OCONEE UNIT 3, CYCLE 10

- Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the tenth 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 10 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 9 and 10 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 10 operation. In those cases where Cycle 10 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 10 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 10 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 10, is the currently operating Cycle 9. Cycle 9 achieved initial criticality on October 6, 1985 and power escalation commenced on October 7, 1985. The fuel cycle design length for Cycle 10 - 440 EFPD - is based on a Cycle 9 length of 349 EFPD.

Cycle 10 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 9.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 4, of the FSAR.[1] The Cycle 10 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 10. Nineteen of the Batch 10 assemblies will be discharged at the end of Cycle 9 along with Batch 9B. The remaining 49 Batch 10 assemblies, designated "10B," and the fresh Batch 12 FAs - with initial enrichments of 3.28 and 3.22 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 11, with an initial enrichment of 3.22 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 10.

Cycle 10 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. The Cycle 10 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 10 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1. CORE LOADING DIAGRAM FOR OCONEE 3, CYCLE 10

X (top, column 8) — Y (right, row H) — Z (bottom, column 8)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						L15 10B	P12 10B	O08 11	P04 10B	L01 10B					
B				O06 11	G14 11	B11 11	12	L11 11	12	B05 11	G02 11	O10 11			
C			N05 11	12	N03 11	12	N07 11	K08 11	N09 11	12	N13 11	12	E04 11		
D		F13 11	12	F11 11	12	F07 10B	12	B10 10B	12	F09 10B	12	M10 11	12	F03 11	
E		P07 11	C12 11	12	B06 10B	12	A07 10B	M08 11	A09 10B	12	F14 10B	12	C04 11	P09 11	
F	R10 10B	M02 11	12	G06 10B	12	N11 11	12	A08 10B	12	M04 11	12	G10 10B	12	M14 11	R06 10B
G	N14 10B	12	G12 11	12	G01 10B	12	O13 10B	12	O03 10B	12	G15 10B	12	G04 11	12	N02 10B
H	H13 11	E10 11	H09 11	F02 10B	H11 11	H01 10B	12	H14 10B	12	H15 10B	H05 11	L14 10B	H07 11	M06 11	H03 11
K	D14 10B	12	K12 11	12	K01 10B	12	C13 10B	12	C03 10B	12	K15 10B	12	K04 11	12	D02 10B
L	A10 10B	E02 11	12	K06 10B	12	E12 11	12	R08 10B	12	D05 11	12	K10 10B	12	E14 11	A06 10B
M		B07 11	O12 11	12	L02 10B	12	R07 10B	E08 11	R09 10B	12	P10 10B	12	O04 11	B09 11	
N		L13 11	12	E06 11	12	L07 10B	12	P06 10B	12	L09 10B	12	L05 11	12	L03 11	
O			M12 11	12	D03 11	12	D07 11	G08 11	D09 11	12	D13 11	12	D11 11		
P				C06 11	K14 11	P11 11	12	F05 11	12	P05 11	K02 11	C10 11			
R						F15 10B	B12 10B	C08 11	B04 10B	F01 10B					

XX	PREVIOUS CYCLE LOCATION
X	BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 10

	8	9	10	11	12	13	14	15
H	3.28 26823	3.22 0	3.28 21269	3.22 13426	3.28 24049	3.22 11694	3.22 13427	3.22 13649
K	3.22 0	3.28 22055	3.22 0	3.28 21556	3.22 0	3.22 13866	3.22 0	3.28 16637
L	3.28 21288	3.22 0	3.22 13776	3.22 0	3.28 24107	3.22 0	3.22 9553	3.28 20295
M	3.22 13423	3.28 21557	3.22 0	3.28 24056	3.22 0	3.22 11241	3.22 11838	
N	3.28 24049	3.22 0	3.28 24090	3.22 0	3.22 13427	3.22 0	3.22 13352	
O	3.22 11690	3.22 13886	3.22 0	3.22 11250	3.22 0	3.22 13761		
P	3.22 13427	3.22 0	3.22 9548	3.22 11839	3.22 13347			
R	3.22 13653	3.28 16640	3.28 20272					

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

FIGURE 3.3 CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 10

X

	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		— Y
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																

Z

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 10

	8	9	10	11	12	13	14	15
H		1.1						
K	1.1		1.1		1.1		0.2	
L		1.1		1.1		0.8		
M			1.1		1.1			
N		1.1		1.1		0.2		
O			0.8		0.2			
P		0.2						
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 10, 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 12 Mark B5Z fuel assemblies are a Mark B5 design with zircaloy intermediate spacer grids. The B5Z design has been previously demonstrated in the Oconee 3 Cycle 9 reload (Reference 5). There are no new features not previously demonstrated. Additionally, the Batch 12 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[5]. All 60 BPRAs will be inserted into Batch 12 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 10 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 10B is more limiting than other batches due to its longer previous incore exposure time. The Batch 10B assembly power histories were assessed against Duke's generic creep collapse analysis which is based on 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 32,900 EFPH, which is greater than the maximum projected residence time of Cycle 10 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 10 core is thermally similar. The fresh Batch 12 fuel inserted for Cycle 10 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 41,158 MWd/MtU and the maximum fuel rod burnup is predicted to be 42,802 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

The Batch 12 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 12 fuel assemblies is identical to those of the present fuel.

Table 4-1.

Fuel Design Parameters and Dimensions

	Batch No.		
	10B	11	12
FA type	Mark B5	Mark B5Z	Mark B5Z
No. of FAs	49	68	60
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	3.28	3.22	3.22
Est. residence time, EOC 10, EFPH	28,776	19,176	10,800
Cladding collapse time, EFPH	>32,900	>27,400	>27,400

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.		
	10B	11	12
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 10.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycles 9 and 10; the values for Cycles 9 and 10 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 tenth cycle at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 9 and 10 are the number of feed assemblies and the different shuffle patterns. 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 10 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 9 Reload Report.[5]

5.2 Analytical Input

The Cycle 10 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 10 as for the reference cycle.

5.3 Changes in Nuclear Design

There are no changes in design methodology between Oconee 3 Cycle 10 and Oconee 3 Cycle 9.

Table 5-1. Oconee 3 Physics Parameters[a]

	Cycle 9[b]	Cycle 10[c]
Cycle length, EFPD	400	440
Cycle burnup, MWd/mtU	12,349	13,763
Average core burnup, EOC, MWd/mtU	23,035	24,631
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, Groups 7 and 8 at nominal positions[d]	1577	1752
HFP, Groups 7 and 8 at nominal positions	1395	1516
Critical boron - EOC (equilibrium xenon), ppm		
HZP, Groups 7 and 8 at nominal positions	402	430
HFP, Groups 7 and 8 at nominal positions	64	57
Control rod worths - HFP, BOC, % Δk/k		
Group 7	1.19	0.96
Group 8 (e)	0.16	0.16
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.24	1.08
Group 8 (e)	0.16	0.22
Max ejected rod worth - HZP, % Δk/k		
BOC, (N12) groups 5-8 inserted	0.69	0.40
EOC, (N12) groups 5-8 inserted	0.51	0.48
Max stuck rod worth - HZP, % Δk/k		
BOC (N12)	1.63	1.55
EOC (N12)	1.92	2.04
Power deficit, HFP to HZP, % Δk/k		
BOC	1.77	1.70
EOC	3.04	3.16
Doppler coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (equilibrium xenon)	-1.31	-1.35
EOC (equilibrium xenon)	-1.63	-1.73

Table 5-1. (Cont'd)

	Cycle 9[b]	Cycle 10[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (equilibrium xenon)	-0.58	-0.47
EOC (equilibrium xenon)	-2.91	-3.30
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	123	124
EOC	110	110
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.43	2.43
EOC (equilibrium)	2.65	2.58
Effective delayed neutron fraction - HFP		
BOC	0.00618	0.00617
EOC	0.00523	0.00518

[a] Cycle 10 data are for the conditions stated in this report. The Cycle 9 core conditions are identified in Reference 5.
[b] Based on a 400-EFPD Cycle 8. (Actual Cycle 8 length 396.59 EFPD).
[c] Based on a Cycle 9 length of 349-EFPD.
[d] Nominal positions are as follows:

	Cycle 9	Cycle 10
HZP (BOC)	Group 7 at 100% WD, 8 at 25.5% WD	Group 7 at 100% WD, 8 at 25.0% WD
HFP (BOC)	Group 7 at 92% WD, 8 at 15% WD	Group 7 at 92% WD, 8 at 35% WD
HZP (EOC)	Group 7 at 100% WD, 8 at 25.5% WD	Group 7 at 100% WD, 8 at 25.0% WD
HFP (EOC)	Group 7 at 100% WD, 8 at 15% WD	Group 7 at 100% WD, 8 at 35% WD

(e) (15% to 100% WD for Cycle 9, 35% to 100% WD for Cycle 10)

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 10

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.80	9.47
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.55	-2.04
Net worth	6.83	7.01
Less 10% uncertainty	-0.68	-0.70
Total available worth	6.15	6.31
Required Rod Worth		
Power deficit, HFP to HZP	1.70	3.16
Max inserted rod worth, HFP	0.35	0.53
Total required worth	2.05	3.69
Shutdown Margin		
Total available worth minus total required worth	4.10	2.62

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

FIGURE 5-1
OCONEE 3, CYCLE 10
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.986	1.198	1.089	1.105	0.996	1.218	1.092	0.572
K	1.198	1.100	1.264	1.081	1.203	1.238	1.233	0.530
L	1.089	1.264	1.274	1.221	0.986	1.264	0.980	0.365
M	1.105	1.081	1.221	1.026	1.185	1.104	0.677	
N	0.996	1.203	0.986	1.185	1.133	1.051	0.446	
O	1.218	1.238	1.264	1.104	1.051	0.547		
P	1.092	1.233	0.980	0.677	0.446			
R	0.572	0.530	0.365					

6. THERMAL-HYDRAULIC DESIGN

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

The Cycle 10 transition core will include 60 fresh Mark-BZ Batch 12 fuel assemblies, all 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 10 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 and Mark-B assemblies in a 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 10 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 10 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 9	Cycle 10
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/BWC	BAW-2/BWC
Min DNBR with densification penalty	2.05/>1.74	>2.05/>1.74
Hot channel factors: Enthalpy rise	1.011/1.011	1.011/1.011
Heat flux	1.014/1.014	1.014/1.014
Flow area	0.98/0.97	0.98/0.97

[a] Generic analyses based on $\geq$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 9 parameters to determine the effect of the Cycle 10 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 12 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 10 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 10 are given in Table 4-2. Table 6-1 compares the Cycle 9 and 10 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 10. The effect of a more negative hot full power end-of-cycle moderator temperature coefficient on the FSAR accident analyses has been analyzed for Oconee Nuclear Station.[16] Table 7-1 has been revised to include the new values for end-of-cycle moderator temperature coefficient and dropped rod worth assumed in these analyses.

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 12 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 10 fuel.

The LOCA kW/ft limits have been reduced for the first 65 EFPDs. This 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 7-2. Table 7-3 shows the bounding values for allowable LHRs for Oconee 3 Cycle 10 fuel after 65 EFPD.

From the examination of Cycle 10 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 10. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 10 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in Cycle 10 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 10 value
BOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.17	-1.35
EOC Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.33[(a)]	-1.73
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[(b)]	-0.47
EOC moderator coeff, HZP, 10^{-4}, $\Delta k/k/°F$	-3.0[(c)]	-2.73
, HFP, 10^{-4}, $\Delta k/k/°F$	-3.5[(c)]	-3.30
All rod bank worth, HZP, % $\Delta k/k$	10.0	9.47
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	88
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.25
Dropped rod worth, HFP, % $\Delta k/k$	0.40	0.13
Initial boron conc, HFP, ppm	1400	1516[(d)]

(a) -1.2×10^{-5} $\Delta k/k/F$ was used for steam line break analysis.
-1.3×10^{-5} $\Delta k/k/F$ was used for cold water accident (pump start-up).

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

(c) The HZP moderator temperature coefficient is one of the key parameters assumed in the steam line break analysis. The HFP moderator temperature coefficient is included since it is one of the parameters assumed in the cold water, rod ejection, and control rod misalignment accident analyses, although none of these accidents are very sensitive to changes in the coefficient.

(d) The combined effect of boron concentration and boron worth is conservative for Cycle 10.

Table 7-2. LOCA Limits, Oconee 3, Cycle 10

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

Table 7-3. LOCA Limits, Oconee 3, Cycle 10, 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

(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 10 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.

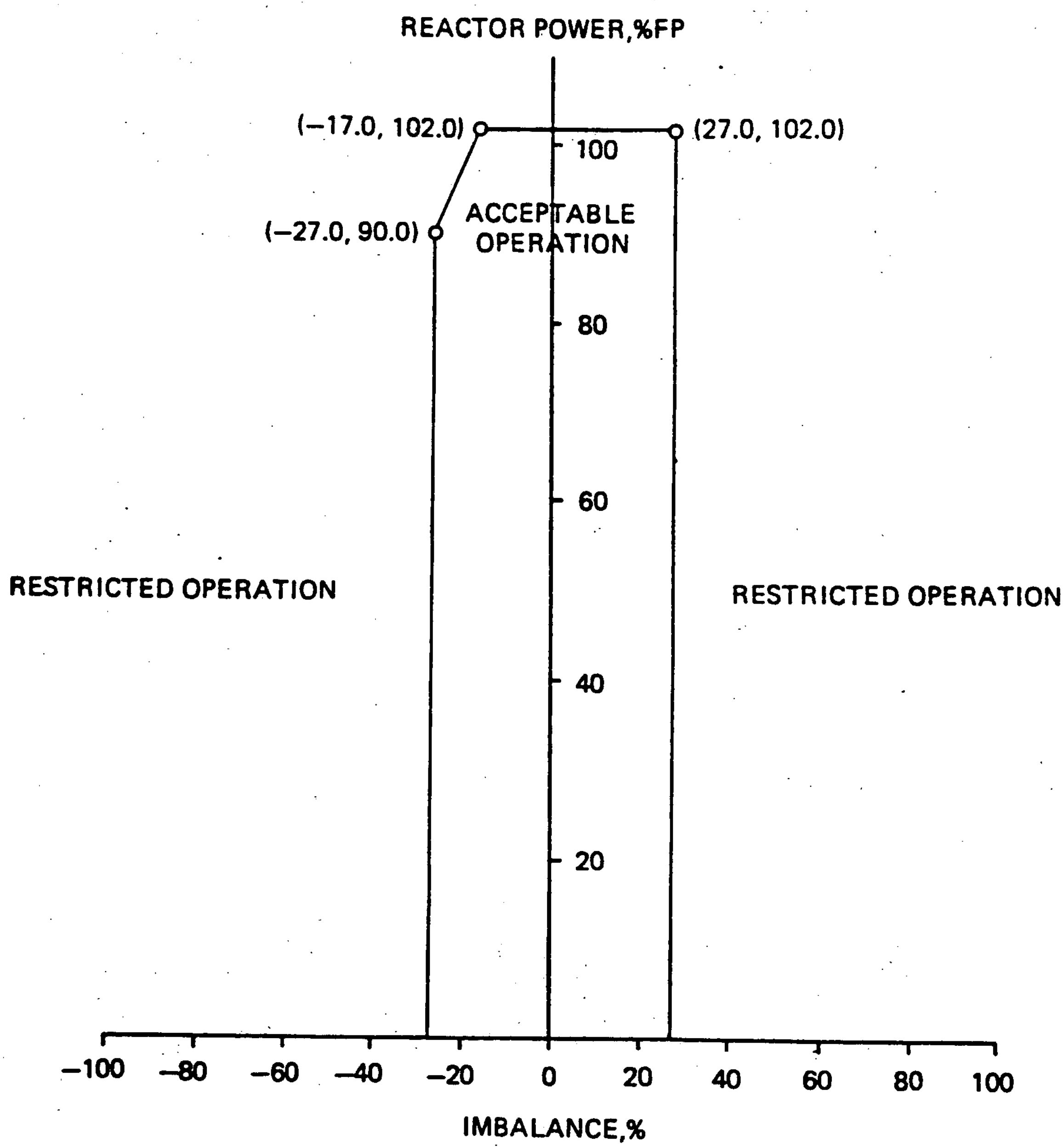
2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 10. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 10 operation.

3. The minimum required boron concentration in the BWST has been raised from 1835 ppm to 2010 pppm in order to ensure that the core is at least 1% $\Delta k/k$ subcritical at 70°F without any control rods in the core.

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. Figure 8-1 is a revision to a previous Technical Specification limit.

Figure 8-1

Operational Power-Imbalance Limits, 0 EFPD to EOC



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