HTGR FUELS AND CORE DEVELOPMENT PROGRAM

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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 early 1980's following the operation of the 330-MW(e) prototype in 1975. 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 (189a 13114)
FISSION PRODUCT MECHANISMS

HTGR CHEMISTRY PROGRAM PLAN

A program plan for development of a base technology to describe fission product behavior and coolant impurity/core materials interactions is being prepared. The primary objectives of the program are (1) to provide adequate calculational methods and input data for the calculation of circulating and plateout activities, (2) to provide adequate data for characterizing coolant impurity/core materials interactions in the primary circuit of HTGR systems, and (3) to validate the calculational methods and input data by comparison of calculated and observed in-pile behavior. The work generally progresses from laboratory tests through validation of codes, data, and design methods by means of large-scale in-pile tests. The program is directed toward the development of adequate fission-product-related design methods to support the final safety analysis report for a large HTGR by 1978. The program plan covers chemistry work to be done at General Atomic Company and at Oak Ridge National Laboratory (ORNL).

The first draft of the program plan, entitled "Program Plan for Work to Establish Fission Product Behavior in HTGR Systems" and dated July 12, 1974, is being revised to include (1) HTGR chemistry work at ORNL, (2) work on the reaction of steam with graphite, and (3) the Peach Bottom Fission Product Surveillance Program. The schedule for completion of the plan is (1) submittal of the revised plan to RRD by December 31, 1974, (2) comments from RRD and ORNL by February 28, 1975, and (3) a target date of May 1, 1975 for the final issue to be in reproduction.

FISSION METAL RELEASE STUDIES

Postirradiation anneal measurements have been performed on fuel particles by the method described in an earlier Quarterly Progress Report (Ref. 4-1) to obtain reduced diffusion coefficients for strontium in fuel kernels and diffusion coefficients for strontium in the pyrocarbon coating of the fuel particles. In these measurements BISO coated particles and one sample of triplex-coated particles were used; the triplex-coated particles had successive coatings of an inner buffer (low density) pyrocarbon, an isotropic pyrocarbon, and an outer granular pyrocarbon. These particles were given a brief reirradiation in the TRIGA facility to form the short-lived strontium-91 isotope; the release of strontium from the particles during the anneal could then be conveniently monitored by gamma counting. For the anneal measurements the particles were placed in graphite crucibles, which were heated in tantalum tubes. Periodically the strontium-91, which collected in the graphite crucible, was counted. Diffusion coefficients were derived from the time profiles of the fractional release of strontium. These coefficients were combined with other data (from Ref. 4-2) to obtain the representations given in Table 4-6 (see section on Fission Product Data Review).

Kernel Diffusion Coefficients for Strontium

The derivation of strontium diffusion coefficients was made on the basis of the idealized release curve illustrated in Fig. 4-1a. For this curve, there is an initial time delay in the observed release followed by a sharp rise to a time region represented by a constant slope. The constant slope corresponds to the steady-state release of the strontium nuclide controlled by diffusion in the kernel. The constant-slope portion of the release curve was used in the following equation to obtain the reduced diffusion coefficient of strontium in the fuel kernel:

$$D' = \frac{D}{a^2} = \frac{\pi(f - f_o)^2}{36t} \quad , \quad (4-1)$$

(a)

(b)

Fig. 4-1. Fission product release profile

where D' = the reduced diffusion coefficient (sec^{-1}),
 D = the diffusion coefficient (cm^2/sec),
 a = the radius of the equivalent sphere of the particle or grain
 through which diffusion occurs (cm),
 f = the fractional release at time t , and
 f_0 = the extrapolated fractional release at zero time.

At zero time, f_0 can be non-zero; this can result from fission products that have recoiled from the fuel kernel, were released prior to the anneal, or were formed from contamination in the particle coatings. Equation 4-1 is valid for $f_0 \leq 0.15$. The derived reduced diffusion coefficients are given in Table 4-1. For the kernels used in the postirradiation anneals, the value of a was equal to or less than approximately 0.001 cm.

Pyrocarbon Diffusion Coefficients for Strontium

The pyrocarbon diffusion coefficients for strontium were derived from the postirradiation anneal measurements using the relation

$$D = \ell^2 / 6\tau \quad , \quad (4-2)$$

where ℓ is the coating thickness in centimeters and τ is the delay time in seconds for the appearance of the diffusant at the outer boundary of the coating. The delay time τ is the value of the intercept on the abscissa of the idealized release curve illustrated in Fig. 4-1b and is obtained by extrapolation of the initial linear portion of the release curve. The analysis of the postirradiation anneal measurements is based on the assumption that the pyrocarbon coating is initially free of diffusant. This assumption requires the irradiation of the particle to be performed at a temperature sufficiently low to prevent all but negligible diffusion of the fission product into the pyrocarbon coating. This requirement is met by the fact that the diffusant, strontium-91, was formed only after the major irradiation period. The derived diffusion coefficients are given in Table 4-2.

TABLE 4-1
 KERNEL DIFFUSION COEFFICIENTS FOR STRONTIUM CALCULATED FROM
 ANNEALING RELEASE CURVES FOR BISO COATED PARTICLES

Sample Ident.	Particle Characteristics		Irradiation Conditions		Anneal Temp. (°C)	Reduced ^(a) Diffusion Coefficient (sec ⁻¹)
	Fuel Type	Kernel Diameter (μm)	Irrad. Temp. (°C)	FIMA (%)		
2702-57E ^(b)	(Th,U)O ₂	240 - 300	1165	3.0	1400	1.2 x 10 ⁻⁹
2702-57E	(Th,U)O ₂	240 - 300	1165	3.0	1550	8.9 x 10 ⁻⁹
2702-57E	(Th,U)O ₂	240 - 300	1165	3.0	1650	3.0 x 10 ⁻⁷
3023-99E	(Th,U)O ₂	300 - 420	1600	2.5	1400	2.9 x 10 ⁻¹⁰
3023-99E	(Th,U)O ₂	300 - 420	1600	2.5	1550	6.8 x 10 ⁻⁹
3222-71E	(Th,U)C ₂	150 - 250	--	0	1400	3.9 x 10 ⁻⁹
3222-71E	(Th,U)C ₂	150 - 250	685	5.7	1400	1.3 x 10 ⁻⁷
3222-71E	(Th,U)C ₂	150 - 250	950	5.6	1400	2.3 x 10 ⁻⁷
3222-105E	(Th,U)C ₂	300 - 420	--	0	1400	4.6 x 10 ⁻⁹
3222-105E	(Th,U)C ₂	300 - 420	950	1.5	1550	4.9 x 10 ⁻⁸
3363-117E	(Th,U)C ₂	300 - 420	1310	5.7	1400	2.6 x 10 ⁻⁸
3363-117E	(Th,U)C ₂	300 - 420	1310	5.7	1550	3.0 x 10 ⁻⁸
4000-632E	(Th,U)C ₂	150 - 250	--	0	1400	5.3 x 10 ⁻⁸
4000-660E	(Th,U)C ₂	150 - 250	975	13	1400	3.4 x 10 ⁻⁸
4000-695E	(Th,U)C ₂	150 - 250	900	0	1400	8.9 x 10 ⁻¹⁰

(a) For strontium in fuel kernel.

(b) Sample 2702-57E consists of triplex-coated particles.

TABLE 4-2
 DIFFUSION COEFFICIENTS FOR STRONTIUM IN PYROCARBON
 (POSTIRRADIATION ANNEAL MEASUREMENTS USING BISO COATED PARTICLES)

Coating Type	Coating Thickness (μm)	Kernel Type	Irrad. Temp. ($^{\circ}\text{C}$)	FIMA (%)	Anneal Temp. ($^{\circ}\text{C}$)	cm^2/sec
LTI	57	(Th,U)C ₂	890	2.3	1200	3.0×10^{-8}
LTI	58	(Th,U)C ₂	975	1.3	1400	3.0×10^{-8}
LTI	58	(Th,U)C ₂	975	1.3	1550	9.0×10^{-9}
HTI	68	(Th,U)C ₂	685	5.7	1400	2.7×10^{-8}
HTI	68	(Th,U)C ₂	685	5.7	1500	5.2×10^{-8}
HTI	68	(Th,U)C ₂	685	5.7	1650	1.3×10^{-7}
HTI	98	(Th,U)C ₂	950	1.5	1400	4.4×10^{-9}
HTI	98	(Th,U)C ₂	950	1.5	1550	5.3×10^{-8}
HTI	68	(Th,U)C ₂	900	2.3	1200	6.1×10^{-10}
HTI	68	(Th,U)C ₂	900	2.3	1400	2.0×10^{-8}
HTI	68	(Th,U)C ₂	900	2.3	1550	4.8×10^{-8}

VAPOR PRESSURE STUDIES

Further vaporization studies to obtain the vapor pressure of strontium over near-isotropic graphite (H-451) have been completed. The vaporization of strontium from strontium-impregnated graphite, having a particle size range of 44 to 74 μm , was measured using a molybdenum Knudsen cell in the mass spectrometer. The temperature range of these experiments was between 1100° and 1900°K.

The isotherm data obtained from these studies for 1300°, 1500°, and 1800°K are shown in Fig. 4-2, along with isotherm curves representing the results of similar experiments in which H-327 graphite was used. Within the errors of the experiments, there is no significant difference between the isotherms derived from observations for H-451 graphite and those found for H-327 graphite. Consequently, current reference data that are based on the isotherms of strontium sorbed on H-327 graphite are adequate for describing the vapor pressure of strontium over the reference graphite.

The data in Fig. 4-2 show a significant deviation at about 0.055 mg strontium/g carbon. This deviation is attributed to an interruption in the experiment at this point, which probably caused a change in the sensitivity of the mass spectrometer.

TRANSPORT CODE DEVELOPMENT

The basis and structure of the TRAFIC code for calculating the release of metallic fission products into the primary coolant circuit of HTGR systems were described in the previous Quarterly Progress Report (Ref. 4-2). During the current report period, efforts were concentrated on verification of the code and minor model improvements.

Fig. 4-2. Isotherm data for H-451 and H-327 graphites

Verification of TRAFIC Code

Verification of the TRAFIC code to date has consisted of a detailed comparison of the submodel calculations for relatively simple test problems with the known analytical solutions of these problems. Integral tests, i.e., verification tests employing the full code, have yet to be performed. Verification of the TRAFIC code by means of the test problems, as presented in this section, has revealed no errors in TRAFIC programming.

Graphite Region Solution

Both the transient and the steady-state solutions for the graphite region have been checked. Because of the similarity between heat conduction and mass transport without thermal diffusion and decay, i.e., $\beta = 0$ and $\gamma = 0$, the heat conduction solutions in Ref. 4-3 (Carslaw and Jaeger) were used to check the transient solution. The steady-state solution was checked by comparison with an independently derived, steady-state analytical solution.

It was found in a careful review of Carslaw and Jaeger's book that there is only one case which can be used to check the results of the TRAFIC code. This case is one with the boundary conditions defined as constant temperature on one side and insulated on the other side (e.g., $T(0) = T_0$, $H(L) = 0$). The equivalent problem in diffusion is a problem with constant concentration on one side and no mass flux on the other side (e.g., $C(0) = C_0$, $J(L) = 0$). A comparison of these two cases is shown in Table 4-3.

It can be seen that the deviation is quite large for the short-time solution. It was determined by hand calculation that this discrepancy is due to a peculiarity of this particular test problem and that this peculiarity is of such a nature that it will not affect an actual TRAFIC run. The problem is that the constant concentration C_0 is applied as a step function at time zero. The numerical approximation to the Duhamel integration requires that non-zero initial boundary conditions (if any) be considered as an "instantaneous response" term of the form

TABLE 4-3
COMPARISON OF HEAT CONDUCTION AND DIFFUSION SOLUTIONS

Effective Time $\rho = \int_0^t \frac{D(\text{or } K)}{L^2} d\tau$	Diffusion $C(L)/C_o$ (TRAFIC)	Heat Conduction $T(L)/T_o$ (Carslaw and Jaeger)	Percentage Difference (%)
0.1	0.0231	0.05	53.8
0.2	0.184	0.23	20.0
0.4	0.497	0.52	4.4
0.6	0.695	0.69	0.7
0.8	0.808	0.82	1.5
1.0	0.891	0.89	0.1
1.5	0.967	0.97	0.3

$$C_o(o) H(\rho_n, \beta, \gamma) \quad , \quad (4-3)$$

where ρ is a "reduced time," β is a thermal diffusion parameter, and γ is a radioactive decay parameter. The rest of the integration, however, contains terms of the form

$$\frac{1}{2} \left[H(\rho_n - \rho_i) + H(\rho_n - \rho_{i+1}) \right] (C_{o_{i+1}} - C_{o_i}) \quad , \quad (4-4)$$

where $i = 1, n-1$.

It has been determined that the most accurate evaluation of term 4-4 is accomplished by integrating by parts, in which case one needs to manipulate the integrated solution functions. The typical term in the summation then becomes

$$\left[H_T(\rho_n - \rho_i) - H_T(\rho_n - \rho_{i+1}) \right] \frac{C_{o_{i+1}} - C_{o_i}}{\rho_{i+1} - \rho_i} \quad , \quad (4-5)$$

where $i = 1, n-1$ and

$$H_T(\rho) = \int H(\rho) d\rho \quad .$$

In the actual TRAFIC application, it was known that the initial concentration is always zero; therefore, term 4-3 is not required. Thus, only the integrated functions are stored and used for TRAFIC calculation. As originally programmed, TRAFIC could not handle non-zero initial conditions. In an effort to reproduce this test result, the program was modified to include a term of type 4-3. However, since the function itself was not known, it was calculated by a numerical differentiation of the integral, i.e.,

$$H(\rho_n) = \frac{H_T(\rho_n) - H_T(\rho_{n-1})}{\rho_n - \rho_{n-1}} \quad . \quad (4-6)$$

This calculation is very inaccurate at small values of ρ .

A hand calculation was performed in which the equations were solved using the correct value of the function, obtained from an early DIFFEQ run. The results at $\rho = 0.1$ were

$$\frac{C(L)}{C_0} = 0.05 \quad ,$$

which is in perfect agreement with the exact solution. This proves two things. First, it proves that all the discrepancy was in the approximation of term 4-3 by term 4-6. Second, since term 4-3 is inconsequential in the actual TRAFIC application, it proves that the TRAFIC transient solution is correct for $\beta = 0$, $\gamma = 0$.

Due to the lack of existing transient heat conduction solutions for the other cases, i.e., different combinations of β , γ , C_0 , and $J_L L/D$, only the long-term solution (steady-state solution) can be compared with known analytical solutions. The analytical steady-state solutions for arbitrary β , γ , C_0 , and $J_L L/D$ are

$$C_L = K_3 [\exp(-K_2) - \exp(K_1)] J_L L/D \\ + \{ \exp(K_1) + [\exp(-K_2) - \exp(K_1)] K_2 K_3 \exp(K_1) \} C_0 \quad , \quad (4-7)$$

and

$$\frac{J_0 L}{D} = K_3 (K_1 + K_2) \frac{J_L L}{D} \\ + [-K_2 + (K_2 + K_1) K_2 K_3 \exp(K_1)] C_0 \quad , \quad \dots \quad (4-8)$$

$$\text{where } K_1 = \frac{-\beta + \sqrt{\beta^2 + 4\gamma}}{2}, \quad (4-9)$$

$$K_2 = \frac{\beta + \sqrt{\beta^2 + 4\gamma}}{2}, \quad (4-10)$$

$$K_3 = \frac{1}{K_1 \exp(-K_2) + K_2 \exp(K_1)} \quad (4-11)$$

Several simplified cases that have been investigated are:

1. $\beta = 0, \gamma = 0, C_o = 0$, and $\frac{J_L L}{D} = 1$.
2. $\beta = -3, \gamma = 0, C_o = 1$, and $\frac{J_L L}{D} = 0$.
3. $\beta = -3, \gamma = 0, C_o = 0$, and $\frac{J_L L}{D} = 1$.
4. $\beta = 0, \gamma = 500, C_o = 1$, and $\frac{J_L L}{D} = 0$.
5. $\beta = 0, \gamma = 500, C_o = 0$, and $\frac{J_L L}{D} = 1$.

A sixth test case (6A and 6B) consisted of a normal TRAFIC run using the power and temperature histories from a typical TREVER calculation. The boundary conditions used to evaluate Eq. 4-7 and Eq. 4-8 were the end-of-life values. This is a non-isothermal case with effective $\beta = -0.1989$ and $\gamma = 0.00299$. The comparison is shown in Table 4-4.

The agreement in cases 1 through 5 is very good, as it should be for an isothermal problem with a long effective time. The discrepancy in case 6A is due to the fact that the transient influence of the past history of temperature and boundary conditions has not died out. In fact, any agreement

TABLE 4-4
COMPARISON OF C_L AND $J_O L/D$

	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6A	Case 6B (Quasi-Steady)
C_L							
TRAFFIC	-1.0	20.1	-6.36	0.391×10^{-9}	-0.0447	0.107	0.108
Eq. 4-7	-1.0	20.1	-6.36	0.388×10^{-9}	-0.0447	0.118	0.105
$J_O L/D$							
TRAFFIC	1.0	0	1.0	22.4	0.391×10^{-9}	1.050	1.043
Eq. 4-8	1.0	0	1.0	22.4	0.388×10^{-9}	1.029	1.041

is simply a proof that the quasi-steady assumption was reasonable in this case. The small discrepancy in case 6B, using the TRAFIC quasi-steady option, is due to the fact that the VECTIT iteration did not converge on the exact solution. The reason it did not converge is that the prior time step (the basis for the initial guess) was at a different temperature with different boundary conditions. Moreover, the convergence tolerance is defined as the ratio of magnitude of the vectors and it allows errors of up to 1%.

Fuel Rod Solution

The general solution for this region is

$$C_o = \frac{r_f^2 \psi}{\phi[1 - \exp(-\psi r_f^2)]} e^{-\lambda t} \int_0^t \left(S - \frac{2}{r_f} \frac{A_c}{A_f} J_o \right) e^{\lambda \tau} d\tau, \quad (4-12)$$

where

$$\psi = \frac{4Q(T_c - T_s)}{BR(T_c + T_s)^2 r_f^2}$$

and

$$\phi = \text{partition factor} = \frac{\text{fuel rod surface concentration}}{\text{graphite concentration on fuel side}}.$$

Two special cases were used to verify the calculations for this region in TRAFIC code:

Case 1: The surface of the fuel rod is insulated and the birth rate is kept constant, i.e., $J_o = 0$ and $S(t) = S^* = \text{constant}$. Assuming $\phi = 1$, Eq. 4-5 becomes

$$C_o(t) = \frac{r_f^2 \psi}{1 - \exp(-\psi r_f^2)} \frac{S^*}{\lambda} (1 - e^{-\lambda t}) \quad (4-13)$$

The comparison is as follows:

ρ	t (10^3 sec)	C_o (Eq. 4-13)	C_o (TRAFIC)
0.05	1.493	0.7386	0.738
0.5	14.93	7.382	7.38
1.0	28.86	14.756	14.7
1.5	44.79	22.122	22.1

Case 2: The outgoing flow rate from the fuel rod surface is equal to the birth rate inside the fuel rod, i.e.,

$$S = \frac{2}{r_f} \frac{A_c}{A_f} J_o$$

By Eq. 4-12, C_o should always be equal to zero. Within the acceptable roundoff error, the TRAFIC code does give $C_o = 0$.

Partition Factor

The partition factor ϕ is defined as

$$\phi = \frac{\text{Surface concentration of fuel rod}}{\text{Graphite concentration on fuel side}} = \frac{C_s}{C_o}$$

This factor governs the transport of fission products from the fuel rod to the graphite web through the gap between them. The fission product first evaporates to the gap and is then adsorbed by the graphite. Depending on the level of concentration, Freundlich or Langmuir sorptive isotherms are used to calculate the vapor pressure of the fission product in the gap and the ϕ factor as follows:

$$\text{Freundlich} \quad p = e^V \left(\frac{C}{\rho} \right)^U \quad \text{if } C > C_{\text{TRAN}} \quad , \quad (4-14)$$

$$\text{Langmuir} \quad p = e^V \left(\frac{C_{\text{TRAN}}}{\rho} \right)^{U-1} \left(\frac{C}{\rho} \right) \quad \text{if } C < C_{\text{TRAN}} \quad , \quad (4-15)$$

where p = vapor pressure (atm),
 C = concentration of fission product inside the medium ($\mu\text{moles/cm}^3$),
 ρ = density of the medium (g/cm^3),
 $V = a + b \left(\frac{10^3}{T} \right)$,
 $U = c + d \left(\frac{10^3}{T} \right)$,
 C_{TRAN} = transition concentration ($\mu\text{moles/cm}^3$),
 T = absolute temperature ($^{\circ}\text{K}$), and
 a, b, c, d = material constants.

Since U and V depend on temperature, ϕ is not a constant for a non-isothermal case.

The value of ϕ was calculated by hand with the aid of Eq. 4-15. The hand calculation result was identical to the TRAFIC result.

Film Mass Transfer

The fission product is released from the graphite web to the coolant main stream through a thin turbulent boundary layer. This is a convective mass transport process that involves both mass transport and momentum transport. Due to the complications of the governing equations, no analytical solution is available. Therefore, the flux across the film is defined as

$$J_L = H(C_a - C_{\infty}) \quad , \quad (4-16)$$

where H is a mass transfer coefficient that is determined by the standard semi-empirical formula

$$H = 0.023 D_c Re^{0.83} Sc^{0.44}/d, \quad (4-17)$$

where Re = Reynolds number of the coolant flow = $(48/\pi d)(\dot{m}/\mu)$,
 Sc = Schmidt number,
 d = coolant hole diameter, and
 D_c = coefficient of diffusion in the coolant given by

$$D_c = \frac{D_{co}}{P_c} \left(\frac{T + 273}{273} \right)^{1.65}, \quad (4-18)$$

where D_{co} = coefficient of diffusion in coolant at standard conditions
 (cm^2/sec),

P_c = pressure of the coolant (atm), and

T = temperature of the coolant ($^{\circ}\text{C}$).

The values of H and D_c were calculated by hand using the above formulae. The results agreed identically with the TRAFIC results.

Combined Model

The previous sections have shown how the calculation of the individual submodels in the TRAFIC code has been verified. The fact that the combined multiregion model works as intended has been demonstrated by the following:

1. The iterative solution procedure converges.
2. The converged solution satisfies all the mathematical conditions imposed by the 5 by 5 system of boundary condition coupling equations.
3. The converged solution is physically reasonable.

It is noted that all the above submodel tests were performed using the combined model as finally programmed. Any deviation from the normal program running condition, such as specially imposed boundary conditions, was accomplished by selectively deleting or bypassing the normal calculation. This demonstrates that the submodels are behaving correctly in the context of the combined model.

Minor Model Improvements

Several minor model improvements in the TRAFIC code have been made. The running time of the TRAFIC code was reduced to about 1.6 sec per space point by experimentation with iteration convergence criteria and accuracy requirements.

An interim version of TRAFIC that is capable of processing data tapes containing power, temperature, and fuel particle failure histories for a number of space points was developed. This is the first stage in preparing the code for prediction of total reactor release.

As a result of running many different space points with the above tape processing version, it was discovered that the nonlinear boundary condition iterations failed to converge to the correct solution in some instances. The reason has been traced to either a bad first guess, with which the Newton-Raphson technique fails, or occasionally a time step that was too long. An effort is now under way to improve the numerical stability by modifying the initial guess and/or subdividing the time steps in special instances. Results to date have been encouraging.

FISSION PRODUCT TRANSPORT CODE VALIDATION PROGRAM

A program plan for verifying HTGR design methods related to fission product transport out of the fuel and around the primary coolant circuit has been drafted. Justification for this program is based on two overriding factors: (1) federal regulations requiring such a program, and (2) the need

to verify the validity of summing individual transport barriers in the predictions of integral fission product release.

Federal Regulation 10 CFR 50, Appendix B, requires that the adequacy of design be verified. Quality Assurance requirement ANSI N45.211-1974 defines the extent of design verification as being dependent upon the safety of the item under consideration, the state of the art, and the similarity with previously proven design. Since radionuclide migration characteristics are very complex, a testing program must be undertaken to prove the adequacy of design methods. Design cannot be verified by alternate or simplified calculational methods alone since some of the basic assumptions used to model transport have not been completely verified.

Large-scale "integral" experiments such as fuel irradiation capsules, fuel test elements, in-pile loops, and operating reactors which present multiple barriers to fission product release are required to validate design codes. Laboratory "differential" tests generally supply information on a single barrier to fission product movement. Incremental effects obtained from differential tests must be summed, or integrated, to yield the desired quantity, the amount of radionuclide migration. Computer codes perform this function. Most code validation to date has been based upon integration of laboratory differential tests.

Two concerns arise from relying on differential data for design information. First, the combined effect of the reactor environment and the various fission product migration processes is not adequately known. Second, because of the need to be conservative in calculating fission product release, conservatively high input data and transport mechanisms are chosen, which may result in grossly overpredicted radionuclide release from the core. At a minimum, integral experiments provide a calibration, if not a validation, of the design methods. By comparing predicted and measured release quantities, pessimistic or optimistic factors can be factored into code predictions. In addition, integral tests allow the measurement of end quantities that result from the interaction of numerous pathways of release, thus reducing

the possibility of neglecting a barrier to release of importance in actual reactor operation. Finally, integral tests take into account those physically dependent phenomena that frequently cannot be considered in laboratory experiments but which are sometimes very important. Laboratory measurements are a prerequisite to understanding integral tests and will continue to be required in the testing of isolated assumptions or for the collection of basic input data for controlled experimental conditions. In addition, laboratory measurements will be increasingly related directly to interpreting integral tests. Integral measurements will be designed to verify the validity of summing laboratory test data.

Three design method computer codes [RAD, FIPER (TRAFIC), and PAD], which model radioactive nuclide transport in the primary coolant circuit, will be verified in large-scale tests. The RAD code is used to estimate fission product inventories in the fuel elements, in the primary coolant, and on surfaces exposed to the primary coolant. RAD predictions should be verified in a large-scale test such as an operating reactor. A great many small-scale tests (e.g., gaseous release from fuel rods and gross capsule measurements) have been conducted to determine release fractions from the fuel. However, these data are used as input to the code and do not confirm the code output or HTGR predicted inventories.

The FIPER (TRAFIC) code is used to calculate the release of metallic fission products from the HTGR core. Efforts to date, as for other codes, have centered primarily on providing accurate assumptions for modeling and accurate input data through laboratory experiments. The predecessor to FIPER, FREVAP, was used to test predicted release against observed values in the PLUTO-8 and -15 loops and the GAIL loop (Ref. 4-4). Metallic fission product migration out of the fuel is among the less understood fission product transport phenomena. Although most metallic species follow well-prescribed characteristics, certain very important metallic elements, such as cesium, are not adequately represented by the fundamental assumption used in the FIPER code, Fickian migration. Laboratory measurements increasingly indicate existence of fast paths of release. Thus, one of the objectives of this testing program is to obtain calibration factors, if not verification of

fundamental assumptions, for these metallic elements from comparison of predicted and observed behavior.

The PAD code calculates the distribution of radioactivity in the primary coolant circuit. The PAD code has been tested against plateout data from deposition loops at General Atomic and Batelle Memorial Institute, Dragon reactor steam generator measurements, and operational measurements from the Peach Bottom HTGR. This verification effort has been moderately successful but has not definitively verified the code. None of the tests contained relevant HTGR materials or operated at HTGR conditions except Peach Bottom. Peach Bottom data are not precise because of the inaccessibility to primary circuit components of the operating reactor. The understanding of transport in the primary coolant circuit is primitive since studies on this began much later and were funded at a much lower level than studies of transport through the fuel.

Table 4-5 lists some of the tests proposed for validation of fission product transport codes. Data from Peach Bottom will be used to validate all three transport codes. Considerable fission product transport data have been obtained by making operational measurements throughout reactor operation. These measurements, which have been carried out by both GA and ORNL, include: (1) fission gas release, (2) diffusion probes, (3) tritium surveys, and (4) external gamma surveys of accessible ducts. In addition, several Core 1 and Core 2 driver fuel elements have been examined at GA and ORNL. These investigations have already yielded valuable fuel performance and fission product transport data and should provide even more valuable insight when coupled with the results of the anticipated EOL program. In terms of transport data for code validations, the most important features of the program are: (1) detailed postirradiation examination (PIE) of five driver elements at ORNL, and (2) determination of plateout distribution (hence total core release) by gamma scanning and by radiochemistry of specimens removed from the steam generator and ducts. Using updated time-temperature-power histories, code calculations can be performed to predict

TABLE 4-5
DESIGN METHOD VALIDATION

	Responsible Group	Code Validated
Reactor		
Peach Bottom	GA-ORNL	RAD, PAD, FIPER
Fort St. Vrain	GA	RAD, PAD, FIPER
Capsules		
F-30, P13R, P13S, P13Q, P13T	GA	RAD, FIPER
Test Elements		
FTE-7, -8, -9, -10, -14, -15, -17	GA	RAD
FTE-16	GA	FIPER

the amount and distribution of particle failure for comparison with PIE-determined failure fractions and to provide input for fission gas and metallic release calculations. In turn, release predictions can be compared with core-average gas release measurements and with circuit inventories (from integrated plateout distributions). Finally, PAD code calculations will be made in order to compare predicted and measured deposition profiles.

Initial full-power operation of FSV offers the best opportunity of all validation tests to verify the gaseous fission product release code (RAD) because all types of fuel in FSV have been well characterized by extensive preirradiation fission gas release studies on fabricated fuel rods. Since the uncertainty associated with the failed fuel particle distribution will be low during initial operation, gaseous releases can be predicted with calculated temperature and power distributions. In addition to RAD code validation, data from the fast gas sampler and iodine monitor will allow verification of two important transport code assumptions: the dependence of fission gas diffusion on the square root of the nuclide half-life and

the analogous release behavior of halogens and noble gases. Plateout probes will enable limited FIPER code validation via sampling of the circulating gases and also PAD code validation via the use of samplers at the entrance and exit of the steam generators.

A series of large HTGR fuel test capsules have been and are being irradiated to study fuel particle and fuel rod performance characteristics. However, in addition to these objectives, capsules also offer an opportunity to verify fission product transport design methods. Through critical analyses of pretest, in-pile, and PIE measurements, predicted fission product migration behavior can be compared to measured values for capsules exposed to a variety of operating conditions (temperature, fast fluence, burnup). Capsules F-30, P13R, and P13S, by virtue of their purged cell design, can be effectively utilized to validate the RAD code. In-pile R/B measurements from each purged cell of the capsules, in conjunction with temperature and fluence measurements, pretest and PIE R/B values, and failed fraction measurements, will be utilized in the comparison of RAD code predictions with observed R/B measurements. Capsules P13Q and P13T, by virtue of their purged capsule design and large HTGR geometry, can be employed to validate the RAD and FIPER codes. Fission gas release data, in conjunction with temperatures and fuel particle failure distribution, will be analyzed for RAD code validation. FIPER code validation will require measurement of mobile fission metal inventories, metallic profiles in graphite, sorption ratios, etc.

A series of 33 fuel test elements (FTEs) have been irradiated in the Peach Bottom reactor. These FTEs were designed to study the irradiation behavior of a wide variety of fuel designs, including FSV reference fuel, candidate large HTGR fuel, and cured-in-place fuel rods. In addition, eight FTEs were fitted with center spine samples utilized to study the irradiation behavior of selected graphite specimens, loose particle beds, mobile fission metals, etc. Data from purge-sampled FTEs will be used to aid validation of the RAD code. In-pile measurements yield the average R/B of all fuel in the element. By entering pretest R/Bs, temperature, fuel particle coating

failure, and flux distributions in the RAD code, a comparison of calculated and measured results can be carried out. FTE-16 will be examined to aid validation of the FIPER code. The PIE measurements of inventory, sorption ratios, profiles, etc., in conjunction with in-pile operating data, can be compared with predicted release characteristics for code verification.

It should be emphasized that all of the validation tests mentioned will aid in verification of input data to the code as well as total transport code validation. Other tests will be considered if it is felt they can contribute to validation of the input data and codes.

One objective of this program is to validate fission product transport design methods in time to be used in support of reactor licensing by 1978. Because of this requirement, only analyses of existing tests have been included in this program. An additional objective of this program is the broadening of the capsule and fuel test element postirradiation examinations to include fission product measurements and analyses. This validation program is meant to provide a systematic approach to analysis of data that will become available and to provide a well-defined, useful, and acceptable goal for fission product studies.

ANALYSIS OF TRANSPORT MECHANISMS

A simplified, yet realistic method of obtaining estimates of the fission product release into the primary circuit of HTGR systems is being developed. In this approach, idealized and simplified models are used to obtain a relatively easy evaluation of the effect of various factors on the end-of-life (EOL) release. The use of approximation is emphasized provided that the uncertainties thereby introduced can be evaluated. Whenever possible, however, the same parameters (i.e., dimensions, sorptivity relations, diffusion coefficients, etc.) as found in codes such as FIPER and TRAFIC are used. The present approach is a supplement to the latter codes but has different objectives and uses different techniques. At present, attention is restricted to cesium release.

The principal simplifications used include:

1. The temperature of a fuel element (block) is assumed to remain constant throughout core life although a core is composed of such blocks having different temperatures. This is termed a "mono-thermal" core.
2. The "liberation" of cesium from fuel particles is assumed to follow simple kinetics, such as a constant rate L or a rate proportional to time (Lt) or to the square of time (Lt^2).
3. Diffusion within the fuel rod is rapid compared to the other processes (as in the other codes).

In the past transport through graphite was assumed to be Fickian, i.e., governed by a single diffusion coefficient D . This meant that the rate at which the graphite was loaded with cesium was coupled to both the steady rate at which cesium permeated through the graphite and to the time lag at the beginning of the permeation. More recently this restriction was relaxed in an attempt to study the effect of the state of affairs suggested by experimental studies of cesium transport in graphite. The new assumptions used are:

1. The rate of permeation is governed by a permeation coefficient π , which may or may not be equal to D .
2. The time lag of permeation is negligible ("fast path").
3. The rate of loading of the graphite is governed by another coefficient σ , which may or may not be equal to π or D .

This can be termed a "decoupled" model since it differs from the Fickian model mainly in that the loading rate and the permeation rates are not related to each other. If both π and σ are assumed equal to D , the difference

from the Fickian model lies only in the neglect of the time lag for permeation, which has only a minor effect on the EOL release.

It was found that the decoupled model can be related to Fickian models under extreme conditions, and a smooth interpolation between them is possible. At sufficiently high temperatures when σ is large enough, the rate of graphite loading is rapid enough to consider the loading always at equilibrium and a standard Fickian model with $D = \pi$ is equivalent to the coupled model, if the time lag is explicitly neglected in the calculations. At low enough temperatures, σ is so small that any loading of the graphite is negligible and a Fickian calculation (omitting time lag) with $D = \pi$ is again valid, provided the sorptivity of the graphite is set at zero, but only in calculating the graphite holding capacity. This last condition indicates that at lower temperature the result is greatly dependent on whether the graphite holding capacity is large or small compared to that of the fuel rod, i.e., on the value of the sorption ratio Φ (ratio of the concentration of sorbent in the fuel rod matrix to that in the graphite).

Thus, π determines the behavior at high and at low temperature and σ determines what temperatures can be considered as "high" and "low" and where the transition between the two regimes lies.

As expected, the computations showed that if both π and σ are set equal to D , the release of a decoupled core is not very different from that of a Fickian one with the difference being larger when Φ is smaller. A tenfold variation in σ in a decoupled system model had a rather minor effect on release, this effect again increasing at smaller Φ . The effect of a similar change in π was dramatic, the change in release exceeding a factor of 5 at low releases where saturation effects are small.

In terms of distribution of the release among different temperature zones of the monothermal core, decoupling produced a shift of importance toward lower temperatures, especially at lower Φ and higher π .

FISSION PRODUCT DATA REVIEW

Efforts on the fission product data review, as outlined in the previous Quarterly Progress Report (Ref. 4-2), are continuing. The report on strontium transport data has been completed and will be issued in December (Ref. 4-5).

A summary of the equations and recommended parameters appearing in the strontium transport data report is given in Table 4-6. This table differs in some respects from the data review summary as presented in the previous Quarterly Progress Report. The principal changes were made to achieve compatibility between experimental data and current methods of computing fission product release. In addition, the following improvements in data have been made: (1) an estimate of the reduced diffusion coefficients* for strontium transport in oxide fuel kernels has been made, which was necessary to obtain more accurate predictions of strontium release into the primary circuit of the HTGR, and (2) data for the diffusion coefficients for strontium transport in LTI pyrocarbon have been included.

The data in Table 4-6 for sorption of strontium on graphite were made primarily on H-327 graphite and not on the reference near-isotropic graphite; however, recent measurements (see section on Vapor Pressure Studies) made on a reference near-isotropic graphite (H-451) show no significant difference from the previous measurements on H-327 graphite.

The strontium transport data report includes discussions of the concentration dependence of the diffusion coefficient of strontium in graphite, the effect of irradiation on sorption isotherms, and the effect of mixed isotopes on sorption and diffusion in graphite.

*See section on Fission Metal Release Studies for definition of reduced diffusion coefficient.

TABLE 4-6
SUMMARY OF EQUATIONS AND PARAMETERS DESCRIBING
REFERENCE DATA FOR STRONTIUM TRANSPORT

DIFFUSION					
$D' = D'_O \exp(-Q/RT)$					
Diffusion In	Parameters		Conditions		
	$D'_O(\text{sec}^{-1})$	$Q(\text{cal/mole})$			
Fuel kernels, oxide	1.96×10^9	142,000 ^(a)	FIMA $\leq$ 20%		
Fuel kernels, carbide	6.43×10^{-2}	49,000			
$D = D_O \exp(-Q/RT)$					
	Parameters		Conditions		
	$D_O(\text{cm}^2/\text{sec})$	$Q(\text{cal/mole})$			
Pyrocarbon, LTI	2.3×10^{-2}	47,000	$C \leq 0.7 \mu\text{mole/g}$		
Graphite	1.66×10^2	64,000			
SORPTION					
$\ln p = (A + B/T) + (D + E/T) \ln C$			$C \geq C_t$		
$\ln p = (A + B/T) + (D - 1 + E/T) \ln C_t + \ln C$			$C < C_t$		
Sorption On	Parameters				
	A	B	D	E	$C_t(\mu\text{mole/g})$
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) This value is unusually large; further experiments to more accurately determine the diffusion coefficient of strontium in oxide fuel kernels are in progress.

TRITIUM PERMEATION RATE THROUGH STEAM GENERATOR MATERIALS

The study of tritium permeation through steam generator tube materials is continuing. Assembly of the experimental apparatus described in the previous Quarterly Progress Report (Ref. 4-2) was completed. In addition to the CuO cold-trap arrangements for determining the tritium permeation rate by liquid scintillation counting, an ionization chamber was added to the helium sweep line in front of the CuO converter for direct reading of the tritium permeation rate. The ionization chamber can be used to follow the permeation rate as a function of time, but it is useful only when the tritium permeation rate is high. There was good agreement between the cold-trap data and the ionization chamber results.

Five helium-tritium mixtures, with specific activities in the range of 10^{-3} to 10^{-5} microcuries per standard cubic centimeter were prepared as the sources for the permeation studies. In addition to the tubular T-22 steel samples of 1-mm wall thickness prepared during the last quarter, two additional tubular T-22 samples, one with a wall thickness of 2 mm and the other with a wall thickness of 3 mm, were fabricated. These samples will be used for the study of the effect of wall thickness on tritium permeability.

Measurements were completed on the tritium permeability through a tubular T-22 sample with a wall thickness of 1 mm (15.24 cm in length, 1.705 cm i.d., 1.905 cm o.d.) in the temperature range 304° to 508°C, using a helium-tritium mixture with a specific activity equal to 8.45×10^{-4} microcuries per standard cubic centimeter. This mixture contains 610 ppm (by volume) H_2 , 28 ppm O_2 , 4.6 ppm H_2O , <1 ppm CO , and <1 ppm CH_4 . Measurements were made at various temperatures while the sample was cooled from 508° to 304°C and then brought back to 500°C. The results obtained are shown in Table 4-7.

The tritium permeation rates P , expressed in microcuries per square centimeter per hour for a wall thickness of 1 mm, are plotted versus

TABLE 4-7
TRITIUM PERMEATION RATE THROUGH T-22

Source specific activity = $8.45 \times 10^{-4} \mu\text{Ci/standard cm}^3$,
equivalent to $P_{\text{HT}} = 5.00 \times 10^{-7}$ torr

Impurity contents: $\text{H}_2 = 610$ ppm, $\text{O}_2 = 28$ ppm, $\text{CO} < 1$ ppm,
 $\text{CH}_4 < 1$ ppm, $\text{H}_2\text{O} = 4.6$ ppm

Run No.	Temp. (°C)	Permeation Rate ($\mu\text{Ci-mm/cm}^2\text{-hr}$)
1	508	4.12×10^{-4}
2	444	2.86×10^{-4}
3	444	2.89×10^{-4}
4	411	2.01×10^{-4}
5	411	1.05×10^{-4}
6	354	2.95×10^{-5}
7	354	3.59×10^{-5}
8	304	1.95×10^{-5}
9	304	1.81×10^{-5}
10	304	2.51×10^{-5}
11	350	4.01×10^{-5}
12	350	4.03×10^{-5}
13	350	4.01×10^{-5}
14	404	1.16×10^{-4}
15	404	1.11×10^{-4}
16	447	2.04×10^{-4}
17	447	1.76×10^{-4}
18	447	1.46×10^{-4}
19	500	4.68×10^{-4}
20	500	5.50×10^{-4}

reciprocal temperature in Fig. 4-3. The least squares line through the data points can be represented by the equation

$$P = 6.09 \exp\left(\frac{-7380}{T}\right)$$

The numbers next to the data points in Fig. 4-3 indicate the sequence of measurements. The direction of changing the temperature had no apparent effect on the data. The measurements will be repeated after the O₂ and H₂O impurities are removed by trapping with a liquid-nitrogen-cooled charcoal trap. The results should indicate the effect the presence of these impurities has on the tritium permeability.

It is planned to determine the tritium permeability through T-22 as a function of (1) tritium specific activity in the helium-tritium mixture, (2) hydrogen impurity concentration in the helium-tritium mixture, and (3) sample wall thickness. Efforts are being made to procure a tritium source containing very low hydrogen impurity concentrations so that the tritium specific activity and the hydrogen impurity concentration in the helium-tritium mixture used for the permeability study can be varied independently.

TRITIUM RETENTION IN CORE MATERIALS

Introduction

A cooperative research effort directed toward tritium retention in irradiated HTGR core materials was conducted in FY-74 by General Atomic and the Los Alamos Scientific Laboratory (LASL). General Atomic supplied samples and characterization data and LASL performed the actual laboratory testing.

The main concern with tritium is the possible release to the environment via diffusion through steam generator tube walls into the secondary

Fig. 4-3. Tritium permeation rate through T-22. Helium-tritium source was 8.45×10^{-4} microcuries per standard cubic centimeter equivalent to $P_{\text{HT}} = 5.00 \times 10^{-7}$ torr

coolant. Tritium (H-3) activity in the core of the HTGR has four principal sources:

1. Ternary fission of U-235 with a fission yield of about 1×10^{-4} .
2. Neutron activation of He-3 in the helium coolant.
3. Neutron activation of Li-6, which occurs as a trace impurity in the core and reflector graphite.
4. Neutron activation of B-10 in control and burnable poison rods.

Neutron reactions with other nuclides occur, but are deemed insignificant contributors due to low reaction cross sections, high threshold energies of the reactions (insignificant neutron flux at high threshold energy), or low concentrations of the reactant species.

A treatment of tritium activity in the primary circuit must therefore consider the aforementioned four sources. An obvious major source is He-3 activation, as no release mechanism is required. For the other potential sources, however, significant barriers are interposed to restrict release to the helium coolant. The objective of the GA/LASL work was to provide additional needed data on the effectiveness of these barriers to H-3 release, with the hope of modifying the conservative assumption used in present predictions that all the H-3 formed in graphite and control materials escapes into the helium coolant.

The results of previous work done at GA and ORNL and by German investigators generally indicated that tritium is retained in pyrocarbon coated fuel materials at normal reactor operating temperatures and that the retention decreases with increasing temperature and burnup.

The cooperative GA/LASL effort was subdivided as follows:

1. Determine the retentive characteristics of TRISO and BISO fuel particle coatings with respect to tritium.

2. Determine the Li and H-3 content of H-327 graphite after a known neutron fluence in order to examine the Li-6 (n, α) reaction and determine the retentive characteristics of H-327 graphite.
3. Determine the H-3 produced and retained in boronated graphite (control) material after a known irradiation.
4. Compare the LASL results with similar experiments previously performed at GA to further validate existing data.

H-3 Retention in Coated Fuel Particles

Five batches of fuel particles (see Table 4-8) were selected at GA for analysis by LASL. These included four batches previously tested for tritium retention at GA by an oxygen train combustion and subsequent liquid scintillation technique described in Ref. 4-6.

The characterization of the particles selected for LASL was conducted in the manner described in Ref. 4-6. The particle batches were first microradiographed to ascertain coating integrity. Those batches containing intact coatings were then counted for fission product inventory by gamma-ray spectrometry using a Li-drifted germanium detector coupled to a Sigma II computer analyzer. For BISO coated particles, the concentration of ruthenium-106, a refractory metal, was used to calculate the total number of fissions that occurred in the kernel. For the TRISO coated particles, cesium-137 was chosen as the reference fission product because of its longer half life and higher activity, and its quantitative retention by TRISO coatings.

From the total fissions, the concentration of tritium in the particle batches was calculated using a ternary fission yield of 0.87×10^{-4} . Ternary fission is the only significant source of tritium in the kernels. The fission yield number is probably uncertain by $\pm 10\%$ and is based on the work of Ray et al. (Ref. 4-7).

TABLE 4-8
FUEL PARTICLE TRITIUM ANALYSIS

Sample Ident.	Avg. Fuel Temp. (°C)	Capsule No.	Burnup (% FIMA)	Batch No.	Kernel Type	Coating Type	Cell Thimble	Calculated H-3 (dpm)	Experimental Results		
									H-3 Found by LASL (dpm)	H-3 Retention (%)	
										LASL	GA
C761	1575	P13L	26	4263-75E	(Th,U)C ₂	TRISO	CIT5	8.54×10^5	3.08×10^5	36	29 (2 runs)
C765	1250	P13L	75	4413-5E	UC ₂	TRISO	C3T3	5.40×10^5	2.19×10^5	40	34 (2 runs)
C768	1165	P19	21	4000-695E	(Th,U)C ₂	BISO	C3T3	6.57×10^5	(a)	(a)	26 (2 runs)
C770	1275	P13L	75	4403-113E	UC ₂	BISO	C2T2	6.18×10^5	(a)	(a)	23 (3 runs)
C772	715	P22	6	4042-129	ThC ₂	TRISO	C2T3	1.50×10^6	1.53×10^6	102	No data

(a) Experimental results are not reported for samples C768 and C770 because the quantities of H-3 found (2.9×10^7 and 1.31×10^8 , respectively) are unreasonably high and clearly incorrect, suggesting tritium contamination in the equipment.

The H-3 activity in each batch was corrected for decay to the counting date, as was the calculation of the fission product inventory. The print-out from the Sigma II computer giving both current H-3 activity and fission product inventory for each batch was sent to LASL with the particles.

The LASL CMB-1 group performed the analyses for H-3 on the irradiated fuel particle samples in their hot cell facility utilizing Procedure No. ANC-DE-1-HC-7,R2. The procedure consists of reacting the sample with moist oxygen at 1000°C and collecting the tritiated water in a cold trap. This trap is then heated, and the tritiated water is swept by argon into a volumetric flask containing distilled water. An aliquot of this water is pipetted into a scintillation vial filled with Aquasol, and the weak β (0.0186 MeV) from the H-3 is counted. The relative standard deviation of this method is $\pm 10\%$ for measuring 0.1 μg of tritium, and the bias is 4.0%. This experimental uncertainty combined with the computational uncertainty of $\pm 10\%$ related to the fission yield gives a sigma of approximately $\pm 15\%$.

The results of the LASL analysis are shown in Table 4-8 and compared with earlier GA work on the same batches (where data exists). For the TRISO particles analyzed at both GA and LASL (C761 and C765), excellent agreement is found, with the LASL results being a few percent higher. The LASL results for the remaining TRISO batch (C772) agree well with the retention of $\sim 100\%$ that GA found in low-temperature low-burnup TRISO coated particles.

The LASL results for the BISO coated particles showed a much higher H-3 content than the corresponding GA results and are clearly incorrect. This unresolved error may have been the result of contaminated equipment during the analyses for these two particle samples. These were the first two analyses.

Analysis of Irradiated H-327 Graphite Samples

Five samples of irradiated graphite were selected at GA for analysis at LASL. The sample identification and characterization are shown in

Table 4-9. The testing at LASL included wet chemistry analysis for Li on two of the H-327 samples (2 and 185) and H-3 determination on the remaining three (33, 143, 156) by the oxygen combustion method used on the fuel particles.

The Li analysis was conducted in the following manner. Each sample was moistened with 1 drop of concentrated H_2SO_4 , ignited, and maintained at 700°C overnight. The residue was then dissolved in 1 ml H_2O to which 1 drop of concentrated H_2SO_4 had been added. This solution was dried to fumes under a heat lamp. This residue was then dissolved in 1/2 ml of H_2O and the absorbance measured at 670.8 nm, using a carbon rod atomizer. A standard was prepared by this same treatment from reagent grade Li_2CO_3 . The precision of measurement for lithium at the 0.10 ppm level is ± 0.05 ppm by this method.

The LASL results shown in Table 4-9 give a lithium impurity in H-327 of between 0.08 and 0.09 ppm. This is in accord with GA values of less than 0.09 ppm. The portion of the total lithium content attributable to Li-6 is 7.56%, giving an impurity level of about 7 ppb for this isotope. This corresponds to approximately 1.3×10^{14} Li-6 atoms in the 360-mg samples used. Neutron activation by the thermal neutron fluences shown in Table 4-9 should have produced approximately 1.14×10^{14} and 9.0×10^{13} tritium atoms for samples 2 and 185, respectively, taking decay between irradiations and the preceding date of analysis into account. This represents in alternative units ~ 1 ppb of H-3 atoms. This was below the limit of detection for the LASL analysis method and is in accord with the reported <0.001 ppm for the three graphite samples analyzed for H-3. The low Li impurity level in the graphite samples therefore precluded any assessment of their retention of tritium.

H-3 Determination in Control Materials

Five boronated graphite samples, irradiated in capsules BG-1 and BG-2, were analyzed at LASL by the same combustion method used for the coated fuel particles. The amounts of tritium found in the samples range from 0.09 to 0.37 ppm.

TABLE 4-9
TRITIUM AND LITHIUM DETERMINATIONS IN GRAPHITE SAMPLES^(a)

Specimen Number	Shape	Approximate Dimensions (in.)	Orientation	G-12 ^(b)		GE-H-13-481 ^(b)		Total Fluence (10^{21} n/cm ²)	Analysis	LASL Exp. Results (ppm)
				Fluence (10^{21} n/cm ²)	Temp. (°C)	Fluence (10^{21} n/cm ²)	Temp. (°C)			
143	Wedge	1/4 x 1/4 x 1	Perp.	5.0	900	4.0	825	9.0	H-3	<0.001
156	Wedge	1/4 x 1/4 x 1	Perp.	5.0	900	4.7	825	9.7	H-3	<0.001
185	Wedge	1/4 x 1/4 x 1	Parallel	5.0	900	4.8	825	9.8	Li	0.09
2	Cylinder	0.18 diam x 0.45 long	Parallel	--	--	4.3	825	4.3	Li	0.08
33	Cylinder	0.18 diam x 0.45 long	Perp.	--	--	5.3	825	5.3	H-3	<0.001

^(a) H-327 preproduction graphite.

^(b) Specimens 143, 156, and 185 were irradiated first in the G-12 irradiation test and then in the GE-H-13-481 irradiation test. The other two specimens were irradiated only in the GE-H-13-481 test. G-12 end-of-irradiation was June 1967; GE-H-13-481 end-of-irradiation was June 1971.

An effort is under way to calculate the quantity of tritium produced in each sample during irradiation. This will enable the determination of tritium retention values.

EFFECT OF TOTAL PRESSURE ON THE RATE OF GRAPHITE OXIDATION BY STEAM

The effect of helium pressure on the rate of reaction of steam with H-327 graphite is being measured over the range 800° to 1000°C and at pressures ranging from 1 to 50 atm. The data are needed for use in theoretical calculations using the OXIDE-3 or GOP computer codes which predict the consequences of a steam inleakage accident on graphite integrity and coolant impurity levels. The present calculations utilize a theoretical formulation which indicates that the overall rate of the oxidation of graphite by steam in the temperature regime of the HTGR is controlled by a combination of in-pore diffusion and the true chemical reaction rate. The equation is of the form

$$\text{Rate} = C_o (k_v D_{\text{eff}})^{1/2} \quad (4-19)$$

where C_o = concentration of water vapor,
 K_v = first order reaction rate constant, and
 D_{eff} = effective diffusion coefficient for water vapor in graphite.

Since the diffusion coefficient for water vapor in graphite, D_{eff} , is inversely proportional to the total gas pressure, it follows that the overall rate of oxidation will be proportional to $(1/P_{\text{He}})^{1/2}$, according to Eq. 4-19. In reality, Eq. 4-19 is precisely true for a rather narrow range of temperatures, where the overall reaction rate is controlled by diffusion of water vapor into the porous substrate. At higher temperatures, the rate is governed by gas phase mass transport at the surface of the graphite and is thus proportional to $1/P$. At low temperatures, the rate is controlled by the true chemical reaction rate and is independent of He pressure. Thus, the rate of graphite oxidation in the HTGR is proportional to

P^{-n} , where n is a variable which theoretically ranges from 0 to 1. It is, therefore, of importance to determine the pressure dependence factor, n , in the temperature range of interest so that calculations of oxidation at reactor conditions are as accurate as possible.

The apparatus used in performing these studies has been described in an earlier Quarterly Progress Report (Ref. 4-8). At that time, measurements of inordinately high rates of oxidation were being reported. Carbon transport, air ingress, and catalysis by the metal reaction tube were believed to be the major contributors to these observed high rates of reaction. After considerable modification of the apparatus (Ref. 4-9) (in particular, installation of a silica liner in the reaction tube), analysis of the effluent gases indicated that these problems had largely been solved, but slightly higher rates of graphite oxidation than those observed in microbalance experiments continued to be evident. This difference in rate data is currently being analyzed.

Other investigators (Ref. 4-10) have clearly demonstrated the inhibiting effect of the reaction product hydrogen on the rate of corrosion of H-327 graphite. During this quarter, studies were aimed at characterizing H-327 oxidation as a function of total pressure with a constant quantity of hydrogen present in the reactant gas. Figure 4-4 summarizes the data obtained when a constant partial pressure of hydrogen was mixed with the process helium, along with the rate data obtained earlier (Ref. 4-9) in the absence of hydrogen added to the reactant gas. In both cases, the rates of reaction are for a temperature of 900°C and a partial pressure of water vapor of 0.03 atm. In the first case, however, with no H_2 added, the H_2 pressure was variable, depending on the reaction rate. In the second case, the H_2 partial pressure was virtually constant. The negative slope of the line drawn through the data gives a value for n , the power relationship between reaction rate and pressure. From the data presented, the values of n for a reactant gas without hydrogen added and with 0.01 atm of hydrogen added are 0.64 and 0.79, respectively. Therefore, integral reactor core

Fig. 4-4. Effect of helium pressure on the rate of reaction of steam and H-327 graphite at 900°C and $P_{H_2O} = 0.03$ atm

oxidation calculations using $n = 0.5$ would conservatively predict somewhat higher overall reaction rates and oxidation at greater depths than the data presented here indicate.

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TASK 8 (189a 13118)

REACTOR PHYSICS

CSEWG THERMAL REACTOR BENCHMARK CALCULATIONS USING ENDF/B VERSION 4 DATA

A series of benchmark calculations recommended by the Cross Section Evaluation Working Group (CSEWG) for thermal reactor data testing have been performed using current GA reactor physics calculational methods and the ENDF/B Version 4 data library. These calculations include:

1. Five unreflected spheres of U-235 (as uranyl nitrate) in H_2O , three of them poisoned with boron (ORNL-1/2/3/4/10).
2. Five unreflected spheres of Pu-239 (as plutonium nitrate) in H_2O (PNL-1/2/3/4/5).
3. Two H_2O -moderated lattices of 1.3%-enriched uranium metal rods in a triangular lattice (TRX-1/2).
4. Three D_2O moderated lattices of natural uranium metal rods in a triangular lattice (MIT-1/2/3).

The basic criticality results are compared with previous GA results using ENDF/B Version 3 data in Tables 8-1 and 8-2. Computed activation parameters for the H_2O and D_2O moderated lattices are given in Table 8-3.

The ENDF/B Version 4 data yielded slightly better agreement between calculation and experiment for the uranium-fueled assemblies than was obtained with ENDF/B Version 3 data. Somewhat poorer agreement was obtained

TABLE 8-1
CRITICALITY (ENDF/B-3/4)

Benchmark	Description	k_{eff}		Difference
		Version 4	Version 3	
ORNL	Unreflected spheres of uranyl nitrate solution			
-4	H/U-235 = 971, R = 34.595 cm	0.9989	0.9963	0.0026
-3	H/U-235 = 1033, R = 34.595 cm	0.9978	0.9952	0.0026
-2	H/U-235 = 1177, R = 34.595 cm	1.0007	0.9982	0.0025
-1	H/U-235 = 1378, R = 34.595 cm	1.0012	0.9987	0.0025
-10	H/U-235 = 1835, R = 61.011 cm	<u>0.9982</u>	<u>0.9952</u>	<u>0.0030</u>
		0.9994 avg	0.9967 avg	0.0027 avg
PNL	Unreflected spheres of plutonium nitrate solution			
-2	H/Pu-239 = 131, R = 19.509 cm	1.0201	1.0149	0.0052
-5	H/Pu-239 = 578, R = 20.1265 cm	1.0169	1.0132	0.0037
-1	H/Pu-239 = 698, R = 19.509 cm	1.0225	1.0193	0.0032
-4	H/Pu-239 = 911, R = 22.70 cm	1.0102	1.0076	0.0026
-3	H/Pu-239 = 1204, R = 22.70 cm	<u>1.0021</u>	<u>1.0000</u>	<u>0.0021</u>
		1.0144 avg	1.0110 avg	0.0034 avg
TRX	H ₂ O moderated uranium metal lattices			
-1	Mod/fuel = 2.35	0.9855	0.9791	0.0064
-2	Mod/fuel = 4.02	<u>0.9961</u>	<u>0.9924</u>	<u>0.0037</u>
		0.9908 avg	0.9858 avg	0.0050 avg
MIT	D ₂ O moderated uranium metal lattices			
-1	Mod/fuel = 20.74	0.9972	0.9888	0.0084
-2	Mod/fuel = 25.88	1.0006	0.9925	0.0081
-3	Mod/fuel = 34.59	<u>1.0076</u>	<u>0.9996</u>	<u>0.0080</u>
		1.0018 avg	0.9936 avg	0.0082 avg

TABLE 8-2
NEUTRON BALANCE SUMMARY
(VERSION 4, MODIFIED P₁ DATA)

Benchmark	ϕ_f/ϕ_{th} (a)	Fission Neutrons	Absorption (b)	Leakage	k_∞	k_{eff}
ORNL-4	1.3903	0.9989	0.8347	0.1653	1.1968	0.9989
-3	1.3152	0.9978	0.8340	0.1660	1.1964	0.9978
-2	1.1622	1.0007	0.8316	0.1684	1.2033	1.0007
-1	1.0026	1.0012	0.8288	0.1712	1.2080	1.0012
-10	0.8198	0.9982	0.9343	0.0658	1.0685	0.9982
PNL-2	16.123	1.0201	0.6218	0.3782	1.6404	1.0201
-5	3.6275	1.0169	0.6363	0.3637	1.5980	1.0169
-1	3.0158	1.0225	0.6370	0.3630	1.6050	1.0225
-4	2.4024	1.0102	0.6806	0.3194	1.4843	1.0102
-3	1.8618	1.0021	0.6881	0.3119	1.4563	1.0021
TRX-1 ^(c)	3.5430	0.9855	0.8518	0.1482	1.1710	0.9855
-2	2.2222	0.9961	0.8677	0.1323	1.1504	0.9961
MIT-1 ^(c)	0.9575	0.9972	0.8626	0.1374	1.1561	0.9972
-2	0.7746	1.0006	0.8456	0.1544	1.1833	1.0006
-3	0.5853	1.0076	0.8317	0.1683	1.2115	1.0076

(a) Fast-to-thermal cutoff energy of 0.625 eV.

(b) Includes transverse leakage in TRX case. The n,2n reaction rate has been subtracted from the listed absorption rates.

(c) Lattice region values in TRX and MIT cases.

TABLE 8-3
ACTIVATION PARAMETERS ^(a)
ENDF/B VERSIONS (V) 3 AND 4

Benchmark	ρ^{28}			δ^{25}			ϵ^{28}			C*		
	Exp.	GA Calc.		Exp.	GA Calc.		Exp.	GA Calc.		Exp.	GA Calc.	
		V-3	V-4		V-3	V-4		V-3	V-4		V-3	V-4
TRX-1	1.311	1.416	1.407	0.0981	0.0981	0.0982	0.0914	0.0910	0.0965	0.792	0.829	0.812
	± 0.020	(1.080) ^(b)	(1.073)	± 0.0010	(1.000)	(1.001)	± 0.0020	(0.996)	(1.056)	± 0.008	(1.047)	(1.025)
TRX-2	0.830	0.877	0.881	0.0608	0.0602	0.0606	0.0667	0.0659	0.0700	0.646	0.661	0.653
	± 0.015	(1.057)	(1.061)	± 0.0070	(0.988)	(0.997)	± 0.0020	(0.988)	(1.049)	± 0.002	(1.023)	(1.011)
MIT-1	0.498	0.534	0.529	0.0447	0.0473	0.0469	0.0597	0.0584	0.0607	1.017	0.987	0.969
	± 0.008	(1.072)	(1.062)	± 0.0019	(1.058)	(1.049)	± 0.0020	(0.978)	(1.017)	± 0.023	(0.971)	(0.953)
MIT-2	0.394	0.435	0.431	0.0310	0.0385	0.0382	0.0596	0.0562	0.0585	0.948	0.929	0.914
	± 0.002	(1.104)	(1.094)	± 0.0030	(1.242)	(1.232)	± 0.0017	(0.943)	(0.982)	± 0.020	(0.980)	(0.964)
MIT-3	0.305	0.334	0.331	0.0248	0.0295	0.0293	0.0583	0.0537	0.0559	0.859	0.868	0.854
	± 0.004	(1.095)	(1.085)	± 0.0010	(1.190)	(1.181)	± 0.0012	(0.921)	(0.959)	± 0.016	(1.010)	(0.994)

(a) Parameters correspond to a thermal cutoff of 0.625 eV.

ρ^{28} = ratio of epithermal-to-thermal U-238 captures.

δ^{25} = ratio of epithermal-to-thermal U-235 fissions.

ϵ^{28} = ratio of U-238 fissions to U-235 fissions.

C* = ratio of U-238 captures to U-235 fissions.

(b) The numbers in parentheses are the calculated values divided by the experimental values.

for the plutonium-fueled assemblies. The GA results obtained with the ENDF/B Version 3 data are consistent with those obtained by other laboratories (Ref. 8-1) except for the PNL-2 benchmark.

In the case of the PNL-2 benchmark, there appears to be some uncertainty in the CSEWG benchmark specification. The specification used for the PNL-2 benchmark calculations reported herein is given in Table 8-4. The specifications used for the other benchmark calculations are given in Ref. 8-2, Appendices A through D.

It should be noted that the ENDF/B Version 3 results for the ORNL and PNL spheres reported here differ slightly from those reported in Ref. 8-2. Group dependent bucklings determined from one-dimensional transport theory calculations were used in the ENDF/B Version 4 spectrum calculations. In order to obtain a consistent comparison, the ENDF/B Version 3 calculations were repeated with the same bucklings.

The longstanding discrepancy between calculated and experimental values of epithermal-to-thermal captures in U-238 persists in these calculations, which were performed using very nearly exact two spatial region resonance cross-section calculations at over 13,000 energies. The calculated epithermal-to-thermal capture ratio in U-238 is about 5 to 8% larger than experimentally determined values. Epithermal fission in Pu-239 also appears to be overestimated in these and other calculations based upon the ENDF/B Version 3 data library (because of the systematically high eigenvalues computed for the harder spectrum systems). The Pu-240 resonance cross sections are probably not at fault because similar results have been obtained for systems containing 42% Pu-240 (Ref. 8-3).

The methods used to perform these calculations are summarized as follows. The evaluated neutron cross-section data source was the ENDF/B Version 4 data file. The GFE4 code was used to generate 99 group [GAM-II

TABLE 8-4
REVISED PNL-2 BENCHMARK SPECIFICATION

H/Pu-239 atom ratio = 131, Pu-240 = 4.6 wt %, and
effective radius = 19.509 cm

Material	Concentration (10^{24} atoms/cm ³)
H	0.054164
O	0.039768
N	4.7204×10^{-3}
Pu-239	4.141×10^{-4}
Pu-240	1.9884×10^{-5}

(Ref. 8-4) group structure] fast neutron data. The GFE4 code is an updated version of the GFE2 code described in Ref. 8-5; the GFE4 code will handle the new ENDF/B Version 4 data formats and also includes more general resolved resonance (including Adler-Adler capability) and nonelastic scattering energy transfer array computational algorithms.

The GAND3 code was used to generate 13,463 energy resonance data covering the 7102 to 2.38 eV energy range (point spacing of 85 meters/sec). The GAND3 code is an updated version of the GAND2 code described in Ref. 8-5 and will handle the new ENDF/B Version 4 formats as well as MLBW and Adler-Adler resolved resonance region representations. The GAND3 code was also used to generate 101 energy [GATHER-II (Ref. 8-6) grid] thermal neutron data for absorber nuclides by condensation from a 900 energy ultra-fine mesh grid (2.38 to 0.001 eV) with a Maxwellian joined to $1/E$ at 10 kT condensation spectrum.

The FLANGM code [a local modification of the FLANG-II code (Ref. 8-7)] properly distributes the ultra-fine mesh normalized integral over all fine groups rather than assigning it entirely to the self-scatter fine group. The FLANGM code was used to prepare 101 energy thermal neutron scattering kernels (P_0 and P_1) for H in H_2O and D in D_2O from the 1969 ENDF/B scattering law data for these nuclides (materials 1002 and 1004). Effective temperatures for use in the "short collision time" approximation used in the FLANGM code for incident energies greater than 1 eV were obtained from the work of Koppel and Houston (Ref. 8-8). Free gas model thermal neutron scattering kernels for nitrogen, oxygen, and U-238 were prepared with the WTFG code (Ref. 8-9).

These cross sections were then used with the MICROX code (Ref. 3-10) to prepare 19 group cross sections for use in 48 interval P_1 , S_4 calculations with the DTEX code, [the GA version of the DTF-IV code (Ref. 8-11)]. The 19 energy group structure is given in Table 8-5.

TABLE 8-5
ENERGY GROUP STRUCTURE

Group Number	Lower Energy (eV)	Lower Lethargy, μ	$\Delta\mu$
1	1.0×10^7	0.0	0.4
2	3.68×10^6	1.0	1.0
3	8.21×10^5	2.5	1.5
4	1.83×10^5	4.0	1.5
5	6.74×10^4	5.0	1.0
6	1.50×10^4	6.5	1.5
7	3.35×10^3	8.0	1.5
8	9.61×10^2	9.25	1.25
9	3.53×10^2	10.25	1.0
10	1.30×10^2	11.25	1.0
11	4.79×10^1	12.25	1.0
12	1.76×10^1	13.25	1.0
13	3.93	14.75	1.5
14	2.37	15.25	0.5
15	6.25×10^{-1}	16.59	1.34
16	4.14×10^{-1}	17.00	0.41
17	1.00×10^{-1}	18.42	1.42
18	4.00×10^{-2}	19.34	0.92
19	0.0	--	--

The MICROX calculations for the uranium and plutonium spheres were one region modified B_0 spectrum calculations using energy-dependent bucklings determined by iteration between MICROX and DTFX transport theory calculations. The P_1 , S_4 transport theory leakages were used to determine bucklings according to the formula

$$\beta_g^2 = 3 \Sigma_{tr,g} L_g / Q_g \quad . \quad (8-1)$$

After one iteration, good agreement between input and output bucklings was obtained.

The MICROX calculations for the H_2O and D_2O moderated lattices were two space region modified B_1 calculations over the 15-MeV to 0.001-eV energy range using the bucklings given in the benchmark specifications. Dancoff correction factors were computed for the H_2O and D_2O moderated lattices with the GADACO code (Ref. 8-12) and are tabulated in Table 8-6.

The aluminum clad was smeared into the H_2O or D_2O . The void region in the TRX-1/2 lattices was explicitly represented in the Dancoff correction factor calculations and smeared into the moderator region in the MICROX calculations.

The P_1 DTFX cross sections prepared by the MICROX code include an "extended transport approximation" correction for higher order scattering events (pages 56-60 of Ref. 8-10). This correction reduced the computed eigenvalue for the PNL benchmarks by about 0.005 (see Table 8-7). This correction would be much smaller for the ORNL spheres in which leakage effects are much smaller than in the PNL spheres.

The effect of changing the Pu-239 fission spectrum only from ENDF/B-3 to ENDF/B-4 was inadvertently obtained and found to increase the computed eigenvalues for the PNL benchmarks by about 0.001 (see Table 8-7).

TABLE 8-6
DANCOFF CORRECTION FACTORS

Benchmark	Dancoff Correction Factor	Moderator Cross Section Used in Dancoff Calculation (cm^{-1})
TRX-1	0.1313	1.3243
TRX-2	0.0549	1.3881
MIT-1	0.0055	0.3479
MIT-2	0.0030	0.34817
MIT-3	0.0013	0.34844

TABLE 8-7
ADDITIONAL PNL SPHERE RESULTS

Benchmark	H/Pu-239	ϕ_f/ϕ_{th} ^(a)	Leakage (Modified P_1)	Δk Due to Transfer Array Option (Modified P_1 - Standard P_1) ^(b)	Δk Due to Pu-239 Fission Spectrum Change (Version 4 to Version 3) ^(c)
PNL-2	131	16.12	0.378	-0.0059	0.0014
-5	578	3.628	0.364	-0.0054	0.0014
-1	698	3.016	0.363	-0.0051	0.0012
-4	911	2.403	0.319	-0.0049	0.0012
-3	1204	1.862	0.312	<u>-0.0046</u>	<u>0.0012</u>
				-0.0052 avg	0.0013 avg

(a) ϕ_{th} is the flux below 0.625 eV; ϕ_f is the flux above 0.625 eV.

(b) ENDF/B Version 4 cross sections and fission spectrum data (modified P_1 transfer array option yields -0.52% smaller k_{eff} values).

(c) ENDF/B Version 3 cross-section data in both cases.

Two-group reaction rates for the TRX lattices computed from 19 group DTFX fluxes and MICROX cross sections are given in Tables 8-8 and 8-9.

In the case of the D_2O moderated lattices, a DTFX calculation was not performed. The results in Tables 8-1 through 8-3 for these lattices are taken directly from the MICROX output. Two group reaction rates for the MIT lattices are given in Tables 8-10 through 8-12.

TABLE 8-8
TRX-1 REACTION RATES
(AT CENTERLINE OF LATTICE)

Group	E_{λ} (eV)	ϕ	$\phi\sigma_c^{238}$	$\phi\sigma_f^{238}$	$\phi\sigma_f^{235}$
1	0.625	0.016465	0.0098894	0.002009	0.14056
2	0.0	0.0043727	0.0070273	0.0	1.43164

TABLE 8-9
TRX-2 REACTION RATES
(AT CENTERLINE OF LATTICE)

Group	E_{λ} (eV)	ϕ	$\phi\sigma_c^{238}$	$\phi\sigma_f^{238}$	$\phi\sigma_f^{235}$
1	0.625	0.0144918	0.0090651	0.0020689	0.12801
2	0.0	0.0062623	0.0102898	0.0	2.11104

TABLE 8-10
MIT-1 REACTION RATES

Group	E_{λ} (eV)	ϕ	$\phi\sigma_c^{238}$	$\phi\sigma_f^{238}$	$\phi\sigma_f^{235}$
1	0.625	78.392	59.289	10.735	1078.9
2	0.0	81.875	112.029	0.0	22998.2

TABLE 8-11
MIT-2 REACTION RATES

Group	E_{λ} (eV)	ϕ	$\phi\sigma_c^{238}$	$\phi\sigma_f^{238}$	$\phi\sigma_f^{235}$
1	0.625	78.651	60.364	12.819	1100.5
2	0.0	101.542	139.963	0.0	28815.9

TABLE 8-12
MIT-3 REACTION RATES

Group	E_{λ} (eV)	ϕ	$\phi\sigma_c^{238}$	$\phi\sigma_f^{238}$	$\phi\sigma_f^{235}$
1	0.625	79.239	62.212	16.381	1135.9
2	0.0	135.382	188.055	0.0	38837.8

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TASK 9 (189a 13119)
FUEL DEVELOPMENT AND ENGINEERING

FUEL IRRADIATIONS

Capsules P13R and P13S

Capsules P13R and P13S are irradiation tests 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 operated at fuel centerline temperatures between 1075° and 1600°C. The capsules were discharged from the GETR on October 31, 1974 with an estimated total accumulated fluence of 1.1×10^{22} n/cm² (E > 0.18 MeV)_{GETR}. The capsules are to be delivered to the GA Hot Cell Facility on December 2, 1974; disassembly is scheduled during the month of December.

Cell 1 of P13S underwent twenty controlled thermal transients between 1075° (soak temperature) and 1600°C (peak temperature). Cycles 11 through 19 were typified by activity spikes during the controlled thermal excursions; i.e., Kr-85m R/B measurements showed a nominal increase of one and a half orders in magnitude during the temperature increase followed by a relaxation back to an equilibrated value approximately twice as great as the initial R/B value. The "activity streaks" during a thermal transient are associated with fuel failure and a sudden release of latent fission products; viz, fuel failure exposes new inner surfaces in the particles and entrapped noble gases diffuse to buffer and kernel free surfaces and contribute to the measured activity. The final equilibrated R/B measurement in this cell was 1.9×10^{-4} , which is approximately an order in magnitude greater than the 3.8×10^{-5} measured in the companion fuel test (cell 1, P13R), excluding thermal cycling. Tentatively, the results indicate

that thermal cycling is an important operation variable which can induce some fuel failure. Final R/B (Kr-85m) measurements for all additional cells were less than 3×10^{-5} , with the exception of cell 2 in P13R and P13S (2×10^{-4}), cell 3 in P13R and P13S (8×10^{-4}), and cell 4 in P13S (2×10^{-4}). The rate of increase in R/B for all cells (defined as the change in natural log of R/B versus the change in irradiation time) has remained effectively constant throughout irradiation.

Capsule P13T

Capsule P13T has been rescheduled for insertion in the ORR on February 24, 1975 to coincide with the removal of capsule P13Q. The decision was made to irradiate P13Q for one additional cycle in order to increase the peak fast fluence of the capsule. The primary objectives of the P13T irradiation test are as follows:

1. Provide a full-exposure qualification test of reference fuel performance in graphite bodies simulating a segment of an HTGR fuel block.
2. Compare and define the irradiation and thermal performance of dense melted UC_2 and weak acid resin (WAR) derived $U \cdot C_x \cdot O_y$ TRISO coated fissile particles at 1100° and 1300°C to peak fluence.
3. Test variations in particle coating attributes as part of the continuing effort to define the permissible range of acceptable coating properties.
4. Characterize the irradiation performance of fuel rods containing alternative shim and matrix materials.

Capsule P13T will have a diameter of 66.5 mm and will contain a total of four graphite bodies in two separate cells. The bottom cell (cell 1) will contain body number 1, a 62-mm-diameter by 200-mm-long graphite (H-451) body loaded with twelve 15.7-mm-diameter fuel rods. Six of the rods will be near the reference length (61 mm long) and six will be 30.5 mm long. The upper cell (cell 2) will house three separate graphite bodies numbered 2, 3, and 4. Bodies 2 and 4 will be TS-1240 graphite and body 3 will be H-451 graphite. The two central bodies (bodies 2 and 3) will be 115 mm long and the uppermost body (number 4) will be 137.5 mm long. Bodies 2 and 3 will each contain six 50-mm-long fuel rods. Body 4 will contain twelve 30.5-mm-long rods. The shorter 30.5-mm rods are placed in the upper and lower ends of the capsule because of steep axial thermal neutron flux gradients at the top and bottom of the core. The shorter rods facilitate adjustment of fuel loading to achieve constant power along the capsule axis, thus eliminating large axial temperature differences within these end rods.

Fuel rod and graphite temperatures will be monitored using Chromel-Alumel and tungsten-rhenium thermocouples. Nine of the 36 fuel rods will be fabricated with central thermocouple holes so their centerline temperature can be measured during the irradiation. These rods will be located at the ends of the capsule and in body 2 (Fig. 9-1).*

Each cell will have a separate purge gas control system to provide a means for temperature control and fission gas release measurements. High purity helium and neon gases will be used as the temperature controlling gases. Periodic fission gas release samples will be taken from each cell in order to determine the R/B values for specified Kr and Xe isotopes. These data will be used to evaluate the in-pile performance of the fuel as a function of fluence and burnup.

Cell 1 will contain only reference fresh fuel. This cell will be exposed to peak temperature-fluence conditions of $1300^{\circ} \pm 50^{\circ}\text{C}$ at $\sim 9.6 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{ORR}. This fast fluence is approximately equivalent

*Figures appear at the end of Section IX.

to an HTGR exposure of $8.5 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{HTGR}. These time-average temperature-fluence conditions are more severe than those experienced by approximately 99% of the fuel in the large HTGR. The range of fluences for this cell will be between 6.0 and $9.6 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{ORR}.

Fuel cell 2 will be irradiated at a peak fuel temperature of $1100^\circ \pm 50^\circ\text{C}$ to 5 to $9 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{ORR}. The peak exposure in this cell will exceed that received by approximately 95% of the fuel in an HTGR core.

Fissile and fertile particles from a large number of coating batches will be tested in capsule Pl3T. These tests will be conducted primarily on particles bonded in fuel rods, although a small number of particles from selected batches will be included as piggyback samples. The piggyback particles will provide samples for subsequent thermal stability work and for burnup analyses.

The fissile particles are UC_2 TRISO and the fertile particles are ThO_2 BISO. Variables to be tested are given in Tables 9-1 and 9-2 for the fissile and fertile particles, respectively. The variables include UC_2 kernel fabrication methods (WAR and VSM), PyC deposition gas (mixed gas and propylene), IPyC density and thickness, SiC thickness, OPyC coating density and coating rate, and buffer seal coating. The WAR kernels are to be completely deoxidized and will contain approximately 2 wt % oxygen. All particle batches for the irradiation tests have been fabricated.

Thirty-six fuel rods will be irradiated in Pl3T. In addition to the coated particle variables, fuel rod variables will include shim type, shim volume loading, and matrix composition. All fuel rods will be cured-in-place and will not be removed from the graphite crucibles prior to irradiation. A duplicate set of fuel bodies will be prepared for the Pl3T experiment for use in preirradiation fuel rod characterization studies. Fission gas release measurements will be made on each body prior to acceptance for use in the capsule.

TABLE 9-1

LIST OF FISSILE PARTICLE VARIABLES TO BE TESTED IN CAPSULE P13T

Sample	Kernel			Buffer Seal	IPyC			SiC	OPyC	Particle Variables
	Fab. Process	Diam. (μm)	Density (g/cm^3)		Thick-ness (μm)	Density (g/cm^3)	Coating Gas	Thick-ness (μm)	Coating Gas	
1	WAR	315	≥ 3.3	No	25	1.90	M-G ^(a)	35	M-G	Recycle ^(b)
2	VSM	200	≥ 10.3	No	35	1.90	M-G	35	M-G	Thick SiC, Thick IPyC
3	VSM	200	≥ 10.3	Yes ^(c)	25	1.90	M-G	25	M-G	Reference fuel
4	VSM	200	≥ 10.3	No	35	1.90	M-G	25	M-G	Thick IPyC
5	VSM	200	≥ 10.3	No	25	1.90	M-G	35	M-G	Thick SiC
6	VSM	200	≥ 10.3	Yes ^(c)	25	1.95	Propylene	35	Propylene	Previous reference ^(d)

(a) M-G = mixed gases (acetylene/propylene).

(b) The WAR UC_2 kernel has been selected as the reference recycle fissile fuel.

(c) Buffer seal coating is not included in reference design although this coating has been routinely applied to particles in previous tests.

(d) Particle batch to compare M-G isotropic PyC performance with propylene pyrocarbons.

TABLE 9-2

LIST OF FERTILE PARTICLE VARIABLES TO BE TESTED IN CAPSULE P13T

Sample	Particle Type	Buffer Seal	OPyC		Total Coating Thickness (μm)	OPyC Coating Rate	Particle Coating Variables
			Coating Gas	Density (g/cm^3)			
1	ThO_2 -BISO	No ^(a)	Mixed-gas	1.85 ^(b)	160	$\geq 3, \leq 5$ ^(b)	Near reference
2	ThO_2 -BISO	No ^(a)	Mixed-gas	1.95 ^(c)	160	$\geq 3, \leq 5$ ^(b)	High-density OPvC
3	ThO_2 -BISO	No ^(a)	Mixed-gas	1.85 ^(c)	160	$\geq 5, \leq 8$ ^(c)	High coating rate
4	ThO_2 -BISO	No ^(a)	Mixed-gas	1.95 ^(c)	160	$\geq 5, \leq 8$ ^(c)	High coating rate, high-density OPyC
5	ThO_2 -BISO	No ^(a)	Mixed-gas	1.90 ^(b)	160	$\geq 5, \leq 8$ ^(c)	High coating rate, medium-density OPyC
6	ThO_2 -BISO	No ^(a)	Mixed-gas	1.85 ^(c)	140	$\geq 3, \leq 5$ ^(b)	Thin coating
7	ThO_2 -BISO	Yes	Mixed-gas	1.85 ^(c)	160	$\geq 3, \leq 5$ ^(b)	Buffer seal
8	ThO_2 -BISO	Yes	Propylene	1.90	160	$\geq 1, \leq 10$	Previous reference

(a) Buffer seal coating is not included in reference design although this coating has been routinely applied to particles in previous tests.

(b) Current most probable midrange value; represents a change from previous reference design.

(c) Values within expected specification limits.

Figure 9-1 shows the fuel rod sample layout for P13T and gives the design irradiation conditions. The rod variables are given in Table 9-3.

All fuel rods for the test have been fabricated and firing and characterization of the fuel bodies are now in process.

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.

The estimated peak fast fluence in capsule P13Q is $8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)_{ORR}. The R/B measurement for Kr-85m is 10^{-6} , which is consistent with R/B values obtained prior to the detected gas leakage between the two sections of the capsule. Current operating procedures have minimized gaseous back-streaming between the GA and ORNL cells, and the R/B measurements represent a maximum value implying no detectable change in the fuel failure fraction during irradiation in the GA cell. P13Q is currently scheduled to complete design irradiation exposure on February 23, 1975. This date is 3 months later than originally planned; the difference is attributed to ORR downtime and revised dosimetry estimates.

Capsules HT-28 and -29

The HT-28 and -29 irradiation series is part of a continued cooperative effort between GA and ORNL and is designed to evaluate the irradiation behavior of unbonded BISO and TRISO coated ThO fertile fuel particles. The experiments will be conducted in the target position of the HFIR at ORNL. The experimental design and operation are similar to those for HT-12 through -15 and HT-17 through -19. Experiments of this type are conducted in the

TABLE 9-3
P13T FUEL ROD TEST VARIABLES

	Hole A	Hole B	Hole C
Cell 1			
<u>Body 1</u> 200 μm (VSM) UC_2 + TRISO + 500 μm ThO_2 BISO	Matrix 1. ^(a) Composites of fertile particle batches.	Matrix 2. ^(b) + alternate additive. Buffer seal on particles.	Matrix 3. ^(c) Alternate shims. ^(d) Single fertile batches.
Cell 2			
<u>Body 2</u> ^(e) Matrix 1. ^(a) High vol % shim <u>Body 3</u> ^(e) Matrix 2. ^(b) + alternate additive Medium vol % shim <u>Body 4</u> ^(e) Matrix 3. ^(c) Low vol % shim	(VSM) UC_2 . Fissile and fertile particle coating variables. Composites of fissile and fertile particle batches.	(VSM) UC_2 . Alternate shims. ^(d) Fissile and fertile particle coating variables. Composites of fissile and fertile particle batches.	Recycle fissile fuel (WAR UC_2). Fertile particle coating variables.

- (a) Matrix 1 is 27% wt % Asbury 6353 natural-flake graphite plus A240 pitch plus ~5 wt % additive.
 (b) Matrix 2 is ≥ 32 wt % Lonza KS-15 graphite plus A240 pitch plus 15 wt % additive.
 (c) Matrix 3 is ≥ 32 wt % Lonza KS-15 graphite plus A240 pitch plus 5 wt % additive.
 (d) Alternate shims are TS-1240X and PoCo X4029A.
 (e) All rods in this body contain these items.

HT (target 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 period of time. The loose particle samples provided by GA for HT-28 and -29 were shipped to ORNL on November 15, 1974 and are currently scheduled for capsule assembly on December 2, 1974.

HFIR target experiments provide a rapid means of evaluating and screening fuel materials and material variables that are being considered for use in HTGRs. The primary attribute being evaluated in capsules HT-28 and -29 is BISO OPyC coatings deposited from mixed-gas (acetylene-propylene). This variation in fuel processing represents an evolutionary change and is considered necessary in order to fabricate coatings on a scale envisioned for the large HTGR. These samples supplement tests of BISO ThO particles with mixed-gas coatings now being tested in capsules P13R and P13S but represent a much broader spectrum of coating variables. The coating variables under investigation in these tests duplicate those investigated in an earlier series of HT-tests (HT-17 through -19) and, therefore, will provide a means for direct comparison of the irradiation performance of PyC coatings deposited from propylene and mixed gas (propylene-acetylene).

In addition, TRISO ThO₂ experiments are planned to obtain evaluation of a potential backup for the BISO ThO₂ design and obtain a quantitative direct comparison of the fission product retention between BISO and TRISO fuel particles.

The prime objectives of the GA samples will be to investigate OPyC coatings deposited from mixed gas (propylene-acetylene) on BISO ThO₂ particles. OPyC variables will be density, coating rate, and OPyC thickness. The effect of these variables on particle integrity and dimensional change will be evaluated. The samples will be selected from particle batches fabricated for other irradiation experiments and will duplicate variables tested in HT-17 through -19, which were propylene-coated ThO₂ BISO particles. One batch of propylene coated particles irradiated in capsules HT-17 through -19 will be included to provide a reference for interpreting and comparing the results of these two series of HT-experiments. Samples of TRISO coated

ThO₂ will be included in this test to evaluate this design as a potential backup for the BISO ThO₂ particles. Three batches will be irradiated to determine the effect of coating rate and coating thickness on OPyC failure. The samples will be irradiated at two nominal temperatures, 900° and 1250°C, over an estimated fast neutron exposure ($E > 0.18 \text{ MeV}$)_{HFIR} between 5.4 to $10.5 \times 10^{21} \text{ n/cm}^2$. The capsules and loading positions for the individual ThO₂ batches, along with estimated fuel burnups and fluences, are given in Table 9-4.

The particle batches were chosen so that only one variable (i.e., coating rate, coating density, or coating thickness) will be tested at a time. To aid in postirradiation analysis, the particles irradiated in HT-28 and -29 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. 9-2. 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 to characterize the OPyC thickness and coating rate of each sample. As a final step, particles are heated in vacuum for 1 hr at 1350°C to ensure volatilization of any of the fluids used for density separation. Preparing samples by these techniques also facilitates measuring particle irradiation-induced dimensional changes since their density and hence volume can be easily determined after irradiation. Confidence in "gross-effects" associated with different particle attributes is desired; the statistical basis for establishing a valid sample size and particle selection is referenced in Attachment 2 of the Approval-in-Principle for HFIR Experiments HT-26 and -27. A summary description of the particle samples irradiated in this series of HT-experiments is given in Table 9-5; Fig. 9-3 delineates the matrix for the BISO ThO₂ particles.

TABLE 9-4
PARTICLE DESIGNATION VERSUS DESIGN
CONDITIONS AND LOCATIONS IN HT-28 AND -29

Fast Fluence ($E > 0.18$ MeV) _{HFIR} ($\times 10^{21}$ n/cm ²)	Burnup (% FIMA)	Organ- ization	Particle Batch Designation ^(a)	Capsule Position	Particle Batch ^(a) Designation
CAPSULE HT-28, DESIGNATED TEMPERATURE 900°C					
4.3	4.5	GA	6252-04	2-51	6542-27
4.9	5.0	GA	6252-03	4-49	6542-34
5.2	5.2	GA	6542-28	5-48	6542-29
5.6	5.6	GA	6252-05	7-46	6542-35
5.8	5.8	ORNL	-	8-45	-
6.2	6.2	ORNL	-	10-43	-
6.5	6.4	GA/ORNL	6542-22	11-42	-
6.9	6.8	GA	6542-01	13-40	6542-24
CAPSULE HT-28, DESIGNATED TEMPERATURE 1250°C					
7.8	7.7	GA	6542-28	15-38	6542-27
8.1	7.9	GA	6542-30	17-36	6542-34
8.2	8.0	GA	6252-04	18-35	6542-29
8.3	8.2	GA	6252-03	20-33	6542-35
8.4	8.2	ORNL	-	21-32	-
8.5	8.3	ORNL	-	23-30	-
8.6	8.4	GA/ORNL	6542-22	24-29	-
8.6	8.4	GA	6542-01	26-27	6542-24
CAPSULE HT-29, DESIGNATED TEMPERATURE 900°C					
5.4	7.0	ORNL	-	2-51	-
6.1	7.5	ORNL	-	4-49	-
6.5	8.0	GA/ORNL	6542-22	5-48	-
7.0	8.6	GA	6542-01	7-46	6542-24
7.2	9.4	GA	6252-04	8-45	6542-27
7.7	9.5	GA	6252-03	10-43	6252-05
8.1	9.7	GA	6542-28	11-42	6542-29
8.4	10.2	GA	6542-30	13-40	6542-34
9.7	11.5	ORNL	-	15-38	-
10.1	11.8	GA	6252-04	17-36	6542-27
10.2	11.9	GA	6252-03	18-35	6542-34
10.3	12.1	GA	6252-05	20-33	6542-29
10.5	12.2	GA	6542-30	21-32	6542-35
10.6	12.4	ORNL	-	23-30	-
10.7	12.5	GA/ORNL	6542-22	24-29	-
10.7	12.5	GA	6542-01	26-27	6542-24

(a) Refer to Table 9-2, ORNL position.

TABLE 9-5

SUMMARY DESCRIPTION OF GA SAMPLES SELECTED
FOR IRRADIATION IN HT-28 AND -29

Sample Number (Parent Batch) ^(a)	Kernel Diameter (μm)	OPyC			Design for HT Particles		OPyC Particle Attributes Investigated
		Density (g/cm^3)	Coating Rate (C.R.) ($\mu\text{m}/\text{min}$)	Coating Thickness (μm)	Diameter (μm)	Density (g/cm^3)	
6542-01 ^(b)	500	1.80	10.0	85	790/833	3.45	Comparative reference ^(c)
6542-22 ^(d)	503	1.80	4.0	81	790/833	3.45	Low C.R., low density ^(c)
6542-24	511	1.94	6.6	83	790/833	3.45	Med. C.R., high density ^(c)
6542-27	512	1.86	4.6	81	790/833	3.45	Low C.R., med. density
6542-28	508	1.72	4.6	81	790/833	3.45	Low C.R., low density
6542-29	499	1.97	4.3	76	790/833	3.45	Low C.R., high density
6542-30	517	1.80	6.5	75	790/833	3.45	Med. C.R., low density
6542-34	505	1.86	6.4	80	790/833	3.45	Med. C.R., med. density
6542-35	505	1.94	5.6	70	790/833	3.45	Med. C.R., high density
6252-03 ^(e)	529	1.82	7.4	37	790/833	3.70	High C.R.
6252-04	528	1.82	5.6	39	790/833	3.70	Med. C.R.
6252-05	519	1.84	4.6	49	790/833	3.56	Low C.R., thick OPyC

(a) FMB-3 Manual.

(b) BISO coating with OPyC deposited from propylene gas. Sample previously irradiated in HT-17 through HT-19.

(c) Particle has buffer seal coating.

(d) BISO coating with OPyC deposited from mixed gas.

(e) TRISO coating with OPyC deposited from mixed gas; all samples have the same substrate from the SiC layer inward.

The HT-28 and -29 capsules will contain a total of 64 different positions, 32 in each capsule. General Atomic experiments will occupy 22 positions in each capsule. The positions in capsule HT-29 are designated for operation at 900°C over a fast fluence increment between 5.4 to 10.7×10^{21} n/cm² ($E > 0.18$ MeV)_{HFIR}. The loose particle locations in capsule HT-28 are divided into two sections: 16 positions at 1250°C to receive an estimated fast fluence between 7.8 and 8.6×10^{21} n/cm² ($E > 0.18$ MeV)_{HFIR} and 16 positions at 900°C bracketed for a 4.3 to 6.9×10^{21} n/cm² ($E > 0.18$ MeV)_{HFIR} fast fluence exposure. The irradiation positions in HT-28 and -29 are uninstrumented, and the designated temperatures in these capsules are design graphite magazine temperatures. A thermal study comparable to the analysis on HT-17, -18, and -19 will be provided by ORNL to specify and control the operating conditions of HT-28 and -29. Irradiation exposure for these capsules is scheduled for 4 to 5 cycles (over a 4- to 5-month period) beginning February 1, 1975.

Postirradiation examination of the particles, to be conducted in the GA Hot Cell Facility, will include the following:

1. Visual observation to determine and categorize OPyC failure fractions.
2. Radiography and metallography.
3. Determination of distribution of fission products and dopants throughout particles (optional).
4. Determination of burnup.

FUEL TEST ELEMENT FTE-14

The main test objective for FTE-14 was the short-term high-temperature irradiation of full-size HTGR fuel rods and loose-bed particle batches containing alternate and near-reference HTGR fuel types. The peak fluence and temperature for FTE-14 were $\sim 1.2 \times 10^{21}$ n/cm² ($E \geq 0.18$ MeV) and $\sim 1450^\circ\text{C}$, respectively. In conjunction with this test, FTE-15, a companion test element containing the same type fuels, received approximately twice the exposure at the same temperatures. Test capsules P13N and P13P, which also contained the same fuels, received moderate and full exposures, respectively, at a variety of temperatures.

Of particular interest in this experiment was the behavior of the reference UC₂ (200 μm) VSM and alternate reference type UC₂ (314 μm) WAR kernels. The other fuel types listed in Table 9-6 and Table 9-9 (see section on spine samples) provided experience with alternate fuel designs and manufacturing conditions. Several of the fuels tested provided in-pile data for determining the initiation points for metallic fission product attack of the SiC and for amoeba migration of the kernels. Particles from thermal stability spine samples are also presently being tested to determine out-of-pile migration rates of irradiated particles.

Fuel Rod Irradiation Performance

Sample Description

The fuel rods were fabricated by the reference Fort St. Vrain hot-injection process using natural-flake graphite filler and pitch binder. None of the rods contained shim or inert particles. Firing was done at 1800°C in alumina-packed beds. The fired rods were nominally 0.615 in. in diameter and 2.5 in. long. The rods were loaded in three H-327 graphite bodies in a six-hole telephone dial arrangement (Table 9-7). The fuel rods selected for initial examination were from the center of body 2, which coincides with the highest fast fluence and temperature conditions. Preliminary data on the irradiation conditions are given in Ref. 9-2.

TABLE 9-6
FTE-14 FUEL PARTICLE SPECIFICATIONS^(a)

Item ^(b)	FMB No.	Kernel		As-Manufactured Coating Parameters						OPyC Density (g/cm ³)	OPTAF ^(c)	SiC Density (g/cm ³)
		Type	Nominal Diameter (μm)	Type	Nominal Thickness (μm)							
					Buffer	IPyC	SiC	OPyC	Total			
12	4162-00-031	UO ₂	200	TRISO	92	26	26	37	175	1.83	1.07	3.18
13	4252-00-011	ThO ₂	400	BISO	73	--	--	70	156	2.00	1.14	--
14	4161-00-011	UC ₂	145	TRISO	47	23	22	25	112	1.82	1.09	3.20
15	4261-00-011	ThC ₂	358	TRISO	54	35	25	38	135	1.81	1.14	3.19
16	4162-01-021	UO ₂	150	TRISO	51	24	26	25	123	1.86	1.09	3.19
17	4251-00-031	ThC ₂	368	BISO	68	--	--	54	122	1.98	1.16	--
18	4262-00-051	ThO ₂	400	TRISO	71	16	35	25	138	1.81	1.08	3.21
21	4161-01-031	UC ₂	200	TRISO	87	28	29	38	162	1.80	1.11	3.20
23	4163-00-011	(Th,U)O ₂ ^(d)	250	TRISO	69	32	36	72	218	1.77	1.09	3.20
24	4251-01-021	ThC ₂	433	BISO	82	--	--	64	133	1.96	1.13	--

(a) All values are from Fuel Materials Branch (FMB) data retrieval.

(b) Refer to Table 9-7.

(c) Optical anisotropy factor.

(d) Th/U ratio = 1.0.

TABLE 9-7

FUEL ROD PARTICLE LOADING COMBINATIONS FOR FTE-14^(a)

Upper End	Fuel Hole Number	Item Number ^(b)	Fissile Particle	Item Number	Fertile Particle
Body No. 3	<u>Body No. 3</u>				
	1	21	200- μm UC_2 TRISO	17	400- μm ThC_2 BISO
	2	12	200- μm UO_2 TRISO	18	400- μm ThO_2 TRISO
	3	12	200- μm UO_2 TRISO	13	400- μm ThO_2 BISO
	4	23	250- μm $(\text{Th},\text{U})\text{O}_2$ TRISO	24	450- μm ThC_2 BISO
	5	16	150- μm UO_2 TRISO	24	450- μm ThC_2 BISO
Body No. 2	6	12	200- μm UO_2 TRISO	15	400- μm ThC_2 TRISO
	<u>Body No. 2</u>				
	1	21	200- μm UC_2 TRISO	17	400- μm ThC_2 BISO
	2	21	200- μm UC_2 TRISO	15	400- μm ThC_2 TRISO
	3	12	200- μm UO_2 TRISO	13	400- μm ThO_2 BISO
	4	23	250- μm $(\text{Th},\text{U})\text{O}_2$ TRISO	24	450- μm ThC_2 BISO
Body No. 1	5	16	150- μm UO_2 TRISO	24	450- μm ThC_2 BISO
	6	12	200- μm UO_2 TRISO	18	400- μm ThO_2 TRISO
	<u>Body No. 1</u>				
	1, 2, 3	14	150- μm UC_2 TRISO	24	450- μm ThC_2 BISO
	4, 5, 6				

^(a) From Ref. 9-1.^(b) Refer to Table 9-6.

FUEL HOLE IDENTIFICATION

Fuel Rod Examination

The purpose of the postirradiation examination was to determine the integrity of both the fuel rods and the fuel particles. Fuel failures from manufacturing defects and irradiation exposure were characterized, and the thermal effects of metallic fission product attack of the SiC and amoeba migration of the kernels were examined. The fuel rod integrity was characterized through visual examination. The extent of particle failure was determined from fission gas release measurements and metallography. Metallography and autoradiography were used to aid in characterizing the fuel failure modes.

Visual Examination. Representative fuel rods for five fuel blends were photographed with the Bausch and Lomb stereomicroscope (Figs. 9-4 through 9-8). In all cases, small cracks were evident on the side surfaces, with larger cracks at the matrix end cap and some spalling at the edge. Some damage to the surface of the fuel rods during unloading in the form of striations along their lengths resulted in debonded and broken particles. An approximate count of broken particles on the surface was made for each rod and is included in Table 9-8.

Fission Gas Release Measurements. Release to birth (R/B) ratios for Kr-85m were determined from fission gas release measurements made in the Triga II facility at 1100°C. The results are given in Table 9-8. Assuming the fractional release to be 1% for a failed fissile particle, the fraction of failed fissile particles in each of the rods that was examined would be <1% except for the UO_2 TRISO - ThO_2 BISO blend with 1.3% failure. These results are based on the assumption that the fertile particles had little effect on the R/B measurement because of low burnups.

The number of broken particles noted in the visual examination is also a significant contribution to the R/B measurement. Assuming an R/B release value of 1 to 4% for a broken particle (Ref. 9-3) and that half of the observed broken particles are fissile particles, the magnitude of surface

failure observed in the visual examination could contribute ~0.1 to 1% fissile failure to the R/B measurement.

Metallographic Examination. Five fuel rods with different fuel blends underwent metallographic examination. In all cases the fuel performed well. Photomicrographs of the radial cross sections and the matrix are shown in Figs. 9-9 through 9-13. The irradiation-induced coating failure was low in most cases and no matrix-particle coating interaction was observed. Failure fractions are shown in Table 9-8. These values are not all attributed to irradiation damage. SiC failure in all cases was partially attributed to polishing. The extent of this contribution could not be estimated metallographically, but because of low fission gas release results, it was considered to be significant.

Fuel rod 2-2-6 had a UC_2 (200 μm) VSM TRISO - ThC_2 TRISO blend. Both the fissile and the fertile particles showed buffer densification and buffer debonding, as indicated in Figs. 9-14a and 9-14b. It was not possible to obtain a significant value for the failure fraction of the ThC_2 particles due to the onset of hydrolysis during the examination. This rod also showed a nonhomogeneity in loading with a high concentration of ThC_2 particles near the matrix end cap. The UC_2 particles showed concentrations of metallic fission products in the IPyC coating, as seen in Fig. 9-14c, but no SiC attack was observed.

SiC attack was observed in a UC_2 (200 μm) VSM particle in fuel rod 2-1-7. However, of the 74 particles examined in the rod, there was only one case, in which the reaction zone extended 3 to 5 μm into the SiC (Fig. 9-15). With this one exception, the fuel looked excellent. Buffer densification and buffer debonding were observed in the UC_2 particles; the fertile ThC_2 BISO particles looked excellent in all respects (Fig. 9-16).

The UO_2 TRISO - ThO_2 BISO blend in rod 2-3-6 looked good. Buffer densification and buffer debonding were evident in the UO_2 particles (Fig. 9-17a). The fertile ThO_2 particles looked excellent (Fig. 9-17b).

TABLE 9-8
SUMMARY OF POSTIRRADIATION EXAMINATION OF FTE-14 FUEL RODS

Fuel Rod Identification (a)	Fissile Particle		Fertile Particle		EOL Peak Fuel Temp (c)	EOL Fast Fluence (d) ($\times 10^{21}$ n/cm ²)	Fissile Particle FIMA (e) (%)	Fission Gas Release (g/g) (Kr-85m) (f)	Dimensional Analysis (g) (% Change)		Visual Examination		Metallographical Examination							
													Fissile Particle				Fertile Particle			
	Type	Item (b)	Type	Item (b)					Diameter	Length	Appearance (h)	Number of Broken Particles on Surface	OPyC Failure (%)	95% Confidence Limits P(%)	SiC Failure (%)	95% Confidence Limits P(%)	OPyC Failure (%)	95% Confidence Limits P(%)	SiC Failure (%)	95% Confidence Limits P(%)
2-1-5	UC ₂ TRISO	21	ThC ₂ BISO	17	1480	1.27		2.68×10^{-4}			Good	78	-	-	-	-	-	-	-	-
2-1-6	UC ₂ TRISO	21	ThC ₂ BISO	17	1480	1.26		3.84×10^{-4}			Good	42	-	-	-	-	-	-	-	-
2-1-7	UC ₂ TRISO	21	ThC ₂ BISO	17	1370	1.24		1.01×10^{-4}			Good	55	1.4	$0.0 < P < 7.5$	0	$0.0 < P < 5.3$	0	$0.0 < P < 1.8$	-	-
2-2-5	UC ₂ TRISO	21	ThC ₂ BISO	17	1480	1.27		5.3×10^{-5}			Good	7	-	-	-	-	-	-	-	-
2-2-6	UC ₂ TRISO	21	ThC ₂ TRISO	15	1480	1.26		4.4×10^{-5}			Good	18	2.4	$0.8 < P < 6.7$	3.9	$1.7 < P < 8.9$	(1)	-	(1)	-
2-2-11	UC ₂ TRISO	21	ThC ₂ TRISO	15	1210	1.12		8.0×10^{-5}			Good	22	-	-	-	-	-	-	-	-
2-3-6	UO ₂ TRISO	12	ThO ₂ BISO	13	1480	1.26		5.3×10^{-5}			Good	1	2.0	$0.7 < P < 6.1$	2.0	$0.7 < P < 6.1$	0	$0.0 < P < 1.4$	-	-
2-4-6	(Th,U)O ₂ TRISO	23	ThC ₂ BISO	24	1480	1.26		-			Good	18	-	-	-	-	-	-	-	-
2-4-7	(Th,U)O ₂ TRISO	23	ThC ₂ BISO	24	1370	1.24		9.8×10^{-6}			Good	19	3.0	$1.0 < P < 8.4$	0	$0.0 < P < 3.7$	0	$0.0 < P < 3.1$	-	-
2-4-11	(Th,U)O ₂ TRISO	23	ThC ₂ BISO	24	1210	1.12		3.4×10^{-5}			Excellent	2	-	-	-	-	-	-	-	-
2-6-5	UO ₂ TRISO	12	ThO ₂ TRISO	18	1480	1.27		1.3×10^{-4}			Excellent	14	-	-	-	-	-	-	-	-
2-6-7	UO ₂ TRISO	12	ThO ₂ TRISO	18	1370	1.24		1.3×10^{-4}			Good	15	2.0	$0.5 < P < 6.9$	0	$0.0 < P < 3.7$	0	$0.0 < P < 4.1$	12.8	$7.6 < P < 20.8$

(a) Fuel body - Hole No. - Fuel Rod No. (No. 11 is at the top of the fuel body).

(b) Refer to Table 9-6.

(c) Calculated value using graphite thermal conductivity = 16 BTU/hr-ft-°F.

(d) Preliminary data.

(e) Physics calculation (under preparation).

(f) Corrected for steady state.

(g) Under evaluation.

(h) Does not take into account unloading damage.

(i) Not determined because of hydrolysis of ThC₂.

Buffer densification and buffer debonding were seen in both the UO_2 TRISO and ThO_2 TRISO particles of rod 2-6-7 (Fig. 9-18a and 9-18b). The 12.7% SiC failure of the ThO_2 particles was concentrated in the matrix end cap portion of the rod. This may have been caused by fabrication damage or polishing. Because of the good integrity of the same particles outside of this area, irradiation-induced failure was not considered to be the cause.

This same fertile particle batch (4262-00-05x) was also irradiated in capsule P13N to $3.4 \times 10^{21} \text{ n/cm}^2$ at 1255°C and showed 4.7% OPyC and 15% SiC failure from metallographic examination. The results in this case were also questionable because of the possible damage during polishing (Ref. 9-4). The susceptibility of these particles to polishing damage is thought to be due to the relative thinness of the coatings.

$(\text{Th,U})\text{O}_2$ kernel migration in several particles was evident in rod 2-4-7, which had a $(\text{Th,U})\text{O}_2$ TRISO - ThC_2 BISO blend (Fig. 9-19a). The direction of the migration tended to be random, which was attributed to a nonuniform temperature gradient throughout the rod. In other respects, the fuel showed low failure and excellent irradiation performance. Buffer densification and buffer debonding were evident in both the fertile and fissile particles, as seen in Figs. 9-19a and 9-19b.

Thermal Stability Spine Samples

Test Configuration

Fuel test element FTE-14 contained nine thermal stability Type 2 crucibles (Fig. 9-20). The loading of these crucibles is shown in Table 9-9. The spine samples that were selected for examination are shown in Table 9-10. Of particular interest were the UC_2 (200 μm) VSM and UC_2 (314 μm) WAR particles.

TABLE 9-9
LOADING OF FTE-14 COATED PARTICLE SPINE SAMPLES

Distance from Reactor Bottom (in.)	Crucible Number ^(a)	Hole Number
17-1/4	TS1	1, 2, 3, 4, 5, 6
18-1/2	TS2	1, 2, 3, 4, 5, 6
18-3/4	TS3	1, 2, 3, 4, 5, 6
37-3/4	TS4	1, 2, 3, 4, 5, 6
46	TS5	1, 2, 3, 4, 5, 6
49-1/4	TS6	1, 2, 3, 4, 5, 6
67-3/4	TS7	1, 2, 3 (empty)
68-1/2	TS8	1, 2, 3, 4, 5, 6
69-3/4	TS9	1, 2, 3, 4, 5, 6

FMB Batch Number ^(b)	Crucible Number - Hole Number	Prior Capsules
4252-01-070	3-4, 6-1	P13P
4252-00-010	1-5, 4-1, 5-1, 9-3	P13N, P13P
4262-00-050	1-2, 5-2, 9-4	P13N, P13P
4162-00-020	1-6, 4-3, 8-4, 5-3	P13P
4162-01-020	2-4, 6-4	P13P, P13N
4163-00-010	2-3, 6-3, 9-2	P13P
5466-37	3-3, 4-5, 5-5	P13N
4413-67	1-1, 7-1	P13L, P13M, FTE-1,2,3
4493-149	2-1, 4-4, 5-4, 8-5	FTE-1,2,3
4413-35	3-1, 8-1	P13L
4413-5	3-2, 7-3	P13L
4161-00-020	1-4, 8-2	P13N, P13P
4161-01-030	3-6, 4-6, 5-6, 8-3	P13N, P13P, FTE-14,15
4000-304	2-2, 6-2, 9-5	P13M
OR-1694	2-6, 6-6, 9-1	P13P
OR-1602	8-6	P13N
4423-89	2-3, 3-5, 4-2, 7-2	P13L
4413-127	2-5, 6-5, 9-6	FTE-1,2,3 spine Thermal Performance Lab

(a) All Type 2 crucibles.

(b) Fuel Materials Branch retrieval number.

TABLE 9-10
SUMMARY OF POSTIRRADIATION EXAMINATION OF FTE-14 SPINE SAMPLES

FMB Particle Batch Number ^(a)	Particle Type	Location ^(b)	EOL Peak Temp ^(c) (°C)	Fast Fluence (x 10 ²¹ n/cm ²)	FIMA (%)	Number Particles in Sample	Stereo Examination ^(d)		Metallographic Examination ^(e)							
							OPyC Failure (%)	SiC Failure (%)	OPyC Failure (%)	95% Confidence Limits P(%)	SiC Failure (%)	95% Confidence Limits P(%)	Metallic Fission Products in IPyC (%)	95% Confidence Limits P(%)	SiC Attack (%)	95% Confidence Limits P(%)
4161-01-030	TRISO UC ₂ (200 μm) VSM	TS3-6	1000	1.11		1056	0	0	0	0.0<P<4.5	0	0.0<P<4.5	19	9.0<P<27	0	0.0<P<4.5
4161-01-030	TRISO UC ₂ (200 μm) VSM	TS5-6	1300	1.25		1026	0	0	0	0.0<P<4.2	3.2	1.0<P<8.5	95	85.0<P<100.0	4.4	1.0<P<10.0
4161-01-030	TRISO UC ₂ (200 μm) VSM	TS4-6	1410	1.28		1049	0	0	0	0.0<P<22.0	5.3	0.0<P<25.0	100	12.0<P<28.0	47	24.0<P<71.0
OR-1694	TRISO UC ₂ (314 μm) WAR	TS2-6	970	1.07		883	7.3	0	8.7	6.0<P<12.6	1	0.1<P<2.0	0	0.0<P<1.3	0	0.0<P<1.3
OR-1694	TRISO UC ₂ (314 μm) WAR	TS6-6	1290	1.22		63	7.0	0.7	3.7	2.0<P<6.9	0	0.0<P<1.6	0	0.0<P<1.6	0	0.0<P<1.6
4252-01-070	BISO ThO ₂	TS6-1	1290	1.22		221	0	-	0	0.0<P<9.0	-	-	-	-	-	-

(a) Fuel Materials Branch retrieval number.

(b) See Table 9-9.

(c) EOL peak fuel temperature (calculated) minus 70°C.

(d) All particles in the spine sample were examined.

(e) Approximately 10 to 20% of samples examined, except for TS4-6 which had only 19 particles.

Spine Sample Postirradiation Examination

Postirradiation examination of the spine samples included visual examination, radiography, and metallography.

Visual Examination. Figures 9-21 through 9-23 are representative stereophotographs of the visual appearance of the spine samples. Of most importance in these photographs is the coating failure of the alternate reference large HTGR UC_2 (314 μm) WAR particles from spine samples TS2-6 and TS6-6. At 970°C (TS2-6) a 7% OPyC failure was identified and at 1290°C (TS6-6) a 7% OPyC failure and a 0.7% SiC failure were identified in the stereomicroscope examination. This same phenomenon was observed in P13P fuel rods with the same particles irradiated to $5 \times 10^{21} \text{ n/cm}^2$ at $\sim 1000^\circ\text{C}$. Density and OPTAF measurements are planned on both the irradiated and unirradiated particles to characterize this problem.

The ThO_2 particles (TS6-1) and the UC_2 (200 μm) VSM particles (TS3-6, TS4-6, and TS5-6) all looked excellent with no observable failure (Figs. 9-22 and 9-23).

Radiography. Contact radiographic plates were made of two batches of 20 particles each of the seven spine samples submitted for initial examination. Representative photographs of these samples are shown in Figs. 9-24 through 9-26.

The UC_2 (200 μm) VSM particles showed metallic fission product buildup in the IPyC coating and SiC attack in some cases, as seen in Fig. 9-24. The UC_2 WAR particles showed a buildup of metallic fission products in the recoil zone of the buffer layer next to the kernel (Fig. 9-25). No migration of fission products beyond this zone was observed. The ThO_2 particles in Fig. 9-26 showed no failure.

Metallography. Metallographic examination was performed on representative particles from the spine samples listed in Table 9-10. The major

observations in this examination were the coating failure in the UC_2 (314 μm) WAR particles and the metallic fission product attack of the SiC in the UC_2 (200 μm) VSM particles.

The UC_2 (200 μm) VSM particles were examined from three longitudinal positions in the test element, corresponding to $\sim 1000^\circ$, 1300° , and 1410°C . In all cases there were some concentrations of metallic fission products in the OPyC coating (Fig. 9-27). SiC attack by the metallic fission products with a reaction zone of several microns was evident in the two high-temperature samples (Fig. 9-28). The extent of this effect is summarized in Table 9-10. It is evident that at higher temperatures the amount of metallic fission products and their attack on the SiC coating increases. This effect also allows some insight into the relative operating temperatures of the test element. The fact that the fuel rods which had the same UC_2 particles and were in approximately the same longitudinal position as the spine sample showed little SiC attack would lead to the conclusion that the spine samples were running at a higher temperature. Preliminary thermal analysis (Ref. 9-2) of this test element, on the other hand, shows the spine region to be running approximately 70°C lower than the fuel rods. A possible explanation could be that the microprofile of the spine region has temperature peaks in the immediate region of the sample, while the average center-line temperature is lower. This is caused by low thermal conductivity in the space between the loose bed of particles and the graphite crucible and the concentration of particles in a relatively small area.

The UC_2 (314 μm) WAR spine samples corresponding to 970°C and 1290°C were examined. The 8.7% OPyC and 1% SiC failures at 970°C and the 3.7% OPyC and 0% SiC failures at 1290°C are consistent with observations in the stereomicroscope examination if the 95% confidence levels are taken into account. Because the whole sample is counted in the stereomicroscope examination, the failure fractions of 7.3% OPyC and 0% SiC at 970°C and 7.0% OPyC and 0.7% SiC at 1290°C are more statistically valid than the determinations from metallography.

Representative photomicrographs of particles from the two spine positions are shown in Fig. 9-29. Dark field views of the particles showed no migration of metallic fission products into the IPyC, which is in agreement with P13P results (Ref. 9-2). It is believed that the ~ 0.8 wt % O_2 inhibits metallic fission product migration by forming stable metal oxides of the rare earth fission products.

The last spine sample examined contained ThO_2 BISO particles (batch 5466-149). These same particles showed close to 100% OPyC failure in capsule P13P fuel rods (Ref. 9-2) at an exposure of 6.0×10^{21} n/cm². Because of the low exposure in this sample (i.e., 1.22×10^{21} n/cm²) no coating failures were observed; however, the dark field microphotograph in Fig. 9-30b reveals a significant amount of anisotropy from interference effects.

Discussion and Conclusions

Most of the fuel tested in FTE-14 behaved in a consistent manner with current fuel design (Ref. 9-5). The extent of pressure vessel failure, fuel kernel migration, and metallic fission product attack was very limited as would be expected at these low burnups (~ 25 to 30% FIMA fissile, $\sim 0.3\%$ FIMA fertile).

In all cases except for the UC_2 (314 μ m) WAR particles, the irradiation-induced failure was low ($<1\%$). The high OPyC failure of the UC_2 (314 μ m) WAR particles is attributed to a pyrocarbon coating problem of this particle batch and is under investigation.

Fuel kernel migration assuming a maximum HTGR thermal gradient of 260°C/cm and the 1490°C maximum temperature of the test element should have caused ThC_2 , UO_2 , and UC_2 kernel migration, based on design curves of irradiated particles. The examination showed no migration in these type kernels. This indicates that at low burnups the current design curves for kernel migration are conservative. Migration that was observed in the

(Th,U)O₂ particles shows that the (Th,U)O₂ kernels have a relatively high kernel migration factor.

Metallic fission product attack was consistent with previous in-pile and out-of-pile data. Capsule P13P showed limited surface attack at 1300°C with full burnups in UC₂ (200 μm) VSM particles (Ref. 9-6). Out-of-pile thermal annealing of particles from 1300° to 1500°C shows that full-burnup UC₂ particles have limited SiC attack (Ref. 9-7). The observations of the SiC attack by metallic diffusion products at 1400° to 1500°C with burnups of ~25 to 30% in FTE-14 will aid in the characterization of the kinetics of the SiC attack when sufficient out-of-pile and in-pile data can be collected. No SiC attack was observed in the UO₂, (Th,U)O₂, or O₂-doped UC₂ (314 μm) WAR kernel types. This is consistent with the fact that oxygen in the kernels ties up rare earth fission products as oxides, which prevents metallic fission product attack of the SiC.

In summary, the following observations were made from the postirradiation examination of FTE-14.

1. Fuel rod integrity and appearance were good in all cases at a peak fluence and temperature of $\sim 1.2 \times 10^{21} \text{ n/cm}^2$ ($E \geq 0.18 \text{ MeV}$) and $\sim 1490^\circ\text{C}$, respectively.
2. Irradiation-induced failure of the fuel at a peak fluence and temperature of $\sim 1.2 \times 10^{21} \text{ n/cm}^2$ ($E \geq 0.18 \text{ MeV}$) and 1490°C were low.
3. The (Th,U)O₂ kernels showed kernel migration of up to 15 μm into the buffer at peak fluence and temperature conditions of $\sim 1.2 \times 10^{21} \text{ n/cm}^2$ ($E \geq 0.18 \text{ MeV}$) and $\sim 1490^\circ\text{C}$.
4. Only one case of SiC attack, with a reaction zone thickness of 5 μm, was observed in all of the fuel rods examined. This was a UC₂ (200 μm) VSM particle at the peak fluence and temperature conditions of $\sim 1.2 \times 10^{21} \text{ n/cm}^2$ ($E \geq 0.18 \text{ MeV}$) and 1490°C .

5. UC_2 (314 μm) WAR particles showed up to $\sim 7\%$ OPyC failure in the spine samples at 970° and 1290°C .
6. Loose UC_2 (200 μm) VSM particles irradiated in the spine samples at longitudinal positions corresponding to 1000° , 1300° , and 1410°C showed SiC attack by metallic fission products up to 5 μm into the surface. With increasing temperature the effect was more widely spread, but in no case did it cause SiC failure.
7. From the fact that no SiC attack was observed in rods containing UC_2 (200 μm) VSM particles that supposedly were operating at a higher temperature than the spine samples, the present thermal analysis (Ref. 9-2) is considered insufficient for characterizing a microprofile in the spine sample region.

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*THERMOCOUPLE RODS

Fig. 9-1. Sample layout for capsule P13T

Fig. 9-2. Steps involved in selecting BISO coated ThO₂ particles for insertion in irradiation experiments HT-28 and -29

OPyC COATING DENSITY (CONSTANT COATING RATE)

OPyC COATING RATE (CONSTANT COATING DENSITY)

*PARTICLES HAVE BUFFER SEAL COATING

[†]OPyC DEPOSITED FROM PROPYLENE GAS

Fig. 9-3. Schematic shim test matrix for BISO ThO_2 particles

S7359 (114,115,120)

S7359 (112,113,119)

S7359-116

S7359-118

Fig. 9-4. Visual examination of FTE-14 fuel rod 2-1-7 (UC_2 TRISO - ThC_2 TRISO blend). Irradiation conditions: $1.24 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1370^\circ\text{C}$. This rod was broken during handling.

S7359 (18-20)

S7359 (14-16)

S7359-11

S7359-12

Fig. 9-5. Visual examination of FTE-14 fuel rod 2-2-6 (UC_2 TRISO - ThC_2 TRISO blend). Irradiation conditions: $1.26 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1480^\circ\text{C}$.

S7359 (56-58)

S7359 (59-61)

S7359-54

S7359-55

Fig. 9-6. Visual examination of FTE-14 fuel rod 2-3-6 (UO_2 TRISO - ThO_2 BISO blend). Irradiation conditions: $1.26 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1480^\circ\text{C}$.

S7359 (62-64)

S7359 (65-67)

S7359-70

S7359-71

Fig. 9-7. Visual examination of FTE-14 fuel rod 2-4-7 [(Th,U)O₂ TRISO - ThC₂ BISO blend]. Irradiation conditions: 1.24×10^{21} n/cm² (E > 0.18 MeV) at ~1370°C.

S7359 (39-41)

S7359 (43-45)

S7359-36

S7359-37

Fig. 9-8. Visual examination of FTE-14 fuel rod 2-6-7 (UO_2 TRISO - ThO_2 TRISO blend). Irradiation conditions: $1.24 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1370^\circ\text{C}$.

L7359-290

(a)

L7359 (276-288)

(b)

Fig. 9-9. Photomicrographs of FTE-14 fuel rod 2-1-7 (UC₂ TRISO - ThC₂ TRISO blend). Irradiation conditions: 1.24×10^{21} n/cm² (E > 0.18 MeV) at ~1370°C. (a) Typical appearance of matrix graphite and (b) composite radial cross section showing ThC₂ beginning to hydrolyze.

L7359-124

(a)

L7359 (109-122)

(b)

Fig. 9-10. Photomicrographs of FTE-14 fuel rod 2-2-6 (UC₂ TRISO - ThC₂ TRISO blend).
Irradiation conditions: 1.26×10^{21} n/cm² ($E > 0.18$ MeV) at $\sim 1480^\circ\text{C}$.
(a) Typical graphite matrix and (b) composite radial cross section.

L7359-254

(a)

L7359 (46-59)

(b)

Fig. 9-11. Photomicrographs of FTE-14 fuel rod 2-3-6 (UO_2 TRISO - ThO_2 BISO blend).
Irradiation conditions: $1.26 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1480^\circ\text{C}$.
(a) Typical graphite matrix and (b) composite radial cross section.

L7359-192

(a)

L7359 (195-208)

(b)

Fig. 9-12. Photomicrographs of FTE-14 fuel rod 2-4-7 [(Th,U)O₂ TRISO - ThC₂ BISO blend].
Irradiation conditions: 1.24×10^{21} n/cm² (E > 0.18 MeV) at ~1370°C.
(a) Typical appearance of matrix graphite and (b) composite radial cross section.

L7359-82

(a)

L7359 (55-68)

(b)

Fig. 9-13. Photomicrographs of FTE-14 fuel rod 2-6-7 (UO_2 TRISO - ThO_2 TRISO blend).
Irradiation conditions: 1.24×10^{21} n/cm² ($E > 0.18$ MeV) at $\sim 1370^\circ\text{C}$.
(a) Typical graphite matrix and (b) composite radial cross section.

Fig. 9-14. Representative particles from FTE-14 fuel rod 2-2-6. Irradiation conditions: $1.26 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1480^\circ\text{C}$. (a) ThC₂ (400 μm) TRISO, (b) UC₂ (200 μm) VSM TRISO, and (c) the same particle as in (b) with dark field showing metallic fission products in the IPyC coating. No SiC interaction was observed.

L7359-259 (a)

L7359-260 (b)

L7359-261 (c)

Fig. 9-15. UC_2 particle in fuel rod 2-1-7 showing SiC attack. Irradiation conditions: $1.24 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1370^\circ\text{C}$. (a) Light field, (b) dark field showing metallic fission products in IPyC coating, and (c) area of SiC attack.

L7359-255

Fig. 9-16. Representative UC_2 TRISO and ThC_2 BISO particles from fuel rod 2-1-7. Irradiation conditions: $1.24 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at $\sim 1370^\circ\text{C}$.

L7359-222

(a)

L7359-220

(b)

Fig. 9-17. Representative particles from fuel rod 2-3-6. Irradiation conditions: 1.26×10^{21} n/cm² ($E > 0.18$ MeV) at $\sim 1480^\circ\text{C}$. (a) UO_2 (200 μm) TRISO and (b) UO_2 TRISO/ ThO_2 (400 μm) BISO.

L7359-34

(a)

L7359-42

(b)

Fig. 9-18. Representative particles from fuel rod 2-6-7. Irradiation conditions: 1.23×10^{21} n/cm² ($E > 0.18$ MeV) at $\sim 1370^\circ\text{C}$.
 (a) UO_2 (200 μm) TRISO and (b) ThO_2 (400 μm) TRISO.

L7359-193

(a)

L7359-188

(b)

Fig. 9-19. Representative particles from fuel rod 2-4-7. Irradiation conditions: 1.23×10^{21} n/cm² ($E > 0.18$ MeV) at $\sim 1370^\circ\text{C}$. (a) ThC₂ (450 μm) BISO that has partially hydrolyzed and (b) (Th,U)O₂ (250 μm) TRISO that showed some kernel migration into the buffer.

Fig. 9-20. Design of type 2 thermal stability crucibles

S7359-90

(a)

S7359-101

(b)

Fig. 9-21. Stereophotographs of UC_2 ($314 \mu\text{m}$) WAR particles from spine samples TS2-6 and TS6-6. Irradiation conditions: (a) $1.07 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at 970°C in location TS2-6 and (b) $1.22 \times 10^{21} \text{ n/cm}^2$ at 1290°C in location TS6-6. Both samples showed $\sim 7\%$ OPyC failure.

Fig. 9-22. Stereophotographs of the UC_2 (200 μm) VSM particles from spine samples TS3-6, TS4-6, and TS5-6. Irradiation conditions: (a) 1.11×10^{21} n/cm 2 ($E > 0.18$ MeV) at 1000°C in TS3-6, (b) 1.25×10^{21} n/cm 2 at 1300°C in TS5-6, and (c) 1.28×10^{21} n/cm 2 at 1410°C in TS4-6.

S7359-99

Fig. 9-23. Stereophotograph of ThO₂ particles from spine sample TS6-1. Irradiation conditions: 1.22×10^{21} n/cm² ($E > 0.18$ MeV) at 1290°C.

Fig. 9-24. Radiographs of representative UC_2 (200 μm) VSM particles from spine samples TS3-6, TS4-6, and TS5-6. (a) TS3-6 irradiated to 1.11×10^{21} n/cm² at 1000°C, (b) TS5-6 irradiated to 1.25×10^{21} n/cm² at 1300°C showing SiC attack, and (c) TS4-6 irradiated to 1.28×10^{21} n/cm² at 1410°C showing metallic fission product buildup in IPyC coating.

HA 322

(a)

HA 320

(b)

Fig. 9-25. Radiographs of representative UC_2 ($314 \mu\text{m}$) WAR particles from spine samples TS2-6 and TS6-6. (a) TS2-6 irradiated to $1.07 \times 10^{21} \text{ n/cm}^2$ at 970°C showing metallic fission products in recoil zone in the buffer and (b) TS6-6 irradiated to $1.22 \times 10^{21} \text{ n/cm}^2$ at 1290°C also showing a buildup fission products in the recoil zone.

HA 318

Fig. 9-26. Radiograph of a representative ThO₂ BISO particles from spine sample TS6-1. Irradiation conditions: 1.22×10^{21} n/cm² at 1290°C.

L7359-149

(a)

L7359-150

(b)

Fig. 9-27. Photomicrographs of representative UC_2 (200 μm) VSM particles from spine sample TS5-6. Irradiation conditions: $1.25 \times 10^{21} \text{ n/cm}^2$ at 1250°C . (a) Bright field and (b) dark field showing metallic fission product concentration in lower section of the IPyC coating.

L7359-158

(a)

L7359-160

(b)

Fig 9-28. Photomicrograph of representative UC_2 (200 μm) VSM particles from spine sample TS5-6. The SiC attack by metallic fission products illustrated was characteristic of particles from both the 1300° and 1410°C spine samples (i.e., TS5-6 and TS4-6).

Fig. 9-29. Photomicrographs of representative UC_2 (314 μm) WAR particles from spine samples TS2-6 and TS6-6. Irradiation conditions: 1.07×10^{21} n/cm² at 970°C (TS2-6) and 1.22×10^{21} n/cm² at 1290°C (TS6-6). (a) TS2-6, (b) TS2-6 particle showing total coating failure, and (c) TS6-6.

L7359-170

(a)

L7359-171

(b)

Fig. 9-30. Photomicrographs of a representative ThO_2 BISO particle from spine sample TS6-1. Irradiation conditions: $1.22 \times 10^{21} \text{ n/cm}^2$ at 1290°C . (a) Bright field and (b) dark field showing anisotropy of OPyC (especially at buffer interface).

TASK 11 (189a 13121)

GRAPHITE RESEARCH

GRAPHITE CHARACTERIZATION

Characterization of grades TS-1240 and H-451, manufactured by Union Carbide Corporation and Great Lakes Carbon Company, respectively, is continuing. Initial characterization data (density, impurities, tensile properties, thermal expansion, and conductivity) were reported previously (Refs. 11-1 through 11-4). Additional purity and tensile property data for both grades are given in Tables 11-1 through 11-7. Additional thermal conductivity data for grade TS-1240 are given in Table 11-8.

The tensile property data presented in Tables 11-1 through 11-5 and Table 11-7 were obtained on specimens 0.505 in. in diameter by 3.0 in. long. The tensile property data presented in Table 11-6 were obtained on specimens 0.25 in. in diameter by 0.90 in. long. Each specimen of both types 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. Additional stress cycles (to 1000 psi) on selected specimens showed no further increases in elastic modulus. Values of ultimate strength, strain to fracture, and permanent deformation in the specimens after the first loading cycle were also obtained from the stress-strain curve.

CAPSULE IRRADIATIONS

Capsule OG-2

Capsule OG-2 continues to operate satisfactorily, although two of the tungsten-rhenium thermocouples (out of a total of 31 thermocouples) have

ceased to give reliable readings. The most recent temperature readings are all within 50°C of the operating temperature listed in Table 11-5 of the previous Quarterly Progress Report (Ref. 11-3). The current schedule calls for removal of the capsule from the Oak Ridge Reactor at the end of cycle 122 (January 5, 1975).

Capsule OG-3

Capsule OG-3 will duplicate the design of capsules OG-1 and OG-2 except for minor changes in the pattern of sample holes in the graphite crucibles and the addition of two thermocouples. Many samples previously irradiated in capsules OG-1 and OG-2 will be transferred to OG-3 after nondestructive tests to obtain isothermal data at increasing neutron exposures.

A Test Plan (TP-325-004) listing the samples and tests to be performed has been written and is being circulated for approval. The plan allocates most of the capsule space to dimensional and mechanical property samples of the two major near-isotropic candidate graphites for HTGR fuel and reflector blocks: Great Lakes Carbon Grade H-451 and Union Carbide Grade TS-1240. Additional space is assigned to side-by-side comparisons of dimensional changes in these two graphites and Airco-Spear Grade SO-818, Pechiney P₃JHAN, and SigriAS2-500 graphites.

THERMAL CONDUCTIVITY OF IRRADIATED GRAPHITES

A topical report (Ref. 11-5) describing the thermal conductivity changes in graphites irradiated in capsules OG-1, GEH-13-421, and GEH-13-422 was issued.

GRAPHITE STANDARD

A draft of an RDT standard E6-1 "Near-Isotropic Petroleum-Coke-Based Graphites for High-Temperature Gas-Cooled Reactor Core Components," was written and reviewed by representatives of the major carbon companies and

TABLE 11-1
IMPURITY CONTENT (IN PPM) OF GRAPHITES H-451 AND TS-1240

Sample Number (Graphite: Log Number)	Location in Log ^(a)	Ash	B	Fe	V	Ti	S
(H-451:5651-86)							
5651-86-3A-L005	MLC	<53	1.0	4.0	<0.5	<1.0	<1.5
-L007	MLC	<72	2.0	<1.0	<0.5	<1.0	<2.0
-L009	MLE	<350 ^(b)	2.0	2.0	<0.5	<1.0	<3.0
-L010	MLE	<500 ^(b)	<u>2.0</u>	<u><1.0</u>	<u><0.5</u>	<u><1.0</u>	<u><2.0</u>
Mean	ML position	-	1.75	<2.0	<0.5	<1.0	<2.1
(TS-1240:5651-73)							
5651-73-3A-L005	MLC	<98	<0.5	<1.0	10.0	10.0	<2.0
-L007	MLC	<39	<0.5	4.0	10.0	10.0	<2.0
-L009	MLE	<30	<0.5	<1.0	20.0	10.0	22.0
-L010	MLE	< <u>31</u>	<u><0.5</u>	<u><1.0</u>	<u>10.0</u>	<u>8.0</u>	<u>10.0</u>
Mean	ML position	<42	<0.5	<1.7	12.5	9.5	<9.0
(TS-1240:5651-72)							
5651-74-3A-L082	MLC	94	1.0	<10.0	20.0	20.0	5
-3B-L094	MLC	120	1.0	<10.0	10.0	20.0	5
-3A-L156	MLE	140	1.0	<10.0	10.0	10.0	10
-3B-L165	MLE	92	1.0	<10.0	10.0	10.0	13
-1A-L031	EC	68	1.0	<10.0	20.0	20.0	22
-1B-L043	EC	69	1.0	<10.0	10.0	10.0	18
-1A-L106	EE	110	<0.5	<10.0	10.0	10.0	<7
-1B-L115	EE	<u>120</u>	<u><0.5</u>	<u><10.0</u>	<u>10.0</u>	<u>10.0</u>	<u>25</u>
Mean	Whole log	89	<0.9	<10.0	12.5	13.7	<13

TABLE 11-1 (Continued)

Sample Number (Graphite: Log Number)	Location in Log ^(a)	Ash	B	Fe	V	Ti	S
(TS-1240:5651-75)							
5651-75-3A-L082	MLC	95	<0.5	<10.0	10.0	20.0	5
-3B-L094	MLC	81	<0.5	<10.0	10.0	20.0	18
-3A-L156	MLE	74	<0.5	<10.0	20.0	20.0	15
-3B-L165	MLE	47	<0.5	<10.0	10.0	20.0	13
-1A-L031	EC	55	<0.5	<10.0	10.0	20.0	9
-1B-L043	EC	89	<0.5	<10.0	20.0	20.0	18
1A-L106	EE	58	<0.5	<10.0	10.0	10.0	17
-1B-L115	EE	<u>39</u>	<u><0.5</u>	<u><10.0</u>	<u>20.0</u>	<u>20.0</u>	<u>18</u>
Mean	Whole log	67	<0.5	<10.0	13.7	18.7	14.2
(TS-1240:5651-74)							
5651-74-3A-L082	MLC	<44	<1.0	<10.0	<10.0	<10.0	<1.5
-3B-L094	MLC	<28	<1.0	<10.0	60.0	40.0	<1.5
-3A-L156	MLE	<35	<1.0	<10.0	40.0	20.0	5.5
-3B-L165	MLE	<27	2.0	<10.0	<10.0	<10.0	3.5
-1A-L031	EC	<50	<1.0	<10.0	<10.0	<10.0	3.5
-1B-L043	EC	<40	<1.0	<10.0	40.0	40.0	2.5
-1A-L106	EE	53	<1.0	<10.0	<10.0	<10.0	3.5
-1B-L115	EE	<u><35</u>	<u><1.0</u>	<u><10.0</u>	<u>40.0</u>	<u>40.0</u>	<u>5.0</u>
Mean	Whole log	<39	<1.1	<10.0	<27.5	<22.5	<3.2

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

(b) Insufficient sample for more accurate analysis.

TABLE 11-2

TENSILE PROPERTIES OF TS-1240 GRAPHITE

Log 5651-72, density = 1.74 g/cm³, 0.505-in.-diameter by 3.0-in.-long samples

Specimen Number	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	EC(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		<u>0.75</u>	<u>0.93</u>	--	<u>0.227</u>	<u>1661</u>
Mean		0.80	0.93	--	0.224	1631
Std. Dev.		0.116	0.090	--	0.053	260
5651-72-1A-L028	EC(1)	0.67	0.84	0.030	0.363	1909
-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		<u>0.92</u>	<u>0.98</u>	--	<u>0.310</u>	<u>2071</u>
Mean		0.84	0.97	--	0.300	2010
Std. Dev.		0.091	0.065	--	0.046	130
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		<u>0.78</u>	<u>1.03</u>	<u>0.023</u>	<u>0.240</u>	<u>1643</u>
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
-L078	↓	0.71	0.88	0.030	0.350	1997
-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		<u>0.79</u>	<u>0.96</u>	<u>0.021</u>	<u>0.320</u>	<u>1997</u>
Mean		0.77	0.97	0.020	0.342	2053
Std. Dev.		0.029	0.057	0.006	0.020	73

TABLE 11-2 (Continued)

Specimen Number	Position and Orientation ^(a)	Elastic Modulus $\times 10^{-6}$ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5615-72-1A-L101	EE(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		0.78	0.99	--	0.228	1712
Mean		0.84	1.05	--	0.275	2008
Std. Dev.		0.031	0.046	--	0.030	150
5651-72-1A-L128	EE(1)	0.74	0.92	0.032	0.361	2050
-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-L144		0.78	0.88	0.020	0.370	2089
-L143		0.78	0.94	--	0.330	2070
-L146		0.84	1.00	--	0.324	2165
-L147		0.84	0.97	--	0.305	2027
Mean		0.83	0.97	--	0.337	2110
Std. Dev.		0.106	0.054	--	0.022	100
5651-72-3A-L152	MLE(11)	0.76	0.96	0.025	0.305	1893
-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-L170		0.82	1.01	0.020	0.288	1912
-L162		0.77	1.00	0.024	0.286	1866
-L164		0.84	1.02	0.017	0.240	1762
-L166		0.81	1.05	0.020	0.220	1624
Mean		0.83	1.05	0.021	0.275	1906
Std. Dev.		0.051	0.080	0.005	0.032	166
5651-72-3A-L178	MLE(1)	0.74	1.05	0.025	0.328	2113
-L181	↓	0.73	0.96	0.031	0.400	2241
-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		0.84	0.99	0.010	0.320	2153
Mean		0.79	0.99	0.016	0.330	2084
Std. Dev.		0.048	0.045	0.009	0.038	146

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

TABLE 11-3

TENSILE PROPERTIES OF TS-1240 GRAPHITE

Log 5651-74, density = 1.79 g/cm³, 0.505-in.-diameter by 3.0-in. long samples

Specimen Number	Position and Orientation ^(a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-74-1A-L001	EC(II) ↓	0.99	1.15	0.018	0.250	1944
-L004		0.90	1.20	0.028	0.333	2418
-L005		1.00	1.18	0.013	0.310	2344
-L009		0.92	1.18	0.019	0.330	2404
-1B-L011		0.96	1.23	0.025	0.295	2256
-L014		0.91	1.14	0.020	0.265	1987
-L015		0.90	1.15	0.023	0.320	2298
-L019		0.95	1.18	0.023	0.315	2342
Mean		0.94	1.18	0.021	0.302	2249
Std. Dev.		0.040	0.030	0.005	0.030	183
5651-74-1A-L025	EC(I) ↓	0.86	1.17	0.032	0.365	2462
-L026		0.86	1.10	0.023	0.310	2145
-L027		0.84	1.02	0.025	0.280	1948
-L030		0.87	1.14	0.024	0.450	2980
-1B-L044		0.83	0.94	0.020	0.260	1946
-L038		0.83	1.11	0.029	0.378	2534
-L039		0.83	1.07	0.030	0.360	2369
-L042		0.82	1.05	0.025	0.450	2843
Mean		0.84	1.08	0.026	0.363	2441
Std. Dev.		0.018	0.073	0.004	0.063	345
5651-74-3A-L050	MLC(II) ↓	0.93	1.10	0.018	0.306	2238
-L053		0.97	1.12	0.014	0.285	2186
-L054		0.89	1.08	0.021	0.275	2010
-L058		0.88	1.11	0.019	0.260	1914
-3B-L061		0.89	1.06	0.020	0.221	1661
-L064		0.83	1.10	0.025	0.271	1890
-L065		0.81	1.05	0.024	0.286	1943
-L069		0.83	1.06	0.024	0.284	1944
Mean		0.88	1.08	0.021	0.273	1973
Std. Dev.		0.054	0.026	0.004	0.025	180
5651-74-3A-L075	MLC(I) ↓	0.74	0.97	0.045	0.206	1445
-L076		0.78	1.02	0.024	0.250	1692
-L077		0.78	1.02	0.026	0.260	1749
-L080		0.78	1.03	0.028	0.265	1505
-3B-L087		0.83	1.03	0.026	0.261	1835
-L088		0.83	1.05	0.022	0.335	2284
-L089		0.84	1.03	0.022	0.221	1630
-L092		0.82	1.01	0.021	0.189	1372
Mean		0.80	1.02	0.027	0.248	1689
Std. Dev.		0.035	0.023	0.008	0.045	287

TABLE 11-3 (Continued)

Specimen Number	Position and Orientation ^(a)	Elastic Modulus $\times 10^{-6}$ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-74-1A-L101	EE(11)	0.97	1.18	0.018	0.308	2369
-L103	↓	1.00	1.18	0.016	0.350	2666
-L105		0.95	1.18	0.022	0.335	2494
-L107		0.88	1.10	0.020	0.295	2110
-1B-L111		0.88	1.12	0.018	0.230	1794
-L112		0.85	1.08	0.023	0.306	2148
-L114		0.84	1.07	0.023	0.325	2197
-L116		<u>0.85</u>	<u>1.11</u>	<u>0.027</u>	<u>0.318</u>	<u>2244</u>
Mean		0.90	1.13	0.021	0.308	2253
Std. Dev.		0.062	0.046	0.004	0.036	264
5651-74-1A-L125	EE(1)	0.89	1.08	0.016	0.372	2544
-L126	↓	0.85	1.07	0.025	0.314	2150
-L129		0.89	1.08	0.019	0.366	2510
-L130		0.85	1.08	0.024	0.327	2300
-1B-L142		0.88	1.11	0.021	0.410	2760
-L143		0.86	1.08	0.021	0.354	2406
-L146		0.83	1.01	0.022	0.337	2247
-L147		<u>0.84</u>	<u>1.06</u>	<u>0.017</u>	<u>0.325</u>	<u>2185</u>
Mean		0.86	1.07	0.021	0.351	2388
Std. Dev.		0.023	0.028	0.003	0.031	208
5651-74-3A-L151	MLE(11)	0.95	1.18	0.020	0.336	2489
-L153	↓	0.94	1.18	0.022	0.356	2586
-L155		0.90	1.12	0.020	0.301	2157
-L157		0.91	1.12	0.016	0.275	2064
-3B-L161		0.83	1.10	0.024	0.292	1995
-L162		0.87	1.10	0.020	0.253	1603
-L164		0.86	1.10	0.021	0.305	2199
-L166		<u>0.96</u>	<u>1.15</u>	<u>0.030</u>	<u>0.332</u>	<u>2458</u>
Mean		0.90	1.13	0.022	0.306	2194
Std. Dev.		0.046	0.034	0.004	0.034	320
5651-74-3A-L175	MLE(1)	0.83	1.02	0.023	0.300	2004
-L176	↓	0.80	1.03	0.020	0.291	1981
-L179		0.80	1.06	0.029	0.259	1785
-L180		0.83	1.05	0.030	0.275	1861
-3B-L193		0.82	1.06	0.026	0.262	1796
-L194		0.86	1.06	0.020	0.292	1987
-L197		0.80	0.98	0.021	0.179	1248
-L198		<u>0.77</u>	<u>0.98</u>	<u>0.025</u>	<u>0.196</u>	<u>1347</u>
Mean		0.81	1.03	0.024	0.257	1754
Std. Dev.		0.027	0.034	0.004	0.045	288

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

TABLE 11-4
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	2338
-L014		1.03	1.25	0.015	0.272	2190
-L015		1.00	1.20	0.016	0.270	2130
-L019		1.00	1.30	0.020	0.245	2038
Mean		1.03	1.25	0.015	0.250	2056
Std. Dev.		0.040	0.041	0.003	0.031	171
5651-75-1A-L025	EC(1) ↓	1.00	1.22	0.008	0.323	2390
-L026		0.90	1.08	0.022	0.280	2007
-L027		0.87	1.07	0.020	0.155	1235
-L030		0.92	1.12	0.019	0.320	2319
-1B-L037		0.89	1.11	0.020	0.387	2600
-L038		0.87	1.08	0.020	0.278	1996
-L039		0.85	1.07	0.023	0.288	1997
-L042		0.95	1.11	0.016	0.397	2793
Mean		0.91	1.11	0.018	0.303	2167
Std. Dev.		0.049	0.049	0.005	0.075	479
5651-75-3A-L051	MLC(11) ↓	1.06	1.23	0.013	0.325	2590
-L053		1.25	1.47	0.005	0.305	2592
-L055		1.05	1.27	0.020	0.275	2281
-L058		1.14	1.41	0.013	0.314	2605
-3B-L062		1.00	1.25	0.025	0.378	2836
-L066		1.08	1.23	0.012	0.335	2851
-L067		1.02	1.25	0.023	0.330	2650
-L068		1.08	1.27	0.015	0.327	2450
Mean		1.08	1.30	0.016	0.323	2606
Std. Dev.		0.079	0.091	0.006	0.029	187
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 11-4 (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-L102	EE(11) ↓	0.97	1.14	0.018	0.270	2050
-L103		0.97	1.16	0.017	0.308	2339
-L105		0.91	1.13	0.020	0.260	1994
-L107		0.94	1.08	0.016	0.183	1467
-1B-L111		0.92	1.11	0.018	0.335	2335
-L112		0.95	1.10	0.020	0.320	2353
-L118		0.86	1.05	0.017	0.335	2341
-L116		0.93	1.10	0.017	0.277	2089
Mean		0.93	1.11	0.286	0.286	2121
Std. Dev.		0.036	0.035	0.001	0.051	303
5651-75-1A-L125	EE(1) ↓	0.90	1.07	0.017	0.335	2301
-L126		0.78	1.07	0.022	0.290	2037
-L129		0.96	1.14	0.009	0.310	2300
-L131		0.92	1.14	0.017	0.293	2125
-1B-L142		0.93	1.00	0.018	0.308	2208
-L143		0.86	1.03	0.020	0.330	2113
-L146		0.88	1.07	0.020	0.340	2310
-L147		0.95	1.05	0.018	0.290	2175
Mean		0.89	1.07	0.018	0.312	2196
Std. Dev.		0.058	0.049	0.004	0.021	102
5651-75-3A-L151	MLE(11) ↓	1.06	1.27	0.013	0.356	2828
-L153		1.10	1.34	0.015	0.340	2703
-L158		0.99	1.17	0.017	0.345	2597
-L160		1.02	1.22	0.019	0.360	2752
-3B-L161		1.15	1.27	0.016	0.400	3152
-L162		1.02	1.29	0.015	0.365	2928
-L167		1.13	1.27	0.015	0.295	2469
-L169		1.14	1.27	0.009	0.295	2578
Mean		1.08	1.26	0.015	0.344	2751
Std. Dev.		0.062	0.050	0.003	0.035	218
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.

TABLE 11-5

TENSILE PROPERTIES OF TS-1240 GRAPHITE

Log 5651-73, density = 1.79 g/cm³, 0.505-in.-diameter by 3.0-in.-long samples

Specimen Number	Position and Orientation ^(a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-73-3A-L002A	MLC(II) ↓	0.95	0.97	0.009	0.242	1837
-L002B		0.90	1.00	0.008	0.255	1873
-L003A		0.94	1.00	0.007	0.230	1658
-L003B		0.90	1.02	0.010	0.230	1724
-3B-L002A		0.94	1.03	0.010	0.231	1714
-L002B		0.97	1.10	0.012	0.258	1969
Mean		0.93	1.02	0.009	0.241	1796
Std. Dev.		0.028	0.044	0.002	0.013	117
5651-73-3A-L005	MLC(I) ↓	0.96	1.00	0.000	0.185	1452
-L006		0.88	0.97	0.005	0.227	1665
-L007		-	-	-	-	1316
-L008		0.83	1.00	0.017	0.158	1233
-3B-L005		0.86	1.22	0.020	0.208	1560
-L007		0.90	1.02	0.017	0.192	1388
Mean		0.87	1.04	0.012	0.194	1436
Std. Dev.		0.049	0.101	0.009	0.026	159
5651-73-3A-L009A	MLE(II) ↓	1.01	1.18	0.018	0.342	2583
-L011A		0.90	1.12	0.030	0.317	2287
-L010A		1.01	1.15	0.015	0.365	2684
-3B-L011A		0.95	1.10	0.031	(b)	(b)
-L009A		0.95	1.17	0.016	0.255	2204
-L009B		0.96	1.15	0.014	0.345	2498
Mean		0.98	1.16	0.016	0.327	2290
Std. Dev.		0.032	0.015	0.002	0.049	357
5651-73-3A-L013	MLE(I) ↓	0.86	0.99	0.015	0.292	2004
-L014		0.82	1.01	0.019	0.257	1730
-L015		0.86	1.02	0.016	0.255	1755
-3B-L013		0.85	1.03	0.019	0.285	1939
-L014		0.93	1.07	0.012	0.248	1790
-L015		0.86	1.18	0.016	0.260	1805
Mean		0.86	1.05	0.016	0.266	1837
Std. Dev.		0.036	0.069	0.003	0.018	109

(a) MLC = midlength center, MLE = midlength edge.

(b) Glue joint failure.

TABLE 11-6

TENSILE PROPERTIES OF TS-1240 GRAPHITE
Log 5651-73, density = 1.79 g/cm³, 0.25-in.-diameter by 0.9-in.-long samples

Specimen Number	Position and Orientation (a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permaent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-73-3A-106B	MLE(11) ↓	0.82	1.06	0.020	0.440	2789
-107B		0.92	1.02	0.006	0.365	2569
-112B		0.86	1.06	0.017	0.383	2400
-113B		0.92	1.03	0.007	0.318	2262
-118B		0.86	1.06	0.022	0.315	2120
-119B		0.72	0.98	0.032	0.375	2267
-124B		0.80	0.97	0.020	0.375	2244
-125B		0.92	0.99	0.010	0.355	2347
-127B		0.82	0.98	0.025	0.365	2183
-128B		0.75	1.01	0.030	0.356	2042
-3B-106B		0.75	0.95	0.070	0.260	1540
-107B		0.82	1.00	0.035	0.290	2082
-112B		0.95	1.12	0.035	0.350	2526
-113B		1.06	1.20	0.015	0.390	2648
-118B		1.00	1.20	0.020	0.360	2588
-119B		0.82	1.06	0.030	0.410	2648
-124B		0.82	1.06	0.040	0.410	2699
-125B		0.86	1.06	0.030	0.380	2608
-127B		0.69	-	-	0.130	971
-128B		0.98	1.29	0.010	0.370	2777
Mean		0.86	1.06	0.025	0.350	2315
Std. Dev.		0.096	0.089	0.014	0.066	440
5651-73-3A-136B	MLE(1) ↓	1.12	1.38	0.020	0.190	1608
-137B		0.64	0.82	0.035	0.275	1524
-142B		0.95	1.12	0.015	0.290	2061
-143B		0.72	0.95	0.040	0.420	2397
-148B		0.90	1.00	0.005	0.355	2302
-149B		0.90	1.06	0.015	0.320	1884
-154B		0.78	1.00	0.040	0.260	1711
-155B		0.67	0.82	0.030	0.325	1732
-160A		0.82	0.90	0.030	0.250	1671
-160B		1.00	1.00	0.015	0.350	2486
-3B-136B		0.86	1.06	0.030	0.320	2119
-137B		0.75	1.00	0.040	0.340	2040
-142B		0.82	0.95	0.030	0.310	1992
-143B		0.69	0.95	0.045	0.390	2207
-148B		0.78	1.00	0.035	0.390	2322
-149B		0.82	1.00	0.020	0.310	2207
-154B		0.86	1.12	0.020	0.240	1848
-155B		0.75	0.90	0.030	0.290	1788
-160A		0.78	0.95	0.035	0.290	1714
-160B		0.95	0.95	0.020	0.320	2205
Mean		0.83	1.00	0.027	0.312	1991
Std. Dev.		0.119	0.121	0.011	0.055	285

TABLE 11-6 (Continued)

Specimen Number	Position and Orientation ^(a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-73-3A-006B	MLC(11) ↓	0.82	0.90	0.015	0.210	1427
-007B		0.78	0.95	0.030	0.270	1711
-012B		0.90	1.00	0.025	0.310	2015
-013B		0.82	0.90	0.025	0.320	1896
-018B		0.75	0.86	0.020	0.200	1469
-019B		0.64	0.69	0.015	0.285	1406
-024B		0.82	0.95	0.010	0.275	1796
-025B		0.78	1.00	0.020	0.250	1691
-027B		0.75	0.86	0.010	0.245	1449
-028B		0.78	0.90	0.020	0.265	1672
-3B-006B		0.89	1.00	0.017	0.323	1976
-007B		0.86	1.00	0.025	0.245	1670
-012B		0.95	1.00	0.010	0.185	1438
-013B		0.82	0.95	0.030	0.145	1141
-018B		0.62	0.60	0.030	0.240	1080
-019B		0.72	0.86	0.022	0.190	1283
-024B		0.78	0.88	0.020	0.215	1487
-025B		0.82	0.90	0.030	0.295	1569
-027B		0.72	0.86	0.015	0.275	1507
-028B		0.75	0.86	0.027	0.335	1731
Mean		0.79	0.90	0.021	0.254	1571
Std. Dev.		0.081	0.102	0.007	0.051	252
5651-73-3A-036B	MLC(1) ↓	0.69	-	-	0.140	957
-037B		0.78	0.95	0.025	0.220	1428
-042B		0.82	1.00	0.035	0.205	1367
-043B		0.69	0.78	0.030	0.300	1266
-048B		0.67	0.72	0.030	0.245	1388
-049B		1.00	1.00	0.020	0.130	1120
-054B		0.75	0.90	0.015	0.210	1450
-055B		0.78	0.95	0.020	0.250	1531
-060A		0.95	1.06	0.040	0.350	2400
-060B		0.67	0.90	0.010	0.175	1170
-3B-036B		0.56	0.75	0.035	0.330	1448
-037B		0.64	-	-	0.165	958
-042B		0.75	0.95	0.030	0.210	1428
-043B		0.67	0.88	0.025	0.195	1287
-048B		0.77	0.90	0.030	0.252	1594
-049B		0.76	0.95	0.017	0.255	1491
-054B		0.71	1.00	0.047	0.205	1384
-055B		0.70	0.91	0.030	0.233	1428
-060A		0.64	0.83	0.025	0.295	1428
-060B		0.59	0.68	0.015	0.255	1144
Mean		0.73	0.90	0.027	0.231	1383
Std. Dev.		0.107	0.105	0.009	0.058	298

(a) MLE - midlength edge, MLC = midlength center.

TABLE 11-7
TENSILE PROPERTIES OF H-451 GRAPHITE
Log 5651-86, density = 1.74 g/cm³, 0.505-in.-diameter by 3.0-in.-long samples

Specimen Number	Position and Orientation (a)	Elastic Modulus x 10 ⁻⁶ (psi)		Permanent Set After First Loading (%)	Fracture Strain (%)	Ultimate Strength (psi)
		First Loading	Second Loading			
5651-86-3A-L002A	MLC () ↓	0.95	1.13	0.010	0.250	1965
-L002B		0.98	1.14	0.010	0.275	2163
-L003A		1.07	1.25	0.008	0.245	2129
-L003B		1.07	1.23	0.010	0.225	2009
-3B-L002A		0.93	0.97	0.022	0.268	2014
-L002B		<u>0.90</u>	<u>0.93</u>	<u>0.014</u>	<u>0.265</u>	<u>1984</u>
Mean		0.98	1.11	0.012	0.254	2044
Std. Dev.		0.072	0.132	0.005	0.018	82
5651-86-3A-L005	MLC (⊥) ↓	0.90	1.07	0.013	0.204	1648
-L006		0.90	1.06	0.013	0.255	1862
-L007		0.85	1.02	0.012	0.145	1135
-L008B		0.88	1.06	0.013	0.205	1623
-3B-L005		0.84	0.99	0.018	0.215	1551
-L007		<u>0.77</u>	<u>0.94</u>	<u>0.022</u>	<u>0.210</u>	<u>1488</u>
Mean		0.86	1.02	0.015	0.206	1551
Std. Dev.		0.049	0.051	0.004	0.035	240
5651-86-3A-L009A	MLE () ↓	1.11	1.32	0.011	0.315	2857
-L009B		1.11	1.32	0.011	0.315	2774
-L010A		1.25	1.38	0.007	0.301	2789
-L010B		1.29	1.36	0.000	0.335	3038
-3B-L009A		1.00	1.18	0.017	0.310	2486
-L012		<u>1.11</u>	<u>1.34</u>	<u>0.020</u>	<u>0.310</u>	<u>2646</u>
Mean		1.14	1.32	0.011	0.314	2765
Std. Dev.		0.106	0.071	0.007	0.011	187
5651-86-3A-L013	MLE (⊥) ↓	0.92	1.10	0.012	0.255	1956
-L014		0.96	1.09	0.010	0.220	1882
-L015		0.92	0.94	0.006	0.195	1629
-3B-L013		0.81	1.00	0.018	0.285	1961
-L014		0.80	0.97	0.022	0.285	1893
-L015		<u>0.80</u>	<u>0.96</u>	<u>0.025</u>	<u>0.245</u>	<u>1724</u>
Mean		0.87	1.01	0.015	0.247	1841
Std. Dev.		0.073	0.069	0.007	0.036	135

(a) MLC = midlength center, MLE = midlength edge.

TABLE 11-8
THERMAL CONDUCTIVITY OF TS-1240 GRAPHITE
(Log 5651-73, density = 1.78 g/cm³)

Specimen Number	Orientation	Location in Log	Thermal Conductivity (cal/cm-sec-°C)				
			22°C	200°C	400°C	600°C	800°C
5651-73-3A-L001M	Parallel ↓	Midlength center ↓	0.211	0.204	0.185	0.159	0.139
-L001N			0.219	0.216	0.174	0.155	0.139
-L001O			0.226	0.213	0.180	0.160	0.143
-L001P			0.230	0.221	0.199	0.164	0.143
-L001Q			0.225	0.221	0.188	0.165	0.144
-L001R			0.245	0.229	0.192	0.163	0.139
-L001S			0.274	0.250	0.218	0.176	0.154
-L001T			<u>0.243</u>	<u>0.233</u>	<u>0.204</u>	<u>0.180</u>	<u>0.157</u>
Mean			0.234	0.223	0.192	0.165	0.145
Std. Dev.			0.020	0.014	0.014	0.008	0.007
5651-73-3A-L008A	Perpendicular ↓	Midlength center ↓	0.242	0.230	0.203	0.166	0.158
-L008B			0.244	0.233	0.188	0.161	0.144
-L008C			0.258	0.246	0.202	0.179	0.154
-L008D			0.230	0.224	0.184	0.161	0.141
-L008E			0.260	0.230	0.207	0.175	0.151
-L008F			0.251	0.231	0.190	0.163	0.143
-L008G			0.246	0.227	0.193	0.168	0.143
-L008H			<u>0.249</u>	<u>0.229</u>	<u>0.197</u>	<u>0.166</u>	<u>0.147</u>
Mean			0.247	0.231	0.195	0.167	0.148
Std. Dev.			0.010	0.006	0.008	0.006	0.006

other designated organizations. Comments from the major carbon companies revealed a conflict in areas where commercial proprietary information was required. The controversial areas centered around testing the raw materials and revealing processing sequences. These subjects were discussed among representatives of RRD, GA, Great Lakes Carbon Company, AirCo Speer, and Union Carbide Corp. in a meeting at RRD Headquarters in August 1974. A new draft will be written and submitted for approval in early 1975.

ETR CELL CALCULATIONS

Introduction

The fast neutron fluence (exposure) for graphite irradiation samples has usually been reported in terms of integrated flux with energies equal to or higher than 180 keV. Recent observations indicate that neutrons in the 50 to 180 keV range contribute to the fast fluence, i.e., to the graphite damage (Ref. 11-6). Also, it is recognized that the spectra of the experimental reactor and a typical HTGR are significantly different, especially in the latter energy range (Ref. 11-7). The flux measurements were carried out with detectors, e.g., Fe-54, with relatively high threshold energies (2 to 3 MeV) for activation reactions and then related to the integrated flux by means of an effective reaction cross section, which was calculated analytically for a desired lower cutoff energy. Consequently, the flux spectrum at a detector plays an important role in determining the fast neutron fluence.

The objectives of this study were (1) to recalculate the effective cross section, using the most up-to-date differential cross sections, and the flux spectrum at the detector location in the experimental reactor, (2) to calculate the flux spectrum at the sample location, and (3) to establish a new criterion for the maximum allowable fast neutron fluence.

Method of Solution

Analytical Model

Out of a multitude of experiments, it was decided to analyze experimental capsule GEH-13-421, located in the I-12 position of the ETR, during cycle 109A. The choice was dictated by the following considerations:

(1) the capsule had been assigned one of the highest priorities for further analysis of fluence data (Ref. 11-8), and (2) the cycle conditions and the availability of information allowed the construction of a meaningful analytical model. The latter consideration is of a critical nature - the ETR core is very complex in structure, has undergone numerous cycles (for which detailed records are not always readily available), and has contained various experiments, often of an undisclosed nature, during each cycle.

The I-12 position of cycle 109A of the ETR was ideal to analyze, since the fuel loading chart, shown in Fig. 11-1, indicated that the capsule was surrounded by identifiable elements; i.e., no other experiment was immediately adjacent. Also, control rods 13 and 15 were not used (except at startup) during this cycle.

The power distribution for this cycle (Fig. 11-2), measured in the ETR critical facility, indicated that the radial fluxes at the core midplane were relatively flat. Consequently, it was decided to set up and analyze a cell (with reflective boundaries) composed of only nine elements. Since the ETR is a water-moderated thermal reactor, it is reasonable to assume that a 9 in. by 9 in. cell is large enough not to be unduly affected by flux perturbations outside the cell boundaries. The flatness of the power distribution also supports the validity of the cell model.

The calculational model is shown in Fig. 11-3. The nuclide concentrations were calculated from the data in Fig. 11-1. It was assumed that fuel

	E	F	G	H	I	J	K	L	M	N
5		E380H 499.2 2.63	XE254G ^b 505.3 3.73	⁵ C185E 226.2 2.05	E365H 499.7 2.62	⁶ C151E 230.2 2.20	E362H 499.5 2.61	XE114G ^b 493.3 3.42	XE242G ^b 497.0 3.41	XE211G ^b 499.9 3.55
6	E388H 499.4 2.66			E387H 499.6 2.66		E361H 499.6 2.60				XE201G ^b 500.3 3.68
7	E369H 499.6 2.67			¹ C249E 228.2 1.08	E371H 499.3 2.66	⁷ C172E 227.1 1.93	E364H 499.5 2.58			E349H 499.7 3.65
8	E373H 499.8 2.64			E390H ^a 499.5 2.70	E346H ^a 499.6 3.61		E252H 499.7 3.75		E391H 499.5 2.66	
9	⁸ C209E 230.3 2.19	E36CH 499.6 2.57	⁹ C169E 226.4 1.85	E366H ^a 499.5 2.58	¹⁰ C156E 230.6 1.88	E351H ^a 499.6 3.66	² C102E 229.2 1.71	E333H 499.6 4.72	¹¹ C68E 229.4 1.73	E367H 499.4 2.67
10	XE200G ^b 499.0 3.43		E370H 499.4 2.62		E228H ^a 499.8 3.74		E368H 499.7 2.69		E382H 499.5 2.70	
11	¹² C165E 228.7 2.21	E308H 499.7 3.65	³ C196E 227.2 2.46	E375H 499.6 2.69	¹³ C128E 229.2 2.51	E130H 499.6 3.51	¹⁴ C202E 229.5 1.95	XE104G ^b 492.7 3.43	⁴ C193E 225.4 2.00	E374H 499.8 2.68
12	XE267G ^b 504.4 3.73	E363H 499.5 2.54		E124H ^a 499.6 3.47	GEH	4X SFR	E384H 499.7 2.68		E415H 499.5 4.71	XE207G ^b 496.8 3.43
13	XE204G ^b 498.7 3.50			¹⁵ C223E 227.9 2.60	4X SFR	XE270G ^b 503.6 3.71	¹⁶ C81E 230.1 2.39			XE180G ^b 504.9 3.44
14		XE217G ^b 495.2 3.70	E337H 499.8 3.68	E300H 499.6 3.75	E377H 499.6 2.68		XE266G ^b 501.7 3.69	XE276G ^b 500.5 3.69	E357H 499.5 3.60

Fuel Element

(position)
Control Rod

Experiment

1st No.: fuel element or control section serial number
 2nd No.: U-235 content, g
 3rd No.: natural boron content, g
 a: location of the six highest power positions
 b: boron foil-type fuel elements

Fig. 11-1. ETR cycle 109A fuel loading

Fig. 11-2. Five-level relative power densities for Cycle 109D mockup (ETRC run 109D-4). The top-to-bottom order in each cell corresponds to +14, +7, 0, -7, and -14 in. with respect to the reactor horizontal midplane.

(1) SW DETECTOR

(2) NE DETECTOR

(S₁) NW GRAPHITE SAMPLE

(S₂) NE GRAPHITE SAMPLE

(S₃) SW GRAPHITE SAMPLE

(S₄) SE GRAPHITE SAMPLE

VALUES GIVEN ARE
 10^{24} ATOMS/CM³

Fig. 11-3. ETR cell model

was fully enriched; i.e., U-238 contamination was neglected. The description of the capsule, given by Price (Ref. 11-9), was as explicit as possible within the limitations of x-y geometry of the cell model. No explicit representation of detectors, or samples, was attempted; instead, the flux readings were taken at their sites. Since only the fast energy neutrons were of interest, the poor representation of fuel elements, detectors, experiment, etc., should not seriously affect the calculated results.

Analytical Procedure

The method of calculation was as follows:

1. For a given typical fuel element composition, the microscopic cross sections were generated with the MICROX code (Ref. 11-10). The calculated spectrum (Fig. 11-4) was compared with the one supplied by Dahl (Ref. 11-11). The spectra are in fairly good agreement, especially considering the sum total of inconsistencies and assumptions made for our model. The broad group energy boundaries and fission source distributions are given in Table 11-9.
2. Each fuel element in the cell (Fig. 11-3) was assigned a uniform external source equal to the product of its relative power factor, taken at the core midplane, and the fission source computed by the MICROX code.
3. The cell calculation was carried out with this external source and the fission spectrum (χ) set to zero in an x-y geometry calculation with the TWOTRAN code (Ref. 11-12) using P_1 scattering and S_4 angular quadrature.

The above procedure allows the calculation of the fast flux spectrum without calculating the thermal fluxes to any degree of accuracy, a

Fig. 11-4. ETR cell calculations, normalized flux versus energy

TABLE 11-9
RESULTS OF SPECTRUM CALCULATION WITH THE MICROX CODE

Group	Lower Energy (eV)	Flux (n/cm ² -sec)	Current (n/cm ² -sec)	Buckling (cm ⁻²)	Fission Source (n/sec)
1	6.06531×10^6	2.06810×10^{-1}	8.57734×10^{-2}	1.00000×10^{-2}	2.32161×10^{-2}
2	4.49329×10^6	4.60985×10^{-1}	1.58033×10^{-1}	1.00000×10^{-2}	5.08030×10^{-2}
3	3.67879×10^6	4.91748×10^{-1}	1.43605×10^{-1}	1.00000×10^{-2}	5.74510×10^{-2}
4	3.01194×10^6	6.88389×10^{-1}	1.83195×10^{-1}	1.00000×10^{-2}	7.52497×10^{-2}
5	2.46597×10^6	8.98760×10^{-1}	2.45254×10^{-1}	1.00000×10^{-2}	8.82473×10^{-2}
6	2.01897×10^6	9.79372×10^{-1}	2.49700×10^{-1}	1.00000×10^{-2}	9.46745×10^{-2}
7	1.00259×10^6	3.18166	5.93807×10^{-1}	1.00000×10^{-2}	3.00117×10^{-1}
8	4.97871×10^5	2.67758	4.19775×10^{-1}	1.00000×10^{-2}	1.78901×10^{-1}
9	3.01974×10^5	1.28658	1.47688×10^{-1}	1.00000×10^{-2}	6.47906×10^{-2}
10	1.83156×10^5	1.06887	1.27271×10^{-1}	1.00000×10^{-2}	3.40418×10^{-2}
11	5.24752×10^4	1.83572	1.68402×10^{-1}	1.00000×10^{-2}	2.79634×10^{-2}
12	1.17088×10^4	1.55007	1.30904×10^{-1}	1.00000×10^{-2}	4.78570×10^{-3}
13	9.61117×10^2	2.22281	2.00551×10^{-1}	1.00000×10^{-2}	5.16634×10^{-4}
14	2.38237	4.63978	4.35690×10^{-1}	1.00000×10^{-2}	1.24412×10^{-5}
15	0.00000	5.57969	0.00000	0.00000	

relatively cheap computation* which yields far more accurate results (uncertainties in the model description, fuel loadings, element structure, etc., amplify with reduced neutron energy).

The fluxes calculated in the ETR cell at the detector and various sample locations are given in Table 11-10.

Discussion of Results

Effective Fe-54(n,p) Cross Sections

Using the fluxes given in Table 11-10 and the differential (n,p) cross sections for iron as reported by various authors (Table 11-11), the effective cross section ($E > 180$ keV) was calculated for the fuel and detector locations (Table 11-12). The effective cross section was also calculated using the spectrum generated by the MICROX code for many more energy groups. It is interesting to observe that σ_{eff} used by BNWL, i.e., 57.1 mb, is in good agreement with the cross section calculated at a fuel location with either the MICROX or the TWOTRAN code. However, a significantly different cross section is calculated for the detector location. The drastic change of spectra as a function of location in the ETR had been observed previously; e.g., Dahl reports an effective cross section as low as 26.6 mb for the E-5 location (Ref. 11-11). A plausible explanation for the basis of $\sigma_{\text{eff}} = 57.1$ mb may be that the fuel spectrum is easier to calculate, and perhaps has been more accurately measured. Furthermore, the fuel spectrum does not change a great deal during the cycle.

The most recent data were generated by McElroy (Ref. 11-15) and independently by Bresesti (Ref. 11-13); consequently, an effective (n,p) cross section for iron of 46.3 mb is used in this study.

*Especially true since only down-scattering needs to be considered.

TABLE 11-10
GROUP ENERGY FLUXES (N/CM²-SEC PER SOURCE NEUTRON), ETR CYCLE 109A

Group	E _L (MeV)	Fuel	Detector ^(a)		Capsule Samples ^(a)			
			SW	NE	S ₃	S ₁	S ₂	S ₄
1	6.065	9.91 x 10 ⁻³	0.51 x 10 ⁻³	0.52 x 10 ⁻³	0.43 x 10 ⁻³	0.47 x 10 ⁻³	0.42 x 10 ⁻³	0.39 x 10 ⁻³
2	4.493	1.93 x 10 ⁻³	1.10 x 10 ⁻³	1.11 x 10 ⁻³	0.96 x 10 ⁻³	1.04 x 10 ⁻³	0.95 x 10 ⁻³	0.87 x 10 ⁻³
3	3.679	1.99 x 10 ⁻³	1.14 x 10 ⁻³	1.14 x 10 ⁻³	0.97 x 10 ⁻³	1.06 x 10 ⁻³	0.97 x 10 ⁻³	0.88 x 10 ⁻³
4	3.011	2.76 x 10 ⁻³	1.57 x 10 ⁻³	1.57 x 10 ⁻³	1.31 x 10 ⁻³	1.43 x 10 ⁻³	1.30 x 10 ⁻³	1.19 x 10 ⁻³
5	2.466	3.64 x 10 ⁻³	2.16 x 10 ⁻³	2.16 x 10 ⁻³	1.89 x 10 ⁻³	2.04 x 10 ⁻³	1.88 x 10 ⁻³	1.73 x 10 ⁻³
6	2.019	3.97 x 10 ⁻³	2.61 x 10 ⁻³	2.63 x 10 ⁻³	2.39 x 10 ⁻³	2.54 x 10 ⁻³	2.38 x 10 ⁻³	2.23 x 10 ⁻³
7	1.003	12.6 x 10 ⁻³	9.43 x 10 ⁻³	9.54 x 10 ⁻³	9.21 x 10 ⁻³	9.49 x 10 ⁻³	9.19 x 10 ⁻³	8.92 x 10 ⁻³
8	0.049	10.7 x 10 ⁻³	8.54 x 10 ⁻³	8.68 x 10 ⁻³	8.39 x 10 ⁻³	8.53 x 10 ⁻³	8.38 x 10 ⁻³	8.26 x 10 ⁻³
9	0.302	5.15 x 10 ⁻³	4.28 x 10 ⁻³	4.37 x 10 ⁻³	4.39 x 10 ⁻³	4.43 x 10 ⁻³	4.39 x 10 ⁻³	4.36 x 10 ⁻³
10	0.183	4.32 x 10 ⁻³	3.82 x 10 ⁻³	3.92 x 10 ⁻³	4.01 x 10 ⁻³	4.02 x 10 ⁻³	4.03 x 10 ⁻³	4.03 x 10 ⁻³
11	0.052	7.48 x 10 ⁻³	6.94 x 10 ⁻³	7.15 x 10 ⁻³	7.56 x 10 ⁻³	7.52 x 10 ⁻³	7.59 x 10 ⁻³	7.65 x 10 ⁻³
12	0.012	6.39 x 10 ⁻³	5.83 x 10 ⁻³	6.00 x 10 ⁻³	6.24 x 10 ⁻³	6.24 x 10 ⁻³	6.27 x 10 ⁻³	6.28 x 10 ⁻³

(a) For location of detectors and samples, see Fig. 11-3.

TABLE 11-11
Fe-54(n,p) CROSS SECTIONS

Group	E_L (MeV)	Differential Cross Section (barns)			
		Bresesti (Ref. 11-13)	Helm (Ref. 11-14)	Dahl (Ref. 11-11)	McElroy (Ref. 11-15)
1	6.065	0.571	0.287	0.479	0.569
2	4.493	0.448	0.242	0.387	0.442
3	3.679	0.309	0.219	0.266	0.300
4	3.011	0.172	0.197	0.185	0.146
5	2.466	0.0771	0.160	0.151	0.0622
6	2.019	0.0113	0.102	0.0640	0.0302
7	1.003	0.0	0.0120	0.0003	0.0073
8	0.498	0.0	0.00	0.00	0.00
9	0.302				
10	0.183				
11	0.053				

TABLE 11-12
EFFECTIVE CROSS SECTIONS FOR FUEL AND DETECTOR LOCATIONS

$$\int_{180 \text{ keV}}^{\infty} \sigma \phi dE / \int_{180 \text{ keV}}^{\infty} \phi dE$$

Calculated By	Effective Cross Section (mb)			
	(a)	(b)	(c)	(d)
MICROX	56.6	58.4	61.6	56.8
TWOTRAN				
At SW detector	45.5	48.2	49.9	46.3 ^(e)
At fuel	58.2	59.2	62.9	58.4

- (a) Based on Bresesti data (see Table 11-11).
(b) Based on Helm data (see Table 11-11).
(c) Based on Dahl data (see Table 11-11).
(d) Based on McElroy data (see Table 11-11).
(e) Best estimate.

Fast Fluence in the ETR

The fast fluence at the detector location can be expressed as follows:

$$\phi_D \cdot \tau = \phi'_D \cdot \tau \cdot \frac{\bar{\sigma}}{\sigma_{\text{eff}}} \quad , \quad (11-1)$$

where ϕ_D = detector flux $\equiv \int_{180 \text{ keV}}^{\infty} \phi dE$

τ = equivalent time at full power

ϕ'_D = reported flux ($E \geq 180 \text{ keV}$)

$\bar{\sigma} = 57.1 \text{ mb (n,p)}$ cross section used to determine the reported fluence

σ_{eff} = effective (n,p) cross section $\equiv 46.3 \text{ mb}$

The fast flux at the sample location can be expressed as follows:

$$\phi_s = \phi_D \times F_1 \quad (E \geq 180 \text{ keV}), \quad (11-2)$$

where

$$F_1 = \frac{\int_{180}^{\infty} \phi dE \text{ (sample)}}{\int_{180}^{\infty} \phi dE \text{ (detector)}}$$

Based on the data in Table 11-10, the above ratio was calculated at various sample positions (Table 11-13). The actual fast fluence, based on the reported maximum allowable fluence of $8 \times 10^{21} \text{ n/cm}^2$, is given in Table 11-14.

TABLE 11-13
RESULTS OF ETR CELL CALCULATION

Description	Fuel	Detector		Sample			
		SW	NE	S ₃	S ₁	S ₂	S ₄
$\int_{180 \text{ keV}}^{\infty} \phi dE$	47.97	35.16	35.64	33.94	35.05	33.89	32.86
$\int_{50 \text{ keV}}^{\infty} \phi dE$	55.45	42.10	42.79	41.50	42.57	41.48	40.51
$\int_{50}^{\infty} \phi dE / \int_{180}^{\infty} \phi dE$	1.155	1.197	1.200	1.222	1.214	1.223	1.232
$\left(\int_{180}^{\infty} \phi dE \right)_S / \left(\int_{180}^{\infty} \phi dE \right)_D$	—	—	—	0.965	0.996	0.963	0.934

TABLE 11-14
FAST FLUENCE AT VARIOUS SAMPLE LOCATIONS

Sample	Fast Fluence (n/cm ²) (E > 180 keV)
S ₃	9.5 x 10 ²¹
S ₁	9.8 x 10 ²¹
S ₂	9.4 x 10 ²¹
S ₄	9.2 x 10 ²¹

Consequently, the fast fluence for graphite samples in the ETR was significantly larger than the reported data. The major cause for this discrepancy was the reliance of the ETR analysts on old data for the effective (n,p) cross section of the fast flux detectors.

Fast Fluence in the HTGR

It was recognized that neutrons in the 50 to 180 keV range contribute to the graphite damage (Ref. 11-6) and that the HTGR spectrum contains many more neutrons in this range than the ETR (Ref. 11-7).

The allowable fast fluence for neutron energies above 50 keV can be equated to the more familiar fluence with E > 180 keV in the following manner:

$$\int_{50}^{\infty} \phi dE / \int_{180}^{\infty} \phi dE \approx 1.4 \text{ (HTGR) (Ref. 11-7)} \quad (11-3)$$

$$\int_{50}^{\infty} \phi dE / \int_{180}^{\infty} \phi dE = F_2 \text{ (ETR sample)} \quad (11-4)$$

It was postulated (Ref. 11-6) that the graphite damage correlates with the neutron flux above 50 keV; i.e.,

$$\left(\int_{50}^{\infty} \phi dE \right)_{\text{HTGR}} \equiv \left(\int_{50}^{\infty} \phi dE \right)_{\text{ETR sample}} .$$

With the above consideration, dividing Eq. 11-4 by Eq. 11-3 gives

$$\left(\int_{180}^{\infty} \phi dE \right)_{\text{HTGR}} = \left(\int_{180}^{\infty} \phi dE \right)_{\text{ETR sample}} \times \left(\frac{F_2}{1.40} \right) . \quad (11-5)$$

Substituting Eq. 11-2 into Eq. 11-5,

$$\left(\int_{180}^{\infty} \phi dE \right)_{\text{HTGR}} = \left(\int_{180}^{\infty} \phi dE \right)_{\text{detector}} \times \left(\frac{F_1 \cdot F_2}{1.40} \right) \quad (11-6)$$

Substituting Eq. 11-1 into Eq. 11-6,

$$\left(\int_{180}^{\infty} \phi dE \right)_{\text{HTGR}} = \phi'_D (\bar{\sigma}/\sigma_{\text{eff}}) (F_1 \cdot F_2 / 1.40) \quad (11-7)$$

The above equation defines the fast fluence dose in an HTGR ($E > 180$ keV) that will produce the same amount of graphite damage as the reported fast fluence for the ETR experiment.

Based on the data in Table 11-10, the necessary factors, F_1 and F_2 , were calculated (Table 11-15). Also in Table 11-15, the "equivalent graphite damage" fast fluence, i.e.,

$$\left(\int_{180 \text{ keV}}^{\infty} \phi dE \right)_{\text{HTGR}} ,$$

of Eq. 11-7 was calculated for the various sample locations.

TABLE 11-15
CALCULATION OF EQUIVALENT DAMAGE FAST FLUENCE DOSE

Description	Sample			
	S ₃	S ₁	S ₂	S ₄
ϕ'_D , nvt	8×10^{21}	8×10^{21}	8×10^{21}	8×10^{21}
$\bar{\sigma}/\sigma_{\text{eff}}$	1.23	1.23	1.23	1.23
F ₁	0.97	1.00	0.96	0.93
F ₂	1.22	1.21	1.22	1.23
$\left(\int_{180 \text{ keV}}^{\infty} \phi dE \right)_{\text{HTGR}}$	8.3×10^{21}	8.5×10^{21}	8.2×10^{21}	8×10^{21}

Conclusions

From the results given in Table 11-15, it can be concluded that the allowable fast fluence of 8×10^{21} n/cm² (E > 180 keV) for graphite is not excessive. However, it should be emphasized that this conclusion is based on limited analytical data. Since the effective detector cross section is very sensitive to its environment, it is difficult to predict how other samples will fare in other cycles and under other conditions.

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APPENDIX

PROJECT REPORTS PUBLISHED DURING THE QUARTER

Price, R. J., "Thermal Conductivity of Neutron-Irradiated Reactor Graphites,"
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