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IRRADIATION OF FUELED GRAPHITE FOR ADVANCED
HIGH-TEMPERATURE GAS-COOLED REACTORS

J.A. Conlin C.W. Cunningham

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IRRADIATION OF FUELED GRAPHITE FOR ADVANCED HIGH-TEMPERATURE GAS-COOLED REACTORS

J. A. Conlin C. W. Cunningham

Abstract

A series of six capsule-irradiation tests was made on cylindrical matrix-type fuel elements containing uncoated UC_2 particles or pyrolytic-carbon-coated UC_2 or $(U-Th)C_2$ particles. Those that contained uncoated fuel were encapsulated in low-permeability graphite as a barrier to fission-gas release. Fuel temperatures ranged from 2000 to 3500°F; power densities ranged from 10 to 300 w/cc; and burnups were up to 21 at. % ^{235}U . One capsule contained loose pyrolytic-carbon-coated fuel particles in sealed tubes, in addition to a matrix element.

Fission-gas release was determined during irradiation by sampling the helium purge gas for analysis by gamma-ray spectrometry. Information on the overall stability of the fuels was determined by postirradiation examination. This report covers the design of the experimental equipment, operation, and fission-gas release data. The results of postirradiation examinations have been reported elsewhere.¹

Introduction

High-temperature gas-cooled reactors are of interest in the development of advanced converter power reactors. In their present concept, these reactors depend on ceramic coating of individual fuel particles for fission-product retention. The OEEC Dragon, Peach Bottom, and Public Service of Colorado reactors are of this type.

Several approaches were made to the problem of fission-product retention in the early development of fuels for this type of reactor. Among these there was, in addition to the coated-particle approach, a concept in which the fuel was encapsulated in a very low-permeability graphite container. The fuel of this latter concept was a dispersion of uncoated fuel particles in a graphite matrix. After fabrication, the fueled matrix was sealed in a low-permeability graphite container for fission-product retention.

This report describes an early series of tests in which variations of these two concepts of fission-product retention by low-permeability graphite encapsulation and pyrolytic-carbon coating of individual fuel particles were tested. The fuel was in the form of UC_2 and $(U-Th)C_2$ particles. This information can add little to the present state of the art but is presented to provide historical background for later tests and to document the work. The results of postirradiation examinations of the assemblies are presented in Ref. 1.

Description of Experimental Fuel Assemblies and Irradiation Equipment

Fuel Assembly

The ORNL-MTR-48 experimental assemblies were irradiated in horizontal beam hole HB-3 of the MTR. A typical assembly containing two fuel elements encapsulated in graphite containers is shown in Figs. 1 and 2. The fueled-graphite cylinders were nominally 1 in. in outside diameter and up to 1 1/2 in. in length. They contained a dispersion of uncoated or pyrolytic-carbon-coated UC_2 or $(U-Th)C_2$ particles in a graphite matrix. The ^{235}U loading for an individual fuel cylinder was determined by the desired power density and the neutron flux at the intended position in the beam hole. In the positions the fuel elements occupied, the perturbed 2200-m/sec thermal flux varied from 1.5 to 5.3×10^{13} neutrons/cm²·sec and the gamma heat from 1.0 to 2.3 w/g.

The graphite containers for the fuel cylinders were 1 15/16 in. OD and 1 in. ID and were made in lengths to suit the particular experiment. The container walls were drilled longitudinally to receive up to eight 1/16-in.-diam stainless-steel-sheathed thermocouples. With the exception of the first assembly, fuel containers or cans of low-permeability graphite were employed for fuel particles that were not coated with pyrolytic carbon. The containers were closed with plugs and sealed by silicon brazing. Unsealed permeable graphite containers were used for the first assembly and for later tests with coated-particle fuels.

Fig. 1. ORNL-MTR-48 Capsule and Plug Assembly for Fuel Element Irradiation Tests.

ORNL-LR-DWG 41751R

Fig. 2. Section Through ORNL-MTR-48-1 Plug and Capsule.

The elements were supported at both ends by porous carbon or graphite pieces that served as thermal insulators to reduce end losses and position the elements radially. There was a uniform annular gas gap between the fuel elements and the water-cooled capsule wall. This gap was sized to the requirements of each experiment to provide the thermal resistance necessary to obtain the desired operating temperature.

Capsule Assembly

The fuel elements were sealed in a water-cooled double-walled aluminum and stainless steel capsule 2 3/8 in. in outside diameter and 24 in. long (see Fig. 1). The fuel element was located in the end nearest the reactor core; the remaining volume behind the fuel was provided to dilute released fission gas as a means of minimizing thermal conductivity changes in the gas gap in experiments such as the first in which no purge gas was supplied. In the region of the fuel assembly, the outer aluminum wall was longitudinally grooved to increase the heat transfer surface and was shrink fitted onto the inner stainless steel capsule wall.

Stainless steel thermocouples and purge-gas leads from the fuel assembly were silver soldered into bulkheads where the leads penetrated the capsule walls. The seal between the aluminum wall and the leads was made with a transition piece consisting of aluminum and copper tubing flash-welded together. The aluminum end of the transition piece was welded to the aluminum capsule bulkhead. The leads passed through the transition tube and were brazed to a stainless steel fitting, which was, in turn, brazed to the copper end of the transition tube.

The leads passed to the rear of the assembly through a conduit that fitted inside a stainless steel capsule support tube. The support tube formed a part of the coolant water passages and positioned the capsule within a special beam-hole plug. An integral lead shield at the rear of the support tube formed a part of the radiation shielding required during assembly installation when the beam-hole plug assembly was not filled with water.

The fuel element capsule, together with the support tube and lead shield, formed the assembly that was replaced with each experiment.

MTR Beam Hole Plug Shield Assembly

The one plug assembly (see Fig. 1), which formed the beam-hole radiation shield section and the coolant water passages, and into which the capsule assembly was inserted, was used for all experiments. The water passages were contained within a stainless steel tube that was 5 1/2 in. in diameter, except for a 3 1/2-in.-diam section in the region of the capsule proper. An internal aluminum baffle directed the coolant water (25 gpm) over the capsule.

The 6 3/4-in.-OD, 2 1/2-in.-ID shield was formed of steel and steel-shot-filled concrete. Wheels were built into the bottom to facilitate insertion and adjustment of position in the beam hole. The external wall of the rear of the plug extended beyond the shield to form a recess that was utilized to house a water flow distribution gland and the capsule assembly lead shield. O-ring gaskets formed the water seal between the plug and capsule assembly. The shielded section of the plug was salvaged from an unused test plug that had been built for another experiment and hence contained some unused features that are shown on Fig. 1.

All service and instrument connections were made at the rear of the plug. The water connections were self-sealing quick-disconnect couplings. Purge line connections were silver brazed. Standard thermocouple connectors joined the thermocouple leads to those of the instrument panel. A remotely operated retractor mechanism was provided in the beam-hole cubicle for changing the position of the plug in the beam hole to vary the neutron flux to which the fuel was exposed. A 7-in. travel provided a factor of 2.5 in flux at the fuel between the fully inserted and the fully retracted positions.

Purge-Gas System

A 200-cm³/hr helium purge was employed in all but the first experiment to sweep fission gases evolved from the fuel past radiation monitors and through a sampling station. Helium with less than 10 ppm by volume of CO₂ and water was passed through a molecular sieve as an added protection against moisture. Before entering the assembly, the helium was also passed through a charcoal trap and capillary tubing within the cubicle

as protection against back diffusion of fission gas at the low flows employed.

The purge gas entered the capsule at the rear support-tube end and passed through a capillary tube in an axial hole in the wall of the graphite fuel container to the forward end of the capsule. The purge gas then flowed back over the graphite fuel containers to the rear of the capsule, where it left by a tube leading to a sample station and activity monitors. After being monitored, the effluent passed through a charcoal trap and a second activity monitor prior to release to the MTR off-gas system.

The rate of fission-gas release to the purge-gas stream from the fuel was determined by gamma-ray spectrographic analysis of gas samples obtained at the sampling station. The gross gamma activity of the purge outlet line at the sample station was continuously recorded by both a Geiger-Mueller tube and an ionization chamber.

The gas-sampling station, Fig. 3, consisted of a shielded and sealed box vented to the off-gas system in which the necessary valves and connections for attaching the metal gas sample bulbs were provided. All valve handles passed through seals in the wall of the box, and the cover was opened only for the installation and removal of sample bulbs. A gas sample was obtained by diverting the gas stream through the sample bulb and isolating the sample by valving. The bulb was removed by clamping the rubber tubing that connected the bulb to the system with pinch clamps and then cutting the rubber tubing. The sample bulb inlet and outlet copper tubes were then removed, and a cap was soft-soldered to the open ends to provide an absolute seal. The elapsed time between passing of the purge gas over the fuel and counting the sample on the gamma-ray spectrometer was approximately 3 hr.

Instrumentation and Controls

The fuel assembly temperature, helium purge flow, and water flow were regulated manually. Alarm annunciation was provided to summon an MTR operator when adjustments became necessary. A reactor "Jr. Scram" action was provided for protection against loss of coolant-water flow, excessive fuel assembly temperature, high purge-gas activity to the MTR stack, and

Fig. 3. Gas-Sampling Station for ORNL-MTR-48 Experiments. (Photograph obtained from NRTS)

loss of instrument air or power. A general listing of the instrumentation and alarm or control actions is given in Table 1.

Temperatures were measured in the graphite bottle or sleeve walls with as many as eight 1 1/16-in.-OD type 316 stainless steel-sheathed, Cromel-P vs Alumel thermocouples. For central fuel temperature measurements, 1/16-in.-diam tantalum-sheathed MgO-insulated Pt-10% Rh vs Pt or BeO-insulated tungsten vs rhenium thermocouples with insulated junctions were used.

Table 1. Instrumentation and Controls^a

Parameter	Sensing Element	Control Action
Temperature		
Fuel element	Thermocouple	Alarm, Jr. scram
Water supply	Thermocouple	None
Water temperature rise	Electrical resistance bulb	None
Fission-gas absorption trap	Thermocouple	Alarm
Helium flow	Pressure drop across capillary tubing	Alarm
Helium pressure		
Supply manifold	Pressure gage and pressure switch	Alarm
Capsule	Pressure transmitter and pressure switch	Alarm
Purge-gas activity		
From capsule	Geiger-Mueller tubes (2)	Alarm
	Ion chamber (1)	Alarm
From charcoal trap (to reactor off-gas system)	Ion chamber	Alarm, Jr. scram
Airborne activity	Geiger-Mueller tubes (5)	Alarm
Water flow	Orifice (upstream and downstream)	Alarm, Jr. scram
Water pressure	Pressure gage and switch	Alarm
Instrument-air pressure	Pressure gage and switch	Alarm, Jr. scram
Loss of instrument power		Jr. scram

^aAll controls to reactor safety circuits were double tracked.

Preparation for Postirradiation Transportation

In the interval between the removal of an experimental assembly from the reactor until delivery to a hot cell for postirradiation examination, the fuel assembly was maintained in a dry atmosphere to protect the carbide fuel particles from moisture. Before removal of the experimental assembly from the reactor beam hole, the purge system was pressurized to about 0.3-psig helium and the soft-copper purge-gas lines from the capsule assembly were sealed with metal-tubing pinch clamps. The assembly was then transported to a plug storage area, where the capsule assembly was removed from the plug. The capsule was cut from the support tube in the MTR hot cell. With the support tube removed the purge gas lines leading to the fuel element were open, and therefore the severed fuel section was sealed in a vessel containing a desiccant to minimize the amount of moisture reaching the fuel. The vessel was then loaded into a shielded carrier for shipment to the examining facility.

Experiment ORNL-MTR-48-1

The first experiment of the series was the irradiation in helium of two unpurged UC_2 -graphite fuel cylinders with a design central temperature of 3500°F and fission power densities of 260 to 285 w/cm^3 . The test was designed to provide data on migration of UC_2 and fission products, dimensional changes, and general stability of the fuel matrix. The fuel assembly, which was supplied by the National Carbon Company, is shown in Figs. 2 and 4, along with an adjacent section of the capsule vessel.²

The fuel cylinders were made by molding a mixture of 100- to 150- μ UO_2 spherical particles and NCC type TS-160 carbonaceous mix. Molding pressure was applied parallel to the cylinder axis to give a radial grain direction. The UO_2 was converted to UC_2 by baking the compact at 2800°C . The UC_2 particle size is estimated to have been 160 to 260 μ .

One 1-in.-OD, 1 1/2-in.-long fuel cylinder was sealed into each of two NCC type R-0020 grade graphite containers. The container seal was made with a screwed graphite plug that was brazed in place with silicon.

Fig. 4. Fuel Test Elements and Containment Vessel for ORNL-MTR-48-1 Experiment.

Preirradiation data are presented in Table 2 for the fuel cylinders and in Table 3 for the graphite containers and supports. The fuel element designations "A" and "B" refer to the position relative to the reactor core, with "A" being the position nearest the reactor. Fuel loading of the cylinders was varied inversely with the anticipated thermal flux at their respective positions in the experiment.

Table 2. Preirradiation Data for ORNL-MTR-48-1
Fuel Cylinders

Fuel cylinder position ^a	A	B
Fuel cylinder identification	NCC 85	NCC 91
Dimensions, in.		
Outside diameter	0.9964	0.9966
Length	1.4688	1.4682
Weight, g		
Total	35.4107	37.8407
UC ₂	3.725	7.315
²³⁵ U	3.148	6.182
Isotopic assay, at. %		
²³⁵ U	93.12	93.12
²³⁸ U	5.39	5.39
²³⁴ U	1.08	1.08
²³⁶ U	0.41	0.41
Bulk density, g/cm ³	1.887	2.017
Fuel loading, g/cm ³		
UC ₂	0.1985	0.3899
²³⁵ U	0.1677	0.3295
Specific electrical resistance, μohm-cm	2185	1460
Thermal conductivity estimated at room temperature, Btu/hr·ft·°F	45	60
Flexural strength estimated at room temperature, psi		
With grain	4200	3000
Against grain	5800	4500
Final baking temperature, °C	2800	2800

^aPosition refers to location of the fuel cylinder in the capsule, with A being nearest the reactor core.

Table 3. Preirradiation Data for ORNL-MTR-48-1 Graphite
Fuel Container and Thermal Insulating Spacers

Fuel element position ^a	A	B
Identification	NCC 3	NCC 4
Encapsulating graphite fuel container		
Material	NCC type R-0020	NCC type R-0020
Dimensions of bottle, in.		
Outside diameter	1.8753	1.8752
Inside diameter	1.000	1.000
Length at 1.87-in. diameter	2.5045-2.5070	2.5000-2.5013
Inside length	1.5	1.5
Overall length	3.2551	3.2543
Bulk density, g/cm ³	1.92	1.92
Weight sealed (including fuel), g	221.7196	222.8360
Silicon in braze seal, g	0.232	0.430
Permeability coefficient of body, before assembly, for helium at 25°C, cm ² /sec	10 ⁻⁴	10 ⁻⁴
Electrical resistivity (typical), μohm-cm		
With grain	1490	1490
Against grain	1640	1640
Thermal conductivity (typical), Btu/hr·ft·°F		
With grain	53	53
Against grain	51	51
Flexural strength (typical), psi		
With grain	4100	4100
Against grain	3900	3900
Compressive strength (typical), psi		
With grain	9100	9100
Against grain	9000	9000
Thermal insulating spacer material	NCC Carbocell graphite grade 60	NCC Carbocell graphite grade 60

^aPosition refers to location of the fuel cylinder in the capsule, with A being nearest the reactor core.

Three temperature measurements were made in the graphite container walls 120° apart at points radially opposite the center of each fuel cylinder. Flux monitor wires of type 302 stainless steel, Inconel, and tungsten were similarly located in the graphite wall parallel to the fuel-cylinder axis.

The irradiation test period was four MTR cycles (12 weeks) between November 2, 1959 and January 25, 1960. Estimates of fission-power density, thermal-neutron flux, and ^{235}U burnup calculated from measured temperatures and geometry are presented in Table 4. The detailed temperature history, including time at temperature and thermal cycles, is presented in Appendix A, Table A-1. The maximum temperature spread for the thermocouples opposite the A and B elements was 100 and 140°F, respectively. One A element thermocouple was lost during the last reactor cycle. The B element temperature was lower than that of the A element, with the thermal flux gradient being steeper than had been estimated.

Postirradiation examination data are reported in Refs. 1, 3, and 4.

Table 4. Operating Conditions^a for Experiment ORNL-MTR-48-1

Fuel cylinder	A	B
Central temperature, ^b °F		
For 1360 hr	3400-3600	2900-3300
For 36 hr	2200-2400	1700-1900
Power generation (for 1360.5-hr period)		
Total fission power, kw	5.3	4.8
Fission power		
Matrix, w/cm ³	285	260
UC ₂ , w/g	1430	660
Thermal-neutron flux (for 1360 hr), neutrons/cm ² ·sec	4.9×10^{13}	2.2×10^{13}
Total burnup (for 1396 hr), at. % ^{235}U	11.8	5.4

^aValues were estimated from geometry and measured temperatures, see Appendix A.

^bThe range in central temperatures corresponds to the variation in the measured temperatures.

Experiment ORNL-MTR-48-2

The second ORNL-MTR-48 experiment was designed to provide data comparable with those of the first test but taken at lower fuel temperature, 2500°F, and power density, 140 w/cm³ of fuel. In addition, the fuel was encapsulated in a low-permeability graphite container, and the assembly incorporated helium purge and gas-sampling systems to permit fission-gas-release measurements.

The fuel consisted of two 1-in.-diam fuel cylinders. The A cylinder, which was positioned closest to the reactor, was 1 1/4 in. long, and the B cylinder, to the rear, was 1/4 in. long. The unequal lengths resulted from the use of available fuel matrices. The fuel matrices were sealed in one encapsulating graphite container with an unthreaded silicon-brazed plug. An x-ray picture of the element is shown in Fig. 5, in which the two fuel cylinders appear as one. The silicon-brazed seal and the access hole in the graphite wall for flux monitor wires, thermocouples, and the purge inlet tube are visible.

Preirradiation data are presented for the fuel cylinders in Table 5 and for the encapsulating graphite fuel container and insulating spacers in Table 6. The container graphite, NCC type R-0020, was treated prior to the silicon braze to reduce its helium permeability at 25°C to 1.5×10^{-8} cm²/sec, as measured by National Carbon Company.

The irradiation test period was eight MTR cycles (24 weeks) between June 20, 1960 and December 5, 1960. The fission power for the fuel is estimated to have been 2.75 kw at design conditions, with the graphite wall temperature at 1500°F. Estimates of fission-power density, thermal-neutron flux, and ²³⁵U burnup are presented in Table 7. The measured graphite wall temperature varied from 1420 to 1590°F and averaged 1500°F (2400 to 2600°F calculated central temperature) during 2466 hr of irradiation. The graphite wall temperature was dropped to 1040 to 1070°F for the last 302 hr to reduce the sampling-station radiation level, which had risen sharply. Temperature variations of any one thermocouple from the average of the three thermocouples did not exceed 60°F. The detailed temperature history, including time at temperature and thermal cycles, is presented in the Appendix, Table A-2.

Fig. 5. X-Ray Photograph of ORNL-MTR-48-2 Fuel Element. (Photograph obtained from Union Carbide Corporation Parma Research Center)

Table 5. Preirradiation Data for ORNL-MTR-48-2
Fuel Cylinders

Fuel cylinder position ^a	A	B
Fuel cylinder identification	NCC 910	NCC 912
Dimensions, in.		
Outside diameter	0.9963	0.9954
Length	1.235	0.2333
Weight, g		
Total	28.4323	5.4272
UC ₂	1.589	0.434
²³⁵ U	1.342	0.367
Enrichment, ^b at. % ²³⁵ U	93.12	93.12
Bulk density, g/cm ³	1.802	1.825
Fuel loading, g/cm ³		
UC ₂	0.1007	0.1460
²³⁵ U	0.0850	0.1234

^aPosition refers to location of the fuel cylinder in the capsule, with A being nearest the reactor core.

^bIsotopic assay was the same as for ORNL-MTR-48-1 fuel.

Table 6. Preirradiation Data for ORNL-MTR-48-2 Graphite
Fuel Container and Thermal Insulating Spacers

Graphite fuel container	
Identification	NCC 5
Material	NCC type R-0020
Outside diameter, in.	1.8750-1.8757
Inside diameter, in.	1.000
Length at 1.87-in. diameter, in.	2.5088-2.5100
Internal length, in.	1.5
Overall length, in.	3.2590
Weight sealed (including fuel), g	223.37
Silicon in braze seal (before facing), g	0.13
Permeability coefficient of body, before assembly, for helium at 25°C, cm ² /sec	1.5 × 10 ⁻⁸
Thermal insulating spacer	
Material	NCC Carbocell carbon grade 60
Thermal conductivity, Btu/hr·ft ² ·°F	1.9

Table 7. Operating Conditions^a for
Experiment ORNL-MTR-48-2

Central temperature, ^b °F	
For 2464 hr	2400–2600
For 29 hr	2100–2300
For 303 hr	1700–1800
Power generation for 2464 hr	
Total fission power, kw	2.75
Fission power	
Matrix, w/cm ³	145
UC ₂ , w/g	1460
Power generation for 303 hr	
Total fission power, kw	1.85
Fission power	
Matrix, w/cm ³	100
UC ₂ , w/g	990
Thermal-neutron flux, neutrons/cm ² .sec	
For 2464 hr	4.6×10^{13}
For 303 hr	3.1×10^{13}
Total burnup for 2797 hr, at. % ²³⁵ U	
A fuel cylinder	23
B fuel cylinder	16

^aValues are estimated from geometry and measured temperatures.

^bThe range in central temperature corresponds to the variation in measured temperatures.

During the test an unexplained intermittent purge flow restriction appeared that caused variations in the helium purge pressure and flow with intermittent flow stoppage. Steady flow rates during the gas-sampling periods were not always attainable.

The isotopes ^{85m}Kr, ⁸⁸Kr, ¹³³Xe, and ¹³⁵Xe were identified in the gas samples. The concentrations were corrected to the time the helium passed the fuel element and are expressed as escape rates (see Fig. 6 and Appendix Table A-3). The fission-gas birth rates were computed from the fission chain yields and the fission power, which was determined from the measured graphite wall temperature. The ratio of escape rate to birth

Fig. 6. Fission-Gas Release from Fuel Cyclinders of Uncoated UC_2 Particles in Experiment ORNL-MTR-48-2 as a Function of Fuel Temperature and ^{235}U Burnup.

rate appeared to level off after 30 to 60 days until near the end of the experiment, when the activity rose and made it necessary to reduce operating conditions and impossible to take further gas samples.

Postirradiation examination data are reported in Refs. 1 and 4 through 8.

Experiment ORNL-MTR-48-3

The ORNL-MTR-48-3 experiment was designed to continue the study of fission-product retention by low-permeability graphite and to obtain further information on the overall stability of graphite-matrix-type fuels. The element supplied by the National Carbon Company, Figs. 7 and 8, contained four fueled cylinders in a single low-permeability graphite container. Only the parting line between the two central cylinders is visible in the radiograph of Fig. 7. The element was designed to operate at a peak fuel temperature of 3500°F and fission power density of 280 w/cm³, with a total fission heat generation of 18 kw. The corresponding mean temperature of the encapsulating graphite wall was 1500°F. Fission-gas release was monitored with the helium purge system.

The four fuel cylinders, 1 in. OD and 1 1/4 in. long, Fig. 9, were sealed in the low-permeability graphite container with a silicon-brazed threaded plug. Axial holes in the graphite wall contained one thermocouple opposite the center of each fuel cylinder and three flux monitors, one of which doubled as the helium purge-gas inlet tube. The helium permeability for the encapsulating graphite with the thermocouple and flux monitor holes present was 5.9×10^{-9} cm²/sec at 25°C, as determined by the National Carbon Company. Preirradiation data are presented for the fuel cylinders in Table 8 and for the encapsulating graphite and spacer in Table 9. Predictions of the fission-gas release were made by the National Carbon Company.⁹

The experiment was terminated after one MTR cycle, December 26, 1960 to January 16, 1961, as a consequence of excessive activity in the purge-gas effluent. The probable cause was a leak that was subsequently located during postirradiation examination in the silicon-brazed seal of the plug in the encapsulating graphite. To limit the purge-gas activity, the

Fig. 7. X-Ray Photograph of ORNL-MTR-48-3 Fuel Element. (Photograph obtained from Union Carbide Corporation Parma Research Center)

Fig. 8. Plug End of Graphite Fuel Element Container Used in Experiment ORNL-MTR-48-3. (Photograph obtained from Union Carbide Corporation Parma Research Center)

Fig. 9. Fuel Cylinder Used in Experiment ORNL-MTR-48-3. (Photograph obtained from Union Carbide Corporation Parma Research Center)

Table 8. Preirradiation Data for ORNL-MTR-48-3 Fuel Cylinders

Fuel cylinder position ^a	A	B	C	D
Fuel cylinder identification	NCC 999	NCC 1000	NCC 1001	NCC 1002
Dimensions, in.				
Outside diameter	0.9958	0.9968	0.9964	0.9963
Length	1.2347	1.2350	1.2352	1.2345
Weight, g				
Total	29.6524	30.6727	31.7620	33.6101
UC ₂	3.232	4.717	6.708	9.481
²³⁵ U	2.734	3.990	5.675	8.022
Enrichment, ^b at. % ²³⁵ U	93.12	93.12	93.12	93.12
Bulk density, g/cm ³	1.882	1.924	2.012	2.131
Fuel loading, g/cm ³				
UC ₂	0.205	0.299	0.425	0.601
²³⁵ U	0.174	0.253	0.360	0.509
Specific axial electrical resistance, μohm-cm	2490	2450	2010	1910
Final baking temperature, °C	2350	2350	2350	2350

^aPosition refers to location in the capsule, with A being at end nearest the reactor core and D at the opposite end.

^bIsotopic assay is the same as for ORNL-MTR-48-1.

experiment was operated in the fully retracted position for the 249-hr duration of the test. For 1 hr, 32 hr after startup, the assembly was inserted until design temperatures were reached to obtain data on the perturbed flux versus element position.

Gas samples were not taken during the test because of the high radiation field at the sampling station. From decay curves obtained from purge-gas-line radiation monitors looking at a stagnant gas sample, the major source of the activity was identified as 2.8-hr ⁸⁸Kr. A single gas sample obtained on the last day of the test and counted after a 14-hr decay period was found to contain ^{85m}Kr, ⁸⁸Kr, ¹³³Xe, and ¹³⁵Xe. The ratios of release rate to birth rate, R/B, were computed to be 5.2×10^{-3} , 0.07×10^{-3} , 9.8×10^{-3} , and 0.27×10^{-3} , respectively.

Table 9. Preirradiation Data for ORNL-MTR-48-3 Graphite Container and Thermal Insulating Spacers

Graphite fuel container	
Identification	NCC 2
Material	NCC type R-0020
Dimensions, in.	
Outside diameter	
Open end	1.8813-1.8815
Middle	1.8820-1.8822
Solid end	1.8824-1.8827
At punch marks ^a	
Open end	1.8822
Solid end	1.8830
Inside diameter	1.000
Length at 1.88-in. diameter	6.017-6.018
Internal length	5
Overall length	6.773-6.775
Weight sealed (including fuel) after finish machining, g	529.63
Silicon in braze seal, g	0.50
Permeability coefficient of body, before assembly, for helium at 25°C, cm ² /sec	5.9×10^{-9}
Electrical resistance between punch marks, ohms	
Before inserting fuel cylinders	0.48
After inserting fuel cylinders	0.479
After sealing	0.488-0.489
Thermal insulating spacers	NCC Carbocell carbon grade 60

^aPunch marks were made at two points on the graphite for identification during postirradiation measurement.

A temperature history is presented in Appendix Table A-4. Fuel matrices are identified as A through D, with the A matrix being that nearest the reactor core. Typical average graphite wall temperatures at low power operation opposite matrices A, B, C, and D were 1100, 1040, 1020, and 920°F, respectively. At the design power level, the temperatures at the same relative locations were 1410, 1520, 1460, and 1360°F. Calculated temperatures are presented in Table 10. The nonuniformity of the wall

Table 10. Operating Conditions^a for Experiment ORNL-MTR-48-3

Fuel cylinder	A	B	C	D
Central temperature, ^b °F				
For 1 hr	3300	3600	3400	3200
For 248 hr	2400-2600	2300-2500	2200-2400	2000-2300
Power generation (for 1-hr period)				
Total fission power, kw	4.3	4.7	4.4	4.1
Fission power				
Matrix, w/cm ³	270	295	280	260
UC ₂ , w/g	1330	990	660	430
Power generation (for 248-hr period)				
Total fission power, kw	3.2	3.1	3.0	2.7
Fission power				
Matrix, w/cm ³	205	195	190	170
UC ₂ , w/g	1000	660	440	290
Thermal-neutron flux, neutrons/cm ² ·sec				
For 1 hr	4.4×10^{13}	3.3×10^{13}	2.2×10^{13}	1.4×10^{13}
For 248 hr	3.2×10^{13}	2.2×10^{13}	1.5×10^{13}	1.0×10^{13}
Total burnup (for 249 hr), at. % ²³⁵ U	1.46	0.97	0.66	0.43

^aValues were estimated from geometry and measured temperatures.

^bThe range in central temperature corresponds to the variation in measured temperatures.

temperature is indicative of thermal losses at the ends and of a steeper axial thermal-neutron flux gradient than was anticipated.

Fission power and ^{235}U burnup estimates are given in Table 10. The estimated total fission power for all fuel cylinders at the reduced operating conditions was 11.6 kw. Estimates of the average fission power densities at the reduced operating conditions varied from 185 w/cm³ for the A matrix to 160 w/cm³ for the D matrix. The corresponding thermal-neutron flux values were 3.1×10^{13} and 0.9×10^{13} neutrons/cm²·sec. The total ^{235}U burnup ranged from 1.4 to 0.43 at. % for the A and D matrices, respectively.

Experiment ORNL-MTR-48-4

The ORNL-MTR-48-4 experiment was a repeat of the third experiment in which the assembly was removed after one MTR cycle because of a high fission-gas release rate. The 48-3 and 48-4 elements were identical with the exception that the four fuel cylinders of 48-4 were coated with pyrolytic carbon as an additional barrier to fission-gas release. The pre-irradiation data are given in Table 11 for the fuel cylinders and in Table 12 for the encapsulating graphite.

This test was terminated after two MTR cycles, March 29, 1961 through May 1, 1961, again as the result of high fission-gas release rate. As in the third test, a leak was found during postirradiation examination in the silicon-brazed seal. After the high fission-gas release appeared about 12 hr after startup, the purge-gas activity increased rapidly to a level that prevented the use of the gas-sampling system. In experiment ORNL-MTR-48-3, on the other hand, the activity appeared in the purge gas almost immediately after reactor startup.

The radiation field at the sampling station was largely due to 2.8-hr ^{88}Kr , as was determined from the decay of an isolated sample of gas. The radioisotopes $^{85\text{m}}\text{Kr}$, ^{88}Kr , ^{133}Xe , and ^{135}Xe were identified in a gas sample that had decayed 14 hr following its isolation on the last day of the test. The ratios of release rate to birth rate, R/B, were computed to be 5.0×10^{-3} , 0.22×10^{-3} , 11×10^{-3} , and 0.22×10^{-3} , respectively.

Table 11. Preirradiation Data for ORNL-MTR-48-4 Fuel Cylinders

Fuel cylinder position ^a	A	B	C	D
Fuel cylinder identification	NCC 999R	NCC 1000R	NCC 1001R	NCC 1002R
Dimensions, in.				
Outside diameter	0.9990	0.9995	0.9984	1.0004
Length	1.2360	1.2358	1.2370	1.2348
Weight, g				
Total	29.370	30.632	31.767	33.968
UC ₂	3.15	4.66	6.71	9.69
²³⁵ U	2.66	3.93	5.67	8.19
Enrichment, ^b at. % ²³⁵ U	93.12	93.12	93.12	93.12
Bulk density, g/cm ³	1.859	1.938	2.012	2.145
Fuel loading, g/cm ³				
UC ₂	0.199	0.295	0.425	0.612
²³⁵ U	0.169	0.249	0.359	0.517
Specific axial electrical resistance, ^c μohm-cm	2590	2400	2360	2630
CCl ₄ pickup, ^d mg	0.7	<0.1	0.2	0.1
Final baking temperature, °C	2350	2350	2350	2350

^aPosition refers to location in the capsule, with A being at end nearest the reactor core and D at the opposite end.

^bIsotopic assay is the same as for ORNL-MTR-48-1.

^cBefore fuel cylinders were pyrolytic-carbon coated.

^dAfter fuel cylinders were pyrolytic-carbon coated.

The temperature history is presented in Appendix Table A-5. The assembly was operated 760 hr in the minimum thermal-neutron flux position (fully retracted), where the graphite wall temperatures opposite the fuel cylinders A, B, C, and D averaged 1220, 1160, 1150, and 1090°F, respectively. When the experiment was inserted to determine the position for design conditions, the average wall temperatures were 1520, 1450, 1460, and 1350°F, respectively. The element was operated in the design position for 19 hr during the first cycle, and then it was retracted to the

Table 12. Preirradiation Data for ORNL-MTR-48-4
Graphite Fuel Container

Material	NCC type ZT-1
Dimensions, in.	
Outside diameter	1.8802-1.8809
Inside diameter	~1.004
Length at 1.88-in. diameter	6.031
Internal length	5
Overall length	6.789
Bulk density, g/cm ³	2.015
Weight (including fuel), g	551.38
Silicon in braze seal, g	0.75
Permeability coefficient, before assembly, for helium at 25°C, cm ² /sec	5.0×10^{-9}

minimum flux position because of the high radiation field at the gas-sampling station.

Central temperature, fission power, and ²³⁵U burnup estimates are given in Table 13. The estimated total fission power at the reduced operating condition was 13.3 kw and the average fission-power densities varied from 215 w/cm³ for the A matrix to 195 w/cm³ for the D matrix. The total ²³⁵U burnup ranged from 5.2 to 1.5 at. % for the A and D matrices, respectively. The corresponding thermal-neutron flux values were 3.6×10^{13} and 1.1×10^{13} neutrons/cm².sec.

Postirradiation examination data are reported in Refs. 1, 4, and 10.

Experiment ORNL-MTR-48-5

The fifth experiment was designed to provide data on the retention of fission products in a graphite matrix fueled with pyrolytic-carbon-coated UC₂ particles. In the preceding 48-2, -3 and -4 tests, uncoated UC₂ particles dispersed in a graphite matrix cylinder were irradiated, and data were collected on the retention of fission products and fuel

Table 13. Operating Conditions^a for Experiment ORNL-MTR-48-4

Fuel cylinder	A	B	C	D
Central temperature, ^b °F				
For 19 hr	3400-3500	3300-3400	3300-3400	3000-3100
For 746 hr	2700-2900	2600-2800	2600-2700	2400-2600
Power generation (for 19-hr period)				
Total fission power, kw	4.5	4.3	4.3	4.0
Fission power				
Matrix, w/cm ³	285	275	275	255
UC ₂ , w/g	1430	930	650	410
Power generation (for 746-hr period)				
Total fission power, kw	3.7	3.4	3.4	3.2
Fission power				
Matrix, w/cm ³	230	215	215	205
UC ₂ , w/g	1170	740	510	330
Thermal-neutron flux, neutrons/cm ² ·sec				
For 19 hr	4.5×10^{13}	3.0×10^{13}	2.2×10^{13}	1.4×10^{13}
For 746 hr	3.7×10^{13}	2.4×10^{13}	1.7×10^{13}	1.1×10^{13}
Total burnup for 765 hr, at. % ²³⁵ U	5.4	3.4	2.4	1.5

^aValues were estimated from geometry and measured temperatures.

^bThe range in central temperature corresponds to the variation in measured temperatures.

materials in capsules of sealed low-permeability graphite. The 48-2 and 48-5 fuel elements were dimensionally similar, but the 48-5 elements were encapsulated in permeable graphite. Both assemblies were designed to operate at a total fission power of 2.75 kw, with estimated central temperatures of 2500°F and fission power densities of 140 w/cm³.

The 48-5 fuel element supplied by National Carbon Company is shown unassembled in Fig. 10. The fuel container and plug were made from permeable graphite, NCC grade ATJ. The plug was threaded into the graphite fuel container and locked with a graphite pin. The 1-in.-OD, 1 1/2-in.-long fuel cylinder fabricated by the National Carbon Company contained 3M Company pyrolytic-carbon-coated UC₂ particles dispersed in a graphite matrix. The outer 1/8-in. thickness of the cylinder was unfueled in order to form a protective graphite shell over the coated particle. The final baking of the fuel cylinder was at 1800°F for 30 min. Preirradiation data are presented for the fuel cylinder in Table 14 and for the graphite container and spacers in Table 15.

The fuel element was irradiated for six MTR cycles (18 weeks) between July 24, 1961 and November 27, 1961. The average graphite wall temperature was between 1500 and 1570°F (2550°F estimated central temperature) for 875 hr of irradiation. At this time, the purge-gas activity was approaching the limit for sampling, and the activity was decreased by retracting the fuel element stepwise into a region of lower neutron flux. This resulted in a graphite wall temperature between 940 and 990°F (1550°F estimated central temperature), which was held for 861 hr. The assembly was partially reinserted to achieve an estimated central temperature of 1750°F for the last 276 hr. The temperature history and estimated power and burnup are given in Table 16. A detailed temperature history is given in Appendix Table A-6.

The ratios of fission-gas release rate to birth rate are presented in Appendix Table A-7 and in Fig. 11 as a function of ²³⁵U burnup. The R/B ratios during the first 99 hr of irradiation with an estimated central temperature of 1750°F at first increased and then decreased. Between the 99th and 981st hr of irradiation, during which the estimated central temperature was maintained at 2550°F, the R/B ratios increased exponentially. The purge-gas activity and R/B ratios were reduced to workable

Fig. 10. Fuel Element Used in Experiment ORNL-MTR-48-5 Before Assembly. (Photograph obtained from Union Carbide Corporation Parma Research Center)

Fig. 11. Fission-Gas Release from ORNL-MTR-48-5 Fuel Elements as a Function of Fuel Temperature and ^{235}U Burnup.

Table 14. Preirradiation Data for ORNL-MTR-48-5
Fuel Cylinder

Fuel cylinder	
Identification	NCC 1080
Dimensions, in.	
Outside diameter	1.025-1.029
Length	1.497-1.501
Weight, g	
Total	35.803
UC_2	2.023
^{235}U	1.710
Bulk density, g/cm ³	1.760
Fuel loading, g/cm ³	
UC_2	0.0994
^{235}U	0.0841
Final baking temperature (30 min), °C	1800
Fuel particles	
Vendor and particle designation	3M-SP2
Uranium, wt %	42.7
Isotopic assay, at. %	
^{235}U	93.29
^{234}U	1.15
^{236}U	0.16
^{238}U	5.40
Fuel particle coating	
Carbon coating type	Lamellar
Nominal coating thickness, μ	82
Nominal core particle size, μ	175-250
Visual appearance	Polygonal, smooth

levels during the interval of 981 to 2113 hr by decreasing the estimated central temperature to 2100 and finally to 1550°F. At this temperature, during the interval 1252 to 2113 hr, the R/B ratios decreased to an apparent constant value ranging from 1/2 to 1/8 of that recorded with a 2550°F fuel temperature. The R/B ratios increased by factors of 2.5 to 10 during the final irradiation interval from 2113 to 2400 hr after the central temperature had been increased to 1750°F.

Interim postirradiation examination data are available in Ref. 4.

Table 15. Preirradiation Data for ORNL-MTR-48-5 Graphite Fuel Container and Thermal Insulating Spacers

Graphite container	
Material	NCC grade ATJ
Dimensions of container, in.	
Outside diameter	1.8750-1.8753
Length, at 1.87-in. diameter	2.508-2.509
Overall length	3.256
Assembled weight with fuel cylinder, g	204.17
Thermal insulating spacer	NCC Carbocell graphite grade 60

Table 16. Operating Conditions^a for ORNL-MTR-48-5

Central temperature, ^b °F	
For 909.5 hr	2400-2600
For 243.5 hr	1900-2100
For 1250 hr	1500-1800
Power generation (for 909.5-hr period)	
Total fission power, kw	2.8
Fission power	
Matrix, w/cm ³	140
UC ₂ , w/g	1390
Power generation (for 243.5-hr period)	
Total fission power, kw	2.3
Fission power	
Matrix, w/cm ³	110
UC ₂ , w/g	1130
Power generation (for 1250-hr period)	
Total fission power, kw	1.7
Fission power	
Matrix, w/cm ³	85
UC ₂ , w/g	870
Thermal-neutron flux, neutrons/cm ² .sec	
For 909.5 hr	4.7×10^{13}
For 243.5 hr	3.8×10^{13}
For 1250 hr	2.9×10^{13}
Total burnup (estimated) for 2430 hr, at. % ²³⁵ U	15.5

^aValues were estimated from geometry and measured temperatures.

^bThe range in central temperature corresponds to the variation in measured temperatures.

Experiment ORNL-MTR-48-6

A purged cylindrical graphite matrix containing pyrolytic-carbon-coated (U,Th)C₂ fuel particles and four sealed tubes containing loose pyrolytic-carbon-coated UC₂ and (U,Th)C₂ fuel particles were irradiated during the ORNL-MTR-48-6 test. The fueled cylindrical matrix and the loose (U,Th)C₂ particles were supplied by General Atomic. The loose UC₂ particles were supplied by National Carbon Company, by High Temperature Materials, and by the 3M Company. The fuel assembly was arranged as shown in Fig. 12. The fueled cylinder was located in the rear or lower thermal-neutron flux position in the capsule, where it operated at a fission-power density of 40 to 61 w/cm³, or specific power of 450 to 690 w per gram of ²³⁵U. The loose fuel particles were positioned between the fueled cylinder and the reactor and operated in an estimated specific power range of 1130 to 2100 w per gram of ²³⁵U. Eight fission-product deposition specimens of four different materials were mounted in the capsule so that a finished surface was exposed to the purge gas after it had passed over the fueled matrix. These were included to obtain information by postirradiation examination on the relative rates of plateout of fission products on different materials and additional information on the release of volatile fission products.

The purged fuel cylinder, 1 in. OD and 1 1/2 in. long, was supported in an AGOT graphite sleeve in place of the usual container. The AGOT graphite sleeve had a central thermocouple well in a 1/4-in.-diam central hole. The central temperature was measured with two 1/16-in.-OD tantalum-sheathed thermocouples; one was tungsten vs rhenium and the other Pt-10% Rh vs Pt. The graphite sleeve mid-wall temperature was measured at five points with Chromel vs Alumel thermocouples. The thermocouples in the graphite sleeve were protected from reaction with carbon by tubular nickel shields. This was deemed desirable because calculations indicated the possibility that graphite sleeve temperatures in excess of 1600°F would be required to achieve the design 2000°F central temperature as a consequence of the low power density of this experiment. No additional protection was attempted for the central thermocouples. The graphite sleeve also contained the helium purge-gas supply tube and flux monitoring

Fig. 12. Section Through Fuel Test Assembly ORNL-MTR-48-6.

wires in longitudinal holes, as in previous designs. Figure 13 shows the parts of the fuel element assembly. Preirradiation data are given in Table 17.

Table 17. Preirradiation Data for Fuel Cylinder, Support Sleeve, and Spacers for Experiment ORNL-MTR-48-6

Fuel cylinder

Vendor and designation

Cylinder	General Atomic BL-119E-3
Fuel particles	General Atomic GA-F1332-45E ^a

Dimensions, in.

Outside diameter	0.9998-1.0003
Inside diameter	0.2540-0.2543
Length	1.4989-1.5006

Weight, g

Total	37.1961
²³⁵ U	1.563

Volume (geometric), cm ³	17.74
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Enrichment, at. % ²³⁵ U	92.87
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Fuel loading

U, wt %	4.5
Th, wt %	9.9
²³⁵ U, g/cm ³	0.088

Support sleeve

Material	NCC type AGOT graphite
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Dimensions of sleeve, in.

Outside diameter	1.8452-1.8473
Inside diameter	1.0023-1.0027
Length at 1.84-in. diameter	1.5037-1.5041
Overall length	2.0039-2.0043

Thermal insulating spacer material	NCC Carbocell carbon grade 60
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Thermocouple protector tube	AGOT graphite
-----------------------------	---------------

^aParticle description is same as for GA-F1332-46E particles described in Table 18.

Fig. 13. Unassembled Fuel Element for Experiment ORNL-MTR-48-6.

The loose fuel particles sealed in helium were doubly contained, as shown in Fig. 14, to minimize the possibility of a fission-gas leak that would make the fission-gas release measurements of the purged element worthless. The inner wall was an Inconel tube with oxidized inner surfaces to minimize any reaction between the particles and the Inconel. The outer type 304 stainless steel tube was a close fit on the inner tube. These small capsules were inserted in close-fitting holes in a graphite cylinder. The graphite cylinder was supported on porous-carbon insulators, and a thermal-insulating gas gap was provided between the graphite wall and the water-cooled capsule vessel similar to that of the purge element. The temperature of the four small capsules was calculated from the graphite temperature, which was measured by two sheathed thermocouples, and the estimated power generation. The preirradiation data for the loose coated particles is given in Table 18.

The eight fission-product deposition specimens were in the form of plugs mounted in holes in a tubular section behind the purged fuel element, Fig. 15. The samples, two each of AGOT graphite, type 347 stainless steel, silver, and 2 1/4% Croloy steel, exposed a 1/2-in.-diam polished face toward the center of the tube in which they were mounted and through which the purge gas effluent passed. One of the stainless steel specimens had a thermocouple mounted on it. A layer of stainless steel shim stock reflective insulation was placed over the outside of these specimens to obtain an operating temperature range of 500 to 1000°F from gamma heat.

The assembly was irradiated for two MTR cycles (six weeks) between January 29 and March 12, 1962. During the initial startup, the central temperature of the purged element was held first at 1300°F and then 1600°F to permit the measurement of fission-gas release at lower temperatures. The assembly was then inserted in the beam hole to the design 2000°F central temperature. During the startup of the second reactor cycle, the assembly was again held at a reduced temperature, 1400°F, to evaluate any change in fission-gas release at the lower temperature. A temperature history is presented in Appendix Table A-8.

During the initial startup, the fission-gas release rate remained constant at 1300°F. It increased rapidly but leveled off within 1 1/2 hr

Fig. 14. Loose Fuel Particle Capsule and Graphite Parts of Fuel Assembly for Experiment ORNL-MTR-48-6.

Table 18. Preirradiation Data for Sealed-Tube Fuel Particle Elements and Support
Cylinder for Experiment ORNL-MTR-48-6

Encapsulating tube identification number				
Outer	2	5	8	4
Inner	1	2	7	8
Vendor and particle designation	NCC-BE	HTM-1	3M-SP2	GA-F1332-46E
Fuel particles				
U, wt %	43.1	47.2	42.7	19.92
Th, wt %	0	0	0	43.7
Isotopic assay (uranium), at. %				
^{235}U	92.61	93.0	93.29	92.87
^{234}U	1.13	1.13	1.15	1.09
^{236}U	0.33	0.16	0.16	0.33
^{238}U	5.93	5.71	5.40	5.70
Nominal particle core diameter, μ	175-246	150-250	175-250	
Fuel particle coating				
Carbon coating type	Duplex	Columnar	Laminar	Laminar
Nominal coating thickness, μ	100-150	82	82	50-60
Visual appearance	Spherical, oolitic	Spherical, oolitic	Polygonal, smooth	Rounded, not spherical
Fuel loading, g				
Particles in tube	0.6167	0.6428	0.6944	0.9477
^{235}U	0.246	0.282	0.286	0.175
^{235}U in 1 cm ³ of loose particles	0.688	0.789	0.772	0.491

PHOTO 37714

Fig. 15. Fuel Assembly for Experiment ORNL-MTR-48-6.

after each temperature increase, first to 1600°F and then to 2000°F. After 140 hr at 2000°F, the release rate began to increase with time. At the start of the second reactor cycle, the central temperature was held at 1400°F for a few days. Upon return to 2000°F, a large increase in release rate occurred, as shown in Fig. 16.

The estimated ^{235}U burnups for the fuel cylinder and loose particles are 2.3 and 6.6 at. %, respectively. Fission power data are presented in Table 19. Complete fission-gas release data are presented in Appendix Table A-9. The $^{85\text{m}}\text{Kr}$, ^{88}Kr , ^{133}Xe , and ^{135}Xe R/B values were 8.1×10^{-3} , 4.0×10^{-3} , 11×10^{-3} , and 1.3×10^{-3} , respectively, at the termination of the experiment. The ^{88}Kr R/B value was 33 times greater at 1400°F at the start of the second reactor cycle than it was at 1300°F at the start of the experiment.

The test was terminated after two cycles to permit the insertion of another test assembly that was scheduled for operation in the beam hole. Originally, it had been planned to reinsert the assembly in the reactor at a later date; however, the fission-gas release rate, although lower than for previous experiments, had become too high to be of interest.

The usual procedure employed for protecting the experimental assembly from moisture during shipping was not considered adequate for this fuel element. The ThC_2 in the fuel would, if the fuel particle coatings had ruptured, have been subject to the severe degradation in the presence of air. To protect the fuel after sectioning in the MTR hot cell, the purge lines were sealed by immersing the cut ends into a container partially filled with epoxy resin immediately after the cut.

Postirradiation examination data are given in Ref. 1.

Acknowledgment

Fuel elements for tests 48-1 through 48-5 were supplied by the National Carbon Company. The 48-6 test element materials were supplied by National Carbon Company, General Atomic, High Temperature Materials, and 3M Company. The tests were designed and programmed with the assistance of M. Janes, National Carbon Company, and F. L. Carlsen, Jr., Oak Ridge National Laboratory. The equipment was constructed under the

Fig. 16. Fission-Gas Release as a Function of Fuel Temperature and ^{235}U Burnup in Experiment ORNL-MTR-48-6.

Table 19. Operating Conditions^a for Experiment ORNL-MTR-48-6

	Fuel Cylinder GA B1-119E-3	Loose Fuel Particles			
		NCC-BE	HTM-1	3M-SP2	GA-F1332-46E
Central temperature of fuel cylinder or loose particles, °F					
For 579 hr	1970-2040	1100-1300	1100-1300	1100-1300	1100-1300
For 137.5 hr	1310-1620	700-900	700-900	700-900	700-900
Power generation (for 579 hr)					
Total fission power, kw	1.06	0.49	0.56	0.55	0.35
Fission power density of fuel cyl- inder or loose particles, w/cm ³	60	1360	1560	1530	970
Specific fission power, w/g					
UC ₂ (U,Th)C ₂	180	1670	1670	1670	520
Power generation (for 137.5 hr)					
Total fission power, kw	0.75	0.32	0.37	0.36	0.23
Fission power density of fuel cyl- inder or loose particles, w/cm ³	42	900	1030	1010	640
Specific fission power, w/g					
UC ₂ (U,Th)C ₂	125	1100	1100	1100	345
Thermal-neutron flux, neutrons/cm ² ·sec					
For 579 hr	1.9 × 10 ¹³	5.6 × 10 ¹³	5.6 × 10 ¹³	5.6 × 10 ¹³	5.6 × 10 ¹³
For 137.5 hr	1.3 × 10 ¹³	3.5 × 10 ¹³	3.5 × 10 ¹³	3.5 × 10 ¹³	3.5 × 10 ¹³
Total burnup (for 716.5 hr), at. % ²³⁵ U	2.3	6.6	6.6	6.6	6.6

^aValues were estimated from geometry and measured temperatures.

direction of R. L. Senn from drawings prepared by ORNL Reactor Division Design Department. The instrumentation was handled by T. L. McLean and G. H. Burger of Oak Ridge National Laboratory and C. C. Hall on loan to Oak Ridge National Laboratory from Lockheed Aircraft Corporation, Marietta, Georgia. Operating arrangements at the MTR reactor were made through Project Engineers R. B. Van Sice, F. R. Vandeweile, and J. A. Cunningham of Phillips Petroleum Company. Fission gas analyses were performed by E. Turk and D. G. Olson of Phillips Petroleum Company and by T. H. Handley of Oak Ridge National Laboratory.

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Appendix A

TEMPERATURE AND FISSION-GAS RELEASE HISTORIES FOR
ORNL-MTR-48-6 EXPERIMENTS 1 THROUGH 6

Tables A-1 through A-9 give the detailed operating time, temperature, and measured fission-gas release rates, where applicable, for the six irradiation experiments of the ORNL-MTR-48 program.

Table A-1. Temperature History for Experiment ORNL-MTR-48-1

Irradiation Interval ^{a, b} (hr)	Encapsulating Graphite Wall Temperature (°F)		Central Fueled-Graphite Cylinder Temperature, ^c (°F)	
	Fuel Cylinder		Fuel Cylinder	
	A	B	A	B
0-18.5	1030-1090	780-802	2200-2300	1700-1800
18.5-60	1560-1590	1300-1340	3400-3500	2900-3000
60-64	1090	820	2300	1800
64-165.5	1590-1610	1320	3500	3000
165.5-167.5	1060	790	2300	1700
167.5-266.5	1580-1600	1320-1340	3500	3000
266.5-272.5	1130	860	2400	1900
272.5-563.5	1540-1580	1300-1360	3400-3500	2900-3100
563.5-565.5	1120	870	2400	1900
565.5-809	1550-1610	1380-1430	3400-3500	3100-3200
809-810.5	1080	860	2300	1900
810.5-991.5	1570-1590	1400-1410	3500	3100-3200
991.5-992.5	1070	850	2300	1800
992.5-1204	1540-1600	1370-1430	3400-3500	3100-3200
1204-1205	1030	810	2200	1800
1205-1396.5	1590-1630	1380-1460	3500-3600	3100-3300

^aReactor down time is not included.

^bThere were 36 reactor power reductions greater than 50% during the test period.

^cCentral temperatures were estimated from geometry and measured temperatures.

Table A-2. Temperature History for Experiment ORNL-MTR-48-2

Irradiation Interval ^{a, b} (hr)	Encapsulating Graphite Wall Temperature (°F)	Central Fueled-Graphite Cylinder Temperature ^c (°F)
0-17	1340-1360	2200-2300
17-290.5	1460-1500	2400-2500
290.5-617	1520-1590	2500-2600
617-674	1480-1520	2500
674-676	1040	1700
676-958	1480-1560	2500-2600
958-1167	1500-1510	2500
1167-1179	1290	2100
1179-1307	1490-1510	2500
1307-1735.5	1440-1510	2400-2500
1735.5-1967.5	1460-1510	2400-2500
1967.5-2395	1480-1520	2500
2395-2495.5	1420-1490	2400-2500
2495.5-2797	1040-1070	1700-1800

^aReactor down time is not included.

^bThere were 56 reactor power reductions greater than 50% during the test period.

^cCentral temperature was estimated from geometry and measured temperatures.

Table A-3. Fission-Gas Escape Rate for Experiment ORNL-MTR-48-2

Time at Power Before Gas Sampling (hr)	Estimated Fuel Cylinder Central Temperature ^a (°F)	Estimated Burnup ^a (at. % ²³⁵ U)	Ratio of Fission-Gas Escape Rate to Birth Rate, R/B			
			^{85m} Kr	⁸⁸ Kr	¹³³ Xe	¹³⁵ Xe
			× 10 ⁻³	× 10 ⁻³	× 10 ⁻³	× 10 ⁻³
64	2500	0.5		0.0073	0.23	
215	2450	1.6			0.51	
283.5	2450	2.1			1.2	
342	2650	2.6	0.76	0.39	18	0.014
507.5	2600	3.9	3.3	2.0	180	
825	2600	6.4	5.7	2.2	390	0.31
827.5	2600	6.4	3.2	1.3	72	0.12
1086.5	2500	8.4	3.1	0.86	240	0.45
2095.5	2500	16.1			330	
2261.5	2500	17.4	5.6	1.4	150	0.32
Experiment terminated		20.7				

^aEstimates based on geometry and bottle wall temperature.

Table A-4. Temperature History for Experiment ORNL-MTR-48-3

Irradiation Interval ^{a,b} (hr)	Encapsulating Graphite Wall Temperature (°F)				Central Fueled-Graphite Cylinder Temperature ^c (°F)			
	Fuel Cylinder				Fuel Cylinder			
	A	B	C	D	A	B	C	D
0-32.5	1040	1080	1040	1020	2400	2500	2400	2300
32.5-33.5	1410	1520	1460	1360	3300	3600	3400	3200
33.5-249	1080-1130	1020-1060	980-1050	900-930	2500-2600	2300-2400	2200-2400	2000-2100

^aReactor down time is not included.

^bThere were seven reactor power reductions greater than 50% during the test period.

^cCentral temperatures were estimated from geometry and measured temperatures.

Table A-5. Temperature History for Experiment ORNL-MTR-48-4

Irradiation Interval ^{a,b} (hr)	Encapsulating Graphite Wall Temperature (°F)				Central Fueled-Graphite Cylinder Temperature ^c (°F)			
	Fuel Cylinder				Fuel Cylinder			
	A	B	C	D	A	B	C	D
0-2.5	1200	1160	1160	1100	2800	2700	2700	2500
2.5-8.5	1450	1500	1410	1300	3400	3300	3300	3000
8.5-55.5	1190-1200	1140	1130-1140	1070	2700-2800	2600	2600	2500
55.5-68.5	1520	1450	1460	1350	3500	3400	3400	3100
68.5-76.5	1200-1250	1120-1210	1120-1180	1060-1130	2800-2900	2600-2800	2600-2700	2400-2600

^aReactor down time is not included.

^bThere were eight reactor power reductions greater than 50% during the test period.

^cCentral temperatures were estimated from geometry and measured temperatures.

Table A-6. Temperature History for Experiment ORNL-MTR-48-5

Irradiation Interval ^{a, b} (hr)	Encapsulating Graphite Wall Temperature (°F)	Central Fueled-Graphite Cylinder Temperature ^c (°F)
0-99	1050-1090	1700-1800
99-761.5	1500-1550	2500-2600
761.5-763.5	1140	1900
763.5-829	1530-1550	2600
829-834	1190	2000
834-981	1510-1560	2500-2600
981-1005	1410-1420	2400
1005-1215	1270-1290	2100
1215-1225.5	1510	2500
1225.5-1252	1240	2000
1252-2113	940-990	1500-1600
2113-2127.5	1010	1700
2127.5-2403	1070-1090	1700-1800

^aReactor down time is not included.

^bThere were 34 reactor power reductions greater than 50% during the test period.

^cCentral temperature was estimated from geometry and measured temperatures.

Table A-7. Fission-Gas Escape Rates for Experiment ORNL-MTR-48-5

Time at Power Before Gas Sampling (hr)	Estimated Fuel Cylinder Central Temperature ^a (°F)	Estimated Burnup ^a (at. % ²³⁵ U)	Ratio of Fission-Gas Escape Rate to Birth Rate, R/B			
			^{85m} Kr	⁸⁸ Kr	¹³³ Xe	¹³⁵ Xe
			× 10 ⁻³	× 10 ⁻³	× 10 ⁻³	× 10 ⁻³
3	1750	0.0	4.5	1.5	7.9	0.6
93	1800	0.5	2.4	0.5	2.7	0.7
95	1800	0.5	2.0	0.5	2.7	0.6
266	2550	2.0	2.9	1.1	4.1	0.4
520	2550	4.2	6.5	3.3	10.1	0.8
688	2600	5.6	14.2	5.3	13.5	1.2
927	2600	7.6	42	18	47	2.6
1095	2100	9.0	19	3.4	95	1.3
1369	1600	10.6	9.5	3.8	29	1.7
1469	1550	11.2	9.7	2.5	13	1.5
1493	1550	11.3	12	4.4	28	2.2
1747	1600	12.5	11	3.6	13	2.2
1888	1600	13.3	13	3.7	15	0.9
2102	1550	14.3	7.3	2.7	4.8	1.2
2126 ^b	1650	14.4	2.0	0.5	9.8	0.4
2150	1800	14.7	36	6.5	65	3.5
2275	1750	15.4	20	6.8	57	3.8
2400	Experiment terminated	16.0				

^aEstimates based on geometry and bottle wall temperature.

^bQuestionable because of sampling difficulties.

Table A-8. Temperature History for Experiment ORNL-MTR-48-6

Irradiation Interval ^{a,b} (hr)	Purged Element Temperature (°F)		Sealed Tube Element Temperature (°F)		Fission Product Deposition Specimen Temperature ^d (°F)
	Graphite Support Sleeve Wall	Fuel Cylinder Center	Graphite Support Cylinder	Coated Particle Maximum ^c	
0-78	1030-1040	1310-1320	400-410	700-800	550
78-103	1250-1260	1610-1620	490	900	650
103-401	1500-1540	1990-2040	580-630	1100-1200	770-800
401-435.5	1080	1420-1430	470	900	590
435.5-716.5	1460-1480	1970-2010	620-650	1200-1300	770-780

^aReactor down time is not included.

^bThere were six reactor power reductions greater than 50% during the test period.

^cMaximum fuel particle temperatures were estimated from geometry and measured temperatures.

^dType 347 stainless steel specimen.

Table A-9. Fission-Gas Escape Rate for Experiment ORNL-MTR-48-6

Time at Power Before Gas Sampling (hr)	Fuel Cylinder Central Temperature ^a (°F)	Estimated Burnup ^b (at. % ²³⁵ U)	Ratio of Fission-Gas Escape Rate to Birth Rate, R/B			
			^{85m} Kr	⁸⁸ Kr	¹³³ Xe	¹³⁵ Xe
			× 10 ⁻³	× 10 ⁻³	× 10 ⁻³	× 10 ⁻³
73	1315	0.17	0.024	0.0070	0.021	0.0051
97	1605	0.23	0.056	0.022	0.028	0.0086
121	1990	0.31	0.24	0.100	0.076	0.016
241	2035	0.72	0.32	0.097	0.13	0.017
407	1425	1.34	0.40	0.22	0.26	0.066
427	1410	1.39	0.46	0.24	0.21	0.13
451	1965	1.46	2.8	1.03	1.9	0.34
570	1985	1.85	6.2	3.3	3.4	1.1
642	1985	2.09	8.1	4.0	11	1.3
717	1995	2.29	Experiment terminated			

^aFrom tungsten versus rhenium thermocouple.

^bEstimates based on geometry and sleeve wall temperature.

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