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AEC RESEARCH AND
DEVELOPMENT REPORT

GA-6950

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

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
DECEMBER 31, 1965

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QUARTERLY REPORT SERIES

GA-6830—July, 1965, through September, 1965

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; this work includes the development of 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
Reactor Vessel	1
PCRV Model Tests	1
Reactor Thermal Barrier	3
Class A Thermal Barrier	3
Metal-Impure Coolant Compatibility Test	3
Nonmetallic Structural Materials	3
Primary Coolant System Flow	5
Compressors and Auxiliary Circulators	5
Compressors	5
Auxiliary Circulators	6
Valves	7
Steam-Water Dump Valve	7
Helium Service Valves	13
Instrumentation and Electrical	13
Moisture Detectors	13
Electrical Penetrations	13
TASK II. NUCLEAR ANALYSIS	15
Fuel Cycle Analysis	15
Over-all Core Dimensions	15
Maximum Fuel Burnup	16
Fuel Zoning and Power Distribution	16
Kinetics Analysis	20
Analytical Studies	20
Fuel Transfer Machine--Criticality Safety	20
TASK III. FUEL DEVELOPMENT	22
Design	22
Mechanical Design	22
Thermal Design	29
Fission-product Transport	45
Materials	50
Particle Beds	50

Bonded Particle Systems	54
Graphite Development	55
Control Materials.	59
Chemistry	66
Fission-product Release	66
Sorption and Diffusion Products	67

Part II

TASK I. FUEL TRANSFER MACHINE.	87
TASK II. SERIES-STEAM-TURBINE-DRIVEN CIRCULATOR	88
Helium Circulator Development	88
Rotor Dynamic Simulation and Lubrication and	
Seal System Test	90
Test Program Outline	93
Static Aerodynamic Components	94
Prototype Test Facility	94
TASK III. CONTROL ROD DRIVE.	95
TASK IV. STEAM GENERATOR	97
Materials	97
Heat Transfer and Fluid Flow.	97
Steam/Water Flow Stability	97
Tube Vibration Studies	98
Design and Analysis	98
Steam Generator Tube Plugging Machine	98
TASK V. COATED PARTICLE DEVELOPMENT	101
Buffer Coating Studies	101
Deposition Studies	101
Diffusivity Measurements	101
High-density Isotropic Coatings	102
Coated Particle Preparation.	102
Dounreay Fast Reactor Irradiation	103
Capsule P14	103
ORNL Irradiations	103
Capsule G11.	105
Coating Evaluation Techniques	105
X-ray Absorption Density Determination	105
Electron Microprobe Density Determination	106
Particle Geometry (T/D) Studies.	106
Thermal Stability Studies.	106
Fission-product Release	107

PART I

TASK I
COMPONENT DEVELOPMENT

REACTOR VESSEL

PCRV Model Tests

The contract for construction work exclusive of tendon installation and tensioning on PCRV Model No. 2 was let to the Robert E. McKee Company. Following completion of the bottom head form work, the liner, which had previously been instrumented, was delivered and installed in position. All the vertical tendon tubes along with the reinforcing steel and the bottom mat of the bottom head, the horizontal tendon tubes, and horizontal reinforcing steel to the elevation of the lower across-the-head tendons were put in place. The SEEE prestressing equipment and material arrived completing the delivery of all material necessary for construction of the model.

The various types of instrument sensors have been selected, and installation of those that are located within the concrete is almost complete. The types of sensors selected include:

1. Electric resistance strain gauges (1/8 in. to 1/2 in. gauge length).
2. Carlson strain meters for certain critical locations. These are individually temperature compensated.
3. Vibrating-wire strain gauges. All are installed so that they can also be used to determine temperature and a portion of them are so wired that temperature is always measured.
4. Load cells. Some are calibrated for use in measuring loads at 24 tendon ends.
5. Linear potentiometers for measuring deformations.
6. Resistance-type moisture monitors. To measure the relative change in moisture content of the concrete.
7. Thermocouples.

One of the problems associated with the use of electric resistance-type strain gauges is that they are subject to drift resulting from adhesive creep, which is a function of time, temperature, and strain. To reduce

this effect the installation procedures were quite elaborate and involved careful temperature control during curing of the adhesive. In key locations, gauges that are not significantly affected by temperature have been installed to supplement the electric resistance type, although these gauges are not available in the gauge lengths most desirable for locations where steep strain gradients are anticipated. The total number of instrument channels for the liner and its penetrations is about 760 and the associated number of locations is 393.

Studies have been made to predict stress distributions with various combinations of loads exerted by tendons combined with pressure and other load conditions. This information was used to complete the design of the conventional reinforcing. Prestressing loads were analyzed for their influence on the vessel. A series of prestressing combinations and varied pressure and thermal loading conditions were calculated and stress contours plotted. Stress calculations have started for a proposed sequence of 18 jacking combinations.

Aggregate selection tests have been made and trial mixes have been prepared for both premixed and preplaced aggregate concrete. For premixed concrete, trial mixes have included aggregate with 3/4 in. top size and 1-1/2 in. top size, respectively, while similar designs for preplaced aggregate concrete include aggregate with one gradation only. Trial mixes for premixed concrete are designed for a range of 28-day strengths between 5500 and 6500 psi.

REACTOR THERMAL BARRIER

Class A Thermal Barrier

A partial review of literature applicable to natural convection into the control rod penetration has been completed. Correlations have been applied to the configuration without the plug at the lower end of the nozzle to give an upper limit on heat-transfer requirements for the penetration cooling system. The resulting heat flux distribution to the cooling system is shown in Fig. 1.1.

Metal-Impure Coolant Compatibility Test

Based on data reported in the previous quarterly report,* the testing was modified as follows:

1. The exposure of Hastelloy X and Stainless Steel 430 at 1650° and 1850°F will be continued for 1500 more hours to provide additional data on compatibility.
2. Four exposed specimens of Hastelloy X have been removed from the experiment for creep testing to provide data on effect of exposure on creep strength. Two of these specimens have been exposed at 1650°F and will be tested at 1650°F. The other specimens have been exposed at 1850°F and will be tested at 1850°F. Stress level is 2800 psi at 1650°F and 850 psi at 1850°F. These stresses should give 0.5% total elongation in 1000 hr. These specimens are now under load.
3. Exposure and testing of other alloys (René 41, Inconel X, Incoloy 800, and Stainless Steel 304) has been discontinued, since these alloys have exhibited undesirable surface attack, structural changes, or changes in strength properties.

Nonmetallic Structural Materials

Work was started on a test program to determine the thermal and mechanical properties of potential nonmetallic materials. Representative alumina, ceramic, and carbon bricks, in both dense and porous form, were obtained, and compressive and modulus-of-rupture strength testing at room temperature was started.

* Public Service Company of Colorado 330-Mw(e) High-temperature Gas-cooled Reactor Research and Development Program, Quarterly Progress Report for the Period Ending September 30, 1965, General Atomic Report GA-6830, December 31, 1965.

Fig. 1.1--Reference design heat-transfer data with no plug at lower end of nozzle

An abrasion and friction test rig was designed, constructed, and assembled, and checkout is now in progress. This rig will be used to determine abrasion rates and coefficients of friction in dry helium at temperatures up to 1400°F.

PRIMARY COOLANT SYSTEM FLOW

Conceptual design for the primary coolant circuit 0.3-scale flow model has been completed. The detail design of the main shell of the model, the arrangements for applying and measuring air flow to the model, and the means for supporting the model internals is almost complete. Procurement of all long-delivery-time items for the flow test program (e.g., blowers, valves, venturis, and instrumentation) is under way.

The first region to be tested is that from the core outlet to the steam generator inlets. Flow analysis of this region has led to a decision on the kind of configurations to be tested. Detail design of the test components for this region is under way.

COMPRESSORS AND AUXILIARY CIRCULATORS

Compressors

Purified Helium Compressor. The purified helium compressor returns the effluent from the helium purification system to the primary coolant system. The machine required is a relatively small, low-flow, high-head gas compressor. The operating conditions have been established as follows:

Flow, lb/hr	900
Suction volume, acfm	35
Suction pressure, psia	650
Suction temperature, °F	107
Discharge pressure, psia	680

In addition, the compressor should not contaminate the process gas and must have a high degree of reliability.

The type of compressor that appears best suited for this service is a small rotating compressor, either centrifugal or regenerative. A gas-bearing machine similar to those built by Rateau or Mechanical Technology, Incorporated, is preferred, but a grease-lubricated compressor, similar to the HECT-2 regenerative compressor built at Oak Ridge, might also be suitable.

An inquiry specification for the prototype compressor is being prepared. This will be used to review the application with potential vendors, so that the type of compressor to be obtained for testing can be selected.

Auxiliary Circulators

Conceptual design studies have been made on the auxiliary circulators whose function is to provide for primary gas circulation when the main steam-turbine-driven circulators are unavailable, either through scheduled or unscheduled shutdowns. Two methods for driving the machine are being examined, and additional work will be required before a selection can be made. Variable speed operation of the auxiliary circulators is required, because wide variations in gas density are possible. Furthermore, the limitation on size imposed by its location within the PCRV and the low molecular weight of helium coolant require high speed operation.

The two types of drive being considered are a high-frequency electric motor and a water turbine of the Francis type. In either case, the torque-speed characteristics of these machines, which have conventional external control systems, allow the fulfillment of the torque-speed requirements imposed by the design conditions. The maximum circulator speed requirement is tentatively established by emergency core cooling requirements at atmospheric helium pressure. A steam turbine drive is not being considered, because of the possible unavailability of steam.

Three circulator design variations have been examined: two have the same blade diameter but different diffusers; the third is a smaller machine. In the first case, the diffuser discharge diameter is larger than the diameter of the PCRV penetration; in the second, the diffuser diameter is limited to 27 in., which is 3 in. less than the penetration diameter to provide for diffuser removal; and in the third case, the design is similar to but smaller than the first so that the diffuser can be removed with the machine. In all cases, a total of four circulators would be provided for the plant, with two in each primary coolant loop.

Since the capacity of an individual compressor at constant speed will vary with the restriction to flow that the system imposes, which in turn is dependent on the total system flow, the individual circulator will, at a given speed, have a different flow depending on the total number of circulators that are operating. This effect is shown in Fig. 1.2 for high- and low-pressure operation. For low-pressure operation, line 1 shows the case in which all four circulators are operating; line 2 shows two circulators operating in one loop; and line 3 shows the case for a single circulator in operation. For high-pressure operation, line 4 shows the results of two circulators operating, and line 5 shows one circulator operating.

Flow rates and power requirements are shown in Table 1.1 for various numbers of circulators operating. Design case No. 2 has been selected as the reference for the preliminary design, since it is highly desirable to be able to remove the diffuser through the PCRV penetration. The compressor characteristic for various tip speeds of this machine are also shown on Fig. 1.2. The torque-speed requirements are shown in Fig. 1.3, and the operating points are further delineated in Table 1.2. The overspeed point (point C) is included, because it is planned to investigate the feasibility of satisfying this condition, although at this time it is not a requirement for the circulator.

As was mentioned earlier, the choice of driver for the compressor lies between a water turbine and a high-frequency electric motor. Preliminary calculations for the designs of each have been made, those for the turbine by General Atomic and those for the motor by a vendor. Either choice is feasible and within the state of the art for the lower design speed, but the overspeed point still remains to be investigated. The preliminary designs are being pursued and installation drawings will be made.

A summary of aerodynamic design data for the compressor is given in Tables 1.3 and 1.4. The design data for the water turbine are given in Table 1.5.

The water-turbine design results in a very small machine that should lead to a relatively simple mechanical design. The flow through the turbine accelerates both in the velocity relative to the blades and in the meridional velocity. The question of cavitation that normally arises with a hydraulic turbine should not be a serious problem here, because the water supply system can be pressurized as required.

VALVES

Steam-Water Dump Valve

A preliminary computer program, WHOOSH, has been written to simulate the conditions during the steam dump. Assuming eight modules per steam generator and one steam generator per loop with two parallel dump lines from the economizer-evaporator header, 6-in. dump valves in 10-in. lines appeared optimum. Under these conditions, 3900 lb out of a total inventory of 4100 lb were dumped within 15 sec after receipt of the signal from the moisture monitors.

The program has been modified to compute the steam in-leakage to the primary system during the dump and to account for the restriction

Fig. 1.2--Plot of compressor and system pressure rise ratio versus normalized mass flow rate for the auxiliary circulator

Fig. 1.3. --Torque versus speed for five operating cases

Table 1.1
OPERATING CONDITIONS
(Design Case No. 2)

No. Circulators Operating	Total Flow (lb/hr)	Speed (rpm)	Power per Circulator (hp)	Blade Stress (psi)	Tip Speed (ft/sec)
4	95,200	11,035	212	20,390	1,300
2 (same loop)	51,600	11,035	203	20,390	1,300
1	31,600	11,035	162	20,390	1,300
1 (overspeed)	45,000	15,550	442	40,300	1,830
2 (same loop)	45,000	9,670	138	15,900	1,150
4	80,000	9,350	128	14,600	1,100

Table 1.2
COMPRESSOR OPERATING CASES

	Line ^a 1	Line ^a 2	Line ^a 3	Line ^a 4	Line ^a 5
Pressure, psia	12	12	12	370	370
Point ^a	d	e	c	b	a
No. of circulators operating	4	2	1	2	1
Temperature, °F	250	250	250	250	250
Mass flow rate per circulator, lb/hr	23,800	25,800	45,000	150,000	300,000
Compressor pressure rise, psia	0.666	0.589	0.636	0.67	0.67
Power per circulator, hp	212	203	442	42	123
Speed, rpm	11035	11035	15550	2115	3400
Tip speed, ft/sec	1300	1300	1830	248	400

^a—Line and point numbers refer to Fig. 1.3.

Table 1.3

HELIUM COMPRESSOR DESIGN

Inlet temperature	250°F
Inlet pressure	12.0 psia
Outlet pressure	12.666 psia
Blade tip speed	1300 ft/sec
Gas through-flow velocity	650 ft/sec
Mass flow rate	6.61 lb/sec
Outer diameter	2.25 ft
Diffuser efficiency	0.8
Diffuser area ratio	0.405

Table 1.4

COMPRESSOR BLADE DATA

	Stator			Rotor		
	Tip	Mean	Hub	Tip	Mean	Hub
Inlet angle	34.3	37.6	41.5	63.4	60.6	57.0
Outlet angle	0.0	0.0	0.0	52.8	45.0	37.4
Solidity	1.20	1.35	1.55	1.05	1.26	1.55
Diffusion factor	0.409	0.433	0.464	0.405	0.454	0.503

NOTE: Total efficiency, 0.836
 Total torque, 102 ft-lb
 Horsepower, 212 hp
 Rotor blade centrifugal stress, 20,392 psi
 Rotor blade bending stress, 1000 psi

Table 1.5

FRANCIS TURBINE DESIGN SUMMARY

Flow:	
W	103 lb/sec
Q	1.65 cfs
Head:	
H	1280 ft
P	554 psi
Power	216 hp
Tip diameter	4 in.
Tip width	0.353 in.
Draft tube entrance diameter	2.17 in.
Draft tube area ratio	4
Tip speed	193 ft/sec
Meridional entrance velocity	52 ft/sec
Meridional exit velocity	61.2 ft/sec
Nozzle angle	15°
Discharge blade angle:	
Tip	27.5°
Mean	34.5°
Root	51.5°

imposed by the flow-limiting orifices that will be included in each steam generator tube. Program runs will also take into account any changes in primary system pressure that will affect the pressure at which the dump is terminated. These results may indicate that valves smaller than 6 in. are optimum.

Based on the results of the revised runs, an inquiry specification will be prepared to obtain proposals on commercially available valves for the dump application.

Investigations have been made to determine whether or not existing facilities are available for the dump tests. The Mishawaka Test Facility does not have sufficient capacity. The facilities proposed for the compressor tests at the Public Service Company of Colorado's Valmont Station are limited to 350 psi steam, which is not adequate for the dump valve tests. The test facilities of Battelle-Northwest at Hanford, Washington, appear adequate from a capacity standpoint, as well as from the standpoint of temperature and pressure. In addition, a wide range of high-speed instrumentation is available to record temperatures, pressures, flows, and strain-gauge measurements during the tests.

Helium Service Valves

A tentative test program comprising static life tests and operational life tests on standard bellows-sealed valves was prepared. Based on the preliminary program, an inquiry specification has been prepared to solicit proposals for the test program. Proposals will be requested from valve vendors who produce a recognized line of bellows-sealed valves and from General Atomic. These competitive bids will form the basis for a determination as to where the tests will be run.

INSTRUMENTATION AND ELECTRICAL

Moisture Detectors

A potential vendor for the dew-point detectors was contacted to discuss the possibility of purchasing a detector which had been developed for a space program. It was found that the design of the detector was not adaptable to PSC duty. Consequently, work at General Atomic was continued on the design and development of a suitable detector.

ELECTRICAL PENETRATIONS

A preliminary survey was made of known electrical cable penetrations of the PCRV to establish the criteria of an electrical nature that will influence the planning for this work. However, the requirements for all of the electrical penetrations have not been defined at the present time.

Fig. 2.1. --Relative fuel cycle cost versus leakage; core power density = 6.25 w/cm^3

TASK II

NUCLEAR ANALYSIS

FUEL CYCLE ANALYSIS

Over-all Core Dimensions

One method for reducing the fuel and coolant temperatures resulting from transverse power peaking is to orifice the core so that the hottest fuel receives the most cooling and the oldest fuel receives the least. Orificing requires either that the pumping power be increased or that the pressure drop in the rest of the system be decreased to accommodate the drop across the orifices. Since the pressure drop in the core varies rapidly with the length-to-diameter ratio (L/D), cores with smaller L/D have been studied.

Two fuel core configurations with reduced L/D have been considered; (1) the addition of one ring of blocks around the periphery and (2) the addition of one ring of refueling regions. The present L/D is 0.8 and the above modifications lead to L/D values of 0.58 and 0.38, respectively. All other things being equal, these changes will increase the over-all leakage and the fuel cycle cost as shown in Fig. 2.1. The axial fuel zoning in all cases was chosen to yield an exponential power distribution which leads to increased leakage in the axial direction. The required uranium and thorium distributions are strongly skewed toward the inlet end and result in some quite heavy loadings at the top of the core. A careful look at the U^{235} cross sections indicates that the value of η drops off very rapidly at carbon to U^{235} ratios (C/U) below 1500. Above this value, the variations are fairly linear. The steep axial power distribution requires fuel zones with C/U in the vicinity of the rapid drop in η . Further, it is difficult to accommodate the required volume of fuel particles, with their present specifications, in the fuel element in the heavily loaded zones. If orificing were used with the present core dimensions and fuel zoning, the maximum fuel temperature is reduced by about 300°F . The elimination of all axial fuel zoning results in a skewed, chopped, cosine with its maximum value below the core midplane. The leakage and fuel cycle costs are both reduced as shown in Fig. 2.1; however, the reduction in the maximum fuel temperature is essentially eliminated. In the coming quarter a modest degree of axial fuel zoning will be considered; it appears feasible to reduce the maximum fuel temperature by 100° to 200°F compared to the no-axial-zoning case, without materially changing either the leakage or the fuel cycle cost.

Maximum Fuel Burnup

The fuel zoning will be accomplished by blending two types of PyC-coated fuel particles-- $(\text{Th}, \text{U})\text{C}_2$ and ThC_2 . The coating characteristics and tests are designed for a maximum of 20% metal atom fission in the Th,U particles and 6% in the Th particles. The fractional fissions in the Th,U particles can be controlled within rather wide limits by the thorium-to-uranium ratio, which is currently set at 3.5. Figure 2.2 shows the maximum fractional fission as a function of irradiation time in the equilibrium core with axial and radial zoning. With the current power distributions, it seems clear that the Th/U should be increased to 4.0; however, it is advantageous for fuel cycle costs, fuel fabrication, and fuel reprocessing to maintain the Th/U as low as possible. This point must be reevaluated with the flatter axial power distribution if it is adopted.

FUEL ZONING AND POWER DISTRIBUTION

The axial and radial distribution of the fissile and fertile material in the equilibrium core was described in the previous quarterly report (GA-6830). The criteria for choosing the distribution are: (1) a flat power profile radially, (2) an exponential power profile axially, and (3) stability of the power distribution during the six-year fuel residence time. The power distributions proved satisfactorily stable, with the greatest variation occurring during the first year owing to the burnout of the unshielded burnable poison.

Two dimensional (R-Z), multigroup diffusion calculations were performed to determine the validity of the method for convoluting the axial and radial fuel zoning. First the nominal composition is zoned radially for a stable, flat power profile with up to five radial zones. These zones correspond to the region boundaries, with the exception of the outermost, which includes only two rings of fuel holes. The radial leakage for each of these zones is computed as a function of energy. Axial zoning is then performed using the nominal feed and radial leakage corresponding to each radial zone. Up to five axial zones are considered, with the zone boundaries corresponding to integral numbers of fuel blocks. Figures 2.3 and 2.4 show the axial power distributions at the beginning of an equilibrium cycle, with a fresh central region, at two radial positions with and without the central control rod pair inserted 2/3 of its length. The only significant deviations from the desired power distribution occur near the inlet end and are characterized by very steep spikes. It is not clear that this power distribution is not acceptable, since axial conduction in the fuel element will tend to flatten the temperature distribution. In any case, the power distribution will be flattened by reducing the loading in the first axial zone. The discontinuities shown within the axial fuel zones in Fig. 2.4 are purely

Fig. 2.2. --Maximum fractional burnup

Fig. 2. 3. --Axial power distribution on core centerline at beginning of an equilibrium cycle

Fig. 2.4. --Axial power distribution in ring 3 at beginning of an equilibrium cycle

calculational and result from splitting the zones when performing the depletion calculations. The effect of partially inserting the central control rod pair is to reduce the power markedly only in the vicinity of the rod pair. The exponential axial power distribution is stable with partial control rod insertion. Figure 2.5 shows the radial power distribution 250 cm from the coolant inlet. Again, it can be seen that the desired flat power distribution was achieved and that the effect of the inserted control rod pair is a very strong reduction in power locally. The perturbation is essentially dissipated within the adjacent regions, and a slight compensating increase in power is evident over the remainder of the core.

KINETICS ANALYSIS

The reference design calls for 37 pairs of 3-in.-OD (poison section) hollow cylindrical rods containing 30 wt-% B₄C in graphite. If a larger core diameter is considered, the number of control rod pairs will have to be raised. The addition of one row of fuel blocks requires 12 additional 3-in. rod pairs, and the addition of one row of fuel regions requires 24 additional rod pairs to obtain a sufficiently strong control system and maintain the necessary symmetry.

The core diameter increase could also be achieved by increasing the size of a fuel block from 14.2 in. across flats to 14.9 in. and 15.7 in., respectively. In either case, the core layout would be as shown in the previous quarterly report, and the number of control rod pairs would be limited to 37. In this case the control rod diameter would have to be increased to 3.6 in. OD and 4.5 in. OD, respectively. In either case, the ID of the hollow poison section would be correspondingly increased to maintain the weight of the rod constant. This appears satisfactory because of the high content of absorber in the rods.

ANALYTICAL STUDIES

Fuel Transfer Machine--Criticality Safety

The criticality safety of a proposed design for the fuel transfer machine has been determined using the following conservative assumptions: (1) the machine is loaded with the most reactive fuel blocks in the reactor and (2) the machine is infinitely long.

Under these conditions, the computed multiplication constant of the machine is 0.5 without water flooding. While flooding would increase this value, the design of the machine makes flooding inconceivable and it was therefore not considered.

Fig. 2.5. --Radial Power distribution 250 cm from coolant inlet at beginning cycle

TASK III

FUEL DEVELOPMENT

DESIGN

Mechanical Design

The phenomenon of graphite contraction as a function of fast-neutron dose and irradiation temperature is expected to produce internal tensile and compressive strains and to cause a net volume change in the fuel element with time. A method has been developed to predict the amount of axial and radial contraction that will occur within the fuel block as a function of its in-core irradiation history. At the present time, the calculated contraction values are considered good engineering predictions but not exact final values. This is due to the uncertainty of some of the graphite contraction data, to the use of preliminary fuel block temperature profiles, and to the use of a preliminary analytical evaluation model. Contraction of the fuel block has been examined in both the axial and radial direction and is discussed below.

Engineering graphite contraction maps have been constructed for use in the contraction analysis. The maps are plots of percent graphite contraction as a function of temperature for lines of constant fast-neutron dose. The maps have been constructed from the available experimental data, and extrapolations have been made to fill in the areas where there are no data. The contraction maps are constantly being reviewed and are modified whenever new data become available. Above 800°C, the quantity of data is presently limited and considerable extrapolation is necessary to define a complete contraction map. The present reference maps being used are Map 3A (see Fig. 3.1) for axial, or parallel, needle-coke graphite contraction and Map 3R (see Fig. 3.2) for radial, or perpendicular, needle-coke graphite contraction.

Map 3A was incorporated into the CONCAL graphite parallel contraction analysis code, and a new set of axial, or parallel, contraction values were calculated. The code calculates the total contraction, tensile and compressive strain, and the tensile strain rate of the fuel element block as a function of time, local fuel block graphite temperature, and temperature gradients and local fast-flux doses. At the present time, a code has not been developed for the analysis of perpendicular, or radial, graphite contraction. This analysis is performed by hand by graphic

Fig. 3.1. --Percent contraction (parallel) versus temperature as a function of total fast-neutron dose for needle-coke graphite (Map 3A)

Fig. 3.2. --Percent contraction (perpendicular) versus temperature as a function of total fast-neutron dose for needle-coke graphite (Map 3R)

plotting on the 3R map. This procedure predicts over-all radial contraction but does not evaluate local radial strain distributions or strain rates. Past evaluations of local radial contraction have indicated that tensile and compressive strains should be very low and are smaller than axial contraction strains. This, however, will have to be reexamined and evaluated in the future to verify the above conclusion.

The axial contraction analysis was performed at specific areas in the core at which maximum contraction effects were expected to occur. Four axial plane locations were selected: the 1.0-, 3.5-, 60-, and 11.5-ft planes (distance from core inlet). The six-year temperature and fast-flux dose at these locations were calculated and used in the contraction analysis. The results are plotted on Figs. 3.3, 3.4, 3.5, and 3.6. The maximum predicted tensile strain occurs at the 1.0-ft plane and has a value of 0.25%. The tensile strains appear to reach their maximum and saturate after about three years of operation in all four of the planes examined. The maximum predicted tensile strain rate is $0.0115\%/10^{20}$ nvt. The maximum tensile strain and tensile strain rates are well below any maximum values that have presently been indicated experimentally. It has been reported by J. L. Jackson* that TS-688 parallel needle-coke graphite irradiated at 625°C at a constant strain rate of $0.13\%/10^{20}$ nvt has displayed a total tensile strain of 0.37% to 0.42% without failure. Since the samples were not broken at the end of the test, the strain limit of that graphite must be above the 0.42% value.

The maximum axial contraction of the fuel block at the end of six years is predicted to be about 2.7%. Based on an analysis of four separate planes along the length of the core, an estimate has been made of a maximum change in column height with time. The active core height of 187.3 in. is reduced by 4.2 in. in six years. The change in column height with time is shown on Fig. 3.7.

An estimate of the radial contraction of the fuel blocks has also been made. The effect of the radial contraction is to reduce the width across the flats of the block, thereby increasing the effective gap between adjacent fuel columns in the core as a function of time and core location. The total net gap for design purposes is a function of (1) the cold assembly clearance between the blocks, (2) the core support thermal expansion, (3) the fuel block radial thermal expansion, and (4) the fuel block radial graphite contraction. The predicted radial contractions across the flats at the end of six years are given in Table 3.1. The net effective gaps due to the contraction and other associated design factors are presently being evaluated.

*J. L. Jackson, The Tensile Fracture Strain of Graphite Determined During Neutron Irradiation, Report BN-SA-56, May 14, 1965.

Fig. 3.3. --Axial graphite contraction versus time at 1.0-ft plane in ring 3

Fig. 3.4. --Axial graphite contraction versus time at 3.5-ft plane in ring 3

Fig. 3.5--Axial graphite contraction versus time at 6.0-ft plane in ring 3

Fig. 3.6.--Axial graphite contraction versus time at 11.5-ft plane in ring 3

Fig. 3.7--Predicted contraction of fuel element column versus core lifetime

Table 3.1
FUEL BLOCK RADIAL GRAPHITE CONTRACTION
AT THE END OF SIX YEARS

Distance from Core Inlet (ft)	Expected Contraction ^a (in.)
0-2.6	0.13
2.6-5.2	0.12
5.2-7.8	0.10
7.8-10.4	0.08
10.4-13.0	0.08
13.0-15.6	0.06

^aDistance across flats.

A study has been initiated to reexamine the fuel-coolant hole patterns to facilitate the center fuel handling hole and determine whether or not a more optimum configuration can be obtained. Consideration is also being given to directing a portion of the coolant along the outside edges of the fuel block. Four specific fuel-coolant hole arrangements have been selected for more detailed thermal, mechanical, and contraction performance analysis. The four selected arrangements are:

1. Ratio of 3:1 fuel to coolant holes with no edge cooling (see (see Fig. 3.8).
2. Ratio of 3:1 fuel to coolant holes with edge cooling (see Fig. 3.9).
3. Ratio of 2:1 fuel to coolant holes with no edge cooling (see Fig. 3.10).
4. Ratio of 2:1 fuel to coolant holes with edge cooling (see Fig. 3.11).

The hole pattern layouts are all based on using the same fuel loading. Some parameters concerning the various layouts are listed in Table 3.2.

Thermal Design

A study was undertaken to determine the performance advantages of orificing the PSC reactor core to (1) produce a more nearly uniform core exit coolant temperature distribution and (2) to reduce the effects

Fig. 3.8. --Partial cross section of a hexagonal-block fuel element having three fuel holes per coolant hole and no edge cooling

Fig. 3.9--Partial cross section of an hexagonal-block fuel element having three fuel holes per coolant hole and edge cooling

Fig. 3.10--Partial cross section of an hexagonal-block fuel element having two fuel holes per coolant hole and no edge cooling

Fig. 3.11--Partial cross section of an hexagonal-block fuel element having two fuel holes per coolant hole and edge cooling

Table 3.2

FUEL ELEMENT FUEL-COOLANT HOLE ARRANGEMENT PARAMETERS

Fuel Element Layout	(A) ^a Fuel Hole Diam (in.)	Large Coolant Hole Diam (in.)	Small Coolant Hole Diam (in.)	Edge Coolant Hole Radius (in.)	Number of Fuel Holes	No. of Internal Coolant Holes	No. of Edge Coolant Holes	(B) ^a Min. Fuel Coolant Hole Web Thickness (in.)	(C) ^a Min. Fuel Hole Web Thickness (in.)	Hole-to-Hole Pitch (in.)
3.8	0.518	0.650	0.550	---	234	90	---	0.172	0.240	0.757
3.9	0.554	0.687	---	0.292	204	60	30	0.197	0.264	0.820
3.10	0.539	0.580	0.492	---	216	108	---	0.198	0.219	0.758
3.11	0.539	0.580	0.492	0.246	216	108	24	0.181	0.202	0.741

^a—See dimensions A, B, and C on Figs. 3.8 through 3.11.

of region age peaking on maximum fuel temperatures. After some initial screening, the study was confined to comparing the performance of the reference 19.5-ft diameter core (15.5-ft active core length) with a 21.75-ft diameter core. The latter was obtained by adding a ring of fuel columns around the 19.5-ft reference core and shortening the active height to 12.6 ft. The flattened core geometry was included in the study to evaluate the possible gains from reduced core pressure drop with orificing and to determine whether the effects of coolant crossflow from un-orificed peak power regions into older severely orificed adjacent regions are reduced with a flattened core geometry. Each region of elements was orificed as a unit, and orifice positions were determined for equal gas exit temperature from each region without considering crossflow. Then, while maintaining these orifice settings, the effects of crossflow between adjacent fuel columns were determined for various assumed gap sizes between the blocks. Two kinds of axial power distribution were studied, the reference five-zone exponential profile and a cosine profile. The conclusions drawn from the results, which are shown in Table 3.3, are as follows:

1. Core temperature conditions, with or without orificing, are not improved by flattening the core.
2. The pressure drop penalty for orificing is less for the larger-diameter core, but increasing the system pressure to 680 psia will accommodate orificing the reference core with no pumping power penalty.
3. Thermal performance of the orificed core with a cosine power distribution (uniform axial loading) is more sensitive to coolant crossflow loss from the hot channel than the exponential power shape because a larger fraction of power is produced at greater axial distance from the channel inlet.
4. The minimum fuel cycle cost is obtained by orificing the reference 19.5-ft-diameter core and utilizing the cosine axial power distribution. (Maximum fuel temperature is unacceptably high for the cosine distribution without orificing.)
5. A slight increase in coolant voidage does not affect the thermal performance. Although primary circuit pumping power can be reduced by increasing the coolant channel voidage, these gains must be compared to the effects on fuel cycle costs of reducing the graphite moderator density within the core. Also, the structural ruggedness of the fuel element block is reduced as coolant channel voidage is increased.

Results were obtained using the core performance code COPE, a program for analyzing a multichannel gas-cooled reactor core. The code has options for investigating an un-orificed core as well as one that is

Table 3 3
PSC ORIFICED CORE DESIGN STUDY

Case	Core Diam (ft)	Axial Power Profile	Sys Press. (psia)	Coolant Voidage (%)	Core ΔP (psi)	Orificed	Gap Height For Crossflow (in.)	Avg Exit Coolant Temp Hot Region ($^{\circ}\text{F}$)	Max Fuel Temp Edge of Hot Region ($^{\circ}\text{F}$)	Cycle Cost Diff From Ref Design (mill/kw-hr)	Remarks
1	19 5	Exp 5 zone	575	16 4	3 9	No	---	1995	2435	0	Ref design
3	19 5	Exp 5 zone	575	16 4	8 1	Yes	0	1482	1960	0	Above design pumping power
6	19 5	Exp 5 zone	680	16 4	6 9	Yes	0	1482	1960	0	Constant reference pumping power
6B-4	19 5	Exp 5 zone	680	16 4	4 6	Yes	0.02	1620	2090	0	Max crossflow gap analyzed
7	19 5	Cosine	575	16 4	3 7	No	---	1945	2797	---	Max fuel temp unacceptable
12	19 5	Cosine	680	16 4	6 5	Yes	0	1483	2185	-0 03	Const ref pumping power
12 B-4	19 5	Cosine	680	16 4	4 4	Yes	0 02	1653	2416	-0 03	Bypass reduces allowable $P/\bar{P}$
13	21 75	Exp 5 zone	575	16 4	2 3	No	---	1963	2484	+0 02	Below ref pumping power
17	21 75	Exp 5 zone	595	16 4	4 6	Yes	0	1482	1989	+0.02	Ref pumping power
17 B-4	21 75	Exp 5 zone	595	16 4	3 0	Yes	0 02	1641	2142	+0 02	Max crossflow gap analyzed
18	21.75	Cosine	575	16 4	2.2	No	---	1950	2845	---	Max fuel temp unacceptable
22	21 75	Cosine	595	16 4	4 4	Yes	0	1484	2212	-0 01	Ref pumping power
22 B-4	21 75	Cosine	595	16 4	3 0	Yes	0.02	1651	2453	-0 01	$P/\bar{P}$ max 1.49 for ref max fuel temp

orificed for equal gas exit temperature in all channels. Crossflow between adjacent channels is, however, not considered by the COPE code. Separate analyses were conducted to determine the effect of crossflow on the overall core pressure drop and on the local coolant and fuel temperatures for the orificed core.

The core model used in the input description for COPE consists of 37 flow channels, one for each radial power region. The 37 regions are strictly applicable only to the 19.5-ft-diameter core, but it is assumed that for the larger diameter core the same 37 power regions occur, i. e., all regions are thought of as having expanded to fill up the larger core area. A flow channel is equal to one coolant hole in the hexagonal block. All calculations were based upon the fuel element design having two fuel holes per coolant hole shown in Fig. 3.10. The void fraction is 16.4% for a coolant hole diameter of 0.580 in. and 18% for a coolant hole dimension of 0.608 in.

The code computes the coolant, channel wall, and maximum fuel temperatures at a selected number of axial locations. However, it does not consider axial conduction in the element. The exact temperature distribution in a plane perpendicular to the coolant hole axis can only be obtained by considering two-dimensional heat flow. For the purpose of calculating fuel temperatures with COPE, a modified, axisymmetrical heat flow model was assumed, allowing linear conduction calculations.

The mixed-mean gas temperature in the core and reflector exit plenum is 1427°F. Assuming that 8% of the coolant bypasses the fuel element coolant channels through the side reflectors and control rod channels and enters the core exit plenum at 800°F, the average gas temperature leaving the fuel coolant channels becomes 1482°F.

Included in the pressure drop calculations are friction losses in the top and bottom reflector, each 2.25 ft long. At each hexagonal block interface a k-factor of 0.2 was applied to account for possible offset of the coolant holes.

Figure 3.12 shows the radial power variation across the core. For the purpose of obtaining the flow rate distribution among the 37 channels using the COPE code, region average power was used for all regions. As may be seen from Fig. 3.12, these region average values range from a minimum of 0.64 times core average to a maximum of 1.53 times core average. Edge peaking within the maximum power region (edge peaking factors are shown by the bracketed numbers in Fig. 3.12) causes the flow rate in an edge hole to be slightly below the region average. Maximum fuel and gas exit temperatures for the edge hole were computed using the

Fig. 3.12--Radial power distribution across core

COPE results and applying appropriate correction factors determined by hand calculations.

The exponential power profile may be optimized to produce a minimum (i. e., axially uniform) fuel centerline temperature. For the orificed cores the resulting ideal profile is such that the power decreases by a factor of about 3.5 between inlet and exit of the core. This ratio of 3.5 is acceptable and, consequently, the optimized profile was used for all orificed core runs. On the other hand, for the unorificed core the optimum power shape displays a maximum/minimum power ratio in excess of five, considered the limit that may be practically achieved. Here, the power distribution shown in Fig. 12 of the previous quarterly report (GA-6830) was used. As can be seen, the exponential shape is approached by means of five axial zones of decreasing fuel loading. As a result of present limitations on the number of axial points used to describe the power distribution, it was not possible to read the actual five-zone distribution into COPE and the smooth exponential was used. However, use of the five-zone "exponential" would have resulted in unrealistically high fuel temperatures at the location of the axial peaks, because COPE does not account for heat conduction in the axial direction. The hot spots resulting from the axial peaks were determined by the RAT code using an R-Z geometry for analysis of the element. The axial fuel temperature distribution obtained with RAT for the actual five-zone power profile and the smooth exponential, as well as the one obtained with COPE, may be compared in Fig. 3.13. These curves all pertain to the unorificed, 19.5-ft-diameter core, which is designated Case 1 in Table 3.3. It shows that the peak fuel temperature occurs about 9 ft from the entrance of the active core. Similar runs on RAT were not made for the other cases, but COPE temperatures were modified by applying a hot-spot factor determined from the RAT calculation at the location of the power peak.

The cosine power distribution may be obtained by uniform fuel loading and represents a benefit in fuel cycle cost. The profile used in the calculations does not correspond exactly to any real situation, but was used to demonstrate approximately what should be expected in terms of fuel temperatures when a uniform axial loading is utilized.

Crossflow Analyses. Because of orificing, a pressure difference exists between the channels in a hot region and those in an adjacent colder region. Consequently, a flow of gas may occur from a hot channel to a colder one at every block interface within the columns, the amount depending on the width of the interface gap and the local pressure difference between channels. The coolant exit temperature from the starved channel is raised above the mixed-mean value, and the fuel temperatures are also increased proportionately.

Fig. 3.13--Comparison of axial fuel temperature profiles obtained by various means; curves pertain to unorificed, 19.5-ft diameter core (Case 1, Table 3.3)

In order to study how crossflow perturbs the temperature distribution, a simplified two-channel model was used to perform the calculations. The number of interconnects between channels was in accordance with the number of fuel blocks; the crossflow resistance was determined by assuming several gap sizes. A set of simultaneous flow network equations was programmed for computer solution and solved to obtain local values for the flow rate in each channel. By using the new flow rates together with COPE calculations, the coolant gas and fuel temperatures along the length of the peak power region were recalculated taking into account the coolant crossflow. It is emphasized that these crossflow estimates are approximate.

Inspection of the crossflow temperatures for Case 6B (exponential profile) and Case 12B (cosine profile) indicates a greater sensitivity to coolant crossflow from the latter. This is due to the fact that for the cosine distribution a larger portion of power is produced toward the lower end of the core, where the gas flow in a hot channel has been depleted. Crossflow results were obtained for four assumed interface gap sizes; only the largest, 0.02 in., is shown on Table 3.3 as the worst case considered.

System Pressure. The reference reactor design analyzed as Case 1 (see Table 3.3) has a core 19.5-ft in diameter with an active core length of 15.5 ft, is unorificed, and operates at a system pressure of 575 psia. The current helium circulator design is based on a 10.3-psi primary circuit ΔP at reference pressure. With core orificing, the total system ΔP is increased by the amount that the core pressure drop increases, raising it for the reference core to 14.4 psi (Case 3). Since this constitutes a penalty in pumping power and might require a different circulator design, it was decided to determine what system pressure would be required to allow orificing the core without any increase in pumping power. Figures 3.14, 3.15, and 3.16 show the results. In Fig. 3.14 the total core ΔP , which is inversely proportional to P , is plotted versus pressure for several core designs. Added to this is the loss in the remaining part of the system, 6.3 psi at 575 psia system pressure. The circuit ΔP is also inversely proportional to P . The characteristic for the compressor at constant pumping power is a line of constant $\Delta P/P$. The highest system pressure required is 680 psia for the 19.5-ft orificed core with 16.4% coolant channel void fraction.

Thermal Performance. Comparison between equivalent cases for the 19.50- and 21.75-ft diameter cores reveals a slight increase in fuel temperatures for the latter core. This is not unexpected. For constant power density and over-all coolant temperature rise, a reduction in the core length causes an almost equal change in the flow rate per channel,

Fig. 3.14--Total core pressure loss versus system pressure for several core designs

Fig. 3.15--Circulator head rise and orificed core ΔP plus primary circuit ΔP versus system pressure (19.5-ft-diam core)

Fig. 3.16--Circulator head rise and orificed core ΔP plus primary circuit ΔP versus system pressure (21.75-ft-diam core)

hence reducing the film coefficient. The temperature gradient in the solid material remains unchanged. How seriously the fuel temperatures are affected by decreasing the core length depends largely on what fraction of the total ΔT occurs in the film. In the present cases, the film temperature drop is a small portion of the over-all ΔT ; hence, the increase in fuel temperature is small, of the order of 2%.

Voidage. Similarly, the effect of increasing the coolant void fraction is almost negligible as far as fuel temperatures are concerned for small changes in voidage (i. e., between 16% and 18%). Core pressure drops are relatively lower for the 18% voidage cases.

Fuel Cycle Cost. Shortening the core from 15.5 to 12.6 ft causes an increase in neutron leakage, with an estimated fuel cycle cost penalty of 0.02 mill/kw-hr. A benefit of 0.03 mill/kw-hr may be realized by using the cosine power distribution instead of the exponential. This benefit results from reduced neutron leakage from the end of the core with the uniform fuel loading of the cosine distribution.

It may be seen that a potential reduction in the cost of the fuel cycle is available for some orificed core cases by noting that the maximum fuel temperatures are well below the tentatively accepted design limit of 1350°C (2462°F), excluding hot-spot factors such as fuel loading and mechanical tolerances (the reference maximum fuel temperature limit including all factors is 1500°C). Allowing a larger region peaking factor would increase the fuel temperature, but it also reduces the fuel cycle cost (see Fig. 3.17). Region age peaking factors are increased by increasing the C/Th ratio of the core.

Distribution of Orificed Core Coolant Flow. The distribution of coolant flow through the 37 core regions of varying age and location is shown in Fig. 3.18 for an ideally orificed core with uniform exit coolant temperature and no crossflow. Work is currently in progress to determine how the flow redistributes throughout the core when crossflow is considered. However, detailed results will not be available until a computer program is completed to analyze multiple parallel-channel flow with multiple flow interconnects. It is expected that up to six months will be required to complete the development of the code.

Fission-product Transport

The exposure of graphite and coated particle specimens to high-temperature steam is continuing. The purpose of this work is to determine the rates of oxidation of coated particles and fuel element graphite to be expected in case of a reactor steam-inleakage accident. Particles

Fig. 3.17--Effect of lighter loadings on fuel cycle cost

TOTAL FLOW RATE IS
 3.385×10^6 LB/HR OF
 WHICH 3% BYPASSES
 ACTIVE CORE ENTIRELY
 AND 5% (THROUGH CONTROL
 ROD OPENINGS) IS DISTRIBUTED
 EVENLY AMONG THE 37 REGIONS

Fig. 3.18--Mass flow distribution in core orificed for equal gas exit temperature of 1482°F from all regions

are exposed to a steam-helium gas mixture at temperatures in the range 1000° to 1500°C . The effluent gas is first dried to remove the excess moisture and then allowed to pass through a gas chromatograph for analysis of carbon-oxidation products (CO and CO_2) as well as carbide-hydrolysis products (hydrocarbons). The gas also passes through an ionization chamber to detect any gaseous radioactive species, thereby yielding a precise measurement of the time of coating failure.

The present experiments were conducted using either 2.5% or 16.5% steam in helium flowing at $300\text{ cm}^3/\text{min}$. The system pressure was 5 psig. Platinum gauze and a closed TS-688 graphite crucible were used as containers for the particles in separate runs.

Experiments at 1000° , 1100° , 1210° , and 1400°C were accomplished using PyC-coated (Th_3U) C_2 particles ($\text{Th:U} = 3:1$, $100\text{ }\mu$ BISO-2000 coating of density 1.97 g/cm^3). The particles were irradiated in TRIGA only to the extent necessary to produce sufficient radioactive Xe^{133} to detect particle rupture. The test results are given in Table 3.4. The two principal conclusions drawn from the experiments so far are:

1. Fuel particle oxidation occurs by pinhole attack. Particles are often ruptured after less than 1% of the total PyC coating is oxidized.
2. Fuel particles enclosed in a graphite crucible are protected until a major portion of the surrounding graphite is oxidized. It is believed, in a conservative extrapolation to reactor conditions, that very little fuel particle oxidation will occur as long as the fuel element sustains oxidative weight losses of less than 5%.

There are several other interesting developments: (1) When PyC-coated particles are exposed to steam, the quantity of CO and CO_2 in the effluent gas is far greater than one would expect on the basis of volume changes apparent from photomicrographs. This is not understood at present and it is hoped that additional experiments will lead to an elucidation of this anomaly. (2) Carbon monoxide is the predominant oxidation product when ordinary graphite is steam-oxidized at about 1000°C under experimental conditions, whereas carbon dioxide is the predominant product when PyC coated particles are steam-oxidized at about 1000°C . This difference in behavior may be related to differences in the steam content of the effluent gas.

Recent experiments in which particles were exposed to a steam concentration of 16.5% have been performed. Preliminary results show a reaction order of near zero (which means that the oxidation rate is

Table 3.4

RESULTS OF STEAM OXIDATION EXPERIMENTS ON PYC-COATED (Th,U)C₂ PARTICLES^a
(Th:U = 3:1, 100-μ BISO-2000 Coatings)

Temp (°C)	Sample Container	Time to Rupture	Oxidation Rate ^b	Comments and Observations
1210	0.15 in. TS-688 graphite crucible	>21.6 hr	0.05%/min	Graphite can almost completely oxidized, particles still intact
1220	0.15 in. TS-688 graphite crucible	14-17 hr	8.2%/hr	
1400	Pt gauze	12-21 min	6%/min	
1410	0.15 in. TS-688 graphite crucible	330-430 min	12%/hr	No pinhole attack observe Very little pinhole attack observed
1000	Pt gauze	>48 hr	0.4%/hr	
1100	Pt gauze	>16.3 hr	1.5%/hr	

^aExperimental conditions: 2.5% steam, 1 atm He, 300 cm³/min purge rate.

^bPercent weight loss per unit time calculated from CO and CO₂ content of effluent gas.

independent of steam concentration) for both coated particles and TS-688 graphite. This result on first analysis is somewhat anomalous, since previous results indicated a reaction of the order of 0.5 to 1.

MATERIALS

Particle Beds

Packing Studies. Packing studies are being conducted to develop practical methods for preparing and loading uniform fuel blends. Several batches of reference coated fuel particles were combined into a single batch for each of the two reference sizes, and the blended particles were characterized. Some of the data are given in Table 3.5. Bed densities were measured for six blends of the two batches, three density measurements being made for each blend. The measurements were made in a 10 cm³ graduated cylinder, and should therefore be considered preliminary measurements (see Table 3.6). The data show that there is some nesting of particles, which results in an increase in vol-% solids over either reference size by itself (0% and 100% blends).

An air pycnometer device was designed and built to measure void volume in a packed bed. The device should allow accurate measurements of the packing density of 30-in. fuel columns.

Studies were conducted to develop means of adding activated charcoal to beds of coated fuel particles because of its possible use as a sorbent for metallic fission products and as a sacrificial material to protect the coated particles in the event of steam leaks. Attempts were made to maximize the activated charcoal concentration without altering the concentration (packing density) of the coated particles. Evaluation of four processes has been completed, and a fifth is being evaluated:

1. Vibration. Activated charcoal was added to the top of a bed of packed coated particles (reference fissile particles) and the assembly was vibrated at intensities ranging up to 35 G's. In every case, the pores between coated particles became clogged at a depth of 4- to 6-in., which stopped the influx of activated charcoal. If this process were used, fuel holes would have to be filled in steps, 2 or 3 in. at a time.
2. Suspension Filtration. A suspension of fine activated charcoal in water was poured into a bed of reference fissile particles. The water was removed by air-pressure filtration through a filter at the bottom of the container, leaving a charcoal sediment on the filter. The process was repeated until the voids between

Table 3 5
DATA ON BISO-COATED FUEL PARTICLES USED FOR STUDIES ON
LOOSE COATED PARTICLES

Type	Batch	Core		Coating	
		Type	Size Range (μ) ^a	Type	Average Thickness (μ) ^a
Fissile	2260-19-1D	(Th, U)C ₂ ^b	155-255	BISO	110
Fertile	2260-19-2	ThO ₂	310-440	BISO	110

^a - Determined by radiography

^b - Th U - 3 5 1

Table 3 6
PACKING DENSITIES FOR BLENDS OF REFERENCE SIZED COATED
PARTICLES

Fertile Particles in Packed Bed (vol-%) ^a	Packing Density	
	(vol-% Solids)	(g/cm ³)
0	64 3	1 63
0	64 3	1 63
0	64 3	1 63
25 7 ^b	64 9	1 825 ^b
25 7 ^b	64 9	1 825 ^b
25 7 ^b	65 5	1 845 ^b
81	66 5	2 26
81	66 5	2 26
81	66 5	2 26
85 8 ^b	67 7	2 33
85 8 ^b	67 7	2 33
85 6 ^b	67 0	2 30
90 6 ^b	67 8	2 37
90 6 ^b	67 8	2 37
90 6 ^b	67 8	2 37
100	64 3	2 40
100	64 3	2 40
100	64 3	2 40

^a - Remainder of solids consists of fissile particles (See Table 3 5 for parameters)

^b - An uncertainty exists in the last digit

coated particles were filled with charcoal. This process resulted in a heavily loaded (13 wt-% activated charcoal) bed of close-packed particles. However, the process became very slow as the sediment accumulated because the sediment provided a high resistance to water flow.

3. Suspension Centrifugation. This process is similar to item (2), except that the water was driven through the particle bed and the filter by centrifugal force rather than air pressure. This was also a slow process because of the resistance to water flow through the activated charcoal sediment. The resulting loaded fuel column was similar to that obtained by suspension filtration.
4. Dry Blending. Mill blending and V-blending of the coated particles with activated charcoal produced very nonuniform mixtures. A process of shaking a mixture of particles and charcoal (13 wt-% activated charcoal) in a container and pouring the mixture into a fuel hole produced better uniformity, but the coated particle content was reduced to 52.4 vol-%. The coated particles alone nominally packed to 64.3 vol-%.
5. Wet Blending. A mixture of 13 wt-% charcoal-87 wt-% reference fissile particles was prepared, just enough water being added to make the blend similar to a viscous fluid. The mixture was blended with a spatula for 1 hr. A 15-in. fuel hole was filled with the blend by loading about 2 in. at a time and tamping the surface after each loading. The column will be dried and packed by vibration for a determination of the coated particle content (by volume), and for a visual (and possibly radiographic) examination of the blend uniformity.

Thermal Expansion. Thermal expansion measurements were made on loose BISO-coated particles in both the reference fissile (small) and fertile (large) sizes (see Table 3.5). The expansion measurements were made relative to TS-688 and 780-S graphite tubes which contained a 30-in. column of the particles. The particles expanded more than the graphite in all the tests. Occasionally the particles exhibited a hysteresis loss of 1 or 2 mils upon returning to room temperature. The data are shown in Table 3.7 along with data on absolute expansion measurements for TS-688 and 780-S graphite and coated particles.

In making absolute measurements on coated particles, the expansion of a single row of particles was measured in a narrow-bore quartz tube. These measurements were made between room temperature and 1000°C, and relative measurements were made between 900°C and 1500°C. The absolute coefficients were constant above 400°C, and the relative coefficients were constant at all temperatures measured. The data represent these constant values.

Table 3.7

THERMAL EXPANSION DATA ON GRAPHITE AND COATED FUEL PARTICLES

Sample	Absolute Coefficient of Thermal Expansion $\times 10^{-6}$ ($^{\circ}\text{C}^{-1}$)	Relative Coefficient of Thermal Expansion $\times 10^{-6}$ ($^{\circ}\text{C}^{-1}$)	
		Relative to TS-688 Graphite	Relative to 780-S Graphite
Reference fertile size BISO- coated ThO_2 (Lot 2269-2)	3.59 ^a	0.326	0.326
Reference fissile size BISO- coated (Th, U) C_2 (Lot 2260-1D)	(b)	2.65	----
TS-688 graphite ^c	2.26	----	----
780-S graphite ^c	2.7	----	----

^aPreliminary data.^bMeasurement in progress.^cParallel to direction of extrusion.

The expansion of the large coated particles was zero up to a temperature between 750° and 850°C. The particles then expanded at a constant rate that was independent of graphite type. The coefficient relative to TS-688 for the large particles was considerably lower than for the small particles. This result could be due to a wide variety of reasons, such as particle bridging in the tube or a real, pronounced difference in the expansion coefficients of the two particle types. If bridging is occurring in the bed of large particles, the observed expansion is probably due only to a shallow layer at the top of the particle column, and additional expansion could occur in a vibration environment. No broken particles were observed.

Thermal Conductivity. An experiment was conducted to determine the effect of charcoal additives on the thermal conductivity of beds of loose coated particles. In an attempt to maintain packing density for the coated particles, the charcoal was sieved to three size ranges: 45 to 75 μ , 75 to 105 μ , and 105 to 150 μ . The mixing technique consisted of filtering the charcoal through the pores between coated particles by vibration. The medium size range allowed the highest loading of charcoal (11.3 wt-% of the blend, or about 4 lb per fuel block), however, the coated particle content by volume was less than that for the coated particles alone. Additional packing studies will be required to determine the maximum charcoal loading attainable without reducing the coated particle content. These studies will require the addition of small (fissile) reference particles. Only large (fertile) reference particles were used in this experiment.

After being loaded in the thermal conductivity apparatus, the blend was outgassed at 1500°C to $>10 \mu$ Hg (the outgassing would be expected to reduce the bed conductivity). The standard apparatus and techniques were used for the comparative thermal conductivity measurements. The results of the measurements indicate that, in this test, the effect of adding charcoal was negligible.

Bonded Particle Systems

Various methods of bonding coated particles into the fuel holes of reference elements are being investigated as a means of preventing fuel spillage in the event of fuel element cracking. The addition of activated charcoal to the bonded particle system is also desirable, both as a sorbent for metallic fission products and as a sacrificial material to protect the coated particles in case of a steam leak. In initial experiments, a mixture of coated particles and charcoal was impregnated with furfuryl alcohol monomer containing 10% ethylcellulose. Fair bonding was obtained, but the particle density appeared to be somewhat below the 65 vol-% density desired for the fuel element.

Graphite Development

The objective of the graphite development program is to evaluate conventional needle-coke graphites for use in fuel elements. The graphites are being evaluated with respect to strength, purity, thermal properties and cost.

Graphite Evaluation. Conventional needle-coke graphites are currently being evaluated for use in fuel elements and as reflector graphites. The graphites of current interest are described in Table 3.8.

Tensile strength testing is under way to determine fracture stress, fracture strain, modulus of elasticity, and Poisson's ratio. Fracture stress data are given in Table 3.9. The tensile test technique, which was developed by BNWL,* involves gluing a 0.5-in.-diameter specimen to metal end grips with epoxy resin. This technique has yielded higher strength values than conventional techniques obtained with "dogbone" samples. This technique was evaluated using electrode-grade graphite (B-30). The resulting strength values from the cylindrical specimens were higher and had less scatter than those from dog-bone specimens (see Table 3.9).

A cooperative tensile testing program is being carried out among General Atomic, BNWL, and UCC using a portion of a TS-688 graphite log supplied by ORNL (see Fig. 3.19). A total of 12 parallel and 12 perpendicular samples of TS-688 were stressed to fracture with 0.5-in.-diameter cylindrical samples as General Atomic's part of the program.

Additional tensile tests were performed on ten parallel samples of TS-688--five from the log selected for the round robin program and five from another log. A comparison of the results shows that the data from log 2369-79 are higher than those from log 2369080. A further comparison of these data with data from ORNL[†] and with the results obtained from 0.5-in.-diameter cylindrical samples from log 2369-80 indicates that at least some of the discrepancy in strength can be traced to the log, or more specifically, to the section of a particular log, from which the samples are selected.

Further clarification of the effect of sample selection on the strength of TS-688 will be evaluated when the results of the round-robin testing program are available. Several additional conclusions that may be drawn from the data of Table 3.9 are:

* BN-SA-55, May 24, 1965.

[†] B. L. Greenstreet, et al., Room-temperature Mechanical Properties of AGOT Graphite, ORNL-3728, January 1965.

Table 3.8

DESCRIPTION OF GRAPHITES BEING EVALUATED

Grade No.	Log No.	Size of Parent Log
TS-688	2369-79 ^a	Probably 18 in. x 18 in. x 20 ft
TS-688	2369-80 ^b	18 in. x 18 in. x 24 in. (20-ft)
H-327	2369-84	12.75 in. Diameter x 24 in.
H-328	2369-83 ^c	17 in. Diameter x 20 in.
9567	2369-82	4.75 in. x 12.4375 in. x 20 in.

^a—Obtained from ORNL as 4.5-in.-diameter, 12-in.-long log; identified as ORNL file No. 266. This piece was cut from the same parent log used for studies reported in ORNL-3728.

^b—Obtained from ORNL; identified as ORNL file No. 325. This portion cut from a 20-ft-long log.

^c—Isotropic graphite, probably not needle-coke.

Table 3 9
TENSILE STRENGTH OF GRAPHITES

Material	Type of Sample	Position in Log	Number of Samples		Fracture Stress (psi)							
					Parallel				Perpendicular			
				⊥	Avg	Min	Max	σ	Avg	Min	Max	σ
B-30	Cylindrical (1/2 in. dia)	Random	6	-	1121	969	1296	136	---	---	---	---
B-30	Dog-bone (1/4 in. x 1/4 in. cross section)	Random	9	-	1059	781	1282	171	---	---	---	---
H-327	Cylindrical (1/2 in. dia)	Random	6	6	>2559	>2227	2949	~251	1085	904	1399	179
H-328	Cylindrical (1/2 in. dia)	Random	6	6	2555	1677	3122	529	>2782	>2379	2929	255
9567	Cylindrical (1/2 in. dia)	Random	6	6	>2185	>1884	2298	~146	1591	1515	1702	64
TS-688 (Log 2369-79)	Dog-bone (1/4 in. dia)	Random	5	-	1509	1230	1660	154	---	---	---	---
TS-688 (Log 2369-80)	Dog-bone (1/4 in. dia)	Random	5	-	1281	880	1650	311	---	---	---	---
TS-688 (Log 2369-80)	Cylindrical (1/2 in. dia)	Row A ^a	5	-	1010	660	1364	254	---	---	---	---
		Row B	-	3	---	---	---	---	726	596	864	110
		Row E	4	-	1076	701	1244	160	---	---	---	---
		Row F	-	3	---	---	---	---	319	263	381	46
		Row H	3	-	973	701	1222	213	---	---	---	---
		Row I	-	3	---	---	---	---	259	242	343	53
		Row N	-	3	---	---	---	---	436	404	160	24
TS-688 (Log 236-80)	Cylindrical (1/2 in. dia)	All Samples	12	12	1023	660	1364	242	443	242	864	184

^a Refers to horizontal rows in the 18 in. x 18 in. cross section of the parent log (see Fig 3.19).

Fig. 3.19--Sampling procedure used for TS-688 graphite (General Atomic log 2369-80)

1. The fracture stresses of Grades 9567 and H-327 were significantly higher than those of TS-688. Grade 9567 showed good strength and low scatter in the perpendicular direction.
2. The fracture stress of Grade H-328 (isotropic) was relatively high in both directions.

Experiments will be continued to evaluate the strength and elastic properties of conventional needle-coke graphites. These evaluations will include CTE, density, and thermal-conductivity measurements.

Surface Area Studies of TS-688 Graphite. Surface area studies of TS-688 graphite have been performed (1) to investigate the uniformity of the large graphite logs, 2369-74A and B, used as a supply of this graphite for research and development work and (2) to determine the effect of particle size on the BET (nitrogen adsorption) surface area of TS-688 graphite. Information obtained on the effect of particle size on surface area will be particularly useful in correlating sorption data measured using graphite samples of different particle size and in applying laboratory sorption data to predicting the sorption of metallic fission products on dust in the PSC reactor.

The studies to investigate the uniformity of the TS-688 graphite logs were prompted by anomalous results (unusually high vapor pressures) obtained in measurements of the sorption of cesium on a sample of TS-688 graphite.

Surface areas were measured using a gas-chromatographic nitrogen adsorption technique. The particle size fractions were obtained by sieving cuttings obtained by drilling into portions of the graphite logs. Surface area measurements were made on nine samples taken from various locations on the graphite logs. Figure 3.20 shows the surface area values obtained and the sources of the samples. Except for the value $8.8 \text{ m}^2/\text{g}$, which may have resulted from contamination in the particular sample, the surface area values are close together, and this suggests that the two graphite logs are fairly uniform. The sample that gave the high surface area value ($8.8 \text{ m}^2/\text{g}$) came from the same cuttings as a sample that gave anomalous results in cesium-sorption measurements.

Figure 3.21 shows the results of surface area measurements performed on fractions having a range of particle size.

Control Materials

Control materials studies are being conducted to determine the availability of materials that can meet the requirements of the PyC reactor. Liaison will be maintained with vendors, and physical and mechanical property evaluations will be conducted on suitable samples procured from vendors.

Fig. 3.20--Surface areas ($\text{m}^2/\text{g C}$) of 150- to 125- μ particles of TS-688 graphite sampled from various locations on logs 2369-74A and 2369-74B

Fig. 3.21--Surface area of TS-688 graphite from log 2369-74B as a function of particle size

Strength measurements are being made on irradiated B_4C -graphite compacts that had been previously irradiated in Capsules 1A1 and 1A2 under the Peach Bottom HTGR program. Because of the small size and hardness of the irradiated samples, some difficulty was encountered in machining modulus of rupture specimens. Efforts are continuing to machine suitable strength specimens from the remaining samples.

Irradiation Testing

A series of irradiation experiments is being conducted to test reference BISO coated particles under PSC design irradiation conditions. These experiments are intended to guide the selection of fuel materials which will meet the performance requirements of the PyC reactor. BISO-coated carbide fuel particles have been selected as the reference design based on their good stability under high-burnup and high-temperature irradiation environments. Current tests are directed toward evaluating the optimum materials parameters for each layer of the BISO coating, with emphasis on evaluating the stability of these particles under high-fast-flux exposures.

Graphite irradiations on the PSC program are directed toward obtaining more extensive data on conventional needle-coke graphites. The objectives of this study are to determine the effect of irradiation temperature on dimensional and property changes in the range 600° to 1200°C , and to relate these changes to current damage theories.

Capsule P14. The objectives of the P14 experiment are to evaluate the effects of density and preferred orientation of the isotropic PyC layer of BISO coatings on radiation stability. The density range covered for the isotropic coatings will be 2.00 to 2.15 g/cm^3 , and the range of anisotropy factors will be 1.0 to 1.8 . (The buffer layer of the coating, and the T/D ratio,* will be kept constant on all particles in these series.) The effect of T/D ratio on the stability of fertile particles with high-density (2.10 g/cm^3) BISO coatings will also be determined. One sample will be tested both loose and bonded, with two different types of binders, to evaluate the radiation stability of bonded beds, and activated charcoal will be included in the binder. A description of the coated particles tentatively selected for irradiation in Capsule P14 is given in Table 3.10.

PyC strips of each of the coating types tested in P14 will also be included in the capsule. These will be irradiated at estimated temperatures of 1070° and 670°C .

* Ratio of coating thickness to particle diameter.

Table 3.10
PRELIMINARY DESCRIPTION OF COATED PARTICLES TO BE TESTED IN CAPSULE P14

Type	Coating			Particle			T/D Ratio	Design Irradiation Conditions ^a		Test Type	Remarks
	Density (g/cm ³)	Anisotropy Factor	Thickness (μ)	Type	Size (μ)	Th:U Ratio		Temperature (°C)	Burnup ^b (% FIMA)		
BISO(2100)	2.00	1.0	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	BISO density series
BISO(2200)	2.05	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.15	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	BISO preferred orientation series
BISO(2200)	2.10	1.4	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.10	1.6	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.10	1.8	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200) ^c	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	BISO crystallite size series
BISO(2000) ^c	2.10	1.0	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	Loose versus bonded particles
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Bonded-Pitch	
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Bonded-Furfuryl	
BISO(2200)	2.05	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1000	20	Loose	Lower temperature BISO density and preferred orientation series
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	150-250	3:1	0.50	1000	20	Loose	
BISO(2200)	2.10	1.4	100	(Th,U)C ₂	150-250	3:1	0.50	1000	20	Loose	
BISO(2200)	2.10	1.2	100	(Th,U)C ₂	300-420	12:1	0.29	1400	6	Loose	BISO fertile T/D series
BISO(2200)	2.10	1.2	130	(Th,U)C ₂	300-420	12:1	0.37	1400	6	Loose	

^a Design fast-flux exposure is 4×10^{21} nvt for four cycles in ETR.

^b During four cycles in ETR.

^c These two samples differ in crystallite size.

Samples of activated charcoal will be tested separately from the particle beds by incorporating a number of small sample holders in the various cells. Some will contain bonded charcoal (pitch and resin binders), and the remainder, loose charcoals. The effects of irradiation on the structure and adsorbent properties of the charcoal will be determined by postirradiation studies.

The capsule is being designed for a minimum of three irradiation cycles in the ETR. If it is performing well at the end of the third cycle, it will be left in up to six cycles in order to get a higher fast-flux exposure. Fast-flux exposure at the end of three cycles is expected to be about 3×10^{21} nvt.

Design burnups are 20% (FIMA) for the fissile particles and 6% for the fertile particles over a four-cycle irradiation. If the capsule is terminated after three cycles, the burnups will be slightly below design, whereas if it should go a full six cycles, burnups will be somewhat higher than design. Th:U ratios of 3:1 and 12:1, respectively, will be used to achieve these burnups.

Design fuel temperatures are 1400° and 1000° C. All samples of coated particles will be tested at 1400° C, and three selected samples will also be tested at 1000° C. The lower temperature is being used because the results from a previous irradiation indicate that maximum dimensional effects may occur in the PyC at about 1000° C.

The design of this capsule incorporates five individual crucible cells, each cell containing three separate particle thimbles. Each particle thimble has its own high-temperature thermocouple. This thermocouple will monitor the highest temperature region of the thimble. End losses will cause an axial gradient of approximately 125° C in each thimble. Thermal calculations indicate that the dual gas control system designed into the capsule will allow the temperature of any individual cell to be varied up to about 1200° C. This control capability will eliminate most of the problems involved in accommodating the $\pm 30\%$ power swing of the ETR, as well as maintaining design temperatures for long periods of time in spite of power reductions resulting from burnup.

Capsule design and construction are proceeding on schedule. Irradiation is scheduled to begin in January, 1966, in ETR cycle 78, in the 113NE position.

Capsule P15. The objective of the P15 experiment is to evaluate the structure of the buffer layer of reference BISO coatings. Planning and engineering for this capsule are in progress, and it is scheduled to begin irradiation in May, 1966.

HTGR Reference Fuel Tests. HTGR reference coated particles (BISO-coated fissile and fertile particles) in a packed bed are being tested in ORNL Loop 1-15 at a maximum design temperature of 1400°C . A low-temperature (600°C) appendage capsule contains a small quantity of both loose- and pitch-bonded coated particles. The irradiation has been extended from four to five ORR cycles so that the fuel will now achieve a burnup of approximately 16% FIMA. The experiment is scheduled to end in March, 1966.

The maximum (center) thermocouple temperature has been fluctuating between 1330° and 1350°C during the current ORR cycle. Because of axial temperature gradients detected on the surface of the capsule, the corresponding range in estimated maximum fuel temperature is 1370° to 1400°C .

The fuel burnup has now reached approximately 4.8% FIMA in the upper capsule and 3.6% in the lower, appendage capsule. Fission-gas release rates (calculated from equilibrium fission gas levels in the loop coolant) are in the 10^{-6} to 10^{-7} range, and have remained low throughout the experiment. All impurities are now being held below 10 ppm; at one point, however, the CO_2 impurity level had reached a maximum of 45 ppm.

Splitting the capsule in half for postirradiation analysis has been shown to be feasible. This operation will be done as a means of checking the uniformity of the blend of fissile and fertile particles.

Capsule G11. A series of capsule experiments has been planned to irradiate conventional needle coke graphites in the temperature range 600° to 1200°C to a maximum exposure of 9×10^{21} nvt ($E > 0.18$ Mev). The first of these capsule experiments is G11.

Details of materials selection were reported in the previous quarterly report. Capsule fabrication was completed; the capsule has been shipped to the ETR; and irradiation is scheduled to begin in January 1966. The experiment is designed to remain in the reactor until an exposure of about 3×10^{21} nvt is achieved (~ 3 ETR cycles).

The G11 capsule was designed using the General Atomic-derived gamma-heating curve. This curve is higher by a factor of about two than the curve derived by ETR (Phillips) for this hole. Measurements made recently by Phillips indicate that the General Atomic curve is more nearly correct than the other. Thus, the decision to use the General Atomic derived curve for the design of the capsule appears to have been a correct one, and narrows the degree of uncertainty as to whether design temperatures will be achieved.

With this capsule the type of gas in the primary specimen container can be changed as well as the gas mixture in the secondary temperature control annulus. This allows somewhat greater temperature control than would be afforded by a single control system, but not quite so much as would be available were the gas in the primary container to be a flowing mixture instead of a closed-off single type. The reason a flowing mixture is not used in the primary containers is that it reduces the possible effects of gas impurities on the graphite specimens (in a flowing mixture the impurity level would be constantly renewed).

CHEMISTRY

Fission-product Release

Work has continued on the release of metallic fission-product nuclides of cesium, strontium, cerium, and barium from coated fuel particles. The purpose of this work is to obtain information for use in evaluating the retention properties of coatings and calculating the release of metallic fission products from PSC fuel. The work involves post-irradiation experiments in which release values are determined for particles irradiated to high burnup. The methods employed in post-irradiation examination of highly irradiated PyC-coated particles are (1) annealing experiments, (2) direct gamma counting, and (3) radiochemical analyses.

In the annealing method, the irradiated particles are placed in a graphite container and annealed using a King (carbon-resistance) furnace for up to 300 hr in a helium stream at temperatures in the range 1100° to 1800° C. During the anneal, the released cesium is collected on a cold finger, and the other metallic nuclides released are collected in the graphite container. Quantities collected are determined using gamma spectrometry and radiochemical analyses. The annealing method is used to determine the fraction of metallic fission products retained by the coated particles during irradiation test and to determine release constants.

In the direct gamma-counting method, quantities of Zr^{95} , Cs^{137} , Ce^{144} , and Ba^{140} (La^{140}) in particles or collector samples are determined using a germanium-crystal detector. This method is particularly useful in the direct determination of the quantities of metallic fission products contained in irradiated particles.

Radiochemical analyses are used primarily for the determination of the strontium nuclides Sr^{89} and Sr^{90} that emit no gammas.

The experimental procedures are described in the previous quarterly report (GA-6830).

Postirradiation measurements have been performed on particles irradiated to about 4% FIMA burnup. Two types of (Th, U) C_2 particles--one with 98- μ BISO(2200) coatings with a density of 2.12 g/cm³ and the other with 105- μ BISO(2100) coatings with a density of 1.99 g/cm³--have been subjected to (1) direct gamma-counting analyses to determine the fraction of metallic fission products retained during the irradiation test and (2) annealing experiments in the range 1300° to 1800°C to determine release constants, as well as fractional retention values. The release data are being calculated.

The data calculated to date indicate that about 90% of the Cs^{137} , 60% of the Ba^{140} , 60% of the Ce^{144} , and 50% of the Sr^{89} were retained by the particles during the irradiation.

Sorption and Diffusion of Fission Products

Vaporization of Metallic Fission Products from Graphite. Measurements are continuing on the vapor pressures of metallic fission products sorbed on TS-688 graphite. The purpose of this work is to obtain basic data for use in calculations to predict the release of metal nuclides from PSC fuel blocks.

At present, studies of the strontium-TS-688 graphite system are being carried out using the mass spectrometric Knudsen-cell method essentially as described in the previous quarterly report. Studies are also being carried out on the strontium-activated carbon system, but it is too early to report the results.

Sorption of Cesium on Graphite. Measurements using an isopiestic method are continuing on the sorption of cesium on TS-688 graphite. The purpose of this work is to obtain data for use in calculations to predict the release of fission-product cesium from PSC fuel.

A sample of graphite having a known BET surface area is placed in the horizontal leg of an L-shaped tube constructed of Pyrex glass and stainless steel. The system is evacuated and cesium tagged with Cs^{137} to a known specific activity is distilled from an auxiliary glass tube into

the Pyrex leg of the tube. The entire system is thoroughly outgassed and sealed off at a pressure of $\sim 10^{-5}$ mm Hg.

During a run, the cesium pressure is fixed by maintaining the cesium metal in the Pyrex leg at a constant temperature. The amount of cesium on the graphite specimen is monitored with a calibrated gamma scintillation counter until an equilibrium condition has been established at the desired specimen temperature and cesium pressure.

Measurements to obtain sorption isotherms for cesium on TS-688 graphite at 800° , 900° , and 1000° C in the cesium pressure range 10^{-4} to 10^{-8} atm have been performed. The samples of graphite (6 to 8 mesh granules) used in the measurements were taken from a sample made up of cuttings taken from various locations on a block (about 18 in. on each side) cut from one end of log 2369-74A, one of two large (20-ft-long) graphite logs being used as a source of this material for research and development work.

Results show that for a particular cesium pressure, cesium loadings obtained using the TS-688 graphite sample are very similar to loadings obtained using HLM-85 graphite when the loadings are expressed on a weight basis (mg Cs/g C); however, when the concentrations are expressed on a surface area basis, an appreciable difference is noted. This difference is due to the spread in BET surface areas ($2.5 \text{ m}^2/\text{g}$ for the TS-688 graphite sample and $0.7 \text{ m}^2/\text{g}$ for HLM-85 graphite samples, using 6 to 8 mesh granules).

Figure 3.22 compares sorption isotherms at 900° C for various types of graphite (TS-688, HLM-85, AGOT, 780-S, S12, CHN, and H315A) and charcoal (activated carbon). Two graphites, CHN and H315A, are needle-coke type, as is TS-688. It is noted that the sorption isotherms fall in three distinct groups that are related to methods of manufacture: (1) nongraphitic charcoal, (2) single impregnation and graphitization (HLM-85, AGOT, CHN), and (3) double impregnation and graphitization cycles (780-S, S12, and H315A). The TS-688 isotherm falls between the charcoal and graphite isotherms and thus shows anomalous behavior.

The anomalous behavior of the TS-688 graphite sample suggests that the particular sample was not representative of the bulk graphite on hand and may have been contaminated with poorly graphitized, high-surface-area carbon. Surface area measurements appear to support this suggestion.

Three sorption runs are in progress using a new sample of graphite

Fig. 3. 22--Sorption of cesium on charcoal and various types of graphite at 900°C

that, on the basis of surface area measurements, is more representative of the bulk TS-688 graphite.

Diffusion of Metallic Fission Products in TS-688 Graphite. Three separate experiments for measuring diffusion coefficients for barium in TS-688 graphite are under way. The three experiments are being conducted in separate furnaces under identical conditions except for operating temperatures, which are 900° , 1000° , and 1200°C .

The diffusion of barium in TS-688 graphite is being studied to obtain data for use in calculations to predict the release of Ba^{140} from PSC fuel blocks. Information on the rate of diffusion of Ba^{140} is particularly important, since this nuclide will decay appreciably (because of its relatively short half-life of 12.8 days) during passage through the graphite of the fuel elements.

The apparatus used in measuring diffusion coefficients consists of a graphite compact held inside a graphite sleeve. The sleeve is mounted in a King furnace and is resistance-heated. The compact is first loaded with a mixture of barium oxide (tagged with 7.5-y Ba^{133}) and graphite powder, and then preconditioned by heating it in a separate graphite tube at 1200°C for several hours. The purpose of this is to react BaO with the graphite powder to form elemental barium uniformly distributed in the graphite compact.

The compact and sleeve are brought to a desired temperature and held for sufficient time to allow the tagged metal to be converted to the elemental form and to diffuse out of the compact into the graphite. Periodically, the experiment is interrupted; the sleeve is removed from the King furnace and cored to obtain a profile of the concentration of metal in the graphite. In coring, a hole is drilled into the sleeve using a special drill, and the activity of the cuttings is determined as a function of depth. Diffusion coefficients are determined from the concentration profiles.

The experiment at 900°C has been run a total of 982 hr. The latest coring of the sleeve indicates a value of about $1.7 \times 10^{-10} \text{ cm}^2/\text{sec}$ for the diffusion coefficient at 900°C . The value has been increasing slightly with each coring. However, the concentration profile is still somewhat indistinct, and further annealing time is required to obtain a reasonable profile.

A total of 746 hr has been logged for the experiment at 1000°C . The latest coring indicates the diffusion coefficient to be $3.4 \times 10^{-9} \text{ cm}^2/\text{sec}$. This value is slightly higher than the value ($2 \times 10^{-9} \text{ cm}^2/\text{sec}$)

found previously for HLM-85 graphite at 1000°C. The concentration of barium on the inner boundary of the test specimen (0.2 mg Ba/g C) seems to be lower than the value (0.8 mg Ba/g C) found previously for HLM-85 graphite, indicating that a fewer number of sorption sites are available on the TS-688 graphite at 1000°C. The graphite compact used in the present experiment has been cored and shows a flat concentration profile at about 0.8 mg Ba/g C.

The experiment at 1200°C has accumulated a total of 32.6 hr. The diffusion coefficient is indicated to be $1.0 \times 10^{-7} \text{ cm}^2/\text{sec}$ at this temperature.

Cesium Plateout on Metal Surfaces. Studies of the plateout of cesium on stainless steel and copper surfaces are continuing. The purpose of this work is to gain information for use in predicting the degree of cesium plateout on the surfaces of reactor components. An isopiestic method is used. The apparatus and experimental procedure are the same as those used in studies of the sorption of cesium in graphite, except that a metal specimen is substituted for the graphite. The specimen is a strip of 0.002-in. stainless steel foil (3 cm by 20 cm) loosely rolled up so that the geometric surface is considered to be 120 cm^2 . Prior to the start of a run the specimen is outgassed at the desired temperature before sealing off the system at a pressure of about 10^{-5} mm Hg .

In an experiment to measure the plateout of cesium on as-received Type 304 stainless steel at 650°F, the equilibrium loading at a cesium pressure of $1.4 \times 10^{-6} \text{ atm}$ was found to be equivalent to 64 monolayers of cesium ($2.56 \times 10^{16} \text{ atoms/cm}^2$). * This run was a duplication of an experiment in which the equivalent of 27 monolayers of cesium was found to be in equilibrium with $1 \times 10^{-6} \text{ atm}$ cesium pressure. The large variation in loading is inexplicable, since both runs are considered reliable. This behavior and similar behavior found in previous studies illustrates difficulty in obtaining "duplicate" samples of steel.

Sorption of Strontium on Steel and Dust Particles. Measurements of the relative sorption of strontium on graphite dust particles, prepared from TS-688 graphite, and steel surfaces are under way. The purpose of this work is to obtain information for use in predicting how the Sr⁸⁹

* One monolayer Cs = $4 \times 10^{14} \text{ atoms/cm}^2$.

and Sr^{90} that escape into the coolant distribute themselves in the PSC reactor.

The loop-type experiment described in the previous quarterly report is continuing. In this experiment, samples of Type-304 stainless steel and TS-688 graphite are held at 800°C and the strontium source is held at 1050°C . (The strontium source consists of a mixture of Sr^{85} -tagged strontium and graphite powder contained in a closed graphite crucible.) A stainless-steel containment tube is used, and the dust particles are located about 15-in. downstream from the strontium source. The strontium vapor pressure at the temperature and strontium loading used was estimated to be 10^{-6} atm.

Helium was recirculated at $200\text{ cm}^3/\text{min}$ for 10 days and no movement of strontium was detected. It was felt that buildup of CO in the system could be suppressing the strontium pressure owing to the reaction $\text{Sr} + \text{CO} = \text{SrO} + \text{C}$. The system was therefore switched to a single pass of helium, flowing at $200\text{ cm}^3/\text{min}$, to ensure a low CO pressure.

The apparatus was disassembled after 33 days, during which no evidence of strontium transport to the dust sample was detected. The strontium carried from the source had plated out in the containment tube a few inches downstream of the source location. The average strontium vapor pressure (calculated from depletion of the source) was determined to be $\sim 10^{-7}$ atm. The system was reassembled and the run continued. After an additional 14 days, the strontium had not yet reached the dust particles.

A new method is being investigated for use in measuring the relative sorption of strontium on dust and stainless steel. In this method a sample of stainless steel foil and a sample of graphite powder (dust) loaded with Sr^{85} -tagged strontium are allowed to equilibrate under vacuum conditions at around 1000°C . The steel and dust samples are contained in a tantalum Knudsen cell and are arranged so that they are not in contact. With the system at equilibrium, vapor pressure measurements are performed and equilibrium loadings of strontium on the steel and graphite samples are obtained by gamma-counting the two samples. This method may prove to be a relatively low-cost and rapid method for obtaining the required sorption information.

Another method, in which graphite particles impregnated with Sr^{85} -tagged strontium are annealed in sealed, steel tubes, was described in the previous quarterly report. However, this method has not been

utilized, because the Knudsen cell method just described appears more attractive.

Mathematical Treatment of Metallic Fission-products Release from HTGR Fuel. A modified FREVAP code, FREVAP-2, was developed to permits the consideration of two absorber regions (e.g., graphite and activated carbon) operating at different temperatures in an HTGR fuel block. This was done so that the effectiveness of adding finely divided charcoal (activated carbon) to fuel particle beds for improving the retention of metallic fission products might be estimated.

The possibility of activated carbon being a good adsorber for metallic fission products at high temperature is based on results of a number of studies at General Atomic. Studies of the sorption isotherms of the cesium-carbon system at 800° to 1100°C showed that activated carbon on a weight basis has several hundred times the capacity of nuclear grade graphite for sorbing cesium. On the basis of BET area (accessible internal surface area as determined by nitrogen adsorption at low temperature), sorption characteristics of graphites and carbons are found to be more nearly comparable. For this reason BET area has been used as a basis for expressing concentrations of cesium and other metallic fission products in graphite in FREVAP calculations.

The physical and logical basis for the code is essentially the same as that presented in a discussion of the FREVAP code (FREVAP-1) given in the previous quarterly report. FREVAP-2 is for the simplified case of a fuel element (or reactor core) at a uniform temperature at a given time and as a first approximation of only very long-lived and stable isotopes, i.e., Cs¹³³, Cs¹³⁵ - Cs¹³⁷, and Sr⁸⁸ - Sr⁹⁰. Total quantity (moles) of a given fission-product element in activated carbon and graphite of the fuel element, M, is

$$M = \frac{KA(C)}{10^6} C(C) + \frac{A(G)}{10^6} C(G) . \quad (1)$$

$C(C) = \sum_i C_i(C)$ = concentration of fission-product element in activated carbon ($\mu\text{moles}/\text{m}^2$ BET area) = sum of long-lived and stable isotopes.

$C(G) = \sum_i C_i(G)$ = concentration of fission-product element in the fuel element graphite ($\mu\text{moles}/\text{m}^2$ BET area) = sum of long-lived and stable isotopes.

It is assumed that the metallic fission-product element is at approximately the same vapor pressure in the activated charcoal and the graphite, though these materials are at different temperatures. The fission-product partial pressure (pp) is then expressed by

$$(pp) = e^{v_j} \{C(C)\}^{u_{1j}} = e^{v_{2j}} \{C(G)\}^{u_{2j}} (\text{atm}) , \quad (2)$$

since both activated carbon and graphite are found to sorb fission-product elements such as Cs and Sr in accordance with the Freundlich isotherm.

The v 's and u 's are given by

$$v_{1j} = a_1 + b_1 \left(\frac{10^3}{T_{Cj}} \right) , \quad (3a)$$

$$u_{1j} = c_1 + d_1 \left(\frac{10^3}{T_{Cj}} \right) , \quad (3b)$$

$$v_{2j} = a_2 + b_2 \left(\frac{10^3}{T_{Gj}} \right) , \quad (3c)$$

$$u_{2j} = c_2 + d_2 \left(\frac{10^3}{T_{Gj}} \right) , \quad (3d)$$

where $a_1, b_1, c_1, d_1, a_2, b_2, c_2,$ and d_2 are empirical constants for the activated carbon and the graphite, respectively (chosen to give partial pressure in atmospheres), and where T_{Cj} is the temperature of activated carbon ($^{\circ}\text{K}$), and T_{Gj} is the temperature of graphite ($^{\circ}\text{K}$). The index j indicates a specific time interval.

The equation for the buildup of the total quantity of a given fission-product element is given by

$$\frac{dM}{dt} = \sum_i S_{ij} - \zeta \left(\frac{\dot{n}_{\text{He}}}{P_{\text{He}}} \right) (\text{pp}) , \quad (4)$$

where t = time (days),

M = total moles of given fission-product element, see Eq. (1),

S_{ij} = release rate of isotope i from fuel particles to graphite in moles/day (j indicates the time interval to which S_{ij} applies),

ζ = the ratio of the partial pressure of the given fission-product element in the coolant stream (in the turbulent region beyond the boundary layer) to the partial pressure at the graphite-coolant interface,

$\dot{n}_{\text{He}}$ = flow rate of He coolant (moles/day),

P_{He} = pressure of He coolant (atm), and

(pp) = partial pressure of the fission-product element as given in Eq. (2).

The source term

$$S_{ij} - R_j N_i \quad (5)$$

is obtained in the same manner as indicated in the previous quarterly report. Thus,

$$\frac{dN_i}{dt} = B_{ij} - (\lambda_i + R_j) N_i , \quad (6)$$

where N_i = the quantity of isotope i in the fuel particles at time t (moles),

B_{ij} = production rate of isotope i in time interval j (moles/day), and

R_j = the release constant of the fuel particles in the time interval j (day^{-1}),

with

$$\ln R_j = \ln R_o - \frac{E_a}{1.987} \left(\frac{10^3}{T_{Fj}} \right) , \quad (7)$$

where R_o , E_a , and T_{Fj} are defined as before.

It is to be noted that, in a more rigorous treatment, Eq. (4) would be replaced by a set of equations.

$$\frac{dM_i}{dt} + \lambda_i M_i = S_{ij} - \zeta \left(\frac{\dot{n}_{\text{He}}}{p_{\text{He}}} \right) (pp_i), \quad (4a)$$

where M_i = total moles of isotope i ,

λ_i = decay constant of isotope i ,

pp_i = partial pressure of the isotope i ,

and

$$\sum_i M_i = M.$$

In FREVAP-2, only the very long-lived and stable isotopes are considered (these greatly outnumber the short-lived isotopes when t is many months to years). In this case λ_i is either nearly zero or zero, and the set of equations represented by Eq. (4a), when summed over the longer-lived and stable isotopes, i , gives Eq. (4) to a good approximation.

Combining Eqs. (1), (2), and (4) yields a differential equation in terms of $C(G)$, the concentration of fission-product element in the graphite of the fuel element,

$$\frac{dC(G)}{dt} = \frac{\sum_i S_{ij} - \zeta \frac{\dot{n}_{\text{He}}}{p_{\text{He}}} \exp v_2 \{C(G)\}^{u_2}}{\left[\frac{KA(C)}{10^6} \exp(v_2 - v_1)/u_1 \left(\frac{u_2}{u_1} \right) \{C(G)\}^{(u_2/u_1 - 1)} + \frac{A(G)}{10^6} \right]}. \quad (8)$$

This nonlinear differential equation is solved using a subroutine, and the integrated fractional release (Int. $F(t)$) is calculated using the equation

$$\text{Int. } F(t) = \frac{\zeta \left(\frac{\dot{n}_{\text{He}}}{p_{\text{He}}} \right) \exp v_2 \{C(G)\}^{u_2}}{\sum_i p_i}, \quad (9)$$

where p_i is the total quantity of a given fission-product isotope, i , at time t (moles), and p_i is calculated using the equation

$$\frac{dp_i}{dt} = B_{ij} - \lambda_i p_i.$$

Application of the FREVAP-2 code was made on the basis that 4 lb of activated carbon (~3.7 wt-% of the graphite) might be added to fuel particles in a PSC fuel block with graphite weighing 107 lb. Calculations were made for long-lived cesium (Cs^{137}) and strontium (Sr^{90}) release for one case of no activated carbon and three cases with activated carbon added, where the carbon was at (1) fuel temperature, (2) graphite temperature, and (3) an intermediate temperature corresponding to the graphite temperature plus 35°C. The assumed temperatures and production rates, B_{ij} , (in moles/day, for the entire PSC core) for time intervals of one-year duration covering a span of six years are given in Table 3.11.

Table 3.11

ASSUMED TEMPERATURES AND PRODUCTION RATES

j	Time Interval (days)	T_{Fj} Fuel (°C)	T_{Gj} Graphite (°C)	Production Rate (Moles/day)			
				Cs		Sr	
				B_{1j} 133+135	B_{2j} 137	B_{1j} 88	B_{2j} 90
1	0-365	1240	1145	0.6242	0.3121	0.2140	0.3013
2	365-730	1220	1130	0.6242	0.3121	0.2140	0.3010
3	730-1095	1140	1060	0.5393	0.2697	0.1849	0.2603
4	1095-1460	1010	940	0.4585	0.2293	0.1572	0.2213
5	1460-1825	945	875	0.3818	0.1909	0.1309	0.1843
6	1825-2190	920	850	0.3070	0.1573	0.1053	0.1482

The R values (R_o and E_a) that were used in the calculations were based on postirradiation anneal release data from P9 and P11 capsule fuel particle; these values are given in Table 3.12.

Table 3.12

R_o AND E_a VALUES BASED ON P9 AND P11 CAPSULE DATA

Capsule	Cs		Sr	
	$\ln R_o$	$E_a \left(\frac{\text{kcal}}{\text{mole}} \right)$	$\ln R_o$	$E_a \left(\frac{\text{kcal}}{\text{mole}} \right)$
P9	15.54	75	15	65
P11	17.49	74	11.2	41

The release constant is:

$$R = R_o \exp \left[-\frac{E_a}{1.987} \left(\frac{10^3}{T(^{\circ}\text{K})} \right) \right] .$$

The results of the calculation of strontium release based on P9 and P11 R values are plotted in Figs. 3.23 and 3.24, respectively. With the activated carbon at fuel temperature, there is more than a hundred-fold reduction in the release fraction* of Sr^{90} in the first case and a somewhat smaller reduction in the second case (the effect of the activated carbon adsorber becomes relatively less for fuel particles with higher release rates). With lower temperatures, the activated carbon becomes more effective (i.e., reduces Sr^{90} release by another factor of ten if it can be maintained at the graphite temperature).

The results of the calculation of cesium release, again based on P9 and P11 R values, are plotted in Figs. 3.25 and 3.26. The effect of activated carbon on cesium retention with the carbon at fuel temperature is seen to be negligible. The carbon has an appreciable effect (about an order of magnitude reduction, depending on the rate of release from the fuel particles) if it is maintained at the graphite temperature.

* The release fraction is the activity outside of the fuel element (i.e., released to coolant) compared to the total activity.

Fig. 3.23--Effect of 3.7 wt-% activated carbon on Sr^{90} release from fuel block (FREVP-2 calculation based on P9 R values)

Fig. 3.24--Effect of 3.7 wt-% activated carbon on Sr^{90} release from fuel block (FREVP-2 calculation based on P11 R values)

Fig. 3.25--Effect of 3.7 wt-% activated carbon on Cs^{137} release from fuel block (FREVP-2 calculation based on P9 R values)

Fig. 3.26--Effect of 3.7 wt-% activated carbon on Cs^{137} release from fuel block (FREVP-2 calculation based on P11 R values)

It is concluded from these results that Sr^{90} will be significantly retained by added activated carbon and that activated carbon in general may be expected to be a useful fuel element additive for trapping fission products that are similar to Sr in being strongly sorbed. These include Y, Ba, La, Ce, Pr, Nd, Pm, Sm, and Eu.

These metallic fission products will be the more strongly held by activated carbon (and, indeed, the graphite of the fuel element) the less they are released by fuel particles. This is due to the nonlinear sorption (Freundlich) isotherms characteristic of these metals in graphite and activated carbon. On the other hand, cesium and other rather volatile or less strongly sorbed metals, such as Rb, Ag, and Cd, cannot be expected to be significantly trapped by activated carbon at fuel temperatures.

It is to be noted that although the absolute values of release fractions calculated by FREVAP-2 are subject to substantial uncertainties--perhaps of an order of magnitude due to various approximations and uncertainties in data--the relative values of the release fractions are believed to be meaningful.

The FREVAP code has been extended (as FREVAP-3) so that a proper calculation can be made of the release of metallic fission products from an entire HTGR core or a test fuel block (such as GAIL), which has pronounced spatial (axial and radial) variations of fuel and graphite temperature. A separate calculation is made for each radial zone (or fuel region) that may be assumed to have approximately uniform temperature at any given axial position. The release of an entire HTGR core is then calculated by summing the release from the several radial zones (regions).

In FREVAP-3 a given radial zone (or fuel region) is divided into a number of axial segments indicated by index k of different temperatures including the reflector. The continuity equation (one for each isotope of total number i) is applied to each segment as follows:

$$\frac{A(G)_k}{10^6} \left[\frac{d \bar{C}_{ik}}{dt} + \lambda_i \bar{C}_{ik} \right] = S_{ijk} - \left(\frac{\dot{n}_{\text{He}}}{P_{\text{He}}} \right) [\zeta_k P_{bk} X_{ik} - \zeta_{(k-1)} P_{b(k-1)} X_{i(k-1)}] \quad (10)$$

The symbols in the equation have the same meaning as defined in the previous quarterly report, but with the added index k . The index $(k-1)$ refers to the adjacent axial zone from which the coolant flows. The second term of the right hand side of the equation represents the net rate of release of a fission product from segment k to the coolant stream. This net rate may be negative--especially in the case of segments in the reflector--and will be determined by temperature and concentration of fission product in the graphite relative to that in the coolant stream. Note that in the reflector $S_{ijk} = 0$. The formulation assumes the fission product in the coolant stream to reach a steady-state equilibrium with fission product in the boundary region of the graphite in each segment k .

In the case of the first segment ($k = 1$), at the point where the coolant stream enters the core, the equation has a special form

$$\frac{A(G)_1}{10^6} \left[\frac{d\bar{C}_{il}}{dt} + \lambda_1 \bar{C}_{il} \right] = S_{ijl} - \left(\frac{\dot{n}_{He}}{p_{He}} \right) [\zeta_1 p_{bl} X_{il} - \alpha \zeta_k p_{bk} X_{ik}], \quad (10a)$$

where α is the fraction of the fission product leaving the last segment $k = k$ that returns with the coolant after passing through the coolant circuit. In the case of 100% plateout (deposition) of the given fission product in the coolant circuit, $\alpha = 0$.

The following equations are substituted into Eqs. (10) and (10a) to obtain differential equations in $\bar{C}_{ik}$:

$$p_{bh} = e^{v_k} \left(\sum_i C_{bik} \right)^{u_k}, \quad (11)$$

$$X_{ik} = \frac{C_{bik}}{\sum_i C_{bik}}, \quad (12)$$

$$C_{bik} = \eta_{ik} \bar{C}_{ik}. \quad (13)$$

Thus, Eq. (10) becomes

$$\frac{A(G)_k}{10^6} \left[\frac{d\bar{C}_{ik}}{dt} + \lambda_i \bar{C}_{ik} \right] = S_{ijk} -$$

$$\frac{\dot{n}_{He}}{p_{He}} \zeta_k \exp(v_{jk}) \{ \eta_{ik} \bar{C}_{ik} \}^{(u_{jk}-1)} \eta_{ik} \bar{C}_{ik} -$$

$$\zeta_{(k-1)} \exp(v_{j(k-1)}) \{ \sum_i \eta_{i(k-1)} \bar{C}_{i(k-1)} \}^{(u_{j(k-1)}-1)} \eta_{i(k-1)} \bar{C}_{i(k-1)} \quad (10b)$$

This leads to $(k \times i)$ simultaneous differential equations that are solved by a subroutine used in the FREVAP code.

FREVAP-3 has been successfully programmed and calculations are under way.

The integrated fractional release of isotope i from a given radial zone (or fuel element region) is calculated from

$$F_i(t)$$

$$P \left(\frac{n_{He}}{p_{He} p_i(t)} \right) \left\{ \sum_{j=1}^{n-1} (1 - \alpha_j) \zeta_{jk} \exp(v_{jk}) \int_{t_{j-1}}^t \left[\sum_i \eta_{ijk} \bar{C}_i(\tau) \right]^{(u_{jk}-1)} \eta_{ijk} \bar{C}_i(\tau) \exp[-\lambda_i(t-\tau)] d\tau + \right. \\ \left. + (1 - \alpha_n) \zeta_{nk} \exp(v_{nk}) \int_{t_{n-1}}^t \left[\sum_i \eta_{ijk} \bar{C}_i(\tau) \right]^{(u_{nk}-1)} \eta_{ijk} \bar{C}_i(\tau) \exp[-\lambda_i(t-\tau)] d\tau \right\} \quad (11)$$

The index j specifies a given time interval (between t_{j-1} and t_j) and n is the total number of intervals involved up to time, t , (with the n^{th} interval, in general, not completed). The index k specifies the segment $k=k$ (i. e., the last segment of the given reactor radial zone--from which the coolant stream passes from the reactor core to the coolant circuit).

ζ_{jk} and η_{ijk} are the quantities ζ and η as defined before. The index j is used because, in general, ζ and η are functions of temperature (which may be different for each time interval j).

HELIUM PURIFICATION SYSTEM

Low-temperature Adsorption Studies

A high-pressure helium loop apparatus for the study of dynamic and equilibrium adsorption at low temperatures has been constructed. The first experiments planned are the determination of the dynamic adsorption coefficient and equilibrium isotherm of tritium gas on activated carbon at -196°C .

The apparatus is ready and experiments will be started on arrival of the tritium gas.

High-temperature Adsorption Studies

A second helium-loop system for studies of the dynamic behavior of fission products and carbon aerosols in a high-temperature activated carbon bed is complete and is presently in the final checkout phase. The system is designed so that conditions of temperature, purge gas velocity, and helium pressure expected in the PSC high-temperature plateout trap can be simulated. It is planned to study (1) dynamic behavior of aerosols, both generated in situ and added externally, (2) dynamic behavior of I_2 , and (3) dynamic behavior of Cs:

Carbon Deposition Studies

Apparatus for use in a study of long-term reaction rates for carbon deposition on reactor metals of interest is presently in the design stage.

PART II

TASK I
FUEL TRANSFER MACHINE

Testing of dry-film lubricants at the environmental conditions of temperature and helium impurity level at which certain critical-service bearings in the fuel transfer machine would be expected to operate are continuing. The test conditions are given in the previous quarterly report.

At the beginning of the quarter, Test No. 6 in this series, the evaluation of DF-700 (a Marlin Rockwell Corporation proprietary coating) was under way. As the quarter ended, the test bearing had logged approximately 600 hr and was continuing to operate without any evidence of increasing torque.

Two additional bearings have been procured. These bearings will be dry-film lubricated with DF-700, and Test No. 6 will be repeated to obtain confirming performance data for this lubricant.

TASK II

SERIES-STEAM-TURBINE-DRIVEN CIRCULATOR

HELIUM CIRCULATOR DEVELOPMENT

Over-all design studies on the circulator have continued for the purpose of incorporating the results of the Mechanical Technology Incorporated and Franklin Institute water-lubricated bearing designs, as well as further facilitating the installation of the machine into a prestressed concrete pressure vessel. Figure II.1 shows the present design concept. The circulator rotor is assembled to a center bearing housing that is attached to a conical head that forms the closure for the PCR.V. The compressor, rotor, stator, inlet, and diffuser are inside the pressure vessel, whereas the steam-turbine driver is outside the closure. The casing for the steam turbine is also attached to the bearing housing.

The primary and secondary fluids, helium and steam, are separated from each other by the water lubricant and seals. On the compressor side, clean helium is used as a buffer. It enters the center of a two-element labyrinth seal and leaks in both directions, into the compressor and into a scavenge chamber outboard of the water seal adjacent to the compressor side journal bearing. The scavenge chamber is connected to the compressor inlet through a gas-liquid mechanical separator and molecular sieve that prevent water and water vapor from entering the primary circuit. The water seal is a viscoseal that uses the shear action of lands on one of the surfaces to create a pressure head. For a given pressure difference between the bearing and the scavenge chamber, and at a particular speed, the water reaches an axial position consistent with the balancing of the forces involved. As the speed is reduced, the water level moves toward the scavenge chamber; and at a sufficiently low speed, there will be water flow into the scavenge chamber. In this design the length of the viscoseal is 2-1/2 in.; at design speed the water engagement is approximately 1 in., and no leakage will occur at speeds in excess of 60% of design. At the lower speeds, the water and buffer helium are scavenged so that there is no leakage into the primary circuit. The vertical orientation of the machine and a slinger between the labyrinth seal and the water seal prevent the leakage of water into the gas portion of the machine.

The seal on the steam side is a hydrostatic seal in which water at a pressure elevated above that of the steam and lubricant systems enters the center of the seal and flows in both directions. An axial clearance is

Fig. II.1--Helium circulator with water-lubricated bearings

maintained by the pressure of the entering water, which leaks into the bearing and into the turbine discharge. This seal also serves as a shutdown seal when the water-pressure differential is removed. There is also another shutdown seal on the helium side between the viscoseal and the bearing, which is activated by a decrease in water pressure.

Four different bearing designs have been made. Two are hydrodynamic with hydrostatic lift-off for operation up to 70% speed, and the other two are hybrid, i. e., essentially hydrostatic with considerable hydrodynamic action at design speed. It is planned to test these bearings in the test program and select the one with the best performance capability. Both of the hydrodynamic thrust bearings have a capability for tolerating misalignment. One of these is of the Kingsbury, tilting-shoe type, and the other is a picket step bearing, which has a spherical seat that is hydrostatically lubricated by the hydrodynamically developed pressure in the step bearing, so that no rubbing takes place. This may, in fact, prove to be an advantage over the tilting-shoe type, in which there is relative motion over the pivots, and hence, a possibility for wear.

ROTOR DYNAMIC SIMULATION AND LUBRICATION AND SEAL SYSTEM TEST

The designs of the simulator test rig, test equipment, test facility, and the water-lubricated bearings have been completed. The detail drawings that are required for parts fabrication are in preparation. The test program has been established. Bids for the manufacture of the test rig, construction of the test facility, and conduct of the test program have been prepared by Mechanical Technology, Inc., Franklin Institute, and internally within General Atomic, and are being evaluated.

In the simulator, the polar moments of inertia and transverse radius of gyration are made identical to those in the actual circulator by disks. The method of attachment of the disks to the shaft is the same as in the circulator. The thrust loads are imposed by pressures applied on either end of the rig within the closure heads. The temperatures are simulated by heaters in either end and by disk friction. The heaters are controlled for adjustment to the temperature levels that will exist at the various plant operating conditions. The flows, temperatures, and pressures of the fluids are all measured, as are the bearing-pad pressures. Radial motion of the rotor will be measured in five axial planes at two positions in each plane 90° apart. Axial motion will be measured on the thrust runner in two positions. Figures II. 2 and II. 3 are schematic diagrams of the test equipment and instrumentation for the test. The equipment, instrumentation, and control devices are sufficient to determine the complete envelope of operation so that the design can be optimized.

Fig. II.2--Instrumentation and flow schematic for the helium circulator rotor dynamics simulator

Test Program Outline

The following is an outline of the planned test procedure:

1. Speed

The performance of the hydrostatic bearings will be tested in the range 0% to 110% of design speed (5700 rpm) at seven separate speeds and four different supply pressures. The performance of the hybrid bearings will be tested under hydrostatic operation at seven separate speeds between 0% and 110% of design speed (5700 rpm) at four different supply pressures. The performance of these bearings under hydrodynamic operating conditions will be determined at seven separate speeds between 50% and 110% of design speed.

2. Thrust

Four separate thrust loads are to be applied at each speed point: one at the design value for each speed, two at lower thrust levels, and one at a thrust higher than design thrust, the exact amount of which is to be determined during test by observation of the minimum film thickness.

3. Dynamic Load

Tests will be performed to determine the effects of four different values of dynamic load over the entire speed range, for both hydrodynamic and hydrostatic bearings and for a total of ten different speeds. The dynamic Load will be produced by appropriate unbalance of the rotating assembly.

4. Special Tests

A simulation of the startup cycle (4 to 6 hr) will consist of a programmed set of speed and thrust conditions over the entire speed range of 0% to 100% speed (8 points).

A simulation of the refueling operation, which is at atmospheric helium pressure and at atmospheric steam pressure will consist of (1) shutdown from design conditions, (2) reduction of pressure to atmospheric, and (3) acceleration to design speed.

A simulation of upset conditions will take into account such possibilities as main turbogenerator trips, control malfunctions, and changes in primary system pressure levels.

5. Start and Stop

Start and stop testing will include twenty-five start and stop cycles between 0% and 100% of design speed.

STATIC AERODYNAMIC COMPONENTS

Preliminary investigations of the program content and the facility requirements are being worked on. Discussions have been held with a number of possible vendors for this phase of the work, but it has not yet been decided how extensive the program should be or whether it will be performed at General Atomic or by a vendor.

PROTOTYPE TEST FACILITY

The preliminary design of the test facility for the prototype circulator is essentially complete. A drawing of the test facility was included in the previous quarterly report.

TASK III

CONTROL ROD DRIVE

The intermittent stall condition of the drive reported last quarter was resolved as being in the shim motor and not the result of improper brake release, as was at first suspected. Examination of the rotor assembly revealed an excessive amount of end play resulting in sufficient bearing radial play to displace the rotor and cause it to rub against the stator laminations. This condition occurred only when the bearings were under load, i. e., when radial loading was imposed on the rotor shaft by pinion gear teeth separation forces.

It was concluded that end play was probably not detectable during assembly of the motor, because the bearings at that time were somewhat tighter in their housings; differential thermal expansion of the aluminum motor end bells would tend to loosen the bearing/housing fit at elevated temperatures and, in the absence of preload, possibly allow the bearings to rotate in the housing causing some wear. Future motors will require either steel sleeves shrinkfitted into the end bells, or, as an alternative, steel end bells.

End play was taken up on the unit under test by the addition of shims, one shim being a wave washer to maintain a constant bearing preload.

Satisfactory operation was achieved following the foregoing rework, brake torque being normal at 12 to 13 in. -lb and starting surge consistent with pretest conditions, namely, 3.0 to 4.0 amp.

Phase II environmental testing resumed on October 20 and proceeded satisfactorily during the rest of the quarter with only one minor interruption when a torque reaction pin in the shim motor brake became loose due to improper staking. Corrective action was taken by welding both pins in place and the test resumed. As the quarter ended, 2569 hr of test time have been logged, of which 484 hr were operational.

A summary of the actual operations performed to date is as follows:

	<u>Number</u>	<u>Rod Travel (ft)</u>
Shim strokes	1,931	20,142
Scrams	582	6,172
Postscram shim rod insertions	422	4,390
Oscillations	118,072	39,000
Cumulative rod travel . . .	---	69,704

Gas chromatograph readings continued to indicate a progressive increase in CO content; consequently, the Raytron purification unit was reinstalled in the loop together with a dryer at the outlet to remove moisture added by the Raytron unit.

TASK IV

STEAM GENERATOR

MATERIALS

Evaluation of the data resulting from the metal-coolant compatibility 18,000-hr exposure tests has continued. The additional testing of the Incoloy 800 material has been completed. In these tests, Incoloy 800 specimens were creep-tested for 500 hr in order to evaluate the effects of heat treatment on the physical properties of this material.

The finalized tube integrity investigation program is under way primarily as a search of the available literature. From the collected information, it is intended that a steam generator tubing specification be developed. In addition, it is expected that appropriate answers to the quality and serviceability of steam generator tubing will be immediately available.

HEAT TRANSFER AND FLUID FLOW

Steam/Water Flow Stability

An analysis that is being used to extend the digital computer dynamic boiling stability code into the once-through regime with superheat has been completed. Transfer functions were developed for the superheat regime and for the transition between vaporizing and superheating regimes. The linearized partial differential conservation equations were solved using Laplace transformation of the temporal terms and integration of the spatial variations. The resulting equations represent the transfer functions for each spatial node. A computer program has been written in the necessary complex variable forms using frequency response techniques. Code debugging is presently under way. Comparisons will be made between code predictions and the few experimental results available regarding once-through boiling flow oscillations.

An analytical development of the static stability index for once-through boiling with superheating was completed. The analysis was based upon the simplifying assumption of constant and uniform heat input. The index represents the normalized slope at any point of the steady-state pressure drop versus flow rate characteristic curve for a heated channel. Various example cases were computed corresponding to typical HTGR steam generator operating conditions. Effects studied were those of interposing intermediate mixing headers and changes in tube length,

feedwater temperature, orificing, and friction correlations. Cases involving more complexity, such as that of a fixed helium temperature profile, are being explored using an existing computer program. In general, the results indicate that use of an intermediate mixing header between evaporator and superheater does not have a large effect on the stability index.

Tube Vibration Studies

The tube vibration test facility and associated instrumentation have been assembled and checked out, and testing has started. Two series of tests are planned during the Phase I testing. In the first series, the tubes will be oriented in a triangular-pitch pattern with respect to the direction of air flow. In the second series, the tube bundle will be rotated 90 deg from its original position, so that the tubes are oriented in an in-line pattern with respect to the direction of flow.

DESIGN AND ANALYSIS

The steam generator supplier design studies have continued during this period, and all studies are essentially completed. The detailed design specification is being prepared on the basis of the progress of the studies to date and will be finalized shortly after their completion. This specification, which will enable the suppliers to complete the design work, will be followed by a specification that will detail fabrication and inspection requirements. The basic design arrangement now being worked on is shown in Fig. IV.1. It is expected that several variations of this arrangement with regard to the type of tube bundle utilized and the design of the penetration area will be presented in the supplier designs.

In addition, other variations of the gas flow arrangement are being considered on a conceptual basis. These variations involve a complete separation of the reheater section from the economizer-evaporator-superheater section, so that the reheater is a complete module in itself. Arrangements of this type would utilize five economizer-evaporator-superheater modules and three reheater modules per steam generator, whereas the unit shown in Fig. IV.1 utilizes eight identical modules. Several variations of the tube terminations in the area of the PCRV penetration are being considered in an effort to simplify the tube termination while retaining a high degree of integrity at the closure.

STEAM GENERATOR TUBE PLUGGING MACHINE

Assembly of the weld preparation tool head was completed in the latter part of this period, and preliminary checkout testing was satisfactorily completed.

Fig. IV.1--Basic design arrangement for the steam generator

Actual cutting tests on tube-to-tubesheet weld specimens have been delayed because the special cutting tools have not yet been received; delivery is being expedited.

Preliminary studies have been made in connection with the miniaturization of the various tool heads in order to achieve a minimal turret diameter. These studies were based on a steam generator with small concentric horizontal tube sheets having tubes 1.0 in. in diameter, or less.

As a result of these studies, it appears feasible to design a tool turret carrying the five special tool heads, together with its supporting machinery and structure, capable of being inserted through a 7-in. diameter aperture.

TASK V

COATED PARTICLE DEVELOPMENT

BUFFER COATING STUDIES

An intensive study of the inner layer of the reference two-layer BISO coated particle is being conducted. This inner layer is interposed between the fuel particle and the dense outer coating to accommodate swelling of the fuel particle and to absorb fission recoils. The objective of the study is to obtain a more complete understanding of the properties required for the inner layer to enhance the radiation stability of coated fuel particles. Information gained during this study will aid in optimization of the reference coating process.

Deposition Studies

Three different approaches to producing a coating that will satisfy the requirements of the inner layer of the BISO fuel particle are being considered. First, the proven concept of using a soft, low-density, isotropic coating (true buffer coating) is being systematically studied. Second, the use of a low-density, anisotropic coating that is not soft is being evaluated. This type of coating should be more economical to produce than the true buffer coating. Third, the laminar inner coating that contains the core particle loosely within a central void will be evaluated.

Diffusivity Measurements

The flash method is being adapted to measuring the thermal diffusivity of low-density buffer coatings. In this technique, one surface of a sample with slab geometry is irradiated with a pulse of energy from an optical source, and the subsequent temperature history is monitored at the opposite surface. From the shape of this transient, the thermal diffusivity of the sample material may be deduced provided the heat-conduction equation has been solved for the boundary conditions of the experiment.

A computer program that has been developed for the reduction of data from the flash method for composite samples has been debugged and successfully tested. This method permits the computation of thermal diffusivity and conductivity in one of the layers of a two-layer sample if the corresponding values are known for the second layer. If the thermal diffusivities of both layers are known beforehand, the method affords qualitative information on the magnitude of the thermal contact resistance

at the interface between the two layers. Through the introduction of an empirical function accurately describing the actual pulse shape of the xenon flash tube employed, the effect of finite pulse length is automatically taken into account, both for the two-layer and the single-layer computations. This feature permits the use of thin samples, thus making it possible to employ the flash method at elevated temperatures for measurements on both one- and two-layer samples without the necessity of correcting for heat losses.

Flash-method oscilloscope traces have been obtained for a series of buffers deposited on ATJ graphite substrates. Traces have been obtained for samples with varying thickness of the buffer layer on the graphite substrate, and comparison of the computed thermal diffusivities of the buffers with the values obtained using samples with the graphite layer removed will provide a qualitative indication of the magnitude of the thermal contact resistance between the two layers. An attempt will be made to correlate the information so gained with the results of a metallographic examination of the interfaces.

HIGH-DENSITY ISOTROPIC COATINGS

An intensive study of high-density, isotropic PyC coatings from methane is being conducted. The results of the G9 irradiation experiment on pyrolytic carbon indicate that this structure should have the greatest radiation stability of the structures tested to date. The objectives of this study are to gain an understanding of the high-density isotropic coating region and to optimize the coating process necessary to obtain the high-density structure.

Part of the study includes an investigation of the effects of variations in coating parameters on the PyC structure. A series of coating runs was carried out at 2000° and 2200°C with variations in methane concentration (5 to 20%), total gas flow (6,000 to 15,000 cm³/min), and bed surface area (800 to 2500 cm²).

COATED PARTICLE PREPARATION

Irradiation testing of promising types of reference coated particles is being carried out in connection with the PSC reactor program. Preparation and evaluation of coated particles for these irradiations is being done as part of this task. These irradiation tests are intended to guide the process development of reference BISO coatings with optimum radiation stability and retentivity and the lowest possible fabrication costs. During the reporting period, coated particles were prepared for the following experiments.

Dounreay Fast Reactor Irradiation

A series of 15 PyC structures has been selected for irradiation in the Dounreay Fast Reactor (DFR). The objectives of this irradiation experiment are to study the effects of coating density and preferred orientation on the radiation stability of PyC under high fast-neutron exposures. Included are nine high-density structures obtained in the systematic study, two additional high-density structures that were deposited over a reference buffer layer, and four control structures that were previously irradiated. For each structure, unrestrained PyC strips, PyC-coated graphite disks, and coated depleted UC_2 particles (except for the control structures) will be irradiated.

Two capsules will be used in the DFR irradiation: one in a core position, the other in a blanket position. Both capsules will operate at 1000° to $1200^{\circ}C$; the core capsule will experience an integrated flux $\sim 1 \times 10^{22}$ nvt, and the blanket capsule will experience an integrated flux of $\sim 4 \times 10^{21}$ nvt. Both capsules will contain all of the structures, but the blanket capsule will not contain any of the coated graphite disks because of the limited number available.

Capsule P14

Twelve types of coated particles to be irradiated in Capsule P14 were prepared and characterized. A description of the particle types is given in Table V.1.

The objective of the P14 experiment is to evaluate the effects of density and preferred orientation of the isotropic PyC layer on the radiation stability of the reference BISO coating. The density range covered for the isotropic coatings will be 1.94 to 2.14 g/cm³. The range of anisotropy factors will be 1.0 to 1.9. The buffer layer of the coating will be kept constant on all particles. The T/D ratio will be varied only on the fertile particles. This test is intended to guide the selection of minimum coating specifications and optimum process conditions for subsequent production.

ORNL Irradiations

Two batches of BISO coated particles have been prepared for irradiation in the ORNL 41-28 high-burnup experiment. The particles have fully enriched 150- to 250- μ UC_2 cores coated with ~ 40 μ of reference buffer. The isotropic overcoats are ~ 2.14 g/cm³ PyC with a Bacon anisotropy factor of ~ 1.2 . The total coating thicknesses are 125 and 150 μ respectively. These particles will experience 25% to 45% FIMA burnup at a maximum fuel temperature of $1400^{\circ}C$.

Table V.1
DESCRIPTION OF COATED PARTICLES TO BE TESTED IN CAPSULE P14

Lot Number	Coating ^a						Particle			T/D Ratio	Design Irradiation Conditions ^b		Test Type	Remarks
	Type	B	I	Total	Density (g/cm ³)	Anisotropy Factor	Type	Size (μ)	Th:U Ratio		Temperature (°C)	Burnup ^c (% FIMA)		
3222-85E	BISO(2100)	27	72	99	1.94	1.0	(Th,U)C ₂	150-250	3:1	0.50	1400	20	Loose	BISO density series
3143-135E	BISO(2200)	31	73	104	2.03	1.1	(Th,U)C ₂	150-250	3:1	0.52	1400	20	Loose	
3222-71E	BISO(2200)	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	
3222-75E	BISO(2200)	32	74	106	2.13	1.2	(Th,U)C ₂	150-250	3:1	0.53	1400	20	Loose	
3222-71E	BISO(2200)	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	BISO preferred orientation series
3222-77E	BISO(2200)	27	74	101	2.11	1.3	(Th,U)C ₂	150-250	3:1	0.51	1400	20	Loose	
3222-145E	BISO(2200)	28	69	97	2.11	1.3 ^d	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	
3222-91E	BISO(2200)	28	73	101	2.10	1.9	(Th,U)C ₂	150-250	3:1	0.51	1400	20	Loose	
3222-71E	BISO(2200) ^e	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	BISO crystallite size series
3143-133E	BISO(2000) ^e	29	68	97	2.12	1.1	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	
3222-71E	BISO(2200)	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Loose	Loose versus bonded particles
3033-151D(P2) ^f	BISO(2200)	32	72	104	2.14	(g)	UC ₂ ^h	150-250	---	0.52	1400	20	Bonded-Pitch	
3222-71E(U1) ^f	BISO(2200)	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1400	20	Bonded-Furfural	
3143-135E	BISO(2200)	31	73	104	2.03	1.1	(Th,U)C ₂	150-250	3:1	0.52	1000	20	Loose	BISO density and preferred orientation series at lower irradiation temperatures
3222-71E	BISO(2200)	29	68	97	2.11	1.2	(Th,U)C ₂	150-250	3:1	0.49	1000	20	Loose	
3222-77E	BISO(2200)	27	74	101	2.11	1.3	(Th,U)C ₂	150-250	3:1	0.51	1000	20	Loose	
3222-129E	BISO(2200)	31	73	104	2.10	1.2	(Th,U)C ₂	300-420	12:1	0.30	1400	6	Loose	BISO fertile particle T/D series
3222-127E	BISO(2200)	34	97	131	2.09	1.2	(Th,U)C ₂	300-420	12:1	0.37	1400	6	Loose	
3222-127E	BISO(2200)	34	97	131	2.09	1.2	(Th,U)C ₂	300-420	12:1	0.37	1400	6	Loose	BISO fertile particle density series (Heater particles for bonded bed P2)
3222-105E	BISO(2200)	34	98	132	2.04	1.1	(Th,U)C ₂	300-420	12:1	0.38	1400	6	Loose	

^a BISO denotes two-layer Buffer-Isotropic PyC coating, the number in parentheses denotes the deposition temperature of the isotropic layer in °C.

B" denotes buffer layer and "I" denotes isotropic layer.

^b Design fast-flux exposure is 4×10^{21} nvt for four cycles in ETR.

^c During four cycles in ETR.

^d Measurement is probably in error. The anisotropy factor is believed to be ~1.6.

^e These two samples differ in crystallite size.

^f Designation for bonded sample.

^g Not determined.

^h Depleted U.

Capsule G11

Two sets of BISO-coated graphite disks were prepared for the G11 capsule. The disks had isotropic coatings of ~ 2.00 and ~ 2.10 g/cm³, respectively, deposited over the reference buffer. These disks will be used to evaluate the accommodation of the buffer layer to the irradiation shrinkage of the isotropic overcoat.

COATING EVALUATION TECHNIQUES

Characterization of PyC coatings is necessary for predicting and understanding the performance of coated particles under irradiation and thermal testing. These studies are intended to improve and add to existing techniques used in determining coating parameters. Methods of determining the density of PyC coatings (in particular, fragile buffer coatings) have been under development during the reporting period.

X-Ray Absorption Density Determination

A new technique for determining coating densities is being developed that shows density variations by the difference in X-ray absorption of the material. A large cross section, about 150 μ thick, is removed from a coating disk, placed on a photographic plate, and exposed to an X-ray beam. The plate is then scanned with a densitometer and the readings are plotted against the log of the coating densities determined by the standard sinkfloat technique.

The determination of coating density by X-ray absorption appears promising. The following observations are relevant:

1. The technique shows density irregularities in the coatings.
2. It is the only method for measuring densities on thin, fragile buffer layers at the present time.
3. The fragile coating need not be removed from the disks and may even be overcoated with a support coating or multiple coatings of any density.
4. The method is currently limited to coatings of ~ 50 μ or greater.

Current work is directed toward determining the penetration characteristics of the X-ray beam with a variable-thickness graphite wedge. The wedge is made of PyC slivers about 10 μ thick that can be stacked to give various thickness layers of a known density material.

Electron Microprobe Density Determination

A feasibility study was performed on the use of the electron microprobe as a tool for determining the density of PyC coatings directly on fuel particles. The microprobe can determine the coating density on a particle by relating it to the voltage required to penetrate the PyC and excite the fuel substrate, which causes the emission of measurable gamma rays.

The results of this feasibility study show that the microprobe can be successfully used to determine the density of PyC carbon coatings; the experimental error is small with the greatest error caused by variations in the particles such as nonuniform coating thickness. It has been found, however, that the method has the following limitations:

1. Only thin coatings can be measured (e. g. , coatings $\leq 20 \mu$ with a density of 2.0 g/cm^3 , or $\leq 40 \mu$ with a density of 1.0 g/cm^3).
2. Fuel migration causes interference.
3. Radial variations in coating density will not show.
4. Analysis time is long compared to present methods.
5. Multiple coatings cannot be analyzed separately.

Because of these limitations, no further development of the microprobe technique for coating density determination is planned.

Particle Geometry (T/D Ratio) Studies

The variation of T/D ratio in selected coated particles used in irradiation experiments is being studied. A brief analysis of four samples indicated that the core diameter as determined by sieving was not significantly different from that determined by radiography, thereby causing no change in the T/D numbers previously obtained.

It is now believed that a more detailed analysis should be made to investigate the variations in T/D ratio within each group of particles and between groups. The variations will be determined by measuring the minimum PyC coating thickness and the average core diameter on each of fifty-five particles in every group.

THERMAL STABILITY STUDIES

Long-term thermal tests of three samples of BISO coated (Th, U)C₂ particles with high-density coatings are in progress. These particles, which are typical of reference BISO-coated particles, were previously

shown to have good radiation stability in P-capsule experiments. This experiment has reached 2180 hr at 1400°C, and examination of the particles indicates negligible migration. The next examination point will be 2500 hr.

Fission-product Release

Postactivation experiments* have been performed on a number of samples of BISO-coated particles. In postactivation experiments, coated particles are given light irradiations (around 10^{13} fissions) in the TRIGA reactor. This degree of irradiation yields sufficient 9.7-hr Sr^{91} and 12.8-d Ba^{140} for counting purposes. (If Cs^{137} release data are required, however, an irradiation corresponding to around 10^{16} fissions must be carried out.) After activation, the particles are placed in a graphite crucible and annealed in a King (carbon-resistance) furnace in a helium stream at 1400°C, or any other desired temperature. During the anneal, the strontium and barium nuclides released from the particles are collected in the graphite crucible. The anneal is interrupted at intervals of about 1, 3, 5, and 8 hr, and the particles are transferred to a new crucible. The exposed crucibles are immediately gamma-counted to determine Sr^{91} release and again after several days (when sufficient La^{140} has grown in) to determine the Ba^{140} release. The fraction of Sr^{91} and Ba^{140} released is the ratio of the quantities of these nuclides released during the anneal and the quantities produced in the irradiation, taking decay into account.

The total number of fissions is determined from the La^{140} activity (measured several days after the irradiation). The amounts of Sr^{91} and Ba^{140} produced are calculated from known fission yields and decay times.

Temperature is measured optically, and the temperature uncertainty is believed to be $\pm 25^\circ\text{C}$.

The experimental results are given in Table V.2. The results, in general, show that (1) all the particles released strontium and barium to an appreciable extent, (2) the total release increased with time as might be expected for a diffusion-controlled process, and (3) strontium releases were higher than barium releases, indicating a higher diffusivity for strontium in agreement with the results of postirradiation experiments on similar samples.

A comparison of the accumulated release fraction values at around 8 hr for the various particles shows that the fractional release values for Sr^{91} differ by as much as a factor of five and those for Ba^{140} by as much

* This is a new designation for annealing experiments using lightly irradiated particles and will be used in subsequent reports.

Table V. 2

FRACTIONAL RELEASE VALUES FOR BISO-COATED
CARBIDE PARTICLES

(Determined by Postactivation Annealing Experiments at 1400°C)

Sample Description	Annealing Time (hr)	Accumulated Fractional Release	
		Sr ⁹¹	Ba ¹⁴⁰
150-250 μ UC ₂ ; 102 μ , BISO(2000), ρ = 2.02 g/cm ³ coatings	1.0	0.014	0.0018
	3.0	0.029	0.0077
	7.0	0.041	0.021
	24	0.050	0.043
250-350 (Th, U)C ₂ ; 102 μ , BISO(2000), ρ = 1.97 coatings	1.0	0.017	0.0022
	3.0	0.036	0.0059
	7.0	0.059	0.022
	24	0.083	0.046
177-250 μ (Th, U)C ₂ ; 70 μ , BISO(2000), ρ = 1.95 coatings	1.0	0.040	0.0096
	3.0	0.081	0.025
	8.5	0.14	0.058
	17.8	0.21	0.11
177-250 μ (Th, U)C ₂ ; 82 μ , BISO(2000), ρ = 1.94 coatings	1.0	0.038	0.0067
	3.0	0.058	0.014
	8.5	0.13	0.055
	17.8	---	0.080
177-250 μ , (Th, U)C ₂ ; 97 μ , BISO(2000), ρ = 1.91 coatings	1.0	0.046	0.011
	3.0	0.076	0.024
	4.8	0.11	0.046
	9.3	0.16	0.070
177-250 μ , (Th, U)C ₂ ; 141 μ , BISO(2000), ρ = 1.88 coatings	1.0	0.017	0.0011
	3.0	0.051	0.010
	4.8	0.077	0.021
	9.3	0.11	0.036
177-250 μ , (Th, U)C ₂ ; 98 μ , BISO(2200), ρ = 2.12 coatings	1.0	0.071	0.010
	4.0	0.12	0.086
	7.3	0.14	0.14
	11.3	0.14	0.15
177-250 μ , (Th, U)C ₂ ; 105 μ , BISO(2100), ρ = 1.99 coatings	1.0	0.037	0.0053
	4.0	0.13	0.067
	7.3	0.19	0.15
	11.3	0.21	0.17

as a factor of five and those for Ba¹⁴⁰ by as much as a factor of seven; however, release values, coating densities, and coating thicknesses do not appear to correlate in any obvious manner. The relatively high release values for the last two samples listed in Table V.2 suggest the possibility of a relatively high degree of uranium contamination in the coatings which, because of the high coating temperatures involved, would not be surprising.