GA Technologies

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HTGR FUEL TECHNOLOGY PROGRAM

SEMIANNUAL REPORT FOR THE PERIOD ENDING MARCH 31, 1983

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ABSTRACT

This document reports the technical accomplishments of the HTGR Fuel Technology Program at GA Technologies Inc. during the first half of FY 83. The activities include the fuel process, fuel materials, fuel cycle, fission product transport, core component verification, and core technology transfer tasks necessary to support the design and development of a steam cycle/cogeneration (SC/C) version of the HTGR.

LIST OF ACRONYMS

AIPA	accident initiation and progression analysis
ASME	American Society of Mechanical Engineers
BOC	beginning of cycle
BOP	balance of plant
CACS	core auxiliary cooling system
CAHE	core auxiliary heat exchanger
CEA	Commissariat a l'Energie Atomique
CWBS	contract work breakdown structure
DBA	design basis accident
DBDA	design basis depressurization accident
DOE	Department of Energy
DV&S	design verification and support
EFPD	effective full power days
EOC	end of cycle
FIMA	fissions per initial metal atom
FRG	Federal Republic of Germany
FSV	Fort St. Vrain
FTE	fuel test element
GA	GA Technologies Inc.
GASSAR	General Atomic Standard Safety Analysis Report
GCRA	Gas Cooled Reactor Associates
HEU	high-enriched uranium
HTGR	high-temperature gas-cooled reactor
HTGR-GT	high-temperature gas-cooled reactor - gas turbine
HTGR-PH	high-temperature gas-cooled reactor - process heat
HTGR-R	high-temperature gas-cooled reactor - reformer
HTGR-SC	high-temperature gas-cooled reactor - steam cycle
HTGR-SC/C	high-temperature gas-cooled reactor - steam cycle/cogeneration

HTR	hochtemperatur reaktor
ILTI	inner low-temperature isotropic
IPyC	inner pyrolytic carbon
KFA	Institut fur Reaktorwerkstoffe der Kernforschungsanlage
KFA/HBK	Institut fur Reaktorwerkstoffe der Kernforschungsanlage Brennstoffkreislauf
LASL	Los Alamos Scientific Laboratory
LEU/Th	low-enriched uranium/thorium
LHTGR	large high-temperature gas-cooled reactor
LMFBR	liquid metal fast breeder reactor
LTR	licensing topical report
LWBR	light water breeder reactor
LWR	light water reactor
NHSDR	nuclear heat source demonstration reactor
NRC	Nuclear Regulatory Commission
NSS	nuclear steam supply
OLTI	outer low-temperature isotropic
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PCRV	prestressed concrete reactor vessel
PIE	postirradiation examination
PSAR	Preliminary Safety Analysis Report
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
R/B	release rate/birth rate
R&D	research and development
SAR	Safety Analysis Report
SWU	separative work unit
VHTR	very high temperature reactor
WAR	weak acid resin

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1. INTRODUCTION

This report is the HTGR* Fuel Technology Program Semiannual Report for the period ending March 31, 1983.

The report is organized around the work breakdown structure (WBS) established to plan, control, and report the programmatic aspects of the technology activities. Figure 1-1 shows this structure. The activities reported include all tasks under WBS-11, along with the core and reactor internals tasks under WBS-14.

The key objectives of the program during this period were as follows:

1. Continue work on development of LEU/Th (<20% enriched LEU) fuel materials and processes.
2. Focus on resolution of design data needs for demonstrating the HTGR steam cycle/cogeneration (HTGR-SC/C) plant concept.
3. Confirm the validity of design methods and assumptions used for HTGR fuel and core components by carefully selected verification testing of appropriate test specimens.
4. Continue international cooperation on HTGR fuel technology.

The summary (Section 2) presents the key activities covered during the reporting period. The subsequent sections of the report provide the details of the work performed and the results obtained.

*A list of acronyms used throughout the report is given on page v.

(a) INCLUDED IN CONSOLIDATED NUCLEAR FUEL REPROCESSING PROGRAM

Fig. 1-1. HTGR fuel technology program work breakdown structure

2. SUMMARY

2.1. GENERAL (0102)

A set of documents defining the planning data base for the HTGR-SC/C technology program was issued to participating contractors. This set will provide the baseline for future change requests in the technical tasks. The planning data base includes:

- Program network resource evaluation and control system (RECS) diagrams and dictionary sheets.
- A cross-referenced index identifying the technology networks that support the parallel design networks.
- A cross-referenced listing of Design Data Need documents and the supporting technology program activities.

2.2. FUEL (11)

2.2.1. Fresh Fuel Process Development (1101)

Scaleup process development studies for fabricating UO₂ fissile fuel kernels continued. Process parameters for broth preparation were updated, as were conditions for sphere formation/gelling processes. Wet kernel drying conditions were further studied, with the objective of establishing reference drying parameters. Testing and an engineering evaluation of candidate materials for the scaleup 1800°C sintering/conversion furnace retort resulted in the selection of graphite for this application. Modifications to the engineering-scale rotary sintering/conversion furnace have resulted in consistent achievement of density and conversion to the desired UC₂ and UO₂ phases.

A series of ten buffer/ILTI coating runs were performed in the 240-mm-diameter coater. The coatings were deposited on 10-kg batches of 500- μ m-diameter ThO₂ kernels to establish current baseline coating capability with respect to HTGR-SC/C specifications. Analysis of the data indicates relatively large thickness variations, both batch-to-batch and within-a-batch, and improvement is required to achieve desired product yields. Otherwise, the coatings fully meet the preliminary HTGR-SC/C fuel specifications.

Additional QC analyses were performed on TRISO coatings deposited by Hochtemperaturreaktor-Brennelement GmbH (HOBEG) as part of the U.S./FRG Cooperative Fuel Development Program. The coatings had been deposited on ThO₂ particles prepared by GA as well as those prepared by HOBEG. GA QC measurements confirmed the HOBEG QC results for the defective SiC coating fraction. However, some discrepancies were observed when comparing GA and HOBEG measurements of SiC coating thickness.

The liquid/gas elutriation unit was used to separate defective ThO₂ kernels from the kernel composite used for the baseline TRISO coating process evaluations. Nitrogen gas elutriation of 120 kg of ThO₂ kernels reduced the fraction of defective kernels from approximately 1% as-received to less than 0.1%, with >98% product yield.

2.2.2. Fuel Materials Development (1102)

Test results were completed on core heatup simulation tests (CHST's) of German LEU TRISO UO₂ fuel particles that had been irradiated to high burnup. The release of fission gas was low until a temperature of 2000°C was reached, indicating that the particle coatings stayed intact until this temperature. The release was complete at 2500°C. This irradiated German fuel performed the same as GA fuel under similar conditions. The results are consistent with the generic model for coated particle performance under accident conditions.

The postirradiation results from capsule HRB-16 provide strong support for other data on the strength of SiC during irradiation to high exposure. The SiC tensile strength is now assumed to be 350 MPa (51 kpsi) in the TRISO coated particle performance model. This is an increase from the previously assumed value of 33 MPa (4.8 kpsi).

The LEU TRISO fuel particles and rods were fabricated for the HRB-17 and HRB-18 capsule irradiation tests. The characterization and delivery to ORNL of these fuel specimens are planned for the next reporting period. HRB-17 and HRB-18 will be important tests for quantifying the effects of moisture impurities on the release of fission gases from defective coated particles.

A test procedure was written for the postirradiation examinations of FSV FTE-2, scheduled for removal from the reactor in 1984.

2.2.3. Fuel Cycle (1105)

Evaluations were made on the input data for fuel cycle component costs. This effort continues into the March-September reporting period, and will be reported in the next semiannual report.

2.2.4. Fission Product and Coolant Chemistry (1106)

Fission product release experiments were conducted in the high-flux isotope reactor (HFIR) irradiations at ORNL (capsules HRB-15A, HRB-16, and HT-35) and in the FRG program (capsule R2-K13). The samples, consisting of niobium-clad graphite crucibles with coated fuel particles, have been returned to GA for gamma counting. The comparisons of measured and calculated inventories show good agreement for cesium, ruthenium, and zirconium fission products. Some uncertainty requires further resolution for silver. The equipment for complete disassembly of the samples has been constructed.

The assembly of a laser optical system for the measurement of cesium and silver vapor pressures over graphite and metals is largely completed.

The remaining work is to perform final calibrations and test runs. This system allows measurement of vapor pressures at less than 10^{-15} atmospheres at 1000 K.

Fission product data were tabulated from fuel irradiation tests conducted from 1977 through 1981 in FRG/KGA capsules. These data will be used in establishing release and transport coefficients for a joint U.S./FRG Fission Product Data Book.

2.2.5. Fuel Trade Studies (1107)

Three trade studies were initiated during this reporting period in response to recommendations made in the HTGR-SC/C Baseline Review held in FY 82. The trade studies are on (1) fuel process unit operations, (2) fuel materials and process modifications that influence fission product release performance, and (3) HEU fuel cycle strategies. Conclusions on these studies are planned for the next reporting period.

2.3. COMPONENT DV&S (14)

2.3.1. Reactor Internals (1417)

The improved Stackpole 2020 core support post graphite was delivered for evaluation. This material will be used for fabrication of posts for the Core Support Performance Test (CSPT) at ORNL.

2.3.2. Reactor Core (1418)

A test specification was written and tests begun for structural load tests to determine the shear strength of the fuel element flange in case of a seismic load. A total of 33 specimens were tested, involving six different flange configurations.

The design of a test apparatus was completed for tests of design corrections to prevent core fluctuations. The 1/4-scale plastic fuel elements were fabricated, and the apparatus was partly assembled.

The test specification was released for a two-block crossflow test. Its purpose is to measure the effectiveness of the fuel element sealing flange in limiting crossflow from the coolant channels to the gaps between fuel columns. Preliminary testing was performed on a small scoping model to test the flow distribution in the plenum area of the flange.

For the development of models to support fuel element structural criteria, an evaluation was initiated on the merits of probabilistic versus deterministic models for predicting structural stress conditions. Post strength data have been evaluated to establish Weibull parameters for use in calculating failure probabilities.

2.4. TECHNOLOGY TRANSFER (15)

2.4.1. International Technology Transfer (1501)

The semiannual Subprogram Management Meeting on the U.S./FRG Umbrella Exchange Subprogram on Fuel, Fission Products, and Graphite was held at GA on February 15 and 16, 1983. Representatives were present from DOE-Washington, KFA, ORNL, General Electric, and GCRA. Revision 6 of the Subprogram Plan was prepared at this meeting. The exchange is proceeding satisfactorily, with a good balance of contributions from each side.

The exchange of personnel between GA and KFA under the Subprogram on Fuel, Fission Products, and Graphite was completed during this period. Dr. Karl Bongartz of KFA completed a one-year assignment at GA on fuel performance modeling in November 1982. Mr. David Hanson of GA completed a 15-month assignment at KFA on fission product methods verification in March 1983.

2.4.2. Fort St. Vrain Core Surveillance (1502)

A destructive post-irradiation examination was begun on one of the FSV fuel elements that was observed to have a cracked web during the visual examination reported last period. All of the fuel rods were pushed out of the graphite block with forces of less than five pounds per fuel stack. All of the fuel rods appeared unchanged in surface condition from when the element was originally assembled. The cracked webs do not appear to have affected the performance of the fuel element in any way.

3. FUEL

3.1. FUEL PROCESS DEVELOPMENT (1101)

3.1.1. Scope

Fuel process development activities during this reporting period included (1) establishment and demonstration of UCO fissile kernel process parameters using engineering-scale equipment for broth preparation/aging and sphere forming, washing, drying, and heat treatments; (2) evaluation of candidate materials for the 1800°C sintering/conversion furnace retort; (3) performance of a baseline-quality series of buffer/ILTI coating runs using the 240-mm development coater; (4) QC evaluation of HOBEG-deposited TRISO coatings as part of the U.S./FRG Fuel, Fission Products, and Graphite Subprogram; and (5) performance defective particle separation tests using gas elutriation, screening, and tabling processes.

3.1.2. Kernel Development

Scaleup process development for fabricating UCO reference fissile kernels continued, along with supporting laboratory studies.

The engineering-scale equipment for sphere forming, washing, and drying and kernel heat-treatment was described in the previous semiannual report, for the period ending September 30, 1982.

The process sequence for fabricating UCO kernels is shown in Fig. 3.1-1, including materials as introduced and key process parameters. During this reporting period, process parameters were established and demonstrated using the engineering-scale equipment for broth preparation and aging; sphere forming, washing, and drying; and dried kernel heat-treatment.

Fig. 3.1-1. UCO kernel process description

The broth composition used in experiments conducted during this reporting period is the same as specified previously, with the exception of an increase in the polyvinyl alcohol component from 21 g/liter to 22.6 g/liter. Methods for consistent preparation of each broth component were established. Best broth aging conditions were determined to be four days at a constant temperature and pressure of 45°C and 1 atm, respectively. Hot aging of the kernels in concentrated NH_4OH solution was found to be unnecessary.

Evaluation of wet kernel drying process parameters is under way. Presently, wet kernels are dried in a 13-cm diameter fluid bed with an initial bed height-to-diameter (L/D) ratio of 2.2. A cone-type gas distributor is used with nitrogen gas preheated between 30° and 60°C and flowing at about 112 liters/min. An average drying time of about 2.5 h reduces the kernel original volume by about 80% and produces dried kernel densities of 2 to 3 Mg/m^3 . Development of the engineering-scale wet kernel drying equipment and process is continuing to produce reliably dried kernels of low breakage and desirable characteristics to facilitate downstream heat-treatment steps.

Low-temperature heat treatment of the kernels in air has been found necessary to provide better control of final kernel composition. Preliminary experiments on air treatment time, temperature, and heat-up rate have not been conclusive as to the effect of these process parameters. An air heat treatment cycle of 16 h at 120°C and 8 h at 150°C, though not optimized, was found to be successful in producing sintered kernels that met all specifications of density, stoichiometry, and high yield. Figures 3.1-2 and 3.1-3 show two good UCO-sintered kernel batches subjected to the air heat treatment cycle described above. In the case in Fig. 3.1-2, the sintered density achieved was 11.0 Mg/m^3 , with a composition of $\text{UC}_{0.19}\text{O}_{1.87}$. In Fig. 3.1-3, the sintered density achieved was 10.84 Mg/m^3 , with a composition of $\text{UC}_{0.22}\text{O}_{1.70}$. Shorter air treatment cycles are currently under evaluation.

(X 100)

(X 200)

Fig. 3.1-2. Sintered UCO fissile kernel product

(X 50)

(X 100)

Fig. 3.1-3. Sintered UCO fissile kernel product

An evaluation of kernel calcining process parameters revealed that sweep-gas flow rates and heat-up rates between 550° and 700°C have little effect on controlling final kernel stoichiometry.

The rotary sintering furnace has been modified to permit improved control and monitoring of furnace retort gas flow and furnace CO and CO₂ partial pressures. The present sintering procedure consists of initially reducing UO₂ to UC₂ under an argon sweep gas flow of about 6 liters/min during heating from 600° to about 1000°C. Above 1000°C, the argon flow rate is increased to keep the CO partial pressure in the furnace below the equilibrium P_{CO} of the reaction $UO_2 + 3C \rightarrow UC + 2CO$, thus favoring the formation of the UC phase. Once the UC phase is formed, a minimum hold time of 2 h at 1600°C is necessary to sinter the kernels to a density of about 11.0 Mg/m³. Subsequently, 100% CO gas is introduced to the furnace retort and the temperature ramped up to 1800°C and held for 0.5 h to convert the UC phase to the UC₂ phase. A cooldown period under CO cover gas to 1450°C appears to be essential for the retention of the UC₂ phase. The phase change of UC₂ to UC appears to be highly reversible, rapid, and totally dependent on the presence of CO atmosphere between the temperatures of 1600° to 1800°C.

A major achievement has been the reduction of the sintering cycle by 7.5 h and consistent production of sintered kernels meeting specifications of composition and density. Figure 3.1-4 shows a batch of kernels sintered to density and of the phase composition UO₂ + UC, with a sintered density achieved of 10.80 Mg/m³ and a composition of UC_{0.16}O_{1.65}. Figure 3.1-3 shows the same batch later heat treated under CO gas to achieve the composition UO₂ + UC₂ and with drastically reduced internal voids.

Current activities seek to optimize the heat-treating process steps and to increase production yield.

Table 3.1-1 summarizes the product properties relative to certain process parameters for several typical study runs. All batch sizes, except batch 8340-146 are approximately 1 kg of nitrogen-dried kernels that were air-heat-treated, calcined, and sintered/converted in rotary furnaces at

(X 50)

(X 100)

Fig. 3.1-4. Sintered UCO fissile kernel product

TABLE 3.1-1
KERNEL PRODUCT DATA SUMMARY

Batch Number	8340-146(a)	8619-73	8619-61	8619-49	8619-75	8619-81	8619-29
Broth aging (days)	4	4	4	4	4	4	4
Warm N ₂ temperature (°C)	50	40	40	--	--	60	--
Dried density (Mg/m ³)	2.40	2.61	2.61	2.31	2.31	3.43	1.9
Dried composition							
U wt%	70.05	63.88	63.88	68.47	68.47	66.44	66.01
C	5.65	6.30	6.30	5.62	5.62	5.30	6.07
O	16.20	19.64	19.64	19.16	19.16	21.77	20.51
Air low heat temperature (°C)	120-150	120-150	120-150	120	120	120	150
Air low heat time (h)	16+8	16+8	16+8	16	16	16	4
Air low heat density (Mg/m ³)	2.69	3.28	3.28	2.88	2.87	3.68	2.56
Air low heat composition							
U wt%	77.3	69.10	69.10	67.66	67.90	69.28	70.24
C	4.1	4.40	4.40	5.27	5.29	4.94	4.90
O	18.6	18.94	18.94	19.12	18.87	17.58	15.53
Calcined density (Mg/m ³)	4.73	5.55	5.55	4.31	4.84	6.00	4.06
Calcined composition							
U wt%	85.86	80.87	80.87	79.10	83.77	81.20	83.30
C	1.57	1.76	1.76	4.29	4.64	4.52	3.14
O	12.57	12.32	12.32	11.44	11.84	13.31	14.19
Argon sintering flow rate (liters/min)	0.3	16.0	16.0	2.9	16.0	16.0	0.3
Argon sintering time (h)	9.5	4.0	4.0	9.5	4.0	2.0	9.5
Conversion atmosphere	CO	CO	Ar	CO	CO	CO	CO
Sintered density (Mg/m ³)	11.0	10.84	10.80	9.71	10.89	11.06	10.20
Sintered composition							
U wt%	88.10	88.70	89.30	87.91	89.04	89.20	88.31
C	0.85	0.97	0.70	2.42	1.92	2.34	1.48
O	11.05	10.13	9.90	9.67	9.04	8.36	10.21
Sintered dominant carbide phase	UC ₂	UC ₂	UC	UC ₂	UC ₂	UC ₂	UC ₂
Sintered UC _x O _y composition	UC _{0.19} O _{1.87}	UC _{0.22} O _{1.70}	UC _{0.15} O _{1.65}	UC _{0.55} O _{1.64}	UC _{0.43} O _{1.51}	UC _{0.52} O _{1.39}	UC _{0.12} O _{1.72}

(a) Batch size = 300 g of N₂-dried kernels.

12 rpm. Alternate material candidates for the UCO kernel conversion/sintering furnace liner for operating conditions of 1800°C and CO gas atmosphere were evaluated. Al_2O_3 , MgO , ZrO_2 , ThO_2 , MoC , POCO graphite, SiC-coated graphite, and H451 graphite materials were tested under the required furnace conditions. All materials except graphite, MoC , and ThO_2 exhibit some chemical reaction with the UCO kernels and/or the CO furnace gas. Graphite was found to be the best-suited material for the purpose. Figure 3.1-5 shows Al_2O_3 , MgO , ZrO_2 , and ThO_2 crucibles after the test.

3.1.3. TRISO Coating Process Development

The objective of TRISO coating process development is to improve fuel particle coating processes to meet the preliminary LEU specifications for LHTGR fuel.

Work during FY 83 involved systematically evaluating current TRISO coating quality capabilities using the 240-mm development coater with the standard 3-gas-path FSV production nozzle. This baseline quality evaluation will serve to define specific areas where further pilot scale development is required.

3.1.3.1. Buffer/ILTI Coating Thickness

A series of ten baseline buffer/ILTI coating runs were performed using 10-kg batches of 500- μm diameter ThO_2 fertile kernels. All ten runs were performed using the same coating process conditions. Prior to the ten baseline runs, two practice runs were performed to establish the process parameters required to meet the HTGR-SC/C fertile particle specifications. Process parameters are listed on Table 3.1-2. Thickness and density QA test data are summarized on Table 3.1-3.

Thickness variations in both the buffer and ILTI exceed the specified limits. Figures 3.1-6 and 3.1-7 show the variations of mean and critical

Fig. 3.1-5. UCO sintering furnace retort materials test
 Al_2O_3 , MgO , ZrO_2 , ThO_2

TABLE 3.1-2
BUFFER/ILTI COATED ThO₂ (500 m) PARTICLE PROCESS CONDITIONS

ThO ₂ kernel substrate	
Kernel batch size	10,000 g
Kernel density	9.68 g/cm ³
Kernel diameter ($\bar{x}$)	505 μ m
Buffer coating conditions	
Coating temperature	1550°C
Coating time	5.9 min
Argon levitation	350 liters/min
Argon auxiliary	70 liters/min
Argon diluent	40 liters/min
Acetylene	414 liters/min
ILTI coating conditions	
Coating temperature	1290°C
Coating time	17.5 min
Argon levitation	300 liters/min
Hydrogen auxiliary	250 liters/min
Hydrogen diluent	250 liters/min
Acetylene	128 liters/min
Propylene	108 liters/min

TABLE 3.1-3
QC DATA FOR BASELINE BUFFER/ILTI COATED ThO₂ PARTICLE RUNS

Run No. 8709	$\bar{x}(a)$ (μm)	Buffer Thickness		Critical (μm)	Buffer Density (g/cm ³)	$\bar{x}(a)$	ILTI Thickness		Critical (μm)	ILTI Density (g/cm ³)	Buffer/ILTI Wt Gain (g)		Coating Eff.	
		$\sigma_{WB}(b)$ (μm)	$\sigma_{WB}(b)$ (%)				$\sigma_{WB}(b)$ (μm)	$\sigma_{WB}(b)$ (%)			Calculated (QC data)	Measured (wt out)	Buffer (%)	ILTI (%)
8709-18	75.09	15.00	19.98	33.0	0.813	32.19	5.49	17.05	13.92	1.92	2408	2680	38.0	26.2
8709-20	78.82	14.36	18.22	38.2	1.020	33.44	4.07	12.17	18.52	1.89	2800	2670	50.7	27.1
8709-22	74.85	14.03	18.74	35.0	1.018	32.55	3.68	11.31	18.56	1.88	2620	2630	47.4	25.4
8709-24	77.09	14.63	18.98	36.0	1.050	32.87	3.36	10.21	19.63	1.88	2753	2680	50.7	26.3
8709-26	83.22	13.77	16.54	44.2	1.001	32.78	3.70	11.28	18.73	1.88	2869	2740	53.3	27.1
8709-28	82.16	14.18	17.26	42.0	1.006	30.57	3.74	12.23	16.43	1.93	2769	2620	52.7	25.5
8709-30	78.99	14.10	17.84	39.0	1.069	32.86	4.00	12.17	18.11	1.90	2846	2700	53.3	26.7
8709-32	79.91	14.43	18.06	39.2	1.049	36.50	4.44	12.16	20.71	1.91	3040	2720	52.1	30.4
8709-34	84.60	14.12	16.69	44.5	1.043	37.90	5.08	13.40	20.60	1.88	3226	2720	56.6	32.0
8709-36	80.05	15.17	18.96	38.0	1.038	38.21	5.17	13.50	20.70	1.90	3106	2740	52.6	31.8
Mean ($\bar{x}$)(a)	79.48	14.38	18.12	38.9	1.011	33.99	4.27	12.55	18.59	1.896	2844	2696	50.84	27.85
Std. dev. (σ_{BB})(c)	3.25	0.44	1.058	3.78	0.073	2.59	0.736	1.85	2.13	0.018	238	42	5.09	2.55
$\sigma_{BB}(c)(\%)$	4.09	3.06	6.00	9.72	7.22	7.62	17.2	14.7	11.46	0.95	8.37	1.56	10.01	9.15

(a) $\bar{x}$ = thickness mean.

(b) σ_{WB} = standard deviation within the batch.

(c) σ_{BB} = standard deviation batch to batch.

Fig. 3.1-6. Plot of thickness data for buffer coating layer

Fig. 3.1-7. Plot of thickness data for ILTI coating layers

thickness values for the ten runs, with respect to specifications. The critical thickness limit is calculated by:

$$\text{Thickness crit.} = \bar{X}_{\text{mean}} - [k \sigma_{\text{WB}} + 2.05 \sigma_{\text{BB}}] ,$$

where $k = 2.36$ for 200 particles measured per batch.

It is believed that fluidization of the 10-kg batch of 500- μm ThO_2 particles is not optimum with the standard gas distributor nozzle. Coating time is apparently too short relative to bed turnover rate. Also, there appears to be some inconsistency in QC thickness measurements relative to observed carbon weight gain, particularly for the last three runs, 8709-32, 8709-34, and 8709-36. Further development efforts will be aimed toward improvements in fluidization to reduce thickness variation and in QC measurement techniques to improve accuracy and consistency.

3.1.3.2. Buffer/ILTI Coating Defect Evaluation

Approximately 65,000 particles from each of the ten coating batches were scanned for coating defects. The results of these evaluations are listed on Table 3.1-4. Observed defect levels are within specification limits.

3.1.3.3. ILTI Coating Microstructure

The ILTI coating microstructure appears to be very dense with little carbon soot inclusion. Figure 3.1-8 presents photomicrographs of typical ILTI structures observed.

Permeability tests of the ILTI coating were also performed. Impregnation with liquid bromoform followed by x-ray examination was used to identify porosity defects. Results of these measurements showed about 1×10^{-4} fraction of permeable ILTI coatings.

TABLE 3.1-4
DEFECTIVE PARTICLE DATA FOR BUFFER/ILTI COATED ThO₂ PARTICLES

Run No. 8709	No. of Particles Examined	Uncoated ThO ₂ Kernels	Buffer Defects			ILTI Defects			Total(a) Number of Defects	Fraction Buffer/ ILTI Defects		
			Missing	Broken	Others	Missing	Broken	Others		Mean (x 10 ⁴)	50% Conf. (x 10 ⁴)	95% Conf. (x 10 ⁴)
8709-18	54,775	4	9	0	1(b)	0	0	0	14	2.56	2.70	3.83
8709-20	60,786	7	0	0	2(b)	0	0	5(c)	14	2.30	2.43	3.45
8709-22	63,240	10	0	0	0	0	0	2(d)	12	1.89	2.53	3.00
8709-24	66,908	1	0	0	1(b)	0	0	2(d)	4	0.60	0.70	1.34
8709-26	67,800	22	0	0	1(b)	0	0	0	23	3.39	3.54	4.57
8709-28	67,800	4	2	1	1	0	1	1(d)	10	1.47	1.59	2.21
8709-30	66,552	3	0	0	0	0	0	0	3	0.45	0.60	1.17
8709-32	67,550	10	3	1	1(b)	0	1	1(c)	17	2.52	2.67	3.67
8709-34	64,505	1	1	0	1(b)	0	1	2(c)	6	0.93	1.09	1.84
8709-36	67,835	1	2	0	1(b)	0	1	5(c)	10	1.47	1.62	2.49
Mean ($\bar{x}$)	64,775	6.3	3.2	0.2	0.9	0	0.4	1.8	11.3	1.76	1.95	2.76
Std. Dev.	--	--	--	--	--	--	--	--	6.1	0.95	0.97	1.13
σ_{BB} (%)	--	--	--	--	--	--	--	--	54.2	53.9	49.5	41.0

(a) Coated particles with surface heavy metal contamination not included.

(b) Kernel migration.

(c) Double coated.

(d) Soot inclusion.

Fig. 3.1-8. Buffer/ILTI coated ThO₂ (500 μ m) particle detailed ILTI coating microstructures

3.1.4. Defective Particle Evaluations

3.1.4.1. U.S./FRG Cooperative Fuel Development Program

Samples of TRISO-coated, 500- μ m diameter ThO₂ that had been coated by HOBEG were received from FRG during this period. The ThO₂ substrate material included batches prepared and provided by GA and batches made at HOBEG. GA performed a QC evaluation of these TRISO coatings.

Combined buffer/ILTI coating thicknesses measured by GA agreed well with those measured by HOBEG.

The mean SiC coating thickness value measured by GA was approximately 17% higher than the values reported by HOBEG for the TRISO coatings deposited on ThO particles made by HOBEG. For GA ThO₂ kernels coated by HOBEG, the GA-determined mean SiC coating thicknesses were in good agreement with the HOBEG values. The SiC coating thickness standard deviations within the batches, as measured by GA, were also significantly larger on the particles made with HOBEG kernels than the values reported by HOBEG. The determination of SiC thickness standard deviation within batches coated on GA kernels showed better agreement between GA and HOBEG.

The OLTi coating mean thickness and standard deviation values measured by GA and HOBEG were in good agreement.

SiC-coated ThO₂ particle defects, as measured by the burn-leach technique, on HOBEG-coated ThO₂ particles made with HOBEG ThO₂ kernels were a factor of 10 to 20 lower than those made by HOBEG on GA-prepared ThO₂ kernels. These data confirm earlier findings by HOBEG that TRISO coatings on GA ThO₂ kernels with low density exhibited higher SiC defects than coatings on higher density ThO₂ kernels made by HOBEG.

Analyses of defective coated particles (buffer, ILTI, and OLTi), coating density, and BAF_o are in progress.

3.1.4.2. Defect Reduction

The gas elutriation system was used to remove defective kernels from the composite of 500- μ m diameter ThO₂ kernels to be used in the baseline coating process evaluations. Approximately 120 kg of kernels were elutriated with nitrogen gas. Fractured kernels and debris were reduced from ~1% to less than 0.1%, with losses less than 2%, resulting in a 98%-yield of usable kernel product.

3.2. FUEL MATERIALS DEVELOPMENT (1102)

3.2.1. Scope

Fuel materials development tasks included fuel performance studies and model development, product specification development, fuel improvement studies, and accelerated and real-time irradiation tests and evaluation.

3.2.2. Fuel Performance

3.2.2.1. Accident Condition Tests

In March 1981, three samples of ~100 high-temperature isotropic (HTI) BISO particles designated for use in core heatup tests were received from KFA as part of the U.S. DOE/FRG umbrella agreement on fuel development. A single batch of the BISO fuels (EO-1084 from R2-K10) was tested at GA under accident conditions (Ref. 3.2-1). An increasing interest in TRISO fuels in the FRG and a continuing interest in the U.S. led to the shipment of two samples of TRISO-coated fuels from the FRG to the U.S. in December 1982 to be tested. Tables 3.2-1, 3.2-2, and 3.3-3 list kernel, coating, and irradiation parameters for these samples. The two TRISO samples were from spheres 19 and 22 of the DR-S6 experiment and have nearly identical exposure conditions. Only the fuel from sphere 19 has been used in high-temperature testing at this time. This fuel batch sample has been designated as "EO 403t-450t, homogenized."

Specific time/temperature test conditions (GA test number 83IHR-B-1-1) are shown in Fig. 3.2-1. The solid line indicates the intended heating ramp, the points indicate actual measured temperatures, and the dotted line indicates the linear interpolation used to report release fractions at intermediate temperatures. Upon reaching ~1750°C, the temperature controller feedback thermocouple of the furnace system failed and was replaced, resulting in an unintended downward transient in temperature to ~950°C. No fission gas release was observed at the time of this transient. At ~2200°C, a temperature feedback failure again resulted in an unattended decline to

TABLE 3.2-1
 KERNEL CHARACTERISTICS
 (COATING BATCH EO 403t-450t HOMOGENIZED)

Kernel composition	UO ₂
U-235 enrichment, %	6.65
Nominal diameter, μm	615
Density, g/cm ³	10.5-10.7
Manufacturer	HOBEG

TABLE 3.2-2
 COATING SPECIFICATIONS
 (COATING BATCH EO 403t-450t HOMOGENIZED)

<u>Layer Number</u>	<u>Type</u>	<u>Deposition Temperature (°C)</u>	<u>Density (g/cm³)</u>	<u>Nominal Thickness (μm)</u>
1	Porous PyC	1450	<1.1	132
2	HDI PyC	1340	1.95 ± 0.05	46
3	SiC	1490	3.21	29
4	HDI PyC	1340	1.95 ± 0.05	38

TABLE 3.2-3
 DR-S6 FUEL IRRADIATION PARAMETERS

Burnup	
% FIMA (design)	4.2-5.5 (max.)
MWd/tonne	11,653
Fast dose	
n/cm ² DNE	2.3 x 10 ²¹
n/m ² (E > 29 fJ) _{HTGR}	3.4 x 10 ²⁵
Design maximum temperature (°C)	
Surface of sphere	1000 ± 50
Fuel	1200 ± 50

Fig. 3.2-1. Time/temperature profile for accident simulation testing of HOBEG fuel

~1800°C (see Fig. 3.2-1), and no fission gas release was indicated as an immediate result of this transient. The lack of detecting particle failure during the unplanned thermal transients implies that the transients had a negligible impact on the test results. At ~2425°C, while large amounts of fission gas were being released from the fuel, one of the three sample-containing crucibles appeared to be displaced from the heating zone by ~0.5 meters. The furnace temperature was decreased to ~1100°C while the sample was repositioned into the heating zone. The temperatures were again increased to intercept the intended ramp, the cooling and heating process taking one hour. Nominal test procedures are given in Ref. 3.2-2, and specific test condition parameters are listed in Table 3.2-4.

Fuel performance during testing was monitored from fission product release measurements made as each test was conducted. Release of Kr-85 and Cs-137 was monitored periodically by replacement of fission product traps. Release data for the nuclides Eu-154, Ru-106, and Ce-144 were determined only at the end of each test by post-test gamma counting of the samples. Tables 3.2-5 and 3.2-6 list the measured releases of fission gas (Kr-85) and fission metals as a function of temperature during testing. The fission gas and volatile metal releases are shown graphically in Fig. 3.2-2.

The gaseous and metallic fission product release from this TRISO fuel during accident simulation testing is in excellent agreement with the releases observed during testing of TRISO fuels fabricated at GA (Refs. 3.2-1 and 3.2-2). This provides additional support for the generic nature of the TRISO coating failure model for accident conditions and adds to the available data base for TRISO fuel performance at high temperatures. The data also support the conclusion that OPyC remains retentive for gas after failure of SiC.

As noted previously at a temperature of ~2450°C, one sample-containing crucible appeared to be displaced ~0.5 meters (in a horizontal direction) from the heating zone of the furnace. Later inspection showed that the crucible lid had been forced away, apparently due to a sudden release of gas

TABLE 3.2-4
CORE HEATUP SIMULATION TEST CONDITIONS FOR KFA
FUEL BATCH NUMBER EO 403t-450t (HOMOGENIZED)

GA test number	83IHR-B-1-1
HOBEG test sample number	EO 403t-450t (homogenized)
Number of particles tested	100
Range of temperatures (°C) (accident ramp)	1115-2605
Time to reach temperature (h)	31.08
Nominal heating rate (°C/h)	47.9

TABLE 3.2-5
Kr-85 RELEASE DATA FROM ACCIDENT SIMULATION
TESTING OF HOBEG FUEL BATCH NUMBER
EO 403t-450t (HOMOGENIZED)

Temperature (°C)	Release Fraction
<2049	<10 ⁻⁵ (a)
2049	1.2 x 10 ⁻⁴
2195	3.7 10 ⁻²
2263	1.4 x 10 ⁻¹
2312	2.4 x 10 ⁻¹
2357	5.6 x 10 ⁻¹
2452	9.9 x 10 ⁻¹
2606	1.0

(a) Estimated detection
limit.

TABLE 3.2-6
METALLIC FISSION PRODUCT RELEASE DATA FROM CHST OF KFA FUEL

Sample Location Number 1 EO 403t-450t					Sample Location Number 2 EO 403t-450t					Sample Location Number 3 EO 403t-450t				
Temperature (°C)	Release Fraction				Temperature (°C)	Release Fraction				Temperature (°C)	Release Fraction			
	Cs-137	Co-144	Ru-106	Eu-154		Cs-137	Co-144	Ru-106	Eu-154		Cs-137	Co-144	Ru-106	Eu-154
1101	3.1-7	N.D.(a)	N.D.	N.D.	1100	2.1-7	N.D.	N.D.	N.D.	1094	2.7-7	N.D.	N.D.	N.D.
1117	2.9-6	N.D.	N.D.	N.D.	1115	2.6-6	N.D.	N.D.	N.D.	1114	2.9-6	N.D.	N.D.	N.D.
1497	4.1-6	N.D.	N.D.	N.D.	1492	4.6-6	N.D.	N.D.	N.D.	1496	5.5-6	N.D.	N.D.	N.D.
1688	4.6-6	N.D.	N.D.	N.D.	1683	5.1-6	N.D.	N.D.	N.D.	1685	6.0-6	N.D.	N.D.	N.D.
1825	4.7-6	N.D.	N.D.	N.D.	1814	5.3-6	N.D.	N.D.	N.D.	1821	6.1-6	N.D.	N.D.	N.D.
1844	3.2-4	N.D.	N.D.	N.D.	1845	5.4-6	N.D.	N.D.	N.D.	1845	1.7-3	N.D.	N.D.	N.D.
1955	1.0-2	N.D.	N.D.	N.D.	1949	6.3-6	N.D.	N.D.	N.D.	1947	1.3-2	N.D.	N.D.	N.D.
2050	3.6-1	N.D.	N.D.	N.D.	2044	5.6-2	N.D.	N.D.	N.D.	2053	8.8-2	N.D.	N.D.	N.D.
2198	7.1-2	N.D.	N.D.	N.D.	2198	3.5-1	N.D.	N.D.	N.D.	2189	3.9-1	N.D.	N.D.	N.D.
2265	6.9-1	N.D.	N.D.	N.D.	2262	7.0-1	N.D.	N.D.	N.D.	2261	6.9-1	N.D.	N.D.	N.D.
2306	8.0-1	N.D.	N.D.	N.D.	2314	8.3-1	N.D.	N.D.	N.D.	2316	8.1-1	N.D.	N.D.	N.D.
2357	8.8-1	N.D.	N.D.	N.D.	2355	9.1-1	N.D.	N.D.	N.D.	2360	9.9-1	N.D.	N.D.	N.D.
2452	9.98-1	N.D.	N.D.	N.D.	2447 ^(b)	9.9-1	N.D.	N.D.	N.D.	2356	9.99-1	N.D.	N.D.	N.D.
2601	1.0	0.04 ^(c)	0.09 ^(c)	1.0 ^(c)	2608	1.0	N.D.	N.D.	N.D.	2610	1.0	0.21 ^(c)	0.39 ^(c)	1.0 ^(c)

(a) N.D. = No Data.

(b) Crucible lid came off at ~2425°C.

(c) Release from both particles and crucibles.

Fig. 3.2-2. Kr-85 and Cs-137 release from HOBEG TRISO LEU UO_2 fuel
(EO 403t-450t)

inside the crucible. A survey of past CHST results (Table 3.2-7) showed that three other tests had crucibles exhibiting the same behavior, and all had included oxide-containing fuels at a temperature of ~2400-2450°C (approximately 10% of the total oxide-containing crucibles tested in this program). It may be speculated that a sudden release of the CO pressure from an oxide particle (i.e., sudden failure) at high temperature can result in violent particle movement, which can damage adjacent particles and cause them to fail (Refs. 3.2-3, 3.2-4). This is in contrast to the case where initially failed fuels slowly convert to oxide as the heatup proceeds with no adverse consequences (Ref. 3.2-1). There is also the possibility that failure of one particle at high temperatures may have exposed kernel material that reacted with adjacent coatings by carbothermic reduction, leading to additional failures (Refs. 3.2-5, 3.2-6, 3.2-7). This would result in an almost instantaneous release of significant amounts of both fission gases and generated CO that could increase the crucible internal pressure to the point where the friction-fit lid would be forcibly ejected. Such mechanisms are only hypothetical at this point, but it is otherwise difficult to rationalize the instantaneous failure of a sufficient number of particles to eject the crucible lid. The impact of this phenomenon on core performance during an accident is not known.

3.2.2.2. Fuel Performance Models

TRISO Pressure Vessel Failure Model. (PWS-FD-8) A series of benchmark failure prediction comparisons was run using the KFA CONVUL and the unified GA/KFA TRISO*SiC stress codes. This work was conducted under PWS-FD8, and the comparisons were initiated by H. Nabielek (HBK/KFA). The results based on KFA irradiation test data showed that code predictions were essentially equivalent for identical input parameters.

H. Nabielek reviewed the documentation (GA-A16807) on the unified KFA/GA stress code (TRISO*SiC). One area identified for continued work under PWS-FD8 is consolidation of KFA and GA data on the release of fission

TABLE 3.2-7
CORE HEATUP TESTS EXHIBITING SUDDEN AND LARGE GAS
PRESSURE RELEASES DURING ACCIDENT SIMULATION TESTS

<u>Test Number</u>	<u>Fuel Type</u>	<u>Temperature (°C)</u>	<u>Time to Reach Temperature (h)</u>
78IHR-A-2-1	(Th,U)O ₂	2440	7.6
80IHR-B-6-1	UC _{0.4} O _{1.6}	2395	27.0
80IHR-B-7-1	UO ₂	2435	28.5
83IHR-B-1-1	UO ₂ (HOBEG)	2425	28.0

gas as a function of kernel type, irradiation temperature, and time. Eventually, this updated information will be incorporated into a later edition of the unified KFA/GA stress model.

Metallic Fission Product Release. Data on the release of metallic fission products from initially intact TRISO and BISO fuels subjected to high-temperature accident-simulation ramp tests were tabulated. This tabulation of the release data will be used to define the parameters for a release model. The performance model will use a diffusion model coupled with an effective thinning of the barrier layer due to fission product - SiC interactions and thermal dissociation of the SiC.

3.2.3. Fuel Irradiation and Specification

3.2.3.1. Data Storage and Retrieval System

Work continued on entering data into the computerized irradiation data storage and retrieval system. Progress was made in storing data for particles, fuel rods, and fuel elements.

Coated Particles. The total number of sample numbers in the coated particle storage program is presently 2789. ORNL data now comprise 175 sample numbers from capsules HRB-14, HT-34, and HRB-15B. Hard-copy worksheets of the computer input data from capsules HT-34 and HRB-15B were shipped to ORNL in December 1982 for review prior to final storage.

Fuel Rods. Work on the fuel rod storage program consisted of correcting previously stored data and putting new information into storage for the 11 capsules shown in Table 3.2-8.

Elements. Since September 1982, preirradiation fuel characterization data from one additional capsule have been entered into the element storage program. That capsule is R2-K13.

TABLE 3.2-8
CAPSULE WITH CORRECTED/ADDITIONAL
FUEL ROD IRRADIATION DATA PLACED INTO
THE COMPUTERIZED DATA STORAGE SYSTEM

<u>Capsule</u>	<u>Pre</u>	<u>Post</u>
P13Q	X	X
P13R	X	X
P13S	X	X
GF-1	X	X
GF-2	X	X
GF-3	X	X
GF-4	X	X
GF-5	X	
R2-K13	X	
HRB-14	X	X
HRB-16	X	X

Data Analysis with the System. A presentation of the analytical tasks possible with the computerized irradiation data storage and retrieval program was made at the U.S./FRG Umbrella Program Coordination Meeting at GA on February 15. The effective use of stored data to create computer-drawn graphs in parametric property versus performance relationships was demonstrated.

3.2.3.2. Capsule R2-K13 (PWS-FD-15)

Iodine Release In Pile. Capsule R2-K13 completed irradiation in the R2 Reactor, Studsvik, Sweden, in September 1982. The iodine release from GA LEU UCO/ThO₂, TRISO fuel (cells 2 and 3) was measured immediately following the reactor shutdown at the irradiation end-of-life (EOL). This was done by monitoring the short-lived xenon daughters of the iodine nuclides present in the sweep gas. The results are shown in Table 3.2-9.

The results showed essentially no change from the iodine release test run in August 1982. The results showed approximately a factor of five difference between cell 2 (1200°C) and cell 3 (1000°C), which is nearly identical to the difference between the EOL Kr-85m R/B for cell 2 and cell 3 when the temperature between the two cells is taken into account. These data support previous work at GA and the assumption that iodine is released in the same manner as xenon and krypton: as an elemental gas and not as a compound.

Fission Gas Release In Pile. Final in-pile fission gas release measurements at the end of irradiation were received from Studsvik. The complete history of Kr-85m fission gas release for R2-K13, cells 2 and 3, is shown in Fig. 3.2-3.

Capsule Disassembly. Following removal from the R-2 reactor, the entire test rig was transported to the Studsvik hot cell facility. The capsule was then opened by sawing the containment tube ends off, which allowed retrieval of the four, sealed primary cells containing the fuel samples. Cells 2 and 3, containing the GA fuel and piggyback specimens, were further

TABLE 3.2-9
R2-K13 IODINE RELEASE AT EOL (R/B)

<u>Cell 2</u>		<u>Cell 3</u>		<u>Ratio of Cell 2 ÷ Cell 3</u>	
<u>Xe-135 (I-135)</u>	<u>Xe-133 (I-133)</u>	<u>Xe-135 (I-135)</u>	<u>Xe-133 (I-133)</u>	<u>Xe-135 (I-135)</u>	<u>Xe-133 (I-133)</u>
3.7 x 10 ⁻⁵	6.4 x 10 ⁻⁵	9.1 x 10 ⁻⁶	1.0 x 10 ⁻⁵	4.1	6.4
				Average = 5.2

Fig. 3.2-3. In-pile and predicted fission gas release of Kr-85m from R2-K13, cells 2 and 3

dismantled by cutting the ends not containing thermocouple penetrations away and pulling the fueled graphite bodies out of the primary containment cans. The fuel bodies were shipped to KFA for destructive examination, but the metallic components of the capsule were retained at Studsvik and examined.

Neither capsule showed any sign of corrosion or other damage. There were a few grams of black powder (carbon or molybdenum oxide) that came out of the cell 2 containment. It was felt that they came from the containment graphite cover plate or from the molybdenum thermal shields since the plate and molybdenum disc were corroded.

Capsule Containment Leach Results. After fuel body removal, the stainless steel containments used in the R2-K13 irradiation capsule were leached.

Three different leach solutions were used: 1 molar NaOH (cesium, barium, strontium, yttrium, and cerium tracers); 1 molar HNO₃ (cesium, barium, strontium, cerium, and silver tracers); and 1 molar HCl-0.3 M HF (cesium, barium, and strontium tracers). Leaching times were 24 h, 6 h, and 5 h, respectively, and the temperatures were 50°, 50°, and 25°C. Portions of the leach solutions were taken from the hot cell for radiochemical analysis. The methods are indicated in Table 3.2-10. A GeLi detector was used for gamma measurements, and an end-window scintillation counter was used for the beta measurement of Sr-90. The results are shown in Table 3.2-11. It is believed that the general increase of activity in cell 2 over that of cell 3 is due to the increased fission product release associated with the higher irradiation temperature of cell 2. One notable and unexplained exception is Sr-90, which showed three times more release in cell 3. Note also that the Ag-110m release is identical in both cells.

Piggyback and Lumped Burnable Poison Samples. The arrangement of piggyback samples, lumped burnable poison, and fuel rods in the R2-K13 fuel bodies is shown in Fig. 3.2-4. Piggyback and lumped burnable poison samples from cells 2 and 3 were removed and shipped to GA in December 1982. The

TABLE 3.2-10
STUDSVIK ANALYTICAL METHODS USED FOR LEACHING OF
CELL 2 AND CELL 3 R2-K13 STAINLESS STEEL CONTAINMENTS

<u>Nuclide</u>	<u>Separation Method</u>
I-131	Extraction with CCl_4
Te-132	Precipitation of Te with SO_2
Cs-134, 137	Contaminants adsorbed on ion exchanger in Cl^- form (bulk of Mo) and in S^{2-} form
Sr-90	As for Cs, precipitation of $\text{Y}(\text{OH})_3$
Ba-140	As for Cs, precipitation of BaCrO_4
Zr-95 Pa-233	Extraction with thenoyltrifluoroacetone
Ag-110m	Precipitation of AgCl

TABLE 3.2-11
NUMBER OF ATOMS OF DIFFERENT NUCLIDES MEASURED IN THE LEACH
SOLUTIONS FROM STEEL CAPSULES OF R2-K13 (CORRECTED TO EOL)

<u>Nuclide</u>	<u>Cell 2</u>	<u>Cell 3</u>	<u>Ratio of Cell 2 to Cell 3</u>
I-131	3.7×10^{13}	1.2×10^{13}	3.1
Te-132	9.8×10^{11}	1.1×10^{11}	8.9
Cs-134	1.0×10^{16}	1.3×10^{15}	7.7
Cs-137	4.6×10^{16}	3.0×10^{15}	15.3
Sr-90	4.8×10^{12}	1.9×10^{13}	0.3
Ba-140	7.9×10^{11}	5.9×10^{11}	1.3
Zr-95(a)	2.4×10^{13}	2.9×10^{13}	0.8
Pa-233	1.7×10^{13}	3.9×10^{12}	4.4
Ag-110m	3.7×10^{13}	3.7×10^{13}	1.0

(a) Zr-95 is produced, in addition to fission, by the reaction $\text{Mo-98} (\text{n}, \alpha) \text{Zr-95}$. Mo was present in the capsules as thermal shields. Nb-95 and Nb-92m were found in considerable quantities in the leach solutions and they are produced by (n, p) reactions in Mo.

(a) SPACERS VOLATILIZED DURING FIRING, LEAVING A GAP

Fig. 3.2-4. Simplified drawing (units of mm) of fuel body for cells 2 and 3 of capsule R2-K13, including cross-section of fuel rod hole

piggyback samples consisted of matrix, graphite, loose fuel particle, dosimetry, burnup, and fission product release samples.

Fuel Rod Removal. Two of the three fuel stacks were removed from each cell (stack C remains in place in both cells for later core heatup simulation testing at KFA). All of the fuel rods appeared to be in excellent condition after removal. The fuel rod starting pushout force was measured for each of the cure-in-place rods and is shown in Table 3.2-12. The pushout force did not exceed 23 kg and averaged 11.4 kg. The upper (no. 2) fuel rod in each hole of both cells had the higher pushout force. This may have been related to the cure-in-place firing position during assembly at GA.

Fission Product Gas Measurement at KFA. A cold gas release test was performed in the KFA hot cells on each of the removed fuel rods and on the graphite body sections containing the "C" fuel stacks. These tests were performed at 60°C by placing each fuel rod or graphite section in a sealed chamber and monitoring the fission gas evolution from the rod. The results indicated a relative gas release of 6 to 8 units per fuel rod. By comparing the extracted fuel rod results with the C stacks that have not been removed, it might be concluded that all of the rods except rod 3B2 were removed without damage. This rod has been designated for metallographic examination at KFA.

Fuel Body Dimensional Change. The GLCC* H-451 graphite fuel bodies were dimensioned at KFA, and the results are shown in Table 3.2-13. The measured results appear to agree with the H-451 design curves as shown in Figs. 3.2-5 and 3.2-6.

Fuel Rod and Matrix Sample Dimensional Change. The fuel rods were dimensioned at KFA; the results are shown in Table 3.2-14. These results compared with the current fuel rod dimensional change model show:

* GLCC = Great Lakes Carbon Company.

TABLE 3.2-12
R2-K13 FUEL ROD PUSHOUT FORCE

Fuel Rod Number	Cell Number	Force Required	
		(kg)	(lb)
A1	2	2	4.4
A2	2	20	44.1
B1	2	1	2.2
B2	2	22	48.6
A1	3	3	6.6
A2	3	23	50.8
B1	3	3	6.6
B2	3	17	37.5

TABLE 3.2-13
R2-K13 GRAPHITE BODY DIMENSIONAL CHANGE

Body Number	Graphite Type	Fast Fluence ($\times 10^{25}$ n/m ²) [(E > 29 fJ)] _{HTGR}	Tempera- ture (°C)	Dimensional Change (%)	
				Diameter	Length
				Average of 9 Measurements	Average of 3 Measurements
2	H-451	7.8	1010	-1.2	-2.7
3	H-451	7.4	820	-1.1	-1.6

Fig. 3.2-5. Design curves for dimensional change of H-451 graphite, radial orientation, as a function of irradiation conditions, compared with R2-K13 fuel rod dimensional change

Fig. 3.2-6. Design curves for dimensional change of H-451 graphite, axial orientation, as a function of irradiation conditions, compared with R2-K13 fuel rod dimensional change

TABLE 3.2-14
R2-K13 FUEL ROD^(a) DIMENSIONAL CHANGE

Sample Number	Cell Number	Fast Fluence ($\times 10^{25}$ n/m ²) [(E > 29 fJ) _{HTGR}]	Temperature (°C)	Dimensional Change (%)(b)	
				Diameter Average of 6 Measurements	Length Average of 2 Measurements
2A1	2	7.8	1190	-0.9	+1.3
2A2	2	7.8	1190	-0.8	+1.3
2B1	2	7.8	1190	-0.9	+1.0
2B2	2	7.8	1190	-0.9	+1.1
3A1	3	7.4	980	-0.8	+1.2
3A2	3	7.4	980	-0.7	+2.4
3B1	3	7.4	980	-1.0	+1.8
3B2	3	7.4	980	-0.7	+1.3

(a) Approximately: 13% fissile TRISO-coated UCO.
6% fertile TRISO-coated ThO₂.

39% shim particles - impregnated H-451 graphite.
(b) Fired preirradiation rod dimensions: 12.48 mm diameter, 25.01 mm long. Dimensions are from rods removed from QC bodies, not rods irradiated in R2-K13.

1. Approximately the predicted amount of fuel rod diametral change.
2. A larger amount of length change than predicted.
3. A correspondinly larger amount of fuel rod total volume change.

Matrix samples were dimensioned at GA, and the results are shown in Table 3.2-15. The results agree well with the existing fuel rod dimensional change model.

Further analysis of both the fuel rod and matrix samples is continuing and will be used to improve the fuel rod dimensional change model.

3.2.3.3. Capsule HRB-16

Irradiation Conditions. The irradiation of capsule HRB-16 was completed in December 1981, and the results of the initial post-irradiation examination are reported in Refs. 3.2-8 and 3.2-9. During this report period, additional information on the detailed HFIR operating history during irradiation, in-pile fission gas release, and calculated fuel temperatures was established, as shown in Tables 3.2-16, 3.2-17, and 3.2-18.

Unbonded Particle Performance. A schematic of fuel sample arrangement in capsule HRB-16 is shown in Fig. 3.2-7. Of particular interest in the unbonded particle trays were several groups of particles that were tested without OPyC coatings. The OPyC's were burned off these particles prior to irradiation to increase the probability of pressure vessel failure, because the OPyC layer exerts a compressive load on the SiC during irradiation, enabling the particle to withstand higher internal gas pressures than would otherwise be possible. Three groups of particles were tested without OPyC layers: (1) $UC_{0.5}O_{1.5}$ particles that were highly faceted (trays 16 and 22) and highly nonfaceted (trays 17 and 23); (2) $(Th,U)O_2$ particles that had batch-typical SiC defects included (trays 3 and 12) and SiC defects removed (trays 2 and 11); and (3) standard UO_2 particles. In each case, a tray of standard particles with OPyC's was tested for comparison.

TABLE 3.2-15
R2-K13 FUEL ROD^(a) MATRIX DIMENSIONAL CHANGE

Sample Number	Cell Number	Fast Fluence ($\times 10^{25}$ n/m ²) [(E > 29 fJ) _{HTGR}]	Temperature (°C)	Dimensional Change (%)	
				<u>Diameter</u>	<u>Length</u>
				Average of 6 Measurements	Average of 5 Measurements
2-A5	2	7.8	1040	-5.0	+2.6
2-B4	2	7.8	1040	-5.4	+0.2
3-A4	3	7.4	840	-7.1	-0.3
3-B3	3	7.4	840	-6.7	-1.2

(a) Matrix Batch MO679
 Ks-15 graphite filler 38%
 A-240 pet. pitch 47%
 Octadecanol 10%
 Polystyrene 5%

TABLE 3.2-16
CAPSULE HRB-16 REACTOR OPERATING HISTORY
(1981)

HFIR Cycle	<u>Cycle Schedule</u>				<u>Irradiation Time (h)(a)</u>	
	<u>Begin</u>		<u>End</u>		<u>In</u>	<u>Total</u>
	<u>Date</u>	<u>Time</u>	<u>Date</u>	<u>Time</u>	<u>Cycle</u>	<u>Accumulated</u>
210(b)	6/21	5:06	7/12	18:33	516.24	516.24
211	7/15	21:11	8/06	12:20	517.92	1034.16
212	8/07	20:00	8/29	01:06	508.80	1542.96
213	8/30	15:17	9/20	09:42	498.24	2041.20
214	9/22	11:15	10/13	08:00	500.88	2542.08
215	10/17	17:31	11/07	23:03	511.20	3053.28
216	11/09	00:13	11/30	08:30	511.68	2564.96
217	12/01	21:55	12/23	11:23	516.72	4081.68

(a) Irradiation time is given in equivalent hours
at 100 MW.

(b) The reactor operated at reduced power (90, 95 MW)
on several occasions due to weather conditions.

TABLE 3.2-17
CAPSULE HRB-16 IN-PILE FISSION GAS RELEASE DATA (1981)

Example No.	Date	Irradiation Time (100 MWh)	Isotope Release Rate (R/B)				
			Kr-85m (x 10 ⁻⁵)	Kr-87 (x 10 ⁻⁵)	Kr-88 (x 10 ⁻⁵)	Xe-133 (x 10 ⁻⁵)	Xe-133 (x 10 ⁻⁴)
1	6/26	125.3	2.44	1.33	1.02	2.11	1.07
2	7/07	391.3	0.10	0.08	0.06	0.07	0.04
3	7/21	650.7	0.72	0.48	0.44	0.36	0.31
4	7/27	795.0	0.84	0.50	0.49	0.37	0.79
5	7/31	890.9	0.81	0.48	0.47	0.35	0.92
6	8/05	1011.6	1.08	0.65	0.52	0.56	0.73
7	8/12	1150.0	0.69	0.46	0.42	0.37	0.42
8	8/17	1266.2	1.26	0.84	0.77	0.58	0.96
9	8/21	1362.5	1.19	0.78	0.73	0.60	1.56
10	8/26	1482.2	0.82	0.52	0.51	0.27	0.41
11	9/03	1637.0	0.78	0.54	0.48	0.31	0.24
12	9/09	1781.8	1.00	0.69	0.64	0.36	0.52
13	9/18	1996.8	1.40	0.96	0.88	0.54	1.01
14	9/24	2094.7	1.05	0.71	0.66	0.29	0.22
15	10/01	2261.3	1.61	1.11	1.08	0.40	0.48
16	10/09	2449.6	1.83	1.25	1.14	0.44	0.60
17	10/20	2607.4	3.75	2.41	2.19	1.10	0.98
18	10/27	2776.0	5.00	2.92	3.23	1.16	1.57
19	11/02	2919.7	5.08	2.95	2.86	1.07	1.05
20	11/06	3015.8	5.12	3.07	2.84	1.37	1.06
21	11/13	3148.7	12.17	7.44	7.40	4.70	7.43
22	11/19	3296.7	15.10	8.76	9.11	4.86	10.02
23	11/24	3413.6	19.76	11.08	11.68	7.00	18.68
24	12/07	3675.4	19.61	10.75	10.10	6.21	11.42
25	12/11	3771.0	20.37	10.79	10.22	6.17	15.85
26	12/17	3914.1	20.84	12.78	13.58	7.33	19.58
27	12/22	4037.7	3.81 ^(a)	12.44	8.51	6.24	0.09

(a) It is not known why the last data set was inconsistent with previous values.

TABLE 3.2-18
PIE TIME-AVERAGED HRB-16 FUEL-ROD CENTERLINE TEMPERATURES (°C)

Rod No.	Thermocouple No.	Cycle I	Cycle II	Cycle III	Cycle IV	Cycle V	Cycle VI	Cycle VII	Cycle VIII	All Cycles
1		923	1045	1060	1079	1072	1049	1081	1046	1044
2	(TE 1, 9)	1040	1035	1157	1049	1060	1074	1130	1093	1080
3		1052	1116	1077	1082	1073	1071	1119	1096	1086
4	(TE 2, 10)	1128	1105	1101	1109	1115	1117	1161	1115	1119
5		1106	1132	1095	1114	1117	1126	1181	1161	1130
6		1128	1138	1106	1134	1141	1151	1205	1183	1148
7	(TE 3, 11)	1239	1169	1137	1142	1147	1132	1188	1172	1166
8		1158	1148	1125	1163	1174	1184	1236	1210	1175
9		1162	1145	1123	1164	1175	1185	1234	1207	1175
10	(TE 4, 5, 6)	1224	1193	1174	1188	1204	1206	1261	1239	1212
11		1176	1152	1131	1174	1187	1198	1249	1221	1186
12		1179	1157	1133	1173	1184	1194	1243	1214	1185
13	(TE 7)	1196	1182	1175	1189	1212	1204	1267	1248	1210
14		1161	1149	1117	1149	1158	1168	1220	1193	1165
15		1133	1133	1097	1123	1130	1140	1193	1168	1141
16(a)	(TE 8)	1189	1228	1226	1215	1242	1250	1321	1296	1246
17		1136	1166	1120	1126	1121	1122	1171	1144	1138
18		1173	1221	1165	1155	1138	1127	1168	1137	1161

(a) TE-temperatures at rod position 16 ran consistently higher than the computer temperatures at that position. One cause may be the metal containers gamma-heating in the "piggyback" experiments. This is not taken into account in the thermal analysis.

Fig. 3.2-7. Lengthwise cross-section of the loaded graphite sleeve of capsule HRB-16

The first observation made on the particles tested without OPyC layers was that the exposed SiC's had not corroded or degraded in any way during irradiation. It was suspected that the SiC might begin to sublime at the peak temperatures encountered in HRB-16, but this did not appear to have occurred in any of these samples. In all cases, the exposed SiC's appeared as shiny and smooth as before irradiation.

Secondly, for the "faceted versus nonfaceted" and "SiC defects versus no SiC defects" particles tested without OPyC's, though predicted failure levels were in the 5% to 20% range, the observed failure levels were less than 5% in all cases, as can be seen in Table 3.2-19. Since few failures were observed in a relatively small population of particles, no correlation of failure with the different particle defects was found. The failure predictions for these particles were based on an assumed SiC strength of 33 MPa (4.8 kpsi). As a result of these findings and other irradiation test evidence that had suggested that the SiC strength might be higher, a new assumed SiC strength of 350 MPa (51 kpsi) is currently being used in the TRISO*SiC fuel particle pressure vessel design models.

The (Th,U)₂O₇ TRISO particles in tray 10 were the only unbonded particles having OPyC failures, in this case 5 out of 107 particles examined, or ~5% of the sample. This particle type was the largest tested in HRB-16 (884 μ m mean total diameter) and tray 10 was the highest exposure position for particles of this type tested with OPyC layers. This level of OPyC failure in these particles is consistent with recent capsule data for large diameter particles.

Fuel Rod Dimensional Measurements. The outside diameter of the POCO graphite sleeve was measured during the capsule disassembly before the sleeve was cut for removal of the fuel samples. After the fuel was retrieved, the fuel rods were also dimensioned for diameter. These data are listed in Table 3.2-20 along with the preirradiation measurements and the calculated dimensional changes. The data have not been analyzed yet but appear to be in general agreement with other recent irradiation data. After the disassembly, the wall thickness of the three graphite sleeve

TABLE 3.2-19
CAPSULE HRB-16 UNBONDED PARTICLE TRAY VISUAL EXAM RESULTS

Tray Number	Particle Batch	Kernel Type	Coating Type	Estimated Exposures		No. of Particles Examined	Total Particle Failures	OPyC Failures	Comments/ Observations	Description
				Fast Fluence ^(a)	Burnup (% FIMA)					
1	CT6B-932	ThC ₂	TRISO	2.9	16.0	72 ^(b)	0	0	(c)	Half of particles obtained from a debonded, fired fuel rod and have outer ~10 µm of OPyC impregnated with matrix; rest of particles not impregnated.
2	6155-01-030	(Th,U)O ₂	TRISO	3.2	3.9	107	5	0	(d)	OPyC coatings removed by burn-back; all particles with SIC defects discarded from this sample.
3	6155-01-030	(Th,U)O ₂	TRISO	3.2	4.0	108	2	0	(d)	OPyC coatings removed by burn-back; particles with defective SIC coatings not removed.
4	6151-23-020	UC ₂	TRISO	3.6	19.5	110	0	0	(c)	Split sample taken from parent-batch.
5	6151-23-020	UC ₂	TRISO	4.1	20.8	109	0	0	(c)	Split sample taken from parent-batch.
6	8423-49	UC ₂	ZrC-TRISO	4.1	20.9	109	0	0	(c)	Split sample taken from parent-batch.
7	6152-03-010	UO ₂ *	TRISO	4.7	22.0	108	0	0	(c)	Split sample taken from parent-batch.
8	6152-04-010	UO ₂	TRISO	5.2	23.0	110	0	0	(c)	Split sample taken from parent-batch.
9	6152-04-010	UO ₂	TRISO	5.2	23.1	110	0	0	(c)	OPyC coatings removed by burn-back.
10	6155-01-030	(Th,U)O ₂	TRISO	5.5	8.0	107	1	5	(d)	Split sample taken from parent-batch.
11	6155-01-030	(Th,U)O ₂	TRISO	5.9	8.4	104	2	0	(d)	Same description as for tray 2.
12	6155-01-030	(Th,U)O ₂	TRISO	5.9	8.5	107	3	0	(d)	Same description as for tray 3.
13	6152-04-010	UO ₂ *	TRISO	6.1	25.2	110	0	0	(c)	Split sample taken from parent-batch.
14	6152-03-010	UO ₂ *	TRISO	6.3	25.5	110	0	0	(c)	Split sample taken from parent-batch.
15	6157-11-010	UC _{0.5} O _{1.5}	TRISO	6.4	25.6	110	1	0	(d)	Split sample taken from parent-batch.
16	6157-11-010	UC _{0.5} O _{1.5}	TRISO	6.3	25.6	104	1	0	(d)	OPyC coatings removed by burn-back; most faceted particles selected for this sample.
17	6157-11-010	UC _{0.5} O _{1.5}	TRISO	6.3	25.6	110	4	0	(d)	OPyC coatings removed by burn-back; roundest, non-faceted particles selected for this sample.
18	6152-03-010	UO ₂ *	TRISO	6.3	25.3	110	0	0	(c)	Split sample taken from parent batch.
19	6157-11-010	UC _{0.5} O _{1.5}	TRISO	6.1	25.0	110	0	0	(c)	Split sample taken from parent-batch.
20	6151-23-020	UC ₂	TRISO	6.1	24.9	109	0	0	(c)	Split sample taken from parent-batch.
21	6152-04-010	UO ₂	TRISO	5.8	24.3	110	0	0	(c)	Split sample taken from parent-batch.
22	6157-11-010	UC _{0.5} O _{1.5}	TRISO	5.5	23.6	105	2	0	(d)	Same description as for tray 16.
23	6157-11-010	UC _{0.5} O _{1.5}	TRISO	5.5	23.5	109	1	0	(d)	Same description as for tray 17.
24	6157-11-010	UC _{0.5} O _{1.5}	TRISO	5.0	22.7	106 ^(b)	0	0	(c)	Split sample taken from parent-batch.
25	6152-04-010	UO ₂	TRISO	4.4	21.7	105 ^(b)	0	0	(c)	Split sample taken from parent-batch.
26	6152-03-010	UO ₂ *	TRISO	4.4	21.6	108	0	0	(c)	Split sample taken from parent-batch.
27	8423-47	UC _x O _y	ZrC-TRISO	3.8	20.3	109	0	0	(c)	Split sample taken from parent-batch.

(a) $\times 10^{25}$ n/m² (E > 29 fJ)HTGR.

(b) Tray lid loose; some particles lost during disassembly.

(c) All particles intact and in good condition unless otherwise noted.

(d) Some bare kernels and/or coating fragments present; number of failed particles estimated from these debris.

TABLE 3.2-20
OD DIMENSIONS AND RELATIVE STRAINS IN THE HRB-16
FUEL RODS AND GRAPHITE SLEEVE

	<u>Rods</u>			<u>Sleeve</u>		
	<u>Pre- irradiation (mm)</u>	<u>Post- irradiation (mm)</u>	<u>Relative strain^(a)</u>	<u>Pre- irradiation (mm)</u>	<u>Post- irradiation (mm)</u>	<u>Relative strain^(a)</u>
1	12.490	12.342	-0.011861	23.825	23.792	-0.00136
2	12.470	12.337	-0.010751	23.825	23.853	0.00117
3	12.470	12.352	-0.009461	23.825	23.876	0.00213
4	12.470	12.362	-0.008646	23.825	23.904	0.00330
5	12.470	12.377	-0.007356	23.823	23.932	0.00458
6	12.470	12.403	-0.005455	23.817	23.973	0.00651
7	12.460	12.408	-0.004113	23.815	23.990	0.00736
8	12.460	12.443	-0.001395	23.815	24.003	0.00789
9	12.480	12.489	0.000803	23.812	24.016	0.00853
10	12.460	12.461	0.000032	23.815	23.025	0.00939
11	12.470	12.461	-0.000635	23.815	24.039	0.00939
12	12.460	12.466	0.000439	23.815	24.041	0.00939
13	12.460	12.438	-0.001735	23.815	24.041	0.00939
14	12.480	12.441	-0.003199	23.815	24.041	0.00939
15	12.490	12.410	-0.006369	23.815	24.041	0.00939
16	12.470	12.385	-0.006745	23.815	24.041	0.00939
17	12.490	12.355	-0.010771	23.817	24.041	0.00928
18	12.470	12.337	-0.010683	23.820	24.041	0.00917

^(a) Relative strain is given by $(D_2 - D_1)/D_1$
where

D_2 = postirradiation diameter
 D_1 = preirradiation diameter.

sections was measured at specific locations along their length. This information allows a determination of the dimensional change of the inner diameter of the sleeve, which is necessary data for the thermal analysis.

Fuel Rod Fission Gas Release Measurements in TRIGA. Several fuel rods, unbonded particle samples, and all the piggyback samples were returned to GA shortly after the capsule disassembly, as called for in the HRB-16 PIE Plan (Ref. 3.2-1). Two of the fuel rods and four of the unbonded particle samples were submitted for postirradiation fission gas release measurements in GA's TRIGA reactor. The samples were irradiated in TRIGA at 1100°C for 30 minutes, during which time any of the short-lived Kr-85m isotope escaping from the fuel was trapped. The traps were then gamma counted, and a determination of the R/B of Kr-85m for each sample was made. The results are presented in Table 3.2-21. In comparison to the in-pile R/B for all fuel samples of $\sim 2.1 \times 10^{-4}$, three of these samples had R/B values lower and three had R/B values ~ 2 to 3 times higher. These data will be correlated with other PIE results to assess the overall performance of the HRB-16 fuel particles.

3.2.3.4. Capsules HRB-17 and HRB-18

This experiment is described in Ref. 3.2-9. During this report period, LEU UCO kernels were manufactured and coated with a single PyC layer designed to fail during the first cycle of exposure in HFIR. Subsequent exposure to moisture is planned to characterize effects of hydrolysis on fission gas release rates.

Fuel rods were fabricated and heat treated using standard ThO₂/UCO TRISO particles along with the special particles with only a single PyC coating. Five of the special "designed to fail" particles were used in each rod, comprising about 3% of the fissile particle loading.

An historical sample of a HEU UC₂ TRISO irradiated in capsule SSL-2 and P13S was selected to be used for comparison with HEU UCO TRISO performance in HRB-17 and HRB-18.

TABLE 3.2-21
CAPSULE HRB-16 END OF LIFE FISSION GAS RELEASE IN TRIGA FOR SEVERAL
FUEL SAMPLES

Capsule Sample	Fissile Particle Type	Fissile Burnup (% FIMA)	Fertile Particle Type	Fertile Burnup (% FIMA)	Fast Fluence (a)	Kr-85m R/B at 1100°C (b)
Fuel rod 3	UO ₂ * TRISO	20.9	ThO ₂ TRISO	2.3	4.2	9.0 x 10 ⁻⁶
Fuel rod 10	UO ₂ * TRISO	27.6	ThO ₂ TRISO	5.4	6.3	4.8 x 10 ⁻⁴
Tray 4(c)	UC ₂ TRISO	19.5	--	-	3.6	4.6 x 10 ⁻⁴
Tray 14(c)	UC ₂ TRISO	25.5	--	-	6.3	6.3 x 10 ⁻⁵
Tray 19(c)	UC ₂ TRISO	25.0	--	-	6.1	7.3 x 10 ⁻⁴
Tray 21(c)	UC ₂ TRISO	24.3	--	-	5.8	4.0 x 10 ⁻⁵

(a) $\times 10^{25}$ n/m² (E > 29 fJ)_{HTGR}.

(b) Measured in GA's TRIGA reactor.

(c) ~100 particle samples.

As part of the U.S. DOE/FRG Umbrella Agreement on Fuel Development, a 2-cm-long section of the capsule was identified for tests of fuel from the KFA/HBK project in Germany for irradiation of FRG LEU UO_2 TRISO particles with variation in SiC thickness. Tentative acceptance of the offer was received.

The capsules were scheduled for insertion in HFIR in September 1983. A decision was made at ORNL to shut down HFIR in October 1983 for three to four months for replacement of the beryllium reflectors. This will result in delay of HRB-17 and HRB-18 insertion to late January 1984. The programmatic impact of capsule delay was not considered significant relative to the risk of long-term HFIR operation without reflector replacement.

3.2.4. Fort St. Vrain Fuel Test Elements

3.2.4.1. FSV FTE-2 Destructive PIE

FSV FTE-2 S/N 7-0206 is one of eight test elements inserted into FSV core segments 2 through 7 during the first refueling in March and April 1979 to allow successive withdrawal of elements after irradiation cycles 2 through 7. FTE-1 through FTE-6 were fully cured-in-place and contain HTGR pilot-plant-produced, dense-melted UC_2 TRISO fissile and ThO_2 TRISO fertile fuel as driver fuel. In addition to the driver fuel, FTE-2, FTE-4, and FTE-6 contain a limited amount of experimental fuel. FTE-7 and FTE-8 contain fuel particles and rods identical to segment 7 reload fuel, blend FSV 14 $[(\text{Th},\text{U})\text{C}_2$ and ThC_2 TRISO]. FSV FTE-2 will be the first of the fuel test elements to undergo PIE at GA. The element will be removed from the core with segment 3 after accumulating a fast neutron fluence of approximately 2.13×10^{25} (E 29 fJ)_{HTGR} in core location 22.06.F.06 for 489 EFPD. PIE objectives for FTE-2 are given in Table 3.2-22.

TABLE 3.2-22
PIE OBJECTIVES FOR FTE-2

<u>Data</u>	<u>Method</u>	<u>Objectives</u>
<u>A. Graphite fuel block</u>		
1. Structural integrity	Perform visual examination.	Verify acceptable performance of H-451 graphite block.
2. Dimensional change	Perform metrological exam. in FSV HSF using metrology robot, and compare results with preirradiation dimensions.	Verify design code and data base strain and bow calculations. Verify acceptable performance of H-451 graphite blocks.
<u>B. Fuel rods^(a)</u>		
1. Structural integrity	Perform visual examination.	Verify acceptable structural performance of fuel rods.
2. Plenum depths and fuel stack push-out forces	Measure plenum depths with depth gauge, and push fuel stacks out of block using a special force measuring device.	Verify the absence of fuel rod - fuel body mechanical interaction.
3. Dimensional change	Measure rods using MICROROBOT, and compare results with preirradiation dimensions.	Verify dimensional stability of rods and compare results with design data.
4. Fission product inventories	Use gamma scanning.	Determine axial and radial power distributions used to verify nuclear design calculations.
5. Fission gas release	Activate rods in TRIGA, and measure fission gas release.	Measure total coating failure used to verify fuel performance predictions.
6. Microstructure	Use metallography.	Determine condition of matrix and fuel particles. Measure coating failure and use to verify fuel performance predictions.
7. Heavy metal inventory and isotopic distribution	Use wet chemistry.	Establish the validity of core physics depletion methodology.
<u>C. Monitor packages</u>		
1. Irradiation temperatures	Use isochronal annealing of SiC temperature monitors.	Verify thermal design calculations.
2. Fast and thermal neutron fluence	Use dosimetry.	Verify nuclear design calculations.
3. Fuel burnup	Use radiochemistry.	Verify nuclear design calculations.

(a) Only three of the seven fuel blends contained in FTE-2 will be examined in detail:

Test array 1 - Blend FSV 12 [(Th,U)C₂, ThC₂ TRISO].

Test array 2 - Blend MEU 12 adj. [(Th,U))₂, ThO₂ TRISO].

Test array 7 - Blend 12 driver (UC₂, ThO₂ TRISO].

Test arrays 3-6 contain UCO WAR TRISO/ThO₂ BISO fuel, which is not of sufficient technical interest to warrant examination at this low fluence level.

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- 3.2-4. Hick, H., "Review of Unusual Particle Bursting Phenomena," DPTN/760, 2 March 1976.
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- 3.2-7. Rollig, K., "Kinetic Model for the Failure Rate of Coated Oxide Particles at High Temperature," Dragon Project Report DPRDD/1699, May 30, 1968.
- 3.2-8. "HTGR Fuel Technology Semiannual Report for the Period Ending March 31, 1982," DOE Report GA-A16757, June 1982.
- 3.2-9. "HTGR Fuel Technolgy Semiannual Report for the Period Ending September 30, 1982," DOE Report GA-A16919, GA Technologies Inc., to be published.

3.3. FUEL CYCLE (1105)

3.3.1. Scope

This task includes all work necessary for establishing the technical data base and updating resource and processing costs that enter fuel cycle cost calculations for HTGR applications.

3.3.2. Economic Data Base

During this period, evaluations were made on the input data for fuel cycle component costs. This effort continues into the March-September reporting period, and will be reported in the next semiannual report.

3.4. FISSION PRODUCT AND COOLANT CHEMISTRY (1106)

3.4.1. Scope

The scope of this task includes all the experimental and analytical work necessary to describe the mechanism for release of fission products from fuel particles and the interaction of fission products and other primary coolant impurities with fuel and other reactor plant components. Specific tasks included (1) coolant impurity/fuel material interactions, (2) fission product release and transport, (3) fission product plateout, and (4) validation of fission product methods.

3.4.2. Coolant Impurity/Fuel Material Interaction

3.4.2.1. Pressure Dependence of Fission Gas Release

A special fuel rod containing a central void and dummy particles was made for the tests to characterize the effect of total gas pressure on fission gas release. UO₂ powder will be placed in the central fuel rod void so that fission gas generated during TRIGA irradiation will be released and

pass through the fuel rod matrix prior to being swept away to the collection system.

3.4.3. Fission Product Release and Transport

3.4.3.1. Release of Fission Products from Fuel Particles

The samples from in-pile fission product release experiments were returned to GA following irradiation in the HFIR reactor at ORNL (capsules HT-35, HRB-15A, and HRB-16) and in the R2-Studsvik reactor (capsule R2-K13). The objectives of these experiments are (1) to measure the in-pile transport of fission products in intact and laser-failed fuel particles as a function of irradiation conditions; (2) to measure the relative sorptivity of fission products on the material components (pyrocarbon, SiC, buffer material) of fuel particles and on fuel rod matrix material, unimpregnated and impregnated H-451 graphite; (3) to assess the effect of hydrolysis on fission product metal release under in-pile conditions; and (4) to provide data for developing and validating release models for silver, cesium, and other fission products. The samples consist of niobium encapsulated graphite crucibles containing fuel particles in various configurations. There are intact particles; laser-failed particles with small holes extending from the particle surface to the surfaces of the kernel, the buffer, and the inner pyrocarbon coatings; and laser-failed particles that have been hydrolyzed prior to encapsulation. The kernel types of these particles are UC_2 , UCO , $(Th,U)O_2$, and ThO_2 . In addition, small pieces of H-451 graphite, fuel rod matrix material, pyrocarbon, and SiC coatings were encapsulated in pairs, along with quantities of cesium or silver, in order to determine the distribution of these fission product elements between the materials under irradiation conditions. There are 68 samples altogether.

All of the samples have now been gamma counted prior to disassembly. Comparison of the measured inventories with the calculated inventories shows, in general, agreement for ruthenium, zirconium, and cesium. This indicates that these elements have been retained in the encapsulated niobium tubes; for silver, however, some uncertainty remains over the cross-sections

used in the inventory calculation. A comparison of measured and calculated silver inventories has not yet been completed.

The equipment to be used for disassembly has been constructed. The samples will be disassembled in a six-port vacuum chamber having a micro-manipulator with a stepping motor for rotary motion, a linear motion feedthrough, a gas port, a quartz window for observation and alignment, and a sapphire window through which a Nd:glass laser beam at $1.06\text{ }\mu\text{m}$ will pass. This assembly is shown in Fig. 3.4-1. In operation, the niobium encapsulation tube is inserted into the chamber affixed to the linear motion feedthrough. The system is evacuated and the tube is transferred to the micro-manipulator, then positioned so that the laser beam can be directed on the tube near the free end. A hole is then made in the tube with a laser pulse and the gases collected from the opened tube. The gases are removed from the chamber and collected in traps. The free end of the tube is then cut by alternately pulsing the laser and rotating the tube. The cut end and the tube and its contents are transferred to the linear feedthrough and then removed from the chamber. The tube and contents will be individually gamma counted so that fission product distributions can be determined.

3.4.3.2. Cesium and Silver Transport in H-451 Graphite

The laboratory studies of the diffusion of cesium in H-451 graphite (Ref. 3.4-1) continued with experiments designed to obtain additional data on the temperature dependence of diffusion, the effect of graphite oxidation, the release of cesium from a graphite surface, and the variability of the diffusion parameters from different graphite logs.

Experiments conducted with a graphite sample oxidized to 3 wt% burnoff yielded the same values of the diffusion parameters as obtained in experiments with a sample oxidized to 6 wt% burnoff. The values of the diffusion and trapping coefficients decreased by factors of 3 and 10, respectively, compared to the values of these parameters in the unoxidized graphite sample. Since oxidation of graphite preferentially involves the binder carbon of the graphite (Ref. 3.4-2), the reduction in the trapping coefficient

Fig. 3.4-1. Vacuum work chamber and manipulator

implies that most traps are located in the binder carbon. The reduction in the diffusion coefficient implies that cesium binds somewhat more strongly on oxidized than on unoxidized surfaces. The similar results for the experiments at 3 and 6 wt% burnoff indicate that the variation due to oxidation is much more rapid at lower burnoffs.

The graphite hexagonal blocks of which the reactor core is composed are each obtained from a graphite log. Since the properties of the graphite vary from log to log, a diffusion experiment was conducted with a graphite sample from a different log than that used for previous experiments. This experiment, conducted at a nominal temperature of 850°C, yielded a cesium profile very much like one obtained at 900°C on a sample from the usual source log. Thus the first estimate of the variability of cesium diffusion in H-451 graphite between source logs is equivalent to a temperature difference of 50°C.

To describe the diffusion of cesium in the graphite core of the reactor, knowledge of the release of cesium from the graphite, as well as transport through the graphite, is required. Previous experiments have addressed the latter aspect. An experiment at 954°C was recently completed in which the cesium diffused to and was released from a surface of an H-451 graphite sample. Analysis of this experiment, together with results of previous diffusion experiments in semi-infinite graphite media, should provide information sufficient to describe, in a first attempt, the transport of cesium through and release from the graphite web of the HTGR core.

3.4.3.3. U.S./FRG Fission Product Data Book

As a part of the cooperative effort to establish a U.S./FRG Fission Product Data Book under the U.S./FRG Umbrella agreement for Cooperation in GCR Development, work was conducted at KFA, Juelich, to define a data base for fission product transport in fuel particles and core materials.

The available KFA data considerably enlarge the data base for developing fission product transport models valid under reactor conditions. Also,

these data were obtained under conditions more closely approaching those established in the normal operation of the high-temperature, gas-cooled reactor than is the case for data obtained by GA. In particular, the neutron flux in the KFA experiments ranged from 0.1 to 3 times the neutron flux encountered under normal operating conditions, whereas for the GA data the range lies above the latter flux by a factor of 8 or more.

The data examined in Juelich consisted of reactor experiments conducted by KFA during the period 1977 to 1981 for which the amount of fission product transport data justified consideration. The experiments included FRJ2-P22, FRJ2-P23 and FRJ2-P25; HFR-K1; R2-K12; and FRJ2-K11. These experiments contained (Th,U)O₂ BISO and TRISO particles, as well as UC₂ and ThO₂ TRISO particles; most of the available fission product data apply to the (Th,U)O₂ particles.

The examination involved familiarization with the KFA experimental data, analysis of these to provide fractional release values, and collation and documentation of the release data and the associated thermal and neutronic data. The documentation, with perhaps minor revisions, should permit calculations to be made for comparison with the experimental results, except in those cases (HFR-K1 and FRJ2-K11) where the thermal analysis is pending.

The KFA experimental data provide part of the data base for establishing the model for fission product metal transport in and release from fuel particles. In addition, some pre-1977 KFA reactor data and GA in-pile fission product release experiments currently being analyzed should also be included in the data base.

On the basis of the work at KFA, the following are recommended:

1. Complete the analysis and documentation of fission product release data from KFA irradiation experiments.

2. Combine KFA, ORNL, and GA data into a source document that will provide the information necessary to perform fission product release calculations under reactor conditions and to develop a mutually acceptable model.
3. Develop the fission product release model to the stage that the documented release data are satisfactorily predicted.

3.4.4. Fission Product Plateout

Preparation for the experiments on the sorptivity of cesium on graphite and primary circuit alloys continued. This work was interrupted for seven weeks by illness. The status of the apparatus for performing the sorptivity measurements is as follows:

1. The photomultiplier tube - optical system has been aligned. In this process, iris diaphragms and mirrors had to be added to the optical system to remove unanticipated spontaneous emission signals and other unwanted contributions to the beam from the Nd:YAG - dye laser system.
2. After negotiations with the contractor over the design of the environmental chamber to be used in providing a controlled temperature environment for the reference cesium fluorescence cell over the temperature range -100° to $+100^{\circ}\text{C}$, a mutually acceptable design has been achieved and the chamber has been ordered.
3. A cesium-filled prototype cell has been fabricated and will be used in the first series of experimental measurements.
4. The system for converting the analog signals from the photomultiplier tube and power meter to digital form for subsequent processing in the computer system is ready. This has involved interfacing the components as well as providing software for processing and analyzing the incoming signals.

5. The computer system has been completed with the acquisition and installation of a printer. Several of the components of the system were tested, were found to be operating below specifications, and had to be returned to the manufacturer and replaced by new components.

The first experiments are expected to be conducted within the next several months.

3.4.5. Oxidation Rate of UCO

In this work, irradiated TRISO UCO particles with coatings failed by laser drilling were heated in helium containing air and moisture levels that may be experienced during fuel reload operations in an HTGR. Results are variable, but most particles showed low reaction rates at hypothetical reload temperatures of 200° to 300°C. Analysis and further tests are under way to quantify results in support of design assumption.

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3.5. TRADE STUDIES (1107)

3.5.1. Scope

This task included technical trade-off evaluations on parts of the program in which manpower and facilities could potentially be shifted for the benefit of the total HTGR program. The trade studies were recommended in the conclusions of the HTGR Baseline Review completed in FY 82.

3.5.2. Initial Trade Studies

Three trade studies were initiated during this reporting period on (1) fuel process unit operations, (2) fuel materials and process modifications that influence fission product release performance, and (3) HEU fuel cycle strategies. Conclusions on these studies are planned for the next reporting period.

4. COMPONENT DV&S

4.1. REACTOR INTERNALS (1417)

4.1.1. Scope

The scope of this task involves coordination with the Core Support Post Test series being conducted in the Core Flow Test Loop (CFTL) at ORNL. Activities in this reporting period included (1) delivery of Stackpole 2020 graphite for use in the CFTL Test Zero and for graphite oxidation tests at GA and (2) release of a test specification for graphite oxidation tests at GA. Both of these activities occurred at the end of the report period.

4.1.2. Oxidation Tests

The objective of this test series is to determine the relative oxidation characteristics of 2020 graphite to be used in CFTL Test Zero at ORNL. The material in these tests is taken from the same lots as are used in Test Zero.

Two types of tests are being performed: microbalance tests and tube tests. The microbalance test procedure involves use of automatic recording microbalance reaction rates with which instantaneous reaction rates, at temperatures to 900°C, are measured by sample weight-loss determinations. The tube tests involve exposing a number of samples in a tube furnace to a H₂O-H₂-He mixture at 700°C for several hundred hours.

Results of this work will be reported in the next semiannual report.

4.2. REACTOR CORE (1418)

4.2.1. Scope

The scope of this task includes structural and thermal test programs to support fuel element design and development of fuel element structural criteria. The structural test programs include (1) tests to determine the strength of the fuel element flange and (2) tests to determine the strength of irradiated Fort St. Vrain fuel elements. The thermal test programs include (1) tests and analyses of corrections to prevent core fluctuations and (2) flanged fuel element crossflow tests. Structural criteria development centered around evaluation of an alternative criterion.

4.2.2. Fuel Element Flange Structural Test Program

A test specification was written for a structural test program to determine shear strength of the fuel element flange in case of a seismic event. The overall test program consists of the following two major test series:

1. A design optimization test series on two-dimensional (2-D) specimens with different flange configurations.
2. A confirmatory test series using the optimum flange configuration from the 2-D test series. This series involves constructing full-scale test models of the entire fuel element flange and loading the test models to failure.

4.2.2.1. Design Optimization Test Series

The 2-D test series was completed within the reporting period. A total of 33 specimens were tested, involving six configurations for the fuel element flange. The effect of fuel and coolant holes adjacent to the flange was investigated. All test specimens were made of H-327 graphite.

Specific objectives of the 2-D test series were as follows:

1. Investigate the effect of the following on flange structural strength:
 - a. Interelement land seal (being considered for use around control rod holes).
 - b. Interelement plenum height.
 - c. Flange thickness.
 - d. Adjacent coolant and fuel holes.
2. Determine failure load and load-deflection characteristics for every specimen tested.
3. Perform preliminary investigations of use of acoustic emission in predicting failure loads.
4. Select a flange configuration for the ensuing full-flange structural test program.

The general approach was to mount a series of graphite specimens in a 4.45 kN (10,000 lb) Instron structural test machine and load into shear failure through a load piece constructed of the same type of graphite as the test specimen. Each specimen was mounted in the test machine so that the specimen flange was oriented in a horizontal direction. A vertical load was applied through a graphite load piece so as to fail the test specimen at the flange, thus simulating loading during a seismic event. Parameters recorded during each test were (1) specimen failure load, (2) flange deflection, by a Baldwin deflectometer, and (3) test machine crosshead movement, by a dial indicator. A schematic of the test installation is shown in Figure 4.2-1. In this test series, the term "test specimen" correctly refers to the above-mentioned "specimen pair," consisting of a male piece (corresponding to the fuel element lower end) and a female piece (corresponding to the fuel element flange). Where the term "specimen" is used hereafter in this report,

Fig. 4.2-1. Installation schematic - 2-D flange structural tests

it refers to the female flanged piece only. The term "loading piece" is used to denote the male piece.

Five different flange configurations were tested in specimens without fuel or coolant holes. Specimens with fuel and coolant holes used six flange configurations, including the five used in the specimens without holes. Three specimens of each different configuration were tested.

Each specimen was made in the form of a block, as shown in Fig. 4.2-2. Each block was 63.5 mm (2.75 in.) high by 81.3 mm (3.20 in.) wide. The length of each block was 127 mm (5.0 in.), including the length of the flange.

Each of the flange configurations is shown in Fig. 4.2-3. Specimens without holes used a "P" designation; specimens with holes used a "Q" designation. The specimen identification system was as follows:

X	_____	X	_____	X
"P" or "Q", depending whether the specimen incorporated fuel and coolant holes	The number of the flange configuration (1 through 6)	A lower case letter designating an indi- vidual specimen of a specific type (a through d)

specimen type				

For example, specimen P3a refers to one of four specimens made with specimen type P3, as shown in Fig. 4.2-2.

The specimen loading pieces are shown in Fig. 4.2-4. Each loading piece was machined with a male flange-mating portion typical of a fuel element design. Loading pieces used for testing P specimens did not contain holes. Loading specimens for testing Q specimens contained fuel and coolant holes, so as to simulate the stiffness of an actual fuel element male portion where it mates with the flange.

Fig. 4.2-2. Pretest photograph of specimen Q1a

ALL DIMENSIONS
ARE IN
MILLIMETERS

Fig. 4.2-3. Flange configurations

Fig. 4.2-4. Load piece QM-2

A summary of failure loads and deflection for all tested specimens is given in Table 4.2-1. Table 4.2-1 shows that the overall range of failure loads was 4.36 kN (980 lb) to 6.32 kN (1420 lb). Deflections at failure, as measured by the deflectometer at 6.4 mm (0.25 in.) behind the flange leading edge, ranged from 0.39 mm (0.0155 in.) to 0.83 mm (0.0325 in.). Dial indicator readings, which measured test machine crosshead deflection relative to the stationary machine baseplate, were higher than the deflectometer readings by approximately 0.30 mm (0.12 in.), due to deflection of the cradles and other portions of the test system.

Prior to testing, finite-element stress analyses were made of specimen types P1 and P3, using the TWOD structural analysis code. A comparison between analytical predictions and mean experimental values of failure loads is as follows:

Specimen Type	Failure Load - kN (lb)	
	Mean Experimental Value	Analytical Prediction
P1	5.56 (1250)	5.20 (1170)
P3	5.28 (1187)	4.40 (890)
P4	5.64 (1267)	--
P5	5.96 (1340)	--
P6	4.68 (1053)	--

The experimental value was higher than the analytical prediction by 6.8% in P1 and 33% in P3.

Mean failure loads for specimens without holes were higher than for specimens with holes in the cases of P1, P3, and P4. Mean failure load for P1 was 19% higher than for Q1, 16% higher for P3 than for Q3, and 16% higher for P4 than for Q4. In the case of P5, mean failure load was 5.98 kN (1340 lb) as compared with 6.15 kN (1383 lb) for Q5. In the case of P6, mean failure load was 4.68 kN (1053 lb) as compared with 4.83 kN (1085 lb) for Q6. Since the presence of holes would serve to weaken, not strengthen, the

TABLE 4.2-1
FUEL ELEMENT FLANGE STRUCTURAL TEST PROGRAM 2-D TEST SERIES

Test Number	Date	Summary of Test Results			Deflection at Failure (in.)(b)		Notes
		Specimen Number	Loading Piece	Failure Load (LBF)(a)	Deflectometer	Dial Indicator	
1	1/21/83	P1a	PM-1	1,250	0.0250	0.0370	Failure at 0.44-in. radius
2	1/24/83	P4a	PM-1	1,140	0.0230	0.0330	Failure at 0.125-in. radius
3	1/25/83	P1b	PM-1	1,420	0.0260	0.0415	Failure at 0.44-in. radius
4	1/26/83	P1c	PM-1	1,080	0.0240	0.0375	
5	1/27/83	P3a	PM-1	1,250	0.0265	0.0370	Failure at 0.030-in. radius
6	1/27/83	P3b	PM-1	1,140	0.0225	0.0345	
7	1/28/83	P4c	PM-1	1,170	0.0245	0.0377	Failure at 0.125-in. radius
8	1/31/83	P4b	PM-1	1,280	0.0255	0.0390	
9	2/01/83	P4c	PM-1	1,380	0.0260	0.0415	Failure at 0.030-in. radius
10	2/01/83	P5a	PM-1	1,250	0.0245	0.0360	
11	2/02/83	P5b	PM-1	1,360	0.0280	0.0445	Failure at 0.44-in. radius
12	2/02/83	P5c	PM-1	1,410	0.0270	0.0371	
13	2/03/83	Q1a	QM-1	1,070	0.0325	0.0472	Failure through lower hole row
14	2/04/83	Q1b	QM-1	1,080	0.0295	0.0425	Failure across middle of bottom hole row
15	2/07/83	Q1d	QM-2	1,050	0.0262	0.0375	Failure at 0.28-in. radius
16	2/07/83	Q2a	QM-2	1,040	0.0230	0.0350	
17	2/08/83	Q2b	QM-2	1,140	0.0230	0.0345	Failure at 0.125-in. radius
18	2/08/83	Q2c	QM-2	1,070	0.0210	0.0314	
19	2/09/83	Q4a	QM-2	1,030	0.0205	0.0293	Failure at middle of bottom hole row
20	2/11/83	Q4b	QM-2	1,045	0.0217	0.0310	
21	2/14/83	Q4c	QM-2	1,190	0.0240	0.0340	Failure at base of bottom hole row
22	2/16/83	Q3b	QM-2	990	0.0185	0.0310	
23	2/16/83	Q3d	QM-2	1,090	0.0207	0.0324	Failure at base of bottom hole row
24	2/17/83	Q3c	QM-2	1,000	0.0235	0.0314	Failure at middle of bottom hole row
25	2/17/83	Q5a	QM-2	1,390	0.0257	0.0378	Failure at base of bottom hole row
26	2/17/83	Q5b	QM-2	1,340	0.0240	0.0366	Failure at 0.030-in. radius
27	2/21/83	Q5c	QM-2	1,420	0.0240	0.0370	Failure at 0.125-in. radius
28	3/14/83	P6a	QM-2	1,080	0.0190	0.0278	Failure at 0.125-in. radius
29	3/14/83	Q6a	QM-2	1,095	0.0180	0.0284	Failure at 0.125-in. radius
30	3/14/83	P6b	QM-2	980	0.0155	0.0248	Failure at middle of bottom hole row
31	3/14/83	P6c	QM-2	1,100	0.0175	0.0274	
32	3/17/83	Q6b	QM-2	1,120	0.0200	0.0298	Failure at middle of bottom hole row
33	3/17/83	Q6c	QM-2	1,040	0.0167	0.0280	

(a)₁ LBF = 4.448 N.

(b)₁ in. = 25.4 mm.

test specimens, it appears that the presence of the holes did not observably weaken the specimens in configurations Q5 and Q6.

Acoustic emission measurements were taken on five specimens to obtain qualitative information on acoustic activity prior to failure. As the specimens were loaded, the acoustic activity typically increased rapidly as the specimen loading was increased above 65% of its failure load. This is shown in Fig. 4.2-5.

A test report containing all results of the 2-D test series is now being prepared. In addition to the test results summarized above, the test report includes the results of material property tests on material left over from the specimen cutting operations. These tests were conducted for proper interpretation of the 2-D structural test results.

Initial examination of the 2-D failure load data and the Q-type specimens showed the sensitivity to flange configuration to be very weak, except in the case of Q5 specimens where the flange thickness had been increased by 5.1 mm (0.20 in.). Though differences in crack patterns were noticed among the specimens, the significance is uncertain due to the weak variation in failure loads. As the flange configuration for the ensuing full-flange test series will be taken from the Q1, Q2, Q3, Q4, or Q6 flange configurations, there did not appear to be a strong structural advantage, based on this test series, for choosing one configuration over another. Configuration Q6 was selected for the full-flange series, partially due to considerations for reducing flow maldistributions within elements.

The use of acoustic emission equipment to predict onset of failure in structural tests appears to be readily feasible. The acoustic emission equipment proved to be easily installed and used, with several methods for presenting results.

Fig. 4.2-5. Acoustic emission trace superimposed on load-deflection diagram

4.2.2.2. Full-Flange Test Series

This test series will involve the use of a 263 kN (60,000 lb.) structural test machine. During this report period, cradles to hold test specimens in the test machine were designed. Design of full-flange test elements, which incorporate complete details of the flange and adjacent coolant and fuel holes, is planned for the next report period. Designs of both control and standard fuel element test models are being completed. Testing in this series will be completed in the next report period.

4.2.3. Irradiated FSV Fuel Element Structural Test Program

The writing of a test specification was initiated for a program of static compression and impact tests on irradiated FSV elements. The planned program consists of two phases. Phase I will consist of static compression and impact tests on one-quarter-length element slices. This slice length permits the use of a standard 263 kN (60,000 lb) test machine for compression testing. Phase II will consist of static compression and impact testing on entire fuel elements. Phase II compression tests will require a large compression test machine in the 300,000 lb range. Impact testing will require specially designed pendulum test facilities.

The above tests will be conducted on identical irradiated and unirradiated elements at ambient temperature. These tests require the use of a hot cell for testing irradiated elements.

Surveys were completed of hot cells at GA, ORNL, Idaho National Engineering Laboratory, and Hanford Engineering Development Laboratory. Initial testing will be performed in the GA hot cell, starting in FY 84.

Six irradiated fuel elements have been selected from FSV segment 3 for testing under this program. The elements were identified as highly stressed elements in layers 3, 4, and 5 of the FSV core. Based on the results of the analyses that used the SURVEY/THERM and SURVEY/STRESS codes, three standard

fuel elements (13/3/5, 33/5/3, and 29/3/3) and two control elements (13/1/5 and 13/1/4) were selected for initial structural testing. All of these elements were made of H-327 graphite produced by the GLCC. In addition, test element FTE-2 will be removed from segment 3 and is slated for postirradiation fuel rod examinations under Task 1102. The graphite element, which is made of GLCC H-451 graphite, will be available for structural testing under this program. Documentation of the selection process is now in progress and will be completed in the next reporting period.

A study was completed on the use of the nondestructive examination (NDE) technique to detect cracks in graphite structures before mechanical tests are made. The present status of NDE testing of graphite was reviewed. Liquid penetrant testing, X-radiography, ultrasonic pulse-echo defect location, sonic velocity and sonic attenuation measurements, and eddy current testing have all been applied to graphite. Two methods that seem appropriate and practicable for the fuel element program were recommended: (1) alcohol wipes on the exposed surfaces to detect cracks that penetrate to the surface and (2) eddy current probing of the coolant and fuel holes with bobbin coils to detect cracked webs.

The optimum test frequency for eddy current probing was calculated to be 70 KHz for unirradiated H-327 graphite, 59 KHz for unirradiated H-451 graphite, 190 KHz for irradiated H-327 graphite, and 149 KHz for irradiated H-451 graphite. This corresponds to the highest sensitivity when using a single test coil, and gives an eddy current skin depth slightly greater than the minimum fuel element web thickness. The dimensions of test coils to operate at these frequencies appear to be compatible with the fuel element geometry. The testing could probably be done with equipment already existing at GA. A test block with both machined cracks and natural cracks would be needed to check out and calibrate the equipment.

Development of equipment and techniques for the NDE will be initiated in FY 84.

4.2.4. Tests on Corrections to Prevent Fluctuations

The scope of this work is to design, construct, and test a thermal-flow model of a column of fuel elements with and without proposed design improvements to demonstrate the effectiveness in the prevention of fluctuations. The initial tests will demonstrate fluctuations in a three-dimensional model. Subsequent tests with design improvements (e.g., lateral constraint devices at the top of the column are vertical and horizontal grooves) will demonstrate the extent to which these changes reduce or eliminate fluctuations.

The design of the test apparatus is complete. The construction of the apparatus is nearly complete. Testing has yet to be done.

The model has been designed and built to simulate a core fuel column at 1/4 scale standing in a flow channel. Air is drawn down the gap between the column and the flow channel walls, into an exit plenum and hence to a pair of electrically driven fans.

It is required that small gaps [typically 1.778 mm (0.070 in.)] be maintained accurately between the simulated fuel column and the flow channel walls over the 2377.4 mm (93.6 in.) height of the column.

The simulated fuel column is hexagonal in section and comprises eight heated and six unheated elements. The heated elements represent core fuel elements, and the unheated elements represent core reflector blocks. As in the reactor core, two of the reflector elements are the same height as the fuel elements, and four are half-height elements. The column has been designed and assembled as in the reactor core with one full-height and one half-height unheated element stacked above the eight heated elements. One full- and three half-height unheated element are stacked below the heated elements.

All elements have been fabricated very accurately from sheets of 12.7 mm (0.5-in.) thick polycarbonate resin. The elements representing the core fuel elements are filled with a silicon heat transfer fluid (HTF), and each has a centrally located electrical heater rod. The simulated core reflector elements do not have heaters or HTF. The column stands in a steel-walled flow channel.

Dowel pins are installed to restrain the relative rotational and lateral movements between all elements of the column and between the column and its support spider.

The simulated core column stands in a steel-walled flow channel. The flow channel is hexagonal in section and made from six bevelled 12.7-mm (0.5-in.) thick steel plates. The six plates are dowelled and keyed together and are held in position by the tube and bulkhead support structure. The bulkheads are made of 1.0-mm (0.04 in.) steel plate, and each has six screw brackets. Each screw bracket has two adjustment screws that bear on the steel flow channel wall plates and hold them in position. With all the flow channel walls in the fully in position, the gap between the column and flow channel is nominally 1.65-mm (0.065 in.). The walls may be spread radially (using shims between their edges) to increase the flow gap up to 6.4-mm (0.25 in.). The flow channel walls stand on the ground upper surface of the top plate of the plenum and are held in position by the support structure. The lowest bulkhead of the support structure is bolted to the top plate of the plenum.

The floor and walls of the plenum chamber are made from 12.7-mm (0.5-in.) thick steel plate, and the top is made from ground 19.05-mm (0.75-in.) thick steel plate. Two transverse screens separate the inlet flow coming from the column gap from the four exit pipes to the two fans. The screens ensure that the sink pressure at exit from the column gap is uniform. The plenum box stands on four adjustable feet for leveling purposes.

The simulated fuel column stands on a steel plate that is welded to a six-legged spider. The spider is a very stiff structure that is suspended

from the plenum top plate by six bolts that permit adjustments to the verticality of the column while a test is in progress.

The program for testing includes the following measurements:

1. Static pressures at the inlet and exit from the column gap using static probes and 1524-mm (60-in.) water manometers.
2. Air temperature into and out of the column, using mercury in glass thermometers and by thermocouples and recorder.
3. Temperature inside the uppermost simulated fuel element, using a thermocouple and recorder.
4. Airflow rate through the model, using two 33.78-mm (1.33-in.) D ASME orifices, upstream total pressure and temperature measurements, and flange tap static pressure. The pressures and $\Delta P_{\text{orifice}}$ will be measured on water manometers, the temperature by thermocouple and recorder.
5. Electrical heat supplied to column by voltmeters and ammeters.
6. The proximity of the column to the flow channel walls, by eddy current proximity probes mounted on the flow channel walls, recorded on a strip chart. The proximity probe traces will determine the period of fluctuation and/or oscillation.

Fluctuation tests are yet to be performed. They are planned to be made over a range of airflows and air temperature rise through the model, which will include the simulation of the range of operating conditions in the reactor core.

The envelope of test conditions that support column fluctuations will be determined within the capability of the test apparatus.

Any unexpected behavior of the model will be investigated and characterized.

The design and test techniques that produce fluctuations in this single column model will be used in subsequent tests to test antifructuation modifications, which include testing the effectiveness of vertical and horizontal grooves that increase the gap area by nominally 50%.

4.2.5. Two-Block Crossflow Tests

A test specification has been released that outlines the requirements for a two-block crossflow test. A portion of the test program described in this test specification will be performed this year. It will determine the effectiveness of the fuel element sealing flange in limiting crossflow between the coolant channel flow and flow in the vertical gaps between fuel columns.

The test model (see Fig. 4.2-6) will be two simulated standard fuel elements approximately twice actual size; a simulated bypass gap; an entry plenum; an exit plenum; a system for regulating the flow in the coolant channels; the ducting, containment vessel; and instrumentation required to measure and regulate flows and pressures. The tests will be performed in air at ambient conditions.

The test fixture will be hinged in the center to allow the simulation of jawed elements. One of the primary objectives of the tests will be to determine the crossflow loss coefficients as a function of jaw gap. Gaps up to 20.32 mm (0.80 in.) will be tested. The maximum gap will be greater than the maximum gap expected in the reactor as a result of irradiation distortions.

Considerable work has been performed in the design of the model and the test fixture. Fabrication is expected to begin in May 1983.

Fig. 4.2-6. Two-block crossflow test rig

Continued testing with this model is planned for FY 84. The present fuel element design includes a plenum between each element interface. The purpose of the plenum is to distribute the incoming crossflow among the coolant channels. Similarly, crossflow leaving the interface would be supplied from a high percentage of the channels. The FY 84 tests will be used to determine the effectiveness of this plenum in distributing crossflow and correcting flow maldistributions that may exist between coolant channels.

Preliminary testing was performed earlier this year using a small scoping model. The primary purpose of this testing was to evaluate the plenum concept discussed above and to establish a plenum height for the large two-block model. Figure 4.2-7 is a sketch of the test fixture. An adjustable plenum was placed between the upper and lower banks of tubes. The base of the test fixture was maintained at subatmospheric pressure. Flow entered the upper tubes and flowed through the plenum and into the lower bank of tubes. Flow rates were measured in each of the upper and lower tubes with PITOT probes.

Two types of tests were performed: flow redistribution tests and crossflow mixing tests. The flow redistribution tests were performed by blocking, or partially blocking, one or more of the upper tubes and observing the degree of flow redistribution in the lower tube bank as a function of plenum height. In the second series of tests, crossflow was simulated by directing flow into or out of one end of the plenum and observing the effect it had on the flow in the upper and lower tubes.

Typical results of the flow redistribution tests are shown in Fig. 4.2-8. This plot shows the effect of a 70% blockage of an upper tube and the correction in the lower tube as a function of plenum height. These results indicate that most of the correction can be achieved with plenum heights of approximately 12.7 mm (0.5 in.). Since the model was twice actual scale, this would be equivalent to a 6.4 mm (.25 in.) plenum in a full-scale fuel element. It should be noted that, although the plenum does not completely correct the flow maldistribution, some correction will take

Fig. 4.2-7. Crossflow proof test rig

Fig. 4.2-8. Preliminary crossflow tests: flow correction as a function of plenum height

place at each level; hence the plenum would promote uniformity as flow moves down the core.

Based on the results of the preliminary testing, a plenum height of 12.7 mm (0.5 in.) [6.4 mm (0.25 in.) full-scale] was selected for the two-block crossflow test.

Considerable data on the effect of crossflow were also collected using this scoping test model. The results of this testing are currently being evaluated.

4.2.6. Support of Fuel Element Structural Criteria

In a companion plant technology task to develop structural criteria, the use of probabilistic criteria had been proposed for the graphite fuel elements. In this period, the relative merits of probabilistic versus deterministic modeling were further evaluated. Two potential positions were identified:

1. Retain the current deterministic criteria, where the maximum graphite stresses are limited by a certain percentage of the maximum ultimate strength of the material (e.g., 70% during normal and upset operating conditions). Justify the allowable limits on a probabilistic basis by showing that fuel elements that operate with stresses at or just below the limits can still satisfy the reliability goals, from both safety and investment-risk standpoints.
2. Express the structural criteria purely on a probabilistic basis. With this approach, a fuel element design would be acceptable when the reliability goal has been met, i.e., the fuel element has a sufficiently low probability of failure.

The tentative conclusion of this evaluation is to favor position 1 because position 2 would make the design process more complex, would deviate

from established practice in structural engineering, and would be difficult to reconcile with licensing requirements.

In addition to the above evaluation, an analysis of data from earlier strength tests was performed. The purpose of this analysis was to establish Weibull parameters to be used to calculate failure probabilities. The best data were for H-451 graphite, which is the present reference material. It was found that the strength variation should properly be separated into three sources: (1) the brittle material variation (the "true" Weibull effect), (2) the log-to-log variation, and (3) the spatial variation within a log. The following approximate formula for the failure probability P_f was developed:

$$P_f = 1 - \exp - \left[\left(\frac{\sigma}{\sigma_0} \right)^m \frac{V}{V_0} \right]^{\frac{m_1}{m+m_1}},$$

where σ = the applied stress,

m and σ_0 = Weibull parameters,

m_1 = parameter expressing the log-to-log variation,

V and V_0 = actual and reference volume, respectively.

By making m and σ_0 functions of location, the systematic spatial variation can be accounted for.

5. TECHNOLOGY TRANSFER (15)

5.1. INTERNATIONAL TECHNOLOGY TRANSFER (1501)

5.1.1. Scope

Cooperative work is being performed in the Fuel, Fission Products, and Graphite Subprogram under the HTR Implementing Agreement of the U.S./FRG Umbrella Agreement for Cooperation in GCR Development.

5.1.2. Umbrella Agreement

The semiannual Subprogram Management Meeting on the Fuel, Fission Products, and Graphite Subprogram was held at GA on February 15 and 16, 1983. Representatives were present from DOE-Washington, KFA, ORNL, General Electric, and GCRA. Revision 5 of the Subprogram Plan was prepared. Table 5.1-1 shows the status of the active project work statements. The status is described more completely in Ref. 5.1-1.

5.1.3. GA/KFA Personnel Exchange

The exchange of personnel between GA and KFA under the Fuel, Fission Products, and Graphite Subprogram was completed during this period. Dr. Karl Bongartz of KFA completed a one-year assignment on fuel performance modeling in November 1982. He was transferred to another exchange task, also at GA, on the modeling of metals corrosion. Mr. David Hanson of GA completed a 15-month assignment at KFA on fission product methods verification in March 1983.

TABLE 5.1-1
PROJECT WORK STATEMENT STATUS

<u>PWS NO.</u>	<u>Title</u>	<u>Status/ Comments</u>
Fuel Development:		
FD-8	Models for Design of Coated Particles	In progress
FD-10	Characterization and QC for SiC	Being completed
FD-11	Testing Fuel Under Accident Conditions	In progress
FD-14	Gamma Exam of Fuel Elements	In progress
FD-15	Fuel Irradiation Test by HBK (R2-K13)	In progress
FD-20	Coated Particle Analysis by IMGA/PGA	In progress
FD-21	Design and Specification Data Files	In progress
FD-22	Coated Particle Process Development	In progress
Fission Product Transport:		
FP-1	Tritium and C-14 in HTR	In progress
FP-2	Fission Product Data Book	In progress
FP-3	Verification of Metallic Fission Product Codes	In progress
FD-4	Fission Product Plateout Verification (GA-HRB)	In progress
FP-5	Fission Gas Release Model	In progress
FP-6	In-Reactor Effects of Moisture on Fission Gas Release	In progress
Graphite:		
GD-1	Graphite Irradiation in HFIR-HTK	Being completed
GD-2	Irradiation of Reflector Graphite in Petten	In progress
GD-4	Graphite Creep Tests in Petten	In progress
GD-5	Structural Design Criteria for Graphite	Implemented prior draft PWS
GD-6	Corrosion Characteristics of Core Support Graphite	Reinstated for CFTL/HOVA tests

TABLE 5.1-1 (Continued)

<u>PWS NO.</u>	<u>Title</u>	<u>Status/ Comments</u>
Graphite:		
GD-9	Irradiation of Reflector Graphite in ORR	ORR tests delayed 1 year
GD-10	Oxidative Strength Loss of ASR Grades	To begin in 1983
GD-11	Characterize KFA Reflector and Support Graphite	In review
GD-12	Characterize Graphites for Purchase Spec	In progress
GD-14	Material Modeling and Fracture Criteria	Implemented prior draft PWS
GD-15	Graphite Corrosion Modeling	Implemented prior draft PWS

REFERENCE

- 5.1-1. Turner, R. F., USA/FRG Fuel, Fission Product, and Graphite Subprogram Quarterly Progress Report for the Period Ending March 31, 1983, DOE Report GA-A17100, GA Technologies Inc., to be published.

5.2. FORT ST. VRAIN CORE SURVEILLANCE (1502)

5.2.1. Scope

The scope of this task includes surveillance examination of FSV core components, analysis of FSV coolant chemistry data, and periodic examination of the FSV plateout probe.

5.2.2. FSV Fuel Surveillance

5.2.2.1. Segment 2, Nondestructive Fuel Surveillance

The nondestructive examination of 54 fuel and reflector elements from segment 2 was described in the previous program report (Ref. 5.2-1). During this report period, a topical report documenting the results of examination was issued (Ref. 5.2-2).

5.2.2.2. Segment 2, Destructive Fuel Examination

During the core 2 surveillance in April 1982 at the FSV reactor site, 54 fuel and reflector elements were metrologically and visually inspected (Ref. 5.2-7). A hairline axial crack was observed extending the entire block length in the center of face B of one element, serial number (S/N) 1-2415 (core location 08.05.F.06). Later inspection of the videotapes and photographs revealed a second element, S/N 1-2415, from core location 08.05.F.07, which also had a hairline axial crack in the center of face B.

In an effort to understand the reason for the cracks, element S/N 1-2415 was selected to undergo destructive postirradiation examination. This block was selected because it exhibited the wider crack and experienced greater fast fluence and metrological changes than the other block with a crack.

Element S/N 1-2415 consisted of a standard H-327 graphite fuel body having 210 fuel holes, 6 burnable poison holes, and 108 coolant channels.

There was no installed lumped burnable poison in the element. The element contained 3132 fuel rods consisting of (Th,U)C₂ TRISO fissile particles and ThC₂ TRISO fertile particles bonded together in a carbonaceous matrix. The test data to be gathered, test methods, and specific test objectives are summarized in Table 5.2-1.

5.2.2.3. Gamma Scan Robot

A robot to gamma scan fuel elements at the FSV site has been designed to verify nuclear codes for predicting power profiles and fission product transport prediction in an operating reactor. During this reporting period, the wiring design was completed and drafting of diagrams was started. Software for controlling high-speed data collection, on both the gamma scan and metrology robots, was brought to 90% completion.

5.2.3. FSV Coolant Chemistry and Fission Product Surveillance

A special test using the iodine monitor at FSV was performed by Public Service Company of Colorado radiochemists. This test was devised to determine whether primary coolant contamination occurs during sampling for circulating gaseous iodine. This is necessary to help resolve the discrepancy between the iodine monitor results and the plateout probe data (Ref. 5.2-3). The discrepancy was that the iodine monitor indicated 200 times more circulating iodine than did the plateout probe. Substantial contamination of the iodine monitor by primary helium coolant would explain the discrepancy.

The results of the first test showed krypton isotopes in the collection trap, indicating that primary coolant contamination of the iodine monitor did indeed occur.

Further tests are planned to determine the degree of contamination and whether the iodine monitor results are compromised by the contamination.

TABLE 5.2-1
TEST DATA, METHODS, AND OBJECTIVES OF EXAMINATION OF FSV
SEGMENT 2 FUEL ELEMENT S/N 1-2415

<u>Data</u>	<u>Method</u>	<u>Objectives</u>
A. Fuel rods		
1. Structural integrity	Visual examination	To verify structural performance and to look for signs of fuel rod-graphite interaction
2. Stack lengths	Fuel-stack-length measuring trough	To measure fuel stack shrinkage and to provide data to compare with assumptions in core design codes
3. Fuel stack pushout force	Force measuring device	To measure the extent of fuel rod-graphite block interaction and to compare with core design assumptions
4. Fission product inventories	Gamma scanning	To determine axial and radial power distributions, to determine fuel burnup, and to compare these results with nuclear design code calculations
5. Micro-structure(a)	Metallography	To verify fuel performance, no kernel migration
B. Graphite block		
1. Structural integrity	Visual examination Also using data from Ref. 5.2-2	To verify performance of H-327 graphite block
2. Thermal expansivity	Silica dilatometer	To verify axial and transverse CTE and identify any differences between analysis and measured values
3. Thermal diffusivity	Heat-pulse	To verify thermal diffusivity and to identify any discrepancies between analysis and measured values
4. Tensile strength	Extensometer	To verify axial and transverse tensile strength and to compare to analysis values

(a) The metallography is not planned under the DOE Program. However, it will be incorporated in the procedure and conducted with funds provided by the PSC.

REFERENCES

- 5.2-1. "HTGR Fuel Technology Semiannual Report for the Period Ending September 30, 1982, DOE Report GA-A16919, GA Technologies Inc., to be published.
- 5.2-2. Saurwein, J. J., "Nondestructive Examination of 54 Fuel and Reflector Elements from Fort St. Vrain Core Segment 2," DOE Report GA-A16829, October 1982.
- 5.2-3. Burnette, R. D., "Radiochemical Analysis of the First Plateout Probe from the Fort St. Vrain High-Temperature Gas-Cooled Reactor," DOE Report GA-A16764, June 1982.