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POSTIRRADIATION EXAMINATION OF CAPSULES P13R AND P13S

by
C. B. SCOTT, D. P. HARMON, and J. F. HOLZGRAF

Prepared under
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ABSTRACT

Capsules P13R and P13S were the seventh and eighth in a series of irradiation tests conducted under the ERDA-sponsored HTGR Fuels and Core Development Program. Reference type LHTGR fuel fabricated with a broad spectrum of property and process variables was irradiated to extreme temperature and fluence conditions. The purpose of these tests was to provide a technical basis for LHTGR fuel product specifications and to provide irradiation data to support and update fuel performance models used in design evaluations.

Postirradiation examination revealed that the bonded fuel rods exhibited good stability after irradiation to fast neutron fluences of 12.4×10^{21} n/cm² (E > 0.18 MeV)_{HTGR}, which is 55% beyond the LHTGR peak design fast neutron fluence of 8.0×10^{21} n/cm². Thermal cycling to high temperatures did not adversely affect fuel rod integrity.

Particle batches with coating designs representative of the design requirements envisioned for the LHTGR exhibited excellent irradiation performance. Ten batches of fissile and fertile particles were irradiated without coating failure to fast neutron exposures which exceeded the LHTGR peak design exposure by 35 to 52%.

A large number of particle batches was tested with coating designs or particle attributes that were not representative of LHTGR design requirements. The irradiation performance of these samples provided important data required to establish fuel particle specification limits.

Capsules P13R and P13S were considered to be very successful qualification tests of LHTGR fuel components. These results provided a substantial data base for the LHTGR Fuel Product Specification and performance models

used in HTGR core design studies, and demonstrated the excellent irradiation performance of reference LHTGR fuel to well beyond peak design exposures.

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1. INTRODUCTION

Development of the high-temperature gas-cooled reactor (HTGR) has been under way in the United States since 1957. The initial fuel development effort for the Peach Bottom and Fort St. Vrain reactors was directed toward demonstrating the coated fuel particle concept and the multihole graphite fuel element design. Development of fuel for the large commercial HTGRs, termed LHTGRs, draws heavily upon General Atomic's experience in developing and manufacturing fuel for the early reactors, especially for the 330-MW(e) Fort St. Vrain power station. However, optimization of the LHTGR fuel has led to a number of changes from Fort St. Vrain fuel, which are summarized in Table 1-1.* These differences are principally the result of three major design goals: (1) complete separation of fissile and fertile material, (2) simplification of the fuel fabrication process, and (3) improved fuel performance.

LHTGR fuel consists of small, right-cylinder rods containing an intimate blend of coated fuel particles and isotropic graphite shim particles bonded together in a close-packed array. Fuel rods are formed by injecting the close-packed particle bed with a matrix material consisting of an organic binder and graphite filler, which when carbonized and heat treated yields a relatively strong carbonaceous bonding medium. The fuel rods are carbonized and heat treated in situ in the graphite fuel element. The use of isotropic graphite shim particles and the cure-in-place process for LHTGR rods represents design improvements that are attractive in both fresh fuel manufacturing and in the remote refabrication process. This commonality between fresh and recycle fuel will reduce overall development costs and reduce risks associated with the refabrication plant.

* Tables and figures appear at the end of each section.

Two types of coated fuel particles are used in LHTGRs: fissile and fertile. The fissile particle has a dense UC_2 fuel kernel consisting of highly enriched (93%) uranium; the fertile particle has a ThO_2 kernel. The design of the coated particles using TRISO coatings on fissile particles and BISO coatings on fertile particles facilitates separation of the bred U-233 from the nuclear poison U-236, generated primarily in the fissile kernel. During reprocessing, a burn-leach step is used to separate thorium and U-233 from the fissile particle.

Since the coatings also serve as the primary barrier to fission product release, it is important to ensure a high degree of coating integrity under all normal operating conditions. This is done in two ways: (1) design and specification of coated particle parameters to ensure a high probability of surviving irradiation-induced stresses, and (2) thermal design of the reactor core to eliminate significant failure due to thermochemical interactions between fuel and fission products and the structural coating layers.

Irradiation tests being conducted by General Atomic (GA) are designed to demonstrate reliable fuel performance to and beyond design LHTGR operating conditions. This report documents the postirradiation examination results for capsules P13R and P13S and discusses the impact of these results on the current fuel specifications and fuel performance models.

TABLE 1-1
PRINCIPAL DIFFERENCES BETWEEN FORT ST. VRAIN AND LARGE HTGR FUEL

	Fort St. Vrain	LHTGR	Purpose of Change
Fuel particles			
Fissile kernel	4:1 (Th,U)C ₂	UC ₂ (melted kernel)	Improve separability
Fertile kernel	ThC ₂	ThO ₂	Simplify process and improve performance
Fissile/fertile coatings	TRISO/TRISO	TRISO/BISO	Improve separability
Fuel rods			
Carbonization technique	In Al ₂ O ₃	In-block	Simplify process
Rod shimming method	Fuel particle blending	Graphite shim particles	Simplify process
Dimensions, cm	Diameter: 1.27 Length: 5.08	Diameter: 1.57 Length: 6.35	Improve core design and simplify process

2. DESCRIPTION OF EXPERIMENT

2.1. OBJECTIVES

These tests have four primary objectives:

1. Characterize reference fresh fuel materials and processes (Ref. 2-1). Results of these tests will be used to support the HTGR fuel specifications (Ref. 2-2). Fuel rod tests are designed to characterize rod integrity and dimensional change as a function of irradiation conditions and rod variables (i.e., shim type and volume loading, heat-treatment temperature, filler type, macro-porosity and microporosity, etc.). The series of unbonded particle tests conducted in cells 3 and 4 of capsules P13R and P13S were selected to provide information on effects of coating property variables and coating design on irradiation performance.
2. Determine the temperature-fluence dependence of in-pile fission gas release and relate it to coated particle integrity. Reference fresh fuel rods (200- μm UC₂ TRISO/500- μm ThO₂ BISO) were designed to be tested at 1075°, 1300°, and 1500°C. In-pile fission gas release data provide a means for monitoring fuel failure as a function of fluence and temperature. These results are used to support and update current fuel performance models (Ref. 2-3).
3. Determine the effect of thermal cycling on fuel integrity. Cell 1 of capsule P13S contained the first in-pile thermal cycling test of reference-type fuel. The thermal cycles simulated the most severe thermal transients to be experienced in an LHTGR during normal load-following conditions. A companion cell in P13R contained the same fuel and was not thermal cycled to provide a basis for performance comparison.

4. Test alternative fuel materials. A number of alternative materials are included in these capsules. These materials include $(\text{Th,U})\text{O}_2$ TRISO fissile particles, Lonza filler and shim particles, and nonimpregnated shim particles. At the time fuel for these capsules was being fabricated (early 1973), $(\text{Th,U})\text{O}_2$ TRISO particles were considered an alternate candidate for the fissile fuel particle. Subsequently, the mixed oxide has been replaced by the weak-acid-resin (WAR) fissile kernel.

2.2. DESCRIPTION OF FUEL SPECIMENS

Capsules P13R and P13S contained 40 fuel rod specimens and 96 unbonded particle samples that were designed to provide data on a broad spectrum of property and process variables for reference LHTGR fuel. Reference-type UC_2 (VSM) TRISO fissile and ThO_2 BISO fertile particles were used in the majority of the fuel samples. Most of the fuel rod specimens were fabricated by the hot-injection process and were carbonized in H-451 graphite tubes to simulate in-block curing. A detailed description of the fuel specimens tested in capsules P13R and P13S is given in Ref. 2-1.

2.2.1. Coated Particles

A general description of the coated particles is given in Tables 2-1 and 2-2. The reference-size 200- μm -diameter UC_2 (VSM) kernel was used in approximately 85% of the fissile particle batches. Most of the UC_2 kernels selected for substrate material were from two large VSM production-size batches. One batch had approximately 1.0 wt % thorium added to improve spherodicity of the kernel. The other batch was not thorium doped; consequently, the kernels were "dimpled" and the coatings on these kernels were faceted. Two alternative fissile kernels, 250- μm -diameter $(\text{Th,U})\text{O}_2$ and 500- μm -diameter $(8\text{Th,U})\text{O}_2$ kernels, fabricated by the sol-gel process, were also tested in order to obtain additional irradiation data on these potential backup fuel kernels. All fissile particles were TRISO coated. The primary variables tested in each particle batch are summarized in Table 2-3, and the coating process data are summarized in Table 2-4.

Reference-size 500- μ m-diameter ThO_2 kernels were used in all fertile particle batches. Fuel kernels from a large number of pilot-scale equipment runs were tested. Many of the ThO_2 kernels were coated and "burned-back" one or more times. To evaluate the irradiation performance of fuel particles containing broken or misshapen kernels, irregularly shaped kernels from a parent batch were separated on a vibrating table and coated.

Fertile particle coating attributes investigated include OPyC coating rate, density, OPTAF, thickness, faceted coatings, no seal coat, and use of N_2 carrier gas during the buffer coating deposition. The majority of OPyC coatings were deposited from propylene, although a number of OPyC coatings were deposited from propylene-acetylene (mixed-gas). Utilization of mixed-gas OPyC coatings is considered necessary in order to fabricate fertile particles on a scale envisioned for the LHTGR. One ThO_2 TRISO particle batch, which is the LHTGR backup fertile particle design, was tested for comparative purposes. A summary of fertile particle variables tested is given in Table 2-5, and the coating process data are summarized in Table 2-6.

2.2.2. Fuel Rods

The fuel rods contained a blend of TRISO coated fissile, BISO coated fertile, and graphite shim particles. Most of the fuel rods also contained varying amounts of TRISO coated inert resin particles in order to achieve close particle packing and the desired fuel particle and shim loadings. The fuel rods were approximately 1.242 cm (0.493 in.) in diameter by 1.905 cm (0.750 in.) long; 16 of the rods tested contained a central hole approximately 0.34 cm (0.13 in.) in diameter through the entire length of the rod to accommodate a thermocouple.

Normally only one attribute within the fuel rod was varied and all other attributes were held constant in order to isolate the effect of that variable on the irradiation performance of the fuel rod. Combining of some variables in given rods was necessary; however, the variables combined were judged not to interact with one another. A general description of the

fuel rods irradiated is given in Tables 2-7 and 2-8, and a summary of the variables tested in each rod is given in Table 2-9.

Fuel rods were fabricated using a prototype LHTGR hot-injection press with the exception of two rods, which were fabricated by the slug-injection process. The majority of the rods were cured-in-place by placing five or six rods in H-451 graphite tubes with Grafoil disks between the rods to prevent sticking. The tubes were heated in an induction furnace in a nitrogen atmosphere at 10°C/min to 1000°C. The final firing was done in an argon or nitrogen atmosphere at 1500° or 1800°C. Two fuel rods were fired in packed beds of alumina at 1800°C in a nitrogen atmosphere. Preirradiation photomicrographs of fuel rods that are representative of the different fabrication and firing processes evaluated are shown in Figs. 2-1 and 2-2.

The graphite shim and filler materials used were comminuted, bulk, near-isotropic graphite supplied by candidate vendors for near-isotropic HTGR graphite. The GLCC 1099^{*} shim impregnated with furfuryl alcohol was used in 32 fuel rods. Four rods contained nonimpregnated GLCC 1099 shim and two rods contained impregnated Lonza^{**} shim particles. The shim volume fraction in the rods tested was varied from 0.10 to 0.36 in order to study the effect of shim content on fuel rod dimensional change.

A matrix composition of 30 wt % 6353 natural-flake graphite,[†] 65 wt % A240 pitch,^{††} and 5 wt % SC011 (octadecanol) additive was used in 34 fuel rods. The remaining rods contained Lonza KS-44 graphite filler, SC003 (stearic acid), or SC027 (dimethyl terephthalate) matrix additives. The matrix additives were incorporated to improve the ease of fuel rod fabrication and to reduce the fuel rod pitch coke content after firing.

* Great Lakes Carbon Corporation.

** Lonza, Ltd.

† Ashbury Graphite Mills.

†† Ashland Oil Company.

2.3. CAPSULE DESIGN

Capsules P13R and P13S were nearly identical in design and each capsule consisted of six individual cells. Fuel rod specimens were tested in cells 1, 2, 5, and 6 and unbonded particle samples were tested in cells 3 and 4. Fuel rod specimens were housed in split graphite crucibles inside Inconel cans which formed the primary containment for each cell, as shown in Fig. 2-3. Each unbonded particle cell contained a graphite crucible assembly, which consisted of 12 annular trays arranged on a threaded graphite stock, as shown in Fig. 2-4. The cells were arranged vertically in an outer containment tube and were numbered 1 through 6 from the bottom of the capsules. The outer containment tube was fabricated from type 304L stainless steel and had an outside diameter of 3.18 cm (1.250 in.) and a wall thickness of 0.089 cm (0.035 in.). An aluminum band spacer was located between the Inconel cans and the outer containment tube. The outer portion of the spacer was slotted to guide the thermocouples and gas tubes to the lead tube.

Each cell was continuously purged with an inert neon/helium gas mixture. Primary temperature control gaps within the cells and secondary control gaps outside the cells were designed to provide the desired range of temperatures by varying the primary and secondary gas mixture composition.

Unbonded particle cells 3 and 4 and fuel rod cells 2 and 6 of both capsules and cell 1 of capsule P13R were designed to operate at a centerline temperature of 1075°C. Cell 5 of capsules P13R and P13S was designed to operate at 1300° and 1500°C, respectively. Cell 1 of capsule P13S was designed to be thermal cycled between a nominal operating temperature of 1075° and 1600°C periodically during capsule operation.

Each capsule contained 35 thermocouples located in or adjacent to the test specimens. Thermocouples used for measuring unfueled area temperatures below 1100°C were Inconel-sheathed Chromel-Alumel wire insulated with magnesium oxide. Fuel and high-temperature areas were monitored with tantalum-sheathed, tungsten-rhenium (W - 3% Re/W - 25% Re) thermocouples

insulated with beryllium oxide. The tungsten-rhenium thermocouples decalibrate under neutron-fluence-induced transmutation; therefore, Chromel-Alumel thermocouples, which do not decalibrate, were placed in conjunction with the tungsten-rhenium thermocouples in the graphite crucible to establish decalibration rates for the experiment.

Standard dosimetry wires (Va - 0.216% Co, V - 0.522% Fe, and pure Fe) were encapsulated in quartz ampoules and positioned along the capsules at seven axial locations at 0° and 180° radial orientations. In addition to the standard dosimetry wires, six ampoules containing purified Mn-54 were employed to evaluate the Mn-54 transmutation cross section.

REFERENCES

- 2-1. Young, C. A., and D. P. Harmon, "Preirradiation Report: Fuel Materials for P13R and P13S Irradiation Capsules," USAEC Report GA-A13026, General Atomic Company, November 1974.
- 2-2. Harmon, D. P., J. T. Ganley, and O. M. Stansfield, "HTGR Fuel Product Specification, Issue B," ERDA Report GA-A13464, General Atomic Company, January 1976.
- 2-3. Smith, C. L., "Fuel Particle Behavior Under Normal and Transient Conditions," USAEC Report GA-A12971 (GA-LTR-15), General Atomic Company, October 1, 1974.

TABLE 2-1
GENERAL DESCRIPTION OF COATED PARTICLES TESTED IN CAPSULE P13R

Particle Batch Data Retrieval Number	Kernel		Coating													Total Coating Thick- ness (μ m)	Total Coated Particle										Particle Disposition	
			Type (a)	Buffer		Seal Coat	Inner Isotropic PyC			SiC		Outer Isotropic PyC					Metal Loading		Fission Gas Release (c)	Exposed Heavy Metal (d)		Fraction Defective SiC Coatings (e)		Used In				
	Type	Mean Diameter (μ m)		Thick- ness (μ m)	Density (g/cm ³)		Thick- ness (μ m)	Density (g/cm ³)	OPTAF (b)	Thick- ness (μ m)	Density (g/cm ³)	Thick- ness (μ m)	Density (g/cm ³)	Coating Rate (μ m/min)	OPTAF (b)		Uranium (wt %)	Thorium (wt %)		Uranium (g U/g U)	Thorium (g Th/g Th)	Fuel Rods	Unbonded Test					
Fissile Particles (f)																												
5466-37	(Th,U)O ₂	258	TRISO	101 (g)	1.12	Yes	29	1.87	1.07	30	3.18	31	1.79	6.0	1.10	197	619	2.37	12.7	12.30	1 x 10 ⁻⁶	2 x 10 ⁻⁷	6 x 10 ⁻⁵	5 x 10 ⁻⁷		X		
6151-00-010	UC ₂ (1)	199	TRISO	95 (g)	1.18	Yes	3 ^c	1.94	1.19	31	3.20	44	1.80	1.1	1.11	207	595	2.28	17.3	0.43	4 x 10 ⁻⁷	0.00	0.00	3 x 10 ⁻⁴		X		
6151-00-025	UC ₂	190	TRISO	90 (g)	1.15	Yes	35	1.93	1.16	31	3.20	33	1.88	1.4	1.17	197	561	2.36	18.0	NA (h)	6 x 10 ⁻⁸	4 x 10 ⁻⁷	ND (j)	2 x 10 ⁻⁴		X		
6151-00-035	UC ₂ (1)	201	TRISO	87 (g)	1.07	Yes	33	1.92	1.22	27	3.20	37	1.85	3.9	1.17	191	577	2.29	18.9	0.20 (k)	3 x 10 ⁻⁷	4 x 10 ⁻⁷	ND	2 x 10 ⁻⁶		X	X	
6151-00-045	UC ₂	196	TRISO	85 (g)	1.15	Yes	35	1.93	1.17	29	3.22	40	1.81	3.8	1.13	192	578	2.33	17.8	NA	2 x 10 ⁻⁷	1 x 10 ⁻⁷	ND	2 x 10 ⁻⁶		X	X	
6151-00-046 (1)	UC ₂	196	TRISO	85 (g)	1.15	Yes	35	1.93	1.17	29	3.22	40	1.81	3.3	1.13	192	578	2.33	17.8	NA	2 x 10 ⁻⁷	1 x 10 ⁻⁷	ND	2 x 10 ⁻⁶		X	X	
6151-01-015	UC ₂ (1)	197	TRISO	99 (g)	1.15	Yes	31	1.94	1.20	25	3.21	42	1.90	0.7	1.15	203	590	2.28	17.4	0.22	1 x 10 ⁻⁸	1 x 10 ⁻⁷	0.00	4 x 10 ⁻⁵		X		
6151-04-015	UC ₂ (1)	202	TRISO	102 (g)	0.87	Yes	33	1.88	1.23	28	3.20	36	1.82	1.4	1.14	202	599	2.45	17.4	0.22	8 x 10 ⁻⁷	1 x 10 ⁻⁶	9 x 10 ⁻⁴	7 x 10 ⁻⁵		X		
6151-08-015	UC ₂ (1)	202	TRISO	95 (g)	1.07	Yes	33	1.92	1.25	26	3.21	36	1.76	3.6	1.09	190	593	2.28	18.6	0.20 (k)	1 x 10 ⁻⁷	9 x 10 ⁻⁷	ND	2 x 10 ⁻⁶		X		
6151-09-015	UC ₂ (1)	197	TRISO	95 (g)	1.15	Yes	31	1.94	1.16	29	3.22	39	1.94	3.6	1.06	204	587	2.30	17.8	0.20 (k)	9 x 10 ⁻⁸	4 x 10 ⁻⁷	ND	4 x 10 ⁻⁶		X	X	
6151-09-025	UC ₂ (1)	197	TRISO	87 (g)	1.18	Yes	35	1.94	1.14	31	3.22	41	1.95	4.0	1.11	204	590	2.37	17.5	0.20 (k)	2 x 10 ⁻⁷	8 x 10 ⁻⁷	0.00	3 x 10 ⁻⁵		X		
Fertile Particles																												
4252-06-018 (m)	ThO ₂ (n)	511	BISO	78	1.10	Yes	NA	NA	NA	NA	NA	77	1.82	5.8	1.14	155	826	2.72	NA	58.90	NA	ND	7 x 10 ⁻⁴				X	
6542-01-010	ThO ₂ (n)	500	BISO	79	1.08	Yes	NA	NA	NA	NA	NA	85	1.80	10.0	1.06	164	828	3.59	NA	56.50	NA	ND	6 x 10 ⁻⁶	NA		X	X	
6542-01-020	ThO ₂ (o)	504	BISO	81	1.17	Yes	NA	NA	NA	NA	NA	74	1.82	2.7	1.10	155	813	3.55	NA	57.50	NA	ND	6 x 10 ⁻⁶	NA		X	X	
6542-02-020	ThO ₂ (n)	481	BISO	87	1.08	Yes	NA	NA	NA	NA	NA	72	1.91	8.5	1.06	159	796	3.59	NA	56.20	NA	ND	2 x 10 ⁻⁵	NA		X	X	
6542-02-030	ThO ₂ (n)	481	BISO	79	1.18	Yes	NA	NA	NA	NA	NA	74	1.89	2.2	1.12	153	786	3.54	NA	56.00	NA	ND	8 x 10 ⁻⁵	NA		X	X	
6542-09-010	ThO ₂ (n)	511	BISO	84	1.06	Yes	NA	NA	NA	NA	NA	75	1.93	5.0	1.13	159	822	3.56	NA	57.60	NA	ND	1 x 10 ⁻⁴	NA		X	X	
6542-11-015	ThO ₂ (o)	497	BISO	87	1.12	No	NA	NA	NA	NA	NA	74	1.83	2.6	1.29	161	819	3.35	NA	57.80	NA	ND	6 x 10 ⁻⁷	NA		X	X	
6542-18-015	ThO ₂ (o)	476	BISO	84	1.11	No	NA	NA	NA	NA	NA	70	1.84	6.5	1.13	154	791	3.53	NA	58.80	NA	ND	3 x 10 ⁻⁶	NA		X	X	
6542-19-015	ThO ₂ (o)	486	BISO	81	1.15	Yes	NA	NA	NA	NA	NA	77	1.78	7.5	1.16	158	824	3.55	NA	58.90	NA	ND	4 x 10 ⁻⁶	NA		X		
6542-20-025	ThO ₂ (o)	495	BISO	76	1.25	Yes	NA	NA	NA	NA	NA	81	1.82	3.5	1.18	157	809	3.57	NA	58.20	NA	ND	3 x 10 ⁻⁶	NA		X	X	
6542-20-035	ThO ₂ (o)	462	BISO	80	1.20	Yes	NA	NA	NA	NA	NA	83	1.84	3.6	1.15	163	810	3.50	NA	57.40	NA	ND	1 x 10 ⁻⁶	NA		X	X	
6542-25-015	ThO ₂ (o)	489	BISO	79	1.21	Yes	NA	NA	NA	NA	NA	76	1.82	7.0	1.11	155	826	3.57	NA	58.20	NA	ND	1 x 10 ⁻⁶	NA		X	X	
Inert Particles																												
6351-01-020	C	253	TRISO	NA	NA	Yes	29	1.76	1.09	29	3.22	36	1.75	ND	1.12	94	573	1.83	NA	NA	NA	NA	NA	NA		X		

(a) TRISO denotes a coating design with an SiC layer and BISO denotes a particle with no SiC layer.

(b) Optical anisotropy factor, relative units.

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

(d) Determined by leach test.

(e) Determined by burn-leach test.

(f) Fissile kernels 93.15% enriched.

(g) Calculated, i.e., determined by subtracting the inner isotropic thickness from the measured buffer plus inner isotropic thickness.

(h) NA = not applicable.

(i) Kernels thorium doped.

(j) ND = not determined.

(k) Estimated.

(l) Monround particles separated from parent batch 6151-00-045 using a vibrating table. Parameters listed are for the parent batch.

(m) Density and size separated from parent batch 4252-06-010.

(n) Derived from a steam demitrated thorium nitrate solution and sintered in a rotary kiln.

(o) Fabricated by oxalate process.

TABLE 2-2
GENERAL DESCRIPTION OF COATED PARTICLES TESTED IN CAPSULE P135

Particle Batch Data Retrieval Number	Kernel		Coating													Total Coating Thick- ness (μm)	Total Coated Particle										Particle Disposition,	
	Type	Mean Diameter (μm)	Type (a)	Buffer		Seal Coat	Inner Isotropic PyC			SiC		Outer Isotropic PyC					Mean Diameter (μm)	Density (g/cm^3)	Metal Loading		Fission Gas Release (c)	Exposed Heavy Metal (d)		Fraction Defective SiC Coatings (e) (g U/g U)	Used In			
				Thick- ness (μm)	Densit (g/cm^3)		Thick- ness (μm)	Density (g/cm^3)	OPTAF (b)	Thick- ness (μm)	Density (g/cm^3)	Coating Rate ($\mu\text{m}/\text{min}$)	OPTAF (b)	Uranium (wt %)	Thorium (wt %)				Uranium (g U/g U)	Thorium (g Th/g Th)		Fuel Rods	Unbonded Test					
Fissile Particles (f)																												
4161-01-021	UC ₂	189	TRISO	82	1.31	Yes	29	1.91	1.16	30	3.20	32	1.82	3.4	1.07	173	541	2.47	20.8	NA (g)	5 x 10 ⁻⁶	6 x 10 ⁻⁷	ND (h)	1 x 10 ⁻⁴			X	
4163-00-011	(Th,U)O ₂	253	TRISO	67 (i)	1.17	Yes	32	1.78	1.14	36	3.20	72	1.77	4.1	1.09	209	690	2.37	9.2	9.20	3 x 10 ⁻⁹	6 x 10 ⁻⁶	3 x 10 ⁻⁵	2 x 10 ⁻⁵			X	
5466-37	(Th,U)O ₂	258	TRISO	101 (i)	1.12	Yes	29	1.87	1.07	30	3.18	31	1.79	6.0	1.10	197	619	2.37	12.7	12.30	1 x 10 ⁻⁶	2 x 10 ⁻⁷	6 x 10 ⁻⁵	5 x 10 ⁻⁵	X		X	
6151-00-035	UC ₂ (j)	201	TRISO	87 (i)	1.07	Yes	33	1.92	1.22	27	3.20	37	1.85	3.9	1.17	191	577	2.29	18.9	0.20 (k)	3 x 10 ⁻⁷	4 x 10 ⁻⁷	ND	2 x 10 ⁻⁶	X		X	
6151-00-045	UC ₂	196	TRISO	85 (i)	1.15	Yes	35	1.93	1.17	29	3.22	40	1.81	3.8	1.13	192	578	2.33	17.8	NA	2 x 10 ⁻⁷	1 x 10 ⁻⁷	ND	2 x 10 ⁻⁶	X			
6151-02-025	UC ₂ (j)	204	TRISO	99 (i)	1.07	Yes	33	1.92	1.25	27	3.22	43	1.50	0.7	1.13	202	608	2.19	18.4	0.22	2 x 10 ⁻⁶	1 x 10 ⁻⁶	8 x 10 ⁻⁴	2 x 10 ⁻⁵			X	
6151-03-015	UC ₂ (j)	199	TRISO	94 (i)	1.10	Yes	31	1.90	1.24	29	3.22	45	1.77	1.5	1.12	204	594	2.40	17.6	0.23	2 x 10 ⁻⁶	2 x 10 ⁻⁶	7 x 10 ⁻⁴	1 x 10 ⁻⁴			X	
6151-08-015	UC ₂ (j)	202	TRISO	95 (g)	1.07	Yes	33	1.92	1.25	26	3.21	36	1.76	3.6	1.09	190	593	2.28	18.6	0.20 (k)	1 x 10 ⁻⁷	9 x 10 ⁻⁷	ND	2 x 10 ⁻⁶	X		X	
6151-09-015	UC ₂ (j)	197	TRISO	95 (i)	1.15	Yes	31	1.94	1.16	29	3.22	39	1.94	3.6	1.06	204	587	2.30	17.8	0.20 (k)	9 x 10 ⁻⁸	4 x 10 ⁻⁷	ND	4 x 10 ⁻⁶	X			
6155-01-020	(Th,U)O ₂	502	TRISO	86 (i)	1.24	Yes	32	1.93	1.17	34	3.22	42	1.81	3.4	1.07	214	885	3.34	5.2	43.60	6 x 10 ⁻⁷	6 x 10 ⁻⁷	7 x 10 ⁻⁵	1 x 10 ⁻⁵			X	
Fertile Particles																												
4252-02-010	ThO ₂ (i)	488	BISO	83	1.08	Yes	NA	NA	NA	NA	NA	73	1.83	4.0	1.16	156	805	3.51	NA	58.80	NA	ND	2 x 10 ⁻⁵	NA			X	
4252-06-010	ThO ₂ (i)	516	BISO	85	1.10	Yes	NA	NA	NA	NA	NA	76	1.82	5.7	1.14	161	841	3.54	NA	58.90	NA	ND	7 x 10 ⁻⁴	NA			X	
6252-00-025	ThO ₂ (i)	512	TRISO	60	1.16	Yes	30	1.87	1.25	31	3.21	42	1.81	4.2	1.24	164	833	3.77	NA	54.00	NA	ND	9 x 10 ⁻⁷	3 x 10 ^{-4 (m)}			X	
6542-01-010	ThO ₂ (i)	500	BISO	79	1.08	Yes	NA	NA	NA	NA	NA	85	1.80	10.0	1.06	164	828	3.59	NA	56.50	NA	ND	6 x 10 ⁻⁶	NA	X			
6542-01-020	ThO ₂ (a)	504	BISO	81	1.17	Yes	NA	NA	NA	NA	NA	74	1.82	2.7	1.10	155	813	3.55	NA	57.50	NA	ND	6 x 10 ⁻⁶	NA	X			
6542-02-020	ThO ₂ (i)	481	BISO	87	1.06	Yes	NA	NA	NA	NA	NA	72	1.91	8.5	1.06	159	796	3.59	NA	56.20	NA	ND	2 x 10 ⁻⁵	NA	X			
6542-09-010	ThO ₂ (i)	511	BISO	84	1.06	Yes	NA	NA	NA	NA	NA	75	1.93	5.0	1.13	159	822	3.56	NA	57.60	NA	ND	1 x 10 ⁻⁴	NA	X			
6542-12-025	ThO ₂ (a)	505	BISO	84	1.10	Yes	NA	NA	NA	NA	NA	70	1.79	6.7	1.22	154	816	3.58	NA	58.00	NA	ND	5 x 10 ⁻⁷	NA			X	
6542-16-010	ThO ₂ (a)	502	BISO	39	1.08	Yes	NA	NA	NA	NA	NA	55	ND	3.7	1.07	94	687	4.84	NA	70.40	NA	ND	1 x 10 ⁻⁴	NA			X	
6542-17-010	ThO ₂ (a)	502	BISO	44	0.95	Yes	NA	NA	NA	NA	NA	122	1.86	2.8	1.27	166	829	3.55	NA	54.50	NA	ND	6 x 10 ⁻⁷	NA			X	
6542-19-015	ThO ₂ (a)	486	BISO	81	1.15	Yes	NA	NA	NA	NA	NA	77	1.78	7.5	1.16	158	824	3.55	NA	58.90	NA	ND	4 x 10 ⁻⁶	NA	X			
6542-19-016 (o,p)	ThO ₂ (a)	486	BISO	81	1.15	Yes	NA	NA	NA	NA	NA	77	1.78	7.5	1.16	158	824	3.55	NA	58.90	NA	ND	4 x 10 ⁻⁶	NA			X	
6542-20-035	ThO ₂ (a)	462	BISO	80	1.20	Yes	NA	NA	NA	NA	NA	83	1.84	3.6	1.15	163	810	3.50	NA	57.40	NA	ND	1 x 10 ⁻⁶	NA	X			
6542-21-015	ThO ₂ (a)	492	BISO	79	1.13	Yes	NA	NA	NA	NA	NA	82	1.73	7.8	1.10	161	833	3.46	NA	57.40	NA	ND	4 x 10 ⁻⁶	NA			X	
6542-21-016 (p,q)	ThO ₂ (a)	492	BISO	79	1.13	Yes	NA	NA	NA	NA	NA	82	1.73	7.8	1.10	161	833	3.46	NA	57.40	NA	ND	4 x 10 ⁻⁶	NA			X	
6542-22-015	ThO ₂ (a)	503	BISO	85	1.12	Yes	NA	NA	NA	NA	NA	81	1.80	4.0	1.07	166	831	3.50	NA	58.00	NA	ND	1 x 10 ⁻⁶	NA			X	
6542-22-025	ThO ₂ (a)	500	BISO	81	1.15	Yes	NA	NA	NA	NA	NA	80	1.81	7.6	1.10	161	833	3.53	NA	57.80	NA	ND	3 x 10 ⁻⁶	NA			X	
6542-23-025	ThO ₂ (a)	486	BISO	82	1.19	Yes	NA	NA	NA	NA	NA	73	1.89	5.7	1.09	155	800	3.58	NA	57.50	NA	ND	4 x 10 ⁻⁶	NA			X	
6542-24-015	ThO ₂ (a)	511	BISO	86	1.13	Yes	NA	NA	NA	NA	NA	83	1.94	6.6	1.11	169	844	3.55	NA	57.10	NA	ND	1 x 10 ⁻⁶	NA			X	
Inert Particles																												
6351-01-020	C	253	TRISO	NA	NA	Yes	29	1.76	1.09	29	3.22	36	1.75	1.05	1.12	94	573	1.83	NA	NA	NA	NA	NA	NA	X			

(a) TRISO denotes a coating design with an SiC layer and BISO denotes a particle with no SiC layer.

(b) Optical anisotropy factor, relative units.

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

(d) Determined by leach test.

(e) Determined by burn-leach test.

(f) Fissile kernels 93.15% enriched.

(g) NA = not applicable.

(h) ND = not determined.

(i) Calculated, i.e., determined by subtracting the inner isotropic thickness from the measured buffer plus inner isotropic thickness.

(j) Kernels thorium doped.

(k) Estimated.

(l) Derived from a steam denitrated thorium nitrate solution and sintered in a rotary kiln.

(m) Fraction defective SiC coatings in g Th/g Th.

(n) Fabricated by oxalate process.

(o) Density separated from parent batch 6542-19-015.

(p) Parameters listed are for the parent batch, except for particle density.

(q) Density separated from parent batch 6542-21-015.

TABLE 2-3
TRISO COATED FISSILE PARTICLE VARIABLES TESTED IN CAPSULES P13R AND P13S

Primary Variables	Batch Data Retrieval Number	Comments ^(a)
Reference TRISO coated UC ₂ ^(b)	6151-00-035	OPyC density 1.85 g/cm ³ , coating rate 4.0 μm/min
OPyC density/coating rate	6151-00-010	OPyC density 1.80 g/cm ³ , coating rate 1.1 μm/min
Faceting and OPyC density/coating rate	6151-00-025	OPyC density 1.88 g/cm ³ , coating rate 1.4 μm/min
Faceting and OPyC density/coating rate	6151-00-045	OPyC density 1.81 g/cm ³ , coating rate 3.8 μm/min
Faceting and OPyC density/coating rate	6151-00-046	Same as 6151-00-045, except the material was tabled to isolate the most severely faceted particles
OPyC density/coating rate	6151-08-015	OPyC density 1.76 g/cm ³ , coating rate 3.6 μm/min
OPyC density/coating rate	6151-02-025	OPyC density measurements vary between 1.50 and 1.66, coating rate 0.7 μm/min
OPyC density/coating rate	6151-01-015	OPyC density 1.90 g/cm ³ , coating rate 0.7 μm/min
OPyC density/coating rate	6151-09-015	OPyC density 1.94 g/cm ³ , coating rate 3.6 μm/min
OPyC density	6151-09-025	OPyC density 1.95 g/cm ³
IPyC coating rate	6151-04-015	IPyC coating rate 0.7 μm/min
Interrupted SiC layer	6151-03-015	
Interrupted SiC layer	4161-01-021	
TRISO coated (Th,U)O ₂	4163-00-011	Th/U ratio of 1/1; 253-μm kernel; 60-μm buffer
TRISO coated (Th,U)O ₂	5466-37	Th/U ratio of 1/1; 258-μm kernel; 107-μm buffer
TRISO coated (Th,U)O ₂	6155-01-020	Th/U ratio of 8/1; 502-μm kernel

^(a) Unless otherwise noted, all particles have 200-μm UC₂ kernels made via the VSM process.

^(b) Reference coatings₃ are: buffer thickness 100 μm, density 1.1 g/cm³; seal coat; IPyC thickness 30 μm, density 1.95 g/cm³; SiC thickness 25 μm, density >3.18 g/cm³; OPyC thickness 40 μm, density 1.80 g/cm³.

TABLE 2-4
FISSILE PARTICLE COATING PROCESS DATA

Batch Number/ Particle Type	Coatings	Charge In (kg)	Coater Size (mm)	Distributor Type	Coating Gases			Batch Number/ Particle Type	Coatings	Charge In (kg)	Coater Size (mm)	Distributor Type	Coating Gases		
					Inert	Diluent	Deposition						Inert	Diluent	Deposition
4161-01-021 UC ₂ TRISO	Buffer	1.500	127.0	Flat frit	Ar	Ar	C ₂ H ₂	6151-02-025 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	SiC	3.1430	127.0	Cone	Ar	H ₂	CH ₃ SiCl ₃		SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.625	127.0	Flat frit	N ₂	N ₂	C ₃ H ₆		OPyC	1.3331	127.0	Multi-nozzle	Ar	Ar	C ₃ H ₆
4163-00-011 (Th,U) ₂ TRISO	Buffer	1.2631	127.0	Flat frit	Ar	Ar	C ₂ H ₂	6151-03-015 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	SiC	2.0693	127.0	Cone	Ar	H ₂	CH ₃ SiCl ₃		SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.800	127.0	Flat frit	N ₂	N ₂	C ₃ H ₆		OPyC	1.3346	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆
5466-37 (Th,U) ₂ TRISO	Buffer	29	38.1	Cone	He	He	C ₂ H ₂	6151-04-015 UC ₂ TRISO	Buffer	401	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂
	Seal (a)	38.1	38.1	Cone	He	He	C ₂ H ₂		Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	IPyC	38.9	38.1	Cone	He	He	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	SiC	49	38.1	Cone	He	H ₂	CH ₃ SiCl ₃		SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	65	38.1	Cone	He	He	C ₃ H ₆		OPyC	1.4905	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆
6151-00-010 UC ₂ TRISO	Buffer	400	127.0	Frit	Ar	Ar	C ₂ H ₂	6151-08-015 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	SiC	962	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃		SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.3437	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		OPyC	1.1075	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6151-00-025 UC ₂ TRISO	Buffer	400	127.0	Frit	Ar	Ar	C ₂ H ₂	6151-09-015 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆
	SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃		SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.3882	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		OPyC	1.058	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6151-00-035 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂	6151-09-025 UC ₂ TRISO	Buffer	400	127.0	Frit	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆
	SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃		SiC (a)	0.962	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.0592	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	0.846	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6151-00-045 UC ₂ TRISO	Buffer	400	127.0	Frit	Ar	Ar	C ₂ H ₂	6155-01-020 (Th,U) ₂ TRISO	Buffer	0.9613	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		Seal (a)	127.0	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	IPyC (a)	127.0	127.0	Frit	Ar	Ar	C ₃ H ₆		IPyC (a)	127.0	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃		SiC	1.269	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃
	OPyC	1.1600	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	1.468	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6151-01-015 UC ₂ TRISO	Buffer	400	127.0	Frit center nozzle	Ar	Ar	C ₂ H ₂								
	Seal (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆								
	IPyC (a)	127.0	127.0	Frit center nozzle	Ar	Ar	C ₃ H ₆								
	SiC (b)	127.0	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃								
	OPyC	1.354	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆								

(a) Continuous from buffer coating run.

(b) Data not available.

TABLE 2-5
FERTILE PARTICLE VARIABLES TESTED IN CAPSULES P13R AND P13S

Primary Variables	Batch Data Retrieval Number	Comments (a)
Reference	4252-02-010	Previously irradiated in HT-12 through -15
	4252-06-010	Previously irradiated in HT-12 through -15
Coating design	6542-16-010	Buffer thickness 39 μm , density 1.08 g/cm^3 ; OPyC thickness 55 μm , density 1.81 g/cm^3 ; irradiated in HT-17 through -19
	6542-17-010	Buffer thickness 44 μm , density 0.95 g/cm^3 ; OPyC thickness 122 μm , density 1.86 g/cm^3 ; irradiated in HT-17 through -19
OPyC density and coating rate	6542-01-010	OPyC density 1.80 g/cm^3 , coating rate 10.0 $\mu\text{m/min}$; irradiated in P13Q and HT-17 through -19
	6542-01-020	OPyC density 1.82 g/cm^3 , coating rate 2.72 $\mu\text{m/min}$; irradiated in HT-17 through -19
	6542-02-020	OPyC density 1.91 g/cm^3 , coating rate 8.5 $\mu\text{m/min}$; irradiated in HT-17 through -19
	6542-02-030	OPyC density 1.89 g/cm^3 , coating rate 2.16 $\mu\text{m/min}$; irradiated in HT-17 through -19
	6542-09-010	OPyC density 1.93 g/cm^3 , coating rate 5.0 $\mu\text{m/min}$
	6542-19-015	OPyC density 1.78 g/cm^3
Constant total particle density	4252-06-018(b)	Density separated from 4252-06-010; OPyC density 1.82 g/cm^3
	6542-19-016	Density separated from 6542-19-010; OPyC density, 1.78 g/cm^3
	6542-21-016	Density separated from 6542-21-010; OPyC density 1.73 g/cm^3
Faceted coatings	6542-20-025	OPyC density 1.80 to 1.90 g/cm^3
	6542-20-035	OPyC density 1.80 to 1.90 g/cm^3
Mixed-gas OPyC layer	6542-21-015	OPyC density 1.73 g/cm^3
	6542-22-015	OPyC density 1.80 g/cm^3 , coating rate 2 to 3 $\mu\text{m/min}$
	6542-22-025	OPyC density 1.81 g/cm^3 , coating rate 7 to 10 $\mu\text{m/min}$
	6542-23-025	OPyC density 1.89 g/cm^3
	6542-24-015	OPyC density 1.94 g/cm^3
N ₂ carrier gas buffer	6542-12-025	
No seal coat	6542-11-015	OPyC density 1.80 to 1.90 g/cm^3
	6542-18-015	OPyC density 1.80 to 1.90 g/cm^3
Nonround kernels	6542-25-015	OPyC density 1.80 to 1.90 g/cm^3
TRISO coating	6252-00-025	Comparison of TRISO and BISO coated ThO ₂

(a) Unless otherwise indicated, desired average particle attributes are: kernel diameter 480 to 520 μm , density 9.5 to 10.0 g/cm^3 ; buffer thickness 80 to 90 μm , density 1.0 to 1.2 g/cm^3 ; seal coat; OPyC thickness 70 to 80 μm , density 1.80 to 1.90 g/cm^3 , OPTAF ≤ 1.20 .

(b) Also screened to a specific size range.

TABLE 2-6
FERTILE PARTICLE COATING PROCESS DATA

Batch Number/ Particle Type	Coatings	Charge In (kg)	Coater Size (mm)	Distributor Type	Coating Gases			Batch Number/ Particle Type	Coatings	Charge In (kg)	Coater Size (mm)	Distributor Type	Coating Gases		
					Inert	Diluent	Deposition						Inert	Diluent	Deposition
4152-02-010 ThO ₂ BISO	Buffer	2.9081	127.0	Flat frit	Ar	Ar	C ₂ H ₂	6542-18-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		Seal (b)	--	--	--	--	--	--
	OPyC	3.3692	127.0	Flat frit	N ₂	N ₂	C ₃ H ₆		OPyC	(b)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
4252-06-010 ThO ₂ BISO	Buffer	2.261	127.0	Flat frit	Ar	Ar	C ₂ H ₂	6542-19-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	2.5638	127.0	Flat frit	N ₂	N ₂	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6252-00-025 ThO ₂ TRISO	Buffer	1.000	127.0	Frit nozzles	Ar	Ar	C ₂ H ₂	6542-20-025 ThO ₂ BISO	Buffer	4.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Frit nozzles	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	IPyC	(a)	127.0	Frit nozzles	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	SiC	1.1846	127.0	Std. FOD cone	H ₂	H ₂	CH ₃ SiCl ₃		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6542-01-010 ThO ₂ BISO	Buffer	2.000	127.0	Nozzles porous cone	Ar	Ar	C ₃ H ₆	6542-20-035 ThO ₂ BISO	Buffer	4.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	2.3349	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	2.3349	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6542-01-020 ThO ₂ BISO	Buffer	2.000	127.0	Multi-nozzle frit	Ar	Ar	C ₂ H ₂	6542-21-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
6542-02-020 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂	6542-22-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	2.3476	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
	OPyC	2.3476	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-02-030 ThO ₂ BISO	Buffer	2.000	127.0	Multi-nozzle frit	Ar	Ar	C ₂ H ₂	6542-22-025 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-09-010 ThO ₂ BISO	Buffer	2.4667	127.0	Flat frit	Ar	Ar	C ₂ H ₂	6542-23-025 ThO ₂ BISO	Buffer	3.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Flat frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	2.7932	127.0	Flat frit	N ₂	N ₂	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-11-015 ThO ₂ BISO	Buffer	2.000	127.0	Multi-nozzle frit	Ar	Ar	C ₂ H ₂	6542-24-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (b)	--	--	--	--	--	--		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-nozzle frit	Ar	Ar	C ₂ H ₂		OPyC	1.163	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-12-025 ThO ₂ BISO	Buffer	2.000	127.0	Multi-nozzle frit	N ₂	N ₂	C ₂ H ₂	6542-25-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Multi-nozzle frit	N ₂	N ₂	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-nozzle frit	N ₂	N ₂	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-16-010 ThO ₂ BISO	Buffer	2.000	127.0	Multi-nozzle frit	Ar	Ar	C ₂ H ₂	6542-25-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-nozzle frit	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆
6542-17-010 ThO ₂ BISO	Buffer	2.000	127.0	Multi-hole frit	Ar	Ar	C ₂ H ₂	6542-25-015 ThO ₂ BISO	Buffer	2.000	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂
	Seal (a)	127.0	127.0	Multi-hole frit	Ar	Ar	C ₃ H ₆		Seal	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₃ H ₆
	OPyC	(a)	127.0	Multi-hole frit	Ar	Ar	C ₃ H ₆		OPyC	(a)	127.0	Moly-nozzle porous cone	Ar	Ar	C ₂ H ₂ /C ₃ H ₆

(a) Continuous from buffer coating run.

(b) No seal coat.

TABLE 2-7
DESCRIPTION OF FUEL RODS(a) TESTED IN CAPSULE P13R

Capsule Position	Data Retrieval Number (7161-004-)	Coated Particles (b)				Shim Particles		Particle Packing (e,f) (Z)	Batch Number	Filler (g) (wt %)	Apparent Density (e,f) (g/cm ³)	Macro- porosity (h,i) (Z)	Coke Content (f)		Fuel Loading Uniformity (k)		Exposed Heavy Metal (R,l) (g heavy metal/ g heavy metal)	Fission Gas Release (m)
		Fissile (c)		Fertile (d)									Yield (i) (Z)	Density (g/cm ³)	11-215	11-212		
		Type	Data Retrieval Number	Type	Data Retrieval Number													
		Type	Data Retrieval Number	Type	Data Retrieval Number													
Cell 1																		
A (n)	01-5	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I (o)	FO 147	--	MO 269	37	--	31	--	--	1.05	1.05	<8 x 10 ⁻⁶	2 x 10 ⁻⁶
B	02-6	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-N (p)	FO 82	56	MO 269	34	0.71	40	39	0.12	1.00	1.02	<8 x 10 ⁻⁶	1 x 10 ⁻⁶
C	03-5	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 136	55	MO 269	33	0.67	42	36	0.13	1.03	1.06	<8 x 10 ⁻⁶	1 x 10 ⁻⁶
D	04-6	UC ₂	6151-00-035	ThO ₂	6542-01-020	1099-I	FO 136	57	MO 269	34	0.68	35	36	0.14	1.04	1.12	<9 x 10 ⁻⁶	3 x 10 ⁻⁶
E (n)	05-6	(Th,U) ₂ O ₇	5466-37	ThO ₂	6542-02-020	1099-I	FO 147	--	MO 269	34	--	32	--	--	1.03	1.07	<9 x 10 ⁻⁶	4 x 10 ⁻⁶
Cell 2																		
A (n)	06-5	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I	FO 147	--	MO 269	32	--	37	--	--	1.05	1.11	<9 x 10 ⁻⁶	3 x 10 ⁻⁶
B	07-5	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-I	FO 136	56	MO 270	42	0.77	28	41	0.13	1.08	1.10	<9 x 10 ⁻⁶	3 x 10 ⁻⁶
C	08-13	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 148	56	MO 269	31	0.65	37	28	0.11	1.02	1.04	<9 x 10 ⁻⁶	4 x 10 ⁻⁶
D	09-5	UC ₂	6151-00-035	ThO ₂	6542-01-020	Lonza	FO 149	57	MO 267	33	0.66	36	31	0.12	1.02	1.04	<1 x 10 ⁻⁵	3 x 10 ⁻⁶
E (n)	10-5	(Th,U) ₂ O ₇	5466-37	ThO ₂	6542-01-010	1099-I	FO 147	--	MO 269	ND (q)	--	37	--	--	1.01	1.08	<1 x 10 ⁻⁵	6 x 10 ⁻⁶
Cell 5																		
A (n)	16-5	(Th,U) ₂ O ₇	5466-37	ThO ₂	6542-01-020	1099-I	FO 147	--	MO 269	35	--	37	--	--	1.01	1.00	<6 x 10 ⁻⁶	4 x 10 ⁻⁶
B	17-5	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-N	FO 82	56	MO 269	34	0.71	35	40	0.14	1.03	1.07	<7 x 10 ⁻⁶	2 x 10 ⁻⁶
C	18-6	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 136	55	MO 269	30	0.69	35	35	0.13	1.09	1.06	<7 x 10 ⁻⁶	3 x 10 ⁻⁶
D	19-7	UC ₂	6151-08-015	ThO ₂	6542-19-015	1099-I	FO 136	56	MO 269	33	0.68	42	37	0.15	1.03	1.09	<7 x 10 ⁻⁶	2 x 10 ⁻⁶
E (n)	20-6	UC ₂	6151-09-015	ThO ₂	6542-09-010	1099-I	FO 147	--	MO 269	32	--	27	--	--	1.04	1.08	<6 x 10 ⁻⁶	3 x 10 ⁻⁶
Cell 6																		
A (n)	21-5	UC ₂	6151-08-015	ThO ₂	6542-19-015	1099-I	FO 147	--	MO 269	33	--	34	--	--	1.07	1.10	<9 x 10 ⁻⁶	4 x 10 ⁻⁶
B	22-5	UC ₂	6151-09-015	ThO ₂	6542-09-010	1099-I	FO 136	56	MO 269	32	0.70	41	38	0.14	1.06	1.09	<8 x 10 ⁻⁶	2 x 10 ⁻⁶
C	23-5	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 136	54	MO 271	30	0.57	43	25	0.09	1.07	1.12	2 x 10 ^{-3(r)}	2 x 10 ⁻⁶
D	24-5	UC ₂	6151-00-035	ThO ₂	6542-02-020	Lonza	FO 149	56	MO 267	31	0.67	40	31	0.12	1.10	1.13	<9 x 10 ⁻⁶	6 x 10 ⁻⁶
E (n)	25-5	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I	FO 147	--	MO 269	33	--	36	--	--	1.03	1.03	<1 x 10 ⁻⁵	3 x 10 ⁻⁶

(a) All rods are approximately 1.25 cm (0.493 in.) in diameter by 1.90 cm (0.75 in.) long.

(b) All rods except cell 1, position B and position C, contain TRISO coated carbon inert particles (batch number 6351-02-020).

(c) All fissile particles are TRISO coated.

(d) All fertile particles are BISO coated.

(e) Calculated from fired rod and mean particle parameters.

(f) Not meaningful to determine value for thermocouple rods due to some uncertainty in green and fired rod parameters.

(g) Measurement made on companion green rod.

(h) Measurement made on companion fired rod.

(i) Determined from metallographic cross section.

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

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

(l) Determined by hydrolysis test. "<" denotes the amount is below the limit of detection of the apparatus. If a detectable quantity, the value is corrected for total conversion of ThO₂ to ThC₂.

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

(n) Thermocouple rod - center hole [approximately 0.34 cm (0.13 in.) in diameter] through the entire length of the rod.

(o) I signifies impregnated with furfuryl alcohol.

(p) N signifies nonimpregnated.

(q) ND = not determined.

(r) Second rod from this batch had a value of <9 x 10⁻⁶.

TABLE 2-8
DESCRIPTION OF FUEL RODS(a) TESTED IN CAPSULE P13S

Capsule Position	Data Retrieval Number (7161-004-)	Coated Particles ^(b)				Shim Particles		Particle Packing ^(e,f) (%)	Batch Number	Matrix						Fuel Loading Uniformity ^(k) U-235 Th-232		Exposed Heavy Metal ^(g,l) (g heavy metal/ g heavy metal)	Fission Gas Release ^(m)
		Fissile ^(c)		Fertile ^(d)						Filler ^(R) (wt %)	Apparent Density ^(e,f) (g/cm ³)	Macro- porosity ^(h,i) (%)	Coke Content ^(f)						
		Type	Data Retrieval Number	Type	Data Retrieval Number	Type	Batch Number						Yield ^(j) (%)	Density (g/cm ³)					
Cell 1																			
A ⁽ⁿ⁾	01-7	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I ^(o)	FO 147	--	MO 269	37	--	31	--	--	1.07	1.10	<8 x 10 ⁻⁶	2 x 10 ⁻⁶	
B	02-5	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-N ^(p)	FO 82	56	MO 269	34	0.67	40	36	0.12	1.07	1.07	<8 x 10 ⁻⁶	1 x 10 ⁻⁶	
C	03-6	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 136	55	MO 269	33	0.66	42	34	0.13	1.10	1.07	<8 x 10 ⁻⁶	1 x 10 ⁻⁶	
D	04-5	UC ₂	6151-00-035	ThO ₂	6542-01-020	1099-I	FO 136	57	MO 269	34	0.68	35	36	0.14	1.11	1.09	<9 x 10 ⁻⁶	2 x 10 ⁻⁶	
E ⁽ⁿ⁾	05-5	(Th,U)O ₂	5466-37	ThO ₂	6542-02-020	1099-I	FO 147	--	MO 269	34	--	32	--	--	1.00	1.02	<9 x 10 ⁻⁶	2 x 10 ⁻⁶	
Cell 2																			
A ⁽ⁿ⁾	11-6	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I	FO 148	--	MO 269	29	--	36	--	--	1.02	1.08	<9 x 10 ⁻⁶	2 x 10 ⁻⁷	
B	12-5	UC ₂	6151-09-015	ThO ₂	6542-09-010	1099-I	FO 136	55	MO 269	32	0.70	32	35	0.14	1.02	1.07	6 x 10 ^{-3(q)}	2 x 10 ⁻⁶	
C	13-6	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 148	53	MO 271	31	0.59	35	25	0.10	1.10	1.10	<9 x 10 ⁻⁶	3 x 10 ⁻⁶	
D	14-5	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-N	FO 82	56	MO 269	36	0.69	36	36	0.14	1.00	1.02	<1 x 10 ⁻⁵	2 x 10 ⁻⁶	
E ⁽ⁿ⁾	15-5	UC ₂	6151-08-015	ThO ₂	6542-19-015	1099-I	FO 147	--	MO 269	36	--	26	--	--	1.05	1.03	<1 x 10 ⁻⁵	5 x 10 ⁻⁶	
Cell 5																			
A ⁽ⁿ⁾	16-6	(Th,U)O ₂	5466-37	ThO ₂	6542-01-020	1099-I	FO 147	--	MO 269	35	--	37	--	--	1.02	1.01	<6 x 10 ⁻⁶	4 x 10 ⁻⁶	
B	17-6	UC ₂	6151-00-035	ThO ₂	6542-02-020	1099-N	FO 82	56	MO 269	34	0.67	35	38	0.14	1.08	1.06	<7 x 10 ⁻⁶	1 x 10 ⁻⁶	
C	18-5	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 136	55	MO 269	30	0.68	35	35	0.13	1.07	1.10	<7 x 10 ⁻⁶	2 x 10 ⁻⁶	
D	19-5	UC ₂	6151-08-015	ThO ₂	6542-19-015	1099-I	FO 136	56	MO 269	33	0.73	42	36	0.15	1.11	1.09	<7 x 10 ⁻⁶	3 x 10 ⁻⁶	
E ⁽ⁿ⁾	20-5	UC ₂	6151-09-015	ThO ₂	6542-09-010	1099-I	FO 147	--	MO 269	32	--	27	--	--	1.09	1.11	<6 x 10 ⁻⁶	3 x 10 ⁻⁶	
Cell 6																			
A ⁽ⁿ⁾	26-5	UC ₂	6151-00-035	ThO ₂	6542-01-020	1099-N	FO 82	--	MO 269	33	--	40	--	--	1.06	1.10	<9 x 10 ⁻⁶	3 x 10 ⁻⁶	
B	27-5	(Th,U)O ₂	5466-37	ThO ₂	6542-02-020	1099-I	FO 136	55	MO 269	32	0.71	29	35	0.15	1.00	1.08	<8 x 10 ⁻⁶	1 x 10 ⁻⁶	
C	28-13	UC ₂	6151-00-035	ThO ₂	6542-01-010	1099-I	FO 148	55	MO 269	29	0.64	38	31	0.12	1.03	1.09	<8 x 10 ⁻⁶	4 x 10 ⁻⁶	
D	29-5	UC ₂	6151-00-035	ThO ₂	6542-01-020	1099-I	FO 136	57	MO 270	44	0.79	32	39	0.13	1.06	1.11	<9 x 10 ⁻⁶	1 x 10 ⁻⁶	
E ⁽ⁿ⁾	30-5	UC ₂	6151-00-045	ThO ₂	6542-20-035	1099-I	FO 147	--	MO 269	ND ^(r)	--	30	--	--	1.03	1.10	<1 x 10 ⁻⁵	2 x 10 ⁻⁶	

(a) All rods are approximately 1.25 cm (0.493 in.) in diameter by 1.90 cm (0.75 in.) long.

(b) All rods except cell 1, position B and position C, contain TRISO coated carbon inert particles (batch number 6351-02-020).

(c) All fissile particles are TRISO coated.

(d) All fertile particles are BISO coated.

(e) Calculated from fired rod and mean particle parameters.

(f) Not meaningful to determine value for thermocouple rods due to some uncertainty in green and fired rod parameters.

(g) Measurement made on companion green rod.

(h) Measurement made on companion fired rod.

(i) Determined from metallographic cross section.

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

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

(l) Determined by hydrolysis test. "<" denotes the amount is below the limit of detection of the apparatus. If a detectable quantity, the value is corrected for total conversion of ThO₂ to ThC₂.

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

(n) Thermocouple rod - center hole [approximately 0.34 cm (0.13 in.) in diameter] through the entire length of the rod.

(o) I signifies impregnated with furfuryl alcohol.

(p) N signifies nonimpregnated.

(q) Second rod from this batch had a value of <9 x 10⁻⁶.

(r) ND = not determined.

TABLE 2-9
FUEL ROD VARIABLES^(a) TESTED IN CAPSULES P13R AND P13S

Rod Location	Fuel Particles			Shim		Matrix		Firing Conditions
	Fissile		Fertile ^(b) Coatings	Particles	Loading	Filler	Additive	
	Kernel	Coatings						
Cell 1								
P13R & P13S								
A ^(c)	WESL ^(d)	Faceted	Faceted	Ref ^(e)	Low	Ref	Ref	Ref
B	WESL	WESL	WESL	Nonimpregnated	High	Ref	Ref	Ref
C	WESL	WESL	WESL	Ref	High	Ref	Ref	Ref
D	WESL	WESL	WESL	Ref	Ref	Ref	Ref	1500°C firing
E ^(c)	(Th,U)O ₂	WESL	WESL	Ref	Ref	Ref	Ref	Ref
Cell 2								
P13R								
A ^(c)	WESL	Faceted	Faceted	Ref	Ref	Ref	Ref	1500°C firing
B	WESL	WESL	WESL	Ref	Ref	Lonza	Ref	Ref
C	WESL	WESL	WESL	Ref	Ref	Ref	Ref	Packed bed ^(f)
D	WESL	WESL	WESL	Lonza	Ref	Ref	SC003	Ref
E ^(c)	(Th,U)O ₂	WESL	WESL	Ref	High	Ref	Ref	Ref
P13S								
A ^(c)	WESL	Faceted	Faceted	Ref	Ref	Ref	Ref	N ₂ atmosphere
B	WESL	High OPyC density	High OPyC density	Ref	Ref	Ref	Ref	Ref
C	WESL	WESL	WESL	Ref	High	Ref	SC027 ^(f)	Ref
D	WESL	WESL	WESL	Nonimpregnated	Ref	Ref	Ref	Ref
E ^(c)	WESL	Low OPyC density	Low OPyC density	Ref	High	Ref	Ref	Ref
Cell 5								
P13R & P13S								
A ^(c)	(Th,U)O ₂	WESL	WESL	Ref	Ref	Ref	Ref	Ref
B	WESL	WESL	WESL	Nonimpregnated	High	Ref	Ref	Ref
C	WESL	WESL	WESL	Ref	Ref	Ref	Ref	1500°C firing
D	WESL	Low OPyC density	Low OPyC density	Ref	Ref	Ref	Ref	N ₂ atmosphere
E ^(c)	WESL	High OPyC density	High OPyC density	Ref	Low	Ref	Ref	Ref
Cell 6								
P13R								
A ^(c)	WESL	Low OPyC density	Low OPyC density	Ref	Ref	Ref	Ref	Ref
B	WESL	High OPyC density	High OPyC density	Ref	Low	Ref	Ref	Ref
C	WESL	WESL	WESL	Ref	High	Ref	SC027 ^(f)	Ref
D	WESL	WESL	WESL	Lonza	Ref	Ref	SC003	Ref
E ^(c)	WESL	Faceted	Faceted	Ref	Ref	Ref	Ref	N ₂ atmosphere
P13S								
A ^(c)	WESL	WESL	WESL	Nonimpregnated	Ref	Ref	Ref	Ref
B	(Th,U)O ₂	WESL	WESL	Ref	Low	Ref	Ref	Ref
C	WESL	WESL	WESL	Ref	High	Ref	Ref	Packed bed ^(f)
D	WESL	WESL	WESL	Ref	Ref	Lonza	Ref	Ref
E ^(c)	WESL	Faceted	Faceted	Ref	Ref	Ref	Ref	1500°C firing

(a) Reference fuel rod attributes:

Fissile particles: UC₂ TRISO, 1.8 g/cm³ OPyC, round

Fertile particles: ThO₂ BISO, 1.85 g/cm³ OPyC, round

Shim: impregnated GLCC 1099.20 to 27 vol %

Matrix: 6353 filler, A240 pitch, SC011 additive

Carbonizing and firing conditions: in-block (H-451) to 1800°C in argon

All rods hot-injected, except P13S Cell 2-C and P13R Cell 6-C, which were slug-injected.

(b) All fertile kernels are within estimated specification limits.

(c) Thermocouple rods.

(d) WESL signifies within estimated specification limits.

(e) Ref denotes LHTGR reference materials at time of sample fabrication (1973).

(f) Variable reduces the coke content in rod matrix.

Fig. 2-1. Representative preirradiation photomicrographs of fuel rods tested in capsules P13R and P13S: (a) fired at 1800°C in argon, which is the reference heat treatment for LHTGR fuel production, and (b) slug-injected fuel rod fired in packed bed of alumina

Fig. 2-2. Representative preirradiation photomicrographs of fuel rods tested in capsules P13R and P13S: (a) fired at 1500°C and (b) fired in a N₂ atmosphere. The Grafoil discs placed on the ends of each rod to prevent sticking during firing in H-451 graphite tubes remained on the rods.

(NOTE: ALL DIMENSIONS ARE IN INCHES)

Fig. 2-3. Schematic showing design of fuel rod cells 1, 2, 5, and 6 of capsules P13R and P13S

Fig. 2-4. Schematic showing design of unbonded particle cells 3 and 4 of capsules P13R and P13S

3. CAPSULE OPERATION AND EVALUATION

3.1. OPERATING HISTORY

Capsules P13R and P13S were inserted into the E-7 position of the General Electric Test Reactor (GETR) at Pleasanton, California on December 14, 1973. Irradiation was completed on October 31, 1974 after 258 days of full power operation (50 MW). Peak fast neutron fluences of 12.4×10^{21} and 12.1×10^{21} n/cm² (E > 0.18 MeV)_{HTGR} were reached by capsules P13R and P13S, respectively. The irradiation proceeded without incident for both capsules.

The capsules were irradiated for a total of 24 cycles. Each cycle was characterized by a gradual shift in the axial neutron flux profile resulting from reactor control rod withdrawal patterns. This necessitated periodic adjustments of the sweep gas mixtures to maintain the proper design temperature for each cell. Figure 3-1 shows the thermal flux spectrum as a function of the GETR control rod bank position. As can be seen in this figure, the greatest changes in flux levels occur in regions near the top and bottom of the reactor core. The cells in these regions (cells 1 and 6) required frequent gas-mixture adjustments during each cycle in order to maintain the desired design fuel temperature. The larger flux level shifts near the top of the core made it impossible to operate cell 6 of both P13R and P13S at a constant temperature throughout irradiation. Fuel rods in these cells reached their design temperature only a fraction of the time, and as a result the time-average temperatures in cell 6 of both capsules were low.

All cells in both capsules operated at a constant temperature, except cell 6 in both P13R and P13S (for the above-mentioned reasons) and cell 1

of P13S. This latter cell was designed for cyclic operation to simulate extreme thermal transient conditions resulting from load-following transients in an HTGR. Immediately following each GETR refueling startup and after the temperatures were stabilized at the nominal design value, the temperature was increased by approximately 400°C. The thermal transient test was performed at this point in the reactor cycle to take advantage of the high flux levels and hence high fuel power densities. The thermal cycle was accomplished in a series of steps of 100°C over a 20-min period. The peak fuel centerline temperature was held at ~1600°C for 1 hr and then decreased to 1075°C over a period of 2 hr, using 100°C steps and spending approximately 30 min at each step. Figure 3-2 depicts the design conditions for thermal cycling fuel in cell 1 of P13S. Measurements of fission gas release (R/B), gross gas activity, gas flow, time, and temperature were recorded during each thermal cycle test. Some problems were encountered in obtaining a fission gas release measurement at the peak of the thermal excursion during the latter part of the irradiation test because of the high activity levels (see Section 3.2).

Fuel temperatures were controlled with the aid of thermocouples which were located in each of the cells. Temperature control set point adjustments were made periodically to account for irradiation-induced emf drift in the tungsten-rhenium (W - 3% Re/W - 25% Re) thermocouples. During the initial part of the irradiation, these corrections were made using empirical data derived from earlier capsule experiments (Refs. 3-1, 3-2). During the course of the irradiation, tests were conducted to compare measured and calculated temperatures. These tests were conducted once every GETR fuel cycle at a constant rod bank position, i.e., constant fuel power. Briefly, the test procedure was as follows. Once every fuel cycle at a 25-in. rod bank position, helium gas was flowed through the cells in a 100% mixture and when temperatures stabilized, they were recorded. This was repeated for 50% and 10% helium mixtures (neon being the other gas). Some problems were encountered in the test procedure in the initial stages, so only six cycles of data were used at the beginning of the study.

The temperature and design GETR thermal fluence data from the test runs were processed by GA's UNIVAC 1108 computer. The computer program

was used to fit the temperature versus fluence data by the method of least squares to first-, second-, and third-order polynomials of the following form:

$$T_i = B_1 + B_2 F_i + B_3 F_i^2 + B_4 F_i^3 \quad i = 1, N ,$$

where B's = coefficients to be estimated from the data,

F_i = i^{th} fluence at time of test,
 T_i = thermocouple reading at the i^{th} fluence,
 N = number of data points.

Decalibration factors derived from these data were used to correct the W/Re drift during the last 90 days of irradiation. These curves and their uncertainty bands are shown in Figs. 3-3 through 3-5.

Periodic gas samples were taken from the primary gas exhaust lines of each cell and analyzed for gaseous fission products. These data were used to determine the fission gas release of the fuel in the six separate cells for each capsule, as discussed in Section 3.2.

A summary of the operating history of each cell is presented in Figs. 3-6 through 3-18. These figures show fission gas release (Kr-85m), reference thermocouple control temperature, neutron fluence, burnup, reactor power, and rod bank profiles throughout the test. More detailed discussions of the first four parameters are presented in the following sections.

3.2. IN-PILE FISSION GAS RELEASE MEASUREMENTS

The inert sweep gas used to control fuel temperatures in each cell was periodically sampled to monitor the fission gas release from the fuel specimens during irradiation. Six short-lived gaseous fission product isotopes (Kr-85m, Kr-87, Kr-88, Kr-89, 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 afford a direct means of determining fuel performance during irradiation.

Approximately 40 measurements were made on each of the six cells, with a total of about 240 measurements for each capsule. Samples were collected daily for the first week of the test and were monitored weekly for the remainder of the test. Additional samples were taken during the thermal cycle tests for cell 1 of P13S. The fission gas release was monitored just before and approximately 2 hr after each thermal cycle. Samples were also obtained at the peak of the cycle, except during the last few cycles. The total gaseous activity of these samples exceeded permissible handling levels and made it impractical to continue making these particular measurements.

The procedure for taking gas samples for the R/B determinations for the GETR was a conventional one similar to that practiced at other reactor sites. The sampling station consisted of a series of shunt lines which allowed the exhaust gas from any of the cells to flow through a line containing a calibrated length of tubing. The tubing was of known volume and could be isolated from the flow system at any time by a set of solenoid valves. The exhaust gas from the cells was normally purged through this calibrated section for 30 min prior to sampling to ensure that the system reached equilibrium. After the flow through the tube was shut off, the entire gas sample was withdrawn into an evacuated vial of special geometry for gamma counting using a Ge(Li) detector coupled with a multichannel analyzer. A standard decay time of 15 min was used to permit transport of the gas samples to the counting room. The daughter product, Rb-89, was used for determining the yield of the short-lived Kr-89 isotope, and the longer-lived isotopes, Kr-85m, Kr-87, Kr-88, Xe-135, and Xe-138, were directly counted. 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. Calibrated Hasting flowmeters were used to monitor the flow rates of the sweep gas entering the capsule.

The resulting R/B data for each of the six isotopes monitored throughout the irradiation period are presented in Figs. 3-19 and 3-20. The broad scatter in some of the data results from variations in reactor operating parameters and the inherent experimental error in making the fission gas release measurements. The gaseous releases showed an upward trend during irradiation, but the R/B values remained within expected levels for the fluence-temperature conditions of the irradiation.

3.3. ANALYSIS OF FLUX DOSIMETERS

Dosimeter wires of V - 0.216% Co, V - 0.522% Fe, and pure Fe were employed in capsules P13R and P13S to monitor long-term thermal (2200 m/sec) and fast ($E > 0.18$ MeV) neutron fluences. The wires were encapsulated in quartz ampoules and positioned along the capsules at seven axial locations at 0° and 180° radial orientations. In addition to the standard dosimetry wires, six ampoules containing purified Mn-54 were employed to evaluate the Mn-54 transmutation cross section.

After removal from the capsule, each wire was cleaned and weighed. The gamma-ray spectrum of each wire was analyzed with a Ge(Li) detector and a Sigma 2 4096-channel computer analyzer. Due to the high radiation levels of the Fe and V-Co dosimetry wires, it was necessary to dissolve and aliquot them prior to gamma-ray spectrometry analysis. Thermal and fast neutron fluxes and fluences were calculated for each axial position with the DOSE-1 computer program using the gamma-ray spectrometry analyses of the dosimeter wires. A third-order polynomial fit was made through the data points to obtain the axial thermal and fast neutron flux and fluence profiles shown in Figs. 3-21 through 3-24. A cobalt cross section of 36.8 barns (2200 m/sec) was used to calculate the thermal fluences and a burnout cross section of 11.6 ± 2.0 barns (696 ± 120 m/sec) for Mn-54, which was determined from analysis of the purified Mn-54 ampoules, was used to evaluate the fast neutron fluence levels.

Average values of the GETR fast and thermal neutron flux and fluence for each fuel specimen are given in Tables 3-1 through 3-4. These data were

obtained from the flux and fluence profiles determined for the 0° and 180° capsule orientations. The neutron spectra of the E-7 position of the GETR experimental reactor and a typical HTGR are different, especially in the 50- to 180-keV range. To be consistent with HTGR design studies, the fast fluence data were converted to equivalent fast fluences in an HTGR environment $[(E > 0.18 \text{ MeV})_{\text{HTGR}}]$ using the following relationship (Ref. 3-3):

$$\Phi (E > 0.18 \text{ MeV})_{\text{HTGR}} = 0.89 \Phi (E > 0.18 \text{ MeV})_{\text{GETR}} \quad . \quad (3-1)$$

The equivalent fast neutron fluence in an HTGR environment for each fuel specimen is given in Tables 3-1 through 3-4.

In order to verify the dosimetry analysis, four V-Fe dosimeter wires were submitted to Oak Ridge National Laboratory (ORNL) for analysis. The wires were weighed and the gamma-ray spectrum was analyzed to obtain the Mn-54 activity of each wire. The fast fluences required to produce the measured Mn-54 activities were then calculated using an ORNL computer code.

The weights of the wires obtained at GA and at ORNL are given in Table 3-5. With the exception of wire C-1632, the agreement in the measured weights was excellent. The weight determined by GA for this wire was selected as the preferred weight. The basis for this decision was that during initial fabrication, four 1-in. lengths had been cut simultaneously from parallel wires and their range of weights was 16.11 to 16.16 mg. Wire C-1632 was one of the four wires cut, and the ORNL weight of 16.639 mg was not consistent with the other weights.

The gamma-ray spectrum of each wire was analyzed to obtain the Mn-54 activity by four separate analytical chemistry groups at ORNL. The independently measured activities (in dps/mg) were corrected to June 5, 1975 and are given in Table 3-6. The average values are reported and the limits mark the range of values obtained within the four chemistry groups. The activities in $\mu\text{Ci/mg}$, corrected to October 31, 1974, are also given in

Table 3-6. The activity reported by ORNL for wire C-1632 was normalized to the correct wire weight.

The activities were determined by GA for two different cases. Case I represents the original analyses completed for capsules P13R and P13S in March 1975. The Sigma 2 computer/analyzer at GA was recalibrated in January 1976, which resulted in a reduction of approximately 6% in the computed activities. These Case II activities were still ~4 to 7% higher than the average values reported by ORNL. There are several sources of error in these analyses. It should be recognized that NBS standards have a total uncertainty of 2 to 3%; also there is a 1 to 2% added error from preparation as Sigma 2 standards. Other sources of error include counting statistics for each wire (1 to 2%) and a weighing error ($\leq 1\%$). Therefore, for these measurements the maximum total uncertainty would be ~4.2%. A similar uncertainty would be associated with the ORNL values.

The fast fluences required to produce the measured Mn-54 activities in each wire are given in Table 3-7. The fast fluences calculated using the GA DOSE 1 computer program and the Mn-54 activities measured by ORNL were ~2 to 4% higher than those calculated by ORNL. This difference is slight and probably resulted from several sources, including: (1) uncertainties in the thermal fluence values used to correct for the iron and manganese burnout, (2) slight differences in the capsule power history used in the calculations, and (3) differences introduced by rounding off numerical values.

The differences between the fluences calculated by GA for Case I and Case II resulted from the Sigma 2 recalibration correction applied to the measured activities, as discussed previously. The fluences determined by GA for Case II are ~5 to 7% greater than the fluences determined using the Mn-54 activities measured at ORNL, which is the result of differences in the activities measured at the two laboratories, as discussed previously.

The neutron fluences given in Tables 3-1 through 3-4 and subsequently used in the analyses of the fuel specimens were determined using the Case I Mn-54 activities and are therefore high by ~6%.

3.4. HEAVY METAL BURNUP ANALYSIS

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. Heavy metal burnup analysis of capsules P13R and P13S was conducted by two techniques: (1) wet chemical analysis and mass spectroscopy and (2) calculations based on gamma-emitting fission product inventory and capsule irradiation history.

Determination of heavy metal burnups by chemical analysis was conducted in accordance with ASTM Procedure E-495-73T, "Spectrophotometric Determination of Fission Zirconium in Irradiated Nuclear Fuels;" ASTM Procedure E-244-69, "Atom Percent Fission in Uranium and Plutonium Fuel (Mass Spectrometric Method);" and General Atomic Procedure ACD-RC-001, "Atom Percent Fission in Fissile and Fertile Fuel Particles." Prior to chemical analysis all fuel particles were cleaned to remove external contamination. The gamma-ray spectrum of the individual fuel particles was analyzed to ensure that the fuel particles had not failed during irradiation, thus releasing a portion of their fission product inventory. The fuel particles were then crushed and dissolved in a perchloric - nitric acid solution. These solutions, which contained zirconium, uranium, and thorium in a strongly acidic chloride medium, were separated from plutonium and fission products by an anion exchange separation. Each fraction (Th, Zr, and U) was determined spectrophotometrically by the absorbance of its respective Arsenazo-III complex.

Total heavy metal atom percent fissions (% FIMA) were calculated from the results of the chemical analysis using the following equations:

$$\% \text{ FIMA (U-235)} = \frac{F}{U_o} \times 100 = \frac{F}{U_R + F} \times 100 \quad , \quad (3-2)$$

$$\% \text{ FIMA (U-233)} = \frac{F}{\text{Th}_O} \times 100 = \frac{F}{\text{Th}_R + \text{U}_R + F} \times 100 \quad , \quad (3-3)$$

where F = fissions per total sample = N'/Y

N' = atoms of Zr or Cs-137 (corrected for decay during and after irradiation)

Y = fractional fission yields of Zr and Cs-137 (Table 3-8)

U_O = initial atoms of uranium

Th_O = initial atoms of thorium

U_R = remaining atoms of uranium

Th_R = remaining atoms of thorium

The heavy metal burnup values determined from the wet chemical analysis using Eqs. 3-2 and 3-3 are shown in Table 3-9 under the headings % FIMA Fiss Zr and % FIMA Fiss Cs-137.

In addition to using cesium and zirconium as fission product monitors, fissile particle burnups were also determined by a mass spectrometric uranium isotopic analysis method. This method measures burnup through changes in uranium isotopic composition and can only be applied to fuel particles that do not contain thorium before irradiation. The uranium isotopic compositions of the irradiated fissile, fertile, and reference archive fissile particle samples determined by mass spectrometry are given in Tables 3-10 and 3-11. The fissile particle burnups were calculated from the data given in Table 3-10 in accordance with ASTM Procedure E244-69, and the results are given in Table 3-9 for each of the three cases. A capture-to-fission cross-section ratio for U-235 of 0.1875 was used in the mass spectrometric calculations. These cross sections satisfied the FISSIN computer program discussed below.

The heavy metal burnup of each sample irradiated in capsules P13R and P13S was calculated using the Sigma 2 FISSIN computer code. This program calculates burnup for both U-235 and thorium (U-233) and the weight of uranium and thorium at the beginning of each reactor cycle, as well as the uranium isotopic inventory at the end of irradiation. Required inputs are

reactor power history, average thermal fluences, initial uranium and thorium loadings, macroscopic cross sections, and fission yields. The cross sections used in the program were adjusted relative to the thermal fluence measured by the cobalt dosimetry in the GETR (see Section 3.3). The burnups calculated for the five particle samples using the FISSIN computer program are given in Table 3-9. Heavy metal burnup values were calculated for all fuel specimens using the FISSIN computer program and are presented in the following sections of this report.

The uranium isotopic compositions of the fissile and fertile samples determined by mass spectrometry and the Sigma 2 FISSIN computer code are compared in Table 3-12. Based on the results from these burnup analyses (Tables 3-9 and 3-12), wet chemistry burnup measurements of high burnup fissile particles overestimate the burnup by approximately 5%. At lower burnups the error would be less. This overestimation occurred because small amounts of U-236 and U-238 were transmuted to neptunium and plutonium and therefore were not recovered during the anion exchange separation. Modification of the chemistry procedure is planned to permit recovery and analysis of these transuranic species.

To obtain a comparison between laboratories, five particle samples were also submitted to ORNL for burnup analysis. Burnups were determined by wet chemistry in accordance with ASTM Procedure E-495-73T and by isotope dilution mass spectrometry (IDMS).

Two particles from each sample were analyzed separately using both techniques. Each particle was examined visually to ensure the coating was intact, and the Zr-95 activity was measured by gamma-counting with a Ge(Li) detector. The particle was then dissolved in a HNO_3 -HF solution at 275°C using a Teflon-lined steel container. After dissolution was effected, the solution was diluted and an aliquot was removed for IDMS measurements. The remaining solution was analyzed for Zr, Th, and U using wet chemistry techniques.

Thorium and uranium were analyzed by IDMS; however, zirconium activity could not be determined due to poor sensitivity. In the IDMS method, the total quantities of Th and U were measured by a double spike isotope dilution technique. Enriched Th-230 and U-238 were used as tracers. After the sample was dissolved, an aliquot was measured for isotopic composition; a second measured aliquot was then equilibrated with a double spike which contained known amounts of Th-230 and U-238. The newly established ratios were measured, and these ratios were then used to calculate the Th and U concentrations of the sample solution.

The above methods of analyses were evaluated using a standard solution which contained 2.32 μg Zr/ml, 400 μg Th/ml, and 58.11 μg U/ml in 4M HNO_3 . An aliquot of the standard solution was analyzed in parallel with each particle sample analysis. The analysis of the standard solution included treatment in the Teflon-lined dissolution apparatus.

The precision and accuracy data obtained for the standard solution are presented in Table 3-13. Based on the Zr data, it was concluded that the precision and accuracy of the burnup measurements were $\pm 5.0\%$ and the precision and accuracy of the Th and U measurements by the wet chemistry or IDMS method were $\pm 2.0\%$.

The uranium isotopic compositions determined for the fissile and fertile particles are given in Table 3-14. These data are in good agreement with the isotopic composition (atom %) determined at GA (see Tables 3-10 and 3-11). The results of the heavy metal burnup analyses conducted using both techniques are given in Table 3-15. These data are in good agreement with the burnup values obtained at GA (Table 3-9).

Heavy metal burnup values reported for all fuel specimens in the following sections of this report were calculated using the GA FISSIN computer program. The fission and capture cross sections used in this program were adjusted relative to the thermal fluences measured for

capsules P13R and P13S using cobalt dosimetry. These burnup values were in good agreement with the values measured on individual fuel particles at both ORNL and GA by wet chemical analysis.

3.5. THERMAL ANALYSIS

One of the most important parameters influencing fuel irradiation behavior is the thermal environment of the fuel. Therefore, to evaluate fuel performance a good knowledge of the fuel's operating temperature history is essential. For this reason temperature profiles in capsules P13R and P13S were extensively monitored by thermocouples. However, difficulties arose in deducing fuel temperatures because the high-temperature thermocouples (W/Re) decalibrate as a function of neutron fluence. Furthermore, in a number of cases the fuel thermocouples failed during the irradiation. To better estimate the fuel temperatures, detailed numerical analyses were used to predict thermal histories for fuel rods and unbonded particle samples in these two capsules. In these analyses computer codes were used which take into account material geometry and properties, reactor operating parameters, irradiation-induced property changes, and the capsule's operational mode.

Capsule operating temperatures below 1100°C in unfueled regions were measured with Chromel-Alumel (C/A) thermocouples. These thermocouples are not affected by the reactor's neutron environment (Ref. 3-1). Therefore, the C/A temperature measurements provide the bases for comparison with calculated values. Fuel and other high-temperature regions ($T > 1100^{\circ}\text{C}$) were monitored with tungsten/rhenium (W - 3% Re/W - 25% Re) thermocouples. These thermocouples undergo irradiation-induced drift resulting from alloy compositional changes caused by transmutations in both metals. Empirical data are available to account for this emf drift; however, relatively large uncertainties are currently associated with the correction factors. One of the primary objectives of the thermal analysis was to more accurately predict irradiation-induced emf changes in the W/Re thermocouples.

3.5.1. Fuel Rod Temperatures

Detailed TAC-2D models for each fuel rod cell were used to calculate beginning-of-life (BOL, first day of operation) and end-of-life (EOL, 258th day of operation) temperature profiles. All calculated fuel and graphite temperatures could be matched with the measured temperatures within $\pm 60^\circ$ at BOL and EOL except for cell 1 in both capsules, as shown in Figs. 3-25 and 3-26. End-of-life W/Re thermocouple temperatures were corrected to account for emf drift according to procedures given in Section 3.1.

To obtain agreement between measured and predicted values, a number of modifications had to be made to the data used in the calculations. Most of these changes were of the nature of refining the input data. However, the most significant problem in the initial calculations was the inability to obtain agreement with the measured primary containment temperature. In almost all cases, the calculated values were about 200°C higher than the measured values. This discrepancy apparently stems from uncertainties in the gap between the secondary containment and the aluminum thermal bond (see Section 2.3). Primary containment temperatures were found to be very sensitive to small changes in this gap; for example, the temperature drop across this gap could change by as much as 150°C for an error of 0.0015 cm in the gap width. It appears from the thermal predictions, as well as from capsule assembly information, that this gap could vary between approximately 0.0015 cm and the design value of 0.008 cm. These gaps were adjusted in the model to make the calculated primary containment temperature agree with the measured value to within 5°C .

The TAC-2D results are used as input to the SHAPE code (Ref. 3-4). This code determines the thermal resistance shape factor which will result in the same radial temperature distributions in the fuel samples when used in a one-dimensional model. A one-dimensional radial heat transfer code is then used to calculate fuel temperature histories on a daily basis. This model is called CAPTEM (Ref. 3-5). CAPTEM analyses were performed on each fuel rod cell (cells 1, 2, 5, and 6) in capsules P13R and P13S. A detailed

discussion of these results is presented in Ref. 3-6. The calculated fuel temperatures and temperature gradients are presented in Tables 3-16 and 3-17.

The uncertainties of the fuel centerline temperatures are calculated on the basis of estimated assigned uncertainties in selected variables. The estimated and calculated uncertainties shown in Table 3-18 (stated as two standard deviations assuming normal distributions) give half the range that is expected to contain 95% of the probable values. For example, if a fuel rod conductivity is predicted to be 7.0 Btu/hr-ft-°F, an uncertainty of 10% would mean that 95% of the probable conductivity values for that rod lie within 7.0 ± 0.7 . Calculations of fuel centerline uncertainties use the outer wall of the primary containment as the reference point (i.e., known temperature). These uncertainty calculations did not account for the fact that reliable temperature measurements were made in the graphite crucibles for most of the cells. Further uncertainties in the calculated temperature arise from: (1) the loss of radial gaps between the rods and graphite crucible and crucible bowing in cell 2 of capsules P13R and P13S, and (2) uncertainties in the gas mixtures in cell 6 of P13S. For item one, the model was modified to account for the fact that the rods contacted the crucibles in cell 2; however, slight differences between calculated and observed graphite crucible temperatures still remained. Regarding the second item, it was noted for cell 6 of P13S that from the 40th to the 140th day of operation, the helium and neon gas flow data were apparently in error (Ref. 3-6). Relatively large discrepancies between calculated and measured temperatures were found during this period, and these discrepancies were the greatest when mixtures of the two gases were used. Good agreement between calculated and measured temperatures was obtained when only He or only Ne was used to control the gap conductivities. Data for this time period have been eliminated from the temperature histories reported in Table 3-17.

Results of the thermal analysis have also shown that the control thermocouples for cell 5 of P13R and P13S were operated too low. The selection of the control temperature was based on preirradiation design studies which

apparently improperly accounted for gamma heating of the W/Re fuel thermocouples. In addition, the fission powers in these cells varied markedly from design conditions which led to different heat flow patterns. These two factors caused the cells to be controlled about 100°C lower than intended.

3.5.2. Unbonded Fuel Particle Temperatures

The CAPTEM code is presently not adaptable to thermal modeling of unbonded particle cells. Consequently, TAC-2D models used in the thermal design were used to evaluate temperature histories of fuel samples in cells 3 and 4 of P13R and P13S. The relationship between the central thermocouple temperatures and individual particle samples was determined as a function of fuel particle power and power distribution, conductivity gas mixture, absolute temperature, etc. The results of the TAC-2D analysis were used to determine the ratios of peak fuel temperatures to those at the reference thermocouple location. These ratios were fit to an empirical equation which accounted for the variables listed above. The indicated temperatures of the W/Re thermocouples were corrected for irradiation-induced drift according to procedures outlined in Sections 3.1 and 3.5.3.

There were twelve particle trays in each cell, each of which contained fissile and fertile samples. These twelve trays were divided into four fuel groups for analysis purposes. The criterion for grouping was to keep the temperature spread less than $\pm 25^{\circ}\text{C}$ from nominal. Tables 3-19 through 3-22 present the calculated average and maximum hot-spot temperatures for each unbonded fuel particle sample contained in capsules P13R and P13S.

3.5.3. Thermocouple Decalibration

In addition to providing fuel thermal histories, the temperature analysis provided information on the degree of irradiation-induced decalibration of the W/Re thermocouples. Direct evidence that the W/Re thermocouples do decalibrate is shown in Fig. 3-27. The end-of-life indicated the fuel rod centerline temperature monitored by a W/Re thermocouple was

as low as 400°C, or about 400°C lower than the surrounding graphite crucible which was monitored by C/A thermocouples. Thermocouple decalibration plots (i.e., difference between indicated and calculated temperatures) are presented in Ref. 3-6 for all W/Re thermocouples in each cell. A linear regression analysis was performed for each set of data and the resulting equations are given in Table 3-23. The slope of most of the curves for thermocouples operating at ~1100°C was nearly constant, yielding an average value for the "decalibration constant" of 75°C per unit of thermal fluence. This value is slightly different from the ones used to control the capsules temperatures; however, these differences did not result in a significant error in maintaining the fuel at design temperatures. For the high-temperature cells (cell 5 of P13R and P13S), the slope of the curve is approximately 170°C per unit of thermal fluence for thermal fluences up to $\sim 2.5 \times 10^{21}$ n/cm². After this time the degree of decalibration appeared to change, as did the fuel temperatures. These changes coincide with a change in the equation used to account for decalibration of the W/Re thermocouples operating at the higher temperatures in these cells. The postirradiation thermal analysis indicated this means of control was incorrect, since the fuel temperatures in these cells were 300° to 400°C below their design values at end-of-life (see Tables 3-16 and 3-17).

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TABLE 3-1
IRRADIATION PARAMETERS FOR CAPSULE P13R UNBONDED FUEL PARTICLE SAMPLES

Capsule Position	Fissile Particle Batch Data Retrieval No.	Fertile Particle Batch Data Retrieval No.	Location ^(a) (in.)	Burnup ^(b) (% FIMA)		GETR Neutron Exposure				Equivalent HTGR Fast Neutron Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)HTGR
						Thermal		Fast		
				Fissile	Fertile	(10 ¹⁴ Flux n/cm ² -sec) (2200 m/sec)	Fluence (10 ²¹ n/cm ²) (2200 m/sec)	(10 ¹⁴ Flux n/cm ² -sec) (E > 0.18 MeV)	Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)	
Cell 3-1	6151-00-010-10	4252-06-018-1	16.20	74	5.1	2.7	6.2	6.1	13.6	12.1
3-2	6151-00-010-11	4252-06-018-2	16.35	74	5.1	2.7	6.2	6.0	13.6	12.1
3-3	6151-00-010-12	6542-01-020-1	16.50	74	5.0	2.7	6.1	6.0	13.6	12.1
3-4	6151-00-035-1	6542-01-020-2	16.65	74	5.0	2.7	6.1	6.0	13.5	12.0
3-5	6151-00-035-2	6542-01-010-1	16.80	74	5.0	2.7	6.1	6.0	13.5	12.0
3-6 ^(c)	6151-00-035-3	6542-01-010-2	16.95	--	--	--	--	--	--	--
3-7	6151-00-045-1	6542-02-020-1	17.10	74	4.9	2.7	6.0	6.0	13.4	11.9
3-8	6151-00-045-2	6542-02-020-2	17.25	74	4.9	2.7	6.0	5.9	13.4	11.9
3-9	6151-00-045-3	6542-02-030-1	17.40	74	4.8	2.6	5.9	5.9	13.3	11.8
3-10	6151-00-046-1	6542-02-030-2	17.55	74	4.8	2.6	5.9	5.9	13.3	11.8
3-11	6151-04-015-1	6542-09-010-1	17.70	74	4.8	2.6	5.9	5.9	13.2	11.7
3-12	6151-04-015-2	6542-09-010-2	17.85	74	4.7	2.6	5.8	5.9	13.2	11.7
Cell 4-1	6151-01-015-1	6542-11-015-1	18.65	73	4.5	2.5	5.7	5.7	12.9	11.5
4-2	6151-01-015-2	6542-11-015-2	18.80	73	4.5	2.5	5.6	5.7	12.9	11.5
4-3	6151-01-015-3	6542-18-015-1	18.95	73	4.4	2.5	5.6	5.7	12.8	11.4
4-4	6151-00-025-1	6542-18-015-2	19.10	73	4.5	2.5	5.5	5.7	12.8	11.4
4-5	6151-00-025-2	6542-02-020-3	19.25	73	4.4	2.4	5.5	5.6	12.7	11.3
4-6	6151-00-025-3	6542-02-020-4	19.40	73	4.3	2.4	5.5	5.6	12.7	11.3
4-7	6151-09-015-1	6542-20-035-1	19.55	73	4.3	2.4	5.4	5.6	12.6	11.2
4-8	6151-09-015-2	6542-20-035-2	19.70	73	4.3	2.4	5.4	5.6	12.5	11.1
4-9	6151-09-015-3	6542-20-025-1	19.85	73	4.2	2.4	5.3	5.5	12.5	11.1
4-10	6151-09-025-1	6542-20-025-2	20.00	73	4.2	2.4	5.3	5.5	12.4	11.0
4-11 ^(c)	6151-09-025-2	6542-25-015-1	20.15	73	4.1	2.3	5.3	5.5	12.3	10.9
4-12	6151-09-025-3	6542-25-015-2	20.30	--	--	--	--	--	--	--

(a) Location in capsule measured from bottom of reactor core to midplane of sample tray.

(b) Heavy metal burnup calculated with FISSIN computer code.

(c) Sample tray was spilled during unloading.

TABLE 3-2
IRRADIATION PARAMETERS FOR CAPSULE P13S UNBONDED FUEL PARTICLE SAMPLES

Capsule Position	Fissile Particle Batch Data Retrieval No.	Fertile Particle Batch Data Retrieval No.	Location (a) (in.)	Burnup (b) (% FIMA)		GETR Neutron Exposure				Equivalent HTGR Fast Neutron Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}
						Thermal		Fast		
				Fissile	Fertile	Flux (10 ¹⁴ n/cm ² -sec) (2200 m/sec)	Fluence (10 ²¹ n/cm ²) (2200 m/sec)	Flux (10 ¹⁴ n/cm ² -sec) (E > 0.18 MeV)	Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)	
Cell 3-1	6151-03-015-1	4252-06-010-1	16.20	73	4.3	2.4	5.4	5.9	13.3	11.8
	3-2	6151-03-015-2	16.35	73	4.3	2.4	5.4	5.9	13.3	11.8
	3-3	6151-03-015-3	16.50	73	4.3	2.4	5.4	5.9	13.2	11.7
	3-4	4161-01-021-1	16.65	73	4.2	2.4	5.4	5.9	13.2	11.7
	3-5	4161-01-021-2	16.80	73	4.2	2.4	5.3	5.9	13.2	11.7
	3-6	4161-01-021-3	16.95	73	4.2	2.4	5.3	5.8	13.1	11.7
	3-7	6151-08-015-1	17.10	73	4.2	2.4	5.3	5.8	13.1	11.7
	3-8	6151-08-015-2	17.25	73	4.1	2.3	5.3	5.8	13.1	11.7
	3-9	6151-08-015-3	17.40	73	4.1	2.3	5.3	5.8	13.0	11.6
	3-10 (c)	6151-02-025-1	17.55	73	4.1	2.3	5.2	5.8	13.0	11.6
	3-11	6151-02-025-2	17.70	--	--	--	--	--	--	--
	3-12	6151-02-025-3	6252-00-025-2	17.85	72	4.1	2.3	5.2	5.7	12.9
Cell 4-1	4163-00-011-3	6542-17-010-1	18.65	40	3.9	2.2	5.0	5.6	12.6	11.2
	4-2	4163-00-011-4	18.80	40	3.9	2.2	5.0	5.6	12.6	11.2
	4-3	4163-00-011-5	18.95	40	3.8	2.2	5.0	5.6	12.5	11.1
	4-4	6155-01-020-10	19.10	11	3.8	2.2	5.0	5.5	12.5	11.1
	4-5 (c)	6155-01-020-11	19.25	--	--	--	--	--	--	--
	4-6 (c)	6155-01-020-12	19.40	--	--	--	--	--	--	--
	4-7	5466-37-9	19.55	38	3.7	2.2	4.9	5.5	12.3	10.9
	4-8	5466-37-10	19.70	38	3.7	2.2	4.8	5.4	12.2	10.9
	4-9	5466-37-11	19.85	38	3.7	2.1	4.8	5.4	12.2	10.9
	4-10	6151-00-035-4	20.00	71	3.6	2.1	4.7	5.4	12.1	10.8
	4-11	6151-00-035-5	20.15	71	3.6	2.1	4.7	5.3	12.0	10.7
	4-12	6151-00-035-6	6542-12-025-2	20.30	71	3.6	2.1	4.7	5.3	12.0

(a) Location in capsule measured from bottom of reactor core to midplane of sample tray.

(b) Heavy metal burnup calculated with FISSIN computer code.

(c) Sample tray was spilled during unloading.

TABLE 3-3
IRRADIATION PARAMETERS FOR CAPSULE P13R FUEL RODS

Data Retrieval No. (7161-004-)	Capsule Position	Location (a) (in.)	Burnup (b) (% FIMA)		GETR Neutron Exposure				Equivalent HTGR Fast Neutron Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)HTGR
					Thermal		Fast		
					Flux (10 ¹⁴ n/cm ² -sec) (2200 m/sec)	Fluence (10 ²¹ n/cm ²) (2200 m/sec)	Flux (10 ¹⁴ n/cm ² -sec) (E > 0.18 MeV)	Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)	
			Fissile	Fertile					
01-5	1A	2.9	71	3.5	2.1	4.7	3.7	8.2	7.3
02-6	1B	3.6	72	3.9	2.2	5.0	4.0	9.1	8.1
03-5	1C	4.4	73	4.2	2.4	5.3	4.4	9.8	8.7
04-6	1D	5.1	73	4.5	2.5	5.6	4.7	10.5	9.3
05-6	1E	5.9	40	4.8	2.6	5.9	5.0	11.1	9.9
06-5	2A	9.8	75	5.5	2.9	6.6	5.9	13.3	11.8
07-5	2B	10.5	75	5.6	3.0	6.6	6.0	13.5	12.0
08-13	2C	11.3	75	5.6	3.0	6.7	6.1	13.7	12.2
09-5	2D	12.0	75	5.6	3.0	6.6	6.1	13.8	12.3
10-5	2E	12.8	41	5.5	2.9	6.6	6.2	13.9	12.4
16-5	5A	23.7	36	3.1	1.9	4.2	4.7	10.6	9.4
17-5	5B	24.4	68	2.8	1.8	4.0	4.5	10.1	9.0
18-6	5C	25.2	67	2.6	1.7	3.8	4.3	9.6	8.5
19-7	5D	25.9	66	2.4	1.6	3.5	4.1	9.2	8.2
20-6	5E	26.7	64	2.1	1.5	3.3	3.9	8.7	7.7
21-5	6A	30.6	54	1.1	1.0	2.2	2.7	6.1	5.4
22-5	6B	31.3	51	1.0	0.9	2.0	2.5	5.6	5.0
23-5	6C	32.1	49	0.8	0.8	1.9	2.3	5.1	4.5
24-5	6D	32.8	47	0.7	0.8	1.7	2.1	4.6	4.1
25-5	6E	33.6	44	0.6	0.7	1.6	1.8	4.1	3.6

(a) Location in capsule measured from bottom of reactor core to fuel rod midplane.

(b) Heavy metal burnup calculated with FISSIN computer code.

TABLE 3-4
IRRADIATION PARAMETERS FOR CAPSULE P13S FUEL RODS

Data Retrieval No. (7161-004-)	Capsule Position	Location ^(a) (in.)	Burnup ^(b) (% FIMA)		GETR Neutron Exposure				Equivalent HTGR Fast Neutron Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}
					Thermal		Fast		
			Fissile	Fertile	Flux (10 ¹⁴ n/cm ² -sec) (2200 m/sec)	Fluence (10 ²¹ n/cm ²) (2200 m/sec)	Flux (10 ¹⁴ n/cm ² -sec) (E > 0.18 MeV)	Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)	
01-7	1A	2.9	70	3.2	1.9	4.4	3.5	7.9	7.0
02-5	1B	3.6	71	3.4	2.0	4.6	3.9	8.7	7.7
03-6	1C	4.4	71	3.7	2.1	4.8	4.2	9.5	8.5
04-5	1D	5.1	72	3.9	2.2	5.0	4.5	10.2	9.1
05-5	1E	5.9	39	4.1	2.3	5.2	4.8	10.8	9.6
11-6	2A	9.8	74	4.6	2.5	5.7	5.8	12.9	11.5
12-5	2B	10.5	74	4.6	2.6	5.8	5.9	13.2	11.7
13-6	2C	11.3	74	4.7	2.6	5.8	5.9	13.4	11.9
14-5	2D	12.0	74	4.6	2.6	5.8	6.0	13.5	12.0
15-5	2E	12.8	74	4.6	2.6	5.7	6.0	13.6	12.1
16-6	5A	23.7	36	2.8	1.8	3.9	4.6	10.3	9.2
17-6	5B	24.4	67	2.6	1.7	3.8	4.4	9.8	8.7
18-5	5C	25.2	66	2.4	1.6	3.6	4.2	9.4	8.4
19-5	5D	25.9	65	2.2	1.5	3.4	4.0	8.9	7.9
20-5	5E	26.7	63	2.0	1.4	3.2	3.8	8.4	7.5
26-5	6A	30.6	54	1.1	1.0	2.2	2.6	5.9	5.3
27-5	6B	31.3	26	1.0	0.9	2.0	2.4	5.4	4.8
28-13	6C	32.1	48	0.8	0.8	1.8	2.2	4.9	4.4
29-5	6D	32.8	46	0.7	0.7	1.6	1.9	4.4	3.9
30-5	6E	33.6	42	0.6	0.7	1.5	1.7	3.9	3.5

(a) Location in capsule measured from bottom of reactor core to fuel rod midplane.

(b) Heavy metal burnup calculated with FISSIN computer code.

TABLE 3-5
WIRE WEIGHTS

Wire Ident. No.	Capsule	Location	ORNL Wire Weight (mg)	GA Wire Weight (mg)
C-1632	P13R	Cell 2/hole 1/0°	16.639	16.16
C-1635	P13R	Cell 5/hole 2/0°	22.411	22.40
C-1675	P13S	Cell 2/hole 3/0°	16.100	16.11
C-1679	P13S	Cell 5/hole 4/0°	22.341	22.30

TABLE 3-6
MEASURED Mn-54 ACTIVITIES

Wire Ident. No.	Measured Mn-54 Activity			
	ORNL (10 ⁵ dps/mg)	ORNL (μCi/mg)	GA - Case I (μCi/mg)	GA - Case II (μCi/mg)
C-1632	3.29 ± 0.02	14.81 ^(a)	16.92	15.98
C-1635	2.42 ± 0.16	10.58	11.79	11.16
C-1675	3.32 ± 0.23	14.51	16.03	15.24
C-1679	2.31 ± 0.09	10.10	11.31	10.79

^(a)Value normalized to corrected weight.

TABLE 3-7
COMPARISON OF FAST NEUTRON FLUENCES

Wire Ident. No.	Calculated Fast Neutron Fluence (10^{21} n/cm ²) (E > 0.18 MeV)GETR			
	ORNL Program	ORNL Mn-54 Activity, GA Dose 1 Program	GA - Case I Mn-54 Activity, GA Dose 1 Program	GA - Case II Mn-54 Activity, GA Dose 1 Program
C-1632	12.5 ^(a)	13.0 ^(a)	14.9	14.0
C-1635	8.9	9.1	10.2	9.6
C-1675	12.2	12.7	14.0	13.4
C-1679	8.5	8.7	9.8	9.2

(a) Activity was normalized to corrected wire weight.

TABLE 3-8
THERMAL FISSION YIELDS OF CESIUM AND ZIRCONIUM ISOTOPES

Isotope	U-233 Yield (%)	U-235 Yield (%)
Zr-91	6.52	5.92
Zr-92	6.59	5.96
Zr-93	7.01	6.37
Zr-94	6.81	6.42
Zr-96	5.68	6.26
Cs-137	6.80	6.27

TABLE 3-9
RESULTS OF HEAVY METAL BURNUP DETERMINATIONS MADE AT GENERAL ATOMIC

Capsule	Capsule Position	Batch No.	Particle Type	Heavy Metal Burnup (% FIMA)					
				Fiss Cs ^(a)	Fiss Zr ^(b)	Case I ^(c)	Case II ^(d)	Case III ^(e)	FISSIN ^(f)
P13R	3-5	6151-00-035-2	UC ₂ TRISO	77.9	78.3	74.37	74.08	74.28	73.64
P13S	4-12	6151-00-035-6	UC ₂ TRISO	76.9	75.7	71.83	71.65	71.76	71.12
P13R	3-7	6542-02-020-1	ThO ₂ BISO	4.21	ND ^(g)	NA ^(h)	NA	NA	4.33
P13S	4-2	4252-02-010-1	ThO ₂ BISO	3.97	ND	NA	NA	NA	3.92
P13S	3-12	6252-00-025-2	ThO ₂ BISO	4.01	ND	NA	NA	NA	4.10

(a) Fiss Cs = atom percent fissions per initial heavy metal atom based on fission product cesium.

(b) Fiss Zr = atom percent fissions per initial heavy metal atom based on fission product zirconium.

(c) Case I = atom percent fissions per initial heavy metal atom based on Case I of ASTM Procedure E244-69.

(d) Case II = atom percent fissions per initial heavy metal atom based on Case II of ASTM Procedure E244-69.

(e) Case III = atom percent fissions per initial heavy metal atom based on Case III of ASTM Procedure E244-69.

(f) FISSIN = atom percent fissions per initial heavy metal atom, Sigma 2 calculation, based on input thermal nvt, reactor power history, and macroscopic cross sections.

(g) ND = not determined.

(h) NA = not applicable.

TABLE 3-10
FISSILE PARTICLE URANIUM ISOTOPIC COMPOSITION DETERMINED BY MASS SPECTROMETRY

Sample	Capsule	Capsule Position	Atom Percent					Weight Percent				
			233	234	235	236	238	233	234	235	236	238
Irradiated Particles												
6151-00-035-2	P13R	3-5	0.183	1.222	24.04	54.48	20.08	0.181	1.211	23.93	54.45	20.23
6151-00-035-2	P13R	3-5	0.172	1.232	23.11	55.33	20.15	0.170	1.220	23.00	55.30	20.31
6151-00-035-6	P13S	4-12	0.135	1.180	31.52	48.09	19.08	0.134	1.170	31.38	48.08	19.23
6151-00-035-6	P13S	4-12	0.132	1.178	32.46	47.81	18.41	0.130	1.168	32.32	47.81	18.57
Reference Archive Particles												
6151-00-035	--	--	Not detected	0.657 ±0.003	93.178 ±0.011	0.3977 ±0.0028	5.767 ±0.010	Not detected	0.654 ±0.003	93.111 ±0.011	0.3991 ±0.0028	5.836 ±0.010

TABLE 3-11
FERTILE PARTICLE URANIUM ISOTOPIC COMPOSITION DETERMINED BY MASS SPECTROMETRY

Sample	Capsule	Capsule Position	Atom Percent					Weight Percent				
			233	234	235	236	238	233	234	235	236	238
6542-02-020-1	P13R	3-7	74.53	20.41	4.229	0.712	0.119	74.43	20.47	4.259	0.720	0.121
			73.25	20.21	4.189	0.710	1.641	73.13	20.27	4.217	0.718	1.673
6252-00-025-2	P13S	3-12	75.48	19.77	4.027	0.637	0.089	75.38	19.83	4.056	0.645	0.091
			75.76	19.49	3.985	0.622	0.139	75.67	19.55	4.014	0.629	0.141
4252-02-010-1	P13S	4-2	76.25	19.11	2.866	0.598	0.177	76.15	19.17	3.894	0.605	0.180
			76.42	19.04	3.81	0.577	0.141	76.33	19.10	3.84	0.584	0.143

TABLE 3-12
COMPARISON OF FISSILE AND FERTILE PARTICLE URANIUM ISOTOPIC COMPOSITION
DETERMINED BY MASS SPECTROMETRY AND THE SIGMA 2 FISSIN COMPUTER PROGRAM

Sample	Capsule	Capsule Position	U-234	U-235	U-236	U-238	Method
Fissile Weight Percent							
6151-00-035-2	P13R	3-5	1.218	23.50	54.97	20.30	Mass Spec. ^(a)
			1.17	23.48	54.88	20.50	Sigma 2
6151-00-035-2	P13S	4-12	1.17	31.89	48.01	18.93	Mass Spec. ^(a)
			1.18	31.91	48.18	18.73	Sigma 2
Fertile Weight Percent							
6542-02-020-1	P13R	3-7	U-233	U-234	U-235	U-236	
			74.45	20.56	4.28	0.726	Mass Spec. ^(b)
			74.62	20.23	4.54	0.80	Sigma 2
6252-00-025-2	P13S	3-12	75.61	19.71	4.04	0.638	Mass Spec. ^(b)
			75.36	19.62	4.30	0.73	Sigma 2
4252-02-010-1	P13S	4-2	76.36	19.17	3.87	0.596	Mass Spec. ^(b)
			76.09	19.13	4.11	0.67	Sigma 2

^(a) U-233 normalized out.

^(b) U-238 normalized out.

TABLE 3-13
METHODS EVALUATION DATA

Standard Sample No. (a)	Metal Concentration Found (µg/ml)					Weight Ratio, Th/U	
	Zr Wet Chem.	Th		U			
		Wet Chem.	IDMS	Wet Chem.	IDMS	Wet Chem.	IDMS
1	3.2	418	389	56.1	56.4	7.45	6.89
2	3.5	400	396	59.8	57.6	6.94	6.87
3	3.4	416	397	59.8	57.7	6.96	6.88
4	3.4	406	399	59.3	59.1	6.84	6.75
5	3.2	411	400	59.1	60.4	6.95	6.62
6	3.5	401	422	59.1	61.3	6.78	6.89
7	3.9	416	(b)	59.3	(b)	7.01	(b)
8	3.7	400	(b)	61.7	(b)	6.48	(b)
9	3.2	400	(b)	57.9	(b)	6.90	(b)
10	3.6	411	(b)	59.6	(b)	6.90	(b)
Average	3.46	408	400	59.2	58.8	6.92	6.82
Relative Std. Dev., %	±6.4	±1.7	±2.5	±2.4	±2.9	±3.2	±1.5
Relative error, %	+4.2	+2.0	0	+1.9	+1.2	+0.6	-0.9

(a) Composition of standard: Zr = $3.32 \mu\text{g/ml}$, Th = $400 \mu\text{g/ml}$, U = $58.1 \mu\text{g/ml}$, and Th/U = 6.88.

TABLE 3-14
URANIUM ISOTOPIIC ANALYSIS

Sample	Sample Code (a)	Isotopic Abundance (atom %)					Avg. Gram Atomic Weight (b)
		233	234	235	236	238	
6151-00-035-2	3-5-(2)	0.473	1.217	25.99	52.77	19.55	236.1
	-(8)	0.220	1.216	23.55	55.00	20.02	236.1
6542-02-020-1	3-7-(1)	73.08	19.93	5.49	0.778	0.728	233.4
	-(6)	74.34	20.42	4.42	0.739	0.074	233.3
6252-00-025-2	3-12-(4)	74.93	19.56	4.62	0.730	0.160	233.3
	-(5)	74.83	19.49	4.72	0.784	0.185	233.3
4252-02-010-1	4-2-(2)	75.97	19.39	4.01	0.588	0.044	233.3
	-(6)	75.51	19.01	4.70	0.662	0.108	233.4
6151-00-035-6	4-12-(2)	0.197	1.202	31.61	48.59	18.39	236.0
	-(4)	0.222	1.194	32.60	47.88	18.10	235.5

(a) Number in parentheses denotes the particle number assigned for the designated particle sample.

(b) Average gram atomic weight for uranium used in the calculation of FIMA.

TABLE 3-15
RESULTS OF HEAVY METAL BURNUP DETERMINATIONS MADE AT
OAK RIDGE NATIONAL LABORATORY

Capsule	Capsule Position	Batch Number	Particle Type	Heavy Metal Burnup (% FIMA)	
				Wet Chem.	IDMS
P13R	3-5	6151-00-035-2	UC ₂ TRISO	75.3 74.6	78.3 75.4
P13R	3-7	6542-02-020-1	ThO ₂ BISO	4.72 4.70	4.96 --
P13S	3-12	6252-00-025-2	ThO ₂ TRISO	4.29 4.44	4.39 4.36
P13S	4-2	4252-02-010-1	ThO ₂ BISO	3.83 3.52	-- 3.61
P13S	4-12	6151-00-035-6	UC ₂ TRISO	71.6 72.5	71.4 72.7

TABLE 3-16
IRRADIATION TEMPERATURES OF CAPSULE P13R FUEL RODS

Data Retrieval No. (7161-004-)	Capsule Position	Centerline Temperature (°C)			Volume Average Temperature (°C)			Fuel Surface Temperature (°C)			Fuel Surface Temperature Gradient (°C/cm)	
		Maximum	Average	EOL	Maximum	Average	EOL (a)	Maximum	Average	EOL	Maximum	Average
01-5	1A	1340	1115	1050	1255	1050	910	1175	985	950	725	455
02-6	1B	1225	1050	1010	1175	1000	890	1135	950	925	480	320
03-5	1C	1235	1065	1020	1190	1015	900	1150	965	940	465	315
04-6	1D	1290	1115	1050	1210	1045	915	1155	975	930	695	450
05-6	1E	1305	1125	1040	1250	1070	920	1205	1010	945	600	405
06-5	2A	1225	1050	1170	1135	975	1045	1050	900	1040	665	515
07-5	2B	1220	1055	1170	1135	985	1050	1065	910	1040	585	470
08-13	2C	1195	1045	1130	1135	980	1025	1085	920	1020	475	395
09-5	2D	1180	1035	1150	1110	965	1035	1050	895	1025	550	455
10-5	2E	1110	965	1110	1060	915	995	1015	860	1015	410	365
16-5	5A	1515	1285	1050	1460	1230	1010	1400	1180	935	450	345
17-5	5B	1460	1225	1020	1400	1175	970	1345	1130	915	420	295
18-6	5C	1475	1235	1055	1410	1180	985	1350	1125	930	510	350
19-7	5D	1480	1225	1050	1410	1170	980	1345	1120	925	505	335
20-6	5E	1590	1280	1055	1505	1215	980	1425	1155	920	685	415
21-5	6A	1220	1065	1015	1185	1035	925	1145	1000	940	350	215
22-5	6B	1215	1055	1050	1170	1020	950	1125	980	955	395	235
23-5	6C	1160	1005	985	1125	975	895	1090	950	915	300	175
24-5	6D	1175	1020	1005	1135	985	905	1095	950	920	355	210
25-5	6E	1155	1000	940	1120	970	855	1085	940	865	350	205

(a) Volume average irradiation temperature for the last cycle of operation.

TABLE 3-17
IRRADIATION TEMPERATURES OF CAPSULE P13S FUEL RODS

Data Retrieval No. (7161-004-)	Capsule Position	Centerline Temperature (°C)			Volume Average Temperature (°C)			Fuel Surface Temperature (°C)			Fuel Surface Temperature Gradient (°C/cm)	
		Maximum	Average	EOL	Maximum	Average	EOL (a)	Maximum	Average	EOL	Maximum	Average
01-7	1A	1545 ^(b)	1030	1015	1465	965	930	1380	900	915	695	445
02-5	1B	1470 ^(b)	975	975	1405	925	915	1345	880	900	435	300
03-6	1C	1505 ^(b)	995	1000	1445	950	935	1380	905	920	435	300
04-5	1D	1555 ^(b)	1035	1015	1470	965	940	1385	900	910	635	420
05-5	1E	1575 ^(b)	1035	1010	1505	980	935	1430	925	920	555	385
11-6	2A	1170	1040	885	1105	970	895	1050	905	765	595	460
12-5	2B	1135	1010	880	1065	945	895	1015	880	770	530	420
13-6	2C	1090	975	860	1040	915	875	995	860	760	435	360
14-5	2D	1100	985	870	1045	920	880	995	855	760	480	400
15-5	2E	1095	960	830	1050	910	850	1010	860	745	385	335
16-6	5A	1645	1375	1125	1595	1330	1055	1545	1285	1030	430	290
17-6	5B	1605	1330	1110	1560	1290	1035	1515	1250	1025	405	255
18-5	5C	1625	1335	1140	1570	1285	1040	1510	1235	1030	535	310
19-5	5D	1610	1325	1130	1560	1280	1035	1505	1230	1030	530	300
20-5	5E	1650	1350	1110	1585	1295	1010	1525	1240	995	670	370
26-5	6A	1250	1085	850	1210	1050	955	1175	1020	770	355	215
27-5	6B	1215	1045	860	1170	1010	935	1130	975	775	360	215
28-13	6C	1215	1050	845	1185	1020	935	1150	995	780	285	170
29-5	6D	1185	1015	825	1150	985	900	1110	955	750	310	190
30-5	6E	1165	995	765	1130	965	855	1095	935	695	315	195

(a) Volume average irradiation temperature for the last cycle of operation.

(b) Does not include maximum temperature achieved during thermal cycling.

TABLE 3-18
VALUES USED IN ESTIMATING UNCERTAINTIES IN FUEL ROD CENTERLINE
TEMPERATURES CALCULATED BY THE CAPTEM CODE

Variable	Assigned Uncertainty ^(a,b) (%)
Fuel rod thermal conductivity	20
Fuel heat generation rate (fission and gamma)	5
Gamma heating of cell components	10
Gap widths	
Fuel - graphite crucible	10
Graphite crucible - primary container	50
Gap conductivities	
Fuel - graphite crucible	10
Graphite crucible - primary container	10
Primary container thermocouple reading	2

(a) Values give half the range that is expected to contain 95% of the probable values.

(b) The total uncertainty is the root mean square of the individual uncertainties.

TABLE 3-19
SUMMARY OF UNBONDED FISSILE PARTICLE TEMPERATURES FOR SAMPLES IRRADIATED
IN CELLS 3 AND 4 OF CAPSULE P13R

Capsule Position	Data Retrieval Number	Peak Fuel Temperature in Tray (°C)	
		Time Average	Maximum
Cell 3-1	6151-00-010-10	1015	1360
3-2	6151-00-010-11	1015	1360
3-3	6151-00-010-12	1075	1400
3-4	6151-00-035-1	1075	1400
3-5	6151-00-035-2	1075	1400
3-6	6151-00-035-3	1075	1400
3-7	6151-00-045-1	1075	1400
3-8	6151-00-045-2	1075	1400
3-9	6151-00-045-3	1075	1400
3-10	6151-00-046-1	1025	1345
3-11	6151-04-015-1	1025	1345
3-12	6151-04-015-2	960	1325
Cell 4-1	6151-01-015-1	1010	1355
4-2	6151-01-015-2	1070	1400
4-3	6151-01-015-3	1070	1400
4-4	6151-00-025-1	1070	1400
4-5	6151-00-025-2	1070	1400
4-6	6151-00-025-3	1070	1400
4-7	6151-09-015-1	1070	1400
4-8	6151-09-015-2	1070	1400
4-9	6151-09-015-3	1070	1400
4-10	6151-09-025-1	1005	1345
4-11	6151-09-025-2	1005	1345
4-12	6151-09-025-3	935	1315

TABLE 3-20
SUMMARY OF UNBONDED FISSILE PARTICLE TEMPERATURES FOR SAMPLES IRRADIATED
IN CELLS 3 AND 4 OF CAPSULE P13S

Capsule Position	Data Retrieval Number	Peak Fuel Temperature in Tray (°C)	
		Time Average	Maximum
Cell 3-1	6151-03-015-1	970	1255
	6151-03-015-2	970	1255
	6151-03-015-3	1020	1280
	4161-01-021-1	1020	1280
	4161-01-021-2	1020	1280
	4161-01-021-3	1020	1280
	6151-08-015-1	1020	1280
	6151-08-015-2	1020	1280
	6151-08-015-3	1020	1280
	6151-02-025-1	980	1230
	6151-02-025-2	980	1230
	6151-02-025-3	920	1200
Cell 4-1	4163-00-011-3	1045	1385
	4163-00-011-4	1090	1430
	4163-00-011-5	1090	1430
	6155-01-020-10	1090	1430
	6155-01-020-11	1090	1430
	6155-01-020-12	1090	1430
	5466-37-9	1090	1430
	5466-37-10	1090	1430
	5466-37-11	1090	1430
	6151-00-035-4	1035	1370
	6151-00-035-5	1035	1370
	6151-00-035-6	975	1345

TABLE 3-21
SUMMARY OF UNBONDED FERTILE PARTICLE TEMPERATURES FOR SAMPLES IRRADIATED
IN CELLS 3 AND 4 OF CAPSULE P13R

Capsule Position	Data Retrieval Number	Peak Fuel Temperature in Tray (°C)	
		Time Average	Maximum
Cell 3-1	4252-06-018-1	960	1055
3-2	4252-06-018-2	960	1055
3-3	6542-01-020-1	1010	1105
3-4	6542-01-020-2	1010	1105
3-5	6542-01-010-1	1010	1105
3-6	6542-01-010-2	1010	1105
3-7	6542-02-020-1	1010	1105
3-8	6542-02-020-2	1010	1105
3-9	6542-02-030-1	1010	1105
3-10	6542-02-030-2	970	1065
3-11	6542-09-010-1	970	1065
3-12	6542-09-010-2	895	975
Cell 4-1	6542-11-015-1	945	1020
4-2	6542-11-015-2	1000	1115
4-3	6542-18-015-1	1000	1115
4-4	6542-18-015-2	1000	1115
4-5	6542-02-020-3	1000	1115
4-6	6542-02-020-4	1000	1115
4-7	6542-20-035-1	1000	1115
4-8	6542-20-035-2	1000	1115
4-9	6542-20-025-1	1000	1115
4-10	6542-20-025-2	940	1050
4-11	6542-25-015-1	940	1050
4-12	6542-25-015-2	870	970

TABLE 3-22
SUMMARY OF UNBONDED FERTILE PARTICLE TEMPERATURES FOR SAMPLES IRRADIATED
IN CELLS 3 AND 4 OF CAPSULE P13S

Capsule Position	Data Retrieval Number	Peak Fuel Temperature in Tray (°C)	
		Time Average	Maximum
Cell 3-1	4252-06-010-1	925	1035
3-2	4252-06-010-2	925	1035
3-3	6542-22-015-1	955	1075
3-4	6542-22-015-2	955	1075
3-5	6542-22-025-1	955	1075
3-6	6542-22-025-2	955	1075
3-7	6542-23-025-1	955	1075
3-8	6542-23-025-2	955	1075
3-9	6542-24-015-1	955	1075
3-10	6542-24-015-2	925	1035
3-11	6252-00-025-1	925	1035
3-12	6252-00-025-2	860	960
Cell 4-1	6542-17-010-1	910	1040
4-2	4252-02-010-1	950	1085
4-3	4252-02-010-2	950	1085
4-4	6542-16-010-1	950	1085
4-5	6542-16-010-2	950	1085
4-6	6542-16-010-3	950	1085
4-7	6542-19-015-1	950	1085
4-8	6542-19-016-1	950	1085
4-9	6542-21-016-1	950	1085
4-10	6542-21-015-1	900	1025
4-11	6542-12-025-1	900	1025
4-12	6542-12-025-2	830	940

TABLE 3-23
DECALIBRATION FACTORS DERIVED FOR W/Re THERMOCOUPLES IRRADIATED IN CAPSULES P13R AND P13S

Capsule-Cell	Thermocouple Location	Average Thermocouple Temperature, $\bar{T}$ (°C)	Coefficients for Linear Equation $\Delta T = a\phi_{Th} + b$		Decalibration Factor, m ($m = b/\bar{T}$) (n/cm ²) ⁻¹
			a (°C)	b (°C cm ² /n)	
P13R-1	Fuel	1150	17	-71	-0.0614
P13R-2	Fuel	1095	77	-70	-0.0636
P13R-5	Fuel	1305	-76	-112	-0.0855
P13R-6	Fuel	980	-69	-73	-0.0747
P13S-1	Fuel	1140	79	-69	-0.0604
P13S-2	Fuel	1100	59	-65	-0.0594
P13S-5	Fuel	1450	-19	-135	-0.0926
P13S-6	Fuel	1010	-57	-97	-0.0963

Fig. 3-1. GETR thermal flux profiles for three control rod bank positions

Fig. 3-2. Design load-following transient for cell 1 of capsule P13S

Fig. 3-3. Decalibration curves used to correct tungsten-rhenium thermocouple drift in cells 1, 2, 4, and 6 in capsules P13R and P13S. Data derived from cell 2 of capsule P13S.

Fig. 3-4. Decalibration curves used to correct tungsten-rhenium thermocouple drift in cell 5 in capsule P13R

Fig. 3-5. Decalibration curves used to correct tungsten-rhenium thermocouple drift in cell 5 in capsule P13S

Fig. 3-6. Operating history for cell 1 of capsule P13R

Fig. 3-7. Operating history for cell 2 of capsule P13R

Fig. 3-8. Operating history for cell 3 of capsule P13R

Fig. 3-9. Operating history for cell 4 of capsule P13R

Fig. 3-10. Operating history for cell 5 of capsule P13R

Fig. 3-11. Operating history for cell 6 of capsule P13R

Fig. 3-12. Operating history for cell 1 of capsule P13S

Fig. 3-13. Thermal cycle operating history for cell 1 of capsule P13S

Fig. 3-14. Operating history for cell 2 of capsule P13S

Fig. 3-15. Operating history for cell 3 of capsule P13S

Fig. 3-16. Operating history for cell 4 of capsule P13S

Fig. 3-17. Operating history for cell 5 of capsule P13S

Fig. 3-18. Operating history for cell 6 of capsule P13S

Fig. 3-19. Ratio of fission gas release to birth rate (R/B) versus fast fluence ($E > 0.18$ MeV)_{HTGR} for each cell in capsule P13R

Fig. 3-20. Ratio of fission gas release to birth rate (R/B) versus fast fluence ($E > 0.18 \text{ MeV}$)_{HTGR} for each cell in capsule P13S

Fig. 3-21. GETR fast and thermal neutron flux profiles for capsule P13R

Fig. 3-22. GETR fast and thermal neutron flux profiles for capsule P13S

Fig. 3-23. GETR fast and thermal neutron fluence profiles for capsule P13R

Fig. 3-24. GETR fast and thermal neutron fluence profiles for capsule P13S

Fig. 3-25. Comparison between measured and calculated (TAC-2D) temperatures for capsule P13R at BOL and EOL

Fig. 3-26. Comparison between measured and calculated (TAC-2D) temperatures for capsule P13S at BOL and EOL

Fig. 3-27. Measured thermocouple readings of fuel and graphite for sample A, cell 2 of capsule P13R

4. POSTIRRADIATION EXAMINATION RESULTS

After removal from the graphite crucibles, the unbonded particle samples and fuel rods were examined visually and dimensional measurements were made on all fuel rods. Fission gas retention of all fuel specimens was evaluated by neutron activation in the TRIGA reactor facility. Gamma-ray spectrometry analysis was performed on all fuel specimens and capsule components to evaluate metallic fission product release and transport during irradiation. Metallographic examination was performed on selected fuel rod and unbonded particle specimens to evaluate irradiation-induced changes in the fuel components. Contact microradiography and measurements of fuel particle and OPyC coating density were performed on selected unbonded particle samples. The results obtained from these examinations are presented in Sections 4.1 through 4.10.

4.1. CAPSULE DISASSEMBLY

The secondary containments were slit at 0° and 180° orientations and removed to expose the aluminum thermal bond spacers. All dosimeter vials, located in grooves of the thermal bond spacers, were recovered with the exception of two secondary dosimeter vials from capsule P13S. Transverse and longitudinal cracking of the protective tantalum thermocouple sheathing was observed, as expected. All purge gas lines were intact. The thermal bond spacers, purge gas lines, and thermocouples were removed, which exposed the primary containment of each cell. Dimensional measurements made on the thermal bond spacers and the primary and secondary containments showed no measurable change in these components during irradiation.

Fuel specimens were housed in interlocking graphite crucibles inside Inconel cans, which formed the primary containment for each cell. After removal of the Inconel cans, the fuel rods were exposed by lifting off the top segment of the graphite crucibles as shown in Fig. 4-1. The fuel rods

were removed by sliding them out of the graphite crucibles into a padded plastic box or by lifting them out with master-slave manipulators. All fuel rods were removed intact and irradiation-induced damage to the fuel rods was considered to be minimal.

Slight bowing of the graphite crucibles in cell 2 of both capsules was observed. The fuel rods were wedged in the graphite crucibles and although the fuel specimens were not damaged, they were removed with considerable difficulty. Subsequent dimensional measurements revealed the fuel specimens were restrained by the graphite crucibles as a result of shrinkage of the graphite crucibles and slight expansion of the fuel rods after irradiation to the severe fast neutron fluences [11.5 to 12.4×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR}] experienced by these cells.

Each unbonded particle cell contained a graphite crucible assembly, which consisted of twelve annular trays arranged on a threaded graphite stock as shown in Fig. 4-2. Neutron radiographs taken of the capsules after irradiation revealed that all four of the crucible assemblies were intact. However, at least one sample tray from each cell was spilled during capsule disassembly or the fuel unloading operation because the threaded graphite stock broke. The sample trays spilled were positions 3-6, 4-1, and 4-12 in capsule P13R and positions 3-11, 4-5, and 4-6 in capsule P13S. Subsequent failure analyses were not performed on these specific samples because only a portion of the fuel particles in each sample was recovered. However, duplicate samples of each particle type were tested in companion trays and were examined.

4.2. VISUAL EXAMINATION

General irradiation performance and integrity of the unbonded coated particle and fuel rod samples after irradiation were assessed by visual examination. The examination was conducted using an in-cell Bausch and Lomb stereomicroscope with a magnification range of 4 to 30X and a camera attachment for photography.

4.2.1. Fuel Rods

The criteria used during the visual examination for judging fuel rod integrity after irradiation are that the fuel rods remain intact and experience negligible particle debonding. The general irradiation performance of the fuel rods was good and excellent fuel rod integrity was demonstrated after irradiation to fast neutron fluences significantly exceeding LHTGR design limits. Very little irradiation-induced damage was observed, even in fuel rods irradiated to a fast neutron fluence of $12.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}, which is 55% beyond the peak LHTGR design fast neutron exposure. The visual appearance of the fuel rods after irradiation is summarized in Tables 4-1 and 4-2. Representative photomicrographs of each rod taken during the visual examination are shown in Figs. 4-3 through 4-22.

4.2.2. Unbonded Particle Samples

Each particle sample was examined visually and a measure of the OPyC coating integrity and pressure vessel failure was determined by counting the number of particles with defective coatings. The irradiation parameters and the results of the visual examination of each particle batch are given in Tables 4-3 through 4-5.

Visual examination of unbonded particle samples provides an accurate assessment of pressure vessel failure levels in BISO coated fertile particles and OPyC coating failure levels in TRISO coated fissile particles. In TRISO coated particles the fragments of failed OPyC coatings often remain attached to the SiC coating substrate as shown in Fig. 4-23. This made an accurate assessment of the pressure vessel failure levels in some TRISO coated particle batches impossible by visual examination. The pressure vessel failure levels reported in Table 4-3 are compared with failure levels determined by fission gas release measurements, which provide a more accurate assessment of TRISO coating failure (see Section 4.4.2).

Particle batches with coating designs representative of the design requirements envisioned for the LHTGR exhibited excellent irradiation performance as shown in Figs. 4-24 through 4-26. A number of fissile and fertile particle batches exhibited 0% failure levels after irradiation to fast neutron exposures which exceeded the LHTGR peak design exposure by 34 to 51%. BISO coated fertile particles with OPyC coatings deposited from acetylene-propylene (mixed gas) exhibited irradiation performance comparable to similarly designed particles with OPyC coatings deposited from propylene.

A large number of particle batches with coating designs or particle attributes that were not representative of LHTGR design requirements were tested. These samples were tested to investigate the effect of variables such as coating deposition rate, density, and faceting on fuel particle irradiation performance.

Fissile particles tested with OPyC coating densities in the range of 1.76 to 1.95 g/cm³ exhibited good integrity after irradiation to fast neutron fluences up to 12.1×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR}. One fissile particle batch (6151-02-025), which had an OPyC coating density of 1.50 g/cm³, exhibited an OPyC coating failure level of 46.3%. The poor irradiation stability of this OPyC coating was probably due to the low deposition rate (0.7 μ m/min) rather than the low density. OPyC coatings with densities as low as 1.53 g/cm³ have been irradiated successfully to low fast neutron exposures in previous capsule tests (Ref. 4-1). BISO coated fertile particles exhibited good integrity during irradiation to the severe fast neutron exposures with OPyC coating densities in the range of 1.73 to 1.94 g/cm³.

For the fissile and fertile particle designs tested, the OPyC coating deposition rate (anisotropy) had a greater influence on particle survival than OPyC coating density or thickness. In UC₂ TRISO batches where the primary variable tested was OPyC coating rate, no failure occurred in OPyC coatings deposited at rates greater than 3.8 μ m/min. As the deposition

rate decreased below 1.5 $\mu\text{m}/\text{min}$, the OPyC coating failure levels increased significantly. The onset of failure shifted to a lower coating rate as the OPyC coating density decreased. BISO coated fertile particles also exhibited similar irradiation performance as a function of OPyC coating deposition rate. As the deposition rate decreased below 4.0 $\mu\text{m}/\text{min}$, the fertile particle OPyC coating failure levels increased. Photographs of fissile and fertile particles with OPyC coatings deposited at relatively low rates are shown in Fig. 4-27.

Fuel particles with severely faceted coatings, nonround fuel kernels, or interrupted SiC layers exhibited higher failure rates than similarly designed fuel particles that did not have these attributes. Photographs of fissile and fertile particles that exhibited failure due to severely faceted coatings are shown in Fig. 4-28.

One TRISO coated ThO_2 fertile particle batch and two alternative TRISO coated fissile kernels, 250- μm -diameter $(\text{Th,U})\text{O}_2$ and 500- μm -diameter $(8\text{Th},1\text{U})\text{O}_2$, were tested to obtain additional irradiation data on these potential backup fuel particles. The TRISO coated 250- μm -diameter $(\text{Th,U})\text{O}_2$ fuel particles (Fig. 4-29) exhibited no coating failure during irradiation to a fast neutron fluence of $11.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. However, the larger diameter $(8\text{Th},1\text{U})\text{O}_2$ fuel particles (Fig. 4-23) exhibited a significant level of OPyC coating failure after irradiation to a fast neutron fluence of $11.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. The TRISO coated fertile particle batch (Fig. 4-30) exhibited OPyC coating and pressure vessel failure levels of 0.4% and 0%, respectively, after irradiation to a fast neutron exposure of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

4.3. DIMENSIONAL CHANGE ANALYSES

4.3.1. Fuel Rods

Preirradiation fuel rod dimensional measurements were made using calibrated micrometers. One length and six diametral measurements were made on each fuel rod to tolerances of $\pm 0.0013 \text{ cm}$. The diameter measurements were

made at 0° and 90° points along the rod; the selection of the 0° and 90° orientation was arbitrary (Ref. 4-2). For a fuel rod having a nominal diameter of 1.250 cm, the precision of the measurements was $\pm 0.1\%$.

The majority of the fuel rods were cured in graphite tubes; to prevent the rods from sticking to one another, pieces of Grafoil were placed on their ends. Normally only a portion of the Grafoil disc remained attached to the rods, although in some instances intact discs adhered to each end of a rod. Preirradiation length measurements included these discs. The nominal thickness of a disc was 0.013 cm.

Postirradiation fuel rod dimensional measurements were performed using a calibrated dial gauge micrometer that could be interpreted to ± 0.0013 cm. For comparative purposes selected rods also were measured using a ratcheted barrel micrometer similar to the one used to make the preirradiation measurements. Both micrometers were specially mounted on platforms to facilitate hot cell measurements (Fig. 4-31). Again six diametral measurements were made at randomly selected 0° and 90° orientations. Two length measurements were made on most of the fuel rods. Typically the standard deviation for the diametral measurements was 0.005 cm, i.e., $\pm 0.4\%$.

Irradiation-induced fuel rod dimensional changes were calculated to show the percent difference between the unirradiated and irradiated states. This was done for diametral, length, and volume changes. These data are summarized in Table 4-6 for capsule P13R and Table 4-7 for capsule P13S. The linear dimensional changes for rods irradiated at 1100°C are plotted in Fig. 4-32 and volumetric dimensional changes for these rods are shown in Fig. 4-33. The length and volume changes for rod 1B in P13R are not plotted in these figures since they appear to be erroneous measurements. The range for predicted dimensional change is also shown on each of these plots. The irradiation exposure for rods tested in cell 2 of P13R and P13S exceeds the limits of current dimensional change data for pyrocarbon and graphite shim; therefore, dimensional changes were not predicted for these rods.

Dimensional changes were predicted using the SHRINK computer code. The model assumes that the particles in the fuel rod are in point-to-point contact (close-packed array) and the fuel rod dimensional change is isotropic. It is also assumed that the percent volume change of a fuel rod during irradiation is equal to the percent volume change of the constituent particle types. The percent linear dimensional change of the rod is therefore equal to one-third the percent volume change and can be calculated using Eq. 4-1 if the dimensional change behavior of the constituent particle types as a function of neutron fluence and temperature is known:

$$\% \Delta D/D_0 = 1/3 \sum_i X_i (\% \Delta V/V_0)_i \quad , \quad (4-1)$$

where $\% \Delta D/D_0$ = percent diameter change of the fuel rod

$\% \Delta V/V_0$ = percent volume change of the i^{th} particle type

X_i = particle volume fraction of the i^{th} particle type

Two general trends can be seen from the plots presented in Figs. 4-32 and 4-33: (1) the rods do not shrink as much as predicted and (2) the rod dimensional changes are anisotropic. Regarding the first point, this trend is similar to that noted in capsules HRB-4, -5, and -6 (Ref. 4-3) where the average fuel rod diametral shrinkage was 36% less than predicted using the SHRINK code. These results are in reasonable agreement with the P13R and P13S results, which show the average diametral shrinkage to be 48% less than predicted. The HRB irradiations were the first tests of cured-in-place rods and graphite-shimmed rods. More recently, similar results for LHTGR type rods have been seen in capsule P13Q (Ref. 4-4).

Regarding the second point, this is the first GA capsule in which the rods exhibited notable anisotropic behavior. In experiments HRB-4, -5, and -6 (Ref. 4-3), as well as all previous tests (Ref. 4-1), the rods behaved isotropically. Although the general profile for the diametral dimensional changes for the P13R and P13S rods irradiated at 1100°C is consistent (Fig. 4-32), the axial dimensional changes appear more random. This

scattering of the axial data could be caused in part by the Grafoil discs attached to the ends of the rods. The Grafoil was noted to have partially flaked off on some of the rods; furthermore, this material is suspected to undergo volumetric expansion, especially in the axial direction. However, more recent irradiation tests (Ref. 4-4) also have shown LHTGR type rods to exhibit anisotropic behavior, and these results support the observed anisotropic behavior of the P13R and P13S fuel rods. The pair of rods that was cured in a packed Al_2O_3 bed behaved nearly isotropically. Measured and predicted dimensions were also in good agreement for the packed-bed rods. However, Grafoil discs were not placed on the ends of these rods.

The degree of anisotropy for rods irradiated in cell 2 of P13R and P13S to fluences of $>10.5 \times 10^{21} \text{ n/cm}^2$ partially resulted from the closing of the radial gap between the graphite crucible and the fuel rod (see Section 4.3.2). The interference between these components apparently caused the rods to expand axially as a result of the lack of accommodation of radial growth of the rod. It was noted that the two highest exposure rods in cell 1 of P13R and P13S (1D and 1E) had significantly greater axial expansions than the other rods in these cells. This may be an indication of interference between the rods and crucible wall at temperature, although room temperature gaps of 0.002 to 0.007 cm were measured for these rods.

4.3.2. Graphite Crucibles

Characterization of the irradiation-induced dimensional change behavior of the H-451 graphite crucibles was required for capsule thermal analysis. Prior to removal of the fuel rods, dimensional measurements were made on the graphite crucibles. Axial and radial measurements both perpendicular and parallel to the plane of the split in the crucibles were made with a dial gauge micrometer.

The measured diametral dimensional changes of the fuel rod graphite crucibles are reported in Table 4-8. Because some of the crucibles had bowed, data obtained by measurement parallel to the crucible split most accurately reflected the irradiation-induced dimensional change of the

crucibles; these data are plotted in Fig. 4-34 as a function of fast neutron fluence. Irradiation-induced dimensional changes ranged from -0.2 to -1.5%. The data were fit to a third-order polynomial given by:

$$\Delta l/l \text{ (\%)} = A\phi + B\phi^2 + C\phi^3 \quad , \quad (4-2)$$

where $A = -1.86 \times 10^{-2}$

$B = -2.49 \times 10^{-2}$

$C = 1.57 \times 10^{-3}$

$\phi = \text{fast neutron fluence (E > 0.18 MeV)}_{\text{HTGR}}$

Also shown in Fig. 4-34 are the predicted dimensional change curves for H-451 graphite in the radial direction for irradiation at 650° and 1050°C (Ref. 4-5). The observed dimensional change of the graphite crucibles (within the precision of the measurements) was in fair agreement with the predicted dimensional change behavior of H-451 graphite. In the capsule thermal analysis, Eq. 4-2 was used to calculate the gaps between the fuel rods and graphite crucibles because the predicted dimensional change curves for H-451 graphite only covered fast neutron fluences to $8.0 \times 10^{21} \text{ n/cm}^2$.

The radial gap between each fuel rod and the wall of the graphite crucible was calculated from the crucible and fuel rod dimensions; the results are given in Tables 4-9 and 4-10. The fuel rods tested in cell 2 of both capsules were wedged in the graphite crucibles after irradiation. The results shown in Tables 4-9 and 4-10 indicate the fuel rods in cell 2 of each capsule were restrained by the graphite crucibles after irradiation as a result of shrinkage of the crucibles and slight expansion of the fuel rods. Since the magnitude of the coefficient of thermal expansion is similar for the fuel rods and H-451 graphite, these fuel rods were probably also in contact with the wall of the graphite crucible during the latter part of the irradiation. No damage to the fuel rods or graphite crucibles was observed as a result of this mechanical interaction. It should be noted that this interaction only occurred in the cell 2 assemblies, which

were irradiated to fast neutron fluences considerably beyond the LHTGR peak design fast neutron fluence of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

4.4. FISSION GAS RELEASE

4.4.1. Postirradiation Fission Gas Release Measurements

Fission gas retention characteristics of fuel rod and unbonded particle samples were evaluated by fission gas release measurements conducted in the GA TRIGA reactor facility. Preirradiation measurements were made as part of the quality control evaluation to determine the amount of heavy metal contamination in the fuel specimens prior to irradiation. The primary objectives of the postirradiation measurements were (1) to cross check the in-pile R/B measurements made at the GETR during capsule irradiation, (2) to obtain R/B values for each fuel sample to better define fuel failure as a function of irradiation conditions, and (3) to compare the more statistically significant particle failure fractions predicted from R/B data with those determined by other postirradiation examination techniques.

Fission gas release measurements were made by irradiating fuel specimens at 1100°, 1300°, 1500°, or 1600°C in a graphite tube furnace designed to fit into a fuel element position of the TRIGA reactor core (Ref. 4-6). The furnace allowed measurements to be made at high temperatures, independent of the power generated in the fuel. The rods were irradiated at appropriate power levels for 0.5 hr to produce approximately 10^{14} fissions in each fuel specimen. During irradiation, the released fission gases were swept with helium into a liquid-nitrogen-cooled charcoal trap. The trapped gases were assayed for fission product isotopes by gamma-ray spectroscopy (Ref. 4-7) using a Ge(Li) detector and a 4096-channel analyzer. The fission product isotope birth fractions were calculated using the neutron flux in the TRIGA core and the end-of-life (EOL) fissionable material loadings. The amount of fissionable material remaining in each fuel rod was calculated using the Sigma 2 FISSIN computer code, which takes into account initial fuel loadings, capsule irradiation conditions, and the GETR power history and cross sections. The adjusted EOL fissile fuel loading for each specimen was

determined by expressing the sum of the weights of all remaining fissionable material (U-233 and U-235) in terms of U-235, producing the equivalent Kr-85m yield (Ref. 4-8).

The results of the preirradiation and postirradiation fission gas release measurements in terms of R/B for Kr-85m are reported in Tables 4-11 through 4-14. The Kr-85m isotope was selected as the reference isotope because it has a short half-life (4.4 hr) and its gamma-ray energy peak can easily be resolved. A reference temperature of 1100°C was used for analysis of fuel failure levels; however, fission gas release measurements were also made at temperatures up to 1600°C on selected fuel rods. The preirradiation R/B data obtained under non-steady-state conditions were corrected to steady state (multiplied by a factor of 1.9). However, the postirradiation R/B values were not corrected to steady state because a study showed that the R/B values obtained for irradiated fuel in the 0.5-hr TRIGA irradiation were essentially steady-state values (Ref. 4-9).

4.4.2. Determination of Fuel Failure Levels

Fuel particle failure levels for each specimen were determined from the fission gas release data. The fission gas released during TRIGA activation comes from two sources: (1) heavy metal contamination outside the particle coatings and (2) failed fuel particles, as shown by the following equation:

$$R/B_{EOI} = R/B_{\text{failed fuel}} + R/B_{c(U)} + R/B_{c(Th)} \quad , \quad (4-3)$$

where R/B_{EOI} = measured postirradiation fission gas release
 $R/B_{\text{failed fuel}}$ = fission gas release from failed fuel particles
 $R/B_{c(U)}$ = fission gas release from uranium contamination
 $R/B_{c(Th)}$ = fission gas release from thorium (U-233) contamination

A certain amount of uranium and thorium contamination in the fuel rods and on the surface of the coated particles is unavoidable during fabrica-

tion. Uranium and thorium contamination levels were determined for each fuel specimen prior to irradiation using fission gas release, thorium hydrolysis, or acid-leach tests (Ref. 4-2). The difference between the fission gas release resulting from contamination and the measured post-irradiation fission gas release (R/B_{EOL}) is the fission gas released from fuel particles that failed during irradiation.

4.4.2.1. Fuel Rod Specimens. The gaseous fission product release resulting from in-service fuel particle failure was calculated for each fuel rod specimen using the equation

$$R/B_{\text{failed fuel}} = R/B_{EOL} - \left[R/B_c(U) + R/B_c(Th) \right] \quad (4-4)$$

The values obtained are given in Tables 4-15 and 4-16 for P13R and P13S, respectively. The fission gas release arising from heavy metal contamination was calculated using the following equations:

$$R/B_c(U) = \left(R/B_{BOL} \right) \left(X_{U-235} \right) \quad (4-5)$$

$$R/B_c(Th) = \left(Th_c \right) \left(r/b_c \right) \left(X_{U-233} \right) \quad , \quad (4-6)$$

where R/B_{BOL} = measured preirradiation fission gas release

Th_c = level of thorium contamination (g Th/g Th)

r/b_c = fractional gaseous release from heavy metal contamination

X_{U-235} = fraction of fissions that occurred in U-235 at EOL

X_{U-233} = fraction of fissions that occurred in U-233 at EOL

A value for the fractional gaseous release from heavy metal contamination of 0.30 was used in these calculations (Ref. 4-9). The fraction of fissions occurring in each isotope at EOL was determined by summing the weights of all remaining fissionable material (U-233 and U-235) normalized to the thermal fission yields of Kr-85m.

When the postirradiation fission gas release (R/B_{EOL}) of nine of the fuel rods was corrected for heavy metal contamination, the irradiation-induced change in the R/B value became negative. This indicates that no fuel failure occurred during irradiation and the level of thorium contamination measured on a companion unirradiated fuel rod was slightly higher than in the fuel rod actually tested.

An equivalent level of in-service fuel failure (F) which would release the amount of fission gas attributed to failed fuel was calculated for each fuel rod using the following equation:

$$F = \frac{R/B_{\text{failed fuel}}}{r/b_f} \quad . \quad (4-7)$$

The values obtained are given in Tables 4-15 and 4-16 for P13R and P13S, respectively. All available data (Ref. 4-9) indicate that the mean fractional release of Kr-85m at 1100°C from failed fuel particles (r/b_f) constrained in a fuel rod is 5.0×10^{-3} . and this value was used in these calculations. However, it should be noted that for conservatism in LHTGR inventory calculations, a value of 0.02 is used (Ref. 4-10). The calculated in-service particle failure levels ranged from 0 to 24%. Twenty-five of the forty rods tested exhibited particle failure levels of less than 1.0%.

Extremely low levels of particle failure can be detected by fission gas release measurements. Each fuel rod contained approximately 2500 coated particles. Assuming a fractional release from failed fuel of 5.0×10^{-3} , an R/B of 2.0×10^{-6} corresponds to one failed particle (a failure level of 0.04%) in a rod.

Although low levels of particle failure in fuel rods can be detected, it is not possible to accurately differentiate between fissile and fertile particle failure using postirradiation R/B data alone. The maximum possible fissile and fertile particle failure levels were calculated for each fuel rod by assuming that all the fission gas recovered was released from

either the fissile or the fertile particles. The fraction of fissions occurring in each particle type, i.e., x_{fissile} or x_{fertile} , were used to calculate the particle failure fractions. Dividing the gaseous release from failed fuel by x_{fissile} in one case and x_{fertile} in the other case gave the maximum R/B for each particle type. For this reason, the particle failure fractions calculated from the R/B data using Eqs. 4-8 and 4-9 are termed maximum failure fractions:

$$F_{\text{fissile}} = \frac{R/B_{\text{failed fuel}}}{(x_{\text{fissile}}) (r/b_f)} , \quad (4-8)$$

$$F_{\text{fertile}} = \frac{R/B_{\text{failed fuel}}}{(x_{\text{fertile}}) (r/b_f)} . \quad (4-9)$$

The maximum in-service fertile particle failure levels calculated using Eq. 4-9 ranged from 0 to 26.3% (Tables 4-15 and 4-16). Since the failure levels calculated from the R/B data represent the maximum possible in-service failure, these failure levels are considered to be an upper limit for fertile particle failure in the fuel rods rather than the upper 95% confidence limit obtained during the metallographic examination (see Section 4.6.3).

Assuming that no fertile particle failure occurred, the maximum possible fissile particle failure levels calculated for the fuel rods using Eq. 4-8 ranged from 0 to 280%. These data were not reported in Tables 4-15 and 4-16 because failure levels greater than 100% indicate that the fissile particles did not contain sufficient fissionable material at EOL to account for all the measured postirradiation fission gas release. When the amount of fissionable material in the fissile particles is low relative to the fertile particles, a much smaller fraction of fissions occurs in the fissile particles than in the fertile particles at EOL, as shown in Tables 4-15 and 4-16. When this is the case, a low level of fertile particle failure gives an R/B equivalent to a much higher fissile particle failure fraction. Unless the fraction of fissions occurring in the fissile particles represents a significant fraction of the total fissions occurring

in the fuel rod at EOL, postirradiation fission gas release data cannot be used to accurately determine fissile particle failure levels.

4.4.2.2. Unbonded Particle Samples. Gaseous fission product release resulting from failed particles was calculated for the unbonded particles in a manner similar to the fuel rods; the values obtained are reported in Tables 4-17 and 4-18. The $R/B_{\text{failed fuel}}$ for UC_2 TRISO particles was calculated using the equation

$$R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - R/B_{\text{BOL}} \quad , \quad (4-10)$$

where R/B_{BOL} represents the contribution from uranium contamination. For the $(Th,U)O_2$ TRISO particles, the $R/B_{\text{failed fuel}}$ was calculated using Eq. 4-4, which takes into account both thorium and uranium contamination. The $R/B_{\text{failed fuel}}$ for the ThO_2 fertile particles was calculated using the following equations:

$$R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - R/B_{\text{c(Th)}} \quad , \quad (4-11)$$

$$R/B_{\text{c(Th)}} = (Th_c) (r/b_c) \quad , \quad (4-12)$$

where Th_c is the level of thorium contamination measured prior to irradiation and $r/b_c = 0.30$.

The in-service particle failure level was calculated for each sample using Eq. 4-7. The results are reported in Tables 4-17 and 4-18. The failure levels for all particle samples except selected UC_2 TRISO particle batches were calculated using a value of 2×10^{-2} for the fractional release from failed fuel (r/b_f) in an unconstrained geometry (Ref. 4-9). This value is applicable only when the particle coating is severely cracked yet remains attached to the fuel kernel. In a number of UC_2 TRISO particle batches, the observed coating failures resulted in bare kernels, i.e., complete loss of coating integrity. In these samples an r/b_f of 3.5×10^{-2} for bare kernels (Ref. 4-9) was used to calculate the failure levels.

4.4.3. High-Temperature Fission Gas Release Measurements

Fission gas release measurements were made on all fuel specimens at a reference temperature of 1100°C to evaluate fuel integrity. Additional measurements at temperatures up to 1600°C were made on eight fuel rods to gain additional data on the temperature dependence of fission gas release from reference-type LHTGR fuel rods and to compare the data obtained at these high temperatures with the fission gas release data obtained from fuel rods in cell 1 of capsule P13S that were thermal cycled to high temperatures periodically during irradiation.

The temperature dependence for steady-state fission gas release of short-lived isotopes can be represented by:

$$\frac{(R/B)_T}{(R/B)_O} = F_{\ell}^O \exp\left[\frac{Q_{\ell}}{R_g}\left(\frac{1}{T_O} - \frac{1}{T}\right)\right] + F_h^O \exp\left[\frac{Q_h}{R_g}\left(\frac{1}{T_O} - \frac{1}{T}\right)\right] \quad , \quad (4-13)$$

where $(R/B)_T$, $(R/B)_O$ = fission gas release at temperature T (°K) and at the reference temperature (1373°K)

F_{ℓ}^O , F_h^O = pre-exponential constants; for Kr-85m $F_{\ell}^O = 0.08$ and $F_h^O = 0.92$

Q_{ℓ} , Q_h = activation energy (kcal/mole) for the low-temperature and high-temperature processes, respectively; for Kr-85m $Q_{\ell} = 0.7$ and $Q_h = 16.5$

R_g = gas constant (1.987×10^{-3} kcal/mole-°K)

This equation is an empirical fit to fission gas release data (steady state) obtained during long-time, high-temperature anneals of Fort St. Vrain fuel particles. Equation 4-13 has been superseded by a semi-empirical equation given in Ref. 4-9. For short-lived isotopes in the temperature range of 1100° to 1500°C, the two equations yield R/B values which are not significantly different.

A summary of the high-temperature fission gas release measurements, including the 1100°C data, obtained for the eight fuel rods irradiated in capsules P13R and P13S is given in Tables 4-19 and 4-20. The data are also plotted as a function of reciprocal temperature in Figs. 4-35 through 4-42. In these figures the closed points indicate the fission gas release measurement was the first or only measurement made during the day and the open points indicate previous measurements had been made on the same day. The arrows indicate the order in which the measurements were made.

The fission gas release data for the three fuel rods in which only two measurements were made are plotted in Figs. 4-35 through 4-37. The dashed line is the predicted temperature dependence for steady-state fission gas release given by Eq. 4-13 and was calculated using the first measurement made on each rod. The agreement between the predicted and measured temperature dependence for fuel rods 7161-004-17-6 and 7161-004-19-5 was good. The fission gas release measured for fuel rod 7161-004-19-7 at 1100°C was slightly higher than the value predicted from 1300°C measurement. This may be due, in part, to the residual gas inventory remaining after the first measurement was made. This effect will be discussed in more detail later in this section.

Fission gas release data obtained for fuel rods 7161-004-03-5 and 7161-004-16-5 at temperatures up to 1600°C are shown in Figs. 4-38 and 4-39, respectively. The two reference R/B values (closed points) obtained for fuel rod 7161-004-16-5 at 1100°C are in good agreement. The fission gas release values obtained for both rods during successive measurements above 1100°C were significantly higher than the predicted levels of fission gas release based on a temperature dependence extrapolation from the 1100°C data. A portion of the observed increase in R/B at the high temperatures was attributed to particle failure induced during the TRIGA measurements.

Fission gas release data obtained for fuel rods 7161-004-14-5 and 7161-004-03-6 at temperatures up to 1600°C are shown in Figs. 4-40 and 4-41, respectively. The two reference R/B values (closed points) obtained for fuel rod 7161-004-14-5 at 1100°C are in good agreement. The R/B values

obtained for both rods at 1300°C were in excellent agreement with the predicted values based on a temperature dependence extrapolation from the 1100°C data. At temperatures above 1300°C the measured R/B values were significantly higher than the predicted values. After the high-temperature measurements, two measurements were again made at 1100°C. The final measurements (closed points) were made several weeks after the 1100°C measurements made upon completion of the high-temperature series (open points). The difference between the two R/B values was attributed to residual fission gases generated during the previous measurements. When the final measurements were made several weeks later, the residual gas inventory would have decayed away. The difference in the R/B values obtained during the final measurements and the original reference value at 1100°C was attributed to ~8.2% particle failure induced in both fuel rods during the heating in TRIGA to 1600°C.

The fission gas release data obtained for fuel rod 7161-004-17-5 at temperatures up to 1600°C are shown in Fig. 4-42. The first measurement was made at 1300°C (point 1) followed by an 1100°C measurement (point 2) on the same day. The R/B value obtained at 1100°C was higher than the predicted value based on the temperature dependence extrapolation. After several weeks a second measurement was made, again at 1100°C (point 3); this value was in excellent agreement with the predicted value. Successive measurements were then made at higher temperatures. The R/B measured at 1300°C (point 4) was almost identical to the value measured previously. At 1500°C the measured R/B (point 5) was close to the predicted value. However, the value obtained at 1600°C (point 6) was significantly higher than the predicted value, which indicates that particle failure occurred between the 1500° and 1600°C measurements. Measurements were made at 1100°C immediately after the high-temperature series (point 7) and several weeks later (point 8). The difference between the two R/B values was attributed to the residual gas inventory. The difference between the final R/B value (point 8) and the reference 1100°C value (point 3) was attributed to ~4.8% particle failure induced during the high-temperature measurements.

Several observations were made from these data. There was excellent agreement between the measured fission gas release for the eight fuel rods and the R/B values predicted from the temperature dependence for steady-state fission gas release (Eq. 4-13). Significant levels of fuel failure can be induced by heating highly irradiated fuel samples to 1600°C in TRIGA. A discussion of the fuel failure induced during in-pile thermal cycling and the TRIGA heat treatments is given in Section 5.1.3. If two or more successive fission gas release measurements are made on a fuel specimen within an 8-hr period, residual Kr-85m fission gas generated during the previous measurement can contribute to the release obtained during the subsequent measurements.

4.4.4. Comparison of In-pile and Postirradiation Fission Gas Release Measurements

In-pile fission gas release measurements (see Section 3.2) were used to evaluate fuel performance during irradiation. One of the objectives of the postirradiation fission gas release measurements was to verify the in-pile data. To accomplish this, an EOL in-pile fission gas release value for each cell containing fuel rods (cells 1, 2, 5, and 6) was determined from the TRIGA postirradiation fission gas release data. The contribution from each fuel rod to the total cell R/B was calculated using the equation

$$R/B_i = (R/B_{EOL})(T_{corr})(X_i) \quad , \quad (4-14)$$

where R/B_i = contribution to the cell R/B from the i^{th} fuel rod

R/B_{EOL} = postirradiation TRIGA fission gas release value (R/B Kr-85m at 1100°C) for the i^{th} fuel rod

T_{corr} = temperature correction factor for Kr-85m determined from Eq. 4-13

X_i = fraction of fissions in cell occurring in the i^{th} fuel rod

The TRIGA fission gas release data (R/B_{EOL}) were corrected to the volume-average irradiation temperature at which the fuel rods operated

when the in-pile fission gas release measurements were made. The contribution from each fuel rod was then added to obtain the calculated TRIGA EOL R/B for each cell. The results of these calculations are given in Tables 4-21 and 4-22.

The cell EOL R/B values calculated from the TRIGA postirradiation data and the EOL cell R/B values monitored in-pile were in good agreement for cells 1, 2, and 5 of P13R and for cells 1 and 5 of P13S. However, the EOL cell R/B values determined from the TRIGA data were significantly lower than the in-pile data measured for cell 6 of P13R and for cells 2 and 6 of P13S.

There are several possible reasons for the large discrepancy between the calculated and measured EOL R/B values for the three cells. The TRIGA R/B data are believed to be the most valid measure of fission gas release from the fuel specimens based on previous experience (Refs. 4-1, 4-9). It is possible the EOL volume-average fuel temperatures used in the calculations were lower than the actual operating temperatures of the fuel specimens at the time the in-pile measurements were made. However, even a temperature difference of 100°C would only increase the calculated TRIGA cell R/B values by a factor of two. When the R/B values were normalized to the fraction of fissions occurring in each fuel rod, a constant neutron flux across the cell was assumed. Although a steep flux gradient existed across cell 6 in each capsule, the effect could not account for such a large discrepancy between the calculated and measured R/B values.

It is possible that a systematic or calibration error was made for some of the cells at the GETR when the in-pile fission gas release was monitored. To evaluate this possibility, a comparison was made between the BOL cell R/B values measured in-pile and those calculated from TRIGA preirradiation data. This calculation was performed in the same manner as the EOL analysis; the results are given in Tables 4-23 and 4-24. At BOL the R/B value monitored in-pile for each cell was lower than the value calculated from the TRIGA fission gas release data. However, at these low

activity levels (in the 10^{-7} range), the differences between the measured and calculated values were not considered significant.

Based on these results, fission gas release data obtained from in-pile measurements at the GETR gave a qualitative indication of fuel performance throughout irradiation. However, to quantitatively evaluate levels of irradiation-induced fuel failure, postirradiation fission gas release measurements were required.

4.5. GAMMA-RAY SPECTROMETRY ANALYSES

Gamma-ray spectrometry was utilized to study fuel rods, unbonded particle samples, and fuel rod graphite crucibles. The analyses were conducted to provide information on fuel rod homogeneity, fuel specimen fission product inventories, and fission product transport.

Gamma counting was performed using a lithium-drifted germanium (Ge-Li) gamma-spectrometer coupled to a 4096-channel Sigma 2 computer analyzer. Single channel analyzers were used to print out relative profiles of individual isotopes. All specimens were counted in the low-level hot cell through a 42-in.-long collimator. Each type of specimen required a different size collimator cross section because of the different sample activities and geometries. Unbonded particle samples were positioned using a Nuclear Chicago automatic sample changer. Fuel rods and graphite crucibles were scanned axially using a special traversing unit that allowed a slow and continuous movement of the sample past the collimator.

4.5.1. Fuel Rods

Fuel rods were gamma counted using a 0.023-in. by 0.875-in. collimator that was positioned with the long axis perpendicular to the long axis of the fuel rod. A column of rods was moved past the collimator continuously at a speed of 0.05 in./min. Single channel scans for Cs-137 and Zr-95 using their respective 661- and 724-keV energy peaks were plotted continuously. These scans are shown in Figs. 4-43 through 4-46 for the fuel rods irradiated

ted in cells 1 and 5. Multichannel scans of 7-min duration over the 0 to 2000-keV energy range were made on 25 fuel rods. This ensured that one complete scan averaged over a 0.35-in. (0.05 in. x 7 min) long fuel zone was obtained for each fuel rod. The fission product isotope inventory for each rod obtained from the multichannel scans and calculated using the FISS-PROD computer program is given in Tables 4-25 and 4-26. The counting error of each fission product isotope is given in Table 4-27. A quantitative fission product isotope inventory was determined from the multichannel scans using the following equations:

$$\mu\text{Ci} = \frac{\text{DPM}}{2.22 \times 10^6} \quad (4-15)$$

and

$$\text{DPM} = \frac{\text{CPM}}{(\text{CE})(\text{AI})} \quad , \quad (4-16)$$

where DPM = disintegrations per minute of each isotope

CPM = counts per minute of each isotope, measured by the detector

CE = counting efficiency of the particular counting geometry

AI = absolute intensity of the various isotopes

Fuel rods 7161-004-18-6 (P13R, 5C) and 7161-004-18-5 (P13S, 5C) were used as internal standards. The theoretical DPM of each isotope in these rods was calculated using the FISS-PROD computer program. A counting efficiency versus peak energy calibration curve was constructed using the measured isotope CPM, the known values of the absolute intensities, and Eq. 4-16. A quantitative value for each isotope was calculated using this calibration curve and the measured CPM. The measured and theoretical fission product isotope inventories are in agreement within $\pm 20\%$. The differences between the two inventories are a composite of the counting error for each isotope and the error involved in the calibration technique used.

4.5.2. Graphite Crucibles

The fuel rod graphite crucibles were gamma counted to aid in the evaluation of metallic fission product release and transport from the fuel rods. The graphite crucibles were counted in a manner similar to the fuel rods except a 0.100-in. by 0.875-in. collimator was used. Plots of the Cs-137 profiles obtained for the crucibles and the positions of the multichannel scans are shown in Figs. 4-47 and 4-48. The only significant isotope activities detected were Ta-182, Cs-134, and Cs-137; these data are presented in Tables 4-28 and 4-29. The concentration of the cesium isotopes was calculated using Eq. 4-15 and Eq. 4-16. The relative activity of Ta-182 (1189 keV) was reported because the calibration curve used was only valid over the energy range of 300 to 750 keV.

The crucible from cell 1 of P13S, which was thermal cycled up to 1675°C during irradiation, exhibited the highest cesium activity. Low cesium levels were detected in the other crucibles, which indicated negligible fission product release from the fuel rods during irradiation.

4.5.3. Unbonded Particle Samples

All unbonded particle batches were gamma counted using an automatic sample changer to position plastic vials containing the particles in front of the collimator. The particle batches were counted using a 0.346-in. by 0.738-in. collimator. Due to the high activity of the particle batches, a 2-in.-thick lead shield was placed between the particle samples and the collimator during counting. Because of the lead shield, a reliable calibration for quantitative analysis of the data could not be made. Instead, a qualitative evaluation of metallic fission product retention of each particle batch was made by comparing the measured and calculated Cs-137/Zr-95 isotope ratios. The Zr-95 isotope was chosen as the reference isotope because of its physical stability and ease of measurement. The theoretical fission product inventory for each batch was calculated with the FISS-PROD computer program.

The measured and calculated Cs-137/Zr-95 isotope ratios of each particle batch examined are given in Tables 4-30 through 4-33. The BISO and TRISO coated fertile particle batches did not exhibit any measurable cesium loss. The measured Cs-137/Zr-95 isotope ratios of each particle batch were within 10% of the calculated theoretical ratio; however, they were systematically higher. This systematic difference was attributed to uncertainties in the input data to the FISS-PROD computer program used to calculate the theoretical ratios. The TRISO coated fissile particle batches retained from 80 to 100% of their cesium inventory. The relationship between metallic fission product loss and particle failure is discussed in more detail in Section 5.3.1.

The majority of particles from each of six different batches were split into groups of one to nine particles each and recounted to evaluate the variation of the Cs-137/Zr-95 ratio from particle to particle. The Cs-137/Zr-95 isotope ratio for each particle aliquot from these six batches is given in Table 4-34. The variation in the isotope ratio between the particle aliquots of most of the particle batches was low, as evidenced by the narrow standard deviation. The measured Cs-137/Zr-95 ratio for scan number 1170 of batch 6151-00-046-1 was approximately 20% lower than the mean isotope ratio. This indicates that at least one of the three particles in the aliquot had lost a significant portion of its Cs-137 inventory. If preparation of the particle aliquots could be automated, gamma-ray spectrometry would be a viable technique for determining failure levels in large particle batches.

4.6. METALLOGRAPHIC EXAMINATION

Metallographic examination was performed on 17 fuel rods and 14 unbonded particle samples to evaluate the irradiation-induced changes in the fuel particle coatings and fuel kernel microstructure and to observe the condition of the carbonaceous fuel rod matrix. Metallographic specimens containing 34 to 210 unbonded coated particles or bonded fuel rods were prepared in the metallographic hot cell using special techniques developed for

high-radiation metallography. After grinding, most specimens were reimpregnated with mounting compound to reduce kernel pullout during polishing. A second polished section of some fuel rods was taken to increase the population size of the fuel particles examined. Prior to examination, all polished specimens were passivated with a solution of 50/50 $\text{HNO}_3\text{-H}_2\text{O}$ to impede hydrolysis of the UC_2 fuel kernels. All specimens were examined under bright-field illumination and polarized light with a Lietz metallograph. Photographs of representative particles from some of the fuel rods examined metallographically are shown in Figs. 4-49 through 4-60.

An evaluation of the fissile, fertile, and inert particle coating failure observed in the fuel rods and unbonded particle samples was made during the metallographic examination; the results are summarized in Tables 4-35 through 4-37. The corresponding 95% confidence intervals for the failure levels, determined using a binomial distribution (Ref. 4-11), are also reported. The 95% confidence statement, which was based on the number of particles examined, means that an infinite population of particles with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

4.6.1. Fuel Rod Matrix

The carbonaceous matrix of all 17 fuel rods examined was judged to be in good condition after irradiation. No significant matrix cracking or other deleterious irradiation effects were observed. Representative photomicrographs of the matrix graphite and composite radial cross sections of the fuel rods are shown in Figs. 4-61 through 4-77.

Matrix macroporosity is made up of the large voids in the injected matrix of a fired rod. The voids are characterized by an absence of continuous matrix and a physical dimension in excess of about 100 μm . Macroporosity, or the lack of matrix, influences the amount of surface area available for bonding between the particles and matrix and thus is related to the strength and thermal conductivity of the fuel rod. Fuel rod matrix

macroporosities, determined by the point-count method (Ref. 4-12) from the diametral composite metallographic cross sections, are reported in Table 4-38.

The postirradiation fuel rod macroporosities ranged from 38 to 60 vol % of the matrix phase. Densification of the carbonaceous matrix during irradiation is reflected by an increase in the fuel rod macroporosity. Irradiation-induced change in the fuel rod macroporosities ranged from 0 to 112%. The precision of the measurements was estimated to be $\pm 2.5\%$ (absolute) at the 95% confidence level based on six measurements made on the same polished section of two different fuel rods. The uncertainty in the fuel rod macroporosity is probably greater than this, however, since macroporosity varies with location in a rod.

4.6.2. Fissile Particles

The condition of the TRISO coated fissile particles was considered to be excellent after irradiation to the severe fast neutron exposures capsules P13R and P13S received. Photomicrographs of representative fissile particles after irradiation are shown in Figs. 4-49 through 4-60 and Figs. 4-78 through 4-80.

A number of particle batches with coating parameters representative of LHTGR design requirements did not exhibit any failure. No coating failure was observed that could be attributed to thermochemical effects such as fuel kernel migration or fission product attack. A majority of the particle batches that exhibited significant failure levels were tested to evaluate the effect of a particular variable such as faceted coatings, PyC deposition rate, or high-temperature transients on fuel particle irradiation performance. Photomicrographs of fuel particles that suffered pressure vessel failure as a result of thermal cycling to 1675°C in-pile and during postirradiation heating to 1600°C in the TRIGA facility are shown in Figs. 4-81 and 4-82, respectively.

An example of fissile particle OPyC coating failure is shown in Fig. 4-83. In particles that did not suffer complete pressure vessel failure the OPyC coating failure was attributed to irradiation-induced dimensional change of the coatings during irradiation to the severe fast neutron exposures. No obvious occurrences of matrix-coating interaction were observed in the fuel rods examined metallographically.

Most of the SiC coating failure appeared mechanical in nature and occurred in the form of fine radial cracks, as shown in Fig. 4-84. It has been observed in previous examinations that the brittle SiC coatings are subject to a finite amount of damage during preparation of the metallographic specimens (Refs. 4-3, 4-13). This effect is illustrated by two unbonded particle samples [batch 6155-01-020 tested in P13S (4-4) and batch 6151-04-015 tested in P13R (3-12)]. The in-service particle failure levels determined by fission gas release measurements for these samples were 2.6% and 0.6% (see Section 4.4.2). These low particle failure levels indicate that the SiC coatings in most of the particles were intact. However, total coating (OPyC plus SiC coating) failure levels of 50 and 95% were recorded for batches 6155-01-020 and 6151-04-015, respectively, during the metallographic examination. These results indicate that a large fraction of the cracked SiC coatings observed was an artifact of sample preparation. For these reasons, the level of cracked SiC coatings observed in the fissile particles was not reported in Table 4-35.

Small laminar inclusions in the SiC coating, of the type shown in Fig. 4-85, were observed in 2 to 3% of the TRISO UC_2 particles examined metallographically. Since only one polished cross section of a particle was examined, the actual population of the inclusions would be greater. The presence of this type of inclusion in the SiC coating is usually observed in most batches of TRISO coated fuel particles. Previous electron microprobe analysis of TRISO particles revealed these inclusions consist almost entirely of carbon (Ref. 4-13). Although no SiC coating failure directly attributable to the presence of these inclusions was observed, if the inclusions were large or occurred frequently enough, they would be expected to adversely affect the mechanical integrity of the SiC coatings.

Levels of IPyC coating debonding and failure observed in the fissile particles are reported in Table 4-35. Relatively high levels of IPyC debonding, an example of which is shown in Fig. 4-86, were observed in most fissile particles. However, actual failure of the IPyC coating, with the exception of particles that suffered complete pressure vessel failure as a result of high-temperature postirradiation heating, was only observed in a few particle batches.

In particles where the IPyC coating remained bonded to the SiC, the buffer coating decoupled from the IPyC coating at the seal coat and densified onto the fuel kernel. In most particles the buffer remained attached to the IPyC coating only in a small localized area, thereby creating a gap at the buffer-IPyC interface which was predominantly on one side of the particle. Although the formation of a gap at the buffer-IPyC interface can affect the temperature profile within a particle (Ref. 4-14), the location of the gap was random with respect to the hot or cold side of the particle.

In one TRISO UC₂ particle, a deposit of carbon was present in the gap formed between the buffer and IPyC coating, as shown in Fig. 4-87. The deposit was optically active and had a columnar-type structure when viewed under polarized light. Similar carbon deposits observed in irradiated TRISO UO₂ particles were attributed to gas phase carbon transport.

Only a small fraction (<2%) of the buffer coatings of the TRISO UC₂ particles were cracked. This occurred only in particles where the fuel kernel had segregated into what appeared to be a fuel phase and regions of rejected carbon, as shown in Fig. 4-88. When the buffer coating cracked, the fuel kernel did not extrude into the resulting gap. In particles where the fuel kernels appeared to have segregated into a fuel phase and regions of rejected carbon, the fission product recoil zone at the buffer-kernel interface was often perforated with holes up to 30 μm in diameter, as shown in Figs. 4-88 and 4-89. The holes in the recoil zone may have resulted from a combination of fission recoil damage and interaction with fragments of the fuel kernel. In some unbonded TRISO UC₂ particle samples,

the structural coatings of a few particles failed, leaving the buffer coating on the fuel kernel as shown in Fig. 4-89.

Less than 1% of the buffer coatings were cracked in the TRISO (1Th,1U)O₂ particles (5466-37); however, 88% of the buffer coatings were cracked in the larger-diameter TRISO (8Th,1U)O₂ particles (6155-01-020). In some of the particles the fuel kernel extruded into the gaps formed in the buffer coating; however, interaction between the kernel and IPyC coating was not observed.

Three distinct types of microstructures were observed in the UC₂ fuel kernels. The kernels that operated in the high-temperature positions (average temperature above 1200°C) were spherical and had a homogeneous appearance, as shown in Fig. 4-90. Under high magnification small flakes of light-colored material with a particle size of ~1.0 μm could be seen evenly dispersed throughout the fuel kernels. Fuel kernels that operated at intermediate temperatures of 1000° to 1100°C appeared to have segregated into a fuel phase and regions of rejected carbon, as shown in Fig. 4-91. The kernels were irregular in shape and the high-carbon regions appeared optically anisotropic when viewed under polarized light.

The fuel kernels that operated at average temperatures of less than 1000°C were more spherical and contained large fission gas bubbles, as shown in Fig. 4-92. In many cases a substance filled the smaller fission gas voids or coated the inner surface of the larger voids. This material was probably carbon, and it appeared optically active under polarized light. The presence of 1 wt % thorium in most of the UC₂ fuel kernels did not have any noticeable effect on the kernel microstructure.

The appearance of the (8Th,1U)O₂ and (1Th,1U)O₂ fuel kernels irradiated as unbonded particle samples at an average temperature of 1090°C was similar to the low-temperature UC₂ particles, as shown in Figs. 4-79 and 4-80. The kernels were spherical and contained small fission gas bubbles dispersed evenly throughout the kernel. The appearance of the (1Th,1U)O₂

fuel kernels irradiated in fuel rod 7161-004-16-6 at 1375°C was similar to the unbonded particle samples; however, the fission gas bubbles had coalesced into large voids as shown in Fig. 4-93. Small white inclusions observed in the kernels are typically found in high burnup oxide fuels and probably consist of molybdenum, ruthenium, and technetium.

Significant diffusion of metallic fission products into the IPyC coating of the TRISO (Th,U)O₂ particles was not observed. Localized concentrations of metallic fission products were observed in the IPyC coatings of TRISO UC₂ particles tested as unbonded particle samples, but attack of the SiC coating did not occur. However, slight degradation of the SiC coating by metallic fission products was observed in TRISO UC₂ particles irradiated in fuel rods at average temperatures above 1050°C. A summary of the metallic fission product interactions observed in the TRISO UC₂ particles during the metallographic examination is given in Table 4-39.

A localized concentration of fission products was observed in the IPyC coating on the cool side, away from the fuel rod centerline, of most particles. If the buildup of fission products at the IPyC/SiC coating interface was large enough, the IPyC coating was forced away from the SiC coating as shown in Fig. 4-94. Reaction of metallic fission products with the SiC coatings was slight. Maximum penetrations of 2 to 5 μm were generally observed, except for two particles in fuel rod 7161-004-18-5 (capsule P13S, position 5C) where the penetration was observed to be slightly less than one-half the thickness of the SiC coating.

Fuel kernel migration was observed in TRISO UC₂ and TRISO (Th,U)O₂ particles irradiated in the high-temperature (cell 5) fuel rods. The direction of migration was toward the hot side of the fuel particles (in a direction perpendicular to the fuel rod centerline), although in a few particles the migration direction was almost parallel to the fuel rod centerline. This deviation probably resulted from a change in the temperature profile within the particles due to localized fuel rod inhomogeneities or fuel rod end effects. The location of the buffer/IPyC coating gap did not have any noticeable effect on the kernel migration direction.

Many of the UC_2 particles had a region of what appeared to be rejected carbon on the cool side of the kernel, as shown in Fig. 4-90. This region was optically active under polarized light and contained small white globules, which were probably a fuel-rich phase.

Photomicrographs of $(\text{Th,U})\text{O}_2$ fuel kernel migration are shown in Fig. 4-93. A crescent-shaped deposit of material, which appeared to be carbon, was present on the cool side of many of the kernels. A carbon deposit with a granular-type structure was also observed in the gap formed between the buffer and IPyC coating.

The magnitude of UC_2 and $(\text{Th,U})\text{O}_2$ fuel kernel migration was measured in six of the fuel rods; the data are presented in Table 4-40. The location of each particle with respect to the fuel rod centerline was established. Migration distances were determined from high-magnification photomicrographs by subtracting the diameter of the fuel kernel perpendicular to the temperature gradient from the distance of the hot kernel surface to the original position of the cool kernel surface. This measurement technique was developed from analyzing microradiographs of UC_2 fuel kernel migration and takes into account migration which has occurred as a result of kernel elongation and rejection of carbon. The difficulty with this measurement technique is that the original position of the cool surface of the fuel kernel cannot be established accurately, and measurements cannot be made if the fuel kernel is irregularly shaped. However, this technique provides a better estimate of the kernel migration distance for small-diameter fissile kernels than a technique where the distances from the hot and cool surfaces of the fuel kernel are measured relative to the SiC coating. In the fissile particles the formation of a gap at the buffer/IPyC coating interface often results in the translation of the fuel kernel toward one side of the particle, which would erroneously influence the migration distance measured.

The observed $(\text{Th,U})\text{O}_2$ kernel migration rate was faster than the UC_2 migration rate over the temperature range 1275° to 1375°C . This result was

expected based on available kernel migration data. The UC_2 kernel migration data presented in Table 4-40 are compared with predicted migration rates based on HTGR core design data in Section 5.1.1.2.

4.6.3. Fertile Particles

Photomicrographs of representative fertile particles irradiated in fuel rods are shown in Figs. 4-49 through 4-60. A number of particle batches with coating parameters representative of LHTGR design requirements did not exhibit any failure. A summary of the coating failure analysis obtained during the metallographic examination is given in Table 4-36. The majority of fertile particle batches that exhibited significant failure levels were tested to evaluate the effect of a particular variable such as faceted coatings, OPyC coating deposition rate, or high-temperature transients on fuel particle irradiation performance.

The BISO OPyC coatings of all particle batches appeared relatively isotropic under polarized light. Photomicrographs of BISO fertile particles with OPyC coatings deposited from propylene and from mixed gas are shown in Figs. 4-95 and 4-96, respectively. No differences were observed in the appearance of the two types of coatings.

The highest levels of OPyC coating failure were observed in particle batch 6542-20-035 (fuel rod 7161-004-01-7) which had faceted coatings and was thermal cycled in-pile (Fig. 4-81) and in particle batch 6542-01-020 (fuel rod 7161-004-16-5) which was heated in TRIGA to 1600°C after irradiation. Particles with severely faceted coatings (Figs. 4-97 and 4-98) exhibited slightly higher failure levels than particles without faceted coatings. Fuel particles with nonround fuel kernels (Fig. 4-99) also exhibited higher failure. Examples of irradiation-induced failure in BISO ThO_2 particles with high- and low-density OPyC coatings are shown in Figs. 4-100 and 4-101, respectively.

Very fine microcracks were observed in the OPyC coatings of two unbonded particle samples (4252-06-010-1 and 6542-20-035-2), as shown in

Figs. 4-102 and 4-103. Microcracks of the size observed in these two particles would probably not be observed during the visual examination.

In all but two BISO ThO₂ particle batches, a thin high-density seal coat was applied between the buffer and OPyC coating. During irradiation the seal coat cracked and densified, leaving numerous small gaps at the buffer/OPyC coating interface as shown in Fig. 4-104. The breakup of the seal coat did not initiate any cracks in the adjacent buffer or OPyC coatings. One BISO ThO₂ particle batch (6542-18-015) was examined that did not have a seal coat, as shown in Fig. 4-105. The absence of the thin seal coat did not have any apparent effect on the BISO coating irradiation performance. In some particles the buffer coating delaminated and shrank away from the OPyC coating.

The buffer coatings densified during irradiation and appeared slightly optically active under polarized light. High levels of buffer cracking were noted in most particle batches; however, this did not appear to be detrimental to the stability of the BISO coatings during irradiation. In many particles the buffer coatings cracked and densified while the OPyC coating remained intact, as shown in Fig. 4-106. Propagation of buffer coating cracks into the adjacent OPyC coating, as shown in Fig. 4-107, was only observed in a few particles. The buffer coatings deposited using N₂ carrier gas (Fig. 4-108) appeared similar to coatings deposited using Ar as the carrier gas. Although a significant fraction of the buffer coatings deposited using N₂ carrier gas (6542-12-025-2) were broken, no apparent differences between the coatings applied using Ar or N₂ carrier gases could be ascertained.

One batch of TRISO ThO₂ particles, shown in Fig. 4-109, was irradiated for comparative purposes. Thirty-seven particles were examined metallographically. The PyC coatings appeared relatively isotropic under polarized light. The OPyC coating of one particle was failed; however, the SiC coating was intact. The SiC coating of one particle was cracked; this was the only instance where the IPyC coating had also debonded from the SiC coating. The buffer coating debonded and shrank away from the IPyC coating in ~50% of

the particles, and in four of the particles the buffer coating was cracked. A reaction zone was observed on the inner surface of the SiC coating of one particle, as shown in Fig. 4-110. The IPyC coating was intact although a portion of the buffer coating pulled out of the particle during polishing. No signs of metallic fission products were observed in the IPyC coating and the reaction probably was a result of oxidation.

Only one defective particle was observed in the fuel rods during the metallographic examination. The BISO fertile particle shown in Fig. 4-111 had a broken OPyC coating during fuel rod fabrication. Matrix material was bonded to the buffer coating in the region where the OPyC coating is missing. During irradiation the coating broke and the fuel kernel converted to carbide and subsequently hydrolyzed.

In some fertile particles where irradiation-induced coating failure occurred early in life, the fuel kernels also partially converted to carbide. An example of this is shown in Fig. 4-107 where partial conversion of a localized region of the ThO_2 kernel near a crack in the OPyC coating has occurred.

Irradiation-induced changes in the ThO_2 fuel kernel microstructure were quite apparent. At low burnup (<3% FIMA) the kernel grain structure was still visible under polarized light. After irradiation to higher burnup, (>5.6% FIMA) formation of pores in the ThO_2 grains occurred as shown in Fig. 4-112. In many particles the ThO_2 grain structure was completely obliterated, although large variations were observed in the microstructures of particles within a given sample. In one particle significant grain boundary thickening occurred on the hot side of the particle, as shown in Fig. 4-113. The grain boundary region was optically active under polarized light, which indicates carbon may have been present (Ref. 4-15).

4.6.4. Inert Particles

All but two fuel rods contained TRISO carbon inert particles in order to achieve close particle packing and the desired fuel particle and shim

loadings. Photomicrographs of a representative inert particle are shown in Fig. 4-114. The OPyC coating of the TRISO inert particles was deposited at a low coating rate of 1.1 $\mu\text{m}/\text{min}$. Consequently significant levels of irradiation-induced OPyC coating failure occurred in the particles, as shown in Table 4-37. Many of the particles exhibited cracked SiC coatings. The broken SiC coatings were attributed to metallographic polishing damage. Since the particles contain inert kernels, irradiation-induced failure resulting from the effects of high burnup would not occur.

4.7. OPyC COATING PERMEABILITY STUDIES

A design requirement for BISO coated particles, in addition to irradiation performance requirements, is that the OPyC coating must be sufficiently dense and impermeable to retain fission gases. Previous results on unirradiated BISO ThO_2 particles without seal coats have indicated that low-density OPyC coatings are permeable to fission gases; however, the fission gas release through OPyC coatings with densities of 1.90 g/cm^3 or higher was immeasurably low (Refs. 4-1, 4-9).

Fission gas release measurements were made on all fuel specimens after irradiation in capsules P13R and P13S (see Section 4.4). A description of the BISO ThO_2 particle batches in which no coating failure was observed during the visual examination and the results of the fission gas release measurements are summarized in Table 4-41. The measurements revealed a large variation in the amount of gaseous release from the unfailed BISO coated particles. A low level of gaseous release ($R/B < 1 \times 10^{-6}$) from intact fuel particles results from heavy metal contamination. However, the higher levels of gaseous release indicate the PyC coatings are permeable to fission gases. The permeability may result from diffusion through a porous coating microstructure or through microcracks that are not visible with a 20X microscope. The fission gas release values obtained for the ten particle batches are plotted in Fig. 4-115 as a function of preirradiation OPyC coating density. The data show an increase in gaseous release as the OPyC coating density decreases, which indicates the level of gaseous release is

probably related to the PyC coating microstructure rather than to the existence of microcracks.

In addition to the results discussed above, the OPyC coating permeability of several BISO ThO₂ particle batches described in Table 4-42 was evaluated using acid-leach and fission gas release measurements. The batches were selected to provide a range of OPyC coating densities and coating rates. Each batch was examined visually using an in-cell stereomicroscope and approximately 200 intact particles were selected. The particle samples were cleaned with diluted nitric acid in an ultrasonic cleaner and rinsed with acetone to remove any surface contamination. Fission gas release measurements were then made on each sample; the results are given in Table 4-42. Visual examination revealed that a low level of coating failure had occurred in two of the particle samples during the fission gas release measurements. The measured fission gas release is plotted in Fig. 4-116 as a function of preirradiation OPyC coating density. The data are consistent with the results obtained for entire particle batches (Fig. 4-115) and indicate that gaseous release increases as the OPyC coating density decreases.

The particles were then leached in an acid (13M HNO₃, 0.1M HF) solution. Aliquots of the acid solutions were taken at regular intervals and analyzed for thorium using a spectrophotometric technique. Results of the analysis, with the exception of batch 6542-01-010-1 in which the acid solution was contaminated during leaching, are presented in Table 4-43. To allow a comparison of data between samples, the leach results from batch 6542-01-020-2 had to be normalized to the other samples since it only contained 55 particles. This was done by multiplying the micrograms of thorium leached from the sample by 200/55.

The condition of the particle samples after leaching was evaluated by visual and metallographic examination; the results are summarized in Table 4-44. Several of the samples, particularly batches 6542-22-015-1 and 6542-01-020-2, exhibited significant levels of OPyC coating failure after leaching. The failure was attributed to the attack of the OPyC coatings

by the acid solution after extended periods of exposure. The rapid rate at which thorium was leached from batch 6542-22-015-1 was attributed to the onset of particle breakage early in the leaching process. To minimize the effect of particle failure contributing to the amount of thorium leached from the particle samples, only the data from aliquots taken during the first 21 hr of hot leaching were analyzed.

Results of the acid leaching experiments, with the exception of particle batches 6542-01-010-1 and 6542-22-015-1, are shown in Fig. 4-117, where the amount of thorium leached is plotted as a function of hot leaching time. The effect of OPyC coating density on the leachability is significant. The entire thorium content of batch 6542-21-016-1, which had an OPyC coating density of 1.73 g/cm^3 , was leached out within 11 hr, whereas only 0.7% of the thorium content of batch 6542-24-015-1, which had an OPyC coating density of 1.94 g/cm^3 , was leached out after 15 hr.

The OPyC coating rate (numbers shown in parentheses in Fig. 4-117) also had an effect on the leachability of the particles, but it was not as pronounced as the effect of OPyC coating density. By comparing the results from particle batch 6542-01-020-2 with 4252-06-018-1 and batch 6542-09-010-2 with 6542-02-020-2 it can be seen that for approximately the same OPyC coating density, the leachability increases slightly as the coating rate increases.

Representative photomicrographs taken during the metallographic examination of these particle batches after acid leaching are shown in Figs. 4-118 through 4-121. The ThO_2 fuel kernels were completely leached from the particle batches with low-density OPyC coatings. In the particle batches with high-density OPyC coatings, approximately 90% of the fuel kernels showed no signs of degradation from the acid leaching, which indicates the PyC coatings were impermeable to the leach solution. The OPyC coatings were affected by the acid leaching. The low-density OPyC coatings, with the exception of batch 6542-01-020-2, appeared more porous and pitted relative to the higher-density coatings, which showed little sign of degradation. The OPyC coating of batch 6542-01-020-2, which had a density of 1.80 g/cm^3

and a coating rate of 2.7 $\mu\text{m}/\text{min}$, had a smooth appearance, similar to the high-density OPyC coatings, except all the fuel kernels were heavily leached.

As mentioned previously the particle samples exhibited varying levels of coating failure after leaching; consequently, only leach aliquots taken during the first 21 hr of hot leaching were analyzed. Coating failure undoubtedly contributed to the levels of thorium leached from the samples, particularly batches 6542-02-020-2, 6542-01-020-2, and 4252-06-018-1. However, in most particle samples the amount of thorium recovered greatly exceeded the amount that would be exposed to the leach solution as a result of the observed level of coating failure.

The leachability of the particle samples correlated with the fission gas release measurements made prior to the acid leaching. The particle samples with low-density OPyC coatings that exhibited the highest thorium leaching rates also exhibited the highest fission gas release.

4.8. SIMULATED REPROCESSING STUDIES

After a 4-year irradiation, spent fuel elements will be removed from the reactor and reprocessed for fuel recovery and fission product removal. Utilization of a TRISO coated all-uranium fissile kernel and a BISO coated fertile particle will improve the recovery and separability of U-233 from U-236 during reprocessing, which is an integral part of the thorium fuel cycle (Ref. 4-16). During reprocessing, the elements will be mechanically crushed and burned to eliminate the fuel element graphite and remove the OPyC coatings from the fuel particles. A leach - solvent extraction process will permit recovery of the U-233 bred in the fertile particles. The fissile particle SiC coating, which remains intact during the burning and leaching operations, prevents the recovered U-233 from being significantly contaminated with U-236 (a neutron poison) that was bred in the fissile particles. To ensure that a sufficient fraction of the fissile particles remain intact, the following criterion has been established (Ref. 4-17): the fissile (TRISO) fuel coating failure from all effects

after irradiation at $\leq 1250^{\circ}\text{C}$ to $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and/or 78% FIMA shall be $< 10\%$ after removal of the OPyC coating by oxidation at $\leq 1000^{\circ}\text{C}$ in the reprocessing facility. To provide data for support of this criterion, two fuel rods were burned in air and fissile particle failure fractions were determined by visual examination and acid leaching.

The fuel rods selected for the burn-back studies were 7161-004-02-6 (P13R, 1B) and 7161-004-11-6 (P13S, 2A). The rods were burned separately in the high-level hot cell using a small muffle furnace. Each rod was placed in an alumina crucible and burned in air at 950°C for 6 hr. After burning the crucibles were allowed to cool overnight before they were removed from the furnace.

Remnants of fuel rod 7161-004-02-6 after the burn-back are shown in Fig. 4-122. All of the fuel rod matrix graphite and fuel particle OPyC coatings were completely oxidized. The remaining fuel particles, ThO_2 kernels and TRISO coated fissile particles, were covered with a white ash. The particles were cleaned in an ultrasonic bath using concentrated nitric acid, rinsed several times in acetone, and air dried. The ThO_2 kernels and TRISO coated fissile particles were then segregated in a separatory funnel using tetrabromoethane with a density of 2.97 g/cm^3 as the solution. The fissile particles were the low-density (2.6 g/cm^3) fraction, and the ThO_2 kernels were the high-density ($> 10.0 \text{ g/cm}^3$) fraction. Photographs of the ThO_2 kernels and TRISO fissile particles recovered from the separatory funnel are shown in Fig. 4-123. During burn-back, fissile particles with failed coatings will completely oxidize leaving SiC coating hulls. Visual examination of the recovered particles revealed 1401 ThO_2 kernels, 1007 intact fissile particles, and very few SiC coating fragments. Based on the original particle weights, approximately 1120 fissile and 1400 fertile particles were estimated to be in the fuel rod. These data indicate that 100% of the ThO_2 kernels were recovered and approximately 90% of the fissile particles were recovered intact. The remaining 10% of the fissile particles were assumed to be failed after the fuel rod burn-back.

The remnants of fuel rod 7161-004-11-6 (P13S, 2A) were boiled for 15 hr in an acid solution consisting of 13M HNO_3 and 0.1M HF. The chemical analysis of the leach solutions is given in Table 4-45. The results indicate that the end point of the leach was reached in a very short time and all the exposed uranium had been dissolved after 3.5 hr of leaching. The fissile particle failure fraction was determined using isotopic dilution mass spectroscopy by the standard additions technique of aliquots of the acid solution taken at regular intervals. The fourth aliquot of the leach solution was submitted for isotopic dilution mass spectroscopy to determine the uranium isotopic composition. These results are given in Table 4-46. The theoretical uranium isotopic composition of the fuel rod at EOL + 120 days was calculated using the FISSIN computer code, and the results are given in Table 4-47. Because the U-238 isotope is unique to the fissile particles, the fissile particle failure fraction (x) can be determined by:

$$x = \frac{\text{weight U-238 in leach solution}}{\text{weight U-238 in fuel rod}} \quad . \quad (4-17)$$

Using Eq. 4-17 and the data given in Tables 4-46 and 4-47, the calculated fissile particle failure fraction for rod 7161-004-11-6 after the burn-back was 4.2%.

After leaching, the particles were examined visually and metallographically. Only three failed TRISO particles were observed out of approximately 2600 TRISO coated fissile and carbon inert particles recovered from the leach solution. Metallographic examination of a portion of the recovered particles revealed no SiC coating attack or failure due to the acid leaching (Fig. 4-124).

In summary, two fuel rods that had been irradiated to fast neutron fluences of 8.2×10^{21} and 11.6×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} at 1050° and 1040°C, respectively, were burned in air at 950°C. After the fuel rod matrix graphite and OPyC coatings were completely oxidized, fissile particle failure levels of 10% and 4.2% were ascertained by visual examination and acid leaching, respectively. These results indicate that the design

criterion required for reprocessing, namely that $\geq 90\%$ of the TRISO coated fissile fuel maintains structural integrity after irradiation to peak conditions followed by oxidation of the fuel rods, can be achieved.

4.9. CONTACT MICRORADIOGRAPHY

Contact x-ray microradiographs of five unbonded particle samples were obtained by exposing high-resolution photographic plates to an x-ray pulse from a Picker 110-kV x-ray unit. The particle images on the radiograph plates were then examined to determine the postirradiation particle dimensions, which are given in Table 4-48. Photographs taken of microradiographs of representative fissile and fertile particles are shown in Fig. 4-125. The light regions within the SiC coating of the fissile particles are concentrations of metallic fission products that have diffused out of the fuel kernels. The irradiation-induced change in the particle dimensions are given in Table 4-49. These data are used to determine kernel swelling and irradiation-induced particle dimensional changes, which must be known for fuel rod dimensional change and coated particle design studies.

4.10. DENSITY MEASUREMENTS

4.10.1. OPyC Coatings from Fissile Particles

Postirradiation density measurements were made on OPyC coatings from nine fissile particle batches irradiated in capsules P13R and P13S to extend the data base on irradiation-induced PyC densification to fast neutron fluences up to $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Coating fragments were stripped from intact TRISO particles and were recovered from failed particles. The fragments were cleaned in an ultrasonic bath and examined by x-ray microradiography. All fragments that contained pieces of SiC coating were removed from the samples. A density column technique similar to the one described in Ref. 4-18 was employed to measure the density of the OPyC coating fragments. Benzene and tetrabromoethane were blended to provide a uniform density gradient column medium, and a density gradient curve for the liquid column was established

with calibrated glass density floats. Postirradiation OPyC coating densities determined for the nine batches of coated particles are reported in Table 4-50.

4.10.2. Fertile Particles

Postirradiation particle density measurements were made on four unbonded fertile particle batches. Sample preparation and density measurements were conducted in the hot cell facility. Small samples were split from the irradiated particle batches and examined using an in-cell stereomicroscope to ensure that all the particles were intact. The samples were cleaned in an ultrasonic bath to remove any surface contamination. A density column technique similar to the one described in Ref. 4-17 was employed to measure the density of the coated particles. Two concentrations of aqueous thallium malonate were blended to provide a uniform density gradient column medium, and a density gradient curve for the liquid column was established with calibrated glass density floats.

Results of the density measurements are given in Table 4-51. Irradiation-induced densification of the BISO coated particles ranged from 7.0 to 15.5%. These data will be used in future BISO particle model and design studies.

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TABLE 4-1
DESCRIPTION AND CONDITION OF FUEL RODS IRRADIATED IN CAPSULE P13R

Fuel Rod Description							Irradiation Parameters		Postirradiation Condition	
Capsule Position	Fuel Rod Data Retrieval No. (7161-004-)	Shim		Matrix			Average Centerline Temperature (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Number of Particles Debonded	Visual Appearance (b)
		Particle Type	Loading (vol %)	Filler	Additive	Firing Conditions (a)				
1A	01-5 (c)	GLCC 1099	10.0	6353	SC011	Ref.	1115	7.3	50	Fair
1B	02-6	GLCC 1099	35.3	6353	SC011	Ref.	1050	8.1	0	Excellent
1C	03-5	GLCC 1099	36.3	6353	SC011	Ref.	1065	8.7	8	Excellent
1D	04-6 (c)	GLCC 1099	20.0	6353	SC011	1500°C	1115	9.3	5	Excellent
1E	05-6	GLCC 1099	25.0	6353	SC011	Ref.	1125	9.9	0	Excellent
2A	06-5 (c)	GLCC 1099	20.0	6353	SC011	1500°C	1050	11.8	0	Excellent
2B	07-5	GLCC 1099	25.0	KS-44	SC011	Ref.	1055	12.0	0	Excellent
2C	08-13 (d)	GLCC 1099	30.0	6353	SC011	Packed bed	1045	12.2	0	Excellent
2D	09-5	Lonza	25.0	6353	SC003	Ref.	1035	12.3	35	Fair
2E	10-5 (c)	GLCC 1099	33.0	6353	SC011	Ref.	965	12.4	0	Excellent
5A	16-5 (c)	GLCC 1099	25.0	6353	SC011	Ref.	1285	9.4	21	Good
5B	17-5	GLCC 1099	30.0	6353	SC011	Ref.	1225	9.0	0	Excellent
5C	18-6	GLCC 1099	20.0	6353	SC011	1500°C	1235	8.5	10	Good
5D	19-7	GLCC 1099	20.0	6353	SC011	N ₂ atmosphere	1225	8.2	10	Good
5E	20-6 (c)	GLCC 1099	10.0	6353	SC011	Ref.	1280	7.7	20	Good
6A	21-5 (c)	GLCC 1099	25.0	6353	SC011	Ref.	1065	5.4	0	Excellent
6B	22-5	GLCC 1099	15.0	6353	SC011	Ref.	1055	5.0	8	Excellent
6C	23-5 (d)	GLCC 1099	30.0	6353	SC027	Ref.	1005	4.5	0	Excellent
6D	24-5	Lonza	20.0	6353	SC003	Ref.	1020	4.1	30	Fair
6E	25-5 (c)	GLCC 1099	25.0	6353	SC011	N ₂ atmosphere	1000	3.6	0	Excellent

(a) Reference firing conditions are carbonization to 1800°C in an argon atmosphere in an H-451 graphite tube to simulate the LHTGR cure-in-place process. Firing conditions that vary from this process are noted.

(b) Relative condition of fuel rod based on visual appearance; e.g., excellent condition indicates rod exhibited no significant matrix spalling, matrix cracking, or particle debonding.

(c) Thermocouple fuel rods.

(d) Fuel rods were fabricated by slug-injection process.

TABLE 4-2
DESCRIPTION AND CONDITION OF FUEL RODS IRRADIATED IN CAPSULE P13S

Capsule Position	Fuel Rod Description						Irradiation Parameters		Postirradiation Condition	
	Fuel Rod Data Retrieval No. (7161-004-)	Shim		Matrix			Average Centerline Temperature (°C)	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}	Number of Particles Debonded	Visual Appearance ^(b)
		Particle Type	Loading (vol %)	Filler	Additive	Firing Conditions ^(a)				
1A	01-7 ^(c)	GLCC 1099	10.0	6353	SC011	Ref.	1030 ^(d)	7.0	15	Fair
1B	02-5	GLCC 1099	35.3	6353	SC011	Ref.	975 ^(d)	7.7	0	Excellent
1C	03-6	GLCC 1099	36.3	6353	SC011	Ref.	995 ^(d)	8.5	2	Excellent
1D	04-5	GLCC 1099	20.0	6353	SC011	1500°C	1035 ^(d)	9.1	3	Excellent
1E	05-5 ^(c)	GLCC 1099	25.0	6353	SC011	Ref.	1035 ^(d)	9.6	1	Excellent
2A	11-6 ^(c)	GLCC 1099	20.0	6353	SC011	N ₂ atmosphere	1040	11.5	0	Excellent
2B	12-5	GLCC 1099	25.0	6353	SC011	Ref.	1010	11.7	2	Excellent
2C	13-6	GLCC 1099	30.0	6353	SC027	Ref.	975	11.9	2	Excellent
2D	14-5	GLCC 1099	25.0	6353	SC011	Ref.	985	12.0	0	Excellent
2E	15-5 ^(c)	GLCC 1099	33.0	6353	SC011	Ref.	960	12.1	8	Good
5A	16-6 ^(c)	GLCC 1099	25.0	6353	SC011	Ref.	1375	9.2	6	Excellent
5B	17-6	GLCC 1099	30.0	6353	SC011	Ref.	1330	8.7	0	Excellent
5C	18-5	GLCC 1099	20.0	6353	SC011	1500°C	1335	8.4	4	Good
5D	19-5 ^(c)	GLCC 1099	20.0	6353	SC011	N ₂ atmosphere	1325	7.9	13	Fair
5E	20-5 ^(c)	GLCC 1099	10.0	6353	SC011	Ref.	1350	7.5	0	Excellent
6A	26-5 ^(c)	GLCC 1099	25.0	6353	SC011	Ref.	1085	5.3	4	Excellent
6B	27-5	GLCC 1099	15.0	6353	SC011	Ref.	1045	4.8	50	Fair
6C	28-13	GLCC 1099	30.0	6353	SC011	Packed bed	1050	4.4	0	Excellent
6D	29-5	GLCC 1099	20.0	KS-44	SC011	Ref.	1015	3.9	0	Excellent
6E	30-5 ^(c)	GLCC 1099	25.0	6353	SC011	1500°C	995	3.5	2	Good

^(a)Reference firing conditions are carbonization to 1800°C in an argon atmosphere in an H-451 graphite tube to simulate the LHTGR cure-in-place process. Firing conditions that vary from this process are noted.

^(b)Relative condition of fuel rod based on visual appearance; e.g., excellent condition indicates rod exhibited no significant matrix spalling, matrix cracking, or particle debonding.

^(c)Thermocouple fuel rods.

^(d)Cell 1 was thermal cycled from 1075° to 1600°C (design) twenty-four times during capsule operation.

TABLE 4-3
RESULTS OF VISUAL EXAMINATION OF UNBONDED FISSILE PARTICLE SAMPLES IRRADIATED IN CAPSULES P13R AND P13S

Capsule	Sample Description(a)						Irradiation Conditions			Visual Examination					Remarks
	Data Retrieval Number	OPyC Thickness (μm)	OPyC Density(b) (g/cm ³)	OPyC Coating Rate (μm/min)	OPyC Anisotropy (BAF ₀)	Total Coating Thickness (μm)									
							Capsule Position	Fast Fluence (10 ²¹ n/cm ²) HTGR	Burnup (% FIMA)	No. of Particles	OPyC Coating Failure (%)	95% Confidence Interval, P(c) (%)	Pressure Vessel Failure (%)	95% Confidence Interval, P(c) (%)	
P13R	6151-00-010	44	1.80	1.1	1.042	200	3-1,2,3	12.1	73	682	32.8	31 ≤ P ≤ 33	0.7	0.3 ≤ P ≤ 1.7	Faceted coatings
	6151-00-035	37	1.85	3.9	1.030	191	3-4,5 ^(d)	12.0	74	459	0	0 ≤ P ≤ 0.8	0	0 ≤ P ≤ 0.8	
	6151-00-045	40	1.81	3.8	1.024	192	3-7,8,9	11.9	74	749	6.8	5.1 ≤ P ≤ 8.1	6.3	4.8 ≤ P ≤ 7.7	
	6151-00-046	40	1.81	3.3	1.024	191	3-10	11.8	74	254	3.5	1.9 ≤ P ≤ 6.6	2.0	0.8 ≤ P ≤ 4.5	Faceted coatings
	6151-04-015	36	1.82	1.4	1.038	199	3-11,12	11.7	73	492	49.0	44.6 ≤ P ≤ 53.4	0	0 ≤ P ≤ 0.8	Faceted coatings
	6151-01-015	42	1.90	0.7	1.077	197	4-2,3 ^(d)	11.5	73	464	100	99.2 ≤ P ≤ 100	2.8	1.6 ≤ P ≤ 4.7	
	6151-00-025	33	1.88	1.4	1.052	188	4-4,5,6	11.3	73	819	83.0	81 ≤ P ≤ 85	0	0 ≤ P ≤ 0.4	
	6151-09-015	39	1.94	3.6	1.028	195	4-7,8,9	11.1	73	785	0	0 ≤ P ≤ 0.5	0	0 ≤ P ≤ 0.5	Interrupted SiC layer
6151-09-025	41	1.95	4.0	1.033	194	4-10,11 ^(d)	11.0	73	491	0	0 ≤ P ≤ 0.8	0	0 ≤ P ≤ 0.8		
P13S	6151-03-015	45	1.77	1.5	1.020	198	3-1,2,3	11.8	72	729	4.3	3.0 ≤ P ≤ 6.0	0.1	0.01 ≤ P ≤ 0.7	Interrupted SiC layer
	4161-01-021	32	1.82	3.4	ND ^(e)	173	3-4,5,6	11.7	73	718	3.9	2.7 ≤ P ≤ 5.6	3.3	2.2 ≤ P ≤ 4.9	Interrupted SiC layer
	6151-08-015	36	1.76	3.6	1.018	190	3-7,8,9	11.7	73	688	0	0 ≤ P ≤ 0.6	0	0 ≤ P ≤ 0.6	250-μm (Th,U)O ₂ , Th/U = 1:1
	6151-02-025	43	1.50	0.7	1.039	202	3-10,12 ^(d)	11.6	72	501	46.3	42.0 ≤ P ≤ 50.7	0.2	0.04 ≤ P ≤ 1.1	
	4163-00-011	72	1.77	4.1	ND	218	4-1,2,3	11.2	27	767	0	0 ≤ P ≤ 0.5	0	0 ≤ P ≤ 0.5	
	6155-01-020	42	1.81	3.4	1.025	194	4-4 ^(d)	11.1	11	163	76.7	69.6 ≤ P ≤ 82.5	1.8	0.6 ≤ P ≤ 3.3	500-μm (Th,U)O ₂ , Th/U = 3:1
	5466-37-9	31	1.79	6.0	ND	182	4-7,8,9	10.9	38	801	0	0 ≤ P ≤ 0.4	0	0 ≤ P ≤ 0.4	250-μm (Th,U)O ₂ , Th/U = 1:1
	6151-00-035	37	1.85	3.9	1.030	191	4-10,11,12	10.7	72	809	0	0 ≤ P ≤ 0.4	0	0 ≤ P ≤ 0.4	

(a) All fissile particle samples are TRISO UC₂ particles with a nominal kernel diameter of 200 μm, except where noted.

(b) Measured using a density gradient column.

(c) An infinite sample population with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(d) Companion sample was spilled during unloading.

(e) ND = not determined.

TABLE 4-4
RESULTS OF VISUAL EXAMINATION OF UNBONDED FERTILE PARTICLE SAMPLES IRRADIATED IN CAPSULE P13R

Sample Description(a)							Irradiation Conditions			Visual Examination			Remarks
Data Retrieval Number	Buffer Thickness (μm)	OPyC Thickness (μm)	OPyC Density(b) (g/cm ³)	OPyC Coating Rate (μm/min)	OPvC Anisotropy (BAF ₀)	Particle Density(c) (g/cm ³)	Capsule Position	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) HTGR	Burnup (%) FIMA	No. of Particles	OPyC Coating Failure (%)	95% Confidence Interval, P(d) (%)	
4252-06-018	78	77	1.82	5.8	1.029	3.54	3-1,2	12.1	5.1	840	0.1	0.01 ≤ P ≤ 0.8	
6542-01-020	81	74	1.82	2.7	1.028	3.55	3-3,4	12.1	5.0	1004	7.8	6.5 ≤ P ≤ 9.0	
6542-01-010	79	85	1.80	10.0	1.028	3.59	3-5(e)	12.0	5.0	455	0.2	0.04 ≤ P ≤ 1.2	
6542-02-020	87	72	1.91	8.5	1.036	3.59	3-7,8	11.9	4.9	1072	0	0 ≤ P ≤ 0.5	
6542-02-030	79	74	1.89	2.2	1.042	3.54	3-9,10	11.8	4.8	1148	10.2	10.0 ≤ P ≤ 10.3	
6542-09-010	84	75	1.93	5.0	1.042	3.56	3-11,12	11.7	4.8	998	0	0 ≤ P ≤ 0.5	
6592-11-015	87	74	1.83	2.6	ND(f)	3.35	4-2(e)	11.5	4.5	492	0.4	0.1 ≤ P ≤ 1.5	No seal coat
6542-18-015	84	70	1.84	6.5	ND	3.53	4-3,4	11.4	4.4	953	0	0 ≤ P ≤ 0.6	No seal coat
6542-02-020	87	72	1.91	8.5	1.028	3.59	4-5,6	11.3	4.4	1149	0	0 ≤ P ≤ 0.5	
6542-20-035	80	83	1.84	3.6	ND	3.50	4-7,8	11.2	4.3	1012	0.6	0.2 ≤ P ≤ 1.2	Faceted coatings
6542-20-025	76	81	1.82	3.5	1.047	3.57	4-9,10	11.1	4.2	1040	1.0	0.6 ≤ P ≤ 1.7	Faceted coatings
6542-25-015	79	76	1.82	10.8	ND	3.57	4-11(e)	10.9	4.1	539	1.7	0.9 ≤ P ≤ 3.1	Nonround kernels

(a) All particle samples are BISO coated ThO₂ particles with a nominal kernel diameter of 500 μm.

(b) Measured using a density gradient column.

(c) Measured using an air pycnometer.

(d) An infinite sample population with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(e) Companion sample spilled during unloading.

(f) ND = not determined.

TABLE 4-5
RESULTS OF VISUAL EXAMINATION OF UNBONDED FERTILE PARTICLE SAMPLES IRRADIATED IN CAPSULE P135

Sample Description(a)							Irradiation Conditions			Visual Examination			Remarks
Data Retrieval Number	Buffer Thickness (μm)	OPyC Thickness (μm)	OPyC Density (g/cm^3) ^(b)	OPyC Coating Rate ($\mu\text{m}/\text{min}$)	OPyC Anisotropy (BAF_0)	Particle Density ^(c) (g/cm^3)	Capsule Position	Fast Fluence ($10^{21} \text{ n}/\text{cm}^2$) $E > 0.18 \text{ MeV}$	Burnup (% FIMA)	No. of Particles	OPyC Coating Failure (%)	95% Confidence Interval, P(d) (%)	
4252-06-010	85	76	1.82	5.8	1.029	3.54	3-1,2	11.8	4.3	872	0	$0 \leq P \leq 0.6$	
6542-22-015	85	81	1.80	4.0	1.020	3.50	3-3,4	11.7	4.3	979	1.0	$0.5 \leq P \leq 1.5$	Mixed-gas OPyC layer
6542-22-025	81	80	1.81	7.6	1.028	3.53	3-5,6	11.7	4.2	964	0.3	$0.1 \leq P \leq 0.9$	Mixed-gas OPyC layer
6542-23-025	82	73	1.89	5.8		3.58	3-7,8	11.7	4.2	992	0	$0 \leq P \leq 0.5$	Mixed-gas OPyC layer
6542-24-015	86	83	1.94	6.6	1.032	3.55	3-9,10	11.6	4.1	942	0	$0 \leq P \leq 0.6$	Mixed-gas OPyC layer
6252-00-025	60	42	1.81	4.2	1.015	3.77	3-12 ^(e)	11.5	4.1	528	0.4	$0.1 \leq P \leq 1.4$	TRISO coated
6542-17-010	44	122	1.86	2.8	ND ^(f)	3.55	4-1	11.2	3.9	458	2.6	$1.5 \leq P \leq 4.5$	
4252-02-010	83	73	1.83	4.0	1.032	3.51	4-2,3	11.2	3.9	925	0	$0 \leq P \leq 0.6$	
6542-16-010	39	55	ND	3.7	ND	4.84	4-4 ^(e)	11.1	3.8	349	5.4	$3.5 \leq P \leq 8.3$	
6542-19-015	81	77	1.78	7.6	1.025	3.55	4-7	10.9	3.7	495	0.6	$0.2 \leq P \leq 1.8$	
6542-19-016	82	71	1.78	7.0	1.026	3.45	4-8	10.9	3.7	502	0.2	$0.04 \leq P \leq 1.1$	
6542-21-016	81	84	1.73	8.0	1.020	3.41	4-9	10.9	3.6	508	0	$0 \leq P \leq 0.8$	Mixed-gas OPyC layer
6542-21-015	79	82	1.73	7.8	1.020	3.46	4-10	10.8	3.6	513	0	$0 \leq P \leq 0.7$	Mixed-gas OPyC layer
6542-12-025	84	70	1.79	6.7	ND	3.58	4-11,12	10.7	3.6	1214	0.2	$0.1 \leq P \leq 0.8$	N ₂ buffer carrier gas

(a) All particle samples are BISO coated ThO₂ particles with a nominal kernel diameter of 500 μm , except where noted.

(b) Measured using a density gradient column.

(c) Measured using an air pycnometer.

(d) An infinite sample population with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(e) Companion sample spilled during unloading.

(f) ND = not determined.

TABLE 4-6
FUEL ROD DIMENSIONAL CHANGE DATA OBTAINED FROM CAPSULE P13R

Capsule Position	Fuel Rod Data Retrieval Number (7161-004-)	Irradiation Conditions		Dimensional Change							
		Volume Average Temperature (°C)	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)HTGR	Measured ^(a)				Calculated ^(b)		Difference ^(c)	
				$\Delta D/D_o$ (%)	Std. Dev.	$\Delta L/L_o$ (%)	$\Delta V/V_o$ (%)	$\Delta D/D_o$ (%)	$\Delta V/V_o$ (%)	Diameter	Volume
1A	01-5	1050	7.3	-0.87	0.08	ND ^(d)	ND	-2.53	--	-65.6	--
1B	02-6	1000	8.1	-1.54	0.34	+6.82 ^(e)	ND	-2.65	-7.95	-41.9	--
1C	03-5	1015	8.7	-1.52	0.09	-0.26	-3.30	-2.28	-6.84	-33.3	-96.2
1D	04-6	1045	9.3	-0.33	0.12	+0.93	+0.27	-1.83	-5.49	-82.0	-104.9
1E	05-6	1070	9.9	-0.60	0.23	+0.29	-0.91	-1.92	-5.76	-68.8	-84.2
2A	06-5	975	11.8	-0.27 ^(f)	0.07	-0.21	-0.75	(g)	(g)	--	--
2B	07-5	985	12.0	+1.13 ^(f)	0.60	+1.97	+4.23	(g)	(g)	--	--
2C	08-13	980	12.2	-0.48	0.16	-0.40	-1.36	(g)	(g)	--	--
2D	09-5	965	12.3	+1.61 ^(f)	0.22	+4.75	+7.97	(g)	(g)	--	--
2E	10-5	915	12.4	-0.12 ^(f)	0.13	+3.89	+3.65	(g)	(g)	--	--
5A	16-5	1230	9.4	-1.19	0.24	+2.34	-0.04	-2.73	-8.19	-56.4	-99.5
5B	17-5	1175	9.0	-1.33	0.44	+0.89	-1.77	-2.62	-7.86	-49.2	-77.5
5C	18-6	1180	8.5	-1.53	0.15	-0.15	-3.21	-2.79	-8.37	-45.2	-61.6
5D	19-7	1170	8.2	-1.33	0.13	+2.62	-0.04	-2.90	-8.70	-54.1	-99.5
5E	20-6	1215	7.7	-1.35	0.16	+0.68	-2.02	-3.87	-11.61	-65.1	-82.6
6A	21-5	1035	5.4	-2.01	0.22	ND	ND	-2.85	-8.55	-29.5	--
6B	22-5	1020	5.0	-1.63	0.13	-0.37	-3.63	-2.40	-7.20	-32.1	-49.6
6C	23-5	975	4.5	-1.77	0.12	-0.67	-4.21	-2.71	-8.13	-34.7	-48.2
6D	24-5	985	4.1	-1.44	0.93	-1.09	-3.97	-2.25	-6.75	-36.0	-41.2
6E	25-5	970	3.6	-1.63	0.17	-0.10	-3.36	-2.47	-7.41	-34.0	-54.7

(a) Measurements made with dial gauge micrometer.

(b) Dimensional changes calculated using SHRINK computer code.

(c) Percent difference between calculated and measured dimensional change:
$$\frac{\Delta x/x_{o\text{ meas}} - \Delta x/x_{o\text{ calc}}}{\Delta x/x_{o\text{ calc}}} \times 100 = \text{difference.}$$

(d) ND = not determined.

(e) Length measurement appears to be erroneous datum point.

(f) Fuel rods were constrained by graphite crucibles at end of the irradiation; therefore, dimensional change of rods may not be representative.

(g) Fast neutron exposure received by these rods exceeds limits of pyrocarbon and graphite shim dimensional change data; therefore, fuel rod dimensional changes cannot be predicted using SHRINK code.

TABLE 4-7
FUEL ROD DIMENSIONAL CHANGE DATA OBTAINED FROM CAPSULE P13S

Capsule Position	Fuel Rod Data Retrieval Number (7161-004-)	Irradiation Conditions		Dimensional Change							
		Volume Average Temperature (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV)HTGR	Measured(a)				Calculated(b)		Difference (c)	
				$\Delta D/D_0$ (%)	Std. Dev.	$\Delta L/L_0$ (%)	$\Delta V/V_0$ (%)	$\Delta D/D_0$ (%)	$\Delta V/V_0$ (%)	Diameter	Volume
1A	01-7	965	7.0	-1.18	0.23	-0.08	-2.44	-2.58	-7.74	-54.3	-68.5
1B	02-5	925	7.7	-1.26	0.26	-0.47	-2.99	-2.50	-7.50	-49.6	-60.1
1C	03-6	950	8.5	-1.04	0.42	-1.57	-3.65	-2.31	-6.93	-55.0	-47.3
1D	04-5	965	9.1	-0.20	0.18	+1.55	+1.55	-2.00	-6.00	-90.0	-80.8
1E	05-5	980	9.6	-0.25	0.43	+1.19	+0.31	-1.89	-5.67	-86.8	-94.5
2A	11-6	970	11.5	-0.27 ^(d)	0.21	+1.09	+0.55	(e)	(e)	--	--
2B	12-5	945	11.7	+0.48 ^(d)	0.23	+2.06	+3.02	(e)	(e)	--	--
2C	13-6	915	11.9	+0.09 ^(d)	0.19	+1.38	+1.56	(e)	(e)	--	--
2D	14-5	920	12.0	+0.28 ^(d)	0.24	+2.28	+2.84	(e)	(e)	--	--
2E	15-5	910	12.1	+0.13 ^(d)	0.21	+2.00	+2.26	(e)	(e)	--	--
5A	16-6	1330	9.2	-1.64	0.26	-0.45	-3.73	-2.73	-8.19	-39.9	-54.5
5B	17-6	1290	8.7	-1.17	0.33	-0.32	-2.66	-2.86	-8.58	-59.1	-69.0
5C	18-5	1285	8.4	-1.58	0.10	ND ^(f)	--	-2.77	--	-43.0	--
5D	19-5	1280	7.9	-1.36	0.27	ND	--	-2.98	--	-54.4	--
5E	20-5	1295	7.5	-1.63	0.28	-0.60	-3.86	-2.95	-8.85	-44.7	-56.4
6A	26-5	1050	5.3	-2.03	0.25	+0.32	-3.74	-2.84	-8.52	-28.5	-56.1
6B	27-5	1010	4.8	-1.52	0.23	ND	--	-2.33	--	-34.8	--
6C	28-13	1020	4.4	-2.34	0.17	-1.97	-6.65	-2.72	-8.16	-14.0	-18.5
6D	29-5	985	3.9	-2.05	0.22	-0.84	-4.94	-2.63	-7.89	-22.1	-37.4
6E	30-5	965	3.5	-1.55	0.65	+0.75	-2.35	-2.44	-7.32	-36.5	-67.9

(a) Measurements made with dial gauge micrometer.

(b) Dimensional changes calculated using SHRINK computer code.

(c) Percent difference between calculated and measured dimensional change:
$$\frac{\Delta x/x_{o_{meas}} - \Delta x/x_{o_{calc}}}{\Delta x/x_{o_{calc}}} \times 100 = \text{difference.}$$

(d) Fuel rods were constrained by graphite crucibles at end of irradiation; therefore, dimensional change of rods may not be representative.

(e) Fast neutron exposure received by these rods exceeds limits of pyrocarbon and graphite shim dimensional change data; therefore, fuel rod dimensional changes cannot be predicted using SHRINK code.

(f) ND = not determined.

TABLE 4-8
MEASURED DIAMETRAL DIMENSIONAL CHANGES OF FUEL ROD GRAPHITE CRUCIBLES

Capsule	Cell	Position	Dimensional Change, $\Delta L/L_0$ (%)		Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Nominal Temperature (°C)
			⊥ to Split	∥ to Split		
P13R	1	Top	-0.8	-1.2	9.9	780
		Center	-0.4	-1.1	8.7	
		Bottom	-0.5	-0.8	7.3	
P13R	2	Top	-0.2	-1.2	12.4	730
		Center	+0.6	-0.8	12.2	
		Bottom	-0.5	-0.7	11.8	
P13R	5	Top	-0.9	-1.2	7.7	960
		Center	-0.9	-1.5	8.6	
		Bottom	-1.1	-1.2	9.4	
P13R	6	Top	-0.2	-0.2	3.6	765
		Center	-0.4	-0.4	4.5	
		Bottom	-0.6	-0.5	5.4	
P13S	1	Top	-0.6	-0.8	9.6	840
		Center	-0.4	-0.7	8.4	
		Bottom	-0.5	-0.6	7.0	
P13S	2	Top	-0.7	-1.2	12.1	750
		Center	-0.3	-1.3	11.9	
		Bottom	-0.8	-1.3	11.5	
P13S	5	Top	-0.8	-0.7	7.5	1150
		Center	-0.6	-0.6	8.4	
		Bottom	-0.6	-1.1	9.2	
P13S	6	Top	-0.2	-0.2	3.5	780
		Center	-0.5	-0.6	4.4	
		Bottom	-0.6	-0.9	5.3	

TABLE 4-9
POSTIRRADIATION RADIAL GAP BETWEEN GRAPHITE CRUCIBLES AND
CAPSULE P13R FUEL RODS

Fuel Rod Data Retrieval No. (7161-004-)	Capsule Position	Postirradiation Dimensions (cm)		
		Crucible I.D.	Fuel Rod O.D.	Radial Gap
01-5	1A	1.258	1.241	0.009
02-6	1B	1.258	1.236	0.011
03-5	1C	1.254	1.232	0.011
04-6	1D	1.254	1.250	0.002
05-6	1E	1.254	1.242	0.006
06-5	2A	1.259	1.250	0.005
07-5	2B	1.259	1.263	0
08-13	2C	1.258	1.241	0.009
09-5	2D	1.256	1.270	0
10-5	2E	1.250	1.252	0
16-5	5A	1.253	1.236	0.009
17-5	5B	1.251	1.238	0.007
18-6	5C	1.249	1.235	0.007
19-7	5D	1.251	1.235	0.008
20-6	5E	1.253	1.234	0.010
21-5	6A	1.262	1.225	0.019
22-5	6B	1.263	1.232	0.016
23-5	6C	1.263	1.230	0.017
24-5	6D	1.265	1.224	0.020
25-5	6E	1.266	1.230	0.018

TABLE 4-10
POSTIRRADIATION RADIAL GAP BETWEEN GRAPHITE CRUCIBLES AND
CAPSULE P13S FUEL RODS

Fuel Rod Data Retrieval No. (7161-004-)	Capsule Position	Postirradiation Dimensions (cm)		
		Crucible I.D.	Fuel Rod O.D.	Radial Gap
01-7	1A	1.261	1.238	0.012
02-5	1B	1.261	1.238	0.012
03-6	1C	1.260	1.237	0.012
04-5	1D	1.260	1.253	0.004
05-5	1E	1.259	1.246	0.007
11-6	2A	1.252	1.247	0.003
12-5	2B	1.252	1.258	0
13-6	2C	1.252	1.254	0
14-5	2D	1.252	1.256	0
15-5	2E	1.253	1.253	0
16-6	5A	1.254	1.230	0.012
17-6	5B	1.258	1.237	0.011
18-5	5C	1.261	1.233	0.014
19-5	5D	1.261	1.234	0.014
20-5	5E	1.259	1.232	0.014
26-5	6A	1.257	1.227	0.015
27-5	6B	1.259	1.233	0.013
28-13	6C	1.261	1.212	0.025
29-5	6D	1.264	1.225	0.020
30-5	6E	1.266	1.230	0.018

TABLE 4-11
TRIGA FISSION GAS RELEASE DATA FOR CAPSULE P13R FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	EOL Fissile Loading (a) (mg U-235)	TRIGA Fission Gas Release Data (R/B Kr-85m at 1100°C)	
			BOL	EOL
01-5	1A	45.1	2×10^{-6}	2.4×10^{-4}
02-6	1B	44.7	1×10^{-6}	3.1×10^{-5}
03-5	1C	43.8	1×10^{-6}	3.3×10^{-6}
04-6	1D	43.4	3×10^{-6}	1.1×10^{-4}
05-6	1E	41.3	4×10^{-6}	3.0×10^{-6}
06-5	2A	43.2	3×10^{-6}	2.5×10^{-4}
07-5	2B	42.6	3×10^{-6}	1.6×10^{-5}
08-13	2C	41.4	4×10^{-6}	1.6×10^{-5}
09-5	2D	39.5	3×10^{-6}	3.9×10^{-4}
10-5	2E	37.7	6×10^{-6}	4.1×10^{-5}
16-5	5A	50.5	4×10^{-6}	4.0×10^{-5}
17-5	5B	49.2	2×10^{-6}	$3.3 \times 10^{-5(b)}$
18-6	5C	52.5	3×10^{-6}	4.7×10^{-7}
19-7	5D	53.2	2×10^{-6}	1.3×10^{-5}
20-6	5E	57.1	3×10^{-6}	6.3×10^{-6}
21-5	6A	39.1	4×10^{-6}	3.1×10^{-6}
22-5	6B	40.8	2×10^{-6}	3.9×10^{-6}
23-5	6C	41.4	2×10^{-6}	5.0×10^{-6}
24-5	6D	44.4	6×10^{-6}	1.8×10^{-5}
25-5	6E	45.8	3×10^{-6}	7.7×10^{-6}

(a) EOL fissile loading is the weight of U-235 which would produce the equivalent Kr-85m yield of all fissionable material (U-233 and U-235) calculated to be in the fuel rod.

(b) Value extrapolated from 1300°C measurement using Eq. 4-13.

TABLE 4-12
TRIGA FISSION GAS RELEASE DATA FOR CAPSULE P13S FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	EOL Fissile Loading ^(a) (mg U-235)	TRIGA Fission Gas Release Data (R/B Kr-85m at 1100°C)	
			BOL	EOL
01-7	1A	44.8	2×10^{-6}	1.1×10^{-3}
02-5	1B	44.3	1×10^{-6}	5.9×10^{-5}
03-6	1C	43.2	1×10^{-6}	1.1×10^{-4}
04-5	1D	42.7	2×10^{-6}	1.2×10^{-3}
05-5	1E	40.6	2×10^{-6}	2.2×10^{-4}
11-6	2A	42.2	2×10^{-7}	1.5×10^{-4}
12-5	2B	41.6	2×10^{-6}	3.4×10^{-7}
13-6	2C	40.4	3×10^{-6}	3.5×10^{-5}
14-5	2D	38.6	2×10^{-6}	1.2×10^{-5}
15-5	2E	36.8	5×10^{-6}	1.0×10^{-4}
16-6	5A	49.9	4×10^{-6}	1.8×10^{-4}
17-6	5B	48.7	1×10^{-6}	$1.0 \times 10^{-4(b)}$
18-5	5C	52.0	2×10^{-6}	2.6×10^{-5}
19-5	5D	52.9	3×10^{-6}	7.4×10^{-5}
20-5	5E	56.8	3×10^{-6}	1.4×10^{-4}
26-5	6A	39.1	3×10^{-6}	2.1×10^{-6}
27-5	6B	40.7	1×10^{-6}	$<10^{-7}$
28-13	6C	41.4	4×10^{-6}	8.4×10^{-7}
29-5	6D	44.5	1×10^{-6}	3.6×10^{-7}
30-5	6E	46.4	2×10^{-6}	1.3×10^{-6}

(a) EOL fissile loading is the weight of U-235 which would produce the equivalent Kr-85m yield of all fissionable material (U-233 and U-235) calculated to be in the fuel rod.

(b) Value extrapolated from 1500°C measurement using Eq. 4-13.

TABLE 4-13
TRIGA FISSION GAS RELEASE DATA FOR UNBONDED FISSIONABLE PARTICLE SAMPLES
TESTED IN CAPSULES P13R AND P13S

Batch No.	Capsule	Capsule Position	EOL Fissile Loading ^(a) (mg U-235)	TRIGA Fission Gas Release Data (R/B Kr-85m at 1100°C)	
				BOL	EOL
6151-00-010	P13R	3-1,2,3	1.09	4×10^{-7}	1.4×10^{-3}
6151-00-035	P13R	3-4,5	0.71	3×10^{-7}	1.2×10^{-4}
6151-00-035	P13S	4-10,11,12	2.58	3×10^{-7}	5.9×10^{-5}
6151-00-045	P13R	3-7,8,9	1.14	2×10^{-7}	6.7×10^{-3}
6151-00-046	P13R	3-10	0.39	2×10^{-7}	4.9×10^{-3}
6151-04-015	P13R	3-11,12	0.88	8×10^{-7}	1.2×10^{-4}
6151-01-015	P13R	4-2,3	0.93	1×10^{-8}	5.0×10^{-4}
6151-00-025	P13R	4-4,5,6	1.55	6×10^{-8}	1.3×10^{-3}
6151-09-015	P13R	4-7,8,9	1.68	9×10^{-8}	3.5×10^{-4}
6151-09-025	P13R	4-10,11	1.19	2×10^{-7}	2.4×10^{-5}
6151-03-015	P13S	3-1,2,3	1.52	2×10^{-6}	6.7×10^{-5}
4161-01-021	P13S	3-4,5,6	1.58	5×10^{-6}	9.4×10^{-3}
6151-08-015	P13S	3-7,8,9	1.64	1×10^{-7}	1.4×10^{-5}
6151-02-025	P13S	3-10,12	1.23	2×10^{-6}	1.5×10^{-3}
4163-00-011	P13S	4-1,2,3	5.11	3×10^{-9}	5.8×10^{-6}
6155-01-020	P13S	4-4	5.23	6×10^{-7}	5.3×10^{-4}
5466-37	P13S	4-7,8,9	3.99	1×10^{-6}	3.0×10^{-6}

^(a) EOL fissile loading is the weight of U-235 which would produce the equivalent Kr-85m yield of all fissionable material calculated to be in the sample.

TABLE 4-14
TRIGA FISSION GAS RELEASE DATA FOR UNBONDED FERTILE PARTICLE SAMPLES
TESTED IN CAPSULES P13R AND P13S

Batch No.	Capsule	Capsule Position	EOL Fissile Loading ^(a) (mg U-235)	TRIGA Fission Gas Release Data, EOL (R/B Kr-85m at 1100°)
4252-06-018	P13R	3-1,2	31.97	1.4×10^{-3}
6542-01-020	P13R	3-3,4	31.85	1.7×10^{-4}
6542-01-010	P13R	3-5	32.31	1.2×10^{-3}
6542-25-015	P13R	4-11	32.31	1.2×10^{-3}
6542-02-020	P13R	3-7,8	31.76	4.4×10^{-5}
6542-02-020	P13R	4-5,6	33.20	3.3×10^{-5}
6542-02-030	P13R	3-9,10	31.67	2.9×10^{-3}
6542-09-010	P13R	3-11,12	34.47	2.4×10^{-5}
6542-11-015	P13R	4-2	46.16	2.4×10^{-3}
6542-19-015	P13S	4-7	46.16	2.4×10^{-3}
6542-19-016	P13S	4-8	46.16	2.4×10^{-3}
6542-18-015	P13R	4-3,4	31.19	2.2×10^{-6}
6542-20-035	P13R	4-7,8	32.95	6.9×10^{-4}
6542-20-025	P13R	4-9,10	32.98	3.7×10^{-4}
4252-06-010	P13S	3-1,2	30.77	6.2×10^{-5}
6542-22-015	P13S	3-3,4	30.79	2.7×10^{-4}
6542-22-025	P13S	3-5,6	30.73	2.4×10^{-4}
6542-23-025	P13S	3-7,8	30.62	2.8×10^{-6}
6542-24-015	P13S	3-9,10	30.59	3.9×10^{-6}
6252-00-025	P13S	3-12	18.00	3.8×10^{-5}
6542-17-010	P13S	4-1	24.44	4.7×10^{-3}
6542-16-010	P13S	4-4	24.44	4.7×10^{-3}
4252-02-010	P13S	4-2,3	27.86	1.1×10^{-3}
6542-21-016	P13S	4-9	31.30	2.1×10^{-3}
6542-21-015	P13S	4-10	31.30	2.1×10^{-3}
6542-12-025	P13S	4-11,12	35.50	2.0×10^{-4}

(a) EOL fissile loading is the weight of U-235 which would produce the equivalent Kr-85m yield of all fissionable material calculated to be in the sample.

TABLE 4-15
GASEOUS FISSION PRODUCT RELEASE (R/B Kr-85m) RESULTING FROM IN-SERVICE FUEL PARTICLE FAILURE IN CAPSULE P13R FUEL RODS

Fuel Rod Data Retrieval No. (7161-004-)	Capsule Position	Thorium Cont. (a) (g Th/g Th)	R/B _{BOL}	Fraction of Fissions at EOL (b)		R/B _c (Th) (c)	R/B _c (U)	R/B _{EOL}	R/B _{failed fuel} (d)	In-Service Particle Failure (e) (%)	Maximum Fertile Particle Failure (e,f) (%)
				Fissile	Fertile						
01-5	1A	8 x 10 ⁻⁶	2 x 10 ⁻⁶	0.108	0.892	2.1 x 10 ⁻⁶	2.2 x 10 ⁻⁷	2.4 x 10 ⁻⁴	2.4 x 10 ⁻⁴	4.8	5.4
02-6	1B	8 x 10 ⁻⁶	1 x 10 ⁻⁶	0.092	0.908	2.2 x 10 ⁻⁶	9.2 x 10 ⁻⁸	3.1 x 10 ⁻⁵	2.9 x 10 ⁻⁵	0.6	0.6
03-5	1C	8 x 10 ⁻⁶	1 x 10 ⁻⁶	0.081	0.919	2.2 x 10 ⁻⁶	8.1 x 10 ⁻⁸	3.3 x 10 ⁻⁶	1.0 x 10 ⁻⁶	0	0
04-6	1D	9 x 10 ⁻⁶	3 x 10 ⁻⁶	0.074	0.926	2.5 x 10 ⁻⁶	2.2 x 10 ⁻⁷	1.1 x 10 ⁻⁴	1.1 x 10 ⁻⁴	2.2	2.4
05-6	1E	9 x 10 ⁻⁶	4 x 10 ⁻⁶	0.094	0.906	2.4 x 10 ⁻⁶	3.8 x 10 ⁻⁷	3.0 x 10 ⁻⁶	2.2 x 10 ⁻⁷	0	0
06-5	2A	9 x 10 ⁻⁶	3 x 10 ⁻⁶	0.062	0.938	2.5 x 10 ⁻⁶	1.9 x 10 ⁻⁷	2.5 x 10 ⁻⁴	2.5 x 10 ⁻⁴	5.0	5.3
07-5	2B	9 x 10 ⁻⁶	3 x 10 ⁻⁶	0.061	0.939	2.5 x 10 ⁻⁶	1.8 x 10 ⁻⁷	1.6 x 10 ⁻⁵	1.3 x 10 ⁻⁵	0.3	0.3
08-13	2C	9 x 10 ⁻⁶	4 x 10 ⁻⁶	0.061	0.939	2.5 x 10 ⁻⁶	2.4 x 10 ⁻⁷	1.6 x 10 ⁻⁵	1.3 x 10 ⁻⁵	0.3	0.3
09-5	2D	1 x 10 ⁻⁵	3 x 10 ⁻⁶	0.062	0.938	2.8 x 10 ⁻⁶	1.9 x 10 ⁻⁷	3.9 x 10 ⁻⁴	3.9 x 10 ⁻⁴	7.8	8.3
10-5	2E	1 x 10 ⁻⁵	6 x 10 ⁻⁶	0.075	0.925	2.8 x 10 ⁻⁶	4.5 x 10 ⁻⁷	4.1 x 10 ⁻⁵	3.8 x 10 ⁻⁵	0.8	0.8
16-5	5A	6 x 10 ⁻⁶	4 x 10 ⁻⁶	0.117	0.883	1.1 x 10 ⁻⁶	4.7 x 10 ⁻⁷	4.0 x 10 ⁻⁵	3.8 x 10 ⁻⁵	0.8	0.9
17-5	5B	7 x 10 ⁻⁶	2 x 10 ⁻⁶	0.109	0.891	2.1 x 10 ⁻⁶	2.2 x 10 ⁻⁷	3.3 x 10 ⁻⁵	3.1 x 10 ⁻⁵	0.6	0.7
18-6	5C	7 x 10 ⁻⁶	3 x 10 ⁻⁶	0.118	0.882	1.9 x 10 ⁻⁶	3.5 x 10 ⁻⁷	4.7 x 10 ⁻⁷	0	0	0
19-7	5D	7 x 10 ⁻⁶	2 x 10 ⁻⁶	0.133	0.867	1.8 x 10 ⁻⁶	2.7 x 10 ⁻⁷	1.3 x 10 ⁻⁵	1.1 x 10 ⁻⁵	0.2	0.2
20-6	5E	6 x 10 ⁻⁶	3 x 10 ⁻⁶	0.157	0.843	1.5 x 10 ⁻⁶	4.7 x 10 ⁻⁷	6.3 x 10 ⁻⁶	4.3 x 10 ⁻⁶	0.09	0.1
21-5	6A	9 x 10 ⁻⁶	4 x 10 ⁻⁶	0.345	0.655	1.8 x 10 ⁻⁶	1.4 x 10 ⁻⁶	3.1 x 10 ⁻⁶	0	0	0
22-5	6B	8 x 10 ⁻⁶	2 x 10 ⁻⁶	0.350	0.650	1.6 x 10 ⁻⁶	7.0 x 10 ⁻⁷	3.9 x 10 ⁻⁶	1.6 x 10 ⁻⁶	0	0
23-5	6C	9 x 10 ⁻⁶	2 x 10 ⁻⁶	0.418	0.582	1.6 x 10 ⁻⁶	8.4 x 10 ⁻⁷	5.0 x 10 ⁻⁶	2.6 x 10 ⁻⁶	0.05	0.09
24-5	6D	9 x 10 ⁻⁶	6 x 10 ⁻⁶	0.464	0.536	1.4 x 10 ⁻⁶	7.8 x 10 ⁻⁶	1.8 x 10 ⁻⁵	1.4 x 10 ⁻⁵	0.3	0.5
25-5	6E	1 x 10 ⁻⁵	3 x 10 ⁻⁶	0.619	0.381	1.1 x 10 ⁻⁶	1.8 x 10 ⁻⁶	7.7 x 10 ⁻⁶	4.8 x 10 ⁻⁶	0.1	0.3

(a) Determined by hydrolysis test on companion unirradiated fuel rods (Ref. 4-2).

(b) Normalized to Kr-85m thermal fission yields.

(c) Calculated assuming a fractional release for contamination of 0.30 (Ref. 4-9).

(d) $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - [R/B_{\text{c}}(\text{Th}) + R/B_{\text{c}}(\text{U})]$.

(e) Calculated assuming a fractional release of 5×10^{-3} for failed fuel in a constrained geometry (Ref. 4-9).

(f) Calculated assuming all fission gas released came from failed fertile particles.

TABLE 4-16
GASEOUS FISSION PRODUCT RELEASE (R/B Kr-85m) RESULTING FROM IN-SERVICE FUEL PARTICLE FAILURE IN CAPSULE P13S FUEL RODS

Fuel Rod Data Retrieval No. (7161-004-)	Capsule Position	Thorium Cont. (a) (g Th/g Th)	R/B _{BOL}	Fraction of Fissions at EOL ^(b)		R/B _c (Th) ^(c)	R/B _c (U)	R/B _{EOL}	R/B ^(d) failed fuel	In-Service Particle Failure ^(e) (%)	Maximum Fertile Particle Failure ^(e,f) (%)
				Fissile	Fertile						
01-7	1A	8 x 10 ⁻⁶	2 x 10 ⁻⁶	0.121	0.879	2.1 x 10 ⁻⁶	2.4 x 10 ⁻⁷	1.1 x 10 ⁻³	1.1 x 10 ⁻³	22.0	24.9
02-5	1B	8 x 10 ⁻⁶	1 x 10 ⁻⁶	0.106	0.894	2.1 x 10 ⁻⁶	1.1 x 10 ⁻⁷	5.9 x 10 ⁻⁵	5.7 x 10 ⁻⁵	1.1	1.3
03-6	1C	8 x 10 ⁻⁶	1 x 10 ⁻⁶	0.094	0.906	2.2 x 10 ⁻⁶	9.4 x 10 ⁻⁸	1.1 x 10 ⁻⁴	1.1 x 10 ⁻⁴	2.2	2.4
04-5	1D	9 x 10 ⁻⁶	2 x 10 ⁻⁶	0.086	0.914	2.5 x 10 ⁻⁶	1.7 x 10 ⁻⁷	1.2 x 10 ⁻³	1.2 x 10 ⁻³	24.0	26.3
06-5	1E	9 x 10 ⁻⁶	2 x 10 ⁻⁶	0.111	0.889	2.4 x 10 ⁻⁶	2.2 x 10 ⁻⁷	2.2 x 10 ⁻⁴	2.2 x 10 ⁻⁴	4.4	4.9
11-6	2A	9 x 10 ⁻⁶	2 x 10 ⁻⁷	0.069	0.931	2.5 x 10 ⁻⁶	1.4 x 10 ⁻⁸	1.5 x 10 ⁻⁴	1.5 x 10 ⁻⁴	3.0	3.2
12-5	2B	9 x 10 ⁻⁶	2 x 10 ⁻⁶	0.068	0.932	2.5 x 10 ⁻⁶	1.4 x 10 ⁻⁷	3.4 x 10 ⁻⁷	0	0	0
13-6	2C	9 x 10 ⁻⁶	3 x 10 ⁻⁶	0.068	0.932	2.5 x 10 ⁻⁶	2.0 x 10 ⁻⁷	3.5 x 10 ⁻⁵	3.2 x 10 ⁻⁵	0.6	0.7
14-5	2D	1 x 10 ⁻⁵	2 x 10 ⁻⁶	0.068	0.932	2.8 x 10 ⁻⁶	1.4 x 10 ⁻⁷	1.2 x 10 ⁻⁵	9.1 x 10 ⁻⁶	0.2	0.2
15-5	2E	1 x 10 ⁻⁵	5 x 10 ⁻⁶	0.069	0.931	2.8 x 10 ⁻⁶	3.4 x 10 ⁻⁷	1.0 x 10 ⁻⁴	9.7 x 10 ⁻⁵	1.9	2.1
16-6	5A	6 x 10 ⁻⁶	4 x 10 ⁻⁶	0.130	0.870	1.6 x 10 ⁻⁶	5.2 x 10 ⁻⁷	1.8 x 10 ⁻⁴	1.8 x 10 ⁻⁴	3.6	4.1
17-6	5B	7 x 10 ⁻⁶	1 x 10 ⁻⁶	0.121	0.879	1.8 x 10 ⁻⁶	1.2 x 10 ⁻⁷	1.0 x 10 ⁻⁴	9.8 x 10 ⁻⁵	2.0	2.2
18-5	5C	7 x 10 ⁻⁶	2 x 10 ⁻⁶	0.129	0.871	1.8 x 10 ⁻⁶	2.6 x 10 ⁻⁷	2.6 x 10 ⁻⁵	2.4 x 10 ⁻⁵	0.5	0.6
19-5	5D	7 x 10 ⁻⁶	3 x 10 ⁻⁶	0.144	0.856	1.8 x 10 ⁻⁶	4.3 x 10 ⁻⁷	7.4 x 10 ⁻⁵	7.1 x 10 ⁻⁵	1.4	1.7
20-5	5E	6 x 10 ⁻⁶	3 x 10 ⁻⁶	0.167	0.833	1.5 x 10 ⁻⁶	5.0 x 10 ⁻⁷	1.4 x 10 ⁻⁴	1.4 x 10 ⁻⁴	2.8	3.3
26-5	6A	9 x 10 ⁻⁶	3 x 10 ⁻⁶	0.350	0.650	1.8 x 10 ⁻⁶	1.0 x 10 ⁻⁶	2.1 x 10 ⁻⁶	0	0	0
27-5	6B	8 x 10 ⁻⁶	1 x 10 ⁻⁶	0.390	0.610	1.5 x 10 ⁻⁶	3.9 x 10 ⁻⁷	<10 ⁻⁷	0	0	0
28-13	6C	8 x 10 ⁻⁶	4 x 10 ⁻⁶	0.425	0.575	1.4 x 10 ⁻⁶	1.7 x 10 ⁻⁶	8.4 x 10 ⁻⁷	0	0	0
29-5	6D	9 x 10 ⁻⁶	1 x 10 ⁻⁶	0.478	0.522	1.4 x 10 ⁻⁶	4.8 x 10 ⁻⁷	3.6 x 10 ⁻⁷	0	0	0
30-5	6E	1 x 10 ⁻⁵	2 x 10 ⁻⁶	0.637	0.363	1.1 x 10 ⁻⁶	1.3 x 10 ⁻⁶	1.3 x 10 ⁻⁶	0	0	0

(a) Determined by hydrolysis test on companion unirradiated fuel rods (Ref. 4-2).

(b) Normalized to Kr-85m thermal fission yields.

(c) Calculated assuming a fractional release for contamination of 0.30 (Ref. 4-9).

(d) $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - [R/B_{\text{c}}(\text{Th}) + R/B_{\text{c}}(\text{U})]$.

(e) Calculated assuming a fractional release of 5×10^{-3} for failed fuel in a constrained geometry (Ref. 4-9).

(f) Calculated assuming all fission gas released came from failed fertile particles.

TABLE 4-17
UNBONDED FISSION PARTICLE FAILURE LEVELS DETERMINED BY FISSION GAS RELEASE MEASUREMENTS FOR
CAPSULES P13R AND P13S

Batch No.	Particle Type(a)	Capsule	Capsule Position	TRIGA Fission Gas Release		R/B Failed Fuel(b)	In-Service Particle Failure(c) (%)
				BOL	EOL		
6151-00-010	UC ₂	P13R	3-1,2,3	4×10^{-7}	1.4×10^{-3}	1.4×10^{-3}	4.0 ^(d)
6151-00-035	UC ₂	P13R	3-4,5	3×10^{-7}	1.2×10^{-4}	1.2×10^{-4}	0.6
6151-00-035	UC ₂	P13S	4-10,11,12	3×10^{-7}	5.9×10^{-5}	5.9×10^{-5}	0.3
6151-00-045	UC ₂	P13R	3-7,8,9	2×10^{-7}	6.7×10^{-3}	6.7×10^{-3}	19.1 ^(d)
6151-00-046	UC ₂	P13R	3-10	2×10^{-7}	4.9×10^{-3}	4.9×10^{-3}	14.0 ^(d)
6151-04-015	UC ₂	P13R	3-11,12	8×10^{-7}	1.2×10^{-4}	1.2×10^{-4}	0.6
6151-01-015	UC ₂	P13R	4-2,3	1×10^{-8}	5.0×10^{-4}	5.0×10^{-4}	1.4
6151-00-025	UC ₂	P13R	4-4,5,6	6×10^{-8}	1.3×10^{-3}	1.3×10^{-3}	6.5
6151-09-015	UC ₂	P13R	4-7,8,9	9×10^{-8}	3.5×10^{-4}	3.5×10^{-4}	1.8
6151-09-025	UC ₂	P13R	4-10,11	2×10^{-7}	2.4×10^{-5}	2.4×10^{-5}	0.12
6151-03-015	UC ₂	P13S	3-1,2,3	2×10^{-6}	6.7×10^{-5}	6.5×10^{-5}	0.2 ^(d)
4161-01-021	UC ₂	P13S	3-4,5,6	5×10^{-6}	9.4×10^{-3}	9.4×10^{-3}	26.9 ^(d)
6151-08-015	UC ₂	P13S	3-7,8,9	1×10^{-7}	1.4×10^{-5}	1.4×10^{-5}	0
6151-02-025	UC ₂	P13S	3-10,12	2×10^{-6}	1.5×10^{-3}	1.5×10^{-3}	4.3 ^(d)
4163-00-011	(Th,U)O ₂	P13S	4-1,2,3	3×10^{-9}	5.8×10^{-6}	0	0
6155-01-020	(Th,U)O ₂	P13S	4-4	6×10^{-7}	5.3×10^{-4}	5.3×10^{-4}	2.6
5466-37	(Th,U)O ₂	P13S	4-7,8,9	1×10^{-6}	3.0×10^{-6}	0	0

(a) All fissile particles were TRISO coated.

(b) For UC₂ particles $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - R/B_{\text{BOL}}$; for (Th,U)O₂ particles $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - [R/B_{\text{c(Th)}} - R/B_{\text{c(U)}}]$.

(c) Calculated assuming a fractional release of 2×10^{-2} for unconstrained failed particles, except where noted.

(d) Calculated assuming a fractional release of 3.5×10^{-2} for bare fuel kernels.

TABLE 4-18
UNBONDED FERTILE PARTICLE FAILURE LEVELS DETERMINED BY FISSION GAS RELEASE MEASUREMENTS
FOR CAPSULES P13R AND P13S

Batch No.	Capsule	Capsule Position	Thorium Cont. (a) (g Th/g Th)	R/B _c (Th) (b)	R/BEOL	R/B Failed Fuel (c)	In-Service Particle Failure (d) (%)
4252-06-018	P13R	3-1,2	7×10^{-4}	2.1×10^{-4}	1.4×10^{-3}	1.2×10^{-3}	6.0
6542-01-020	P13R	3-3,4	0	0	1.7×10^{-4}	1.7×10^{-4}	0.9
6542-01-010	P13R	3-5	6×10^{-6}	1.1×10^{-6}	1.2×10^{-3}	1.2×10^{-3}	6.0
6542-25-015	P13R	4-11	1×10^{-6}	1.1×10^{-6}	1.2×10^{-3}	1.2×10^{-3}	6.0
6542-02-020	P13R	3-7,8	0	0	4.4×10^{-5}	4.4×10^{-5}	0.2
6542-02-020	P13R	4-5,6	0	0	3.3×10^{-5}	3.3×10^{-5}	0.17
6542-02-030	P13R	3-9,10	8×10^{-5}	2.4×10^{-5}	2.9×10^{-3}	2.9×10^{-3}	14.5
6542-09-010	P13R	3-11,12	1×10^{-4}	3.0×10^{-5}	2.4×10^{-5}	0	0
6542-11-015	P13R	4-2	6×10^{-7}	8.6×10^{-7}	2.4×10^{-3}	2.4×10^{-3}	12.0
6542-19-015	P13S	4-7	4×10^{-6}	8.6×10^{-7}	2.4×10^{-3}	2.4×10^{-3}	12.0
6542-19-016	P13S	4-8	4×10^{-6}	8.6×10^{-7}	2.4×10^{-3}	2.4×10^{-3}	12.0
6542-18-015	P13R	4-3,4	3×10^{-6}	9.0×10^{-7}	2.2×10^{-6}	1.3×10^{-6}	0
6542-20-035	P13R	4-7,8	1×10^{-6}	3.0×10^{-7}	6.9×10^{-4}	6.9×10^{-4}	3.5
6542-20-025	P13R	4-9,10	3×10^{-6}	9.0×10^{-7}	3.7×10^{-4}	3.7×10^{-4}	1.9
4252-06-010	P13S	3-1,2	7×10^{-4}	2.1×10^{-4}	6.2×10^{-5}	0	0
6542-22-015	P13S	3-3,4	1×10^{-6}	3.0×10^{-7}	2.7×10^{-4}	2.7×10^{-4}	1.4
6542-22-025	P13S	3-5,6	3×10^{-6}	9.0×10^{-7}	2.4×10^{-4}	2.4×10^{-4}	1.2
6542-23-025	P13S	3-7,8	4×10^{-6}	1.2×10^{-6}	2.8×10^{-6}	1.6×10^{-6}	0
6542-24-015	P13S	3-9,10	1×10^{-6}	3.0×10^{-7}	3.9×10^{-6}	3.6×10^{-6}	0
6252-00-025 ^(e)	P13S	3-12	0	0	3.9×10^{-5}	3.9×10^{-5}	0.2
6542-17-010	P13S	4-1	0	0	4.7×10^{-3}	4.7×10^{-3}	23.5
6542-16-010	P13S	4-4	0	0	4.7×10^{-3}	4.7×10^{-3}	23.5
4252-02-010	P13S	4-2,3	1×10^{-4}	3.0×10^{-5}	1.1×10^{-3}	1.1×10^{-3}	5.5
6542-21-016	P13S	4-9	4×10^{-6}	1.2×10^{-6}	2.1×10^{-3}	2.1×10^{-3}	10.5
6542-21-015	P13S	4-10	4×10^{-6}	1.2×10^{-6}	2.1×10^{-3}	2.1×10^{-3}	10.5
6542-12-025	P13S	4-11,12	0	0	2.0×10^{-4}	2.0×10^{-4}	1.0

(a) Determined by acid-leach test.

(b) Fission gas release from heavy metal contamination, $R/B_c(\text{Th}) = (\text{Th}_c)(r/b_c)$ where $r/b_c = 0.30$.

(c) $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - R/B_c(\text{Th})$.

(d) Calculated assuming a fractional release of 2×10^{-2} for unconstrained failed particles.

(e) TRISO coated.

TABLE 4-19
SUMMARY OF HIGH-TEMPERATURE FISSION GAS RELEASE MEASUREMENTS MADE ON
SELECTED CAPSULE P13R FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	Date of Measurement	Measurement Temperature (°C)	Fission Gas Release (R/B Kr-85m)
03-5	1C	Feb. 21, 1975	1100	3.3×10^{-6}
		July 9, 1975	1300	1.9×10^{-4}
		July 9, 1975	1500	2.0×10^{-2}
		July 9, 1975	1600	2.9×10^{-2}
		July 9, 1975	1100	4.3×10^{-3}
16-5	5A	March 6, 1975	1100	3.9×10^{-5}
		Oct. 16, 1975	1100	1.7×10^{-5}
		Oct. 16, 1975	1300	5.9×10^{-4}
		Oct. 16, 1975	1500	1.5×10^{-3}
		Oct. 16, 1975	1600	5.1×10^{-3}
		Oct. 16, 1975	1100	7.3×10^{-4}
17-5	5B	March 6, 1975	1300	6.8×10^{-5}
		March 6, 1975	1100	7.6×10^{-5}
		Oct. 10, 1975	1100	2.0×10^{-5}
		Oct. 10, 1975	1300	6.9×10^{-5}
		Oct. 10, 1975	1500	2.6×10^{-4}
		Oct. 10, 1975	1600	9.0×10^{-3}
		Oct. 10, 1975	1100	9.1×10^{-4}
		Dec. 19, 1975	1100	2.9×10^{-4}
19-7	5D	March 12, 1975	1300	2.5×10^{-5}
		March 12, 1975	1100	3.4×10^{-5}

TABLE 4-20
SUMMARY OF HIGH-TEMPERATURE FISSION GAS RELEASE MEASUREMENTS MADE ON
SELECTED CAPSULE P13S FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	Date of Measurement	Measurement Temperature (°C)	Fission Gas Release (R/B Kr-85m)
03-6	1C	April 11, 1975	1100	1.1×10^{-4}
		July 9, 1975	1300	3.1×10^{-4}
		July 9, 1975	1500	2.2×10^{-3}
		July 9, 1975	1600	1.2×10^{-2}
		July 9, 1975	1100	1.8×10^{-3}
		Dec. 17, 1975	1100	5.2×10^{-4}
14-5	2D	May 7, 1975	1100	1.2×10^{-5}
		Oct. 13, 1975	1100	1.7×10^{-5}
		Oct. 13, 1975	1300	3.2×10^{-5}
		Oct. 13, 1975	1500	1.6×10^{-3}
		Oct. 13, 1975	1600	1.6×10^{-2}
		Oct. 13, 1975	1100	1.0×10^{-3}
		Dec. 17, 1975	1100	4.2×10^{-4}
17-6	5B	May 7, 1975	1500	3.0×10^{-4}
		May 7, 1975	1100	5.9×10^{-5}
19-5	5D	May 7, 1975	1500	4.8×10^{-4}
		May 8, 1975	1100	7.4×10^{-5}

TABLE 4-21
COMPARISON OF CAPSULE P13R EOL IN-PILE AND TRIGA FISSION GAS RELEASE DATA (R/B Kr-85m)

Data Retrieval Number (7161-004-)	Capsule Position	Volume Average Temp. (a) (°C)	Measured TRIGA EOL R/B at 1100°C	Temp. Correction Factor (b)	TRIGA R/B at Vol. Avg. Temp. (c)	Fraction of Fissions in Sample (d)	Contribution to Total Cell R/B	Calculated TRIGA EOL Cell R/B	Measured In-Pile EOL Cell R/B
01-5	1A	910	2.4×10^{-4}	0.43	1.0×10^{-4}	0.207	2.1×10^{-5}	3.4×10^{-5}	3.4×10^{-5}
02-6	1B	890	3.1×10^{-5}	0.39	1.2×10^{-5}	0.205	2.5×10^{-6}		
03-5	1C	900	3.3×10^{-6}	0.41	1.3×10^{-6}	0.200	2.6×10^{-7}		
04-6	1D	915	1.1×10^{-4}	0.44	4.8×10^{-5}	0.199	9.6×10^{-6}		
05-6	1E	920	3.0×10^{-6}	0.44	1.3×10^{-6}	0.189	2.5×10^{-7}		
06-5	2A	1045	2.5×10^{-4}	0.79	2.0×10^{-4}	0.211	4.2×10^{-5}	1.1×10^{-4}	3.3×10^{-4}
07-5	2B	1050	1.6×10^{-5}	0.82	1.3×10^{-5}	0.208	2.7×10^{-6}		
08-13	2C	1025	1.6×10^{-5}	0.72	1.2×10^{-5}	0.203	2.4×10^{-6}		
09-5	2D	1035	3.9×10^{-4}	0.76	3.0×10^{-4}	0.193	5.8×10^{-5}		
10-5	2L	995	4.1×10^{-5}	0.64	2.6×10^{-5}	0.185	4.8×10^{-6}		
16-5	5A	1010	4.0×10^{-5}	0.68	2.7×10^{-5}	0.192	5.2×10^{-6}	1.1×10^{-5}	3.9×10^{-5}
17-5	5B	970	3.3×10^{-5}	0.57	1.9×10^{-5}	0.187	3.6×10^{-6}		
18-6	5C	985	4.7×10^{-7}	0.61	2.9×10^{-7}	0.200	5.8×10^{-8}		
19-7	5D	980	1.3×10^{-5}	0.59	7.7×10^{-6}	0.203	1.6×10^{-6}		
20-6	5E	980	6.3×10^{-6}	0.59	3.7×10^{-6}	0.218	8.1×10^{-7}		
21-5	6A	925	3.1×10^{-6}	0.46	1.4×10^{-6}	0.185	2.6×10^{-7}	3.2×10^{-6}	3.1×10^{-5}
22-5	6B	950	3.9×10^{-6}	0.51	2.0×10^{-6}	0.193	3.9×10^{-7}		
23-5	6C	895	5.0×10^{-6}	0.40	2.0×10^{-6}	0.196	3.9×10^{-7}		
24-5	6D	905	1.8×10^{-5}	0.42	7.5×10^{-6}	0.210	1.6×10^{-6}		
25-5	6E	855	7.7×10^{-6}	0.32	2.5×10^{-6}	0.216	5.4×10^{-7}		

(a) Volume average irradiation temperature for the last cycle of operation.

(b) Temperature correction factor was calculated using Eq. 4-13.

(c) Measured TRIGA fission gas release data corrected to volume average irradiation temperature.

(d) Fraction of fissions in individual fuel rods equals the amount of fissionable material in the fuel rod divided by the total amount of fissionable material in the cell at EOL.

TABLE 4-22
COMPARISON OF CAPSULE P13S EOL IN-PILE AND TRIGA FISSION GAS RELEASE DATA (R/B Kr-85m)

Data Retrieval Number (7161-004-)	Capsule Position	Volume Average Temp. (a) (°C)	Measured TRIGA EOL R/B at 1100°C	Temp. Correction Factor(b)	TRIGA R/B at Vol. Avg. Temp. (c)	Fraction of Fissions in Sample(d)	Contribution to Total Cell R/B	Calculated TRIGA EOL Cell R/B	Measured In-Pile EOL Cell R/B
01-7	1A	930	1.1×10^{-3}	0.47	5.2×10^{-4}	0.208	1.1×10^{-4}	2.7×10^{-4}	3.9×10^{-4}
02-5	1B	915	5.9×10^{-5}	0.43	2.5×10^{-5}	0.206	5.2×10^{-6}		
03-6	1C	935	1.1×10^{-4}	0.48	5.3×10^{-5}	0.200	1.1×10^{-5}		
04-5	1D	940	1.2×10^{-3}	0.50	6.0×10^{-4}	0.198	1.2×10^{-4}		
05-5	1E	935	2.2×10^{-4}	0.48	1.1×10^{-4}	0.188	2.1×10^{-5}		
11-6	2A	895	1.5×10^{-4}	0.39	5.9×10^{-5}	0.212	1.3×10^{-5}	2.2×10^{-5}	3.2×10^{-4}
12-5	2B	895	3.4×10^{-7}	0.39	1.3×10^{-7}	0.208	2.7×10^{-8}		
13-6	2C	875	3.5×10^{-5}	0.36	1.3×10^{-5}	0.203	2.6×10^{-6}		
14-5	2D	880	1.2×10^{-5}	0.37	4.4×10^{-6}	0.193	8.5×10^{-7}		
15-5	2E	850	1.0×10^{-4}	0.31	3.1×10^{-5}	0.184	5.7×10^{-6}		
16-6	5A	1055	1.8×10^{-4}	0.84	1.5×10^{-4}	0.192	2.9×10^{-5}	7.9×10^{-5}	2.4×10^{-5}
17-6	5B	1035	1.0×10^{-4}	0.75	7.5×10^{-5}	0.187	1.4×10^{-5}		
18-5	5C	1040	2.6×10^{-5}	0.78	2.0×10^{-5}	0.200	4.0×10^{-6}		
19-5	5D	1035	7.4×10^{-5}	0.76	5.6×10^{-5}	0.203	1.1×10^{-5}		
20-5	5E	1010	1.4×10^{-4}	0.68	9.5×10^{-5}	0.218	2.1×10^{-5}		
26-5	6A	955	2.1×10^{-6}	0.53	1.1×10^{-6}	0.184	2.0×10^{-7}	4.1×10^{-7}	2.9×10^{-5}
27-5	6B	935	$<10^{-7}$	0.48	4.9×10^{-8}	0.192	9.4×10^{-9}		
28-13	6C	935	8.4×10^{-7}	0.48	4.0×10^{-7}	0.195	7.8×10^{-8}		
29-5	6D	900	3.6×10^{-7}	0.41	1.5×10^{-7}	0.210	3.2×10^{-8}		
30-5	6E	855	1.3×10^{-6}	0.32	4.2×10^{-7}	0.219	9.2×10^{-8}		

(a) Volume average irradiation temperature for the last cycle of operation.

(b) Temperature correction factor was calculated using Eq. 4-13.

(c) Measured TRIGA fission gas release data corrected to volume average irradiation temperature.

(d) Fraction of fissions in individual fuel rods equals the amount of fissionable material in the fuel rod divided by the total amount of fissionable material in the cell at EOL.

TABLE 4-23
COMPARISON OF CAPSULE P13R BOL IN-PILE AND TRIGA FISSION GAS RELEASE DATA (R/B Kr-85m)

Data Retrieval Number (7161-004-)	Capsule Position	Volume Average Temp. (a) (°C)	Measured TRIGA BOL R/B at 1100°C	Temp. Correction Factor (b)	TRIGA R/B at Vol. Avg. Temp. (c)	Fraction of Fissions in Sample (d)	Contribution to Total Cell R/B	Calculated TRIGA BOL Cell R/B	Measured In-Pile BOL Cell R/B
01-5	1A	1180	2×10^{-6}	1.36	2.7×10^{-6}	0.227	6.1×10^{-7}	2.6×10^{-6}	5.9×10^{-8}
02-6	1B	1095	1×10^{-6}	0.98	1.0×10^{-6}	0.213	2.1×10^{-7}		
03-5	1C	1100	1×10^{-6}	1.00	1.0×10^{-6}	0.195	2.0×10^{-7}		
04-6	1D	1135	3×10^{-6}	1.15	3.5×10^{-6}	0.187	6.5×10^{-7}		
05-6	1E	1165	4×10^{-6}	1.29	5.2×10^{-6}	0.178	9.3×10^{-7}		
06-5	2A	920	3×10^{-6}	0.44	1.3×10^{-6}	0.211	2.7×10^{-7}	1.5×10^{-6}	1.3×10^{-7}
07-5	2B	925	3×10^{-6}	0.46	1.3×10^{-6}	0.206	2.7×10^{-7}		
08-13	2C	935	4×10^{-6}	0.48	1.9×10^{-6}	0.200	3.8×10^{-7}		
09-5	2D	915	3×10^{-6}	0.43	1.3×10^{-6}	0.194	2.5×10^{-7}		
10-5	2E	850	6×10^{-6}	0.31	1.9×10^{-6}	0.189	3.6×10^{-7}		
16-5	5A	1205	4×10^{-6}	1.50	6.0×10^{-6}	0.178	1.1×10^{-6}	3.7×10^{-6}	4.9×10^{-8}
17-5	5B	1140	2×10^{-6}	1.17	2.3×10^{-6}	0.186	4.3×10^{-7}		
18-6	5C	1125	3×10^{-6}	1.11	3.3×10^{-6}	0.195	6.4×10^{-7}		
19-7	5D	1130	2×10^{-6}	1.13	2.3×10^{-6}	0.204	4.7×10^{-7}		
20-6	5E	1215	3×10^{-6}	1.55	4.7×10^{-6}	0.237	1.1×10^{-6}		
21-5	6A	1020	4×10^{-6}	0.71	2.8×10^{-6}	0.176	4.9×10^{-7}	2.0×10^{-6}	9.2×10^{-7}
22-5	6B	980	2×10^{-6}	0.59	1.2×10^{-6}	0.169	2.0×10^{-7}		
23-5	6C	950	2×10^{-6}	0.51	1.0×10^{-6}	0.187	1.9×10^{-7}		
24-5	6D	960	6×10^{-6}	0.54	3.2×10^{-6}	0.206	6.6×10^{-7}		
25-5	6E	980	3×10^{-6}	0.59	1.8×10^{-6}	0.262	4.7×10^{-7}		

(a) Volume average irradiation temperature at the time the in-pile fission gas release measurement was made.

(b) Temperature correction factor was calculated using Eq. 4-13.

(c) Measured TRIGA fission gas release data corrected to volume average irradiation temperature.

(d) Fraction of fissions in individual fuel rods equals the amount of fissionable material in the fuel rod divided by the total amount of fissionable material in the cell at BOL.

TABLE 4-24
COMPARISON OF CAPSULE P13S BOL IN-PILE AND TRIGA FISSION GAS RELEASE DATA (R/B Kr-85m)

Data Retrieval Number (7161-004-)	Capsule Position	Volume Average Temp. (a) (°C)	Measured TRIGA BOL R/B at 1100°C	Temp. Correction Factor (b)	TRIGA R/B at Vol. Avg. Temp. (c)	Fraction of Fissions in Sample (d)	Contribution to Total Cell R/B	Calculated TRIGA BOL Cell R/B	Measured In-Pile BOL Cell R/B
01-7	1A	1000	2×10^{-6}	0.65	1.3×10^{-6}	0.227	3.0×10^{-7}	8.7×10^{-7}	6.9×10^{-8}
02-5	1B	905	1×10^{-6}	0.42	4.2×10^{-7}	0.213	8.9×10^{-8}		
03-6	1C	925	1×10^{-6}	0.46	4.6×10^{-7}	0.195	9.0×10^{-8}		
04-5	1D	935	2×10^{-6}	0.48	9.6×10^{-7}	0.187	1.8×10^{-7}		
05-5	1E	975	2×10^{-6}	0.58	1.2×10^{-6}	0.178	2.1×10^{-7}		
11-6	2A	955	2×10^{-7}	0.53	1.1×10^{-7}	0.211	2.3×10^{-8}	9.5×10^{-7}	1.0×10^{-7}
12-5	2B	915	2×10^{-6}	0.43	8.6×10^{-7}	0.206	1.8×10^{-7}		
13-6	2C	870	3×10^{-6}	0.35	1.1×10^{-6}	0.200	2.2×10^{-7}		
14-5	2D	885	2×10^{-6}	0.38	7.6×10^{-7}	0.194	1.5×10^{-7}		
15-5	2E	890	5×10^{-6}	0.39	2.0×10^{-6}	0.189	3.8×10^{-7}		
16-6	5A	1330	4×10^{-6}	2.27	9.1×10^{-6}	0.178	1.6×10^{-6}	5.5×10^{-6}	4.6×10^{-7}
17-6	5B	1295	1×10^{-6}	2.03	2.0×10^{-6}	0.186	3.7×10^{-7}		
18-5	5C	1280	2×10^{-6}	1.94	3.9×10^{-6}	0.195	7.6×10^{-7}		
19-5	5D	1280	3×10^{-6}	1.94	5.8×10^{-6}	0.204	1.2×10^{-6}		
20-5	5E	1320	3×10^{-6}	2.20	6.6×10^{-6}	0.237	1.6×10^{-6}		
26-5	6A	960	3×10^{-6}	0.54	1.6×10^{-6}	0.176	2.8×10^{-7}	1.0×10^{-6}	2.7×10^{-7}
27-5	6B	900	1×10^{-6}	0.41	4.1×10^{-7}	0.169	6.9×10^{-8}		
28-13	6C	935	4×10^{-6}	0.48	1.9×10^{-6}	0.187	3.6×10^{-7}		
29-5	6D	890	1×10^{-6}	0.39	3.9×10^{-7}	0.206	8.0×10^{-8}		
30-5	6E	900	2×10^{-6}	0.41	8.2×10^{-7}	0.262	2.1×10^{-7}		

(a) Volume average irradiation temperature at the time the in-pile fission gas release measurement was made.

(b) Temperature correction factor was calculated using Eq. 4-13.

(c) Measured TRIGA fission gas release data corrected to volume average irradiation temperature.

(d) Fraction of fissions in individual fuel rods equals the amount of fissionable material in the fuel rod divided by the total amount of fissionable material in the cell at BOL.

TABLE 4-25
FISSION PRODUCT INVENTORY OF FUEL RODS IRRADIATED IN CAPSULE P13R

Data Retrieval Number (7161-004-)	Capsule Position	Fission Product Isotope Inventory ^(a) ($\mu\text{Ci} \times 10^6$)													
		Pa-233		Ru-103		Ru-106		Cs-134		Cs-137		Ce-144		Zr-95	
		Meas.	Calc. ^(b)	Meas.	Calc. ^(b)	Meas.	Calc. ^(b)	Meas.	Calc. ^(b)	Meas.	Calc. ^(b)	Meas.	Calc. ^(b)	Meas.	Calc. ^(b)
01-5	1A	131.72	137.52	3.26	3.35	0.32	0.24	0.38	ND ^(c)	0.25	0.21	3.63	4.44	10.75	9.50
02-6	1B	143.12	144.41	2.98	3.42	0.30	0.25	0.37	ND	0.23	0.22	3.09	4.52	10.34	9.81
03-5	1C	152.44	148.29	3.26	3.43	0.24	0.24	0.37	ND	0.22	0.22	3.39	4.49	10.58	9.91
04-6	1D	155.40	152.84	3.20	3.49	0.35	0.24	0.40	ND	0.23	0.22	2.43	4.55	10.58	10.15
05-6	1E	183.52	149.37	3.81	3.42	0.34	0.24	0.45	ND	0.25	0.22	3.58	4.46	11.72	9.96
16-5	5A	ND	148.24	3.01	3.21	0.24	0.21	0.33	ND	0.20	0.19	2.54	3.96	9.13	9.10
17-5	5B	174.64	138.02	3.16	3.07	0.30	0.21	0.35	ND	0.23	0.18	2.74	3.82	10.42	8.63
18-6	5C	138.68	141.13	3.01	3.13	0.29	0.21	0.34	ND	0.22	0.18	3.04	3.86	9.45	8.73
19-7 ^(d)	5D	--	136.40	--	3.08	--	0.21	--	ND	--	0.18	--	3.78	--	8.48
20-6	5E	140.75	138.15	3.07	3.25	0.15	0.22	0.31	ND	0.21	0.19	2.69	4.00	9.45	8.83
21-5	6A	88.36	62.12	2.30	1.96	0.15	0.13	0.15	ND	0.15	0.11	2.04	2.33	6.32	4.85
22-5	6B	66.16	62.25	1.97	1.86	0.17	0.12	0.12	ND	0.12	0.10	1.93	2.15	5.53	4.55
23-5	6C	66.90	54.96	2.30	1.87	0.15	0.12	0.12	ND	0.13	0.10	1.65	2.16	5.62	4.48
24-5	6D	58.02	52.93	2.09	1.94	0.18	0.13	0.11	ND	0.12	0.10	1.69	2.20	5.31	4.55
25-5	6E	50.62	37.85	2.30	2.15	0.21	0.14	0.10	ND	0.14	0.11	1.68	2.44	5.85	4.86

(a) Fission product isotope inventory was decayed back to EOL.

(b) Fission product isotope inventory was calculated using FISS-PROD computer code.

(c) ND = not determined.

(d) Fuel rod was not gamma counted.

TABLE 4-26
FISSION PRODUCT INVENTORY OF FUEL RODS IRRADIATED IN CAPSULE P13S

Data Retrieval Number (7161-004-)	Capsule Position	Fission Product Isotope Inventory ^(a) ($\mu\text{Ci} \times 10^6$)													
		Pa-233		Ru-103		Ru-106		Cs-134		Cs-137		Ce-144		Zr-95	
		Meas.	Calc. (b)	Meas.	Calc. (b)	Meas.	Calc. (b)	Meas.	Calc. (b)	Meas.	Calc. (b)	Meas.	Calc. (b)	Meas.	Calc. (b)
01-7	1A	119.58	129.87	3.10	3.21	0.31	0.23	0.30	ND ^(c)	0.20	0.20	4.44	4.23	9.80	9.01
02-5	1B	152.44	134.28	3.53	3.22	0.25	0.23	0.35	ND	0.23	0.20	4.21	4.24	10.61	9.14
03-6	1C	145.78	136.26	3.32	3.18	0.22	0.23	0.39	ND	0.23	0.20	4.95	4.15	10.20	9.10
04-5	1D	167.24	139.10	3.26	3.20	0.29	0.23	0.33	ND	0.22	0.20	4.71	4.16	11.34	9.21
05-5	1E	182.04	119.19	3.50	2.84	0.18	0.21	0.44	ND	0.26	0.18	5.26	3.79	11.50	8.18
16-6	5A	156.88	139.32	3.44	3.03	0.37	0.20	0.31	ND	0.23	0.18	4.55	3.73	10.45	8.52
17-6	5B	102.42	130.40	2.64	2.92	0.24	0.20	0.34	ND	0.21	0.17	4.03	3.62	9.64	8.14
18-5	5C	110.41	134.55	2.73	3.00	0.25	0.20	0.30	ND	0.18	0.17	3.89	3.68	7.92	8.26
19-5	5D	112.48	130.86	2.77	2.97	0.26	0.20	0.30	ND	0.19	0.17	4.71	3.63	7.98	8.14
20-5	5E	155.45	133.29	3.23	3.15	0.32	0.21	0.30	ND	0.21	0.18	4.10	3.87	9.96	8.53

(a) Fission product isotope inventory was decayed back to EOL.

(b) Fission product isotope inventory was calculated using FISS-PROD computer code.

(c) ND = not determined.

TABLE 4-27
FISSION PRODUCT ISOTOPE COUNTING ERROR

Isotope	Counting Error, ^(a) 2σ (%)
Pa-233	5.0
Ru-103	15.0
Ru-106	10.0
Cs-134	1.6
Cs-137	1.6
Ce-144	6.0
Zr-95	1.6

(a) The 2σ counting error is calculated by:

$$2\sigma = \pm \frac{1.96}{\text{PEAK} \cdot \text{LT}} \left\{ \text{CNT} + (1 - \text{CH}/2)^2 \cdot (\text{LC} - \text{RC}) / \sqrt{\text{PTS}} \right\}^{1/2} \cdot 100 (\%)$$

where PEAK = counts in the peak (min⁻¹)

CNT = total counts in peak (peak + background)

CH = number of channels in peak,

LC = counts in lowest-energy channel

RC = counts in highest-energy channel

PTS = number of smoothing points

LT = analog-to-digital converter lifetime (min)

TABLE 4-28
GAMMA SCAN RESULTS OF CAPSULE P13R FUEL ROD CRUCIBLES

Spectrum No. (a)	Isotope Concentration ($\mu\text{Ci} \times 10^2$)		Relative Activity(b) (cpm) Ta-182 (1189 keV)	Isotope Ratio $\frac{\text{Cs-137}}{\text{Cs-134}}$
	Cs-134 (605 keV)	Cs-137 (661 keV)		
1	6.50	5.29	150	0.81
2	9.96	6.17	146	0.62
3	7.93	5.84	152	0.74
4	5.59	4.40	150	0.79
5	6.71	5.73	163	0.85
6	8.44	8.37	106	0.99
7	7.83	6.61	104	0.88
8	13.92	7.27	123	0.52
9	30.29	15.31	103	0.51
10	14.53	9.14	98	0.63
11	16.26	10.79	84	0.66
12	17.99	11.34	69	0.63
13	11.89	9.36	43	0.79
14	12.71	9.36	34	0.74
15	3.35	4.73	0	1.41
16	2.34	2.09	82	0.89
17	1.83	2.20	62	1.20
18	0	1.54	60	--
19	0	1.98	56	--
20	0	0	50	--
21	0	1.65	37	--
22	0	1.54	81	--
23	0	0	106	--
24	0	1.54	104	--
25	0	2.20	74	--
26	0	1.32	85	--
27	0	1.98	110	--
28	0	2.09	90	--

(a) Location of spectrum number is given in Fig. 4-47. Activity was decayed back to EOL.

(b) Relative activity includes background counts.

TABLE 4-29
GAMMA SCAN RESULTS OF CAPSULE P13S FUEL ROD CRUCIBLES

Spectrum No. (a)	Isotope Concentration ($\mu\text{Ci} \times 10^2$)		Relative Activity(b) (cpm) Ta-182 (1189 keV)	Isotope Ratio $\frac{\text{Cs-137}}{\text{Cs-134}}$
	Cs-134 (605 keV)	Cs-137 (661 keV)		
1	85.16	44.04	0	0.52
2	193.28	96.46	115	0.50
3	107.92	51.64	114	0.48
4	81.91	38.10	302	0.47
5	119.10	53.07	214	0.45
6	212.28	90.84	166	0.43
7	125.81	56.93	156	0.45
8	19.00	8.81	55	0.46
9	14.13	8.15	72	0.58
10	14.33	9.14	113	0.64
11	12.30	7.82	71	0.64
12	14.43	9.58	54	0.66
13	11.58	7.27	73	0.63
14	29.67	13.65	0	0.46
15	27.13	13.32	0	0.49
16	22.25	11.34	0	0.51
17	14.73	8.04	0	0.57
18	7.93	4.95	0	0.62
19	2.95	2.42	0	0.82
20	2.03	3.41	0	1.68
21	0	0	0	--
22	0	0	94	--
23	0	0	0	--
24	0	1.21	37	--
25	0	1.43	74	--
26	0	5.84	79	--
27	0	0	153	--

(a) Location of spectrum number is given in Fig. 4-48. Activity was decayed back to EOL.

(b) Relative activity includes background counts.

TABLE 4-30
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13R UNBONDED
FISSILE PARTICLE SAMPLES

Batch No.	Capsule Position	Kernel Type(a)	Number of Particles	Fission Product Isotope Ratio ^(b) (Cs-137/Zr-95)		
				Meas. (c)	Calc. (d)	Difference(e) (%)
6151-00-010-10	Cell 3-1	UC ₂	225	0.0231	0.0253	-8.7
6151-00-010-11	3-2	UC ₂	233	0.0228	0.0253	-10.0
6151-00-010-12	3-3	UC ₂	224	0.0236	0.0253	-6.7
6151-00-035-1	3-4	UC ₂	231	0.0267	0.0249	+7.2
6151-00-035-2	3-5	UC ₂	228	0.0235	0.0249	-5.6
6151-00-045-1	3-7	UC ₂	251	0.0208	0.0250	-16.8
6151-00-045-2	3-8	UC ₂	246	0.0228	0.0250	-8.8
6151-00-045-3	3-9	UC ₂	252	0.0212	0.0250	-15.2
6151-00-046-1	3-10	UC ₂	254	0.0223	0.0246	-9.3
6151-04-015-1	3-11	UC ₂	225	0.0226	0.0244	-7.4
6151-04-015-2	3-12	UC ₂	267	0.0240	0.0244	-1.6
6151-01-015-2	Cell 4-2	UC ₂	232	0.0209	0.0238	-12.2
6151-01-015-3	4-3	UC ₂	232	0.0217	0.0238	-8.8
6151-00-025-1	4-4	UC ₂	260	0.0215	0.0234	-8.1
6151-00-025-2	4-5	UC ₂	284	0.0212	0.0234	-9.4
6151-00-025-3	4-6	UC ₂	275	0.0240	0.0234	-2.6
6151-09-015-1	4-7	UC ₂	261	0.0225	0.0228	-1.3
6151-09-015-2	4-8	UC ₂	251	0.0210	0.0228	-7.9
6151-09-015-3	4-9	UC ₂	273	0.0225	0.0228	-1.3
6151-09-025-1	4-10	UC ₂	246	0.0214	0.0225	-4.9
6151-09-025-2	4-11	UC ₂	245	0.0211	0.0225	-6.2

(a) All particle samples were TRISO coated.

(b) Fission product isotope inventory was decayed back to EOL.

(c) The measured isotope ratios were normalized using intrinsic efficiency curve factors and absolute intensities of the isotopes.

(d) Calculated ratio was determined using the FISS-PROD computer code and attenuation calculations.

(e) Difference between the measured and theoretical isotope ratio.

TABLE 4-31
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13S UNBONDED
FISSILE PARTICLE SAMPLES

Batch No.	Capsule Position	Kernel Type ^(a)	Number of Particles	Fission Product Isotope Ratio ^(b) (Cs-137/Zr-95)		
				Meas. (c)	Calc. (d)	Difference (e) (%)
6151-03-015-1	Cell 3-1	UC ₂	245	0.0221	0.0229	-3.5
6151-03-015-2	3-2	UC ₂	239	0.0226	0.0229	-1.3
6151-03-015-3	3-3	UC ₂	245	0.0236	0.0229	-3.1
4161-01-021-1	3-4	UC ₂	238	0.0190	0.0229	-17.0
4161-01-021-2	3-5	UC ₂	239	0.0188	0.0229	-17.9
4161-01-021-3	3-6	UC ₂	241	0.0181	0.0229	-21.0
6151-08-015-1	3-7	UC ₂	229	0.0226	0.0225	+0.4
6151-08-015-2	3-8	UC ₂	234	0.0238	0.0225	+5.8
6151-08-015-3	3-9	UC ₂	225	0.0238	0.0225	+5.8
6151-02-025-1	3-10	UC ₂	236	0.0230	0.0222	+3.6
6151-02-025-3	3-12	UC ₂	265	0.0235	0.0222	+5.6
4163-00-011-3	Cell 4-1	(Th,U)O ₂	255	0.0212	0.0186	+14.0
4163-00-011-4	4-2	(Th,U)O ₂	254	0.0201	0.0186	+8.0
4163-00-011-5	4-3	(Th,U)O ₂	258	0.0217	0.0186	+16.7
6155-01-020-10	4-4	(Th,U)O ₂	163	0.0152	0.0141	+7.8
5466-37-9	4-7	(Th,U)O ₂	269	0.0206	0.0186	+10.8
5466-37-10	4-8	(Th,U)O ₂	269	0.0204	0.0186	+9.7
5466-37-11	4-9	(Th,U)O ₂	263	0.0198	0.0186	+6.5
6151-00-035-4	4-10	UC ₂	247	0.0233	0.0210	+11.0
6151-00-035-5	4-11	UC ₂	246	0.0214	0.0210	+1.9
6151-00-035-6	4-12	UC ₂	316	0.0206	0.0210	-1.9

(a) All particle samples were TRISO coated.

(b) Fission product inventory was decayed back to EOL.

(c) The measured isotope ratios were normalized using intrinsic efficiency curve factors and absolute intensities of the isotopes.

(d) Calculated ratio was determined using the FISS-PROD computer code and attenuation calculations.

(e) Difference between the measured and theoretical isotope ratio.

TABLE 4-32
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13R UNBONDED
FERTILE PARTICLE SAMPLES

Batch No.	Capsule Position	Kernel Type (a)	Number of Particles	Measured Fission Product Isotope Ratio (b) (Cs-137/Zr-95)	Difference (c) (%)
4252-06-018-1	Cell 3-1	ThO ₂	422	0.0086	0
6542-01-020-1	3-3	ThO ₂	502	0.0087	+1.2
6542-01-020-2	3-4	ThO ₂	502	0.0090	+4.7
6542-01-010-1	3-5	ThO ₂	455	0.0091	+5.8
6542-02-020-1	3-7	ThO ₂	539	0.0092	+7.0
6542-02-020-2	3-8	ThO ₂	533	0.0094	+9.3
6542-02-030-1	3-9	ThO ₂	574	0.0092	+7.0
6542-02-030-2	3-10	ThO ₂	574	0.0092	+7.0
6542-19-010-1	3-11	ThO ₂	457	0.0093	+8.1
6542-19-010-2	3-12	ThO ₂	541	0.0092	+7.0
6542-18-015-1	Cell 3-3	ThO ₂	477	0.0090	+4.7
6542-18-015-2	3-4	ThO ₂	476	0.0099	+15.1
6542-02-020-3	3-5	ThO ₂	576	0.0094	+9.3
6542-02-020-4	3-6	ThO ₂	573	0.0091	+5.8
6542-20-035-1	3-7	ThO ₂	502	0.0092	+7.0
6542-20-035-2	3-8	ThO ₂	510	0.0094	+9.3
6542-20-025-1	3-9	ThO ₂	503	0.0089	+3.5
6542-20-025-2	3-10	ThO ₂	537	0.0096	+11.6
6542-25-015-1	3-11	ThO ₂	539	0.0093	+8.1

(a) All particle samples are BISO coated.

(b) Fission product isotope inventory was decayed back to EOL. The isotope ratios were normalized using intrinsic efficiency curve factors and absolute intensities of the isotopes.

(c) Difference between the measured and calculated isotope ratio. The calculated isotope ratio for all particle samples was 0.0086. This ratio was determined using the FISS-PROD computer code and attenuation calculations.

TABLE 4-33
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13S UNBONDED FERTILE
PARTICLE SAMPLES

Batch No.	Capsule Position	Kernel Type (a)	Number of Particles	Measured Fission Product Isotope Ratio (b) (Cs-137/Zr-95)	Difference (c) (%)
4252-06-010-1	Cell 3-1	ThO ₂	436	0.0088	+2.3
4252-06-010-2	3-2	ThO ₂	436	0.0094	+9.3
6542-22-015-1	3-3	ThO ₂	491	0.0086	0
6542-22-015-2	3-4	ThO ₂	488	0.0090	+4.7
6542-22-025-1	3-5	ThO ₂	483	0.0086	0
6542-22-025-2	3-6	ThO ₂	481	0.0087	+1.2
6542-23-025-1	3-7	ThO ₂	496	0.0094	+9.3
6542-23-025-2	3-8	ThO ₂	496	0.0090	+4.7
6542-24-015-1	3-9	ThO ₂	475	0.0091	+5.8
6542-24-015-2	3-10	ThO ₂	467	0.0093	+8.1
6252-00-025-2 ^(d)	3-12	ThO ₂	528	0.0087	+1.2
6542-17-010-1	Cell 4-1	ThO ₂	458	0.0095	+10.5
4252-02-010-1	4-2	ThO ₂	462	0.0093	+8.1
4252-02-010-2	4-3	ThO ₂	463	0.0089	+3.5
6542-16-010-1	4-4	ThO ₂	349	0.0094	+9.3
6542-19-015-1	4-7	ThO ₂	495	0.0090	+4.7
6542-19-016-1	4-8	ThO ₂	502	0.0091	+5.8
6542-21-016-1	4-9	ThO ₂	508	0.0091	+5.8
6542-21-015-1	4-10	ThO ₂	513	0.0094	+9.3
6542-12-025-1	4-11	ThO ₂	538	0.0089	+3.5
6542-12-025-2	4-12	ThO ₂	676	0.0088	+2.3

(a) All particle samples were BISO coated except where noted.

(b) Fission product isotope inventory was decayed back to EOL. The isotope ratios were normalized using intrinsic efficiency curve factors and absolute intensities of the isotopes.

(c) Difference between the measured and calculated isotope ratio. The calculated isotope ratio for all particle samples was 0.0086. This ratio was determined using the FISS-PROD computer code and attenuation calculations.

(d) TRISO coated particle batch.

TABLE 4-34
GAMMA SCAN RESULTS OF SELECTED LOOSE PARTICLE BATCHES IN CAPSULES P13R AND P13S

6151-09-025-1 P13R Cell 4-10 UC ₂ TRISO			6151-00-035-4 P13S Cell 4-10 UC ₂ TRISO			6151-09-015-2 P13R Cell 4-8 UC ₂ TRISO			6151-00-046-1 P13R Cell 3-10 UC ₂ TRISO			4252-02-010-2 P13S Cell 4-3 ThO ₂ BISO			6252-00-025-2 P13S Cell 3-12 ThO ₂ TRISO		
Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95	Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95	Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95	Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95	Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95	Scan Number	No. of Particles Scanned	Cs-137 ^(a) Zr-95
2043	3	0.037	1184	3	0.038	2070	3	0.037	1157	1	0.042	1104	9	0.016	1130	3	0.016
2045	3	0.038	1185	3	0.038	2071	3	0.038	1158	3	0.039	1105	3	0.017	1131	3	0.016
2046	3	0.038	1186	3	0.037	2072	3	0.039	1159	2	0.040	1106	1	0.016	1132	3	0.016
2047	3	0.035	1187	3	0.039	2073	3	0.038	1160	3	0.038	1107	1	0.017	1134	3	0.016
2048	3	0.038	1188	3	0.038	2075	3	0.039	1161	2	0.044	1108	1	0.016	1135	3	0.016
2049	3	0.039	1189	3	0.039	2076	3	0.039	1162	6	0.039	1109	1	0.017	1136	3	0.016
2050	3	0.038	1190	3	0.036	2077	3	0.037	1164	3	0.040	1110	1	0.015	1137	3	0.016
2051	3	0.038	1191	3	0.036	2078	3	0.039	1165	1	0.043	1111	1	0.017	1138	3	0.017
2052	3	0.039	1192	3	0.038	2079	3	0.040	1166	1	0.043	1112	1	0.016	1139	3	0.016
2053	3	0.038	1194	3	0.039	2080	3	0.040	1167	3	0.042	1114	3	0.016	1140	3	0.016
2055	3	0.038	1195	3	0.037	2081	3	0.038	1168	3	0.040	1115	3	0.016	1141	3	0.016
2056	3	0.038	1196	3	0.038	2082	3	0.037	1169	3	0.041	1116	3	0.016	1142	3	0.017
2057	3	0.038	1197	3	0.040	2083	3	0.037	1170	3	0.033	1117	5	0.016	1144	3	0.016
2058	3	0.038	1198	3	0.037	2085	3	0.039	1171	3	0.041	1118	4	0.016	1145	3	0.016
2059	3	0.036	1199	3	0.037	2086	3	0.040	1172	3	0.040	1119	4	0.015	1146	3	0.016
2060	3	0.038	1200	3	0.039	2087	3	0.039	1174	3	0.042	1120	1	0.016	1147	3	0.017
2061	3	0.039	2035	3	0.036	2088	3	0.038	1175	3	0.042	1121	1	0.016	1148	3	0.016
2062	3	0.036	2036	3	0.037	2089	3	0.039	1176	3	0.040	1122	3	0.016	1149	3	0.016
2063	3	0.037	2037	3	0.038	2090	3	0.038	1177	3	0.042	1124	1	0.016	1150	3	0.016
2065	3	0.038	2038	3	0.040	2091	3	0.037	1178	3	0.043	1125	3	0.016	1151	3	0.016
2066	3	0.037	2039	3	0.039	2092	3	0.038	1179	3	0.040	1126	4	0.016	1152	3	0.016
2067	3	0.039	2040	3	0.037	2093	3	0.039	1180	3	0.040	1127	1	0.017	1154	3	0.016
2068	3	0.038	2041	3	0.039	2095	3	0.039	1181	3	0.040	1128	3	0.016	1155	3	0.016
2069	3	0.038	2042	3	0.038	2096	3	0.038	1182	3	0.041	1129	4	0.016	1156	3	0.016
Mean		0.038			0.037			0.038			0.041			0.016			0.016
Std. dev.		0.0009			0.001			0.001			0.002			0.0003			0.0004
Calculated ratio ^(b)		0.042 ±10%			0.038 ±10%			0.039 ±10%			0.045 ±10%			0.016 ±10%			0.016 ±10%

(a) Fission product inventory was decayed back to EOL. The measured isotope ratios were normalized using intrinsic efficiency curve factors and absolute intensities of the isotopes.

(b) Theoretical isotope ratio calculated using the FISS-PROD computer code and attenuation calculations.

TABLE 4-35
RESULTS OF METALLOGRAPHIC EXAMINATION OF FISSILE PARTICLES IRRADIATED IN CAPSULES P13R AND P13S

Batch No.	Fuel Rod Data Retrieval No. (7161-004-)	Capsule	Capsule Position	Average Temp ^(a) (°C)	Fast Fluence (10 ²¹ n/cm ²) ² (E > 0.18 MeV)HTGR	Burnup (% FIMA)	Number of Particles Examined	OPyC Coating		IPyC Coating Failure (%)	IPyC Coating Debonded (%)	Total Coating ^(c)		Remarks
								Failure (%)	95% Confidence Interval, P ^(b) (%)			Failure (%)	95% Confidence Interval, P ^(b) (%)	
6151-00-035	26-5	P13S	6C	1050	4.4	48	77	0	0 ≤ P ≤ 4.9	0	13.0	0	0 ≤ P ≤ 4.9	Heated in TRIGA to 1600°C
6151-00-035	23-5	P13R	6C	1005	4.5	49	33	0	0 ≤ P ≤ 10.9	0	3.0	0	0 ≤ P ≤ 10.9	
6151-00-035	18-5	P13S	5C	1335	8.4	66	69	2.9	0 ≤ P ≤ 10.4	0	29.0	0	0 ≤ P ≤ 5.2	
6151-00-035	03-5	P13R	1C	1065	8.7	73	30	66.7	47 ≤ P ≤ 83	66.7	ND ^(d)	66.7	47 ≤ P ≤ 83	
6151-00-035	--	P13S	4-12	975	10.7	71	84	10.7	5.0 ≤ P ≤ 18.9	ND	9.5	10.7	5.0 ≤ P ≤ 18.9	
6151-00-035	--	P13R	3-5	1075	12.0	74	78	0	0 ≤ P ≤ 4.7	0	15.4	0	0 ≤ P ≤ 4.7	
6151-00-035	13-6	P13S	2C	975	11.9	74	57	0	0 ≤ P ≤ 6.2	0	15.8	0	0 ≤ P ≤ 6.2	
6151-00-035	07-5	P13R	2B	1055	12.0	75	54	0	0 ≤ P ≤ 6.5	0	13.0	0	0 ≤ P ≤ 6.5	Faceted coatings, thermal cycled in-pile
6151-00-035	08-13	P13R	2C	1045	12.2	75	71	0	0 ≤ P ≤ 5.1	0	16.9	0	0 ≤ P ≤ 5.1	
6151-00-045	01-7	P13S	1A	1030	7.0	70	72	15.3	8.0 ≤ P ≤ 26.5	15.3	34.3	15.3	8.0 ≤ P ≤ 26.5	
6151-00-045	06-5	P13R	2A	1050	11.8	75	16	6.3	0 ≤ P ≤ 30.2	0	43.8	0	0 ≤ P ≤ 20.6	
6151-00-046	--	P13R	3-10	1025	11.8	74	77	5.2	1.5 ≤ P ≤ 11.5	13.0	45.5	3.9	2.0 ≤ P ≤ 10.6	
6151-08-015	19-5	P13S	5D	1325	7.9	65	53	17.0	8.3 ≤ P ≤ 29.3	0	54.7	3.8	0 ≤ P ≤ 13.3	
6151-08-015	19-7	P13R	5D	1225	8.2	66	78	5.1	1.5 ≤ P ≤ 11.3	2.2	11.1	2.6	0 ≤ P ≤ 9.3	
6151-08-015	15-5	P13S	2E	960	12.1	74	19	0	0 ≤ P ≤ 17.8	21.1	26.3	0	0 ≤ P ≤ 17.8	High OPyC coating density
6151-09-015	20-5	P13S	5E	1350	7.5	63	62	1.6	0 ≤ P ≤ 9.0	4.8	40.3	1.6	0 ≤ P ≤ 9.0	
6151-09-015	20-6	P13R	5E	1280	7.7	64	55	0	0 ≤ P ≤ 6.5	0	ND	0	0 ≤ P ≤ 6.5	
6151-09-015	12-5	P13S	2B	1010	11.7	74	58	0	0 ≤ P ≤ 6.1	0	25.9	0	0 ≤ P ≤ 6.1	High OPyC coating density
6151-09-025	--	P13R	4-11	1005	10.9	73	71	0	0 ≤ P ≤ 5.1	12.7	52.1	0	0 ≤ P ≤ 5.1	
6151-04-015	--	P13R	3-12	960	11.7	74	71	100	44.6 ≤ P ≤ 70.1	1.4	4.2	95.0 ^(e)	ND	
5466-37	16-6	P13S	5A	1375	9.2	36	28	7.1	0.9 ≤ P ≤ 23.0	0	0	0	0 ≤ P ≤ 12.9	250-μm (Th,U)O ₂
5466-37	16-5	P13R	5A	1285	9.4	36	35	5.7	0.5 ≤ P ≤ 19.6	0	2.9	2.9	0 ≤ P ≤ 14.8	Heated in TRIGA to 1600°C, 250-μm (Th,U)O ₂
5466-37	--	P13S	4-8	1090	10.9	38	57	0	0 ≤ P ≤ 6.2	0	0	0	0 ≤ P ≤ 6.2	250-μm (Th,U)O ₂
6155-01-020	--	P13S	4-4	1090	11.1	11	34	85.3	64.2 ≤ P ≤ 92.9	44.1	0	50.0 ^(e)	ND	500-μm (8Th,1U)O ₂

(a) Time-averaged fuel centerline temperature.

(b) The 95% confidence statement means that an infinite population of particles with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(c) Total coating failure is defined as the failure of both the OPyC and SiC structural coatings in a fuel particle that would result in the release of gaseous fission products.

(d) ND = not determined.

(e) A large fraction of the cracked SiC coatings observed in these particle batches was an artifact of sample preparation. R/B values yielded failure levels of 2.6% for batch 6155-01-020 and 0.6% for batch 6151-04-015.

TABLE 4-36
RESULTS OF METALLOGRAPHIC EXAMINATION OF FERTILE PARTICLES IRRADIATED IN CAPSULES P13R AND P13S

Batch No.	Fuel Rod Data Retrieval No. (7161-004-)	Capsule	Capsule Position	Average Temp.(a) (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	Number of Particles Examined	OPyC Coating		Buffer Coating Cracked (%)	Remarks
								Failure (%)	95% Confidence Interval, P ^(b) (%)		
6542-20-035	01-7	P13S	1A	1030	7.0	3.2	135	51.9	$43.5 \leq P \leq 60.1$	60.0	Faceted coatings, thermal cycled in-pile
6542-20-035-2	--	P13R	4-8	1000	11.1	4.3	174	2.9	$1.2 \leq P \leq 6.5$	1.7	Faceted coatings
6542-20-035	06-5	P13R	2A	1050	11.8	5.5	57	1.8	$0.2 \leq P \leq 9.8$	1.8	Faceted coatings
6542-02-020	07-5	P13R	2B	1055	12.0	5.6	161	0	$0 \leq P \leq 2.3$	1.9	
6542-01-010	28-13	P13S	6C	1050	4.4	0.8	112	0	$0 \leq P \leq 3.3$	0	High OPyC coating rate
6542-01-010	23-5	P13R	6C	1005	4.5	0.8	76	0	$0 \leq P \leq 4.8$	0	High OPyC coating rate
6542-01-010	18-5	P13S	5C	1335	8.4	2.4	165	0	$0 \leq P \leq 2.3$	7.9	High OPyC coating rate
6542-01-010	03-5	P13R	1C	1065	8.7	4.2	69	0	$0 \leq P \leq 5.2$	15.9	Heated in TRIGA to 1600°C, high OPyC coating rate
6542-01-010	13-6	P13S	2C	975	11.9	4.7	175	0	$0 \leq P \leq 2.1$	4.2	High OPyC coating rate
6542-01-010	08-13	P13R	2C	1045	12.2	5.6	144	0	$0 \leq P \leq 2.6$	4.2	High OPyC coating rate
6542-19-015	19-5	P13S	5D	1325	7.9	2.2	194	5.7	$3.2 \leq P \leq 9.9$	14.9	Low OPyC coating density
6542-19-015	19-7	P13R	5D	1225	8.2	2.4	206	0.5	$0.1 \leq P \leq 2.7$	0.5	Low OPyC coating density
6542-19-015	15-5	P13S	2E	960	12.1	4.6	36	0	$0 \leq P \leq 9.9$	8.3	Low OPyC coating density
6542-09-010	20-5	P13S	5E	1350	7.5	2.0	247	0	$0 \leq P \leq 1.5$	0	High OPyC coating density
6542-09-010	20-6	P13R	5E	1280	7.7	2.1	144	0.7	$0.1 \leq P \leq 3.8$	0.7	High OPyC coating density
6542-09-010	12-5	P13S	2B	1010	11.7	4.6	129	0.8	$0.1 \leq P \leq 4.3$	0.8	High OPyC coating density
6542-01-020	16-6	P13S	5A	1375	9.2	2.8	89	5.6	$2.1 \leq P \leq 12.6$	ND ^(c)	
6542-01-020	16-5	P13R	5A	1285	9.4	3.1	62	16.1	$7.8 \leq P \leq 27.5$	ND	Heated in TRIGA to 1600°C
6542-18-015-1	--	P13R	4-3	1000	11.4	4.4	165	0	$0 \leq P \leq 2.3$	1.2	No seal coat
6542-25-015-1	--	P13R	4-11	940	10.9	4.1	209	1.4	$0.5 \leq P \leq 4.1$	6.2	Nonround kernels
4252-06-010-1	--	P13S	3-1	925	11.8	4.3	158	2.5	$1.0 \leq P \leq 6.3$	0	
6542-23-025-1	--	P13S	3-7	955	11.7	4.2	181	0	$0 \leq P \leq 2.1$	14.4	Mixed-gas OPyC coating
6252-00-025-2	--	P13S	3-12	860	11.5	4.1	37	2.7	$0 \leq P \leq 14.1$	10.8	TRISO coated
6542-12-025-2	--	P13S	4-12	830	10.7	3.6	177	1.1	$0.3 \leq P \leq 4.0$	18.6	N ₂ carrier gas buffer

(a) Time-averaged fuel centerline temperature.

(b) The 95% confidence statement means that an infinite population of particles with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(c) ND = not determined.

TABLE 4-37
RESULTS OF METALLOGRAPHIC EXAMINATION OF INERT PARTICLES IRRADIATED IN P13R AND P13S FUEL RODS

Fuel Rod Data Retrieval No. (7161-004-)	Capsule	Capsule Position	Average Centerline Temperature (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Number of Particles Examined	OPyC Coating Failure (%)	SiC Coating Cracked (%)
28-13	P13S	6C	1050	4.4	101	1.0	3.0
23-5	P13R	6C	1005	4.5	90	0	1.1
20-5	P13S	5E	1350	7.5	260	36.9	3.1
20-6	P13R	5E	1280	7.7	193	14.0	ND ^(a)
19-5	P13S	5D	1325	7.9	237	22.8	0.8
19-7	P13R	5D	1225	8.2	277	11.6	2.9
18-5	P13S	5C	1335	8.4	211	28.9	2.8
12-5	P13S	2B	1010	11.7	259	0.7	2.4
06-5	P13R	2A	1050	11.8	149	3.3	12.1
13-6	P13S	2C	975	11.9	123	0.8	3.3
15-5	P13S	2E	960	12.1	44	2.3	ND
07-5	P13R	2B	1055	12.0	251	1.6	0.4
08-13	P13R	2C	1045	12.2	224	0	0.4

(a) ND = not determined.

TABLE 4-38
IRRADIATION-INDUCED CHANGE IN MACROPOROSITY OF P13R AND P13S FUEL RODS

Data Retrieval Number (7161-004-)	Capsule	Capsule Position	Average Centerline Temp. (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV)HTGR	Macroporosity ^(a) (%)			Matrix Composition		
					Pre-Irrad. ^(b)	Post-Irrad.	$\Delta x/x_0$	Filler	Additive	Firing Conditions ^(c)
28-13	P13S	6C	1050	4.4	38	46	+21	6353	SC011	Packed bed
23-5	P13R	6C	1005	4.5	43	43	0	6353	SC027	Ref.
01-7	P13S	1A	1030	7.0	31	38	+23	6353	SC011	Ref.
20-5	P13S	5E	1350	7.5	27	45	+67	6353	SC011	Ref.
20-6	P13R	5E	1280	7.7	27	51	+89	6353	SC011	Ref.
19-5	P13S	5D	1325	7.9	42	44	+5	6353	SC011	N ₂ atm
19-7	P13R	5D	1225	8.2	42	49	+17	6353	SC011	N ₂ atm
18-5	P13S	5C	1335	8.4	35	53	+51	6353	SC011	1500°C
03-5	P13R	1C	1065	8.7	42	54	+29	6353	SC011	Ref.
16-6	P13S	5A	1375	9.2	37	52	+41	6353	SC011	Ref.
16-5	P13R	5A	1285	9.4	37	49	+32	6353	SC011	Ref.
12-5	P13S	2B	1010	11.7	32	58	+81	6353	SC011	Ref.
06-5	P13R	2A	1050	11.8	37	48	+30	6353	SC011	1500°C
13-6	P13S	2C	975	11.9	35	47	+34	6353	SC027	Ref.
15-5	P13S	2E	960	12.1	36	55	+112	6353	SC011	Ref.
07-5	P13R	2B	1055	12.0	28	53	+89	Lonza	SC011	Ref.
08-13	P13R	2C	1045	12.2	37	60	+62	6353	SC011	Packed bed

(a) Determined by point-count method (Ref. 4-12) on 75X diametral composite cross section of fuel rod.
Macroporosity = $V_{\text{void}} / (V_{\text{void}} + V_{\text{matrix}})$.

(b) Measured on companion fuel rods.

(c) Reference firing conditions: in-block (H-451 graphite tubes) to 1800°C in argon. Deviations from these conditions are noted.

TABLE 4-39
SUMMARY OF METALLIC FISSION PRODUCT INTERACTIONS OBSERVED IN TRISO UC₂ PARTICLES

Sample Description			Irradiation Conditions						Results of Metallographic Examination			
Particle Batch Data Retrieval Number	Mean Kernel Diameter (μm)	Buffer + IPyC Coating Thickness (μm)							No. of Par- ticles	IPyC Coating Reaction ^(c) (%)	SiC Coating Reaction ^(d) (%)	Pressure Vessel Failure ^(e) (%)
			Capsule	Capsule Position	Average Temp. (a) (°C)	Maximum Temp. (b) (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)				
6151-00-035	201	120	P13S	6C	1050	1215	4.4	48	77	75	0	0
6151-00-035	201	120	P13R	6C	1005	1160	4.5	49	33	39	0	0
6151-00-035	201	120	P13S	5C	1335	1625	8.4	66	69	100	88	0
6151-00-035	201	120	P13S	2C	975	1090	11.9	74	57	100	0	0
6151-00-035	201	120	P13R	2B	1055	1220	12.0	75	54	100	1.9	0
6151-00-035	201	120	P13R	2C	1045	1195	12.2	75	71	100	8.5	0
6151-00-045	196	120	P13S	1A	1030	1545	7.0	70	72	100	2.9	15.3
6151-00-045	196	120	P13R	2A	1050	1225	11.8	75	16	90	0	6.3
6151-08-015	202	128	P13S	5D	1325	1610	7.9	65	53	100	100	3.8
6151-08-015	202	128	P13R	5D	1225	1480	8.2	66	78	100	5.1	2.6
6151-08-015	202	128	P13S	2E	960	1095	12.1	74	19	100	0	0
6151-09-015	197	126	P13S	5E	1350	1650	7.5	63	62	100	86	1.6
6151-09-015	197	126	P13R	5E	1280	1590	7.7	64	55	100	0	0
6151-09-015	197	126	P13S	2B	1010	1135	11.7	74	58	95	3.4	0

(a) Time-averaged fuel rod centerline irradiation temperature.

(b) Maximum fuel rod centerline temperature during irradiation.

(c) Fraction of particles with large concentrations of metallic fission products in the IPyC coating.

(d) Fraction of particles that exhibited metallic fission product attack of the SiC coating less than one-half the thickness of the SiC coating.

(e) Failure of the structural OPyC and SiC coatings that would result in the release of gaseous fission products.

TABLE 4-40
SUMMARY OF FUEL KERNEL MIGRATION DATA

Particle Batch Data Retrieval Number	Kernel Type	Fuel Rod Data Retrieval No. (7161-004-)	Capsule	Capsule Position	Average Centerline Temperature (°C)	Particle Ident. Number (a) (L7440-)	Particle Location (b) (cm)	Migration Distance (µm)
6151-08-015	UC ₂	19-7	P13R	5D	1225	-381	0.5113	ND ^(c)
						-383	0.5808	2.0
						-385	0.2015	ND
						-393	0.1180	2.0
						-397	0.0085	6.0
						-401	0.5095	ND
6151-00-035	UC ₂	18-5	P13S	5C	1335	-403	0.2300	15.0
						-428	0.5915	2.0
						-431	0.5715	2.0
						-434	0.1725	6.0
						-437	0.3730	3.0
						-444	0.4028	12.0
6151-08-015	UC ₂	19-5	P13S	5D	1325	-601	0.1253	9.0
						-316	0.3259	4.0
						-319	0.2269	6.0
						-331	0.5961	1.0
						-325	0.5100	ND
6151-00-035	UC ₂	28-3	P13S	6C	1050	-466	0.5000	ND
						-468	0.4157	ND
						-470	0.4319	ND
						-476	0.2748	ND
						-478	0.0951	ND
						-480	0.4634	ND
						-484	0.1541	ND
						-486	0.1159	ND
						-490	0.5818	ND
5466-37	(1Th,1U)O ₂	16-5	P13R	5A	1285	-1009b	0.2223	ND
						-1011	0.2110	4.0
						-1025	0.3378	1.0
						-1027	0.5720	ND
5466-37	(1Th,1U)O ₂	16-6	P13S	5A	1375	-1041	0.5936	13.0
						-1043	0.5395	23.0
						-1051	0.5336	15.0
						-1045	0.3391	6.0
						-1055	0.2795	17.0
						-1059	0.0735	8.0
						-1063	0.4925	12.0
						-1061	0.2606	ND
						-1067	0.5917	19.0
						-1065	0.4626	7.0
						-1070	0.3022	4.0

(a) Identification number of laboratory photograph of fuel particle.

(b) Distance of particle from fuel rod centerline.

(c) ND = kernel migration was not detectable.

TABLE 4-41
SUMMARY OF FISSION GAS RELEASE MEASUREMENTS ON UNFAILED BISO ThO₂ PARTICLE BATCHES

Data Retrieval Number	OPyC Coating			Capsule	Capsule Position	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	No. of Particles	OPyC Coating Failure ^(b) (%)		Fission Gas Release ^(c) (R/B Kr-85m)
	Density ^(a) (g/cm ³)	Rate (μm/min)	Gas						Pre-FGR	Post-FGR	
6542-09-010	1.93	5.0	Propylene	P13R	3-11,12	11.7	4.8	998	0	0.3	2.4 x 10 ⁻⁵
6542-02-020	1.91	8.5	Propylene	P13R	3-7,8	11.9	4.9	1072	0	0	4.4 x 10 ⁻⁵
6542-18-015	1.84	6.5	Propylene	P13R	4-3,4	11.4	4.5	953	0	0	2.2 x 10 ⁻⁶
6542-02-020	1.91	8.5	Propylene	P13R	4-5,6	11.3	4.4	1149	0	0	3.3 x 10 ⁻⁵
4252-06-010	1.82	5.8	Propylene	P13S	3-1,2	11.8	4.3	872	0	0	6.2 x 10 ⁻⁵
6542-23-025	1.89	5.8	MG ^(d)	P13S	3-7,8	11.7	4.2	992	0	0	2.8 x 10 ⁻⁶
6542-24-015	1.94	6.6	MG ^(d)	P13S	3-9,10	11.6	4.1	942	0	0	3.9 x 10 ⁻⁶
4252-02-010	1.83	4.0	Propylene	P13S	4-2,3	11.2	3.9	925	0	0	1.1 x 10 ⁻³
6542-21-016	1.73	8.0	MG ^(d)	P13S	4-9	10.9	3.7	508	0	0.6	2.1 x 10 ⁻³
6542-21-015	1.73	7.8	MG ^(d)	P13S	4-10	10.8	3.6	513	0	0	2.1 x 10 ⁻³

(a) Preirradiation density gradient column measurement.

(b) Determined by visual examination before and after fission gas release (FGR) measurements.

(c) Fission gas release measurements made in TRIGA at 1100°C.

(d) OPyC coating was deposited from mixed gas (MG) (propylene-acetylene).

TABLE 4-42
SUMMARY DESCRIPTION OF ThO₂ PARTICLE SAMPLES SELECTED FOR ACID LEACHING AND FISSION GAS RELEASE MEASUREMENTS

Data Retrieval Number	OPyC Coating			Capsule	Capsule Position	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV)HTGR	Burnup (% FIMA)	No. of Par- ticles	OPyC Coating Failure ^(b)	Fission Gas Release ^(c) (R/B Kr-85m)
	Density ^(a) (g/cm ³)	Rate (μm/min)	Gas							
6542-21-016-1	1.73	8.0	MG ^(d)	P13S	4-9	10.9	3.7	197	1	3.0 x 10 ⁻³
6542-22-015-1	1.80	4.0	MG ^(d)	P13S	3-3	11.7	4.3	207	0	6.7 x 10 ⁻⁴
6542-24-015-1	1.94	6.6	MG ^(d)	P13S	3-9	11.6	4.1	200	0	1.4 x 10 ⁻⁵
6542-01-020-2	1.82	2.7	Propylene	P13R	3-4	12.0	5.0	55	0	1.6 x 10 ⁻³
4252-06-018-1	1.82	5.8	Propylene	P13R	3-1	12.1	5.1	196	5 ^(e)	3.0 x 10 ⁻³
6542-01-010-1	1.80	10.0	Propylene	P13R	3-5	12.0	5.0	202	0	1.5 x 10 ⁻³
6542-02-020-2	1.91	8.5	Propylene	P13R	3-8	11.9	4.9	205	0	5.6 x 10 ⁻⁵
6542-09-010-2	1.93	5.0	Propylene	P13R	3-12	11.7	4.7	201	0	8.5 x 10 ⁻⁶

(a) Preirradiation density gradient column measurement.

(b) Number of particles in sample with failed coatings. Failure was detected by visual examination of sample after fission gas release measurements but before acid leaching.

(c) Fission gas release measurements made on particle sample at 1100°C prior to acid leaching.

(d) OPyC coating was deposited from mixed gas (MG) (propylene-acetylene).

(e) Particles contained very fine hair-line cracks.

TABLE 4-43
RESULTS OF SPECTROPHOTOMETRIC ANALYSIS OF LEACH SOLUTIONS

Aliquot	6542-21-016-1			6542-22-015-1			6542-24-015-1			6542-01-020-2			4252-06-018-1			6542-02-020-2			6542-09-010-2		
	Leach Time ^(a) (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)	Leach Time (hr)		Thorium Leached (μg)
	Hot	Cold		Hot	Cold		Hot	Cold		Hot	Cold		Hot	Cold		Hot	Cold		Hot	Cold	
1	4	15.5	1.0×10^4	6	--	2.5×10^3	5.5	--	2.7×10^2	3.5	--	6.3×10^2	5.5	--	2.2×10^4	2.5	--	2.5×10^2	2.5	--	2.4×10^2
2	11	31	1.1×10^5	9	16	1.0×10^4	7.0	16.0	2.5×10^2	5.5	16.0	4.1×10^3	7.0	16.0	3.6×10^4	5.5	16.0	2.4×10^2	5.5	16.0	2.4×10^2
3	17	31	1.1×10^5	13	16	4.5×10^4	12.5	16.0	3.7×10^2	13.0	16.5	2.2×10^4	12.5	16.0	1.2×10^5	9.0	16.0	8.7×10^2	9.0	16.0	4.8×10^2
4	21	46	1.1×10^5	17.5	31.5	1.1×10^5	15.0	32.5	6.5×10^2	16.5	32.0	2.9×10^4	15.0	32.5	1.2×10^5	12.0	79.5	3.6×10^3	12.0	79.5	2.9×10^3
5	--	--	--	22.5	31.5	NA ^(b)	--	--	--	20.5	47.5	3.3×10^4	--	--	--	18.0	79.5	1.8×10^4	18.0	79.5	NA
6	--	--	--	26.0	47.0	NA	--	--	--	24.5	63.0	NA	--	--	--	18.0	85.0	2.6×10^4	--	--	--

(a) All leach times are cumulative. Hot is the boiling time and cold is with the hot plate turned off at night.

(b) Leach solution was not analyzed.

TABLE 4-44
RESULTS OF VISUAL AND METALLOGRAPHIC EXAMINATIONS OF BISO COATED PARTICLES AFTER ACID LEACHING

Data Retrieval Number	Capsule	Capsule Position	Visual Examination		Metallographic Examination		Remarks
			Number of Particles Recovered	Number of Particles with Failed OPyC Coatings	Number of Particles Examined	Number of Particles with Failed OPyC Coatings	
6542-21-016-1	P13S	4-9	190	0 ^(a)	148	2	All kernels were heavily leached
6542-22-015-1	P13S	3-3	202	43	144	68	Severe degradation of OPyC coating and all kernels were heavily leached
6542-24-015-1	P13S	3-9	195	1	157	2	Mild degradation of OPyC coating; ~90% of kernels were unleached
6542-01-020-2	P13R	3-4	44	9	27	16	OPyC coating appeared pitted and porous; all kernels were heavily leached
4252-06-018-1	P13R	3-1	187	6	151	21	OPyC coating appeared porous; all kernels were heavily leached
6542-01-010-1	P13R	3-5	155	5	143	2	OPyC coating appeared porous; all kernels were heavily leached
6542-02-020-2	P13R	3-8	204	22	158	31	Mild degradation of OPyC coating; ~90% of kernels were unleached
6542-09-010-2	P13R	3-12	200	0	161	0	Mild degradation of OPyC coating; ~90% of kernels were unleached

(a) The surface of all particles was crazed; however, during the metallographic examination it was observed that the microcracks did not penetrate the thickness of the OPyC coating.

TABLE 4-45
CHEMICAL ANALYSIS OF LEACH SOLUTIONS

Aliquot	Time (hr)	Volume of Aliquot (mℓ)	Total U (mg/mℓ)
1	3.5	5.9	247 ± 4.5%
2	7.0	4.5	255 ± 0.2%
3	13.0	4.7	245 ± 0.5%
4	15.0	6.8	259 ± 2.7%

TABLE 4-46
URANIUM ISOTOPIC COMPOSITION OF THE LEACH SOLUTION

Isotope	Weight (mg)	Weight (%)
U-233	17.92 ± 0.025	71.930 ± 0.08
U-234	5.367 ± 0.007	21.739 ± 0.004
U-235	1.110 ± 0.0015	4.496 ± 0.02
U-236	0.385 ± 0.007	1.558 ± 0.01
U-238	0.069 ± 0.002	0.279 ± 0.002

TABLE 4-47
URANIUM ISOTOPIC COMPOSITION OF THE FUEL ROD^(a)

Isotope	Weight (mg)		
	Fissile	Fertile	Total
U-233	0	16.72	16.72
U-234	0.10	5.97	6.07
U-235	1.61	1.29	2.90
U-236	4.97	0.23	5.20
U-238	1.67	0	1.67

^(a) Composition at EOL + 120 days calculated using FISSIN computer program.

TABLE 4-48
POSTIRRADIATION PARTICLE DIMENSIONS DETERMINED BY CONTACT MICRORADIOGRAPHY

Data Retrieval Number	Capsule	Capsule Position	Radio-graph Plate No.	Number of Particles	Kernel			Buffer			SiC			OPyC			Total Particle		
					Diam (um)	Std. Dev. (um)	Range (um)	Thick-ness (um)	Std. Dev. (um)	Range (um)	Thick-ness (um)	Std. Dev. (um)	Range (um)	Thick-ness (um)	Std. Dev. (um)	Range (um)	Diam (um)	Std. Dev. (um)	Range (um)
6151-00-035-2	P13R	3-5	HA370, 377	57	ND ^(a)	ND	ND	ND	ND	ND	26	2.6	20-31	35	5.5	25-60	580	31.0	490-660
4252-06-018-1	P13R	3-1	HA349, 352, 354	69	529	12.8	508-576	69	6.5	55-91	--	--	--	73	6.0	58-87	810	25.6	730-877
4252-06-018-2	P13R	3-2	HA353, 355, 358	59	532	20.9	510-657	67	6.3	49-82	--	--	--	73	5.5	59-81	812	26.1	728-872
6542-19-016-1	P13S	4-8	HA359, 363, 369	87	506	17.8	462-542	71	6.7	54-89	--	--	--	72	6.5	55-88	796	27.6	722-844
6542-21-016-1	P13S	4-9	HA364, 368	75	505	17.9	463-538	66	6.2	53-85	--	--	--	78	7.1	60-97	796	29.8	730-855

^(a)ND = not determined.

TABLE 4-49
IRRADIATION-INDUCED PARTICLE DIMENSIONAL CHANGE DETERMINED BY MICRORADIOGRAPHY

Data Retrieval Number ^(a)	Capsule	Capsule Position	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	Particle Type	Kernel Diameter		Buffer Thickness		OPyC Thickness		Total Particle Diameter	
						ΔD (μm)	$\Delta D/D_0$ (%)	ΔT (μm)	$\Delta T/T_0$ (%)	ΔT (μm)	$\Delta T/T_0$ (%)	ΔD (μm)	$\Delta D/D_0$ (%)
6151-00-035-2	P13R	3-5	12.0	74	UC ₂ TRISO	ND ^(b)	ND	ND	ND	-1	-2.8	0	0
4252-06-018-1	P13R	3-1	12.1	5.1	ThO ₂ BISO	+8	+1.5	-10	-12.7	-9	-11.0	-30	-3.6
4252-06-018-2	P13R	3-2	12.1	5.1	ThO ₂ BISO	+11	+2.1	-13	-16.3	-8	-9.9	-30	-3.6
6542-19-016-1	P13S	4-8	10.9	3.7	ThO ₂ BISO	+9	+1.8	-11	-13.4	-10	-12.2	-31	-3.7
6542-21-016-1	P13S	4-9	10.9	3.7	ThO ₂ BISO	+8	+1.6	-14	-17.5	-11	-12.4	-41	-4.9

(a) Preirradiation dimensions of the actual sample tested, not the parent batch, were used to calculate dimensional change.

(b) ND = not determined.

TABLE 4-50
RESULTS OF OPyC DENSITY MEASUREMENTS OF UNBONDED FISSILE PARTICLE SAMPLES IRRADIATED IN CAPSULES P13R AND P13S

Sample Description						Postirradiation Density Measurements (a)						
Data Retrieval Number	Preirradiation Density (a) (g/cm ³)	Capsule	Capsule Position	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) HTGR	Average Irrad. Temp. (°C)	Fragments Obtained From		Number of Fragments	Mean Density (g/cm ³)	Density Range (g/cm ³)	Standard Deviation (g/cm ³)	Densification (%)
						Failed Particles	Intact Particles					
6151-00-010-10	1.80	P13R	3-1	12.1	1015	X	X	13	2.07	2.01 - 2.09	0.021	+15.3
6151-00-035-1	1.85	P13R	3-4	12.0	1075		X	5	2.01	2.01 - 2.02	0.007	+8.6
6151-04-015-1	1.82	P13R	3-11	11.7	960	X		16	2.06	1.97 - 2.09	0.025	+13.2
6151-04-015-1	1.82	P13R	3-11	11.7	960	X		9	2.08	2.07 - 2.08	0.003	+14.3
6151-01-015-2	1.90	P13R	4-2	11.5	1070	X		31	2.01	1.92 - 2.03	0.019	+5.8
6151-00-025-2	1.88	P13R	4-4	11.4	1070	X		19	2.02	1.97 - 2.09	0.038	+7.4
6151-09-015-1	1.94	P13R	4-7	11.2	1070		X	17	2.07	2.03 - 2.10	0.019	+6.7
6151-08-015-1	1.76	P13S	3-7	11.7	1020		X	10	2.01	2.00 - 2.02	0.007	+14.2
6151-02-025-1	1.50	P13S	3-10	11.6	980	X		20	2.12	1.84 - 2.16	0.070	+41.3
6155-01-020-10	1.81	P13S	4-4	11.1	1090	X		15	2.00	1.97 - 2.03	0.002	+10.5

(a) Measured using a density gradient column.

TABLE 4-51
RESULTS OF DENSITY MEASUREMENTS OF UNBONDED FERTILE PARTICLE SAMPLES IRRADIATED IN CAPSULES P13R AND P13S

Sample Description					Preirradiation (a)				Postirradiation (a)				
Data Retrieval Number	Capsule	Capsule Position	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	No. of Par- ticles	Mean Density (g/cm ³)	Density Range (g/cm ³)	Standard Deviation (g/cm ³)	No. of Par- ticles	Mean Density (g/cm ³)	Density Range (g/cm ³)	Standard Deviation (g/cm ³)	Densification (%)
4252-06-018-1	P13R	3-1	12.1	5.1	25	3.72	3.69 - 3.97	0.053	30	3.98	3.96 - 4.01	0.010	+7.0
4252-06-018-2	P13R	3-2	12.1	5.1	25	3.72	3.69 - 3.97	0.053	30	4.00	3.99 - 4.19	0.050	+7.5
6542-19-016-1	P13S	4-8	10.9	3.7	ND ^(b)	3.45	ND	ND	33	3.98	3.96 - 3.98	0.005	+15.4
6542-21-016-1	P13S	4-9	10.9	3.7	ND	3.41	ND	ND	11	3.94	3.85 - 3.96	0.034	+15.5

(a) Measured using a density gradient column.

(b) ND = not determined.

K7440-80b

K7440-81

Fig. 4-1. Photomicrographs showing P13S cell No. 2 fuel rods during removal from split graphite crucible

K7425-95

(a)

K7425-102

(b)

Fig. 4-2. Representative photomicrographs showing (a) unbonded particle crucible assembly after removal from primary containment, and (b) unloading of unbonded particle samples from crucible

S7440-11

S7440-9

0.5 cm

S7440-101

S7440-98

Fig. 4-3. Photomicrographs of fuel rod 7161-004-01-5 (top) after irradiation in capsule P13R (position 1A) to a fast neutron fluence of $7.3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1115°C and of fuel rod 7161-004-01-7 (bottom) after irradiation in capsule P13S (position 1A) to a fast neutron fluence of $7.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Cell No. 1 of capsule P13S was thermal cycled 24 times during irradiation.

S7440-7

S7440-5

0.5 cm

S7440-104

S7440-103

Fig. 4-4. Photomicrographs of fuel rod 7161-004-02-6 (top) after irradiation in capsule P13R (position 1B) to a fast neutron fluence of $8.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C and of fuel rod 7161-004-02-5 (bottom) after irradiation in capsule P13S (position 1B) to a fast neutron fluence of $7.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Cell No. 1 of capsule P13S was thermal cycled 24 times during irradiation.

S7440-20

S7440-19

0.5 cm

S7440-108

S7440-107

Fig. 4-5. Photomicrographs of fuel rod 7161-004-03-5 (top) after irradiation in capsule P13R (position 1C) to a fast neutron fluence of $8.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1065°C and of fuel rod 7161-004-03-6 (bottom) after irradiation in capsule P13S (position 1C) to a fast neutron fluence of $8.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Cell No. 1 of capsule P13S was thermal cycled 24 times during irradiation.

S7440-15

S7440-13

0.5 cm

S7440-114

S7440-111

Fig. 4-6. Photomicrographs of fuel rod 7161-004-04-6 (top) after irradiation in capsule P13R (position 1D) to a fast neutron fluence of 9.3×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} at 1115°C and of fuel rod 7161-004-04-5 (bottom) after irradiation in capsule P13S (position 1D) to a fast neutron fluence of 9.1×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR}. Cell No. 1 of capsule P13S was thermal cycled 24 times during irradiation.

S7440-1

S7440-3

0.5 cm

S7440-117

S7440-116

Fig. 4-7. Photomicrographs of fuel rod 7161-004-05-6 (top) after irradiation in capsule P13R (position 1E) to a fast neutron fluence of $9.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1125°C and of fuel rod 7161-004-05-5 (bottom) after irradiation in capsule P13S (position 1E) to a fast neutron fluence of $9.6 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Cell No. 1 of capsule P13S was thermal cycled 24 times during irradiation.

S7440-26

S7440-24

0.5 cm

S7440-122

S7440-121

Fig. 4-8. Photomicrographs of fuel rod 7161-004-06-5 (top) after irradiation in capsule P13R (position 2A) to a fast neutron fluence of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C and of fuel rod 7161-004-11-6 (bottom) after irradiation in capsule P13S (position 2A) to a fast neutron fluence of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1040°C

S7440-53

S7440-52

0.5 cm

S7440-127

S7440-125

Fig. 4-9. Photomicrographs of fuel rod 7161-004-07-5 (top) after irradiation in capsule P13R (position 2B) to a fast neutron fluence of $12.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1055°C and of fuel rod 7161-004-12-5 (bottom) after irradiation in capsule P13S (position 2B) to a fast neutron fluence of $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1010°C

S7440-58

S7440-56

0.5 cm

S7440-130

S7440-129

Fig. 4-10. Photomicrographs of fuel rod 7161-004-08-013 (top) after irradiation in capsule P13R (position 2C) to a fast neutron fluence of $12.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1045°C and of fuel rod 7161-004-13-6 (bottom) after irradiation in capsule P13S (position 2C) to a fast neutron fluence of $11.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 975°C

S7440-63

S7440-60

0.5 cm

S7440-134

S7440-132

Fig. 4-11. Photomicrographs of fuel rod 7161-004-09-5 (top) after irradiation in capsule P13R (position 2D) to a fast neutron fluence of $12.3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1035°C and of fuel rod 7161-004-14-5 (bottom) after irradiation in capsule P13S (position 2D) to a fast neutron fluence of $12.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 985°C

S7440-66

S7440-64

0.5 cm

S7440-138

S7440-136

Fig. 4-12. Photomicrographs of fuel rod 7161-004-10-5 (top) after irradiation in capsule P13R (position 2E) to a fast neutron fluence of $12.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 965°C and of fuel rod 7161-004-15-5 (bottom) after irradiation in capsule P13S (position 2E) to a fast neutron fluence of $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 960°C

S7440-73

S7440-68

0.5 cm

S7440-145

S7440-142

Fig. 4-13. Photomicrographs of fuel rod 7161-004-16-5 (top) after irradiation in capsule P13R (position 5A) to a fast neutron fluence of $9.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1285°C and of fuel rod 7161-004-16-6 (bottom) after irradiation in capsule P13S (position 5A) to a fast neutron fluence of $9.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1375°C

S7440-80

S7440-76

0.5 cm

S7440-150

S7440-147

Fig. 4-14. Photomicrographs of fuel rod 7161-004-17-5 (top) after irradiation in capsule P13R (position 5B) to a fast neutron fluence of $9.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1225°C and of fuel rod 7161-004-17-6 (bottom) after irradiation in capsule P13S (position 5B) to a fast neutron fluence of $8.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1330°C

S7440-84

S7440-83

0.5 cm

S7440-153

S7440-152

Fig. 4-15. Photomicrographs of fuel rod 7161-004-18-6 (top) after irradiation in capsule P13R (position 5C) to a fast neutron fluence of $8.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1235°C and of fuel rod 7161-004-18-5 (bottom) after irradiation in capsule P13S (position 5C) to a fast neutron fluence of $8.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1335°C

S7440-89

S7440-88

0.5 cm

S7440-160

S7440-157

Fig. 4-16. Photomicrographs of fuel rod 7161-004-19-7 (top) after irradiation in capsule P13R (position 5D) to a fast neutron fluence of $8.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1225°C and of fuel rod 7161-004-19-5 (bottom) after irradiation in capsule P13S (position 5D) to a fast neutron fluence of $7.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1325°C

S7440-94

S7440-92

0.5 cm

S7440-163

S7440-161

Fig. 4-17. Photomicrographs of fuel rod 7161-004-20-6 (top) after irradiation in capsule P13R (position 5E) to a fast neutron fluence of $7.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1280°C and of fuel rod 7161-004-20-5 (bottom) after irradiation in capsule P13S (position 5E) to a fast neutron fluence of $7.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1350°C

S7440-30

S7440-29

0.5 cm

S7440-169

S7440-166

Fig. 4-18. Photomicrographs of fuel rod 7161-004-21-5 (top) after irradiation in capsule P13R (position 6A) to a fast neutron fluence of $5.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1065°C and of fuel rod 7161-004-26-5 (bottom) after irradiation in capsule P13S (position 6A) to a fast neutron fluence of $5.3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1085°C

S7440-36

S7440-34

0.5 cm

S7440-173

S7440-171

Fig. 4-19. Photomicrographs of fuel rod 7161-004-22-5 (top) after irradiation in capsule P13R (position 6B) to a fast neutron fluence of $5.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1055°C and of fuel rod 7161-004-27-5 (bottom) after irradiation in capsule P13S (position 6B) to a fast neutron fluence of $4.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1045°C

S7440-40

S7440-38

0.5 cm

S7440-176

S7440-174

Fig. 4-20. Photomicrographs of fuel rod 7161-004-23-5 (top) after irradiation in capsule P13R (position 6C) to a fast neutron fluence of $4.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1005°C and of fuel rod 7161-004-28-13 (bottom) after irradiation in capsule P13S (position 6C) to a fast neutron fluence of $4.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C

S7440-43

S7440-41

0.5 cm

S7440-180

S7440-179

Fig. 4-21. Photomicrographs of fuel rod 7161-004-24-5 (top) after irradiation in capsule P13R (position 6D) to a fast neutron fluence of $4.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1020°C and of fuel rod 7161-004-29-5 (bottom) after irradiation in capsule P13S (position 6D) to a fast neutron fluence of $3.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1015°C

S7440-50

S7440-47

0.5 cm

S7440-185

S7440-182

Fig. 4-22. Photomicrographs of fuel rod 7161-004-25-5 (top) after irradiation in capsule P13R (position 6E) to a fast neutron fluence of $3.6 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1000°C and of fuel rod 7161-004-30-5 (bottom) after irradiation in capsule P13S (position 6E) to a fast neutron fluence of $3.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 995°C

S7440-372

Fig. 4-23. Photograph of TRISO (8Th,1U)O₂ particles (6155-01-020) after irradiation in position 4-4 of capsule P13S to a fast neutron fluence of 11.1×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR}. In some TRISO coated particle batches, the fragments of failed OPyC coatings remained attached to the SiC coating substrate, which made an accurate assessment of the total coating failure level impossible by visual examination.

S7440-273

S7440-274

Fig. 4-24. Representative photographs of LHTGR reference design type fissile particles. The particle batch (6151-00-035) exhibited 0% ($0 \leq P \leq 0.8$) failure after irradiation in positions 3-4 and 3-5 of capsule P13R to a fast neutron fluence of $12.0 \times 10^{21} \text{ n/cm}^2$ and burnup of 74% FIMA at 1075°C.

S7440-266

S7440-267

Fig. 4-25. Representative photographs of LHTGR reference design type fertile particles. The particle batch (6542-02-020) exhibited 0% ($0 \leq P \leq 0.5$) failure after irradiation in positions 3-7 and 3-8 of capsule P13R to a fast neutron fluence of $11.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HGTR} and burnup of 4.9% FIMA at 1010°C.

S7440-311

S7440-312

Fig. 4-26. Representative photographs of LHTGR reference design type fertile particles with the OPyC coating deposited from mixed gas. The particle batch (6542-23-025) exhibited 0% ($0 \leq P \leq 0.5$) failure after irradiation in positions 3-7 and 3-8 of capsule P13S to a fast neutron fluence of $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.2% FIMA at 955°C.

S7440-244

(a)

S7440-256

(b)

Fig. 4-27. Photographs of TRISO UC_2 and BISO ThO_2 particles after irradiation in capsule P13R. The high OPyC coating failure levels of 49.0 and 10.2% for the fissile and fertile particles, respectively, were due to the poor irradiation stability of the OPyC coatings, which were deposited at relatively low coating rates and therefore had high BAF_0 values. (a) TRISO UC_2 particles (6151-04-015) irradiated to $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and (b) BISO ThO_2 particles (6542-02-030) irradiated to $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

S7440-224

(a)

S7440-204

(b)

Fig. 4-28. Photographs of TRISO UC_2 and BISO ThO_2 particles after irradiation in capsule P13R. Coating failure of both particle types was due to the poor irradiation stability of the severely faceted coatings. (a) TRISO UC_2 particles (6151-00-025) irradiated to $11.3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and (b) BISO ThO_2 particles (6542-20-025) irradiated to $11.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

S7440-383

S7440-384

Fig. 4-29. Representative photographs of TRISO (Th,U)O₂ fissile particles. The particle batch (4163-00-011) exhibited 0% ($0 \leq P \leq 0.5$) failure after irradiation in positions 4-1, 4-2, and 4-3 of capsule P13S to a fast neutron fluence of 11.2×10^{21} n/cm² ($E > 0.18$ MeV) HTGR and burnup of 27% FIMA at 1045°C.

S7440-294

S7440-296

Fig. 4-30. Representative photographs of TRISO ThO_2 fertile particles. The particle batch (6252-00-025) exhibited 0.4% ($0.1 < P < 1.4$) OPyC coating failure after irradiation in position 3-12 of capsule P13S to a fast neutron fluence of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.1% FIMA at 860°C.

K7440-31

(a)

K7440-32

(b)

Fig. 4-31. Photographs showing fuel rod dimensional measurements being made using (a) a barrel micrometer and (b) a dial gauge micrometer

Fig. 4-32. Measured and predicted linear dimensional change data for P13R and P13S fuel rods irradiated at 1100°C (centerline temperature): (a) $\Delta D/D_0$ and (b) $\Delta L/L_0$

Fig. 4-33. Measured and predicted volume change data for P13R and P13S fuel rods irradiated at 1100°C (centerline temperature)

Fig. 4-34. Irradiation-induced diametral change of graphite crucibles as a function of fast neutron fluence

Fig. 4-35. Fission gas release versus reciprocal temperature for fuel rod 7161-004-19-5 irradiated in capsule P13S (position 5D)

Fig. 4-36. Fission gas release versus reciprocal temperature for fuel rod 7161-004-17-6 irradiated in capsule P13S (position 5B)

Fig. 4-37. Fission gas release versus reciprocal temperature for fuel rod 7161-004-19-7 irradiated in capsule P13R (position 5D)

Fig. 4-38. Fission gas release versus reciprocal temperature for fuel rod 7161-004-03-5 irradiated in capsule P13R (position 1C)

Fig. 4-39. Fission gas release versus reciprocal temperature for fuel rod 7161-004-16-5 irradiated in capsule P13R (position 5A)

Fig. 4-40. Fission gas release versus reciprocal temperature for fuel rod 7161-004-14-5 irradiated in capsule P13S (position 2D)

Fig. 4-41. Fission gas release versus reciprocal temperature for fuel rod 7161-004-03-6 irradiated in capsule P13S (position 1C)

Fig. 4-42. Fission gas release versus reciprocal temperature for fuel rod 7161-004-17-5 irradiated in capsule P13R (position 5B)

Fig. 4-43. Single-channel Cs-137 and Zr-95 gamma-ray scans of fuel rods irradiated in cell 1 of capsule P13R

Fig. 4-44. Single-channel Cs-137 and Zr-95 gamma-ray scans of fuel rods irradiated in cell 1 of capsule P13S

Fig. 4-45. Single-channel Cs-137 and Zr-95 gamma-ray scans of fuel rods irradiated in cell 5 of capsule P13R

Fig. 4-46. Single-channel Cs-137 and Zr-95 gamma-ray scans of fuel rods irradiated in cell 5 of capsule P13S

Fig. 4-47. Single-channel gamma scan of Cs-137 (661 keV) and location of multichannel gamma scan spectra for P13R graphite fuel rod crucibles

Fig. 4-48. Single-channel gamma scan of Cs-137 (661 keV) and location of multichannel gamma scan spectra for P13S graphite fuel rod crucibles

L7440-41

(a)

L7440-42

(b)

Fig. 4-49. Photomicrographs of representative particles from fuel rod 7161-004-06-5 irradiated in capsule P13R (position 2A) to a fast neutron fluence of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C. (a) Bright field and (b) polarized light.

L7440-289

(a)

L7440-290

(b)

Fig. 4-50. Photomicrographs of representative particles from fuel rod 7161-004-07-5 irradiated in capsule P13R (position 2B) to a fast neutron fluence of $12.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1055°C . (a) Bright field and (b) polarized light.

L7440-254

(a)

L7440-255

(b)

Fig. 4-51. Photomicrographs of representative particles from fuel rod 7161-004-08-13 irradiated in capsule P13R (position 2C) to a fast neutron fluence of $12.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1045°C . (a) Bright field and (b) polarized light.

L7440-113

(a)

L7440-114

(b)

Fig. 4-52. Photomicrographs of representative particles from fuel rod 7161-004-19-7 irradiated in capsule P13R (position 5D) to a fast neutron fluence of $8.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1225°C. (a) Bright field and (b) polarized light.

Fig. 4-53. Photomicrographs of representative particles from fuel rod 7161-004-20-6 irradiated in capsule P13R (position 5E) to a fast neutron fluence of $7.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1280°C. (a) Bright field and (b) polarized light.

L7440-879

(a)

L7440-880

(b)

Fig. 4-54. Photomicrographs of representative particles from fuel rod 7161-004-01-7 irradiated in capsule P13S (position 1A) to a fast neutron fluence of $7.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. This rod was thermal cycled to high temperatures twenty-four times during irradiation. (a) Bright field and (b) polarized light.

L7440-620

(a)

L7440-621

(b)

Fig. 4-55. Photomicrographs of representative particles from fuel rod 7161-004-12-5 irradiated in capsule P13S (position 2B) to a fast neutron fluence of $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1010°C. (a) Bright field and (b) polarized light.

L7440-536

(a)

L7440-537

(b)

Fig. 4-56. Photomicrographs of representative particles from fuel rod 7161-004-13-6 irradiated in capsule P13S (position 2C) to a fast neutron fluence of $11.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 975°C . (a) Bright field and (b) polarized light.

L7440-1068

(a)

L7440-1069

(b)

Fig. 4-57. Photomicrographs of representative particles from fuel rod 7161-004-16-6 irradiated in capsule P13S (position 5A) to a fast neutron fluence of $9.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1375°C. (a) Bright field and (b) polarized light.

L7440-209

(a)

L7440-210

(b)

Fig. 4-58. Photomicrographs of representative particles from fuel rod 7161-004-19-5 irradiated in capsule P13S (position 5D) to a fast neutron fluence of $7.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1325°C. (a) Bright field and (b) polarized light.

L7440-187

(a)

L7440-188

(b)

Fig. 4-59. Photomicrographs of representative particles from fuel rod 7161-004-20-5 irradiated in capsule P13S (position 5E) to a fast neutron fluence of $7.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1350°C. (a) Bright field and (b) polarized light.

L7440-496

(a)

L7440-497

(b)

Fig. 4-60. Photomicrographs of representative particles from fuel rod 7161-004-28-13 irradiated in capsule P13S (position 6C) to a fast neutron fluence of $4.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1050°C . (a) Bright field and (b) polarized light.

L7440-553

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3400-3

(b)

Fig. 4-61. Representative photomicrographs of fuel rod 7161-004-03-5 after irradiation in capsule P13R (position 1C) to a fast neutron fluence of $8.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR. After irradiation this rod was heated to 1600°C in TRIGA for fission gas release measurements. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-52

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3398-2

(b)

Fig. 4-62. Representative photomicrographs of fuel rod 7161-004-06-5 after irradiation in capsule P13R (position 2A) to a fast neutron fluence of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-300

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3400-4

(b)

Fig. 4-63. Representative photomicrographs of fuel rod 7161-004-07-5 after irradiation in capsule P13R (position 2B) to a fast neutron fluence of $12.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1055°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-272

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3399-4

(b)

Fig. 4-64. Representative photomicrographs of fuel rod 7161-004-08-13 after irradiation in capsule P13R (position 2C) to a fast neutron fluence of $12.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1045°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-1032

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3401-1

(b)

Fig. 4-65. Representative photomicrographs of fuel rod 7161-004-16-5 after irradiation in capsule P13R (position 5A) to a fast neutron fluence of $9.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. After irradiation this rod was heated to 1600°C in TRIGA for fission gas release measurements. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-135

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3399-5

(b)

Fig. 4-66. Representative photomicrographs of fuel rod 7161-004-19-7 after irradiation in capsule P13R (position 5D) to a fast neutron fluence of $8.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1225°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-20

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3398-5

(b)

Fig. 4-67. Representative photomicrographs of fuel rod 7161-004-20-6 after irradiation in capsule P13R (position 5E) to a fast neutron fluence of $7.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1280°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-161

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3398-3

(b)

Fig. 4-68. Representative photomicrographs of fuel rod 7161-004-23-5 after irradiation in capsule P13R (position 6C) to a fast neutron fluence of $4.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1005°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-843

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3401-2

(b)

Fig. 4-69. Representative photomicrographs of fuel rod 7161-004-01-7 after irradiation in capsule P13S (position 1A) to a fast neutron fluence of $7.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. This rod was thermal cycled to high temperatures 24 times during irradiation. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-581

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3400-2

(b)

Fig. 4-70. Representative photomicrographs of fuel rod 7161-004-12-5 after irradiation in capsule P13S (position 2B) to a fast neutron fluence of $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1010°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-523

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3400-5

(b)

Fig. 4-71. Representative photomicrographs of fuel rod 7161-004-13-6 after irradiation in capsule P13S (position 2C) to a fast neutron fluence of $11.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 975°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-94

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3398-4

(b)

Fig. 4-72. Representative photomicrographs of fuel rod 7161-004-15-5 after irradiation in capsule P13S (position 2E) to a fast neutron fluence of $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 960°C . (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-1072

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3398-1

(b)

Fig. 4-73. Representative photomicrographs of fuel rod 7161-004-16-6 after irradiation in capsule P13S (position 5A) to a fast neutron fluence of $9.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1375°C . (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-463

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3399-3

(b)

Fig. 4-74. Representative photomicrographs of fuel rod 7161-004-18-5 after irradiation in capsule P13S (position 5C) to a fast neutron fluence of $8.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1335°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-218

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3399-1

(b)

Fig. 4-75. Representative photomicrographs of fuel rod 7161-004-19-5 after irradiation in capsule P13S (position 5D) to a fast neutron fluence of $7.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. After irradiation this rod was heated to 1500°C in TRIGA for fission gas release measurements. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-202

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3399-2

(b)

Fig. 4-76. Representative photomicrographs of fuel rod 7161-004-20-5 after irradiation in capsule P13S (position 5E) to a fast neutron fluence of $7.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1350°C . (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-499

(a)

FUEL ROD
SURFACE

FUEL ROD
CENTERLINE

75K3400-1

(b)

Fig. 4-77. Representative photomicrographs of fuel rod 7161-004-28-13 after irradiation in capsule P13S (position 6C) to a fast neutron fluence of $4.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} at 1050°C. (a) Typical appearance of matrix and (b) composite showing radial cross section.

L7440-833

(a)

L7440-834

(b)

Fig. 4-78. Representative photomicrographs of TRISO UC₂ particles (6151-00-035) irradiated in capsule P13S (position 4-12) to a fast neutron fluence of $10.7 \times 10^2 \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 71% FIMA at 975°C. (a) Bright field and (b) polarized light.

L7440-800

(a)

L7440-801

(b)

Fig. 4-79. Representative photomicrographs of TRISO (8Th,1U)O₂ particles (6155-01-020) irradiated in capsule P13S (position 4-4) to a fast neutron fluence of $11.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 11% FIMA at 1090°C. (a) Bright field and (b) polarized light.

L7440-816

(a)

L7440-817

(b)

Fig. 4-80. Representative photomicrographs of TRISO (Th,U)O₂ particles (5466-37) irradiated in capsule P13S (position 4-8) to a fast neutron fluence of $10.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 38% FIMA at 1090°C. (a) Bright field and (b) polarized light.

L7440-881

L7440-885

L7440-857

Fig. 4-81. Photomicrographs of fissile and fertile particles from fuel rod 7161-004-01-7 that failed during irradiation in capsule P13S (position 1A), which was thermal cycled to high temperatures 24 times during irradiation. Both the fissile (6151-00-045) and fertile (6542-20-035) particles had severely faceted coatings.

L7440-538

(a)

L7440-539

(b)

Fig. 4-82. Photomicrographs showing a failed fissile particle (6151-00-035) from fuel rod 7161-004-035 irradiated in capsule P13R (position 1C) to a fast neutron fluence of $8.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 73% FIMA. After irradiation the rod was heated in the TRIGA facility to 1600°C. (a) Bright field and (b) polarized light.

L7440-599

(a)

L7440-600

(b)

Fig. 4-83. Photomicrographs showing OPyC coating failure of fissile particle in fuel rod 7161-004-18-5 that was irradiated in capsule P13S (position 5C) to a fast neutron fluence of $8.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR at 1335°C . (a) Bright field and (b) polarized light.

L7440-356

L7440-358

Fig. 4-84. Photomicrographs showing the appearance of a typical cracked SiC coating in a TRISO coated particle. A significant fraction of the cracked SiC coatings observed was an artifact of sample preparation.

L7440-385

Fig. 4-85. Photomicrograph showing a small laminar carbon inclusion in SiC coating of a TRISO UC₂ particle irradiated in fuel rod 7161-004-19-7 in capsule P13R (position 5D)

Fig. 4-86. Photomicrographs showing an example of IPyC coating debonding that occurred in a TRISO UC₂ particle during irradiation to a fast neutron fluence of $8.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}

Fig. 4-87. Photomicrographs showing a carbon deposit in the gap between the buffer and IPyC coating of a TRISO UC₂ particle from fuel rod 7161-004-19-7 irradiated in capsule P13R (position 5D) at 1225°C. (a) Bright field and (b) polarized light.

L7440-31

L7440-32

Fig. 4-88. Photomicrographs showing an example of a cracked buffer coating that occurred in a TRISO UC_2 particle during irradiation to a fast neutron fluence of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}

L7440-719

(a)

L7440-720

(b)

Fig. 4-89. Photomicrographs showing a TRISO UC_2 particle from batch 6151-04-015-2 in which the structural coatings failed during irradiation leaving the buffer coating intact on the fuel kernel. (a) Bright field and (b) polarized light.

L7440-444

(a)

L7440-445

(b)

L7440-446

(c)

L7440-447

(d)

Fig. 4-90. Photomicrographs showing an example of UC_2 fuel kernel migration that was observed in fuel rod 7161-004-18-5 irradiated in capsule P13S (position 5C) at $1335^\circ C$. The kernel microstructure appeared homogeneous except for small flakes of light-colored material dispersed evenly throughout. (a,c) Bright field and (b,d) polarized light.

L7440-291

L7440-293

Fig. 4-91. Photomicrographs showing the microstructure of a UC_2 fuel kernel irradiated in fuel rod 7161-004-07-5 at $1055^\circ C$. The kernel apparently segregated into a fuel phase and regions of rejected carbon.

L7440-147

L7440-158

Fig. 4-92. Photomicrographs showing the microstructure of a UC_2 fuel kernel irradiated in fuel rod 7161-004-23-5 at $1005^\circ C$. The kernel was relatively spherical and contained large fission gas bubbles.

L7440-1063

(a)

L7440-1064

(b)

L7440-1073

(c)

L7440-1074

(d)

Fig. 4-93. Photomicrographs showing an example of $(1\text{Th},1\text{U})\text{O}_2$ kernel migration that was observed in fuel rod 7161-004-16-6 irradiated in capsule P13S (position 5A) at 1375°C . The kernel was relatively spherical and contained large fission gas bubbles. (a,c) Bright field and (b,d) polarized light.

L7440-416

(a)

L7440-417

(b)

L7440-418

(c)

Fig. 4-94. Photomicrographs showing metallic fission product attack of the SiC coating in a TRISO UC₂ particle irradiated in fuel rod 7161-004-18-5 at an average centerline temperature of 1335°C. (a,c) Bright field and (b) polarized light.

Fig. 4-95. Photomicrographs of BISO ThO₂ particles (4252-06-010-1) with an OPyC coating deposited from propylene. The particle batch was irradiated in capsule P13S (position 3-1) to a fast neutron fluence of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.3% FIMA at 925°C. (a) Bright field and (b) polarized light.

L7440-769

(a)

L7440-770

(b)

Fig. 4-96. Photomicrographs of BISO ThO₂ particles (6542-23-025-1) with an OPyC coating deposited from mixed gas. The particle batch was irradiated in capsule P13S (position 3-7) to a fast neutron fluence of $11.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.2% FIMA at 955°C. (a) Bright field and (b) polarized light.

L7440-697

(a)

L7440-698

(b)

Fig. 4-97. Photomicrographs of severely faceted BISO ThO_2 particles (6542-20-035-2) after irradiation in capsule P13R (position 4-8) to a fast neutron fluence of $11.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.3% FIMA at 1000°C. (a) Bright field and (b) polarized light.

L7440-47

(a)

L7440-48

(b)

Fig. 4-98. Photomicrographs of a severely faceted BISO ThO₂ particle (6542-20-035) from fuel rod 7161-004-06-5 that survived irradiation in capsule P13R (position 2A) to a fast neutron exposure of $11.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 5.5% FIMA at 1050°C. (a) Bright field and (b) polarized light.

L7440-735

(a)

L7440-736

(b)

Fig. 4-99. Photomicrographs of BISO ThO_2 particles with nonround fuel kernels (6542-25-015-1) after irradiation in capsule P13R (position 4-11) to a fast neutron fluence of $10.9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.1% FIMA at 940°C. (a) Bright field and (b) polarized light.

L7440-574

(a).

L7440-575

(b)

Fig. 4-100. Photomicrographs showing irradiation-induced failure in a BISO ThO₂ particle (6542-09-010) with a high-density (1.93 g/cm³) OPyC coating. The particle was irradiated in fuel rod 7161-004-12-5 of capsule P13S (position 2B) to a fast neutron fluence of 11.7×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} and burnup of 4.6% FIMA at 1010°C. (a) Bright field and (b) polarized light.

L7440-119

(a)

L7440-120

(b)

Fig. 4-101. Photomicrographs showing irradiation-induced failure in a BISO ThO₂ particle (6542-19-015) with a low-density (1.78 g/cm³) OPyC coating. The particle was irradiated in fuel rod 7161-004-19-7 of capsule P13R (position 5D) to a fast neutron fluence of 8.2×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} and burnup of 2.4% FIMA at 1225°C. (a) Bright field and (b) polarized light.

L7440-760

OPyC COATING
MICROCRACKS

L7440-762

Fig. 4-102. Photomicrographs showing microcracks in OPyC coating of BISO ThO₂ particle (4252-06-010-1) irradiated in capsule P13S (position 3-1)

L7440-705

OPyC COATING
MICROCRACKS

L7440-707

Fig. 4-103. Photomicrographs showing microcracks in OPyC coating of BISO ThO₂ particle (6542-20-035-2) irradiated in capsule P13R (position 4-8)

L7440-89

(a)

L7440-90

(b)

Fig. 4-104. Photomicrographs of a BISO ThO₂ particle showing numerous small gaps formed at the buffer/OPyC coating interface where the seal coat cracked and densified during irradiation. (a) Bright field and (b) polarized light.

L7440-654

L7440-650

Fig. 4-105. Photomicrographs of BISO ThO₂ particles (6542-18-015-1) with no seal coat between the buffer and OPyC coatings. The particle batch was irradiated in capsule P13R (position 4-3) to a fast neutron fluence of $11.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.4% FIMA at 1000°C.

L7440-281

(a)

L7440-282

(b)

Fig. 4-106. Photomicrographs of a BISO ThO_2 particle where the buffer coating cracked and densified while the OPyC coating remained intact. (a) Bright field and (b) polarized light.

PARTIAL
CONVERSION
OF ThO_2 TO ThC_2

PROPAGATION OF
CRACK INTO OPyC
COATING

L7440-788

Fig. 4-107. Photomicrograph of a BISO ThO_2 particle (6542-12-025) which shows the propagation of a buffer coating crack into the adjacent OPyC coating and the partial conversion of a localized region of the fuel kernel to carbide. The presence of a seal coat on this particle batch did not prevent propagation of the crack from the buffer coating into the OPyC coating.

L7440-784

(a)

L7440-785

(b)

Fig. 4-108. Photomicrographs of BISO ThO₂ particles (6542-12-025-2) with the buffer coating applied with N₂ carrier gas. The sample was irradiated in capsule P13S (position 4-12) to a fast neutron fluence of $10.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 3.6% FIMA at 830°C. (a) Bright field and (b) polarized light.

L7440-808

(a)

L7440-809

(b)

Fig. 4-109. Photomicrographs of TRISO ThO_2 particles (6252-00-025-2) irradiated in capsule P13S (position 3-12) to a fast neutron fluence of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR and burnup of 4.1% FIMA at 860°C . (a) Bright field and (b) polarized light.

L7440-822

REACTION ZONE
IN SiC COATING

L7440-824

Fig. 4-110. Photomicrographs showing reaction zone observed in TRISO ThO_2 particle (6252-00-025-2) irradiated in capsule P13S (position 3-12) to a fast neutron fluence of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} and burnup of 4.1% FIMA at 860°C

L7440-176

Fig. 4-111. Photomicrograph of a defective BISO ThO₂ particle in fuel rod 7161-004-20-5. Matrix material is bonded to the buffer coating in the region where the OPyC coating is missing. During irradiation the coating broke and the fuel kernel converted to carbide and subsequently hydrolyzed.

L7440-374

(a)

L7440-375

(b)

L7440-376

(c)

Fig. 4-112. Photomicrographs showing the microstructure of a ThO_2 kernel from particle batch 6542-01-010 irradiated in fuel rod 7161-004-08-13 of capsule P13R (position 2C) to a fast neutron fluence of $12.2 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) HTGR and burnup of 5.6% FIMA at 1045°C . The formation of pores has occurred in the outer region of the fuel kernel. (a,c) Bright field and (b) polarized light.

L7440-378

(a)

L7440-379

(b)

L7440-380

(c)

Fig. 4-113. Photomicrographs showing the microstructure of a ThO_2 kernel from particle batch 6542-01-010 irradiated in fuel rod 7161-004-08-13 of capsule P13R (position 2C). Grain boundary thickening occurred on the hot side of the fuel kernel. (a,c) Bright field and (b) polarized light.

L7440-269

(a)

L7440-270

(b)

Fig. 4-114. Photomicrographs of TRISO inert particle (6351-01-020) irradiated in fuel rod 7161-004-08-13 of capsule P13R (position 2C) to a fast neutron fluence of 12.2×10^{21} n/cm² ($E > 0.18$ MeV) HTGR at 1045°C. (a) Bright field and (b) polarized light.

Fig. 4-115. Postirradiation fission gas release versus unirradiated OPyC coating density (density gradient column) for intact BISO ThO₂ particles

Fig. 4-116. Postirradiation fission gas release versus unirradiated OPyC coating density (density gradient column) for intact BISO ThO₂ particles. The particles were cleaned with diluted nitric acid and examined visually prior to fission gas release measurements.

Fig. 4-117. Amount of thorium leached from irradiated BISO ThO₂ particle samples as a function of hot leaching time. The OPyC coating density in g/cm³ and OPyC coating rate in μm/min (numbers in parentheses) are also shown for each batch.

L7440-923

(a)

L7440-990

(b)

Fig. 4-118. Photomicrographs showing the condition of irradiated BISO ThO₂ particles after acid leaching. (a) Batch 6542-21-016-1 with an OPyC coating density of 1.73 g/cm³ and a coating rate of 8.0 μm/min and (b) batch 6542-22-015-1 with an OPyC coating density of 1.80 g/cm³ and a coating rate of 4.0 μm/min.

L7440-957

(a)

L7440-976

(b)

Fig. 4-119. Photomicrographs showing the condition of irradiated BISO ThO₂ particles after acid leaching. (a) Batch 6542-24-015-1 with an OPyC coating density of 1.94 g/cm³ and a coating rate of 6.6 μm/min and (b) batch 6542-02-020-2 with an OPyC coating density of 1.91 g/cm³ and a coating rate of 8.5 μm/min.

L7440-951

(a)

L7440-911

(b)

Fig. 4-120. Photomicrographs showing condition of irradiated BISO ThO₂ particles after acid leaching. (a) Batch 6542-09-010-2 with an OPyC coating density of 1.93 g/cm³ and a coating rate of 5.0 μm/min and (b) batch 6542-01-020-2 with an OPyC coating density of 1.82 g/cm³ and a coating rate of 2.7 μm/min.

L7440-965

(a)

L7440-937

(b)

Fig. 4-121. Photomicrographs showing condition of irradiated BISO ThO_2 particles after acid leaching. (a) Batch 4252-06-018-1 with an OPyC coating density of 1.82 g/cm^3 and a coating rate of 5.8 μm/min and (b) batch 6542-01-010-1 with an OPyC coating density of 1.80 g/cm^3 and a coating rate of 10.0 μm/min .

S7440-457

(a)

S7440-458

(b)

Fig. 4-122. Photographs of fuel particles recovered from rod 7161-004-02-6 (P13R, 1B) after it was burned in air at 950°C for 6 hr

S7440-463

(a)

S7440-465

(b)

Fig. 4-123. Photographs showing ThO_2 kernels and SiC coated fissile particles recovered from fuel rod 7161-004-02-6 (P13R, 1B). The two particle types were segregated using a separatory funnel. (a) ThO_2 kernels and (b) SiC coated fissile particles.

L7440-1002

Fig. 4-124. Photomicrograph showing condition of TRISO coated fissile particle from fuel rod 7161-004-11-6 (P13S, 2A). After the fuel rod was burned, the remnants were boiled in an acid solution for 15 hr. No SiC coating attack or failure due to the acid leaching was observed.

HA-372

(a)

HA-364

(b)

Fig. 4-125. Photographs taken of microradiographs of fissile and fertile particles. (a) Fissile particles irradiated in capsule P13R (position 3-5) and (b) fertile particles irradiated in capsule P13S (position 4-9).

5. DISCUSSION

5.1. COATED PARTICLE IRRADIATION PERFORMANCE

5.1.1. Fissile Particles

5.1.1.1. Failure Analyses. Quantitative failure levels of the unbonded fissile particle samples were ascertained by visual examination (see Section 4.2.2) and by fission gas release measurements (see Section 4.4.2). The results of these examinations are summarized in Table 5-1. The agreement between the pressure vessel failure levels observed during the visual examination and those calculated from the fission gas release data was excellent for 11 of the particle batches. Three batches (6151-00-010, 6151-02-025, and 6151-00-025) exhibited a high fraction of OPyC coating failure. An accurate assessment of pressure vessel failure (failure of the OPyC and SiC coating) could not be made during the visual examination because fragments of the failed OPyC coatings remained attached to the SiC substrate. Fission gas release measurements revealed pressure vessel failure levels of 4.0, 6.5, and 4.3% for these particle batches. Low levels of OPyC coating and pressure vessel failure were observed in the other three particle batches (6151-00-045, 6151-00-046, and 4161-01-21); however, the samples exhibited high fission gas release. The samples were reexamined visually after the TRIGA measurements and no significant additional particle failure had occurred. The reason for the large discrepancy between the observed and calculated failure levels for these three particle batches is not apparent from the results of these examinations.

Fissile particle failure levels determined during the metallographic examination are reported in Table 4-35. The irradiation performance of fissile particles tested in the fuel rods was similar to the performance

of the unbonded particle samples. Failure levels determined by metallographic examination of the fuel rods were similar to the failure levels determined for the unbonded particle samples by visual examination and fission gas release measurements. There is a larger uncertainty in the metallography failure data, however, because only a small portion of the sample population was examined. Fission gas release measurements were made on each fuel rod. However, accurate fissile particle failure fractions could not be determined from the data because only a small fraction of the total fissions occurred in the fissile particles at end-of-life.

5.1.1.2. Influence of Product and Process Variables

5.1.1.2.1. OPyC Coating Rate, Anisotropy, and Density

The OPyC layers were deposited on the TRISO fissile particles using coating rates varying between 0.7 and 4.0 $\mu\text{m}/\text{min}$. This produced coatings having a range of anisotropy values. This series also included particles having different coating densities. The objective of these tests was to determine the influence of these properties on OPyC coating performance at peak HTGR fluences. This section is concerned primarily with coating rate and anisotropy; however, the interaction of density with these variables is mentioned briefly.

At the time the P13R and P13S fissile TRISO samples were fabricated, coating rate was considered the primary process variable controlling coating anisotropy. Since previous irradiation results had shown a good correlation between coating performance and coating rate (Refs. 5-1 and 5-2), this parameter was a key factor in selecting samples for this test series.

Data for the nine different UC_2 TRISO particle batches included in this study are given in Table 4-3 along with the results of the postirradiation examination. Each of these samples was irradiated in unbonded particle trays, and typically particles from each batch were tested in two or

three trays. The OPyC failure levels are plotted in Fig. 5-1 as a function of the batch mean coating rate. The data show that the failures occurred for coatings having apparent deposition rates of less than 2 $\mu\text{m}/\text{min}$. Also, onset of failure is shifted toward lower coating rates for the lower-density samples. These results are very consistent with data from earlier irradiation tests of fissile TRISO particles (Ref. 5-2), and indicate a dependence of OPyC failure on both coating rate and coating density. However, when care is taken to precisely measure PyC coating anisotropy values, there is a noticeable effect of density as well as coating rate (Fig. 5-2). Data presented in Fig. 5-2 were obtained using an optical anisotropy measuring apparatus developed by OSGAE (Osterreichischen Studien Gesellschaft für Atomenergie), Seibersdorf, Austria (Ref. 5-3). This instrument has greater accuracy and precision than most alternate methods used to characterize the particle coatings during their manufacture. It can be seen from Fig. 5-2 that at a given coating rate, the lower-density OPyC coatings have lower anisotropies. Plotting OPyC coating failure against BAF_o optical anisotropy values reveals that the initiation of failure occurs in batches having mean anisotropies around 1.02 and that failures increase as the batch mean BAF_o increases (Fig. 5-3). There also appears to be a trend for the low-density coatings to have higher failure levels at a given anisotropy. These two observations are in agreement with results from characterization studies of PyC irradiation behavior (Refs. 5-4 and 5-5) which show (1) coating stresses increase with increasing coating anisotropy and (2) higher-density pyrocarbons are stronger than lower-density ones.

These trends become even clearer when the particle performance data from the fuel rod tests are evaluated. These tests contained UC_2 TRISO particles having OPyC coatings with low to medium anisotropies and densities ranging between 1.76 and 1.95 g/cm^3 . In addition, most of the rods contained fissile size (600- μm diameter) TRISO inert particles which had low-density coatings that were moderately anisotropic ($\text{BAF}_o \sim 1.03$). OPyC failure levels were determined from metallographic examination (see Tables 4-35 and 4-37). The data have been plotted in Fig. 5-4. These plots show

the importance of temperature as well as fluence on the coating irradiation performance. These results also magnify the effects of anisotropy and density on TRISO OPyC coating survival. Comparing Figs. 5-4a and 5-4b illustrates the influence of anisotropy for coatings having nearly identical densities. The coatings represented in Fig. 5-4a had relatively high anisotropies for their low density. These coatings behaved well at low temperature; however, as the irradiation temperature increased to 1250°C and above, their performance was poor. A fissile TRISO sample having an OPyC coating with nearly identical density but lower anisotropy had better performance (Fig. 5-4b), although the extent of the OPyC failure is still considered excessive at the higher temperatures. The influence of density on performance is seen when Fig. 5-4c is compared with Figs. 5-4a and b. Particles having OPyC coating densities between 1.85 and 1.95 g/cm³ had no failures in samples irradiated to full fluence at 1250°C and approximately 2% failure at 1350°C. Since the samples irradiated at an average temperature of 1350°C had operated as high as 1600°C for a period of time, their performance was judged to be excellent.

In summary, the results from this series of unbonded particle and fuel rod tests have provided an excellent basis for setting the specification limit for the TRISO OPyC coating. Results from the unbonded particles irradiated at 1000°C illustrate the need to maintain low anisotropy values. Batch mean anisotropies of <1.03 BAF₀ values are required. These results are very consistent with tests conducted on fuel rods at similar temperatures. However, once the lowest practical anisotropy values have been achieved (i.e., the plateau in the coating rate - optical anisotropy curve, Fig. 5-2), it becomes necessary to restrict the range of allowable coating density to achieve satisfactory performance at higher temperatures. This conclusion is drawn from the high-temperature fuel rod tests, which show coatings having densities between 1.85 and 1.95 g/cm³ exhibited the most satisfactory performance. Coatings having densities below approximately 1.75 g/cm³ had excessive failure, even when the optical anisotropy values were very low (~1.02 BAF₀ units).

5.1.1.2.2. OPyC Coating Irradiation-Induced Densification

Postirradiation density measurements were made on OPyC coating fragments from nine fissile particle batches, as discussed in Section 4.10.1, and the data are summarized in Table 4-50. The postirradiation densities of OPyC coatings with initial densities in the range of 1.76 to 1.82 g/cm³ and 1.85 to 1.94 g/cm³ are plotted in Fig. 5-5 as a function of fast neutron fluence. Also shown in the figure are the densification curves obtained from previous measurements on OPyC coatings irradiated to lower fast neutron fluences (Ref. 5-2). The degree of irradiation-induced densification of OPyC coatings with initial densities in the range of 1.76 to 1.82 g/cm³ was in good agreement with the lower-fluence data. Postirradiation densities of the OPyC coatings having an initial density in the range of 1.85 to 1.94 g/cm³ were consistent with studies by Kaae (Ref. 5-6) which indicate that high-density, unrestrained PyC densifies a small amount initially and then expands during subsequent irradiation.

5.1.1.2.3. SiC Coating Performance

The SiC coating is the primary load-bearing member of the TRISO coating as well as an absolute barrier to certain metallic fission products. Low-density (<3.18 g/cm³) SiC is undesirable as a coating material because it can contain free silicon, which has a high vapor pressure and forms low-melting eutectic compounds. Also, low-density SiC shows a decrease in strength at temperatures above 1000°C, whereas higher-density (>3.18 g/cm³) SiC exhibits an increase in strength at these temperatures (Ref. 5-7). For these reasons, SiC coatings with densities ≥ 3.18 g/cm³ (>99% theoretical density) are specified for TRISO particles.

The SiC coating density of fissile particles tested in capsules P13R and P13S was ≥ 3.18 g/cm³. The irradiation performance of these coatings was good. Failure of the SiC coating in particles which suffered pressure vessel failure was attributed to other variables, such as faceted coatings, rather than to properties of the SiC coating.

Tensile stresses in the OPyC coating generated by irradiation-induced densification help maintain the SiC coating in compression. Failure of the OPyC coating during irradiation would therefore be expected to increase the probability of SiC coating failure. To evaluate the effect of OPyC coating failure on SiC coating performance, the failure levels in several TRISO coated UC₂ particle batches with coating parameters (except the OPyC coating) representative of LHTGR design requirements were analyzed. Particle batches that had a specific attribute which would be expected to increase the probability of pressure vessel failure, i.e., nonround kernels, faceted coatings, or interrupted SiC coatings, were not considered. The irradiation performance of these particle batches is summarized in Table 5-2. The same SiC substrate number for more than one particle indicates that one batch was split after deposition of the SiC coating and different OPyC coatings were applied. One batch that was split from each of the parent batches had the OPyC coating deposited at a very low coating rate in order to establish the specification limit for this parameter. These batches, 6151-02-025, 6151-01-015, and 6151-00-010, experienced OPyC coating failure levels of 46.3%, 100%, and 32.8%, respectively. Even though high levels of OPyC coating failure occurred, the total coating failure levels for these batches only ranged from 1.4 to 4.0%. These results indicate the conservative nature of the LHTGR reference fissile particle design.

During fabrication, deposition of the SiC may be inadvertently interrupted. To determine whether it was acceptable to interrupt the SiC coating process if the final product met specification, two batches (6151-03-015 and 4161-01-021) of TRISO UC₂ particles with interrupted SiC coatings were tested in capsule P13S. Results of the visual examination and fission gas release measurements performed on these two particle batches are summarized in Table 5-1. Batches 6151-03-015 and 4161-01-021 exhibited pressure vessel failure levels of 0.1 and 3.3%, respectively, as determined by visual examination after irradiation to fast neutron fluences of 11.8 and 11.7 x 10²¹ n/cm² (E > 0.18 MeV)_{HTGR}. The equivalent failure level of 26.9% determined from the fission gas release data for batch

4161-01-021 was anomalous. The high release could have resulted from contamination pickup in the hot cell during handling, from microcracks in the particles that were not observed during the visual examination, or from failure induced during the TRIGA measurement. Visual examination of the batch after the TRIGA measurement indicated significantly higher failure levels than were originally observed. Batch 4161-01-021 had a high metal density (thin buffer coating) and severely faceted coatings. These two variables probably contributed more to the high failure observed in this batch than the presence of an interrupted SiC coating. The low failure level exhibited by batch 6151-03-015 indicated that an interruption in the SiC coating deposition may not be detrimental to coated particle irradiation performance. However, these limited results do not justify use of interrupted SiC coatings.

5.1.1.2.4. IPyC Coating Performance

The function of the IPyC layer of the UC_2 TRISO particle is to protect the SiC layer from detrimental chemical reactions during manufacturing and irradiation. For manufacturing purposes a dense, impermeable coating is most desirable in order to prevent infiltration of gaseous by-products into the buffer during the SiC deposition step. Process studies have shown that the presence of Cl-compounds in the buffer can cause extensive fuel dispersion. Although quantitative data relating the effect of fuel dispersion on particle performance are not available, the occurrence of such dispersion is specified to be kept to a minimum during fuel production (Ref. 5-8).

The IPyC layer also appears to act as a sink for some of the more mobile fission products during irradiation (see Section 5.1.1.2). Its presence may slow down the migration of these fission products (primarily the rare earths) or deter them from reacting with the SiC layer. However, under the more severe irradiation conditions these fission products do alter the structure of the IPyC layer (for example, see Fig. 4-94), and in

some cases cause the IPyC layer to fail. Failure of the IPyC layer under these circumstances presumably should be independent of the physical properties of the PyC.

Results of metallographic examination of a large number of fissile particles tested in capsules P13R and P13S revealed that the majority of the IPyC layers survived. Only six of the 24 batches examined metallographically exhibited a significant fraction of cracked IPyC coatings (see Table 4-35). In no case was failure of the IPyC layer found to be detrimental to particle performance; i.e., no deleterious chemical reactions occurred as a result of the failure of this layer. A correlation of the failures with coating properties (density and anisotropy) indicated that the batches in which failure occurred had the most anisotropic coatings. These samples had optical anisotropy values (measurements made after the SiC and OPyC layers were deposited) of $>1.08 \text{ BAF}_0$. Thus, if other tests show it is necessary to have the IPyC layer retain its integrity during irradiation, this anisotropy value (measured after the SiC deposition) should be a reasonable limit. Results from these tests do not dictate a need for this layer to remain intact; therefore, manufacturing considerations probably should influence property specification limits for this coating.

5.1.1.2.5. Buffer Coating Performance

The purpose of a low-density buffer coating immediately adjacent to the fuel kernel is to attenuate fission recoils and to provide a void volume to accommodate fission gases and fuel kernel swelling. The thickness and density of buffer coatings required for TRISO particles irradiated to peak conditions are interrelated and are a function of the fuel density in the particle and the coating design.

TRISO coated UC_2 particles with buffer coating thicknesses ranging from 82 to 105 μm and densities ranging from 0.87 to 1.31 g/cm^3 were tested

in capsules P13R and P13S. During irradiation the buffer coatings decoupled from the IPyC coatings and densified around the fuel kernels. From 0 to 2.0% of the buffer coatings were cracked in the particles examined metallographically. The cracking was not a function of the buffer coating thickness or density. It occurred only in particles irradiated in the temperature range of 1000° to 1100°C where the fuel kernel had segregated into a two-phase structure as shown in Fig. 4-88. In some particles irradiated as unbonded particle samples, the structural coatings failed and the buffer coating remained on the fuel kernel as shown in Fig. 4-89. Results from capsules P13R and P13S and from previous irradiation experiments will be analyzed using a TRISO particle stress model to establish the buffer coating thickness and density limits for the LHTGR fuel specifications.

Less than 1.0% of the buffer coatings were cracked in the TRISO (1Th,1U)O₂ particles (5466-37) with 250-μm-diameter kernels. However, 88.0% of the buffer coatings were cracked in the TRISO (8Th,1U)O₂ particles (6155-01-020) with larger (500 μm) diameter kernels. Batch 6155-01-020 had a 5-μm-thick seal coat which allowed the buffer coating to decouple from the IPyC coating during irradiation and shrink onto the fuel kernels. Cracking of the buffer coating occurred as a result of buffer densification and fuel kernel swelling. In contrast, the TRISO ThO₂ particles with 500-μm-diameter kernels (batch 6252-00-025) exhibited buffer cracking in only 11.0% of the particles. This batch did not have a seal coat, and the buffer coatings remained bonded to the IPyC coating. Buffer cracking occurred only in the particles that debonded from the IPyC coating and shrank onto the fuel kernel. Buffer coating density may also have had an effect on the levels of cracking observed. Batch 6155-01-020 had a buffer density of 1.24 g/cm³, whereas batches 5466-37 and 6252-00-025 had buffer densities of 1.12 and 1.16 g/cm³, respectively. Higher-density buffer coatings may be more susceptible to irradiation-induced densification and cracking.

5.1.1.2.6. Faceted Particles

Severely faceted particles would be expected to have a higher probability of failure than more spherical particles because the smaller radius of curvature would enhance the stresses in the coating. Two batches (6151-00-045 and 6151-00-025) of fissile particles were tested to evaluate the effect of faceting on TRISO particle performance. In addition, one sample (6151-00-046) of severely misshaped particles segregated from batch 6151-00-045 was also irradiated. Particles from each of these three samples were tested as unbonded particle samples, and particles from batch 6151-00-045 also were tested in six fuel rods. Results from metallographic evaluation (Table 4-35) and from visual and fission gas release evaluations (Table 5-1) all show that these particles performed poorly.

Fission gas release measurements revealed that batches 6151-00-045, 6151-00-046, and 6151-00-025, tested as unbonded particle samples, experienced failure levels of 19.1, 14.0, and 6.5%, respectively. No failure was observed in batch 6151-00-045 when it was irradiated in fuel rod 7161-004-06-5 at 1050°C; however, this batch experienced 15.3% failure in rod 7161-004-01-7, which was thermal cycled up to 1650°C in-pile. These results indicate that severely faceted TRISO particles are more susceptible to pressure vessel failure during irradiation than similarly designed, non-faceted fuel particles (e.g., compare with results of batches 6151-09-025 and 6151-00-035). However, the effect of faceting on the irradiation performance of these particle batches could not be evaluated quantitatively because both particle batches had other attributes that would have been expected to contribute to the failure levels observed. Batch 6151-00-045 was non-conservatively designed since it had a buffer coating thickness of 85 μm , and batch 6151-00-025 had an anisotropic OPyC coating (deposited at a coating rate of 1.4 $\mu\text{m}/\text{min}$) which resulted in a high level of OPyC coating failure, independent of the pressure vessel failure observed in the particle batch.

5.1.1.3. Thermochemical Stability

5.1.1.3.1. Metallic Fission Product Interactions

Metallic fission products diffuse from UC_2 kernels during irradiation at high temperatures. In TRISO particles concentrations of metallic fission products build up in the IPyC coating on the cooler side of the fuel particles and subsequently react with the SiC coating at the SiC/IPyC coating interface. Reactions of this type have been observed in UC_2 TRISO particles irradiated in capsules P13L, P13M, P13N, and P13P. Electron microprobe analysis revealed the only heavy metals preferentially diffusing out of the fuel kernel to the SiC coating were the rare earth fission products cerium, neodymium, lanthanum, samarium, praseodymium, and europium (Ref. 5-2).

Slight degradation of the SiC coating by metallic fission products was observed in TRISO UC_2 particles irradiated in capsule P13R and P13S fuel rods at average temperatures above 1050°C. The extent of attack increased with increasing temperature. A photomicrograph of fission product attack in a UC_2 TRISO particle irradiated in fuel rod 7161-004-18-5 at an average center-line temperature of 1335°C is shown in Fig. 4-94. In this particle the buildup of metallic fission products at the SiC/IPyC coating interface was large enough to mechanically force the IPyC coating away from the SiC coating. A summary of the metallic fission product interactions observed in UC_2 TRISO particles during the metallographic examination (see Section 4.6.2) is given in Table 4-39. In particles where metallic fission products reacted with the SiC coating, maximum depths of penetration into the SiC coating generally ranged from 2 to 5 μm . No coating failure was observed as a result of metallic fission product interaction. Although concentrations of metallic fission products were observed in the IPyC coatings, no SiC coating reaction was observed in the UC_2 TRISO unbonded particle samples which were irradiated at lower temperatures than the fuel rods. No metallic fission product interactions were observed in the TRISO (Th,U) O_2 particles. This observation is consistent with previous results which indicated that in oxide fuel particles,

the rare earth metallic fission products are retained in the fuel kernel in the form of stable oxides.

Metallic fission product attack in TRISO UC_2 particles is dependent on the concentration of fission products in the particle or fission density, time, and temperature. The kinetics of this reaction are not well known. Consequently, extensive out-of-pile annealing studies are being conducted to evaluate the degradation of TRISO coated fissile particles by metallic fission products. In order to provide irradiation data that can be used in conjunction with out-of-pile postirradiation heating data to determine the kinetics of metallic fission product attack, a more quantitative assessment has been made of reactions observed in some of the particles (see Table 5-3). The depth of fission product penetration in the SiC coating of these particles was measured from high-magnification photomicrographs taken of localized regions of SiC attack. The location of each particle with respect to the fuel rod centerline was established. An average irradiation temperature and temperature gradient for each particle were calculated using the following equations:

$$T = T_{C.L.} - bx^2 \quad , \quad (5-1)$$

$$\frac{\Delta T}{\Delta x} = -2bx \quad , \quad (5-2)$$

$$b = +\frac{\Delta T(F.R.)}{r^2} \quad , \quad (5-3)$$

where

T = particle irradiation temperature ($^{\circ}C$)

$T_{C.L.}$ = average fuel rod centerline irradiation temperature ($^{\circ}C$)

x = distance of particle from fuel rod centerline (cm)

$\Delta T/\Delta x$ = temperature gradient in fuel rod at location of particle ($^{\circ}C/cm$)

r = radius of fuel rod (cm)

$\Delta T(F.R.)$ = time-averaged temperature drop from fuel rod centerline to surface

These temperatures and temperature gradients were calculated using time-averaged irradiation conditions and therefore only approximate the temperature history of the fuel particles.

5.1.1.3.2. Fuel Kernel Migration

During operation at high temperatures, carbide 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. The temperature and thermal gradient dependence of carbide fuel kernel migration and its impact on HTGR core thermal design have been discussed in Ref. 5-9.

Fuel kernel migration was observed in TRISO UC_2 and TRISO $(1Th, 1U)O_2$ particles irradiated in the high-temperature (cell 5) fuel rods of capsules P13R and P13S. A quantitative evaluation of kernel migration in six fuel rods was made, as discussed in Section 4.6.2, and the data are presented in Table 4-40.

Time-averaged particle temperatures and temperature gradients were calculated using Eqs. 5-1, 5-2, and 5-3 for the particles noted in Table 4-40 that exhibited kernel migration. Kernel migration coefficients calculated for the UC_2 TRISO particles were in good agreement with migration data obtained for UC_2 TRISO particles irradiated in capsule P13N (Ref. 5-9). The use of time-averaged irradiation temperatures provides only approximate kernel migration coefficients because the rate of migration follows an Arrhenius type of dependence rather than a linear temperature dependence. It is concluded from a comparison of the in-pile data with kernel migration coefficients obtained from out-of-pile annealing studies on unirradiated and irradiated fuel particles that the rate of kernel migration is not significantly enhanced during irradiation.

To compare fuel kernel migration rates observed in capsules P13R and P13S with rates predicted for LHTGR operation, fuel performance subroutines which are similar to those contained in the SURVEY code were incorporated into the thermal analysis CAPTEM code. The SURVEY code is used in LHTGR core thermal design to predict the extent of kernel migration in an LHTGR core at any time during operation. Using the UC_2 kernel migration coefficient design curve, the CAPTEM code predicted the fuel kernel migration distance in 10 equally divided annular rings of a fuel rod for the 5, 50 (nominal), and 95% confidence levels on a daily basis over the operating life of the capsules. The total predicted migration distance at the end of irradiation as a function of the distance from the fuel rod centerline was obtained for each rod. The predicted migration distances for kernels in fuel rods 7161-004-18-5, 7161-004-19-7, and 7161-004-19-5 are plotted in Figs. 5-6, 5-7, and 5-8, respectively, as a function of radial location in the fuel rod. The solid line represents the predicted migration distance using the mean of the 90% confidence level, and the dashed lines represent the predicted migration distance using the range at the 90% confidence level of the kernel migration coefficient design curve. Also plotted are the measured migration distances from Table 4-40.

There was good agreement between the measured and predicted kernel migration distances. The scatter in the observed migration distances results from a number of factors, as discussed in Ref. 5-9. The migration distance of 6.0 μm measured for one particle in fuel rod 7161-004-19-7 (Fig. 5-7) is significantly greater than the predicted value. This occurred because the particle was located at the end of the fuel rod on the centerline. The direction of migration was up the fuel rod centerline, which indicates a steep axial thermal gradient in the rod. The CAPTEM code used in thermal analysis takes into account two-dimensional (radial and axial) effects in calculating the radial temperature distribution in the fuel rod; however, axial temperature distributions are not calculated. Consequently, migration distances based on a radial temperature distribution were calculated and the severe axial temperature gradient was not taken into account. Grafoil discs were placed between the fuel rods;

consequently, the end thermal effects were more severe than would be expected for fuel rods in the LHTGR. Fuel kernel migration was not detected in fuel rod 7161-004-28-3 (Table 4-40). This observation is consistent with the maximum migration distance of 0.03 μm predicted by the CAPTEM code.

Kernel migration was also observed in TRISO (Th,U)O₂ particles, and the data are reported in Table 4-40. Design kernel migration coefficient curves were not available for (Th,U)O₂ particles. Therefore, in-pile migration distances were not calculated. However, the observed (Th,U)O₂ kernel migration rates were greater than the UC₂ migration rates. This result was expected based on available migration data.

5.1.2. Fertile Particles

5.1.2.1. Failure Analyses. Quantitative failure levels of the unbonded fertile particle samples were ascertained by visual examination (see Section 4.2.2) and by fission gas release measurements (see Section 4.4.2). The results of these examinations are summarized in Table 5-4. The in-service failure levels determined from the fission gas release data were generally higher than the total coating failure levels determined during the visual examination. This discrepancy was attributed to the partial permeability of lower-density OPyC coatings to gaseous fission products, as discussed in Sections 4.7 and 5.1.2.2.3. As a result of this phenomenon, the failure analysis performed by visual examination provided the more accurate measure of pressure vessel failure in the unbonded fertile particle samples.

Failure levels of fertile particles tested in fuel rods were determined by metallography and from fission gas release data. The results of these analyses are presented in Table 5-5. There is a large uncertainty in the metallography failure data because only a small sample population was examined. Metallography data on fuel rods 7161-004-03-5 and 7161-004-16-5 are not presented in Table 5-5 because the examination was performed after the fuel rods were heated to 1600°C in the TRIGA facility. Evaluation of fuel failure levels in these rods is discussed in Section 5.1.3. Maximum

fertile particle failure levels were calculated using fission gas release data by assuming all the fission gas released during the TRIGA measurement came from failed fertile particles. This is a good approximation since more than 80% of the total fissions occurred in the fertile particles during the TRIGA measurement, with the exception of fuel specimens from cell 6, and metallographic examination revealed low fissile particle failure levels (Table 4-36).

There was good agreement between the failure levels observed during the metallographic examination and the levels calculated from the fission gas release data. In 11 of the 15 rods examined metallographically, the failure level determined from the R/B data was within the 95% confidence interval for failure observed metallographically. Fuel rod 7161-004-20-5 was the only rod for which the maximum failure level calculated from the R/B data was greater than the upper 95% confidence level calculated on the basis of metallographic observations.

Comparison of failure levels revealed that the irradiation performance of fertile particles tested in fuel rods was similar to the performance of the unbonded particle samples. However, there was a discrepancy in the failure levels determined from the R/B data for two particle batches. Failure levels of 6.0 and 12.0% were determined from R/B data for batches 6542-01-010 and 6542-19-015, respectively, tested as unbonded particle samples. However, in 12 fuel rods maximum failure levels ranging from only 0 to 2.1% were determined for these batches based on R/B data. The high fission gas release measured for the unbonded particle samples was attributed to the permeability of the OPyC coatings since (1) the coatings were of relatively low density, (2) they had high coating rates, and (3) low failure levels were observed during the visual examination. The permeability of OPyC coatings is discussed further in Section 5.1.2.2.2.

Based on these results the fission gas release measurements gave a fairly accurate measure of pressure vessel failure that occurred in the fertile particles tested in the bonded fuel rods. However, there is a

level of uncertainty of at least a factor of two in the reported failure fractions because of the assumptions made in calculating the R/B failure levels and the effects of OPyC coating permeability on gaseous fission product release.

5.1.2.2. Influence of Property and Process Variables

5.1.2.2.1. OPyC Coating Rate, Anisotropy, and Density

An analysis similar to that presented in Section 5.1.1.2.1 for the behavior of fissile particle OPyC coatings was performed for the ThO_2 BISO fertile particles. Thirteen different batches were included in this evaluation (see Tables 4-4 and 4-5). All of these batches had kernel diameters and buffer and OPyC coating thicknesses within the limits of the current HTGR Fuel Product Specification (Ref. 5-8). OPyC coating rates ranged between 2.2 and 10.0 $\mu\text{m}/\text{min}$ and densities ranged between 1.73 and 1.94 g/cm^3 . Batch mean BAF_O values varied from about 1.02 to 1.04.

A plot of OPyC optical anisotropy as a function of coating rate (Fig. 5-9) did not reveal as good a correlation as was found for fissile particle outer coatings (see Fig. 5-2). It should be noted that within some batches, significant variation in BAF_O values was obtained. For example, in batch 6542-09-010 the maximum and minimum values were 1.06 and 1.02, respectively; the batch mean BAF_O was 1.04. Since this batch apparently had a moderately high coating rate, 5.0 $\mu\text{m}/\text{min}$, one would expect the BAF_O values to be much lower and have an insignificant amount of scatter.

The reason for the poor correlation between anisotropy and coating rate is not known at this time. Since these coatings are much thicker than the OPyC layers on the TRISO fissile particles, one might suspect gradients in density and/or anisotropy to be the cause for part of the scatter. Also, a contributing factor could be related to the fact that these measurements were not made at the same time as those for the fissile particles. Since

these measurements require care and precision, part of the scatter could be introduced by different techniques employed in gathering the data. Both of the above possibilities will be investigated further.

Plotting OPyC failure levels as a function of apparent coating rate and optical anisotropy (BAF_o) reveals correlations much like those for TRISO fissile particle OPyC coatings. In the case of coating rate (Fig. 5-10a), onset of failure appears to begin around $4 \mu\text{m}/\text{min}$, at least for the low-density samples. The correlation between anisotropy and failure relies heavily upon samples 6542-01-020 and 6542-02-030, since these were the only two batches which exhibited any significant amounts of failure (Fig. 5-10b). Although the trends parallel those observed for fissile particles, care must be taken not to overemphasize these results in view of the apparent scatter observed in the optical anisotropy measurements. It is concluded from these unbonded particle tests, however, that the 1.85 to $1.95 \text{ g}/\text{cm}^3$ OPyC coatings should perform better than low-density ones, for much the same reasons given earlier in connection with fissile particle coatings.

Fertile particle failure levels in fuel rods calculated from TRIGA fission gas measurements (Table 5-5) are plotted as a function of temperature and fluence in Fig. 5-11. Fuel burnup is not considered here, although the fertile kernel burnup generally increases exponentially with fluence. Several trends can be noted from these plots. First, OPyC coating failure increased with both fluence (burnup) and temperature. At the higher temperatures ($>1250^\circ\text{C}$), there is not a significant difference in coating performance for particles with different densities and different coating rates (anisotropy). However, at 1000°C there is a notable difference in OPyC coatings. That is, higher-density coatings perform better than lower-density ones (compare Figs. 5-11a and 5-11b), and for a given density coatings having higher coating rates (and presumably lower anisotropies) perform better than those having lower coating rates (compare Figs. 5-11b and 5-11c). As can be seen, the OPyC performance results from

these fuel rods yield conclusions similar to those drawn from the constant-temperature unbonded particle tests. However, in reviewing these data it should become obvious that this simplistic, straightforward treatment of OPyC coating performance on BISO particles will not yield results as consistent as those obtained for TRISO particles. This is because stress levels in OPyC coatings on TRISO particles depend primarily on the irradiation temperature and fluence. For TRISO particles, burnup need not be considered a variable because the rigid SiC layer shields the OPyC from any stresses resulting from fission gas pressure. BISO OPyC coating failure, on the other hand, depends on fuel burnup, buffer coating properties, and OPyC thickness, as well as the irradiation temperature and fluence. Rigorous interpretation of coating anisotropy and density requirements can be performed only with the aid of particle stress models which account for effects of these coating properties. Such models are currently being developed. However, in the absence of these analytical tools a recommendation of appropriate property limits based on this analysis alone must be made. To reiterate previous statements, OPyC coatings on BISO particles exhibiting optimum performance are those having densities between 1.85 and 1.95 g/cm³ and anisotropies below 1.04 BAF₀ units (i.e., coating rates >4 μ/min).

5.1.2.2.2. OPyC Permeability

In addition to maintaining structural integrity during irradiation to peak LHTGR fast neutron exposures, BISO OPyC coatings must remain impermeable to gaseous fission products. A number of BISO fertile particle batches were irradiated to fast neutron exposures up to 12.1×10^{21} n/cm² (E > 0.18 MeV)_{HTGR} without any apparent loss of structural integrity of the OPyC coatings. However, these particle batches exhibited varying levels of fission gas release and loss of thorium when leached in acid solutions (see Section 4.7). Fission gas release levels and thorium leachability appeared to increase with decreasing OPyC coating density (unirradiated). These results indicate that the OPyC coatings were permeable to gases and

liquid solutions and that the degree of permeability was not altered significantly by irradiation-induced densification of the OPyC coatings.

Pyrolytic carbons contain several density defects, including large inter-growth feature pores, intercrystalline microporosity, and low-density carbon inclusions (Ref. 5-10). Any surface-connected porosity available to gases or liquids will undoubtedly be related to the fraction of large inter-growth feature pores rather than the smaller intercrystalline pores. The amount of surface available porosity can be ascertained by evaluating the differences between the PyC density measured using mercury porosimetry and a sink-float technique. Mercury does not wet the carbon and at low pressure does not tend to penetrate the porosity, while the liquid solution used in the sink-float measurement does. Thus, the difference between the two densities will increase with increasing available porosity (Ref. 5-10).

The unirradiated OPyC coating densities of the fertile particles were determined using a sink-float technique. The bulk density of the OPyC coatings was subsequently remeasured by mercury porosimetry, and the results are given in Table 5-6. The amount of surface-connected porosity or the difference in density ($\rho_{SF} - \rho_{bulk}$) determined for the unirradiated OPyC coatings is plotted in Fig. 5-12 as a function of coating rate. The amount of surface-connected porosity increases with increasing coating rate for the propylene and acetylene-propylene OPyC coatings, which is consistent with previous observations (Ref. 5-10).

The amount of inter-growth feature porosity in PyCs is not significantly altered by irradiation-induced densification (Ref. 5-6). Consequently, the levels of fission gas released from the fertile particles should be a function of the amount of surface-connected porosity in the unirradiated coatings. However, when the fission gas release data were analyzed as a function of the calculated OPyC porosity, a clear relationship between the two sets of data could not be established. The lack of correlation might be attributed to an increase in porosity of the more anisotropic

OPyC coatings during irradiation. During irradiation to high fast neutron fluences, anisotropic PyCs exhibit a turnaround in the densification and begin to expand. This expansion or decrease in density is attributed to tearing and cracking of the microstructural growth features of the PyC (Ref. 5-6). The development of these cracks increases the porosity in the PyC and consequently would result in an increase in fission gas release regardless of the initial level of surface-connected porosity in the OPyC coating. The permeability of the BISO coatings evaluated in this study might be attributed to a combination of the initial surface-connected porosity and an increase in porosity due to microstructural cracking after irradiation to the severe fast neutron fluences ranging from 10.7 to $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

Four of the fertile particle batches which exhibited permeable OPyC coatings (Table 5-6) were also tested in bonded fuel rods. Fuel rods containing batches 6542-01-020 and 6542-02-020 exhibited levels of fission gas release similar to those of the unbonded samples after irradiation to nominally the same conditions. However, rods which contained batches 6542-09-010 and 6542-01-010 exhibited two orders of magnitude lower fission gas release than the unbonded particle samples. The reason for the better retention of fission gas in the particles irradiated in the bonded fuel rods is not clear. It is possible that impregnation of the surface of the OPyC coating with pitch during fuel rod fabrication and the subsequent carbonization reduce the permeability of the OPyC coating.

5.1.2.2.3. Mixed-Gas OPyC Coatings

The OPyC coatings of Fort St. Vrain TRISO fuel particles and of BISO ThO₂ particles irradiated in previous evaluation tests were deposited from propylene. However, to fabricate coated particles in a 240-mm-diameter coater, it is necessary to deposit OPyC coatings from acetylene and propylene mixed in the proper proportions to avoid severe thermal transients during the coating operation. Data have previously been obtained on mixed-gas OPyC coatings irradiated to moderate fast neutron fluences. BISO

coated carbide fuel particles with OPyC coatings deposited from mixed gas exhibited good irradiation performance in capsules F-27 and F-28 (Ref. 5-11) and in Peach Bottom Core 2 (Ref. 5-12).

Five BISO coated fertile particle batches with OPyC coatings deposited from mixed gas were tested as unbonded particle samples in capsule P13S to evaluate the performance of these coatings upon irradiation to peak LHTGR fast neutron fluences. The OPyC coatings of the remaining BISO fertile particle batches tested in capsules P13R and P13S were deposited from propylene.

Results of the visual examination and fission gas release measurements made on the fertile unbonded particle samples are summarized in Table 5-4. The irradiation performance of the mixed-gas OPyC coatings was good. OPyC coating failure levels determined by visual examination ranged from 0 to 1.0% after irradiation to severe fast neutron fluences from 10.8 to 11.7×10^{21} n/cm^2 ($E > 0.18 \text{ MeV}$)_{HTGR}. Equivalent in-service failure levels calculated from fission gas release data for four of the batches ranged from 0 to 1.4%, which indicated the coatings were relatively impermeable. A high level of fission gas release was measured for the other particle batch (6542-21-015) even though no OPyC coating failure was observed during the visual examination. The high fission gas release was attributed to the permeability of the low-density (1.73 g/cm^3) OPyC coating. Within the range of coating parameters tested, the irradiation performance of BISO OPyC coatings deposited from mixed gas was comparable to that of coatings deposited from propylene.

5.1.2.2.4. Buffer Coating Performance

The purpose of a low-density buffer coating immediately adjacent to the fuel kernel is to attenuate fission recoils and to provide a void volume to accommodate fission gases and fuel kernel swelling. The thickness and density of buffer coatings required for BISO particles irradiated to peak conditions are interrelated and are a function of kernel diameter and

OPyC coating design. BISO coated ThO_2 particles with buffer coating thicknesses ranging from 39 to 87 μm and densities ranging from 0.95 to 1.25 g/cm^3 were tested in capsules P13R and P13S.

The buffer coatings densified during irradiation and appeared slightly anisotropic when viewed under polarized light. Densification resulted from a combination of irradiation-induced PyC shrinkage and fuel kernel swelling. Cracked buffer coatings were observed in most particle batches examined metallographically, as indicated in Table 4-36. Cracking of the buffer coating is not considered detrimental to the performance of BISO coatings unless the cracks propagate into the OPyC coating. This phenomenon, an example of which is shown in Fig. 4-107, was observed only in a couple of particles. In many particles the buffer coating cracked while the OPyC coating remained intact, as shown in Fig. 4-106. In most instances where the OPyC coating failed during irradiation, the cracks propagated into the buffer coating. Because of the interdependence of the buffer and OPyC coating failure, it was not possible to quantitatively evaluate buffer coating performance relative to coating thickness and density. The effects of these variables can be dealt with quantitatively only by using a model which can calculate free void volume in the buffer coating and for a specific coating design can calculate stress levels in the OPyC coating as a function of irradiation parameters. The results from capsules P13R and P13S and from previous irradiation experiments (Ref. 5-2) will be analyzed using a BISO particle performance model to establish the buffer coating thickness and density limits for the LHTGR fuel specifications.

Buffer coatings are deposited from acetylene using argon as a diluent carrier gas. One particle batch (6542-12-025) was tested to determine if there were any detrimental irradiation effects from using N_2 as the buffer carrier gas. The buffer coatings deposited using N_2 carrier gas (Fig. 4-108) appeared similar to coatings deposited using argon as the carrier gas. Although a significant fraction of the buffer coatings deposited using N_2 were cracked, no apparent differences between the coatings applied using argon and those applied using N_2 carrier gas could be ascertained by metallographic examination.

5.1.2.2.5. Seal Coat

A thin, high-density, impermeable PyC seal coat was applied between the buffer and IPyC coatings of Fort St. Vrain carbide fuel. The purpose of this coat was to prevent hydrolysis of the carbide kernels during subsequent handling and storage of incompletely coated fuel particles and to allow the buffer and IPyC coatings to decouple during irradiation. Since ThO_2 fuel kernels are not subject to hydrolysis, a protective seal coat over the buffer coating is not required. From the standpoint of irradiation performance, it has not been established whether the buffer and OPyC coatings need to be decoupled by a seal coat during irradiation if the buffer coating is deposited from acetylene. Previous irradiation results have indicated that buffer coatings deposited from acetylene form a weaker bond with the OPyC coating than buffer coatings deposited from propylene, which was used for Fort St. Vrain fertile particles.

All but two BISO particle batches tested in capsules P13R and P13S had seal coats. During irradiation the highly anisotropic seal coats cracked and densified, leaving numerous small gaps at the buffer/OPyC coating interface, as shown in Fig. 4-104. The breakup of the seal coat effectively decoupled the buffer and OPyC coating, as shown in Fig. 4-106, and did not initiate any cracks in the adjacent buffer or OPyC coating.

Particle batches 6542-11-015 and 6542-18-015 were tested without seal coats and exhibited OPyC coating failure levels of 0.4 and 0% after irradiation to fast neutron fluences of 11.5 and $11.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}, respectively. The OPyC failure in batch 6542-11-015 was attributed to the relatively high anisotropy rather than the absence of a seal coat. Representative photomicrographs of particles from batch 6542-18-015 are shown in Fig. 4-105. The absence of a seal coat did not have any visible effect on the irradiation behavior of the BISO coatings. In approximately 3% of the particles, the buffer coating delaminated and shrank away from the OPyC coating. This indicates that buffer coatings

deposited from acetylene may not form a strong bond with the OPyC coating. However, the strength of the buffer/OPyC coating bond is also a function of the buffer coating density. If a seal coat were not present, lower-density buffer coatings would be more susceptible to penetration of carbon during the deposition of the OPyC coating, which would make the buffer/OPyC coating interface less distinct.

In summary, a seal coat effectively decoupled the buffer and OPyC coating during irradiation. This retarded but did not completely eliminate propagation of cracks from the buffer into the OPyC coating. Two BISO ThO₂ particle batches without seal coats performed satisfactorily during irradiation to severe fast neutron exposures. The absence of a seal coat did not have any apparent detrimental effect on BISO coating irradiation performance.

5.1.2.2.6. Coating Design

Results from capsules P13R and P13S indicate that for the particle designs tested, OPyC coating properties such as coating rate or anisotropy are more important for particle survival than OPyC coating thickness. For example, particle batches 6542-16-010 and 6542-17-010 had the same nominal kernel diameters and buffer coating parameters; however, there was a large difference (55 μm versus 122 μm) in the OPyC coating thickness of the two batches. After irradiation to similar conditions, the batches exhibited similar failure levels of 2.6 and 5.4%. The observed failure was attributed to the effects of high anisotropy of the coatings because the OPyC coating rate of both particle batches was less than 4.0 $\mu\text{m}/\text{min}$. As discussed previously in this section, a significant increase in OPyC coating failure was observed for particle batches with OPyC coatings deposited at coating rates less than 4.0 $\mu\text{m}/\text{min}$ owing to the increase in anisotropy at the lower deposition rates. In particle batches with OPyC coatings deposited at rates greater than 4.0 $\mu\text{m}/\text{min}$, no coating failure was observed that could be attributed to variations in coating design, although a much narrower range of OPyC coating thicknesses was tested. Irradiation data from

capsules P13R and P13S and previous screening tests (Ref. 5-13) will be analyzed using a BISO particle performance model to establish the range of coating thicknesses acceptable for LHTGR BISO fertile particles.

5.1.2.2.7. Faceted and Nonround Particles

Three batches of ThO_2 BISO particles were tested to determine the effect of faceted coatings and nonround coatings on irradiation performance. The faceted particles were characterized by their multifaced coatings. Normally, the particles appeared like dodecahedrons (see Fig. 4-28). Faceted coatings predominantly occur when large batch loads and/or low levitating gas flow rates are used. The nonround particles tested were caused by badly misshapen kernels. This batch was produced from a kernel charge which had a high fraction of cracked kernels. An example of these particles can be seen in Fig. 4-99. Each of these batches was tested in the unbonded particle cells, and particles from batch 6542-20-035 were also tested in six fuel rods.

To date, quantitative techniques have not been developed to evaluate the fuel particle failures observed. However, these techniques are scheduled to be ready before firm specifications are set limiting the fraction of misshapen kernels (and/or coated particles) and faceted coatings. These results show a need for such specifications, since between 0 and 5.4% particle failures were observed in the fuel rod and unbonded particle tested at $\sim 1000^\circ\text{C}$. Only one sample, batch 6542-20-035, was tested at temperatures higher than 1000°C , and it was irradiated in the thermal-cycled cell. Particle failures on the order of 25% were observed for this batch.

5.1.2.2.8. TRISO ThO_2 Performance

A TRISO coated ThO_2 particle is the LHTGR backup fertile particle design and is also being considered for use in Fort St. Vrain reload segments. One TRISO coated ThO_2 particle batch (6252-00-025) was tested in capsule P13S for comparative purposes.

The particle batch exhibited good performance during irradiation to a severe fast neutron exposure of $11.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. Total coating and OPyC failure levels of 0 and 0.4%, respectively, were observed during the visual examination, and the postirradiation fission gas release was equivalent to a failure level of 0.2%. It should be noted, however, that the batch was irradiated at a relatively low temperature of 860°C.

Representative photomicrographs of particle batch 6252-00-025 taken during the metallographic examination are shown in Fig. 4-109. The PyC coatings appeared relatively isotropic under polarized light. The IPyC coating debonded from the SiC layer in approximately 3.0% of the particles, and 10.8% of the particles had cracked buffer coatings. A reaction zone was observed on the inner surface of the SiC coating of one particle, as shown in Fig. 4-110. The IPyC coating was intact, although a portion of the cracked buffer coating pulled out during polishing. The reaction was not attributed to metallic fission products, since the irradiation temperature was low and signs of metallic fission products were not observed in the adjacent IPyC coating. A similar attack of the SiC coating was also observed in TRISO ThO_2 particles irradiated in capsule P13P at 1230°C. The reaction probably resulted from oxidation since oxidation of the SiC coating had been previously observed in irradiated TRISO UO_2 particles and unirradiated TRISO $(\text{Th,U})\text{O}_2$ particles during high-temperature thermal gradient annealing studies.

5.1.2.3. Thermochemical Stability. BISO coated fertile particles are not as susceptible to degradation by metallic fission products as BISO and TRISO coated fissile particles (Ref. 5-2). Radiographs of TRISO UC_2 and BISO ThO_2 particles irradiated in capsules P13R and P13S at temperatures of 1075° and 950°C are shown in Fig. 4-125. Evidence of diffusion of metallic fission products from the UC_2 fuel kernels can be observed; however, the BISO ThO_2 particles do not have any localized concentrations of fission products in the PyC coatings. No evidence of metallic fission product degradation of BISO ThO_2 particles was observed in the fuel rod specimens

irradiated at temperatures up to 1310°C. Although the fertile particle BISO coatings allow a controlled diffusive release of volatile metallic fission products, such as cesium and strontium, the quantities are not sufficient to result in degradation of the PyC coatings.

No evidence of ThO_2 fuel kernel migration was observed in capsule P13R and P13S fuel specimens. If kernel migration had occurred, elongation of the fuel kernel or thinning of the PyC coating on the hot side of the particles would have been detected. As discussed in Section 5.1.1.3, UC_2 fuel kernel migration was observed in fuel rods irradiated in cell 5 at average centerline temperatures ranging from 1225° to 1335°C. The absence of ThO_2 kernel migration in the P13R and P13S fuel specimens can be explained on the basis of a delay in the onset of kernel migration which has been observed in postirradiation heating experiments of low burnup samples (Ref. 5-14). Toward the end of the irradiation when the delay time was exceeded and migration was expected, the irradiation temperatures were too low for detectable migration to occur. Characterization of the temperature dependence of ThO_2 migration in a manner which accounts for the delay in onset of migration has not been completed. Therefore, predicted migration distances for ThO_2 kernels were not calculated using the CAPTEM code.

5.1.3. Effects of High-Temperature Transients

During the first year of operation under normal LHTGR steady-state operating conditions, 50 vol % of the fuel will experience peak centerline temperatures of $\leq 1150^\circ\text{C}$ and only 1 vol % will experience peak centerline temperatures of $\geq 1425^\circ\text{C}$. Fuel temperatures gradually drop with increasing age, and in the fourth year of operation they are approximately 150°C lower than during the first year. However, load-following transients result in increased localized fuel temperatures (Ref. 5-15). Thermal-cycling conditions during load-following are dependent on the maximum allowable region outlet temperature, load-following range, and reactor

design. In the core design described in Ref. 5-15 for the 1160-MW(e) LHTGR, load-following results in power and fuel temperature increases toward the center of the core. The peak fuel temperatures that were expected for the 100-60-100 load-following transient ranged from 1680°C at beginning-of-life to 1375°C at an end-of-life fluence of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} (Ref. 5-16). However, subsequent improvements in core loading patterns and rod programming have substantially reduced the temperatures expected for the 100-60-100 load-following transient.

Previous irradiation capsules have been operated at a constant design temperature (Ref. 5-2). Capsule P13S was the first attempt to simulate temperature cycling that results from load-following and/or control rod pattern changes in the LHTGR. Cell 1 of capsule P13S, which contained five fuel rods, was designed to be thermal cycled from a nominal operating temperature of 1075°C to 1600°C once every GETR fuel cycle. Identical fuel rods were irradiated in cell 1 of capsule P13R at a constant design temperature of 1075°C for comparative purposes. In addition to this test, eight fuel rods were heated to temperatures ranging from 1300° to 1600°C in the TRIGA facility to evaluate the effects of high-temperature transients on severely irradiated fuel.

5.1.3.1. In-pile Thermal Cycling. Decalibrated temperatures of the P13S cell 1 control thermocouple are plotted in Fig. 5-13 as a function of fast neutron fluence. Thermal cycles, where thermocouple data were obtained at the peak of the cycle, are represented by the spikes. Also shown is a temperature envelope of the LHTGR peak fuel temperature versus fast fluence that were assumed to occur under load-following conditions (Ref. 5-16). The fuel temperatures obtained in P13S cell 1 during thermal cycling were representative of the peak LHTGR fuel temperatures expected, except during the latter part of the irradiation, when the P13S fuel temperatures were approximately 100°C higher.

The irradiation parameters and fuel failure levels, determined by postirradiation fission gas release measurements, for the fuel rods tested in cell 1 of capsules P13R and P13S are given in Table 5-7. The same fuel

specimens were tested in both cells, and the specimens were irradiated under similar conditions. The only difference was that the P13S fuel rods were thermal cycled 24 times to temperatures 200° to 400°C higher than the nominal irradiation temperature. The effects of thermal cycling clearly resulted in increased fuel failure in all specimens during irradiation. However, the effect was most pronounced in fuel rods which contained out-of-specification fuel. The 1A and 1D fuel rods, which contained severely faceted fuel particles (1A) and fertile particles with the OPyC coating applied at a low deposition rate (1D), exhibited the highest failure levels of 22.0 and 24.0%, respectively. Faceting and low OPyC coating deposition rates also resulted in relatively high failure levels of 4.8 and 2.2% in the non-thermal-cycled fuel rods (1A and 1D) tested in capsule P13R.

Fuel rods 1B and 1C contained fuel particles more representative of LHTGR design requirements. These rods exhibited low average failure levels of 0.6 and 0%, respectively, after irradiation to peak LHTGR fluence conditions in capsule P13R. After thermal cycling in P13S, the average in-service particle failure levels in rods 1B and 1C were 1.1 and 2.2%, respectively. If all the measured fission gas is assumed to have been released from failed fertile fuel, then the maximum fertile particle failure levels in rods 1B and 1C were 1.3 and 2.4%. These failure levels are below a failure level of approximately 7.5% predicted for BISO ThO₂ particles that experience a thermal transient to 1550°C after irradiation to a fast neutron fluence of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} (Ref. 5-17). Because of the low fission rates in the fissile particles, a comparable fissile failure level was not determined.

5.1.3.2. Postirradiation Heat Treatment. After irradiation eight fuel rods were heated to temperatures ranging from 1300° to 1600°C in the TRIGA facility (see Section 4.4.3). The results of the fission gas release measurements made on these fuel rods are given in Tables 4-19 and 4-20 and Figs. 4-35 through 4-42. Six of the fuel rods did not experience any additional fuel failure during heating to temperatures 100° to 300°C higher than their average irradiation temperature. This was evidenced by the absence of an increase in fission gas release during the heat treatment other than the

increase resulting from the R/B temperature dependence. However, all five rods heated to a maximum temperature of 1600°C did experience additional fuel failure as a result of the heat treatment. Because of the possible contribution of the residual fission gas inventory from successive measurements to the final R/B level, induced fuel failure could only be treated quantitatively in three of the fuel rods when the final R/B measurement was made several days after the heat treatment. The results of the fission gas release measurements made on these three rods are given in Table 5-8.

Particle failure levels were calculated from the R/B data by assuming a fractional release of 5.0×10^{-3} for failed fuel. The results indicate that a significant level of particle failure was induced during the 1600°C heat treatment in the TRIGA facility. Fuel rod 7161-004-03-6, which was thermal cycled in-pile, also exhibited a significant increase in fuel failure during the heat treatment. Fuel rods 7161-004-14-5 and 7161-004-17-5 contained the same fissile and fertile particle batches as rod 7161-004-02-5, which was thermal cycled in-pile (Table 5-7). Total particle failure levels in the two fuel rods after heating to 1600°C in the TRIGA facility were 8.3 and 5.8%, while fuel rod 7161-004-02-5 exhibited a failure level of only 1.1% after being thermal cycled in-pile. It is not clear why the postirradiation heat treatment resulted in such a significant increase in fuel failure relative to the in-pile thermal cycling.

5.2. FUEL ROD IRRADIATION PERFORMANCE

5.2.1. Integrity

A large number of pitch-bonded fuel rods have exhibited satisfactory performance in previous capsule irradiation experiments. The integrity of bonded fuel rods was found to be independent of irradiation temperature and fast neutron fluence within the realm of LHTGR operating conditions (Ref. 5-2).

The irradiation performance of bonded fuel rod specimens tested in capsules P13R and P13S was good. Very little irradiation-induced damage

was observed, even in fuel rods irradiated to a fast neutron fluence of $12.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}, which is 55% beyond the peak LHTGR design fast neutron exposure. The visual appearance and amount of particle debonding observed in the fuel rods are summarized in Tables 4-1 and 4-2. The type of fuel rod fabrication process (hot-injection and slug-injection) and the firing process variables tested (1500° and 1800°C firing temperatures, firing in argon or N₂ atmosphere, and carbonization in H-451 graphite tubes or packed beds of alumina) did not influence fuel rod irradiation performance. The postirradiation integrity of all the fuel rods fabricated or fired under these conditions was good.

The fuel rods irradiated in P13S cell 5 at a design temperature of 1500°C were in relatively good condition, although the rods appeared somewhat weak around the edges, especially those in positions 5C and 5D, which exhibited slight circumferential cracking on the ends. Fuel rods tested in P13S cell 1, which was thermal cycled from a nominal operating temperature of 1075°C to temperatures ranging from 1500° to 1650°C twenty-four times during capsule operation, were in good condition. These results indicate that high-temperature operation or thermal cycling to high temperatures is not detrimental to fuel rod integrity.

5.2.2. Influence of Product and Process Variables

5.2.2.1. Cure-in-Place. The majority of the rods were fired in H-451 graphite tubes to simulate the cure-in-place process. Two fuel rods were fired in packed beds of alumina for comparative purposes.

The postirradiation integrity of the cure-in-place fuel rods was excellent. Seven of the rods exhibited slight matrix cracking and particle debonding, but this was attributed to the matrix variables tested in these rods rather than to the fabrication process. These results are consistent with other capsule tests which also demonstrated the excellent irradiation stability of cure-in-place fuel rods (Refs. 5-18, 5-19).

5.2.2.2. Shim Particles. LHTGR fuel rods contain from 25.3 to 42.7 vol % coated fuel particles. The remaining portion of the total particle packing volume (34.7 to 17.3 vol %) is made up of uncoated graphite shim particles. The graphite shim particles used in the P13R and P13S fuel rods were comminuted, bulk, near-isotropic graphite supplied by candidate vendors for near-isotropic HTGR graphite. The GLCC 1099 shim impregnated with furfuryl alcohol was used in 32 fuel rods. Four rods contained nonimpregnated GLCC 1099 shim, and the remaining two rods contained impregnated Lonza shim particles.

Fuel rods fabricated with GLCC 1099 shim particles (impregnated or nonimpregnated) which contained medium to high (20 to 36 vol %) shim contents exhibited excellent integrity after irradiation to fast neutron fluences up to $12.3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. However, six fuel rods fabricated with Lonza shim particles or with low (10 to 15 vol %) shim contents exhibited slight matrix cracking and particle debonding. Most of the particle debonding occurred predominantly on one end of the rods. This indicates that the matrix content at the end of the rod was not adequate to provide a good matrix/particle bond rather than that the matrix itself was weak.

In the hot-injection process, the matrix (pitch plus filler) is injected into the particle bed from the bottom end of the rod. During the fabrication of the P13R and P13S fuel rods, it was difficult to achieve proper filtration of the matrix into the particle beds; consequently, a matrix inhomogeneity from one end of the rod to the other often existed. A matrix-poor region at one end of a rod would be more susceptible to particle debonding. Matrix nonuniformity could have been more prevalent in the rods containing low shim contents and Lonza shim. The Lonza shim was more porous than the GLCC 1099 shim and would therefore be expected to absorb more pitch during the injection process, which could result in a matrix-poor region at one end of the rod. The rods with lower shim contents had a higher particle packing fraction. This lower void space would restrict the flow of matrix during the injection process.

These observations indicate the particle debonding observed on the ends of some of the fuel rods resulted from improper filtration of the matrix into the particle bed during rod fabrication, rather than from a characteristic of the type of shim particle used in the rods. The matrix injection problem during fabrication of fuel rods has been eliminated for subsequent capsule tests by using additives to decrease the viscosity of the matrix during injection.

5.2.2.3. Matrix Type. Fuel rods fabricated with natural-flake graphite filler and A240 petroleum pitch binder have been irradiated successfully to peak LHTGR fast neutron fluences in previous capsules (Ref. 5-18). However, in some of the rods tested, high levels of OPyC coating failure occurred as a result of excessive matrix/coating interaction due to the strong bond formed between the matrix binder and the OPyC coatings. Formation of the strong bond was attributed to the porous surface texture of the OPyC coatings and the high pitch coke content of the matrix. A matrix blend of 30 wt % 6353 natural-flake graphite filler, 65 wt % A240 petroleum pitch binder, and 5 wt % SC011 additive was used in 34 of the rods in capsules P13R and P13S. The remaining six rods contained Lonza KS-44 graphite filler and SC003 or SC027 matrix additives. The volatile matrix additives were incorporated to improve the ease of fuel rod fabrication and to reduce the fuel rod pitch coke content after firing.

No evidence of matrix/coating interaction was observed in any of the P13R and P13S fuel rods examined metallographically. No differences were observed between the matrices containing the 6353 natural-flake or the Lonza KS-44 graphite filler. The higher fraction of particle debonding observed in some of the rods was attributed to inadequate matrix injection rather than the matrix type, as discussed in Section 5.2.2.2.

The good irradiation stability of the P13R and P13S fuel rods indicates that the matrix combinations tested are suitable for LHTGR fuel rods. For these matrix combinations, pitch coke yields of 22 to 39% and levels of matrix macroporosity ranging from 26 to 43 vol % are adequate to ensure good fuel rod integrity during irradiation to peak LHTGR fast neutron

exposures. Nonuniform injection of the matrix material during fuel rod fabrication can result in loss of integrity and subsequent particle debonding.

5.2.2.4. Firing Conditions. All fuel rods were carbonized by heating to 1000°C at 10°C per minute in an induction furnace in a nitrogen atmosphere. Thirty rods were high-fired at 1800°C in an argon atmosphere. The remaining 10 rods were either fired at 1800°C in a nitrogen atmosphere or at 1500°C in an argon atmosphere. Neither the lower firing temperature of 1500°C nor the nitrogen atmosphere had any apparent effect on the irradiation stability of the fuel rods. However, dimensional change measurements revealed that the fuel rods fired at 1500°C shrink less than companion rods fired at 1800°C (see Fig. 4-33).

5.3. METALLIC FISSION PRODUCT BEHAVIOR

From the analysis of the gamma-ray spectroscopy results of the fuel rods, graphite crucibles, and unbonded particle samples, the magnitude of metallic fission product release and redistribution in the capsule has been determined and a qualitative evaluation of fuel homogeneity in the rods made. A value for the fractional release of Cs-137 from failed fuel in the rods and unbonded fissile particle samples has also been estimated.

5.3.1. Fuel Rod Homogeneity

Fuel loadings in the rods were fairly homogeneous, as evidenced by the relatively uniform Cs-137 and Zr-95 fission product profiles measured in the rods shown in Figs. 4-43 through 4-46. There was evidence, however, of small localized regions of fuel inhomogeneity in some of the fuel rods. Regions of high fuel concentrations, the majority of which occurred near the ends of the fuel rods, are indicated by peaks in the fission product profiles. No correlation was obtained between these regions of slightly higher fuel concentration and fuel performance as assessed by metallographic examination, fission gas release measurements, or visual examination.

5.3.2. Release from Failed Fuel

5.3.2.1. Fuel Rods. The measured and calculated fission product isotope inventories of the fuel rods that were gamma counted are given in Tables 4-25 and 4-26. The two fission product isotope inventories are in agreement within $\pm 20\%$, which indicates that metallic fission product release from the fuel rods during irradiation was low. However, the best measure of fission product release from the fuel rods was obtained by evaluating the difference between the measured and calculated Cs-137/Zr-95 ratios. This is because (1) Cs-137 is more mobile than Zr-95, so that a low measured ratio indicates cesium loss, and (2) by taking a ratio of the two fission product activities, the uncertainty associated with calculating the theoretical inventory is removed. The difference between the calculated and measured Cs-137/Zr-95 ratios given in Tables 5-9 and 5-10 indicates that cesium loss from the fuel rods was low. The only rods which exhibited a measurable cesium loss were those tested in P13R position 1C and P13S positions 1A and 1D. The cesium losses for these rods were 10.0, 9.5, and 15.8%, respectively.

A measure of the cesium loss was also obtained by examining the fuel rod graphite crucibles. Relative Cs-137 profiles obtained for all the graphite crucibles are shown in Figs. 4-47 and 4-48. The regions of the graphite crucible adjacent to the fuel rods tested in P13S positions 1A and 1D showed a high Cs-137 activity, as would be expected since these rods exhibited a relative high level of fuel failure. Cesium concentrations in the other graphite crucibles were relatively low. The graphite crucible adjacent to the fuel rod in P13R position 1C also exhibited a low activity, which indicates that cesium loss from the 1C fuel rod was probably not as high as the 10% value given above. The low fission gas release ($R/B \text{ Kr-85m} = 1.0 \times 10^{-6}$) obtained for this rod, which indicates a fuel failure level of 0%, supports this conclusion.

Most of the metallic fission product release observed was attributed to release from failed particles rather than diffusive release from intact BISO coated particles. This conclusion is supported by comparison of the measured cesium release of individual fuel rods with the fuel failure

levels determined from the fission gas release measurements (see Section 4.4.2, Tables 4-15 and 4-16). Fuel rods tested in P13S positions 1A and 1D, which exhibited significantly higher levels of cesium release than the other rods, had average fuel failure levels of 22.0 and 24.0%, which were significantly higher than fuel failure levels observed in the other rods. The cesium release from P13S cell 5 (indicated by the graphite crucible cesium activity) was higher than the release from P13R cell 5, which is consistent with the higher level of fuel failure observed in P13S cell 5 than in P13R cell 5. Fuel rod 2A of capsule P13R exhibited a higher cesium release and also a higher level of fuel failure than the other rods tested in cell 2.

From the above results, the fractional release of Cs-137 from failed fuel in the P13S 1A and 1D fuel rods was estimated. Fractional Cs-137 release values were calculated for failed fuel in the rods by dividing the percent of cesium lost from the fuel rods determined from the measured and calculated Cs-137/Zr-95 ratios (Table 5-10) by the failure fractions determined from the fission gas release measurements (Table 4-16). These results are summarized in Table 5-11. The values of 0.43 and 0.66 probably underestimate the actual fractional Cs-137 release values from failed fuel for two reasons: (1) the total Cs-137 inventory in the bonded fuel rods was used in the calculations; however, a portion of the cesium released from the failed fuel has been sorbed by the graphite matrix; and (2) the measured Cs-137/Zr-95 ratios in Table 5-10 of fuel rods that did not exhibit any cesium loss are systematically higher than the calculated ratios by 5 to 10%. If the measured ratios were decreased by 10%, the cesium loss from rods 1A and 1D would be 17.8% and 22.2%, which would correspond to a fractional Cs-137 release from failed fuel of 0.81 and 0.93, respectively.

5.3.2.2. Unbonded Particles. Because the unbonded particle batches were gamma-counted through a 2-in.-thick lead shield, a reliable calibration for quantitative analysis could not be made. Instead, a qualitative evaluation of the metallic fission product retention of each particle batch was made by comparing the measured and calculated Cs-137/Zr-95 ratios.

The Cs-137/Zr-95 ratios obtained for each fertile particle batch examined are given in Tables 4-32 and 4-33. The BISO and TRISO coated fertile particle batches did not exhibit any measurable cesium loss. The measured Cs-137/Zr-95 isotope ratios were within 10% of the calculated theoretical ratio; however, the measured ratios were systematically higher. Reasons for this discrepancy are discussed in Section 4.5.3. The fertile particle batches exhibited failure levels ranging from 0 to 10.2% (see Tables 4-4 and 4-5). No cesium release was observed, even in particle batches containing failed particles, within the experimental uncertainties of the measurements. These results indicate that the irradiation temperatures, which ranged from 830° to 1010°C, were low enough to prevent the diffusive release of cesium from the ThO₂ particles even if failure occurred during the 6200-hr irradiation. These results are in accord with cesium release levels predicted for these samples using the FIPER code, which is used for calculating the release of metallic fission products into the primary coolant circuit of an LHTGR (Ref. 5-14).

The unbonded fissile particle batches were gamma-counted using the same procedure, and the Cs-137/Zr-95 ratios obtained for each particle batch are given in Tables 4-30 and 4-31. Comparison of the measured and calculated fission product ratios revealed that some of the batches lost up to 20% of their cesium inventory. Since the measurement uncertainty in the Cs-137/Zr-95 ratio is $\pm 10\%$, a comparison was made only between the cesium loss and failure level in particle batches that exhibited significant failure. In Table 5-12, the cesium loss is compared with the failure level of these particle batches determined from the fission gas release measurements. To evaluate the fractional release of Cs-137 from failed fissile fuel, the percent Cs-137 loss was plotted as a function of failure fraction (see Fig. 5-14). The slope of this curve (least-squares fit forced through zero) is 0.70, which gives the fractional release of Cs-137 from failed fissile particles irradiated in the temperature range of 1015° to 1075°C. This value is in good agreement with the normalized values of 0.81 and 0.93 for the fractional Cs-137 release from failed fuel in the two bonded rods. Although there is a large measurement uncertainty associated with these values, the data indicate that the fractional cesium release is

high. It should be noted that in calculating metallic fission product release in an LHTGR core, the assumption is made that 100% of the metallic fission product inventory in failed fissile fuel is released to the fuel rod matrix (Ref. 5-20).

5.3.3. Fission Product Transport

An approximate cesium mass balance was obtained by comparing the Cs-137 isotope inventories observed in the fuel rods and graphite crucibles. Because of the large uncertainty associated with the gamma-counting data and the low cesium activity levels observed in most of the graphite crucibles, only cell 1 of P13S, which had the highest cesium release, was analyzed. Fuel rods 1A and 1D were the only rods in this cell which exhibited measurable cesium loss (see Section 5.3.2.1). The calculated Cs-137 losses from rods 1A and 1D were 9.5 and 15.8%, which corresponds to 1.9×10^4 μCi and 3.2×10^4 μCi , respectively. The measured cesium activity of the P13S graphite crucibles is given in Table 4-29. The total Cs-137 activity of the cell 1 crucible (spectrum numbers 1 through 7 in Table 4-29) was 4.3×10^4 μCi , which represents 84.0% of the total Cs-137 lost from fuel rods 1A and 1D.

The fact that most of the cesium released from the fuel rods was present in the graphite crucibles would be expected. Cesium released from the fuel rods is sorbed in the graphite and transported primarily by a surface or bulk diffusion mechanism rather than by gas phase transport. The graphite crucible operated at an average temperature of 760°C and a maximum temperature of 1140°C during irradiation. The mean migration distance of cesium in the graphite crucible calculated using a diffusion coefficient of 1.6×10^{-9} cm^2/sec at 760°C would be 2.7 mm, which is slightly less than one-half the thickness of the graphite crucible wall.

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TABLE 5-1
SUMMARY DESCRIPTION AND FAILURE ANALYSIS OF UNBONDED FISSION PARTICLE SAMPLES

Data Retrieval Number	Particle Type (a)	Sample Description				No. of Particles Examined	Irradiation Conditions					Visual Examination					In-Service Particle Failure (d) (%)	Variables Tested
		OPvC Coating Thickness (μm)	OPvC Coating Density (g/cm ³)	OPvC Coating Rate (μm/min)	OPvC Coating Anisotropy (RA ₁₀)		Capsule	Cell Position	Average Temp. (b) (°C)	Maximum Temp. (b) (°C)	Fast Fluence (10 ²¹ n/cm ²) (F > 0.18 MeV) HTGR	Burnup (% FIMA)	OPvC Coating Failure (%)	95% Confidence Interval, (c) P (%)	Pressure Vessel Failure (%)	95% Confidence Interval, (c) P (%)		
6151-00-010	UC ₂	44	1.80	1.1	1.042	682	P13R	3-1, 2, 3	1015-1075	1360-1400	12.1	74	32.8	31 < P ≤ 33	(e)	0.3 < P < 1.7	4.0 ^(f)	Low OPvC coating rate
6151-00-035	UC ₂	37	1.85	3.9	1.030	459	P13R	3-4, 5	1075	1400	12.0	74	0	0 < P < 0.8	0	0 < P < 0.8	0.6	
6151-00-035	UC ₂	37	1.85	3.9	1.030	809	P13S	4-10, 11, 12	975-1035	1345-1370	10.7	71	0	0 < P < 0.7	0	0 < P < 0.7	0.3	
6151-00-045	UC ₂	40	1.81	3.8	1.024	749	P13R	3-7, 8, 9	1075	1400	11.9	74	6.8	5.1 < P ≤ 8.1	6.5	4.8 < P ≤ 7.7	19.1 ^(f)	Faceted coatings
6151-00-046	UC ₂	40	1.81	3.3	1.024	254	P13R	3-10	1025	1345	11.8	74	3.5	1.9 < P < 6.6	2.0	0.8 < P ≤ 4.5	14.0 ^(f)	Faceted coatings
6151-04-015	UC ₂	36	1.82	1.4	1.038	492	P13R	3-11, 12	960-1025	1325-1345	11.7	74	49.0	44.6 < P ≤ 53.4	0	0 < P < 0.8	0.6	Low OPvC coating rate
6151-01-015	UC ₂	42	1.90	0.7	1.077	464	P13R	4-2, 3	1070	1400	11.5	73	100	99.2 < P ≤ 100	2.8	1.6 < P < 4.7	1.4	Low OPvC coating rate
6151-00-025	UC ₂	33	1.88	1.4	1.052	819	P13R	4-4, 5, 6	1070	1400	11.3	73	83.0	81 < P < 85	(e)	0 < P < 0.7	6.5	Faceted and low OPvC coating rate
6151-09-015	UC ₂	39	1.94	3.6	1.028	785	P13R	4-7, 8, 9	1070	1400	11.1	73	0	0 < P < 0.7	0	0 < P < 0.7	1.8	High OPvC coating density
6151-09-025	UC ₂	41	1.95	4.0	1.033	491	P13R	4-10, 11	1005	1345	11.0	73	0	0 < P < 0.8	0	0 < P < 0.8	0.12	High OPvC coating density
6151-03-015	UC ₂	45	1.77	1.5	1.020	729	P13S	3-1, 2, 3	970-1020	1255-1280	11.8	73	4.3	3.0 < P ≤ 6.0	0.1	0 < P < 0.7	0.2 ^(f)	Interrupted SiC layer
4161-01-021	UC ₂	32	1.82	3.4	ND ^(g)	718	P13S	3-4, 5, 6	1020	1280	11.7	73	3.9	2.7 < P ≤ 5.6	3.3	2.2 < P ≤ 4.9	26.9 ^(f)	Interrupted SiC layer, faceted
6151-08-015	UC ₂	36	1.76	3.6	1.018	688	P13S	3-7, 8, 9	1020	1280	11.7	73	0	0 < P < 0.6	0	0 < P < 0.6	0	Low OPvC coating density
6151-02-025	UC ₂	43	1.50	0.7	1.039	501	P13S	3-10, 12	920-980	1200-1230	11.6	73	46.3	42 < P ≤ 51	(e)	0.04 < P ≤ 1.1	4.3 ^(f)	Low OPvC coating rate and density
4163-00-011	(Th,U)O ₂	72	1.77	4.1	ND	767	P13S	4-1, 2, 3	1045-1090	1385-1430	11.2	40	0	0 < P < 0.5	0	0 < P < 0.5	0	250-μm kernel diameter
6155-01-020	(Th,U)O ₂	42	1.81	3.4	1.075	163	P13S	4-4	1090	1430	11.1	11	76.7	69 < P < 83	1.8	0.6 < P < 5.3	2.6	500-μm kernel diameter
5466-37	(Th,U)O ₂	31	1.79	6.0	ND	801	P13S	4-7, 8, 9	1090	1430	10.9	38	0	0 < P < 0.7	0	0 < P < 0.7	0	250-μm kernel diameter

(a) All particles were PRISO coated.

(b) Range of time averaged and maximum fuel bed centerline temperatures.

(c) An infinite sample population with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(d) Calculated from fission gas release data assuming a fractional fission gas release (r/b_f) of 2×10^{-2} for unconstrained failed fuel particles except where noted.

(e) An accurate assessment of the pressure vessel failure level could not be made during the visual examination because fragments of the failed OPvC coatings remained bonded to the SiC substrate.

(f) Calculated assuming a fractional release of 3.5×10^{-2} for bare kernels.

(g) ND = not determined.

TABLE 5-2
COMPARISON OF FISSILE PARTICLE OPyC AND TOTAL COATING FAILURE LEVELS

Data Retrieval Number	SiC Coating Substrate Number ^(a)	Average Temperature (°C)	Burnup (% FIMA)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Coating Failure (%)	
					OPyC ^(b)	Total ^(c)
6151-00-035	5862-1115	1025	74	12.0	0	0.6
6151-00-035		1045 - 1090	71	10.7	0	0.3
6151-08-015		980	73	11.6	0	0
6151-02-025		970 - 1020	73	11.6	46.3	4.3
6151-09-015	5862-1095	1005	73	11.1	0	1.8
6151-01-015		935 - 1010	73	11.5	100	1.4
6151-09-025	5862-975	1070	73	11.0	0	0.1
6151-00-010		960 - 1025	74	12.1	32.8	4.0

(a) The same SiC substrate number indicates one particle batch through deposition of the SiC layer. The batch was then split, and different OPyC coatings were applied.

(b) OPyC coating failure level was determined by visual examination.

(c) Total coating failure level was calculated from fission gas release data.

TABLE 5-3
ANALYSIS OF METALLIC FISSION PRODUCT ATTACK OF SiC COATING IN TRISO UC₂ PARTICLES

Particle Batch Data Retrieval Number	Fuel Rod Data Retrieval Number (7161-004-)	Capsule	Position	Average Fuel Rod Temp. (a) (°C)	Particle I.D. Number (b) (L7440-)	Particle Location (c) (cm)	Average Particle Temp. (°C)	Particle Temp. Gradient (°C/cm)	Reaction Depth (μm)
6151-00-035	07-5	P13R	2B	1055	343	0.2200	1037	161	2.5
					340	0.3876	1000	283	<1.0
6151-00-035	08-13	P13R	2C	1045	350	0.4431	981	287	<1.0
					353	0.5290	954	343	<1.0
					356	0.4587	977	297	<1.0
					363	0.3979	994	258	1.5
					368	0.1487	1038	96	<1.0
					371	0.4511	979	292	<1.0
6151-00-035	18-5	P13S	5C	1335	403	0.2300	1321	121	3.0
					406	0.4838	1274	254	5.5
					411	0.1053	1332	55	1.5
					416	0.4835	1274	254	4.0
					419	0.1950	1325	102	3.0
6151-08-015	19-5	P13S	5D	1325	422	0.1378	1330	72	4.0
					428	0.5915	1243	311	3.0
					431	0.5715	1249	300	3.0
					434	0.1725	1327	91	4.5
					437	0.3730	1298	196	1.5
					609	0.1253	1331	66	14.0
					316	0.3259	1299	161	4.0
					319	0.2269	1312	112	6.0
					322	0.2731	1307	135	6.0
					331	0.5961	1237	294	6.0
					333	0.4345	1278	215	7.5
					335	0.3895	1288	192	6.5

(a) Time-averaged fuel rod centerline temperature.

(b) Identification number of laboratory photograph of fuel particle.

(c) Distance of particle from fuel rod centerline.

TABLE 5-4
SUMMARY DESCRIPTION AND FAILURE ANALYSIS OF UNBONDED FERTILE PARTICLE SAMPLES

Sample Description							Irradiation Conditions						Visual Examination		In-Service Particle Failure ^(c) (%)	Variables Tested
Date Retrieval Number	Buffer Thickness (μm)	OPyC Coating Thickness (μm)	OPyC Coating Density (g/cm ³)	OPyC Coating Rate (μm/min)	OPyC Coating Anisotropy (BAF ₀)	No. of Particles Tested	Capsule	Cell Position	Average Temp. (°C)	Maximum Temp. (°C)	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	Total Coating Failure ^(a) (%)	95% Confidence Interval ^(b) , P (%)		
4252-06-018	78	77	1.82	5.8	1.029	840	P13R	3-1, 2	960	1055	12.1	5.1	0.1	0.01 ≤ P ≤ 0.8	6.0	
6542-01-020	81	74	1.82	2.7	1.028	1004	P13R	3-3, 4	1010	1105	12.0	5.0	7.8	6.5 ≤ P ≤ 9.0	0.9	Low OPyC coating rate
6542-01-010	79	85	1.80	10.0	1.028	455	P13R	3-5	1010	1105	12.0	5.0	0.2	0.04 ≤ P ≤ 1.2	6.0	High OPyC coating rate
6542-25-015	79	76	1.82	10.8	ND ^(d)	539	P13R	4-11	940	1050	10.9	4.1	1.7	0.9 ≤ P ≤ 3.1		High OPyC coating rate, nonround kernels
6542-02-020	87	72	1.91	8.5	ND	1072	P13R	3-7, 8	1010	1105	11.9	4.9	0	0 ≤ P ≤ 0.5	0.2	
6542-02-020	87	72	1.91	8.5	ND	1149	P13R	4-5, 6	1000	1115	11.3	4.4	0	0 ≤ P ≤ 0.5	0.17	
6542-02-030	79	74	1.89	2.2	1.042	1148	P13R	3-9, 10	970-1010	1065-1105	11.8	4.8	10.2	10.0 ≤ P ≤ 10.3	14.5	Low OPyC coating rate
6542-09-010	84	75	1.93	5.0	1.042	998	P13R	3-11, 12	895-970	975-1065	11.7	4.8	0	0 ≤ P ≤ 0.5	0	
6542-11-015	87	74	1.83	2.6	ND	492	P13R	4-2	1000	1115	11.5	4.5	0.4	0.1 ≤ P ≤ 1.5	12.0	No seal coat
6542-19-015	81	77	1.78	7.6	1.026	495	P13S	4-7	950	1085	10.9	3.7	0.6	0.2 ≤ P ≤ 1.8		
6542-19-016	82	71	1.78	7.0	1.026	502	P13S	4-8	950	1085	10.9	3.7	0.2	0.04 ≤ P ≤ 1.1		
6542-18-015	84	70	1.84	6.5	ND	953	P13R	4-3, 4	1000	1115	11.4	4.5	0	0 ≤ P ≤ 0.6	0	No seal coat
6542-20-035	80	83	1.84	3.6	ND	1012	P13R	4-7, 8	1000	1115	11.2	4.3	0.6	0.2 ≤ P ≤ 1.2	3.5	Faceted coatings
6542-20-025	76	81	1.82	3.5	ND	1040	P13R	4-9, 10	940-1000	1050-1115	11.1	4.2	1.0	0.6 ≤ P ≤ 1.7	1.9	Faceted coatings
4252-06-010	85	76	1.82	5.8	1.029	872	P13S	3-1, 2	925	1035	11.8	4.3	0	0 ≤ P ≤ 0.6	0	
6542-22-015	85	81	1.80	4.0	1.020	979	P13S	3-3, 4	955	1075	11.7	4.3	1.0	0.5 ≤ P ≤ 1.5	1.4	Mixed-gas OPyC coating
6542-22-025	81	80	1.81	7.6	1.028	964	P13S	3-5, 6	955	1075	11.7	4.2	0.3	0.1 ≤ P ≤ 0.9	1.2	Mixed-gas OPyC coating
6542-23-025	82	73	1.89	5.8	ND	992	P13S	3-7, 8	955	1075	11.7	4.2	0	0 ≤ P ≤ 0.5	0	Mixed-gas OPyC coating
6542-24-015	86	83	1.94	6.6	1.032	942	P13S	3-9, 10	925-955	1035-1075	11.6	4.1	0	0 ≤ P ≤ 0.6	0	High-density mixed-gas OPyC coating
6252-00-025	60	42	1.81	4.2	1.015	528	P13S	3-12	860	960	11.5	4.1	0	0 ≤ P ≤ 1.4	0.2	TRISO coated
6542-17-010	44	122	1.86	2.8	ND	458	P13S	4-1	910	1040	11.2	3.9	2.6	1.5 ≤ P ≤ 4.5	23.5	Coating design, low OPyC coating rate
6542-16-010	39	55	ND	3.7	ND	349	P13S	4-4	950	1085	11.1	3.8	5.4	3.5 ≤ P ≤ 8.3		Coating design
4252-02-010	83	73	1.83	4.0	1.032	925	P13S	4-2, 3	950	1085	11.2	3.9	0	0 ≤ P ≤ 0.6	5.5	
6542-21-016	81	84	1.73	8.0	1.020	508	P13S	4-9	950	1085	10.9	3.7	0	0 ≤ P ≤ 0.8	10.5	Low OPyC coating density
6542-21-015	79	82	1.73	7.8	1.020	513	P13S	4-10	900	1025	10.8	3.6	0	0 ≤ P ≤ 0.7		Low-density mixed-gas OPyC coating
6542-12-025	84	70	1.79	6.7	ND	1214	P13S	4-11, 12	830-900	940-1025	10.7	3.6	0.2	0.1 ≤ P ≤ 0.8	1.0	H ₂ buffer carrier gas

(a) Failure of the OPyC coating of BISO particles or failure of the OPyC and SiC coating of TRISO particles that would result in the release of gaseous fission products.

(b) An infinite sample population with attributes of the particle batch tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(c) Calculated from fission gas release data assuming a fractional release (r/b_f) of 2 × 10⁻² for unconstrained failed fuel particles.

(d) ND = not determined.

TABLE 5-5
SUMMARY DESCRIPTION AND FAILURE ANALYSIS OF FERTILE PARTICLES TESTED IN FUEL RODS

Sample Description					Irradiation Conditions					Metallographic Examination			Maximum Fertile Particle Failure ^(c) (%)	Variables Tested
Fertile Particle Batch Data Retrieval No.	Fuel Rod Data Retrieval No. (7161-004-)	OPyC Coating Density (g/cm ³)	OPyC Coating Rate (μm/min)	OPyC Coating Anisotropy (BAF _o)	Capsule-Position	Average Temp. ^(a) (°C)	Maximum Temp. ^(b) (°C)	Fast Fluence (10 ²¹ n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)	Number of Particles Examined	Pressure Vessel Failure (%)	95% Confidence Interval, P (%)		
6542-20-035	01-5	1.84	3.6	(d)	P13R-1A	1115	1340	7.3	3.5	NE ^(e)	--	--	5.4	Faceted coatings
6542-20-035	06-5	1.84	3.6	(d)	P13R-2A	1050	1225	11.8	5.5	157	1.8	0.2 ≤ P ≤ 9.8	5.3	
6542-20-035	25-5	1.84	3.6	(d)	P13R-6E	1000	1155	3.6	0.6	NE	--	--	0.3	
6542-20-035	01-7	1.84	3.6	(d)	P13S-1A	1030	1545 ^(f)	7.0	3.2	135	51.9	43.5 ≤ P ≤ 60.1	24.9	
6542-20-035	11-6	1.84	3.6	(d)	P13S-2A	1040	1170	11.5	4.6	NE	--	--	3.2	
6542-20-035	30-5	1.84	3.6	(d)	P13S-6A	1085	1250	5.3	1.1	NE	--	--	0	
6542-02-020	02-6	1.91	8.5	(d)	P13R-1B	1050	1225	8.1	3.9	NE	--	--	0.6	
6542-02-020	05-6	1.91	8.5	(d)	P13R-1E	1125	1305	9.9	4.8	NE	--	--	0	
6542-02-020	07-5	1.91	8.5	(d)	P13R-2B	1050	1225	12.0	5.6	161	0	0 ≤ P ≤ 2.3	0.3	
6542-02-020	17-5	1.91	8.5	(d)	P13R-5B	1225	1460	9.0	2.8	NE	--	--	0.7	
6542-02-020	24-5	1.91	8.5	(d)	P13R-6D	1020	1175	4.1	0.7	NE	--	--	0.5	
6542-02-020	02-5	1.91	8.5	(d)	P13S-1B	975	1470 ^(f)	7.7	3.4	NE	--	--	1.3	
6542-02-020	05-5	1.91	8.5	(d)	P13S-1E	1035	1575 ^(f)	9.6	4.1	NE	--	--	4.9	
6542-02-020	14-5	1.91	8.5	(d)	P13S-2D	985	1100	12.0	4.6	NE	--	--	0.2	
6542-02-020	17-6	1.91	8.5	(d)	P13S-5B	1330	1605	8.7	2.6	NE	--	--	2.2	
6542-02-020	27-5	1.91	8.5	(d)	P13S-6B	1045	1215	4.8	1.0	NE	--	--	0	High OPyC coating rate
6542-01-010	03-5	1.80	10.0	1.028	P13R-1C	1065	1235	8.7	4.2	(g)	--	--	0	
6542-01-010	08-13	1.80	10.0	1.028	P13R-2C	1045	1195	12.2	5.6	144	0	0 ≤ P ≤ 2.6	0.3	
6542-01-010	10-5	1.80	10.0	1.028	P13R-2E	965	1110	12.4	5.5	NE	--	--	0.8	
6542-01-010	18-6	1.80	10.0	1.028	P13R-5C	1235	1475	8.5	2.6	NE	--	--	0	
6542-01-010	23-5	1.80	10.0	1.028	P13R-6C	1005	1160	4.5	0.8	76	0	0 ≤ P ≤ 4.8	0	
6542-01-010	03-6	1.80	10.0	1.028	P13S-1C	995	1505 ^(f)	8.5	3.7	NE	--	--	26.3	
6542-01-010	13-6	1.80	10.0	1.028	P13S-2C	975	1090	11.9	4.7	175	0	0 ≤ P ≤ 2.1	0.7	
6542-01-010	18-5	1.80	10.0	1.028	P13S-5C	1335	1625	8.4	2.4	165	0	0 ≤ P ≤ 2.3	0.6	
6542-01-010	28-13	1.80	10.0	1.028	P13S-6C	1050	1215	4.4	0.8	112	0	0 ≤ P ≤ 3.3	0	
6542-01-020	04-6	1.82	2.7	1.028	P13R-1D	1115	1290	9.3	4.5	NE	--	--	2.4	
6542-01-020	09-5	1.82	2.7	1.028	P13R-2D	1035	1180	12.3	5.6	NE	--	--	8.3	
6542-01-020	16-5	1.82	2.7	1.028	P13R-5A	1285	1515	9.4	3.1	(g)	--	--	0.9	
6542-01-020	04-5	1.82	2.7	1.028	P13S-1D	1035	1555 ^(f)	9.1	3.9	NE	--	--	4.9	Low OPyC coating rate
6542-01-020	16-6	1.82	2.7	1.028	P13S-5A	1375	1645	9.2	2.8	89	5.6	2.1 ≤ P ≤ 12.6	4.1	
6542-01-020	26-5	1.82	2.7	1.028	P13S-6A	1085	1250	5.3	1.1	NE	--	--	0	
6542-01-020	29-5	1.82	2.7	1.028	P13S-6D	1015	1215	3.9	0.7	NE	--	--	0	
6542-19-015	19-7	1.78	7.6	1.026	P13R-5D	1225	1480	8.2	2.4	206	0.5	0.1 ≤ P ≤ 2.7	0.2	
6542-19-015	21-5	1.78	7.6	1.026	P13R-6A	1065	1220	5.4	1.1	NE	--	--	0	Low OPyC coating density
6542-19-015	15-5	1.78	7.6	1.026	P13S-2E	960	1095	12.1	4.6	36	0	0 ≤ P ≤ 9.9	2.1	
6542-19-015	19-5	1.78	7.6	1.026	P13S-5D	1325	1610	7.9	2.2	194	5.7	3.2 ≤ P ≤ 9.9	1.7	
6542-09-010	20-6	1.93	5.0	1.042	P13R-5E	1280	1590	7.7	2.1	144	0.7	0.1 ≤ P ≤ 3.8	0.1	
6542-09-010	22-5	1.93	5.0	1.042	P13R-6B	1055	1215	5.0	1.0	NE	--	--	0	
6542-09-010	12-5	1.93	5.0	1.042	P13S-2B	1010	1135	11.7	4.6	129	0.8	0.1 ≤ P ≤ 4.3	0	High OPyC coating density
6542-09-010	20-5	1.93	5.0	1.042	P13S-5E	1350	1650	7.5	2.0	247	0	0 ≤ P ≤ 1.5	3.3	

(a) Time average fuel rod irradiation temperature.

(b) Maximum fuel rod centerline temperature during irradiation.

(c) Calculated assuming a fractional release (r/b_f) of 5 × 10⁻³ for failed fuel particles in a constrained geometry and that all the fission gas released came from failed fertile particles.

(d) Not determined.

(e) NE = not examined.

(f) Thermal cycled in-pile.

(g) Metallography was performed after the fuel rod was heated to 1600°C in TRIGA.

TABLE 5-6
FERTILE PARTICLE OPyC COATING DENSITY

Data Retrieval Number	OPyC Coating					Fission Gas Release (R/B Kr-85m)
	Rate ($\mu\text{m}/\text{min}$)	Gas	Density, $\rho_{\text{SF}}^{(a)}$ (g/cm^3)	Density, $\rho_{\text{bulk}}^{(b)}$ (g/cm^3)	$\rho_{\text{SF}} - \rho_{\text{bulk}}$ (g/cm^3)	
6542-09-010	5.0	Propylene	1.93	1.83	0.10	2.4×10^{-5}
6542-02-020	8.5	Propylene	1.91	1.82	0.09	4.4×10^{-5}
6542-18-015	6.5	Propylene	1.84	1.73	0.11	2.2×10^{-6}
6542-02-020	8.5	Propylene	1.91	1.82	0.09	3.3×10^{-5}
4252-06-010	5.8	Propylene	1.82	1.74	0.08	6.2×10^{-5}
6542-23-025	5.8	Mixed-gas	1.89	1.77	0.12	2.8×10^{-6}
6542-24-015	6.6	Mixed-gas	1.94	1.84	0.10	3.9×10^{-6}
4252-02-010	4.0	Propylene	1.83	1.76	0.07	1.1×10^{-3}
6542-21-016	8.0	Mixed gas	1.73	1.59	0.14	} 2.1×10^{-3}
6542-21-015	7.8	Mixed gas	1.73	1.59	0.14	
6542-22-015	4.0	Mixed gas	1.80	1.70	0.10	6.7×10^{-4}
6542-01-020	2.7	Propylene	1.82	1.76	0.06	1.6×10^{-3}
4252-06-018	5.8	Propylene	1.82	1.74	0.08	3.0×10^{-3}
6542-01-010	10.0	Propylene	1.80	1.71	0.09	1.5×10^{-3}

(a) Unirradiated OPyC coating density determined by a sink-float technique.

(b) Unirradiated OPyC coating density determined by low-pressure mercury intrusion.

TABLE 5-7
IRRADIATION PARAMETERS AND FUEL FAILURE LEVELS IN FUEL RODS TESTED IN CELL 1 OF CAPSULES P13R AND P13S

Fuel Rod Data Retrieval No. (7161-004-)	Capsule- Position	Average Temp. (°C)	Maximum Temp. (°C)	Fast Fluence (10^{21} n/cm ²) (E > 0.18 MeV) _{HTGR}	Burnup (% FIMA)		R/B Failed Fuel (a)	In-Service Particle Failure ^(b) (%)	Maximum Fertile Particle Failure ^(b,c) (%)	Remarks
					Fissile	Fertile				
01-7	P13S-1A	1030	(d)	7.0	70	3.2	1.1×10^{-3}	22.0	24.9	Faceted fuel particles
02-5	P13S-1B	975	(d)	7.7	71	3.4	5.7×10^{-5}	1.1	1.3	
03-6	P13S-1C	995	(d)	8.5	71	3.7	1.1×10^{-4}	2.2	2.4	Fertile particles had a high OPyC coating rate.
04-5	P13S-1D	1035	(d)	9.1	72	3.9	1.2×10^{-3}	24.0	26.3	Fertile particles had a low OPyC coating rate.
06-5	P13S-1E	1035	(d)	9.6	39	4.1	2.2×10^{-4}	4.4	4.9	(Th,U)O ₂ fissile particles
01-5	P13R-1A	1115	1340	7.3	71	3.5	2.4×10^{-4}	4.8	5.4	Faceted fuel particles
02-6	P13R-1B	1050	1225	8.1	72	3.9	2.9×10^{-5}	0.6	0.6	
03-5	P13R-1C	1065	1235	8.7	73	4.2	1.0×10^{-6}	0	0	Fertile particles had a high OPyC coating rate.
04-6	P13R-1D	1115	1290	9.3	73	4.5	1.1×10^{-4}	2.2	2.4	Fertile particles had a low OPyC coating rate.
05-6	P13R-1E	1125	1305	9.9	40	4.8	2.2×10^{-7}	0	0	(Th,U)O ₂ fissile particles

(a) R/B for Kr-85m measured at 1100°C. $R/B_{\text{failed fuel}} = R/B_{\text{EOL}} - R/B_{\text{contamination}}$.

(b) Calculated by assuming a fractional release of 5×10^{-3} for failed fuel in a constrained geometry.

(c) Calculated by assuming all the fission gas release came from failed fertile particles.

(d) P13S cell 1 fuel rods were thermal cycled 24 times during irradiation.

TABLE 5-8
FAILURE ANALYSIS OF FUEL RODS HEATED TO 1600°C IN THE TRIGA FACILITY

Data Retrieval Number (7161-004-)	Capsule- Position	Fission Gas Release (R/B Kr-85m at 1100°C)		Fuel Particle Failure ^(b) (%)			
				Pre-HT ^(a)		Post-HT ^(a)	
		Pre-HT ^(a)	Post-HT ^(a)	Total	Maximum Fertile ^(c)	Total	Maximum Fertile ^(c)
14-5	P13S-2D	1.2×10^{-5}	4.2×10^{-4}	0.2	0.2	8.3	8.9
03-6	P13S-1C	1.1×10^{-4}	5.2×10^{-4}	2.2	2.4	10.4	11.4
17-5	P13R-5B	2×10^{-5}	2.9×10^{-4}	0.6	0.7	5.8	6.5

(a) Pre-HT and post-HT indicate before and after heating to 1600°C in the TRIGA facility.

(b) Fuel failure levels were calculated from the fission gas release data (corrected for heavy metal contamination) by assuming fractional release of 5.0×10^{-3} for failed fuel.

(c) Calculated by assuming all fission gas was released from failed fertile particles.

TABLE 5-9
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13R FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	Measured $\frac{\text{Cs-137}}{\text{Zr-95}}$	Calculated $\frac{\text{Cs-137}}{\text{Zr-95}}$	Difference ^(a) (%)	Measured ^(b) $\frac{\text{Cs-137}}{\text{Cs-134}}$
01-5	1A	0.024	0.023	+4.2	0.668
02-6	1B	0.022	0.022	0	0.603
03-5	1C	0.020	0.022	-10.0	0.590
04-6	1D	0.022	0.022	0	0.577
05-6	1E	0.021	0.022	-4.8	0.560
16-5	5A	0.022	0.021	+4.5	0.618
17-5	5B	0.022	0.021	+4.5	0.648
18-6	5C	0.023	0.021	+8.7	0.631
19-7	5D	(c)	0.021	--	(c)
20-6	5E	0.022	0.021	+4.5	0.681
21-5	6A	0.023	0.022	+4.3	0.948
22-5	6B	0.022	0.022	0	1.008
23-5	6C	0.023	0.022	+4.3	0.976
24-5	6D	0.023	0.022	+4.3	1.410
25-5	6E	0.025	0.022	+12.0	1.142

(a) Difference between measured and calculated $\frac{\text{Cs-137}}{\text{Zr-95}}$ ratio.

(b) A theoretical Cs-134 inventory was not computed.

(c) Fuel rod 7161-004-19-7 was not gamma-counted.

TABLE 5-10
FISSION PRODUCT ISOTOPE RATIOS OF CAPSULE P13S FUEL RODS

Data Retrieval Number (7161-004-)	Capsule Position	Measured $\frac{\text{Cs-137}}{\text{Zr-95}}$	Calculated $\frac{\text{Cs-137}}{\text{Zr-95}}$	Difference ^(a) (%)	Measured ^(b) $\frac{\text{Cs-137}}{\text{Cs-134}}$
01-7	1A	0.021	0.023	-9.5	0.671
02-5	1B	0.022	0.022	0	0.650
03-6	1C	0.023	0.022	+4.3	0.603
04-5	1D	0.019	0.022	-15.8	0.544
05-5	1E	0.023	0.022	+4.3	0.558
16-6	5A	0.022	0.021	+4.5	0.765
17-6	5B	0.022	0.022	0	0.634
18-5	5C	0.023	0.021	+8.7	0.601
19-5	5D	0.024	0.021	+12.5	0.644
20-5	5E	0.021	0.021	0	0.715

(a) Difference between measured and calculated $\frac{\text{Cs-137}}{\text{Zr-95}}$ ratio.

(b) A theoretical Cs-134 inventory was not computed.

TABLE 5-11
FRACTIONAL Cs-137 RELEASE FROM FAILED FUEL IN FUEL RODS

Rod I.D.	Equivalent Fuel Failure ^(a) (%)	Cs-137 Loss (%)	Fractional Cs-137 Release From Failed Fuel
P13S-1A	22.0	9.5	0.43
P13S-1D	24.0	15.8	0.66

(a) Percent of fissions occurring in failed fuel (fissile and fertile) calculated from fission gas release data.

TABLE 5-12
COMPARISON OF Cs-137 LOSS WITH FISSILE PARTICLE FAILURE LEVELS

Batch Data Retrieval No.	Capsule ^(a)	Particle Failure (%)	Cs-137 Loss (%)
6151-00-025	P13R	6.5	4.4
6151-00-010	P13R	4.0	8.5
6151-00-045	P13R	19.1	13.6
6151-00-046	P13R	14.0	8.6
4161-01-021	P13S	26.9	18.9

(a) Particles were irradiated in the temperature range of 1015° to 1075°C.

Fig. 5-1. OPyC coating failure versus coating rate and density for UC₂ TRISO particles irradiated in capsules P13R and P13S to fast neutron fluences of 10.7 to 12.1×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} at 1075°C (design)

Fig. 5-2. Optical anisotropy measured on individual OPyC coatings of UC_2 TRISO particles versus apparent coating rate. Anisotropy values were measured by using a 4 x 24 μm slit and radial scanning the coatings. A Seibersdorf unit was used and anisotropies are in BAF_0 units.

Fig. 5-3. OPyC coating failure versus optical anisotropy for UC₂ TRISO particles irradiated in unbonded beds at 1075°C to fast neutron fluences of 10.7 to 12.1×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR}. Optical anisotropies were measured by using a Seibersdorf unit and radially scanning the coatings with a 4×24 μ m slit.

Fig. 5-4. OPyC performance on 600- μm -diameter TRISO particles irradiated in P13R and P13S fuel rods

Fig. 5-5. Density change of OPyC coatings on TRISO fuel particles irradiated at 950° to 1250°C: (a) initial densities of 1.76 to 1.82 g/cm³ and (b) initial densities of 1.85 to 1.94 g/cm³

Fig. 5-6. Measured and predicted UC_2 fuel kernel migration distance as a function of radial location in fuel rod 7161-004-18-5 irradiated in capsule P13S (position 5C)

Fig. 5-7. Measured and predicted UC_2 fuel kernel migration distance as a function of radial location in fuel rod 7161-004-19-7 irradiated in capsule P13R (position 5D)

Fig. 5-8. Measured and predicted UC_2 fuel kernel migration distance as a function of radial location in fuel rod 7161-004-19-5 irradiated in capsule P13S (position 5D)

Fig. 5-9. Optical anisotropy measured on individual OPyC coatings of ThO₂ BISO particles versus apparent coating rate. Optical anisotropy values were obtained by using a Seibersdorf unit and radially scanning the coatings with a 4 x 24 μm slit. Note the trend toward higher anisotropies as the coating rates decrease. Compare with Fig. 5-2.

Fig. 5-10. OPyC coating failure of ThO₂ BISO particles versus (a) apparent coating rate and (b) optical anisotropy (BAF_0) for coatings of various densities. Particles were irradiated as unbonded samples in capsules P13R and P13S at $\sim 1075^\circ\text{C}$ to fast neutron fluences of 10.7 to $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

Fig. 5-11. Performance of OPyC coatings on ThO₂ BISO particles irradiated in P13R and P13S fuel rods. (a) High-density coatings deposited at medium to high coating rates; (b) low-density coatings deposited at high coating rates; and (c) low-density coatings deposited at low coating rates.

Fig. 5-12. Difference between sink-float and bulk density of fertile particle OPyC coatings as a function of coating rate

Fig. 5-13. Capsule P13S cell 1 decalibrated control thermocouple temperature versus fast neutron fluence. Also shown is the peak LHTGR fuel temperature envelope expected for load-following transients.

Fig. 5-14. Release of Cs-137 versus fissile particle failure level calculated from R/B values

6. IMPLICATIONS FOR LHTGR FUEL DESIGN AND PERFORMANCE

6.1. FUEL PARTICLE IRRADIATION PERFORMANCE

One of the primary goals of capsules P13R and P13S was to evaluate the performance of LHTGR reference-type fissile and fertile fuel particles during irradiation to LHTGR peak design conditions. A requirement of fuel performance derived from core design is that fuel meeting all requirements of particle design with no defective coatings shall retain at least one intact structural coating layer (i.e., retain gaseous fission products) on 99.5% of all fuel particles at the 95% confidence level after irradiation at 1250°C to 8.0×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} and/or 78% FIMA (fissile) or 7.5% FIMA (fertile) for times up to 4 yr (Ref. 6-1).

The irradiation performance of fissile particle batches with coating designs representative of the design requirements envisioned for the LHTGR is summarized in Table 6-1. The particle batches exhibited excellent stability during irradiation to peak burnups and fast neutron fluences which exceeded the LHTGR peak design exposure by 34 to 50%. Visual examination and postirradiation fission gas release measurements revealed that no coating failure occurred in these particle batches with an upper 95% confidence level of 0.2%. These particle batches also exhibited good performance in bonded rods; however, the failure levels could not be evaluated quantitatively. These results demonstrate that the fuel performance requirement discussed above is achieved with TRISO fissile particles having coating parameters representative of LHTGR design requirements.

These results are also consistent with LHTGR fuel performance models presented in Ref. 6-1. Methods for prediction of fuel performance given in Ref. 6-1 do not allow an evaluation of fuel failure levels at the severe

fast neutron exposures achieved during the capsule P13R and P13S irradiation. However, these performance models predicted the particle batches given in Table 6-1 would have exhibited a failure level of 0.5% after reaching a fast neutron exposure of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. These results indicate the performance models given in Ref. 6-1 are clearly conservative in this case.

Fertile particles with coating parameters representative of LHTGR design requirements also exhibited excellent irradiation performance under the severe conditions of capsules P13R and P13S, as shown in Table 6-2. No coating failure was observed during the visual examination, and all but two of the particle batches had OPyC coatings that were completely impermeable to the release of gaseous fission products. Two of the batches exhibited levels of fission gas release that were comparable to an in-service failure level of 0.2%. These results were also consistent with fuel performance models which predict a failure level of 0.5% for these particle batches after irradiation to a fast neutron fluence of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.

The excellent irradiation performance of these fissile and fertile particle batches indicates that high levels of fuel particle integrity can be achieved during irradiation to peak LHTGR conditions by proper specification of the fuel particle design parameters and illustrates the conservative nature of the fuel performance models presented in Ref. 6-1.

6.2. PyC COATING BEHAVIOR

The OPyC coating on HTGR fuel particles is one of the most important components of the fuel. In the fissile TRISO design, the OPyC performs several functions. It provides mechanical support to the SiC coating, which is the primary load-bearing layer; it protects the SiC from external impurity attack; and it also provides a means to mechanically bond the particles in the rod with the matrix. The OPyC on a BISO particle is, of course, the primary containment for fission gases and thus its survival is essential in preventing high release into the reactor coolant.

Results from the P13R and P13S capsules have provided a sound basis for a product specification for the OPyC layer. The recommended limits for coating rate, anisotropy, and density are presented in Table 6-3. These results draw heavily upon the fissile particle tests, because with the TRISO design it is less difficult to interpret the results of PyC irradiation stability since parameters which influence the pressure vessel performance can be separated out in the analyses. This is not true with the BISO design since stress levels in the OPyC depend upon fission gas pressure as well as the irradiation-induced structural changes. However, it seems logical to conclude that the coatings exhibiting optimum performance in the TRISO design also will perform best in BISO particles--provided they are sufficiently impermeable to fission gases.

Data from these tests tend to support the above supposition and show that OPyC coatings having very low anisotropies with densities in the range of 1.85 to 1.95 g/cm³ will survive to peak fluence at 1250°C with essentially no failure. Coatings with densities as low as 1.80 g/cm³ appear to have adequate performance on TRISO particles. However, the permeability of fission gases through coatings of these densities needs to be further investigated before the coatings can be recommended for use on the BISO particle.

Tests are now under way in capsules P13T (Ref. 6-2) and P13V (Ref. 6-3), which are designed to verify at the 95% confidence level that particles with these attributes will survive peak design conditions with ≥99.5% of the particles intact. Demonstration of this goal is one of the primary objectives of the capsule irradiation program.

6.3. FUEL KERNEL MIGRATION

Fuel kernel migration is one of four phenomena that control fuel particle performance during irradiation at high temperatures (Ref. 6-1). The HTGR core thermal design limits kernel migration to an acceptably low rate that will not result in significant coating degradation and subsequent

coating failure. The same techniques that are utilized in core thermal design to predict the extent of fuel kernel migration were applied in the thermal analysis of capsules P13R and P13S. There was excellent agreement between the predicted migration distances and the distances observed during the metallographic examination. These results give a high degree of confidence in the kernel migration coefficient design curves and the technique used in core thermal design to evaluate the extent of kernel migration during normal and transient core operation.

6.4. FUEL ROD IRRADIATION PERFORMANCE

The criteria used for evaluating fuel rod performance are as follows (Ref. 6-4):

1. To maintain integrity to full HTGR lifetime exposure.
2. To exhibit dimensional change compatible with the graphite fuel block so that no detrimental fuel-graphite mechanical interaction occurs.
3. To have bonding characteristics that do not cause fuel failure as a result of matrix-coating interaction.
4. To provide a heat transfer medium that will avoid excessive temperatures and thermal gradients in the fuel. Thermal conductivity shall be at least 4 Btu/hr-ft-°F to end of design life.

The postirradiation integrity of the bonded fuel rod specimens tested in capsules P13R and P13S was excellent and demonstrated that the performance criteria of integrity and bonding characteristics noted above can be achieved using a variety of processing conditions and matrix components. Very little irradiation-induced damage was observed even in fuel rods thermal cycled during irradiation or irradiated to a fast neutron fluence of 12.4×10^{21} n/cm² (E > 0.18 MeV)_{HTGR}, which is 55% beyond the peak LHTGR design fast

neutron exposure. The different fuel rod matrix and shim materials tested exhibited satisfactory performance, and no evidence of interaction between the fuel rod matrix and fuel particle coatings was observed.

These results are consistent with previous capsule tests (Ref. 6-4) which indicate that the integrity of pitch-bonded fuel rods is independent of irradiation temperature and fast neutron fluence within the realm of LHTGR operating conditions.

6.5. FUEL ROD DIMENSIONAL CHANGE BEHAVIOR

Characterization of irradiation-induced fuel rod dimensional change is referenced for the proper thermal and mechanical design of LHTGR fuel elements. An evaluation of the radial clearance between the fuel rods and the fuel holes is necessary for core thermal analyses. Also, irradiation-induced fuel rod dimensional changes must be compatible with the dimensional change behavior of the graphite fuel elements (GLCC Grade H-451) to ensure that excessive stresses are not built up as a result of dimensional interferences between the fuel rod columns and graphite fuel element during reactor operation.

Data on irradiation strain of H-451 graphite obtained from capsule irradiation tests have been statistically fit to produce dimensional change contours used in core design. These contours are a function of mean irradiation temperature and fast neutron fluence (Refs. 6-5 and 6-6). The irradiation strain data obtained from fuel irradiated in capsules P13R and P13S are compared with graphite dimensional change curves in Fig. 6-1. For comparison of fuel rod with graphite dimensional changes, graphite strain contours for temperatures approximately 200° to 300°C below the fuel centerline temperature should be used.

Fuel rod diametral changes appear to be quite compatible with those of the graphite (radial direction) to and slightly beyond peak HTGR fast neutron fluences (i.e., 8×10^{21} n/cm², E > 0.18 MeV). Since a 0.08- to 0.012-mm

radial gap typically exists between the rods and the fuel element block, interference between these two components would not be expected until fast neutron fluences in excess of $\sim 10^{22}$ n/cm² (E > 0.18 MeV)_{HTGR} are reached, as was the case for capsule P13R and P13S rods.

As noted earlier in this report, the diametral dimensional changes measured for the P13R and P13S fuel rods were approximately 45% less than predicted using current codes. A similar discrepancy had been noted from other tests of LHTGR rods (Ref. 6-7). An empirically derived correction factor is now being used in core design calculations, and this factor takes into account these P13R and P13S dimensional data (Ref. 6-4).

Data supplied for use in core design calculations have treated the rods as behaving isotropically. The results from these tests and from Peach Bottom test element irradiations (Ref. 6-8) have shown that fuel rods exhibit anisotropic dimensional changes, and at high fast fluences net expansion in the axial direction occurs. The cause for this anisotropic behavior has not been defined from these tests; however, recently reported results from a screening test of candidate materials indicate the graphite shim could be a contributing factor (Ref. 6-9). A detailed study of results from all of the most current tests of LHTGR fuel rods is now being undertaken to more thoroughly characterize the dimensional change mechanism. Input to core design calculations will be modified to reflect the anisotropic behavior of the rods.

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TABLE 6-1
PERFORMANCE OF LHTGR REFERENCE TYPE FISSILE PARTICLES

Batch Number	Number of Particles Tested(a)	Pressure Vessel Failure(b) (%)	95% Confidence Interval(c), P (%)	R/B Failure Level(d) (%)
6151-00-035	459	0	$0 \leq P \leq 0.8$	0
6151-00-035	809	0	$0 \leq P \leq 0.6$	0
6151-08-015	<u>688</u>	<u>0</u>	<u>$0 \leq P \leq 0.7$</u>	<u>0</u>
Total	1956	0	$0 \leq P \leq 0.2$	0

(a) Particle samples were irradiated to fast neutron fluences of 10.7 to 12.0×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} and burnups of 72 to 74% FIMA at temperatures ranging from 975° to 1075°C.

(b) Determined by visual examination.

(c) The 95% confidence statement for the failure levels determined by visual examination means that an infinite population of particles with attributes of these batches tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(d) Calculated from postirradiation fission gas release data by assuming a fractional release of 2×10^{-2} for failed fuel in an unconstrained geometry.

TABLE 6-2
PERFORMANCE OF LHTGR REFERENCE TYPE FERTILE PARTICLES

Batch Number	Number Particles Tested (a)	Pressure Vessel Failure (b) (%)	95% Confidence Interval (c), P (%)	R/B failure Level (d) (%)
6542-02-020	1149	0	$0 \leq P \leq 0.5$	0.2
6542-02-030	1072	0	$0 \leq P \leq 0.5$	0.2
6542-09-010	948	0	$0 \leq P \leq 0.5$	0
6542-18-015	953	0	$0 \leq P \leq 0.5$	0
4252-06-010	872	0	$0 \leq P \leq 0.6$	0
6542-23-025	992	0	$0 \leq P \leq 0.5$	0
6542-24-015	<u>942</u>	<u>0</u>	<u>$0 \leq P \leq 0.5$</u>	<u>0</u>
Total	6928	0	$0 \leq P \leq 0.1$	0.06

(a) Particle samples were irradiated to fast neutron fluences of 11.3 to 11.9×10^{21} n/cm² ($E > 0.18$ MeV)_{HTGR} and burnups of 4.1 to 4.9% FIMA at temperatures ranging from 895° to 1010°C.

(b) Determined by visual examination.

(c) The 95% confidence statement for the failure levels determined by visual examination means that an infinite population of particles with attributes of these batches tested to these conditions will have a failure level at the 95% confidence level bounded by this interval.

(d) Calculated from postirradiation fission gas release data by assuming a fractional release of 2×10^{-2} for failed fuel in a constrained geometry.

TABLE 6-3
RECOMMENDED OPyC COATING ATTRIBUTE SPECIFICATIONS FOR HTGR FISSILE TRISO AND FERTILE BISO PARTICLES

Fuel Particle	Property (units)	Inspection Method	Acceptance Criteria	
			Mean	Critical Region
Fissile: (200 μm) UC_2 TRISO	Density (g/cm^3)	Liquid density gradient	1.80 - 1.95	≤ 1.75
	Optical anisotropy (BAF_o)	Seibersdorf unit oil immersion, 4 x 24 μm slit, radial scan	≤ 1.03	≥ 1.04
	Coating rate ($\mu\text{m}/\text{min}$)	Calculation (coating thickness $\div$ coating time)	$3.0 \leq x \leq^{(a)}$	≤ 2.0
Fertile: (500 μm) ThO_2 BISO	Density (g/cm^3)	Liquid density gradient column	$1.85^{(b)} - 1.95$	$\leq 1.80^{(b)}$
	Optical anisotropy (BAF_o)	Seibersdorf unit oil immersion, 4 x 24 μm slit, radial scan	≤ 1.04	≥ 1.05
	Coating rate ($\mu\text{m}/\text{min}$)	Calculation (coating thickness $\div$ coating time)	$4.5 \leq x \leq^{(a)}$	≤ 3.0

(a) The upper limit for the coating rate was not defined from these tests.

(b) The lower limit and critical region need to be defined by coating permeability measurements.

Fig. 6-1. Irradiation strain of H-451 graphite, (a) radial and (b) axial, as a function of fast fluence compared with measured P13R and P13S fuel rod dimensional changes

7. SUMMARY

Capsules P13R and P13S demonstrated the integrity of LHTGR reference type fresh and alternate fuels over a wide range of irradiation conditions. Evaluation of these results with respect to specific fuel properties provided a substantial data base for the LHTGR Fuel Product Specification and performance models used in core design studies. A brief summary of the irradiation results is presented below:

1. Bonded fuel rods were successfully irradiated to fast neutron exposures of $12.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}, which is 55% beyond the LHTGR peak design fast neutron fluence of $8.0 \times 10^{21} \text{ n/cm}^2$.
2. Thermal cycling to high temperatures did not adversely affect fuel rod integrity.
3. Postirradiation integrity of fuel rods fired at 1500°C or in a nitrogen atmosphere was comparable to that of fuel rods fired under the LHTGR reference conditions of 1800°C in an argon atmosphere. These results contribute to the qualification of a lower firing temperature and nitrogen cover gas.
4. Irradiation-induced diametral fuel rod dimensional changes were compatible with the dimensional change behavior of LHTGR near-isotropic moderator graphite.
5. Particle batches with coating designs representative of the design requirements envisioned for the LHTGR exhibited excellent irradiation performance. A number of fissile and fertile particle batches were irradiated without coating failure to fast neutron exposures which exceeded the LHTGR peak design exposure by 34 to 51%.

6. A large number of particle batches were tested with coating designs or particle attributes that were not representative of LHTGR design requirements. The irradiation performance of these samples provided important data required to establish fuel particle specification limits.
7. Fuel particles with severely faceted coatings, nonround fuel kernels, or interrupted SiC layers exhibited higher failure rates than similarly designed fuel particles that did not have these attributes.
8. Survival of OPyC coatings is dependent on anisotropy over a wide density range. However, once low OPyC coating anisotropies have been achieved, it becomes necessary to restrict the range of OPyC coating density to ensure survival at high irradiation temperatures.
9. In-pile thermal cycling to temperatures up to 1650°C resulted in increased fuel failure levels relative to fuel failure levels in fuel specimens that were not thermal cycled. The effects of thermal cycling on fuel particle integrity were the most pronounced in fuel particles having out-of-specification coating parameters. Fertile particles more representative of LHTGR design requirements exhibited failure levels well below the level predicted for BISO ThO₂ particles that experience a thermal transient to 1550°C after irradiation to a fast neutron fluence of $8.0 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}.
10. A number of BISO fertile particle batches were irradiated to fast neutron fluences up to $12.1 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR} without any apparent coating failure. However, these batches exhibited varying levels of permeability to gaseous fission products (up to an R/B of 2×10^{-3} for Kr-85m at 1100°C). The OPyC permeability was attributed to a combination of the intergrowth feature porosity after deposition and an increase in the surface-connected porosity due to irradiation-induced cracking of the PyC microstructure.

11. Measured UC_2 fuel kernel migration distances were in good agreement with migration distances predicted with techniques and design data used in core thermal design studies to evaluate the extent of fuel kernel migration in the LHTGR core during normal and transient operation.
12. An addition of 1.0 wt % thorium to UC_2 kernels to improve the kernel sphericity did not alter the performance of the fuel relative to kernels without the thorium addition.
13. Deletion of the seal coat previously applied to fertile particle buffer coatings was shown to have no effect on coating irradiation performance.

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