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AEC RESEARCH AND
DEVELOPMENT REPORT

GA-9875
UC-2 General,
Miscellaneous, and
Progress Reports

PUBLIC SERVICE COMPANY OF COLORADO
330-MW(E) HIGH-TEMPERATURE GAS-COOLED REACTOR
RESEARCH AND DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
DECEMBER 31, 1969

Prepared under
Contract AT(04-3)-633
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

January 30, 1970

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INTRODUCTION

The objective of the research and development program work reported here is to develop and verify the information required to design, construct, operate, and maintain the Public Service Company of Colorado power plant as provided in U.S. Atomic Energy Commission Contract AT(04-3)-633.

Part I of this report includes the work described in Appendix B of the contract; this work consists largely of component development and testing, nuclear analysis, and fuel development and testing. Part II covers the work described in Appendix K of the contract on the fuel transfer machine, the series-steam-turbine-driven circulator, the control rod drives, the steam generator, and coated particles.

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PART I

AEC-SUPPORTED RESEARCH AND DEVELOPMENT PROGRAM
FOR THE PUBLIC SERVICE COMPANY OF COLORADO HTGR

Task I

COMPONENT DEVELOPMENT

No significant events took place in this task during the report period. Much of the work has been completed, and the remaining efforts will be reported as activities take place.

Task II
NUCLEAR ANALYSIS

FUEL CYCLE ANALYSIS

The objective of this subtask is to establish the fuel management program for the reactor based on analyses of the effects of the loadings in the initial and replacement segments, including recycle segments in an equilibrium core, on local power generation, reactor control, reactor safety, and economics.

Status of Nuclear Design

Some modifications have been made to the previously established Reference Design 9. Of particular concern were the power distribution and control rod worth. The adoption of the thin-buffer-zone design results in a fairly high worth for single rod pairs such that core shutdown margins with one rod pair removed were less than the applicable reactivity uncertainty. In addition, although refueling region peaking factors were acceptable, it was clear that some improvements could easily be achieved. As a result, the Reference Design 9 fuel zoning was slightly modified, leading to improved power distributions and increased shutdown margins.

This modified design, Reference Design 10, is intended to form the basis for the final calculations leading to fuel loading specifications. The analysis of the final core design is currently under way. During this design iteration special emphases will be put on the initial core design; it is not planned to do any more analysis of refueling segments before the initial core loadings are completely specified.

Before proceeding with the final analysis for the initial core, Reference Design 10, all basic input data have been reviewed. Initial core loadings remain unchanged at 880 kg of uranium and 19,458 kg of thorium.

The change of the reference design from bonded fuel rods to blended beds resulted in a 20% carbon loss in the fuel rods, or about a 5% carbon loss in the core. However, the reactivity loss due to the reduced core carbon content is just about compensated for by the reduction of the effective thorium resonance integral resulting from the lower carbon density in the fuel beds. This very desirable feature indicates that the initial core excess reactivity is not very sensitive to the carbon density in the fuel beds.

All core and reflector cross sections have been reviewed. Besides the thorium cross sections, all data from the previous design iteration appear to be up to date.

FUEL ZONING AND POWER SHAPING

The objective of this subtask is to establish the fuel and burnable poison distribution and the control rod programming based on analyses of the transverse and axial power distribution as a function of fuel and poison loading, burnup, and control rod positions and considerations of fuel temperatures and economics.

Modification of Reference Design 9

Reference Design 9 was adopted after it became apparent that the thick-buffer-zone design (Reference Design 7) was unacceptable because of large power tilts across single refueling regions. As reported in the previous quarterly report (GA-9720), the initial core loading for Reference Design 9 consists of 13 fuel compositions to partially mock up the equilibrium cycle core loading and contains a thin buffer zone with five rows of fuel holes at the core-reflector interface. Although the calculated unrodded power distributions appeared to be quite acceptable, further analysis showed that the design limit of a 1.83 peaking factor for any refueling region was reached during the initial cycle with some control rod patterns investigated. In addition, the calculated shutdown margin that was available with one rod pair removed was almost within the uncertainty band for the core excess reactivity and control rod worth at the beginning of the cycle. At the middle of the initial cycle when the core excess reactivity reaches its maximum, the core shutdown margin was less than the applicable reactivity uncertainty. Therefore, in order to obtain a more uniform radial power distribution for the initial core and in order to reduce the worth of the maximum worth control rod, it was decided to radially rezone the initial core loading.

A favorable radial rezoning could be achieved by removing 6% of the uranium load from compositions 7 and 8 and transferring it to composition 1 and 2 (see the previous quarterly report, GA-9720). Early analysis indicated that shifting fuel from a single refueling region to reduce the rod worth on the order of 0.01 Δk yields extremely high flux tilts in the region where the fuel has been relocated. This information, coupled with the fact that the maximum worth control rod for the initial core was nearly the same for any of the refueling regions containing compositions 7 and 8, suggested that the uranium loading in these compositions be reduced for all of the corresponding regions. To preserve nearly the same reactivity performance for the total core and keep the same number of fuel element compositions, the uranium was relocated in those compositions 1 and 2 which have the lowest uranium loading. The majority of this is located within the central refueling regions of the core, resulting in an over-all shift of the power to the center. However, this can be minimized by selective control rod programming during operation.

The primary disadvantage to this method of rezoning is that it results in a small perturbation to the uranium inventory during the approach to equilibrium for the reference design because the uranium removed from composition 7 and 8 is scheduled to remain in the core through reloads four to six, whereas the regions in which it is relocated are scheduled for removal with reloads 1 and 2. However, a depletion analysis through the approach to equilibrium showed this perturbation to be insignificant. The calculated maximum rod worth

for the modified design is $0.01 \Delta k$ lower than in Reference Design 9, correspondingly, shutdown margins with one rod pair removed are increased. In addition, even with the shift in power from outer refueling regions toward the center, the maximum control rod worth for "inner" control rod pairs is still considerably lower than that for the outer regions.

The shift in the radial power distribution caused by transferring uranium from compositions 7 and 8 into compositions 1 and 2 of the initial core is shown in Fig. 2.1 for the modified as well as for the reference design. For the reference design, the maximum region power of 1.34 is located in refueling regions 30 and 31, whereas the value for these regions decreases to 1.19 for the modified design. Conversely, the power in refueling region 2 increases from 1.21 in the reference design to 1.30 in the modified design.

In the new design the maximum power never exceeds 1.80 with 0, 4, 7, or 10 control rod pairs inserted and is below 1.65 for these configurations. An example of region peaking factors with 10 inserted rod pairs is also shown in Fig. 2.1.

The beginning-of-cycle excess reactivity for the modified initial core was identical to that for the reference design, namely $0.053 \Delta k$ for the hot, clean core. This was not surprising, in that the total uranium and thorium load was kept constant and the total amount of uranium that was transferred between the compositions was less than 15 kg. Not only is the beginning-of-cycle excess reactivity identical with the reference, but the excess reactivity during the first year, from an unrodded burnup analysis, is the same.

Integrated fast-flux doses and region tilting factors of the modified design compare well with previously reported data for Reference Design 9 and the modified design was adopted as the reference design. The following design iteration will form the basis for the initial core fuel loading specification.

Lumped Burnable Poison Design

About two thirds of all fuel elements in the initial core contain two poison rods located in opposite corners of the element. To minimize interference between poison rods, they should be so located that each corner where three elements adjoin contains only one poison rod.

Consistent with this procedure, the lumped poison rods will be located in the fuel elements in the following manner:

1. No lumped poison rods will be located within fuel elements loaded with compositions 1 and 2 (see GA-9720), which partially mock up for four- and five-year-old fuel. These regions have a relatively high thorium and relatively low uranium content and are refueled after one and two years.
2. No lumped poison will be located within the central (control rod) elements for any refueling region.
3. Each of the regular fuel elements for the remaining compositions will contain two poison rods per element. They will be loaded into the available location at the corner of the element as shown in Fig. 2.2.

Fig. 2.1--Power distributions for Reference Design 9 and modified design

Fig. 2.2--Typical refueling region in initial core with poison rods in selected locations

Located in this manner, any of the fuel elements are interchangeable within a refueling region, except for the regions at the outer edge of the core. In those regions, some of the outer elements contain the buffer zones with a heavier thorium loading. In order to prevent interference between poison rods, it is important that the poison rod nearest the control rod element be located in the left corner of the side that faces the central element.

The resulting distribution of burnable poison rods in the core is shown in Fig. 2.3.

Detailed studies have been performed in the past to evaluate various methods of analytically treating the lumped burnable poison in two-dimensional diffusion calculations (see earlier quarterly progress report GA-8879). These studies, however, were primarily concerned with the reactivity worth of the poison in static calculations disregarding depletion effects.

The usual method used to calculate the poison reactivity worth is to smear the poison density uniformly over the entire volume of a fuel element and to apply appropriate disadvantage factors. Since the poison rods are located at the corners of the fuel elements, some of them are in the proximity of material discontinuities where the neutron flux may vary drastically in amplitude and energy spectrum. Smearing the poison over a whole fuel element may thus result in an incorrect calculated reactivity worth for the poison and an incorrect depletion behavior.

In order to show the effect of internal material discontinuities on the initial worth and the depletion characteristics of the lumped burnable poison, a cell was constructed that contained the same geometrical configuration as existed in the core for a refueling region at the core-reflector interface. The schematic of the calculational model is shown in Fig. 2.4. This particular configuration was chosen because (1) a refueling region located at the core edge sees the greatest flux gradient and spectrum shift, (2) two dissimilarly loaded regions could be shown with a background of average fuel, and (3) control rods could be inserted in a fuel element that was either in the center of a refueling region, or adjacent to the reflector. Furthermore, the model permitted control rod insertion to be represented by either of the two methods: control poison smeared over the central hexagonal element, or blackness boundary conditions at the perimeter of a control rod channel.

The mesh grid lines were so chosen that a minimum volume of $1/48$ of a fuel element was available to locally smear the poison rods at their proper location. Thus, localized effects of material discontinuities could be studied in more detail.

Comparative calculations were carried out for a model in which the lumped burnable poison was smeared over the whole fuel element volume with the control rod poison smeared over the central hexagonal element (the method most commonly used in the modeling of the Fort St. Vrain reactor) and, for a model in which the lumped burnable poison was smeared over the appropriate rhombi, with the control rod insertion treated explicitly by assignment of appropriate boundary conditions.

Fig. 2.3--Lumped poison distribution in the initial core

Fig. 2.4--Cell model for poison depletion study

The results of static calculations are summarized in Table 2.1, which compares the k_{eff} and the region peaking factors for the two methods of poison representation. Both reactivity and power peaking factors agree very well.

The depletion of the cell, which indicates that the reactivity change as a function of time is the same for both methods of poison representation, is shown in Table 2.2. The power peaking factors for the refueling regions as a function of burnup time are also given in this table. As for the static calculations, the agreement in the average refueling region peaking factors is very good, the local power factors showing a discrepancy of about 5%.

On the basis of this study, it may be concluded that the present method of control rod and lumped burnable poison worth calculations (i.e., by smearing the poison over the whole fuel element volume) is adequate for static as well as for depletion calculations and that the effect of material discontinuities on the poison worth is negligible.

KINETIC ANALYSIS

The objective of this subtask is to predict the required strength of the control system, the control rod worth, the magnitude of the temperature coefficients, and the kinetic behavior of the core including insertions of reactivity and spatial xenon stability by means of analyses under normal operating and accident conditions.

Temperature Coefficient Analysis

The negative temperature coefficient is one of the inherent safety features of the Fort St. Vrain reactor. The coefficients are negative over the entire range of realizable temperatures and effective in limiting reactivity excursions.

In connection with safety analyses, temperature coefficients were calculated for the initial cycle, the reference six-year equilibrium cycle and a four-year equilibrium cycle. A summary of the results is shown in Figs. 2.5, 2.6, and 2.7. Figure 2.5 shows the isothermal and fuel (Doppler) coefficients during the initial cycle. At the beginning of the cycle before Xe^{135} and Sm^{149} have equilibrated, the isothermal coefficients are most negative. As the depletion of the core progresses, the coefficients become less negative. The same trend can be seen on Figs. 2.6 and 2.7 for the two equilibrium cycles. Fuel coefficients stay essentially unchanged during each cycle.

The main contributors to the negative temperature coefficient are Th^{232} , U^{235} , and lumped burnable poison. The significant positive contributors are U^{233} , Xe^{135} , and Sm^{149} . This is illustrated in Tables 2.3 and 2.4, comparing isothermal coefficients for the six- and four-year equilibrium fuel cycles.

The negative fuel coefficient is provided almost entirely by the Doppler broadening of the thorium resonances and is proportional to the total thorium loadings. This explains the decreasing fuel coefficient from the heavily loaded initial cycle to the more lightly loaded six-year equilibrium cycle and the even more lightly loaded four-year equilibrium cycle. The fact that

Table 2.1
RESULTS OF POISON WORTH STUDY^a
(Static Calculations)

Case	Poison Smeared Over Fuel Element			Locally Smeared Poison		
	k_{eff}	Region	Peaking Factor	k_{eff}	Region	Peaking Factor
No control rods	1.104	1	1.27	1.105	1	1.29
		2	1.16		2	1.15
		3	0.46		3	0.45
Control rods in region 2	1.033	1	1.55	1.030	1	1.56
		2	0.89		2	0.88
		3	0.52		3	0.51
Control rods in region 3	1.091	1	1.38	1.092	1	1.40
		2	1.22		2	1.21
		3	0.25		3	0.24

^aTotal reactivity worth of the poison was 0.028 Δk .

Table 2.2
RESULTS OF POISON WORTH STUDY
(Depletion Calculations)

Time (Days)	Poison Smeared over Fuel Element			Locally Smeared Poison		
	k_{eff}	Region	Peaking Factor	k_{eff}	Region	Peaking Factor
0	1.104	1	1.27	1.105	1	1.29
		2	1.16		2	1.16
		3	0.46		3	0.45
103.5	1.046	1	1.25	1.046	1	1.26
		2	1.16		2	1.16
		3	0.48		3	0.48
207	1.041	1	1.24	1.041	1	1.25
		2	1.16		2	1.16
		3	0.50		3	0.49
310.5	1.034	1	1.23	1.034	1	1.24
		2	1.15		2	1.15
		3	0.52		3	0.51

Fig. 2.5--Initial core temperature coefficients

Fig. 2.6--Temperature coefficients for the reference 6-year equilibrium cycle

Fig. 2.7--Temperature coefficient for the backup 4-year equilibrium cycle

Table 2.3
MAIN CONTRIBUTORS TO THE ISOTHERMAL TEMPERATURE
COEFFICIENT AT 1200°K
(6-year Equilibrium Cycle, C/Th = 225)

	$\frac{1}{k} \frac{\Delta k}{\Delta t} (10^{-5}/^{\circ}\text{K})$	
	BOC ^a	EOC ^a
Th ²³²	-2.10	-2.09
Pa ²³³	-0.02	-0.06
U ²³³	-0.00	+0.32
U ²³⁵	-2.15	-1.72
Np ²³⁷	-0.22	-0.36
Xe ¹³⁵	0.0	+0.87
Rh ¹⁰³	-0.09	-0.15
Sm ¹⁴⁹	+0.20	+0.26
Sm ¹⁵¹	+0.10	+0.15
LBP ^b	-0.22	---
Leakage	-0.19	-0.19
Total	-4.35	-2.59

^aBOC = beginning of cycle;
EOC = end of cycle.

^bLumped burnable poison.

Table 2.4
MAIN CONTRIBUTORS TO THE ISOTHERMAL TEMPERATURE
COEFFICIENT AT 1200°K
(4-year Equilibrium Cycle, C/Th = 275)

	$\frac{1}{k} \frac{\Delta k}{\Delta t} (10^{-5}/^{\circ}\text{K})$	
	BOC ^a	EOC ^a
Th ²³²	-1.94	-1.69
Pa ²³³	-0.05	-0.16
U ²³³	+0.42	+0.81
U ²³⁵	-2.06	-1.88
Np ²³⁷	-0.14	-0.31
Xe ¹³⁵	+0.98	+1.08
Rh ¹⁰³	-0.12	-0.23
Sm ¹⁴⁹	+0.18	+0.28
Sm ¹⁵¹	+0.09	+0.16
LBP ^b	-0.40	---
Leakage	-0.34	-0.33
Total	-3.19	-1.89

^aBOC = beginning of cycle;
EOC = end of cycle.

^bLumped burnable poison

only a small amount of Th^{232} burns out during each cycle explains why the Doppler coefficient stays essentially constant during each cycle.

The isothermal coefficients are most negative at the beginning of each cycle, when fission product inventory is lowest and U^{235} and lumped poison loadings are largest. As U^{233} and fission products build up and U^{235} and the lumped poison deplete, the coefficients become progressively less negative. However, they stay negative over the entire temperature range of interest. Least-negative coefficients are calculated at the end of the cycle for the four-year equilibrium fuel cycle. This set of coefficients was used in the final safety analysis report.

Sensitivity Studies

The maximum fuel, moderator, and gas outlet temperatures encountered during a rod withdrawal accident depend on many parameters. Since the uncontrolled withdrawal of a control rod pair is the worst reactivity accident in the Fort St. Vrain HTGR, it is desirable to evaluate the sensitivity of maximum temperatures to the various system parameters. This information is required because all parameters are known only to a certain accuracy and because it is necessary to assess the effect of possible deviations from the nominal values used in the hazards analysis.

Uncertainties affecting the results of rod withdrawal accident calculations stem from various sources. First, the accuracy with which physics parameters like control rod worths and temperature coefficients can be calculated has to be considered. Second, the tolerance limits for mechanical equipment such as the control rod drive units has to be taken into account, since they affect the reactivity insertion rates. In addition, there are tolerance limits in the sensitivity of the protective nuclear instrumentation that act to terminate uncontrolled transients.

The sensitivity of system temperatures to the worth of a withdrawn control rod pair was discussed in the previous quarterly report (GA-9720), assuming various protective actions to terminate the transient. Control rod worths considerably higher than actually calculated were used to evaluate limits on acceptable system performance.

A similar approach was taken in the studies reported below. The variation of system parameters was taken to be considerably larger than expected in the actual plant.

For all cases, except where noted differently, equilibrium end-of-cycle temperature coefficients were used (see Fig. 2.7). For the base case, a 0.012 Δk rod withdrawal accident and a 0.08 Δk scram was assumed. Nominal rod withdrawal and insertion times were 180 sec and 150 sec, respectively. For all cases, scram action was assumed to occur 105 sec after accident initiation. Normalized reactivity insertion rates are shown in Fig. 2.8.

Sensitivity to Reactivity Insertion Rate. The results of point kinetics calculations studying the sensitivity to reactivity insertion rates are summarized in Table 2.5. Case 1 is the base case using nominal data for the hazards analysis. Cases 2 and 3 show the effect of faster and slower control

Fig. 2.8--Reactivity insertion rate during rod pair withdrawal and scram actions

Table 2.5

SENSITIVITY TO REACTIVITY INSERTION RATES

Case	Ramp Insertion (sec)	Scram Insertion (sec)	Average Fuel Temp Rise (°C)	Average Gas Outlet Temp Rise (°F)	Reactor Thermal Power (Mw)
1	180	150	383	502	2700
2	160	150	459	597	3110
3	200	150	318	425	2440
4	180	100	354	423	2700
5	180	175	405	540	2700

NOTE: Accident: $\Delta K = 0.012$.

Scram: $\Delta K = 0.08$ at 105 sec after accident initiation.

rod withdrawal. The average rise in fuel and gas outlet temperature change by $\pm 70^{\circ}\text{C}$ and $\pm 90^{\circ}\text{F}$ if the insertion time is changed by ± 20 sec. Clearly, the exact knowledge of the rod withdrawal time is not too important if it is within these limits. Similarly, the calculated results are not sensitive to the scram insertion time as shown in Cases 4 and 5.

Sensitivity to Temperature Coefficients. The maximum reactivity worth calculated at the end of the cycle, when temperature coefficients are least negative, is $0.012 \Delta k$. At the beginning of the cycle, when the temperature coefficients are more negative (Fig. 2.7), a rod worth of 0.016 is calculated. As shown in Table 2.6, the higher rod worth at the beginning of the cycle is more than compensated for by the more negative coefficients. As a result, the end-of-cycle condition was used as the worst condition for accident analysis.

Table 2.6

SENSITIVITY STUDIES FOR ROD WORTH AND TEMPERATURE COEFFICIENTS

Case	Coeff.	Reactivity Insertion (Δk)	Average Fuel Temp Rise ($^{\circ}\text{C}$)	Average Gas Outlet Temp Rise ($^{\circ}\text{F}$)	Maximum Reactor Power (Mw)
1	EOC	0.012	383	502	2700
3	BOC	0.016	345	462	2460

NOTE: Reactivity insertion in 180 sec. Scram insertion: $0.08 \Delta k$ in 150 sec, 105 sec after accident initiation.

Control Rod Worths

The reactivity worth of the control rods in the Fort St. Vrain reactor is determined primarily with two-dimensional diffusion calculations in XY geometry, in which the control rods are either fully withdrawn or fully inserted. This calculational model does not consider the effect of the top and bottom reflectors nor the axial fuel loading variations that are present in the reactor; it effectively assumes an infinite core height where the control rods are also of infinite height. In reality a completely inserted rod pair extends only over the active core and in the withdrawn position the tips of the rods are about 4 in. above the active core.

Detailed calculations have shown that by neglecting the finite height of the core, the total calculated bank worth is overestimated by about 7%. Several calculations were performed to determine what factors produced this drop in rod worth. A unit cell was constructed comprising one average refueling region in cylindrical RZ geometry and containing a single control rod pair. The poison body itself was treated as a nondiffusion region. Appropriate factors for the boundary conditions were calculated with one-dimensional transport calculations.

From the analysis it is concluded that three effects primarily contribute to the total reactivity worth discrepancy. The fact that the control rods are withdrawn to only 4 in. above the core makes the unrodded situation appear less reactive and, indeed, reduces the total core excess reactivity. Conversely, the over-all core reactivity change from the completely withdrawn to the completely inserted position must be less in actuality than calculated solely with two-dimensional diffusion calculations. This effect accounts for about 40% of the total discrepancy.

A similarly large effect results from the fact that the inserted control rods extend only over the active core. Calculations indicate that again about 40% of the reactivity discrepancy would be recovered by allowing the inserted control rods to cover both active core and reflectors.

The remainder of the reactivity loss appears to result from the axial fuel loading variations and the different effectiveness of the control rods in the top and bottom half of the core.

On the basis that the calculated control rod worth is too high because of the combined effects mentioned above, the available reactivity control with rods stuck out was calculated for the modified Reference Design 9 (Reference Design 10). The calculations were made by reducing the homogenized boron density of the control rods in a GAUGE (Ref. 1) calculation in which (1) all the rods were in, (2) all the rods except the maximum-worth rod pair were in, and (3) all the rods except two of the maximum-worth pairs were in. This combined with the results from the unrodded case provided the control rod bank worth for the different configurations.

The results of this work are summarized in Table 2.7 and Fig. 2.9. The data in Table 2.7 show the control rod bank and the maximum rod worth for different total control rod worths (control rod densities). The second column gives results for the unmodified case in which the total bank worth is probably too high by about 5% to 10%. The third column gives results for that case in which the total rod worth has been reduced to match the most probable value. The fourth and fifth columns show the results where extreme changes are introduced in the total control rod worth and are presented mainly for comparison. Figure 2.9 shows essentially the same data.

From these results it can be concluded that an uncertainty of $\pm 0.02 \Delta k$ in the calculated bank worth leads to an uncertainty of only $\pm 0.01 \Delta k$ in the shutdown margin with one rod pair withdrawn.

Data Review

Before proceeding with the analysis for the final design of the initial core, a review of all the cross-section data has been made. This review included the effective cross sections for the core and reflector materials as well as the fission-product yield sets for depletion analysis.

In Reference Design 10 for the initial core, the graphite fuel elements will be loaded with loose fissile, fertile, and coke particles instead of the bonded fuel rods as in some earlier designs. This results in about 20%

Table 2.7

CALCULATED CONTROL ROD BANK WORTH FOR INITIAL FORT ST. VRAIN CORE

Configuration	Ref. Results, Control Rods Unmodified (Δk)	Control Rod Density ^a Reduced (0.9) for Expected Worth (Δk)	Control Rod Density Reduced (0.75) (Δk)	Control Rod Density Reduced to Match Worth of Reserve Shutdown
Total control rod bank, 37 rod pairs	0.230	0.216	0.192	0.130
Maximum worth rod pair	0.064	0.056	0.043	0.017
Rod bank with maximum worth rod pair out	0.166	0.160	0.149	0.103
Two maximum worth pairs	0.110	0.100	0.082	0.041
Rod bank with two maximum rod pairs out	0.120	0.116	0.110	0.089

^aBest estimate for reference design.

Fig. 2.9--Worth of a single rod pair as a function of control system bank worth

less carbon within the fuel rod and consequently about 5% less carbon within the fuel element. For the bonded fuel rod, all of the space between the particles is filled with a carbon matrix having a density of about 1.2 g/cm^3 , whereas with the loose particles about 25% of the volume is occupied by void. In addition, the packing fraction of the fuel particles has been reduced slightly to allow space for the coke particles.

This loss of carbon from the core reduces the initial excess reactivity by about $0.012 \Delta k$. However, reducing the carbon within the fuel rod also reduces the effective resonance cross section for thorium by about 4%, and this change almost compensates for the reactivity loss due to carbon removal. As a result, using both the modified thorium cross sections and the carbon atom densities for the loose fuel, the initial core excess reactivity is reduced only about $0.003 \Delta k$ at the beginning of the cycle and about $0.004 \Delta k$ at the end of the cycle for the first year. This indicates a desirable design feature, namely that the core excess reactivity for the first core is rather insensitive to changes of the carbon density in the fuel rod.

Reactivity Worth of Helium Coolant

One of the advantages of gas cooling is the low reactivity worth associated with the coolant. In the Fort St. Vrain reactor, the helium coolant mainly contributes to the moderation of neutrons. By increasing the resonance escape probability and contributing to a better thermalization of the undermoderated core, the addition of helium increases the core reactivity. However, since the helium occupies only about 20% of the core volume and because its effective density is low, its total contribution to neutron moderation is very small compared to the graphite moderator and the coolant reactivity worth is almost negligible.

The total reactivity worth of the complete helium inventory is calculated to be $1.45 \times 10^{-4} \Delta \rho$, which is very small compared to other reactivity sources.

ANALYTICAL STUDIES

The objective of this subtask is to provide support to other tasks of the program by means of analyses of such items as neutron and flux distributions, nuclear safety of auxiliary equipment, loadings and nuclear properties of irradiation experiments, defining of the need for physics measurements during reactor startup, and evaluating the core response to reserve shutdown system operation.

Criticality of Fresh Fuel Shipping Container

Results of the criticality analysis of fresh fuel shipping containers were discussed in the previous quarterly report (GA-9720). Conservative estimates show that up to 120 containers containing U^{233} as fissile material can be stored and arranged in any array without exceeding a multiplication constant of 0.85.

The assumption of having U^{233} fissile material, rather than U^{235} , is conservative considering the fact that the initial fuel loading for the Fort St. Vrain reactor contains no U^{233} . Table 2.8 gives a comparison of multiplication constants for infinite layers of shipping containers stacked on top of

For an infinite bare and unflooded layer with no reflector on either side, a multiplication constant of 0.76 is calculated. With a 12-in. water reflector on one side, the multiplication constant increases to 0.90. This latter case simulates the storage of fuel on hydrogeneous material such as concrete during fuel loading and fuel handling.

It is concluded that any handling of bare fuel elements will not lead to a criticality hazard if other than within the PCRV, fuel handling machine, shipping cask, or fuel storage wells, no bare element is placed on top of another.

References

1. Wagner, M. R., GAUGE, A Two-Dimensional Few Group Neutron Diffusion-Depletion Program for a Uniform Triangular Mesh, USAEC report GA-8307, Gulf General Atomic Incorporated, March 15, 1968.

Task III
FUEL DEVELOPMENT

FUEL ELEMENT DESIGN

The objective of this subtask is to develop the design of the fuel elements to be used in the plant by means of mechanical and flow testing, analysis of temperatures under normal and abnormal conditions, analyses of stresses, including effects of temperature, irradiation, and creep, and analysis of fission-product concentrations in the reactor system.

Mechanical Design

Fuel Element Drawings. Work is continuing to complete fuel element assembly drawings, core layer loading maps, and a core and reflector arrangement drawing. The initial reactor core is loaded with 83 different types of fuel elements. This number is the result of a combination of seven different graphite blocks, thirteen types of fuel loadings, and two types of burnable poison rod loadings (some contain poison rods and some do not). In addition, there are nine different configurations of the two types of fuel loadings in outer buffer ring of fuel columns. The arrangement drawing and loading drawings will show the location and orientation of all the types of fuel elements and reflector elements in the reactor core.

Particle Traps. The use of blended fuel beds makes it desirable to consider a system to trap fuel particles that, by some mechanism, might enter the primary coolant flow stream from a fuel element before they enter the steam generators and helium circulators. Particles accelerated by the coolant flow would be trapped in the reflector without appreciable bypass or breakage. The system would provide some type of reservoir to catch particles falling into open coolant holes in the event a fuel element were sufficiently damaged during refueling to allow fuel particle spillage.

A trap has been designed to meet these requirements. The trap replaces an exit reflector element, is 95% or more efficient, and introduces an additional pressure drop to the core of no more than 0.5 psi.

The trapping system is made up of 247 particle traps located in the core exit reflector. Of these, 210 replace the permanent hexagonal reflector elements and 37 replace the replaceable reflector element at the bottom of the control rod channels. This location in a typical region is shown in Fig. 3.1. Particles entering the coolant holes anywhere in the active core are entrapped as they enter the permanent reflector layer. Coolant passes through the trap, into the transition element, and out of the core.

The design utilizes a stagnation region and a simple bypass. Particles would enter the particle trap through 0.625-in.-diameter holes in the top cover that match the coolant holes in the reflector block above it. By their inertia,

Fig. 3.1--View showing location of particle trap in a typical region; typical particle trap is shown at right (LC-78407 and LC-78408)

the particles are carried into cavities in the center of the element (see Fig. 3.2). The entering coolant, which is somewhat stagnated, turns and enters the 1.25-in.-wide opening below the 45° inclined surface. The coolant is exhausted through 1.875-in.-diameter holes to the transition block below. Particles entering above the open cavity enter the cavity directly. Particles entering above the inclined surfaces rebound to the center support and again rebound back toward the coolant exit but strike below the opening and continue into the cavity. Particles that are deflected to either side of the center support fall into cavities elsewhere in the block. Holes 1.25 in. in diameter are drilled through compartment partitions to prevent particles from accumulating in any one compartment.

The exterior geometry of the trap elements is identical to the hexagonal reflector elements they replace. The handling hole profile is also unchanged.

The particle traps beneath the control rods function in the same way as the standard ones. Cavities are provided to entrap the particles while the coolant is deflected and exhausted through 1.75-in. slots along the side. The gas then flows into six 1.875-in.-diameter holes to the transition block and out of the core. The impact surfaces at the bottom of the control rod channel were not altered, and thus should resist dropped control rods in the same manner as the original design.

Both types of particle traps are constructed similarly from H-327 graphite. They are assembled from two pieces. The bottom part has a funnel-shaped inner perimeter that leads to a hollow space in the center portion of the block. Ribs 0.625 in. thick extend radially from the center of the block to the periphery and to the bottom of the block. This separates the hollow space into cavities (six for the regular element and four for the control rod element as shown in Figs. 3.2 and 3.3). Each cavity extends to a depth of 14.5 in. below the top of the block, leaving 1.125 in. of material at the bottom. The center portion is shaped to allow normal use of the fuel handling machine and extends to the same height as the outer perimeter. This gives center support to the top cover. Holes extend from the bottom of the element to the opening just beneath the sloping outer surface and blend into the 1.25-in. opening just below the 45° sloping surface. This opening extends completely around the inner perimeter of the block and penetrates to the center line of the 1.875-in.-diameter coolant exit holes.

The top face is a 2.0-in.-thick slab of graphite with a coolant hole pattern matching the hole pattern of the reflector element (or fuel block for the control rod element) above it. The three locating dowels are threaded and cemented in place in the standard pattern. The top and bottom sections are assembled with cement and threaded dowels.

Since the particle traps exhaust coolant through enlarged, 1.875-in.-diameter holes, the transition elements must have matching holes. These are turned toward the block center, so that coolant enters the support blocks as before. This change is necessary for all of the transition blocks. The new element designs required by the system are shown in Figs. 3.4 and 3.5.

Fig. 3.2--Exit reflector particle trap

Fig. 3.3--Reflector, lower control rod

Fig. 3.4--Bottom transition reflector element

Fig. 3.5--Bottom transition control rod

A series of tests was run to determine trap efficiency, pressure drop, and breakage of particles. The parameters of interest were the effects due to tube location, the change in efficiency as a function of gas flow, the effect of filling up the trap with particles, and the effect of a change in the exit opening. Practical considerations dictated using different type particles, starting with coated ZrO_2 , glass beads, and fuel particles of different mixes of interest. It is expected that the range of particle diameters may well play a role.

The apparatus for the fuel particle trapping experiment is schematically shown in Fig. 3.6. The driving gas is provided by the plant air supply. A molecular sieve column is used to eliminate water vapor and oil. The air pressure is adjusted with a high-throughput regulator. The air then flows through a 1-in.-ID copper tube that can be used for flow calibration. Two pressure pickoff points, 54 in. apart, are provided for a differential manometer. Next, the air enters an input manifold connecting the six tubes that represent a portion of the coolant tubes of the reactor fuel blocks. On top of the six tubes is the particle injection assembly, which can be positioned to drop the particle load into any selected tube. A pressure pickoff is available at the manifold.

The particle trap, which is attached to the bottom of the six tubes 6 ft below the injection point (see Fig. 3.6), contains a deflector block that shields the exhaust gas opening. The impact surface of the graphite deflector block is at an angle of 45° to the vertical. The effective trap depth is adjustable by vertical positioning of a rectangular graphite block. The exhaust gas window is designed for a height of about 1.8 in. but can be closed down. Data were obtained also for decreased openings of 1.48 and 0.73 in. The sides of the trap were made of a transparent plastic to permit the particle trajectories to be seen. The details of the trap model itself are shown in Fig. 3.7.

The weight of each charge of particles was chosen to be approximately equivalent to a 12-in. length of a fuel column in a 5/8-in. ID fuel hole. The particles were preweighed and loaded into the injection assembly, and the assembly was then installed on the input manifold at the desired tube position. The air flow was started and adjusted to the desired velocity. When a steady state was reached, the particles were injected by opening a ball valve, a typical load drop taking about 20 sec. The valve to the injection assembly was then closed and the gas flow stopped. The particles were then collected from the trap and bypass and the individual segments weighed to obtain the net amounts so that the trap efficiency could be calculated.

The trap efficiency for this preliminary test rig is shown in Fig. 3.8 as a function of tube position for coated ZrO_2 particles under no-flow and flow conditions. The no-flow conditions shows a minimum efficiency for the tube where particles partly hit the deflector and partly bypass. The efficiency under flow conditions continues to drop as the particles bounce from the deflector and rebound from the opposite wall. This same pattern was true for fuel particles, as shown in Fig. 3.9. The primary result from the examination of the trap efficiency as a function of tube position is that ricocheting and interaction with walls and particles play a major role. This was also demonstrated very clearly by high-speed motion pictures.

Fig. 3.6--Schematic view of trapping efficiency apparatus

Fig. 3.7--Schematic view of particle trap model

Fig. 3.8--Collection efficiency for coated ZrO_2 particles versus tube location - data from preliminary tests on particle trap model

Fig. 3.9--Collection efficiency for fuel particles versus tube location - data from preliminary tests on particle trap model

As part of the particle trap evaluation, high-speed motion pictures were made of coated ZrO_2 particles entering the particle trap test section. These pictures were analyzed for particle velocities and dynamic behavior during trapping.

Measured particle velocities varied from 13.5 to 30.6 ft/sec, with an average of 20.3 ft/sec (see Fig. 3.10). The air velocity for all measurements was maintained at about 110 ft/sec, and there was no significant preferential collimation of the discharging stream of particles. The envelope enclosing all but a few stray particles is a cone with an inclination angle of $15.5^\circ \pm 5^\circ$. Rebounding from surfaces of the trap was significant in determining the particle paths.

A lower efficiency was observed for holes No. 3 and No. 6. This was attributed to the large exit gas port. The opening was subsequently reduced to 1.25 in., where the pressure drop was just starting to be affected. This increased the efficiency by up to 10% for these hole locations. Therefore, the actual trap design was changed to utilize this opening width.

For tube position No. 3, the trap efficiency was determined as a function of gas flow and is shown in Fig. 3.11. It is not a very strong function of the flow. The flow in the 5/8-in.-diameter coolant tubes corresponding to a duct flow of 200 ft/min is about 100 ft/sec. Tube position No. 3 was chosen because this was the tube where the particles going straight just miss the deflector but are closest to the exit gas opening. This was done with the original exhaust opening of 1.781 in.

About 80 g of coke particles (64 to 125 μm) were injected in tube No. 2 with an air flow of about 110 ft/sec in the 5/8-in.-diameter tubes. The over-all coke collection efficiency was 93.5%.

Two mixtures were prepared each consisting of about 91 g of fuel particles with about 9 g of coke. Assuming a fuel packing fraction of 0.65, a particle density of 3 g/cm³, and a coke density of 1.93 g/cm³ gives a mixture of 0.18 g of coke per cubic centimeter of charge volume.

For one of the charges injected in tube No. 2 under the same flow conditions as above, the over-all efficiency was 93.1%. However, from the coke-only results, coke dust of considerably smaller diameter than 64 μm appears present originally. In addition, the material collected in the trap itself was sized for particles with diameters greater and less than 295 μm . Fuel accounted for 92.6% of the trapped material, the balance being coke. Analysis of the bypass collector showed 99.4% fuel with only 0.6% coke. The adjusted collection efficiency for fuel is then about 96% and about 76% for coke. This compares with 96.2% for fuel only obtained in independent tests on fuel particles. Both tests were run with an exhaust opening of 1.781 in.

An analysis was performed of the core exit including the support block region for the original design and with the particle trap included. The results indicated that the pressure drop will be increased from 0.1 to 0.2 psi by including the particle trap. This was approximately confirmed by data collected from the test. The analysis will be checked experimentally with a full-scale test of a particle trap and transition block.

Fig. 3.10--Velocity distribution of 100 coated ZrO_2 particles

Fig. 3.11--Collection efficiency for tube No. 3 versus flow velocity

Results from the test investigation are summarized below with no attempt to order them according to possible significance for design decisions.

1. Coated ZrO_2 particles provide a realistic dummy vehicle for initial trap evaluation.
2. Details of trap geometry strongly influence collection efficiency.
3. Particles make multiple collisions within the trap, which adversely affects trap efficiency.
4. No fuel particles were broken with an air flow of 100 ft/sec.
5. Particles show a pronounced "fanning" as they leave the coolant tube rather than streaming in a well-collimated beam.
6. Actual particle exit speeds are considerably less than expected from simple entrainment. Ricocheting within the coolant tube is assumed to be responsible.
7. A small pressure drop results from replacing a reflector element with the particle trap.

A full-scale test has been planned to verify trapping efficiencies and pressure drops for the entire range of scaled reactor flow conditions. The test will utilize a fuel element, the trap, and the transition element.

Thermal and Coolant Flow Design

Crossflow Correlations for COPE. The crossflow factors incorporated into the new code POKE have been refined and now appear to be resulting in a correct value being calculated for a core pressure drop. The new code has had some difficulties in converging to a solution when the orifice values are being calculated for balanced region outlet temperatures. Progress is being made as new methods of obtaining quick convergence are being tried.

Comparisons have been made between core temperatures and pressures computed by POKE and similar data computed by means of the flow analysis FLAC. Investigation has shown the discrepancies are due to differences in the geometric assumptions concerning certain crossflow gaps.

Regional Radial Power Factor Measurement at Startup. The feasibility of operating during low power at startup with the orifice valves wide open has been investigated. The purpose is to use the exit gas temperature from each region as an indicator of the region radial power factors. These data are desirable as an experimental check on the physics power calculations.

Such a procedure should not result in excessive fuel temperatures in the high-powered regions or unacceptable balance in steam generator gas inlet temperatures. The study has shown that up to 31% flow can be achieved at 18% power with a maximum fuel temperature of 1280°F. At 6% power, two of the steam generator modules receive helium at 538° and 876°F, and these values represent the extremes. At 10% power this becomes 653° and 918°F. At 18% power this balance further improves to 745° and 994°F. Although the imbalance is more than could be tolerated at full power, at these low temperatures the wide-open orifice procedure appears feasible.

Thermal Power Measurement and Error Analysis. During the normal operation of the plant, it is important to know the thermal power level of the reactor to ensure that the power limit set by the reactor operating license is not exceeded, to calculate the plant efficiency, and to estimate the fuel inventory in the core from the power history. There are a number of methods by which the reactor power can be calculated. These methods involve measuring various temperatures, pressures, flows, and other parameters and then calculating the reactor power. The following are typical methods of determining reactor thermal power:

1. Power Measurement from Fission Rate. Every fission that occurs in the reactor contributes on an average a fixed amount of energy to the thermal power. If the total number of fissions occurring in the reactor per unit time is measured, the reactor power can be calculated by multiplying this value by the average energy per fission. The fission rate in the reactor core can be estimated either from the neutron flux or the gamma flux at each point of the reactor as measured by neutron or gamma detectors. In the PSC reactor, there are six neutron detectors. These cannot be used to measure the absolute power of the reactor but can be used to provide an indication of the power level by calibrating them against the reactor power as calculated by some other method. These detectors need to be calibrated from time to time as the power distribution in the reactor changes as a result of burnup.
2. Power Measurement from Heat Absorbed by Helium in Core. The heat energy produced within the active core and reflector is removed by a downward flow of helium coolant. A large fraction of the coolant flows through the coolant holes in 37 regions, and a small portion bypasses the coolant holes and flows through the core in other flow paths. A part of this bypass flow enters the coolant holes of regions due to crossflow effects. The amount of crossflow to any particular region depends upon the orifice setting, the radial power factor, the age of that region, and the location of the region in the core. The total heat produced in the reactor is the sum of the heat absorbed by helium in each of the regions, the heat absorbed by the bypass flow, and the heat lost to the surrounding containment. The reactor power can thus be calculated by measuring the inlet and outlet conditions of helium in each of the regions and estimating heat flow to bypasses and containment.
3. Power Measurement from Heat Released by Helium in Steam Generators. The hot helium from the reactor core flows to 12 steam generator modules. The helium loses heat and its temperature drops as it flows through the steam generators. The reactor power can be calculated by making a heat balance on the helium flowing through the steam generators. The temperature of helium at both the inlet and outlet of each module is measured, and the mass flow of helium in each of the steam generator modules is estimated from the total flow rate of helium from the circulators.
4. Power Measurement from Heat Absorbed by Water and Steam in Steam Generators. Feedwater flows to the steam generators in two loops, and each loop feeds six steam generator modules. The feedwater, as

it flows through the steam generator, absorbs heat from the helium and is converted into steam. The main steam, after driving the high-pressure turbine, enters the steam generator reheater sections, where it absorbs additional heat before driving the intermediate- and low-pressure turbines. By measuring the inlet and outlet conditions of both main steam and reheat steam for each loop, the reactor thermal power can be calculated.

In each of the above methods, the parameters are measured within a certain error limit, and the merit of each method depends on the accuracy with which the required measurements can be made. A detailed analysis of the required measurements and their errors was made, and the results were used to estimate the errors for the three heat-balance methods of calculating the reactor power. The errors at full load for each method are given in Table 3.1. The errors in each of the methods depend on the load conditions.

Table 3.1
PERCENTAGE ERROR IN REACTOR POWER DETERMINATIONS AT FULL LOAD

<u>Method</u>	<u>Percent Error</u>
Reactor power from heat balance of helium in the core	22
Reactor power from heat balance of helium in steam generator	6.1
Reactor power from heat balance of steam in steam generator	2.55

The $\pm 22\%$ error in the reactor power calculated from the heat balance of helium in the reactor core is mainly due to crossflow effects and also to an uncertainty in flow measurement through the orifices.

The main contribution to the $\pm 6.1\%$ error in the reactor power calculated from the heat balance of helium in the steam generators is the error in helium flow measurement through each of the steam generator modules. The assumption of equal flow of helium through each module is not valid at part loads.

The main contribution to the $\pm 2.55\%$ error in the reactor power calculated from the heat balance of steam in the steam generators error is the error in feedwater flow measurements. This method of calculating reactor power is more accurate than the other two methods.

Fine-mesh Core Thermal Calculations

Fine-mesh radial power density peaking factors have recently been calculated for Reference Design 9 (thin-buffer-zone design). For the new data, individual core columns have been divided into 24 triangular sections, with power histories available on tape for each triangular section.

An analysis is in progress to determine the effect on core operating temperatures resulting from using the finer-mesh radial power density peaking factors in core thermal calculations. In previous calculations column-average power densities have been used to determine local core power. To date the analysis has been limited to selected triangular sections in the peripheral core regions (Regions 20-37), where maximum temperatures are expected to occur.

Fission-product Transport

Primary Coolant Plateout Monitor. Construction of the test probe and the autoclave fixtures for the environmental test is under way. The hot O-ring sealing tests will be performed prior to the long-term environmental tests. The drawings have been updated to reflect minor design improvements made during the machining of this prototype probe.

The probe locking concept has been successfully tested for engagement and release at ambient temperature. The reverse torque necessary to override the ratchet lock will be tested as soon as the test probe is assembled. No major difficulties have been encountered, although certain modifications and improvements have suggested themselves in the course of the operation and are being incorporated into the design.

The construction drawings for the plateout probe and the penetration plug have been initiated. These drawings will be updated as experience is obtained from the assembly and testing of the test unit. Figure 3.12 shows the plateout probe with the five diffusion tubes, the plateout sleeve, and the coiled charcoal trap at the sample outlet. The Fiberfrax insulation and the graphite thermal bond are arranged to keep the diffusion tube temperatures close to the return coolant gas temperature.

Considerable effort has been devoted to determining the best method of controlling the flow through the diffusion tubes. The design requires a 3-psi pressure loss in the outlet coolant samples and a 13-psi loss in the return coolant sample. Restrictors of various types have been considered and, at present, the prime candidate is a porous plug of sintered stainless steel. Further efforts are being made to determine the best pore size, method of fabrication and calibration, and over-all design for reliability of operation in the reactor environment. Currently, variations in the pore structure affecting calibration and operation are being studied. These occur in the material, the shaping process, the joining process, and possibly in operation because of the entrapment of particulate matter. An effort is being made to minimize the possibility of plugging during normal reactor operation by placing Fiberfrax filters in front of the sintered stainless steel plugs.

It has been tentatively decided to operate four of the five diffusion tubes at the same flow rate. One of these will be the coolant return sample. The other three will be core outlet samples with one sample inlet facing upstream, one downstream, and one perpendicular to the coolant flow. The remaining inlet tube (perpendicular to the flow) will be operated at the minimum obtainable flow rate. It is felt that this arrangement will result in a considerable amount of information about particulate-borne fission products.

Fig. 3.12--Plateout probe assembly showing five diffusion tubes, plateout sleeve, and activated charcoal trap

MATERIALS

The objectives of this subtask are to:

1. Determine the characteristics of beds of bonded coated particles by means of fabrication technique investigations, thermal property tests, vibration and impact tests, and measurements of packing density and loading uniformity.
2. Select the type of particle bed to be used in the reactor based on the above work and work carried out in other parts of the program, taking into account the expected characteristics and relative ease of reprocessing, and determine the suitability of the selected system by means of further tests of the type described above.
3. Determine the characteristics of various candidate graphites for use in the reactor fuel elements by means of measurements of physical, thermal, and mechanical properties of these graphites.
4. Determine the characteristics of various control materials by means of measurements of physical, mechanical, chemical, and thermal properties.
5. Determine the irradiation stability of candidate graphites for use in the fuel elements by means of irradiation tests conducted in the Engineering Test Reactor at the temperatures and to the integrated fast flux expected in the plant, followed by postirradiation examination. Approximately five capsules are planned for this part of the program.
6. Establish the effect of fast flux on fuel particle coating structural integrity as a function of coating structure and exposure, by means of irradiation tests in the Engineering Test Reactor and other reactors on unfueled samples of particle coatings, postirradiation measurements of coating properties, such as dimensional changes, flexural strength and crystallite spacing, and subsequent reirradiation to design integrated fast flux and remeasurement. These irradiations will be carried out in the graphite capsules used in 5 above, and in the coated particle capsules used in 7 below.
7. Determine the irradiation stability of coated fuel particles for use in the reactor by means of a series of (1) short-term tests on coated fuel particles to select a coated fuel particle(s) having over its design life a high degree of structural integrity suitable for use in a nonvented fuel element, followed by (2) longer-term irradiation tests conducted at design temperature, to design burnup and to a significant fraction of the integrated fast flux expected in the plant. The "significant" integrated fast dosage will be determined from the results obtained in 6. Capsule tests will be followed by postirradiation examination of the particles.

Particle Fuel Systems

A study was made to investigate methods of further improving the radiation stability of the matrix material of bonded fuel rods. Included were studies of filler materials, binder materials, matrix blends, and some process modifications.

Filler Materials. One of the main components of the fuel rod matrix is the graphite powder filler material. The properties of several candidate filler materials were evaluated to determine how close they came to meeting the following requirements: high L_c (crystallite size), isotropic structure on the micro scale, high density, reasonable purity, correct particle size for injection ($\sim 30 \mu\text{m}$), and availability in plentiful supply. Nine carbon and graphite materials were evaluated in the first phase of the characterization program, each being subjected to the tests listed in Table 3.2. The various materials are listed in Table 3.3.

After initial detailed evaluation of these materials, some additional materials were obtained. These materials were subjected only to the most significant tests, which included bright-field and sensitive-tint metallography, electron microscopy, X-ray diffraction, and helium pycnometer density. These materials (also listed in Table 3.3) are referred to as Phase 2 filler materials. The details of the test procedures used are reported in GA-9374 (Ref. 1).

The natural flake materials are very flaky and acicular in cross section (Fig. 3.13), give high aspect ratios (see Table 3.4), and are highly anisotropic under sensitive-tint metallography. The GP-38 and AGL-1242 materials are somewhat more isotropic (Fig. 3.14), however, and AXF-5Q graphite is almost completely isotropic (Fig. 3.15). The thin flat flakes of the graphite flake materials are quite different from GP-38, which is also flaky; the AGL-1242 is a more chunky material (Figs. 3.16 and 3.17). Carbon black, a possible filler material additive, has a structure of agglomerated submicron-size spheres typical of that material (see Fig. 3.18).

The as-received calcined coke has an "onion-skin" structure but breaks down into fairly isotropic chunks (see Fig. 3.19). The H-386 coke appears fairly isotropic in the large particles, but on grinding breaks up into acicular, anisotropic particles (see Fig. 3.20). Even after heat treatment, this condition is generally true. The H-378 material is the only other graphite that appears reasonably isotropic after grinding.

The size distribution of the various filler materials were determined by three main methods, not all of which were applied to all of the materials. In particular, the Phase 2 cokes were sized only visually from the transmission electron photomicrographs. The methods used were (1) holography, (2) the micromerograph, and (3) the Coulter counter, discussed in detail in GA-9374. Instrument No. 1 measures linear dimensions of the individual particles, No. 2 integrates weight distributions, and No. 3 measures individual particle volumes.

The results of the holographically determined size distributions are shown in Table 3.5. It should be noted when comparing the data listed in this table with results obtained by other methods that these measurements are based on linear dimensions and can be affected by the shape of the particles (whether platelike or isotropic, etc.).

The micromerograph results are shown in Table 3.6, and the two curves obtained for AGL-1242 and AXF-5Q in Figs. 3.21 and 3.22. The curve for AGL-1242 is of the normal shape, indicating a normal distribution of particle sizes. The curve for AXF-5Q, however, has an extended linear portion, which indicates

Table 3.2

CHARACTERIZATION TESTS FOR FUEL ROD FILLER MATERIALS

Phase 1

Metallography (bright field and sensitive tint)
 Scanning electron microscopy
 Transmission electron microscopy
 Particle sizing by holography
 Particle sizing by micromerograph
 Particle sizing by Coulter counter
 Tap density
 Liquid-immersion density
 Helium-pycnometer density
 Mercury-porosimeter density versus pressure
 Bulk density versus pressure
 Spectrochemical analysis
 Sulphur analysis
 Ash content
 X-ray diffraction
 Specific surface area

Phase 2

Metallography (bright field and sensitive tint)
 Transmission electron microscopy
 Helium-pycnometer density
 Spectrochemical analysis
 X-ray diffraction

Table 3.3

FILLER MATERIALS CHARACTERIZED

Phase 1

Natural flake	Natural graphite, ground
M-850	Natural graphite, finely ground
M-975	Natural graphite, finely ground
M-6353	Natural graphite, finely ground
P-6353	Natural graphite, finely ground
GP-38	Graphitized coke
AGL-1242	Powdered isotropic coke
AXF-5Q	Ground isotropic coke
Carbon black	---

Phase 2

Calcined coke	---
H-378	Isotropic graphite (log)
H-386	Calcined petroleum coke flour
H-387	As above, heat treated
H-388	As H-389, heat treated
H-389	Purified calcined petroleum coke flour

M-29274-1

500X

M-29272-1

500X

Fig. 3.13--Photomicrographs of two graphite flake materials: M-975 (above) and M-850 (below). The only apparent difference between the two materials is size range, as may be seen by comparing the two photomicrographs

M-29058-3

500X

M-29059-3

500X

Fig. 3.14--Microstructures of two candidate filler materials: GP-38 (above) and AGL-1242 (below). GP-38, a graphitized coke, was PSC reference material at outset of program. AGL-1242 is a powdered isotropic coke of somewhat larger size

M-29062-3

500X

Fig. 3.15--Microstructure of AXF-5Q isotropic ground coke. Sensitive-tint metallography showed material to be truly isotropic, whereas materials pictured in Fig. 3.15 were somewhat anisotropic. Majority of the AXF-5Q material appeared to be less than about 15 μm

Fig. 3.16--Scanning electron photomicrographs of graphite flake filler materials: natural flake (above) and M-975 (below); the very thin, platelike structure of this type of material is quite obvious in the M-975

Fig. 3.17--Scanning electron photomicrographs of filler materials; GP-38 (above) and AGL-1242 (below). The GP-38 is a flaky material, but not like natural flake graphite; AGL-1242 has a more chunky appearance

P1556-2

6000X

Fig. 3.18--Transmission electron photomicrograph of carbon black, individual spherical particles may be readily seen, even in agglomerated masses

M-29717-2

50X

M-29718-1

250X

Fig. 3.19--Microstructure of calcined coke investigated in second phase of the program: as-received (above) and $-30\ \mu\text{m}$ fraction. Onion-skin structure is apparent in as-received material but is not present in the $-30\ \mu\text{m}$ fraction; sensitive-tine metallography showed fine material to be almost isotropic

M-29869-1

250X

M-29870-1

750X

Fig. 3.20--Photomicrographs of H-386 coke: as-received (above) and $-30\text{ }\mu\text{m}$ fraction. Individual light and dark areas on the particles showed anisotropy under sensitive-tint metallography; also, $-30\text{ }\mu\text{m}$ particles are more anisotropic in shape than those pictured in Fig. 3.20. H-389 coke appeared practically identical to H-386

Fig. 3.21 --Micromerograph size distribution analysis for AGL-1242 graphite

Fig. 3.22--Micromerograph size distribution analysis for AXF-5Q graphite

Table 3.4
ASPECT RATIOS OF FILLER MATERIALS

<u>Material</u>	<u>Aspect Ratio^a</u>
Natural flake	18 to 20
M-850, -975, -6353	18 to 20
GP-38	~5
AGL-1242	~4
AXF-5Q	1 to 2
Carbon black	1

^aThe ratio of the maximum to the minimum dimension on an average particle.

abnormal size distribution and is probably indicative of agglomerated particles. This is reflected in Table 3.6 for AXF-5Q and carbon black, for which it is known that all the individual spherical particles are less than 1 μm in diameter (see Fig. 3.18).

A Coulter counter size distribution was obtained only for AGL-1242; this is shown in Fig. 3.23. As can be seen, it is the inverse of Fig. 3.21. The distributions obtained by the two methods are very similar; that they are not identical may be due to the fact that Stokes law is used to obtain the micromerograph results, and this assumes sphericity of the particles.

A comparison of the density determinations on the initial filler materials obtained by five different methods are presented in Table 3.7. For the Phase 2 materials, only the helium pycnometer densities were determined for each material both as-received and after screening to $-30\ \mu\text{m}$ (see Table 3.8).

Of the values reported in Table 3.7, the helium pycnometer, liquid immersion, and mercury porosimeter density data are comparable, representing the densities of individual particles. The tap and bulk densities, however, are measures of the density of the bulk material (the bulk density is the tap density at atmospheric pressure).

In most cases the helium pycnometer and liquid immersion densities are very similar, suggesting that there is little porosity accessible to helium at atmospheric pressure (GP-38 seems to be the one major exception). The mercury porosimeter densities, however, in many cases are higher, suggesting that intrusion of mercury into voids or cracks does take place at 2000 psi and above (e.g., natural flake graphite, M-975, and GP-38). It will be noted that the three values listed for carbon black are extremely close, as would be expected for a material composed of submicron-sized spheres.

Porosimetry curves for natural flake graphite, AXF-5Q, and carbon black, shown in Fig. 3.24, illustrate the variation in pressure required to wet the

Table 3.5

HOLOGRAPHICALLY DETERMINED SIZE DISTRIBUTIONS OF FILLER PARTICLES

Material	Percentage by Number of Particles in Size Range Indicated							
	<1 μm	1-5 μm	6-10 μm	11-15 μm	16-20 μm	21-30 μm	31-40 μm	
GP-38	0	93.7	3.45	0.8	1.25	0	0.8	
AGL-1242	0	98.8	0.45	0.45	0.19	0.05	0	
AXF-5Q	0	93	4.7	0.94	1.36	0	0	
M-850	0.8	94.8	4.4	0	0	0	0	
Carbon black	100	0	0	0	0	0	0	
Material	<1 μm	1-5 μm	6-20 μm	21-40 μm	41-100 μm	100-200 μm	200-300 μm	300-500 μm
Natural flake	18.13	36.7	14.1	2.82	5.65	14.1	4.25	4.25

Table 3.6

RESULTS OF MICROMEROGRAH SIZE DISTRIBUTION
ANALYSES OF FILLER MATERIALS

Material	Percentage of Sample Below Size Indicated (μm)		
	5%	50%	95%
M-850	2.4	5.7	11.3
M-975	2.4	6.5	13.0
M-6353	2.1	5.4	12.0
P-6353	2.2	6.5	13.5
GP-38	3.4	19.0	47.0
AGL-1242	2.8	11.8	29.5
AXF-5Q	2.9 ^a	10.5 ^a	88.0 ^a
Carbon black	2.1 ^a	5.4 ^a	15.5 ^a

^aNot fully deagglomerated.

Table 3.7

DENSITY DETERMINATIONS ON PHASE 1 FILLER MATERIALS

Material	Tap Density (g/cm^3)	Helium Pycnometer Density (g/cm^3)	Liquid Immersion Density (g/cm^3)	Mercury Porosimeter Density at 2000 psi (g/cm^3)	Bulk Density at 5000 psi (g/cm^3)
Natural flake	0.69	2.18	2.13	2.26	2.11
M-850	0.35	2.07	2.18	2.15	1.66
M-975	0.40	1.98	2.00	2.29	1.91
M-6353	0.37	2.06	2.05	2.13	1.82
P-6353	0.36	2.05	2.04	2.18	1.85
GP-38	0.32	2.05	1.93	2.21	1.71
AGL-1242	0.66	2.11	2.13	1.98	1.55
AXF-5Q	0.57	2.13	2.08	2.18	1.56
Carbon black	0.48	1.80	1.81	1.81	1.25

Fig. 3.23--Coulter counter size distribution analysis for AGL-1242 graphite

Fig.3.24 --Density of filler materials versus pressure during porosimetry

Table 3.8
HELIUM PYCNOMETER DENSITIES OF PHASE 2 FILLER MATERIALS

Material	Density (g/cm ³)	
	As Received	-30 μ m Fraction
Calcined coke	1.92	1.99
Calcined coke after 1 hr at 2800°C	---	2.22
AGL-1242 ^a	2.19	2.18
H-378	---	2.23
H-386	2.03	2.07
H-387	2.17	2.24
H-388	2.21	2.23
H-389	1.99	1.99

^aNew material.

different size particles. The large, natural flake graphite is being wetted at atmospheric pressure, the 5- to 15- μ m sized AXF-5Q only above about 150 psi, and the submicron-sized carbon black only above about 800 psi.

Bulk density curves versus pressure are shown in Fig. 3.25 for several of the filler materials. It can be seen that the flat, platelike particles of natural flake and M-975 materials pack down much more efficiently than the AGL-1242 particles, for example.

The spectrochemical analyses of the various filler materials are listed in Tables 3.9 and 3.10, and the sulphur and ash contents of the Phase 1 materials appear in Table 3.11. The X-ray diffraction parameters of the various materials appear in Tables 3.12 and 3.13, and the specific surface areas of the Phase 1 materials are also listed in Table 3.12. AXF-5Q graphite and carbon black are the two purest materials tested, whereas the most impure are the natural flake graphites and calcined coke. As anticipated, the natural flake graphites have very large crystallite sizes, while that of carbon black is extremely small. The BET measurements indicate that GP-38 and AGL-1242 both have large surface areas, whereas the more isotropic AXF-5Q and carbon black materials are considerably less. The very low value found for natural flake graphite is due to the large size of its particles.

Various trials have shown that fillers, when mixed with the binder, will inject into the die providing the +30 to -43 μ m size fraction does not exceed about 5%. P-6353, as received, fulfills this requirement, but grade H-378 is delivered in 18-in.-diameter by 24-in.-long blocks. A method has been developed for comminuting the H-378 blocks to injectable powder. The blocks are sawed into small pieces and fed into a 3-in. jaw crusher that in turn provides

Fig. 3.25--Bulk density of various filler materials as a function of pressure

Table 3.9

SPECTROCHEMICAL ANALYSES OF PHASE 1 FILLER MATERIALS

Material	Concentration of Given Element (ppm)																
	Ag	Al	B	Ba	Ca	Cr	Cu	Fe	Mg	Mn	Mo	Ni	Pb	Si	Ti	V	W
Natural flake	20	100	<5	<5	(a)	<100	10	1000	20	<100	<10	<40	60	600	<10	<5	<400
GP-38	8	80	40	1	200	100	1	1000	10	200	10	20	<6	400	20	1	100
AGL-1242	2	40	<0.5	4	400	<10	8	200	20	<10	<1	6	<6	100	10	<0.5	<40
AXF-5Q	2	2	<0.5	<0.5	(a)	<10	<1	6	1	<10	<1	<4	<6	<10	<1	<0.5	<40
Carbon black	2	8	<0.5	1	(a)	<10	1	10	10	<10	<1	<4	<6	200	<1	<0.5	<40
M-975	1	>1000	<0.5	10	40	20	20	>1000	400	200	20	20	20	>1000	100	40	<40
M-850	1	200	<0.5	4	80	<10	20	>1000	100	100	<1	8	40	1000	40	2	<40
P-6353	0.1	40	<0.5	4	40	<10	20	400	10	10	<1	6	<6	600	20	<0.5	<40

^aNot determined.

Table 3.10

SPECTROCHEMICAL ANALYSES OF PHASE 2 FILLER MATERIALS

Material	Concentration of Given Element (ppm)																				
	Ag	Al	B	Ca	Co	Cr	Cu	Fe	Mg	Mn	Mo	Na	Ni	P	Pb	Si	Sn	Sr	Ti	V	Zr
Calcined coke (unmilled)	<0.1	600	2	<1000	100	10	8	1000	400	20	40	<1000	1000	400	<6	>1000	<6	40	60	40	4
Calcined coke (milled)	<0.1	200	<0.5	600	100	<10	8	1000	80	10	20	1000	200	400	<6	>1000	<6	40	20	8	<0.5
AGL-1242	0.1	40	<0.5	100	<4	<10	10	400	20	<10	10	<60	10	<100	<6	600	<6	<40	10	8	<0.5
H-386	0.1	100	<0.5	>1000	20	200	400	1000	200	40	20	>1000	1000	<100	80	200	40	<40	40	1000	4
H-387	0.1	10	1	<100	<4	<10	<1	10	<0.5	<10	<1	<60	100	<100	<6	<10	<6	<40	20	100	<0.5
H-388	<0.1	8	40	<100	<4	60	8	400	2	<10	<1	<60	40	<100	<6	60	<6	<40	100	200	6
H-389	<0.1	100	2	200	<4	200	400	1000	20	20	<1	1000	200	<100	<100	200	40	<40	10	200	<0.5
H-378	<0.1	20	8.0	200	<4	10	20	80	40	<10	10	---	20	---	20	400	---	---	40	10	---

Table 3.11
SULPHUR AND ASH CONTENT OF PHASE 1 FILLER MATERIALS

Material	Sulphur Content (ppm)	Ash Content (%)
Natural flake	75.8	0.58
M-850	50	0.61
M-975	75	5.00 ^a
M-6353	40	0.11
P-6353	30	0.42
GP-38	80.9	0.09
AGL-1242	66.7	0.07
AXF-5Q	<1	<0.02
Carbon black	2	0.04

^aAfter oxidation this material left behind a red residue thought to be Fe_2O_3 .

Table 3.12
STRUCTURAL PARAMETERS OF PHASE 1 FILLER MATERIALS

Material	Crystallite Size L_c (Å)	Interlayer Spacing C_o (Å)	Specific Surface Area (m^2/g)
Natural flake	~8000	6.71	0.63
M-850	~1800	6.71	8.7
M-975	~2300	6.70	6.8
M-6353	~2300	6.72	6.2
P-6353	~3200	6.71	5.4
GP-38	1000	6.73	16
AGL-1242	405	6.75	16
AXF-5Q	540	6.74	5.3
Carbon black	20	7.09	7.3

Table 3.13
X-RAY DIFFRACTION PARAMETERS OF PHASE 2 FILLER MATERIALS

Material	Crystallite Size L_c (Å)		Interlayer Spacing C_o (Å)	
	As Received	-30 μ m Fraction	As Received	-30 μ m Fraction
Calcined coke	---	27	---	6.89
Calcined coke after 1 hr at 2800°C	---	463	---	6.73
AGL-1242 ^a	405	425	6.74	6.74
H-378	505	463	6.74	6.74
H-386	28	30	6.94	6.92
H-387	705	---	6.72	---
H-388	650	---	6.72	---
H-389	27	---	6.93	---

^aNew material.

feed to a micropulverizer. The powdered product is ball-milled for 12 hr and then screened to -43 μ m, the +43 μ m material being returned to the ball-mill. The -43 μ m material is separated in an air cyclone into coarse and fine fractions. The fines are tested for the fraction of +30 μ m and, if it is less than 5%, it is used for injection. If the 5% is exceeded, it is returned for ball-milling. To date, the fraction of +30 μ m material has been about 0.5%. Sufficient filler has been produced to satisfy the demand for all F-24 capsule specimens and replicates, but further tests are being performed to reduce the recycle fraction.

The studies of filler materials discussed above were designed to narrow the choice to one candidate in each of the two main types of graphite powder: natural flake graphite, and manufactured isotropic graphite. After a careful evaluation of the data, P-6353 was selected as the natural flake filler material. Although this material is quite anisotropic (as are all natural flakes), the large crystallite size ($L_c = 3200$ Å) and high density should give good radiation stability. The material is apparently in abundant supply. Its main disadvantage is an iron impurity level of 400 to 600 ppm, but it appears that this can be tolerated.

AXF-5Q isotropic graphite powder showed promising technical features, however, availability appears to be a problem.

Although the graphitized cokes, H-387 and H-388, and the calcined coke appear to be isotropic on a macro scale, the individual particles tended to

be somewhat anisotropic when the material was ground to the required 30- μ m size.

After careful evaluation, ground H-378 graphite was selected as the isotropic graphite filler. The individual ground particles of this material appear to have superior isotropic structure based on sensitive-tint metallography. In addition, the crystallite size, density, and purity are as good as or better than that of the other materials. H-378 also appears to be in abundant supply.

Binder Materials. One of the main efforts in the fuel rod improvement program was directed toward increasing the graphitic character of the binder coke that is the continuous phase of the matrix material in order to give the matrix greater radiation stability. One measure of the degree of graphitic character of the binder coke is the crystallite size, L_c . This parameter was determined by X-ray diffraction on several candidate binder cokes that were heat-treated at various temperatures. The materials are listed in Table 3.14 with their coking yield. Measured crystallite size versus heat-treatment temperature for the various binder cokes is plotted in Fig. 3.26.

One of the most graphitizable materials appears to be catalyzed acenaphthylene (ACN). This material can be cured at about 150°C to form a fairly hard and strong green rod, and subsequently can be heat-treated to a fairly graphitic material. The other graphitizable binders, decacyclene, isotruxene, and the two coal tar pitches, all are thermoplastic materials and are cured or hardened by carbonization. The phenolic and furfuryl resins, which are easily cured, appear to be nongraphitizing.

Additional information on crystallite size as a function of heat-treatment time (at 2000°C) for the ACN and No. 30 pitch binders is shown in Fig. 3.27; it can be seen that about 80% of the crystallite size growth is completed in the first 15 to 30 min of heat treatment.

The graphitizable binders at room temperature are solid materials. Blending with filler materials and rod injection must therefore be done at elevated temperatures. The feasibility of hot injection has been demonstrated, however, and a hot-injection molding process using either ACN or pitch binder has been developed.

Polymerization of ACN is initiated by the addition of a peroxide catalyst to the hot, melted material. As heating is continued (at 95° to 100°C), increased polymerization becomes evident by the increase in viscosity of the melt. It was found that the hot injection of the binder mix is greatly helped by allowing the polymerization to proceed and the viscosity to build up to a syrupy consistency before the filler is added and the hot injection is started.

The hot mix has a limited pot life (about 1 hr) at the melt temperature, 95° to 100°C. However, it can be cooled to room temperature and solidified at any point. The solidified mix appears to be stable for several days and can be heated and remelted.

Table 3.14

COKING YIELD OF CANDIDATE BINDER MATERIALS		
Sample No.	Binder Material	Coking Value (%)
B-1	R-72S phenolic resin + catalyst	34.4
B-2	8251 furfuryl resin + catalyst	53.5
B-3	Decacyclene (organic compound)	88.1
B-4	Truxene (organic compound)	63.8
B-5	Acenaphthylene (ACN) + catalyst	36.9
B-6	Divinyl benzene (DVB) + catalyst	24.7
B-7	Divinyl benzene and acenaphthylene + catalyst	29.9
B-8	Isotruxene resin	69.7
B-9	No. 30 coal tar pitch	60.7
B-10	No. 15V coal tar pitch	53.4

Although ACN is available in laboratory quantities from special organic chemical suppliers, there appears to be some difficulty in finding sources of the material in the large quantities required for production. On the other hand, coal tar pitch is plentiful. The two pitches being considered, No. 30 and 15V, appear to be quite similar, except that the 15V has a somewhat lower softening temperature and is supposed to contain less insoluble material. This might possibly permit lower processing temperatures and a slightly higher filler content.

Matrix Materials and Hot-injection Process. Early in the development program a number of rod samples representing various combinations of binder and filler materials were fabricated. The binders used were R72S (the initial reference mix) and 8251 furfuryl alcohol resin. The filler materials were GP-38 (-30 μ m fraction), GP-38 and 15 wt-% carbon black, AXF-5Q graphite (ground to -44 μ m), M-975, and P-6353 in concentrations of 21, 30, and 35 to 40 wt-% of the binder mix. While 35 to 40 wt-% appeared to be the upper limit for injection of most of the materials, formulations containing 50 wt-% AXF-5Q were successfully injected.

After fabrication, the rods were examined and the dimensions measured. A metallographic mount was also made of each type of rod and it was examined metallographically. In general, the test results tend to confirm earlier observations. With only occasional exceptions, the general trends observed from the examinations are:

1. Increasing amounts of filler increase the matrix density (a maximum of 1.13 was obtained with 50 wt-% AXF-5Q filler and 8251 binder).
2. Increasing amounts of filler caused more diametral shrinkage and carbonization and more resultant broken particles.
3. The finer the filler particle size used, the more fine-grained and dense (on a micro scale) the matrix material.

Fig. 3.26--Crystallite size, L_c , of candidate binder cokes as a function of heat-treatment temperature

Fig. 3.27--Crystallite size, L_c , of ACN and No. 30 pitch binders as a function of duration of heat treatment at 2000°C in an atmosphere of 45-psig helium

4. A slightly higher broken-particle trend, along with what appeared to be more bonding to the fuel particle surfaces, was noted when the 8251 resin binder was used.

Subsequent to the fabrication of these rods, efforts were directed toward the development of the hot-injection molding process in order to utilize the ACN and pitch binders. A hot-injection molding process was successfully developed and a number of rod samples were fabricated. These samples included both ACN and pitch binders with natural flake and with isotropic graphite fillers. The rods were carbonized with the standard 18-hr heatup to 1000°C cycle. After carbonization, the rods were heat-treated for 1 hr at 2000°C for graphitization. In general, the rods held together well, and the matrix appeared to be fairly uniform. Very few broken particles were noted.

One problem encountered with the ACN-binder rods is that of maintaining dimensional stability during carbonization. The initial 1-in.-long rod samples were carbonized with no restraint in a graphite holder and showed a tendency to swell or "barrel" in the middle. Also, slight swelling or "dishpanning" effects were noted on the ends. Although the ACN resin system is cured to a thermosetting condition at 150°C and remains solid at higher temperatures it appears to go through a soft or plastic stage during carbonization. It is most likely that the swelling occurs at this stage. This problem was resolved by carbonizing the rods in a loosely restraining graphite holder. For rods with a diameter of 0.512 in., graphite holders with hole diameters of 0.518 to 0.520 in. were satisfactory. The rods carbonized in this manner are fairly uniform, with diameters of 0.513 to 0.515 in., and do not have a significant number of broken particles.

The dimensional stability of the pitch-binder rods during carbonization is maintained by packing them in graphite flour for this operation. These rods also appear to have very few or no broken particles and are uniform and dense. The pitch binder rods could not be carbonized in the graphite holders because of tight bonding to the holder walls. With the graphite-powder packing technique, excess pitch binder is absorbed by the powder, which is brushed away after carbonization.

The thermal conductivity measurements made on specimens with the new fuel rod matrix formulations are listed in Table 3.15. These values are higher than the values for specimens with the previous matrix formulations by a factor of almost two. This is believed to be due to three conditions: higher filler loading, greater over-all matrix density, and greater degree of graphitization of the matrix.

The coefficient of thermal expansion of a rod of the ACN formulation (see Table 3.15) was also measured and found to be $4.82 \times 10^{-6} \text{ }^{\circ}\text{C}^{-1}$. This compares with a value of $5.40 \times 10^{-6} \text{ }^{\circ}\text{C}^{-1}$ for an earlier matrix formulation.

Following the selection of the binder material, additional work was done on the hot-injection process with No. 15V pitch. Since pitch is a thermoplastic material, the matrix mix can be reheated to its softening temperature

Table 3.15
THERMAL CONDUCTIVITY AT 900°C (1652°F) OF TRISO PARTICLES
BONDED WITH VARIOUS MATRIX MATERIALS^a

Specimen No.	Matrix for TRISO Particles	Thermal Conductivity	
		Cal/cm-sec-°C at 900°C	Btu/ft-hr-°F at 1652°F
1	65% std ACN binder, 35% P-6353 (natural flake filler)	0.038	9.2
2	65% std ACN binder, 35% P-6353 (natural flake filler)	0.039	9.4
3	70% No. 15V pitch, 30% P-6353	0.044	10.6
5	75% No. 30 pitch, 25% P-6353	0.0355	8.6
6	50% std ACN binder, 50% H-378 filler	0.028	6.8
7	60% No. 30 pitch, 40% H-378 filler	0.035	8.4

^aMatrix was hot-injected, carbonized, and heat-treated 1 hr at 2000°C.

repeatedly with no danger of the material setting up. Two basic formulations were developed for use with the hot-injection process. One consisted of 70 wt-% No. 15V pitch with 30 wt-% 6353 natural flake graphite, and the other consisted of 55 wt-% No. 15V pitch with 45 wt-% H-378 graphite powder (95%, -30 μ m). The binder and filler were thoroughly mixed while heating at approximately 175°C. The mix was transferred to the heated chamber of the injection apparatus and injected into a mold loaded with the required fuel particles maintained at 175°C. The mold was cooled to room temperature, and the green rods were ejected. For carbonization, the green rods were packed in a bed of graphite flour and heated to 1000°C over a period of 18 hr. After carbonization the rods were removed and the excess graphite flour brushed off. The rods were then given the desired heat treatment (1800° or 2000°C for 1/2 hr) to effect a degree of graphitization in the binder coke. This appears to be about the maximum temperature to which the particles can be exposed with no evidence of core migration.

Since these formulations were designed to contain the maximum practical amount of filler material, the resulting mixes are slightly more viscous than previous mixes. In a related area, the matrix mix has shown somewhat different behavior with different types of fuel particles. Using the reference mix, solid, uniform rods were obtained with fuel particles prepared by propane coating gas. However with propylene gas particles, some problems with incomplete injection of rods were encountered. These were overcome for the most part by adjusting the matrix formulation to 27% natural flake filler-73% pitch and using a higher injection pressure.

Studies were made to determine the maximum length of fuel rod that could be injected with the new mixes. Subsequent trials with 6-in.-long columns of fuel particles indicated that the matrix mix penetrated only about 3 in. down the particle column. This, however, would be satisfactory since one design calls for 2-in.-long rods. When the filler material in each formulation was decreased by 5 wt-%, the full 6 in. of rod could easily be injected.

Another approach to fuel rods with greater irradiation stability would be to use a matrix material having much higher filler content and density, such as that obtained with the hot-pressing process. Hot-pressed fuel compacts of the Peach Bottom type have a matrix that is 85 to 90 wt-% filler and has a density of around 1.85 g/cm³. However, these compacts have a fuel loading of only 19 to 21 vol-%, which is far below that required for Fort St. Vrain.

One possible method of obtaining a high-density matrix and a high-volume loading of fuel particles is by overcoating the fuel particles with a minimum amount of green matrix material, then hot pressing to form the fuel body. The individual overcoat around each particle should provide a cushion to prevent point-to-point contact and therefore allow a much higher volume loading of particles without breakage than the admixing technique. The possibilities of this approach were investigated briefly.

Several samples were fabricated including some by the regular admix technique for comparison, some with a buffer-PyC coating over the regular particle, and some by the matrix-overcoat process. These samples and some of their characteristics are listed in Table 3.16. Using 6353 natural flake filler (80%) and 15V pitch binder (20%), matrix overcoats of 20 to 50 wt-% pickup were applied to TRISO fertile particles by the revolving-drum technique. These represent matrix overcoats of approximately 50 to 250 μ m thick. The application of the 250- μ m overcoat required only 15 min compared to 60 min for some previous formulations.

When formed into compacts by the usual hot-pressing techniques, the best samples were obtained from the 250- μ m overcoated particles. The compact had an apparent fuel packing fraction of 53%, fairly good integrity, and only a few broken particles on the outer surface. Although this would indicate the feasibility of the process, further development and optimization is required.

A variation of the injection-molding process was investigated in an effort to introduce maximum loadings of graphite powder into particle beds. In this process, fine (about 1- μ m) graphite powder suspended in an acetone-dilute resin solution is injected into the particle bed under pressure. This pressure causes a filter "cake" of the graphite powder to build up in the channels between the fuel particles. When the particle bed is completely filled with matrix material, the remaining acetone is flushed out with air pressure. It is important that enough binder remain to hold the rod together for removal from the mold and subsequent curing and carbonization. From some of the initial samples prepared in this manner, it appears that at least 10% resin is necessary to hold the rod together through carbonization.

Fuel Rod Evaluation by Burning. As a test supplementary to evaluations by microscopic examination and metallography, a rod-burning technique was developed. The objective of rod burning is to determine the number of particles

Table 3.16

FUEL ROD DEVELOPMENT USING HOT-PRESSED COMPACTS

Rod No.	Compacting Pressure (psi)	Process	Matrix ^a Formulation	Fuel Packing Fraction (vol-%)	Matrix Density (g/cm ³)	Fuel Particles	Compact Length ^b (in.)	Remarks (Broken Particles Noted)
1	2500	Matrix admix	GP-38-30% 15V	47.5	1.52	HTDK 4624	1.043	18 on sides (compact end broken off) 8 broken inside
2	2000	Matrix admix	6353-30% 15V	50.5	1.62	HTDK 4624	0.978	4 side, 6 bottom edge
3	1500	Matrix admix	6353-30% 15V	48.5	1.56	HTDK 4624	1.020	9 on sides
4	2500	Matrix admix	6353-15% 15V	47.5	1.52	HTDK 4624	1.040	4 on side
5	1500	Matrix admix	6353-30% 15V	50.4	1.60	HTDK 4624	0.983	2 broke 1/4 in. from bottom; one 1/4 in. from top; same side
6	1000	Matrix admix	6353-30% 15V	50.0	1.56	4000-201	0.988	About 12 broken bottom edge; 2 on side; overnight breaking apart all over
7	1000	Matrix admix	328-30% 15V	50.0	1.46	4000-201	0.978	Insufficient matrix voids; many broken particles all over
8	1000	PyC buffer overcoat	6353-30% 15V	48.8	1.55	4000-149 buffer 4483-145D OC	1.014	No signs of breaking
9	1000	PyC buffer overcoat	6353-30% 15V	49.0	1.54	4000-201 4493-3 46- μ m buffer OC	1.007	Starting to show signs of breakage; broke up completely
10	1000	PyC buffer overcoat	6353-30% 15V	50.0	1.66	4000-201 50% 4000-149 50%	0.987	Looks good; overnight about 12 broken on sides
110C	1000	Matrix overcoat	6353-20% 15V	---	---	4000-201 20 wt-% OC	---	90% broken particles
120C	1000	Matrix overcoat	6353-20% 15V	33	---	4000-201 26 wt-% OC	0.625	Held together but weak
130C	1000	Matrix overcoat	6353-20% 15V	53	---	4000-201 50 wt-% OC	0.685	Looks good; few broken around top and couple on side

^aPercent binder given; percent filler is the difference to 100%.^bCompact diameter = nominal 1 in. for all samples.

that are cracked through the SiC coating and to substantiate the result by calculating the weight fraction of broken particles.

The test is carried out by weighing the rod, burning down to the SiC coat on an Inconel tray at 700°C in an air-atmosphere muffle furnace, weighing the residue, counting the number of white (broken) particles under a microscope, screening the particles on 246- μ m mesh after tumbling (to separate the broken particle pieces from the whole ones), and weighing the residue. The results, which are listed in Table 3.17, show that (1) rod manufacture, curing, and carbonizing do not break particles and (2) graphitizing for 1 hr at 2000°C does break a small fraction of fertile particles. The results in Table 3.18 show that 2000°C heat treatment causes some breakage of loose particles and that broken particles in the graphitized rods are not necessarily the result of matrix volume change.

The heat treatment at 2000°C for 1 hr drives off some impurities from rods, and this is easily observable. A graphitized rod, after burning, leaves a residue of clean black particles, but a nongraphitized rod leaves a residue of particles coated with a fine brown dust, probably ferric oxide.

Evaluation of extruded rods has also been carried out as for injected fuel rods. The results are shown in Table 3.19. It can be seen that the fraction of broken particles is virtually the same as in the injected fuel rods both when carbonized and graphitized. This is not surprising in view of the evidence that it is the fertile particles that are broken by the 2000°C heat treatment.

Alternative Fuel Rod Fabrication Processes. A number of different approaches to rod fabrication, including some modifications of the injection-molding process, have been investigated. These included the infiltration technique, addition of chopped graphite fiber, incorporation of graphite cloth, and incorporation of graphite foil. Since the most promising of these appeared to be extrusion, the main effort was subsequently directed toward this process.

Fuel rods with high fuel particle loadings were successfully extruded with no broken particles. The formulations included rods with both ACN and pitch binders, ethyl cellulose prebinder, and natural flake filler. Attempts to fabricate rods with two different types of isotropic graphite filler were unsuccessful, however. After carbonization, the pitch rods have a fair amount of strength, but the ACN rods are relatively weak and require a reimpregnation and another carbonization to attain good handling strength.

Sorption of Strontium on Charcoal-type Materials. Mass-spectrometric experiments have been performed to determine the strontium sorptive capacity (relative strontium loadings at equivalent pressure and temperature) of several carbons in order to evaluate their potential for use in blended fuel beds. The samples, after an initial high-temperature pretreatment, were loaded with SrC₂. They were treated in a sealed tantalum container at 1620°K for about 3 hr to distribute the strontium uniformly throughout the sample as well as to remove oxygen and then quickly put into the mass spectrometer. Some oxygen contamination remained, but, by comparison with previous experiments with the Sr-O-graphite system, this should have had, at worst, only a small influence on the strontium pressures.

Table 3.17
EVALUATION OF INJECTION-MOLDED FUEL RODS BY BURNING

Rod No.	Rod Formulation	Status	Weight Loss on Burning (wt-%)	-246 μ m Mesh Ash (wt-%)	Number of Broken Particles per Gram
A-146 } A-147 }	70% ACN (1/2% MEKO) 30% P-6353	Carbonized	30.7	0.08	2.5
		Graphitized	31.1	0.23	4.5
A-153 } A-154 }	70% ACN (1% MEKO) 30% P-6353	Carbonized	30.6	0.11	3.6
		Graphitized	30.8	0.21	4.5
A-171 } A-172 }	70% ACN (1/2% BPO) 30% P-6353	Carbonized	30.5	0.10	2.5
		Graphitized	30.5	0.23	7.9
A-177 } A-178 }	70% ACN (1% BPO) 30% P-6353	Carbonized	30.3	0.10	1.6
		Graphitized	30.2	0.14	3.2
A-207 } A-208 }	50% ACN 50% H-378	Carbonized	32.7	0.08	0.7
		Graphitized	33.4	0.41	5.2
A-235 } A-236 }	75% No. 30 pitch 25% P-6353	Carbonized	30.9	0.13	0.7
		Graphitized	30.0	1.0	7.0
A-237 } A-242 }		Injected & cooled	38.0	0.11	0.9
		Injected & cooled	38.6	0.10	0.9
A-243 } A-244 }	70% 15V pitch 30% P-6353	Graphitized	31.2	0.54	0.5
		Carbonized	45.6	0.04	0
C-104 } C-105 }	65% ACN 35% P-6353	Carbonized	34.7	0.04	1.0
		Graphitized	30.0	---	0
C-110 } C-111 }	75% No. 30 pitch 25% P-6353	Carbonized	34.3	0.05	0
		Graphitized	34.2	---	4.0

Table 3.18
EVALUATION OF LOOSE PARTICLES

Batch No.	Status	Weight Loss on Burning (wt-%)	-246 μ m Mesh Ash (wt-%)	No. of Broken Particles per Gram
4000-147	Fissile particles, as received	25.2	ND ^a	0.9
4000-151	Fertile particles	22.1	0.04	1.0
4000-149	Fissile particles heat treated at 2000°C	25.1	ND ^a	0.5
4000-151	Fertile particles heat treated at 2000°C	18.2	0.48	8.4

^aNot determined.

Table 3.19
EVALUATION OF EXTRUDED FUEL RODS BY BURNING

Rod No.	Status	Weight Loss on Burning (wt-%)	-200 μ m Mesh Ash (wt-%)	No. of Broken Particles per Gram
5.49.P52	Carbonized	24.6	0.13	1.8
5B.P52	Carbonized	28.0	0.16	~1.8
6.P52	Carbonized	26.5	0.13	~1.8
1.P52	Carbonized	26.8	0.11	~1.8
59#1.P60	Graphitized	26.7	0.21	~6.2
57A#4.P63	Carbonized	32.6	0.08	~1.6
57A#1.P60	Graphitized	30.2	0.19	3.2
#59#1.P60	Graphitized	24.9	0.24	8.3
59#5.P60	Carbonized	25.4	0.09	1.7
57A#2.P60	Graphitized	32.0	0.17	3.8

The samples were compared according to their 1500°K isotherms as shown in Fig. 3.28. The concentration range was about 4 to 40 mg Sr/g C. The materials studied, in the form of 62- to 125- μ m grains, were:

1. MI-6736 activated charcoal; 1620°K vacuum pretreatment for 3 hr (800 ppm sulfur after pretreatment).
2. Furfuryl alcohol coke, 1620°K vacuum pretreatment for 3 hr (<2 ppm sulfur after pretreatment).
3. Calcined petroleum coke; 1775°K vacuum pretreatment for 1 hr (0.9% sulfur after pretreatment).
4. Calcined petroleum coke, 2275°K in helium flow for 1 hr (0.4% sulfur after pretreatment).
5. Calcined petroleum coke, 1775°K fluidized bed for 10 hr (0.9% sulfur after pretreatment).

The isotherms for samples 1 and 2 are approximately parallel to the line in Fig. 3.28 obtained previously for 37- to 74- μ m grain H-327 graphite but are displaced to higher strontium loadings by factors of 40 and 60, respectively. Applying a factor of four (to account for the difference in sorptive capacity between 37- to 74- μ m grain graphite and bulk graphite), it is deduced that the sorptivities of the respective carbon materials are 160 and 240 times greater than for bulk graphite over the range investigated.

The isotherms for the three petroleum coke samples are all roughly parallel to each other but show a stronger concentration dependence in comparison with samples 1 and 2. The two coke samples that were pretreated to 1775°K have identical isotherms and the strontium capacity is as much as a factor of ten greater than for samples 1 and 2. Although sample 4 shows a capacity similar to samples 1 and 2 in the range of investigation, its capacity is greater for the lower part of the range investigated where conditions are more similar to the concentrations of interest for the reactor. The greatly enhanced sorptivity of the petroleum cokes seems to correlate with the sulfur concentrations, which are quite large. This is not surprising, since from thermodynamic considerations, sulfur should be still more effective in stabilizing strontium on the carbon surface than was the case with oxygen. The reduced capacity of sample 4 was probably caused by a reduction in sulfur content resulting from the pretreatment at 2275°K.

Impurity Content of Fuel Element Materials. Work was performed to measure the content of hydrogen and other impurities in charcoal-type material, core graphite, and TRISO fuel particles, all of which are candidates for use in the reactor core.

Because of its effect on nuclear reactivity, the initial content of hydrogen in the core must be known. Hydrogen in carbonaceous materials occurs both as sorbed water and as chemically bound hydrogen. Analyses of both forms of hydrogen have been performed on candidate as-received activated carbons, H-327 graphite, and fuel particles. In addition, analyses were performed on samples annealed at temperatures up to 2000°C.

Fig. 3.28--Strontium sorption isotherms at 1500°K for several carbons and H-327 graphite

Other impurities that may be of particular concern are iron, nickel, and sulfur. A high level of iron and nickel impurities in the fuel bed is deemed undesirable because of the possibility that these impurities may cause particle coating degradation. High sulfur-bearing materials are undesirable because of the possibility that sulfur-containing gases in the main coolant may cause corrosion of metal surfaces.

Emission spectrometric determinations and sulfur analyses were performed on coke and activated carbon materials before and after annealing in static or fluidized beds; the results are given in Table 3.20. It was found that the iron and nickel impurities could be greatly reduced by heating the material in a helium atmosphere to 1500° or 2000°C. Sulfur content was also reduced but to a lesser extent.

A preliminary screening of candidate materials was performed for hydrogen content and water uptake; the results are given in Table 3.21. Furfuryl alcohol coke and MI-6736 activated carbon were selected as candidates for further testing. Samples of these two materials and a commercial calcined petroleum coke (CPC) were outgassed at four different temperatures (1000°, 1500°, 1750°, and 2000°C) and were allowed to equilibrate in a moist atmosphere. The results (see Table 3.22) show that water uptake is reduced considerably when the materials are heated above 1000°C with further reduction at the higher temperatures. The hydrogen content of each carbon material was lower (see Table 3.22) by about an order of magnitude after 1 hr at 1500°C.

TRISO fuel particles and H-327 graphite pieces were similarly tested and, as shown in Tables 3.23, 3.24, and 3.25, were found to pick up negligible amounts of water at 62% humidity.

On the basis of these results and assuming that the fuel-loading operations will be performed at a relative humidity of 62% or less, the hydrogen in the core carbonaceous materials will consist largely of chemically bound hydrogen; the total hydrogen content of the core graphite, coke filler, and fuel particles at startup is estimated to be about 17 lb.

Graphite Development

The solid graphite matrix of Fort St. Vrain fuel elements is machined from logs of extruded needle coke grade H-327 material approximately 18 in. in diameter and 34 in. long. Over 60% of the material may be removed in manufacturing the fuel block configuration. Large extrusions of graphite have variations in mechanical strength throughout their bulk, and the acceptable strength criterion for the fuel element material is a minimum value of 1000 psi in tension in samples machined from the centers of logs in the direction parallel to extrusion. Each sample is removed from the fuel element handling tool access hole position, which does not permit a sample of sufficient size to test the material in the weaker transverse plane of the log. This mechanical strength criterion, together with controls on chemical purity and bulk density, is a condition of acceptance of the log of graphite from the manufacturer.

The general arrangement of a total of 324 closed-ended, 0.5-in.-diameter fuel holes and continuous 0.625-in.-diameter coolant holes in a fuel element

Table 3.20
IMPURITY CONTENT OF CARBON MATERIALS

Material	Condition	Time (hr)	Impurities (ppm)		
			S	Ni	Fe
CPC	As received	--	9300	1000	1000
CPC	1500°C, static bed ^a	1	--	200	200
CPC	1750°C, static bed	1	--	200	100
CPC	2000°C, static bed	1	4100	--	60
CPC	1500°C, static bed	14	--	600	400
CPC	2000°C, fluidized bed ^b	1	2300	80	40
CPC	1500°C, fluidized bed	10	8900	200	80
CPC	2000°C, static bed	1	--	100	400
Furf. alc. coke	1500°C, static bed	1	--	--	<1
Phenolic coke	1500°C, static bed	1	--	--	10
Furf. alc. BC589	1500°C, static bed	1	--	8	160
Phenolic BC589	1500°C, static bed	1	--	8	160
MI-6736 carbon	1500°C, static bed	1	--	--	40
P-110 carbon	1500°C, static bed	1	--	--	20
MI-6736 carbon	As received	--	800	--	--
Furf. alc. coke	As received	--	<2	--	--
Ion Ex. coke	As received	--	9300	4	10
H-327 graphite	As received	--	<1	<10	<10
PGX graphite ^c	As received	--	360	--	--
ATJ graphite ^c	As received	--	13	--	--

^aSamples heated in closed graphite crucible.

^bCoke particles heated in bed fluidized with helium.

^cCandidates for core supports and side reflectors.

Table 3.21
RESULTS OF PRELIMINARY SCREENING TESTS

Material	Wt. Loss to 1000°C (%)	Wt. Loss to 1500°C (%)	Wt. Gain After Outgassing at 1000°C and 3 Days at 100% RH ^a	Wt. Gain After Outgassing at 1500°C and Exposure to 100% RH ^a		Hydrogen Content (as H) As-received (ppm)	Hydrogen Content (as H) After 3 hr at 1500°C (ppm)
				(hr) ^b	(%)		
Furfuryl alcohol coke	3.9	5.5	11.45	22	0.13	4300	110
Phenolic coke	4.0	5.4	10.1	86	0.95	5200	140
BC589 furfuryl alcohol	4.4	10.4 ^c	15.1	139	6.8	3000	150
BC589 phenolic	6.0	11.0	16.7	86	6.5	3800	91 ^d
MI-6736 activated carbon	11.7	15.9	25.6	22	28.3	4700	630
P-110 activated carbon	19.7	24.7	29.3	22	24	21,000	470

^aRH = relative humidity.

^bTime of exposure to 100% RH.

^cToo high because piece of crucible missing.

^dA low value because some of sample lost.

Table 3.22

EFFECT OF TEMPERATURE ON WATER-SORPTION QUALITIES AND HYDROGEN
CONTENT OF CARBON MATERIALS

Materials	Outgassing Temp (°C)	Loss After 1 hr at Temp (wt-%)	Gain at RT and 100% RH ^b (wt-%)		Gain at 85°F and 62% RH ^b (wt-%)		Hydrogen ^c (as H) (ppm)
			114 hr	165 hr	116 hr	186 hr	
Furfuryl alcohol coke	1000	6.4	14.8	15.2	5.4	13.8	NM
	1500	6.9	0.10	0.41	NM	NM	110
	1750	7.2	0.25	0.55	NM	NM	NM
	2000	7.1	0	0.24	0	0	NM
CPC petroleum coke	1000	0.57	0.37	0.59	0	0	NM
	1500	1.50	0.13	0.39	NM	NM	81
	1750	2.94	0.10	0.07	NM	NM	NM
	2000	3.61	0	0.07	0	0	NM
MI-6736 activated carbon	1000	12.82	62.1	62.3	5.0	25.4	NM
	1500	14.44	49.4	49.8	NM	NM	630
	1750	16.01	41.0	42.1	NM	NM	NM
	2000	18.18	20.6	21.6	0	0.5	NM

^aAs-received contents of chemically bound hydrogen determined after water removal were 4300 ppm for furfuryl alcohol coke, 1040 ppm for CPC petroleum coke, and 4700 ppm for MI-6736 activated carbon.

^bRT = room temperature; RH = relative humidity; NM = not measured.

^cChemically bound hydrogen after 1 hr at 1500°C.

Table 3.23

WEIGHT LOSS, WATER UPTAKE, AND HYDROGEN CONTENT OF H-327 GRAPHITE SAMPLES

Test No.	Loss After 20 hr at 250°C (wt-%)	Loss After 3 hr at 1000°C (wt-%)	Loss After 16 hr at 300°C (wt-%)	Gain After 72-95 hr at 100% RH and RT ^a (wt-%)	Gain After 7-20 Days at 62% RH and 85°F ^a (wt-%)	Hydrogen Content (as H) ^b (ppm)
1 ^c	---	0.01	---	0.004	---	10
2 ^d	0.01	0.025	---	0.019	Neg	--
3 ^e	0.015	0.035	---	0.022	Neg	--
4 ^f	---	---	0.012	---	Neg	--

^aRH = relative humidity; RT = room temperature; Neg = negligible.

^bContent of chemically bound hydrogen. Average of analyses performed by the vacuum-fusion technique on two different as-received samples.

^cAverage of several samples taken from unknown positions in graphite log.

^dAverage of four samples taken from center of graphite log.

^eAverage of two samples taken from outer edge of graphite log.

^fAverage of three samples taken from unknown position in graphite log.

Table 3.24
LOSS AND GAIN OF WATER FOR TRISO COATED
FERTILE FUEL PARTICLES

	Weight Change (wt-%)
Lot 4483-950:	
1. Loss after 5 hr at 1000°C	0.075
2. Loss after 3 hr at 1500°C	0.089 ^a
3. Gain after steps 1, 2, and 3-day exp to 100% RH at RT	0.081
Lot 4000-194:	
1. Loss after 1 hr at 1000°C	0.071
2. Gain after step 1 and 165-hr exp to 100% RH at RT	0.024
3. Gain after step 1 and 40-day exp to 62% RH at 85°F	Neg

NOTE: RH = relative humidity; RT = room temperature;
Neg = negligible.

^a-Includes weight loss that occurred after 5 hr at 1000°C.

Table 3.25
HYDROGEN CONTENT OF TRISO COATED FERTILE
FUEL PARTICLE SAMPLES

Lot No.	Hydrogen ^a Content (as H) ^b (ppm)	Hydrogen ^b in Outer Coating (as H) (ppm)	Hydrogen in Remainder of Particle (as H) (ppm)
4483-950	180	---	--
4000-193	---	292	64
4000-194	---	523	46

NOTE: Samples were first outgassed in vacuum at 150°C to remove water.

^aHydrogen determined by combustion of crushed particles and analysis of evolved gases.

^bHydrogen in outer coating, which comprised about 16% of the particle weight, determined by combustion and analysis of evolved gases. Remaining SiC-coated kernels were then crushed and combusted.

block may be described as a basic array consisting of one fuel hole surrounded by six equidistant alternate fuel and coolant holes about a 0.740-in.-diameter circle. The ligament between fuel and coolant holes is a web of graphite 0.177 in. thick.

In service, the fuel block graphite acts as a moderator, structural member, and container for the blended beds of fuel particles. It is important, therefore, that no large discontinuities or structural weaknesses exist in the graphite block after machining. Current test work on H-327 graphite is designed to study and improve nondestructive testing procedures where necessary and to maintain a continuous survey of the physical, chemical, and mechanical properties to verify that no unscheduled changes occur in the product.

Mechanical Strength. Currently, approximately half the total quantity of graphite logs required for the core have been produced and accepted for machining. The mean tensile strength of the sample taken from the fuel handling hole of each of the logs is 1640 psi with a standard deviation of 429 psi.

During the course of the batch manufacture of grade H-327 graphite logs five logs, each selected at a different period in the production schedule, have been given more extensive tensile tests. It has been previously established that the largest differences in mechanical strength of positions in the extruded logs were in the extreme outer and inner fibers and at the end and center length positions (see earlier quarterly report GA-9440). Table 3.26 lists the mechanical properties of test samples machined from these positions in a log selected at the mid-production stage of manufacture. Also given is a comparison of the mean test values for individual logs and cumulative values for tests on three previous logs. These values show that at the mid-production stage the graphite product shows increase in ultimate tensile strength (UTS) of up to 10% at the weakest position in a log and that the elastic modulus is slightly lower. In summary:

1. The variation in mean UTS within a log of fuel element graphite is 874 to 2498 psi.
2. The weakest position in a fuel block is at the inner fibers at the center of the 34-in. extruded length in the transverse-to-extrusion direction.
3. The strongest position in a fuel block is at the outer fibers at the center of the 34-in. extruded length in the parallel-to-extrusion direction.
4. The chord modulus of elasticity of the graphite at stresses between 250 and 500 psi ranges between 1.28 and 1.91×10^6 psi in the parallel direction and 0.54 to 0.707×10^6 psi in the transverse orientation.

Homogeneity of Fuel Block Graphite. Hitherto, the machined surface appearance of H-327 graphite has been used as a guide to its homogeneity. The method gives a practical appraisal of the results of machining the material, since voids are produced through the tearing out of particles during machining. Such voids are mostly hemispherical in shape with major axes ranging between 40 and 60 mils. To assess the extent of internal voidage, a whole log was sectioned into 1-1/2-in. slabs (see Fig. 3.29) and radiographed to a sensitivity of 1%. A visual interpretation of the maximum voids (over and above the general

Table 3.26

TENSILE TEST DATA ON MID-PRODUCTION LOG NO. 42/2034 85/0069 9.2

Property	Sample Position							
	Parallel Orientation				Transverse Orientation			
	End Edge	Mid- Length Edge	Mid- Length Center	End Center	End Edge	Mid- Length Edge	Mid- Length Center	End Center
UTS, psi:								
Max.	2566	2667	1920	2290	1678	1359	986	1760
Min.	2214	2566	1810	2097	1600	1344	960	1580
Mean	2390	2616	1865	2193	1639	1351	973	1670
Cum. mean	2394	2498	1517	1781	1383	1180	874	1407
Current std deviation	256	158	350	384	179	170	82	182
Strain at Frac., 10^{-3} in./in.:								
Max.	2.35	2.98	1.73	2.34	3.70	3.22	2.23	3.94
Min.	2.31	2.46	1.52	2.01	3.0	2.53	1.70	3.71
Mean	2.33	2.68	1.62	2.17	3.35	2.87	1.96	3.82
Cum. mean	2.12	2.20	1.47	1.69	2.67	2.32	1.78	2.9
Current std deviation	0.50	0.55	0.498	0.457	0.75	0.415	0.86	0.806
Secant Modulus, 10^6 psi:								
Max.	1.22	1.1	1.22	0.99	0.55	0.536	0.565	0.446
Min.	0.98	0.9	1.13	0.425	0.43	0.42	0.447	0.426
Mean	1.10	1.0	1.17	0.707	0.46	0.478	0.506	0.436
Cum. mean	1.35	1.23	1.32	0.968	0.56	0.618	0.496	0.476
Current std deviation	0.226	0.205	0.31	0.197	0.171	0.155	0.085	0.197
Chord Modulus between 250 and 500 psi, 10^6 psi:								
Max.	1.388	1.606	2.08	1.250	0.714	0.625	0.625	0.625
Min.	1.250	0.926	1.923	1.250	0.5	0.5	0.5	0.595
Mean	1.319	1.366	2.001	1.250	0.607	0.562	0.562	0.610
Cum. mean	1.57	1.698	1.91	1.28	0.707	0.669	0.543	0.627
Current std deviation	0.234	0.458	0.506	0.068	0.274	0.12	0.051	0.033

Fig. 3.29--Arrangement of slab sectioning for 100% radiography of fuel block graphite

Table 3.27
VOIDAGE DETECTED BY RADIOGRAPHY OF TYPICAL FUEL BLOCK
GRAPHITE SECTIONED IN 1-1/2-IN. SLABS

Slab No.	Largest Dim. Observed Discontinuity (in.)	Slab No.	Largest Dim. Observed Discontinuity (in.)
1	0.100	16	0.120
2	0.090	17	0.080
3	0.075	18	0.090
4	0.050	19	0.090
5	0.050	20	0.090
6	0.075	21	0.090
7	0.090	23	0.105
8	0.080	24	0.110
10	0.075	25	0.140
11	0.075	26	0.100
12	0.100	27	0.080
13	0.080	28	0.100
14	0.075	29	0.100
15	0.110		

level of fine dispersed voids which give a typical density of 1.75 to 1.85 g/cm³) is listed in Table 3.27. The sizes of the biggest discontinuity in each slab ranged between 50 and 140 mils, and all but one slab contained defects in this size range.

The largest void was shown to be ellipsoidal in shape, and when radiographed from two positions at 90° orientation (see Fig. 3.30), was shown to have a major axis of 140 mils.

Each slab showed superficial voids of the type previously described (see Fig. 3.31), with a network of discontinuities becoming more pronounced in the center of the extrusion. These discontinuities, or flow-line effects, were macroscopic in scale and showed on the radiographs. There are no unusual differences in the microstructure of the log at the inner and outer fiber positions (see Figs. 3.32 and 3.33). The voids in the structure are laminates generally of the order of 10⁻³ cm in major dimensions in the outer fibers of the log with fewer and larger laminate type voids in the center. This latter condition possibly explains the reduced strength of the production grade graphites in the center of the log, which have differences in measured bulk densities (see Table 3.28) of 2% across the section.

In summarizing the size of voids in a typical PSC fuel element graphite, it should be noted that manufacturing quality control procedures are arranged to detect voids that breach the web between any two holes. Thus, the major

Fig. 3.30--Prints from two radiographs taken at 90° from each other showing maximum-sized void observed in single-shot radiographs of 1-1/2-in.-thick sections of graphite log

HT-7717 5

Fig. 3.31--Part of the cross section of a H-327 graphite log showing typical size and distribution of voids. The discontinuities are highlighted by acetone staining and are concentrated toward the center of the log (upper center of photograph)

M-31726-1

(75X)

M-31728-1

(75X)

Fig. 3.32--Structure of H-327 graphite log produced at mid-stage of production. Samples transverse to extrusion at mid-length of log; upper sample from center of log, lower from edge

M-31727-1

(75X)

M-31729-1

(75X)

Fig. 3.33--Structure of H-327 graphite log produced at mid-stage of production. Samples parallel to extrusion at mid-length of log; upper sample from center of log, lower from edge

Table 3.28

BULK DENSITY VARIATIONS IN PRODUCTION-GRADE GRAPHITE LOG

Sample Location and No.	Diam. (in.)	Length (in.)	Density (g/cm ³)
End edge:			
1	0.502	3.007	1.79
2	0.502	3.003	1.80
3	0.502	3.006	1.80
End center:			
1	0.052	2.996	1.78
2	0.502	3.007	1.79
3	0.502	3.003	1.79
Mid-length edge:			
1	0.502	2.997	1.81
2	0.502	3.004	1.80
3	0.502	3.002	1.80
Mid-length center:			
1	0.502	3.006	1.77
2	0.502	2.998	1.77
3	0.502	3.001	1.77

dimension of the smallest hole detectable would be approximately 170 mils. The distribution and size of voids in a typical fuel block graphite are such that 95% of any 1-1/2-in.-thick section would contain a void with a dimension of at least 50 mils, 25% of the slabs may contain voids with a maximum size of 75 mils, and 14% larger than 100 mils. No internal void sufficient to breach the web between a fuel and coolant hole was observed.

Permeability of Production-grade H-327 Graphite. From a series of helium flow measurements using graphite samples machined from various positions within logs of production grade H-327 graphite, Creutz (Refs. 2-4) has tabulated the derived values of average pore size and effective number of pores (see Tables 3.29 through 3.33). The flow rate per unit area over a given length of graphite varies by a factor of two within samples in any one log and by a factor of four between samples from different logs. The permeability at a pressure of 1 atm is relatively closely matched in samples within a log, but there is a spread of values between logs of a factor of three. The calculated flow constants for the production grade materials (see Tables 3.29 through 3.32) cover a smaller spread of slightly lower values than the preproduction material (see Table 3.33, Ref. 5), upon which most of the fission-product transport experiments were conducted. From the foregoing, it is concluded that no significant deleterious changes in permeability are present in fuel block samples of graphites representative of manufacture up to the mid-production stage.

Table 3.29

H-327 GRAPHITE PERMEABILITY, HELIUM FLOW CONSTANTS FOR PRODUCTION
LOG 2, LOG NO. 83/9976-C 42/2163 3-1

	End ^a		Mid-length ^a	
	Sample 69-1, Parallel	Sample 69-2, Transv.	Sample 69-3, Parallel	Sample 69-4, Transv.
Sample radius squared R^2 (cm ²)	15.7	15.7	15.8	14.9
Helium permeability at 1 atm, K_1 , cm ² /sec	3.4	1.24	4.2	1.11
$(p\Delta p)_0$, atm ²	130	150	97	160
F_0 atm, cm ³ /sec	2800	1300	1750	1100
Avg effective pore radius, r , cm	8.7×10^{-4}	7.1×10^{-4}	4.2×10^{-4}	6.8×10^{-4}
Effective No. of pores, n	1.1×10^5	1.05×10^5	1.7×10^6	1.10×10^5
Effective No. of pores per cm ² , $n/\pi r^2$	2.2×10^3	2.1×10^3	3.4×10^3	2.1×10^3

^a"End" and "mid-length" denote mid-radial position in log;
"parallel" and "transv." denote orientation with respect to extrusion
direction.

Table 3.30

H-327 GRAPHITE PERMEABILITY, HELIUM FLOW CONSTANTS FOR PRODUCTION
LOG 3, LOG NO. 83/9976 6-1

	End ^a		Mid-length ^a	
	Sample 6-1-1, Parallel	Sample 6-1-2, Transv.	Sample 6-1-3, Parallel	Sample 6-1-4, Transv.
Helium permeability at 1 atm, K_1 , cm^2/sec	4.9	2.1	2.6	2.7
$(p\Delta p)_o$, atm^2	103	315	100	180
F_o , atm, cm^3/sec	2900	2300	1550	2600
Avg effective pore radius, r , cm	4.2×10^{-4}	2.4×10^{-4}	1.36×10^{-3}	1.07×10^{-3}
Effective No. of pores, n	1.5×10^6	6.1×10^6	1.6×10^4	1.8×10^4
Effective No. of pores per cm^2 , $n/\pi r^2$	3.4×10^4	1.3×10^5	3.6×10^3	4.0×10^3

^a"End" and "mid-length" denote mid-radial position in log;
"parallel" and "transv." denote orientation with respect to extrusion
direction.

Table 3.31
H-327 GRAPHITE PERMEABILITY, HELIUM FLOW CONSTANTS FOR
MID-PRODUCTION LOG 42/2034

	End ^a		Mid-length ^a	
	Sample 4-4, Parallel	Sample 4-2, Transv.	Sample 4-3, Parallel	Sample 4-1, Transv.
Helium permeability at 1 atm, K_1 , cm^2/sec	1.09	0.64	1.39	0.77
$(p\Delta p)_0$, atm^2	140	185	100	190
F_0 , $\text{atm, cm}^3/\text{sec}$	1000	700	1000	900
Avg effective pore radius, r , cm	5.2×10^{-4}	2.8×10^{-4}	2.8×10^{-4}	4×10^{-4}
Effective No. of pores, n	3.2×10^5	1.7×10^6	3.1×10^6	5.5×10^5
Effective No. of pores per cm^2 , $n/\pi r^2$	6.0×10^3	3.2×10^4	5.8×10^4	1.0×10^4

^a"End" and "mid-length" denote mid-radial position in log;
"parallel" and "transv." denote orientation with respect to extrusion
direction.

Table 3.32

H-327 GRAPHITE PERMEABILITY, HELIUM FLOW CONSTANTS FOR MID-PRODUCTION
LOG 40/9977C 42/1960

	Sample No., Position, and Orientation ^a							
	12-1-1, End, Parallel	12-1-2, End, Parallel	12-1-3, End-center Parallel	12-1-4, Mid-length Edge, Parallel	12-1-5, Mid-length Center, Parallel	12-1-6, Mid-length Edge, Transv.	12-1-7, Mid-length Edge, Parallel	12-1-8, Mid-length Center, Parallel
Helium permeability at 1 atm, K_1 , cm^2/sec	0.76	1.07	1.29	1.04	0.94	0.97	1.40	1.77
$(p\Delta p)_0$, atm ²	150	150	150	160	155	150	145	285
$(p\Delta p)_0/\text{cm length}$	2.54	2.58	2.56	2.75	2.85	2.56	2.49	3.4
F_0 , atm, cm^3/sec	620	930	1010	1000	890	870	1020	2120
Avg effective pore radius, r , cm	3.3×10^{-4}	5.8×10^{-4}	7.9×10^{-4}	8.2×10^{-4}	1×10^{-3}	8.8×10^{-4}	1.8×10^{-3}	5×10^{-4}
Effective No. of pores, n	1.1×10^6	1.3×10^4	5.9×10^4	4.5×10^4	1.8×10^4	3.4×10^4	3.0×10^3	4.3×10^5
Effective No. of pores per cm^2 , $n/\pi r^2$	2.5×10^4	3.0×10^2	1.4×10^3	1.0×10^3	4.2×10^2	8.0×10^2	7.0×10^1	1.0×10^4

^aOrientation in block with respect to extrusion direction.

Table 3.33

H-327 GRAPHITE PERMEABILITY, HELIUM FLOW CONSTANTS FOR PILOT
PRODUCTION GRADE, LOT 862, LOG 114

	Sample No., Position, and Orientation ^a								
	1, End Edge, Parallel	2, End Center, Parallel	3, Middle Edge, Parallel	4, Middle Center, Parallel	5, End Edge Parallel	6, End Center, Parallel	7, Middle Edge, Transv.	8, Middle Center, Transv.	9, End Center, Transv.
Helium permeability at 1 atm, K_1 , cm^2/sec	0.28	0.71	0.72	0.78	0.88	0.66	0.31	3.6	0.28
$(p\Delta p)_0$, atm^2	320	260	56	180	51	175	155	30 ^b	370
F_0 , $\text{atm, cm}^3/\text{sec}$	620	1500	390	880	360	900	420	1100 ^b	920
Avg effective pore radius, r , cm	4.4×10^{-4}	1.0×10^{-3}	1.0×10^{-3}	4.8×10^{-4}	9.2×10^{-4}	5.5×10^{-4}	8.5×10^{-4}	1.0×10^{-3b}	5.0×10^{-4}
Effective No. of pores per cm^2 , $n/\pi r^2$	9×10^3	9.5×10^2	9.5×10^2	1.78×10^4	1.2×10^3	9.5×10^3	8.35×10^2	5.1×10^{3b}	5.7×10^3

^aOrientation with respect to extrusion direction.

^bQuestionable values; sample cracked.

Effect of Defects on Fuel Blocks. As a preliminary to a quantitative series of tests, cylindrical test specimens of H-327 graphite consisting of a single fuel and coolant hole array (see Fig. 3.34) were subjected to a proof load test. The test length of the specimens was 3 to 4-1/2 in., and the configuration was machined concentric about a central fuel hole position with the axis of the specimen parallel to the direction of graphite extrusion. The outside of the test section was machined to a diameter of 1.97 in. to expose the bores of each of the fuel and coolant holes pitched about the central hole. The specified maximum acceptable defect in the web between fuel and coolant holes is a 14-mil-diameter through hole. Since defects greater than 14 mils in diameter through the web of a block will be cause for rejection of the material, proof tests were conducted to establish the effects of acceptable through-holes not greater than this order of size. One, two, or three 15-mil holes were drilled through the web of the test configuration to penetrate into the central fuel hole in the thinnest section of the web. The assemblies were then loaded by a uniaxial tensile load applied to the longitudinal axis or subjected to hoop-stress-only loading by internally pressurizing a softened copper or rubber tube in the central hole.

In eight hoop burst tests, only one fracture propagated through a defect position. No increases or reductions in the burst pressures were apparent as a result of the presence or absence of the artificial defects. Burst-test pressures were 1000 to 2600 psi in the hoop-stress-only tests, and in the one particular instance where fracture in the web coincided with the defect-bearing web, the fracture path completely circumvented the defect. At the above burst pressures, the graphite body remained intact, and the fracture was local to the area of the applied load. Where additional graphite matrix is present, there is sufficient constraint to arrest the crack (see Fig. 3.35). Where a continued increase in radial pressure is applied, a second fuel hole web fracture occurs without any catastrophic breakup of the graphite body (see Figs. 3.36 and 3.37).

In a series of uniaxial tensile tests, the fuel block hole array as described above was aligned with the axis of the holes parallel to the axis of extrusion and in the direction of the applied load. Dumbbell-shaped ends to the test specimens were used to reinforce the cross section of the body cemented into the flexible tensile test fixture. One, two, or three defects in the form of 15-mil holes were drilled through the web and no fracture occurred at these positions. Prior failure at stresses of 625 to 835 psi and extensions of 1 to 1.25×10^{-3} in. occurred in all the samples at the point where the test length commenced. This highlights the fact that a sharp change of section normal to the principal load direction is a much more significant type of defect than cylindrical holes through the web.

Further evidence of the small significance of voids in the outer fibers of a thick-shelled graphite cylinder subjected to internal pressure is shown in Fig. 3.38 where failure occurred at an inner fiber stress in excess of 1200 psi. The resultant fracture, however, bypassed a defect that had reduced the wall thickness by 25% (see Fig. 3.39) and progressed by linking up those discontinuities and weaknesses in the plane of the specimen normal to the principal stress direction (see Fig. 3.40).

HT-77452

Fig. 3.34--End view of specimen showing fuel and coolant hole arrangement used in proof tests

M-31424-3

Fig. 3.35--View of fracture in web of fuel block array showing constraint of fracture by outer fibers (left of picture) of full body of graphite. The burst pressure was 2600 psi on softened copper tube in center hole

HT-77451

Fig. 3.36--Fracture in web between coolant hole and fuel hole at 1400 psi internal pressure (hoop stress only) at the inside of centrally located fuel hole

HT-77450

Fig. 3.37--Fracture in second coolant/fuel hole web after initial fracture (see Fig. 3.36) with accompanying fracture in heavier web section between fractured sections encompassing one third of circumference

Fig. 3.38--Fracture mode due to hoop stress only in 0.75-in. OD, 0.5-in. ID graphite tube (H-327) (axis parallel to log extrusion). Estimated failure stress in excess of 1200 psi (M-31717-3)

In conclusion, the fuel hole arrangement in a fuel block is a thick shell configuration; thus, when subjected to internal loading, defects on the face of the bore would be the most significant. Defects that breach the web between any two holes, but that are equal to or below the acceptable maximum diameter, do not precipitate failure under static loads. The fracture within a fuel hole remains local to the area of applied load when excessive hoop stresses are created at the inner fibers (i.e., at the ultimate stress, fracture of a coolant hole web occurs without total breakup of the material).

Control Materials

Experiments on the behavior of boronated graphite materials at conditions calculated to occur during the hypothetical loss-of-forced-circulation (LOFC) accident have continued (see earlier quarterly reports GA-9130 and GA-9440 for results of earlier work). The tests are being performed to study the transport of boron from control rod materials and to measure the degree of compaction or slumping of boronated compacts subjected to a compressive load.

The conditions chosen for these experiments are those calculated to occur in a small region at the center of the core during the LOFC accident. Temperatures up to 2980°C are predicted; and, due to melting of metal components and subsequent slumping of control rods, the central boronated compacts could be subjected to a compressive load of 16 psi.

Test samples were prepared from production boronated graphite compacts and from compacts prepared at Gulf General Atomic. In addition, tests were performed on boronated graphite spheres prepared at Gulf General Atomic (i.e., reserve shutdown material). Measurements of weight loss, boron loss, and structural integrity (i.e., slumping) were performed at 2950°C, at compressive loads of 16 psi, and for exposure times up to 120 hr.

The compact samples were right circular cylinders (1.6 cm diameter by 3 cm long) with the exception of sample 4705-14, which was 5 cm long. Each

M-31717-2

(50X)

Fig. 3.39--Structure through outer fiber section of tube (Fig. 3.39) showing relatively strong matrix devoid of fracture. Wall thickness reduced at defect position by 25%

M-31717-1

(75X)

Fig. 3.40--Section through wall of tube (Fig. 3.39) showing mode of fracture by linkup of discontinuities normal to principal stress direction

sample rested on the bottom of a surrounding graphite crucible and was held at the top by a weighted graphite ram. There was a radial clearance of about 0.5 cm between each sample and the inside wall of the graphite crucible.

Two types of spheres were tested in one experiment. Four spheres, two of each type were stacked vertically, the two types being separated by a graphite wafer. The sphere column was subjected to a 16-psi compressive load (based on the area of the equatorial plane of the spheres).

The experiments were conducted in an induction furnace, the temperatures being measured with an optical pyrometer that has been calibrated to 2300°C with a standard lamp. Temperature calibration runs utilizing the thermal arrest of melting ZrC were performed. Definite indications of melting occurred at an apparent temperature of 2845°C, and resolidification was observed at 2870°C. The published melting points for ZrC range from 2800° to 2911°C, with several investigators agreeing on 2850°C. It is concluded that temperatures measured during these tests are accurate to within $\pm 50^\circ\text{C}$.

The experimental results of all of the tests are shown in Table 3.34, including data on slumping behavior, weight loss, and boron loss. In addition, estimates of the diffusion coefficient for boron loss from the compact samples are included as well as the apparent boron/carbon ratio in the effusing species.

All of the test samples slumped relatively little (see Col. 8, Table 3.34) as a result of the combined conditions of compressive load and high temperature (with the exception of sample 4705-48 which compressed about 30%). It is of interest that in all of the tests, sample compression was essentially complete after the first 5 or 6 hr of the run. This was true even for the 120-hr test in which the sample (No. 4705-14) slumped 8% during the first 5 hr and an additional 4% during the next 45 hr. The sample then showed no further sample compression throughout the remainder of the test period.

Boron losses are shown in Col. 7 of Table 3.34. These results are also shown in Fig. 3.41, where log boron loss is plotted versus log time. The data roughly fit a square-root-of-time relationship, which indicates a diffusion-controlled mechanism.

At the conclusion of the 120-hr test, all parts of the furnace internal components were assayed to obtain a boron material balance and to ascertain where the major portion of the effusing boron was collected. The data given in Table 3.35 indicate that significant boron redistribution can occur in a graphite system where an efficient sink is provided (i.e., graphite at lower temperatures).

Boron transport in laboratory-scale tests such as these is expected to be much higher than in the reactor core. This is because the increase in the boron content in the core graphite surrounding the control material is not duplicated in the laboratory.* Moreover, the diffusion path through the

*Boron in the core graphite would cause a back-pressure effect. This effect would not occur in the laboratory because cold regions in the furnace would act as sinks for the boron and would tend to limit the increase in boron content of the graphite surrounding the sample region.

Table 3.34

BEHAVIOR OF BORONATED GRAPHITE AT 2950°C AT A COMPRESSIVE LOAD OF 15 PSI

Sample No. ^a	Density (g/cm ³)	Initial Boron Content (%)	Anneal Time (hr)	Total Weight Loss (%)	Final Boron Content (%)	Total Boron Loss (%)	Sample Compression (%)	B/C ^b	D _{eff} ^c (10 ⁻⁷ cm ² /sec)
Production Compacts:									
4705-46	1.7	28.6	1.5	7.1	27.0	12.1	6.7	0.9	3.3
4705-38	1.7	28.6	4	8.6	23.1	26.7	3.9	8.0	6.0
4705-40	1.7	28.6	10	34.4	13.8	68.8	5.4	1.6	16.7
4705-29	1.7	28.6	12.5	29.6	17.3	57.6	9.4	1.5	10.8
4705-34	1.7	28.6	27	25.5	21.0	45.6	4.0	1.2	2.6
4705-42	1.7	28.6	38	35.6	10.2	80.2	6.8	1.2	5.9
4705-14	1.7	28.6	120	38.2	0.16	99.7	12	3.5	---
GGA Compacts:									
4705-48	1.76	30.7	10	34.1	20.0	57.5	29.5	1.0	11
4705-58	1.59	20.9	10	9.0	17.8	25.1	2.5	1.3	2.2
4705-62	1.68	20.3	10	8.8	16.9	23.8	1.3	1.5	2.0
4705-64	1.78	21.5	10	9.5	15.0	36.9	1.9	6.0	4.8
GGA 1/2-in.-diam. spheres: ^d									
4705-56L	1.49	20.7	10	--	12.9	44.5	3.7	---	3.7
4705-56H	2.14	18.5	10	--	12.2	33.6	2.1	---	2.1

^aAll compact samples were right circular cylinders (1.6 cm diam. by 3 cm long) with the exception of 4705-14, which was 5 cm long.

^bAtom ratio (boron/carbon) in transported species.

^cD_{eff} calculated from $F = A/L \sqrt{D_{eff}t}$ (see text).

^dTwo of each type of sphere were used in the same experiment (see text).

Fig. 3.41--Boron loss from boronated graphite at 2950°C

graphite is much greater in the core than in the experiment. Accordingly, an important objective of these experiments is to use the boron loss data to determine diffusion coefficients from which boron distribution in the core can be calculated.

Effective diffusion coefficients for boron losses from the test specimens were calculated using the relationships:

$$F = \sqrt{Dt} \, 2.256/L \text{ for cylindrical compacts, and}$$

$$F = \sqrt{Dt} \, 3.385/L \text{ for spherical specimens,}$$

where F = fraction boron lost,

D = effective diffusion coefficient, cm^2/sec ,

t = time, sec, and

L = radius of sphere or cylinder, cm.

These relationships, which are not exact because geometry factors are not precisely taken into account, yield diffusion coefficients that should be of sufficient accuracy for reactor calculations.

The diffusion coefficients thus calculated are listed in Col. 10 of Table 3.34. They range from 2×10^{-7} to $1.7 \times 10^{-6} \text{ cm}^2/\text{sec}$. The diffusion coefficient for boron in graphite at 2950°C used in safety analyses is 3×10^{-6} . This value is greater than the above experimental values, indicating that the boron redistribution calculations for the LOFC accident based on diffusion through graphite are conservative.

Table 3.35
DISTRIBUTION OF BORON IN FURNACE FOLLOWING 120-HR ANNEAL
OF SAMPLE 4705-14 IT 2950°C

Specimen	Specimen Weight (g)	Boron Content (%)
Compact sample	0.0225	0.38
Graphite ram, above sample	0.0585	0.99
Compact crucible, around sample	0.194	3.28
Crucible holder	0.266	4.51
Graphite susceptor:		
Top cap	0.156	2.65
Middle	0.388	6.56
Bottom cap	0.087	1.47
Lampblack, 3 cm around susceptor	3.363	56.93
Remaining lampblack	1.372	23.22

In all of these high-temperature anneals, total sample weight losses invariably exceeded the amount of boron transported. Moreover, it was observed that nonboronated graphite control samples, heated under similar conditions, lost relatively little weight. This suggests that the vapor above boron carbide in graphite at high temperatures is composed of species containing both boron and carbon. Column 9 of Table 3.34 gives the apparent atom ratio (B/C) for the transported species. It is seen that the B/C ratio ranges from 0.9 to 3.5. (The B/C values of 8.0 and 6.0 obtained for samples 4705-38 and 4705-64, respectively, are believed to be anomalous.) This phenomenon is consistent with experimental data of other workers where the species B, BC, B₂C, and BC₂ have been observed to exist in the vapor phase at high temperatures.

Tests planned for the future include measurements of boron diffusion coefficients in H-327 graphite and boron vapor phase transport in a mocked-up control rod system.

Irradiation Testing

Irradiation experiments are being conducted to proof test TRISO and BISO coated particles, blended beds, and fuel rods under Fort St. Vrain design irradiation exposures. These experiments are intended to confirm the selection of the reference fuel designs and to demonstrate the performance of these materials under service conditions. To date a total of 96 TRISO samples and 129 BISO samples have demonstrated successful irradiation performance, as shown in Table 3.36.

During this report period, two full-exposure tests (P20 and P22) of TRISO coated particles completed irradiation, and postirradiation examination of one (P20) was completed. Five proof tests of TRISO particles, blended beds, and fuel rods to half (F-25, F-27, and F-28) and full exposure (F-26 and F-29) began irradiation. One test of a full-size blended bed (F-30) is being designed.

Capsule P20. Capsule P20 is the highest exposure test of TRISO coated particles ever conducted with significant burnup. The particles were irradiated to fast fluences up to 8.7×10^{21} n/cm² and to burnups up to 27% FIMA, both of which exceed the reactor maximums of 8×10^{21} n/cm² and 20% FIMA, respectively.

The capsule contained twelve different batches of TRISO particles. Each was tested at the maximum temperature (1300°C programmed down to 1100°C) and three were also irradiated at the median temperature (approximately 900°C), making a total of 15 tests.

The particles were designed to compare the irradiation performance of TRISO-I and TRISO-II coating designs with LTI and HTI outer PyC coatings. Various SiC layer thicknesses were included. Two of the samples had outer isotropic PyC coatings doped with silicon to determine the effectiveness of this advanced concept for further improving irradiation stability of the PyC layer. All of the particles were coated in laboratory-scale equipment.

Table 3.36
 NUMBER OF COATED PARTICLE SAMPLES SUCCESSFULLY
 TESTED TO INDICATED EXPOSURE

Fast Fluence (10^{21} n/cm ²)											
Type	1	2	3	4	5	6	7	8			
BISO	31	18	31	16	13	9	2	9			
TRISO	21	21	19	22	3	—	5	5			
									↑		
Burnup (% FIMA)											
Type	≤2	4	6	8	10	12	14	16	18	20	>20
BISO	9	11	22	9	2	8	9	5	12	5	37
TRISO	11	2	8	7	8	19	9	6	8	1	17
											↑
Temperature (°C)											
Type	500	600	700	800	900	1000	1100	1200	≥1300		
BISO	--	1	--	3	9	11	1	14	90		
TRISO	6	3	3	--	19	14	6	9	36		
										↑	(~1260)

NOTE: Arrows indicate maximum values for Fort St. Vrain reactor.

The P20 test had been originally designed to be irradiated to a fast fluence of 4×10^{21} n/cm², but the irradiation was extended to 8×10^{21} n/cm² after the P21 test was prematurely terminated. Capsule P20 was very similar to P21 and contained nearly identical samples, but it was designed for a lower thermal-neutron exposure. The burnup values for capsule P20 were therefore considerably higher than the design value and exceeded the expected burnups of the reactor.

Dosimetry and burnup results show this to be the highest exposure test of TRISO particles conducted to date; the fast fluences ranged between 6.5 and 8.7×10^{21} n/cm² ($E > 0.18$ Mev) and burnups between 21% and 27% FIMA.

Postirradiation examination showed that reference design TRISO coatings successfully survived at both the maximum and median HTGR fuel temperatures (1300°C programmed down to 1100°C, and 900°C, respectively). All of the reference LTI coatings survived, whereas all but one of the HTI coatings failed; the reference TRISO-II coatings performed significantly better than the alternative TRISO-I design. In many cases, the SiC layer remained intact, even when the outer PyC layer failed. These results are extremely encouraging and demonstrate that TRISO coated particles should perform successfully in the reactor.

The test results are summarized in Table 3.37, and representative photomicrographs of each sample are shown in Figs. 3.42 through 3.53. Some observations made from the postirradiation stereographic and metallographic examinations are described below.

1. All LTI particle batches had $\leq 1\%$ failure of the outer PyC coating. The appearance of these coatings was good, and they did not display the Maltese-cross pattern under polarized light examination (see Fig. 3.54). Three TRISO-I LTI batches, which had SiC layers 15 μ m thick or less, had a slightly higher failure rate of the SiC layer than of the PyC coating, which resulted in the batch being classified as failed. SiC layer failures were caused principally by kernel-SiC chemical reactions resulting in penetration of the SiC layer.
2. Five of the six particle batches with HTI PyC coatings experienced high failures. Four of these samples were tested at high temperature and exhibited nearly 100% failure of the outer PyC; the other failed HTI was tested at low temperature and had approximately 10% outer PyC failure. All of the HTI coatings had an intense Maltese cross pattern, indicating considerable reorientation had occurred during the irradiation (see Fig. 3.55). Judging from the intensity of the Maltese cross observed on the metallographically sectioned coatings of the one HTI sample that did survive the irradiation (3923-141E on the right in Fig. 3.55), it was inferred that this sample may have been on the verge of failure.

Considerable distortion of the broken outer HTI coatings was very evident in at least one of the batches (3923-149E). In this case the coatings broke in a fashion that resembled a water lily opening up; a number of these coating fragments were no longer spherical in shape, but had begun to straighten out (see Fig. 3.56).

Table 3.37
DESCRIPTION AND POSTIRRADIATION EXAMINATION OF COATED PARTICLES TESTED IN CAPSULE P20
(8-cycle ETR Irradiation)

Lot No.	Type ^(a)	COATING									PARTICLE ^(e)			IRRADIATION CONDITIONS						POSTIRRADIATION EXAMINATION				Remarks
		Buffer		Isotropic		SiC		Isotropic			Total Coating Thickness (μm)	Type	Size (μm)	Th U Ratio	Burnup (% FIMA)	Temperature(°C) ^(g)			Fast Dose (10 ¹⁹ n/cm ²)	Micro Exam (% Fail)	Met Exam (% Fail)			
		Thick (μm)	Density (g/cm ³)	Thick (μm)	Density (g/cm ³)	Thick (μm)	Density ^(c) (g/cm ³)	Thick (μm)	Density (g/cm ³)	B.A.F. (d)						Avg	Max	BOL ^(h)			O PyC	SiC		
3923-05E	TRISO-I LTI	46	1.15	—	—	20	3.17	48	1.93	1.06	114	(Th,U)C ₂	150-250	2.25	23	1125	1300	1100	7.8	0	0.6	0.4	TRISO-I LTI Series	
3923-09E	TRISO-I LTI	46	1.15	—	—	15	3.17	47	1.96	1.00	108	(Th,U)C ₂	150-250	2.25	23	1125	1300	1100	7.8	0.7	0.7	1.3		
3923-07E	TRISO-I LTI	46	1.15	—	—	10	3.17	49	1.95	1.03	105	(Th,U)C ₂	150-250	2.25	23	1125	1300	1100	7.8	0.1	0.2	3		
4032-149E	TRISO-I LTI (Si-doped)	46	1.15	—	—	15	3.17	46	2.08	1.05	107	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.4	0.5	1	2		
3923-127E	TRISO-I HTI	46	1.15	—	—	20	3.17	47	1.96	.99	113	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.7	100(7) ^(f)	100	10	TRISO-I HTI Series	
3923-131E	TRISO-I HTI	46	1.15	—	—	15	3.17	54	1.86	.96	115	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.4	100(45) ^(f)	100	48		
3923-137E	TRISO-I HTI	46	1.15	—	—	10	3.17	50	1.88	.97	106	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.4	100(85) ^(f)	99	96		
3923-07E	TRISO-II LTI	49	1.26	23	(b)	20	3.17	30	1.95	1.01	122	(Th,U)C ₂	150-250	2.25	21	1125	1300	1100	6.5	0	0	0	TRISO-II LTI Series	
3923-101E	TRISO-II LTI	49	1.26	24	(b)	15	3.17	29	1.98	1.10	117	(Th,U)C ₂	150-250	2.25	21	1125	1300	1100	6.5	0	0	0		
4042-7E	TRISO-II LTI (Si-doped)	49	1.26	24	(b)	15	3.17	32	2.09	1.02	120	(Th,U)C ₂	150-250	2.25	21	1125	1300	1100	6.5	0	0	0		
3923-141E	TRISO-II HTI	49	1.26	24	(b)	15	3.17	31	1.84	1.01	119	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.7	0.01	0.6	0.2	TRISO-II HTI Series	
3923-149E	TRISO-II HTI	49	1.26	25	(b)	21	3.17	32	1.86	1.03	127	(Th,U)C ₂	150-250	2.25	27	1125	1300	1100	8.7	100(0) ^(f)	100	0		
3923-05E	TRISO-I LTI	46	1.15	—	—	20	3.17	48	1.93	1.06	114	(Th,U)C ₂	150-250	2.25	25	900	900	900	7.0	1.0	0.9	0.9	Lower Irradiation Temperature Series	
3923-149E	TRISO-II LTI	49	1.26	25	(b)	21	3.17	32	1.86	1.03	127	(Th,U)C ₂	150-250	2.25	25	900	900	900	7.0	1.7	8.6	0		
3923-07E	TRISO-II LTI	49	1.26	23	(b)	20	3.17	30	1.95	1.01	122	(Th,U)C ₂	150-250	2.25	25	900	900	900	7.0	1.2	0.4	0.4		

(a) TRISO-I denotes a coating with a coating sequence of buffer, SiC, and LTI or HTI isotropic PyC. TRISO-II denotes a coating sequence of buffer, LTI isotropic PyC, SiC, and HTI or LTI isotropic PyC. LTI denotes

low-temperature isotropic outer pyrolytic carbon coating, and HTI denotes outer isotropic pyrolytic carbon coating.

(b) Not determined.

(c) Estimated value.

(d) Bacon Anisotropy Factor.

(e) All particles are spheroidal and >90% of theoretical density.

(f) Number in parentheses indicates percent failed SiC layers observed.

(g) Design conditions, programmed to simulate the thermal history of the reactor.

(h) End-of-life temperature.

Fig. 3.42--TRISO-I LTI coated particles from lot 3923-85E before irradiation (left), after irradiation at 900°C (center), and after irradiation at 1300°C (right) in capsule P20 (K-76597, 100X)

(100X)

(160X)

Fig. 3.43--TRISO-I LTI coated particles from lot 3923-99E before (left) and after (right) irradiation in capsule P20 (K-76599)

Fig. 3.44--TRISO-I LTI coated particles from lot 3923-97E before (left) and after (right) irradiation in capsule P20 (K-76592, 100X)

Fig. 3.45--TRISO-I LTI coated particles from lot 4032-149E before (left) and after (right) irradiation in capsule P20 (K-76600, 100X)

(~100X)

(~125X)

Fig. 3.46--TRISO-I HTI coated particles from lot 3923-127E before (left) and after (right) irradiation in capsule P20 (K-76598)

(100X)

(150X)

Fig. 3.47--TRISO-I HTI coated particles from lot 3923-131E before (left) and after (right) irradiation in capsule P20 (K-76595)

Fig. 3.48--TRISO-I HTI coated particles from lot 3923-137E before (left) and after (right) irradiation in capsule P20 (K-76594, 100X)

Fig. 3.49--TRISO-II LTI coated particles from lot 3923-87E before irradiation (left), after irradiation at 900°C (center), and after irradiation at 1300°C (right) in capsule P20 (K-76651, 200X)

Fig. 3.50--TRISO-II LTI coated particles from lot 3923-101E before (left) and after (right) irradiation in capsule P20 (K-76590, 200X)

Fig. 3.51--TRISO-II LTI coated particles from lot 4042-7E before (left) and after (right) irradiation in capsule P20 (K-76593, 200X)

Fig. 3.52--TRISO-II HTI coated particles from lot 3923-141E before (left) and after (right) irradiation in capsule P20 (K-76591, 200X)

(100X)

(~150X)

(~125X)

Fig. 3.53--TRISO-II HTI coated particles from lot 3923-149E before irradiation (left), after irradiation at 900°C (center), and after irradiation at 1300°C (right) in capsule P20 (K-76596)

M-251-62

100X

M-251-134

125X

Fig. 3.54--Polarized light photomicrographs of irradiated particles with LTI coatings. Note absence of Maltese cross pattern in outer isotropic coat, which indicates little or no reorientation had occurred during irradiation. Left, sample 4032-149E, 8.4×10^{21} n/cm²; right sample 3923-87E, 6.5×10^{21} n/cm²

M-251-68

125X

M-251-88

125X

Fig. 3.55--Polarized light photomicrographs of irradiated particles with HTI coatings. Note intense Maltese cross pattern in the outer isotropic coating, indicating that extensive reorientation occurred during irradiation. Left, sample 3923-149E, 8.7×10^{21} n/cm²; right sample 3923-141E, 8.7×10^{21} n/cm²

M-251-23

18X

Fig. 3.56--Photomacrograph showing broken outer isotropic coatings from irradiated sample 3923-149E (1300°C). Note how coatings have distorted as a result of irradiation effects

3. SiC failure percentages were inversely proportional to the SiC coating thickness for the three TRISO-I HTI samples that had total failure of the outer isotropic layer (see Fig. 3.57). The 10- μ m SiC had about 100% failure, the 15- μ m SiC about 50%, and the 20- μ m SiC about 10%. In the one TRISO-II HTI (20- μ m SiC) that had total failure of the outer isotropic coating, none of the SiC layers were broken.
4. In the TRISO-I particles, a small number of reactions between the kernel (and/or fission products) and SiC were observed (see Fig. 3.58). The incidence of reaction in particles with 10- and 15- μ m SiC coatings was greater than in those with 20- μ m SiC coatings. Moreover, it was noted that the particles that showed signs of reaction also had cracked SiC coatings. (There were, of course, many particles with cracked SiC layers that did not show evidence of a reaction.) Interestingly, one batch of TRISO-II particles had a small percentage of buffer and inner isotropic coatings cracked and a few cases where the fuel kernel extruded into the crack(s) and was in contact with the SiC without reacting with it (see Fig. 3.59). The SiC coatings on these particles were not cracked. From the observations it could be speculated that the crack in the SiC preceded and/or enhanced the reactions; however, the reason why this should occur cannot be given, especially in view of the fact that the reaction did not appear to occur in the region of the crack(s) but almost always diagonally opposite the crack.
5. Preirradiation metallography showed a difference in the appearance of SiC coatings deposited at 1400°C compared with those deposited at 1500°C. The 1400°C SiC coatings were optically active when viewed in polarized light, but the 1500°C SiC coatings appeared dark. After irradiation, both structures looked identical and gave no response when viewed in polarized light (i.e., appeared dark), which indicates that the SiC deposited at 1400°C underwent a structural change, possibly densification, during the irradiation (see Fig. 3.60).
6. The three samples irradiated at the median temperature (900°C) performed as well as, if not better than, the high-temperature (1300°C) samples. The HTI sample (3923-149E) experienced considerably less failure than its high-temperature counterpart, and little difference in the over-all performance of the LTI samples was noted.
7. Both of the Si-doped LTI coatings (7% and 14% Si) survived the irradiation. Sample 4032-149E was irradiated at high temperature to a fast fluence of 8.4×10^{21} n/cm², which was the highest LTI exposure. The postirradiation metallographic examination indicated no apparent differences between doped and undoped PyC coatings.
8. No bonding or sticking of any of the irradiated TRISO samples to the graphite thimbles in which they were contained during irradiation was observed.

The P20 irradiation has provided much important information about the irradiation stability of TRISO coated particles. Of primary significance is the successful performance of TRISO LTI particles to fast fluence and burnup values that exceed the full six-year Fort St. Vrain core exposures. The reference TRISO-II LTI particles survived at both the median and maximum

Fig. 3.57--Percent failed silicon carbide coatings versus SiC coating thickness for TRISO-I HTI particles with 100% failure of the outer isotropic coating; irradiation conditions were fast fluence of 8.4 n/cm^2 or more, burnup of 27% FIMA, and temperature of 100°C or greater

M-251-104

200X

Fig. 3.58--TRISO-I particle exhibiting a reaction between kernel and silicon carbide. Note cracks in the SiC opposite the reaction zone

M-251-15

750X

Fig. 3.59--TRISO-II particle in which fuel and silicon carbide are in contact, but show little or no signs of an interaction. There were no cracks in the SiC coating of this particle

M-24736-5

100X

M-251-68

~100X

Fig. 3.60--Polarized light photomicrographs of particles from sample 3923-141E before irradiation (left) and after irradiation (right). Note change in optical response of the silicon carbide coating. SiC is light before irradiation and dark after irradiation.

temperatures to burnups that exceeded the maximum value by as much as 30% and at fast fluences greater than 80% of the maximum expected dose.

The superior performance of the LTI outer pyrolytic coatings compared to HTI coatings was an unexpected bonus; the LTI coatings now appear more stable during irradiation than HTIs in addition to being more economical to produce. The reasons for the differences in performance have not been conclusively ascertained; however, a clue to the failure of the HTI coatings is thought to be given by the presence of the Maltese cross pattern in the outer isotropic layers of the HTIs but not the LTIs. It is inferred from this observation that the HTI coatings underwent considerably more reorientation than the LTIs. This additional amount of orientation experienced by the HTIs could be irradiation induced, stress induced, or both. An increase in orientation would then cause the coating to shrink even more rapidly in the circumferential direction because of the anisotropic dimensional changes of pyrocarbons, thus increasing the stress level in the coating. An autocatalytic process would be initiated that would ultimately result in the failure of the coating.

As a result of the HTI failures, some very interesting results concerning the integrity of the SiC layer were obtained. It was observed for TRISO-I particles that the failure percentage of the silicon carbide was inversely proportional to the SiC layer thickness. A TRISO-I HTI with a 10- μ m SiC coating had almost total failure of the SiC, whereas a 20- μ m SiC coating experienced only 10% failure. More significantly, a TRISO-II with a 20- μ m SiC coating had no SiC failures, which illustrates the value of the inner isotropic layer in improving the performance of the particles. These data allow a direct calculation of the ultimate tensile strength of SiC coatings under irradiation. The results show that the 5000-psi tensile stress limit currently imposed on SiC coatings in coated particle design calculations is conservative, although realistic (see Task V in Part II of this report).

In conclusion, the P20 results add significantly to the confidence that Fort St. Vrain fuel will perform satisfactorily in reactor service. Most encouraging was the observation that the reference fuel design (TRISO-II LTI coated particles) appears to be the best design.

Capsule P22. Capsule P22, a full-exposure test of TRISO coated particles, has been discharged from the ETR after completing irradiation and is being returned to the Gulf General Atomic hot cell for postirradiation examination. It received an estimated fast fluence of 7.8×10^{21} n/cm² with an estimated burnup of 23% FIMA. Most of the TRISO samples in the capsule are early production samples that are not representative of current production materials.

Capsule F-25. Capsule F-25 is a half-exposure (4×10^{21} n/cm²) proof test in ETR of nine TRISO coated particle and 15 fuel rod samples. All coated particle and most fuel rod samples were made in production equipment. This capsule is a backup proof test (to F-28) to control the fuel specification for the first half of the core. The principal TRISO coated particle variables being tested in this experiment are LTI type (propane, propylene, and mixed gas) and isotropic layer density (1.54 to 1.91 g/cm³ for propane LTIs). Six fissile and three fertile samples are included. The principal fuel rod variables are binder (pitch and Si), filler (natural flake graphite and isotropic

graphite), fabrication process (hot injection, extrusion, and SiC sintering), heat treatment (2000°C for 0.5 hr, 1800°C for 0.5 hr, and 1000°C for 1 hr), and TRISO particle type (isotropic layers deposited from propane, propylene, and mixed gas). All fuel rod samples are 0.5 in. in diameter and 0.75 in. long.

The fuel samples being tested in capsule F-25 are summarized in Table 3.38; a detailed description of the samples is given in Table 3.39.

Virgin and previously irradiated samples of H-327 graphite are being irradiated in cell 6 (the lowest flux cell) at approximately 650°C to 2×10^{21} n/cm².

The F-25 capsule is in its second of four cycles in the ETR (F-13 core position). The capsule is operating very satisfactorily, with most fuel rod samples close to the design temperature of 1250°C and all thermocouples still operative. The fissile particle thimble samples are also close to the 1250°C design temperature, but the fertile particle thimble samples are operating about 200° to 250°C below design. The fuel has achieved an estimated fast fluence of 2.1×10^{21} n/cm² to date with an estimated burnup of 11% FIMA. Design exposures are 4×10^{21} n/cm² and 18% FIMA, respectively. The capsule is scheduled to complete irradiation in the Spring of 1970.

Capsule F-26. Capsule F-26 is a full-exposure (8×10^{21} n/cm²) proof test in ETR of four blended beds, eleven fuel rod, and nine TRISO coated particle samples. All coated particle and most fuel rod samples were made in production equipment. This capsule is a backup proof test (to F-29) to control fuel specification for the second half of the core. The blended bed variables being tested are filler (two types of charcoal), restraint (graphite felt and temporary binder), and TRISO coating type (propylene and propane LTI). All bed samples are 0.5 in. in diameter and 0.75 in. long. The coated particle thimble samples are similar to those used in capsule F-25, except for one new fertile sample. The fuel rod samples are similar to those used in F-25, except for the deletion of the four extruded samples, since extrusion was no longer a candidate process. The fuel rod samples are also 0.5 in. in diameter and 0.75 in. long.

The fuel samples being tested in capsule F-26 are summarized in Table 3.40; a detailed description of the samples is given in Table 3.41.

Samples of virgin and previously irradiated H-327 graphite are being tested in cell 6 at approximately 750°C to 4×10^{21} n/cm².

The F-26 capsule is in its second cycle of an eight-cycle irradiation in the ETR in the I-13 core position. The capsule is operating extremely satisfactorily with all monitored blended bed, fuel rod, and coated particle thimble samples close to the design temperature of 1250°C and all thermocouples operative. To date the capsule has received an estimated fast fluence of 1.0×10^{21} n/cm² with an estimated fuel burnup of 6% FIMA. Design values are 8×10^{21} n/cm² and 23% FIMA, respectively. The capsule is expected to complete irradiation in early 1971.

Table 3.38
SUMMARY OF FUEL SAMPLES BEING TESTED IN CAPSULE T-25

Capsule Position and Design Temp	Sample Type	Bed/Rod			Coating Type	Coated Particles					
		Binder/ Restraint	Filler ^a	Heat Treat. (°C)		Iso Layer ^b	Density		Lot No.		
							Gas ^c	Fissile	Fertile	Fissile	Fertile
Cell 1, 1250°C:											
A	Fuel rod	Pitch	NF	2000	TRISO	MG	1.79	1.75	4000-175	4000-187	
B	Fuel rod ^d	Pitch	NF	1000	TRISO	PP	1.83	1.76	4000-182	4000-176	
C	Fuel rod ^d	Pitch	NF	1800	TRISO	PP	1.83	1.76	4000-182	4000-176	
D	Fuel rod ^d	Pitch	NF	2000	TRISO	PP	1.83	1.76	4000-182	4000-176	
E	Fuel rod	Pitch	Iso	1800	TRISO	PP	1.83	1.76	4000-182	4000-176	
Cell 2, 1250°C:											
1	Thimble	--	--	--	TRISO	MG	1.79	--	4000-175	--	
2	Thimble	--	--	--	TRISO	PP	1.68	--	4000-179	--	
3	Thimble	--	--	--	TRISO	PL	1.78	--	4000-183	--	
4	Thimble	--	--	--	TRISO	PP	1.83	--	4000-182	--	
5	Thimble	--	--	--	TRISO	PP	1.54	--	4000-181	--	
6	Thimble	--	--	--	TRISO	PP	1.91	--	4000-180	--	
Cell 3, 1250°C:											
1	Thimble	--	--	--	TRISO	MG	--	1.75	--	4000-187	
2	Thimble	--	--	--	TRISO	PP	--	1.76	--	4000-176	
3	Thimble	--	--	--	TRISO	PL	--	1.76	--	4000-184	
Cell 4, 1250°C:											
A	Fuel rod	Pitch	NF	2000	TRISO	PP	1.83	1.76	4000-182	4000-176	
B	Fuel rod	Pitch	NF	1800	TRISO	PP	1.83	1.76	4000-182	4000-176	
C	Fuel rod	Pitch	NF	1000	TRISO	PP	1.83	1.76	4000-182	4000-176	
D	Fuel rod ^d	Pitch	NF	1800	TRISO	MG	1.79	1.74	4000-175	4000-187	
E	Fuel rod	Pitch	Iso	2000	TRISO	PP	1.83	1.76	4000-182	4000-176	
Cell 5, 1250°C											
A	Fuel rod	Pitch	NF	2000	TRISO	PL	1.78	1.76	4000-183	4000-184	
B	Fuel rod	Si	NF	1850	TRISO	MG	1.79	1.73	4632-3	4452-147	
C	Fuel rod	Si	NF	1850	TRISO	PP	1.83	1.76	4452-151	4452-149	
D	Fuel rod	Si	NF	1850	TRISO	PP	1.83	1.76	4452-151	4452-149	
E	Fuel rod	Pitch	Iso	1800	TRISO	MG	1.79	1.75	4000-175	4000-187	

^aNF = natural flake graphite; Iso = isotropic graphite.

^bOuter isotropic layer.

^cPL = propylene; PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^dMade by extrusion; all others made by hot injection or SiC sintering.

Table 3.39
DESCRIPTION OF COATED PARTICLE SAMPLES IN CAPSULE F-25

Lot No.	Coating													Kernel			Used In		
	Type	Buffer		Isotropic			SiC		Isotropic			Total Coating Thick. (μm)							
		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF	Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF		Dep From ^a	Type	Size (μm)	Th:U Ratio	Blen. Beds	Fuel Rods	Thimble
4000-183	TRISO	50	1.15	20	1.83	1.06	20	3.18	43	1.78	1.04	PL	131	(Th,U)C ₂	175-275	2.75		X	X
4000-181	TRISO	50	1.08	20	1.74	1.06	20	3.22	48	1.59	1.02	PP	135	(Th,U)C ₂	175-275	2.75		X	X
4000-179	TRISO	52	1.26	16	1.85	1.12	20	3.18	42	1.68	1.05	PP	130	(Th,U)C ₂	175-275	2.75		X	X
4000-182	TRISO	52	1.11	25	1.82	1.05	16	3.17	39	1.83	1.02	PP	128	(Th,U)C ₂	175-275	2.75	X	X	X
4000-180	TRISO	45	1.38	23	1.91	1.01	21	3.19	37	1.91	1.01	PP	124	(Th,U)C ₂	175-275	2.75		X	X
4000-175	TRISO	43	1.10	18	1.86	1.05	17	3.19	44	1.79	1.03	MG	126	(Th,U)C ₂	175-275	2.75	X	X	X
4000-184	TRISO	56	1.00	32	1.84	1.07	21	3.18	51	1.76	1.05	PL	145	ThC ₂	450-590	--	X	X	X
4000-176	TRISO	46	1.26	24	1.82	1.08	20	3.19	55	1.76	1.00	PP	146	ThC ₂	450-590	--	X	X	X
4000-187	TRISO	55	1.20	28	1.88	1.08	20	3.20	45	1.75	1.07	MG	145	ThC ₂	450-590	--	X	X	X
4452-151 ^b	TRISO-Si	52	1.11	25	1.82	1.05	16	3.17	39	1.83	1.02	PP	132	(Th,U)C ₂	175-275	2.75		X	
4632-3 ^c	TRISO-Si	43	1.10	18	1.86	1.05	17	3.19	44	1.79	1.03	MG	130	(Th,U)C ₂	175-275	2.75		X	
4452-149 ^d	TRISO-Si	46	1.26	24	1.82	1.08	20	3.19	55	1.76	1.00	PP	154	ThC ₂	450-590	--		X	
4452-147 ^e	TRISO-Si	52	1.19	20	1.81	1.02	19	3.19	54	1.73	1.02	MG	145	ThC ₂	450-590	--		X	

^aPL = propylene, PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^bLot 4000-182 overcoated with 4 μm of Si/SiC.

^cLot 4000-175 overcoated with 4 μm of Si/SiC.

^dLot 4000-176 overcoated with 8 μm of Si/SiC.

^eLot 4000-178 overcoated with 8 μm of Si/SiC.

Table 3.40
SUMMARY OF FUEL SAMPLES BEING TESTED IN CAPSULE F-26

Capsule Position and Design Temp	Sample Type	Bed/Rod			Coated Particles					
		Binder/ Restraint ^a	Filler ^b	Heat Treat. (°C)	Coating Type	Iso Layer			Lot No.	
						Gas ^c	Density			
							Fissile	Fertile	Fissile	Fertile
Cell 1, 1250°C:										
A	Fuel rod	Pitch	NF	2000	TRISO	MG	1.79	1.75	4000-175	4000-187
B	Blen. bed	GF	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
C	Blen. bed	Temp	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
D	Blen. bed	GF	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
E	Fuel rod	Pitch	Iso	1800	TRISO	PP	1.83	1.76	4000-182	4000-176
Cell 2, 1250°C:										
1	Thimble	---	---	--	TRISO	MG	1.79	--	4000-175	---
2	Thimble	---	---	--	TRISO	PP	1.68	--	4000-179	---
3	Thimble	---	---	--	TRISO	PP	1.54	--	4000-181	---
4	Thimble	---	---	--	TRISO	PP	1.83	--	4000-182	---
5	Thimble	---	---	--	TRISO	PL	1.78	--	4000-183	---
6	Thimble	---	---	--	TRISO	PP	1.91	--	4000-180	---
Cell 3, 1250°C:										
1	Thimble	---	---	--	TRISO	PL	--	1.74	---	4000-194
2	Thimble	---	---	--	TRISO	PP	--	1.76	---	4000-176
3	Thimble	---	---	--	TRISO	MG	--	1.75	---	4000-187
Cell 4, 1250°C:										
A	Fuel rod	Pitch	NF	2000	TRISO	PP	1.83	1.76	4000-182	4000-176
B	Fuel rod	Pitch	NF	1800	TRISO	PP	1.83	1.76	4000-182	4000-176
C	Fuel rod	Pitch	NF	1000	TRISO	PP	1.83	1.76	4000-182	4000-176
D	Blen. bed	GF	Charc.	--	TRISO	PP	1.83	1.75	4000-182	4000-185
E	Fuel rod	Pitch	Iso	2000	TRISO	PP	1.83	1.76	4000-182	4000-176
Cell 5, 1250°C:										
A	Fuel rod	Pitch	NF	1800	TRISO	PL	1.78	1.74	4000-183	4000-194
B	Fuel rod	Si	NF	1850	TRISO	MG	1.79	1.73	4632-3	4452-147
C	Fuel rod	Si	NF	1850	TRISO	PP	1.83	1.76	4452-151	4452-149
D	Fuel rod	Si	---	1850	TRISO	PP	1.83	1.76	4452-151	4452-149
E	Fuel rod	Pitch	Iso	1800	TRISO	MG	1.79	1.75	4000-175	4000-187

^aGF = graphite felt; temp = temporary.

^bNF = natural flake graphite; Iso = isotropic graphite; charc. = charcoal.

^cPL = propylene; PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

Table 3.41
DESCRIPTION OF COATED PARTICLE SAMPLES IN CAPSULE F-26

Lot No.	Coating												Total Coating Thick. (μm)	Kernel			Blow Bed	Used In	
	Buffer			Isotropic			SiC		Isotropic			Type		Size (μm)	Th:U Ratio	Fuel Rods		Thimble	
	Type	Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF	Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF								Dep From ^a
4000-183	TRISO	50	1.15	20	1.83	1.06	20	3.18	43	1.78	1.04	PL	131	(Th,U)C ₂	175-275	2.75	X	X	X
4000-181	TRISO	50	1.08	20	1.74	1.06	20	3.22	48	1.59	1.02	PP	135	(Th,U)C ₂	175-275	2.75	X	X	X
4000-179	TRISO	52	1.26	16	1.85	1.12	20	3.18	42	1.68	1.05	PP	130	(Th,U)C	175-275	2.75	X	X	X
4000-182	TRISO	52	1.11	25	1.82	1.05	16	3.17	39	1.83	1.02	PP	128	(Th,U)C	175-275	2.75	X	X	X
4000-180	TRISO	45	1.38	23	1.91	1.01	21	3.19	37	1.91	1.01	PP	124	(Th,U)C ₂	175-275	2.75	X	X	X
4000-175	TRISO	43	1.10	18	1.86	1.05	17	3.19	44	1.79	1.03	MG	126	(Th,U)C ₂	175-275	2.75	X	X	X
4000-194	TRISO	55	1.26	24	1.81	1.12	21	3.19	53	1.74	1.10	PL	145	ThC ₂	450-590	---	X	X	X
4000-176	TRISO	46	1.26	24	1.82	1.08	20	3.19	55	1.76	1.00	PP	146	ThC ₂	450-590	---	X	X	X
4000-185	TRISO	53	1.20	28	1.88	1.08	20	3.19	58	1.75	1.04	PP	159	ThC ₂	450-590	---	X	X	X
4000-187	TRISO	55	1.20	28	1.88	1.08	20	3.20	45	1.75	1.07	MG	145	ThC ₂	295-450	---	X	X	X
4452-151 ^b	TRISO-Si	52	1.11	25	1.82	1.05	16	3.17	39	1.83	1.02	PP	132	(Th,U)C ₂	175-275	2.75	X	X	X
4632-3 ^c	TRISO-Si	43	1.10	18	1.86	1.05	17	3.19	44	1.79	1.03	MG	130	(Th,U)C ₂	175-275	2.75	X	X	X
4452-149 ^d	TRISO-Si	46	1.26	24	1.82	1.08	20	3.19	55	1.76	1.00	PP	154	ThC ₂	450-590	---	X	X	X
4452-147 ^e	TRISO-Si	52	1.19	20	1.81	1.02	19	3.19	54	1.73	1.02	MG	145	ThC ₂	450-590	---	X	X	X

^aPL = propylene; PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^bLot 4000-182 overcoated with 4 μm of Si/SiC.

^cLot 4000-175 overcoated with 4 μm of Si/SiC.

^dLot 4000-176 overcoated with 8 μm of Si/SiC.

^eLot 4000-178 overcoated with 8 μm of Si/SiC.

Capsule F-27. Capsule F-27 is the first test of reference production samples to half exposure (4×10^{21} n/cm²). Included are nine blended bed, nine fuel rod, and nine TRISO coated particle samples. All coated particle (and most fuel rod) samples were made in production equipment and are expected to be representative of materials selected for the core. The principal blended bed variables being tested are restraint (graphite felt, carbon felt, and temporary binder) and coated particle type (TRISO-LTI deposited from propylene and propane and BISO-LTI deposited from propylene). All samples contain charcoal filler. The blended beds are 0.5 in. in diameter and 0.75 in. long.

The fuel samples being tested in capsule F-27 are summarized in Table 3.42; a detailed description of the samples is given in Table 3.43.

The principal fuel rod variables being tested are binder (pitch and Si), filler (natural flake graphite and isotropic graphite), process (hot injection, hot pressing, and SiC sintering), heat treatment (2000°C for 0.5 hr, 1800°C for 0.5 hr, and 1000°C for 1 hr), and coating type (TRISO-LTI deposited from propylene, propane, and mixed gas). All fuel rod samples are 0.5 in. in diameter and 0.75 in. long.

The principal coated particle variables being tested are coating type (BISO and TRISO), HTI layer type (deposition from propylene and propane for TRISO and from propylene and mixed gas for BISO), and HTI layer density (low and medium for TRISO propane samples). Included also are TRISO samples having no outer isotropic layer in order to evaluate the necessity for this layer since some samples in previous capsule tests (see the section above on capsule P20 and previous quarterly report GA-9440) have survived with the SiC layer intact despite failure of the outer isotropic layer. Six fissile and three fertile samples are being tested.

The F-27 capsule, which is beginning its fourth and final cycle in the GETR (E-5 core position), is operating satisfactorily. The monitored blended bed, fuel rod, and coated particle samples in cells 1 to 5 are operating at temperatures of 1130° to 1250°C, close to the design temperature of 1250°C. The bed and rod samples in cell 6 are operating at about 750°C, somewhat below their design temperature of 900°C. Seven of the 12 fuel thermocouples are still operative, and the fuel has reached an estimated fast fluence of 4.4×10^{21} n/cm² to date with an estimated burnup of 16% FIMA. Design exposures are 4×10^{21} n/cm² and 18% FIMA, respectively. The capsule is scheduled to complete irradiation in January 1970.

Capsule F-28. Capsule F-28 is a proof test of reference production fuel samples to half exposure (4×10^{21} n/cm²). The results from this experiment will control the start of fuel production for the first half of the core. Six blended bed, eight fuel rod, and nine TRISO coated particle thimble samples are being tested. Two of the blended bed samples are 0.4 in. in diameter and 4.0 in. long and contain reference TRISO particles with low- and medium-density coatings; they will be irradiated at 1250°C. The other four beds are 0.5 in. in diameter by 0.75 in. long and contain TRISO particles with low-, medium-, and high-density LTI coatings and BISO particles with medium-density coatings; these beds will be irradiated at 900°C. All beds contain coke filler particles and are restrained by a graphite felt plug and a thin layer of temporary binder at the top of the bed.

Table 3.42
SUMMARY OF FUEL SAMPLES BEING TESTED IN CAPSULE F-27

Capsule Position and Design Temp	Sample Type	Bed/Rod			Coating Type	Coated Particles				
		Binder/ Restraint ^a	Filler ^b	Heat Treat. (°C)		Gas ^c	Iso Layer		Lot No.	
							Density			
							Fissile	Fertile	Fissile	Fertile
Cell 1, 1250°C:										
A	Blen. bed	GF	Charc.	--	BISO	PL	1.87	1.71	4000-189	4000-188
B	Fuel rod	Si	---	1850	TRISO	PL	1.78	1.76	4632-47	4632-41
C	Blen. bed	GF	Charc.	--	TRISO	PL	1.83	1.71	4000-192 ^d	4000-193
D	Blen. bed	GF	Charc.	--	TRISO	PP	1.83	1.75	4000-182	4000-183
E	Blen. bed	CF	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
Cell 2, 1250°C:										
1	Thimble	--	---	--	BISO	MG	1.88	1.88	4000-174 ^e	4000-173
2	Thimble	--	---	--	BISO	PL	1.87	--	4000-189	---
3	Thimble	--	---	--	Bare SiC	PL	--	--	4000-190	---
4	Thimble	--	---	--	TRISO	PL	1.78	--	4000-183	---
5	Thimble	--	---	--	TRISO	PP	1.83	--	4000-182	---
6	Thimble	--	---	--	TRISO	PP	1.54	--	4000-181	---
Cell 3, 1250°C:										
1	Thimble	--	---	--	TRISO	PL	--	1.74	---	4000-194
2	Thimble	--	---	--	TRISO	PP	--	1.75	---	4000-185
3	Thimble	--	---	--	Bare SiC	PL	--	--	---	4000-195
Cell 4, 1250°C:										
A	Blen. bed	GF	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
B	Fuel rod	Pitch	NF	2000	TRISO	PP	1.81	1.76	4000-177	4000-176
C	Blen. bed	Temp	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
D	Fuel rod	Pitch	NF	1800	TRISO	PL	1.78	1.74	4000-183	4000-194
E	Fuel rod	Pitch	Iso	2000	TRISO	PP	1.81	1.76	4000-177	4000-176
Cell 5, 1250°C:										
A	Fuel rod	Pitch	NF	1000	TRISO	PP	1.81	1.76	4000-177	4000-176
B	Compact	Pitch	GP-38	1800	BISO	MG	1.88	1.86	4000-174	4000-160
C	Compact	Pitch	GP-38	1800	BISO	MG	1.88	1.86	4000-173	4000-160
D	Compact	Pitch	NF	1800	BISO	MG	1.88	1.86	4000-174	4000-160
E	Fuel rod	Pitch	Iso	1800	TRISO	MG	1.79	1.73	4000-175	4000-178
Cell 6, 900°C:										
A	Fuel rod	Pitch	Iso	2000	TRISO	PP	1.81	1.76	4000-177	4000-176
B	Fuel rod	Pitch	NF	1800	TRISO	PL	1.78	1.74	4000-183	4000-194
C	Blen. bed	GF	Charc.	--	TRISO	PL	1.78	1.74	4000-183	4000-194
D	Compact	Pitch	GP-38	1800	BISO	MG	1.88	1.86	4000-174	4000-160
E	Blen. bed	GF	Charc.	--	TRISO	PP	1.83	1.75	4000-182	4000-185
F	Fuel rod	Si	---	1850	TRISO	PL	1.78	1.76	4632-47	4632-41
G	Blen. bed	GF	Charc.	--	TRISO	PL	1.83	1.71	4000-192 ^d	4000-193

^aGF = graphite felt; CF = carbon felt.

^bNF = natural flake graphite; Charc. = charcoal; Iso = isotropic graphite.

^cPL = propylene; PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^dHigh BAF.

^eLow/high Th, a mixture of fissile and fertile particles.

Table 3.43
DESCRIPTION OF COATED PARTICLE SAMPLES IN CAPSULE F-27

Lot No.	Coating												Total Coating Thick. (μm)	Kernel			Used In		
	Type	Buffer		Isotropic		BAF	SiC		Isotropic			Dep From ^a		Type	Size (μm)	Th:U Ratio	Blen. Beds	Fuel Rods	Thimble
		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF								
4000-192	TRISO	50	1.30	22	1.79	1.12	20	3.20	31	1.83	1.08	PL	120	(Th,U)C ₂	105-175	2.75	X		
4000-183	TRISO	50	1.15	20	1.83	1.06	20	3.18	43	1.78	1.04	PL	131	(Th,U)C ₂	175-275	2.75	X	X	X
4000-181	TRISO	50	1.08	20	1.74	1.06	20	3.22	48	1.54	1.02	PP	135	(Th,U)C ₂	175-275	2.75			X
4000-177	TRISO	45	1.30	19	1.87	1.02	19	3.22	39	1.81	1.01	PP	120	(Th,U)C ₂	175-275	2.75		X	
4000-182	TRISO	52	1.11	25	1.82	1.05	16	3.17	39	1.83	1.02	PP	128	(Th,U)C ₂	175-275	2.75	X		X
4000-175	TRISO	43	1.10	18	1.86	1.05	17	3.19	44	1.79	1.03	MG	126	(Th,U)C ₂	175-275	2.75		X	
4000-193	TRISO	49	1.28	17	1.90	1.30	21	3.20	51	1.71	1.14	PL	130	ThC ₂	450-590	--	X		
4000-194	TRISO	55	1.26	24	1.81	1.12	21	3.19	53	1.74	1.10	PL	145	ThC ₂	450-590	--	X	X	X
4000-185	TRISO	53	1.20	28	1.88	1.08	20	3.19	58	1.75	1.04	PP	159	ThC ₂	450-590	--	X		X
4000-176	TRISO	46	1.26	24	1.82	1.08	20	3.19	55	1.76	1.00	PP	146	ThC ₂	450-590	--		X	
4000-178	TRISO	52	1.19	20	1.81	1.02	19	3.19	54	1.73	1.02	MG	137	ThC ₂	450-590	--		X	
4000-190	Bare SiC	50	1.15	20	1.83	1.06	20	3.18	--	--	--	PL	90	(Th,U)C ₂	175-275	2.75			X
4000-195	Bare SiC	55	1.26	24	1.81	1.12	22	3.19	--	--	--	PL	101	ThC ₂	450-590	--			X
4000-189	BISO	63	1.20	--	--	--	--	--	64	1.87	1.07	PL	127	(Th,U)C ₂	175-275	2.75	X		X
4000-188	BISO	68	1.29	--	--	--	--	--	76	1.71	1.04	PL	144	ThC ₂	450-590	--	X		
4000-174	BISO	54	1.34	--	--	--	--	--	82	1.90	1.02	MG	136	(Th,U)C ₂	350-450	5		X	X
4000-173	BISO	55	1.31	--	--	--	--	--	58	1.86	1.03	MG	113	(Th,U)C ₂	420-495	18		X	X
4000-160	BISO	44	1.35	--	--	--	--	--	88	1.86	1.03	MG	132	ThC ₂	350-450	--		X	
4632-47 ^b	TRISO-Si	50	1.15	20	1.83	1.06	20	3.18	43	1.78	1.04	PL	135	(Th,U)C ₂	175-275	2.75		X	
4632-41 ^c	TRISO-Si	56	1.00	32	1.84	1.07	21	3.18	51	1.76	1.05	PL	153	ThC ₂	450-590	--		X	

^aPL = propylene; PP = propane; MG = mixed gas.

^bLot 4000-183 overcoated with 4 μm Si/SiC.

^cLot 4000-184 overcoated with 8 μm Si/SiC.

All fuel rod samples were made by the hot-injection process with pitch binder and natural flake graphite filler. They contain reference TRISO particles of low, medium, and high density, and high density with high BAF. Five fuel rods are being tested at 1250°C, and three are being tested at 900°C. All rods are 0.5 in. in diameter and 0.75 in. long.

The coated particle samples are intended to demonstrate the stability of reference TRISO coated particles. TRISO coatings of low, medium, and high density are included, along with one high-density sample with a high BAF. Six fissile and three fertile samples are being tested. Because of the small fuel loadings used in the capsule, the thorium-to-uranium ratio in the fissile particle kernels was increased to 18 from the 2.75 used in capsules F-25 through F-27, in order to allow the use of a larger number of fissile particles (at constant fuel loading) in the blended bed and fuel rod samples and thus achieve a more representative blend of fissile to fertile particles. The higher thorium content results, of course, in lower fuel burnup. Thus, the thimble samples include particles both with the high ratio (and low burnup) that was used in the beds and with the low ratio to test reference particles to full burnup.

The fuel samples being tested in capsule F-28 are summarized in Table 3.44; a detailed description of the samples is given in Table 3.45.

The F-28 capsule is beginning its second of four scheduled cycles in the GETR (E-5 core position) and is operating very satisfactorily. The monitored bed, rod, and thimble samples in cells 1 to 5 are operating at 1160° to 1260°C, which are close to the design temperature of 1250°C. The bed and rod samples in cell 6 are operating at 740° to 880°C, close to their design temperature of 900°C. Eleven of the 12 fuel thermocouples are operative. The fuel has reached an estimated fast fluence of 1.5×10^{21} n/cm² with an estimated burnup of 8% FIMA (high-uranium-content samples). Design exposures are 4×10^{21} n/cm² and 18% FIMA (high-uranium samples) or 4% FIMA (low-uranium samples), respectively. The capsule is scheduled to complete irradiation in March 1970.

Capsule F-29. Capsule F-29 is a proof test of reference production fuel samples to full exposure (8×10^{21} n/cm²). The results from this experiment will control the start of fuel production for the second half of the core. The samples being tested in F-29 are identical to those in F-28, except that two sintered SiC matrix rods have been included.

The F-29 samples are described in Tables 3.46 and 3.47.

The F-29 capsule is beginning its second of eight scheduled cycles in the GETR (E-5 core position) and is also operating very satisfactorily. The monitored blended bed, fuel rod, and coated particle samples in cells 1 to 5 are operating at 1230° to 1260°C, extremely close to the design temperature of 1250°C. The beds and rods in cell 6 are operating at 730° to 780°C, somewhat below their design temperature of 900°C. The fuel has reached an estimated fast fluence of 1.5×10^{21} n/cm² with an estimated burnup of 8% FIMA (high-uranium-content samples). Design exposures are 8×10^{21} n/cm² and 23% FIMA (high-uranium samples) or 5% FIMA (low-uranium samples). The capsule is scheduled to complete irradiation in August 1970.

Table 3.44
SUMMARY OF FUEL SAMPLES BEING TESTED IN CAPSULE F-28

Capsule Position and Design Temp	Sample Type	Bed/Rod			Coating Type	Coated Particles				
		Binder/ Restraint	Filler ^a	Heat Treat. (°C)		Gas ^b	Iso Layer		Lot No.	
							Density			
							Fissile	Fertile	Fissile	Fertile
Cell 1, 1250°C: -	Blen. bed ^c	Temp/Felt	Coke	--	TRISO	PL	1.80	1.64	4000-216	4000-199
Cell 2, 1250°C: 1	Thimble	---	---	--	TRISO	PL	1.80	--	4000-216 ^d	---
2	Thimble	---	---	--	TRISO	PL	1.77	--	4000-211 ^d	---
3	Thimble	---	---	--	TRISO	PL	1.93	--	4000-219 ^e	---
4	Thimble	---	---	--	TRISO	PL	1.72	--	4000-223 ^f	---
5	Thimble	---	---	--	TRISO	PL	1.75	--	4000-218	---
6	Thimble	---	---	--	TRISO	PL	1.86	--	4000-220	---
Cell 3, 1250°C: 1	Thimble	---	--	--	TRISO	PL	---	1.64	---	4000-199
2	Thimble	---	--	--	TRISO	PL	---	1.78	---	4000-217
3	Thimble	---	--	--	TRISO	PL	---	1.88	---	4000-215
Cell 4, 1250°C: -	Blen. bed ^c	Temp/Felt	Coke	--	TRISO	PL	1.77	1.78	4000-211	4000-217
Cell 5, 1250°C: A	Fuel rod	Pitch	NF	1800	TRISO	PL	1.77	1.78	4000-211	4000-217
B	Fuel rod	Pitch	NF	1800	TRISO	PL	1.86	1.88	4000-212	4000-215
C	Compact	Pitch	GP-38	1800	BISO	MG	1.88	1.86	4000-173 ^g	4000-160
D	Fuel rod	Pitch	NF	1800	TRISO	PL	1.80	1.64	4000-216	4000-199 ^h
E	Fuel rod	Pitch	NF	1800	TRISO	PL	1.97	1.90	4000-213	4000-210 ^h
Cell 6, 900°C: A	Blen. bed	Temp/Felt	Coke	--	TRISO	PL	1.75	1.78	4000-218 ^h	4000-217
B	Fuel rod	Pitch	NF	1800	TRISO	PL	1.75	1.78	4000-218 ^h	4000-217
C	Blen. bed	Temp/Felt	Coke	--	TRISO	PL	1.72	1.64	4000-223 ^h	4000-199
D	Compact	Pitch	GP-38	1800	BISO	MG	1.88	1.86	4000-174 ⁱ	4000-160
E	Blen. bed	Temp/Felt	Coke	--	TRISO	PL	1.86	1.88	4000-220 ^h	4000-215
F	Fuel rod	Pitch	NF	1800	TRISO	PL	1.72	1.64	4000-223 ^h	4000-199
G	Blen. bed	Temp/Felt	Coke	--	BISO	MG/PL	1.88	1.84	4000-174	4000-224

^aNF = natural flake graphite.

^bPL = propylene; PP = propane; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^c0.4 in. in diameter by 4.0 in. long; all other blended beds and fuel rods are 0.5 in. in diameter by 0.75 in. long.

^dLow burnup.

^eHigh burnup; high BAF.

^fHigh burnup.

^gHigh thorium.

^hHigh BAF.

ⁱLow thorium.

Table 3.45

DESCRIPTION OF COATED PARTICLE SAMPLES IN CAPSULE F-28

Lot No.	Type	Coating												Total Coating Thick. (μm)	Kernel			Used In		
		Buffer		Isotropic		SiC		Isotropic												
		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF	Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF	Dep From ^a								
4000-223	TRISO	57	1.07	22	1.79	1.05	20	3.21	40	1.72	1.03	PL	131	(Th,U)C ₂	175-275	2.75	X	X	X	
4000-218	TRISO	52	1.30	18	1.85	1.00	19	3.21	37	1.75	1.03	PL	122	(Th,U)C ₂	175-275	2.75	X	X	X	
4000-220	TRISO	54	1.16	15	1.90	1.03	22	3.21	39	1.86	1.12	PL	130	(Th,U)C ₂	175-275	2.75	X		X	
4000-219	TRISO	54	1.20	23	1.93	1.21	19	3.21	42	1.93	1.15	PL	132	(Th,U)C ₂	175-275	2.75			X	
4000-211	TRISO	54	1.20	20	1.83	1.03	21	3.20	39	1.77	1.03	PL	135	(Th,U)C ₂	175-275	18	X	X	X	
4000-216	TRISO	58	1.22	28	1.84	1.05	18	3.20	41	1.80	1.04	PL	142	(Th,U)C ₂	175-275	18	X	X	X	
4000-212	TRISO	53	1.20	30	1.88	1.10	22	3.20	39	1.86	1.01	PL	140	(Th,U)C ₂	175-275	18		X		
4000-213	TRISO	53	1.26	22	1.89	1.06	23	3.20	39	1.97	1.05	PL	136	(Th,U)C ₂	175-275	18		X		
4000-217	TRISO	56	1.30	21	1.82	1.07	21	(b)	45	1.78	1.07	PL	135	ThC ₂	295-450	--	X	X	X	
4000-199	TRISO	50	1.24	21	1.75	1.06	19	3.19	52	1.64	1.01	PL	144	ThC ₂	450-590	--	X	X	X	
4000-215	TRISO	51	1.27	22	1.85	1.08	20	3.19	51	1.88	1.00	PL	142	ThC ₂	450-590	--	X	X	X	
4000-210	TRISO	49	1.27	19	1.90	1.14	19	3.19	56	1.90	1.11	PL	138	ThC ₂	450-590	--		X		
4000-174	BISO	54	1.34	--	--	--	--	--	82	1.90	1.02	MG	136	(Th,U)C ₂	350-450	5	X	X		
4000-173	BISO	55	1.31	--	--	--	--	--	58	1.86	1.03	MG	113	(Th,U)C ₂	420-495	18		X		
4000-160	BISO	44	1.35	--	--	--	--	--	88	1.86	1.03	MG	132	ThC ₂	350-450	--		X		
4000-224	BISO	73	1.16	--	--	--	--	--	78	1.84	1.08	PL	151	ThC ₂	295-450	--	X			

^aPL = propylene; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.^bNot available.

Table 3.46
SUMMARY OF FUEL SAMPLES BLING TESTED IN CAPSULE F-29

Capsule Position and Design Temp	Sample Type	Bed/Rod		Heat Treat. (°C)	Coating Type	Coated Particles				
		Binder/ Restraint	Filler ^a			Gas ^b	Iso Layer		Lot No.	
							Fissile	Fertile	Fissile	Fertile
Cell 1, 1250°C: —	Blen. bed ^c	Temp/Felt	Coke	--	TRISO	PL	1.80	1.64	4000-216	4000-199
Cell 2, 1250°C										
1	Thimble	---	---	--	TRISO	PL	1.80	--	4000-216 ^d	---
2	Thimble	---	---	--	TRISO	PL	1.77	--	4000-211 ^d	---
3	Thimble	---	---	--	TRISO	PL	1.93	--	4000-219 ^e	---
4	Thimble	---	---	--	TRISO	PL	1.72	--	4000-223 ^f	---
5	Thimble	---	---	--	TRISO	PL	1.75	--	4000-218	---
6	Thimble	---	---	--	TRISO	PL	1.86	--	4000-220	---
Cell 3, 1250°C:										
1	Thimble	---	--	--	TRISO	PL	--	1.64	---	4000-199
2	Thimble	---	--	--	TRISO	PL	--	1.78	---	4000-217
3	Thimble	---	--	--	TRISO	PL	--	1.88	---	4000-215
Cell 4, 1250°C: —	Blen. bed ^c	Temp/Felt	Coke	--	TRISO	PL	1.77	1.78	4000-211	4000-217
Cell 5, 1250°C:										
A	Fuel rod	Pitch	NF	1800	TRISO	PL	1.77	1.78	4000-211	4000-217
B	Fuel rod	Pitch	NF	1800	TRISO	PL	1.86	1.88	4000-212	4000-215
C	Fuel rod	Si	--	1850	TRISO	PL	--	--	4632-67	4632-71
D	Fuel rod	Pitch	NF	1800	TRISO	PL	1.80	1.64	4000-216	4000-199
E	Fuel rod	Pitch	NF	1800	TRISO	PL	1.97	1.90	4000-213	4000-210 ^g
Cell 6, 900°C:										
A	Blen bed	Temp/Felt	Coke	--	TRISO	PL	1.75	1.78	4000-218 ^f	4000-217
B	Fuel rod	Pitch	NF	1800	TRISO	PL	1.75	1.78	4000-218 ^f	4000-217
C	Blen. bed	Temp/Felt	Coke	--	TRISO	PL	1.72	1.64	4000-223 ^f	4000-199
D	Fuel rod	Si	--	1850	TRISO	PL	--	--	4632-67	4632-71
E	Blen. bed	Temp/Felt	Coke	--	TRISO	PL	1.86	1.88	4000-220 ^f	4000-215
F	Fuel rod	Pitch	NF	1800	TRISO	PL	1.72	1.64	4000-223 ^f	4000-199
G	Blen. bed	Temp/Felt	Coke	--	BISO	MG, PL	1.88	1.84	4000-174	4000-224

^aNF = natural flake graphite.

^bPL = propylene; PP = propane, MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^c0.4 in. in diameter by 4.0 in. long; all other blended beds and fuel rods are 0.5 in. in diameter by 0.75 in. long.

^dLow burnup.

^eHigh burnup; high BAF.

^fHigh burnup.

^gHigh BAF.

Table 3.47
DESCRIPTION OF COATED PARTICLE SAMPLES IN CAPSULE F-29

Lot No.	Type	Coating												Total Coating Thick (μm)	Kernel			Used In		
		Buffer		Isotropic		BAF	SiC		Isotropic											
		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)		Thick. (μm)	Density (g/cm ³)	Thick. (μm)	Density (g/cm ³)	BAF	Dep From ^a								
4000-223	TRISO	57	1.07	22	1.79	1.05	20	3.21	40	1.72	1.03	PL	131	(Th,U)C ₂	175-275	2.75	X	X	X	
4000-218	TRISO	52	1.30	18	1.85	1.00	19	3.21	37	1.75	1.03	PL	122	(Th,U)C ₂	175-275	2.75	X	X	X	
4000-220	TRISO	54	1.16	15	1.90	1.03	22	3.21	39	1.86	1.12	PL	130	(Th,U)C ₂	175-275	2.75	X		X	
4000-219	TRISO	54	1.20	23	1.93	1.21	19	3.21	42	1.93	1.15	PL	132	(Th,U)C ₂	175-275	2.75			X	
4000-211	TRISO	54	1.20	20	1.83	1.03	21	3.20	39	1.77	1.03	PL	135	(Th,U)C ₂	175-275	18	X	X	X	
4000-216	TRISO	58	1.22	28	1.84	1.05	18	3.20	41	1.80	1.04	PL	142	(Th,U)C ₂	175-275	18	X	X	X	
4000-212	TRISO	53	1.20	30	1.88	1.10	22	3.20	39	1.86	1.01	PL	140	(Th,U)C ₂	175-275	18		X		
4000-213	TRISO	53	1.26	22	1.89	1.06	23	3.20	39	1.97	1.05	PL	136	(Th,U)C ₂	175-275	18		X		
4000-217	TRISO	56	1.30	21	1.82	1.07	21	(b)	45	1.78	1.07	PL	135	ThC ₂	295-450	--	X	X	X	
4000-199	TRISO	50	1.24	21	1.75	1.06	19	3.19	52	1.64	1.01	PL	144	ThC ₂	450-590	--	X	X	X	
4000-215	TRISO	51	1.27	22	1.85	1.08	20	3.19	51	1.88	1.00	PL	142	ThC ₂	450-590	--	X	X	X	
4000-210	TRISO	49	1.27	19	1.90	1.14	19	3.19	56	1.90	1.11	PL	138	ThC ₂	450-590	--		X		
4632-67	TRISO-Si	54	1.20	20	1.83	1.03	21	3.20	39	1.77	1.03	PL	139 ^c	(Th,U)C ₂	175-275	18		X		
4632-71	TRISO-Si	56	1.30	21	1.82	1.07	21	(b)	45	1.78	1.07	PL	143 ^d	ThC ₂	295-450	--		X		
4000-174	BISO	54	1.34	--	--	--	--	--	82	1.90	1.02	MG	136	(Th,U)C ₂	350-450	5	X			
4000-224	BISO	73	1.16	--	--	--	--	--	78	1.84	1.08	PL	151	ThC ₂	295-450	--	X			

^aPL = propylene; MG = mixed gas (propane-acetylene). Inner isotropic is deposited from same gas as outer isotropic.

^bNot available.

^cLot 4000-211 overcoated with 4 μm of Si/SiC.

^dLot 4000-217 overcoated with 8 μm of Si/SiC.

Capsule F-30. Capsule F-30 will be an irradiation test of a full-sized (0.5 in. in diameter by 30 in. long) blended bed of TRISO particles. The bed will be irradiated to a fast fluence of 6 to 8×10^{21} n/cm² at a temperature of 1250°C (maximum). In addition to the usual temperature monitoring, the capsule is expected to contain instrumentation to measure the following:

1. Neutron-induced shrinkage of the H-327 graphite tube containing the bed.
2. Relative movement between the bed and the graphite tube.
3. Any failure of the graphite tube.
4. The vibration environment encountered during irradiation.

Design of the capsule is currently in progress. Studies are being conducted to develop the instrumentation techniques necessary to accomplish the above objectives. The capsule is scheduled to begin irradiation in February in GETR cycle 115.

Coating Densification Measurements. The densification of PyC coatings under fast-neutron fluences up to 4.3×10^{21} n/cm² has been measured on nine samples of TRISO coated particles that were irradiated in capsules P21 and P23. The preirradiation densities of the outer PyC coatings varied between 1.55 and 2.01 g/cm³; both LTI and HTI coatings were included. The majority of the particles were irradiated at 1100° to 1300°C, although three samples were tested in duplicate at 900°C. The PyC coatings were stripped from the TRISO particles by crushing. A sink-float technique utilizing mixtures of benzene and tetrabromoethane was employed to measure the density of the coating fragments. The two liquids were blended to provide a uniform-density-gradient column and a density gradient curve for the liquid column was determined with calibrated glass floats; the position of the PyC coatings relative to the glass floats established their density.

Densities were measured on unirradiated as well as on irradiated samples. Coatings from approximately 1000 particles were used for the preirradiation measurements, and coatings from about 300 particles were used for the post-irradiation determinations. The PyC coatings normally exhibited a rather narrow range of densities (± 0.03 g/cm³).

The results of the measurements are presented in Table 3.48 along with the particle descriptions, PyC coating parameters, and irradiation conditions. (A rather large variation between densities measured on the coated particles and those measured on coatings deposited on graphite disks was found in some cases, indicating that the properties of the disk coatings may not always be identical to the properties of the particle coatings). The PyC densification for various initial densities as a function of fast fluence is shown in Fig. 3.61 for LTI PyC coatings and in Fig. 3.62 for HTI PyC.

These data were compared to irradiation data on density changes of unrestrained PyC strips. The data obtained from the PyC coated TRISO particles appear to follow the general patterns of density changes for unrestrained PyC; however, the degree of densification was, in most cases, less than that found with unrestrained samples. Recent data have indicated that restrained pyro-carbon coatings undergo less densification than unrestrained strips.

Table 3.48
COATING DENSITIES OF IRRADIATED PARTICLES

Sample No.	Type	Irradiation Conditions					Coating Parameters (Outer Isotropic)				
		Capsule	Temp (°C)			Fast Dose (10 ²¹ n/cm ²)	Thick- ness (μm)	BAF	Density (g/cm ³)		
			Avg.	Max.	EOL ^a				Unirradiated		Irradiated Particle
									Disk	Particle	
4263-77E	TRISO-II LTI	P23	905	930	910	2.0	31	1.00	1.53	1.54	1.83
4263-77E	TRISO-II LTI	P23	1240	1280	1250	2.3	31	1.00	1.53	1.54	1.92
4263-59E	TRISO-II LTI	P23	1240	1240	1240	2.3	30	1.00	1.73	1.68 ^b	1.95
4263-75E	TRISO-II LTI	P23	1230	1260	1220	2.1	30	1.00	1.83	1.76	2.00
3923-113E	TRISO-II LTI	P21	1240	1375	1125	3.0	29	1.01	1.95	2.01 ^b	2.07
4263-91E	TRISO-II HTI	P23	950	980	960	1.8	28	1.07	1.55	1.64	1.91
4263-91E	TRISO-II HTI	P23	1245	1270	1260	2.1	28	1.07	1.55	1.64	1.96
3923-139E	TRISO-II HTI	P21	1220	1350	1130	4.3	29	1.01	1.84	1.69 ^b	1.99
4263-83E	TRISO-II HTI	P23	1245	1270	1260	2.1	31	1.07	1.86	1.82	2.02
4263-81E	TRISO-II HTI	P23	950	950	950	1.8	30	1.07	1.76	1.83	2.03
4263-81E	TRISO-II HTI	P23	1245	1265	1250	2.1	30	1.07	1.76	1.83	2.07
3923-71E	TRISO-II HTI	P21	1260	1380	1170	4.3	48	1.00	1.98	1.93	1.99

^aEnd-of-life temperature.

^bDuplicate measurements made.

Fig. 3.61--Estimated density - fast-fluence relationship for LTI PyC coatings irradiated at 1170° to 1250°C

Fig. 3.62--Estimated density - fast-fluence relationship for HTI PyC coatings irradiated at 1170° to 1250°C

From the limited data presented here, the restrained PyC coatings irradiated at about 1200°C appeared to densify 2% to 3% less than HTI unrestrained strips of comparable initial density. There are currently no high-temperature irradiation data available for LTI coatings.* There did not appear to be a marked difference between the amount of densification for LTI and HTI restrained coatings. Coating density measurements were performed on only three samples irradiated at a low temperature (900°C), but the data appeared to be consistent with the expected densification profile.

The conclusions drawn from the initial portion of this study are:

1. Outer PyC coating densities can be measured directly using a sink-float technique. This method is applicable to both unirradiated and irradiated particles.
2. The outer PyC coating density appears to be fairly uniform within a particular batch, but there was not always a good correlation between this value and the density obtained from coatings taken from disks.
3. The initial data gathered using irradiated TRISO particles appear to correlate reasonably well with values currently accepted in the literature.

Further measurements are being made to extend these data to the full value of fast fluence (8×10^{21} n/cm²).

Graphite Irradiations. In addition to the fuel specimens discussed above, capsules F-27, F-28, and F-29 contain graphite samples in piggyback locations within the walls of the crucibles containing the fuel at temperatures below 800°C. Cell 4 in capsule F-27 contains 0.105-in.-diameter specimens of a graphite log manufactured at the mid-production stage. Cells 1, 4, and 6 in capsules F-28 and F-29 also contain production-grade graphite samples similar in size taken from the mid-length center and edge and the end center and edge locations of a full-size log.

CHEMISTRY

The objectives of this subtask are to determine:

1. The release behavior of solid fission products that might be important plateout sources and gaseous fission products from the fuel particles being considered for use in the plant. Release behavior will be determined as a function of time, temperature, and irradiation by means of analysis and postirradiation annealing and in-TRIGA furnace tests on irradiated fuel particles. The data will be used to determine the effects of such variables as particle diameter, coating thickness, and

*Recent data have supplied new information about the densification rates of LTI and HTI pyrocarbons. LTI pyrocarbons appear to stabilize after experiencing their maximum amount of shrinkage, whereas HTI pyrocarbons turn around and expand anisotropically. This observation indicates that LTI coatings should be more stable than HTI coatings at high fast fluences, which was evidenced by the P20 irradiation experiment.

coating structure and as input for the calculation of fission-product release to the coolant circuit. The particles used in these tests will be irradiated under other tasks of this program or will be obtained from other sources.

2. The sorption and diffusion behavior of solid fission products in graphites, being considered for use in the plant, as a function of temperature and concentration by means of analysis and out-of-reactor chemical experiments on graphite samples doped with "fissium."
3. The deposition (sorption) behavior of metallic fission products on coolant circuit metals and the relative distribution of fission products between carbon dust and metal surfaces.
4. The extent and effects of oxidation attack on both fuel particles and fuel element graphite as the result of steam leakage into the coolant circuit.

Impurity Attack on Fuel Materials

Steam Graphite Reaction Rate Studies. Experiments to determine the rate of steam oxidation of H-327 graphite are continuing. Specimens have been prepared from one of the latest production logs. So far, two specimens from this log have been run in the recording electrobalance to measure reaction rate data and neither has shown the anomalous behavior (initially high reaction rate) observed for specimens from earlier pilot logs.

The H-327 production specimens show a reaction rate similar to that of AGOT graphite. The reaction rate data for AGOT graphite are essentially those used for calculating the degree of graphite oxidation in reactor systems. This means that, based on existing rate data for H-327 graphite no revision needs to be made in reaction calculations to take into account the use of H-327 production graphite.

An attempt has been made to determine the cause of the previously reported anomalous behavior. Microprobe examinations were performed on specimens of the H-327 production log and of an earlier specimen that underwent steam attack. These examinations have so far revealed (within the sensitivity of the measurements) no highly concentrated Fe, Ti, or V spots that would be expected to exhibit catalytic effects. Both the earlier and later graphite material were also checked for the degree of graphitization and were observed to be more completely graphitic than typical graphite material examined in the past.

Reaction of Water with Fuel Material. Experiments to measure the consequences of steam hydrolysis of exposed fuel in fuel channels are continuing. Tests have been performed (1) on unrestrained fuel particle beds to determine expansion characteristics of exposed kernels and (2) on fuel beds contained in H-327 graphite crucibles to measure possible expansion or cracking of webs as a result of fuel hydrolysis. The fuel beds consisted of mixtures of failed and intact BISO coated particles. In some tests charcoal-type material was mixed with the particles.

Six expansion tests were performed on unrestrained fuel beds, and the results are shown in Table 3.49. It is seen that appreciable expansion of loose beds containing failed particles occurred as a result of kernel hydrolysis.

Table 3.49
EXPANSION OF UNRESTRAINED FUEL BEDS
FOLLOWING HYDROLYSIS AT 260°C

Exp. No.	Failed Particles (%)	Charcoal Filler (g/cm ³)	Expansion (%)
1147-103-1	3	None	<1
1147-103-2	15	None	12
1147-103-3	62	None	55
1147-105-4	3	0.1	0.4
1147-105-5	15	0.1	0.9
1147-105-6	62	0.1	59

Five tests were performed on closed crucibles containing fuel undergoing hydrolysis. These tests were designed to help estimate possible stresses generated in the fuel element webs. Accordingly, crucibles were manufactured of H-327 graphite with walls of sufficient thickness to simulate the strength of the weakest webs in the fuel block. The resultant crucible dimensions were 0.5 ID, 0.813 OD, and 3 in. long. Inside-threaded screw lids were provided on each end so that the crucible was essentially a hollow tube with end plugs. With this arrangement expansion or failure would not be hindered by having a solid end on the crucible.

The experimental apparatus was designed so that crucible expansion could be measured during the steam exposure by using a dial gage attached to a platinum wire that was looped three times around the crucible. Thus, diametral expansion was amplified by a factor of 3π .

The crucibles were tightly packed with a mixture of so-called strong and weak BISO coated particles with ThC₂ kernels and in some cases charcoal filler material. The strong particles had BISO coatings consisting of a 70- μ m buffer layer and a 113- μ m buffer layer and a relatively thin (21- μ m) outer isotropic layer. The crucibles were annealed at 2400° to 2450°C for 10 min. This initial anneal caused failure of essentially all the weak particles and in most cases none of the strong particles.

The crucibles were then exposed to He containing 0.03 atm H₂O for times up to 268 hr at 240°C. At the conclusion of the exposures, the crucible contents were analyzed for unfailed (strong) particles by a sieving technique. In this way, the failed fraction was accurately determined. This procedure was not performed for test 4705-10; in this case a photomicrograph technique was employed, and the exact failed fraction could not be determined.

The experimental results are shown in Table 3.50. It is seen that crucible failures occurred only in the cases where the failed fraction was

Table 3.50

RESULTS OF HYDROLYSIS OF FUEL PARTICLE BEDS CONTAINING BROKEN PARTICLES AND HELD
IN CLOSED H-327 GRAPHITE CRUCIBLES

Sample No.	Failed Particle Content (%)	Charcoal Filler Content (g/cm ³)	Time (hr)	Results
1147-109	12	0.1	168	No expansion
4705-24	33	0	262	No expansion
4705-10	~62 ^a	0.1	119 ^b	No expansion
1147-108	75	0.2	<70	Crucible failed
1147-107	100	0	<22	Crucible failed

NOTE: Tests were performed at 260°C; the water content of the helium flowing over the samples was 3%.

^aThe percent failure was calculated (not actually measured) based on expected behavior of particles used (see text). The actual percent failed could be somewhat lower.

^bSample 4705-10 was exposed to 3% steam for 78 hr and to about 80% steam for an additional 41 hr.

quite large. No expansion was observed on the dial gage during the exposures in which failures did not occur. The dial gage sensitivity is about 5×10^{-4} in.; thus the diametral expansion for these runs was less than about 6×10^{-5} in. Calculations for stresses in thick-walled hollow cylinders show that the internal pressure required to cause a 6×10^{-5} diametral expansion in these test crucibles is 44 psi. It is concluded, therefore, that the pressure generated in the unfailed crucibles was less than 44 psi.

References

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Task IV

FUEL ELEMENT PROOF TESTING

The objective of this task is to demonstrate suitable performance of a test fuel element(s) based on the plant fuel element design by means of irradiation testing at expected fuel operating temperatures at greater than expected temperature gradients and up to or exceeding approximately one third of the planned fuel element burnup in the plant.

The Peach Bottom reactor was shut down on October 3, 1969 for the scheduled 450 EFPD fuel element inspection. At the time of shutdown, the proof test element had generated a total of 788 Mw-hr of energy. The Chromel-Alumel thermocouple was operating reliably and read typically 1530° to 1674°F as the reactor power level ranged from 35% to 94%.

PART II

PRIVATELY SPONSORED RESEARCH AND DEVELOPMENT PROGRAM

(Not the Subject of AEC Reimbursement)

Task I

FUEL TRANSFER MACHINE

The objective of this task is to establish the suitability and maintainability of the fuel transfer machine for use in the plant by means of testing of the fuel transfer machine with a segment of a dummy core, and by carrying out tests in helium at the temperature and chemical impurity level expected in the plant of critical components, as necessary.

Testing to develop the automatic programming and operating of the fuel handling machine utilizing the computer console was accomplished during the quarter. It consisted of four parts: (1) automatic operation of each individual motion, (2) tracking operation of the grapple head, (3) search pattern testing of first handled element, and (4) handling of offset elements.

Also accomplished were tests to develop the safe speeds and stopping distances of the vertical motion. The safe speeds were developed from determining permissible overtravel, as sensed by the grapple head. Overtravel is related to element pickup and deposit speeds, as well as to travel speeds in the core where the possibility of contact with various surfaces exists.

Tests were then performed to appraise handling techniques for various elements. The first portion of this testing was done to establish a method of handling 30° rotated border elements, and the second to assess wall tracking by the grapple head versus direct placement of elements. The tests on the 30° rotated border element established that these elements could be placed in their core positions only from above. The tests showed that the core wall could not be used to cam an element into its final orientation. Because there is insufficient clearance over the top of the core to handle a full-length 30° rotated border reflector element, the test showed that these elements must be half length. The effect of providing half length reflector blocks at these locations is being evaluated.

The new hydraulic motors were received and installed. Initial operation with these motors has been satisfactory and they thus far have performed as expected. Final tuning of the drive system with respect to acceleration and balance between units is in progress.

Some operation of the vertical motion was rough and erratic during this period. This was traced in one case to the hydraulic pump yoke potentiometer; in another case, to the servo command and position potentiometers. Investigation of the first problem showed that the yoke potentiometer shaft was bound up and would not rotate. It is believed that the mounting configuration caused this situation, and improved mounting schemes are being considered. The erratic motion caused by the command and position potentiometers was investigated, and it was discovered that the signal from these potentiometers was electrically noisy, resulting in rough motion. The resistance elements of

these units were wiped with solvent and cleaned of foreign matter, which returned the vertical motion to normal operation.

Continued operation within the fuel storage rack revealed additional interferences with the rack structure and portions of elements, grapple head, and index arm. These interferences with the rack were observed during operation by personnel. The major cause of interference was with the remaining portions of the element restraint rails. The removal of these remaining portions resulted in five of the seven columns being clear for element handling. The other two columns have additional interference, and plans to rework and eliminate this are being developed.

Preliminary analysis of the handling of bowed elements showed that these elements could be handled with the existing grapple if the handling hole were enlarged to allow insertion of the grapple head probe. A simulated element was modified to this new configuration, and a test showed that the element could be successfully grappled at an angle simulating the maximum bow. Plans are being developed for additional tests to demonstrate actual pickup and deposit of bowed elements.

An analysis was completed to provide increased assurances of maintaining a grappled condition with fuel elements. It was found that a mechanically actuated valve could be incorporated into the grapple head to prevent the element from being released unless all weight was relieved from the grapple probe.

An evaluation was performed to determine the ability of the index arm actuating link to collapse under a given external load should the index arm be jammed in the extended position. It was shown that a series of shear pins would allow the jammed index arm to retract when it is pulled against the refueling penetration. An evaluation of the effect of the loads imposed on the lower end of the refueling penetration is proceeding.

A Fortran program was prepared for the 1108 computer to generate complete refueling cycle data tapes. It is called DATA II and has been used to produce the data tapes for the automatic testing performed to date.

It is planned to replace the signal conditioning units for the strain gage transducers of the fuel handling machine because of inadequate electronics packaging and component installation. New units were procured and received, but after installation it was found that these units were subject to electrical noise and false signal output. The units were returned to the vendor for reworking to eliminate this problem.

Task II

SERIES-STEAM-TURBINE-DRIVEN HELIUM CIRCULATOR

The objective of this task is to establish the suitability of the main helium circulators and steam turbine drives for use in the plant by means of analysis, component tests, and prototype testing under conditions of temperature and speed expected in the plant and up to at least 50% of design horsepower. Compressor blade and rotor vibration measurements will be made during these tests. The duration of the full-scale prototype testing will be approximately 200 hr.

The final evaluation of results from the tests on the Mark II prototype helium circulator has been completed. This includes performance data on both the helium compressor and the steam turbine. Sufficient data were obtained from the tests to permit an evaluation of the performance capability of the compressor and turbine.

The pressure-level fluctuations (noise) generated by the prototype main helium circulator have been measured in both helium and air during development testing at the Valmost Circulator Test Facility and the results are being reviewed.

Task IV

STEAM GENERATOR

The objective of this task is to establish the suitability and maintainability of the design of the steam generators for the conditions of temperature and pressure expected in the plant by means of conceptual design, analyses, and testing of the critical aspects of the design. The tests are expected to include gas flow distribution and pressure drops, helium-side heat transfer, steam-water flow stability, tube vibration, and experimental investigation of material compatibility with coolant impurities. The development of a high-temperature design basis and investigation of experience relative to tube integrity is also planned.

STEAM-WATER FLOW STABILITY TEST PROGRAM

The records of recent experimental runs of the boiling stability test loop are being examined to determine the effect of the addition of an orifice restriction at the coil outlet. This recent test series was performed to simulate the behavior of the steam generator module encompassing circuits between feedwater and outlet steam ringheaders. Results will be used in control and transient analyses.

A study was completed on the behavior of deposits in orifices and other devices involving high-velocity liquid flow. This effort is related to the design of stabilizing orifices employed in the steam generator tube circuits.

MODULE AIR-FLOW TEST

The assembly of the blowers and water system was completed. The data acquisition system is being assembled.

The required air and water connections to the test module are being designed.

MATERIALS TESTS

Stress Rupture Tests on Sanicro 31

Tests are nearing completion on the Sanicro 31 tube material that has been cold-worked to varying degrees and subjected to several transient temperature conditions, though three specimens are still being tested. An evaluation of the results so far obtained shows that the strain hardening caused by the cold working has given the highest stress-to-rupture times but with a subsequent loss in ductility, the values for reduction of area and elongation

being low. With 20% cold work the stress-to-rupture times are satisfactory up to and including the 1600°F transient. This general trend is shown for the 10% cold work, but a substantial scatter is apparent for the duplicate specimens. The as-received material before being subjected to cold working shows some very scattered results on duplicate specimens, which casts doubt on the validity of the test. However, it appears that a transient metal temperature between 1500° and 1600° lasting 1 hr would be required to affect the stress-to-rupture life of the steam generator Ni/Fe/Cr tube materials.

Fatigue Testing

Creep tests on Incoloy 800 conducted at Gulf General Atomic are nearing completion. From a first evaluation of this data, it appears that additional tests may be necessary to confirm some of the test results involving large deviations that occurred under duplicate conditions.

Hold-time effects are to be studied at Battelle Memorial Institute. The specimens have been delivered, and the test equipment at Battelle has been checked out.

TUBE PLUGGING MACHINE

The primary design effort during the quarter was concentrated on the construction drawings for the tube plugging tools, which were released for fabrication. The design work on this task was brought near to completion, with only the installation drawings and other minor design tasks still in process.

The test program for the seal-weld tools was completed successfully, and refurbishment and final checkout are under way before shipment to the site. The results of the welding tests were presented in the previous quarterly report, GA-9720.

The test objective for the cutting tool was that it cut completely through a sample without a change of cutters. This objective was satisfied on three weld samples. The lower level of rigidity in the mounting system for the cutter limited the feed rates to 1.2 in./min in circumferential motion and a radial movement of about 0.050 in. per circumferential pass. A total of four passes were required to cut through the sample, the purpose of the fourth pass being to verify that the seal was completely cut at all points.

During the development of the cutting parameters, it was learned that an air-line oiler was required to maintain air motor efficiency through the cut. In addition, the tool steel cutters had to be nitrided to obtain sufficient life so that a complete cut could be made without changing cutters.

Task V

COATED PARTICLE DEVELOPMENT

The objective of this task is to determine the process variables for fabrication of coated fuel particles suitable for use in the plant by means of investigation of various methods of fabricating fuel particles, investigation of coating process variables, and preirradiation evaluation of coated particles.

COATED PARTICLE EVALUATION

Alpha-counting of bulk particles is being investigated as a means of determining the level of fuel contamination on or near the surface of TRISO coated fuel particles. Levels of fuel contamination can be related to fission-gas release behavior (see earlier quarterly report GA-9130). The goal of this work is to develop the alpha-counting technique as a low-cost method for establishing that the Fort St. Vrain production fuel particles meet fission-product release criteria.

As described in an earlier quarterly progress report (GA-9440), the results of tests using the bulk-particle alpha-counting techniques have not been as encouraging as those obtained with the thin-film alpha-counting technique.* The data for samples tested using the bulk-particle technique showed a large spread in a plot of surface uranium contamination versus alpha count, and the slope of the line was not unity. The large spread in the data may be partially attributed to the poor resolution and sensitivity of the alpha spectrometer that was used. An alpha spectrometer with more sensitivity and better resolution has been obtained.

The deviation from a unity slope for the bulk-particle alpha-count data may have been the result of attenuation of the alpha particles in the surface material of the particles, and the degree of self-attenuation may have varied with the degree of uranium surface contamination. Examination of the alpha spectra showed a low-energy continuum on some of the spectra indicating the occurrence of attenuation. This made integrations of the spectra to obtain surface fuel contamination somewhat uncertain. The continuums could be attributed to fuel contamination in (rather than on) the PyC coating.

On the basis of these results, it appears that bulk-particle alpha-counting techniques may be more meaningful than the thin-film (surface leach) technique for predicting the release of fission gases. The fission gases originating

*In the thin-film alpha-counting technique, the surface fuel contamination is leached into solution and counted in a thin-film geometry using appropriate standards.

in the outer 10 to 20 μm of the PyC coatings would not be accounted for using the thin-film technique. In those samples in which the uranium contamination in the coating is high compared to that on the surface, most of the fission gases would evolve by recoiling out of the particle coatings rather than by recoiling from their surface.

It is doubtful whether the alpha-counting technique will be satisfactory for detecting particles with defective coatings (e.g., particles with no SiC layer). The method presently employed for determining defective TRISO coated particles consists of burning off the outer PyC layer and leaching the particles. This procedure is considered undesirable because of the destructive nature of the test and the time-consuming procedure.

THERMAL STABILITY OF COATED PARTICLES

The purpose of this study is to determine the mechanisms for chemical interaction of kernel and coating in TRISO coated particle fuel, and ultimately, to establish more accurate time-temperature operating limits for TRISO particles. Effort has been concentrated on out-of-pile heating experiments in which nominal temperature and temperature gradient magnitude are the primary variables. The condition of the particles is monitored by periodic radiographic analysis.

Thermal Gradient Experiments

Beds of TRISO coated particles are being heated in a thermal gradient and radiographically examined at intervals to detect changes in distribution of Si, C, Th, and U.

The apparatus and specimens for experiment 4283-72 were described in a previous quarterly report (GA-9261). Heating of particles continued in a thermal gradient of 0.01 to 0.04°C/ μm at a nominal temperature of 1400° to 1550°C. The experiment was temporarily interrupted during this report period for maintenance work on the crucibles and heater electrode at about 6000 hr of heating. At that time the P-capsule thimble samples were removed.

All but one of the four P-capsule thimble specimens were inadvertently destroyed during removal from the holder. The one surviving thimble with particles contained in the annulus was impregnated with resin and sectioned for metallographic examination. The examination is still in progress.

Heating of these specimens is being continued, time at temperature having now reached 6580 hr. The next examination will be at 8000 hr.

Experiment 4464-25 is a long-term intermediate-temperature (1000° to 1300°C) experiment in which samples of 14 different TRISO coating batches are being heated in a thermal gradient of 0.01 to 0.06°C/ μm . The apparatus and specimens were described in detail in the previous quarterly report (GA-9720).

The specimens have been examined at 500 and 1000 hr of heating, and no significant changes have been observed in the particles. Heating is being continued, time at temperature now being 1460 hr. The next examination is scheduled at 2000 hr.

The iron and nickel content (1000 ppm each) of calcined petroleum coke used in the fuel beds of the F-28 and F-29 irradiation experiments was considered a potential thermal compatibility problem. This concern was based on prior observation of Fe attack of SiC coatings heated in Fe contaminated crucibles. The combined Ni and Fe content in the coke loaded at 0.15 g coke per cubic centimeter of fuel bed would be about 3×10^{-4} g/cm³. Dispersed with the coke in fuel particles would be about 0.2 g Si/cm³ of bed in TRISO-II particles. The relatively small amount of Fe and Ni present would not create a thermal compatibility problem if thermal gradients in the bed do not cause local concentration of the Fe and Ni to much higher levels.

A high-temperature thermal gradient experiment (4464-35) was conducted to determine (1) whether thermal gradients could produce deleteriously high concentrations of Ni and Fe within a small F-capsule fuel bed, and (2) the beneficial effects of thermal purification of the calcined petroleum coke. Particles from six different TRISO-II batches were heated in holes 0.265 in. in diameter by 0.31 in. deep in H-327 graphite with a thermal gradient of about 0.03°C/μm at a temperature of about 1650°C to 1820°C. The particle batches and experimental conditions are shown in Table V.1. Particles were heated both with and without calcined petroleum coke additions of 0.15 g coke per cubic centimeter of bed. The coke was used either as-received or purified by heating at 1500°C or 2000°C for 1 hr in helium to remove Fe and Ni impurities. After heating for 10 and 30 hr under the conditions outlined, no indication of iron attack was evident in the radiographs of particles taken from the hot and cold side of each bed. From these results, it is probable that iron attack of the SiC coatings would not be significant in F-capsules with either the as-received or the purified coke. However, the use of a purified coke in the fuel beds of both the F-capsules and the reactor would maintain a conservatively low inventory of potentially reactive elements in the core.

ANALYTICAL STUDIES OF COATING DESIGN

Studies of the integrity of TRISO coated fuel are being conducted using stress-analysis models. The model used in these TRISO studies was developed by Kaae (Refs. 1 and 2). The Kaae model is a mathematical analysis of coating stresses induced by the buildup of fission-gas pressure, by anisotropic radiation-induced dimensional changes in the inner and outer pyrolytic carbon (PyC) layers, and by differential thermal expansion of the SiC and PyC layers. The analysis also accounts for the influence of radiation-induced creep in the PyC layers. Irradiation-induced dimensional change and creep parameters for high-temperature isotropic (HTI) PyC are those reported by Bokros, *et al.* (Ref. 3). The remainder of the input parameters are the same as those reported in an earlier quarterly report (GA-8879).

Stress design limits in each layer of the coating were established by calculating the stresses in samples irradiated in the F-capsules and then determining the correlations between calculated stresses and the observed performance of the samples.

Table V.1

COATED PARTICLES AND CONDITIONS FOR EXPERIMENT 4464-35^a

TRISO Particle Batch	Calcined Petroleum Coke ^b			Without Coke Addition
	As-received, 1000 ppm Fe 1000 ppm Ni	Heated 1 hr in Helium at 1500°C, 200 ppm Fe 200 ppm Ni	Heated 1 hr in Helium at 2000°C, 60 ppm Fe Ni not Determ.	
4423-3	X	X	X	X
4000-220	X	X	X	X
3592-35	X	X		
4263-91	X	X	X	X
4000-218	X			X
4423-53				X

^aParticles heated 30 hr in gradient of 0.03°C/μm between 1650° and 1820°C.

^bParticle size 61 to 125 μm; 0.15 g coke per cubic centimeter of bed.

Stress Calculations for Irradiated Coated Particles

Coating stresses were calculated for the TRISO coated particles used in the capsule FR-1 irradiation experiment. Because no LTI PyC dimensional change data were available for input to the model, all calculations were performed using HTI data. Furthermore, since the irradiation temperatures of the fuel rods and the batches of loose particles were different, stress calculations were performed at a number of temperatures for each type of particle used in FR-1. The models used in the calculation cannot predict stresses in bonded particles, so the following comments apply only to the loose batches of particles.

At 1200°C, calculated stresses in the outer PyC coating of both fertile samples (4000-711 and -712) exceeded the 20,000-psi design limit at a fluence of 3.6×10^{21} n/cm², which is lower than the actual fluences received (3.8 to 5.3×10^{21} n/cm²). Even at lower temperatures, calculated stresses in the outer PyC coating exceeded the design limit, but at fluences on the order of 6×10^{21} n/cm² (higher than the actual fluence). Calculated stresses in both fissile particle samples (3923-87 and -149) were generally acceptable out to 8×10^{21} n/cm² with one exception, again at 1200°C.

Coating stresses were calculated for the TRISO particles used in the capsule P22 full-exposure test. The majority of these particles had LTI coatings and were again treated as HTI. Silicon carbide layer stresses were found to be acceptable in all cases. High stresses were calculated in those

PyC coatings (both inner and outer) that were high-density coatings; these samples are predicted to fail. There are also a number of borderline cases in the calculations (outer layer stresses slightly exceeding 20,000 psi late in life). Coating stresses exceed the PyC tensile stress limit at a dose below the maximum, but the rate of stress buildup is very slow. These samples should prove of interest in further establishing the validity of the existing criteria for prediction of coating failure. After the postirradiation examination of the P22 capsule, a comparison will be made between predicted and observed performance.

Calculated Strength of SiC Coatings

The results of the capsule P20 irradiation experiment have allowed a direct calculation of the ultimate tensile strength of SiC coatings under irradiation. The only uncertainty that still exists is the actual fission-gas release from the kernel. The calculated fission-gas pressure was assumed to be the yield pressure; a lower fractional release implies a lower yield pressure and hence a lower calculated tensile strength of the SiC. For a fission-gas release of 100%, the calculated tensile strength was 10,000 psi; for a release of 15%, the value was 5500 psi. Nevertheless, the P20 irradiation experiment has confirmed that the current design basis tensile strength limit for the SiC layer of coated particle design calculations (5000 psi) is realistic, although conservative.

FISSION-PRODUCT RELEASE

Effect of Fuel Migration on Fission-product Release

A series of experiments was initiated (1) to evaluate the rate of fuel (uranium and thorium) migration through the coatings of intact fuel particles in the temperature range of 1100° to 1600°C and (2) to establish the relationship between fuel migration and the release of fission products from the particles.

In the initial studies, approximately 1000 TRISO coated particles are being used for each anneal. It is planned to measure the degree of fuel migration and/or the level of fission-gas release at intervals of 5, 10, 20, 30, 50, 100, 300, 500, 1000, and 2000 hr. The particles used in these tests were hand-picked (with the aid of radiographs) to eliminate defectives. Selected particles are used so that particle characteristics can be related to fuel migration and fission-gas release behavior.

Presently, samples are being annealed at 1100° and 1600°C, and the levels of fuel migration are being determined after a 200-hr interval.

Effect of Temperature on Fission-gas Release

Experimental work designed to determine the effect of temperature on fission-gas release values of TRISO coated fissile fuel particles has continued. As reported in an earlier quarterly report (GA-9440), fission-gas release values were obtained at room temperature and over the temperature range

1000° to 1700°C for unleached BISO and TRISO fuel particles. Additional data have now been obtained for temperatures between room temperature and 1000°C using thermocouples to measure the lower temperatures. Samples from TRISO fuel batch 4000-745 were used in the experiments. The same sample used for the earlier measurements reported in GA-9440 was also used in the current experiments.

The samples were irradiated for 1 hr at each temperature at a constant reactor power level and were analyzed as previously described. Fractional release values for the unleached particles have been found to be approximately constant from room temperature to about 800°C, as shown in Table V.2. Arrhenius plots of fractional release versus $1/T$ ($^{\circ}\text{K}^{-1}$) for $\text{Kr}^{85\text{m}}$ and Xe^{138} from room temperature up to the maximum experimental temperature of about 1700°C are shown in Figs. V.1 and V.2. The experimental (apparent) activation energy for these TRISO particles is the same as that obtained for a sample of BISO particles in a similar study within experimental error. This indicates a similar release mechanism for both types of particles. If the observed release values are corrected for the low-temperature recoil release (which makes a significant contribution to the release in the range of room temperature to 1400°C), then the activation energy for the steep part of the curve is about 20 kcal/mole.

This value is close to one-half of reported values of the activation energy for the diffusion of xenon in pyrolytic carbon (e.g., see Ref. 4), as would be predicted if fission-product diffusion in the pyrocarbon was the important process.* However, it should be noted that the observed temperature dependence of the fractional release is influenced by the distribution of uranium contamination on and within the outer pyrolytic carbon coating. The model suggested above assumes a uniform coating contamination.

The fission-gas release values from room temperature to about 800°C tend to remain reasonably constant. This is consistent with a fission recoil type release mechanism resulting from uranium contamination on the surface or in the outer PyC coating of the fuel particles. The changes in slopes of the curves shown in Figs. V.1 and V.2 at about 800°C indicate a change in controlling mechanism. It was postulated earlier (quarterly report GA-9440) that a second mechanism, possibly diffusion of the uranium contamination found in the outer pyrolytic carbon layer, was responsible for the increased release. However, if the increased releases at temperatures above 800°C were due to diffusion of uranium through the PyC coating, the releases would be expected to remain at the higher values observed in experiments at 1600° to 1800°C when the temperature is reduced, due to increased uranium activity near the surface of the particle. Since the data were obtained first at room temperature, then from 1000° to 1700°C, and finally from room temperature to 1000°C, using the identical sample of particles for all experiments, it is clear from Figs. V.1 and V.2 that uranium diffusion is not responsible for the increased release.

*In the case of fission-product release by diffusion from an ideal coated fuel particle (where the uranium is uniformly distributed in the outer coating), the release fraction is proportional to the square root of the diffusion coefficient; it follows that the activation energy for release is one-half of the activation energy for diffusion.

Table V.2
EFFECT OF TEMPERATURE ON THE FISSION-GAS RELEASE OF TRISO (4000-745)
FISSILE FUEL PARTICLES^a

Exp. No.	Temp (°C)	Reciprocal Temp $10^4/T$ (°K)	Fractional Release					
			Kr ^{85m}	Kr ⁸⁸	Kr ⁸⁷	Kr ⁸⁹	Xe ¹³⁸	Xe ¹³⁹
A	RT	33.3	3.2×10^{-7}	1.3×10^{-7}	9.2×10^{-8}	2.0×10^{-7}	1.6×10^{-7}	1.7×10^{-7}
B	410	14.65	4.1×10^{-7}	2.0×10^{-7}	1.8×10^{-7}	2.1×10^{-7}	1.5×10^{-7}	3.2×10^{-7}
C	710	10.15	4.3×10^{-7}	3.0×10^{-7}	2.1×10^{-7}	2.2×10^{-7}	2.1×10^{-7}	4.4×10^{-7}
D	1000	7.85	6.6×10^{-7}	5.2×10^{-7}	3.8×10^{-7}	3.6×10^{-7}	2.6×10^{-7}	6.3×10^{-7}
E	1060	7.50	1.9×10^{-6}	1.4×10^{-6}	8.5×10^{-7}	4.2×10^{-7}	3.6×10^{-7}	5.1×10^{-7}
F	1315	6.30	2.3×10^{-6}	2.2×10^{-6}	1.6×10^{-6}	6.9×10^{-7}	4.4×10^{-7}	7.5×10^{-7}
G	1470	5.75	3.7×10^{-6}	3.5×10^{-6}	3.0×10^{-6}	8.3×10^{-7}	7.6×10^{-7}	8.7×10^{-7}
H	1700	5.08	6.9×10^{-6}	5.8×10^{-6}	4.3×10^{-6}	1.9×10^{-6}	1.4×10^{-6}	1.3×10^{-6}

^aParticles were not leached.

Fig. V.1-- $\text{Kr}^{85\text{m}}$ release from TRISO coated (4000-745) unleached fissile fuel particles

Fig. V.2-- Xe^{138} release from TRISO coated (4000-745) unleached fissile fuel particles

Experiments are in progress using leached particles to determine what effect surface contamination has on the fission-gas release at temperatures up to 1700°C.

Effect of Graphite Crucible on Fission-gas Release Values

Fission-gas release measurements in the TRIGA King furnace are being obtained, as previously described in quarterly report GA-9440, using closed AGOT graphite crucibles to contain the fuel particles or uranium stearate standard. The fuel particles are irradiated at temperature to 1800°C at constant reactor power levels to determine fractional release values. The uranium stearate standard is also irradiated at constant reactor power levels, but only at room temperature, because of its low melting point (about 70°C). The fission-gases released from the fuel or standard sample are then collected and gamma counted. The stearate is assumed to be essentially 100% releasing for fission gases with half lives longer than 1 sec, allowing fractional release values for the isotopes of interest to be determined from the ratio of the counts per minute per fission of the sample to that of the standard.

The assumption that the uranium stearate standard is 100% releasing requires that essentially all of the recoiling fission fragments terminate in the stearate sample material and not in the surrounding graphite crucible. In addition, it must also be assumed that diffusion of the fission gases through the graphite container is not rate limiting. In view of the fact that fission-gas release from TRISO fuel particles exhibits a measurable temperature dependence only above 800°C and shows an apparent leveling off of the release at lower temperatures, as shown in Figs. V.1 and V.2, it appeared desirable to conduct experiments designed to determine whether the release at low temperatures is limited by the rate of diffusion out of the crucible.

Experiments were conducted using 50-mg samples of uranium stearate in (1) a standard closed graphite container and (2) a graphite container with 90% of the cylindrical wall removed and a porous paper extraction thimble inserted into the crucible to contain the stearate. The samples were each irradiated for 1 hr at room temperature and the same reactor power level. The values used to calculate fractional release values for the krypton and xenon isotopes of interest are given in Table V.3 for the standard and modified crucibles. The fact that the values are in reasonably good agreement indicates that diffusion of fission gases through the graphite crucible is not rate controlling. In addition, gross gamma counts for the krypton and xenon release from the modified and standard crucible are within the experimental error, thus indicating that recoil losses into the graphite crucible are minimal. It has been assumed and calculations have shown that the total recoil loss from the stearate standard into the standard graphite crucible is approximately 10% of the total recoiling atoms under the conditions of irradiation. It has been assumed that this small loss would not appreciably affect the results, and this assumption is indicated to be valid by the present experimental data.

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The assumption that the uranium stearate standard is 100% releasing requires that essentially all of the recoiling fission fragments terminate in the stearate sample material and not in the surrounding graphite crucible. In addition, it must also be assumed that diffusion of the fission gases through the graphite container is not rate limiting. In view of the fact that fission-gas release from TRISO fuel particles exhibits a measurable temperature dependence only above 800°C and shows an apparent leveling off of the release at lower temperatures, as shown in Figs. V.1 and V.2, it appeared desirable to conduct experiments designed to determine whether the release at low temperatures is limited by the rate of diffusion out of the crucible.

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Table V.3

COMPARISON OF VALUES USED TO CALCULATE FRACTIONAL RELEASE VALUES

Crucible Type	Reciprocal of CPM/Fission					
	Kr ^{85m}	Kr ⁸⁸	Kr ⁸⁷	Kr ⁸⁹	Xe ¹³⁸	Xe ¹³⁹
Standard	5.9x10 ⁵	7.1x10 ⁵	3.0x10 ⁵	3.0x10 ⁵	3.2x10 ⁵	2.5x10 ⁶
Modified	5.7x10 ⁵	6.1x10 ⁵	2.7x10 ⁵	2.7x10 ⁵	3.1x10 ⁵	2.7x10 ⁶

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