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POST-IRRADIATION EXAMINATION OF OCONEE 1 FUEL
— CYCLE 1 DESTRUCTIVE TEST PHASE —

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Babcock & Wilcox
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Report BAW-1535

July 1979

Post-Irradiation Examination of Oconee 1 Fuel -
Cycle 1 Destructive Test Phase

Key Words: UO₂ Fuel, Zircaloy-4, Irradiation
Testing, Oconee 1

ABSTRACT

Standard B&W Mark-B (15 × 15) pressurized water reactor fuel rods were destructively examined after one cycle of irradiation in the Oconee 1 reactor. Fuel rod average burnup ranged from 10,603 to 11,270 MWd/mtU for the rods examined. Data obtained included fuel rod extraction loads, rod dimensional changes, cladding tensile properties, fuel pellet gap length, fission product distribution, fission gas and crud composition, fuel densification, chemical burnup analysis, and fuel and cladding microstructure. As expected, parametric changes were well within the design envelope. Superficial corrosion and wear were found at spacer grid contact points. However, the 19 rods examined were structurally sound and exhibited no indications of cladding defects associated with pellet-cladding interactions. Both plate- and pin-like (or spherical) forms of hydride precipitation were observed. The cladding exhibited a 37% increase in yield strength while retaining 69% of its original ductility.

SUMMARY

The end-of-cycle (EOC) 1 post-irradiation destructive examination of Oconee 1 fuel was conducted at B&W's Lynchburg Research Center. Nineteen fuel rods were extracted from the central core fuel assembly, which had experienced a calculated burnup of 10,852 MWd/mtU. The 19 extracted rods were individually examined using nondestructive methods. Three of these rods were selected for additional destructive sectioning and examination. Data were obtained in the following areas:

Nondestructive — Fuel rod extraction loads; visual surface appearance; fuel rod length and diameter; cladding wall thickness and eddy-current indications; fuel pellet gap length, fuel stack length, and gross burnup; fission product distribution; crud characteristics; fission gas composition.

Destructive — Fuel density and burnup, fuel microstructure, cladding tensile properties, cladding hydrogen content, cladding microstructure and inner-surface (ID) features.

Fuel rod extraction loads indicated that the average grid spring force was within the range expected. The general surface appearance of the fuel rods was quite good after extraction. Fuel rod axial growth was well within the design envelope with an average axial growth of 0.135%. An average rod ovality of 0.004 inch was determined, rod creepdown averaged 0.002 inch, and an apparent average decrease of 0.0003 inch in wall thickness was noted. The number of cladding eddy-current indications was large; however, thorough metallographic examinations revealed no structural damage in the cladding other than superficial corrosion and wear at the grid contact points. Fuel pellet gap sizes and locations were variable, having been affected somewhat by fuel handling. Fuel stack length decreased an average of 1.14% $\Delta L/L$. No unexpected trends were observed in the fission product distributions. Crud thickness was 70 to 90% less than anticipated from site poolside examinations; the crud could have been sloughed off during handling and storage. Fission gas release was quite small, averaging only 0.67% by volume.

Fuel pellet densification averaged 2.8% but was somewhat variable with the lower density fuel pellets exhibiting greater densification. Slight gradients in densification and grain growth were observed within the fuel pellets. No indications of cladding defects associated with fuel-cladding interactions were observed. Both plate- and pin-like (or spherical) forms of hydride precipitates were found with highly variable axial and circumferential distributions. Hydrogen concentrations within the cladding varied from 55 to 257 ppm with no distinct distribution pattern. The cladding exhibited a 37% increase in yield strength while retaining 69% of its original ductility.

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1. INTRODUCTION

The B&W Mark-B 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 continuously 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 testing of the fuel assemblies. Nondestructive testing (NDT) is performed at the Oconee site in the Oconee 1 and 2 spent fuel storage pool. Destructive examination is performed in B&W's hot cell laboratory at the Lynchburg Research Center (LRC). The results of the extensive EOC 1 NDT (poolside) phase were summarized previously.¹ This report summarizes the results of the EOC 1 destructive examination.

During the fall of 1975, a fuel assembly was transferred from the Oconee site to the LRC for post-irradiation destructive examination. Fuel assembly 1A16 was selected for this initial destructive examination because it was special in two aspects: (1) it operated in the central core position, and (2) an extensive characterization data package was generated during its fabrication. Prior to being returned to the LRC, this fuel assembly had also been subjected to a thorough NDT examination during the EOC poolside operations at Oconee.¹

Upon receipt at the LRC hot cell laboratory, fuel assembly 1A16 was examined visually and 19 fuel rods were extracted. Each of these extracted rods was subject to the following additional nondestructive examinations:

1. Extraction load determination.
2. Visual surface examination.
3. Length measurement.
4. Axial gamma-scans for fuel stack length, fuel pellet gap length, gross burnup profile determinations, and isotopic distribution.
5. Diameter, creepdown, and ovality measurements.
6. Cladding wall thickness measurement.
7. Eddy-current measurement.
8. Crud analysis.
9. Fission gas analysis.

Of the 19 fuel rods extracted, three were selected for destructive sectioning at the LRC hot cell laboratory. The destructive examinations conducted at the LRC included the following:

1. Fuel burnup analysis.
2. Fuel density measurement.
3. Fuel metallographic examination.
4. Cladding metallographic examination.
5. Cladding hydrogen analysis.
6. Cladding inner surface examination.
7. Cladding tensile testing.

The results of these hot cell examinations, along with the relevant fuel assembly design parameters and operating characteristics, are summarized and discussed in the following sections of this report.

2. MARK-B FUEL DESIGN AND CORE CHARACTERISTICS

A cutaway version of a typical Mark-B fuel assembly is shown in Figure 1. For reference, the six intermediate spacer grids (Inconel-718) are numbered sequentially from 1 to 6 from top to bottom. The structural cage of the assembly consists of 16 control rod guide tubes (Zircaloy-4) permanently attached to the upper and lower end fittings; the spacer grids form the structural link between the guide tubes and the Zircaloy-4-clad fuel rods but are free to move with the fuel rods as the rods grow. During fuel rod insertion, the grids are keyed open to prevent mechanical damage to the surface of the rods or to the grids. After positioning, the grids are unkeyed, holding the fuel rods securely in their lattice position.

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 incore instrumentation tube. The remaining 208 lattice positions contain fuel rods. The fuel is UO_2 in the form of dished-end cylindrical pellets. Figure 2 is a schematic cross section of fuel assembly 1A16, which was selected for destructive examination. The locations of the 19 fuel rods extracted for examination are indicated in the figure.

Nominal design parameters and operating conditions of Oconee 1 cycle 1 fuel are summarized in Table 1. During this cycle, fuel assembly 1A16 operated in the central core position (as shown schematically in Figure 3) and experienced an average calculated burnup of 10,852 MWd/mtU. This figure also displays the locations of the other fuel assemblies examined at the end of cycle 1.¹

Table 2 lists the average relative power levels at various times during the cycle for each of the 19 fuel rods extracted from fuel assembly 1A16. The average power distribution across the assembly was relatively flat except for that period of time during which the transient control rod assembly (CRA) was inserted (from approximately 92 to 196 EFPD). During the remainder of the cycle, this fuel assembly contained a safety CRA, which was essentially 100% withdrawn at all times.

The calculated rod average burnup values for the 19 extracted fuel rods are listed in Table 3. These values were obtained from two-dimensional PDQ² calculations and ranged from 10,630 to 11,272 MWd/mtU. The corresponding PDQ-calculated burnup was 10,852 MWd/mtU for assembly 1A16, which agreed favorably with the burnup values obtained from incore detector monitoring (11,105 MWd/mtU) and FLAME³-calculated (10,530 MWd/mtU) axial burnup profiles for this assembly. The axial profile representing the calculated (FLAME) cycle 1 burnup distribution for assembly 1A16 is shown in Figure 4. In this figure, the assembly burnup is normalized to the peak axial burnup of 13,430 MWd/mtU.

The average linear heat rates of the 19 extracted fuel rods were estimated to a first approximation by ratioing the rod average burnup and core average heat rate. The values calculated are given in Table 4. The calculated average heat rate for these rods was 6.32 kW/ft during the first cycle. The average of the peak linear heat rates for these rods was calculated to be 8.23 kW/ft, using the rod average-to-peak burnup values.

Cladding temperature profiles for fuel assembly 1A16 fuel rods were calculated for several time periods during cycle 1. Typical examples are shown in Figures 5 through 8.

Table 1. Oconee 1 Nominal Fuel Parameters

Core power rating, MWt	2568
Average linear heat rate, kW/ft	5.66
Cycle 1 length, EFPD	310
EOC core average burnup, MWd/mtU	9600
No. of assemblies in core	177
No. of fuel rods per assembly	208
Cladding OD, in.	0.430
Cladding ID, in.	0.377
Cladding thickness, mils	26.5
Total fueled length, in.	144
Fuel pellet diameter, in.	0.370
Fuel pellet length, in.	0.7
Fuel density, % TD	93.5
Initial average enrichment, % ²³⁵ U	
Batch A	2.00
Batch B	2.10
Batch C	2.15

Table 2. Assembly 1A16 Average Relative Fuel Rod Power Levels During Cycle 1

Time, EFPD	Ass'y power relative to core avg ass'y power	Individual rod power relative to core average power, rod No.										
		47101 47102 47103 47104 47111	47105 47106 47107 47108	47004	47005	47010	47015	47021	47110 47114	47118	47125	AB
0	1.664	1.687	1.680	1.631	1.631	1.658	1.740	1.716	1.642	1.630	1.733	1.617
2	1.579	1.601	1.594	1.547	1.547	1.573	1.652	1.629	1.557	1.547	1.645	1.536
4	1.593	1.615	1.608	1.560	1.560	1.586	1.667	1.644	1.570	1.560	1.661	1.548
50	1.543	1.564	1.557	1.509	1.511	1.534	1.618	1.593	1.516	1.510	1.610	1.503
100	0.669	0.532	0.550	0.701	0.738	0.568	0.490	0.548	0.690	0.800	0.483	0.972
150	0.653	0.525	0.541	0.685	0.719	0.560	0.483	0.540	0.681	0.777	0.477	0.935
200	1.087	1.117	1.110	1.058	1.056	1.093	1.164	1.144	1.080	1.044	1.158	1.026
250	1.104	1.131	1.124	1.076	1.074	1.106	1.176	1.156	1.091	1.065	1.170	1.050
285	1.383	1.410	1.403	1.349	1.349	1.378	1.468	1.444	1.363	1.340	1.459	1.329
310	1.275	1.299	1.293	1.244	1.246	1.269	1.351	1.329	1.255	1.238	1.343	1.230

Table 3. Calculated Average Burnup Values
for 1A16 Fuel Rods

<u>Rod</u>	<u>Burnup, MWd/mtU</u>
47004	10,783
47005	10,868
47010	10,630
47015	10,775
47021	10,775
47101	10,630
47102	10,630
47103	10,630
47104	10,630
47105	10,630
47106	10,630
47107	10,630
47108	10,630
47110	10,775
47111	10,630
47114	10,775
47118	11,021
47125	10,775
AB	11,272

Table 4. Calculated Linear Heat Rates for 1A16 Fuel Rods

<u>Rod No.</u>	<u>Rod avg heat rate, kW/ft</u>	<u>Rod avg/peak burnup</u>	<u>Rod avg peak heat rate, kW/ft</u>
47004	6.35	0.778	8.16
47005	6.40	0.785	8.15
47010	6.26	0.779	8.04
47015	6.34	0.745	8.51
47021	6.34	0.747	8.51
47101	6.26	0.765	8.18
47102	6.26	0.781	8.02
47103	6.26	0.749	8.36
47104	6.26	0.745	8.40
47105	6.26	0.759	8.25
47106	6.26	0.760	8.24
47107	6.26	0.754	8.30
47108	6.26	0.753	8.31
47110	6.34	0.767	8.27
47111	6.26	0.781	8.02
47114	6.34	0.785	8.08
47118	6.49	0.797	8.14
47125	6.34	0.746	8.50
AB	<u>6.63</u>	<u>0.836</u>	<u>7.93</u>
$\bar{X}$	6.32	0.769	8.23
S	± 0.10	0.023	0.17

Figure 1. Mark B Fuel Assembly

Figure 2. Schematic Cross Section of Fuel Assembly 1A16

Figure 4. Fuel Assembly 1A16 Axial Burnup Profile

(a) Normalized to the assembly peak axial (nodal) burnup of 13,430 MWd/mtU.

Figure 5. Typical Cladding Temperature Profile for
Assembly 1A16 Fuel Rods at 80 EFPD

Figure 6. Typical Cladding Temperature Profile for
1A16 Fuel Rod at 100 EFPD

Figure 7. Typical Cladding Temperature Profile for
1A16 Fuel Rod at 200 EFPD

Figure 8. Typical Cladding Temperature Profile for
1A16 Fuel Rod at 310 EFPD

3. RESULTS AND DISCUSSION

The examination of 19 fuel rods comprising the EOC 1 destructive examination phase was performed in two stages. The first stage consisted of the extraction of 19 fuel rods from fuel assembly 1A16 (after removal of the upper end fitting), thorough visual and nondestructive examinations of the extracted fuel rods, and the puncturing of each extracted rod for fission gas sampling and for pressure and void volume determinations. The second stage consisted of the destructive sectioning of three of the extracted rods (fuel rods 47004, 47005, and 47106) for metallographic examination of the fuel and cladding, density and burnup determinations of the fuel, and tensile testing and hydrogen analysis of the cladding. This work was performed at B&W's Lynchburg Research Center from September 1975 to February 1977.

3.1. Fuel Rod Extraction Loads

Of the 19 fuel rods extracted from assembly 1A16, 17 were removed from interior lattice positions (see Figure 2). The fuel rods were removed by directly pulling them upward through the unkeyed fuel assembly grids. The force required to do this provided an indication of grid spring relaxation during operation when compared to design levels and experimental data derived from similar tests performed on unirradiated assemblies. During rod-pulling operations on fuel assembly 1A16, the load required to extract each rod was continuously monitored by a calibrated load cell and strip chart recorder.

Grid relaxation at EOC 1 was expected to be approximately 60%. For the 17 interior rods, the average measured lift-off force indicated a decrease of 53% in spring force during the first cycle of operation. The extraction lift-off force for mid-face peripheral rod 47118 indicated a relaxation of 54%; the load required for the AB corner rod suggested a relaxation of about 67%. The reason for the increased relaxation of the corner cell is not apparent and may be the result of normal scatter associated with a single data point.

The grid frictional forces were also determined for each extracted fuel rod by subtracting the free hanging rod weight from the measured lift-off force. The distribution of the individual grid frictional forces relative to assembly lattice position is shown in Figure 9. No obvious trends in the distribution are apparent.

3.2. Visual Examinations

A thorough visual examination of the surface of individual fuel rods extracted from fuel assembly 1A16 was conducted prior to any further nondestructive or destructive examination. An optical periscope was used, and photographs were taken for evaluation and documentation.

In general, all the rods were similar in appearance. As was noted in the pool-side examinations,¹ the crud patterns and discolorations were highly variable and nonuniform. On several rods, distinct circumferential rings were evident in the crud and oxide patterns. Marks of varying degree were consistently seen at each grid elevation, and superficial longitudinal scratches were present on all rods from contact with the grid springs during the rod extraction process. None of the observed features were judged to be indications of damage to the fuel rods.

3.3. Fuel Rod Length Measurements

Fuel rod length measurements were taken on each of the extracted rods, and the results were compared to the nominal pre-irradiation length of 153.688 inches. Length was measured by comparing dial indicator readings (to the nearest 0.001 inch) to a similar reading taken from a standard rod of known length. The measurement error was estimated to be less than ± 0.010 inch. The average linear growth for the 19 extracted rods was determined to be 0.209 ± 0.045 (1 σ) inch; the maximum observed increase in length was 0.311 inch. The average axial strain for these rods was $0.135 \pm 0.030\%$ (1 σ). The measured rod growth is shown relative to the operating lattice position of each rod in the fuel assembly in Figure 10. No obvious trends were observed.

3.4. Fuel Rod Gross Gamma Scans

Full length axial gross gamma-scan profiles were obtained on each of the 19 extracted fuel rods using a doubly adjustable (apertures at both ends) collimator, scintillation (NaI) detection system, and X-Y recorder. Each profile was a continuous plot of gamma count rate (energies ≥ 0.3 MeV) as a function

of axial position along the rod. Two types of gross gamma-scans were made: (1) a fast gamma-scan (~ 4.00 in./min.) was run on all rods, and (2) if more than two significant fuel pellet gaps were located, a slow scan (~ 0.80 in./min.) was run. Length along the rod was known to within ± 0.050 inch, and gap lengths of 0.10 inch were easily detected. The profiles were analyzed to determine fuel pellet gap lengths, fuel stack lengths, and gross burnup profiles.

Fuel pellet gaps present in the fuel stacks were generally small, with the number and size of gaps per rod varying in a random manner. The average total gap length per rod was 0.46 ± 0.60 (1 σ) inch.

Fuel stack length was determined by subtracting the total gap length from the post-irradiated uncorrected stack length. Except for the uncharacterized AB corner rod, the fuel stack length was compared with precharacterization data to determine the change in stack length at EOC 1. The accumulated data on gap and stack length, and the amount of change in stack length for each rod are given in Table 5. The average decrease in stack length for the 19 rods was 1.65 ± 0.51 inch (1.143% $\Delta l/l$). If densification is assumed to occur isotropically ($\Delta \rho/\rho = -3 \Delta l/l$), the average increase in density was approximately 3.43% $\Delta \rho/\rho$. This value is in fair agreement with the measured density changes (2.81% average as noted in section 3.12).

Gross burnup profiles were obtained by direct correlation with the gross gamma activity. For each rod, the gamma-scan results indicated a peak burnup location about 30 inches from the bottom of the rod.

The fast gamma-scans of adjacent rod pairs were compared to determine whether secondary peaks in the burnup profile were located in the rod adjacent to a pellet gap. No peaks of sufficient magnitude were observed, suggesting that no pellet gaps of significant length were present during incore operation. Furthermore, gamma-scans taken before and after handling (Figures 11 and 12) indicated that changes in gap length and location may be attributed to handling operations. Ease of fuel movement within the fuel rods was also indicated during removal of the fuel from sections of the cladding that were selected for tensile tests. Thus, it is quite probable that many of the detected fuel pellet gaps resulted from post-irradiation handling of the fuel and were not present while the rods were in the core.

3.5. Fuel Rod Isotopic Distribution

Each extracted fuel rod was scanned for individual isotopic distribution utilizing a doubly adjustable collimator and Ge(Li) detection system having a combined resolution of about 4 keV. Gamma-ray spectra were obtained at seven axial locations on 16 of the rods and at 12 axial locations on the remaining three rods; all locations were between spacer grids. Isotopic distributions were analyzed to assess the extent of cesium migration and to determine the plutonium fission fraction, relative neutron flux, and burnup. No quantitative comparisons of isotopic content between rods were possible because of the non-reproducibility of the collimator aperture openings.

The fission product radionuclides identified in the fuel rod spectra were cerium-144, cesium-134, cesium-137, ruthenium-103, ruthenium-106, and zirconium-95. The ^{103}Ru data were not processed because of the relatively long decay time (about 10 half-lives) and resultant data scatter. A typical plot of the relative activity levels of the remaining five radionuclides as a function of distance from the bottom of the fuel stack is shown in Figure 13. The ratios of ^{137}Cs to ^{144}Ce and ^{134}Cs to ^{95}Zr indicate a slightly lower relative amount of cesium at the cooler areas (or ends) of the fuel stack. This is the reverse of what would be expected if cesium migration had occurred.

Relative neutron flux and burnup were calculated from the ratio of ^{134}Cs to ^{137}Cs at the various axial locations along the fuel rods. The ratios at any axial plane were within $\pm 10\%$ for all of the rods, which is consistent with the calculated integrated power distribution.

The fraction of fissions resulting from plutonium was calculated from the ratio of ^{106}Ru to ^{137}Cs by assuming that the fraction of ^{235}U fissions was constant (0.075) and that the fissions of ^{241}Pu were 10% of those from ^{239}Pu . These results, shown in Figure 14 for a typical fuel rod, exhibit a trend that is very similar to that for ^{137}Cs activity and for the ^{134}Cs to ^{137}Cs ratio.

3.6. Fuel Rod Diameter, Creepdown, and Ovality

The diameter of each extracted fuel rod was measured using a scanning device that contained a linear variable displacement transducer (LVDT). The LVDT output data were recorded on a strip chart recorder and manually digitized for analysis. Several scanning modes were available; however, it was found that maximum and minimum diameters for a given cross-section could be determined

with better statistical representation of creepdown and ovality by using step-scans in which the fuel rod is moved to a fixed axial position, whereupon the scanning head is rotated 180°. Each of the 19 fuel rods was scanned in this manner using an axial separation of 4 inches between steps. The accuracy of the average diameter measurements was estimated to be ± 0.0008 inch. The measurement accuracy (and statistical representation) was further improved to ± 0.0002 inch by decreasing the axial separation between steps to 0.25 inch or less. Axial position along the fuel rod was determined from the rod hoist odometer to an accuracy within 0.125 inch for each method.

Seven rods were rescanned using the more time-consuming procedure with axial steps of 0.250 inch or less. Five of those rods were also scanned later in another program⁴ by Battelle Columbus using a spiral scan with a pitch of 0.125 inch; these results, which have comparable accuracy to those produced in the LRC rescans, are included in this report. The remaining seven fuel rods extracted from the assembly were characterized by the 4-inch axial step-scan data.

Figure 15 is an example of the detailed changes in the dimensional envelope of fuel rod 47102 along a segment of its axial length with a step-scan of 0.125 inch. From measurements such as these the maximum, minimum, and average diameters and ovality ($D_{\max} - D_{\min}$) along the full length of each fuel rod were determined. The results of these determinations for fuel rod 47102 are shown in Figure 16 as an example. Figure 17 displays a scan made on a typical section of as-received (archive) cladding for comparison.

A summary of the diameter and creepdown results for all of the extracted fuel rods is given in Table 6. The final average diameter values include the 4-inch step-scan averages for each rod. Average creepdown values (initial minus final fuel rod average diameter) were calculated from the tight pitch scans where they were available. Rescans on rods 47102 and 47114 were made with 0.125-inch step-scans. Mean values ($\bar{X}$) and standard deviations (S) for the 19 rods were determined for each dimensional parameter and are included in Table 6. The average initial diameter of the fuel rods was 0.4294 inch; this value was used for calculational purposes as the initial value for the AB corner rod, which had not been characterized during fabrication.

The average creepdown calculated using the best available (tightest pitch) final diameters for all 19 rods was 0.0020 inch; the largest creepdown was

0.0033 inch. The average creepdown for each rod is shown in Figure 18 as a function of assembly lattice position during cycle 1. No obvious pattern to the distribution within the assembly and no visible bias related to peripheral positions or to positions adjacent to control rods are apparent.

The axial profile for average creepdown (determined from the seven tight pitch LRC rescanned rods) is compared to the axial burnup profile in Figure 19. In this comparison the peak in the burnup profile is roughly normalized to the peak of the creepdown profile. Since the temperature profiles typically peaked closer to the top of the assembly, the creepdown behavior of the cladding is apparently more strongly dependent on irradiation effects than on temperature effects.

Peak creepdown averaged 0.0028 inch and was typically located approximately 30 to 50 inches from the bottom of the rods, near or between spacer grids 5 and 6. The peak creepdown was coincident with the approximate peak burnup location. On two rods, secondary peak creepdown locations occurred about 105 inches from the bottom, near the peak cladding temperature location.

A summary of the ovality results for all of the extracted fuel rods is given in Table 7. The results tabulated are those obtained from the 4-inch step-scans. The rod average final ovalities ranged from 0.0026 to 0.0057 inch, with a mean of 0.0039 inch. The peak ovalities observed ranged from 0.0054 to 0.0120 inch and averaged 0.0083 inch. The peak ovality locations varied along the axial length of the rods but tended to occur with a greater frequency between spacer grids 1 and 2 and grids 4 and 6. No correlations were noted between the initial and final ovalities.

To identify the potential effect of grid spring forces on fuel rod deformation, expanded plots of the diameter envelope were studied for two fuel rods at each spacer grid elevation. No definite correlation between spacer grid position and the ovality envelope or the average creepdown behavior was noted for fuel rod 47114, which operated in an interior lattice position where grid spring loads were relatively low. For fuel rod 47118, ovality and creepdown peaks were apparent at each spacer grid location. This rod operated in a mid-face peripheral assembly lattice position, had the highest extraction forces of any of the 19 extracted rods, and had higher grid spring forces due to the stiffer outer grid strip. The interaction of the grid loads with average creepdown

behavior was apparently a local effect and did not influence the creepdown or ovalization behavior of the rods between the spacer grids.

3.7. Fuel Rod Eddy-Current Examinations

Each fuel rod extracted from the assembly (except rod 47101) was scanned with a continuous encircling eddy-current device at an axial scan rate of 4 in./min. Data were collected on an X-Y recorder and analyzed to obtain the frequency and location of cladding eddy-current indications.

The system provided 100% inspection of the cladding but was not arranged to permit differentiation between eddy-current indications received from the inside (ID) and the outside (OD) of the tubing. Also, although the system was calibrated with machined defect standards, eddy-current indications could not be correlated accurately or consistently with cladding structural perturbations because of the difficulty in anticipating the various types and geometries of perturbations that could occur.

On each of the 18 rods, eddy-current indications were obtained at the spacer grid locations. Occasionally, one grid area did not produce an indication, particularly at the bottom (sixth) grid location. A typical X-Y plot of the eddy-current data for a relatively clean fuel rod (47010) is shown in Figure 20 with indications at the top five spacer grid regions. No indication is present at the sixth grid location. The increasing background signal in the high-burnup region (~120 inches from the top of the rod) was consistently observed and was attributed to fuel/cladding bonding on the inner surface of the cladding.

A histogram showing the total distribution of eddy-current indications occurring between spacer grids for the 18 rods scanned is given in Figure 21. The highest frequency of indications occurred in the region between the fourth and sixth spacer grids, and the overall distribution generally followed the gross burnup profiles (obtained from the gross gamma-scans).

Expanded eddy-current plots were obtained to study the distribution of indications at the spacer grid positions. A histogram of the distribution of indications at the spacer grid locations for the 18 rods is given in Figure 22. Visual and metallographic characterization (discussed in section 3.14) suggests a probable correlation of these indications with superficial wear and corrosion marks found under the grid contact points.

In Figure 23 the number of eddy-current indications received between spacer grids for each rod is shown versus the lattice position of the rod in the fuel assembly. No obvious correlations can be seen, but rods operating close to one another seem to have similar numbers of indications.

Other examinations (particularly the Maine Yankee fuel rod performance evaluation⁵) have revealed that various structures and configurations in irradiated fuel rods, other than cracks and holes in the cladding, produce eddy-current indications. For example, fuel/cladding bonding on the inner surface of the cladding and high localized hydride concentrations have been noted to produce indications. With the inherent inaccuracy of the calibration of the system used in this examination and insufficient information on the response of the system to various cladding perturbations, it is not possible to judge whether or not a specific eddy-current indication represents a true irregularity in the cladding. As presented in section 3.14, metallographic sectioning of the fuel rods at locations where significant eddy-current indications were obtained failed to reveal structural damage of any type in the cladding.

3.8. Cladding Wall Thickness Measurements

Each of the 19 fuel rods was scanned for thickness variations of the cladding wall using dual, fine-focusing ultrasonic transducers incorporated in the multi-component head of the scanning device. At least two of three scanning modes (line, rotational step, or continuous rotational) were used on each rod to characterize the cladding wall thickness variations along the length of each rod. The major uncertainty in the calibration of this system is the system response to crud and irradiation damage effects in the cladding being tested. The sections of cladding material used as calibration standards were clean, unirradiated Zircaloy tubing sections, while the irradiated fuel cladding had surface crud layers of varying thickness and morphology. Neglecting the uncertainty of these effects, the measurement system is considered to be accurate to within 0.1 to 0.2 mil.

From these measured thickness variations, values were tabulated for average ($\bar{T}$), average maximum ($\bar{T}_{\max}$), and average minimum ($\bar{T}_{\min}$) wall thickness; maximum eccentricity ($\omega = \bar{T}_{\max} - \bar{T}_{\min}$); and the axial pitch of the spiraling eccentric cross section. The pitch of the eccentricity of the cladding was taken to be the axial distance between indicated maximum (or minimum) wall

thickness values. A single average wall thickness value for each rod was then determined and compared to the characterization data obtained during fabrication to calculate the change in the average wall thickness of each rod after the first cycle of operation. A summary of values for each rod is given in Table 8.

The average cladding wall thicknesses calculated before and after irradiation were 26.2 ± 0.2 and 25.9 ± 0.3 mils, respectively. Although there was a large scatter in the data, the average decrease in apparent wall thickness of about 0.3 mil is well within the range expected to result for general surface corrosion during one cycle of operation in a PWR.⁶

No correlation could be found between the measured geometric variations in the fuel cladding and the observed variations in cladding deformation occurring during the first cycle of operation.

3.9. Fuel Rod Crud Analysis

Twelve crud samples were collected from fuel rods AB, 47118, 47114, and 47101 during the destructive phase of the examination. Samples were obtained at known locations between spacer grids (primarily grids 3, 4, and 5) using cotton swiped over a sample area of about 17.4 cm².

Gamma isotopic analyses were conducted utilizing a Ge(Li) gamma detection system with an estimated accuracy of within $\pm 10\%$. Surface concentrations were determined for ⁵⁸Co, ⁶⁰Co, and ⁵⁴Mn; the activities for ⁵¹Cr, ⁵⁹Fe, and ⁹⁵Zr (all with half-lives of less than 70 days) had decayed below minimum levels of detection due to the relatively long decay time prior to counting (~ 18 months). From the 12 samples, the average radionuclide surface concentrations determined were 2.1 $\mu\text{Ci } ^{54}\text{Mn}/\text{cm}^2$, 42 $\mu\text{Ci } ^{58}\text{Co}/\text{cm}^2$, and 10 $\mu\text{Ci } ^{60}\text{Co}/\text{cm}^2$. These values are approximately a factor of seven less than the best estimate of the surface concentrations based on the results of the Oconee 1 EOC 1 nondestructive examination (poolside).¹

The chemical compositions of the fuel rod crud samples were determined quantitatively by x-ray fluorescence with an accuracy within $\pm 5\%$ after dissolving the cotton swabs in boiling HNO₃ (conc.) and H₂SO₄ (conc.). The average elemental composition, determined from nine samples, consisted of 39% Ni, 34% Fe, and 27% Cr.

The specific activities of the crud samples were calculated from the results of the gamma isotopic and elemental analyses. The average values determined from nine samples were 1.2×10^6 dps $^{54}\text{Mn}/\text{mg Fe}$ and 1.2×10^7 dps $^{58}\text{Co}/\text{mg Ni}$. These values are about 20% higher than the average specific activities reported previously.¹

Fuel crud thickness was calculated for each sample by using the radionuclide surface concentration (mg/cm^2) to obtain the oxide weight per unit area and by assuming a crud density of $1.0 \text{ g}/\text{cm}^3$. The average calculated crud thickness was 0.1 mil. This is about 25% of the average thickness calculated by weight previously and only 10% of the thickness observed for the crud flake samples reported previously.¹

The results of this crud analysis indicated that approximately 70 to 90% of the fuel crud had sloughed off the fuel rods before the crud samples were collected. Consequently, no additional analyses of the variations in crud composition, concentration, and thickness with respect to axial location along the fuel rod were pursued.

3.10. Fuel Rod Fission Gas Analysis

In order to obtain data on fission gas release under actual operating conditions, each of the 19 extracted fuel rods was punctured and its fission gas collected for isotopic chemical analysis. Internal rod void volume and pressure were also determined for each extracted rod during the gas collection process. Isotopic gas analysis was performed using a quadrupole gas analyzer calibrated with a known mixture of helium, krypton, xenon, and argon. Rod void volume and pressure were calculated using a gas volume expansion technique (with an argon backfill) and applying the ideal gas law with known system volumes, pressures, and temperatures.

Table 9 is a summary of the results of the isotopic gas analyses. Difficulties were encountered with contamination of the fission gas samples by argon, which resulted from argon adsorption on the system walls either during calibration of the gas analyzer or during fuel rod backfilling. Because of the uncertainty in the origin of the argon, the analyses are reported for only those samples in which no argon was detected. The average rod composition based on these uncontaminated data was $99.27 \pm 0.51\%$ He and $0.73 \pm 0.44\%$ fission gas, by volume.

The average volume percent of fission gas release for the seven rods was estimated to be $0.67 \pm 0.48\%$ with a population average burnup of 10.7 GWd/mtU, assuming a fission yield of 0.31 gas atoms per thermal fission of ^{235}U and neglecting the short-lived isotopes of xenon and krypton.

Fuel rod pressures decreased during the first cycle of operation by an average of 14.3%, while void volume increased by an average of 19.3%. The EOC pressure calculated using the ideal gas law was in good agreement with the average gas pressure obtained experimentally.

3.11. Fuel Burnup Analysis

Samples of the UO_2 fuel pellets were taken from the high burnup region of each selected fuel rod for chemical burnup analysis. The sections of fuel and cladding were approximately 0.5 inch long and were removed from each rod at about 30 inches from the rod bottom.

The samples were analyzed for burnup by two methods: (1) based on an evaluation of the fission product content (using ASTM E-321) and (2) based on changes in the uranium and plutonium content.⁷ As shown in Table 10, very good agreement was obtained between the two methods. The uncertainty in the measured burnup values expressed as fissions per initial atom of uranium is about $\pm 2.5\%$, and the overall error resulting from expressing burnup in GWd/mtU is about $\pm 4\%$.

The experimentally determined post-irradiation isotopic distribution and the calculated number of fissions per initial uranium atom are given in Table 11.

3.12. Fuel Density Measurements

Preirradiation density measurements were taken during fuel rod fabrication for selected fuel pellets in each of the three rods sectioned in the destructive examination. The densities were determined by a water-immersion technique based on ASTM C-373. Pellets from two fuel lots were involved. After irradiation, the selected fuel pellets were located by gamma-scans, and the fuel rods were sectioned to obtain sizeable fragments of nine characterized pellets for post-irradiation density measurements in the hot cell.

The amount of densification that occurred in the pellets during the first cycle of irradiation was correlated with burnup at the axial position of each selected pellet. The burnup was determined by correlating the burnup determined at the

positions of the chemical burnup samples with the gross gamma-scan activities. The results of the density measurements are tabulated along with the corresponding burnup information in Table 12.

Despite their variability, the tabulated results show a clear distinction in densification between the two lots of fuel; the maximum increase in density of lot 16 fuel (2.29%) was close to the minimum increase in density of lot 13 fuel (2.20%), suggesting that the lot 13 fuel is more prone to densification. Furthermore, although the average change in density calculated from the fuel stack length changes ($\sim 3.43\%$ as noted in section 3.4, assuming isotropic densification) was in fair agreement with the average change in immersion density ($2.81 \pm 1.28\%$), the majority of the tabulated fuel pellet densities increased by 2.65% or less. This apparent incomplete agreement between fuel stack length changes and pellet densification may be due either to nonuniform densification caused by the lower burnup at the ends of the fuel rods or to anisotropic densification.

3.13. Fuel Metallography

At least two samples were taken for metallographic examination from each of the three rods sectioned. The types, locations, burnups, and operating temperatures of the specimens are listed in Table 13. The microstructures of these specimens were compared to the microstructures of respective archive pellets selected from each of the two fuel lots. Porosity distributions were determined, and estimates of grain size were obtained by calculating the grain boundary linear intercept averages.

Photomicrographs of unirradiated (archive) lot 13 fuel exhibited a matrix of low density material having a fine porosity and containing islands of higher density fuel distributed within it; the grains were fine and reasonably uniform. Lot 16 archive fuel exhibited a matrix of lower density material with islands of higher density fuel surrounded by slightly larger pores than lot 13 fuel. A greater variation in grain size was also evident in the lot 16 fuel.

The irradiation-induced changes in the porosity distributions are summarized in Table 14 for those observed pores of 5 μm equivalent diameter or less. For lot 13 fuels, there was a general decrease in porosity in moving from the pellet edge to center, with this gradient decreasing as pellet peak temperature increased. The minimum porosity at the mid-radius of sample 11E suggests that

the peak temperature seen by this pellet was high enough to encourage the formation of fission gas bubbles at grain boundaries in the center region of the pellet. Although the porosity distributions for samples from lot 16 fuels did not exhibit significant change, the fine intragranular porosity present before irradiation was located predominantly on the grain boundaries after irradiation.

The results of the grain size measurements are summarized in Table 15. With the exception of samples 25MT and 11E, the irradiated pellets exhibited only a slight increase in grain size toward their center. For sample 25MT, the grain size at the center was more than double that at the outer edge, while for sample 11E there was no significant change. There is no explanation for this observed difference in grain size behavior between the two samples.

3.14. Cladding Metallography

Nineteen fuel-cladding cross sections were prepared for metallographic examination from the three fuel rods sectioned. The samples were selected to represent a good distribution of power histories, and several samples were specifically selected at locations where eddy-current indications had suggested the presence of cladding defects. Both transverse and longitudinal sections were prepared, and sequential polishing procedures were used on some samples to better locate and define probable cladding defects.

Metallographic examination of the various samples at both high and low magnifications revealed no indications of cladding defects. No cracks or pits were observed on the cladding inner surfaces; and the cladding cross sections were generally uniform and unperturbed. Even in those samples taken from locations that had produced significant eddy-current indications during the nondestructive examinations, sequential sectioning revealed no cladding structural damage.

On the outer cladding surface, complex grid-cladding interaction phenomena were revealed at the spacer grid locations after removal of the flow-induced crud depositions. Light fretting, due to abrasion by the grid stops, and some localized pitting were noted at the grid contact points. The maximum depth of this pitting was estimated to be approximately 2 mils, which is not unexpected. Localized pitting of 5 mils over three cycles of irradiation was considered in the fuel rod design.

Two distinct morphologies of hydride precipitation were observed in these metallographic specimens. The majority of the hydride regions exhibited circumferentially oriented, plate-like precipitates, as shown in Figures 24 and 25. However, in several transverse sections taken from near the bottoms of the rods, pin-like or spherical morphologies were observed, as shown in Figure 26. Unfortunately, these could not be classified in more detail because equivalent longitudinal sections were not available for metallographic analysis. The degree of axial and circumferential variation and the relative concentrations of the two morphological classes were not determined because the quantities of hydride precipitates varied considerably in both axial and circumferential locations. This variability substantiated the variability in the results of the hydrogen analyses reported in section 3.15.

3.15. Cladding Hydrogen Analysis

Hydrogen analyses were performed on ring cladding samples taken from three sections (top, middle, and bottom) of fuel rod 47106. Each ring section was divided into four segments of 0.1 gram each for a total of 12 samples. Hydrogen analyses were conducted on 11 of these samples using vacuum extraction (LECO) techniques with an estimated accuracy of ± 10 ppm.

The results of these analyses displayed a wide range of hydrogen concentrations, from 55 to 257 ppm, with a mean of 124 ppm. Considerable variation in hydrogen content was indicated circumferentially within each ring section; a somewhat smaller, yet notable, variation was indicated axially between each of the sections. In view of the wide variations in concentration, these results are considered too limited to be statistically significant. However, they essentially corroborate the metallographic results summarized previously in section 3.14.

Increases of approximately 50 ppm hydrogen per cycle are expected to result from waterside corrosion during normal PWR operation. Since the as-received hydrogen content of the cladding material was typically less than 20 ppm, it is believed that the higher hydrogen concentrations found in this examination resulted from contaminant moisture that was liberated early in life from undried UO_2 fuel (a problem which has been corrected in more recent fuel by improved manufacturing processes). Thus, additional increases on the order of 50 ppm hydrogen per subsequent cycle are expected. These increases will not

present a problem since hydrogen concentrations in the range of 500 to 600 ppm have been stated in literature as being an acceptable upper operating limit.⁸

3.16. Cladding ID Examination

The ID of a section of EOC 1 fuel cladding, cut and quartered from fuel rod 47004, was examined using scanning electron microscopy (SEM). The 1-inch-long cylindrical section (20E) was removed from the rod at a location approximately 69 to 70 inches from the bottom of the fuel rod. An estimate of the power history at this axial location indicated a local burnup of 13,600 MWd/mtU and a higher than average linear heat rate of 8 kW/ft. Qualitative chemical analyses using the energy-dispersive x-ray analysis (EDAX) feature of the SEM were attempted; however, these were greatly limited because of the high levels of background radiation emanating from the irradiated samples. The surface features observed in this examination were also compared to those of a typical as-received ID surface of standard Mark-B cladding.

Notably obvious in this examination were dark linear markings (image contrast areas) that corresponded to fuel pellet cracks and pellet-to-pellet interfaces, similar to those previously documented in the examination of Maine Yankee fuel.⁵ Several interesting surface features, described below, were observed by SEM on or near the linear markings. However, no evidence of incipient cladding cracks was found. The markings appear to have resulted from local variations in chemical deposits or sorption/diffusion layers of different chemical species rather than from cladding deformation. Uranium and zirconium were the only elements detected (by EDAX) on the cladding ID, but the presence of other chemical species may have been masked by the relatively high background count rate of the specimens.

Figures 27 and 28 display the markings found on the cladding ID surface before and after ultrasonic cleaning, respectively. Figure 29 is a SEM photographic mosaic of the major circumferential linear marking shown in the center of Figure 28.

Two types, or forms, of deposits were identified within or near the linear markings. One type is readily observable in Figure 29 as isolated particles. These deposits are approximately circular and vary from 3 to 10 mils in diameter. Figure 30a is an enlargement of section 9 of Figure 29, and Figure 30b is a magnification of the deposit. The deposit has a fragmented crystalline appearance. Around the fringes of the deposit are apparent reaction

regions that may be indicative of bond formation between the cladding and the deposits. Despite the high background count rate of the sample, uranium was detected in these deposits. An isolated particle, also found to contain uranium, is shown in Figure 31.

The other type of oxide observed is shown in Figure 32. This oxide deposit is considerably smaller than the fragmented deposits and is nodular in form. This nodular deposit was also located in the region of the circumferential linear marking, but it appeared to be more adherent to the cladding surface and exhibited little or no evidence of the fragmentation observed in the deposit shown in Figure 30. Figures 32a and 32b illustrate the tendency of these oxide modules to have nucleated and grown in an approximately linear alignment parallel to the axial direction of the cladding. Features of the alignment and the location relative to the linear markings suggest that the alignment is related to either the topography or the migration of fission product vapor phases away from the fuel pellet cracks. Only zirconium was detected in the deposits, suggesting that the nodular deposits were comprised of zirconium oxide.

At higher magnifications ($\geq 1000\times$) both light and dark regions were apparent within some linear markings. The distinctly darker regions of the markings shown in Figure 29 exhibited indications of local attack of the oxide layer. Segment 10 of the mosaic of Figure 29 is shown in Figure 33a. The nodular type of oxide deposit apparent in Figure 33b appears to have been attacked by the spherical particles residing on it. Some of the particles had been removed from the deposit, leaving crater-like reaction regions. Figures 33c and 33d illustrate the presence of the particulate matter and its apparent attack on the underlying deposit. In Figure 33d several prominent crater-like regions are apparent in which the inside structure of the crater appears to be quite different from the structure of the surrounding deposit, indicating a possible reaction product structure. Figures 33b and 33c also display some regions of possible micro-pitting that appear to have resulted from chemical attack. The attack is generally aligned in the direction of the circumferential linear marking shown in Figure 29.

Additional features, both of a pitting type attack and of shallow elongated (or grooved) regions of attack, are shown in Figure 34. This figure is a higher magnification of the axially-aligned dark-stained region in the lower right corner of Figure 28. It is apparent from Figure 34 that the difference in

contrast between the light and dark regions is not due to topological differences. This raises the possibility of local variations in chemistry on the cladding surface due to an absorption or diffusion layer of different chemical species.

3.17. Cladding Tensile Properties

Uniaxial and ring tensile tests were performed at $650 \pm 5F$ on samples cut from various locations on each of the three rods sectioned (47004, 47005, and 47106). Each of the 17 irradiated uniaxial test specimens was strain-rate paced at 0.005 min.^{-1} until reaching the maximum load and then switched over to cross-head pacing at 0.1 in./min. Ring test specimens with a nominal width-to-length ratio of 9.4 were also taken from each of the three irradiated rods (two specimens per rod); all were adjacent to axial specimens obtained from the higher operating temperature regions of the rods. For the ring tests, a constant cross-head speed of 0.002 in./min. (corresponding to an initial strain rate of 0.005 min.^{-1}) was maintained through the maximum load and then increased to 0.02 in./min. until failure. Six uniaxial and two ring tensile tests were also conducted on unirradiated (archive) material selected from the same manufacturing lot as used in this fuel assembly, 1A16.

The uniaxial tensile test data for each specimen are tabulated in Table 16. The data indicate that significant strengthening of the cladding had occurred during the first cycle of irradiation (average fluence of $\sim 3 \times 10^{21} \text{ nvt}$, $\phi > 0.1 \text{ MeV}$). The 0.2% yield strength increased by 25 to 52%, and the ultimate tensile strength increased by 16 to 36% as compared to mean values of $54,365 \pm 1,108$ and $70,163 \pm 959 \text{ psi}$, respectively, for the unirradiated cladding. The irradiated uniaxial strength values were observed to generally increase with relative burnup; however, this effect was greatly overshadowed by a very strong dependence on irradiation temperature as shown in Figure 35. The straight lines shown on this figure are the results of a linear least-squares analysis of these data, yielding the following:

$$\text{UTS} = -269(T_{\text{irr}}) + 259,829 \text{ psi}$$

and

$$0.2\% \text{ YS} = -259(T_{\text{irr}}) + 239,745 \text{ psi}$$

where T_{irr} is the irradiation temperature in °F. The ductility of the irradiated cladding exhibited only a 31% decrease in the mean value of uniform elongation ($2.5 \pm 0.5\%$ Vs $3.6 \pm 0.8\%$) and a 22% decrease in total elongation ($10.9 \pm 1.9\%$ Vs $13.9 \pm 1.4\%$) as compared with the unirradiated cladding; no correlations were observed with respect to either burnup or irradiation temperature.

The uniaxial tensile data were also compared to that of other B&W programs involving irradiated cladding. The percentage increase in strength from EOC 1 was greater than the increases observed in the other programs, indicating that the fluences seen in the other programs (1.07×10^{20} to 4.2×10^{20} nvt, $\phi > 0.1$ MeV) were not sufficient to saturate the irradiation-enhanced strength of B&W cladding in the 650F irradiation temperature range. However, it appears that the ductility decreases might have saturated at fluences of about 1.07 to 3.18×10^{20} nvt ($\phi > 0.1$ MeV).

The ring tensile test results are listed in Table 17. The ring test results and those from the adjacent uniaxial test specimens (high-temperature region of the rods only) are summarized in Table 18. Although there were slight differences in irradiation temperature and flux among the rings, the six rings were considered to be from the same population and treated accordingly. The irradiation effects determined by the two test methods compare reasonably well, considering the differences in the methods, indicating that the EOC 1 cladding had undergone considerable irradiation strengthening while retaining a large percentage of its original ductility.

Table 5. Results of Fuel Stack Length Measurements^(a)

Fuel rod	Fuel lot No.	Gamma- scan type	Overall stack length, in.	No. of gaps	Total length of gaps, in.	Corrected stack length, in. ^(c)	Original stack length, in. ^(c)	Δ length ($L_o - L_f$), in.
47004	13	Slow	144.577	14	1.871	142.706	143.76 ^(b)	1.054
47005	13	Slow	144.353	9	2.070	142.283	143.95 ^(b)	1.667
47010	13	Fast	141.630	4	0.140	141.490	143.673	2.183
47015	13	Fast	141.970	0	0	141.970	144.173	2.203
47021	13	Fast	141.970	2	0.280	141.690	143.704	2.014
47101	16	Fast	142.370	0	0	142.370	144.485	2.115
47102	16	Slow	143.336	4	0.550	142.786	144.360	1.574
47103	16	Fast	142.650	2	0.200	142.450	144.298	1.848
47104	16	Slow	143.564	14	0.558	143.006	143.985	0.979
47105	16	Fast	141.970	2	0.200	141.770	143.985	2.215
47106	16	Fast	142.410	3	0.220	142.190	143.798	1.608
47107	16	Slow	143.264	9	1.173	142.091	143.798	1.707
47108	16	Fast	142.550	1	0.060	142.490	144.110	1.620
47110	16	Fast	143.090	3	0.300	142.790	143.50 ^(b)	0.710
47111	16	Slow	143.761	16	0.438	143.323	144.05 ^(b)	0.727
47114	16	Fast	142.770	0	0	142.770	144.60 ^(b)	1.830
47118	16	Slow	142.664	9	0.290	142.374	143.70 ^(b)	1.326
47125	16	Fast	142.050	0	0	142.050	144.30 ^(b)	2.250
AB corner	--	Slow	141.776	6	0.451	141.325	--	--

(a) Estimated measurement accuracy, both pre- and post-irradiation, is ± 0.10 inch.

(b) Manufacturing as-built stack lengths.

(c) Corrected stack length, L_f ; original stack length, L_o .

Table 6. Summary of Fuel Rod Diameter Data and Creepdown

Fuel rod	Initial avg dia, in.	Final avg diameter, in.			Avg creepdown	Peak creepdown	Peak creepdown location
		LRC	LRC 0.25	BMI 0.125			
47004	0.4296	0.4280	--	--	0.0016	0.0023	29.7
47005	0.4294	0.4275	--	--	0.0019	0.0025	35.4,31.4,23.8
47010	0.4291	0.4268	--	0.4270	0.0021	0.0029	28.8
47015	0.4293	0.4274	--	0.4275	0.0018	0.0026	36.6
47021	0.4288	0.4270	--	0.4268	0.0020	0.0027	104.6
47101	0.4300	0.4278	0.4279	--	0.0021	0.0029	31.18,34.19
47102	0.4299	0.4270	0.4279	--	0.0020	0.0028	22.89
47103	0.4290	0.4274	--	0.4273	0.0017	0.0025	50.6,39.2,23.6
47104	0.4297	0.4281	0.4278	--	0.0019	0.0026	45.24
47105	0.4291	0.4269	--	--	0.0022	0.0032	48.2
47106	0.4292	0.4269	--	--	0.0023	0.0029	105.6,43.3,39.4 35.4,29.8
47107	0.4293	0.4269	--	--	0.0024	0.0033	52.8,29.8
47108	0.4289	0.4274	--	--	0.0015	0.0025	56.9
47110	0.4293	0.4284	0.4278	--	0.0015	0.0022	26.20
47111	0.4297	0.4278	0.4282	--	0.0015	0.0025	45.86
47114	0.4291	0.4267	0.4270	--	0.0021	0.0030	34.03
47118	0.4297	0.4279	0.4275	--	0.0022	0.0030	33.03
47125	0.4293	0.4281	--	0.4270	0.0023	0.0029	31.0
AB	(0.4294)	0.4274	--	--	0.0020	0.0029	--
$\bar{X}$	0.4294	0.4274	0.4277	0.4271	0.0020	0.0028	--
S	± 0.00033	± 0.00053	± 0.00038	± 0.00028	± 0.00029	± 0.00029	--

**Table 7. Summary of Fuel Rod Ovality Results --
LRC 4-Inch Step-Scans**

Fuel rod	Initial average (a) ovality, in.	Final average (a) ovality, in.	Average ovality change, in.	Peak ovality, in.	Peak ovality (b) location, in.
47004	0.0006	0.0035	0.0029	0.0095	100.8
47005	0.0002	0.0041	0.0039	0.0099	120.2
47010	0.0003	0.0030	0.0027	0.0069	109.5
47015	0.0003	0.0035	0.0032	0.0069	28.4
47021	0.0004	0.0041	0.0037	0.0096	64.2
47101	0.0002	0.0032	0.0030	0.0085	128.9
47102	0.0003	0.0039	0.0036	0.0085	116.3
47103	0.0002	0.0050	0.0048	0.0097	39.2
47104	0.0004	0.0045	0.0041	0.0084	132.0
47105	0.0001	0.0041	0.0040	0.0088	48.2
47106	0.0001	0.0047	0.0046	0.0086	47.1
47107	0.0003	0.0044	0.0041	0.0081	109.9
47108	0.0003	0.0037	0.0034	0.0067	60.8
47110	0.0003	0.0029	0.0026	0.0054	50.0, 21.2, 17.4
47111	0.0002	0.0026	0.0024	0.0055	70.8
47114	0.0003	0.0057	0.0054	0.0120	64.5
47118	0.0002	0.0042	0.0040	0.0095	66.6
47125	0.0003	0.0039	0.0036	0.0085	131.8
AB	--	0.0038	--	0.0070	63.2
$\bar{X}$	0.0003	0.0039	0.0037	0.0083	
S	± 0.0002	± 0.0016	± 0.0018	± 0.0016	

(a) The rod average ovality is given as the sum of the local ovalities ($OD_{\max} - OD_{\min}$) divided by the number of local ovalities measured.

(b) Distance from the bottom of the fuel rod.

Table 8. Wall Thickness Measurements (mils)

<u>Fuel rod</u>	<u>Initial</u>	<u>Final</u>	<u>Δ</u>	<u>Max</u>	<u>Min</u>	<u>ω</u>	<u>Pitch</u> ^(a)
47004	26.3	25.5	0.8	26.0	25.1	0.9	46
47005	26.2	25.8	0.4	26.2	25.1	1.0	43
47010	26.3	26.0	0.3	26.4	25.3	1.1	54
47015	25.9	25.9	0.0	26.2	25.7	0.5	0
47021	25.9	25.8	0.1	26.3	25.3	1.0	36
47101	26.4	26.0	0.4	26.3	25.5	0.8	56
47102	26.4	25.9	0.5	26.2	25.2	1.0	0
47103	25.9	25.9	0.0	26.1	25.5	0.6	54
47104	26.6	26.4	0.2	26.7	26.0	0.7	0
47105	25.8	25.5	0.3	26.0	25.0	1.0	50
47106	26.2	25.4	0.8	25.6	25.1	0.5	40
47107	26.1	26.1	0.0	26.3	25.7	0.6	41
47108	25.9	25.9	0.0	25.9	25.3	0.6	42
47110	26.1	25.9	0.2	26.9	25.7	0.2	0
47111	26.4	26.3	0.1	26.7	25.7	1.0	38
47114	26.1	25.7	0.4	26.5	25.2	1.1	38
47118	26.5	26.1	0.4	26.5	25.4	1.1	50
47125	26.1	25.8	0.3	26.5	25.0	1.5	51
AB	--	25.9	--	--	--	1.4	60
$\bar{X}$	26.2	25.9	0.3	26.2	25.4	0.9	47

(a) Pitch measurements are given in inches.

Table 9. Results of Fission Gas Analysis

<u>Fuel rod</u>	<u>Kr, vol %</u>	<u>Xe, vol %</u>	<u>He, vol %</u>	<u>Avg burnup, GWd/mtU</u>	<u>Peak burnup, GWd/mtU</u>	<u>Gas release, %</u>
47004	0.03	0.14	99.8	10.8	13.9	0.18
47010	0.3	1.4	98.3	10.6	13.6	1.6
47005	0.03	0.3	99.7	10.9	13.9	0.26
47101	--	1.0	99.0	10.6	13.9	0.89
47106	<0.1	0.6	99.4	10.6	13.9	0.55
47110	--	0.5	99.5	10.8	14.1	0.46
47125	--	0.8	99.2	10.8	14.5	0.74
$\bar{X}$	0.05	0.68	99.27	10.7	14.0	0.67
S	± 0.11	± 0.43	± 0.51	± 0.1	± 0.7	± 0.48

Table 10. Measured Burnup Values for EOC 1 Fuel Samples

	<u>Rod No.</u>		
	<u>47004</u>	<u>47005</u>	<u>47106</u>
<u>Fission Product Method</u>			
Fissions per 100 initial U atoms	1.57	1.42	1.44
Burnup, GWd/mtU	15.1	13.6	13.8
<u>Heavy Element Method</u>			
Fissions per 100 initial U atoms	1.59	1.46	1.45
Burnup, GWd/mtU	15.3	14.0	13.9
<u>Average of Two Methods</u>			
Burnup, GWd/mtU	15.2	13.8	13.8

**Table 11. Post-Irradiation Isotopic Distribution
of EOC 1 Fuel Samples**

	Rod No.		
	<u>47004</u>	<u>47005</u>	<u>47106</u>
<u>Post-Irradiation Uranium Isotopic Composition, atom %</u>			
²³⁴ U	0.008	0.012	0.012
²³⁵ U	0.967	0.981	0.924
²³⁶ U	0.204	0.196	0.189
²³⁸ U	98.820	98.811	98.876
<u>Post-Irradiation Plutonium Isotopic Composition, atom %</u>			
²³⁸ Pu	0.317	0.368	0.355
²³⁹ Pu	68.074	69.014	68.908
²⁴⁰ Pu	20.562	20.230	20.228
²⁴¹ Pu	9.048	8.584	8.672
²⁴² Pu	1.999	1.804	1.837
<u>Post-Irradiation Atoms per 100 Initial Uranium Atoms</u>			
²³⁴ U	0.009	0.012	0.012
²³⁵ U	0.927	0.946	0.936
²³⁶ U	0.195	0.197	0.201
²³⁸ U	96.329	96.599	96.614
²³⁹ Pu	0.656	0.555	0.555
²⁴⁰ Pu	0.198	0.163	0.1630
²⁴¹ Pu	0.094	0.075	0.075
²⁴² Pu	0.019	0.014	0.015
<u>Fissions per Initial Uranium Atoms</u>			
²³⁵ U	0.886	0.878	0.863
²³⁸ U	0.114	0.104	0.103
²³⁹ Pu	0.540	0.436	0.438
²⁴¹ Pu	0.054	0.041	0.041

Table 12. Density and Burnup Results for EOC 1 Fuel

Fuel lot	Fuel rod	Sample	Original density, (a) % TD	Final density, (a) % TD	Density change, % TD	Fissions, 10 ²⁰ /cm	Burnup, MWd/mtU (b)
13	47005	16-D	94.30	96.65 ± 0.91	2.35	3.05	12,600
		19-D	94.68	96.88 ± 0.37	2.20	2.88	11,900
	47004	5-D	92.01	97.12 ± 0.41	5.11	2.25	9,300
		11-D	93.21	96.42 ± 1.59	3.21	2.83	11,700
		16-D	93.34	95.99 ± 1.89	2.65	2.98	12,300
		18-D	91.92	96.37 ± 0.77	4.45	3.03	12,500
16	47106	21-D	93.23	95.31 ± 0.82	2.08	3.20	13,200
		27-D	93.33	95.62 ± 1.06 (c)	2.29	3.32	13,700
		34-D	93.82	94.75 ± 0.63	0.93	1.65	6,800

(a) 10.97 gm/cm³ used as the theoretical density (TD) of low-enriched and depleted UO₂; fuel density is the average of four determinations (two dry and two wet).

(b) Burnup extrapolated from chemical burnup analysis via gross gamma-scans.

(c) Final density is the average of six determinations (three dry and three wet).

Table 13. Fuel Metallographic Specimens

<u>Fuel rod</u>	<u>Fuel lot</u>	<u>Specimen No.</u>	<u>Type</u>	<u>Location, in. (a)</u>	<u>Burnup, GWd/mtU (b)</u>	<u>Avg heat rate, kW/ft (c)</u>	<u>Peak center temp, C (d)</u>	<u>Peak heat rate, kW/ft (d)</u>
47004	13	004-4-MT	Transverse	129	8.8	5.2	1020	6.4
		004-25-MT	Transverse	42	14.1	8.3	1315	9.6
47005	13	005-7-MT	Transverse	107	11.7	7.0	1270	9.0
		005-11-E	Transverse	82	11.7	7.0	1375	10.0
		005-22ML	Longitud'l	30.5 (e)	13.5	8.0	1280	9.2
47106	16	106-8-MT	Transverse	114	10.3	6.2	1240	8.7
		106-29-MT	Transverse	24.5	12.8	7.6	1225	8.8

(a) Location measured from bottom of rod.

(b) Burnup extrapolated from chemical burnup analysis via gross gamma scans.

(c) Time average heat rate based on 310-EFPD cycle.

(d) Based on fuel pin performance analysis using fuel assembly 1A16 average power distribution.

(e) Axial center of sample.

Table 14. Summary of Porosity Distribution Data

<u>Fuel lot</u>	<u>Sample</u>	<u>Peak temp, C^(a)</u>	<u>Relative post-irradiation porosity content^(b)</u>
13	47004-4MT	1020	Edge > MR > center ^(c)
13	47004-25MT	1315	Edge > MR ~ center
13	47005-7MT	1270	Edge > MR ~ center
13	47005-11E	1375	Edge > MR < center
16	47106-8MT	1240	Edge ~ MR > center
16	47106-29MT	1225	Edge ~ MR ≥ center

(a) Linear interpolation from fuel pin performance analysis code using fuel assembly 1A16 average power distribution.

(b) Based on distribution data of pores less than or equal to 5 μm equivalent diameter.

(c) MR is mid-radius.

Table 15. Grain Size Data for EOC 1 Fuel

<u>Sample</u>	<u>Burnup, GWd/mtU</u>	<u>Avg linear heat rate, kW/ft</u>	<u>Avg linear intercept, μ</u>		
			<u>Edge</u>	<u>Mid-radius</u>	<u>Center</u>
Lot 13	0	0	4.0	3.7	3.6
Lot 16	0	0	6.9	7.6	6.2
47004-4MT	8.8	5.2	3.0	3.4	4.2
47004-25MT	14.1	8.3	2.5	3.8	5.7
47005-7MT	11.7	7.0	3.8	4.5	4.8
47005-11E	11.7	7.0	5.3	5.3	5.0
47005-22ML	13.5	8.0	3.6	5.0	5.0
47106-8MT	10.3	6.2	4.2	3.7	5.0
47106-29MT	12.8	7.6	3.8	4.5	5.2

Table 16. Uniaxial Tensile Properties at 650F of Irradiated (Ocone 1, EOC 1) and Unirradiated Zircaloy-4

Rod No.	Distance from top of rod, in.	Irradia'n temp, F(a)	Ultimate tensile strength, psi(b)	0.2% yield strength, psi(b)	Uniform elongation, %	Total elongation, %	Total strain of yield, %			
Rod No.	Distance from top of rod, in.	Irradia'n temp, F(a)	Ultimate tensile strength, psi(b)	0.2% yield strength, psi(b)	Uniform elongation, %	Total elongation, %	Total strain of yield, %	Elastic modulus, 10 ⁶ psi(c)	Avg relative burnup(d)	Comments
47106	23-7/8 - 30-7/8	659	84,821	71,429	1.9	13.5	0.81	11.4	0.65	
47106	30-7/8 - 37-7/8	661	81,101	67,708	2.1	11.3	0.79	11.0	0.73	
47106	91-7/8 - 98-7/8	646	88,095	72,619	2.0	7.0	0.78	12.5	0.84	
47106	98-7/8 - 105-7/8	642	83,929	71,875	2.3	9.0	0.86	10.9	0.86	
47106	131-3/8 - 138-3/8	621	94,494	82,589	2.2	10.4	0.96	10.9	0.89	
47106	138-3/8 - 145-3/8	610	86,756	79,613	2.6	15.2	0.94	10.9	0.69	
47004	16-7/8 - 23-7/8	634	87,798	72,917	2.1	11.4	0.88	10.0	0.55	Sixth specimen cut too short.
47004	29-7/8 - 36-7/8	662	82,589	68,452	2.1	10.1	0.83	10.0	0.76	
47004	99-7/8 - 106-7/8	641	87,054	72,321	3.0	10.6	0.86	11.2	0.90	
47004	130-3/8 - 137-3/8	622	95,238	81,548	2.5	10.9	0.92	11.2	0.89	
47004	137-3/8 - 144-3/8	612	94,048	81,101	2.2	9.4	0.91	11.4	0.91	
47005	24-7/8 - 31-7/8	660	80,357	68,452	2.1	11.2	0.84	10.6	0.66	
47005	31-7/8 - 38-7/8	662	84,226	69,643	3.0	9.2	0.77	12.3	0.77	
47005	90-5/8 - 97-5/8	646	84,077	69,345	2.2	13.2	0.74	13.0	0.90	
47005	99-7/8 - 106-7/8	641	87,054	72,619	2.7	11.7	0.83	11.5	0.91	
47005	130-7/8 - 137-7/8	622	93,006	82,738	3.3	10.1	0.94	11.2	0.92	
47005	137-7/8 - 142-7/8	612	95,238	78,274	3.6	10.0	0.91	10.8	0.72	
Archive-1	NA	NA	69,196	53,571	2.8	12.6	0.69	10.9	NA	Unirradiated
Archive-1	NA	NA	69,940	55,357	4.5	13.8	0.73	10.1	NA	Unirradiated
Archive-1	NA	NA	69,345	52,976	4.7	16.5	0.69	10.2	NA	Unirradiated
Archive-2	NA	NA	70,833	54,167	3.0	14.2	0.66	11.4	NA	Unirradiated
Archive-2	NA	NA	71,726	55,952	2.9	12.9	0.74	10.4	NA	Unirradiated
Archive-2	NA	NA	69,940	54,167	3.6	13.2	0.68	11.2	NA	Unirradiated

(a) Calculated at mid location of specimen.

(b) Based on nominal cladding dimensions with cross-sectional area equal to 0.0336 inch².

(c) Approximate elastic modulus only since extensometer is not class A.

(d) Relative burnup values are valid only on one rod; value represents relative burnup at center of specimen based on gamma scan data.

Table 17. 650F Irradiated and Unirradiated Ring Tensile
Properties of Oconee 1 Fuel Rod Cladding

<u>Rod No.</u>	<u>Specimen width, in.</u>	<u>Ultimate tensile strength, psi</u>	<u>0.2% yield strength, psi</u>	<u>Uniform elongation, %</u>	<u>Total elongation, %</u>
Archive	0.2435	63,566	56,202	2.1	24.8
Archive	0.251	63,158	54,887	2.5	17.8
47004	0.280	83,904	70,548	2.1	15.3
47004	0.2515	80,273	68,045	1.7	13.5
47005	0.260	79,826	68,116	1.5	10.9
47005	0.226	79,167	71,667	1.6	12.4
47106	0.254	81,711	74,074	1.6	13.6
47106	0.228	81,099	71,901	1.5	14.1

Table 18. Summary of Effects of Oconee 1 Cycle 1 Operation
on Tensile Properties of Fuel Rod Cladding

	<u>Unirradiated</u>		<u>Irradiated</u>		<u>Change, %</u>
	<u>$\bar{X}$</u>	<u>S</u>	<u>$\bar{X}$</u>	<u>S</u>	
<u>Ring Tensile Test</u>					
UTS, psi	63,362	--	80,997	1,686	+28
0.2% YS, psi	55,544	--	70,725	2,345	+27
Uniform elong., %	2.3	--	1.7	0.2	-26
Total elong., %	21.3	--	13.3	1.5	-38
<u>Uniaxial Tensile Test</u> (a)					
UTS, psi	70,163	959	82,619	1,927	+18
0.2% YS, psi	54,365	1,108	69,137	1,457	+27
Uniform elong., %	3.6	0.8	2.2	0.4	-38
Total elong., %	13.9	1.4	11.1	1.6	-20

(a) Results from specimens taken from the high temperature region of the fuel rods adjacent to the ring tensile test specimens.

Figure 9. Rod/Grid Frictional Force as Function of Assembly Position

Figure 10. Rod Growth Relative to Fuel Assembly Lattice Position

- 00 Fuel rod growth expressed in inches
- X Control rod guide tube
- X Instrument tube

Figure 11. Initial Gamma-Scan of Fuel Rod 47004

Figure 12. Gamma-Scan of Fuel Rod 47004 After Additional Handling and Storage

Figure 13. Gamma Activity Profile of Fuel Rod 47005

Figure 14. Comparison of ^{134}Cs to ^{137}Cs , Fraction of Fissions From Plutonium, and ^{137}Cs Activity

Figure 15. LVDT Output Data for Diameter Step-Scan on Fuel Rod 47102

Figure 16. 0.125-Inch Step-Scan Data for Fuel Rod 47102

Figure 17. 0.50-Inch Step-Scan Data for Archive Cladding

Figure 18. Fuel Rod Average Creepdown Relative to Fuel Assembly Lattice Position

Figure 19. Creepdown Profile Vs Relative Burnup Profile

(a) Normalized to the assembly peak axial (nodal) burnup of 13,430 MWd/mtU (FLAME-calculated).

Figure 20. Eddy-Current Trace of Fuel Rod 47010

Figure 21. Histogram of Eddy-Current Indications
Between Spacer Grids

Figure 22. Histogram of Eddy-Current Indications at Spacer Grid Locations

Figure 23. Number of Eddy-Current Indications Between Spacer
Grids as a Function of Assembly Lattice Position

- 00 Number of indications on fuel rod
- X Control rod guide tube
- X Instrument tube

Figure 24. Transverse Section, Hydride Platelet Distribution
Near Cladding ID — Fuel Rod 47005, Sample 005-11-E,
Polarized Light

7X

Figure 25. Longitudinal Section, Axial Hydride
Distribution in Polarized Light —
Fuel Rod 47005, Sample 005-22-ML

50X

Figure 26. Transverse Section, Hydride Distribution Near
Bottom of Rod in Polarized Light -- Fuel Rod
47106, Sample 106-29-MT

7X

Figure 27. Low Magnification View of Half the ID of
Top Inch of Section 20E From Fuel Rod
47004

Figure 28. Mosaic of SEM Photographs Showing
Cladding in Section 20E

20X

Figure 29. SEM Photomosaic of Circumferential Linear Marking Seen in Center of Figure 28

Figure 30. SEM Photographs of Adherent Crystalline
Deposits Believed to be UO_2

a. 100X

b. Deposit in Upper Right
Region of a at 500X

Figure 31. Isolated UO_2 Particle on Cladding

2000X

Figure 32. Adherent Oxide From Structure in
Left Center of Figure 30

a. 200X

b. Region in Upper Left
of a at 10,000X

Figure 33a. Photograph 10 From Mosaic in
Figure 29

100X

Figure 33b. Higher Magnification of Central
Region of Figure 33a

1000X

Figure 33c. View of Structural Detail in Left
Center of Figure 33b

5000X

Figure 33d. View of Structural Detail in Right
Center of Figure 33b

5000X

Figure 34. Views of Stained Region in Lower
Right Corner of Figure 28

a. 100X

b. Central Region in
a at 1000X

Figure 35. Effect of Irradiation Temperature on 650F Uniaxial Tensile Strength of Fuel Rod Cladding at EOC 1

4. CONCLUSIONS

Summarized below are pertinent comments on the results of the examinations.

1. Fuel rod extraction loads indicated that the average grid spring relaxation was slightly less than the 60% relaxation anticipated for the internal and peripheral fuel rods, but slightly greater for the one corner rod extracted.
2. Visual examinations verified fuel rod integrity and revealed highly variable and nonuniform crud distribution patterns.
3. Fuel rod axial growth exhibited an average axial strain of $0.135 \pm 0.030\%$, which was well within the design limit.
4. The total fuel column gap length per rod was small but variable. Changes in the length and location of individual gaps noted before and after handling and storage operations and the ease of fuel removal during fuel rod sectioning suggest that many of the gaps between pellets were formed during post-irradiation handling. Furthermore, a comparison of the gamma-scan burnup profiles of adjacent rods suggests that no fuel pellet gaps of significant length were present during incore operation.
5. Fuel stack length decreased an average of 1.65 in. ($1.14\% \Delta l/l$) from beginning of life.
6. Analysis of the axial distribution of fission product isotopes revealed no unexpected trends in the relative neutron flux, burnup, and plutonium fission fraction distributions; no evidence of cesium migration was found.
7. Fuel rod diametral creepdown and ovality averaged 0.0020 and 0.0039 in., respectively. The average creepdown axial profile correlated well with axial burnup. A weaker correlation was noted between creepdown and the calculated cladding operating temperature.

8. A number of indications were obtained during the eddy-current examinations. However, metallographic examinations failed to reveal any type of cladding structural damage that correlated directly with the indications. Superficial grid-cladding interaction effects were observed at various spacer grid contact points, which contributed to a significant number of the eddy-current indications.
9. Cladding wall thickness measurements were somewhat uncertain due to the presence of nonuniform crud deposits. However, the apparent decrease in average wall thickness of 0.0003 in. was well within the range expected to result from general surface corrosion of Zircaloy-4.
10. The composition of the fuel rod crud was about as expected. However, the average surface isotopic concentrations and calculated crud thickness values were considerably less than expected, indicating that about 70 to 90% of the crud had sloughed off of the rods during handling and storage operations.
11. The average volume of fission gas released was quite small — well within the range expected for fuel irradiated at low heat rates.
12. The chemical burnup analyses displayed very good agreement between the two independent methods used, with an uncertainty of about $\pm 2.5\%$ in the values.
13. Fuel pellet densification averaged 2.8% but varied considerably among the pellets selected from two fuel lots. These results were complicated by the effects of differences in the average linear heat rates and burnups between each sample.
14. Fuel pellet metallographic examination revealed slight gradients in grain growth and densification. Evidence of fission gas coalescence at grain boundaries was noted in one sample having a fine grain structure.
15. Cladding metallographic examination revealed no indication of cladding defects associated with fuel-cladding interaction. Superficial effects of corrosion and wear on the cladding OD were noted at spacer grid contact points.
16. Hydride precipitation in the cladding exhibited both plate- and pin-like (or spherical) morphologies with highly variable axial and circumferential distributions.

17. The hydrogen concentration in the cladding varied considerably, from 55 to 257 ppm, with a mean of 124 ppm.
18. The tensile test results indicated that the EOC 1 cladding had undergone an average increase of 37% in yield strength while retaining 69% of its original ductility.

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