HEAVY WATER ORGANIC COOLED REACTOR

POSTIRRADIATION EXAMINATION OF THE SAP-UC FUEL BUNDLES OF THE NRU-305 EXPERIMENT

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ABSTRACT

The U-305 Experiment was conducted to evaluate SAP-UC and Zircaloy- UO_2 fuel systems for the HWOCR program. The U-305 was irradiated in the NRU Reactor at Chalk River, Ontario, Canada, to peak UC fuel burnup near 10,000 Mwd/MTU and was removed due to defects in the Zircaloy- UO_2 portion of the experiment. This report describes the postirradiation examination of SAP-UC Bundle LTA, which experienced the highest burnup.

Bundle LTA was disassembled and all fuel rods were subjected to visual examination, fission gas and gamma scan measurements, fouling film sampling, and profilometer measurements. Fully characterized Fuel Rods B, H, J, and P were destructively examined, including disassembly, visual examination of declad fuel and cladding inner surface, measurement of fuel growth and absolute burnup, and fuel and cladding metallography.

No conclusive evidence of defects or radiation-induced degradation was observed in the SAP-UC. The SAP-UC fuel system demonstrated satisfactory operation at the temperature and burnup levels experienced in the U-305 Experiment.

TABLE 1
COMPARISON OF U-305 AND HWO CR PARAMETERS

Parameter	HWO CR	U-305
Physical Configuration	37-pin hexagonal array	18-pin circular array
Fuel and Cladding	SAP-UC	SAP-UC
Coolant	Santowax OM + 10% HB	Santowax OM + 10-20% HB
Fuel U-235 Enrichment (%)	1.3	1.5
Peak Unperturbed Thermal Neutron Flux (n/cm ² -sec)	2.8×10^{14}	2.4×10^{14}
Peak Initial Specific Power (Mw/MTU)	48.1	46.8
Bulk Average Power Density (Mw/ft ²)	4.11	3.96
Peak Fuel Temperature (°F)	2100	2000
Peak Cladding Temperature (°F)	870	850
Bulk Average Coolant Temperature (°F)	662	661
Coolant Velocity (ft/sec)	30	30
Coolant Pressure (psi)	217	200

I. INTRODUCTION

EXP-NRU-305 (U-305) was an irradiation experiment conducted as a materials evaluation test in support of the joint Atomics International-Combustion Engineering (AI-CE) program on the Heavy Water Organic Cooled Reactor (HWOCR). The experiment consisted of five 18-fuel-rod bundles mounted in tandem: three bundles contained Sintered Aluminum Powder (SAP) clad uranium carbide fuel (UC), and two bundles contained Zircaloy-4 clad uranium oxide fuel (UO_2). The primary objectives of the U-305 test⁽¹⁾ were: evaluation of the irradiation performance of SAP-UC and Zircaloy- UO_2 systems under HWOCR operating conditions; and development of data concerning UC fuel growth and fission gas release, SAP cladding behavior, coolant chemistry and fouling film formation. The ability of the U-305 Test to meet these objectives required similarity between test design⁽²⁾ and design of the reference HWOCR demonstration plant.⁽³⁾ A comparison of operating and performance parameters of HWOCR and the SAP-UC system in U-305, presented in Table 1, indicates HWOCR and U-305 were very similar.

This report describes the post-irradiation examination of the SAP-UC bundles contained in U-305. Comprehensive examination was restricted to the bundle that experienced the highest burnup, LTA, while the other bundles were visually examined and selected fuel rods used in some phases of the examination. A detailed description of the U-305 Experiment's design, analysis, and irradiation may be found elsewhere.⁽⁴⁾

Figure 1. U-305 Test Assembly

II. EXPERIMENT DESCRIPTION

A. U-305 ASSEMBLY

The U-305 Test Assembly contained three 16-5/8-in. long SAP-UC bundles and two 25-1/2-in. Zircaloy- UO_2 bundles, mounted in tandem, as shown in Figure 1. Each of the fuel bundles contained 18 fuel rods. These rods were arranged in a circular array with the center fuel rod removed to make room for a 3/8-in. diameter stainless steel hanger rod that passed through all of the bundles (Figure 2). The hanger rod was contained in a dummy cladding tube to preserve hydraulic characteristics and house instrumentation leads. An Inconel-X spring, located at the bottom of the assembly, maintained a compressive load to assure nesting of the bundles and allow for thermal expansion. The end plates, which held the fuel rods in place, were keyed together to prevent rotation of the bundles. The top end plate of the uppermost bundle was keyed to the upper retainer assembly. This assembly was welded to a hanger tube which supported the experiment in the reactor.

B. SAP-UC FUEL BUNDLES

The SAP-UC fuel bundles were 16-5/8-in. long, including the 1/16-in. stainless steel end plates. Diameter over the fins was 3.175 in. A typical SAP-UC bundle is shown in Figure 3.

The cladding material was SAP containing 5 to 8 wt % Al_2O_3 , extruded into finned tubing with the geometry indicated in Figure 4. The tubes were twisted after extrusion so that the fins spiraled $90^\circ/\text{ft}$. Cladding inner diameter was 0.5045 ± 0.0005 -in., yielding a nominal 0.0050-in. diametral gap between fuel and cladding. End plugs were fabricated from Type 1100-S aluminum, eutectic bonded to the cladding tube in a helium atmosphere. Fabrication procedures and pre-irradiation data for the cladding are presented in Reference 5.

The fuel material was arc-cast hyperstoichiometric UC, enriched to 1.5 wt % U-235. Fuel slugs were of random length (1 to 6 in.), with an outer diameter of 0.4995 ± 0.0005 in., dished at one end to a spherical radius of 0.055 in. Fuel slugs were loaded in cladding tubes to give a fuel stack length of approximately 13.6 in., as shown in Figure 4. The fuel compositions used in

Figure 2. U-305 SAP-UC Bundle
LTA Schematic

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Figure 3. Pre-Assembly Condition of U-305 SAP-UC Bundle LTA

Figure 4. U-305 SAP Cladding and Fuel Rod Schematic

U-305 are given in Table 2. Fabrication procedures and pre-irradiation data for the fuel are presented in Reference 6. All three SAP-UC bundles (LTA, LSA, LRA) were fabricated at AI.

TABLE 2
UC FUEL COMPOSITIONS IN U-305

Bundle	Fuel Rod Identification	Fuel Composition Range (wt % C)
LRA, LSA, LTA	B, C, D	4.82 - 4.92
	E, F, H	4.92 - 5.00
	J, K, L	5.00 - 5.10
	M, N, P	5.10 - 5.20
	R, S, T, W, X, Y	4.92 - 5.20

C. ZIRCALOY- UO_2 FUEL BUNDLES

The two Zircaloy-4/ UO_2 bundles were 25-1/2-in. long. One bundle, JRA, contained fuel rods with six spacer wires attached to the outside of the tube wall

by electron beam welding; the other bundle, JPA, contained fuel rods with spacer wires attached by spot welding. The Zircaloy-4 tubing was twisted to give a $90^\circ/\text{ft}$ spiral to the spacer wires in each bundle. The Zircaloy cladding material conformed to ASTM Specification B-35362T. Tubing inner diameters ranged from 0.531 to 0.534 in., yielding diametral gaps of 0.009 to 0.012 in. Clearance between end plug and fuel stack was 0.160 in. (nominal).

The UO_2 fuel was in the form of cored pellets, approximately $1/2$ -in. long, with an inner diameter of 0.184 in. and an outer diameter of 0.522 ± 0.0005 in. Each pellet was dished at one end to a spherical radius of 0.055 in. The fuel was enriched to 1.12 wt % U-235 in the top bundle, 1.52 wt % U-235 in the bottom. Each fuel rod contained a fuel stack $24-3/8$ -in. long, including solid pellets of depleted UO_2 , $3/8$ -in. long, located on both ends of the fuel stack. The UO_2 fuel conformed to CEND Specification NL-102.

One of the Zircaloy- UO_2 bundles, JRA, was fabricated at CE; while the other, JPA, was fabricated by Westinghouse of Canada for CE.

III. OPERATING HISTORY

The U-305 irradiation facility was the organic cooled U-3 loop of the NRU reactor,^(4,7) located at Chalk River National Laboratory (CRNL), Chalk River, Ontario, Canada. The U-305 Experiment was assembled at CRNL from AI and CE components and inserted during July of 1965. The U-305 Test attained a peak UC fuel burnup near 10,000 Mwd/MTU between startup on July 29, 1965 and April 8, 1966, when a defect was detected. The experiment was operated in the defected condition until April 16, 1966, when it was removed from the reactor and transferred to the universal cells at CRNL.

The bundles were separated, and a stereomicroscopic examination was conducted for each bundle. Obvious defects were noted in four Zircaloy- UO_2 rods: D and H in Bundle JRA, C and E in Bundle JPA.⁽⁴⁾ No other defects or anomalies were noted in the remainder of the components. All five bundles were shipped to the AI Hot Laboratory (AIHL). Subsequently, the two Zircaloy- UO_2 bundles were returned to CRNL.

BUNDLE LTA - BOTTOM END PLATE
A - A

2X

RODS B, J AND H SHOWING FINS DAMAGED DURING HANDLING. NOTE THAT ONE FIN IN J CONTAINED A TC.

BUNDLE LTA - TOP END PLATE
C - C

2X

RODS M AND K SHOWING TYPICAL AREAS OF BARE CLADDING AT OTHER ORIENTATIONS OF THE BUNDLE. THE DARKER AREAS OF ROD M ARE CLADDING DISCOLORATIONS

2X

RODS H, P AND D SHOWING FILM ON THESE RODS AT THE MID-PLANE OF THE BUNDLE.

2X

RODS B, J AND H AT THE INTERFACE BETWEEN THE FUELED AND UNFUELED PORTIONS OF THE ROD. NOTE THE CHANGE IN FILM CONDITION

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Figure 5. Post-irradiation Condition of U-305 SAP-UC Bundle LTA at the 0° Orientation

IV. POST-IRRADIATION EXAMINATION

A. SCREENING EXAMINATION

The screening examination consisted of: (1) a thorough visual inspection supported by a photographic record, (2) fission gas sampling of all the fuel rods, (3) gamma scanning of the rods, (4) measurement of fuel rod bowing, and (5) fouling film examination.

1. Visual Inspection

Bundle LTA was visually examined by Kollmorgen periscope and stereo-microscope. The excellent condition of the bundle noted at Chalk River⁽⁴⁾ was confirmed. Figure 5 shows a Kollmorgen composite of Bundle LTA at 0° orientation (hot side) with higher magnification stereophotos of selected locations on the bundle.

After completion of the visual examination, the bundle was disassembled by drilling out the screws and holding pins of each rod with an appropriately sized center drill and removing the end plates. Each rod was identified and placed into 6-rod storage and transport fixtures for further examination.

Window photos were taken of all rods at three orientations within their transport fixtures, and all rods were examined by Kollmorgen periscope. A high power periscope composite was made of Rods H, J and T; other photos were taken of selected areas as required. No anomalous conditions were noted on any rods.

Figures 6 and 7 show the Kollmorgen composites of Rods J and T, respectively, with higher magnification photos of selected areas on the rods. Definite trends in film conditions were observed. A double-layer variety of film, which can be seen most readily in Figure 6 (Rod J), has been noted on the surfaces of the outer ring of rods closest to the outer process tube. The surfaces of the outer rods next to other fuel rods exhibit a flaky single layer-type film. The film formations on the inner rods are very light. Only a very thin single layer flaky film was noted on these rods. Rod T, shown in Figure 7, is representative of the inner rods. The film seemed to follow the temperature distribution of the bundle in that the heaviest deposits of film appeared at the hotter areas,

66-7680-101-2

Figure 6. Post-irradiation Condition of U-305 Bundle LTA Fuel Rod J at the 90° Orientation

ROD T AT THE INTERFACE BETWEEN THE
FUELED AND UNFUELED PORTION OF THE
ROD

ROD T SHOWING THE SMOOTH AND FLAKY
FILM CHARACTERISTIC OF THE INNER RODS

ROD T TYPICAL AT 180° ORIENTATION

66-7680-101-4

Figure 7. Post-irradiation Condition of U-305 Bundle LTA Fuel Rod T at the 0° Orientation

and conversely, the lightest at the colder areas. It is also believed that some of the film conditions may be the result of overnight exposure to air inside a fuel handling cask during the examination at CRNL; this could have resulted in cladding surface temperatures up to 320°F.⁽⁴⁾

Visual examination of Bundles LSA and LTA revealed no defects or anomalies. Fuel rod conditions were similar to those observed for Bundle LTA fuel rods.

2. Fission Gas Analysis

A fission gas sample was taken from each element by drilling through the upper end cap while the element was in a vacuum chamber. The gas released was collected on activated charcoal in glass traps and subsequently analyzed by gamma spectrometry for Kr⁸⁵ content. The fission gas data are presented in Table 3 and Figure 8.

TABLE 3
FISSION GAS RELEASE FOR U-305 BUNDLE LTA

Fuel Rod	Gas Release (% Theoretically Produced)	Remarks
Outer Ring		
D	1.642×10^{-2}	
P	1.729×10^{-2}	
M	1.532×10^{-2}	
B	1.648×10^{-2}	
K	2.739×10^{-2}	
J	3.209×10^{-2}	
H	1.609×10^{-2}	
E	2.033×10^{-2}	
N	-	Drill broke off in specimen
L	2.501×10^{-2}	
F	2.233×10^{-2}	
C	-	Trap broke
Inner Ring		
Y	8.530×10^{-3}	
X	8.225×10^{-3}	
W	7.971×10^{-3}	
T	-	Lost system vacuum
S	8.355×10^{-3}	
R	8.305×10^{-3}	

a. Outer Ring Fuel Rods

b. Inner Ring Fuel Rods

Figure 8. Fission Gas Release for U-305 Bundle LTA

TABLE 4
SUMMARY OF GAMMA SCAN DATA FOR FUEL RODS FROM
U-305 BUNDLE LTA

Rod	Orientation (Degrees)	Average counts/sec	Peak counts/sec	End-Peaking (%)	
				Top	Bottom
B	0	2104	2623	25.5	0
	90	2265	2725		
C *	0	2232	2568	17.0	0
	270	2051	2487		
J	0	2296/2273 *	2703/2533 *	11.1	0
	90	2372	2671		
	270 *	2337	2517		
P	0	2303/2164 *	2672/2504 *	14.2	0
	90	2371	2677		
	270 *	2095	2443		
F	0	2342/2021 *	2507/2425 *	17.5	0
	90	2316	2444		
	270	1955	2325		
N	0	2273	2387	9.0	0
	90	2226	2372		
W	0	1481	2215	50.0	29.0
	90	1569	2310		
Y	0	1536	2437	62.0	23.0
	90	1457	2338		
T	0	1578	2395	55.0	26.0
	90	1581	2289		
	180	1559	2298		
	270	1560	2320		

*The 270° orientations and the second 0° orientations were scanned approximately 14 days after the original 0° and 90° orientations.

Examination of the data shows the average fission gas release of rods in the outer ring to be about 2-1/2 times that of the rods in the inner ring, reflecting the somewhat higher burnup of the former. The plot of fission gas release versus carbon content for the lower burnup, lower release level inner rods (Figure 8b) shows no correlation. However, the plot for outer ring rods (8a) shows lowered fission gas release in the 5.05 wt % carbon region.

3. Gamma Scans

Fuel rods were gamma scanned to evaluate power distribution and to allow determination of burnup at any point in any rod by correlation with quantitative burnup data. Gamma scanning used a scintillation counter that counted and discriminated the Zr^{95} photopeak as the rod was moved past a small counting window.

A summary of the results from the gamma scans of selected rods is given in Table 4. These scans indicated that the inner ring of rods ran at approximately 65% the power level of the outer ring. A few rods were gamma scanned at 180° and 270° orientations to show the azimuthal variation in activity for an individual rod. The total variation was approximately 3% for the outer rods and 9% for the inner rods. Representative scans of an inner and outer rod are shown in Figure 9. With reference to Figure 9, end peakings of approximately 11% at the top of the outer rods, and approximately 55% at the top, and 26% at the bottom of the inner rods were observed. It is of interest to note that all these observations are very close to those predicted during design nuclear analyses.⁽⁴⁾

4. Bow Measurements

Rod bowing was measured on the in-cell profilometer device. The profilometer is normally capable of measuring rod bow and diameter, but in the case of the U-305 rods, the closely spaced spiral fins prevented the transducer shoes from contacting the outside of the cladding. Measurements, therefore, were made along the top of the fins and were converted to rod bowing data. Subsequently, methods were developed to measure cladding outside diameters for another examination.⁽⁸⁾

Figure 9. Typical Gamma Scans for Outer (J) and Inner (T) Fuel Rods of U-305 Bundle LTA

Figure 10. Typical Profilometer Scan Over the Fins of U-305 Bundle LTA Fuel Rod N

The results of the profilometer measurements indicate that all the rods are bowed out from the center of the bundle. A summary of the maximum bow measured for each rod at four orientations is given in Table 5, and a typical series of scans is shown in Figure 10. The maximum bow measured was 44 mils for Rod L. There was a nominal 50 mils clearance between the rods and the outer process tube, indicating no contact during operations.

5. Film Examination

Film flakes were taken from all but two of the LTA rods, and a composite sample from the outer rods and a composite from the inner rods were measured and analyzed. Film thickness data given in Table 6 indicate a mean thickness of 1.22 mils for outer rods and 0.65 mil for inner rods. A statistical test shows that the difference between the inner and outer rods is significant at the 98% level.

The gross beta-gamma activity of the flakes varied from 250 to 5000 mr/hr for the outer rods, and from 250 to 400 mr/hr for the inner rods. This agrees with the lower thermal neutron flux experienced by the inner rods. The most important radionuclides were Cs^{137} , Mn^{54} , and Co^{60} .

Only the outer rod composite was large enough to permit chemical analysis. Carbon was 69.4%, hydrogen 2.2%, and ash 4.8%, leaving 24% unaccounted for. It was not possible to determine oxygen by fast neutron activation, because of the radiation background of the samples. The most important ash constituents were aluminum and iron. Zirconium was only 0.02%, in spite of the large amount of zirconium in the U-3 system.

6. Spring Test

A spring deflection test was performed in-cell on the Inconel-X compression spring. Figure 11 shows the spring before irradiation, and Figure 1 shows the spring's location in the U-305 assembly. The spring, located at the bottom of the test assembly, was immersed in 661°F coolant while accumulating a thermal flux of about 10^{20} nvt. The spring was placed between the jaws of a remote-controlled Instron TT-C testing machine that applied equal increments of compressive load. Figure 12 shows a plot of spring deflection versus load. This plot shows a linear decrease in spring length until the nested, or solid length,

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Figure 11. Inconel-X Compression Spring From the Bottom of the U-305 Before Assembly

Figure 12. Post-irradiation Spring Deflection vs Load for Inconel-X Compression Spring

TABLE 5
SUMMARY OF BOW MEASUREMENTS FOR FUEL RODS
FROM U-305 BUNDLE LTA

Ring	Rod	Maximum Bow (mils)			
		0°	180°	90°	270°
Outer	B	25	22	22	19
	C	17	17	18	15
	D	22	17	22	17
	E	28	28	6	4
	F	22	28	8	10
	H	42	42	12	10
	J	42	32	41	41
	K	35	34	15	19
	L	44	38	42	30
	M	32	31	23	20
	N	15	13	31	32
	P	29	24	10	8
Inner	R	21	13	10	10
	S	13	17	16	16
	T	20	20	14	12
	W	7	4	10	12
	X	10	10	17	18
	Y	19	25	11	13

TABLE 6
FILM THICKNESS DATA FOR U-305 BUNDLE LTA

	Composite Sample From Outer Rods	Composite Sample From Inner Rods
Number of Measurements	19	8
Range (mils)	0.2 - 2.39	0.2 - 1.89
Median (mils)	1.2	0.5
Mean (mils)	1.22	0.65
Standard Deviation	0.55	0.52

was attained. The slope of the linear portion of the curve represents the spring constant, which was determined as 204 lb/in. The design spring constant was 180 lb/in; however, this value was calculated, not measured.⁽⁵⁾ The pretest free length was 5.848 ± 0.002 in., the post-test free length was 5.845 ± 0.002 in., and the solid length was 3.998 ± 0.002 in.

No damage due to reactor operations was noted in the Inconel-X spring, as evidenced by the insignificant change in the spring constant and the lack of change in the spring free length before and after testing, indicating elastic behavior.

B. DETAILED EXAMINATION

Selected fuel rods from Bundle LTA were subjected to a detailed destructive examination. This examination consisted of: (1) disassembly of fuel rods, (2) visual examination of fuel and cladding inner surface, (3) measurement of fuel growth, (4) fuel burnup determination, and (5) fuel and cladding metallography. Elevated temperature stress-rupture tests of SAP cladding tubes were conducted but are separately reported.⁽⁸⁾

Fuel Rods B, H, J, and P of Bundle LTA were selected for detailed examination. These rods were the only ones in this bundle containing fully characterized fuel, and they therefore offered the opportunity to make a thorough comparison of pre- and post-irradiation characteristics. Subsequently, LTA Rod N was sampled for metallography to provide backup information.

1. Fuel Rod Disassembly

Fuel Rods B, H, and J were disassembled by removing the end caps and slitting the cladding into two halves by Elox machining. The fuel and inner cladding surfaces of these rods were then exposed for examination. Rod P was disassembled by sectioning into 1-in. wafers from various locations. These wafers were then used for fuel and cladding metallography, and fuel-cladding compatibility study. Subsequently, the cladding was removed from the remaining sections of Rod P by Elox machining, and the fuel and cladding were subjected to detailed examination.

2. Visual Examination

Composite photographs were made of the fuel and the inner surfaces of the cladding, and higher magnification stereophotographs were taken of selected areas. Figures 13, 14, and 15 show the fuel and cladding from Rods B, H, and J, respectively. The fuel is relatively intact and the individual slugs are easily discernible. The fuel cracks seen in Figure 13 occurred during separation of the fuel and cladding. The inner surfaces of the cladding exhibited stains and light colored deposits (the white or silver dots seen in the above mentioned figures). The stains appeared to correspond to the interfaces of the individual fuel slugs and to a surface cracking pattern observed on the fuel. This surface cracking pattern is characteristic of a stress-relieved condition in a cast rod. Some large fuel pieces were also observed on the cladding at the interfaces between slugs, but these were not adherent to the cladding and are thought to be the result of fuel shifting during handling and not an in-pile condition. The small spherical deposits were tightly adherent to the cladding and distributed generally over the fuel-cladding interface. However, there was a concentration of deposits in regions of fuel slug interfaces and cracks. The amount of these deposits observed appears to increase with lower carbon level (i. e. Rods B and H exhibited a higher incidence of these deposits than Rods J and P). There were no detrimental anomalies observed on any fuel or cladding.

3. Fuel Growth

Diameter and limited length measurements were taken of the fuel slugs. Not many length measurements were taken because the slugs were either cracked or chipped and no consistent measurements could be obtained. The diameters were measured at 1/2-in. intervals and two orientations along the slugs using a dial indicator V-block system. The measurements were taken by comparing the dial indicator readings from a tip in contact with a standard of known diameter and a tip in contact with the fuel slug. The accuracy of these measurements was ± 0.0005 in. This accuracy was comparable to that of the pre-irradiation tolerances⁽⁶⁾ on the fuel diameter (0.499 ± 0.0005 in.), i. e. $\pm 0.1\%$. With these accuracies for the individual measurements, the most probable error in the comparison, $\Delta D/D_i$, is then 20%.

2-10-67 UNC

7680-25974

Figure 13. Fuel and Cladding From U-305 Bundle LTA Rod B

2-10-67 UNC

7680-25975

Figure 14. Fuel and Cladding From U-305 Bundle LTA Rod H

2-10-67 UNC

7680-25976

Figure 15. Fuel and Cladding From U-305 Bundle LTA Rod J

The densities were measured by the Archimedes Principle, using calibrated V-60 oil as the weighing liquid. The accuracy of the pre- and post-irradiation measurements is approximately 0.07%. With these accuracies the most probable error in the comparison, $\Delta\rho/\rho_f (\approx \Delta V/V_i)$, is then 8.3%.

Density and dimensional data are tabulated by slug for Rods B, H, J, and P in Tables 7, 8, 9, and 10, respectively. Figure 16 presents the percentage volume changes calculated from the density measurements ($\Delta\rho/\rho_f$) plotted against those calculated from dimensional measurements ($3 \Delta D/D_i$). Isotropic growth would be indicated by a one-to-one relationship between the two measurements. The fuel volume changes calculated from density changes were plotted as a function of carbon content in Figure 17, using the average burnup.

Comparison of the density and dimensional measurements shown in Figure 16 indicates that the fuel diameters and lengths, where available, grew more than predicted by density measurements, so that $\Delta\rho/\rho_f = 2.5 \Delta D/D_i$. This comparison, however, is clouded by the poor accuracy of the dimensional measurements. It is for this reason that the density measurements are used for the correlation of swelling with carbon content shown in Figure 17.

4. Fuel Burnup

Fuel samples taken from various points in the four clad rods were analyzed for isotopic uranium content by mass spectrographic methods. The isotopic uranium content of each irradiated sample was compared with that of nonirradiated archive samples to yield a measure of U^{235} burnup. Burnup results are presented in Table 11 and Figure 18. The results in the figure show a very good correlation between gamma scan and mass spectrographic data, indicating gamma scan data can be used to determine burnup values not determined by analysis.

5. Fuel Metallography

Transverse sections were removed from three to five positions along the length of LTA Fuel Rods B, H, J, N, and P. Samples were vacuum impregnated with epoxy resin mounting compound, polished with oxalic acid slurries of 6μ diamond and 0.3μ Linde-B, and etched with a 1:1:1 mixture of nitric acid, acetic acid and water.

TABLE 7
U-305 BUNDLE LTA FUEL ROD B DENSITY
AND DIMENSIONAL DATA

Slug	Carbon (wt %)	Diameter (0°)	$\Delta D/D_i$ (%)	Diameter (90°)	$\Delta D/D_i$ (%)	ρ_i	ρ_f	$\Delta\rho/\rho_f$ (%)
B11-7A	4.90	0.5033	0.86	0.5035	0.90	13.52	13.27	1.88
		0.5026	0.72	0.5030	0.80			
		0.5027	0.74	0.5021	0.62			
		0.5013	0.46	0.5023	0.66			
		0.5022	0.64	0.5025	0.70		13.30	1.65
		0.5021	0.62	0.5022	0.64			
		0.5018	0.56	0.5024	0.68			
		0.5022	0.64	0.5022	0.64			
B11-27A	4.88	0.5015	0.50	0.5020	0.60	13.51	13.28	1.73
		0.5020	0.60	0.5027	0.74			
		0.5023	0.66	0.5021	0.62			
		0.5025	0.70	0.5022	0.64			
		0.5017	0.54	0.5020	0.60		13.28	1.73
		0.5017	0.54	0.5020	0.60			
		0.5015	0.50	0.5020	0.60			
		0.5020	0.60	0.5018	0.56			
B11-10A	4.85	0.5024	0.68	0.5022	0.64	13.52	13.29	1.73
		0.5024	0.68	0.5025	0.70			
		0.5022	0.64	0.5024	0.68			
		0.5023	0.66	0.5026	0.72			
		0.5017	0.54	0.5018	0.56			
		0.5024	0.68	0.5022	0.64			
		0.5025	0.70	0.5027	0.74			
		0.5024	0.68	0.5026	0.72			
B11-7	4.90	0.5015	0.50	0.5024	0.68	13.53	13.27	1.95
		0.5018	0.56	0.5027	0.74			
		0.5024	0.68	0.5025	0.70			
		0.5014	0.48	0.5024	0.68			
B11-29	4.87	0.5021	0.62	0.5017	0.54		13.32	1.57
		0.5020	0.60	0.5018	0.56			
		0.5016	0.52	0.5023	0.66			

TABLE 8
U-305 BUNDLE LTA FUEL ROD H DENSITY AND DIMENSIONAL DATA

Slug	Carbon (wt %)	Diameter (0°)	$\Delta D/D_i$ (%)	Diameter (90°)	$\Delta D/D_i$ (%)	ρ_i	ρ_f	$\Delta\rho/\rho_f$ (%)
H-7-9	4.97	0.5018	0.56	0.5014	0.48	13.51	13.29	1.66
		0.5017	0.54	0.5010	0.40			
		0.5020	0.60	0.5023	0.66			
H-7-23	4.95	0.5018	0.56	0.5013	0.46	13.51	13.28	1.73
		0.5023	0.66	0.5023	0.66			
		0.5022	0.64	0.5022	0.64		13.27	1.81
		0.5024	0.68	0.5023	0.66			
		0.5021	0.62	0.5021	0.62			
H-7-29	4.98	0.5021	0.62	0.5022	0.64	13.50	13.27	1.73
		0.5021	0.62	0.5026	0.72			
		0.5023	0.66	0.5026	0.72			
		0.5018	0.56	0.5022	0.64			
		0.5021	0.62	0.5023	0.66			
		0.5024	0.68	0.5027	0.74			
H-7-26	4.93	0.5020	0.60	0.5023	0.66	13.51	13.29	1.66
		0.5018	0.56	0.5018	0.56			
		0.5023	0.66	0.5019	0.58		13.27	1.81
		0.5019	0.58	0.5013	0.46			
		0.5022	0.64	0.5008	0.36			
		0.5017	0.54	0.5020	0.60			
H-7-18	4.94	0.5032	0.84	0.5044	1.08	13.51	13.24	2.04
		0.5030	0.80	0.5043	1.06			
		0.5033	0.86	0.5031	0.82			
		0.5012	0.44	0.5015	0.50		13.26	1.88
		0.5010	0.40	0.5008	0.36			
		0.5025	0.70	0.5024	0.68		13.28	1.73
		0.5021	0.62	0.5023	0.66			

TABLE 9
U-305 BUNDLE LTA FUEL ROD J DENSITY AND DIMENSIONAL DATA

Slug	Carbon (wt %)	Diameter (0°)	$\Delta D/D_i$ (%)	Diameter (90°)	$\Delta D/D_i$ (%)	ρ_i	ρ_f	$\Delta\rho/\rho_f$ (%)
J-6-8A	5.01	0.5047	1.14	0.5035	0.90	13.49	13.20	2.19
		0.5041	1.02	0.5032	0.84			
		0.5038	0.96	0.5034	0.88			
		0.5042	1.04	0.5031	0.82		13.25	1.81
		0.5018	0.56	0.5033	0.86			
		0.5024	0.68	0.5028	0.76			
		0.5022	0.64	0.5026	0.72		13.25	1.81
J-6-1A	5.04	0.5021	0.62	0.5026	0.72	13.43		
		0.5018	0.56	0.5027	0.74			
		0.5013	0.46	0.5022	0.64		13.18	1.89
		0.5016	0.52	0.5020	0.60			
		0.5022	0.64	0.5023	0.66			
		0.5026	0.72	0.5022	0.64			
J-6-8B	5.01	0.5039	0.98	0.5029	0.78	13.45	13.22	1.73
		0.5030	0.80	0.5031	0.82			
		0.5037	0.94	0.5035	0.90			
		0.5030	0.80	0.5036	0.92			
		0.5037	0.94	0.5032	0.84		13.23	1.66
		0.5033	0.86	0.5038	0.96			
J-6-20B	5.08	0.5024	0.68	0.5027	0.74	13.40		
		0.5027	0.74	0.5022	0.64			
		0.5028	0.76	0.5036	0.92		13.18	1.66
		0.5033	0.86	0.5031	0.82			
		0.5030	0.80	0.5030	0.80			
		0.5032	0.84	0.5028	0.76			
J-6-6C	5.05	0.5024	0.68	0.5031	0.82	13.41	13.18	1.74
		0.5030	0.80	0.5034	0.88		13.18	1.74
		0.5028	0.76	0.5032	0.84		13.18	1.74
		0.5033	0.86	0.5035	0.90		13.18	1.74

TABLE 10
U-305 BUNDLE LTA FUEL ROD P
DENSITY DATA

Slug	Carbon (wt %)	ρ_i	ρ_f	$\Delta\rho/\rho_f$ (%)
P17-29A	5.20	13.38	13.13	1.90
P17-27	5.22	13.39	13.14	1.90
P17-7	5.24	13.37	13.10	2.06
P17-17	5.19	13.38	13.15	1.75
		13.38	13.14	1.83
		13.38	13.15	1.75

TABLE 11
U-305 BUNDLE LTA BURNUP DATA

Sample	Fuel Rod	Rod Location (From Top) (in.)	Burnup (Mwd/MTU)
1	B	Top End	10,310
2	B	7	8,045
3	B	12.6	8,244
4	J	2	9,932
5	J	6	8,329
6	J	9	8,706
7	J	11.6	6,946
8	H	2	9,474
9	H	8	8,839
10	H	Bottom End	8,084
11	P	1	8,357
12	P	3	8,158

Figure 16. Comparison of Density and Dimensional Measurements of UC Fuel Growth in U-305 Bundle LTA Rods B, H, J, and P

Figure 17. UC Fuel Swelling vs Carbon Content for Fuel Slugs from U-305 Bundle LTA Rods B, H, J, and P

Figure 18. Comparison of Mass Spectrograph and Gamma Scan Burnup Data for U-305 Bundle LTA Fuel Rods B, J, and P

Typical microstructures for Rods P and N (5.16 to 5.28 wt % C before irradiation) are shown in Figure 19. The Widemanstätten pattern of UC_2 platelets precipitated in a UC matrix appears to be identical with the pre-irradiation structure.⁽⁶⁾

Typical microstructures for Rods B, H, and J (4.88 to 5.04 wt % C before irradiation) are shown in Figure 20. The center and mid-radius area (Figure 20a), shows UC_2 platelets in a UC matrix, similar to the unirradiated structure.⁽⁶⁾ The center is partially obscured by an etching phenomenon. The periphery of the sample (Figure 20b), however, appears to be very close to stoichiometric UC, with no UC_2 platelets visible. This apparent gradient in carbon content was noted in all samples from B, H, and J (with an initial 4.88 to 5.04 wt % carbon content) and not in Rods P and N (with an initial carbon content of 5.16 to 5.28 wt %).

An unsuccessful effort was made to confirm the existence of this carbon gradient by chemical analysis. Small central and peripheral fuel samples were taken from wafers immediately adjacent to wafers used for metallography; these small samples were gravimetrically analyzed for carbon content. The results

ETCHED

91-8-1

a. UC_2 Platelets in a UC Matrix. LTA Rod P,
4-in. From Top, at Periphery

ETCHED

96-4-5

b. UC_2 Platelets in a UC Matrix, and Microcracking
LTA Rod N, 12-in. From Top, at Center

Figure 19. Typical Microstructure for Irradiated LTA Rods P and N
(5.16 to 5.28 wt % Carbon)

ETCHED

91-2-2

a. Center and Mid-Radius Area, UC_2 Platelets in a UC Matrix, Obscured by Etching Phenomenon

ETCHED

91-2-1

b. Periphery, Showing UC Near Stoichiometric Area

Figure 20. Microstructures Typical of Irradiated LTA Rods B, H, and J (4.88 to 5.04 wt % Carbon) Showing Apparent Gradient. Sample Shown From LTA Rod B, 10-in. From Top

were scattered and probably indicate the need for more precisely located fuel samples and a more accurate analytical method.

6. Cladding Examination

a. Metallography

Transverse sections were removed from cladding halves of LTA Fuel Rods H, J, and B. Transverse sections, containing fuel and cladding, were taken from LTA Rods P and N. Samples were mounted in epoxy resin, polished with water slurries of 6μ diamond and 0.3μ Linde-B, and etched in 10% NaOH.

All samples from areas containing a fuel/cladding interface exhibited a tightly adherent, discontinuous deposit on the inner surface of the cladding, as shown in Figures 14 and 21. The deposits extended from the fuel/cladding interface into the fuel, not into the cladding. An exception to this was found in areas of high mechanical interference between fuel and cladding, as in Figure 21c, which shows an embedded fuel particle surrounded by the reaction product. The discontinuous reaction product was noted only in areas of fuel/cladding interference, about 30-45%, and not in other areas, as shown in Figure 22. The reaction product showed an average thickness of 0.0008 in. in LTA Rods B, H, J, N, and P. Table 12 presents a summary of measurements taken from 500X photomicrographs. Accuracy of these measurements was conservatively placed at ± 0.0001 in.

The reaction product responded to metallographic processing in a manner that indicated it was an intermetallic compound of high aluminum content. Unfortunately, no analyses were performed to support or refute this hypothesis. Extensive work performed on UC-SAP reaction zones observed in studies with unirradiated materials are reported separately.⁽⁹⁾

Aside from the UC-SAP reaction product, no anomalies were noted in the SAP microstructure following irradiation. Comparison of the unirradiated control specimen with an irradiated specimen, illustrated in Figure 23, indicated basically identical material.

b. Analyses

Cladding samples from declad Rods B, J, and H were chemically and spectrographically analyzed. Spectrographic analysis results are listed in

ETCHED

85-2-2

a. Reaction Product 0.5 to 1.0-mil Thick on LTA Rod H, 7-1/4-in. From Top

ETCHED

98-2-2

b. Reaction Product 0.5-mil Thick on LTA Rod J, 14-1/4-in. From Top

ETCHED

85-4-2

c. UC Fuel Particle Embedded in SAP Cladding, Reaction Product Surrounding Particle. LTA Rod H, 10-1/2-in. From Top

Figure 21. Typical Deposits of UC-SAP Reaction Product in U-305 Bundle LTA

ETCHED

84-2-2

a. Area of Fuel/Cladding Interference Showing UC-SAP Reaction Product

ETCHED

84-2-6

b. Area 90° From That Shown in View a. No Fuel/Cladding Interference; no UC-SAP Interaction

Figure 22. Fuel/Cladding Interface 8-1/2-in. From Top of LTA Rod P

a. Unirradiated, Processed In-Cell as Control

b. 14-1/2 in. From Top of LTA Rod J

Figure 23. Comparison of Outside Edges of Unirradiated and Irradiated SAP Cladding From U-305 Experiment

AI-CE-76

TABLE 12

THICKNESS OF UC-SAP INTERACTION ZONES

LTA Fuel Rod	Carbon (wt %)	Distance From Top (in.)	Thickness (in.)
B	4.85 - 4.90	3-1/2	0.0009
		8-1/2	0.0010
		15	0.0009
			Average = 0.0009
H	4.94 - 4.98	1	0.0000*
		5-1/2	0.0006
		7-1/4	0.0005
		10-1/2	0.0010
		10-1/2	0.0010
			Average = 0.0008
J	5.04 - 5.08	3-1/2	0.0010
		5	0.0008
		14-1/4	0.0005
			Average = 0.0008
N	5.16 - 5.28	2-1/2	0.0011
		6	0.0006
		9	0.0006
		12	0.0006
		14-1/2	0.0007
			Average = 0.0007
P	5.19 - 5.22	4	0.0011
		8-1/2	0.0009
		12-1/2	0.0011
			Average = 0.0010
		Average for B, H, J, N, P = 0.0008	

*Above fuel stack

Table 13 and chemical analysis results are presented in Table 14. The data indicated no significant change in cladding carbon content, but hydrogen content increased approximately eightfold.

Cladding samples were also analyzed to determine the uranium content of the SAP surface. Samples were subjected to a water wash followed by an HNO_3 wash and the washings were analyzed for uranium. In all cases, results indicated uranium content less than $10\mu\text{g}/\text{mg}$ SAP.

TABLE 13

SPECTROGRAPHIC ANALYSIS OF IRRADIATED SAP CLADDING

Element	Weight Percent			
	Rod B at 3 in.	Rod B at 10 in.	Rod J at 8 in.	Rod H at 8 in.
B	<0.002	<0.002	<0.002	<0.002
Ca	<0.003	<0.003	<0.003	<0.003
Cd	<0.003	<0.003	<0.003	<0.003
Cr	0.002	0.002	0.002	0.002
Cu	0.002	0.002	0.002	0.002
Fe	0.2	0.25	0.3	0.25
Ga	0.03	0.03	0.03	0.03
Mg	0.005	0.005	0.005	0.005
Mn	0.002	0.002	0.002	0.002
Ni	0.001	0.002	0.001	0.002
Pb	<0.004	<0.004	<0.004	<0.004
Si	0.07	0.07	0.08	0.08
Sn	<0.005	<0.005	<0.005	<0.005
Ti	<0.01	<0.01	<0.01	<0.01
Zn	0.004	0.01	0.09	0.004

TABLE 14

CHEMICAL ANALYSIS OF IRRADIATED SAP CLADDING

Fuel Rod	Postirradiation		Pre-irradiation*	
	Carbon (wt %)	Hydrogen (wt %)	Carbon (wt %)	Hydrogen (wt %)
B at 3 in.	0.61	0.031	0.59 and 0.68	0.0017 and 0.0030
B at 10 in.	0.66	0.015		
J at 8 in.	0.71	0.022		
H at 8 in.	0.46	0.010	0.59 and 0.68	0.0017 and 0.0030

*Each pre-irradiation number is an average of three samples and each represents a tubing lot from which cladding tubes were cut.

V. SUMMARY AND DISCUSSION

Visual examination of assembled bundle LTA and the individual fuel rods revealed no defect or damage. Film formation was heavier on the outer rods (B-P), averaging 1.2 mils, than the inner rods (R-Y), which averaged 0.5 mil. The average fission gas release for outer rods was $2.09 \times 10^{-2} \%$, while the average for inner rods was $0.83 \times 10^{-2} \%$. The higher release for the outer rods reflected the higher burnup, which was shown by the gamma scans. The gamma scans indicated inner rods ran at about 65% of the power level of the outer rods. The gamma scans also indicated burnup end peaking in all rods. These observations are close to those predicted as part of the design analysis.⁽⁴⁾

The clad fuel displayed limited cracking, but no gross damage. The summary of fuel burnup and growth, presented in Table 15, indicates values close to those expected.⁽⁴⁾ Fuel metallography revealed no severe structural damage.

TABLE 15
SUMMARY OF FUEL GROWTH AND BURNUP FOR U-305
BUNDLE LTA RODS B, H, J, and P

Fuel Rod	Carbon Content (wt %)	Growth ($\Delta\rho/\rho_f$) (%)	Burnup (Mwd/MTU)
B	4.85 to 4.90	1.75	8866
H	4.94 to 4.98	1.78	8799
J	5.04 to 5.08	1.79	8478
P	5.19 to 5.22	1.87	8258

The microstructures exhibited by Rod P (5.19 to 5.22 wt % C) appeared identical to the unirradiated condition. Rods B, H, and J (4.88 to 5.04 wt % C) showed a carbon gradient, with center and mid-radius portions appearing higher in carbon content than portions at the periphery. Attempts to confirm the presence of this carbon gradient by chemical analysis yielded inconclusive results. Termination of the HWO CR Program precluded additional effort to study this phenomenon.

Visual examination of the interior surfaces of the SAP cladding revealed deposits formed at the fuel/cladding interfaces. Subsequent metallographic

examination indicated formation of an interaction product in a discontinuous manner over 30 to 45% of the inner circumference of the cladding, with an average thickness of 0.8 mil. The reaction product appeared to be an inter-metallic compound high in aluminum, but was not identified due to time limitations imposed by termination of the HWOCR Program. It is significant that the reaction product formed on the cladding surface growing into the fuel, not into the cladding, and the thickness rarely exceeded one mil. It is doubtful, therefore, that the deposit formed on these samples would have a detrimental effect on the SAP cladding.

VI. CONCLUSIONS

The significant conclusions of the post-irradiation examination are listed below.

- 1) No evidence of defects or radiation induced degradation was noted in the SAP-UC portion of the U-305 Experiment.
- 2) The SAP-UC interaction product formed on the cladding surface and the apparent carbon gradient observed in the fuel appeared to have no deleterious effect on the fuel rods.
- 3) Satisfactory operation of the SAP-UC fuel system was demonstrated at the burnup and temperature levels experienced by the U-305 Experiment.

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