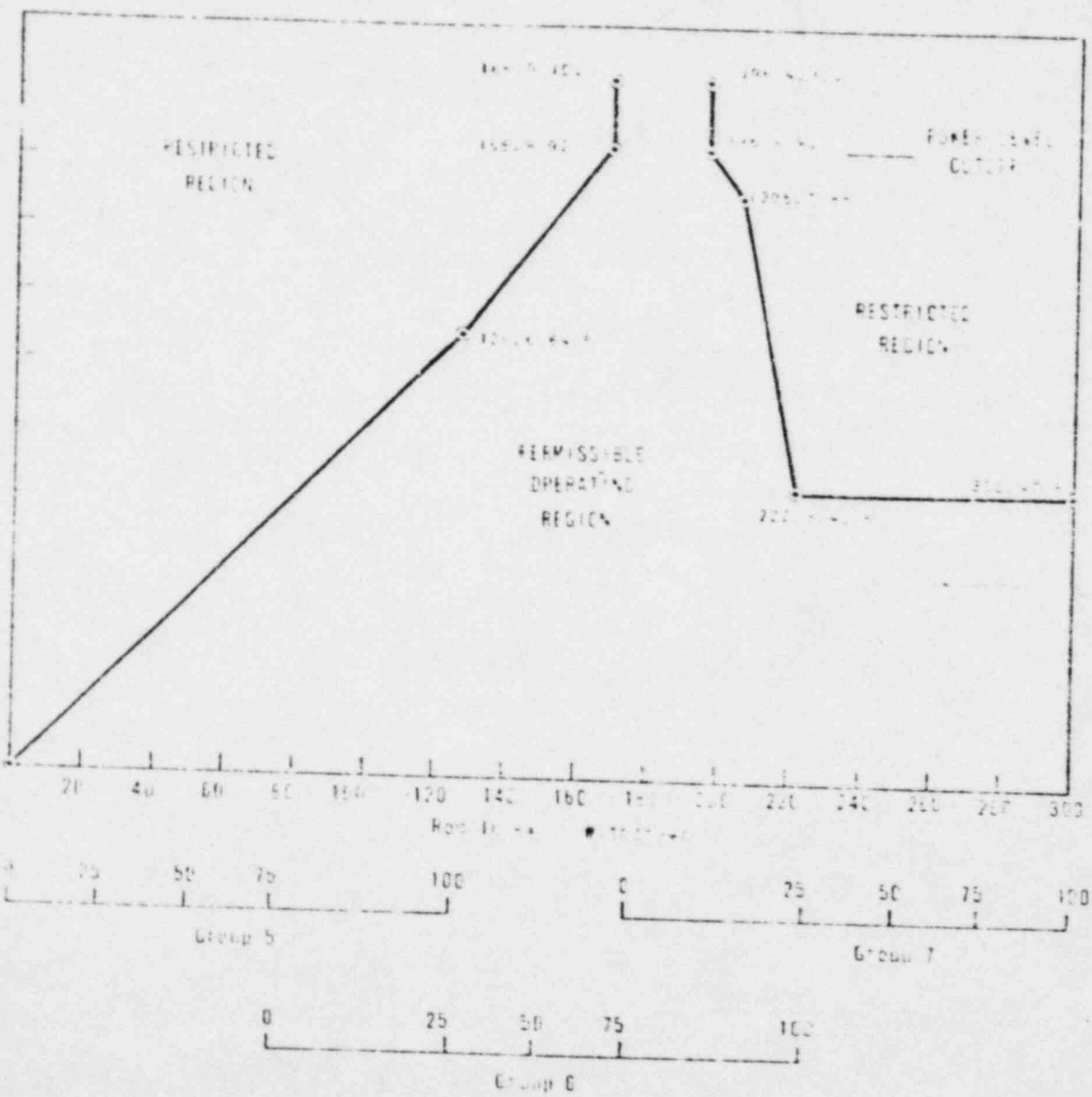


Figure 8-7. [illegible] 2, Cycle 2 — Rod Position Limits for Four-Pump Operation From [illegible] to [illegible] EFPD

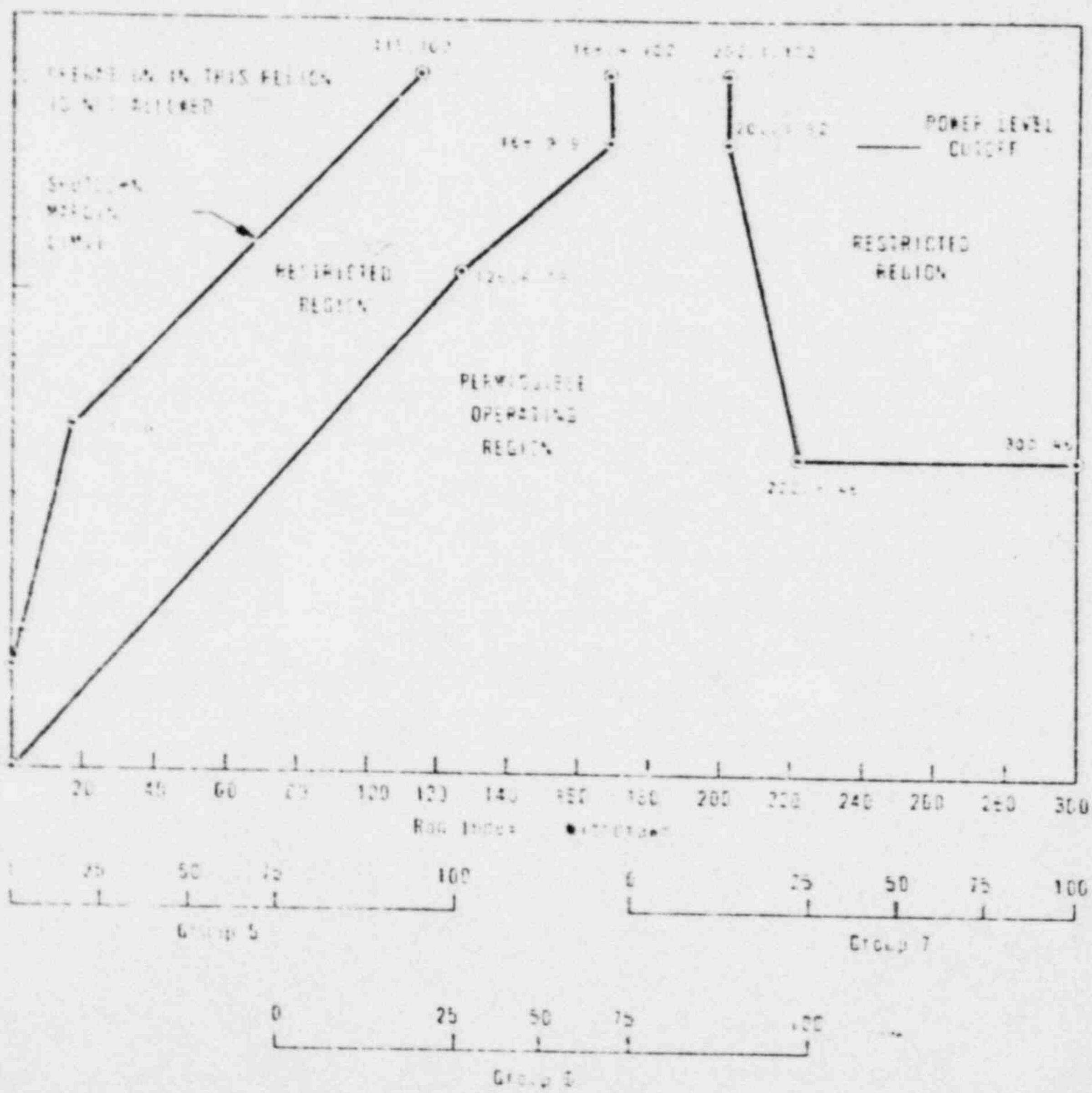


Figure 8-9. Oconee 2, Cycle 7 — Rod Position Limits for Four-Pump Operation After 267 (±10) EFPD

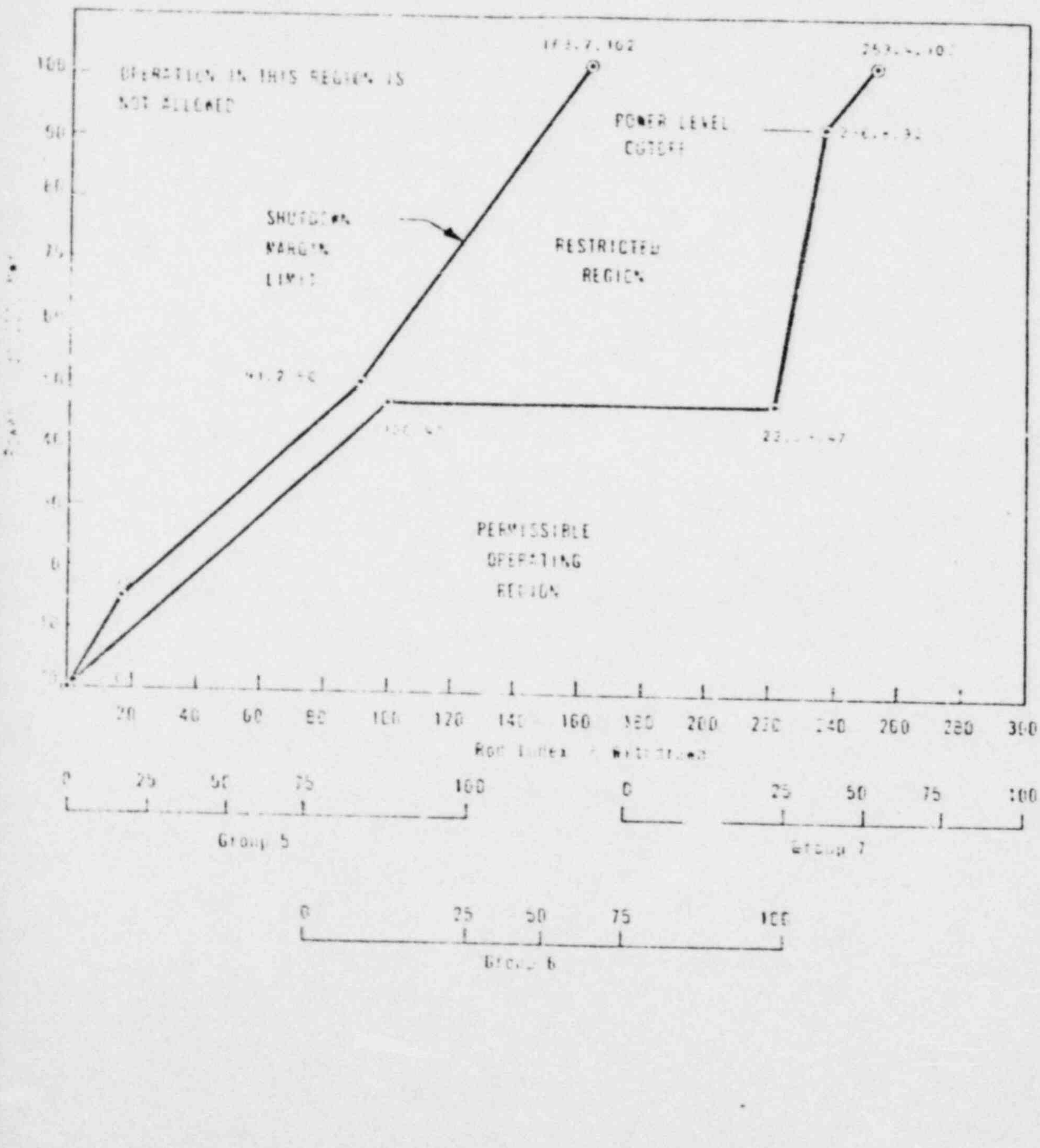


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Figure 8-9. Oconee 2, Cycle 2 — Rod Position Limits for Two- and Three-Pump Operation From 0 to 150 (±10) EFPD

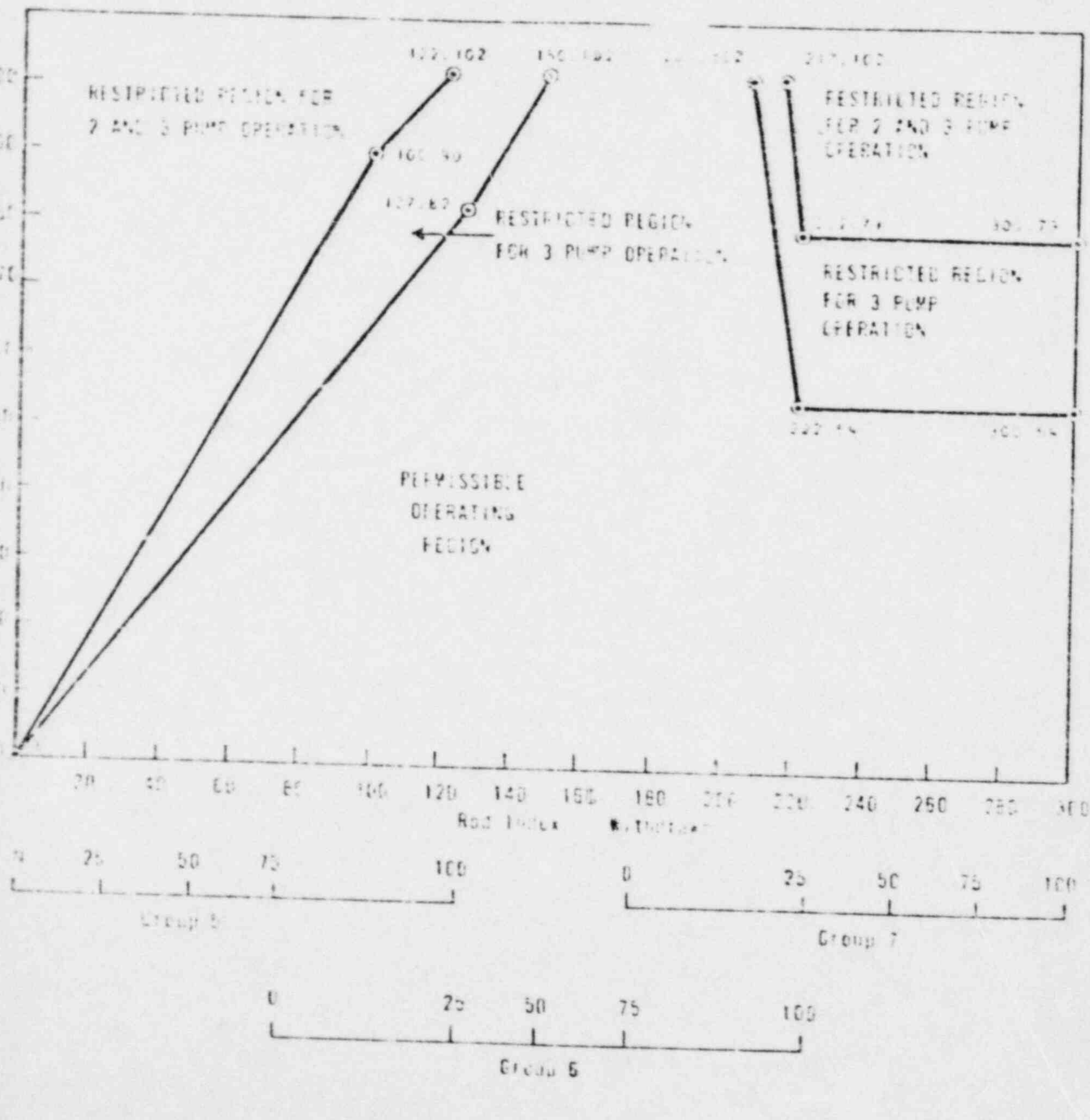


Figure 8-10. [illegible], Cycle 2 — Rod Position Limits for Two- and Three-Pump Operation From 150 (+10) to 267 (+10) EFPD

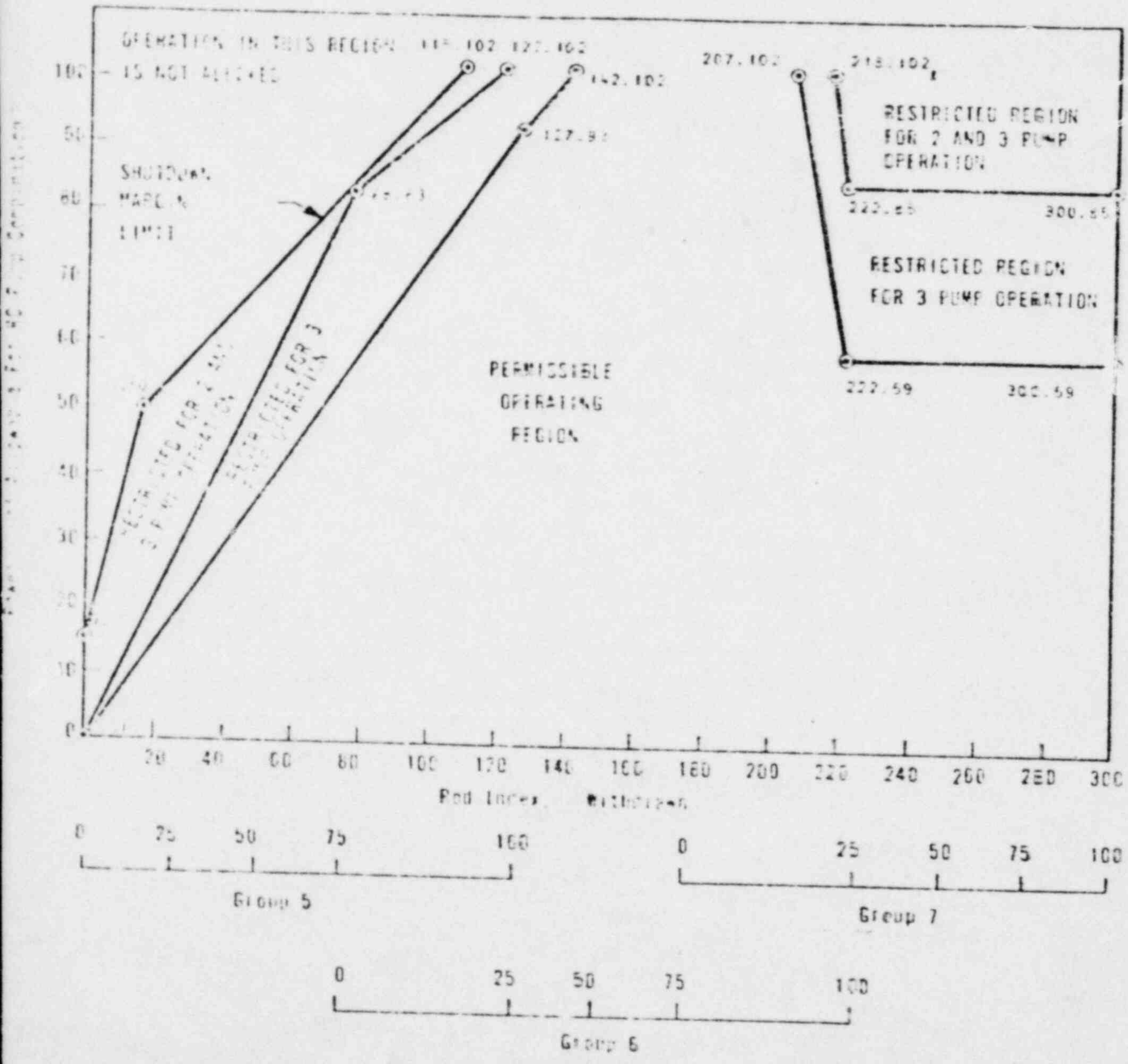


Figure 8-11. Oconee 2, Cycle 2 — Rod Position Limits for Two- and Three-Pump Operation After 267 (±10) EFPD

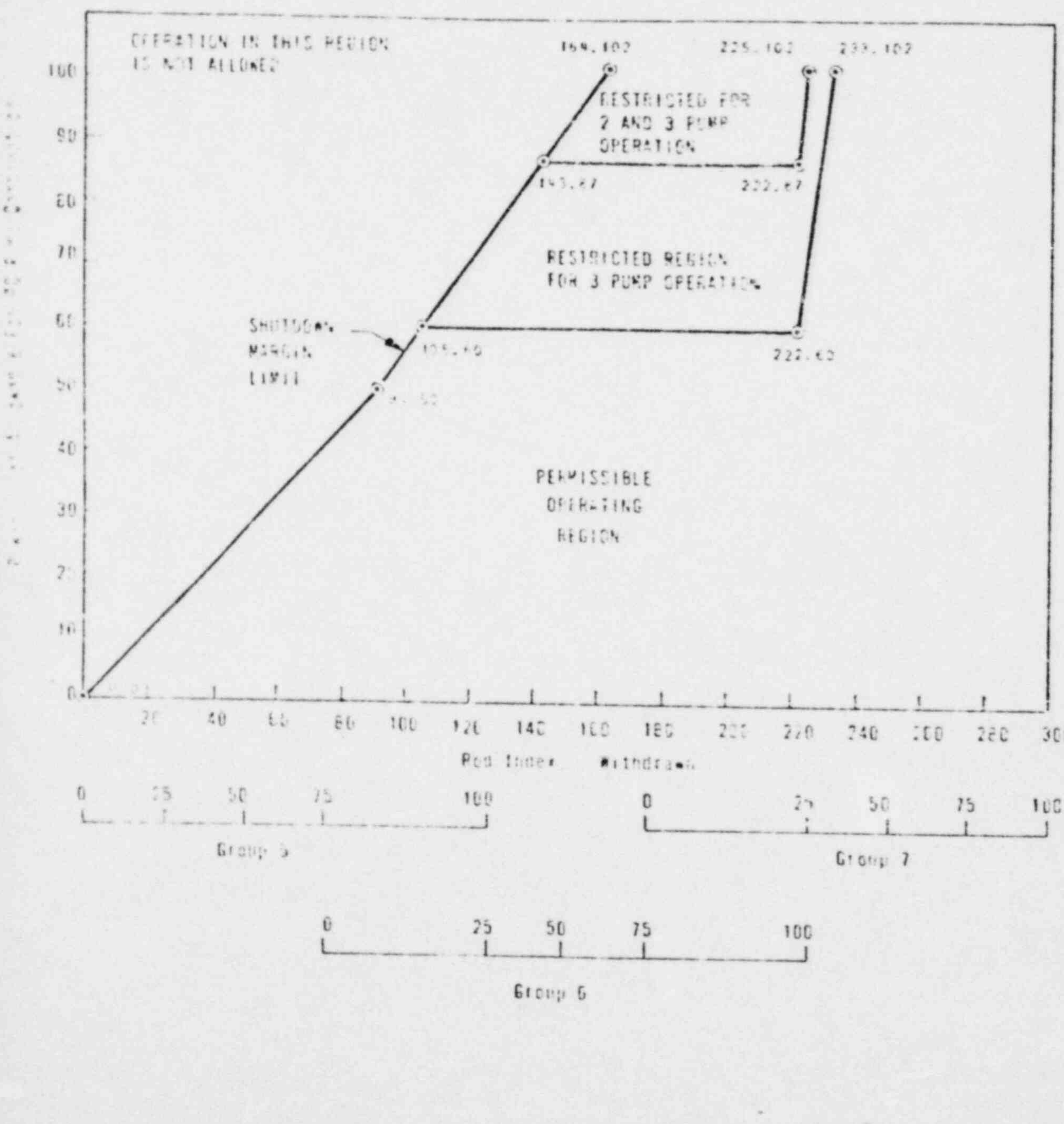


Figure 8-12. Oconee 2, Cycle 2 -- Operational Power Imbalance Envelope for Operation From 0 to 150 (±10) EFPD

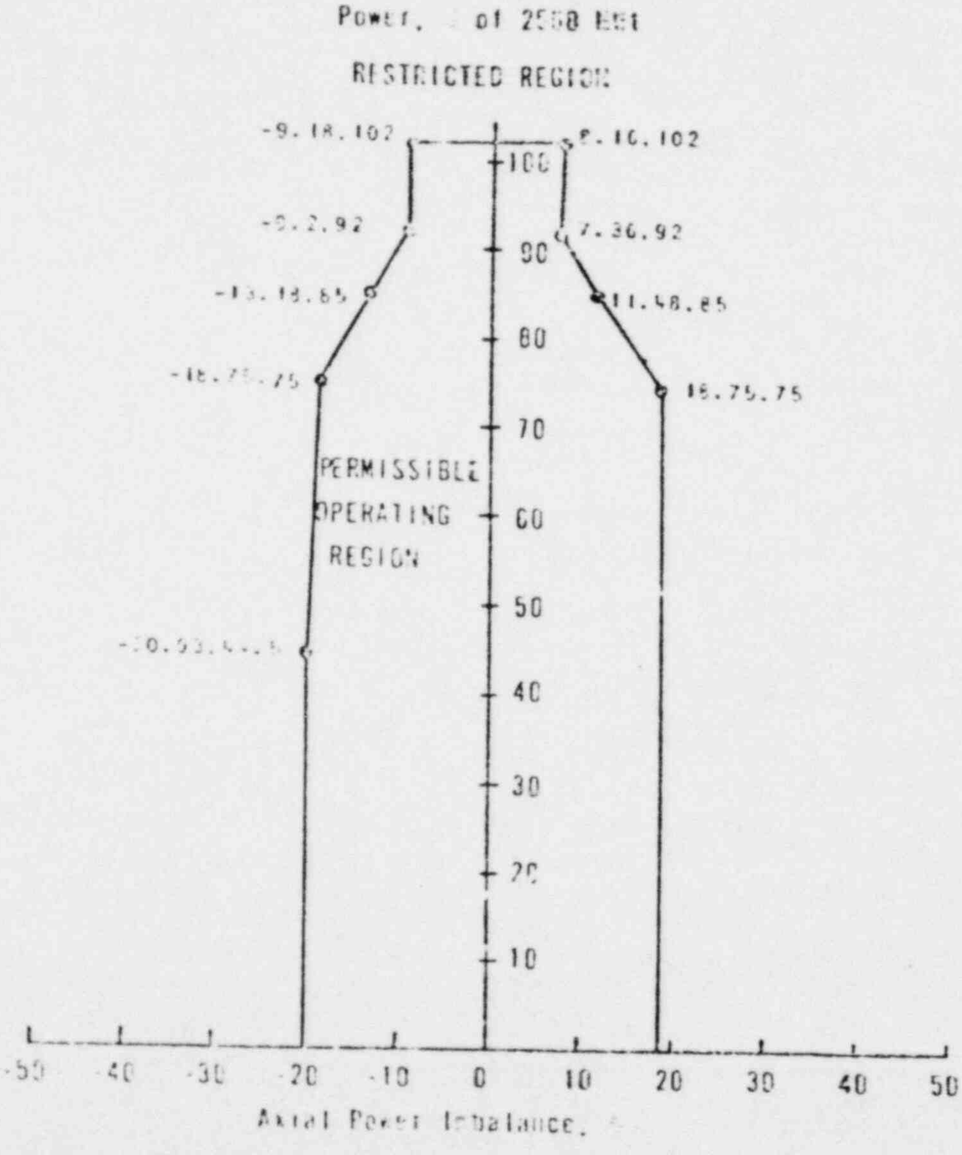


Figure 8-15. Oconee 2, Cycle 2 — Operational Power Imbalance Envelope for Operation from 150 (±10) to 267 (±10) EFPD

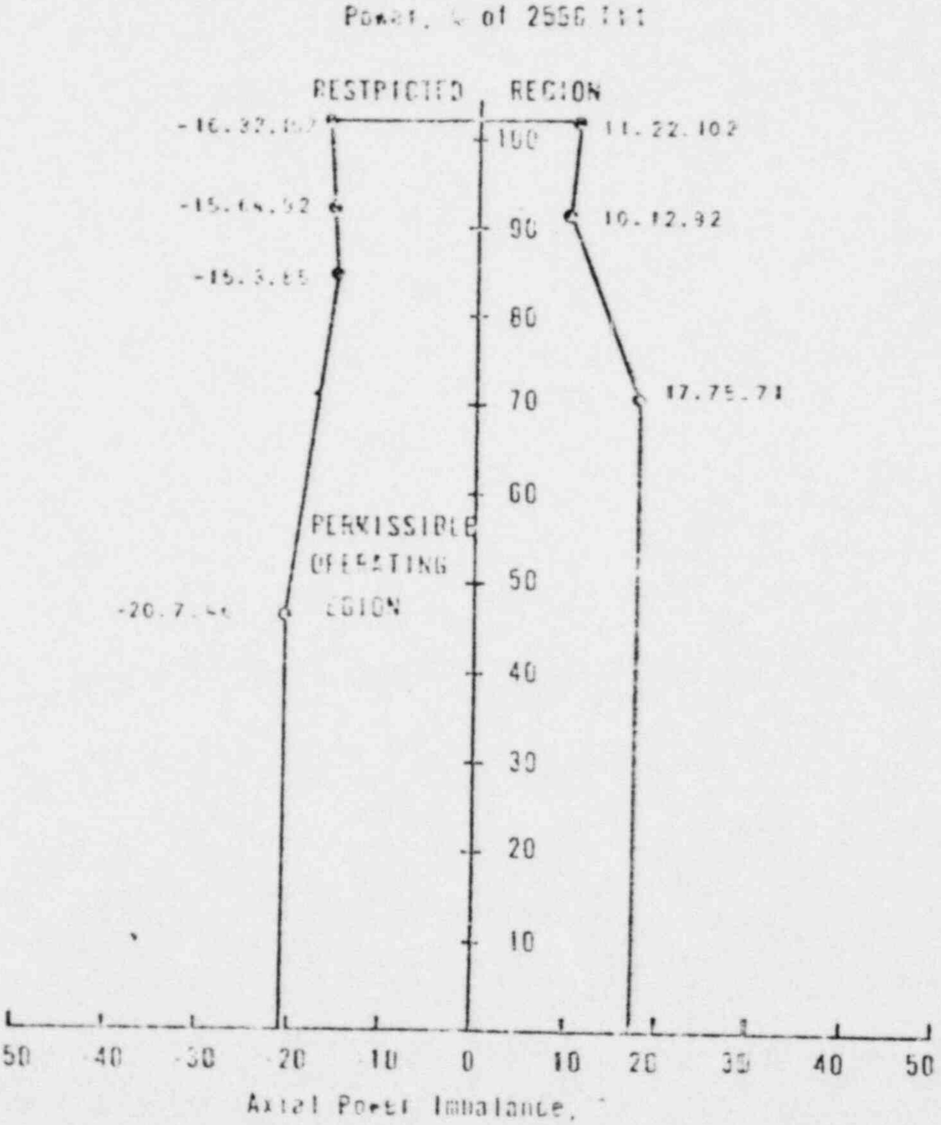
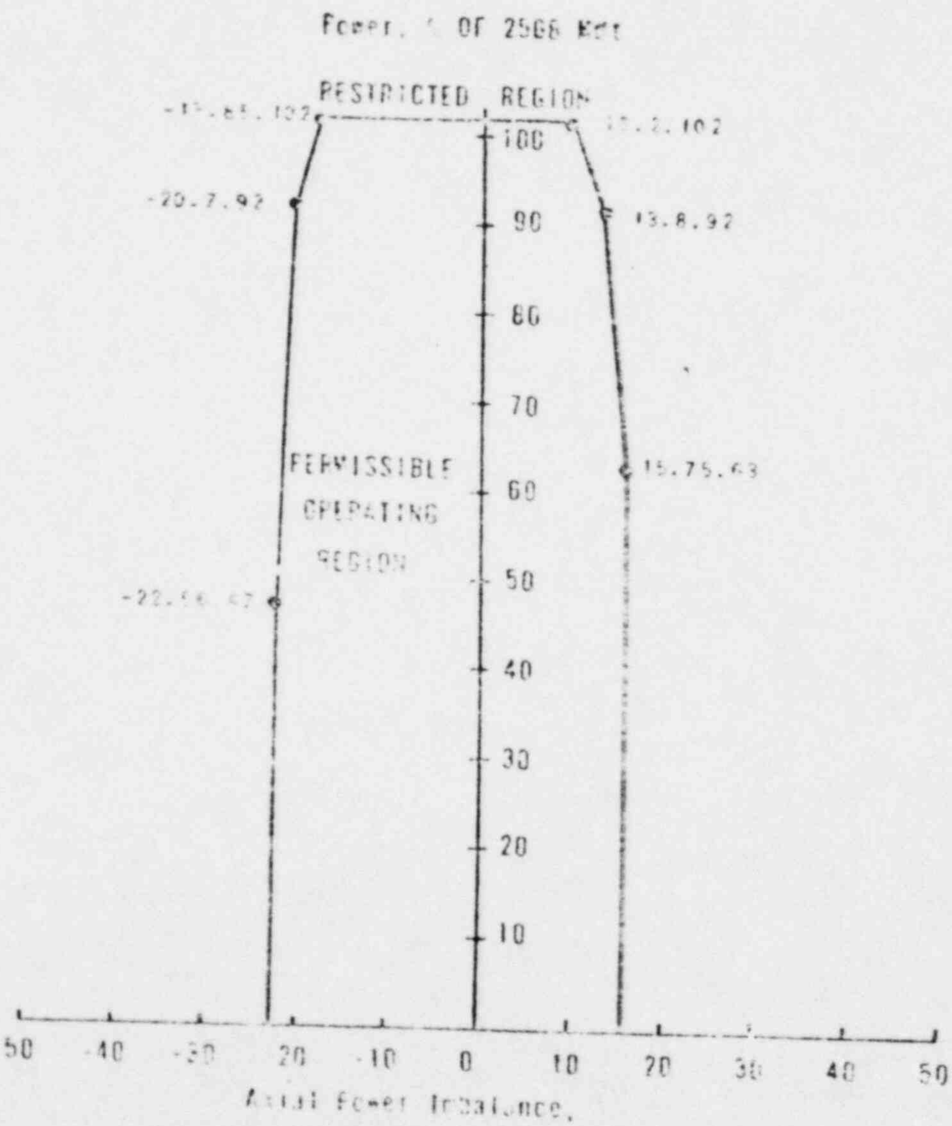


Figure 8-14. Oconee 2, Cycle 2 — Operational Power Imbalance Envelope for Operation After 267 (±15) EFPD



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