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HTGR FUELS AND CORE DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
AUGUST 30, 1974

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

DATE PUBLISHED - SEPTEMBER 30, 1974

MASTER

GENERAL ATOMIC

GENERAL ATOMIC COMPANY
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ABSTRACT

This publication continues the quarterly report series on the HTGR Fuels and Core Development Program. The Program covers items of the base technology of the High-Temperature Gas-Cooled Reactor (HTGR) system. The development of the HTGR system will, in part, meet the greater national objective of more effective and efficient utilization of our national resources. The work reported here includes studies of reactions between core materials and coolant impurities, basic fission product transport mechanisms, core graphite development and testing, the development and testing of recyclable fuel systems, and physics and fuel management studies. Materials studies include irradiation capsule tests of both fuel and graphite. Experimental procedures and results are discussed and, where appropriate, the data are presented in tables, graphs, and photographs. More detailed descriptions of experimental work are presented in topical reports; these are listed at the end of the report.

INTRODUCTION

This report covers the work performed by the General Atomic Company under U.S. Atomic Energy Commission Contract AT(04-3)-167, Project Agreement No. 17. This Project Agreement calls for support of basic technology associated with the fuels and core of the gas-cooled, nuclear power reactor systems. The program is based on the concept of the High-Temperature Gas-Cooled Reactor (HTGR) developed by the General Atomic Company.

Large HTGR systems will be placed in operation starting in the late 1970's following the operation of the 330-MW(e) prototype in 1974. Characteristics of these advanced systems include:

1. A single-phase gas coolant allowing generation of high-temperature, high-pressure steam with consequent high-efficiency energy conversion and low thermal discharge.
2. A prestressed concrete reactor vessel (PCRv) offering advantages in field construction, primary system integrity, and stressed member inspectability.
3. Graphite core material assuring high-temperature structural strength, large temperature safety margins, and good neutron economy.
4. Thorium fuel cycle leading to U-233 fuel which allows good utilization of nuclear resources and minimum demands on separative work.

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TASK 4

FISSION PRODUCT MECHANISMS

DIFFUSION STUDIES

Diffusion coefficient data for fission product metals, such as cesium and strontium, in fuel element graphite are required for use in reactor calculations. Diffusion coefficient data for cesium in graphite have been measured by several investigators. However, additional measurements are needed to obtain data for the reference (near-isotropic) graphite. Also, there is a great need for more information on the transport process of cesium and strontium in graphite and the factors that affect this process. It appears that a complicated mechanism is involved in the transport of these metals, with at least a relatively rapid surface diffusion and a relatively slow uptake by the bulk graphite. The rate and relative importance of these processes depend on the metal and the graphite, as well as the temperature, the concentration, and possibly irradiation.

An experimental study is under way to provide the needed basic data to define cesium transport through the reference graphite. No a priori assumption (such as the assumption of Fickian diffusion) is being made. A first step is the development of a reliable experimental method. The method should be simple and capable of providing reproducible information on cesium takeup by the graphite, on flux through the graphite, and on the profile within the graphite. The method should also be adaptable to in-core experiments. Some progress has been made along this line.

The primary approach is to use a small high-capacity cylindrical cesium source, a cylindrical sleeve of reference graphite around the source, and a highly sorptive cylindrical sink surrounding the graphite sleeve. End plugs and caps complete the two layers around the source.

Char-loaded (high sorptivity) graphite serves as the sink and as the basis for the source. It is loaded with cesium tracer from the oxalate by heating for 36 hr at 1100°C. All fits between the graphite parts are tight so as to maximize surface and gap resistances. The overall dimensions are 5/8-in. diameter by 9/8 in. long, and can readily be increased as desired. After annealing under the desired conditions, the activity of all parts is determined to verify the material balance. The activity of the sink (and caps) measures the flux through the graphite; the activity of the sleeve and plugs measures the takeup. The sleeve can be machined to obtain concentration profiles or sectioned to obtain the profiles by radioautography.

Exploratory experiments showed that the flux from source to sink was indeed determined primarily by the resistance of the sleeve and not by source or sink effects. For this purpose the sleeve was eliminated and a snugly fitting sink was placed directly around the source. The flux was thus consistently increased by 10 to 20 fold, showing that the graphite resistance was more than 90 to 95% of the total in the standard arrangement.

Reproducibility, even using the same source and sleeve, was poor. Experiments in which two experimental assemblies were annealed at the same time indicated that the annealing conditions were responsible for much of the variance in the results because the flux in both assemblies tended to vary in unison. Originally annealing was conducted under continuous evacuation with a diffusion pump protected by a charcoal - liquid nitrogen trap. It was suspected that the problem arose from insufficient definition of the residual atmosphere, and experiments were conducted in the presence of helium, hydrogen, and carbon monoxide. Significant differences in the flux were found, with helium (presumably due to impurities) and carbon monoxide giving faster fluxes than vacuum and hydrogen. A new series using a mixture of 10% CO and 90% H₂ has been started in an attempt to eliminate this variable while maintaining conditions close to those of the HTGR.

Takeup by the graphite showed much less dependence on the atmosphere than the flux and was also surprisingly small compared with the flux. The takeup varied from 1/4 to 1/50 of the flux for a 1/8-in. (3.2 mm) cylindrical wall in about 2 hr at 900°C. This clearly confirms that takeup and flux are separate processes which occur simultaneously but at different rates.

An alternative approach, better suited to obtaining insight by radioautography and conducting comparative experiments on different graphite samples, involves a stack of flat discs held in a tight graphite tube fitted with end screw caps. One of the discs is the source. Samples can be placed on both sides of the source followed by sinks. It is also possible to press fit small sample cylinders into axial holes in one of the sample discs and to observe the pattern of cesium penetration and uptake on radioautographs of the faces of the sample and the sink. Some of the results obtained by this method are quite suggestive but others show that the geometry, which is not as well defined in this arrangement, and the effects of the sample surface require additional study before drawing conclusions.

TRANSPORT CODE DEVELOPMENT

Introduction

The FIPER series of codes are used for calculating the release of metallic fission products into the primary coolant circuit of HTGR systems. These codes were developed at GA under the AEC Base Program. They take into account the combined effects of (1) the retention of fission product metals by fuel particles, (2) sorption of the metals by the fuel rod matrix material and graphite of the fuel elements, (3) evaporation of the metals from the graphite surface into the helium coolant through a boundary layer, (4) decay during diffusion transport, and (5) changes in fission rate (power), fuel temperature, and graphite temperature with time. Two forms of the FIPER code are presently utilized: (1) the FIPER Q code, which is

the basic, rigorous version, and (2) the FIPER S code, which is an interim version.

The FIPER S code was designed to rapidly compute metallic fission product release at a great many spatial positions in an HTGR core. The need for such a code, as opposed to the more rigorous FIPER Q solution, is due to the fact that release is a highly nonlinear function of temperature and temperature history. Temperatures vary in an irregular manner with space and time in the HTGR core, and a great many space points must be sampled in order to integrate the results and to locate critical areas for design purposes.

Experience in using FIPER S has disclosed that the code contains a number of questionable assumptions and approximations that can lead to incorrect and very conservative release calculations. Shortcomings have been identified with respect to boundary conditions, mass conservation, radioactive decay, thermal diffusion, and transverse flow. As a result of these deficiencies, a completely new model called TRAFIC (TRANsport of FISSION products in the Core) is being developed. TRAFIC suffers from none of the above shortcomings. In addition, it will eventually also account for mixed species interactions on the coolant hole boundary and axial stacking of the blocks.

The fundamental differences between TRAFIC and FIPER S are (1) a more comprehensive differential equation that includes more effects, and (2) a more rigorous solution with realistic, nonlinear boundary conditions. As with FIPER S, the solution is closed form in order to enhance calculational efficiency. As a result, a few simplifications have been required to remove nonlinearities and to make a closed-form solution possible. However, in the case of TRAFIC the simplifications are justifiable and physically reasonable.

The first stand-alone version of TRAFIC has been completed. This version uses the same thermal models and particle release models as were used in the interim FIPER S code. Transverse flow (a minor effect) has not yet been included.

General Description of TRAFIC Model

TRAFIC computes block-normalized releases using a typical symmetrical web unit as the geometrical model (Fig. 4-1).

Fig. 4-1. Geometry of symmetrical web unit

The fuel rod is a compact of many small coated fuel particles bonded in a carbon matrix. The particles consist of TRISO coated fissile particles, which achieve a high burnup ($\sim 75\%$ FIMA), and BISO coated fertile particles, which achieve a relatively low burnup ($\sim 7.5\%$ FIMA). Fission products accumulate in both types of particles. They can be released due to particle failure or by diffusion through the particle coatings (BISO only). The particle release models in the current version of TRAFIC are the same as those in the interim FIPER S code: (1) empirical failure fraction curves for TRISO and BISO particles, and (2) a BISO diffusion calculation based on Duhamel superposition of a response function calculated by FIPER Q.

Fission products released from the particles diffuse quite rapidly through the fuel rod matrix material. The TRAFIC model takes advantage of the fact that the diffusion coefficient in the matrix material is considerably greater than that in the graphite region. The consequence of this is that one can derive a quasi-steady concentration profile in the rod that is independent of the current mass flux. The average concentration in the rod depends on the integrated past history of the source, the decay, and the boundary leakage. The shape of the concentration profile depends on the current temperature distribution as a result of thermal diffusion effects.

From the fuel rod solution, plus a knowledge of the source, decay, and leakage effects, the concentration C_s at the surface of the fuel rod can be determined. This surface concentration is in a state of thermodynamic equilibrium with the vapor pressure within the gap. Similarly the same vapor pressure, which is uniform throughout the gap, is also in equilibrium with the surface concentration at the fuel hole surface of the graphite. There can be large differences between the concentrations at the fuel rod surface (C_s) and at the graphite surface (C_o). These differences are caused by the fact that these surfaces are at different temperatures and the fact that the fuel rod and the graphite can have greatly different sorbtivity properties. These sorbtivity properties (vapor pressure versus concentration functions — the so-called Freundlich and Langmuir isotherms) are input into TRAFIC, and the code automatically computes the concentration ratio

$$\phi' = \frac{C_o}{C_s} \quad . \quad (4-1)$$

This ratio is often designated the sorption ratio.*

* The concentration (or sorption) ratio is given the prime notation because the fuel rod and graphite surfaces are at different temperatures. The symbol ϕ without the prime applies where the two surfaces are at the same temperature.

Having been transported through the gas gap by the above mechanism, the fission product concentration C_o accumulates on the graphite fuel rod hole surface. Since the concentration on the coolant side C_L is lower (initially zero), this introduces a concentration gradient in the graphite web. The concentration gradient acts as a driving force that results in a diffusive mass flux across the graphite. TRAFIC employs a reasonably rigorous transient diffusion calculation in the graphite region, which includes the effects of thermal diffusion and radioactive decay. A quasi-steady solution, which is about a factor of two cheaper, is also an option.

Having reached the coolant hole surface by diffusion through the graphite, the fission products have one more barrier to cross before they escape into the coolant system. This is the coolant boundary layer. Transport through the boundary layer depends upon the vaporization characteristics of the graphite surface and upon the mass transport characteristics of the boundary layer. The former are computed in TRAFIC using the previously discussed adsorption isotherms. The boundary layer mass transport coefficient is computed from a standard empirical correlation.

TRAFIC performs diffusion and mass-transport calculations for three separate subregions: the fuel rod, the graphite web, and the coolant boundary layer. The solution in each of these subregions depends upon boundary conditions that are initially unknown. These can be reduced to five independent quantities:

1. ϕ' - concentration ratio (see Eq. 4-1).
2. C_o - graphite fuel surface concentration.
3. J_o - graphite fuel surface mass flux.
4. C_L - graphite coolant surface concentration.
5. J_L - graphite coolant surface mass flux.

These five boundary conditions are related to one another through the solutions in the three subregions and through the nonlinear adsorption isotherms. It is possible to reduce the problem at any given time step to

the solution of five simultaneous nonlinear equations in five unknowns. At each time step TRAFIC sets up this system of equations and solves them by an iterative, Newton-Raphson type of procedure. The resulting solution is one that satisfies all the differential equations with a consistent set of boundary conditions.

Summary of Theory

The details of the TRAFIC model will be thoroughly derived and explained in a forthcoming topical report. This section gives a brief summary of the theory.

Basic Diffusion Equation

The basic diffusion equation is a modification of Fick's Law that accounts for diffusion due to both a concentration gradient and a temperature gradient:

$$\vec{J} = - \left(D \vec{\text{grad}} C + \frac{CQD}{BRT^2} \vec{\text{grad}} T \right) , \quad (4-2)$$

where J = mass flux

D = diffusion constant

C = mass concentration

B = thermal diffusion parameter

R = gas constant

T = temperature

Q = activation energy

D, B, and Q are defined through empirical formulas.

Conservation of Mass

By considering the conservation of mass in a small differential volume of the system, the following differential equation is obtained:

$$\frac{\partial}{\partial t} C(x,t) = S(x,t) - \lambda C(x,t) - \text{div } \vec{J}(x,t) \quad , \quad (4-3)$$

where S = source rate of mass creation

λ = decay constant

Assuming that the temperature field is known, Eqs. 4-2 and 4-3 can be combined to give a second order partial differential equation with the concentration $C(x,t)$ as the only unknown.

Fuel Rod Solution

The fuel rod is represented by an axisymmetric cylindrical geometry. The diffusion constant D_1 in the fuel rod region is assumed to be very large compared with the diffusion constant D_2 in the graphite region. As a result, the mass flux $\vec{J}$ is limited almost entirely by resistance in the graphite, and the steady-state concentration profile, neglecting thermal diffusion, appears as shown in Fig. 4-2.

Fig. 4-2. Steady-state concentration profile

The mass flux $\vec{J}$ is, for all practical purposes, unaffected by the magnitude of D_1 . Considering D in Eq. 4-2 to be very large and arbitrary and J to be known and limited, it is clear that

$$\overrightarrow{\text{grad}} C_1 + \frac{C_1 Q}{BRT^2} \overrightarrow{\text{grad}} T = - \frac{J}{D_1} \approx 0 \quad . \quad (4-4)$$

Equation 4-4 can be used to solve for the spatial shape of the concentration profile in the rod if the concentration at the fuel rod surface, C_s , is known. Solving Eq. 4-4 in cylindrical coordinates, the result is

$$C_1(r,t) = C_s \exp \left[\psi(r^2 - r_s^2) \right] \quad , \quad (4-5)$$

where

$$\psi = \frac{4Q(T_c - T_s)}{BRr_s^2(T_c + T_s)} \quad , \quad (4-6)$$

where T_c is the fuel rod centerline temperature and T_s is the fuel rod surface temperature.

The constant ψ in Eq. 4-6 is derived with the aid of the assumption that the temperature profile in the rod is parabolic. Other profiles will give slightly different values of ψ .

Having found the spatial dependency of $C_1(r,t)$ from Eq. 4-5, the time dependency can be found by integrating Eqs. 4-3 and 4-5 over the volume of the rod and computing the mass balance for the rod as a whole. The result of integrating Eq. 4-3 is

$$\frac{d\bar{C}}{dt} = S(t) - \lambda \bar{C}(t) - \frac{2}{r_s} J_s(t) \quad , \quad (4-7)$$

where J_s is the mass flux at the surface of the rod.

The result of integrating Eq. 4-5 is

$$C_s = \bar{C} \frac{\psi r_s^2}{1 - \exp(-\psi r_s^2)} \quad (4-8)$$

Equation 4-7 can be solved for $\bar{C}$ and then, with the aid of Eqs. 4-1 and 4-8, the concentration at the graphite surface, C_s , can be computed. The result is

$$C_o(t) = \left\{ \frac{\psi r_s^2}{\phi [1 - \exp(-\psi r_s^2)]} \right\} e^{-\lambda t} \int_0^t \left[S - \frac{2}{r_s} \left(\frac{A_c}{A_f} \right) J_o \right] e^{\lambda t} dt \quad , \quad (4-9)$$

where A_c is 1/12 of the coolant hole area and A_f is 1/6 of the fuel hole area. The reason for the area ratio A_c/A_f in Eq. 4-9 is that the flux J_o is calculated from a slab-geometry model for the graphite that uses A_c as the reference area. On the fuel side, J_o is a fictitious flux, and J_o times A_c is the correct total mass flow rate. This becomes more apparent in the following section.

Equation 4-9 defines a relationship between the present C_o , the present ϕ , and the past history of J_o . It is one of the five nonlinear equations used in the boundary value solution.

Graphite Web Solution

The irregular, two-dimensional web shape is modeled in TRAFIC by a one-dimensional slab geometry, as illustrated in Fig. 4-3.

The reference area for the slab is arbitrarily taken to be A_c , 1/12 of the coolant hole area for the web. The effective length L is then computed, from a two-dimensional thermal calculation, as that length which gives the same thermal conductance as the correct two-dimensional model. Thus, two-dimensional geometrical effects are modeled correctly in an overall sense, even if the fine detail of the spatial distribution is not.

Fig. 4-3. Web idealization

This assumption is also used in the more rigorous FIPER Q calculations done at GA. It is believed that local two-dimensional effects have no real influence on the gross transport phenomena, especially in view of material property uncertainties.

All the production of fission products takes place in the fuel; thus, the source term S in Eq. 4-3 vanishes in the graphite web region. Equation 4-3 then becomes

$$\frac{\partial}{\partial t} C(x,t) = -\lambda C(x,t) - \text{div } \vec{J}(x,t) \quad . \quad (4-10)$$

In slab geometry, the combination of Eqs. 4-2 and 4-10 results in the differential equation

$$\frac{\partial}{\partial t} C(x,t) = D(t) \frac{\partial^2}{\partial x^2} C(x,t) + \alpha(t) \frac{\partial}{\partial x} C(x,t) - \lambda C(x,t) \quad , \quad (4-11)$$

where

$$\alpha(t) \equiv \frac{QD(t)}{BRT(t)^2} \frac{\partial \bar{T}}{\partial x}(t) \quad (4-12)$$

is the influence of thermal diffusion in Eq. 4-2. Equation 4-11, with arbitrary time-dependent coefficients, cannot be solved in closed form. However, with a few judicious manipulations and assumptions, it can be transformed into a form where the coefficients are constant. This is done by first nondimensionalizing the equation to obtain

$$\frac{\partial C}{\partial \rho} = \frac{\partial^2 C}{\partial \xi^2} + \beta \frac{\partial C}{\partial \xi} - \gamma C \quad , \quad (4-13)$$

where

$$\rho(t) = \frac{1}{L^2} \int_0^t D(\tau) d\tau \quad , \quad (4-14)$$

$$\xi = \frac{X}{L} \quad , \quad (4-15)$$

$$\beta = \frac{\alpha L}{D} \quad , \quad (4-16)$$

$$\gamma = \frac{\lambda L^2}{D} \quad . \quad (4-17)$$

In these equations, ρ and ξ are the new independent variables and β and γ are effective material constants. It is seen that this transformation forces all time dependency to lie in the β and γ terms, which control the thermal diffusion and the decay effects, respectively. Any errors in β and γ influence only these effects, and if $\beta = \gamma = 0$ (as was assumed in the original FIPER S model), the equation is exact.

In TRAFIC it is assumed that β and γ are both constant, equal to the time-averaged values computed by a numerical integration. This gives a reasonable estimate of the average thermal diffusion and decay effects, which in any case are not large for the most important isotopes (cesium and strontium).

Having reduced the equation to one of constant coefficients, an exact closed-form solution is possible. The general solution is

$$C(\rho, \xi) = e^{-\beta\xi/2} \left(A_o e^{\theta\xi} + B_o e^{-\theta\xi} \right) + e^{-\beta\xi/2} e^{-\theta^2\rho} \sum_{i=1}^{\infty} \left(A_i \cos \omega_i \xi + B_i \sin \omega_i \xi \right) , \quad (4-18)$$

where

$$\theta = \sqrt{\frac{\beta^2}{4} + \gamma} .$$

Application of the boundary conditions and initial conditions is a tedious process and is not reported in detail here. Assuming a known flux J_L on the right and a known concentration C_o on the left, one can obtain particular solutions and then evaluate these solutions for the two unknown boundary conditions. The results have the form

$$C_L(\rho) = G_1(\rho) C_o + G_2(\rho) \frac{J_L L}{D(t)} \quad (4-19)$$

and

$$J_o(\rho) = H_1(\rho) \frac{C_o D(t)}{L} + H_2(\rho) J_L . \quad (4-20)$$

Equations 4-19 and 4-20 are strictly applicable only to the case of constant, time-independent boundary conditions. In our case, the boundary conditions vary with time. The time-dependent problem can be solved with the Duhamel superposition principle. The results are as follows:

$$\begin{aligned}
C_L(\rho) &= \int_0^\rho G_1(\rho - \rho') \frac{d}{d\rho'} C_o(\rho') d\rho' \\
&+ \int_0^\rho G_2(\rho - \rho') \frac{d}{d\rho'} \left(\frac{J_L L}{D(t)} \right) d\rho'
\end{aligned} \tag{4-21}$$

and

$$\begin{aligned}
J_o(\rho) &= \int_0^\rho H_1(\rho - \rho') \frac{d}{d\rho'} \left(\frac{C_o(\rho') D(t)}{L} \right) d\rho' \\
&+ \int_0^\rho H_2(\rho - \rho') \frac{d}{d\rho'} [J_L(\rho')] d\rho'
\end{aligned} \tag{4-22}$$

Due to accuracy problems, and because the H_1 function is singular at $\rho = 0$, the evaluation of the integrals is accomplished by integrating by parts. The G and H functions are integrated in closed form, removing the singularity.

In solving for the unknown boundary conditions at any time step, the entire past history is known. If the integrals in Eqs. 4-21 and 4-22 are evaluated by the trapezoidal rule, and if all known quantities are then lumped together, the following two expressions are obtained at the n^{th} time step:

$$C_{L_n} = A_1 + A_2 C_{o_n} + A_3 \left(\frac{J_{L_n} L}{D_n} \right) \tag{4-23}$$

$$J_{o_n} = B_1 + B_2 \left(\frac{C_{o_n} D_n}{L} \right) + B_3 J_{L_n} \tag{4-24}$$

Equations 4-23 and 4-24 constitute an application of the Crank-Nicholson implicit method for incrementally solving a differential equation in time. These equations are two more of the five simultaneous equations used to solve for the unknown boundary conditions.

Coolant Hole Evaporation

The fourth boundary condition equation relates the coolant hole mass flux, J_L , to the boundary layer concentration difference and is given by

$$J_L = h(C_c - C_\infty) \quad , \quad (4-25)$$

where h is a mass transfer coefficient given by

$$h = 0.023 D_{He} Re^{0.83} Sc^{0.44}/D \quad , \quad (4-26)$$

and where

D_{He} = helium diffusion coefficient

Re = Reynolds number

Sc = Schmidt number

D = coolant hole diameter

The concentrations in Eq. 4-25 are:

C_c = concentration in the helium adjacent to the hole surface

C_∞ = free stream concentration

C_∞ is given as an external boundary condition and C_c is computed from the adsorption-vapor pressure relationships for the graphite. The details are not given here.

Calculation of ϕ'

The last of the five boundary condition equations is an expression for calculating ϕ' in Eq. 4-1. This is computed from the adsorption-vapor pressure relationships for the fuel rod and the graphite surfaces in the gap. Thus, the vapor pressure can be calculated at the graphite surface by

$$P = f_1(C_o, T_G) \quad . \quad (4-27)$$

It can also be computed at the fuel rod surface by

$$P = f_2(C_s, T_s) \quad , \quad (4-28)$$

where

$$C_s = \phi' C_o \quad . \quad (4-29)$$

By appropriately combining Eqs. 4-27 through 4-29, it is possible to calculate

$$\phi' = \phi(C_o, T_G, T_s) \quad . \quad (4-30)$$

This is the final equation used to solve for the unknown boundary conditions.

Solution for Boundary Conditions

The five unknown boundary quantities ϕ' , C_o , J_o , C_c , and J_L are found by a simultaneous solution of Eqs. 4-9, 4-23, 4-24, 4-25, and 4-30. This is accomplished by a subroutine VECTIT, which solves an arbitrary system of nonlinear equations using the tangent-secant method, a numerical application of the Newton Raphson method.

FISSION PRODUCT DATA REVIEW

Considerable experimental data exist on fission product transport, including release, diffusion, and vaporization data. These data have been accumulated over many years as a result of measurements in the United States and Europe and are scattered in a variety of documents. In addition, continuing experiments in both the United States and Europe are providing new data. These data are critically reviewed and evaluated (1) to assure that they are being utilized to maximum advantage, and (2) to determine the adequacy of the data and the specific need for more work. Results of this work are used (1) to update the reference fission product transport data used in reactor analyses, (2) to serve as a guide for future experimental work on fission product transport, (3) to set up a reference data file, and (4) to provide recommended data for use in reactor calculations.

Presently available fission product transport data have been reviewed. Documentation of the results, in the form of an informal report, is nearly complete. The information in this report is being used in the preparation of topical reports covering (1) strontium transport data, (2) fission gas transport data, and (3) cesium transport data. The strontium report is now being written.

Strontium transport data derived from the review are summarized in Table 4-1. The materials used in the measurements are not the current reference materials for fuel kernels and graphite; thus, additional measurements on diffusion of strontium in fuel kernels and in graphite and sorption of strontium on graphite are needed for ThO_2 as a fuel kernel reference material and for near-isotropic graphite. The reference fuel rod matrix material is expected to be essentially the same as that for Fort St. Vrain on which the measurements have been made. Therefore, the sorption data for Fort St. Vrain matrix material in Table 4-1 can be considered as representative of measurements on reference material.

TABLE 4-1
SUMMARY OF PARAMETERS AND EQUATIONS DESCRIBING STRONTIUM TRANSPORT DATA^(a)

DIFFUSION ^(b)			
$D = D_o \exp(-Q/RT)$			
Diffusion In	Parameters		Conditions
	D_o (cm ² -sec ⁻¹)	Q (cal-mole ⁻¹)	
Fuel kernels	8.9×10^{-6}	50,000	FIMA $\leq$ 20%
Graphite	165.6	64,000	$C_d \ 0.7 \leq \mu\text{mole g}^{-1}$
Graphite	$337.9 C_d^2$	64,000	$C_d \ 0.7 \leq \mu\text{mole g}^{-1}$

SORPTION ^(b)					
$\ln p = (A + B/T) + (D - 1 + E/T) \ln C_d + \ln C \quad C_d \geq C_t$ $\ln p = (A + B/T) + (D - 1 + E/T) \ln C_t + \ln C \quad C_d < C_t$					
Sorption On	Parameters				
	A	B	D	E	C_t ($\mu\text{mole g}^{-1}$)
Fuel rod matrix	42.8	-1.49×10^5	-8.52	2.85×10^4	22.8
Graphite	9.44	-4.24×10^4	-0.57	4.59×10^3	0.12

(a) T = temperature ($^{\circ}\text{K}$); $R = 1.987 \text{ cal-mole}^{-1} \text{ }^{\circ}\text{K}^{-1}$.

p = vapor pressure of strontium in atm.

C_d = concentration of sorbed strontium plus barium in $\mu\text{mole per gram of graphite}$.

C = concentration of sorbed strontium in $\mu\text{mole per gram of graphite}$.

C_t = concentration of sorbed strontium plus barium at transition between
Freundlich and Henrian regions.

(b) The diffusion and sorption data were derived primarily from measurements using carbide fuel kernels, Fort St. Vrain fuel rod matrix material, and H-327 graphite.

A mixed isotope effect for strontium diffusion in graphite and sorption in graphite and fuel rod matrix material is taken into account for concentration levels of strontium plus barium that exceed the concentration limits prescribed below. Neglecting the mixed isotope effect would lead to an underestimate (i.e., a nonconservative estimate) of strontium release. For graphite, barium is more strongly bound than strontium and for this reason is assumed to have a significant effect on the diffusion and sorption of strontium. Cesium, which is bound less strongly than strontium, is neglected. The relative binding strengths of strontium, barium, and cesium are assumed to apply also to the fuel rod matrix material.

For the diffusion of strontium in graphite, a dependence of the diffusion coefficient on the concentration is taken into account when the concentration of strontium plus barium exceeds $0.7 \mu\text{mole}$ of diffusant per gram of graphite. In this case the second set of parameters for diffusion, as listed in Table 4-1, is used in calculating the diffusion coefficient of strontium in graphite. The pre-exponential term in the expression for the diffusion coefficient has a dependence on the square of the concentration.

For the sorption of strontium, the concentration range of strontium is divided into two regions in which Freundlich and Henrian descriptions of sorption are considered to apply. The concentration of strontium, $C = C_t$, divides these regions. The assumption is made that C_t is independent of temperature. When the concentration of strontium plus barium exceeds C_t , the equation of Table 4-1 for the vapor pressure of strontium includes the mixed isotope effect as outlined above.

TRITIUM PERMEATION RATE THROUGH STEAM GENERATOR MATERIALS

The principal sources of tritium in HTGR systems are ternary fission and neutron activation of (1) He-3 present in the helium coolant, (2) Li-6 and Li-7 present as impurities in the core and reflector materials, and (3) B-10 in the control rods and reflector materials. Other tritium-producing reactions are relatively unimportant. The main concern with

tritium is its release to the environment via permeation through the steam generator tube walls into the secondary coolant.

The permeation of tritium through steam generator tube walls is often calculated by using hydrogen permeation data (Refs. 4-1, 4-2), with appropriate corrections for the mass difference between tritium and hydrogen and taking into consideration the tritium-to-hydrogen ratio in the helium coolant. Assumptions are made that the permeation rate is inversely proportional to the thickness of the diffusion path and no surface film exists on either the helium side or the steam side of the steam generator tubes. In the actual case, these assumptions may not be true.

The objective of the work in this subtask is to measure the tritium permeation rates from a tritium-helium mixture through T-22 steel, Incoloy-800, iron-chromium alloys of various chromium content, and low-carbon steel at temperatures of HTGR interest in order to provide a direct basis for calculation of the tritium permeation rate through steam generator tubes. Steam will be present on the tritium exit side in some of the experiments. Both the tritium-helium mixture and the steam will be at 1 atm pressure. Variables to be included in the study are temperature, hydrogen partial pressure, tritium-to-hydrogen ratio, impurity contents (e.g., H_2O , CO , and CH_4) in the helium, and wall thickness of the tubular samples.

The work planned for FY-75 will include (1) setup and checkout of the experimental arrangement, (2) study of the effect of tritium concentration in helium and sample wall thickness on the permeation rate through T-22, (3) study of the effect of hydrogen partial pressure and impurity contents in helium on the permeation rate through T-22, and (4) study of the effect of the presence of steam on the tritium exit side on the permeation rate through T-22.

Experimental

Figure 4-4 shows the experimental arrangement for the study of the tritium permeation rate from a tritium-helium mixture through a tubular sample into a stream of helium carrier gas. The tubular samples used in these studies will be about 15 cm long and 2 cm outside diameter, with wall thicknesses of 1, 2, and 3 mm. The samples will be prepared from reference HTGR steam generator materials by swaging, machining, appropriate heat treatments, and surface cleaning.

During measurement of the tritium permeation rate, the sample will be contained in a ceramic tube that is maintained at the desired temperature in a resistance furnace. The temperature distribution will be measured by four sets of chromel-alumel thermocouples equally spaced along the length of the sample. The temperature gradient will be minimized by adjusting the Nichrome winding of the resistance furnace and by using suitable heat chokes along the sample support. The sample and the ceramic container will be thoroughly outgassed at high temperature prior to the introduction of the tritium-hydrogen-helium mixture, which flows continuously around the sample during the run.

The impurity contents (e.g., H_2O , CO , H_2 , and CH_4) of the helium will be measured with a gas chromatograph and a moisture monitor. The tritium content in the helium will be determined by passing the gas through a CuO converter, trapping the tritium water formed in a cold trap, and counting the activity of the content of the trap with a liquid scintillation counter. These measurements will be made at both the entrance and the exit of the ceramic tube containing the sample.

The tritium that permeates through the wall of the sample will be swept out with a stream of high-purity helium gas and analyzed for impurity and tritium contents by the techniques described above. The permeation rate will then be calculated from the sample geometry and the tritium activity in the cold trap as a function of the duration of the run. The

Fig. 4-4. Apparatus for study of tritium permeation through steam generator materials

impurity contents, the hydrogen partial pressure, and the tritium-to-hydrogen ratio in the helium will be varied to provide information on the effect of these parameters on the tritium permeation rate in the temperature range of interest to HTGR steam generator application. For some of the experiments, steam will be introduced into the tritium exit side of the sample as a carrier gas for the tritium that permeates through the wall of the sample. The tritium activity in the condensate will be determined by liquid scintillation counting. This will serve as a check of the effect of the steam-metal reaction, if any, on the tritium permeation rate.

Current Work

The main efforts during this quarter were devoted to design and assembly of the experimental apparatus and preparation of the T-22 samples.

Assembly of the tritium permeation apparatus is about 80% completed. The permeation chamber and the associated valves and gas lines have been leak checked and found to be leak free. The moisture monitor has been received, and the gas chromatograph is scheduled for delivery by the middle of September 1974.

The furnace for the permeation study was specially wound to minimize the temperature gradient along the sample. The temperature control and monitoring systems have been connected to the furnace and checked out.

The tritium trapping system and the helium-tritium source have been connected to the permeation chamber. The entire system should be ready for shakedown tests during the third week of September 1974.

Two T-22 samples have been prepared. Prior to machining, the material was heat treated at 1575°F for 1 hr, furnace-cooled at a rate of 50°F/hr to 1200°F, and then removed from the furnace and air-cooled. The finished samples were 6 in. long with a wall thickness of 0.0394 in. One of these samples has been installed in the diffusion chamber.

EFFECT OF FUEL HYDROLYSIS ON FUEL EXPANSION

A potential consequence of carbide hydrolysis or oxidation is expansion of the affected fuel. This effect is believed to be of little significance for normal plant operations because only 5% of the total fuel loading is UC_2 (the remaining 95% is ThO_2) and the design value for failed fuel is 10%. On the average less than 0.5% of the fuel is subject to hydrolysis; therefore, the effects of expansion should be negligible. There is a potential, however, for local fuel particle failure fractions greater than the core average, and should failed fertile ThO_2 particles convert to carbide via the reaction $ThO_2 + 4C = ThC_2 + 2CO$, hydrolysis could produce significant stresses in graphite elements in local areas of the core.

The fuel hydrolysis reaction is represented by

The hydrate product is less dense than the original ThC_2 , which gives rise to expansion of the affected fuel. The degree of expansion is a function of failed fraction, temperature, water vapor concentration, and burnup.

An experimental program is planned to study the expansion characteristics of hydrolyzing HTGR fuels and the concomitant stresses produced in constraining graphite fuel elements. This information is needed to define operational or technical specification conditions for plant operation, particularly in the event of steam ingress accidents.

Experimental Procedure

The experimental apparatus is shown schematically in Figs. 4-5 and 4-6. The overall arrangement is shown in Fig. 4-5, and details of the load cell arrangement are shown in Fig. 4-6. The test graphite crucible is machined and drilled to simulate a section of a large HTGR fuel element and includes a central fuel hole surrounded by three fuel holes and three

Fig. 4-5. Fuel hydrolysis/expansion apparatus

1. LOAD CELL	6. CLAMSHELL FURNACE
2. STEEL ROD	7. FUEL SAMPLE
3. VACUUM PORT	8. GRAPHITE RODS (SLIP FIT)
4. GAS OUTLET	9. GRAPHITE CRUCIBLE
5. QUARTZ ENCLOSURE	10. GAS INLET PLENUM

Fig. 4-6. Fuel expansion/load cell apparatus

coolant channels. The central fuel hole will contain the fuel specimen and the other fuel holes will be filled with dummy graphite rods. The test crucible is contained in a vertical quartz tube, which is heated from the outside by ordinary clamshell heaters and is fitted on the top and bottom with water-cooled metal plates. Gas-tight seals between the plates and the quartz tube are accomplished by flexible silicone sealout, or a similar means.

The crucible rests on a hollow steel pedestal which in turn rests on the bottom plate. A graphite rod is inserted into the top of the central fuel hole and rests directly on the fuel sample. The graphite rod is connected to a steel rod, which penetrates the upper plate through a sliding O-ring seal and is attached to the load cell. The load cell is held firmly in place by a steel plate, which is connected to the bottom plate by threaded 3/4-in. steel rods. With this arrangement, loads induced in the fuel sample are communicated directly to the stationary load cell via the central steel rod. The dc response of the load cell will be calibrated for the 1 to 2000 lb range and will be continuously recorded on a strip chart recorder.

The quartz enclosure is provided with a gas inlet in the lower plate so that He-H₂O mixtures can be purged through the coolant channels of the test crucible. The outlet of the system is plumbed to a sensitive flame ionization hydrocarbon analyzer to permit analysis of the hydrolysis products methane, ethane, and propane. With this arrangement the rate of hydrolysis, as measured by gas analysis, can be directly compared with the rate of load development.

A further refinement of this method is to utilize slightly irradiated fuel and to monitor noble gas release with an ionization chamber simultaneously with hydrocarbon release and load development. This arrangement would yield additional information on the hydrolysis process and would also yield data on hydrolysis-induced release of noble gases from a fuel geometry which simulates that of the reactor core.

Work Scope

The work scope as presently conceived includes measurements of hydrolysis rates and load development as a function of temperature (100° to 900°C), water vapor concentration (100 to 30,000 ppm), and the fraction of failed particles (10 to 100%). To obtain data on all of these parameters, three test rigs will be built with automatic stream selection and gas sampling equipment for continuous gas chromatographic analyses. This equipment has been designed, and all long-lead items have been ordered.

Preliminary Results

Preliminary experiments are under way using a temporary rig and a borrowed load cell. Tests with this apparatus have proven the feasibility of the experimental design. The results of a calibration run using a bed of pure ThC_2 particles at 300°C are shown in Fig. 4-7. In this test the relatively thin-wall graphite crucible (wall thickness = 0.18 in.)* failed after an exposure of 64 hr at a measured load of 2700 psi. This load can be converted to a hoop stress of 6194 psi, which is close to the predicted failure point. The hydrolysis rate as measured by ethane formation was quite constant throughout the test until the time of failure, where the rate increased by a factor of 15. This result clearly indicates that the rate of reaction is mass transport limited, which is expected for unirradiated fuel.

EFFECT OF FUEL HYDROLYSIS ON FISSION GAS RELEASE (R/B)

Previous studies (Ref. 4-3) have determined the effect that high-temperature (>700°C) fuel hydrolysis has on the release of Kr-85m. These studies utilized special fuel rods prepared from (1) unirradiated, bare, uncoated UC_2 kernels (93% enrichment) or (2) laser-failed (drilled), unirradiated TRISO UC_2 particles (93% enrichment). It was shown that

*The wall thickness of a simple graphite crucible equivalent in strength to the large HTGR fuel element web would be 0.25 in.

Fig. 4-7. Pressure versus time during hydrolysis of pure ThC_2 particles contained in graphite crucible

exposure of these fuel rods to 100-ppm water vapor for up to 200 hr resulted in an increase in the steady-state release of Kr-85m.

During this quarter these fuel rods have been exposed to water vapor for an additional 350 hr. In addition, an experiment has been initiated to examine the effect of hydrolysis at lower temperatures. In this experiment fuel rods have been exposed to wet helium at 300°, 400°, and 500°C for up to 100 hr.

In another new experiment, the effect of prior irradiation is being examined. A sample of irradiated fuel compact (D1305-29) material from Peach Bottom Core 1 has been exposed to 100-ppm water vapor at 900°C for 663 hr. Essentially all of the particles were failed in the compact material. The particles were PyC-coated (1U, 5Th)C₂ irradiated to ~4% FIMA burnup.

In all the experiments, the steady-state Kr-85m release rate (R/B) at 1100°C was measured prior to the hydrolysis exposure and then at intervals during the exposure.

Experimental

The fuel rods were prepared as described in Ref. 4-4. The quantity of U-235 in each rod was calculated from the number of fissions during irradiation as determined by the content of La-140. The apparatus and procedure used to conduct the hydrolysis reactions of the fuel rods are similar to those previously described (Refs. 4-3, 4-4).

The sample of Peach Bottom fuel compact was hydrolized in a TRIGA pool-side King furnace, which had been plumbed to allow continuous volumetric flow of wet (100-ppm water) helium past the sample. The water concentration was measured with an electrolytic-type water analyzer. A flow diagram of the King furnace and auxiliary apparatus is shown in Fig. 4-8.

1. HELIUM SUPPLY
2. LOW-PRESSURE SOLENOID VALVE, MELETRON CONTROLLED
3. LN₂-COOLED He CLEANUP TRAP
4. MANUALLY OPERATED VALVE
5. MOISTURE INJECTION APPARATUS
6. MOLECULAR SIEVE
7. MILLIPORE FILTER
8. LN₂-COOLED FISSION GAS TRAP, LN₂ LEVEL ELECTRICALLY CONTROLLED
9. LOW LN₂ LEVEL SOLENOID VALVE, MELETRON CONTROLLED
10. MOISTURE MONITOR

Fig. 4-8. Auxiliary flow system connected to TRIGA King furnace for use in hydrolysis tests

Results

Figure 4-9 shows the effect of hydrolysis on the R/B value for fuel rods containing unirradiated kernels as a function of time and hydrolysis temperature. Similarly, Fig. 4-10 shows the effect on rods prepared from laser-drilled^{*} TRISO particles. As can be seen, continued annealing under 100-ppm H₂O resulted in a decrease in the R/B value for the high-temperature runs (>700°C). This effect has been observed before and is believed to be a sintering effect due to agglomeration of the finely divided hydrated UO₂. This hypothesis is supported by the observation that the exposure time necessary for the R/B value to begin to decrease was inversely dependent on temperature: 900°C ≈ 50 hr; 800°C ≈ 100 hr; and 700°C ≈ 200 hr.

The preliminary results obtained for low-temperature hydrolysis show that even at 300°C there is an increase in R/B. As was observed for the high-temperature runs, this increase is greater for bare kernel rods than for rods prepared with laser-failed particles. The increase is greater for the bare kernel rods because the exposed surface in these rods is greater than in the rods prepared from laser-failed particles.

Figure 4-11 shows the effect of hydrolysis on the sample of irradiated Peach Bottom compact containing (U,Th)C₂. The data have been normalized to an initial R/B value of 0.01, which is a representative R/B value for Peach Bottom Core 1 material.

^{*} Particles are laser-drilled to simulate failed coated particles.

Fig. 4-9. Effect of hydrolysis on steady-state Kr-85m release (R/B) at 1100°C for fuel rods containing bare UC₂ particles ($P_{H_2O} = 10^{-4}$ atm): (a) 700°, 800°, and 900°C anneal temperature

Fig. 4-9. Effect of hydrolysis on steady-state Kr-85m release (R/B) at 1100°C for fuel rods containing bare UC₂ particles ($P_{H_2O} = 10^{-4}$ atm): (b) 300°, 400°, and 500°C anneal temperature

Fig. 4-10. Effect of hydrolysis on steady-state Kr-85m release (R/B) at 1100°C for fuel rods containing laser-drilled TRISO UC₂ particles ($P_{H_2O} = 10^{-4}$ atm): (a) 700°, 800°, and 900°C anneal temperature

Fig. 4-10. Effect of hydrolysis on steady-state Kr-85m release (R/B) at 1100°C for fuel rods containing laser-failed TRISO UC₂ particles ($P_{H_2O} = 10^{-4}$ atm): (b) 300°, 400°, and 500°C anneal temperature

Fig. 4-11. Effect of hydrolysis at 900°C on the steady-state release (R/B) at 1100°C from irradiated (U,Th)C₂ in Peach Bottom compact material ($P_{H_2O} = 10^{-4}$ atm)

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- 4-4. "HTGR Base Program Quarterly Progress Report for the Period Ending February 28, 1974," USAEC Report GA-A12916, General Atomic Company, March 29, 1974.

TASK 8

PHYSICS AND FUEL MANAGEMENT

ENDF/B PARTICIPATION

Detailed comments on minor problems with the ENDF/B-III data sets for Pa-233 and U-233 were forwarded to the Cross Section Evaluation Working Group (CSEWG). Revised ENDF/B format thermal and resonance cross-section data for Pa-233 were also supplied to the CSEWG.

A number of ENDF/B-IV data sets were received. The data sets for the uranium and plutonium isotopes were processed for use in data testing activities.

A General Atomic Company representative was sent to the June 1974 CSEWG meeting.

XENON STABILITY ANALYSIS IN LARGE HTGRs

The scoping of this study has been completed and the study itself was initiated. Current effort is centered on an assessment of the relative ability of two linear models to accurately predict the stability characteristics of HTGRs.

HTLTR ANALYSIS

The use of the ENDF/B-III data for U-233 makes essentially no change in previously reported GA calculated values for the HTLTR lattices containing U-233. This means that the use of the presumably more accurate Adler-Adler representation of the Doppler broadening of the resonance cross sections for U-235 does not seem to be needed in HTGR applications.

TASK 9
FUEL MATERIALS DEVELOPMENT

FUEL THERMAL STABILITY

A series of thermal gradient tests (Experiment 6440) was conducted on large HTGR reference TRISO coated UC_2 particles to characterize UC_2 kernel migration as a function of temperature. The reference fuel was coated for irradiation in capsule tests P13Q, P13R, and P13S. The results of this test series will be compared with data obtained from postirradiation thermal gradient tests, which will be conducted after capsules P13Q, P13R, and P13S have completed irradiation. Also included in this thermal gradient test series were TRISO coated UC_2 particles having 110- μm kernels. The average coating properties for the particles tested are given in Table 9-1. This experiment is part of a continuing test sequence designed to broaden the data base used to estimate the kernel migration of fissile fuel during HTGR core operation.

Kernel migration of carbide kernels in a thermal gradient takes place by the process of carbon solution on the hot side of the kernel and rejection of carbon on the cold side of the kernel (Ref. 9-1). The rate of advance of the hot side of a carbide kernel in a thermal gradient is used to calculate a kernel migration coefficient (KMC), which is defined by the expression

$$KMC = \frac{dx}{dt} T^2 \left(\frac{\Delta T}{\Delta X} \right)^{-1} = K D_o e^{(-\Delta H/RT)} \quad , \quad (9-1)$$

TABLE 9-1
TRISO VSM PARTICLE PARAMETERS, EXPERIMENT 6440

			Coating											Total Coating Thick- ness (μm)
Sample No.	Kernel Type Diam.			Buffer		IPyC			SiC		OPyC			
			Type	Thick- ness (μm)	Density (g/cm^3)	Thick- ness (μm)	Density (g/cm^3)	OPTAF	Thick- ness (μm)	Density (g/cm^3)	Thick- ness (μm)	Density (g/cm^3)	OPTAF	
4413-21	UC ₂	110	TRISO	44	1.22	21	1.78	--	21	3.17	19	1.91	1.00	105 ^(a)
6151-04-010	UC ₂	202	TRISO	105	0.87	33	1.88	1.23	28	3.20	36	1.82	1.14	199
6151-00-016	UC ₂	200	TRISO	97	1.18	35	1.94	1.19	31	3.20	44	1.80	1.11	200
6151-02-020	UC ₂	201	TRISO	99	1.07	33	1.92	1.25	28	3.22	43	1.50	1.13	202

(a) Bacon anistropy factor.

where $\frac{dx}{dt}$ = rate of advance of the hot kernel surface

T = temperature, °K

$\frac{\Delta T}{\Delta X}$ = temperature gradient across the kernel

K = constant which depends on material properties

D_0 = pre-exponential constant

ΔH = activation energy for carbon diffusion in the fuel kernel

R = gas constant (cal/deg mole)

Kernel migration coefficients were calculated from the data obtained during this test series, and the variation in KMC with temperature was defined using Eq. 9-1.

The experimental arrangements and procedures used for thermal gradient testing have been described previously (Ref. 9-2). Kernel migration was induced by heating particles in the temperature range 1300° to 1750°C. The temperatures, temperature gradients, total heating times, and total number of particles heated at each temperature was described in Table 9-2, along with the range of kernel migration distances observed.

The kernel migration distance was determined as a function of time for each particle heated in these experiments by measuring the advance of the hot side of the migrating kernels. The range in migration behavior observed in this test series is illustrated in Fig. 9-1, which shows the migration distance as a function of annealing time at 1600°C for two particles from batch 6151-00-016. These data are typical of the results obtained throughout the test. In some cases, the migration distance increases steadily with time (e.g., particle 3, Fig. 9-1) and, in these cases, the migration rate in Eq. 9-1 was defined as the maximum migration distance divided by the total test time. In other cases (particle 1, Fig. 9-1) the rate of kernel migration decreased markedly at some point in time. This was generally observed in a number of particles heated at any

TABLE 9-2
EXPERIMENTAL CONDITIONS AND RANGE OF KERNEL MIGRATION
DISTANCES, EXPERIMENT 6440

Kernel Diameter (μm)	Temp. ($^{\circ}\text{C}$)	Temp. Gradient ($^{\circ}\text{C}/\text{cm}$)	Time (hr)	No. of Particles	Kernel Migration (μm)
110	1664	7510	32	8	19 to 36
	1484	6880	250	8	13 to 28
	1345	6110	501	8	3 to 11
200	1750	5500	18	12	13 to 86
	1745	5130	18	12	18 to 88
	1652	4750	32	12	6 to 59
	1644	4480	32	7	19 to 96
	1500	3816	531	13	25 to 59
	1484	4200	250	7	15 to 35
	1345	3607	501	7	7 to 12
	1338	3851	501	12	2 to 9

Fig. 9-1. Range of kernel migration distance versus time at 1600°C for particles 1 and 3

given temperature. The decreased migration rate is generally related more to the total heating time than to migration distance and is attributed to a loss of thermal gradient across the coated particles. The loss of thermal gradient is assumed to be caused by shrinkage of the binder, which holds the particles in graphite thermal stability crucibles, away from the crucible walls. When behavior such as illustrated for particle 1 in Fig. 9-1 is observed, the kernel migration rate is assumed to be the ratio of migration distance to time calculated at the last measurement showing a steady increase in migration distance with time.

The variation in log KMC with $1/T$ is shown for particles having 100- and 200- μm kernels in Figs. 9-2 and 9-3, respectively. Comparison of the two figures shows the measured values for KMC to be independent of kernel size. The values for KMC measured during this test are compared with KMC values used in large HTGR core design in Fig. 9-4. The data points in this figure were measured in experiment 6440 from particles having 200- μm kernels; the linear least-squares curve fit and the upper 95% confidence limit are for the data used to establish the UC_2 KMC curve used in HTGR core design. The comparison shows that the design curve is conservative with respect to the data obtained in this experiment at normal HTGR operating temperatures ($T \leq 1350^\circ\text{C}$).

FUEL IRRADIATIONS

Capsules P13N and P13P

Capsules P13N and P13P were the fourth and fifth capsules in a series of irradiation tests of candidate HTGR recyclable-type fuels. Both capsules were purged with an inert gas mixture during operation to provide a means of temperature control and of monitoring in-pile fission gas release. Each capsule contained five cells in which carbide and oxide fuels in the form of fuel rods and unbonded particles were tested. Postirradiation examination results of fuel rod samples irradiated in capsules P13N and P13P have been published previously (Refs. 9-3, 9-4). Thermal analysis of these

Fig. 9-2. Kernel migration coefficient versus $1/T$, 100- μ m kernels

Fig. 9-3. Kernel migration coefficient versus $1/T$, 200- μ m kernels

Fig. 9-4. Comparison of KMC data from experiment 6440 with least-squares curve fit and upper 95% confidence limit for KMC used in large HTGR design studies

capsules using the CAPTEM computer code is currently in progress. A more detailed analyses of the postirradiation examination results of capsules P13N and P13P will be reported in topical reports upon completion of the capsule thermal analyses.

Capsules P13R and P13S

Capsules P13R and P13S are being irradiated in the GETR to provide data on a broad spectrum of property and process variables for reference fuel in the large HTGR. The cells in these two capsules are operating at fuel centerline temperatures between 1075° and 1600°C. The capsules have received 76% of the design fast fluence ($E > 0.18$ MeV) of 9.6×10^{21} n/cm². Provided there are no unscheduled shutdowns, dosimetry evaluation indicates the design fluence in this experiment will be attained by November 1, 1974.

Set control temperatures in these capsules have been reduced by nominally 30° to 150°C in accordance with a statistical analysis of decalibration data on tungsten/rhenium thermocouples. The revised thermocouple decalibration indicates that the actual cell temperature, i.e., based on a mean time-integrated temperature, exceeded the design temperatures by approximately 20° to 70°C.

Cell 1 of P13S has undergone its sixteenth thermal transient cycle from 1075° to 1600°C, with an indicated R/B (Kr-85m) of 2.5×10^{-4} at the end of the cycle. This is approximately an order in magnitude greater than the R/B measurements in the companion fuel test (cell 1, P13R), which has not undergone thermal cycling. The R/B (Kr-85m) measurements for all cells at the equilibrated soak temperature was less than 4×10^{-5} with the exception of cell 2 in P13R and P13S. The R/B for these cells is currently at a level of $\sim 10^{-4}$, and the rate of increase in R/B (defined as the change in the natural log of R/B versus the change in irradiation time) has been effectively constant throughout irradiation.

Capsule P13T

Capsule P13T was scheduled for insertion in the ORR on November 3, 1974; however, a delay associated with fuel fabrication has caused rescheduling of the insertion date to December 29, 1974. Manufacturing UC_2 kernels by the VSM process was initially unsuccessful due to improper kernel shape and carbon content. This work was originally started in February, and acceptable kernels were produced by the end of August. The delay was compounded by a defective glove box, which caused the loss of 600 g U-235 WAR kernels by oxidation. Replacement WAR kernels have been produced, and both the VSM and WAR kernels are scheduled for TRISO coatings during September. Fuel rod fabrication combined with curing the rods in-place is scheduled for September and October.

GA-ORNL Cooperative Irradiation Capsules

Capsule P13Q

Capsule P13Q is a large-diameter capsule designed to evaluate representatively processed fuel bodies in the ORR. The capsule is a combined GA/ORNL experiment in which GA supplied a fueled cell to be incorporated in an ORNL-fabricated capsule.

An unscheduled 7-week delay in capsule irradiation (between 5/24/74 and 7/15/74) resulted from two separate leaks that developed in the gas lead tubes. The cause of the gas leaks was attributed to corrosion enhanced by a high-moisture atmosphere around the inlet tube. The condition was corrected by installing a water monitoring system and purging the outside of the lead tubes with dry helium.

The capsule resumed its irradiation on July 15 and has now reached an estimated peak fluence of 7.0×10^{21} n/cm² ($E > 0.18$ MeV)_{ORR}. Fission gas release measurements (Kr-85m) indicate a marked increase in R/B values after the last cycle (from 7×10^{-6} to 5.85×10^{-5}). The increase was

was attributed to gas leakage between the two sections of the capsule as a result of bulkhead failure. This condition was partially corrected by imposing a slight pressure gradient between the GA and ORNL cells, and the R/B value was reduced to 1.16×10^{-5} . Reactor problems at the start of the last cycle (8/15/74) necessitated an unscheduled shutdown, which will probably delay completion of the full irradiation exposure until early December.

Capsules HT-12, -13, -14, and -15

The HT-12, -13, -14, and -15 series was a set of four irradiation experiments designed to evaluate the irradiation behavior of unbonded BISO coated ThO_2 fertile fuel particles. The experiments were conducted in the target position (HT) of the High Flux Isotope Reactor (HFIR) at ORNL. Experiments of this type are conducted in the HT position of the HFIR because the high fluxes in the reactor allow the accumulation of high fast neutron doses and high thorium burnups in a short time. The irradiation schedule for these experiments is summarized in Table 9-3.

TABLE 9-3
IRRADIATION SCHEDULE FOR HT-12, -13, -14, AND -15

Experiment	Insertion Date	Removal Date	Accumulated Irradiation Time at 100-MW Reactor Power (days)
HT-12	Sept. 10, 1972	Oct. 23, 1972	38.6
HT-13	Oct. 7, 1972	Jan. 10, 1973	85.5
HT-14	Dec. 13, 1972	Apr. 15, 1973	137.0
HT-15	Dec. 13, 1972	May 11, 1973	160.0

Coating variables evaluated were buffer and outer PyC (OPyC) thickness, as well as OPyC density and OPTAF. The objectives of this series of HT irradiation experiments are summarized below:

1. Evaluate the general irradiation performance of unbonded ThO_2 BISO particles as a function of coating design.
2. Evaluate BISO particle dimensional and density changes as a function of fast neutron exposure and heavy metal burnup.
3. Provide particle samples for fission product retention studies.
4. Provide particle samples for postirradiation annealing experiments designed to evaluate high-temperature survival rates and fuel kernel migration.

The particle batches were chosen so that only one variable (i.e., coating density, OPTAF, or thickness) was tested at a time. To aid in postirradiation analysis, the particles irradiated in HT-12, -13, -14, and -15 were selected so that the particles within each individual batch had a constant total coated particle density and a narrow distribution of kernel and total coated particle diameters. The steps used to separate desired particles from their respective parent batches are outlined in Fig. 9-5. Briefly, coated particles were sieved to plus or minus one standard deviation in particle diameter. They were then further separated by selecting, via constant density columns, only those particles having the batch-average total coated particle density. Following density separation, particles were individually radiographed and picked so that both kernel and total particle diameters were within 20 μm of their respective average values. As a final step, particles were heated in vacuum for 1 hr at 1350°C to ensure volatilization of any of the fluids used for density separation.

A summary description of the particle samples irradiated in this series of HT-experiments is given in Tables 9-4 through 9-7. The high-temperature samples contained 75 particles and the low-temperature samples contained 53 particles. Data on the experimental irradiation conditions

Fig. 9-5. Steps involved in selecting BISO coated ThO₂ particles for insertion in irradiation experiments HT-12 through HT-15

TABLE 9-4
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-12

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure (g) (%)	Postirradiation Examinations Performed		
Batch No.	Kernel ^(a)		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density ^(f)	Metallography	X-ray Radiography
	Diam ^(b) (μm)	Density ^(c,d) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,e) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,f) (g/cm ³)	OPTAF ^(c)	Coating Rate (μm/min)	Diam ^(b) (μm)	Density ^(b,f) (g/cm ³)								
4252-00-013-3 ^(h)	404	9.93	77	1.26	79	2.00	1.14	2.3	703	3.30	20	1250	3.8	1.8	3	X		X
4252-00-013-7 ^(h)	397	9.93	79	1.26	79	2.00	1.14	2.3	703	3.30	7	900	2.5	1.2	0	X		X
4252-01-071-3	494	9.86	62	1.13	75	2.02	1.16	1.6	768	4.07	33	1250	3.8	1.8	33			
4252-01-071-7	491	9.86	62	1.13	73	2.02	1.16	1.6	764	4.07	46	900	2.5	1.2	0			
4252-02-015-1 ⁽ⁱ⁾	510	9.95	86	1.08	70	1.84	1.07	3.9	820	3.58	27	1250	3.9	1.9	0			
4252-02-015-5	505	9.95	85	1.08	76	1.84	1.07	4.2	827	3.58	40	900	3.1	1.5	0	X	X	X
4252-03-012-1	503	9.90	86	1.00	70	2.00	>1.5 ^(j)	0.7	812	3.60	30	1250	3.9	1.9	100			
4252-03-012-5	505	9.90	92	1.00	68	2.00	>1.5 ^(j)	0.7	824	3.60	43	900	2.8	1.4	0			
4252-06-012-1	513	9.94	78	1.10	74	1.82	1.14	5.6	818	3.81	23	1250	3.9	1.9	0	X		X
4252-06-012-5	515	9.94	76	1.10	78	1.82	1.14	5.9	820	3.81	10	900	2.8	1.4	2	X		X
4252-07-016-1	491	9.90	45	1.03	48	1.96 ^(k)	1.20	1.3	674	4.84	36	1250	3.6	1.8	27			
4252-07-016-5	492	9.90	45	1.03	51	1.96 ^(k)	1.20	1.4	680	4.84	49	900	2.2	1.1	0			X
4252-08-014-1	488	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	830	3.84	15	1250	3.5	1.7	59			
4252-08-014-5	494	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	823	3.84	2	900	2.0	1.0	0	X		X

(a) 500-μm-diameter ThO₂ kernels fabricated at ORNL.
(b) Measured on particle sample that was irradiated.
(c) Measured on parent particle batch.
(d) Kernel densities measured by Hg porosimetry.
(e) Calculated.
(f) Measured using density gradient column.
(g) Determined during visual examination.
(h) Subjected to constant density separation but not constant size selection.
(i) Sample spilled in hot cell during PIE density measurements.
(j) Maximum measurable OPTAF is 1.5.
(k) Buffer-OPyC layers could not be separated for OPyC density measurement. A value of 1.96 g/cm³ was measured from an as-coated grab sample.

TABLE 9-5
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-13

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure ^(g) (%)	Postirradiation Examinations Performed		
Batch No.	Kernel ^(a)		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density ^(f)	Metallo-graphy	X-ray Radio-graphy
	Diam ^(b) (μm)	Density ^(c,d) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,e) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,f) (g/cm ³)	OPTAF ^(c)	Coating Rate (μm/min)	Diam ^(b) (μm)	Density ^(b,f) (g/cm ³)								
4252-00-013-4 ^(h)	406	9.93	81	1.26	79	2.00	1.14	2.3	718	3.30	20	1250	8.3	8.2	100			
4252-00-013-8 ^(h)	404	9.93	78	1.26	80	2.00	1.14	2.4	721	3.30	7	900	5.6	5.6	8	X		X
4252-01-071-4	494	9.86	65	1.13	73	2.02	1.16	1.6	769	4.07	33	1250	8.3	8.2	100			
4252-01-071-8	496	9.86	60	1.13	77	2.02	1.16	1.6	768	4.07	46	900	5.6	5.6	34			
4252-02-015-2	502	9.95	89	1.08	67	1.84	1.07	3.7	807	3.58	27	1250	8.6	8.4	88			
4252-02-015-6	508	9.95	84	1.08	75	1.84	1.07	4.1	828	3.58	40	900	6.9	6.8	0	X	X	X
4252-03-012-2	498	9.90	95	1.00	64	2.00	>1.5 ⁽ⁱ⁾	0.6	821	3.60	30	1250	8.5	8.3	100			
4252-03-012-6	505	9.90	92	1.00	66	2.00	>1.5 ⁽ⁱ⁾	0.7	826	3.60	43	900	6.2	6.2	100			
4252-06-012-2	517	9.94	79	1.10	77	1.82	1.14	5.8	824	3.81	23	1250	8.5	8.3	80			
4252-06-012-6	517	9.94	78	1.10	74	1.82	1.14	5.8	821	3.81	10	900	6.2	6.2	0	X		X
4252-07-016-2	483	9.90	42	1.03	48	1.96 ^(j)	1.20	1.3	671	4.84	36	1250	8.1	7.9	100			
4252-07-016-6	492	9.90	45	1.03	50	1.96 ^(j)	1.20	1.4	683	4.84	49	900	4.9	5.0	42			X
4252-08-014-2	465	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	833	3.84	15	1250	7.8	7.7	100			
4252-08-014-6	484	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	831	3.84	2	900	4.3	4.5	94			

(a) 500-μm-diameter ThO₂ kernels fabricated at ORNL.

(b) Measured on particle sample that was irradiated.

(c) Measured on parent particle batch.

(d) Kernel densities measured by Hg porosimetry.

(e) Calculated.

(f) Measured using density gradient column.

(g) Determined during visual examination.

(h) Subjected to constant density separation but not constant size selection.

(i) Maximum measurable OPTAF is 1.5.

(j) Buffer-OPyC layers could not be separated for OPyC density measurement. A value of 1.96 g/cm³ was measured from an as-coated grab sample.

TABLE 9-6
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-14

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC(g) Failure (%)	Postirradiation Examinations Performed		
Batch No.	Kernel (a)		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density(f)	Metallo- graphy	X-ray Radio- graphy
	Diam (b) (μm)	Density (c,d) (g/cm ³)	Thickness (b) (μm)	Density (c,e) (g/cm ³)	Thickness (b) (μm)	Density (c,f) (g/cm ³)	OPTAF (c)	Coating Rate (μm/min)	Diam (b) (μm)	Density (b,f) (g/cm ³)								
4252-00-013-5 ^(h)	403	9.93	77	1.26	82	2.00	1.14	2.4	708	3.30	20	1250	13.4	16.1	100 ⁽ⁱ⁾			
4252-00-013-9 ^(h)	404	9.93	78	1.26	81	2.00	1.14	2.4	724	3.30	7	900	9.0	11.7	98			
4252-01-071-5	494	9.86	62	1.13	75	2.02	1.16	1.6	773	4.07	33	1250	13.4	16.1	100 ⁽ⁱ⁾			
4252-01-071-9	495	9.86	64	1.13	73	2.02	1.16	1.6	770	4.07	46	900	9.0	11.7	100			
4252-02-015-3	508	9.95	85	1.08	74	1.84	1.07	4.1	826	3.58	27	1250	13.9	16.6	90			
4252-02-015-7	509	9.95	88	1.08	68	1.84	1.07	3.8	819	3.58	40	900	11.0	13.7	0	X	X	X
4252-03-012-3	499	9.90	90	1.00	72	2.00	>1.5 ^(j)	0.7	825	3.60	30	1250	13.7	16.4	100 ⁽ⁱ⁾			
4252-03-012-7	501	9.90	95	1.00	63	2.00	>1.5 ^(j)	0.6	818	3.60	43	900	10.0	12.8	100			
4252-06-012-3	515	9.94	76	1.10	79	1.82	1.14	6.0	821	3.81	23	1250	13.7	16.4	85			
4252-06-012-7	513	9.94	80	1.10	75	1.82	1.14	5.7	822	3.81	10	900	10.0	12.8	0	X	X	X
4252-07-016-3	490	9.90	46	1.03	47	1.96 ^(k)	1.20	1.3	670	4.84	36	1250	12.9	15.7	100 ⁽ⁱ⁾			
4252-07-016-7	487	9.90	44	1.03	48	1.96 ^(k)	1.20	1.3	668	4.84	49	900	7.9	10.4	98			
4252-08-014-3	461	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	826	3.84	15	1250	12.5	15.3	100 ⁽ⁱ⁾			
4252-08-014-7	492	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	827	3.84	2	900	6.9	9.6	100			

(a) 500-μm-diameter ThO₂ kernel fabricated by ORNL.

(b) Measured on particle sample that was irradiated.

(c) Measured on parent particle batch.

(d) Kernel densities measured by Hg porosimetry.

(e) Calculated.

(f) Measured using density gradient column.

(g) Determined during visual examination.

(h) Subjected to constant density separation but not constant size selection.

(i) Sample not examined. 100% failure assumed based on HT-13 irradiation results.

(j) Maximum measurable OPTAF is 1.5.

(k) Buffer-OPyC layers could not be separated for OPyC density measurement. A value of 1.96 g/cm³ was measured from an as-coated grab sample.

TABLE 9-7
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-15

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure ^(g) (%)	Postirradiation Examinations Performed		
Batch No.	Kernel ^(a)		Buffer		Outer Pyrocarbon (OPyC)			Total Particle		Design Temp (°C)		Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)	Particle Density ^(f)		Metallo- graphy	X-ray Radio- graphy	
	Diam ^(b) (μm)	Density ^(c,d) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,e) (g/cm ³)	Thickness ^(b) (μm)	Density ^(c,f) (g/cm ³)	OPTAF ^(c)	Coating Rate (μm/min)	Diam ^(b) (μm)									Density ^(b,f) (g/cm ³)
4252-00-013-6 ^(h)	404	9.93	78	1.26	77	2.00	1.14	2.3	723	3.30	20	1250	15.6	19.6	100 ⁽ⁱ⁾			
4252-00-013-10 ^(h)	393	9.93	78	1.26	79	2.00	1.14	2.3	716	3.30	7	900	10.5	14.4	87			
4252-01-071-6	495	9.86	61	1.13	74	2.02	1.16	1.6	766	4.07	33	1250	15.6	19.6	100 ⁽ⁱ⁾			
4252-01-071-10	494	9.86	63	1.13	76	2.02	1.16	1.7	765	4.07	46	900	10.5	14.4	98 ⁽ⁱ⁾			
4252-02-015-4	510	9.95	86	1.08	69	1.84	1.07	3.8	818	3.58	27	1250	16.2	20.3	91			
4252-02-015-8	509	9.95	86	1.08	75	1.84	1.07	4.1	827	3.58	40	900	12.8	16.8	45	X	X	X
4252-03-012-4	505	9.90	90	1.00	71	2.00	>1.5 ^(j)	0.7	831	3.60	30	1250	16.0	20.0	100 ⁽ⁱ⁾			
4252-03-012-8	503	9.90	96	1.00	64	2.00	>1.5 ^(j)	0.6	821	3.60	43	900	11.7	15.7	100 ⁽ⁱ⁾			
4252-06-012-4	515	9.94	76	1.10	78	1.82	1.14	5.9	820	3.81	23	1250	16.0	20.0	82			
4252-06-012-8	509	9.94	77	1.10	75	1.82	1.14	5.6	820	3.81	10	900	11.7	15.7	0	X		X
4252-07-016-4	492	9.90	45	1.03	51	1.96	1.20	1.4	680	4.84	36	1250	15.1	19.1	100			
4252-07-016-8	494	9.90	46	1.03	48	1.96	1.20	1.3	677	4.84	49	900	9.3	13.0	100 ⁽ⁱ⁾			
4252-08-014-4	504	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	813	3.84	15	1250	14.6	18.7	100 ⁽ⁱ⁾			
4252-08-014-8	503	9.95	39 ^(c)	1.08	119 ^(c)	1.95	1.14	1.3 ^(c)	821	3.84	2	900	8.1	11.9	100 ⁽ⁱ⁾			

(a) 500-μm-diameter ThO₂ kernel fabricated by ORNL.
(b) Measured on particle sample that was irradiated.
(c) Measured on parent particle batch.
(d) Kernel densities measured by Hg porosimetry.
(e) Calculated.
(f) Measured using density gradient column.
(g) Determined during visual examination.
(h) Subjected to constant density separation but not constant size selection.
(i) Sample not examined. 100% failure assumed based on HT-13 and HT-14 irradiation results.
(j) Maximum measurable OPTAF is 1.5.
(k) Buffer-OPyC layers could not be separated for OPyC density measurement. A value of 1.96 g/cm³ was measured from as-coated grab samples.

were supplied by ORNL. The reported temperatures are the design graphite magazine temperature. A thermal analysis is currently in progress at ORNL. The actual fuel particle irradiation temperatures are expected to be 100° to 200°C higher than the graphite design temperature. The OPyC coating failure rates were determined during the visual examination conducted at GA. Postirradiation particle density measurements, metallography, and contact X-ray radiography examinations were performed on selected particle samples as noted in Tables 9-4 through 9-7.

Irradiation performance of the BISO coated ThO₂ particles irradiated in the HT-12, -13, -14, and -15 capsule series revealed a correlation between OPyC coating rate, OPyC coating OPTAF, and particle survival. BISO particle samples from particle batches 4252-02-015 and 4252-06-012 exhibited the lowest rate of OPyC coating failure during irradiation. These particle batches had medium density (1.80 to 1.85 g/cm³) and low-to-medium OPTAF (≤1.15) OPyC coatings that were applied at relatively high coating rates of 3.7 to 5.8 μm/min. No particle failure was observed in batch 4252-02-015 irradiated in the low-temperature positions to fast neutron exposures up to 11.0×10^{21} n/cm² (E > 0.18 MeV) and heavy metal burnups up to 13.7% FIMA. With the exception of one particle in HT-12, no particle failure was observed in batch 4252-06-012 irradiated in the low-temperature positions to fast neutron exposures up to 11.7×10^{21} n/cm² (E > 0.18 MeV) and heavy metal burnups up to 15.7% FIMA. These irradiation conditions are significantly beyond the peak irradiation conditions of 8×10^{21} n/cm² and 7.0% FIMA that fertile fuel will experience in the large HTGR.

In contrast, particle failure rates in batch 4252-01-071 reached 34% and 100% during irradiation to fast neutron exposures of 5.6 and 9.0×10^{21} n/cm² (E > 0.18 MeV), respectively. This particle batch had high density (2.02 g/cm³) and medium OPTAF (1.16) OPyC coatings that were applied at a relatively low coating rate of 1.6 μm/min. Although batch 4252-01-071 had a high OPyC coating density, the failure was attributed to irradiation instability of the OPyC coating as a result of high PyC anisotropy rather than high density. The reason for this conclusion is that BISO particles

irradiated in HT-17, -18, and -19 survived irradiation to high fast neutron exposures with high-density OPyC coatings that were applied at high coating rates. Low coating rates can be directly correlated with high PyC anisotropy or preferred orientation (Ref. 9-5).

Although the measured OPTAF values for all three particle batches were similar, the data are not considered to be very quantitative. Since the preirradiation characterization of these particle batches was conducted, improvements have been made in evaluation of PyC coating optical anisotropy. The relationship between OPyC coating rate, OPTAF, and particle failure is discussed in further detail in the discussion of the HT-17, -18, and -19 capsule irradiation results.

Relatively high failure rates were observed in the remaining four particle batches irradiated in the HT-12, -13, -14, and -15 capsule series. These BISO particle batches had large optical anisotropy gradients and possibly density gradients across the OPyC coatings. The poor irradiation performance of these coatings masked any remaining correlations that may have been drawn between particle irradiation performance and coating variables. Preirradiation photomicrographs of the parent particle batches irradiated in the HT-12, -13, -14, and -15 capsule series are shown in Figs. 9-6 through 9-12. The OPyC coatings in particle batches 4252-02-015, 4252-06-012, and 4252-01-071 are relatively free of anisotropy or density gradients. However, under polarized light extreme optical anisotropy gradients are visible in the OPyC coatings of the remaining four particle batches.

Particle samples from the parent batch of 4252-00-013 were also irradiated in capsule P13N. The optical anisotropy across the OPyC coating was measured for eight particles from this batch using the ÖSGAE technique (Refs. 9-6, 9-7). The data are plotted in Fig. 9-13 in terms of the optical Bacon anisotropy factor (BAF) versus radial position across the OPyC coating. As shown in Fig. 9-13, the inner 20- μ m region of the OPyC coating is highly anisotropic and becomes relatively isotropic near the outer surface. Optical anisotropy values (OPTAF) reported in Tables 9-4

M38811-5 (a)

M38811-6 (b)

M38811-3 (c)

M38811-4 (d)

Fig. 9-6. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-00-013: (a) and (c) bright field; (b) and (d) polarized light. High-density regions in the OPyC coating appear bright under polarized light

M38810-4 (a)

M38810-5 (b)

M38810-1 (c)

M38810-2 (d)

Fig. 9-7. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-01-071: (a) and (c) bright field; (b) and (d) polarized light

M38812-5 (a)

M38812-6 (b)

M38812-1 (c)

M38812-2 (d)

Fig. 9-8. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-01-015: (a) and (c) bright field; (b) and (d) polarized light

M38813 5 (a)

M38813-6 (b)

M38813-1 (c)

M38813-2 (d)

Fig. 9-9. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-03-012: (a) and (c) bright field; (b) and (d) polarized light. High-density regions in the OPyC coating appear bright under polarized light.

M38814-5 (a)

M38814-6 (b)

M38814-1 (c)

M38814-2 (d)

Fig. 9-10. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-06-012: (a) and (c) bright field; (b) and (d) polarized light

M38815-7 (a)

M38815-8 (b)

M38815-1 (c)

M38815-2 (d)

Fig. 9-11. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-07-016: (a) and (c) bright field; (b) and (d) polarized light. High-density regions in the OPyC coating appear bright under polarized light

M38816-7 (a)

M38816-8 (b)

M38816-3 (c)

M38816-4 (d)

Fig. 9-12. Preirradiation photomicrographs of BISO coated ThO_2 particles from batch 4252-08-014. (a) and (c) bright field; (b) and (d) polarized light. High-density regions in the OPyC coating appear bright under polarized light

Fig. 9-13. Optical Bacon anisotropy factor $(BAF)_0$ versus radial position in OPyC coating in BISO particle batch 4252-00-010

through 9-7 were measured at the mid-radius of the OPyC and do not reflect the existence of an anisotropy gradient across the coating. The OPTAF values will be remeasured on these particle batches, and the data will be reported when available. The capsule P13N irradiation results revealed that particle coatings that contain optical anisotropy gradients are unstable during irradiation to fast neutron exposures beyond $\sim 5.0 \times 10^{21}$ n/cm² (Ref. 9-3). At higher irradiation exposures, cracks generated in the highly anisotropic region of the OPyC coating propagate through the coating causing it to fail. This type of coating defect is due to a large temperature gradient in the coater during initial application of the OPyC coating. Failure of BISO coatings resulting from these properties can be eliminated in optimized coating designs and by proper control of the coating operation.

During irradiation BISO coated particles undergo irradiation-induced shrinkage, which can be measured experimentally by radiography (measuring particle diameter) or by measuring the total particle density before and after irradiation. Characterization of BISO particle dimensional change as a function of fast neutron exposure and heavy metal burnup is required for fuel rod dimensional change models and BISO particle irradiation performance models. One of the objectives of the HT-12, -13, -14, and -15 capsule irradiation series was to evaluate BISO particle dimensional and density changes as a function of fast neutron exposure and heavy metal burnup.

Postirradiation particle densities were measured using a density gradient column on thirteen particle samples irradiated in this HT-series. In addition, fifteen particle samples were X-ray radiographed, and the postirradiation particle dimensions were measured. Comparison of postirradiation total coated particle densities with postirradiation particle densities predicted using the BISO particle stress code is in progress. These comparisons will be reported in a future Quarterly Progress Report when the analyses have been completed.

Fission product retention studies are currently in progress on selected particle samples. Fission product inventories in the particle samples are being determined by gamma-ray spectroscopy. These data are compared with theoretical fission product inventories, which are calculated by taking into account heavy metal inventories, fuel burnup, and irradiation history to evaluate the retention of metallic fission products during irradiation. Particle samples have also been selected for postirradiation annealing studies to evaluate high-temperature survival rates and fuel kernel migration.

Capsules HT-17, -18, and -19

The HT-17, -18, and -19 irradiation series is part of a continued cooperative effort between GA and ORNL. This series was designed to evaluate the irradiation behavior of unbonded BISO and TRISO coated ThO₂ fertile fuel particles. The experiments were conducted in the target position of the HFIR at ORNL; the experimental design and operation was identical to the HT-12, -13, -14, and -15 capsule irradiation series. The irradiation schedule for these experiments is summarized in Table 9-8.

TABLE 9-8
IRRADIATION SCHEDULE FOR HT-17, -18, AND -19

Experiment	Insertion Date	Removal Date	Accumulated Irradiation Time at 100-MW Reactor Power (days)
HT-17	July 22, 1973	Sept. 8, 1973	46.4
HT-18	July 22, 1973	Oct. 28, 1973	92.9
HT-19	July 22, 1973	Dec. 17, 1973	139.3

Each experiment had high-temperature regions and low-temperature regions. The experiments were designed so that the particles (41 particles per sample) in the low-temperature regions operated at $\sim 1100^{\circ}\text{C}$, while the particles (58 particles per sample) in the high-temperature regions operated at $\sim 1500^{\circ}\text{C}$.

The objectives of the HT-17, -18, and -19 experiments are as follows:

1. Compare postirradiation integrity of BISO coated ThO_2 .
The variables are: (a) coating design, (b) OPyC density, (c) OPyC coating rate, (d) fast neutron exposure, and (e) irradiation temperature.
2. Compare neutron-induced BISO particle dimensional change as a function of the variables listed in (1).
3. Examine the effect of neutron exposure on kernel dimensions.
4. Irradiate TRISO coated ThO_2 for postirradiation fission gas pressure/composition studies.

The particle batches were chosen so that only one variable (i.e., coating density, OPTAF, or thickness) was tested at a time. Particle selection was based on constant size and density separations in the same manner as the HT-12, -13, -14, and -15 samples were selected (outlined in Fig. 9-5).

A summary description of the particle samples irradiated in this series of HT-experiments is given in Tables 9-9 through 9-12. Data on the experimental irradiation conditions were supplied by ORNL. The reported temperatures are the design graphite magazine temperatures. A thermal analysis is currently in progress at ORNL. The actual fuel particle irradiation temperatures are expected to be 100° to 200°C higher than the graphite design temperature. The OPyC coating failure rates were determined during the visual examination conducted at GA. Postirradiation particle density measurements and contact X-ray radiography examinations were performed on selected particle samples, as noted in Tables 9-9 through 9-12.

Irradiation performance of BISO coated ThO_2 particles irradiated in the HT-17, -18, and -19 capsule series revealed a correlation between OPyC

coating rate, OPyC coating OPTAF, and particle survival similar to that observed in the HT-12, -13, -14, and -15 capsule series. BISO particle samples with the higher OPyC coating rates exhibited lower coating failure rates as a function of fast neutron exposure. BISO particle samples from particle batches 6542-01-013 and 6542-02-023 exhibited the lowest rate of OPyC coating failure during irradiation. These particle batches had low OPTAF (1.06) OPyC coatings that were applied at high coating rates of 8.2 and 10.5 $\mu\text{m}/\text{min}$. The difference in OPyC coating density (1.82 and 1.92 g/cm^3) in these two particle batches apparently did not affect the irradiation stability of the OPyC coatings.

The remaining four particle batches had medium density (1.81 to 1.86 g/cm^3) and medium-to-high OPTAF (1.07 to 1.27) OPyC coatings that were applied at relatively low OPyC coating rates of 2.0 to 3.5 $\mu\text{m}/\text{min}$. These particle batches exhibited higher OPyC coating failure rates than the two particle batches with OPyC coatings applied at rates of 8.2 and 10.5 $\mu\text{m}/\text{min}$. Low coating rates can be directly correlated with high PyC preferred orientation (Ref. 9-5). This relationship can also be observed for the particle samples tested in HT-17, -18, and -19, as shown in Fig. 9-14. The optical anisotropy (in relative units) rapidly increases for OPyC coatings applied at rates of less than $\sim 4.0 \mu\text{m}/\text{min}$.

OPyC coatings with higher preferred orientation or anisotropy are more dimensionally unstable during irradiation than isotropic OPyC coatings. OPyC coating failure as a function of fast neutron exposure for BISO particles irradiated in the low-temperature positions of the HT-12, -13, -14, and -15 and HT-17, -18, and -19 capsule series has been plotted in Fig. 9-15. The failure data have been grouped into four regions of OPyC coating rate. The three particle batches tested in the HT-12, -13, -14, and -15 capsule series, which did not exhibit significant optical anisotropy gradients in the OPyC coatings, are also included in this analysis. As shown in Fig. 9-15, as the OPyC coating rate increases the particle survival line moves progressively to higher fast neutron exposures. These results are a further indication that OPyC coating rate is an important

TABLE 9-9
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-17

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure ^(f) (%)	Postirradiation Examinations Performed	
Batch No.	Kernel		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	* Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density ^(e)	X-ray Radio- graphy
	Diam ^(a) (μm)	Density ^(b,c) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,d) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,e) (g/cm ³)	OPTAF ^(b)	Coating Rate (μm/min)	Diam ^(a) (μm)	Density ^(a,e) (g/cm ³)							
6542-01-013-7	509	10.04	79	1.08	89	1.82	1.06	10.5	842	3.44	40	900	3.7	2.2	0	X	X
6542-01-013-10	512	10.04	79	1.08	91	1.82	1.06	10.7	845	3.44	27	1250	4.7	2.9	0	X	X
6542-01-023-1	490	9.75	79	1.17	73	1.83	1.10	2.7	793	3.54	42	900	3.6	2.1	0	X	X
6542-01-023-4	483	9.75	79	1.17	70	1.83	1.10	2.6	759	3.54	29	1250	4.7	2.9	0	X	X
6542-02-023-1	475	10.06	85	1.08	71	1.94	1.06	8.4	784	3.45	43	900	3.4	2.0	0	X	X
6542-02-023-4	477	10.06	87	1.08	70	1.94	1.06	8.2	787	3.45	30	1250	4.7	2.9	0	X	X
6542-02-033-1	471	9.75	74	1.18	67	1.88	1.12	2.0	763	3.54	45	900	3.2	1.9	0	X	X
6542-02-033-4	471	9.75	77	1.18	72	1.88	1.12	2.1	773	3.54	32	1250	4.6	2.8	0	X	X
6542-16-013-1	484	9.75	48	1.08	52	1.81	1.07	3.5	676	4.78	46	900	3.1	1.8	0		X
6542-16-013-4	488	9.75	44	1.08	52	1.81	1.07	3.5	656	4.78	33	1250	4.6	2.8	2	X	X
6542-17-013-1	497	9.75	48	0.95	107	1.86	1.27	2.5	819	3.58	48	900	2.8	1.7	0	X	X
6542-17-013-4	501	9.75	47	0.95	112	1.86	1.27	2.6	819	3.58	35	1250	4.5	2.7	0	X	X

(a) Measured on particle sample that was irradiated.

(b) Measured on parent particle batch.

(c) Kernel densities measured by Hg porosimetry.

(d) Calculated.

(e) Measured using density gradient column.

(f) Determined during visual examination

TABLE 9-10
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-18

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure ^(f) (%)	Postirradiation Examinations Performed	
Batch No.	Kernel		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density ^(e)	X-ray Radio- graphy
	Diam ^(a) (μm)	Density ^(b,c) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,d) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,e) (g/cm ³)	OPTAF ^(b)	Coating Rate (μm/min)	Diam ^(a) (μm)	Density ^(a,e) (g/cm ³)							
6542-01-013-8	509	10.04	80	1.08	87	1.82	1.06	10.2	840	3.44	40	900	7.5	7.8	2	X	X
6542-01-013-11	511	10.04	82	1.08	90	1.82	1.06	10.6	848	3.44	27	1250	9.5	9.7	9	X	X
6542-01-023-2	486	9.75	80	1.17	73	1.83	1.10	2.7	788	3.54	42	900	7.1	7.4	46	X	X
6542-01-023-5	479	9.75	79	1.17	74	1.83	1.10	2.7	779	3.54	29	1250	9.4	9.6	2	X	X
6542-02-023-2	476	10.06	84	1.08	70	1.94	1.06	8.2	783	3.45	43	900	6.8	7.2	15	X	X
6542-02-023-5	474	10.06	84	1.08	70	1.94	1.06	8.2	781	3.45	30	1250	9.4	9.6	0	X	X
6542-02-033-2	469	9.75	74	1.18	71	1.88	1.12	2.1	761	3.54	45	900	6.4	6.8	24		
6542-02-033-5	474	9.75	76	1.18	72	1.88	1.12	2.1	769	3.54	32	1250	9.2	9.5	74		
6542-16-013-2	489	9.75	46	1.08	54	1.81	1.07	3.6	686	4.78	46	900	6.1	6.5	17		X
6542-16-013-5	481	9.75	47	1.08	53	1.81	1.07	3.5	668	4.78	33	1250	9.1	9.4	100		
6542-17-013-2	497	9.75	45	0.95	110	1.86	1.27	2.6	814	3.58	48	900	5.7	6.0	14	X	X
6542-17-013-5	500	9.75	46	0.95	108	1.86	1.27	2.5	821	3.58	35	1250	9.0	9.2	95		

(a) Measured on particle sample that was irradiated.

(b) Measured on parent particle batch.

(c) Kernel densities measured by Hg porosimetry.

(d) Calculated.

(e) Measured using density gradient column.

(f) Determined during visual examination.

TABLE 9-11
SUMMARY DESCRIPTION OF ThO₂ BISO PARTICLES IRRADIATED IN CAPSULE HT-19

Particle Description											Capsule Container No.	Irradiation Conditions			OPyC Failure ^(f) (%)	Postirradiation Examinations Performed	
Batch No.	Kernel		Buffer		Outer Pyrocarbon (OPyC)				Total Particle			Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)		Particle Density ^(e)	X-ray Radio graphy
	Diam ^(a) (μm)	Density ^(b,c) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,d) (g/cm ³)	Thickness ^(a) (μm)	Density ^(b,e) (g/cm ³)	OPTAF ^(b)	Coating Rate (μm/min)	Diam ^(a) (μm)	Density ^(a,e) (g/cm ³)							
6542-01-013-9	511	10.04	81	1.08	90	1.82	1.06	10.6	846	3.44	40	900	11.2	14.2	17	X	X
6542-01-013-12	510	10.04	80	1.08	89	1.82	1.06	10.5	847	3.44	27	1250	14.2	17.2	97		
6542-01-023-3	490	9.75	81	1.17	69	1.83	1.10	2.5	786	3.54	42	900	10.7	13.6	73		
6542-01-023-6	480	9.75	79	1.17	72	1.83	1.10	2.6	784	3.54	29	1250	14.1	17.1	98		
6542-02-023-3	472	10.06	87	1.08	68	1.94	1.06	8.0	783	3.45	43	900	10.3	13.2	3	X	X
6542-02-023-6	474	10.06	85	1.08	71	1.94	1.06	8.4	786	3.45	30	1250	14.0	17.0	0	X	X
6542-02-033-3	471	9.75	77	1.18	72	1.88	1.12	2.1	774	3.54	45	900	9.6	12.5	66		
6542-02-033-6	468	9.75	76	1.18	72	1.88	1.12	2.1	760	3.54	32	1250	13.8	16.8	100		
6542-16-013-3	481	9.75	46	1.08	52	1.81	1.07	3.5	670	4.78	46	900	9.2	12.1	74		
6542-16-013-6	483	9.75	45	1.08	55	1.81	1.07	3.7	641	4.78	33	1250	13.7	16.7	100		
6542-17-013-3	500	9.75	45	0.95	113	1.86	1.27	2.6	816	3.58	48	900	8.5	11.3	98		
6542-17-013-6	497	9.75	44	0.95	115	1.86	1.27	2.7	818	3.58	35	1250	13.4	16.4	100		

(a) Measured on particle sample that was irradiated.
(b) Measured on parent particle batch.
(c) Kernel densities measured by Hg porosimetry.
(d) Calculated.
(e) Measured using density gradient column.
(f) Determined during visual examination.

TABLE 9-12
SUMMARY DESCRIPTION OF ThO₂ TRISO PARTICLES IRRADIATED IN CAPSULES HT-17, -18, AND -19

Batch No.	Particle Description							Irradiation Conditions				Postirradiation	
	Kernel Diam(a) (μm)	Coating Thickness (μm)				Total Particle		Capsule Position	Design Temp (°C)	Fast Fluence x 10 ⁻²¹ (n/cm ²) (E > 0.18 MeV)	Burnup (% FIMA)	OPyC ^(d) Failure (%)	Total Particle Failure ^(d) (%)
		Buffer ^(b)	IPyC ^(b)	SiC ^(a)	OPyC ^(a)	Diam ^(a) (μm)	Density ^(a,c) (g/cm ³)						
6252-01-023-1	498	95	28	26	37	884	3.27	HT-17-49	900	2.7	1.6	56	0
6252-01-023-4	495	95	28	26	37	879	3.27	HT-17-36	1250	4.4	2.7	78	53
6252-01-023-7	491	95	28	28	36	865	3.27	HT-17-51	900	2.4	1.4	83	0
6252-01-023-10	487	95	28	29	33	868	3.27	HT-17-38	1250	4.3	2.6	74	50
6252-01-023-2	496	95	28	26	37	877	3.27	HT-18-49	900	5.4	5.8	83	66
6252-01-023-5	497	95	28	25	37	880	3.27	HT-18-36	1250	8.8	9.1	86	79
6252-01-023-8	489	95	28	29	35	858	3.27	HT-18-51	900	4.7	5.2	61	7
6252-01-023-11	478	95	28	30	34	843	3.27	HT-18-38	1250	8.5	8.9	86	81
6252-01-023-3	502	95	28	26	38	891	3.27	HT-19-49	900	8.2	10.8	88	80
6252-01-023-6	497	95	28	27	37	884	3.27	HT-19-36	1250	13.2	16.3	85	71
6252-01-023-9	489	95	28	28	35	874	3.27	HT-19-51	900	7.1	9.9	74	58
6252-01-023-12	492	95	28	27	35	849	3.27	HT-19-38	1250	12.8	15.9	86	81

(a) Measured on particle sample that was irradiated.
(b) Measured on parent particle batch.
(c) Measured using density gradient column.
(d) Determined during visual examination.

Fig. 9-14. Optical anisotropy versus coating rate of BISO coated ThO_2 particles irradiated in the HT-17, -18, and -19 capsule series

Fig. 9-15. OPyC coating failure versus fast neutron exposure ($E > 0.18$ MeV) for BISO particles irradiated in the low-temperature positions of the HT-12, -13, -14, and -15 and HT-17, -18, and -19 capsule series. The failure data have been grouped into four regions of OPyC coating rate (CR)

parameter relative to particle survival when other parameters such as OPyC coating density are within acceptable limits.

The OPyC coating properties such as coating rate or anisotropy appear to be more important for particle survival than certain coating parameters, i.e., coating thickness, for the particle designs tested. Particle batches 6542-16-013 and 6542-17-017 had similar buffer coating thicknesses and OPyC coating rates. The irradiation performance of these two particle batches was similar; however, there was approximately a 50- μm difference in the OPyC coating thickness.

The particle performance of the samples irradiated in the high-temperature positions of the HT-17, -18, and -19 capsule series followed the same trends as the lower-temperature samples except that failure occurred earlier as would be expected at these high temperatures. It should be mentioned that a temperature of 1450°C is higher than 99.5% of the HTGR core fuel volume will experience during its lifetime.

One batch of TRISO coated ThO_2 particles was also tested in the HT-17, -18, and -19 capsule series to provide samples for postirradiation fission gas pressure and fission gas composition studies. A summary description of these particle samples is given in Table 9-12.

The TRISO coated ThO_2 particles exhibited a rather high failure rate. The SiC coating failure as a function of heavy metal burnup in these particles is shown in Fig. 9-16. As indicated in Table 9-12, the OPyC failure fractions were higher than the SiC coating failure fractions. Failure of the OPyC coating evidently preceded the loss of integrity of the SiC coating, which probably failed by a pressure vessel mechanism. Recent data on large TRISO particles have indicated that failure of the OPyC coating can have a significant effect on the SiC coating integrity during irradiation to full burnup. A coated particles stress model is currently being used to evaluate the SiC coating stresses generated during irradiation in these particle samples, both with and without an intact OPyC coating.

Fig. 9-16. SiC coating failure versus heavy metal burnup for ThO₂ TRISO particles irradiated in the HT-17, -18, and -19 capsule series

The TRISO particle OPyC coatings were applied at a coating rate of 3.0 $\mu\text{m}/\text{min}$. The HT BISO particle irradiation results discussed previously revealed that OPyC coating rates of 3.0 $\mu\text{m}/\text{min}$ may result in sufficiently high PyC coating anisotropy, such that integrity of the OPyC coating cannot be ensured. These results indicate a specification limit for the OPyC coating rate of 3 $\mu\text{m}/\text{min}$ may not be high enough to ensure adequate performance of large fertile TRISO particles during irradiation to the large HTGR peak fast neutron fluence of $8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$).

BISO particle dimensional and density changes as a function of fast neutron exposure and heavy metal burnup are also being evaluated for selected particle samples tested in the HT-17, -18, and -19 capsule series. Postirradiation particle densities have been determined using a density gradient column and X-ray radiography for selected particle samples. These particle dimensional change data are currently being compared with post-irradiation particle densities predicted using a BISO particle irradiation performance model.

FUEL TEST ELEMENTS

Fuel Test Element FTE-4

Irradiation History

Fuel test element FTE-4 was irradiated at Peach Bottom for 449 EFPD in position A14-08 with BOL on July 11, 1971 at 252 EFPD (Core 2) and EOL on September 14, 1973 at 701 EFPD. The calculated EOL fast fluence, thermal fluence, and maximum fuel temperature as a function of distance from the bottom of the core are plotted in Fig. 9-17. The normalized fast and thermal flux (Figs. 9-18 and 9-19, respectively) and the coolant, sleeve outside surface, spine sample, and fuel centerline temperatures (Fig. 9-20) are plotted as a function of axial position averaged for 449 EFPD of irradiation. The calculated fuel temperatures are for reactor operation at 95% power [109 MW(t)] and 105% flow (500,000 lb/hr). These values characteristically represent operation conditions prevailing throughout the lifetime of FTE-4.

Fig. 9-17. Temperature and fluence for FTE-4

Fig. 9-18. Normalized axial fast fluence for FTE-4 (average for 449 EFPD)

Fig. 9-19. Normalized axial thermal fluence for FTE-4 (average for 449 EFPD)

Fig. 9-20. FTE-4 calculated axial temperatures (EOL)

Linear Heat Rate and Temperature Calculation. The changes in the linear heat rate during irradiation because of different control rod configurations are given in Table 9-13.

TABLE 9-13
POWER FACTOR HISTORY FOR FTE-4

Time Phase No.	Time Period ^(a)		FTE-4		Linear Heat Rate Change (%)
			EFPD	Average Power Factor	
	From	To			
1	252	342	90	0.95	Base
2	342	384	132	0.935	+10.6
3	384	438	186	0.92	+10.6
3	438	507	255	0.90	+7.8
4	507	587	335	0.88	+7.8
4-5	587	647	395	0.854	-30.1
5	647	701	449	0.835	-30.1

(a) Effective full power days of Core 2.

It has been shown that the thermal conductivity of graphite decreases upon irradiation. At BOL, the unirradiated graphite thermal conductivity has been assumed as 25 Btu/hr-ft-°F at 1800°F. Upon irradiation, this value decreases rapidly to approximately 16 Btu/hr-ft-°F, where it remains relatively constant until EOL. Therefore, calculations were made for BOL linear heat rates using a graphite thermal conductivity of 25 Btu/hr-ft-°F and for EOL values using 16 Btu/hr-ft-°F. Both temperature profiles are shown in Fig. 9-17.

The fuel centerline temperatures were obtained from the TES code (one-dimensional code for Peach Bottom test elements), which does not treat the axial interruptions in the fuel where the individual fuel bodies meet. Although these end effects are localized, previous calculations have predicted steep axial thermal gradients at the ends of the fuel bodies, as indicated in Fig. 9-17.

Temperature Measurements. FTE-4 was assembled with one W/Re thermocouple and one C/A thermocouple at 75.5 in. total height, (i.e., 49.9 in. active core height). The W/Re thermocouple was near the center spine samples and the C/A thermocouple was near the outer sleeve position. The C/A thermocouple had no temperature readings from the beginning of irradiation, signifying a malfunction.

Readings, which were recorded on a weekly basis, were corrected for:

1. Changes in cold junction temperature.
2. Adjustment in calibration equipment for W/Re type only, where readings are 40°F to high.
3. Decalibration of W/Re type because of permutation of emf output due to neutron bombardment.

The uncorrected thermocouple readings plotted in Fig. 9-21 took into account correction factors (1) and (2). The corrected thermocouple readings in Fig. 9-21 include the permutation correction. This correction term has been deduced from experimental results by a mathematical expression as shown in Ref. 9-8.

The corrected thermocouple readings and the element power factor history have been used for determination of the fuel centerline temperatures at the thermocouple axial position. The power factors are given in Table 9-13.

Average thermocouple readings for five time phases are shown in Table 9-14. Thermocouple readings are lower by a certain ΔT than the fuel centerline temperature at the same axial position. This correction term is given for different power factors and both thermocouple types for BOL thermal conductivities ($K = 25 \text{ Btu/hr-ft-}^\circ\text{F}$) in Fig. 9-22 and approximately EOL ($K = 16 \text{ Btu/hr-ft-}^\circ\text{F}$) in Fig. 9-23. These correction terms are shown in

Fig. 9-21. Temperature history from thermocouple readings for FTE-4: (a) July 1971 through June 1972

Fig. 9-21. Temperature history from thermocouple readings for FTE-4: (b) July 1972 through February 1973

Fig. 9-21. Temperature history from thermocouple readings for FTE-4: (c) March 1973 through September 1973

TABLE 9-14
FTE-4 AVERAGE THERMOCOUPLE READINGS

Time Phase	Irradiation Period			Total Readings	W/Re Reading (°C)	Average Power Factor
	From	To	EFPD			
1	7/11/71	10/31/71	0-90	13	1092	0.95
2	11/22/71	1/8/72	90-132	4	1149	0.935
3	7/5/72	11/25/72	132-255	20	1072	0.90-0.92
4	1/12/73	5/18/73	255-335	18	1191	0.88-0.854
5	6/9/73	9/14/73	335-449	13	1216	0.854-0.835

Fig. 9-22. Correction to thermocouple reading to obtain maximum fuel temperature as a function of test element to core average element power, BOL

Fig. 9-23. Correction to thermocouple reading to obtain maximum fuel temperature as a function of test element to core average element power, EOL

Table 9-15 for the 5 time phases (time phases 2, 3, and 4, called middle-of-life (MOL), use an arithmetic average). Table 9-16 gives the peak fuel temperatures as derived from thermocouple readings. The thermocouple position nearly coincides with the overall axial peak of the fuel temperature, as shown in Fig. 9-17. Calculated and measured peak fuel temperatures are included in Fig. 9-21.

Temperature and Fluence Evaluation Results

The following peak values can be quoted:

EOL peak fast fluence	$\sim 1.90 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)
EOL peak thermal fluence	$\sim 2.51 \times 10^{21} \text{ n/cm}^2$ ($E < 2.38 \text{ eV}$)
EOL peak fuel temperature (physics calculations)	2077°F (1136°C)
EOP peak center sample temperature	1957°F (1069°C)
Average peak fuel temperature (derived from corrected thermocouple readings)	2210°F (1210°C $\pm 100^\circ\text{C}^*$) (W/Re)

*Bracket of temperature readings at full power during irradiation history.

The following conclusions can be drawn from the temperature evaluation:

1. The overall tendency is increasing temperatures with time.
2. Corrected thermocouple readings indicate an average during life peak fuel temperature of $1210^\circ\text{C} \pm 100^\circ\text{C}$, which is 76°C greater than the calculated EOL peak fuel temperature of 1136°C .

TABLE 9-15
 ΔT CORRECTION FOR THERMOCOUPLE READINGS AND PEAK FUEL TEMPERATURE (AT SAME AXIAL LEVEL)^(a)

	Power Factor = 0.95			Power Factor = 0.935			Power Factor = 0.91			Power Factor = 0.87			Power Factor = 0.84		
	BOL K=25 ^(b)	MOL	EOL K=16	BOL K=25	MOL	EOL K=16	BOL K=25	MOL	EOL K=16	BOL K=25	MOL	EOL K=16	BOL K=25	MOL	EOL K=16
Thermocouple to peak fuel correction, W/Re ΔT ($^{\circ}\text{C}$)	84	74	64	83	73	62	80	70	60	76	66	56	72	63	53
Time phase (in Tables 9- and 9-)	1				2			3			4				5

^(a) From Figs. 9-22 and 9-23.

^(b) Btu/hr-ft- $^{\circ}\text{F}$.

TABLE 9-16
FTE-4 PEAK FUEL TEMPERATURE FROM W/Re THERMOCOUPLE
READING (TABLE 9-14) AND CORRECTION (TABLE 9-15)

	Peak Fuel Temp. (°C)
Time phase 1 0-90 EFPD	1176
Time phase 2 90-132 EFPD	1222
Time phase 3 132-255 EFPD	1142
Time phase 4 255-335 EFPD	1257
Time phase 5 335-449 EFPD	1269
Time phases 1-5 0-449 EFPD	1210 ^(a)

(a) Balanced via EFPD.

Hot Cell Examination and Results

Upon arrival from Peach Bottom on March 29, 1974, the helium-filled aluminum canister containing fuel test element FTE-4 was removed from the Hallam cask and injected into the high-level cell without accident. The canister was girdle cut approximately 3/16 in. deep and 3 ft from the top end of the canister and the piece removed. The element was pulled out until clear of the canister, thereby exposing the thermocouple contacts. The remaining portion of the canister was discarded. Thermocouple resistance readings were measured and recorded (Table 9-17). Preirradiation measurements were not available; therefore, no comparison could be made. A composite photograph of the total element is shown in Fig. 9-24.

In order to section the fuel element, it was supported horizontally by a series of V-rollers, which facilitated horizontal and rotational movement. A special extension drill was inserted through the purge gas inlet hole in the upper reflector and a hole was drilled through the porous graphite plug. The fuel bodies were then seated against the bottom connector by using a push rod through this hole and applying force. While the bodies were held down, the upper reflector was removed by plunge cutting through the sleeve at a point about 2 in. below the sleeve-reflector joint. The sleeve was then girdle cut at a point about 16 in. above the sleeve-bottom connector joint. After inspecting the top surface of the graphite cloth at the upper end of the internal trap and finding no fuel particles, the bottom section of the lower reflector was removed. The two thermocouples were pulled out and the entire bottom assembly was discarded.

The fuel bodies were subsequently removed from the sleeve by inserting a push rod into the upper opening, holding the sleeve, and applying a force. Each individual fuel body was carefully examined externally, and the graphite integrity was assessed (see Figs. 9-25 through 9-27). Following visual inspection, the fuel bodies were placed in a special holding fixture to facilitate the removal of the center samples. A hole was drilled through the bottom sample hole plug and a push rod was inserted. When the upper

TABLE 9-17
FTE-4 THERMOCOUPLE RESISTANCE MEASUREMENTS^(a)

	+ to Ground Circuit (Ω)		- to Ground Circuit (Ω)		Loop (Ω)	
	Pre- Irrad.	Post- Irrad.	Pre- Irrad.	Post- Irrad.	Pre- Irrad.	Post- Irrad.
W-3% Re/W-25% Re thermocouple (ARI S/N 38247)	N/A	7.0	N/A	19.2	N/A	20.4
C/A thermocouple (ARI S/N 37047)	N/A	35.9	N/A	16.2	N/A	52.2

^(a) Accuracy $\pm 2\Omega$.

K7410 (3-14)

Fig. 9-24. Full-length mosaic of FTE-4

K7410 (Top 21-24, Bottom 25-28)

Fig. 9-25. Composite photograph of FTE-4 fuel body 1; top, 180° line and bottom, 0° line

K7410 Top 35-42, Bottom 43-48)

Fig. 9-26. Composite photograph of FTE-4 fuel body 2; top, 90° line and bottom, 270° line

K7410 (Top 69-74, Bottom 77-82)

Fig. 9-27. Composite photograph of FTE-4 fuel body 3: top, 0° line and bottom, 180° line

sample hole plug was unscrewed, the distances from the top of the spine samples to the edge of the holes were measured. These measurements are compared with preirradiation measurements in Table 9-18. Changes are in the range to be expected for H-327 graphite except for the spine stack in body 3, which indicates an increase in length. However, this can be related to stack perturbation (e.g., a small chip between two spine samples could have caused this effect). The spine samples were pushed out one at a time, identified, and placed in a pan. All these operations were carried out over this pan in order to catch any loose fuel particles from the drilled holes. However, none were found. No problems were encountered during the removal of the spine samples. The inscribed identity of all spine samples was the same as recorded in the drawings except for the thermal stability samples of fuel body 3 where the type 2 samples had a prefix of 2- on the drawings but no prefixes were observed on the actual sample. The activity of each spine sample was measured and recorded. All spine samples for each body were placed in separate, custom-built purge containers. The spine samples are described in Table 9-19.

Upon discharge of the spine samples, the graphite plugs capping the fuel holes were easily removed, and the distances from the top of the fuel rods to the edge of the fuel holes were measured. These measurements are compared with preirradiation measurements in Table 9-20.

Holes were drilled on the bottom of the fuel bodies to enable the fuel rods to be pushed out with an 1/8-in.-diameter push rod. After all of the fuel rods had been removed, the fuel bodies were sectioned as shown in Fig. 9-28. No cracks could be found on the sectioned faces, which are shown in Figs. 9-29 through 9-31. Therefore, it can be concluded that the structural behavior of the graphite fuel bodies was excellent. The numbers on the photographs are fuel hole identification numbers. The graphite slices will be used for autoradiography and some cesium profile measurements. Residual stress examination will be carried out on some of the graphite bodies.

TABLE 9-18
DISTANCE d FROM TOP OF SPINE TO EDGE OF HOLE IN FTE-4

Body No.	Spine Stack Length (in.)	Preirrad., d_1 (in.)	Postirrad., d_2 (in.)	Δd (in.)	$\Delta G^{(a)}$ (in.)	$\frac{-\Delta d + \Delta G}{L}$ (%)
1	28.487	2.125	2.266	+0.141	-0.0314	-0.605
2	28.361	2.261	2.313	+0.052	-0.1068	-0.560
3	28.562	2.058	1.914	-0.144	-0.0759	+0.238

(a) ΔG = length change of graphite fuel body.

TABLE 9-19
FTE-4 SPINE SAMPLES

Position	Pre-Irrad. Length (in.)	Pre-Irrad. Weight (g)	Sample Type	Identification Number	Activity Reading
Body 1					
1	3.000	--	Boronated stainless steel	21	10 r (1 ft)
2	1.502	32.60	Stress relaxation	21	48 r (C) ^(a)
3	3.815	78.82	Mechanical test assembly	IV	10 r (C) ^(a)
4	1.258	25.04	Diffusion	61	7 r (1 ft)
5	1.261	24.80	Diffusion	52	10 r (2.5 ft)
6	1.257	24.69	Diffusion	41	4 r (1 ft)
7	1.256	24.59	Diffusion	31	9 r (C) ^(a)
8	1.255	24.88	Diffusion	12	2.8 r (1 ft)
9	1.258	24.68	Diffusion	10	10 r (C) ^(a)
10	1.256	24.70	Diffusion	4	10 r (2.5 ft)
11	1.258	25.01	Diffusion	64	5 r (1 ft)
12	1.261	24.65	Diffusion	54	10 r (1 ft)
13	1.257	24.63	Diffusion	46	10 r (1.5 ft)
14	1.260	24.71	Diffusion	34	3.5 r (1 ft)
15	1.253	24.86	Diffusion	35	8 r (C) ^(a)
16	1.255	24.69	Diffusion	27	4.5 r (1 ft)
17	1.255	24.62	Diffusion	18	5 r (1 ft)
18	1.257	24.80	Diffusion	6	25 r (1 ft)
19	1.313	29.83	Graphite spacer	26	8 r (C) ^(a)
	28.487				
Body 2					
1	2.495	57.00	Graphite spacer	27	20 r (C) ^(a)
2	1.263	25.00	Diffusion	58	10 r (1.5 ft)
3	1.256	24.65	Diffusion	21	4 r (1 ft)
4	1.257	25.05	Diffusion	67	4 r (1 ft)
5	1.258	24.39	Diffusion	28	65 r (1 ft)
6	1.257	24.10	Diffusion	37	3 r (C) ^(a)
7	1.256	22.80	Diffusion	49	100 mr (C) ^(a)
8	1.256	24.89	Diffusion	22	5 r (1 ft)
9	4.020	92.55	Advanced fuel rods	1-B	10 r (1.5 ft)
10	4.000	92.12	Advanced fuel rods	1-A	9 r (1.5 ft)
11	1.255	101.89	Fission product release	NB-2	8 r (1 ft)
12	1.250	24.75	Fission product release	10-P	1.5 r (1 ft)
13	1.258	102.03	Fission product release	NB-9	8 r (1 ft)
14	1.260	24.88	Fission product release	42-P	1 r (1 ft)
15	1.262	102.21	Fission product release	NB-13	8 r (1 ft)
16	1.259	24.61	Fission product release	146-P	2.5 r (1 ft)
17	1.499	--	Stress relaxation	10	0
	28.361				
Body 3					
1	1.800	40.29	Graphite spacer	29	250 mr (C) ^(a)
2	5.760	137.63	Thermal stability	1-G	15 r (2 ft)
3	5.750	137.38	Thermal stability	1-H	15 r (2 ft)
4	1.244	26.29	Thermal stability	2-16	2 r (1 ft)
5	1.250	24.06	Thermal stability	2-22	2 r (1 ft)
6	1.250	--	Thermal stability	2-28	1.5 r (1 ft)
7	1.248	--	Thermal stability	2-4	10 r (1 ft)
8	1.255	--	Thermal stability	2-10	8 r (1 ft)
9	1.253	--	Stress relaxation	45	300 mr (C) ^(a)
10	4.002	--	Hollow graphite spacer	28	100 mr (C) ^(a)
11	3.750	--	Absorber rod		10 r (2 ft)
	28.562				

(a) Contact.

TABLE 9-20
FTE-4 FUEL STACK LENGTH^(a)

Hole No.	Stack Length, L (in.) (accuracy: +0.03, -0.01)		ΔL (in.)	$\Delta L/L$ (%)	Plenum P ^(b) (in.)		ΔP (in.)	$\frac{-\Delta P + \Delta G^{(c)}}{L}$ (%)
	Pre-Irrad.	Post-Irrad.			Pre-Irrad.	Post-Irrad.		
Bottom Body								
1	27.165	26.6626	-0.5024	-1.85	2.638	3.078	+0.440	-1.735
2	27.179	26.6966	-0.4824	-1.77	2.619	3.203	+0.504	-2.264
3	27.224	26.9253	-0.2987	-1.10	2.576	2.828	+0.252	-1.041
4	27.204	26.8769	-0.3271	-1.20	2.581	2.859	+0.267	-1.097
5	27.216	27.0121	-0.2039	-0.75	2.581	2.758	+0.177	-0.766
6	27.255	27.0233	-0.2317	-0.85	2.541	2.766	+0.225	-0.941
7	27.164	26.8531	-0.3109	-1.14	2.636	2.096	+0.270	-1.110
8	<u>27.166</u>	<u>26.8373</u>	<u>-0.3287</u>	<u>-1.21</u>	<u>2.634</u>	<u>2.914</u>	<u>+0.312</u>	<u>-1.263</u>
Avg.	27.197	26.8609	-0.3357	-1.23	2.602	2.914	+0.312	-1.263
Center Body								
1	27.168	26.4649	-0.7031	-2.59	2.626	2.836	+0.210	-1.166
2	27.168	26.4537	-0.7143	-2.63	2.626	3.047	+0.421	-1.943
3	27.206	26.7423	-0.4637	-1.70	2.590	2.828	+0.238	-1.267
4	27.194	26.7735	-0.4205	-1.55	2.601	2.695	+0.094	-0.074
5	27.258	26.8752	-0.3828	-1.40	2.539	2.797	+0.258	-1.338
6	27.246	26.8714	-0.3746	-1.38	2.551	2.820	+0.269	-1.379
7	27.151	26.6253	-0.5257	-1.94	2.645	2.867	+0.222	-1.211
8	<u>27.162</u>	<u>26.6140</u>	<u>-0.5480</u>	<u>-2.02</u>	<u>2.634</u>	<u>2.875</u>	<u>+0.241</u>	<u>-1.280</u>
Avg.	27.194	26.6775	-0.5166	-1.90	2.602	2.846	+0.244	-1.290
Top Body								
1	27.177	26.5563	-0.6207	-2.28	2.628	2.398	-0.230	+0.567
2	27.193	26.5560	-0.6370	-2.34	2.612	2.289	-0.323	+0.909
3	27.217	26.7784	-0.4386	-1.61	2.588	2.625	+0.037	-0.415
4	27.210	26.7858	-0.3557	-1.56	2.593	2.242	-0.351	+1.011
5	27.246	26.8903	-0.3557	-1.31	2.558	2.461	-0.097	+0.077
6	27.238	26.8958	-0.3422	-1.26	2.566	2.617	+0.051	-0.466
7	27.154	26.7093	-0.4447	-1.64	2.649	2.359	-0.290	+0.788
8	<u>27.156</u>	<u>26.6942</u>	<u>-0.4618</u>	<u>-1.70</u>	<u>2.647</u>	<u>2.852</u>	<u>+0.205</u>	<u>-1.034</u>
Avg.	27.199	26.7333	-0.4657	-1.71	2.605	2.480	-0.125	+0.181

(a) Nominal depth = 29.79 (+0.03, -0.01) in. including plenum, minus compact pusher thickness. Nominal stack length = 27.16 (+0.25, -0.16) in. Nominal compact pusher thickness = 0.12 in.

(b) Plenum P = distance from top of fuel rod to edge of hole.

(c) ΔG = length change of graphite fuel body (see Table 9-18).

Fig. 9-28. Sampling of the bodies for measurement and autoradiography

K7410-101 (a)

K7410-99 (b)

K7410-98 (c)

K7410-96 (d)

Fig. 9-29. FTE-4 fuel body 1: (a) top of section 1-1, (b) top of section 1-2, (c) top of section 1-3, and (d) bottom of section 1-4

K7410-105 (a)

K7410-103 (b)

K7410-104 (c)

K7410-102 (d)

Fig. 9-30. FTE-4 fuel body 2: (a) top of section 2-1, (b) top of section 2-2, (c) bottom of section 2-2, and (d) bottom of section 2-3

K7410-106 (a)

K7410-107 (b)

K7410-108 (c)

K7410-109 (d)

Fig. 9-31. FTE-4 fuel body 3: (a) top of section 3-1, (b) top of section 3-2, (c) bottom of section 3-2, and (d) bottom of section 3-3

Gamma Scan Results

Upon removal during the 701 EFPD shutdown, FTE-4 was gamma scanned at Peach Bottom for fueled length determinations. This was done by the Peach Bottom standard procedure, where the fuel element is carefully raised into (or lowered from) the charge machine, passing a collimator slot in the radial shielding. Gamma radiation that traverses the collimator path is detected by a scintillation crystal multiplier assembly and amplified. Both the count rate and the element position are recorded simultaneously on a strip chart from which the actual measurement of the fuel length is obtained.

The gamma scan strip chart for FTE-4 is shown in Fig. 9-32. The gamma scan results are summarized in Table 9-21 together with competitive measurements. All three fuel bodies show excellent agreement for length changes, deduced from averaged fuel stack length, and for axial and radial fuel rod measurements (i.e., shrinkage values). However, there is some discrepancy in the plenum measurements (i.e., free distance between fuel stack and end of the graphite body). The lower amount of shrinkage in the center body and the expansion in the top fuel body may be due to perturbation of the fuel stack. Good results were obtained for a comparison of the average fuel stack length measurement for the bottom fuel body.

TABLE 9-21
FTE-4 FUEL STACK DIMENSIONAL CHANGES

	FTE-4/1, Bottom	FTE-4/2, Center	FTE-4/3, Top
Average fuel stack length (L), preirradiation, in.	27.197	27.194	27.199
Gamma scan ($\Delta L/L$), %	-1.61	-2.28	-2.01
Plenum measurement ($\Delta L/L$), %	-1.26	-1.29	(+0.18)
Average fuel stack length measurement ($\Delta L/L$), %	-1.23	-1.90	-1.71
Axial fuel rod measurement ($\Delta L/L$), %	-1.63	-2.19	-2.02
Radial fuel rod measurement ($\Delta D/D$), %	-1.58	-2.30	-2.00

Fig. 9-32. FTE-4 gamma scan trace

The following conclusions can be drawn from the gamma-scan results:

1. Gamma scanning gives an accurate indication of fuel stack length changes during irradiation.
2. Average axial and radial fuel rod measurements agree very well with the gamma-scan results.

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TASK 11

GRAPHITE RESEARCH

INTRODUCTION

Work during the current quarter was accomplished under the following major tasks: (a) characterization of production-size graphites, (2) graphite irradiations, and (3) statistical measurements of graphite strength.

GRAPHITE CHARACTERIZATION

Preliminary evaluation of grade TS-1240, manufactured by Union Carbide Corporation, is continuing. Density, impurity, and initial tensile strength data were reported previously (Refs. 11-1, 11-2). Additional tensile property data are given in Tables 11-1 and 11-2.

An in-depth evaluation of grade TS-1240 is under way. Initial thermal expansivity and thermal conductivity data were reported previously (Ref. 11-2). Additional thermal expansivity data are given in Tables 11-3 and 11-4.

The tensile property data presented in Tables 11-1 and 11-2 were obtained on specimens 0.505 in. in diameter by 3 in. long. Each specimen was loaded to 1000 psi, unloaded to zero stress, and reloaded to fracture while recording the stress-strain curve. The elastic modulus for each loading was taken as the chord modulus between 100 and 1000 psi. In each test, the elastic modulus for the second loading was increased by 10 to 20% over the initial loading value. Values of ultimate strength, strain to fracture, and the permanent deformation in the specimen after the first loading cycle were also obtained from the stress-strain curve.

TABLE 11-1
 TLNSILE PROPERTIES OF TS-1240 GRAPHITE
 Log 5651-72, density = 1.74 g/cm³, 0.505-in.-diameter by 3.0-in.-long samples

Specimen No.	Position and Orientation ^(a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-72-1A-L001	LC(11) ↓	0.83	1.07	--	0.260	1980
-L004		0.78	0.94	--	0.170	1443
-L005		1.05	0.82	--	0.156	1210
-L009		0.75	0.92	--	0.304	1941
-1B-L011		0.85	0.99	--	0.234	1742
-L014		0.78	1.00	--	0.176	1454
-L015		0.65	0.80	--	0.266	1618
-L019		0.75	0.93	--	0.227	1661
Mean		0.80	0.93	--	0.224	1631
Std. Dev.		0.116	0.090	--	0.053	260
5651-72-1A-L025	LC(1) ↓	1.02	1.01	--	(b)	2121
-L026		0.92	1.02	--	0.308	2179
-L027		0.83	0.95	--	0.326	2079
-L030		0.85	1.05	--	0.200	1776
-1B-L037		0.78	0.96	--	0.312	1979
-L038		0.83	0.95	--	0.295	1964
-L039		0.95	1.02	--	0.288	2127
-L042		0.92	0.98	--	0.310	2071
Mean		0.88	0.99	-	0.291	2037
Std. Dev.		0.078	0.038	--	0.042	128
5651-72-3A-L050	MLC(11) ↓	0.79	1.02	0.024	0.252	1718
-L053		0.79	0.95	0.013	0.287	1906
-L054		0.73	0.95	0.028	0.275	1734
L058		0.83	1.11	0.026	0.320	2078
-3B-L061		0.81	1.11	0.034	0.292	1969
-L064		0.80	1.12	0.023	0.236	1753
-L065		0.80	1.11	0.031	0.236	1694
-L069		0.78	1.03	0.023	0.240	1643
Mean		0.79	1.05	0.025	0.267	1812
Std. Dev.		0.029	0.073	0.006	0.031	154
5651-72-3A-L079	MLC(1) ↓	0.80	0.98	0.013	0.356	2098
-L076		0.75	0.96	0.023	(b)	1667
-L077		0.78	0.98	0.017	0.307	1943
-L080		0.78	1.05	0.016	0.337	2035
-3B-L087		0.80	0.99	0.018	0.355	2150
-L088		0.77	1.01	0.028	0.345	2138
-L089		0.77	0.89	0.016	0.370	2073
-L092		0.79	0.96	0.021	0.320	1997
Mean		0.78	0.98	0.019	0.341	2012
Std. Dev.		0.017	0.046	0.005	0.022	156

TABLE 11-1 (Continued)

Specimen No.	Position and Orientation (a)	Elastic Modulus $\times 10^{-6}$ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-72-1A-L101	EL(11) ↓	0.84	1.03	--	0.295	2044
-L103		0.85	1.05	--	0.282	2057
-L105		0.87	1.00	--	0.314	2083
-L107		0.88	1.07	--	0.310	2240
-1B-L111		0.86	1.10	--	0.264	1996
-L112		0.85	1.07	--	0.249	1931
-L114		0.82	1.12	--	0.260	2000
-L116		<u>0.78</u>	<u>0.99</u>	--	<u>0.228</u>	<u>1712</u>
Mean		0.84	1.05	--	0.275	2008
Std. Dev.		0.031	0.046	--	0.030	150
5651-72-1A-L125	EE(1) ↓	0.81	0.99	--	(b)	1915
-L126		0.81	1.01	--	0.354	2294
-L129		1.08	1.05	--	0.323	2242
-L130		0.78	0.96	--	0.330	2050
-1B-L142		0.89	1.00	--	(b)	1569
-L143		0.78	0.94	--	0.330	2070
-L146		0.84	1.00	--	0.324	2165
-L147		<u>0.84</u>	<u>0.97</u>	--	<u>0.305</u>	<u>2020</u>
Mean		0.85	0.99	--	0.328	2040
Std. Dev.		0.098	0.034	--	0.016	226
5651-72-3A-L151	MLE(11) ↓	0.73	0.97	0.030	(b)	1585
-L153		0.83	1.10	0.031	0.276	2010
-L155		0.86	1.08	0.016	0.315	2167
-L157		0.92	1.22	0.016	0.268	2019
-3B-L161		0.83	1.15	0.030	(b)	2042
-L162		0.77	1.00	0.024	0.286	1866
-L164		0.84	1.02	0.017	0.240	1762
-L166		<u>0.81</u>	<u>1.05</u>	<u>0.020</u>	<u>0.220</u>	<u>1624</u>
Mean		0.82	1.07	0.023	0.267	1884
Std. Dev.		0.057	0.082	0.007	0.033	210
5651-72-3A-L175	MLE(1) ↓	0.81	1.00	0.020	(b)	1640
-L176		0.80	0.99	0.023	(b)	1925
-L179		0.83	0.96	0.006	0.304	1978
-L180		0.80	0.95	0.010	0.298	1918
-3B-L193		0.80	0.96	0.010	0.284	1870
-L194		0.76	0.99	0.022	0.358	2145
-L197		0.86	1.07	0.017	0.353	2261
-L198		<u>0.84</u>	<u>0.99</u>	<u>0.010</u>	<u>0.320</u>	<u>2153</u>
Mean		0.81	0.99	0.015	0.319	1986
Std. Dev.		0.030	0.037	0.006	0.030	196

(a) EC = end center, MLC = midlength center, EE = end edge, MLE = midlength edge.

(b) Broke at epoxy joint.

TABLE 11-2
TENSILE PROPERTIES OF TS-1240 GRAPHITE
Log 5651-75, density = 1.81 g/cm³, 0.505-in.-diameter by 3.0-in.-long samples

Specimen No.	Position and Orientation(a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-75-1A-L001	EC(11) ↓	0.95	1.22	0.020	0.270	2060
-L004		1.05	1.23	0.014	0.222	1920
-L005		1.05	1.23	0.012	0.200	1770
-L009		1.08	1.30	0.012	0.230	2007
-1B-L011		1.05	1.30	0.015	0.290	2350
-L014		1.03	1.25	0.015	0.272	2200
-L015		1.00	1.20	0.016	0.270	2140
-L019		1.00	1.30	0.020	0.245	2038
Mean		1.03	1.25	0.015	0.250	2061
Std. Dev.		0.040	0.041	0.003	0.031	176
5651-75-1A-L025	EC(1) ↓	1.00	1.22	0.008	0.323	2390
-L026		0.90	1.08	0.022	0.280	2020
-L027		0.87	1.07	0.020	0.155	1240
-L030		0.92	1.12	0.019	0.320	2325
-1B-L037		0.89	1.11	0.020	0.387	2600
-L038		0.87	1.08	0.020	0.278	1990
-L039		0.85	1.07	0.023	0.288	2010
-L042		0.95	1.11	0.016	0.397	2800
Mean		0.91	1.11	0.018	0.303	2171
Std. Dev.		0.049	0.049	0.005	0.075	478
5651-75-3A-L050	MLC(11) ↓	1.18	1.30	0.000	(b)	1821
-L053		1.25	1.47	0.005	0.305	2592
-L054		1.12	1.36	0.006	(b)	1225
-L058		1.14	1.41	0.013	0.314	2605
-3B-L061		1.10	1.34	0.013	(b)	2589
-L064		1.14	1.45	0.013	(b)	2485
-L065		1.05	1.27	0.016	(b)	2318
-L069		1.17	1.36	0.012	(b)	2313
Mean		1.14	1.37	0.010	0.309	2243
Std. Dev.		0.059	0.070	0.005	0.006	486
5651-75-3A-L075	MLC(1) ↓	0.88	--	--	0.108	996
-L076		1.25	--	--	0.150	596
-L077		0.87	--	--	0.130	865
-L080		0.93	1.15	0.016	0.235	1821
-3B-L087		0.82	1.03	0.020	0.140	1056
-L096		1.05	1.14	0.008	0.360	2666
-L089		0.93	--	--	0.084	833
-L092		0.95	1.14	0.007	0.390	2696
Mean		0.96	1.11	0.013	0.199	1441
Std. Dev.		0.135	0.057	0.006	0.117	844

TABLE 11-2 (Continued)

Specimen No.	Position and Orientation (a)	Elastic Modulus $\times 10^{-6}$ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-75-1A-L101	EE(11) ↓	0.98	1.10	0.014	(b)	1250
-L103		0.97	1.16	0.017	0.308	2345
-L105		0.91	1.13	0.020	0.260	2000
-L107		0.94	1.08	0.016	0.183	1475
-1B-L111		0.92	1.11	0.018	0.335	2335
-L112		0.95	1.10	0.020	0.320	2350
-L114		0.93	1.12	0.020	(b)	1950
-L116		0.93	1.10	0.017	0.277	2085
Mean		0.94	1.11	0.018	0.280	1973
Std. Dev.		0.024	0.024	0.002	0.055	413
5651-75-1A-L125	EE(1) ↓	0.90	1.07	0.017	0.335	2310
-L126		0.78	1.07	0.022	0.290	2035
-L129		0.96	1.14	0.009	0.310	2300
-L130		0.86	1.12	0.023	(b)	1684
-1B-L142		0.93	1.00	0.018	0.308	2225
-L143		0.86	1.03	0.020	0.330	2110
-L146		0.88	1.07	0.020	0.340	2325
-L147		0.95	1.05	0.018	0.290	2175
Mean		0.89	1.07	0.018	0.314	2145
Std. Dev.		0.059	0.040	0.004	0.021	213
5651-75-3A-L151	MLE(11) ↓	1.06	1.27	0.013	0.356	2840
-L153		1.10	1.34	0.015	0.340	2703
-L155		1.08	1.32	0.013	(b)	1855
-L157		1.10	1.34	0.014	(b)	2679
-3B-L161		1.15	1.27	0.016	0.400	3152
-L162		1.02	1.29	0.015	0.365	2925
-L164		1.18	1.45	0.006	(b)	2776
-L166		1.10	1.32	0.010	(b)	2175
Mean		1.10	1.32	0.013	0.365	2638
Std. Dev.		0.050	0.058	0.003	0.025	420
5651-75-3A-L175	MLE(1) ↓	0.89	1.20	0.009	0.215	1740
-L176		0.87	1.10	0.024	0.255	1884
-L179		0.90	1.12	0.015	0.216	1692
-L180		0.87	1.12	0.016	0.196	1615
-3B-L193		0.87	1.12	0.022	0.150	1315
-L194		0.96	1.12	0.010	0.165	1497
-L197		0.93	1.15	0.020	0.367	2571
-L198		0.98	1.20	0.010	0.360	2600
Mean		0.91	1.14	0.016	0.240	1864
Std. Dev.		0.043	0.039	0.006	0.082	476

(a) EC = end center, MLC = midlength center, EE = end edge, MLE = midlength edge.

(b) Broke at epoxy joint.

TABLE 11-3
THERMAL EXPANSIVITY OF TS-1240 GRAPHITE
(Log 5651-72, density = 1.74 g/cm³)

Specimen No.	Orientation	Location in Log	Mean Coefficient of Thermal Expansion, $\alpha \times 10^6 \text{ }^\circ\text{C}^{-1}$ (22°-500°C)
5651-72-1A-007A	Parallel ↓	End Center ↓	4.51
-007B			4.43
-010A			4.60
-010B			4.45
-1B-032A			4.36
-032B			4.40
-035A			4.64
-035B			4.52
Mean			4.48
Std. Dev.			0.097
5651-72-1A-063A	Perpendicular ↓	End Center ↓	4.70
-063B			5.08
-066A			4.75
-066B			4.61
-1B-073A			4.80
-073B			4.83
-076A			4.66
-076B			5.33
Mean			4.84
Std. Dev.			0.243
5651-72-1A-052A	Parallel ↓	End Edge ↓	3.96
-052B			4.45
-054A			4.32
-054B			4.75
-1B-057A			4.28
-057B			4.48
-059A			4.96
-059B			4.73
Mean			4.49
Std. Dev.			0.317
5651-72-1A-081A	Perpendicular ↓	End Edge ↓	4.77
-081B			5.57
-083A			5.07
-083B			5.04
-1B-085A			5.00
-085B			5.16
-087A			4.31
-087B			4.83
Mean			4.96
Std. Dev.			0.360

TABLE 11-3 (Continued)

Specimen No.	Orientation	Location in Log	Mean Coefficient of Thermal Expansion, $\alpha \times 10^6 \text{ }^\circ\text{C}^{-1}$ (22°-500°C)
5651-72-3A-107A	Parallel ↓	Midlength Center ↓	4.16
-107B			4.29
-110A			4.00
-110B			4.35
-3B-132A			3.83
-132B			4.10
-135A			3.78
-135B			<u>3.90</u>
Mean			4.05
Std. Dev.			0.209
5651-72-3A-163A	Perpendicular ↓	Midlength Center ↓	4.64
-163B			4.45
-166A			4.42
-166B			4.63
-3B-173A			4.50
-173B			4.31
-176A			4.55
-176B			<u>4.52</u>
Mean			4.50
Std. Dev.			0.109
5651-72-3A-152A	Parallel ↓	Midlength Edge ↓	3.99
-152B			4.02
-154A			4.45
-154B			3.92
-3B-157A			4.16
-157B			4.19
-159A			4.17
-159B			<u>4.03</u>
Mean			4.12
Std. Dev.			0.165
5651-72-3A-181A	Perpendicular ↓	Midlength Edge ↓	4.90
-181B			4.44
-183A			5.07
-183B			4.65
-3B-185A			4.65
-185B			4.50
-187A			4.65
-187B			<u>4.63</u>
Mean			4.68
Std. Dev.			0.205

TABLE 11-4
THERMAL EXPANSIVITY OF TS-1240 GRAPHITE
(Log 5651-75, density = 1.81 g/cm³)

Specimen No.	Orientation	Location in Log	Mean Coefficient of Thermal Expansion, $\alpha \times 10^6 \text{ }^\circ\text{C}^{-1}$ (22°-500°C)
5651-75-1A-052A	Parallel ↓	End Edge ↓	4.59
-052B			4.38
-054A			4.40
-054B			4.27
-1B-057A			4.28
-057B			4.42
-059A			4.69
-059B			<u>4.56</u>
Mean			4.44
Std. Dev.			0.151
5651-75-1A-081A	Perpendicular ↓	End Edge ↓	5.11
-081B			4.80
-083A			5.26
-083B			4.81
-1B-085A			4.91
-085B			4.86
-087A			4.84
-087B			<u>5.03</u>
Mean			4.95
Std. Dev.			0.166
5651-75-3A-152A	Parallel ↓	Midlength Edge ↓	4.21
-152B			4.31
-154A			4.20
-154B			4.30
-3B-157A			3.94
-157B			4.09
-159A			4.31
-159B			<u>4.12</u>
Mean			4.18
Std. Dev.			0.130
5651-75-3A-181A	Perpendicular ↓	Midlength Edge ↓	4.68
-181B			4.67
-183A			2.58
-183B			4.72
-3B-185A			4.72
-185B			4.73
-187A			4.57
-187B			<u>4.94</u>
Mean			4.70
Std. Dev.			0.114

CAPSULE IRRADIATIONS

Capsule OG-1

A final report (Ref. 11-3) on the graphite irradiation experiment OG-1 has been issued.

Capsule OG-2

Capsule OG-2 was built according to the same design as capsule OG-1 and contains physical property samples of H-451, H-327, H-429, TS-1240, P₃JHAN, 2020, and other miscellaneous graphites. The type and number of samples were reported previously (Ref. 11-1). A complete sample inventory, together with preirradiation measurements, has been prepared.

The capsule was inserted into the C-3 position of the Oak Ridge Reactor on July 15, 1974 and startup took place on July 19. Full-power operation was achieved on July 20. All 31 thermocouples operated normally and indicated temperatures were close to those of the OG-1 capsule. The operating temperatures on August 12, 1974, after 534 full-power hours and a peak fast fluence of 6×10^{21} n/cm² ($E > 0.18$ MeV), are shown in Table 11-5, together with the mean operating temperatures in OG-1 for comparison. The radial temperature gradients in OG-2 are lower than the mean OG-1 values because the neutron fluences in OG-2 are as yet too low to greatly reduce the thermal conductivity of the graphite crucibles. However, the duplication of OG-1 conditions shown in Table 11-5 is excellent.

STATISTICAL MEASUREMENTS OF GRAPHITE STRENGTH

The statistical distribution of strengths and the dependence of strength on specimen size and location in a log are being studied on material from a preproduction log of H-451 graphite. The specimen dimensions and sampling plan were described previously (Ref. 11-2).

TABLE 11-5
OPERATING TEMPERATURES OF CAPSULE OG-2 AFTER 534 FULL-POWER HOURS

Crucible No.	Location in Crucible ^(a)	Design Temperature (°C)	OG-1 Mean Temperature (°C)	OG-2 Operating Temperature, 8/12/74 (°C)
1	C	700	657	690
	P	--	558	620
2	C	800	754	780
	P	--	610	680
3	C	1000	993	1010
	P	--	887	930
4	C	1200	1172	1150
	P	--	1024	1060 ^(b)
5	C	1400	1395	1290
	P	--	1327	1225
6	C	1100	1078	1045
	P	--	938	910
7	C	1000	992	990
	P	--	884	900
8	C	900	928	940
	P	--	836	840
9	C	800	860	830
	P	--	753	760
10	C	600	650	660
	P	--	597	615 ^(b)

(a) C = centerline, P = periphery.

(b) No thermocouple in this location; estimated from adjacent thermocouples.

Tensile tests were performed on 0.25-in.-diameter x 0.90-in.-long specimens from slabs 1A and 1B, close to one end of the log. The specimens were cemented between metal end pieces and tested to failure at room temperature in an Instron testing machine. Tests have been completed on axial and radial tensile samples from the center region of the slab; the strength values are listed in Tables 11-6 and 11-7. The core locations are shown in Fig. 11-9 of Ref. 11-2. Tables 11-6 and 11-7 also show the mean and standard deviation of the measurements within each batch, and the value of the Weibull modulus, m , obtained by fitting the data to the Weibull function, which expresses the probability of failure F at stress σ in terms of the sample volume V and two material parameters, σ_o and m :

$$F = 1 - \exp \left[- \int_V \left(\frac{\sigma}{\sigma_o} \right)^m dV \right] .$$

The tensile strength in the axial direction was consistently higher than in the radial direction (mean strength 2300 psi compared with 1770 psi), and showed no systematic variation with position in the slab. The radial strengths showed some systematic variation with position, with the specimens from the A section of the slab having a significantly lower mean strength than those from the B section. The variation of strength with location is depicted in Fig. 11-1, which shows the specimens with high, low, and about average strengths as a function of position in the slab. The densities of the samples are shown in a similar way in Fig. 11-2. A comparison of Figs. 11-1 and 11-2 shows that the weak radial samples from Section A came from a low-density part of the slab.

TABLE 11-6
TENSILE STRENGTH OF H-451 NEAR-ISOTROPIC GRAPHITE, AXIAL ORIENTATION
(0.25-in. diameter samples from end-center location of slab 1, Log 5651-90)

Section A				Section B			
Core No.	Density (g/cm ³)	Failure Load (lb)	Strength (psi)	Core No.	Density (g/cm ³)	Failure Load (lb)	Strength (psi)
C1A1	Not determined ↓	95	1935	1A1	1.754	102	2078
C1A2		127	2587	1A2	1.760	105	2139
C2A1		93	1895	2A1	1.753	120	2444
C2A2		138	2811	2A2	1.750	122	2485
C3A1		118	2404	3A1	1.742	125	2546
C3A2		98	1996	3A2	1.759	127	2587
C4A1		131	2669	4A1	1.759	98	1996
C4A2		127	2587	4A2	1.753	112	2281
C5A1		117	2383	5A1	1.756	92	1874
C5A2		110	2241	5A2	1.748	87	1772
C6A1		129	2628	6A1	1.752	131	2669
C6A2		108	2200	6A2	1.751	102	2078
C7A1		118	2404	7A1	1.762	84	1711
C7A2		136	2770	7A2	1.749	105	2139
C8A1		119	2424	8A1	1.749	120	2444
C8A2		139	2832	8A2	1.755	148	3015
C9A1		81	1650	9A1	1.754	103	2098
C9A2		117	2384	9A2	1.751	90	1833
C10A1		120	2445	10A1	1.752	118	2404
C10A2		77	1569	10A2	1.755	125	2545
Mean strength = 2341 psi Standard deviation = 359 psi Coefficient of variation = 0.153 Weibull Modulus, m = 6.8				Mean strength = 2257 psi Standard deviation = 332 psi Coefficient of variation = 0.147 Weibull Modulus, m = 7.5			

Joint mean (A and B sections): Mean strength = 2299 psi
Standard deviation = 348 psi
Coefficient of variation = 0.151
Weibull modulus, m = 7.5

TABLE 11-7
TENSILE STRENGTH OF H-451 NEAR-ISOTROPIC GRAPHITE, RADIAL ORIENTATION
(0.25-in.-diameter samples from end-center location of slab 1, Log 5651-90)

Section A				Section B			
Core No.	Density (g/cm ³)	Failure Load (lb)	Strength (psi)	Core No.	Density (g/cm ³)	Failure Load (lb)	Strength (psi)
C300A1	1.715	79	1609	300A1	1.752	84	1711
C301A1	1.737	118	2404	301A1	1.748	105	2139
C302A1	1.729	76	1548	302A1	1.751	83	1691
C303A1	1.726	72	1467	303A1	1.759	87	1772
C304A1	1.730	69	1406	304A1	1.755	90	1833
C305A1	1.725	38 (a)	774 (a)	305A1	1.754	112	2282
C306A1	1.711	71	1446	306A1	1.742	64	1304
C307A1	1.736	67	1365	307A1	1.737	106	2159
C308A1	1.720	74	1508	308A1	1.727	112.5	2292
C309A1	1.728	78	1589	309A1	1.732	103	2098
C310A1	1.735	80	1630	310A1	1.756	111	2261
C311A1	1.737	102	2078	311A1	1.732	93	1895
C312A1	1.742	83	1691	312A1	1.749	98	1996
C313A1	1.742	71	1446	313A1	1.744	100	2037
C314A1	1.746	82	1670	314A1	1.750	67	1365
C315A1	1.749	85	1732	315A1	1.746	78	1589
C316A1	(b)	(b)	(b)	316A1	1.744	89	1813
C317A1	1.742	83	1691	317A1	1.745	87	1772
C318A1	1.743	62	1263	318A1	1.745	109	2221
C319A1	1.743	57	1161	319A1	1.740	109	2221
Mean Strength = 1595 psi Standard deviation = 278 psi Coefficient of variation = 0.174 Weibull modulus, m = 6.4				Mean strength = 1922 Standard deviation = 297 psi Coefficient of variation = 0.155 Weibull modulus, m = 7.0			
Joint means (A and B sections): Mean strength = 1767 psi Standard deviation = 328 psi Coefficient of variation = 0.186 Weibull modulus, m = 6.2							

(a) Sample flawed - excluded from average.

(b) Sample broken during machining.

Fig. 11-1. Variation of strength of 0.25-in.-diameter tensile samples with location in central section of slab 1, log 5651-90, H-451 graphite

Fig. 11-2. Variation of density of 0.25-in.-diameter tensile samples with location in central section of slab 1, log 5651-90, H-451 graphite

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- 11-1. "HTGR Base Program Quarterly Progress Report for the Period Ending February 28, 1974," USAEC Report GA-A12916, General Atomic Company, March 29, 1974.
- 11-2. "HTGR Base Program Quarterly Progress Report for the Period Ending May 31, 1974," USAEC Report GA-A13030, General Atomic Company, June 28, 1974.
- 11-3. Price, R. J., and L. A. Beavan, "Final Report on Graphite Irradiation Test OG-1," USAEC Report GA-A13089, General Atomic Company, August 1, 1974.

APPENDIX

PROJECT REPORTS PUBLISHED DURING THE QUARTER

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