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OCONEE UNIT 1

CYCLE 3 RELOAD REPORT

Babcock & Wilcox

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OCONEE UNIT 1
CYCLE 3 RELOAD REPORT

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the third cycle of Oconee Nuclear Station, Unit I, 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 3 operation of the Oconee Nuclear Station, Unit I, this report employs analytical techniques and design bases established in reports which were previously submitted and accepted by the USNRC and its predecessor (see references).

A brief summary of cycle 1 and cycle 3 reactor parameters that are related to power capability is included in Section 7 of this report. All of the accidents analyzed in the FSAR have been reviewed for cycle 3 operation. In those cases where cycle 3 characteristics proved to be conservative with respect to those analyzed for cycle 1 operation, no new analysis was performed.

The Technical Specifications have been reviewed and the modifications required for cycle 3 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 Unit I, Cycle 3, can be safely operated at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal hydraulic analyses of Oconee Nuclear Station Unit I is the presently operating cycle 2. Cycle 2 power escalation commenced on March 11, 1975, following the completion of the zero power physics testing. The rated power level of 2568 MWt was achieved on April 13, 1975. A control rod interchange was performed at 53 effective full power days (EFPD). The design fuel cycle of 290 EFPD is scheduled for completion in January of 1976. No operating anomalies occurred during the second cycle which would adversely affect the fuel performance during the third cycle.

The nuclear and thermal-hydraulic analyses of cycle 2 utilized the BAW-2 critical heat flux correlation and the measured core flow. The cycle 3 analyses also employed these features which have the combined or singular effect of increasing margin to DNB.

Operation of cycle 3 is scheduled to begin in March of 1976. The design cycle length is 292 EFPD and one control rod interchange is planned at 100 ± 10 EFPD.

3. GENERAL DESCRIPTION

The Oconee Unit I reactor core is described in detail in Section 3 of the Oconee Nuclear Station, Unit I, Final Safety Analysis Report (reference 1).

The cycle 3 core consists of 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 pin 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 Table 4.1-1 and Table 4.2-1 for additional data.) The fuel assemblies in batch 3 have an average nominal fuel loading of 468.6 kg of uranium whereas the batch 4 and 5 assemblies maintain an average nominal fuel loading of 463.6 kg of uranium. The undensified nominal active fuel lengths and theoretical densities also vary between batches and are presented in Tables 4.1-1 and 4.2-1.

Figure 3-1 is the core loading diagram for Oconee Unit I, Cycle 3. The initial enrichments of batches 3, 4A and 4B were 2.15, 2.60 and 3.20 wt % ^{235}U, respectively. Batch 5 is enriched to 2.75 wt % ^{235}U. All of the batch 2 assemblies and 24 of the batch 3 assemblies will be discharged at the end of cycle 2. The remainder of batch 3 assemblies and the batch 4A and 4B assemblies will be shuffled to new locations at the beginning of cycle 3. The fresh batch 5 assemblies will occupy primarily the periphery of the core and 4 major axes positions slightly interior to the core. Figure 3-2 is an eighth core map showing the assembly burnup and enrichment distribution at the beginning of cycle 3.

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 axial power shaping rods are provided for additional control of the axial power distribution. The cycle 3 locations of the 69 control rods and the group designations are indicated in Figures 3-3 and 3-4. The core locations of the total pattern (69 control rods) for cycle 3 are identical to those of the reference cycle indicated in the Oconee I, Cycle 2 Reload Report (reference 2).

The group designations, however, differ between cycle 3 and the reference cycle in order to minimize power peaking. One control rod interchange is planned at 100 $\pm$ 10 EFPD.

The nominal system pressure is 2200 psia, and the densified nominal heat rate is 5.78 kw/ft at the rated core power of 2568 MWt.

Figure 3-1 Oconee I, Cycle 3
Core Loading Diagram

FUEL TRANSFER CANAL →

X

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

Z

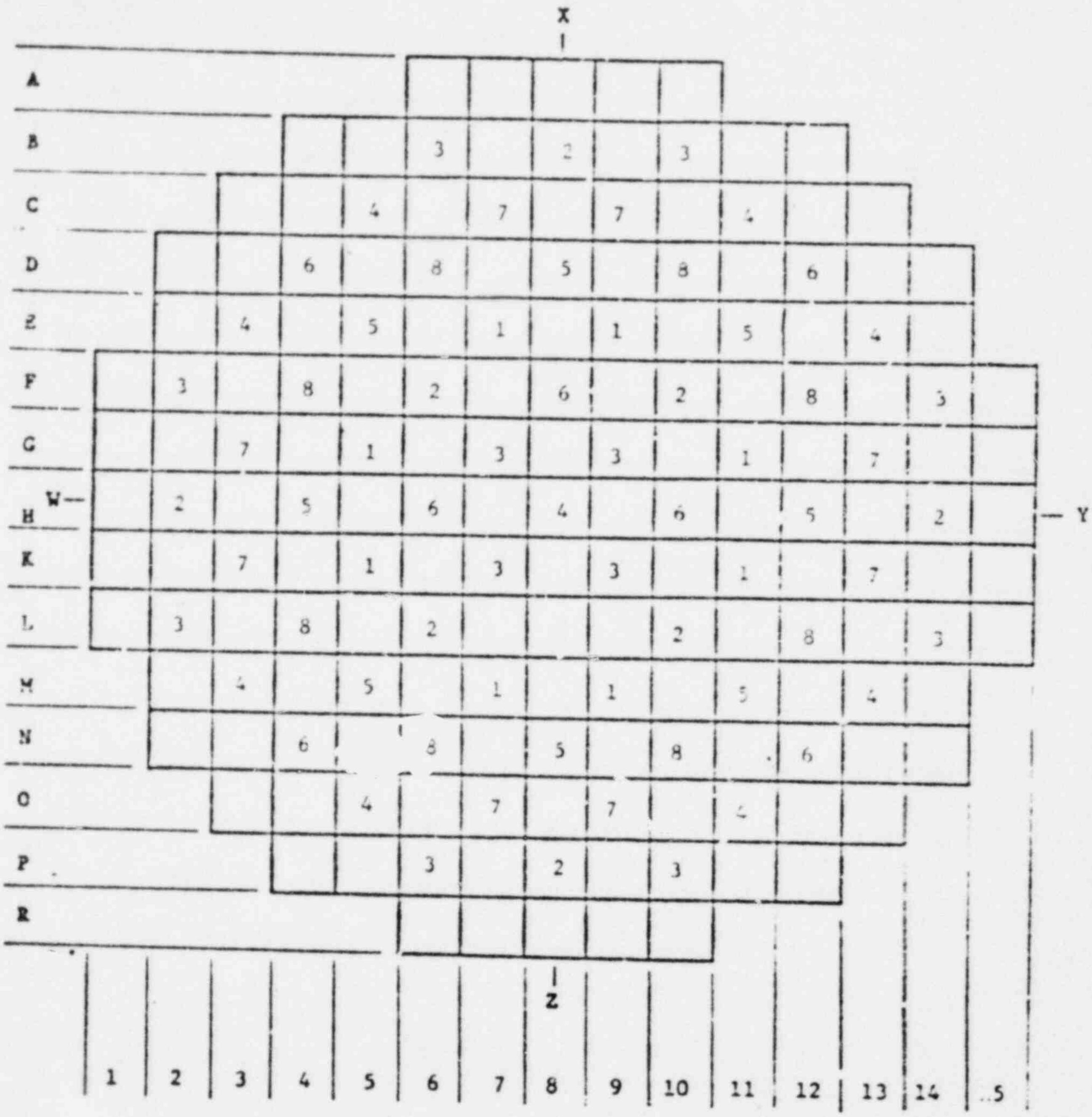
Batch Location
Cycle 2 Core Location

FIGURE 3-2 OCONEE I ENRICHMENT AND BURNUP DISTRIBUTION FOR CYCLE 3

	8	9	10	11	12	13	14	15
H	2.60 12152	2.15 13460	2.60 10877	3.20 9612	2.15 15569	2.75 0	2.15 14357	2.75 0
K		3.20 7656	2.15 15755	3.20 11913	3.20 9430	3.20 10349	2.15 18245	2.75 0
L			2.15 15788	3.20 7974	2.15 12803	3.20 7273	2.75 0	2.75 0
M				3.20 13345	2.15 12720	2.15 16419	2.75 0	
N					3.20 11010	2.75 0	2.75 0	
O						2.75 0		
P								
R								

XXX XXXX	Initial Enrichment BOC Burnup (MWD/MTU)

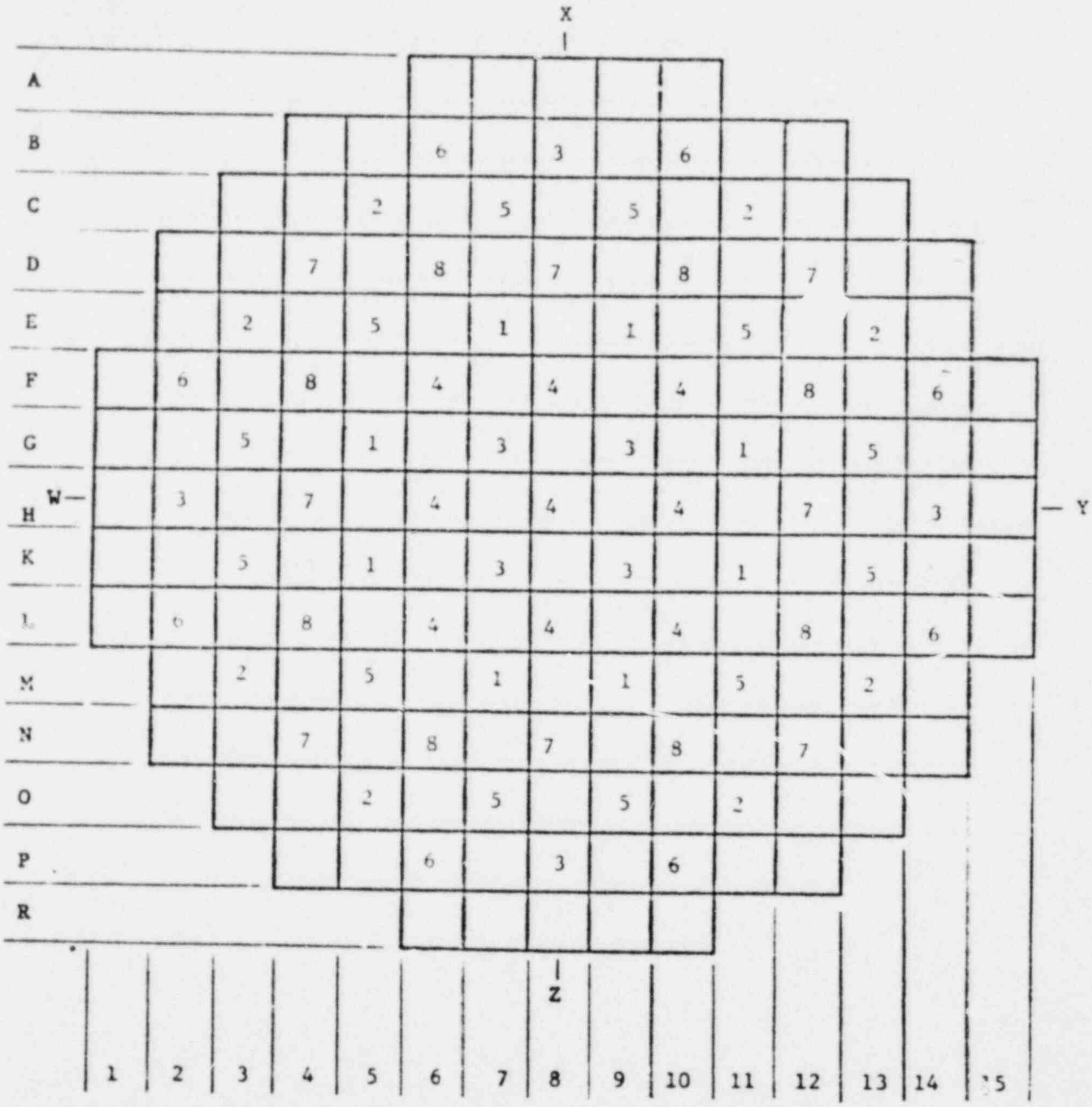
Figure 3-3 Oconee 1, Cycle 3 Control Rod Locations Before Interchange



X Group Number

Group	Number of Rods	Function
1	8	Safety
2	8	Safety
3	12	Safety
4	9	Safety
5	8	Control
6	8	Control
7	8	Control
8	8	APSR's
TOTAL	69	

Figure 3-4 Oconee 1, Cycle 3 Control Rod Locations After Interchange



X Group Number

Group	Number of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	9	Safety
5	12	Control
6	8	Control
7	8	Control
8	8	APSR'
TOTAL	69	

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

Pertinent fuel design paramaters are listed in Table 4.1-1. All fuel assemblies are identical in concept and are mechanically interchangeable. The new fuel assemblies incorporate minor modifications to the end fittings, primarily to reduce fuel assembly pressure drop and to increase holddown margin. 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 4.2-1. The mechanical evaluation of the fuel rod is discussed below.

Cladding Collapse:

Creep collapse analyses were performed for three-cycle assembly power histories for Oconee I. The batch 3 fuel is more limiting than batch 4 fuel due to the lower prepressurization and lower pellet density. A summary of the batches 3, 4, and 5 fuel rod designs is contained in Table 4.2-2. The batch 3 assembly power histories were analyzed and the most limiting assembly for cycle 3 was determined. 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 (reference 3). Measured power distribution data obtained during cycle 2 operation confirmed the accuracy of the cycle 2 design calculations used for the collapse analysis.

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, as demonstrated in reference 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 LTL (lower tolerance limit) of the as-built measurements. The initial ovality of the cladding used was the UTL (upper tolerance limit) of the as-built measurements. These values were taken from a statistical sampling of the cladding.
4. Batch 3 cladding temperatures were calculated using assembly outlet temperatures. This results in cladding temperatures which are conservatively high when combined with the maximum axial peak.

The most limiting assembly was found to have a collapse time greater than the maximum projected cycle 3 life of 21,500 hours (see Table 4.1-1). This analysis was performed using the assumptions on densification described in reference 3.

Cladding Stress:

Since the batch 3 fuel is the most limiting from a cladding stress point of view due to the low prepressurization and low density, the calculations performed in the Oconee I Fuel Densification Report, (reference 4), are the most limiting.

Fuel Pellet Irradiation Swelling:

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is set such that the plastic cladding strain is less than 1% at 55,000 MWD/MTU. The conservatisms in this analysis are listed below.

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 3 operation is shown in Figure 3-1. There are 60 fresh (batch 5) fuel assemblies, 61 once-burned (batches 4a and 4B) assemblies and 56 twice-burned (batch 3) fuel assemblies. These assemblies are thermally and geometrically similar. Limitations on the linear heat rate were established utilizing full fuel densification penalties. This results in a minimum linear heat rate capability of 20.15 kw/ft.

Power Spike Model

The power spike model utilized in this analysis is identical to that presented in BAW-10055[5] except for two modifications. The modifications have been applied to F_g and F_k[6]. 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 distribution to a linear distribution, which reflects a decreasing frequency with increasing gap size.

The power spike and maximum gap size have been calculated both for batch 4 and batch 5 fuel. The maximum gap size versus axial position is shown for both batches in Figure 4.3-1 and the power spike factor versus axial length is shown in Figure 4.3-2.

For those analyses where centerline fuel melt is limiting, the higher power spike of batch 5 fuel has been used; however. for DNBR analyses (Section 6.2), the batch 4 power spike has been used. The factor, when combined with the shorter active length of batch 4 fuel, results in the worst DNBR densification penalty.

Fuel Temperature Analysis

Thermal analysis of the fuel rods assumed in-reactor fuel densification to 96.5% theoretical density (TDF). The basis for the analysis utilized is given in BAW-10055 and BAW-10044[7] with the following modifications:

1. The option in the code for no restructuring of fuel has been used in the analysis presented here in accordance with the NRC interim evaluation of TAFY.
2. The calculated gap conductance was reduced by 25% in accordance with the NRC interim evaluation of TAFY.

During Cycle 3 operation, the highest relative assembly power levels occur in batches 4 and 5 fuel. Fuel temperature analysis for batches 1, 2, and 3 fuel is documented in the Oconee I Fuel Densification Report. This analysis is also applicable to batches 4 and 5 because they have the same linear heat rate capabilities to centerline melt as batches 1, 2 and 3 (20.15 kw/ft). The maximum hot spot centerline fuel temperature is predicted on the basis of the reference design peaking conditions as shown in Table 4.3-1.

4.4 Material Design

The batch 5 fuel assemblies are not new in concept and they do not utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 5 fuel assemblies are identical to those of the present fuel.

4.5 Operating Experiences

B&W's operating experience has been demonstrated in the operation of six 177 fuel assembly plants utilizing this fuel assembly design.

Table 4.1-1. Fuel Design Parameters

	Residual Fuel Assembly		New Fuel Assembly
	Batch 3	Batch 4	Batch 5
1. Fuel Assembly Type	Mk-B2	Mk-B3	Mk-B4
2. Number	56	61	60
3. Initial Fuel Enrichment	2.15	3.20/2.60	2.75
4. Initial Fuel Density, % Theoretical	93.5	> 94.5	93.5
5. Initial Fill Gas Pressure (Minimum specified), psia	*	*	*
6. Batch Burnup, BOL, MWD/MTU	15076	9798	0
7. Clad Collapse Time, Effective Full Power Hours	> 26,000	> 30,000**	> 26,000**

Table 4.2-1. Fuel Rod Dimensions

Component	Residual Fuel Batch 3	Residual Fuel Batch 4	New Fuel Assembly Batch 5
1. Fuel Rods			
O.D. Inches	.430	.430	.430
I.D. Inches	.377	.377	.377
2. Fuel Pellet			
O.D. Inches	.370	.3685 (mean)	.370
Density, % Theoretical	93.5	> 94.5	93.5
3. Undensified Active Fuel Length, inches	144	142	142.6
4. Flexible Spacers, Type	Corrugated Spacer	Spring	Spring
5. Solid Spacers, Material	ZrO_2	Zr-4	Zr-4

* PROPRIETARY

**A conservative power history envelope was used for batches 4 and 5 rather than specific histories.

Table 4.2-2. Input Summary for Cladding Creep Collapse Calculations

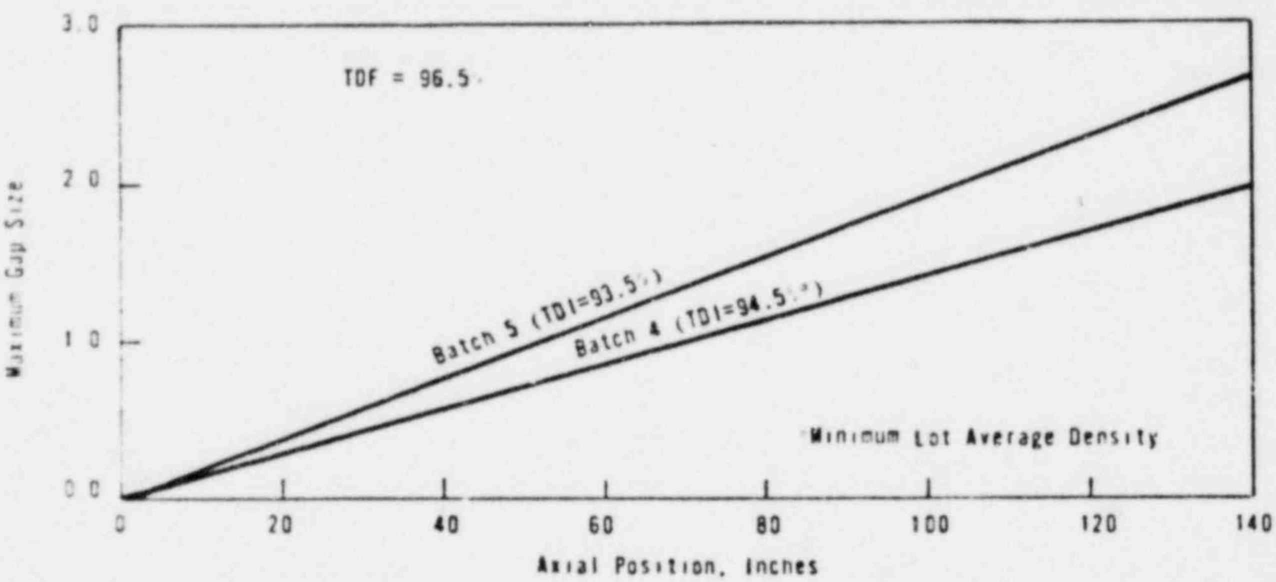
	Batch 3	Batch 4	Batch 5
Pellet OD (mean specified), in.	.3700	.3685	.3700
Pellet Density (mean specified), % TD	93.5	94.5	93.5
Densified Pellet OD, in.	.3663	.3661	.3663
Cladding ID (mean specified), in.	.377	.377	.377
Cladding Ovality, (UTL), in.	*	*	*
Cladding Thickness (LTL), in.	*	*	*
Prepressure (minimum specified), psia	*	*	*
Post Densification Prepressure (cold), psia	*	*	*
Reactor System Pressure, psia	2200	2200	2200
Stack Height (undensified), in.	144	142	142.6

* PROPRIETARY

TABLE 4.3-1

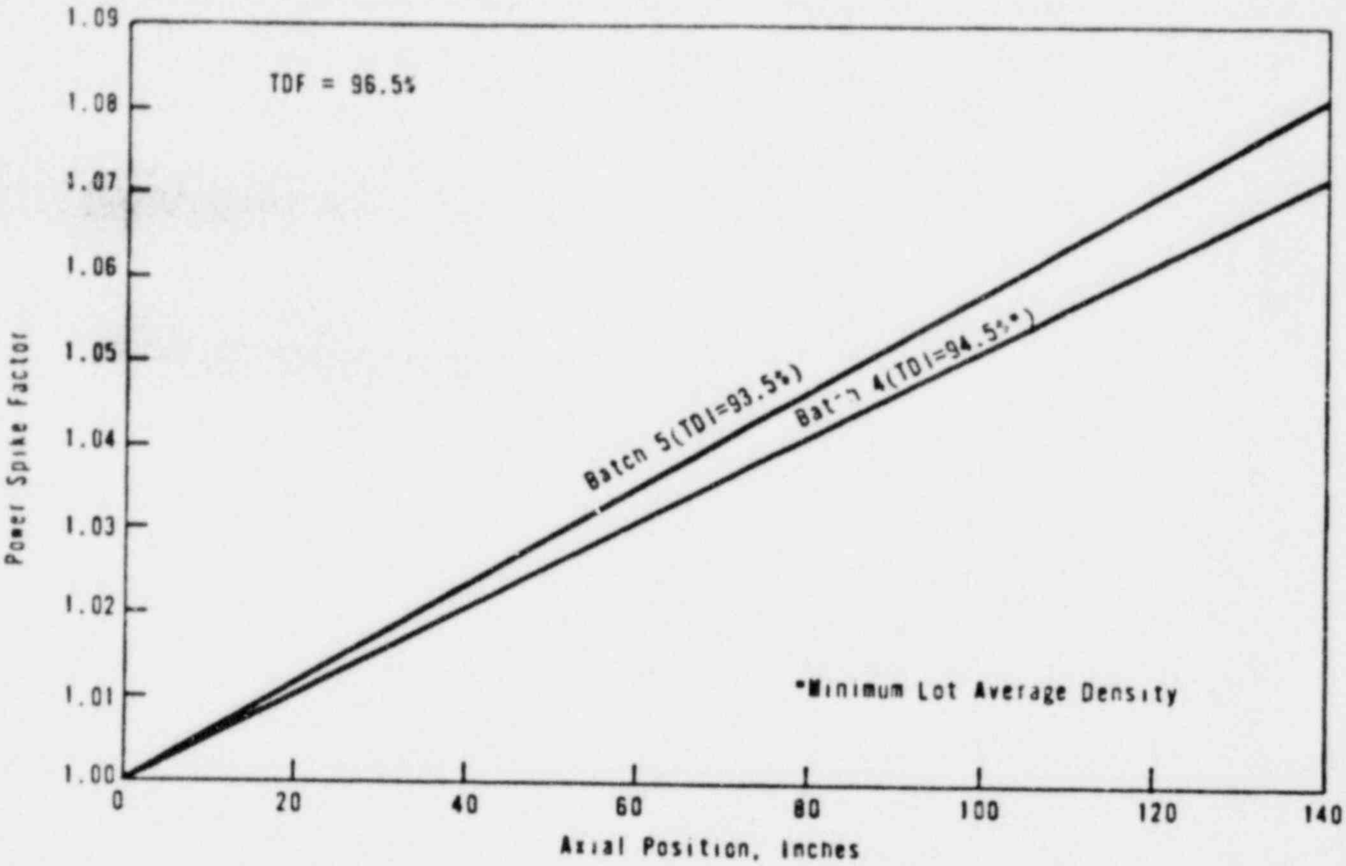
DENSIFIED FUEL TEMPERATURE ANALYSIS PARAMETERS FOR CYCLES 2 AND 3

Reactor Core Power Level, MW_t	2568
System Pressure, psia	2200
Reactor Vessel Coolant Temperature, F	579
Fraction of Heat Generated in Fuel and Cladding	.973
F_H^N	1.78
F_Z^N	1.70
Fq (Nuclear)	3.03
Fq (Nuclear and Mechanical)	3.12
Average Thermal Output kw/ft - Batch 3	5.742
Batch 4	5.805
Batch 5	5.799
Average Fuel Temperature, F	1350
Maximum Fuel Centerline Temperature at Hot Spot, F	4710
Densified Active Fuel Length, in. - Batch 3	141.84
Batch 4	140.30
Batch 5	140.46
Linear Heat Rate to Central Fuel Melt, kw/ft	20.15
Initial Theoretical Density (TDI) - Batches 3, 5	93.5
Batch 4	95.5



MAXIMUM GAP SIZE VS AXIAL POSITION

Figure 4.3-1



POWER SPIKE FACTOR VS AXIAL POSITION

Figure 4.3-2

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5.1-1 compares the core physics paramaters of cycles 2 and 3. 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 longer cycle 3 will produce a slightly larger cycle differential burnup than that for the cycle 2. The accumulated average core burnup will be higher in cycle 3 than in cycle 2 because of the presence of the once-burned batch 4 fuel and the twice burned batch 3 fuel. Figure 5.1-1 illustrates a representative relative power distribution for the beginning of the third cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 3 are approximately the same as those for cycle 2 but vary slightly due to cycle length differences, radial power distributions, etc. The control rod worths for hot full power differ between cycles due to changes in group designations as well as changes in radial flux distributions and isotopics. The ejected rod worths in Table 5.1-1 are the maximum calculated values within the allowable rod insertion limits. It is difficult to compare values between cycles or between rod patterns since neither the rod patterns from which the CRA is 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 position limits presented in Section 8. The maximum stuck rod worth for cycle 3 is lower than for cycle 2 at the beginning of cycle but higher at end of cycle. However, no adverse safety implications are associated with this higher worth since the adequacy of the shutdown margin with cycle 3 stuck rod worths is demonstrated in Table 5.1-2. For the shutdown calculations the following conservatisms were applied.

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 3 is analyzed at approximately 230 EFPD's. This is the latest time (± 5 days) in core life in which the transient bank is nearly fully inserted. After 230 EFPD's the transient bank will be almost fully withdrawn thus increasing the available shutdown margin. Reference fuel cycle shutdown margin is presented in the Oconee I, Cycle 2 Reload Report.

The cycle 3 power deficits from hot zero power to hot full power are higher than those for cycle 2 due to a more negative moderator coefficient in cycle 3. The differential boron worths and total xenon worths for cycle 3 are the same or lower than for cycle 2 due to depletion of the fuel and the associated buildup of fission products. Effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2 Analytical Input

The cycle 3 incore measurement calculation constants to be used for computing core power distributions were prepared in the same manner as the reference cycle.

5.3 Changes in Nuclear Design

There were no relevant changes in core design between the reference and reload cycles. The same calculational methods and design information were used to obtain the important nuclear design parameters. In addition, no significant operational procedure changes exist 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.

A fuel melt limit of 20.15 kw/ft has been employed in calculating the Reactor Protection System (RPS) setpoints and is the same as in cycles 1 and 2. The batch 5 fuel assemblies will be loaded as in Figure 3-1. As-built data have been used to ensure eighth-core symmetry in ^{235}U loading. The three batch 4 assemblies that had been assigned a maximum linear power rating of 20.02 kw/ft based on as-built data will again be placed in lower power core locations. These locations (E-10, L-11, and M-6) have been investigated and it has been determined that after the fuel has been shuffled to these cycle 3 core locations, they will not experience greater than 19.4 kw/ft through cycle 3. Thus, a sufficient fuel melt margin will be maintained through cycle 3. See reference 2 for a detailed outline of the methods involved.

In addition, assembly 1D61, which contains simulated fuel column gaps, will be placed in core location F-13 in conjunction with B&W's continuing program to evaluate fuel performance. Contained in one fuel rod of assembly 1D61 are three ceramic spacers which simulate fuel densification gaps. The description of the irradiation program for this special assembly in Oconee Unit I was presented in a letter (6/18/74) to Angelo Giambusso, USAEC. Continuation of the irradiation of assembly 1D61 will not adversely affect fuel or reactor performance during cycle 3.

Table 5.1-1 Cycle 2 and Cycle 3 Physics Parameters

	Cycle 2	Cycle 3
Cycle length, EFPD	290	292
Cycle burnup, MWd/mtU	9000	9107
Average core burnup - EOC, MWd/mtU	14,550	17,254
Initial core loading, mtU	82.6	82.3
Critical boron - BOC, ppm		
HZP* - all rods out	1285	1332
HZP - groups 7 and 8 inserted	1159	1169
HFP - groups 7 and 8 inserted	1028	977
Critical boron - EOC, ppm		
HZP - all rods out	285	364
HFP - group 8 (37.5% withdrawn, equil. Xe)	75	56
Control rod worths - HFP, BOC, %Δk/k		
Group 6	1.12	1.30
Group 7	1.14	1.30
Group 8 (37.5% wd)	0.36	0.45
Control rod worths - HFP, EOC, %Δk/k		
Group 7	1.97	1.31
Group 8 (37.5% wd)	0.41	0.46
Maximum ejected rod worth - HZP, %Δk/k		
Rod configuration 1	0.71	0.82
Rod configuration 2	0.80	0.27
Maximum stuck rod worth - HZP, %Δk/k		
BOC	2.55	2.34
EOC	1.96	2.71
Power deficit, HZP to HFP, %Δk/k		
BOC (groups 7 and 8 inserted)	1.34	1.45
EOC (groups 7 and 8 inserted)	1.99	2.6[illegible]**
Doppler coeff. - BOC, 10^{-5} Δk/k/F		
100% power (0 Xe)	-1.60	-1.60
Doppler coeff. - EOC, 10^{-5} Δk/k/F		
100% power (equil. Xe)	-1.62	-1.62

Table 5.1-1 (Continued)

	Cycle 2	Cycle 3
Moderator coeff. - HFP, 10^{-4} $\Delta k/k/F$		
BOC (0 Xe, 1000 ppm, groups 7 and 8 inserted)	-0.79	-0.89
EOC (equil. Xe, 17 ppm, group 8 inserted)	-2.35	-2.42
Boron worth - HFP, ppm/%$\Delta k/k$		
BOC (1000 ppm)	97	103
EOC (17 ppm)	91	92
Xenon worth - HFP, %$\Delta k/k$		
BOC (4 days)	2.64	2.64
EOC (equilibrium)	2.69	2.67
Effective delayed neutron fraction (HFP)		
BOC	.00602	.00582
EOC	.00520	.00518

* HZP denotes Hot Zero Power/HFP denotes Hot Full Power

** Power deficit at 230 EFPD's

Table 5.1-2 Shutdown Margin Calculation

Oconee 1, Cycle 3

	BOC, % k/k	EOC*, % k/k
I. Available Rod Worth		
a. Total rod worth, HZP	9.43	9.21
b. Worth reduction due to burnup of poison material	-0.33	-0.37
c. Maximum stuck rod, HZP	-2.34	-2.64
d. Net worth	6.76	6.20
e. Less 10% uncertainty	-0.68	-0.62
f. Total available worth	6.08	5.58
II. Required Rod Worth		
a. Power deficit, HFP to HZP	1.45	2.00
b. Maximum allowable inserted rod worth	1.53	1.39
c. Flux redistribution	.40	0.69
d. Total required worth	3.38	4.08
III. Shutdown Margin		
(I.f. minus II.d.)	2.70	1.50

NOTE: Required shutdown margin is 1.00% k/k

*For shutdown margin calculations, this is defined as ~ 230 EFPD, the latest time in core life in which the transient bank is nearly full in.

FIGURE 5.1-1 BOC (4 EFPD), Cycle 3 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	1.25	1.16	1.28	1.47	1.00	1.01	.62	.59
K	1.16	1.44	1.09	1.40	1.22	7 .68	.62	.61
L	1.28	1.09	1.07	1.41	8 .91	1.11	1.03	.57
M	1.47	1.40	1.41	1.29	.97	.90	.94	
N	1.00	1.22	8 .91	.97	1.29	1.23	.72	
O	1.01	7 .68	1.11	.90	1.23	.58		
P	.62	.62	1.03	.94	.72			
R	.59	.61	.57					

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

6. THERMAL-HYDRAULIC DESIGN

6.1 Thermal-Hydraulic Design Calculations

Thermal-hydraulic design calculations for support of cycle 3 operation utilized the same analytical methods previously documented in references 1 and 2. Adjustments to these calculations recognize the introduction of the Mk-B4 assemblies in batch 5 and account for modifications in the use of the B&W-2 critical heat flux correlation.[8,9] The B&W correlation was utilized in the licensing of the Oconee I Cycle 2 core. In the application of the B&W-2 CHF correlation to the Oconee I, Cycle 3 core, two modifications in the use of the correlation have been instituted. The following modifications have also been applied to the TMI-I, Cycle 2 core.

1. The limiting design DNBR of 1.30, representing a 95 percent confidence level for 95 percent population protection, was used in the analysis. A limiting DNBR of 1.32 representing a 99 percent confidence level for a 95 percent protection was used in previous design analyses. This change is consistent with industry practice and statistical standards associated with limiting design DNBR values as accepted by the NRC Staff and ACRS.
2. The pressure range applicable to the correlation has been extended downward from 2000 psia 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.

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 total Mark B as-built data were taken into account. This resulted in a minimum DNBR of 2.0 at 112 percent power for undensified fuel.

The effects due to densification can be divided into two categories: (1) the result of reduced stack height and (2) power spiking caused by densification induced gaps in the fuel column. As input to the DNBR analysis for batch 4 fuel (most limiting), the minimum lot average density and the densified as-built stack height were used. Using this input and the corresponding power spike, the most limiting DNBR conditions were calculated for cycle 3 operation.

The axial flux shape which resulted in the maximum change in DNBR from the original design value was an outlet peak with a core offset of +11.8%. The spike magnitude and the maximum gap size are discussed in Section 4.3 and the values used in the analysis are 1.07 and 1.96 inches, respectively. The results of the two effects are -5.4% and -3.0% change in minimum hot channel DNBR and peaking margin, respectively. These numbers are summarized in Table 6.2-1 which includes comparisons of other pertinent cycle 2 and cycle 3 data.

Table 6.2-1 Cycle 2 and Cycle 3 Design Conditions

	Cycle 2	Cycle 3
Power Level, MW_t	2568	2568
System Pressure, psia	2200	2200
Reactor Coolant Flow, % Design Flow	107.6	107.6
Vessel Inlet Coolant Temperature - 100% Power, F	555.9	555.9
Vessel Outlet Coolant Temperature - 100% Power, F	602.26	602.26
Ref. Design Radial - Local Power Peaking Factor	1.78	1.78
Ref. Design Axial Flux Shape	1.5 cosine	1.5 cosine
Densified Active Fuel Length*	139.64	139.64
Average Heat Flux (100% Power), $Btu/h\text{-}ft^2$	176472	176472
Maximum Heat Flux (100% Power), $Btu/h\text{-}ft^2$ (for DNBR calculations)	471180	471180
CHF Correlation	B&W-2	B&W-2
Minimum DNBR (Max. Design Conditions, No Densification Penalties)	2.0 (112% power)	2.0 (112% Power)
Hot Channel Factors		
Enthalpy Rise	1.011	1.011
Heat Flux	1.014	1.014
Flow Area	0.98	0.98
Densification Effects		
Change in DNBR Margin, %	-5.4	-5.4
Change in Power Peaking Margin, %	-3.0	-3.0

* Number used for DNBR analysis (batch 4 length). See Table 4.3-1 for values for batches 3 & 5.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR accident analysis has been examined with respect to changes in cycle 3 parameters to determine the effects of the cycle 3 reload and to ensure that thermal performance during hypothetical transients is not degraded.

Core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. A comparison of cycle 2 values of core thermal parameters with parameters used in cycle 3 analysis is given in Table 6.2.1. Cycle 2 and cycle 1 core thermal parameters are compared in reference 2. These are parameters common to all of the accident analyses presented herein. For each accident of the FSAR, a discussion of the accident and the key parameters are provided. A comparison of the key parameters (See Table 7.1-1) from the FSAR and cycle 3 is provided with the accident discussions to show that the initial conditions of the transient are bounded by the FSAR analysis.

The effects of fuel densification on the FSAR accident analysis results have been evaluated and are reported in reference 4. Since batch 5 reload fuel assemblies do not contain fuel rods whose theoretical density is lower than those considered in reference 4, the conclusions in that reference are still valid.

Calculational techniques and methods for cycle 3 analysis remain consistent with those used for the FSAR. Additional DNBR margin is shown for cycle 3 due to use of the B&W-2 CHF correlation instead of the W-3 CHF correlation.

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

7.2 Rod Withdrawal Accidents

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

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, ignoring multiple alarms, interlocks and trips that normally preclude this type of incident.

For positive reactivity addition indicative of these events, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were -1.17×10^{-5} $\Delta k/k/F$ for the Doppler coefficient $+0.5 \times 10^{-4}$ $\Delta k/k/F$ for the moderator temperature coefficient and rod group worths up to and including a 10.0% $\Delta k/k$ rod worth. Comparable cycle 3 parametric values are -1.60×10^{-5} $\Delta k/k/F$ for Doppler coefficient, -0.89×10^{-4} $\Delta k/k/F$ for moderator temperature coefficient, and maximum rod bank worth of 9.4% $\Delta k/k$. Therefore, cycle 3 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 and the fuel densification report.

7.3 Moderator Dilution Accident

Boron in the form of boric acid is utilized to control exess 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 reactor coolant system under conditions of full power operation, hot shutdown and during refueling.

The key parameters in this analysis are the initial boron concentration, boron reactivity worth, and moderator temperature coefficent 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 $+0.94 \times 10^{-4}$ $\Delta k/k/F$ for the moderator temperature coefficient. Comparable cycle 3 values are 997 ppm for the initial boron concentration, 78 ppm/1% $\Delta k/k$ boron reactivity worth and -0.89×10^{-4} $\Delta k/k/F$ for the 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 3 parameter values of importance to 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

The NSS does not contain any check or isolation values in the reactor coolant system piping, therefore, the classical cold water accident is not possible. However, when the reactor is operated with one or more pumps not running, and the idle pumps are started, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, reactivity will be added to the core and a power increase will occur.

Protective interlocks and administrative procedures exist to prevent the starting of idle pumps if the reactor power is above 22%. However, these restrictions were not assumed 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 -1.3×10^{-5} $\Delta k/k/F$. The corresponding most negative moderator temperature coefficient and least negative Doppler coefficient predicted for cycle 3 are -2.42×10^{-4} $\Delta k/k/F$ and -1.60×10^{-5} $\Delta k/k/F$, respectively. As the predicted cycle 3 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

A reduction in the reactor coolant flow can occur from mechanical failure or from a loss of electrical power to the pumps. With four independent pumps available, a mechanical failure in one pump will not affect operation of others. With the reactor at power, the effect of loss of coolant flow is a rapid increase in coolant temperature due to reduction of heat removal capability. This increase could result in DNB if corrective action were not taken immediately. The key parameters for 4-pump coastdown or locked rotor incident are the flow rate, flow coastdown characteristics, Doppler coefficient, moderator temperature coefficient, and hot channel DNB peaking factors. The conservative initial conditions assumed for the densification report were: FSAR values of flow and coastdown, -1.2×10^{-5} $\Delta k/k/F$ Doppler coefficient, $+0.5\times10^{-4}$ $\Delta k/k/F$ moderator temperature coefficient, with densified fuel power spike and peaking. The results showed the DNBR remained above 1.3 (W-3) for the 4-pump coastdown and the fuel cladding temperature remained below criteria limits for the locked rotor transient.

The predicted parametric values for cycle 3 are -1.60×10^{-5} Δk/k/F Doppler coefficient, -0.89×10^{-4} Δk/k/F moderator temperature coefficient and peaking factors as shown in Table 6.2-1. Since the B&W-2 CHF correlation was used for cycle 3 and the predicted cycle 3 values are bounded by those used in the densification report, the results of that analysis represents the most severe consequences from a loss of flow incident.

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

If a control rod is dropped into the core while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution may be distorted due to a new control rod pattern. Therefore, under these conditions a return to rated power may lead to localized power densities and heat fluxes in excess of design limitations.

The key parameters for this transient are moderator temperature coefficient, worth of dropped rod, and local peaking factors. The FSAR analysis was based on 0.46% Δk/k and 0.36% Δk/k rod worths with a moderator temperature coefficient of -3.0×10^{-4} Δk/k/F. For cycle 3, the maximum worth rod at power is 0.20% Δk/k and the moderator temperature coefficient is -2.42×10^{-4} Δk/k/F. Since the predicted rod worth is less and the moderator temperature coefficient 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: (i) a loss of load condition, caused by separation of the unit from the transmission system, and (ii) a hypothetical condition which results in a complete loss of all system and unit power except 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 resulted 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 and, 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 blowdown, causing a sudden decrease in 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 to be -3.0×10^{-4} $\Delta k/k/F$ in the FSAR. The cycle 3 predicted value of moderator temperature coefficient is -2.42×10^{-4} $\Delta k/k/F$. This value is bounded by the value used in the FSAR analysis and hence, the results in the FSAR represent the worst situation.

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 severence 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 type of accident is considered the maximum potential source of activity release during fuel handling activity. The primary concern is over radiological releases which are independent of core loading and, therefore, the results of the FSAR are applicable to all reloads.

7.11 Rod Ejection Accident

For reactivity to be added to the core at a more rapid rate than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the control 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. 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 of -1.17×10^{-5} $\Delta k/k/F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $\Delta k/k/F$ moderator temperature coefficient, and ejected rod worth of 0.50% $\Delta k/k$ represent the maximum possible transient. The corresponding cycle 3 parametric values of -1.60×10^{-5} $\Delta k/k/F$ Doppler and -0.59×10^{-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 0.25% Δk/k ensure that the results will be less severe than those presented in the FSAR and densification report.

7.12 Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur, since this 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. Therefore, 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 and, therefore, the results reported in the FSAR are applicable to any reload.

7.14 LOCA Analysis

A generic LOCA analysis for B&W 177 FA lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103 (reference 10). The analysis in BAW-10103 is generic in nature since the limiting values of key parameters for all plants in the category were used. Furthermore, the average fuel temperature as a function of the linear heat rate and the lifetime pin pressure data used in the BAW-10103 LOCA limits analysis are conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee I, Cycle 3.

Table 7.14-1 shows the bounding values for allowable LOCA peak linear heat rates for Oconee I, Cycle 3 fuel.

TABLE 7.1-1 Comparison of Key Parameters for Accident Analysis

Parameter	FSAR & Densification Report Value	Predicted Cycle 3 Value
Doppler Coefficient, BOL	$-1.17\times10^{-5}\ \Delta k/k/F$	$-1.60\times10^{-5}\ \Delta k/k/F$
, EOL	$-1.33\times10^{-5}\ \Delta k/k/F$	$-1.62\times10^{-5}\ \Delta k/k/F$
Moderator Coefficient, BOL	$+0.5\times10^{-4}\ \Delta k/k/F$	$-0.89\times10^{-4}\ \Delta k/k/F$
EOL	$-3.0\times10^{-4}\ \Delta k/k/F$	$-2.42\times10^{-4}\ \Delta k/k/F$
All Rod Bank Worth (HZP)	10.0%Δk/k	9.4[illegible]%Δk/k
Initial Boron Concentration	1400 ppm	977 ppm
Boron Reactivity Worth	75 ppm/1%Δk/k	78 ppm/1%Δk/k
Max. Ejected Rod Worth (HFP)	0.50%Δk/k	0.25[illegible]Δk/k
Dropped Rod Worth, HFP	0.46%Δk/k	0.20[illegible]Δk/k

Table 7.14-1 ALLOWABLE LOCA PEAK LINEAR HEAT RATE

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

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 3 operation. The changes made are as a result of:

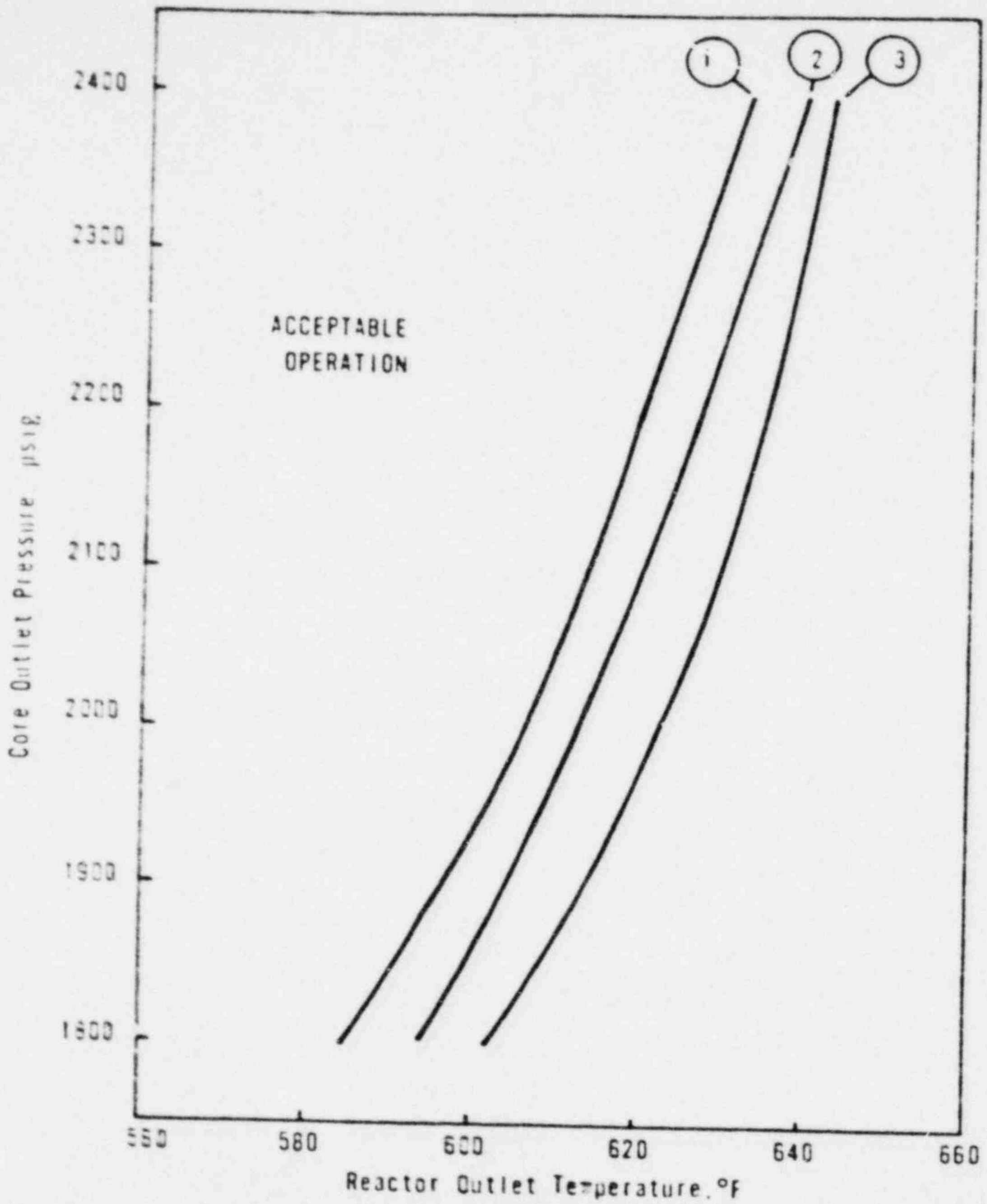
(1) The use of a 95/95 confidence level rather than 99/95 as discussed in Section 6.1.

(2) The increase in range of applicability of the B&W-2 CHF correlation as discussed in Section 6.1.

(3) The use of the Final Acceptance Criteria LOCA analyses for restricting peaks during operation as discussed in Section 7.14.

(4) A revision to the assumptions upon 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.

Based upon the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded and the thermal design criteria will not be violated.

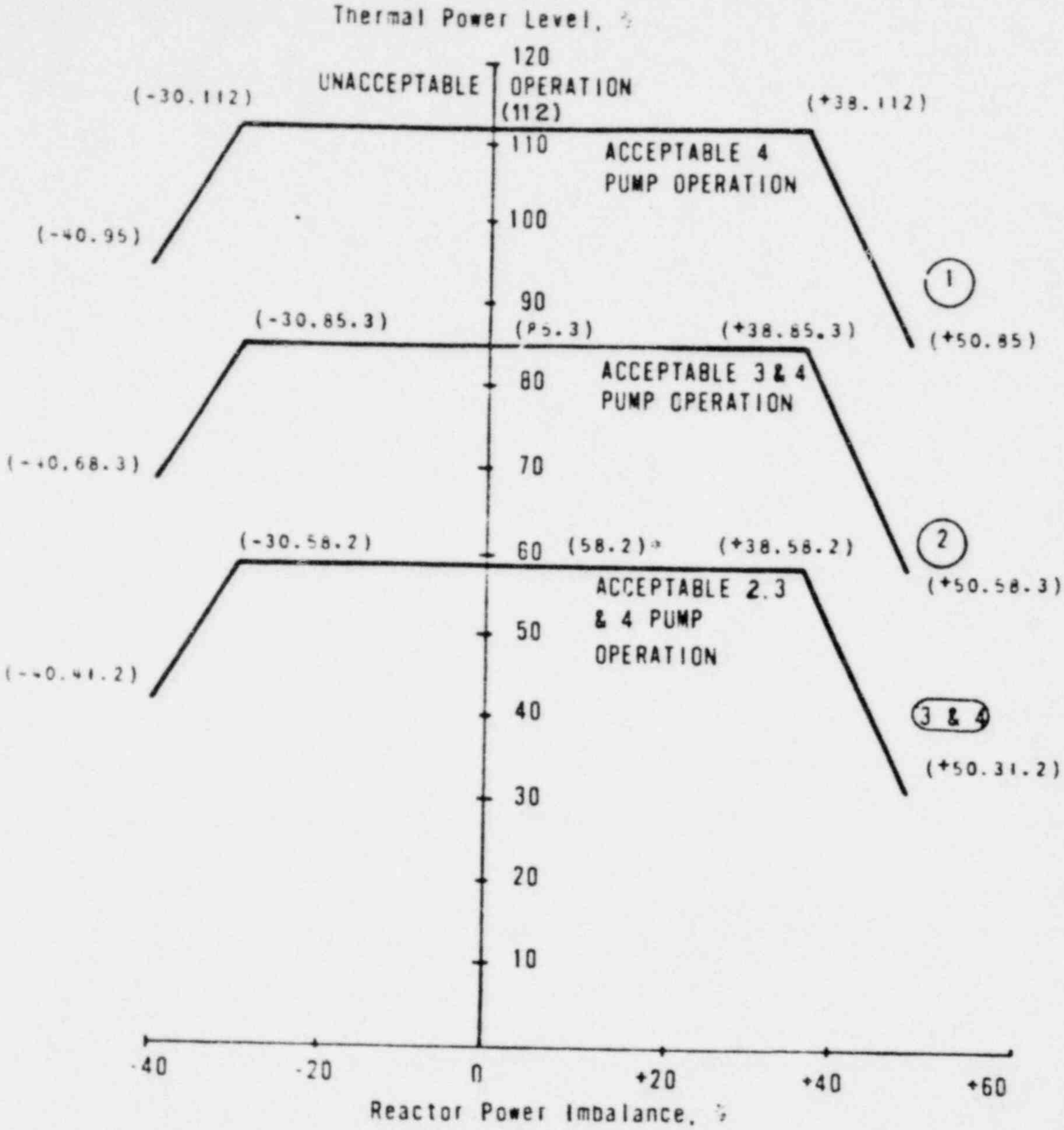


CURVE	REACTOR COOLANT FLOW (lb/hr)
1	141.3×10^6 DNBR Limit
2	105.6×10^6 DNBR Limit
3	69.3×10^6 Quality Limit

UNIT 1, CYCLE 3

CORE PROTECTION SAFETY LIMITS

Figure 8-1

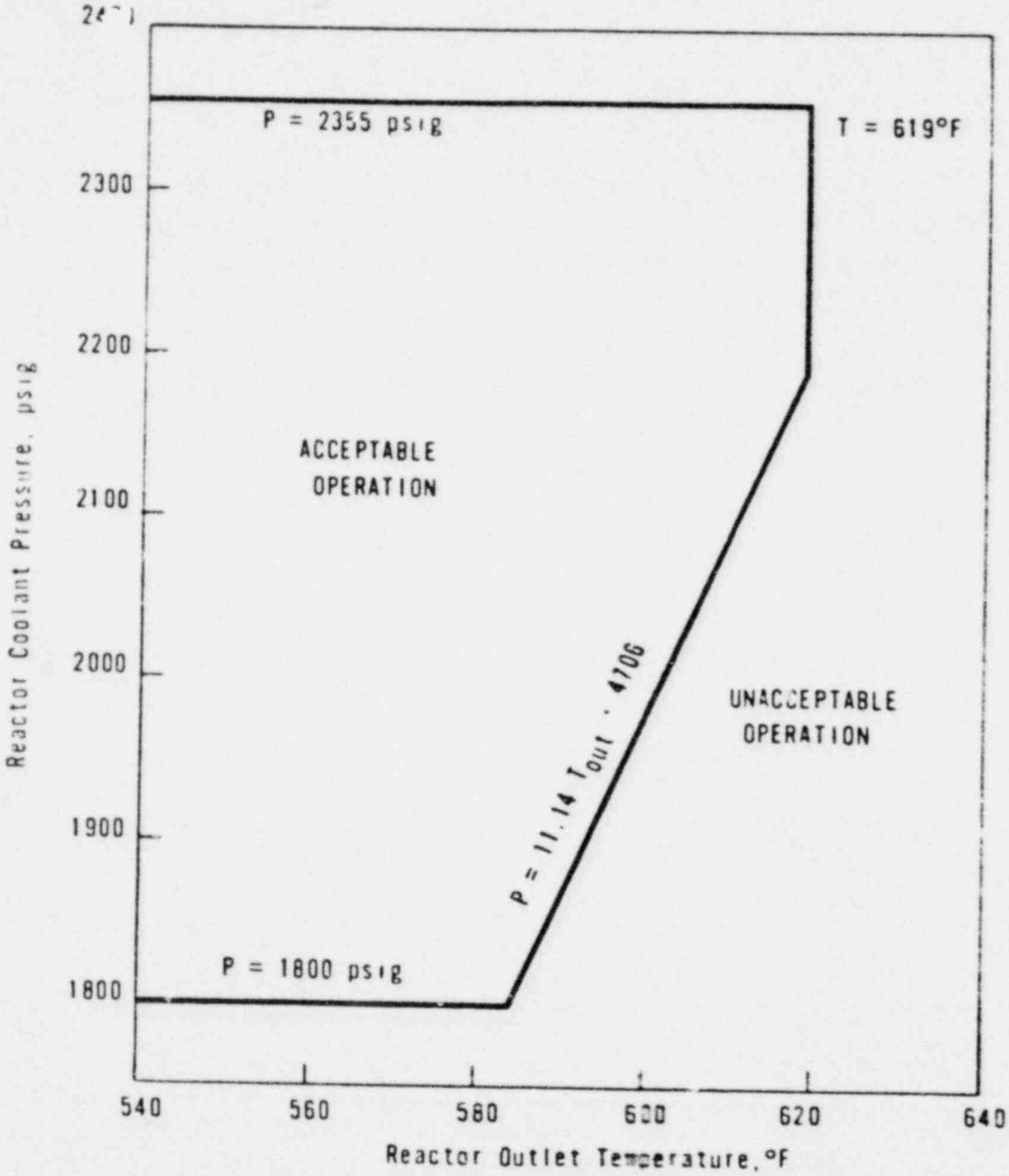


CURVE	REACTOR COOLANT FLOW (lb hr)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6
4	64.7×10^6

THE FLUX FLOW SETPOINT FOR 2 0 PUMP OPERATION MUST BE SET AT 0.949

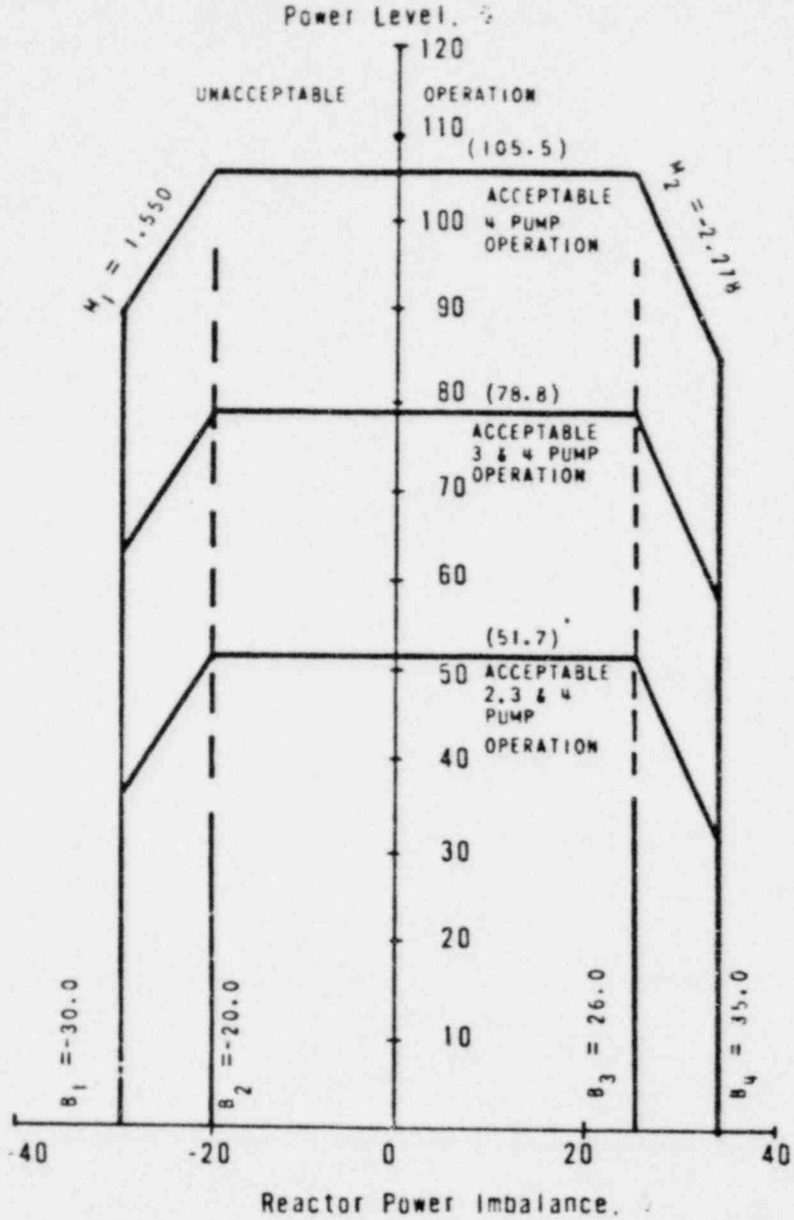
UNIT 1, CYCLE 3
CORE PROTECTION SAFETY LIMITS

Figure 8-2



UNIT 1, CYCLE 3
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS

Figure 8-3



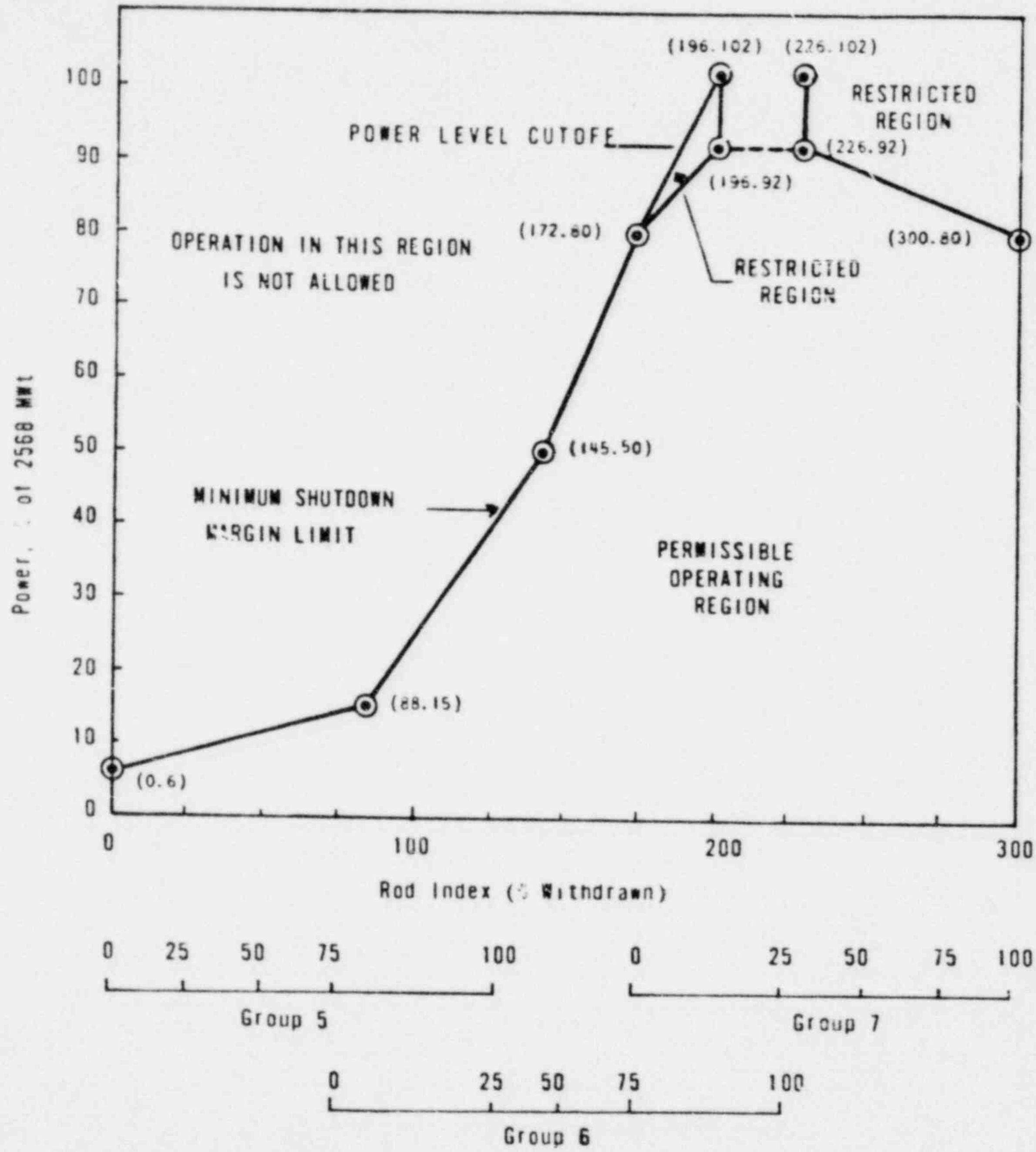
*THE FLUX FLOW SETPOINT FOR 2.0 PUMP OPERATION MUST BE SET AT 0.949

UNIT 1, CYCLE 3

PROTECTION SYSTEM MAXIMUM ALLOWABLE SET POINTS

Figure 8-4

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 0 TO 230 ± 5 EFPD

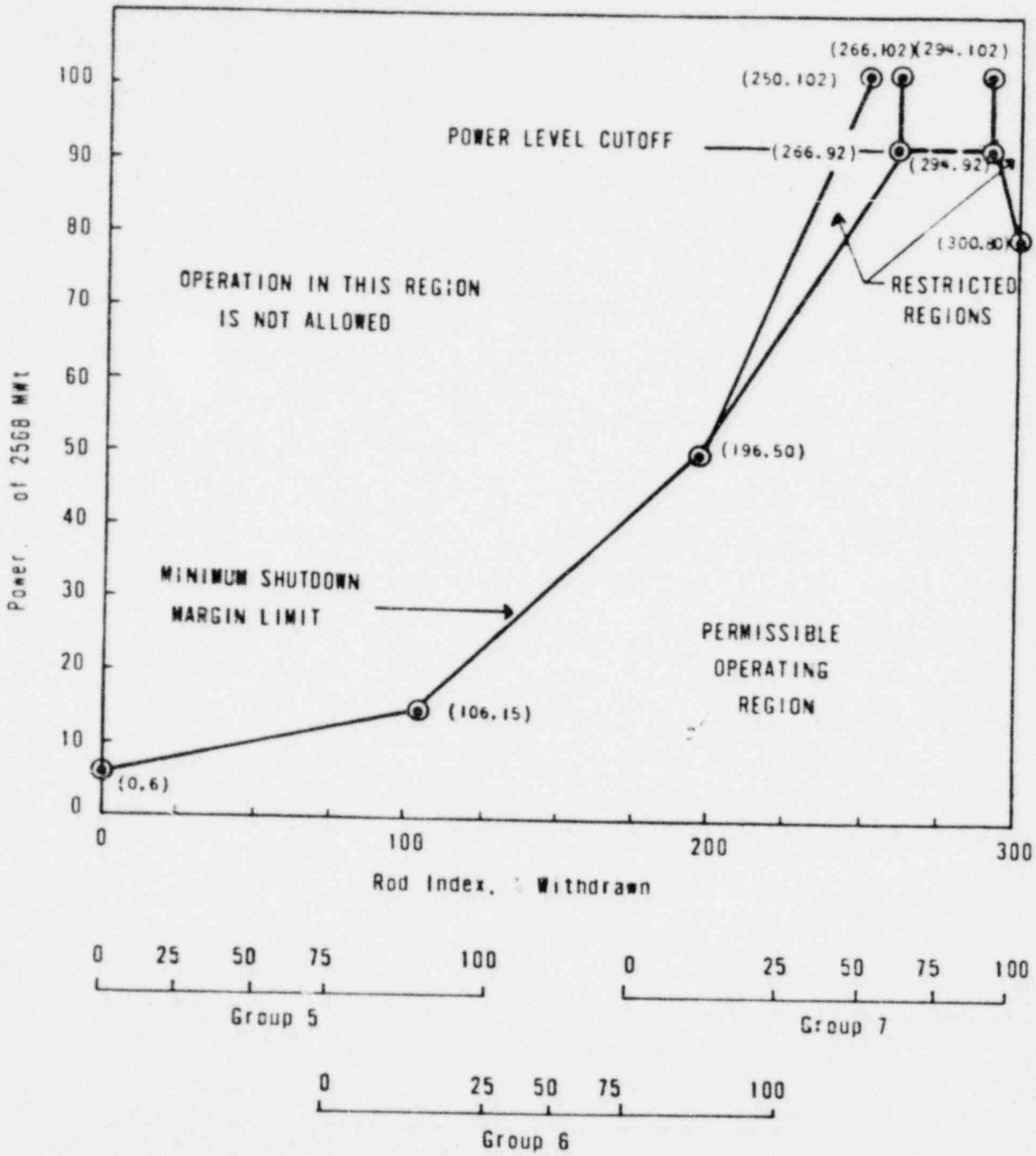


Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7.

UNIT 1, CYCLE 3
ROD POSITION LIMITS

Figure 8-5

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE
TO THE PERIOD AFTER 230 ± 5 EFPD



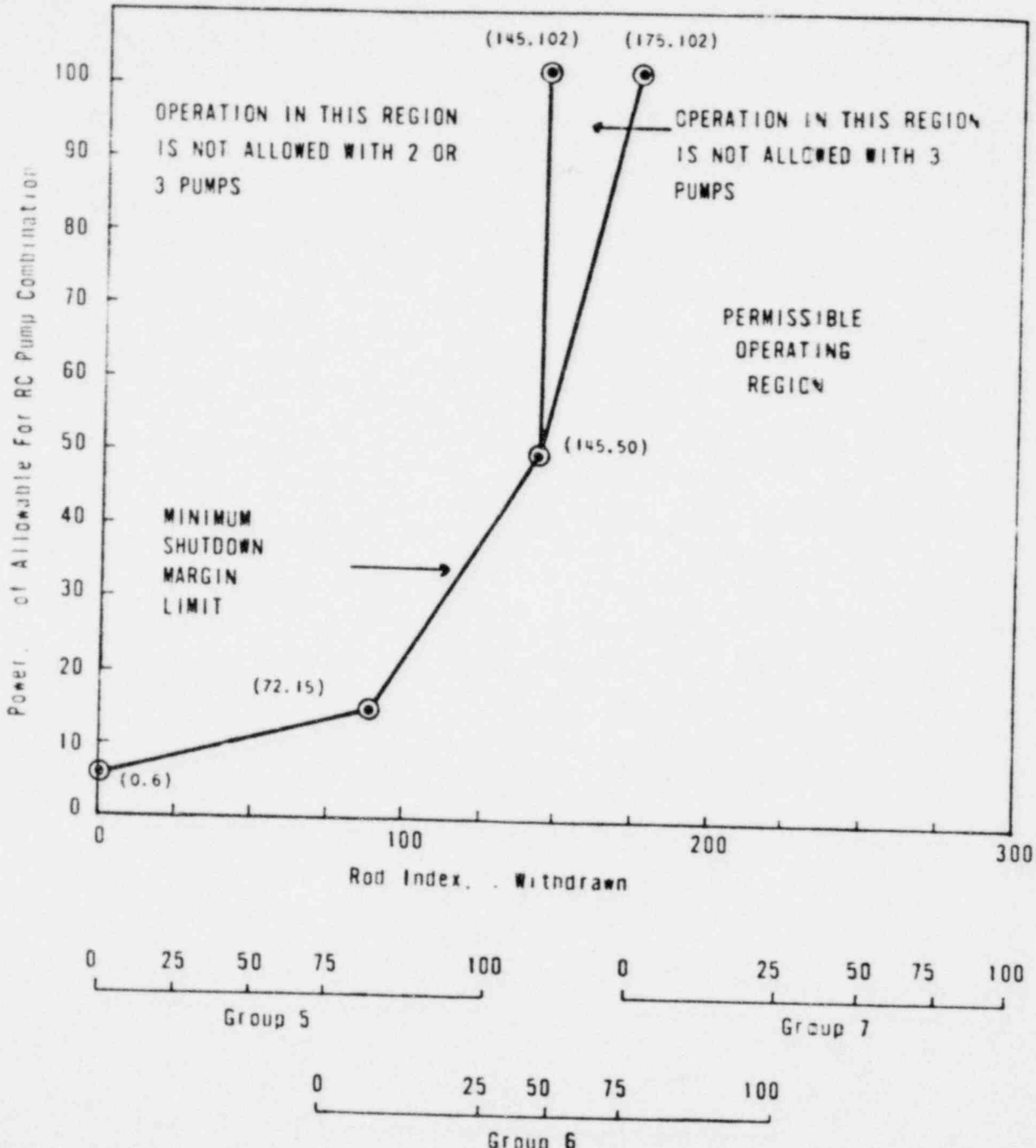
Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

UNIT 1, CYCLE 3

ROD POSITION LIMIT

Figure 8-6

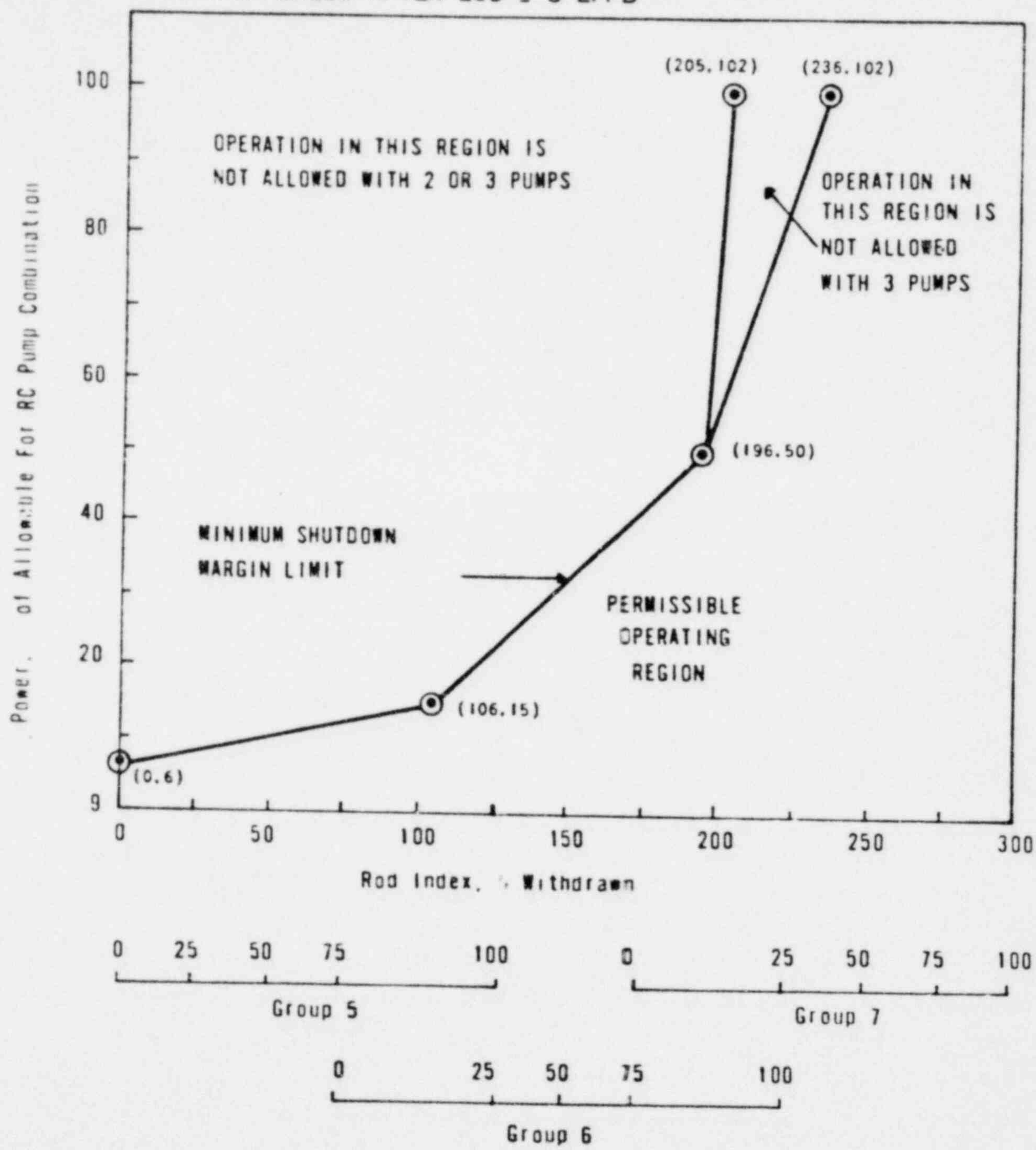
ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 0 TO 230 ± 5 EFPD



Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

UNIT 1, CYCLE 3
ROD POSITION LIMITS
Figure 8-7

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE TO THE PERIOD AFTER 230 ± 5 EFPD

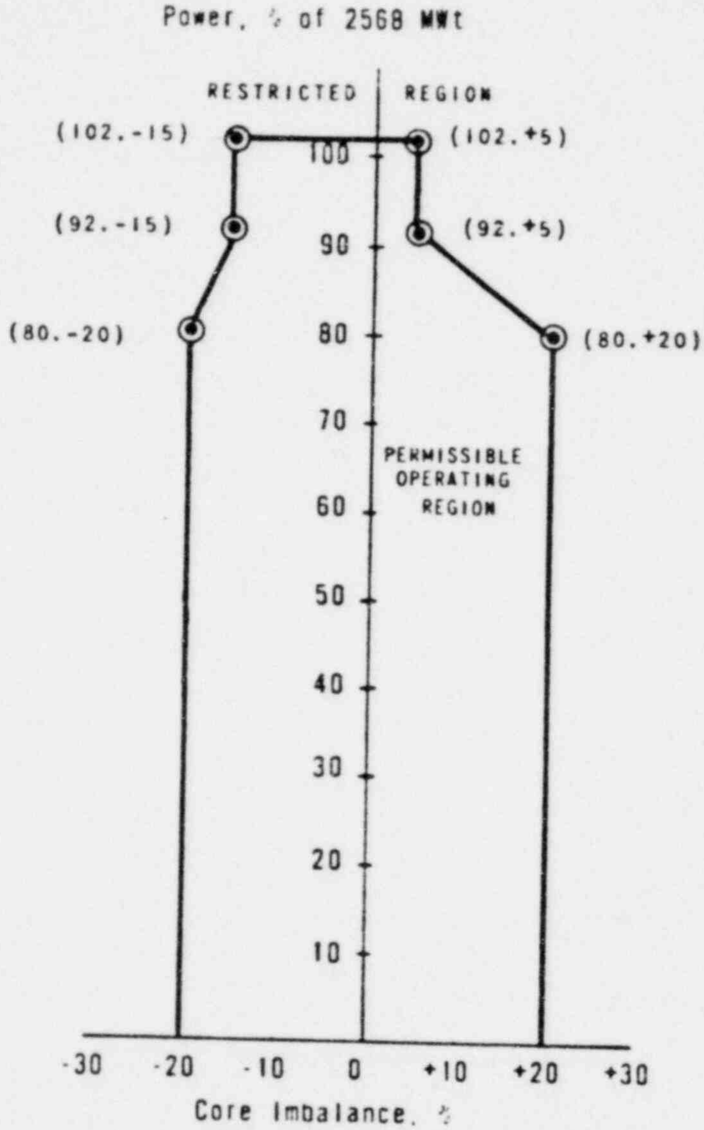


Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7.

UNIT 1, CYCLE 3

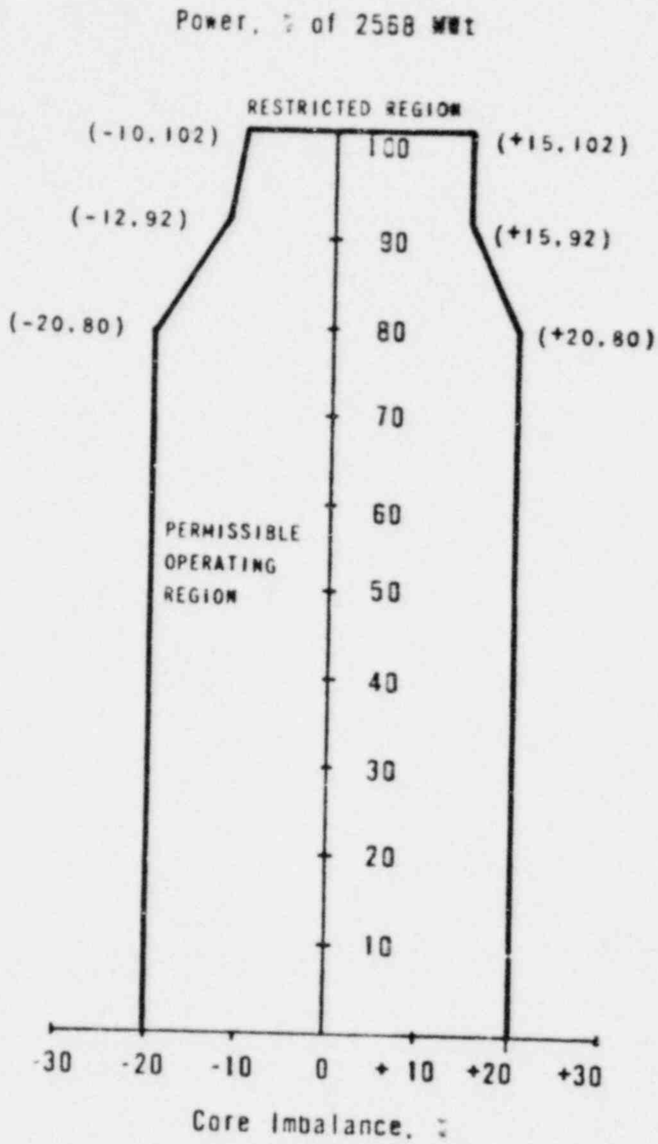
ROD POSITION LIMITS

Figure 8-8



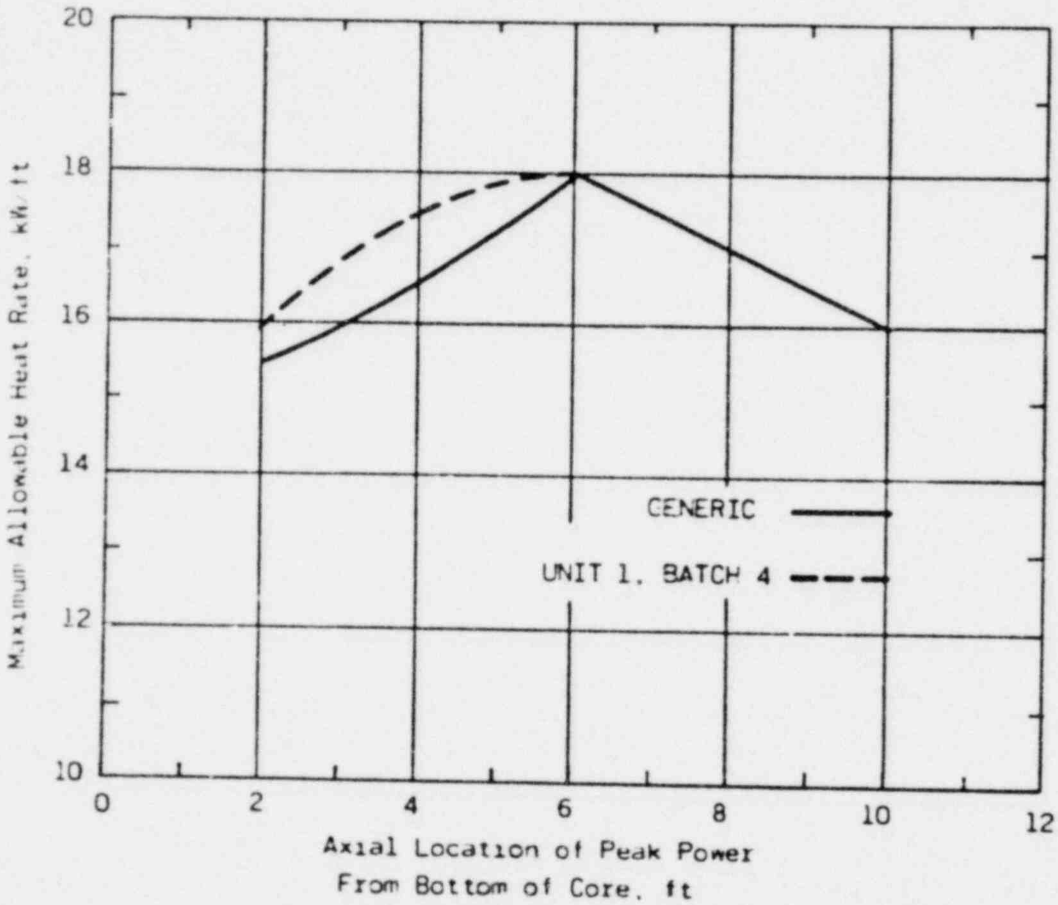
UNIT 1, CYCLE 3
OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 230 ± 5 EFPD

Figure 8-9



UNIT 1, CYCLE 3
OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION AFTER
230 ± 5 EFPD

Figure 8-10



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE

Figure 8-11

9. STARTUP PROGRAM

The planned startup testing associated with core performance are provided below. These tests verify that core performance is within the assumptions of the safety analysis and provide the necessary data for continued safe plant operation.

Pre-Critical Tests

1. Control Rod Drive Trip Time Testing

Zero Power Tests

1. Critical Boron Concentration
2. Temperature Reactivity Coefficient
3. Control Rod Group Worth
4. Ejected Rod Worth

Power Tests

1. Core Power Distribution Verification at Approximately 40, 75, and 100% FP Normal Control Rod Group Configuration
2. Core Power Distribution Verification at Approximately 40% FP With Worst Case Dropped Rod Fully Inserted
3. Incore/Out-of-Core Detector Imbalance Correlation Verification at Approximately 75% FP
4. Power Doppler Reactivity Coefficient at Approximately 100% FP
5. Temperature Reactivity Coefficient at Approximately 100% FP

10. REFERENCES

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