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IRRADIATION EFFECTS ON UC_2 DISPERSED IN GRAPHITE

(ORNL-MTR-48-1), INTERIM REPORT NO. 1

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ABSTRACT

A dispersion of enriched (93 wt % U^{235}) UC_2 in graphite was irradiated in the MTR as a part of the advanced fuels testing program. The fuel was irradiated to a burnup of 7560 Mwd/T of fueled matrix, with average fission power densities of 270 and 300 w/cm³.

Preliminary examinations in the hot cell showed no external dimensional changes to the graphite cans enclosing the fuel cylinders. One fuel cylinder showed a 2.4 to 5.1% decrease in diameter and a 3.5% increase in length.

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INTRODUCTION

A dispersion of enriched UC₂ in a graphite matrix is included among the materials being tested under the advanced materials program of the Gas-Cooled Reactor Project. Two small cylinders of this fuel material were sealed in graphite containers and irradiated in the MTR.¹ Following irradiation, the experiment was returned to ORNL for postirradiation examination in the Solid State Division hot cells.

The hot-cell operations were conducted by H. E. Robertson, J. W. Gooch, and J. E. Van Cleve. Photography was done by J. C. Gower and R. M. Wallace.

DESCRIPTION OF EXPERIMENT

The fueled graphite samples and the graphite containers were fabricated by the National Carbon Company Research Laboratories.² Spherical UO₂ shot, of 93 wt % U²³⁵ enrichment, were added to a carbonaceous mix, and cylinders were molded, pressed, and baked. The UO₂ was converted to UC₂ by final baking at 2800°C, and the final particle size was estimated to range from 160 to 260 μ . Autoradiographs of the preirradiated cylinders indicated uniform distribution of the hollow spheres of UC₂ through the graphite (Fig. 1). The autoradiographs were obtained by using contrast process ortho film in direct contact with the end of fuel cylinder. An 18-hr exposure was used.

Type R-0020 graphite, with a permeability of about 10^{-4} cm²/sec (helium at room temperature), was used for the fuel containers. The threaded graphite end plugs were sealed into the cans with Si-SiC.

¹Irradiation performed by Reactor Projects Division personnel: D. B. Trauger, Irradiation Engineering Department; C. W. Cunningham, Program Engineer.

²Memo from M. Janes, National Carbon Research Laboratory to W. D. Manly, ORNL, February 10, 1960.

Fig. 1. Autoradiograph of UC_2 Dispersed in Graphite. 4X.

The two graphite cans, separated by a carbon spacer, were installed in a stainless steel containment vessel. Figure 2 is a schematic of the assembly. Six thermocouples, located in the walls of the graphite cans at appropriate locations around the fuel, monitored the temperatures. A purified helium atmosphere protected the graphite from oxidation during irradiation.

IRRADIATION HISTORY

The experiment was installed in the MTR on November 2, 1959, and irradiated for 1440 hr prior to withdrawal on January 25, 1960. Test conditions estimated from operating data appear in Table 1.

Table 1. ORNL-MTR-48-1 Test Conditions

Conditions	Element A	Element B
Avg temperature, graphite can wall, °F	1570	1310
Avg thermal neutron flux	4.6×10^{13}	1.9×10^{13}
Avg fission power density, w/cm ³	270	300
Burnup, % U ²³⁵ atoms	12	5.6
Central fuel temperature, °F	3355	3616
Weight % UC ₂	10.52	19.33

POSTIRRADIATION EXAMINATION

The gas tube protruding from the steel containment vessel was pierced, and the mixture of helium and fission gases was collected in an evacuated system. The analyses are not yet complete. The relative gamma radioactivity was monitored by a shielded ion chamber and indicated the flux distribution along the apparatus. Figure 3 shows the gamma activity scan of ORNL-MTR-48-1.

The layout of the cell equipment is shown in Fig. 4. A circumferential cut has just been made on the outer metal tubes, freeing the end of the assembly containing the fuel. The thermocouples can be seen protruding from the outer can. The ribbed-aluminum end section containing the fuel can be

① POROUS CARBON-60	⑦ HELIUM GAS GAP, 0.010 in.
② R-0020-TYPE GRAPHITE	⑧ INNER CONTAINMENT VESSEL, TYPE 304 STAINLESS STEEL
③ U^{235} -FUELED GRAPHITE MATRIX	⑨ OUTER CONTAINMENT VESSEL, 6061 ALUMINUM (WITH LONGITUDINAL GROOVES)
④ GRAPHITE THREADS SEALED WITH Si-SiC	⑩ 1/8-in. WATER FLOW ANNULUS
⑤ TYPE 316 STAINLESS STEEL-SHEATHED MgO INSULATED CHROMEL-ALUMEL THERMOCOUPLES (THREE 120° APART PER MATRIX)	⑪ WATER FLOW DIVIDER, 6061 ALUMINUM
⑥ FLUX MONITOR WIRES 120° APART ALONG FULL LENGTH OF BOTH MATRICES	⑫ OUTER WATER JACKET, TYPE 304L STAINLESS STEEL

Fig. 2. Experimental Assembly for Irradiation of Graphite-Clad Fuel Elements.

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Fig. 3. Gamma Radioactivity Scan of Experiment MTR-48-1.

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Fig. 4. Hot-Cell Disassembly Equipment.

seen in front of the meter stick. This end section was then scanned for radioactivity by using the apparatus to the right rear of the picture. The two graphite fuel cans were removed, and visual inspection revealed no apparent change in the cans or carbon spacers (see Fig. 5). The ends of the graphite can, element B, were removed and the fuel cylinder examined. Figure 6 shows the appearance of the irradiated fuel cylinder compared with one unirradiated. Micrometer measurements showed a decrease in diameter and an increase in length (Table 2). The surface of the fuel cylinder was quite friable, making more precise measurements difficult. The outside dimensions of the encapsulating graphite cans remained unchanged (Table 3).

Table 2. MTR-48-1 Dimensional Data for the Fueled Graphite Cylinder, Element B

	Preirradiation	Postirradiation	Change (%)
Avg diameter, in.			
End toward reactor	0.9966	0.973	-2.4
End away from reactor	0.9966	0.946	-5.1
Avg length, in.	1.4682	1.521	+3.5

Table 3. MTR-48-1 Dimensional Data for the Graphite Fuel Containers

	Element A (Identifying No. 3)		Element B (Identifying No. 4)	
	Preirradiation	Postirradiation	Preirradiation	Postirradiation
Avg length, capsule body, in.	2.5055	2.506	2.5006	2.501
Avg diameter, capsule body, in.	1.8753	1.875	1.8752	1.875
Avg diameter, end projections, in.				
Plug end		0.7505	0.7503	0.7510
Solid end		0.7501	0.7503	0.7507
Total length, in.	3.2551	3.253	3.2543	3.254

Fig. 5. Irradiated Fuel Assembly, Exploded View.

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IRRADIATED TO 6 wt %
URANIUM BURNUP

Fig. 6. Appearance of Irradiated and Unirradiated Fuel Cylinders.

Samples of the graphite can, in the form of sawdust, were submitted for qualitative radiochemical analysis. The results are shown in Table 4.

Table 4. Radionuclides Detected in Graphite Dust from MTR-48-1

Nuclide	Ratio to Ce^{144}	
	Observed	Normal Ratio in Fissioning U^{235}
Ce^{144}	1	1
Ce^{141}	0.6	0.6
Zr^{95}	0.08	1.6
Ba^{140}	0.0015	0.02
Ru^{103}	< 0.02	0.4
Cs^{137}	< 0.02	0.04

Figures 7 and 8 show the end views of the two irradiated graphite cans and the stainless-steel-sheathed thermocouples which were in contact with the graphite. The Si-SiC sealing material can be seen around the end plug. The holes accommodated the thermocouple and flux monitor wires.

FURTHER EXAMINATION

The other graphite-canned element (A) will also be opened and the fuel cylinder examined. In addition, radial samples will be core-drilled from the fuel to determine the extent of fuel migration. Metallographic examinations will be made as well as density determinations. The Si-SiC seal will be sectioned and examined.

SUMMARY

Postirradiation examinations were made on a fuel cylinder of UC_2 dispersed in graphite. The cylinders were canned in low-permeability graphite. The fuel was irradiated at a calculated center temperature of $3600^\circ F$ to a burnup of 7560 Mwd/T of fueled matrix. The preliminary examination showed no outside dimensional changes on the graphite can. One fuel cylinder showed a 2.4 to 5.1% decrease in diameter and a 3.5% increase in length.

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ELEMENT A

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ELEMENT B

Fig. 7. End Views of Irradiated Graphite Cans, Showing Si-SiC Plug Seal.

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Fig. 8. Sheathed Thermocouple, Irradiated.

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