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**ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM
QUARTERLY PROGRESS REPORT NOS. 6 AND 7
FEBRUARY - JULY 1969**

COMPILED BY
F.H. MEGERTH

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

REACTOR FUELS AND REPROCESSING DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM

Quarterly Progress Report Nos. 6 and 7
February - July 1969

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REACTOR, FUELS AND REPROCESSING DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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SECTION I

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/cc (10,000 MWd/tU). To demonstrate at the earliest possible time the capability of Zircaloy-clad fuel to achieve average burnups of at least 6×10^{20} fissions/cc (20,000 MWd/tU) 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 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 has been operated continuously from June 1964 to the present time 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 ten rods is in progress.

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

1. "Quarterly Progress Report No. 1, November 1967 - January 1968," GEAP-5598.
2. "Quarterly Progress Report No. 2, February 1968 - April 1968," GEAP-5624.
3. "Quarterly Progress Report No. 3, May 1968 - July 1968," GEAP-5667.
4. "Quarterly Progress Report No. 4, August 1968 - October 1968," GEAP-5702.
5. "Quarterly Progress Report No. 5, November 1968 - January 1969," GEAP-5746.

SECTION II

SUMMARY

Assembly SA-1 continued to operate in the Dresden reactor throughout the report period.

Longitudinal tensile coupon specimens obtained from the cladding of several of the removed fuel rods were tested at room temperature, 343°C (650°F), 399°C (750°F), and 454°C (850°F). The integrated fast neutron exposure of this material is estimated to be 3×10^{21} to 4×10^{21} nvt > 1 MeV. The high temperature ultimate strengths ranged from about 5000 kg/cm² (70 ksi) at 343°C to about 3000 kg/cm² (40 ksi) at 454°C. These values represent a total radiation-induced strength increase of approximately 1000 kg/cm² (at 454°C) to 2500 kg/cm² (at 343°C). Increases in yield strength were comparatively greater; the differences between ultimate strength and yield strength in the irradiated material were generally ≤ 70 kg/cm² (1 ksi) in the high temperature tests. The uniform elongations were typically 0.5 percent or less. Total elongations were 1 to approximately 1.5 percent at 343°C, up to 2 percent at 399°C, and ranged from 5 to 12 percent at 454°C. The average uniform elongations of the unirradiated archive material ranged from 10 percent at 454°C to 13 percent at 343°C, and the total elongations ranged from 17 percent at 454°C to 22 percent at 343°C.

The average 343°C ultimate strength of the present irradiated specimens was about 700 kg/cm² (10 ksi) greater than the 343°C ultimate

strength of comparable specimens irradiated in the VBWR to $\sim 1 \times 10^{21}$ nvt > 1 MeV, and the total elongation was ~ 0.3 percent lower. There were no distinct differences in the ultimate-yield strength differential or in the uniform elongation.

Similarly, microhardness measurements on current SA-1 cladding specimens indicated some increase in hardness when compared with specimens from Fuel Cycle Program fuel rods whose exposure was limited to VBWR operation.

Samples of SA-1 fuel rod cladding that had been exposed to a fast neutron fluence of approximately 4×10^{21} nvt > 1 MeV were examined by transmission with a 100 keV electron microscope. So far, no evidence of radiation-induced voids has been observed.

The radial dependence of burnup, uranium and plutonium isotopic contents, and the abundances of Am-241 and -242, Cm-244, Nd-148, Cs-137 and -134, Ru-106, and Ce-144 was determined from analyses of fuel samples taken at different radial locations from one of the relatively low exposure fuel rods. These data are presented in Table 5.

The Pu²³⁶ contents were measured on full cross section fuel samples from four rods. The atom ratios of Pu²³⁶ to U²³⁸ ranged from about 3×10^{-10} to 5×10^{-10} . The fuel sample burnups were between 7×10^{20} and 8×10^{20} fissions/cc (25,000 and 30,000 MWd/tU).

SECTION III

ASSEMBLY OPERATION

The Dresden reactor operated throughout the months of February through July. The average and maximum thermal powers for the reactor during this period were approximately 540 MW and 700 MW, respectively. Estimated thermal performance and exposure data on Assembly SA-1 are given in Table 1. Reactor water chemistry data, presented in Table 2, were extracted from the reports of the Common-

wealth Edison Company on Dresden operation for the months of December 1968 through June 1969 and are the most recent that are presently available.

The irradiation of Assembly SA-1 will be terminated at the end of the current reactor operating cycle, which is now scheduled for September 1969.

TABLE 1
THERMAL PERFORMANCE AND EXPOSURE OF SA-1
(UNREVISED)

Reactor Power (MWT)	<u>February - July 1969</u>			
	<u>540 (Average)</u>		<u>700 (Maximum)</u>	
	<u>Average</u>	<u>Peak</u>	<u>Average</u>	<u>Peak</u>
SA-1 Thermal Performance				
Heat Flux, W/cm ² (Btu/h-ft ²)	26 (83,000)	47 (147,000)	37 (118,000)	66 (209,000)
Specific Power, kW/kgU	13.7	24.2	19.5	34.5
Power Density, kW/l	24.1	42.6	34.4	60.9
<hr/>				
	<u>January 31, 1969</u>		<u>July 31, 1969</u>	
Assembly Average Exposure fissions/cc x 10 ⁻²⁰ (MWd/tU)	8.3 (30,300)		8.8 (32,000)	

TABLE 2
DRESDEN REACTOR WATER CHEMISTRY

	<u>December 1968</u>			<u>January 1969</u>			<u>February 1969</u>			<u>March 1969</u>		
	<u>Min</u>	<u>Avg</u>	<u>Max</u>	<u>Min</u>	<u>Avg</u>	<u>Max</u>	<u>Min</u>	<u>Avg</u>	<u>Max</u>	<u>Min</u>	<u>Avg</u>	<u>Max</u>
pH	6.1	6.8	8.5	5.8	6.87	7.8	6.00	7.00	7.90	6.5	7.5	8.0
Conductivity (μ mhos)	0.27	0.38	0.55	0.255	0.35	0.49	0.30	0.41	0.60	0.285	0.41	0.77
Turbidity (APHA)	1.2	4.1	19.0	2.2	4.05	5.6	2.20	5.20	15.60	2.0	3.9	6.0
Chloride (ppm)	0.01	0.032	0.071	0.016	0.045	0.083	0.02	0.05	0.15	0.020	0.044	0.122
	<u>April 1969</u>			<u>May 1969</u>			<u>June 1969</u>					
	<u>Min</u>	<u>Avg</u>	<u>Max</u>	<u>Min</u>	<u>Avg</u>	<u>Max</u>	<u>Min</u>	<u>Avg</u>	<u>Max</u>			
pH	6.64	7.62	8.23	6.5	7.2	8.3	5.80	6.62	7.50			
Conductivity (μ mhos)	0.29	0.47	0.84	0.300	0.375	0.565	0.28	0.43	0.58			
Turbidity (APHA)	2.00	4.30	14.80	2.0	3.4	5.4	2.00	5.77	23.20			
Chloride (ppm)	0.026	0.070	0.088	0.018	0.034	0.067	0.013	0.036	0.076			

SECTION IV

EXAMINATIONS AND ANALYSES OF FUEL RODS

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

4.1 FUEL ROD PERFORMANCE EVALUATION

4.1.1 Cladding Mechanical Properties Testing4.1.1.1 Tensile Tests on Longitudinal Miniature Coupon Specimens - G. P. Ferguson
G. F. Rieger

Longitudinally oriented tensile coupon specimens were obtained from the cladding of several irradiated SA-1 fuel rods. To prepare a specimen, the fuel rod was sectioned transversely to the desired length with an abrasive wheel. This section was then slit longitudinally to obtain two tensile coupon blanks. A hot cell-adapted Tensilkut machine and special holding fixtures were used to machine a reduced gauge length region on each coupon blank. A sketch of a tensile specimen is shown in Figure 1. The gauge width and cladding thickness were measured with a micrometer.

Similar specimens were prepared from unirradiated archive tubing to provide comparative data. Some of these specimens were prepared remotely, while others were prepared out-of-cell with either a Tensilkut machine or milling machine.

All of the irradiated specimens and some of the unirradiated specimens were tested with an Instron Model TT-C-L hot cell tensile testing machine. All specimens were pulled at a strain rate of either 0.05 or 0.065 cm/cm/min (0.02 or 0.025 in./in./min). The uniform and total extension values on the remotely tested specimens were determined from traces of load versus time and the crosshead speed of the testing machine. It was hoped that these specimens could be tested in a shielded out-of-cell facility where a photographic method of

extensometry can be applied. However, attempts to reduce the radioactive contamination on these specimens to a level low enough for handling in this facility were not successful. Although improvements in the decontamination procedure may permit such out-of-cell tests, it was felt that further efforts in this direction on this group of specimens were not warranted.

Specimens were tested at room temperature, 343°C (650°F), 399°C (750°F), and 454°C (850°F). The specimens were normally held at the testing temperature 15 minutes before tension was applied.

The integrated fast flux exposure for the irradiated material is estimated to be between 3×10^{21} and 4×10^{21} nvt > 1 MeV.

The results of the tests are presented in Table 3. As expected, the irradiated material showed significantly increased strength and decreased ductility compared to the unirradiated Fuel Cycle Program archive material. The least effect of irradiation on yield strength, ultimate strength, and total elongation was shown at 454°C (850°F).

At room temperature (22°C or 72°F) the average yield and ultimate strengths of the irradiated specimens were 6900 kg/cm² (97 ksi) and 7700 kg/cm² (110 ksi), and the uniform and total elongations were about 1 percent and 3 percent. The high temperature ultimate strengths ranged from about 5000 kg/cm² (70 ksi) at 343°C (650°F) to about 3000 kg/cm² (40 ksi) at 454°C (850°F). The differences between ultimate strength and yield strength were generally ≤ 70 kg/cm² (1 ksi) in the high temperature tests, although there were some exceptions at 343°C. The uniform elongations were typically 0.5 percent or less. Total elongations were 1 to approximately 1.5 percent at 343°C, up to 2 percent at 399°C (750°F), and ranged from 5 to 12 percent at 454°C.

FIGURE 1. SCHEMATIC DIAGRAM OF TENSILE SPECIMENS OBTAINED FROM SA-1 FUEL CLADDING

TABLE 3
TENSILE TESTS ON LONGITUDINAL COUPON CLADDING SPECIMENS

Specimen Identification	Tubing Lot Number	Estimated Fast Fluence Exposure ₂₁ [nvt (>1 MeV) x 10 ⁻²¹]	Test Temperature [°C (°F)]	0.2% Offset Yield Strength		Ultimate Tensile Strength		Elongation [% in 2.03 cm (0.8 in.)]	
				kg/cm ²	ksi	kg/cm ²	ksi	Uniform	Total
1	CA-3	0	Room ↓	2570	36.6	5170	73.6	14.4	24.5
5	CA-2	0		3160	45.0	4500	64.0	14.8	>23.6
6	CA-2	0		3300	47.0	4650	66.2	17.2	27.4
13	CA-4	0		3620	51.5	5050	71.8	18.0	31.6
14	CA-3	0		3800	54.0	5200	74.0	15.8	28.0
A13-I2	CA-2	3.8		6920	98.4	7870	111.9	1.1	3.0
A41-G1	CA-4	3.8		6550	90.3	7450	106.0	0.8	3.2
A41-G2	CA-4	3.8		7250	103.1	7920	112.7	0.8	3.0
2	CA-3	0		1450	20.6	2410	34.3	11.8	19.6
7	CA-2	0		1550	22.0	2710	38.5	14.7	22.2
8	CA-2	0	343 (650) ↓	1480	21.0	2460	35.0	11.0	20.5
15	CA-4	0		1650	23.5	2460	35.0	15.2	26.8
A13-B1	CA-2	3.8		4060	57.8	4640	66.0	0.7	1.2
A13-B2	CA-2	3.8		4650	66.1	4900	69.7	0.4	1.4
A41-B1	CA-4	3.8		4660	66.3	4680	66.5	0.3	1.2
A41-B2	CA-4	3.8		3890	55.3*	5210	74.1	1.1*	2.2*
A46-D1	CA-2	3.4		4710	67.0	4750	67.5	0.3	1.1
A46-D2	CA-2	3.4		3300	46.9*	4650	66.2	1.7*	3.0*
A46-M1	CA-2	3.2		4800	68.3	4800	68.3	0.2	1.0
B97-F1	CA-2	3.6		5320	75.7	5360	76.2	0.3	1.1
A46-L2	CA-2	3.2	343 (650) [†]	4630	65.9	4750	67.6	0.3	1.2
A46-O2	CA-2	3.1	343 (650) [#]	4810	68.4	4810	68.4	0.2	1.7
3	CA-3	0	399 (750) ↓	1420	20.2	2180	31.0	7.7	18.3
9	CA-2	0		1490	21.2	2070	29.5	17.0	21.2
10	CA-2	0		1620	23.0	2250	32.0	14.6	19.5
17	CA-3	0		1790	25.4	2400	34.2	12.9	19.8
18	CA-3	0		2170	30.8	2710	38.5	10.8	19.2
A46-C1	CA-2	3.4		4190	59.6	4210	59.9	0.4	2.0
A46-C2	CA-2	3.4		4360	62.0	4390	62.4	0.3	0.6
A46-L1	CA-2	3.2		4330	61.6	4330	61.6	0.2	1.9
4	CA-3	0		1400	19.9	1980	28.1	6.8	19.9
11	CA-2	0		1300	18.5	2140	30.5	12.4	16.6
12	CA-2	0	454 (850) ↓	1320	18.8	2100	29.8	9.6	16.2
19	CA-3	0		1510	21.5	2180	31.0	12.2	19.5
20	CA-3	0		2060	29.3	2430	34.6	7.0	14.2
A46-J1	CA-2	3.2		2690	38.3	2730	38.8	0.5	12.1
A46-J2	CA-2	3.2		3220	45.8	3290	46.8	0.5	5.4
A46-M2	CA-2	3.2		2490	35.4	2560	36.4	0.6	9.7
A46-O1	CA-2	3.1	454 (850) [†]	2720	38.7	2860	40.7	0.6	5.0

*Accuracy of these values is in doubt. Specimens may have slipped in grips.

[†]Held at designated test temperature 1 hour prior to testing.

[#]Held at designated test temperature 16-1/2 hours prior to testing.

NOTE: Unirradiated specimens 1 through 4 were prepared out of cell with a Tensilkut machine and were tested in cell. Specimens 5 through 12 were prepared out of cell with a milling machine and were tested out of cell. Specimens 13 through 15, and 17 through 20 were prepared in cell with a Tensilkut machine and were tested out of cell.

The average 343°C ultimate strength of the present specimens was about 700 kg/cm² (10 ksi) greater than the average 343°C ultimate strength of comparable specimens irradiated in the VBWR to $\sim 1 \times 10^{21}$ nvt > 1 MeV, and the total elongation was ~ 0.3 percent lower. There were no distinct differences in the ultimate-yield strength differential or in the uniform elongation. Thus, the present data agree with the contention of earlier investigators that the effect of irradiation on the ductility of this material, as measured by elongation, essentially reached saturation during the initial 10^{21} nvt of exposure⁽¹⁾.

The ultimate strength, ultimate-yield strength differential, and uniform elongation of one of the current specimens held at 343°C for 16-1/2 hours and then tested at that temperature fell within the ranges of values for specimens held at temperature for the normal 15-minute soaking time. The total elongation was slightly higher, 1.7 percent. The strength and elongation values for one specimen held at 454°C for 1 hour and then tested at 454°C fell within the range of values obtained on comparable specimens held at temperature for 15 minutes.

4.1.1.2 Microhardness Measurements on Metallography Specimens - R. E. Smith

Measurements of diamond pyramid hardness (DPH) were made on the cladding of two transverse and one longitudinal fuel rod section metallography specimens. Three-by-three arrays of indentations were made on each specimen at two locations approximately 180° apart, or on opposite wall sections in the case of the longitudinal specimen. A typical array of indentations is shown in Figure 2.

The DPH values obtained are given in Table 4. The average hardnesses were 217 and 223 DPH for the transverse specimens and 246 DPH for the longitudinal specimen. Although the exposure of the longitudinal specimen was somewhat higher than the exposures of the transverse specimens, an additional directional difference in hardness was apparent.

The present measurements are probably most significant when compared with data on lower exposure and unirradiated Fuel Cycle Program cladding. The microhardness of an unirradiated transverse specimen was 158 DPH, and that of a longitudinal specimen was 195 DPH. Hardness measurements made on several transverse cladding sections exposed in the VBWR to between 1×10^{21} and about 1.5×10^{21} nvt > 1 MeV showed a maximum value of 201 DPH and an overall average value of 193 DPH.⁽¹⁾

No general difference in microhardness was noted with respect to the outside, center, and inside locations of the cladding wall. With the possible exception of the longitudinal specimen, the hardnesses measured at the two locations of a given specimen were about the same. However, the locations of the arrays of indentations with respect to the orientation of the rods in the fuel assembly during operation are not known.

4.1.2 Electron Microscopic Examination of Cladding and UO₂

H. S. Rosenbaum, Technical Director
U. E. Wolff
W. H. Kimball

4.1.2.1 Cladding

Thin foils of unirradiated archive cladding were prepared for transmission electron microscopic examination by first slitting a tube section longitudinally and flattening the resulting half rings. The pieces were ground on both sides to remove the oxide and then prethinned chemically to a thickness of about 0.013 to 0.025 cm (0.005 to 0.010 in.). Discs 3 mm in diameter were punched from these pieces. Finally, the centers of the discs were thinned electrolytically in a twin-jet electro-polisher.

Samples of SA-1 cladding that had been exposed to a fast neutron fluence of approximately 4×10^{21} nvt > 1 MeV were prepared by first slitting and straightening sections of tubing from fuel rod A13 and punching 3 mm diameter discs from these sections in a hot cell. The

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FIGURE 2. TYPICAL ARRAY OF DIAMOND PYRAMID HARDNESS INDENTATIONS (SPECIMEN NO. 9957).

TABLE 4
MICROHARDNESS OF SA-1 CLADDING

Fuel Rod	Metallography Specimen Number	Orientation	Estimated Fast Fluence Exposure nvt (>1 MeV) $\times 10^{-21}$	Location	DPH, 0.5 kg Load			
					Outside Diameter	Center	Inside Diameter	Average
A41	9956	Longitudinal	3.8	1	240	246	228	240
				2	253	246	254	251
A41	9957	Transverse	3.6	1	218	224	221	221
				2	228	226	222	225
A46	9958	Transverse	3.3	1	219	219	213	217
				2	219	214	217	217

discs then were prethinned by hand grinding and finally thinned with the twin-jet electro-polisher.

Examinations of the thinned material were initiated at the Vallecitos Nuclear Center with a 100 KeV electron microscope. Structures typical of the unirradiated Zircaloy are shown in Figure 3, and representative areas of the irradiated material are shown in Figure 4. Evidence of long dislocations can be seen in both the unirradiated and irradiated material. In the irradiated cladding there also appear to be areas containing defects. However, it has not yet been established whether these are surface defects (pits or oxides) or internal defects associated with irradiation. So far no evidence of radiation-induced voids has been observed.

4.1.2.2 UO₂

Samples of unirradiated UO₂ were thinned both to help establish the most feasible method of preparation that can be applied to irradiated UO₂ and to provide a reference for interpreting the transmission electron micrographs that are to be obtained on the irradiated material.

Ion bombardment was one technique employed to obtain thin foils of UO₂. First, a thin slice from a UO₂ pellet was mechanically ground and polished to a thickness of approximately 0.013 cm (0.005 in.). Discs 3 mm in diameter were trepanned from this wafer with an ultrasonic drill. Each disc was reduced to a thickness of approximately 20 μ m by mounting it in a recess that had been etched into a glass slide and then grinding, polishing, and attack-polishing, first on one side and then on the other.

An Ion Micro Milling Instrument (IMMI), manufactured by Commonwealth Scientific Corporation, was used to thin the mechanically prepared discs to electron transparency (about 1000 to 2000 Å). The main part of the machine is a vacuum chamber with two ion guns operated at approximately 5 KV and with an argon pressure about 10^{-4} torr. The streams of argon ions produced in the ion guns impinge onto both sides

of the sample, which is mounted on a turntable with the sample surface forming an angle of 15° with the ion beams. During bombardment the sample is rotated in its own plane.

In addition to the above method, a technique similar to one that has been used by Katz was employed to produce pieces of unirradiated UO₂ thin enough for electron transmission.⁽²⁾ A sintered UO₂ pellet was ground through 600 grit SiC paper, cellulose acetate replicating tape was applied to the surface, and indentations were made with 1000 gm on a Tukon hardness tester through the acetate layer. Successive replicas were then applied to the surface and stripped. Although the first replicas lifted out UO₂ chips that were too thick for electron transmission, later replicas contained thin slivers which on a secondary carbon replica could be used for transmission electron microscopy. In addition to the chips, the hardness indenter produced fracture surfaces which were replicated with a two-stage carbon replica for electron microfractography.

Such a fracture surface is shown in Figure 5. It is partly intergranular (smooth areas) and partly transgranular (areas with cleavage steps). Numerous sintering pores in a wide range of sizes can also be seen. Figure 6 is an example of the internal structure of a UO₂ sliver. One pore (lighter than the background) is present in this area.

4.2 FUEL BURNUP AND ISOTOPIC COMPOSITION ANALYSES

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Fuel samples taken at different radial locations from the maximum burnup location of a relatively low exposure fuel rod (B70) were analyzed during the period. The initial enrichment of the UO₂ in this rod was 2.76 weight percent U-235, and the cross-sectional average burnup at the sample location was 6.9×10^{20} fissions/cc (25,300 MWd/tU). [In the case of the set of radial samples from rod A13,

(a)

10,500x

(b)

30,000x

FIGURE 3. UNIRRADIATED SA-1 (FUEL CYCLE PROGRAM) ARCHIVE CLADDING MATERIAL

(a)

30,000x

(b)

30,000x

**FIGURE 4. SA-1 FUEL ROD CLADDING EXPOSED TO APPROXIMATELY
 4×10^{21} nvt >1 MeV**

4500x

FIGURE 5. FRACTURE SURFACE ON UNIRRADIATED UO_2 (REPLICA)

45,000x

FIGURE 6. TRANSMISSION MICROGRAPH OF THIN UNIRRADIATED UO_2 SLIVER

reported in Quarterly Progress Report No. 5, the initial enrichment was 3.5 percent, and the average burnup was 8.1×10^{20} fissions/cc (29,700 MWd/tU).⁽³⁾ Presented in Table 5 as functions of radial location are burnup, uranium and plutonium isotopic compositions, and the atom ratios relative to U-238 of Pu-239

and -238, Am-241 and -242, Cm-244, Nd-148, Cs-137 and -134, Ru-106, and Ce-144.

In addition to the above analysis, the Pu²³⁶ contents were measured on full cross-section fuel samples from four fuel rods and are given in Table 6.

TABLE 5
ROD B-70
ANALYSES OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS

Sample Number	Distance From Pellet Periphery (cm (inch))	Burnup			Uranium Isotopes		Plutonium Isotopes						Atom Ratios Relative to U^{238} (Decay corrected to time of fuel discharge)									
		Fission (atom percent)	Fissions/cc $\times 10^{-20}$	$\frac{MWd}{tU}$	Relative Concentrations (atom percent)								$\frac{Pu^{239}}{U^{238}}$ ($\times 10^3$)	$\frac{Pu^{238}}{U^{238}}$ ($\times 10^4$)	$\frac{Am^{241}}{U^{238}}$ ($\times 10^5$)	$\frac{Cm^{242}}{U^{238}}$ ($\times 10^5$)	$\frac{Cm^{244}}{U^{238}}$ ($\times 10^5$)	$\frac{Nd^{148}}{U^{238}}$ ($\times 10^3$)	$\frac{Cs^{137}}{U^{238}}$ ($\times 10^3$)	$\frac{Cs^{134}}{U^{238}}$ ($\times 10^3$)	$\frac{Ru^{106}}{U^{238}}$ ($\times 10^3$)	$\frac{Ce^{144}}{U^{238}}$ ($\times 10^3$)
					U^{234}	U^{235}	U^{236}	U^{238}	Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}										
93	0.000-0.005 (0.000-0.002)	4.25	10.2	37,400	0.011	0.768	0.363	98.858	53.47	25.92	13.72	6.89	8.26	1.68	7.6	2.92	3.34	0.763	3.1	0.24	0.28	0.54
94	0.005-0.010 (0.002-0.004)	3.79	9.1	33,300	0.013	0.599	0.395	98.993	54.14	26.06	13.47	6.33	6.74	1.26	6.7	2.47	2.28	0.691	2.7	0.19	0.33	0.45
95	0.010-0.018 (0.004-0.007)	3.25	7.8	28,600	0.013	0.596	0.391	99.000	54.32	26.52	12.95	6.21	4.74	1.01	4.6	1.46	1.44	0.593	2.1	0.15	0.21	0.37
96	0.018-0.025 (0.007-0.010)	2.96	7.1	25,900	0.013	0.594	0.385	99.008	54.03	27.14	12.78	6.50	4.16	1.08	3.5	1.25	1.26	0.543	2.1	0.15	0.18	0.34
97	0.025-0.038 (0.010-0.015)	2.87	6.9	25,100	0.013	0.595	0.374	99.018	54.23	27.31	12.55	5.90	3.77	1.01	3.3	1.15	1.14	0.511	2.1	0.15	0.26	0.35
98	0.038-0.051 (0.015-0.020)	2.74	6.5	23,900	0.013	0.614	0.381	98.993	54.16	27.80	12.28	5.78	3.21	0.95	---	1.04	0.73	0.487	1.7	0.14	0.14	0.28
99	0.051-0.076 (0.020-0.030)	2.86	6.8	25,000	0.013	0.618	0.380	98.989	54.69	28.24	11.61	5.45	3.24	0.96	1.6	1.12	0.88	0.513	1.9	0.14	0.13	0.30
100	0.076-0.127 (0.030-0.050)	2.85	6.8	24,900	0.013	0.629	0.374	98.984	54.07	28.68	11.49	5.75	3.72	1.01	1.8	0.92	0.81	0.464	2.0	0.14	0.15	0.32
101	0.127-0.254 (0.050-0.100)	2.79	6.7	24,400	0.013	0.642	0.372	98.974	54.43	29.27	11.12	5.16	3.37	0.88	2.8	0.89	0.84	0.493	2.0	0.14	0.28	0.33
102	0.254-center* (0.100-center)	2.59	6.2	22,600	0.013	0.657	0.376	98.955	54.29	29.47	11.20	5.05	3.19	0.79	2.0	0.77	0.68	0.458	1.5	0.11	0.12	0.25

*Nominal pellet radius = 0.478 cm (0.188 inch)

TABLE 6

Pu²³⁶ CONTENTS OF FULL CROSS SECTION FUEL SAMPLES

Rod Number	Sample Number	Burnup			Atom Percent U ²³⁸	Atom Ratio (x 10 ¹⁰)	Pu ²³⁶ * U ²³⁸
		Fission (Atom Percent)	Fissions/cc (x 10 ⁻²⁰)	MWd tU			
A13	67	3.402	8.1	29,700	94.66	4.9	
A41	68	3.266	7.8	28,500	94.84	5.5	
A46	69	2.894	6.9	25,300	95.52	3.8	
B70	70	2.889	6.9	25,300	95.45	2.5	

* Decay corrected to time of fuel discharge.

NOTE: Nominal initial uranium enrichments were 3.5 weight percent for rods A13 and A41 and 2.76 weight percent for rods A46 and B70.

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