95
4-27-77
D. 955

ORNL-5214

Distribution of Fission Products in Peach Bottom HTGR Fuel Element E11-07

R P Wichner
F F Dyer
W J Martin
L C Bate

MASTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$7.50; Microfiche \$3.00

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Energy Research and Development Administration/United States Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

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

Contract No. W-7405-eng-26

CHEMICAL TECHNOLOGY DIVISION

HTGR BASE TECHNOLOGY PROGRAM
HTGR Chemistry Studies (189a 0329)

DISTRIBUTION OF FISSION PRODUCTS IN PEACH BOTTOM HTGR
FUEL ELEMENT E11-07

R. P. Wichner
F. F. Dyer*
W. J. Martin
L. C. Bate*

*Analytical Chemistry Division.

Date Published: April 1977

— NOTICE —
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	1
1. INTRODUCTION	1
2. FUEL-ELEMENT DESCRIPTION	4
2.1 Design Dimensions and Specifications	4
2.2 Operating History — Power, Temperature, and Fluence	15
3. PHOTOGRAPHIC DESCRIPTION AND METROLOGY	16
3.1 Visual and Photographic Examination	16
3.2 Metrology of Sleeve, Spine, and Fuel Compacts	60
4. DISTRIBUTION OF GAMMA-EMITTING NUCLIDES	73
4.1 Equipment and Methods	73
4.1.1 Measurement of gamma emitters in fuel compacts ...	73
4.1.2 Measurement of axial distributions of gamma emitters in the sleeve, spine, and bottom reflector	74
4.2 Axial Distribution of Gamma Emitters	80
4.2.1 Fuel compacts	80
4.2.2 Axial distributions of gamma-emitting nuclides in sleeve and spine	89
4.2.3 Axial distributions in the upper and lower reflector regions	102
4.3 Radial Distributions in the Sleeve and Spine	105
4.3.1 Facility for remote acquisition of radial graphite samples	105
4.3.2 Radial distributions of gamma emitters in the sleeve and spine	111
5. BURNUP ANALYSES AND BETA-EMITTING NUCLIDE DISTRIBUTIONS	128
5.1 Burnup Determinations	128
5.2 ^3H , ^{14}C , and ^{90}Sr Determinations in Fuel Particles	129
5.3 Radial Distribution of ^3H , ^{14}C , and ^{90}Sr in Sleeve and Spine	132

	<u>Page</u>
6. FUEL PARTICLE EXAMINATION	148
6.1 Metallographic Examination of Fuel Compacts	148
6.2 Failed Particle Fraction Measurements	179
6.2.1 Test of method on archive compact. Determination of FPF and "tramp fuel" level in archive compact	180
6.2.2 Failed particle fraction in Ell-07 fuel compacts	182
7. REFERENCES	184

DISTRIBUTION OF FISSION PRODUCTS IN PEACH BOTTOM HTGR
FUEL ELEMENT ELL-07

R. P. Wichner
F. F. Dyer
W. J. Martin
L. C. Bate

ABSTRACT

This is the second in a projected series of six post-irradiation examinations of Peach Bottom High-Temperature Gas-Cooled Reactor driver fuel elements. Element Ell-07, the subject of this report, received an equivalent of 701 full-power days of irradiation prior to scheduled withdrawal.

The examination procedures emphasized the determination of fission product distributions in the graphite portions of the fuel element. Continuous axial scans indicated a ^{137}Cs inventory of 17 Ci in the graphite sleeve and 8.3 Ci in the spine at the time of element withdrawal from the core. In addition, the nuclides ^{134}Cs , $^{110\text{m}}\text{Ag}$, ^{60}Co , and ^{154}Eu were found in the graphite portions of the fuel element in significant amounts. Radial distributions of these nuclides plus the distribution of the beta emitters ^3H , ^{14}C , and ^{90}Sr were obtained at six axial locations, four within the fueled region and one each above and below. The radial dissection was accomplished by use of a manipulator-operated lathe in a hot cell. These profiles reveal an increased degree of penetration of ^{134}Cs relative to ^{137}Cs , evidently due to a longer time spent as xenon precursor.

In addition to fission product distribution, the appearance of the element components was recorded photographically, fuel compact and graphite dimensions were recorded at numerous locations, and metallographic examinations of the fuel were performed.

1. INTRODUCTION

A series of postirradiation examinations of driver fuel elements withdrawn from the Peach Bottom High-Temperature Gas-Cooled Reactor (HTGR)

is being undertaken as a part of a surveillance program for this reactor being carried out in conjunction with the General Atomic Company (GAC). The overall objective of the surveillance work is the verification of design procedures used for predicting fission product and temperature distributions in an HTGR.

This report presents results of the examination of fuel element Ell-07, which received 701 equivalent full-power days of irradiation. Emphasis was placed on determining fission product distributions in the element, especially as a result of migration to the graphite sleeve and spine from the fuel compacts. Inventories of gamma-emitting nuclides in the fuel were obtained by scanning each of the 30 compacts individually. Distributions of gamma-emitting nuclides and the beta emitters ^3H , ^{14}C , and ^{90}Sr were obtained for the graphite portions of the element by axial scanning and radial dissection in a hot cell.

In addition to the measurement of fission product distributions, examinations in the following general categories were performed:

1. The appearance of the component parts of the element was photographically recorded.
2. Critical dimensions, such as the inner and outer sleeve and fuel compact diameters, were measured. The recorded metrology data form an essential part of the effort to validate the predictive temperature codes for HTGRs.
3. Portions of four fuel compacts were examined metallographically.
4. The failed fraction of fuel particles in seven compacts was determined by a hot-chlorine leach procedure.

General information regarding the construction and operation of driver element Ell-07 was obtained from GAC. Portions of the engineering drawings supplied by GAC were redrawn and included in Sect. 2, which is devoted to fuel-element description. Other general descriptive material was obtained from the Peach Bottom HTGR Final Hazards Summary Report (FHSR),¹ the core 2 supplement to the FHSR,² a report by Turner et al.³

which discussed the fuel performance of core 2 after 630 equivalent full-power days (EFPDs), and an internal GAC memorandum⁴ which details the irradiation exposure and the first estimate of the thermal history. A more recent estimate of temperature distributions within the fuel element, which takes into account measured dimensional changes and improved flow and power shape information, is provided by ref. 5. Refined estimates of internal fuel element temperatures, which include the most recent PIE information, are being prepared at GAC, and hence the values presented in Sect. 2 must be regarded as preliminary.

As currently envisaged, the fuel-element examination portion of the Peach Bottom Surveillance Program currently in progress includes one 385 EFPD element which has been reported,⁶ one 701 EFPD element reported here, and four end-of-life elements which received 900 EFPDs of reactor service. The surveillance program at ORNL has also been concerned with the measurement of fission product distributions in the primary coolant circuit during the operation of core 2, from 1971 through most of 1974. Results from these measurements have been reported by F. F. Dyer, R. P. Wichner, et al.⁷

In addition to providing fuel-element construction drawings and a comprehensive data package, GAC assisted the examination work described here by supplying funds to partially cover the physical examination procedures described in Sect. 3. These funds also enabled improvement of tools employed for metrology and hence also provide benefit to subsequent fuel-element examinations.

The authors wish to gratefully acknowledge the support provided by the Analytical Chemistry Division, particularly to personnel in the Reactor Projects group headed by D. A. Costanzo and the Radiochemical Analyses group headed by R. A. Rickard. J. A. Botts developed the useful method for determining failed particle fractions of fuel compact samples. Hugh Parker and P. S. Gouge performed the beta-emitter analyses on the graphite samples. W. F. Rogers skillfully operated the lathe used to acquire the radial graphite specimens in a hot cell.

2. FUEL-ELEMENT DESCRIPTION

2.1 Design Dimensions and Specifications

Fuel element Ell-07 was one of the 56⁴ so-called "light-rhodium" elements of Peach Bottom core 2 and, as such, fell into the class comprising the majority of the 80⁴ driver elements. The reader is referred to ref. 6 for a description of the four basic types of Peach Bottom driver fuel elements wherein descriptive material from the Final Hazards Summary Report¹ and its revisions² are summarized.

The fuel-element assembly drawing shown in Fig. 2.1-1 and the more detailed section of Ell-07 shown in Figs. 2.1-2 to 2.1-6 are given here because of minor differences between Ell-07 and a similar set of figures given in ref. 6 for element EO6-01. The primary difference between these two elements is that EO6-01 possessed thermocouples in two of its three spine pieces, whereas Ell-07 is not instrumented. Hence the three spine pieces are solid cylinders, and the 1/2-in.-long endpiece on EO6-01 for electrical terminals is absent here. For a general summary of the purpose and function of element components, refer to Sect. 2.1 of ref. 1.

An additional difference between these two light-rhodium elements is that preirradiation dimensional data were obtained from Ell-07 prior to irradiation service, enabling accurate determination of dimensional changes due to irradiation service. These data specifically relating to element Ell-07 are summarized in Sect. 3.

Specifications for the various graphites comprising the fuel-element components, given in Sect. 2.2 of ref. 1 for element EO6-01, apply equally to element Ell-07. Also, the specifications for the BISO-coated particles given in Table 2.2 of ref. 1 are universally applicable to all core 2 driver elements. Beginning-of-life (BOL) and end-of-life (EOL) heavy-metal loadings for Ell-07, calculated using the GAUGE code,⁴ are given in Table 2.1-1.

Impurity levels in the sleeve and spine could be of significance in the interpretation of observed radionuclide distributions in the graphite. Measurements of impurity levels for an archive sleeve specimen (AQ No. 3290)

Fig. 2.1-1. Peach Bottom core 2 driver fuel element assembly, element Ell-07.

ORNL DWG 75-5037 R1

Fig. 2.1-2. Top reflector assembly.

Fig. 2.1-3. Details of Peach Bottom core 2 driver element sleeve.

ORNL DWG 75-5032

Fig. 2.1-4. Peach Bottom core 2 fuel compact.

ORNL DWG 75-15193

Fig. 2.1-5. Details of Peach Bottom core 2 Ell-07 driver element spine pieces; 1 of 3.

Fig. 2.1-6. Details of internal fission product trap.

Table 2.1-1. Heavy-metal loadings for Ell-07^a

Nuclide	BOL (0.0 EFPD) nuclide weights (g)	EOL (701.2 EFPD) nuclide weights (g)
Th-232	1373.757	1324.348
Pa-233	0	0.140
U-233	0	30.655
U-234	0	2.626
U-235	0	0.307
U-234	3.738	2.571
U-235	232.451	105.816
U-236	1.251	25.539
U-238	12.110	10.641
Np-237	0	1.832
Np-239	0	
Pu-238	0	0.421
Pu-239	0	0.328
Pu-240	0	0.104
Pu-241	0	0.147

^aFirst five rows pertain to inventories due to Th-232.

were reported in Table 2.5 of ref. 6. Some of these measurements have been duplicated, and also, specimens from two additional archive sleeves (QA Nos. 1010 and 1716) were analyzed for microconstituents. These results are listed in Table 2.1-2. Impurity levels reported in Table 2.1-2 were obtained using spark source spectrometry, except for Th and U which were obtained by neutron activation analysis. Table 2.1-3 records a more complete list of trace elements found in Peach Bottom archive sleeve graphite obtained using neutron activation analysis exclusively.

Table 2.1-4 lists measured levels of impurities in the archive sleeve (QA No. 1716) as a function of radial location in the sleeve specimen. Each value given is an average of two determinations (performed by neutron

Table 2.1-2. Trace-element levels (ppm) in Peach Bottom
archive sleeve specimens^a

Element	QA number		
	3290	1010	1716
B	2.0	6.0	2.0
Ba	~15.0	~15.0	~15.0
Be	<1.0	<1.0	<1.0
Br	<10.0	<10.0	<10.0
Ca	20.0	5.0	5.0
Co	<5.0	<5.0	<5.0
Cs	<20.0	<10.0	<10.0
Eu	<30.0	<30.0	<30.0
Fe	150.0	5.0	150.0
I	<10.0	<10.0	<10.0
K	10.0	10.0	30.0
Li	<1.0	<1.0	<1.0
Sr	<10.0	<10.0	<10.0
Th	0.178	0.068	0.021
U	0.084	0.066	0.036

^aObtained by spark source spectrometry except for Th and U which were obtained by neutron activation analysis.

Table 2.1-3. Trace-element levels (ppm) in archive
Peach Bottom sleeve graphite obtained
by neutron activation analysis

Element	QA number		
	1010	1716	3290
Al	0.96 ± 0.49	1.72 ± 0.26	3.57 ± 0.62
Ag	<0.023	<0.016	<0.030
As	$<2.0E-3$	$<1.8E-3$	$<4.5E-3$
Au	$<1.5E-4$	$<1.2E-4$	$<3.0E-4$
Ba	<0.26	<0.50	<0.78
Br	0.010 ± 0.001	$8.3E-3 \pm 0.0008$	0.020 ± 0.001
Ce	0.73 ± 0.01	0.44 ± 0.01	1.06 ± 0.02
Cl	6.90 ± 0.36	8.28 ± 0.28	16.5 ± 1.6
Co	0.03 ± 0.07	0.012 ± 0.01	0.070 ± 0.010
Cr	2.12 ± 0.15	1.87 ± 0.07	5.13 ± 0.18
Cs	$<5.0E-3$	$<3.0E-3$	$<6.0E-3$
Cu	<0.11	<0.060	<0.41
Dy	0.043 ± 0.001	0.015 ± 0.001	0.062 ± 0.003
Eu	$<3.0E-4$	$<2.2E-4$	$6.2E-4 \pm 0.00005$
Fe	5.59 ± 1.15	3.18 ± 0.77	15.6 ± 3.3
Hf	0.017 ± 0.001	$4.4E-3 \pm 0.0011$	0.051 ± 0.01
Hg	0.092 ± 0.002	0.038 ± 0.007	0.13 ± 0.01
I	<0.046	<0.014	<0.18
K	0.72 ± 0.21	0.49 ± 0.10	3.39 ± 0.25
La	0.38 ± 0.01	0.27 ± 0.01	0.52 ± 0.01
Lu	0.010 ± 0.001	$1.7E-3 \pm 0.0001$	0.015 ± 0.001
Mn	0.016 ± 0.003	0.20 ± 0.03	0.051 ± 0.024
Na	0.85 ± 0.03	0.80 ± 0.06	8.74 ± 0.38
Nd	0.28 ± 0.07	0.15 ± 0.05	0.40 ± 0.10
Rb	<0.10	<0.10	<0.18
Sb	$<5.0E-3$	$<4.5E-3$	$<7.0E-3$
Sc	0.058 ± 0.001	0.010 ± 0.001	0.11 ± 0.01
Se	<0.025	<0.020	<0.030
Sr	<0.52	<1.0	<0.70
Ta	$<4.0E-3$	$<3.0E-3$	$6.0E-3 \pm 0.0005$
Tb	0.014 ± 0.002	$3.7E-3 \pm 0.0007$	0.019 ± 0.004
Ti	16.5 ± 0.4	2.60 ± 0.17	30.4 ± 0.21
Th	0.065 ± 0.001	0.017 ± 0.0020	0.19 ± 0.01
U	0.063 ± 0.002	<0.017	0.079 ± 0.011
V	3.79 ± 0.09	0.98 ± 0.01	11.6 ± 1.6
W	0.055 ± 0.002	0.026 ± 0.001	0.078 ± 0.004
Zn	<0.30	<0.16	<0.40

Table 2.1-4. Radial distribution of impurities in archive sleeve (QA No. 1716) by neutron activation analysis

Radial width (mm)	2.54	2.54	2.54	2.54	2.54	2.54	2.54	2.54	2.54	2.54	2.54	2.54
Distance from OD to midsample (mm)	1.27	3.81	6.35	8.89	31.75	62.23	81.28	83.82	86.36	88.90	91.44	93.98
Impurity concentration (µg/g)												
Ag	0.063	<0.029	<0.028	<0.022	<0.031	a	<0.028	<0.027	0.058	<0.030	0.14	0.12
Al	11.0	1.25	3.13	2.61	1.75	7.85	7.43	8.32	15.8		320.0	162.0
Au	1.6 E-3	<2.0 E-4	3.4 E-3	4.9 E-3	<1.4 E-3	3.2 E-4	4.5 E-4	4.0 E-4	1.2 E-3	5.2 E-4	4.3 E-3	1.3 E-3
Br	0.089	0.017	0.015	0.023	7.5 E-3	0.013	0.042	0.021	0.044	0.033	0.039	0.11
Ce	0.43	0.75	0.39	0.57	0.71	0.55	0.40	0.34	0.30	0.32	0.28	0.32
Cl	249.0	126.0	125.0	744.0	131.0	106.0	605.0	299.0	165.0	346.0	559.0	523.0
Co	0.45	0.031	0.20	0.18	0.21	0.28	0.26	0.36	0.25	0.31	0.45	0.58
Cr	3.92	4.21	3.09	2.81	2.85	4.32	3.25	3.51	2.62	3.14	5.52	8.36
Cs	<7.0 E-3	<7.2 E-3	<6.4 E-3	<5.8 E-3	<6.7 E-3	<6.8 E-3	<7.0 E-3	<5.8 E-3	<6.0 E-3	<6.5 E-3	<7.1 E-3	<9.6 E-3
Cu	1.06	<0.44	<0.38	<0.18	b	a	1.30	<0.27	c	<0.25	d	<3.1
Fe	9.17	7.29	8.92	5.93	<4.5	11.6	9.70	7.99	13.1	<4.80	37.1	23.4
Hg	0.39	0.038	0.025	0.026	0.017	0.018	0.036	0.27	0.31	0.37	0.55	1.55
K	10.3	<0.64	1.06	0.93	0.99	0.93	1.46	1.07	3.39	0.98	4.14	3.31
La	0.22	0.43	0.24	0.36	0.47	0.36	0.24	0.20	0.17	0.19	0.15	0.17
Mn	0.10	0.037	0.13	0.074	0.016	0.088	0.10	0.063	0.21	<0.03	0.43	0.25
Na	23.2	0.85	1.03	1.66	0.57	1.08	1.74	1.28	2.71	1.29	3.15	5.50
Sc	9.3 E-3	0.010	0.010	0.010	0.017	0.012	0.010	0.010	0.012	0.011	0.011	0.010
Ta	<6.4 E-3	<6.7 E-3	<3.8 E-3	<2.8 E-3	<4.4 E-3	0.012	<3.8 E-3	<5.5 E-3	4.6 E-3	<6.0 E-3	0.015	0.025
Th	0.027	0.019	0.023	0.020	0.022	0.013	0.015	0.015	0.022	0.014	0.017	0.014
Ti	<1.6	1.11	1.27	<2.7	2.45	4.70	<2.7	<2.3	3.3	<4.1	<7.0	<7.0
U	<	<	<	<	<	<	<	<	<	<	<	<
V	0.29	0.34	0.39	0.48	0.96	1.34	1.17	1.28	1.34	1.22	1.51	1.24
W	0.21	0.065	0.013	0.012	0.039	0.12	0.088	0.11	0.24	0.12	0.53	0.27
Zn	1.68	<0.3	<0.3	1.25	b	a	<0.3	0.86	1.24	<0.3	3.44	2.92

^aVariable sample. Concentration ranges: Ag, <0.02 - 0.11; Cu, <1.1 - 760; Zn, <0.02 - 0.11.

^bConcentration ranges: Cu, <0.44 - 2.39; Zn, <0.27 - 2.38.

^cConcentration range: <0.45 - 3.75.

^dConcentration range: <2.1 - 7.45.

activation analysis), which in most cases agreed to within $\pm 25\%$. In some instances, higher variability was observed, as noted in the table; particularly, copper and zinc levels were frequently widely divergent within individual samples.

Referring to Fig. 2.1 of ref. 6, one may note the locations of element E11-07 in the eleventh row of sector E of the Peach Bottom core, one full element removed from control rod element E12-09.

2.2 Operating History -- Power, Temperature, and Fluence

The power operating history of element E11-07, begun at the start-up of core 2 on July 15, 1970, to withdrawal on September 14, 1973, is given in Table 2.2-1. Employing the full operating power level of 115.5 MW(t) indicates that this fuel element experienced 701 EFPDs of operation. Note that during the 162-day shutdown period, fuel element E06-01, reported in ref. 1, was removed from the reactor after 384 EFPDs.

Table 2.2-1. Operating history of element E11-07

Core power [MW(t)]	Time at power (days)	EFPDs	Cumulated EFPDs	Radial power factor	Channel flow (kg/hr)
(Initial condition)			0	1.076	248.0
108	29	27	27	1.069	248.0
107	39	36	63	1.077	248.0
92	7	6	69	1.072	248.0
109	141	133	202	1.078	248.0
96	60	50	252	1.100	248.0
Shutdown	(80)				
94	56	46	298	1.109	248.0
93	56	45	343	1.1096	248.0
Shutdown	(22)				
103	47	42	385	1.105	248.0
Shutdown	(162)				
91	146	115	500	1.097	248.0
Shutdown	(46)				
92	81	64	564	1.089	248.0
103	52	46	610	1.106	248.0
Shutdown	(20)				
106	99	91	710	1.267	248.0

Present best estimates of the axial distribution of the maximum fuel temperature are shown in Fig. 2.2-1 for BOL, midlife (MOL), and EOL conditions. These temperatures are still considered as preliminary, even though element dimensional change data and newer values of the power profiles and coolant flow have been taken into account. Fuel temperatures shown here represent a volume-average value, including fuel particles plus matrix graphite. Some uncertainty exists for an appropriate average value for the thermal conductivity of compact material, especially as a function of degree of irradiation. Future fuel-element temperature estimates may entail incorporation of improved values for the effective fuel compact thermal conductivity. Estimated EOL axial temperature distributions in the spine and sleeve, along with the EOL thermal and fast fluence, are shown in Fig. 2.2-2. End-of-life temperatures at the inside diameter of the compact ranged from about 610°C at the bottom of the core, up to a maximum of about 1330°C near the center of the core, and then back down to about 1090°C at the top of the core. Figure 2.2-1 shows that the fuel temperatures were substantially less than those given above during most of the irradiation time. End-of-life temperatures inside the sleeve ranged from about 470°C at the bottom of the core to about 850°C at the top. The maximum thermal and fast fluence of about 4.1×10^{21} and 3.2×10^{21} neutrons/cm², respectively, occurred at compact 15.

3. PHOTOGRAPHIC DESCRIPTION AND METROLOGY

3.1 Visual and Photographic Examination

Element Ell-07 was visually examined before and during dissection, and representative normal features and appearances were noted and photographed. In addition, all nonnormal features, such as compact cracks and graphite spallation, were photographically recorded.

Figures 3.1-1 to 3.1-4 show external views of the element prior to dissection. The first of these is a view of the top of the element showing the so-called "loading marks" used to fix the 0° orientation of the element. The remaining three external views, Figs. 3.1-2 to 3.1-4 show the element top reflector, bottom spacer ring, and purge-flow exit regions respectively; the element identification number is shown in Fig. 3.1-4.

Fig. 2.2-1. Estimated peak fuel temperatures in Ell-07.

Fig. 2.2-2. Computed axial temperature and fluence profiles for E11-07 at end of life.

Fig. 3.1-1. External view of top of element Ell-07, showing "loading marks" which designate 0° orientation.

Fig. 3.1-2. External view of element in the top reflector region, showing the spacer ring above the fuel zone.

Fig. 3.1-3. External view at bottom spacer ring.

Fig. 3.1-4. External view of bottom of Ell-07, showing element identification number.

Figures 3.1-1 to 3.1-4 illustrate that no significant abnormalities of the elements' externals were observed. Some "soot" deposits were seen,* most notably downstream from the standoff ring at the 4-ft 3-in. location shown in Fig. 3.1-2. In addition, a longitudinal wear mark of unknown origin was observed along most of the cylindrical portion of the element. This is most clearly seen in Figs. 3.1-3 and 3.1-4.

Following completion of the external examination, the fuel element was disassembled using the remote-operated hacksaw shown in Fig. 3.1-5. The saw was designed for operation by hot-cell manipulators to cut partially through the sleeve graphite without damaging the fuel compacts. Following an initial partial cut, the element was rotated by the knob shown in the top middle portion of the photograph to accomplish a complete peripheral cut of the sleeve. Four such cuts through the sleeve were made at compact locations 1, 11, 21, and 30, allowing complete fuel-element disassembly.

The compacts were canned in groups of ten, placed in a nitrogen atmosphere, and transferred to the High-Radiation-Level Examination Laboratory (HRLLEL) for gamma scanning, metrology, and photography. After the compacts were removed from the sleeve, 1/2-in.-long sleeve specimens were cut at compact locations 1, 11, 21, and 30, to be used for wall thickness measurements.

Figures 3.1-6 through 3.1-76 document the appearance of fuel compacts and spine upon disassembly. The first set, Figs. 3.1-6 through 3.1-9, illustrate the bottom of the fuel element from compact locations 1 through 10 in groups of three to four compacts and associated spine. Figures 3.1-10 through 3.1-13 show four close-up views of compact 1. The large crack seen in each of the four figures was not observed on initial disassembly. Hence, it probably was a hairline failure which separated as a result of subsequent handling. A common feature shown by compact 1 is the presence of exposed fuel particles which are clearly visible along the edges.

*

An examination of soot deposits taken from two driver elements is reported in Sect. 5 of ref. 7.

Fig. 3.1-5. Hacksaw used for fuel-element dissection.

Fig. 3.1-6. Compacts 1, 2, and 3 plus spine.

Fig. 3.1-7. Compacts 3, 4, and 5 plus spine.

Fig. 3.1-8. Compacts 5, 6, 7, and 8 plus spine.

Fig. 3.1-9. Compacts 8, 9, and 10 plus spine.

Fig. 3.1-10. Top view of compact 1.

Fig. 3.1-11. Bottom view of compact 1.

Fig. 3.1-12. Side view of compact 1 from 0° direction.

Fig. 3.1-13. Side view of compact 1 from 180° direction.

Figures 3.1-14 through 3.1-17 show four close-up views of compact 2. Figure 3.1-16 shows a longitudinal crack extending the length of the compact. As for compact 1, this crack was not seen directly on disassembly. Therefore, it is thought to have initially been a hairline crack which subsequently opened further. Only compacts 7, 11, and 21 were observed to have cracks immediately following element disassembly. All other subsequently recorded cracks were probably too minute to show up on initial examination.

Figures 3.1-18 through 3.1-21 show four close-up views of compact 5. Figure 3.1-22 shows a lateral view of compact 7, where one of the three compact cracks observed during initial element disassembly is shown. Four views of compact 9 are shown in Figs. 3.1-23 to 3.1-26.

A similar sequence of photographs shown in Figs. 3.1-27 through 3.1-48 record the appearance of the central set of ten fuel compacts and associated spine. Figures 3.1-27 to 3.1-30 show the appearance directly upon disassembly, whereas the remaining photographs in this sequence are close-up views obtained in the HRLEL at a somewhat later date. As noted above, compact 11 was one of three compacts in which cracks were noted at the time of disassembly. Figures 3.1-31 and 3.1-32 show close-up views of the failed region of compact 11. One may also note hairline cracks in compacts 12 and 15.

Figures 3.1-49 through 3.1-76 show various views of compacts 21 through 30 and associated spine. The first four figures in this group illustrate these compacts in groups of three directly on removal from the sleeve. A discoloration of the spine was noted in the vicinity of compacts 21 and 22 and spallation of the spine in regions adjacent to compacts 23 to 25. A close-up of this spalled zone on the spine is shown in Fig. 3.1-53.

The crack in compact 21, clearly visible in Fig. 3.1-49, is one of the three noted during disassembly. A close-up view is shown in Fig. 3.1-54; Figs. 3.1-55 and 3.1-56 are photographs of the same crack edge taken after the compact was deliberately broken. Figures 3.1-57 through 3.1-76 show photographs of compacts 22, 24, 27, 28, and 30. Minor cracking in compacts 22, 24, 27, and 30 may be noted; however, the surface finish appears to be excellent.

Fig. 3.1-14. Top view of compact 2.

Fig. 3.1-15. Bottom view of compact 2.

Fig. 3.1-16. Side view of compact 2 from 0° direction.

Fig. 3.1-17. Side view of compact 2 from 180° direction.

Fig. 3.1-18. Top view of compact 5.

Fig. 3.1-19. Bottom view of compact 5.

Fig. 3.1-20. Side view of compact 5 from 0° direction.

Fig. 3.1-21. Side view of compact 5 from 180° direction.

Fig. 3.1-22. Lateral view of compact 7, showing 1 of the 3 initially seen cracks.

Fig. 3.1-23. Top view of compact 9.

Fig. 3.1-24. Bottom view of compact 9.

Fig. 3.1-25. Side view of compact 9 from 0° direction.

Fig. 3.1-26. Side view of compact 9 from 180° direction.

Fig. 3.1-27. Compacts 11, 12, 13, and associated spine.

Fig. 3.1-28. Compacts 14, 15, 16, and associated spine.

Fig. 3.1-29. Compacts 16, 17, 18, and associated spine.

Fig. 3.1-30. Compacts 18, 19, 20, and associated spine.

Fig. 3.1-31. Close-up view of crack in compact 11.

Fig. 3.1-32. Close-up view of crack in compact 11.

Fig. 3.1-33. Top view of compact 12.

Fig. 3.1-34. Bottom view of compact 12.

Fig. 3.1-35. Side view of compact 12 from 0° direction.

Fig. 3.1-36. Side view of compact 12 from 180° direction.

Fig. 3.1-37. Top view of compact 15.

Fig. 3.1-38. Bottom view of compact 15.

Fig. 3.1-39. Side view of compact 15 from 0° direction.

Fig. 3.1-40. Side view of compact 15 from 180° direction.

Fig. 3.1-41. Top view of compact 16.

Fig. 3.1-42. Bottom view of compact 16.

Fig. 3.1-43. Side view of compact 16 from 0° direction.

Fig. 3.1-44. Side view of compact 16 from 180° direction.

Fig. 3.1-45. Top view of compact 18.

Fig. 3.1-46. Bottom view of compact 18.

Fig. 3.1-47. Side view of compact 18 from 0° direction.

Fig. 3.1-48. Side view of compact 18 from 180° direction.

Fig. 3.1-49. Compacts 21, 22, 23, and associated spine.

Fig. 3.1-50. Compacts 23, 24, 25, 26, and associated spine.

Fig. 3.1-51. Compacts 26, 27, 28, and associated spine.

Fig. 3.1-52. Compacts 28, 29, 30, and associated spine.

Fig. 3.1-53. View of top spine piece, showing discoloration in compact regions 21 and 22, and spallation in compact regions 23 to 25.

Fig. 3.1-54. Close-up view of crack in compact 21.

Fig. 3.1-55. View of cracked surface of compact 21.

Fig. 3.1-56. Additional view of cracked surface of compact 21 near inside diameter.

Fig. 3.1-57. Top view of compact 22.

Fig. 3.1-58. Bottom view of compact 22.

Fig. 3.1-59. Side view of compact 22 from 0° direction.

Fig. 3.1-60. Side view of compact 22 from 180° direction.

Fig. 3.1-61. Top view of compact 24.

Fig. 3.1-62. Bottom view of compact 24.

Fig. 3.1-63. Side view of compact 24 from 0° direction.

Fig. 3.1-64. Side view of compact 24 from 180° direction.

Fig. 3.1-65. Top view of compact 27.

Fig. 3.1-66. Bottom view of compact 27.

Fig. 3.1-67. Side view of compact 27 from 0° direction.

Fig. 3.1-68. Side view of compact 27 from 180° direction.

Fig. 3.1-69. Top view of compact 28.

Fig. 3.1-70. Bottom view of compact 28.

Fig. 3.1-71. Side view of compact 28 from 0° direction.

Fig. 3.1-72. Side view of compact 28 from 180° direction.

Fig. 3.1-73. Top view of compact 30.

Fig. 3.1-74. Bottom view of compact 30.

Fig. 3.1-75. Side view of compact 30 from 0° direction.

Fig. 3.1-76. Side view of compact 30 from 180° direction.

3.2 Metrology of Sleeve, Spine, and Fuel Compacts

Preirradiation dimensional data were obtained by GAC during the assembly of element Ell-07. These data, obtained from ref. 8, are summarized in Tables 3.2-1 through 3.2-7 together with the postirradiation dimensions. These include (1) the overall element length; (2) the distance between reference marks (termed fiducial marks) on two lands on the sleeve exterior; (3) the so-called "compact expansion gap," which is the free distance between the upper surface of compact 30 and the lower surface of the upper reflector assembly; (4) the outer sleeve diameter at ten locations, each in two planes at 0 and 90° orientations; (5) the wall thickness of the sleeve at four locations; (6) the diameter of the three spine pieces, each at three locations; (7) the length of each spine piece; (8) the outer and inner diameters of ten fuel compacts at 0 and 90° orientation; and (9) the length of ten fuel compacts.

Postirradiation dimensional measurements were determined by employing the hot-cell equipment shown in Figs. 3.2-1 to 3.2-4. Figure 3.2-1 shows the device used to determine the inner and outer diameters of the fuel compacts at various desired angular orientations. Fuel compact lengths were determined using the simple device shown in Fig. 3.2-2. Sleeve diameters and wall thicknesses were determined using the device shown in Fig. 3.2-3, and the spine lengths and distance between fiducial marks were measured using the device shown in Fig. 3.2-4. Spine diameters were determined by a manipulator-operated micrometer caliper.

The overall element length and distance between fiducial marks are given in Table 3.2-1 for pre- and postirradiation conditions. As noted, these dimensions decreased as a result of irradiation service. The axial expansion gap diminished 6.9%, primarily because of the axial expansion experienced by the fuel compacts.

Table 3.2-2 lists the pre- and postirradiation outside sleeve diameters at the indicated compact locations in each of two designated planes. As noted, the sleeve outer diameters decreased on irradiation service from 0.14% to 0.44%, with the largest reduction occurring at compact location 18.

Fig. 3.2-1. Dial gage comparator for determining fuel compact inner and outer diameters.

Fig. 3.2-2. Device for measuring compact length.

Fig. 3.2-3. Device for measurement of sleeve wall thickness and outer diameter.

Fig. 3.2-4. Device for measuring spine length and distance between fiducial marks on sleeve.

Table 3.2-1. Fuel-element dimensional changes due to irradiation

	Preirradiation (in.)	Postirradiation (in.)	Percent change ^a
Overall element length	143.9375	143.375	-0.39
Axial distance between fiducial marks	38.00	37.7282	-0.715
Axial compact expansion gap	4.2825	3.9844	-6.96

^a Negative value denotes a decrease in dimension.

Table 3.2-2. Outside diameter of Ell-07 sleeve

Compact	Distance from bottom of element (in.)	Azimuth ^a					
		0°			90°		
		Pre- irradiation (in.)	Post- irradiation (in.)	Percent change ^b	Pre- irradiation (in.)	Post- irradiation (in.)	Percent change ^b
1	27.5	3.4867	3.4822	-0.15	3.4869	3.4820	-0.14
5	39.5	3.4861	3.4765	-0.27	3.4863	3.4775	-0.25
9	51.5	3.4840	3.4755	-0.24	3.4840	3.4750	-0.26
12	60.5	3.4863	3.4775	-0.25	3.4866	3.4765	-0.29
15	69.5	3.4862	3.4750	-0.320	3.4861	3.4750	-0.315
18	78.5	3.4862	3.4710	-0.436	3.4867	3.4725	-0.407
21	87.5	3.4863	3.4728	-0.387	3.4863	3.4702	-0.462
24	96.5	3.4856	3.4745	-0.318	3.4858	3.4720	-0.395
27	105.5	3.4863	3.4750	-0.324	3.4862	3.4750	-0.320
30	114.5	3.4869	3.4821	-0.14	3.4872	3.4813	-0.17

^aZero degree denotes plane of serial number on sleeve.

^bNegative value denotes decrease in dimension.

Table 3.2-3 records postirradiation inner diameter and wall thickness of the sleeve at four compact locations, three of which correspond to locations where preirradiation data exist. The postirradiation measurements included the outer diameter at two orientations and the wall thickness at four, from which the inner diameters were computed. Conversely, prior to irradiation the inner and outer diameters were directly measured and the wall thicknesses calculated therefrom. We note that irradiation service caused a reduction in the inner diameter of from 0.56% to 0.17%.

Table 3.2-4 records the change in spine diameter and length as a result of irradiation. The table shows that spine diameters and lengths diminished, with the central portions showing the greatest change.

Table 3.2-5 records the postirradiation inner and outer diameters of ten fuel compacts at a top, middle, and bottom axial location, and in two planes. Comparison with initial dimensions, given for eight of these compacts, shows that both the compact inner and outer diameters experienced irradiation-induced shrinkage from 0.2% to 3.9%.

Measured compact wall thicknesses at four angular locations following irradiation service are recorded in Table 3.2-6. The change in compact length as a result of irradiation is listed in Table 3.2-7; in general, compact lengths were found to increase up to approximately 1%.

One of the critical dimensions in the Peach Bottom fuel element is the gap between outer compact diameter and inner sleeve. This dimension sensitively affects the volume rate of purge flow in this area and, consequently, axial fission product transport as well. In addition, since the nuclear heat is transferred by conduction across this helium gap, it also sensitively affects the temperature level of the fuel compacts. Figures 2.1-3 and 2.1-4 indicate that a nominal gap width prior to irradiation of 0.009 in. was intended. This checked well with the data near compact 30 given in Tables 3.2-3 and 3.2-5, which is the only compact location where direct determination of the gap width can be made. Using average values for the data listed for compact 30, a initial gap width of 0.00503 in. is calculated. Following irradiation service, again using

Table 3.2-3. Wall thickness and inside diameter of Ell-07 sleeve

Compact	Azimuth ^a							
	0°			90°			180°	270°
	Pre- irradiation (in.)	Post- irradiation (in.)	Percent change ^b	Pre- irradiation (in.)	Post- irradiation (in.)	Percent change ^b	Post- irradiation (in.)	
Wall thickness								
1	0.3661	0.37445	2.3	0.3672	0.37655	2.5	0.36045	0.35765
11		0.37215			0.37465		0.36015	0.35535
21	0.36665	0.36445	-0.60	0.3659	0.36125	-1.3	0.36615	0.36975
30	0.36595	0.36375	-0.60	0.3671	0.36435	-0.75	0.36885	0.36855
Inside diameter								
1	2.7545	2.7473	-0.26	2.7525	2.7478	-0.17		
11		2.7431			2.7448			
21	2.7530	2.7422	-0.39	2.7545	2.7392	-0.56		
30	2.7550	2.7495	-0.20	2.7530	2.7484	-0.17		

^aZero degree denotes plane of serial number.

^bNegative number denotes decrease in dimension.

Table 3.2-4. Diameter and length of Ell-07 spines

		Diameter		Percent change ^b
	Position ^a	Pre- irradiation (in.)	Post- irradiation (in.)	
Bottom spine	T	1.6625	1.648	-0.87
	M	1.6645	1.653	-0.69
	B	1.6630	1.6605	-0.15
Middle spine	T	1.6640	1.646	-1.08
	M	1.6620	1.6445	-1.05
	B	1.6640	1.6470	-1.02
Top spine	T	1.6640	1.6535	-0.63
	M	1.6635	1.6495	-0.84
	B	1.6640	1.648	-0.96

		Length		Percent change ^b
		Pre- irradiation (in.)	Post- irradiation (in.)	
Top spine		30.015	29.5789	-1.45
Middle spine		30.016	29.2978	-2.39
Bottom spine		30.016	29.7395	-0.92

^aT, M, and B denote top, middle, and bottom of spine.

^bNegative value denotes a decrease in dimension.

Table 3.2-5. Outside and inside diameters of Ell-07 fuel compacts

Compact	Position ^b	Azimuth ^a											
		0°						90°					
		Outside diameter			Inside diameter			Outside diameter			Inside diameter		
		Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^c	Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^c	Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^c	Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^c
2	T		2.7336			1.7222			2.7386			1.7436	
	M		2.7348			1.7419			2.7374			1.7442	
	B		2.7359			1.7425			2.7386			1.7453	
5	T	2.7390	2.7140	-0.913	1.7500	1.7305	-1.114	2.7410	2.7110	-1.094	1.7510	1.7283	-1.296
	M	2.7400	2.7157	-0.887	1.7490	1.7099	-2.236	2.7420	2.7112	-1.123	1.7500	1.7288	-1.211
	B	2.7400	2.7171	-0.836	1.7490	1.7319	-0.978	2.7410	2.7127	-1.032	1.7490	1.7293	-1.126
9	T	2.7410	2.7334	-0.277	1.7520	1.7467	-0.30	2.7450	2.7346	-0.379	1.7520	1.7474	-0.263
	M	2.7410	2.7350	-0.219	1.7515	1.7465	-0.29	2.7440	2.7375	-0.237	1.7510	1.7483	-0.154
	B	2.7390	2.7343	-0.172	1.7530	1.7470	-0.34	2.7440	2.7381	-0.215	1.7525	1.7496	-0.165
12	T	2.7410	2.6565	-3.083	1.7472	1.6892	-3.320	2.7400	2.6916	-1.766	1.7476	1.7207	-1.539
	M	2.7400	2.6426	-3.555	1.7470	1.6793	-3.875	2.7390	2.6829	-2.048	1.7470	1.7197	-1.563
	B	2.7400	2.6542	-3.131	1.7474	1.6868	-3.468	2.7400	2.6981	-1.529	1.7465	1.7258	-1.185
15	T	2.7410	2.6637	-2.820	1.7480	1.6975	-2.889	2.7410	2.6442	-3.532	1.7485	1.6791	-3.969
	M	2.7430	2.6565	-3.153	1.9490	1.6938	-3.153	2.7420	2.6352	-3.895	1.7490	1.6704	-4.494
	B	2.7440	2.6710	-2.660	1.7495	1.7039	-2.606	2.7435	2.6368	-3.889	1.7510	1.6738	-4.409
18	T	2.7430	2.6566	-3.150	1.7500	1.6935	-3.23	2.7455	2.6653	-2.921	1.7510	1.7053	-2.610
	M	2.7400	2.6451	-3.464	1.7500	1.6901	-3.42	2.7435	2.6505	-3.390	1.7485	1.6970	-2.945
	B	2.7410	2.6502	-3.313	1.7500	1.6894	-3.46	2.7435	2.6591	-3.076	1.7500	1.6988	-2.928
22	T		2.6571			1.6886			2.6587			1.6921	
	M		2.6479			1.6831			2.6519			1.6664	
	B		2.6603			1.6912			2.6722			1.7014	
24	T	2.7405	2.6610	-2.901	1.7500	1.6897	-3.446	2.7440	2.6840	-2.187	1.7500	1.7049	-2.577
	M	2.7400	2.6537	-3.150	1.7500	1.6831	-3.823	2.7430	2.6714	-2.610	1.7500	1.6980	-2.971
	B	2.7410	2.6432	-3.568	1.7490	1.6870	-3.545	2.7450	2.6824	-2.281	1.7500	1.7048	-2.583
27	T	2.7425	2.6981	-1.619	1.7520	1.7201	-1.821	2.7455	2.6963	-1.792	1.7520	1.7100	-2.397
	M	2.7400	2.6892	-1.854	1.7500	1.7157	-1.960	2.7430	2.6866	-2.056	1.7495	1.7061	-2.481
	B	2.7438	2.6908	-1.932	1.7480	1.7119	-2.065	2.7400	2.6917	-1.763	1.7480	1.7107	-2.134
30	T	2.7410	2.7220	-0.693	1.7490	1.7277	-1.218	2.7410	2.7266	-0.525	1.7485	1.7359	-0.721
	M	2.7450	2.7172	-1.013	1.7490	1.7277	-1.218	2.7425	2.7226	-0.726	1.7480	1.7327	-0.875
	B	2.7445	2.7175	-0.984	1.7510	1.7271	-1.365	2.7465	2.7254	-0.768	1.7480	1.7343	-0.784

^aZero degree denotes plane of + mark on end of predimensioned compacts.^bT, M, and B denote top, middle, and bottom of compact.^cNegative value denotes decrease.

Table 3.2-6. Wall thickness of Ell-07 fuel compacts

Compact	Location ^b	Wall thickness (in.) at azimuths ^a of:			
		0°	90°	180°	270°
2	T	0.4950	0.4974	0.5164	0.4842
	M	0.4946	0.4989	0.4983	0.4943
	B	0.4947	0.5003	0.4987	0.4930
5	T	0.4948	0.4910	0.4887	0.4917
	M	0.4954	0.4924	0.5104	0.4900
	B	0.4953	0.4929	0.4899	0.4905
9	T	0.4938	0.4964	0.4929	0.4908
	M	0.4927	0.4979	0.4958	0.4913
	B	0.4917	0.4990	0.4956	0.4895
12	T	0.4842	0.4795	0.4831	0.4914
	M	0.4821	0.4774	0.4812	0.4858
	B	0.4869	0.4818	0.4805	0.4905
15	T	0.4838	0.4797	0.4824	0.4854
	M	0.4815	0.4784	0.4812	0.4864
	B	0.4848	0.4792	0.4823	0.4838
18	T	0.4775	0.4787	0.4856	0.4813
	M	0.4731	0.4756	0.4819	0.4779
	B	0.4763	0.4771	0.4845	0.4832
22	T	0.4866	0.4861	0.4819	0.4805
	M	0.4840	0.4850	0.4808	0.5005
	B	0.4868	0.4896	0.4823	0.4785
24	T	0.4859	0.4906	0.4854	0.4885
	M	0.4856	0.4848	0.4850	0.4886
	B	0.4676	0.4881	0.4886	0.4895
27	T	0.4896	0.4945	0.4884	0.4918
	M	0.4884	0.4923	0.4881	0.4882
	B	0.4901	0.4924	0.4888	0.4886
30	T	0.4955	0.4967	0.4988	0.4940
	M	0.4960	0.4970	0.4935	0.4929
	B	0.4957	0.4982	0.4947	0.4929

^aZero degree denotes plane of + mark on end of predimensioned compacts.

^bT, M, and B denote top, middle, and bottom of compact.

Table 3.2-7. Lengths of Ell-07 fuel compacts

Compact	Azimuth ^a					
	0°			90°		
	Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^b	Pre-irradiation (in.)	Post-irradiation (in.)	Percent change ^b
2		2.9745			2.9697	
5	2.9900	2.9894	-0.02	2.9910	2.9917	0.02
9	3.0260	3.0427	0.55	3.0270	3.0430	0.53
12	3.0250	3.0310	0.19	3.0220	3.0350	0.43
15	3.013	3.0415	0.95	3.014	3.0422	0.93
18	3.0450	3.0633	0.60	3.0520	3.0641	0.40
22		3.1062			3.1068	
24	3.0220	3.0270	0.16	3.0180	3.0292	0.37
27	2.9590	2.9708	0.40	2.9590	2.9670	0.27
30	3.0060	3.0011	-0.16	3.0070	3.0032	-0.13

^aZero degree denotes first land clockwise of + position on compact.

^bNegative value denotes decrease in length.

average values, the gap width expanded to 0.0135 in. in the vicinity of compact 30. Since radial shrinkage was greatest for the central compacts, gap increases were undoubtedly higher in the central core region than those observed for this one case.

Table 3.2-8 records the observed curvature of the top spine piece. The other two spine pieces exhibited no measurable curvature.

Table 3.2-8. Fuel-element dimensional changes as a result of irradiation

	Pre- irradiation (in.)	Post- irradiation (in.)	Percent change ^a
Overall element length	143.9375	143.375	-0.39
Axial distance between fiducial marks	38.00	37.7282	-0.715
Axial compact expansion gap	4.2825	3.9844	-6.96

^aNegative value denotes a decrease in dimension.

4. DISTRIBUTION OF GAMMA-EMITTING NUCLIDES

4.1 Equipment and Methods

4.1.1 Measurement of gamma emitters in fuel compacts

Axial distributions and inventories of gamma-emitting radionuclides in the fuel compacts of E11-07 were determined by gamma spectrometry methods similar to those used for fuel element E06-01 as described in ref. 6. The fuel compacts, hermetically sealed under a nitrogen atmosphere in groups of ten in aluminum containers, were measured by gamma spectroscopy in a hot cell of the HREEL. The aluminum containers had a bail at the top to permit them to be hung on a scanning mechanism, centered over and moved vertically past a collimator slit. Because the activity levels in the E11-07 compacts were higher than in E06-01, a 4.4-cm-thick aluminum gamma

attenuator was interposed in the gamma-ray beam between the collimator and the detector to reduce the total count rates to manageable levels. In other respects the collimator system was the same as used previously.

Prior to the acquisition of gamma spectra of individual compacts, continuous scans of ^{95}Zr and ^{137}Cs levels were obtained. This was accomplished by routing the signal from the gamma spectroscopy amplifier to two single-channel analyzers (SCAs). The energy window of one SCA was set over both the ^{95}Zr photopeaks (724 and 756 keV), and the window of the other SCA was set over the 662-keV photopeak of ^{137}Cs . The signals from the SCAs were routed through count-rate meters to a dual-channel strip-chart recorder. Strip-chart recordings were made while each of the groups of compacts was moved continuously past the collimator slit at a rate of about 0.09 mm/sec. It should be noted that when SCAs are used in this manner to measure photopeak count rates, Compton-scattered radiation from gamma rays of higher energies underlies the peak of interest and is also measured. The strip-chart recordings for ^{95}Zr and ^{137}Cs are thus semiquantitative representations of the axial distributions of these radionuclides.

Following the continuous scans, one gamma spectrum of each compact was acquired with the compacts centered over the collimator slit. These spectra, recorded on magnetic tape, were processed later to obtain activities for each compact after counting efficiencies were determined.

Counting efficiencies were computed from photopeak count rates of spectra acquired with the collimator. Specific activities ($\mu\text{Ci/g}$) were measured on specimens core-drilled from three compacts. Five core-drilled specimens from compacts 9, 16, and 28 were used for this calibration. The procedure was previously described.⁶

4.1.2 Measurement of axial distributions of gamma emitters in the sleeve, spine, and bottom reflector

The axial distribution of gamma emitters in the sleeve, spine, and bottom reflector was measured with an automatic gamma scanning system. The sleeve was cut into three sections to correspond in length to the

three spine pieces, each associated with a group of ten compacts. Each sleeve and spine piece and the bottom reflector were individually placed in aluminum containers and gamma scanned with the system shown in Fig. 4.1-1.

The gamma scanning system consisted of a shield (central part of figure) with a collimator slit, a Ge(Li) detector (lower left), a motor-driven drum (on wall at top right of shield), and electrical controls for the motor (on wall at right of shield). The shield, made from lead-filled steel shells, consisted of a bottom section, 86 cm high and 50 cm in diameter, and a top section, 142 cm high and 41 cm in diameter. The collimator slit was 0.62 cm high, 2.54 cm wide, and 17.8 cm long. Each shield section had a mating central annulus, 15 cm in diameter, into which a canned graphite component could be placed for gamma scanning. A flexible steel cable wound onto the motor-driven drum was lowered through the shield into a hot cell below. The shield was then attached to the bail of a container, and the graphite component was brought up into the top section of the shield. The component was located in the shield such that the bottom of the can was just visible through the collimator slit. The electrical controls, consisting of a motor speed and direction controller, a counting location timer, and a total experiment timer, were then set for the experiment.

Gamma scanning consisted in acquisition of a gamma spectrum while the graphite fuel component remained stationary. Upon completion of the gamma spectrum, the data were recorded on magnetic tape, and the graphite specimen was moved 0.622 cm ($\sim 1/4$ in.). The cycle was then repeated. A 300-sec gamma spectrum was thereby acquired at each location along the specimen length. The movement of the component by an exact distance was provided by a cam mounted on the shaft of the motor that turned the drum. Two cam lobes affixed to the cam 180° apart actuated a microswitch mounted under the cam. Each time the cam turned one-half revolution, the component was lowered 0.622 cm before the microswitch was contacted, causing the counting timer to be reset. Movement to a new counting location was adjusted by the motor speed controller to about the same time, ~ 1 sec, required to record the gamma spectrum. Automatic scanning was initiated by an

Fig. 4.1-1. Axial gamma scanning system for sleeve and spine.

operator located at the scanner and another operator located at a computer-based pulse-height analyzer. Simultaneous starting of the scanner and the analyzer was begun by the operators in contact by telephone. Gamma spectra were acquired in real time for 300 sec; therefore, the acquisition of spectra remained synchronized with the scanner. The total scan durations were set to stop the scanner at the end of an automatic run. At the beginning of a scanning run the detector was placed at one of four locations at 5, 66, 170, and 353 cm from the scanning shield. These locations provided for measurement of graphite components whose activity levels ranged from a few millicuries to a few curies per 3-in. section of component. Although the activity levels of the middle sleeve and spine sections were the highest of all components examined, they were the least variable. It was thus possible to leave the detector at location 4 during scanning of the entire section. The top and bottom sleeve and spine sections were highly variable in activity, and it was necessary to interrupt the scanning occasionally and move the detector to a different location.

The calibration of the counting efficiencies was carried out with radioactive mock-ups of a sleeve and spine section. A section of blotter paper was uniformly coated with paint in which ^{110m}Ag has been homogeneously dispersed. Sections of the paper were cut out, and cylinders were formed that would fit snugly into a nonradioactive archive length of sleeve and over an archive spine piece. The lengths of the paper cylinders for both the sleeve and spine mock-ups were each 29.2 cm. The specific surface activities of the paper cylinders were determined by measuring the total amounts of ^{110m}Ag on each and dividing these values by the corresponding surface areas. The paper cylinders were then combined with the archive graphite pieces, and a gamma spectrum of each was acquired with the mock-ups in the scanning shield and with the detector placed at counting location 1. Each spectrum was acquired for 8000 sec, and the mock-up was moved 2.54 cm every 800 sec to yield gamma spectra averaged at ten locations over each mock-up. Counting efficiencies for each major gamma ray of ^{110m}Ag were computed by dividing photopeak count rates by the corresponding photon emission rate per square centimeter.

Because the specific surface activity of the mock-ups was low compared with that of Ell-07 graphite, it was not possible to acquire gamma spectra of the mock-ups with the detector at counting locations 2, 3, and 4. The counting efficiencies for counting location 2 were derived by selecting locations on the Ell-07 sleeve and spine pieces that could be counted at both detector locations 1 and 2. The specific surface activities of the radionuclides measured with the detector at location 1 were then used to derive counting efficiencies for detector location 2. This procedure was then repeated for the other detector locations.

Activity levels in Ell-07 graphite components were derived according to the principles outlined in Appendix A of ref. 7. Activities were computed on the basis of a 3-in. section of sleeve and spine to correspond to the length of a fuel compact. Activities contained per 3-in. length of graphite were computed by the relation

$$A \text{ (Ci/g in.)} = \frac{NC \cdot e^{\lambda t}}{W \cdot T_L \cdot EF \cdot BR (1 - A_K R_T) 3.7 \text{ E10}} ,$$

where

NC = net counts in an appropriate photopeak,

λ = decay constant of the radionuclide, sec^{-1} ,

t = decay time, sec,

W = reciprocal of the surface area of 3 in. of component, cm^{-2} ,

T_L = counting time, sec,

A_K = amplifier resolving time, 7 sec ,

R_T = count rate of the entire gamma spectrum, counts/sec,

3.7 E10 = conversion factor, disintegrations/ μCi .

Gamma spectra were acquired and processed with a Tennecomp TP-5000 computer-based pulse-height analyzer shown in Fig. 4.1-2. The system consisted of a Digital Equipment Corporation PDP-11/05 computer with a 28 K, 16-bit memory; a Teletype; a high-speed paper-tape punch and reader; an industry-standard reel magnetic-tape deck made by Cipher Data Products; a Tennecomp "DataPour" cartridge magnetic-tape deck; and an ORTEC model 800 analog-to-digital converter.

Fig. 4.1-2. Computer-based pulse-height analyzer used to acquire and process gamma spectra of Ell-07 components.

4.2 Axial Distribution of Gamma Emitters

4.2.1 Fuel compacts

The measured inventories in curies as of September 16, 1973, of 11 gamma-emitting nuclides in each of the 30 fuel compacts are listed in Table 4.2-1. Each spectrum was taken through a 0.125-in.-diam collimator such that the view from the detector was a 0.5-in.-diam circle through the center of each compact. Total inventories are listed in the last row of the table.

Figure 4.2-1 illustrates the axial variation of ^{233}Pa and ^{95}Zr inventories in the fuel compacts. As anticipated, concentrations of both these nuclides generally follow the expected axial dependency of neutron flux, since neither protactinium nor zirconium is expected to migrate under these irradiation conditions. The lack of mobility of ^{95}Zr is corroborated by comparing it with the ^{144}Ce profile, as is done in Fig. 4.2-2. Neither is ^{144}Ce expected to be mobile in the fuel element under these conditions, and the level value of the $^{144}\text{Ce}/^{95}\text{Zr}$ ratio with axial location is consistent with the view that neither isotope migrated substantially during irradiation service. The indicated variation of $^{141}\text{Ce}/^{95}\text{Zr}$ in Fig. 4.2-2 is shown also to be level on the average, with a greater local variation caused by poorer counting statistics for this shorter-half-life isotope of cerium. A similar comparison of ^{106}Ru to ^{95}Zr levels illustrates that ^{106}Ru also did not significantly migrate from Ell-O7 fuel compacts during its stay in the core.

Hence, any of the isotopes, ^{95}Zr , ^{144}Ce , ^{106}Ru , or ^{233}Pa , could be used as a "mobility monitor;" that is, the value of the inventory ratio of a particular nuclide to any of the cited nuclides would be indicative of its mobility. For example, Fig. 4.2-3 illustrates the variation of ^{137}Cs and ^{134}Cs inventories relative to ^{95}Zr in each compact. If cesium did not migrate in the fuel element, one would expect the $^{137}\text{Cs}/^{95}\text{Zr}$ ratio to be level like the $^{144}\text{Ce}/^{95}\text{Zr}$ ratio, and the value of $^{134}\text{Cs}/^{95}\text{Zr}$ to have roughly a cosine dependency, paralleling the thermal flux profile. However, Fig. 4.2-3 indicates that both ^{134}Cs and ^{137}Cs migrated significantly in the fuel element during its irradiation in the reactor. Evidently the

Table 4.2-1. Fission product inventories in element E11-07 fuel compacts
(curies/compact on September 16, 1973)

Compact												
No.	ZR95	%ER ^a	RU103	%ER	RU106	%ER	AG110M	%ER	SB125	%ER	CS134	%ER
1	1.32E+02	0.2	5.65E+01	1.6	6.41E+00	2.0	1.80E-02	* ^b	3.00E-01	14.0	4.58E+00	0.3
2	1.46E+02	0.3	6.72E+01	2.0	6.96E+00	3.0	6.69E-02	16.0	3.00E-01	20.0	7.71E+00	0.3
3	1.57E+02	0.3	6.34E+01	1.6	7.74E+00	3.0	3.63E-02	36.0	3.00E-01	18.0	9.24E+00	0.4
4	1.99E+02	0.3	8.60E+01	1.4	9.69E+00	3.6	5.40E-02	*	5.70E-01	15.0	1.43E+01	0.2
5	2.03E+02	0.3	9.67E+01	1.4	1.04E+01	2.7	3.90E-02	*	5.60E-01	15.0	1.70E+01	0.2
6	2.44E+02	0.2	1.14E+02	1.3	1.26E+01	2.5	3.90E-02	*	7.70E-01	18.0	2.38E+01	0.2
7	2.15E+02	0.3	1.00E+02	1.4	1.12E+01	2.8	3.90E-02	*	4.00E-01	18.0	2.53E+01	0.2
8	2.38E+02	0.1	1.12E+02	1.6	1.24E+01	1.6	3.90E-02	*	6.10E-01	9.4	2.32E+01	0.2
9	2.54E+02	0.1	1.04E+02	0.9	1.30E+01	2.2	3.90E-02	*	7.70E-01	11.0	3.08E+01	0.1
10	2.75E+02	0.1	1.14E+02	0.4	1.41E+01	1.0	3.90E-02	*	8.00E-01	3.8	2.15E+01	0.1
11	2.64E+02	0.2	1.17E+02	1.1	1.32E+01	3.0	3.90E-02	*	7.90E-01	11.0	9.32E+00	0.3
12	2.69E+02	0.1	1.22E+02	0.8	1.34E+01	2.1	3.90E-02	*	8.30E-01	7.8	2.21E+01	0.2
13	2.66E+02	0.2	1.19E+02	1.2	1.33E+01	2.3	3.90E-02	*	8.70E-01	11.0	1.92E+01	0.2
14	2.56E+02	0.2	1.05E+02	1.2	1.38E+01	2.2	3.90E-02	*	7.10E-01	15.0	1.46E+01	0.4
15	2.82E+02	0.1	1.20E+02	0.3	1.37E+01	0.9	3.90E-02	*	8.70E-01	3.2	8.72E+00	0.2
16	2.82E+02	0.2	1.26E+02	1.1	1.44E+01	2.1	3.90E-02	*	8.90E-01	8.8	8.95E+00	0.4
17	2.82E+02	0.1	1.24E+02	0.7	1.40E+01	1.8	3.90E-02	*	8.40E-01	6.7	7.17E+00	0.3
18	2.93E+02	0.1	1.38E+02	0.7	1.47E+01	1.8	3.90E-02	*	9.10E-01	6.4	8.77E+00	0.4
19	2.80E+02	0.1	1.33E+02	0.7	1.44E+01	1.5	3.90E-02	*	9.00E-01	7.0	8.35E+00	0.4
20	3.07E+02	0.1	1.35E+02	0.3	1.45E+01	1.1	3.90E-02	*	9.20E-01	3.3	8.95E+00	0.2
21	2.70E+02	0.2	1.24E+02	1.1	1.32E+01	2.2	3.90E-02	*	7.20E-01	11.0	7.10E+00	0.7
22	2.99E+02	0.1	1.39E+02	0.3	1.49E+01	1.1	3.90E-02	*	9.20E-01	3.1	6.28E+00	0.2
23	2.54E+02	0.1	1.21E+02	0.3	1.24E+01	0.7	3.90E-02	*	6.80E-01	3.9	6.65E+00	0.2
24	2.55E+02	0.1	1.20E+02	0.7	1.25E+01	2.8	3.90E-02	*	7.30E-01	7.4	1.05E+01	0.3
25	2.16E+02	0.1	1.02E+02	0.4	1.03E+01	0.9	3.90E-02	*	5.90E-01	5.0	1.07E+01	0.1
26	2.02E+02	0.4	9.53E+01	2.0	9.25E+00	4.1	3.90E-02	*	5.00E-01	5.0	1.08E+01	0.5
27	2.05E+02	0.2	9.84E+01	1.0	9.39E+00	2.2	3.90E-02	*	4.40E-01	14.0	9.79E+00	0.3
28	1.82E+02	0.1	8.88E+01	0.4	8.45E+00	1.2	3.90E-02	*	4.60E-01	5.1	7.65E+00	0.2
29	1.76E+02	0.1	8.48E+01	0.4	7.96E+00	0.9	3.90E-02	*	4.00E-01	5.8	6.21E+00	0.2
30	1.52E+02	0.1	7.36E+01	0.4	6.94E+00	1.0	3.90E-02	*	3.90E-01	5.9	4.66E+00	0.2
SUM	7.07E+03		3.20E+03		3.50E+02				1.97E+01		3.74E+02	

^aThe 1-sigma counting statistics error.

^bThe * signifies upper limit value.

Table 4.2-1 (continued)

Compact No.	CS137	%ER	CE141	%ER	CE144	%ER	EU154	%ER	PA233	%ER	CS137/CS134
1	8.01E+00	0.2	9.76E+01	4.2	7.58E+01	1.0	1.00E-01	5.9	6.66E+02	1.0	1.75
2	1.00E+01	0.3	1.00E+02	6.6	8.70E+01	1.5	1.50E-01	7.4	8.75E+02	1.5	1.30
3	1.03E+01	0.3	1.14E+02	6.0	9.21E+01	1.5	1.90E-01	6.0	9.96E+02	1.4	1.12
4	1.41E+01	0.3	1.24E+02	5.5	1.22E+02	1.4	2.70E-01	5.1	1.16E+03	1.2	0.98
5	1.50E+01	0.3	1.39E+02	5.8	1.23E+02	1.4	3.00E-01	4.8	1.25E+03	1.2	0.88
6	1.90E+01	0.2	1.67E+02	5.3	1.44E+02	1.2	3.70E-01	4.5	1.58E+03	1.1	0.80
7	1.88E+01	0.2	1.70E+02	5.7	1.22E+02	1.4	3.70E-01	4.6	1.43E+03	1.2	0.74
8	1.77E+01	0.1	1.75E+02	3.2	1.42E+02	0.8	4.30E-01	2.6	1.68E+03	0.7	0.76
9	2.19E+01	0.1	1.88E+02	3.3	1.46E+02	0.8	4.30E-01	2.8	1.71E+03	0.6	0.71
10	1.64E+01	0.1	2.05E+02	1.4	1.62E+02	0.3	4.00E-01	1.4	1.91E+03	0.3	0.76
11	9.00E+00	0.3	1.94E+02	4.0	1.50E+02	1.2	1.60E-01	7.8	1.91E+03	0.8	0.97
12	1.62E+01	0.1	2.10E+02	2.9	1.57E+02	0.7	4.10E-01	2.6	1.99E+03	0.6	0.73
13	1.44E+01	0.2	2.02E+02	4.4	1.54E+02	1.2	3.40E-01	4.7	2.01E+03	0.9	0.75
14	1.14E+01	0.3	2.00E+02	4.2	1.49E+02	1.2	3.80E-01	4.1	1.86E+03	0.9	0.78
15	8.41E+00	0.1	2.33E+02	1.2	1.61E+02	0.3	1.50E-01	2.7	1.95E+03	0.2	0.96
16	8.61E+00	0.3	2.34E+02	3.9	1.65E+02	1.1	1.70E-01	8.2	2.10E+03	0.8	0.96
17	7.54E+00	0.2	2.32E+02	2.3	1.62E+02	0.7	2.00E-02	8.0	2.10E+03	0.5	1.05
18	8.77E+00	0.2	2.34E+02	2.4	1.71E+02	0.6	1.30E-01	2.1	2.20E+03	0.5	1.00
19	8.38E+00	0.2	2.37E+02	2.3	1.61E+02	0.7	1.40E-01	6.3	2.04E+03	0.5	1.00
20	9.08E+00	0.1	2.43E+02	1.0	1.66E+02	0.3	2.70E-01	1.8	2.40E+03	0.2	1.01
21	7.61E+00	0.3	1.85E+02	4.0	1.54E+02	1.1	2.00E-01	6.0	1.85E+03	0.9	1.07
22	7.51E+00	0.1	2.12E+02	1.2	1.69E+02	0.3	2.00E-01	10.7	2.02E+03	0.2	1.20
23	7.27E+00	0.1	2.09E+02	1.2	1.44E+02	0.3	1.20E-01	3.4	1.87E+03	0.2	1.09
24	1.01E+01	0.2	2.06E+02	2.5	1.41E+02	0.7	1.50E-01	4.0	1.76E+03	0.5	0.96
25	1.01E+01	0.1	1.87E+02	1.7	1.21E+02	0.4	3.00E-01	1.6	1.53E+03	0.3	0.94
26	1.05E+01	0.4	1.42E+02	8.4	1.13E+02	2.0	2.40E-01	6.0	1.36E+03	1.6	0.98
27	1.07E+01	0.2	1.44E+02	4.5	1.15E+02	1.1	2.30E-01	4.7	1.32E+03	0.9	1.10
28	9.45E+00	0.1	1.41E+02	1.6	1.00E+02	0.4	1.90E-01	2.1	1.22E+03	0.3	1.24
29	9.02E+00	0.1	1.31E+02	1.7	9.80E+01	0.4	1.60E-01	2.3	1.02E+03	0.4	1.45
30	8.05E+00	0.1	1.14E+02	1.8	8.62E+01	0.5	1.20E-01	2.7	8.24E+02	0.4	1.73
SUM	3.43E+02		5.38E+03		4.06E+03		7.09E+00		4.87E+04		

OR 447-81

Fig. 4.2-1. Element Ell-07 inventories of ^{95}Zr and ^{233}Pa per compact as of September 16, 1973.

ORI L DWG 7c-83

Fig. 4.2-2. Inventory ratios of $^{141}\text{Ce}/^{95}\text{Zr}$ and $^{144}\text{Ce}/^{95}\text{Zr}$ in Ell-07 fuel compacts.

Fig. 4.2-3. Inventory ratios of $^{134}\text{Cs}/^{95}\text{Zr}$ and $^{137}\text{Cs}/^{95}\text{Zr}$ in Ell-07 fuel compacts.

means for transport was provided by the internal purge flow, which proceeds from the top of the element at compact 30 down toward compact 1. The level value of the $^{137}\text{Cs}/^{95}\text{Zr}$ ratio at 0.05 for the lowest five compacts, which experienced temperatures in the range 1100 to 1230°C at end of life, indicates no movement of ^{137}Cs in these compacts. Above compact 25, fuel temperatures gradually increase to a maximum range from 1230 to 1330°C at compact 16 (see Fig. 2.2-2). The dip in the $^{137}\text{Cs}/^{95}\text{Zr}$ ratio shown in Fig. 4.2-3 evidently results from a transport of cesium vaporized by the high temperatures which existed in the central portion of the fuel element, caused by the purge flow to cooler regions approximately below compact location 14. If the value 0.05 observed for the $^{137}\text{Cs}/^{95}\text{Zr}$ ratio in compacts 26 to 30 is taken to be typical for cases where cesium has not migrated, then graphical integration of the profile shown in Fig. 4.2-3 indicates that about 54 Ci of ^{137}Cs migrated, amounting to about 16% of the fuel-element inventory.

A behavior similar to that of ^{137}Cs , though somewhat more erratic, is demonstrated by ^{154}Eu , as seen in Fig. 4.2-4. The greater irregularity is due to the poorer counting statistics exhibited in this case, where the count rates are much lower. A further complication is that as an activation product of a fission product (like ^{134}Cs), ^{154}Eu levels depend on the square of the flux, and hence the $^{154}\text{Eu}/^{95}\text{Zr}$ ratio would tend to show a cosine dependency with axial location. Nevertheless, ^{154}Eu shows a marked drop in inventory in compact regions between 24 and 15, as does ^{137}Cs . However, contrary to the case concerning cesium, no deposition zone is clearly apparent.

The continuous scans of the photopeak count rates of ^{137}Cs and ^{95}Zr in the fuel compacts are presented in Fig. 4.2-5. These scans were obtained using the method described in Sect. 4.1. Three principal features of the distribution of these fission products are easily recognizable. Most interesting is the drop of ^{137}Cs relative to the ^{95}Zr in the region of compacts 11 through 20, where the fuel temperature is highest. Also of interest are some localized and pronounced variations within single compacts, for example, compacts 6, 7, 8, 18, and 19. Generally the localized variations of ^{95}Zr are larger than those for ^{137}Cs . A possible reason for

ORNL DW-76-65

Fig. 4.2-4. Inventory ratios of $^{154}\text{Eu}/^{95}\text{Zr}$ and $^{125}\text{Sb}/^{95}\text{Zr}$ in Ell-07 fuel compacts.

ORNL DWG 74-6370R1

Fig. 4.2-5. Continuous scans of photopeak count rates of ^{95}Zr and ^{137}Cs in E11-07 fuel compacts.

this behavior is that ^{95}Zr variations represent loading nonuniformities which would be less evident in the cesium distribution because of its tendency to more readily migrate. Last, the interfaces between compacts, where compact end leveling results in lower amounts of fuel per unit volume, are clearly indicated by dips in the count rate. This feature was used to locate the center of compacts for the acquisition of gamma spectra and compact inventory.

4.2.2 Axial distributions of gamma-emitting nuclides in sleeve and spine

The sleeve, spine, and lower reflector regions were axially scanned for gamma-emitting radionuclides, employing the apparatus described in Sect. 4.1. Two planes, 90° apart, were scanned with spectra acquired at 0.25-in. intervals in each plane. As usual practice in these examinations, plane 1 is defined by the serial number stamped on the exterior of the sleeve, and plane 2 is obtained by 90° counterclockwise rotation of the upright element.

Some raw computer plots are illustrated in Figs. 4.2-6 through 4.2-9. Planes 1 and 2 are designated, respectively, by squares and circles; where significant, vertical error bands computed from the photon counting statistics are plotted. The ordinate is given in terms of curies for each 3 in. of graphite, which is the length of a fuel compact. In viewing these axial distributions, one should recall that the bottom of compact 1, which is the reference for the abscissa scale, is located at the entry of the primary coolant; however, the direction of the fuel-element purge flow is opposite to that of the primary coolant, that is, from right to left on the abscissa.

Figures 4.2-6 and 4.2-7 illustrate, respectively, the ^{60}Co and ^{137}Cs axial profiles in the sleeve. The data for planes 1 and 2 seem closely coincident for ^{60}Co , whereas the ^{137}Cs distribution displays a significant peripheral dependency, especially between axial locations from 40 to 60 in. from the bottom. The $^{110\text{m}}\text{Ag}$ sleeve profile in Fig. 4.2-8 illustrates the the relatively poorer counting statistics for this nuclide. Here the small, triangular points signify upper-limit values at locations where $^{110\text{m}}\text{Ag}$ was

Fig. 4.2-6. Axial distribution of ^{60}Co in Ell-07 sleeve as of September 14, 1973.

Fig. 4.2-7. Axial distribution of ^{137}Cs in Ell-07 sleeve as of September 14, 1973.

ORNL DWG 76-685

Fig. 4.2-8. Axial distribution of ^{110m}Ag in Ell-07 sleeve as of September 14, 1973.

Fig. 4.2-9. Axial distribution of ^{154}Eu in Ell-07 sleeve as of September 14, 1973.

not detected. Figure 4.2-9 illustrates the axial distribution of activation product ^{154}Eu in the sleeve.

In addition to ^{60}Co , ^{137}Cs , $^{110\text{m}}\text{Ag}$, and ^{154}Eu , ^{134}Cs was normally observed at all locations in the sleeve and spine, and ^{54}Mn , ^{106}Ru , and ^{144}Ce were occasionally observed. A summary of the principal gamma-emitting nuclides found in the sleeve and spine is given in Tables 4.2-2 and 4.2-3. The nuclide inventories are given in terms of curies per unit of compact length for each of the two planes. Therefore, each value in the tables is an average of 12 determinations. Those values indicated by an asterisk are an average of 12 upper-limit determinations in that 3-in. span; that is, no positive detection was made of the nuclide in 12 spectra acquired in the 3-in. length of graphite. In cases where upper-limit estimates and real data both exist in a sequence of 12 determinations, the upper-limit values are simply ignored, and the value listed is the average of the real data.

A pictorial summary of the ^{137}Cs axial distribution is illustrated in Fig. 4.2-10. Smoothed profiles in the sleeve and spine are drawn from the detailed computer plots such as shown in Fig. 4.2-7. The calculated (but still preliminary) inside sleeve and spine temperature profiles are also shown. The horizontal dashed lines display the inventory ratios, $^{137}\text{Cs}/^{95}\text{Zr}$, observed for each fuel compact at the indicated axial location. As discussed in Sect. 4.2-1, ^{95}Zr is used here as a mobility monitor; that is, it is known to remain immobile in the fuel under these irradiation conditions; therefore any departure of the inventory ratio from normal signifies movement of ^{137}Cs . The normal value of 0.052 is given at the purge flow inlet at the right in the figure. It is apparent that ^{137}Cs deposition on the sleeve begins (viewing from right to left in the direction of the purge) where ^{137}Cs loss from the fuel first becomes apparent--at approximately the 75-in. location. Here the volume average compact temperature at the inner compact diameter is estimated to be 1250°C, and the inner sleeve temperature, where cesium is depositing, is at an estimated 920°C. Further in the direction of the purge flow, ^{137}Cs deposition on the sleeve increases throughout the region where its removal from the fuel is observed (by lower than normal $^{137}\text{Cs}/^{95}\text{Zr}$ ratios), and diminishes beyond the region of maximum removal from the fuel.

Table 4.2.2. Axial distribution of radionuclides in Ell-07 sleeve
Curies/compact length September 14, 1973

COMPACT NO.	CS-137		CS-134		AG-110		CO-60		EU-154	
	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2
1	1.02E -3	3.86E -4	1.23E -3	3.91E -4	1.04E -4	1.26E -4	2.86E -4	2.62E -4	1.25E -4	1.18E -4
2	6.81E -3	6.66E -4	9.94E -3	7.59E -4	4.61E -4	4.78E -4	3.16E -4	3.14E -4	1.41E -4	1.47E -4
3	1.69E -2	2.99E -3	2.65E -2	4.92E -3	1.34E -3	1.01E -3	3.34E -4	3.53E -4	1.44E -4	1.72E -4
4	1.90E -2	6.00E -3	3.25E -2	9.90E -3	3.23E -3	1.99E -3	3.42E -4	3.86E -4	1.39E -4	1.57E -4
5	5.11E -2	1.36E -2	8.92E -2	2.34E -2	7.13E -3	3.77E -3	4.37E -4	4.12E -4	1.94E -4	1.75E -4
6	9.34E -2	8.02E -2	1.66E -1	1.40E -1	5.07E -3	6.61E -3	4.55E -4	4.75E -4	2.44E -4	3.15E -4
7	4.06E -1	3.09E -1	7.21E -1	5.40E -1	1.42E -2	2.05E -2	5.84E -4	7.14E -4	5.53E -4	8.86E -4
8	9.00E -1	6.67E -1	1.61E 0	1.17E 0	2.79E -2	2.86E -2	8.77E -4	1.08E -3	2.31E -3	1.64E -3
9	9.82E -1	7.67E -1	1.74E 0	1.33E 0	3.56E -2	4.01E -2	1.06E -3	1.09E -3	2.55E -3	1.67E -3
10	8.65E -1	9.34E -1	1.55E 0	1.64E 0	5.60E -2	6.08E -2	1.35E -3	1.55E -3	2.37E -3	3.16E -3
11	1.21E 0	1.16E 0	2.16E 0	2.09E 0	4.07E -2	5.42E -2	1.48E -3	1.59E -3	4.36E -3	6.32E -3
12	1.20E 0	1.11E 0	2.11E 0	1.97E 0	4.43E -2	6.37E -2	1.86E -3	1.78E -3	5.10E -3	6.43E -3
13	1.51E 0	1.38E 0	2.64E 0	2.44E 0	3.79E -2	3.76E -2	2.58E -3	1.65E -3	1.32E -2	8.59E -3
14	1.70E 0	1.44E 0	2.94E 0	2.54E 0	3.67E -2	2.53E -2	2.45E -3	2.41E -3	3.69E -2	6.38E -2
15	1.82E 0	1.32E 0	3.19E 0	2.37E 0	3.82E -2	2.68E -2	3.56E -3	4.70E -3	4.99E -2	8.57E -2
16	1.96E 0	1.38E 0	3.45E 0	2.49E 0	3.68E -2	3.34E -2	2.88E -3	4.41E -3	6.87E -2	9.91E -2
17	1.59E 0	1.04E 0	2.74E 0	1.87E 0	3.59E -2	1.89E -2	6.14E -3	6.69E -3	1.91E -1	2.41E -1
18	1.54E 0	9.92E -1	2.66E 0	1.77E 0	3.02E -2	3.29E -2	6.22E -3	7.01E -3	1.52E -1	1.56E -1
19	1.14E 0	7.61E -1	1.93E 0	1.34E 0	3.45E -2	2.18E -2	8.02E -3	1.18E -2	1.71E -1	1.87E -1
20	1.13E 0	5.42E -1	1.90E 0	9.40E -1	1.63E -2	1.82E -2	1.53E -2	1.82E -2	1.96E -1	2.01E -1
21	5.81E -1	3.65E -1	9.28E -1	6.04E -1	1.57E -2	9.96E -3	1.63E -2	1.75E -2	2.21E -1	1.77E -1
22	3.91E -1	2.05E -1	6.17E -1	3.47E -1	1.26E -2*	1.25E -2	1.43E -2	1.54E -2	2.13E -1	2.29E -1
23	2.42E -1	1.55E -1	3.84E -1	2.44E -1	8.96E -3	7.48E -3	1.92E -2	1.86E -2	1.34E -1	1.86E -1
24	1.42E -1	1.43E -1	2.08E -1	2.09E -1	5.94E -3	7.50E -3	6.55E -3	7.04E -3	6.36E -2	9.15E -2
25	4.95E -2	3.06E -2	6.45E -2	4.36E -2	3.70E -3	2.98E -3	2.76E -3	3.18E -3	1.35E -2	2.65E -2
26	6.97E -3	1.34E -2	8.83E -3	1.91E -2	4.08E -3	4.29E -3	1.35E -3	1.91E -3	4.54E -3	6.68E -3
27	2.41E -3	4.88E -3	3.01E -3	5.93E -3	3.94E -3	3.65E -3	1.45E -3	1.28E -3	9.18E -4	1.36E -3
28	6.51E -4	1.02E -3	8.82E -4	1.18E -3	3.01E -3	3.08E -3	6.48E -4	6.36E -4	4.11E -4	3.56E -4
29	5.17E -4	8.76E -4	5.88E -4	8.81E -4	2.18E -3	2.34E -3	4.13E -4	3.69E -4	1.67E -4	1.57E -4
30	1.68E -4	3.32E -4	1.74E -4	3.35E -4	1.36E -3	2.07E -3	3.66E -4	3.23E -4	1.01E -4	1.51E -4
SUM	1.95E 1	1.48E 1	3.39E 1	2.61E 1	5.64E -1	5.53E -1	1.20E -1	1.33E -1	1.55E 0	1.78E 0

*Indicates upper limit value.

Table 4.2.3. Axial distribution of radionuclides in Ell-07 spine
Curies/compact length September 14, 1973

COMPACT NO.	CS-137		CS-134		AG-110		CO-60		EU-154	
	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2	PLANE 1	PLANE 2
1	9.61E -3	2.06E -2	1.46E -2	3.09E -2	2.41E -3	4.27E -3	1.67E -4	1.89E -4	1.93E -4	2.10E -4
2	1.85E -2	4.65E -2	3.14E -2	7.43E -2	8.24E -3	1.11E -2	4.81E -4	4.38E -4	2.12E -4	2.32E -4
3	6.33E -2	9.33E -2	1.16E -1	1.51E -1	1.21E -2	8.21E -3	5.76E -4	5.10E -4	2.75E -4	3.19E -4
4	9.58E -2	1.51E -1	1.63E -1	2.52E -1	8.25E -3	4.89E -3	7.64E -4	7.07E -4	3.18E -4	3.72E -4
5	2.08E -1	2.33E -1	3.63E -1	3.99E -1	5.60E -3	5.94E -3	9.93E -4	9.25E -4	4.51E -4	5.08E -4
6	3.16E -1	3.08E -1	5.67E -1	5.33E -1	4.93E -3	7.75E -3	1.23E -3	1.04E -3	7.56E -4	7.24E -4
7	6.47E -1	5.52E -1	1.20E 0	9.68E -1	1.09E -2*	8.17E -3	1.48E -3	1.16E -3	1.40E -3	8.26E -4
8	7.56E -1	6.60E -1	1.35E 0	1.17E 0	1.18E -2*	1.22E -2*	1.28E -3	1.51E -3	1.72E -3	2.39E -3
9	7.39E -1	7.61E -1	1.30E 0	1.34E 0	1.13E -2*	1.46E -2	1.46E -3	1.86E -3	1.52E -3	3.24E -3
10	6.97E -1	6.60E -1	1.23E 0	1.17E 0	1.10E -2*	1.05E -2	2.18E -3	2.50E -3	4.47E -3	3.25E -3
11	7.39E -1	6.19E -1	1.34E 0	1.09E 0	1.32E -2	7.79E -3	9.18E -4	6.03E -4	2.14E -3	2.29E -3
12	7.23E -1	6.15E -1	1.30E 0	1.08E 0	1.34E -2	1.25E -2*	8.30E -4	6.72E -4	5.10E -3	3.50E -3
13	6.98E -1	6.50E -1	1.24E 0	1.13E 0	1.40E -2	1.31E -2*	1.19E -3	9.30E -4	5.48E -3	4.89E -3
14	5.89E -1	5.86E -1	1.05E 0	1.01E 0	1.06E -2	1.06E -2	1.05E -3	1.01E -3	9.46E -3	9.28E -3
15	4.07E -1	4.02E -1	7.31E -1	6.94E -1	8.99E -3	7.76E -3*	6.06E -4	4.34E -4	1.37E -2	1.50E -2
16	3.89E -1	3.86E -1	6.98E -1	6.64E -1	7.70E -3	6.58E -3	5.72E -4	4.80E -4	1.11E -2	1.35E -2
17	3.25E -1	3.21E -1	5.80E -1	5.46E -1	6.11E -3*	6.00E -3*	3.45E -4	3.73E -4	1.49E -2	1.46E -2
18	2.54E -1	2.56E -1	4.59E -1	4.32E -1	3.99E -3	5.13E -3	2.49E -4	1.84E -4	1.45E -2	1.36E -2
19	2.20E -1	2.26E -1	3.97E -1	3.77E -1	3.10E -3	3.63E -3	3.34E -4	2.75E -4	1.40E -2	1.37E -2
20	1.99E -1	1.94E -1	3.57E -1	3.25E -1	4.62E -3	4.06E -3*	6.47E -4	4.48E -4	1.43E -2	1.35E -2
21	2.40E -1	2.05E -1	3.95E -1	3.31E -1	4.33E -3*	4.03E -3*	2.47E -4	2.46E -4	1.36E -2	1.44E -2
22	9.03E -2	7.99E -2	1.47E -1	1.31E -1	1.90E -3*	2.92E -3	7.74E -5	6.10E -5	1.06E -2	1.16E -2
23	2.41E -2	1.81E -2	3.79E -2	2.87E -2	3.69E -5*	3.29E -5	5.88E -6	5.96E -6	5.90E -4	6.36E -4
24	2.45E -2	1.55E -2	3.71E -2	2.32E -2	2.61E -5	3.00E -5*	2.01E -5	1.83E -5	4.32E -4	4.87E -4
25	1.45E -2	1.03E -2	2.16E -2	1.43E -2	1.07E -4*	2.20E -4*	3.77E -4	6.38E -4	1.44E -3	3.13E -3
26	4.74E -3	4.42E -3	7.20E -3	5.75E -3	2.54E -4*	2.16E -4	1.03E -3	9.55E -4	3.64E -3	2.97E -3
27	2.26E -3	2.54E -3	3.19E -3	3.04E -3	1.89E -4*	1.82E -4	8.50E -4	7.62E -4	2.19E -3	1.71E -3
28	7.49E -4	8.95E -4	1.09E -3	1.10E -3	1.03E -4	1.42E -4	8.34E -4	7.19E -4	1.18E -3	1.06E -3
29	2.17E -4	2.60E -4	4.24E -4	4.06E -4	1.18E -4*	1.17E -4*	4.87E -4	4.38E -4	4.40E -4	3.96E -4
30	1.54E -4	1.73E -4	3.27E -4	3.15E -4	1.06E -4	1.04E -4	2.85E -4	2.62E -4	2.12E -4	2.08E -4
SUM	8.50E 0	8.08E 0	1.51E 1	1.40E 1	1.79E -1	1.73E -1	2.16E -2	2.04E -2	1.50E -1	1.53E -1

*Indicates upper limit value.

Fig. 4.2-10. Axial distribution of ^{137}Cs in element Ell-07 sleeve and spine as of September 14, 1973.

The behavior described above is consistent with the view that cesium is redistributed in fuel element ELL-07 as a result of boil-off from the fuel compact and subsequent convection in the direction of the purge flow. The observed spine profiles are qualitatively in accord with this model. Cesium deposition peaks in the spine are smaller than in the sleeve and also occur farther downward of the purge. Both of these features evidently result from (1) the larger spine-to-compact gap width relative to the sleeve-to-compact gap, thus rendering radial transport to the spine more difficult, and (2) the higher spine temperatures, which reduce the driving force for condensation of cesium vapor.

The smoothed axial distribution of ^{134}Cs , shown in Fig. 4.2-11, appears to be approximately like that shown for ^{137}Cs despite the differing distribution of production rates for these two nuclides. The production rate for fission products like ^{137}Cs varies with the flux, while secondary nuclides like ^{134}Cs vary as the flux squared. Cesium-134 deposition on the sleeve also appears to begin at the axial location where ^{134}Cs loss from the fuel begins, that is, at approximately the 75-in. location on the abscissa.

Smoothed $^{110\text{m}}\text{Ag}$ axial deposition profiles in the sleeve and spine are shown in Fig. 4.2-12. No fuel compact inventory profiles are superimposed in this case because $^{110\text{m}}\text{Ag}$ detection in the fuel was completely masked by its relatively low inventory. In particular, high ^{134}Cs levels in long-decayed fuel would serve to mask the fewer and lower-energy $^{110\text{m}}\text{Ag}$ photons. As for cesium, the $^{110\text{m}}\text{Ag}$ profiles are also strongly suggestive of a boil-off and convection mechanism for determining the axial distribution within the element. However, the very much lower vapor pressure of silver in comparison with cesium suggests that transport occurs via some species other than elemental silver.

Smoothed axial profiles of ^{60}Co and ^{154}Eu in the sleeve and spine are shown in Fig. 4.2-13. The source of ^{60}Co is evidently both activation of various iron isotopes and ^{59}Co present as contaminants in the graphite as well as deposition of corrosion debris carried along with the helium. It is quite interesting to note the significantly higher

Fig. 4.2-11. Axial distribution of ^{134}Cs in element Ell-07 sleeve and spine as of September 14, 1973.

Fig. 4.2-12. Axial distribution of ^{110}mAg in element E11-07 sleeve and spine as of September 14, 1973.

Fig. 4.2-13. Axial distribution of ^{60}Co and ^{154}Eu in element E11-07 sleeve and spine.

levels of ^{60}Co found on the sleeve relative to the spine. From the radial profile data presented in Sect. 4.3, it is clear that the major portions of the ^{60}Co found in both sleeve and spine came from a surface deposit with subsequent diffusion radially inward. The sleeve apparently was exposed to a ^{60}Co source at both the outer diameter by primary coolant and internally via the fuel-element purge, with the latter source predominating. The radial profile data show that ^{60}Co levels due to activation of impurities in the graphite generally equal about $1\text{ }\mu\text{Ci/g}$, compared with levels nearer the exposed surfaces, which range from about 7 to $90\text{ }\mu\text{Ci/g}$.

Europium-154 activity levels similarly result from a dual source--either activation of ^{153}Eu present as a contaminant in graphite, or transport from the fuel of fission product ^{153}Eu and the resulting activation product ^{154}Eu . As for ^{60}Co , radial profile data in Sect. 4.3 clearly show that the latter source is the dominant one; that is, the radial distributions indicate that ^{154}Eu in the graphite resulted from a surface deposit which diffused radially inward.

4.2.3 Axial distributions in the upper and lower reflector regions

Axial distributions of observed gamma-emitting nuclides in the upper element region are given in Table 4.2-4. The top portion of this table records nuclide levels found in graphite samples drawn from the inner surface of the purge-flow inlet tube by drilling the $1/4$ -in.-diam (0.635-cm) tube out to $5/16$ in. (0.794 cm). These samples thus represent deposited activity from the primary coolant, from which the purge flow is drawn directly at this point.

The lower portion of the table indicates the nuclide levels observed in core-drilled samples taken from the porous plug. Evidently, some cesium deposition from the purge-gas flow is affected by the porous plug.

Axial distributions in the region of the bottom reflector and the fission product trap are illustrated in Fig. 4.2-14. These measurements were performed using the axial scanning cask described in Sect. 4.1. The $^{110\text{m}}\text{Ag}$ curve is a dashed line, signifying that at most locations, only upper-limit values of $^{110\text{m}}\text{Ag}$ levels were obtained. The large ^{60}Co peak

Table 4.2-4. Axial distributions in top reflector region
(corrected to September 14, 1973)

Axial location ^a (cm)	Section length (cm)	Radionuclide concentration (μCi/g)								
		⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
Samples from purge tube ^b										
4.0	8.0		0.30			1.21	0.62	0.21	0.089	0.51
11.75	7.5		0.028			0.23	0.11			0.48
19.23	7.45	0.008	0.028		0.028	0.26	0.16			0.62
26.80	7.7		0.024		0.075	0.39	0.20			0.51
34.45	7.6		0.042		0.17	0.45	0.23			0.51
41.33	6.15		0.041		0.23	0.41	0.21			0.51
Samples from porous plug ^c										
45.18	1.55		6.44	0.31	2.74	0.46	0.25	0.2	1.45	0.54
46.75	1.6		12.72	1.17	12.0	0.44	0.23	0.66	1.50	0.52
48.28	1.45		3.75		0.16	0.12	0.028		1.42	0.23
50.1	2.2		6.02			0.15	0.026		1.88	0.17
52.15	1.9	0.25	12.0			0.15	0.027		2.00	0.18
53.88	1.55	0.49	16.8			0.16	0.025		2.19	0.16
55.49	1.68	0.52	16.5			0.15	0.022		2.28	0.15
56.88	1.1	0.88	22.3	0.67		0.49	0.63	5.87	2.57	1.29

^aFrom the top of the element to the center of the section.

^bObtained by drilling purge hole [0.635 cm (1/4 in.)] to 0.794 cm (5/16 in.).

^cObtained by core drilling porous plug.

Fig. 4.2-14. Axial distribution of gamma emitters in bottom reflector assembly of element Ell-07 as of September 14, 1973.

is caused by the stainless steel screen supporting the charcoal trap and serves to determine precisely the axial location of the readings.

One may observe that the charcoal trap serves to decrease the observed cesium levels by about two orders of magnitude in the first 2 in. of the charcoal trap. However, total inventory of cesium in the trap is quite low in comparison with observed deposits on the sleeve and spine--approximately 0.04 Ci of ^{137}Cs in the trap compared with about 17 Ci on the spine. Therefore it appears that the fuel-element graphite itself serves as the major cesium trap within the element.

The fission product trap was sliced into five axial sections, and the charcoal from each section was collected for analysis. As recorded in Table 4.2-5, four of the five sections were approximately 5.1 cm long, and the one farthest downstream was approximately 10.2 cm long. The masses of charcoal and the contained gamma activity collected from each section are also recorded in Table 4.2-5. As noted in the last column of the table, the decrease in the ratio $^{137}\text{Cs}/^{134}\text{Cs}$ with distance down the trap indicates the greater penetrating power of the mass 134 isotope as a result of the relatively longer time it spends as the xenon precursor. A further interesting feature is the degree of activity retention on the nut and the cloth, the two graphite structural pieces at the trap entry. The preponderant portion of the trap activity is contained on these two pieces of trap structure.

4.3 Radial Distributions in the Sleeve and Spine

4.3.1 Facility for remote acquisition of radial graphite samples

Gamma activity in the sleeve and spine graphite of element Ell-07 was sufficiently high so as to preclude radial sectioning in the glove-box facility used for the earlier, 384-day irradiation element.⁶ Therefore a remote-operated lathe, shown in overall view in Fig. 4.3-1, was installed in hot cell 3 of Building 3019 for the acquisition of radial samples of sleeve and spine in subsequent fuel elements.

A conventional 5-ft bed lathe was adapted for remote acquisition of radial graphite samples by the following alterations:

Table 4.2-5. Axial distribution of gamma emitters in the charcoal of the fission product trap

Sample	Distance from top of trap (cm)	Mass of charcoal (g)	Nuclide concentration ($\mu\text{Ci/g}$)								$^{137}\text{Cs}/^{134}\text{Cs}$
			^{54}Mn	^{60}Co	^{106}Ru	^{110m}Ag	^{134}Cs	^{137}Cs	^{144}Ce	^{154}Eu	
1	0-5.1	7.159	<1.1	10.7	<11.1	<2.7	249.0	129.0	35.7	0.9	0.52
2	5.1-10.2	17.319	<0.5	11.6		2.5	78.2	16.1		0.4	0.21
3	10.2-15.2	20.804	1.1	12.6		2.1	44.8	7.8		0.3	0.17
4	15.2-20.3	20.066	0.3	9.2		<0.9	35.0	3.9	<24.0	0.2	0.11
5	20.3-30.5	32.958	0.4	7.4		0.8	24.5	3.7	<17.0	0.2	0.15
<hr/>											
Nuclide inventory (μCi)											
Graphite nut and cloth at trap entry			3.9 E2	1.1 E2	4.8 E3	1.0 E3	1.34 E5	1.1 E5	5.4 E4

Fig. 4.3-1. Manipulator-operated lathe for acquiring radial graphite samples.

1. The motor drive was relocated from beneath to the rear of the main axis, and changed from single- to variable-speed drive. The best cutting speed was found to be about 200 rpm, which is as slow as possible under load without stalling.
2. The carriage movement and cross-feed drive handles (shown center front, Fig. 4.3-1) were adapted for easy manipulator handling.
3. Instead of the normal chuck used for metallic specimens, special collets were selected to hold the sleeve and spine specimens. Remote specimen centering is performed more easily with the collet rather than with the chuck holder.
4. Dial indicators for centering the specimens and for the measurement of radial feed were installed as shown in Fig. 4.3-2.
5. A Lucite sample retainer, shown in place in Fig. 4.3-1 and open in Fig. 4.3-2, was installed to collect the graphite cuttings. The Lucite retainer was equipped with a sliding door arrangement which allowed entry of the cutting tool and the necessary axial motion.
6. Some simple, special tools were fabricated. The sample collectors shown in Fig. 4.3-3 were designed to fit in the Lucite retainer and were used to transfer the sample to preweighed bottles. Other tools, shown hanging at the right in Fig. 4.3-1, were used for tightening the collet and centering the specimen.

The first step in the radial sampling procedure was vacuum cleaning of the sampling area, Lucite retainer, and steel sample holder, using the vacuum and hose attachment shown in Fig. 4.3-1. After installing the sample in the collet, the dial indicator was dropped in place, as shown in Fig. 4.3-2, and the lathe was manually turned using the holes provided in the handwheel. The dial indicator was then removed, and the sample holder and Lucite retainer were installed.

Fig. 4.3-2. Close-up view of radial profile lathe with specimen of sleeve in collet.

PHOTO 0328-76

Fig. 4.3-3. Graphite sample collectors. Sample specimens of sleeve.

The sleeve samples were potted in epoxy, with a handling bar installed as shown in Fig. 4.3-2. The first cut from any specimen fixed in the collet was along the exposed right face for cleanup purposes. Following this facing operation, the cutting tool was located at the specimen surface by means of a feeler gage (not shown), moved axially to the right of the specimen, and then radially inward some specified degree of travel noted on the radial feed dial indicator. An approximately 1/2-in. axial cut was usually taken, terminated automatically by a prepositioned stop. Each subsequent radial cut was taken over progressively smaller axial travels to preclude the possibility of cross-contaminating the samples.

4.3.2 Radial distributions of gamma emitters in the sleeve and spine

Radial distributions, using the previously described apparatus, were obtained at the following axial locations: in the graphite bottom-connector region directly below the charcoal trap; in the sleeve and spine at compact locations 5, 12, 18, and 28; and in the upper reflector region, 16 cm from the top of the element. These data are recorded in Tables 4.3-1 through 4.3-10. As indicated in these tables, radial sections 5 to 10 mils (0.13 to 0.25 mm) wide were obtained near surfaces, and thicker sections of from 30 to 60 mils (0.77 to 1.54 mm) were used for interior graphite regions. The two central spine samples were acquired by drilling, using successively 1/4- and 3/8-in. drills.

Radial distributions for the observed gamma-emitting nuclides are shown graphically in Figs. 4.3-4 through 4.3-7 for the four locations in the fueled region of the element and in Fig. 4.3-8 for the top reflector region. The dashed lines shown in some of the figures signify that only upper-limit values were obtained for that nuclide in the indicated region. The minima evident in all of the sleeve profiles indicate exposure to diffusing nuclides both from the coolant and purge-gas sides of the sleeve.

Figure 4.3-9 illustrates the manner in which the ratio $^{137}\text{Cs}/^{134}\text{Cs}$ varies radially in the sleeve and spine at axial locations 5, 12, 18, and 28. Radial variation of this ratio may be interpreted in terms of either differences in half-life of the two cesium isotopes, 30.1 vs 2.1 years

Table 4.3-1. Radial distribution of radionuclides in top reflector region,
16 cm from the top of the element
(corrected to September 14, 1973)

Radial width of sample		Distance to center of cut from OD (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)				
(mils)	(cm)		^{60}Co	^{110m}Ag	^{134}Cs	^{137}Cs	^{154}Eu
							$^{137}\text{Cs}/^{134}\text{Cs}$
20	0.0508	0.0254	0.29		4.05	2.37	4.71 E-2
20	0.0508	0.0762	4.25 E-3		1.06 E-1	6.65 E-2	0.63
20	0.0508	0.127	1.6 E-3		7.45 E-2	4.66 E-2	6.08 E-4
90	0.229	0.267	5.33 E-4		4.88 E-2	3.13 E-2	0.64
100	0.254	0.508	4.01 E-4		2.66 E-2	1.72 E-2	0.65
200	0.508	0.889	2.62 E-4		1.1 E-2	7.27 E-3	0.66
200	0.508	1.397	1.97 E-4		3.18 E-3	2.23 E-3	0.70
842.25	2.139	3.79					
62.5	0.159	3.869			9.4 E-3	5.95 E-3	0.63
31.25	0.0794	3.988	1.16 E-3		4.34 E-2	2.75 E-2	0.63
31.25	0.0794	4.068	2.40 E-2	2.80 E-2	2.58 E-1	1.6 E-1	0.62

Table 4.3-2. Radial distribution of radionuclides in sleeve at compact 5
(corrected to September 14, 1973)

Radial width of sample (mils)	(cm)	Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)								$^{137}\text{Cs}/^{134}\text{Cs}$
			^{54}Mn	^{60}Co	^{106}Ru	^{110m}Ag	^{134}Cs	^{137}Cs	^{144}Ce	^{154}Eu	
5	0.0127	0.0064	1.61	1.40	42.56	5.01	1500.0	913.0	48.1	0.98	0.61
5	0.0127	0.0191		1.09			1620.0	975.0			0.61
5	0.0127	0.0318	< 2.72	1.13	29.71	5.92	1300.0	779.0	< 305.0		0.60
10	0.0254	0.0509	< 1.97	0.69	23.83	4.7	910.0	545.0	< 205.0		0.60
15	0.0381	0.0826	< 1.16	0.81	15.38	< 2.57	526.0	315.0	173.0	0.48	0.60
30	0.0762	0.140	< 0.87	0.84	11.44	< 1.95	403.0	241.0	< 87.7	0.36	0.60
30	0.0762	0.216	0.49	1.11	6.65	< 1.37	275.0	164.0	< 59.9	0.49	0.60
30	0.0762	0.292	0.38	1.53	4.59	< 0.85	165.0	95.5	26.6	0.62	0.60
30	0.0762	0.368									
30	0.0762	0.445	0.28	1.91	3.63	< 0.41	48.6	28.6	< 16.0		0.59
30	0.0762	0.521									
30	0.0762	0.597	0.17	1.86	0.40	< 0.087	9.64	5.80	< 2.76		0.60
30	0.0762	0.673	0.16	1.62	0.35	< 0.084	3.83	2.30	< 2.44	0.57	0.60
30	0.0762	0.749	0.11	1.35		< 0.056	1.21	0.75	0.27		0.62
30	0.0762	0.826	0.058	0.96	0.19	< 0.023	0.23	0.15	0.23	0.48	0.66
15	0.0381	0.883	0.067	1.26	0.043	< 0.046	0.18	0.13	0.17	0.54	0.69
15	0.0381	0.971	0.12	1.59		< 0.033	0.29	0.20	0.24	0.49	0.69
8	0.0203	0.950	2.83	8.22		2.35	2.59	1.78	0.44	0.67	0.69

Table 4.3-3. Radial distribution of radionuclides in spine at compact 5
(corrected to September 14, 1973)

Radial width of sample (mils)	(cm)	Distance to center of cut from OD (cm)	Radionuclide concentration (μCi/g)								
			⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
5	0.0127	0.0064	81.7	16.81	1202.0	140.2	3.93 E4	2.41 E4	4.64 E2		0.61
15	0.0381	0.0318	12.1	5.95	421.0	671.3	1.47 E4	8.95 E4	1.22 E3		0.61
15	0.0381	0.0699	<31.6	5.7	<560.0	178.5	1.06 E4	6.36 E3	3.88 E3		0.60
30	0.0762	0.127	<20.3	5.31	357.0	<47.3	7.29 E3	4.35 E3	2.55 E3		0.60
30	0.0762	0.203		3.42			3.78 E3	2.27 E3			0.60
30	0.0762	0.279		3.21			2.13 E3	1.27 E3			0.60
30	0.0762	0.356		3.46			1.21 E3	7.2 E2			0.60
30	0.0762	0.432		3.09			6.64 E2	3.93 E2			0.59
30	0.0762	0.508		3.56			3.9 E2	2.31 E2		0.76	0.59
30	0.0762	0.584		4.03			2.21 E2	1.31 E2		0.32	0.59
30	0.0762	0.660		3.71			1.35 E2	79.5		0.63	0.59
60	0.152	0.775	0.44	3.62			68.5	40.4		0.80	0.59
60	0.152	0.927	0.58	3.52			26.7	15.4		0.76	0.58
26	0.157	1.08	0.53	3.54			10.8	5.93		0.81	0.56
188	0.478	1.64	-	-							
625	0.159	1.72	0.51	3.08			1.78	0.58		0.75	0.33
125	0.318	1.96	0.51	3.07			1.70	0.51	0.33	0.83	0.30

Table 4.3-4. Radial distribution of radionuclides in sleeve at compact 12
(corrected to September 14, 1973)

Radial width of sample (mils)	Distance to center of cut from ID (cm)	Radionuclide concentration (μCi/g)									
		⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs	
5	0.0127	0.0064	<1.62 E2		<3.17 E3	2.65 E3	4.47 E4	2.84 E4	<2.24 E4	1.92 E2	0.64
5	0.0127	0.0191	<1.36 E2	42.96	<2.70 E3	2.12 E3	4.1 E4	2.54 E4	<1.89 E4		0.62
5	0.0127	0.0318	<1.34 E2	32.04	2.37 E3	2.44 E3	3.84 E3	2.34 E4	<1.87 E4		0.61
10	0.0254	0.0509	65.7	7.43	1.10 E3	1.76 E2	2.49 E4	1.49 E4	<1.06 E4		0.60
20	0.0508	0.089	<50.5	7.05	<8.74 E2	<1.13 E2	1.87 E4	1.12 E4	<6.58 E3		0.60
30	0.0762	0.152	<33.3	7.74	3.28 E2	<73.7	1.31 E4	7.75 E3	<3.61 E3		0.59
30	0.0762	0.229	31.8	2.42	<3.28 E2	<50.6	8.7 E3	4.97 E3	<2.5 E3		0.57
18	0.0457	0.290	<17.1	4.15	<3.05 E2	<39.0	6.01 E3	3.43 E3	<2.45 E3		0.57
30	0.0762	0.351									
30	0.0762	0.427	8.94	3.15	47.96	<10.7	2.37 E3	1.34 E3	710.0		0.57
30	0.0762	0.503									
30	0.0762	0.579	1.75	2.62	29.22	3.55	683.2	377.0	230.0		0.55
30	0.0762	0.655	1.03	2.40	9.42	<1.8	368.7	204.0	<78.3	0.42	0.55
30	0.762	0.732	0.32	2.01	9.92	<0.9	176.8	96.6	<37.8	0.58	0.55
30	0.762	0.809	0.17	1.83	1.79	<0.4	71.88	39.0	<16.4	0.56	0.54
15	0.0381	0.865	0.12	2.23		<0.046	27.15	15.0	0.18	0.69	0.55
15	0.0381	0.903	0.19	3.56		<0.14	12.38	6.97	4.2	0.87	0.56
5	0.0127	0.929	8.12	27.92	4.76	2.63	59.62	36.2	4.81	8.46	0.61

Table 4.3-5. Radial distribution of radionuclides in spine at compact 12
(corrected to September 14, 1973)

Radial width of sample		Distance to center of cut from OD (cm)	Radionuclide concentration (μCi/g)								
(mils)	(cm)		⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
8	0.0203	0.0101	<15.8	14.25	5.85 E2	<36.4	2.18 E4	1.38 E4	1.11 E3	51.9	0.63
15	0.0381	0.0292	<48.1	5.99	<7.39 E2	<1.09 E2	1.82 E4	1.11 E4	5.41 E3	38.0	0.61
15	0.0381	0.0673	<44.0	4.49	7.27 E2	<98.3	1.64 E4	9.75 E3	<5.33 E3	41.0	0.59
30	0.0762	0.124	<34.8	1.41	4.64 E2	<78.1	1.34 E4	7.95 E3	<3.91 E3	32.0	0.59
30	0.0762	0.201	<27.5	6.22	<4.18 E2	<60.9	1.02 E4	5.92 E3	<3.13 E3	28.7	0.58
30	0.0762	0.277	<21.6	6.39	<3.58 E2	<49.0	7.76 E3	4.5 E3	<2.69 E3	30.9	0.58
30	0.0762	0.353	<18.0	5.16	2.67 E2	<41.3	5.86 E3	3.34 E3	<2.29 E3	27.5	0.57
30	0.0762	0.429	<14.5	4.45	<2.73 E2	<33.7	4.37 E3	2.48 E3	1.38 E3	28.5	0.57
30	0.0762	0.505	<12.2		1.91 E2	<29.5	3.28 E3	1.85 E3	<1.77 E3		0.56
30	0.0762	0.582	<10.3	4.59	2.02 E2	<26.2	2.53 E3	1.41 E3	<1.54 E3	17.2	0.56
30	0.0762	0.658	< 4.62	3.24	<60.0	<10.3	1.77 E3	9.92 E2	<5.08 E2	16.1	0.56
60	0.1524	0.772	< 3.13	1.95	<39.8	< 6.81	1.22 E3	6.72 E2	<3.33 E2	13.2	0.55
60	0.1524	0.925	< 1.57	1.23	27.4	< 4.6	7.1 E2	3.9 E2	<2.32 E2	11.3	0.55
68	0.1727	1.087	< 1.22	0.73	17.5	< 2.85	3.71 E2	2.00 E2	<1.58 E2	6.75	0.54
62.5	0.159	1.70	0.67	1.55	<11.2	< 2.06	53.6	27.9	<89.7		0.52
125	0.3175	1.960	0.54	1.31	< 7.15	< 1.33	37.25	17.8	<58.4	1.70	0.48

Table 4.3-6. Radial distribution of radionuclides in sleeve at compact 18
(corrected to September 14, 1973)

Radial width of sample (mils)	Distance to center of cut from ID (cm)	Radionuclide concentration (μCi/g)									
		⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs	
5	0.0127	0.0064	<78.1	84.3	<9.7 E2	<1.8 E2	5.26 E4	3.22 E4	<8.72 E3	9.48 E3	0.61
5	0.0127	0.0191	<2.87 E2	79.0	<3.2 E3	<6.59 E2	3.86 E4	2.36 E4	<3.04 E4	1.01 E4	0.61
5	0.0127	0.0318	<2.56 E2	53.1	<2.65 E3	<5.8 E2	3.21 E4	1.95 E4	<2.53 E4	1.16 E4	0.61
5	0.0127	0.0445	<1.99 E2	26.6	<2.13 E3	3.09 E2	2.59 E4	1.58 E4	<2.1 E4	9.6 E3	0.61
15	0.0381	0.0699	<1.05 E2	5.1	<1.2 E3	1.5 E2	2.51 E4	1.54 E4	<1.11 E4	4.24 E3	0.61
30	0.0762	0.127	<58.2	3.7	<6.9 E2	<1.28 E2	2.03 E4	1.24 E4	4.86 E3	2.15 E2	0.61
30	0.0762	0.203	<35.9	2.5	<5.2 E2	<79.2	1.31 E4	7.83 E3	<4.1 E3		0.60
30	0.0762	0.279	<20.4	3.4	<3.6 E2	<46.6	6.92 E3	4.15 E3	<2.68 E3		0.60
30	0.0762	0.356									
30	0.0762	0.432		2.5			1.78 E3	1.05 E3			0.59
30	0.0762	0.508									
30	0.0762	0.584		2.9			440.0	255.0			0.58
30	0.0762	0.661		2.7			185.8	105.0			0.57
30	0.0762	0.737		2.0			60.0	34.0		0.34	0.57
30	0.0762	0.813		2.0			21.9	12.0		0.57	0.56
15	0.0381	0.870	0.10	2.3			6.5	4.0		0.58	0.56
15	0.0381	0.908	0.21	3.5			3.7	2.0	0.45	0.69	0.57
9	0.0229	0.939	3.56	14.6			8.5	5.0		1.05	0.60

Table 4.3-7. Radial distribution of radionuclides in spine at compact 18
(corrected to September 14, 1973)

Radial width of sample (mils)	Distance to center of cut from OD (cm)	Radionuclide concentration (μCi/g)									
		⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs	
10	0.0254	0.0127	< 8.1	64.7	82.6	<19.6	4.76 E3	3.28 E3	1.41 E3	3.03 E2	0.69
15	0.0381	0.0445	<18.0	25.9	<3.27 E2	<43.7	4.3 E3	2.93 E3	<2.5 E3	94.3	0.68
15	0.0381	0.0826	<18.6	27.7	<3.26 E2	<45.0	4.09 E3	2.76 E3	-	86.3	0.67
30	0.0762	0.1397	<13.3	14.8	<2.29 E2	<31.2	3.84 E3	2.53 E3	<1.8 E3	95.2	0.66
30	0.0762	0.2159	<12.6	15.4	<2.17 E2	<29.8	3.27 E3	2.08 E3	<1.7 E3	80.9	0.64
30	0.0762	0.2921	9.0	18.8	<2.09 E2	<29.0	2.89 E3	1.8 E3	<1.6 E3	73.8	0.62
30	0.0762	0.3683	<11.8	17.1	1.58 E2	<28.7	2.48 E3	1.52 E3	<1.5 E3	70.7	0.61
30	0.0762	0.4445	<12.1	19.1	<1.9 E2	<28.4	2.15 E3	1.3 E3	<1.5 E3	66.8	0.60
30	0.0762	0.5207	<11.6	18.6	<1.8 E2	<27.7	1.87 E3	1.08 E3	<1.5 E3	67.0	0.58
30	0.0762	0.5969	< 6.4	3.7	<1.1 E2	<15.8	1.57 E3	9.28 E2	<8.5 E2	62.5	0.59
60	0.1524	0.7112	< 4.7	2.5	<74.7	<10.8	1.25 E3	7.21 E2	<5.8 E2	58.7	0.58
60	0.1524	0.8636	< 4.3	2.5	<69.5	<10.2	9.37 E2	5.25 E2	<5.5 E2	54.0	0.56
60	0.1524	1.016	< 2.7	1.8	<45.5	< 6.3	7.24 E2	3.87 E2	<3.5 E2	38.9	0.53
40	0.1016	1.143	< 2.6	3.2	<52.4	< 8.1	5.64 E2	3.0 E2	<4.2 E2	36.1	0.53
62.5	0.159	1.70	< 2.4	2.6	<27.6	< 5.8	1.31 E2	67.9	<2.7 E2	22.3	0.52
125	0.3175	1.9873	2.4	7.0		< 5.5	1.89 E2	1.0 E2	<2.2 E2	56.3	0.53

Table 4.3-8. Radial distribution of radionuclides in sleeve at compact 28
(corrected to September 14, 1973)

Radial width of sample (mils)	width of (cm)	Distance to center of cut from ID (cm)	Radionuclide concentration (μCi/g)								
			⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
5	0.0127	0.0064	7.46	80.19		340.0	54.5	66.2	197.3	65.37	1.21
5	0.0127	0.0191	1.07	17.06		369.0	19.35	22.6	39.1	6.76	1.17
5	0.0127	0.0318	0.75	8.31		153.0	16.87	20.0	30.6	5.12	1.18
5	0.0127	0.0445		1.86		141.0	12.19	14.6	0.85	0.45	1.20
15	0.0381	0.0699		1.24		9.1	8.57	9.76		0.329	1.14
30	0.0762	0.127		1.07		0.25	6.29	6.95		0.532	1.10
30	0.0762	0.203	0.053	1.02	0.14	<0.009	2.90	3.00	<0.30	0.517	1.03
30	0.0762	0.279	0.061	1.39			1.58	1.57		0.546	0.99
30	0.0762	0.356									
30	0.0762	0.432	0.131	2.08			0.603	0.564	0.18	0.631	0.94
30	0.0762	0.508									
30	0.0762	0.584	0.130	1.83			0.193	0.184	0.14	0.600	0.95
30	0.0762	0.660	0.109	1.76	0.078		0.133	0.128	0.21	0.665	0.96
30	0.0762	0.737	0.094	1.46			0.097	0.093	0.24	0.676	0.96
30	0.0762	0.813	0.077	1.29	0.15		0.093	0.072	0.19	0.666	0.77
15	0.0381	0.870	0.108	1.70	0.21		0.159	0.116	0.18	0.788	0.73
15	0.0381	0.908	0.196	2.47			0.335	0.218	0.14	0.735	0.65
10	0.0254	0.940	1.85	6.52			1.93	1.21	0.79	1.55	0.63

Table 4.3-9. Radial distribution of radionuclides in spine at compact 28

(corrected to September 14, 1973)

Radial width of sample (mils) (cm)		Distance to center of cut from OD (cm)	Radionuclide concentration (μCi/g)							
			⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
8	0.0203	0.0102	20.23	86.4		54.56	61.04	1106.5	17.46	1.12
15	0.0381	0.0394	8.0	0.15	0.059	3.82	3.32	9.9	12.15	0.87
15	0.0381	0.0775	7.23			3.34	2.47		10.28	0.74
30	0.0762	0.135	6.21			2.80	2.11		8.47	0.75
30	0.0762	0.211	5.22			2.58	1.85		7.17	0.72
30	0.0762	0.287	4.84			2.91	2.09		6.26	0.72
30	0.0762	0.363	3.47			2.26	1.52		4.46	0.67
30	0.0762	0.439	2.70			2.03	1.33		3.78	0.66
30	0.0762	0.516	2.14			1.94	1.21		3.18	0.62
30	0.0762	0.592	1.73			1.76	1.05		2.60	0.60
30	0.0762	0.668	1.49			1.81	1.04		2.19	0.57
30	0.0762	0.744	1.37			1.61	0.89		1.87	0.55
28	0.0711	0.818	1.26			1.49	0.79		1.57	0.53
309	0.785	1.638								
62.5	0.159	1.718	0.90			1.03	0.35		0.80	0.34
125	0.318	1.956	0.93			1.00	0.30		0.88	0.30

Table 4.3-10. Radial distribution of gamma-emitting radionuclides found in
bottom connector of Ell-07
(corrected to September 14, 1973)

Radial width of sample (mils)	Distance to center of cut from OD (cm)	Distance to center of cut from OD (cm)	Radionuclide concentration (μCi/g)								
			⁵⁴ Mn	⁶⁰ Co	¹⁰⁶ Ru	^{110m} Ag	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu	¹³⁷ Cs/ ¹³⁴ Cs
20	0.0508	0.025	7.1 E-2	0.74	<0.14	0.13	1.75	1.54		<9.6 E-3	0.88
40	0.1016	0.102	<1.9 E-3	9.3 E-3	<1.0 E-2	<3.8 E-3	1.97 E-2	2.25 E-2		<2.1 E-4	1.14
40	0.1016	0.203	<1.8 E-3	6.4 E-3	<1.3 E-2	<3.2 E-3	6.17 E-2	4.66 E-2		<2.4 E-4	0.75
60	0.1524	0.330	1.8 E-2	1.9 E-1	<3.6 E-2	3.3 E-2	0.44	0.39		<2.4 E-3	0.88
60	0.1524	0.635	<6.4 E-4	3.4 E-3	<5.1 E-3	<1.1 E-3	1.13 E-2	8.51 E-3		<1.0 E-4	0.75
60	0.1524	0.940	<2.0 E-3	7.5 E-3	<6.2 E-3	<2.6 E-2	1.13 E-2	9.62 E-3	<6.7 E-2	<1.9 E-4	0.85
60	0.1524	1.245	<1.0 E-3	3.7 E-3	<7.5 E-3	<2.1 E-3	2.73 E-2	2.12 E-2	<6.8 E-2	<1.3 E-4	0.78
60	0.1524	1.549	<6.6 E-4	2.2 E-3	<4.6 E-3	<1.6 E-3	4.1 E-3	3.28 E-3	<4.3 E-2	<9.8 E-5	0.8
60	0.1524	1.854	<1.1 E-3	5.7 E-3	<8.5 E-3	<2.7 E-3	2.54 E-2	1.97 E-2	<7.7 E-2	<1.8 E-4	0.78
60	0.1524	2.159	<7.9 E-4	6.7 E-3	<7.9 E-3	<2.4 E-3	1.83 E-2	1.41 E-2	<8.3 E-2	<1.7 E-4	0.77
60	0.1524	2.464	<1.0 E-3	2.6 E-3	<9.1 E-3	<2.2 E-3	3.05 E-2	2.41 E-2	<8.3 E-2	<1.8 E-4	0.79
62.5	0.1588	3.553	2.7 E-3	9.2 E-3	<1.8 E-2	<4.3 E-3	0.17	9.4 E-2		2.4 E-3	0.81
62.5	0.1588	3.712	1.5 E-2	9.8 E-2	<3.7 E-2	<3.8 E-2	0.18	0.16		3.0 E-3	0.94

Fig. 4.3-4. Radial distribution of some gamma emitters in sleeve and spine at compact 5 (corrected to September 14, 1973).

Fig. 4.3-5. Radial distribution of gamma emitters at compact 12
(corrected to September 14, 1973).

Fig. 4.3-6. Radial distribution of gamma emitters at compact 18
 (corrected to September 14, 1973).

Fig. 4.3-7. Radial distribution of gamma emitters at compact 28
(corrected to September 14, 1973).

Fig. 4.3-8. Radial distribution of gamma-emitting radionuclides in E11-07 top reflector 16 cm from top end (corrected to September 14, 1973).

Fig. 4.3-9. Radial distribution of the nuclide ratio $^{137}\text{Cs}/^{134}\text{Cs}$ in the sleeve and spine (corrected to September 14, 1973).

for mass 137 and 134, respectively, or differences in the time spent as the xenon precursor, 3 min vs 5 days for mass 137 and 134 respectively. The general fall in the $^{137}\text{Cs}/^{134}\text{Cs}$ ratio observed proceeding inward toward the center of the spine is due to the greater penetrating ability that ^{134}Cs gains by virtue of the greater time spent as the xenon precursor. This view is consistent with the minimum shown in the sleeve interior of compact locations 5, 12, and 18; that is, the interior sleeve being least accessible contains proportionally more of the ^{134}Cs isotope. The rather unique behavior of the $^{137}\text{Cs}/^{134}\text{Cs}$ ratio at compact 5 is difficult to explain.

5. BURNUP ANALYSES AND BETA-EMITTING NUCLIDE DISTRIBUTIONS

5.1 Burnup Determinations*

The number of fissions per initial heavy-metal atom (FIMA) was determined for six samples, each consisting of a pair of fuel particles from compacts 9, 16, and 28. The method consisted in first determining the amount of stable zirconium in the sample colorimetrically and then determining the uranium and thorium contents by both colorimetric and isotope dilution mass spectrographic (IDMS) methods. The colorimetric procedures, described in ref. 6, are based on the method of Buzzelli and Smith.⁹ Initially, some difficulty was experienced in obtaining a satisfactory separation of Th (Zr, U, and Th must each be separately determined for the colorimetric methods); hence, IDMS analyses were performed on both U and Th for most samples as a check.

However, the chief limitation on the accuracy of the derived value of FIMA rests on the zirconium determination, since there was so little of this material--only 1 to 3 μg for each particle pair. Significant increase in precision will be obtained when IDMS methods are developed for zirconium in addition to heavy metal, at which time the difficult separation steps in the analysis would not be necessary. To increase the

*Measurements performed by J. L. Botts of the Analytical Chemistry Division.

accuracy of the zirconium assay, the ^{95}Zr level was determined for each sample by gamma spectroscopy prior to and following dissolution and separation. By this means a determination of the separation efficiency of zirconium was obtained.

Table 5.1-1 lists burnup results from duplicate samples taken from compacts 9, 16, and 28. FIMA were calculated from the relation

$$\text{FIMA} = \frac{N_{\text{Zr}}/Y}{N_{\text{Th}} + N_{\text{U}} + N_{\text{Zr}}/Y},$$

where

N = moles of Zr, Th, or U in sample,

$Y = 0.0031$, atoms of stable Zr per fission.

The tabulated results show a good agreement in most cases between the colorimetric and the IDMS determinations of thorium and uranium. Therefore, the large differences in FIMA between samples from the same compact are undoubtedly due to inaccurate zirconium-level determinations.

Calculated fissile and fertile burnups are listed in Table 5.1-2 for each fuel compact. According to ref. 10, where these burnups are precisely defined, fissile-FIMA signifies the mole fraction of initial fissile material (in this case only ^{235}U) that fissions; fertile-FIMA signifies the mole fraction of initial ^{232}Th that ultimately fissioned in transmuted forms. In other words, fertile-FIMA does not count heavy-metal transmutations of ^{232}Th as being lost. Calculated relative axial distributions of the major heavy-metal nuclides as a function of irradiation time are presented in Figs. 33 through 40 of ref. 10.

5.2 ^3H , ^{14}C , and ^{90}Sr Determinations in Fuel Particles

The beta emitters ^3H and ^{14}C were determined for both fuel particle and graphite samples by initially burning the sample at 850°C in wet oxygen. The tritium is converted to HTO and collected in a trap cooled with dry ice and acetone. The ^{14}C is converted to $^{14}\text{CO}_2$ and passes through the cold trap to be absorbed in Hyamine. Strontium-90 determinations for graphite were performed by initially wet-ashing the samples in concentrated nitric

Table 5.1-1. Burnup analyses results

Compact	Sample ^a	Colorimetric				IDMS		
		Zr (μg)	Th (μg)	U (μg)	FIMA (%)	Th (μg)	U (μg)	FIMA (%)
9	1	1.18	172	30.2	4.54	197	25.8	4.14
	2	1.70	160	19.1	7.19	155	19.4	7.35
	3	2.00	228	27.5	6.00			
	4	2.55	191	23.7	8.86	186	24.4	8.98
	5	2.50	184			152	22.6	8.99
	6	2.07	165	18.9	8.40			
16	1	1.05	160	17.3	4.60	162	19.5	4.75
	2	1.80	140	16.5	8.57	138	17.9	8.60
	3	2.20	246	29.7	6.10	233	29.7	6.38
	4	2.55	194	22.1	8.76			
	5	2.35	180			176	22.3	8.80
	6	3.10	270			246	32.6	8.31
28	1	1.49	186	26.8	5.40	203	29.0	4.96
	2	1.80	178	23.1	6.80	156	27.2	7.41
	3	1.00	157	22.3	4.35	150	22.9	4.48
	4	1.50	209	23.2	5.00	204	23.8	5.09
	5	1.75	216			219	32.1	5.37
	6	1.76	232			233	34.7	5.07

^aTwo fuel particles.

Table 5.1-2. Calculated FIMA's for Ell-07 fuel compacts^a

Compact	Axial location (in.)	Fissile- FIMA (%)	Fertile- FIMA (%)	Mixed (5.5 Th, 1 U)C ₂ FIMA (%)
1	1.5	28.71	0.35	4.74
2	4.5	28.71	0.35	4.47
3	7.5	34.45	0.58	5.76
4	10.5	38.31	0.80	6.56
5	13.5	41.63	1.00	7.21
6	16.5	43.89	1.61	7.72
7	19.5	45.73	1.38	8.16
8	22.5	46.96	1.52	8.45
9	25.5	47.99	1.64	8.74
10	28.5	48.60	1.72	8.89
11	31.5	49.06	1.76	8.96
12	34.5	49.42	1.80	9.04
13	37.5	49.51	1.80	9.12
14	40.5	49.53	1.80	9.10
15	43.5	49.22	1.78	9.04
16	46.5	48.60	1.73	8.95
17	49.5	47.91	1.66	9.74
18	52.5	47.09	1.59	8.53
19	55.5	46.14	1.52	8.38
20	58.5	45.12	1.43	8.16
21	61.5	43.89	1.31	7.90
22	64.5	42.25	1.19	7.51
23	67.5	40.48	1.07	7.07
24	70.5	38.51	0.95	6.70
25	73.5	36.09	0.81	6.27
26	76.5	34.25	0.69	5.83
27	79.5	31.58	0.55	5.39
28	82.5	29.54	0.43	4.95
29	85.5	27.89	0.32	4.59
30	88.5	27.89	0.32	4.59
Average		41.02	1.15	7.29

^aInternal correspondence, GAC, Frances McCord to C. F. Wallroth, "Peach Bottom Core 2 Driver Element Ell-07 Irradiation Summary," June 25, 1975.

acid and separating the strontium, using standard techniques. After the indicated separations, inventories of ^3H , ^{14}C , and ^{90}Sr were obtained by beta counting.

Results are given in Table 5.2-1 for samples consisting of a pair of particles taken from compacts 2, 13, 20, and 24. The left portion of the table lists the inventory of beta emitters in each sample. In the right-hand portion of the table, the beta-emitter inventories are normalized relative to the observed ^{144}Ce level in each sample. Since ^{144}Ce is retained within an intact particle, its inventory as determined by gamma spectroscopy may be used as a measure of fuel mass in samples of equal burnup. For completeness, the observed inventories of gamma-emitting nuclide for each sample, plus some additional particle pairs, are given in Table 5.2-2.

5.3 Radial Distribution of ^3H , ^{14}C , and ^{90}Sr in Sleeve and Spine*

The radial distribution of the beta emitters ^3H , ^{14}C , and ^{90}Sr in the sleeve and spine at compact locations 5, 12, 18, 28, and the bottom connector are presented in Tables 5.3-1 through 5.3-9. These samples were acquired using the manipulator-operated lathe installed in a hot cell described in Sect. 4.1.

Figures 5.3-1 through 5.3-3 illustrate the observed radial distributions of ^3H , ^{14}C , and ^{90}Sr found in the sleeve and spine at compacts 18 and 28 and in the bottom connector. Table 5.3-10 lists the observed levels of these beta emitters found in the charcoal trap as a function of axial location in the trap.

Standard procedures were employed to determine concentrations of these beta-emitting nuclides. For tritium and ^{14}C , a graphite sample was burned in wet oxygen, the tritiated water being collected by condensation and the carbon dioxide in an alkaline solution. Strontium-90 was determined by wet ashing in concentrated nitric acid and chemically separating ^{90}Sr plus carbon by standard analytical procedures. Concentration levels for each separate nuclide were then determined by beta counting. About half

* Measurements performed by Hugh Parker and P. S. Gouge of the Analytical Chemistry Division.

Table 5.2-1. Tritium, ^{14}C , and ^{90}Sr in fuel particle samples as of September 14, 1973

Compact	Sample ^a	Radionuclide concentration ($\mu\text{Ci/sample}$)			$^3\text{H}/^{144}\text{Ce}$	$^{14}\text{C}/^{144}\text{Ce}$	$^{90}\text{Sr}/^{144}\text{Ce}$
		^3H	^{14}C	^{90}Sr			
2	1,2			18.3			0.073
	3,4	0.067	2.9 E-4		3.2 E-4	1.4 E-6	
	5,6	0.048	2.7 E-4		1.5 E-4	8.7 E-7	
	7,8			13.2			0.110
	9,10	0.122	7.6 E-4		4.10 E-4	2.5 E-6	
	11,12			33.2			0.079
13	1,2			23.3			0.100
	3,4			7.3			0.040
	5,6			18.3			0.056
	7,8	0.069	2.77 E-3		1.8 E-4	7.39 E-6	
	9,10	0.185	5.63 E-3		2.40 E-4	7.32 E-6	
	11,12	0.062	2.55 E-3		1.8 E-4	7.26 E-6	
20	1,2			5.3			0.018
	3,4	<1.4 E-3	3.04 E-3		<3.8 E-6	8.23 E-6	
	5,6	<6.6 E-4	2.11 E-3		<3.0 E-6	9.45 E-6	
	7,8			14.7			0.049
	9,10	<7.0 E-4	1.65 E-3		<2.7 E-6	6.39 E-6	
	11,12			6.6			0.030
24	1,2			19.8			0.069
	3,4			20.8			0.063
	5,6			19.9			0.081
	7,8	0.037	2.79 E-3		7.6 E-5	5.72 E-6	
	9,10	0.047	2.79 E-3		1.3 E-4	6.18 E-6	
	11,12	0.068	3.45 E-3		1.1 E-4	5.67 E-6	

^aEach sample consists of two particles.

Table 5.2-2. Gamma-emitting nuclide inventories in fuel particle samples as of September 14, 1973

Compact	Sample ^a	Radionuclide concentration ($\mu\text{Ci/sample}$)					
		¹⁰⁶ Ru	¹²⁵ Sb	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁵⁴ Eu
2	1,2	21.7	1.08	16.9	24.5	252	0.517
	3,4	18.9	0.91	15.4	22.1	212	0.354
	5,6	24.4	1.02	15.0	20.8	312	0.280
	7,8	12.8		10.8	16.0	119	
	9,10	25.5		27.8	40.7	297	
	11,12	33.7	1.65	26.2	38.5	421	0.636
	13,14	32.9	1.24	26.7	39.3	396	0.497
	15,16	28.2	1.23	20.5	29.9	309	0.468
	17,18	22.5	1.14	17.0	25.5	239	0.314
13	1,2	27.3	1.34	36.5	31.4	233	0.343
	3,4	25.0	1.65	34.4	27.7	180	
	5,6	41.0	1.65	52.2	45.8	329	
	7,8	40.0	2.33	70.6	50.2	375	0.916
	9,10	76.3	4.32	142.1	98.4	769	2.57
	11,12	34.2	1.90	59.0	41.3	351	0.51
	13,14	36.4	2.05	62.9	44.3	400	1.01
	15,16	37.0	1.51	62.2	44.2	389	0.99
	17,18	34.7	1.94	61.4	44.1	430	0.96
20	1,2	49.8	2.86	17.3	21.8	288	
	3,4	60.3	4.08	22.6	28.0	369	
	5,6	39.1	2.24	14.1	17.6	223	
	7,8	52.6	3.25	21.6	25.7	299	
	9,10	46.9	2.84	18.2	21.6	258	
	11,12	32.2	2.21	13.2	15.7	221	
	13,14	40.1	2.85	13.5	16.7	250	
	15,16	31.7	1.81	11.4	14.0	281	
	17,18	55.9	3.23	18.4	24.0	379	
24	1,2	36.6	2.03	32.2	30.5	285	0.299
	3,4	39.4	2.03	48.9	41.2	330	0.226
	5,6	28.7	1.77	37.2	31.8	246	0.337
	7,8	65.6	3.08	77.7	66.6	487	0.357
	9,10	42.9	2.30	56.0	47.8	445	0.674
	11,12	66.3	2.89	81.3	68.3	608	0.496
	13,14	26.7	1.25	36.2	31.2	219	
	15,16	43.0	1.74	53.4	46.1	376	0.530
	17,18	51.1	2.01	65.5	56.3	424	0.768

^aEach sample consists of a pair of particles. Particle numbers (col. 2) are same as for beta emitters given in Table 5.2-1.

Table 5.3-1. Radial distribution of beta-emitting radionuclides
in sleeve at compact 5
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
			^3H	^{14}C	^{90}Sr
5	0.0127	0.0064	12.9	0.147	25.42
5	0.0127	0.0191	28.6	0.14	15.65
5	0.0127	0.0318	71.3	0.126	7.44
10	0.0254	0.0509	93.4	0.134	3.33
30	0.0762	0.1398	29.2	0.114	0.037
30	0.0762	0.292	18.0	0.135	0.0766
30	0.0762	0.446	11.4	0.112	0.044
30	0.0762	0.597	11.3	0.131	0.111
30	0.0762	0.749	9.5	0.136	0.068
15	0.0381	0.883	11.0	0.156	0.103
8	0.0203	0.95	21.6	0.721	1.88

^aThickness of sample milled from sleeve.

Table 5.3-2. Radial distribution of beta-emitting radionuclides
in spine at compact 5
(corrected to September 14, 1973)

Thickness ^a		Distance to center of cut from OD (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
(mils)	(cm)		^3H	^{14}C	^{90}Sr
5	0.0127	0.0064	4.19	0.561	182.0
15	0.0381	0.0318	5.14	0.537	24.6
15	0.0381	0.0699	3.88	0.473	0.140
30	0.0762	0.127	4.66	0.487	0.085
30	0.0762	0.203	5.01	0.423	0.101
30	0.0762	0.356	4.65	0.413	0.074
30	0.0762	0.508	4.40	0.526	0.074
30	0.0762	0.660	3.50	0.467	0.060
60	0.152	0.927	3.48	0.512	0.058
62.5	0.159	1.72	2.79	0.535	0.115
125	0.318	1.96	2.50	0.488	0.157

^aThickness of sample milled from spine.

Table 5.3-3. Radial distribution of beta-emitting radionuclides
in sleeve at compact 12
(corrected to September 14, 1973)

Thickness ^a		Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
(mils)	(cm)		^3H	^{14}C	^{90}Sr
5	0.0127	0.0064	31.3	0.40	6010.0
5	0.0127	0.019	27.5	0.262	2878.0
5	0.0127	0.032	27.6	0.604	2231.0
10	0.0254	0.051	23.4	0.20	762.0
30	0.0762	0.153	87.4	0.262	5.42
18	0.0457	0.2897	55.4	0.319	3.81
30	0.0762	0.427	60.2	0.152	0.19
30	0.0762	0.579	71.7	0.186	0.19
30	0.0762	0.732	12.76	0.218	1.54
15	0.0381	0.865	11.17	0.23	2.86
5	0.0127	0.93	12.37	1.40	102.0

^aThickness of sample milled from sleeve.

Table 5.3-4. Radial distribution of beta-emitting radionuclides
in spine at compact 12
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from OD (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
			^3H	^{14}C	^{90}Sr
8	0.0203	0.010	16.6	1.12	1920.0
15	0.0381	0.0394	11.1	0.75	1882.0
15	0.0381	0.0774	8.64	0.75	1490.0
30	0.0762	0.135	7.33	0.82	1652.0
30	0.0762	0.211	6.15	0.7	1205.0
30	0.0762	0.363	4.66	0.68	1242.0
30	0.0762	0.516	3.59	0.75	905.0
30	0.0762	0.668	2.58	0.72	731.0
60	0.1524	0.935	2.03	0.78	406.0
62.5	0.159	1.699	0.86	0.72	34.2
125	0.318	1.94	0.80	0.79	20.9

^aThickness of sample milled from spine.

Table 5.3-5. Radial distribution of beta-emitting radionuclides
in sleeve at compact 18
(corrected to September 14, 1973)

Thickness ^a		Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
(mils)	(cm)		^3H	^{14}C	^{90}Sr
5	0.0127	0.0064	32.5	0.799	8.62 E4
5	0.0127	0.019	23.3	0.718	1.02 E5
5	0.0127	0.032	13.2	0.391	1.15 E5
5	0.0127	0.045	3.8	0.197	1.59 E5
30	0.0762	0.127	4.57	0.235	2.9 E4
30	0.0762	0.279	4.75	0.174	15.6
30	0.0762	0.432	5.7	0.193	0.046
30	0.0762	0.584	9.1	0.149	0.071
50	0.0762	0.737	14.3	0.180	0.135
15	0.0381	0.87	24.7	0.193	0.62
9	0.0229	0.94	28.4	0.878	8.37

^a Thickness of sample milled from sleeve.

Table 5.3-6. Radial distribution of beta-emitting radionuclides
in spine at compact 18
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from OD (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
			^3H	^{14}C	^{90}Sr
10	0.0254	0.0127	7.47	1.29	12471
15	0.0381	0.0445	5.49	0.758	2597
15	0.0381	0.0826	7.23	0.722	2767
30	0.0762	0.1397	4.93	0.735	1746
30	0.0762	0.2159	4.02	0.717	3069
30	0.0762	0.3683	3.14	0.716	2027
30	0.0762	0.5207	2.35	0.713	2118
60	0.1524	0.7112	1.90	0.717	2130
60	0.1524	1.016	3.01	1.09	1618
62.5	0.159	1.70	1.01	0.735	2540
125	0.3175	1.94	0.99	0.731	3187

^aThickness of sample milled from spine.

Table 5.3-7. Radial distribution of beta-emitting radionuclides
in sleeve at compact 28
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
			^3H	^{14}C	^{90}Sr
5	0.0127	0.0064	3.16	0.082	2990.0
5	0.0127	0.0191	10.6	0.172	432.0
5	0.0127	0.0318	7.71	0.179	424.0
5	0.0127	0.0445	5.74	0.146	1.1
15	0.0381	0.0699	3.34	0.132	0.364
30	0.0762	0.127	2.41	0.107	0.023
30	0.0762	0.279	1.96	0.109	0.020
30	0.0762	0.432	2.05	0.107	0.017
30	0.0762	0.584	2.26	0.107	0.020
30	0.0762	0.737	3.22	0.111	0.018
15	0.0381	0.870	4.38	0.116	0.031
10	0.0254	0.940	7.74	0.566	11.9

^a Thickness of sample milled from sleeve.

Table 5.3-8. Radial distribution of beta-emitting radionuclides
in spine at compact 28
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from OD (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)		
			^3H	^{14}C	^{90}Sr
8	0.0203	0.0102	13.6	2.22	690.0
15	0.0381	0.0394	3.72	0.372	477.0
15	0.0381	0.0775	2.38	0.324	356.0
30	0.0762	0.135	2.27	0.335	363.0
30	0.0762	0.211	2.09	0.320	298.0
30	0.0762	0.363	2.03	0.359	273.0
30	0.0762	0.516	1.78	0.363	149.0
30	0.0762	0.668	1.58	0.347	112.0
28	0.0711	0.818	1.61	0.368	78.9
62.5	0.159	1.718	1.02	0.360	3.67
125	0.318	1.956	1.03	0.392	2.24

^aThickness of sample milled from the spine.

Table 5.3-9. Radial distribution of beta-emitting radionuclides
in bottom connector
(corrected to September 14, 1973)

Thickness ^a (mils)	(cm)	Distance to center of cut from ID (cm)	Radionuclide concentration ($\mu\text{Ci/g}$)	
			^3H	^{14}C
20	0.0508	0.025	4.51	0.047
40	0.1016	0.102	0.28	0.02
40	0.1016	0.203	0.24	0.02
60	0.1524	0.33	0.55	0.022
60	0.1524	0.94	0.29	0.02
60	0.1524	1.55	0.49	0.02
60	0.1524	2.16	0.22	0.02
60	0.1524	2.46	0.33	0.02
62.5	0.1588	3.55	0.26	0.02
62.5	0.1588	3.7	0.60	0.02

^aThickness of sample milled from sleeve.

Fig. 5.3-1. Radial distribution of beta-emitting radionuclides found at compact 18 in fuel-element E11-07 (corrected to September 14, 1973).

Fig. 5.3-2. Radial distribution of beta-emitting radionuclides found at compact 28 in fuel-element Ell-07 (corrected to September 14, 1973).

Fig. 5.3-3. Radial distribution of beta-emitting radionuclides found in E11-07 bottom connector (corrected to September 14, 1973).

Table 5.3-10. Axial distribution of beta-emitting radionuclides
in fission product trap
(corrected to September 14, 1973)

Weight of charcoal (g)	Radionuclide concentration ($\mu\text{Ci/g}$)		
	^3H	^{14}C	^{90}Sr
7.1587	333	46.3	30.5
17.3193	292	56.7	9.12
20.8035	250	39.5	9.37
20.0655	310	23.7	8.47
32.9581	243	12.1	7.97

the samples submitted contained sufficiently high levels of gamma emitters to necessitate running these procedures in a hot cell.

6. FUEL PARTICLE EXAMINATION

6.1 Metallographic Examination of Fuel Compacts*

A representative portion of the photomicrographs obtained from samples of fuel compacts 9, 16, 18, and 30 is presented in this section. The reader should keep in mind that any such collection of photographs would naturally overemphasize the failed particles or other nonnormal features, since it is desired to display the modes of failure. Although a number of typical particles are included in this series, one does not obtain a proper estimate of the failed fuel fraction by scanning through this series of photomicrographs. Determinations of the failed particle fraction by a hot-chlorine leaching procedure are presented in Sect. 6.2.

The fuel particles in all Peach Bottom HTGR driver elements were of the type designated BISO with kernels composed of mixed $(\text{Th}, \text{U})\text{C}_2$. Compacts 1 through 9 (numbered from the bottom of the core) and 27 through 30 were the so-called "standard compacts" with an initial Th:U ratio of 5.5 and no burnable poison. The central compacts, numbered 10 through 26, had equal amounts of Th and U, and additionally 0.342 g Rh per compact. More complete specifications of the fuel compacts are given in ref. 1 as well as in the summary presented in Sect. 2.2 of ref. 6. In addition, kernel migration rates observed in compacts 16 and 18 have been separately studied and are reported by Lindemer and Pearson.¹¹

The photomicrographs are presented in four sets, one for each of the four compacts, with each set beginning with a complete radial section. The operating conditions for these four compacts at end of life may be observed from Fig. 2.2-2 where it is seen that the two intermediate compacts sustained the highest fast and thermal fluence and highest temperature. Compact 9 at end of life was at the lowest temperature of the four, but compact 30 sustained the lowest fluence.

* These examinations were performed by the Remote Metallography Group of the Metals and Ceramics Division in the High-Radiation-Level Examination Laboratory.

While the fluence profiles presented in Fig. 2.2-2 may be considered to be reasonably accurate, the temperatures presented in this figure are highly preliminary. At present, ongoing work at GAC is endeavoring to more accurately specify Peach Bottom fuel service temperatures.

Figure 6.1-1 illustrates a complete radial section through compact 9 magnified 20 times. The clearly visible curvature denotes the inner diameter surface on the right and the outer diameter surface on the left of the field. Some noteworthy features displayed in this cross section are: (1) the seeming nonuniform distribution of fuel particles within the matrix; (2) the very rough inner surface appearance, almost as if it were abraded; (3) the appearance of cracks in the matrix (top third on the right and top left), circumventing fuel particles; and (4) missing fuel kernels undoubtedly removed in the polishing process. Evidently, the pyrocarbon coatings were more firmly held in the matrix than the kernels were held in the buffer.

Figure 6.1-2 illustrates a fuel particle in compact 9 located at the outer diameter of the compact. The right view employs normal light, while polarized light was used to obtain the darker photograph shown on the left. Polarized light views are used to bring out detail features within the pyrocarbon layer which may result from an improper fabrication technique. As seen in Fig. 6.1-2, no such features appear in this particle, and none appeared in any of the fuel particles viewed. In the normal light view, one clearly sees the recoil zone surrounding the kernel, which appears to be abnormally cracked. There is some small degree of migration of fuel material toward the buffer zone (toward the left in the figure).

Figure 6.1-3 illustrates two more-or-less typical particles located in the middle zone (right) and on the inner diameter (left) of compact 9. The right view shows small penetrations of fuel into the buffer in an approximately isotropic or random fashion. Knowledgeable specialists^{12,13} have designated this behavior as intermittent plastic flow of the kernel through the recoil zone.

Figure 6.1-4 illustrates a doublet particle (a manufacturing fault) which failed through the buffer and pyrocarbon coating at several locations

RADIAL SECTION THROUGH FUEL COMPACT No.9, PEACH BOTTOM DRIVER ELEMENT E11-07 (701 EFPD)

Fig. 6.1-1. Radial section through compact 9. Magnification 20X.

Fig. 6.1-2. Fuel particle in compact 9 located at outer diameter.

Fig. 6.1-3. Two fuel particles in compact 9 at midplane (right)
and on inner diameter (left).

Fig. 6.1-4. Doublet fuel particle in compact 9.

because of excess internal pressure or fast-neutron-induced densification of the pyrocarbon layer.^{12,13} The smooth-appearing material which fills some of the crack area is epoxy resin used to mount the specimen.

Figure 6.1-5 illustrates the effect of densification of the matrix (magnifications of 200X and 500X), which then separates from the particles, indicating that the particle strength exceeds the matrix-to-particle bond. Again, the smooth, gray material in the crack is the resin employed in specimen preparation.

A set of photomicrographs representing fuel compact 16 is presented in Figs. 6.1-6 through 6.1-13. The first is a composite showing an entire radial segment at about 20X magnification. Some noteworthy features shown in this composite are the following: (1) note again a crack progression which appears to skirt fuel particles through the matrix at the upper right; (2) contrary to the appearance of the previously shown radial segment (compact 9, Fig. 6.1-1), the appearance of the inner diameter is quite smooth, which leads one to think that perhaps the inner diameter of compact 9 was machined; (3) again contrary to the appearance of compact 9, whole particles rather than kernels have been lost in the specimen preparation process, indicating the densification of the matrix around those particles and the looser bond between pyrocarbon and matrix for compact 16; (4) also the appearance of the fuel kernels is darker. Evidently the hotter kernels react more readily with air or moisture during the specimen preparation process than those experiencing less severe conditions.

Figure 6.1-7 shows two particles in compact 16, one at the extreme outer diameter (right) and the other located very near the inner boundary (left). Note the darker appearance of the kernel in the inner-boundary particle, consistent with the observation cited above that the hotter kernels darken to a greater degree during sample preparation. It is also interesting to note the differing degrees of kernel migration between these two particles. The outer particle is subject to the largest temperature gradient within the compact, and it shows a migration distance of about one-half a buffer-layer thickness in the direction of increasing temperature. On the other hand, the interior particle (left) would have been subject to a far smaller thermal gradient and indeed shows far less kernel migration.

Fig. 6.1-5. Two views of a crack progression through the matrix of compact 9.

RADIAL SECTION THROUGH FUEL COMPACT No.16, PEACH BOTTOM DRIVER ELEMENT E11-07 (701 EFPD)

Fig. 6.1-6. Radial section through compact 16. Magnification 20X.

Fig. 6.1-7. Fuel particles in compact 16. Right view near outside diameter; left view near inner diameter.

Figure 6.1-8 shows two views of a particle at mid-radius with kernel migration part way through the buffer at a number of locations. The light-appearing boundary of the kernel viewed with polarized light indicates a zone of graphite material surrounding the kernel.

Figure 6.1-9 shows two similar types of particles in compact 16; that is, the kernel has migrated into the buffer in a number of locations, and there appears to be some graphitized material surrounding the kernel.

Figure 6.1-10 illustrates a matrix crack in compact 16 at 100X magnification.

Figure 6.1-11 shows two types of particle fabrication defects. The particle at right contains one complete buffer, one partial buffer, and a partial pyrocarbon coat. The view at bottom left shows a particle containing matrix carbon in place of a fuel kernel.

Figure 6.1-12 illustrates a fuel particle with distinctive buffer-pyrocarbon separation. An approximately isotropic movement of kernel material into the buffer zone is shown in Fig. 6.1-13. The view at right is a highly magnified view of a leading edge of kernel material penetrating the buffer.

The radial segment of compact 18 in Fig. 6.1-14 shows similar features to those seen for compact 16. There is nonuniform dispersion of fuel particles; some matrix cracking is evident. Figure 6.1-15 illustrates two particles on the outer diameter with some kernel migration, as anticipated, in the direction of increasing temperatures. Figure 6.1-16 shows two particles from the mid-radius location; the view at right indicates almost complete kernel penetration of the buffer. In Fig. 6.1-17 are shown two more-or-less typical fuel particles from compact 18.

Two types of manufacturing faults are illustrated in Fig. 6.1-18. The particle at right possesses an irregular kernel, perhaps because of an "amoeba effect," whereas the left view shows a particle devoid of a buffer layer and which has, as a consequence, cracked because of internal pressure. The matrix graphite of compact 18 is shown in Fig. 6.1-19, viewed at 250X and 750X magnification.

Fig. 6.1-8. Natural and polarized light (left) views of a particle at mid-radius in compact 16.

Fig. 6.1-9. Two particles in compact 16 showing some degree of kernel migration.

Fig. 6.1-10. View of a matrix crack in compact 16.

Fig. 6.1-11. Two types of manufacture faults appearing in compact 16.

Fig. 6.1-12. Particle in compact 16 showing buffer-pyrocarbon separation.

50 100 MICRONS 200X 300 350
0.005 INCHES 0.010 0.015

20 40 60 MICRONS 500X 120 140
0.001 INCHES 0.005

Fig. 6.1-13. Particle in compact 16 showing isotropic kernel migration into the buffer layer.

RADIAL SECTION THROUGH FUEL COMPACT No.18, PEACH BOTTOM DRIVER ELEMENT E11-07 (701 EFPD)

Fig. 6.1-14. Radial segment through compact 18. Magnification 20X.

R-65016

R-65017

Fig. 6.1-15. Two particles at the outer diameter of compact 18.

R-65063

50 100 MICRONS 200X 300 350
0.005 INCHES 0.010 0.015

R-65018

50 100 MICRONS 200X 300 350
0.005 INCHES 0.010 0.015

Fig. 6.1-16. Two particles from mid-radial region of compact 18.
Magnification 20X.

Fig. 6.1-17. Two typical fuel particles from compact 18.

Fig. 6.1-18. Two types of fabrication faults seen in compact 18.

Fig. 6.1-19. Matrix graphite of compact 18.

The final set of photomicrographs shows views of compact 30. The first of this sequence, Fig. 6.1-20, is a complete radial segment illustrating a number of interesting features. Note the deposited layer at the inner diameter. Examination under higher magnification (Fig. 6.1-21) reveals this as a pyrocarbon layer deposited from the purge flow. A second feature of interest is the more uniform dispersion of fuel particles than previously seen for compacts 9, 16, and 18. No matrix cracks are seen, and the fuel kernels are as light-appearing as those seen in compact 9.

A 500X magnified view of the inner diameter layered deposit is shown in Fig. 6.1-21 under natural (left) and polarized (right) light. Several layerings (at least four) are clearly visible at left, while the polarized light view shows distinctive columnar growth patterns typical for some types of pyrocarbon deposits. Evidently this coating was produced as a result of occasional hydrocarbon contamination of the coolant stream during core 2 operation. The purge flow, bled from the coolant flow and hence compositionally similar, enters the top of the element through the purge hole, reaching compact 30 before the other compacts. Hence it appears that the upper moderator region and upper compact zone served to clear the purge flow of hydrocarbon contaminants by deposition as a pyrocarbon layer. No deposit is clearly seen on the outer compact surface because only about 1% of the total purge flows between the outside diameter and the sleeve.

Figure 6.1-22 illustrates two typical fuel particles in compact 30 located at the outer (right) and inner diameter (left). The particle at left shows some faceting of the outer pyrocarbon layer. Neither particle visibly exhibits kernel migration, indicative of the small temperature gradient experienced at this core location. Figure 6.1-23 illustrates a fuel particle exhibiting neutron-induced densification of the buffer and pyrolytic graphite coating. The densified portion is shown at 500X magnification at left. The pyrocarbon coat is also cracked, possibly due to excess internal pressure.

Figure 6.1-24 to 6.1-26 illustrate six particles in compact 30 exhibiting cracked coatings due to manufacturing defects causing excessive internal pressure, or radiation-induced stress. The particle at left in Fig. 6.1-26

RADIAL SECTION THROUGH FUEL COMPACT No.30, PEACH BOTTOM DRIVER ELEMENT E11-07 (701 EFPD)

Fig. 6.1-20. Radial segment through compact 30. Magnification 20X.

Fig. 6.1-21. Pyrocarbon deposit on inside diameter of compact 30 viewed with normal (left) and polarized light (right).

50 100 MICRONS 200X 300 350
0.005 INCHES 0.010 0.015

50 100 MICRONS 200X 300 350
0.005 INCHES 0.010 0.015

Fig. 6.1-22. Particles from compact 30 at the compact outer diameter (right) and inner diameter (left).

20 40 60 MICRONS 120 140
0.001 500X INCHES 0.005

50 100 MICRONS 300 350
0.005 200X INCHES 0.010 0.015

Fig. 6.1-23. Particle in compact 30 showing pyrocarbon and buffer densification and pyrocarbon cracking.

Fig. 6.1-24. Two cracked particles in compact 30.

Fig. 6.1-25. Two faulted particles in compact 30.

Fig. 6.1-26. Two particles in compact 30 exhibiting cracked coatings.

exhibits a classic excess internal pressure crack. The adjacent particle shows no kernel because the metallographic plane passed through the buffer zone.

6.2 Failed Particle Fraction Measurements*

A method was developed[†] for determining the failed particle fraction (FPF) in samples of irradiated Peach Bottom fuel compacts, consisting of the following steps:

1. An approximately 1-cm-square sample containing about 10^4 fuel particles is cut from the irradiated compact, using a remotely operated saw in a hot cell.
2. The cut surfaces are cleaned (i.e., removed of debris formed by the cutting operation) by localized deconsolidation in dilute nitric acid at low temperatures. This is accomplished by placing an anode near the surface to be cleaned and allowing current to flow directly to the cut surface.
3. The sample is then leached with chlorine gas at 1000°C. This process removes from the sample, as a mixture of various gaseous uranium chlorides and as thorium tetrachloride, all heavy metal not protected by an impervious pyrocarbon coating.
4. The gaseous uranium and thorium chlorides are collected in a cold trap and quantitatively determined by standard techniques.
5. The sample is then fully deconsolidated, and the total mass of the fuel particles is determined. This yields the number of particles in the sample, using a known average mass per particle.

* Results presented in this section are tentative. Fuller discussion of failed particle fraction determinations in Peach Bottom fuel compacts will be reported later.

[†] Work performed by J. A. Botts, Analytical Chemistry Division.

6. The FPF is then determined using data from steps 4 and 5 and the known average inventory of uranium and thorium per particle.

6.2.1 Test of method on archive compact. Determination of FPF and "tramp fuel" level in archive compact

As an initial step, the particles in the archive compact were characterized by determining an average kernel diameter, buffer, and pyrocarbon layer thickness. This was done by radiographing 200 particles; results are given in Table 6.2-1. In addition to the average dimensions given in the table, the mean and the standard deviation of the mass of 200 fuel particles were $198 \pm 2.2 \mu\text{g}$.

Table 6.2-1. Average dimensions of Peach Bottom fuel particles

Group	No. of particles	Mean and standard deviation (μm)		
		Fuel kernel diameter	Buffer layer thickness	Pyrocarbon layer thickness
1	50	272.4 ± 27.6	46.9 ± 9.2	55.5 ± 4.9
2	50	283.6 ± 33.3	43.4 ± 5.9	55.5 ± 5.3
3	50	283.3 ± 27.2	45.7 ± 7.6	55.7 ± 4.2
4	50	<u>277.4 ± 30.4</u>	<u> </u>	<u> </u>
		279.2 ± 29.7	45.3 ± 7.7	55.6 ± 4.8

Using the given average kernel diameter of $279.2 \mu\text{m}$, the kernel mass was calculated to be $106.2 \mu\text{g}$. This value was determined assuming 8.96 and 11.28 g/cm^3 to be the densities of ThC_2 and UC_2 , respectively, which yields a kernel density of 9.316 g/cm^3 for the Th/U ratio of 5.5/1 existing in the archive compact. An average mass of $81.45 \mu\text{g}$ of Th and $14.81 \mu\text{g}$ of U per kernel is thereby obtained.

Results for four tests on samples drawn from an archive fuel compact are summarized in Table 6.2-2. The sample masses listed in the first column record values obtained after electrolytic cleansing to remove surface debris caused by the cutting operation to acquire the sample. The masses of

Table 6.2-2. Failed particle fractions in archive Peach Bottom fuel by hot-chlorine leach

Sample mass (g)	Chlorine leach				Th/U	Particle mass ^a (g)	No. of particles ^b	No. of failed particles, ^c based on		% failed particles ^d	
	Th leached (mg)	U leached (μg)	Leach time (hr)	Leach temperature (°C)				Th	U	Th	U
8.6586	3.47	635	46	1000	5.46	2.8585	14,436	42.6	42.9	0.295	0.297
8.4610	3.31	581	46	1000	5.70	2.8573	14,430	40.6	39.2	0.281	0.272
9.9006	3.25	610	6	1500	5.25	3.3384	16,860	39.9	41.8	0.237	0.248
9.3094	4.36	800	6	1500	5.45	3.4566	17,458	53.5	54.0	0.309	0.309
					av = 5.47						
										av of eight = 0.281 ± 0.022	

^aDetermined after leaching and deconsolidation.

^bBased on 198 μg per particle.

^cBased on 81.45 μg Th and 14.81 μg U per particle.

^dIncludes "tramp fuel."

thorium and uranium removed in the leaching process, the mass of the particles determined after leaching and deconsolidation, and the calculated number of particles are given in the table for each sample. The columns at the right record the calculated number of failed particles based on the amount of thorium and uranium removed in the leach process, and the calculated failed particle fraction.

One should note that the Th/U ratio leached from the fuel equaled closely the nominal reported value of 5.5, which lends some confidence to the method. As a direct result, the calculated fractions based on thorium and uranium removed agree closely. Also, the two leach temperatures of 1000 and 1500°C yielded essentially identical results. Therefore, the system installed in a hot cell for the irradiated-compact determinations, which can only achieve 1000°C, should nevertheless produce valid results.

The results in Table 6.2-2 indicate that the archive compact had an average of $0.281 \pm 0.022\%$ failed particle.

Subsequent tests on this archive compact were performed to determine approximately what portion of the heavy metal leached originated from matrix contamination instead of failed fuel particles. These tests showed that a maximum of 64% of the thorium and uranium leached from the archive compact came from matrix contamination. In other words, the actual percent of failed particles in the archive specimen was 0.10%, with matrix contamination yielding a mass of heavy metal equivalent to 0.18% failed particles.

6.2.2 Failed particle fraction in Ell-07 fuel compacts

Determinations of the failed particle fraction in irradiated compacts 2, 13, 16, and 28 are summarized in Table 6.2-3. The method used was the same as for the archive compacts except that the procedures were carried out in a hot cell. The average mass of thorium and uranium in fuel particles from each of these four compacts was calculated from the axial average burnups given in Table 2.1-1 and the axial form factors given in ref. 10. An initial loading of 81.45 μg Th per particle and 14.81 μg U per particle was assumed, as determined using the archive compact. As anticipated,

Table 6.2-3. Failed fuel particle determination
by hot chlorine leach

	Compact number			
	2	13	16	28
Sample mass (g)				
Initial	12.6087	9.0386	14.8927	6.3881
After electrolytic cleansing	11.0634	5.5520	7.6122	5.1636
Calculated heavy metal ^a (μg/particle)				
Th	79.3	77.7	77.7	79.7
U	11.6	10.3	10.3	11.8
Th/U	6.86	7.55	7.51	6.73
Leached heavy metal (mg)				
Th	5.510	1.930	1.734	1.927
U	0.721	0.400	0.181	0.523
Th/U	7.64	4.83	9.58	3.68
Mass of particles (g)	2.0232	1.9642	2.0413	1.4523
No. of particles ^b	10,218	9920	10,309	7334
No. of failed particles				
Based on Th	69.5	24.8	22.3	24.2
Based on U	62.2	38.8	17.6	44.3
Percent failed particles				
Based on Th	0.68	0.25	0.21	0.33
Based on U	0.61	0.39	0.17	0.60

^aAssuming 81.45 and 14.81 μg/particle Th and U initially, and calculating depletion using Table 2.1-1 and form factors given in ref. 10.

^bBased on 198 μg/particle.

reactor service elevates the Th/U ratio at each location from an initial value of 5.5 to the given range, 6.73 to 7.55. The table shows that the Th/U ratios determined by leaching depart significantly from the anticipated values. The reason for this variance is not known at this time. As noted in the previous section, Th/U ratios determined by hot-chlorine leach of the archive compact were quite close to the anticipated pre-irradiation value of 5.5.

Despite the uncertainty generated by the deviation of the Th/U ratio from the expected value, the results indicate that failed particle rates in element Ell-07 are quite low. In five of the eight determinations, values comparable to that determined for the archive were measured, and three of the eight determinations were about a factor of 2 above the archive value of 0.28%.

7. REFERENCES

1. Peach Bottom Atomic Power Station, Final Hazards Summary Report, Part C, vol. 2, NP-9115 (March 1964).
2. Proposed Facility Change and Technical Specifications Change No. 13 to the Final Hazards Summary Report (January 1970).
3. R. F. Turner et al., HTGR Fuel Performance in the Peach Bottom Reactor, GA-A 12675 (July 1973).
4. Internal correspondence, GAC, Frances McCord to C. F. Wallroth, "Peach Bottom Core 2 Driver Element Ell-07 Irradiation Summary," June 25, 1975.
5. Keith Assmussen (GAC), personal communication, Sept. 16, 1976.
6. F. F. Dyer, R. P. Wichner et al., Post-Irradiation Examination of Peach Bottom HTGR Driver Fuel Element EO6-01, ORNL-5126 (March 1976).
7. F. F. Dyer, R. P. Wichner et al., Distribution of Radionuclides in the Peach Bottom HTGR Primary Circuit During Core 2 Operation, ORNL-5188 (March 1977).

8. Letter, C. M. Miller (GAC) to W. J. Martin (ORNL), Jan. 22, 1974, Attachment 3.
9. G. Buzzelli and R. H. Smith, Spectrophotometric Determination of Fission Zirconium in Nuclear Fuels, GAMD-10072 (February 1973).
10. C. F. Wallroth et al., Postirradiation Examination and Evaluation of Peach Bottom Molded Test Element FTE-18, GA-A13699 (June 1976).
11. T. B. Lindemer and R. L. Pearson, Kernel Migration in HTGR Fuels from the Th, U, C, O, N System, ORNL/TM-5207 (April 1976).
12. E. L. Long, Metals and Ceramics Division, ORNL, personal communication, March 1976.
13. M. J. Kania, Metals and Ceramics Division, ORNL, personal communication, November 1976.

INTERNAL DISTRIBUTION

1. J. L. Anderson	61. R. A. Lorenz
2. S. J. Ball	62. A. L. Lotts
3. D. E. Bartine	63. W. S. Lyon
4. L. C. Bate	64. R. E. MacPherson
5. C. D. Baumann	65. J. C. Mailen
6. R. L. Beatty	66. A. P. Malinauskas
7. J. P. Callahan	67. W. J. Martin
8. D. O. Campbell	68. W. R. Martin
9. D. A. Canonico	69. W. J. McDowell
10. C. E. Clifford	70. K. J. Notz
11. J. A. Conlin	71. L. C. Oaks
12. J. H. Coobs	72. A. R. Olsen
13. D. A. Costanzo	73. G. W. Parker
14. F. L. Culler, Jr.	74. W. H. Pechin
15. F. C. Davis	75. H. Postma
16. J. R. DiStefano	76. M. F. Pruitt
17. B. C. Duggins	77. R. H. Rainey
18-27. F. F. Dyer	78. D. P. Reid
28. W. P. Eatherly	79. R. R. Rickard
29. J. F. Emery	80. A. D. Ryon
30. D. E. Ferguson	81. J. P. Sanders
31. Uri Gat	82. J. D. Sease
32. J. H. Goode	83. J. H. Shaffer
33. A. G. Grindell	84. R. L. Shepard
34. W. S. Groenier	85. W. D. Shults
35. P. A. Haas	86. J. W. Snider
36. W. R. Hamel	87. P. R. Stein
37. F. E. Harrington	88. R. S. Stone
38. C. C. Haws	89. M. L. Tobias
39. L. C. Hensley	90. D. B. Trauger
40. R. F. Hibbs	91. J. E. Van Cleve
41. R. M. Hill, Jr.	92. V. C. A. Vaughan
42. F. J. Homan	93-102. R. P. Wichner
43. J. R. Horton	103. R. G. Wymer
44. L. D. Hulett	104-105. Central Research Library
45. J. D. Jenkins	106. Document Reference Section
46. D. R. Johnson	107. Laboratory Records, ORNL-RC
47-56. P. R. Kasten	108-109. Laboratory Records
57. W. J. Lackey	110. ORNL Patent Section
58. K. H. Lin	111. E. L. Gaden, Jr. (consultant)
59. T. B. Lindemer	112. C. H. Ice (consultant)
60. E. L. Long, Jr.	113. R. B. Richards (consultant)

EXTERNAL DISTRIBUTION

- 114. Research and Technical Support Division, ERDA-ORO, P. O. Box E, Oak Ridge, Tn. 37830
- 115. Director, Reactor Division, ERDA-ORO, P. O. Box E, Oak Ridge, Tn. 37830
- 116-117. Director, Division of Nuclear Fuel Cycle and Production, ERDA, Washington, D.C. 20545
- 118-119. Director, Division of Reactor Research and Development, ERDA, Washington, D.C. 20545
- 120-296. Given distribution as shown in TID-4500 under category UC-77 - Gas-Cooled Reactor Technology