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**ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM
PROGRESS REPORT NO. 13
NOVEMBER 1970-MARCH 1971**

COMPILED BY
F.H. MEGERTH

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

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**ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM**

**Progress Report No. 13
November 1970—March 1971**

Compiled by
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U. S. Atomic Energy Commission
Contract AT(04-3)-189
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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 Unit No. 1 for additional irradiation. The assembly consisted 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, ten 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 remaining rods operated a total of 1850 days. The two 44-rod subassemblies were shipped to VNC in March 1970, where the final phase of the program is now 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).
- h. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 9, November 1969-January 1970," February 1970 (GEAP-10160).
- i. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 10, February-April 1970," May 1970 (GEAP-10204).
- j. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 11, May-July 1970," August 1970 (GEAP-10217).
- k. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 12, August-October 1970," November 1970 (GEAP-10287).

2. SUMMARY

Earlier in the examinations, a neutron radiograph of one of the fuel rods showed a small region in the cladding that was believed to be possibly defective. Recent metallographic examination of this region, however, failed to reveal any defect. On another rod, a relative increase in cladding corrosion was noted near a prominent spacer spring contact mark. The external ZrO_2 layer was continuous and approximately $100\ \mu$ (0.004 in.) thick in the contact area. The corrosion layer around the rest of the rod section typically consisted of a $\sim 15\ \mu$ thick film with numerous intermittent oxide patches 60 to $90\ \mu$ thick. Metallographic examination of a section of a rod that showed considerable crud spalling indicated that the circumferential-average crud thickness was $28\ \mu$ (0.0011 in.), which amount was the maximum measured on any specimen. Although the external ZrO_2 on this specimen was above average in thickness, a few other specimens showed thicker oxide layers.

Cladding samples from selected fuel rods were analyzed for total hydrogen and other gaseous constituents by vacuum hot extraction followed by gas chromatography. The total volumes of gas evolved from the samples ranged from approximately 700 to 2200 microliters per gram of cladding. The analyses showed total hydrogen contents of 72 to 96%, carbon monoxide contents of 5 to 24%, and smaller amounts of carbon dioxide, nitrogen, oxygen, and methane. In terms of parts per million, the total hydrogen contents ranged from 46 to 143 ppm. The maximum absorption of corrosion-generated hydrogen was about 4%.

Samples of irradiated fuel rod cladding that were burst tested at room temperature and at 343°C (650°F), 399°C (750°F), and 454°C (850°F) showed respective hoop stresses at rupture of approximately $8000\ \text{kg/cm}^2$ (114 kpsi), $5100\ \text{kg/cm}^2$ (72 kpsi), $4900\ \text{kg/cm}^2$ (69 kpsi), and $4400\ \text{kg/cm}^2$ (62 kpsi). Irradiation increased the stress values 1400 to $1800\ \text{kg/cm}^2$ (19 to 25 kpsi). Strain to fracture values ranged from 2 to 8% among the irradiated samples, and did not show any clear temperature dependence between 343°C and 454°C .

Several full cross-section fuel samples were analyzed for burnup by measuring the neodymium isotopes, and for uranium and plutonium isotopic composition. Most of the samples were taken from the peak exposure locations of the rods and ranged from 8.4×10^{20} to 10.4×10^{20} fissions/ cm^3 (30,800 to 37,900 MWd/tU) in burnup. The ratios of peak burnup to average burnup, obtained from gamma scans, were applied to determine the rod-average burnups. These values ranged from 7.6×10^{20} to 9.3×10^{20} fissions/ cm^3 (27,900 to 33,900 MWd/tU). Samples also were taken from near the ends of the fuel columns of two high-exposure rods and analyzed for burnup and isotopic composition.

3. EXAMINATIONS AND ANALYSES OF FUEL RODS

3.1 FUEL ROD PERFORMANCE EVALUATION

3.1.1 Metallographic Examination of Rod Sections (D. L. Orton, R. E. Smith)

Three transverse rod section metallography specimens were examined in addition to those discussed in the preceding progress report.¹ One of the current specimens (No. A626 from rod A9) was prepared, to examine a possibly defective area in the cladding of the rod. The area showed on a neutron radiograph print as a $\sim 0.1\text{-cm}$ ($\sim 0.04\text{ in.}$)-long spot slightly darker than the surrounding background. Metallographic examination failed to reveal any reason for the spot. Nothing unusual was noted during the earlier visual examinations of the rod surface or on the diameter profile traces, and analyses of the contained gases indicated that the rod was sound.

During the visual examinations of the fuel rods, markings frequently were noted near points of contact with the SA-1 subassembly spacer springs.^{2,3} The most prominent of these are shown by corner rod B50. The surface appearance of this contact area can be seen in Figure 1.* Specimen No. A625 was prepared so that the area could be viewed in cross section. In general, the external corrosion layer around the specimen consisted of a $\sim 15\ \mu$ thick film with numerous intermittent oxide patches 60 to $90\ \mu$ thick. A relative increase in corrosion was noted in the vicinity of the prominent contact area in octant 3 and part of octant 2 (see Table 1). In this region the external ZrO_2 layer was continuous and 90 to $100\ \mu$ (0.0035 to 0.004 in.) thick.

*All illustrations are included in Section 4.

Table 1
THICKNESS OF CRUD AND ZrO₂ ON THE CLADDING OF TRANSVERSE METALLOGRAPHY SPECIMENS

Rod Designation Specimen Number Location, Operation, and Tubing Lot Number*	Octant	Even Film Average Thickness (microns)	ZrO ₂ on Outside Surface Patches		Overall Average Thickness (microns)	Weight Gain (mg/dm ² O ₂)**	Crud Average Thickness (microns)	ZrO ₂ on Inside Surface Average Thickness (microns)
			Average Thickness (microns)	Relative Amount				
A9 A626	1(0°)	8	21	0.11	9			8
a) 29.4(11-9/16)	2	8	14	0.04	8			9
b) 124.6(49-1/16)	3	10	18	0.06	10			9
c) 1860	4	9	15	0.06	9			9
d) 9.6(35,100)	5	10	25	0.09	11		Crud chemically removed	8
e) 40.3(127,000)	6	10	23	0.16	12			9
f) 140(440,000)	7	11	39	0.34	21			8
g) CA-2	8	10	32	0.20	14			9
	1 through 8	10	23	0.13	12	190		9
B50 A625	1(0°)	16	74	0.44	42			
a) 68.3(26-7/8)	2	13(0.8)	71	0.50	53			
		97(0.2)	—	0.00				
b) 223.5(88)	3	89	—	0.00	89			
c) 1874	4	16	79	0.60	54		crud chemically removed	
d) 7.4(27,200)	5	16	94	0.66	67			
e) 31.3(99,000)	6	14	84	0.30	35			
f) 54(170,000)	7	14	71	0.37	30			
g) CA-2	8	14	61	0.27	27			
	1 through 8	26	76	0.38	50	780		low
B56 A624	1(0°)	29	57	0.40	40		34	10
a) 36.5(14-3/8)	2	41	75	0.35	53		29	10
b) 191.8(75-1/2)	3	30	54	0.29	37		29	8
c) 1874	4	26	65	0.46	44		31	8
d) 9.4(34,300)	5	22	51	0.33	32		34	8
e) 39.0(123,000)	6	34	93	0.35	55		22	9
f) 117(370,000)	7	46	78	0.13	50		21	9
g) CA-2	8	43	139	0.20	62		20	9
	1 through 8	34	76	0.31	47	730	28	9

- *a) Distance from prepared surface of specimen to shoulder of short end plug (cm [in.]).
 b) Distance from bottom of Dresden Reactor active fuel zone (cm [in.]). (The SA-1 fuel column was approximately 192 cm [75-9/16 in.] long, which dimension includes a ~3 cm [1-1/4 in.] nonfueled gap between the top and bottom subassemblies. The bottom of the column was 58.6 cm [23-1/16 in.] above the bottom of the reactor active fuel zone, and the top of the column was 28.7 ± 4.4 cm [11-5/16 ± 1-3/4 in.] below the top of the reactor active fuel zone.)
 c) Total days of operation (hours VBWR thermal power greater than 5 MW ÷ 24 + hours of Dresden turbine-generator operation ÷ 24).
 d) Specimen burnup (fissions/cm³ × 10⁻²⁰ [MWd/tU]).
 e) Life-averaged heat flux for specimen (W/cm² [Btu/h-ft²]).
 f) Estimated maximum heat flux attained during operation (W/cm² [Btu/h-ft²]).
 g) Tubing lot.

** Weight gain in mg per dm² of rod surface = 15.6 × ZrO₂ thickness in microns.

Similar areas on rod B70 were examined during the initial phase of the program. The ZrO_2 ranged from 70 to 100 μ in thickness near the most prominent contact area on the earlier specimen.⁴ Thus, although rod B50 operated 1500 days in Assembly SA-1, compared with 840 days for rod B70, the corrosion shown by rod B50 in the spring contact region was not much greater than that shown by rod B70.

The prepared surface of specimen No. A625 included a second spring-rod contact area 90 degrees from the one just discussed. This second region showed some enhancement of corrosion, but not to the degree shown by the first contact area.

Specimen No. A624 was taken from a corner rod (B56) which appeared to have an unusually thick crud layer. The crud had spalled down to a continuous white ZrO_2 surface at several locations, one of which is shown in Figure 2. A portion of the crud and ZrO_2 layers are shown in cross-sectional view in Figure 3. The average thickness of crud on this specimen was 28 μ (0.0011 in.), which amount was the maximum measured on any specimen and approximately twice the average thickness measured on the specimens taken as a group. The weight gain shown by specimen No. A624 was 230 mg/dm² greater than the average for all 16 specimens, but it was still lower than the weight gains shown by 4 specimens.

3.1.2 Cladding Hydrogen Analysis (D. E. Rey)

Six cladding samples from four fuel rods were analyzed for total hydrogen and other gaseous constituents by vacuum hot extraction, followed by gas chromatography. The details of the procedures used in this work were presented in an earlier progress report.³ The results of the current analyses are given in Table 2. The total volumes of gas evolved from the samples ranged from approximately 700 to 2200 microliters per gram of cladding. The analyses showed total hydrogen contents of 72 to 96%, carbon monoxide contents of 5 to 24%, and carbon dioxide contents of 0.3 to 2%. The samples also contained up to 0.6% nitrogen and oxygen, and up to 0.05% methane. The maximum total gas content and % CO value are about twice the corresponding maximums determined on samples from rods that were removed from SA-1 after 1200 days of VBWR and Dresden operation.³ However, the % CO values are no greater than some of those obtained in the analyses of samples from rods that saw only VBWR operation under the Fuel Cycle Program.⁵ In terms of parts per million, the total hydrogen contents of the current set of samples ranged from 46 to 143 ppm. The maximum content of any of the earlier samples was 89 ppm. The maximum absorption of corrosion-generated hydrogen was still only about 4%.

3.1.3 Burst Testing of Cladding Samples (G. P. Ferguson)

3.1.3.1 Procedure

Samples of cladding that had been exposed to estimated fast fluences of 4×10^{21} to 5×10^{21} nvt ($E > 1$ MeV) were obtained for burst testing by sectioning out 20-cm (8 in.) lengths from selected fuel rods. All of the irradiated cladding samples came from tubing lot CA-2. One-half of the SA-1 rods were clad with tubing from this lot; hence it was the most representative and afforded the best selection of samples. For comparison, unirradiated control samples were cut from Type J Zircaloy-2 Fuel Cycle Program tubing that had been autoclaved. All of the control samples came from tubing lot CA-3; unfortunately, only tubing from this lot was in sufficient supply to prepare the required number. The unirradiated samples were tested in the hot cell according to the same procedure used on the irradiated samples.

Before an irradiated sample was tested, the crud and some of the external ZrO_2 were removed from ~ 2 cm ($\sim 3/4$ in.) of each end with emery cloth. Most of the fuel was left intact within the cladding; approximately 0.6 to 1.0 cm ($1/4$ to $3/8$ in.) was removed from one end so that wall thickness measurements could be made with a micrometer. The outside diameter of each sample was measured with a micrometer at the middle and at 5.1 cm (2 in.) from each end in two orientations 90 degrees apart. One end of the sample was sealed with a stainless steel Swagelok cap-type tube fitting. The other end was connected to a 1400 kg/cm² (20,000 psi) air-driven gas booster through 0.6 cm ($1/4$ in.) o.d. by 0.21 cm (0.083 in.) wall high-pressure tubing which was joined to the sample by means of another stainless steel Swagelok tube fitting. The length of sample between the fittings was approximately 16 cm ($6-1/4$ in.). The sample was placed in the center of a 3.8 cm ($1-1/2$ in.) i.d. by 0.6 cm ($1/4$ in.) wall by ~ 25 cm (10 in.) long steel shroud tube which was capped at each end.

Table 2
ANALYSES OF GASES EVOLVED FROM FUEL ROD CLADDING SAMPLES

Sample Number and Location*	Total Gas Volume (Microliters per Gram at STP)	Composition (percent by volume) (Normalized to Total 100 Percent)						Volumes of Constituents (microliters per gram at STP)						Hydrogen Content (ppm)		Weight Gain (mg/dm ² O ₂)***	Hydrogen Absorption**** (percent)
		H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	Terminal	Preirradiation**		
A15F(a)	659.7	77.57	ND	0.03	ND	21.34	0.79	511.7	<0.1	2.0	<0.1	140.8	5.2	46	28	180	3.4
(b)	738.9	78.47	0.09	0.60	0.01	20.08	0.75	579.8	0.7	4.4	0.1	148.4	5.5	52			
1) 88.4(34-13/16)																	
2) 65.6(25-13/16)																	
A15B(a)	1344.4	91.15	0.26	0.42	0.05	7.41	0.70	1225.4	3.5	5.6	0.7	99.6	9.4	109	28	440	4.5
(b)	988.4	93.49	0.13	0.51	0.03	5.33	0.50	924.1	1.3	5.0	0.3	52.7	4.9	83			
1) 24.3(9-9/16)																	
2) 129.7(51-1/16)																	
A49D(a)	2240.7	73.11	0.65	ND	0.05	23.99	2.20	1638.1	14.6	≤0.2	1.1	537.5	49.3	146	27	870	3.7
(b)	1678.2	85.88	0.19	ND	0.05	12.71	1.17	1441.2	3.2	<0.2	0.8	213.3	19.6	129			
1) 75.4(29-11/16)																	
2) 78.6(30-15/16)																	
B50D(a)	1029.7	94.75	0.01	0.44	ND	4.47	0.33	975.6	0.1	4.5	≤0.1	46.0	3.4	87	27	800	2.7
(b)	1380.3	93.51	0.21	0.26	0.01	5.59	0.41	1290.7	2.9	3.6	0.1	77.2	5.7	115			
1) 73.5(28-15/16)																	
2) 228.8(90-1/16)																	
B85B(a)	1290.5	89.17	0.49	0.63	0.04	9.01	0.66	1150.7	6.3	8.1	0.5	116.3	8.5	103	28	1092	1.9
(b)	1135.3	93.70	0.04	0.38	0.02	5.46	0.41	1063.8	0.5	4.3	0.2	62.0	4.7	95			
1) 24.3(9-9/16)																	
2) 179.5(70-11/16)																	
B85F(a)	983.4	79.94	0.29	0.33	0.03	18.55	0.85	786.1	2.9	3.2	0.3	182.4	8.4	70	28	550	2.2
(b)	900.2	85.85	ND	0.58	0.02	12.95	0.59	772.8	<0.1	5.2	0.2	116.6	5.3	69			
1) 88.4(34-13/16)																	
2) 243.7(95-15/16)																	

ND= Not Detected (<0.01%)
*1) Distance from center of specimen to shoulder of short end plug (cm[in.]).
2) Distance from bottom of Dresden Reactor active fuel zone (cm[in.]).
** Determined by General Electric Company under the AEC Fuel Cycle Program
on samples of tubing from corresponding lots
*** Weight gain in mg/dm² = 15.6 X circumferential average ZrO₂ thickness in microns measured on adjacent
metallography specimens¹
**** Based on average hydrogen content of corresponding "a" and "b" samples.

The sample then was pressurized to failure with argon at a rate of $53 \pm 21 \text{ kg/cm}^2 \cdot \text{min}$ ($750 \pm 300 \text{ psi/min.}$). (However, the rate at the time of rupture was unintentionally increased to $320 \text{ kg/cm}^2 \cdot \text{min}$ [4600 psi/min] in the case of sample B98C-1, and to $89 \text{ kg/cm}^2 \cdot \text{min}$ [1260 psi/min] in the case of the unirradiated 454°C [850°F] control sample.) The pressure was measured by both a Bourdon tube gage and a transducer. The transducer output was fed into a strain gage signal conditioner, and the readout was continuously recorded in terms of pressure on a strip chart recorder.

Tests were performed at nominal temperatures of 343°C (650°F), 399°C (750°F), and 454°C (850°F), in addition to room temperature. To carry out the high-temperature tests, the shrouded sample was placed in an electric resistance tube furnace. The sample temperature was measured by means of three chromel-alumel thermocouples located at the middle and at 5.1 cm (2 in.) from each end of the sample. The middle thermocouple was connected to a saturable reactor temperature controller. The sample was brought to the test temperature and held there 1/2 hour. It was then pressurized to failure. The temperature at the middle of the sample was somewhat higher than at the other locations measured. The three temperature readings were generally within $5\text{--}1\frac{1}{2}^\circ\text{C}$ to $8\text{--}1\frac{1}{2}^\circ\text{C}$ (10°F to 15°F) of each other throughout the test. In the case of the unirradiated 454°C control sample, the temperature at one end was $\sim 17^\circ\text{C}$ (30°F) lower than at the middle.

The diameter of a tested sample was measured with the micrometer at locations just outside of the burst region and, in most cases, at locations somewhat removed from the burst region as well. The measurements were made in two orientations, 90 degrees apart.

In addition to the foregoing measurements, the distance around a tested sample from one edge of the primary (longitudinal) fracture to the other was measured by using strips of 0.0025-cm (0.001 in.)-thick stainless steel foil, 0.16 cm (1/16 in.) to 0.32 cm (1/8 in.) wide by approximately 25 cm (10 in.) long. While the sample was held in a horizontal position, one of these foil strips—with small lead weights attached to each end—was wrapped around the fracture region at the location of apparent maximum circumferential elongation. Pulling the strip taut about the sample resulted in a crimping of the strip at the edges of the fracture. The distance between the crimp marks (C_f) was then measured out of cell by means of a scale and magnifying glass. The measured distances between crimp marks on duplicate foil strips were found to be within 0.025 to 0.05 cm (0.01 to 0.02 in.) of each other on each of the irradiated samples and on the room-temperature control sample. Reproducibility was poorer on the high-temperature control samples due to the irregular nature of the fractures shown by these samples.

3.1.3.2 Results

The control and irradiated samples are shown, after testing, in Figures 4 through 8, and the values of burst pressure, hoop stress at rupture, and percent strain to fracture are listed in Table 3. The $\Delta D/D_i$ values in the table were calculated from selected post-test diameter measurements made near the burst regions and the corresponding pre-test measurements. The locations at which these values were determined for the high test temperature samples are indicated in Figures 9 through 11. $\Delta D/D_i$ values determined at locations away from the burst regions also are designated. The circumferential elongation values $(C_f - C_i)/C_i \times 100$ in Table 3 were calculated from measurements made on the foil strips and the corresponding pre-test diameter measurements.

The average stresses at rupture for irradiated samples tested at room temperature and at nominal temperatures of 343°C , 399°C , and 454°C , were 8000 kg/cm^2 (114 kpsi), 5100 kg/cm^2 (72 kpsi), 4900 kg/cm^2 (69 kpsi), and 4400 kg/cm^2 (62 kpsi), respectively. Based on these average values for each pair of irradiated samples, the largest difference between irradiated and unirradiated control samples was shown in the 343°C tests (1800 kg/cm^2 or 25 kpsi) and the smallest difference was shown in the room temperature and 454°C tests (1400 kg/cm^2 or 19 kpsi).

It should be pointed out that while one of the room temperature samples was tested in a normal manner, the second was held at the highest test temperature investigated in the current series (i.e., 454°C) for the standard holding time before being pressurized at room temperature. Although the stress to rupture shown by the second sample was somewhat lower than that of the first, the difference in values probably was no greater than one might expect to obtain between two comparable non-heat-treated samples.

Table 3
BURST TESTS ON SA-1 CLADDING SAMPLES

Sample Number*	Test Temperature °C (°F)	Burst Pressure kg/cm ² (kpsi)	Hoop Stress kg/cm ² (kpsi)	Strain to Fracture Percent	
				$\frac{D_f - D_i}{D_i} \times 100$	Circumferential Elongation
Unirradiated (0)	Room	716 (10.19)	6650 (94.6)	9	8
B98A-2 (1)	Room	792 (11.27)	8320 (118.4)	—	8
A42C-1 (2)	Room**	828 (11.77)	7660 (109.0)	—	6
Unirradiated (3)	343 (650)	356 (5.07)	3330 (47.3)	27	≥30
B96A-2 (4)	348 (659)	564 (8.02)	4980 (70.8)	4	3
A47D-2 (5)	348 (659)	586 (8.34)	5200 (74.0)	8	7
Unirradiated (6)	403 (757)	338 (4.81)	3150 (44.8)	16	≥12
B96C-1 (7)	403 (758)	533 (7.58)	4960 (70.6)	3.4	5
B98C-1 (8)	393 (740)	532 (7.56)	4770 (67.8)	4	5
Unirradiated (9)	454 (850)	315 (4.48)	3010 (42.8)	23	≥16
B96C-2 (11)	454 (850)	488 (6.94)	4490 (63.8)	2.4	4
B50A-2 (13)	459 (859)	478 (6.80)	4270 (60.7)	6	6

* Numbers in parentheses refer to sample designations used in Figure 4.

** Sample was held at 454°C for 30 minutes, cooled to room temperature and pressurized to failure.

— could not be measured.

Most of the strain to fracture values shown by the unirradiated high test temperature samples were considerably greater than those shown by the room temperature test sample. However, irradiation markedly decreased the strains to fracture among the high temperature samples; it showed a comparatively small effect at room temperature. The percent strain to fracture did not show any clear temperature dependence between 343°C and 454°C.

It is of interest to note in Figures 9 through 11 that diametral expansions of 2 to 7% occurred at locations removed from the burst regions on the various irradiated high test temperature samples.

3.2 FUEL BURNUP AND ISOTOPIC COMPOSITION ANALYSES

(C. P. Ruiz, Technical Director; J. P. Peterson, F. R. Smith)

Burnup analyses, and uranium and plutonium isotopic composition analyses, were completed on 24 full cross-sectional fuel samples from 16 rods that had been selected from both subassemblies A-1 (bottom) and B-1 (top). The results are given in Table 4. The burnup values listed were determined by measuring the neodymium isotopes.

Most of the samples analyzed were taken from the peak exposure locations of the rods (i.e., 25 to 30 cm [10 to 12 in.] above or below the middle of the assembly) and ranged from 8.4×10^{20} to 10.4×10^{20} fissions/cm³ (30,800 to 37,900 MWd/tU). Samples from near the ends of the fuel columns of high exposure rods A37 and B62 also were analyzed. The axial locations of these samples are indicated in Figure 12.

The distribution of fissions among U-235, Pu-239, Pu-241, and U-238 are given for the different samples in Table 5.

The average burnups for the 16 rods ranged from 7.6×10^{20} to 9.3×10^{20} fissions/cm³ (27,900 to 33,900 MWd/tU). These values were obtained by applying the Cs-137 gamma scan results to the sample burnup data. The peak-to-average burnup ratios ranged from 1.06 to 1.11 among the "A" rods and from 1.14 to 1.17 among the "B" rods.³

Table 4
SAMPLE BURNUPS AND URANIUM AND PLUTONIUM ISOTOPIC COMPOSITIONS

Rod Number	Sample Number	Nominal Cold Enrichment (at.% ²³⁵ U)	Distance From Short End Plug Shoulder (cm[in.])	Burnup		Atom Percent (Normalized to Total 100%)								
				Fissions/cm ³ (X 10 ⁻²⁰)	MWd/tU	Fission	²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu
A9	100	3.53	26.4(10-3/8)	9.60	35,080	4.010	0.011	0.422	0.486	94.318	0.358	0.231	0.096	0.069
A11	101	3.53	↓	10.03	36,650	4.188	0.012	0.373	0.500	94.227	0.305	0.233	0.086	0.074
A33	102	3.24	↓	8.99	32,875	3.760	0.013	0.518	0.483	94.493	0.361	0.224	0.090	0.057
A34	103	3.24	↓	8.82	32,230	3.683	0.011	0.431	0.459	94.708	0.343	0.218	0.088	0.060
A35	104	2.79	↓	8.42	30,780	3.510	0.010	0.312	0.395	95.053	0.345	0.218	0.092	0.066
A37	105	3.53	26.4(10-3/8)	10.21	37,330	4.264	0.012	0.359	0.500	94.133	0.331	0.232	0.092	0.076
↓	106	↓	1.6(5/8)	9.96	36,400	4.160	0.012	0.301	0.497	94.346	0.309	0.216	0.085	0.074
↓	107	↓	4.1(1-5/8)	8.80	32,190	3.684	0.013	0.517	0.485	94.616	0.338	0.211	0.084	0.053
↓	108	↓	92.2(36-5/16)	7.25	26,510	3.040	0.013	0.859	0.445	95.055	0.318	0.174	0.065	0.030
↓	109	↓	94.8(37-5/16)	7.97	29,130	3.339	0.014	0.651	0.472	94.934	0.300	0.185	0.067	0.039
A40	110	3.24	26.4(10-3/8)	9.24	33,790	3.850	0.009	0.230	0.403	94.789	0.321	0.226	0.091	0.081
A42	111	2.79	↓	9.04	33,050	3.850	0.010	0.243	0.405	94.849	0.335	0.222	0.093	0.076
A43	112	2.79	↓	8.75	31,980	3.655	0.012	0.450	0.454	94.706	0.356	0.219	0.090	0.058
A47	113	2.79	↓	9.04	33,060	3.768	0.009	0.260	0.402	94.820	0.348	0.224	0.095	0.074
A48	114	2.79	↓	9.18	33,560	3.825	0.009	0.245	0.404	94.771	0.347	0.226	0.095	0.078
A49	115	2.79	↓	9.19	33,590	3.828	0.009	0.212	0.408	94.799	0.341	0.226	0.096	0.082
B62	116	3.53	29.4(11-9/16)	10.37	37,900	4.328	0.011	0.365	0.505	94.044	0.333	0.241	0.094	0.078
↓	117	↓	1.6(5/8)	9.89	36,130	4.131	0.012	0.329	0.502	94.345	0.305	0.222	0.084	0.071
↓	118	↓	4.1(1-5/8)	8.86	32,370	3.704	0.013	0.532	0.479	94.600	0.326	0.211	0.082	0.052
↓	119	↓	92.2(36-5/16)	6.19	22,620	2.596	0.015	1.118	0.403	95.238	0.332	0.151	0.058	0.020
↓	120	↓	94.8(37-5/16)	6.97	25,460	2.920	0.014	0.944	0.434	95.132	0.307	0.160	0.061	0.026
B91	121	2.79	29.4(11-9/16)	8.87	32,420	3.696	0.009	0.272	0.415	94.876	0.347	0.223	0.095	0.073
B96	122	2.79	↓	8.78	32,080	3.658	0.009	0.279	0.398	94.910	0.358	0.220	0.096	0.071
B98	123	2.79	↓	9.10	33,250	3.789	0.009	0.224	0.403	94.796	0.362	0.231	0.102	0.083

Table 5
DISTRIBUTION OF FISSIONS
AMONG URANIUM AND PLUTONIUM ISOTOPES

Rod Number	Sample Number	Nominal Cold Enrichment (at. % ^{235}U)	Distance From Short End Plug Shoulder (cm [in.])	Integrated Effective 2200 m/sec Flux (nvt $\times 10^{-22}$)	Atom Percent Fission					Burnup*	
					^{235}U	^{239}Pu	^{241}Pu	^{238}U	Total	Fissions/cm ³ ($\times 10^{-20}$)	MWd/tU
A9	100	3.53	26.4 (10-3/8)	3.265	2.546	1.038	0.186	0.263	4.033	9.65	35.280
A11	101	3.53	↓	3.453	2.588	1.107	0.198	0.269	4.162	9.97	36,430
A34	103	3.24	↓	3.018	2.300	0.994	0.134	0.226	3.653	8.74	31,960
A35	104	2.79	↓	3.253	2.032	1.076	0.175	0.217	3.500	8.39	30,680
A37	105	3.53	26.4 (10-3/8)	3.515	2.600	1.145	0.201	0.271	4.217	10.10	36,910
↓	106	↓	1.5 (5/8)	3.791	2.648	1.045	0.197	0.255	4.144	9.92	36,250
↓	107	↓	4.1 (1-5/8)	2.951	2.468	0.858	0.133	0.226	3.685	8.81	32,200
↓	108	↓	92.2 (36-5/16)	2.164	2.184	0.596	0.072	0.186	3.039	7.25	26,500
↓	109	↓	94.8 (37-5/16)	2.591	2.356	0.681	0.095	0.205	3.337	7.96	29,110
A42	111	2.79	26.4 (10-3/8)	3.624	2.089	1.191	0.207	0.233	3.720	8.93	32,630
A47	113	2.79	↓	3.525	2.075	1.196	0.200	0.233	3.704	8.89	32,490
A48	114	2.79	↓	3.612	2.087	1.226	0.211	0.236	3.761	9.03	33,000
A49	115	2.79	↓	3.835	2.116	1.224	0.227	0.236	3.802	9.13	33,360
B62	116	3.53	29.4 (11-9/16)	3.488	2.596	1.190	0.204	0.273	4.264	10.22	37,340
↓	117	↓	1.5 (5/8)	3.649	2.624	1.040	0.188	0.266	4.118	9.86	36,020
↓	118	↓	4.1 (1-5/8)	2.904	2.455	0.864	0.131	0.237	3.688	8.82	32,220
↓	119	↓	92.2 (36-5/16)	1.664	1.912	0.481	0.046	0.159	2.597	6.19	22,630
↓	120	↓	94.8 (37-5/16)	2.017	2.115	0.549	0.062	0.186	2.913	6.95	25,390
B91	121	2.79	29.4 (11-9/16)	3.429	2.061	1.161	0.196	0.228	3.646	8.75	31,980
B96	122	2.79	↓	3.417	2.059	1.138	0.188	0.226	3.611	8.66	31,670
398	123	2.79	↓	3.749	2.087	1.218	0.226	0.250	3.782	9.08	33,180

*from heavy element data

4. ILLUSTRATIONS

This section contains Figures 1 through 12, as referenced in the preceding text.

Figure 1. Subassembly Spacer Spring Marking on Rod 850

Figure 2. Region of Rod B56 Where Crud Has Spalled from Surface

Figure 3. Portion of Rod B56 Cladding Showing Thick Crud and External ZrO_2 (Specimen No. A624)

Figure 4. SA-1 Cladding Samples After Burst Testing
(See Table 3 for key to sample designation)

Figure 5. Burst Samples Tested at Room Temperature: Unirradiated (above); Irradiated Sample A42C-1 (right) Which Was Held at 454°C (850°F) for 1/2 Hour Before Testing

*Figure 6. Burst Samples Tested at $\sim 343^{\circ}\text{C}$ (650°F):
Unirradiated Sample No. 3 (left); Irradiated Sample A47D-2 (right)*

Figure 7. Burst Samples Tested at $\sim 399^{\circ}\text{C}$ (750°F):
Unirradiated Sample No. 6 (left); Irradiated Sample B98C-1 (right)

Figure 8. Burst Samples Tested at $\sim 454^{\circ}\text{C}$ (850°F):
Unirradiated Sample No. 9 (left, top); Irradiated Samples B50A-2 (left, bottom) and B96C-2 (right)

Figure 9. Values of $(D_f - D_i)/D_i \times 100$ at Various Points Along Samples Tested at a Nominal Temperature of 343°C (650°F). (Two determinations 90 degrees apart were made at each axial location indicated.)

Figure 10. Values of $(D_f - D_i) / D_i \times 100$ at Various Points Along Samples Tested at a Nominal Temperature of 399°C (750°F)

Figure 11. Values of $(D_f - D_i) / D_i \times 100$ at Various Points Along Samples Tested at a Nominal Temperature of 454°C (850°F)

Figure 12. Locations of Burnup and Isotopic Composition Samples from Rods A37 and B62 in Terms of Distance from the Bottom of the Reactor Active Fuel Zone

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