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POSTIRRADIATION EXAMINATION OF CAPSULE GF-4

by
W. J. KOVACS and B. J. SEDLAK

Prepared under
Contract DE-AT03-76ET35300
for the San Francisco Operations Office
Department of Energy

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ABSTRACT

The GF-4 capsule test was irradiated in the SILOE reactor at Grenoble, France between April 8, 1975 and July 26, 1976. High-enriched uranium (HEU) UC_2 and weak acid resin (WAR) UC_xO_y fissile and ThO_2 fertile particles were tested over the following exposure conditions:

1. 5.6 to 9.5×10^{25} n/m^2 ($E > 29$ fJ_{HTGR}).
2. 1010° to $1120^\circ C$ time-volume average temperature.
3. UC_2 and UC_xO_y burnup between 69.1% and 77.2% fissions per initial metal atom (FIMA) and ThO_2 burnup between 2.0% and 4.8% FIMA.

Postirradiation examination of cured-in-place fuel rods showed no fuel rod/graphite element interaction. In addition, all rods exhibited adequate structural integrity. Irradiation-induced dimensional changes for rods containing all TRISO-coated fuel were consistent with model predictions; however, rods containing BISO-coated fuel exhibited greater volumetric contractions than predicted.

A comparison of predicted and measured in-pile Kr-85m release/birth (R/B) values for TRISO UC_xO_y /BISO ThO_2 (cell I) and TRISO UC_2 /TRISO ThO_2 (cell II) showed good agreement, which implies $\leq 0.1\%$ pressure vessel failure. Furthermore, the performance of the all-TRISO fuel system tested in cell II provides strong empirical support for licensing TRISO HEU UC_2/ThO_2 in the Fort St. Vrain high-temperature gas-cooled reactor (HTGR) plant. Kr-85m R/B for laser-failed WAR UC_xO_y particles was measured between 7.5% and 30%. The high release is attributed to porous kernels (typically 65% porosity) and provides additional support for the decision made in 1976 to drop

further development of low-density WAR UC_xO_y kernels. Stoichiometric limits of $0.6 \leq \text{oxygen/uranium ratio} \leq 1.7$ for UC_xO_y kernels are supported by rare-earth fission product distributions measured in GF-4 fuel.

No amoeba migration was observed in UC_2 , UC_xO_y , or ThO_2 fuel tested in GF-4, and limited ($< 7 \mu m$) of lanthanide/silicon carbide (SiC) attack was observed in UC_2 . These results are consistent with fuel performance models.

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1. INTRODUCTION.

A General Atomic/Commissariat a l'Energie Atomique (GA/CEA) cooperative program was instituted in 1973 to qualify the French CEA and its industrial partners to manufacture and market high-temperature gas-cooled reactor (HTGR) fuel, based on technology licensed from GA. The program was also designed to significantly contribute to the HTGR fuel development already under way at GA.

The GF-4 capsule test, irradiated in the SILOE reactor at Grenoble, France was the fourth in a series of instrumented irradiation tests planned under the cooperative GA/CEA program.

Capsule GF-4 was designed to provide a portion of the qualification data for the candidate Fort St. Vrain (FSV) VSM^{*} TRISO/ThO₂ TRISO) fuel system. In addition, GF-4 was designed to serve with later GF-series capsules as a benchmark for comparing GA-and CEA-fabricated fuel [i.e., HEU^{**} UC₂ TRISO-coated fissile particles with kernels fabricated by the weak acid resin (WAR) and VSM processes and ThO₂ BISO- and TRISO-coated fertile particles]. The GF-4 capsule is expected to define the irradiation performance of fissile carbide (UC₂ VSM and UC_xO_y WAR) and fertile BISO ThO₂ fuel particles manufactured to reference particle specifications. In addition, advanced particle concepts were evaluated to define the optimum carbon and oxygen contents of WAR kernels and the efficacy of different doping agents on restricting fission product release and kernel migration.

GF-4, irradiated between April 8, 1975 and July 26, 1976, contained three separate cells (namely, cells I and II contained three fuel rods each,

^{*} Patented GA process for production of dense UC₂ kernels.

^{**} HEU refers to high-enriched uranium (93 wt % U-235) UC₂ and UC_xO_y kernels.

and cell III contained 113 loose particle planchets). The cells were continuously monitored for surface temperature of fuel samples and periodically purged for fission gas release measurements. The range of irradiation conditions for GF-4 is summarized as follows:

1. Fast fluence exposure of 5.6 to $9.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}.
2. Time-volume average fuel rod temperature between 1010° and 1120°C with loose particle samples between 1070° and 1120°C .
3. UC_2 and UC_xO_y fissile particle burnups between 65.4% and 77.2% fissions per initial metal atom (FIMA) and ThO_2 burnups of between 2.5% and 4.8% FIMA.

This report will detail the GF-4 irradiation experiment, the results of postirradiation examination, and the analysis of results relative to fuel material properties and performance. In addition, the report will emphasize the relationship between particle performance and the generic TRISO-coated particle data base.

2. DESCRIPTION OF EXPERIMENTS

2.1. OBJECTIVES

The planned objectives for GF-4, in decreasing order of priority, were as follows:

1. Evaluate the irradiation performance of HEU UC_2 (VSM) TRISO particles in combination with ThO_2 TRISO particles.
2. Evaluate the irradiation performance of HEU UC_xO_y (WAR) TRISO particles in combination with BISO ThO_2 particles in fuel rods and loose particles.
3. Evaluate the effect of different methods of carbonizing on the irradiation performance of fuel rods. The different methods used for carbonizing the GF-4 fuel rod matrix were (a) cure-in-place, (b) cure in alumina beds, (c) cure in graphite tubes, then remove prior to irradiation.
4. Evaluate the irradiation performance of a number of developmental reference and advanced fuel particles:
 - a. Resin UC_xO_y kernels with varying oxygen and carbon ratios:

$$3.0 < x < 4.3 \text{ and } 0.2 < y < 2.0$$

The kernels had both TRISO and BISO coatings. The TRISO coating tested silicon carbide (SiC) and zirconium carbide (ZrC) coating layers as the primary containment of fission products. The BISO coatings tested how doping the outer

pyrocarbon (OPyC) with silicon or zirconium affected the ability of the particles to improve fission product retention.

- b. Particles with ThO_2 kernels doped with varying amounts of SiO_2 , Al_2O_3 , and SiC . The particles had a reference BISO coating adjusted to the different kernel sizes.
5. Evaluate the irradiation effect of varying the OPyC coating rate between 4 and 6 $\mu\text{m}/\text{min}$, as part of a continuing effort to define material specifications for fissile and fertile particles.
6. Determine the compatibility of the fuel rods with two nearly isotropic graphite fuel crucibles (Great Lakes Carbon Corporation H-451 and Union Carbide TS-1240) by measuring the fuel rod dimensional change.

2.2. DESCRIPTION OF FUEL SPECIMENS

2.2.1. Unbonded Particles

Cell III contained loose fissile and fertile particles which were tested as either advanced fuel designs or developmental reference fuel designs. Advanced particle designs were tested to define potential improvements for fission product retention of BISO fertile and TRISO and BISO fissile fuel particles. Tables 2-1* and 2-2 list the unbonded particle samples tested and show the particle batch numbers, the individual particle loadings, the position in the cell, and the critical particle properties.

2.2.1.1. Fissile Particles. The fissile particle variables tested included UC_2 fissile kernels fabricated by two different processes: (1) reference fresh, designated VSM UC_2 (kernels solidified from the liquid phase) and (2)

* Tables and figures are grouped at the end of each section.

WAR UC_xO_y (synthesized from Diamond Shamrock C464, a magnesium WAR). VSM UC_2 and WAR UC_xO_y kernels were typically 98% and 35% of theoretical density, respectively. Table 2-3 generally describes these two fissile particle types which were tested in fuel rods (cells I and II) and loose particles (cell III). The loose fissile particles tested in cell III (refer to Table 2-2) investigated the stoichiometry of UC_xO_y kernels over the ranges $3.0 < x < 4.3$ and $0.2 < y < 2.0$. Other fissile particle design variations tested in cell III were

1. A comparison of the irradiation performance of particles with SiC or ZrC TRISO coatings.
2. A comparison of the irradiation performance of BISO coated particles with OPyC layer doped with either silicon or zirconium.

2.2.1.2. Fertile Particles. The fertile particles tested in GF-4 include ThO_2 kernels with BISO and TRISO coatings. Table 2-3 lists four particle batches tested in fuel rods and as loose particles. The three TRISO particle batches were designed to test the effects on irradiation performance of OPyC thickness and coating rate. One batch of BISO ThO_2 particles was included to compare with the TRISO ThO_2 particles.

A number of BISO ThO_2 batches were tested as loose particles in cell III (refer to Table 2-2). These particles were designed to test the ability of dopants added to the kernel or the OPyC of BISO particles to stabilize and increase the retention of fission products in the particle. The OPyC layers were doped with silicon and zirconium, and the ThO_2 kernels were doped with silicon, aluminum, and SiC.

2.2.1.3. Inert Particles. One batch of TRISO-coated carbon kernels (refer to Table 2-3) was used in fuel rods tested in cells I and II. The purpose of the TRISO-coated inert particles was to allow simulation of the representative particle volume loading without exceeding metal loading restrictions.

2.2.1.4. Coated Particle Properties. Reference 2-1 details the advanced and developmental reference fuel particles tested in GF-4.

2.2.2. Fuel Rods

2.2.2.1. Description. Six fuel rods were irradiated in cells I and II of GF-4. Table 2-4 summarizes the critical rod properties. Cell I contained three $\text{UC}_3\text{O}_{0.6}$ (WAR) TRISO/ ThO_2 BISO fuel rods, and cell II contained three UC_2 (VSM) TRISO/ ThO_2 TRISO fuel rods. The fuel rods in cell I were fabricated with three separate curing treatments: (1) cured-in-place in H-451 graphite, (2) cured in a bed of Al_2O_3 , and (3) cured in a H-451 graphite tube, then removed and reinserted in a graphite crucible. All of the fuel rods in cell II were cured-in-place in TS-1240 graphite and were not removed from the graphite crucibles before irradiation. Reference 2-1 details all rod properties.

2.2.2.2. Fuel Rod Constituent Variable. Fuel rods were fabricated by injecting a matrix phase into a close-packed bed of particles (namely, fissile, fertile, inert, and shim particles). The particle packing fraction in a rod was typically 0.57. Table 2-3 describes fissile, fertile, and inert particles. The shim particles were 1099 (Great Lakes Carbon Corporation) isotropic graphite nominally 800 μm in diameter and impregnated with furfuryl alcohol. A single matrix blend of 47 wt % A-240 petroleum pitch, 25 wt % LONZA Ks-15 graphite filler, and 10 wt % octadecanal +5 wt % polystyrene additives was used in fabricating all fuel rods.

2.2.2.3. Fabrication. Fuel rods 1.573 cm in diameter by 5.0 cm long were fabricated using a prototype HTGR hot-injection press. All rods except IB were carbonized in-place in TS-1240 crucibles or H-451 tubes. The loaded crucibles and tubes were heated in an induction furnace at the rate of $\sim 15^\circ\text{C}/\text{min}$ in nitrogen to 1000°C . The final firing was done in argon at $1800^\circ\text{C} \pm 50^\circ\text{C}$ for ~ 30 min with a heatup rate of $\sim 10^\circ\text{C}/\text{min}$ between 1000° and 1800°C . After firing, the rods were pushed out of the graphite crucibles and tubes with a small hydraulic press. The total load required to push out

the fuel rods ranged between 10.3 and 27.2 kg (25 and 60 lb), which was the equivalent of 45 to 108 kN/m² (6.54 to 15.70 lb/in.²) shear strength bond at the fuel rod/graphite interface. One set of rods designated for position IB was carbonized by initially heating the rods (at ~6°C/min) in a packed bed of Al₂O₃ to 1000°C in nitrogen followed by a high fire to 1800°C in argon.

REFERENCE

- 2-1. Kovacs, W. J., and D. P. Harmon, "Preirradiation Report GA Fuel Materials for GF-4," General Atomic Report GA-A13475, September 1, 1975.

TABLE 2-1
UNBONDED FISSILE AND FERTILE PARTICLES TESTED IN CELL III OF GF-4

Tray Designation	Location in Capsule (mm from core bottom)	Design	Kernel Type	Data Retrieval No.	Particle Loading		Distribution of Different Particle Types per Tray Position	Total No. of Particles per Tray Position	X-ray Plate I.D.
					(μg U-235/ particle)	(μg Th-232/ particle)			
1	460.3	(a)	10.4:1 (Th,U) O_2	6332-63E	36.6	375	1/5	101	--
1	460.3	(a)	10.4:1 (Th,U) O_2	6332-67E	36.6	375	1/5	101	--
1	460.3	(a)	10.4:1 (Th,U) O_2	6332-55E	36.6	375	1/5	101	--
1	460.3	(a)	10.4:1 (Th,U) O_2	6332-57E	36.6	375	1/5	101	--
1	460.3	(a)	10.4:1 (Th,U) O_2	6411-101E	36.6	375	1/5	101	--
2	461.6	(a)	Th O_2	6721-31E	--	503	1/3	101	--
2	461.6	(a)	Th O_2	6562-97E	--	503	1/3	101	--
2	461.6	(a)	Th O_2	6562-101E	--	503	1/3	101	--
3	462.8	(a)	8:1(Th,U) O_2	6171-105E	31.8	242	1/5	108	--
3	462.8	(a)	8:1(Th,U) O_2	6171-107E	31.8	242	1/5	108	--
3	462.8	(a)	8:1(Th,U) O_2	6171-115E	31.8	242	1/5	108	--
3	462.8	(a)	8:1(Th,U) O_2	6171-117E	31.8	242	1/5	108	--
3	462.8	(a)	8:1(Th,U) O_2	6171-67E	31.8	242	1/5	108	--
4	464.1	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
5	465.4	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
6	466.5	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
7	467.8	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
8	468.9	(a)	UC $_4$.302.0	6562-7E	40.6	--	1	128	--
9	470.2	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
10	471.3	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
11	472.6	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
12	473.9	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
13	475.0	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
14	476.3	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
15	477.4	(a)	UC $_3$.200.8	6552-141E	45.8	--	1	128	--
16	478.7	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
17	480.0	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
18	481.1	(a)	UC $_4$.001.4	6551-53E	46.3	--	1	128	--
19	482.4	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
20	483.5	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
21	484.8	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
22	485.9	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
23	487.2	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
24	488.3	(a)	UC $_4$.302.0	6551-67E	40.6	--	1	128	--
25	489.6	(b)	Th O_2	6252-05-010	--	642	1	101	LA510
26	490.7	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
27	492.0	(b)	Th O_2	6252-05-010	--	642	1	101	LA510
28	493.3	(b)	Th O_2	6252-05-010	--	642	1	101	LA510
29	494.4	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
30	495.7	(b)	Th O_2	6252-05-010	--	642	1	101	LA510
31	496.8	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
32	498.1	(b)	Th O_2	6252-05-010	--	642	1	101	LA510
33	499.4	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
34	500.5	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
35	501.8	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
36	503.1	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
37	504.2	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
38	505.5	(b)	Th O_2	6252-04-010	--	673	1	101	LA437
39	506.6	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
40	507.9	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6721-3	--	586	1	101	--
41	509.2	(a)	+1 wt % SiC 1:0.07:0.28 (Th,Si,Al) O_2	6721-3	--	586	1	101	--
42	510.3	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
43	511.6	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
44	512.9	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
45	514.0	(a)	UC $_3$.200.8	6551-51E	45.8	--	1	128	--
46	515.3	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
47	516.6	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
48	517.7	(a)	UC $_3$.200.8	6551-37E	45.8	--	1	128	--
49	519.0	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
50	520.1	(b)	UC $_3$.000.5	6157-02-010	35.5	--	1	128	LA426
51	521.4	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
52	522.5	(a)	UC $_3$.200.8	6552-135E	45.8	--	1	128	--
53	523.8	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--

TABLE 2-1 (continued)

Tray Designation	Location in Capsule (mm from core bottom)	Design	Kernel Type	Data Retrieval No.	Particle Loading		Distribution of Different Particle Types per Tray Position	Total No. of Particles per Tray Position	X-ray Plate I.D.
					(μg U-235/particle)	(μg Th-232/particle)			
54	525.1	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
55	526.1	(a)	UC $_3$.0 O_2 .2	6552-123E	45.8	--	1	164	--
56	527.4	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
57	528.7	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--
58	529.7	(a)	UC $_3$.0 O_2 .2	6551-9E	45.8	--	1	164	--
59	531.0	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--
60	532.3	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
61	533.4	(a)	UC $_4$.0 O_2 1.4	6551-55E	46.3	--	1	128	--
62	534.7	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
63	536.0	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-21EDS	24.8	308	1/4	101	--
63	536.0	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-E	24.8	308	3/4	101	--
64	537.3	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
65	538.6	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-23EDS	24.8	308	1/4	101	--
65	538.6	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-23E	24.8	308	3/4	101	--
66	539.9	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
67	541.0	(a)	UC $_4$.0 O_2 1.4	6552-143E	46.3	--	1	128	--
68	542.3	(b)	Th O_2	6542-27-010	--	598	1	101	LA109-1, 2
69	543.6	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-25EDS	24.8	308	1/4	101	--
69	543.6	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-25E	24.8	308	3/4	101	--
70	544.9	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
71	546.1	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-27EDS	24.8	308	1/4	108	--
71	546.1	(a)	12:1:1:1.7 (Th,U,Si,Al) O_2	6562-27E	24.8	308	3/4	108	--
72	547.4	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
73	548.7	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
74	549.8	(a)	UC $_4$.0 O_2 1.4	6552-149E	46.3	--	1	128	--
75	551.1	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-111	--	502	1	101	--
76	552.4	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--
77	553.5	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
78	554.8	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--
79	555.9	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
80	557.2	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
81	558.5	(a)	10.4:1 (Th,U) O_2	6332-63EDS	36.6	375	1/5	101	--
81	558.5	(a)	10.4:1 (Th,U) O_2	6332-67EDS	36.6	375	1/5	101	--
81	558.5	(a)	10.4:1 (Th,U) O_2	6332-55EDS	36.6	375	1/5	101	--
81	558.5	(a)	10.4:1 (Th,U) O_2	6552-29EDS	36.6	375	1/5	101	--
81	558.5	(a)	10.4:1 (Th,U) O_2	6411-101EDS	36.6	375	1/5	101	--
82	559.8	(a)	1:0.07:0.28 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
83	561.0	(a)	8:1 (Th,U) O_2	6171-105EDS	31.8	242	1/5	108	--
83	561.0	(a)	8:1 (Th,U) O_2	6171-107EDS	31.8	242	1/5	108	--
83	561.0	(a)	8:1 (Th,U) O_2	6171-115EDS	31.8	242	1/5	108	--
83	561.0	(a)	8:1 (Th,U) O_2	6171-117EDS	31.8	242	1/5	108	--
83	561.0	(a)	8:1 (Th,U) O_2	6171-67EDS	31.8	242	1/5	108	--
84	562.3	(b)	Th O_2	6252-03-010	--	676	1	101	LA438
85	563.6	(a)	1:0.07:0.07 (Th,Si,Al) O_2	6291-119	--	537	1	101	--
86	564.7	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
87	566.0	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--
88	567.1	(b)	UC $_2$	6151-17-020	40.2	--	1	128	LA425
89	568.4	(a)	1:0.07:0.28 (Th,Si,Al) O_2 +1 wt % SiC	6721-3	--	586	1	101	--

TABLE 2-1 (continued)

Tray Designation	Location in Capsule (mm from core bottom)	Design	Kernel Type	Data Retrieval No.	Particle Loading		Distribution of Different Particle Types per Tray Position	Total No. of Particles per Tray Position	X-ray Plate I.D.
					(μ g U-235/ particle)	(μ g Th-232/ particle)			
90	569.5	(b)	UC ₂	6151-17-020	40.2	—	1	128	LA425
91	570.8	(a)	1:0.07:0.07 (Th,Si,Al) ₂ O ₂	6291-111	—	502	1	101	—
92	571.9	(b)	UC ₂	6151-17-020	40.2	—	1	128	LA425
93	573.2	(b)	ThO ₂	6252-05-010	—	642	1	101	LA510
94	574.3	(b)	UC ₂	6151-17-020	40.2	—	1	128	LA425
95	575.6	(a)	1:0.01:0.01 (Th,Si,Al) ₂ O ₂	6551-147	—	547	1	101	—
96	576.8	(a)	ThO ₂	6721-31EDS	—	503	1/3	108	—
96	576.8	(a)	ThO ₂	6721-31E	—	503	2/3	108	—
97	577.9	(a)	UC _{3.0} O _{0.2}	6551-7E	45.8	—	1	128	—
98	579.1	(a)	ThO ₂	6562-97EDS	—	503	1/3	108	—
98	579.1	(a)	ThO ₂	6562-97E	—	503	2/3	108	—
99	580.4	(b)	ThO ₂	6252-04-010	—	673	1	101	LA437
100	581.5	(a)	UC _{3.0} O _{0.2}	6552-131E	45.8	—	1	128	—
101	582.8	(a)	ThO ₂	6562-191EDS	—	503	1/3	101	—
101	582.8	(a)	ThO ₂	6562-101	—	503	2/3	101	—
102	583.9	(a)	UC _{4.3} O _{0.2}	6551-65E	40.6	—	1	128	—
103	585.2	(a)	1:0.03:0.03 (Th,Si,Al) ₂ O ₂	6551-149	—	529	1	101	—
104	586.3	(a)	UC _{4.3} O _{2.0}	6562-7E	40.6	—	1	128	—
105	587.6	(a)	1:0.07:0.28 (Th,Si,Al) ₂ O ₂	6291-119	—	537	1	101	—
106	588.9	(a)	10.4:1 (Th,U) ₂ O ₂	6332-63EDS	36.6	375	1/5	101	—
106	588.9	(a)	10.4:1 (Th,U) ₂ O ₂	6552-29EDS	36.6	375	1/5	101	—
106	588.9	(a)	10.4:1 (Th,U) ₂ O ₂	6411-101EDS	36.6	375	1/5	101	—
107	590.2	(a)	1:0.07:0.28 (Th,Si,Al) ₂ O ₂ +1 wt % SiC	6721-3	—	586	1	101	—
108	591.4	(a)	8:1 (Th,U) ₂ O ₂	6171-105EDS	31.8	242	1/5	108	—
108	591.4	(a)	8:1 (Th,U) ₂ O ₂	6171-107EDS	31.8	242	1/5	108	—
108	501.4	(a)	8:1 (Th,U) ₂ O ₂	6171-115EDS	31.8	242	1/5	108	—
108	591.4	(a)	8:1 (Th,U) ₂ O ₂	6171-117EDS	31.8	242	1/5	108	—
108	591.4	(a)	8:1 (Th,U) ₂ O ₂	6171-EDS	31.8	242	1/5	108	—
109	592.7	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6252-21EDS	24.8	308	1/4	101	—
109	592.7	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-21E	24.8	308	3/4	101	—
110	594.0	(a)	1:0.07:0.07 (Th,Si,Al) ₂ O ₂	6291-111	—	502	1	101	—
111	595.3	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-23EDS	24.8	308	1/4	101	—
111	595.3	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-23E	24.8	308	3/4	101	—
112	596.6	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-25EDS	24.8	308	1/4	101	—
112	596.6	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-25E	24.8	308	3/4	101	—
113	597.8	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-27EDS	24.8	308	1/4	108	—
113	597.8	(a)	12:1:1:1.7 (Th,U,Si,Al) ₂ O ₂	6562-27E	24.8	308	3/4	108	—

(a) Advanced fuel design.

(b) Developmental reference design.

TABLE 2-2
GENERAL DESCRIPTION OF ADVANCED FUEL PARTICLES TESTED AS LOOSE
LOOSE PARTICLES IN CELLS III

Particle Batch Data Retrieval No.	Kernel			Coating										Total Coated Particle				Selective Dopants in BISO Particles(d) (wt %)	
	Type(a)	Mean Diameter (μm)	Density (g/cm^3)	Buffer		Isotropic IPyC		Carbide		Isotropic OPyC		Coating Rate ($\mu\text{m}/\text{min}$)	Mean Diameter (μm)	Density of EDS Batches(c) (g/cm^3)	Metal Loading				
				Type(b)	Thickness (μm)	Density (g/cm^3)	Thickness (μm)	Density (g/cm^3)	Type	Thickness (μm)	Density (g/cm^3)				Thickness (μm)	Density (g/cm^3)	Uranium ($\mu\text{g U-235}/\text{particle}$)		Thorium ($\mu\text{g Th}/\text{particle}$)
Fissile Particles (W8)																			
6551-37E	UC ₃ 2 ⁰ 0.42	338	2.37	TRISO	47	1.00	32	1.91	SiC	30.2	ND(e)	32	1.86	3.3	620	ND	45.8	NA(f)	NA
6551-51E		338	2.37	TRISO	48	0.96	34	1.96	ZrC	44.4	ND	40	ND	3.8	674	ND	45.8	NA	NA
6552-135E		338	2.37	BISO	47	0.93	34	1.95	NA	NA	NA	76	2.12	0.7	652	ND	45.8	NA	28 Si
6552-141E		338	2.37	BISO	48	0.96	34	1.96	NA	NA	NA	71	ND	2.4	648	ND	45.8	NA	41 Zr
6551-53E	UC ₄ 0 ⁰ 0.98	331	2.67	TRISO	50	0.94	34	1.93	ZrC	45.4	ND	36	1.85	5.9	666	ND	46.3	NA	NA
6551-55E		331	2.67	TRISO	50	0.94	34	1.92	SiC	35	ND	29	1.85	4.1	631	ND	46.3	NA	NA
6552-143E		331	2.67	BISO	50	0.94	36	1.93	NA	NA	NA	62	ND	2.0	626	ND	46.3	NA	NA
652-149E		331	2.67	BISO	47	0.96	36	1.92	NA	NA	NA	66	2.13	0.7	628	ND	46.3	NA	NA
6551-9E	UC ₃ 0 ⁰ 0.43	304	3.11	TRISO	48	1.11	32	1.91	ZrC	29.2	ND	31	ND	4.2	548	ND	45.8	NA	NA
6551-7E		304	3.11	TRISO	45	1.20	42.5	1.90	SiC	34.6	ND	36	1.742	5.6	620	ND	45.8	NA	NA
6552-131E		304	3.11	BISO	45	1.20	42.5	1.90	NA	NA	NA	73	2.09	0.7	625	ND	45.8	NA	28 Si
6552-123E		304	3.11	BISO	48	1.13	32	1.91	NA	NA	NA	55	ND	1.5	574	ND	45.8	NA	81 Zr
6551-67E	UC ₄ 3 ⁰ 1.6	327	3.40	TRISO	55	ND	36	ND	ZrC	46	ND	35	ND	4.4	671	ND	40.6	NA	NA
6551-65E		327	3.40	TRISO	55	ND	34	1.916	SiC	32	ND	32	1.803	4.9	633	ND	40.6	NA	NA
6562-7E		327	3.40	BISO	55	ND	36	ND	NA	NA	NA	70	ND	2.0	648	ND	40.6	NA	40 Zr
6562-3E		327	3.40	BISO	55	ND	34	1.916	NA	NA	NA	67	2.17	0.7	639	ND	40.6	NA	20 Si
Fertile Particles																			
6332-63EDS, g(c)	10:4:1 (Th,U)O ₂	450	9.7	BISO	137	0.95	25	1.97	NA	NA	NA	64	2.23	3.3	902	2.60	36.6	375	NA
6332-67EDS, g(c)		450	9.7	BISO	104	0.93	25	1.98	NA	NA	NA	70	2.29	3.5	848	2.721	36.6	375	NA
6332-55EDS, g(c)		450	9.7	BISO	137	0.95	25	1.97	NA	NA	NA	63	2.39	2.8	900	2.570	36.6	375	NA
6332-57E		450	9.7	BISO	104	0.93	25	1.98	NA	NA	NA	43	2.23	ND	793	ND	36.6	375	NA
6411-101EDS	8:2 (Th,U)O ₂	450	9.7	BISO	130	0.92	NA	NA	NA	NA	NA	91	1.87	4.6	892	2.655	36.6	375	NA
6171-105EDS, g(c)		400	9.7	BISO	97	1.21	17	1.86	NA	NA	NA	84	2.15	13.6	796	2.963	31.8	242	19 Si
6171-127EDS, g(c)		400	9.7	BISO	97	1.21	17	1.86	NA	NA	NA	81	2.20	10.1	710	2.819	31.8	242	25 Si
6171-115EDS, g(c)		400	9.7	BISO	97	1.21	17	1.86	NA	NA	NA	78	2.32	5.33	784	2.738	31.8	242	39 Si
6171-117EDS, g(c)	10:1 (Th,U)O ₂	400	9.7	BISO	97	1.21	17	1.86	NA	NA	NA	85	2.48	4.0	784	2.964	31.8	242	48 Si
6171-67EDS, g(c)		400	9.7	BISO	97	1.21	NA	NA	NA	NA	NA	80	1.78	11.24	784	2.940	31.8	242	NA
6411-101EDS, g(c)		450	9.86	BISO	130	0.91	NA	NA	NA	NA	NA	91	1.87	4.6	892	2.656	34.0	366	NA
6291-119		535	8.12	BISO	79	1.11	NA	NA	NA	NA	NA	74	1.88	ND	841	ND	NA	537	NA
6291-111	1:0.07:0.07 (Th, Si, Al)O ₂	518	9.05	BISO	85	1.04	NA	NA	NA	NA	NA	83	1.84	ND	868	ND	NA	502	NA
6562-21E		440	8.78	BISO	94	0.85	27	1.92	NA	NA	NA	64	ND	1.9	810	ND	24.8	308	37 Si
6562-21EDS		440	8.78	BISO	94	0.85	27	1.92	NA	NA	NA	64	ND	1.9	810	2.922	NA	NA	NA
6562-27EDS, g(c)		440	8.78	BISO	94	0.85	27	1.92	NA	NA	NA	64	ND	2.4	824	2.949	24.8	NA	NA
6562-21EDS, g(c)	ThO ₂	440	8.78	BISO	94	0.85	27	1.92	NA	NA	NA	71	ND	2.3	820	3.939	24.8	NA	NA
6562-27EDS, g(c)		440	8.78	BISO	94	0.85	27	1.92	NA	NA	NA	48	ND	1.2	778	3.538	24.8	308	NA
6562-97EDS, g(c)		482	9.76	BISO	79	1.04	35	1.96	NA	NA	NA	45	ND	ND	800	3.547	NA	503	NA
6562-101EDS, g(c)		482	9.76	BISO	79	1.04	35	1.96	NA	NA	NA	45	ND	ND	800	3.995	NA	503	NA
6551-149	1:0.03:0.03 (Th, Si, Al)O ₂	498	9.38	BISO	86	1.05	NA	NA	NA	NA	NA	78	1.80	ND	828	ND	NA	529	NA
6551-147		495	9.83	BISO	82	1.03	NA	NA	NA	NA	NA	78	1.82	ND	817	ND	NA	547	NA
6721-3		561	7.85	BISO	84	0.96	NA	NA	NA	NA	NA	82	1.79	ND	893	ND	NA	586	NA
6721-31EDS, g(c)		482	9.76	BISO	79	1.04	NA	NA	NA	NA	NA	(g)	(g)	(g)	(g)	3.382	NA	503	NA

- (a) Described on stoichiometric basis for W8 fissile particles and weight ratio basis for fertile particles.
 (b) TRISO denotes a laminated coating layer design including a carbide layer, and BISO is a comparable design without a carbide layer.
 (c) Batches designated EDS were density separated from the parent E batches; both E and EDS particles have comparable attributes.
 (d) Silicon or zirconium dopants were homogeneously deposited in the OPyC layers to inhibit fission product release.
 (e) ND = not determined.
 (f) NA = not applicable.
 (g) Not currently available.

TABLE 2-3
GENERAL DESCRIPTION OF DEVELOPMENTAL REFERENCE FUEL PARTICLES TESTED IN FUEL RODS AND
AS LOOSE PARTICLES IN CELLS I, II, AND III

Particle Batch Data Retrieval No.	Kernel		Coating															Total Coating Thick- ness (μ m)	Total Coated Particle						
			Type(a)	Buffer		Seal Coat	Isotopic IPyC			SiC		Isotopic OPyC							Mean Diameter (μ m)	Density (g/cm ³)	Metal Loading		Fission Gas Release(c)	Exposed Heavy Metal(d)	
	Thick- ness (μ m)	Density (g/cm ³)		OPTAP(b)	Thick- ness (μ m)		Density (g/cm ³)	Thick- ness (μ m)	Density (g/cm ³)	Coating Rate (μ m/min)	Microporosity (ml/kg OPyC)	Active Coating Gas Volume Fraction	OPTAP(b)	Uranium (wt %)	Thorium (wt %)	Uranium (g U/g U)	Thorium (g Th/g Th)								
																		Type			Mean Diameter (μ m)	Thick- ness (μ m)		Density (g/cm ³)	Thick- ness (μ m)
Fissile Particles(e)																									
6151-17-020	UC ₂ (VSM)	204	TRISO	100(f)	1.23	No	35	1.87	1.14	30	3.21	33	1.84	3.57	58.3	0.53	1.14	196	602	2.23	16.5	0.28	9.5 x 10 ⁻⁶	2.3 x 10 ⁻⁶	ND(g)
6157-02-010	UC ₃ O ₈ (WAR)	297	TRISO	38(f)	1.09	No	32	1.96	1.15	34	3.22	30	1.89	3.57	34.8	0.35	1.17	132	572	2.35	16.26	NA(h)	5.49 x 10 ⁻⁶	8.65 x 10 ⁻⁶	ND
Fertile Particles																									
6252-03-010	ThO ₂	529	TRISO	63(f)	1.04	No	30	1.86	1.09	33	3.22	37	1.82	5.3	59.6	0.18	1.13	161	861	3.71	NA	53.38	ND	ND	3.4 x 10 ⁻⁵
6252-04-010	ThO ₂	528	TRISO	63(f)	1.04	No	30	1.86	1.15	33	3.22	39	1.82	7.8	59.8	0.23	1.12	166	864	3.73	NA	53.47	ND	ND	3.0 x 10 ⁻⁵
6252-05-010	ThO ₂	519	TRISO	63(f)	1.04	No	30	1.86	1.18	33	3.22	49	1.84	4.5	30.3	0.15	1.12	176	874	3.59	NA	51.16	ND	ND	3.6 x 10 ⁻⁵
6542-27-010	ThO ₂	512	BISO	84	1.09	No	NA	NA	NA	NA	NA	81	1.86	4.6	22.7	0.29	1.09	164	836	3.45	NA	56.70	ND	ND	8.1 x 10 ⁻⁶
Inert Particles																									
6351-03-010	C	358	TRISO	ND	ND	ND	ND	ND	1.25	28	3.20	32	1.916	4.05	ND	0.35	1.16	ND	656	1.80	NA	NA	NA	NA	NA

(a) TRISO denotes a laminated coating layer, design with a SiC layer, and BISO is a comparable design without the SiC layer.

(b) Optical anisotropy factor (relative units). The following expression defines equivalence to RAP₀ measurement on Selbersdorf unit: $RAP_0 = 0.27 \times OPTAP + 0.733$.

(c) Release/birth rate for Kr-85m at 1100°C.

(d) Determined by leach test.

(e) Fissile kernels 93.15 wt % U-235 enriched.

(f) Calculated by subtracting the inner isotropic thickness from the measured buffer plus inner isotropic thickness.

(g) ND = not determined.

(h) NA = not applicable.

TABLE 2-4
AVERAGE PROPERTIES OF FUEL RODS FABRICATED FOR CF-4

Data Retrieval No. (7101-009-)	Rod Fabrication No. (RC Series)	Rod Capsule No.	Dimensions (cm)						Weight (g)	Particle Packing (%)	Matrix										Fuel Loading Uniformity (f)		Heavy Metal Content (b) (g)		Visual Exam of Surface (d)	Exposed Heavy Metal (g heavy metal/g heavy metal)	Fission Gas Release (1)	Fraction Defective SIC (2)		Push-Out Load Per Fuel Rod, CIP Processing (h) [lb (kg)]
			Green				Fired				Type	Filler (b) (%)			Apparent Density (g/cm ³)		Coke Content		Macro-porosity (e) (%)											
			Diam	Length	Diam	Length	Uniformity	Green (c)				Fired (c)	Yield (d)	Density (g/cm ³ rod)																
															Top Half	Bottom Half	Average													
O1	3391, 3395, 3402	1A	1.572	5.018	1.575	4.978	20.448	17.579	56	ND386	44	35	40	1.309	0.702	23	0.077	44	1.04	1.05	0.156	5.140	ND (1)	<1 x 10 ⁻⁶	3 x 10 ⁻⁶	ND	ND	222.4 (50)		
O2	3387-3389	1B	1.572	5.016	1.574	4.940	19.938	17.059	57	ND386	41	42	42	1.312	0.714	22	0.074	30	1.03	1.04	0.137	4.526	19	<1 x 10 ⁻⁶	4 x 10 ⁻⁶	ND	ND	NA (6)		
O3	3390-3392	1C	1.572	5.023	1.575	5.002	19.603	16.815	56	ND386	42	43	43	1.309	0.702	22	0.075	38	1.02	1.05	0.125	4.148	19	7 x 10 ⁻⁵	3 x 10 ⁻⁶	ND	ND	266.9 (60)		
D4	3396-3398	1CA	1.572	5.053	1.575	4.997	19.847	16.987	56	ND386	42	43	43	1.305	0.714	23	0.079	34	1.05	1.03	0.114	3.787	ND	<1 x 10 ⁻⁶	2 x 10 ⁻⁶	5.7 x 10 ⁻⁴	0	111.2 (25)		
D5	3399-3401	1CB	1.572	5.052	1.575	4.984	19.856	16.945	56	ND386	42	43	43	1.309	0.706	23	0.078	37	1.06	1.05	0.113	3.764	ND	<1 x 10 ⁻⁶	2 x 10 ⁻⁶	3.5 x 10 ⁻⁴	0	133.4 (30)		
D6	3403-3405	1CC	1.573	4.962	1.575	4.965	19.584	16.878	56	ND386	43	44	44	1.309	0.700	22	0.075	33	1.04	1.04	0.113	3.746	ND	1 x 10 ⁻⁶	2 x 10 ⁻⁶	5.5 x 10 ⁻⁴	1.8 x 10 ⁻⁴	133.4 (30)		

^(a) Calculated from fired rod and mean particle parameters.

^(b) Measurement made on one green rod.

^(c) Calculated from green rod and mean particle parameters.

^(d) Calculated by dividing the coke weight by the initial pitch plus additive weight.

^(e) Determined from metallographic cross section of one fired rod.

^(f) Determined by gamma counting both ends of rod and calculating ratio of maximum and mean values.

^(g) The grading guide specified in FRS-3 document No. 7.1.5.2 was used to evaluate fired rods. The maximum value is 20.

^(h) Determined by hydrolysis test. "C" denotes the amount is below the limit of detection of apparatus. If detectable quantity, the value is corrected for total conversion of ThO₂ to ThO₂.

⁽ⁱ⁾ Release/birth rate for Kr-85m at 1100°C.

^(j) Determined by burn-leach technique on fuel rods containing all TRISO-coated particles.

^(k) Maximum push-out load required to break fuel rod/crucible bond formed by curling-in-place.

^(l) ND = not determined.

^(m) NA = not applicable.

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3. IRRADIATION DESCRIPTION

3.1. IRRADIATION FACILITIES

Capsule GF-4 was irradiated in the 45G position of the SILOE reactor in Grenoble, France for 323 effective full power days (EFPD) OR 7750 h. The capsule began irradiation on April 8, 1975 and finished operation on July 26, 1976 after 15 cycles. GF-4 was irradiated in a TUMULT G device, which consisted of three independent cylindrical compartments, each ~150 mm long with a maximum usable diameter of 15.76 mm. Figure 3-1 is a schematic diagram which shows the individual components of the TUMULT G device and the specific SILOE core locations for GF-4. The temperature in each of the individual cells is monitored by a tungsten-sheath chromel-alumel (C/A) thermocouple and is regulated by a common neon-helium gas mixture. Figure 3-2 is a schematic diagram which shows the fuel rod and loose particle cell arrangements in the TUMULT G device and thermocouple and dosimeter placements for irradiation capsule GF-4. The flux dosimeters used for this capsule were copper, copper-cobalt (CuCo), and niobium. Figures 3-3 and 3-4 show measured thermal and fast flux axial profiles for SILOE core location 45G. Table 3-1 summarizes the irradiation test conditions for capsule GF-4. The next sections will also briefly describe the thermal and neutron fluence calculations; however, Ref. 3-1 includes more detailed information. Figures 3-5 through 3-10 plot a summary of the irradiation history (temperature, power, and fluence) versus time for capsule GF-4.

3.1.1. Thermal Measurements and Calculations

Thermal analysis of capsule GF-4 was based on in-pile temperature measurements using C/A thermocouples and computer codes which described heat fluxes and thermal resistances. This procedure enables fuel temperatures to be calculated as a function of capsule position and time during irradiation.

The controlling thermocouples were located at the inner containment wall and in the graphite bodies adjacent to the fuel compacts (refer to Fig. 3-2). These measured temperatures were below 1100°C and were monitored with C/A thermocouples in an Inconel sheath. Figure 3-2 shows that cells I and II each had three thermocouples in the graphite adjacent to the fuel compacts and two thermocouples in the inner containment wall. Cell III had four thermocouples in the graphite adjacent to the loose particle planchets and two thermocouples in the containment wall. C/A thermocouples are stable during irradiation and are not affected by alloy composition changes caused by transmutation (Ref. 3-2); consequently, no irradiation correction factors are required for calibration. The calculation of fuel temperatures was based on the following considerations:

1. Mean gamma and fission power in fuel for each cycle.
2. Linear decrease with fluence of radial gap between fuel rod and graphite crucible.
3. Fuel rod thermal conductivity of 0.087 W/cm-°C (5 Btu/h-ft-°F).
4. Material thermal properties for neon-helium gas mixtures and graphite consistent with irradiation capsules GF-1, -2, and -3 (Ref. 3-3).

The temperature at the fuel rod axis (T_a) was calculated as

$$T_a = T_s + \Delta T_{(k)} \quad , \quad (3-1)$$

where T_s is the surface temperature and $T_{(k)}$ is the calculated fuel rod radial temperature drop for a given power level. Figures 3-5 through 3-10 show calculated mean fuel surface and axial centerline temperatures for each cycle of capsule GF-4. As a point of reference, the fuel rod volume average temperature (T_v) is $(T_a - T_s)/2$.

The corresponding time-volume average temperature is defined as

$$T_{\bar{v},t} = 1/t \int_0^t T_{\bar{v}}(t) dt \quad , \quad (3-2)$$

where t is the irradiation time. Average temperatures are calculated in a like manner for the center of the fuel rod ($T_{a,t}$) and for the surface ($T_{s,t}$) of the fuel rod. Table 3-2 lists these time averaged temperatures for each of the irradiation cells in capsule GF-4. The two sigma uncertainty bound on reported temperatures is estimated at $\sim 100^\circ\text{C}$. This accounts for uncertainties associated with fuel rod thermal conductivity, fuel heat generation rate (fission and gamma), and fuel rod/graphite radial gap widths. Reference 3-4 discusses thermal analysis uncertainties in irradiation capsule tests in more detail.

3.1.2. Neutron Fluence Calculations

Neutron fluence in the SILOE reactor is measured in terms of graphite damage units which are then transposed to equivalent HTGR ($E > 29$ fJ) fluence units. This equivalence is established by utilizing the effect of irradiation-induced swelling in graphite, which is a result of atomic displacements due to fast neutron ($E > 29$ fJ) damage. Different reactor spectra can then be normalized against a constant amount of graphite damage. Reference 3-3 details the equivalent fission fluence for damage in graphite (ϕ_G) and shows that the SILOE and HTGR reactor spectra have the following fluence equivalence:

$$\phi(E > 161 \text{ fJ})_{\text{SILOE}} \times 2 = \phi_G \quad . \quad (3-3)$$

$$\phi(E > 29 \text{ fJ})_{\text{HTGR}} \times 1.25 = \phi_G \quad . \quad (3-4)$$

These equations imply the following fluence equivalence:

$$\phi(E > 161 \text{ fJ})_{\text{SILOE}} \times 1.6 = \phi(E > 29 \text{ fJ})_{\text{HTGR}} \quad . \quad (3-5)$$

Figures 3-3 and 3-4 plot the axial profiles for thermal and fast flux and fluence measured for the SILOE reactor. The flux dosimeters used were copper, copper-cobalt (CuCo), and niobium. Equation 3-5 above converted a measured SILOE fluence ($E > 161$ fJ) to a corresponding HTGR fluence ($E > 29$ fJ).

3.1.3. Heavy Metal Burnup Calculations

Heavy metal burnup is a measure of the total fissions which have occurred in a fuel sample during irradiation and is proportional to the amount of fission products generated. The heavy metal burnup of each fuel sample irradiated in GF-4 was calculated using the APPOLLO computer code (i.e., a code which subdivides thermal and epithermal neutron energy levels into 99 separate groups) (Refs. 3-3 and 3-5).

Required inputs are reactor power history, average thermal fluences, initial uranium and thorium loadings, macroscopic cross sections, and fission yields. Isotopic fission yields were obtained from a compilation of M. E. Meek and B. F. Rider (Ref. 3-6). The cross sections used in the program were normalized relative to the thermal fluence measured by cobalt dosimetry and are listed as follows:

<u>Isotope</u>	<u>Fission Cross Section (b)</u>	<u>Capture Cross Section (b)</u>
U-235	602	127
U-233	695	72
Th-232	--	22

The APPOLLO code calculated the burnup for both U-235 and thorium (U-233) at the end of irradiation. However, because of the complexity and computer run time, a simplified computer code (BUAPO) calculated burnups at the end of each irradiation cycle (Ref. 3-1). This simplified code provided a basis for noble fission gas release [release/birth rate (R/B)]

measurements during each cycle. The simplified code could calculate isotopic concentrations and burnups to within $\pm 5\%$ of the more exact APPOLLO calculations (Ref. 3-3).

3.1.4. Fission Gas Release Measurements

The inert sweep gas (neon-helium) used to control fuel temperatures in each cell was sampled approximately every 10 days to monitor fission gas release from fuel specimens during irradiation. Seven gaseous fission product isotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138) were measured. These isotopes were of primary interest because they constitute the main source of gaseous activity released from failed fuel and, thereby, can directly determine fuel performance during irradiation. R/B determinations were made by allowing gaseous activity to reach equilibrium through a standard tube prior to sampling. After equilibration, the gas sample was withdrawn into an evacuated vial and gamma counted using a germanium (lithium) [Ge(Li)] detector coupled with a multichannel analyzer. A standard decay time permitted the gas samples to be transported to the counting room. Absolute quantities of all isotopes were calculated to determine the amount of gas released at the surface of the samples. This was accomplished by knowing the flow rates of the sweep gas from each cell to the sampling station. In-pile fission gas release measurements provide a direct measure of fuel performance; specifically, gas release due to exposed uranium and/or thorium contamination and gas release due to fuel failure are measured. The following equation defines individual contributors to R/B:

$$\begin{aligned}
 R/B = & \left(\frac{U_c^F U-235 Y_{U-235} + T_c^F U-233 Y_{U-233}}{F_{U-235} Y_{U-235} + F_{U-233} Y_{U-233}} \right)_{\text{contamination}} \\
 & + \left(\frac{X_{\text{fissile}} R_{\text{fissile}}^F U-235 Y_{U-235} + X_{\text{fertile}} R_{\text{fertile}}^F U-233 Y_{U-233}}{F_{U-235} Y_{U-235} + F_{U-233} Y_{U-233}} \right)_{\text{fuel failure}}
 \end{aligned} \tag{3-6}$$

where R/B = total release per birth,

U_c = beginning of life (BOL) uranium contamination measured by TRIGA,

T_c = BOL thorium contamination measured on companion rods using LINAC or thorium hydrolysis test,

F_{U-235} = fissions from U-235,

F_{U-233} = fissions from U-233,

Y_{U-235} = yield of a particular isotope per U-235 fission
(Y_{U-235} for Kr-85m = 0.013),

Y_{U-233} = yield of a particular isotope per U-233 fission
(Y_{U-233} for Kr-85m = 0.022),

$X_{fissile}$ = fissile particle failure fraction,

$X_{fertile}$ = fertile particle failure fraction,

$R_{fissile}$ = fractional release of a certain isotope per failed TRISO particle,

$R_{fertile}$ = fractional release of a certain isotope per failed BISO particle.

The following points are worth noting regarding Eq. 3-6 and in-pile R/B measurements:

1. At BOL and in the absence of fuel failure, the contamination term dominates R/B measurements.
2. At BOL, the fraction of fissions in fissile fuel is much greater than in fertile fuel ($F_{U-235} > F_{U-233}$) and reverses at end of life (EOL) ($F_{U-233} > F_{U-235}$). This implies that either fissile contamination and/or failure dominates the R/B measurement during the early part of irradiation and that fertile contamination and/or failure dominates R/B near EOL.

3. Short-lived isotopes, such as Kr-85m, measure instantaneous fission gas release and do not account for time-dependent diffusive release. However, long-lived isotopes, such as Xe-133, account for the cumulative time-dependent diffusive release of gases occurring in surface contamination or in failed fuel.

Figure 3-11 plots the resulting in-pile Kr-85m R/B measurements for cells I and II. Loose particle cell measurements are not included on Fig. 3-11 because of the wide variety of fuel types tested in this cell and the inability to correlate R/B release with a particular fuel type. In addition, Figs. 3-12 and 3-13 plot in-pile R/B measurements on the long-lived isotopes Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, and Xe-138 for cells I and II, respectively.

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Fig. 3-1. Schematic diagram of TUMULT G device and SILOE reactor core penetrations

Fig. 3-2. Schematic diagram of thermocouple and dosimeter placements in capsule GF-4

Fig. 3-3. Axial flux profiles for position 45G in SILOE reactor

Fig. 3-4. Axial fluence profiles for position 45G in SILOE reactor

Fig. 3-5. Irradiation history of GF-4, rod 1A

Fig. 3-6. Irradiation history of GF-4, rod 1B

Fig. 3-7. Irradiation history of GF-4, rod IC

Fig. 3-8. Irradiation history of GF-4, rods IIA, IIB, and IIC

Fig. 3-9. Irradiation history of GF-4, cell IIIA

Fig. 3-10. Irradiation history of CF-4, cell IIIB

Fig. 3-11. Kr-85m R/B measurements versus fluence for cells I and II

Fig. 3-12. Isotope R/B measurements versus fluence for cell I

Fig. 3-13. Isotope R/B measurements versus fluence for cell II

TABLE 3-1
SUMMARY DESCRIPTION OF IRRADIATION CONDITIONS FOR CAPSULE GF-4

Cell No.	Designation	Test Variable	Fuel Particles	Irradiation Conditions ^(a)				
				Thermal Flux (10^{14} n/cm ² -s)	Fluence [(10^{25} n/m ²) (E > 29 fJ) _{HTGR}]	Particle Burnup (% FIMA)		Time-Volume Average Temperature (°C)
						Fissile	Fertile	
I (fuel rods)	IA	Cured-in-place	WAR UC ₃ O _{0.6}	1.04	7.2	69.1	2.5	1010
	IB	Cured in Al ₂ O ₃	TRISO/ThO ₂ BISO	1.36	8.0	73.5	3.4	1020
	IC	Cured in H-451 graphite		1.53	5.6	76.6	4.5	1120
II (fuel rods)	IIA	Cured-in-place	VSM UC ₂ TRISO	1.58	8.9	77.2	4.6	
	IIB	Cured-in-place	ThO ₂ TRISO	1.62	9.5	77.2	4.8	1020
	IIC	Cured-in-place		1.61	9.4	77.2	4.6	1020
III (loose particles; 113 trays)	IIIA (top)	Different particle types	ThO ₂ BISO	1.27	7.5	72.4	3.3	1120
	IIIB (bottom)		WAR UC _x O _y TRISO	0.92	5.7	65.4	2.0	1070
			VSM UC ₂ TRISO					

(a) Irradiation began April 8, 1975 and ended July 26, 1976 for a total of 7750 h and 15 cycles.

TABLE 3-2
IRRADIATION TEMPERATURES OF CAPSULE GF-4

Cell No.	Fuel Description	Fuel Centerline Temperature (°C)			Volume Average Temperature (°C)			Fuel Surface Temperature (°C)			Average Fuel Surface Temperature Gradient (°C/cm)
		Maximum	Time Average	EOL	Maximum	Time Average	EOL	Maximum	Time Average	EOL	
IA	WAR UC ₃ O _{0.6}	1130	1080	1080	1060	1010	1000	990	930	930	190
IB	TRISO/ThO ₂	1120	1100	1100	1040	1020	1020	960	940	940	205
IC	BISO	1260	1200	1160	1180	1120	1070	1090	1020	990	230
IIA	VSM UC ₂ TRISO/										
IIB	ThO ₂ TRISO	1130	1110	1130	1050	1020	1050	960	940	960	215
IIC											
IIIA	WAR UC _x O _y TRISO	1190	1140	1180	1170	1120	1160	1130	1110	1130	--
IIIB	VSM UC ₂ TRISO										
	ThO ₂ BISO	1130	1080	1130	1110	1070	1110	1090	1050	1090	--

4. POSTIRRADIATION EXAMINATION

4.1. CAPSULE DISASSEMBLY AND VISUAL EXAMINATION

Figure 4-1 shows neutronography and radiography of capsule GF-4. Neutronography was performed at EOL, and radiography was done when the capsule arrived at the hot cell. The purpose of both examinations was to determine whether the capsule was damaged in transit from the reactor to the hot cell, and the French concluded that the capsule was not damaged during the move to the hot cell. Reference 4-1 presents a detailed description of capsule disassembly, fuel rod and graphite metrology, and the loose particle visual examination.

4.1.1. Graphite Sleeves

The capsule was filled with argon and sectioned 90 cm from the base. Immediately after sectioning, a plug was placed into the capsule to prevent air from coming into contact with the samples. The fuel rods in their graphite sheaths were then removed from the capsule. Figures 4-2 and 4-3 show the rod ends and their sheaths. All of the fuel rods were tight in their sheaths, except IB which moved freely.

4.1.2. Fuel Rods

The fuel rods were pushed out of their sheaths, except for IIC, which was later sectioned and its graphite fuel rod gap measured for thermal conductivity calculations. Table 4-1 lists the extraction forces, radial gaps, and irradiation conditions for each rod. Maximum push out force and radial gap have no apparent correlation; however, radial gap measurements are averages and do not reflect minimum clearances between the fuel rod and graphite sleeve.

After disassembly, the general irradiation performance and integrity of the loose particle samples and fuel rod compacts were assessed by visual examination. The criteria for judging fuel rod integrity after irradiation are that fuel rods remain intact with minimal cracking and experience negligible particle debonding. The examination was conducted with a stereomicroscope at a magnification range of approximately 10X to 20X (Ref. 4-1). Figures 4-4 and 4-5 show photographs of each of the irradiated rods. In general, all rods exhibited good to excellent structural integrity with no irradiation-induced fractures evident. The rods exhibited superficial microcracks on the surface and were fragile; also, the rod ends had a tendency to chip or crumble. However, all of the visible damage to the rods, including the breakage of IB, was caused by handling after the rods were removed from their graphite sheaths and not during irradiation.

4.1.3. Loose Particles

Cell III contained two sections with a total of 113 loose particle trays. The sections were removed from the capsule without any problem. Each planchet was then taken off the support rod and examined individually to determine visual failure levels.

The developmental reference fuel designs which were used in the fuel rods were also irradiated in loose particle trays. As shown in Table 4-2, the developmental reference particles exhibited no pressure vessel failure and 0% to 2% OPyC failure. The WAR UC₂ TRISO and the ThO₂ BISO particles showed no visual failure.

Particles that appear to have intact coatings under visual examination may have failed coatings which escaped detection. To more accurately determine particle failure, autoradiography was performed on the empty particle holders after the visual examination was completed. In most cases, the autoradiography confirmed the visual examination results. This was the case for the UC₂ WAR TRISO particles from particle holder 6; Fig. 4-6 shows the particles in the holder, and Fig. 4-7 shows the autoradiograph. Figures

4-8 and 4-9 show the visual appearance of UC_2 VSM TRISO particles in plate 22 and the autoradiograph of the plate. The visual exam results showed only OPyC failure; however, the autoradiograph clearly shows one of the particles released fission products; therefore, the SiC must have failed during irradiation.

The advanced fuel designs tested in GF-4 can be classified in two categories:

1. Reference coatings on resin kernels with varying oxygen/uranium (O/U) ratios.
2. Oxide kernel particles.

4.1.3.1. Resin Kernel Particles. The kernels in this group of particles have four O/U ratios: 0.42, 0.45, 0.98, 1.6. The coatings are either TRISO (SiC or ZrC) or BISO (with the OPyC doped with either silicon or zirconium). Table 4-3 shows the visual failure levels for these particle batches. TRISO-coated particles exhibited no pressure vessel or OPyC failure. In addition, autoradiography confirmed zero pressure vessel failure for SiC TRISO particles. In contrast, the BISO particles exhibited pressure vessel failure between 0% and 26% and OPyC failure levels between 0% and 70%. The failure levels of the BISO particles did not appear to be correlated to (1) the concentration of silicon or zirconium in the OPyC, (2) the density of the OPyC, (3) the OPyC coating rate, or (4) the O/U ratio of the kernel.

4.1.3.2. BISO-Coated Oxide Particles. Oxide kernel particle types tested in GF-4 were (1) mixed oxide (8, 10.4 Th,U) O_2 kernels (some kernel types were doped with SiO_2 or Al_2O_3 and OPyC layers with silicon or zirconium), (2) ThO_2 kernels doped with SiO_2 , Al_2O_3 , or SiC and OPyC layers doped with zirconium or silicon.

The undoped mixed oxide kernels with OPyC layers containing 0 to 48 wt % SiC showed a partial scaling (delamination) of OPyC layers for the

higher weight percent doped particles (refer to Table 4-4 and Fig. 4-10). Less scaling was seen in particle samples that were exposed to a lower fluence. Autoradiography (Fig. 4-11) showed that fission product release was not necessarily associated with particles that exhibited scaling of the OPyC. Metallography (Fig. 4-12) showed that OPyC layer delaminations during irradiation were attributed to structural inhomogeneity in the SiC-doped OPyC layer (namely, concentric ring striations in the OPyC layers of alternating high and low silicon concentrations).

Four batches were tested that had doped mixed oxide kernels and doped OPyC layers (refer to Table 4-4). The two batches with silicon-BISO coatings showed an equivalent per cent OPyC scaling as the undoped mixed oxide particles. However, the zirconium-BISO coatings were completely intact.

The three batches of undoped ThO_2 kernels exhibited OPyC failures between 0% and 27%: the undoped sample had no failure, the silicon-BISO had a 0.7% failure, and the zirconium-BISO had a 27% failure.

Zero failure was observed in the five batches of doped ThO_2 kernels with reference BISO coatings (refer to Table 4-4).

4.2. METROLOGY

4.2.1. Graphite Sleeves

Postirradiation dimensional change measurements were made on TS-1240 and H-451 graphite sleeves using a calibrated dial gauge micrometer that had an accuracy of $\pm 10^{-4}$ mm. Table 4-5 lists the measured axial and radial dimensional changes for the near-isotropic graphic sleeves. Figures 4-13 (Ref. 4-2) and 4-14 compare these values with predicted values. Predicted and observed dimensional changes for H-451 graphite sleeves are in excellent agreement. Currently, the irradiation data base on TS-1240 graphite is limited, and predictions are restricted to a fast fluence of $< 5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}; however, the TS-1240 irradiated in GF-4 is consistent with

extrapolation of the existing data base. Figures 4-13 and 4-14 indicate that TS-1240 graphite exhibits less axial and radial dimensional changes compared to H-451.

4.2.2. Fuel Rods

Postirradiation measurements on fuel rods were made with a calibrated dial gauge micrometer ($\pm 10^{-4}$ mm accuracy). Three diametral measurements (one on each end and one in the middle) and one length measurement were made on each fuel rod. The averages of each set of diameter and length measurements before and after irradiation calculated the percent dimensional change due to irradiation. In addition, the volumetric dimensional change of fuel rods was calculated on the basis of a cylindrical fuel rod geometry as follows:

$$\Delta V/V = 2 \Delta D/D + \Delta l/l \quad , \quad (4-1)$$

where $\Delta V/V$, $\Delta D/D$, and $\Delta l/l$ are incremental volume, diameter, and length changes due to irradiation, respectively. Table 4-6 lists values for $\Delta V/V$, $\Delta D/D$, and $\Delta l/l$ and irradiation exposure conditions for each fuel rod tested in capsules GF-1, -2, and -3. Also, the last column in Table 4-6 lists fuel rod anisotropy, which is arbitrarily defined as $(\Delta l/l - \Delta D/D)$. Figures 4-15 through 4-18 plot volumetric and length changes versus fast fluence for GF-4 fuel rods. These figures show separate plots for measured and predicted volumetric and dimensional changes for rods containing either all TRISO-coated fuel or TRISO/BISO-coated fuel.

Fuel rod volumetric and dimensional changes were predicted using the model described in Ref. 4-3. The model assumes that the fuel rod structure is described as follows: point-to-point contact of the particle phases with an interconnecting matrix network through the interstitial voids. The model accounts for the volume fraction and irradiation-induced volumetric changes of all components in a fuel rod (i.e., fuel particles, shim, matrix, and porosity). Figures 4-15 and 4-16 present a comparative evaluation of

predicted and measured volumetric changes for fuel rods irradiated in cells I and II. Figure 4-15 shows that the observed volumetric changes for fuel rods containing TRISO/BISO-coated fuel are 2% to 5% less than predicted values. This discrepancy is attributed to the relatively poor predictability between calculated and observed BISO-coated particle dimensional changes. Specifically, calculated BISO-coated volumetric changes are typically 2% to 6% less than observed for fluences between 7 and $9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR} (Ref. 4-3). This trend is consistent with the underprediction of fuel rod volumetric changes shown in Fig. 4-15 when fuel rod volumetric changes are considered to be proportional to the volume fraction times coated particles in a rod. Refinements are currently being made in the BISO-coated particle stress model (Ref. 4-4) to account for the effects of a changing creep rate and OPyC anisotropy on irradiation-induced dimensional change. These modifications are expected to improve the degree of predictability for BISO particle volumetric changes and, in turn, for fuel rods containing BISO-coated particles.

In contrast, Fig. 4-16 shows excellent agreement between predicted and measured volumetric changes for cell II rods containing TRISO/TRISO-coated fuel. This is attributed to the improved predictability of TRISO- compared to BISO-coated particle dimensional change. Figure 4-16 shows fuel rod expansion (0.07 vol %) at a fluence of $9.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}. This behavior is indicative of matrix expansion, since the other components in a fuel rod (TRISO fuel and shim) undergo volumetric contractions of approximately -3% to -4% for fluences of $9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR} at 1100°C or less.

Figures 4-17 and 4-18 plot predicted and measured diametral and length changes for GF-4 fuel rods. Predicted diameter and length dimensional changes in GF-4 fuel rods are defined on the following basis:

$$\Delta l/l = 1/3 (\Delta V/V + 2\Delta\epsilon) \quad , \quad (4-2)$$

$$\Delta D/D = 1/3 (\Delta V/V - \Delta\epsilon) \quad , \quad (4-3)$$

where $\Delta l/l$, $\Delta D/D$, and $\Delta V/V$ are length, diameter, and volume changes, respectively, and $\Delta \epsilon$ is strain anisotropy, which is defined as $(\Delta l/l - \Delta D/D)$. If a fuel rod exhibits isotropic behavior, $\Delta \epsilon = 0$ and Eqs. 4-2 and 4-3 reduce to $\Delta l/l = \Delta D/D = 1/3 \Delta V/V$. However, fuel rods containing matrix and shim particles exhibit anisotropic behavior, which is described by the $\Delta \epsilon$ versus fast fluence relationship plotted in Fig. 4-19 (solid line). This relationship and predicted volumetric fuel rod changes were used with Eqs. 4-2 and 4-3 to predict length and diameter changes in Figs. 4-17 and 4-18. The predicted diametral and length dimensional changes for rods containing TRISO/BISO-coated fuel (Fig. 4-17) do not agree with observed values. This is a consequence of the poor volumetric change ($\Delta V/V$) predictability and the corresponding effect on diameter ($\Delta D/D$) and length ($\Delta l/l$) changes in Eqs. 4-2 and 4-3. Fuel rods containing TRISO/TRISO-coated fuel particles show excellent agreement between predicted and observed diametral and length changes (refer to Fig. 4-18).

Figure 4-19 plots the expected strain anisotropy for GF-4 fuel rods, based on the empirical model presented in Ref. 4-3. Superimposed on this figure is the strain anisotropy data observed for GF-4 fuel rods. The GF-4 data are bounded by the 90% confidence bounds of the model. Figure 4-19 shows that fuel rod strain anisotropy exhibits the following trends:

1. A positive correlation with fast fluence exposure (i.e., a progressive increase in strain anisotropy).
2. A preferential bias between diametral and length changes, with $\Delta \epsilon$ being consistently greater than unity.

These observations are consistent with anisotropy observations in GF-1, -2, and -3 fuel rods (Ref. 4-5) and may be rationalized on the basis of systematic preferential alignment of anisotropic components within a fuel rod. The most plausible explanation for the strain anisotropy trends evident in Fig. 4-19 is that the preferred alignment of graphite shim and/or filler particle basal planes perpendicular to the rod axial direction

resulted in expansion or less shrinkage in the axial direction compared with the radial direction. This has been partially substantiated by empirical results presented in Ref. 4-3.

4.3. FUEL ROD THERMAL CONDUCTIVITY MEASUREMENTS

The relative thermal conductivity of fuel rod IIB was measured after irradiation at 600°, 700°, and 800°C. The relative thermal conductivity measurements were converted to absolute values by comparing all measurements to graphite standards of known thermal conductivities. This technique is referred to as a LECI thermal conductivity measurement and is documented in Ref. 4-6. The implicit assumption in applying this technique is that differences in thermal conductivity between the sample and the standard do not result in significant changes in the response of thermal flux meters in the apparatus. This condition is not generally met; consequently, large uncertainties in the measured irradiated thermal conductivities exist. Preirradiation thermal conductivity measurements were made using a Kolraush technique (Ref. 4-6) on a companion IIB fuel rod. The Kolraush technique measures thermal conductivity directly; consequently, it is more accurate and can be considered an absolute thermal conductivity measurement compared to the LECI measurement. Table 4-7 lists pre-and postirradiation thermal conductivity measurements on fuel rod IIB. The general trends evident from this table are summarized as follows:

1. A 22% difference in thermal conductivity between each half of the preirradiated fuel rod. This difference is attributed to structural inhomogeneities in the fuel rod (e.g., matrix macroporosity variations). Macroporosity variations between 25% and 35% have been measured along the axial direction of HTGR fuel rods. In addition, the correlation between macroporosity and thermal conductivity in Ref. 4-7 indicates that a 10% (absolute) variation in macroporosity is sufficient to cause a 20% difference in thermal conductivities.

2. A 29% to 51% decrease in the irradiated compared to unirradiated thermal conductivity. This decrease is reasonably consistent with the 42% to 64% decrease reported for GF-1 and GF-2 fuel rods (Ref. 4-5).

A further point worth noting is that the friable character of irradiated fuel rods (surface microcracks) made plane or parallel surfaces impossible to obtain on samples which were needed for the measuring device. This experimental difficulty, combined with the inaccuracies associated with a comparative standard measurement in the LECI technique, results in large uncertainties in the reported thermal conductivities in Table 4-7. However, an experimental program is under way to recalibrate the LECI thermal conductivity device to define a more accurate basis for irradiated fuel rod conductivity measurements. Specifically, standards are being used on the LECI device which bracket the range of expected fuel rod conductivities and minimize experimental inaccuracies when balancing heat fluxes between the standard and the fuel rod sample.

4.4. FISSION GAS RELEASE MEASUREMENTS

4.4.1. Reirradiated Fuel Rods in Cells I and II

Postirradiation fission gas release measurements were made on the six fuel rods from cells I and II. The rods were reirradiated in their graphite sleeves about four months after the end of irradiation in the SILOE reactor. Reirradiation in the graphite sleeves was necessary to minimize particle damage during handling and to eliminate the potential for fuel damage during rod extraction from the graphite. Each fuel compact was reirradiated at 820°C, and the activity values for Kr-85m, Kr-88, Kr-87, Xe-138, Xe-137, and Kr-89 were measured and converted to R/B values. Figures 4-20 and 4-21 show the reirradiated R/B values for each isotope measured at 820°C. In addition, for comparative purposes, Figs. 4-20 and 4-21 show the in-pile EOL R/B values for cells I and II corrected for temperature. The progressive decrease in R/B with decreasing half-life is related to the rapid decay of

the isotope prior to diffusive release through the kernel and coatings. Comparing the reirradiated gaseous isotope measurements with the in-pile EOL values shows that in-pile EOL fractional gaseous release values at 820°C for cells I and II are between one third to an order of magnitude less than the average fuel rod fractional release during reirradiation.

This implies that additional fuel hydrolysis and/or failure occurred between the end of irradiation and reirradiation as a result of handling or exposure to ambient conditions.

4.4.2. Fission Gas Release from Failed WAR Particles

Fuel particle failure levels may be determined from in-pile or postirradiation fission gas release measurements if the fractional release of fission gases from failed fuel $(R/B)_f$ is known. A determination of $(R/B)_f$ for WAR particles was made by laser failing ~10 WAR particles irradiated in cell III (planchet No. 6) of GF-4. All particle handling operations were conducted in an inert atmosphere to prevent hydrolysis, and particles were transferred to a reirradiation furnace and isochronally heat treated at different temperatures. Three successive periods of reirradiation were conducted at heat treatment times of 148, 72, and 48 h. Figure 4-22 summarizes the results of the isochronal heat treatment studies on laser-failed WAR particles. The following trends are evident from this figure:

1. During the first irradiation period (148-h heat treatment), the release rate is high at 680°C (20%) and decreases by a factor of three when the temperature increases to 1000°C. As the heat treatment temperature increases above 1000°C, the fractional gaseous release continuously increases to a maximum value of 31% at 1300°C.
2. During the second and third reirradiation heat treatments, the fractional release values between 680° and 1000°C vary from over

an order of magnitude less at 680°C to approximately one-half of the values measured during the first heat treatment.

The trends observed in Fig. 4-22 during the first heat treatment phase can be rationalized on the basis of increased densification of the WAR kernel structure with temperature. The rate of kernel densification dominates between 680° and 1000°C and results in a tighter structure and reduced gaseous release due to a more tortuous diffusion bath. However, for heat treatment temperatures between 1000° and 1300°C, the exponential increase in gaseous diffusivity within the kernel structure offsets kernel densification. Finally, the reduction in fractional gas release measured during the second and third heat treatment phases is a result of kernel densification which occurred during the first heat treatment phase.

Superimposed on Fig. 4-22 is a plot (dashed line) of the expected (Ref. 4-8) Kr-85m R/B versus temperature for failed unhydrolyzed VSM UC₂ kernels. Kr-85m (R/B)_f for VSM UC₂ fuel is <1% up to 1300°C, and the low gaseous release compared to the WAR UC_xO_y kernels is attributed to the dense melted UC₂ kernel structure. The high fission gas release values measured on failed WAR UC_xO_y fuel were instrumental in eliminating this kernel type from developmental reference HTGR fuel systems.

4.4.3. EOL Fuel Failure Determinations

Fuel failure at EOL may be determined by using Kr-85m R/B measurements and accounting for the fractional gaseous release from failed fuel. In-pile EOL Kr-85m R/B measurements are preferred over reirradiated out-of-pile measurements because of the additional fuel failure and/or hydrolysis which occurred during handling (refer to Section 4.4.1). Equation 3-6 determined fuel failure by assuming equivalent fissile and fertile particle failure fractions. Without this assumption, an infinite number of allowable fissile and fertile particle failure fractions would satisfy a given Kr-85m R/B measurement. At EOL, the fraction of fissions in fertile and fissile fuel

are 0.88 and 0.12, respectively (Ref. 4-9). This allows the following approximation to Eq. 3-6 to be made:

$$(R/B)_{EOL} \cong U_C \times \bar{F} + T_C + X R_{fissile} \times \bar{F} + R_{fertile} \quad , \quad (4-4)$$

where $(R/B)_{EOL}$ = EOL R/B for given isotope,

U_C = BOL uranium contamination (2×10^{-6} ; Ref. 4-9),

T_C = BOL thorium contamination ($\sim 3 \times 10^{-6}$; Ref. 4-9),

X = fissile and fertile particle failure fraction,

$R_{fissile}$ = fractional release per failed fissile particle
(refer to Fig. 4-22),

$R_{fertile}$ = fractional release per failed fertile particle
(dependent on burnup and temperature; Ref. 4-8),

$\bar{F} = \left(F_{U-235} \times Y_{U-235} / F_{U-233} \times Y_{U-233} \right) = 0.077$
for GF-4 at EOL.

The calculated fuel failure fractions based on in-pile Kr-85m R/B measured at EOL and using Eq. 4-4 are $\sim 0.12\%$ for cell I and $\sim 0.08\%$ for cell II. These calculated values are well below the allowable pressure vessel failure design limit of 0.5% for fuel exposed to peak conditions; 1250°C, $8 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) HTGR, and 78% FIMA for fissile fuel and 7.5% FIMA for fertile fuel (Ref. 4-10).

4.5. METALLOGRAPHIC EXAMINATION

4.5.1. Fuel Rods

Metallographic examination was performed on rods IC and IIC to characterize the irradiation-induced changes in the fuel coatings, the fuel kernel, and the fuel rod matrix. Figures 4-23 through 4-28 are representative metallographic cross sections of fuel rods IC and IIC.

The metallographic cross sections in Figs. 4-12 and 4-15 are macroscopic views which show four predominant fuel rod components (fuel particles, graphite shim, matrix, and macroporosity*). The spherical particles are either TRISO- or BISO-coated fuel, and the shim is a near-isotropic-equiaxed graphite particle nominally 800 mm in diameter. Matrix and porosity phases are interconnecting and distinguished as a mottled (black and white) area and a light grey phase which is the impregnated metallographic mount resin in preexisting voids. The macroporosity* for irradiated fuel rod types IC and IIC is 41% and 32%, respectively, which is comparable to preirradiation values. Figure 4-24 is a metallographic cross section of fissile particles irradiated in fuel rod IC. These particles have WAR kernels ($UC_3O_{0.6}$) and the irradiated structure is characterized as follows:

1. The uranium in the kernel (white phase) occupies only a small fraction of the original kernel volume. Furthermore, the kernel has apparently dispersed and segregated into uranium-rich (white modules) and carbon-rich (grey phase) regions (refer to Fig. 4-24d).
2. The kernel apparently reacted with the buffer layer, which results in anisotropic carbon-rich deposits at the kernel perimeter (refer to Fig. 4-24b).
3. IPyC, SiC, and OPyC layers appear intact and unaffected by irradiation.
4. Microprobe analysis (refer to Section 4.6) of small white phases at the SiC-IPyC interface shows the presence of lanthanide and palladium fission products (Fig. 4-24d); however, degradation of the SiC was not observed.

* Macroporosity is defined as void percent in the matrix phase which is $>50 \mu m$.

The above structural observations on irradiated WAR ($\text{UC}_3\text{O}_{0.6}$) particles in GF-4 are consistent with other irradiation experience (Ref. 4-11). Also, the WAR kernel stoichiometry tested in GF-4 ($\text{O/U} \approx 0.6$) is the lowest O/U ratio limit consistent with retention of lanthanide fission products in the kernel structure (Ref. 4-7). At O/U ratios less than ~ 0.6 , oxygen is insufficient to completely react with lanthanide fission products to form stable-immobile oxides; consequently, below this range, lanthanide fission product/SiC attack is possible.

BISO-coated particles in fuel rod IC appeared to be in excellent condition after irradiation (refer to Figs. 4-25a and b). The polarized light photomicrograph shows an anisotropic OPyC structure; however, no OPyC failure was detected in this particle type (6542-27-010). In addition, Fig. 4-25 shows representative cross sections of the matrix and shim components in GF-4 fuel rods. The structures of these components are comparable to the preirradiated structures.

Figure 4-26 is a metallographic cross section of fuel rod IIC. All of the particles in this fuel rod had TRISO coatings and are distinguished by their characteristic kernel appearance. The fissile kernels are white with a large amount of porosity, fertile kernels appear grey and are the largest kernel type, and inert particles have bright white kernels.

The cross section of fuel rod IIC contained only nine fissile particles, and Fig. 4-27 shows a representative fissile particle. The IPyC is partially separated from the SiC, and polarized photomicrographs show substantial lanthanide and palladium fission product accumulations at the inner SiC surface. Figure 4-27c focuses on $\sim 7 \mu\text{m}$ of lanthanide/palladium-SiC corrosion which has occurred. The buffer and the IPyC are anisotropic, but do not show irradiation-induced cracking. No OPyC failure was noted, and all microcracks observed in SiC layers were attributed to polishing artifacts.

Forty fertile particles were observed in fuel rod IIC cross section. Figure 4-28 shows two fertile particles. The buffer and IPyC usually separated, and 35% of the particles had fractured buffers, as shown in Fig. 4-28b. One OPyC and SiC failure and two OPyC failures were observed.

The cured-in-place fuel rod/graphite sheath interface shown in Fig. 4-28d is $\sim 25 \mu\text{m}$.

4.5.2. Loose Particle Metallographic Evaluation

Table 4-8 lists the loose particle batches which were examined metallographically. This section summarizes the structural characterization for each of the batches listed in Table 4-8.

4.5.2.1. TRISO-Coated UC₂ VSM Fuel. Fifty particles of TRISO-coated UC₂ VSM fuel (6151-17-020) irradiated in planchet No. 22 were examined metallographically (refer to Fig. 4-29). This was the same batch as used in rod IIC, and similar structural features are present in Figs. 4-27 and 4-29 (namely, fission products have migrated into the IPyC layer, as defined by the optically active regions under polarized light and in radiographs). The fission products are concentrated at the IPyC-SiC interface, and microprobe analysis showed these deposits to contain lanthanides and palladium. In addition, evidence of lanthanide/palladium-SiC attack was noted (~ 5 and $7 \mu\text{m}$ SiC penetration in Fig. 4-29c and 4-27c, respectively). The kernel is defined as a mottled grey phase with uranium- and carbon-rich regions. No irradiation-induced SiC or OPyC failure was observed in this batch.

4.5.2.2. TRISO-Coated WAR UC_xO_y Fuel. Four separate TRISO-coated WAR UC_xO_y kernel stoichiometries with O/U ratios of 0.45, 0.60, 0.98, and 1.6 were evaluated in the loose particle tests. Figures 4-30 through 4-33 show the metallographic characterization for these particle types. Structural features in the WAR loose particles are summarized as follows:

1. Batch 6151-7E with an O/U ratio = 0.45 showed a two-phase kernel structure [uranium- (white) and carbon-rich (dark grey) areas; refer to Fig. 4-30]. No kernel migration was observed, and the carbon-rich phase was randomly dispersed in large colonies within the kernel and as a concentrated band around the kernel perimeter. In addition, the buffer layer is thinner and, apparently, has been partially consumed by the kernel phase. No metallic fission products were observed in the IPyC layer of this kernel; also, no OPyC or SiC failure was observed.
2. Batch 6157-02-010 with an O/U ratio = 0.60 was the same batch of fissile particles tested in fuel rod IC. The structural features in this loose particle type, shown in Fig. 4-31, are comparable to those observed in fuel rod IC (refer to Fig. 4-24). Figure 4-31 shows localized metallic fission products concentrated at the IPyC/SiC interface; however, no SiC attack was observed.
3. Batch 6551-55E with an O/U ratio = 0.98 shows the kernel phase (Fig. 4-32) to be more concentrated at the center of the particle than was observed in kernels with an O/U ratio = 0.45 (refer to Fig. 4-30). Figure 4-32 shows that the kernel separated into uranium- and carbon-rich phases, with the carbon phase tending to be spherical and surrounded by a uranium-rich matrix. In most particles, the buffer has been partially and sometimes totally consumed by the kernel. The IPyC, SiC, and OPyC did not appear damaged by irradiation. However, fission products are visible at the IPyC-SiC interface (Fig. 4-32d).
4. Batch 6551-85E with an O/U ratio = 1.6 exhibits extensive fuel migration through buffer and into the IPyC. In some cases, the dispersed kernel phase reached the SiC layer; however, the migration did not seem to be caused by a temperature gradient. Metallographic evaluation showed two different structural features in

the kernel phase (refer to Figs. 4-33a and d). The structure shown in Fig. 4-33a shows a clearly defined two-phase region (light grey uranium oxide phase which is surrounded by a carbon phase). In Fig. 4-33d, the two phases are less distinctly defined. A possible explanation for the difference in kernel appearance may be the temperature difference experienced during irradiation by the different particles. The temperature could vary from particle to particle by as much as 40°C, depending upon the location of the particle in the planchets. The IPyC shows significant deterioration, probably attributable to reaction with the dispersed kernel phase. The SiC and OPyC layers appear in good condition; however, failure levels could not be established by visual examination.

4.5.2.3. ZrC-TRISO-Coated WAR UC_xO_y Fuel. A ZrC-TRISO-coated WAR UC_xO_y particle batch (6551-67E) with an O/U ratio = 1.6 was evaluated in the loose particle tests (refer to Fig. 4-34). The fuel kernels in these particles are identical to those of batch 6551-65E (refer to Fig. 4-33) and exhibited similar structural features. The variation of ZrC layer thickness between particles was quite large, and in most particles, radial cracking of the ZrC layer was observed (refer to Fig. 4-35). No irradiation-induced OPyC failures were observed in this batch.

4.5.2.4. Zirconium-Doped BISO-Coated Particles. One loose particle batch (6552-12E) with a 68 wt % zirconium-doped BISO coating was evaluated metallographically (refer to Figs. 4-36 and 4-37). The kernel was a $UC_xO_{0.43}$ and showed separate uranium- and carbon-rich phases. Structurally, the kernel was similar in appearance to WAR $UC_xO_{0.45}$ batch 6551-7E (refer to Fig. 4-30) (namely, no kernel migration was observed, and a dense carbon-rich phase encompassed the kernel perimeter). The particles appeared intact under visual examination; however, many radial cracks were observed in the outer coating that sometimes penetrated the IPyC layer. These cracks were attributed to metallographic preparation and the fragile nature of the irradiated coatings. The polarized micrographs in Fig. 4-36 show an inhomogeneous OPyC coating with apparently zirconium-rich and zirconium-poor zones.

4.5.2.5. Silicon-Doped BISO-Coated Particles. Four loose particle batches of silicon-doped BISO coatings between 22 and 48 wt % silicon were evaluated metallographically.

Batch 6552-149E was a silicon (22 wt %) BISO-coated WAR $UC_xO_{0.98}$ kernel (refer to Fig. 4-38). Structurally, the kernel is similar in appearance to batch 6551-55E (refer to Fig. 4-32). In addition, Fig. 4-38 shows radial cracks in the OPyC layer which were not detected during visual examination. The polarized light micrograph in Fig. 4-38b shows a highly anisotropic OPyC coating, which is attributed to inhomogeneous layers of alternating silicon-rich and silicon-poor deposits.

Figure 4-39 shows batch 6552-135E, a 28 wt % silicon-doped BISO coating with a $UC_xO_{0.42}$ kernel. The kernels in these particles showed no migration and had distinct carbon- and uranium-rich phases. The OPyC layer showed localized voids at the inner surface and radial fissures comparable to those observed in Fig. 4-38.

Two batches (6332-EDS and 6171-117EDS) of BISO-coated (U,Th) O_2 kernels with 40 wt % silicon and 48 wt % silicon-doped OPyC layers were evaluated. Figures 4-40 and 4-41 are representative metallographic cross sections of these particle types which show highly anisotropic OPyC structures. Anisotropy is attributed to alternating layers of silicon-rich and silicon-depleted regions (i.e., defined as circumferential banding in OPyC layers of Figs. 4-40 and 4-41). This structural inhomogeneity resulted in partial OPyC delaminations along the banded structures (refer to Figs. 4-40d and 4-41d). Partial delamination of silicon-doped OPyC layers was observed in capsules GF-1, -2, and -3 (Ref. 4-5), where the severity of OPyC delaminations appeared to increase with silicon-dopant levels.

4.6. MICROPROBE ANALYSIS

The purpose of the microprobe evaluation was to determine the effect of the initial O/U ratio of the kernel on the dispersion of fission products in

the particle. Three batches of TRISO-coated UC_xO_y WAR kernels irradiated as loose particles were chosen for the microprobe analysis. The O/U ratio of the batches were 0.42, 0.98, and 1.6. Four particles from each batch were examined.

In general, particles with a low O/U ratio (<0.6) behave similarly to particles with UC_2 kernels (i.e., lanthanides tend to migrate up the thermal gradient to the SiC coating). Particles with a high O/U ratio (>1) correspond to particles with UO_2 kernels where the lanthanides tend to remain in the kernel. Reference 4-12 presents a detailed evaluation of the microprobe results.

4.6.1. $UC_xO_{0.42}$ (Batch 6551-37E)

Figures 4-42 and 4-43 are back-scattering electron microprobe images of different fission product distributions in a TRISO-coated WAR $UC_xO_{0.42}$ particle batch.

The lanthanides (lanthanum, cesium, praseodymium, neodymium) are found in carbon-rich areas of the kernel. A significant quantity of lanthanides migrate out of the kernel and segregate into bands between the buffer and IPyC and SiC coating layers (refer to Fig. 4-43). The band between the IPyC and SiC was measured at 4 wt % cesium and neodymium, 1.5 wt % lanthanum, and <1 wt % praseodymium.

Cesium was found in the kernel, buffer, and IPyC. About 1.5 wt % cesium was found halfway through the IPyC, and 1 wt % was measured at the IPyC-SiC interface. Approximately 0.5 wt % cesium was found in the uranium phase of the kernel and 3 to 5 wt % in the carbon phase of the kernel.

Zirconium, molybdenum, and ruthenium are primarily distributed within the uranium phase of the kernel. Molybdenum and ruthenium do not migrate out of the kernel; however, zirconium migrates up to the buffer-IPyC interface.

Palladium migrates to the SiC, and xenon and iodine tend to remain in the kernel and its immediate surroundings.

4.6.2. $\text{UC}_x\text{O}_{0.98}$ (Batch 6551-55E)

Figure 4-44 is a back-scattering electron image defining different fission product distributions in TRISO-coated WAR $\text{UC}_x\text{O}_{0.98}$ fuel. The figure shows that lanthanides are concentrated within the kernel phase. The amount of lanthanides at the IPyC-SiC interface is ~ 0.1 wt %. In contrast, the previous batch, which had an O/U ratio of 0.42, had several weight percent in the IPyC-SiC interface. Thus, increasing the O/U ratio from 0.42 to 0.98 improves kernel retention of lanthanides.

The migration of cesium into the coatings was less than for the previous batch (O/U ratio = 0.42) and was measured at ~ 0.5 wt % halfway through the IPyC and ~ 0.2 wt % at the IPyC-SiC interface. The kernel content and distribution of cesium was about the same as for the previous batch. In addition, transition metal (zirconium, molybdenum, ruthenium) distributions were about the same as the previous batch.

Palladium migrates to the SiC, xenon is outside of the uranium kernel area, and iodine is distributed through the kernel and carbon areas.

4.6.3. $\text{UC}_x\text{O}_{1.6}$ (Batch 6551-65E)

Figure 4-45 defines fission product distributions in TRISO-coated WAR $\text{UC}_x\text{O}_{1.6}$ fuel. The fission product distributions in this fuel type are substantially different compared to the other WAR kernels with lower O/U ratios (refer to Figs. 4-42 through 4-44). Figure 4-45 shows that the kernel has three phases, one rich in uranium, and two carbon phases, one of which is very porous. The higher oxygen content made it possible to determine the presence of oxygen in the uranium-rich areas. Table 4-9 summarizes the concentrations of different fission products in each of the kernel phases.

The lanthanides were found mainly in the uranium-rich area of the kernel. Only traces of cesium were found in the coatings; ~0.2 wt % was in the buffer, and less than 0.1% was at the IPyC-SiC interface. The increase in oxygen content in the WAR kernel decreases the lanthanide mobility.

The examination of cesium, barium, and strontium shows cesium to be outside of the uranium phase, strontium mostly inside the uranium phase, and barium uniformly distributed in all three phases of the kernel. The quantity of cesium measured in the coatings was comparable to that in the previous batch (10^3 ppm in IPyC).

The transition metals remained in the kernel area, except zirconium, which migrated into the buffer.

Palladium was again found in the IPyC-SiC interface. Xenon remained primarily in the carbon phase of the kernel, and iodine was not detected.

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(a)

(b)

(c)

Fig. 4-1. Neutronography and radiography of cells I, II, and III in capsule GF-4: (a) neutronography (cell III loose particles), (b) radiography (cell I compacts), (c) radiography (cell II compacts)

COMPACT IC

COMPACT IB

COMPACT IA

Fig. 4-2. Visual appearance of the graphite sheaths and the ends of the fuel rods, compacts IA, IB, and IC. Note: Cuts on IC were caused by unloading.

COMPACT IIC

COMPACT IIB

COMPACT IIA

Fig. 4-3. Visual appearance of the graphite sheaths and the ends of the fuel rods, compacts IIA, IIB, and IIC. Note: Cuts on IIC were caused by unloading.

IC

IB

IA

Fig. 4-4. Visual appearance of cell I fuel rods. Note: Rod IB was broken during PIE handling.

IIA

IIB

Fig. 4-5. Visual appearance of cell II fuel rods. Note: Rod IIC was not removed from sheath.

Fig. 4-6. Visual appearance of TRISO-coated WAR $UC_{x0.6}$ fuel (batch 6157-02-010) irradiated in planchet No. 6 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 72.4% FIMA.

Fig. 4-7. Autoradiograph of TRISO-coated WAR $UC_{x0.6}$ fuel (batch 6157-02-010) irradiated in planchet No. 6 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 72.4% FIMA.

Fig. 4-8. Visual appearance of TRISO-coated VSM UC_2 fuel (batch 6151-17-020) irradiated in planchet No. 22 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 72.4% FIMA.

Fig. 4-9. Autoradiograph of TRISO-coated VSM UC₂ fuel (batch 6151-17-020) irradiated in planchet No. 22 of cell III. Exposure conditions: 1120°C, 7.5×10^{25} n/m² ($E > 29$ fJ)_{HTGR}, and 72.4% FIMA.

Fig. 4-10. Visual appearance of 37% silicon-BISO-coated $(\text{Th,U,Si,Al})\text{O}_2$ fuel (batch 6562-21 EDS) irradiated in planchet No. 63 of cell III. Exposure conditions: 1070°C , $5.7 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 2.0% FIMA.

Fig. 4-11. Autoradiograph of 37% silicon-BISO-coated $(\text{Th,U,Si,Al})\text{O}_2$ fuel (batch 6562-21 EDS) irradiated in planchet No. 63 of cell III. Exposure conditions: 1070°C , $5.7 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 2.0% FIMA.

(a)

(b)

(c)

Fig. 4-12. Visual and metallographic appearance of the delaminations of the SiC-doped OPyC in 48% silicon-BISO-coated (Th:U) fuel (batch 6171-117 EDS) irradiated in planchet No. 6 of cell III. Exposure conditions: 1120°C, $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 3.3% FIMA.

Fig. 4-13. Comparative evaluation of predicted and observed axial dimensional changes in TS-1240 and H-451 graphites

Fig. 4-14. Comparative evaluation of predicted and observed radial dimensional changes in TS-1240 and H-451 graphites

Fig. 4-15. Predicted and measured volumetric changes versus fluence for rod compacts IA, IB, and IC containing TRISO/BISO-coated fuel

Fig. 4-16. Predicted and measured volumetric changes versus fluence for rod compacts IIA and IIB containing TRISO/TRISO-coated fuel.

Fig. 4-17. Predicted and measured dimensional changes versus fluence for rod compacts IA, IB, and IC containing TRISO/BISO-coated fuel

Fig. 4-18. Predicted and measured dimensional changes versus fluence for rod compacts IIA, IIB, and IIC containing TRISO/TRISO-coated fuel

Fig. 4-19. Predicted and measured strain anisotropy versus fluence for fuel compacts irradiated in GF-4

Fig. 4-20. Reirradiation measurements at 820°C for fuel rods irradiated in cell I, GF-4

Fig. 4-21. Reirradiation measurements at 820°C for fuel rods irradiated in cell II, GF-4

Fig. 4-22. Fractional Kr-85m release versus heat treatment temperature for laser failed WAR particles irradiated at 1120°C to 72% FIMA

Fig. 4-23. Representative metallographic cross section of fuel rod IC. Exposure conditions: 1120°C , $8.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, fissile 76.6% FIMA, fertile 4.5% FIMA. Fissile particles: $\text{UC}_{x0.6}$ WAR TRISO, batch 6157-02-010; fertile particles: ThO_2 sol-gel BISO, batch 6542-27-010; inert particles: TRISO, batch 6351-03-010.

(a)

(b)

(c)

(d)

Fig. 4-24. Metallographic cross section of WAR $UC_{x0.6}$ fissile particles (batch 6157-02-010) in GF-4 fuel rod IC. Exposure conditions: $1120^{\circ}C$, $8.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 77% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-25. Metallographic appearance of fertile fuel particles, matrix, and shim in fuel rod IC. Exposure conditions: 1120°C, $8.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 4.5% FIMA. (a) bright field, (b) polarized light, (c) matrix, (d) shim particle.

Fig. 4-26. Representative metallographic cross section of fuel rod IIC. Exposure conditions: 1020°C, $9.4 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) HTGR, fissile 77.2% FIMA, fertile 4.6% FIMA. Fissile particles: UC_2 VSM TRISO, batch 6151-17-020; fertile particles: ThO_2 sol-gel TRISO, batch 6252-05-010; inert particles: TRISO, batch 6351-03-010.

(a)

(b)

(c)

(d)

Fig. 4-27. Metallographic appearance of VSM UC_2 fissile particles (batch 6151-17-020) in GF-4 fuel rod IIC. Exposure conditions: 1020°C , $9.4 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 77.2% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-28. Metallographic appearance of TRISO-coated ThO_2 fertile particles (batch 6252-05-010) in GF-4 fuel rod IIC and fuel rod-graphite sheath interface. Exposure conditions: 1020°C , $9.4 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) HTGR, 4.6% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-29. Metallographic evaluation of TRISO-coated UC₂ VSM fuel particles (batch 6151-17-020) irradiated in planchet No. 22 of cell III. Exposure conditions: 1120°C, 7.5×10^{25} n/m² ($E > 29$ fJ)_{HTGR}, 72% FIMA. (a) bright field, (b) polarized light, (d) radiograph.

(a)

(b)

(c)

(d)

Fig. 4-30. Metallographic evaluation of TRISO-coated WAR $UC_{xO_{0.6}}$ ($O/U = 0.45$) fuel particles (batch 6551-7E) irradiated in planchet No. 97 of cell III. Exposure conditions: $1070^{\circ}C$, 5.7×10^{25} n/m^2 ($E > 29$ fJ)_{HTGR}, 65% FIMA. (a) bright field, (b) polarized light.

(a)

(b)

(c)

(d)

Fig. 4-31. Metallography of TRISO-coated UC_xO_y WAR ($O/U = 0.60$) fuel particles (batch 6157-02-010) irradiated in planchet No. 6 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($F > 29 \text{ fJ}$)_{HTGR}, 74% FIMA. (a) bright field, (b) polarized light.

(a)

(b)

(c)

(d)

Fig. 4-32. Metallography of TRISO-coated UC_xO_y WAR ($O/U = 0.98$) fuel particles (batch 6551-55E) irradiated in planchet No. 61 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) HTGR, 68% FIMA. (c) polarized light.

(a)

(b)

(c)

(d)

Fig. 4-33. Metallographic appearance of TRISO-coated UC_xO_y WAR ($O/U = 1.6$) fuel particles (batch 6551-65E) irradiated in planchet No. 102 of cell III. Exposure conditions: 1070°C , $5.7 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 64% FIMA. (a) bright light, (b) polarized light.

Fig. 4-34. Representative photomicrographs of samples from batch 6551-67E, ZrC-TRISO-coated WAR UC_xO_{1.60} kernels at two magnifications in both bright field and polarized dark illumination. Irradiated in planchet No. 24 at 1120°C, 7.5×10^{25} n/m² ($E > 29$ fJ)_{HTGR} and 73% FIMA.

Fig. 4-35. Metallographic appearance of ZrC-TRISO-coated WAR UC_xO_y ($O/U = 1.6$) fuel particles (batch 6551-67E) irradiated in planchet No. 24 of cell III. Exposure condition: $1120^{\circ}C$, $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 73% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-36. Metallographic appearance of 68% zirconium-BISO-coated $\text{UC}_{x0.43}\text{O}_{1-x}$ WAR fuel particles (batch 6552-123E) irradiated in planchet No. 55 of cell III. Exposure conditions: 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 68% FIMA.

Fig. 4-37. Representative photomicrographs of samples from batch 6552-123E, 68% zirconium-BISO-coated WAR UC_xO_{0.43} kernels at two magnifications in both bright field and polarized dark illumination

(a)

(b)

(c)

(d)

Fig. 4-38. Metallographic appearance of 22% silicon-BISO-coated UC_xO_y WAR ($O/U = 0.98$) fuel particles (batch 6552-149E) irradiated in planchet No. 74 of cell III. Exposure conditions: 1070°C , $5.7 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 67% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-39. Metallographic appearance of 28% silicon-BISO-coated UC_{xO_y} WAR ($O/U = 0.42$) fuel particles (batch 6552-135E) irradiated in planchet No. 52 of cell III. Exposure conditions: $1120^{\circ}C$, $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 70% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-40. Metallographic appearance of 40% silicon-BISO-coated (8 Th,U)O₂ fuel particles (batch 6332-55EDS) irradiated in planchet No. 1 of cell III. Exposure conditions: 1120°C, 7.5×10^{25} n/m² (E > 29 fJ)_{HTGR}, 10% FIMA.

(a)

(b)

(c)

(d)

Fig. 4-41. Metallographic appearance of 48% silicon-BISO-coated $(8 \text{ Th,U})\text{O}_2$ fuel particles (batch 6171-117 EDS) irradiated in planchet No. 3 of cell III. Exposure conditions. 1120°C , $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, 10% FIMA.

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4-64

METALLOGRAPHIC
CROSS SECTION

Cs

Nd

Mo

Xe

ELECTRON IMAGE > 0

Rb

La

Zr

I

U

Sr

Ce

Ru

C

Ba

Pr

50 μm

Fig. 4-42. Back-scattering electron image defining fission product distributions in TRISO-coated WAR $UC_{x0.42}$ kernel (batch 6551-37E) irradiated in loose particle planchet No. 48 to $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) HTGR and a burnup of 72% FIMA at 1120°C .

Fig. 4-43. Back-scattering electron image defining metallic fission product distributions in TRISO-coated WAR $\text{UC}_{x0.42}$ (batch 6551-37E) irradiated in loose particle planchet No. 48 to 7.5×10^{25} n/m^2 ($E > 29$ fJ)_{HTGR} and a burnup of 72% FIMA at 1120°C

Fig. 4-44. Back-scattering electron image defining fission product distributions in TRISO-coated WAR $UC_{x0.98}$ fuel (batch 6551-55E) irradiated in loose particle planchet No. 61 to 7.5×10^{25} n/m² ($E > 29$ fJ)_{HTGR} and a burnup of 68% FIMA at 1120°C: (a) kernel region, (b) coating layers

METALLOGRAPHIC SECTION

ELECTRON IMAGE > 0

U

Cs

Nd

Mo

Xe

C

La

Zr

Pd

O

Ce

Ru

Si

Ba

Pr

Tc

100 μm

Fig. 4-45. Back-scattering electron image defining fission product distributions in TRISO-coated WAR $UC_{x0}1.6$ fuel (batch 6551-65E) irradiated in loose particle planchet No. 102 to 5.7×10^{25} n/m^2 ($E > 29$ fJ)_{HTGR} and a burn-up of 64% FIMA at 1070°C

TABLE 4-1
EXTRACTION FORCES FOR CURED-IN-PLACE FUEL RODS IRRADIATED IN CAPSULE GF-4

Fuel Rod Designation	Processing Variable	Irradiation Conditions		Radial Fuel Rod/Graphite Gap (mm)	Maximum Pushout Force (kg/m-s)
		Fuel Rod Time-Volume Average Temperature (°C)	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]		
IA	Cured-in-place in H-451	1010	7.2	0.005	15
IB	Cured-in-bed in Al ₂ O ₃	1020	8.0	0.005	0
IC	Cured-in-place in H-451	1120	8.9	0	20
IIA	Cured-in-place in TS-1240, extracted, and placed in sheath	1020	8.9	0	49
IIB	Cured-in-place in TS-1240	1020	9.5	0.020	64
IIC	Cured-in-place in TS-1240	1020	9.4	(a)	(a)

(a) Rod not removed (metallographic evaluation in graphite sheath).

TABLE 4-2
VISUAL FAILURE IN DEVELOPMENTAL REFERENCE FUEL IRRADIATED AS LOOSE
PARTICLES IN CELL III OF CAPSULE GF-4 AT 1090°C

Batch No.	Particle Type	No. of Plates	Total No. of Particles	Tray Position(s)	Fluence ^(a) [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Failure (%)	
						OPyC	Total ^(b) Coating
6151-17-020	UC ₂ VSM TRISO	10	1280	22,34,37,77, 79,86,88,90, 92,94	2.9,2.7,2.7,2.2, 2.2,2.1,2.1,2.0, 2.0,2.0	2	0 ^(c)
6157-02-010	UC ₂ WAR TRISO	10	1280	6,10,13,20, 26,29,31,39, 42,50	3.1,3.0,3.0,2.9, 2.8,2.8,2.8,2.7, 2.6,2.5	0	0
6252-03-010	ThO ₂ TRISO	7	707	4,5,7,9, 11,12,84	3.1,3.1,3.1,3.1, 3.0,3.0,2.1	1	0
6252-04-010	ThO ₂ TRISO	7	707	14,16,17,19, 21,38,99	3.0,3.0,3.0,2.9, 2.9,2.7,1.9	0.4	0
6252-05-010	ThO ₂ TRISO	6	606	25,27,28, 30,32,93	2.9,2.8,2.8, 2.8,2.8,2.0	0.5	0
6542-27-010	ThO ₂ BISO	10	1010	43,44,46,47, 49,60,62,64, 66,68	2.6,2.6,2.6,2.6, 2.6,2.4,2.4,2.4, 2.3,2.3	0	NA ^(d)

(a) Assumes linear fluence and back-to-back trays; fluence = 3.17 - (0.71/57) x (tray No.).

(b) Termed pressure vessel failure (failed SiC plus OPyC layer).

(c) One particle with an intact OPyC layer exhibited metallic fission product release (determined autoradiographically).

(d) NA = Not applicable.

TABLE 4-3
VISUAL FAILURE IN ADVANCED FUEL IRRADIATED AS LOOSE PARTICLES IN CELL III
OF CAPSULE GF-4 AT 1090°C

Batch No.	Coating Type	O/U Ratio	No. of Particles Tested	Tray Position(s)	Fluence ^(a) [[10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Failure (%)	
						OPyC	Total Coating
6551-37E ^(b)	TRISO SiC	0.42	128	48	2.6	0	0
6551-51E ^(c)	ZrC TRISO	0.42	128	45	2.6	0	0
6552-135 ^(d)	Si-BISO (28%)	0.42	128	52	2.5	1.6	1.6
6552-141E ^(e)	Zr BISO (41%)	0.42	128	15	3.0	25.8	46.9
6551-9E ^(c)	ZrC TRISO	0.43	164	58	2.4	0	0
6551-7E ^(b)	SiC TRISO	0.43	128	97	2.0	0	0
6552-131E ^(d)	Si BISO (28%)	0.43	128	100	1.9	21.1	21.1
6552-123E ^(e)	Zr BISO	0.43	164	55	2.5	0	0
6551-53E ^(c)	ZrC TRISO	0.98	128	18	2.9	0	0
6551-55E ^(b)	TRISO SiC	0.98	128	61	2.4	0	0
6552-143E ^(e)	Zr BISO (41%)	0.98	128	67	2.3	14.1	69.5
6552-149E ^(d)	Si BISO (22%)	0.98	128	74	2.2	10.2	10.2
6551-67E ^(c)	ZrC TRISO	1.6	128	24	2.9	0	0
6551-65E ^(b)	TRISO SiC	1.6	128	102	1.9	0	0
6562-7E ^(e)	Zr BISO (40%)	1.6	128	104	1.9	0	2.3
6562-3E ^(d)	Si BISO (20%)	1.6	128	8	3.1	50.8	50.8

(a) Assumes linear fluence and back-to-back trays; fluence = $3.17 - (0.71/57) \times (\text{tray No.})$.

(b) TRISO SiC: classic TRISO coating.

(c) ZrC TRISO: TRISO coating with ZrC replacing SiC.

(d) Si BISO (X%): BISO coating with X% Si in OPyC.

(e) Zr BISO (X%): BISO coating with X% Zr in OPyC.

TABLE 4-4
VISUAL FAILURE IN ADVANCED FUEL PARTICLE DESIGNS IRRADIATED AS LOOSE PARTICLES
IN CELL III OF CAPSULE GF-4 AT 1090°C

Particle Characteristics			External Layer	No.		Fluence ^(a)	
Kernel	Metal Coating (wt %)	Batch No.	Deposit Rate (m/min)	Particles Irradiated	Tray Position(s)	[(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Failure ^(b)
Mixed Oxide Undoped Th:U							
10,4:1	30 Si BISO	6332-63-EDS	3.3	60	81,106	2.2, 1.8	8.3+
10,4:1	32.5 Si BISO	6332-67-EDS	3.5	63	81,106	2.2, 1.8	0
10,4:1	40 Si BISO	6332-55-EDS	2.8	60	81,106	2.2, 1.8	0
10,2:1	28 Si BISO	6552-29-EDS	ND ^(c)	60	81,106	2.2, 1.8	0
10,4:1	0 BISO	6411-101-EDS	4.6	60	81,106	2.2, 1.8	0
8:1	19 Si BISO	6171-105-EDS	13.6	66	83,108	2.1, 1.8	0
8:1	25 Si BISO	6171-107-EDS	10.1	66	83,108	2.1, 1.8	0
8:1	39 Si BISO	6171-115-EDS	5.3	66	83,108	2.1, 1.8	10.6*
8:1	48 Si BISO	6171-117-EDS	4.8	63	83,108	2.1, 1.8	46.0*
8:1	(x Si) BISO	(6171-67-EDS ^(d))	11.3	63	83,108	2.1, 1.8	7+
Doped Th:U:Si:Al							
12,6:1:0,93:1,74	37 Si BISO	6562-21-E(DS)	1.9	202	63,109	2.4, 1.8	15.8*
12,6:1:0,93:1,74	30 Si BISO	6562-23-E(DS)	2.4	202	65,111	2.4, 1.8	47.0*
12,6:1:0,93:1,74	39 Zr BISO	6562-25-E(DS)	2.3	202	69,112	2.3, 1.8	0
12,6:1:0,93:1,74	37 Zr BISO	6562-27-E(DS)	1.2	216	71,113	2.3, 1.8	0
Thorine Undoped							
ThO ₂ pure	30 Si BISO	6562-97-E(DS)	4.0	142	2,98	3.1, 1.9	0.70+
ThO ₂ pure	70 Zr BISO	6562-101-E	3.8	135	2,101	3.1, 1.9	27.2+
ThO ₂ pure	0 BISO	6721-31	4.4	141	2,96	3.1, 2.0	0
Doped Th:Si:Al							
1:0,03:0,03	BISO	6551-149	3.5	101	103	1.9	0
1:0,01:0,01	BISO	6551-147	3.6	101	95	2.0	0
1:0,07:0,28 + 1 wt % SiC	BISO	6291-119	ND	707	3,35,51,53, 70,72,80,82, 85,105	2.8,2.7,2.5,2.5, 2.3,2.3,2.2,2.1, 2.1,1.9	0
1:0,07:0,07 + 1 wt % SiC	BISO	6291-111	ND	1111	23,36,54,56, 73,75,91,110	2.9,2.7,2.5,2.5, 2.3,2.2,2.0,1.8	0
1:0,07:0,28 + 1 wt % SiC	BISO	6721-3	3.8	909	40,41,57,59 76,78,87,89, 107	2.7,2.7,2.5,2.4 2.2,2.2,2.1,2.1 1.8	

(a) Assumes linear fluence and back-to-back trays. Fluence = $3.17 - (0.071/57) \times (\text{tray No.})$.

(b) Failure categorized as complete rupture (+) and partial OPyC delamination(*).

(c) ND = Not determined.

(d) Metallography showed that the particles contain silicon in the external layer. They therefore do not belong to batch 6171-67-EDS as indicated in the loading plan.

TABLE 4-5
IRRADIATION-INDUCED DIMENSIONAL CHANGES IN NEAR-ISOTROPIC GRAPHITE
SLEEVES USED IN CELLS I AND II

Fuel Rod No. Encapsulated by Graphite	Graphite Type	Time-Average Temperature (°C) ^(a)	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Dimensional Change (%)	
				Axial	Radial
IA	Great Lakes Carbon H-451	825	7.2	-1.70	-1.00
IB	Great Lakes Carbon H-451	850	8.0	-2.62	-1.07
IC	Great Lakes Carbon H-451	925	8.9	-2.90	-0.75
IIA	Union Carbide TS-1240	875	8.9	0.26	-0.47
IIB	Union Carbide TS-1240	875	9.5	0.50	-0.35
IIC	Union Carbide TS-1240	875	9.4	0.42	-0.23

(a) Refer to Figs. 3-5 through 3-8.

TABLE 4-6
FUEL ROD DIMENSIONAL CHANGE DATA FOR FUEL RODS
IRRADIATED IN CELLS I AND II

Fuel Rod No.	Irradiation Conditions		Dimensional		Volumetric Change (%) (2 $\Delta D/D$ + $\Delta L/L$)	Fuel Rod Anisotropy ($\Delta L/L$ - $\Delta D/D$)(%)
	Time-Volume Average Temperature (°C)	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Diametral, $\Delta D/D$ (%)	Axial, $\Delta L/L$ (%)		
IA	1010	7.2	-3.04	-2.51	-8.59	0.53
IB	1020	8.0	-2.22	(b)	-6.66	(b)
IC	1120	8.9	-1.46	-0.67	-3.59	0.79
IIA	1020	8.9	-0.57	0.61	-0.53	1.18
IIB	1020	9.5	-0.44	0.81	0.07	1.25
IIC	1020	9.4	(a)	1.13	(a)	(a)

- (a) Fuel rod was not removed from graphite sheath (metallographic examination of cured-in-place fuel rod).
 (b) Fuel rod inadvertently broken during disassembly.

TABLE 4-7
THERMAL CONDUCTIVITY MEASUREMENTS ON GF-4 FUEL ROD IIB

Rod Type	Time-Volume Average Temperature (°C)	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Measurement Temperature (°C)	Unirradiated Conductivity, λ_0 (cal/cm-°C-s)(a)		Irradiated Conductivity, λ (Cal/cm-°C-s)	Conductivity Change (%) ($\Delta\lambda/\lambda \times 100$)(b)
				Section 1	Section 2		
TRISO VSM	1020	9.5	600	0.0373	0.0310	0.0166	51
UC ₂ /BISO	--	--	700	0.0358	0.0295	0.0200	38
ThO ₂	--	--	800	0.0345	0.0281	0.0222	29

(a) Sections 1 and 2 refer to thermal conductivity measurements on each half of the preirradiated fuel rod.

(b) Based on the average fuel rod thermal conductivity of sections 1 and 2.

TABLE 4-8
METALLOGRAPHICALLY EXAMINED LOOSE PARTICLES IRRADIATED
IN CELL III OF CAPSULE GF-4

Batch No.	Kernel Type	Coating Type	Planchet No.	Irradiation Conditions		Burnup (% FIMA)	Metallographic Figure No.
				Temp (°C)	Fluence [[10^{25} n/m ²] (E > 24 fJ) _{HTGR}]		
6151-17-020	UC ₂ VSM	TRISO	22	1120	7.5	72	4-29
6551-7E	UC _x O _y WAR O/U = 0.45	TRISO	97	1070	5.7	65	4-30
6157-02-010	UC _x O _y WAR O/U = 0.6	TRISO	6	1120	7.5	74	4-31
6551-55E	UC _x O _y WAR O/U = 0.98	TRISO	61	1120	7.5	68	4-32
6551-65E	UC _x O _y WAR O/U = 1.6	TRISO	102	1070	5.7	64	4-33
6551-67E	UC _x O _y WAR O/U = 1.6	ZrC TRISO	24	1120	7.5	73	4-34, 4-35
6552-123E	UC _x O _y WAR O/U = 0.43	Zr BISO (68%)	55	1120	7.5	68	4-36, 4-37
6552-149E	UC _x O _y WAR O/U = 0.98	Si BISO (22%)	74	1070	5.7	67	4-38
6552-135E	UC _x O _y WAR O/U = 0.42	Si BISO (28%)	52	1120	7.5	70	4-39
6332-55E	(8 Th,U)O ₂	Si BISO (40%)	1	1120	7.5	10	4-40
6171-117E	(8 Th,U)O ₂	Si BISO (48%)	3	1120	7.5	10	4-41

TABLE 4-9
ELECTRON MICROPROBE EVALUATION OF FISSION PRODUCT DISTRIBUTIONS
IN THE WAR $UC_{x0.6}$

Fission Product	Kernel Phases(a)		Porous Carbon Phase
	Uranium Phase	Dense Carbon Phase	
Uranium	X		
Carbon		X	X
Oxygen	X	tr	
Neodymium	X	tr	
Lanthanum	X	tr	
Praseodymium	X	tr	
Cerium	X	tr	
Cesium	tr	X	X
Barium	tr	tr	tr
Strontium	X		
Technetium	X	X	
Ruthenium	X	X	
Molybdenum	X	X	
Zirconium	X	X	
Xenon	tr	tr	X

(a) Refer to Fig. 4-45. X = 1 to several weight percent.
tr = up to 0.1 wt %.

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5. DISCUSSION

5.1. COATED PARTICLE IRRADIATION PERFORMANCE

5.1.1. TRISO-Coated Fuel Pressure Vessel Performance

Cell II of capsule GF-4 was an integral test of TRISO-coated UC_2 and ThO_2 fuel in cured-in-place bodies. This system and its performance in GF-4 is particularly important in HTGR fuels development, since TRISO-coated UC_2 was recently licensed (Ref. 5-1) for the FSV plant and the TRISO-coated ThO_2 licensing application is currently pending (Ref. 5-2). The licensing effort for these fuels was strongly supported by a comparative evaluation between the measured and predicted in-pile Kr-85m R/B values (refer to Fig. 5-1).

The predicted Kr-85m R/B values in Fig. 5-1 are based on Eq. 3-6 and pressure vessel performance models which imply zero fissile and fertile particle failure. Consequently, Kr-85m R/B predictions are determined exclusively by uranium and thorium contributions from heavy metal contamination and from defective particles. Good agreement between predicted and measured Kr-85m R/B values in Fig. 5-1 implies no irradiation-induced fuel failure, which is taken as indirect confirmation of the pressure vessel performance model.

Reference 5-3 details this model and calculates a SiC layer stress, which is a function of dimensions and densities of individual coating components and the temperature, burnup, and fluence to which the particle is subjected. A Monte Carlo calculational routine is then used to repeat particle stress calculations a large number of times (~1000) and define a particle stress density distribution. The predicted failure fraction is then defined as that fraction of fuel with calculated particle stresses greater than the apparent SiC failure stress (AFS). The AFS value is empirically determined

from irradiation capsule data and is 262 MPa (38,000 psi) for spherical TRISO-coated UC_2 or UC_xO_y fuel <600 μm in diameter (Ref. 5-3) and -20 MPa (-2800 psi) for TRISO-coated ThO_2 fuel (Ref. 5-4).

Table 5-1 lists the predicted pressure vessel failure levels for TRISO-coated fissile and fertile fuel tested in cells I and II. A comparison of predicted (0.4%) and measured (0.08%) UC_2 fuel failure in Table 5-1 indicates that the AFS design basis [262 MPa (38,000 psi)] for this particle type is relatively conservative.

A similar comparison for TRISO-coated ThO_2 fuel tested in cell II indicates that an AFS value of -20 MPa (-2800 psi) is apparently conservative. Table 5-1 indicates that the expected pressure vessel failure is 0.1% and 0.03%, based on AFS values of -20 MPa (-2800 psi) and 33 MPa (4800 psi), respectively. Consequently, the calculated AFS value for TRISO ThO_2 tested in GF-4 is between -20 and 33 MPa (-2800 and 4800 psi). This conservatism is particularly important, because the generic HTGR program is currently emphasizing achievement of recycle separability in a TRISO-coated low-enriched uranium (LEU) (<20% enriched) UC_2/ThO_2 fuel system (Ref. 5-5).

Table 5-2 compares the TRISO-coated ThO_2 designs tested in cell II of GF-4 and the current recycle separability design for fertile fuel. Table 5-2 shows that the TRISO-coated ThO_2 particles tested in GF-4 are less conservative than the proposed recycle separability design (namely, the GF-4 particles have larger kernel diameters and smaller buffer thicknesses). However, the most significant factor is that the AFS value established for TRISO-coated ThO_2 irradiated in GF-4 is approaching the currently proposed target AFS value of 33 MPa (4800 psi). Confirming this target AFS value is a current goal of the R2-K13 irradiation proof test for TRISO-coated medium-enriched uranium (MEU) UC_xO_y/ThO_2 fuel (Ref. 5-6). This increase in the calculated AFS value compared to -20 MPa (-2800 psi) is attributed to one or both of the following effects:

1. <0.5% OPyC failure observed for TRISO ThO_2 fuel in cell II (refer to Table 4-2). Originally, the AFS value of -20 MPa (-2800 psi)

was determined for TRISO-coated fuel with 0% to 2.5% OPyC failure (Ref. 5-4). However, an intact OPyC coating provides an added compressive component to the SiC layer. Consequently, if OPyC failure is reduced, less of the TRISO-coated particle population would be subjected to high SiC tensile stresses, and pressure vessel failure would be decreased. This would increase the apparent SiC failure stress (AFS value).

2. Potential improvement in the SiC strength of GF-4 fuel particles, which is attributed to a reduction in the fraction of defective SiC layers. Recent characterization of the fraction of defective SiC layers by mercury intrusion/radiography in GF-4 fuel particles indicates <0.1% defective SiC. In contrast, GF-1, -2, and -3 fuel (Ref. 5-7) showed that 2% to 10% of the SiC layers contained localized microfissures. This type of defect was shown to be not readily detectable by the standard burn-leach technique. Furthermore, as internal fission gas pressure increases during irradiation, microfissures in defective SiC layers were noted to be able to open up to form larger macroscopic cracks. This would reduce the calculated AFS value.

Cell II of GF-4 contained TRISO-coated WAR UC_3O_6 and BISO-coated ThO_2 . Development of this fuel system is not actively being pursued in the HTGR generic fuel program as a consequence of the following performance considerations:

1. High fission gas release (7.5% to 30%) from failed TRISO-coated WAR kernels (refer to Fig. 4-22). Acceptance of WAR UC_xO_y kernels with high gaseous release would require an impractically low fuel-failure design value compared to the current design value of <0.5% pressure vessel failure for UC_2 fuel.

2. Higher metallic fission product retention in an all-TRISO fuel system than in a TRISO/BISO fuel system. This retention is particularly important, because advanced gas-turbine HTGR designs require low metallic fission product release for hands-on maintenance of turbines.

Figure 5-2 plots the measured and predicted in-pile Kr-85m R/B in cell I versus fast fluence. Predicted Kr-85m R/B values are based on Eq. 3-6, zero failure in TRISO-coated WAR fuel, and BISO-coated ThO₂ failure fractions consistent with the design basis defined in Ref. 5-8. Figure 5-2 shows that the measured gaseous release values are systematically less than predicted values. Consequently, the pressure vessel performance models and design basis for TRISO-coated WAR UC_xO_y and BISO-coated ThO₂ fuel are conservative with respect to performance in GF-4 cell I.

5.1.2. TRISO-Coated Particle OPyC Performance

Currently, HTGR fuel specifications assure the irradiation stability of OPyC layers on TRISO-coated fuel by defining critical limits on the following properties and process conditions (Ref. 5-9):

1. Thickness (30 to 50 μm).
2. Density (1.80 to 1.95 Mg/m^3).
3. Optical anisotropy [$<1\% > 1.075$ Bacon anisotropy factor (BAF_O)].
4. Microporosity (13 to 32 ml/kg OPyC).
5. Active coating gas ratio (volume fraction $\text{C}_2\text{H}_2 + \text{C}_3\text{H}_6 > 0.25$).

TRISO-coated fissile and fertile fuels tested in GF-4 were consistent with the above specification requirements, except the upper OPyC microporosity limit and the active coating gas ratio (refer to Table 2-3). The UC₂ and UC₃O_{0.6} fuel, tested in cells I and II, and TRISO-coated ThO₂ batches 6252-03 and 04-010, tested in cell II, had OPyC microporosities between 35 and 60 ml/kg OPyC. The upper OPyC microporosity specification limit is defined to limit OPyC/matrix coating interactions; however, no

matrix/coating interactions were observed metallographically in fuel rods IC and IIC. Mechanistically, the degree of matrix intrusion into the OPyC layer can be rationalized to be dependent on the amount of surface-connected porosity. This is supported by scanning electron microscopy which shows extensive matrix intrusion. Matrix intrusion into the OPyC layer can result in an added irradiation-induced tensile component from two possible mechanisms:

1. Differential volume contraction between the OPyC layer and the intruded matrix phase.
2. Differential volume change of the OPyC coating and the strongly bonded matrix phase between particles.

Recent evaluations of irradiation capsule data on fuel rods (Ref. 5-3) have shown that OPyC microporosity is the most significant property capable of explaining differences in observed failure levels. Figure 5-3 plots TRISO-coated particle OPyC failure in HTGR fuel rods versus microporosity. Figure 5-3 shows that OPyC failure levels increase markedly above 32 ml/kg-OPyC (the recommended upper specification limit). Superimposed on Fig. 5-3 are the metallographic OPyC failure observations made on TRISO-coated fuel in rods IC and IIC (refer to Section 4.5.1). The observations in GF-4 fuel rods (dark circles) are generally consistent with the empirical data base (open symbols), except one point showing zero OPyC failure at a microporosity of 58 ml/kg OPyC. This point represents the TRISO-coated UC₂ batch (6151-17-020) tested in fuel rod IIC. However, only nine fissile particles were observed metallographically in rod IIC. Based on zero OPyC failures in nine observations, the upper 95% confidence level on the actual OPyC failures in this rod is <25%. Consequently, the statistical uncertainty associated with this measurement encompasses OPyC failure levels consistent with the larger body of empirical data.

In addition to the observations of OPyC failure in fuel rods IC and IIC, the GF-4 data on loose particle tests in cell III provide added support to a lower OPyC microporosity specification (>13 ml/kg OPyC). The TRISO ThO_2 loose particles batches tested in cell III had OPyC microporosities between 30 and 60 ml/kg OPyC; however, the batches had active coating ratios <0.25 (refer to Table 2-3). The limit on active coating gas ratio during OPyC deposition is based on empirical data which supports correlations between coating gas ratios, microporosities, and OPyC structures (Ref. 5-3). However, the GF-4 data is inconsistent with the positive correlation which was developed between OPyC microporosity and active coating gas ratios (Ref. 5-3) (namely, for active coating gas ratios <0.25 , the expected OPyC microporosity is <20 ml/kg OPyC).

In considering the deposition process, the volume fraction of hydrocarbon gases (C_2H_2 and C_3H_6) in the total coating gas, the bed surface area, and the coating temperature determine the relative amounts of pyrocarbon crystallite nucleation and growth in the vapor phase and on the levitated particle bed substrate. OPyC structural characterizations (Ref. 5-3) indicate that oriented pyrocarbon crystallites within conical growth features are expected as the deposition coating rate or hydrocarbon concentration is reduced. The crystallites in the conical growth features show a strong tendency for orientation of the carbon layer planes parallel to the surface of the substrate. No intergrowth feature pores are present because the cones impinge uniformly on one another, and the crystallites tend to lie with the layer planes perpendicular to the axis of the conical growth feature. Consequently, the interconnecting microporosity for these type structures is expected to be low. As the deposition coating rate and hydrocarbon gas concentration are increased, the growth features progress from conical to spherical, and individual growth pores develop, which result in higher measured microporosities. In addition, the preferred orientation of the pyrocarbon crystallites between different spherical growth features is randomly disposed, which results in a more isotropic structure. These structural changes account for the increase in OPyC microporosity with increasing volume fraction of active coating gases during deposition.

The inferior irradiation performance of pyrocarbon structures deposited under low hydrocarbon gas concentrations (<0.25) and with low microporosities (<13 ml/kg OPyC) is attributed to excessive orientation of pyrocarbon crystallites. Figure 5-4 plots OPyC failure versus volume fraction of active coating gases for the empirical data base (open circles) used to establish the specification limit and for the three TRISO ThO_2 batches (dark circles) tested in GF-4. Based on the existing correlation in Fig. 5-4, the TRISO ThO_2 batches tested in GF-4 would have been expected to exhibit between 4% and 20% OPyC failure. This apparent lack of a systematic correlation with GF-4 fuel indicates that active coating gas ratio is not a necessary and sufficient process control to assure OPyC performance.

5.2. THERMOCHEMICAL STABILITY

5.2.1. Fuel Kernel Migration

During operation at high temperatures, fuel kernels migrate into the surrounding PyC coatings under the influence of a temperature gradient. Kernel migration up the temperature gradient results from carbon transport in the fuel phase and is characterized by a rejected graphite layer on the cool side of the kernel. Reference 5-10 describes the temperature and thermal gradient dependence of fuel kernel migration in oxide and carbide fuel systems. Using the fuel kernel migration design basis in Ref. 5-10 and the thermal exposure conditions listed in Table 3-2, the predicted kernel migrations in $\text{UC}_3\text{O}_{1.6}$, UC_2 , and ThO_2 are less than $1\text{ }\mu\text{m}$. Predicted kernel migration for UC_xO_y fuel was based on UC_2 data. This is consistent with the data presented in Ref. 5-11, which indicate that WAR UC_xO_y kernels containing $>15\%$ UC_2 should exhibit kernel migration consistent with pure UC_2 kernels. Predicted kernel migrations of $<1\text{ }\mu\text{m}$ are not detectable with standard metallographic techniques; consequently, the observation of no kernel migration in oxide and carbide fuels is consistent with current performance models used in HTGR core design.

5.2.2. Metallic Fission Product Distributions in UC_2 and UC_xO_y

Metallographic and electron microprobe evaluation of metallic fission product distributions in TRISO-coated UC_2 and UC_xO_y fuel showed the following:

1. Rare-earth fission products diffuse from UC_2 kernels and concentrate in the IPyC coating on the cooler side of the fuel particles. The concentrated rare-earth fission products have the potential for reacting with the SiC coating at the SiC/IPyC coating interface (refer to Figs. 4-27 and 4-29).
2. The mobility of rare-earth fission products outside of WAR UC_xO_y kernels is dependent on the O/U ratio of the kernels (namely, as the O/U ratio is increased from ~0.5 to 1.6, sufficient oxygen becomes available in the kernel phase to stabilize rare-earth fission products as oxides, which eliminates the potential for rare-earth/SiC attack) (refer to Figs. 4-30 through 4-33).

The kinetics of rare-earth/SiC attack in UC_2 fuel are dependent on temperature, and Fig. 5-5 plots the rate of change of SiC thickness versus reciprocal temperature. This figure presents a large body of data (Ref. 5-12) on SiC attack and highlights where the UC_2 loose particle tests in GF-4 (cells II and III) are with respect to the empirical data base. Figure 5-5 shows that the rates of SiC attack observed in GF-4 are within the statistical uncertainty associated with the overall empirical data base.

TRISO-coated WAR UC_xO_y fuel has been eliminated from further development in the HTGR fuels program. This decision was based on the high fission gas release (~7.5% to 30%) from failed fuel (refer to Fig. 4-22). The high release was attributed to the porosity of the WAR kernels (typically 35% of theoretical density). However, understanding how WAR UC_xO_y kernel stoichiometry affects fuel failure is important in the generic HTGR program because of the current reference high-density MEU $UC_{0.3}O_{1.5}$ fuel being irradiated in

R2-K13 (Ref. 5-6). Two fuel failure mechanisms related to UC_xO_y stoichiometry are thermal migration of the kernel through the coating layers and chemical interaction between rare-earth fission products and the SiC layer. Thermal migration appears to be most severe for oxide fuels, while chemical interaction is most severe with carbide systems.

Empirical observations in GF-4 and Ref. 5-11, in conjunction with thermodynamic calculations (Ref. 5-11), indicated that oxide-carbide fuel kernels may permit a stoichiometry that reduces both failure mechanisms to manageable levels consistent with HTGR plant design criteria. Stoichiometric adjustments are based on varying the O/U ratio of WAR-derived UC_xO_y kernels and are possible over the complete spectrum from UO_2 to UC_2 . Thermodynamic calculations in Ref. 5-11 define acceptable stoichiometric limits on UC_xO_y kernels based on the following considerations:

1. Upper O/U ratio limit of <1.7 ($UC_xO_{1.7}$), because UO_2 exceeds 85% of kernel phase. This results in excessive CO overpressures within the particle during irradiation and limits the pressure vessel design basis for this particle concept. In addition, excessive kernel migration for these stoichiometries becomes a limiting design consideration.
2. Lower O/U ratio limit of >0.6 ($UC_xO_{0.6}$) is based on kernels which have $<30\%$ UO_2 and contain insufficient oxygen to oxidize rare-earth fission products in fuel subjected to 75% FIMA. In this range of kernel stoichiometries, rare-earth dicarbides form and are observed to segregate at the IPyC/SiC interface and react with the SiC layer.

The electron microprobe results on WAR UC_xO_y particles tested in cell III of GF-4 provide additional support to the thermodynamically defined stoichiometric limits of $0.6 < O/U \text{ ratio} < 1.7$. Figures 4-42 and 4-43 show that rare-earth fission products in $UC_xO_{0.42}$ kernels are associated with

carbon and that the fission products migrate to and concentrate at the inner SiC surface. In addition, Fig. 4-43 shows diffusion of the metallic fission products through the SiC layer. In contrast, Figs. 4-44 and 4-45 show that rare-earth fission products in $UC_xO_{0.98}$ and $UC_xO_{1.6}$ kernels are associated with the SiC surface. In addition, Fig. 4-43 shows diffusion of the metallic fission products through the SiC layer. In contrast, Figs. 4-44 and 4-45 show that rare-earth fission products in $UC_xO_{0.98}$ and $UC_xO_{1.6}$ kernels are associated with the kernel oxide phase and do not migrate into the coating layers. Palladium migrated in $UC_xO_{1.6}$ kernels and formed a band at the IPyC/SiC interface; however, no SiC attack was observed for an exposure of 64% FIMA, $5.7 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}, and 1070°C.

5.2.3. Silicon-Doped OPyC Layers

Four separate batches of BISO-coated UC_xO_y and $(8Th,U)O_2$ particles with silicon-doped OPyC layers (22 to 48 wt % silicon) were irradiated in cell III at temperatures between 1070° to 1120°C and fluences between 5.7 to $7.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}. Metallographically, all of the batches exhibited either fine radial cracks traversing the OPyC layer or partial circumferential delaminations of the outer OPyC layer. This type of failure may be attributed to the structural appearance of silicon-doped OPyC layers, which are characterized as concentric laminated rings rich in silicon content (refer to Figs. 4-38 through 4-41).

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Fig. 5-1. Measured and predicted Kr-85m R/B versus fast fluence for capsule GF-4, cell II containing VSM UC_2 TRISO/ ThO_2 TRISO fuel

Fig. 5-2. Measured and predicted Kr-85m R/B versus fast fluence for capsule GF-4, cell I containing WAR $\text{UC}_{30.6}^{\text{O}}$ TRISO/ ThO_2 BISO fuel

Fig. 5-3. Matrix/OPyC interaction in TRISO-coated fissile and fertile fuel as a function of OPyC microporosity

Fig. 5-4. OPyC failure versus the volume fraction of active coating gases

Fig. 5-5. Rate of change of SiC thickness in TRISO-coated UC₂ fuel versus reciprocal temperature. Comparative evaluation between GF-4 results and empirical data base.

TABLE 5-1
PREDICTED AND OBSERVED PRESSURE VESSEL FAILURE LEVELS FOR TRISO-COATED
FISSILE AND FERTILE FUEL TESTED IN CELLS I AND II

Cell No.	TRISO-Coated Fuel Type (Batch Designation)	Irradiation Conditions			Pressure Vessel Failure (%)			
		Temperature (°C)	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Burnup (% FIMA)	Measured ^(a)	Predicted Based on AFS Value of		
						-2800	4800	3800
I	WAR UC ₃ O _{0.6} (6157-02-010)	1010 to 1120	5.6 to 8.0	69 to 77	0.12	--	--	0
II	VSM UC ₂ (6151-17-020)	1020	8.9 to 9.5	77	0.08	--	--	0.4
II	ThO ₂ 6252-03-010	1020	8.9 to 9.5	4.6 to 4.8	0.08	0.1	0	--
	6252-04-010	1020	8.9 to 9.5	4.6 to 4.8	0.08	0.1	0	--
	6252-05-010	1020	8.9 to 9.5	4.6 to 4.8	0.08	0.1	0.1	--

(a) Based on in-pile EOL Kr-85m R/B (refer to Section 4.4.3).

TABLE 5-2
COMPARATIVE EVALUATION OF TRISO-COATED ThO₂ PARTICLE DIMENSIONS TESTED IN GF-4 AND
ThO₂ DESIGN PROPOSED FOR RECYCLE SEPARABILITY

TRISO-Coated ThO ₂ Particle Design	Particle Dimensions (μm)					Pressure Vessel Failure Design Basis			
	Kernel Diameter	Buffer Thickness	IPyC Thickness	SiC Thickness	OPyC Thickness	Fluence [(10 ²⁵ n/m ²) (E > 29 fJ) _{HTGR}]	Burnup (% FIMA)	AFS [MPa (psi)]	Failure (%)
GF-4, Cell II									
6252-03-010	529	63	30	33	37	9	4.5	-19.3(-2800)	≤0.5
6252-04-010	528	63	30	33	39	9	4.5	-19.3(-2800)	≤0.5
6252-05-010	519	63	30	33	49	9	4.5	-19.3(-2800)	≤0.5
Recycle separability	500	80	35	35	40	9	7.0	33.1 (4800)	≤0.5

6. SUMMARY AND CONCLUSIONS

Capsule GF-4 demonstrated the integrity of a TRISO-coated HEU UC_2/ThO_2 fuel system subjected to exposure conditions in excess of nominal FSV operating conditions. In addition, GF-4 provided extensive empirical support for defining the optimum kernel stoichiometry for a TRISO-coated HEU WAR UC_xO_y fuel system. Evaluation of fuel performance with respect to generic product and process specifications additionally confirmed HTGR specification requirements and performance models used in core design studies. The irradiation results are briefly summarized below:

1. HTGR fuel rods cured-in-place in Union Carbide (TS-1240) and Great Lakes Carbon (H-451) graphite crucibles were successfully irradiated to exposures of 7.2 to $9.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR} and 1010° to 1120°C . No detrimental fuel rod/graphite element interactions were observed, and all rods exhibited adequate structural integrity. The maximum pushout force for cured-in-place rods was 64 kg/m-s .
2. GF-4 fuel rods fabricated with all TRISO-coated fuel particles exhibit volumetric changes of -0.5% and 0.1% for fluences between 8.9 and $9.5 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}. This behavior is consistent with irradiation-induced expansion in the matrix phase. In contrast, fuel rods fabricated with TRISO- and BISO-coated fuel exhibited volumetric contractions of -3.6% to -8.6% for fluences between 7.2 and $8.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}. Predicted and observed fuel rod dimensional changes agreed well for rods containing all TRISO-coated fuel; however, agreement was poor for rods containing TRISO- and BISO-coated fuel. This inconsistency was attributed to calculated BISO-coated particle volumetric

changes which are typically 2% to 6% less than observed for fluences between 7 and $9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$)_{HTGR}. In addition, GF-4 fuel rods exhibited strain anisotropy consistent with the empirical model, showing that axial contractions are less than diametral contractions.

3. Irradiation-induced dimensional changes in Union Carbide (TS-1240) and Great Lakes Carbon (H-451) graphite crucibles are consistent with HTGR core design values.
4. Thermal conductivity measured in each half of a preirradiated fuel rod differed 19% to 22%. This difference is attributed to matrix macroporosity structural inhomogeneities in the fuel rod. The irradiated fuel rod thermal conductivity decreased between 29% to 51%, compared to the unirradiated rod. This decrease is reasonably consistent with the 42% to 64% decrease reported for GF-1 and GF-2 fuel rods. The measured reductions in thermal conductivity are subject to large uncertainties as a result of inaccuracies associated with a comparative standard measurement in the LECI thermal conductivity technique.
5. Heat treatment studies on laser-failed WAR UC_xO_y fuel indicated that Kr-85m R/B values varied between 7.5% and 30%, depending on temperature and prior heat treatment conditions. This relatively high fractional gaseous release for UC_xO_y compared to UC_2 fuel provides added support for the 1976 decision to drop development of low-density WAR UC_xO_y kernels in the generic HTGR fuels program.
6. Predicted and measured in-pile Kr-85m R/B values for TRISO UC_xO_y /BISO ThO_2 (cell I) and TRISO UC_2 /TRISO ThO_2 (cell II) agreed well. This implies $<0.1\%$ pressure vessel failure for fuel exposed to the following conditions: 5.9 to $9.5 \times 10^{25} \text{ n/m}$ ($E > 29 \text{ fJ}$)_{HTGR},

1010° to 1120°C, 69% to 77% fissile FIMA, and 2.5% to 4.8% fertile FIMA.

7. Thermodynamically defined stoichiometric limits of $0.6 \leq O/U \leq 1.7$ for UC_xO_y kernels are supported by a microprobe evaluation of rare-earth fission product distributions in TRISO-coated fuel. Specifically, O/U ratios of ~ 1 and 1.6 for UC_xO_y kernels tested in GF-4 showed rare-earth fission products associated with the kernel oxide phase and no migration into the coating layers. These stoichiometries reduce the potential for rare-earth/SiC attack. Kernels with O/U ratios $\lesssim 0.6$ have excess CO pressures which limit the pressure vessel design and have the potential for excessive amoeba migration.
8. No amoeba migration was observed in WAR UC_xO_y , VSM UC_2 , or ThO_2 fuel tested in GF-4 fuel rods. These results are consistent with current performance models.
9. Five to seven micrometers of lanthanide/SiC attack were observed in UC_2 fuel tested in rods and as loose particles. The rates of SiC attack observed in GF-4 are consistent with empirical models, based on a square root of time dependence.
10. BISO-coated UC_xO_y and $(8Th,U)O_2$ with silicon-doped OPyC layers in the range of 22 to 48 wt % exhibited 0% to 47% pressure vessel failure. No apparent correlation existed between silicon-dopant level and failure; however, all batches exhibited an OPyC structural inhomogeneity characterized as concentric laminated rings rich in silicon content.

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