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GULF GENERAL ATOMIC

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10288

HTGR BASE PROGRAM
QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
AUGUST 31, 1970

Prepared under
Contract AT(04-3)-167
Project Agreement No. 17
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

September 30, 1970

GULF GENERAL ATOMIC COMPANY, P.O. BOX 608, SAN DIEGO, CALIFORNIA 92112

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GA-9660-June, 1969, through August, 1969
GA-9815-September, 1969, through November, 1969
GA-9944-December, 1969, through February, 1970
GA-10088-March, 1970, through May, 1970

INTRODUCTION

This report covers the work performed by Gulf General Atomic Incorporated under U.S. Atomic Energy Commission Contract AT(04-3)-167, Project Agreement No. 17. This Project Agreement calls for preliminary design and related exploratory development of a 1000-MW(e), gas-cooled, nuclear power reactor incorporating advanced technology, and a 150-300 MW(e) prototype. This program is based on the concept of the High-temperature Gas-cooled Reactor (HTGR) system developed by Gulf General Atomic.

The related exploratory development effort is aimed at a 1000-MW(e) gas-cooled reactor plant that could be placed in operation in the 1970's, with operation of the prototype at an earlier date. The plant would demonstrate such new design features as:

1. Breeding or near breeding of fuel in a high-temperature, gas-cooled, thermal reactor using a Th-U²³³ fuel cycle.
2. Advanced reactor fuel elements, which may include new and simpler configurations and new methods of processing and fabricating to extend fuel lifetime and reduce fuel costs.
3. Alternative plant arrangements to reduce plant capital costs, including such possibilities as alternative materials (e.g., concrete pressure vessels) and the assembly of the reactor and steam-generating equipment in a single containment.

Physics, metallurgical, chemical, electronic, and analytical design and development engineering programs support the primary reference design objective.

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Task IV
FISSION-PRODUCT MECHANISMS

FISSION-PRODUCT TRANSPORT IN HTGR SYSTEMS

FIPER Code Analyses

The FIPER I code was developed to permit the description of fission-product transport in HTGR systems (Ref. 1). The volatile metal atoms released from fuel particles diffuse through the surrounding fuel rod matrix into the graphite and then traverse internal surfaces and defects into the coolant hole (Ref. 2).^{*} The primary assumption in FIPER I is one-dimensional geometry, which permits fast computer solution of the diffusion equation by iterating the corresponding finite difference equations in one space and one time dimension.

The development of the FIPER II code is proceeding and at this time works for cylinder geometry. The space- and time-dependent concentration profiles and the diffusion currents are found by solving the following equation:

$$\frac{\partial c(\underline{r}, t)}{\partial t} = S(\underline{r}, t) - \lambda c(\underline{r}, t) - \text{div } \underline{J}(\underline{r}, t) \quad , \quad (1)$$

where c is the concentration, S is the source, and $\underline{J}$ the current, t the time, and λ the decay constant. To make the mathematical problem given by Eq. (1) a definitive one, the interface and boundary conditions must be stated. They depend on the geometry of the system. The cylinder geometry shown below was chosen:

^{*}References will be found at the end of each task.

The geometry is given by a two-region cylinder oriented in the z -direction and bounded at $z = 0$ and $z = L$. The interface between region 1 and region 2 is given by $r = R_1$ and the outer boundary is given by $r = R_2$.

At the interface, the mass current is continuous,

$$J_r^- = J_r^+ \quad . \quad (2)$$

The continuity of the chemical potential,

$$\mu^- = \mu^+ \quad , \quad (3)$$

yields a relation between the concentrations at the interface,

$$\frac{c^-}{c^+} = \phi(z) \quad . \quad (4)$$

Here, ϕ is a distribution coefficient that is in general determined by the adsorption isotherms in regions 1 and 2.

The diffusion coefficients in regions 1 and 2 have the values D_1 and D_2 , respectively, and the boundary conditions are as follows:

At $r = R_2$, the current normal to the surface vanishes.

At $z = 0$, the concentration is a constant, c_0 , independent of r and t .

At $z = L$, the concentration vanishes.

According to these boundary and interface conditions, the following physical picture can be obtained. At time $t = 0$, there are no metal atoms inside the cylinder. At $t > 0$, the atoms diffuse from the circular boundary at $z = 0$ into the cylinder. Eventually, some atoms reach the circular surface at $z = L$, and the release current at this boundary is determined under the condition that the outside presents a perfect sink. The cylinder consists of region 1 having a small cross section and a large diffusion coefficient, and region 2 having a large cross section and a small diffusion coefficient. Region 1 represents a pore or some other defect region characterized by fast diffusion. Region 2 represents graphite with inner surfaces in the form of grain boundaries along which the slow diffusion takes place. The length of the cylinder is of the order of 1 cm, whereas the radius of the inner region is of the order of 10^{-3} cm, corresponding to the radius of a large defect. The outer radius, $R_2 = 10^{-1}$ to 10^{-2} cm, is determined by the number of defects per cubic centimeter. It may be emphasized that the geometry is flexible and that more than two regions can easily be taken into account. Furthermore, release currents can be calculated for different geometries, and each geometry can be weighted with a distribution characteristic for a particular graphite. The adequate parameterization of reactor graphites is difficult, but it is necessary for accurate answers to fission-product-release problems.

For the first model calculations of the concentration profiles and the release currents, the following geometry has been chosen:

$$\begin{aligned} L &= 0.5 \text{ cm,} \\ R_1 &= 0.0015 \text{ cm, and} \\ R_2 &= 0.037 \text{ cm.} \end{aligned}$$

The interface condition (4) is given by $\phi = 1$; $0 \leq z \leq L$. The boundary concentration at $z = 0$ is $c_0 = 2.28 \times 10^{17}$ Sr atoms/cm³ = 0.379 moles of Sr/cm³. The diffusion coefficients are given by

$$\begin{aligned} D_1 &= 1.5 \times 10^{-5} \text{ cm}^2/\text{sec,} \\ D_2 &= 2 \times 10^{-8} \text{ cm}^2/\text{sec.} \end{aligned}$$

These parameter values are chosen in a first and crude attempt towards a resemblance of the experimental situation of a strontium diffusion experiment (see earlier quarterly report GA-9815). Some results for the time development of the concentration profiles are shown in Figs. 4.1 through 4.3. In Fig. 4.1, the concentration at time $t = 10$ sec is plotted on a logarithmic scale. The origin at $r = 0$ and $z = 0$ is indicated by a dot, and the arrows indicate the positive r and z directions. Regions 1 and 2 are of equal size in this plot; in other words, region 1 is enlarged in the r direction. It is seen that the concentration in region 1 is independent of r , the reason being that the time $t \gg R_1^2/D_1$. In region 2 the concentration rapidly decreases in the r as well as the z direction, the decrease becoming even more pronounced at a later time as can be seen in Fig. 4.2. Eventually the concentration profile in region 2 becomes flat in the r direction (see Fig. 4.3); the characteristic time is $t = R_2^2/D_2$. The release rates from regions 1 and 2 are plotted in Fig. 4.4. In this particular case, the release rate from region 1 dominates at all times the release from region 2. A slight decrease of the diffusion coefficient in region 1 changes this picture. In Fig. 4.5 the release rates are plotted for the same geometry and for the same value of D_2 , but $D_1 = 10^{-5} \text{ cm}^2/\text{sec}$. At time $t = 10^4$ sec, the release from region 2 becomes larger than the release from region 1, because of the larger cross section of region 2.

In summary, the FIPER II code is intended to make possible the numerical evaluation of time-dependent concentration profiles and release currents in two space dimensions for highly anisotropic geometries and for coupled regions with diffusion coefficients that differ by many orders of magnitude. Currently the code must be brought into a more flexible form with regard to geometry and boundary conditions. Then an adequate parameterization of reactor graphite from a combined experimental and computational effort may be attempted. This effort should lead to more accurate determination of metallic fission-product release from HTGR systems.

Fig. 4.1. Concentration profile of Sr at time $t = 10$ sec

Fig. 4.2. Concentration profile of Sr at time $t = 10^3$ sec

Fig. 4.3. Concentration profile of Sr at time $t = 2 \times 10^5$ sec

Fig. 4.4. Release rates from regions 1 and 2 as a function of time
for $D_1 = 1.5 \times 10^{-5} \text{ cm}^2/\text{sec}$ and $D_2 = 2 \times 10^{-8} \text{ cm}^2/\text{sec}$

Fig. 4.5. Release rates from regions 1 and 2 as a function of time for $D_1 = 10^{-5} \text{ cm}^2/\text{sec}$ and $D_2 = 2 \times 10^{-8} \text{ cm}^2/\text{sec}$

VAPORIZATION OF STRONTIUM FROM GRAPHITE

Two additional experiments on the vaporization of strontium from 37- to 74- μm grain H-327 graphite have been completed using the mass spectrometric-Knudsen cell technique. The samples were initially loaded with strontium in the form of the nitrate. Each sample, contained in a closed tantalum cell, was heated to 1350°C for 3 hr to remove oxygen. The sample was then loaded into the Knudsen cell in a dry box and quickly transferred to the mass spectrometer.

The strontium was vaporized from the cell over an extended period of time resulting in a decrease in strontium concentration of one to two orders of magnitude. The data consist of heat and entropy data calculated from many consecutive Arrhenius plots, each obtained at nearly constant concentrations. The linear dependences of these two quantities on $\log C$ are sufficient conditions for Freundlich behavior (see earlier quarterly report GA-9372). The family of isotherms may then be expressed in the form

$$\log (P/P_0) = A + B \frac{10^3}{T} \log (C/C_0) \quad , \quad (5)$$

where P is in atmospheres, C is in mg Sr/g graphite (37 to 74 μm), and T is in °K.

The parameters for these two experiments, and also for the first experiment (which was reported in GA-9372), are given in Table 4.1. The differences in these parameters for the various experiments reflect experimental error in taking the measurements. The actual experimental reproducibility becomes more clear on plotting the isotherms as shown in Fig. 4.6. The actual measurements covered the ranges from about 1000° to 1800°K and 5 to 0.003 mg Sr/g C. The measured pressures were in reasonably good agreement, but the derived functional dependences lead to some divergence in the extrapolated curves.

Fig. 4.6. Strontium partial pressure over 37- to 74- μ m grain H-327 graphite

One other experiment had been performed but was not included here because of a suspicion that kinetic limitations were involved (see the previous quarterly report, GA-10088). The average rate of strontium loss from the Knudsen cell was larger than for the other experiments. The derived isotherms showed only slightly worse agreement; the isotherms extrapolated toward low concentration were somewhat lower than those given in Fig. 4.1.

Below 1450°K the upper bound to the extrapolated pressures at low concentration are the values given by experiment 1. These are the same values (with a slight correction) as given in GA-9372. Above this temperature, the extrapolated pressures at low concentration for experiment 5 become the larger.

The current experiment is on cesium sorption on H-327 graphite. This system is being investigated as a prelude to the initiation of studies of adsorption in mixed metallic systems.

TABLE 4.1
PARAMETERS FOR THREE STRONTIUM VAPORIZATION EXPERIMENTS

Exp. No.	Log P_o	Log C_o	A	B
1	1.92	3.02	-0.51	4.52
5	0.36	2.42	-1.24	5.58
6	1.43	2.21	-0.84	5.70

DIFFUSION OF STRONTIUM IN GRAPHITE

Work on strontium permeation experiments 4370-143 and 4370-144 has been resumed. The graphite crucibles, which had been taken out of the furnaces previously to carry out modifications and repairs on the system, were reinserted and the anneals continued (see earlier quarterly report GA-10088 for a description of the modifications). The strontium released during the first 300 hr was collected and is presently being analyzed.

CALCULATION OF Cs-134 AND Cs-137 CONCENTRATION PROFILES IN FUEL ELEMENT GRAPHITE

An attempt has been made to mathematically simulate the diffusional transport of Cs-134 and Cs-137 in fuel element graphite. This has been done in order to explain the similarity of concentration profiles for these two isotopes of cesium in Peach Bottom fuel element graphite (see Ref. 3). One would expect the concentration profiles for the two isotopes to be somewhat dissimilar because Cs-134 has a much shorter half life than Cs-137 and because Cs-134 is formed by neutron activation of Cs-133, whereas Cs-137 is a fission product.

The mathematical treatment simulates decay of Cs-137 and Cs-134 and the in situ birth of Cs-134 from diffusing Cs-133, where Cs-133, Cs-134, and Cs-137 have the same diffusion parameters and where the amount of these isotopes available to diffuse is essentially linearly increasing at the surface facing the reactor fuel. This was done by adaptation of the Korts and Norman approximation system for diffusion in a sphere (Ref. 3). The diffusion coefficient for cesium in the graphite sphere was taken as 10^{-8} cm/sec. The simulated reactor was operated steadily for 1.6×10^6 sec causing a linearly increasing fission-product source, and the half life for Cs-137 was taken as 1×10^8 sec and for Cs-134 as 2.25×10^6 sec. Of course, none of these numbers are exact, and they were selected strictly for demonstration purposes. The birth rate of Cs-134 from Cs-133 was chosen to be equal to the decay constant of Cs-137. This rate should not make an appreciable difference unless it represents an appreciable conversion of Cs-133.

Figure 4.7 shows the result of this calculation. There is a moderate difference between the calculated curves. The difference is shown graphically by a plot of the ratio of concentrations. Other calculations exhibiting flatter profiles (as a result of using larger diffusion coefficients) showed similar differences.

Fig. 4.7. Calculated concentration profiles for Cs-137 and Cs-134 in graphite in simulated reactor operation

Before this calculation was made, it was not known which of the following two effects would predominate: (1) steepening of the Cs-134 profile with respect to the Cs-137 profile because the decay of the Cs-134 is much faster than the decay of Cs-137 or (2) flattening of the Cs-134 profile with respect to the Cs-137 profile because of the higher neutron dose experienced by the older Cs-133. It appears from this calculation that the latter effect exceeded the former for the given conditions.

References

1. Appel, J. C., and B. W. Roos, Nucl. Sci. Eng. 34, 201 (1968).
2. Vanslager, F. E., W. E. Bell, O. Sisman, and M. T. Morgan, "Fission Product Transport in HTGR Systems - A Summary," Proceedings of the ORNL Gas-cooled Reactor Information Meeting, Oak Ridge, Tennessee, April 27-29, 1970, CONF-700401-1; also USAEC Report GA-10073, Gulf General Atomic Incorporated, April 22, 1970.
3. "40-MW(e) Prototype High-temperature Gas-cooled Reactor Postconstruction Research and Development Program, Quarterly Progress Report for the Period Ending July 31, 1969," USAEC Report GA-9494, Gulf General Atomic Incorporated, August 29, 1969.
4. Korts, R. F., and J. H. Norman, "A Calculational Model for Condensed-state Diffusion-controlled Fission-product Absorption During Fallout Formation," USNRDL Report GA-7598, OCD Work Unit 3111A, General Dynamics, General Atomic Division, January 10, 1967.

Task V
RECYCLE FUEL STUDIES

RECYCLE TEST ELEMENTS

Test Element Design

The fuel loading of recycle test element 7 was modified since the previous quarterly report (GA-10088); the changes are the new loadings shown in Table 5.1 (see also Table 5.2). The change was made in bodies 1, 2, 3, and 4. Fuel rods manufactured at Oak Ridge National Laboratory were substituted for Gulf General Atomic rods since the ORNL rods were considered representative.

Test Element Fabrication

Six of the eight planned recycle test elements have been fabricated, accepted by ORNL, shipped, and inserted with Peach Bottom Reactor Core 2; the remaining two (1 and 3) are scheduled for insertion in early 1971. Recycle test elements 1 and 3 are scheduled for one-year and two-year irradiation, respectively. All of the elements are designed to provide irradiated fuel for use in HTGR fuel reprocessing studies.

The fuel rods for the recycle test elements were manufactured both at Gulf General Atomic and at Oak Ridge National Laboratory. The Gulf General Atomic fuel rods, which were manufactured using production equipment and techniques, consist of closely packed beds of blended fissile and fertile coated fuel particles bonded together with a carbonaceous matrix.

Table 5.1
RTE LOADING COMBINATIONS^a

Position of Fuel Body in Element	Centerline Temp Range (°C)	First Discharge		Second Discharge		Final Discharge			
		RTE 1 ^b	RTE 2	RTE 3 ^b	RTE 4	RTE 5 ^b	RTE 6	RTE 7 ^b	RTE 8
6 (top of reactor)	2050-1950	bdei	f	bdei	e*	bdei	g	acfg	f
5	2250-2050	behd	a*	behd	d	behd	f	acfg	i
4 (max flux)	2250-2300	acfg	e	acfg	f	acfg	f	bdgh	d
3 (max 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*

^aLoading compositions are indicated by letters a through j as defined in Table 5.2. The fuel compositions marked with an asterisk (*) are blended beds; all others are bonded rods.

^bIn these elements, each pair of adjacent fuel holes contains one of the four fuel compositions indicated.

Table 5.2
RTE LOADING COMPOSITIONS^a

Composition	Fissile	Fertile
a	(Th,U)O ₂ ^b	None
b	UO ₂	ThC ₂
c	(Th,U)O ₂ ^c	ThO ₂
d	(Th,U)O ₂ ^c	ThC ₂
e	UC ₂	ThC ₂
f	UC ₂ TRISO	ThC ₂
g	UO ₂	ThO ₂
h	UC ₂	ThO ₂
i	UC ₂ TRISO	ThC ₂ TRISO

^aUnless otherwise indicated, all particles are BISO coated.

^bTh/U = 4.2.

^cTh/U = 2.0.

The matrix used for the completed test elements was a blend of 27%-30% natural-flake graphite flour and 73%-70% coal tar pitch. The matrix was prepared by hot mixing the ingredients to form a homogeneous blend, then cooling the mixture to room temperature and grinding the solid matrix into granules suitable for use in the fuel-rod injection equipment. The impurities in the matrix are kept in the low parts per million range.

The closely packed bed of coated fuel particles is blended in a manner similar to the blending of blended beds described in the previous quarterly (GA-10088), although bed length was limited to two inches for proper matrix injection.

The hot matrix was injected into the particle bed with a prototype injection machine. After cooling, the fuel rods are removed from the injection die and are referred to, at this point in manufacture, as "green" rods. Before further reprocessing, the green rods are subjected to a quality control check for size, fuel homogeneity, fuel loading, integrity, and external appearance.

After the preliminary inspection the green rods are carbonized and heat treated to convert the green matrix into fully carbonaceous matrix. Two processes were used on the recycle test element fuel rods to carbonize and heat treat them. The rods used in the quartered elements were carbonized and heat treated in individual graphite holders. The green rods used in the single combination fuel bodies were loaded directly into the fuel bodies and carbonized and heat treated in place.

The fully heat treated rods that were processed in the individual holder were given a final inspection to ensure the requirements of the specification were met. Each fuel body that was loaded with green rods carried a fuel rod sample tube during carbonization and heat treatment. After heat treatment, the tube was cut open and the rods removed and inspected.

The qualified rods were loaded into fuel bodies. The green rods were loaded by pushing the rod stack directly from its plastic shipping tube into the graphite body as shown in Fig. 5.1. The fully heat treated rods were loaded singly into the fuel holes to prevent damage to the rod. All the rods were manufactured and then preassembled in stacks of the correct length to fill one fuel hole. This allowed more flexibility on the length of individual rods.

A thin cap of liquid polystyrene was poured and cured on the top of the fully heat treated rod stacks after loading to prevent shipping vibrations. A test specimen of this process is shown on Fig. 5.2.

The central hole of each fuel body was loaded with a solid graphite spine except for two bodies, RTE 5-2 and RTE 5-4, which also contained six center samples each. The center samples consist of graphite crucibles loaded with special fuel rods. They were provided for loading by ORNL.

The fully assembled fuel bodies were loaded into a standard Peach Bottom Core 2 fuel element sleeve assembly. This assembly consists of a noninstrumented graphite bottom connector that was silicon brazed to a graphite sleeve.

HT81711

Fig. 5.1. Loading of green fuel rods into fuel body

HT81704

Fig. 5.2. Fully heat treated rod stack showing polystyrene transit bonding

A graphite internal trap loaded with charcoal is inserted first into the sleeve. This is followed by a solid graphite cylinder called the lower reflector. The length of the cylinder determines the start of the active fuel in a particular element. The assembly is then loaded with six fuel bodies as shown in Fig. 5.3. The upper reflector assembly, a standard Core 2 element component, is threaded and cemented into the top end of the sleeve and completes the element assembly. During assembly of the various components into the sleeve, axial measurements are taken for accurate determination of the position of the various fuel bodies.

HTGR FUEL RECYCLE PLANT STUDY

Work has continued on the combined reprocessing-refabrication facility design and economic study. Facility designs for HTGR economies of 15,000, 30,000, and 45,000 MW(e) are being developed in this study. Block flow diagrams (see Figs. 5.4 through 5.12) and material balances have been completed.

The material balances are based on a fueling system having two element types and bonded-bed BISO-TRISO fuels. In this system, U-233 materials are separated from U-235 materials by putting all recycle uranium (U-233) in one type of fuel element and all makeup uranium (U-235) in another type. The makeup uranium is isolated from U-233 bred from thorium by adding a SiC coating on the U-235 particle. This coating will make the particle resistant to oxidation in head-end burning operations. The SiC coating also permits separation of uranium types should makeup and recycle fuel elements be mixed in reprocessing operations.

Fuel particles will consist of BISO coated ThC_2 fertile particles, TRISO coated UC_2 (U-235) fissile particles, and BISO coated $(\text{Th,U})\text{O}_2$ (U-233) mixed particles. U-233 is incorporated into the fuel elements as a thorium-diluted kernel. Since the fissile content of the U-233 stream will be about 60% to 70%, a thorium uranium ratio of 3:1 in the kernel is adequate to limit burnup to about 20% FIMA, thereby ensuring particle survival. U-235

HT81710

Fig. 5.3. Loading of fuel bodies into sleeve

Fig. 5.4. Graphite disposal flowsheet

Fig. 5.5. Uranium-233 and thorium recovery flowsheet

Fig. 5.6. Uranium-235 particle processing flowsheet

Fig. 5.7. Radioactive waste processing flowsheet

Fig. 5.8. Sol production flowsheet

Fig. 5.9. Fuel particle kernel production flowsheet

Fig. 5.10. Fuel particle coating flowsheet

Fig. 5.11. Recycle fuel element production flowsheet

Fig. 5.12. Scrap recovery and waste disposal flowsheet

particle survival is dependent on small kernel size and buffer carbon dilution. Thorium carbide particles have low burnup and therefore have no burnup survival problem. Carbide kernels for the thorium and the U-235 particles are employed as representative of current fabrication technology. U-233 fuel particles are sol-gel oxides.

In reprocessing spent fuel, a crush, burn, leach head-end followed by acid Thorex solvent extraction is used. Both makeup (U-235) and recycle (U-233) uranium are recovered and kept separated. This can be done either by using separate processing lines or by consecutive recovery campaigns. Methods for utilizing recycle thorium and the determination of the number of solvent extraction cycles necessary to prepare recycle uranium (U-233) for refabrication will also be studied. Although one solvent extraction cycle is probably adequate for neutron poison decontamination, additional cycles may be necessary for chemical decontamination for sol-gel feed. Radioactive decontamination is less important in U-233 recovery, because the buildup of gamma-emitting U-232 daughters during the refabrication of U-233 requires shielded facilities.

Refabrication of recycle U-233 is by the ORNL amine extraction-hexanol dehydration process. Only fuel elements containing recycle U-233 are fabricated in this facility. It is assumed that cold facilities are available for makeup U-235 fuel fabrication and for the production of ThC₂ fuel particles.

Previously developed equipment layout criteria drawings (Refs. 1, 2) and the new flow diagrams and material balances will be used to describe and size equipment. From the equipment layout criteria drawings and flow diagrams, building layouts will be developed and capital costs estimated for the facility and equipment. Recycle plant staffing and other operating expenses will also be determined.

Based on throughput considerations and capital and operating costs, financial studies using discounted-cash-flow - return-on-investment methods will be undertaken to determine economic facility size and startup times.

In addition to the basic design calculations, the study will be used to determine the effect of a range of plant sizes on costs and to guide process and equipment development work to be done under the National HTGR Fuel Recycle Development Program.

NATIONAL HTGR RECYCLE DEVELOPMENT PROGRAM PLAN

Review and revision of the National HTGR Recycle Program Plan continued during the quarter. The compositions of fresh and spent Fort St. Vrain fuel and of the 1000-MW(e) reactor fuel were supplied to ORNL together with reactor operation data sufficient for calculating the fission product and recovery element content of spent fuel.

The sections of the program plan revised by ORNL were reviewed.

HEAD-END PROCESSING

Fluid Bed Burning

Work with the 4-in. fluid bed burner continued during the quarter in three major areas: TRISO/TRISO crushed fuel feed, TRISO/BISO crushed fuel feed, and crushed graphite-only feed. In addition, several miscellaneous runs were carried out, and are also reported on here.

Crushed TRISO/TRISO Fuel Feed. Runs 33 and 35 were carried out in a last attempt to burn TRISO/TRISO Rezolin-bonded fuel rods without agglomeration. The fuel elements were crushed, mixed with alumina, and fed into the 4-in. fluid bed burner. Although it was possible to maintain operation for as long as 4 hr, temperature control became very difficult. When the results of the runs were analyzed, it was found that the SiC coats of the particles were covered with one layer of alumina grains. An example of this effect is shown in Fig. 5.13. Analysis of the material forming the agglomerating bond showed the presence of sodium. Microscopic examination showed the presence of a glass-like material that had at some time been molten.

M-32711-1

Fig. 5.13. Rezolin-bonded fuel particles covered with single layer of alumina grains after burning in 4-in. fluidized bed burner

It was inferred from this evidence that SiO_2 was formed in the burner and that this reacted with a sodium compound from the Rezolin to form sodium silicate.

These latest results, together with those reported previously, are sufficient evidence to conclude that the agglomeration during burning is connected with Rezolin and that trouble-free operation cannot be achieved when fuel rods bonded with this resin are burned.

Crushed TRISO/BISO Fuel Feed. TRISO fissile fuel particles and BISO fertile fuel particles bonded with pitch into rods is the reference fuel for large HTGR reactors. It is therefore the most important feed to a full-scale reprocessing plant.

Experiments have been carried out to determine the effect of feeding crushed TRISO/BISO pitch-bonded fuel rods to the 4-in. fluid bed burner. The rods were 2.75 in. in diameter and 15 in. long. They were prepared by mixing pitch, filler, and particles in a heated bowl and then tamping the mixture into hot graphite molds. The rods were then carbonized under the standard conditions.

Run 36 was an attempt to burn -1/4-in. crushed rods continuously. Operation was trouble-free from 100 min, at which point the temperature began to drop until combustion could no longer be maintained. Examination of the product and final bed gave the results shown in Table 5.3. It can be seen that there was an adequate fraction of burnable carbon present, so a shortage of combustible material was not the reason for the enforced shutdown. Two other factors could have been responsible: the moderately large fraction of coarse material, or the large fraction of fines.

Run 37 was done to test the hypothesis that there was too high a fraction of coarse material in the burner. The feed size was therefore reduced to -1/8-in. crushed rods. After 50 min of operation, combustion could not be maintained and the run had to be terminated. Microscopic examination revealed that the product and final bed consisted of some whole white spheres,

bits of broken white spheres, white powder, powdered matrix, fissile particles, and a large fraction of BISO particles in the process of losing their PyC coats by combustion. The last group ranged from particles that had not been burned at all to some that still had a thin perforated PyC shell through which could be seen the white oxidized kernel. It was concluded that the -250- μ m material, and not an overabundance of coarse material, was creating the combustion difficulties. The results of examining the product and final bed are shown in Table 5.3.

Table 5.3
RESULTS OF SCREENING AND BURNING THE PRODUCT
AND FINAL BED OF F4B-36 AND -37

Screen Size (μ m)	Product Taken During Run		Final Bed (4.8 Kg)	
	Weight (%)	Burnable Carbon in Size Fraction (%)	Weight (%)	Burnable Carbon in Size Fraction (%)
F4B-36:				
+ 869	0.8	81.4	36.0	94.5
-869 + 420	35.7	13.3	22.7	23.0
-420 + 250	20.0	5.3	6.4	19.5
-250	43.4	10.6	34.9	26.2
F4B-37:				
+869	10.0	88.4	34.8	60.5
-869 + 420	56.5	24.0	46.4	16.5
-420 + 250	21.7	13.8	14.2	16.6
-250	11.9	22.8	4.5	79.6

To test this theory, run 38 was carried out using a mixture of loose TRISO fissile and BISO fertile particles only as feed. From past experience, it was known that a mixture of TRISO particles will burn to the SiC coats with no operating difficulties, and produce a product containing less than 1% burnable carbon. It was reasoned that if ThO₂ dust causes combustion to cease, it could be observed better without the presence of graphite dust or particle agglomerates. The run proceeded well and burned out after two hours. Examination of the product and final bed showed that only 30% of the

possible ThO_2 had been released from the particles. The remainder was still in the form of fertile particles in various stages of PyC attack. The fraction of -250- μm material in the final bed was 21%. It was concluded that it is this fine material that causes combustion to cease.

To test this conclusion the +250- μm material from run 38 was used as the feed in run 39. Combustion ceased after 70 min and the final bed contained 36% -250- μm material. The +250 material from run 39 was used as the feed in run 40, and after 30 min of operation the final bed contained 26% -250- μm material. It was concluded from this series of runs that it is the fines that prevent the bed from burning down to a low concentration of carbon.

Run 40 was carried out to determine the effect of feeding graphite mixed with crushed fuel rods. A 50/50 mixture of -3/16-in. graphite and crushed pitch-bonded rods was used as feed and the run was started. Operation was trouble free for 153 min, so the feed was changed to a 50/50 mixture of -3/16-in. crushed rods and -1/4-in. graphite. During the following 75 min it was a continuous struggle to maintain operation and the run had to be terminated. The product samples, taken at intervals during operation, were screened on a 420- μm sieve and the results are given in Table 5.4. They show that there is a large fraction of -420- μm material in the product and that each sieve fraction contains a considerable quantity of burnable carbon. It was concluded from this run that the difficult operating period had been due to inadequate fluidization and that a once-through continuous system of burning, as in this run, is no use for plant operation because of the high concentration of carbon in the product. In order to burn the carbon concentration down to an acceptably low value, two methods are available:

1. Batch operation using an inert fluidizing medium and heating the wall of the reactor. An internal grate may be used. This arrangement allows oxidation to continue when the heat of combustion is insufficient to balance the heat-loss from the reactor and when the powder buildup would cause fluidization difficulties in the absence of the inert medium. The concentration of CO_2 in the off-gas indicates when oxidation is complete.

Table 5.4

RESULTS OF SCREENING AND TRAY BURNING SAMPLES OF PRODUCT
FROM RUN 41 IN THE 4-IN. FLUID BED BURNER

Sample No.	Fraction of -420- μ m Product In Sample (wt-%)	Burnable Carbon in the -420- μ m Sample (wt-%)	Burnable Carbon in the +420- μ m Sample (wt-%)
1	82.2	9.7	30.4
2	50.7	7.1	36.0
3	39.0	16.3	28.6
4	63.8	21.1	49.9
5	65.9	23.3	51.5
6	41.2	13.0	45.4
7(a)	79.4	30.7	49.6
8	80.0	19.7	39.5

NOTE: Feed was 50/50 mixture of -3/16-in. graphite and crushed TRISO/BISO pitch-bonded rods.

(a) The change of feed to -1/4-in. graphite plus -3/16-in. crushed rods was made when sample 7 was taken. Sample 8 was taken 30 min later. The change of feed appears to have made a small difference to the -420- μ m burnable carbon content.

- Continuous operation, screening the product on a 420- μ m sieve. The +420- μ m material is returned to the reactor, and the -420- μ m material is burned in a smaller fluidized bed that is operated batchwise, contains an inert medium, and is heated. The 420- μ m screen size allows the fissile particles to follow the main process stream.

Both these methods are being examined in order to obtain more details and to discover the relative advantages.

Run 42 was carried out to check that the +420- μ m material could be recycled through the 4-in. burner without difficulty. Four kilograms of +450- μ m product from run 41 was mixed with 2 kg of SiC-exposed fissile particles

and burned without difficulty in 70 min. The run was terminated when the heat loss was greater than the heat of combustion. The -420- μm product contained 6.5% carbon, and the final bed contained 4% carbon at the top, and 3.2% carbon at the bottom. It was concluded that recycle of +420- μm material presented no problems in the fluid bed.

Some of the -450- μm material from run 41 was batch burned (run 29) in the heated 2-in. fluid bed reactor. The SiC-exposed fissile particles behaved as the inert medium so there was no need to add, for example, alumina. The run was terminated after 90 min. The final bed, weighing 1695 g, contained 0.1% carbon. It was concluded that there is no difficulty in burning the carbon down to a low concentration. A second burning experiment was carried out in the 2-in. fluid bed but without the fissile particles present. The result was high local temperatures in the bed, hot spots on the reactor wall, and finally, a hole in the wall. It was concluded that it is essential to have a fluidizing medium present when BISO fuel particles are burned to a low carbon concentration.

Graphite-only Feed. Numerous runs were carried out in the 4-in. fluid bed to determine the important variables that affect the combustion of a graphite-only feed.

The results of runs 55 to 68 are shown in Table 5.5. It was found that a reasonable minimum bed weight is about 2 kg. A bed weighing less than this is subject to wide and rapid temperature fluctuations. The maximum feed size using the tabulated gas flows is 1/4 in.; the sieve analyses are given in Table 5.6. The duration of the runs with a controlled shutdown could have been extended indefinitely but was curtailed when the objective of the run had been attained. The temperature of operation is much higher than the 750°C reported by ORNL and BNL. The reason for this is not understood, but it may be the result of having no inert fluidizing medium present. It has been found that it is most difficult to revive combustion if the temperature drops below 850°C. In the early runs, the

Table 5.5
RESULTS OF BURNING GRAPHITE ONLY

Run No. F4B-	Feed Size (a) (in.)	Bed Weight (g)	Duration (min)	Temp Center Bed (°C)		Approx. Air Rate (l/min)	Approx. Oxygen Rate (l/min)	Over-all Burning Rate (g C/min)	Calculated CO Conc. in CO/CO ₂ (mole-%)	Approx. O ₂ Conc. in Off-gas (mole-%)	Remarks
				Low	High						
55	-3/16	460	55	930	>1020	200	18	24	---	2	Controlled shutdown
56	-3/16	760	38	930	980	150	42	29	---	1	Flameout (b)
57	-3/16	540	40	950	1000	170	25	24	---	3	Flameout (b)
58	-3/16	660	95	910	1020	160	32	32	---	2	Flameout (b)
59	-3/16	142	40	900	1000	(c)	(c)	25	---	2	Flameout (b)
60	-3/16	200	70	920	1040	70	48	41	---	1	Controlled shutdown.
62	-3/16	2050	45	920	1040	120	52	44	---	1	Controlled shutdown
63	-3/16	2020	80	950	1040	160	48	46	33	1	Controlled shutdown
64	-1/4	1840	70	930	1040	170	50	43	22	---	Controlled Shutdown
65 (d)	-1/4	---	---	---	---	---	---	---	---	---	---
66	-1/4	1500	100	880	1020	170	48	40	0	1	Controlled shutdown
67 (d)	-1/4	1500	---	---	---	---	---	---	---	---	---
68	-1/4	1880	50	940	1040	170	48	52	47	1	Controlled shutdown

(a) Sieve analysis given in Table 5.6.

(b) Temperature dropped below that at which combustion can be revived.

(c) Variable.

(d) Defluidized because of too high fraction of -1/4 + 3/16 graphite.

Table 5.6
SIEVE ANALYSIS OF FEED GRAPHITE

Screen Size	Feed Graphite (wt-%)				
	Runs 55-63	Run 64	Run 65(a)	Runs 66, 68(b)	Run 67(a)
-1/4 + 3/16 in.	0	23.08	70.00	30.00	40.35
-3/16 + 1/8 in.	18.52	14.25	5.56	11.22	9.57
-1/8 in. + 869 μ m	41.99	32.30	12.60	34.78	29.66
-869 + 525 μ m	10.63	8.18	3.19	8.42	7.19
-525 + 420 μ m	1.71	1.31	0.51	1.10	0.94
-420 + 327 μ m	4.98	3.83	1.49	3.58	3.05
-327 μ m	22.16	17.07	6.65	10.83	9.23

(a) Defluidized at 220 l/m STP.

(b) Coarsest mixture used with no difficulties.

air and oxygen flows were continually adjusted to maintain constant temperature, but in the later runs the air flow was kept constant and only the oxygen was adjusted. The overall burning rate has been increased to 52 g/min, which is in contrast to 3 kg/min required on a 15,000 MW(e) reprocessing plant. The CO/CO₂ concentration was calculated from the molar ratio of oxygen and carbon consumed. The oxygen concentration in the offgas varied during operation, but the general level indicated that the oxygen efficiency was high.

Miscellaneous Runs. Runs 44 to 47 were carried out to determine whether secondary oxygen fed in just above the bed would reduce the fraction of carbon fines carryover. The results showed very clearly that it made no difference at all, the carryover remaining at 8% to 11%.

Runs 48 and 51 showed that when fines collected from the cyclone were screw-fed into the bottom of the bed, all were consumed.

Discussion. The work done to date indicates that if the feed to the reprocessing plant is TRISO/TRISO (as it will be in TURF), then the conceptual flowsheet could be as illustrated in Fig. 5.14. The graphite and particle burners do not require an inert fluidizing medium because fluidization is satisfactory without it. When burning the fertile material after grinding, a fluidizing medium is required. This burning stage will also require heat input in order to burn the carbon down to a low concentration before leaching.

The results of work done on the TRISO/BISO bonded fuel has shown that the conceptual flowsheet could be as illustrated in Fig. 5.15. The particle burner requires a fluidizing medium to maintain fluidization, and for this purpose the +420- μ m material is recycled. The -420- μ m material also requires a fluidizing material and in this case, alumina is used.

Fig. 5.14. Reprocessing block diagram for TRISO/TRISO bonded fuel

Fig. 5.15. Reprocessing block diagram for TRISO/BISO bonded fuel

References

1. Shefcik, J. J., "Central Reprocessing Plant for HTGR Fuels," USAEC Report GAMD-8382, General Dynamics, General Atomic Division, October 16, 1967.
2. Stelling, H. E., "Central Refabrication Plant for HTGR Fuels," USAEC Report GAMD-9145, Gulf General Atomic Incorporated, March 15, 1969.

Task VIII
PHYSICS AND FUEL MANAGEMENT

HTLTR SUPPORT ACTIVITIES

Calculations using both Gulf General Atomic and ENDF/B cross-section data sets were completed for HTGR Lattice 2 (C/Th=200, C/U²³³=11000). The calculational results compare reasonably well with preliminary experimental data for this lattice. Reduced experimental data for this lattice are not yet available.

Reduced experimental data for HTGR Lattice 1 (C/Th=200, C/U²³⁵=6000) became available and indicated a larger temperature defect (1% to 2% in $\Delta k/k$) between room temperature and 1000°C than had been predicted by either Gulf General Atomic or Battelle Northwest Laboratory (see Table 8.1). A review of the theory used for the correction of the experimental data for leakage and flux-mismatch effects was initiated. Revised Gulf General Atomic calculations including up to 210- μ m-diameter ThO₂ particles indicated up to 1% in $\Delta k/k$ increase in k'_{∞} , but only insignificant changes in computed temperature defects. A review of the Gulf General Atomic calculations, including atom densities and UC₂ particle self-shielding effects, was initiated. Revised experimental data incorporating a corrected flux-mismatch effect was received from Battelle Northwest Laboratory. Although the revised experimental data is in better agreement with the Gulf General Atomic and Battelle Northwest Laboratory temperature defect calculations for Lattice 1, a discrepancy still remains.

FUEL TEST ELEMENT

The design of the fuel test element was discussed in detail in the previous quarterly report (GA-10088). The design work is complete, and the element graphite components are ready for assembly.

Table 8.1
HTGR LATTICE 1 RESULTS

Temp (°C)	k'_{∞}			
	HTLTR Experiment		BNWL Calc.	GGA Calc.
	Old Mismatch Correction	New Mismatch Correction		
20	1.111	1.102	1.093	1.094
150	1.093	1.085	1.078	1.080
300	1.076	1.072	1.065	1.064
550	1.049	1.044	1.048	1.046
700	1.038	1.034	1.042	1.037
1000	1.024	1.022	1.029	1.024

The fuel test element spine samples were fabricated and assigned positions within the element. These spine samples are of the general designs shown in Ref. 1 and include metallic diffusion, fission-product release, boronated graphite, and thermal stability samples, plus some advanced fuel rods. Because of difficulties in fabricating the rods originally planned for fuel test element 6, it was decided that the element would not be assembled until improved fuel rods are fabricated. The fuel element is now scheduled for insertion in the Peach Bottom core early in 1971. Photographs of the spine samples showing their proposed positions in the element are shown in Fig. 8.1.

Reference

1. Steward, K. P., "Objectives and Plans for Fuel Testing in the Peach Bottom HTGR," USAEC Report GA-10065, Gulf General Atomic Incorporated, April 23, 1970.

HT-81488

HT-81489

HT-81490

Fig. 8.1. Photographs of the spine samples showing their proposed positions in the element

Task IX
ADVANCED COATED PARTICLE FUELS

COATED PARTICLE IRRADIATIONS

A series of capsule irradiations is being conducted to evaluate both startup and recycle fuel systems for a large HTGR under the expected irradiation conditions. These tests are directed toward the evaluation of fuel particles that contain only uranium (diluent particles) and toward evaluation of coated oxide fuel. Emphasis is also being placed on irradiation testing at temperatures above the maximum in a large HTGR (1350°) to better define the operating limits of the fuel, and for safety analysis applications.

Capsule P13L

Capsule P13L is the second test of coated diluent particles and the first to full exposure (8×10^{21} n/cm², $E > 0.18$ MeV). Both diluent and fertile particles having both oxide and carbide kernels are being tested with BISO and TRISO coatings on the diluent particles and BISO coatings on the fertile particles. (Samples of TRISO coated fissile and fertile carbide particles from another program included for comparison.) All diluent samples are of the buffer-diluted design. Some of the samples are designed to evaluate the effect of higher fuel loadings in coated particles (i.e., thinner coatings).

The capsule has been discharged from the ETR, after attaining a maximum fluence of 8×10^{21} n/cm², and maximum burnup in the diluent particles of 75% FIMA. Most samples in the high-temperature cells apparently operated considerably below their design temperatures, partly because of fuel depletion, but probably mainly due to thermocouple decalibration. Temperatures in the 1500°C cell (cell 1) were estimated at 900° to 1100°C (no thermocouples still in operation), and temperatures in the 1300°C cells (cells 2 through 4)

were 950° to 1070°C. Samples in the 900°C cell (cell 5), however, operated close to design temperature.

Postirradiation examination of capsule P13L is scheduled to begin early next quarter.

Capsule P13M

Capsule P13M will be the second full-exposure test of coated diluent particles. Prototype coated oxide recycle fuel is also planned to be included. Many of the samples will be tested in pitch-bonded fuel rods to properly evaluate the full range of interactive stresses experienced by initial and recycle core fuel for a large HTGR. Many of the samples tested will be duplicates of those in the Peach Bottom test element which still remain to be fabricated (see Task VIII of this report), and the P13M capsule will provide full-exposure data on these important materials. (The maximum exposure received in the test elements will be 4.2×10^{21} n/cm² after 900 effective full-power days.) Due to the importance of determining thermocouple decalibration data to apply to subsequent capsules, capsule P13M may also contain one or more thermocouple decalibration cells, originally planned for P13N.

The P13M samples will be irradiated at 1500°, 1350°, and 1000°C to burnups of 75% FIMA for diluent fissile particles, 20% FIMA for recycle fissile particles, and 7% FIMA for fertile particles (full HTGR burnups for all three types of particles). The design fast fluence is 8×10^{21} n/cm² (full HTGR exposure).

Task XI
GRAPHITE RESEARCH

CONTROL MATERIALS IRRADIATIONS

Capsule BG-2

The primary purpose of the BG-2 experiment (GA-19-2) is to measure the effects of neutron irradiation on boronated (B or B_4C plus graphite) and hafnated (HfC plus graphite) compacts. Dimensional changes in boron fibers and density changes in loose granules of B_4C and HfC will also be determined in piggyback specimens to enhance the knowledge of irradiation effects in these poison materials in the absence of a bonding matrix material. A better understanding of irradiation effects in boronated and hafnated graphite should result from this investigation.

Irradiation-induced dimensional change was the phenomenon of primary interest; however, a correlation of thermal expansion coefficient, strength, and microstructure with neutron exposure was planned. Some specimens irradiated previously in capsule BG-1 (reported in earlier quarterly report GA-9372) continued irradiation exposure in BG-2. A detailed summary of samples being tested in BG-2 was given in earlier quarterly report GA-9660. The design of BG-2 was basically the same as BG-1 as described in Ref. 1.

The capsule was inserted in the ETR and started irradiation in July 1969 (cycle 103) at temperatures of 600° to 700°C. The capsule remained in the ETR through cycle 106 and was removed in May 1970. During this report period the capsule was disassembled and the specimens were prepared for dimensional measurement. The experimental results reported here are observations made during disassembly and comments on the general appearance of the specimens.

Capsule Disassembly

The capsule disassembly began on June 17, 1970 in the Gulf General Atomic hot cell. The dosimetry was recovered and neutron exposures were determined as shown in Table 11.1.

Table 11.1
NEUTRON EXPOSURE OF SPECIMENS IN CAPSULE BG-2

Capsule Position	Fast Neutron Dose ^(a) (E>0.18 MeV)	Thermal Neutron Exposure ^(b) (E<0.4 eV)	
		Neutron Window	Shielded
Bottom	3.5×10^{21}	1.9×10^{21}	0.7×10^{21}
Middle	5.9×10^{21}	3.0×10^{21}	1.0×10^{21}
Top	4.2×10^{21}	2.0×10^{21}	0.7×10^{21}

(a) Determined from the Fe-54(n,p) Mn-54 reactions with a 51.2-mb cross section.

(b) Determined from the Co-59(n, γ) Co-60 reaction with a 36.8-b cross section.

The gadolinium wafers placed between specimens as thermal-neutron shields showed evidence of severe corrosion in cells 1, 2, and 4. In cells 1 and 3, a white powder deposit was observed on graphite components that had been near the purge gas inlet and adjacent to the metal walls of the cell. A sample of the powder taken from cell 1 was examined by X-ray diffraction and found to be Li_2CO_3 .

Apparently the cells had been purged with gas containing oxidant which attacked the Gd wafers, carbon, and Li produced by the $\text{B}^{10}(\text{n},\alpha)\text{Li}^7$ reactions. The source of the oxygen in the cells has not been determined. Helium leak detector inspection of the purge lines during shutdown between the last two cycles at the ETR had not revealed a leak. Construction of the capsule and observations during disassembly rule out the possibility of water leakage into the system from the ETR coolant.

The Inconel-sheathed Chromel/Alumel thermocouples and the tungsten wire wrapping did not show signs of oxidation or other corrosion with the exception of the single thermocouple at the center of cell 2. That thermocouple had failed about half way through the irradiation experiment and a subsequent operator error caused the cell temperature to approach the melting point of the Inconel sheath for a period of several hours. About 1/2 in. of the tip of that thermocouple had the appearance of having been melted and about 1/8 in. was missing as shown in Fig. 11.1. Metallography confirmed that the sheath had been near the melting point from observations of large grain size and apparent plastic deformation of the metal. The tungsten wire wrap had been pressed into the softened inconel with no apparent chemical reaction.

A qualitative indication of the room-temperature ductility of the Inconel sheath material on the thermocouples was obtained by bending the thermocouple about 180° with a 3/8-in. radius. No brittle cracking of the sheath was observed, and this was taken as a confirmation of the expected compatibility of the nickel base alloy with boronated graphite at high B^{10} burnup at reactor service temperatures.

Specimen Examination

All of the piggyback specimens were recovered from the experiment, and these will be examined when dimensioning, crush-strength, and thermal expansion measurements, and chemistry have been completed on the boronated and hafnated specimens.

The boronated and hafnated graphite specimens were taken from the graphite crucibles and examined. The hafnated samples were examined in the hot cell because of high radioactivity (about 100 R per sample $[B + \gamma]$ at 2 in.). They appeared unchanged from the irradiated condition shown in Fig. 11.2.

The boronated graphite specimens were also essentially unchanged in appearance during the irradiation experiment. Those specimens were

K-367-60

Fig. 11.1. Inconel-sheathed Chromel/Alumel thermocouples from cell 2 of BG-2. The central thermocouple tip had apparently been melted and about 1/8 in. was missing from the end. Three other thermocouples in cell performed satisfactorily throughout the experiment. Tungsten wire was wrapped on the thermocouples to prevent intimate contact with the graphite and specimens. Note where tungsten wire produced grooves in central thermocouple near tip.

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Fig. 11.2. Crucible and specimen No. 4 from cell 2 of BG-2. Specimen contained 90 wt-% Hf (HfC + graphite) and was irradiated to a fast fluence of about 5.9×10^{21} ($E > 0.18$ MeV) at a center-line temperature of 700°C . A piggyback crucible can be seen in one of the holes in the crucible wall.

removed from the crucibles and transferred to the low-level cell at Gulf General Atomic for further examination. Dimensional measurements are currently in progress.

Reference

1. Stansfield, O. M., "Irradiation-induced Dimensional Change in Boronated Graphite, Results of the BG-1 (GA-19-1) Experiment," USAEC Report GAMD-9276, Gulf General Atomic Incorporated, July 29, 1969.