OCONEE UNIT 2, 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 2, 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 2, 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 2 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 2, Cycle 9, is the currently operating Cycle 8. Cycle 8 achieved initial criticality on April 20, 1985 and power escalation commenced on April 21, 1985. The fuel cycle design length for Cycle 9 - 400 EFPD - is based on Cycle 8 length of 421 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 2 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 2, Cycle 9. Twenty-three of the Batch 9 assemblies will be discharged at the end of Cycle 8 along with Batch 8B. The remaining 49 Batch 9 assemblies, designated "9B," and the fresh Batch 11 FAs - with initial enrichments of 3.24 and 3.22 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 10, with an initial enrichment of 3.22 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 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 2, CYCLE 9

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						L15 9B	P12 9B	O08 10	P04 9B	L01 9B					
B				O06 10	G14 10	B11 10	11	L11 10	11	B05 10	G02 10	O10 10			
C			N05 10	11	N03 10	11	N07 10	K08 10	N09 10	11	N13 10	11	E04 10		
D		F13 10	11	F11 10	11	F07 9B	11	M08 10	11	F09 9B	11	M10 10	11	F03 10	
E		P07 10	C12 10	11	B06 9B	11	A07 9B	B10 9B	A09 9B	11	F14 9B	11	C04 10	P09 10	
F	R10 9B	M02 10	11	G06 9B	11	N11 10	11	A08 9B	11	M04 10	11	G10 9B	11	M14 10	R06 9B
G	N14 9B	11	G12 10	11	G01 9B	11	O13 9B	11	O03 9B	11	G15 9B	11	G04 10	11	N02 9B
HW —	H13 10	E10 10	H09 10	H11 10	F02 9B	H01 9B	11	O11 9B	11	H15 9B	L14 9B	H05 10	H07 10	M06 10	H03 10 — Y
K	D14 9B	11	K12 10	11	K01 9B	11	C13 9B	11	C03 9B	11	K15 10	11	K04 10	11	D02 9B
L	A10 9B	E02 10	11	K06 9B	11	E12 10	11	R08 9B	11	D05 10	11	K10 9B	11	E14 10	A06 9B
M		B07 10	O12 10	11	L02 9B	11	R07 9B	P06 9B	R09 9B	11	P10 9B	11	O04 10	B09 10	
N		L13 10	11	E06 10	11	L07 9B	11	E08 10	11	L09 9B	11	L05 10	11	L03 10	
O			M12 10	11	D03 10	11	D07 10	G08 10	D09 10	11	D13 10	11	D11 10		
P				C06 10	K14 10	P11 10	11	F05 10	11	P05 10	K02 10	C10 10			
R						F15 9B	B12 9B	C08 10	B04 9B	F01 9B					

Z

XX	PREVIOUS CYCLE LOCATION
X	BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 2, CYCLE 9

	8	9	10	11	12	13	14	15
H	3.24 28799	3.22 0	3.24 23648	3.24 23244	3.22 16198	3.22 15443	3.22 15921	3.22 16648
K	3.22 0	3.24 23361	3.22 0	3.24 23196	3.22 0	3.22 16858	3.22 0	3.24 21933
L	3.24 23614	3.22 0	3.22 16175	3.22 0	3.24 27501	3.22 0	3.22 11159	3.24 21522
M	3.24 23244	3.24 23155	3.22 0	3.24 23251	3.22 0	3.22 12842	3.22 13861	
N	3.22 16217	3.22 0	3.24 27552	3.22 0	3.22 15930	3.22 0	3.22 16204	
O	3.22 15515	3.22 16844	3.22 0	3.22 12836	3.22 0	3.22 16163		
P	3.22 15921	3.22 0	3.22 11149	3.22 13885	3.22 16187			
R	3.22 16671	3.24 21885	3.24 21590					

S_L

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

FIGURE 3.3 CONTROL ROD LOCATIONS FOR OCONEE 2, CYCLE 9

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 2, CYCLE 9

	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 2, Cycle 9, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. The Batch 11 fuel assemblies are of the Mark B5Z design[18]. Retainers will be used on two Batch 10 fuel assemblies that contain regenerative neutron sources (RNS). The justification for the design and use of the RNS retainers is described in References 3 and 4. These retainers will be exposed for a fourth cycle of irradiation during Cycle 9, as justified in Reference 15.

Cycle 9 retains one Advanced Cladding Pathfinder (ACP) assembly from Cycle 8. This assembly is a reconstitutable design with 12 special advanced cladding rods that were previously described in Reference 5. Cycle 9 will be the third cycle of exposure for the ACP assembly.

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). The creep collapse analysis of the beta-quenched fuel rods contained in the ACP assembly has been revised from Reference 5 based on creep tests performed on this cladding.

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. 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 computer code. The nominal rod design parameters for each batch of fuel 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 41,830 MWD/MTU and the maximum fuel rod burnup is predicted to be 43,492 MWD/MTU. Fuel rod internal pressure has been evaluated using TACO2 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 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(a)	10	11
FA type	Mark B4	Mark B4Z	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	3.24	3.22	3.22
Est. residence time, EOC 9, EFPH	30,288	19,944	9,840
Cladding collapse time, EFPH	>31,400	30,800	30,800

(a) All Batch 9B values are acceptable for the one ACP assembly.

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No. 9B	Batch No. 10	Batch No. 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[c]	20.15	20.15
Linear heat rate capability[b] >1000 MWD/MTU, kW/ft	21.20[c]	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.

(c) ACP rods have been assessed against these limits and found to be bounded.

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 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 the Core Operational Limits Report for Oconee 2 Cycle 9. 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 2, 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

There are no changes in design methodology between Oconee 2 Cycle 9 and Oconee 2 Cycle 8.

Table 5-1. Oconee 2 Physics Parameters[a]

	Cycle 8[b]	Cycle 9[c]
Cycle length, EFPD	400	400
Cycle burnup, MWd/mtU	12,346	12,512
Average core burnup, EOC, MWd/mtU	22,789	25,063
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, Groups 7 and 8 at nominal positions[d]	1575	1570
HFP, Groups 7 and 8 at nominal positions	1394	1311
Critical boron - EOC (equilibrium xenon), ppm		
HZP, Groups 7 and 8 at nominal positions	430	395
HFP, Groups 7 and 8 at nominal positions	81	31
Control rod worths - HFP, BOC, % Δk/k		
Group 7	1.08	1.04
Group 8 (e)	0.18	0.17
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.17	1.12
Group 8 (e)	0.23	0.21
Max ejected rod worth - HZP, % Δk/k		
BOC, (N12) Groups 5-8 inserted	0.29	0.51
EOC, (N12) Groups 5-8 inserted	0.42	0.46
Max stuck rod worth - HZP, % Δk/k		
BOC (N12)	1.51	1.70
EOC (N12)	1.82	2.05
Power deficit, HFP to HZP, % Δk/k		
BOC	1.74	1.89
EOC	3.06	3.19
Doppler coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (equilibrium xenon)	-1.32	-1.39
EOC (equilibrium xenon)	-1.63	-1.74

Table 5-1. (Cont'd)

	Cycle 8[b]	Cycle 9[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (no xenon)	-0.53	-0.71
EOC (equilibrium xenon)	-2.90	-3.28
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	123	122
EOC	111	109
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.42	2.45
EOC (equilibrium)	2.63	2.59
Effective delayed neutron fraction - HFP		
BOC	0.00619	0.00611
EOC	0.00523	0.00519

(a) Cycle 9 data are for the conditions stated in this report. The Cycle 7 core conditions are identified in Reference 5.

(b) Based on a 421-EFPD Cycle 7. (Actual Cycle 7 length 430.9 EFPD).

(c) Based on a Cycle 8 length of 421-EFPD.

(d) Nominal positions are as follows:

	Cycle 8	Cycle 9
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 35% WD	Group 7 at 92% WD, 8 at 25% 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 35% WD	Group 7 at 100% WD, 8 at 25% WD

(e) (35% to 100% WD for Cycle 8, 35% to 100% WD for Cycle 9)

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

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.94	9.60
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.70	-2.05
Net worth	6.82	7.13
Less 10% uncertainty	-0.68	-0.71
Total available worth	6.14	6.42
Required Rod Worth		
Power deficit, HFP to HZP	1.89	3.19
Max inserted rod worth, HFP	0.37	0.55
Total required worth	2.26	3.74
Shutdown Margin		
Total available worth minus total required worth	3.88	2.68

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

FIGURE 5-1
OCONEE 2, CYCLE 9
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.968	1.207	1.039	0.949	1.140	1.167	1.043	0.528
K	1.207	1.096	1.270	1.050	1.248	1.206	1.228	0.478
L	1.039	1.271	1.267	1.251	0.964	1.287	0.974	0.351
M	0.949	1.050	1.251	1.083	1.234	1.125	0.685	
N	1.141	1.248	0.964	1.234	1.149	1.097	0.445	
O	1.167	1.206	1.288	1.125	1.097	0.555		
P	1.043	1.228	0.974	0.685	0.445			
R	0.528	0.478	0.351					

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 60 fresh Mark-BZ Batch 11 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 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 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 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.

For Cycle 9 operation a flux to flow setpoint of 1.07 is maintained. 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 setpoint analysis 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]	7.8	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 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 15 are characteristic for Oconee 2 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. 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.[19] 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[12,13] 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[16]. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 2, Cycle 9 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[14,17]. The limits for the first 65 EFPD are shown in Table 7-2. Table 7-3 shows the bounding values for allowable LOCA peak LHRs for Oconee 2 Cycle 9 fuel after 65 EFPD.

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 2 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}, $\Delta k/k/°F$	-1.17	-1.39
EOC Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.33[(a)]	-1.74
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[(b)]	-0.71
EOC moderator coeff, HZP, 10^{-4}, $\Delta k/k/°F$	-3.0[(c)]	-2.72
HFP, 10^{-4}, $\Delta k/k/°F$	-3.5[(c)]	-3.28
All rod bank worth, HZP, % $\Delta k/k$	10.0	9.60
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	87
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	1311

(a) -1.2×10^{-5} $\Delta k/k/F$ was used for the steam line break analysis.
-1.3×10^{-5} $\Delta k/k/F$ was used for the 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.

Table 7-2. LOCA Limits, Oconee 2, 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

Table 7-3. LOCA Limits, Oconee 2, 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

(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

A review of pertinent reload dependent parameters (as described in Reference 10) has indicated that no Technical Specification revisions are necessary to support operation of Oconee 2, Cycle 9. The figures for the operating limits on rod index and axial power imbalance have been removed from Section 3 of the Technical Specifications and will be included in the cycle specific Core Operational Limits Report.

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. These operational limits are provided in the Oconee 2 Cycle 9 Core Operational Limits Report.
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.

In accordance with the guidelines set forth in 10CFR50.59, a safety evaluation has been performed and shown that there are no unreviewed safety questions associated with the operation of the ninth cycle of Oconee Unit 2.

REFERENCES

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

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19. H. B. Tucker, (Duke Power Company) to J. F. Stolz (NRC), Letter, April 23, 1986.