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P O Box 608, San Diego, California 92112

INFORMAL AEC RESEARCH AND
DEVELOPMENT REPORT

GAMD-8743

MASTER

POSTIRRADIATION EXAMINATION OF PEACH BOTTOM FUEL ELEMENTS EO5-05 AND CO5-05, AND RELATED ANALYSES

Written Under the Direction of:

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U. S. Atomic Energy Commission
Contract AT(04-3)-314
Projects 390 and 166

March 17, 1969

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ERRATUM

GAMD-8743, POSTIRRADIATION EXAMINATION OF PEACH BOTTOM FUEL
ELEMENTS EO5-05 AND CO5-05 AND RELATED ANALYSES

Owing to an oversight, the name of a major contributor to GAMD-8743 was omitted from the cover. Please substitute the attached cover page for the one appearing on the report.

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1. INTRODUCTION

Power operation of the Peach Bottom Atomic Power Station began in January 1967. Full power was reached on May 26, and commercial operation began on June 1. Despite normal shakedown and maintenance problems plus continued test operations, the plant record for availability and capacity factor was excellent for the two extended operating periods ending respectively in January 1968 and October 1968, when shutdowns were scheduled to permit an inspection and evaluation of the nuclear steam supply equipment and components.

The primary coolant system activity of the Peach Bottom reactor up to January 1968 was less than 1 curie (C). On January 9, a step increase of about 4 C was noted and interpreted to indicate a loss of purge flow in one fuel element. However, reactor operations were not affected by this small activity increase since a level of 4225 C is permitted under the Technical Specification.

Following this modest increase in coolant activity the reactor was shut down on January 11, after 168 equivalent full power days (EFPD) of operation, for the first of the scheduled inspection programs mentioned above.

The nonpurging fuel element was subsequently found to be in location C05-05 by the failed fuel element locator (FFEL), a device which measures the purge flow in each element. On April 1, C05-05 was removed from the reactor and a circumferential crack in the sleeve was visually observed. During April, the failed element and an element that was believed to be sound but that had suspicious markings on it (E05-05) were sent to the Gulf General Atomic hot cell for postirradiation examination. The details and results of those examinations are reviewed in this report.

Reactor power operations commenced again in early May, and until the time of the next scheduled shutdown for fuel element inspection in October, after about 300 EFPD of operation, eleven new apparent sleeve failures were observed. The approximate timing of these failures is shown on Table 1.1. Two failures occurred after about 179 EFPD, and nine additional failures occurred more or less uniformly over the next 130 EFPD.

While sleeve cracks have been considered as fuel element failures, the occurrence of these cracks and the gradual increase in coolant activity did not have a significant affect on plant operations. It would be possible to operate with many tens of such failures in the core without exceeding applicable safety limits or technical specifications.

The eleven fuel elements which failed after the 168-EFPD shutdown were identified during the 300-EFPD shutdown, and these together with C05-05 are listed on Table 1.2. There is no correlation between the ordering of the lists on Tables 1.1 and 1.2.

Table 1.1
TIMING OF FUEL ELEMENT FAILURES IN CORE 1

Date (1968)	Change in Activity Level (Curies)	Core Exposure (EFPD)	Comments
1/9	4	168	C05-05
6/1	3	170	
6/2	3	170	
7/29	1	230	
8/14	1	240	
9/9	2	260	
9/14	3.5	270	
9/17	4.5	270	
10/4	3.2	280	
10/5	5.4	280	
10/22- 10/23	8	300	2 failures

NOTE: Total main loop activity as of October 23, 1968: 36.2 C.

The location of the 12 failed elements in the core is shown on Fig. 1.1. These failures occurred at radiation exposures well below the design exposures, which are (after 900 EFPD):

Average fast fluence ..... 3.1×10^{21} nvt ($E > 0.18$ Mev)
 Peak fast fluence ..... 4.7×10^{21} nvt ($E > 0.18$ Mev)
 Fissions per initial
   metal atom (average) 0.069
 Fissions per cm^3
   of compact (average) ..... 7.4×10^{19}
 Average fuel exposure ..... 60,000 Mwd/T

The average fast and thermal fluxes during the three-year life of the fuel are 4.0×10^{13} and 4.3×10^{13} n/cm²-sec, respectively.

This report is primarily a summary of the postirradiation examination of the two fuel elements E05-05 and C05-05 and an analysis of the results of the examination. The conclusions of these analyses are presented in Section 2.

Background data on the fuel element design and fabrication are given in Section 3. The irradiation data that were available to influence the design of the element are also given in Section 3. A limited amount of relevant

Table 1.2
FAILED ELEMENT INFORMATION SUMMARY

Ser. No.	Location (Type)	Time of Removal (EFPD)	Avg. Fast Fluence At Removal (10^{21} nvt)	Avg. Exposure At Removal (Mwd/T)	Radial Peaking Factor	Axial Peaking Factor	Avg. Fast Flux, $E > 0.18$ Mev (n/cm^2 -sec)
43	C05-05 (II)	168	0.55	9,800	0.85- 0.95	1.30	3.8×10^{13}
807	E02-01 (I)	300	0.89	14,800	0.800	1.30	3.4×10^{13}
805	D06-01 (II)	300	0.96	15,200	0.83	1.30	3.7×10^{13}
451	D08-04 (II)	300	1.1	19,000	1.03	1.30	4.2×10^{13}
759	E08-06 (II)	300	1.1	19,000	1.03	1.30	4.2×10^{13}
870	D10-03 (III)	300	1.1	18,100	0.98	1.30	4.4×10^{13}
159	B12-12 (II)	300	1.2	20,300	1.10	1.30	4.4×10^{13}
198	B12-02 (II)	300	1.2	20,300	1.10	1.30	4.5×10^{13}
320	A13-07 (II)	300	1.2	21,200	1.15	1.30	4.7×10^{13}
265	C14-07 (II)	300	1.2	20,800	1.13	1.30	4.6×10^{13}
151	C15-09 (II)	300	1.1	21,000	1.14	1.30	4.2×10^{13}
115	B15-09 (II)	300	1.1	21,000	1.14	1.30	4.2×10^{13}

Fig. 1.1--Failed fuel element positions

irradiation data has been obtained relatively recently, and this is presented and discussed in Section 7 with reference to the fuel element failure problem.

The action taken at Peach Bottom relative to the first fuel element failure is presented in Section 4. The postirradiation examination is described in Sections 5 and 6 and in the appendices.

In Section 7, the postirradiation examination results and site investigations are reviewed and analyzed. An analysis of the purge flow data is presented, and the results of a statistical analysis of the fuel element failure rate are reviewed. Finally, an evaluation of the behavior of the fuel element component materials under irradiation is given, and the most probable cause of the fuel element failures is presented.

2. SUMMARY

The postirradiation examination of C05-05 indicated that the sleeve fracture in C05-05 was due to the radial growth in some of the fuel compacts. Compact 18 in particular appears to have swelled tightly against the sleeve preventing axial expansion of the fuel compacts below it. The resulting combination of hoop stresses and axial tensile stresses resulted in the sleeve fracture noted on the television screen at the site. The relatively massive sleeve fracture observed in the Gulf General Atomic hot cell, as well as some of the compact fragmentation, is believed to have been caused by a preliminary examination of C05-05 at the reactor site.

The coated particles used in the Peach Bottom compacts were rather primitive in the light of today's technology, and they were probably the principal factor in the failure mechanism. It is believed that radial expansion of the compacts began with the cracking and curling of the pyrolytic carbon coatings on the fuel particles. These coatings are anisotropic and monolithic and would be expected to break from radiation-induced dimensional changes and from damage due to fission-product recoils. The broken coatings, in the process of curling and changing dimensions, would cause the compact to distort and swell. Moreover, the unrestrained metal carbide kernels would then be free to expand from fission-product-gas pressure, adding to the compact distortion. The randomly oriented microcracking observed in the fueled compact is indicative of the stresses generated in the compact by the distorted particle coatings.

The contribution of failed fuel particles to expansion of the fuel compacts was not unknown at the time of fabrication of the Peach Bottom core. In the initial design of the Peach Bottom compacts, the compact matrix was known to shrink radially under irradiation, while the fuel particles were expected to show a small volume expansion after failure. The net effect in a fueled matrix was expected to be a small shrinkage. Nevertheless some room for radial expansion of the compact was provided between the compact and sleeve, about 0.009 in. in the operating core. However, failure of C05-05 indicates that the volume expansion and distortion of the failed fuel particles was greater than expected, and that the net effect could be a small radial expansion of the compact of sufficient magnitude to cause failure of the fuel element sleeve. A growth in the radius of a compact of 0.011 to 0.012 in. in 168 EFPD would probably be enough to cause excessive hoop stresses at operating temperatures at the inner surface of the sleeve.

The measurement of the purge flow in an element has proved to be a highly effective method for identifying failed elements. However, the purge flow data have not been useful in predicting the elements most likely to fail in the future.

3. FUEL ELEMENT DESIGN, EXPECTED PERFORMANCE, AND FABRICATION

3.1. FUEL ELEMENT DESIGN

The design power of the Peach Bottom reactor (Fig. 3.1) is 115.5 Mw(t) giving an average power density of 8.3 kw/liter. The active core is approximately cylindrical in shape with an effective diameter of 9.1 ft and a height of 7.5 ft. The basic core component is the fuel element (Fig. 3.2), which has the outward appearance of a graphite cylinder 3.5 in. in diameter and 12 ft long. The core consists of 804 elements, 36 control rod guide tubes, and 19 emergency shutdown rod guide tubes all oriented vertically in a closely packed triangular array. Thirty six of the fuel elements are instrumented with two thermocouples each.

In the fuel element, graphite serves as the moderator, reflector, cladding, fuel matrix, and structure. The fuel element consists of an upper reflector section, a fuel-bearing middle section, a bottom reflector section, and an internal fission-product trap. A sleeve of low-permeability graphite contains the fuel compacts, a portion of the bottom reflector, and the fission-product trap. The sleeve is joined to the upper reflector at one end and to a bottom connector fitting at the other end. Within the fuel element, annular fuel compacts are stacked on cylindrical graphite spines as shown in Fig. 3.3. These compacts contain coated fuel particles in a graphite matrix.

The main coolant in the core passes upward through tricuspid passageways formed by the triangular packing of the cylindrical fuel elements. The fuel element pitch is designed to be 3.57 in. at operating temperature. Spacer rings machined onto the outside surface of the fuel elements at three axial locations serve to maintain the pitch and prevent line contact along the length of the elements. The tricuspid coolant channels occupy 12.8% of the core cross-sectional area where the fuel element diameter is 3.50 in. At the plane where spacer rings are in contact, the coolant channels occupy 9.5% of the core cross-sectional area.

A small fraction of the primary helium (about 1.1 lb/hr per element) enters the fuel element through a hole in the upper reflector. The gas then flows downward around the fuel compacts, sweeping fission products out of the space between the fuel compacts and the graphite sleeve. Flow of this purge gas is abetted by grooves molded in the outer surface of the fuel compacts. The purge gas then flows through the internal trap where some of the fission products either plate out or are absorbed. At the bottom end of the internal trap the gas passes through a hole in the standoff pin to a manifold in the core support plate and thence to either of two purge lines leading to the external fission-product trapping system. An inleakage of helium around the standoff connectors could contribute a maximum of 0.12 lb/hr per element to the purge flow. The design approach was not to attempt hermetic containment

Fig. 3.1--Isometric view of reactor as installed in cavity (LC-22023-A)

Fig. 3.2--Fuel element (LC-66221)

Fig. 3.3--Fuel compact assembly (LC-23294)

of fission products within the fuel elements, but rather to control the escape of these fission products and retain them in traps in such a way that the activity in the primary circuit was maintained at a satisfactorily low level.

Each fuel element rests upon its own stainless steel standoff support pin which is screwed into the core support plate. A female sealing surface within the bottom end of the fuel element slips over a mating male portion of the standoff pin, thereby providing a gas seal for restricting fission-product release as well as forming a rigid structural support for preventing lateral movement of the fuel elements relative to the core support plate.

The bottom connector and the sleeve are joined by a silicon braze, and together they form the main barrier against fission-product leakage from the fuel element. The dimensions and tolerances of the bottom connector and the sleeve are shown in Fig. 3.4. The two components are made of graphite which has a helium permeability* of 3×10^{-3} cm²/sec or less, and an effective permeability to gaseous fission products of approximately 10^{-5} cm²/sec at reactor conditions.

The screen, internal trap assembly, lower reflector piece, and fuel compacts with spines are stacked in that order within the sleeve. The weight of these components is supported by the bottom connector. The lower reflector piece is a 3-in.-long graphite cylinder made of reactor-grade graphite. The annular fuel compacts fit over cylindrical graphite spine sections. These spine sections are approximately 30 in. long and about 1-3/4 in. in diameter. There are two types of spines, one of solid graphite, and one of graphite with a 0.89-in.-diameter axial hole to contain burnable poison compacts. The screen is made of 18-8 stainless steel. Its purpose is to retain any charcoal granules that might be released from the graphite body of the internal trap.

The diametric clearance between the fuel compacts and the sleeve was 0.009 +0.007 or -0.005 in., cold, at the time of manufacture. The nominal clearance hot was 0.011 in. The fuel compact has a slightly smaller thermal expansion coefficient than the sleeve in the radial direction.†

The upper reflector assembly is a graphite piece that is threaded and cemented into the sleeve of the fuel element. The cement is a carbonaceous material that is furnace cured. The joint between the sleeve and the upper reflector does not need to be of a low permeability, because negligible fission-product concentrations exist at this location. For this reason the upper reflector is not fabricated of low-permeability graphite. The upper end of this reflector piece is machined for engagement with the fuel handling machines. A 1/4-in.-diameter hole down the centerline of the reflector serves as an inlet channel for purge gas. A porous plug cemented and retained within the upper reflector provides a controlled pressure drop for inflowing purge gas.

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*Measured at room temperature and one-half atmosphere mean pressure. These are conditions accepted within the graphite industry as standard for comparing permeability.

†Engle, G. V., High Energy Neutron-Induced Changes in Carbon and Graphite, USAEC report GA-2802, General Dynamics, General Atomic Division, September 20, 1962.

Fig. 3.4--Fuel element details (LC-23293)

The uranium and thorium within the fuel compacts are in the form of carbides uniformly dispersed as coated particles in the graphite matrix. The fuel compacts are fabricated by first warm-pressing to about 750°C then sintering at 1800°C in vacuum. There are four types of compact loadings, as shown in Table 3.1.

The fuel particles consist of uranium-thorium carbide substrates 100 to 485 μm in diameter coated with $55 \pm 10 \mu\text{m}$ of pyrolytic carbon. The total carbon within the substrate is between 11 and 16 wt-%. The uranium was initially full enriched, i.e., 93.15% U^{235} . The size distribution of the particles was selected so that the volume fraction of coated particles in the finished compacts did not exceed 30% of the total compact volume.

Table 3.1
FUEL COMPACT LOADINGS
(Loading Per 3 In. of Compact (g))

Description	Compact Type			
	A	B	C	D
	Standard	Heavy Rhodium	Light Rhodium	Heavy Thorium
Th^{232}	52.100	52.100	52.100	115.360
$\text{U}^{234\text{a}}$ (max)	0.156	0.156	0.156	0.082
U^{235}	9.700	9.700	9.700	5.140
$\text{U}^{236\text{a}}$ (max)	0.052	0.052	0.052	0.028
U^{238}	0.505	0.505	0.505	0.268
Rh^{103}	0.000	1.028	0.342	0.000
Carbon	285.000	285.000	285.000	273.000

^aThese are the maximum amounts expected in the fully enriched feed material.

The normal tolerance on loading of thorium and of uranium was $\pm 5\%$ for individual compacts. However, this tolerance was sometimes relaxed in the following two ways: (1) fuel bodies exceeding the specification on the high side, and being not greater than +8% of nominal uranium and +10% of nominal thorium loadings, were accepted if used in the lower 33 in. of the active fuel region of a fuel element and (2) fuel bodies not meeting the specification on the low side, and being not less than -8% of nominal uranium and -10% of nominal thorium loadings were accepted if used in the upper 36 in. of the active fuel region.

The tolerance on dispersion of fuel within any compact was that the thorium and uranium loading in any 10 cm^3 volume sectioned perpendicular to the axis of the compact must be within $\pm 10\%$ of the mean loading for that compact.

The tolerance of rhodium loading was $\pm 20\%$ for individual compacts.

The density of graphite matrix material in any compact (not including fuel particles) was uniform within $\pm 5\%$ for any 10 cm^3 volume and was not lower than 1.70 g/cm^3 of matrix volume.

The manufacturing tolerance on broken coatings was a maximum of 1% of the coated particles in that compact.

Cylindrical burnable poison compacts are located within the spines of some of the fuel elements. Each compact initially contained $0.436 \pm 0.030 \text{ g}$ of natural boron in the form of zirconium diboride pressed into a graphite matrix. The maximum particle size of the zirconium diboride was $100 \text{ }\mu\text{m}$.

Four types of fuel elements were required to achieve the desired power density zoning. These are defined on Table 3.2. They are loaded with the four types of fuel compacts described above. The relative location of the fuel element types within the core is shown schematically on Fig. 3.5.

Table 3.2
TYPES OF FUEL ELEMENTS BASED ON NUCLEAR PROPERTIES

Description	Fuel Element Type			
	1	2	3	4
	Heavy Rh	Light Rh	Light Rh with Burnable Poison	Heavy Th, Light U
Spine	solid graphite	solid graphite	hollow with poison	solid graphite
Compact type:				
In upper 9 in.	A	A	A	D
In middle 54 in.	B	C	C	D
In lower 27 in.	A	A	A	D
No. for nominal core loading	54	588	60	102

The total design loadings within the active core are given in Table 3.3. The tolerance on loading of thorium was $\pm 3.5\%$ and that of uranium was $\pm 2.5\%$ for any completely assembled fuel element (which was accomplished by selecting compacts), and $\pm 1\%$ for both thorium and uranium in the nominal core loading of 804 fuel elements. The tolerance for the boron content in poison spines was $\pm 5\%$ in any fuel element and $\pm 3\%$ for the 60 poisoned fuel elements specified for a core loading. The total carbon content in the 804 fuel elements constituted a core loading of 43,400 lb, based on a sleeve density of 1.90 g/cm^3 , a spine density of 1.85 g/cm^3 , and a fuel compact graphite density of 1.71 g/cm^3 of matrix. This does not include end reflectors.

The internal trap is a graphite cylinder 2.75 in. in diameter by 12 in. long machined with 16 slots, 0.13 in. wide by 0.81 in. deep, to hold the trapping

Fig. 3.5--Fuel zoning arrangement (LC-21840)

Table 3.3
NOMINAL CORE LOADINGS

<u>Element</u>	<u>Loading (kg)</u>
Th ²³²	1450.0
U ²³⁴	3.5
U ²³⁵	220.0
U ²³⁶	1.18
U ²³⁸	11.46
Rh ¹⁰³	5.00
B (natural)	1.10

material, activated carbon made from coconut shells. The heat of decay of the absorbed fission products and the sensible heat of the helium purge gas is carried radially through the graphite sleeve to the primary helium coolant. At the bottom end of the internal trap the gas stream is filtered through a porous graphite cylinder.

Each fuel element has a serial number and a loading mark engraved on the surface. The loading mark indicates the type of fuel element loading (from Table 3.3). In addition, elements which are instrumented have an additional marking.

The fuel elements are restricted from moving within the core by action of their weight, by the restraint of the standoff pins, and by clamping action of tilting side reflectors. The weight of each fuel element is 90 lb compared with a net vertical upward coolant pressure force per element of 26 lb at full-power operation. Therefore, a net downward force of 64 lb/element assures that no vertical movement can take place under any mode of operation. Lateral movement of the fuel elements is restricted at the lower end of the core by the standoff pins. The portion of the fuel element that slips over the standoff pin is 10 in. long, which is sufficient to allow each fuel element to stand vertically by itself within the core. At the operating temperature, the clearance between a fuel element and its standoff pin connector is 0.001 +0.002, -0.001 in.; this close fit allows negligible lateral motion at the lower end. At the upper end of the core, each fuel element is restrained by lateral contact forces from adjacent elements. The plane of contact is at the spacer rings located 137.9 in. above the bottom of the fuel element. Contact forces on the outer ring of fuel elements are applied by side reflectors which pivot at the lower end and are forced against the fuel elements at the upper end. The radial compressive force acting on the fuel elements by the side reflectors is equivalent to 475 lb per outermost reflector element and results from the helium coolant pressure differential across the tilting side reflectors.

3.2. EXPECTED FUEL ELEMENT PERFORMANCE

3.2.1. Power Generation

The design life of the HTGR fuel element is three years at a load factor of 80%, or about 900 days at full power. The fuel elements are designed to generate 112.7 Mw(t), which with 2.8 Mw(t) produced by nuclear heating in other reactor internals, gives a total thermal power of 115.5 Mw(t). The power generated in an average element is 140 kw(t) and peak power generated in the hottest element is about 174 kw(t), estimated to occur after about 900 effective full-power days (EFPD). Expected fuel element exposures at the end of life were listed in Section 1.

3.2.2. Expected Irradiation Effects

Changes in fuel element characteristics are expected over the design life of the core. Table 3.4 lists some of the important beginning-of-life and expected end-of-life characteristics. Changes with irradiation and temperature were predicted from tests on representative materials, and in no case did the expected change constitute a limiting condition.

The dimensional changes that occur with neutron exposure involve the over-all fuel length, the outside sleeve diameter, the compact-sleeve gap, the axial clearances between fuel compacts and reflectors, and the bowing of the elements from asymmetric dimensional changes. Capsule irradiation experiments performed by Gulf General Atomic and by Hanford up to the time of fuel element fabrication had indicated that axial and radial contraction could be expected in graphite sleeves irradiated under Peach Bottom conditions. Representative data are presented in Figs. 3.6, 3.7, 3.8, and 3.9.

Figures 3.6 and 3.7 show irradiation data on EGCR graphite, a needle-coke graphite similar to HLM-85, the sleeve material for the Peach Bottom elements. The data refer, respectively, to the longitudinal (or axial) and radial directions in the sleeve. The EGCR graphite contracted steadily with increasing fast-neutron exposure, at least at the relatively low temperatures existing in the irradiation tests.

These data together with those shown on Figs. 3.8 and 3.9 were the basis for the design of the sleeve and to some extent the fuel compact. Figures 3.8 and 3.9 were derived largely from uninstrumented capsule experiments performed at Gulf General Atomic. Data on HLM-85 graphite and NPR graphite are shown for both the radial and longitudinal (axial) directions. NPR graphite is another needle-coke graphite similar to HLM-85. As with the EGCR graphite, these materials contracted in both directions. Dimensional changes were expected to decrease in magnitude from 400° to 600°C and then increase again above 800°C. Data were obtained to exposures similar to those expected at about half the maximum expected Core 1 exposure, and extrapolations (using EGCR data) were made to full exposure.

The predicted changes in sleeve length or outside diameter, using these data, did not pose serious problems. The calculated maximum total axial contraction was less than 0.6 in. at the end of three years of operation.

Fig. 3.6--Contraction of EGCR graphite (parallel) as a function of exposure. Data pertain to axial direction of sleeve-type graphite (LC-21648)

Fig. 3.7--Contraction of EGCR graphite (transverse) as a function of exposure. Data pertain to radial direction of sleeve-type graphite

Fig. 3.8--Dimensional changes (radial) in sleeve graphite samples from capsules GA311-10 and -11 and previously obtained data on fuel matrix material (LC-21975)

Fig. 3.9--Dimensional changes (axial) in-sleeve graphite samples from capsules GA311-10 and -11 and from two Hanford capsules (LC-21914)

Table 3.4
FUEL ELEMENT CHARACTERISTICS

	<u>Beginning of life</u>	<u>Expected After 3 yr</u>
Over-all length at room temp., in.	144.00	143.45 (min)
Outside diam at room temp., in.	3.485±0.005	3.450 (min)
Diametric clearance between fuel compacts and sleeve at operating temp., in.	0.009 +0.007 -0.005	0 to 0.040
Clearance between spacer rings, at operating temp., in.	0 to 0.010	0 to 0.030
Helium permeability of sleeve, ^a cm ² /sec	3 x 10 ⁻³	7 x 10 ⁻³
Avg tensile strength of sleeve, psi	2500	2500
Minimum flexural strength of sleeve, psi	3500	3500
Tilting reflector restraining force, average lb/element	475	475
Max force between warped elements, lb	0	163
Purge gas flow rate per element, lb/hr	1.1±0.2	1.1±0.3
Max standoff seal leakage per element, lb/hr	0.1	0.1

^aMeasured at room temperature and one-half atmosphere mean pressure. Mean pressure is the average of the absolute pressure inside and outside the sleeve during measurement.

Also shown on Fig. 3.8 are data pertinent to the radial dimension of the fuel compact matrix material. Again, contraction was measured, comparable in magnitude to that observed for sleeve graphite.

Contraction data on sleeve and (unfueled) matrix material appeared to indicate a reasonable match except for the possible increased contraction of the matrix material at the peak core temperatures of about 1400°C. In the limiting case, these data indicated that the gap between the compact and the sleeve might gradually open to as much as 0.020 in.

Based on the data obtained on unfueled compacts and graphite sleeve material, closure of the gap was not expected to occur provided dimensional changes in the fuel particles did not significantly affect fueled compact dimensions. The initial nominal diametral clearance between the fuel compacts and the sleeve was 0.009 in. (cold). However, it was recognized that the presence of fuel could conceivably alter compact dimension and result in a closing of the gap, in which case sleeve failure could occur in one of three ways:

1. A sudden increase in power which would raise the compact temperature but not the sleeve temperature could lead to a thermal expansion of the compact and stress concentrations in the sleeve.
2. A continued gradual expansion of the compact against the sleeve could lead to a sleeve failure.
3. A cooling of both the compact and sleeve temperatures could lead to stress concentrations in the sleeve due to differences in the thermal expansion properties of sleeve and compact.

A study of the first possibility indicated that a fuel compact in contact with a sleeve must be rapidly heated through a step increase of 1550°F before sleeve fracture would occur due to diametral interference. (Each radial gap increment of 0.001 in. allows an additional 550°F increase in the threshold step increase.) Such changes were not expected to occur during operation and have not occurred.

The relatively unknown behavior of fueled compacts under irradiation, and the conceivability of gross compact growth in spite of the data indicating contraction of the matrix material, motivated a limited irradiation program with small fuel compact bodies. The data available at time of fuel element manufacture were not as extensive or as definitive as desired, lacking the combination of high burnup and high fast fluence, but they were interpreted to indicate that the fuel compacts were reasonably stable and well behaved. These data are summarized below.

A series of capsules containing graphite matrix fuel bodies was irradiated to determine the deformation that might occur to these bodies under thermal neutron exposure. Pertinent data are presented on Table 3.5. In reviewing these data, reference should be made to the expected average exposures in Peach Bottom given in Section 1.

The first capsule contained six fuel compacts, three produced by National Carbon and three by Gulf General Atomic. The fuel was $(\text{Th,U})\text{C}_2$, converted from the oxide in situ. The U^{235} content was 3 wt-% and the Th content was 7 wt-%. The temperature of the can containing these fuel compacts ranged from 575°C at one end of the capsule to 830°C at the other end; the fuel compact temperatures were estimated to be 260°C higher than the can temperatures. The maximum calculated burnup was 16.6% of U^{235} , which is equivalent to 4.25×10^{19} fissions/ cm^3 .

The general physical appearance of all the fuel bodies in the first capsule was excellent. The National Carbon bodies, whose fuel particle sizes were approximately 4 μm , had the greatest dimensional changes, i.e., contraction in a longitudinal and diametral direction. (The fuel particles in the Gulf General Atomic fuel bodies were 110 to 250 μm in diameter.) The contraction was attributed primarily to the fission-recoil damage to the graphite, since the total fast-neutron flux, $E > 0.18 \text{ Mev}$, was small, only about 10^{19} nvt. The average range of fission fragments in graphite is about 20 μm .

The fuel bodies for the second capsule (Table 3.5) had approximately the same outer dimensions as those for the first capsule, but were annular in shape

Table 3.5
IRRADIATION EFFECTS IN GRAPHITE BASE FUEL BODIES

Capsule No.	Fuel Compact No.	Mfr.	Fuel Particle Size (μm)	Max. Burnup (10^{19} fissions per cm^3)	Avg. Temp. of Graphite Can ($^{\circ}\text{C}$)	Calculated Fuel Body Centerline Temp. ($^{\circ}\text{C}$)	Dimensional Change (%)		Remarks
							ΔD	ΔL	
1	1	NCC	4	4.25	575	840	-1.7	-0.7	Post irradiation appearance excellent, 3 wt-% U^{235} and 7 wt-% Th as solid solution dicarbide formed from oxide conversion <u>in situ</u> .
1	2	NCC	4	4.25	690	955	-1.7	-0.7	
1	3	NCC	4	4.25	750	1150	-1.5	-1.4	
1	4	GGA	110-250	4.25	815	1205	-0.2	0	
1	5	GGA	110-250	4.25	830	1205	-0.1	-0.5	
1	6	GGA	110-250	4.25	775	1040	0	0	
2	1	GGA	250-500	6.87	650/750	1205	0	+0.2	Appearance good, 6.5 wt-% U^{235} and 15.5 wt-% Th as solid solution dicarbide.
2	2	GGA	110-250	6.87	730/840	1315	-0.2	+0.2	
2	3	GGA	<50	6.87	840/900	1400	-1.7	+2.2	
2	4	GGA	<50	6.87	900/925	1480	-2.2	+3.0	
2	5	GGA	110-250	6.87	760/840	1210	-0.3	-0.1	
2	6	GGA	250-500	6.87	705/790	1150	0	+0.1	
3	1	GGA	<50	6.01	750/760	1095	-0.9	+3.0	Appearance good.
3	2	GGA	250-500	6.01	705/790	1290	0	+0.3	Appearance good.
3	3	GGA	<50	6.01	790/870	1370	-0.8	+0.8	Surface roughened.
3	4	GGA	110-250	6.01	870/925	1480	-0.1	+0.1	Appearance good.
3	5	GGA	250-500	6.01	790/840	1230	0	+0.1	Slight longitudinal crack.
3	6	GGA	110-250	6.01	730/815	1175	0	+0.1	

NOTE: All group-3 bodies contained 22 wt-% U, Th as dicarbides converted from silicide in the ratio of Th/U = 2.5. Experimental error in measurements: ± 0.0001 in. or $\pm 0.01\%$. Fast-neutron exposure: approximately 10^{19} nvt ($E > 0.18$ Mev).

and contained 22 wt-% metal (6.5% U^{235} , 15.5% Th) as $(Th,U)C_2$. The burnup attained was 18%, or 6.87×10^{19} fissions/cm³. The diameter of these bodies contracted while the longitudinal dimension increased. Furthermore, the largest variations occurred in compacts whose fuel particles were less than 50 μm in diameter. The results indicated that the minimum size of particles should be in the range of 110 to 250 μm in order to minimize the change in dimensions caused by fission-fragment recoils.

The spines of the fuel elements, which were made from graphite of nuclear purity and low permeability, suffered no physical degradation. Melt wires contained in the spines indicated that the temperature ranged up through the melting point of platinum (1774°C).

The fuel bodies used in the third capsule were approximately identical in composition and size to those in the second capsule, except that they were prepared from silicide fuels. Conversion of the silicide to dicarbide was accomplished by sintering. The exposure of these fuel bodies was 6×10^{19} fissions/cm³, and the calculated centerline temperature ranged up to 1480°C. These fuel bodies contained 22 wt-% of combined uranium-thorium with a Th/U ratio of 2.5. The particle sizes ranged from less than 50 to 500 μm . Except for the surface roughening of compact 3 and a crack in compact 5, the appearance of the remaining compacts of this capsule was comparable to that prior to irradiation. One of the bodies with small fuel particles was placed in the cold region of the capsule and another in a hot region. The contractive effect was greater in both cases than for the bodies with large fuel particles.

The data obtained from these capsules would indicate that, from the standpoint of uranium burnup, the compacts were generally acceptable in terms of dimensional and physical integrity. While the dimensional stability was affected by fuel particle size, it seemed to be somewhat temperature insensitive, at least at these low fast-neutron exposures. The burnups for these capsules are in the range of 60% to 75% of that expected for Peach Bottom fuel. The starting material, whether oxide or silicide, performed satisfactorily. It must be emphasized, however, that the fast fluence was only about 10^{19} nvt, considerably lower than that expected in the reactor.

In a second series of tests, sample compacts containing coated fuel particles were irradiated in six purge capsules (GA309-4 through -9). These capsules were primarily designed to study the release of gaseous fission products from graphite-matrix fuel compacts at elevated temperatures. Pertinent exposure characteristics are shown in Table 3.6.

All of the compacts were annular, 1 in. outside diameter, 1/2 in. inside diameter, and 1.75 in. long, and contained 18 to 24 wt-% metal in the fuel body. The fuel used in capsules GA309-4, -5, and -6 was some of the first pyrolytic carbon-coated carbide fuel produced. Particle densities were low, and coatings were thin (25 μm). The compacts in the other three capsules were prepared with improved coated particles. The coating thickness in capsules GA309-8 and -9 was 50 to 60 μm , while that in GA309-7 was 10 to 15 μm to simulate possible end-of-life conditions. All six capsules were exposed to a thermal flux of 4.0 to 4.8×10^{13} neutrons/cm²-sec.

Table 3.6
IRRADIATION DATA FOR SIX CAPSULES

Capsule	Fuel Temp (°C)	Time Above 1400°C (hr)	Exposure (nvt)	
			Thermal	Fast
GA309-4	850-1700	1760	5.7 to 6.9x10 ²⁰	3.8 to 4.6x10 ¹⁹
GA309-5	1000-1950	2430	5.7 to 6.9x10 ²⁰	3.8 to 4.6x10 ¹⁹
GA309-6	950-1950	2180	5.7 to 6.9x10 ²⁰	3.8 to 4.6x10 ¹⁹
GA309-7	1000-1700	820	3.9 to 4.7x10 ²⁰	2.6 to 3.1x10 ¹⁹
GA309-8	1200-2050	1460	3.9 to 4.7x10 ²⁰	2.6 to 3.1x10 ¹⁹
GA309-9	1250-1600	1210	3.9 to 4.7x10 ²⁰	2.6 to 3.1x10 ¹⁹

When the capsules were being removed from the test reactor, the gas-sampling lines were not properly sealed off, and water entered all the capsules. Postirradiation examination showed that the compacts had suffered some deterioration typical of the effects caused by moisture reacting with (Th,U)C₂ particles with broken coatings. It could be observed under microscopic examination that some of the particles in capsules GA309-4, -5, and -6 (early coated particles) had coatings that had deteriorated. The majority of the central regions of the compacts from GA309-8 and -9 (improved coated particles) were intact and in good condition; only 4% broken particles were counted on a polished cross section of GA309-8, and only 15% on GA309-9.

In addition to the above data on the irradiation stability of fuel compacts, some data from the GAIL irradiation loop in the GETR were available to influence the fuel element design. In the GAIL III-A element, 15 compacts similar in dimension to the Peach Bottom compacts were irradiated to approximately 20% of the Peach Bottom exposure. Examination of several of these compacts revealed no changes in compact dimensions.

A second GAIL element, GAIL III-B, was later irradiated in the GETR for a longer period of time, but as will be discussed in Section 7, little information on compact diameters was obtainable from it.

This body of data appeared to show a possibility for axial expansion of the compacts but relatively little possibility for significant radial expansion at low fast fluences. To accommodate axial growth, a 5.3-in. expansion space was provided within the sleeve at the top of the column of fuel compacts, adequate for a 6% growth in the column of compacts.

The bowing conditions in the fuel elements were studied in detail. Exposure to fast neutrons will result in dimensional changes in all types of nuclear grade graphite. At operating temperatures the change was expected to be a contraction rather than an expansion. In regions of fast-flux gradients, differential contraction strains result and, if unrestrained, the elements would take a bowed shape. Regional thermal gradients also have the

same effect. The results of the analysis indicated that the fast-neutron and thermally induced bowing stresses were within the graphite strength limits.

The most severe fast-flux and thermal gradient conditions exist at the edge of the reactor core. Fuel elements and side reflector hexagonal elements were analyzed by considering the elements to be cantilevered beams fixed at the standoff pin and subjected to concentrated loads at the spacer locations. The input data to the analysis included axial temperature profiles in the sleeve wall at two locations 180° apart, fast-flux gradients across the element at different axial locations, and the spacer force locations. The output included the unrestrained deflection, the forces at spacer locations under restrained conditions, and the flexural stress in the sleeve. The results of the bowing analyses are presented in Table 3.7. The maximum calculated bending stress was 1587 psi, and occurred in a core-reflector interface element at the end of core life. Added to this, on the tension side of the wall, was a 414-psi local thermal stress. The combined maximum stress, 2001 psi, was similar to a flexural stress condition. No relaxation from creep was considered in arriving at the sleeve stress values. Available data on creep in graphite indicated that some relaxation would take place. A creep rate of 7×10^{-11} (in./in.)/psi-hr, as observed by Greenstreet at ORNL*, would result in the 1587-psi stress due to bending being reduced to less than 700 psi. Assuming the local thermal stress distribution superimposed on the bending stress remained constant, the maximum combined stress would become approximately 1100 psi. The flexural strength of the sleeve graphite is 3500 psi.

Interaction forces between individual core elements due to restraint of nonuniform deformation are balanced by the circumferential restraint of the tilting side reflectors. An analysis was made for the forces exerted by all bowed fuel and reflector elements at the end of the three-year core life. Distribution and direction of these forces are shown in Fig. 3.10. The interaction forces from these elements induce a net tipping moment outward (the pivot point is at the standoff pin) of 4.37×10^4 in.-lb per outermost reflector element. The net inward moment due to a pressure differential of 2.7 psi across the upper end of the tilting side reflectors induces an inward moment of 6.55×10^4 in.-lb per circumferential element width. Thus, the shape of the core is not expected to be altered by bowed elements at the end of core life.

The bottom connector of the fuel element, which fits closely over the standoff support, is expected to maintain its dimensions throughout the core life. This component operates at a temperature of 700°F within a region of the bottom reflector where the fast flux is reduced to about 1×10^{11} n/cm²-sec. The total contraction of the bottom connector should be less than 0.1% in three years. A contraction of 0.1% would reduce the radial gap between the connector and the standoff pin by only 0.0002 in. This reduction would cause no adverse effects on the connector.

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* R. E. Nightingale, Hanford Newsletter, No. 10 (HW-76446).

Table 3.7

FUEL ELEMENT AND REFLECTOR ELEMENT BOWING ANALYSIS

Case	Description	Unrestrained Max. Deflection Top of Element (in.)	Max. Restraint Flexural Stress In Sleeve (psi)	Location of Max. Stress (in. above bottom)	Max. Restraint Force at Spacer Rings (lb)
I	Fuel element at reflector interface. Thermal gradient only (new element).	1.02	857	92	89
II	As in I except 900-day end of core life fast-flux exposure.	3.03	1587	54	135
III	As in II except reactor shutdown. Residual neutron damage.	2.01	1351	54	163
IV	Fuel element adjacent to inserted control rod. Thermal gradient only (new element).	0.93	646	92	59
V	As in IV except 400-day exposure to fast-flux gradient.	1.74	864	54	60

Table 3.7 (continued)

Case	Description	Unrestrained Max. Deflection Top of Element (in.)	Max. Restraint Flexural Stress In Sleeve (psi)	Location of Max. Stress (in. above bottom)	Max. Restraint Force at Spacer Rings (lb)
VI	As in V except reactor shutdown. Residual neutron damage.	0.81	559	54	64
VII	First reflector row. New element. Thermal gradient only.	-0.60	630	92	154
VIII	As in VII except end of adjacent fuel life fast-neutron exposure. 900 days.	1.59	1069	54	287
IX	As in VIII except reactor shut down. (Residual neutron damage)	2.18	1156	54	221
X	Second row reflector. Restraint imposed at 12-in. axial increments. New element.	-0.29	368	132	284
XI	As in X except end of adjacent core life exposure. 900 days.	0.51	794	36	408

- A - FORCES SHOWN ARE THOSE EXERTED BY ELEMENT WHEN RESTRAINED AT THE FORCE LOCATION, (LB)
- B - DEFLECTIONS & FORCES EVALUATED FOR ELEMENTS AFTER 900 DAYS FULL POWER OPERATION.
- C - UNRESTRAINED SHAPE SHOWN. (EXPANDED SCALE) NO CREEP RELAXATION.

Fig. 3.10--Core element interaction restraint forces (LC-21477)

3.3. FABRICATION

3.3.1. Fuel Element Components

All fuel elements (Fig. 3.2) contain the following basic components and materials:

1. Graphite sleeve.
2. Graphite bottom connector.
3. Silicon braze ring.
4. Stainless steel screen.
5. Internal trap assembly, containing:
  - a. Porous graphite filler.
  - b. Graphite pin.
  - c. Internal trap (graphite).
  - d. Four graphite cloth washers.
  - e. Activated charcoal (about 130 g).
  - f. Graphite trap nut.
6. Lower reflector.
7. Three graphite spines.
8. Thirty fuel compacts.
9. Upper reflector assembly, containing:
  - a. Graphite upper reflector.
  - b. Porous graphite filter.
  - c. Graphite retaining ring.
10. C-6 cement.

3.3.2. Fuel Compacts

The nominal fresh fuel compact is about 3 in. long, 2.743 in. in outside diameter, and 1.750 in. in inside diameter. The compact is beveled on both ends and contains 16 equispaced grooves, 0.054 in. deep by 0.108 in. wide, running its length. The compacts were made by hot-pressing a mixture of fuel particles, graphite powder, and a pitch/ethylcellulose binder. The detailed procedure is given in Appendix A.

Many of the compacts were stored for several months or more in 3-in.-diameter metal tubes sealed with polyethylene bags. Other compacts were stored in metal trays sealed in polyethylene bags. If compacts are stored in a normal atmospheric environment, and if some particle coatings are broken, the carbide will be converted to an oxide and a weight loss will occur. It was found that in a period of eight weeks, after a complete vacuuming of a compact, the weight loss on the average was only 0.15%. Checks after storage in the vault for several months showed there was no detectable change in compact dimensions.

3.3.3. Assembly Data

The sleeves were dimensionally inspected at Gulf General Atomic using full-length, plug-type air gaging for the inside diameter and a micrometer for the outside diameter. They were also ultrasonically inspected for internal defects. Based on a 275-psi internal bladder pressure test done at Great Lakes Carbon, the sleeve hoop strength was proof-tested to be greater than 1170 psi.

A ring gage and a plug gage were used to check the outside and inside diameters of compacts just prior to assembly. In some instances, compacts were sanded or machined in order to meet dimensional tolerances. This fabrication procedure was accepted following studies which showed that after vacuuming of the oxidized machined surface, the compact was dimensionally stable. The compacts were assembled on a spine prior to insertion into a sleeve. The space between the compact and the top end of the sleeve was determined by actual measurement at the time of the assembly.

A standard element assembly check list was made out for each element.

The total compact length and fuel loadings for elements E05-05 and C05-05 are shown on Table 3.8.

Table 3.8

FUEL ELEMENT COMPACT LENGTH AND URANIUM DATA

Reactor position	C05-05	E05-05
Serial number	43	46
Total compact length, in. ..	88-3/4	88-11/16
Total uranium loading, g ...	311.87	311.59

3.3.4. Fuel Element Component Properties

Some of the properties of the fuel element components are listed in Table 3.9. All graphite components in the element were graphitized at 2800°C. The total thermal neutron absorption cross-section equivalent of the graphite impurities was less than 5 ppm natural B by weight.

3.3.5. Flow Data for Fuel Elements

All fuel elements were flow-tested after assembly. The flow through the element had to be maintained between 3 and 6 ft³/hr at a ΔP of 11 in. of water. The flow for element 43 (C05-05) was 4.7 ft³/hr; that for element 46 (E05-05) was 5.2 ft³/hr.

Table 3.9
FUEL ELEMENT COMPONENT PROPERTIES

	Sleeve	Compact
Avg permeability	1×10^{-3} cm ² /sec at 1/2 atm ΔP	---
Radial tensile strength	1600 psi	---
Axial tensile strength	2000 to 3000 psi	---
Radial coefficient of expansion	4.5×10^{-6} in./in.-°C	$\sim 2.2 \times 10^{-6}$ in./in.-°C
Axial coefficient of expansion	2.9×10^{-6} in./in.-°C	$\sim 7.5 \times 10^{-6}$ in./in.-°C
Thermal conductivity	25 Btu/hr-ft-°F at 1200°C	22 Btu/hr-ft-°F at 1400°C
Density, g/cm ³	1.94	2.11
Final heat treatment temperature, °C	2800	1800

3.4. SHIPPING AND STORAGE

After the elements were assembled and bagged in plastic, they were stored in steel shipping containers that held 25 elements. When these containers were sealed shut, they were partially evacuated and back-filled with nitrogen that is estimated to have contained 20 ppm water. About 10% air still remained in the shipping container when the evacuation and filling with nitrogen was completed.

The elements were shipped to Peach Bottom by truck after it had been determined by a test shipment of 25 elements to Salt Lake City, Utah, and return, that the method of shipment would not damage or change the dimensions of the fuel compacts or any of the fuel element components.

3.5. INSPECTION AT PEACH BOTTOM

When the elements arrived at Peach Bottom, they were inspected by Philadelphia Electric and Gulf General Atomic personnel for chips and cracks, bowing, dimensional changes, and the condition of the bottom and top of the element. The records indicate that C05-05 and E05-05 were in good condition at the time of this inspection. When this inspection was completed, the elements were again stored in nitrogen.

Prior to loading into the reactor, the elements were again inspected and the record indicates that both elements were acceptable.

4. INVESTIGATION AT PEACH BOTTOM

4.1. PRESHUTDOWN OPERATIONS

On January 9, 1968 the reactor was operating at full power with an indicated primary coolant activity of 330 mC. At approximately 10:20 PM, a sharp increase in primary coolant activity was indicated on the Cary ionization chamber. Shortly thereafter, the area monitor at the No. 1 dust collector alarmed because of an increase from 2 to 10 mr/hr; the No. 2 dust collector increased from 20 to 40 mr/hr. Other plant parameters remained normal; no increase in stack release was noted, nor did general radiation levels around the plant increase. A survey of the air room did not reveal any unusual radiation levels.

In an effort to determine the cause of the increased activity, the liquid-nitrogen-cooled traps were bypassed at 11:40 PM without a noticeable effect on the activity level. They were returned to service at 1:15 AM, again without effect. Primary coolant samples were also taken. Activity levels of the samples confirmed the Cary reading, while a gamma spectrometer indicated the presence of short-lived fission products not normally present in detectable quantities. Samples of the effluents from the low-temperature delay bed and the liquid-nitrogen-cooled system showed normal activity values. At 2:00 AM, on January 10 the total primary system activity level appeared to be stabilizing at about 4 C. The reactor power was decreased by 2%. Subsequent changes in activity level were minor, indicating that an equilibrium had been reached.

Reactor operation was continued until January 11, 1968 without further incident. During this period, additional data were collected, and radio-chemistry samples were taken to determine the cause of the increase.

The following operations were also performed during this period:

1. The fuel element purge was transferred from line A to line B. No effect was noted.
2. An increase in purge flow was obtained by reducing the steam generator baffle flow. This did not result in any change.
3. A second purified helium compressor was started to increase the purge flow and cause a decrease in primary coolant activity, but the change was only consistent with increased purification system cleanup rate for the new flow. It had been hoped that if a reduced purge flow through one element was the cause of the increase in activity, then increasing the total purge flow might decrease the release to the main coolant system or even dislodge the material causing the blockage.

On January 11, 1968 at 5:00 PM, plant shutdown was commenced.

Radiochemistry data indicated that release from the core, rather than penetration of the trapping system was responsible for the main coolant activity increase. The presence of short-lived Kr^{89} (3.2m) and Xe^{138} (17m) in the main loop grab samples had not been detected before during normal operation. The absence of high activity in the effluents of the low-temperature delay bed and the liquid-nitrogen-cooled traps indicated that these systems were not contributing to the main loop activity. Pertinent data are shown on Table 4.1. Radiation surveys in the transfer compressor compartments did not show an increase at the time of the excursion, thus eliminating the low-pressure tank as the source.

A calculation was made to determine the expected equilibrium activity in the main loop if all the purge flow of one element entered the main coolant directly. The results of this are shown in Table 4.2; nuclide half lives are included for convenience.

The 6 C expected approximates the 4 C measured on the Cary ion chamber, especially if some subtraction for decay is allowed on the Kr^{89} and Xe^{138} contributions to the total. It was therefore tentatively concluded that one fuel element had lost its purge flow. Comparison of the time-dependent measured main loop activity with Fig. II-45 of the Peach Bottom Final Hazards Summary Report, which shows the predicted activity ratios for failed fuel elements, also favored this conclusion.

4.2. FAILED FUEL ELEMENT LOCATION

The failed fuel element locator (FFEL) consists of a heated-thermopile gas flowmeter contained in the grapple head of the fuel transfer machine. When seated on the fuel element grapppling knob, a reading of the purge flow rate into or out of each fuel element can be made over a wide range. The initial FFEL test of the newly loaded core in May, 1966 (GAMD-7360) demonstrated that, as designed, variations of purge flow in individual fuel elements could be measured, and improperly purging elements could be readily located.

Beginning on February 10, 1968 the transfer machine FFEL head was utilized to determine the purge flow through each fuel element. All 804 elements were checked by February 19.

The initial FFEL operations were run with one main helium circulator operating at 1000 rpm and with a ΔP between the reactor and the exit of the water-cooled delay beds of 0.6 psi. During these operations, the normal elements had purge flows between 1.4 and 4.8 on the high scale of the flowmeter (10 indicates a flow of 3000 cm^3/min). Under these conditions, the element at location C05-05 exhibited zero purge flow. When all 804 elements had been tested, the plant conditions were changed to 3000 rpm on the main circulator and 0.3 psi between the reactor and the exit of the water-cooled delay beds, these conditions being the same as those existing during the initial FFEL measurements.

Selected elements exhibiting low flow, high flow, and normal flow were tested. The correlation of the flows between the two FFEL operations was

Table 4.1
KRYPTON AND XENON CONCENTRATIONS
($\mu\text{C}/\text{std cm}^3$)

Sample	Date	Kr ^{85m}	Kr ⁸⁸	Kr ⁸⁷	Kr ⁸⁹	Xe ¹³⁸	Xe ¹³⁵	Remarks
Main loop	11/21/67	6.6x10 ⁻⁶	7.7x10 ⁻⁶	3.8x10 ⁻⁶	---	---	3.1x10 ⁻⁶	Typical analysis
Main loop	1/10/68	4.9x10 ⁻⁴	3.2x10 ⁻⁴	3.0x10 ⁻⁴	7.6x10 ⁻⁴	5.8x10 ⁻⁵	2.7x10 ⁻⁴	
Fuel element purge stream	12/27/67	0.35	0.75	0.87	11.0	3.4	0.23	Typical analysis
Fuel element purge stream	1/10/68	0.24	0.45	0.51	8.8	1.7	0.13	
Low-temperature delay bed exit	1/10/68			No significant activity				
Liquid-nitrogen- cooled trap exit	1/10/68			No significant activity				

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Table 4.2
LOOP ACTIVITY IF PURGE FLOW FROM ONE ELEMENT ENTERED COOLANT

	Kr ^{85m}	Kr ⁸⁸	Kr ⁸⁷	Kr ⁸⁹	Xe ¹³⁸	Xe ¹³⁵	Total
Activity at equilibrium, C	0.58	0.98	0.99	1.7	1.36	0.41	6.0
Half life	4.4h	2.8h	1.3h	3.2h	17m	9.2h	---

good, except for C05-05. This element, which had exhibited zero flow previously, exhibited an upward flow of 0.36 on the high scale. A test was run with zero purge line flow to determine the backflow rate that would be obtained from a normal element. The backflow of a normal element was found to be about 0.03, or about 10% of that shown by the failed element. The backflow through a good element when lifted off the standoff pin was measured to be about 0.7, or twice the failed element flow on the standoff pin. The results of these tests indicated that every element except C05-05 had an acceptable amount of purge flow in the proper direction. C05-05 showed a slight amount of flow in the improper direction, indicating a possible crack in the sleeve.

An analysis of all purge flow data is presented in Section 7. It was originally hoped that sets of purge flow data could be correlated to indicate potentially troublesome fuel elements. As described in Section 7.1, however, this did not prove possible.

4.3. REMOVAL PROCEDURES

Since all test data indicated a crack in the fuel element sleeve, the element could not be removed with the normal fuel handling machinery, which lifts the element by a fuel handling knob at the top and depends on the integrity of the upper reflector, sleeve, and lower reflector to support the remainder of the element. If a crack in the sleeve existed, handling this element in a normal manner could result in removing only a part of the element. It was therefore decided to use the broken fuel element removal tool, which was available and had been successfully tested in the autoclave reactor mockup at Gulf General Atomic in 1966.

The broken element removal tool consisted of a cylindrical sleeve made of 35-mil-wall stainless steel having a diameter slightly larger than that of a fuel element. The cylinder was provided with a normal fuel handling knob at the top and with spring loaded fingers at the bottom. The six springs and fingers are contained in housings outside the tubular sleeve and are oriented to fit into the tricusps. The fingers are pushed into their housings during insertion over an element, and spring outward into the engaged position when the removal tool is fully inserted. When the tool is then raised, the fingers engage the tapered portion of the fuel element bottom connector and lift the element.

The use of the broken element removal tool for the removal of C05-05 was somewhat complicated by the adjacent control rod guide tube, which had a graphite key in one of the six tricusps surrounding C05-05. A segment of the core was subsequently mocked up utilizing a core plate, standoff pins, and an array of full-size dummy fuel elements. The control rod guide tube with the graphite key was also simulated. All dimensions of the mockup were made the same as the actual core. With this mockup, a test program was run to determine the hazards and problems associated with the removal operation. As a result of this program and a study of all as-built dimensions and tolerances, several modifications were made to the broken element removal tool. These included slotting the lower half of the tool to increase its flexibility

and reducing the wall thickness near the bottom where the tool passed between C05-05 and the neighboring elements or the control rod guide tube. The tests also indicated the necessity of rotating the key of the guide tube away from C05-05 to permit insertion of the removal tool without developing horizontal stress in the guide tube at the guide tube nut.

To permit a partial on-site inspection of the failed fuel element, special tools and procedures had to be developed. An on-site inspection was desired by Philadelphia Electric Company to determine the condition of the element at the earliest possible time, before exposing the element to possible damage during shipment to the Gulf General Atomic hot cell for examination. To accomplish this inspection, the following items were procured or fabricated, installed, and tested prior to removal of C05-05 from the core:

1. Release Tool for Broken Element Removal Tool. This tool was designed to release the spring-loaded fingers of the broken element removal tool in the spent fuel can in the canning machine. With the fingers released, the broken element removal tool could then be pulled over the element and be completely removed. This tool was fabricated and tested numerous times prior to actual use.
2. Shielding Spool. A shield pipespool piece was fabricated and tested in position above the canning machine isolation valve. Use of this shielding permitted manual removal of the broken element removal tool via a pulley arrangement to a modified manual grapple.
3. Manual Grapple Modification. The new fuel element manual grapple was modified with remote operation cables and a jaw locking mechanism to permit operation of the grapple from outside the shielding spool and to assure positive jaw closure.
4. Spent Fuel Can Lip. A removable lip was fabricated to protect the upper sealing surface of the spent fuel can during most of the operations at the canning machine. A tool for installing and removing the lip with an element in the spent fuel can was also provided.
5. Canning Machine Television. A television viewing port and additional lighting and mirrors were installed in the canning machine to permit observation of about 80% of the circumference of the element as it passed by the television camera situated outside the canning machine cavity.
6. Element Rotation Tool. A tool was fabricated that would reach through lead shielding and the canning machine isolation valve and rotate the element if more detailed viewing was desirable.
7. Padded Spent Fuel Can Liners. Split liners with asbestos padding were made to protect the element from further damage during off-site shipment. These liners were inserted over a dummy element in a spent fuel can by remote manual means for test purposes prior to use on the actual fuel element.

All of these items were fabricated and tested individually. A final practice run of all handling external to the reactor was then completed. All necessary changes were made to the procedure and equipment prior to handling of C05-05.

4.4. ELEMENT REMOVAL AND INSPECTION

On March 29 and 30, five hexagonal reflectors were removed from the reactor and placed in dry temporary storage on the refueling floor. The maximum radiation level on contact with the center of the hexagonal reflector was approximately 3.5 r/hr. Temporary shielding was placed near the storage location to reduce radiation levels in the vicinity.

On March 30, 1968 five elements from the circle of elements one row removed from C05-05 were removed from the core and temporarily placed in the vacant holes in the core or reflector regions. This gave the elements immediately surrounding C05-05 additional freedom as the broken element removal tool was inserted. Element D05-01, immediately next to C05-05, was then removed and temporarily placed in a vacant reflector position. The control rod guide tube was rotated 30° clockwise to clear the tricusps and permit insertion of the removal tool. Element D05-01 was left in the reflector position for the duration of the removal operation.

Before actually removing C05-05 with the broken element removal tool, the tool was inserted over C05-05 with the gripping fingers locked in a retracted position. The operation went very smoothly and very little force was required to insert or remove the tool. This gave assurance that the tool would fit down over the entire element and be able to grip the bottom of the element on removal.

On April 1, 1968 the broken element removal tool with its fingers released was inserted over C05-05 with the transfer machine. The tool succeeded in gripping the element and transporting it intact to a special canister in the parking hole. This canister containing the element and removal tool was then transported to the pickup cell by the charge machine.

At the pickup cell location, the canister was released and the removal tool grappled. The element inside the tool was then gamma-scanned as it was raised into the charge machine.*

From the pickup cell, the tool containing the element was taken to the canning machine and lowered into a spent fuel can containing the broken element removal tool release collar. The bottom of the element protruding from the tool was observed via the television system installed in the canning machine during this operation. This verified that the entire fuel element was contained in the removal tool. The shielded spool was then installed over the canning machine, the isolation valve was opened, and the remote-manual operated grapple was lowered into the canning machine. Radiation levels were less than 50 mr/hr in all areas during this operation. The grapple was then attached to the removal tool, and the tool was successfully pulled over the element out of the canning machine.

Following this, the charge machine was returned to the canning machine location to grapple the bare element to lift it for inspection via the television system. On raising the element, the weight indication appeared slightly

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* See Section 6.2 for gamma-scan data.

low. At approximately 62 in. from the top of the element, a 3- to 4-in. jagged vertical crack appeared in the sleeve of the element, a picture of which is shown in Fig. 6.30. At a distance of 71 in. from the top of the element, there appeared a jagged circumferential crack which completely severed the element. A picture of this is shown in Fig. 4.1. The center spine of the element was seen protruding from the top portion of the sleeve. The bottom of the center spine was not seen. At this point, the decision was made to terminate the inspection. Weight readings taken on the charge machine hoist load cells indicated that the fuel compacts were still in the upper portion of the sleeve at this time.

Fig. 4.1--Fuel element C05-05 as seen by the canning machine television. Note protruding spine and mirror view at left side

The element was then reinserted into the spent fuel canister and released, but the top was found to be some 2 to 3 in. higher than before it was lifted. Since this would prevent the canister cap from being installed, the element rotation tool was used. It was manipulated through the canning machine valve, and after several attempts, the element top and the tool dropped a few inches. Padded liners were placed into the canister by remote-manual means to protect the element from further damage during shipment. The spent fuel can cap was then swaged closed and leak-tested, and the assembly was lowered into the spent fuel pit. During these operations, the fuel element was exposed to air with some moisture for about one to three hours.

4.5. GAMMA SCAN

The equipment and methods used for the fuel element gamma scans at Peach Bottom are fully described in GAMD-8703.* A collimated detector is used to measure gross gamma count rate from the fuel element as it is moved vertically in the charge machine. Both the count rate and the element position are recorded simultaneously on a strip chart recorder, giving an accurate measure of fuel compact stack height. With properly adjusted electronics and collimators, individual compacts can be seen on the gamma trace.

The original scan of the element from position C05-05 at Peach Bottom is shown in Fig. 6.12. Compacts 10, 13, 14, and 19 (as numbered from the bottom of the element) showed above-normal gamma peaks. The original fuel column height had changed from 88.75 to 89.75 in., or a growth of 1.0 ± 0.10 in. This was the largest column growth recorded of the 18 fuel elements measured at this shutdown. Except for the E05-05 element which measured 0.1 in., all others showed 0.3 to 0.7 in. of growth.

4.6. E05-05

After the removal of element C05-05 from the reactor, fourteen additional elements were gamma-scanned and visually inspected between April 4 and 11, 1968. All of these elements appeared normal, with the exception of the element from location E05-05, which showed a very small fuel compact stack growth (~0.1 in.) and a surface marking that could possibly be a crack. This element was removed from the core, canned, and shipped to San Diego for closer examination in the hot cell.

The details of the investigations and work performed at the site on E05-05 will not be described here since they were comparatively minimal and not too pertinent to the main theme of this report.

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* W. J. Scheffel, J. R. Brown, B. F. Disselhorst, and W. P. Wallace, Peach Bottom 150 Full-power Day Core Examinations (Results of Peach Bottom Postconstruction R & D Test Procedures, JM-2, JM-3, JM-4, and JM-6), USAEC report GAMD-8703, Gulf General Atomic, December 31, 1968.

5. GULF GENERAL ATOMIC HOT-CELL PREPARATION FOR PEACH BOTTOM

COMPONENT EXAMINATION

5.1. CELL PREPARATION

The design and construction of the Gulf General Atomic hot cell facility is directed toward the remote examination of high-temperature graphite reactor fuels. The facility contains three different cells. The high-level cell, capable of handling 1,000,000 C of activity, is 18.08 ft long and 8 ft deep with three operating stations. The low-level cell, capable of handling 10,000 C, is 8.92 ft long and 8.5 ft deep. These two cells are adjacent, with both the low-level and high-level cell doors opening into the back wall. The cells are in line with the service gallery and loading dock to facilitate handling fuel elements, like the Peach Bottom elements, which are 12 ft long. The third cell is a 10,000 C metallography cell.

Specific and detailed planning for the postirradiation examination of a Peach Bottom element was initiated in 1967. The type of examination and information needed to evaluate the fuel characteristics were determined by specialists in the various disciplines.

In preparation for the in-cell examination, the following tasks of significance were completed:

1. A cask sufficient to handle the Peach Bottom element was obtained.
2. The hot cell facility was prepared to handle the large cask and transfer the radioactive fuel into the cell.
3. In-cell equipment to properly handle and evaluate the Peach Bottom element was designed and built.
4. A major high-level cell decontamination and cleanup was conducted so that the in-cell equipment could be installed.
5. Detailed disassembly procedures were prepared and a mockup examination was conducted to test these procedures and the in-cell equipment.

Gulf General Atomic was fortunate in obtaining an AEC-owned lead shipping cask constructed for fuel shipments from the Hallam reactor. Necessary licenses were obtained after some modifications were made to the cask.

It was necessary to modify the hot cell dock facility and add a motorized dolly-track system to move the 48,000-lb cask into the service area and against the high-level cell door prior to ejecting the fuel element from the cask.

The design of the in-cell equipment commenced in June 1967, and construction of the various devices paralleled the cell cleanup period. Special cell fixtures and equipment are described in Appendix B.

The cell cleanup and decontamination were completed in September 1967, and all the new equipment had been installed by November 1967.

The mockup examination to test the new devices was conducted in January 1968.

5.2. MOCKUP EXAMINATION

A nonfueled element was used to test and check out the equipment and postirradiation examination procedures. Most operations went very well. The motorized dolly had to be modified several times before it operated properly. The wheel track surface and guide rails had to ground so that no discontinuities existed.

A mobile crane was used to lift the cask from the cradle to a horizontal position. The mockup element was sealed into a canister. The canister was set into the cask guide frame. The cask was lowered into the guide frame, and the grappler mechanism was used to pull the element into the cask. The plug valve was closed, the grappler secured, and the cask lowered into the cradle.

The crane was used to place the cask and cradle onto the carrier dollies. The dollies were moved into the service gallery. The shielding donut, the canister removal device, and the plug valve open-and-close mechanism were attached to the cask. The cask was moved into the low-level cell. The door plug in the high-level cell door was removed using the door plug removal device. The cask was then rolled against the high-level cell door. The canister was discharged from the cask into the high-level cell using the canister removal device. After this successful test, the cask, cask equipment, and Peach Bottom handling hardware were shipped to the reactor site.

The disassembly of the mockup element in the cell went very well. The canister was removed by girdle cutting the ends and then slitting the aluminum longitudinally. Various cuts were made on the element to test the techniques of plunge-cutting, girdle-cutting, slitting, and coring.

Test measurements of the graphite sleeve, compacts, and graphite spine were successfully made with dial gages.

The testing of the inert gas protection system, including purge collar, caps, and boxes, was not included in the mockup examination. The mockup fuel element did not contain fuel-bearing compounds.

5.3. INERT GAS PROTECTION SYSTEM

It was originally planned to conduct the examination in the hot-cell air atmosphere. This decision was based on hydrolysis rate studies on

unirradiated fuel and observations of the stability of uranium-thorium carbide kernels to hydrolysis in hot-cell examinations of loop experiments and of fuel particles from irradiation capsules. Although the hydrolysis rate of the specific type of irradiated fuel in the Peach Bottom fuel elements was not known, it was not expected to be rapid. Moreover, it was felt that since two fuel elements were to be examined, the hydrolysis problems could be studied with the first element and, if necessary, additional measures could be taken to protect the second element from environmental moisture.

The hydrolysis problem was mitigated, however, by developing a method of passing a dry cover gas over the fuel compacts similar to the way in which reactor helium passes over the fuel compacts to pick up and trap the fission products. A purge collar was designed to form a tight seal over the graphite sleeve of the fuel elements for either end. In the case of the C05-05 fuel element, known to have a cracked sleeve, two purge collars were used. The purge collars were placed at the top and bottom end of the fuel element to allow the purge gas to flow toward the center and out through the cracked region. For an unbroken fuel element, only one purge collar was needed.

The inert purge gas selected was commercially available nitrogen of high purity with an H_2O vapor limit of less than 5 ppm. Such a supply system was installed in the hot cell, all lines were baked out, and appropriate moisture monitors were installed. The nitrogen gas actually averaged between 0.5 and 3.5 ppm moisture during examination of both fuel elements. This moisture level was not enough to cause significant hydrolysis. Calculations indicated (see Appendix C) that with 3 ppm moisture, the reaction rate of the purge gas with the fuel would be only 0.033 g metal carbide per day per fuel element. This is only 0.002% of the carbide in a fuel element.

6. HOT-CELL EXAMINATION OF E05-05 AND C05-05 FUEL ELEMENTS

As described earlier, equipment and procedures for the hot-cell examinations were developed and tested successfully. Detailed procedures were devised to obtain the following information:

1. For the E05-05 fuel element, to identify the suspected area that was thought to be a scratch or crack and check the validity of the gamma-scan technique and system at the reactor facility.
2. For the C05-05 fuel element, to conduct a detailed failure analysis to identify the mode of failure and its effect on the continued operation of the reactor system. Observations at the site had shown that this element had a complete circumferential break in the sleeve.

In general, the procedures called for the performance of the following over-all evaluations of these two fuel elements:

1. Check the effects of shipment of the fuel elements from the reactor site to the Gulf General Atomic hot cell by gamma-scanning, observations of the shipping canister, and measurements to check the integrity of the bottom weld joint and upper Magneform joint.
2. Visually observe the condition of the outer surfaces, components, and assembly joints.
3. Measure longitudinal bowing characteristics.
4. Measure component dimensions.
5. Determine the stability of the fuel compacts.
6. Perform metallographic examinations of the fuel compacts and determine the conditions of the particles.
7. Determine fission-product distributions by removing and analyzing coring samples from the sleeve, fuel compacts, reflectors, and spine.
8. Photograph the various operations as they occur.

For the E05-05 examination, the procedures as developed were generally used and the examination was conducted with great success. These procedures also worked well for the C05-05 element up to the point where the gross fracture in the central region of the element was revealed. Upon this discovery, modified procedures were developed to handle the subsequent disassembly in order to uncover, if possible, the failure mode.

6.1. POSTIRRADIATION EXAMINATION OF FUEL ELEMENT E05-05

6.1.1. Operations

The loading of fuel element E05-05 into the shipping cask occurred at the Peach Bottom site on April 8, 1968. The shipment was not made until April 10 because of the license requirement to inspect the cask 40 hr after insertion of the element. Shock load indicators were mounted directly on the cask housing to indicate shock during the cross-country trip. The minimum shock indicator, which was set for 5 g, was not tripped during the journey. Figure 6.1 shows the cask loaded on the truck just prior to leaving the reactor site.

The unloading of the cask, its placement in the proper position for fuel element discharge, and the discharge of the element into the high-level cell were performed without difficulty.

The canister was thoroughly examined for scratches, dents, hot spots, or any mishandling and appeared in excellent condition.

Prior to any disassembly steps, the E05-05 fuel element was gamma-scanned in the hot cell. The hot-cell gamma scan was in agreement with the scan taken at the reactor, indicating that no movement of the fuel compacts within the sleeve had occurred during the cross-country shipment. An orientation mark was placed on the canister to circumferentially identify the position of this gamma scan, and the mark was carried over to the fuel element when the canister and internal shock sleeve were removed.

6.1.2. Disassembly

The first penetration of the canister was made on April 29, 1968 after a purge band was placed on the lower end of the canister. A puncturing device in the purge band was immediately connected to the inert N₂ purge gas system to apply a pressure of 1.0 psig within the canister. Testing of the weld and Magneform end seals indicated that the hermetic seal had not been broken during storage and shipment.

Next, the grappler end (top) of the canister was girdle-cut and removed, and a purge cap was slipped onto the upper end of the fuel element to provide the inert gas (N₂) purge over the fuel compacts. This inert gas flows down the internal part of the fuel element and back over the outer portions. During the girdle cut, inert gas flowed through the original canister penetration and out the saw cut space until the girdle cut was completed. It took about 90 sec to remove this upper canister section and place the purge cap over the upper end of the fuel element (Fig. 6.2). The bottom end of the canister was also removed by a girdle cut, and a purge plug was placed in the bottom end of the fuel element. This purge plug was not connected to an inert gas flow line but had a perforated rubber balloon detector to observe gas flow through the element.

Fig. 6.1--Cask loaded on the trailer at Peach Bottom and ready for shipment to the Gulf General Atomic hot-cell facility

Fig. 6.2--Fuel element E05-05 in high-level hot cell. Grappler end of canister removed and a purge cap installed

The canister was then split by two longitudinal cuts at 180°. The iron shock sleeve liner insert for the fuel element shipment also had to be longitudinally cut at 180° intervals for removal (Fig. 6.3).

6.1.3. Inspection Results

The complete surface area of the E05-05 fuel element was carefully inspected with short-focus binoculars and the in-cell Kollmorgen, a periscope with magnifications to 2.5X and photographic capabilities. The fuel element did not have a crack but rather a surface blemish, caused by rubbing of the sleeve surface. Figure 6.4 is a photograph of the television screen view of the suspected area at the reactor. Figures 6.5 and 6.6 are views of the area identified by project personnel as the same area. If one looks closely, the original machining marks are visible, and the edges of these marks have been disturbed by abrasion which reflects light differently than does the surrounding surface.

The length of the element including the upper and lower joints was measured and the results are compared to the as-built dimensions taken before insertion into the reactor in Fig. 6.7. Diameter measurements were then taken every 6 in. at 120° rotations and are shown in Fig. 6.8. These measurements indicate the diameter of the fuel element is within the original maximum-minimum tolerance band.

The bowing of this element was measured with the transit and offset bow measurement tool, and the results are shown in Fig. 6.9. The fuel element had a definite bow of 0.140 in. at approximately 20° from the zero degree orientation. The direction of bowing was such that the middle portion of the element was closer to the control rod F04-01 than were the ends of the element. The element orientation is illustrated in Fig. 6.10.

Following these measurements, corings were removed from the sleeve at eight locations along one axial line (Fig. 6.10). This line is closer to the maximum activity side of the element than the side adjacent to the control rod. The axial positions of the corings are given in Table 6.1. To prevent exposure of the fuel to the atmosphere by completely penetrating the sleeve (0.3665 ± 0.0035 in. thick), the coring tool was limited to a depth of 1/4 in. by a shoulder. Coring weights ranged from 0.16 to 0.24 g (Table 6.1). Except for one or two cases, corings in the fueled region were taken from positions thought to be opposite the centers of compacts, and the spacer ridges were avoided. Each coring was rebottled in the laboratory, weighed, and gamma-counted with the Ge(Li) detector and a 4096-channel analyzer. Gamma spectrometry results are reported in Tables 6.2 and 6.3. The values reported should be accurate within a factor of about two.

Almost all of the nuclides detected were activation products of impurities in the graphite, the principal ones being Sc^{46} , Mn^{54} , Co^{60} , Nb^{95} , and Tb^{160} . Very low fission-product concentrations in the outer half of the sleeve indicated that the sleeve and purge stream had been very effective in limiting the migration of fission products into the primary system.

Fig. 6.3--Fuel element E05-05 with canister and upper half of liner also removed

Fig. 6.4--Photograph of closed-circuit television monitor at Peach Bottom showing suspected diagonal line thought to be a scratch or crack

Fig. 6.5--Hot-cell photograph showing surface blemish. This was identified as the suspect area in the television scan. (Note: Kollmorgen camera takes a mirror image)

Fig. 6.6--Same surface blemish as in Fig. 6.5 with different lighting. Blemish is in lower right section of photograph

Fig. 6.7--E05-05 fuel element length measurements. Dimensions in parentheses are measurements at time of manufacture

Fig. 6.8--E05-05 sleeve outside diameter measurements

Fig. 6.9--E05-05 bow determinations

Fig. 6.10--E05-05 element orientation

Table 6.1
CORING DATA FOR E05-05 FUEL ELEMENT

Coring Designation	Distance from Bottom of Fuel Element (in.)	Distance from Bottom of Fuel Column (in.)	Position	Coring Weight (g)
Ref. point	143.9	--	Top of fuel element	0.1605
A	130.0	--	Upper reflector assembly	
Ref. point	122.0	--	Upper joint slot	0.1718
B	117.6	--	Opposite center of space for expansion for compacts	
Ref. point	114.7	88.7	Top of fuel column	0.1994
C	99.5	73.5	Opposite upper part of fueled region	
D	87.5	61.5	Close to center of range within which maximum temperature occurs	0.1811
E	80.0	54.0	---	0.1687
F	76.0	50.0	Close to power peak and maximum count rate on gamma scan	0.2442
G	51.5	25.5	Opposite lower part of fueled region	0.2108
Ref. point	26.0	0	Bottom of fuel column	0.2469
H	24.5	--	Opposite center of lower reflector compact	

Table 6.2
ACTIVITY CONCENTRATIONS IN E05-05 SLEEVE CORINGS

Nuclide	Activity Concentration ($\mu\text{C/g}$ graphite)							
	Sample A	Sample B	Sample C	Sample D	Sample E	Sample F	Sample G	Sample H
Sc ⁴⁶	---	1.2	2.2	13	8.1	7.2	1.6	1.7
Cr ⁵¹	0.1 ^a	---	0.4 ^a	---	3.4	2.9	3.1	---
Mn ⁵⁴	---	---	---	---	(b)	0.041	---	0.006 ^a
Co ⁵⁸	---	---	0.11	---	0.19	0.13	0.21	0.063
Co ⁶⁰	0.014	0.22	0.38	0.53	0.65	0.66	0.96	0.71
Nb ⁹⁵	---	0.40	(b)	(b)	(b)	(b)	0.51	0.43
Cs ¹³⁴	0.007	---	0.037	---	0.029	0.044	---	0.014
Cs ¹³⁷	0.021	0.027	0.13	0.12	(b)	0.11	---	0.037
Ce ¹⁴¹	---	0.15	0.18	1.2	1.1	1.2	0.24	---
Ce ¹⁴⁴	---	0.018	0.009	---	---	---	0.059	0.13
Tb ¹⁶⁰	---	0.17	0.67	4.6	3.5	3.5	1.2	0.22
Ta ¹⁸²	0.01 ^a	0.026	0.17	0.25	0.21	0.42	0.89	0.23
Pa ²³³	---	---	0.1 ^a	1.5	1.3	0.62	2.0	2.2

^aQuestionable value.

^bDetected.

Table 6.3
CONCENTRATIONS OF NUCLIDES IN E05-05 SLEEVE CORINGS

Nuclide	Concentration (mg/g graphite)							
	Sample A	Sample B	Sample C	Sample D	Sample E	Sample F	Sample G	Sample H
Sc ⁴⁶	---	3.5x10 ⁻⁸	6.5x10 ⁻⁸	4.0x10 ⁻⁷	2.4x10 ⁻⁷	2.1x10 ⁻⁷	4.7x10 ⁻⁸	5.1x10 ⁻⁸
Cr ⁵¹	1x10 ^{-9a}	---	4x10 ^{-9a}	---	3.7x10 ⁻⁸	3.1x10 ⁻⁸	3.4x10 ⁻⁸	---
Mn ⁵⁴	---	---	---	---	(b)	5.3x10 ⁻⁹	---	8x10 ^{-10a}
Co ⁵⁸	---	---	3.5x10 ⁻⁹	---	6.0x10 ⁻⁹	4.1x10 ⁻⁹	6.7x10 ⁻⁹	2.0x10 ⁻⁹
Co ⁶⁰	1.2x10 ⁻⁸	1.9x10 ⁻⁷	3.4x10 ⁻⁷	4.7x10 ⁻⁷	5.7x10 ⁻⁷	5.8x10 ⁻⁷	8.6x10 ⁻⁷	6.3x10 ⁻⁷
Nb ⁹⁵	---	1.0x10 ⁻⁸	(b)	(b)	(b)	(b)	1.3x10 ⁻⁸	1.1x10 ⁻⁸
Cs ¹³⁴	5.4x10 ⁻⁹	---	2.9x10 ⁻⁸	---	2.2x10 ⁻⁸	3.4x10 ⁻⁸	---	1.1x10 ⁻⁸
Cs ¹³⁷	2.4x10 ⁻⁷	3.2x10 ⁻⁷	1.5x10 ⁻⁶	1.4x10 ⁻⁶	(b)	1.2x10 ⁻⁶	---	4.3x10 ⁻⁷
Ce ¹⁴¹	---	5.3x10 ⁻⁹	6.4x10 ⁻⁹	4.3x10 ⁻⁸	4.0x10 ⁻⁸	4.2x10 ⁻⁸	8.6x10 ⁻⁹	---
Ce ¹⁴⁴	---	5.3x10 ⁻⁹	2.8x10 ⁻⁹	---	(b)	---	1.9x10 ⁻⁸	4.0x10 ⁻⁸
Tb ¹⁶⁰	---	1.5x10 ⁻⁸	5.9x10 ⁻⁸	4.1x10 ⁻⁷	3.1x10 ⁻⁷	3.1x10 ⁻⁷	1.1x10 ⁻⁷	1.9x10 ⁻⁸
Ta ¹⁸²	2x10 ^{-9a}	4.2x10 ⁻⁹	2.7x10 ⁻⁸	4.0x10 ⁻⁸	3.4x10 ⁻⁸	6.7x10 ⁻⁸	1.4x10 ⁻⁷	3.7x10 ⁻⁸
Pa ²³³	---	---	5x10 ^{-9a}	7.2x10 ⁻⁸	6.4x10 ⁻⁸	3.0x10 ⁻⁸	9.6x10 ⁻⁸	1.1x10 ⁻⁷

^a-Questionable value.

^b-Detected.

6.1.4. Storage

At the conclusion of the above examination, the fuel element was re-canned in a special aluminum storage tube, purged for 24 hr, and then valved off under 1.0 psig of N_2 gas. A gamma scan was again taken to observe the fuel column height, which indicated no change from the original gamma scan. This tended to confirm the fact that the purge gas had been effective in preventing hydrolysis. If significant hydrolysis and compact degradation had occurred, fuel chips, flakes, and dust would probably have fallen into the void cavity, a condition that would have been revealed by the gamma scan.

6.2. POSTIRRADIATION EXAMINATION OF THE C05-05 FUEL ELEMENT

The orientation of C05-05 in the reactor is shown in Fig. 6.11. Many postirradiation examination measurements were made with reference to this orientation.

6.2.1. Operations

The removal of the C05-05 fuel element at the reactor facility was made with the failed fuel element removal tool as described in Section 4. The element was gamma-scanned at the site while still inside the removal tool. The tool was removed and the bare element was then lifted by the top knob for inspection with a closed-circuit television system. This revealed that the element sleeve was circumferentially broken about half way from the top. A portion of the central graphite spine was seen to extend below the sleeve fracture, but no fuel compacts were observed (Fig. 4.1).

Once the circumferential crack on the sleeve was revealed, the element was lowered back into the canning machine. Visual examination of the top of the fuel element indicated the element was 2 to 3 in. higher than before. The element was rotated until the upper half of the element dropped a few inches, and the element was then canned and shipped.

It became evident later in the postirradiation examination that the top half of the element was positioned 180° from the bottom half at this point in time. Furthermore, the handling and rotation of the top half had apparently resulted in additional damage to the sleeve. These points are covered in detail later.

The reception and insertion of the C05-05 fuel element into the high-level hot cell was accomplished without incident on May 6, 1968. Visual observation of the canister indicated it was in excellent condition.

A gamma-scan of this element was conducted on the opposite side of the element from the first scan before any disassembly operations. The canister was marked to reference the position of this scan. The original Peach Bottom gamma scan is shown on Fig. 6.12, and the second scan, performed at Gulf General Atomic, is shown on Fig. 6.13. Both show high gamma activity from compacts 10 and 19, measured from the bottom of the fuel element. However, two discrepancies between the scans were noted:

Fig. 6.11--C05-05 fuel element orientation diagram

Fig. 6.12--Gamma scan information from fuel element C05-05 recorded at Peach Bottom

Fig. 6.13--Gamma scan information from fuel element C05-05 recorded at the Gulf General Atomic hot cell

1. Compacts 13 and 14 showed an abnormal decrease in gamma intensity in the Gulf General Atomic scan. The Peach Bottom gamma scan had indicated an above-normal gamma peak for compact 13.
2. The fuel column height (88.75 in. original) was measured from the Peach Bottom gamma scan to be 89.75 in., indicating a growth of about 1.0 in. The Gulf General Atomic gamma scan indicated a fuel column length of 90.6 in. or a growth of almost another inch.

Because of these anomalies, two additional gamma scans were made at 90° and 270° to complement the original gamma scans taken at 0° and 180°. The only significant features displayed by these two additional gamma scans were the variance in abnormal peak heights of compacts 10 and 19.

6.2.2. Disassembly

The penetration and testing of the C05-05 fuel canister revealed the canister to be sound. The girdle cuts on both ends and the attachments of purge caps to the upper and lower ends of the fuel element was done sequentially following the same procedure as in the E05-05 disassembly. The canister was sectioned and removed by making two longitudinal cuts at 180° intervals. Since the shock insert liner was in two half sections, it was not necessary to cut this liner longitudinally.

When the upper half section of the shock liner was removed, the condition of the fuel element was much worse than the expected transverse sleeve fracture and vertical crack. Instead, as shown in Figs. 6.14 and 6.15, there was extensive fracture and separation in the front part of the graphite sleeve, the separation being 1-1/4 in. at the widest point in front. The fracture included a completely broken triangular portion of the graphite sleeve and two wandering longitudinal fractures from 14 to 24 in. long. Because of the severity of the fracturing in the central region, immediate action was taken to prevent hydrolysis by placing the inert gas purge box over the entire element.

The first of several revisions to the step-by-step procedures was instituted to attempt to resolve the nature of these fractures and to provide a failure analysis. The immediate concern was to preserve the condition of the element, and until additional inert purge gas chambers could be built, continued surveillance was maintained to check on the fuel element condition.

After two and one-half days under the long storage box, the openings at the broken triangular section of the sleeve appeared to be widening, which indicated that some hydrolysis might be occurring. Immediate action was taken to place the element in a safer condition by first removing the portion of broken sleeve and then removing the exposed fuel compacts. The portion of sleeve was replaced to close the fracture gap, which permitted the cover purge gas system to function more efficiently. Pertinent photographs are shown in Figs. 6.16, 6.17, and 6.18.

Because of the separation, and evidence that the upper and lower sections of the graphite sleeve were touching on the back side of the element (Fig. 6.16), it appeared that the handling of the upper portion of the fuel

Fig. 6.14--C05-05 fuel element with upper half of shock liner removed showing massive fracture and separation

Fig. 6.15--C05-05 fuel element; closeup view of separation showing broken fuel compact pieces (2.5X)

Fig. 6.16--View of back side of break showing contact of graphite sleeve (2.5X)

Fig. 6.17--View of broken area with sleeve sections removed (0.33X)

element at Peach Bottom had probably rotated this upper section approximately 180° with respect to the bottom portion. The rotation of the top portion of the element at Peach Bottom was confirmed by counter-rotating the top portion about 180° and refitting the original transverse fractured surfaces (Fig. 6.19). During these operations, the bottom half of the shock sleeve liner was removed, which exposed one of the more prominent longitudinal fractures on the sleeve (Fig. 6.20). It may be seen in these photographs of the fuel compacts that the general surface conditions were good, though radial fractures are evident and compact 14 has partly disintegrated.

The subsequent postirradiation examination of the C05-05 fuel element was conducted in three phases. In the first phase, the highly fractured central region of the fuel element was examined (fuel compacts 14 through 20 are included, as numbered from the bottom of the fuel element). In the second phase, the upper portion of the fuel element, including compacts 21 through 30 and the upper reflector was disassembled. The third phase involved disassembly of the remaining lower portion of the fuel element including the bottom connector, trap, bottom reflector, and compacts 1 through 13 to the transverse fracture.

Phase I - Grossly Fractured Section. The Phase I disassembly began after the highly fractured area was covered by a purge box to protect it from hydrolysis during handling operations (Fig. B.2). Then the element was separated at the transverse fractured area so that the lower portion of the fuel element could be protected during operations in Phases I and II. Curatory examinations of the fractured sleeve, the compact, and the spine were made before a purge cap was fitted over the fractured portion to protect it. Several interesting features were revealed.

First, each fuel element contains three spines, each about 30 in. long, on which 10 compacts are installed. This should place the spine boundaries between compacts 10 and 11 and 20 and 21. The fracture was located near the top of compact 13, which should result in about 8 in. of spine extending below the transverse fracture. However, the bottom of the middle spine was protruding only about 3 to 4 in. below the transverse fracture, as shown in Fig. 4.1. Second, observations of the spine cavity and photographs of the void above the lower spine indicated that no fuel residue or particles had entered this void space (Fig. 6.21). Third, observations and photographs (Fig. 6.22) of the transverse fracture indicated that fractured surfaces of compact 13 were generally in the same plane as the fractured surface of the sleeve.

When the two halves had been separated and the lower section secured with the purge cap, the graphite sleeve was girdle-cut at approximately the junction between compacts 20 and 21, thus removing the highly fractured section from the upper part of the fuel element. It was then evident that the middle spine and upper spine were not in their original position, but were instead vertically displaced about 5 in. This displacement will be discussed later in the report. Before the protective purge cap was installed on the cut end of the upper section, a photograph was taken of the portion of compact 21 extending below the girdle cut (Fig. 6.23).

Fig. 6.18--Closeup view of compacts 14, 15, and 16; compacts were disturbed during removal of the broken triangular portion of the graphite sleeve (2.5X)

Fig. 6.19--Transverse fracture area after counter-rotating upper fuel element section approximately 180° with respect to bottom portion and then refitting the element (2.5X)

Fig. 6.20--View of broken area after removing fuel compacts and rotating element to remove bottom half of shock sleeve liner (0.33X)

Fig. 6.21--End view of transverse fracture showing top end of spine, approximately 8 in. below (2.5X)

Fig. 6.22--End view of transverse fracture of both sleeve and compact; note sharpness of fission gas channels (2.5X)

Fig. 6.23--End view of compact 21 at circumferential (girdle) cut-off area, upper fuel element section

Continued disassembly of the fractured section involved making two longitudinal cuts to the two longitudinal fractures. The highly fractured central portion was then moved from the saw table to a table in front of a high-level cell window, where removal of the upper half portion of the sleeve revealed compacts 16, 17, 18, 19, and 20 (see Fig. 6.24). Compact 18 had a definite barrel shape in the lower midportion of the fuel compact, and compact 19 had a rather dark surface blotch on part of its exposed surface. The chamfer areas between compact edges, normally 1/8 in. wide, were crushed together at the boundaries of compacts 16 and 17, and compacts 17 and 18. The chamfer areas between compacts 18 and 19 and compacts 19 and 20 appeared normal.

The removal of the lower half of the sleeve was difficult because the longitudinal fractures were not exactly 180° apart. Only compacts 19 and 20 could be removed as complete compacts; compacts 16, 17, and 18 were removed in three or four longitudinal sections. Physical measurements were made of compact dimensions before storage in the individual gas purge containers. Measurements were also made of the graphite sleeves and spine. Detailed photographs were made of the compacts, sleeve internals, and fractured surfaces.

Phase II - Upper Fuel Element Section. The Phase II portion of the disassembly on the upper section of the fuel element required girdle cutting the upper reflector from the fuel sleeve at the fuel expansion void cavity. Physical measurements of this void cavity are shown in Fig. 6.25. The upper sleeve section was cut longitudinally at 0°, 120°, and about 240° to facilitate removal of the fuel compacts. The dimensions of the compacts, sleeve, and spines were recorded, and detailed photographs were made of all components.

Three areas of interest were encountered during this part of the disassembly: (1) the upper spine had moved so that it extended completely into the expansion void area; (2) the surfaces of the fuel compacts were not all the same texture or condition and could not be related to their respective position in the fuel element (Fig. 6.26); and (3) the inner surfaces of the graphite sleeve had a shadowing phenomena that provided "fingerprints" of the compacts' surfaces. This was also noted for the compacts removed during Phase I. The interesting feature of these signatures was that for compacts 19 through 30, the edges of the signatures had shifted in a vertical direction about 0.5 in. The general condition of the compacts in this upper section was very good. Measurements of the outside diameters showed, however, that four compacts were smaller than the original manufacturing tolerance band, two compacts were within the band, and four compacts were larger.

As part of Phase II, two parallel experiments were conducted. The first was to resolve the extent of hydrolysis of a compact in the hot-cell air without inert gas protection. Subsequent to the dimensional measurements, compact 21 was left in the hot-cell air for a period of 122 hr immediately after removal from the graphite sleeve, and a series of photographs was taken over grid paper to indicate dimensional changes. The compact suffered no gross hydrolysis, the only noticeable change being that some surface material flaked off in small pieces and dropped onto the grid paper (Fig. 6.27). Final measurements indicated no abnormal growth.

Fig. 6.24--Upper half of graphite sleeve removed from central region of fuel element exposing compacts 16, 17, 18, 19, and 20 (from right). Note barrel shape of compact 18.

In the second experiment, continued measurements over many days were taken of several compacts in this upper portion. The compacts that were within or below the tolerance band showed no increase or change of dimensions. The compacts that were larger than the tolerance band experienced some continued diametral growth, by as much as 0.010 to 0.015 in. in some cases. This growth was not believed to be due to hydrolysis, although a positive statement to that effect was not possible. Gradual relaxation of the previously restrained compact was believed to be the principal cause of the continued growth.

Phase III - Lower Fuel Element Section. For Phase III of the disassembly, the sleeve around compact 13 was retained in the as-fractured condition. Circumferential cuts were made at compact 12 and compact 11, leaving the compacts in their original position in the graphite sleeve. The spine inside these compacts had been removed during Phase II. The force necessary to slip compacts 11 and 12 from the inside of their respective graphite sleeves was measured. A dead-weight loading device indicated that compact 12 required 7.7 lb to be pushed from the graphite sleeve. An Instron remote testing machine, which registered a force of 845 lb, was required to push compact 11 out of the graphite sleeve.

The continued disassembly of the lower section by longitudinal sleeve cuts proceeded normally (Fig. 6.28), with the exception of compact 10, which developed a circumferential fracture near the center. This fracture (Fig. 6.29) developed overnight while six compacts were in the inert protection

Fig. 6.25--Fuel expansion void cavity measurements

Fig. 6.26--Compacts 22 through 30 (from right); compact 21 was removed for hydrolysis studies

Fig. 6.27--Compact 21 after 122-hr hydrolysis test in hot-cell air

Fig. 6.28--Compacts 1 through 10 (from right) after removal of graphite sleeve

Fig. 6.29--View of compact 10 rupturing in center after overnight period in inert-gas purge box. Note there are no deleterious effects to the adjacent five compacts. Compact 10 had exhibited abnormally high gamma scan in center of compact

box and was not observed until the following morning. The five other compacts (9 through 5), although exposed to the same conditions as compact 10, showed no deleterious effects. Compact 10 was one of three compacts that registered high activity in the original gamma-scan examination.

No shadows or fingerprints were noted on the inside of the lower section of sleeve.

Throughout these operations, the original orientation 0° mark was transferred to all of the sleeve parts, compacts, and spine sections. Each compact was also marked to note the top portion of the compact with respect to the top portion of the fuel element.

6.2.3. Sleeve Fracture Reconstruction

An investigation was performed to determine the extent of the sleeve failure during irradiation. For the analysis, the fractured midsection of the C05-05 fuel element graphite sleeve was reassembled on an aluminum tube with the same outside diameter as the fuel compacts. This assembled section was placed in front of the hot-cell window and a mirror was placed behind the element. The setup was made as near to the conditions that existed at the Peach Bottom canning machine as possible. The element was rotated, and the observations were compared to a photograph of the site television picture. One of the wandering longitudinal cracks was found to be in such a position that it matched identically with the lines observed in the television picture. Figures 6.30 and 6.31 illustrate this view. Figure 6.32 is a photograph taken from the front. Since the Peach Bottom pictures did not show the apparent massive fracture observed initially in the hot cell at Gulf General Atomic, it is evident that the massive fracturing of the sleeve did not occur during irradiation. Rather, the massive fracture was due to the handling of the halves of the fuel element at Peach Bottom.

6.2.4. Inspection Results

Physical and chemical measurements of the various fuel element components were made. The details of the physical measurements on components such as the sleeve, compacts, and spine are summarized in Fig. 6.33. Specific areas of interest are discussed in the following subsections.

Fractured Section. The details of the fractured sleeve section are also illustrated in Fig. 6.33. As discussed in Section 6.2.3, the massive fracturing of the sleeve did not occur during the irradiation. The comparison of gamma scans taken at the reactor site, with the element in the failed fuel element removal tool and before subsequent handling, versus the gamma scan taken at the hot cell indicated no gross separation of the fuel element prior to removal from the reactor vessel. The physical measurements of the void cavity (Fig. 6.25) and the indicated fuel column expansion of 5/8 in. are not abnormal.

Macroscopic examinations of the fractured graphite surfaces appeared normal with no indications of flaws or voids. The characteristics of the fuel element sleeve fracture were comparable to fractures developed in laboratory experiments which resulted from internal fuel compact expansion. This point is discussed more fully in Section 7.

Fig. 6.30--Closed circuit television photograph taken at Peach Bottom showing suspected crack in mirrored portion of picture

Fig. 6.31--Fractured sleeve assembled on Al tube and oriented in same position as suspected at Peach Bottom examination

Fig. 6.32--Front view of longitudinal crack identified at Peach Bottom; compact 18 is located approximately at the 12- to 15-in. mark on the ruler

Graphite Sleeve. The outside diameter of the graphite sleeve was within the design tolerances. Outside diameter measurements were not taken in the fractured area because of the expanded openings of the cracks. The measurements of the sleeve wall thickness revealed some areas that were thinner or thicker than nominal values. These areas indicate that the inner void area was not completely concentric with the outside of the graphite sleeve. These dimensions were taken at various angular positions from the zero orientation marks.

Sleeve bowing specimens were taken in areas in the upper portion of the fuel region opposite compacts 21, 22, and 23, as well as the lower fuel region opposite compacts 8, 9, and 10. These specimens were approximately 9 in. long by 3/8 in. wide. These long narrow specimens were set adjacent to a certified straight edge. A light source placed behind the specimen and straight edge gave no indications of bowing of these graphite specimens.

Fuel Compacts. Some of the outside diameters of the thirty fuel compacts were below the original tolerance, some were within, and others exceeded the original design and manufacturing tolerances. The wall thickness measurements also indicated that the inner spine hole was not concentric with the outside of the fuel compact.

The length of the compacts as manufactured were generally 3.0 ± 0.1 in. The only abnormal compact length observed was that of compact 18, which was 3.19 in.

Gamma-scanning, gamma-spectrum measurements, and metallographic examinations of selected compacts were conducted subsequent to the physical measurements. The individual rotational gamma scans and gamma-spectrum measurements were done on compacts 6, 8, 9, 10, 11, 13, 18, 19, 23, 27, and 28. The rotational gamma scans of selected compacts show a circumferential (0° to 180°) variation of from a few percent to over 30% in the gamma activity. These differences may be due entirely to the local flux and power gradient produced by the adjacent control rod, or may be due in part to nonhomogeneities in the fuel particle concentrations in the compacts.

Gamma-spectrum analyses of the ratio of the 0.722-Mev Zr^{95} peak activity to the 0.312-Mev Pa^{233} peak activity indicates that the U/Th ratios are reasonably constant throughout the fuel regions measured.

Metallographic examination was performed in order to determine the condition of the fuel compacts and to relate this information to the failure analysis of Peach Bottom fuel element C05-05. It was also hoped that this investigation would clarify the differences observed in the conditions of the compacts as well as provide some insight into the greater-than-expected radial dimensional changes that were measured for a few of the samples.

Four compacts, 10, 18, 25, and 27, were selected for study because of their location and condition. Compact 25 was selected because it appeared to be generally in very good condition and was located near compact 27, which exhibited an abnormal expansion subsequent to dismantling of the sleeve. Fuel compact 18 appeared to have expanded radially upon first inspection and was indicated to have an excessive wall thickness; also, it was located in

Fig. 6.33--Postirradiation dimensional analysis of fuel element C05-05

the center spine section where the fracture of the graphite sleeve occurred. Compact 10 was from the lower spine assembly and was located in the proximity of the transverse fracture of the sleeve. Its condition was judged somewhat abnormal because of its cracked condition; in addition, this compact exhibited a higher-than-normal gamma scan. Compacts 13 through 16, which were located in the badly broken portion of the sleeve of the element, were not examined because of their badly damaged condition.

A piece approximately 1/2 in. by 1/2 in. was taken from each of the compacts. An attempt was made to maintain the location and orientation of these sections with respect to the original compact orientation; however, because of the highly fractured condition of compacts 10 and 18 at the time of sampling, the orientation could not be maintained (Fig. 6.34). The handling of the material to be examined was controlled to assure minimal exposure to the atmosphere. All transfer operations were made in inert-atmosphere containers, and the pieces were impregnated with epoxy as soon as feasible, which inherently protected the material until it was ready to be ground and polished for viewing. Standard techniques used for metallographic preparation of carbide-fueled materials were employed, and, again, extreme care was exercised to protect the material from moisture.

The condition of compact 25 (Figs. 6.35 through 6.37) was considered to be generally good; it was thought to be representative of other compacts that have a similar appearance. Compact 25 was used as a control specimen to ensure that the proposed examination procedures produced satisfactory results.

The microscopic examination of this compact revealed that about 20% of the coatings were unbroken, 35% were damaged,* and 45% were broken. This was the lowest number of damaged and broken coatings encountered among the four samples investigated, as shown in Table 6.4. About 40% to 50% of the coatings appeared to be more anisotropic than the remaining coatings, since under polarized light the Maltese cross pattern for these coatings showed more contrast than did the pattern for the remainder of the coated particles.

The reason for the apparent difference in anisotropy between coatings is that they were from different coating batches, i.e., fuel particles from various coating runs were mixed together during the production of the fuel compacts in order to obtain the proper fuel loading for each compact.

The graphite matrix had a number of small cracks; these appeared to originate at the inside surface of the compact and to run radially into the compact.

The segment of compact 10 (Figs. 6.38 through 6.41) that was examined contained a large number of small matrix cracks, many of which emanated from broken particles, and some of these were relatively wide (as wide as about 200 μm). The cracks were observed to run in all directions; however, the

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* A damaged coating was pitted and cracked (usually on the inside) but unbroken in the plane of the metallographic section.

COMPACT 25

COMPACT 27

Fig. 6.34--Figures show location of specimen used for metallographic examination. Asterisk (*) indicates surface examined

(M255-2)

(50X)

Fig. 6.35--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 25

(M255-11)

(75X)

Fig. 6.36--Photomicrograph showing appearance of fuel particles and matrix material of compact 25. Concentration of fuel particles shown here is higher than normal for this compact. Note presence of unbroken and broken particles

(M255-10)

(75X)

Fig. 6.37--Polarized-light photomicrograph of field shown in Fig. 6.36

(M255-81)

(75X)

Fig. 6.38--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 10

(M255-79)

(75X)

Fig. 6.39--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 10.

(M255-78)

(75X)

Fig. 6.40--Polarized-light photomicrograph of field shown in Fig. 6.39

majority appeared to lie in the radial direction. These microcracks were also observed to some extent in the other compacts.

Table 6.4
SUMMARY OF PARTICLE CONDITIONS

Compact	Compact Condition	Coatings			Particle Density ^a (particles/cm ²)
		Unbroken (%)	Damaged (%)	Broken (%)	
25	Good	21	34	45	270 to 300
27	Fair, expanded	16	26	58	270 to 300
10	Fair, cracked	10	34	56	270 to 300
18	Badly fractured	7	9	84	400

^a-From metallography examination.

A somewhat greater percentage of the PyC coatings were observed to be fractured in this compact than in compact 25; on the other hand, this compact contained the highest ratio of less anisotropic to more anisotropic coatings as observed under polarized light, about 5:1.

The matrix of compact 27 contained only one or two small cracks; at higher magnification it appeared to be less dense than the other three specimens. Pertinent photographs are shown in Figs. 6.42 through 6.44. The number of unbroken, damaged, and broken PyC coatings approximately parallels the numbers observed in compact 10. The ratio of less anisotropic to more anisotropic coatings was intermediate between compacts 10 and 25.

Of the compacts examined metallographically, compact 18 exhibited the worst condition (Figs. 6.45 through 6.49). The matrix was badly fractured and 85% or more of the PyC coatings were broken. A considerable percentage of the particles with broken coatings, which were adjacent to the large matrix cracks, had a second phase on the surface of the kernels. This phase was tentatively identified as an oxide. It was not possible, however, to determine the means by which it formed, i.e., whether it formed as a result of hydrolysis from exposure to the atmosphere during handling or as a result of oxidation during the reactor operation. In any event, this oxide phase was not extensive and was not believed to be related to the gradual compact expansion observed in the hot cell in Phase II of the disassembly.

A discussion of these results is presented in Section 7.

Graphite Spine. The physical measurements taken on the upper, middle, and lower graphite spine included diametral and length measurements. The upper and middle spine showed contraction in both the diametral and longitudinal directions whereas the lower spine remained within the tolerance bands.

(M255-83)

(75X)

Fig. 6.41--Photomicrograph showing fuel particles adjacent to a large crack in the matrix of compact 10

(M255-59)

(75X)

Fig. 6.42--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 27

(M255-56)

(75X)

Fig. 6.43--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 27. Note larger unbroken particles showing fission recoil damage ("spearhead" attack)

(M255-55)

(75X)

Fig. 6.44--Polarized-light photomicrograph of field shown in Fig. 6.43

(M255-43)

(150X)

Fig. 6.45--Broken particle showing gray phase around (Th,U)C₂ kernel; phase is tentatively identified as an oxide (compact 18)

(M255-27)

(75X)

Fig. 6.46--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 18

(M255-33)

(75X)

Fig. 6.47--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 18. Note gray phase around carbide kernels

(M255-32)

(75X)

Fig. 6.48--Polarized-light photomicrograph of field shown in Fig. 6.47

(M255-29)

(75X)

Fig. 6.49--Photomicrograph showing typical appearance of fuel particles and matrix material of compact 18

6.2.5. Chemistry Analysis

Chemistry data obtained from E05-05 has provided information on fission-product distributions within the fuel element graphite. Corings were also taken from C05-05 for storage and possible future analysis.

Coring specimens in the fueled length of the element were made in nine areas of interest, and in each of these nine areas corings were taken from the sleeve, the fuel compact, and the spine. The compacts designated were 6, 9, 10, 11, 13, 14 (area nearest the fractured region), 18, 19, 23, and 27. Coring specimens were also taken from the reflector compact just below the fuel column and from the upper reflector.

6.3. POSTIRRADIATION EXAMINATION OF C05-05 AT LASL

Los Alamos Scientific Laboratory was requested by AEC-DRDT to examine materials from C05-05 in their large and very complete hot-cell facilities. This was done in the summer of 1968, following shipment of fuel compacts and portions of the fuel element sleeve on July 10, 1968. An edited summary of their findings, presented in CMB-1494, Examination of Materials from Peach Bottom Fuel Element C05-05, November 7, 1968, is given below.

1. A burst test on a section of unirradiated sleeve indicated that the initial crack that formed was longitudinal. Examination at

low magnification of fracture surfaces produced in tension, in compression, and in the burst test did not reveal differences.

2. Chemical analyses of unirradiated Peach Bottom coated particles for Th and U, by the procedures used for the irradiated compact materials, agreed with those reported by Gulf General Atomic.
3. No differences in appearance were detected between fracture surfaces in the irradiated sleeve from the center section of fuel element C05-05 and fracture surfaces produced at LASL in unirradiated sleeve material. Wall-thickness measurements on the center section of the irradiated sleeve agreed with the Gulf General Atomic values, showing nonconcentricity. Metallographic examination of irradiated sleeve material showed no anomalies. Spectrochemical analyses showed that the irradiated and unirradiated sleeve graphite had comparable high purity. The "shadow" or "signature" material from the inside of the irradiated sleeve was mainly carbonaceous and contained significantly more B, Si, Ca, Al, Fe, Mg, Ni, Th, and Cr than the irradiated sleeve. The most probable source of some of these elements, such as Si, Ca, Al, and Fe, could have been the compacts; measurable quantities of these impurities, as well as Th, were found in a compact that was analyzed.
4. Portions of eight irradiated compacts (10, 11, 13, 18, 19, 23, 25, and 27) from fuel element C05-05 were analyzed for Th and U. The results gave Th/U concentration ratios near the expected value of 5.0 for all compacts except one, 27, which had a concentration ratio of 5.5. The ranges in the concentration of the Th and U were larger and the concentrations themselves were lower in most cases than expected from analyses of entire compacts performed previously (before assembly) at Gulf General Atomic by gamma-counting. These differences may be due to uncertainties in the sampling method. Material from each of the eight compacts was examined metallographically, and counts were made of undamaged, damaged, and broken particles. The percentages of undamaged particles were in good agreement with those obtained by Gulf General Atomic on the same compacts (10, 18, 25, and 27). Compacts 18 and 19 had the smallest percentage of surviving particles (9%), whereas compact 11 had the highest (40%).
5. Pieces from compacts 18 and 11 were subjected to partial dissolution in nitric plus perchloric acids. The undissolved residue from compact 18 was mainly shells of pyrolytic carbon, with very few intact particles. Most of the PyC shells were bright and optically active under polarized light. The density of these shells by the sink-float method was sharply defined at $2.04 \pm 0.02 \text{ g/cm}^3$, indicating that most of the original particles in compact 18 had the same type of PyC, which after irradiation was oriented and dense.
6. Correlation of the Th and U analyses with the results of metallographic counting gave a measure of the relative amount of exposed fuel that was highest for compacts 18 and 19. However, the values were not excessively high compared to several of the other compacts.

The results obtained independently at LASL substantiate the data obtained at Gulf General Atomic and reinforce the conclusions derived from those data. The range in uranium and thorium concentrations observed by LASL in the compacts is at variance with the original loading data and the subsequent gamma-scan data taken at Peach Bottom and at Gulf General Atomic; it is possible that the apparent discrepancy is due to sampling problems.

7. DATA ANALYSIS AND EVALUATION

7.1. ANALYSIS OF PURGE FLOW MEASUREMENTS

7.1.1. Introduction and Summary*

The failed fuel element locator (FFEL) system was designed to detect a fuel element with a broken sleeve by obtaining an estimate of the purge flow in the element. In a failed element, the purge flow would be expected to diminish, stop, or change direction, depending on whether the failure occurred at the top or the bottom of the element.

There are three main components to the 3.5-psi pressure drop within the element. The porous graphite filter and flow limiter in the top of the element account for 3.0 psi, the fuel channel for 0.25 psi, and the trap and filter in the bottom for 0.25 psi. Thus, the total purge flow is quite insensitive to small changes in fuel channel pressure drop. For example, the resistance has to increase by a factor of four to produce a 20% decrease in total purge flow. Changes smaller than this would be within the statistical variation of the measurement and would not in general be observable. While a failed sleeve can be readily detected by measuring purge flow rate, purge flow measurements cannot be expected to be a sensitive indicator of the general condition of the element. It will be shown below, in fact, that no information of significance relative to general fuel element performance has been derivable from the purge flow data taken through 300 equivalent full-power days (EFPD) of operation.

7.1.2. Data Evaluation

Four sets of data are available on the purge flow through the elements. These data were taken at (1) time of manufacture, (2) before initial operation at power, (3) after 168 EFPD, and (4) after 300 EFPD. Set 1 was not used in this analysis.

To compare the FFEL sets, the data must be normalized. Set 2 was defined as the basis, and the measurements of sets 3 and 4 were adjusted so that the average flow through an element in each set was identical. The normalized flow for element i in sets 2, 3, and 4 has been denoted by $f_i^{(2)}$, $f_i^{(3)}$, and $f_i^{(4)}$, respectively.

A ring dependence to the relative flow between FFEL sets was noticed, a phenomenon due probably to plant flow conditions; it is not believed to be significant to the question of fuel element integrity. Let $R_j^{(3)}$ be the

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* The analysis of the FFEL data was performed by J. E. Larson and G. R. Hopkins.

average flow in the ring j element for data set 3, and similarly for the other sets. An adjusted flow parameter, F_i , defined as

$$F_i = f_i - R_j \quad ,$$

for i and j consistently chosen, can then be used to compare FFEL sets. Some of these comparisons are discussed below.

The relative ring dependence between sets 3 and 2 and between sets 4 and 2 are shown on Fig. 7.1, where the differences,

$$R_j^{(3)} - R_j^{(2)} \quad \text{and} \quad R_j^{(4)} - R_j^{(3)} \quad ,$$

are plotted as a function of the ring number, j . The data are consistent, showing no significant change during the 132 days between the assembly of data sets 3 and 4. The ring dependence to the difference in purge flows between sets is, at most, about 20%.

A comparison of the adjusted flow parameters, F_i , between sets 3 and 2 is shown on Fig. 7.2, where the number of elements is shown for a given value of the difference in adjusted flows, $F_i^{(3)} - F_i^{(2)}$. A similar comparison is shown on Fig. 7.3 for sets 4 and 2. A significant feature of the set 3 data (Fig. 7.2) is the relatively narrow distribution in value for most of the elements. Of the 804 elements analyzed, five showed flow decreases that would not appear to follow the statistical distribution. These are the elements with flow decreases greater than 1.6 shown on the right edge of the histogram. The locations and relative flow values for these elements, and others, are given in Table 7.1.

If low flow is indicative of incipient failure, these are the elements that would be expected to fail in subsequent operation. This, in fact, did not happen. During the next 132 EFPD of operation, several fuel elements did fail but none of them were low-flow elements in set 3. A list of the failed elements, excluding C05-05, is shown on Table 7.2. They do not exhibit remarkable purge flow data.

For future reference, the low-flow elements from set 4 are shown in Table 7.3.

Further analyses with the purge flow data at both Gulf General Atomic and Philadelphia Electric Company have indicated that there is no significant dependence of the purge flow data on azimuth. There does appear to be a slight dependence of purge flow on proximity to a control rod element, but the change in purge flow with time does not correlate with that proximity.

Fig. 7.1--Ring dependence of the purge flow difference

Fig. 7.2--FFEL flow distribution at 168 EFPD, normalized data

Fig. 7.3--FFEL flow distribution at 300 EFPD, normalized data

Table 7.1
LOW-FLOW ELEMENTS AT 168 EFPD

Position	f_i Values Normalized to Set 2 Data			$f_i^{(2)} - f_i^{(3)}$	Adjusted Data	$f_i^{(2)} - f_i^{(4)}$	Adjusted Data
	Set 2	Set 3	Set 4		$F_i^{(3)} - F_i^{(2)}$		$F_i^{(4)} - F_i^{(2)}$
E09-02	5.9	4.4	4.7	1.5	-1.05	1.2	-0.93
F09-09	5.0	2.8	3.2	2.2	-1.77	1.8	-1.6
A09-09	4.8	3.2	3.2	1.6	-1.16	1.6	-1.4
B09-09	5.3	3.8	4.0	1.5	-1.06	1.4	-1.1
E11-06	8.3	6.8	7.1	1.5	-1.22	1.2	-1.3
C11-04	6.6	5.2	5.5	1.4	-1.14	1.1	-1.2
A13-10	5.2	3.2	4.7	2.0	-2.17	0.46	-0.5
A14-03	7.6	5.2	7.9	2.4	-2.67	-0.30	0.17
C15-13	7.4	5.4	4.7	2.0	-2.68	2.7	-3.1
E16-04	6.7	5.0	7.1	1.7	-1.92	-0.41	-0.09
D17-13	5.2	4.0	5.1	1.2	-1.16	0.064	-0.26

Table 7.2
168 EFPD FLOW DATA FOR BROKEN ELEMENTS

Position	f_i Values Normalized Data		$f_i^{(2)} - f_i^{(3)}$	Adjusted Data
	Set 2	Set 3		$F_i^{(3)} - F_i^{(2)}$
E02-01	5.6	5.4	0.2	0.28
D06-01	7.6	8.0	-0.6	0.79
D08-04	6.3	6.2	0.1	0.09
E08-06	4.9	4.8	0.1	0.07
D10-03	4.8	4.6	0.2	-0.09
B12-12	6.5	7.0	-0.5	0.50
B12-02	6.6	7.2	-0.6	0.61
A13-07	5.0	5.4	-0.4	0.26
C14-07	5.1	4.8	0.3	-0.57
B15-09	4.9	5.2	-0.3	-0.30
C15-09	5.2	6.0	-0.8	0.21

Table 7.3
NORMALIZED PURGE FLOW OF LOW-FLOW ELEMENTS AT 300 EFPD^a

Position	f_i Values		
	Set 2	Set 3	Set 4
D07-05	5.9	4.0	4.08
A07-01	7.6	6.4	5.4
B07-01	5.9	4.6	4.7
B13-05	6.8	7.2	4.7
E14-03	---	---	0
C15-13	7.4	5.4	4.7
C17-11	6.9	9.4	5.0

^aLow flow is defined as a large decrease from set 2 to set 4.

7.2. STATISTICAL ANALYSIS OF FUEL ELEMENT FAILURES

7.2.1. Introduction and Summary

A statistical model of Peach Bottom fuel element failures was developed for use in evaluating the important parameters in the failure mechanism and for projecting fuel element performance into the future. A detailed discussion of the model is presented in Appendix D. The analysis is based on the hypothesis that fuel compact expansion is the cause of the failure of the sleeves. A brief description of the model is presented below in which the important parameters are identified. The evaluation of these parameters was made with reference to data obtained during the postirradiation examination, but it was concluded that the model is still essentially empirical.

About 200 EFPD of operation are anticipated for calendar year 1969. The model predicts about 20 to 40 failures during this period of time. These predictions are, at this point in time, very approximate, but additional observations of fuel element failures may allow an increasingly more accurate prediction of the subsequent failure rate.

7.2.2. Statistical Model

It was postulated that the primary cause of failure is radial expansion of a compact to a point beyond a critical value at which the sleeve cracks. It was also assumed that the compacts exhibit "mixed" behavior under irradiation, i.e., some expand radially while others contract. This mixed behavior is supported in part by hot-cell measurements of C05-05. Several C05-05 compacts were outside the stated radial tolerances after 168 days of irradiation, on both the high and low sides. The compact-to-sleeve radial clearance is approximated by a Gaussian distribution that broadens during the life of the reactor (due to the mixed behavior mentioned above). As the distribution broadens, an ever-increasing number of compacts expand beyond the critical value causing the sleeve to crack. Thus the important parameters in the model are:

r_f = the compact radius that leads to failure of the sleeve.

μ = the average radius of the compact.

$\sigma(t)$ = the standard deviation of the radius.

$p(t)$ = the probability that at time t a compact chosen at random causes a failure of the sleeve; i.e., μ exceeds r_f .

The quantity $\sigma(t)$ is assumed to vary linearly with time, $\sigma(t) = \sigma_0 + ct$. The critical grouping of parameters is

$$x(t) = \frac{r_f - \mu}{\sigma(t)},$$

which measures the gap between the sleeve and compact in standard deviations.

7.2.3. Data Analysis

As shown in Appendix D, the most probable value of the parameter $1/x_0$ appears to be about 0.250; when used with $p(168 \text{ days}) = 1.6 \times 10^{-4}$, the observed failure rates are predicted. The value of 1.6×10^{-4} for $p(168)$ yields a probability of 50% that 12 elements are broken at $t = 300$ days.

Two items of information from the postirradiation examination relate to the computational model described here. The first is the maximum diameter of the compacts measured in this element. The maximum measured value was 2.783 in. with a mean for that compact of approximately 2.772 in. If the initial mean diameter were the nominal value of 2.743 in. then the change in mean diameter would be about 0.030 in., or a change in radius of 0.015 in. This compares favorably with the values indicated above as being required for failure, namely about 0.01 in.

The diameters of 23 compacts were measured in detail and six individual readings for each compact were recorded. These data are shown plotted in a histogram in Fig. 7.4. The initial average diameter and its limits are also shown for comparison. The features of significance are that an equal number of compacts did not expand and contract although some mixed behavior is shown. The actual case shows a skew distribution with an increase in average radius as well as a large increase in the standard deviations. The median value for this distribution is 2.745 in. (50% of readings above and below) which give an increase in average diameter of 0.002 in. The standard deviation for the compact diameters, σ_d , for this distribution is 0.0125 in., or 0.00625 in. for the distribution in radii. If it is assumed that the sleeves did not change in dimension with irradiation, then the time dependence of σ_d is equal to the time dependence of $\sigma(t)$. Hence, a value of c can then be deduced from these data, namely 0.00225 per 168 EFPD, assuming the initial standard deviation in the radii of the compacts as high as 0.004. The calculated value for c that best fits the element failure rate is 0.000378 per 168 EFPD, a value much smaller than the one inferred from the measured data.

Some improvement in this comparison can be achieved with a modified interpretation of the hot-cell measurements. Two peaks occur in the distribution shown on Fig. 7.4, one with a mean of 2.745 in. and one around 2.7675 in. It can be argued that the second peak should be ignored; those compacts with values greater than 2.7625 in. (about two in this sample) could have caused the breakage of the sleeve. The other compacts then could be assumed to be typical of the rest of the core. The value of $2\sigma_d'$ obtained in this case is 0.0175 in., or 0.0044 in. for the standard deviation of the radius. The inferred value of c could then be as low as 0.0004, much closer to the best computational value of 0.000378.

Hence, it appears that the predictions of the statistical model are in marginal agreement with the results of other analyses, including the results of the postirradiation examination. The calculational results are very sensitive to small changes in the initial parameters, and small changes in these parameters could yield agreement with almost any failure rate observation. At present, the model is still basically empirical.

Fig. 7.4--Fuel compact diameter distribution after 168 EFPD. Results of a total of 138 measurements on 23 compacts from element C05-05

7.3. EVALUATION OF MATERIALS PERFORMANCE

This section covers the evaluation of materials components with respect to the postirradiation examination of elements E05-05 and C05-05. The role of the materials in the failure of C05-05 is discussed.

7.3.1. Irradiation Effects on the Fuel Element Components

A brief summary of the irradiation effects on fuel element components and the pertinent results from irradiation experiments conducted at Gulf General Atomic and elsewhere* is given below. This discussion duplicates to some extent that presented in Section 3 and is included here for completeness. Recent irradiation data on graphites not available at the time of the Peach Bottom fuel element design are also presented.

Crystal Structure in Graphite. The carbon atoms in a graphite lattice are arranged in hexagonal rings and are bonded together chemically in the basal or layer plane. The crystal structure is depicted in Fig. 7.5.[†] These planes are stacked together to form an anisotropic graphite crystallite. The short distance between atoms in the layer plane is one reason for the strong covalent bonding within the layer planes and the relatively high strength in the planar, or a, direction. The weak bonding between planes, due in part to their relatively large separation (6.71 Å), results in a relatively low strength between the layer planes and thus a relatively low tensile strength in the c direction.

In the manufacture of conventional commercial graphite, calcined petroleum coke and coal-tar pitch are used. The coke particles are semicrystalline and the pitch is amorphous. If the body is formed by extrusion, as in the case of the sleeves of the Peach Bottom fuel elements, the a-axes of the crystallites tend to align preferentially in the direction of extrusion; if the body is formed by pressing, as in the case of the fuel compacts, the a-axes of the crystallites tend to be aligned perpendicular to the forming (or axial) direction. The extent of which the bulk properties reflect the orientation of crystallites depends on the relative number of layer planes that align in any one direction. If the crystallites are randomly oriented, the contribution of crystallites is averaged and the bulk properties are isotropic. However, if the layer planes are oriented more in any one direction than in another, the bulk properties will show anisotropy.

Pitch contains about 50% volatile matter. In the subsequent operations of baking and graphitizing, the volatile matter is released, and the finished material consists of graphitized coke particles bonded with pitch coke. The graphite will contain considerable porosity resulting from release of the volatile matter during baking. In addition, the graphite crystallites are

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* W. V. Goeddel and J. N. Siltanen, Annual Review of Nuclear Science, Vol. 17, 1967, pp. 189-252. Also see Section 3 of this report.

[†] R. E. Nightingale, Ed., Nuclear Graphite, Academic Press, New York, 1962, p. 547.

Fig. 7.5--Crystal structure of graphite

far from perfect in their structure because of impurity atoms, lattice defects, dislocations, and microporosity.

Neutron irradiation tends to produce a contraction of the graphite crystallite in the a direction and an expansion in the c direction, as shown schematically in Fig. 7.6.* However, the bulk dimensions begin to contract immediately upon irradiation at all temperatures. The expansion of the crystallites in the c direction is initially accommodated by the microporosity, and therefore will not contribute to the bulk expansion of the body at low exposures. In fact, a net contraction is usually observed during this period. However, once the microporosity is "used up," the bulk dimensions in the c direction will cease to contract and start to expand.

Sleeve and Reflector Components. The irradiation-induced dimensional changes of a few bulk graphite bodies are illustrated in Figs. 7.7[†] and 7.8.** Figure 7.7 illustrates changes typical of needle-coke graphites, the type of graphite used for the Peach Bottom reflector. Figure 7.8 represents dimensional changes for impregnated graphites, the type of graphite used for Peach Bottom sleeves. The notations parallel (||) and perpendicular (⊥) refer to extrusion direction, or, in case of grade H-283 graphite, to the pressing direction. Since the maximum fast-flux exposure for sleeves is estimated to be 4.7×10^{21} nvt, the maximum longitudinal contraction, from Fig. 7.8, would thus be expected to be about 1%. Since the average temperature is less than that shown in Fig. 7.8, the actual contraction should be less than 1%.

Irradiation-induced contraction will induce stresses in the graphite sleeve due to flux gradients and temperature gradients. Analysis indicates that the strain rates are accommodated by irradiation-induced creep at stress levels below the structural limits of the graphite.

Fuel Compacts. Changes in the fuel compact dimensions occur during irradiation for several reasons, two of them being that after the monolithic laminar pyrolytic carbon coatings have failed, the fuel kernels tend to swell upon irradiation due to entrapped fission gases, and that the laminar PyC coatings tend to change dimension anisotropically in a fast-neutron flux. This latter effect is dependent on the original degree of orientation of the PyC polycrystalline material; the higher the degree of orientation of the crystallites in the coating the greater, in general, will be the dimensional change of the coating during irradiation. The coatings used in the fuel compact had a high degree of orientation in the as-produced condition, and during reactor life these coatings would therefore be expected to exhibit marked anisotropic dimensional changes. Also, since the single-layer coatings were applied directly to the fuel kernels, they are subjected to fission recoil

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* R. J. Price and J. C. Bokros, J. Appl. Phys., Vol. 63, 1965, pp. 1897-1906.

† G. B. Engle, Effect of Fast Neutron Irradiation from 495° to 1035°C on Nuclear Graphites, GA-6888, February 11, 1966.

** G. B. Engle, G. R. Tully, B. F. Disselhorst, and W. P. Wallace, Effect of Furfuryl Alcohol Impregnation on Irradiation Stability of Graphite, GA-4224 (Rev.), November 6, 1964.

Fig. 7.6--Schematic diagram of the dimensional changes produced in a graphite crystallite by fast-neutron irradiation: (a) before irradiation; (b) after irradiation

Fig. 7.7--Irradiation-induced changes for needle-coke graphites (Engle, GA-6888)

Fig. 7.8--Irradiation-induced dimensional changes for furfuryl-impregnated graphites (Engle, et al., GA-4224 (Rev.))

bombardment. The fission-fragment recoils damage the PyC coatings causing the phenomenon known as "spearhead" attack. The result of these effects is that, during irradiation, the coatings will shrink and distort around the expanding fuel kernel and eventually fracture because of the increase in stresses. This is shown schematically in Fig. 7.9.

The data on the effect of irradiation on fuel compacts are not nearly as extensive as those for graphite itself. However, some data were presented in Section 3, and additional data are described below.

The dimensional change of hot-pressed graphite fuel compacts tested in the GA2-284-6F2 capsule, and reported by Vanslager, *et al.*,* were +3% to +6% in the c-axis or longitudinal direction and 0% to +0.6% in the a-axis or radial direction. The irradiations were carried out at about 1300°C for more than 2600 full-power hr; the fraction of U²³⁵ that fissioned varied from 0.25 to 0.35, or 50% to 70% of the expected maximum burnup in a Core-1 element, and the corresponding fast-flux exposure varied from 1.7 to 2.5 x 10²¹ nvt.

In addition to the GA2-284-6F2 capsule, the GAIL III-B test conducted by Gulf General Atomic is pertinent. The GAIL III-B[†] experiment was similar to GAIL III-A (see Section 3) except that the test element was in the GETR test reactor for a longer period. The burnup of metal atoms (U + Th) ranged from 2.8% to 5.5%, with an average of 4.5%, which can be compared to an average of 6.9% for a depleted Peach Bottom element. The fast-flux exposure was a relatively low 0.9 x 10²¹ nvt, and the maximum temperature for the 15 compacts ranged from 1250° to 1750°C, with an average temperature around 1300° to 1400°C. The maximum sleeve temperature was 1125°C. The expansion along the longitudinal axis of the fuel compacts was 1.3% shortly after completion of irradiation. An expansion of 5% was measured in the unopened element after arrival at Gulf General Atomic; however, it was known that water had entered during the discharge from the reactor. Unfortunately, the fuel element was dropped during disassembly operations. Upon examination after the drop, seven of the fifteen fuel compacts had expanded an average of 9% in length with little change (<1%) in diameter.

The above experimental evidence indicates again that the major dimensional changes in the compact should be in the longitudinal rather than the radial direction. This longitudinal expansion was considered in the element design, and a large expansion space was provided above the fuel compact stack to accommodate the compact growth. The GA2-284-6F2 capsule data indicate, however, that radial swelling may be more important than originally believed, an observation which is, of course, consistent with results of the postirradiation examination.

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* F. E. Vanslager, G. B. Engle, D. R. Lofing, and C. S. Luby, Final Report on the GA2-284-6F2 Fuel Irradiation Experiment, GAMD-5925, February 16, 1965.

† C. S. Luby, H. S. Shoemaker, and G. R. Tully, Postirradiation Examination of the GAIL III-B Fuel Element, GAMD-5605, August 17, 1964.

Fig. 7.9--Drawing representing fuel particle with broken PyC coating. Arrows indicate direction of anisotropic dimensional changes of PyC coatings that occur during irradiation. Coatings contract in the circumferential direction and expand in the radial direction. Cross-hatched lines in kernel indicate the swelling of the kernel caused by entrapped fission gases and other irradiation effects. Fission-fragment recoil damage ("spear-head" attack) is illustrated in inner surface of pyrolytic-carbon coating. Volume change in the fuel compact results from the swelling of the kernel and from the distortion of the PyC coating

7.3.2. Discussion and Correlation of Results in the Hot-cell Examination of E05-05 and C05-05

Graphite Sleeve. The effects of irradiation on the graphite sleeve material, as discussed above and in Section 3, should pose no serious problems to the fuel element integrity. The results of the postirradiation measurements of the dimensions of both the E05-05 and C05-05 sleeves tend to confirm this prediction, since no significant changes from the assembled dimensions could be determined. However, the analysis of the C05-05 fuel element was complicated by the fact that it was probably damaged during the examination at the site. In particular, the 180° rotation and subsequent 2- and 3-in. drop of the top part of the element apparently caused additional damage to the sleeve and some of the compacts. The following pertinent information substantiates this hypothesis:

1. The fact that the failed fuel element removal tool could be removed in the canning machine indicated that the initial sleeve cracks were probably less than 1/16 in. wide.
2. The television pictures taken at the site, and later compared with the reproduced fractured section photographs, did not indicate the frontal cracks observed in the first hot-cell exposure.
3. The lateral forces generated as a result of the vertical drop could have been sufficient to break up parts of compacts 13 and 14 and create the additional cracks in the sleeve noted above. The upper and lower portions of the sleeve, 180° out of phase, made contact during the 2- to 3-in. drop on only one side of the sleeve. The resultant horizontal forces, being partially restrained by the middle spine seated in the spine hole of the lower fuel element section, caused additional fracturing and breaking of compacts. The drop itself could have occurred after the act of rotation had enabled the middle spine to drop into the vacant central spine hole. The fact that no fuel compact residue was found in the spine cavity between the bottom and middle spine indicates fractures of the compacts occurred at the end of the drop.

Item 3 above could also be pertinent to the fact that the middle and top spine segments were displaced about 5 in. vertically. When the upper portion of the fuel element was being lowered after the television examination at the site, the middle spine could have contacted a compact (probably compact 13) with the weight of the upper portion of the fuel element and twisting tool (about 100 lb) causing the middle and upper spines to be shoved upward until stopped by the upper reflector. This is believed to have been the condition of the element when it was found to be too high to place the cap on the canister. During the rotation of this upper portion of the element, the spine was rotated in a nonconcentric manner until it fit into the central hole of the fuel compact. It is important to note that the compact edges were strong enough to resist spalling during the rotation.

In the region away from the fracture, the outside diameter of the cylindrical sleeve section of the C05-05 element was found to be within the specified tolerance band. Outside diameter measurements of the E05-05 sleeve were also within the manufactured tolerance.

The inside diameter could not be measured directly, but after the disassembly of the C05-05 element, these dimensions could be deduced from the wall thicknesses that were measured at selected angular positions. The inside diameter dimensions were found to be within tolerance. However, the wall thickness measurements revealed that at some sections of the sleeve an eccentricity existed between the axis of the drilled hole and the axis of the sleeve outside surface. This condition resulted in a variance between the maximum and minimum thickness of 0.344 to 0.391 in., or a difference of about 0.046 in. The eccentricity was attributed to radial drift during manufacture of the deep boring tool over the midlength of the sleeve.

The region of greatest wall thickness variation was along the middle portion of the graphite sleeve. However, the wall thickness dimensions near

the prima longitudinal crack did not indicate that the thinner portion of the wall was involved in the failure of the sleeve. This point was confirmed by work at LASL (see Section 6), which indicated that the "prime crack" was located in the thicker portion of the sleeve.

Measurements performed on E05-05 indicated a bow of about 0.14 in. This measurement was, of course, impossible to make on the entire C05-05 element because of its fractured condition. Measurements were made on short segments of the upper sleeve and no evidence of bowing was observed. There were no mechanical property tests performed on the sleeve material. Previous irradiation studies had indicated that significant changes in the physical properties should not occur for the dose that the material received. Stereographic examination of the cracked region of the C05-05 sleeve did not reveal any flaws or abnormalities that could have been responsible for the failure.

Density and porosity measurements were performed at LASL on both unirradiated and irradiated sleeve material, and no significant differences between the two materials were found.

In summary, the examination of the graphite sleeves of both E05-05 and C05-05 indicated that they had not undergone any detrimental dimensional changes due to irradiation in the reactor and that the massive fracture of C05-05 was not a result of any defects or abnormalities in the sleeve.

Graphite Spine. The graphite spine was manufactured from Speer 711GSXY graphite, and even though this exact material has not been tested in an irradiation capsule, it was expected from tests on a similar grade of graphite that the spine column would experience a decrease in volume during irradiation. The dimensional measurements made in the Gulf General Atomic hot cell show that the upper and middle spine section contracted in both the diametral and longitudinal direction, i.e., 29.81 to 29.69 in. in length and 1.733 to 1.720 in. in diameter. The lower section exhibited approximately the as-produced dimensions. The appearance and performance of the spines were judged to be satisfactory.

Fuel Compacts. As described above and in Section 3, the fuel compacts were expected to experience significant axial expansion but no appreciable radial expansion. It is clear, however, that significant expansion also occurred in the radial direction, at least for some of the compacts. The results of one test showed that it required considerable force to push the fuel compacts out of the C05-05 sleeve, a graphic indication that significant radial growth had occurred.

The metallographic examination indicated that most of the particles were broken or damaged in all compacts. For compact 18, about 84% of the particles were broken, and this compact was believed to have exhibited the greatest amount of expansion in the radial direction.* On the other hand, compact 25 was in apparently good condition and showed little, if any, radial

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* Axial dimensional measurements were not recorded for individual compacts since the tolerances in this dimension were not critical to the performance of the element.

expansion; the number of broken coatings in this case was only 45%. Compacts 10 and 27 were considered to be in an intermediate state between 18 and 25, and the number of broken particles in each of these two compacts was also found to fall between those of 18 and 25. The rough correlation between compact condition and fraction of broken particles is qualitatively consistent with the hypotheses that cracked coatings were the principal cause of compact (radial) expansion.

An interesting result reported by LASL also relates the exposed fuel to the degree of compact swelling. Their results show that samples which had the largest exposed fuel mass also exhibited the greatest degree of compact swelling (compacts 18 and 19). These results reinforce the belief that coating failure and kernel swelling are the principal causes of radial growth in the compact at these low exposures.

The metallographic examination also indicated the presence of micro-cracks in the matrix material. These cracks were believed to have developed when the coatings cracked and became distorted, and indicate the strong disruptive forces generated by the coating distortion. The cracks typically ran between two or more particles and were generally randomly oriented. Radial growth of the compact would be another symptom of these processes.

The coatings were known to have had varying degrees of anisotropy. It would be expected that the more oriented coatings would undergo greater distortion at lower exposures than the less oriented coatings. Therefore, a time element must be considered in compact performance. Compacts containing many particles with highly oriented coatings would tend to exhibit dimensional changes earlier in life than compacts that contain particles with more isotropic coatings. It is not surprising, therefore, that different compacts (and fuel elements) may show a quite different behavior to the same neutron exposure.

Fuel compact 18 is considered the principal factor in the sleeve failure of C05-05. One of the main reasons for this conclusion is related to the "shadowing" phenomenon that occurred on the inside of the graphite sleeve opposite compacts 19 through 30 (see Section 6). A black deposit was noted along the inside of the graphite sleeve when it was cut open to allow removal of the fuel compacts. The vertical and horizontal markings in the shadowed area corresponded exactly to the location of the purge grooves and chamfered edges, respectively, of the adjacent fuel compacts. Since the purge grooves of the compacts were not exactly aligned with one another, the change in the axial position of the compacts could be inferred from the overlap of these shadowed regions. This overlap was constant, about 0.5 in. for each compact. Compacts 1 through 18, however, showed a relatively small and almost insignificant degree of shadowing. The significance of these observations is not fully known, but it seems clear that the compacts above compact 18 had more freedom to move than did those below compact 18. This was tentatively interpreted to indicate that compact 18 was binding against the graphite sleeve and thus inhibiting the movement of the lower compacts. Significant hoop stress in the sleeve could have been generated by the expansion of compact 18 which, when coupled with axial tensile forces generated by expanding lower compacts, was sufficient to fracture the sleeve.

7.3.3. Conclusions

The results of the hot-cell examination of fuel element C05-05 support the conclusion that the radial expansion and consequent binding of fuel compact 18 against the sleeve was instrumental in the failure of the C05-05 fuel element. The predominant factor in this binding and the ultimate fracture of the sleeve is believed to be the expansion of the coated particles caused by the distortion and cracking of the PyC coatings. Since these coatings are anisotropic and monolithic, they are susceptible to irradiation-induced dimensional changes as well as fission-product recoil damage and would eventually break. The broken coating would not provide a restraining force around the fuel kernels, and the kernels would be free to swell from the pressure of the entrapped fission gases. The hoop stresses created by the bound compact plus the tensile stresses resulting from restraining the lower fuel compacts were evidently sufficient to cause the fracture of the C05-05 sleeve.

The current analysis did not yield sufficient information for the prediction of the radial growth rates of the compacts, and therefore it is not possible to accurately forecast how many and at what rate other fuel elements might fail. Pertinent data in this regard may result from future postirradiation examinations on both failed and unfailed elements.

Appendix A

FABRICATION PROCEDURE FOR FUEL COMPACTS

Appendix A
FABRICATION PROCEDURE FOR FUEL COMPACTS

A batch of fuel compact material would yield 18 fuel compacts. The following procedure was followed for each batch.

1. Add the following to an 8-qt Vee blender and blend for 10 min:
  - a. 200 g of GP-38.
  - b. 908 g of fuel particles.
  - c. Rhodium, if required (9.26 g for B-type compacts and 3.09 g for C-type compacts).
2. Blend dry for 10 min, and carefully load into a 30-qt Hobart bowl.
3. Repeat steps 1 and 2 above.
4. Add the following to a 1-gal-capacity low-speed Waring blender and mix for 1 min:
  - a. 3 liters of Triclene.
  - b. 540 g of pitch.
  - c. 36 g of ethylcellulose.
5. Add the contents of blender to a Hobart bowl.
6. Rinse the Waring blender by adding 500 ml of Triclene and mix for 30 sec, and add the contents to Hobart bowl.
7. Run the Hobart for 5 min using pastry mixing blade.
8. Repeat step 7, using a flat mixing blade.
9. Mix for 7 min with a wire whip.
10. Load agglomerates into pans and dry for 6 hr at 160°C.
11. Weigh batches for material accountability and determine whether the Triclene has all evaporated.
12. Weigh eighteen 355-g charges for pressing into 18 compacts.
13. Gamma-count the charges for U content prior to pressing into compacts.

In the pressing process, the pelletized material was loaded into the die at a temperature of 1000° to 1050°F and pressed for 20 min with the temperature of the die going to 1400°F. The compact was then ejected and cooled.

In the sintering process, the compacts were heated to 1800°C and held for one hour in a vacuum. The furnace was cooled to 200°F before unloading. The completed compacts were inspected for size, fuel uniformity, condition of the particles, thermal expansion characteristics, thermal conductivity, and density.

Appendix B

SPECIAL HOT-CELL FIXTURES AND EQUIPMENT FOR POSTIRRADIATION

EXAMINATION OF E05-05 AND C05-05

Appendix B

SPECIAL HOT-CELL FIXTURES AND EQUIPMENT FOR POSTIRRADIATION EXAMINATION OF EO5-05 AND CO5-05

The disassembly and examination of a Peach Bottom element required the design and manufacture of special equipment for use in the high-level cell.* Guidelines for the equipment necessary for the evaluation of the fuel element were established, and the following equipment was designed and constructed:

1. Wall Frame Supports. Four C-frame brackets were attached to the back hot-cell wall. These brackets were used to support the rails for the saw head frame and were the support arms for the table section.
2. Table with Positioning Brackets. A metal table, 16 ft long by 14 in. wide by 7/8 in. thick, was positioned and leveled on the C-frame brackets attached to the back wall of the cell. It was aligned in two coordinates with the saw travel through the X and Y axes. It was leveled and aligned with the saw head mechanism by a transit in the cell to ± 0.010 in. for the full length of the saw. The positioning brackets consisted of Vee rollers sliding in a wide Tee slot that could clamp the Vee rollers in place. The elevation of the Vee rollers was coincident with the hole in the high-level cell door. When a canister was discharged from the cask, it was aligned with the rollers.
3. Saw Travel Attachments and Operating Console. A saw head, mounted on rails, could be driven by a variable-speed electric motor through a rack and pinion gear along the full length of the table. It was supported on the rails by four bearings. The rails, which were mounted on the upper arms of the C-frames, were aligned with a transit to ± 0.010 in. A control console outside the high-level cell was connected to the saw by electrical leads that penetrated the cell through a wall plug. The cut-off saw head motor, the vacuum cleaner, and the saw head drive motor were controlled from the external control console.
4. Saw Head Mechanism. The need to cut the element radially and axially required a saw device that could be rotated through 90° . The saw positioning mechanism was designed for a 6-in. travel in the X, Y, and Z directions. The saw was powered by a Dayton 0.4-hp electric motor. It was designed for the airframe industry, but its small size and high speed were ideal for this use. The saw blade was a bonded abrasive disk. The saw head could accommodate 5-in. to 12-in. blades. A vacuum cleaner was attached to the saw head frame to collect the cutting debris from the saw.

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*The largest device previously disassembled was the GAIL IV fuel element, which was 4 ft long. The equipment used here was too small to handle a Peach Bottom element.

5. Door Plug Removal Device. The spent fuel cask was too long to fit into the low-level cell, and it was necessary to discharge the element with the low-level cell door open. This was accomplished by remotely pulling the door plug in the high-level cell door. A counterbalance device was constructed and threaded into the door plug, to permit a gentle removal of the door.
6. Purge Boxes. Because the fuel is susceptible to hydrolysis, it was necessary to protect the fuel element as much as possible from exposure to the hot-cell atmosphere. The following plastic nitrogen-purged boxes were made to provide a working atmosphere with less than 5 ppm of moisture:
  - a. A large purge box to cover the entire C05-05 element (Fig. B.1).
  - b. A small purge box to cover the fracture section of C05-05 (Fig. B.2).
  - c. A segment purge box to cover the segments removed from the element onto the window table for disassembly.
  - d. A disassembly and photo purge box for photographing the compacts and disassembling the compacts from the spine (Fig. B.3).
  - e. A dimensioning purge box (Fig. B.4).
  - f. A small box for sectioning individual fuel compacts for metallography specimens (Fig. B.5).
7. Coring Tool Drills. Core drills were made to take samples from the sleeve, spine, and reflector sections. The coring drills were operated in the PAR manipulator.
8. Bowing Device. To measure the bowing of a fuel element, a reference transit line was established from the operating gallery to the low-level cell. This transit line was parallel to the element in its position on the table-mounted Vee block. The transit line became a reference for a vertical indicator which was moved on the table until it touched the element. The indicator was then made perpendicular to the tangent point on the element. A scale at the top of the indicator was used to measure the deviation from the transit line. By sighting several points along the length of the element, a bow measurement was obtained to ± 0.010 in.
9. Dial Gages. The following measurements were made using dial gages:
  - a. Element outside diameter.
  - b. Element sleeve thickness.
  - c. Compact outside diameter.
  - d. Compact inside diameter.
  - e. Compact wall thickness.
  - f. Compact length.

A reference standard was made for each dimension to be measured to establish a zero for the dial gage. The deviation from this zero was recorded when the particular parts were measured.

Appendix C

HYDROLYSIS RATE OF FUEL COMPACT IN HOT-CELL EXAMINATIONS

Fig. B.1--Large clear plastic inert gas storage box used during nonworking hours

Fig. B.2--Special purge box covering highly fractured area

Fig. B.3--Special purge box for viewing and photographing sectioned portions of fuel element

Fig. B.4--Special purge box for dimensional measurements of fuel compacts

10. Compact Storage and Transfer Containers. A bank of 30 compact storage containers was built to keep the individual compacts under nitrogen-gas purge after the compacts were removed from the element sleeve (Fig. B.6). When the compacts were transferred from one area to another, they were placed in the individual canisters. The transfer canisters had poppet valves which opened at 0.33 psi. They were purged through a shutoff quick-disconnect coupler. When transferred, there was $1/3$ psi of N_2 pressure in the canister.
11. Purge Caps and Collar. When the spent fuel canister was received at the hot cell, it was leak-checked. This was done by placing a purge collar at one end of the canister and penetrating the canister through a special fitting with a $1/4$ -in. drill. The special fitting was used to pressurize the canister. After the canister ends were girdle-cut, purge caps were placed on the top and bottom reflectors, and dry N_2 gas with less than 5 ppm of H_2O was purged through fittings in the caps. The gas flowed over the fuel compacts while the compacts were still in the graphite sleeve. As the element was being disassembled, purge caps were placed over the open end sections of the graphite sleeve that contained fuel compacts.
12. Cask Plug Valve Open-and-close Device. The spent fuel shipping cask that was used to bring the fuel elements to the cell had a large plug valve at the discharge end of the cask. The cask was loaded at the reactor site with the plug valve under water. A small cable drum winch system operated above water was used to open and close the plug valve. The same unit was used to open the plug valve when the element was discharged at the hot cell.
13. Cask Canister Removal Device. When the cask was positioned at the hot-cell door, a ratcheting device was attached to the grapppler end of the cask. The operator stood to one side of the cask and operated a lever arm that ratcheted the element into the high-level cell.
14. Gamma-scan Tables. Two tables were constructed for gamma-scanning, one for scanning the element length, the other for scanning compacts. The high-level cell is 18 ft long; the element in the canister was 13 ft long. To scan the length of the canister, it was necessary to open the high-level cell door and push the canister into the low-level cell. A support table was made for the low-level cell to support the canister as it was pushed into the cell. The compact scanning table was designed to receive the transfer canister and rotate it through 360° . The scanning plane could be changed through 3 in. of elevation. The table face could be tilted at 90° to scan the axial position of a compact (Fig. B.7).
15. Removable Table Panels. There was a 24-in. space between the positioning table and the window working surfaces. The area was covered with removable table panels, giving the operating personnel additional working surface. These panels could be removed for access to the storage holes in the floor or storage of components under the working table surfaces.

Fig. B.5--Inert-gas purged fuel compact sample sectioning box for metallography specimens

Fig. B.6--Inert-gas purged aluminum canisters for individual fuel compact storage

Fig. B.7--Individual mobile fuel-compact canister on rotating gamma-scan table

Appendix C

HYDROLYSIS RATE OF FUEL COMPACT IN HOT-CELL EXAMINATIONS

The maximum hydrolysis of a typical Peach Bottom fuel compact in the purified nitrogen purge gas is estimated below. The N₂ contained 3 ppm, or less, H₂O.

The predicted reaction is

Hence, one mole of H₂O will react with 1/2 mole of metal carbide. The nitrogen gas flow rate is 0.2 ft³/hr in the purge system under study, corresponding to 135.7 liters/day. This transports 2.57×10^{-5} moles of H₂O per day. Since there are about 260 g of metal carbide per mole, only 1.28×10^{-5} moles/day could react with the moisture in the nitrogen. This is 0.0334 g of metal carbide, or about 0.05% of the carbide in one compact.

Appendix D

STATISTICAL MODEL OF FUEL ELEMENT FAILURE

Appendix D
STATISTICAL MODEL OF FUEL ELEMENT FAILURE

MODEL DEVELOPMENT

Let $p(t)$ be the probability at time t that a compact chosen at random has a radius in excess of the failure radius r_f . A sleeve failure would then result. The assumption that a fuel element failure occurs when one compact grows excessively is conservative, but only slightly so, for this analysis. It is possible, of course, to have two fast-growing compacts within one rather than two elements, but for analyses pertinent to Core-1, this effect is small. Let μ and $\sigma(t)$ be the average radius and the standard deviation of radii, respectively. Then

$$p(t) = \int_{r_f}^{\infty} \frac{1}{\sqrt{2\pi}} \frac{1}{\sigma(t)} \exp \left[-\frac{(r - \mu)^2}{2\sigma(t)^2} \right] dr = \int_{x_f(t)}^{\infty} \frac{1}{\sqrt{2\pi}} \exp \left[-\frac{x^2}{2} \right] dx$$

where $x_f(t) = (r_f - \mu)/\sigma(t)$.

Let $P_{n,m}(t)$ be the probability that exactly m compacts of the total n compacts ($n = 24,120$) have exceeded r_f at time t . Then

$$P_{n,m}(t) = \frac{[p(t)]^m [1-p(t)]^{n-m} n!}{m! (n-m)!} .$$

The probability that m or more compacts have exceeded r_f at time t is $P_{n,m>}(t)$.

$$P_{n,m>}(t) = P_{n,(m-1)>}(t) - P_{n,m-1}(t) .$$

Hence $P_{n,m>}(t)$ depends only on $p(t)$. Let m' be the number of failures that will occur with a 10% probability, and m'' the number of failures with a 90% probability. This range of failures then forms the 80% confidence limit, that is

$$P_{n,m'>}(t) = 0.1 \quad P_{n,m''>}(t) = 0.9 .$$

and it is 80% certain that the number of failures at time t will be in the range from m'' to m' .

If the time dependence of $\sigma(t)$ is specified, the problem is uniquely determined. In this study it was assumed that $\sigma(t)$ varies linearly in time, that is, $\sigma(t) = \sigma_0 + ct$, as shown on Fig. D.1.

If r_f , μ , σ_0 , and c were accurately known, this analysis would be complete. That is, $x_f(t)$ would be specified and since it uniquely determines $p(t)$ and $P_{n,m>}(t)$, the 80% confidence range m'' to m' for each time t could be determined and used to predict (with 80% confidence) the number of failures for each time t .

These parameters are not well known, however, but observed failure rates can be used to improve the accuracy of future predictions. This is done in the following manner:

1. First, a range of values of $p(t = 168 \text{ days})$, the probability that any given compact has a radius in excess of r_f at the end of 168 days, is chosen which predicts a failure probability of three or more in the range $P_{n,3>}(168) = 0.1$ to 0.9 . (Note that three failures actually occurred during the first 168 days and nine more during the next 132 days.) Such a curve has been determined and is shown on Fig. D.2.
2. Second, a reasonable range of $1/x_0 = \sigma_0/(r_f - \mu)$ is chosen from a knowledge of σ_0 , r_f , and μ .
3. Third, the 80% confidence limit curves are obtained for each set of values of $p(168)$ and $1/x_0$ for subsequent times.
4. Finally, the scope of acceptable values of $1/x_0$ and $p(168)$ are narrowed by looking at the predicted failures at $t = 300$ days. The total number of failures at $t = 300$ is 12. Hence, any set of values of $1/x_0$ and $p(168)$ that predict a much different number of failures at $t = 300$ can be ruled out even though they are allowed by the uncertainty in r_f , μ , σ_0 , and c .

Note that the failure rate is uniquely determined by just two parameters, $1/x_0$ and $p(168)$, since

$$\frac{1}{x_f(t)} = \frac{\sigma_0}{r_f - \mu} + \frac{ct}{r_f - \mu} = \frac{1}{x_0} + \frac{c}{r_f - \mu} t$$

and it can be easily shown that $1/x_0$ and $p(168)$ uniquely determine $c/r_f - \mu$.

NUMERICAL RESULTS

The appropriate range of values for $1/x_0$ is bracketed by 0.200 and 0.275. The low value (0.200) gives a breakage rate much too high for all values of $p(168)$. This judgement depends on the fact that 12 elements failed by $t = 300$ days. The maximum value of $1/x_0$ is derived by considering the initial tolerances and requiring that c be positive, that is, that the distribution becomes broader after irradiation, it does not shrink.

Fig. D.1--Compact radius distribution

Fig. D.2--Single failure probabilities versus probability of three or more failures

Fig. D.3--Number of failures for which $P_{n,m>}(t) = 0.5$ as a function of time

Figure D.3 shows the number of failures m for which $P_{n,m>}(t) = 0.5$ as a function of time. That is, for a given set of values of $1/x_0$ and $p(168)$, m or more failures will occur at time t with a 50% probability. Figure D.3 shows the probable number of failures for all combinations of $1/x_0$ and $p(168)$ which agree with the failure rate to $t = 300$ days. If the statistical model described previously truly represents the physical situation, then the two curves $1/x_0 = 0.225$, $p(168) = 8 \times 10^{-5}$ and $1/x_0 = 0.257$, $p(168) = 1.9 \times 10^{-4}$, should indicate the limits of the failure rate. The value of $p(168) = 1.6 \times 10^{-4}$ corresponds to $P_{n_{12}>}(300) = 0.5$, the most probable value, for a total of 12 elements broken at 300 days.