RECEIVED BY DTIE SEP 9 1970

MASTER

Gulf General Atomic
Incorporated

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10202

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

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
JUNE 30, 1970

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

July 31, 1970

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

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

LEGAL NOTICE

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

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

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

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

Gulf General Atomic

Incorporated

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

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10202

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

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
JUNE 30, 1970

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

LEGAL NOTICE

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

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

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

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

July 31, 1970

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

QUARTERLY REPORT SERIES

GA-6830-July, 1965, through September, 1965
GA-6950-October, 1965, through December, 1965
GA-7086-January, 1966, through March, 1966
GA-7314-April, 1966, through June, 1966
GA-7453-July, 1966, through September, 1966
GA-7634-October, 1966, through December, 1966
GA-7939-January, 1967, through March, 1967
GA-8038-April, 1967, through June, 1967
GA-8270-July, 1967, through September, 1967
GA-8420-October, 1967, through December, 1967
GA-8600-January, 1968, through March, 1968
GA-8725-April, 1968, through June, 1968
GA-8879-July, 1968, through September, 1968
GA-9130-October, 1968, through December, 1968
GA-9261-January, 1969, through March, 1969
GA-9440-April, 1969, through June, 1969
GA-9720-July, 1969, through September, 1969
GA-9875, October, 1969, through December, 1969
GA-10010, January, 1970, through March 31, 1970

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.

CONTENTS

INTRODUCTION	i
Part I	
TASK I. COMPONENT DEVELOPMENT	1
Helium Seal Development	1
Test Fixture	3
Test Program	5
Summary of Results	5
Auxiliary Equipment and Special Tools	5
Reserve Shutdown Vacuum Tool	7
Special Tools Test	7
TASK II. NUCLEAR ANALYSIS	8
Fuel Cycle Analysis	8
Status of Nuclear Design	8
Review of Data and Calculational Methods	8
Fuel Zoning and Power Shaping	15
Three-Dimensional Analysis of the Initial Core	15
Transverse Power Distribution	16
Axial Power Distribution	16
Core Multiplication	23
Analytical Studies	23
Initial Core Reactivity Adjustment	23
Neutron Flux Distribution Measurements	26
Differential Rod Worth Measurements	30
TASK III. FUEL DEVELOPMENT	32
Fuel Element Design	32
Mechanical Design	32
Fission-product Transport	33
Materials	36
Graphite Development	37
Graphite Evaluation	40
Control Materials	44
Irradiation Testing	44
Chemistry	101
Impurity Attack on Fuel Materials	101

Part II

TASK I.	FUEL TRANSFER MACHINE	103
TASK IV.	STEAM GENERATOR	107
	Sleeve and Wedge Test	107
	Fatigue Testing	109
	Chemical Cleaning Test of Feedwater Ringheader	111
	Module Air-Flow Test	111
	Tube Plugging Machine	111
TASK V.	COATED PARTICLE DEVELOPMENT	113
	Coated Particle and Fuel Rod Evaluation	113
	Distribution of Uranium and Thorium in Particle Coatings	113
	Effect of Fuel Migration on Fission-Product Release from TRISO Particles	114
	Thermal Stability of Coated Particles	118
	Thermal Gradient Experiments	118
	Fission-Product Release	125
	Fission-Gas Release from Mixtures of Fuel Particles and Coke Matrix Material	125
	Fission-gas Release from Fuel Rod	131
	Temperature Dependence of Fission-gas Release	135
	TRIGA Mark I Fission-gas Release Measurements	138

PART I

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

Task I
COMPONENT DEVELOPMENT

HELIUM SEAL DEVELOPMENT

The objective of this task is to develop information necessary to confirm the design of (1) bellows-type helium seals for the steam generator modules, helium circulators, and core support columns to accommodate the movements between these components and the core support floor and/or the lower floor, which form parts of the reactor internal structure, and (2) flexing and/or sliding helium seals between the lower floor structures and the PCRV liner while preventing significant leakage of helium past the seal.

A test was conducted for the purpose of aiding in the selection of a metallic seal to be used as the steam generator module primary closure seal. The seal has two purposes, the main function being to form a seal between the reactor primary coolant (helium) and the penetration interspace, which is normally pressurized with clean helium to 31 psig higher than the coolant pressure. The second function is to prevent excessive leakage of the primary coolant if there should be a loss of penetration interspace pressure because of a penetration secondary closure failure. The seal must therefore function with pressure applied to either side. Figure 1.1 shows the location of the seal in the steam generator module.

The thermal and pressure transients imposed on the steam generator primary closure seal during the life of the reactor may cause relative radial movements between the seal, the steam generator flange sealing surface, and the PCRV closure flange sealing surface. This test consisted of measuring the helium leakage rate across the seal while the effects of these transients were imposed on it.

Fig. 1.1. Section of steam generator module area

A silver-plated metallic O-ring and a gold-plated metallic K-seal were selected for testing. Relative leakage rates of the seals under simulated reactor operating conditions of temperature and pressure were to be prime factors in the final selection.

Test Fixture

Relative movement between the primary closure and its support flange under operating conditions will result from operating temperature differences between these items. Since it would have been very difficult to simulate these conditions, the test was designed to be performed with both flanges at the same temperature. The relative movements were obtained by fabricating the test flanges from steels having different coefficients of thermal expansion.

The test fixture, shown in Fig. 1.2, is so designed that the two seals under consideration (a K-seal and a self-energizing metal O-ring seal) can be tested together. The upper ring shown in the drawing represents the penetration flange and is made from SA-105 Grade II low-carbon steel. The lower flange represents the steam generator primary closure flange and is made from ASTM-A-357-58 (5% Cr, 0.5% Mo) steel plate. The two rings of bolts provide the necessary resistance to gas pressure and the force required to compress the seals.

The exterior horizontal surfaces of both rings are fitted with electrical heaters that were designed to provide a maximum test temperature of 550°F. The gaps between the flange mating surfaces and the inside and outside of the rings are shrouded to form cooling air passages. Both rings had passages drilled in them to permit helium pressure to be applied to both sides of either test seal.

The mating surfaces of the lower ring were treated with a solid-film lubricant. This treatment prevented the rings from welding together in the purified helium atmosphere. This treatment was kept clear of the seal grooves.

Fig. 1.2. Section of test fixture assembly and seals

TEST PROGRAM

Two types of seals were selected for testing, a gold-plated, K-type, metal flange seal for external pressure, and a self-energizing metal O-ring seal for internal pressure. Both seals had a 64 rms sealing surface finish with circumferential tooling. On the actual penetration closures, the seals used will be the external-pressure sealing type. During the test, the metal O-ring seal was reversed to permit testing of both seals together under the same conditions.

The temperature and pressure cycles of the test are presented in Table 1.1. The specified temperatures are calculated to give the correct relative radial movements between the flanges based on the thermal expansion coefficients of the two flange materials. Reverse pressure testing, simulating the effects of a secondary closure failure, was performed at the times indicated in Table 1.1. Recordings of seal leakage for each seal were taken at convenient times, the minimum number of leakage measurements for each set of test conditions being as specified in the table. The helium used for the testing was purified helium containing less than 10 ppm of oxidants.

Summary of Results

The O-ring seal performed satisfactorily and the average leak rate was less than $1/50$ the maximum of $10 \text{ cm}^3/\text{sec}$ required by the test specifications during the cold startup tests, less than $1/10$ the specified maximum during the hot startup tests, and less than $1/2$ the specified maximum after the load-change test. No test was performed on the K-seal because of an unsatisfactory gold plating on the seal and a misaligned hole in the test fixture. On the basis of these test results, the self-energizing O-ring seal was selected.

AUXILIARY EQUIPMENT AND SPECIAL TOOLS

The objective of this subtask is to establish the suitability of the manipulators, tools, and reactor viewing device for use in the plant by means of analysis and full-scale testing.

TABLE 1.1
STEAM GENERATOR PRIMARY CLOSURE SEAL TEST CYCLES

	Simulating Cold Startup	Simulating Hot Startup	Simulating 25% to 100% Load Cycle
<u>Press. and Temp. Testing</u>			
No. cycles	125	480	4000
Press. cycle, psig:			
Port A	0-674	674-674	690-690
Port B	0-705	705-705	721-721
Port C	0-705	705-705	721-721
Temp. cycle, °F	210-500	362-500	436-500
<u>Reverse Press. Testing</u>			
Frequency	After 60, 125 cycles	Every 100 cycles	Every 200 cycles
Press., psig:			
Port A	690	690	690
Port B, C	0	0	0
Min. No. Measurements	5	8	20

Reserve Shutdown Vacuum Tool

The reserve shutdown vacuum tool blower, including the related motor-alternator (M-A) set, have been manufactured and shop tested. The test results exceed the specification requirements. With a required helium flow of 800 cfm at 250°F, the blower data were found to be as follows:

	Specification Requirements	Test Results from Supplier
Blower operating time, min	4	5
Blower speed, rpm	24,000	18,700
M-A output characteristics:		
Voltage	440	440
Cycles per second	400	340
Phase	3	3

The blower and motor-alternator package will be sent to San Diego for equipment tests in conjunction with the reserve shutdown vacuum tool.

Special Tools Test

As a result of the special tools tests, the following tools have been redesigned and are being rebuilt: (1) the element pickup probe, (2) the special element pickup (stripped-hole) probe, (3) the cable cutoff tool, and (4) the element uprighting tool.

The manipulator wrist rotation potentiometer was found to be nonlinear and was returned to the supplier for repair. During the period when the manipulator was disassembled, lifting lugs were added to permit safe handling.

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

The analysis of the Reference 10 Nuclear Design, the design iteration that led to the preliminary specification of initial core fuel loadings, showed that the nominal initial core excess reactivity would have to be increased in order to meet the desired reactivity lifetime of the initial cycle, if all uncertainties occur in one adverse direction. Before proceeding with a corresponding loading adjustment, however, considerable time and effort has been spent in reviewing data and calculational methods that may affect the initial core loadings. In particular, the review of cross-section evaluation methods showed that improvements in the accuracy of the Dancoff factor and the grain shielding calculations have become necessary. In addition, more data became available on the actual carbon density of the fuel elements and on the impurity level in the graphite. Most of the available information has been reviewed and evaluated in preparation for the required final fuel loading adjustment.

Review of Data and Calculational Methods

Before proceeding with the final adjustment of the initial core loadings, basic data, as available from the current status of fuel element material production and testing, have been reviewed to evaluate their effects on core

excess reactivity and reactivity lifetime. In addition to the data review, some of the most important calculational methods, such as cross-section evaluations and multidimensional diffusion calculations, have been reexamined to detect possible systematic or random errors that might affect the initial core excess reactivity. Results of these investigations are discussed below.

Graphite Impurity. Delta-inhour (DIH) measurements have been performed on graphite samples to evaluate the effective impurity level in the fuel element graphite to be used for the reactor. Because the total impurity level in the fuel element graphite may have a significant effect on core reactivity and core lifetime, these measurements were undertaken as a part of Gulf General Atomic's production verification test program. Three graphite samples were prepared by the manufacturer of the fuel element graphite for DIH testing in the Hanford Test Reactor. Taken from production graphitization furnace runs, these samples should be representative of the material used for fabrication of the Fort St. Vrain fuel elements. Results of these measurements are discussed below.

The interpretation of DIH tests is discussed in Refs. 1, 2, and 3. For the analysis of the three test samples, the equations published by Nichols (Ref. 1) were used to determine the effective carbon capture cross section:

$$\sigma_a = 4.47 \pm 0.04 - (0.733 \text{ dih} \pm 0.03) \\ - 1.466 \left(\frac{1}{d} - \frac{1}{2.25} \right) (1.0 \pm 0.1) \text{ mb} ,$$

where σ_a = effective thermal (2200 m/sec) capture cross section per carbon atom of the graphite sample (which includes the effect of impurities),
dih = measured DIH value, and
d = carbon density of the sample, g/cm³.

The carbon density was calculated from available data on weight and nominal dimensions of the provided samples.

Measured DIH values for the three samples are shown in Table 2.1 together with the resulting effective thermal capture cross section per carbon atom.

The equivalent boron impurity level, which is also listed in Table 2.1, was calculated assuming a 3.4-mb thermal cross section for carbon and a 755-b thermal cross section for natural boron. All three graphite samples give an impurity level equivalent to about 0.9 ppm by weight of natural boron.

The impurity level in the fuel element graphite appears to be well below the maximum specified value of 5 ppm equivalent boron. Chemical analyses by the manufacturer for boron, iron, vanadium, and titanium, showed an average boron impurity level of about 0.5 ppm and a total boron equivalent of these four materials of less than 0.55 ppm. Chemical analyses by Gulf General Atomic for the same materials showed very similar results. Considering the fact that DIH measurements account for all impurities, it appears that the results of DIH tests are in good agreement with chemical analyses.

Table 2.1
MEASURED DIH VALUES FOR THREE CARBON SAMPLES

Sample	Sample Weight (kg)	Measured DIH	Effective Cross Section 2200 (mb)	B Equivalent of Impurities (ppm by weight)
1	23.126	+0.160	4.58±0.09	0.90
2	23.296	+0.196	4.137±0.09	0.88
3	23.353	+0.210	4.129±0.09	0.87

Graphite Density. One of the most important parameters affecting core fuel loading and reactivity lifetime is the average core graphite density. Most of the core graphite (about 80%) is contained in the fuel element structure; the remaining 20% is contributed by the particle coatings, the fuel rod matrix, and the graphite in the uranium and thorium carbide. Since a 1% change in the average core graphite density changes the core reactivity by about 0.0025 Δk , an accurate knowledge of the core graphite loading is fairly important.

Nineteen fuel elements that were almost completely machined have been weighed to evaluate the average carbon density in the elements. The machined elements were hexagonal in shape and contained all fuel and coolant holes

and the fuel handling holes, but did not include the machining of the chamfers on the top or bottom of the blocks or the chamfer on the handling hole. In addition, the dowels or holes for the dowels had not been machined when the element was weighed. These minor deviations from completely machined fuel elements, however, should not affect the validity of the determined carbon density. They rather facilitate the calculation of the graphite volume in the partly machined fuel elements.

The weights and block numbers of the weighed fuel elements are listed in Table 2.2, where weights are quoted to the nearest 1/2 lb. The linear average weight of the 19 elements is about 196.8 lb, the lightest element being about 194.5 lb and the heaviest about 199.5 lb.

TABLE 2.2
FUEL ELEMENT WEIGHTS

Block No.	Weight ^(a) (lb)	Block No.	Weight ^(a) (lb)
1004	194.5	0157	197
1387	196.5	0323	197.5
0707	196.5	0938	197.5
2441	196.5	1920	195.5
1459	197	0309	196.5
1718	198	0771	196.5
1378	196	2552	196
2319	199.5	2161	198.5
2489	197.5	1215	196
1420	196.5		

(a) Average weight, 196.82 lb.

The graphite density depends on both fuel element weight and fuel element volume. Detailed calculations show that the minimum possible fuel element volume, based on fuel block drawing tolerances with the largest diameter holes of the longest length in the smallest block, differs about 2% from the maximum volume, based on smallest diameter holes of shortest length in the largest, longest block. Thus, there is quite a variation in the expected fuel element volume.

Statistical calculations were used to determine the average fuel element volume assuming different statistical distributions for all linear dimensions within the specified tolerance band. Since it is not known which distribution is really applicable, the average fuel element was calculated for three assumptions:

1. Linear dimensions follow a normal distribution around the midpoint between the tolerances; all tolerances are 3σ limits.
2. All linear dimensions have a normal distribution; all tolerances are 5σ limits.
3. All dimensions within the specified tolerance band are equally probable, i.e., a uniform distribution that is constant over the tolerance interval.

The calculated average fuel element graphite volumes are as follows:

<u>Method</u>	<u>Volume (cm³)</u>
1	50822.3±108.0
2	50822.5±64.9
3	50821.7±188.3

It appears, that the average fuel element volumes for all three cases agree very well. The calculated carbon density in the fuel elements, using the average element weight and the average element volume, is calculated to be about 1.76 g/cm³. This compares to an average density of 1.78 g/cm³ that was measured in the unmachined graphite logs. It appears that there is a density gradient across raw graphite logs, with the maximum density at the outside. During machining, part of this high-density graphite is cut off, and a lower effective carbon density remains in the machined elements.

Dancoff Correction. The Dancoff correction factor is one of the important parameters that affects the effective thorium resonance integral in the fuel lattice. To date, the Dancoff factor has been calculated using a unit cell that comprises only one fuel rod with the underlying assumption that the reactor consists of an infinite array of unit cells. This assumption,

however, neglects the fact that the regular fuel-rod - coolant-hole pattern does not extend beyond the boundaries of a single fuel element and that an extra layer of graphite is present at the boundary between elements.

The Dancoff correction factor for the fuel rods in the reactor lattice has been recalculated taking into account the irregularities of the fuel rod array (boundary regions and central holes of the fuel element blocks). The evaluation procedure was based upon direct numerical integration of the moderator transmission probability over the chord length distribution according to the geometry of Fig. 2.1. An hexagonal block of moderator material contains an hexagonal array of coolant holes and fuel rods that are surrounded by an empty gap. The six holes at the block corners are assumed to be filled with moderator material. The hexagonal block may contain a central cylinder of reduced density where no holes (fuel or coolant) are allowed.

The Dancoff factor for the single fuel rods indicated in Fig. 2.1 are summarized in Table 2.3. For an infinite array of fuel rods and coolant holes, the Dancoff factor was calculated to be about 0.47. Note that the Dancoff factor for rods, which are not close to either the element surface or the fuel handling hole, approaches the value calculated for an infinite array. However, rods located at the fuel element boundaries show a considerably smaller Dancoff factor because of the additional moderator of the outside of an element, resulting in an average Dancoff factor for a fuel element of 0.42, about 0.05 lower than the previously computed value of 0.47. Further evaluations have shown that this reduction increases the thorium resonance integral by about 2.5%, which is equivalent to a 0.005 Δk reduction of the initial core excess reactivity.

Grain Shielding. A reevaluation of the grain shielding calculation procedure has shown that previous analyses overestimated the effect of grain shielding. The change in the effect thorium resonance integral reduces the core excess reactivity by about 0.002 Δk . Details of this investigation are summarized in Ref. 4.

$A = 0.667 \text{ IN.}$

$\text{PITCH} = 0.74 \text{ IN.}$

$\text{RADIUS (FUEL HOLE)} = 0.25 \text{ IN.}$

$\text{RADIUS (COOLANT HOLE)} = 0.3125 \text{ IN.}$

$\text{GAP (BETWEEN FUEL ROD}$
 $\text{AND FUEL HOLE)} = 0.005 \text{ IN.}$

Fig. 2.1. Model for Dancoff factor calculation

TABLE 2.3
DANCOFF FACTOR FOR FUEL RODS AS A FUNCTION OF LOCATION
(Fuel Element Graphite Density, 1.78 g/cm³)

Rod Location	Dancoff Factor ^(a)	Rod Location	Dancoff Factor ^(a)
1	0.263	11	0.456
2	0.339	12	0.457
3	0.341	13	0.462
4	0.399	14	0.462
5	0.421	15	0.462
6	0.424	16	0.459
7	0.427	17	0.456
8	0.457	18	0.462
9	0.457	19	0.414
10	0.457	20	0.360

(a) Average factor 0.420

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.

Three-Dimensional Analysis of the Initial Core

The entire nuclear design work for the initial core of the Fort St. Vrain reactor to date was based on two-dimensional diffusion theory calculations and single-channel synthesis methods. Recently, a three-dimensional diffusion theory code, GATT-2 (Ref. 8), became available which is particularly suitable for the hexagonal fuel element and refueling region structure of the core and thus is considered to be an excellent tool to evaluate the validity of previously used approximations. To compare a true three-dimensional analysis

to the single-channel synthesis methods, various calculational tools were used to compute core reactivity and power distribution for an approximate critical rod configuration in the hot clean initial core. All cases used the same loading, with 10 control rod pairs completely inserted and 13 pairs 50% inserted.

Effective few-group macroscopic cross sections for the three-dimensional calculations were determined by using the results of seven-group BUGTRI calculations for the top and bottom slice to flux and volume weight multi-group macroscopic cross sections.

Transverse Power Distribution

Figure 2.2 shows a comparison between refueling region peaking factors predicted by synthesized BUGTRI-GAMBLE/RZ (Refs. 5,6) calculations, four-group SCANAL (Ref. 7), and GATT-2 (Ref. 8) calculations. As can be seen from Fig. 2.2, the agreement is quite good. All three models agree within 4% of each other in the central part of the core. The maximum discrepancy is found in a few outer regions, where the agreement is only within 10%. These results clearly indicate that the single-channel synthesis methods are an excellent tool to obtain good and reliable estimates of region peaking factors with symmetrical rod configurations.

Axial Power Distribution

Figure 2.3 compares the average axial power distributions from GATT-2, SCANAL, and GAMBLE/RZ. Since an average axial power distribution is not easily obtained from GAMBLE/RZ, a one-dimensional FEVER (Ref. 9) calculation was run to obtain the average axial power distribution with seven-group constants. Core-average cross sections for FEVER were obtained using flux- and volume-weighted BUGTRI data. Group-dependent bucklings were obtained by using the average core leakage and flux from the BUGTRI neutron balance. As might be expected, SCANAL and BUGTRI/FEVER agree quite well. GATT-2 shows a slightly higher power fraction in the top slice and a slightly lower power fraction in the bottom slice. This increase in the top slice

Fig. 2.2. Comparison of refueling region peaking factors

Fig. 2.3. Average axial power distribution

power fraction may be associated with the better treatment of the neutron leakage by GATT-2 in the area of partially inserted rods. It is interesting that GAMBLE/RZ shows about 68% of the power in the top slice, whereas BUGTRI/FEVER shows only 65%. This would probably bring the GAMBLE/RZ and GATT-2 results in closer agreement than is implied by the BUGTRI/FEVER results. It should be pointed out, however, that the GATT-2, GAMBLE/RZ, and the SCANAL and FEVER treatments of the control rods in the upper reflector are not entirely consistent. SCANAL and FEVER represent all 37 control rod pairs smeared in the upper reflector. The RZ representation shows all 30 wt-% rods smeared in the inner part of the upper reflector and all 40 wt-% rods smeared in the outer part of the upper reflector. GATT-2 shows the presence of all control rods in the reflector (except for those that are inserted in the core) by smearing over each refueling region. It is not known what effect these different representations have on the axial power distributions.

One other point worth noting is the BUGTRI/FEVER power peaking in the vicinity of the axial reflectors. This is probably a result of the seven-group treatment. Notice that the four-group calculations in GATT-2 do not show this effect.

GATT-2 and SCANAL power distributions for several refueling regions are compared in Figs. 2.4, 2.5, and 2.6. In rings 1, 2, and 3 (see Fig. 2.4) both models are in fairly good agreement. In refueling regions 20 and 22 of ring 4, the SCANAL power distributions are also in good agreement with GATT-2. In refueling regions 23 and 24, however, SCANAL shows very poor agreement with GATT-2, since, by the single-channel synthesis method, the effect of the axial leakage is ignored in the vicinity of the partially inserted rod (region 24). Clearly, the single-channel synthesis method is not very successful in predicting detailed axial power distributions. This fact, however, is not really surprising; especially in the vicinity of partially rodded regions, the radial and axial distributions are not separable and the basic assumptions break down.

Fig. 2.4. Axial power distribution, regions 1 and 2

Fig. 2.5. Axial power distribution, regions 20 and 22

Fig. 2.6. Axial power distribution, regions 23 and 24

Core Multiplication

The eigenvalues calculated with the different codes for the approximate critical rod position are summarized in Table 2.4. The results of the different analyses agree extremely well and indicate that reactivity uncertainties due to core modeling should be less than $0.01 \Delta k$.

TABLE 2.4
COMPARISON OF CORE MULTIPLICATION
COMPUTED BY MEANS OF VARIOUS CODES^(a)

	SCANAL	BUGTRI (RZ-GAMBLE)	GATT
No. of neutron energy groups	4	7	4
Core multiplication, k_{eff} :			
Top core half, no transverse leakage	1.027	1.036	--
Bottom core half, no transverse leakage	0.990	0.992	--
Total core (synthesized RZ or 3D)	0.996	1.001	1.003

^(a) Initial core; 10 rod pairs full inserted, 13 rod pairs 50% inserted.

ANALYTICAL STUDIES

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

Initial Core Reactivity Adjustment

In the initial core, about $0.05 \Delta k$ of reactivity are controlled by burnable poison. Holes are drilled into the six corners of each fuel element for poison insertion. In the initial core, however, only two of the six locations in about two thirds of the fuel elements contain poison

rods. The remaining unoccupied holes may be used if the zero-power measurements indicate that a change of the initial core excess reactivity is necessary. The prime criterion for a reactivity reduction, i.e., addition of burnable poison, will be the shutdown margin with one rod pair withdrawn. Although care has been taken during the whole nuclear design to make a reactivity reduction more probable, the core reactivity could be increased, if necessary, by removing burnable poison from selected locations, in order to ensure an adequate reactivity lifetime.

The addition of burnable poison to the initial core will affect core reactivity and the initial power distribution as well as the time variations of the power distribution. Preliminary calculations have been performed to evaluate both the amount and location of burnable poison required to shim a given amount of reactivity and the effect of the poison on the core power distribution. Clearly, there are two conflicting points to be observed. In order to minimize the time to be spent for the actual shimming operation, it is desirable to add poison to as few regions as possible. On the other hand, if poison is added to only a few regions, the poison will be more concentrated, and the perturbations of the design power distribution will be greater.

A series of static initial core calculations where the poison concentration was altered in selected refueling regions is summarized in Table 2.5. Control rod patterns were chosen to obtain approximate critical conditions and to evaluate the poison effects on power distribution in the critical core. In all cases, the burnable poison was added to the refueling regions that are refueled at the first and second reload. These regions, in the reference design, comprise the one third of the core that does not contain poison rods. It was assumed that, for shimming purposes, reference poison rods, or rods with a higher poison concentration, would be inserted to obtain the desired reactivity change.

It appears that a reactivity change of up to about $0.02 \Delta k$ can be achieved by poisoning only the regions that are refueled early in the reactor

TABLE 2.5
REACTIVITY CHANGE VERSUS BURNABLE POISON ADDITION

Case	Number of Rod Groups Inserted	Regions to Which Poison Was Added	Relative ^(a) Density of Poison	Avg. Core Temp. (°K)	k_{eff}	Reactivity Worth of Added Poison (Δk)
1	9	---	--	300 ^(b)	0.9819	---
2	9	5,10,17,21,28,35	1.0	300	0.9761	0.0058
3	9	5,10,17	2.5	300	0.9761	0.0058
4	9	4,5,8,10,15,17 21,25,28,32,35,36	1.0	300	0.9676	0.0143
5	9	5,10,17,21,28,35	3.0	300	0.9724	0.0095
6	5	---	--	1100 ^(b)	0.9933	---
7	5	5,10,17,21,28,35	1.0	1100	0.9838	0.0095
8	5	5,10,17	2.5	1100	0.9820	0.0113
9	5	4,5,8,10,15,17 21,25,28,32,35,36	1.0	1100	0.9756	0.0177
10	5	5,10,17,21,28,35	3.0	1100	0.9767	0.0166

(a) A relative density of 1.0 indicates the same density as the reference poison density on initial core poison rods.

(b) Base case.

life. Since burnable poison is nothing but a parasitic neutron absorber, it is desirable to remove the poison as early as possible from the core; the fuel loading of reload segments can be adjusted correspondingly.

It is interesting that the worth of the burnable poison changes quite significantly when going from room to operating temperature. Although the increasing poison worth enhances the negative temperature coefficient, care has to be taken not to overpoison the cold core.

Power distributions for the cases at 1100°K (operating temperature) are shown in Fig. 2.7. In none of the cases does a drastic change in power distribution occur, although refueling region peaking factors generally increase as poison is added to selected regions. It should also be pointed out that the control rod configurations have not yet been optimized to reflect the changes in power distribution.

An unrodded depletion calculation was performed for Case 5, which has the highest poison concentration in adjusted regions and thus should give an upper limit for the power distribution variations to be expected. As can be seen in Fig. 2.8, region peaking factors are well below design limits. It is believed that acceptable results would be obtained even with inserted control rods. The heavy lumping of the burnable poison also prevents a too rapid depletion of the poison. Toward the end of the cycle, at least half of the reactivity shim worth is still present.

Neutron Flux Distribution Measurements

Flux distribution measurements might be taken for the following reasons:

1. To measure the gross radial power distribution (mainly refueling region peaking factors).
2. To measure the axial power distribution both with and without partially inserted rods.
3. To measure local power peaking at fuel loading discontinuities.

Fig. 2.7. Power distribution with poison additives (regions with rodded poison shown with heavy outlines)

0 DAYS,
 $k_{eff} = 1.039$

207 DAYS,
 $k_{eff} = 1.000$

10 DAYS,
 $k_{eff} = 1.0021$

310.5 DAYS,
 $k_{eff} = 1.000$

103.5 DAYS,
 $k_{eff} = 0.994$

Fig. 2.8. Refueling region peaking factor versus time for case 5, unrodded

4. To measure the effect of misaligned control rods on the core power distributions.
5. To detect misplaced fuel elements and loading errors.
6. To check calculational methods.

Since the Fort St. Vrain reactor is a commercial power plant, rather than a research reactor, only those tests are justified that give information necessary for the safe and reliable operation of the plant. This practical criterion eliminates many measurements that might be of interest only to the physicist or that would not affect the performance requirements of the reactor.

A check of the calculational methods and their agreement with the experiment would be useful and convenient but could not justify any flux measurement. The detection of misplaced fuel elements by flux measurements would be only fortuitous, if possible at all. Besides, as was shown in the FSAR, single misplaced fuel elements do not affect safe operation of the plant, and flux measurements are therefore not necessary. In addition, the determination of local power peaks alone could not justify neutron flux measurements.

The gross radial power distribution will be determined at a lower power level using flow and temperature measurements to determine region peaking factors. This measurement will also indicate gross loading errors and core power tilts. The feasibility of this measurement has been established, and no flux measurements would be required for any reason other than for detecting gross loading errors before the fuel is irradiated. In addition, even unexpected radial power tilts could be compensated for or corrected by modifying the control rod program. During power operation, some information will always be available about region peaking factors using temperature measurements at the region outlets and flow estimates from the orifice settings. It thus appears that flux measurements for determining only region peaking factors can be eliminated.

The axial core power distribution has a significant effect on fuel performance. Little information will be available during full-power operation, but it appears that good information about the axial flux (or power) distribution can be obtained during the zero-power test program. Since this is the first Gulf General Atomic reactor employing axial power shaping by fuel zoning, a check on the axial flux distribution appears to be well justified.

Knowledge of an unexpectedly tilted axial distribution could affect both the technical specifications and the operating procedures, which might be modified in order to obtain proper fuel performance. One could also modify the axial distribution of burnable poison to obtain an acceptable power shape. It appears that axial flux distribution measurements are highly desirable.

The axial flux distribution will be measured within the reserve shutdown hole of selected core regions with various control rod configurations. In-core nuclear detectors will be used, with drive units mounted on the control rod drive units, to permit the remote positioning of a detector at selected axial positions within a reserve shutdown hole. It is planned that at least six refueling regions, at various radial positions, will be traversed to evaluate the over-all axial power distribution. These measurements will be performed in the vicinity of completely and partially inserted control rod pairs.

Differential Rod Worth Measurements

This test will establish procedures for calibrating control rods by differential rod worth measurements using a reactivity computer and normal plant instrumentation. Verification of the reactivity measurement methods will be accomplished by comparing the instantaneous response of the reactivity computer as a result of step reactivity changes with the asymptotic period produced by that reactivity change. Reactivity measurements will be made during control rod withdrawals and subsequent insertions of approximately 9 cents.

REFERENCES

1. Nichols, P. F., "Absorption Cross Section of Graphite," Nucl. Sci. Eng. 7, 395-399 (1960).
2. The Industrial Graphite Engineering Handbook, National Carbon Company, Rev. June 1961.
3. Davis, M. U. and H. A. Fowler, "Hanford Test Pile," Hanford Atomic Products Operation, HW-43547, June 7, 1956.
4. Walti, P., "On the Evaluation of Grain Shielding Factors," Gulf General Atomic Report GAMD-10134, May 19, 1970.
5. Dorsey, J. P., et al., "BUG-2/BUGTRI, Two Dimensional Multigroup Burnup Codes for Rectangular and Hexagonal Geometry," USAEC Report GA-8272, Gulf General Atomic Incorporated, August 22, 1969.
6. Dorsey, J. P., et al., "GAMBLE-5, A Program for the Solution of the Multigroup Neutron Diffusion Equation in Two Dimensions with Arbitrary Group Scattering for the UNIVAC-1108 Computer," USAEC Report GA-8188, Gulf General Atomic Incorporated, December 4, 1967.
7. Traylor, R. C., et al., "SCANAL, A Single Channel Synthesis Depletion Code with Triangular Mesh in the Horizontal Plane," USAEC Report GA-9423, Gulf General Atomic Incorporated, July 1, 1969.
8. Kraetsch, H. and M. R. Wagner, "GATT, A Three-Dimensional Few-Group Neutron Diffusion Theory Program for a Hexagonal-Z Mesh," USAEC Report GA-8547, Gulf General Atomic Incorporated, January 1, 1969.
9. Todt, F. W., et al., "FEVER/M1, A One-Dimensional Depletion Program for Reactor Fuel Cycle Analysis," Gulf General Atomic Incorporated Report GA-9780, October 22, 1969.

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

Upper and Lower Reflection Revisions. Studies have been completed to determine the effect and feasibility of increasing the flow holes in the upper and lower reflectors from 5/8 to 3/4 in. in order to reduce core pressure drop. Results show that this is feasible with one layer of blocks in the top reflector and two layers in the bottom reflector with a resulting saving in pressure loss of about 0.6 lb/in.²

Flow holes near the dowels required special attention, because increasing them to 3/4 in. would have resulted in a probable breakthrough to the dowel hole. Keeping the flow holes at 5/8 in. would result in a temperature increase of about 40°F in the hot channel, but keeping these holes at 5/8 in. only at the dowel hole (for 1-1/2 in. in length) results in a 20°F increase in the hot channel.

Side Reflector Spacer Blocks. Results of work on the use of steel boronated pins in the side reflector to replace the boronated graphite have shown that at maximum power peaking values, the temperature drop across a 1/16 in. gap between the pin and its support hole in the spacer

block is on the order of 100°F. From a thermal standpoint, the design appears to be feasible, and work has been initiated to analyze the side reflector according to the latest configurations utilizing the TAC2D code.

Neutron Sources. No special design features are required to cool the neutron sources. Adequate cooling for the approximately 300-Ci sources is available with the standard fuel element design. The maximum source temperature in this case is found to be 1100°F, which is a short-term (several days) and has a margin of at least 60°F with respect to the closest materials limit for the sources considered.

Effect of Fuel Inhomogeneity in Fuel Rods. A two-dimensional TAC model of a PSC fuel column has been used to calculate the effect of fuel inhomogeneity of fissile and fertile fuel rods on temperatures and temperature gradients. Calculations have been made for various distributions of uranium loading within a 2-in.-long fuel rod. It has been assumed that one-half of the 2-in.-long rod contains U_1 mass of uranium and the other half contains U_2 mass of uranium. Initially, all fissions are in the U^{235} , so that the effect of inhomogeneity is greatest initially.

Preliminary results of the effect of fuel temperature on fuel inhomogeneity in all 2-in.-long fuel rods in the bottom 15 in. of the bottom fuel block are shown in Table 3.1.

Fission-product Transport

Primary Coolant Plateout Probe. As a result of construction experience, several improvements have been incorporated in the probe design. The first of these was a change from tungsten-inert-gas welding to electron-beam welding for the main O-ring seal grooves. This is expected to result in less distortion and consequently in improved sealing characteristics at lower torque values and in the elimination of several postwelding remachining operations. It was also thought that this may be the most satisfactory

TABLE 3.1
EFFECT OF FUEL ROD METAL LOADING INHOMOGENEITY
ON FUEL TEMPERATURE

Inhomogeneity $\frac{ U_1 - U_2 }{U_1 + U_2} \times 100$	Temperature ^(a) Increase (°F)
0	0
15	15
25	30
50	60
100	125

(a) For average power region, $P/\bar{P} = 1.0$.

assembly technique for the flow control units that are to be used on the diffusion tubes. An additional advantage is realized from the high degree of repeatability possible with this method of welding, which should result in greater probe uniformity. The test probe components were therefore modified to permit electron-beam welding.

Test specimens were made to develop the welding procedures to be used in the reassembly of the test probe and for inclusion in the plateout probe fabrication specification. All required probe weld configurations were successfully made, and examination of the test welds indicated the method to be far superior to tungsten-inert-gas welding for this application. The final assembly welding on the test probe by the electron-beam process was completed, and no problems were encountered. A weld test specimen indicated that all welds met the specifications. There was no indication of metal warpage and the O-ring fit was much better than on previous test probes.

To improve the diffusion-tube temperature profile and simplify construction, several changes were made in the internal design of the probe. A small activated charcoal trap has been included in the flow control unit to be placed in each diffusion tube, thus eliminating the need for the costly

hand-coiled charcoal trap previously located in the critical heat-transfer-head end of the probe cavity. This permits the graphite thermal sheath to be extended into the coolest region of the probe and for it to be reshaped for more efficient cooling of the diffusion tube array.

Preliminary flow-pressure drop tests were run on the new diffusion-tube flow control concept with the charcoal trap included in the prefilter upstream of the porous metal plug. Results, using a 5- μ m porous plug, indicated no appreciable change in operating characteristics with the addition of the charcoal bed. Porous filter material of the 5- and 10- μ m types in full 0.250-in. thickness is currently on order and additional tests will be made to develop fabrication techniques.

The latch plate of the probe locking mechanism has been reshaped to provide a shorter bending arm that, together with the increase in the material thickness to 20 gage, will significantly increase the reverse-torque capabilities of this mechanism. A new heavy-duty latch plate was heat treated in the 5-ft autoclave using a specially designed form to prevent warping. The finished plate showed excellent spring characteristics. The latch plate now requires a force of 12 lb to actuate and will withstand 100 in.-lb reverse torque without damage. All assembly on the test probe, which contains all the modifications described above, is now complete, and it is ready for the temperature profile and long-term environmental tests.

The new autoclave test stand was received and the autoclave mounted in it to check the mechanics of the system. Internal outfitting of the autoclave will be completed shortly.

The rough draft of the probe fabrication specification has been completed and editing is in progress. The plug specification will be a modified version of the probe specification with unrelated sections deleted.

Review and comments were made on the test specification for the on-site demonstration of the handling tool system. No serious difficulties were found in relation to the plateout probe.

MATERIALS

The continuing objectives of this subtask have been to:

1. Determine the characteristics of beds of 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 including 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 fuel elements by means of measurements of the 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 post-irradiation examination.
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 fuel elements by means of a series of (1) short-term

tests on coated fuel particles to select a coated fuel particle (or particles) having over its design life a high degree of structural integrity, 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.

Graphite Development

The standard containment for irradiation testing HTGR fuels in capsules F-27 and F-28 consisted basically of a graphite crucible surrounding the test pieces with the assembly doubly contained in a metal sheath. A controlled helium gas flow maintained the required temperatures, which were continuously monitored with thermocouples in the wall of the crucible, in the center of the fuel, and on the outer wall of the primary metal cladding. In similar positions to the thermocouples around the periphery of some graphite crucibles in the capsules were a number of "piggyback" holes which gave ideal temperature monitored locations for irradiating graphite test specimens.

One cell in capsule F-28 irradiated for four cycles in GETR to a total measured fast neutron fluence of $5.6 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) between temperatures of 750° to 800°C contained 16 graphite samples located in a uniformly distributed axial flux position. Capsule F-27 had three cells each containing 16 PSC graphite piggyback specimens irradiated within flux gradients which gave mean fast neutron fluences of 1.73, 2.6, and $3.4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) between temperatures of 670° and 800°C . In addition to the piggyback samples, the crucibles themselves also provided irradiation data on the dimensional changes in grade H-327 graphite test pieces.

The graphite piggyback samples selected for the above irradiation tests were machined from a log of graphite manufactured at the mid-production stage of the total quantity of fuel element blocks required for the core. It had been previously established that the largest differences in the mechanical properties in fuel block graphite are at the end of the block across the section and across the section in the mid-length of the original log. The

positions are described here as (1) end edge, (2) end center, (3) mid-length edge, and (4) mid-length center. The graphite samples for the above irradiation tests were selected from these positions in the log.

The dimensional changes in graphite samples taken from the various positions in a log have a similar trend in contraction rate to that previously established at fast fluences up to $3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) (see Fig. 3.1). However, beyond this dose level there is a greater divergence in dimensional change between different log positions oriented either parallel or perpendicular to extrusion. Normally the spread in data is not wholly apparent because the dimensional change plots are generally of different graphite samples at different flux levels and probably different thermal histories which were thought to account for most of the spread in data. However, some divergence of the dimensional change plots for various log positions is apparent at fluences approaching the turnaround point in the perpendicular direction, and this anomaly is substantiated by the single dose dimensional changes of the one cell in F-28 at approximately median fast-neutron dose levels. This cell was located in a uniform dose region of the GETR core and thus required no fast-neutron dose averaging to establish the dimensional change plots.

There are still insufficient data to show any correlation between any of the variations in thermo-physical properties of the various positions of graphite fuel blocks with the order of greater dimensional change. Preliminary indications are that where the ultimate tensile strength anisotropy of the graphite is highest, the differences in dimensional change between the two extrusion directions are greatest. There is an approximately linear relationship between these two parameters at fast fluences beyond $4 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$). The largest dimensional strain differences in an H-327 graphite block are along the center of the log for the parallel extrusion, and at approximate median fluences the contraction is 2.25 times greater in the mid-length position to that at the end. The biggest disparity in dimensional changes for the perpendicular orientation is across the section where the difference is again 2.25 times greater in the center of the log than at the outer section.

Fig. 3.1. Dimensional changes in fuel block graphites, effects of log location at irradiation temperatures of 670° to 800°C

Graphite Evaluation

Conventional electrode-grade graphite will be used in the Fort St. Vrain core to fill the areas within the core barrel surrounding the hexagonal side reflector elements. These large side reflector blocks have a minimum specified graphitization temperature of 2600°C, and an ultimate tensile strength of 1000 psi in the direction parallel to extrusion and 700 psi in the perpendicular direction. The compressive strength is specified to be not less than 4500 psi in the direction parallel to extrusion and 5000 psi in the perpendicular direction. Machining of the large side reflector blocks will be from a rectangular sectioned extrusion having a 24 by 30 in. cross section and minimum length of 47 in.

In place, the permanent side reflector elements will be arranged with the extrusion direction tangential to the reactor core. Internal strains and stresses in the structure of the graphite will be low owing to the absence of large thermal gradients and fast-neutron exposures. To check the effects of unforeseen bearing loads in the radial orientation of the block assembly, tensile and compressive tests were conducted at locations around the periphery of a typical, machined, large side reflector block (see Fig. 3.2). The test samples were taken from locations in the block at which there were sharp changes in section and at which points some local strain may be induced from the possible misfit of any two contiguous faces.

Mechanical testing was carried out on an Instron TTC-M-1 using a B-3M Hi Mag extensometer for tensile strain measurements and PD-1-M Hi Mag strain measuring instrument for the compressive load tests. The results are given in Tables 3.2 and 3.3. Due to the anomalies in establishing the elastic constants of graphite in a uniaxial test, the chord modulus between stresses of 250 and 500 psi is given for a comparative property of the graphite test sections. The elastic constant derived in a compressive test serves mostly as a means of comparison between the graphite samples. Two test lengths were used and the effects are seen in the enhanced strain at fracture of the shorter specimen length. This effect is thought due to

Fig. 3.2. Permanent reflector (graphite) stockpart relation in original extruded log and location of mechanical test samples

TABLE 3.2
MEASURED TENSILE PROPERTIES OF PERMANENT SIDE REFLECTOR GRAPHITE

Property	Orientation of Log Sample and Related Reactor Orientation									
	Perpendicular to Extrusion, Core Radial Direction					Perpendicular to Extrusion, Core Axial Direction				Parallel to Extrusion, Core Tangential Dir.
	Pos. 1	Pos. 2	Pos. 3	Pos. 4	Pos. 5	Pos. 1	Pos. 2	Pos. 3	Pos. 4	Pos. 5
Ultimate tensile strength, psi:										
Max.	2140	2260	2030	1780	2130	1550	1300	1330	1550	2200
Min.	2020	1830	1750	1760	2000	1380	1160	1100	1390	1640
Mean	2090	2046	1871	1766	2053	1483	1230	1223	1466	1973
Strain at fracture, 10^{-3} in./in.:										
Max.	2.95	3.4	2.8	2.4	3.2	2.6	2.85	2.9	2.95	2.25
Min.	2.65	2.8	2.2	2.0	2.8	1.8	2.2	2.4	2.8	1.8
Mean	2.82	3.0	2.35	2.1	3.65	2.3	2.46	2.66	2.9	1.93
Secant modulus, 10^6 psi:										
Max.	0.79	0.685	0.816	0.89	0.713	0.765	0.536	0.46	0.525	1.13
Min.	0.73	0.603	0.725	0.73	0.674	0.595	0.457	0.46	0.495	0.91
Mean	0.765	0.647	0.78	0.81	0.70	0.656	0.506	0.46	0.503	1.03
Chord modulus, 10^6 psi (250-500 psi):										
Max.	1.32	0.892	1.25	1.5	1.19	1.0	0.77	0.77	0.832	1.66
Min.	1.00	0.78	1.14	1.04	0.83	0.71	0.625	0.47	0.605	1.0
Mean	1.17	0.835	1.21	1.35	1.02	0.85	0.70	0.60	0.718	1.27

TABLE 3.3
MEASURED COMPRESSIVE STRENGTH PROPERTIES OF PERMANENT SIDE REFLECTOR GRAPHITE

Property	Orientation of Log Sample and Related Reactor Orientation									
	Perpendicular to Extrusion, Core Radial Direction					Perpendicular to Extrusion, Core Axial Direction				Parallel to Extrusion, Core Tangential Direction
	Pos. 1	Pos. 2	Pos. 3	Pos. 4	Pos. 5	Pos. 1	Pos. 2	Pos. 3	Pos. 4	Pos. 5
Compressive strength, psi:										
Nom. length 1 in.										
Max.	6110	5980	6110	6030	5790	5950	5480	5640	6130	5450
Min.	5820	5560	6000	6000	5870	5950	5200	5380	6000	5050
Mean	5965	5770	6055	6015	5830	5950	5340	5510	6065	5250
Nom. Length 1/2 in.										
Max.	6180	5820	6160	6110	5970	6000	5460	5950	5950	5605
Min.	6110	5770	6050	6060	5810	5970	5440	5820	5900	5290
Mean	6140	5795	6105	6085	5890	5985	5550	5885	5925	5445
Strain at fracture, 10 ⁻³ in.:										
Nom. length 1 in.										
Max.	23	24	26.5	26	22	34	33	31.5	37	18
Min.	19.5	23	16	24	15	32	29	27.5	35	18
Mean	22.25	23.5	21	25	18.5	33	31	29.3	36	18
Nom. length 1/2 in.										
Max.	68	30	38	33	29	44	42	40	35	46
Min.	44	26	32	26	26	38	36	40	35	42
Mean	56	28	35	29.5	27.5	41	39	40	35	44
Chord modulus, 10 ⁶ psi (250-500 psi):										
Nom. length 1 in.										
Max.	3.85	2.96	3.12	3.58	3.5	3.6	3.35	4.5	3.85	3.58
Min.	3.85	2.5	2.78	3.3	2.78	3.34	3.35	1.94	2.8	3.35
Mean	3.85	2.73	2.96	3.44	3.15	3.47	3.35	3.22	3.3	3.47
Nom. length 1/2 in.										
Max.	3.58	3.85	4.17	3.85	4.17	4.15	5.0	4.15	5.0	3.75
Min.	3.34	3.85	3.33	2.5	3.32	3.56	3.85	3.58	3.85	3.3
Mean	3.56	3.85	3.75	3.17	3.75	3.86	4.41	3.86	4.42	3.53

the relatively greater end support of the smaller volume test piece (from the onset of fracture). It is also a possible reason for an apparent 1% to 5% increase in ultimate compressive strength. However, the tests provide a basis for comparing the properties of the conventional electrode graphite across the section of the block, and the results show:

1. That the tensile strengths of a typical, permanent, side reflector block are similar in core radial and tangential directions and are 1.75 to 2 times larger than specified. The strength in the weakest plane (i.e., core axial direction) is a minimum of 1.75 times the minimum specified value.
2. That the variation in tensile strength across the axial and radial direction of a log is less than 15%.
3. That the compressive strength in a typical large side reflector block is a maximum in the core radial plane of the graphite and is 20% greater than specified. The compressive strength in the core axial plane is normally 5% to 10% lower in the axial direction of the core, and the tangential compressive strength is the weakest orientation of the graphite block and is approximately 10% to 25% higher than specified.

Control Materials

Studies have been made in preparation for the production of burnable poison rods. The burnable poison will be fabricated as 0.410-in.-diameter by 26-in.-long boronated graphite rods in accordance with specification 396-FO-9M. Rods of three different boron contents will be fabricated to produce boron densities of 0.02, 0.04, and 0.1 g B/cm³. The two latter densities (0.04 and 0.1 g B/cm³) are tentative and will not be confirmed until fuel loading characteristics are more precisely defined.

Irradiation Testing

Irradiation experiments are being conducted to proof test TRISO and BISO coated particles, fuel rods, and blended beds 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 Fort St. Vrain service conditions.

During this report period, two half-exposure capsules (F-25 and F-28) completed irradiation and were undergoing postirradiation testing. Capsules F-25 and F-28 are proof tests of production TRISO coated particles and fuel rods. Postirradiation examination of a third capsule (F-27), a higher-exposure test of TRISO particles, was essentially completed. The metallographic examination of the capsule P22 samples, whose irradiation was completed during the previous quarter (see quarterly report GA-10010), will be made following the completion of the postirradiation examination of capsule F-27.

Capsule F-25. Capsule F-25, a half-exposure (4×10^{21} n/cm²) proof test of nine TRISO coated particle and 15 fuel rod samples, completed its scheduled irradiation in the ETR during the quarter. The postirradiation examination of the fuel samples is currently under way at the Gulf General Atomic hot-cell facility.

All the coated particle and most fuel rod samples were made in production equipment. This capsule is a backup proof test (to capsule F-28) to control fuel specifications for the first half of the core. The fuel achieved an estimated burnup of 19% FIMA and an estimated fast fluence of 5×10^{21} n/cm².

Capsule F-26. Capsule F-26 is a full-exposure (8×10^{21} n/cm²) proof test in the ETR of eleven fuel rod, four blended bed, 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 capsule F-29) to control fuel specification for the second half of the core.

The F-26 capsule is beginning its fourth cycle in the I-13 core position of the ETR. The capsule is operating satisfactorily with the monitored fuel rod, blended bed, and coated particle thimble samples close to the design temperature of 1125°C. To date the capsule has received an estimated fast

fluence of 3.8×10^{21} n/cm², with an estimated fuel burnup of 17% FIMA. Design exposures are 8×10^{21} n/cm² and 23% FIMA, respectively. The capsule is expected to complete irradiation in early 1971.

Capsule F-27. The postirradiation examination of capsule F-27 was essentially completed during this reporting period. The capsule contained the first test to half exposure of reference Fort St. Vrain fuel samples. The peak fast fluence of 6.0×10^{21} n/cm² ($E > 0.18$ MeV) reported in the previous quarterly report (GA-10010) has been revised downward to 5.6×10^{21} n/cm² based on interpretation of the capsule dosimetry orientation during irradiation. (The dosimeter wires were located near the outside of the capsule at a radial position that should have experienced a higher fast fluence than the samples in the center of the capsule due to a shorter mean coolant water path with resultant lower neutron thermalization. The F-28 dosimetry, discussed later, was located in a radial position 180° opposite that of F-27, which should have received a lower fast fluence than the samples, and the sample fluences were revised upward accordingly.) The revised dosimetry values for the individual sample are given in Tables 3.4 through 3.6.

Postirradiation results obtained this quarter for fuel rod, particle, and blended bed samples are presented in Tables 3.4 through 3.6, respectively. (Tables describing the individual sample properties were given in earlier quarterly reports GA-9875 and GA-10010.) Metallographic results that were not presented in previous quarterly report GA-10010 are given in Figs. 3.3 through 3.22.

An up-to-date summary of the F-27 capsule experiment results is given below:

1. The Fort St. Vrain prototype pitch-bonded fuel rods were generally in very good condition (see Table 3.4 and GA-10010). Visual inspection showed the rods with isotropic filler to be in slightly better condition than the rods having natural-flake filler. A high-temperature preirradiation anneal ($\geq 1800^\circ\text{C}$) of the rods appeared

TABLE 3.4
FUEL RODS IRRADIATED IN CAPSULE F-27

Rod No.	Type	Matrix			Coated Particles ^(b)	Irradiation			Postirradiation Examination							Remarks
		Binder	Filler ^(a)	Heat Treat. (°C)		Temp (°C)	Fissile Burnup (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	Appearance ^(b)	Dimensional Changes		Metallographic Exam (% Failure)				
										Diam. (%)	Length (%)	Fissile		Fertile		
												PyC	SiC	PyC	SiC	
6A	Hot injection	Pitch	Iso	2000	TRISO-PP	700	12	3.8	Large crack	-0.04 ^(d)	-0.3 ^(d)	0	0	0	0	Rod slightly cracked as fabricated
5E	Hot injection	Pitch	Iso	1800	TRISO-MG	1165	14	4.7	Very good	-1.7	-0.8	0	0	0	0	
4E	Hot injection	Pitch	Iso	2000	TRISO-PP	1135	17	5.3	Very good	-1.1	-0.9	0	0	0	0	
6B	Hot injection	Pitch	NF	1800	TRISO-PL	700	11	3.5	Some loose CPs	-1.4 ^(d)	-1.4 ^(d)	0	0	0.6	0	Portions of rod inadequately injected
4D	Hot injection	Pitch	NF	1800	TRISO-PL	1135	17	5.3	Debonded; CP failure	--	--	0 ^(d)	0 ^(d)	75	15	Bad fertile lot
4B ^(c)	Hot injection	Pitch	NF	2000	TRISO-PP	1325	17	5.3	Good	-1.4	-1.0 to -1.7	0	0	0	0	
5A ^(c)	Hot injection	Pitch	NF	1000	TRISO-PP	1255	16	5.2	Dark; cracks	-1.6	-2.2	0	0	0.5	0.5	Low-temp heat treat
1B	Diffusion bonding	Si-SiC	--	1850	TRISO-PL	1245	15	4.1	Debonded	--	--	3.5	26	0.5	0.5	Matrix dis-integrated
6F	Diffusion bonding	Si-SiC	--	1850	TRISO-PL	735	8	2.4	Debonded	--	--	0.4	0.4	0.8	0.8	Matrix dis-integrated

(a) Iso = isotropic, NF = natural flake.

(b) All coated particles fabricated in production equipment in 5-in. coaters. PP = propane, PL = propylene, MG = mixed gas, CP = coated particle.

(c) Produced in production equipment; all others produced in laboratory equipment.

(d) See remarks column.

TABLE 3.5
COATED PARTICLES IRRADIATED IN CAPSULE F-27^(a)

Lot No.	Coating			Kernel Type	Irradiation			Postirradiation Examination (% Failure)				Remarks
	Type	Density (g/cm ³)			Temp (°C)	Burnup (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	Macro Exam		Met Exam		
		Disk	CP ^(b)					PyC	SiC	PyC	SiC	
181	TRISO-PP	1.54	1.59	Fissile-B	1210	17	5.6	0.5	0.1	0.9	0.2	Lot 183 without outer PyC
182	TRISO-PP	1.83	1.88	Fissile-B	1220	17	5.6	0	0	0	0	
183	TRISO-PL	1.78	1.84	Fissile-B	1135	17	5.6	0.9	0	1.8	0	
190	Bare SiC	--	--	Fissile-B	1130	17	5.6	--	0.1	--	0.4	
189	BISO-PL	1.87	--	Fissile-B	1170	17	5.6	0	--	0	--	
185	TRISO-PP	1.75	1.86	Fertile-B	1225	1.5	5.3	11	7	8	6	Lot 194 without outer PyC
194	TRISO-PL	1.74	1.67	Fertile-B	1170	1.5	5.3	~50	~40	>25	20	
195	Bare SiC	--	--	Fertile-B	1155	1.5	5.3	--	~10	--	3.5	
174	BISO-MG	1.90	--	Low Th ^(c)	1210	12	5.6	0	--	0	--	
173	BISO-MG	1.86	--	High Th ^(c)	1210	4	5.6					

(a) All coated particles prepared in production equipment in 5-in. coaters.

(b) Density determined on outer PyC removed from coated particle (CP).

(c) Peach Bottom Core 2 prototype fuel.

TABLE 3.6
BLENDED BEDS IRRADIATED IN CAPSULE F-27

Bed No.	Bed Design		Coated Particles (a)	Irradiation			Postirradiation Examination								Remarks
				Temp (°C)	Fissile Burnup (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	Type	Macro Exam (% Fail)		Met Exam (% Fail)		Dimensional Changes (b)		Volume Change (e) (%)	
	Filler	Restraint						Diam (c) (%)	Length (d) (%)						
6G	Charcoal	G Felt	TRISO-PL	700	7	1.8	Fiss Fert	0.1 0.1	0 0	0.7 1.6	0 0	0	N.D.	N.D.	
6E	Charcoal	G Felt	TRISO-PP	700	9	2.7	Fiss Fert	(f)	(f)	(f)	(f)	N.D.	N.D.	N.D.	
6C	Charcoal	G Felt	TRISO-PL	700	11	3.2	Fiss Fert	0 0.1	0 0	0 0.5	0 0	0	N.D.	N.D.	
1A	Charcoal	G Felt	BISO-PL	1245	14	3.6	Fiss Fert	0 4.5	-- --	0 11	-- --	0	12-20	12-20	
1C	Charcoal	G Felt	TRISO-PL	1185	15	4.5	Fiss Fert	3 7	3 4.5	1.5 6	12 12	0	7-16	7-16	Particles stuck into large lump; high BAF fissile. Bad fertile lot (-194T) Bad fertile lot (-194T) Bad fertile lot (-194T)
1E	Charcoal	G Felt	TRISO-PL	1125	16	4.8	Fiss Fert	0.1 6.5	0 4	0 2	0 0	0	6-14	6-14	
4C	Charcoal	Temp	TRISO-PL	1250	17	5.3	Fiss Fert	2.5 50	1 50	0 60	4 27	0	0-4	0-4	
4A	Charcoal	G Felt	TRISO-PL	1225	17	5.3	Fiss Fert	4.5 50	2 50	0 25	0 20	0	2-12	2-12	
1D	Charcoal	G Felt	TRISO-PP	1185	15	4.7	Fiss Fert	8.5 7.5	8.5 6.5	6 3	16 3	0	7-14	7-14	

(a) All coated particles prepared in production equipment in 5-in. coaters.

(b) N.D. = not determined.

(c) Determined from measured ID of crucible.

(d) Determined from neutron radiography.

(e) Calculated from (c) and (d).

(f) In progress.

M340-446

(a)

75X

M340-453

(b)

200X

Fig. 3.3. (a) Representative photomicrograph of fuel particles from fuel rod No. 1B. Rod was Si-SiC diffusion bonded and debonded during irradiation: (b) Photomicrograph of fissile particle (batch 4632-43) showing kernel migration and reaction with the SiC coating (amoeba effect)

M340-461

60X

Fig. 3.4. Photomicrograph of fuel particles from fuel rod No. 6F. Rod was Si-SiC diffusion bonded and debonded during irradiation

M31253-1

(a)

200X

M340-362

(b)

200X

Fig. 3.5. Batch 4000-181 TRISO coated fissile particles (propane outer isotropic) (a) before and (b) after irradiation at 1210°C to 5.6×10^{21} n/cm² and 17% FIMA

M31254-8

(a)

200X

M340-353

(b)

200X

Fig. 3.6. Batch 4000-182 TRISO coated fissile particles (propane outer isotropic) (a) before and (b) after irradiation at 1220°C to 5.6×10^{21} n/cm² and 17% FIMA

M31255-6

(a)

150X

M340-34

(b)

150X

Fig. 3.7. Batch 4000-183 TRISO coated fissile particles (propylene outer isotropic) (a) before and (b) after irradiation at 1135°C to 5.6×10^{21} n/cm² and 17% FIMA

M31245-6

(a)

125X

M340-479

(b)

150X

Fig. 3.8. Batch 4000-190 TRISO-III coated fissile particles
(a) before and (b) after irradiation at 1130°C
to 5.6×10^{21} n/cm² and 17% FIMA

M31533-5

(a)

125X

M340-404

(b)

125X

Fig. 3.9. Batch 4000-189E BISO coated fissile particles
(a) before and (b) after irradiation at 1170°C
to 5.6×10^{21} n/cm² and 17% FIMA

M31531-7

(a)

100X

M340-463

(b)

125X

Fig. 3.10. Batch 4000-185 TRISO coated fertile particles (propane outer isotropic) (a) before and (b) after irradiation at 1225°C to 5.3×10^{21} n/cm² and ~2% FIMA

M32494-2

(a)

75X

M340-488

(b)

75X

Fig. 3.11. Batch 4000-194 TRISO coated fertile particles (propylene outer isotropic) (a) before and (b) after irradiation at 1170°C to 5.3×10^{21} n/cm^2 and $\sim 2\%$ FIMA

M32494-1

(a)

75X

M340-487

(b)

75X

Fig. 3.12. TRISO coated fertile particle batch 4000-194 (see Fig. 3.09) under polarized light (a) before and (b) after irradiation. Note intense Maltese cross in outer isotropic after irradiation, indicating significant reorientation occurred during the irradiation

M31246-9

(a)

90X

M340-183

(b)

75X

Fig. 3.13. TRISO-III coated fertile particle batch 4000-195 (a) before and (b) after irradiation at 1155°C to 5.3×10^{21} n/cm² and 2% FIMA

M30461-1

(a)

M30460-7

(b)

M340-346

(c)

Fig. 3.14. BISO coated particle batches (a) 4000-173T, high Th, and (b) 4000-174E, low Th, before irradiation. Photo (c) shows particles after irradiation at 1210°C to 5.6×10^{21} n/cm² at burnups of 4% and 12% FIMA

M31535-5

(a)

M31534-7

(b)

M340-320

(c)

Fig. 3.15. TRISO coated particles of blended bed 6G. (a) Preirradiation fertile batch 4000-193, (b) preirradiation fissile batch 4000-192, (c) postirradiation

M340-192

75X

Fig. 3.16. TRISO coated particles of blended bed 6C after irradiation. Fissile particle from batch 4000-183 and fertile particle from batch 4000-194

M340-223

(a)

M340-218

(b)

M340-221

(c)

Fig. 3.17. Bonded particle from center of bed 6C: (a) bonded lump; (b) and (c) show bonding material tentatively identified as pyrolytic carbon

M340-270

(a)

100X

M340-267

(b)

125X

Fig. 3.18. TRISO coated particles of blended bed 1C after irradiation: (a) Fissile particle from batch 4000-192 and fertile particle from batch 4000-193; (b) fertile particle exhibiting kernel migration (amoeba effect) and reaction between kernel and the silicon carbide layer

S340-71

(a)

M340-232

(b)

M340-175

(c)

Fig. 3.19. TRISO coated particles from blended bed 1E: (a) Representative photomicrograph of fissile (4000-183) and fertile (4000-194) particles after irradiation; (b) fissile particle with kernel migration (amoeba effect); and (c) fertile particle with fractured PyC layers

M340-106

(a)

M340-107

(b)

M340-103

(c)

Fig. 3.20. TRISO coated particles from blended bed 4C: (a) fissile (4000-183) and fertile (4000-194) particle after irradiation; (b) polarized light photograph of (a) showing intense Maltese cross pattern in the fertile outer coating indicating considerable reorientation occurred during the irradiation; and (c) failed fertile particle from bed 4C

M340-420

(a)

75X

M340-418

(b)

75X

Fig. 3.21. TRISO coated particles from blended bed 4A:
 (a) representative photomicrograph of fissile (4000-183) and fertile (4000-194) particles after irradiation; (b) fissile particles exhibiting different kernel morphology, indicating particles experienced different thermal conditions in the bed

M340-285

(a)

100X

M340-293

(b)

200X

Fig. 3.22. TRISO coated particles from blended bed 1D:
(a) fissile (4000-182) and fertile (4000-185)
after irradiation; (b) fissile particle showing
reaction between kernel and silicon carbide

to enhance stability. Two rods (4B and 6B), which were prepared in prototype production equipment, were inadequately injected with matrix material; as a result, they partially debonded during irradiation. One of these rods also contained a fertile particle sample (lot 4000-194) that had significant failure as a thimble sample and in blended beds.

2. Dimensional change measurements indicated that the fuel rods shrank 1% to 2% in both length and diameter (see Table 3.4). The rods with isotropic filler had slightly better dimensional stability.
3. Initial metallographic results on the fuel rods showed no evidence of coating failure from matrix-to-coating interactions in any rod. The matrices were in good condition, with those containing isotropic graphite filler slightly denser than those with natural-flake filler; however, little difference in performance was noted between the two filler types.
4. The two silicon-bonded fuel rods were debonded and many of the SiC coatings were observed to have failed. Many of the fissile particles from the rod irradiated at high temperature exhibited kernel migration and kernel-silicon carbide interaction. The kernel migration is indicated to have been caused by higher temperatures and increased thermal gradients resulting from the decrease in thermal conductivity when the rod debonded (see Figs. 3.3 and 3.4).
5. The three batches of TRISO fissile particles all survived the irradiation. The two batches of TRISO fertile (fertile-B) particles had about 10% (lot 4000-185) and 50% (lot 4000-194) failure of both the outer PyC and SiC coatings. Similarly, a TRISO-III (TRISO without the outer PyC) fissile survived, whereas a TRISO-III fertile had about 10% failure. Fertile particle outer isotropic coating failure results appear to correlate reasonably well with initial optical anisotropy values (and/or BAFs) and irradiation exposure, with the more isotropic coatings showing the better performance.

Results are given in Table 3.5 and photographs of samples are shown in Figs. 3.5 through 3.8 and 3.10 through 3.13.

6. The three BISO particle batches survived the irradiation without failure. The results are given in Table 3.5 and photographs of the samples are shown in Figs. 3.9 and 3.14.
7. The five blended beds containing TRISO particles irradiated between 1185° and 1250°C all had significant failure of the fertile particles; these samples (with one exception) also had fissile particles that had greater than 1% failure. The particles in beds irradiated at 800°C survived the irradiation. The blended bed metallography is presented in Figs. 3.16 through 3.23.

Bed No. 6C, which was irradiated at a low temperature, was found to have some fuel particles in the center stuck together into a lump (see Fig. 3.18). In-situ pyrolytic carbon deposition was found to have been the cause of the particles bonding together. This pyrolysis most probably occurred as a result of impurity water and oxygen absorbed by the charcoal filler material in the bed. No sticking was observed in any of the other beds.

8. Dimensional measurements of the blended beds had a high degree of uncertainty resulting from difficulty in interpreting the neutron radiograph from which the data was obtained. Bed height shrinkages between approximately 2% to 15% for TRISO beds and 12% to 20% for the BISO bed were observed, and the data are summarized in Table 3.6.
9. Metallographic examination of TRISO blended bed samples, revealed, in the two high-temperature cells (1 and 4), some particle failures resulting from kernel migration and subsequent reaction with the silicon carbide layer (amoeba effect). Of the five TRISO beds from cells 1 and 4, three had significant failure (>10%) of the fissile and/or fertile particles.

Because of the small size of the blended bed samples (they are 0.4 in. in diameter by 0.7 in. high), only moderately high particle packing fractions were achieved during fabrication. This, coupled

with the relatively low thermal conductivity of the charcoal filler, is considered to have caused a more severe thermal environment (i.e., higher thermal gradients, and higher center line temperatures) than for beds with higher packing fractions and coke as filler material as has been considered for the Fort St. Vrain reactor.

Capsule F-28. Capsule F-28 is a proof test of production fuel samples to half exposure (4×10^{21} n/cm²). The capsule was removed from the GETR during the quarter after completing its scheduled irradiation. Dosimetry results (adjusted for dosimeter-wire orientation) showed the peak fast fluence to have been 4.1×10^{21} n/cm² ($E > 0.18$ MeV), the design value. The capsule is currently undergoing postirradiation examination in the hot-cell facility.

Capsule F-28 contained eight fuel rod, nine TRISO coated particle thimble, and six blended bed samples.

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 were tested at approximately 1250°C, and three were tested at about 900°C. All rods are 0.5 in. in diameter and 0.75 in. long.

The coated particle samples were intended to demonstrate the stability of reference TRISO coated particles. TRISO coatings of low, medium, and high density were included, along with one high-density sample with a high BAF. Six fissile and three fertile samples were tested. Because of the small fuel loading 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.

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 were irradiated at approximately 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 were irradiated at approximately 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.

Postirradiation analysis of the F-28 capsule samples is still in progress, and only preliminary results can be presented now.

Tables 3.7 and 3.8 summarize the irradiation condition and performance of fuel rod and particle samples. A brief summary of the results to date is given below. Photographs of fuel rod samples are shown in Figs. 3.23 through 3.35; photos of fuel particle samples are shown in Figs. 3.36 through 3.44. Work on blended beds is in progress and will be reported at a later date.

1. Four of the six fuel rods prepared in prototype production equipment survived the irradiation in very good condition. One rod (5A), which was wholly intact and appeared to be in good condition upon removal from the capsule, partially debonded while it was being measured with a micrometer (see Fig. 3.23). Another rod (5D) was badly cracked but intact (see Fig. 3.27).
2. Dimensional measurements of rods 5B, 5E, 6B, and 6F showed shrinkages of about 1% in both diameter and length. Postirradiation metallographic analyses of these rods revealed that they were very well injected with matrix and that the matrix was uniformly distributed and dense and exhibited very little cracking. There were no adverse coating-matrix interactions observed in any of these rods.
3. Metallographic examination of rod 5A showed it to be inadequately injected at the end that broke, and this is the apparent reason for its partial debonding during dimensioning (see Fig. 3.24). The

Table 3.7
FUEL RODS IRRADIATED IN CAPSULE F-28

Rod No. ^(a)	Type	Matrix			Coated Particles ^(b)	Irradiation		Postirradiation Examination							Remarks
						Temp ^(c) (°C)	Fast Fluence (10 ²¹ n/cm ²)	Dimensional Changes		Metallographic Exam					
		Diam (%)	Length (%)	Coated Particles						OPyC (% Fail)	SiC (% Fail)				
				Type				Batch No.							
6B	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	720	2.5	Excellent	-1.0	-0.9	Fissile Fertile	4000-218 4000-217	0 0.5	0 0	Small chip on one end Top portion of rod broke during dimensioning Small chip on one end
6F	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	730	1.8	Excellent	-1.1	-1.0	Fissile Fertile	4000-223 4000-199	0 0	0 0	
5A	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	1185	3.7	Good	(-1.5)	--	Fissile Fertile	4000-211 4000-217	1 10	0 1.5	
5B	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	1195	3.5	Very good	-1.1	-0.9	Fissile Fertile	4000-212 4000-215	1 1	0 0	
5D	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	1185	3.3	Poor, badly cracked	--	--	Fissile Fertile	4000-216 4000-199	3 15-25	0 0	
5E	Hot injected	Pitch	Nat. Flake	1800	TRISO-PL	1180	3.2	Very good	-1.1	-0.9	Fissile Fertile	4000-213 ^(d) 4000-210	1 80 ^(d)	0 0	

(a) Fuel rods are 0.5-in. in diameter by 0.75-in. long; all were produced in production equipment.

(b) All coated particles fabricated in production equipment in 5-in. coaters; PL = propylene.

(c) Average integrated temperature.

(d) High BAF particles.

TABLE 3.8
COATED PARTICLES IRRADIATED IN CAPSULE F-28

Lot No.	Coating ^(a)			Kernel		Irradiation			Postirradiation Examination (% Failure)			
	Type	Density (g/cm ³)							Macro Exam		Met Exam	
		Disk	CP ^(b)			Type	Th:U Ratio	Temp (°C)	Burnup ^(c) (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	PyC	SiC
211	TRISO-PL	1.77	1.84	Fissile-B	18	950	3	3.7	0.1	0.1	0.8	0
216	TRISO-PL	1.80	1.71	Fissile-B	18	950	3	3.7	0.2	0.1	0	0
218	TRISO-PL	1.75	1.79	Fissile-B	2.75	950	15	3.7	0	0	0	0
219	TRISO-PL	1.93	1.93	Fissile-B	2.75	950	15	3.7	0.1	0	0	0
220	TRISO-PL	1.86	1.93	Fissile-B	2.75	950	15	3.7	0	0	0	0
223	TRISO-PL	1.72	1.68	Fissile-B	2.75	950	15	3.7	0	0	0	0
199	TRISO-PL	1.64	1.68	Fertile-B	--	1185	1	4.0	6	1	4	1
215	TRISO-PL	1.88	1.94	Fertile-B	--	1170	1	4.0	2	0.5	4	0
217	TRISO-PL	1.78	1.84	Fertile-A	--	1155	1	4.0	6	0.5	2	0

(a) All coated particles prepared by production equipment in 5-in. coaters.

(b) Density determined on outer PyC removed from coated particle.

(c) Estimated.

M31431-1

M31431-3

(a)

K349-29

(b)

Fig. 3.23. Photomicrograph of fuel rod 5A irradiated at 1180°C to a fast fluence of $3.7 \times 10^{-21} \text{ n/cm}^2$ in capsule F-28 (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles): (a) preirradiation side and end views, and (b) view of irradiated rod broken during dimensioning; diametral change was approximately -1.5%

K81909

(a)

K81911

(b)

Fig. 3.24. Composite photomicrographs showing radial cross sections of fuel rod 5A after irradiation in capsule F-28 to 3.7×10^{21} n/cm² at 1180°C: (a) end that broke during dimensioning showing lack of matrix graphite and (b) section away from break

M31434-4

M31434-2

Preirradiation

S349-53

S349-54

Postirradiation

Fig. 3.25. Photomacrograph of fuel rod 5B irradiated in capsule F-28 to 3.5×10^{21} n/cm² at 1190°C (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles); diametral change was -1.1%

M349-201

(a)

K81913

(b)

Fig. 3.26. Photomicrographs of fuel rod 5B after irradiation to 3.5×10^{21} n/cm² at 1190°C in capsule F-28: (a) typical appearance of matrix graphite, (b) composite radial cross section

M31435-3

M31435-1

Preirradiation

S349-38

S349-40

Postirradiation

Fig. 3.27. Photomicrograph of fuel rod 5D irradiated to 3.3×10^{21} n/cm² at 1180°C in capsule F-28 (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles); diametral change was not determined

K82163

Fig. 3.28. Composite photomicrograph showing cross section of fuel rod 5D after irradiation to 3.3×10^{21} n/cm² at 1180°C in capsule F-28

M349-76

(a)

M349-69

(b)

Fig. 3.29. Photomicrographs of fuel rod 5D irradiated in capsule F-28 and showing (a) fertile particle (-199) with broken outer isotropic coating and (b) representative appearance of matrix graphite

M31437-4

M31437-2

Preirradiation

S349-43

S349-45

Postirradiation

Fig. 3.30. Photomacrograph of fuel rod 5E irradiated in capsule F-28 to 3.2×10^{21} n/cm² at 1180°C (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles); diametral change was -1.1%

M349-236

(a)

K81910

Fig. 3.31. Photomicrographs of fuel rod 5E after irradiation in capsule F-28 to 3.2×10^{21} n/cm² at 1180°C:
 (a) typical appearance of matrix graphite, and (b) composite showing radial cross section

M31439-4

M31439-1

Preirradiation

S349-24

S349-26

Postirradiation

Fig. 3.32. Photomacrograph of fuel rod 6B irradiated in capsule F-28 to 2.5×10^{21} n/cm² at 720°C (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles); diametral change was -0.9%

M349-239

(a)

K81908

(b)

Fig. 3.33. Photomicrographs of fuel rod 6B after irradiation to 2.5×10^{21} n/cm² at 720°C in capsule F-28; (a) typical appearance of matrix graphite, and (b) composite showing radial cross section

M31442-3

M31442-2

Preirradiation

S349-11

S349-12

Postirradiation

Fig. 3.34. Photomacrograph of fuel rod 6F irradiated in capsule F-28 to 1.8×10^{21} n/cm² at 730°C (pitch binder, natural-flake graphite filler, 1800°C preirradiation heat-treatment, TRISO-propylene coated particles); diametral change was -1.0%

M349-218

(a)

K81917

(b)

Fig. 3.35. Photomicrographs of fuel rod 6F after irradiation to $1.8 \times 10^{21} \text{ n/cm}^2$ at 730°C in capsule F-28: (a) typical appearance of matrix graphite, and (b) composite showing radial cross section

M31607-1

Before irradiation

M349-37

After irradiation

Fig. 3.36. Batch 4000-215 TRISO coated fertile particles (propylene outer isotropic) irradiated to 4.0×10^{21} n/cm² (1% FIMA) at 1170°C

M31609-9

Before irradiation

M349-18

After irradiation

Fig. 3.37. Batch 4000-217 TRISO coated fertile particles (propylene outer isotropic) irradiated to $4.0 \times 10^{21} \text{ n/cm}^2$ (1% FIMA) at 1155°C

M31602-7

Before irradiation

M341-11

After irradiation

Fig. 3.38. Batch 4000-199 TRISO coated fertile particles (propylene outer isotropic) irradiated to 4.0×10^{21} n/cm² (1% FIMA) at 1185°C

M31612-6

Before irradiation

M349-186

After irradiation

Fig. 3.39. Batch 4000-220 TRISO coated fissile particles (propylene outer isotropic) irradiated to 3.7×10^{21} n/cm² (15% FIMA) at 950°C

M31613-6

Before irradiation

M349-158

After irradiation

Fig. 3.40. Batch 4000-223E TRISO coated fissile particles (propylene outer isotropic) irradiated to $3.7 \times 10^{21} \text{ n/cm}^2$ (15% FIMA) at 950°C

M31611-3

Before irradiation

M349-148

After irradiation

Fig. 3.41. Batch 2000-219 TRISO coated fissile particles (propylene outer isotropic) irradiated to 3.7×10^{21} n/cm² (15% FIMA) at 950°C

M31604-7

Before irradiation

M349-51

After irradiation

Fig. 3.42. Batch 4000-211 TRISO coated fissile particles (propylene outer isotropic) irradiated to 3.7×10^{21} n/cm² (3% FIMA) at 950°C

M31608-5

Before irradiation

M349-46

After irradiation

Fig. 3.43. Batch 4000-216 TRISO coated fissile particles (propylene outer isotropic) irradiated to $3.7 \times 10^{21} \text{ n/cm}^2$ (3% FIMA) at 950°C

M31610-7

Before irradiation

M349-177

After irradiation

Fig. 3.44. Batch 4000-218 TRISO coated fissile particles (propylene outer isotropic, density = 1.79 g/cm^3) irradiated to $3.7 \times 10^{21} \text{ n/cm}^2$ (15% FIMA) at 950°C

major part of the rod, which remained intact, appeared well injected with matrix and was in very good condition.

4. Reasons for the poor performance of rod 5D are still being investigated. The 5D matrix appeared to be different from the other fuel rod matrices, and differences in process conditions that may influence matrix integrity are currently being examined. The failure of both fissile and fertile outer PyC coatings was observed to be higher in the rod than for the same batch in thimble samples. (The fissile particles had 0.2% failure in thimble sample and 3% failure in the rod; the fissile thimble sample was irradiated at a lower temperature, however: 950°C for the thimble compared to 1185°C for the rod. The fertile particles had 6% failure in thimble sample and 15% to 25% failure in the rod.) It is not certain, however, that the coating failures were the prime cause of the poor condition of the rod, since the matrix was more cracked and contained more voids than the matrixes of intact rods.

A very small fraction of the fissile particle outer PyC coatings appeared to have cracked as a result of possible matrix-coating interactions. However, preirradiation analysis of a companion rod also revealed incipiently cracked outer PyC coatings on a very small number of fissile particles. This cracking was apparently caused by the matrix bonding tightly to the outer isotropic coating of these few particles. This effect was not seen in any other rod and appears to be peculiar to this fissile particle batch (4000-216)-matrix combination. This matter is being investigated further.

5. Rod 5E was found to have 80% failure of the outer PyC coatings on the fertile particles. (The fertile particles occupied about 12% of the rod by volume.) This rod had a well injected matrix. It was intact, in excellent condition, and exhibited 1% dimensional shrinkage in both directions. These data show that high fertile particle failure is not necessarily detrimental to the performance of a well injected

fuel rod at these intermediate exposures (3.2×10^{21} n/cm²). Failure of the fertile particles was not unexpected, since these coatings were intentionally prepared with a high BAF (1.11) to evaluate anisotropy limits. There were no failures observed that were attributed to interactions with the matrix.

6. The six TRISO fissile thimble samples survived the irradiation. The two TRISO Fertile-B thimble samples had 2% and 6% failure of the outer PyC coatings; the TRISO Fertile-A sample also had 6% outer PyC failure.
7. Metallographic examination of the fertile thimble samples indicated that high preferred orientation in the outer PyC coatings of all samples is the most probable reason for the failures observed.

Sample batch 4000-215 appeared to have the most isotropic coating (as determined by examination in polarized light) and had the best performance.

8. Average integrated temperatures for the fissile particle thimble samples were about 950°C, or approximately 200°C below design (the thermocouples in these samples were lost early in life).

Several preliminary conclusions can be drawn from the investigation of capsule F-28 in conjunction with the F-27 results:

1. Fuel rods having a dense, uniform matrix retain their integrity and exhibit dimensional shrinkage even if there is a high failure of particle coatings. This has been demonstrated by rod F28-5E, which had 80% failure of the fertile particles with no apparent detrimental effects upon the rod stability. This rod exhibited 1% shrinkage on both the length and diameter as did other rods irradiated under similar conditions.

2. Fuel rods with nonuniform matrix (extensive voids) debond easily in matrix-lean regions, even if there is no coating failure. Examples of this have been rods F28-5A and F27-6B (see earlier quarterly report GA-10010); both had regions with inadequate matrix material and in each case particles debonded in these regions.
3. Coatings with preferred orientation higher than specification (BAF > 1.09) have very high failure percentages (up to 80%). Fertile batches 4000-194 (F-27) and 4000-210 (F-28) both had BAFs of greater than 1.09, and these batches exhibited substantially higher outer pyrolytic coating failures than the other batches (at 1100°C or higher). These coatings also exhibit a marked increase in the degree of preferred orientation during irradiation.
4. Rods that have both a nonuniform matrix and particles with highly oriented coatings have poor irradiation stability (as evidenced by rod F27-4D, which had particles from batch 4000-194 and was also poorly injected).

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 capsule is in its seventh and last scheduled cycle in the GETR (E-5 core position) and is operating satisfactorily. No fuel thermocouples are still operative. However, the fuel rod, coated particle, and blended bed samples in cells 1 to 5 are estimated to be operating at 1090° to 1120°C, very close to the design temperature of 1125°C. The beds and rods in cell 6 are operating at 580°C, considerably below their design temperature of 900°C. The fuel has reached an estimated fast fluence of 7.2×10^{21} n/cm² with an estimated burnup of 20% 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.

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 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 deposition (sorption) behavior of metallic fission products on coolant circuit metals and the relative distribution of fission products between carbon dust and metal surfaces.
3. 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

Reaction of Water with Fuel Materials. Experiments on the effects of water vapor on fuel beds with varying failed particle fractions have continued (see earlier quarterly report GA-9875 for description of earlier work). Four additional samples containing varying amounts of failed fuel particles (15%, 25%, 47%, and 62.5%) were hydrolyzed for one week at 240°C and 2.5% to 3% water vapor in helium. Weak particles (particles with thin coatings) were mixed with strong particles and 0.1 g/cm³ CPC coke filler material and tightly packed in screw-end H-327 graphite crucibles of proper dimensions

(0.5 in. i.d., 0.815 in. o.d., and 3 in. long) to simulate the strength of the weakest web in a fuel block. The crucibles were annealed at 2400°C to fail the weak particles prior to the steam exposure.

Only the carbide fuel mixture containing 47% broken particles cracked the graphite. The 62.5% sample did not break, although the debris mixture was tightly packed in the crucible.

All of the results to date are summarized in Table 3.9. The 47%, 75%, and 100% samples fractured the graphite crucibles, yet two at 62% and 62.5% did not. This result indicates that the crucible used for the 47% run was abnormally weak (i.e., flawed); and it is concluded that for normal graphite, particle failures up to 62% would not break the graphite web.

TABLE 3.9
RESULTS OF HYDROLYSIS OF FUEL PARTICLE BEDS
CONTAINING BROKEN PARTICLES

Sample No.	Failed Particle Content (%)	Charcoal Filler Content (g/cm ³)	Time (hr)	Results
1147-109	12	0.1	168	Crucible intact
1147-113-16	15	0.1	168	Crucible intact
1147-113-25	25	0.1	168	Crucible intact
4705-24	33	0	262	Crucible intact
1147-113-47	47	0.1	168	Crucible failed
4705-10	~62(a)	0.1	119(b)	Crucible intact
1147-113-57	62.5	0.1	168	Crucible intact
1147-108	75	0.2	<70	Crucible failed
1147-107	100	0	<22	Crucible failed

(a) The percent failure was calculated (not actually measured) based on expected behavior of particles used.

(b) 4705-10 was exposed to 2.5-3% steam for 78 hours and ~80% steam for an additional 41 hours.

Further work on these blended bed experiments has been suspended, but similar runs are planned using bonded fuel rods containing various known amounts of failed particles.

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 by testing it with a segment of a dummy core and by testing critical components, as necessary, in helium at the temperature and chemical impurity level expected in the plant.

The in-air refueling testing of the fuel handling machine has now been completed with the exception of the 10-column region-28 offset at 1-5/8 in. at 10°. During these tests a number of problems were encountered that required either hardware or software modifications.

The guide rails were removed from the fuel storage rack compartment. These rails were removed to prevent interference with the keyed metal elements in compartment No. 1 and simulated bowed elements in the remaining compartments. The bowed elements were simulated by adding 0.093-in. shims on all sides. This simulates the maximum bow plus 0.013 in. of clearance.

While transferring elements from the fuel storage rack into the simulated core, a grapple head (ϕ) rotation of 3° to 4° was required to deposit a border reflector element that normally required no ϕ rotation. Investigation showed the grapple head lower plate to be rotated counterclockwise with respect to the normal head rotation zero reference position. The grapple head was removed from the fuel handling machine, and a detailed inspection was performed. It was found that the clearance allowed for tolerance buildup, also allowed the rotation to occur. The subassemblies of the grapple head were match drilled and doweled as the head was reassembled.

Testing was later stopped when the lower plate was again found to be rotated. An investigation determined that the dowels had been bent permitting

rotation. The cause of these bent dowels was attributed to the head rotation system. During deposit of border reflector elements into the core, severe oscillation of the grapple head occurred. Several tests were conducted to determine the cause. Adjustments of the control gain to provide a slower response made a marked improvement, but some oscillation continued. Various sized orifices were next tried in the fill and bleed lines of the head rotation pneumatic system in order to provide damping. These orifices caused a loss of system response time and control.

It was concluded that the system without orifices was unstable, and with orifices the physical arrangement of the tubing permitted pressures to build up locally in the system that were in excess of that required to index the grapple head to any desired position of head rotation.

Without the orifices, as long as the grapple head is engaged to an element in the core, no effect is noticed because the buildup in torque is restrained by adjacent elements and no motion occurs. Once the element is free of its position of residence, the restraint on the element is removed and rotation occurs. The control system senses this and, in attempting to correct, oscillation results.

To prevent this situation the control was converted to an open-loop pressure/position command system. The accuracy for this type of control is reduced from that of a closed loop type; however, it is insensitive to the problems of external over-riding forces. Initial tests indicate that the accuracy of this control will be satisfactory for all refueling conditions. The automatic refueling computer program was modified to check that the proper grapple head position is obtained before continuing with other motions. This was done to prevent lowering of the machine in a pickup or deposit operation before the correct grapple head position is obtained.

Two grapple head pneumatic hoses failed and were replaced with a different and more flexible metal hose. One of these hoses was the grapple probe hose.

This hose attaches to a swivel that has an inclined face. The shape of the face was determined to be causing excessive bending of the hose. A new swivel with a compatible shape is being fabricated.

Handling of the metal control rod element has proved difficult because of the cupped hole. The grapple probe retainer diameter has been increased to provide a better register than is presently provided. This probe has not as yet been tested.

Throughout the tests when operating in the automatic mode a number of emergency stops (clamps) occurred during the pickup and deposit of elements. These stoppages were the result of exceeding the maximum plate vertical displacement sensor setting. The tolerance between the maximum "working" plate position and the over-travel sensor was found to be less than 0.05 in. This tolerance was increased to 0.15 in. by raising the over-travel sensor setting to its highest safe point. Further testing is required before the adequacy of this revision can be fully assessed.

At the end of the reporting period during testing of the 10-column region-28 offset 1-5/8 in. at 10°, the R drive motor current became excessive. The fuel handling machine arm drive unit was removed from the machine and disassembled to determine the cause of this failure and the bearings and ball screw were examined. The ball screw was found to have a broken ball and debris causing rough operation. Evidence of corrosion was present on the thrust bearings. Methods of preventing a recurrence of this situation are presently being evaluated to determine corrective action.

The Data II program, which produces refueling data tapes on the UNIVAC 1108 computer, has been modified to provide for additional vertical positions to be specified by its input data, which comes from "Order Program" punched cards. This was done to allow for the special handling of some top layer elements in the reactor core. After having trouble reading the refueling data tapes produced on the UNIVAC 1108, it was found that the punch was intermittently incorrectly punching the refueling tapes. This trouble has

been investigated and some repairs have been made, but the problem is still not completely resolved. The present error rate has been reduced to about one error per 500 ft of data tape. Equipment has been purchased to manually correct the errors which were made, thus allowing some faulty tapes to be salvaged.

Preparation for environmental tests is under way. The autoclave and fuel handling machine have been leak checked. A few leaks were found in the rack and have been repaired. Other minor leaks were found at the top housing of the fuel handling machine. These latter leaks were at gasketed joints and were arrested by retorquing.

A specification for the fuel handling machine disassembly and refurbishment has been issued, and a shipping specification has been issued for review and comments.

The fuel handling machine field assembly and field test specifications were issued, and a comprehensive contractor's checkout procedure is being written as an appendix to the fuel handling machine field assembly specification.

A reevaluation of the need for the indexing arm link to collapse under a given external load was made. It was determined that arm-drive motor currents can be monitored and sufficient indication obtained of any potential problem to permit preventive maintenance to be effected. Further work on a collapsing link design has therefore been discontinued.

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 tube integrity are also planned.

SLEEVE AND WEDGE TEST

In response to a question by the AEC Compliance Division concerning the adequacy of serrated sleeve-wedge combinations, a test series was carried out to measure the pull-off strength of the serrated sleeves and wedges incorporated in the design. The tests were conducted at service operating temperature in a helium atmosphere on approximately 40 specimens. The results are presented in Table IV.1. The maximum expected end loads on the sleeve-wedge assemblies were 110 lb for the superheater II outlet and 360 lb for the inlet; for the economizer-evaporator-superheater they were 530 lb at the feedwater inlet and at the steam outlet. For superheater II, the end loads are low and would not cause the serrated sleeve and wedge to be pulled apart even if the flange were bearing against the support plate.

The end load problem does not exist for the reheater because the sleeves and wedges are so located that the flanges could not come in contact with the support plate during service.

TABLE IV.1
RESULTS OF SLEEVE-WEDGE TESTS

Serial No.	Assembly Loading (Preload) (lb)	Pull-off Load (lb)
Reheater Test at 1300°F ^(a)		
2	---	370
3	---	660
5	---	735
16	---	380
R1	993	1685
R2	372	675
R3	1737	1580
R4	1737	4050
R5	2358	2950
R6	1737	1935
R7A	2600	1335
R8A	2600	1135
R9A	2600	2200
R10A	2600	2550
R11A	2600	1120
R12A	2600	460(b)
Superheater II at 1300°F ^(c)		
7	---	1285
6	---	530(b)
8	---	1500
10	---	1720
11	---	1410
14	---	1430
R1A	2480	1260
R2A	2480	810
R3A	2480	(d)
R4A	2480	2580
R5A	2480	295
R6A	2480	890
Economizer-Evaporator-Superheater at 1000°F ^(e)		
9	---	1335
12	---	1310
13	---	415
15	---	295
17	---	1335
R13A	2480	470
R14A	2480	1275
R15A	2480	2675
R16A	2480	3355
R17A	2480	2925
R18A	2480	(d)

(a) Average load 1490 lb.

(b) Broke in first serration on sleeve.

(c) Average load 1230 lb.

(d) Would not fit in test rig.

(e) Average load 1540 lb.

The economizer-evaporator-superheater bundle feedwater inlet and steam outlet tubes have predicted end loads that are somewhat higher than a few of the individual test pull-off loads obtained, but the average test load is a factor of three above the predicted end loads. The amount of axial movement that would be necessary to totally relieve the load is only 0.025 in., and therefore even small movements of the tube would effectively reduce the end load. To cause any problem, the flange of the sleeve or wedge would have to be flat and hard against the support plate, but this does not occur. Furthermore, the number of tubes having end loads is limited, the total number having end loads over 300 lb being 45 per module out of 3280. The last sleeve-wedge assembly on each tube coil would have the maximum end load and those in the bundle would have essentially none.

The pull-off test did not permit the sleeve-wedge assembly to slide, as can be seen in Fig. IV.1, the load being applied directly to the flanges. During plant operation, one flange would come up against the support plate and sliding could occur and relieve the end load. Should a sleeve-wedge assembly separate, it can not come off the tube; the sleeves and wedges will either be retained between the two support plates or, in the case of the last assembly on a coil, it could move to the feedwater subheader but would still be completely retained.

It is concluded that there will be no separation of sleeves and wedges as a result of end loads in superheater II and the reheater tube bundles. In the economizer-evaporator-superheater, it is very unlikely that any separations would occur, and even in the event that separation should occur, it does not seem possible for a sleeve or wedge to come off the tube. The loss of a sleeve-wedge from a tube in the tube support plate does not necessarily mean that the functional life of the tube would be impaired, but only that some wear might occur on the outside surface.

FATIGUE TESTING

Low cycle strain controlled fatigue tests with hold-time effects are progressing at Battelle on Incoloy 800 material. A progress report is expected shortly from BMI.

Fig. IV.1. Sleeve-wedge assembly during testing; material Hastelloy X

CHEMICAL CLEANING TEST OF FEEDWATER RINGHEADER

A program justification document and a test procedure for the chemical cleaning of a Sulzer feedwater ringheader was issued. The test is designed to establish the severity of chemical attack on the Sulzer valve internals from the cleaning solution. The test will be conducted using an industrial chelating agent.

MODULE AIR-FLOW TEST

The air flow test module was readied for shipping to San Diego, the shipping procedure was approved, and shipment occurred at the end of May. A preliminary instrumentation procedure has been received, and the remainder of the test procedure is being prepared.

Work has begun on developing a procedure for extrapolating tube temperatures from the air-flow test to reactor operating conditions.

All test systems are complete with the exception of the dry air purge system. Completion of a mockup of the reheater dome installation cover is expected early in June.

TUBE PLUGGING MACHINE

Installation drawings of the tube plugging machine and "as built" changes of drawings pertaining to the seal weld configuration were completed and issued.

As a result of a cooling water leak, the vertical drive motor brake and the drive-screw ball-bearing assembly were replaced; following cleaning and reassembly, the machine was successfully checked out and shipped to Fort St. Vrain. In lieu of the inflatable seals, a piston was constructed and shipped with the tube plugging machine for leak testing the seal welds made as part of the initial steam generator installation. The inflatable seal vendor has designed new tooling, and plans to deliver the seals during the next reporting period.

Fabrication of tube plugging parts is about 80% complete, and assembly of these parts has been initiated.

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 investigation of fabrication methods and coating process variables and by preirradiation evaluation of coated particles.

COATED PARTICLE AND FUEL ROD EVALUATION

Studies have indicated further that alpha counting is promising for use in predicting uranium contamination on and in the outer 10 to 20 μm of coated fuel particles. A drawback is that counting times associated with 1 to 2 g of fuel particles in standard alpha counting equipment appear to be too long (3 to 8 hr) for the technique to be useful for testing production particles. Gas proportional counters and scintillation counters are therefore being investigated for possible use in reducing counting times.

The burn-leach technique (see earlier quarterly report GA-9440 for description) appears to be the most satisfactory method for evaluating production fuel rods containing TRISO coated particles. This technique can be used to determine the fraction of fuel outside of intact silicon carbide coatings.

DISTRIBUTION OF URANIUM AND THORIUM IN PARTICLE COATINGS

The uranium and thorium contamination in the outer pyrolytic carbon coating of TRISO coated particles (Lot 4000-218E) was measured. (The particles are described in the next section.)

Five aliquots of approximately 1000 carefully selected fuel particles were surface leached at 90°C using an acid solution ($\text{HNO}_3 + \text{HF}$). The leach solutions were activated and counted to determine uranium and thorium content. Two consecutive leaches were carried out, and the uranium and thorium content of the second leach solution was below the level of detection. After the surface leach, the outer PyC layer was burned off at 800°C in an oxygen atmosphere, and the oxidized material was dissolved in acid solution at 90°C. The solutions were activated and counted for U and Th. The results are given in Table V.1.

The uranium contamination on the surface of the particle is a factor of approximately 10 lower than the amount of uranium within the outer pyrolytic carbon layer. The thorium contamination differs by a factor of approximately 5. The data indicate that selecting the particles was done quite satisfactorily because no broken or defective particles were observed.

EFFECT OF FUEL MIGRATION ON FISSION-PRODUCT RELEASE FROM TRISO PARTICLES

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. This work involves long term (months) isothermal anneals of selected particles in vacuum with periodic measurements of the degree of fuel migration and the level of fission-gas release. Related work involving isothermal and thermal gradient anneals of particles with periodic determination of the degree of fuel migration is also being performed.

A batch of TRISO coated carbide particles (Lot 4000-218E)* was selected for the experiments. These particles were considered representative of production particles at the current state of the art.

*Particles were TRISO coated carbide particles Lot 4000-218E: kernels were 225 μm (avg), $(\text{Th,U})\text{C}_2$, Th:U = 2.8:1; coatings were TRISO-II, 52 μm buffer, 18 μm inner PyC, 19 μm SiC, 37 μm outer PyC; uranium content, 8.29 wt-%.

TABLE V.1
URANIUM AND THORIUM CONTAMINATION ON AND IN THE OUTER PYROLYTIC
CARBON COATING OF TRISO COATED PARTICLES
(Lot 4000-218E)

Aliquot Weights (g)	Uranium (g U/g Particles)		Thorium (g Th/g Particles)	
	Surface Contam	Coating Contam	Surface Contam	Coating Contam
0.1203	4.3×10^{-7}	5.4×10^{-6}	0.8×10^{-5}	4.7×10^{-5}
0.1200	5.6×10^{-7}	5.4×10^{-6}	1.1×10^{-5}	4.2×10^{-5}
0.1200	5.6×10^{-7}	5.8×10^{-6}	1.6×10^{-5}	5.1×10^{-5}
0.1260	5.5×10^{-7}	3.5×10^{-6}	2.0×10^{-5}	4.8×10^{-5}
0.1205	8.1×10^{-7}	4.9×10^{-6}	(a)	4.2×10^{-5}
Average Values	5.8×10^{-7}	5.0×10^{-6}	1.4×10^{-5}	4.6×10^{-5}

(a) Not determined.

Approximately 1000 particles are used for each anneal. The particles are mixed with calcined petroleum coke material and contained in graphite crucibles. The graphite crucibles are placed in open tantalum cylinders, and tantalum lids are pressed on the top and bottom of the cylinder by means of a screw cap on an outer graphite crucible. This is heated above 1000°C, and the tantalum self-welds and forms a leak-tight containment for the inner graphite crucible. Therefore, no migration of uranium or thorium into the furnace system can occur. After an anneal, the particles are separated from the coke material and graphite crucible, and the latter are analyzed for uranium and thorium content by activation (by irradiation in TRIGA) and gamma counting. Different aliquots of particles are annealed for different lengths of time and the release to the coke and graphite is taken as the total release for this time period.

Particles used in these tests were hand-picked (with the aid of X-ray radiographs) to eliminate defective particles. Selected particles were used so that particle characteristics can be related to fuel migration and fission-gas release behavior.

Samples have been annealed at 1660°, 1550°, and 1140°C for various time periods up to 2000 hr. The experimental results are given in Table V.2. The results for thorium are not complete. The data are somewhat scattered mainly because of variation in particles.

Except at 1660°C and 1000 hr, the fractional release values for uranium and thorium were consistently low (around 10^{-5}), indicating that only the heavy metal in the outer pyrolytic carbon layer migrated into the coke matrix material.

The heavy metal fractional release data for the TRISO coated particles are much lower than data for BISO coated particles measured at similar conditions in other laboratory anneal tests conducted at Gulf General Atomic, indicating that the silicon carbide layer in the TRISO coating is relatively effective as a barrier to the migration of the heavy metals. In the tests using BISO coated particles, uranium releases become appreciable after about 20 hr at 1600°C.

TABLE V.2
FRACTIONAL RELEASE OF URANIUM AND THORIUM FROM TRISO COATED
FUEL PARTICLES IN LABORATORY ANNEAL TESTS^(a)

Time (hr)	Anneal at 1660°C ^(b)		Anneal at 1550°C		Anneal at 1140°C	
	U	Th	U	Th ^(c)	U	Th
5	3.4×10^{-6}	8.3×10^{-6}	2.2×10^{-6}	---	8.6×10^{-7}	1.1×10^{-5}
10	5.3×10^{-6}	1.8×10^{-5}	1.2×10^{-6}	---	6.9×10^{-7}	1.1×10^{-5}
20	9.1×10^{-6}	1.5×10^{-5}	3.2×10^{-6}	---	9.5×10^{-7}	1.3×10^{-5}
30	1.5×10^{-5}	1.4×10^{-5}	1.1×10^{-6}	---	6.9×10^{-7}	1.4×10^{-5}
50	1.5×10^{-5}	1.6×10^{-5}	1.4×10^{-6}	---	1.0×10^{-6}	9.3×10^{-6}
100	2.2×10^{-5}	4.8×10^{-6}	2.7×10^{-6}	---	8.2×10^{-7}	1.3×10^{-5}
200	1.2×10^{-5}	3.8×10^{-6}	5.4×10^{-6}	---	7.0×10^{-7}	9.5×10^{-6}
300	2.7×10^{-5}	2.0×10^{-4}	7.5×10^{-6}	---	1.2×10^{-5}	3.8×10^{-5}
500	2.4×10^{-5}	3.7×10^{-5}	1.9×10^{-5}	---	---	---
1,000	1.4×10^{-2}	1.3×10^{-2}	---	---	3.2×10^{-6}	1.8×10^{-5}
1,490	---	---	1.6×10^{-5}	---	---	---
2,000	---	---	---	---	---	1.1×10^{-5}

(a) Values represent fractional release of U and Th from particles into coke matrix material.

(b) Temperatures are within $\pm 20^\circ$ of the indicated values.

(c) Data not yet available.

The high release of heavy metals in the 1000-hr anneal at 1660°C (see Table V.2) is attributed to the failure of silicon carbide coatings. This has been confirmed by X-ray radiography.

THERMAL STABILITY OF COATED PARTICLES

The purpose of the thermal stability studies is to determine the maximum fuel operating limits of time and temperature that are imposed by chemical interactions between the kernel and coating in the coated fuel particles. There are three features of the fuel system that can be characterized with regards to nuclear and thermal history and used to determine fuel design limits. These features are (1) mechanical integrity of the fuel particle coating, (2) mechanical effects in fuel rods and elements following loss of coating integrity, and (3) fission-product release of coated particles.

Determination of time-temperature limits for loss of coating mechanical integrity has been the object of experimental work in which coated particles are heated for long times in a thermal gradient with periodic radiographic monitoring of kernel migration and kernel-coating interaction. Expansion of this work is planned to include determination of structural effects in the fuel rod and element as a consequence of loss of coating integrity. There has also been a secondary effort in support of experimental studies directed toward the determination of fission-product release from particles as a function of time and temperature.

Thermal Gradient Experiments

In this work, beds of coated particles are being heated in a thermal gradient and examined at periodic intervals to detect kernel migration and changes in distribution of Si, C, Th, and U. Both loose beds and monolayers of resin-bonded particles are being heated. The loose beds will be destructively examined by metallography when a portion of the bed has exhibited loss of coating integrity. The bonded monolayer of particles is examined nondestructively by contact microradiography at periodic intervals.

Experiment 4283-72. The apparatus and specimens for this experiment were described in an earlier quarterly report, GA-9261. Particles from five TRISO particle batches are being heated in a thermal gradient of about $0.03^{\circ}\text{C}/\mu\text{m}$ in a temperature range of 1250° to 1350°C or 1350° to 1450°C across the 1/8-in.-diameter loose beds and 1375° to 1425°C across the 3/32-in.-diameter bonded monolayer of particles. These temperatures differ somewhat from those reported previously and represent the results of a recent temperature calibration of the experimental assembly.

During this report period, specimens in the resin-bonded monolayer of particles were radiographically examined after 10,000 hr of heating at 1375° to 1425°C . As observed in earlier examinations, kernel migration toward the hot side of the particle (amoeba effect) was observed in 50% to 75% of the particles. Maximum kernel migration of $25\ \mu\text{m}$ was observed at 10,000 hr in a particle from TRISO batch 3923-111E (Th:U ratio 2.75). There has been no loss of coating integrity in any of the specimens examined radiographically.

During the examination at 8217 hr, the crucibles containing loose beds of particles were inadvertently tipped over. During this accident, the particles heated in the high-temperature section (1350° to 1450°C) did not fall out of the holes in the crucible because they were firmly cemented in place by a deposit of pyrocarbon that was deposited during heating. The particles heated in the lower-temperature section (1250° to 1350°C) were cemented by pyrocarbon only on the cool side of the 1/8-in.-diameter bed. Particles from the high-temperature side of these beds fell from the holes and their identity was lost. Pyrocarbon deposits in these experiments have been reported previously (see earlier quarterly report GA-9130) and probably result from a complex reaction between the hot surface of the graphite electrode, moisture in the helium atmosphere, and the relatively cool surfaces of the particles and crucibles. The effect of the pyrocarbon deposit has been to change the originally loose particle beds to a form of bonded bed. This effect will tend to reduce thermal gradients and will have to be taken into account when final analysis of the bed particles is performed.

This experiment was returned to the furnace and heating continued, the time at temperature is now 10,250 hr.

Experiment 4464-25. This is a long-term intermediate temperature (1150° to 1325°C) experiment in which samples of 14 different TRISO coating batches are being heated in a thermal gradient of 0.05°C/μm. The apparatus and specimens were described in detail in an earlier quarterly report (GA-9720).

Specimens were examined radiographically at 4450 hr of heating, and no kernel migration was detected. The only notable effect of the heating was the loss of silicon from the hot side of particles with a silicon or silicon-rich SiC overcoat (Lots 4452-149, 4632-43 D) at temperatures of 1200° to 1350°C. The TRISO particles that have an outer PyC coating do not exhibit this effect.

The silicon overcoat for particles was a concept explored in an attempt to develop particles that would bond to each other with a proper heat treatment to form fuel rods. The loss of Si from the outer coating, demonstrated in these experiments at peak reactor operating temperatures, suggests that the system may lack the required stability for long-term fuel rod integrity. No further work is planned with these particles other than to periodically report their condition in this test.

Experiment 4464-57. This is an experiment conducted to determine the relative magnitude of the driving force for kernel migration derived first from the thermal gradient and second from the difference in free energy between pyrocarbon and the graphite rejected on the cool side of the kernel during the amoeba effect.

Particles from TRISO batch 3592-35 E were mounted in five separate mono-layer radiograph mounts of the kind used in experiment 4283-72 described above. The mounts were heated for 143 hr in a thermal gradient of about 0.05°C/μm in a temperature range of 1450° to 1550°C. The mounts were radiographed at

143 hr. Three of the mounts were returned to heating under the condition discussed above, but one was intentionally rotated 180° and another 90° from its former position relative to the thermal gradient. At the end of an additional 1002 hr of heating under these conditions, the mounts were again radiographed.

Examination of the radiographs at 0, 143, and 1145 hr of heating showed that the kernel migrated toward the hot side of the particle at a rate apparently affected only by temperature at constant gradient. This was true even when it was necessary for the kernel to move through graphite that had been rejected on what was previously the cool side.

The effect is shown graphically in Figs. V.1 and V.2 for a single particle that was in the mount rotated 180° at 143 hr of heating. This mount was examined metallographically and the particle cross section shown in Fig. V.3 exhibits the typical "amoeba" effect microstructure.

The remaining four mounts were placed in a graphite crucible and heated in helium at 1700°C isothermally for 1000 hr. At the end of 1000 hr the mounts were radiographed, and the results showed that migration started during heating in a thermal gradient had not progressed a measurable amount (greater than 2 μm) during 1000 hr heating at 1700°C. The mounts were returned to the furnace for additional heating at 1700°C.

These results clearly show that (1) the thermal gradient provides the primary driving force for kernel migration, (2) the free energy difference between PyC and rejected graphite does not provide sufficient driving force to maintain kernel migration in the absence of a thermal gradient, and (3) the rejected graphite developed during kernel migration does not provide a barrier to kernel migration in the opposite direction when the thermal gradient is reversed.

M32184-7

(300x)

Fig. V.1. Radiograph of TRISO-I coated $(\text{Th,U})\text{C}_2$ particle batch 3592-35E) as-coated and mounted for heating in a thermal gradient

Cool
Side

M32185-7

(300x)

M32186

(300x)

Initial Heating
143 hr, $1525 \pm 50^\circ\text{C}$
Migration rate, $0.05 \mu\text{m/hr}$

Reversed Thermal Gradient
1002 hr, $1475 \pm 50^\circ\text{C}$
Migration rate, $0.01 \mu\text{m/hr}$

Fig. V.2. Radiographs of the particle shown in Fig. V.1 for two experimental heating conditions. The rejected graphite can be seen as a dark band adjacent to the kernel on the cool side of the kernel in each condition. The kernel migrated through graphite previously rejected on the cool side when the thermal gradient was reversed. Temperature difference across the entire particle was about 16°C ; it was about $\sim 3^\circ\text{C}$ across the kernel

Cool
Side of
Particle

M32269-13

(300x)

Fig. 3. Metallographic section of the particle shown in Figs. V.1 and V.2 after heating under reversed thermal gradient conditions. Carbon rejected as graphite during the initial heating was transported across the kernel and rejected again as graphite when the thermal gradient was reversed

FISSION-PRODUCT RELEASE

Fission-Gas Release from Mixtures of Fuel Particles and Coke Matrix Material

Fission-gas release measurements have been performed on samples of coated fuel particles mixed with matrix material contaminated with uranium. The mixture of fuel particles and coke material simulates a fuel rod. The purpose of this work was to establish the relationship between the level of heavy metal contamination in fuel rod matrix material and the release of fission-product gases.

In this work, samples of BISO coated carbide particles (Lot 4000-174E) were mixed with petroleum coke matrix material and annealed for various time periods at high temperature as described earlier in this task. This resulted in various degrees of uranium migration into the matrix material.

For use in the fission-gas release experiments, a weighed portion of the coke matrix material, which had been previously separated from the fuel particles to determine uranium and thorium concentrations in the coke matrix material, was recombined with the particles inside the graphite crucible used for the annealing studies, and this was inserted into a clean graphite crucible. Routine fission-gas release measurements were carried out by irradiating the sample for 1 hr at 1200°C in a King-type graphite tube furnace mounted in the core of the TRIGA Mark I reactor core (see earlier quarterly report GA-8879 for a description of the technique for measuring fission-gas release in the TRIGA reactor).

The fission-gas release (R/B) values (corrected for steady state using the equations given by Zumwalt, et al., Ref. 1) are given in Table V.3 together with the fractional uranium contents in the matrix material. The fission-gas release values at four different fractional uranium contents are plotted as a function of half life in Fig. V.4, and the Kr-85m R/B values are plotted as a function of the fraction of uranium released into the coke material in Fig. V.5.

TABLE V.3
FISSION-GAS RELEASE FROM MIXTURES OF BISO COATED FUEL PARTICLES
AND COKE MATRIX MATERIAL CONTAMINATED WITH URANIUM^(a)

Fraction U in Coke	Fractional Release, R/B					
	Kr-85m	Kr-88	Kr-87	Kr-89	Xe-138	Xe-139
5.7×10^{-5}	1.9×10^{-4}	1.6×10^{-4}	1.1×10^{-4}	2.2×10^{-5}	9.3×10^{-6}	7.6×10^{-6}
2.0×10^{-4}	3.6×10^{-4}	3.6×10^{-4}	2.4×10^{-4}	5.2×10^{-5}	2.2×10^{-5}	1.9×10^{-5}
2.3×10^{-4}	6.1×10^{-4}	5.4×10^{-4}	3.7×10^{-4}	8.8×10^{-5}	2.4×10^{-5}	1.1×10^{-5}
1.8×10^{-3}	2.1×10^{-3}	1.9×10^{-3}	1.2×10^{-3}	2.9×10^{-4}	1.1×10^{-4}	1.7×10^{-5}
4.5×10^{-3}	3.3×10^{-3}	3.3×10^{-3}	2.1×10^{-3}	5.7×10^{-4}	2.2×10^{-4}	9.2×10^{-5}
2.1×10^{-2}	1.8×10^{-2}	1.7×10^{-2}	1.2×10^{-2}	3.2×10^{-3}	8.8×10^{-4}	3.4×10^{-4}
1.7×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.0×10^{-2}	3.2×10^{-3}	9.1×10^{-4}	6.4×10^{-4}
1.7×10^{-2}	1.3×10^{-2}	1.3×10^{-2}	8.4×10^{-3}	3.0×10^{-3}	7.4×10^{-4}	3.9×10^{-4}
2.2×10^{-2}	2.1×10^{-2}	2.1×10^{-2}	1.3×10^{-2}	4.0×10^{-3}	1.3×10^{-3}	9.5×10^{-4}

(a) Measurements were made at 1200°C in the TRIGA reactor and a steady-state correction was applied to the data.

Fig. V.4. Fission gas release (R/B) values for mixtures of BISO coated fuel particles and coke matrix material contaminated with uranium

Fig. V.5. Kr-85m R/B values plotted as a function of uranium contamination of the matrix material

Figure V.4 shows a distinct half-life dependence for the krypton and xenon R/B values, indicating that the release at 1200°C is dependent on a diffusion mechanism. This half-life dependence does not change when the matrix contamination increases, indicating that the fission-gas release even from the small coke particles is diffusion limited. This is important, because the uranium contamination in the coke is a major contributor to the fission-gas release.

To establish the contribution of the matrix material to the fission-gas release, a uranium migration sample was run as received, the particles were separated from the coke, and R/B values were measured for both. The particles were then mixed with clean coke matrix material, and the fission-gas release was measured. The results are given in Table V.4. The matrix alone showed an R/B value of 0.47 for the long-lived krypton isotope Kr-85m, whereas the particles alone had a much lower release (a factor of about 10^4 lower). The R/B values increased by approximately a factor of four when the particles were mixed with clean matrix material.

It can be seen in Fig. V.5 that there is a correlation between the fission-gas release and uranium content of the matrix material. The data are not quite consistent with an expected slope of 45° in the given plot; however, there is some contribution to the release from the contamination on the surface of the fuel particles that will make the R/B values appear higher at the low uranium contamination levels in the matrix material.

The data in Fig. V.5 are sufficiently uncertain to be misleading in one respect. At the highest levels of contamination in the coke matrix material, R/B values for Kr-85m appear to be numerically equivalent to the fractional contamination values, suggesting, incorrectly, 100% release of Kr-85m from the matrix material. Other studies, as indicated above, have shown that the fractional release of Kr-85m from the matrix material is about 0.50 or less (see, for example, the fractional release value of 0.47 in Table V.4).

TABLE V.4
FISSION-GAS RELEASE FROM COATED PARTICLES AND COKE MATRIX
MATERIAL, SEPARATE AND COMBINED

Samples	Fractional Release, R/B					
	Kr-85m	Kr-88	Kr-87	Kr-89	Xe-138	Xe-139
Particles plus matrix (a)	1.3×10^{-2}	1.3×10^{-2}	8.4×10^{-3}	3.0×10^{-3}	7.4×10^{-4}	3.9×10^{-4}
Particles alone	2.3×10^{-5}	1.2×10^{-5}	1.3×10^{-5}	4.0×10^{-6}	3.6×10^{-6}	4.3×10^{-6}
Matrix alone	4.7×10^{-1}	4.5×10^{-1}	3.1×10^{-1}	8.9×10^{-2}	2.6×10^{-2}	8.6×10^{-3}
Particles plus new matrix	7.2×10^{-5}	7.5×10^{-5}	4.8×10^{-5}	1.3×10^{-5}	4.6×10^{-6}	8.3×10^{-6}

(a) Fraction of uranium in matrix was 0.017.

Fission-gas Release from Fuel Rod

The steady-state release of fission gases from a fuel rod sample of the type to be used in the Fort St. Vrain reactor has been measured in a long-term release experiment carried out in the TRIGA Mark III reactor utilizing a King (graphite tube) furnace and a loop-type system. The fuel rod was 2 in. long by 1/2 in. in diameter, contained TRISO coated particles, and was made using pitch binder and natural-flake graphite filler material. The rod was representative of those manufactured for use in the Peach Bottom fuel test elements. The manufacturing procedure included a heat-treatment at 1800°C for 1/2 hr.

The furnace was placed alongside the graphite-filled thermal column in the reactor where the flux was approximately 10^9 n/cm²-sec. The sample, held in a graphite crucible, was irradiated in this position. Power and temperature levels for the entire period of the run are given in Fig. V.6. As indicated in Fig. V.6, the flux levels changed in the early period of the experiment.

Helium at about 20 psi was circulated in the loop-type system using a circulating pump. The helium gas was passed over the sample, and released fission gases were carried into a liquid-nitrogen-cooled charcoal trap.

Due to the very rapid He flow, there was a considerable temperature gradient across the length of the graphite crucible (and sample). The average temperatures in the sample were therefore higher than indicated on the pyrometer. The reported temperatures are average temperatures and take into account the temperature gradient. The average temperature during a large portion of the run was approximately 1460°C. During the later part of the experiment the temperature was reduced to about 30°C (the pool temperature) and then raised to approximately 1600°C.

The helium flow rate was fast enough so that the delay time from the sample to the trapping system was less than 30 sec. During the run, flow

Fig. V.6. Power, temperature, and time history for experiment to measure long-term fission-gas release from fuel rod

rates changed by a factor of approximately five, apparently because of the plugging of the charcoal trap with water.

To establish the fission-gas release, the fission gases were collected in the liquid-nitrogen-cooled charcoal trap for a time period long enough for the gaseous activities to reach a steady state. The trap was then counted using Ge(Li) detectors in conjunction with an SDS Sigma II analyzer-computer. One detector was mounted so that the trap could be counted in place. The analyzer-computer processed the data and computed the activity (in dpm) for each fission-gas isotope.

The R/B value for each isotope was taken to be the ratio of the steady-state activity for the isotope in the trap and the total activity for the isotope. The activity for each isotope in the trap was calculated from the count-rate data and the known efficiency of the counting system. Total activity values for each isotope were calculated from (1) the activity of Zr-95 in the sample at the end of the experiment, (2) the power history of the reactor, (3) fission yields, (4) half-lives, and (5) the efficiency of the counting system.

The steady-state release (R/B) values were determined at 30°, 1470°, and 1590°C, and the data are given in Table V.5. As indicated in the table, the R/B values are very low. The values at 1470°C are comparable to the beginning-of-life values for GAIL IV, even though the fuel temperatures in GAIL IV were about 100° to 200°C lower. This indicates that the fission-gas release performance of the fuel rod was quite satisfactory.

The R/B data do not show a square-root-of-half-life dependence. The longer half-life isotopes exhibit essentially no half-life dependence (compare the R/B values for 5.3-d Xe-133 and 9.2-hr Xe-135 given in Table V.5). This is to be expected when the half-life becomes long compared to the diffusion time.

TABLE V.5
STEADY-STATE FISSION-GAS RELEASE DATA FOR FSV-TYPE FUEL ROD SAMPLE

Time from Start of Exp. (hr)	Average Sample Temp. (°C) (a)	Release Fraction, R/B						
		Kr-85m	Kr-87	Kr-88	Kr-89	Xe-133	Xe-135	Xe-138
171	1470	2.9×10^{-6}	2.6×10^{-6}	3.0×10^{-6}	1.4×10^{-6}	3.0×10^{-6}	2.8×10^{-6}	1.2×10^{-6}
230	30	$< 2 \times 10^{-7}$	(b)	(b)	(b)	(b)	(b)	(b)
285	1590	1.7×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	1.1×10^{-5}	1.3×10^{-5}	1.7×10^{-5}	6.4×10^{-6}

(a) Temperature uncertainty is $\pm 25^\circ\text{C}$.

(b) Too low to detect.

Temperature Dependence of Fission-gas Release

Measurements have been conducted on the effect of temperature on fission-gas release from BISO coated fuel particles mixed with coke matrix material. This admixture was designed to simulate a fabricated fuel rod. The particles and matrix were annealed at 1600°C for 200 hr before irradiation, and migration of uranium during the anneal led to contamination of the matrix with uranium. Preliminary data for measurements from room temperature to 1500°C are given in Table V.6. A plot of the data for Kr-85m fractional release versus $1/T$ (°K) is given in Fig. V.7. The shape of the curve is similar to that reported earlier (see earlier quarterly report GA-9875) for gaseous fission-product release from loose particles. A steady-state correction has not been applied to the data, since it was difficult to separate matrix and particle diffusional release into appropriate steady-state corrections.

Duplicate data points at 1200°C indicate high reproducibility of the measurements.

In a system consisting of coated particles in a bed of contaminated matrix, recoil and diffusional release from both matrix and fuel particles must be considered. A speculative model for the gas release from an ensemble of this type is as follows:

1. At room temperature, the release is controlled by recoil from fissions in the uranium contamination in both the matrix and the coatings.
2. At intermediate temperatures (800° to 1200°C), a diffusional component of the release becomes predominant.
3. At high temperatures (above 1200°C) the controlling diffusional mechanism may change. This is indicated by the apparent decrease in the slope of the release curve in the high-temperature region. Speculatively, diffusion in the matrix material may be controlling

TABLE V.6
EFFECT OF TEMPERATURE ON RELEASE FROM MIXTURE OF ANNEALED PARTICLES
AND COKE MATRIX MATERIAL^(a)

Sample No.	Irrad. Temp (°C)	$10^4/T$ (°K)	Release Fraction, R/B					
			Kr-85m	Kr-88	Kr-87	Kr-89	Xe-138	Xe-139
4770-4-11-1	RT	33.3	8.4×10^{-5}	9.1×10^{-5}	7.8×10^{-5}	6.5×10^{-5}	6.0×10^{-5}	4.5×10^{-5}
4770-4-11-2	RT	33.3	6.8×10^{-5}	7.6×10^{-5}	6.2×10^{-5}	5.5×10^{-5}	5.7×10^{-5}	5.3×10^{-5}
4770-4-11-3	842	8.97	2.2×10^{-4}	2.5×10^{-4}	2.2×10^{-4}	1.4×10^{-4}	1.1×10^{-4}	7.2×10^{-5}
4770-4-11-4	1029	7.68	4.2×10^{-4}	5.2×10^{-4}	5.0×10^{-4}	2.5×10^{-4}	9.8×10^{-5}	5.6×10^{-5}
4770-4-11-5	1133	7.11	2.3×10^{-3}	2.9×10^{-3}	2.7×10^{-3}	1.2×10^{-3}	2.6×10^{-4}	1.6×10^{-4}
4770-4-11-6	1216	6.72	5.0×10^{-3}	5.8×10^{-3}	5.4×10^{-3}	3.2×10^{-3}	5.5×10^{-3}	3.4×10^{-4}
4770-4-11-7	1205	6.76	5.3×10^{-3}	6.5×10^{-3}	5.8×10^{-3}	3.3×10^{-3}	6.0×10^{-4}	1.8×10^{-4}
4770-4-11-8	1369	6.09	9.1×10^{-3}	1.1×10^{-2}	9.7×10^{-3}	5.2×10^{-3}	1.8×10^{-3}	6.6×10^{-4}
4770-4-11-9	1504	5.63	1.3×10^{-2}	1.7×10^{-2}	1.4×10^{-2}	7.7×10^{-3}	3.8×10^{-3}	6.3×10^{-3}

(a) A steady-state correction was not applied to the data (see text).

Fig. V.7. Effect of temperature on fission-gas release from mixture of BISO coated particles and coke matrix material contaminated with uranium; the steady-state correction was not applied to the data (see text)

in an intermediate region, whereas diffusion from the particles becomes important at higher temperatures. The matrix diffusional process should saturate at least at a release corresponding to the fraction of uranium in the matrix, in this case 0.021.

TRIGA Mark I Fission-gas Release Measurements

Fission-gas release (R/B) measurements are routinely performed utilizing a King-type furnace mounted in the TRIGA Mark I reactor. The procedure formerly used for determining the fission gas release values is described in an earlier quarterly report (GA-8879).

An alternative improved method of determining the release values utilizing a Ge(Li) solid-state detector in conjunction with the SDS Sigma II 4096-channel analyzer-computer has been developed. The procedure is reliable, and a computer program has been developed to give fractional release (R/B) values for the various krypton and xenon isotopes of interest directly from the Sigma II computer.

REFERENCES

1. Zumwalt, L. R., P. E. Gethard, and E. E. Anderson, Nucl. Sci. Engr. 21, 1 (1965).