OCONEE UNIT 2, CYCLE 7

\- Reload Report -

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the seventh 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 7 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 6 and 7 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 7 operation. In those cases where cycle 7 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 7 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 7 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 7, is the currently operating cycle 6. Cycle 6 achieved initial criticality on May 17, 1982 and power escalation commenced on May 18, 1982. The fuel cycle design length for cycle 7 - 421 EFPD - is based on cycle 6 length of 400 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in cycle 7.

Cycle 7 will operate in a feed-and-bleed mode for its entire design length, as did cycle 6.

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 7 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 7. Thirty-five of the batch 7 assemblies will be discharged at the end of cycle 6 along with batch 6B. The remaining 32 batch 7 assemblies, designated "7C," and the fresh batch 9 FAs - with initial enrichments of 3.07 and 3.24 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 8, with an initial enrichment of 3.17 wt % ^{235}U, will occupy primarily the core periphery. The remaining assembly is a reinserted batch 6 assembly, discharged after cycle 5, at location H-08 with an initial enrichment of 2.91 wt % ^{235}U. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 7.

Cycle 7 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 64 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 7 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The cycle 7 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3–1. CORE LOADING DIAGRAM FOR OCONEE 2 CYCLE 7

X (axis above column 8); Y (axis right of row H); Z (axis below column 8)

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

XX	PREVIOUS CYCLE LOCATION
X	BATCH NO.

FIGURE 3–2. ENRICHMENT AND BURNUP DISTRIBUTION FOR OCONEE 2 CYCLE 7

	8	9	10	11	12	13	14	15
H	2.91 20192	3.17 16651	3.07 19788	3.24 0	3.07 21624	3.24 0	3.17 15729	3.17 10167
K	3.17 16769	3.17 16175	3.24 0	3.07 21397	3.24 0	3.17 15918	3.24 0	3.17 12538
L	3.07 19788	3.24 0	3.07 19796	3.24 0	3.07 22326	3.24 0	3.17 16325	3.17 14961
M	3.24 0	3.07 21431	3.24 0	3.07 16670	3.24 0	3.17 15590	3.24 0	
N	3.07 21592	3.24 0	3.07 22350	3.24 0	3.17 16149	3.24 0	3.17 13225	
O	3.24 0	3.17 16094	3.24 0	3.17 15597	3.24 0	3.17 10156		
P	3.17 15724	3.24 0	3.17 16363	3.24 0	3.17 13214			
R	3.17 10167	3.17 12588	3.17 14962					

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

FIGURE 3–3. CONTROL ROD LOCATION FOR OCONEE 2 CYCLE 7

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

X — Y — Z

$\boxed{X}$ GROUP NO.

GROUP	NO. of RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	12	CONTROL
8	8	APSRs
TOTAL	69	

FIGURE 3–4. BPRA ENRICHMENT AND DISTRIBUTION FOR OCONEE 2 CYCLE 7

	8	9	10	11	12	13	14	15
H				1.1		1.1		
K			1.1		1.1		0.2	
L		1.1		1.1		0.8		
M	1.1		1.1		1.1			
N		1.1		1.1		0.2		
O	1.1		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 7, 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 Mk B4 fuel assemblies. Retainers will be used on two batch 8 fuel assemblies that contain regenerative neutron sources (RNS), and on 64 batch 9 assemblies containing BPRA's. The justification for the design and use of the BPRA retainers is described in reference 3 and 21, which is also applicable to the RNS retainers of Oconee 2, Cycle 7.

Batch 9 contains one Advanced Cladding Pathfinder (ACP) assembly. This assembly is a reconstitutable design with 12 special advanced cladding rods. The ACP assembly is designed to be reconstitutable to allow future removal of selected rods for examination. The assembly reconstitutable features are designed so that reactor safety and performance are not adversely affected.

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 7 design. All methods are consistent with the approved methodologies of reference 16 except where specifically stated.

4.1.1 Advanced Cladding Irradiation Program

The ACP fuel rod design is identical to the standard MK-B design. Six zirconium lined tubes and six beta quenched tubes will be used for 12 test rods. These tube modifications are expected to provide improved resistance to water-side corrosion and/or pellet-cladding interaction.

In assessing the capabilities of these rods, all generic calculations were reviewed. In the areas of Creep Collapse and Strain Analysis, the irradiation and thermal creep term was doubled for the Beta-Quenched cladding. This fuel rod was analyzed using TACO2[22], to assess the necessary changes in the creep

expression. In the case of the Zirconium-lined cladding, the liner-thickness was removed for the evaluation of Creep Collapse. Additional calculations for Linear Heat Rate to Melt and Pin pressure were made which identified that these special rods are bounded by the Batch 9 fuel design limits.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of batch 7C is more limiting than other batches due to its longer previous incore exposure time. The batch 7C 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 TACO[4] code was used to calculate internal pin pressures and clad temperatures used as input to CROV. The collapse time for the most limiting assembly was conservatively determined to be 33,600 EFPH, which is greater than the maximum projected residence time of Cycle 7 fuel (Table 4-1).

4.2.2 Cladding Stress

Duke has performed a generic and conservative fuel rod cladding stress analysis. This analysis is consistent with the methodology described in Reference 16 with the following exception: the fuel rod total stress (primary plus secondary) was permitted to exceed the unirradiated yield strength. Two times the minimum unirradiated yield strength (2.0 Sy) has been used as a criterion for the total stress calculation, as permitted by Section III, Article NB-3000 of the ASME Boiler and Pressure Vessel Code. Approximately 0.35 Sy margin remains in this total stress calculation.

Primary membrane plus primary bending stresses are limited to 1.0 Sy, and primary membrane stress is limited to 2/3 Sy. Substantial margin exists in both of these evaluations.

The following conservatisms exist in the generic cladding stress calculation:

- Specification cladding dimensions which result in highest stress
- A low internal pressure (HZP);
- A high external pressure (110 percent of design pressure);

- A large through wall cladding temperature gradient (fuel melt conditions), and
- BOL grid loads for worst grid cell type (based on as-built cladding diameter and spacer grid cell size)

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO in accordance with the approved methodology[16]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3 Thermal Design

All fuel in the Cycle 7 core is thermally similar. The fresh batch 9 fuel inserted for Cycle 7 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 TACO computer code. The individual fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The input shown includes the following conservatisms:

1. Lower Tolerance Limit (LTL) initial density
2. LTL initial pellet diameter.
3. A maximum gap based on as-fabricated pellet and cladding data.
4. 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 35,882 MWD/MTU and the maximum fuel rod burnup is predicted to be 40,225 MWD/MTU. Fuel rod internal pressure has been evaluated using TACO 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 9 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 9 fuel assemblies is identical to those of the present fuel.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch No.			
	6C	7C	8	9[a]
FA type	Mark B4	Mark B4	Mark B4	Mark B4
No. of FAs	1	32	72	72
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4
Undensified active fuel length, in.	142.2	142.2	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3695	0.3695	0.3686	0.3686
Fuel pellet initial density (mean spec), %TD	94.0	94.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	2.91	3.07	3.17	3.24
Est residence time, EOC 7, EFPH	28,176	29,304	19,704	10,104
Cladding collapse time, EFPH	33,600	33,600	33,600	33,600

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

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.			
	6C	7C	8	9
Initial density, % TD	93.50(a)	93.55	94.52	94.63
Max. In-reactor densification, % TD	2.39(a)	2.22	2.17	1.73
Burnup corresponding to max. densification, MWd/mtU	3041.(a)	3447.	3704.	3665.
Initial pellet diameter, in.	0.3692(b)	0.3691	0.3682	0.3684
Initial clad ID, in.	0.3772(b)	0.3772	-0.3773	0.3776
Initial clad OD, in.	0.4302(b)	0.4302	0.4303	0.4306
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.73	5.73	5.74	5.74
Linear heat rate capability(d) from 0-1000 MWD/MTU, kW/ft	20.4	20.4	20.4	20.4(e)
Linear heat rate capability(d) >1000 MWD/MTU, kW/ft	21.2	21.2	21.2	21.2(e)
Average fuel temp. @ nominal linear heat rate, °F	1250(c)	1250	1240	1240

(a) Basis: Batch specific pellet resinter data

(b) Basis: Pellet and cladding as-fabricated dimensions (95/95 tolerances)

(c) Basis: TACO, 96.5% TD @ 4000 MWD/mtU, nominal pellet and cladding dimensions

(d) These values are utilized as fuel design limits for Cycle 7.

(e) All special rods have been assessed against these limits and found to be bounding.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design cycles 6 and 7; the values for cycle 6 were generated by B&W [6, 7, 8, 13, 15] using PDQ07 while the values for cycle 7 were generated by Duke Power Company using methods described in Reference 16. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The longer cycle 7 will produce a higher cycle burnup than that for the design cycle 6. Figure 5-1 illustrates a representative relative power distribution for the beginning of the seventh cycle at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between cycles 6 and 7 are the longer cycle 7 design length, different BPRA loadings, and different shuffle patterns. The control rod worths differ between cycles because of changes in the radial flux and burnup distributions. This also accounts for differences in ejected and stuck rod worths. 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 7 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 6 reload report.[5]

5.2 Analytical Input

The cycle 7 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for cycle 7 as for the reference cycle, however, CASMO[17] was used to derive the F-factors.

5.3 Changes in Nuclear Design

There is only one significant core design change between the reference cycle and the reload cycle. Duke Power calculational methods[16] are used to obtain the important nuclear design parameters for this cycle.

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

	Cycle 6[b]	Cycle 7[c]
Cycle length, EFPD	400	421
Cycle burnup, MWd/mtU	12,518	12,961
Average core burnup, EOC, MWd/mtU	21,771	22,743
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, group 7 at 100% WD, 8 at 37.5% WD	1547	1552
HFP, group 7 at 87% WD, 8 at 25% WD	1338	1363
Critical boron - EOC (equilibrium xenon), ppm		
HZP, group 7 at 100% WD, 8 at 37.5% WD	394	376
HFP, group 7 at 100% WD, 8 at 25% WD	69	9
Control rod worths - HFP, BOC, % $\Delta k/k$		
Group 7	1.46	1.51
Group 8 (25% to 100% WD)	0.40	0.32
Control rod worths - HFP, EOC, % $\Delta k/k$		
Group 7	1.53	1.64
Group 8 (25% to 100% WD)	0.49	0.28
Max ejected rod worth - HZP, % $\Delta k/k$		
BOC, (N12) groups 5-8 inserted	0.63	0.59
EOC, (N12) groups 5-8 inserted	0.57	0.58
Max stuck rod worth - HZP, % $\Delta k/k$		
BOC (N12)	1.78	1.43
EOC (N12)	1.80	1.85
Power deficit, HFP to HZP, % $\Delta k/k$		
BOC	1.56	1.78
EOC	2.38	3.10
Doppler coeff - BOC, 10^{-5} ($\Delta k/k$-°F)		
100% power (equilibrium xenon)	-1.51	-1.35
Doppler coeff - EOC, 10^{-5} ($\Delta k/k$-°F)		
100% power (equilibrium xenon)	-1.77	-1.67

Table 5-1. (Cont'd)

	Cycle 6[b]	Cycle 7[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (equilibrium xenon)	-0.65	-1.15
EOC (equilibrium xenon)	-2.97	-2.92
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC (979 ppm)	124	122
EOC (50 ppm)	107	109
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.56	2.47
EOC (equilibrium)	2.70	2.68
Effective delayed neutron fraction - HFP		
BOC	0.00634	0.00622
EOC	0.00527	0.00522

[a] Cycle 7 data are for the conditions stated in this report. The cycle 6 core conditions are identified in reference 5.

[b] Based on a 390-EFPD cycle 5. (Actual cycle 5 length 400 EFPD).

[c] Based on a cycle 6 length of 400-EFPD.

Table 5-2. Shutdown Margin Calculation for Oconee 2, Cycle 7

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.40	9.35
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.43	-1.85
Net worth	6.55	7.08
Less 10% uncertainty	-0.66	-0.71
Total available worth	5.89	6.37
Required Rod Worth		
Power deficit, HFP to HZP	1.78	3.10
Max inserted rod worth, HFP	0.23	0.59
Total required worth	2.01	3.69
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 7
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.879	1.009	1.011	1.239	1.056	1.312	1.030	0.614
K	1.007	1.092	1.251	1.039	1.249	1.182	1.219	0.580
L	1.010	1.250	1.066	1.231	0.933	1.232	0.896	0.409
M	1.237	1.038	1.230	1.108	1.226	1.055	0.874	
N	1.054	1.246	0.931	1.225	1.067	1.068	0.491	
O	1.308	1.177	1.229	1.054	1.067	0.595		
P	1.026	1.215	0.894	0.873	0.491			
R	0.612	0.579	0.408					

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 9 fuel and the one reinserted batch 6C fuel assembly are hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The generic thermal-hydraulic design analysis supporting cycle 7 operation was performed by Duke Power Company and employed the methods and models described in references 1, 5, 9 and 16.

The maximum core bypass flow for cycle 6 was 7.6% of the total system flow. For cycle 7 operation, 64 BPRAs will be inserted, and two assemblies will contain regenerative neutron sources, leaving 42 FAs with open guide tubes. This results in a core bypass flow of 7.8% of the total system flow. The bypass flow is less than that assumed in the generic thermal-hydraulic design analysis and the consequent increase in core flow establishes the generic analysis as conservative for cycle 7 operation. The cycle 6 and 7 maximum design conditions are summarized in Table 6-1.

A net rod bow DNBR penalty of 0.0% was calculated for cycle 7, taking credit for the flow area reduction hot channel factor used in all DNBR calculations. Based on the procedure approved in reference 10, the penalty is based on the maximum assembly burnup for the fuel batch containing the limiting (highest power) assembly. For cycle 7, the DNBR penalty was based on the maximum batch 9 assembly burnup, 16,819 MWd/mtU.

An analysis has been performed to conservatively determine the minimum allowable reduction in pin peak as a function of burnup required to offset rod bow DNBR penalty, reference 18. The result was used to demonstrate that the increase in DNBR associated with the lower pin peaks (relative to the limiting batch 9 pin peak) for the limiting batch 6, 7, and 8 assemblies, more than offsets the increased rod bow DNBR penalty that would be calculated for the higher assembly burnups of batch 6, 7, or 8 fuel.

For cycle 7 operation, a flux to flow setpoint of 1.07 is established. The minimum DNBR value determined for the flux to flow setpoint analysis is greater than the BAW-2 CHF correlation limit of 1.30. All other plant operating limits based on a minimum DNBR limit include a minimum of 10.2% DNBR margin from the design limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 6	Cycle 7
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	7.6	7.8
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
Hot channel factors: Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	(a)	(a)
Avg heat flux at 100% power, 10^3 Btu/h-ft^2 [a]	176[b]	176[b]
CHF correlation	BAW-2	BAW-2
Min DNBR with densification penalty	2.05	>2.05

[a] See Table 4-1.

[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 6 parameters to determine the effect of the cycle 7 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 9. Since batch 9 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 9, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in reference 20 are characteristic for Oconee 2 cycle 7 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 7 are given in Table 4-2. Table 6-1 compares the cycle 6 and 7 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and cycle 7.

A generic LOCA analysis for the B&W 177-FA, lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103, Rev. 3.[11] 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[11,12] is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 2, cycle 7 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 2 cycle 7 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[19]. The limits for the first 65 EFPD are shown in Table 7-3.

From the examination of cycle 7 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 7. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 7 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 7 are bounded by the FSAR and/or the fuel densification report.[9]

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted cycle 7 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]	-1.67
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[b]	-1.15
EOC moderator coeff, 10^{-4}, $\Delta k/k/°F$	-3.0	-2.92
All rod bank worth, HZP, % $\Delta k/k$	10.0	9.35
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	87
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.19
Dropped rod worth, HFP, % $\Delta k/k$	0.46	0.12
Initial boron conc, HFP, ppm	1400	1363

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

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

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	17.5	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 7 operation in accordance with the methods of reference 16 to account for minor changes in power peaking and control rod worths inherent with an extended, lumped burnable poison cycle. Cycle 6 Technical Specifications were generated in accordance with the methods described in Reference 14.

In addition:

1. The Reactor Protective System (RPS) instrumentation string errors associated with the high flux and power-imbalance trips have been reanalyzed using a Monte-Carlo simulation methodology. The resulting string errors were used to establish the high flux trip setpoint at 105.5% of rated power and the power-imbalance maximum allowable setpoints given in Figure 8-4.

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

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 8-16 are revisions to previous Technical Specification limits.

REFERENCES

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21. J. H. Taylor (B&W) to S. A. Varga (NRC), Letter, "BPRA Retainer Reinsertion," January 14, 1980.

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