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THORIUM FUEL CYCLE DEVELOPMENT PROGRESS REPORT

JUNE 1970

NO. 11

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CHEMICAL TECHNOLOGY DIVISION
METALS AND CERAMICS DIVISION

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AUGUST 1970

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THORIUM FUEL CYCLE DEVELOPMENT PROGRESS REPORT

June 1970

No. 11

Compiled by

The Staff of the Chemical Technology and
Metals and Ceramics Divisions

INTRODUCTION

The thorium fuel cycle development at ORNL is directed almost solely at HTGR fuels. These fuels consist of large blocks of graphite containing coolant channels and fuel and blanket holes. The fuel and blanket are made of microspheres of uranium or thorium compounds separately, or of mixtures of them in a single microsphere. The microspheres are coated with layers of pyrolytically deposited carbon and in some cases silicon carbide. The microspheres are retained in the holes in the graphite blocks in a bonded state.

Development work on all aspects of HTGR fuel recycle is in progress at ORNL. In addition, a major recycle development facility, the Thorium-Uranium Recycle Facility (TURF), has been built at ORNL, and the Coated Particle Development Laboratory (CPDL) has been put into operation in Building 4508. TURF is intended to be used as a development facility for fuel recycle. The CPDL is for engineering development studies leading to design of the equipment to be used in TURF.

I. HEAD-END REPROCESSING DEVELOPMENT

(R. E. Blanco, W. E. Unger)

The objective of this program is to evaluate head-end processes for converting irradiated HTGR fuels to a form suitable for recovery and decontamination of the thorium and uranium by a solvent extraction process. Small samples of irradiated and unirradiated fuel are studied to determine

effects of irradiation on fuel reprocessing steps and to correlate with metallographic studies. An important objective is the determination of the amount of breakage of particle coatings and the resultant amount of cross-contamination of the fertile Th-^{233}U and fissile ^{235}U components in alternative reprocessing steps. Mechanical systems are being developed for degrading the fuels and providing a material suitable for use in studies of the burn-leach steps using fluidized-bed or fixed-bed burners. The mechanical and burn-leach engineering development work is carried out using full-scale unirradiated fuel of the type to be used in PSC, and is designed to provide scale-up data for use in the design of pilot or full-scale processing plants.

1. Studies with Irradiated Fuels

(V. C. A. Vaughen, J. H. Goode, G. Davis)

1.1 Procurement of Irradiated Dragon Fuels

As mentioned last month, we have been in negotiations with the UKAEA for the procurement of 9 irradiated Dragon compacts of the $(Th/U)C_2$, ThC_2, UC_{10}, UC_{20}, and $(U/Th)O_2$ varieties, and 7 unirradiated controls for our head-end studies. It now appears that the unirradiated controls will be received in July, but that the irradiated compacts cannot possibly be received before August, and perhaps may arrive even later.

1.2 Further Studies with Irradiated Dragon Compact #8375

Compact No. 8375 is a sample of "standard" Dragon fuel which we have reported on earlier.[1] We have now completed our hot cell studies with it. TRISO coatings on the particles in this compact are buffer pyrolytic carbon/silicon carbide/isotropic pyrocarbon. The coating thicknesses are 27 μm/17 μm/64 μm, respectively. Using ^{137}Cs analyses, the compact was calculated to have been irradiated to approximately 15,000 Mwd/ton. The purposes of our hot cell tests were (1) to test sawing and various methods of compact breakage to confirm that crusher jaw opening was the controlling factor in the breakage of the SiC-coated particles and (2) to measure ^{85}Kr and ^{3}H releases during head-end reprocessing steps. In the course of these studies, other data on leaching, fission product behavior,

[1] Chem. Technol. Div. Ann. Progr. Rept. May 31, 1969, ORNL-4422, p. 117 (Table 3.5) (October 1969).

and losses to residues were obtained. Although the experiments with irradiated Dragon Compact No. 8375 are now complete, not all of the data have been analyzed.

The hollow-cylinder compact was sawed into two rings. One ring was further sawed into 4 sections of about equal size. All saw dust was brushed off the cut edges and collected for leaching. Two of the four sawed sections were burned as large pieces (static bed) and the whole bead fraction and fines residues were combined and leached for breakage. The final two sawed sections were recrushed at 1/8-in. jaw opening, burned in a static bed and leached (without grinding) for breakage.

The other half cylinder was crushed into 9 large pieces in the 5/8-in. opening jaw crusher. Three samples (R5, 6, 7) were burned in a fluidized bed with Al_2O_3, ground to −60 mesh, burned again in a static bed, and leached to obtain off-gas ^3H and ^{85}Kr releases in each step and to obtain the uranium and thorium content, fission product content, and residue losses.

Crushing Studies with Irradiated Dragon Fuel

The hot cell crusher (DFC Model 20830) has a nominal 5/8-in. jaw opening and can be shimmed to smaller openings. For the standard crushing procedure, the jaw opening was shimmed to give a product size distribution using machined, hollow graphite cylinders, similar to the laboratory crusher shimmed to a nominal 1/8 in. (a good size for our small-scale burner tests).

With the jaws wide open, the compacts are crushed to a few large lumps. For irradiated compact No. 8375, the half compact ring broke into 9 pieces; consequently, no size distribution can be defined. However, two of the sawed sections were recrushed at 1/8-in. jaw opening and analyzed by screening. The results are given in Table 1, with some earlier data for comparison. The important points to note are: (1) the fueled compacts release a larger fraction +42-mesh material (free beads) than that obtained from machined cylinders, (2) except for some variability in the amounts of +4 and +9 mesh material, the distributions are quite similar.

Table 1. Crushing Size Distributions in the Hot Cell Crusher at 1/8-in. Jaw Opening

	% of Total on Screen of Size Shown					
	+4	+9	+20	+42	+100	−100
Machined cylinders of graphite (no fuel)	∿9	∿60	12.6	6.6	4.9	6.9
Unirradiated Compact #8232	2.6	62.6	9.1	18.1	2.9	4.8
Irradiated Compact #8368	5.9	48.0	12.2	22.8	3.8	7.3
Irradiated Compact #8375						
R-3	18.3	44.1	11.5	17.6	3.9	4.6
R-4	9.1	52.4	12.9	18.0	3.1	4.5

The amount of broken particles present in this compact was estimated at 2.3% from static bed burn-leach tests on two sawed segments (R-1, R-2). The burner product was leached twice with 11 M HNO_3--0.1 M Al--0.05 M HF without grinding the SiC-coated particles, and compared to the prorated amount of U and Th present in the crushed fuel samples as determined by analyses of fractions R5, 6, and 7.

The results are presented in Table 2. Results from studies of other Dragon compacts are included for comparison purposes.

Compact 8375 confirmed that the jaw crusher opening was a significant variable in the breakage of fuel particles. In general, the fines (−42 mesh) were representative of the compact composition, indicating no great preference in cleavage planes between fuel and matrix. With unirradiated fuels (and #8368) significantly fewer particles were ruptured, indicating that the cleavage was around (rather than through) the particles dispersed in the hot pressed matrix.

About 2.3% of the fuel particles were probably cracked before crushing since hacksawing and the static bed burning are not expected to fracture particles inside the fuel matrix, and the sawed surfaces were brushed free of broken fines before burning.

Table 2. Comparison of Two Size Reduction Techniques, Crushing and Hacksawing, as Related to Dragon Fuel Particle Breakage (Irradiated Fuels)

	Fraction of Sample Reduced to Fines (%)	Fraction of Loose Broken Particles (%)	Fraction of Other Broken Particles (%)	Total Fraction of Broken Particles (%)
Compact 8375 (5/8-in. jaw)	1.4	∿1.0	2.3	3.3[a]
Compact 8375 (1/8-in. jaw)	8.1[a]	∿8.1[c]	∿5.5[c]	13.6[a]
Compact 8375 Hacksaw (1/16-in. blade)	7.3[b]	6.7	2.3	9.0
Compact 8368 (1/8-in. jaw)	10.6	3.9[a]	3.7[a]	7.6[a]

[a]Average of two determinations.

[b]Includes 0.2% +42-mesh material (intact particles).

[c]Estimated.

Crushing to 5/8-in. jaw opening produced a −42-mesh fines fraction containing 1.4% of the sample weight and about 1% of the calculated uranium and thorium present in the half cylinder. Crushing to a nominal 1/8-in. jaw opening produced a fines fraction averaging 8.1% of the sample weight. After burning in a static bed and leaching the residue (without grinding), 13.6% of the calculated fuel particles present were broken. These results indicate that a significant number of fractured particles were retained with the larger lumps (+42 mesh) of fuel. Breakage due to crushing to 1/8-in. jaw opening is estimated at 11.3%, of which 28% was retained above the +42 mesh.

Crushing to 1/8-in. jaw opening is necessary for fluidized-bed burning studies in the small-scale burner. In a full-scale burner, the fuel would be crushed to larger particles, with less breakage.

Release of ^{3}H and ^{85}Kr During Head-End Reprocessing Studies

Three samples of fuel crushed in the 5/8-in. opening jaw crusher (R-5, 6, 7) were put through the complete burn-grind-burn-leach flowsheet. In each step, a purge of either 4% H_2 in argon, or moist oxygen-nitrogen mixtures was used to provide adequate water for trapping the ^{3}H on molecular sieves. The off-gases passed over a hot copper oxide bed to convert all hydrogen to water before trapping. The entire gas output was collected and sampled for direct counting of ^{85}Kr at the end of each run.

The results are given in Table 3. About 1% of the total ^{85}Kr was collected in the first burner off-gas, representing a release of ^{85}Kr from broken particles. There was some variability in the percents released during grinding and second burning; however, the sum of these two steps average 98% release. The remainder was released on dissolution. The residual ^{85}Kr released during dissolution (1-2%) probably indicates that the burning conditions were not optimum for complete removal in this step.

Tritium release data are also given in Table 3. As noted before,[1] a large fraction of the total tritium (∿30-40%) exists outside the SiC coated particles and is released in the first burning. This tritium is

Table 3. Release of ^{3}H and ^{85}Kr in Head-End Steps (HDC 8375)

	^{85}Kr			^{3}H		
	R-5 (%)	R-6 (%)	R-7 (%)	R-5 (%)	R-6 (%)	R-7 (%)
1st burn O. G.	0.4[a]	1.04	0.98	32.3	b	29.7
Grind O. G.	27.6	81.1	16.7	0.3	-	5.6
2nd burn O. G.	71.7	15.7	81.3	60.6	-	62.0
1st leach O. G.	0.3	2.1	1.0	3.8	-	2.2
1st leach liquor	-	-	-	3.0	-	0.4
Total dpm	4.13×10^{10}	3.34×10^{10}	3.32×10^{10}	7.45×10^{8}	5.44×10^{8}	1.80×10^{8}
Sample wt., g	16.35	14.55	11.11	16.35	14.55	11.11

[a]Collected on dry ice-charcoal trap. CO_2 displaced some of the ^{85}Kr. Subsequent samples were taken from the collected gases.

[b]Percent values are not realistic because second burn off-gas ^{3}H value is very low.

believed to come from neutron reactions with impurities in the bonding matrix and ^{3}He in the coolant gas. The ternary-fission-produced tritium is retained within the SiC coating and is released mainly in the second heatup and burning step. Residual tritium in the second burner product (2.6 to 6.8%) was recovered during leaching. As with the ^{85}Kr, the residual tritium probably indicates a lack of optimum burner conditions.

2. Head-End Engineering Studies
(C. D. Watson, R. S. Lowrie)

The head-end engineering studies are comprised of two principal areas of investigation - "mechanical" and "burn-leach." The general approach followed has entailed the mechanical dissection, by sawing, of full-sized graphitic blocks containing fuel, followed by comminution of size fractions suitable for the burn-leach process. However, the recent GGA decision to use bonded fuel sticks rather than loose particles as the reference form has necessitated changes in the approach.

Work on the head-end engineering studies program has resumed. Efforts during this report period continued on rewriting the National HTGR Recycle Program Plan to reflect changes which have occurred in the overall HTGR program.

II. REFABRICATION DEVELOPMENT

1. Particle Preparation
(R. G. Wymer - Coordinator)

The fuel material of primary interest for ORNL recycle studies is the ThO_2-UO_2 particle, which has a thorium-to-uranium ratio of about 4.2. The uncoated fuel particles are to be microspheres 350 $\pm$ 100 μ in diameter, made by the sol-gel process. Fuel preparation includes development and demonstration of all process steps involved in making remotely the ThO_2-UO_2 microspheres. The steps include demonstrating reliable, remote processes for reproducibly mixing $Th(NO_3)_4$ and $UO_2(NO_3)_2$ solutions in the desired thorium-to-uranium ratio, preparing the mixed, stable ThO_2-UO_3 sol in concentrations exceeding 1 g-mole of oxides per liter, forming ThO_2-UO_3 gel microspheres, and converting them to dense ThO_2-UO_2 in good yield.

1.1 Sol and Microsphere Preparation Development (P. A. Haas)

Experimental engineering studies of processes and equipment for preparation of sols and microspheres are reported here. The present emphasis is on processes, procedures, and prototype equipment for preparation of oxide microspheres in the Thorium-Uranium Recycle Facility (TURF). Tests of flowsheets and procedures to be used for test materials preparation are also included.

Preparation of Gel Spheres Without Fluidization

The properties of n-hexanol are intermediate to those of 2-ethyl-1-hexanol and isoamyl alcohol, and n-hexanol has given good ThO_2 gel spheres when used for mass transfer studies. Therefore, the use of n-hexanol was tested in the 10-ft-high, non-fluidized sphere forming column. The solubility of water in n-hexanol is about 6 mg/cc. For a 2 M ThO_2 sol, 400 and 500 micron sol droplets give good gel spheres while 600 micron sol droplets give slightly distorted gel particles in dry n-hexanol and fall out incompletely gelled in n-hexanol containing 3 mg H_2O/cc. In the same column, ThO_2 sol droplets of 300 microns distort in isoamyl alcohol and those of 400 microns fall out incompletely gelled when 2-ethyl-1-hexanol is used. For 1 M CUSP UO_2 sol, the n-hexanol does not result in any great increase in the maximum diameter of gel spheres; 300 micron sol droplets give gel particles with a deep dimple or pit. The experimental results agree with mass transfer calculations for n-hexanol with respect to which conditions result in incomplete gelation.

2.0 Fueled Graphite Fabrication Development

(F. J. Furman, W. H. Pechin, and J. D. Sease)

We are developing processes and equipment for the refabrication of HTGR fuel as detailed in the National HTGR Recycle Development Program Plan. The fuel consists of microspheres of thorium and/or uranium as the oxide or carbide and coated with multiple layers of pyrolytic carbon and silicon carbide. The particles are bonded together into fuel sticks and these sticks are located into hexagonal graphite logs which contain both fuel and coolant holes.

Our work is divided into particle coating, particle handling and inspection, particle blending, fuel stick making, fuel element assembly of the Recycle Test Element (RTE) segments. We are also starting the modifications of the prototype remotely operable coating furnace.

2.1 Particle Coating

After analysis of the particles prepared for the RTE's showed that the coatings were acceptable, we began decontaminating the prototype remote coating furnace.

The decontamination of the prototype remotely operable coating furnace, necessary because of the ^{238}Pu contamination of the recently completed RTE fuel, is proceeding well. The furnace has been completely disassembled and the furnace enclosure cleaned sufficiently that gas masks are no longer necessary. Double coveralls and shoe covers are still required but only as a precaution to prevent spread of alpha active material.

Some disruption to our normal operations is being caused by construction of an alpha laboratory in the area surrounding the coating equipment. However, we plan to take advantage of this disruption by modifying some problem areas of the coating system. We plan to install a fluidized bed particle cooling hopper which should considerably accelerate cooling the particles compared to the previous water-cooled hopper. Also, we will install a critically safe effluent system designed to be easier to handle in the hot cell and have more reliable

gas-tight connections. In addition, we will correct some defects in the exhaust pumping system. Under certain conditions water could spill down the exhaust line and clog the absolute filters. Also, the weather protection (heat tapes and insulation) proved to be unreliable and a handicap when repairs are needed. The water will be trapped out of the line and an insulated shed built over the pumping system.

2.2 Fuel Element Assembly

The loading of the two loose particle RTE segments was completed. These segments were loaded with mixtures of coated (Th—20% U)O_2, coated ThC_2, and graphite flour. This particle mixture is the reference recycle fuel; however, the reference fuel is bonded in the fuel sticks while this fuel is loaded as loose particles.

The three types of particles were loaded concurrently from three Syntron vibrators on which were mounted v-trough feeders. Filling rate was approximately 1 in./min. After all eight holes of the fuel segment were loaded 65 g steel cylinders were placed on top of each hole to hold the particle bed in place and the segment was vibrated by a Syntron vibrator to settle the bed. The method of mounting the segment onto the vibrator is shown in Fig. 1. A glass dummy fuel hole is mounted on the side and filled with low density ($\sim$2.4 g/cm^3) coated particles and uncoated ThO_2. After extensive vibration only a small area at the top of the column appears to have partially segregated. This demonstrates the practicality of settling loose beds by vibration.

After the initial settling, additional graphite flour was added and infiltrated to fill the bed with graphite. (Loading of all the graphite during the initial feeding causes the bed to expand beyond the volume required for the particles.)

When the filling was completed the beds were sealed in place by the addition of 2 cm^3 of methyl-ethyl-ketone + 10% polystyrene. After the sealing mixture dried the segments were checked for contamination and shipped to Gulf General Atomic for loading into Peach Bottom fuel elements. A dummy hole for each segment was filled with the identical mixture as the segment holes and retained at ORNL for future reprocessing cold tests prior to testing the irradiated segments.

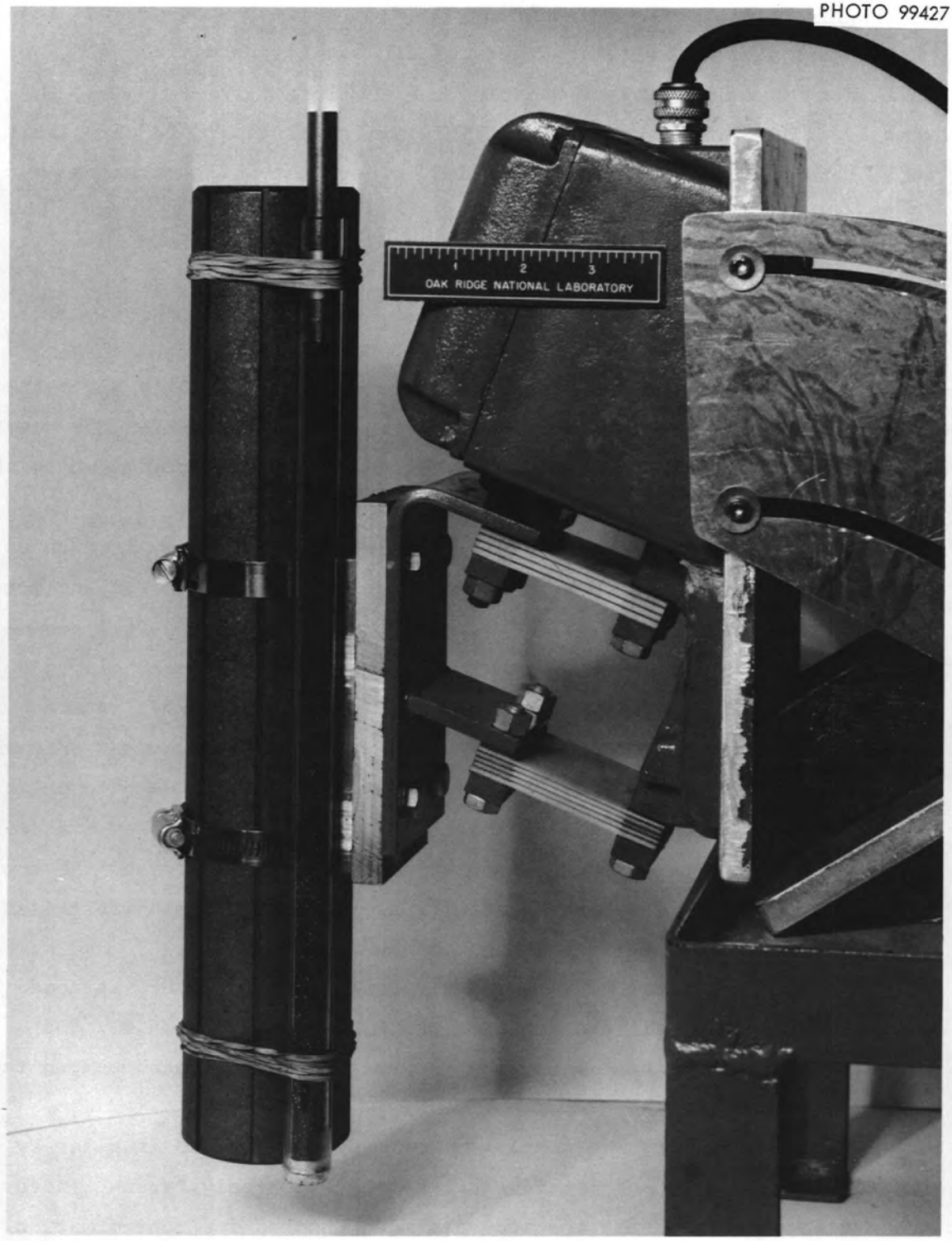

Fig. 1. RTE Segment Vibrator.

2.3 Particle Inspection

The equipment for preparation of solid samples for analysis on the MS-7 spark-excited mass spectrometer is now in operation so samples of the fuel loaded into the RTE's will be submitted for impurity analysis on this machine.

2.4 Development of Bonded Beds of Coated Particles for HTGR Fuel Elements

(J. M. Robbins)

This report period has been devoted entirely to the production of fuel sticks for the Recycle Test Elements (RTE's), to the fabrication of the spine samples that will be tested with the RTE's in Peach Bottom, and to the preparation of other irradiation specimens. The required fuel sticks and spine samples were completed and shipped to GGA. The quantities of fuel sticks prepared are given in the next section.

We have started fabrication of specimens for HFIR tests HRB-2 and HT-6. These experiments are designed to test new filler materials (other than Poco graphite) and the new resin-derived fuel particles. Fillers to be tested are shown in Table 1. All compositions will be prepared with 15V pitch as the binder.

Table 1. Filler Materials to be Tested in HRB-2 and/or HT-6

Filler	Amount Used, Wt. %	Size Flour, μm
Poco[a]	35	−40
Thermax[b]	50	−40
Santa Maria[c]	40	−40
Robinson[d]	40	−40
JOZ[e]	40	−40
Asbury[f]	35	−40
6353[g]	27	−40
H-378[h]	35	−27

[a]An isotropic graphite flour from Poco Graphite, Inc.

[b]A soft spherical carbon black from R. T. Vanderbilt Company.

[c]Graphite flour from Collier Carbon and Chemical Company.

[d]An air-blown graphite flour from Carbon Products Division, UCC.

[e]Graphite flour from Great Lakes Carbon Corporation.

[f]Natural flake graphite from Asbury Graphite, Inc.

[g]Natural flake graphite from Charles Pettinos Graphite, Inc.

[h]An isotropic graphite from Great Lakes Carbon Corporation.

III. Material Irradiation

(A. L. Lotts, T. N. Washburn, J. D. Sease, and J. H. Coobs)

1.0 HTGR Recycle Fuels Irradiation

(T. N. Washburn, R. B. Fitts, and A. R. Olsen)

The irradiation tests on the HTGR recycle program have two main objectives: (1) to provide irradiated fuel for head-end process studies, and (2) to provide irradiation proof tests of the products of coated particle process development for the Thorium-Uranium Recycle Facility. The test conditions of interest include fuel temperatures between 600 and 1300°C, burnup to 20% FIMA in the $(Th,U)O_2$ particles, and fast fluence exposures up to 8×10^{21} neutrons/cm^2.

The first two stages in this program are being implemented this year. They are: (1) accelerated burnup data capsule irradiations, and (2) eight test fuel elements to be irradiated in the Peach Bottom Reactor.

1.1 Capsule Irradiation

(A. R. Olsen)

The fuel performance in an HTGCR is a complex interaction of fuel burnup and fast neutron irradiation damage to the pyrolytic carbon coatings on the fuel and the carbonaceous bonding material. The relative proportion of burnup and fast neutron damage effects is determined by the time and temperature of irradiation and the neutron flux energy spectrum. In a commercial HTGR the flux energy spectrum is such that the thermal neutron flux and the fast (> 1 Mev) neutron flux are essentially equal. In the available positions in thermal flux test reactors such as the ORR, ATR, and ETR, the thermal flux is normally 4 to 12 times the fast flux. Therefore, we have designed a capsule which incorporates thermal neutron filter shrouds to reduce the thermal to fast flux ratio by a factor of 2.6 to more nearly simulate the HTGR conditions. These shrouds are fabricated from

material obtained from the Bettis Atomic Power Laboratory.[1] The capsule design is shown schematically in Fig. 2. Two capsules are being fabricated for irradiation in the ETR starting in the first quarter of FY-1971. Four fissile-fertile particle combinations will be irradiated at four temperatures to fast fluence levels between 6 and 8×10^{21} neutrons/cm^{-2}. The loadings for the two capsules are described in Table 2.

These capsules will provide small quantities of irradiated fuels for chemical reprocessing studies in advance of the material from the Peach Bottom Reactor irradiations and samples for evaluation of the irradiation performance characteristics. Additional capsule tests will be made only as the remote reprocessing and fuel fabrication reaches critical stages of development, such as testing of the remote coating furnace product.

In the design work physics calculations have been delayed during the acquisition of cross-section data for hafnium. These data are now available and are currently being reduced to the 36 group cross-sectional sets we use for most ANISN capsule calculations. The ETR personnel have obtained new flux monitor data for us for the J-8 position. These data indicate peak thermal flux (2200 neutrons/sec) of 6.42×10^{14} neutrons/cm^2-sec and a measured peak fast flux (E > 1.0 Mev) of 3.31×10^{14}. These flux measurements will be used together with the ANISN calculations to establish the desired hafnium content in the thermal flux shields.

1.2 Large Scale Irradiation

(R. B. Fitts)

Eight Recycle Test Elements (RTE's) will be operated in the Peach Bottom Reactor with the first six starting during this month. The fuel elements contain samples of fuels proposed for use in 1100 Mw(e) HTGR's. All of the fuel for the six elements for the initial insertion has been completed and shipped to GGA where the elements have been completed. The

[1] Richard M. Lieberman, "A System for Calculation of Fission Power in Irradiation Tests," National Symposium on Development in Irradiation Testing Technology, CONF-690910, pp. 473–490 (1969).

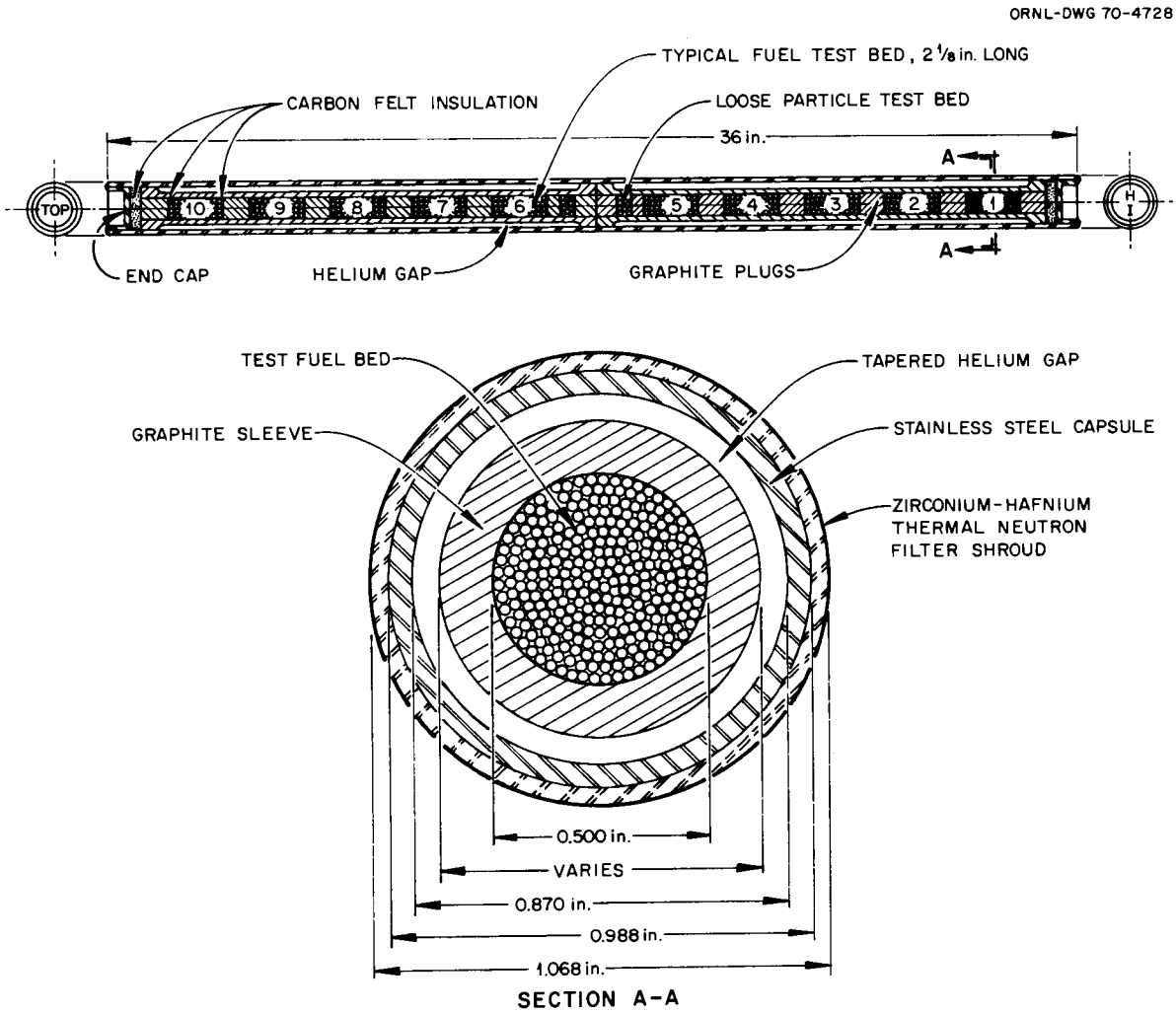

Fig. 2. Schematic of Proposed HTGR Recycle Irradiation Capsule.

Table 2. Fuel Loadings for HTGR Irradiation Test Capsules H-1 and H-2

Position	Capsule H-1		Capsule H-2	
	Temperature (°C)	Particle Mix[a]	Temperature (°C)	Particle Mix[a]
1	1050	f	950	f
2	1050	g	950	g
3	1050	c	950	c
4	1050	a	950	a
5	1050	f	950	f
LP[b]	1050	a	950	a
LP[b]	1050	x	950	x
LP[b]	1300	a	750	a
6	1300	f	750	f
7	1300	a	750	f
8	1300	c	750	c
9	1300	g	750	g
10	1300	g[c]	750	g[c]

[a]Particle mix: a = (4.2 Th, U)O_2 BISO plus ThO_2 BISO
c = (2 Th, U)O_2 BISO plus ThO_2 BISO
f = UC_2 TRISO plus ThC_2 BISO
g = UO_2 BISO plus ThO_2 BISO
x = (2Th, U)O_2 from carbon containing sol process (nonspherical)

[b]LP are loose particle containers for extensive metallographic examination and performance analysis (approximately 1-in.-long beds)

[c]Blended beds (all others are bonded beds)

final collection and correlation of the fabrication data on this fuel is in progress.

All of the fuel for the remaining two elements, RTE's 1 and 3, has been completed and shipped to GGA except for 102 in. of fuel sticks made of UO_2 and ThO_2 coated particles. This is a new batch of fuel necessitated by a recent change in the fuel element loading plan. The present plan is given in Table 3. These last fuel sticks are scheduled for delivery at GGA by September 30, 1970. The last two RTE's are scheduled for insertion in Peach Bottom Reactor around November 1970.

Table 3. RTE Loading Combinations*

Position of Fuel Body in Element	Center-Line Temperature Range (°F)	1-year Irradiation		2-year Irradiation		3-year Irradiation			
		RTE 1†	RTE 2	RTE 3†	RTE 4	RTE 5	RTE 6	RTE 7	RTE 8
6 (Top of reactor)	1950–2050	bdei	f	bdei	e‡	bdei	g	acfg	f
5	2050–2250	bdeh	a‡	bdeh	d	bdeh	f	acfg	i
4 (Maximum flux)	2250–2300	acfg	e	acfg	f	acfg	f	bdgh	d
3 (Maximum flux)	2050–2250	acfg	f‡	acfg	a‡	acfg	d	bdgi	e
2	1650–2050	acfg	d	acfg	f‡	acfg	f	bdgi	i
1 (Bottom of reactor)	1050–1650	acfg	e‡	acfg	e	acfg	e	bdgi	f‡

*Loadings are indicated by letters a, b, c, d, e, f, g, h, and i, and are defined in Table 2.

†To be inserted in the reactor approximately 150 days after the core II startup; all others inserted at core II startup.

‡Blended beds.

1-year elements will see a maximum dose of $\cong 1.4 \times 10^{21}$ neutrons/cm^2.

2-year elements will see a maximum dose of $\cong 2.8 \times 10^{21}$ neutrons/cm^2.

3-year elements will see a maximum dose of $\cong 4.2 \times 10^{21}$ neutrons/cm^2.

IV. URANIUM-233 REPROCESSING

(J. R. Parrott, R. G. Nicol, W. A. Shannon)

ORNL serves as a national distribution center for ^{233}U. The facility, which contains a small batch leacher, a batch dissolver, and a single-cycle solvent extraction system, is capable of purifying ^{233}U at the rate of 25 kg per week. It includes storage systems with capacities of 500 kg ^{233}U as nitrate solution and 455 kg of ^{233}U as solid oxide having a density of 1.0 g/cc.

1. Dissolution and Purification

Revisions to the thermosyphon oxide dissolver and associated equipment were made to meet the requirements of the Nuclear Safety Review Committee. These revisions included: (1) installing a solution sampler, (2) disconnecting pipe lines from the dissolver and storage tanks to the feed adjustment tank, (3) replacing leaking valves, (4) installing a locked switch to the air supply for the air-operated valve in the pipe line draining the dissolver, and (5) recalibrating the dissolver vessel.

One dissolution of oxide, amounting to 4900 g of uranium was completed. The contents of eight cans were added and the dissolver was sampled after each addition to determine the dissolution rate. The final concentration was 400 g/liter U and 0.0 M HNO_3 which makes the material ideal for feed to the ion exchange system after dilution to 100 g/liter U.

2. Storage and Distribution

The facility presently contains 291 kg of ^{233}U, which varies in isotopic purity between 84 and 98% and ^{232}U content between 4 and 250 ppm. One shipment of 6,418 g of ^{233}U, as oxide, was made during the month. One shipment of 34 containers, containing a total of 28.5 kg of ^{233}U as oxide, was received for the LWBR support program from Savannah River. The material was stored in the dry wells in the penthouse until it can be dissolved.

Besides the solution stored for the LWBR program (in tank U-3), the facility contained 123.5 kg ^{233}U as $UO_2(NO_3)_2$ solution in two 400-liter storage tanks (U-1 and U-2).

The empty storage tanks (U-4 and U-5) were recalibrated and the presence of the neutron absorbers (glass Raschig rings) was confirmed by the ultransonic inspection technique. The calibration solution (water) is being boiled to concentrate the U that redissolved after having deposited on the glass rings.

The facility also contains a solution storage system (TRUST-Thorium Reactor Uranium Storage Tank) in which 1047 kg of highly enriched uranium (76.5% ^{235}U, 9.7% ^{233}U) in the form of a uranyl nitrate solution are stored. This material is the uranium product from the Indian Point reactor fuel, which was purified by solvent extraction at the Nuclear Fuels Services Plant. The solution will be stored indefinitely since the ^{232}U content (120 ppm) prohibits its direct refabrication into fuel elements, and the low ^{233}U content makes it of little interest for reactors demonstrating the thorium fuel cycle.

The sampling of this solution will be delayed until lead shielding is installed around the off-gas filter.

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