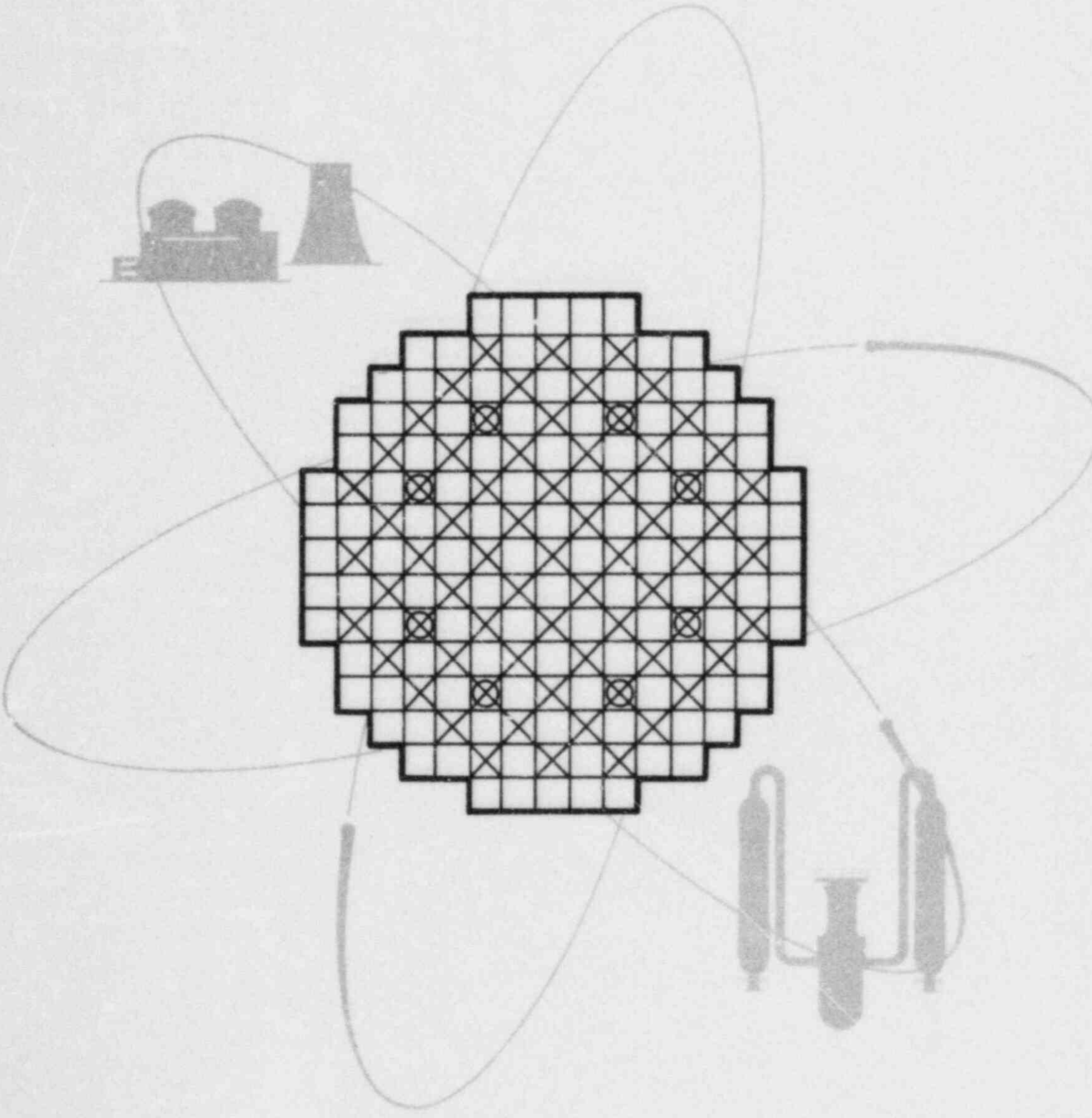
BAW-1841

August 1984

OCONEE UNIT 1, CYCLE 9

-- Reload Report --



Babcock & Wilcox

a McDermott company

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the ninth cycle of Oconee Nuclear Station, Unit 1, at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975.

To support cycle 9 operation of Oconee 1, this report employs analytical techniques and design bases established in reports that have been submitted to and accepted by the USNRC and its predecessor (see references).

A brief summary of cycle 8 and 9 reactor parameters related to power capability is included in section 5 of this report. All of the accidents analyzed in the Final Safety Analysis Report (FSAR)[1] have been reviewed for cycle 9 operation. In those cases where cycle 9 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

Four of the batch 10 assemblies are gadolinia lead test assemblies (LTAs). These assemblies are part of a joint Duke Power/Babcock & Wilcox (B&W)/Department of Energy (DOE) program to develop and demonstrate an advanced fuel assembly design incorporating UO_2-Gd_2O_3 for extended burnup in pressurized water reactors (PWRs). Reference 2 describes the LTAs. Four Mark BZ demonstration fuel assemblies containing Zircaloy-4 intermediate spacer grids will be reinserted for a third cycle of irradiation. The Mark BZ demonstration assemblies are described in reference 3. The sixty-four batch 11 fuel assemblies are the Mark BZ type. The analyses supporting full batch implementation of Mark BZ fuel are presented in reference 4. Gray axial power shaping rods (APSRs) have also been included in the cycle 9 design. The gadolinia LTAs, the Mark BZ assemblies, and the gray APSRs will not adversely affect cycle 9 operation.

The Technical Specifications have been reviewed, and the modifications required for cycle 9 operation are justified in this report.

Based on the analyses performed, which account for the postulated effects of fuel densification and the final acceptance criteria for emergency core cooling systems (ECCS), it has been concluded that Oconee Unit 1 can be operated safely for cycle 9 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee 1, cycle 9 is the currently operating cycle 8. The cycle 9 design length of 410 effective full power days (EFPD) is based on a planned cycle 8 length of 410 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in cycle 9.

3. GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design basis are described in detail in section 3 of the FSAR for Oconee Nuclear Station, Unit 1.[1] The cycle 9 core contains 177-fuel assemblies, each of which is a 15x15 array of 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The standard Mark B fuel assemblies in all batches have an average fuel loading of 463.6 kg of uranium. The undensified active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Tables 4-1 and 4-2.

Figure 3-1 is the core loading diagram for Oconee 1, cycle 9. Nineteen of the batch 9 assemblies will be discharged at the end of cycle 8, along with 44 batch 8C assemblies and the batch 10A assembly. The remaining 49 batch 9 assemblies (designated 9B) and the fresh batch 11 Mark BZ assemblies[4] with initial enrichments of 3.28 and 3.314 wt % ^{235}U, respectively, will be loaded into the central portion of the core. The four batch 10B gadolinia LTAs,[2] with an initial enrichment of 4.00 wt % ^{235}U, are in locations symmetrical to H13. The batch 10C fuel, with an initial enrichment of 3.41 wt % ^{235}U, will mainly occupy the core periphery. Figure 3-2 is an eighth core map showing the assembly burnup and enrichment distribution at the beginning of cycle (BOC) 9.

Reactivity is controlled by 61 full-length Ag-In-Cd control rods, 60 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length control rods, eight Inconel gray APSRs are provided for additional control of the axial power distribution. Since gray APSRs are being utilized, there are eight control rods in group 7 and twelve in group 5 to reduce the negative offset response to the group 7 rod movement. The cycle 9 locations of the 69 control rods and the group designations are

indicated in Figure 3-3. The core locations with the exception of group 5 and group 7 are identical to those of the reference cycle. The cycle 9 locations and concentrations of the BPRAs are shown in Figure 3-4.

The system pressure is 2200 psia and the core average densified nominal heat rate is 5.80 kW/ft at the rated power of 2568 MWt for the standard Mark B fuel assemblies.

Figure 3-1. Core Loading Diagram for Oconee 1, Cycle 9

FUEL TRANSFER CANAL →

X (top) — Y (right) — Z (bottom) — W (left, at row H)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						10C LØ5	10C KØ4	11 F	10C K12	10C L11					
B				10C LØ3	10C KØ2	10C NØ3	11 F	10C MØ8	11 F	10C N13	10C K14	10C L13			
C			10C N11	11 F	10C KØ6	11 F	9B P1Ø	10B CØ8 *	9B PØ6	11 F	10C K1Ø	11 F	10C MØ4		
D		10C C1Ø	11 F	9B BØ4	11 F	9B AØ7	11 F	9B RØ8 **	11 F	9B AØ9	11 F	9B D14	11 F	10C CØ6	
E		10C BØ9	10C FØ9	11 F	9B AØ6	11 F	9B P11	11 F	9B PØ5	11 F	9B F15	11 F	10C FØ7	10C BØ7	
F	10C E1Ø	10C C12	11 F	9B GØ1	11 F	9B O13	11 F	9B RØ6	11 F	9B OØ3	11 F	9B G15	11 F	10C CØ4	10C EØ6
G	10C DØ9	11 F	9B L14	11 F	9B M14	11 F	9B M14	10C NØ5	9B PØ4	11 F	9B MØ2	11 F	9B LØ2	11 F	10C DØ7
H	11 F	10C H11	10B HØ3 *	9B H15 **	11 F	9B L15	10C M12	9B H14	10C EØ4	9B FØ1	11 F	9B HØ1 **	10B H13 *	10C HØ5	11 F
K	10C NØ9	11 F	9B F14	11 F	9B E14	11 F	9B B12	10C O11	9B DØ2	11 F	9B EØ2	11 F	9B FØ2	11 F	10C NØ7
L	10C M1Ø	10C O12	11 F	9B KØ1	11 F	9B C13	11 F	9B A1Ø	11 F	9B CØ3	11 F	9B K15	11 F	10C OØ4	10C MØ6
M		10C PØ9	10C LØ9	11 F	9B LØ1	11 F	9B B11	11 F	9B BØ5	11 F	9B R1Ø	11 F	10C LØ7	10C PØ7	
N		10C O1Ø	11 F	9B NØ2	11 F	9B RØ7	11 F	9B AØ8 **	11 F	9B RØ9	11 F	9B P12	11 F	10C OØ6	
O			10C E12	11 F	10C GØ6	11 F	9B B1Ø	10B OØ8 *	9B BØ6	11 F	10C G1Ø	11 F	10C DØ5		
P				10C FØ3	10C GØ2	10C DØ3	11 F	10C EØ8	11 F	10C D13	10C G14	10C F13			
R						10C FØ5	10C GØ4	11 F	10C G12	10C F11					

Batch I.D.
Previous Cycle Location

* Gadolinia LTA
** Mark BZ LTA

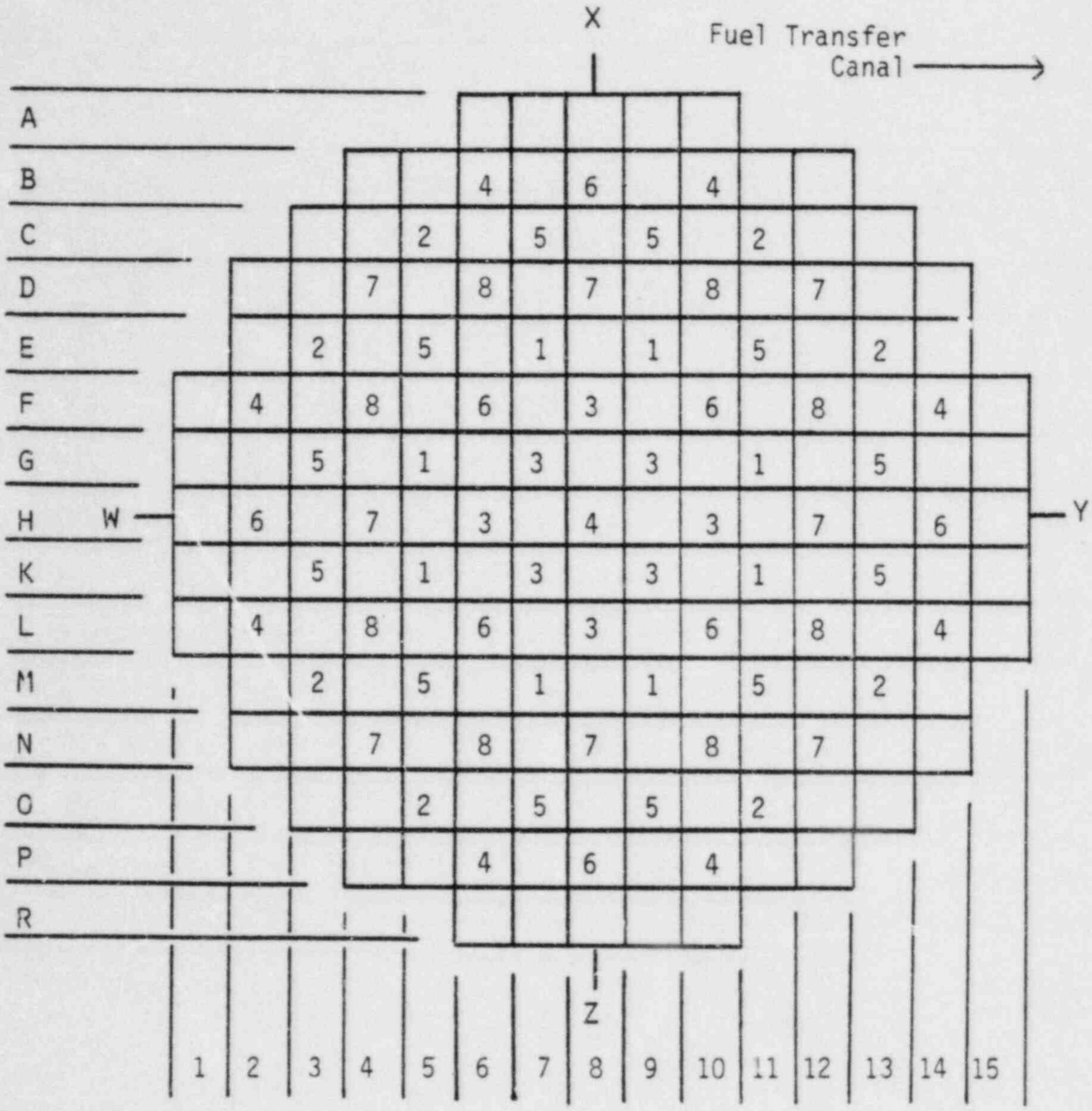
Figure 3-2. Enrichment and Burnup Distributions for Oconee 1, Cycle 9

	8	9	10	11	12	13	14	15
H	3.280 29,099	3.410 16,462	3.280 21,669	3.314 0	3.280 Mark BZ* 23,414	4.000 LTA* 15,840	3.410 17,365	3.314 0
K		3.280 21,633	3.314 0	3.280 21,594	3.314 0	3.280 22,811	3.314 0	3.410 16,423
L			3.280 22,649	3.314 0	3.280 20,926	3.314 0	3.410 13,150	3.410 17,014
M				3.280 21,666	3.314 0	3.410 17,292	3.410 14,266	
N					3.280 21,634	3.314 0	3.410 16,015	
O						3.410 16,492		
P								
R								

*Demonstration assemblies.

X.XXX	Initial Enrichment
XXXXX	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Oconee 1, Cycle 9



X Group Number

Group	No. 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	APSRs
Total	69	

Figure 3-4. BPRA Concentration and Distribution for Oconee 1, Cycle 9

	8	9	10	11	12	13	14	15
H				1.40				
K			1.40		1.40		0.20	
L		1.40		1.40		0.50		
M	1.40		1.40		1.40			
N		1.40		1.40		0.00		
O			0.50		0.00			
P		0.20						
R								

X.XX	BPRA Concentration, wt % B_4C in Al_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 1, cycle 9 are listed in Table 4-1. All the fuel assemblies are mechanically interchangeable. Four twice-burned Mark BZ demonstration fuel assemblies are included in batch 9B. The Mark BZ uses Zircaloy as the material for the six intermediate spacer grids. The Mark BZ demonstration assembly is described in reference 3, which concludes that reactor safety and performance are not adversely affected by the presence of the four demonstration assemblies. The four fuel assemblies in batch 10B are gadolinia LTAs. The mechanical design of the LTAs is described in reference 2. The 64 fuel assemblies in batch 11 are also Mark BZ assemblies. The mechanical design of the Mark BZ fuel is described in reference 4. The balance of the fuel assemblies in the cycle 9 core are the standard Mark B type.

Retainer assemblies will be used on the two fuel assemblies that contain regenerative neutron source assemblies and on the 60 batch 11 assemblies that contain BPRAs. The 62 retainers will be exposed for a fourth cycle of irradiation during cycle 9. This additional cycle of irradiation is justified based on an examination of retainers which have undergone three cycles of irradiation. The results of the examination meet criteria developed earlier in terms of wear and holddown force. These criteria ensure that the retainers will perform in a safe and adequate manner in the areas of holddown force, stress, and fatigue during a fourth cycle of in-reactor use. These criteria were developed from analyses similar to those performed in the original justification of the design and use of the retainer assemblies in references 5 and 6. The justification for the fourth cycle of irradiation is given in reference 7.

4.2. Fuel Rod and Gray APSR Designs

Mechanical evaluations of the fuel rods and gray APSRs are discussed below.

4.2.1. Cladding Collapse

The fuel assemblies of batch 9B are more limiting than those of other batches because of their longer previous incore exposure time. The power history and fuel design parameters for the most limiting batch 9B fuel assembly were compared with those used in the generic Mark-B creep collapse analysis and were found to be enveloped. The generic analysis was based on the methods and procedures described in reference 8 and is applicable to the batch 9B fuel design. The generic analysis predicts a collapse time of more than 35,000 effective full power hours (EFPH), which exceed the maximum projected residence time of 29,359 EFPH (Table 4-1).

A detailed creep analysis was performed on the gadolinia-bearing fuel rods in the LTAs. The collapse time for these rods was greater than the maximum projected residence time.

The gray APSRs that are to be used in cycle 9 were designed to improve creep life. Cladding thickness and rod ovality control, which are the primary factors controlling the creep life of a stainless steel material, have been improved to extend the creep life of the gray APSR. The minimum design cladding thickness of the Mark B APSR is 18 mils, while that of the gray APSR is 24 mils. Additionally, the gap width between the end plug and Inconel absorber material was reduced. Finally, the ovality in the gap area will also be controlled to tighter tolerances. The gray APSR is shown in Figure 4-1.

4.2.2. Cladding Stress

The stress parameters for the Oconee 1 standard fuel rods and the gadolinia-bearing fuel rods are enveloped by a conservative fuel rod stress analysis. The following four assumptions were used in this analysis:

1. A lower post-densification internal pressure.
2. A lower initial pellet density.
3. A higher system pressure.
4. A higher thermal gradient across the cladding.

For design evaluation, the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses (primary and secondary) must be less than the minimum specified un-irradiated yield strength. In all cases, the margin is in excess of 30%.

The gray APSR design was analyzed to demonstrate that it meets specified design requirements. The APSR was analyzed for cladding stress due to pressure, temperature, and ovality. It was found that the gray APSR nas sufficient cladding and weld stress margins.

4.2.3. Cladding Strain

The fuel design criteria specify that the cladding average circumferential strain is not to exceed 1% inelastic strain. The pellet design is established for a plastic cladding strain of less than 1% at the maximum design local pellet burnup and heat generation rate. These values are higher than the values the Oconee 1 UO_2 fuel is expected to see. A strain analysis of the gadolinia fuel showed that the calculated strains for these rods are also below design limits. Thus, fuel rod cladding strain will not affect cycle 9 fuel performance.

The gray APSR was analyzed for cladding strain due to thermal and irradiation swelling. The results of this analysis showed that no cladding strain is induced due to thermal expansion or irradiation swelling of the Inconel absorber.

4.3. Thermal Design

All fuel rods in cycle 9 are thermally similar, except for the urania-gadolinia fuel in four LTAs. The analysis for reinserted batches 9B, 10B, and 10C and feed batch 11 fuel was performed with the TACO2[9] code using the methodology described in reference 10. Nominal undensified input parameters used in this methodology are presented in Table 4-2. Densification effects were accounted for in TACO2.

Centerline fuel melt (CFM) limits of 20.5 kW/ft for 95% theoretical density (TD), pure UO_2 fuel, and 17.6 kW/ft for UO_2-Gd_2O_3 fuel were predicted using the thermal code TACO2. The fuel internal pressure in the highest burnup rod is predicted to reach the nominal reactor coolant system pressure of

2200 psia after 45,000 MWd/mtU for both pure UO_2 and UO_2-Gd_2O_3. The maximum burnup of any UO_2 fuel rod in cycle 9 is less than 45,000 MWd/mtU; the highest burnup of any UO_2-Gd_2O_3 fuel rod is less than 25,000 MWd/mtU.

4.4. Material Design

The batch 11 fuel assemblies are not new in concept, nor do they utilize different component materials, except for the Zircaloy spacer grids and the Inconel 718 holddown springs. These materials have been used with success in similar reactor environments. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 11 fuel assemblies is acceptable.

4.5. Operating Experience

B&W operating experience with the Mark B 15x15 fuel assembly has verified the adequacy of its design. As of April 30, 1984, the following experience has been accumulated for eight B&W 177-fuel assembly (FA) plants using the Mark B fuel assembly:

Reactor	Current cycle	Max FA burnup,[a] MWd/mtU: Incore	Max FA burnup,[a] MWd/mtU: Discharged	Cumulative net electric output,[b] MWh
Oconee 1	8	34,499	50,598	48,808,138
Oconee 2	7	27,035	36,800	43,444,856
Oconee 3	8	35,123	35,463	45,200,486
Three Mile Island	5	25,200	32,400	23,840,053
Arkansas Nuclear One, Unit 1	6	31,450	36,540	38,872,852
Rancho Seco	6	30,500	38,268	33,923,457
Crystal River 3	5	23,170	29,900	27,083,428
Davis-Besse	4	28,520	32,790	19,237,628

[a]As of April 30, 1984.

[b]As of January 31, 1984.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch 9B	Batch 10B/10C	Batch 11
FA type	Mark B/ Mark BZ	Mark GdB/ Mark B	Mark BZ
No. of FAs	45/4	4/60	64
Fuel rod OD, in.	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4
Undensified active fuel length (nom.), in.	141.8	143.5/141.8	141.8
Fuel pellet initial density (nom.), % TD	95	95	95
Fuel pellet OD (mean specification), in.	0.3686	0.3686	0.3686
Initial fuel enrichment, wt % ^{235}U	3.28	4.0/3.41	3.314
BOC burnup (avg), MWd/mtU	22,090	15841/15911	0
Cladding collapse time, EFPH	>35,000	>35,000	>35,000
Estimated residence time (max.), EFPH	29,359	19,680	9,840

Table 4-2. Fuel Thermal Analysis Parameters -- Oconee 1, Cycle 9

	Batch			
	9B[a]	10B[b]	10C	11[c]
No. of assemblies	49	4	60	64
Nominal pellet density, % TD	95	95	95	95
Pellet diameter, in.	0.3686	0.3686	0.3686	0.3686
Stack height, in.	141.8	143.5	141.8	141.8
Nominal LHR[d] @ 2568 MWt, kW/ft	5.74	5.68	5.74	5.74
LHR to C_L fuel melt, kW/ft	20.5	17.6[e]	20.5	20.5

(a) Includes four Mark BZ demonstration assemblies.

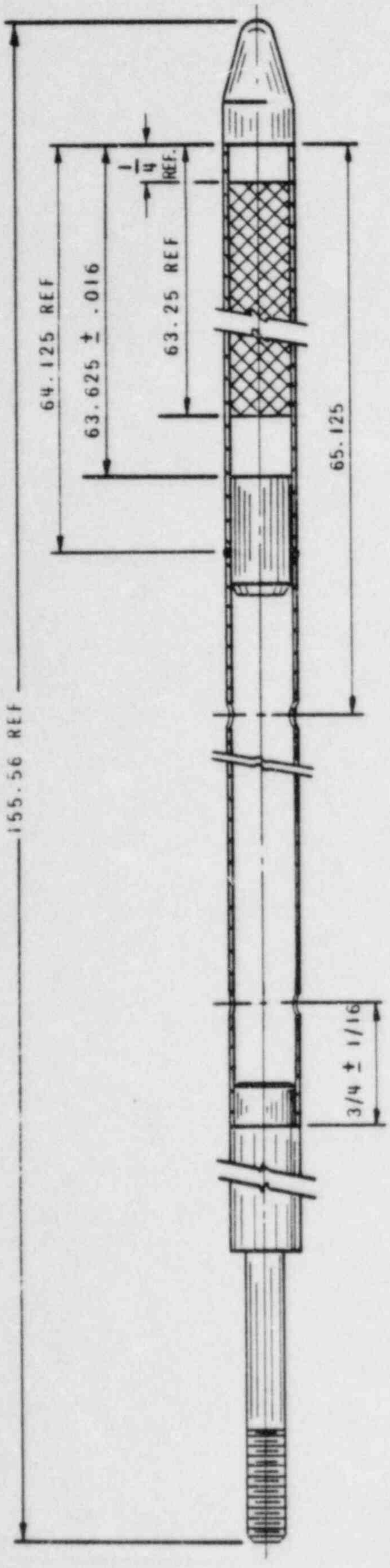
(b) Gadolinia LTAs.

(c) Sixty-four Mark BZ assemblies, including 60 BPRAs.

(d) LHR denotes linear heat rate.

(e) Reduced for gadolinia fuel.

Figure 4-1. Gray Axial Power Shaping Rod



5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of design cycle 9 with those of the reference cycle 8. The values for both cycles were generated using PDQ07[11-13]. The average cycle 9 burnup will be the same as that of the cycle 8 design. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 9 at full power with equilibrium xenon and normal rod positions.

Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The critical boron concentrations for cycle 9 are higher because of the gray APSRs and previous cycle reactivity carryover. The control rod worths differ between cycles due to the gray APSRs, changes in rod groupings for transient banks 5 and 7, changes in radial flux, and burnup distributions. This also accounts for the smaller ejected and stuck rod worths in cycle 9 compared to cycle 8 values. 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. These rod worths meet all safety criteria. The adequacy of the shutdown margin with cycle 9 stuck rod worths is demonstrated in Table 5-2. The following assumptions were applied for the shutdown calculations:

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

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The reference fuel cycle shutdown margin is presented in the reload report for Oconee 1, cycle 8[14]. The cycle 9 power deficits, differential boron worths, and effective delayed

neutron fractions differ from those for cycle 8 because of the higher critical boron concentrations.

5.2. Analytical Input

The constants used to compute core power distributions from incore detector measurements were obtained in the same manner for cycle 9 as for the reference cycle 8. The monitoring of power distributions in the LTAs is discussed in reference 3.

5.3. Changes in Core Design

Core design changes for cycle 9 are the use of gray APSRs and the introduction of 64 Mark BZ assemblies in addition to the four Mark BZ LTAs loaded in cycle 7. Gray APSRs, which are longer and use a weaker Inconel absorber, replace the silver-indium-cadmium APSRs used in all previous cycles. Calculations with the standard three-dimensional model verified that these APSRs provide adequate axial power distribution control.

Table 5-1. Oconee 1 Cycles 8 and 9 Physics Parameters[a]

	Cycle 8[b]	Cycle 9[c]
Cycle length, EFPD	410	410
Cycle burnup, MWd/mtU	12,858	12,858
Avg. core burnup, EOC, MWd/mtU	24,183	24,724
Initial core loading, mtU	82.1	82.1
Critical boron, BOC, (No Xe), ppm		
HZP[d], group 8 inserted	1602	1682
HFP[d], group 8 inserted	1365	1454
Critical boron, EOC, (eq Xe), ppm		
HZP, group 8 inserted	401	457
HFP, group 8 inserted	60	121
Control rod worths, HFP, BOC, % $\Delta k/k$		
Group 6	0.98	1.09
Group 7	1.47	0.93
Group 8	0.42	0.10
Control rod worths, HFP, EOC, % $\Delta k/k$		
Group 7	1.54	1.00
Group 8	0.49	0.10
Max ejected rod worth, HZP, % $\Delta k/k$[e]		
BOC (N12)	0.59	0.35
EOC (L10)	0.48	0.36
Max stuck rod worth, HZP, % $\Delta k/k$		
BOC (H14)	1.68	1.31
EOC (N12)	1.72	1.35
Power deficit, HZP to HFP, % $\Delta k/k$		
BOC	1.62	1.51
EOC	2.36	2.29
Doppler coeff., HFP, 10^{-5} ($\Delta k/k/F$)		
BOC, 100% power, no Xe	-1.54	-1.53
EOC, 100% power, eq Xe	-1.78	-1.77
Moderator coeff., HFP, 10^{-4} ($\Delta k/k/F$)		
BOC, (0 Xe, crit ppm, gp 8 ins)	-0.67	-0.47
EOC, (eq Xe, 17 ppm, gp 8 ins)	-2.85	-2.75
Boron worth, HFP, ppm/% $\Delta k/k$		
BOC (1425 ppm)	129	130
EOC (17 ppm)	110	111

Table 5-1. (Cont'd)

	Cycle 8[b]	Cycle 9[c]
Xenon worth, HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.54	2.53
EOC (equilibrium)	2.67	2.66
Effective delayed neutron fraction, HFP		
BOC	0.00625	0.00621
EOC	0.00526	0.00526

[a]Cycle 9 data are for the conditions stated in this report. The cycle 8 core conditions are identified in reference 14.

[b]Cycle 8 data are based on a cycle 7 length of 420 EFPD.

[c]Cycle 9 data are based on a cycle 8 length of 410 EFPD.

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

[e]Ejected rod worth for groups 5 through 7 inserted.

Table 5-2. Shutdown Margin Calculations for Oconee 1, Cycle 9

	BOC, % $\Delta k/k$	EOC, % $\Delta k/k$
Available Rod Worth		
Total rod worth, HZP[a]	8.14	8.82
Worth reduction due to burnup of poison material	-0.42	-0.42
Maximum stuck rod, HZP[a]	-1.31	-1.35
Net worth	6.41	7.05
Less 10% uncertainty	-0.64	-0.71
Total available worth	5.77	6.34
Required Rod Worth		
Power deficit, HFP to HZP[a]	1.51	2.29
Max allowable inserted rod worth	0.30	0.50
Flux redistribution	0.81	1.20
Total required worth	2.62	3.99
Shutdown margin (total available worth minus total required worth)	3.15	2.35

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

[a] HZP denotes hot zero power and HFP denotes hot full power.

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

	8	9	10	11	12	13	14	15
H	0.848	1.059	1.042	1.203	1.061 Mark BZ*	1.193 LTA*	1.130	0.850
K		0.996	1.169	1.077	1.210	1.073	1.193	0.639
L			1.053	1.198	8 1.096	1.277	0.969	0.432
M				1.097	1.213	1.098	0.702	
N					1.040	1.039	0.465	
O						0.572		
P								
R								

*Demonstration assemblies.

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

6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 9 operation utilized the methods described in references 1, 5, and 15. Section 5 of reference 4 demonstrates that a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions and that the current pressure-temperature trip envelope is conservative for a full Mark BZ core.

The cycle 9 transition core includes 64 fresh Mark BZ batch 11 fuel assemblies, 60 of which will contain BPRAs. Batch 10B LTAs and the 4 demonstration assemblies in batch 9B also incorporate Mark BZ spacer grids. The effect of the higher pressure drop caused by the Mark BZ grids and by the BPRA retainers[7] in a predominantly Mark B core is a slightly lower flow in the Mark BZ assemblies. The DNB margin for the Mark BZ assemblies is reduced as a result. To preserve the DNB margin, the radial-local design peaking is reduced to 1.67 for the Mark BZ assemblies. This peaking reduction ensures a comparable DNB margin for the limiting transient and for steady-state operations for the cycle 9 transition core and the Mark B reference core. The 1.71 radial-local peaking still applies to the Mark B assemblies. The 1.67 value is applicable to cycle 9 only.

The maximum expected peaking during cycle 9 is 1.416. No fuel rod bow penalty has been incorporated into the core departure from nucleate boiling ratio (DNBR) limits, as justified by reference 16.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 8	Cycle 9
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Vessel inlet coolant temp. at 100% power, F	555.6	555.6
Vessel outlet coolant temp. at 100% power, F	602.4	602.4
Ref. design axial flux shape	1.5 cos	1.5 cos
Ref. design radial-local peaking factor	1.71	1.67[a]
Active fuel length	140.3[b]	140.3[b]
Average heat flux at 100% power, 10^3 Btu/h-ft^2	176[c]	176[c]
Maximum local heat flux at 100% power, 10^3 Btu/h-ft^2	451[d]	441[e]
Critical heat flux (CHF) correlation	B&W-2	BWC
CHF correlation limit	1.30	1.18
Hot channel factors -- Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.97
Minimum DNBR (112% power)	2.05[f]	1.74

(a) Applicable to cycle 9 only.

(b) Used in DNBR calculations and to calculate the next two items in this table.

(c) Based on an expanded rod OD of 0.43075 in.

(d) Based on a length of 140.3 in., rod OD of 0.43075 in., axial peak of 1.5, and radial-local peak of 1.71.

(e) Based on a length of 140.3 in., rod OD of 0.43075 in., axial peak of 1.5, and radial-local peak of 1.67.

(f) Based on 52 open assemblies.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 9 parameters to determine the effects of the cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in BAW-1388.[15] Since batch 11 reload fuel assemblies contain fuel rods with a higher theoretical density than those considered in the reference 15 report, the conclusions in the reference are still valid.

7.2. Accident Evaluation

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

Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Fuel thermal analysis values for each batch in cycle 9 are compared in Table 4-2. The cycle 9 thermal-hydraulic maximum design conditions are compared to the previous cycle 8 values in Table 6-1. These parameters are common to all of the accidents considered in this report. The key kinetics parameters from the FSAR and cycle 9 are compared in Table 7-1.

A generic LOCA analysis for the B&W 177-FA, lowered-loop nuclear steam supply (NSS) system has been performed using the final acceptance criteria ECCS evaluation model. This study is reported in BAW-10103, Rev. 3.[17] The analysis in BAW-10103 is generic since the limiting values of key parameters for all plants in this category were used. Furthermore, the

combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used in the BAW-10103 loss-of-coolant accident (LOCA) limits analysis is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results. Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 1, cycle 9 as a function of burnup. The LOCA kW/ft limits have been reduced for low burnups to ensure conservative limits based upon an interim bounding analytical assessment of NUREG 0630 on LOCA and operating kW/ft limits[21].

The Oconee 1, cycle 9 core contains four Mark BZ demonstration assemblies, four gadolinia LTAs, and 64 Mark BZ assemblies. As a result of material and geometrical differences, these assemblies have LOCA kW/ft limits that are lower in some cases than the standard Mark B limits. The four gadolinia LTAs are non-limiting. The Mark BZ LTAs were analyzed using LOCA kW/ft limits consistent with the 64 fresh Mark BZ assemblies.

It is concluded from the examination of cycle 9 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the ability of the Oconee 1 plant to operate safely during cycle 9. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 9 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 9 are bounded by the FSAR[1], the fuel densification report[15], and/or subsequent cycle analyses.

The radiological dose consequences of the accidents presented in chapter 15 of the FSAR were recalculated using the specific parameters applicable to cycle 9. The bases used in the dose calculations are identical to those in the FSAR except that updated dose conversion factors were used. The use of the updated dose conversion factors resulted in reduced whole body dose values.

Table 7-3 compares the revised FSAR dose values with those calculated specifically for cycle 9. As can be seen from the table, some cycle 9 doses vary slightly from the FSAR values. However, all cycle 9 doses are either bounded by the values presented in the FSAR or are a small fraction of the 10 CFR 100 limits, i.e. below 30 rem to the thyroid or 2.5 rem to the whole

body. Thus, the radiological impact of the accidents during cycle 9 are not significantly different than those described in chapter 15 of the FSAR.

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

Parameter	FSAR and densification report value	Predicted cycle 9 value
Doppler coeff, 10^{-5} $\Delta k/k/F$		
BOC	-1.17	-1.53
EOC	-1.33	-1.77
Moderator coeff, 10^{-4} $\Delta k/k/F$		
BOC	+0.5	-0.47
EOC	-3.0	-2.75
All-rod group worth at HZP, % $\Delta k/k$	10	8.14
Initial boron conc'n at HFP, ppm	1400	1454[a]
Boron reactivity worth at 70F, ppm/1% $\Delta k/k$	75	91
Max ejected rod worth at HFP, % $\Delta k/k$	0.65	0.22
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20

[a] The combined effect of boron concentration and boron worth is conservative for cycle 9.

Table 7-2. LOCA Limits for Oconee 1, Cycle 9

Elevation, ft	Linear heat rate, kW/ft		
	0-30 +10/-0 EFPD	30 +10/-0 to 250 ±10 EFPD	250 ±10 EFPD to EOC
2	13.5	15.0	15.5
4	16.1	16.6	16.6
6	17.5[a]	18.0	18.0
8	17.0	17.0	17.0
10	16.0	16.0	16.0

[a] For the Mark BZ assemblies the LOCA limit is 16.5 kW/ft at the 6-ft elevation for the 0-30 +10/-0 EFPD window only.

Table 7-3. Comparison of FSAR and Cycle 9 Accident Doses

	FSAR doses,[a] rem	Cycle 9 doses, rem
1. Fuel Handling Accident		
Thyroid dose at EAB, 2 h	0.50	0.50
Whole body dose at EAB, 2 h	0.028	0.010
2. Steam Line Break		
Thyroid dose at EAB, 2 h	0.20	0.20
Whole body dose at EAB, 2 h	0.002	0.001
3. Steam Generator Tube Failure		
Thyroid dose at EAB, 2 h	0.31	0.31
Whole body dose at EAB, 2 h	0.058	0.027
4. Waste Gas Tank Rupture		
Thyroid dose at EAB, 2 h	0.27	0.28
Whole body dose at EAB, 2 h	0.17	0.08
5. Control Rod Ejection Accident		
Thyroid dose at EAB, 2 h	1.44	1.40
Whole body dose at EAB, 2 h	0.004	0.002
Thyroid dose at LPZ, 30 days	1.57	1.55
Whole body dose at LPZ, 30 days	(b)	0.002
6. Loss-of-Coolant Accident		
Thyroid dose at EAB, 2 h	5.0	4.97
Whole body dose at EAB, 2 h	0.010	0.005
Thyroid dose at LPZ, 30 days	5.5	5.51
Whole body dose at LPZ, 30 days	0.010	0.007
7. Maximum Hypothetical Accident		
Thyroid dose at EAB, 2 h	193	193
Whole body dose at EAB, 2 h	1.4	1.11
Thyroid dose at LPZ, 30 days	180	180
Whole body dose at LPZ, 30 days	0.62	0.44

(a)FSAR changed since cycle 7 reload.

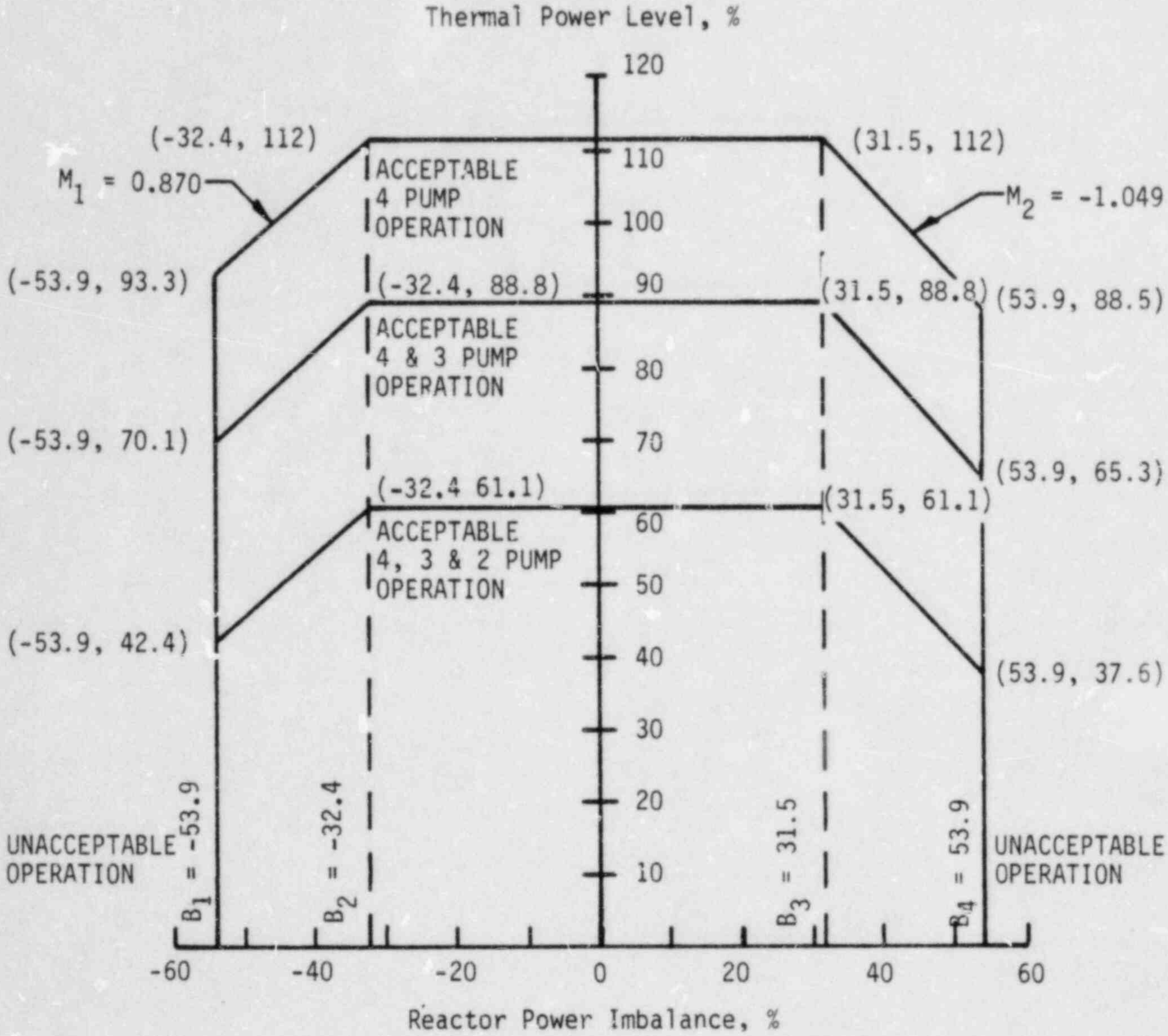
(b)Not listed in FSAR.

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 9 operation in accordance with the methods of references 18 through 21 to account for changes in power peaking and control rod worths as well as the inclusion of gray APSRs in the design.

Based on 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. Figures 8-1 through 8-14 are revisions to previous Technical Specification limits. The allowable withdrawal limit for gray APSRs is 0-100% withdrawn at any time during cycle 9. Therefore the figures pertaining to APSR withdrawal limits have been deleted.

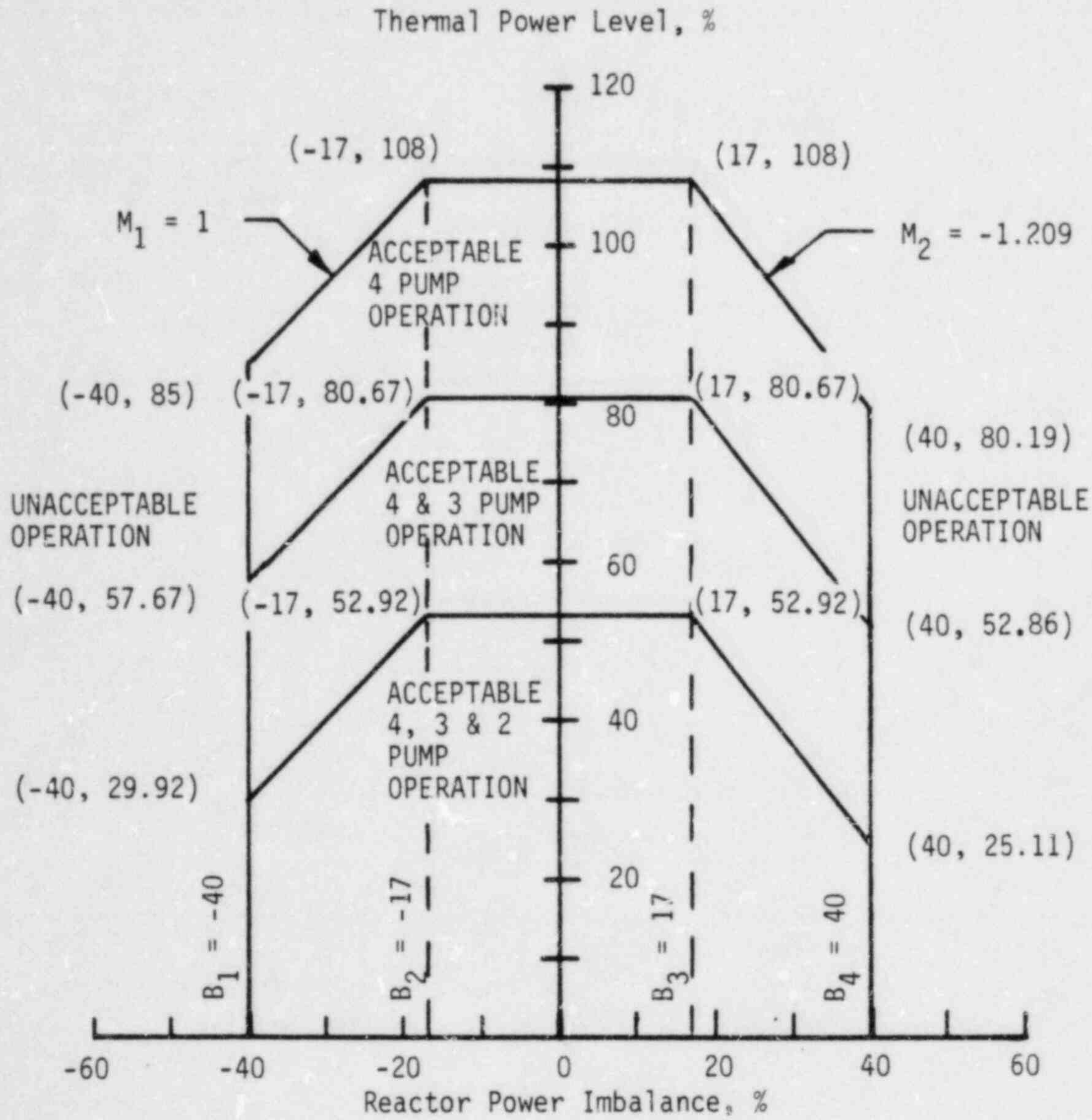
Figure 8-1. Core Protection Safety Limits for Oconee Unit 1, Cycle 9 (Technical Specification Figure 2.1-2A)



CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

2.1-7

Figure 8-2. Protective System Maximum Allowable Setpoints for Oconee Unit 1, Cycle 9 (Technical Specification Figure 2.3-2A)



2.3-8

Figure 8-3. Rod Position Limits for Four-Pump Operation,
0 to 30 +10/-0 EFPD, Oconee 1, Cycle 9
(Technical Specification Figure 3.5.2-1A1)

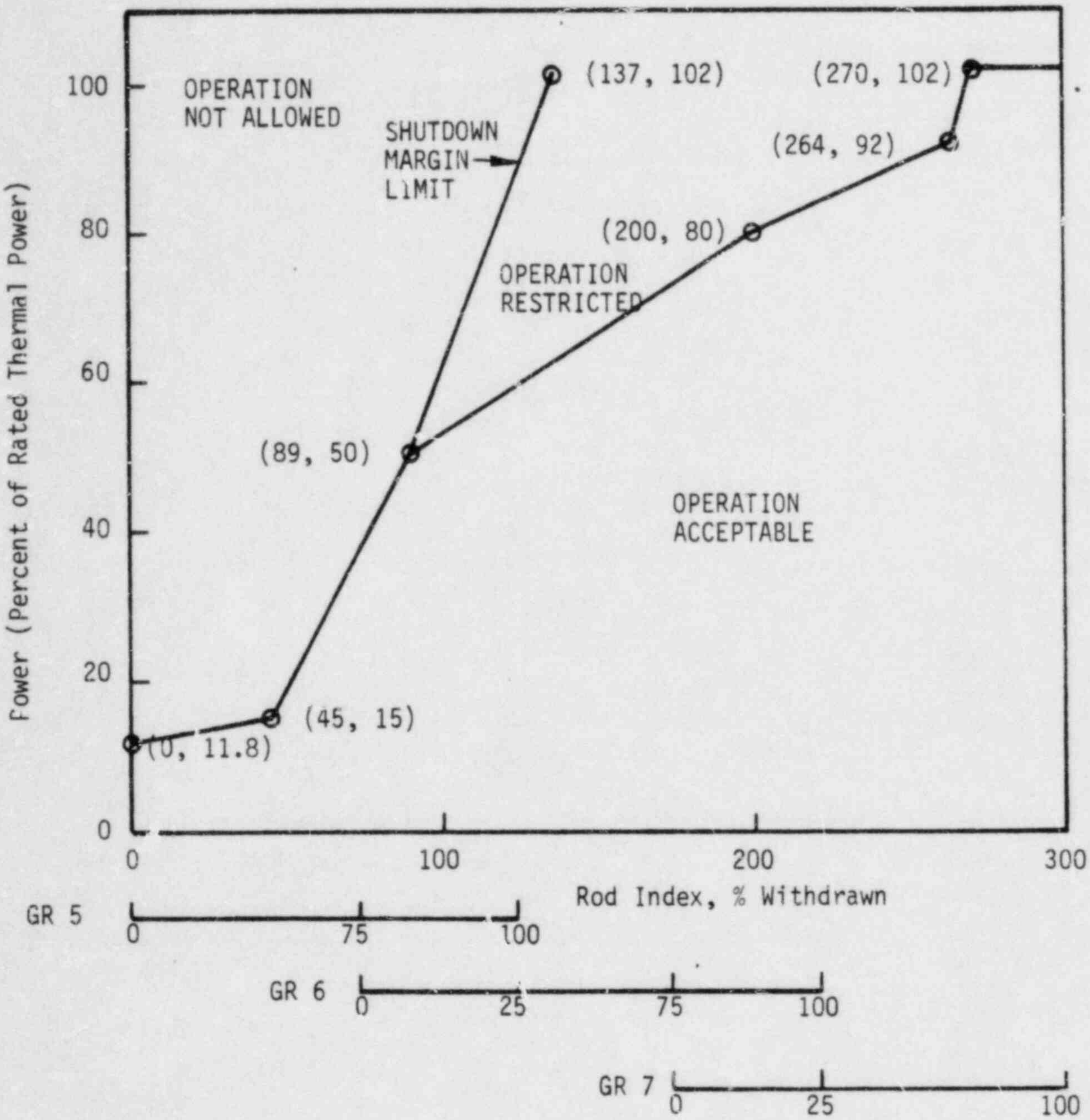
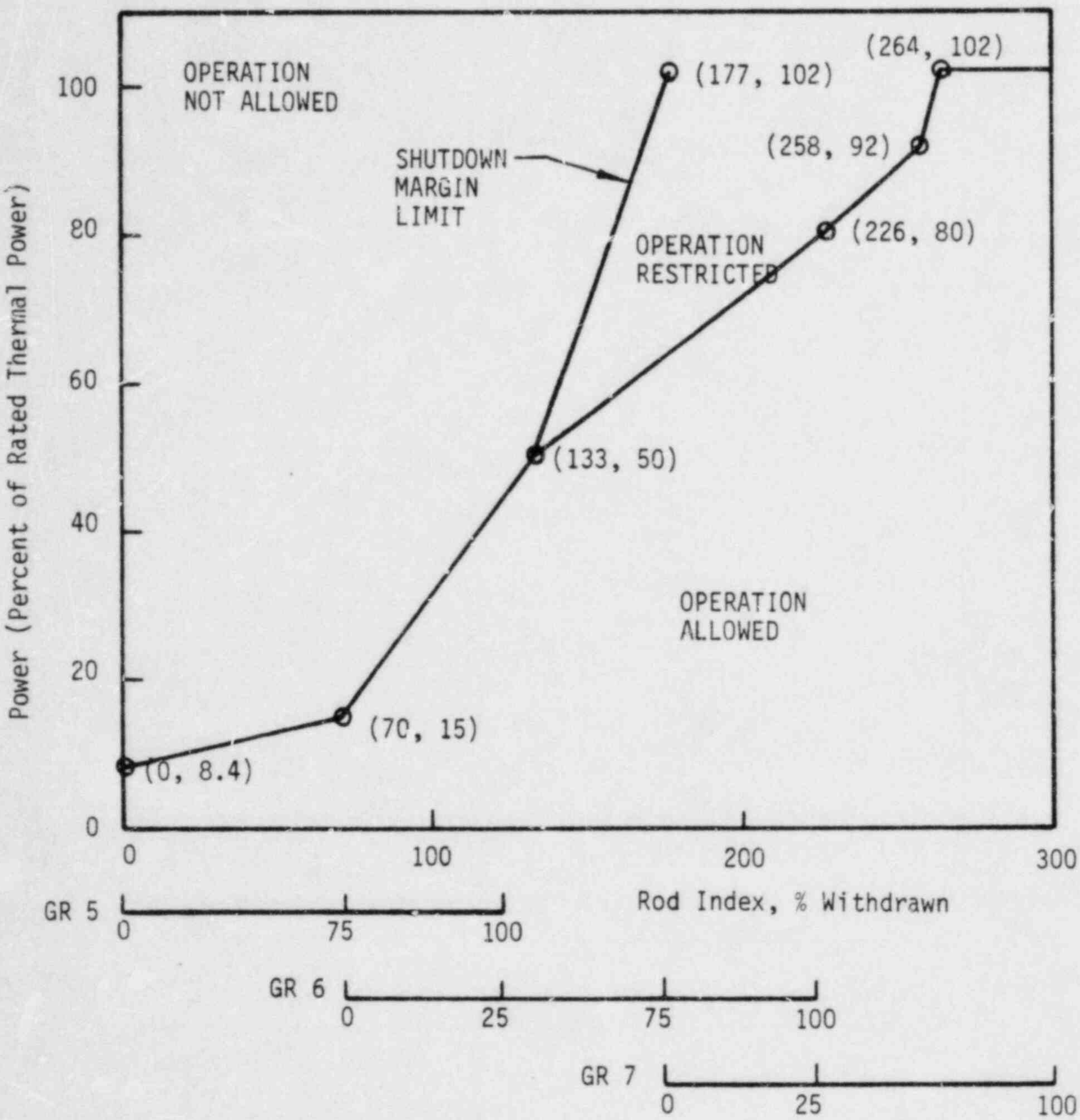


Figure 8-4. Rod Position Limits for Four-Pump Operation, 30 +10/-0 to 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-1A2)



3.5-15a

Figure 8-5. Rod Position Limits for Four-Pump Operation After 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-1A3)

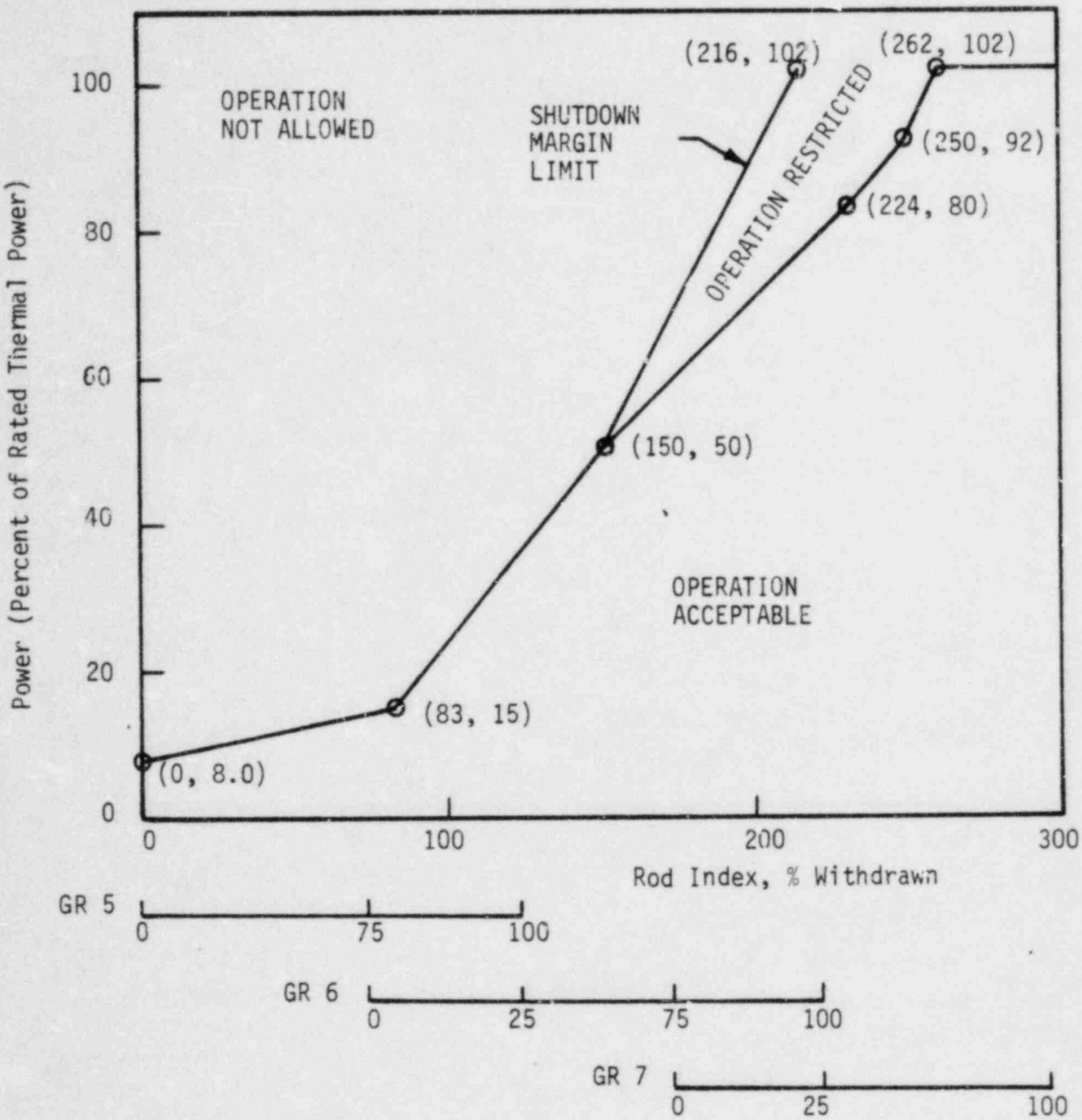
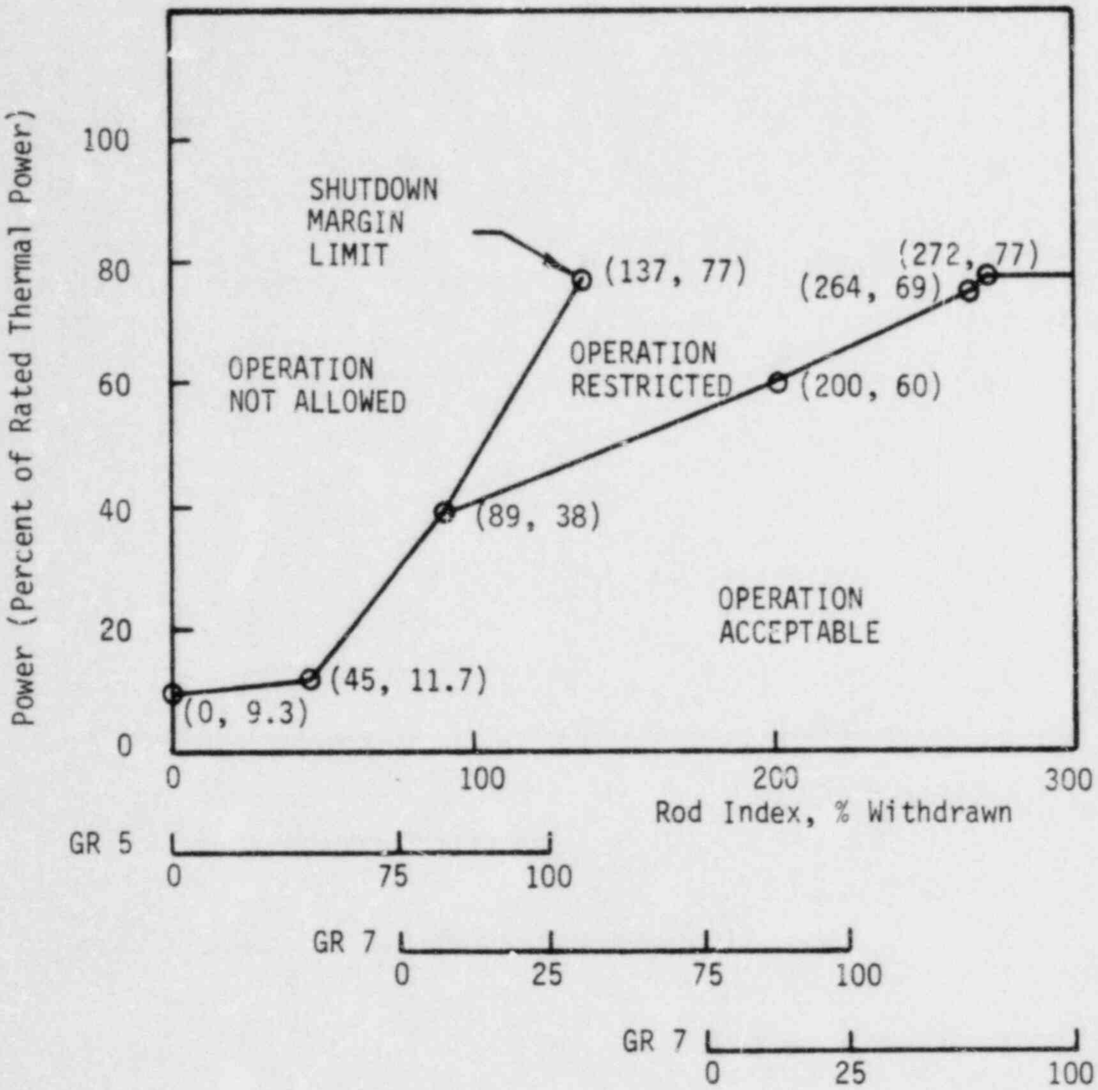
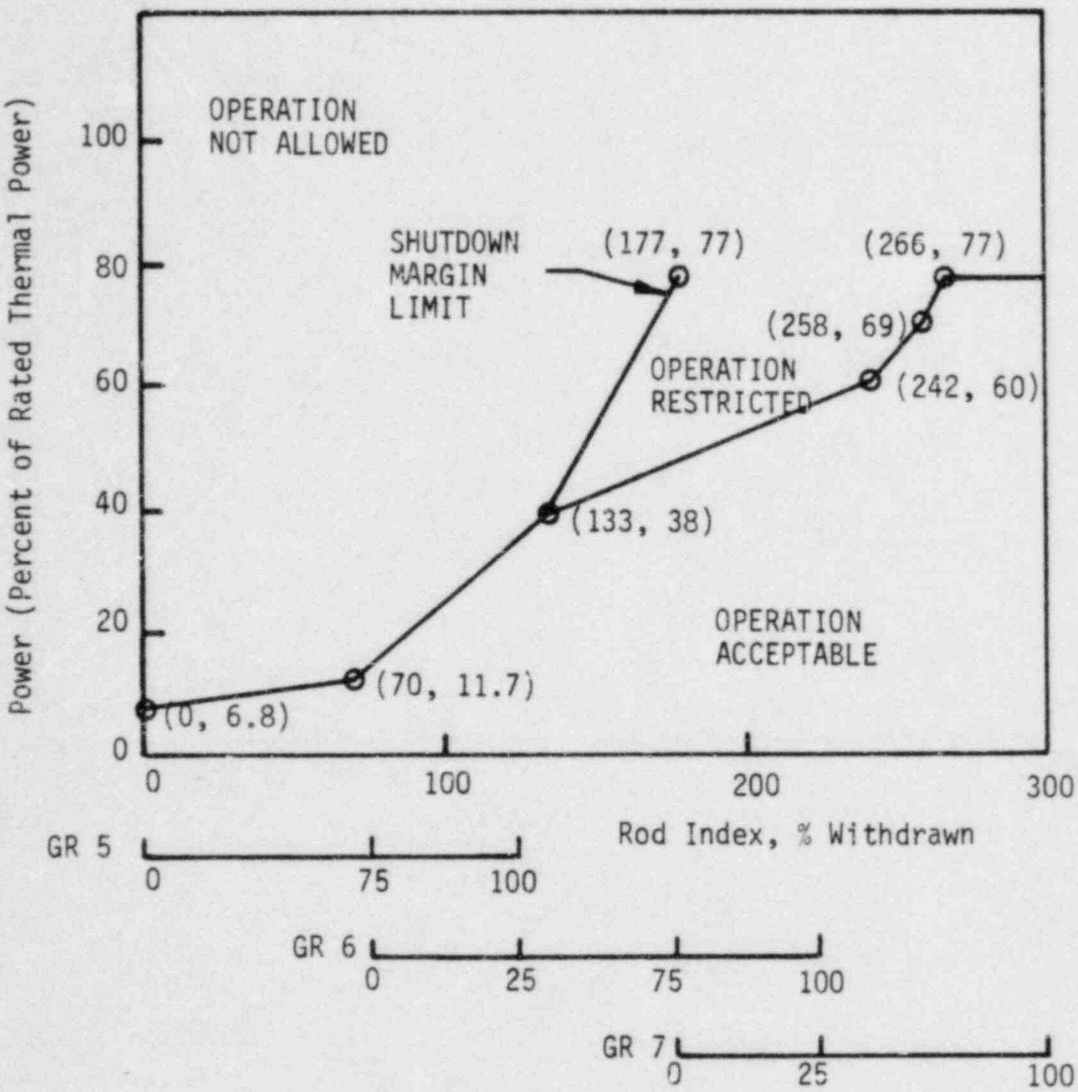


Figure 8-6. Rod Position Limits for Three-Pump Operation,
0 to 30 +10/-0 EFPD, Oconee 1, Cycle 9
(Technical Specification Figure 3.5.2-2A1)



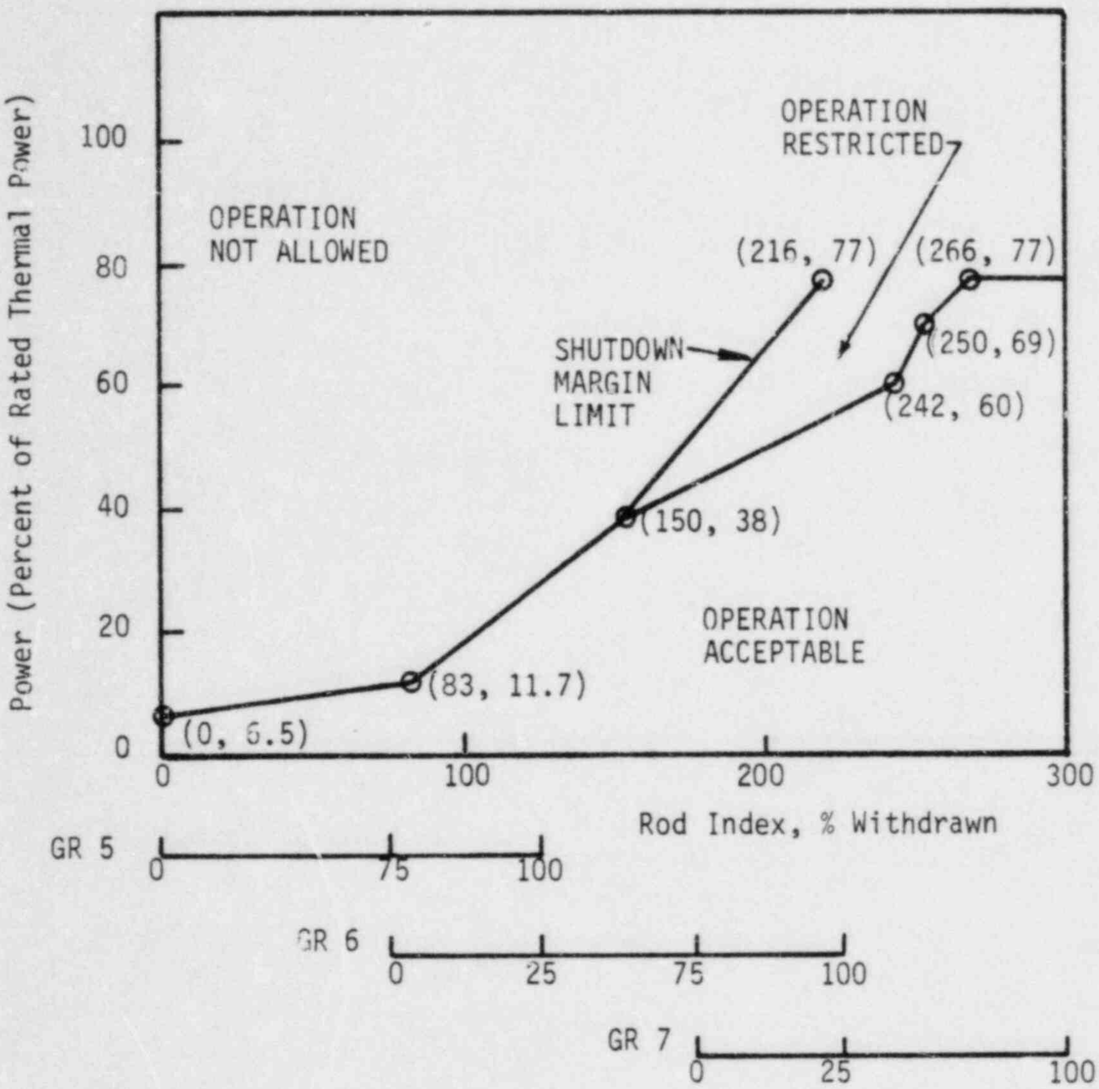
3.5-18

Figure 8-7. Rod Position Limits for Three-Pump Operation, 30 +10/-0 EFPD to 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-2A2)



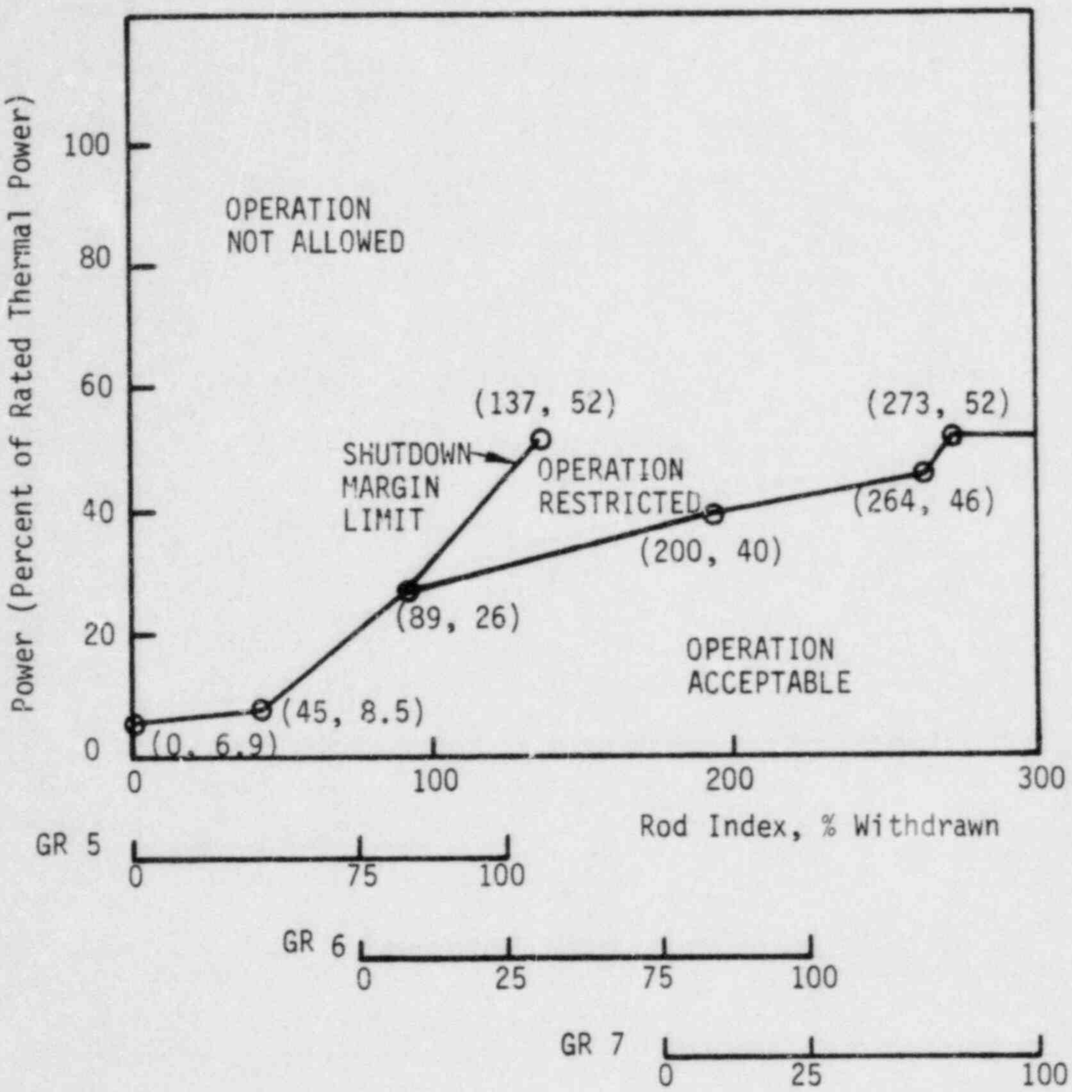
3.5-18a

Figure 8-8. Rod Position Limits for Three-Pump Operation After 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-2A3)



3.5-18b

Figure 8-9. Rod Position Limits for Two-Pump Operation, 0 to 30 +10/-0 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-2A4)



3.5-18c

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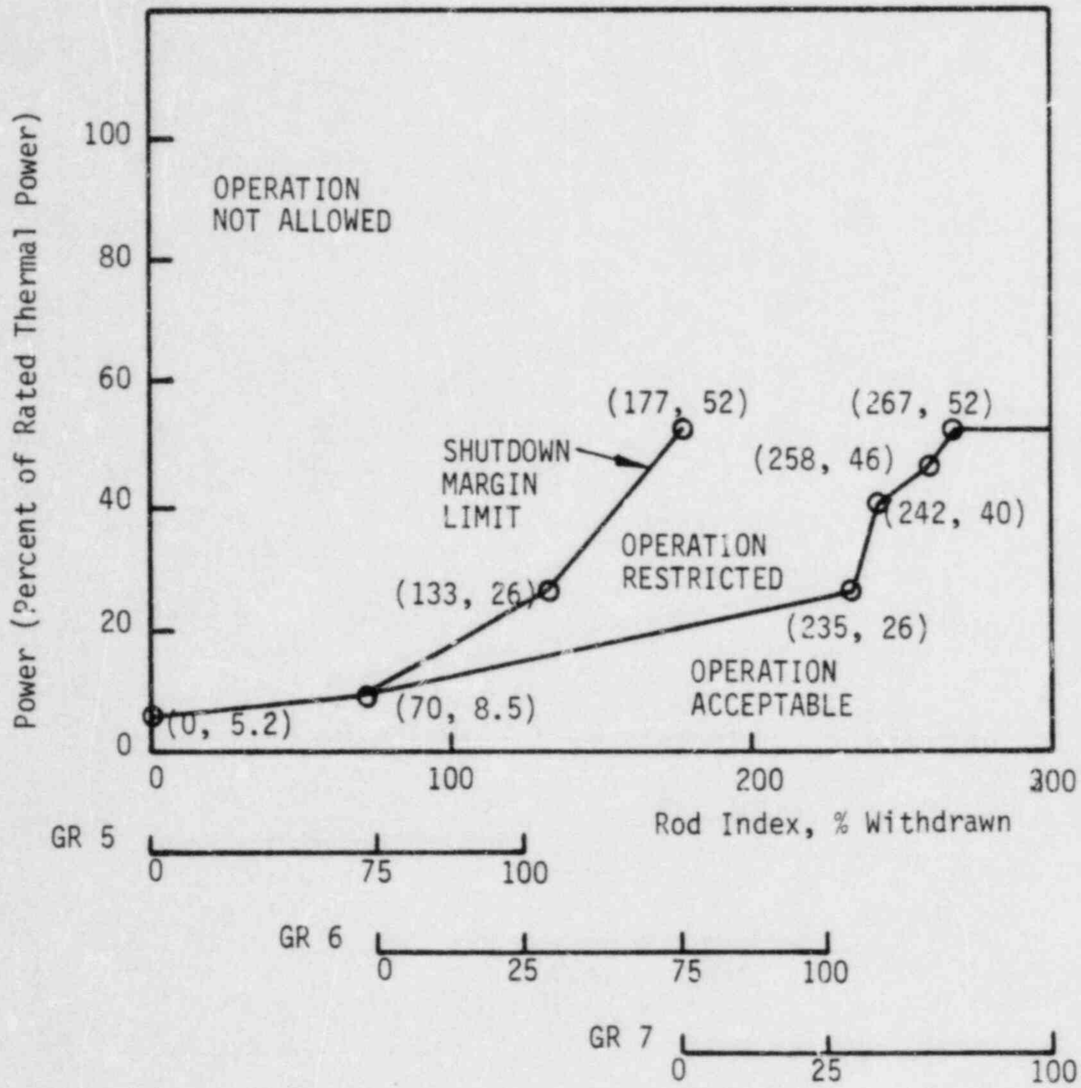
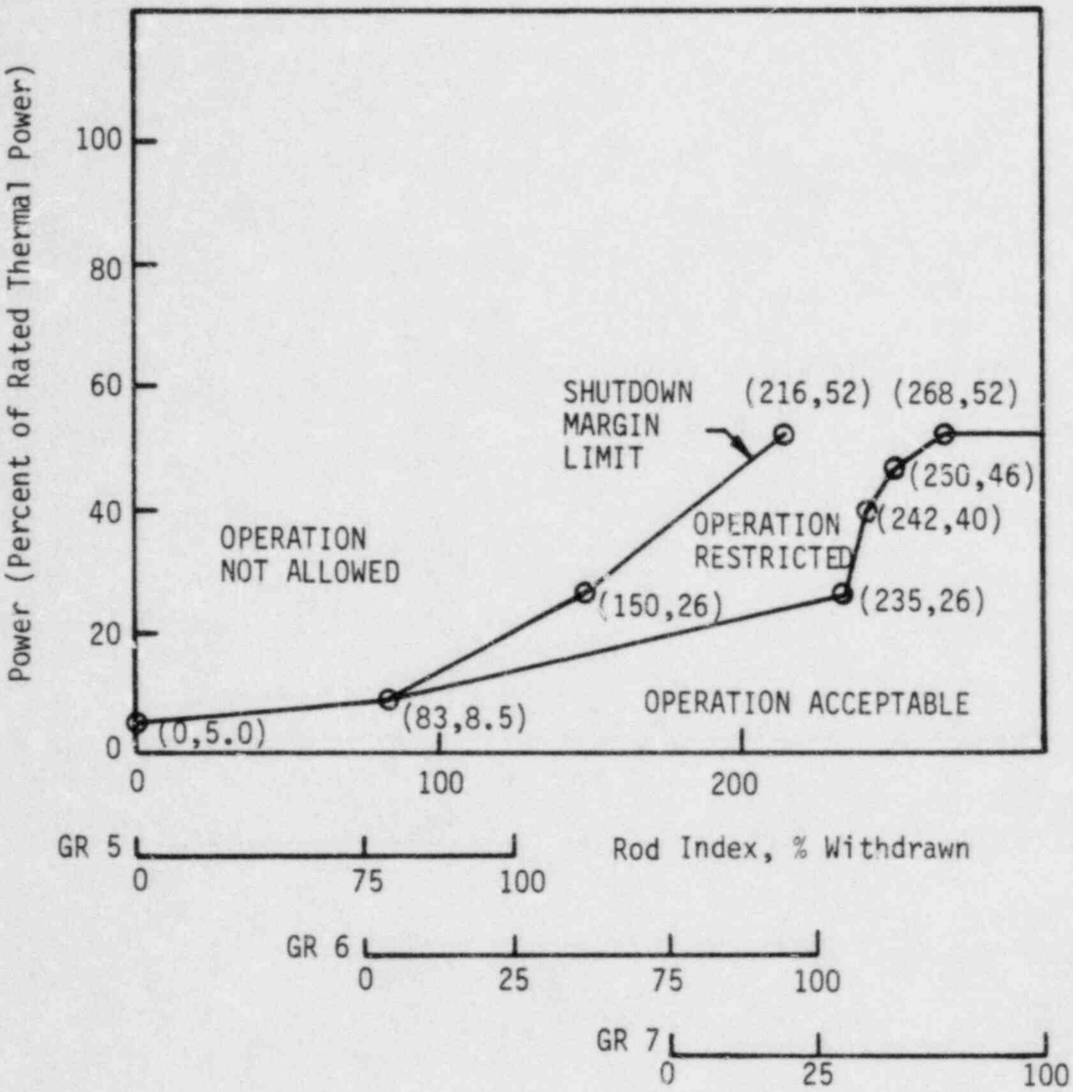
Figure 8-10. Rod Position Limits for Two-Pump Operation, 30 +10/-0 to 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-2A5)



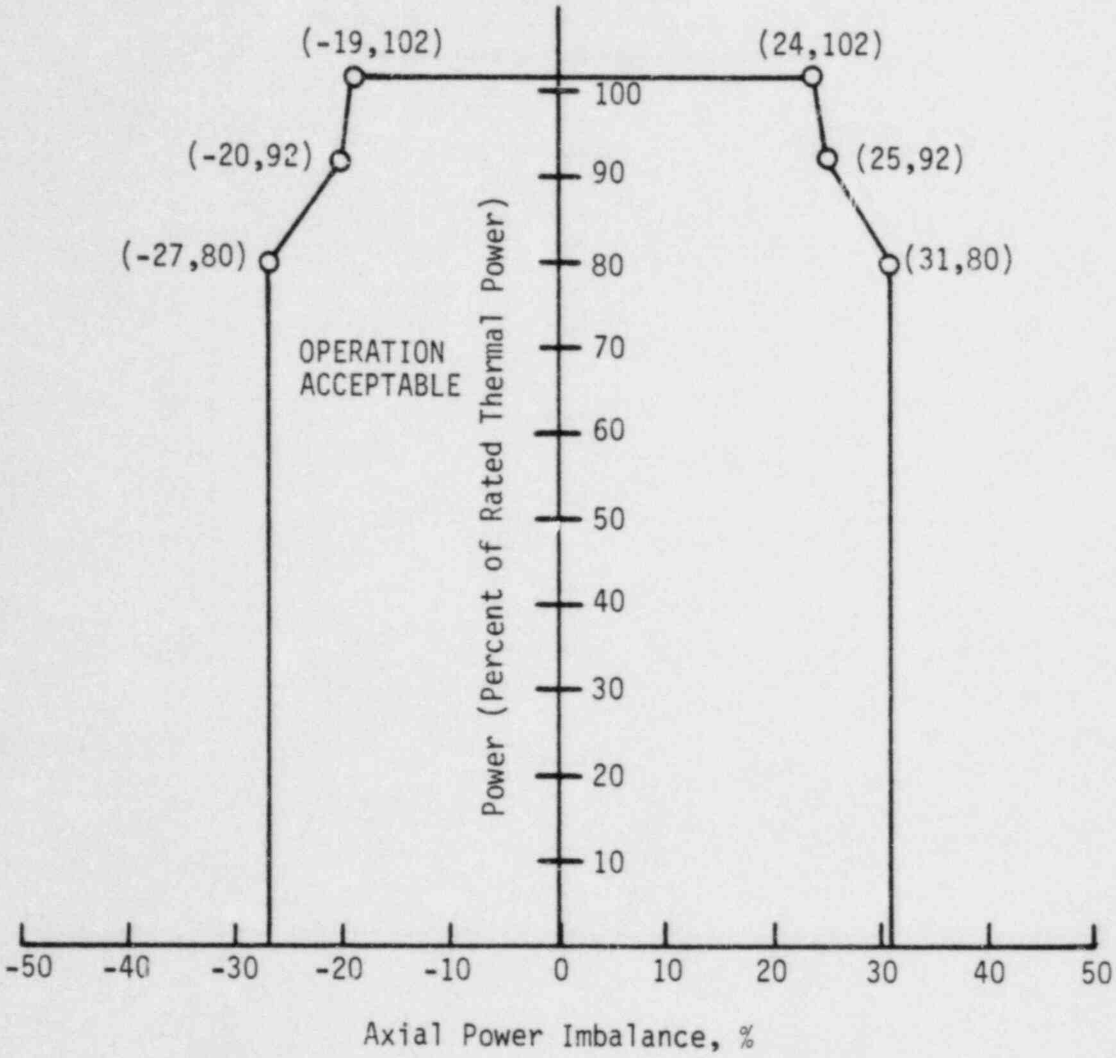
3.5-18d

Figure 8-11. Rod Position Limits for Two-Pump Operation After 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-2A6)



3.5-18e

Figure 8-12. Power Imbalance Limits for 0 to 30 +10/-0 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-3A1)



3.5-21

Figure 8-13. Power Imbalance Limits, 30 +10/-0 to 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-3A2)

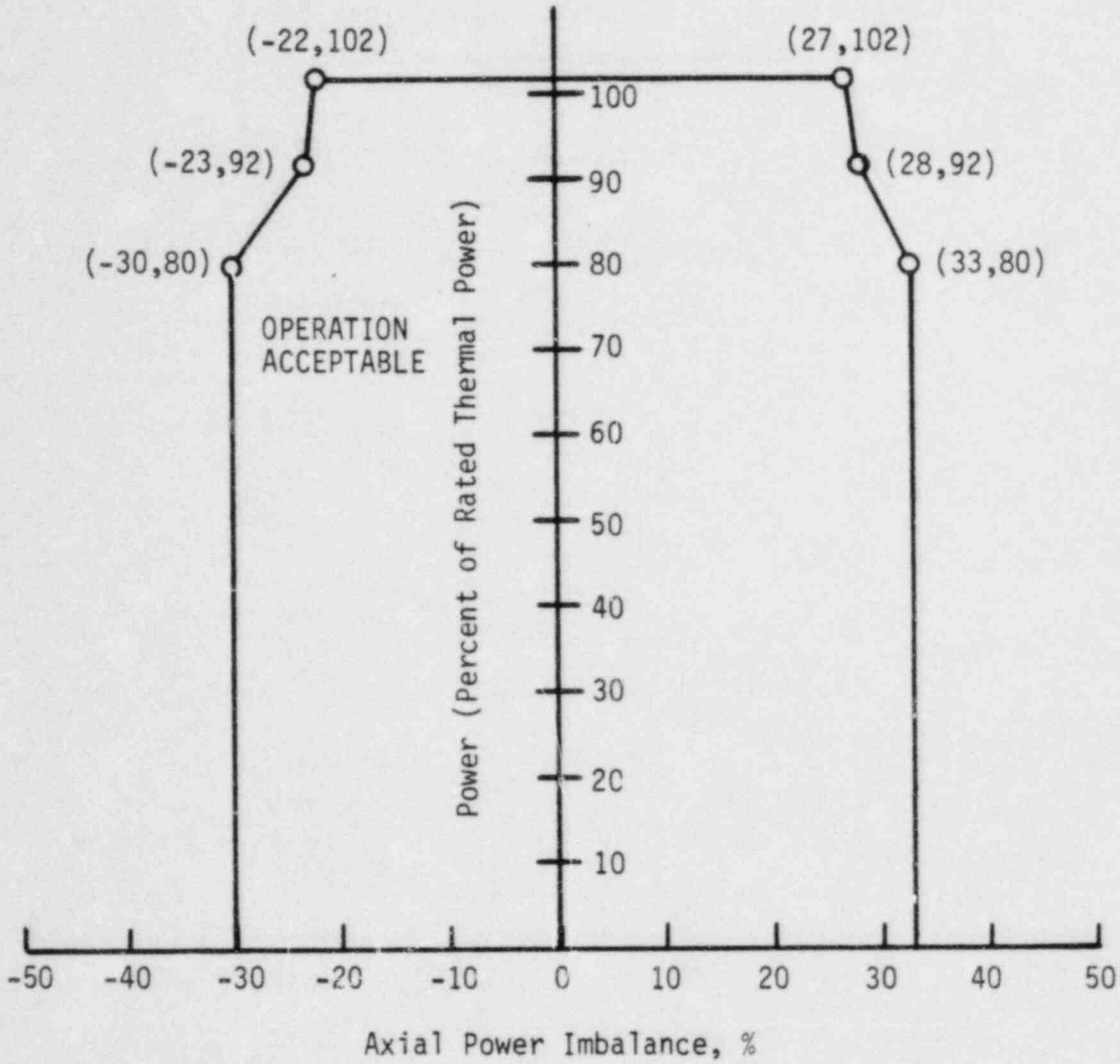
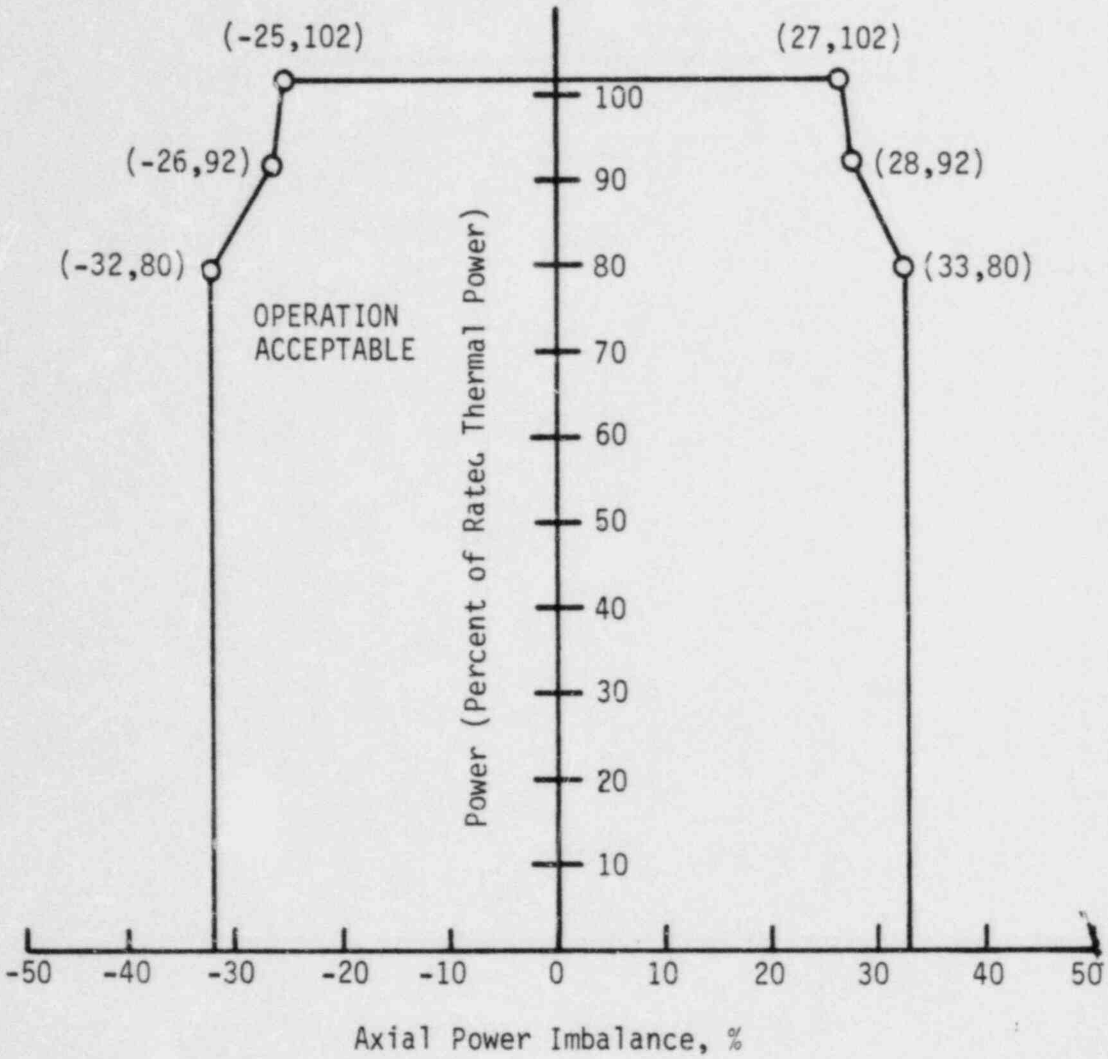


Figure 8-14. Power Imbalance Limits After 250 ±10 EFPD, Oconee 1, Cycle 9 (Technical Specification Figure 3.5.2-3A3)



3.5-21b

Figure 3.5.2-4A1
(Deleted)

Figure 3.5.2-4A2
(Deleted)

Figure 3.5.2-4A3
(Deleted)

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