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**ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM
QUARTERLY PROGRESS REPORT NO. 8
AUGUST - OCTOBER 1969**

COMPILED BY
F. H. MEGERTH

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

P4726

REACTOR FUELS AND REPROCESSING DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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AEC Research and
Development Program
November 1969

ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM

Quarterly Progress Report No. 8
August-October 1969

Compiled By
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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 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 ten rods is in progress.

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

1. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 1, November 1967 - January 1968, " March 1968, (GEAP-5598).
2. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 2, February 1968 - April 1968, " May 1968, (GEAP-5624).
3. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress

Report No. 3, May 1968 - July 1968, " August 1968, (GEAP-5667). 4. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 4, August 1968 - October 1968, " November 1968, (GEAP-5702).	5. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 5, November 1968 - January 1969, " February 1969, (GEAP-5746). 6. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report Nos. 6 and 7, February - July 1969, " August 1969, (GEAP-10079).

2. SUMMARY

The irradiation of Assembly SA-1 was terminated in September 1969, after 1800 days of VBWR and Dresden operation. The final bundle average exposure of the SA-1 fuel rods is approximately 8×10^{20} fissions/cm³ (30,000 MWd/t U). Post-irradiation leak testing of the assembly indicated that it is sound.

A number of high-temperature tensile tests were run on longitudinal coupon cladding specimens in which the specimens were held at the test temperatures for periods longer than the normal 15-minute soaking time before tension was applied. Holding specimens at 343°C (650°F) for up to 16-1/2 hours had no significant effect on any of the 343°C tensile properties. However, a specimen held for 16-1/2

hours at 399°C (750°F) showed ultimate and yield strengths approximately 950 kg/cm² (13.5 ksi) and 1200 kg/cm² (17 ksi) lower than any specimen tested at 399°C by the normal procedure, and, in addition, it showed somewhat greater uniform and total elongations. The 454°C (850°F) tensile properties of the irradiated cladding apparently were not affected significantly by increasing the soaking time to 1 hour at that temperature. The ultimate strengths of three irradiated specimens held at 454°C for 16-1/2 hours before testing approached that of the unirradiated material. These same specimens showed wide variations in elongation.

The atom ratios of Am-243 to U-238 ranged from 0.41×10^{-4} to 0.56×10^{-4} in full-cross-section fuel samples with

exposures from 6.9×10^{20} to 7.8×10^{20} fissions/cm³ (25,300 to 28,500 MWd/t U). The ²⁴³Am/²³⁸U ratios at pellet sur-

faces were determined to be approximately three times the ²⁴³Am/²³⁸U ratios at corresponding pellet centers.

3. ASSEMBLY OPERATION

The Dresden reactor operated throughout August and during the first five days of September, at average and maximum thermal powers of approximately 515 MWt and 540 MWt, respectively. The reactor was shut down on September 5, 1969 for scheduled refueling and maintenance operations.

The irradiation of Assembly SA-1 was concluded as planned with this reactor shutdown. Leak testing of SA-1 by the wet sipping technique indicated that the assembly is sound. SA-1 continued in operation in Dresden approximately 1500 days (hours of turbine-generator operation ÷ 24), and

the previous VBWR operation of the fuel rods averaged about 360 days (hours reactor thermal power greater than 5 MWt ÷ 24). Estimated thermal performance and exposure data on Assembly SA-1 are given in Table 1. Chemical burnup analyses, which will be carried out during the final phase of the evaluation program, may indicate that revisions to these data are necessary.

Chemical data on reactor water, presented in Table 2, were extracted from the reports of the Commonwealth Edison Company on Dresden operation for the months of July and August 1969.

4. 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 Testing, G. P. Ferguson

4.1.1.1 Tensile Tests on Longitudinal, Miniature Coupon Specimens

The tensile testing of longitudinally oriented coupon specimens obtained from

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

August and September 1969

<u>Reactor Power (MWt)</u>	<u>515 (Average)</u>		<u>540 (Maximum)</u>	
<u>SA-1 Thermal Performance</u>	<u>Average</u>	<u>Peak</u>	<u>Average</u>	<u>Peak</u>
Heat Flux, W/cm ² (Btu/h-ft ²)	20(62,000)	36(110,000)	21(66,000)	37(117,000)
Specific Power, kW/kg U	10.4	18.3	10.9	19.4
Power Density, kW/l	18.2	32.1	19.3	34.0
	<u>July 31, 1969</u>		<u>September 5, 1969</u>	
Assembly Average Exposure fissions/cm ³ x 10 ⁻²⁰ (MWd/t U)	8.8 (32,000)		8.9 (32,200)	

TABLE 2
DRESDEN REACTOR WATER CHEMISTRY

	<u>July 1969</u>			<u>August 1969</u>		
	<u>Min.</u>	<u>Avg.</u>	<u>Max.</u>	<u>Min.</u>	<u>Avg.</u>	<u>Max.</u>
pH	5.00	6.00	7.00	6.1	6.7	7.4
Conductivity (μ mhos)	0.29	0.38	0.80	0.29	0.39	0.56
Turbidity (APHA)	2.00	5.10	19.40	2.0	3.9	8.6
Chloride (ppm)	0.016	0.034	0.069	0.01	0.041	0.158

the SA-1 cladding was initiated during the preceding quarter.⁽¹⁾ A few additional high temperature tests have been run in which the specimens were held at the test temperatures for periods longer than the normal 15-minute soaking time before tension was applied.

Recent test results are given in Table 3, along with some earlier data for comparison. Holding specimens at 343°C (650°F) for up to 16-1/2 hours had no significant effect on any of the 343°C tensile properties. The approximate mean operating temperature of the cladding at the end of irradiation was 343°C. The ultimate and yield strengths of one specimen held at 399°C (750°F) for 7 hours before testing at that temperature were, respectively, about 800 kg/cm² (11 ksi) and 850 kg/cm² (12 ksi) lower than the minimum values obtained on specimens tested at 399°C after the normal soaking period. The elongations did not change. A specimen held for 16-1/2 hours at 399°C showed ultimate and yield strengths approximately 950 kg/cm² (13.5 ksi) and 1200 kg/cm² (17 ksi) lower than any specimen tested by the normal procedure, and, in addition, it showed somewhat greater uniform and total elongations (1.2 percent and 3.6 percent). The 454°C tensile properties of the

irradiated cladding apparently were not affected significantly by increasing the soaking time at that temperature to 1 hour. The ultimate strengths of three irradiated specimens held at 454°C for 16-1/2 hours before testing approached that of the unirradiated material. One of the three showed increased uniform elongation and ultimate-yield strength differential, while another showed a total elongation at least as high as those of the unirradiated specimens and was of relatively high strength. There are no certain reasons for these differences in behavior.

4.1.1.2 Ring Tests

A cladding ring test of the type carried out under the Fast Ceramic Reactor Development Program [Contract AT(04-3)-189, Project Agreement 10] is being investigated for possible application to the SA-1 cladding.⁽²⁾ The specimen in this test is a transverse cladding section, generally somewhat less in length than the ID of the cladding. This ring of cladding is placed over a metal cylinder that is somewhat longer than the ring and of a diameter slightly smaller than the ID of the cladding. The diameter of the ring is increased to the point of fracture by compressing the ends of the cylindrical mandrel with an Instron testing machine.

TABLE 3

TENSILE TESTS ON LONGITUDINAL COUPON CLADDING SPECIMENS
HELD AT TEST TEMPERATURES FOR VARIOUS TIME PERIODS

Specimen Identification	Tubing Lot Number	Estimated Fast Fluence Exposure [nvt (>1MeV)x10 ⁻²¹]	Test Temperature [°C (°F)]	Holding Time at Test Temperature (h)	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
*	CA-2, -3, -4	0	343(650)	1/4	1530(1450-1650)	21.8(20.6-23.5)	2510(2410-2710)	35.7(34.3-38.5)	13.2(11.0-15.2)	22.3(19.6-26.8)
*	CA-2, -4	3.6(3.2-3.8)		1/4	4700(4060-5320)	66.9(57.8-75.7)	4860(4640-5360)	69.0(66.0-76.2)	0.4(0.2-0.7)	1.2(1.0-1.4)
A46-L2**	CA-2	3.2		1	4630	65.9	4750	67.6	0.3	1.2
A46-O2**	CA-2	3.1		16-1/2	4810	68.4	4810	68.4	0.2	1.7
A46-P2	CA-2	3.1		16-1/2	4630	65.9	4680	66.6	0.3	1.4
*	CA-2, -3	0	399(750)	1/4	1700(1420-2170)	24.1(20.2-30.8)	2320(2070-2710)	33.0(29.5-38.5)	12.6(7.7-17.0)	19.6(18.3-21.2)
*	CA-2	3.3(3.2-3.4)		1/4	4290(4190-4360)	61.1(59.6-62.0)	4320(4210-4390)	61.3(59.9-62.4)	0.3(0.2-0.4)	1.9(1.9-2.0)
B97-G1	CA-2	3.6		7	3340	47.5	3420	48.7	0.4	2.0
B97-G2	CA-2	3.6		16-1/2	3000	42.7	3260	46.3	1.2	3.6
*	CA-2, -3	0	454(850)	1/4	1520(1300-2060)	21.6(18.5-29.3)	2170(1980-2430)	30.8(28.1-34.6)	9.6(5.8-12.4)	17.3(14.2-19.9)
*	CA-2	3.2		1/4	2800(2490-3220)	39.8(35.4-45.8)	2860(2560-3290)	40.7(36.4-46.8)	0.5(0.5-0.6)	9.1(5.4-12.1)
A46-O1**	CA-2	3.1		1	2720	38.7	2860	40.7	0.6	5.0
A46-R1	CA-2	3.0		16-1/2	2280	32.4	2350	33.4	0.7	8.5
A46-R2	CA-2	3.0		16-1/2	1840	26.1	2040	29.0	2.7	9.5
B97-F2	CA-2	3.6		16-1/2	2610	37.1	2660	37.8	0.6	20.0

*Values shown are averages and ranges obtained on from three to six specimens on which data were reported previously⁽¹⁾.

**First reported in Reference 1.

The diameter or circumference of the ring at fracture is measured, and the value $\Delta D/D_0$ is calculated.

Preliminary trial runs have been made to determine the compression characteristics of Zircaloy-2 mandrels between room temperature and 454° C. It appears that rings of irradiated SA-1 cladding can be tested at temperatures between 343° C and 454° C, and probably at room temperature as well. Unirradiated control specimens probably cannot be expanded to the point of fracture with these mandrels.

4.1.2 Electron Microscopic Examination of UO_2 and Cladding,
H. S. Rosenbaum, Technical Director
U. E. Wolff

An unirradiated UO_2 specimen, thinned to electron transparency by mechanical grinding and polishing followed by ion bombardment, was examined with the high voltage electron microscope at the University of California, Berkeley. Extensive areas that were not transparent to 100 keV electrons were transparent to 650 keV electrons. A 650 keV electron micrograph is shown in Figure 1. Grain boundaries and sintering pores are indicated. Some of the fine detail observed was found to be surface defects, probably caused by the ion bombardment. A method to remove this disturbed surface layer should be developed. However,

the presence of these surface defects does not preclude detection and interpretation of internal features.

Development of remote handling procedures for preparing thin, irradiated UO_2 foils was initiated in the hot cells of the Radioactive Materials Laboratory.

4.2 FUEL BURNUP AND ISOTOPIC COMPOSITION ANALYSIS,
C. P. Ruiz, Technical Director

With the exception of Np-237 the isotopic analyses on the fuel removed from Assembly SA-1 in January 1967 have now been completed. During the present quarter the Am-243 contents were measured on full-cross-section fuel samples from three fuel rods and on radial samples from two rods. The results of these analyses are given in Tables 4 and 5.

Table 4 of GEAP-5746 gave ⁽³⁾ the analysis of a radial sample from high-exposure fuel rod A13 for burnup, uranium and plutonium isotopic compositions, and the atom ratios relative to U-238 of Pu-239 and Pu-238, Am-241, Cm-242 and Cm-244, Nd-148, Cs-137 and Cs-134, Ru-106, and Ce-144. Some of the data presented in the original table have been revised recently. The complete table, with revisions, is given in the appendix.

Examination and analyses of the SA-1 fuel rods whose operation was terminated in September 1969 are at present scheduled to begin next January.

Figure 1. Transmission electron micrograph of a specimen from an unirradiated sintered UO_2 pellet

TABLE 4Am-243 CONTENTS OF FULL CROSS SECTION FUEL SAMPLES

Rod Number	Sample Number	Burnup			Atom Percent U-238	Atom Ratio ($\times 10^4$) $^{243}\text{Am}/^{238}\text{U}$
		Fission (atom percent)	Fissions/cm ³ ($\times 10^{-20}$)	MWd/t U		
A41	68	3.266	7.8	28,500	94.84	0.56
A46	69	2.894	6.9	25,300	95.52	0.48
B70	70	2.889	6.9	25,300	95.45	0.41

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.

TABLE 5Am-243 CONTENTS OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS

FUEL ROD A13						
Sample Number	Distance From Pellet Periphery [cm(inch)]	Burnup			Atom Percent U-238	Atom Ratio ($\times 10^4$) $^{243}\text{Am}/^{238}\text{U}$
		Fission (atom percent)	Fissions/cm ³ ($\times 10^{-20}$)	MWd/t U		
82	0.000-0.005 (0.000-0.002)	4.84	11.6	42,500	92.715	-
83	0.005-0.010 (0.002-0.004)	4.39	10.6	38,600	93.333	2.0
84	0.010-0.018 (0.004-0.007)	3.85	9.2	33,700	94.125	1.1
85	0.018-0.025 (0.007-0.010)	3.83	9.1	33,400	94.206	1.0
86	0.025-0.038 (0.010-0.015)	3.61	8.6	31,600	94.484	0.5
87	0.038-0.051 (0.015-0.020)	3.49	8.3	30,400	94.560	0.4
88	0.051-0.076 (0.020-0.030)	3.27	7.8	28,500	94.949	0.5
89	0.076-0.127 (0.030-0.050)	3.17	7.6	27,700	94.925	0.5
90	0.127-0.254 (0.050-0.100)	3.42	8.2	29,900	94.741	0.5
91	0.254-center* (0.100-center)	3.30	7.9	28,800	94.820	0.6

*Nominal pellet radius = 0.478 cm (0.188 inch)

TABLE 5 (continued)

Am-243 CONTENTS OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS

FUEL ROD B70						
Sample Number	Distance From Pellet Periphery [cm(inch)]	Burnup			Atom Percent U-238	Atom Ratio (x 10 ⁴) 243Am/238U
		Fission (atom percent)	Fissions/cm ³ (x 10 ⁻²⁰)	MWd/t U		
93	0.000-0.005 (0.000-0.002)	4.25	10.2	37,400	93.234	0.6
94	0.005-0.010 (0.002-0.004)	3.79	9.1	33,300	94.082	0.5
95	0.010-0.018 (0.004-0.007)	3.25	7.8	28,600	94.962	0.3
96	0.018-0.025 (0.007-0.010)	2.96	7.1	25,900	95.348	0.5
97	0.025-0.038 (0.010-0.015)	2.87	6.9	25,100	95.519	0.3
98	0.038-0.051 (0.015-0.020)	2.74	6.5	23,900	95.718	0.2
99	0.051-0.076 (0.020-0.030)	2.86	6.8	25,000	95.598	0.2
100	0.076-0.127 (0.030-0.050)	2.85	6.8	24,900	95.513	0.2
101	0.127-0.254 (0.050-0.100)	2.79	6.7	24,400	95.625	0.2
102	0.254-center* (0.100-center)	2.59	6.2	22,600	95.834	0.2

*Nominal pellet radius = 0.478 cm (0.188 inch)

REFERENCES

1. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program Quarterly Progress Report Nos. 6 and 7, February - July 1969," August 1969, (GEAP-10079).
Program, Twenty-Sixth Quarterly Report, February - April 1968, June 1968, (GEAP-5631).
2. "Sodium Cooled Reactors, Fast Ceramic Reactor Development
Evaluation Program Quarterly Progress Report No. 5, November 1968 - January 1969," February 1969, (GEAP-5746).
3. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program Quarterly Progress Report No. 5, November 1968 - January 1969," February 1969, (GEAP-5746).

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APPENDIX

TABLE 4 OF GEAP-5746 (REVISED)

ANALYSES OF FUEL SAMPLES FROM DIFFERENT RADIAL LOCATIONS

FUEL ROD A13

Burnup					Composition (atom percent)												Atom Ratios Relative to ²³⁸ U (Corrected to time of fuel discharge)									
																	²³⁹ Pu	²³⁸ Pu	²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	¹⁴⁸ Nd	¹³⁷ Cs	¹³⁴ Cs	¹⁰⁶ Ru
Sample Number	Distance From Pellet Periphery [cm (inch)]	Fission (atom percent)	Fissions/cm ³ (x10 ⁻²⁰)	MWd/t U	Uranium				Plutonium				²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U	²³⁸ U			
					²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	(x10 ³)	(x10 ⁴)	(x10 ⁴)	(x10 ⁴)	(x10 ⁵)	(x10 ⁵)	(x10 ³)	(x10 ³)	(x10 ³)	(x10 ³)	(x10 ³)	(x10 ³)	(x10 ³)	
82	0.000-0.005 (0.000-0.002)	4.84	11.6	42,500	0.016	0.776*	0.483*	98.725*	53.43	26.07	13.50	6.99	8.01	1.59	0.88	-	3.00	3.33	0.850	3.1	0.23	0.60*	0.46*			
83	0.005-0.010 (0.002-0.004)	4.39	10.6	38,600	0.016	0.757	0.481*	98.75*	53.76	26.35	13.23	6.66	6.41	1.39	-	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)	3.85	9.2	33,700	0.014*	0.755*	0.490*	98.74*	54.05	26.68	12.89	6.39	4.90	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)	3.83	9.1	33,400	0.016	0.887	0.431	98.6	53.93	27.42	12.44	6.21	3.95	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)	3.61	8.6	31,600	0.016	0.802	0.452	98.73	53.38	27.38	13.28	5.96	3.91	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)	3.49	8.3	30,400	0.017	0.921	0.436	99.06	54.21	28.23	11.86	5.70	3.66	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)	3.27	7.8	28,500	0.016	0.790	0.448	98.75	54.28	28.87	11.43	5.43	3.29	0.92	0.40	0.5*	1.10	1.01	0.582*	1.9*	0.13*	0.10*	0.27*			
89	0.076-0.127 (0.030-0.050)	3.17	7.6	27,700	0.018	0.958	0.430	99.02	53.96	28.89	11.56	5.58	3.17	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)	3.42	8.2	29,900	0.016	0.842	0.439	98.70	54.04	29.26	11.43	5.26	3.39	1.0	0.66	0.5*	1.3	1.1	0.593	1.5*	0.13*	0.15*	0.28*			
91	0.254-center ** (0.100-center)	3.30	7.9	28,800	0.016	0.883	0.428	98.67	54.38	29.56	10.99	5.08	3.47	1.14	0.58	0.6*	0.98	1.00	0.566	1.8*	0.11*	0.17*	0.35*			

*Corrected or additional values
**Nominal pellet radius = 0.478 cm (0.188 inch)

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