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POSTIRRADIATION EXAMINATION OF PEACH BOTTOM FUEL TEST ELEMENT FTE-4

MASTER

by
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ABSTRACT

This report presents the irradiation results and their evaluation for Peach Bottom fuel test element FTE-4. It describes in detail the efforts by General Atomic Company over the last two years to establish a system for extracting meaningful performance information from a fuel test element. This has been done with the goal of making direct comparisons between as-measured data and core design code predictions. Special emphasis has been placed on determining the 95% confidence limits on most of the preirradiation and postirradiation measurements in order to allow a better comparison with GAUGE, FEVER, and TREVER code calculations which are used in HTGR core thermal and mechanical design.

The new or updated methods employed in this irradiation experiment are:

1. Statistical analysis of metrology data to determine the structural performance of graphite and fuel rod materials as well as to prove the hypothesis of isotropic irradiation deformation of fuel rods.
2. Fuel performance analysis by the use of fission gas release measurements, metallography, and gamma spectroscopy.
3. Multi-isotopic gamma spectroscopy to establish simple means for nondestructive burnup estimations as well as fission product distributions. Radial cesium migration profiles have been determined in special spine samples and also for the first time in the structural graphite bodies containing the test fuel.
4. Recalibrations of irradiated thermocouples out of a common reactor environment, which were employed for the first time to improve the accuracy of the thermocouple measurements.

The collected data were compared with design code predictions. Comparison with the GAUGE and FEVER design codes showed that fissile nuclide concentrations are accurately predicted within $\pm 2\%$, but non-fissile isotopes could be off by $\pm 20\%$. Radiochemical burnup measurements as well as axial burnup calculations need to be improved.

A major effort had been made to apply the large HTGR design code TREVER to the Peach Bottom test element configuration for predicting temperatures correctly and establishing means of direct comparison of structural design information with measurements. This has been done for the comparison of the strains and interfaces of the sleeve, fuel body, and fuel rod.

This examination proved favorable in that the absolute and integrated effects are concurrent with calculations based on the median values of the applied material properties. However, detail effects caused by unfueled zones or at core boundaries are presently not predictable for the structural graphite. The hypothesis of isotropic fuel strain was accepted within the uncertainty of the measurements for three of the four fuel blend varieties tested. Fuel strain can be predicted within $\pm 30\%$ of the experienced value.

A stress calculation was carried out, which revealed in-plane shutdown stresses with a maximum stress/strength ratio of 1.42 at EOL shutdown. Stress concentrations were very localized. No structural damage was found during destructive graphite examination. This may be explained by conservatism in the uniaxial strength criteria or the calculational methods.

An area of needed improvement is the reduction of the uncertainties for the preirradiation and postirradiation measurements. For the preirradiation phase, this is very difficult to achieve. Grouping of several elements will possibly reduce the uncertainty. For the postirradiation phase, better sampling would have a positive impact. Sensitivity analyses are required for the design prediction in order to allow for material property variances, which may explain deviations from the expected performance.

In summary, fuel test element FTE-4 has proven to be a very useful test of the HTGR concept and its applied design techniques. No area of critical performance defect or performance mismatch has been found. With further application of the presented test scheme and procedure to other Peach Bottom test and driver elements, it is believed that the design code verification goal can be met with a high degree of confidence. This will allow improved HTGR performance predictions, so that design conservatism can be relaxed.

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1. INTRODUCTION

Fuel test element FTE-4 is one of a group of 33 test elements irradiated in the Peach Bottom Unit 1 High-Temperature Gas-Cooled Reactor (HTGR). FTE-4 was part of a test program which had been designed to provide information on the behavior of HTGR fuels subjected to representative irradiation conditions (Ref. 1). For this purpose, Peach Bottom Unit 1 was a unique facility with regard to the HTGR neutron spectrum, pressurized helium environment, and temperature regime. The main test objectives were:

1. Performance evaluation of various types of HTGR fuels over a wide range of operating conditions and lifetimes.
2. Design information about the performance of the compound system fuel rod - structural graphite (or alternative concepts) under realistic reactor conditions from statistically representative samples.
3. Controlled irradiation experiments on fuel particle thermal stability, metallic fission product diffusion rates, and dimensional, structural, and chemical stability of fuel rods, graphite, and absorber materials.
4. HTGR design code validation with respect to:
 - a. Fluence and depletion.
 - b. Thermal performance.
 - c. Thermal and Wigner strain.
 - d. Fission product behavior predictions.

This report summarizes the irradiation history and the postirradiation examination and evaluation of FTE-4. Errors were progressed so that 95%

confidence limits were estimated for derived quantities (such as strain) as well as for the actual measurements, which then were compared with HTGR design code predictions.

2. EXPERIMENT DESCRIPTION

FTE-4 is one of 33 test elements which had been inserted into Core 2 of the Peach Bottom HTGR. It is the second of four nearly identical experiments analyzed, with FTE-3 already completed (Ref. 2) and FTE-5 and FTE-6 still to come. A layout of the general assembly is shown in Fig. 1.* The exterior design and appearance resembled a standard driver fuel element utilizing such components as top reflector, sleeve, and bottom connector, which also formed the secondary containment for the test element. These parts were made from H-381, an extruded medium-grain petroleum coke graphite with multiple impregnation and low permeability, manufactured by Great Lakes Carbon Corporation (GLCC). (See also Table 1.)

The active portion of the test element was made of three cylindrical H-327 graphite bodies of 31-in. length, which replaced the 30-fuel-compact arrangement in a standard Core 2 element. H-327 is a coarse-grain, impregnated needle coke graphite manufactured by GLCC and was also used for the initial core of the Fort St. Vrain HTGR. The experimental fuel rods had been loaded in an eight-hole "telephone dial" arrangement and closed with threaded graphite plugs to provide complete primary containment to the fuel as shown in Fig. 2. A cross section is shown in Fig. 3 together with specified dimensions.

FTE-4 had been instrumented with two thermocouples: one Chromel-Alumel (C/A) type for graphite body temperature measurements and one tungsten-rhenium (W/Re) type for fuel temperature measurements at 49.5 in. (1257.3 mm) active core height.[†] Axial and radial locations are indicated in Figs. 1 and 3, respectively.

*Figures and tables appear at the end of each section.

[†]Active core height commences 26.00 in. (660.4 mm) above the core bed (= start of total core height) and goes to 90 in. (2286 mm), which is 116 in. (2946.4 mm) above the core bed plate (see Fig. 1) The terms active core height and total core height are used interchangeably in this report.

FTE-4 contained 336 fuel rods that were carbonized in Al_2O_3 beds. These rods consist of a closely packed bed of blended fissile and fertile coated fuel particles bonded together in a carbonaceous matrix. The matrix is a blend of 27% to 30% natural flake graphite flour and 70% to 73% coal tar pitch. The matrix was prepared by hot-mixing the ingredients to form a homogeneous blend, cooling to room temperature, and grinding the solid matrix into granules suitable for use in the fuel rod injection equipment. The impurities in the matrix were kept in the low ppm range.

Both the fuel particles and fuel rods were made in production equipment using available production processes and quality control techniques. The hot matrix was injected into the particle bed with a prototype injection machine similar to those built for manufacturing Fort St. Vrain initial core fuel. After cooling, the fuel rods were removed from the injection die; at this point, the rods were referred to as "green" rods. The green rods were then packed into Al_2O_3 beds and subjected to a carbonization treatment.

The carbonization cycle consisted of heating the fuel rods in a flowing nitrogen atmosphere for 2 hr to 750°C and holding at that temperature for 0.5 hr. The fuel rods were then heat treated by passing them through a furnace, which had a 72-in. (1829 mm) long hot zone, at 1800°C and at a rate of 1 in. (25.4 mm) per minute. The atmosphere in the furnace was argon.

The cured rods were measured for length and diameter and loaded into the three graphite fuel bodies.

The eight "teledial" holes contained four different kinds of fuel rods:

Holes 1 and 2	UO_2 TRISO, ThO_2 BISO
Holes 3 and 4	$(2.75 \text{ Th:U})\text{C}_2$ TRISO, ThC_2 BISO
Holes 5 and 6	$(2.75 \text{ Th:U})\text{C}_2$ TRISO, ThC_2 TRISO
Holes 7 and 8	UC_2 TRISO, ThC_2 BISO

FTE-4 contained a total of 43 spine samples of the following types:

- 22 metallic diffusion samples
- 6 fission product release samples
- 3 stress relaxation samples
- 1 mechanical test assembly
- 2 advanced fuel rods
- 7 thermal stability samples
- 1 boronated stainless steel sample
- 1 absorber rod sample

FTE-4 was assembled at General Atomic Company and shipped to Peach Bottom, where it commenced irradiation on July 11, 1971.

TABLE 1
GRAPHITE MATERIALS USED IN PEACH BOTTOM FUEL ELEMENTS

Fuel Element Component	Core 1 Driver Graphite Type		Core 2 Driver Graphite Type		FTE-3, 4, 5, 6 Graphite Type	
	Base Stock	Manufacturer	Base Stock	Manufacturer	Base Stock	Manufacturer
Top reflector	H-253 ^(a)	GLCC	H-381 ^(b)	GLCC	H-381 ^(b)	GLCC
Top spine	711-GSXY	SPEER	711-T	SPEER	H-327	GLCC
Lower two spines (with T/C slots for instru- mented drivers)	711-GSXY	SPEER	711-GSXY	SPEER	H-327	GLCC
Fuel compact or rod filler binder	GP-38 flour Barrett No. 30	UCC Allied Chemical	GP-38 flour Barrett No. 30	UCC Allied Chemical	6353 15V	Ashbury Allied Chemical
Graphite fuel body	N/A		N/A		H-327	GLCC
Sleeve	H-253 ^(a)	GLCC	H-382 ^(b)	GLCC	H-382 ^(b)	GLCC
Internal trap	806 RL	SPEER	580	SPEER	580	SPEER
Lower reflector	806 RL	SPEER	580	SPEER	H-327	GLCC
Bottom connector	H-253 ^(a)	GLCC	H-381	GLCC	H-253 ^(a)	GLCC

(a) Also referred to as HLM-85-10; made with Texas Lockport coke.

(b) Also referred to as HLM-85-10; made with "Y" coke. Discrimination into H-381 and H-382 lacks explanation. It should be noted that most of the Core 1 and all Core 2 HLM-85-10 material was graphitized in a standard Acheson type furnace for about 28 days with temperatures up to 2800°C. However, some of the Core 1 material went through a rapid graphitization process in a tube furnace for about 1 hr at 2800°C (which also has been used for impregnation purposes of Core 1 and 2 sleeves after machining). About 40 elements in Core 2 have been made out of leftover Core 1 rapid graphitized material. During the Core 2 unloading exercise, it was discovered that this material happened to shrink more than the Core 2 standard material.

Fig. 1. Typical fuel test element assembly

Fig. 2. Graphite fuel body for Peach Bottom fuel test elements

ALL DIMENSIONS
IN MILLIMETERS

Fig. 3. Peach Bottom fuel test elements, eight-hole "teledial" cross section

3. IRRADIATION CONDITIONS

3.1. IRRADIATION HISTORY

Fuel test element FTE-4 was irradiated for 448.8 EFPD* in Peach Bottom Core 2 position A 14-08 from 252.4 EFPD through 701.2 EFPD. Beginning of life (BOL) was July 11, 1971, and end of life (EOL) occurred on September 14, 1973. Peach Bottom Core 2 operating parameters are given in Fig. 4 for the TREVER simulation. The reactor power history is shown in Figs. 5 and 6, which present thermal balance and TREVER (Ref. 3) simulation data, respectively. Figure 7 presents the power history for FTE-4. Table 2 summarizes the power, flow, and flux history used for nuclear and thermal modeling of the irradiation history of FTE-4. These data have been derived from GAUGE code calculations (Ref. 4). The average power factor was 0.91 and the total accumulated thermal energy was 56.9 MWd for the test element. A normalized axial power profile is shown in Fig. 8, as applied to the radial power, calculated by the GAUGE code, and used for the thermal analysis with the TREVER code.

The time- and element-averaged values[†] are as follows [for a definition of the root mean square (RMS) deviation, see Section 3.2]:

*1 EFPD is defined as 1 equivalent full-power day of 115-MW(t) operation of the reactor, which contained 804 elements.

[†]Time average via real time at operation.

	Mean	RMS
Irradiation time, equivalent (EFPD)	448.8	--
Irradiation time, real	535.8	--
Total elapsed time of irradiation(a)	796.0	--
Element power (kW)	106.2	±5.0
Element power factor(b)	0.911	±0.042
Coolant power factor(b)	1.018	±0.046
Coolant flow(c) (kg/s)	0.0316	±0.012
Flux (10^{17} n/m ² /s)		
Fast ($E > 28.8$ fJ)	3.31	±0.17
Fast ($13.8 \text{ fJ} \leq E \leq 28.8 \text{ fJ}$)	0.58	±0.03
Epithermal ($2.8 \text{ aJ} \leq E \leq 13.8 \text{ fJ}$)	6.53	±0.35
Epithermal ($0.38 \text{ aJ} \leq E \leq 2.8 \text{ fJ}$)	1.06	±0.06
Thermal ($0 \leq E \leq 0.28 \text{ fJ}$)	4.63	±0.48

(a) Total time in calendar days between BOL and EOL consisting of real time at operation and shutdown intervals.

(b) Relative to core average.

(c) For one tricusp channel.

3.2. FLUENCE

The fast and thermal fluences have been calculated with the two-dimensional four-group neutron diffusion depletion program GAUGE (Ref. 4) for the specific radial core location and have been axially distributed by means of the FEVER code (Ref. 5) applying the fast flux distribution in Fig. 9a and the thermal flux distribution in Fig. 9b.

The calculated fluence values for FTE-4 are:

EOL Fluences	Peak	Mean	RMS	Mean/Gradient Fuel Body Sequence(a)
Fast fluence [10^{25} n/m ² ($E > 29$ fJ)]	1.98	1.53	±0.42	213/231
Thermal fluence [10^{25} n/m ² ($E < 38$ aJ)]	2.86	2.14	±0.62	213/231

(a) Example: 213/231 = body 2 - body 1 - body 3 highest to lowest ranking for average value along body sequences 2, 1, 3 and lowest to highest ranking for gradient or RMS deviation along body sequence 2, 3, and 1.

Detailed fluence information for each of the 42 axial fuel rod locations is given in Table 3. All values are calculated and have not been verified by measurements except for a qualitative comparison with fuel gamma-scan data, reported in Section 4.6.7. Nuclear calculations are based on the latest GAUGE power information, incorporating Peach Bottom EOL fuel gamma-scan results (Ref. 6).

The root mean square deviation (RMS) quoted in Table 3 and above is derived as follows:

$$\text{RMS} = \left[\frac{1}{n} \sum (x_i - \bar{x})^2 \right]^{1/2}, \quad (1)$$

where n = number of calculations, combined in one strata,

x_i = calculation within one strata,

$\bar{x}$ = strata mean.

A weighted RMS variation can be determined from

$$\text{RMS} = \left[\frac{1}{\sum n_i} \sum n_i (x_i - \bar{x})^2 \right]^{1/2}, \quad (2)$$

where n_i = number (or length interval) of observations in stratum i .

RMS terms within an interval, RMS_j , and along all considered intervals, RMS , can also be postulated. The average of these RMS terms are the first and second terms, respectively, in the following equation:

$$\text{RMS} = \left[\frac{1}{\sum n_j} \sum n_j \text{RMS}_j^2 + \frac{1}{\sum n_j} \sum n_j (\bar{x}_j - \bar{x})^2 \right]^{1/2} = \left[\overline{\text{RMS}_j^2} + \overline{\text{RMS}^2} \right]^{1/2}. \quad (3)$$

The number of calculations n is arbitrarily selected. In the presented case, n was determined by the number of fuel rods in an axial arrangement per body. The fluence data, in turn, had been determined by linear interpolation between the 30 nodal points, used in the TREVER calculations.

3.3. TEMPERATURE

3.3.1. Temperature Calculations

Figures 10a through 10h show the TREVER (Ref. 3) calculated maximum and minimum axial temperature distribution for a TRISO/BISO fuel rod type with respect to fuel center line, fuel body, sleeve, and coolant gas for different time intervals of the irradiation history. The presented temperature range is calculated from the TREVER simulation and represents temperature brackets under normal operation, excluding any short-term transient conditions. Tuning of radial and axial temperature effects has been done with the TAC2D code (Ref. 7). Figure 11 shows a typical radial temperature distribution at axial peak location. Figure 12a gives the temperature history at the radial plane of thermocouple location with a 49.5-in. (1257 mm) active core height location, equivalent to a 75.5-in. (1917 mm) total core height, which coincides with fuel rod No. 10. This location is close to the peak axial fuel temperature, which is actually reached at fuel rod No. 12, as can be seen in Fig. 10.

Tables 4 through 6 present the temperature data for all four types of fuel rods in the eight-hole "teledial" arrangement. The $(2.75 \text{ Th:U})\text{C}_2$ TRISO, ThC_2 BISO fuel in holes 3 and 4 and the UC_2 TRISO, ThC_2 BISO fuel in holes 7 and 8 are grouped together because of their similar volume loading characteristics (given in Table 36), which are thought to be primarily responsible for irradiation-induced dimensional changes. The variation in fuel hole - fuel body gap from fuel hole to fuel hole caused by non-constant dimensional change of the rods is responsible for the observed variation in fuel temperature in the eight fuel holes. The fuel hole - fuel body gap is discussed in Section 4.6.4.

It can be seen that the TRISO/TRISO fuel rod variant in holes 5 and 6 experienced the lowest temperatures (Table 6), with the low-volume-fraction BISO fueled rods in holes 3, 4, 7, and 8 (Table 5) having the next higher temperature and the high-volume-fraction BISO fueled rod type in holes 1 and 2 (Table 4) being highest. Absolute differences of 40°C between the

peak temperatures of these fuel rod types are probably lower than the present uncertainty on the fuel temperature calculations. However, the relative grading is believed to be accurate, based on the evidence from fuel rod strain predictions and measurements and associated heat transfer gaps. Uncertainties on the calculated temperatures are estimated to be $\pm 100^{\circ}\text{C}$, for which no confidence level can be given. A systematic sensitivity analysis is planned which will take into account the variances on the material properties as well as asymmetrical behavior in the gap arrangements.

The latest H-327 graphite material properties were used (Ref. 8) together with Fort St. Vrain (FSV) and large HTGR fuel rod shrinkage models as described in the large HTGR fuel design data manual (Ref. 9). Thermal conductivities of the fuel rods are expressed as a function of the fast fluence, ϕ , by the relationship discussed in the fuel design data manual (Ref. 9).

The following fuel rods are identified as having the highest time-weighted temperatures, based on the thermal calculations:

<u>Highest Irradiation Temperature</u>	
Body 1	Rod 12
Body 2	Rod 12, also element high
Body 3	Rod 3

The following maximum, average, and minimum time-averaged fuel center line temperatures can be quoted for the axial peak temperature position at a 79.7-in. (2024 mm) core height, coinciding with fuel rod No. 12 in fuel body 2. The estimated uncertainty on the calculations is $\pm 100^{\circ}\text{C}$.

Hole	Fuel	Time-Averaged Peak Fuel Center Line Temperatures		
		Maximum (°C)	Mean (°C)	Minimum (°C)
1,2	UO ₂ TRISO, ThO ₂ BISO	1187	1115	1044
3,4	(2.75 Th,U)C ₂ TRISO, ThC ₂ BISO	1177	1105	1034
7,8	UC ₂ TRISO, ThC ₂ BISO			
5,6	(2.75 Th,U)C ₂ TRISO, ThC ₂ TRISO	1165	1093	1021

Calculated spine sample temperatures are given in Table 7.

Spine sample temperatures and fuel rod temperatures have been averaged by two different time-weighing methods:

$$\bar{T} = \sum (W_i T_i) \quad , \quad (4)$$

$$RMS_{\bar{T}} = \left\{ \sum [W_i (T_i - \bar{T})^2] \right\}^{1/2} \quad , \quad (5)$$

where W_i = weighing factor for time interval Δt_i ;

linear weighing:

$$W_i = \frac{\Delta t_i}{\sum \Delta t_i} \quad , \quad (6)$$

exponential weighing:

$$W_i = \frac{\Delta t_i \exp(-Q/RT_i)}{\sum [\Delta t_i \exp(-Q/RT_i)]} \quad , \quad (7)$$

where Q = activation energy or equivalent,

R = gas constant.

The first weighing procedure is appropriate for general fuel and material performance correlations. The second one is applicable for any kind of kinetic effect (i.e., amoeba, SiC attack, fission product diffusion). Temperatures are given in Table 7 for cases with activation energies of 143,700 J/mole (Cs diffusion in graphite) and 267,000 J/mole (Sr, Ba, Eu, Sm diffusion in graphite).

3.3.2. Temperature Measurements

FTE-4 was assembled with two thermocouples at 75.5-in. (1917.7 mm) total core height [i.e., 49.5-in. (1257.3 mm) active core height]:

TC-A: W/Re type thermocouple near spine sample

TC-B: C/A type thermocouple near outer sleeve location

For detailed axial and radial thermocouple locations, see Figs. 1 and 3.

Readings were recorded on a weekly basis and had to be corrected for:

1. Changes in cold junction temperatures ("standoff pin temperature") for both thermocouple types, which had been automatically done at Peach Bottom.
2. Adjustment in calibration equipment for the W/Re type only, where readings were 40°F (22°C) too high because of a change from the W-5% Re/W-26% Re type used for Core 1 to the W-3% Re/W-25% Re type used for Core 2.
3. Decalibration of the W/Re type only for permutation effect on emf output by neutron bombardment.

This decalibration can be correlated with the thermal fluence in addition to probable fluence and temperature gradient effects. Recalibration has been carried out on a group (not including FTE-4) of Peach

Bottom test elements, from which the following recalibration formula has been derived (Ref. 10):

$$T_a = -31.638 \Psi + 2.1181 \Psi^2 + (1 + 0.12172 \Psi) (T_i - T_s) + T_s, \quad (8)$$

where T_a = actual irradiation temperature (corrected signal) ($^{\circ}\text{C}$),
 T_i = indicated temperature (uncorrected signal) ($^{\circ}\text{C}$),
 T_s = standoff pin temperature (cold junction) ($^{\circ}\text{C}$),
 Ψ = thermal fluence in (10^{25} n/m², $E < 38$ aJ), dimensionless.

Figure 12b and Table 8 give the measured and corrected thermocouple readings and the calculated temperatures. The following values can be quoted:

Temperature Source	Peak ($^{\circ}\text{C}$)	Mean ($^{\circ}\text{C}$)	Long-Term RMS ($^{\circ}\text{C}$)	Medium-Term RMS ($^{\circ}\text{C}$)
TREVER prediction	1221	1146	± 39	N.D.
W/Re recalibrated thermocouple	1092	1047	± 26	± 47

The conclusions from the thermal analysis are as follows:

1. The C/A thermocouple was not functional.
2. The W/Re thermocouple reading at BOL is in very close agreement ($\sim 51^{\circ}\text{C}$) with the TREVER calculation.
3. The mismatch of calculated temperatures and corrected measurements at EOL is $\sim +135^{\circ}\text{C}$; the integrated mismatch, based on the number of thermocouple readings, is 99°C with an RMS deviation of $\pm 35^{\circ}\text{C}$. (See Table 8.)
4. Secondary effects such as eccentricity or thermocouple malfunction should not be excluded.

5. The thermocouple recalibration will be evaluated for associated uncertainties and eventual gradient effects in forthcoming post-irradiation examination reports.
6. The calculated temperatures are believed to be more representative for the thermal endurance characterization of the FTE-4 test element than information which can be extracted from this single thermocouple. It is possible that after evaluation of all 88 Peach Bottom thermocouple histories and associated postirradiation examination results, some adjustment would be appropriate.

3.4. BURNUP

3.4.1. Burnup Calculation

Burnup is defined as fissions per initial metal atom (FIMA) and can be calculated as follows:

$$\text{FIMA} = \frac{W_1/Z_1 - W_2/Z_2}{W_1/Z_1} = 1 - \frac{W_2}{W_1} \cdot \frac{Z_1}{Z_2} \quad (\text{atomic ratio}) \quad , \quad (9)$$

where W_1 = heavy metal nuclide weight at BOL (g),

W_2 = heavy metal nuclide weight at time t, i.e., EOL (g),

Z_1 = heavy metal atomic weight at BOL (g/mole),

Z_2 = heavy metal atomic weight at time t, i.e., EOL (g/mole).

A composite atomic weight from i components with the total weight W_j is calculated at a time point j by

$$Z_j = \frac{\sum (W_{j,i} \cdot Z_i)}{\sum W_{j,i}} \quad . \quad (10)$$

By selection of W_1 and Z_1 , the character of the FIMA value is defined:

$$W_{\text{fissile}} \text{ for FIMA}_{\text{fissile}} = F_5 \quad ,$$

$$W_{\text{fertile}} \text{ for FIMA}_{\text{fertile}} = F_3 ,$$

$$W_{\text{total}} \text{ for FIMA}_{\text{total}} = F_c .$$

Discrimination between fissile and fertile fuel at BOL is usually done by accounting for all isotopes from U-234 upward for fissile fuel and Th-232 for fertile fuel. For EOL, the U-234 amount has to be split into feed and breed fuel portions as obtained from GAUGE calculations. The same applies to the U-235 (and higher) isotopes. The GAUGE code discriminates feed and breed fuel portions only for U-234 and U-235.

The following different FIMA values can be quoted from GAUGE calculations:

1. For the fissile material with all uranium isotopes, excluding U-232, U-233, bred U-234, and bred U-235, but including plutonium and neptunium, for $U_j = W_j/Z_j$:

$$\text{FIMA}_{\text{fissile}} = \frac{U_1 - U_2}{U_1} \equiv F_5 . \quad (11)$$

2. For the fertile material with all thorium isotopes and Pa-231, Pa-233, U-232, U-233, bred U-234, and bred U-235 isotopes for $\text{Th}_j = W_j/Z_j$:

$$\text{FIMA}_{\text{fertile}} = \frac{\text{Th}_1 - \text{Th}_2}{\text{Th}_1} \equiv F_3 . \quad (12)$$

- 3a. For the total heavy metal material (composite burnup),

$$\text{FIMA}_{\text{total}} = F_5 \cdot x + F_3 \cdot (1 - x) \equiv F_c , \quad (13)$$

$$\text{where } x = \frac{U_1}{U_1 + \text{Th}_1} , \quad (14)$$

and if different blends, k, with different thorium loadings are used,

$$\text{FIMA}_{\text{total},k} = \text{FIMA}_{\text{total}} \cdot y \quad , \quad (15)$$

where

$$y = \frac{F_5 \cdot x_k + F_3 (1 - x_k)}{F_5 x + F_3 (1 - x)} \quad . \quad (16)$$

Fissile FIMA F_5 and fertile FIMA F_3 do not change with different blend types.

3b. For kernels with mixed fuel of a $\text{Th:U}_{\text{total}} = N$ atomic ratio,

$$\text{FIMA}_{\text{mixed}} = \frac{1}{1 + N} \cdot (F_5 + N \cdot F_3) \equiv F_c \quad . \quad (17)$$

Fuel loadings can be discriminated for either specific particles or specific isotopes, for which the above formulas must be used accordingly. The fuel loadings for FTE-4 are given in Table 9 for the four different blend types with one blend per two fuel holes. The amount of fissile material was the same in all four blends; the amount of fertile material was different. Additional heavy metal was located in the center spine samples. EOL total element loadings had been taken from nuclear physics calculations, using the GAUGE code. Linear corrections to GAUGE code calculations had to be made for the thorium loading, which was 4.26% too low for BOL in the calculation compared with the as-loaded figures.

GAUGE calculations are integrated values over the total length of the element. For any specific axial location, normalized axial FEVER distributions have to be applied to the fictive EOL sample loading, $\bar{w}'_2$, which is determined from the total element loadings w_1 and w_2 :

$$\bar{w}'_2 = w'_1 \cdot \frac{w_2}{w_1} \quad . \quad (18)$$

The true EOL loading of a specific sample, e.g., a fuel rod, will then be:

$$W'_2 = \bar{W}'_2 \cdot FFF_x, \quad (19)$$

where FFF_x = FEVER form factor at axial location x , which may be an average over the sample length ΔX .

The FEVER form factor is different for each isotope and duration time within the reactor. A detailed description will be given in a future report (Ref. 11). Examples for Th-232, U-233, U-235, and U-238 and the irradiation duration of FTE-4 are given in Figs. 13a through 13d, respectively.

The results of the burnup calculations are given in Tables 10 and 11 for average and individual axial fuel rod locations, respectively.

The following burnup calculations can be quoted:

Burnup Calculation	Peak	Mean	RMS	Mean/Gradient Fuel Body Sequence
Fissile FIMA (%)	36.40	29.54	± 6.26	213/231
Fertile FIMA (%)	0.75	0.45	± 0.24	213/231
Mixed FIMA (5.54 Th,U) (%)	6.21	4.91	± 1.16	213/231
Composite FIMA (%)				
Holes 1,2	4.19	3.26	± 0.81	213/231
Holes 3,4	7.11	5.64	± 1.31	213/231
Holes 5,6	7.48	5.94	± 1.37	213/231
Holes 7,8	8.92	7.12	± 1.61	213/231

The composite FIMA values are calculated using Eq. 17, the fissile and fertile FIMA values in Table 10, and the following N values for the different fuel blends:

Holes	N (Th/U Ratio)
1,2	9.37
3,4	4.52
5,6	4.30
7,8	3.36

3.4.2. Burnup Measurements

Four different techniques have been employed to measure the fuel burnup in FTE-4:

1. Radiochemistry.	} of loose particles
2. Mass spectrometry.	
3. Isotope dilution mass spectroscopy.	
4. Gamma spectroscopy of fuel rods.	

The first three methods have been applied to loose particles from spine sample locations; the last one used results from fuel rod gamma spectroscopy, reported in Section 4.6.7. Loose particles were only contained in body 3, whereas fuel rods from all three bodies have been analyzed.

3.4.2.1. Radiochemical Burnup Determination. Two radiochemical methods have been used: one utilizes Cs-137 and the other employs the stable fission products of zirconium as burnup monitors. Both radiochemical methods were performed on the same particle. The first method is an ASTM standard procedure (Ref. 12), and the second one is being developed (Ref. 13). Each particle was cleaned and measured by gamma spectroscopy with a calibrated system for prominent fission products plus Pa-233 where applicable. Particles showing a low Cs-137/Zr-95 ratio, which indicates fuel failure, were not used in the analysis.

The particles were crushed and dissolved in perchloric-nitric acid mixture. These solutions, then containing zirconium, uranium, and thorium (if the particles were fertile) in a strongly acidic chloride medium, were

separated from plutonium and fission products by an anion exchange separation. The thorium, zirconium, and uranium were separated by absorbing zirconium and uranium on a Dowex 1 x 8 column from 12 N HCl. Finally, uranium was eluted using 0.1 N HCl. Any plutonium, neptunium, or iron eluted with the uranium was reduced by passing through a lead reductor column, and in their reduced state these elements did not interact with Arsenazo III. Each fraction (thorium, zirconium, and uranium) was spectrophotometrically measured by the absorbance of its respective Arsenazo III complex.

Burnup is determined using the following equations:

$$FIMA_{\text{fissile}} = F_5 = \frac{F}{U^0} = \frac{F}{U + F + N_p + Pu} \quad (\text{atomic ratio}) \quad , \quad (20)$$

$$FIMA_{\text{fertile}} = F_3 = \frac{F}{Th^0} = \frac{F}{Th + U + F} \quad (\text{atomic ratio}) \quad , \quad (21)$$

where F = fissions per total sample,

$$= N/y \text{ (atoms)}, \quad (22)$$

N = atoms of stable zirconium (Zr-95 activity had been subtracted) for Eq. 20, or Cs-137, corrected for decay during and after irradiation, for Eq. 21, respectively,

y = fractional fission yields of appropriate fission product,

U^0 = initial atoms of uranium,

Th^0 = initial atoms of thorium,

U = remaining atoms of uranium,

N_p = remaining atoms of neptunium, $\left. \begin{array}{l} \text{not being used in the FTE-4} \\ \text{burnup evaluation; only significant at high burnups} \end{array} \right\}$

Pu = remaining atoms of plutonium,

Th = remaining atoms of thorium.

To determine the error in Eqs. 20 and 21 it is first necessary to progress the error in the determination of the number of atoms of uranium, thorium, zirconium, and Cs-137. The method of determining these values is taken from Ref. 13, and the results are given in Tables 12 and 13.

The number of Cs-137 atoms is determined by gamma scanning the particle and correcting the DPMs for decay during and after life. The 1σ error of $N_{\text{Cs-137}}$ is assumed to be $\pm 2.5\%$, which is the estimated calibration error of the SIGMA II Ge(Li) detector.

The number of Zr-95 atoms is determined from the following equation:

$$N_{\text{Zr}} = \frac{E_{\text{Zr}} \cdot D_{\text{Zr}} \cdot G_{\text{Zr}}}{F_{\text{Zr}} \cdot R_{\text{Zr}}} , \quad (23)$$

where N_{Zr} = atoms of zirconium per total sample,
 E_{Zr} = corrected sample reading, in absorbance units,
 F_{Zr} = calibration factor, absorbance units/ μg zirconium,
 D_{Zr} = dilution factor,
 G_{Zr} = 6.44×10^{15} atoms/ μg from fission of U-235 or U-233,
 $\quad = 6.60 \times 10^{15}$ atoms/ μg from natural zirconium,
 R_{Zr} = recovery fraction of zirconium from the column.

The error progression of Eq. 23 is as follows (G_{Zr} assumed to be a constant conversion factor):

$$dN_{\text{Zr}}^2 = N_{\text{Zr}}^2 \left[\left(\frac{dE_{\text{Zr}}}{E_{\text{Zr}}} \right)^2 + \left(\frac{dD_{\text{Zr}}}{D_{\text{Zr}}} \right)^2 + \left(\frac{dR_{\text{Zr}}}{R_{\text{Zr}}} \right)^2 + \left(\frac{dF_{\text{Zr}}}{F_{\text{Zr}}} \right)^2 \right] . \quad (24)$$

The number of atoms of uranium left in the sample is determined by Eq. 25:

$$N_{\text{U}} = \frac{E_{\text{U}} \cdot D_{\text{U}} \cdot G_{\text{U}}}{F_{\text{U}} \cdot R_{\text{U}}} , \quad (25)$$

where N_{U} = atoms of uranium per total sample,
 E_{U} = corrected sample reading (absorbance units),
 F_{U} = calibration factor (absorbance units/ μg uranium),

D_U = dilution factor,
 $G_U = 2.53 \times 10^{15}$ atoms/ μg uranium,
 R_U = recovery fraction of uranium from the column.

The error progression on the number of uranium atoms is as follows assuming G_U as constant conversion factor:

$$dN_U^2 = N_U^2 \left[\left(\frac{dE_U}{E_U} \right)^2 + \left(\frac{dD_U}{D_U} \right)^2 + \left(\frac{dR_U}{R_U} \right)^2 + \left(\frac{dF_U}{F_U} \right)^2 \right] \quad (26)$$

The only remaining value to be determined for fertile FIMA is the number of atoms of thorium, which is determined using Eq. 27:

$$N_{Th} = \frac{E_{Th} \cdot D_{Th} \cdot G_{Th}}{F_{Th} \cdot R_{Th}} \quad (27)$$

where N_{Th} = atoms of thorium per total sample,
 E_{Th} = corrected sample reading (absorbance units),
 F_{Th} = calibration factor (absorbance units/ μg thorium),
 D_{Th} = dilution factor,
 $G_{Th} = 2.595 \times 10^{15}$ atoms/ μg thorium,
 R_{Th} = recovery fraction of thorium from the column.

The error progression on the number of thorium atoms is as follows, assuming G_{Th} is a constant:

$$dN_{Th}^2 = N_{Th}^2 \left[\left(\frac{dE_{Th}}{E_{Th}} \right)^2 + \left(\frac{dD_{Th}}{D_{Th}} \right)^2 + \left(\frac{dR_{Th}}{R_{Th}} \right)^2 + \left(\frac{dF_{Th}}{F_{Th}} \right)^2 \right] \quad (28)$$

Using the number of Cs-137 atoms or Zr-95 atoms, the number of fissions is calculated using Eq. 22. The progressed error on the number of fissions is as follows:

$$dF^2 = F^2 \left[\left(\frac{dN}{N} \right)^2 + \left(\frac{dy}{y} \right)^2 \right] \quad (29)$$

With the errors calculated from Eqs. 25 through 28, the 1σ error can be progressed in Eqs. 30 and 31 assuming neptunium and plutonium are not included in the analysis because of the relatively low burnups.

The progressed error for Eq. 20 is

$$dF_5^2 = \left[-\frac{F}{(U + F)^2} \right]^2 (dU)^2 + \left[\frac{U}{(U + F)^2} \right]^2 (dF)^2 \quad . \quad (30)$$

The progressed error for Eq. 21 is

$$\begin{aligned} dF_3^2 = & \left[\frac{Th + U}{(Th + U + F)^2} \right]^2 (dF)^2 + \left[-\frac{F}{(Th + U + F)^2} \right]^2 (dTh)^2 \\ & + \left[-\frac{F}{(Th + U + F)^2} \right]^2 (dU)^2 \quad . \quad (31) \end{aligned}$$

In cases where multiple measurements are available, individual signals F_j and their standard deviations, s_j , can be combined to the mean value (see also Section 4.1.3.5):

$$\bar{F} = \frac{1}{m} \sum_j F_j \quad (32)$$

and the combined variance

$$s_F^2 = \left[\frac{1}{m^2} \sum s_{F_j}^2 + \frac{1}{m^2} \sum (F_j - \bar{F})^2 \right] \quad . \quad (33)$$

The 95% confidence limits can be constructed using

$$dF = \pm 1.96 s_F \quad . \quad (34)$$

The radiochemical burnup measurements and their associated 2σ errors are given in Table 14 and compared with mass spectrometric, gamma-spectrographic, IDMS, and GAUGE calculated burnups. It should be noted that the radiochemical measurements significantly exceed the mass spectrometric and calculated burnups. The applied rigorous error analysis does not explain the mismatch. Therefore, it must be concluded that the radiochemical measurements are systematically in error; e.g., undetected contamination, additional zirconium leaching of glass equipment (beyond the effects already detected and compensated for prior to these measurements), and underestimated thorium and uranium concentrations could have caused these overestimations.

3.4.2.2. Mass Spectrometric Burnup Determination. Uranium solutions of the particles analyzed using wet radiochemistry have been analyzed by a mass spectrometric uranium isotopic analysis method. This technique measures burnup through changes in uranium isotopic composition and can be applied only to uranium- and plutonium-containing fuel. For thorium-containing fuel, burnup can be determined only if the Th:U for the irradiated fuel is known. A further benefit obtained from uranium isotopic ratio measurements is a direct comparison with GAUGE-calculated ratios. Therefore, the fissile and fertile fuel particles had been subjected to mass spectrometry.

The method is standardized as ASTM E 244-69 (Ref. 19). Archive material of the same origin must be measured at the same time and in the same equipment. The atom percentages of U-235 and U-238, the ratios of U-235 and U-236 to U-238 before irradiation, and the isotopic ratios of U-235, U-236, Pu-239, Pu-240, Pu-241, and Pu-242 to U-238 after irradiation must be determined. In the presented case, the plutonium isotopes have been ignored because of the small amount of U-238 present; however, the U-233 and U-234 isotopes have been added. The results of these measurements and their uncertainties are given in Table 14 (burnup) and Table 15 (isotopic concentration).

The ASTM standard allows for three different methods of employing the isotopic data for burnup determination. The analysis functions are presented below together with the error analysis functions for the accumulated variances.

The symbols used in the following equations are:

- N_i^0 = heavy metal atom percent of uranium isotope i (last digit of mass number used) related to the total uranium atoms, in the preirradiated fuel,
- N_i = same as above for the irradiated fuel,
- R_i^0/j = atom ratio of U-23 i to U-23 j (i, j last digit of mass number) in the preirradiated fuel,
- R_i/j = same as above for the irradiated fuel,
- α_5 = effective ratio of U-235 (n, γ) capture-to-fission cross sections;

$$\alpha_5 = 0.242 \pm 0.009 \text{ (from GAUGE calculation),} \quad (35)$$

- S_K^0 = standard deviation of uncertainty of subject K at the preirradiation stage,
- S_K = same as above after irradiation.

At high exposures, where more than 10% of the U-235 is consumed, the FIMA_{fissile} can be calculated under the assumption of no loss of U-236 or U-238 during irradiation with ASTM Equation 1:

$$F_5 = N_8^0 [(R_{5/8}^0 - R_{5/8}) - (R_{6/8} - R_{6/8}^0)] \text{ (atom \%)} \quad (36)$$

In units of atom percent concentration, Eq. 36 is as follows:

$$F_5 = N_5^0 + N_6^0 - \frac{N_8^0}{N_8} \cdot (N_5 + N_6) \text{ (atom \%)} \quad (37)$$

The analysis function for the variances is

$$s^2 = s_5^2 + s_6^2 + s_5^2 + s_6^2 + \left[s_8^2 + s_8^2 \left(\frac{N_8^0}{N_8} \right)^2 \right] \cdot \left(\frac{N_5 + N_6}{N_8} \right)^2 (\text{atom } \%)^2 \quad (38)$$

Equation 36 is extremely sensitive to and adversely affected by natural uranium contamination in the irradiated sample, especially in fully enriched uranium after short exposure.

At low exposures, ASTM Equation 2, which assumes no loss of U-236 during irradiation, is recommended. This equation is equally valid for high exposures. ASTM Equation 2 is as follows:

$$F_5 = N_5^0 \{ (R_{6/5} - R_{6/5}^0) / [R_{6/5} + \alpha_5 (1 + R_{6/5})] \} (\text{atom } \%) \quad (39)$$

The atom percent concentration is in the form

$$F_5 = (N_5^0 \cdot N_6 - N_6^0 \cdot N_5) / (N_6 + N_6 \cdot \alpha_5 + N_5 \cdot \alpha_5) (\text{atom } \%) \quad (40)$$

with the analysis function for the variances being

$$s^2 = \frac{1}{(N_6 + N_6 \cdot \alpha_5 + N_5 \cdot \alpha_5)^2} \left\{ s_5^2 \cdot N_6^2 + s_5^2 (-N_6^0 - \alpha_5 \cdot F_5)^2 + s_6^2 (-N_5)^2 + s_6^2 [N_5^0 - (1 + \alpha_5) \cdot F_5]^2 + s_{\alpha_5}^2 [-F_5 \cdot (N_6 + N_5)]^2 \right\} (\text{atom } \%)^2 \quad (41)$$

Because the change in the ratio $R_{6/5}$ is always greater than the corresponding change in the ratio $R_{5/8}$, ASTM Equation 2 is a more sensitive measure of the fissile FIMA than either of the other two ASTM equations. Because of the independence of U-238, it is 139 times less sensitive to natural uranium contamination than the other two. However, it is very sensitive to knowledge of α_5 .

For recycled uranium, the initial U-236 content could be large as well as the variance from fuel to fuel. The following equations allow the burnup estimation without knowledge of the U-236.

ASTM Equation 3 is as follows:

$$F_5 = N_8^0 [(R_{5/8}^0 - R_{5/8}) / (1 + \alpha_5)] \quad (\text{atom } \%) \quad . \quad (42)$$

The atom percent concentration is in the form

$$F_5 = \left(N_5^0 - N_5 \cdot \frac{N_8^0}{N_8} \right) / (1 + \alpha_5) \quad (\text{atom } \%) \quad . \quad (43)$$

The analysis function for the associated variances is

$$s^2 = \frac{1}{(1 + \alpha_5)^2} \left\{ s_5^0{}^2 + \frac{1}{N_8^2} \left[s_5^2 \cdot (-N_8^0)^2 + s_8^0{}^2 \cdot (-N_5)^2 + s_8^2 \left(\frac{N_8^0 \cdot N_5}{N_8} \right)^2 \right] + s_{\alpha}^2 (-F_5)^2 \right\} \quad (\text{atom } \%)^2 \quad . \quad (44)$$

This third group of equations is also dependent on knowledge of α_5 , but the dependence is less sensitive, as can be seen from comparison of the last term in Eqs. 41 and 44.

In Table 15, the GAUGE-calculated uranium concentrations are compared with the mass spectrometric and IDMS results. GAUGE/FEVER data have been adjusted for the as-measured BOL nuclide concentration. Comparison is made by normalization to the GAUGE code data. Based on this definition, the agreement between measured and calculated compositions for fissile fuel and U-235 is within $\pm 12\%$, for U-236 and U-238 it ranges between -2% and -7% , and U-234 is underpredicted by -15% . The bias for fertile fuel and U-233 is within -1% , for U-234 $+22\%$, and for U-235 $+38\%$. Based on these results it can be concluded that the fissile material is well defined by the GAUGE/FEVER calculations and that the uncertainties in burnup estimates are more likely related to the non-fissile isotopes.

3.4.2.3. Isotope Dilution Mass Spectroscopy Burnup Determination. The isotope dilution mass spectroscopy (IDMS) method is a very accurate technique for determining quantitative amounts of uranium with a mass spectrometer. The analysis is described in detail in Ref. 20. The results of the analysis are given in Table 16, and FIMA values calculated from the IDMS results are compared with those obtained from other methods in Table 14.

Using the uranium IDMS values, the fissile FIMA can be calculated with any of the three ASTM equations (36, 39, or 42) or with the radiochemistry equation (20).

When the IDMS data are used to calculate the fissile FIMA with the mass spectroscopy equations (36, 39, and 42), the isotope fraction is calculated as follows:

$$U_F = \frac{U_I}{U_T} \quad , \quad (45)$$

where U_F = uranium isotope fraction,

U_I = uranium isotope concentration from IDMS (ppm),

U_T = total uranium isotope concentration from IDMS (ppm).

The error progression is as follows:

$$dU_F^2 = U_F^2 \left[\left(\frac{dU_I}{U_I} \right)^2 + \left(\frac{dU_T}{U_T} \right)^2 \right] \quad (46)$$

The error on each of the uranium isotope fractions calculated from Eq. 46 contributes to the error on the mean uranium isotope:

$$dU_T^2 = \sum dU_I^2 \quad (47)$$

An alternative, but less accurate, procedure is*

$$dU_T = \sum dU_I \quad (48)$$

These variances are then used in Eqs. 38, 41, and 44 to determine the variance for the fissile FIMA from Eqs. 36, 39, and 42.

IDMS uranium values can be used in the radiochemical evaluation of the fissile FIMA:

$$U_{\text{corr}} = \frac{U_I D_U X_U}{R_U} \quad (49)$$

where U_{corr} = uranium atoms in the total sample,

U_I = uranium atoms from IDMS measurements,

R_U = recovery fraction of uranium from the column (used in Eq. 25),

D_U = dilution factor,

$X_U = 2.545 \times 10^{15}$ atoms U/mg (FTE-4 value; will vary).

*Earlier laboratory data were used with Eq. 48 to determine the total error. Equation 48 is the procedure used in this report.

The progressed error is under the assumption of a constant conversion factor X_U :

$$dU_{\text{corr}}^2 = U_{\text{corr}}^2 \left[\left(\frac{dU_I}{U_I} \right)^2 + \left(\frac{dR_U}{R_U} \right)^2 + \left(\frac{dD_U}{D_U} \right)^2 \right] \quad (50)$$

Using Eqs. 49 and 20, the fissile FIMA can then be calculated using either zirconium or Cs-137 as a fission monitor. The error progression for the calculated FIMA is calculated using Eq. 30 in combination with Eqs. 24, 29, and 50 and the assumed 2.5% error on Cs-137. The results of the IDMS associated fissile FIMA values and uncertainty limits are shown in Table 14.

3.4.2.4. Gamma Spectrometric Burnup Determination. Fuel gamma spectrometric measurements performed as described in Section 4.6.7 have been employed to estimate the composite burnup in the examined fuel rods by using Cs-137 as a burnup monitor. The GAUGE-calculated and gamma-scan-measured values are compared in Table 17.

The analysis function is similar to the radiochemistry evaluation (Eqs. 20 through 27). It differs in the respect that the historically reported heavy metal loadings for the fuel rods are applied, for which an arbitrary $\pm 5\%$ uncertainty (1σ) has been estimated. Rod-to-rod variations will be determined further by evaluating gamma spectroscopic data. Specifically for one blend (holes 7 and 8), significant inhomogeneities have been found during fuel metallography (see Section 4.6.6), which may also affect rod-to-rod variations.

The composite burnup is calculated using Cs-137 as a monitor for the number of fissions:

$$F_c = \frac{N}{y (U^0 + Th^0)} \quad (51)$$

where F_c = composite burnup,
 y = fission yield of Cs-137 from U-235,

N = number of Cs-137 atoms corrected for decay during and after irradiation,
 U^0 = initial uranium loading,
 Th^0 = initial thorium loading.

The number of Cs-137 atoms at the time of counting is determined by

$$N_o = \frac{(GF)(CPM)}{(DRE)(AI) \lambda} \quad , \quad (52)$$

where GF = geometry factor, postirradiation length of rod/effective columnator width,
 CPM = counts per minute of Cs-137,
 DRE = detector efficiency at 662 keV,
 AI = absolute intensity,
 λ = Cs-137 decay constant (7.3×10^{-10} sec).

The number of atoms produced during the life of the element is approximated as follows:

$$N = N_o \frac{\lambda t}{e^{-\lambda t'} (1 - e^{-\lambda t})} \quad , \quad (53)$$

where N = atoms of Cs-137 in the total sample corrected for decay during and after irradiation,
 N_o = atoms of Cs-137 in the total sample at the time of measurement,
 λ = Cs-137 decay constant, 7.3×10^{-10} sec⁻¹,
 t' = elapsed time from the end of irradiation to measurement, in seconds,
 t = irradiation time (sec).

Equation 53 assumes that the element experienced a steady-state operation during its life. A rigorous approach with corrections for shutdown periods during the element's life is not necessary because the shutdown times are insignificant compared with the 30-yr half-life of Cs-137.

The error statement for the composite FIMA, F_c , proceeds as follows:

$$dF_c = F_c \left\{ \left[\frac{d(GF)}{GF} \right]^2 + \left[\frac{d(CPM)}{CPM} \right]^2 + \left[\frac{d(DRE)}{DRE} \right]^2 + \left[\frac{d(AI)}{AI} \right]^2 + \left(\frac{d\lambda}{\lambda} \right)^2 + \left(\frac{dy}{y} \right)^2 + \left[\frac{d(Th^0 + U^0)}{(Th^0 + U^0)} \right]^2 \right\}^{1/2} \quad (54)$$

The following assumptions are made in this error progression:

1. F_c does not include the $[\lambda t / (1 - e^{-\lambda t})]$ term to correct for time during irradiation. This term is assumed to have an insignificant error and is near 1 for Cs-137 in FTE-4. [However, the $\lambda t / (1 - e^{-\lambda t})$ term is used in the calculation of the burnup using Cs-137 as a monitor.]
2. DPM is corrected for decay to EOL.
3. $(Th^0 + U^0)$ metal loadings are included together because of small differences in atomic weight.

In cases where multiple measurements were available, e.g., several spectra taken from the same rod, the results were combined in a manner analogous to Eqs. 32 and 33. Calculated combined FIMAs were determined from the GAUGE/FEVER fissile and fertile FIMA values and Eq. 13 for the four fuel varieties. Both the gamma-spectroscopy-measured and calculated combined burnups are given in Table 17. In Figs. 14 through 16, three of the fuel rod types are shown with available gamma scan burnups and the GAUGE/FEVER calculated combined burnup profiles. No fuel from holes 1 and 2 was included in the comparison because at the time of gamma scanning, all the surplus FTE-4 fuel had been sent to Oak Ridge National Laboratory (ORNL) for head-end hot cell studies. At ORNL five rods were scanned (Ref. 21) as described in Section 4.6.7. Fortunately, two fuel rods from stack 1 were available for this exercise.

In order to compare the gamma scan burnups with radiochemistry, IDMS, and mass spectroscopy data, a composite burnup was calculated for each case using Eq. 13, and the associated variances were determined using Eqs. 22 through 31. The results of this comparison are given in Table 18, and highlights are listed below:

1. For the bottom end of the test element, i.e., body 1, there is very good agreement between gamma-spectroscopy-measured and calculated burnups. In the middle and top sections, i.e., body 2 and 3, the gamma-spectroscopy-measured burnups are consistently higher than the GAUGE/FEVER calculated values, which are exceeded by up to 20%.
2. The mass spectrometric composite FIMA values are in line with calculated values and are 15% to 20% lower than the gamma spectroscopy values.
3. Radiochemistry composite FIMAs were higher than comparable mass spectroscopy and gamma spectroscopy values. The radiochemistry values were up to 30% higher than associated mass spectroscopy values.
4. The individual burnup results can be summarized as follows:
 - a. For the fissile burnup,

$$F_5 \text{ calculated} < F_5 \frac{\text{mass spec.}}{\text{IDMS}} < F_5 \text{ radiochemistry}$$

Spine Sample	Particle Type	$\left(\frac{F_5 \text{ Radio. (a)}}{F_5 \text{ Mass Spec. + IDMS}} \right) - 1$	Error (1 σ)
TS4-5	UC ₂	+0.08	±0.06
TS28-6	UO ₂	+0.22	±0.04

(a) Cs-137 monitor.

b. For the fertile burnup,

$$F_3 \text{ calculated} < F_3 \text{ radiochemistry}$$

$$\frac{F_3 \text{ calc.}}{F_3 \text{ meas.}} - 1 = -0.28 \pm 0.10 (1\sigma)$$

The mismatch between radiochemical and calculated burnups may be due to larger variances and bias in the uranium loading of the fuel rods and, more likely, in the thorium loading, which had not been so severely controlled as the uranium loading. Attempts will be made to analyze a few rods which had been fueled with thorium only in order to determine non-composite burnups for the fuel rod size geometry. In summary, it can be stated that:

1. Mass spectrometry is the most precise burnup method, which gave good justification for the fissile to fertile capture ratio of $\alpha_5 = 0.242 \pm 0.009$ used in depletion calculations for the Peach Bottom HTGR.
2. Mass spectrometry confirms the fissile isotopic concentrations as determined by GAUGE/FEVER; for some non-fissile isotopes, this must be further evaluated.
3. Radiochemistry is presently the only way to measure fertile burnup. This method needs methodical improvements for the fissile and the fertile FIMA determination.
4. GAUGE/FEVER calculated burnups may underestimate the burnup for FTE-4 in the upper half of the fuel element.
5. Heavy metal loadings of the fuel rods may be slightly in bias.

6. Gamma spectroscopy has been applied as a valid method to determine fission product distribution and does allow some degree of certainty in a burnup determination. The calibration of the system must still be improved in order to reduce the major area of uncertainty in determination of the cesium activities.

TABLE 2
FTE-4 POWER, FLOW, AND FOUR-GROUP FLUX HISTORY^(a)

Beginning of Interval (EFPD)	Real Time of Interval (days)	Absolute Radial Power (kW)	Relative Radial Power	Relative Radial Power Coolant	Channel Flow (kg/s)	Four Flux Groups ^(b)				
						Group 1A ^(c) (10^{17} n/m ² /s)	Group 1B (10^{17} n/m ² /s)	Group 2 (10^{17} n/m ² /s)	Group 3 (10^{17} n/m ² /s)	Group 4 (10^{17} n/m ² /s)
252.41	56.40	110.9460	0.9840	1.0970	0.0311	3.4204	0.6036	6.6157	1.0771	4.2181
298.00	55.62	105.3400	0.9360	1.0450	0.0309	3.3400	0.5894	6.5130	1.0605	4.1707
342.95	46.98	117.8230	0.9340	1.0440	0.0325	3.6500	0.6441	7.1374	1.1612	4.8079
385.44	0.00	117.0860	0.9280	1.0380	0.0325	0.0000	0.0000	0.0000	0.0000	0.0000
385.44	145.87	102.4190	0.9290	1.0400	0.0301	3.1406	0.5542	6.1538	1.0005	4.2327
499.59	80.59	99.8910	0.8950	1.0050	0.0320	3.1143	0.5496	6.1815	1.0045	4.5597
564.07	51.56	108.3230	0.8970	0.9740	0.0329	3.4795	0.6140	6.9705	1.1330	5.2258
610.25	98.73	107.8950	0.8390	0.9450	0.0333	3.3628	0.5934	6.7959	1.1056	5.3924
701.23	0.00	107.2520	0.8340	0.9410	0.0333	$\sim 3 \times 10^{-8}$	$\sim 6 \times 10^{-8}$	$\sim 7 \times 10^{-8}$	$\sim 1 \times 10^{-8}$	$\sim 5 \times 10^{-8}$
Average	535.75 ^(d)	106.1677	0.9112	1.0177	0.0316	3.3050	0.5832	6.5271	1.0616	4.6336
RMS	--	± 5.0136	± 0.0421	± 0.0460	± 0.0012	± 0.1684	± 0.0297	± 0.3488	± 0.0570	± 0.4770

(a) All time weighing was done using real time at operation.

(b) The energy ranges for the flux groups are as follows:

- Group 1A: 28.8 fJ - 2.4 pJ
- Group 1B: 13.8 fJ - 28.8 fJ
- Group 2: 2.8 aJ - 13.8 fJ
- Group 3: 0.38 aJ - 2.8 aJ
- Group 4: 0.38 aJ - 0

(c) Group 1A, the fast flux group, has been modified from the GAUGE output by a factor of 0.85. This accounts for the Peach Bottom energy cutoff of 13.8 fJ instead of 28.8 fJ.

(d) Total real time at operation used for time-weighted averaging.

TABLE 3
FTE-4 THERMAL AND FAST FLUENCE FOR FUEL RODS

Fuel Body	Fuel Rod	Mean Core Height (mm)	Fast Fluence (10^{25} n/m ² /s)	Thermal Fluence (10^{25} n/m ² /s)
1	1	685	0.595	0.838
	2	735	0.784	1.008
	3	785	0.952	1.181
	4	835	1.097	1.357
	5	885	1.227	1.532
	6	934	1.345	1.707
	7	984	1.445	1.879
	8	1034	1.534	2.047
	9	1084	1.612	2.195
	10	1134	1.683	2.336
	11	1184	1.744	2.450
	12	1233	1.799	2.551
	13	1283	1.841	2.622
	14	1333	1.877	2.679
Body mean			1.395	1.885
RMS			±0.396	±0.601
2	1	1475	1.938	2.779
	2	1525	1.954	2.805
	3	1575	1.967	2.828
	4	1625	1.974	2.843
	5	1675	1.976	2.854
	6	1725	1.973	2.857
	7	1774	1.968	2.857
	8	1824	1.958	2.849
	9	1874	1.953	2.835
	10	1924	1.946	2.806
	11	1974	1.938	2.711
	12	2024	1.890	2.727
	13	2074	1.851	2.674
	14	2123	1.824	2.610
Body mean			1.937	2.793
RMS			±0.046	±0.073
3	1	2266	1.725	2.412
	2	2315	1.674	2.318
	3	2365	1.623	2.223
	4	2415	1.562	2.122
	5	2465	1.499	2.021
	6	2515	1.430	1.909
	7	2565	1.360	1.797
	8	2615	1.278	1.681
	9	2664	1.195	1.568
	10	2714	1.101	1.378
	11	2764	1.000	1.375
	12	2814	0.875	1.301
	13	2864	0.737	1.228
	14	2914	0.552	1.167
Body mean			1.258	1.757
RMS			±0.351	±0.405
Element mean			1.530	2.145
RMS			±0.424	±0.624

TABLE 4
FTE-4 IRRADIATION TEMPERATURES (HOLES 1 AND 2)

Fuel Rod I.D.	Mean Pos. Core Ht. (mm)	Temperature Envelope (°C)						Time-Weighted Averages (°C)									Coolant Temp. (°C)
		Fuel Max.	Fuel Min.	Body Max.	Body Min.	Slv. Max.	Slv. Min.	Fuel Max.	Fuel Avg.	Fuel Min.	Body Max.	Body Avg.	Body Min.	Slv. Max.	Slv. Avg.	Slv. Min.	
Fuel Body 1																	
1	685	615	502	513	457	441	384	572	546	521	487	479	472	418	406	393	340
2	735	692	550	569	501	485	413	642	610	579	537	528	518	458	440	423	344
3	785	766	594	621	543	525	440	709	672	635	584	573	561	493	472	450	349
4	835	834	636	666	579	556	462	771	728	685	626	613	600	522	497	473	355
5	885	887	674	702	612	581	484	825	776	728	662	648	633	548	521	493	362
6	934	931	708	739	642	602	500	870	817	764	693	678	662	570	541	512	369
7	984	978	740	771	670	625	514	909	852	796	721	705	688	590	560	530	377
8	1034	1020	768	801	695	647	528	944	884	824	746	729	712	609	578	546	386
9	1084	1056	793	828	717	668	543	976	913	850	769	752	735	627	595	563	395
10	1134	1089	817	852	738	687	556	1005	940	875	791	773	756	644	611	578	404
11	1184	1120	839	875	755	705	569	1032	964	897	811	793	775	660	627	594	414
12	1233	1146	859	896	770	722	582	1055	985	916	829	811	793	675	641	608	424
13	1283	1101	856	892	767	720	581	1051	981	912	826	808	790	673	640	607	414
14	1333	1114	838	872	753	705	570	1029	980	892	810	792	774	660	628	546	445
Mean	1009	956	727	757	657	619	509	885	831	777	707	692	676	582	554	526	385
Range/±RMS	698	168	115	121	101	88	62	154	139	125	108	105	102	81	74	68	33
Fuel Body 2																	
1	1475	1128	851	884	765	717	582	1043	974	905	822	804	787	671	640	608	455
2	1525	1186	898	932	809	758	617	1099	1026	954	888	849	831	710	677	645	466
3	1575	1225	930	965	840	787	643	1137	1063	988	900	881	863	738	705	671	477
4	1625	1236	942	977	853	798	654	1150	1075	1000	912	893	874	749	716	683	488
5	1675	1244	953	988	862	809	665	1159	1084	1009	921	903	884	759	727	694	499
6	1725	1252	962	995	871	818	675	1167	1092	1017	931	912	893	769	737	704	510
7	1774	1257	970	1002	880	828	685	1174	1100	1025	939	921	902	778	746	714	521
8	1824	1261	978	1009	888	836	695	1180	1105	1031	947	928	910	787	755	724	531
9	1874	1261	983	1013	896	843	704	1183	1109	1036	952	934	916	794	764	733	542
10	1924	1261	986	1016	902	850	713	1185	1112	1039	957	940	922	802	771	741	553
11	1974	1280	993	1019	907	857	722	1187	1115	1043	962	945	928	808	779	749	563
12	2024	1256	994	1021	909	861	730	1186	1115	1044	965	948	931	814	785	758	574
13	2074	1224	971	997	889	845	720	1157	1089	1021	945	929	913	800	773	745	584
14	2123	1177	933	957	854	816	699	1109	1045	981	910	895	879	774	748	722	594
Mean	1799	1231	953	984	868	818	679	1151	1079	1007	924	906	888	768	737	706	526
Range/±RMS	698	39	39	37	39	39	41	40	39	38	39	39	39	39	40	41	43
Fuel Body 3																	
1	2266	1175	928	955	853	817	705	1103	1041	978	909	894	879	777	752	727	604
2	2315	1215	965	992	888	853	739	1142	1078	1015	945	930	915	812	787	762	614
3	2365	1236	985	1013	907	875	762	1161	1098	1034	966	951	936	833	808	784	623
4	2415	1227	978	1011	903	875	766	1150	1089	1028	962	948	933	834	811	787	632
5	2465	1215	974	1016	900	876	769	1139	1080	1021	958	944	931	835	812	790	641
6	2515	1200	986	999	895	875	772	1125	1069	1012	952	939	926	834	813	792	649
7	2565	1183	956	991	888	872	774	1108	1055	1001	944	931	919	832	812	793	657
8	2615	1166	945	982	881	869	775	1091	1040	989	935	924	912	829	811	793	665
9	2664	1145	935	971	874	866	775	1072	1025	977	926	915	905	826	809	792	672
10	2714	1121	922	959	866	863	774	1052	1007	963	916	906	896	822	807	791	679
11	2764	1094	907	944	856	858	772	1029	988	947	903	894	885	817	803	789	685
12	2814	1065	894	929	846	853	770	1006	968	931	891	883	875	812	799	787	691
13	2864	1028	874	911	831	838	755	979	943	906	870	863	856	796	784	773	696
14	2914	993	850	888	809	816	732	948	913	877	844	837	831	772	761	751	702
Mean	2590	1147	934	968	871	858	760	1079	1028	977	923	911	900	816	798	779	658
Range/±RMS	698	74	40	38	28	20	20	65	55	46	35	33	30	20	19	19	30
Element mean	1799	1111	871	903	798	764	649	1038	979	920	851	836	822	722	896	671	523
Element range/ ±RMS	2278	158	126	129	119	118	114	150	139	130	123	123	122	114	115	117	117

TABLE 5
FTE-4 IRRADIATION TEMPERATURES (HOLES 3, 4, 7, AND 8)

Fuel Rod I.D.	Mean Pos. Core Ht. (mm)	Temperature Envelope (°C)						Time-Weighted Averages (°C)									Coolant Temp. (°C)
		Fuel Max.	Fuel Min.	Body Max.	Body Min.	Slv. Max.	Slv. Min.	Fuel Max.	Fuel Avg.	Fuel Min.	Body Max.	Body Avg.	Body Min.	Slv. Max.	Slv. Avg.	Slv. Min.	
Fuel Body 1																	
1	685	612	500	513	457	441	384	570	545	519	487	479	472	418	406	393	340
2	735	686	549	569	501	485	413	639	607	576	537	528	518	458	440	423	344
3	785	758	594	621	543	525	440	704	667	630	584	573	561	493	472	450	349
4	835	824	636	666	579	556	462	766	722	679	626	613	600	522	497	473	355
5	885	875	674	702	612	581	484	818	769	720	662	648	633	548	521	493	362
6	934	924	708	739	642	602	500	862	809	756	693	678	662	570	541	512	369
7	984	970	740	771	670	625	514	900	843	786	721	705	688	590	560	530	377
8	1034	1011	768	801	695	647	528	934	874	814	746	729	712	609	578	546	386
9	1084	1047	793	828	717	668	543	966	903	840	769	752	735	627	595	563	395
10	1134	1080	817	852	738	687	556	995	929	864	791	773	756	644	611	578	404
11	1184	1109	839	875	755	705	569	1021	953	885	811	793	775	660	627	594	414
12	1233	1136	859	896	770	722	582	1043	974	904	829	811	793	675	641	608	424
13	1283	1130	856	892	767	720	581	1039	970	900	826	808	790	673	640	607	434
14	1333	1104	838	872	753	705	570	1018	949	881	810	792	774	660	628	596	445
Mean	1009	948	727	757	657	619	509	877	823	768	707	692	676	582	554	526	385
Range/±RMS	698	167	115	121	101	88	62	150	136	122	108	105	102	81	74	68	33
Fuel Body 2																	
1	1475	1118	851	884	765	717	582	1032	963	894	822	804	787	671	640	608	455
2	1525	1175	898	932	809	758	617	1088	1015	942	868	849	831	710	677	645	466
3	1575	1214	930	965	800	787	643	1126	1051	976	900	881	863	738	705	671	477
4	1625	1226	942	977	853	798	654	1138	1063	988	912	893	874	749	716	683	488
5	1675	1234	953	986	862	809	665	1147	1072	997	921	903	884	759	727	694	499
6	1725	1242	962	995	871	818	675	1156	1080	1005	931	912	893	769	737	704	510
7	1774	1247	970	1002	880	828	685	1163	1088	1013	939	921	902	778	746	714	521
8	1824	1251	978	1009	888	836	695	1169	1094	1019	947	928	910	787	755	724	531
9	1874	1252	983	1013	896	843	704	1172	1098	1024	952	934	916	794	764	733	542
10	1924	1252	986	1016	902	850	713	1175	1102	1028	957	940	922	802	771	741	553
11	1974	1251	990	1019	907	857	722	1177	1104	1032	962	945	928	808	779	749	563
12	2024	1247	991	1021	909	861	730	1176	1105	1034	965	948	931	814	785	758	574
13	2074	1213	968	997	889	845	720	1148	1079	1011	945	929	913	800	775	745	584
14	2125	1162	929	957	854	816	699	1100	1036	971	910	895	879	774	748	722	594
Mean	1799	1220	952	984	866	816	679	1140	1068	995	924	906	888	768	737	706	526
Range/±RMS	698	40	38	37	39	39	41	40	39	38	39	39	39	39	40	41	43
Fuel Body 3																	
1	2266	1161	925	955	853	817	705	1095	1032	969	909	894	879	777	752	727	604
2	2315	1201	962	992	888	853	739	1133	1070	1006	945	930	915	812	787	762	614
3	2365	1222	982	1013	907	875	762	1152	1089	1026	966	951	936	833	808	784	623
4	2415	1214	976	1011	903	875	766	1142	1081	1020	962	948	933	834	811	787	632
5	2465	1202	971	1006	900	876	789	1132	1073	1014	958	944	931	835	812	790	641
6	2515	1188	964	999	895	875	772	1118	1061	1005	952	939	926	834	813	792	649
7	2565	1172	954	991	888	872	774	1102	1048	994	944	931	919	832	812	793	657
8	2615	1155	942	982	881	869	775	1085	1034	983	935	924	912	829	811	793	665
9	2664	1135	931	971	874	866	775	1067	1019	971	926	915	905	826	809	792	672
10	2714	1112	919	959	866	863	774	1047	1003	958	916	906	896	822	807	791	679
11	2764	1087	904	944	856	858	772	1025	984	943	903	894	885	817	803	789	685
12	2814	1059	892	929	848	853	770	1002	965	927	891	883	875	812	799	787	691
13	2864	1023	872	911	831	838	755	976	940	904	870	863	856	796	784	773	696
14	2914	992	848	888	809	816	732	946	911	875	844	837	831	772	761	751	702
Mean	2590	1137	932	968	871	858	760	1073	1022	971	923	911	900	816	798	779	658
Range/±RMS	698	70	39	38	28	20	20	63	53	44	35	33	30	20	19	19	30
Element mean	1799	1102	870	903	798	764	649	1030	971	912	851	836	822	722	696	671	523
Element range/ ±RMS	2278	156	126	129	119	118	114	148	138	128	123	123	122	114	115	117	117

TABLE 6
FTE-4 IRRADIATION TEMPERATURES (HOLES 5 AND 6)

Fuel Rod I.D.	Mean Pos. Core Ht (mm)	Temperature Envelope (°C)						Time-Weighted Averages (°C)									Coolant Temp. (°C)
		Fuel Max.	Fuel Min.	Body Max.	Body Min.	Slv. Max.	Slv. Min.	Fuel Max.	Fuel Avg.	Fuel Min.	Body Max.	Body Avg.	Body Min.	Slv. Max.	Slv. Avg.	Slv. Min.	
Fuel Body 1																	
1	685	606	498	513	457	441	384	568	542	516	487	479	472	418	406	393	340
2	735	677	547	569	501	485	413	634	603	571	537	528	518	458	440	423	344
3	785	748	592	621	543	525	440	698	661	624	584	573	561	493	472	450	349
4	835	812	636	666	579	556	462	758	714	671	626	613	600	522	497	473	355
5	885	861	674	702	612	581	484	809	760	711	662	648	633	548	521	493	362
6	934	913	706	739	642	602	500	852	799	746	693	678	662	570	541	512	369
7	984	958	740	771	670	625	514	889	832	775	721	705	688	590	560	530	377
8	1034	999	768	801	695	647	528	923	863	802	746	729	712	609	578	546	386
9	1084	1034	793	828	717	668	543	954	891	827	769	752	735	627	595	563	395
10	1134	1068	817	852	738	687	556	982	916	851	791	773	756	644	611	578	404
11	1184	1095	839	875	755	705	569	1008	940	872	811	793	775	660	627	594	414
12	1233	1121	859	896	770	722	582	1031	961	890	829	811	793	675	641	608	424
13	1283	1116	856	892	767	720	581	1027	957	887	826	808	790	673	640	607	434
14	1333	1090	838	872	753	705	570	1005	937	868	810	792	774	660	628	596	445
Mean	1009	936	726	757	657	619	509	867	812	758	707	692	676	582	554	526	385
Range/±RMS	698	165	116	121	101	88	82	147	133	118	108	105	102	81	74	68	33
Fuel Body 2																	
1	1475	1104	851	884	765	717	582	1020	950	881	822	804	787	871	640	608	455
2	1525	1161	897	932	809	758	617	1074	1001	928	868	849	831	710	677	645	466
3	1575	1200	929	965	840	787	643	1112	1037	962	900	881	863	738	705	671	477
4	1625	1211	941	977	853	798	654	1125	1049	974	912	893	874	749	716	683	488
5	1675	1220	950	986	862	809	665	1134	1059	983	921	903	884	759	727	694	499
6	1725	1227	958	995	871	818	675	1143	1067	991	931	912	893	769	737	704	510
7	1774	1233	966	1002	880	828	685	1150	1075	999	939	921	902	778	746	714	521
8	1824	1237	972	1009	888	836	695	1156	1081	1006	947	928	910	787	755	724	531
9	1874	1238	977	1013	898	843	704	1160	1085	1011	952	934	916	794	764	733	542
10	1924	1238	980	1016	902	850	713	1162	1089	1015	957	940	922	802	771	741	553
11	1974	1238	984	1019	907	857	722	1165	1092	1019	962	945	928	808	779	749	563
12	2024	1234	985	1021	909	861	730	1164	1093	1021	965	948	931	814	785	756	574
13	2074	1201	963	997	889	845	720	1136	1067	999	945	929	913	800	773	745	584
14	2123	1148	925	957	854	816	699	1090	1025	960	910	895	879	774	748	722	594
Mean	1799	1207	908	984	866	816	679	1128	1055	982	924	906	888	768	737	706	526
Range/±RMS	698	40	37	37	39	39	41	40	39	38	39	39	39	39	40	41	43
Fuel Body 3																	
1	2266	1146	921	955	853	817	705	1084	1021	958	909	894	879	777	752	727	604
2	2315	1185	958	992	888	853	739	1123	1059	995	945	930	915	812	787	762	614
3	2365	1207	978	1013	907	875	762	1142	1079	1015	966	951	936	833	808	784	623
4	2415	1199	971	1011	903	875	766	1133	1071	1010	962	948	933	834	811	787	632
5	2465	1188	966	1006	900	876	769	1122	1063	1004	958	944	931	835	812	790	641
6	2515	1174	957	999	895	875	772	1109	1053	996	952	939	926	834	813	792	649
7	2565	1159	947	991	888	872	774	1094	1040	986	944	931	919	832	812	793	657
8	2615	1143	938	982	881	869	775	1077	1026	975	935	924	912	829	811	793	665
9	2664	1124	925	971	874	866	775	1060	1012	964	926	915	905	826	809	792	672
10	2714	1102	913	959	86	863	774	1041	996	951	916	906	896	822	807	791	679
11	2764	1077	899	944	856	858	772	1019	978	937	903	894	885	817	803	789	685
12	2814	1050	888	929	848	853	770	997	960	922	891	883	875	812	799	787	691
13	2864	1016	869	911	831	838	755	972	936	900	870	863	856	796	784	773	696
14	2914	990	846	888	809	816	732	943	907	872	844	837	831	772	761	751	702
Mean	2590	1126	927	968	871	858	760	1065	1014	963	923	911	900	816	798	779	658
Range/±RMS	698	67	38	38	28	20	20	61	51	42	35	33	30	20	19	19	30
Element mean	1799	1089	867	903	798	784	649	1020	961	901	851	836	822	722	696	671	523
Element range/ ±RMS	2278	155	124	129	119	118	114	146	136	127	123	123	122	114	115	117	117

TABLE 7
FTE-4 SPINE SAMPLE TEMPERATURES

Sample Type	Sample I.D.	Mean Core Height (mm)	Temperature Envelope		Linear Weighted Average Temp. (°C)	Exponential Weighted Average Temp.	
			Maximum (°C)	Minimum (°C)		Q = 143,700 J/mole (K)	Q = 267,500 J/mole (K)
Body 1							
Boronated stainless steel	21	678	592	531	553	840	848
Stress relaxation sample	21	735	675	592	630	922	933
Mechanical test assembly	IV	803	768	662	714	1011	1024
Diffusion sample	61	867	841	727	785	1082	1095
Diffusion sample	52	899	873	758	818	1113	1126
Diffusion sample	41	931	903	785	844	1136	1149
Diffusion sample	31	963	932	810	868	1159	1170
Diffusion sample	12	995	959	834	890	1180	1191
Diffusion sample	10	1027	985	857	911	1200	1211
Diffusion sample	4	1059	1009	879	932	1220	1230
Diffusion sample	64	1091	1030	898	951	1239	1249
Diffusion sample	54	1123	1051	917	969	1257	1267
Diffusion sample	46	1155	1069	935	986	1273	1283
Diffusion sample	34	1187	1088	953	1002	1288	1299
Diffusion sample	35	1219	1104	970	1016	1302	1313
Diffusion sample	27	1250	1120	986	1030	1316	1327
Diffusion sample	18	1282	1107	976	1019	1305	1316
Diffusion sample	6	1314	1091	963	1005	1291	1302
Graphite spacer	26	1347	1074	951	991	1277	1287
Body 2							
Graphite spacer	27	1462	1079	956	997	1282	1293
Diffusion sample	58	1510	1133	1007	1049	1335	1346
Diffusion sample	21	1542	1170	1041	1084	1370	1381
Diffusion sample	67	1573	1189	1061	1103	1389	1400
Diffusion sample	28	1605	1196	1069	1111	1397	1408
Diffusion sample	37	1638	1203	1077	1118	1404	1416
Diffusion sample	49	1669	1208	1084	1124	1410	1421
Diffusion sample	22	1701	1215	1089	1129	1416	1427
Advanced fuel rod sample	1-B	1768	1222	1100	1140	1426	1438
Advanced fuel rod sample	1-A	1870	1227	1108	1150	1437	1448
Fission product release sample	NB-2	1937	1227	1111	1153	1441	1453
Fission product release sample	10-P	1969	1227	1112	1155	1443	1455
Fission product release sample	NB-9	2000	1225	1111	1155	1443	1455
Fission product release sample	42-P	2032	1223	1110	1155	1443	1456
Fission product release sample	NB-13	2064	1200	1090	1135	1425	1437
Fission product release sample	146-P	2096	1167	1061	1106	1396	1409
Stress relaxation sample	10	2132	1134	1028	1073	1364	1378
Body 3							
Graphite spacer	29	2243	1123	1011	1059	1351	1366
Thermal stability sample	1-G	2339	1199	1085	1133	1425	1439
Thermal stability sample	1-H	2486	1178	1062	1110	1403	1418
Thermal stability sample	2-16	2574	1151	1033	1082	1376	1392
Thermal stability sample	2-22	2606	1141	1022	1072	1367	1382
Thermal stability sample	2-28	2638	1130	1011	1062	1356	1371
Thermal stability sample	2-4	2669	1117	1001	1051	1374	1359
Thermal stability sample	2-10	2701	1104	991	1039	1332	1346
Stress relaxation sample	45	2733	1088	978	1026	1317	1331
Hollow graphite spacer	28	2800	1053	953	997	1286	1298
Absorber rod sample		2898	988	910	945	1228	1236

TABLE 8
FTE-4 COMPARISON OF PREDICTED AND TIME-AVERAGED W/Re THERMOCOUPLE TEMPERATURES (°C)

TREVER Simulation							Thermocouple Data						Comparison of TREVER and Corrected Thermocouple Temp.			
							W/Re Measured Temp.			W/Re Corrected Temp.						
Real Days		Elapsed Time (days)	EFPD		Temp. TA	Error ±EA ^(a)	N	Mean TB	RMS ±SB	Mean TC	RMS ±SC	Recal. Error ±EC ^(a)	Diff. TA-TC	Error ±ED ^(a)	Rel. Change TA/TC-1	Error ±ER ^(a)
From	To		From	To												
276.12	332.52	56.40	252.00	298.00	1116.0	0.0	7	1057.6	9.3	1064.6	8.6	0.0	51.4	0.00	4.8	0.0
332.52	388.14	55.62	298.00	343.00	1107.0	0.0	6	1044.2	6.4	1064.5	4.9	0.0	42.5	0.00	4.0	0.0
388.14	435.12	46.98	343.00	365.40	1208.0	0.0	4	1057.4	11.4	1092.1	12.5	0.0	115.9	0.00	10.6	0.0
435.12	435.12	0.00	385.40	385.40	1221.0	0.0	1	1052.8	0.0	1091.7	0.0	0.0	129.3	0.00	11.8	0.0
435.12	580.99	145.87	385.40	499.60	1116.0	0.0	20	959.5	57.2	1008.1	62.4	0.0	107.9	0.00	10.7	0.0
580.99	661.58	80.59	499.60	504.10	1128.0	0.0	10	980.7	25.0	1059.0	29.3	0.0	69.0	0.00	6.5	0.0
661.58	713.14	51.56	564.10	610.20	1198.0	0.0	8	958.1	7.1	1049.1	5.7	0.0	148.9	0.00	14.2	0.0
713.14	811.87	98.73	610.20	701.20	1189.0	0.0	13	937.3	18.4	1054.1	38.1	0.0	134.9	0.00	12.8	0.0
Grand Total or Grand Mean		535.75			1146.3	0.0	68	986.1	55.9	1047.3	46.6	0.0	98.9	0.00	0.5	0.0
±RMS					38.5			45.2		26.2			35.1		3.4	

(a) To be determined.

TABLE 9
FTE-4 AVERAGE NUCLIDE WEIGHTS^(a) FOR FUEL RODS, SPINE SAMPLES, AND TOTAL ELEMENT
(in grams)

Nuclide	Atomic Weight (g/mole)	Fuel Rods								Spine Samples		Total Element (With Spine)	
		Holes 1 & 2		Holes 3 & 4		Holes 5 & 6		Holes 7 & 8		BOL	EOL	BOL	EOL
		BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL				
Th-232	232.03807	5.184	5.078	2.549	2.497	2.378	2.329	1.860	1.822	65.830	64.486	1071.394	1049.520
Pa-231	231.03590	--	0.00002	--	0.000010	--	0.000009	--	0.000007	--	0.00026	--	0.00417
U-232	232.03715	--	0.000015	--	0.000007	--	0.000007	--	0.000005	--	0.00019	--	0.00313
U-233	233.03965	--	0.07824	--	0.03847	--	0.03589	--	0.02807	--	0.99357	--	16.170
U-234 ^{bred}	234.04098	--	0.00444	--	0.00218	--	0.00204	--	0.00159	--	0.05637	--	0.917
Th ₀ /Th _i	232.038/232.055	5.184	5.161	2.549	2.538	2.378	2.367	1.860	1.852	65.830	65.536	1071.394	1066.614
U-234 ^{fed}	234.04098	0.008292	0.003286	0.0082935	0.0032867	0.0082935	0.003286	0.0082935	0.003286	0.03930	0.01557	2.82579	1.11980
U-235	235.04394	0.514933	0.314071	0.515026	0.314128	0.515026	0.314128	0.515026	0.314128	2.44053	1.48854	175.48156	107.03070
U-236	236.04558	0.002764	0.040648	0.0027645	0.040612	0.0027645	0.040612	0.0027645	0.040612	0.01310	0.19245	0.94193	13.83750
U-238	238.05082	0.026811	0.029955	0.026816	0.029960	0.026816	0.029960	0.026816	0.029960	0.12707	0.14197	9.13672	10.20812
Np-237	237.04819	--	0.001760	--	0.001761	--	0.001761	--	0.001761	--	0.00834	--	0.59989
Pu-239	239.05218	--	0.000733	--	0.00734	--	0.007734	--	0.000734	--	0.00348	--	0.24995
Pu-240	240.05383	--	0.000205	--	0.00205	00	0.000205	--	0.000205	--	0.00097	--	0.06999
Pu-241	241.05687	--	0.000235	--	0.000235	--	0.000235	--	0.000235	--	0.00111	--	0.07998
Pu-242	242.05877	--	0.000044	--	0.000044	--	0.080044	--	0.000044	--	0.00029	--	0.01500
U ₀ /U _i	235.180/235.416	0.5528	0.3909	0.5529	0.3910	0.5529	0.3910	0.5529	0.3910	2.620	1.853	188.386	133.211
X ₀ /X _i	232.038/232.292	5.7368	5.5519	3.1019	2.9290	2.9309	2.7580	2.4129	2.2430	68.450	67.389	1259.754	1199.825

(a) BOL as-manufactured and loaded values. EOL values are from GAUGE calculations. The total thorium weight at BOL was corrected by +4.26% against GAUGE code calculations. EOL thorium, protactinium, and bred uranium data are adjusted by the same factor. Pa-233 is included in U-233.

TABLE 10
FTE-4 AVERAGE CALCULATED BURNUP

Fuel Rod No.	Core Height (mm)	Fertile FIMA (%)	Fissile FIMA (%)	Mixed FIMA (%)
Body 1				
1	685	0.060	14.207	2.227
2	735	0.123	18.005	2.862
3	785	0.185	21.306	3.421
4	835	0.249	23.862	3.866
5	885	0.311	26.210	4.278
6	934	0.372	28.045	4.611
7	984	0.432	29.755	4.924
8	1034	0.493	30.969	5.161
9	1084	0.551	32.139	5.390
10	1134	0.591	32.916	5.542
11	1184	0.630	33.692	5.694
12	1233	0.658	34.271	5.807
13	1283	0.686	34.850	5.919
14	1333	0.706	35.264	5.999
Mean	1009	0.432	28.249	4.693
Range/±RMS	698	0.210	6.431	1.160
Body 2				
1	1475	0.724	35.659	6.075
2	1525	0.737	35.964	6.133
3	1575	0.749	36.248	6.186
4	1625	0.752	36.362	6.207
5	1675	0.752	36.404	6.213
6	1725	0.738	36.268	6.180
7	1774	0.715	36.014	6.122
8	1824	0.692	35.696	6.054
9	1874	0.669	35.315	5.975
10	1924	0.645	34.917	5.895
11	1974	0.621	34.494	5.810
12	2024	0.596	34.015	5.715
13	2074	0.565	33.408	5.596
14	2123	0.533	32.776	5.472
Mean	1799	0.678	35.253	5.974
Range/±RMS	697	0.072	1.127	0.233
Body 3				
1	2266	0.496	32.039	5.327
2	2315	0.458	31.285	5.179
3	2365	0.417	30.353	5.003
4	2415	0.377	29.421	4.826
5	2465	0.337	28.193	4.604
6	2515	0.297	26.965	4.382
7	2565	0.260	25.737	4.162
8	2615	0.222	24.508	3.942
9	2664	0.183	23.359	3.733
10	2714	0.144	22.230	3.527
11	2764	0.107	21.066	3.318
12	2814	0.073	19.894	3.109
13	2864	0.042	18.773	2.911
14	2914	0.014	17.686	2.721
Mean	2590	0.245	25.108	4.053
Range/±RMS	698	0.152	4.599	0.833
Grand Mean	1799	0.452	29.537	4.907
Grand Range/±RMS	2279	0.236	6.264	1.156

(a) Th/U ratio is 2.54 for total element composition.

TABLE 11
GAUGE CALCULATED TOTAL FIMA (%)

Fuel Rod No.	Axial Core Height (mm)	Stacks 1&2	Stacks 3&4	Stacks 5&6	Stacks 7&8
Body 1					
1	685	1.42	2.58	2.73	3.30
2	735	1.85	3.31	3.50	4.22
3	785	2.22	3.95	4.17	5.02
4	835	2.52	4.46	4.70	5.66
5	885	2.81	4.93	5.20	6.25
6	934	3.04	5.30	5.59	6.71
7	984	3.26	5.66	5.96	7.15
8	1034	3.43	5.93	6.24	7.48
9	1084	3.59	6.18	6.51	7.79
10	1134	3.71	6.35	6.69	8.00
11	1184	3.82	6.52	6.87	8.21
12	1233	3.90	6.65	7.00	8.36
13	1283	3.98	6.76	7.13	8.51
14	1333	4.04	6.87	7.23	8.62
Mean		3.11	5.39	5.68	6.81
RMS		±0.81	±1.32	±1.38	±1.63
Body 2					
1	1475	4.09	6.95	7.31	8.73
2	1525	4.13	7.02	7.38	8.81
3	1575	4.17	7.08	7.45	8.88
4	1625	4.18	7.10	7.47	8.91
5	1675	4.19	7.11	7.48	8.92
6	1725	4.16	7.07	7.44	8.88
7	1774	4.12	7.01	7.37	8.80
8	1824	4.06	6.93	7.30	8.71
9	1874	4.01	6.84	7.20	8.61
10	1924	3.95	6.75	7.11	8.50
11	1974	3.89	6.66	7.01	8.38
12	2024	3.82	6.55	6.90	8.25
13	2074	3.73	6.42	6.76	8.09
14	2123	3.64	6.28	6.62	7.92
Mean		4.01	6.84	7.20	8.60
RMS		±0.17	±0.26	±0.28	±0.31
Body 3					
1	2266	3.54	6.12	6.45	7.72
2	2315	3.43	5.95	6.27	7.52
3	2365	3.30	5.75	6.06	7.28
4	2415	3.18	5.55	5.86	7.03
5	2465	3.02	5.30	5.59	6.72
6	2515	2.87	5.05	5.33	6.41
7	2565	2.71	4.80	5.07	6.10
8	2615	2.56	4.55	4.80	5.79
9	2664	2.42	4.31	4.56	5.49
10	2714	2.27	4.08	4.31	5.20
11	2764	2.13	3.84	4.06	4.91
12	2814	1.98	3.61	3.81	4.61
13	2864	1.85	3.38	3.58	4.33
14	2914	1.72	3.16	3.35	4.06
Mean		2.64	4.68	4.94	5.94
RMS		±0.58	±0.94	±0.99	±1.17
Grand Mean		3.26	5.64	5.94	7.12
RMS		±0.81	±1.31	±1.37	±1.61

TABLE 12
THERMAL FISSION YIELDS, DECAY CONSTANTS, AND UNCERTAINTY LIMITS USED IN BURNUP CALCULATIONS

Nuclear Constants						Uncertainty Limits		
Isotope	Thermal Fission Yield(a)		Decay Constant, λ (sec ⁻¹)	Absolute Intensity(a)	Gamma Ray Energy (keV)	Technique	Attribute	$\pm 1\sigma$ Error (%)
	U-233 (%)	U-235 (%)						
Zr-91	6.52	5.92				Mass spectroscopy (c)	Atom % U-235, U-236, U-238 α_5 (0.242)	Table 14 3.9
Zr-92	6.59	5.96						
Zr-93	7.01	6.37						
Zr-94	6.81	6.42						
Zr-96	5.68	6.25						
Total stable Zr	32.61	30.92				Radiochemistry	Atoms Cs-137 Zr U (EOL) Th (EOL)	5.0 7.2 4.7 2.2
Zr-95	6.25	6.46	1.225×10^{-7}	0.436	724			
Cs-137	6.80	6.27	7.302×10^{-10}	0.846	661			
Cs-134	1.36×10^{-3}	4.5×10^{-5}	1.067×10^{-8}	0.961	604			
Ru-103	1.70	3.14	2.026×10^{-7}	0.890	497			
Ru-106	0.257	0.392	2.174×10^{-8}	0.210	512	IDMS	Atoms Cs-137 U (EOL)	5.0 Table 15
Ce-141	6.58	5.89	2.466×10^{-7}	0.480	145			
Ce-144	4.64	5.46	2.821×10^{-8}	0.030	133			
Pa-233			--	0.0033	312			
La-140(d)	6.43(d)	6.32(d)	4.786×10^{-6}	0.956	1596			
I-131			6.723×10^{-7}	0.820	364	Gamma spectroscopy (e,f) (e) (a)	Cs-137 counting error	Table 48
							Intrinsic photo-peak efficiency of Ge (Li) detector	10.0
							Cs-137 absolute intensity	1.0
							Cs-137 decay constant	2.0
							Cs-137 fission yield	1.0
							Initial U or Th loading	2.5
							Fuel rod length	0.1

(a) Ref. 14.

(b) Ref. 15.

(c) Ref. 16.

(d) Precursor Ba-140 is the direct fission yield isotope, measured through its daughter La-140.

Direct yield La-140 is decayed at the time of gamma scan because of its shorter half life
[$\lambda(\text{La-140}) = 4.786 \cdot 10^{-6} \text{ sec}^{-1}$].

(e) Ref. 17.

(f) Ref. 18.

TABLE 13
FTE-4 SUMMARY OF IDMS AND RADIOCHEMISTRY RESULTS

Sample I.D.	Atoms/Particle x 10 ¹⁶										Fissions/Particle x 10 ¹⁶			
	Cs-137, N _{Cs}	Error (1σ), ±s _{Cs}	Zr, N _{Zr}	Error (1σ), ±s _{Zr}	Total U Radiochem., N _U	Error (1σ), ±s _U	Total U IDMS, N _U [*]	Error (1σ), ±s _U [*]	Total Th , N _{Th}	Error (1σ), ±s _{Th}	Cs-137, N _{Cs} /y _{Cs} ^(a)	Error ±(1σ)	Zr, N _{Zr} /y _{Zr} ^(a)	Error ±(1σ)
Fissile fuel														
UC₂ TRISO (4000-302) (b)														
C1536-1	0.02577	0.0013			1.113	0.093	--	--	--	--	0.411	0.021	--	--
C1536-3	0.03824	0.0019	0.1675	0.0072	1.722	0.145	--	--	--	--	0.611	0.031	0.542	0.023
C1536-4, 5	0.08240	0.0041	0.3115	0.0134	2.888	0.098	--	--	--	--	1.314	0.066	1.007	0.044
C1536-6, 7	0.04159	0.0021	0.1733	0.0075	1.578	0.054	--	--	--	--	0.663	0.033	0.560	0.024
C1536-8, 9, 10	0.09710	0.0049	0.4168	0.0179	3.081	0.105	--	--	--	--	1.549	0.077	1.348	0.058
C2050-2	0.04388	0.0021	--	--	2.261	0.093	--	--	--	--	0.700	0.0350	--	--
C2050-3 F1933	0.04385	0.0022	--	--	2.464	0.101	2.489	0.0045	--	--	0.699	0.0350	--	--
C2050-4 F1937	0.03030	0.0015	--	--	1.760	0.072	1.675	0.0069	--	--	0.483	0.0242	--	--
Fissile fuel														
UO₂ TRISO (4413-67) (b)														
C1538-22	0.04047	0.0020	--	--	1.333	0.112	--	--	--	--	0.646	0.032	--	--
C1538-23	0.03594	0.0018	0.1771	0.0076	1.216	0.102	--	--	--	--	0.573	0.029	0.573	0.025
C1538-24, 25	0.07114	0.0036	0.3218	0.0138	2.426	0.082	--	--	--	--	1.135	0.057	1.041	0.045
C1538-26, 27	0.06888	0.0034	0.3128	0.0135	2.426	0.082	--	--	--	--	1.099	0.055	1.012	0.044
C1538-28, 29, 30	0.09360	0.0047	0.4290	0.0184	3.633	0.124	--	--	--	--	1.493	0.075	1.388	0.060
Fertile fuel														
ThC₂ TRISO (4000-242) (b)														
C1537-11	0.04995	0.0025	--	--	2.993	0.093	--	--	215.9	5.18	0.7346	0.037	--	--
C1537-12	0.04522	0.0023	--	--	2.632	0.082	--	--	181.9	4.37	0.6650	0.033	--	--
C1537-13	0.06373	0.0032	--	--	3.745	0.116	--	--	242.8	5.83	0.9371	0.047	--	--
C1537-14, 15	0.09002	0.0045	--	--	4.519	0.203	--	--	367.1	14.68	1.324	0.066	--	--
C1537-16, 17	0.1044	0.0052	--	--	5.682	0.256	--	--	336.1	13.44	1.536	0.077	--	--
C1537-18, 19	0.1117	0.0056	--	--	6.506	0.293	--	--	373.3	14.93	1.643	0.082	--	--

(a) y - fission yield: y_{Cs-137} = 6.27%; y_{Zr} = 30.92%.

(b) Fuel Materials Branch data retrieval number.

TABLE 14
PTE-4 FISSILE AND FERTILE BURNUP MEASUREMENTS AND COMPARISON WITH CALCULATIONS

Particle Batch No. (a)	Core and Spine Location (b) (mm)	Particle I.D. No. (c)	Subject	Symbol	Mass Spectrometry FIMA (atom %)				IDMS (d) FIMA (atom %)				Radiochemistry (atom %)		Radiochemistry (atom %)		Selected Measured FIMA Meas*(e)	Calculated FIMA GAUGE/FEVER
					ASTM Eq. 1	ASTM Eq. 2	ASTM Eq. 3	Mean	ASTM Eq. 1	ASTM Eq. 2	ASTM Eq. 3	Mean	FIMA Cs-137	FIMA Zr	FIMA Cs-137	FIMA Zr		
Fissile Fuel UO ₂ TRISO																		
4000-302	TS4-5 (2669)	C1536-1	FIMA Error (1σ)	F ₅ S _{F₅}	ND	ND	ND	ND	ND	--	--	--	26.97 ±1.93	--	--	--	--	23.38
		C1536-3	FIMA Error (1σ)	F ₅ S _{F₅}	ND	ND	ND	ND	ND	--	--	--	26.19 ±1.90	23.94 ±1.72	--	--	--	23.38
		C1536-4,5	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	31.27 ±1.30	25.85 ±1.06	--	--	--	23.38
		C1536-6,7	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	29.59 ±1.26	26.19 ±1.06	--	--	--	23.38
		C1536-8,9,10	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	33.46 ±1.34	30.44 ±2.00	--	--	--	23.38
		C2050-2	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	23.64 ±1.17	--	--	--	--	23.38
		C2050-3	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	ND	ND	ND	ND	22.10 ±1.11	--	21.93 ±0.86	--	--	23.38
		C2050-4	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	ND	ND	ND	ND	21.53 ±1.41	--	22.38 ±0.87	--	--	23.38
		Mean																
		FIMA Error(f) (1σ)	F ₅ S _{F₅}	25.56 ±0.32	24.03 ±0.69	25.15 ±0.30	24.91 ±0.49	27.42 ±0.36	24.16 ±0.69	26.57 ±0.33	26.05 ±1.66	26.84 ±1.53	26.61 ±1.41	22.16 ±0.63	--	24.91 ±0.49	23.38	
		Meas Meas* -1 Error (1σ)	μ*	+0.026 ±0.024	-0.035 ±0.034	+0.010 ±0.023	(g) ±0.026	+0.101 ±0.034	-0.030 ±0.025	+0.067 ±0.025	+0.045 ±0.069	+0.077 ±0.065	+0.068 ±0.060	-0.110 ±0.031	--	(g) ±0.018	--	
Calc Meas -1 Error (1σ)	μ	-0.085 ±0.011	-0.027 ±0.028	-0.070 ±0.011	-0.061 ±0.018	-0.147 ±0.012	-0.032 ±0.028	-0.120 ±0.011	-0.102 ±0.057	-0.129 ±0.050	-0.121 ±0.047	+0.055 ±0.030	--	-0.061 ±0.018	--			
Fissile Fuel UO ₂ TRISO																		
4413-67	TS28-6 (2638)	C1538-21	FIMA Error (1σ)	F ₅ S _{F₅}	ND	ND	ND	ND	--	--	--	--	32.64 ±2.14	--	--	--	--	23.76
		C1538-22	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	32.03 ±2.13	32.03 ±2.06	--	--	--	23.76
		C1538-23	FIMA Error (1σ)	F ₅ S _{F₅}	ND	ND	ND	ND	--	--	--	--	31.87 ±1.31	30.00 ±2.10	--	--	--	23.76
		C1538-24,25	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	31.18 ±1.30	29.44 ±1.14	--	--	--	23.76
		C1538-26,27	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	29.13 ±1.25	27.64 ±1.10	--	--	--	23.76
		C1538-28,29,30	FIMA Error (1σ)	F ₅ S _{F₅}	--	--	--	--	--	--	--	--	--	--	--	--	--	23.76
		Mean																
		FIMA Error(f) (1σ)	F ₅ S _{F₅}	26.24 ±0.19	25.10 ±0.74	25.94 ±0.24	25.78 ±0.53	--	--	--	--	31.37 ±0.93	29.78 ±1.14	--	--	25.78 ±0.53	23.76	
		Meas Meas* -1 Error (1σ)	μ*	+0.018 ±0.022	-0.026 ±0.035	+0.006 ±0.023	(g) ±0.023	--	--	--	--	+0.217 ±0.044	+0.155 ±0.050	--	--	(g) ±0.019	--	
Calc Meas -1 Error (1σ)	μ	-0.095 ±0.007	-0.053 ±0.028	-0.084 ±0.008	-0.078 ±0.019	--	--	--	--	-0.243 ±0.022	-0.202 ±0.031	--	--	-0.078 ±0.019	--			
Fertile Fuel ThO ₂ TRISO																		
4000-242	TS16-3 (2574)	C1537-11	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.33 ±0.02	--	--	--	--	0.28
		C1537-12	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.36 ±0.02	--	--	--	--	0.28
		C1537-13	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.38 ±0.02	--	--	--	--	0.28
		C1537-14,15	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.36 ±0.02	--	--	--	--	0.28
		C1537-16,17	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.45 ±0.03	--	--	--	--	0.28
		C1537-18,19	FIMA Error (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.43 ±0.03	--	--	--	--	0.28
		Mean																
		FIMA Error(f) (1σ)	F ₃ S _{F₃}	--	--	--	--	--	--	--	--	0.39 ±0.06	--	--	--	0.39 ±0.06	0.28	
		Meas Meas* -1 Error	μ*	--	--	--	--	--	--	--	--	--	--	--	--	--	--	
Calc Meas -1 Error	μ	--	--	--	--	--	--	--	--	-0.28 ±0.10	--	--	--	-0.28 ±0.10	--			

(a) Fuel Material Branch data retrieval number.

(b) TS6-5 = thermal stability type II, crucible 6, hole 5.

(c) C1536-1 = hot cell "C" bottle No. C1536, particle No. 1.

(d) IDMS = isotope dilution mass spectroscopy.

(e) Fissile FIMA = average of mass spectroscopy and IDMS; fertile FIMA = radiochemistry with Cs-137 monitor.

(f) $S_{\bar{F}_5} = \left[\frac{1}{n} \sum s_i^2 + \frac{1}{n} \sum (F_5 - \bar{F}_5)^2 \right]^{1/2}$

(g) Normalizer point.

TABLE 15
FTE-4 COMPARISON OF MASS SPECTROMETRIC AND CALCULATED ISOTOPIC CONCENTRATIONS

Particle Batch No. (a)	Sample No. (b)	Analysis Procedure	Relative Atomic Concentration (atom %)										
			U-233		U-234		U-235		U-236		U-238		
			N ₃	±S ₃ (1σ)	N ₄	±S ₄ (1σ)	N ₅	±S ₅ (1σ)	N ₆	±S ₆ (1σ)	N ₈	±S ₈ (1σ)	
UC ₂ TRISO Fissile													
4000-302 TS4-5	Archive	Mass spec. BOL	--	--	0.794	±0.006	93.199	±0.015	0.275	±0.003	5.731	±0.014	
		GAUGE BOL	--	--	0.740	--	93.150	--	0.270	--	5.840	--	
		GAUGE/FEVER EOL ^(c)	--	--	0.727	--	84.766	--	7.464	--	7.054	--	
	C1536-1	Mass spec.	--	--	0.885	±0.0024	84.08	±0.020	7.83	±0.010	7.21	±0.019	
		C1536-3	Mass spec.	--	--	0.886	±0.005	84.13	±0.050	7.84	±0.040	7.15	±0.03
		Mean mass spec.	--	--	0.885	±0.0028	84.10	±0.032	7.84	±0.021	7.18	±0.028	
	C2050-3	(Calc/Meas - 1) μ	--	--	-0.179	--	+0.008	--	-0.048	--	-0.018	--	
		Error (1σ) S _μ	--	--	±0.0026	--	±0.0004	--	±0.0026	--	±0.0038	--	
		C2050-4	IDMS ^(d)	--	--	0.846	±0.046	83.84	±0.074	7.95	±0.046	7.36	±0.046
			IDMS	--	--	0.847	±0.010	83.89	±0.22	7.89	±0.043	7.38	±0.044
			Mean IDMS	--	--	0.847	±0.024	83.87	±0.117	7.92	±0.037	7.37	±0.033
			(Calc/Meas - 1) μ	--	--	-0.142	--	-0.011	--	-0.058	--	-0.043	--
			Error (1σ) S _μ	--	--	±0.024	--	±0.0014	--	±0.0044	--	±0.0043	--
UO ₂ TRISO Fissile													
4413-67 TS28-6	Archive	Mass spec. BOL	--	--	0.994	±0.009	92.948	±0.015	0.503	±0.004	5.555	±0.012	
		GAUGE BOL	--	--	0.740	--	93.15	--	0.270	--	5.840	--	
		GAUGE/FEVER EOL ^(c)	ND	--	0.936	--	84.184	--	8.008	--	6.871	--	
	C1538-21	Mass spec.	0.0077	±0.0028	1.108	±0.004	83.18	±0.023	8.61	±0.019	7.089	±0.016	
	C1538-23	Mass spec.	--	--	1.103	±0.007	83.19	±0.03	8.62	±0.03	7.092	±0.018	
		Mean mass spec.	0.0077	±0.0028	1.106	±0.0044	83.19	±0.019	8.62	±0.018	7.091	±0.012	
		(Calc/Meas - 1) μ	ND	--	-0.154	--	+0.012	--	-0.071	--	-0.031	--	
		Error (1σ) S _μ	--	--	±0.0034	--	±0.0002	--	±0.0019	--	±0.0016	--	
ThC ₂ TRISO Fertile													
4000-242 TS16-3		GAUGE/FEVER EOL	94.388	--	5.111	--	0.501	--	--	--	--	--	
	C1537-11	Mass spec.	95.345	±0.027	4.173	±0.026	0.402	±0.0015	0.01902	±0.00018	0.0607	±0.0007	
		C1537-13	Mass spec.	95.431	±0.023	4.192	±0.022	0.326	±0.0013	0.01149	±0.00009	0.0400	±0.0004
		Mean mass spec.	95.388	±0.035	4.183	±0.018	0.364	±0.027	0.01526	±0.0027	0.0504	±0.0073	
		(Calc/Meas - 1) μ	-0.010	--	+0.222	--	+0.376	--	ND	--	ND	--	
		Error (1σ) S _μ	±0.00036	--	±0.0053	--	±0.102	--	--	--	--	--	

(a) Fuel Materials Branch data retrieval number.

(b) Hot-cell "C" bottle number - particle number.

(c) Corrected for as-measured BOL concentrations.

(d) IDMS - isotope dilution mass spectrometry.

TABLE 16
FTE-4 ISOTOPIC DILUTION MASS SPECTROMETRY RESULTS

Particle Batch No. (a)	Spine and Core Location	Sample I.D. (b)	Absolute Atomic Concentration (ppm)										Total U	Error (c) $\pm(1\sigma)$
			U-233	Error $\pm(1\sigma)$	U-234	Error $\pm(1\sigma)$	U-235	Error $\pm(1\sigma)$	U-236	Error $\pm(1\sigma)$	U-238	Error $\pm(1\sigma)$		
4000-302	TS4-5	C2050-3	--	--	0.00702	0.00004	0.6958	0.0014	0.0660	0.0004	0.0611	0.0004	0.8299	0.0015
4000-302	(2669)	C2050-4	--	--	0.00473	0.00006	0.4686	0.0023	0.0441	0.0003	0.0412	0.0003	0.5586	0.0023

(a) Fuel Materials Branch data retrieval number.

(b) Hot cell "C" bottle number - particle number.

(c) $du_T^2 = \sum du_i^2$.

TABLE 17
FTE-4 COMPARISON OF GAMMA-SPECTROSCOPY-MEASURED
AND GAUGE-CALCULATED FIMA VALUES

Rod I.D.	Core Height (mm)	Gamma Spectroscopy (%)		GAUGE-Calculated Total FIMA (%)	Loading Factor, X ^(a)
		FIMA	±2σ Error		
1-3-5	885	4.92	1.02	4.93	0.1782
1-5-5	885	5.05	1.05	5.20	0.1886
1-3-6	934	5.24	1.09	5.30	0.1782
1-5-6	934	4.63	0.96	5.59	0.1886
1-3-7	984	5.62	1.17	5.66	0.1782
1-5-7	984	6.18	1.28	5.96	0.1886
1-7-7	984	7.32	1.51	7.15	0.2291
2-3-3	1575	7.92	1.64	7.08	0.1782
2-3-4	1625	8.48	1.79	7.10	0.1782
2-3-5	1675	8.43	1.78	7.11	0.1782
2-4-5	1675	8.65	1.79	7.11	0.1782
2-5-5	1675	9.11	1.88	7.48	0.1886
2-3-6	1725	7.53	1.56	7.07	0.1782
2-4-6	1725	8.57	1.77	7.07	0.1782
2-5-6	1725	9.20	1.90	7.44	0.1782
2-7-6	1725	9.47	1.96	8.88	0.2291
2-8-6	1725	10.40	2.15	8.88	0.2291
2-3-7	1774	8.26	1.71	7.01	0.1782
2-4-7	1774	8.63	1.78	7.01	0.1782
2-7-7	1774	9.52	1.97	8.80	0.2291
2-4-8	1824	8.71	1.80	6.93	0.1782
2-5-8	1824	8.97	1.86	7.30	0.1886
2-6-8	1824	8.78	1.82	7.30	0.1886
2-7-8	1824	10.00	2.07	8.71	0.2291
2-8-8	1824	10.06	2.08	8.71	0.2291
2-4-9	1874	8.17	1.69	6.84	0.1782
2-7-9	1874	9.68	2.00	8.61	0.2291
2-8-9	1874	12.41	2.57	8.61	0.2291
2-4-10	1924	8.19	1.69	6.75	0.1782
2-8-10	1924	10.64	2.20	8.50	0.2291
2-8-11	1974	9.92	2.05	8.38	0.2291
3-3-5	2465	6.53	1.35	5.30	0.1782
3-5-5	2465	6.82	1.41	5.59	0.1886
3-3-6	2515	6.77	1.40	5.05	0.1782
3-1-7	2565	10.94 ^(b)	--	2.71	0.0964
3-3-7	2565	6.14	1.27	4.80	0.1782
3-7-7	2565	6.67	1.38	6.10	0.2291
3-1-8	2615	4.11 ^(b)	--	2.56	0.0964
3-3-8	2615	5.57	1.15	4.55	0.1782
3-7-8	2615	6.76	1.40	5.79	0.2292

(a) Total FIMA = X(FIMA)_{Fissile} + (1 - X) (FIMA)_{Fertile};
X = U/(U + Th) (atomic ratio).

(b) ORNL data. No error available.

TABLE 18
FTE-4 COMPARISON OF BURNUP MEASUREMENTS AND CALCULATED VALUES

Sample I.D.	Core Location (mm)	Radiochemistry										Mass Spectroscopy + IDMS							Gamma Spectroscopy FIMA (%)		GAUGE-Calculated FIMA (%)		
		Fissile FIMA				Fertile FIMA		Interpolated FIMA (%)				Fissile FIMA		Interpolated FIMA (%)				Total	Error	Fissile	Fertile	Total(c)	
		$\bar{F}_5$	1 σ	$\bar{F}_5$	1 σ	$\bar{F}_5$	1 σ	Fissile(a)	Fertile(b)	Total(c)	Loading Factor (x)	%	1 σ	Fissile(d)	Fertile(b)	Total(c)	Loading Factor (x)						
		Cs-137	Error	Zr	Error	Cs-137	Error					%	Error										Fissile(d)
Comparison with Stack 3 Fuel Rods																							
Rod 3-3-7	2565	--	--	--	--	--	--	41.77	0.40	7.77	--	--	--	27.83	0.40	5.29	--	6.14	±0.65	25.74	0.26	4.80	
TS16-3	2574	--	--	--	--	0.39	±0.055	40.39	0.39	7.52	0.1782	--	--	27.58	0.39	5.23	0.1782	--	--	--	--	--	
Rod 3-3-8	2615	--	--	--	--	--	--	34.52	0.34	6.43	--	--	--	26.42	0.34	4.99	--	5.57	±0.59	24.51	0.22	4.55	
TS28-6	2638	31.37	±0.93	29.78	±1.14	--	--	31.37	0.31	5.84	0.1782	25.78	±0.53	25.78	0.31	4.85	0.1782	--	--	--	--	--	
Rod 3-3-9	2664	--	--	--	--	--	--	27.27	0.28	5.09	--	--	--	25.05	0.28	4.69	--	--	--	23.36	0.18	4.31	
TS4-5	2669	26.84	±1.53	26.61	±1.41	--	--	26.84	0.27	5.00	0.1782	24.91	±0.49	24.91	0.27	4.66	0.1782	--	--	--	--	--	
Comparison with Stack 7 Fuel Rods																							
Rod 3-7-7	2565	--	--	--	--	--	--	41.77	0.40	9.88	--	--	--	27.83	0.40	6.68	--	6.08	±0.70	25.74	0.26	6.10	
TS16-3	2574	--	--	--	--	0.39	±0.055	40.39	0.39	9.55	0.2291	--	--	27.58	0.39	6.62	0.2291	--	--	--	--	--	
Rod 3-7-8	2615	--	--	--	--	--	--	34.52	0.34	8.17	--	--	--	26.42	0.34	6.31	--	6.76	±0.71	24.51	0.22	5.79	
TS28-6	2638	31.37	±0.93	29.78	±1.14	--	--	31.37	0.31	7.42	0.2291	25.78	±0.53	25.78	0.31	6.15	0.2291	--	--	--	--	--	
Rod 3-9-9	2664	--	--	--	--	--	--	27.27	0.28	6.46	--	--	--	25.05	0.28	5.95	--	--	--	23.36	0.18	5.49	
TS4-5	2669	26.84	±1.53	26.61	±1.41	--	--	26.84	0.27	6.36	0.2291	24.91	±0.49	24.91	0.27	5.92	0.2291	--	--	--	--	--	

- (a) Interpolated from radiochemistry fissile FIMA values from TS28-6 and TS4-5.
 (b) Interpolated from GAUGE calculated values and radiochemistry measured value at TS16-3.
 (c) $FIMA_{total} = x(FIMA)_{fissile} + (1 - x)(FIMA)_{fertile}$, where $x = \frac{U}{(U + Th)}$ (atomic ratio).
 (d) Interpolated from mass spectroscopy fissile FIMA values from TS28-6 and TS7-5.

Fig. 4. Peach Bottom core 2 operating parameters, TREVER simulation

Fig. 5a. Peach Bottom core 2 1970 power history from thermal balance

Fig. 5b. Peach Bottom core 2 1971 power history from thermal balance

Fig. 5c. Peach Bottom core 2 1972 power history from thermal balance

Fig. 5d. Peach Bottom core 2 1973 power history from thermal balance

Fig. 5e. Peach Bottom core 2 1974 power history from thermal balance

Fig. 6. Peach Bottom core 2 thermal power history, TREVER simulation

Fig. 7. FTE-4 power history, TREVER simulation

Fig. 8. Normalized axial power distribution for Peach Bottom

Fig. 9a. Normalized axial fast flux distribution for Peach Bottom

Fig. 9b. Normalized axial thermal flux distribution for Peach Bottom

Fig. 10a. FTE-4 axial temperature profiles (BOL-298 EFPD)

Fig. 10b. FTE-4 axial temperature profiles (298-343 EFPD)

Fig. 10c. FTE-4 axial temperature profiles (343-385 EFPD)

Fig. 10d. FTE-4 axial temperature profiles (385-499 EFPD)

Fig. 10e. FTE-4 axial temperature profiles (499-564 EFPD)

Fig. 10f. FTE-4 axial temperature profiles (564-610 EFPD)

Fig. 10g. FTE-4 axial temperature profiles (610-EOL)

Fig. 10h. FTE-4 axial temperature profiles (EOL)

Fig. 11. Radial temperature distribution at the point of axial peak temperature in a typical test element

Fig. 12a. FTE-4 calculated radial temperature history for thermocouple location of 49.5 in. (1257.3 mm) active core height

Fig. 12b. FTE-4 measured and calculated temperature history

Fig. 13a. Peach Bottom normalized axial distribution for Th-232 from FEVER calculations

Fig. 13b. Peach Bottom normalized axial distribution for U-233 from FEVER calculations

Fig. 13c. Peach Bottom normalized axial distribution for U-235 from FEVER calculations

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Fig. 13d. Peach Bottom normalized axial distribution for U-238 from FEVER calculations

Fig. 14. FTE-4 fuel hole 3 and 4, comparison of measured and calculated composite burnup

Fig. 15. FTE-4 fuel hole 5 and 6, comparison of measured and calculated composite burnup

Fig. 16. FTE-4 fuel hole 7 and 8, comparison of measured and calculated composite burnup

4. POSTIRRADIATION EXAMINATION AND EVALUATION

4.1. ERROR ANALYSIS FOR PREIRRADIATION AND POSTIRRADIATION EXAMINATION

A complete derivation of error analysis methods is included in this report so that a reference model will be available for the examination and reporting of later test elements.

4.1.1. Analysis Definition

Irradiation experiments are directed toward the determination of irradiation-induced changes. Therefore, preirradiation and postirradiation measurements are compared by means of the following function:

$$u = \frac{\Delta x}{x_1} = \frac{x_2 - x_1}{x_1} = \frac{x_2}{x_1} - 1, \quad (55)$$

where $x_1 \equiv$ measurement prior to irradiation (index 1),

$x_2 \equiv$ measurement after irradiation (index 2).

In this case the irradiation-induced strain is derived from one set of data, e.g., fuel rod diameter. The mean strain can be determined in two ways:

$$\bar{u} = \frac{\frac{1}{n_2} \sum_{i=1}^{n_2} x_{2,i}}{\frac{1}{n_1} \sum_{i=1}^{n_1} x_{1,i}} - 1 = \frac{\bar{x}_2}{\bar{x}_1} - 1 \quad \text{(procedure 1, preferred method)}, \quad (56)$$

$$\bar{u}' = \frac{1}{n} \sum_i^n \left(\frac{x_2}{x_1} \right)_i - 1 \quad \text{(procedure 2, for comparison only)} \quad (57)$$

where $n = n_1 = n_2$ = number of measurements before and after irradiation
 ($n_1 \neq n_2$ is also possible for Eq. 56),
 i = measurement index.

It has been shown by Leary (Ref. 22) that a mean value $\bar{u}$ calculated using Eq. 56 is superior to one calculated using Eq. 57 in that the effect of the systematic error is smaller.

Equation 56 (procedure 1) represents the case of comparing two statistical means, e.g., from a representative sample taken prior to and an independent sample taken after irradiation. This is believed to be a good procedure where irradiation conditions are relatively uniform and the population from which the samples can be drawn is large.

Whenever individual measurements are available, a comparison of the specific measurements can be made by applying Eq. 55 and a mean value of a strata can then be determined using Eq. 57 (procedure 2).

A further method of stratification is to determine, e.g., for a specific fuel rod, the irradiation strain for each of the top, middle, and bottom end locations separately using Eq. 56 (or Eq. 57 when more than one measurement is available) in order to identify eventual local effects and also to determine a sample mean, in this case for the total fuel rod, using Eq. 56. If additional grouping is justified because of, for example, uniform irradiation conditions, Eq. 56 can be further applied. This third procedure has been used in this report.

In Section 4.1.3.5, several strata and their associated variances are combined into one grand mean and a combined variance is presented.

A different method of analysis is to combine two sets of data in one mean value, e.g., dimensional strain in the radial x and axial y directions, via arithmetic average:

$$\bar{v} = \frac{1}{2} \left(\frac{\bar{x}_2}{\bar{x}_1} + \frac{\bar{y}_2}{\bar{y}_1} - 2 \right) . \quad (58)$$

A third method of analysis determines the anisotropy by comparing axial with radial strains:

$$\bar{w} = \left(\frac{\bar{y}_2}{\bar{y}_1} - 1 \right) / \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right) . \quad (59)$$

In case of total disorientation, the anisotropy equals 1.

4.1.2. Error Definition and Determination

Any measurement has the following error terms:

$$x_i = \mu + r_{s,i} + r_{p,i} + c + r_c , \quad (60)$$

where $x_i = i^{\text{th}}$ measured value,

μ = true value,

$r_{s,i}$ = random error due to sample variation,

$r_{p,i}$ = random error due to equipment, operator or other variation,
and precision limits,

c = bias of measurement from calibration,

r_c = random error due to uncertainty in c .

The combination of all random variances gives the "noise" or the square of the total standard deviation on x_i :

$$\sigma_{x_i}^2 = \sigma_{r_{s,i}}^2 + \sigma_{r_{p,i}}^2 + \sigma_{r_c}^2 , \quad (61)$$

where $\sigma_{r_{s,i}}$ = standard deviation of $r_{s,i}$,
 $\sigma_{r_{p,i}}$ = standard deviation of $r_{p,i}$,
 σ_{r_c} = standard deviation of r_c .

Sixty-seven percent confidence limits (1σ) are constructed using

$$\mu = x_i \pm \sigma_{x_i} \quad (62)$$

and 95% confidence limits (2σ) using

$$\mu = x_i \pm 1.96 \sigma_{x_i} \quad (63)$$

The different error terms have to be determined or estimated. This is done as described below:

1. Systematic Error c

This is a correction term common to all measurements and could be a bias, determined by repeated measurement of the calibration standard. It could be associated with the use of two different modes of measurement techniques applied to the same subject, as in the case of air gauging and dial gauging, where, according to Ref. 23, air gauging produces a systematical error of -3 mils (0.076 mm) (because surface roughness effects are accounted for by air gauging) with a standard deviation of 0.6 mil (0.015 mm). The variation could be called equipment random error r_c .

The systematic error can be accounted for before any further data reduction and is therefore a correction term which does not need any further attention.

2. Precision e

Limitation in the readout system, i.e., the finest graduation on the scale, determines the precision of one measurement. For hot cell operations it is postulated that the precision is determined on the one hand by the affiliation of a certain measurement to a calibration standard. Normally, secondary standards are used to "zero" dial gauges, with which a certain object is then compared. Based on this it is estimated that the precision is at least 3 times as high as the precision of the calibration standard involved. On the other hand, a precision of less than 0.0001 in. (0.0025 mm) is believed not to be achievable with standard hot cell techniques:

$$e \geq 0.0001 \text{ in. (0.0025 mm)} \quad (64)$$

Precision can be described as the confidence associated with the last digit in a measurement.

3. Equipment Random Error $r = r_c$

This is a random error associated with the calibration standard, which may have its own sample random variation, e.g., because of manufacturing limitations. It may also be measured only a finite number of times, and there could be some spread of these measurements. A conservative estimate is

$$r = r_c \cong s_c + 3e, \quad (65)$$

where s_c = sample random error of calibration standard,
 e = precision of measurement apparatus.

Another way to estimate r would be to measure a calibration standard repeatedly, say n_c times. Let $\hat{s}_c$ be the standard deviation of these n_c measurements. Then the estimate would be

$$\sigma_{r_c} \cong \sqrt{\hat{s}_c^2 / n_c} ; \quad (66)$$

i.e., the standard deviation of the mean of n_c measurements is assumed to be a 67% confidence estimate of the equipment random error r .

4. Sample Random Error s

This random error is associated with the sample's irregularities, undulations, etc., and is caused by material, process, and other product-related variations. Its standard deviations can be calculated by

$$s^2 = \frac{1}{n-1} \left(\sum_{i=1}^n x_i^2 - n \cdot \bar{x}^2 \right) , \quad (67)$$

where n = number of measurements.

5. Combined Sample Random Error $\hat{s}$

Equation 64 describes, in addition to the sample random variation $r_{s,i}$, an additional random error term $r_{p,i}$ due to measuring apparatus variations, operator variations, temperature changes, etc., which also includes precision limits. An estimate for the combination of both standard deviations is given by

$$\hat{s}^2 = \sigma_{r_{s,i}}^2 + \sigma_{r_{p,i}}^2 = \frac{s^2}{n} + \frac{e^2}{12n} , \quad (68)$$

where the variance on the precision e is determined by the assumption that the operator reads to the nearest scale division e , and that the true value should be within $\pm 1/2 e$ of the scale mark,

ignoring other errors. This true value will be uniformly distributed in this interval, so the variance of the precision is given by

$$r_e^2 \equiv \int_{-e/2}^{+e/2} x^2 \cdot \frac{1}{e} dx = \frac{e^2}{12} \quad (69)$$

6. Combined Random Error ϵ

The combined random error is defined as the estimate of standard deviation from x_1 in Eq. 60:

$$\epsilon = \pm \left(r^2 + \frac{s^2}{n} + \frac{e^2}{12n} \right)^{1/2} \quad (70)$$

7. Total Error dx

The total error dx is bounded with 95% confidence by the limits

$$dx = \pm 1.96\epsilon = \pm 1.96 \left(r^2 + \frac{s^2}{n} + \frac{e^2}{12n} \right)^{1/2} \quad (71)$$

8. Accuracy α

Accuracy may be defined as a 95% confidence statement on the error of the measurement system obtained from the sum of the precision plus 2 (more precisely 1.96) standard deviations of the equipment random error:

$$\alpha \cong \pm(e + 2r) \quad (72)$$

9. Tolerances τ

Tolerances are assumed to be a 3σ (3 standard deviations representing 99% confidence) statement on the equipment random error:

$$\tau \cong \pm 3r \quad .$$

(73)

Accuracy and tolerance are identical for

$$e = r \quad .$$

Nominal values and tolerance statements are occasionally the only information available as preirradiation data (examples are the sleeve inner and outer diameter measurements on Peach Bottom elements; see Table 19). An error approximation can be constructed through the above assumptions.

Table 19 summarizes the error estimates for several kinds of dimensional measurements carried out on Peach Bottom elements. In cases where preirradiation measurements are lacking, the same error as for postirradiation measurements is assumed (and vice versa). This is a conservative estimate for the first case, as shown by the typical values for FTE-4 in Table 19.

4.1.3. Error Combination

4.1.3.1. Combining One Set of Data. Irradiation experiments are evaluated on the basis of irradiation-induced changes; therefore, errors from preirradiation and postirradiation must be combined. Usually, the total error of a specific result

$$u = f(x_1, x_2 \dots) \quad ,$$

which is determined from several single measurements $x_1, x_2 \dots$ with the errors $dx_1, dx_2 \dots$, is defined by its partial derivatives (error progression law):

$$du = dx_1 \frac{\partial u}{\partial x_1} + dx_2 \frac{\partial u}{\partial x_2} + \dots \quad . \quad (74)$$

In the case of one set of data (Eq. 55) with the errors e_1 and e_2 , the following total error is obtained:

$$e_u = -e_1 \frac{x_2}{x_1^2} + e_2 \frac{1}{x_1} .$$

When the variable x_j is estimated by the mean of its measured value $\bar{x}_j$, the result is

$$e_u \cong \pm \frac{1}{\bar{x}_1} \left(e_1 \frac{\bar{x}_2}{\bar{x}_1} + e_2 \right) . \quad (75)$$

If the variables $x_1, x_2 \dots$ are independent, the standard deviation of e_u is seen from Eq. 59 to be

$$\epsilon_u = \pm \sqrt{\left(\epsilon_1 \frac{\partial u}{\partial x_1} \right)^2 + \left(\epsilon_2 \frac{\partial u}{\partial x_2} \right)^2 + \dots} . \quad (76)$$

This, combined with Eq. 71, gives

$$\epsilon_u = \pm \frac{1}{\bar{x}_1} \left[\left(r_1^2 + \frac{s_1^2}{n_1} + \frac{e_1^2}{12 n_1} \right) \left(\frac{\bar{x}_2}{\bar{x}_1} \right)^2 + r_2^2 + \frac{s_2^2}{n_2} + \frac{e_2^2}{12 n_2} \right]^{1/2} . \quad (77)$$

For the specific case where $r = r_1 = r_2$, $n = n_1 = n_2$, $e = e_1 = e_2$, and $s = s_1 = s_2$,

$$\epsilon_u = \pm \frac{1}{\bar{x}_1} \left\{ \left[\left(\frac{\bar{x}_2}{\bar{x}_1} \right)^2 + 1 \right] \left(r^2 + \frac{s^2}{n} + \frac{e^2}{12} \right) \right\}^{1/2} . \quad (78)$$

The total (2 σ) error obtained using Eq. 63 is

$$du = \pm 1.96 \cdot \epsilon_u .$$

A formula summary is given in Table 20.

4.1.3.2. Combining Several Sets of Data. In cases where errors have to be determined from several different kinds of measurements, as in the case of fuel stack length determination via plenum measurements, the error combination formulas have to include the sum of the individual error terms.

Precision

$$e_u = \pm \frac{1}{\bar{x}_1} \left[\Sigma e_1 (\bar{u} + 1) + \Sigma e_2 \right] \quad (79)$$

Equipment Random Error

$$r_u = \pm \frac{1}{\bar{x}_1} \left[\Sigma r_1^2 (\bar{u} + 1)^2 + \Sigma r_2^2 \right]^{1/2} \quad (80)$$

Sample Random Error

$$s = \pm \frac{1}{\bar{x}_1} \left[\frac{\Sigma s_1^2}{n_1} (\bar{u} + 1)^2 + \frac{\Sigma s_2^2}{n_2} \right]^{1/2} \quad (81)$$

Combined Random Error

$$\epsilon_u = \frac{1}{\bar{x}_1} \left[\left(\Sigma r^2 + \frac{\Sigma s_1^2}{n_1} + \frac{\Sigma e_1^2}{12 n_1} \right) (\bar{u} + 1)^2 + \Sigma r_2^2 + \frac{\Sigma s_2^2}{n_2} + \frac{\Sigma e_2^2}{12 n_2} \right]^{1/2} \quad (82)$$

Equations 79 through 82 will have to be further modified for $n_{1,i} \neq \text{const.}$ and $n_{2,i} \neq \text{const.}$ as indicated in Table 20.

4.1.3.3. Combining Sets of Data by Averaging. When two or more sets of measurements are combined in one expression, as for instance by arithmetically

averaging the fuel rod length x and diameter y in one strain figure via Eq. 58, the errors must be combined accordingly.

The precision follows from the error progression law (Eq. 74):

$$e_v = \pm \frac{1}{2} (e_x + e_y) \quad ; \quad (83)$$

e_x and e_y are determined using Eq. 75.

The combination of the random errors follows from Eq. 76:

$$\epsilon_v = \pm \frac{1}{2} \left(\epsilon_x^2 + \epsilon_y^2 \right)^{1/2} \quad ; \quad (84)$$

ϵ_x and ϵ_y are determined using Eq. 77.

A different method of combination is a weighting average form, which may be considered especially when n_x and n_y differ significantly:

$$\epsilon_v = \pm \left[\frac{\epsilon_x^2 \cdot n_x^2 + \epsilon_y^2 \cdot n_y^2}{(n_x + n_y)^2} \right]^{1/2} \quad . \quad (85)$$

In this specific case, the analysis function (Eq. 58) would have to read as follows:

$$\bar{v} = \frac{\bar{x}_1}{\bar{x}_2} \cdot \left(\frac{n_x}{n_x + n_y} \right) + \frac{\bar{y}_1}{\bar{y}_2} \left(\frac{n_y}{n_x + n_y} \right) - 1 \quad . \quad (86)$$

An error analysis formula summary is given in Table 20.

4.1.3.4. Combining Two Sets of Data by Ratio. The analysis function $\bar{w}$ for anisotropy (Eq. 59) results from Eq. 74 in the following precision:

$$e_w = \pm \left(e_y + e_x \cdot \bar{w} \right) \cdot \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} ; \quad (87)$$

e_y and e_x have to be determined using Eq. 76.

In case of identical errors $e = e_{y_1} = e_{y_2} = e_{x_1} = e_{x_2}$,

$$e_w = \pm e (1 + \bar{w}) \cdot \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} ; \quad (88)$$

and for isotropy with $\bar{y}_2/\bar{y}_1 = \bar{x}_2/\bar{x}_1$,

$$e_w = \pm 2e \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} . \quad (89)$$

The random errors follow from Eq. 70:

$$e_w = \pm \sqrt{\varepsilon_y^2 + \varepsilon_x^2 \bar{w}^2} \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} . \quad (90)$$

ε_y and ε_x must be calculated using Eq. 71. For identical errors with $\varepsilon = \varepsilon_{y_1} = \varepsilon_{y_2} = \varepsilon_{x_1} = \varepsilon_{x_2}$, the result simplifies to

$$\varepsilon_w = \pm \varepsilon \sqrt{1 + \bar{w}^2} \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} ; \quad (91)$$

and in the case of isotropy with $\bar{y}_2/\bar{y}_1 = \bar{x}_2/\bar{x}_1$,

$$\varepsilon_w = \pm \varepsilon \sqrt{2} \left(\frac{\bar{x}_2}{\bar{x}_1} - 1 \right)^{-1} . \quad (92)$$

A formula summary for this error analysis is given in Table 20.

4.1.3.5. Combining Strata Into Grand Mean. When Combining j strata of n_j elements with the strata mean x_j and standard deviation s_j into one grand mean, the calculations are equivalent to averaging all associated measurements and to determining their standard deviation. The grand mean is calculated in the following way for

$$n_j \neq \text{const.} \quad ,$$

$$\bar{x} = \frac{1}{\Sigma n_j} \Sigma (n_j \cdot x_j) \quad : \quad (93)$$

$$s^2 = \frac{1}{\Sigma n_j} \left\{ \Sigma (n_j \cdot s_j^2) + \Sigma \left[n_j \cdot (x_j - \bar{x})^2 \right] \right\} \quad . \quad (94)$$

The term $n_j / \Sigma n_j$ is the weighing factor for mean x_j and standard deviation s_j . In the case of constant elements per strata, the mean and standard deviation are determined with

$$n_j = k = \text{const.} \quad ,$$

and

$$\Sigma n_j = j \cdot k \quad ,$$

$$\bar{x} = \frac{1}{j} \Sigma x_j \quad , \quad (95)$$

$$s^2 = \frac{1}{j} \Sigma s_j^2 + \hat{s}^2 \quad , \quad (96)$$

where $\hat{s}$ is the standard deviation of all x_j ,

$$\hat{s}^2 = \frac{1}{j} \Sigma (x_j - \bar{x})^2 \quad . \quad (97)$$

Equations 94, 96, and 97 assume that Σn_j or $j \gg 1$; i.e., sampling is approaching total population.

In other cases, sampling is very limited and covers only partial population. This must be accounted for as follows:

$$s_{\text{total}}^2 = s_{\text{partial}}^2 \cdot \frac{n - 1}{n} \quad , \quad (98)$$

where n = total number of measurements (Σn_j or j).

By applying Eq. 98 to Eq. 94, the standard deviation is then calculated for

$$n_j \neq \text{const.} \quad ,$$

$$s_{\text{partial}}^2 = \frac{1}{\Sigma n_j - 1} \left\{ \Sigma \left[(n_j - 1) s_{\text{partial}}^2 \right] + \Sigma \left[n_j (x_j - \bar{x})^2 \right] \right\} \quad . (99)$$

With Eq. 93, the last term can be modified, giving

$$s_{\text{partial}}^2 = \frac{1}{\Sigma n_j - 1} \left\{ \Sigma \left[(n_j - 1) s_{j,\text{partial}}^2 \right] + \Sigma (n_j \cdot x_j^2) - \frac{1}{\Sigma n_j} \Sigma \left[(n_j \cdot x_j)^2 \right] \right\} \quad . \quad (100)$$

In the case where the elements n_j per strata are constant, the standard deviation is calculated in the following way for

$$n_j = k = \text{const.}$$

and

$$\Sigma n_j = j \cdot k \quad :$$

$$s_{\text{partial}}^2 = \frac{1}{j \cdot k - 1} \left[(k - 1) \Sigma s_{j,\text{partial}}^2 + k \cdot \Sigma (x_j - \bar{x})^2 \right] \quad . \quad (101)$$

The last term can be modified using Eq. 95:

$$s_{\text{partial}}^2 = \frac{1}{j \cdot k - 1} \left[(k - 1) \Sigma s_{j,\text{partial}}^2 + k \cdot \Sigma x_j^2 - \frac{k}{j} (\Sigma x_j)^2 \right] \quad . \quad (102)$$

Occasionally, strata means x_j are known along with their associated error s_j , which can be interpreted as 1σ statements, although the number of associated measurements n_j are unknown. It can further be postulated that these measurements are representative for the total population of the strata:

$$n_j = k \gg 1, \quad s_j \approx \sigma_j \quad ,$$

$$s_{\text{partial}}^2 = \frac{1}{j - 1} \Sigma s_j^2 + \hat{s}_{\text{partial}}^2 \quad , \quad (103)$$

where $\hat{s}_{\text{partial}}$ = standard deviation of $\bar{x}$,

$$\hat{s}_{\text{partial}}^2 = \frac{1}{j - 1} \Sigma (x_j - \bar{x})^2 \quad . \quad (104)$$

4.2. GAMMA SCAN AT PEACH BOTTOM

Upon removal during the 701-EFPD shutdown, FTE-4 was gamma scanned at Peach Bottom for fueled-length determinations. This was done by the Peach Bottom standard procedure, wherein the fuel element is carefully lowered from the charge machine, passing a collimator slot in the radial shielding. Gamma radiation that traverses the collimator path is detected by a scintillation crystal multiplier assembly and amplified. The count rate and the element position are recorded simultaneously on a strip chart from which the actual measurement of the fuel length is obtained.

The gamma scan strip chart for FTE-4 is shown in Fig. 17. The gamma-scan-determined fuel stack length change (Table 21) compares favorably with the changes determined from plenum, i.e., free distance between the fuel stack and the end of the graphite body, fuel stack, individual fuel rod length, and diameter measurements. It is slightly (5% to 15%) above the ground mean value of all five different methods; this is well within the 2σ noise level of $\pm 20\%$ to $\pm 30\%$ for all grand mean values. Only the plenum-determined length changes for the center and top body do not support the overall tendency. It can be concluded that the smaller amount of observed shrinkage in the center body and the expansion in the top fuel body may be due to perturbation of at least some of the eight fuel stacks, for which the measurements are representative.

The following conclusions can be drawn from the gamma-scan results:

1. Gamma scanning gives an accurate indication of fuel stack length changes during irradiation.
2. Average axial and radial fuel rod measurements agree very well with the gamma scan results.

4.3. DISASSEMBLY OPERATIONS

Upon arrival from Peach Bottom on March 29, 1974, the helium-filled aluminum canister containing FTE-4 was removed from the Hallam cask and injected into the high-level cell. The disassembly and postirradiation examination were performed according to a special procedure (Ref. 24). The canister was girdle cut approximately $3/16$ in. (4.7 mm) deep and 3 ft (914 mm) from the top end of the canister, and the piece was removed. The element was pulled out until clear of the canister, thereby exposing the thermocouple contacts. The remaining portion of the canister was discarded. Thermocouple resistance readings were measured and recorded (Table 22). The log resistancy values of the C/A thermocouple add up to the loop resistancy measurement, but W/Re values do not. This could have been caused by short-circuiting of the positive log to ground during loop

measurement, because the negative-to-ground reading is only slightly less than the loop reading of the grounded thermocouple. Preirradiation measurements are not available, so no further comparison can be made. A composite photograph of the total element is shown in Fig. 18.

In order to section the fuel element, it was supported horizontally by a series of V-rollers, which facilitated horizontal and rotational movement. A special extension drill was inserted through the purge gas inlet hole in the upper reflector, and a hole was drilled through the porous graphite plug. The fuel bodies were then seated against the bottom connector by using a push rod through this hole and applying force. While the bodies were held down, the upper reflector was removed by plunge cutting through the sleeve at a point about 2 in. (51 mm) below the sleeve-reflector joint. The sleeve was then girdle cut at a point about 16 in. (406 mm) above the sleeve-bottom connector joint. After inspection of the top surface of the graphite cloth at the upper end of the internal trap revealed no fuel particles, the bottom section of the lower reflector was removed. The two thermocouples were pulled out and the entire bottom assembly was discarded.

The fuel bodies were subsequently removed from the sleeve by inserting a push rod into the upper opening, holding the sleeve, and applying a force. Each individual fuel body was carefully examined externally, and the graphite integrity was assessed (see Figs. 19 through 21). Following visual inspection, the fuel bodies were placed in a special holding fixture to facilitate the removal of the center samples. A hole was drilled through the bottom sample hole plug and a push rod was inserted. When the upper sample hole plug was unscrewed, the distances from the top of the spine samples to the edge of the holes were measured. These measurements are compared with preirradiation measurements in Table 23. Changes are in the range to be expected for H-327 graphite except for the spine stack in body 3, which indicates an increase in length. However, this can be related to stack perturbation (e.g., a small chip between two spine samples could have caused this effect). These measurements are only meaningful for accurate determination of spine sample location. Therefore, no error analysis has

been performed. The spine samples were pushed out one at a time, identified, and placed in a pan. All these operations were carried out over this pan in order to catch any loose fuel particles from the drilled holes. However, none were found. No problems were encountered during the removal of the spine samples. The inscribed identity of all spine samples was the same as recorded in the drawings except for the thermal stability samples of fuel body 3, where the type 2 samples had a prefix of 2- on the drawings but no prefixes were observed on the actual samples. The activity of each spine sample was measured and recorded. All spine samples for each body were placed in separate, custom-built purge containers. The spine samples are described in Table 24.

Upon discharge of the spine samples, the graphite plugs capping the fuel holes were easily removed, and the distances from the top of the fuel rods to the edge of the fuel holes were measured. These measurements are compared with preirradiation measurements in Table 25.

Holes were drilled on the bottom of the fuel bodies to enable the fuel rods to be pushed out with a 1/8-in. (3.2 mm) diameter push rod. After all of the fuel rods had been removed, the fuel bodies were axially sectioned (Fig. 22); faces are shown in Figs. 23 through 25. From these figures, it can be concluded that the structural behavior of the graphite fuel bodies was sound. The numbers on the photographs are fuel hole identification numbers. The graphite slices have been used for autoradiography and some cesium profile measurements. Residual stress examination will be carried out on some of the graphite bodies.

4.4. FUEL ELEMENT AND SLEEVE EXAMINATION

Photographs of the entire length of the element were taken prior to examination of the element exterior with short-focus binoculars and the in-cell Kollmorgan periscope (see the mosaic photograph in Fig. 18). No cracks or abnormalities were found. The overall length of the element is shown in Table 26. FTE-4 increased in length, as had FTE-3. All strain evaluation data were taken from Ref. 3. The strain data are summarized below:

Total Fuel Element	Strain (%)	Accuracy (%)	Exposure (EFPD)
FTE-3	+0.07	+0.04	133
FTE-4	+0.35	+0.04	449

Sleeve outside diameters were determined with calibrated snap-on dial indicator gauges; inside diameters were measured from slices in the hot cell hood. Preirradiation measurements were used from the measurement of archive samples (Ref. 25). Table 27 gives both the preirradiation and postirradiation sleeve measurements and the associated strain. Figure 26 compares the TREVER-calibrated and measured sleeve strains. For the small number of measurements made, the calculated and measured values appear to be in fair agreement.

The following conclusions may be drawn from the results of the fuel element and sleeve examination:

1. The length of the element appears to increase slightly at low exposures as shown in both FTE-3 and FTE-4.
2. The sleeve strain from both inside diameter and outside diameter measurements shows fair agreement with TREVER-predicted strains.

4.5. FUEL BODY EXAMINATION

4.5.1. Fuel Body Weight Changes

The weight changes of the three graphite fuel bodies were $<\pm 0.2\%$ in all cases. Table 26 shows the preirradiation and postirradiation measurements, the relative differences, and the associated errors.

4.5.2. Graphite Fuel Body Dimensional Changes

After the fuel bodies were removed from the graphite sleeve and weighed, each was placed on a granite surface plate and examined.

Essentially no bowing was observed. Each body was photographed and visually inspected at high magnification, as shown in Figs. 19, 20, and 21. No cracks or abnormalities were found.

The dimensional changes for the longitudinal and transverse directions of the fuel bodies are presented in Tables 28 through 31. In Table 32, the TREVER-calculated and measured results are compared. These results are summarized below:

FTE-4 Fuel Body	Graphite Strain Measurements (%)					
	Axial	$\pm 2\sigma$	Radial	$\pm 2\sigma$	Anisotropy	$\pm 2\sigma$
Body 1 (bottom)	-0.10	± 0.02	-0.09	± 0.04	113	± 53
Body 2 (center)	-0.34	± 0.04	-0.22	± 0.05	159	± 39
Body 3 (top)	-0.24	± 0.03	-0.19	± 0.05	130	± 36
Total element	-0.23	± 0.08	-0.17	± 0.04	138	± 60

Comparison of the calculated and measured results in Table 32 and Fig. 27 shows an overprediction of the radial strain in the center of the element and reasonably close agreement in body 3 and the bottom of body 1. It is also obvious from the data that unfueled areas of the bodies showed significant end effects. Turnaround in the strain direction was obvious for the bottom body, which will have to be compared with other test elements to test its significance.

The fuel body strain comparison is summarized below:

FTE-4	Calculated vs. Measured Radial Graphite Strain			
	C/M-1 (%)	$\pm 2\sigma$ (%)	$\pm \text{RMS}$ (%)	χ^2 , Student "t," and Axial Variability Test Results
Body 1 (bottom)	111	± 57	± 126	Significant ^(a)
Body 2 (center)	68	± 18	± 67	Significant
Body 3 (top)	23	± 13	± 35	Significant
Total element	65	± 19	± 90	Significant

(a) Insignificant for axial variability test due to large relative measurement uncertainties.

Based on the previous discussion, the following conclusions can be drawn from the fuel body examination:

1. Weight changes of the fuel bodies were low ($<\pm 9$ g).
2. Visual appearance of the fuel bodies was good, with no apparent bowing.
3. There appear to be significant end effects in the unfueled zones of the graphite bodies which caused deviations in the fuel body strains from the predicted performance (Fig. 27).
4. The strain in bodies 1 and 2 was less than predicted, which may have been caused by an overestimation of the irradiation temperatures in the bottom of the element or, more likely, by limited knowledge of the graphite-strain characteristic at the low fluence range experienced by FTE-4.

4.5.3. Sleeve - Fuel Body Gaps

By comparing the preirradiation sleeve data in Ref. 25 with the post-irradiation fuel body and sleeve measurements in Tables 27 through 32, the sleeve - fuel body gap can be determined at BOL and EOL. These data are presented in Table 33 (outside diameter measurements) and are plotted in Fig. 28. The measurements are summarized below:

FTE-4 Fuel Element	BOL		EOL		BOL/EOL-1 (%)	Std. Dev. (%)
	Gap (mm)	Std. Dev. (mm)	Gap (mm)	Std. Dev. (mm)		
Body 1 (bottom)	0.155	± 0.004	0.155	± 0.035	0	± 23
Body 2 (center)	0.155	± 0.004	0.168	± 0.031	+8	± 20
Body 3 (top)	0.155	± 0.014	0.197	± 0.042	+27	± 23
Total element	0.155	± 0.005	0.176	± 0.042	+13	± 27

On an element average, there was a 13% increase in the sleeve - fuel body gap during irradiation. In the comparison of the measured and predicted values in Fig. 28, the measurements taken from the inside diameter of the sleeve show good agreement with predictions. The outside diameter measurements on the sleeve, on the other hand, are scattered about the predicted values.

4.5.4. Autoradiography

Beta-gamma autoradiography was employed to examine the fission product distributions in the graphite bodies. One cross-sectional slice was cut from each body. Each slice was ground with sandpaper until it was approximately 1/8 in. (3.2 mm) thick and flat. Exposures were made by placing the specimen on Kodak type M industrial film for about 70 min.

Figures 29, 30, and 31 show the autoradiography results for the three fuel body slices. In bodies 2 and 3, which had relatively high temperatures, the greatest release was from holes 1 and 2, which contained the UO_2 TRISO/ ThO_2 BISO fuel. The $(2.75 \text{ Th:U})\text{C}_2/\text{ThC}_2$ TRISO fuel in holes 5 and 6 had the lowest release, as would be expected. Body 1, which had relatively low temperatures compared with bodies 2 and 3, showed relatively little fission product release. All these results were consistent with FTE-3 results (Ref. 2).

4.5.5. Radial Cesium Determination

4.5.5.1. Destructive Graphite Web Sectioning. As an adjunct to the work discussed in Section 4.8, diffusion profiles of Cs-134 and Cs-137 in FTE-4 fuel body 2 have been measured. This work was performed to confirm that profile sectioning techniques developed for the analysis of diffusion spine samples were applicable to actual fuel element bodies and to provide data suitable for fitting with the FIPERQ code.

A 0.625-in. (15.90 mm) thick cross-sectional graphite slice, irradiated at an active core height of 57.3 in. (1.455 m), was cut from fuel

body 2. Beta-gamma autoradiography of the slice revealed a highly contaminated web adjacent to fuel hole 2 (see Fig. 30 and Section 4.5.4). This hole contained fuel rods with a TRISO UO_2 /BISO ThO_2 particle blend in matrix material. After the slice was milled to a thickness of 0.500 in. (12.70 mm), the graphite web was sectioned in a glove-box-enclosed milling machine. The samples were then gamma counted and analyzed with the FIPERQ code.

Figures 32 and 33 show the profiles obtained from sectioning. The 2 σ error bars are shown according to the convention described in Section 4.8. The concentration of Cs-137 is seen to be higher than the Cs-134 concentration, partly because the latter is an activation product of Cs-133 and thus experiences a time delay after fission before it is activated to Cs-134. The profiles of the two isotopes are seen to be similar, indicating that gaseous precursors do not play a significant role in establishing the profiles; the differences in precursor half-lives (Xe-133, 5.2 d; Xe-137, 3.8 m) would yield different profile shapes if holdup of the fission gases were sufficiently long to allow decay of Xe-137 to Cs-137.

The Cs-137 profile has been fit using FIPERQ, the reference design code used for modeling fission metal release from fuel elements (Refs. 26, 27). This code solves the differential equation modeling birth, decay, and diffusion of metals in the fuel rod and graphite regions of the fuel element. Calculated time-dependent temperatures are input into the code along with fuel element dimensions. No attempt is made to accurately model the source term since the time-dependent variation is uncertain. This does not affect the shape of the predicted profile since the Fickian diffusion coefficient input into the FIPERQ code is assumed to be independent of concentration. The computed profile, shown in Fig. 33, has been superimposed over the observed data points to allow ready comparison of the profiles. The FIPERQ profile is much flatter than the measured profile. This is an indication that the calculated temperatures are too high. Accordingly, a FIPERQ run was made using temperatures that were lower than calculated by 100°C. The fit, also shown in Fig. 33, more nearly coincides with the bulk of the measured data points, indicating that lower temperatures of

this magnitude are not unreasonable. This assumption has some experimental basis. The slice selected for analysis was cut off the bottom end of body 2. This region is known to experience lower temperatures as a result of a small fuel-free region at the interface. This temperature depression has not been accurately predicted, but values lower by at least 100°C can be expected.

4.5.5.2. Nondestructive Gamma Scanning. Fuel body 2, which had been sectioned near its center at approximately 58.07-in. (1475 mm) active core height, was scanned across several radial planes to determine the fission product release in various fuel holes. The 0.12 in. by 0.12 in. (3 mm by 3 mm) collimator was used with a slow traversing speed. Because this geometry is not calibrated for quantitative results, a comparison of the counts-per-minute of the volatile Cs-134 and Cs-137 isotopes gave a qualitative release of these fission products in various regions of the fuel body cross section (see Table 34). The gamma scanning shows insignificant cesium release near holes 3, 6, and 7 and relatively high release near hole 2. This is verified by the autoradiography of a thin section of body 2 (see Fig. 30).

A plot of the Cs-137 and Cs-134 activity across hole 2 is shown in Figs. 34 and 35. These data have not been corrected for the columnation effect, which tends to smear the data at the edges of the graphite. The purpose of these plots is only to give a qualitative picture of the cesium profile in the graphite web. The shape of the Cs-134 and Cs-137 profile from the outside edge of the fuel body to the fuel hole is similar to that found by a core drilling method (Figs. 32 and 33). The space between the fuel hole and the surface of the spine hole shows a lower Cs-137 activity and a significantly higher Cs-134 activity than the outside web.

4.6. FUEL ROD IRRADIATION PERFORMANCE

The purpose of the postirradiation examination is to determine fuel rod and fuel particle integrity and to characterize any irradiation effects. Visual examination with a stereomicroscope is used to determine

fuel rod integrity. Fuel rod irradiation-induced strain was measured, and the hypothesis of anisotropic behavior was tested. Fission gas release measurements give an estimate of the extent of fuel failure, while metallography and autoradiography characterize the fuel failure modes and thermal effects in the fuel. Selected fuel rods had been gamma spectroscopically analyzed for homogeneity, fission product distribution, and burnup estimation.

4.6.1. Sample Description

FTE-4 had four fuel rod types that were loaded into three H-327 graphite bodies of eight-hole "teledial" design. Fuel loadings in all three bodies were the same (see Table 35).

All fuel rods were fabricated by the reference FSV hot-injection process using natural-flake graphite filler and pitch binder followed by an 1800°C heat treatment in an alumina packed bed. The nominal dimensions of the rods were 0.490 in. (12.4 mm) in diameter and 1.94 in. (49.3 mm) long. The fuel rods in this test element that were examined and the extent of the examination are presented in Table 36. Most of the examination was done in body 2, which showed the highest EOL fluence [1.9×10^{25} n/m² ($E \geq 29$ fJ)] and highest time-averaged temperatures (1090° to 1115°C).

4.6.2. Visual Examination

Representative fuel rods were examined and photographed immediately after unloading using a Kollmorgan periscope. Rods selected for further examination were examined with a Bausch and Lomb stereomicroscope. In all cases irradiation-induced damage of the rods was low except for slight debonding of particles at the end caps and spalling at the edges. In all cases the rods showed unloading damage. This damage was characterized by striations on the sides of the rods that, in many cases, caused extensive debonding and breaking of particles on the surface. Stereophotographs of four rods that represent the four fuel types from body 2 (Figs. 36 through 39) give a clear idea of the extent of this damage. Particles that were

clearly broken on the surface were counted for each rod and are listed in Table 36.

4.6.3. Fuel Rod Dimensional Changes

Radial and axial dimensional measurements were taken on 80 out of 336 fuel rods. The preirradiation measurements and their associated errors were determined according to the procedure and algorithm outlined in Section 4.1. Figures 40 through 43 give the as-measured fuel rod strain in the axial and radial directions. The former was measured twice and the latter six times during the postirradiation examination. Only three preirradiation measurements were taken.

Fuel blends in holes 1, 2, 5, and 6 show a slightly narrower band than holes 3 and 4 and holes 7 and 8. In the majority of cases, the diametral fuel shrinkage was larger at the center location than at the two ends of the rod. All fuel rod strain data follow a general pattern which can be correlated to the temperature and fluence profiles. The shrinkage was more pronounced at the top end of the element than at the bottom end; this is related to the higher temperatures at the gas exit side of the element, i.e., top end.

Figures 44, 45, and 46 and Tables 37 through 43 present the strain measurements with 95% confidence limits and the TREVER-predicted strain. The TREVER predictions are based on FSV and large HTGR design code subroutines, which calculate the fuel rod dimensional changes as a sum of its subcomponent irradiation deformation (Ref. 3). The contribution of each subcomponent is a function of its volume in the fuel rod. Fuel from stacks 3 and 4 and fuel from stacks 7 and 8 are similar in that respect (see Table 35) and therefore are shown together on one plot (Fig. 45).

Selected data from Tables 37 through 43 are compared with FTE-3 data below:

FTE-4 Fuel Rod Strain After 449 EFPD

	Fuel Blend	Axial Strain (%)	$\pm 2\sigma$ (%)	Radial Strain (%)	$\pm 2\sigma$ (%)	Axial/Radial Anisotropy (%)	$\pm 2\sigma$ (%)
Holes 1 & 2	TRISO/BISO 22.4:77.6	-2.57	± 0.11	-2.63	± 0.25	97.7	± 10.2
Holes 3 & 4	TRISO/BISO 78.1:21.9	-1.62	± 0.17	-1.86	± 0.27	87.5	± 15.7
Holes 5 & 6	TRISO/TRISO 78.6:21.4	-1.48	± 0.09	-1.47	± 0.26	100.5	± 18.5
Holes 7 & 8	TRISO/BISO 64.1:35.9	-1.73	± 0.16	-2.10	± 0.27	82.5	± 13.2
Total element		-1.89	± 0.11	-2.02	± 0.25	93.7	± 12.8

FTE-3 Fuel Rod Strain After 133 EFPD

	Fuel Blend	Axial Strain (%)	$\pm 2\sigma$ (%)	Radial Strain (%)	$\pm 2\sigma$ (%)	Axial/Radial Anisotropy (%)	$\pm 2\sigma$ (%)
Holes 1 & 2	TRISO/BISO 24.2/75.8	-1.58	± 0.14	-1.55	± 0.26	102.3	± 19.0
Holes 3 & 4	TRISO/BISO 85.4/14.6	-0.96	± 0.12	-1.06	± 0.25	91.3	± 24.6
Holes 5 & 6	TRISO/TRISO 85.8:14.2	-0.84	± 0.11	-0.90	± 0.25	93.4	± 28.7
Holes 7 & 8	TRISO/BISO 74.0:26.0	-0.97	± 0.09	-1.26	± 0.25	76.9	± 17.0
Total element		-1.13	± 0.07	-1.23	± 0.25	91.6	± 19.2

The reference design code models for fuel rod performance (Ref. 9) are based on the assumption that a fuel rod performs identically in the radial and axial directions; i.e., the subcomponents are fully disoriented, giving an anisotropy of 1. This hypothesis has been tested on each of the examined fuel rods and on the whole population of each fuel rod stack in all three fuel bodies. These results are presented in Figs. 47 through 50. The hypothesis of anisotropy can be accepted for three out of the four fuel rod blends with 95% confidence. However, for the UC_2 TRISO/ ThC_2 BISO fuel

in holes 7 and 8, the hypothesis of isotropy has to be rejected because its mean value of $83\% \pm 13\%$ (2σ) is definitely below 100% anisotropy at the 95% confidence level.

Element average strains for each fuel blend from Tables 37 through 39 are compared below:

FTE-4 Total Element Fuel Rod Strains After 449 EFPD

	Fuel Blend	Calculated Vs. Measured Fuel Rod Strain		
		C/M-1 (%)	$\pm 2\sigma$ (%)	$\pm$ RMS (%)
Holes 1 & 2	TRISO/BISO 22.4:77.6	33.7	± 2.3	± 12.4
Holes 3 & 4*	TRISO/BISO 78.1:21.9	17.8	± 4.7	± 18.5
Holes 5 & 6	TRISO/TRISO 78.6:21.4	-24.2	± 2.5	± 14.4
Holes 7 & 8*	TRISO/BISO 64.1:35.9	17.8	± 4.7	± 18.5

In order to differentiate axially, the results were tabulated per body (error and axial variation statements RMS have been omitted for simplicity):

FTE-4 Fuel Rod Strains Per Fuel Body

	Fuel Blend	Bias (C/M-1) Calculated Vs. Measured Fuel Rod Strain			
		Body 1 (%)	Body 2 (%)	Body 3 (%)	Total Element (%)
Holes 1 & 2	TRISO/BISO	46.8	32.9	23.1	33.7
Holes 3 & 4*	TRISO/BISO	38.3	12.5	2.7	17.8
Holes 5 & 6	TRISO/TRISO	-4.7	-26.2	-37.2	-24.2
Holes 7 * 8*	TRISO/BISO	38.3	12.5	2.7	17.8

*Combined because of similar loadings.

Figures 44 through 46 present the measured and calculated strain characteristics along the axis of the experiment. In all three cases (two blends were grouped together because of similar loadings), the strain was either significantly overpredicted or underpredicted. The TRISO/BISO blend in holes 1 and 2 is about 30% overpredicted on an element basis. The all-TRISO blend in holes 5 and 6 is underpredicted by about 20%. The TRISO/BISO fuel blend in holes 3 and 4 and holes 7 and 8 is overpredicted by about 20%.

The axial variation for the bias between prediction and measurements follows a clear trend on a fuel body average basis: the effect is declining from the bottom to the top of the element, with a large positive bias becoming smaller and a negative bias becoming even more negative from bottom to top.

The results of the fuel rod strain examination are summarized below:

1. The fuel rod strain reveals a pattern which generally follows the temperature and fluence distribution.
2. There are less pronounced indications of end effects in the fuel rod strain than in the fuel body strain.
3. The predicted fuel rod strain is less than the measured values in the case of the TRISO/TRISO fuel blend and greater than the measured values in the TRISO/BISO fuel blends. In all cases the measured and predicted axial profiles disagree in a varying amount along the axis of the experiment. A lack of proper modeling of fluence and temperature dependence in the strain characteristics is believed to be the cause.
4. All the fuel rods except rods from holes 7 and 8 showed isotropic fuel rod strain behavior. No fluence or temperature dependence can be detected for the anisotropy.

4.6.4. Fuel Body - Fuel Rod Gaps

Using the BOL fuel hole diameter, the graphite fuel body outside diameter strain data, and the preirradiation and postirradiation fuel rod information, the EOL fuel body - fuel rod gap can be calculated. Tables 44 through 46 compare the calculated and measured gaps, and these data are plotted in Figs. 51 through 53. The comparison of the measured and predicted gaps was not expected to yield consistent results because both the fuel rod and graphite fuel body strain models were already found to be in disagreement with the measured results.

In holes 1 and 2, the measured gaps were between 0.21 and 0.30 mm, with an element average of 0.28 mm. The predicted element average was 0.33 mm, i.e., a 17% overprediction. In holes 3, 4, 7, and 8, the measured gap was 0.22 mm with a 10% overprediction on the element average. Measured results from holes 5 and 6 were higher on the element than the predicted results by about 9%, with an EOL measured average gap of 0.20 mm.

The gap predictions are close enough to the measurements to guarantee reasonable fuel temperatures. Gap contours can be used to test the validity of graphite and fuel rod strain correlations. Modifications to these correlations can be tested against the FTE-4 measurements when they become available.

4.6.5. Fission Gas Release

The FTE-4 fuel performed well during irradiation as shown by fission gas release measurements (Table 36). With the exception of the rods damaged during removal, the R/B Kr-85m was found to be of the same magnitude as values measured prior to irradiation for similar fuel manufactured for the FSV reactor (Ref. 28). It is reasonable to assume that the fission gas release from FTE-4 remained unchanged during irradiation with no fuel failure. These results are consistent with expected behavior based on previous work (Refs. 28,29). Several rods (2-1-7, 2-2-7, and 2-7-7) which received extensive surface damage during unloading had higher fission gas

release measurements than companion rods (2-1-8 and 2-8-7). This indicates that the unloading damage was a significant contribution to the R/B measurements.

4.6.6. Metallography

Metallography was done on at least one fuel rod from each fuel blend; the results are given in Table 36. All of these rods were located in the center of body 2, which had the highest fast fluence [about 1.9×10^{25} n/m² ($E \geq 29$ fJ)] and temperature (1090°C). Radial cross sections of each of these rods and high-magnification microphotographs of the matrix are shown in Figs. 54 through 58. In general, the fuel had little coating failure, and there was no matrix cracking or interaction.

Fuel rods 2-3-8 [(Th,U)C₂ TRISO/ThC₂ BISO], 2-5-7 [(Th,U)C₂ TRISO/ThC₂ TRISO], and 2-8-7 (UC₂ TRISO/ThC₂ BISO) all showed very good performance. Failure fractions were low and were attributed mostly to polishing damage. The TRISO particles all showed buffer densification and debonding. No fuel kernel migration or metallic fission product attack was observed. Representative photomicrographs of particles are shown in Figs. 59 through 61.

Rods 2-1-7 and 2-2-7, which contained a UO₂ TRISO/ThO₂ BISO blend, showed significant kernel mobility (Fig. 62). No fission product attack or coating failure was associated with this kernel migration. The UO₂ kernel has shown high mobility in previous work, and for that reason it has been rejected as a candidate HTGR fuel material (Ref. 29). The ThO₂ BISO fuel in these rods showed excellent performance with no visible kernel migration or failure.

Several of the rods (2-8-7, 2-1-7, and 2-2-7) showed an inhomogeneity in particle loading. A simple analysis of a radial cross section of fuel rod 2-1-7 was done to determine the extent of this inhomogeneity. A count of the number of UO₂ fissile and ThO₂ fertile particles was made for four equal pie-shaped sections. A summary of these results is given in Table 47A.

From the analysis it can be seen that there is a slight discrepancy between the total actual loading and the theoretical loading at this plane. However, of greater importance is the considerable inhomogeneity in loading in the various sections. Almost all the fissile particles were located in sections 1 and 4 and near the outer edge of the rod.

Another rod (2-2-7) that contained the same UO_2 TRISO/ ThO_2 BISO blend was also analyzed for inhomogeneity in an axial cross section and divided into 12 sections as shown in Table 47B. Examination of these data shows that almost 50% of the fissile particles are located in sections 1 and 5.

Other fuel homogeneity determinations from the axial gamma scanning of the fuel rods are discussed in Section 4.6.7.

4.6.7. Fuel Gamma Scanning

Fuel rods, fuel bodies, and spine samples were gamma-scanned using a Ge(Li) spectrometer with a 6.3% efficiency. All the samples were transversed past a 42-in. (1066.9 mm) columnator at slow speeds using a custom-built rig (Fig. 63). The size and the shape of the collimator cross section were changed for various activities and shapes of specimens being scanned. All the data were analyzed by the Sigma II computer facility, which was directly interfaced with the Ge(Li) detector.

Representative fuel rods from all three fuel bodies were scanned. An 0.85 in. by 0.023 in. (22.23 mm by 0.58 mm) collimator was used in all cases. The rods were passed by the collimator so that the long axis of the collimator was perpendicular to the length of the rod. This counting geometry was calibrated with a standard rod of known burnup from capsule P13R (Ref. 30). This allowed a quantitative FIMA calculation based on knowledge of the fission product inventories of the fuel rods. These data are presented in Table 48. Because these fuel rods were counted almost 2 yr after being taken out of the Peach Bottom reactor, many of the short-lived isotopes had decayed away. Also, the count rates of most of the

isotopes were low, which resulted in substantial counting errors as shown in Table 48.

The results of the fuel rod inventories were substantiated by the analysis of several similar rods at both GA and ORNL (Ref. 21) (see Table 49). Good agreement was found between the GA and ORNL measurements for all isotopes except Zr-95. This can be attributed to the low count rates of Zr-95 and subsequent high counting errors. Burnup calculations for both the ORNL and GA fuel rods are included in Table 17.

The activities of Cs-137 measured in fuel rods from stack 3 are plotted against core location in Fig. 64. Because of the long half-life of the Cs-137 isotope (i.e., 30.2 y), the time-averaged thermal flux (i.e., fluence) can be correlated with the cesium profile which was overlaid on the fluence data in Fig. 64. Reasonable agreement was found between the calculated thermal fluence and measured Cs-137 inventory.

In addition to the multichannel analysis of the fuel rods, single-channel scans of Cs-137 were generated to determine the relative axial homogeneity of the fuel rods. Only a few representative rods from holes 3, 4, 5, 6, 7, and 8 have been examined. A brief examination of the data shows that fuel rods loaded in holes 3 through 6 were fairly homogeneous and fuel rods loaded in holes 7 and 8 were inhomogeneous. Single-channel scans of representative fuel rods from holes 5 and 8 given in Fig. 65 show this difference in fuel loading uniformity. Fuel rods from holes 1 and 2 have not been scanned because these samples were not available at the time of the gamma scanning exercise; however, metallographic examination showed these rods to be inhomogeneous (see Section 4.6.5).

The results of the fuel gamma scanning may be summarized as follows:

1. The measured isotopic inventory agreed well with independent ORNL measurements.

2. The Cs-137 inventory in the fuel rods agreed with the predicted thermal fluence profile.
3. Total FIMA measured from the Cs-137 fuel rod inventories was consistent with predictions and radiochemical determinations (see Section 3.4.2).
4. Fuel rods in holes 3, 4, 5, and 6 had homogeneous fuel loadings as determined from single-channel gamma scan plots of the rods. Fuel rods in holes 7 and 8 had inhomogeneous fuel loadings. Rods in holes 1 and 2 were not scanned but were examined metallographically and showed inhomogeneous fuel loadings.

4.7. THERMAL STABILITY SAMPLES

4.7.1. Test Configuration

Thermal stability samples were contained in both type 1 and type 2 crucibles. These crucibles were loaded into a 1.1-in. (27.9 mm) diameter hole in the center of the fuel bodies as indicated in Table 50. The types of fuels examined in these crucibles are also listed in Table 50.

The two types of crucibles were used to provide both an isothermal experiment (i.e., type 2) and an experiment that had similar irradiation conditions but a constant high thermal gradient (i.e., type 1). This thermal gradient was achieved in the type 1 crucibles by placing one particle in each hole of the small inner crucibles and then packing TRISO fuel in the void space of the outer crucible. Samples in the type 2 crucible were simply loaded into the six small holes, which guaranteed a relatively isothermal condition for all the particles in a particular hole. Only one sample from a type 2 crucible has been examined to date (i.e., TS4-5) (Fig. 66).

Postirradiation examination of the spine samples included visual examination, radiography, metallography, and fission gas release.

4.7.2. Visual Examination

Thermal stability sample TS4-5 containing the 100- μm (VSM) UC_2 TRISO particles (batch 4000-302) showed no observable coating failures when examined with a Bausch and Lomb stereomicroscope. Figure 67 is a photograph of several representative particles.

4.7.3. Radiography

Radiographs of two sets of 20 particles were taken of spine sample TS4-5. No failures were observed in either case. Figure 68 is a photograph of one of these radiographs.

4.7.4. Metallography

Metallography of spine sample TS4-5 is represented in Fig. 69. The particles looked excellent in all cases. No SiC failures were observed, and the 2.7% PyC failures were attributed to polishing. Buffer densification and debonding were evident in all cases.

4.7.5. Fission Gas Release

The entire batch of particles was measured for the release of Kr-85m at 1100°C in the TRIGA reactor facility after disassembly. The measured R/B was 3.12×10^{-6} , which indicates insignificant failure or contamination.

4.8. METALLIC DIFFUSION SPINE SAMPLES

4.8.1. Test Description

Twenty-two fission metal diffusion samples were irradiated in the center spine position of FTE-4 (see Table 51). The spine samples were designed to provide data on the transport properties of fission product metals, including diffusion and sorption, in HTGR fuel materials. They are particularly well suited for providing data on metal behavior in

graphite because (1) the temperature of crucibles is uniform throughout, (2) the fission metal source concentrations in the matrix material remain essentially constant during the irradiation, and (3) the geometry of the samples is simple and well characterized (see Fig. 70). All of these features are important when modeling the observed profiles. Specific effects that could be studied included:

1. Determination of the transport properties of selected metallic fission products in graphite under HTGR irradiation conditions as a function of metal concentration and temperature.
2. Measurement of the distribution of metallic fission products between matrix and graphite (partition factor).
3. Evaluation of the effects of other fission product metals on the migration rate of individual metals.

Studies of the FTE-4 diffusion samples were directed toward characterization of the fission metals cesium, strontium, barium, and europium by analysis of the concentration profiles established in the graphite crucible center posts and annular walls and by measurement of the fission metal partition coefficients. This work is an extension of that initiated in FTE-3 (Ref. 2), where samples irradiated to 133 EFPD were analyzed to determine fission metal behavior.

4.8.2. Experimental

The FTE-4 diffusion samples were located in fuel bodies 1 and 2. Based partly on a review of the data presented in the FTE-3 report, six samples were selected for analysis (see Table 52). The samples were initially loaded with strontium (samples No. 10, 12, 18, 22),* barium (31),*

*Samples 10, 12, 18, 22, and 31 had no cesium loaded into their matrix materials prior to irradiation. The cesium found in the graphite and matrix during sectioning was the result of in-diffusion from adjacent fuel compacts in the element.

or mixtures of cesium, strontium, barium, and europium isotopes in varying concentrations (41). The metallic source materials and their respective gettering agents were loaded as carbides into previously out-gassed H-327 graphite crucibles using inerted-glove-box techniques. The carbides were formed by sorbing the metal isotopes of interest on matrix material and converting to the carbide at 1400°C in a vacuum furnace. The conversion was accomplished by heating the loaded matrix material in a tantalum crucible lined with graphite foil to minimize contamination of the matrix with tantalum. Following conversion to carbide, the matrix was loaded into the annulus of the H-327 graphite crucible and sealed with a screw-top lid. After loading, the crucibles were coated on the outside with polystyrene to protect the metals and carbides from exposure to moist air (thereby preventing hydrolysis or oxidation) prior to insertion in the reactor.

Fission product concentration profiles of Sr-85, Ba-133, Cs-134, and Eu-154 were obtained from radial sections of the diffusion spine sample crucibles. The observed species were activation tracers of cesium, strontium (enriched in Sr-84), barium (tagged with Ba-133), and samarium (enriched in Sm-152) as shown in Table 53.

After irradiation to 448.9 EFPD and a suitable cooling period, the diffusion spine crucibles were removed from the fuel test element, the matrix material was removed and gamma counted, and the samples were sectioned by turning the crucible walls and center posts in a glove-box-enclosed lathe, uniformly removing a layer of graphite for each section. Samples of known weight were gamma counted using a Ge(Li) crystal detector and a 4096-channel pulse height analyzer. A 370 IBM computer, Model 165, was used in conjunction with the program CPROFIT (Concentration PROfile FIT) to assist in the analysis of the concentration profiles that were analyzed at North Carolina State University (Ref. 31).

4.8.3. Analysis and Results

4.8.3.1. Strontium. The concentration profiles for Sr-85 in the center post and crucible wall of samples 10, 12, 18, and 22B are shown in Figs. 71

through 77. The error bars associated with each point in the figures represent the following:

1. The horizontal error bars represent the absolute RMS 2σ uncertainty associated with the radial position of each data point. The absolute uncertainty in each case is $50.4 \mu\text{m}$.
2. The vertical error bars represent the relative 2σ uncertainty in the fission metal concentration resulting from counting errors (assuming a Poisson distribution). The uncertainty in the concentration introduced from weighing errors was found to be significantly less than 5% of the counting error in all cases and was neglected in the analyses. If no error bars are shown, the uncertainty was found to lie inside the area outlined by the data point.

Previous analyses of metal profiles in graphite (Refs. 2,31) have revealed that strontium generally exhibits classical Fickian diffusion profiles which can be fitted with the complementary error function equation (Ref. 32):

$$C = C_0 \operatorname{erfc} (x/2\sqrt{Dt}) \quad , \quad (105)$$

where C = metal concentration in the graphite at x (mg/g),

C_0 = metal concentration in the graphite region adjacent to the matrix (mg/g),

x = depth of penetration (cm),

D = Fickian bulk diffusion coefficient (cm^2/s),

t = irradiation time (s).

Since the strontium profiles in the crucible walls and center posts were Fickian in nature, they were fit with the CPROFIT non-linear least squares complementary error function options. The results of this analyses are presented in Table 54. An Arrhenius plot of these data, along with the reference data line (Ref. 31) for strontium and diffusion coefficients

derived from FTE-3 spine sample analyses, is presented in Fig. 78. It should be noted that the FTE-3 spine sample temperatures were revised subsequent to the issuance of the FTE-3 report (Ref. 2). Table 55 presents the revised activation-energy-weighted temperatures for these samples.

A review of Fig. 78 shows that the diffusion coefficient data derived from FTE-3 sample analyses is in good agreement with the reference line. This is not surprising since the revised temperatures from Table 55 were derived in part from the assumption that the data should be in good agreement with the reference line. This was dictated because the strontium data for diffusion in graphite show a high degree of precision (Ref. 31). Any significant deviation from the reference line is assumed to result principally from uncertainty in the temperature, rather than from uncertainty in the calculated value of D . Using this criterion, the FTE-4 crucible temperatures appear to be too high by $\sim 150^\circ\text{C}$. In addition, a least squares fit of the data provides an activation energy, E_a , of 83,700 cal/mole, significantly larger than the reference value of 64,000 cal/mole. The former value is subject to considerable uncertainty since only seven values were available for use in deriving E_a .

4.8.3.2. Barium. The profiles of barium in the center post and crucible wall of diffusion sample 31 are shown in Figs. 79 and 80. The results of fitting the profile, as presented in Table 54, indicated that the use of a complementary error function analysis as utilized in the CPROFIT code was appropriate. These results are plotted in Fig. 81 along with similar results from FTE-3 and the reference Arrhenius line for strontium. (The barium and europium results are compared with the reference strontium line since, for reactor design purposes, the three metals are assumed to behave in a like manner.) The revised data for FTE-3 (see Table 55) again appear to cluster about the reference line, although the data exhibit almost two orders of magnitude uncertainty. The barium diffusion coefficients derived from FTE-3 substantiate the assumption that the calculated spine temperatures are too high by $\sim 150^\circ\text{C}$.

4.8.3.3. Europium. The europium profiles found in the center post and crucible wall of spine sample 41 are shown in Figs. 82 and 83. The crucible wall profile could be adequately modeled with the complementary error function option in the CPROFIT code, indicating that Fickian diffusion was occurring. The center post profile, however, showed evidence of a "fast path" tail. Both the fast and slow path diffusion coefficients were calculated using CPROFIT (see Section 4.8.3.4). The results are presented in Table 54 and Fig. 81. Whether this "tail" represents a fast path mechanism is uncertain. Further similar results will be required to substantiate this conclusion.

4.8.3.4. Cesium. Concentration profiles for Cs-134 in crucibles 10, 12, 18, 22, and 31 are presented in Figs. 84 through 96. In a number of cases, a figure has two parts, "a" and "b." The "a" part presents the original data, while the "b" part gives the data and fit points which result from the CPROFIT computer code analysis and plotting. Note that the data points in part "b" representing the slow component of diffusion (squares) have the contribution of the fast component (triangles) subtracted. This results in a lower value for the slow component data points in part "b" than are plotted in part "a." No cesium was initially loaded into the matrix material of the first four crucibles; the cesium profiles found result from in-diffusion from the adjacent fuel rods surrounding the center spine. Even in the case of crucible 41, which had matrix material preloaded with cesium, permeation from the fuel rods was found to be the dominant source. These profiles were not fit with CPROFIT since a constant source term could not be assumed (a necessary assumption for the correct application of the CPROFIT code).

A review of the cesium profiles from FTE-4 confirms the results found in FTE-3, namely that the diffusion of cesium probably occurs by a non-Fickian mechanism. In many cases, two profile regions are seen, one adjacent to the source region, characterized by a very steep slope (attributed

to a "slow path" diffusion mechanism*), and a second flatter profile extending through the graphite (attributed to a "fast path" diffusion mechanism). Several authors have proposed models to treat this behavior, including "uncoupled" (Ref. 2), "coupled" (Refs. 33,34), and "decoupled" (Ref. 35) fast and slow path diffusion mechanisms. Chandra and Norman (Ref. 36) have also proposed a coupled diffusion mechanism based on the Fisher model (Ref. 37). In order to permit direct comparison with the metal diffusion results derived from FTE-3 diffusion samples, the current data have been analyzed with the CPROFIT code. Later analyses of these results may indicate that another model more closely describes the physical phenomena of cesium diffusion and may be more appropriate for predictions of cesium release from the HTGR core.

The CPROFIT analysis of cesium diffusion is based on the assumption that the fast component of cesium is very likely due to in-pore surface diffusion of atoms, which is accompanied by an activated diffusion and sorption process whereby atoms are fixed to sites where they have a low mobility. Time constants are such that a quasi-steady state is reached in a few hours (or tens of hours) and then, assuming the adsorption rate coefficient K to remain constant, a concentration profile is established whose shape does not change with time even though its magnitude continues to increase. This model is discussed at length in Ref. 2. Briefly, the observed fast path profile is fit with the best value of K/D , where D is the diffusion coefficient of cesium migrating along the pores. When $Kt \gg 1$,

$$D_{\text{eff}} = \frac{1}{(K/D)t} \quad , \quad (106)$$

where D_{eff} = "effective" fast path diffusion coefficient (cm^2/s),
 K = adsorption rate coefficient (s^{-1}),

*The "slow path" diffusion mechanism is not always prominent and is characterized by a diffusion coefficient of low magnitude ($D < 10^{-11} \text{ cm}^2/\text{s}$) and little temperature dependence. Whether this portion of the profile is a bulk diffusion process or, in fact, is an artifact of outer surface effects or experimental sectioning techniques is uncertain.

D = diffusion coefficient for migration along the pore (cm^2/s),
 t = irradiation time (s).

The results of CPROFIT fits of the cesium data from each crucible are presented in Table 56 and Figs. 84b, 86b, 88b, 90, and 95. In using the CPROFIT code to model the center post profiles, it was assumed that the loading of the matrix was sufficiently rapid to supply a steady source term for diffusion into the center post. It can be seen that all the profiles exhibited a fast path component that could be fit with the uncoupled diffusion model in CPROFIT. The presence of the slow component was not always evident, a finding that tends to confirm that this phenomenon may result from simple loading of the surface at the boundary, an effect that could be expected to differ for different samples.

In addition to calculations of D_{eff} from CPROFIT analyses, additional values have been calculated from measurements of the cesium loading in the matrix material (for those crucibles which initially had no cesium present in the matrix). The method has been discussed in Ref. 2. Briefly, the flux, $\bar{J}$, to the matrix is determined by measuring the cesium loading arising from in-diffusion and dividing by the irradiation time, t . This value is then coupled with the concentration gradient across the crucible wall to establish D_{eff} :

$$\bar{J} = -D_{\text{eff}} \frac{\Delta(C_1 - C_g)}{\Delta X}, \quad (107)$$

where $\bar{J}$ = time-averaged flux ($\text{g}/\text{cm}^2\text{-s}$),
 D_{eff} = effective diffusion coefficient (cm^2/s),
 C_1 = extrapolated cesium concentration at the outer crucible wall boundary (mg/g),
 C_g = extrapolated cesium concentration in the graphite crucible wall adjacent to the matrix (mg/g),
 ΔX = wall thickness (cm).

Values of D_{eff} calculated in this manner are shown in Table 56. Note that in all but one case (sample 22b), the values of D_{eff} (flux) are greater than the comparable D_{eff} (profile) values. No explanation for this discrepancy is offered.

The results of both types of analyses are included in Fig. 97, where the effective cesium diffusion coefficients are plotted as a function of temperature along with data derived from FTE-3 spine sample analysis. Also shown is the GA reference Arrhenius line for cesium diffusion in graphite (Ref. 38). The reasonable agreement of the FTE-3 data is expected because of adjustments in the calculated spine temperatures (see Section 4.8.3.1). As plotted, the FTE-4 data derived from diffusion profiles are in good accord with the reference line. This agreement may be fortuitous since the crucible temperatures may actually be lower by $\sim 150^\circ\text{C}$, as proposed following a review of the strontium, barium, and europium data. If the data were plotted using reduced temperatures, virtually all would lie above the reference line. Later use of these data to help establish the behavior of cesium in graphite should take account of the uncertainty present in all the spine sample temperatures.

The partition factors, ϕ (cesium concentration in matrix/cesium concentration in graphite), presented in Table 56 show a general trend of decreasing ϕ with increasing temperature. The trend is significantly greater than that expected from design calculations (Ref. 38) where values of 14.5 ± 0.5 (900° to 1100°C) are computed using the reference isotherms shown in Table 57 (assuming Henrian behavior). It should be noted, however, that the reference matrix isotherm is not that for furfuryl alcohol/resin coke, the matrix material used in the crucibles. This difference introduces uncertainty into the calculated absolute value of ϕ but cannot explain the lack of temperature dependence noted for the calculated ϕ values. The large observed ϕ values are surprising since irradiation is felt to increase the sorptive capacity of the graphite significantly more than that of matrix material, resulting in a lowered ϕ following irradiation. (The reference isotherms were obtained from unirradiated materials.) Similar results were found in FTE-3 analyses.

4.8.3.5. Summary. Additional data have been gathered on the diffusive behavior of the fission metals cesium, strontium, barium, and europium. The behavior of cesium was analyzed using a two-component model that included non-Fickian behavior. Strontium and barium profiles were adequately fit using a Fickian model. Some question as to the existence of a "fast" component for europium exists; therefore, it was fit with the two-component model. This work has again highlighted the importance of knowing accurate irradiation temperatures. The results of strontium profile fitting showed that the calculated spine temperatures were probably too high by $\sim 150^{\circ}\text{C}$. This uncertainty prevents the inclusion of FTE-4 diffusion sample data in the reference data base at this time. A later independent reassessment of the temperatures that provides more accurate results will permit the inclusion of these data in the reference data base.

4.9. BORONATED GRAPHITE SPINE SAMPLES

4.9.1. Test Material Description

The boronated graphite spine sample irradiated in FTE-4 was a shortened lower reflector shield pin assembly fabricated by standard FSV production techniques using reference FSV materials. The assembly consisted of four B_4C -graphite pellets 0.44 in. (11.2 mm) in diameter by 0.44 in. (11.2 mm) long, encapsulated in a 0.500-in. (12.7 mm) O.D., 2.5-in. (63.5 mm) long Hastelloy X tube. The tube was sealed at both ends with Hastelloy X end caps welded to the tube. One end cap contained a 0.12-in. (305 mm) diameter hole to vent the closed assembly. Figure 98 shows the shield pin assembly.

4.9.2. Postirradiation Examination and Discussion

The four B_4C -graphite pellets were removed from the Hastelloy X canister by cutting off the end cap. The B_4C graphite pellets were observed to be intact, although a light surface discoloration was evident at free surfaces of the pellets. The weights and dimensions of the irradiated pellets were measured as reported in Table 58.

One pellet was sectioned for microprobe examination, which revealed that the light discoloration observed was present only at the surfaces of the pellets. The material in the light area was identified as a boron-nitrogen compound, most likely BN.

The small measured dimensional changes in the pellets confirm that the B_4C -graphite composite is stable under neutron irradiation. However, the existence of the surface coating of BN and the attendant weight loss in the pellets limit the accuracy of the dimensional analysis to at best $\pm 1\%$ of the dimensions. Thus, the data show only that no significant dimensional change occurred in the B_4C graphite pellets.

BN was formed at the free surfaces of the pellets by reaction of B_4C in the pellets with excessive amounts of N_2 in the Peach Bottom coolant, which in some cases exceeded 20 ppm (Ref. 39). Limited amounts of N_2 apparently diffused through the canister vent hole to react at the pellet surface.

The performance of the shield pin assembly is unaffected by the formation of BN since BN is as stable as B_4C . BN is stable in the presence of water at temperatures below $400^\circ C$, while at high temperatures BN will compete with B_2O_3 as a reaction product of B_4C and coolant impurities.

Examination of the B_4C graphite pellets shows that the compacts are dimensionally stable under irradiation. Although some chemical interaction between B_4C and coolant impurities occurred, the reaction was limited to free surfaces and the reaction product BN does not compromise the performance of the neutron absorber pellet.

4.10. BORONATED STAINLESS STEEL SAMPLE

4.10.1. Test Sample Description

A 5-in.-long, 3/4-in.-diameter boronated stainless steel rod was obtained from Esco Corporation during the FSV production run. The rod was

fabricated to meet FSV specification 17-R-12(B) with a natural boron content of 2.0 to 2.5 wt %.

A sample was cut from the rod and machined to the configuration shown in Fig. 99. The ends were turned to a smaller diameter to provide locations for precise diameter measurements. Preirradiation characterization measurements are included in Fig. 99. Archive samples are in the possession of the Fuel Test Element Activity Branch.

The sample was inserted into an H-327 graphite crucible (see Fig. 99). The crucible was placed in test element FTE-4 at the bottom of the spine column in body 1.

Estimated fuel and spine temperatures and EOL thermal and fast fluences for FTE-4 are given below:

EOL weighted average temperature, °C	553
Spine sample maximum temperature, °C	592
Spine sample minimum temperature, °C	531
Thermal fluence, 10^{25} n/m ² (E < 38 aJ)	0.80
Fast fluence, 10^{25} n/m ² (E ≥ 29 fJ).	0.45

4.10.2. Postirradiation Examination and Discussion

4.10.2.1. Visual Examination. The graphite crucible appeared to be in mint condition with no evidence of deterioration, cracking, distortion, etc. (see Fig. 100). Length and diameter measurements were made and are included in Fig. 101. No preirradiation measurements were made. The sample was easily removed from the graphite crucible with no evidence of sample-crucible mechanical interaction. A thick black layer was evident over most of the surface of the sample (see Fig. 100). This surface layer was tightly adherent and could not be removed by wiping or dipping in acetone. No other unusual observations were noted.

4.10.2.2. Dimensional Characterization. Postirradiation length and diameter measurements were performed using calibrated micrometer devices. The measurements and calculated dimensional changes are given in Table 59. All dimensions increased, indicating swelling, with average dimensional increases of:

$$\begin{array}{lcl} & \frac{\overline{\Delta L}}{L} = +1.0\% & , \\ \text{Top} & \frac{\overline{\Delta D}}{D} = +0.9\% & \left. \vphantom{\begin{array}{l} \text{Top} \\ \text{Bottom} \end{array}} \right\} \frac{\overline{\Delta D}}{D} = 1.2\% \\ \text{Bottom} & \frac{\overline{\Delta D}}{D} = +1.5\% & . \end{array}$$

Using the above average dimensions, the volume increase is calculated to be $+3.4\% = (\Delta L/L) + 2 (\Delta D/D)$. The average diameter and length values are about the same, indicating nearly isotropic swelling.

The large difference in swelling ($\sim 0.6\%$) between the "top" and "bottom" diameters should be noted. Unfortunately, the orientation of the sample was not noted on the drawing and was lost during disassembly of the crucible. However, some evidence does suggest the "top" diameter was oriented toward the bottom of the reactor. Thus, at least part of the swelling difference could be attributed to the fluence gradient from "top" to "bottom" of the sample (see Section 3.2). Other major factors which could influence swelling are temperature and boron concentration gradients.

4.10.2.3. Weight and Density Measurements. The weight of the sample was determined using a balance calibrated to ± 0.1 mg. The weight increased 0.0136 g (0.015%). The density measurement was performed using immersion-type density determination methods. The sample was weighed in air and acetone, and the density was calculated according to the following relationship:

$$P = \frac{W_a \cdot P_{ac}}{W_a - W_{ac}} , \quad (108)$$

where p = density of sample,
 p_{ac} = density of acetone,
 W_a = weight of sample in air,
 W_{ac} = weight of sample in acetone.

A density decrease of 2.7% was determined. This value agrees favorably with the calculated volume increase of 3.4% (weight change is negligible).

4.10.2.4. Boron Depletion and Predicted Dimensional Change. The B_{10} depletion was calculated using the following relationship:

$$\frac{N}{N_o} = 1 - \exp (-\sigma_a \cdot \phi_t) \quad , \quad (109)$$

where N = number of B_{10} atoms per cc remaining,
 N_o = initial number of B_{10} atoms per cc,
 σ_a = thermal neutron microscopic absorption cross section for B_{10}
 (1550 barns),
 ϕ_t = thermal fluence.

The B_{10} depletion contribution from fast flux was neglected, and the calculation assumes constant thermal flux (ϕ) throughout the irradiation period.

Calculations were performed for maximum, minimum, and average composition and fluence ranges and are summarized in Table 59.

The examination can be summarized as follows:

1. A black surface layer and a slight weight increase indicated that a graphite-stainless steel reaction may be occurring. The extent of the reaction is not known.

2. A density decrease of 2.7% and average diameter and length increases of 1.2% and 1.0%, respectively, were observed. Swelling appeared to be nearly isotropic.
3. The amount of swelling at the calculated boron depletion levels is less than design predictions.

Further work should determine the source(s) and extent of the reaction layer on the surface of the sample. Extrapolation of these relatively short time exposure data (~ 2 yr) to expected core life exposure time (~ 30 yr) will require precise definition of irradiation temperatures.

4.11. OTHER SPINE SAMPLES

Several spine samples from each fuel body were gamma scanned using the 0.12 in. by 0.12 in. collimator. A plot of Cs-137, Cs-134, and Ce-144 along the length of each of the spine samples is shown in Figs. 102, 103, and 104 for fuel bodies 1, 2, and 3. Because this counting geometry is not calibrated for quantitative results, the scans give only relative activity differences in the various samples. The scans show that spine samples which contained no fuel had no volatile cesium plateout on them.

Spine samples not covered by an analysis description within this report will be examined and reported elsewhere.

TABLE 19
ERRORS ASSOCIATED WITH FUEL TEST ELEMENT METROLOGY

	Symbol	Fuel Rod Diameter (in.)	Fuel Rod Length (in.)	Fuel Hole Diameter (in.)	Spine Hole Diameter (in.)	Fuel Body Outer Dia. (in.)	Sleeve Inner Dia. (in.)	Sleeve Outer Dia. (in.)	Fuel Stack Length (in.)	Plenum Length (in.)	Fuel Body Length (in.)	Element Length (in.)
Specified value	Pre X_1	0.490	1.940	0.500	1.095	2.740	2.752	3.485	27.16	2.63	31	144
Tolerance (3 σ)	Pre τ_1	± 0.004		± 0.001	+0.010, -0	+0, -0.002	± 0.002	± 0.005	(+25, -0.16) ± 0.02 from plenum or nominal hole depth	± 0.02 for stack		
Precision error	Pre e_1	$-0.003^{(a)} \pm 0.0001$	± 0.0001	± 0.0002	± 0.0002	± 0.0003	$\pm 0.0006^{(b)}$	$\pm 0.0016^{(b)}$	$\pm 0.006^{(b)}$	± 0.0003	± 0.0006	± 0.01125
	Post e_2	± 0.0001	± 0.0001	± 0.0002	± 0.0002	± 0.0003	± 0.0003	± 0.0003	± 0.0006	± 0.0003	± 0.0006	± 0.01125
Equipment random error (1 standard deviation)	Pre r_1	$\pm 0.0006^{(a)}$	± 0.0001	± 0.00015	± 0.00015	± 0.00035	$\pm 0.0007^{(b)}$	$\pm 0.0017^{(b)}$	$\pm 0.007^{(b)}$	± 0.00035	± 0.0020	± 0.01
	Post r_2	± 0.0001	± 0.0001	± 0.00015	± 0.00015	± 0.00035	± 0.00035	± 0.00035	± 0.002	± 0.00035	± 0.0020	± 0.01
Accuracy ($e + 2\sigma$)	Pre α_1	± 0.0013	± 0.0003	± 0.0005	± 0.0005	± 0.001	$\pm 0.002^{(b)}$	$\pm 0.005^{(b)}$	$\pm 0.020^{(b)}$	± 0.001	± 0.0046	± 0.03125
	Post α_2	± 0.0003	± 0.0003	± 0.0005	± 0.0005	± 0.001	± 0.001	± 0.001	± 0.0046	± 0.001	± 0.0046	± 0.03125
Typical FTE-4 Sample random errors	Pre s_1	± 0.0006	(c)	± 0.0001	0	± 0.0003	(c)	(c)	± 0.0266	± 0.0366	(c)	(c)
(1 standard deviation)	Post s_2	± 0.0013	± 0.0013	± 0.0004	(c)	± 0.0022	± 0.0015	± 0.0035	± 0.1392	± 0.1534	± 0.0034	(c)

(a) Fuel rod preirradiation metrology done by air gauging, which is -0.003 ± 0.0012 (2 σ) smaller than dial gauge metrology. Equipment random error may have included some sample random error.

(b) Derived from preirradiation tolerances, which are assumed to be 3 σ confidence statements.

(c) No preirradiation information available; postirradiation variances have been assumed as conservative estimate (and vice versa).

TABLE 20
FORMULA SUMMARY ON ERROR ANALYSIS

Error	Symbol	Measurements		Analysis Function			
		Preirradiation	Postirradiation	Difference	Relative Change (Radial Strain)	Average (Radial and Axial Strain)	Ratio (Anisotropy = Axial Versus Radial Strain)
		x_1	x_2	$x_{21} = x_2 - x_1$	$u = \frac{x_2}{x_1} - 1$	$v = \frac{1}{2} \left(\frac{x_2}{x_1} + \frac{y_2}{y_1} - 2 \right)$	$w = \left(\frac{y_2}{y_1} - 1 \right) \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
<u>Measuring System</u> Precision	e	$e_1 =$ $\pm e_1$	$e_2 =$ $\pm e_2$	$e_{21} =$ $\pm (e_1 + e_2)$	$e_u = e_x =$ $\pm \frac{1}{x_1} \left(e_1 \frac{x_2}{x_1} + e_2 \right)$	$e_v =$ $\pm \frac{1}{2} (e_x + e_y)$	$e_w =$ $\pm (e_y + e_x \cdot w) \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
Standard Error, Precision	p	$p_1 = \pm \frac{e_1}{\sqrt{12}}$	$p_2 = \pm \frac{e_2}{\sqrt{12}}$	$\pm (p_1^2 + p_2^2)^{1/2}$	$p_u = \pm \frac{1}{x_1} \left[p_1^2 \left(\frac{x_2}{x_1} \right)^2 + p_2^2 \right]^{1/2}$	$p_v = \pm \frac{1}{2} (p_x^2 + p_y^2)^{1/2}$	$p_w = \pm (p_y^2 + p_x^2 w^2)^{1/2} \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
Standard Error, Calibration at 67% Confidence (1 σ)	r	$r_1 =$ $\pm \left[\sum_i (r_i)^2 \right]^{1/2}$	$r_2 =$ $\pm \left[\sum_i (r_i)^2 \right]^{1/2}$	$r_{21} =$ $\pm (r_1^2 + r_2^2)^{1/2}$	$r_u = r_x =$ $\pm \frac{1}{x_1} \left[r_1^2 \left(\frac{x_2}{x_1} \right)^2 + r_2^2 \right]^{1/2}$	$r_v =$ $\pm \frac{1}{2} (r_x^2 + r_y^2)^{1/2}$	$r_w =$ $\pm (r_y^2 + r_x^2 \cdot w^2)^{1/2} \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
Accuracy (assumption) at 95% Confidence (2 σ)	α	$\alpha_1 =$ $\pm (e_1 + 1.96 r_1)$	$\alpha_2 =$ $\pm (e_2 + 1.96 r_2)$	$\alpha_{21} =$ $\pm (e_{21} + 1.96 r_{21})$	$\alpha_u = \alpha_x =$ $\pm (e_x + 1.96 r_x)$	$\alpha_v =$ $\pm (e_v + 1.96 r_v)$	$\alpha_w =$ $\pm (e_w + 1.96 r_w)$
Tolerance (assumption) at 99% Confidence (3 σ)	τ	$\tau_1 =$ $\pm 3 r_1$					
<u>Sample</u>		$j = 1$	$j = 2$				
Number of Measurements	n	n_1	n_2	$n_1; n_2$	$n_1; n_2$	$n_{x1}; n_{x2}; n_{y1}; n_{y2}$	$n_{x1}; n_{x2}; n_{y1}; n_{y2}$
Sample Mean ^(a)							
Column average	x, u, v, w	$\bar{x}_1$	$\bar{x}_2$	$\bar{x}_{21} = \bar{x}_2 - \bar{x}_1$	$\bar{u} = \left(\frac{\Delta x}{x_1} \right)$	$\bar{v} = \frac{1}{2} \left[\left(\frac{\Delta x}{x_1} \right) + \left(\frac{\Delta y}{y_1} \right) \right]$	$\bar{w} = \left(\frac{\Delta y}{y_1} \right) \left(\frac{\Delta x}{x_1} \right)^{-1}$
Line average				$\bar{x}_{21} = \bar{x}_2 - \bar{x}_1$	$\bar{u} = \frac{\bar{x}_2}{x_1} - 1$	$\bar{v} = \frac{1}{2} \left(\frac{\bar{x}_2}{x_1} + \frac{\bar{y}_2}{y_1} - 2 \right)$	$\bar{w} = \left(\frac{\bar{y}_2}{y_1} - 1 \right) \left(\frac{\bar{x}_2}{x_1} - 1 \right)^{-1}$
Standard Error on Sample ^(b,c) at 67% Confidence (1 σ)	s	$s_j =$ $\pm \left[\frac{1}{n_j - 1} \left(\sum_i (x_i^2) - n_j \bar{x}_j^2 \right) \right]^{1/2}$		s_3 from $\bar{x}_{21}$; s_{21} from $\bar{x}_{21}$ $s_{21} = \pm (s_1^2 + s_2^2)^{1/2}$	s_4 from $\bar{u}$; $s_u = s_x$ from $\bar{u}$ $s_u = \pm \frac{1}{x_1} \left[s_1^2 \left(\frac{x_2}{x_1} \right)^2 + s_2^2 \right]^{1/2}$	$s_v =$ $\pm \frac{1}{2} (s_x^2 + s_y^2)^{1/2}$	$s_w =$ $\pm (s_y^2 + s_x^2 w^2)^{1/2} \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
Combined Standard Error on Mean ^(c) at 67% Confidence (1 σ)	ϵ	$\epsilon_j =$ $\pm \left[r_j^2 + \sum_i \left(\frac{s_j^2}{n_j} \right) \right]^{1/2}$		$\epsilon_{21} =$ $\pm (e_1^2 + e_2^2)^{1/2}$	$\epsilon_u = \epsilon_x =$ $\pm \frac{1}{x_1} \left[e_1^2 \left(\frac{x_2}{x_1} \right)^2 + e_2^2 \right]^{1/2}$	$\epsilon_v =$ $\pm \frac{1}{2} (e_x^2 + e_y^2)^{1/2}$	$\epsilon_w =$ $\pm (e_y^2 + e_x^2 w^2)^{1/2} \left(\frac{x_2}{x_1} - 1 \right)^{-1}$
Total Error at 95% Confidence (2 σ)	d	dx_1 $\pm (1.96 \epsilon_1)$	dx_2 $\pm (1.96 \epsilon_2)$	dx_{21} $\pm (1.96 \epsilon_{21})$	du $\pm (1.96 \epsilon_u)$	dv $\pm (1.96 \epsilon_v)$	dw $\pm (1.96 \epsilon_w)$

(a) Line average preferred over column average.

(b) s_3 and s_4 from different averaging procedure; error progression is analog.

(c) Includes precision limits. Minimum standard errors on sample and sample mean are, respectively,

$$s_{\min} = \pm \frac{e}{\sqrt{12}} \text{ and } \epsilon_{\min} = \pm \left(e^2 + \frac{e^2}{12n} \right)^{1/2}$$

TABLE 21
FTE-4 COMPARISON OF FUEL STACK LENGTHS DERIVED FROM METROLOGY AND GAMMA SCANNING

	FTE-4-1 Bottom		FTE-4-2 Center		FTE-4-3 Top	
	Mean	2σ	Mean	2σ	Mean	2σ
Fuel stack length, L_1 (in.)	27.197	±0.0658	27.194	±0.022	27.199	±0.068
Relative change (%) via						
1. Gamma scan	-1.61	ND	-2.28	ND	-2.01	ND
2. Plenum, P	-1.46	±0.41	-1.28	±0.27	+0.35 ^(a)	±0.53 ^(a)
3. Fuel stack length, L	-1.23	±0.35	-1.90	±0.41	-1.71	±0.35
4. Fuel rod diameter, X	-1.58	±0.28	-2.10	±0.26	-2.00	±0.27
5. Fuel rod length, Y	-1.48	±0.20	-2.26	±0.17	-1.90	±0.19
6. Fuel rod mean $(X+Y)/2$ ^(a)	-1.53	±0.17	-2.18	±0.16	-1.95	±0.16
Grand Mean Change (%)	-1.47	±0.43 ^(b)	-1.96	±0.41 ^(b)	-1.90	±0.39 ^(b)

(a) Excluded from grand mean change.

(b)

$$\epsilon = \pm 1.96 \left\{ \frac{1}{n-1} \sum s_i^2 + \hat{s}^2 \right\}^{1/2}$$

TABLE 22
FTE-4 THERMOCOUPLE RESISTANCE MEASUREMENTS^(a)

	- to Ground Circuit (Ω)		- to Ground Circuit (Ω)		Loop (Ω)	
	Pre- irrad.	Post- irrad.	Pre- irrad.	Post- irrad.	Pre- irrad.	Post- irrad.
W-3% Re/W-25% Re thermocouple (ARI S/N 38247)	N/A	7.0	N/A	19.2	N/A	20.4
C/A thermocouple (ARI S/N 37047)	N/A	35.9	N/A	16.2	N/A	52.2

(a) Accuracy $\pm 2\Omega$.

TABLE 23
FTE-4 DISTANCE d FROM TOP OF SPINE TO EDGE OF HOLE

Body No.	Spine Stack Length (in.)	Preirrad., d_1 (in.)	Postirrad., d_2 (in.)	Δd (in.)	$\Delta G^{(a)}$ (in.)	$\frac{-\Delta d + \Delta G}{L}$ (%)
1	28.487	2.125	2.266	+0.141	-0.03145	-0.605
2	28.361	2.261	2.313	+0.052	-0.10685	-0.560
3	28.562	2.058	1.914	-0.144	-0.07590	+0.238

(a) ΔG = length change of graphite fuel body.

TABLE 24
FTE-4 SPINE SAMPLES

Position	Preirrad. Length (in.)	Preirrad. Weight (g)	Sample Type	Identification Number	Activity Reading
Body 1					
1	3.000	--	Boronated stainless steel	21	10 r (1 ft)
2	1.502	32.60	Stress relaxation	21	48 r (C)(a)
3	3.815	78.82	Mechanical test assembly	IV	10 r (C)(a)
4	1.258	25.04	Diffusion	61	7 r (1 ft)
5	1.261	24.80	Diffusion	52	10 r (2.5 ft)
6	1.257	24.69	Diffusion	41	4 r (1 ft)
7	1.256	24.59	Diffusion	31	9 r (C)(a)
8	1.255	24.88	Diffusion	12	2.8 r (1 ft)
9	1.258	24.68	Diffusion	10	10 r (C)(a)
10	1.256	24.70	Diffusion	4	10 r (2.5 ft)
11	1.258	25.01	Diffusion	64	5 r (1 ft)
12	1.261	24.65	Diffusion	54	10 r (1 ft)
13	1.257	24.63	Diffusion	46	10 r (1.5 ft)
14	1.260	24.71	Diffusion	34	3.5 r (1 ft)
15	1.253	24.86	Diffusion	35	8 r (C)(a)
16	1.255	24.69	Diffusion	27	4.5 r (1 ft)
17	1.255	24.62	Diffusion	18	5 r (1 ft)
18	1.257	24.80	Diffusion	6	25 r (1 ft)
19	1.313	29.83	Graphite spacer	26	8 r (C)(a)
	28.487				
Body 2					
1	2.495	57.00	Graphite spacer	27	20 r (C)(a)
2	1.263	25.00	Diffusion	58	10 r (1.5 ft)
3	1.256	24.65	Diffusion	21	4 r (1 ft)
4	1.257	25.05	Diffusion	67	4 r (1 ft)
5	1.258	24.39	Diffusion	28	65 r (1 ft)
6	1.257	24.10	Diffusion	37	3 r (C)(a)
7	1.256	22.80	Diffusion	49	100 mr (C)(a)
8	1.256	24.89	Diffusion	22	5 r (1 ft)
9	4.020	92.55	Advanced fuel rods	1-B	10 r (1.5 ft)
10	4.000	92.12	Advanced fuel rods	1-A	9 r (1.5 ft)
11	1.255	101.89	Fission product release	NB-2	8 r (1 ft)
12	1.250	24.75	Fission product release	10-P	1.5 r (1 ft)
13	1.258	102.03	Fission product release	NB-9	8 r (1 ft)
14	1.260	24.88	Fission product release	42-P	1 r (1 ft)
15	1.262	102.21	Fission product release	NB-13	8 r (1 ft)
16	1.259	24.61	Fission product release	146-P	2.5 r (1 ft)
17	1.499	--	Stress relaxation	10	0
	28.361				
Body 3					
1	1.800	40.29	Graphite spacer	29	250 mr (C)(a)
2	5.760	137.63	Thermal stability	1-G	15 r (2 ft)
3	5.750	137.38	Thermal stability	1-H	15 r (2 ft)
4	1.244	26.29	Thermal stability	2-16	2 r (1 ft)
5	1.250	24.06	Thermal stability	2-22	2 r (1 ft)
6	1.250	--	Thermal stability	2-28	1.5 r (1 ft)
7	1.248	--	Thermal stability	2-4	10 r (1 ft)
8	1.255	--	Thermal stability	2-10	8 r (1 ft)
9	1.253	--	Stress relaxation	45	300 mr (C)(a)
10	4.002	--	Hollow graphite spacer	28	100 mr (C)(a)
11	3.750	--	Absorber rod		10 r (2 ft)
	28.562				

(a) Contact.

TABLE 25
FTE-4 FUEL STACK DATA

		FUEL STACK LENGTH				FUEL HOLE PLFNUM LENGTH				INDIRECT FUEL STACK LENGTH			
FUEL BODY	FUEL HOLE	(IN) L1	(IN) L2	(IN) L21	% UL	(IN) P1	(IN) P2	(IN) P21	(IN) G2	(IN) G21	(IN) L1	% UL1	
1	1	27.1650	26.6026	-5.024	-1.6404	2.6380	3.0780	.4400	31.0735	29.8865	26.8085	-1.3122	
1	2	27.1790	26.6966	-4.824	-1.7749	2.6190	3.2030	.5840	31.0735	29.8865	26.6835	-1.8229	
1	3	27.2240	26.9253	-2.297	-1.0972	2.5760	2.4280	.2520	31.0735	29.8865	27.0585	-.6078	
1	4	27.2000	26.8769	-3.271	-1.2024	2.5920	2.4590	.2670	31.0735	29.8865	27.0275	-.6487	
1	5	27.2160	27.0121	-2.639	-.7492	2.5810	2.7580	.1770	31.0735	29.8865	27.1285	-.3214	
1	6	27.2556	27.0233	-2.317	-.8501	2.5410	2.7660	.2250	31.0735	29.8865	27.1205	-.4933	
1	7	27.1640	26.8531	-3.109	-1.1445	2.6360	2.9010	.2650	31.0735	29.8865	26.9855	-.6570	
1	8	27.1660	26.8373	-3.287	-1.2100	2.6340	2.9140	.2800	31.0735	29.8865	26.9725	-.7121	
STRATUM	L,P,G,U (IN OR %)	27.1966	26.8009	-3.357	-1.2344	2.6021	2.9134	.3112	31.0735	29.8865	26.9732	-.8216	
AVG	E (IN OR %)	.0060	.0006	.0066	.0240	.0003	.0003	.0006	.0006	.0006	.0009	.0251	
	R (IN OR %)	.0070	.0020	.0073	.0265	.0003	.0003	.0005	.0020	.0019	.0020	.0265	
STRATUM	S (IN OR %)	.0340	.1314	.1357	.4986	.0353	.1546	.1545	.0011	.0010	.1546	.5818	
1	EP (IN OR %)	.0139	.0465	.0485	.1783	.0125	.0547	.0561	.0021	.0021	.0547	.2074	
	D (IN OR %)	.0273	.0911	.0951	.3454	.0244	.1071	.1099	.0042	.0040	.1072	.4065	
2	1	27.1600	26.0649	-6.951	-2.5593	2.6260	2.8360	.2100	31.0021	29.8179	26.9819	-.6559	
2	2	27.1680	26.4537	-7.143	-2.6292	2.6260	3.0470	.4210	31.0021	29.8179	26.7709	-1.4618	
2	3	27.2660	26.7423	-4.637	-1.7044	2.5900	2.8280	.2380	31.0021	29.8179	26.9899	-.7944	
2	4	27.1900	26.7735	-4.205	-1.5463	2.6010	2.6950	.0940	31.0021	29.8179	27.1229	-.2616	
2	5	27.2580	26.8752	-3.628	-1.4004	2.5390	2.7970	.2580	31.0021	29.8179	27.0209	-.8700	
2	6	27.2060	26.8714	-3.746	-1.3749	2.5510	2.8200	.2690	31.0021	29.8179	26.9979	-.9107	
2	7	27.1510	26.6253	-5.257	-1.9302	2.6450	2.8670	.2220	31.0021	29.8179	26.9509	-.7371	
2	8	27.1620	26.6140	-5.480	-2.0175	2.6340	2.8750	.2410	31.0021	29.8179	26.9429	-.8068	
STRATUM	L,P,G,U (IN OR %)	27.1931	26.6775	-5.156	-1.8960	2.6015	2.8456	.2441	31.0021	29.8179	26.9722	-.8123	
AVG	E (IN OR %)	.0060	.0006	.0066	.0239	.0003	.0003	.0006	.0006	.0006	.0009	.0251	
	R (IN OR %)	.0070	.0020	.0073	.0263	.0003	.0003	.0005	.0020	.0019	.0020	.0265	
STRATUM	S (IN OR %)	.0413	.1658	.1709	.6274	.0392	.0986	.1061	.0060	.0058	.0987	.3931	
2	EP (IN OR %)	.0162	.0586	.0608	.2235	.0139	.0348	.0375	.0047	.0045	.0351	.1421	
	D (IN OR %)	.0318	.1149	.1193	.4380	.0272	.0683	.0735	.0092	.0088	.0689	.2785	
3	1	27.1770	26.5563	-6.207	-2.2839	2.6280	2.3980	-.2300	31.0311	29.8457	27.4477	.9961	
3	2	27.1930	26.5560	-6.370	-2.3025	2.6120	2.2890	-.3230	31.0311	29.8457	27.5567	1.3375	
3	3	27.2170	26.7784	-4.386	-1.5115	2.5880	2.6250	.0370	31.0311	29.8457	27.2207	.0136	
3	4	27.2100	26.7858	-4.202	-1.5590	2.5930	2.2420	-.3510	31.0311	29.8457	27.6037	1.4469	
3	5	27.2460	26.8903	-3.557	-1.3055	2.5580	2.4610	-.0970	31.0311	29.8457	27.3847	.5091	
3	6	27.2380	26.8958	-3.422	-1.2563	2.5660	2.6170	.0510	31.0311	29.8457	27.2287	-.0341	
3	7	27.1540	26.7093	-4.447	-1.6377	2.6490	2.3590	-.2900	31.0311	29.8457	27.4867	1.2253	
3	8	27.1540	26.6942	-4.618	-1.7005	2.6470	2.8520	.2050	31.0311	29.8457	26.9937	-.5976	
STRATUM	L,P,G,U (IN OR %)	27.1989	26.7333	-4.456	-1.7119	2.6051	2.4804	-.1247	31.0311	29.8457	27.3653	.6120	
AVG	E (IN OR %)	.0060	.0006	.0066	.0239	.0003	.0003	.0006	.0006	.0006	.0009	.0254	
	R (IN OR %)	.0070	.0020	.0073	.0263	.0003	.0003	.0005	.0020	.0019	.0020	.0269	
STRATUM	S (IN OR %)	.0355	.1314	.1361	.4999	.0347	.2046	.2075	.0045	.0044	.2046	.7638	
3	EP (IN OR %)	.0184	.0465	.0487	.1787	.0123	.0723	.0734	.0036	.0036	.0724	.2715	
	D (IN OR %)	.0282	.0912	.0954	.3503	.0241	.1418	.1438	.0074	.0071	.1420	.5322	
TEST	L,P,G,U (IN OR %)	27.1962	26.7572	-4.390	-1.6141	2.6029	2.7465	.1435	31.0356	29.8500	27.1036	-.3406	
ELEMENT	E (IN OR %)	.0060	.0006	.0066	.0239	.0003	.0003	.0006	.0006	.0006	.0009	.0252	
AVERAGE	R (IN OR %)	.0070	.0020	.0073	.0264	.0003	.0003	.0005	.0020	.0019	.0020	.0266	
	S (IN OR %)	.0355	.1582	.1621	.5056	.0309	.2464	.2488	.0323	.0311	.2483	.9224	
	EP (IN OR %)	.0101	.0324	.0339	.1244	.0071	.0503	.0508	.0132	.0127	.0519	.1943	
	D (IN OR %)	.0198	.0634	.0669	.2439	.0140	.0986	.0996	.0259	.0249	.1017	.3808	

TABLE 26
FTE-4 TOTAL ELEMENT DATA

Parameter	Value	Accuracy
Preirradiation length, L1 (in.)	144.3030	0.03125
Postirradiation length, L2 (in.)	144.8125	0.03125
Difference, L2 - L1 (in.)	0.50950	0.06250
Strain, L2/L1 - 1) (%)	0.35308	0.04339

Fuel Body	Fuel Body Weight							
	Preirradiation		Postirradiation		Difference		Relative Change	
	Weight (g)	Accuracy (g)	Weight (g)	Accuracy (g)	Weight (g)	Accuracy (g)	Value (%)	Accuracy (%)
1	4940.00	1.0000	4931.30	1.0000	-8.70	2.0000	-0.1761	0.0405
2	5112.00	1.0000	5107.80	1.0000	-4.20	2.0000	-0.0822	0.0391
3	4925.00	1.0000	4927.30	1.0000	+2.30	2.0000	+0.0467	0.0406

TABLE 27A
FTE-4 SLEEVE DATA (BASED ON O.D. MEASUREMENTS)

PRECISION:

DIAMETER = .00020

EQUIPMENT RANDOM ERROR:

DIAMETER = .00015

PRECISION:

DIAMETER = .00030

EQUIPMENT RANDOM ERROR:

DIAMETER = .00035

CORE HEIGHT	PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN					
	IN X1	NX1	MIL (a) SX1	MIL EX1	IN X2	NX2	MIL SX2	MIL EX2	IN X21	MIL SX21	MIL DX21	% UX	% EX	% RX	% SX	% EPX	% DX
7.16	3.4868	2	.577	.437	3.4892	2	.577	.541	.0024	.816	1.363	.070	.014	.011	.023	.020	.039
31.16	3.4868	2	.623	.467	3.4863	2	.623	.566	-.0005	.881	1.438	-.014	.014	.011	.025	.021	.041
55.16	3.4868	2	.354	.294	3.4809	2	.354	.434	-.0058	.500	1.029	-.168	.014	.011	.014	.015	.029
79.16	3.4868	2	.849	.620	3.4806	2	.849	.697	-.0062	1.200	1.829	-.178	.014	.011	.034	.027	.052
103.16	3.4868	2	.461	.361	3.4844	2	.461	.482	-.0024	.652	1.181	-.067	.014	.011	.019	.017	.034
119.73	3.4868	2	.597	.552	3.4900 ^(b)	2	.597 ^(c)	.552	.0032	.844	1.531	.092	.014	.011	.024	.022	.044
TEST ELEMENT AVG.	3.4868	12	.437	.237	3.4852	12	3.851	1.166	-.0016	3.876	2.332	-.046	.014	.011	.111	.034	.067

(a) Postirradiation sample random error used in absence of preirradiation data.

(b) Value of 3.4847 was rejected as being in error since it is so far from the expected value at this axial location.

(c) Average sample random error for remaining five sleeve O.D. measurement locations:

$$S_{\bar{X}} = \left(\frac{\sum_{j=1}^5 S_j^2}{5} \right)^{1/2}$$

TABLE 27B
FTE-4 SLEEVE DATA (BASED ON I.D. MEASUREMENTS)

PRECISION:

DIAMETER = .00020

EQUIPMENT RANDOM ERROR:

DIAMETER = .00015

PRECISION:

DIAMETER = .00030

EQUIPMENT RANDOM ERROR:

DIAMETER = .00035

CORE HEIGHT	PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN					
	IN X1	NX1	MIL SX1	MIL EX1	IN X2	NX2	MIL SX2	MIL EX2	IN X21	MIL SX21	MIL DX21	% UX	% EX	% RX	% SX	% EPX	% DX
24.75	2.7522	2	.283	.253	2.7516	2	.283	.408	-.0006	.400	.941	-.022	.018	.014	.015	.017	.034
72.24	2.7522	2	.283	.253	2.7500	2	.283	.408	-.0022	.400	.941	-.080	.018	.014	.015	.017	.034
119.73	2.7522	2	.696	.516	2.7533	2	.696	.607	.0011	.984	1.561	.040	.018	.014	.036	.029	.057
TEST ELEMENT AVG.	2.7522	6	.359	.211	2.7516	6	1.519	.713	-.0006	1.561	1.457	-.021	.018	.014	.057	.027	.057

TABLE 28
FTE-4 FUEL BODY 1 DIMENSIONAL CHANGE

PRECISION:					PRECISION:															
LENGTH = .00043					LENGTH = .00060															
DIAMETER = .00030					DIAMETER = .00030															
EQUIPMENT RANDOM ERROR:					EQUIPMENT RANDOM ERROR:															
LENGTH = .00200					LENGTH = .00200															
DIAMETER = .00035					DIAMETER = .00035															
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN							
CORE	OUTER	IN	MTL	MIL	IN	MTL	MIL	IN	MTL	MIL	%	%	%	%	%	%				
WEIGHT	DIAMETER	Y1	MX1	SX1	FX1	Y2	MX2	SX2	FX2	X21	SX21	DX21	UX	EX	RX	SX	FPX	DX		
29.59		2.7400	2	.000	.355	2.7379	2	.428	.467	-.0021	.428	1.150	-.077	.022	.018	.016	.021	.042		
31.59		2.7400	2	.000	.355	2.7374	2	.566	.535	-.0026	.566	1.259	-.095	.022	.018	.021	.023	.046		
33.59		2.7400	2	.000	.355	2.7375	2	.814	.677	-.0025	.814	1.498	-.091	.022	.018	.030	.028	.055		
35.59		2.7400	2	.000	.355	2.7368	2	1.061	.830	-.0033	1.061	1.769	-.119	.022	.018	.039	.033	.065		
36.59		2.7400	2	.000	.355	2.7380	2	.000	.355	-.0020	.000	.985	-.073	.022	.018	.000	.018	.036		
38.45		2.7400	2	.000	.355	2.7392	2	.537	.520	-.0008	.537	1.234	-.027	.022	.018	.020	.023	.045		
39.96		2.7400	2	.000	.355	2.7390	2	.000	.355	-.0010	.000	.985	-.036	.022	.018	.000	.018	.036		
41.96		2.7400	2	.000	.355	2.7381	2	.428	.467	-.0019	.428	1.150	-.069	.022	.018	.016	.021	.042		
43.96		2.7400	2	.000	.355	2.7382	2	.537	.520	-.0018	.537	1.234	-.064	.022	.018	.020	.023	.045		
45.96		2.7400	2	.000	.355	2.7385	2	.707	.613	-.0015	.707	1.389	-.055	.022	.018	.026	.026	.051		
47.96		2.7400	2	.000	.355	2.7381	2	.428	.467	-.0019	.428	1.150	-.069	.022	.018	.016	.021	.042		
49.96		2.7400	2	.000	.355	2.7360	2	1.414	1.061	-.0040	1.414	2.194	-.146	.022	.018	.052	.041	.080		
51.96		2.7400	2	.000	.355	2.7368	2	1.061	.830	-.0033	1.061	1.769	-.119	.022	.018	.039	.033	.065		
52.96		2.7400	2	.000	.355	2.7371	2	.639	.575	-.0029	.639	1.524	-.104	.022	.018	.023	.025	.048		
53.96		2.7400	2	.000	.355	2.7362	2	.537	.520	-.0038	.537	1.234	-.137	.022	.018	.020	.023	.045		
54.96		2.7400	2	.000	.355	2.7357	2	.537	.520	-.0043	.537	1.234	-.155	.022	.018	.020	.023	.045		
RDY	G	31.1050	2	1.061	2.140	31.0735	2	1.061	2.140	-.0315	1.500	5.930	-.101	.004	.009	.005	.010	.019		
AVG.	X	2.7400	32	.000	.350	2.7375	32	1.137	.404	-.0025	1.137	1.048	-.090	.022	.018	.041	.020	.038		
(R+X)/2														-.095	.013	.010	.021	.011	.021	
G/Y														112.64	31.8	24.8	52.3	26.8	52.5	

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FTE-4 FUEL BODY 2 DIMENSIONAL CHANGE

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TABLE 30
FTE-4 FUEL BODY 3 DIMENSIONAL CHANGE

PRECISION:				PRECISION:			
LENGTH = .00060				LENGTH = .00060			
DIAMETER = .00030				DIAMETER = .00030			
EQUIPMENT RANDOM ERROR:				EQUIPMENT RANDOM ERROR:			
LENGTH = .00200				LENGTH = .00200			
DIAMETER = .00035				DIAMETER = .00035			

CORF HEIGHT	OUTER DIAMETER	PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN					
		IN X1	NX1	MIL SX1	MIL FX1	IN X2	NX2	MIL SX2	MIL FX2	IN X21	MIL SX21	MIL DX21	% UX	% EX	% RX	% SX	% EPX	% DX
88.06		2.7400	2	.000	.355	2.7360	2	1.414	1.061	-.0040	1.414	2.194	-.146	.022	.018	.052	.041	.080
89.06		2.7400	2	.000	.355	2.7355	2	.707	.613	-.0045	.707	1.389	-.164	.022	.018	.026	.026	.051
90.06		2.7400	2	.000	.355	2.7335	2	2.121	1.542	-.0065	2.121	3.101	-.237	.022	.018	.077	.058	.113
91.06		2.7400	2	.000	.355	2.7327	2	2.475	1.786	-.0073	2.475	3.569	-.265	.022	.018	.090	.066	.130
93.06		2.7400	2	.000	.355	2.7320	2	.707	.613	-.0080	.707	1.389	-.292	.022	.018	.026	.026	.051
95.06		2.7400	2	.000	.355	2.7330	2	.000	.355	-.0070	.000	.985	-.255	.022	.018	.000	.018	.036
97.06		2.7400	2	.000	.355	2.7315	2	.707	.613	-.0085	.707	1.389	-.310	.022	.018	.026	.026	.051
99.06		2.7400	2	.000	.355	2.7320	2	.000	.355	-.0080	.000	.985	-.292	.022	.018	.000	.018	.036
101.06		2.7400	2	.000	.355	2.7340	2	1.471	1.099	-.0060	1.471	2.264	-.219	.022	.018	.054	.042	.083
103.06		2.7400	2	.000	.355	2.7320	2	1.471	1.099	-.0080	1.471	2.264	-.292	.022	.018	.054	.042	.083
104.56		2.7400	2	.000	.355	2.7327	2	.354	.434	-.0073	.354	1.109	-.265	.022	.018	.013	.020	.040
106.19		2.7400	2	.000	.355	2.7355	2	.707	.613	-.0045	.707	1.389	-.164	.022	.018	.026	.026	.051
108.19		2.7400	2	.000	.355	2.7362	2	.537	.520	-.0038	.537	1.234	-.137	.022	.018	.020	.023	.045
110.19		2.7400	2	.000	.355	2.7362	2	.537	.520	-.0038	.537	1.234	-.137	.022	.018	.020	.023	.045
112.19		2.7400	2	.000	.355	2.7370	2	.000	.355	-.0030	.000	.985	-.109	.022	.018	.000	.018	.036
113.19		2.7400	2	.000	.355	2.7377	2	.537	.520	-.0023	.537	1.234	-.082	.022	.018	.020	.023	.045
114.19		2.7400	2	.000	.355	2.7377	2	.537	.520	-.0023	.537	1.234	-.082	.022	.018	.020	.023	.045
115.19		2.7400	2	.000	.355	2.7380	2	.000	.355	-.0020	.000	.985	-.073	.022	.018	.000	.018	.036
116.19		2.7400	2	.000	.355	2.7385	2	.707	.613	-.0015	.707	1.389	-.055	.022	.018	.026	.026	.051
BODY AVG.	G X	31.1070	2	4.526	3.776	31.0311	2	4.526	3.776	-.0759	6.400	10.465	-.244	.004	.009	.021	.017	.034
	(G+X)/2	2.7400	38	.000	.350	2.7348	38	2.455	.530	-.0052	2.455	1.246	-.188	.022	.018	.090	.023	.045
	G/X												-.216	.013	.010	.046	.014	.028
													129.62	17.1	13.3	62.7	18.4	36.0

TABLE 31
FTE-4 FUEL BODY STRAIN SUMMARY

TEST ELEMENT FTE-4 FUEL BODY DATA																		
PRECISION:									PRECISION:									
LENGTH = .00030									LENGTH = .00030									
DIAMETER = .00030									DIAMETER = .00030									
EQUIPMENT RANDOM ERROR:									EQUIPMENT RANDOM ERROR:									
LENGTH = .00035									LENGTH = .00035									
DIAMETER = .00035									DIAMETER = .00035									
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN					
FUEL	LENGTH	IN	MI	ML		IN	MI	ML		IN	MI	ML	X	X	X	X	X	X
BODY	DIAMETER	G1	NG1	SG1	EG1	G2	NG2	SG2	EG2	G21	SG21	EG21	UG	EG	RG	SG	EPG	DG
		X1	MX1	SX1	EX1	X2	MX2	SX2	EX2	X21	SX21	EX21	UX	EX	RX	SX	EPX	DX
	LEN + DIAM ANISOTROPY												V	EV	RV	SV	EPV	DV
													W	EW	RW	SW	EPW	DW
1	G	31.1050	2	1.061	2.140	31.0735	2	1.061	2.140	-.0315	1.500	5.930	-.101	.004	.009	.005	.010	.019
	X	2.7400	32	.000	.350	2.7375	32	1.137	.404	-.0025	1.137	1.048	-.090	.022	.018	.041	.020	.038
	(G+X)/2												-.095	.013	.010	.021	.011	.021
	G/X												112.64	31.8	24.8	52.3	26.8	52.5
2	G	31.1090	2	5.992	4.687	31.0021	2	5.992	4.687	-.1069	8.474	12.992	-.343	.034	.009	.027	.021	.042
	X	2.7400	32	.000	.350	2.7341	32	2.273	.533	-.0059	2.273	1.250	-.217	.022	.018	.083	.023	.046
	(G+X)/2												-.280	.013	.010	.044	.016	.031
	G/X												158.50	17.8	13.8	62.0	19.6	38.5
3	G	31.1070	2	4.526	3.776	31.0311	2	4.526	3.776	-.0759	6.400	10.465	-.244	.004	.009	.021	.017	.034
	X	2.7400	38	.000	.350	2.7348	38	2.455	.530	-.0052	2.455	1.246	-.188	.022	.018	.090	.023	.045
	(G+X)/2												-.216	.013	.010	.046	.014	.028
	G/X												129.62	17.1	13.3	62.7	18.4	36.0
TEST	G	31.1070	6	3.830	1.604	31.0356	6	32.299	13.191	-.0714	32.526	26.045	-.230	.002	.002	.105	.043	.084
ELEMENT	X	2.7400	102	.000	.350	2.7354	102	2.515	.430	-.0046	2.515	1.086	-.166	.022	.018	.092	.020	.040
AVERAGE	(G+X)/2												-.198	.012	.009	.070	.024	.046
	G/X												138.04	19.3	15.0	98.8	30.7	60.1

TABLE 32

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL BODY STRAIN

FUEL BODY	CORE HEIGHT	STRAIN %	CALCULATED			MEASURED			RELATIVE DIFFERENCE			COMPARISON	
			S(1 SIGMA) %	E(1 SIGMA) %		S(1 SIGMA) %	E(1 SIGMA) %		CALC/MEAS-1 2(%)	S(1 SIGMA) 2(%)	E(1 SIGMA) 2(%)	TEST1 D-2/EZ	TEST2 D-2
1	29.59	-.005				-.077	.010	.021	-93.22	.91	1.86	-50.10	*****
	31.59	-.031				-.095	.015	.024	-67.14	5.36	8.27	-8.12	65.89
	33.59	-.061				-.091	.021	.027	-32.96	15.17	20.19	-1.63	2.67
	35.59	-.092				-.117	.028	.034	-21.47	19.09	22.64	-.95	.90
	36.59	-.105				-.073	.000	.018	43.28	.00	35.48	1.22	1.49
	38.45	-.129				-.029	.013	.022	338.12	193.61	333.18	1.01	1.03
	39.96	-.145				-.036	.000	.018	297.49	.00	196.75	1.51	2.29
	41.96	-.167				-.069	.010	.021	140.44	35.79	72.13	1.95	3.79
	43.96	-.187				-.066	.013	.022	184.26	55.83	96.04	1.92	3.68
	45.96	-.205				-.055	.018	.026	274.55	123.59	174.77	1.57	2.47
	47.96	-.222				-.069	.010	.021	220.11	47.65	96.02	2.29	5.25
	49.96	-.238				-.146	.036	.040	62.95	40.33	45.08	1.40	1.95
	51.96	-.253				-.117	.028	.034	116.60	52.65	62.39	1.87	3.49
	52.96	-.261				-.106	.015	.024	146.14	36.01	55.32	2.64	6.98
	53.96	-.269				-.139	.013	.022	93.29	17.98	30.93	3.02	9.10
	54.96	-.276				-.157	.013	.022	75.62	14.44	24.83	3.05	9.27
BODY AVERAGE													
		-.165				-.090	.018	.025	111.13	64.36	116.41	-2.33	164.40
RMS/ERROR/TESTS		.085				.037	.004	.006	125.67	31.42	16.09	9.34	1.64
2	59.96	-.279				-.091	.021	.027	205.90	69.22	91.96	2.24	5.01
	60.96	-.285				-.084	.013	.022	239.18	52.14	89.68	2.67	7.11
	61.96	-.290				-.139	.013	.022	109.38	19.48	33.50	3.27	10.66
	62.96	-.296				-.139	.013	.022	113.50	19.86	34.16	3.32	11.04
	64.96	-.306				-.182	.000	.016	67.92	.00	16.61	4.09	16.73
	66.96	-.315				-.201	.021	.027	57.10	16.16	21.46	2.66	7.08
	68.96	-.324				-.201	.039	.043	61.48	31.14	34.36	1.79	3.20
	70.96	-.330				-.237	.016	.026	39.30	10.61	14.99	2.62	6.87
	72.96	-.335				-.219	.000	.018	52.86	.00	12.60	4.20	17.61
	74.96	-.337				-.230	.013	.022	46.64	8.23	14.15	3.30	10.87
	76.96	-.339				-.248	.028	.034	36.63	15.63	18.52	1.98	3.91
	78.96	-.339				-.248	.013	.022	36.72	7.11	12.22	3.00	9.03
	80.96	-.338				-.266	.010	.021	26.90	4.92	9.90	2.72	7.38
	92.96	-.336				-.274	.021	.027	22.63	9.25	12.28	1.84	3.39
	85.21	-.333				-.328	.039	.043	1.36	11.95	13.18	.10	.01
	86.21	-.332				-.394	.028	.034	-15.84	6.06	7.18	-2.21	4.86
BODY AVERAGE													
		-.320				-.218	.021	.028	68.85	25.35	37.29	2.35	7.80
RMS/ERROR/TESTS		.021				.079	.005	.007	66.59	16.65	6.34	9.40	1.64

TABLE 32 (Continued)

3	88.06	-.333	-.146	.036	.040	127.89	56.40	63.05	2.03	4.11
	89.06	-.329	-.164	.018	.026	100.12	22.01	31.12	3.22	10.35
	90.06	-.325	-.237	.054	.057	36.84	31.26	32.95	1.12	1.25
	91.06	-.321	-.266	.065	.067	20.33	29.14	30.26	.67	.45
	93.06	-.312	-.292	.018	.026	6.70	6.60	9.33	.72	.52
	95.06	-.300	-.255	.000	.018	17.50	.00	8.30	2.11	4.45
	97.06	-.288	-.310	.018	.026	-7.32	5.40	7.63	-.96	.92
	99.06	-.272	-.292	.000	.018	-6.83	.00	5.76	-1.19	1.41
	101.06	-.254	-.219	.039	.043	16.14	20.53	22.65	.71	.51
	103.06	-.236	-.292	.039	.043	-19.22	10.71	11.82	-1.63	2.65
	104.56	-.221	-.266	.010	.021	-16.93	3.22	6.48	-2.61	6.82
	106.19	-.205	-.164	.018	.026	24.89	13.74	19.42	1.28	1.64
	108.19	-.181	-.139	.013	.022	30.73	12.16	20.92	1.47	2.16
	110.19	-.155	-.139	.013	.022	11.97	10.42	17.91	.67	.45
	112.19	-.127	-.109	.000	.018	15.81	.00	19.10	.83	.69
	113.19	-.111	-.084	.013	.022	32.24	20.33	34.96	.92	.85
	114.19	-.095	-.064	.013	.022	13.44	17.44	29.99	.45	.20
	115.19	-.079	-.073	.000	.018	8.82	.00	26.92	.33	.11
	116.19	-.064	-.055	.018	.026	16.26	38.36	54.25	.38	.09
	BODY AVERAGE	-.221	-.189	.027	.032	22.60	21.45	28.26	.55	2.09
	RMS/ERROR/TESTS	.090	.084	.006	.007	35.04	8.04	4.92	6.48	2.39
	ELEMENT AVERAGE	-.235	-.167	.023	.029	64.88	40.90	70.61	.21	54.80
	RMS/ERROR/TESTS	.096	.089	.003	.004	90.25	12.64	5.73	9.89	1.49

TABLE 33
FTE-4 SLEEVE-FUEL BODY GAPS (O.D.)

FUEL BODY	SLEEVE DIAMETER		FUEL BODY DIAMETER		GAP1		SLEEVE DIAMETER		FUEL BODY DIAMETER		GAP2		DIFFERENCE GAP2-GAP1		RELATIVE CHANGE GAP2/GAP1 - 1	
	BOL IN.	S.D. IN.	BOL IN.	S.D. IN.	BOL MIL	S.D. MIL	EOL IN.	S.D. IN.	EOL IN.	S.D. IN.	EOL MIL	S.D. MIL	VALUE MIL	S.D. MIL	VALUE %	S.D. %
1	2.7522	.0003	2.7400	.0000	6.10	.14	2.7497	.0025	2.7375	.0011	6.09	1.38	-.01	1.39	-.21	22.78
2	2.7522	.0003	2.7400	.0000	6.10	.14	2.7473	.0009	2.7341	.0023	6.60	1.23	.50	1.24	8.25	20.30
3	2.7522	.0007	2.7400	.0000	6.10	.35	2.7503	.0009	2.7348	.0025	7.75	1.31	1.65	1.35	27.10	22.60
ELEMENT AVERAGE																
X, SX	2.7522	.0004	2.7400	.0000	6.10	.18	2.7492	.0021	2.7354	.0025	6.92	1.64	.82	1.65	13.48	27.14
EX		.0002		.0003		.20		.0008		.0004		.46		.51		8.51
7.16	2.7522	.0003	2.7400	.0000	6.10	.14	2.7541	.0007	2.7400	.0000	7.05	.56	.95	.59	15.53	6.54
31.16	2.7522	.0003	2.7400	.0000	6.10	.14	2.7518	.0007	2.7374	.0006	7.20	.47	1.10	.49	18.08	8.20
55.16	2.7522	.0003	2.7400	.0000	6.10	.14	2.7475	.0005	2.7357	.0005	5.92	.36	-.16	.39	-2.93	6.37
79.16	2.7522	.0003	2.7400	.0000	6.10	.14	2.7473	.0009	2.7332	.0005	7.05	.53	.95	.55	15.62	9.10
103.16	2.7522	.0007	2.7400	.0000	6.10	.35	2.7503	.0009	2.7320	.0015	9.15	.87	3.05	.94	50.05	16.67
119.73	2.7522	.0007	2.7400	.0000	6.10	.35	2.7526	.0043	2.7400	.0000	6.30	2.15	.20	2.18	3.23	35.73
24.75	2.7522	.0003	2.7400	.0000	6.10	.15	2.7516	.0003	2.7400	.0000	5.80	.15	-.30	.21	-4.92	3.39
72.24	2.7522	.0003	2.7400	.0000	6.10	.15	2.7500	.0003	2.7340	.0000	8.00	.15	1.90	.21	31.15	4.06
119.73	2.7522	.0007	2.7400	.0000	6.10	.35	2.7533	.0007	2.7400	.0000	6.65	.35	.55	.49	9.02	8.49
EOL SLEEVE DIA. = BOL SLEEVE I.D. + EOL SLEEVE O.D./BOL SLEEVE O.D.																

TABLE 34
FTE-4 GRAPHITE WEB RADIAL GAMMA SCANNING RESULTS

Hole 7-3 ^(a)					Hole 2-6 ^(b)				
Scan Number	Radial Location (in.)	Isotope Relative Activity (cpm)			Scan Number	Radial Location (in.)	Isotope Relative Activity (cpm)		
		Cs-137 (661 keV)	Cs-134 (604 keV)	Cs-134 (795 keV)			Cs-137 (661 keV)	Cs-134 (604 keV)	Cs-134 (795 keV)
2229	0.0555	6	0	0	2116	0.063	14	22	12
2230	0.0115	5	0	0	2117	0.125	30	27	25
2231	0.081	6	0	0	2118	0.188	64	65	44
2232	0.151	8	0	0	2119	0.250	77	100	59
2233	0.221	7	0	0	2120	0.313	60	91	70
2234	0.292	8	0	7	2121	0.375	37	75	45
2235	0.363	4	18	7	2122	0.438	14	33	28
2236	0.436	5	0	0	2123	0.500	8	21	12
2237	0.508	6	0	0	2124	0.563	8	16	14
2238	0.581	5	10	0	2125	0.625	11	15	16
2239	0.655	5	12	11	2126	0.688	19	24	23
2240	0.729	11	14	0	2127	0.750	23	42	23
2241	0.803	6	11	9	2128	0.813	23	133	91
2242	0.877	7	32	8	2129	0.875	22	150	98
2243	0.952	7	34	26	2130	0.938	13	99	57
2244	1.027	6	6	6	2131	1.000	6	15	14
2245	1.102	7	0	0	2132	1.063	5	0	7
2246	1.177	5	0	6	2133	1.125	7	0	0
2247	1.243	6	0	0	2134	1.188	6	0	0
2248	1.328	8	0	0	2135	1.250	10	0	0
2249	1.404	4	0	0	2136	1.313	0	0	0
2250	1.480	0	0	0	2137	1.375	3	0	0
2251	1.556	6	0	0	2138	1.438	5	0	0
2252	1.632	0	0	0	2139	1.500	5	0	0
2253	1.708	8	0	0	2140	1.563	0	0	0
2254	1.784	9	19	18	2141	1.625	5	0	0
2255	1.861	8	25	18	2142	1.688	5	0	0
2256	1.936	10	0	6	2143	1.750	7	0	0
2257	2.006	7	0	9	2144	1.813	5	0	0
2258	1.082	8	0	8	2145	1.875	7	0	0
2259	2.158	7	10	8	2146	1.938	7	0	8
2260	2.234	7	0	0	2147	2.00	5	0	0
2261	2.310	8	0	7	2148	2.063	5	0	0
2262	2.386	9	0	0	2149	2.125	0	0	0
2263	2.462	0	0	0	2150	2.188	0	0	0
2264	2.538	6	0	0	2151	2.250	6	0	0
2265	2.614	10	0	0	2152	2.313	7	0	7
2266	2.690	8	0	0	2153	2.375	6	0	0
2267	2.766	4	0	0	2154	2.438	5	0	0
	2.842				2155	2.500	5	0	0
					2156	2.563	5	0	0
Counting error, σ (%)		30-60	30-80	30-80	2157	2.625	5	0	7
					2158	2.688	9	14	0
					2159	2.750	5	8	5
					2160	2.813	0	0	0

TABLE 35
FTE-4 COMPOSITION AND LOCATION OF BONDED FUEL RODS^(a)

Body	Hole	FOD Number	FMB Number	Kernel			Coating Type	As-Manufactured Coating Parameters							Total Particle Parameters				Fuel Rod Volume Fraction (%)	
				Type	Density (g/cm ³)	Nominal Diameter (μm)		Nominal Thickness (μm)					OPyC Density (g/cm ³)	OPyC OPTAF	SiC Density (g/cm ³)	Density (g/cm ³)	Diameter (μm)	Heavy Metal Content		
								Buffer	IPyC	SiC	OPyC	Total						wt % U		wt % Th
Fertile Particle																				
1,2,3	1,2	ED1258BIL(A)SL	4000-355	UO ₂	10.23	201	TRISO	53	27	23	37	140	1.80	1.07	3.20	2.62	472	25.70	--	13.8
	3,4	ET1272BIL(A)SL	4000-357	(Th,U)O ₂ ^(b)	9.07	211	TRISO	56	28	27	44	147	1.80	1.06	3.19	2.47	509	7.70	17.75	48.4
	5,6	ET1271BIL(A)SL	4000-357	(Th,U)C ₂	9.07	211	TRISO	56	28	27	44	147	1.80	1.06	3.19	2.47	509	7.70	17.75	48.7
	7,8	ET1274BIL(A)SL	4000-358	UC ₂	12.50	99	TRISO	57	27	27	33	137	1.79	1.11	3.20	2.26	382	9.85	--	35.9
Fissile Particle																				
1,2,3	1,2	T01236BL	4000-339	ThO ₂	9.95	410	BISO	62	--	--	81	142	1.78	1.09	--	3.42	690	--	53.40	47.8
	3,4	T1254BL	4000-345	ThC ₂	8.78	351	BISO	63	--	--	69	135	1.81	1.15	--	3.04	629	--	51.23	13.7
	5,6	CT6A56L	4000-335	ThC ₂	8.80	360	TRISO	56	26	28	41	145	1.79	1.13	3.20	3.15	663	--	44.15	13.3
	7,8	T1254BL	4000-345	ThC ₂	8.78	351	BISO	63	--	--	69	135	1.81	1.15	--	3.04	629	--	51.23	20.0

(a) All values are from data retrieval (FMB).

(b) Th/U ratio = 2.75.

TABLE 36
FTE-4 SUMMARY OF POSTIRRADIATION EXAMINATION OF FUEL RODS

Fuel Rod I.D. (a)	Fissile Particle (b)	Fertile Particle	Avg. Fuel Temp. (c) (°C)	Fast Fluency (n/m ² × 10 ²⁵) (E ≥ 29 fJ)	FIMA Fissile (%)	Fission Gas Release (d) (R/B)	Visual Appearance	Broken Particles on Surface	Dimensional Change (%)		Fissile Particles				Fertile Particles			
											OPyC Failure (%)	95% Confidence Limits P (%)	SiC Failure (%)	95% Confidence Limits P (%)	OPyC Failure (%)	95% Confidence Limits P (%)	SiC Failure (%)	95% Confidence Limits P (%)
									Diameter	Length								
2-1-7 (e)	(201 μm) UO ₂ TRISO	(410 μm) ThO ₂ BISO	1100	1.97	35.7	6.9 × 10 ⁻⁴	Good	43	-2.892	-2.871	0	0.0 ≤ P ≤ 8.3	4.5	0.3 ≤ P ≤ 15.7	0	0.0 ≤ P ≤ 9.2	--	--
2-1-8	(201 μm) UO ₂ TRISO	(410 μm) ThO ₂ BISO	1105	1.96	35.4		Excellent	19	-2.968	-2.734	--	--	--	--	--	--	--	--
2-2-7	(201 μm) UO ₂ TRISO	(410 μm) ThO ₂ BISO	1100	1.97	35.7	1.0 × 10 ⁻³	Good	57	-2.820	-2.759	0.6	0.1 ≤ P ≤ 4.1	12.9	9.8 ≤ P ≤ 21.7	0	0.0 ≤ P ≤ 2.0	--	--
2-3-7	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1088	1.97	35.7	3.9 × 10 ⁻⁵	Broken in two	--	-2.022	-1.719	--	--	--	--	--	--	--	--
2-3-8	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1094	1.96	35.4	1.6 × 10 ⁻⁵	Excellent	0	N/A	N/A	0.1	0.0 ≤ P ≤ 0.9	0.1	0.0 ≤ P ≤ 0.9	0	0.0 ≤ P ≤ 2.2	--	--
2-4-7	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1088	1.97	35.7	3.9 × 10 ⁻⁶	Excellent	1	-1.975	-1.775	--	--	--	--	--	--	--	--
2-4-8	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1094	1.96	35.4	2.9 × 10 ⁻⁶	--	--	N/A	N/A	--	--	--	--	--	--	--	--
2-4-9	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1098	1.95	35.0	3.7 × 10 ⁻⁶	--	--	N/A	N/A	--	--	--	--	--	--	--	--
2-4-10	(211 μm) (Th,U)C ₂ TRISO	(351 μm) ThC ₂ BISO	1102	1.95	34.7	4.4 × 10 ⁻⁶	--	--	N/A	N/A	--	--	--	--	--	--	--	--
2-5-7	(211 μm) (Th,U)C ₂ TRISO	(360 μm) ThC ₂ TRISO	1075	1.97	35.7	1.7 × 10 ⁻⁵	Excellent	3	-1.628	-1.658	5.0	2.4 ≤ P ≤ 10.0	1.4	0.4 ≤ P ≤ 5.1	4.0	2.0 ≤ P ≤ 8.0	0	0.05 ≤ P ≤ 2.1
2-6-7	(211 μm) (Th,U)C ₂ TRISO	(360 μm) ThC ₂ TRISO	1075	1.97	35.7	5.3 × 10 ⁻⁶	Good	6	-1.537	-1.803	--	--	--	--	--	--	--	--
2-7-7	(99 μm) UC ₂ TRISO	(351 μm) ThC ₂ BISO	1088	1.97	35.7	1.4 × 10 ⁻⁴	Good Cracked	10	-2.549	-2.013	--	--	--	--	--	--	--	--
2-8-7	(99 μm) UC ₂ TRISO	(351 μm) ThC ₂ BISO	1088	1.97	35.7	2.5 × 10 ⁻⁵	Excellent	0	-2.322	-2.312	0.7	0.1 ≤ P ≤ 1.6	0.5	0.2 ≤ P ≤ 2.0	1.2	0.4 ≤ P ≤ 3.5	--	--

(a) Body No. - Hole No. - Position No.

(b) Th/U ratio = 2.75. See Table 35.

(c) Calculated.

(d) Corrected for steady state.

(e) Radial section. All other rods were longitudinal sections.

TABLE 37
FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD STRAIN
HOLES 1 AND 2; UO₂ TRISO (22.4% FUEL VOLUME) AND ThO₂ BISO (77.6% FUEL VOLUME) FUEL

FUEL BODY	FUEL ROD	STRAIN %	CALCULATED		STRAIN %	MEASURED		RELATIVE DIFFERENCE CALC/MEAS-1 2(%)	RELATIVE DIFFERENCE		COMPARISON	
			S(1 SIGMA) %	E(1 SIGMA) %		S(1 SIGMA) %	E(1 SIGMA) %		S(1 SIGMA) %	E(1 SIGMA) %	TEST1 D=2/E2	TEST2 D**2
1	2	-1.909			-1.523	.033	.070	25.36	2.75	5.74	4.42	19.52
	5	-2.966			-1.693	.075	.097	75.22	7.76	10.02	7.51	56.37
	7	-3.311			-2.240	.056	.083	47.81	3.73	5.48	8.73	76.15
	10	-3.567			-2.428	.051	.079	46.93	3.07	4.80	9.78	95.72
	13	-3.679			-2.650	.132	.145	38.83	6.93	7.62	5.10	25.97
BODY AVERAGE												
RMS/ERROR/TESTS		-3.087			-2.107	.077	.099	46.83	5.27	6.99	7.11	54.75
		.638			.431	.035	.044	16.32	7.30	3.13	15.89	2.21
2	1	-3.720			-2.530	.049	.078	47.07	2.88	4.56	10.33	106.77
	2	-3.730			-2.767	.124	.138	34.82	6.04	6.72	5.18	26.86
	3	-3.739			-2.876	.057	.083	30.00	2.57	3.75	8.00	64.00
	4	-3.746			-2.625	.106	.122	42.70	5.76	6.63	6.44	41.45
	5	-3.748			-3.019	.077	.098	24.16	3.17	4.03	6.00	35.98
	6	-3.751			-2.844	.077	.098	31.87	3.57	4.54	7.02	49.26
	7	-3.752			-2.327	.091	.109	32.70	4.26	5.12	6.38	40.75
	8	-3.747			-2.852	.094	.112	31.39	4.32	5.14	6.11	37.32
	9	-3.739			-2.802	.082	.102	33.43	3.90	4.85	6.89	47.50
	10	-3.729			-2.856	.049	.078	30.57	2.26	3.57	8.56	73.24
	11	-3.719			-2.694	.128	.142	28.50	5.70	6.30	4.52	20.46
	12	-3.704			-2.853	.168	.179	29.83	7.66	8.14	3.66	13.42
	13	-3.687			-2.870	.076	.097	28.45	3.38	4.33	6.57	43.16
	14	-3.668			-2.727	.065	.089	34.50	3.22	4.38	7.87	61.99
BODY AVERAGE												
RMS/ERROR/TESTS		-3.727			-2.910	.095	.112	32.86	4.45	5.30	6.68	47.30
		.024			.116	.025	.030	5.64	1.51	1.42	25.00	1.70
3	2	-3.599			-2.781	.044	.075	29.42	2.04	3.48	8.45	71.34
	5	-3.452			-2.645	.035	.070	30.51	1.74	3.46	8.81	77.69
	7	-3.293			-2.674	.074	.096	23.15	3.43	4.42	5.23	27.40
	10	-2.927			-2.292	.086	.105	27.71	4.77	5.85	4.73	22.41
	13	-2.113			-2.019	.123	.138	4.65	6.40	7.14	.65	.42
BODY AVERAGE												
RMS/ERROR/TESTS		-3.077			-2.482	.079	.100	23.09	4.06	5.08	5.58	39.85
		.531			.284	.035	.045	9.55	4.27	2.27	12.47	2.21
ELEMENT AVERAGE												
RMS/ERROR/TESTS		-3.458			-2.595	.088	.107	33.73	4.56	5.65	6.54	47.30
		.495			.378	.018	.022	12.36	2.52	1.15	32.04	1.52

TABLE 38

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD STRAIN

HOLES 3 AND 4; UC₂ TRISO (78.1% FUEL VOLUME) AND ThC₂ BISO (21.9% FUEL VOLUME) FUELHOLES 7 AND 8; UC₂ TRISO (64.1% FUEL VOLUME) AND ThC₂ BISO (35.9% FUEL VOLUME) FUEL

FUEL BODY	FUEL ROD	STRAIN %	CALCULATED			STRAIN %	MEASURED			RELATIVE DIFFERENCE			COMPARISON	
			S(1 SIGMA) %	E(1 SIGMA) %			S(1 SIGMA) %	E(1 SIGMA) %		CALC/MEAS-1 Z(1)	S(1 SIGMA) Z(1)	E(1 SIGMA) Z(1)	TEST1 D=Z/E2	TEST2 D**2
1	2	-1.310				-1.799	.049	.079		64.04	10.00	16.13	3.97	15.76
	5	-1.907				-1.244	.121	.135		53.33	14.87	16.68	3.20	10.22
	7	-2.092				-1.585	.112	.127		32.01	9.30	10.59	3.02	9.14
	10	-2.234				-1.861	.059	.085		20.04	3.81	5.48	3.66	13.40
	13	-2.299				-1.887	.078	.099		21.87	5.06	6.40	3.42	11.67
BODY AVERAGE														
		-1.968				-1.475	.088	.107		38.26	9.46	12.01	3.45	12.04
RMS/ERROR/TESTS		.255				.410	.040	.048		17.50	7.83	4.23	7.72	2.21
2	2	-2.331				-2.122	.077	.098		9.87	7.98	5.08	1.95	3.78
	5	-2.343				-2.234	.103	.120		4.88	4.85	5.63	.87	.75
	7	-2.346				-2.087	.089	.108		12.39	4.77	5.79	2.14	4.58
	10	-2.330				-1.987	.057	.083		17.31	3.35	4.91	3.52	12.41
	13	-2.304				-1.955	.248	.255		17.84	14.94	15.38	1.16	1.35
BODY AVERAGE														
		-2.331				-2.077	.134	.147		12.46	7.70	8.39	1.93	4.57
RMS/ERROR/TESTS		.015				.100	.060	.066		4.83	2.16	3.44	4.31	2.21
3	2	-2.252				-2.288	.178	.188		-1.59	7.67	8.10	-.20	.04
	5	-2.169				-2.015	.109	.125		7.64	5.81	6.66	1.15	1.32
	7	-2.082				-2.080	.080	.101		.13	3.85	4.84	.03	.00
	10	-1.886				-1.731	.065	.089		8.94	4.06	5.59	1.60	2.56
	13	-1.432				-1.459	.059	.085		-1.87	3.96	5.71	-.33	.11
BODY AVERAGE														
		-1.964				-1.914	.107	.123		2.65	5.29	6.28	.45	.80
RMS/ERROR/TESTS		.293				.289	.048	.055		4.68	2.09	2.36	2.81	2.21
ELEMENT AVERAGE														
		-2.088				-1.822	.111	.127		17.79	7.68	9.20	1.94	5.81
RMS/ERROR/TESTS		.317				.390	.029	.033		18.51	4.78	1.98	7.53	1.67

TABLE 39

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD STRAIN
 HOLES 5 AND 6; (Th,U)C₂ TRISO (78.6% FUEL VOLUME) AND ThC₂ BISO (21.4% FUEL VOLUME) FUEL

FUEL BODY	FUEL ROD	STRAIN %	CALCULATED		STRAIN %	MEASURED		RELATIVE DIFFERENCE			COMPARISON	
			S(1 SIGMA) %	E(1 SIGMA) %		S(1 SIGMA) %	E(1 SIGMA) %	CALC/MEAS-1 Z(%)	S(1 SIGMA) Z(%)	E(1 SIGMA) Z(%)	TEST1 D=Z/EZ	TEST2 D**2
1	2	-1.585			-1.464	.030	.069	25.91	8.27	18.68	1.39	1.92
	5	-1.899			-1.958	.059	.085	4.87	7.17	10.40	.47	.22
	7	-1.007			-1.206	.092	.111	-16.50	6.37	7.66	-2.16	4.64
	10	-1.098			-1.259	.099	.116	-12.86	6.83	8.04	-1.60	2.56
	13	-1.145			-1.526	.111	.127	-24.98	5.45	6.23	-4.01	16.09
BODY AVERAGE												
RMS/ERROR/TESTS												
2	1	-1.165			-1.486	.081	.101	-21.59	4.25	5.34	-4.04	16.32
	2	-1.171			-1.563	.038	.072	-25.08	1.81	3.45	-7.27	52.88
	3	-1.176			-1.575	.073	.095	-25.33	3.45	4.51	-5.62	31.53
	4	-1.180			-1.661	.035	.071	-28.95	1.51	3.03	-9.56	91.45
	5	-1.182			-1.528	.048	.078	-22.65	2.43	3.94	-5.75	33.01
	6	-1.183			-1.665	.034	.070	-28.94	1.44	2.98	-9.72	94.46
	7	-1.184			-1.658	.062	.087	-28.58	2.68	3.76	-7.61	57.87
	8	-1.181			-1.722	.179	.189	-31.43	7.12	7.52	-4.18	17.48
	9	-1.176			-1.510	.074	.096	-22.14	3.84	4.97	-4.46	19.86
	10	-1.170			-1.602	.056	.083	-26.95	2.56	3.79	-7.12	50.64
	11	-1.165			-1.476	.023	.066	-21.09	1.25	3.52	-6.00	35.97
	12	-1.157			-1.675	.049	.078	-30.93	2.02	3.23	-9.56	91.41
	13	-1.149			-1.703	.038	.072	-32.54	1.49	2.64	-11.45	131.09
	14	-1.140			-1.541	.023	.066	-26.06	1.11	3.14	-8.29	68.75
BODY AVERAGE												
RMS/ERROR/TESTS												
3	2	-1.110			-1.587	.079	.100	-30.06	3.48	4.41	-6.82	46.55
	5	-1.055			-1.634	.063	.088	-35.45	2.49	3.47	-10.21	104.25
	7	-1.001			-1.418	.027	.067	-29.38	1.34	3.33	-9.82	77.71
	10	-1.067			-1.437	.041	.074	-38.28	1.77	3.17	-12.06	145.48
	13	-1.645			-1.363	.179	.189	-52.68	6.20	6.55	-8.04	64.66
BODY AVERAGE												
RMS/ERROR/TESTS												
ELEMENT AVERAGE												
RMS/ERROR/TESTS												

TABLE 40
FTE-4 HOLES 1 AND 2 FUEL ROD STRAIN SUMMARY

PRECISION:										PRECISION:									
LENGTH = .00010										LENGTH = .00010									
DIAMETER = .00010										DIAMETER = .00010									
EQUIPMENT RANDOM ERROR:										EQUIPMENT RANDOM ERROR:									
LENGTH = .00010										LENGTH = .00010									
DIAMETER = .00060										DIAMETER = .00010									
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN						
BUDDY HOLE	LENGTH DIAMETER LEN + DIAM ANISOTROPY	IN	NY1	MTL	MY1	IN	NY2	MTL	MY2	IN	MY21	MTL	UY	EY	RY	SY	EPY	DY	
		X1	NY1	SY1	EX1	X2	NY2	SY2	EX2	X21	SY21	DY21	UX	EX	RX	SX	EPX	DX	
1	1	Y	1.9450	10	2.674	.853	1.9042	10	2.484	2.685	-.041	8.897	5.521	-2.098	.010	.007	.457	.145	.283
		X	.4896	15	.601	.620	.4792	30	2.619	.488	-.010	2.687	1.547	-2.115	.040	.122	.548	.159	.312
		(Y+X)/2												-2.107	.025	.061	.357	.107	.211
		Y/X												99.19	2.4	5.7	33.6	10.1	19.8
2	1	Y	1.9465	28	3.152	.605	1.8930	28	4.304	.820	-.053	5.339	1.997	-2.747	.010	.007	.272	.052	.102
		X	.4894	43	.673	.609	.4753	84	1.668	.208	-.014	1.799	1.261	-2.874	.040	.121	.366	.128	.251
		(Y+X)/2												-2.810	.025	.061	.228	.069	.135
		Y/X												95.57	1.7	4.0	15.4	4.6	9.1
2	2	Y	1.9480	2	.319	.247	1.8942	2	.319	.247	-.054	.450	.685	-2.759	.010	.007	.023	.018	.035
		X	.4894	3	.578	.687	.4755	6	2.993	1.226	-.014	3.049	2.755	-2.823	.040	.121	.622	.285	.559
		(Y+X)/2												-2.791	.025	.061	.311	.143	.280
		Y/X												97.73	1.8	4.2	21.6	9.9	19.4
3	1	Y	1.9468	10	1.290	.020	1.8977	10	4.347	1.378	-.049	4.534	2.824	-2.524	.010	.007	.232	.074	.145
		X	.4891	15	.311	.605	.4771	30	2.675	.499	-.012	2.693	1.537	-2.440	.040	.121	.550	.158	.310
		(Y+X)/2												-2.482	.025	.061	.299	.087	.171
		Y/X												103.45	2.1	5.2	25.2	7.4	14.4
STRATHAM AVERAGE	Y	Y	1.9463	50	2.767	.404	1.8962	50	6.800	.967	-.050	7.342	2.054	-2.573	.010	.007	.376	.054	.105
		X	.4894	76	.613	.604	.4765	150	2.650	.238	-.013	2.720	1.273	-2.633	.040	.121	.555	.130	.254
		(Y+X)/2												-2.603	.025	.061	.335	.070	.138
		Y/X												97.71	1.9	4.5	25.1	5.2	10.2

TABLE 41
FTE-4 HOLES 3 AND 4 FUEL ROD STRAIN SUMMARY

		PRECISION:				PRECISION:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00010				DIAMETER = .00010													
		EQUIPMENT RANDOM ERROR:				EQUIPMENT RANDOM ERROR:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00060				DIAMETER = .00010													
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN						
BODY HOLE	LENGTH DIAMETER LEN + DIAM ANTISOTROPY	I*	MTI	MTI	MTI	IN	MTI	MTI	MTI	JN	MTI	MTI	%	%	%	%	%	%	
		Y1	NY1	SY1	EY1	Y2	NY2	SY2	EY2	Y21	SY21	OY21	UY	FY	RY	SY	EPY	DY	
		X1	FX1	SX1	EX1	X2	FX2	SX2	EX2	X21	SX21	OY21	UX	FX	RX	SX	EPX	DX	
				LEN + DIAM		ANTISOTROPY								V	EV	RV	SV	EPV	DV
														W	EW	RW	SW	EPW	DW
1	3	Y	1.9486	10	3.722	1.181	1.9233	10	7.300	2.311	-.025	8.194	5.087	-1.296	.010	.007	.419	.133	.260
		Y	.4914	16	1.671	.731	.4440	30	2.300	.432	-.007	2.843	1.644	-1.506	.040	.122	.576	.171	.335
		(Y+X)/2												-1.401	.025	.061	.356	.108	.212
		Y/X												86.08	3.0	7.0	43.1	13.2	25.8
2	3	Y	1.9526	10	4.834	1.532	1.9191	10	10.150	3.211	-.034	11.242	6.974	-1.716	.010	.007	.574	.182	.356
		Y	.4906	15	.795	.634	.4802	30	1.061	.218	-.010	1.326	1.315	-2.131	.040	.121	.268	.134	.263
		(Y+X)/2												-1.924	.025	.061	.317	.113	.221
		Y/X												80.50	2.0	4.6	28.8	9.9	19.4
2	4	Y	1.9490	2	.990	.707	1.9144	2	.990	.707	-.035	1.400	1.961	-1.775	.010	.007	.071	.051	.100
		Y	.4498	3	.183	.610	.4801	6	1.465	.607	-.010	1.477	1.686	-1.977	.040	.122	.301	.174	.341
		(Y+X)/2												-1.876	.025	.061	.155	.091	.178
		Y/X												89.80	2.4	5.5	14.2	8.3	16.3
3	3	Y	1.9472	10	2.075	.664	1.9116	10	4.370	1.385	-.036	4.837	3.011	-1.828	.010	.007	.248	.079	.154
		Y	.4906	16	1.783	.747	.4813	30	1.583	.306	-.009	2.384	1.583	-1.902	.040	.122	.481	.162	.317
		(Y+X)/2												-1.865	.025	.061	.270	.090	.176
		Y/X												96.11	2.6	6.2	27.6	9.2	18.0
STATION	Y	1.9404	32	4.143	.719	1.9178	32	8.632	1.529	-.032	9.575	3.329	-1.623	.010	.007	.490	.087	.170	
AVERAGE	Y	.4908	50	1.490	.635	.4817	96	2.324	.257	-.009	2.755	1.344	-1.856	.040	.122	.558	.137	.269	
	(Y+X)/2													-1.740	.025	.061	.371	.081	.159
	Y/X													87.47	2.5	5.7	37.3	8.0	15.7

TABLE 42
FTE-4 HOLES 5 AND 6 FUEL ROD STRAIN SUMMARY

		PRECISION:				PRECISION:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00010				DIAMETER = .00010													
		EQUIPMENT RANDOM ERROR:				EQUIPMENT RANDOM ERROR:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00060				DIAMETER = .00010													
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE			STRAIN						
BODY HOLE		TL	MTL	MTL	TL	MTL	MTL	MTL	IN	MJL	MJL	X	X	X	X	X			
		Y1	SY1	SY1	EY1	Y2	SY2	SY2	FY2	Y21	SY21	DY21	UY	EY	RY	SY	EPY	DY	
		X1	MX1	SX1	EX1	X2	MX2	SX2	FX2	X21	SX21	DX21	UX	EX	RX	SX	EPX	DX	
	LEN + DIAM ANISOTROPY												V	EV	RV	SV	EPV	DV	
													W	FW	RW	SW	EPW	DW	
1	5	Y	1.9512	10	2.110	.675	1.9297	10	6.565	2.078	-.022	6.896	4.283	-1.103	.010	.007	.353	.112	.219
		X	.4906	15	.501	.616	.4856	30	2.623	.489	-.005	2.679	1.542	-1.023	.041	.123	.546	.159	.312
		(Y+X)/2											-1.063	.025	.061	.325	.097	.191	
		Y/X											107.85	5.3	13.0	67.1	20.1	39.3	
2	5	Y	1.9531	28	2.624	.506	1.9217	28	3.766	.719	-.031	4.590	1.723	-1.609	.010	.007	.234	.045	.088
		X	.4901	43	1.260	.630	.4823	84	1.349	.178	-.008	1.846	1.283	-1.595	.040	.122	.374	.132	.258
		(Y+X)/2											-1.602	.025	.061	.220	.070	.136	
		Y/X											100.87	3.2	7.7	27.8	8.8	17.2	
2	6	Y	1.9520	2	.370	.281	1.9168	2	.370	.281	-.035	.524	.779	-1.803	.010	.007	.027	.020	.040
		X	.4905	3	.468	.658	.4829	6	1.531	.633	-.008	1.601	1.790	-1.539	.040	.122	.326	.185	.362
		(Y+X)/2											-1.671	.025	.061	.163	.093	.182	
		Y/X											117.15	3.7	9.3	24.9	14.1	27.7	
3	5	Y	1.9530	10	2.530	.806	1.9250	10	5.030	1.594	-.028	5.630	3.501	-1.435	.010	.007	.287	.091	.179
		X	.4900	16	2.165	.808	.4827	30	1.619	.312	-.008	2.703	1.698	-1.573	.040	.122	.546	.174	.341
		(Y+X)/2											-1.504	.025	.061	.308	.098	.193	
		Y/X											91.25	3.0	7.1	36.5	11.7	22.8	
STRATUM	Y	1.9527	50	2.529	.371	1.9238	50	5.675	.809	-.029	6.213	1.744	-1.481	.010	.007	.317	.045	.089	
AVERAGE	X	.4903	77	1.381	.620	.4830	150	2.139	.201	-.007	2.546	1.278	-1.474	.040	.122	.517	.131	.257	
	(Y+X)/2												-1.477	.025	.061	.303	.069	.136	
	Y/X												100.48	3.5	8.4	41.3	9.5	18.5	

TABLE 43
FTE-4 HOLES 7 AND 8 AND TOTAL ELEMENT FUEL ROD STRAIN SUMMARY

		PRECISION:				PRECISION:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00010				DIAMETER = .00010													
		EQUIPMENT RANDOM ERROR:				EQUIPMENT RANDOM ERROR:													
		LENGTH = .00010				LENGTH = .00010													
		DIAMETER = .00060				DIAMETER = .00010													
		PRE-IRRADIATION				POST-IRRADIATION				DIFFERENCE					STRAIN				
BODY HOLE	LENGTH	IN	MIL	MIL	MIL	IN	MIL	MIL	MIL	%	%	%	%	%	%				
	DIAMETER	Y1	NY1	SY1	FY1	Y2	NY2	SY2	FY2	Y21	SY21	NY21	UX	EX	RX	SY	EPY	DY	
	LEN & DIAM	Y1	NY1	SY1	FY1	Y2	NY2	SY2	FY2	Y21	SY21	NY21	UX	EX	RX	SY	EPX	DX	
	ANISOTROPY												V	EV	RV	SV	FPV	DV	
													W	EW	RW	SW	EPW	DW	
1	7	Y	1.9466	10	2.483	.792	1.9190	10	7.752	2.453	-.028	8.140	5.053	-1.419	.010	.007	.418	.132	.259
		Y	.4910	15	.641	.622	.4627	30	2.615	.488	-.008	2.692	1.550	-1.686	.040	.122	.548	.159	.312
		(Y+X)/2												-1.553	.025	.061	.344	.104	.203
		Y/X												84.20	2.6	6.1	36.9	11.2	21.9
2	7	Y	1.9465	8	2.170	.774	1.9068	8	1.812	.648	-.040	2.827	1.979	-2.039	.010	.007	.144	.051	.100
		Y	.4906	12	.352	.609	.4782	24	1.963	.413	-.012	1.994	1.442	-2.534	.040	.121	.406	.147	.289
		(Y+X)/2												-2.287	.025	.061	.215	.078	.153
		Y/X												80.46	1.7	3.8	14.1	5.1	10.0
2	8	Y	1.9510	2	.000	.102	1.9059	2	.000	.102	-.045	.000	.283	-2.312	.010	.007	.000	.007	.014
		Y	.4906	3	1.106	.915	.4792	6	1.121	.469	-.011	1.640	2.015	-2.324	.040	.121	.330	.206	.403
		(Y+X)/2												-2.318	.025	.061	.165	.103	.202
		Y/X												99.49	2.2	5.2	14.1	8.8	17.3
3	7	Y	1.9467	8	2.914	1.035	1.9143	8	8.159	2.886	-.032	8.664	6.010	-1.668	.010	.007	.444	.157	.308
		Y	.4902	12	.671	.631	.4798	24	2.213	.463	-.010	2.312	1.533	-2.123	.040	.122	.471	.157	.308
		(Y+X)/2												-1.896	.025	.061	.324	.111	.218
		Y/X												78.57	2.0	4.5	27.2	9.4	18.5
STRUT	Y	1.9469	28	2.610	.503	1.9132	28	8.176	1.548	-.034	8.582	3.191	-1.731	.010	.007	.440	.083	.164	
AVERAGE	Y	.4906	42	.673	.609	.4803	84	2.894	.331	-.010	2.972	1.359	-2.099	.040	.121	.605	.139	.272	
	(Y+X)/2													-1.915	.025	.061	.374	.081	.159
	Y/X													82.50	2.1	4.8	31.7	6.8	13.2

TABLE 44

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD-FUEL HOLE GAPS
HOLES 1 AND 2; UO₂ TRISO (22.4% FUEL VOLUME) AND ThO₂ BISO (77.6% FUEL VOLUME) FUEL

FUEL BODY	CORE HEIGHT	CALCULATED			MEASURED			RELATIVE DIFFERENCE			COMPARISON	
		GAP	S(1 SIGMA)	E(1 SIGMA)	GAP	S(1 SIGMA)	E(1 SIGMA)	CALC/MEAS-1	S(1 SIGMA)	E(1 SIGMA)	TEST1	TEST2
		MIL	MIL	MIL	MIL	MIL	MIL	Z(X)	SZ(X)	EZ(X)	D=Z/EZ	D**2
1	29.06	9.677			8.117	.140	.227	19.22	2.05	3.34	5.75	33.08
	34.89	12.065			9.083	.345	.389	32.83	5.05	5.69	5.77	33.26
	38.76	12.783			10.185	.223	.287	25.51	2.75	3.53	7.22	52.14
	44.61	13.258			10.561	.174	.250	25.54	2.06	2.97	8.60	73.99
	50.43	13.409			11.360	.330	.376	18.03	3.43	3.90	4.62	21.36
BODY AVERAGE												
		12.239			9.861	.256	.313	24.22	3.27	4.01	6.39	42.77
RMS/ERROR/TESTS		1.364			1.139	.114	.140	5.31	2.37	1.46	1.79	14.29
2	58.23	13.441			11.039	.240	.313	21.76	2.65	3.45	6.31	39.84
	60.17	13.437			11.206	.242	.314	19.92	2.59	3.36	5.93	35.13
	62.12	13.431			11.554	.221	.298	16.25	2.23	3.00	5.42	29.33
	64.06	13.421			11.095	.428	.473	20.97	4.67	5.15	4.07	16.56
	66.00	13.406			11.357	.341	.395	18.04	3.55	4.11	4.39	19.27
	67.94	13.391			11.574	.278	.343	15.70	2.78	3.42	4.58	21.01
	69.89	13.374			11.445	.337	.392	16.86	3.44	4.00	4.21	17.76
	71.83	13.349			11.608	.367	.418	15.00	3.64	4.14	3.62	13.12
	73.77	13.321			11.662	.312	.371	14.22	3.06	3.63	3.92	15.35
	75.71	13.291			11.818	.245	.316	12.47	2.33	3.01	4.15	17.19
	77.66	13.262			11.289	.586	.619	17.48	6.10	6.45	2.71	7.35
	79.60	13.227			11.547	.211	.291	14.55	2.10	2.88	5.05	25.46
	81.54	13.190			11.452	.295	.356	15.18	2.97	3.58	4.24	17.97
	83.48	13.150			11.467	.268	.334	14.68	2.68	3.34	4.39	19.26
BODY AVERAGE												
		13.335			11.436	.327	.383	16.65	3.36	3.93	4.50	21.04
RMS/ERROR/TESTS		1.094			.212	.087	.102	2.61	.70	.90	1.05	16.83
3	91.27	13.017			11.610	.261	.316	12.12	2.52	3.06	3.97	15.74
	97.10	12.738			11.059	.157	.238	15.19	1.63	2.48	6.12	37.41
	100.99	12.431			11.220	.377	.418	10.79	3.72	4.12	2.62	6.85
	106.82	11.677			10.040	.408	.445	16.30	4.72	5.16	3.16	9.99
	112.65	9.878			9.302	.604	.630	6.20	6.89	7.19	.86	.74
BODY AVERAGE												
		11.948			10.646	.391	.430	12.12	4.30	4.71	3.34	14.15
RMS/ERROR/TESTS		1.128			.849	.175	.192	3.57	1.60	1.93	2.11	7.48
ELEMENT AVERAGE												
		12.818			10.944	.328	.380	17.28	3.56	4.12	4.65	24.13
RMS/ERROR/TESTS		1.020			.922	.175	.192	5.32	1.09	1.93	2.11	7.48

TABLE 45

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD FUEL HOLE GAPS
 HOLES 3 AND 4; UC₂ TRISO (78.1% FUEL VOLUME) AND ThC₂ BISO (21.9% FUEL VOLUME) FUEL
 HOLES 7 AND 8; UC₂ TRISO (64.1% FUEL VOLUME) AND ThC₂ BISO (35.9% FUEL VOLUME) FUEL

FUEL BODY	CORE HEIGHT	GAP MIL	CALCULATED		GAP MIL	MEASURED		RELATIVE DIFFERENCE			COMPARISON	
			S(1 SIGMA) MIL	E(1 SIGMA) MIL		S(1 SIGMA) MIL	E(1 SIGMA) MIL	CALC/MEAS-1 Z(%)	S(1 SIGMA) SZ(%)	E(1 SIGMA) EZ(%)	TEST1 D=Z/EZ	TEST2 D+2
1	29.06	8.209			6.113	.192	.263	34.29	4.23	5.78	5.93	35.19
	34.89	9.469			6.671	.215	.280	41.94	4.57	5.96	7.04	49.58
	38.76	9.797			8.052	.256	.312	21.67	3.86	4.72	4.59	21.09
	44.61	9.991			8.619	.171	.248	15.92	2.31	3.34	4.77	22.74
	50.43	10.029			8.902	.233	.294	12.66	2.95	3.72	3.40	11.56
BODY AVERAGE		9.499			7.672	.216	.280	25.30	3.68	4.82	5.15	28.03
RMS/ERROR/TESTS		.675			1.094	.096	.125	11.12 4.97	1.65	2.16	11.51	2.21
2	60.17	10.010			9.934	.348	.402	.76	3.53	4.07	.19	.03
	66.00	9.963			9.898	.373	.423	.65	3.79	4.30	.15	.02
	69.89	9.928			9.780	.221	.298	1.51	2.30	3.09	.49	.24
	75.71	9.865			9.567	.157	.254	3.11	1.70	2.74	1.14	1.29
	81.54	9.802			9.630	.243	.314	1.79	2.56	3.32	.54	.29
BODY AVERAGE		9.913			9.762	.280	.344	1.57	2.88	3.55	.50	.38
RMS/ERROR/TESTS		.073			.144	.125	.154	.89 .40	1.29	1.59	1.12	2.21
3	91.27	9.718			8.901	.395	.434	9.17	4.85	5.33	1.72	2.97
	97.10	9.596			9.438	.252	.310	1.68	2.72	3.33	.50	.25
	100.99	9.464			9.295	.263	.319	1.82	2.88	3.49	.52	.27
	106.82	9.126			8.744	.252	.310	4.36	3.01	3.69	1.18	1.40
	112.65	8.209			8.152	.186	.259	.70	2.30	3.20	.22	.05
BODY AVERAGE		9.222			8.906	.278	.331	3.55	3.27	3.89	.83	.99
RMS/ERROR/TESTS		.544			.454	.124	.148	3.06 1.37	1.46	1.74	1.85	2.21
ELEMENT AVERAGE												
RMS/ERROR/TESTS		9.545			8.780	.260	.320	10.14	3.24	4.12	2.16	9.80
		.577			1.100	.124	.148	12.66 3.27	1.46	1.74	1.85	1.67

TABLE 46

FTE-4 COMPARISON OF MEASURED AND CALCULATED FUEL ROD FUEL HOLE GAPS
HOLES 5 AND 6; (Th,U)C₂ TRISO (78.6% FUEL VOLUME) AND ThC₂ BISO (21.4% FUEL VOLUME) FUEL

FUEL BODY	CORE HEIGHT	CALCULATED			MEASURED			RELATIVE DIFFERENCE			COMPARISON	
		GAP	S(1 SIGMA)	E(1 SIGMA)	GAP	S(1 SIGMA)	E(1 SIGMA)	CALC/MEAS-1	S(1 SIGMA)	E(1 SIGMA)	TEST1	TEST2
		MIL	MIL	MIL	MIL	MIL	MIL	Z(%)	SZ(%)	EZ(%)	D=Z/EZ	D**2
1	29.06	6.432			4.875	.154	.236	31.92	4.15	6.39	4.99	24.95
	34.89	7.001			5.900	.299	.349	18.65	6.02	7.02	2.66	7.06
	38.76	7.139			7.110	.194	.264	.40	2.73	3.73	.11	.01
	44.61	7.207			6.653	.476	.509	5.17	7.31	7.81	.66	.44
	50.43	7.201			7.786	.492	.524	-7.51	5.84	6.22	-1.21	1.46
BODY AVERAGE												
		6.996			6.505	.352	.395	9.73	5.45	6.38	1.44	6.78
RMS/ERROR/TESTS		.292			1.015	.157	.177	13.98	6.25	2.44	3.23	2.21
2	56.23	7.182			8.189	.367	.418	-12.30	3.93	4.48	-2.75	7.55
	60.17	7.168			8.655	.171	.263	-17.19	1.64	2.52	-6.83	46.67
	62.12	7.152			8.195	.298	.358	-12.72	3.17	3.82	-3.33	11.11
	64.06	7.135			8.727	.163	.258	-18.25	1.53	2.41	-7.56	57.15
	66.00	7.117			8.315	.083	.216	-14.41	.66	2.23	-6.47	41.82
	67.94	7.100			8.140	.128	.237	-12.78	1.37	2.54	-5.03	25.34
	69.89	7.082			8.207	.249	.319	-13.71	2.61	3.35	-4.09	16.73
	71.83	7.061			7.207	.606	.638	-2.02	8.23	8.67	-.23	.05
	73.77	7.042			8.311	.290	.352	-15.27	2.95	3.58	-4.26	18.15
	75.71	7.023			8.300	.175	.265	-15.39	1.78	2.70	-5.70	32.44
	77.66	7.004			8.180	.109	.227	-14.37	1.14	2.38	-6.04	36.44
	79.60	6.987			8.440	.110	.228	-17.60	1.07	2.22	-7.94	63.05
	81.54	6.971			8.309	.109	.227	-16.11	1.10	2.30	-7.01	49.18
	83.48	6.955			8.058	.118	.232	-13.69	1.27	2.48	-5.51	30.35
BODY AVERAGE												
		7.070			8.234	.253	.322	-13.99	2.99	3.65	-5.20	31.15
RMS/ERROR/TESTS		.073			.339	.068	.086	3.77	1.01	.80	.98	19.44
3	91.27	6.920			7.960	.249	.307	-13.06	2.72	3.36	-3.89	15.14
	97.10	6.866			8.500	.173	.249	-19.23	1.64	2.37	-8.11	65.84
	100.99	6.815			7.886	.135	.224	-13.58	1.48	2.46	-5.52	30.50
	106.82	6.679			8.065	.110	.210	-17.18	1.13	2.16	-7.95	63.19
	112.65	6.281			6.860	.278	.331	-8.43	3.72	4.42	-1.91	3.64
BODY AVERAGE												
		6.712			7.854	.200	.269	-14.30	2.34	3.07	-5.48	35.66
RMS/ERROR/TESTS		.230			.541	.089	.120	3.72	1.66	1.05	1.37	12.25
ELEMENT AVERAGE												
		6.980			7.795	.267	.329	-9.11	3.54	4.27	-3.87	27.01
RMS/ERROR/TESTS		.227			.896	.089	.120	12.05	2.46	1.05	1.37	12.25

TABLE 47A
FUEL NONHOMOGENEITY ANALYSIS OF FUEL ROD 2-1-7

		Fissile Particles	Fertile Particles
Section 1, %		40	60
Section 2, %		12	88
Section 3, %		7	93
Section 4, %		66	62
Total loading, %		39	62
Total particles		98	155
Theoretical loading, %		47	53

TABLE 47B
FUEL NONHOMOGENEITY ANALYSIS OF FUEL ROD 2-2-7

Section	Number of Particles	
	Fissile	Fertile
1	80	21
2	21	62
3	28	53
4	16	42
5	54	69
6	15	95
7	10	120
8	24	90
9	12	35
10	6	50
11	6	50
12	8	51
Total	280	738
Total loading, %	27.5	72.5
Theoretical loading, %	47	53

1	2	3	4
5	6	7	8
9	10	11	12

TABLE 48
FTE-4 FUEL ROD GAMMA SPECTROSCOPY SUMMARY

FUEL ROD I.D.	MEAN CORR HT (MM)	PA-233 ^(a) (311.9KEV) (CT)	RU-103 ^(a) (497.1KEV) (CT)	RU-106 (511.9KEV) (CT)	CS-134 (604.7KEV) (CT)	CS-137 (661.6KEV) (CT)	CE-144 (695.6KEV) (CT)	ZP-95 (724.2KEV) (CT)	CS-137 /ZR-95 (CT/MIN)	TOTAL FIMA %
1-3-5	788.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	5.62584-01 1.63541-01	2.86348-01 5.93121-02	4.76645-01 9.68704-02	6.92737+00 2.06436+00	2.21038+01 1.35683+01	.04584 .02662	4.92288 1.02372
1-3-6	936.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	5.59959-01 1.40442-01	3.05370-01 6.25852-02	5.06945-01 1.03028-01	7.84143+00 2.11579+00	3.92282+01 1.51577+01	.02747 .00910	5.23582 1.08880
1-3-7	985.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	6.08733-01 1.64250-01	3.73208-01 7.73036-02	5.44557-01 1.10672-01	8.41850+00 2.32850+00	2.88343+01 1.71559+01	.04014 .02251	5.62429 1.16958
2-3-3	1578.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.17560-01 2.05050-01	7.14119-01 1.45133-01	7.67056-01 1.54945-01	1.17967+01 3.10556+00	4.91233+01 2.78449+01	.03319 .01760	7.92231 1.63838
2-3-4	1627.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	9.31818-01 2.28255-01	7.81069-01 1.58740-01	8.39804-01 1.69640-01	1.25078+01 3.21342+00	3.81134+01 2.16041+01	.04684 .02484	8.47718 1.79377
2-3-5	1676.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.47132-01 2.07511-01	7.71332-01 1.56761-01	8.34783+01 1.68626-01	1.32459+01 3.32216+00	5.06167+01 3.15494+01	.03506 .02069	8.42649 1.78304
2-3-6	1726.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.55784-01 2.04860-01	6.93300-01 1.40902-01	7.28790-01 1.47216-01	1.19051+01 3.13409+00	4.04555+01 2.33104+01	.03829 .02069	7.52709 1.55665
2-3-7	1775.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.82948-01 2.16284-01	7.52059-01 1.52844-01	7.99906-01 1.61581-01	1.11093+01 2.85410+00	4.24967+01 2.32959+01	.04001 .02042	8.26159 1.70855
2-4-5	1676.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.43911-01 2.06722-01	7.65553-01 1.55586-01	8.37066-01 1.69087-01	1.25406+01 3.07190+00	4.69897+01 2.40201+01	.03787 .01781	8.64538 1.78792
2-4-6	1726.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	9.01229-01 2.15739-01	7.75401-01 1.57598-01	8.29628-01 1.67585-01	1.31523+01 3.14844+00	5.17669+01 2.55134+01	.03407 .01534	8.56856 1.77203
2-4-7	1775.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.96829-01 2.09950-01	7.59668-01 1.54390-01	8.35219-01 1.68714-01	1.27137+01 3.04344+00	3.83236+01 2.17232+01	.04633 .02457	8.62631 1.78398
2-4-8	1824.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	9.04873-01 2.16611-01	7.92036-01 1.60969-01	8.43321-01 1.70351-01	1.32355+01 3.16835+00	4.72720+01 2.37305+01	.03792 .01746	8.70999 1.80128
2-4-9	1874.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.71746-01 2.08681-01	7.40084-01 1.50410-01	7.91106-01 1.59803-01	1.19438+01 2.92571+00	3.33427+01 2.04672+01	.05043 .02927	8.17070 1.68975
2-4-10	1923.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	8.58541-01 2.00987-01	7.36025-01 1.49595-01	7.93094-01 1.60205-01	1.22453+01 2.99956+00	4.23352+01 2.36018+01	.03982 .02072	8.19123 1.69400
3-3-5	2467.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	6.95939-01 1.63210-01	4.83605-01 9.91142-02	6.32687-01 1.27803-01	9.81284+00 2.71417+00	2.80774+01 1.91030+01	.04790 .02115	6.53451 1.35138
3-3-6	2516.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	7.82296-01 1.96205-01	4.85291-01 9.94598-02	6.55422-01 1.32395-01	1.05003+01 2.63355+00	3.43529+01 2.27172+01	.04055 .02556	6.76933 1.39994
3-3-7	2565.00 2SIG ERR	0.00000 0.00000	0.00000 0.00000	6.55687-01 1.72614-01	4.14014-01 8.48516-02	5.94645-01 1.20116-01	9.77826+00 2.57419+00	2.81547+01 1.91556+01	.04489 .02920	6.14161 1.27012

TABLE 48 (Continued)

FUEL ROD 1.0-	PEAN CORE HT (MM)	PA-233 (a) (311.9KEV) (CI)	RU-103 (a) (497.1KEV) (CI)	RU-106 (511.9KEV) (CI)	CS-134 (604.7KEV) (CI)	CS-137 (661.6KEV) (CI)	CE-144 (695.6KEV) (CI)	ZR-95 (724.2KEV) (CI)	CS-137 /ZR-95 (CT/MIN)	TOTAL FIMA %
3-3-8	2615.00	0.00000	0.00000	5.95771-01	3.59641-01	5.39756-01	8.67955+00	3.89665+01	.02944	5.57470
	2SIG ERR	0.00000	0.00000	1.53061-01	7.37080-02	1.09031-01	2.22989+00	2.13608+01	.01503	1.15288
1-5-5	986.00	0.00000	0.00000	4.95867-01	2.83912-01	4.61739-01	7.13059+00	0.00000	.00000	5.04784
	2SIG ERR	0.00000	0.00000	1.33796-01	5.88074-02	9.38409-02	1.97227+00	0.00000	.00000	1.04971
1-5-6	936.00	0.00000	0.00000	4.26644-01	2.41406-01	4.23143-01	6.69711+00	1.80283+01	.04989	4.62590
	2SIG ERR	0.00000	0.00000	1.09610-01	5.00070-02	8.59970-02	1.80703+00	1.41726+01	.03795	.96197
1-5-7	985.00	0.00000	0.00000	6.34317-01	3.90421-01	5.64852-01	8.68415+00	3.19109+01	.03763	6.17510
	2SIG ERR	0.00000	0.00000	1.59091-01	8.00164-02	1.14100-01	2.23107+00	1.69009+01	.01845	1.27705
2-5-5	1676.00	0.00000	0.00000	9.06111-01	7.80584-01	8.33639-01	1.22446+01	3.33122+01	.05319	9.11354
	2SIG ERR	0.00000	0.00000	2.21958-01	1.58641-01	1.68395-01	3.07103+00	2.23466+01	.03406	1.88474
2-5-6	1726.00	0.00000	0.00000	9.18018-01	7.75767-01	8.41708-01	1.26447+01	5.13430+01	.03485	9.20176
	2SIG ERR	0.00000	0.00000	2.14910-01	1.57662-01	1.70025-01	3.24857+00	2.71926+01	.01709	1.90298
2-5-8	1824.40	0.00000	0.00000	8.79277-01	7.61011-01	8.29570-01	1.16927+01	3.90775+01	.04463	8.97066
	2SIG ERR	0.00000	0.00000	2.10483-01	1.54663-01	1.65755-01	2.93260+00	2.17856+01	.02323	1.85519
2-6-8	1824.04	0.00000	0.00000	8.97114-01	7.67152-01	6.03481-01	1.22064+01	4.59162+01	.03720	8.78385
	2SIG ERR	0.00000	0.00000	2.25002-01	1.55911-01	1.62303-01	3.13599+00	3.16790+01	.02456	1.81656
3-5-5	2467.00	0.00000	0.00000	7.05166-01	4.97071-01	6.23872-01	1.02008+01	3.25837+01	.04070	6.82032
	2SIG ERR	0.00000	0.00000	1.76860-01	1.01022-01	1.26022-01	2.62070+00	2.09274+01	.02484	1.41049

TABLE 48 (Continued)

FUEL POD I.D.	MEAN CORE HT (MM)	PA-233 (311.9KEV) (CT)	RU-103 (a) (427.1KEV) (CT)	RU-106 (511.9KEV) (CT)	CS-134 (604.7KEV) (CT)	CS-137 (661.6KEV) (CT)	CE-144 (695.6KEV) (CT)	ZR-95 (724.2KEV) (CT)	CS-137 /ZR-95 (CT/MIN)	TOTAL FIMA %
1-7-7	985.00	0.00000	0.00000	6.21719-01	3.92855-01	5.50781-01	9.28353+00	2.63996+01	.04435	7.31770
	2SIG ERR	0.00000	0.00000	1.52275-01	8.05151-02	1.11258-01	2.32837+00	1.67050+01	.02662	1.51335
2-7-6	1726.00	0.00000	0.00000	7.77410-01	6.88770-01	7.12689-01	1.12560+01	4.47305+01	.03387	9.46882
	2SIG ERR	0.00000	0.00000	1.99726-01	1.39991-01	1.43963-01	2.96321+00	2.74575+01	.01965	1.95821
2-7-7	1775.00	0.00000	0.00000	7.44674-01	6.95501-01	7.16669-01	1.17334+01	5.97948+01	.02548	9.52170
	2SIG ERR	0.00000	0.00000	1.96040-01	1.41349-01	1.44767-01	3.08890+00	3.05659+01	.01198	1.96915
2-7-8	1824.00	0.00000	0.00000	8.24892-01	7.26501-01	7.52897-01	1.18292+01	5.70917+01	.02803	10.00303
	2SIG ERR	0.00000	0.00000	2.06888-01	1.47650-01	1.52085-01	3.03906+00	3.12966+01	.01431	2.06869
2-7-9	1874.00	0.00000	0.00000	8.13804-01	6.93497-01	7.29232-01	1.12797+01	5.01266+01	.03088	9.67532
	2SIG ERR	0.00000	0.00000	2.04108-01	1.40942-01	1.47103-01	2.89788+00	3.26710+01	.01916	2.00092
2-8-6	1726.00	0.00000	0.00000	8.51562-01	7.68268-01	7.82932-01	1.22194+01	4.27370+01	.03894	10.40207
	2SIG ERR	0.00000	0.00000	2.13577-01	1.56138-01	1.58152-01	3.21683+00	2.62338+01	.02260	2.15122
2-8-8	1824.00	0.00000	0.00000	8.72591-01	7.44006-01	7.57246-01	1.17548+01	3.94013+01	.04085	10.06080
	2SIG ERR	0.00000	0.00000	2.24180-01	1.51207-01	1.52964-01	3.09453+00	2.56806+01	.02534	2.08064
2-8-9	1874.00	0.00000	0.00000	8.51562-01	7.69357-01	9.36160-01	1.26653+01	3.62973+01	.05482	12.40692
	2SIG ERR	0.00000	0.00000	2.03849-01	1.56359-01	1.89104-01	3.17655+00	2.26241+01	.03236	2.56583
2-8-10	1923.00	0.00000	0.00000	7.75777-01	7.74796-01	8.00706-01	1.21557+01	3.65212+01	.04660	10.63822
	2SIG ERR	0.00000	0.00000	1.94570-01	1.57465-01	1.61743-01	3.12295+00	2.07016+01	.02472	2.20005
2-8-11	1973.00	0.00000	0.00000	8.27936-01	7.14026-01	7.46634-01	1.10715+01	3.20716+01	.04948	9.91982
	2SIG ERR	0.00000	0.00000	2.02821-01	1.45114-01	1.50820-01	2.91465+00	1.81794+01	.02625	2.05148
3-7-7	2565.00	0.00000	0.00000	5.96123-01	3.69392-01	5.02019-01	7.95484+00	3.22871+01	.03305	6.66985
	2SIG ERR	0.00000	0.00000	1.46025-01	7.57065-02	1.01408-01	2.14639+00	2.28944+01	.02248	1.37937
3-7-8	2615.00	0.00000	0.00000	5.71925-01	3.53209-01	5.08596-01	8.00373+00	3.29706+01	.03279	6.75724
	2SIG ERR	0.00000	0.00000	1.43443-01	7.23897-02	1.02736-01	2.15958+00	1.89976+01	.01772	1.39744

(a) THESE ISOTOPES HAD DECAYED AWAY
AT THE TIME OF COUNTING.

TABLE 49
FTE-4 COMPARISON OF GAMMA SPECTROSCOPY DONE AT ORNL AND AT GA

Isotope	2 σ Counting Error at GA ^(a) (%)	Curies per Fuel Rod			
		3-5-5 ^(b) GA	3-5-6 ORNL	3-5-7 ORNL	3-5-8 ORNL
Zr-95	55	3.30	16.58	18.60	16.85
Ru-106	14	0.69	0.59	0.65	0.59
Cs-134	3	0.42	0.34	0.39	0.33
Cs-137	2	0.62	0.52	0.56	0.51
Ce-144	15	10.10	8.87	8.87	8.32

(a)

$$\% \text{ } 2\sigma \text{ error} = \frac{2[\text{CNT} + (1 - \text{CH}/2)^2] \cdot [(\text{LC} - \text{RC})/\sqrt{\text{PTS}}]^{-1/2}}{\text{PEAK}} \times 100,$$

where PEAK = counts in the peak,

CNT = total counts in peak (peak + background),

CH = number of channels in peak,

LC = counts in lowest-energy channel,

RC = counts in highest-energy channel,

PTS = number of smoothing points.

(b)

Fuel rod identification number.

TABLE 50
FTE-4 THERMAL STABILITY SPINE SAMPLES

Position ^(a)	Batch Number	Particle Type
1-G-1	4000-300	UC ₂ BISO
1-G-7	4503-63	UC ₂ TRISO
1-G-13	4423-3	UC ₂ TRISO
1-G-19	4000-229	(Th,U)C ₂ BISO
1-G-25	4503-27	(Th,U)C ₂ TRISO
1-G-31	4423-35	(Th,U)C ₂ BISO
1-G-37	4423-53	(Th,U)C ₂ TRISO
1-H-49	4000-238	ThC ₂ BISO
1-H-55	4503-25	ThC ₂ TRISO
1-H-61	4413-149	ThC ₂ BISO
1-H-67	4903-5	UC ₂ BISO
1-H-73	4632-137	(Th,U)C ₂ BISO
1-H-79	4493-149	ThO ₂ BISO
1-H-85	4000-195	ThC ₂ TRISO
1-H-4	4000-300	UC ₂ BISO
2-16-1	4000-238	ThC ₂ BISO
2-16-2	4000-225	ThC ₂ BISO
2-16-3	4000-242	ThC ₂ TRISO
2-16-4	4000-195	ThC ₂ TRISO
2-16-5	4503-25	ThC ₂ TRISO
2-16-6	4493-149	ThO ₂ BISO
2-22-1	4423-33	(Th,U)C ₂ BISO
2-22-2	4423-35	(Th,U)C ₂ BISO
2-22-3	4423-41	(Th,U)C ₂ TRISO
2-22-4	4423-53	(Th,U)C ₂ TRISO
2-22-5	4000-211	(Th,U)C ₂ TRISO
2-22-6	4000-200	(Th,U)C ₂ TRISO
2-28-1	4413-149	ThC ₂ BISO
2-28-2	4413-127	ThC ₂ BISO
2-28-3	3592-35	(Th,U)C ₂ TRISO

TABLE 50 (Continued)

Position ^(a)	Batch Number	Particle Type
2-28-4	4423-3	UC ₂ TRISO
2-28-5	4413-137	UC ₂ TRISO
2-28-6	4413-67	UO ₂ TRISO
2-4-1	4000-246	UC ₂ BISO
2-4-2	4000-300	UC ₂ BISO
2-4-3	4902-5	UC ₂ BISO
2-4-4	4503-63	UC ₂ TRISO
2-4-5 ^(b)	4000-302	UC ₂ TRISO ^(c)
2-4-6	4503-59	UO ₂ BISO
2-10-1	4000-229	(Th,U)C ₂ BISO
2-10-2	4000-227	(Th,U)C ₂ BISO
2-10-3	4000-232	(Th,U)C ₂ BISO
2-10-4	4632-137	(Th,U)C ₂ BISO
2-10-5	4503-27	(Th,U)C ₂ TRISO
2-10-6	4503-53	(Th,U)O ₂ BISO

(a) Crucible type - crucible number - hole number; all crucibles were located in body 3.

(b) Selected for initial examination. All other samples will be saved as historical samples until the end of the postirradiation examination.

(c) 100 μ m (VSM).

TABLE 51
FTE-4 METALLIC DIFFUSION SAMPLES

Sample No.	Element(s)	Approximate Metal Concentrations (mg/g matrix material)	Mean Activation-Energy-Weighted Temperature ^(a) (°C)	Mean Active Core Height (in.)
4	Cs	16.0	947	15.69
6	Cs	16.0	1018	25.74
10	Sr	1.0	938	14.43
12	Sr	10.0	918	13.17
18	Sr	10.0	1043	24.49
21	Sr	1.0	1108	34.70
22	Sr	10.0	1154	40.98
27	Sm	10.0	1054	23.23
28	Sm	10.0	1135	37.21
31	Ba	6.0	897	11.92
34	Ba	0.6	1026	20.72
35	Ba	6.0	1040	21.98
37	Ba	6.0	1143	38.48
41	Cs, Sr, Sm, Ba	Medium ^(b)	863 (Cs) 875	10.67
46	Cs, Sr, Sm, Ba	High ^(b)	999 (Cs) 1010	19.46
49	Cs, Sr, Sm, Ba	High ^(b)	1137 (Cs) 1148	39.72
52	Cs, Sr, Sm, Ba + 1% MgF	High ^(b)	840 (Cs) 853	9.41
54	Cs, Sr, Sm, Ba + 1% MgF	Medium ^(b)	983 (Cs) 994	18.20
58	Cs, Sr, Sm, Ba + 1% MgF	High ^(b)	1062 (Cs) 1072	33.44
61	Cs, Sr, Sm, Ba + 0.5% S	High ^(b)	808 (Cs) 822	8.14
64	Cs, Sr, Sm, Ba + 0.5% S	High ^(b)	965 (Cs) 970	16.95
67	Cs, Sr, Sm, Ba + 0.5% S	High ^(b)	1116 (Cs) 1127	35.95

(a) For Cs, $Q = 34,300$ cal/mole; for Sr, Ba, and Eu (Sm), $Q = 64,000$ cal/mole. See Section 3.3.1 for discussion of activation-energy-weighted temperatures.

(b) Same as above for each particular isotope.

TABLE 52
FTE-4 METALLIC DIFFUSION SAMPLE DATA

Crucible No.	Mean Active Core Height (cm)	Sample Temperature (a)				Neutron Fluence		Crucible Load		Matrix Source Loading		
		RMS Min. (°C)	Activation-Energy Weighted		RMS Max. (°C)							
			(°C)	10 ⁴ /T (K)		Thermal (b) (x 10 ²¹ nut)	Fast (c) (x 10 ²¹ nvt)	Metric Type	Element	Metal Conc. (d) (mg/g)	Metal Isotope	Isotope Conc. (d) (mg/g)
10	36.65	908 (893)	938 (927)	8.26 (8.33)	968 (941)	1.479	1.950	RC ^(e)	Sr (Cs) (g)	0.830 0	Sr-85 (Cs-134)	(f) (1.86-2)
12B	33.45	883 (872)	917 (907)	8.40 (8.47)	951 (941)	1.424	1.847	RC	Sr (Cs) (g)	1.59 0	Sr-85 (Cs-134)	(f) (1.14-2)
18B	62.20	1008 (996)	1043 (1032)	7.60 (7.66)	1078 (1068)	1.795	1.795	RC	Sr (Cs) (g)	1.59 0	Sr-85 (Cs-134)	(f) (1.52-2)
22B	104.09	1116 (1103)	1154 (1142)	7.01 (7.07)	1192 (1181)	1.936	2.786	RC	Sr (Cs) (g)	1.59 0	Sr-85 (Cs-134)	(f) (5.01-4)
31	30.28	868 (851)	897 (886)	8.55 (8.63)	926 921	1.364	1.743	RC	Ba (Cs) (g)	8.28 0	Ba-133 (Cs-134)	6.92-4 (1.19-2)
41	27.10	828 847 847 847	863 875 875 875	8.80 8.71 8.71 8.71	898 903 903 903	1.300	1.638	RC	Cs Sr Sm Ba	0.921 0.46 6.42 0.62	Cs-134 Sr-85 Eu-154 Ba-133	6.68-2 (f) 2.36-2 (f)

(a) For Cs, Q = 34,300 cal/mole; for Sr, Ba, and Eu, Q = 64,000. See Section 3.3.1 for a discussion of activation-energy-weighted temperatures.

(b) Thermal flux 0 < E < 2188 eV.

(c) Fast flux E > 0.18 MeV.

(d) Total metal loading.

(e) Furfuryl alcohol/resin coke matrix.

(f) Not detectable above background.

(g) Not initially loaded into matrix. Loading resulted from in-diffusion from adjacent fuel compacts.

TABLE 53
NUCLEAR TRANSFORMATIONS OF METAL DIFFUSANTS

Loaded Species	Isotopic Conc. (%)	Nuclear Reaction	Product Species
Cs-133	100	n, γ	Cs-134
Sr-84	82 (enriched)	n, γ	Sr-85
Sm-152	99 (enriched)	n, γ Sm-153 β Eu-153 n, γ	Eu-154
Ba (natural)	Ba-133 tagged	None	Ba-133

TABLE 54
DIFFUSION COEFFICIENTS FOR STRONTIUM, BARIUM, AND EUROPIUM
DERIVED FROM FTE-4 DIFFUSION SPINE SAMPLES

Crucible Number	Isotope	Activation- Energy- Weighted Temperatures (a)		$Dt^{(b)}$ (cm^2)	D_{slow} (cm^2/s)	$K/D^{(c)}$ (cm^{-2})	$D_{\text{eff}} \text{ (fast)}^{(d)}$ (cm^2/s)
		($^{\circ}\text{C}$)	$10^4/T$ (K)				
10	Sr-85	938	8.26	2.25-4	5.8-12	(e)	(e)
10				1.01-4	2.6-12	(e)	(e)
12	Sr-85	917	8.40	4.65-4	1.1-11	(e)	(e)
18	Sr-85	1043	7.60	8.14-3	2.1-10	(e)	(e)
18				5.04-3	1.3-10	(e)	(e)
22	Sr-85	1154	7.01	7.37-2	1.9-9	(e)	(e)
22				3.88-2	1.0-9	(e)	(e)
31	Ba-133	897	8.55	1.24-4	3.2-12	(e)	(e)
31				2.21-4	5.7-12	(e)	(e)
41	Eu-154	863	8.80	5.82-5	1.5-12	(e)	(e)
41				5.04-5	1.3-12	51.5	4.1-10

(a) Assuming $Q = 64,000$ cal/mole. See Section 3.3.1 for discussion of activation-energy-weighted temperatures.

(b) $t = 448.9$ days = 3.878×10^7 sec.

(c) Derived from CPROFIT "fast path" analysis.

(d) $D_{\text{eff}} = 1/(K/D)t$.

(e) Not applicable.

TABLE 55
REVISED ACTIVATION-ENERGY-WEIGHTED TEMPERATURES FOR FTE-3 SPINE SAMPLES^(a)

Crucible Number	Isotope	Activation-Energy-Weighted Temperature			
		Lower RMS (°C)	°C	10 / T (K)	Upper RMS (°C)
3	Cs-134	733	766	9.62	799
5	Cs-134	906	945	8.21	984
11	Sr-85	793	813	9.21	833
	Cs-134	768	803	9.29	838
16	Sr-85	926	945	8.21	964
	Cs-134	898	936	8.27	974
24	Eu-154	820	841	8.98	862
	Cs-134	795	831	9.06	867
30	Ba-133	847	868	8.76	889
42	Sr-85	872	893	8.58	914
	Ba-133	872	893	8.58	914
	Eu-154	872	893	8.58	914
	Cs-134	843	882	8.66	921
45	Sr-85	915	934	8.29	953
	Eu-154	915	934	8.29	953
	Cs-134	887	925	8.35	963
55	Sr-85	891	910	8.45	929
	Ba-133	891	910	8.45	929
	Eu-154	891	910	8.45	929
	Cs-134	866	902	8.51	938
63	Sr-85	904	923	8.36	942
	Ba-133	904	923	8.36	942
	Eu-154	904	923	8.36	942
	Cs-134	877	914	8.42	951
66	Sr-85	1045	1068	7.46	1091
	Ba-133	1045	1068	7.46	1091
	Eu-154	1045	1068	7.46	1091
	Cs-134	1009	1057	7.52	1105

^(a) For Cs, $Q = 34,300$ cal/mole; for Sr, Ba, and Eu, $Q = 64,000$ cal/mole.
See Section 3.3.1 for a discussion of activation-energy-weighted temperatures.

TABLE 56
DIFFUSION COEFFICIENTS FOR CESIUM DERIVED FROM FTE-4 DIFFUSION SPINE SAMPLES

Crucible Number(a)	Activation- Energy- Weighted Temperatures(b)		$C_m(c)$ (mg/g)	J (mg/cm ² -s)	$K/D(d)$ (cm ⁻²)	$C_l(e)$ (mg/g)	D_{eff} (cm ² /s)	$C_g(f)$ (mg/g)	$\phi(g)$ (cm/C _g)	$D(h)$ (cm ² /s)
	°C	$10^4/T$ (K)								
10(P) ^(h) Profile	927	8.33	1.86-2	--	4.8	5.0-4	4.5-9	3.0-5	6.20+2	4.9-12
Flux				6.83-11			3.6-8			
12(P) Profile	907	8.47	1.14-2	--	4.0	7.0-4	5.3-9	6.8-6	1.67+3	6.5-12
Flux				4.18-11			1.6-8			
18(P) Profile	1032	7.66	1.52-2	--	2.9	3.0-4	7.3-9	5.0-5	3.04+2	--
Flux				5.78-11			5.0-8			
22b(P) Profile	1142	7.07	5.01-4	--	1.5	2.0-4	1.4-8	9.0-6	5.60+1	--
Flux				1.84-12			2.4-9			
31(P) Profile	886	8.63	1.19-2	--	4.5	4.0-4	4.7-9	7.5-6	1.59+3	2.8-12
Flux				4.37-11			2.9-8			
41(P) Profile	863	8.80	6.68-2	--	6.4	0--	3.3-9	1.7-5	3.93+3	1.8-12

(a) P = center post; Profile = D_{eff} derived from CPROFIT profile analyses; Flux = D_{eff} derived from time-averaged flux calculations.

(b) Assuming $Q = 34,300$ cal/mole. See Section 3.3.1 for a discussion of activation-energy-weighted temperatures.

(c) Matrix concentration of cesium.

(d) Derived from CPROFIT analysis: $D_{eff} = [(K/D)(t)]^{-1}$.

(e) Extrapolated cesium concentration at the outer crucible wall boundary.

(f) Extrapolated cesium concentration in the graphite crucible wall adjacent to the matrix material.

(g) Partition factor.

(h) Slow component of cesium profile.

TABLE 57
REFERENCE ISOTHERM^(a) CONSTANTS FOR GA MATRIX MATERIAL AND GRAPHITE

Material	A	B	D	E	C_t ($\mu\text{mole/g}$)
Matrix	8.39	-3.83+4	-0.50	4.10+3	4.0
Graphite	7.09	-2.76+4	-0.50	4.10+3	0.30

(a) The relationship between vapor pressure and concentration for metallic fission products on graphite and fuel rod matrix is given by the Freudlich (F) isotherm:

$$\ln p(F) = (A + B/T) + (D + E/T) \ln C \quad , \quad (a)$$

where $p(F)$ = equilibrium partial pressure of the metallic element (atm),

C = concentration ($\mu\text{mole/g}$ carbon),

A, B, D, E = constants.

Equation (a) is valid for concentrations greater than a fixed value, C_t , the transition concentration. Below this concentration, the relationship between vapor pressure and concentration for metallic fission products in graphite and fuel rod matrix is given by the Henrian (H) isotherm:

$$\ln p(H) = (A + B/T) + (D - 1 + E/T) \ln C_t + \ln C \quad . \quad (b)$$

TABLE 58
FTE-4 DIMENSIONS AND WEIGHTS OF B₄C-GRAPHITE PELLETS

Pellet Number	Weight (g)	Length at Center Line (in.)	Diameter (in.)			
			Top		Bottom	
			0°	90°	0°	90°
1 } Preirradiation } Postirradiation Δ(%) ^(b)	1.6625	0.4337	0.4480 ^(a)	0.4175	0.4183	0.4157
	1.6203	0.4266	0.4152	0.4130	0.4150	0.4127
	-2.57	-1.65	ND	-4.08	-0.79	-0.72
2 } Preirradiation } Postirradiation Δ(%)	1.7503	0.4432	0.4352	0.4360	0.4352	0.4362
	1.7215	0.4376	0.4291	0.4307	0.4297	0.4306
	-1.65	-1.26	-1.40	-1.22	-1.22	-1.23
3 } Preirradiation } Postirradiation Δ(%)	1.5352	0.4446	0.4212	0.4200	0.4216	0.4216
	1.5137	0.4364	0.4174	0.4173	0.4163	0.4164
	-1.40	-1.84	-0.90	-0.64	-1.26	-1.23
4 } Preirradiation } Postirradiation Δ(%)	1.6314	0.4370	0.4087	0.4190	0.4079	0.4141
	1.6177	0.4361	0.4063	0.4189	0.4061	0.4135
	-0.84	-0.21	-0.59	-0.02	-0.44	-0.14

(a) Believed to be in error based on other measurements at different locations on the same compact.

(b) Δ is the percentage change from preirradiation to postirradiation measurements.

TABLE 59
FTE-4 BORONATED STEEL SAMPLE: DEPLETION CALCULATION AND PREIRRADIATION
AND POSTIRRADIATION CHARACTERIZATION

A. Depletion Calculation					
Boron (wt %)	Number of B atoms/cm ³	Number of B ₁₀ atoms/cm ³ (a)	Thermal Fluence	B ₁₀ Depletion (%)	B ₁₀ Depletion (atoms/cm ³)
2.0	81 x 10 ²⁰	16 x 10 ²⁰	0.7 x 10 ²¹	66	11 x 10 ²⁰
2.5	101 x 10 ²⁰	20 x 10 ²⁰	0.7 x 10 ²¹	66	13 x 10 ²⁰
2.0	81 x 10 ²⁰	16 x 10 ²⁰	0.9 x 10 ²¹	75	12 x 10 ²⁰
2.5	101 x 10 ²⁰	20 x 10 ²⁰	0.9 x 10 ²¹	75	15 x 10 ²⁰
Avg. 2.25	91 x 10 ²¹	18 x 10 ²⁰	0.80 x 10 ²¹	71	13 x 10 ²⁰

(a) Assumes natural boron is 19.8% B₁₀.

B. Preirradiation and Postirradiation Examination					
Dimension	Preirrad.	Postirrad. (a)	Change	% Change	
L ₁ in. (cm)	1.884 (4.785)	1.9034 (4.8346)	+0.019 +(0.50)	+1.0	$\left\{ \frac{\overline{\Delta D}_1}{D_1} = 0.88\% \right.$
L ₂ in. (cm)	1.902 (4.831)	1.9232 (4.8849)	+0.021 +(0.053)	+1.1	
D ₁ Top, in. 0° (cm)	0.6062 (1.540)	0.6116 (1.554)	+0.0054 +(0.014)	+0.89	
90°	0.6062 (1.529)	0.6115 (1.553)	+0.0053 +(0.013)	+0.87	
D ₂ Mid., in. 0° (cm)	(b)	0.7724 (1.962)	--	--	$\left\{ \frac{\overline{\Delta D}_3}{D_3} = 1.5\% \right.$
90°	(b)	0.7604 (1.931)	--	--	
D ₃ Bot., in. 0° (cm)	0.5915 (1.502)	0.6012 (1.527)	+0.0097 +(0.024)	+1.6	
90°	0.5915 (1.502)	0.5998 (1.523)	+0.0083 +(0.021)	+1.4	
Weight, g	92.8245	92.8381	+0.0136	+0.0147	
Density, g/cm ³	7.5109	7.31	-0.20	-2.7	

(a) Accuracies: Diameter ±0.0002 in.
Length ±0.001 in.

(b) No preirradiation measurements performed. Nominal diameter = 0.750 in.

Fig. 17. FTE-4 gamma scan trace

K7410 (3-14)

Fig. 18. FTE-4 full-length mosaic

K7410 (Top 21-24, Bottom 25-28)

Fig. 19. FTE-4 composite photograph of fuel body 1; top, 180° line and bottom, 0° line

K7410 Top 35-42, Bottom 43-48)

Fig. 20. FTE-4 composite photograph of fuel body 2; top, 90° line and bottom, 270° line

K7410 (Top 69-74, Bottom 77-82)

Fig. 21. FTE-4 composite photograph of fuel body 3; top, 0° line and bottom, 180° line

Fig. 22. FTE-4 fuel body sectioning plan

K7410-101 (a)

K7410-99 (b)

K7410-98 (c)

K7410-96 (d)

Fig. 23. FTE-4 fuel body 1 sections

K7410-105 (a)

K7410-103 (b)

K7410-104 (c)

K7410-102 (d)

Fig. 24. FTE-4 fuel body 2 sections

K7410-106 (a)

K7410-107 (b)

K7410-108 (c)

K7410-109 (d)

Fig. 25. FTE-4 fuel body 3 sections

Fig. 26. FTE-4 measured and calculated sleeve strain

Fig. 27. FTE-4 measured and calculated fuel body strain

Fig. 28. FTE-4 measured and calculated sleeve - fuel body gap

Fig. 29. FTE-4 autoradiography of slice 1-3 from body 1, located ~ 18 in. from bottom of active core

Fig. 30. FTE-4 autoradiography of slice 2-2 from body 2, located ~34 in. from bottom of active core

Fig. 31. FTE-4 autoradiography of slice 3-2 from body 3, located ~ 75 in. from bottom of active core

Fig. 32. FTE-4 Cs-134 profile in fuel body 2 graphite web at 57.5 in. (1455 mm) active core height

Fig. 33. FTE-4 Cs-137 profile in fuel body 2 graphite web at 57.5 in. (1455 mm) active core height

Fig. 34. FTE-4 Cs-137 gamma spectroscopic profile scanned across hole 2 of body 2 at 57.5 in. (1455 mm) active core height

Fig. 35. FTE-4 Cs-134 gamma spectroscopic profile scanned across hole 2 of body 2 at 57.5 in. (1455 mm) active core height

S7410 (61-62)

S7410 (63-64)

S7410-59

S7410-60

Fig. 36. Visual examination of FTE-4 fuel rod 2-2-7 UO₂ TRISO/ThO₂ BISO blend); irradiated to 1.9×10^{25} n/m² ($E > 29$ fJ) at 1100°C

S7410 (48-49)

S7410 (50-51)

S7410-46

S7410-47

Fig. 37. Visual examination of FTE-4 fuel rod 2-3-8 [(Th,U)C₂ TRISO/ThC₂ BISO blend]; irradiated to 1.9×10^{25} n/m² ($E > 29$ fJ) at 1094°C

S7410 (24-25)

S7410 (26-27)

S7410-29

S7410-30

Fig. 38. Visual examination of FTE-4 fuel rod 2-5-7 [(Th,U)C₂ TRISO/ThC₂ TRISO blend]; irradiated to 1.9×10^{25} n/m² ($E > 29$ fJ) at 1075°C

S7410 (11-12)

S7410 (9-10)

S7410-13

S7410-14

Fig. 39. Visual examination of FTE-4 fuel rod 2-8-7 (UC_2 TRISO/ ThC_2 BISO blend); irradiated to $1.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1088°C

Fig. 40. FTE-4 as-measured fuel rod strain for UO_2 TRISO, ThO_2 BISO (holes 1 and 2)

Fig. 41. FTE-4 as-measured fuel rod strain for $(2.75 \text{ Th:U})\text{C}_2$ TRISO, ThC_2 BISO (holes 3 and 4)

Fig. 42. FTE-4 as-measured fuel rod strain for $(2.75 \text{ Th:U})\text{C}_2$ TRISO, ThC_2 BISO (holes 5 and 6)

Fig. 43. FTE-4 as-measured fuel rod strain for UC_2 TRISO, ThC_2 BISO (holes 7 and 8)

Fig. 44. FTE-4 measured and calculated fuel rod strain (holes 1 and 2)

Fig. 45. FTE-4 measured and calculated fuel rod strain (holes 3 and 4 and holes 7 and 8)

Fig. 46. FTE-4 measured and calculated fuel rod strain (holes 5 and 6)

Fig. 47. FTE-4 fuel rod anisotropy (holes 1 and 2)

Fig. 48. FTE-4 fuel rod anisotropy (holes 3 and 4)

Fig. 49. FTE-4 fuel rod anisotropy (holes 5 and 6)

Fig. 50. FTE-4 fuel rod anisotropy (holes 7 and 8)

Fig. 51. FTE-4 measured and calculated fuel rod - hole gap (holes 1 and 2)

Fig. 52. FTE-4 measured and calculated fuel rod - hole gap (holes 3 and 4 and holes 7 and 8)

Fig. 53. FTE-4 measured and calculated fuel rod - hole gap (holes 5 and 6)

L7410-195

(a)

L7410 (183-193)

(b)

Fig. 54. Photomicrographs of fuel rod 2-1-7 from FTE-4; irradiated to $1.87 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) at 1100°C : (a) typical graphite matrix and (b) composite radial cross section

L7410-121

(a)

L7410 (122-132)

(b)

Fig. 55. Photomicrographs of fuel rod 2-2-7 from FTE-4; irradiated to $1.87 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) at 1100°C : (a) typical graphite matrix and (b) composite radial cross section

L7410-101

(a)

L7410 (89-99)

(b)

Fig. 56. Photomicrographs of fuel rod 2-3-8 from FTE-4; irradiated to $1.87 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) at 1105°C : (a) typical graphite matrix and (b) composite radial cross section

L7410-25

(a)

L7410 (13-23)

(b)

Fig. 57. Photomicrographs of fuel rod 2-5-7 from FTE-4; irradiated to $1.87 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) at 1075°C : (a) typical graphite matrix and (b) composite radial cross section

L7410-64

(a)

L7410 (52-62)

(b)

Fig. 58. Photomicrographs of fuel rod 2-8-7 from FTE-4; irradiated to $1.87 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) at 1088°C : (a) typical graphite matrix and (b) composite radial cross section

L7410-72

(a)

L7410-81

(b)

Fig. 59. Photomicrographs of representative particles from fuel rod 2-3-8 in FTE-4; irradiated to $1.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1094°C : (a) $(\text{Th,U})\text{C}_2$ TRISO and (b) ThC_2 BISO beginning to hydrolyze

L7410-7

(a)

L7410-3

(b)

Fig. 60. Photomicrographs of representative particles from fuel rod 2-5-7 in FTE-4; irradiated to $1.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1075°C : (a) ThC_2 TRISO and (b) $(\text{Th,U})\text{C}_2$ TRISO

L7410-46

(a)

L7410-42

(b)

Fig. 61. Photomicrographs of representative particles from fuel rod 2-8-7 in FTE-4; irradiated to $1.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1088°C : (a) ThC_2 BISO with UC_2 TRISO and (b) UC_2 TRISO

L7410-110 (a)

L7410-142 (b)

L7410-108 (c)

Fig. 62. Photomicrographs of representative particles from fuel rod 2-2-7 in FTE-4; irradiated to $1.9 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1100°C : (a) UO_2 particle showing extrusion, (b) migrating UO_2 kernel showing plasticity next to a ThO_2 BISO particle

Fig. 63. Axial gamma-scanning rig at General Atomic hot cell

Fig. 64. FTE-4 comparison of Cs-137 inventory in stack 3 fuel rods with calculated thermal fluence profile

Fig. 65. FTE-4 typical Cs-137 single-channel scans, demonstrating fuel homogeneity

6-HOLE CRUCIBLE

MATERIAL: AGOT GRAPHITE

Fig. 66. Design of type 2 thermal stability crucibles

S7410-67

Fig. 67. Stereophotograph of UC_2 100- μm (VSM) particles from spine sample TS4-5 in FTE-4; irradiated to $1.2 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1051°C

HA314

Fig. 68. Radiograph of representative UC_2 100- μm (VSM) particles from spine sample TS4-5 in FTE-4; irradiated to $1.2 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1051°C

L7410-147

Fig. 69. Typical microstructure of UC_2 100- μm (VSM) particles from spine sample TS4-5 in FTE-4; irradiated to $1.2 \times 10^{25} \text{ n/m}^2$ ($E > 29 \text{ fJ}$) at 1051°C

ALL DIMENSIONS IN INCHES

Fig. 70. Design of diffusion spine sample

Fig. 71. Sr-85 profile in FTE-4 MDS-10 center post

Fig. 72. Sr-85 profile in FTE-4 MDS-10 crucible wall

Fig. 73. Sr-85 profile in FTE-4 MDS-12 center post

Fig. 74. Sr-85 profile in FTE-4 MDS-18B center post

Fig. 75. Sr-85 profile in FTE-4 MDS-18B crucible wall

Fig. 76. Sr-85 profile in FTE-4 MDS-22B center post

Fig. 77. Sr-85 profile in FTE-4 MDS-22B crucible wall

Fig. 78. Strontium diffusion coefficients as a function of activation-energy-weighted crucible irradiation temperatures for data derived from FTE-3 and FTE-4

Fig. 79. Ba-133 profile in FTE-4 MDS-31 center post

Fig. 80. Ba-133 profile in FTE-4 MDS-31 crucible wall

Fig. 81. Barium and europium diffusion coefficients as a function of activation-energy-weighted crucible irradiation temperatures for data derived from FTE-3 and FTE-4

Fig. 82. Eu-154 profile in FTE-4 MDS-41 center post

Fig. 83. Eu-154 profile in FTE-4 MDS-41 crucible wall

Fig. 84a. Cs-134 profile in FTE-4 MDS-10 center post - raw data

Fig. 84b. Cs-134 profile in FTE-4 MDS-10 center post - fitted data

Fig. 85. Cs-134 profile in FTE-4 MDS-10 crucible wall

Fig. 86a. Cs-134 profile in FTE-4 MDS-12 center post - raw data

Fig. 86b. Cs-134 profile in FTE-4 MDS-12 center post - fitted data

Fig. 87. Cs-134 profile in FTE-4 MDS-12 crucible wall

Fig. 88a. Cs-134 profile in FTE-4 MDS-18B center post - raw data

Fig. 88b. Cs-134 profile in FTE-4 MDS-18B center post - fitted data

Fig. 89. Cs-134 profile in FTE-4 MDS-18B crucible wall

Fig. 90. Cs-134 profile in FTE-4 MDS-22B center post

Fig. 91. Cs-134 profile in FTE-4 MDS-22B crucible wall

Fig. 92. Cs-134 profile in FTE-4 MDS-31 center post

Fig. 93. Cs-134 profile in FTE-4 MDS-31 crucible wall

Fig. 94. Cs-134 profile in FTE-4 MDS-41 center post - raw data

Fig. 95. Cs-134 profile in FTE-4 MDS-41 center post - fitted data

Fig. 96. Cs-134 profile in FTE-4 MDS-41 crucible wall

Fig. 97. "Effective" diffusion coefficient of cesium (D_{eff}) plotted as a function of activation-energy-weighted irradiation temperatures

Fig. 98. FTE-4 shield pin assembly

POST IRRADIATION MEASUREMENT

O.D.	
TOP	0 - 1.0002 90 - 0.9991
MIDDLE	0 - 0.9994 90 - 0.9989
BOTTOM	0 - 0.9992 90 - 0.9999
LENGTH	- 2.9988

Fig. 99. FTE-4 boronated steel sample and sample holder

Fig. 100. FTE-4 boronated steel sample

Fig. 101. FTE-4 comparison of measured versus predicted swelling for boronated stainless steel sample

GA HOT CELL GAMMA SCAN

Fig. 102. FTE-4 spine sample gamma scan from Body 1

GA HOT CELL GAMMA SCAN

Fig. 103. FTE-4 spine sample gamma scan from Body 2

GA HOT CELL GAMMA SCAN

Fig. 104. FTE-4 spine sample gamma scan from Body 3

5. STRESS ANALYSIS

5.1. METHOD AND MATERIAL PROPERTIES

Stress calculations have been carried out to determine:

1. The distribution of thermal- and irradiation-induced operating stresses as a function of time, fluence, and axial position.
2. The maximum operating and shutdown stresses for the irradiation period as well as for BOL and EOL.

Temperature and stress distributions at the midsection of each fuel body were calculated using GTEPC-2D (Ref. 40), a computer program for viscoelastic analysis of graphite. For stress calculations GTEPC uses a standard viscoelastic solid, where the coefficients of the springs and dashpots are functions of temperature and fluence.

The finite element model of the fuel body and sleeve was a 22.5° sector of unit thickness taken at the midplane of each of the three eight-hole "teledial" fuel bodies. The finite element mesh, which consists of 334 nodes and 524 elements, is shown in Fig. 105a. The full mesh system was used for the thermal calculations, whereas stresses were calculated in the graphite fuel body only. Thermal boundary conditions were determined from one-dimensional TREVER calculations. GTEPC temperature field calculations have been verified with TAC2D and with TREVER calculations within $\pm 40^\circ\text{C}$.

Material properties used in the stress analysis as well as the thermal analysis of the H-327 graphite bodies were taken from Ref. 8; irradiation-induced creep data were taken from Ref. 41. The thermal conductivity and irradiation strain for the H-381/2 sleeve graphite were taken from Ref. 42, and the coefficient of thermal expansion for this graphite came from Ref. 43.

5.2. INITIAL THERMAL STRESS

A summary of the initial thermal stress results is given in Tables 60A, B, and C for each midplane position of the three fuel bodies together with maximum temperature differences within each plane at BOL (6624 hr of reactor operation). This temperature difference is an indicator of the general level of initial stresses to be found at start-up.

Maximum initial tensile stresses occurred at the midplane of body 2 at element position 288 (Fig. 105b). This is on the periphery of the fuel hole adjacent to the slot cut into the inside of the element. The slot was a safety feature, designed to confine any graphite breakage due to swelling of fuel rods to the inside diameter of the elements.

Maximum compression stresses are located in the region at the tip of the safety slot (i.e., element 359). Comparison of the stress results with the ultimate tensile strength (UTS) and ultimate compressive strength (UCS) values for H-327 graphite in Table 61 shows that initial thermal stresses are less than the ultimate strength values. The in-plane tensile and compressive stresses are 20% and 18% of the corresponding strength.

The strength history is obtained by correcting the initial strength at room temperature for the effects of temperature and fast neutron irradiation according to (Ref. 8):

$$UTS_t = UTS_o (E_t/E_o)^n, \quad (110)$$

where $n = 0.4$ for radial direction,

$= 0.67$ for axial direction for H-327 graphite,

E_o = elastic modulus at BOL and room temperature (25°C),

E_t = elastic modulus at time t as a function of fluence and irradiation temperature with the additional correction for elevated temperatures by

$$E_T = E_O (1 + 1.5 \times 10^{-4} \Delta T), \quad (111)$$

where ΔT = difference between the operating temperature and 25°C.

The initial tensile strength at room temperature is obtained from Table 61 for the midlength edge of the log. The midlength edge location was selected because the test element bodies were known to have been machined from the outer portion of the H-327 graphite log. This may overestimate the strength somewhat since some of the test elements were taken from a position intermediate between the outer portion of the log (designated as edge) and the central portion of the log. On the other hand, the test elements were machined from logs that are likely to be somewhat stronger than the mean. The tensile strengths of the single specimens taken from the midlength center of each of the logs from which bodies in FTE-4 were machined were measured to be 12.6 to 13.0 MPa, well above the mean strength of all the logs of 11.2 MPa. In all, the midlength edge strength is believed to be an unbiased best estimate of tensile strength at the midplanes of the test element bodies. For the initial compressive strength, which has not been characterized as a function of position in the log, a conservative minimum value of 27.6 MPa (4000 psi) was used.

Compressive stress values have been found to be small in comparison with the UCS value.

The following peak initial thermal stresses can be quoted for FTE-4:

Initial Thermal Stress	Peak	Mean ^(a)	±RMS ^(a)	Fuel Body Sequence
Tensile stress				
In-plane (element 288) (MPa)	3.24	2.73	±0.37	231
Stress - strength ratio	0.33	0.28	±0.04	231
Axial (element 297) (MPa)	2.30	1.92	±0.28	231
Stress - strength ratio	0.13	0.11	±0.02	231

(a) The mean and RMS are used to describe the variation of calculated peak stresses between the three fuel body midplanes and are not a representative statement for mean body stresses nor of uncertainty in calculations.

Initial Thermal Stress	Peak	Mean ^(a)	±RMS ^(a)	Fuel Body Sequence
Compression stress				
In-plane (element 359) (MPa)	-5.19	-4.39	±0.59	231
Stress - strength ratio	0.18	0.15	±0.02	231
Axial (element 10) (MPa)	-4.10	-3.44	±0.50	231
Stress - strength ratio	0.13	0.11	±0.02	231

(a) The mean and RMS are used to describe the variation of calculated peak stresses between the three fuel body midplanes and are not a representative statement for mean body stresses nor of uncertainty in calculations.

Body 2 experiences the highest BOL stresses. The stress level is well below the ultimate strength values.

Temperature and stress contour plots for start-up at BOL are presented in Figs. 106 through 109, for the center region of each fuel body. Figure 109 gives the stress direction in each plane, which is the same for all three bodies.

5.3. PEAK STRESS DURING LIFE

The stress distribution changes rapidly with time owing to increased irradiation strain in the hotter areas of the fuel body. The higher-temperature areas reverse from initial compressive stress to a state of tensile stress, while the cooler areas near the sleeve enter a compressive state. This stress reversal is predicted by the code at all three axial locations.

In general, peak EOL tensile stress occurs in the hot positions toward the center of the body, while peak EOL compressive stresses are found in the low-temperature areas near the outer edge of the element.

Tables 60A, B, and C contain the peak operating stresses found during the irradiation simulation. The time points (core 2 reactor hours at

operation) and finite elements, where the peak stresses occur, are also included. Time points refer to TREVER simulation intervals, and element locations to the finite element mesh in Fig. 105b. Temperature and stress contour plots for EOL operation are presented in Figs. 110 through 112, and stress directions are shown in Fig. 113 for all three planes. Figures 114 through 116 give the history of the elements with the maximum and minimum temperatures and maximum and minimum tensile stresses (in-plane and axial) during operation and shutdown for an axial position near the midplane of each body. Finite elements with a peak-in-time condition have been selected for the presentation.

The following peak operational and shutdown stress values during the irradiation life of FTE-4 can be quoted:

Operational Stress	Peak	Mean	±RMS	Fuel Body Sequence
Tensile stress				
In-plane (element 359) (MPa)	4.53	3.42	±0.83	231
Stress - strength ratio	0.37	0.28	±0.07	231
Axial (element 20) (MPa)	6.27	4.98	±0.91	231
Stress - strength ratio	0.24	0.19	±0.04	231
Compression stress				
In-plane (element 288) ^(a) (MPa)	-2.06	-1.54	±0.44	123
Stress - strength ratio	0.06	0.04	±0.01	123
Axial (element 297) ^(a) (MPa)	-3.41	-2.47	±0.72	231
Stress - strength ratio	0.08	0.06	±0.01	231

(a) Element 359 for body 1.

Shutdown Stress	Peak	Mean	±RMS	Fuel Body Sequence
Tensile stress				
In-plane (element 359) (MPa)	15.82	14.37	±1.41	123
Stress - strength ratio	1.42	1.27	±0.18	123
Axial (element 20) (MPa)	15.69	14.46	±1.34	213
Stress - strength ratio	0.69	0.61	±0.08	213

Shutdown Stress	Peak	Mean	±RMS	Fuel Body Sequence
Compression stress				
In-plane (element 20) ^(a) (MPa)	-7.66	-7.22	±0.39	213
Stress - strength ratio	0.23	0.21	±0.01	213
Axial (element 297) (MPa)	-11.19	-9.67	±1.08	231
Stress - strength ratio	0.27	0.24	±0.03	231

(a) Element 359 for body 3.

Peak operational stresses occur in body 2, with the exception of peak in-plane compression occurring in body 1. Peak shutdown stresses are found in body 2, with the exception of the peak in-plane tension in body 1, where the peak stress-strength ratio also occurs in body 2. The tabulated stress values are the peak values seen at any time during the irradiation of FTE-4, as obtained from the stress predictions. The actual time of occurrence covers the period between 275 and 701 EFPD, which is about 23 EFPD after BOL through EOL. The highest tensile stress occurred at the EOL shutdown in the in-plane direction, which gives a stress-strength ratio of 1.42. This is the highest stress-strength ratio calculated for FTE-4; it was found for the inner safety slot location. Although the predicted stress exceeds the predicted strength of the irradiated material, no defects have been detected during destructive postirradiation examination.

It should be noted that the stress at the edge of the safety slot is not calculated reliably by the present models or methods. The stress has been shown (Ref. 44) to be sensitive to the fineness of the mesh used at this point and is also expected to be sensitive to the exact shape of the notch, which is modelled as a sharp corner but will actually have a finite radius of curvature. The results would also be expected to be sensitive to the shape of the stress-strain curve for graphite, which has been approximated in the present calculations by a straight line.

5.4. PEAK STRESS AT EOL

Peak EOL operation and shutdown stresses have been determined and are included in Tables 60A, B, and C for each of the three bodies. The finite element locations are also presented. As mentioned earlier, an inversion took place between BOL and EOL for the peak stress locations: the highly compressed areas became subject to tensile stresses and vice versa. On EOL shutdown, these tensile stresses and compression stresses increased nearly three-fold in both cases, compared with the EOL operating stresses.

Temperature and stress contour plots for shutdown stresses are presented in Figs. 117 through 119. Shutdown temperature is 20°C; for better comparison, the temperature field prior to shutdown has been included in these figures, which are also presented in the contour plots for EOL operation (Figs. 110 through 112). Figure 120 contains the stress directions for EOL shutdown, which vary slightly from body to body.

The peak EOL stresses can be summarized as follows:

EOL Operational Stress	Peak	Mean	±RMS	Fuel Body Sequence
Tensile stress				
In-plane (element 359) (MPa)	5.18	4.16	±0.80	231
Stress - strength ratio	0.43	0.34	±0.07	231
Axial (element 20) (MPa)	7.10	5.89	±0.86	231
Stress - strength ratio	0.28	0.23	±0.03	231
Compression stress				
In-plane (element 288) (MPa)	-1.61	-1.38	±0.21	231
Stress - strength ratio	0.03	0.03	±0.01	231
Axial (element 297) ^(a) (MPa)	-3.94	-3.01	±0.77	231
Stress - strength ratio	0.06	0.05	±0.01	231

(a) Element 333 for body 1.

EOL Shutdown Stress	Peak	Mean	±RMS	Fuel Body Sequence
Tensile stress				
In-plane (element 359) (M a)	15.82	14.37	±1.41	123
Stress - strength ratio	1.42	1.27	±0.18	123
Axial (element 20) (MPa)	15.69	14.46	±1.34	213
Stress - strength ratio	0.69	0.61	±0.08	213
Compression stress				
In-plane (element 91) ^(a) (MPa)	-6.43	-5.53	±1.05	123
Stress - strength ratio	0.14	0.12	±0.02	123
Axial (element 297) (MPa)	-11.73	-10.23	±1.07	231
Stress - strength ratio	0.20	0.17	±0.02	231

(a) Element 359 for body 1, element 10 for body 2.

Peak operational stresses all occur in body 2 at EOL and are always lower than the strength. Tensile shutdown stresses, however, are significant for the inner edge of the safety slot, where the in-plane stress-strength factor reaches 1.42 for body 1, and for the inner location of the fuel hole surface, where the axial stress-strength factor is determined to be 0.69. However, no structural damage was found during destructive post-irradiation examination.

5.5. STRESS ANALYSIS CONCLUSIONS

1. FTE-4 was a high-stress experiment, reaching EOL shutdown stresses of up to 142% of the UTS. Compressive stresses were small in comparison with the UCS of the material.
2. Highest in-plane stresses were found for the inner edge of the safety slot for EOL operation (43%) and shutdown (142%). For the fuel hole surface closest to the center, the highest axial stresses were found during shutdown (69%).

3. Visual examination of the fuel bodies and several slices taken during postirradiation examination did not reveal any structural damage of the graphite, although the predicted stresses are significantly higher than measured strength values of the H-327 material. The in-plane stresses at the safety slot, in particular, were predicted to be 142% above the ultimate strength. The fact that these stresses did not cause damage is probably due to the relatively low notch sensitivity of graphite (Ref. 45) and the very small area near the slot that was highly stressed. It should also be noted that the actual stresses near the slot may be significantly different from the calculated stresses, as discussed in Section 3.5.3.

TABLE 60A
FTE-4 STRESS SUMMARY FOR MIDPLANE OF BODY 1

Time (hr)	Operating Condition						Shutdown Condition			
	Temperature (°C)		In-Plane Stress (MPa)		Axial Stress (MPa)		In-Plane Stress (MPa)		Axial Stress (MPa)	
	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.
0	255	255	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	0	0
6624	1125	1015	2.36	-5.21	1.63	-3.00	.04	-.07	.01	-.01
	10	297	288	359	297	10	288	100	288	359
7992	1148	1011	1.03	-2.34	.87	-1.52	4.67	-1.90	3.65	-1.92
	10	297	288	359	297	359	359	359	10	297
9312	1275	1072	.86	-1.95	.60	-.93	9.58	-3.87	8.20	-4.33
	10	297	288	359	297	359	359	20	10	297
10440	1263	1051	.31	-.50	1.04	-.38	11.29	-4.55	9.95	-5.28
	10	297	269	359	1	332	359	20	10	297
10441	1214	1019	1.29	-.59	2.00	-.68	11.28	-4.54	9.94	-5.28
	10	297	359	288	9	314	359	20	10	297
13944	1223	1012	.95	-.41	2.37	-.76	12.68	-5.11	11.40	-5.91
	10	297	359	288	20	308	359	359	20	297
15888	1288	1050	1.55	-.68	3.21	-1.06	15.02	-6.05	13.72	-7.30
	10	297	359	288	20	314	359	359	20	297
17112	1277	1042	2.53	-1.11	4.25	-1.60	15.44	-6.21	14.39	-7.73
	10	297	359	288	20	322	359	359	20	297
19488	1274	1040	3.24	-1.42	5.20	-2.11	15.82	-6.36	15.10	-8.20
	10	297	359	288	20	333	359	359	20	297

TABLE 60B
FTE-4 STRESS SUMMARY FOR MIDPLANE OF BODY 2

Time (hr)	Operating Condition						Shutdown Condition			
	Temperature (°C)		In-Plane Stress (MPa)		Axial Stress (MPa)		In-Plane Stress (MPa)		Axial Stress (MPa)	
	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.
0	255	255	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	0	0
6624	1283	1144	3.24	-7.15	2.30	-4.27	.02	-.04	.00	-.00
	10	297	288	359	297	10	288	100	288	359
7992	1296	1136	.49	-1.11	.48	-.21	6.98	-2.83	6.36	-3.52
	10	297	288	359	1	359	359	20	10	297
9312	1426	1205	.46	-.20	1.89	-1.00	11.77	-4.74	11.21	-6.39
	10	297	359	288	10	297	359	20	10	297
10440	1402	1179	2.06	-.90	3.42	-1.89	12.86	-5.16	12.48	-7.19
	10	297	359	288	20	297	359	359	20	297
10441	1347	1143	3.62	-1.59	4.50	-2.50	12.85	-5.16	12.47	-7.19
	10	297	359	288	20	297	359	359	20	297
13944	1361	1135	3.46	-1.50	5.25	-2.75	13.72	-5.51	13.87	-8.12
	10	297	359	288	20	297	359	359	20	297
15888	1438	1182	2.48	-1.06	4.79	-2.48	14.51	-5.83	14.75	-8.78
	10	297	359	288	20	297	359	359	20	297
17112	1425	1174	4.53	-1.98	6.27	-3.53	15.26	-6.12	15.61	-9.43
	10	297	359	288	20	297	359	359	20	297
19488	1423	1175	5.18	-2.26	7.10	-4.09	14.82	-5.94	15.69	-9.58
	10	297	359	288	20	297	359	91	20	297

TABLE 60C
FTE-4 STRESS SUMMARY FOR MIDPLANE OF BODY 3

Time (hr)	Operating Condition						Shutdown Condition			
	Temperature (°C)		In-Plane Stress (MPa)		Axial Stress (MPa)		In-Plane Stress (MPa)		Axial Stress (MPa)	
	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.	Max.	Min.
0	255	255	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	0	0
6624	1316	1206	2.60	-5.74	1.82	-3.35	.02	-.03	.00	-.00
	10	297	288	359	297	10	288	100	288	359
7992	1306	1188	.41	-.92	.23	-.26	4.63	-1.88	4.08	-2.27
	10	297	288	359	1	359	359	20	10	297
9312	1405	1253	.06	-.06	.93	-.52	7.62	-3.07	7.05	-4.01
	10	297	269	320	1	297	359	20	10	297
10440	1370	1219	.92	-.40	1.85	-1.03	8.28	-3.33	7.78	-4.46
	10	297	359	288	20	297	359	359	20	297
10441	1321	1182	1.86	-.82	2.48	-1.48	8.29	-3.34	7.78	-4.46
	10	297	359	288	20	297	359	359	20	297
13944	1389	1203	1.62	-.70	2.98	-1.62	10.56	-4.24	10.21	-5.89
	10	297	359	288	20	297	359	359	20	297
15888	1464	1254	2.06	-.89	3.47	-1.95	11.99	-4.82	11.57	-6.86
	10	297	359	288	20	297	359	359	20	297
17112	1454	1246	3.19	-1.39	4.42	-2.57	12.37	-4.97	12.15	-7.27
	10	297	359	288	20	297	359	359	20	297
19488	1457	1249	4.05	-1.76	5.36	-3.17	12.46	-5.00	12.59	-7.61
	10	297	359	288	20	297	359	91	20	297

TABLE 61A
H-327 GRAPHITE ULTIMATE TENSILE AND COMPRESSION STRENGTH

Block Position, Orientation	Ultimate Strength (Ref. 38)			
	Mean		Std. Deviation	
	psi	MPa	psi	MPa
Ultimate tensile strength				
End center, axial	2180	15.03	435	3.00
End edge, axial ^(a)	2395	16.51	480	3.31
Midlength center, axial	1630	11.24	325	2.24
Midlength edge, axial ^(b)	2395	16.51	480	3.31
End center, radial	1350	9.31	270	1.86
End edge, radial ^(a)	1350	9.31	270	1.86
Midlength center, radial	940	6.48	190	1.31
Midlength edge, radial ^(b)	1290	8.90	260	1.79
Ultimate compressive strength ^(b)	≥4000	≥27.58		

(a) End edge values were obtained by taking the midlength edge value for the axial and the end center value for the radial orientation. The standard deviation was calculated from 20% of the mean UTS value based upon characterization and QA test results.

(b) Used for unirradiated strength at room temperature in computing stress-to-strength ratios.

TABLE 61B
H-327 GRAPHITE ELASTIC MODULUS

Pointwise Block Position, Orientation	Elastic Modulus			
	Mean		Std. Deviation (Ref. 38)	
	10 ⁶ psi	10 ³ MPa	10 ⁶ psi	10 ³ MPa
End center, axial	1.42	9.79	0.3	2.07
End edge, axial ^(a)	2.02	13.93	0.6	4.14
Midlength center, axial	1.42	9.79	0.4	2.76
Midlength edge, axial ^(b)	2.02	13.93	0.6	4.14
End center, radial	0.754	5.20	0.3	2.07
End edge, radial ^(a)	0.669	4.61	0.3	2.07
Midlength center, radial	0.571	3.94	0.2	1.38
Midlength edge, radial ^(b)	0.774	5.34	0.3	2.07

(a) End edge values were obtained by taking the midlength edge value for the axial and the end center value for the radial orientation.

(b) Average used in stress calculation.

Fig. 105a. Finite element model for thermal analysis

Fig. 105b. Finite element model for residual stress analysis

Fig. 106. FTE-4 fuel body 1 temperature and stress contour plots for start-up at BOL

Fig. 107. FTE-4 fuel body 2 temperature and stress contour plots for start-up at BOL

Fig. 108. FTE-4 fuel body 3 temperature and stress contour plots for start-up at BOL

Fig. 109. FTE-4 fuel bodies 1, 2, and 3: stress directions for start-up at BOL

Fig. 110. FTE-4 fuel body 1 temperature and stress contour plots for operation at EOL

Fig. 111. FTE-4 fuel body 2 temperature and stress contour plots for operation at EOL

Fig. 112. FTE-4 fuel body 3 temperature and stress contour plots for operation at EOL

Fig. 113. FTE-4 fuel bodies 1, 2, and 3: stress directions for operation at EOL

Fig. 114. FTE-4 fuel body 1 maximum-minimum temperatures and maximum-minimum stress histories for operation and shutdown

Fig. 115. FTE-4 fuel body 2 maximum-minimum temperatures and maximum-minimum stress histories for operation and shutdown

Fig. 116. FTE-4 fuel body 3 maximum-minimum temperatures and maximum-minimum stress histories for operation and shutdown

Fig. 117. FTE-4 fuel body 1 temperature at EOL operation and stress contour plots for EOL shutdown

Fig. 118. FTE-4 fuel body 2 temperature at EOL operation and stress contour plots for EOL shutdown

Fig. 119. FTE-4 fuel body 3 temperature at EOL operation and stress contour plots for EOL shutdown

Fig. 120. FTE-4 fuel bodies 1, 2 and 3: stress directions for EOL shutdown

6. DISCUSSION AND CONCLUSIONS

The postirradiation examination of FTE-4 has made a significant contribution to the verification of nuclear, thermal, fission product, and structural computer codes for the large HTGR.

The irradiation conditions of FTE-4 were calculated by the latest GAUGE computer codes. The test element was irradiated in Peach Bottom Core 2 position A-14-08 for 449 EFPD [July 11, 1971 (BOL) to September 14, 1973 (EOL)]. The average power factor was 0.91, and the total accumulated thermal energy was 58.4 MWd. The element experienced a $1.53 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) EOL fast fluence and a $2.145 \times 10^{25} \text{ n/m}^2$ ($E < 0.38 \text{ aJ}$) thermal fluence. Axial fuel rod gamma scanning substantiated the shape of the calculated thermal fluence profile.

The time-averaged maximum fuel temperatures for FTE-4 calculated by the TREVER code were between 1165° and 1187°C. FTE-4 had both a W/Re and a C/A thermocouple, although the latter was not functional. From recent Peach Bottom thermocouple decalibrations, the integrated temperatures from the thermocouple are $99^\circ \pm 35^\circ\text{C}$ (RMS) below calculated values. Until more Peach Bottom EOL evidence is available, the calculated values will be considered more representative of FTE-4's thermal history.

An extensive burnup analysis was performed on FTE-4 with the following techniques:

GAUGE/FEVER calibration

Radiochemistry

Mass spectroscopy

Isotope dilution mass spectroscopy (IDMS)

Gamma spectroscopy

The GAUGE/FEVER computer codes calculated an average 29.54% fissile FIMA and 0.45% fertile FIMA. A general statement of the comparison of the various burnup measurements is given below:

For the fissile burnup

$$F_5 \text{ calculated} < F_5 \text{ IDMS} < F_5 \text{ radiochemical}$$

For the fertile burnup

$$F_3 \text{ radiochemical} > F_3 \text{ calculated}$$

A detailed comparative analysis and an error determination of the various methods are given in earlier sections of this report.

The structural integrity of the fuel element sleeve and graphite fuel bodies was sound. The EOL gamma scan at Peach Bottom confirmed axial fuel rod measurements taken during the postirradiation examination. The fuel body and sleeve strain measurements were in agreement with TREVER predictions by $\pm 30\%$. There was less shrinkage measured in the unfueled zone of the fuel bodies than predicted by TREVER. However, this is related to axial heat conductance which is presently not included in the thermal modeling of the element.

Radial Cs-137 and Cs-134 profiles were determined in the outer fuel hole webs of the graphite fuel bodies by a sectioning technique. FIPERQ code fits of the Cs-137 reference data indicated an approximately 100°C overcalculation of the temperatures in the region of the measurement.

FTE-4 fuel performance was consistent with current HTGR design data (Ref. 9) and the results from the previous companion test element FTE-3 (Ref. 2). Irradiation-induced fuel failure was low and there was no matrix-particle interaction. Except for the UO_2 kernel movement caused by the gross inhomogeneity of several fuel rods, there were no thermochemical cases of either SiC attack by metallic fission products or kernel migration.

This is as predicted for these low burnups (30% FIMA fissile and 0.4% FIMA fertile) and low temperatures (1090°C) (time-weighted average of the examined fuel rod).

In summary, the following conclusions and observations can be made based on the postirradiation examination of the fuel in FTE-4:

1. The physical integrity and appearance of all the fuel rods examined were good except for unloading damage.
2. Irradiation-induced failure of the fuel as determined by fission gas release measurement and metallography was low (<1%) at peak fluence and temperature conditions of $1.9 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) and 1090°C.
3. No metallic fission product attack of the SiC or kernel migration occurred at the peak temperature and fluence conditions of 1190°C and $1.9 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) in any of the fuels tested except those noted in item 4, below.
4. UO_2 kernels showed significant fuel kernel migration and deformation, which was not detrimental to the integrity of the particles.
5. Unloading damage to the surface of the fuel rods can definitely affect fission gas R/B measurements.
6. Behavior of 100- μm (VSM) UC_2 particles in spine samples was very good at a 1051°C and $1.2 \times 10^{25} \text{ n/m}^2$ ($E \geq 29 \text{ fJ}$) irradiation exposure.

Twenty-two metallic diffusion samples were analyzed to determine the diffusion and sorption characteristics of Cs-134, Cs-137, Sr-85, Ba-133, and Eu-154 in H-327 graphite. The data were consistent with FTE-3 results.

The Cs-134 and Cs-137 and possibly the Eu-154 isotopes showed a two-component diffusion mechanism (fast and slow). The strontium and barium diffusion cases were all of the classical Fick's law type.

A spine sample containing B_4C graphite pellets was examined to determine the effects of neutron exposure. Results show the B_4C to be dimensionally stable and to react with the coolant impurities to form a BN layer which is not considered detrimental to the B_4C performance.

Boronated stainless steel was also tested in a spine sample. Visual examination showed a black deposit on the exterior of the sample, and dimensional measurements showed an $\sim 1\%$ expansion.

The stress analysis of FTE-4 indicates that the highest in-plane tensile stresses occurred at the safety slot in the fuel bodies (Fig. 3). During operation and shutdown the maximum probable in-plane stress reached 43% (36% median) and 142% (120% median) of BOL UTS, respectively. In all cases the compressive stresses were less than half the UTS of H-327 graphite. The stress analysis was considered conservative because no cracking or structural damage was evident in the highly stressed regions.

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