OCONEE UNIT 3, CYCLE 9

- Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the ninth cycle of Oconee Nuclear Station, Unit 3, 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 Unit 3, this report employs analytical techniques and design bases established in reports that were previously submitted 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 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.

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

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems, it has been concluded that Oconee Unit 3 can be operated safely for Cycle 9 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 3, Cycle 9, is the currently operating Cycle 8. Cycle 8 achieved initial criticality on May 25, 1984 and power escalation commenced on May 26, 1984. The fuel cycle design length for Cycle 9 - 400 EFPD - is based on Cycle 8 length of 400 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in Cycle 9.

Cycle 9 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 8.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 3, of the FSAR.[1] The Cycle 9 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 consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 3, Cycle 9. Thirty-one of the Batch 9 assemblies will be discharged at the end of Cycle 8 along with Batch 8B. The remaining 41 Batch 9 assemblies, designated "9B," and the fresh Batch 11 FAs - with initial enrichments of 3.18 and 3.22 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 10, with an initial enrichment of 3.28 wt % ^{235}U, will occupy primarily the core periphery. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 9.

Cycle 9 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 60 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of axial power distribution. Since gray APSRs are being utilized, there are eight control rods in group seven and twelve in group five to reduce the negative offset response to the group seven rod movement. The Cycle 9 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 9 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1. 'CORE LOADING DIAGRAM FOR OCONEE 3, CYCLE 9

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						F11 10	F13 10	K08 10	F03 10	F05 10					
B				E14 10	11	B07 10	11	N07 10	11	B09 10	11	E02 10			
C			H13 10	11	E12 10	11	A06 9B	11	A10 9B	11	E04 10	11	O08 10		
D		P05 10	11	N09 10	11	B06 9B	11	M08 10	11	B10 9B	11	K04 10	11	P11 10	
E		11	N05 10	11	K15 9B	11	B04 9B	11	B12 9B	11	R07 9B	11	N11 10	11	
F	M06 10	G02 10	11	F02 9B	11	R09 9B	D13 10	R08 9B	D03 10	K01 9B	11	F14 9B	11	G14 10	M10 10
G	O06 10	11	F01 9B	11	D02 9B	O04 10	O13 9B	11	O03 9B	O12 10	D14 9B	11	F15 9B	11	O10 10
HW —	H09 10	K12 10	11	H11 10	11	H15 9B	11	P08 9B	11	H01 9B	11	H05 10	11	G04 10	H07 10 — Y
K	C06 10	11	L01 9B	11	N02 9B	C04 10	C13 9B	11	C03 9B	C12 10	N14 9B	11	L15 9B	11	C10 10
L	E06 10	K02 10	11	L02 9B	11	G15 9B	N13 10	A08 9B	N03 10	A07 9B	11	L14 9B	11	K14 10	E10 10
M		11	D05 10	11	A09 9B	11	P04 9B	11	P12 9B	11	G01 9B	11	D11 10	11	
N		B05 10	11	G12 10	11	P06 9B	11	E08 10	11	P10 9B	11	D07 10	11	B11 10	
O			C08 10	11	M12 10	11	R06 9B	11	R10 9B	11	M04 10	11	H03 10		
P				M14 10	11	P07 10	11	D09 10	11	P09 10	11	M02 10			
R						L11 10	L13 10	G08 10	L03 10	L05 10					

Z

XX X	PREVIOUS CYCLE LOCATION BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 9

	8	9	10	11	12	13	14	15
H	3.18 28896	3.22 0	3.18 23868	3.22 0	3.28 15500	3.22 0	3.28 15945	3.28 15237
K	3.22 0	3.18 23648	3.28 12331	3.18 19307	3.22 0	3.18 21783	3.22 0	3.28 15701
L	3.18 23871	3.28 12319	3.18 21432	3.22 0	3.18 23010	3.22 0	3.28 14019	3.28 15780
M	3.22 0	3.18 19296	3.22 0	3.18 21433	3.22 0	3.28 15553	3.22 0	
N	3.28 15522	3.22 0	3.18 23009	3.22 0	3.28 15958	3.22 0	3.28 10822	
O	3.22 0	3.18 21780	3.22 0	3.28 15547	3.22 0	3.28 15976		
P	3.28 15945	3.22 0	3.28 14024	3.22 0	3.28 10821			
R	3.28 15263	3.28 15705	3.28 15806					

X.XX	INITIAL ENRICHMENT, wt% ^{235}U
XXXXX	BOC BURNUP, MWd/mtU

FIGURE 3-3. CONTROL ROD LOCATIONS FOR OCONEE 3,CYCLE 9

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

X — GROUP NO.

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 ENRICHMENT & DISTRIBUTION FOR OCONEE 3, CYCLE 9

	8	9	10	11	12	13	14	15
H		1.1		1.1		1.1		
K	1.1				1.1		0.2	
L				1.1		0.8		
M	1.1		1.1		1.1		NONE	
N		1.1		1.1		0.2		
O	1.1		0.8		0.2			
P		0.2		NONE				
R								

X.X	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 3 Cycle 9, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Two regenerative neutron sources will be used in Mark B5 fuel assemblies.

The Batch 11 Mark B5Z fuel assemblies are a Mark B5 design with zircaloy intermediate spacer grids. Both the B5 design and the zircaloy grid concept have been previously demonstrated separately (References 5 and 3, respectively), and there are no new features not previously demonstrated. Additionally, the Batch 11 fuel rods have a slightly reduced prepressurization level to provide a small increase in fuel rod burnup. This level of prepressurization has also been previously implemented[3]. All 60 BPRAs will be inserted into Batch 11 fuel assemblies.

Other results presented in the FSAR[1] fuel assembly mechanical discussions and in previous reload reports, are applicable to the reload fuel assemblies. Duke has performed generic mechanical analyses, as described below, which envelope the Cycle 9 design. All methods are consistent with the approved methodologies of Reference 10 except where specifically stated.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of Batch 9B is more limiting than other batches due to its longer previous incore exposure time. The Batch 9B assembly power histories were analyzed, and the most limiting assembly was used to perform the creep collapse analysis using the CROV computer code and procedures described in topical report BAW-10084, Rev. 2[2]. The TACO2[6] code was used to calculate internal pin pressure and clad temperatures used as input to CROV. The

collapse time for the most limiting assembly was conservatively determined to be 31,400 EFPH, which is greater than the maximum projected residence time of Cycle 9 fuel (Table 4-1).

4.2.2 Cladding Stress

As described in Reference 10, Duke has performed a generic and conservative fuel rod cladding stress analysis in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB, of the ASME Boiler and Pressure Vessel Code. All methods are consistent with Reference 10 except that the static stress analysis uses design stress intensity limits on mechanical properties based on the requirements of ASME Code Article III-2000. Compliance with ASME Code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

The following conservatisms exist in the generic cladding stress calculation:

- high external cladding pressure (110% of design system pressure)
- low internal pressure (HZP - min. specified pre-pressure)
- maximum possible radial temperature gradient through clad (fuel melt conditions)
- conservative cladding dimensions with regards to stress

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2[6] in accordance with the approved methodology [10]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3 Thermal Design

All fuel in the Cycle 9 core is thermally similar. The fresh Batch 11 fuel inserted for Cycle 9 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The linear heat rate to melt capability based on centerline fuel melt was determined separately for each batch of fuel using the TACO2[6] computer code. The fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The input shown includes the following bounding, generic conservatisms:

1. A maximum gap based on as-fabricated pellet and cladding data.
2. Maximum incore densification based on resinter test results.

The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2.

The maximum assembly average burnup is predicted to be 39,758 MWd/MtU and the maximum fuel rod burnup is predicted to be 40,912 MWd/mtU. Fuel rod internal pressure has been evaluated using TACO2[6] with a conservative pin power history, and the maximum pressure is less than the nominal reactor coolant (RC) system pressure of 2200 psia.

4.4 Material Design

With the exception of zircaloy intermediate spacer grids, the Batch 11 fuel assemblies are not unique in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the Batch 11 fuel assemblies is identical to those of the present fuel.

Table 4-1.

Fuel Design Parameters and Dimensions

	Batch No.		
	9B	10	11
FA type	Mark B5	Mark B5	Mark B5Z
No. of FAs	41	68	68
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, in.	141.8	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3686	0.3686	0.3686
Fuel pellet initial density (mean spec), %TD	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.18	3.28	3.22
Est. residence time, EOC 9, EFPH	30,460	19,680	9,840
Cladding collapse time, EFPH	>31,400	>31,400	>20,000

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.		
	9B	10	11
Nominal initial density, % TD	95.0	95.0	95.0
Nominal initial pellet diameter, in.	0.3686	0.3686	0.3686
Nominal initial clad ID, in.	0.377	0.377	0.377
Nominal initial clad OD, in.	0.430	0.430	0.430
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.74	5.74
Linear heat rate capability[b] from 0-1000 MWD/MTU, kW/ft	20.15	20.15	20.15
Linear heat rate capability[b] >1000 MWD/MTU, kW/ft	21.20	21.20	21.20
Average fuel temp. @ nominal linear heat rate, °F	1240[a]	1240[a]	1240[a]

(a) Basis: TACO2, 96.5% TD @ 4000 MWD/MTU, nominal pellet and cladding dimensions

(b) These values are utilized as fuel design limits for Cycle 9.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycles 8 and 9; the values for Cycles 8 and 9 were generated by Duke Power Company using methods described in Reference 10. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of the ninth cycle at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 8 and 9 are the different control rod pattern, different BPRA loadings, and different shuffle patterns. The control rod worths differ primarily due to the reduced number of Group 7 control rods and the introduction of gray APSRs. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. 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. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin with Cycle 9 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.

Flux redistribution was explicitly accounted for since the shutdown analysis was calculated using a three-dimensional model. The reference fuel cycle shutdown margin is presented in the Oconee 3, Cycle 8 Reload Report.[5]

5.2 Analytical Input

The Cycle 9 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 9 as for the reference cycle.

5.3 Changes in Nuclear Design

Core design changes for Cycle 9 are the use of gray APSRs, the new control rod group pattern, and the introduction of 68 Mark BZ assemblies. 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 3 Physics Parameters[a]

	Cycle 8[b]	Cycle 9[c]
Cycle length, EFPD	400	400
Cycle burnup, MWd/mtU	12,307	12,349
Average core burnup, EOC, MWd/mtU	22,818	23,035
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, groups 7 and 8 at nominal positions[d]	1550	1577
HFP, groups 7 and 8 at nominal positions	1358	1395
Critical boron - EOC (equilibrium xenon), ppm		
HZP, groups 7 and 8 at nominal positions	404	402
HFP, groups 7 and 8 at nominal positions	13	64
Control rod worths - HFP, BOC, % Δk/k		
Group 7	1.41	1.19
Group 8 (e)	0.33	0.16
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.57	1.24
Group 8 (e)	0.30	0.16
Max ejected rod worth - HZP, % Δk/k		
BOC, (N12) groups 5-8 inserted	0.42	0.69
EOC, (N12) groups 5-8 inserted	0.51	0.51
Max stuck rod worth - HZP, % Δk/k		
BOC (N12)	1.12	1.63
EOC (N12)	1.76	1.92
Power deficit, HFP to HZP, % Δk/k		
BOC	1.83	1.77
EOC	3.12	3.04
Doppler coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (equilibrium xenon)	-1.37	-1.31
EOC (equilibrium xenon)	-1.68	-1.63

Table 5-1. (Cont'd)

	Cycle 8[b]	Cycle 9[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (equilibrium xenon)	-0.62	-0.58
EOC (equilibrium xenon)	-2.92	-2.91
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	122	123
EOC	110	110
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.48	2.43
EOC (equilibrium)	2.68	2.65
Effective delayed neutron fraction - HFP		
BOC	0.00617	0.00618
EOC	0.00522	0.00523

[a] Cycle 9 data are for the conditions stated in this report. The Cycle 8 core conditions are identified in Reference 5.

[b] Based on a 440-EFPD cycle 7. (Actual Cycle 7 length 449.15 EFPD).

[c] Based on a Cycle 8 length of 400-EFPD.

[d] Nominal positions are as follows:

	Cycle 8	Cycle 9
HZP (BOC)	group 7 at 100% WD, 8 at 37.5% WD	group 7 at 100% WD, 8 at 25.5% WD
HFP (BOC)	group 7 at 87% WD, 8 at 25% WD	group 7 at 92% WD, 8 at 15% WD
HZP (EOC)	group 7 at 100% WD, 8 at 37.5% WD	group 7 at 100% WD, 8 at 25.5% WD
HFP (EOC)	group 7 at 87% WD, 8 at 25% WD	group 7 at 100% WD, 8 at 15% WD

(Changes in Group 7 and Group 8 nominal rod positions between Cycle 8 and Cycle 9 are due to the introduction of gray APSRs in Cycle 9)

(e) (25% to 100% WD for Cycle 8, 15% to 100% WD for Cycle 9)

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 9

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.53	9.42
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.63	-1.92
Net worth	6.48	7.08
Less 10% uncertainty	-0.65	-0.71
Total available worth	5.83	6.37
Required Rod Worth		
Power deficit, HFP to HZP	1.77	3.04
Max inserted rod worth, HFP	0.38	0.59
Total required worth	2.15	3.63
Shutdown Margin		
Total available worth minus total required worth	3.68	2.74

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

FIGURE 5-1
OCONEE 3 CYCLE 9
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.802	1.014	0.936	1.229	1.247	1.262	1.061	0.555
K	1.014	0.894	1.069	1.086	1.248	1.089	1.199	0.538
L	0.936	1.069	0.994	1.179	0.995	1.255	0.966	0.410
M	1.229	1.086	1.179	1.069	1.234	1.143	0.938	
N	1.247	1.248	0.995	1.234	1.171	1.127	0.550	
O	1.262	1.089	1.255	1.143	1.127	0.571	S_L	
P	1.061	1.199	0.966	0.938	0.550			
R	0.555	0.538	0.410					

6. THERMAL-HYDRAULIC DESIGN

The generic Mark-B and Mark-BZ thermal-hydraulic design analyses supporting Cycle 9 operation were performed by Duke Power Company using the methods described in References 1, 5, 8, and 10. The Cycle 8 and Cycle 9 maximum design conditions are summarized in Table 6-1.

The Cycle 9 transition core will include 68 fresh Mark-BZ Batch 11 fuel assemblies, 60 of which will contain BPRAs. Two assemblies will contain regenerative neutron sources, leaving 46 fuel assemblies with open guide tubes. This results in a core bypass flow of 7.9% of the total system flow. This bypass flow is less than that assumed in the generic thermal-hydraulic analyses and the consequent increase in core flow establishes the generic analyses as conservative for Cycle 9 operation.

The Mark-BZ fuel assembly has a slightly higher pressure drop than the Mark-B assembly as a result of the increased flow resistance of the Zircaloy spacer grids. The presence of Mark-BZ assemblies in a predominantly Mark-B core results in less coolant flow in the Mark-BZ fuel then would occur in an all Mark-BZ core. The generic Mark-BZ analyses conservatively account for this transition core effect.

In a Mark-BZ transition core the limiting Mark-B hot channel will receive more coolant and yield better DNB performance than would be predicted for a full Mark-B core. Thus, the generic Mark-B analyses, based on the B&W-2 CHF correlation, are bounding and are applicable to the Cycle 9 transition core.

No fuel rod bow penalty was included in the DNBR limit used in the generic Mark-BZ analyses, as justified in Reference 9. The rod bow topical report concludes that a DNBR penalty is no longer required for thermal-hydraulic analyses. Nevertheless, to account for fuel rod bow, the generic Mark-B analyses used for determining plant operating limits (except the flux to flow setpoint analysis) were based on a DNBR criteria including 10.2% margin from

the 1.30 design limit. Primarily due to this conservatism, the current pressure-temperature envelope and design radial x local peaking have been shown to be conservative for a full and transition Mark-BZ core.

A flux to flow setpoint of 1.07 will be used for Cycle 9 operation. A conservative transition core pump coastdown analysis was performed based on a 1.08 flux to flow setpoint and the reference design radial-local peaking factor, $F_{\Delta H}$ = 1.714. The minimum DNBR determined in the Mark-BZ transition core flux to flow analysis is greater than the BWC CHF correlation limit of 1.18, Reference 11. The minimum DNBR determined in the generic Mark-B flux to flow analysis, also based on a 1.08 flux to flow setpoint, is greater than the BAW-2 CHF correlation limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 8	Cycle 9
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Core bypass flow, % total flow[a]	7.9	7.9
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	555.6/602.4
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine
Active fuel length, in.	141.8	141.8
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	176[b]	176[b]
CHF correlation	BAW-2	BAW-2/BWC
Min DNBR with densification penalty	2.05	>2.05/>1.74
Hot channel factors: Enthalpy rise	1.011	1.011/1.011
Heat flux	1.014	1.014/1.014
Flow area	0.98	0.98/0.97

[a] Generic analyses based on ≥8.0% core bypass flow.

[b] Heat flux based on a conservative minimum densified length of 140.3 in.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in Cycle 8 parameters to determine the effect 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 Reference 8. Since Batch 11 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in Reference 8, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in Reference 13 are characteristic for Oconee 3 Cycle 9 based upon comparisons of key parameters which determine radionuclide inventories.

7.2 Accident Evaluations

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

Fuel thermal analysis parameters for each batch in Cycle 9 are given in Table 4-2. Table 6-1 compares the Cycle 8 and 9 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 9.

A generic LOCA analysis for the B&W 177-FA, lowered-loop NSSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103, Rev. 3.[12] 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 LOCA limits analysis[7,12] is conservative compared to those calculated for this reload. In addition, it has been determined that the slightly lower prepressurization of the Batch 11 fuel rods has a negligible impact on the LOCA analyses[14]. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 3, Cycle 9 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 3 Cycle 9 fuel after 65 EFPD. The LOCA kW/ft limits have been reduced for the first 65 EFPDs. The reduction will ensure conservative limits based upon an interim bounding analytical assessment of NUREG 0630 on LOCA and operating kW/ft limits performed by Babcock and Wilcox[4,15]. The LOCA kw/ft limits for the first 65 EFPD are shown in Table 7-3.

From the examination of Cycle 9 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the safe operation of Oconee 3 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 of the transients in Cycle 9 are bounded by the FSAR and/or the fuel densification report.[8]

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted Cycle 9 value
BOC Doppler coeff, 10^{-5}, Δk/k/°F	-1.17	-1.31
EOC Doppler coeff, 10^{-5} Δk/k/°F	-1.33[(a)]	-1.63
BOC moderator coeff, 10^{-4}, Δk/k°F	+0.5[(b)]	-0.58
EOC moderator coeff, 10^{-4}, Δk/k/°F	-3.0	-2.91
All rod bank worth, HZP, % Δk/k	10.0	9.42
Boron reactivity worth, 70°F ppm/1% Δk/k	75	88
Max. ejected rod worth, HFP, % Δk/k	0.65	0.21
Dropped rod worth, HFP, % Δk/k	0.46	0.13
Initial boron conc, HFP, ppm	1400	1395

(a) -1.2×10^{-5} Δk/k/F was used for steam line break analysis.

-1.3×10^{-5} Δk/k/F was used for cold water accident (pump start-up).

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

Table 7-2. LOCA Limits, Oconee 3, Cycle 9, After 2600 MWD/mtU(b)

Elevation, ft	LHR limits, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

Table 7-3. LOCA Limits, Oconee 3, Cycle 9

Elevation, ft	LHR Limits, kW/ft	
	0-1000 MWd/mtU[a]	1000-2600 MWd/mtU[b]
2	13.5	15.0
4	16.1	16.6
6	16.5	18.0
8	17.0	17.0
10	16.0	16.0

(a) 1000 MWd/mtU corresponds to approximately 25 EFPD for the most limiting assembly

(b) 2600 MWd/mtU corresponds to approximately 65 EFPD for the most limiting assembly

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for Cycle 9 operation in accordance with the methods of Reference 10 to account for minor changes in power peaking and control rod worths.

In addition:

1. The operating limits on rod index and axial power imbalance were developed in accordance with the LOCA linear heat rate limits discussed in Chapter 7.

2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 9. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 9 operation.

3. The minimum required boron concentration in the CBAST has been changed from 8700 ppm to 11,000 ppm to ensure a 1.0% $\Delta K/K$ shutdown margin when borating from hot full power to cold shutdown.

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 8-6 are revisions to previous Technical Specification limits.

Figure 8-1

Core Protection Safety Power-Imbalance Limits

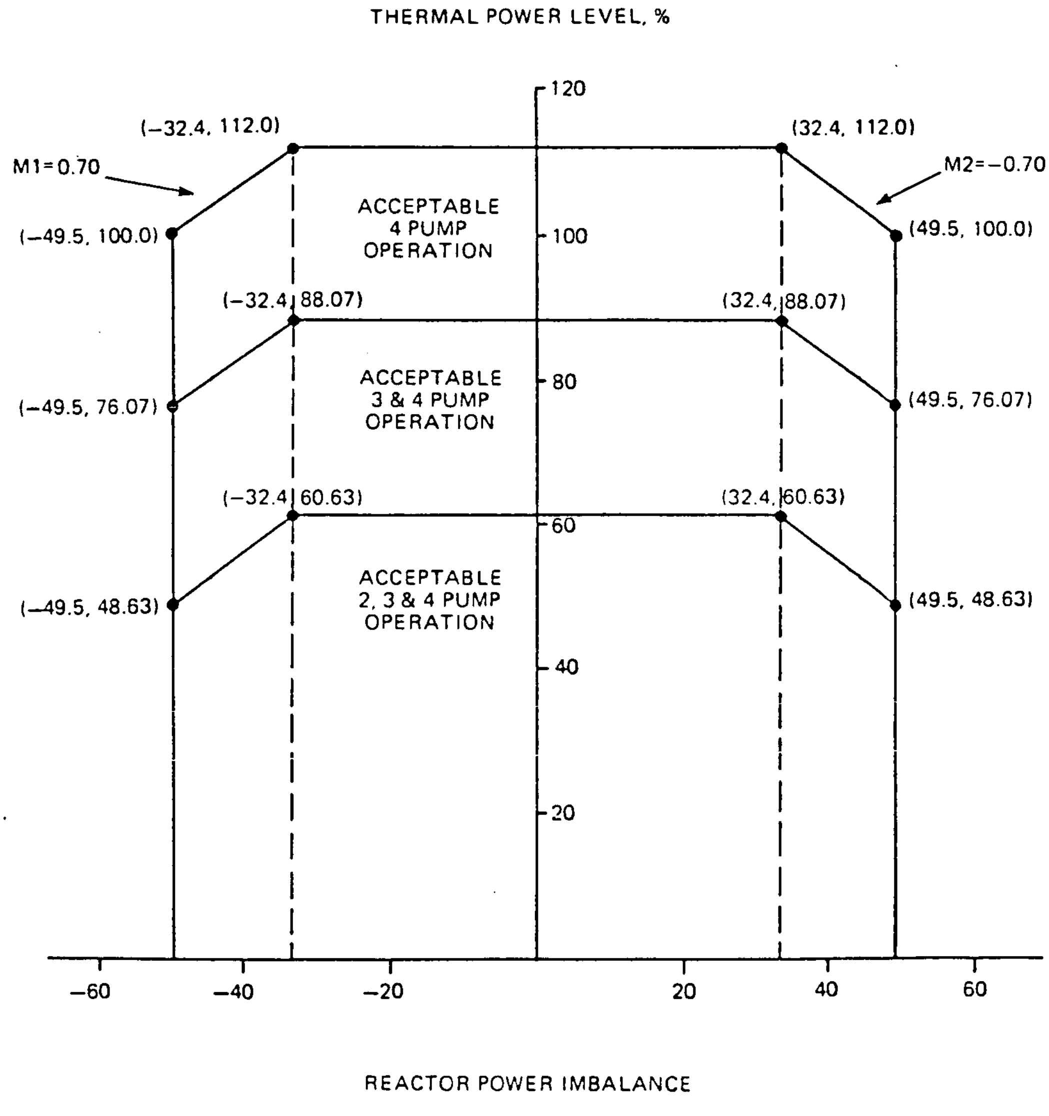


Figure 8-2

Maximum Allowable Power-Imbalance Setpoints

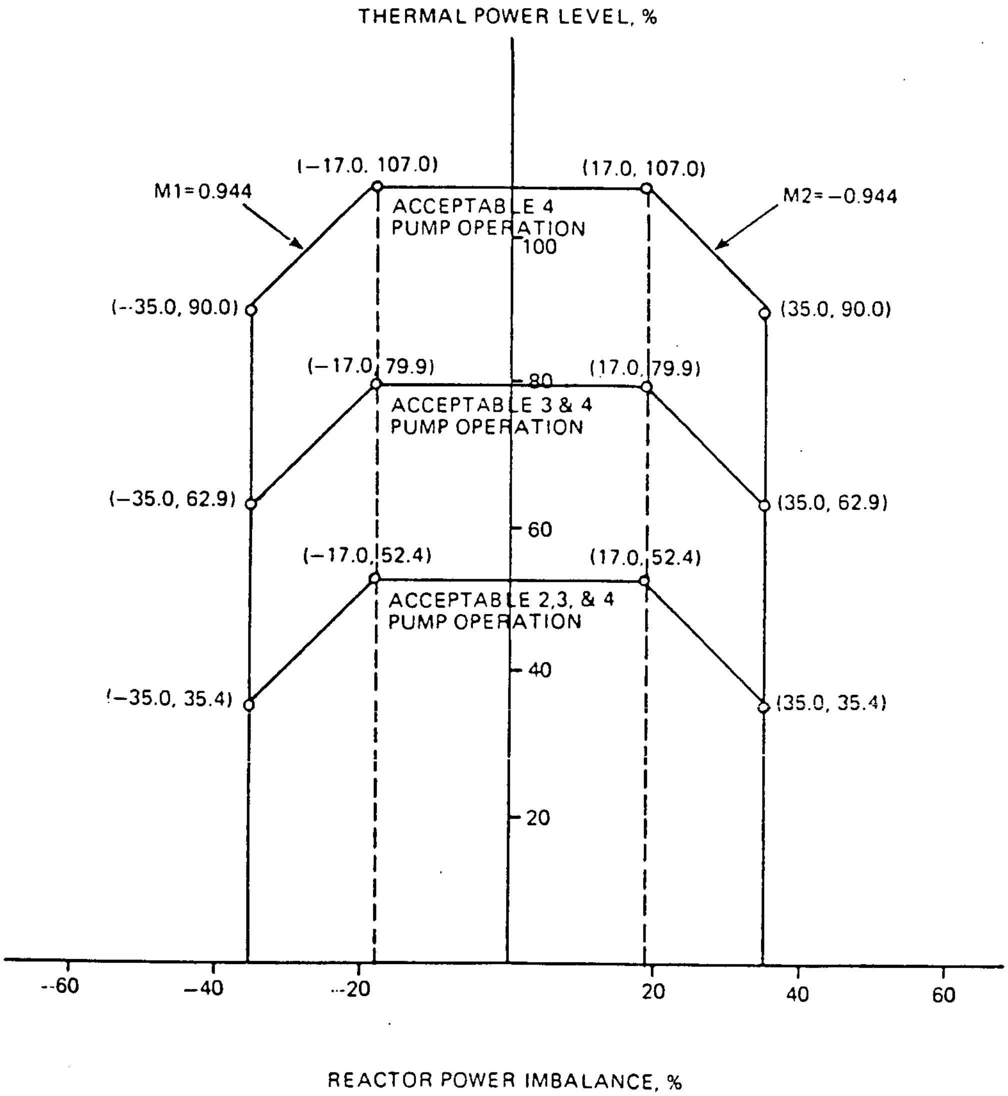


Figure 8-3

Operational Power-Imbalance Limits, 0 EFPD to EOC

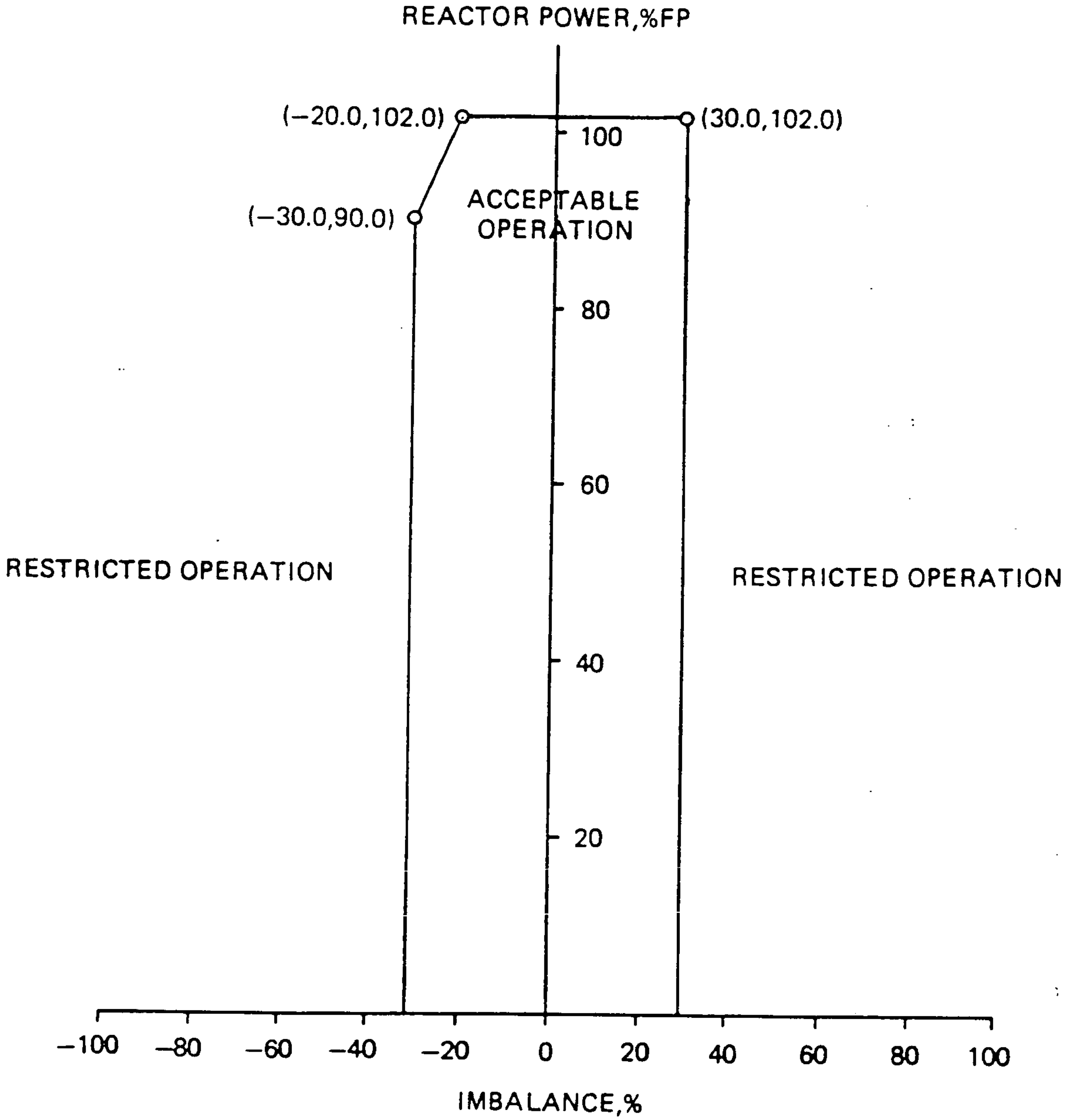


Figure 8-4

Control Rod Position Limits, 4 Pumps, 0 EFPD to EOC

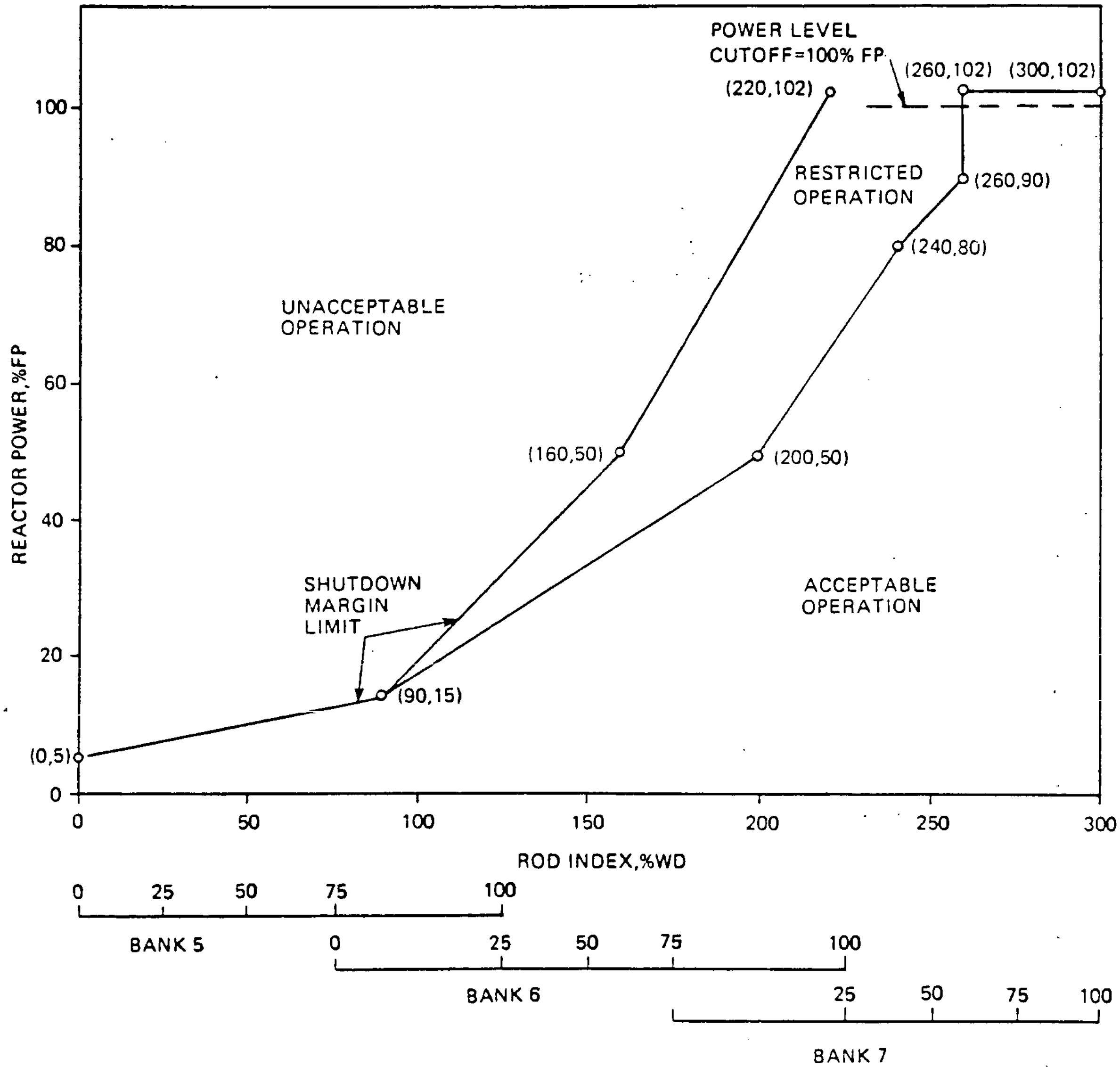


Figure 8-5

Control Rod Position Limits, 3 Pumps, 0 EFPD to EOC

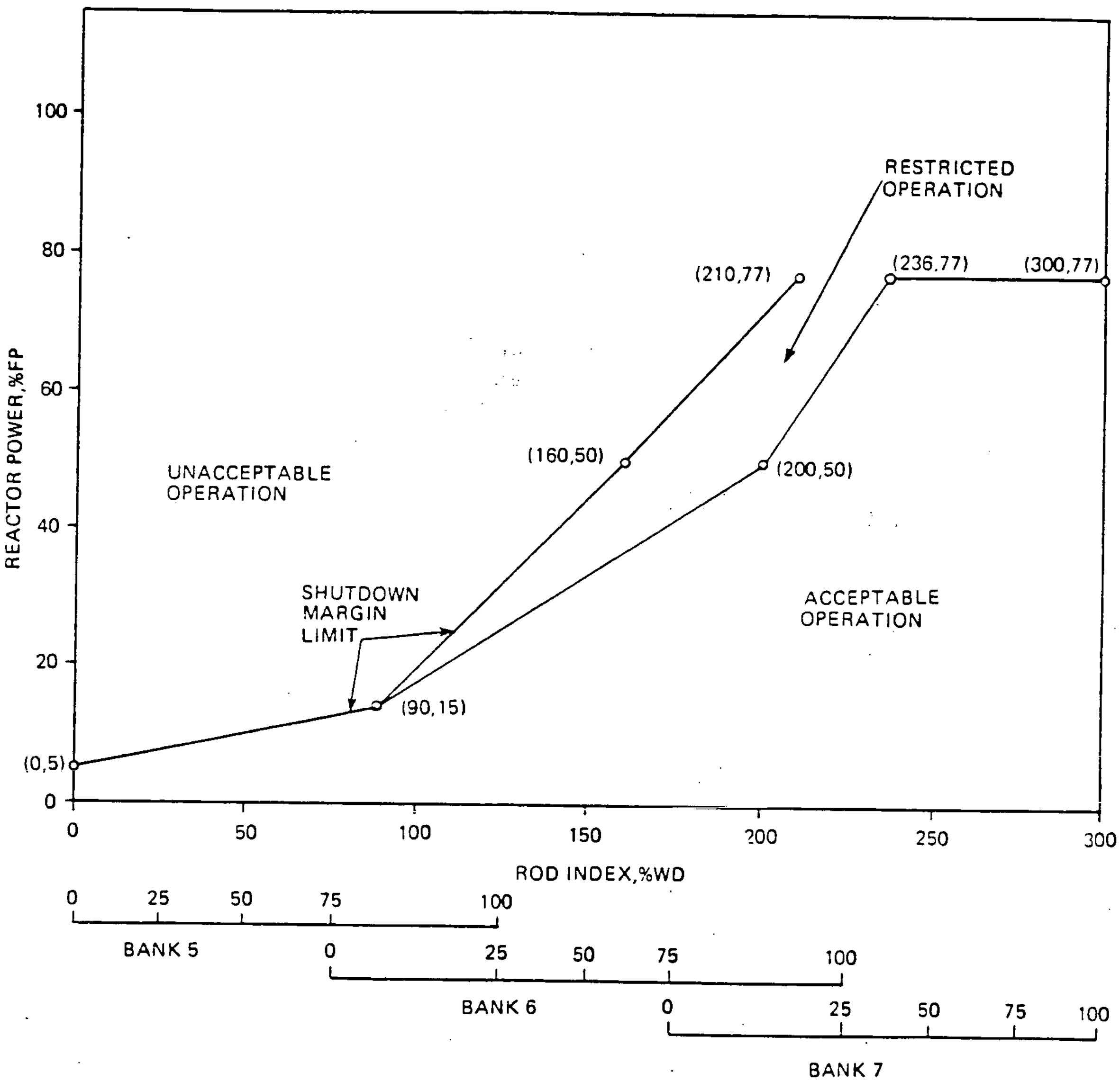
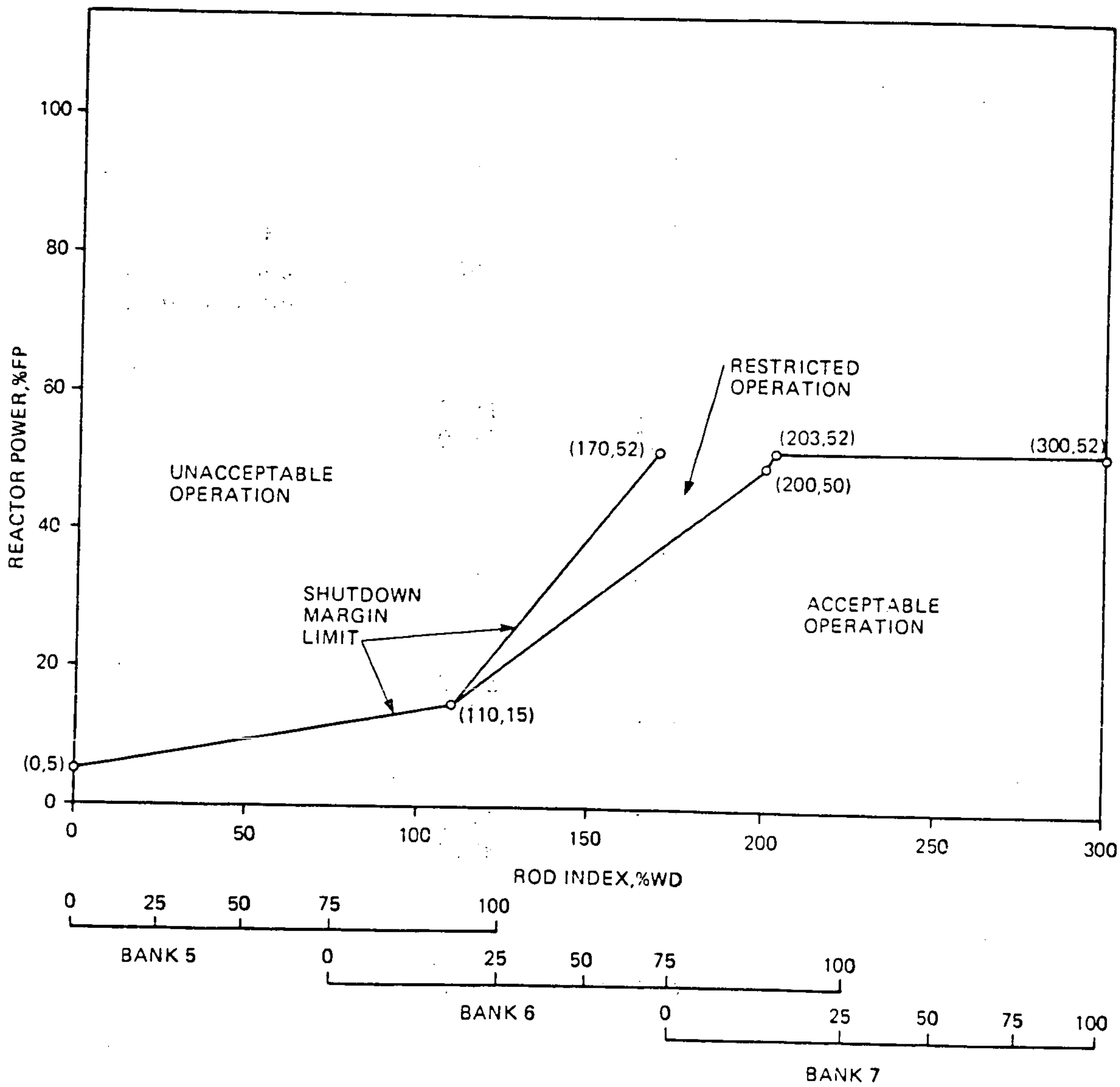


Figure 8-6

Control Rod Position Limits, 2 Pumps, 0 EFPD to EOC



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

3. Oconee Unit 2, Cycle 8 - Reload Report, DPC-RD-2004, Duke Power Company, December 1984.

4. Bounding Analytical Assessment of NUREG 0630 on LOCA and Operating kw/ft Limits, B&W Document No. 77-1141256-00, Babcock & Wilcox.

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6. TACO2 - Fuel Performance Analysis, BAW-10141P-A, Rev. 1, Babcock & Wilcox, June 1983.

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8. Oconee 3 Fuel Densification Report, BAW-1399, Babcock & Wilcox, November 1973.

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

10. Oconee Nuclear Station Reload Design Methodology Technical Report NFS-1001A, Duke Power Company, Charlotte, North Carolina, April 1984.

11. Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143, Part 2, Babcock & Wilcox, Lynchburg, Virginia, March 1980.

12. ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 3, Babcock & Wilcox, July, 1977.

13. Oconee Unit 1, Cycle 9 - Reload Report, BAW-1841, Babcock & Wilcox, August 1984.

14. R. J. Walker (B&W) to K. S. Canady (Duke Power Company), Letter, February 18, 1985.

15. BOL LOCA Limits, B & W Document No. 86-1153360-00, Babcock & Wilcox, April 1985.

Parameters[a]

for Proposed[b] License Amendments Relating to Refueling," June, 1975. The Reload Report employs analytical techniques, and design bases established in reports that were previously submitted and accepted by the USNRC and its predecessor. They are listed in the Reload Report references.

The following evaluation demonstrates by reference to previously performed analysis, that when measured against the three significant safety hazards consideration standards in 10 CFR 50.92, the circumstances of this reload amendment would not involve nor create the conditions described.

First Standard

Involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 8 parameters to determine the effect of the Cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 9 is considered to be bound by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

Create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses performed in support of this reload are in accordance with the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975. The analysis found that the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

Involve a significant reduction in a margin of safety.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations
2. Nuclear Design considerations
3. Thermal-Hydraulic Design considerations

Sections 4, 5, and 6 of the Oconee 3, Cycle 9 Reload Report address the above areas, respectively. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. The evaluations are summarized below: