RECEIVED BY DTIE MAY 18 1970

Gulf General Atomic
Incorporated

RECEIVED BY DTIE MAY 18 1970

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-9944
UC-80 Reactor
Technology

HTGR BASE PROGRAM
QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
FEBRUARY 28, 1970

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

March 31, 1970

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Printed in the United States of America

Available from

Clearinghouse for Federal Scientific and Technical Information

National Bureau of Standards, U. S. Department of Commerce

Springfield, Virginia 22151

Price: Printed Copy \$3.00, Microfiche \$0.65

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Gulf General Atomic Incorporated

P. O. Box 608, San Diego, California 92112

MASTER

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-9944
UC-80 Reactor
Technology

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States nor the Commission nor any person acting on behalf of the Commission

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe upon privately owned rights, or

B. Assumes any liabilities with respect to the use of, or for damages resulting from, the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, person acting on behalf of the Commission includes a contractor, employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor, prepares, disseminates, or provides access to any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

HTGR BASE PROGRAM
QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
FEBRUARY 28, 1970

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

March 31, 1970

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Key

QUARTERLY REPORT SERIES

GA-4072-December, 1962, through February, 1963
GA-4350-March, 1963, through May, 1963
GA-4569-June, 1963, through August, 1963
GA-4937-September, 1963, through November, 1963
GA-5104-December, 1963, through February, 1964
GA-5366-March, 1964, through May, 1964
GA-5618-June, 1964, through August, 1964
GA-5866-September, 1964, through November, 1964
GA-6113-December, 1964, through February, 1965
GA-6418-March, 1965, through May, 1965
GA-6671-June, 1965, through August, 1965
GA-6869-September, 1965, through November, 1965
GA-7010-December, 1965, through February, 1966
GA-7181-March, 1966, through May, 1966
GA-7396-June, 1966, through August, 1966
GA-7553-September, 1966, through November, 1966
GA-7801-December, 1966, through February, 1967
GA-7981-March, 1967, through May, 1967
GA-8200-June, 1967, through August, 1967
GA-8356-September, 1967, through November, 1967
GA-8530-December, 1967, through February, 1968
GA-8662-March, 1968, through May, 1968
GA-8860-June, 1968, through August, 1968
GA-9090-September, 1968, through November, 1968
GA-9227-December, 1968, through February, 1969
GA-9372-March, 1969, through May, 1969
GA-9660-June, 1969, through August, 1969
GA-9815-September, 1969, through November, 1969

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 developmental engineering programs support the primary reference design objective.

CONTENTS

INTRODUCTION	1
TASK IV. FISSION-PRODUCT MECHANISMS	1
Diffusion of Strontium in Graphite	1
Mathematical Treatment of Fission-product Transport	8
Diffusion of Fission Products	8
Metallic-fission-product Release	11
TASK V. RECYCLE FUEL STUDIES	13
Recycle Test Elements	13
HTGR Fuel Recycle Plant Study	13
Head-end Reprocessing	13
Block Crushing	13
Tumbling	15
Fluid Bed Burning	34
Alumina-particle Separation	49
Vibratory Screener	53
Dust Collection Facility	53
TASK VIII. PHYSICS AND FUEL MANAGEMENT	55
HTLTR Support Activities	55
Fuel Test Element	55
Fuel Body	59
Fuel Beds	61
Central Irradiation Samples	62
TASK IX. ADVANCED COATED PARTICLE FUELS	69
Coated Particle Irradiations	69
Capsule P13L	69
TASK XI. GRAPHITE RESEARCH	71
Irradiation Effects in Carbon and Graphite	71
Control Materials Irradiations	71
Capsule BG-2	71

Task IV
FISSION-PRODUCT MECHANISMS

DIFFUSION OF STRONTIUM IN GRAPHITE

Strontium permeation experiments 4370-143 and 4370-144 with initial source concentrations of 20.3 mg Sr/g matrix (source) material have now accumulated 1780 hr at 850°C and 1400 hr at 1000°C, respectively. The experimental procedures and apparatus have been described in an earlier quarterly report (GA-9660). Tables 4.1 and 4.2 show the experimental results to date; and Figs. 4.1 through 4.4 show the total strontium releases and release rates for both experiments.

The preliminary results of these two experiments indicate that the temperature dependence of the rate of strontium release from the graphite crucibles is relatively small (about 10 kcal). This suggests an activation energy for the diffusion process that is small compared to literature values of 30 to 60 kcal/mole.

The release rates in both experiments decreased during the first 800 hr. This is consistent with earlier runs where a similar effect was observed. One explanation for this behavior is that sulfur or some other material migrates into the graphite with subsequent plugging of strontium flow paths. However, this or any other explanation needs to be verified experimentally.

It is of interest that the release rates for experiment 4370-144 (1000°C) are very similar to the release rates of earlier experiment 4370-16 (see GA-9660) even though the source concentrations differ by a factor of about 10. One would expect a considerable change in release rate with changing concentration. The probable explanation for this apparent discrepancy is a difference in the strontium sorptive capacity of the source coke materials used in the two experiments. In experiment 4370-16, the matrix material used in the source was a furfuryl alcohol coke with low sulfur content (<1000 ppm), whereas the matrix material of experiment 4370-144 was calcined petroleum coke containing about 1% sulfur. In recent vapor pressure measurements on these materials, it was found that the vapor pressure of strontium at similar strontium loadings, which were near the loadings for the diffusion experiments, was a factor of approximately 10 lower for the petroleum coke material. This means that the distribution coefficient for strontium between the matrix material and graphite at the source boundary is about 100 for experiment 4370-16 (measured) and about 1000 for experiment 4370-144 (estimated). The driving force for migration of strontium through graphite in these permeation experiments is dependent on the strontium concentration in the graphite at the source boundary. Due to the different distribution coefficients, both experiments probably had similar strontium concentrations in the graphite; therefore, similar release values would be expected.

Table 4.1

PRELIMINARY RESULTS FOR STRONTIUM PERMEATION
EXPERIMENT NO. 4370-143 PERFORMED AT 850°C

Run No.	Run Time (hr)	Total Time (hr)	Sr Release in Time Period (mg)	Release Rate (mg Sr/hr)	Total Sr Release (mg)
1	1.0	1.0	6.9×10^{-7}	6.9×10^{-7}	6.9×10^{-7}
2	2.0	3.0	5.6×10^{-7}	2.8×10^{-7}	1.2×10^{-6}
3	4.0	7.0	1.1×10^{-6}	2.7×10^{-7}	2.3×10^{-6}
4	10.0	17.0	1.8×10^{-6}	1.8×10^{-7}	4.1×10^{-6}
5	22.7	39.7	6.3×10^{-6}	2.8×10^{-7}	1.0×10^{-5}
6	26.0	65.7	3.4×10^{-6}	1.3×10^{-7}	1.4×10^{-5}
7	40.5	106.2	5.6×10^{-6}	1.4×10^{-7}	1.9×10^{-5}
8	45.5	151.7	7.4×10^{-6}	1.6×10^{-7}	2.7×10^{-5}
9	47.5	199.2	6.1×10^{-6}	1.3×10^{-7}	3.3×10^{-5}
10	34.0	233.2	7.6×10^{-6}	2.2×10^{-7}	4.0×10^{-5}
11	21.0	254.2	3.9×10^{-6}	1.9×10^{-7}	4.4×10^{-5}
12	35.2	289.5	3.7×10^{-6}	1.1×10^{-7}	4.8×10^{-5}
13	48.0	337.5	5.3×10^{-6}	1.1×10^{-7}	5.3×10^{-5}
14	48.0	385.5	5.7×10^{-6}	1.2×10^{-7}	5.9×10^{-5}
15	36.7	422.2	3.8×10^{-6}	1.1×10^{-7}	6.3×10^{-5}
16	64.0	486.2	6.6×10^{-6}	1.0×10^{-7}	7.0×10^{-5}
17	112.0	598.2	8.6×10^{-6}	7.7×10^{-8}	7.8×10^{-5}
18	72.0	670.2	4.8×10^{-6}	6.6×10^{-8}	8.3×10^{-5}
19	137.7	808.0	4.0×10^{-6}	2.9×10^{-8}	8.7×10^{-5}
20	192.0	1000.0	3.3×10^{-6}	1.7×10^{-8}	9.0×10^{-5}
21	233.0	1233.0	4.0×10^{-6}	1.7×10^{-8}	9.4×10^{-5}
22	262.6	1495.6	4.5×10^{-6}	1.7×10^{-8}	9.9×10^{-5}
23	164.5	1660.1	2.8×10^{-6}	1.7×10^{-8}	1.0×10^{-4}
24	121.0	1781.1	1.7×10^{-6}	1.4×10^{-8}	1.0×10^{-4}

Table 4.2
PRELIMINARY RESULTS FOR STRONTIUM PERMEATION EXPERIMENT
NO. 4370-144 PERFORMED AT 1000°C

Run No.	Run Time (hr)	Total Time (hr)	Sr Release in Time Period (mg)	Release Rate (mg Sr/hr)	Total Sr Release (mg)
1	1.0	1.0	2.2×10^{-6}	2.2×10^{-6}	2.2×10^{-6}
2	2.0	3.0	2.2×10^{-6}	1.1×10^{-6}	4.4×10^{-6}
3	4.0	7.0	3.6×10^{-6}	9.0×10^{-7}	8.0×10^{-6}
4	8.2	15.2	5.7×10^{-6}	7.0×10^{-7}	1.4×10^{-5}
5	17.7	32.9	1.7×10^{-5}	9.4×10^{-7}	3.0×10^{-5}
6	22.2	55.2	1.6×10^{-5}	7.4×10^{-7}	4.7×10^{-5}
7	19.7	74.9	1.0×10^{-5}	5.2×10^{-7}	5.7×10^{-5}
8	26.5	101.4	1.4×10^{-5}	5.4×10^{-7}	7.1×10^{-5}
9	23.0	124.4	1.0×10^{-5}	4.6×10^{-7}	8.2×10^{-5}
10	23.5	147.9	6.4×10^{-6}	2.7×10^{-7}	8.8×10^{-5}
11	65.0	212.9	1.3×10^{-5}	2.0×10^{-7}	1.0×10^{-4}
12	72.0	284.9	1.3×10^{-5}	1.8×10^{-7}	1.1×10^{-4}
13	47.0	331.9	5.4×10^{-6}	1.2×10^{-7}	1.2×10^{-4}
14	114.0	445.9	8.5×10^{-6}	7.5×10^{-8}	1.3×10^{-4}
15	96.0	541.9	1.3×10^{-5}	1.4×10^{-7}	1.4×10^{-4}
16	88.5	630.4	3.6×10^{-6}	4.1×10^{-8}	1.4×10^{-4}
17	216.0	846.4	7.9×10^{-6}	3.6×10^{-8}	1.5×10^{-4}
18	269.0	1115.4	1.3×10^{-5}	4.7×10^{-8}	1.7×10^{-4}
19	161.6	1277.1	1.1×10^{-5}	6.9×10^{-8}	1.8×10^{-4}
20	121.8	1398.9	5.4×10^{-6}	4.4×10^{-8}	1.8×10^{-4}

Fig. 4.1--Total strontium release as a function of time (permeation experiment 4370-143 conducted at 850°C)

Fig. 4.2--Strontium release rate as a function of time (experiment 4370-143 conducted at 850°C)

Fig. 4.3--Total strontium release as a function of time (permeation experiment 4370-144 conducted at 1000°C)

Fig. 4.4--Strontium release rate as a function of time (experiment 4370-144 conducted at 1000°C)

Some of the explanations given above will be verified after the two experiments have been terminated and concentration gradients and distribution coefficients have been established.

These experiments should also yield data on the migration of sulfur through graphite at 850° and 1000°C. The question of sulfur migration is of importance because it determines whether sulfur, originally introduced in the reactor as an impurity in the fuel bed matrix or filler material, will stay in the material during reactor operation. Matrix or filler material with a relatively high sulfur content could decrease significantly the degree of strontium migration.

MATHEMATICAL TREATMENT OF FISSION-PRODUCT TRANSPORT

Development, including documentation, of FIPER X, the one-dimensional version of the FIPER code, is now complete, and the code has already proved to be very valuable for use in predicting the release of fission-product metals from HTGR fuel elements.

Development of the two-dimensional version of the FIPER code is continuing. The code is being applied to the multiregion diffusion problem describing the simultaneous and coupled diffusion of metallic fission products through a hollow graphite cylinder containing a small core of a substance characterized by a large diffusion coefficient. Concentration profiles have been evaluated as the function of the cylinder coordinates r and z and of the time t . At the interface between the graphite and the high-diffusivity material, the concentration ratio is given by the distribution coefficient ϕ , which can be calculated as a function of r and z once the pressure isotherms are known in both substances. The code works for diffusion coefficients in the graphite and in the core material, respectively, that differ by a factor of as much as a thousand. The axial release current at a given time t and the time-integrated release current will be calculated, assuming that the concentration at one end of the cylinder is given by a constant, independent of time.

DIFFUSION OF FISSION PRODUCTS

Under certain circumstances the diffusion profile for an arbitrary half-life unstable species will have a shape identical to that of a stable species. This fact is of importance in interpreting the profiles obtained in some experiments. The above assertion is proved in the following.

For a stable species diffusing in a one-dimensional medium, the pertinent differential equation for the concentration C is

$$\frac{\partial C}{\partial t} = D \frac{\partial^2 C}{\partial x^2} + S_s(x,t) \quad , \quad (1)$$

where $S_s(x,t)$ is an arbitrary source term, and the other terms have their usual meaning. Attention in what follows is restricted to initial and boundary conditions of the form

$$C(0,x) = f(x) \quad (2)$$

and

$$\left[g(x,t)C + h(x,t) \frac{\partial C}{\partial x} \right]_{\partial} = 0 \quad (3)$$

For an unstable species with a decay constant, λ , the appropriate differential equation for the concentration, U , is

$$\frac{\partial U}{\partial t} = D \frac{\partial^2 U}{\partial x^2} + S_u(x,t) - \lambda U \quad (4)$$

Assume temporarily, that $U = Ce^{-\lambda t}$ and $S_u = S_s e^{-\lambda t}$ and substitute into Eq. (4). The resulting expression,

$$e^{-\lambda t}(-\lambda C + \frac{\partial C}{\partial t} = D \frac{\partial^2 C}{\partial x^2} + S_s - \lambda C) \quad ,$$

is identical to Eq. (1). The initial condition, Eq. (2), is satisfied, since $C(0,x) = U(0,x)$. Also, the homogeneous boundary conditions, Eq. (3), are likewise satisfied. Thus, the solutions for a stable and unstable species will have identical geometric shapes. That is,

$$U = C(x,t)e^{-\lambda t} \quad ,$$

as long as

$$S_u = S_s(x,t)e^{-\lambda t} \quad .$$

In words, for an arbitrary initial concentration distribution and an arbitrary source, the concentration profile of a stable and unstable species will always be identical in shape as long as one has homogeneous boundary conditions and the unstable source is an exponential decay factor less than the stable source.

The above conditions can obviously be satisfied in some annealing experiments where the source term is zero. Of more importance, however, is that these conditions are approximately satisfied in other situations. As an example, the curve of Fig. 4.5 shows a FIPER calculation giving cesium migration in the sleeve of the Peach Bottom D06-01 fuel element. The upper curve was obtained by using a constant source in the fuel region and using a Cs^{137} half life of 30 years. The other two curves were obtained by employing exactly the same source and using cesium half lives of 8 years and 2.1 years. Notice that the curves have almost exactly the same shapes even though the source was not exponentially decayed as required by the theory (a relatively small correction). This discrepancy probably accounts for the fact that the curves are not exactly displaced by $e^{-\lambda t}$.

Fig. 4.5--Metal concentration profile of sleeve from Peach Bottom fuel element D06-01, opposite compact 25, after 300 EFPD

An interesting sidelight is that the curves for the 8-year half life closely approximates the calculated Cs^{133} concentration curve (the average yield is approximately the same as Cs^{137}). Although Cs^{133} is a stable isotope, its average $\phi\sigma$ value of $2.7 \times 10^{-9} \text{ sec}^{-1}$ gives it an effective half life of 8 years at the average flux of the Peach Bottom reactor. One can develop equations for the 2.1-year Cs^{134} activation product of Cs^{133} . However, it does not appear that strong over-all conclusions can be drawn - only local conclusions. One thing apparent from the data points shown in Fig. 4.5 is that the Cs^{134} concentration is almost exactly a factor of $(1 - e^{-\phi\sigma t})$ less than the Cs^{133} concentration (approximately the same as the Cs^{137} concentration points). Also, other data indicate that the variation of the $\text{Cs}^{133}/\text{Cs}^{134}$ ratio along the length of the element appears to resemble the epithermal $\phi\sigma$ variation. It is therefore likely that the Cs^{134} profile is essentially determined by activation of the Cs^{133} profile.

METALLIC-FISSION-PRODUCT RELEASE

Strontium-graphite diffusion experiments have been carried out by Gulf General Atomic for some time in an effort to characterize the transport of strontium in fuel element graphite. Concentration profile data have been obtained from these experiments over a range of strontium source concentration profiles. The object of the calculations is to show (1) whether the experimental concentration profiles may be described in whole or in part as a simple diffusion process, (2) whether the experimental profiles are the result of transient or steady-state conditions, and (3) the degree of fit that may be obtained using the Gulf General Atomic reference diffusion coefficient (temperature dependent).

Calculations made to date, using a transient diffusion model and Gulf General Atomic reference diffusion data, generally are in reasonable agreement with the experimental profiles for strontium-graphite concentrations in the range of interest. For experiments of up to 15 hr duration, the portion of the concentration profile about two orders of magnitude less than the initial concentration, characteristically has a tail that is flatter than the calculated reference design diffusion profile. This tail, however, is not apparent in the long-term experiments carried out from 300 to 4500 hr, i.e., the calculated reference design diffusion profile is in reasonable agreement with the experimental profiles over the whole cross section of the test piece. However, it is not readily apparent whether the low-level tail has moved up, or whether it has been masked by movement of the high-level transient portion. In any case, it does appear that the diffusion process is in a transient state, and that any attempt to apply "permeation" results could be misleading when applied to the strongly time-varying reactor situation.

Task V
RECYCLE FUEL STUDIES

RECYCLE TEST ELEMENTS

Eight recycle test elements (RTE) are being designed and assembled by Gulf General Atomic for irradiation as part of Peach Bottom Core two. Following irradiation they will be used to test reprocessing equipment at ORNL. The objectives of this work are to prove out reprocessing equipment designs, to demonstrate the separability of particle types, and to determine chemical and physical characteristics of irradiated fuels significant to recycle operations.

In cooperation with ORNL and AEC, a basic RTE design envelope and fuel loading requirements have been developed. The design configuration of the RTE is similar to that of the fuel test element described in detail in Task VIII of this report, except that six 15.5-in.-long fuel bodies are used and the central hole contains a solid graphite spine. The eight fuel holes in the fuel body will contain combinations of bonded rods and blended beds as indicated in Table 5.1. The various combinations of fissile and fertile particles with oxide or carbide kernels are given in Table 5.2.

Particles for loading the fuel elements are being provided by both ORNL and Gulf General Atomic. In general, ORNL is supplying sol-gel oxide particles and Gulf General Atomic is providing sintered carbide particles. Work is under way on particle preparation and on the development of fabrication procedures for fuel rods.

HTGR FUEL RECYCLE PLANT STUDY

Material balances have been completed on both the reprocessing and re-fabrication portions of the HTGR fuel recycle plant. Work is in progress on equipment design and sizing. When complete, the study will provide cost and other data for three sizes of integrated reprocessing-refabrication plants.

HEAD-END REPROCESSING

Block Crushing

A jaw-crusher has been designed for crushing 12-rod fuel elements. The jaw angle is 11° , and the opening is adjustable from zero to 1-1/2 in. The feasibility of procuring such a machine is being evaluated as a possible alternative to modifying the National Lead Company of Ohio jaw-crusher.

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	behj	a*	behj	d	behj	f	acfg	i
4 (max flux)	2250-2300	acfg	e	acfg	f	acfg	f	behj	j
3 (max flux)	2050-2250	acfg	f*	acfg	a*	acfg	d	bdhi	e
2	1650-2050	acfg	d	acfg	f*	acfg	f	bdhi	i
1 (bottom of reactor)	1050-1650	acfg	e*	acfg	e	acfg	e	bdhi	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
j	(Th,U)O ₂ ^c TRISO	ThC ₂

^aUnless otherwise indicated, all particles are BISO coated.

^bTh/U = 4.2.

^cTh/U = 2.0.

Tumbling

The tumbler described in the previous quarterly report has been delivered, operated, and modified as discussed below. A diagram of the modified machine is shown in Fig. 5.1. The drum covered with 1/4-in.-mesh woven cloth, is rotated at 24 rpm. Flights inside the drum lift the feed and allow it to fall, thus causing attrition. The -1/4-in. pieces fall through the mesh into the fines hopper, whereas the +1/4-in. pieces are retained in the drum until the disk is withdrawn away from the drum end. The coarse material then falls into the coarse hopper. When a mixture of crushed graphite and bonded rods is fed, some of the graphite is recovered free of particles, and the rods, which attrite more readily to single particles and particle agglomerates, are recovered containing some graphite. Dust created during operation is removed through the exhaust duct.

Four runs have been carried out. The first, Run DT-001, was carried out before the disk was fitted; feed was thus able to pass out through the drum end. The objective was to determine the fraction of broken particles and to learn whether the fuel rods could be broken down to -1/4-in. mesh continuously. It was found that they could not. Large quantities of +1/4-in. fuel particle agglomerates appeared in the coarse product hopper. The product was manually screened 1/4-in. mesh, and the oversize refed. This procedure was followed until there were no fuel particles in the +1/4-in. fraction. All the -1/4-in. product was burned, and the residue was screened through 250- μ m

Fig. 5.1--Rotating drum tumbler

mesh to determine the fraction of broken particles. The result is given in Table 5.3.

Table 5.3
PARTICLE BREAKAGE DURING TUMBLING
(Disk Not Fitted)

Run No.	Graphite Weight (g)	Rod Weight (g)	Broken Particles (%)	Fraction Fissile Particles (%)
DT-001	773.5	104.8	1.5	---
DT-002	804.6	888.0	1.4	41.0 Expected: 43.1

NOTE: Feed Size: +1/4-in. -1/2-in., TRISO/TRISO fuel particles.

Run DT-002 was also carried out without the disk. The objective was to determine the rate of attrition, the fraction of broken particles, and the ratio of broken fertile to broken fissile particles. The mode of operation was to feed, tumble for 5 min, screen through 1/4-in. mesh, refeed the oversize, and again tumble for 5 min. The run was stopped when no particles could be detected in the +1/4-in. mesh fraction. The -1/4-in. mesh product was burned, and the residue was sieved through a 420- μ m screen to separate fertile from fissile particles and through a 250- μ m screen to determine the fraction of broken particles. The result is given in Table 5.3. The -250- μ m residue was analyzed for uranium and thorium, from which values the ratio of broken fertile to broken fissile particles was calculated to be 2.6. (These broken particles represent about 1.0% of the total particles.) The ratio of fissile to fertile particles in the feed was 5.0. The attrition results are shown in Fig. 5.2.

These two runs showed that the tumbler cannot be operated continuously because the residence time is uncontrollable. The disk was therefore fitted to enable batch operation to be carried out. After the disk was added, the product was removed at prescribed time intervals during the tumbling, and the disk was withdrawn to collect oversize graphite when fuel particles could no longer be observed in the product.

Run DT-003 results are given in Fig. 5.3. From these values, a curve showing the fraction of feed fuel particles in the product versus time was constructed and is given in Fig. 5.4. Another curve, the fraction of initial graphite in the product versus time, is shown in Fig. 5.5. Particle breakage was 1.05%.

Some fuel elements were crushed at Gulf General Atomic with 7/16-in. open side, 3/16-in. closed side. The crushed product was fed to the tumbler in Run DT-004, and the results are shown in Fig. 5.6. As for Run DT-003, the curves of fuel rod and graphite attrition versus time have been drawn

Fig. 5.2--Attrition of crushed fuel element in the drum tumbler, Run DT-002. Feed composition was graphite, 805 g, and fuel rods, 888 g; feed size, +1/4-in. -1/2-in.; product was a mixture of graphite and fuel particles; product size, -1/4-in.

Fig. 5.3--Flowsheet of Run DT-003 on the drum tumbler

Fig. 5.3--Run DT-003, continued

Fig. 5.3--Run DT-003, continued

$$\% \text{ BROKEN PARTICLES} = \frac{0.1454}{13.866} = 1.05\%$$

Fig. 5.3--Run DT-003, continued

WEIGHT OF BURNED PARTICLES: 13.8659 G
 WEIGHT OF ASH: 0.1454 G
 $\Sigma = 14.0113$
 WEIGHT EXPECTED: 13.99
 ± 0.07 G

Fig. 5.3--Run DT-003, continued

Fig. 5.4--Attrition of fuel rods in the drum tumbler, Run DT-003. Feed composition was graphite, 196.26 g, and fuel rods, 19.42 g; feed size, +1/4-in. -1/2-in; product is a mixture of graphite and fuel particles; product size, -1/4-in.

Fig. 5.5--Attrition of graphite in the drum tumbler, Run DT-003

Fig. 5.6--Flowsheet of Run DT-004 on the drum tumbler

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

Fig. 5.6--Run DT-004, continued

and are shown in Figs. 5.7 and 5.8, respectively. An additional curve that can be constructed from the data is the fraction of broken particles in each sample. This is shown in Fig. 5.9. Particle breakage was 1.17%. From an analysis of the residue, the ratio of broken fertile to broken fissile particles was found to be 2.8. This result compared well with the 2.6 obtained in Run DT-002. As in that run, the initial fissile to fertile particle ratio was 5:1.

A noteworthy point during all four runs was that blinding of the screen was virtually zero. Two lumps of graphite had to be pressed out from the screen at the end of the four runs. Vibration of the drum appeared to shake lumps out of the holes during rotation.

These preliminary results show that the tumbling operation is best carried out in batches.

The broken particle fraction depends on the number of tumbles the agglomerates receive, and it follows that a larger hole size in the mesh will lead to a smaller fraction. The tumbler mesh size depends on the maximum size feed the burner will accept. The results of burner operation will therefore dictate the tumbler mesh size.

If the fluid bed burner can be fed with the -1/4-in. tumbler product, then, based on the results from Run DT-004, 99.4% of the feed particles would be fed to the burner after a 35-min tumble. The burner feed would also contain 15.2% of the graphite fed to the tumbler. The tumbling time is too long; a better time would be 25 min or less, because the fuel block feed rate to a reprocessing plant is about one block every 30 min, and it would be helpful to synchronize the batch block feed with batch tumbler feed. The 15.2% graphite feed to the burner seems to be within reason.

The ratio of broken fertile to broken fissile particles suggests that the -250- μ m ash should be recycled to the fertile particle stream.

The program planned is to operate the burner to determine the maximum size feed it will accept and then to investigate the tumbling step using that size of mesh. The tumbler has been returned to the manufacturer for two spare drums to be made, 3/8-in. and 1/2-in. mesh. With this equipment, optimum conditions for operation of the tumbler will be sought, i.e., minimum particle breakage, minimum fraction of graphite, maximum product size, and maximum throughput.

Fluid Bed Burning

Before retiring the 2-in. fluid bed burner, the last seven runs were directed toward evaluating several methods of heating the bed for ignition. An electric furnace was used for early runs, but because of scaleup problems additional methods were tested. The most promising alternative to electric heating was found to be an in-bed CO burner. CO was chosen to obviate the formation of water vapor during combustion. Startup is as follows: CO and O₂ are introduced into a mixing chamber where two spark plugs cause ignition (see Fig. 5.10). Combustion takes place in the mixing chamber and the flame

Fig. 5.7--Attrition of fuel rods in the drum tumbler, Run DT-004. Feed composition was graphite, 1662 g, and fuel rods, 941 g; feed size, as received from jaw crusher 7/16-in., 3/16-in. open/closed; product is a mixture of graphite and fuel particles; product size, -1/4 in.

Fig. 5.8--Attrition of graphite in the drum tumbler, Run DT-004. Feed composition was 1662 g, fuel rods, 941 g; feed size, as received from jaw crusher 7/16-in., 3/16-in. open/closed; product size, -1/4 in.

Fig. 5.9--Fraction of broken particles in each sample from the drum tumbler, Run DT-004

Fig. 5.10--4-in. stainless steel fluid bed burner

enters the fluid bed. It heats the bed to ignition temperature and is then made progressively richer in oxygen until the bed ignites. The CO is turned off when fluid bed burning is under way. Since quick ignition is facilitated by a small initial bed, feed addition begins just after ignition, and the bed is readily built up to the desired height.

The results and experience gained from the 2-in. fluid bed were used to incorporate several improvements into the 4-in. fluid bed design. The major improvements are:

1. Five thermocouples to give continuous temperature recordings at various depths in the bed.
2. An enlarged off-gas line to reduce pressure surges.
3. Continuous feeder to eliminate temperature surges.
4. Top and bottom feed capability to study the effect of each.
5. Four product lines to study burning at various bed levels.
6. A clam-shell jacket around the reaction zone to allow forced-air convection heat transfer.
7. Studded reaction zone to increase the effective heat-transfer area.
8. A control board for the in-bed CO burner.

The 4-in. fluid bed burner (see Fig. 5.10) has been operated 19 times.

The effect of initial bed composition and bed size on the time required for ignition has been studied in each run (see Table 5.4). The bed size versus ignition time for various bed compositions has been plotted and is shown in Fig. 5.11. These initial results indicate that alumina in the startup bed increases the ignition time, small beds with wide particle size distribution are advantageous, and the ignition time increases with bed weight until, in the limit, heat lost from the reactor wall balances heat input below 600° and the bed will never ignite. As the tests continue, additional startup data should better define the influencing variables and their effects.

An advantage of this method of startup is that the reactor wall is not covered by an electric heater or lagging and can be observed during operation. A disadvantage is that if combustion ceases, the run has to be terminated because there is no supplementary heating available. The CO/O₂ flame will not ignite a bed having a weight greater than 600 g.

The first 12 runs of the 4-in. fluid bed burner are listed in Table 5.5. These runs were directed toward establishing the technique of bed ignition, buildup, and control. The average temperature of the center of the bed during burning was 900°C. Runs 1 through 6 were conducted using alumina as the inert material while burning -1/8-in. graphite.

Runs 7 through 12 burned -1/4-in. graphite in an alumina bed. In these runs, feeding and taking-off was done batchwise to determine the carbon in the take-off samples. The carbon concentrations in the samples ranged from

Table 5.4

4-IN. FLUID BED IGNITION CONDITIONS AT THE START OF EACH RUN

Run F4B	Alumina (g)	Graphite (g)		Coke -20, +100 mesh (g)	Ignition Time (min)
		-420, +125 μ m	-1/8 in.		
1	53.4	54.0	0	0	13
2	54.0	185.0	0	0	10
3	0	0	441.1	0	18
4	452.2	0	415.2	0	>40 ^a
5	218.6	0	442.9	0	>40 ^a
6	99.3	0	348.1	0	37
7	0	0	328.0	0	12
8	0	0	230.0	0	--
9	0	0	0	335.0	24
10	0	0	260.0	0	8
11	0	0	256.0	0	11
12	0	0	0	415.0	22
13	0	0	0	513.3	25
14	0	0	512.0	0	25
15	0	214.0	0	0	13
16	0	0	0	612.4	33
17	0	0	0	220.0	29
18	0	0	400	0	9
19	0	193.6	162.7	0	25

^aNo ignition.

Fig. 5.11--Boundary conditions for ignition with CO/O_2 in the 4-in. fluid bed burner

Table 5.5
INITIAL RESULTS OF BURNING GRAPHITE IN THE 4-IN. FLUID BED

Run F4B	Initial Bed (g)		Material Added (g)		Final Bed (g)		Burning Time (min)	Burning Rate (g/min)	Remarks
	Al ₂ O ₃	Graphite	Al ₂ O ₃	Graphite	Al ₂ O ₃	Graphite			
1	53.4	54.0	50.0	152.0	103.4	15.4	---	---	---
2	54.0	185.3	492.4	214.0	546.4	86.5	30	10.4	---
3	0	441.1	1520.0	934.3	1520.0	70.0	86	15.2	---
4	452.2	415.2	0	0	452.2	382.0	---	---	Ignition time >40 min; terminated run
5	218.6	442.9	0	0	218.0	327.0	---	---	Ignition time >40 min; terminated run
6	99.3	348.1	670.0	709.0	769.0	417.0	32	20.0	O ₂ cylinder emptied
7	0	328.0	1583.0	1660.0	1582.0	38.0	95	20.5	Samples contained 2% graphite
8	0	230	0	0	0	0	---	---	Bed burned out before complete ignition achieved
9	0	335	3540.0	2774.0	3540.0	1270.0	53	35.0	O ₂ cylinder emptied
10	0	260	142.0	137.0	142.0	125.0	11	24.7	Early addition of excessive -1/4-in. graphite terminated run
11	0	256	1430.0	1731.0	1430.0	193.0	60	29.9	---
12	0	415	2680.0	1600.0	2680.0	420.0	110	14.5	---

0.55% to 2.04%. The bed did burn the -1/4-in. graphite well, but some difficulty was experienced when the bed was allowed to burn to a graphite concentration of less than 2%. As the feed becomes larger, the ratio of surface area to mass becomes smaller. Because the combustion of graphite in oxygen is largely a surface-area-limited reaction, a greater mass of large graphite pieces is needed to obtain the same reaction rate as a specific mass of small graphite pieces.

Since the presence of alumina in the fluid bed necessitates an alumina-particle separation step, consideration is being given to alumina-free burning. Two all-graphite beds (crushed to -1/8 in.) were burned in the 2-in. fluid bed, and operation was as easy as burning with an alumina-graphite bed. Runs 13 through 19 with the 4-in. fluid bed (see Table 5.6) were conducted without alumina; the beds contained TRISO particles and graphite, with the exception of Run 14, which burned only -1/4-in. graphite.

Run 13 was conducted using crushed fuel blocks with bonded rods (screened to -1/4 in.). Large gas flows were required to keep the porous bed fluidized; allowing the bed to burn without fluidization (shaft kiln) resulted in particle agglomeration and very high center-bed temperatures (above 1350°C). Figure 5.12 shows an agglomeration of particles taken from the final bed of Run 13.

Run 14 was a study of burning only -1/4-in. graphite. The flowmeter capacity of the unit was insufficient to fluidize the bed; so, to prevent melting of the reactor wall, the run was terminated.

Run 15 was a repeat of Run 13 but the feed crushed to -1/8 in. Particle agglomeration resulted again when the bed became static. Figures 5.13 and 5.14 show this agglomeration.

A large rotameter for compressed air was installed before continuing with Run 16 to assure sufficient gas flow to fluidize the bed. Run 16 was carried out by burning a small bed of loose TRISO fertile particles. Since no agglomeration was apparent and the reactor exhibited a uniform red color (indicating fluidized burning), Run 17 was also conducted with loose TRISO fertile particles. Figure 5.15 shows a sample of the fines collected in the deentrainment drum. These fines are thin carbon shells that broke loose from the SiC coat and were elutriated. The results of Run 17 are shown in Fig. 5.16; the run terminated because the graphite concentration fell to 0.8% before feed addition. The bed did not agglomerate. The fines in the off-gas deentrainment drum were burned, and the ash, which weighed 0.82 g, was analyzed for thorium. The value was 61 ppm. The -250- μ m material from the final bed was burned and the ash (7.4 g) analyzed for thorium. The value was 2460 ppm. Both these results confirm the result in Table 5.6 that only a very small fraction of fuel particles was lost during burning.

Run 18 was also carried out with loose TRISO fertile particles. The bed was permitted to burn down to 1% graphite, but the addition of more particles failed to revive combustion. There was no agglomeration.

Run 19 was an extended run to locate some of the boundaries of particle-graphite burning. The run was initiated by burning loose TRISO fertile particles. Then a mixture of crushed graphite, screened to -1/8-in., and loose

Table 5.6

INITIAL RESULTS OF BURNING FUEL PARTICLE-GRAPHITE MIXTURES IN THE 4-IN. FLUID BED

Run F4B	Material Added (g)		Final Bed (g)		Burning Time (min)	Burning Rate (g/min)	Approximate Particle Loss (%)	Remarks
	TRISO Particles ^a	Graphite	TRISO Particles ^b	Graphite				
13	593.3	1587.0	--	--	46	20.9	--	Particle agglomeration when bed was allowed to go static
14	0	2590.0	0	1273.2	79	14.4	--	Terminated due to hot spots on reactor wall
15	503.0	1112.5	--	--	49	21.0	--	Particle agglomeration when bed was allowed to go static
16	365.3	697.4	--	--	37	18.5	--	No particle agglomeration
17	3215	1131	3230	165.5	72	16.0	0	No particle agglomeration
18	--	695.0	--	--	38	18.0	--	Allowed bed to burn out
19	3335.6	2392.0	3308.1	140.0	124	19.5	0.8	No particle agglomeration, but high center bed temperatures resulted from static bed periods

44

^aWeight of particles in SiC coats calculated from a preliminary burning test.^bWeight of particles in SiC coats recovered.

M31930-1

Fig. 5.12--Particle agglomeration from bonded rods (Run F4B13)

M31931-1

Fig. 5.13--Particle agglomeration from bonded rods (Run F4B15)

M31931-2

4X

Fig. 5.14--Closeup of the black/white area in the agglomeration above

M32013-1

16X

Fig. 5.15--Macrophotograph of the remains of outer pyrocarbon coats elutriated from the bed (Run F4B17)

Fig. 5.16--Temperatures recorded by various thermocouples during intermittent batch feed and take-off (run F4B17)

TRISO fertile particles was added. Next, crushed graphite, screened to -1/4-in., was added. The bed burned well with no particle agglomeration; however, graphite concentrations of less than 2% in the bed resulted in difficulty in maintaining operating temperatures. Samples were obtained with less than 0.5% graphite, but continued burning required that the bed be allowed to go static during feed addition, after which the bed was returned to fluidization.

Particle agglomeration is assumed to result from rod matrix material, since it did not occur when loose TRISO particles were burned in a like manner.

Since initial results show that alumina-free particle burning can be achieved, additional tests will be conducted to obtain a quantitative picture of the process. Variables such as feed size, distribution, use of air or oxygen (or a mixture of the two), particle breakage, particle agglomeration, fines elutriation, minimum graphite concentration, and burning efficiency must be studied further. These results will be compared with those from burning in an inert bed to determine the best method for continued study.

Alumina-particle Separation

If an inert material such as alumina is used in fluid bed burning, it must be separated from the particles and recycled to the burner. Since the alumina currently in use blinds the screens during the screening, another separation method may be required if satisfactory spherical alumina is not obtainable.

Several air elutriation tests with the equipment shown in Fig. 5.17 indicate that:

1. Alumina-particle separation is slow.
2. Air flow requirements are large.
3. Particle carryover is troublesome.

Figure 5.18 shows the rate of alumina separation at the optimum air flow rate. Increased air flow resulted in increased particle carryover, while decreased air flow decreased the alumina-particle separation. ORNL has reported a similar experiment in which much longer deentrainment sections were used and alumina-particle separation was achieved. The particles used, however, were larger than the fissile particles used in the Gulf General Atomic mixture. Further development work is required on this topic.

A method of "fluid bed" separation (see Fig. 5.19) makes use of the difference in velocities required to fluidize particles and alumina. The advantage of this method is the lower gas flow rate required. The two sections are designed so that fluidization begins in each section at the same time, particles in the lower section and alumina in the top section. By controlling the air flow, an alumina-free particle fraction can be maintained, but the equilibrium concentration of particles in alumina is about 25%. This method would work quite well if the particle-alumina mixture could be recycled to the burner.

Fig. 5.17--Glass elutriator

Fig. 5.18--Rate of elutriation of alumina at the optimum gas flow rate. The air flow rate was 67.5 L/min, take-off from top. The initial charge contained 50.0 g of alumina and 46.6 g of fissile + fertile particles, giving a total of 96.6 g

Fig. 5.19--Fluid bed separator

A type of spherical alumina has been obtained for comparison with the material currently being used. The results of the comparison indicate that:

1. Under microscopic examination, this spherical alumina looks "milky" and soft whereas the other looks clear and hard.
2. The ratio of tap density of the present alumina to that of the spheres is 1.67.
3. The spherical alumina reduces screen blinding.
4. Fluidization characteristics of the two types of alumina are very similar.
5. Attrition during fluidization is quite severe, as shown in Fig. 5.20.

The search for better spherical alumina will continue.

Vibratory Screener

A vibratory screener is being set up for determining the rates of separation of fuel particles during continuous operation.

Dust Collection Facility

The dust collection facility designed during the previous quarter has been received and is being set up to collect airborne dust from the tumbler and the fluid bed burner.

Fig. 5.20--Spherical alumina attrition during fluidization

Task VIII
PHYSICS AND FUEL MANAGEMENT

HTLTR SUPPORT ACTIVITIES

The measurements on the first HTGR lattice in the HTLTR have been completed for temperatures up to 1000°C. This lattice contained U^{235} and thorium with a C/Th ratio of 200 and a C/ U^{235} ratio of 6000. The data from these measurements will be available in the next quarter.

Measurements have started on the second HTGR lattice in the HTLTR. This lattice contains U^{233} and thorium with a C/Th ratio of 200 and a C/ U^{233} ratio of 11,000.

A plutonium-thorium lattice loading of interest to the HTGR program has been recommended for the HTLTR program. The utilization of plutonium in an HTGR system was discussed in a previous quarterly report, GA-9660. The initial plutonium lattice would contain plutonium and thorium with subsequent lattices having U^{233} added. The plutonium would be in the form of grains 200 μ m in diameter, coated with graphite, and having a C/Pu atom ratio of about 7500. The plutonium has the isotopic ratio 239:240:241:242 = 68:21:9:2 for this study. The C/Th atom ratio would be in the range of 250.

Some additional calculations have been performed to provide a basis for the recommendation for the first plutonium lattice in the HTLTR. These calculations have evaluated the effect on k_{∞} of variations in plutonium grain size and thorium loading for the HTLTR lattice geometry being used for the HTGR program. The results of these calculations are given in Figs. 8.1 and 8.2 and are summarized in Table 8.1. The choice of the 200- μ m plutonium grain size will provide a satisfactory test of the ability to calculate the Pu^{240} self-shielding effects and is in the range of the 100- to 300- μ m grain size that might possibly be used in an HTGR system.

FUEL TEST ELEMENT

Design work on the fuel test element for irradiation in the Peach Bottom reactor continued during the quarter. The major purpose of this element is to evaluate the performance of advanced fuels over a wide range of operating conditions and lifetimes. The time-at-temperature lifetimes of various types of fuel will be determined, and data will be obtained to identify causes of particle failure in order to improve fuel performance and lifetimes. The element will also make possible controlled irradiation experiments on:

1. Fuel particle thermal stability.
2. Metallic diffusion rates in an HTGR environment.

Fig. 8.1--Effect of plutonium grain size on k_{∞} for an HTLTR Pu-Th lattice with a C/Th ratio of 300 and a C/Pu ratio of 7500

Fig. 8.2--Effect of plutonium grain size on k_{∞} for an HTLTR Pu-Th lattice with a C/Th ratio of 250 and a C/Pu ratio of 7500

Table 8.1
HTLTR PLUTONIUM THORIUM LATTICE

Lattice Composition				Temp (°K)	k_{∞}	Thermal η_f	Fast Region (E > 2.38 ev)		
Pu(68:21:9:2)		U ²³³	Th				Σ_a	$v\Sigma_f$	Σ_{out}
300- μ m grains	C/Pu 7500	None	C/Th 300	300	1.170	1.4174	0.886×10^{-3}	3.317×10^{-4}	2.847×10^{-3}
				600	1.151	1.4250	0.950×10^{-3}	3.288×10^{-4}	2.797×10^{-3}
				900	1.135	1.4234	0.991×10^{-3}	3.270×10^{-4}	2.765×10^{-3}
				1200	1.113	1.4080	1.021×10^{-3}	3.256×10^{-4}	2.743×10^{-3}
				1500	1.081	1.3754	1.044×10^{-3}	3.246×10^{-4}	2.725×10^{-3}
300- μ m grains	C/Pu 7500	None	C/Th 250	300	1.114	1.3803	0.963×10^{-3}	3.292×10^{-4}	2.798×10^{-3}
				600	1.097	1.3937	1.038×10^{-3}	3.258×10^{-4}	2.741×10^{-3}
				900	1.083	1.3975	1.085×10^{-3}	3.236×10^{-4}	2.704×10^{-3}
				1200	1.061	1.3849	1.120×10^{-3}	3.219×10^{-4}	2.677×10^{-3}
				1500	1.030	1.3546	1.147×10^{-3}	3.206×10^{-4}	2.656×10^{-3}
200- μ m grains	C/Pu 7500	None	C/Th 300	300	1.160	1.4048	0.886×10^{-3}	3.317×10^{-4}	2.847×10^{-3}
				600	1.138	1.4071	0.950×10^{-3}	3.288×10^{-4}	2.797×10^{-3}
				900	1.119	1.4013	0.991×10^{-3}	3.270×10^{-4}	2.765×10^{-3}
				1200	1.091	1.3789	1.021×10^{-3}	3.256×10^{-4}	2.743×10^{-3}
				1500	1.059	1.3461	1.044×10^{-3}	3.246×10^{-4}	2.725×10^{-3}
200- μ m grains	C/Pu 7500	None	C/Th 250	300	1.108	1.3709	0.963×10^{-3}	3.292×10^{-4}	2.798×10^{-3}
				600	1.086	1.3786	1.038×10^{-3}	3.258×10^{-4}	2.741×10^{-3}
				900	1.068	1.3770	1.085×10^{-3}	3.236×10^{-4}	2.704×10^{-3}
				1200	1.046	1.3633	1.120×10^{-3}	3.219×10^{-4}	2.677×10^{-3}
				1500	1.011	1.3274	1.147×10^{-3}	3.206×10^{-4}	2.656×10^{-3}
100- μ m grains	C/Pu 7500	None	C/Th 300	300	1.132	1.3680	0.886×10^{-3}	3.317×10^{-4}	2.847×10^{-3}
				600	1.106	1.3638	0.950×10^{-3}	3.288×10^{-4}	2.797×10^{-3}
				900	1.082	1.3523	0.991×10^{-3}	3.270×10^{-4}	2.765×10^{-3}
				1200	1.053	1.3268	1.021×10^{-3}	3.256×10^{-4}	2.743×10^{-3}
				1500	1.017	1.2872	1.044×10^{-3}	3.246×10^{-4}	2.725×10^{-3}
100- μ m grains	C/Pu 5000	None	C/Th 300	300	1.120	1.3411	0.982×10^{-3}	4.876×10^{-4}	2.771×10^{-3}
				1200	1.015	1.2593	1.115×10^{-3}	4.791×10^{-4}	2.669×10^{-3}

3. Fission-product release of various types of fuel particles.
4. Graphite mechanical and creep properties.

The test element, which is designed to replace an instrumented Core 2 fuel element, has the same external appearance as a standard Core 2 fuel element. The top reflector, sleeve, and bottom connector components that form the outer containment envelope of the test element are standard Core 2 fuel element components. The lower reflector and internal trap are very similar to the standard fuel element components. The active fuel portion is made up of three 31-in.-long cylindrical fuel bodies that replace the fuel compacts in the Core 2 fuel element. The fuel bodies contain eight fuel holes equally spaced on a circular pattern and are filled with coated fuel particles. The fuel holes are closed with graphite plugs to provide complete containment of the fuel. The sleeve, top reflector, and bottom connector assembly are also completely closed to form a second containment of the fuel to be tested.

The test element is cooled by the flow of helium in the tricuspid channels formed by it and the adjacent elements as in the remainder of the core. It is purged by a flow of helium between the sleeve and the fuel bodies similar to the purge flow in the standard fuel element.

The assembly of the test element is very similar to that of standard elements. Braze sleeve/bottom connector assemblies are obtained from standard Core 2 assemblies. The internal trap and lower reflector are loaded into the bottom of the sleeve, followed by the stack of fuel bodies. The upper reflector assembly is then threaded and cemented to the sleeve.

The uranium loading of the element is 237 g, and the irradiation is scheduled for a total of 900 effective full-power days (EFPD). It is currently planned that, for the first 300 EFPD, the element will be in core location A07-07, and for the remaining 600 EFPD, it will be in core location C02-01. The maximum heat generation rate is 140 kw, and the maximum burnups are predicted to be about 20% FIMA in (Th,U)C₂ fissile particles with a Th/U ratio of 2 and about 60% FIMA in some buffer-diluted UC₂ fissile particles. The maximum fast fluence at the end of three years is calculated to be about 4.2×10^{21} n/cm², about half that to be accumulated during the core lifetime of a large HTGR. The maximum fuel bed temperatures will be 1250°C.

Fuel Body

The fuel body shown in Fig. 8.3 is a graphite circular cylinder with an array of eight fuel holes in a telephone-dial pattern. The cylinder is 31.0 in. long and 2.739 ± 0.001 in. in diameter. The eight fuel holes, each with a diameter of 0.500 ± 0.001 in., are equally spaced on a 1.80-in. bolt circle. The fuel holes are approximately 29.9 in. deep. A central hole, 1.1 in. in diameter, is provided for testing various small samples each enclosed in a graphite container, under controlled temperature and irradiation conditions.

Basically the test element is purged in the same manner as are the Core 2 elements. Eight equally spaced axial purge grooves on the outer surface of the fuel body provide adequate flow area for the helium purge gas. The grooves are each about 0.05 in. deep and 0.10 in. wide. The ends of the fuel

Fig. 8.3--Cross section of the fuel test element

body cylinders are chamfered to provide a continuous axial flow path irrespective of the azimuthal orientation of the fuel bodies. All the purge gas flows in the purge grooves and the annular clearance gap between the fuel body and the sleeve; none flows down through the central irradiation hole. The total purge flow rate is determined by the pressure drop in the porous plug and is not limited by the cross-sectional area of the purge channel.

A design objective for the fuel bodies is the accommodation and relief of possible high interactive forces between the fuel beds in the fuel hole and the graphite fuel body, while still maintaining total containment of the fuel within the fuel body. The fuel body has therefore been designed to yield toward the central irradiation hole should the graphite be subjected to stresses that exceed its tensile strength. This preferential failure mode is obtained by the addition of eight axial grooves in the central irradiation hole to control the relative ligament thicknesses between the fuel holes and the outside and inside surfaces of the fuel body. The inner graphite ligament (radial distance between central hole and fuel hole) has less than half the thickness of the outer graphite ligament (radial distance between outer surface and fuel hole). The grooves, which are equally spaced around the central hole, greatly reduce the compressive strength of the inner ligament or cylinder of the fuel body and provide a definite path for failure in the event of high loads.

Each fuel hole of the fuel body is closed at the top with a 0.25-in.-thick threaded graphite plug. The plug provides positive containment of the fuel particles in the graphite fuel body. The central irradiation hole is also closed with a threaded graphite plug, 0.38 in. thick. The minimum graphite thickness between the fuel holes and the outside surface of the fuel body is 0.2 in. The minimum thickness of the graphite below the fuel hole or central hole is 0.3 in. All of the central irradiation samples are in individual closed graphite containers. Thus, the fuel in the fuel holes is doubly contained from the helium coolant stream, and the central irradiation test samples are triply contained.

Fuel Beds

The fuel holes in the fuel test element will be loaded with fuel rods 0.490 in. in diameter by 2 in. long, and a different fuel combination will be employed in each of the bodies. The uppermost fuel body will contain TRISO coated $(\text{Th,U})\text{C}_2$ and ThC_2 particles; the center body will contain TRISO coated UC_2 particles and BISO coated ThC_2 particles; the lowermost body will contain BISO coated $(\text{Th,U})\text{C}_2$ and ThC_2 particles. These three combinations are the ones that are likely to be bonded in a large HTGR. All the fuel rods will be made by hot injection molding in production equipment, and three different bonding matrices, consisting of two kinds of manufactured isotropic graphite and a natural flake graphite filler, all with pitch as the binder, will be used.

The loading of fissile and fertile particles in each rod type is dictated by the over-all loading of 237 g of uranium. Approximately 67 g of buffer-diluted UC_2 particles or 83 g of $(\text{Th,U})\text{C}_2$ particles with a Th/U ratio of 2 will be required per 30-in. hole. After irradiation, representative samples of each of the types of fuel rods will be examined visually and metallographically to determine the effects of irradiation dose and temperature.

Central Irradiation Samples

Experimental samples will be contained in the central 1.10-in.-diameter hole of the element. In general, these samples can be divided into five types:

1. Thermal stability (fuel particles).
2. Fission-product release (fuel particles).
3. Metallic diffusion.
4. Fuel rods.
5. Graphite.

Although the designs of the various specimens are different, they will all be contained within cylindrical H-327 graphite crucibles having sufficient clearance that there will be no interference with the graphite body at any time during irradiation. By using these crucibles, all the center samples are in effect triply contained, and the more critical experiments (thermal stability and fission-product release) will employ smaller crucibles inside these containments. The purpose and description of each type of sample are presented below. The experimental variables being investigated in each type of sample are presented in Table 8.2.

Thermal Stability. The purpose of the thermal stability experiments is:

1. To expose fuel particles to a wide range of operating variables, particularly high temperatures and temperature gradients.
2. To determine the time-at-temperature lifetimes of various types of fuel.
3. To obtain data that can be integrated with other work to identify causes of particle failure and improve particle lifetimes.

High temperatures and high thermal gradients are achieved with the sample configuration shown in Fig. 8.4. Particles are loaded individually into separate holes of the rectangular graphite crucibles in the upper portion of Fig. 8.4, and these are assembled as a cruciform. This cruciform shape slides into the cylindrical crucible, which is then loaded with loose particles and a lid screwed into place. The amount of uranium in these particles and the axial position of the crucible in the element determines the maximum temperature and temperature gradient seen by the particles in the cruciform. Several of the 1/2-in.-high crucibles will be assembled within each 1-in.-diameter crucible. The maximum temperature of these crucibles will be 1500°C, and temperature gradients will be up to 1000°C/in.

Particle Fission-product Release. The purpose of the fission-product release experiments is to obtain:

1. In-pile fission-product release data for BISO and TRISO fuel particles.
2. Out-of-pile fission-product release data for BISO and TRISO particles.
3. Distribution coefficients of fission products between kernel and coatings.
4. The effect of different types of kernels on fission-product release (oxides versus carbides).

Table 8.2

FUEL TEST ELEMENT EXPERIMENTAL VARIABLES

Thermal Stability

BISO, TRISO
Oxide, carbide
ThC₂, UC₂, (Th,U)C₂
Processing variables
H₂ content
Fe content
Th:U ratios

Metallic Diffusion

Elements (Sr, Cs, Ba, Sm)
Element concentration
Effect of other metals on migration rates of individual metals
Effect of getters

Fission-product Release

ThC₂, UC₂, (Th,U)C₂
ThO₂, UO₂, (Th,U)O₂
Silicon doping

Fuel Rods

High and low packing fraction
Various filler materials

Graphite

Isotropic, H-327 graphite
Parallel and transverse orientation
Edge and center of log

Fig. 8.4--Thermal stability samples

5. Thorium and uranium migration in kernel and coating as a function of temperature.
6. The effect of metal doping (silicon) of the coatings on the fission-product release.

The design of the crucibles for the fission-product release samples is shown schematically in Fig. 8.5. A few hundred loose particles will be loaded into a graphite crucible approximately 3/8 in. in diameter and 1/2 in. high. This will be loaded into another larger graphite crucible and will be surrounded by graphite flour to absorb fission products. The larger crucible will in turn be contained within a can of niobium. The objective of the Nb can is to retain all the fission products released during irradiation within the crucible. Each Nb can assembly will fit within a graphite crucible approximately 1.25 in. long by 1 in. in diameter.

Metallic Diffusion. The purpose of the metal diffusion studies is to obtain:

1. Migration rates for Sr, Cs, Ba, Sm, etc., in graphite under reactor conditions as a function of metal concentration and temperature.
2. Data on the effects of the presence of other fission-product metals on the migration rate of individual metals (especially Sr).
3. Data on the ability of getter materials to reduce metal migration.
4. Data on the integrity of getter materials.

The graphite crucibles with an annular space for the diffusion samples are shown in Fig. 8.6. The graphite crucibles will be 1.25 in. long, will have an outside diameter of 1 in., and will probably be coated with vapor-deposited niobium on both ends. The annular space will contain various metals (such as Sr, Cs, Ba) in the carbide form, and will be mixed in known proportions in a graphite matrix material.

Fuel Rods. The purpose of these fuel rod experiments is to evaluate the performance of various types of fuel rod concepts to a dose of 4×10^{21} n/cm² and to determine the best over-all fuel rod concept.

The bonded rods will be 1/2 in. in diameter by 3/4 in. long. Each rod will be inserted in a 1-in.-diameter graphite crucible with a screw-on top.

Graphite. The purpose of the graphite experiments is to obtain additional data on:

1. The effect of temperature and irradiation dose to 4×10^{21} n/cm² on the tensile properties of various graphites.
2. The dimensional change behavior of various graphite samples.
3. The transient creep characteristics of various graphites after irradiation to 4×10^{21} n/cm².

The tensile test specimens will be cylindrical in shape, while the dimensional change specimens will be flat rectangular solids. The transient creep specimens are shown in Fig. 8.7. All the specimens will be contained in 1-in.-diameter graphite crucibles as shown in the plan view of Fig. 8.7.

Fig. 8.5--Fission-product release samples

Fig. 8.6--Metallic diffusion samples

Fig. 8.7--Graphite samples

Task IX
ADVANCED COATED PARTICLE FUELS

COATED PARTICLE IRRADIATIONS

A series of capsule irradiations is being conducted to evaluate advanced types of coated particles under potential HTGR irradiation environments. These tests are directed toward the evaluation of fuel particles containing diluents other than thorium, which could lead to reductions in HTGR fuel cycle costs.

Capsule P13L

Capsule P13L is the second test of coated diluent particles. Both carbide and oxide diluent and fertile particles are being tested, with both BISO and TRISO coatings on the diluent particles and BISO coatings on the fertile particles. (Samples of reference TRISO coated fissile and fertile carbide particles are 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). Details of the sample selection were given in previously quarterly reports (GA-9227, GA-9372, and GA-9815).

The capsule is in its fifth cycle of irradiation in the ETR. Most samples in the high-temperature cells continue to operate considerably below their design temperatures. Temperatures in the 1500°C cell (cell 1) are 925° to 1125°C; and temperatures in the 1300°C cells (cells 2 through 4) are 925° to 1175°C. Samples in the 900°C cell (cell 5) are close to design temperature. The capsule has received an estimated fast fluence of 5.7×10^{21} n/cm² (E > 0.18 Mev) to date, with an estimated burnup of 74% FIMA for the UC₂ particles. The capsule is scheduled to be irradiated for eight cycles in the ETR to a fast fluence of 8×10^{21} n/cm² and a burnup of 75% FIMA for the diluent particles.

Task XI
GRAPHITE RESEARCH

IRRADIATION EFFECTS IN CARBON AND GRAPHITE

Comparative dimensional data from irradiation experiments that included both high-temperature isotropic (HTI) and low-temperature isotropic (LTI) carbons show that at high fluences the volume expansion and the accelerations in the dimensional change rates are considerably more pronounced for HTI carbons than for LTI carbons. Alloying LTI carbons with up to 15 wt-% silicon enhances the dimensional stability slightly at about 600°C but has no effect on the behavior in irradiations at higher temperatures. For HTI carbons the effects of the alloying additions are more pronounced. In neither case, however, are the improvements thought to be of practical importance.

Data from both multiple irradiations at different temperatures and irradiations in which the temperatures were programmed down during the course of the irradiation show that prior irradiation at high temperature markedly increases the dimensional changes that occur in subsequent irradiations at lower temperature. The implications of these findings to reactor design are discussed in a topical report (Ref. 1) which presents the detailed results of the investigation.

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 B₄C plus graphite) and hafnated (HfC plus graphite) compacts. Dimensional changes in boron fibers and density changes in loose granules of B₄C and HfC will also be determined in piggyback samples 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 or hafnated graphite should result from this investigation.

Irradiation-induced dimensional change is the phenomenon of primary interest; however, there will also be correlation of thermal-expansion coefficient, strength, and microstructure with neutron exposure. Some specimens irradiated previously in capsule BG-1 (reported in a previous quarterly report, GA-9372) continue irradiation exposure in BG-2. A detailed summary of samples being tested in BG-2 was given in another quarterly report, GA-9660.

The capsule was inserted in the ETR and started irradiation in July 1969 (cycle 103) at temperatures of 600° to 700°C.

During this report period, irradiation of BG-2 continued at design conditions in the ETR. The capsule is currently in its fourth cycle and has accumulated a fast fluence of about 4.2×10^{21} n/cm² ($E > 0.18$ Mev). The irradiation had been originally scheduled for a total of four cycles, but it is now planned to extend it for a fifth cycle to a maximum fast fluence of about 5×10^{21} n/cm².

References

1. Bokros, J. C., and D. W. Stevens, Irradiation Behavior of Isotropic Carbons, USAEC report GA-9900, Gulf General Atomic Incorporated, to be published.