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GEAP-10160
AEC RESEARCH AND
DEVELOPMENT REPORT
FEBRUARY 1970

**ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM
QUARTERLY PROGRESS REPORT NO. 9
NOVEMBER 1969 - JANUARY 1970**

COMPILED BY
F. H. MEGERTH

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 41

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GEAP-10160
AEC Research and
Development Report
February 1970

ZIRCALOY-CLAD UO₂ FUEL ROD EVALUATION PROGRAM

Quarterly Progress Report No. 9
November 1969—January 1970

Compiled By
F. H. Megerth

Approved:

F. H. Megerth
F. H. Megerth
Project Engineer

Approved:

Mark H. Lyons
M. F. Lyons, Manager
Uranium Fuel Development

Prepared for the
U.S. Atomic Energy Commission
Contract AT(04-3) - 189
Project Agreement 41

*Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia
Price: \$3.00 per copy*

REACTOR FUELS AND REPROCESSING DEPARTMENT • GENERAL ELECTRIC COMPANY
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1. INTRODUCTION

At the time of the final shutdown of the Vallecitos Boiling Water Reactor (VBWR) and the termination of the AEC Fuel Cycle Program (Contract AT(04-3)-189, Project Agreement 11), the leading Zircaloy-2-clad UO_2 fuel rods of the program's Type J basic assemblies had attained rod-average exposures as high as 3×10^{20} fissions/cm³ (10,000 MWd/t U).

To demonstrate at the earliest possible time the capability of Zircaloy-clad fuel to achieve average burnups of at least 6×10^{20} fissions/cm³ (20,000 MWd/t U) and to take advantage of the available lead time in achieving these burnup targets, 98 of the higher exposure Fuel Cycle Program rods were incorporated in a special assembly designated as SA-1 and inserted in the Dresden reactor for additional irradiation. The assembly consists of two 7-by-7 subassemblies, stacked one on top of the other in a channel of standard dimensions for the Dresden reactor.

Assembly SA-1 was operated continuously from June 1964 to September 1969, with the exception of reactor shutdowns for refueling and maintenance.

To investigate in detail the performance of the SA-1 fuel rods and to determine the isotopic composition of the highly exposed fuel, a two-phase program of destructive examination and analysis was drawn up. For the initial phase of the program, 10 rods were removed from the assembly in January 1967, after 1200 days of VBWR and Dresden operation, and were shipped to the Vallecitos

Nuclear Center. The examination of these 10 rods is in progress.

Prior reports to the Atomic Energy Commission under Project Agreement 41 comprise the following:

- a. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 1, November 1967-January 1968," March 1968 (GEAP-5598).
- b. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 2, February 1968-April 1968," May 1968 (GEAP-5624).
- c. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 3, May 1968-July 1968," August 1968 (GEAP-5667).
- d. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 4, August 1968-October 1968," November 1968 (GEAP-5702).
- e. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 5, November 1968-January 1969," February 1969 (GEAP-5746).
- f. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report Nos. 6 and 7, February-July 1969," August 1969 (GEAP-10079).
- g. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 8, August-October 1969," November 1969 (GEAP-10121).

2. SUMMARY

Several cladding ring tests of the type carried out under the Fast Ceramic Reactor Development Program were performed on irradiated SA-1 cladding samples to determine some of the transverse tensile properties, particularly ductility. In this type of test, a ring of cladding is placed over and centered on a metal cylinder that is somewhat longer than the ring. The diameter of the ring is increased to the point of fracture by compressing the ends of the cylindrical mandrel.

The irradiated ring specimens were prepared from two relatively high exposure, ~15-cm (6-inch)-long sections of fuel rods A13 and A41. The fast fluence exposures are estimated to be between 3.5×10^{21} and 4×10^{21} nvt > 1 MeV for all specimens tested.

Most of the tests were carried out at average specimen strain rates of 0.01 to 0.02 per minute. A few tests were run at average strain rates of 0.001 to 0.003 per minute. Different specimens were tested at room temperature,

343°C (650°F), and 454°C (850°F). In the high-temperature tests, the specimens were held at temperature for 15 minutes before the load was applied to the mandrels.

Strains to fracture ($\Delta D/D_0 \times 100$) of 13 to 27% were obtained at room temperature on the irradiated specimens. The strains to fracture for specimens from rod A13 were considerably lower than those from rod A41, although the operating histories of these two rods were quite similar. An unirradiated specimen prepared from SA-1 (Fuel Cycle Program) archive tubing was strained 50% without fracture in a room temperature test.

Specimens from rod A41 were increased in diameter as much as 69% without fracturing when tested with normal mandrels at 343°C (650°F). [A large portion of the tests, particularly those run at 343°C, had to be terminated before specimen fracture occurred because the load limit of the testing machine was reached or because the height of the mandrel closely approached the 0.445-cm (0.175-inch)

ring width.] A specimen from rod A41 tested at 343°C on a mandrel that contained a longitudinal groove in its cylindrical surface was expanded 55% before breaking at the location of the groove. The groove was intended to simulate a radial crack in a fuel pellet.

Two specimens tested at 454°C (850°F) fractured at strains of 34% and 45%.

The results were inconclusive with respect to the possible effects of variation in strain rate.

Data on the americium and curium isotopic contents of full-cross-section fuel samples with burnups of 7×10^{20} to 8×10^{20} fissions/cm³ (25,000 to 30,000 MWd/t U) were extended to include Am-242, and Cm-243, Cm-245, and Cm-246. Np-237 contents also were measured. The atom ratios of these isotopes relative to U-238 were as follows:

Np-237:	$2.8 - 3.4 \times 10^{-4}$
Am-242:	$3.9 - 5.3 \times 10^{-7}$
Cm-243:	$2.3 - 3.3 \times 10^{-7}$
Cm-245:	$3.4 - 5.6 \times 10^{-7}$
Cm-246:	$3.8 - 6.8 \times 10^{-8}$

In addition, two sets of radial fuel samples were analyzed for Np-237. The atom ratios of Np-237 to U-238 ranged from 3.2×10^{-4} at the pellet center (7.9×10^{20} fissions/cm³) to 5.4×10^{-4} at the pellet surface (11.6×10^{20} fissions/cm³) for one set of samples, and 2.4×10^{-4} at the center (6.2×10^{20} fissions/cm³) to approximately 5×10^{-4} at the surface (10.2×10^{20} fissions/cm³) for the other set.

The complete, final, isotopic composition data obtained during the initial phase of the program are compiled in Subsection 3.2.

3. EXAMINATIONS AND ANALYSES OF FUEL RODS

The following work was carried out on the 10 fuel rods that were permanently removed from Assembly SA-1 in January 1967.

3.1 FUEL ROD PERFORMANCE EVALUATION

3.1.1 Cladding Mechanical Properties Testing—Ring Tests (G. P. Ferguson, F. H. Megerth, D. S. Vash)

The tensile properties of irradiated fuel cladding in the transverse direction are of interest because of the possibility of mechanical interaction between the fuel and the cladding during the course of operation. It is expected that for a given fuel rod power level, the likelihood (or degree) of interaction will increase with time through the mechanism of fission-product-induced fuel swelling. Several cladding ring tests of the type carried out under the Fast Ceramic Reactor Development Program [Contract AT(04-3)-189, Project Agreement 10]¹ were performed on irradiated SA-1 cladding samples to determine some of the transverse tensile properties, particularly ductility. The specimen in this test was a transverse cladding section, generally somewhat less in length than the inside diameter of the cladding. This ring of cladding was placed over and centered on a metal cylinder that was somewhat longer than the ring, and of a diameter slightly smaller than the inside diameter of the cladding. The diameter of the ring was increased to the point of fracture by compressing the ends of the cylindrical mandrel. Pressure was applied to the mandrel by means of a compression cage device used in conjunction with an Instron Model TT-C-L tensile testing machine. The diameter or circumference of the ring at fracture was measured, and the value $\Delta D/D_0$ was calculated.

During the present and preceding quarters, some preliminary trial runs were performed to determine the compression characteristics of Zircaloy-2 mandrels of various heights between room temperature and 454°C (850°F). From the results of these runs, it appeared that 1.3 cm (0.5 inch) was about the tallest mandrel that could be used without the occurrence of buckling or "hour-glassing" around the specimen. Furthermore, the amount of diametral expansion that could be attained was limited by the 4500-kg (10,000-lb) load capacity of the Instron machine. Nevertheless, based on these preliminary trials, it was predicted that the amounts of expansion attained with 1.3-cm (0.5-inch)-high, 0.960-cm (0.378-inch)-diameter Zircaloy-2 mandrels would be adequate to test irradiated SA-1 cladding ring specimens successfully within the aforementioned temperature range with ring widths up to about 0.5 cm (0.2 inch). (Further work showed that this was not the case and, as a result, mandrels of other materials and heights had to be used.)

A ring width of 0.445 cm (0.175 inch) was selected for the present series of tests. This width was the same as the width of the reduced section in the longitudinal tensile coupons tested previously.² The nominal inside diameter and wall thickness of the SA-1 cladding were 0.965 cm (0.380 inch) and 0.056 cm (0.022 inch), respectively.

The irradiated ring specimens were prepared from two relatively-high-exposure, ~15-cm (6-inch)-long sections of fuel rods A13 and A41. The fast fluence exposure was estimated to be between 3.5×10^{21} and 4×10^{21} nvt > 1 MeV for all specimens tested.

The ring specimens were cut from the tubing with a small, thin, sectioning-type cut-off wheel. The fuel was

removed from the rod sections by dissolving it in nitric acid before cutting these sections into rings. The external surfaces of rods A13 and A41 were treated earlier with hydrochloric acid to remove the crud deposits.³

A trace of load on the mandrel versus time was made for each ring test. A given test was run to the point of specimen failure—indicated by a drop in load on the strip chart—to the 4500-kg machine load limit, or to a point where the mandrel was estimated to have been compressed to a height only 0.050 cm (~ 0.020 inch) or so greater than the ring width.

Most of the tests were performed with a machine crosshead speed of 0.025 cm (0.01 inch) per minute, which generally resulted in average specimen strain rates of 0.01 to 0.02 cm per cm/min. A few tests were run with crosshead speeds of 0.0025 cm or 0.005 cm (0.001 inch or 0.002 inch) per minute and correspondingly lower average strain rates. The average specimen strain rate was determined by dividing $D_f - D_0 / D_0$ by the total test time.

Different specimens were tested at room temperature, 343°C (650°F), and 454°C (850°F). In the high-temperature tests, the specimens were held at temperature for 15 minutes before the load was applied to the mandrels. The values of $\Delta D / D_0$ were determined by measuring the diameters with a micrometer. The test results are given in Table 1. A large portion of the tests had to be terminated before specimen fracture occurred because of the limitations of machine load or mandrel height. These premature terminations happened most often in tests run at 343°C (650°F).

Strains to fracture ($\Delta D / D_0 \times 100$) of 13 to 27% were obtained at room temperature on the irradiated specimens. Two of these specimens are shown in Figure 1 after testing. An unirradiated specimen prepared from SA-1 (Fuel Cycle Program) archive tubing was strained 50% without fracture, in a room temperature test.

Specimens from rod A41 were increased in diameter as much as 69% without fracturing when tested with normal mandrels at 343°C (650°F). [The maximum wall-average operating temperature of the SA-1 cladding is estimated to have been between 329°C and 343°C (625°F and 650°F).] Specimen A41-13 ($\Delta D / D_0 = 69\%$) is shown after testing, in Figure 2, with a typical unirradiated specimen that has not been tested. A specimen from rod A13 tested at 343°C fractured at a $\Delta D / D_0$ of 69%. However, because the ring was not centered on the mandrel, it underwent an abnormal deformation, and at the end of the test one edge of the ring was flush with the flat surface of the mandrel (Figure 3). These conditions may have caused premature fracturing.

A specimen from rod A41 tested at 343°C on a mandrel that contained a longitudinal groove in its cylindrical surface was expanded 55% before breaking at the location of the groove. The groove was intended to simulate a radial crack in a fuel pellet. The ring and mandrel are shown at the end of the test, in Figure 4. The groove shows considerable deformation. It was not possible to observe what changes in form the groove may have undergone while the test was in progress, but it is believed that an unsupported circumferential increment was maintained across the width of the ring.

Two specimens from rod A41 tested at 454°C (850°F) fractured at strains of 34% and 45%. One of these specimens, A41-4, is shown in Figure 5.

The strains to fracture for specimens from rod A13 were considerably lower than those for specimens from rod A41, in tests at room temperature. The operating histories of these two fuel rods in terms of exposure time, power, and integrated neutron exposure, and such effects of operation as corrosion and increased hydrogen content, were quite similar. The cladding for the two rods was fabricated from two different lots of tubing; how this might be responsible for the differences in room temperature ductility has not yet been thoroughly investigated. No valid comparison of 343°C (650°F) ductility between specimens from the two rods can be made.

The effect of strain rate is uncertain. The greatest strains were realized in some of the highest strain rate tests, but none of the tests performed at one of the lower rates was carried out to specimen fracture.

Investigators with the Fast Ceramic Reactor Development Program, who have used the ring-on-compressed-cylinder test to measure the tensile ductility of irradiated Types-304 and -316L stainless steel cladding, have pointed out that in this type of test the friction between the mandrel and the ring develop surface shear forces which support the entire gage length of the specimen and thus suppress localized necking.* When mechanical interaction occurs during reactor operation, the fuel may support the cladding in a manner similar to the support provided the ring test specimens by the mandrels, but the relative extents of support are uncertain. In any event, it is expected that the strains to fracture obtained from ring tests will be

* In the present series of tests on the SA-1 cladding, the 1100 aluminum alloy mandrels used in most of the room temperature tests and the brass mandrels used in many of the 343°C tests folded over some fraction of the edge, or wall, surfaces of the specimens and, in so doing, may have caused the development of some stresses at these surfaces in addition to those at the inner surface of the rings. This "folding" did not occur in tests run with Zircaloy mandrels.

TABLE 1

SA-1 CLADDING RING TESTS

Specimen Number	Test Temperature °C (°F)	Average Strain Rate min ⁻¹	Mandrel Height cm (in.)	Mandrel Material	Specimen Strain $\frac{\Delta D}{D_0} \times 100$ (percent)	Specimen Fracture?
A41-1	Room ↓	0.017	1.3 (0.5) ↓	Zircaloy-2	4	No (a)
A41-9		0.012		Zircaloy-2	15	Yes ↓
A41-11		0.0013		6061 Al	18	
A41-14		0.011		1100 Al	25	
A41-17		0.012			27	
A13-25		0.009			13	
A13-23		0.010			17	No (a) ↓
Unirradiated		0.017			50	
A41-2	343 (650) ↓	0.015	1.3 (0.5) ↓	Zircaloy-2	25	No (a)
A41-5		0.013			19	Yes ↓
A41-6		0.0032			30	
A41-7		0.013		6061 Al	42 (c)	No (a)
A41-8		0.0015		Zircaloy-2	25	No (a)
A41-10		~0.01		304 S.S.	1	No (a)
A41-12		0.017		Brass	50	No (b)
A41-13		0.017			69	No (b)
A41-15		~0.001 (d)]			(d)	No
A41-16		0.018			55 (e)	Yes
A13-22		0.017			69 (f)	Yes
A41-3	454 (850)	0.015	1.3 (0.5)	Zircaloy-2	34	Yes
A41-4	454 (850)	0.017	1.3 (0.5)	Zircaloy-2	45	Yes

(a) Test terminated when machine load limit was reached.

(b) Test terminated when minimum mandrel height was reached.

(c) Mandrel folded over edges of specimen, and premature fracture may have resulted.

(d) Mandrel folded over edges of specimen to the extent that specimen could not be measured.

(e) Mandrel contained a 0.075-cm (0.03-inch)-wide by 0.09-cm-(0.035-in.)-deep longitudinal groove in its cylindrical surface.

(f) Specimen may have fractured prematurely. Specimen was not centered on mandrel and underwent abnormal deformation. One edge of ring was flush with flat surface of mandrel at end of test.

(a)

(b)

(c)

Figure 1. Ring Specimens and Mandrels After Testing at Room Temperature:
 (a) Specimen A41-14 (Side View),
 (b) Specimen A13-25 (Side View),
 (c) Specimen A13-25 (Top View).

(a)

(b)

Figure 2. Specimen A41-13 After Testing at 343°C (650°F): (a) Side View, and (b) Top View, shown with an Unirradiated Specimen that has not been tested.

Figure 3. Specimen A13-22 After Testing at 343°C (650°F).

Figure 4. Specimen A41-16 After Testing at 343°C (650°F) with Grooved Mandrel.

Figure 5. Specimen A41-4 After Testing at 454°C (850°F).

greater than both the total elongations obtained from those tensile tests in which the gage region is not supported, and the diametral deformations to fracture obtained from internal gas pressure (burst) tests in which necking instabilities are similarly unsupported.^(1,4)

The results from the tests on SA-1 cladding ring specimens do, in fact, contrast markedly with the results obtained from the tensile testing of longitudinally oriented miniature coupon specimens prepared from comparable irradiated SA-1 fuel rod cladding. The total elongations shown by the longitudinal coupons were only 3% in room temperature tests, approximately 1 to 1.5% at 343°C, and 5 to 12% at 454°C.²

It should be recalled, however, that in work performed under the AEC Fuel Cycle Program on the mechanical properties of irradiated Zircaloy-2 fuel rod cladding, burst specimens that were internally pressurized to failure with a liquid medium at room temperature also showed $\Delta D/D_0$ values that were considerably greater than the room-temperature total elongations shown by longitudinal coupon specimens cut from comparable cladding material. (The rods from which these specimens were prepared and the rods chosen to be incorporated in assembly SA-1 came from the same group of Zircaloy-2-clad VBWR fuel assemblies.) Burst specimens that were irradiated to a maximum fast fluence of about 1×10^{21} nvt >1 MeV showed strains to fracture of 13 to 16% in room-temperature tests, while longitudinal coupon specimens exposed to the same level showed total elongations of 2 to 5% at room temperature and 1 to 2% at 343°C.^{5,6} Investigators with the Fuel Cycle Program stated that this difference in ductility

between the transverse and longitudinal directions might be explained, in part, by differences in crystal orientation, as determined by x-ray diffraction on samples of unirradiated material.⁶

The single ring test of the present series that was run at 343°C with a grooved mandrel apparently indicates that the presence of an unsupported increment which is small with respect to the total length over which deformation is occurring does not necessarily "drastically reduce" the strain to fracture of the specimen. Additional, more refined tests of this general type would have to be run, however, before the effects and significance of "notched mandrels" could be fully evaluated.

3.2 FUEL BURNUP AND ISOTOPIC COMPOSITION ANALYSES (C. P. Ruiz, Technical Director)

Data on the americium and curium isotopic contents of full-cross-section fuel samples from rods A41, A46, and B70 were extended to include Am-242 and Cm-243, Cm-245, and Cm-246. The atom ratios of Am-242 to Am-243, and of the above curium isotopes to Cm-244, were measured by mass spectrometry. They were then converted to ratios relative to U-238 by applying the previously determined Am-243 and Cm-244 contents relative to U-238.

The Np-237 contents of full-cross-section fuel samples from rods A13, A41, and B70, and of two sets of radial samples—one from rod A13 and one from rod B70—also were measured.

The foregoing measurements complete the burnup and isotopic composition analyses scheduled to be made during the initial phase of the program. The complete, final data are compiled in Tables 2, 3, and 4.

TABLE 2

ANALYSES OF FULL CROSS-SECTION FUEL SAMPLES FROM
FUEL RODS A13, A41, A46, AND B70

Rod Number	Sample Number	Burnup		Atom Percent (Normalized to Total 100%)								
		Fissions/cm ³ (× 10 ⁻²⁰)	MWd/tU	Fission	²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu
A13	67	8.1	29,700	3.402	0.016	0.777	0.442	94.663	0.376	0.198	0.086	0.041
A41	68	7.8	28,500	3.266	0.015	0.736	0.484	94.838	0.347	0.192	0.080	0.041
A46	69	6.9	25,300	2.894	0.012	0.572	0.382	95.516	0.331	0.181	0.075	0.037
B70	70	6.9	25,300	2.889	0.012	0.622	0.372	95.449	0.352	0.188	0.079	0.036

Atom Ratios Relative to ²³⁸U (Corrected to time of fuel discharge)

Rod Number	Sample Number	²³⁷ Np ²³⁸ U	²³⁶ Pu ²³⁸ U	²³⁸ Pu ²³⁸ U	²⁴¹ Am ²³⁸ U	²⁴² Am ²³⁸ U	²⁴³ Am ²³⁸ U	²⁴² Cm ²³⁸ U	²⁴³ Cm ²³⁸ U	²⁴⁴ Cm ²³⁸ U	²⁴⁵ Cm ²³⁸ U	²⁴⁶ Cm ²³⁸ U	¹⁴⁸ Nd ²³⁸ U	¹³⁷ Cs ²³⁸ U	¹³⁴ Cs ²³⁸ U	¹⁰⁶ Ru ²³⁸ U	¹⁴⁴ Ce ²³⁸ U
		($\times 10^4$)	($\times 10^{10}$)	($\times 10^4$)	($\times 10^5$)	($\times 10^7$)	($\times 10^5$)	($\times 10^5$)	($\times 10^7$)	($\times 10^5$)	($\times 10^7$)	($\times 10^8$)	($\times 10^3$)	($\times 10^3$)	($\times 10^3$)	($\times 10^3$)	($\times 10^3$)
A13	67	3.4	4.9	1.14	6.5	NA	NA	1.58	NA	1.62	NA	NA	0.606	2.4	0.18	0.16	0.41
A41	68	3.4	5.5	1.06	5.5	5.3	5.6	1.52	3.32	1.48	5.59	6.8	0.573	2.3	0.16	0.15	0.42
A46	69	NA	3.8	0.87	2.7	4.4	4.8	1.11	2.33	0.93	2.81	3.8	0.522	1.9	0.13	0.17	0.31
B70	70	2.8	2.5	0.70	3.2	3.9	4.1	1.17	2.27	1.01	3.35	3.8	0.507	1.7	0.12	0.14	0.34

NOTES:

NA = not analyzed.

Nominal initial uranium enrichments were 3.53 at.% for rods A13 and A41, and 2.79 at.% for rods A46 and B70.

TABLE 3
ANALYSES OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS, FUEL ROD A13

Sample Number	Distance From Pellet Periphery cm(inch)	Burnup		Fission	Atom Percent (Normalized to Total 100%)								Atom Ratios Relative to ²³⁸ U (Corrected to time of fuel discharge)										
													²³⁷ Np	²³⁸ Pu	²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	¹⁴⁸ Nd	¹³⁷ Cs	¹³⁴ Cs	¹⁰⁶ Ru	¹⁴⁴ Ce
		Fissions/cm ³ (X 10 ⁻²⁰)	MWd/tU		²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	(X 10 ⁻⁴)	(X10 ⁻⁴)	(X10 ⁻⁴)	(X10 ⁻⁴)	(X10 ⁻⁵)	(X10 ⁻⁵)	(X10 ⁻³)	(X10 ⁻³)	(X10 ⁻³)	(X10 ⁻³)	(X10 ⁻³)
82	0.000-0.005 (0.000-0.002)	11.6	42,500	4.84	0.015	0.728	0.453	92.577	0.742	0.362	0.187	0.097	5.4	1.59	0.88	NA	3.00	3.33	0.850	3.1	0.23	0.60	0.46
83	0.005-0.010 (0.002-0.004)	10.6	38,600	4.39	0.015	0.716	0.454	93.314	0.598	0.293	0.147	0.074	4.5	1.39	NA	2.0	2.75	3.26	0.780	3.2	0.17	0.46	0.42
84	0.010-0.018 (0.004-0.007)	9.2	33,700	3.85	0.013	0.720	0.467	94.097	0.461	0.228	0.110	0.055	4.3	1.26	0.74	1.1	1.93	1.89	0.681	2.7	0.20	0.27	0.40
85†	0.018-0.025 (0.007-0.010)	9.1	33,400	3.83	0.015	0.726	0.465	94.274	0.372	0.189	0.086	0.042	3.9	1.16	0.80	1.0	1.74	1.83	0.656	2.4	0.18	0.13	0.38
86†	0.025-0.038 (0.010-0.015)	8.6	31,600	3.61	0.015	0.737	0.459	94.488	0.369	0.189	0.092	0.041	4.0	1.29	0.52	0.5	1.41	1.47	0.640	2.3	0.17	0.13	0.33
87†	0.038-0.051 (0.015-0.020)	8.3	30,400	3.49	0.015	0.742	0.453	94.660	0.346	0.181	0.076	0.036	3.9	1.18	0.58	0.4	1.20	1.25	0.616	2.1	0.16	0.10	0.32
88†	0.051-0.076 (0.020-0.030)	7.8	28,500	3.27	0.015	0.758	0.456	94.925	0.312	0.166	0.066	0.031	3.5	0.92	0.40	0.5	1.10	1.01	0.502	1.9	0.13	0.10	0.27
89†	0.076-0.127 (0.030-0.050)	7.6	27,700	3.17	0.015	0.758	0.454	95.043	0.301	0.161	0.065	0.031	3.6	1.02	0.32	0.5	1.16	1.07	0.555	1.5	0.11	0.11	0.23
90†	0.127-0.254 (0.050-0.100)	8.2	29,900	3.42	0.015	0.771	0.443	94.757	0.321	0.173	0.068	0.031	3.5	1.0	0.66	0.5	1.3	1.1	0.593	1.5	0.13	0.15	0.28
91†	0.254-center*	7.9	28,800	3.30	0.015	0.842	0.443	94.796	0.329	0.179	0.066	0.030	3.2	1.14	0.58	0.6	0.98	1.00	0.566	1.8	0.11	0.17	0.35

Notes:

NA = not analyzed.
Nominal initial uranium enrichment for rod A13 was 3.53 atom %.
*Nominal pellet radius = 0.478 cm (0.188 inch).

†Some of the uranium isotopic analyses on the designated samples were reported erroneously in the appendix of GEAP-10121.⁷ The necessary corrections have been applied to the values given in this table.

TABLE 4
ANALYSES OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS, FUEL ROD B70

Sample Number	Distance From Pellet Periphery [cm(inch)]	Burnup		Fission	Atom Percent (Normalized to Total 100 Percent)								Atom Ratios Relative to ²³⁸ U (Corrected to time of fuel discharge)										
													²³⁷ Np	²³⁸ Pu	²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	¹⁴⁸ Nd	¹³⁷ Cs	¹³⁴ Cs	¹⁰⁶ Ru	¹⁴⁴ Ce
		Fissions/cm ³ (× 10 ⁻²⁰)	MWd/tU		²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁸ U (×10 ⁴)	²³⁸ U (×10 ⁴)	²³⁸ U (×10 ⁴)	²³⁸ U (×10 ⁴)	²³⁸ U (×10 ⁵)	²³⁸ U (×10 ⁵)	²³⁸ U (×10 ³)	²³⁸ U (×10 ³)	²³⁸ U (×10 ³)	²³⁸ U (×10 ³)	²³⁸ U (×10 ³)
93	0.000-0.005 (0.000-0.002)	10.2	37,400	4.25	0.010	0.724	0.342	93.234	0.770	0.373	0.198	0.099	NA	1.68	0.76	0.6	2.92	3.34	0.763	3.1	0.24	0.28	0.54
94	0.005-0.010 (0.002-0.004)	9.1	33,300	3.79	0.012	0.569	0.375	94.082	0.634	0.305	0.158	0.074	4.5	1.26	0.67	0.5	2.47	2.28	0.691	2.7	0.19	0.33	0.45
95	0.010-0.018 (0.004-0.007)	7.8	28,600	3.25	0.012	0.572	0.375	94.962	0.450	0.219	0.107	0.051	3.1	1.01	0.46	0.3	1.46	1.44	0.593	2.1	0.15	0.21	0.37
96	0.018-0.025 (0.007-0.010)	7.1	25,900	2.96	0.012	0.572	0.371	95.348	0.397	0.199	0.093	0.048	3.6	1.08	0.35	0.5	1.25	1.26	0.543	2.1	0.15	0.18	0.34
97	0.025-0.038 (0.010-0.015)	6.9	25,100	2.87	0.012	0.574	0.361	95.519	0.360	0.181	0.083	0.039	2.8	1.01	0.33	0.3	1.15	1.14	0.511	2.1	0.15	0.26	0.35
98	0.038-0.051 (0.015-0.020)	6.5	23,900	2.74	0.012	0.593	0.369	95.718	0.307	0.158	0.070	0.033	NA	0.95	NA	0.2	1.04	0.73	0.487	1.7	0.14	0.14	0.28
99	0.051-0.076 (0.020-0.030)	6.8	25,000	2.86	0.012	0.597	0.367	95.598	0.310	0.160	0.066	0.031	3.4	0.96	0.16	0.2	1.12	0.88	0.513	1.9	0.14	0.13	0.30
100	0.076-0.127 (0.030-0.050)	6.8	24,900	2.85	0.012	0.607	0.361	95.513	0.355	0.188	0.075	0.038	3.3	1.01	0.18	0.2	0.92	0.81	0.464	2.0	0.14	0.15	0.32
101	0.127-0.254 (0.050-0.100)	6.7	24,400	2.79	0.012	0.621	0.360	95.625	0.322	0.173	0.066	0.031	3.1	0.88	0.28	0.2	0.89	0.84	0.493	2.0	0.14	0.28	0.33
102	0.254-center* (0.100-center)	6.2	22,600	2.59	0.012	0.636	0.364	95.834	0.306	0.166	0.063	0.029	2.4	0.79	0.20	0.2	0.77	0.68	0.458	1.5	0.11	0.12	0.25

Notes:

NA = not analyzed.

Nominal initial uranium enrichment for rod B70 was 2.79 at. %.

*Nominal pellet radius = 0.478 cm (0.188 inch).

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ACKNOWLEDGMENT

The Zircaloy-Clad UO₂ Fuel Rod Evaluation Program is sponsored by the Atomic Energy Commission, Division of Reactor Development and Technology, and is conducted by General Electric Company, with the following individuals contributing to the program during the report period:

G. P. Ferguson

IPO Materials Investigation

R. E. Smith }
D. S. Vash }

IPO Remote Handling Operation

C. P. Ruiz

NL Chemistry and Chemical Engineering Laboratory

J. P. Peterson }
F. R. Smith }

NL Analytical Chemistry

E. S. Darlin }
H. S. Rosenbaum }
U. E. Wolff }

NL Materials Structure and Properties

M. E. Carino }
G. E. Robinson }

IPO Applications Engineering

F. H. Megerth

RF&RD Uranium Fuel Development

DISTRIBUTION

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 Attn: B. Lustman

1

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 Oak Ridge Operations Office
 P.O. Box E
 Oak Ridge, Tennessee 37830
 Attn: D.E. Ferguson

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