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MASTER

POST-IRRADIATION EXAMINATION OF OCONEE 1 FUEL —
END-OF-CYCLE 2 NONDESTRUCTIVE TEST PHASE

PREPARED FOR THE
U.S. DEPARTMENT OF ENERGY
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Lynchburg, Virginia

Report BAW-1574

November 1979

Post-Irradiation Examination of Oconee 1 Fuel -
End-of-Cycle 2 Nondestructive Test Phase

Key Words: UO₂ Fuel, Zircaloy-4, Irradiation Testing, Fuel
Performance, Fuel Cladding Performance

ABSTRACT

Standard B&W Mark B (15 × 15) pressurized water reactor fuel assemblies were nondestructively examined at the end of the second cycle of Oconee 1 reactor operation. Burnups of the 16 fuel assemblies examined ranged from 13,100 to 20,000 MWd/mtU. The examinations were conducted in the Oconee 1 and 2 spent fuel storage pool using the installed underwater test equipment. Data obtained included fuel rod and fuel assembly dimensions, water channel spacings, holddown spring forces, fuel rod crud characteristics, and fuel column axial gap and stack lengths. Visual examinations revealed no evidence of significant rod bowing, cladding deformation, cocked grids, or rod defects. The results, summarized in this report, indicate that the assemblies performed well through two cycles of reactor operation.

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1. INTRODUCTION

The B&W Mark B (15 × 15) fuel assembly design was first used in Duke Power Company's Oconee 1 nuclear reactor during its initial cycle of operation. Criticality was first achieved on April 19, 1973. The value of obtaining operating performance information on this fuel was recognized well before plant startup. Therefore, a cooperative post-irradiation examination (PIE) program with the owner/utility, Duke Power Company, was initiated to measure important fuel assembly parameters over three cycles of reactor operation. The information obtained from this program is being evaluated to ensure that the fuel is performing as designed and to provide a basis for future design improvements in B&W's nuclear fuel product.

The PIE program at Oconee 1 covers the first three reactor cycles and includes both destructive and nondestructive examinations of the fuel assemblies. Destructive examinations are performed in B&W's hot cell laboratory at the Lynchburg Research Center (LRC). Nondestructive testing (NDT) is conducted at the Oconee site in the Oconee 1 and 2 spent fuel storage pool where an extensive array of underwater test equipment has been designed, built, and installed for this purpose. Results obtained from the first cycle nondestructive and destructive examinations are summarized in references 1 and 2, respectively. This report summarizes the NDT results obtained from the end-of-cycle (EOC) 2 poolside examination. A brief description of the poolside PIE equipment and NDT methods used is contained in reference 3.

The following examinations and measurements were performed on 16 fuel assemblies at EOC 2:

1. Television examination and videotaping of fuel assemblies.
2. Periscope examination and 35-mm photography of fuel assemblies.
3. Fuel assembly holddown spring force versus load deflection.
4. Rod and assembly length and spacer grid axial locations.
5. Fuel assembly bow.
6. Water channel spacings.

7. Crud sampling.
8. Fuel column axial gamma scans.
9. Fuel rod diameters.

The 16 fuel assemblies examined at EOC 2 and the particular examinations conducted on each are presented in Table 1-1. The relevant fuel assembly design and core characteristics and the results of the EOC 2 poolside examinations are summarized in this report.

Table 1-1. Fuel Assemblies Examined at Oconee 1 EOC 2

Assembly ID	Assembly burnup, MWd/mtU	Examinations Conducted								
		Visual	Assembly length	Rod length	Assembly bow	Water channel spacing	Rod ^(b) diameter	Gamma scan	Holddown spring force	Crud ^(c) analysis
1B04	19,800	X	--	--	--	X	--	--	--	X(7)
1B05	19,700	X	--	--	--	X	--	--	--	--
1B08	18,700	X	--	--	X	--	X(4)	--	--	--
1B10	17,800	X	--	--	--	--	--	--	--	X
1B19	17,800	X	--	--	X	--	X(4)	--	X	X(7)
1B29	20,000	X	--	--	X	X	X(4)	X	X	X(7)
1B30	20,000	X	--	--	X	--	X(3)	--	X	X
1C01	17,300	X	--	--	--	--	--	X	--	X
1C30	15,900	X	--	--	X	X	--	--	--	--
1C44	19,300	X	X	X	X	--	X(4)	X	X	X
1C56	18,700	X	X	X	--	X	X(16)	X	X	X
1C63	16,600	X	X	--	X	X	--	X	--	--
1C66	18,500	X	X	--	X	X	--	X	--	--
1C73	18,500	X	X	X	X	X	--	--	--	--
1C76	18,500	X	X	X	--	--	--	--	--	--
1D40 ^(d)	13,100	X	X	--	X	X	--	--	--	--

(a) FLAME-calculated⁶ assembly average burnup.

(b) Four corner rods were measured in each assembly except 1B30; numbers in parenthesis indicate the number of fuel rods examined.

(c) One sample was obtained from each assembly unless noted otherwise in parenthesis.

(d) Assembly 1D40 was inserted after EOC 1 operation and thus was examined after only one cycle of irradiation.

2. SUMMARY

This report summarizes the end-of-cycle (EOC) 2 phase of the nondestructive (poolside) post-irradiation examination of Oconee 1 fuel assemblies. The examination was completed in July 1976. A total of 16 B&W Mark B (15 × 15) fuel assemblies were examined. Fifteen of the assemblies had been in the core through two cycles of operation; the other had been inserted during refueling at EOC 1. Assembly average burnups for the 15 two-cycle assemblies ranged from 15,900 to 19,800 MWd/mtU and averaged 18,500 MWd/mtU. The assembly average burnup for the one-cycle assembly was 13,100 MWd/mtU.

2.1. Examinations Conducted

Various examinations were conducted on each of the 16 fuel assemblies. The data obtained included the following:

1. Visual fuel assembly appearance (television and periscope examination).
2. Fuel rod and assembly growth (length measurements).
3. Fuel assembly bow (line scan profilometry).
4. Fuel rod bow (water channel spacing measurements).
5. Fuel cladding creepdown and ovality (diameter profilometry).
6. Fuel column gap and stack lengths (gamma scans).
7. Holddown spring preload force (load-deflection measurements).
8. Fuel rod crud characteristics (thickness and composition analyses).

2.2. Examination Results

The results of the poolside examination indicated that the assemblies performed well through two cycles of irradiation. Visual examination of the peripheral rods of the 16 fuel assemblies revealed no indication of fuel rod defects; the general appearance of the assemblies was very good.

The following summary of results is relevant to the assemblies that experienced two cycles of irradiation:

1. Fuel assembly and fuel rod growth averaged 0.22 and 0.28% $\Delta L/L$, respectively, from beginning of life (BOL).
2. Fuel rod bow, determined from the standard deviation of water channel spacing measurements and expressed conservatively as percent water channel closure (95% UTL), averaged 12.3%. Worst-span water channel closures (the maximum closure per assembly) averaged 14.4%.
3. The fuel assemblies exhibited a tendency to bow outward radially from the center of the core. The maximum assembly bow magnitudes averaged 0.20 in.
4. Cladding diametral creepdown averaged 0.0027 in. A distinct axial variation in average creepdown was apparent along the length of the fuel rod. The average creepdown increased monotonically from a minimum of 0.0019 in. at the top to a maximum of 0.0031 in. near the bottom of the rod.
5. The mean value of the statistically determined average maximum cladding ovality was 0.0057 in. A bimodal distribution of ovality was noted along the axial length of the average fuel rod; maxima of 0.0060 in. and 0.0065 in. occurred near the top and bottom, respectively.
6. Minor indications of fuel pellet-cladding contact at various fuel pellet interfaces were inferred from the diameter profilometry traces of 12 of the 35 rods profiled.
7. Fuel stack length decreased an average of 0.74% from BOL, agreeing favorably with the predicted decrease of 0.92% $\Delta L/L$.
8. Of 24 rods subjected to gamma scans, eight contained no observable gaps in their fuel columns. Approximately 60% of the 64 gaps (≥ 0.1 inch length) found in the other 16 rods were located within the top 12 in. of the fuel column, and all of the larger gaps (≥ 0.25 in.) were contained within 20 in. from the top of the column.
9. Holddown spring preload forces and spring rates exhibited no significant change after two cycles of irradiation.
10. Crud flake thickness averaged 0.00051 in. and exhibited no distinct axial variation. Crud metallic composition consisted of iron (55%) nickel (39%), and chrome (6%, by weight); the only compound identified was NiFe_2O_4 .

3. MARK B FUEL ASSEMBLY DESIGN AND OPERATING HISTORY

3.1. Fuel Assembly Design

Duke Power Company's Oconee Unit 1 is a Babcock & Wilcox designed pressurized water reactor with a core composed of 177 Mark B (15 × 15) fuel assemblies. A cutaway version of a typical Mark B fuel assembly is shown in Figure 3-1. The structural cage of the assembly consists of 16 control rod guide tubes (cold-worked, stress-relieved Zircaloy-4) permanently attached to the upper and lower end fittings. The spacer grids (Inconel 718) form the structural link between the guide tubes and the fuel rods (cold-worked, stress relieved, Zircaloy-4 clad), but are free to move with the fuel rods as the rods grow. Fuel assembly lift resulting from coolant flow is prevented by the helical holddown spring contained within the upper end fitting of the assembly.

Each fuel assembly consists of a 15 × 15 lattice arrangement with 16 positions occupied by guide tubes and one position, the center lattice site, occupied by an in-core instrumentation tube (cold-worked, stress-relieved Zircaloy-4). The remaining 208 lattice positions contain fuel rods. A schematic cross-section of a typical fuel assembly displaying the fuel rod numbering convention is presented in Figure 3-2.

The fuel contained within the rods is low enriched UO_2 in the form of dished-end, chamfered, cylindrical pellets (sintered). The nominal design parameters and operating conditions of Oconee 1 fuel are summarized in Table 3-1. Additional details are contained in section 3 of the Final Safety Analysis Report⁴ for the Oconee Nuclear Station, Unit 1, and in the Oconee 1, Cycle 2 Reload Report⁵.

3.2. Cycles 1 and 2 Operating Histories

Oconee 1 achieved cycle 1 initial criticality on April 19, 1973, and zero power physics testing was completed on April 30. The unit officially began commercial operation on July 15, 1973. This cycle was operated such that control rod group 7 was 16 to 27% withdrawn throughout the entire period of operation. Control rod interchanges, wherein the individual control rod assemblies comprising the control groups were reassigned to different groups, took place at 92 EFPD (November 20, 1973) and at 196 EFPD (May 1, 1974). Assembly 1C01 was in a group 7 location during the first (0-92 EFPD) and third (196-309.3 EFPD) rod configurations, and assembly 1B04 was in a group 7 location during the second configuration (92-196 EFPD). Assembly 1C44 was in a group 6 location for the entire cycle; this group was approximately 90-95% withdrawn for most of the cycle. Cycle 1 was completed on October 18, 1974, after 309.3 effective full power days (EFPDs) of operation. The power history for cycle 1 is displayed in Figure 3-3. The cycle 1 core locations of the 16 fuel assemblies examined at the end of cycle 2 are shown in Figure 3-4. The EOC 1 axial burnup profiles of the examined assemblies are given in Table 3-2. These tabulated burnup distributions were calculated using the three-dimensional nodal code FLAME⁶ with one node per assembly in the radial direction and 32 nodes per assembly in the axial direction. The actual reactor operations during each cycle were modeled as closely as practicable using this code.

Cycle 2 initial criticality was achieved on February 11, 1975, and power operation began on March 11. This cycle had a control rod interchange at 53 EFPD (May 17, 1975). However, this rod interchange did not affect the positions of control rods in any of the examined assemblies. Control rod groups 6 and 7 were withdrawn at 237 EFPDs (December 5, 1975). Group 7 was 3 to 10% withdrawn prior to the rod withdrawal and was approximately 85% withdrawn thereafter. Control rod group 6 was approximately 85% withdrawn prior to 237 EFPDs and was completely withdrawn for the remainder of the cycle. Cycle 2 operation was completed at 292.2 EFPDs on February 7, 1976. Assemblies 1B10 and 1B19 were in a group 7 location for the entire cycle. The other examined assemblies did not contain control rods during this cycle. The power history for cycle 2 is shown in Figure 3-5. The cycle 2 core locations of the examined assemblies are shown in Figure 3-6, and the EOC 2 FLAME-calculated axial burnup profiles are listed in Table 3-3.

Table 3-1. Oconee 1 Nominal Fuel Design
Parameters

Reactor Core

Power rating, MWt	2,568
Average linear heat rate, kW/ft	5.656
Inlet coolant temperature, F	555
Outlet coolant temperature, F	606
Operating pressure, psia	2,200

Fuel Assemblies and Fuel Rods

Number of fuel assemblies in core	177
Number of fuel rods/assembly	208
Number of control rod guide tubes/assembly	16
Fuel rod outside diameter, in.	0.430
Fuel tubing (cladding) thickness, in.	0.0265
Fuel rod pitch, in.	0.568
Fuel assembly pitch, in.	8.587

Fuel

Material	UO ₂ (sintered)
Fuel pellet length, in.	0.700
Fuel pellet diameter, in.	
Assembly batches B and C	0.370
Assembly batch D	0.367
Active fuel length, in.	
Assembly batches B and C	144.0
Assembly batch D	141.0
Fuel density, % theoretical	
Assembly batches B and C	93.5
Assembly batch D	95.5

Table 3-2. Oconee 1 EOC 1 FLAME-Calculated Axial Burnup Profiles

Fuel assembly ID	1C44	1C30	1B29 1B30	1B10 1B19	1B08	1C56	1B04	1B05	1C66 1C73 1C76	1C01	1C63
Location ^(a)	H-14	H-15	K-13	L-11	L-13	L-14	M-11	M-12	M-13	N-12	N-13
Burnup ^(b)	19,418	6,338	11,907	11,761	10,991	8,995	10,638	10,494	5,769	8,104	7,550
Elevation, in. ^(c)											
141.8 (top)	1,903	1,473	3,206	3,264	3,035	2,352	2,813	2,842	2,589	1,904	1,890
137.2	4,254	2,873	6,001	5,084	5,672	4,426	5,260	5,315	4,852	3,591	3,562
132.8	5,369	4,095	8,257	5,334	7,794	6,130	7,229	7,306	6,690	4,978	4,938
128.2	3,141	5,092	9,981	10,030	9,411	7,453	8,725	8,825	8,102	6,056	6,013
123.8	9,398	5,851	11,244	11,258	10,596	8,436	9,817	9,939	9,146	6,861	6,818
119.2	10,425	6,415	12,142	12,126	11,441	9,143	10,596	10,738	9,896	7,454	7,406
114.8	11,078	6,806	12,765	12,735	12,030	9,637	11,146	11,308	10,426	7,923	7,833
110.2	11,502	7,068	13,187	13,170	12,433	9,972	11,533	11,718	10,794	8,436	8,149
105.8	11,779	7,240	13,471	13,457	12,706	10,197	11,795	11,994	11,043	8,731	8,364
101.2	11,960	7,352	13,662	13,639	12,887	10,346	11,966	12,171	11,208	8,907	8,507
96.8	12,079	7,424	13,790	13,746	13,000	10,445	12,070	12,279	11,316	9,012	8,599
92.2	12,159	7,471	13,872	13,800	13,066	10,509	12,123	12,338	11,384	9,073	8,658
87.8	12,217	7,504	13,923	13,813	13,093	10,551	12,137	12,354	11,425	9,104	8,696
83.2	12,260	7,528	13,953	13,790	13,086	10,578	12,121	12,334	11,446	9,116	8,719
78.8	12,294	7,546	13,973	13,769	13,080	10,596	12,092	12,313	11,456	9,121	8,734
74.2	12,322	7,561	13,984	13,737	13,064	10,607	12,051	12,280	11,457	9,122	8,742
69.8	12,344	7,572	13,984	13,684	13,024	10,612	11,997	12,224	11,447	9,117	8,746
65.2	12,361	7,581	13,974	13,631	12,984	10,610	11,935	12,169	11,428	9,107	8,743
60.8	12,372	7,586	13,952	13,563	12,929	10,599	11,864	12,098	11,399	9,088	8,734
56.2	12,378	7,588	13,913	13,469	12,842	10,578	11,783	11,999	11,356	9,059	8,720
51.8	12,375	7,584	13,858	13,367	12,734	10,543	11,707	11,870	11,301	9,023	8,704
47.2	12,359	7,572	13,778	13,244	12,580	10,490	11,657	11,762	11,231	8,991	8,692
42.8	12,323	7,545	13,663	13,069	12,328	10,413	11,724	11,587	11,147	9,002	8,694
38.2	12,256	7,495	13,521	12,922	12,038	10,312	12,275	11,458	11,064	9,105	8,717
33.8	12,134	7,404	13,361	12,853	11,833	10,182	12,492	11,448	10,990	9,379	8,764
29.2	11,917	7,246	13,135	12,776	11,623	9,988	12,683	11,478	10,876	10,062	8,802
24.8	11,540	6,979	12,763	12,563	11,312	9,667	12,561	11,369	10,631	10,402	8,700
20.2	10,911	6,550	12,133	12,100	10,793	9,136	12,106	10,986	10,145	10,191	8,344
15.8	9,910	5,893	11,095	11,207	9,897	8,289	11,191	10,173	9,287	9,561	7,649
11.2	3,400	4,936	9,479	9,688	8,479	7,012	9,646	8,776	7,921	8,244	6,505
6.8	6,247	3,624	7,112	7,361	6,392	5,201	7,292	6,498	5,920	6,196	4,835
2.2 (Bottom)	3,401	1,948	3,903	4,088	3,528	2,824	4,025	3,680	3,234	3,396	2,624

(a) Eighth-core equivalent location.

(b) FLAME-calculated⁶ assembly average burnup, MWd/mtU.

(c) Elevation of the fuel column axial node; based on an active fuel length of 144 in.

Table 3-3. Oconee 1 EOC 2 FLAME-Calculated Axial Burnup Profiles

Fuel assembly ID	1C44	1C30	1B04	1B08	1B10 1B19	1C66 1C73 1C76	1C56	1B29 1B30	1D40	1B05	1C63	1C01
Location ^(a)	H-12	H-14	K-9	K-12	K-13	K-14	L-11	L-13	M-11	M-12	M-13	N-12
Burnup ^(b)	19,289	15,913	19,777	18,722	17,791	18,485	18,678	20,022	13,074	19,743	16,603	17,288
Elevation, in. ^(c)												
141.8 (Top)	4,859	3,922	5,846	5,732	4,818	4,705	5,743	5,584	3,982	6,045	4,621	4,941
137.2	9,526	7,367	10,685	10,460	8,878	8,772	10,490	10,272	7,273	11,038	8,509	9,075
132.8	13,253	10,172	14,339	14,023	12,033	11,928	14,065	13,918	9,720	14,806	11,525	12,234
128.2	16,076	12,363	16,948	16,574	14,393	14,428	16,605	16,667	11,419	17,498	13,762	14,514
123.8	18,017	14,036	18,750	16,360	16,193	16,394	18,345	18,738	12,554	19,358	15,388	16,105
119.2	19,475	15,305	19,979	19,609	17,804	17,927	19,510	20,290	13,297	20,622	16,554	17,195
114.8	20,412	16,221	20,805	20,444	18,814	19,030	20,277	21,364	13,783	21,476	17,377	17,964
110.2	21,020	16,863	21,354	20,994	19,475	19,792	20,775	20,078	14,107	22,057	17,957	18,629
105.8	21,416	17,307	21,714	21,351	19,892	20,310	21,099	22,547	14,334	22,438	18,348	19,025
101.2	21,677	17,615	21,946	21,584	20,154	20,664	21,312	22,859	14,505	22,687	18,613	19,274
96.8	21,854	17,835	22,095	21,731	20,314	20,910	21,458	23,072	14,647	22,851	18,797	19,439
92.2	21,982	18,001	22,188	21,825	20,408	21,087	21,564	23,220	14,774	22,960	18,930	19,552
87.8	22,080	18,133	22,243	21,879	20,453	21,217	21,644	23,325	14,894	23,025	19,032	19,634
83.2	22,161	18,245	22,271	21,897	20,459	21,318	21,706	23,402	15,007	23,052	19,113	19,694
78.8	22,228	18,343	22,287	21,911	20,463	21,400	21,754	23,463	15,112	23,073	19,180	19,743
74.2	22,282	18,432	22,292	21,908	20,452	21,468	21,784	23,504	15,199	23,071	19,233	19,780
69.8	22,320	18,512	22,282	21,865	20,414	21,520	21,785	23,519	15,258	23,027	19,268	19,797
65.2	22,336	18,583	22,262	21,791	20,365	21,558	21,743	23,494	15,276	22,945	19,279	19,789
60.8	22,326	18,646	22,227	21,654	20,287	21,580	21,636	23,410	15,238	22,795	19,261	19,748
56.2	22,286	18,700	22,171	21,440	20,169	21,581	21,462	23,264	15,138	22,561	19,209	19,665
51.8	22,216	18,743	22,105	21,161	20,030	21,561	21,218	23,056	14,976	22,259	19,125	19,542
47.2	22,118	18,772	22,040	20,769	19,860	21,520	20,877	22,750	14,761	21,854	19,015	19,386
42.8	22,010	18,782	22,046	20,294	19,646	21,458	20,519	22,410	14,547	21,399	18,907	19,249
38.2	21,909	18,760	22,435	19,898	19,489	21,389	20,253	22,144	14,386	21,076	18,827	19,188
33.8	21,790	18,680	22,546	19,672	19,458	21,311	20,045	21,927	14,271	20,921	18,767	19,259
29.2	21,596	18,493	22,629	19,501	19,553	21,164	19,821	21,687	14,162	20,882	18,675	19,644
24.8	21,219	18,108	22,394	19,255	19,564	20,811	19,475	21,301	13,989	20,583	18,387	19,677
20.2	20,486	17,386	21,737	18,736	19,133	20,060	18,826	20,571	13,621	19,994	17,721	19,109
15.8	19,144	16,125	20,401	17,667	18,216	19,682	17,607	19,238	12,852	18,759	16,470	17,915
11.2	16,882	14,034	17,988	15,739	16,963	16,361	15,492	16,983	11,395	16,545	14,336	15,661
6.8	13,008	10,742	13,960	12,343	13,453	12,584	12,012	13,194	8,877	12,871	10,974	12,031
2.2 (Bottom)	7,340	6,005	7,909	7,036	7,697	7,066	6,782	7,465	5,019	7,302	6,135	6,752

(a) Eighth-ccre equivalent location.

(b) FLAME-calculated⁶ assembly average burnup, MWd/mtU.

(c) Elevation of the fuel column axial node; based on an active fuel length of 144 in.

Figure 3-1.
Mark B Fuel Assembly

Figure 3-2. Fuel Assembly Cross Section and Numbering Conventions

Figure 3-3. Power History - Oconee 1, Cycle 1

Figure 3-4. Fuel Assembly Locations During Cycle 1 Operation

←
North

Figure 3-5. Power History — Oconee 1, Cycle 2

Figure 3-6. Fuel Assembly Locations During Cycle 2 Operation

4. RESULTS AND DISCUSSION

The examination of the 16 assemblies comprising the end-of-cycle (EOC 2) NDT phase was conducted in two parts. The first part was an examination of six assemblies scheduled to be reinserted in Oconee 1 for cycle 3, and was performed during the refueling outage in February 1976. The second portion was the examination of 10 assemblies that were discharged at EOC 2, and was completed in July 1976. Of the 16 assemblies in the program, several had been examined previously at EOC 1.¹ A summary of the EOC 2 examination scope is given in Table 1-1.

4.1. Visual Examinations

All 16 fuel assemblies were extensively examined with underwater television and/or optical periscope. Videotaping and 35-mm photography were used to record areas of interest.

The overall appearance of the assemblies was quite good, showing no evidence of significant rod bowing, cladding deformation, cocked grids, or rod defects. Fuel rod crud patterns were generally similar to those seen in the EOC 1 examination¹ of 36 assemblies, but the EOC 2 patterns were not as pronounced, i.e., the contrast or variation between the different areas was less noticeable. One new feature seen in the EOC 2 examination was thin crud film in the process of peeling off the rod. This feature occurred on several of the rods and was generally restricted to the upper one-third of the assembly. The peeling crud had the appearance of very thin and shiny foil. It was obviously quite flexible, since the adherent layer could often be seen flapping back and forth in the convective current. Beneath the shiny foil was a whitish surface of low reflectivity.

4.2. Fuel Assembly and Rod Length

Fuel assembly length and intermediate spacer grid locations were measured on all four faces of six fuel assemblies and on two faces of assembly 1D40 using a measuring tape and the installed optical periscope. Fuel assembly lengths

and spacer grid locations were measured to an accuracy of ± 0.043 in. and ± 0.040 in., respectively. All of the measurements were corrected for tape stretch and thermal expansion.

Individual fuel rod lengths were similarly obtained by measuring the distance from the visible tops of the rods to a reference position on the lower end fitting. These measurements were taken on 25 fuel rods which were located in peripheral lattice positions of four fuel assemblies which had their upper end fittings modified to contain a "window" for viewing the rod tops. Considering the parallax in sighting along the fuel rod tops, the estimated accuracy of the rod length measurements was ± 0.072 in.

Three types of growth data were obtained from these length measurements: (1) fuel assembly (guide tube) growth, (2) assembly average fuel rod growth, and (3) individual fuel rod growth. Fuel assembly and individual fuel rod growth results were obtained directly from the corrected length measurements by subtracting the nominal as-built lengths. Assembly average fuel rod growth results were obtained indirectly from the corrected grid location measurements. Since the fuel rods are restricted from moving relative to the spacer grids by the friction forces between the rods and grids, the change in grid locations (measured minus the nominal as-built values) provides a measure of the average growth of the fuel rods in the assembly. The assembly average fuel rod growth data obtained by this method is considered to be more accurate than that obtained from the individual fuel rod length measurements because of less parallax error and the greater reproducibility of the grid location measurements.

4.2.1. Fuel Assembly Growth

The results of the fuel assembly length measurements, expressed in terms of fuel assembly growth, are summarized in Table 4-1. The percent fuel assembly growth, $\% \Delta L/L$, was determined by dividing the measured growth by the nominal as-built guide tube length. For the six fuel assemblies examined after two cycles of exposure, fuel assembly growth averaged $0.22\% \Delta L/L$. The average burnup of these six assemblies was 18,300 MWd/mtU, which corresponds to a fast neutron fluence ($E_n > 1$ MeV) of 3.06×10^{21} n/cm². These results and the results obtained from the Oconee 1 EOC 1 poolside examination are displayed in Figure 4-1. As indicated by this figure, fuel assembly growth was less than $0.26\% \Delta L/L$ after two cycles of operation (having increased approximately linearly with fluence).

4.2.2. Fuel Rod Growth

The individual fuel rod and assembly average fuel rod growth data are summarized in Table 4-2. The individual fuel rod growth results averaged 0.30% $\Delta L/L$ at a rod average fast neutron fluence ($E > 1$ MeV) of 3.02×10^{21} n/cm² (corresponding to a rod average burnup of 18,100 MWd/mtU) for the 25 rods examined. Assembly average fuel rod growth averaged 0.28% $\Delta L/L$ for the six two-cycle assemblies at an assembly average fast neutron fluence of 3.06×10^{21} n/cm² (burnup of 17,600 MWd/mtU). These results and the results of the Oconee 1 EOC 1 poolside examination are shown in Figure 4-2. As seen in this figure, the length measurements indicate a trend of increasing fuel rod growth with fluence. Although there is a large amount of scatter in the data, it is also evident from this figure that the fuel rod growth determined by measuring the individual fuel rods is slightly greater than that determined from the spacer grid location measurements. The scatter in the data is attributed both to measurement inaccuracies and to individual differences in fuel rod growth.

4.3. Fuel Assembly Bow

Fuel assembly bow data were obtained from 10 fuel assemblies during the Oconee 1 EOC 2 PIE. All but one (1D40) had two cycles of exposure. The nine 2-cycle assemblies had an average EOC 2 burnup of 18,400 MWd/mtU. Fuel assembly bow is defined as the deviation of the assembly centerline from a line connecting the centers of the upper and lower end fittings. Assembly bow data were obtained from measurements of the lateral displacement of each spacer grid in the plane of each of the four faces. The individual measurements were combined vectorially to obtain bow magnitude and direction at each spacer grid location.

4.3.1. Results

Fuel assembly bow generally occurred as a 0.10 to 0.33 in. lateral displacement of the assembly in a direction outward from the core center. Figure 4-3 shows a typical example of assembly bow measured at the axial elevations of fuel assembly 1B19. For this assembly the maximum bow occurred at grid 4 with a magnitude of 0.175 in. and was directed -127° radially from reference face A (north side) of the assembly. The direction and magnitude of the maximum bow vectors for each assembly are listed in Table 4-3. Assembly bow maxima ranged from 0.100 to 0.328 in. and averaged 0.198 in. for the 10 assemblies. The maximum bow of each assembly occurred near the middle of the assembly at the third or fourth spacer grid; in general, it was directed diagonal to the assembly

axes and radially outward from the core. The axial distribution of assembly bow did not exhibit a strong correlation with the axial burnup profile.

The maximum bow vectors are shown in Figure 4-4 as a function of the core location of the respective assemblies. Also shown in this figure are the bow vectors determined for the assemblies examined at the end of cycle 1¹, three of which were also examined at the end of cycle 2. The bow reversed direction from cycle-to-cycle in two of the three fuel assemblies (1C30, 1C63, 1C73) that were cross-core shuffled, without being rotated, between operating cycles.

4.3.2. Discussion

Since the tight assembly spacing in-core serves to restrain the fuel assemblies, thereby reducing the in-core bow magnitudes, the fuel assembly bow measured out of core does not correspond directly to the in-core bow. The measured bow is, however, representative of the fuel assembly distortion which may be encountered when the assembly is inserted or withdrawn from the core during fuel handling operations.

4.4. Water Channel Spacing

Water channel spacing measurements were taken on nine fuel assemblies to obtain an indirect measurement of fuel rod bow. Eight of the assemblies had been exposed for two cycles of irradiation, achieving a mean assembly average burnup of 18,500 MWd/mtU for the group. The ninth assembly (1D40) had been exposed for one cycle.

The spacing measurements were obtained, as described in reference 3, using a leaf spring probe with strain gages attached. The probe was inserted between rods in two orthogonal directions at the midspan level between spacer grids. Thus, 420 spacings (352 rod-to-rod, and 68 rod-to-guide tube and instrument tube) were characterized at each midspan level per assembly. These measurements were obtained with a precision (1σ) of ± 0.00007 in.

A measure of fuel rod bow was obtained by correcting the water channel spacing data for the effects of the measurement probe, which spreads the rods slightly as it measures the spacings, and computing the standard deviation of the corrected data for each midspan level of the respective fuel assembly. Fuel rod bow, expressed as the standard deviation (σ) of the corrected spacings, was converted to percent water channel closure (one-sided 95% UTL) using the expression:

$$\% \text{ closure} = (1.645 \sigma / 0.138) \times 100\%.$$

The nominal pre-irradiation water channel spacing was 0.138 in. The percent water channel closure calculated using this expression is a conservative estimate of statistical fuel rod bow (σ). The percent closure represents the one-sided 95% upper tolerance (or confidence) level (UTL) of the bow magnitude (σ), below which all except 5% of the bow values are expected to fall. The 1.645 factor is based on the assumption of a normally distributed population of infinite size.

The data obtained from the EOC 2 spacing measurements were reduced and averaged by assembly and by midspan level. The results are summarized in Table 4-4 in terms of average and maximum (worst-span) closure. Water channel closure averaged 12.3% for the eight 2-cycle assemblies and 10.9% for assembly 1D40. Worst-span closure for the eight assemblies averaged 14.4%. As indicated in Table 4-4, the worst-span closure occurred slightly more frequently towards the bottom of the assemblies at midspan level 12. The maximum bow (σ) observed for any one midspan level was 0.0140 in., which corresponds to a worst-span closure of 16.7%. This closure occurred in assembly 1C63 at midspan level 12, which is near the bottom of the assembly at the axial midspan level between spacer grids 5 and 6. The mean water channel spacing determined at this level for assembly 1C63 was 0.1398 in.

The axial distribution of water channel closure is shown in Figure 4-5. The values plotted are the averages determined at each midspan level for the five fully-characterized, 2-cycle, assemblies (which had a group average burnup of 18,800 MWd/mtU). It is apparent in this figure that the closure is greater in the lower portion of the fuel assemblies, below grid 2.

The worst-span water channel closure is shown in Figure 4-6 as a function of assembly average burnup. Also, included in this figure are the results obtained from the EOC 1 poolside examination. The increase both in closure and scatter at the higher burnups is apparent.

4.5. Fuel Rod Diameter

Diameter profilometry traces were obtained for 35 different fuel rods located on the periphery of six fuel assemblies. On five of these assemblies, the four corner rods were profiled in two orthogonal directions. Thus, a total of 55 profilometry traces were obtained. Assembly 1C56 was characterized more extensively than the others since it was selected for a subsequent

destructive examination in the LRC hot cells. Three rods (5, 8, and 11) on each face of this assembly were profiled in addition to the four corner rods. The rod average burnups of each of the 35 rods examined ranged from 15,000 to 21,200 MWd/mtU.

The poolside profilometry traces were single-orientation line scans which measured the diameter of the rod in the plane of the assembly face (described in reference 3). Although a continuous motion was used to trace the diameter of a rod along 132 in. of its length, data were obtained only on the segments of the fuel rod exposed between the spacer grids. Based on an analysis of variance conducted on 335 previous sets of data, the probable error in the diameter measurements is ± 0.0001 in.

The diameter data obtained from the poolside profilometry traces were digitized at axial intervals of 1.0 or 0.8 in. The digitized data were then treated statistically to obtain a mean diameter and its standard deviation (σ) for each grid-to-grid segment of the fuel rod. Creepdown values were calculated over these segments using the nominal pre-irradiation fuel rod diameter of 0.430 in. A relative measure of the cladding ovality was calculated from the amplitude variations in the diameter measurements. This statistically calculated ovality, termed "average maximum ovality," is defined as the 4σ range of the mean diameter, i.e., 95% confidence interval. While this is not the conventionally defined ovality ($OD_{\max} - OD_{\min}$) at a given transverse plane, the average maximum ovality (4σ) does provide a useful means for comparing the single-orientation line scan data. Reduced profilometry traces of the corner rods of each assembly were also redrawn by minicomputer, overlaying the orthogonal traces on the same axes for comparison. Reduced traces of the corner rods of assembly 1B19 are shown in Figure 4-7 as an example.

The interpretation of the single-orientation profilometry traces is illustrated in Figure 4-8. This figure shows a portion of a typical profilometry trace obtained during the EOC 1 destructive examination², in which a step-scan trace is compared with a zero-degree orientation trace. The zero-degree scan is analogous to the single-orientation line scan taken at poolside. In the step-scan mode, the diameter profilometry head is rotated 180° about the fuel rod at each axial step (0.25 in.) to obtain a well-defined ovality envelope along the length of the rod. In Figure 4-8 the zero-degree trace appears as a continuous solid line which wanders within the ovality envelope of the rod and which corresponds to either the maximum or the minimum diameter at various

axial locations. The average diameter, shown by the dashes through the center of the envelope, remains relatively constant along the rod segment. Thus, the diameter variations in the single-orientation traces are seen to result from the ovalities along the length of the rod. It should also be noted that the occurrence of non-spiraling ovalities over a large section of a rod will tend to bias the average diameter and ovality values obtained from a single-orientation trace. Therefore, it is more accurate to determine the rod average diameter and average maximum ovality values from the combined orthogonal scans of the corner rods.

4.5.1. Cladding Creepdown

A summary of the average diameter and creepdown values for the corner rods of the five fuel assemblies is given in Table 4-5. The average creepdown values are shown as a function of rod average burnup in Figure 4-9. The creepdown results obtained from the EOC 1 destructive (hot cell) examination of assembly 1A16 fuel rods² are also included in this figure for comparison. Diameter profilometry measurements were not obtained during the Oconee 1 EOC 1 nondestructive (poolside) PIE. Although a large amount of scatter is seen in the poolside data, a general trend of increased creepdown with burnup is evident. The average diameter of the 35 rods examined from six fuel assemblies at the EOC 2 was 0.4273 in. for an average creepdown of 0.0027 in. This average value is the same as that for the 20 corner fuel rods tabulated in Table 4-5.

Average creepdown as a function of grid span location was also determined. A distinct axial variation in creepdown was noted. The averages for all of the diameter data as a function of axial location are shown in Figure 4-10. The minimum observed average creepdown of 0.0019 in. occurred near the top of the fuel column, above the first spacer grid, while the maximum average of 0.0031 in. occurred near the bottom of the fuel column, below grid 5.

4.5.2. Pellet-Clad Contact

The profilometry traces were also examined closely for indications of contact between the fuel pellets and cladding. Pellet-clad contact was inferred from small ripples in the profilometry traces which approximately coincided with the periodicity of the fuel pellet interfaces, as shown in Figure 4-11. These ripples were more apparent in the regions of low cladding ovality and correlated reasonably well in both orthogonal traces of the respective corner rods.

The typical peak height of the ripples was in the range of 0.00075 to 0.0010 in.

Inferences of pellet-clad contact were noted in 12 of the 35 rods examined. The axial locations of the contact areas are listed in Table 4-6. The majority of the ripples were located near the upper end of the rods, above the second spacer grid. The clearest indication of contact was observed between grids 1 and 2 of the A-D corner fuel rod in assembly 1B19. The orthogonal traces of this rod segment, overlayed for comparison, are shown in Figure 4-11.

4.5.3. Cladding Ovality

Relative values of cladding ovality are expressed in terms of the average maximum ovality (4σ) measured between spacer grids. The average maximum ovality values ranged from 0.0012 to 0.0140 in. with an overall mean of 0.0057 in. The statistical averages for the 35 rods examined were calculated as a function of axial location and are shown in Figure 4-12. Note that a bimodal distribution of ovality is indicated. The maximum ovality (≤ 0.0065 in.) appears to have occurred in the axial region between grids 3 and 6. A secondary maximum (≤ 0.0060 in.) is located between grids 1 and 2.

Reduced traces of the corner rod orthogonal scans were redrawn to facilitate identifying and measuring the peak ovality nodes. Histograms displaying the measured peak ovality nodes as a function of axial location are shown in Figure 4-13. It is apparent from this figure that the peak ovalities ($OD_{\max} - OD_{\min}$) occurred most frequently in two locations along the rods — between spacer grids 4 and 5, and between grids 1 and 2. This bimodal distribution is seen both for the various discriminator levels of peak ovality (0.008, 0.010, and 0.012 in.) and for the average maximum ovalities presented previously in Figure 4-12. Note that the peak ovality values obtained at these peak nodes are not necessarily the true peak ovality values in each case because the major and minor axes of the respective oval may not necessarily coincide with the orientation of the profilometry measurements. However, the true peak ovalities cannot be smaller than the corresponding observed peak values. Thus, the measured values indicate the minimum amount of peak ovalization that has occurred in the cladding.

4.6. Gamma Scans

Full-length axial gamma scan profiles were obtained (as described in reference 3) on each of the four corner rods of six fuel assemblies. Each profile was a continuous plot of gamma count rate ($E_\gamma > 0.2$ MeV) as a function of axial distance along a rod. From these profiles fuel column axial gap length and fuel stack length data were obtained for the respective rods.

The profiles analyzed were those obtained from slow scans (1.8 to 4.0 in./min.) taken while lowering the fuel assembly past the gamma-ray detector. The vertical slit width of the detector's collimator was typically set between 0.103 and 0.301 in. for this examination. Axial distance along the length of the fuel rod was determined more accurately than at the EOC 1 by correlating the measured spacer grid separation distance with that determined from the gamma scan traces. Fuel stack lengths were measured to an accuracy of within ± 0.1 in., and fuel column axial gap lengths of 0.05 in. or greater were easily resolved from normal fuel pellet interfaces. This improved method and the re-alignment of the assembly guide frame (to reduce gamma interference from the storage rack) considerably increased the accuracy and resolution of the gamma scan profiles in comparison to the profiles obtained at the EOC 1.

4.6.1. Fuel Column Gap Lengths

The axial distribution of fuel column gaps (gap length versus location in the fuel column) was determined from the gamma scan profiles of the 24 rods scanned. This distribution is shown in Figure 4-14. The minimum gap length measured was 0.1 in. A total of 64 gaps were found which were approximately 0.1 in. or greater in length. Of these gaps, 22 were 0.25 in. or greater. The longer gaps were distributed among the 24 rods as follows:

- 11 rods with no gaps ≥ 0.25 in.
- 7 rods with one gap ≥ 0.25 in.
- 4 rods with two gaps ≥ 0.25 in.
- 1 rod with three gaps ≥ 0.25 in.
- 1 rod with four gaps ≥ 0.25 in.

The frequency distributions of size and location for all of the gaps detected (≥ 0.1 in.) are shown in Figure 4-15. Approximately 60% of all gaps were located within 12 in. from the top of the fuel column. All of the larger gaps (≥ 0.25 in.) were located within 20 in. from the top of the fuel column. The distribution of total fuel column gap length per rod is shown in Figure 4-16. Note that eight of the 24 rods contained no gaps of 0.1 in. or greater.

4.6.2. Fuel Stack Lengths

Fuel stack lengths were determined from the respective gamma scan profiles and corrected by subtracting the total fuel column gap lengths measured for each rod. The corrected fuel stack lengths are given in Table 4-7. Figure 4-17 displays the change in stack length as a function of burnup, compared to the curve of predicted stack length change. The change in stack length is based on the nominal pre-irradiation stack length of 144 in. The predicted length change assumes an initial fuel density of 93.5% theoretical density (TD) and a peak densification of 96.5% TD occurring at 4,000 MWd/mtU. The average burnup of the 24 rods scanned was 17,800 MWd/mtU. The average decrease in stack length for these rods was 0.73% $\Delta L/L$, which agrees favorably with the predicted decrease of 0.92% $\Delta L/L$ at a rod average burnup of 17,800 MWd/mtU. The large spread in the data is attributed to the fact that the initial core of Oconee 1 contained 65 different fuel lots.

4.6.3. Discussion

A comparison of the EOC 1 and 2 gamma scan results indicates a major difference in the axial distribution of the fuel column gaps. At the EOC 1, the larger gaps (≥ 0.25 in.) were found at various axial locations along the rods. However, the EOC 2 data show no gaps of this size below the top 20 in. of the fuel columns. This comparison reinforces the conclusion expressed previously in references 1 and 2 that many of the gaps found during the EOC 1 examinations, particularly the larger gaps found near the bottom of the rods, resulted from the fuel handling and transfer operations conducted after the first cycle of irradiation. (The fuel assemblies are transferred from the containment building to the spent fuel storage pool in the horizontal position.) Many of the gaps thus formed were subsequently closed after the assemblies were returned to the core for the second cycle of irradiation.

Sixteen of the fuel rods examined at the EOC 2 were examined previously at the EOC 1. Although the EOC 1 results are slightly less accurate than those obtained at the EOC 2, a one-to-one comparison of the gaps showed that, for 24 gaps at EOC 1 and 36 gaps at EOC 2, 13 correlated between cycles. Of these 13, 10 were in the top 12 in. of the fuel column. Two gaps apparently increased in size, while four decreased and seven did not change measurably. The uncorrelated gaps exhibited similar axial distributions below the top 12 in. of the fuel column. However, within the top 12 in. there was a substantially higher number of gaps present at the EOC 2 (11 versus 2).

A possible scenario for this gap behavior is given below. During the first cycle of irradiation, some gaps were formed throughout the fuel column as the result of fuel densification and pellet hang-up. However, the majority of the gaps, particularly those near the bottom of the fuel column, were formed during the EOC 1 fuel transfer and handling operations. In the early stages of cycle 2 irradiation, the majority of the gaps near the bottom of the fuel column were closed by pellets settling as the result of flow-induced vibration and the weight of the fuel stack. Throughout cycle 2, cladding creepdown and fuel pellet swelling increased with greater creepdown occurring near the bottom of the fuel rod. At the EOC 2, the pellets near the bottom of the rod were more firmly held in place than those near the top. Thus, the EOC 2 handling operations resulted in the formation of gaps primarily near the top of the fuel column, and a top-heavy gap frequency distribution resulted.

4.7. Holddown Spring Force

Holddown spring load-deflection curves were obtained (as described in reference 3) for five fuel assemblies which were discharged from the Oconee 1 reactor after two cycles of irradiation. Values of spring preload force and spring rate were determined from the loading portion of the curves with an uncertainty of within ± 5 lb and $\pm 5\%$, respectively.

4.7.1. Results

A summary of the holddown spring preload force measurements is given in Table 4-8. Average values for the five assemblies were 397 lb and 183 lb/in. for the preload force and spring rate, respectively. Burnups for these five assemblies averaged 19,200 MWd/mtU. These measurements indicate an average decrease in preload force of 5% from the beginning-of-life (BOL) nominal mean of 418 lb. The results are well within the BOL tolerance range (2σ) of 350 to 495 lb. Also, the slight difference in spring rate of -2.1% from the BOL nominal mean of 187 lb/in. (for the entire spring population) is well within $\pm 5\%$ accuracy of the measurements, indicating no significant change after two cycles of irradiation.

4.7.2. Discussion

The preload force measurements obtained at the EOC 2 are compared in Figure 4-18 with those obtained for other assemblies at the EOC 1¹. A linear least

squares fit of this data extrapolates to a preload force of 385 lb at an assembly average burnup of 30,000 MWd/mtU, which is within the BOL tolerance range.

4.8. Crud Sampling

Cladding crud samples were taken using a steel-tipped scraper and vacuum pump apparatus identical to that used in the EOC 1 examination (reference 1). As before, crud flakes were collected separately on filters for flake thickness measurements, and the area of each sample was estimated from photographs.

A total of eight fuel assemblies were sampled for crud. Five of the assemblies were sampled only once, at the center between spacer grids 3 and 4. For the other three, several samples were taken at various locations to determine any axial trends or patterns. A total of 26 samples plus spent fuel pool water background samples were collected and analyzed. Isotopic analysis of the latter showed a negligible contribution from the pool water.

4.8.1. Crud Thickness

The thickness of the crud layer was estimated by two methods. The first was a direct microscope measurement on the crud flakes separated from some of the samples. The second was a calculation based on sample weight and sample area. Details concerning the two methods can be found in reference 1.

A total of 15 flake specimens were measured for thickness using the microscope technique. These specimens covered six different assemblies and included full axial profiles on two of the assemblies. The flakes varied in thickness from 0.25 to 0.89 mils, with an average of 0.51 mils. In contrast, the EOC 1 flakes averaged 1.03 mils, with a range of 0.73 to 1.35 mils. The EOC 2 data showed no strong correlation between thickness and axial location on the assembly.

A crud specific thickness (mg/cm^2) was also determined using another approach. The total sample weight was determined by chemical analysis on the major metallic species and a knowledge of the oxide forms. The weights were then divided by the sampled area. The 21 samples calculated by this method averaged $0.58 \text{ mg}/\text{cm}^2$, although the range was fairly large. This compares with an average of $1.1 \text{ mg}/\text{cm}^2$ for EOC 1 samples.

4.8.2. Crud Composition

Chemical and radiochemical analyses were conducted to determine the composition of the crud samples. X-ray fluorescence showed the average metallic composition to be 39% Ni, 55% Fe and 6% Cr (by weight). This compares with 57% Ni, 40% Fe and 3% Cr for EOC 1 samples. No consistent axial trend was seen in the crud composition data. The only compound identified (through X-ray diffraction) was nickel ferrite (NiFe_2O_4 , spinel structure). Nickel oxide, a major constituent in EOC 1 samples, was not seen (detectability limit was 5 to 10%). The decrease in the amount of this compound in EOC 2 samples is consistent with the drop in Ni composition found.

Gamma isotopic analysis, utilizing a Ge(Li) detector and multi-channel analyzer, was employed to identify the major radionuclide constituents. The results (normalized to ^{58}Co) are compared with EOC 1 values in the tabulation below.

<u>Isotope</u>	<u>Normalized activity ($^{58}\text{Co}=1$)</u>	
	<u>EOC 1</u>	<u>EOC 2</u>
^{51}Cr	0.29	0.40
^{54}Mn	0.067	0.073
^{59}Fe	0.039	0.054
^{60}Co	0.15	0.13
^{95}Zr	0.084	0.17

Specific activities (disintegrations/s per unit mass of element) were very similar to EOC 1 values.

4.8.3. Discussion

Crud sampling after two cycles showed about a 50% decrease in crud thickness from EOC 1. Crud metallic and radionuclide composition changed slightly, the relative amount of Ni decreased while the relative proportions of Fe and NiFe_2O_4 was the only compound identified.

Table 4-1. Summary of Fuel Assembly Growth Results
for Oconee 1 EOC 2

<u>Fuel assembly</u>	<u>Ass'y avg fluence, ^(b) 10⁻²¹ n/cm²</u>	<u>Assembly growth, % $\Delta L/L$</u>
1C44	3.222	0.257
1C56	3.120	0.226
1C63	2.774	0.191
1C66	3.088	0.218
1C73	3.088	0.231
1C76	3.088	0.212
Average	3.063	0.222
1D40 ^(a)	2.184	0.050

(a) Assembly 1D40 length measurements were taken on only two faces; the tabulated growth value is considered to be in error.

(b) $E > 1$ MeV.

Table 4-2. Summary of Fuel Rod Growth Results for
Oconee 1 EOC 2

Fuel assembly	Ass'y avg fluence ^(d) , 10 ⁻²¹ n/cm ²	Ass'y avg rod growth, ^(a) % ΔL/L	Fuel rod	Rod avg fluence ^(d) , 10 ⁻²¹ n/cm ²	Rod growth, ^(b) % ΔL/L
1C44	3.222	0.317	A5	3.301	0.303
			A11	3.301	0.294
			B5	3.180	0.270
			B11	3.036	0.295
			C5	2.987	0.286
			C11	2.987	0.289
			D5	3.036	0.306
			Average	3.118	0.292
1C56	3.120	0.230	A5	2.890	0.316
			A11	2.647	0.239
			B5	2.650	0.315
			B11	2.878	0.369
			C5	3.040	0.346
			C11	3.237	0.322
			D5	3.304	0.300
			D11	3.130	0.277
1C73	3.088	0.289	Average	2.972	0.310
			A5	3.216	0.32
			A11	3.101	0.32
			C5	2.662	0.28
			C11	2.793	0.28
			D5	3.121	0.29
			D11	3.274	0.31
			Average	3.028	0.30
1C76	3.088	0.275	C5	3.274	0.30
			C11	3.121	0.29
			D5	2.793	0.28
			D11	2.662	0.31
			Average	2.962	0.30
1C63	2.774	0.310	--	--	--
1C66	3.088	0.264	--	--	--
Average	3.063	0.281	--	3.025	0.30
1D40 ^(c)	2.184	0.179	--	--	--

(a) Determined from fuel assembly spacer grid location measurements on four faces.

(b) Determined from individual fuel rod length measurements.

(c) Fuel assembly 1D40 was exposed for only one cycle of irradiation; measurements were taken on two faces.

(d) E > 1 MeV.

Table 4-3. Summary of Oconee 1 EOC 2 Fuel Assembly Bow Results

Fuel assembly	Ass'y avg burnup MWd/mtU	Max. bow magnitude, inches	Max. bow direction ^(a) (θ), degrees	Axial location, grid position
1B08	18,700	0.100	128	3
1B19	17,800	0.175	-127	4
1B29	20,000	0.280	-169	4
1B30	20,000	0.173	-71	4
1C30	15,900	0.098	64	4
1C44	19,300	0.301	159	4
1C63	16,600	0.244	-121	3
1C66	18,500	0.328	-139	4
1C73	18,500	0.115	-127	4
1D40	13,100	0.162	-145	3

(a) Bow direction, θ , is given relative to assembly face A (north):

Table 4-4. Summary of Water Channel Closure Results

Fuel assembly	Ass'y avg burnup, MWd/mtU	Assembly ^(a) average closure, %	Maximum ^(a) channel closure, %	Midspan ^(b) level of max closure	Maximum ^(c) fuel rod bow (σ), in.	Mean ^(d) channel spacing, in.
1B04	19,800	11.8	13.8	8	0.0116	0.1395
1B05	19,700	12.0	13.9	8	0.0117	0.1384
1B29	20,000	11.1	13.0	8	0.0109	0.1395
1C30	15,900	13.0	16.6	12	0.0139	0.1393
1C56 ^(e)	18,700	12.0	12.4	6	0.0104	0.1383
1C63 ^(e)	16,600	13.7	16.7	12	0.0140	0.1398
1C66	18,500	12.8	16.3	12	0.0137	0.1413
1C73 ^(e)	18,500	12.3	12.8	14	0.0107	0.1374
Average	18,500	12.3	14.4	--	0.0121	0.1392
1D40 ^(f)	13,100	10.9	13.7	12	0.0115	0.1379

(a) Statistically, 95% of the water channels have a closure less than the value tabulated.

(b) Midspan levels are evenly numbered from top to bottom between spacer grids, e.g., level 2 is located between the upper end fitting and the first spacer grid.

(c) Expressed as the standard deviation (σ) of the corrected water channel spacings measured at the midspan level of maximum channel closure.

(d) This is the mean of the water channel spacings measured at the midspan level of maximum channel closure.

(e) Measurements were obtained at only two of the seven midspan levels.

(f) Assembly 1D40 was exposed for only one cycle of irradiation.

Table 4-5. Average Diameter and Creepdown of Corner Rods

<u>Fuel assembly</u>	<u>Fuel rod</u>	<u>Rod avg burnup, MWd/mtU</u>	<u>Average diameter, inch</u>	<u>Average creepdown, (a) mils</u>
1B08	A1,D15	17,900	0.4277	2.3
	B1,A15	18,100	0.4277	2.3
	C1,B15	18,700	0.4279	2.1
	D1,C15	17,900	0.4270	3.0
1B19	A1,D15	17,900	0.4282	1.8
	B1,A15	18,100	0.4277	2.3
	C1,B15	19,200	0.4267	3.3
	D1,C15	18,800	0.4257	4.3
1C29	A1,D15	21,200	0.4274	2.6
	B1,A15	19,500	0.4274	2.6
	C1,B15	17,700	0.4279	2.1
	D1,C15	19,400	0.4262	3.8
1C44	A1,D15	19,200	0.4272	2.8
	B1,A15	19,200	0.4271	2.9
	C1,B15	17,600	0.4269	3.1
	D1,C15	17,600	0.4265	3.5
1C56	A1,D15	17,800	0.4275	2.5
	B1,A15	15,100	0.4274	2.6
	C1,B15	17,600	0.4282	1.8
	D1,C15	19,800	0.4278	2.2
	Average	18,400	0.4273	2.7

(a) Calculated using as-fabricated nominal diameter of 0.4300 inch.

Table 4-6. Locations of Possible Pellet-Clad Contact
From Diameter Profilometry Results

<u>Assembly</u>	<u>Rod</u>	<u>Location</u>	<u>Diameter, in.</u>
1B19	AD	G1-G2	0.429
	B15	G2-G3	0.425 (ovality minimum)
	A15	G1-G2	0.426-0.428
1B29	D15	G2-G3	0.424 (ovality minimum)
1B08	None		
1C44	AB	G1-Top	0.426-0.430 (ovality max. and min.)
	A15	G1-G2	0.423-0.427
	D15	G1-Top	0.427-0.428
	D15	G2-G3	0.429
	D1	G1-Top	0.425-0.428
	CD	G4-G3	0.425-0.429 (ovality max. and min.)
	BC	G2-G3	0.424-0.429
1C56 ^(a)	A8	G4-G5	0.428

(a) Most traces of 1C56 were 1/10th scale which do not readily reveal the indications of pellet-clad contact.

Table 4-7. Summary of Gamma Scan Fuel Stack Length Results

<u>Fuel assembly/rod^(b)</u>	<u>Rod avg burnup, MWd/mtU</u>	<u>Corrected stack length, in.</u>	<u>Change^(a) in stack length, % $\Delta L/L$</u>
1B29/AB	19,500	141.9	-1.46
BC	17,700	142.3	-1.18
CD	19,400	141.5	-1.74
AD	21,200	143.1	-0.62
1C01/AB	17,900	142.6	-0.97
BC	17,700	142.2	-1.25
CD	16,700	141.5	-1.04
AD	17,700	142.7	-0.97
1C44/AB	19,200	142.2	-1.25
BC	17,600	142.8	-0.83
CD	17,600	142.7	-0.90
AD	19,200	142.2	-1.25
1C56/AB	15,100	142.9	-0.76
BC	17,600	142.2	-1.25
CD	19,800	142.2	-1.25
AD	17,800	143.9	0.00
1C63/AB	13,300	143.7	-0.21
BC	16,400	143.7	-0.21
CD	17,400	143.6	-0.28
AD	16,300	143.7	-0.21
1C66/AB	17,500	143.9	0.00
BC	19,300	143.5	-0.35
CD	18,100	144.4	0.28
AD	16,100	144.2	0.14
Average	17,800	142.9	-0.73

(a) Based on a nominal pre-irradiation length of 144.0 in.

(b) Designates rod at corner of assembly, e.g., rod AB is the rod at the corner of faces A and B, rod A15/B1 as shown in Figure 3-2.

Table 4-8. Summary of Holddown Spring Preload
Force Measurements

<u>Fuel assembly</u>	<u>Ass'y avg burnup, MWd/mtU</u>	<u>Spring rate, lb/in.</u>	<u>Preload force, lb</u>	<u>Change in preload, (a) % $\Delta F/F$</u>
1B19	17,800	184	385	-7.9
1B29	20,000	180	415	-0.7
1B30	20,000	194	400	-4.3
1C44	19,300	164	385	-7.9
1C56	18,700	191	400	-4.3
Average	19,200	183	397	-5.0

(a) Using a nominal as-built preload force, F, of 418 lb.

Figure 4-1. Fuel Assembly Growth as Function of Assembly Average Fluence

Figure 4-2. Fuel Rod Growth as Function of Average Fluence

Figure 4-3. Axial Distribution of Fuel Assembly Bow

ASSEMBLY 1B19:
BOW DIRECTION AND
MAGNITUDE

$\theta = 127^\circ$

Figure 4-4. Fuel Assembly Bow as Function of Assembly Lattice Position

NOTE: The bow vectors are shown in the lattice positions of their respective fuel assemblies.

Figure 4-5. Axial Distribution of Average Water Channel Closure

(a) STATISTICALLY, 95% OF THE FUEL RODS HAVE A CLOSURE LESS THAN THE PLOTTED VALUE. NOTE THAT THE TOP DATA POINT WAS MEASURED ~3 IN. BELOW MIDSPAN LEVEL 2.

Figure 4-6. Worst-Span Water Channel Closure Versus Assembly Average Burnup

Figure 4-7. Orthogonal Traces for Corner Rods on Assembly 1B19

Figure 4-8. Example of Step-Scan Profilometry With 180° Rotation at Each Step

(a) The zero degree orientation scan is analogous to the traces obtained at poolside.

Figure 4-9. Creepdown Versus Burnup for EOC 2 Corner Rods

Figure 4-10. Rod Diameter Versus Axial Location for all EOC 2 Data

Figure 4-11. Segment of Traces for AD Corner Rod of Assembly 1B19

(a) Note ripples which occur with a periodicity equal to fuel pellet lengths.

Figure 4-12. Average Rod Ovality Versus Axial Location for all EOC 2 Data

Figure 4-13. Histograms Showing Peak Ovality Locations Along Length of Rods

████████████████████

Figure 4-15. Frequency Distributions of Fuel Column Gaps, Oconee 1 EOC 2

Figure 4-16. Frequency Distribution of Fuel Column
Total Gap Length, Oconee 1 EOC 2

Figure 4-17. Fuel Stack Length Change as Function of Rod Average Burnup

Figure 4-18. Holddown Spring Preload Force

5. CONCLUSIONS

Summarized below are the results of the Oconee 1 EOC 2 poolside examination.

1. Visual examination of 16 fuel assemblies confirmed the structural integrity of the fuel rods and assemblies. No evidence of rod defects was observed.
2. Fuel assembly (guide tube) growth averaged 0.22% $\Delta L/L$ for six assemblies having a group burnup of 18,300 MWd/mtU.
3. Assembly average fuel rod growth averaged 0.28% $\Delta L/L$ for six assemblies having a group average fast fluence ($E > 1$ MeV) of 3.06×10^{21} n/cm². Individual fuel rod growth averaged 0.30% $\Delta L/L$ for 25 fuel rods having an average fast fluence of 3.02×10^{21} n/cm².
4. Maximum fuel assembly bow ranged from 0.10 to 0.33 in. and averaged 0.20 in. The direction of bow was, in general, diagonal to the axes of the assemblies and directed radially outward from the center of the core.
5. Water channel closure (95% UTL), which is a conservative estimate of fuel rod bow, averaged 12.3% for eight assemblies having a group average burnup of 18,500 MWd/mtU. The maximum bow (σ) observed at any midspan level was 0.0140 in., which corresponds to a worst-span channel closure of 16.7%. The average of the worst-span closures was 14.4%.
6. Cladding creepdown and ovality values were obtained from the diameter profilometry traces of 35 fuel rods having rod average burnups ranging from 15,000 to 21,200 MWd/mtU. Creepdown averaged 0.0027 in. A distinct axial variation in creepdown was noted with a minimum average creepdown (~ 0.0019 in.) occurring above the first spacer grid and a maximum average creepdown (~ 0.0031 in.) occurring below the fifth spacer grid.

Ripples in the traces which correlated with the periodicity of fuel pellet interfaces in regions of low cladding ovality implied that pellet-clad contact had occurred in 12 of the 35 rods. The majority of the ripples were located near the upper end of the rods, above the second spacer grid.

Relative values of cladding ovality, expressed in terms of average maximum ovality (4 σ range of the mean diameter), were obtained along the midspan between spacer grids. The average maximum ovalities ranged from 0.0012 to 0.0140 in. with an overall mean of 0.0057 in. The axial variation in ovality displayed a bimodal distribution with maxima of 0.0065 and 0.0060 in. occurring between grids 3 and 6 and between grids 1 and 2, respectively.

7. Gamma scans of the corner rods of six fuel assemblies indicated that the fuel stack length had decreased an average of 0.74% $\Delta L/L$ from BOL. Although there was a considerable amount of scatter in the data due to the various lots of fuel used, the observed results agreed favorably with the predicted stack length decrease of 0.92% $\Delta L/L$.

Fuel column gap lengths and their axial locations were also determined from the gamma scans. A total of 64 gaps (≥ 0.1 in.) were found in the 24 rods scanned; eight of the rods contained no measurable gaps. Approximately 60% of all of the gaps were located within the top 12 inches of the fuel column; all of the larger gaps (≥ 0.25 in.) were located within the top 20 in. A greater percentage of gaps were located near the top of the fuel column at the EOC 2 in comparison to the EOC 1.

8. Holddown spring preload forces averaged 397 lb (5% below the BOL nominal mean) and remained well within the BOL nominal tolerance range. The spring rates averaged 183 lb/in., which is 2.1% less than the BOL nominal value and within the $\pm 5\%$ accuracy range of the measurements. Thus, no significant change in the holddown spring characteristics were noted in the five assemblies examined which had a group average burnup of 19,200 MWd/mtt.
9. Crud flake thickness ranged from 0.00025 to 0.00089 in. and averaged 0.00051 in., a decrease of about 50% from EOC 1. Crud specific thickness averaged 0.58 mg/cm². No strong correlation between thickness and axial location was observed. The metallic composition of the crud was composed of 55% Fe, 39% Ni, and 6% Cr (by weight); the only compound identified was NiFe₂O₄. The major radionuclide specific activities and compositions were similar to those observed at the EOC 1.

6. REFERENCES

- ¹ Post-Irradiation Examination of Oconee 1 Fuel - Cycle 1 Nondestructive Test Phase, BAW-1519, Babcock & Wilcox, Lynchburg, Virginia, June 1979.
- ² Post-Irradiation Examination of Oconee 1 Fuel - Cycle 1 Destructive Test Phase, BAW-1535, Babcock & Wilcox, Lynchburg, Virginia, July 1979.
- ³ Nondestructive Examination of Oconee 1 Fuel Assemblies After Three Cycles of Irradiation, C00-4711-20 (BAW-1553), Babcock & Wilcox, Lynchburg, Virginia, September 1979.
- ⁴ Oconee Nuclear Station, Units 1, 2, and 3 - Final Safety Analysis Reports, Docket Nos. 50-269, 50-270, and 50-287, Duke Power Company.
- ⁵ Oconee 1, Cycle 2 - Reload Report, BAW-1409, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, October 1974.
- ⁶ C. W. Mays and M. Furtney, FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions, BAW-10124A, Babcock & Wilcox, Lynchburg, Virginia, August 1976.

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