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May 1979

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OCONEE 1, CYCLE 5 DESIGN REPORT

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May 1979

OCONEE 1, CYCLE 5 DESIGN REPORT

T. A. Coleman, Editor

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Oconee 1, Cycle 5 Design Report

T. A. Coleman, Editor

Key Words: UO₂ Fuel, Fuel Cycle

ABSTRACT

The Oconee 1, cycle 5 fuel cycle was designed to irradiate five fuel assemblies to a burnup of approximately 40,000 MWd/mtU. The fuel cycle design was performed using standard Babcock & Wilcox calculational techniques for nuclear fuel cycles. All applicable design criteria were satisfied. Valuable experimental data on the performance characteristics of high-burnup fuel assemblies will be obtained from these assemblies in subsequent post-irradiation examinations.

CONTENTS

	Page
1. INTRODUCTION	1-1
2. CORE DESCRIPTION	2-1
3. OPERATING HISTORY	3-1
4. NUCLEAR ANALYSES	4-1
4.1. Fuel Cycle Description	4-1
4.2. Power Distributions	4-2
4.3. Control Rod Worths	4-2
4.4. Physics Parameters	4-2
5. MECHANICAL ANALYSIS	5-1
5.1. Fuel Assembly	5-1
5.2. Fuel Rod	5-2
5.2.1. Cladding Collapse	5-2
5.2.2. Cladding Stress	5-2
5.2.3. Cladding Strain	5-3
6. THERMAL-HYDRAULIC ANALYSIS	6-1
6.1. Core	6-1
6.2. Fuel Rod	6-1
7. ACCIDENT AND TRANSIENT ANALYSIS	7-1
7.1. General Safety Analysis	7-1
7.2. Accident Evaluation	7-1
REFERENCES	A-1

List of Tables

Table	Page
2-1. Oconee 1 Design Data and Performance Characteristics	2-3
4-1. Oconee 1, Cycle 5 - Fuel Cycle Depletion	4-3
4-2. Oconee 1, Cycle 5 - Control Rod Worths	4-4
4-3. Oconee 1, Cycle 5 - Stuck Rod Worths	4-4
4-4. Oconee 1, Cycle 5 - Shutdown Margin Calculation	4-5
4-5. Oconee 1, Cycle 5 - Ejected Rod Worths	4-6
4-6. Oconee 1, Cycle 5 Physics Parameters	4-7
5-1. Oconee 1 Cycle 5 - Fuel Design Parameters and Dimensions . . .	5-3
7-1. Oconee 1 Cycle 5 - Comparison of Key Parameters for Accident Analysis	7-3
7-2. Oconee 1 Cycle 5 LOCA Limits	7-3

List of Figures

Figure	
2-1. Reactor Coolant System Arrangement - Elevation	2-4
2-2. Reactor Coolant System Arrangement - Plan View	2-5
2-3. Reactor Vessel and Internals - General Arrangement	2-6
2-4. Mark B Fuel Assembly	2-7
2-5. Oconee 1, Cycle 5 - Full Core Loading Diagram	2-8
2-6. Oconee 1, Cycle 5 - Control Rod Assembly Locations	2-9
2-7. Incore Detector Locations	2-10
2-8. Reactor Vessel and Internals - Cross Section	2-11
4-1. Oconee 1, Cycle 5 Two Dimensional Relative Power Distri- bution - 4EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions	4-8
4-2. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution - 100 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions	4-9
4-3. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution - 200 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions	4-10
4-4. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution - 300 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions	4-11
4-5. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution - 330 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions	4-12
4-6. Oconee 1, Cycle 5 Assembly and Batch Burnups, MWd/mtU	4-13

1. INTRODUCTION

Near-term benefits in uranium resource utilization and fuel cycle costs can be achieved by extending the burnup of current-design fuel assemblies. Increasing the batch average discharge burnups would result in lower uranium requirements, better fuel cycle efficiency, and smaller spent fuel storage requirements for nuclear reactors. Extended fuel burnup also facilitates conversion to longer fuel cycles, which reduce the number of refueling outages and increase plant availability. In addition, extending burnup would increase flexibility in batch size variation to accommodate a utility's varying energy requirements. Fuel assemblies can be left in the core for another cycle or reinserted from spent fuel storage when a reduction in fresh fuel loading is desired.

In order to obtain valuable experimental data on the performance characteristics of fuel assemblies at high exposures, five fuel assemblies are being irradiated for a fourth cycle in Oconee 1, cycle 5, as part of a Department of Energy-sponsored research and development program with Duke Power Company and The Babcock & Wilcox Company. These fuel assemblies will be examined at poolside before and after the fourth cycle of irradiation. One of the fuel assemblies will be returned to Babcock & Wilcox's Lynchburg Research Center hot cell facility following the fourth cycle of irradiation for detailed examination of the fuel rods.

This report describes the analyses conducted to justify utilization of five fuel assemblies in Oconee 1 for a fourth cycle. Information is presented on the fuel cycle design and licensing for Oconee 1, cycle 5.

2. CORE DESCRIPTION

The Duke Power Company's Oconee Unit 1 is a Babcock & Wilcox-designed pressurized water reactor fueled with low-enriched uranium. The reactor coolant system comprises two loops with one hot outlet leg and two cold inlet legs per loop. Four reactor coolant pumps mounted in the cold legs pump subcooled water under a pressure of 2200 psia through the core where it absorbs the heat generated by the uranium fuel. After passing through the core, the coolant flows from the vessel out the hot leg and into a once-through steam generator (one per loop) where heat is transferred to the low-pressure secondary system. The superheated steam, at approximately 585F and 910 psig, is then used in the generation of electricity.

Figures 2-1 and 2-2 are simplified diagrams of the reactor coolant (RC) system arrangement, and Figure 2-3 shows the reactor vessel and internals.

The core consists of 177 fuel assemblies, each containing 208 fuel rods, 16 control rod guide tubes, and one incore instrumentation tube. The fuel rods are loaded with sintered, dished-end, cylindrical pellets of low-enriched uranium dioxide. The fuel pellets are positioned by spring and tubular spacers and are clad in cold-worked Zircaloy-4 tubing. The fuel rods are pressurized in a helium atmosphere and sealed with Zircaloy-4 end caps. The fuel assemblies have an average nominal fuel loading of 463.6 kg of uranium. The important design data and performance characteristics of the core are given in Table 2-1. Section 3 of the Final Safety Analysis Report for the Oconee Nuclear Station, Unit 1, provides additional details.¹

Figure 2-4 shows a typical Mark-B fuel assembly. It is composed of 208 fuel rods, 16 control rod guide tubes, one instrumentation tube, seven segmented spacer sleeves, eight spacer grids, and two end fittings. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the fuel rods in a 15 by 15 array. The center position is reserved for incore instrumentation.

Spacer grids are constructed of Inconel 718 strips, which are slotted and fitted together in egg-crate fashion. Each grid has 32 strips, 16 perpendicular to 16, which form the 15 by 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the walls of each square opening are integrally punched in the strips. On each of the two end spacer grids, the peripheral strip is extended and rigidly attached to the respective end fitting. The spacer grids are evenly spaced along the active fuel column length of the fuel assembly.

Figure 2-5 is the core loading diagram for Oconee 1, cycle 5. The initial enrichment of the fresh batch 7 fuel is 3.02 wt % ^{235}U . The remaining batches 4D, 5, and 6 were initially enriched to 3.20, 2.75, and 2.795 wt % ^{235}U , respectively. Five batch 4B assemblies were retained in cycle 5 for a fourth cycle of irradiation and are redesignated batch 4D. The batch 4D, 5, and 6 assemblies were shuffled to new locations at the beginning of cycle 5. The fresh batch 7 assemblies occupy the periphery of the core and eight interior locations.

Reactivity control is accomplished by 61 full-length silver-indium-cadmium (Ag-In-Cd) control rod assemblies (CRAs) and soluble boron shim. In addition to the full-length assemblies, eight partial-length CRAs shape the axial power distribution. Each CRA comprises 16 rods, a stainless steel spider, and a female coupling. Each control rod consists of stainless steel tubing and end caps, spring spacers, and cylindrical Ag-In-Cd rods in an unpressurized helium atmosphere. Full-length control rods have 134-inch absorber lengths, and 16 rods as a unit are termed control rod assemblies. The eight CRAs with partial length absorber in the lower 36 inches of each rod are designated axial power shaping rod assemblies (APSRAs). The location and operating group number for each CRA are shown in Figure 2-6.

The power distribution within the core is monitored at 364 locations (seven axial positions in 52 fuel assemblies) by the incore monitoring system's self-powered neutron detector (SPND) assemblies (see Figure 2-7). The outputs of this system are connected to the plant computer, which provides the operator with three-dimensional power distributions. The relationship of the detector locations to the control rods and other related reactor components is shown in Figure 2-8.

Two regenerative neutron sources (consisting of antimony-beryllium rods) provide a sufficient neutron count rate during fuel loading and reactor startup for monitoring reactor criticality. For cycle 5, the regenerative neutron sources are placed in core locations B-12 and P-4.

Table 2-1. Oconee 1 Design Data and Performance Characteristics

Reactor

Rated heat output, MWt	2568
Vessel coolant inlet temperature, F	554
Vessel coolant outlet temperature, F	603.8
Core outlet temperature, F	606.2
Core operating pressure, psig	2185

Core and Fuel Assemblies

Total number of fuel assemblies in core	177
Number of fuel rods per fuel assembly	208
Number of control rod guide tubes per assembly	16
Number of incore instrumentation positions per FA	1
Fuel rod outside diameter, in.	0.430
Clad thickness, in.	0.0265
Fuel rod pitch, in.	0.568
Fuel assembly pitch spacing, in.	8.587
Unit cell metal/water ratio (volume basis)	0.82
Cladding material	Zircaloy-4 (cold-worked)

Fuel

Material	UO ₂
Form	Dished-end, cylindrical pellets
Pellet diameter, in.	0.370
Active length, in.	~142
Density, % TD	>93.5

Control Data

Control rod material	Ag-In-Cd
Number of full-length rods	61
Number of APSRs	8
Control rod cladding material	SS-304

Figure 2-1. Reactor Coolant System Arrangement — Elevation

Figure 2-2. Reactor Coolant System Arrangement — Plan View

Figure 2-3. Reactor Vessel and Internals — General Arrangement

Figure 2-4. Mark B Fuel Assembly

Figure 2-5. Oconee 1, Cycle 5 – Full Core Loading Diagram

A						7	7	7	7	7					
B			7	7	7	E7 5	R8 6	E9 5	7	7	7	#			
C	7	M2 6	C6 5	F7 5	L2 6	M5 5	L14 6	F9 5	C10 5	M14 6	7				
D	7	B11 6	O13 6	L1 6	N3 6	D5 5	7	D11 5	N13 6	L15 6	O3 6	B5 6	7		
E	7	F3 5	A10 6	O9 5	K1 6	N2 6	O5* 4D	N14 6	K15 6	K3 5	A6 6	F13 5	7		
F	7	7	G6 5	C12 6	A9 6	K13 5	D6 5	N8 5	D10 5	O7 5	A7 6	C4 6	G10 5	7	7
G	7	G5 5	B10 6	E4 5	B12 6	F4 5	7	K8 5	7	F12 5	B4 6	E12 5	B6 6	G11 5	7
H	7	H15 6	M11 5	7	M13* 4D	H12 5	H9 5	K14* 4D	H7 5	H4 5	E3* 4D	7	E5 5	H1 6	7
K	7	K5 5	P10 6	M4 5	P12 6	L4 5	7	G8 5	7	L12 5	P4 6	M12 5	P6 6	K11 5	7
L	7	7	K6 5	O12 6	R9 6	C9 5	N6 5	D8 5	N10 5	G3 5	R7 6	O4 6	K10 5	7	7
M	7	L3 5	R10 6	G13 5	G1 6	D2 6	C11* 4D	D14 6	G15 6	C7 5	R6 6	L13 5	7		
N	7	P11 6	C13 6	F1 6	D3 6	N5 5	7	N11 5	D13 6	F15 6	C3 6	P5 6	7		
O	7	E2 6	O6 5	L7 5	F2 6	E11 5	F14 6	L9 5	O10 5	E14 6	7				
P			#	7	7	7	M7 5	A8 6	M9 5	7	7	7			
R						7	7	7	7	7					
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

*Location of thrice-burned batch 4D assemblies.

#Location of regenerative neutron source assemblies and retainer assemblies.

xxx	Previous cycle location.
x	Batch No.

Figure 2-6. Oconee 1, Cycle 5 — Control Rod Assembly Locations

x ← Group Number

Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	9	Safety
4	8	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total	69	

Figure 2-7. Incore Detector Locations

Figure 2-8. Reactor Vessel and Internals. — Cross Section

3. OPERATING HISTORY

The Oconee 1 reactor achieved initial criticality on April 19, 1973. Cycle 1 of Oconee 1 operated for 309.6 EFPD and was shut down for refueling on October 18, 1974. On February 11, 1975, initial cycle 2 criticality occurred. The five batch 4D fuel assemblies, which are being irradiated for a fourth cycle in cycle 5, were part of the fresh fuel batch for cycle 2. Cycle 2 operated for 292.2 EFPD. The refueling outage began on February 7, 1976, and initial cycle 3 criticality occurred on March 31, 1976. Cycle 3 was 308.3 EFPD in duration, and the unit was shut down for refueling on August 5, 1977. Oconee 1 achieved initial cycle 4 criticality on October 14, 1977. Cycle 4 ended on September 2, 1978, having achieved a cycle length of 245.9 EFPD.

Oconee 1 cycle 4 was the reference cycle for the licensing of cycle 5. The cycle 5 design was based on a planned cycle 4 length of 235 EFPD rather than the initial design length of 292 EFPD. Cycle 5 will operate in a feed-and-bleed mode for its entire design length of 330 EFPD.

4. NUCLEAR ANALYSES

4.1. Fuel Cycle Description

The fuel cycle design for Oconee 1 cycle 5 was performed with the PDQ07 code. The major characteristics of the PDQ07 model were as follows:

1. Two-dimensional (XY) quarter-core geometry.
2. Discrete (pin-by-pin) representation.
3. Thermal-hydraulic feedback.

Fuel placement was performed using the SHUFFLE computer code, and fuel pin relative power densities were calculated from the PDQ partition power files using the PINPOW code.

A homogenized one-zone PDQ07 model was used to generate most of the reactivity parameters for cycle 5. The PDQ07 and FLAME design models used at Babcock & Wilcox are discussed in detail in reference 2.

The Oconee 1 cycle 5 fuel cycle design meets the criteria applicable to radial power peaking, cycle lifetime, ejected rod worth, and shutdown margin. The cycle 5 design characteristics are as follows:

1. An out-in shuffle scheme with a feed batch of 56 assemblies (at 3.02 wt % ^{235}U) loaded primarily on the core periphery.
2. A mixture of once-, twice-, and thrice-burned assemblies loaded in the interior of the core.
3. Feed-and-bleed mode of operation.
4. No transient or Doppler control rod bank interchanges.

The fuel shuffle scheme and fuel batch designations are shown in Figure 2-5. The significant parameters associated with the fuel cycle depletion are summarized in Table 4-1. The fuel cycle depletion was performed with the regulating bank (group 7) fully withdrawn and the APSRs (group 8) centered.

4.2. Power Distributions

The cycle 5 design met the design criteria for radial pin peaking with a 7% margin to the design limits. The assembly relative power densities at various times during cycle 5 are presented in Figures 4-1 through 4-5.

Figure 4-6 shows the assembly exposures at the beginning and end of cycle 5. The maximum BOC 5 assembly burnup is 31,185 MWd/mtU for the four assemblies in locations H-5, H-11, E-8, and M-8, which are four of the batch 4D fuel assemblies being irradiated for a fourth cycle. These assemblies will achieve a burnup of 40,395 MWd/mtU at the end of cycle 5 and will be the maximum-burnup assemblies.

4.3. Control Rod Worths

Figure 2-6 identifies the control rods by group, number of rods in each group, group location, and function. Control rod groups are withdrawn according to the numbering sequence of the banks shown in the figure.

Control rod bank worths were calculated using the PDQ code in two-dimensional quarter-core geometry. Rod worth calculations were performed at hot zero power and hot full power conditions at the beginning and end of cycle 5. Table 4-2 summarizes control rod group worths.

A two-dimensional, full-core PDQ model was used to determine the worth of the maximum-worth stuck rod; the results are given in Table 4-3. Verification of shutdown margin, assuming maximum worth stuck rod conditions, is shown in Table 4-4.

Ejected rod worths were determined at hot zero power and hot full power using a two-dimensional, full-core PDQ model. Table 4-5 summarizes these calculated worths. The maximum calculated values for cycle 5 are within design limits.

4.4. Physics Parameters

Table 4-6 compares the core physics parameters of cycle 5 with those of reference cycle 4. The values for both cycles were generated using PDQ07. The average cycle burnup will be higher in cycle 5 than in the design cycle 4 because of the longer length of cycle 5. The power deficits from hot zero power to hot full power differ from those for the design cycle 4 because of this burnup difference. The differential boron worths and total xenon worths for cycle 5 are greater than or equal to those for the design cycle 4 because of fuel depletion and the associated buildup of fission products. Effective delayed neutron fractions for both cycles show a decrease with burnup.

4. NUCLEAR ANALYSES

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The fuel cycle design for Oconee 1 cycle 5 was performed with the PDQ07 code. The major characteristics of the PDQ07 model were as follows:

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A homogenized one-zone PDQ07 model was used to generate most of the reactivity parameters for cycle 5. The PDQ07 and FLAME design models used at Babcock & Wilcox are discussed in detail in reference 2.

The Oconee 1 cycle 5 fuel cycle design meets the criteria applicable to radial power peaking, cycle lifetime, ejected rod worth, and shutdown margin. The cycle 5 design characteristics are as follows:

1. An out-in shuffle scheme with a feed batch of 56 assemblies (at 3.02 wt % ^{235}U) loaded primarily on the core periphery.
2. A mixture of once-, twice-, and thrice-burned assemblies loaded in the interior of the core.
3. Feed-and-bleed mode of operation.
4. No transient or Doppler control rod bank interchanges.

The fuel shuffle scheme and fuel batch designations are shown in Figure 2-5. The significant parameters associated with the fuel cycle depletion are summarized in Table 4-1. The fuel cycle depletion was performed with the regulating bank (group 7) fully withdrawn and the APSRs (group 8) centered.

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Ejected rod worths were determined at hot zero power and hot full power using a two-dimensional, full-core PDQ model. Table 4-5 summarizes these calculated worths. The maximum calculated values for cycle 5 are within design limits.

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Table 4-1. Oconee 1, Cycle 5 - Fuel
Cycle Depletion

Core Parameter

No. of fuel assemblies in core	177
Type of fuel assembly	Mark B
Fuel loading, kgU/FA	463.6
Fuel density, % TD	94.0
No. of control rod assemblies	69
No. of full-length rods	61
No. of APSRs	8
Type of control rods	Ag-In-Cd

Operating Conditions

Rated power level, MWt	2568
Avg moderator temp at HFP, F	582
Hot zero power temperature, F	532
Primary system pressure, psia	2200
Core flow, lbm/ft ² -h	2.94×10^6

Fuel Cycle Parameters

Cycle No.	1	2	3	4	5
Full-power days	310	292.2	309	235	330
Power level, MWt	2568	2568	2568	2568	2568
kW/ft nuclear	5.81	5.81	5.81	5.81	5.81

Table 4-2. Oconee 1, Cycle 5 — Control Rod Worths^(a)

Bank(s) inserted	Sequential worth, % $\Delta\rho$			
	0 EFPD		330 EFPD	
	HFP	HZP	HFP	HZP
1-4	--	5.41	--	5.23
5	1.11	1.05	1.14	1.07
6	1.21	1.13	1.24	1.13
7	1.45	1.32	1.53	1.36
1-7	--	8.91	--	8.79
8	0.43	0.42	0.48	0.47

(a) All calculations were performed with APSRs (bank 8) inserted except for the bank 8 worth calculations.

Table 4-3. Oconee 1, Cycle 5 — Stuck Rod Worths

Conditions	Stuck rod location	Worth, % $\Delta\rho$
BOC, HZP	L14	1.99
BOC, HZP	N12	2.17
EOC, HZP	L14	1.95
EOC, HZP	N12	2.01

Table 4-4. Oconee 1, Cycle 5 - Shutdown
Margin Calculation

	<u>BOC, % $\Delta k/k$</u>	<u>EOC, % $\Delta k/k$</u>
Available rod worth		
Total rod worth, HZP	8.91	8.79
Worth reduction due to burnup of poison material	-0.36	-0.42
Maximum stuck rod, HZP	<u>-2.17</u>	<u>-2.01</u>
Net worth	6.38	6.36
Less 10% uncertainty	<u>-0.64</u>	<u>-0.64</u>
Total available worth	5.74	5.72
Required rod worth		
Power deficit, HFP to HZP	1.31	2.12
Max allowable inserted rod worth	0.40	0.60
Flux redistribution	<u>0.59</u>	<u>1.20</u>
Total required worth	2.30	3.92
Shutdown margin (total available worth minus total required worth)	3.44	1.80

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

Table 4-5. Oconee 1, Cycle 5 - Ejected
Rod Worths

<u>Conditions</u>	<u>Inserted</u>	<u>Rod location</u>	<u>Worth, % $\Delta\rho$</u>
BOC, HZP	5-8	N12	0.57
BOC, HFP	7-8	K13	0.21
BOC, HFP	7-8	L10	0.21
EOC, HZP	5-8	N12	0.70
EOC, HFP	7-8	K13	0.25
EOC, HFP	7-8	L10	0.20

Table 4-6. Oconee 1, Cycle 5 Physics Parameters^(a)

	Cycle 4 ^(b)	Cycle 5 ^(c)
Cycle length, EFPD	292	330
Cycle burnup, MWd/mtU	9,136	10,327
Average core burnup, EOC, MWd/mtU	19,034	19,027
Initial core loading, mtU	82.1	82.1
Critical boron, BOC (no Xe), ppm		
HZP, group 8 37.5% wd ^(d)	1415	1458
HZP, groups 7 and 8 inserted	1335	1324
HFP, group 8 inserted	1145	1276
Critical boron, EOC (eq Xe), ppm		
HZP, group 8 37.5% wd	373	343
HFP, group 8 37.5% wd	88	44
Control rod worths, HFP, BOC, % $\Delta k/k$		
Group 6	1.07	1.21
Group 7	0.93	1.45
Group 8 37.5% wd	0.50	0.43
Control rod worths, HFP, EOC, % $\Delta k/k$		
Group 7	1.16	1.53
Group 8 37.5% wd	0.47	0.48
Max ejected rod worth, HZP, % $\Delta k/k$ ^(e)		
BOC (N-12)	0.68	0.57
EOC (N-12)	0.61	0.70
Max stuck rod worth, HZP, % $\Delta k/k$		
BOC (N-12)	1.74	2.17
EOC (N-12)	2.02	2.01
Power deficit, HZP to HFP, % $\Delta k/k$		
BOC	1.49	1.31
EOC	2.07	2.12
Doppler coeff, $10^{-5}(\Delta k/k-^{\circ}F)$		
BOC, 100% power, no Xe	-1.45	-1.45
EOC, 100% power, eq Xe	-1.55	-1.62
Moderator coeff, HFP, $10^{-4}(\Delta k/k-^{\circ}F)$		
BOC (0 Xe, crit ppm, gp 8 ins)	-1.00	-0.45
EOC (eq Xe, 17 ppm, gp 8 ins)	-2.55	-2.64
Boron worth, HFP, ppm/% $\Delta k/k$		
BOC (1150 ppm)	109	109
EOC (17 ppm)	101	97
Xenon worth, HFP, % $\Delta k/k$		
BOC (4 EFPD)	2.60	2.62
EOC (equilibrium)	2.61	2.73
Eff delayed neutron fraction, HFP		
BOC	0.00593	0.00598
EOC	0.00530	0.00521

(a) Cycle 5 data are for the conditions stated in this report. The cycle 4 core conditions are identified in reference 4.

(b) Based on 292 EFPD at 2568 MWt, cycle 3.

(c) Cycle 5 data are based on a "planned" cycle 4 length of 235 EFPD; the cycle 4 "design" lifetime is 292 EFPD.

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

(e) Ejected rod worth for groups 5 through 8 inserted.

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

	8	9	10	11	12	13	14	15
H	0.83	0.93	0.96	0.90	1.37	1.03	1.09	0.87
K		1.35	1.07	1.21	0.98	1.09	0.93	0.83
L			1.05	1.25	1.03	0.95	1.15	0.67
M				1.09	1.23	0.89	0.91	
N					1.21	0.94	0.61	
O						0.70		
P								
R

Inserted Rod Group No.

Relative Power Density

Figure 4-2. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution — 100 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.83	0.92	0.95	0.89	1.34	1.03	1.11	0.92
K		1.31	1.04	1.17	0.97	1.10	0.96	0.88
L			1.01	1.19	1.01	0.96	1.17	0.71
M				1.05	1.19	0.90	0.95	
N					1.18	0.95	0.65	
O						0.73		
P								
R

Inserted Rod Group No.

Relative Power Density

Figure 4-3. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution — 200 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.85	0.93	0.96	0.90	1.30	1.03	1.11	0.93
K		1.28	1.03	1.15	0.97	1.09	0.96	0.89
L			1.01	1.17	1.00	0.96	1.16	0.74
M				1.03	1.17	0.91	0.96	
N					1.16	0.96	0.69	
O						0.77		
P								
R								

x
x.xx

Inserted Rod Group No.

Relative Power Density

Figure 4-4. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution - 300 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.87	0.94	0.97	0.90	1.27	1.02	1.10	0.93
K		1.26	1.03	1.14	0.97	1.09	0.96	0.90
L			1.01	1.15	1.00	0.97	1.15	0.75
M				1.02	1.15	0.92	0.97	
N					1.15	0.97	0.72	
O						0.79		
P								
R

Inserted Rod Group No.

Relative Power Density

Figure 4-5. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution — 330 EFPD, Full Power, Equilibrium Xenon, Normal Rod Positions

	8	9	10	11	12	13	14	15
H	0.88	0.94	0.97	0.90	1.27	1.02	1.10	0.93
K		1.26	1.03	1.14	0.97	1.09	0.96	0.90
L			1.01	1.14	0.99	0.97	1.15	0.76
M				1.02	1.15	0.93	0.97	
N					1.14	0.97	0.72	
O						0.80		
P								
R

Inserted Rod Group No.

Relative Power Density

Figure 4-6. Oconee 1, Cycle 5 Assembly and Batch Burnups, MWd/mtU

	8	9	10	11	12	13	14	15
H	28,479	20,488	16,053	31,135	0	15,903	5,889	0
	* 37,208	30,108	25,982	* 40,395	13,667	26,513	17,283	9,403
K	20,479	0	14,270	5,138	19,206	8,537	16,345	0
	30,098	13,457	25,055	17,210	29,254	19,837	26,197	9,013
L	16,057	14,276	17,336	5,853	8,262	15,846	0	0
	25,984	25,057	27,844	18,130	18,694	25,743	11,974	7,400
M	31,135	5,141	5,855	17,341	5,011	18,348	0	
	* 40,393	17,209	18,125	28,175	17,261	27,717	9,771	
N	0	19,212	8,268	5,012	5,846	7,092	0	
	13,664	29,257	18,699	17,268	17,990	16,948	6,852	
O	15,093	8,540	15,845	18,359	7,095	0	BOC	
	27,511	19,838	25,741	27,726	16,950	7,699	EOC (330 EFPD)	
P	5,891	16,347	0	0	0			
	17,283	26,197	11,973	9,770	6,852			
R	0	0	0					
	9,402	9,012	7,400					

Batch	No. FAs	E _o	BOC 5	EOC 5	ΔBU
* 4D	5	3.20	30,604	39,757	9,153
5	60	2.75	17,011	24,104	10,093
6	56	2.79	6,539	17,960	11,421
7	56	3.02	0	9,589	9,589
Core	177		8,700	19,027	10,327

5. MECHANICAL ANALYSIS

5.1. Fuel Assembly

The types of fuel assemblies and pertinent fuel design parameters and dimensions for Oconee 1 cycle 5 are listed in Table 5-1. All fuel assemblies are identical in concept and are mechanically interchangeable.

Five batch 4D Mark-B3 fuel assemblies will remain in the core for their fourth cycle of irradiation and will experience burnups less than 41,000 MWd/mtU. The paragraphs below discuss the adequacy of the design relative to assembly growth, fuel rod growth, and holddown spring relaxation.

The fuel assembly is designed with a gap between the upper end fitting and the reactor internals to accommodate guide tube irradiation growth. An analysis performed for the fuel assemblies in question demonstrates that the gap is sufficient for fuel assembly average burnups up to 48,000 MWd/mtU. This analysis conservatively accounts for fuel assembly and internals tolerances and maximum expected growth based on a 95/95 upper tolerance limit. The growth curve has been developed from post-irradiation data from Mark-B fuel assemblies.

In a similar manner, fuel rod growth is accommodated by a gap between the upper end of the rod and the grillage of the upper end fitting. The size of the gap was designed assuming minimum guide tube growth and maximum fuel rod growth. Post-irradiation data indicate that fuel rod and guide tube growth are nearly equal. Thus, there is little gap reduction at 41,000 MWd/mtU.

The holddown spring is also influenced by irradiation, which causes stress relaxation. Because the holddown spring is several inches away from the active fuel, its irradiation exposure is relatively low, resulting in limited relaxation. In fact, guide tube growth causes additional spring compression that compensates for any spring relaxation. The net result is that holddown force does not change under irradiation, eliminating any holddown force concern because of the extended burnup. The spring stress analysis accounts for spring compression greater than expected for these assemblies.

5.2. Fuel Rod

The pertinent fuel rod and pellet dimensions for cycle 5 fuel are listed in Table 5-1. Cycle 5 fuel was analyzed for cladding stress, strain, and creep collapse. The results of these analyses are discussed in the following subsections.

5.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories as well as for the four-cycle assembly power histories of batch 4D. For cycle 5, the batch 5 fuel is more limiting than all other batches except 4D because of its previous incore exposure time. The batch 5 and 4D assembly power histories were analyzed, and the most limiting assembly from each batch was determined.

The power histories for the most limiting assemblies were used to calculate the fast neutron flux level for the energy range above 1 MeV. The collapse time for the most limiting assembly from each batch was conservatively determined to be more than 30,000 EFPH (effective full-power hours), which is longer than the maximum projected batch 5 exposure life of 21,456 EFPH and maximum the projected batch 4D exposure life of 28,469 EFPH (see Table 5-1). The creep collapse analyses were performed based on the conditions set forth in references 3 and 4.

5.2.2. Cladding Stress

The Oconee 1 stress parameters are enveloped by a conservative fuel rod stress analysis. This includes batch 4D fuel, primarily because the worst-case stress conditions are at BOL (beginning of life). For design evaluation, the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses (primary and secondary) must be less than the minimum specified unirradiated yield strength. In all cases, the margin is more than 30%. The following conservatisms with respect to Oconee 1 fuel were used in the analysis:

1. Low post-densification internal pressure.
2. Low initial pellet density.
3. High system pressure.
4. High thermal gradient across the cladding.

5.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is established for plastic cladding strain of less than 1% at values of maximum design local pellet burnup and heat generation rate, which are higher than the values the Oconee 1 fuel is expected to see, including batch 4D. The strain analysis is also based on the maximum specification value for the fuel pellet diameter and density and the lowest permitted specification tolerance for the cladding inside diameter.

Table 5-1. Oconee 1 Cycle 5 -- Fuel Design Parameters and Dimensions

	Thrice- burned FAs, <u>batch 4D</u>	Twice- burned FAs, <u>batch 5</u>	Once- burned FAs, <u>batch 6</u>	Fresh FAs, <u>batch 7</u>
FA type	Mark B3	Mark B4	Mark B4	Mark B4
Number of FAs	5	60	56	56
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
Undensif. active fuel length (nominal), in.	142.0	142.6	142.25	142.25
Fuel pellet initial density (nominal), % TD	>94.5	93.5	94.0	94.0
Fuel pellet OD (mean specif.), in.	0.3685	0.3700	0.3695	0.3695
Initial fuel enrich., wt % ^{235}U	3.20	2.75	2.79	3.02
BOC burnup (avg), MWd/mtU	30,604	17,011	6,539	0
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>30,000
Estimated residence time (max), EFPH	28,469	21,456	22,440	26,496

6. THERMAL-HYDRAULIC ANALYSIS

6.1. Core

A thermal-hydraulic analysis was performed in support of Ocone 1, cycle 5 operation. Core integrity during steady-state operation is ensured by maintaining the minimum departure from nucleate boiling ratio (MDNBR) for the highest power assembly in the core (the "hot" assembly) above a statistical minimum. Conservatism in the analysis is ensured by the use of design peaking (greater than the reference design value), reduced flow, and plant overpower reflecting instrumentation errors.

The batch 4D assemblies are hydraulically similar to all other fuel assemblies in the cycle 5 core; therefore, the presence of the assemblies has no impact on core thermal-hydraulic performance.

The potential effect of fuel rod bow on DNBR was considered by incorporating suitable margins into the core safety limits and reactor protection system (RPS) setpoints. The fuel cycle design calculations show that the maximum radial-local peak during cycle 5 is never located in the batch 4D assemblies. The rod bow penalty for cycle 5 is conservatively based on the burnup of the highest-burnup assembly in the batch containing the hot assembly. The method of calculating the maximum rod bow penalty has been reviewed and approved by the NRC.⁵ In addition, no credit is claimed for the difference between calculated peaking for the "hot" assembly and the reference design peaking used for the analysis.

6.2. Fuel Rod

Fuel pins in batch 4D were analyzed with the TAFY-3 fuel pin performance computer code⁶, which is the currently licensed code for calculating the pin pressure and temperature used for accident analysis. The results of this analysis demonstrated acceptable internal fuel rod pressures for burnups to 44,000 MWd/mtU based on the actual power histories of these assemblies for

their first three cycles of operation and a conservative power history for their fourth cycle.

A comparative analysis was performed using the TACO2 fuel rod thermal performance code.⁷ TACO2 is currently scheduled for NRC review and is technologically superior to the TAFY-3 code. The following supplemental models are present in TACO2: a burnup-dependent fission gas release model based on PWR type fuel, gap closure due to pellet cracking, fuel pellet densification and Zircaloy cladding creepdown.

TAFY-3 is a very conservative basis for licensing, as demonstrated by comparisons of the internal fuel rod pressure and temperature predicted by TACO2 and TAFY-3. Since TACO2 itself is a conservative code for licensing, as demonstrated by both thermal and fission gas release benchmarks, future licensing applications using this code are expected to justify the even higher design burnups for the Mark-B fuel assembly.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR¹ accident analysis has been examined with respect to changes in cycle 5 parameters to determine the effect of the cycle 5 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 the Oconee 1 fuel densification report.⁸ Since batch 7 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the fuel densification report, the conclusions in that reference are still valid.

7.2. Accident Evaluation

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, thermal-hydraulic, and kinetics, including the reactivity feedback coefficients and control rod worths.

The core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Table 7-1 is a comparison of the key accident analysis parameters from the FSAR and cycle 5.

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. 1.⁹ The BAW-10103 analysis 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 linear heat rate (LHR) and the lifetime pin pressure data used in the reference 9 LOCA limits analysis is conservative compared to those calculated for this reload, including the batch 4D assemblies. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 1

cycle 5 fuel. Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 1 cycle 5 fuel.

The radiological consequences of the fuel handling accident are directly dependent on the fission product inventory within a single fuel assembly. Since the five batch 4D fuel assemblies will receive a higher burnup than normal, an evaluation was performed to determine whether or not the FSAR radiological consequences of the fuel handling accident represent a conservative analysis for each of these five high burnup fuel assemblies. The dose results of the evaluation confirm that the radiological consequences of the fuel handling accident for each of these five high burnup fuel assemblies are bounded by the FSAR analysis of the accident. Although these high-burnup assemblies receive extended burnup, the FSAR doses are larger because the FSAR fuel handling accident analysis assumed a larger than actual assembly power for the high-burnup assemblies. The two-hour thyroid dose for the worst-case fuel assembly is 0.29 Rem, which is a factor of 1.48 smaller than the corresponding FSAR thyroid dose. The two-hour whole-body dose for the worst-case assembly is 0.013 Rem, which is a factor of 2 smaller than the FSAR whole-body dose.

It is concluded from the examination of cycle 5 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the Oconee 1 plant's ability to operate safely during cycle 5. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 5 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 5 are bounded by the FSAR¹, the fuel densification report⁸, and/or subsequent cycle analyses.

Table 7-1. Oconee 1 Cycle 5 — Comparison of Key Parameters for Accident Analysis

Parameter	FSAR and densification report value	Predicted cycle 5 value
Doppler coeff, $10^{-5} \Delta k/k/^{\circ}F$		
BOC	-1.17	-1.45
EOC	-1.33	-1.62
Moderator coeff, $10^{-4} \Delta k/k/^{\circ}F$		
BOC	+0.5	-0.45
EOC	-3.0	-2.64
Total control rod worth, HZP, % $\Delta k/k$	10	8.91
Initial boron conc, HFP, ppm	1400	1276
Boron reactivity worth at 70F, ppm/1% $\Delta k/k$	75	76
Max ejected rod worth, HFP, % $\Delta k/k$	0.65	0.25
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20

Table 7-2. Oconee 1 Cycle 5
LOCA Limits

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

REFERENCES

- ¹ Oconee Nuclear Station, Units 1, 2, and 3 — Final Safety Analysis Reports, Docket Nos. 50-269, 50-270, and 50-287, Duke Power Company.
- ² Core Calculational Techniques and Procedures, BAW-10118, Babcock & Wilcox, Lynchburg, Virginia, October 1977.
- ³ Oconee Unit 1, Cycle 4 Reload Report, BAW-1447, Babcock & Wilcox, Lynchburg, Virginia, March 1977.
- ⁴ Program to Determine In-Reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, July 1973.
- ⁵ Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment 14 to Facility Operating License No. DPR-54, Rancho Seco Nuclear Generation Station, Sacramento Municipal Utility District, Docket No. 50-312.
- ⁶ C. D. Morgan and H. S. Kao, TAFY — Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, Lynchburg, Virginia, May 1972.
- ⁷ Y. H. Hsii, et al., TACO2 — Fuel Pin Performance Analysis, BAW-10141P, Babcock & Wilcox, Lynchburg, Virginia, February 1979.
- ⁸ Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, July 1973.
- ⁹ ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, September 1975.

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