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**SPECIFIC ZIRCONIUM ALLOY
DESIGN PROGRAM
QUARTERLY PROGRESS REPORT
Nos. 18 AND 19**

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U.S. ATOMIC ENERGY COMMISSION
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GENERAL ELECTRIC
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and Development Report
January 1968

SPECIFIC ZIRCONIUM ALLOY DESIGN PROGRAM QUARTERLY PROGRESS REPORT NOS. 18 AND 19

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ABSTRACT

A Zr + 2.0 at. % Cr + 0.16 at. % Fe alloy (Zr-Cr) is being evaluated as nuclear fuel cladding. Zr-Cr alloy-clad fuel rods are being irradiated in the Consumers Big Rock Point reactor to a target exposure of 15,000 to 20,000 MWd/T. The rate of fuel burnup has been approximately 800 MWd/TU (average) per month of full-power operation. The Zr-Cr alloy-clad fuel rods have accumulated a total of approximately 6000 MWd/TU average exposure as of January 31, 1968. This corresponds to about 5100 hours of high-temperature, full-power, reactor operation.

The long-term, out-of-pile, loop corrosion tests are continuing satisfactorily. Zr-Cr alloy samples are being tested in the loop under heat transfer conditions, and have accumulated about 3800 hours exposure as of January 17, 1968.

Mechanical test results, burst and tensile tests, indicate that the fine-grain Zr-Cr tube material is stronger and less ductile than the usual Zircaloy-2 tubing; however, the high-temperature stress rupture performance of the fine-grain size Zr-Cr is poorer than that of the Zircaloy-2 materials.

SECTION I

INTRODUCTION

This is the combined eighteenth and nineteenth Quarterly Progress Report on Project Agreement 24 of USAEC Contract AT(04-3)-189, covering the period from July 1967 through December 1967.

A summary report⁽¹⁾ was issued covering the Specific Zirconium Alloy Design Program from January 1962 to May 1964. The project was taken up again in November 1964 for the evaluation of a Zr + 2.0 at. % Cr + 0.16 at. % Fe alloy (Zr-Cr) as cladding for water reactor fuel.

The continuation program provides for ex-reactor corrosion tests, fuel rod and tensile coupon irradiations of the alloy, and detailed post-irradiation examinations of the fuel clad-

ding. Zr-Cr alloy-clad fuel rods were successfully fabricated by the contractor for irradiation in the Consumers Big Rock Point reactor.

Before committing the fuel irradiation, the program provided for a detailed evaluation of the cladding to determine if the potential of the alloy composition was actually realized after fabrication of the fuel cladding.

A topical report⁽²⁾ was prepared, entitled "The Fabrication and Characterization of Zr-Cr Alloy Tubing," covering the technical aspects of obtaining satisfactory Zr-Cr tubing to be used as fuel rod cladding in the irradiation proof test. The irradiation tests are underway in a boiling water power reactor. Concurrent out-of-pile tests on the Zr-Cr tubing materials used as fuel rod cladding are also in progress.

SECTION II

ALLOY DESIGN CONTINUATION

2.1 Zr-Cr TUBING FABRICATION

A topical report, "The Fabrication and Characterization of Zr-Cr Alloy Tubing," GEAP-5589, by H. J. Busboom, G. W. Cleveland, H. H. Klepfer, and H. L. Shannon, was prepared; it describes the fabrication of significant quantities of tubing for fuel cladding and bar stock for end plugs from a large ingot of Zr-Cr alloy. Characterization tests on these wrought products confirmed the excellent

high-purity water corrosion resistance for this composition at temperatures above 300°C.⁽³⁾

Also described in this report is the irradiation test program where the performance of Zr-Cr alloy-clad fuel rods is being compared directly with similar Zircaloy-2 clad fuel rods in a boiling water power reactor environment. Included in this experiment are nonfueled dummy rods to separate the effects of heat transfer during irradiation on the corrosion of the Zr-Cr alloy.

SECTION III

FUEL CLADDING EVALUATION

Evaluation of the mechanical and corrosion properties of the Zr-Cr alloy tubing being used as fuel rod cladding in the irradiation proof tests continued. The tests are being performed to verify that the full potential of the alloy previously established by laboratory tests⁽¹⁾ can be realized on tubing produced by conventional processes.⁽²⁾

An out-of-pile, boiling water, heat transfer corrosion test is being performed as a control for the irradiation test program. This test will be run for the same duration as the initial phase of the irradiation test (the first group of fuel rods will be removed after irradiation to 5000 to 7500 MWd/TU or after 4500 to 6500 hours).

3.1 CORROSION TESTS

Corrosion testing of the Zr-Cr tubing and bar stock has been performed at 300, 400, and 500°C in three different corrosion facilities. Weight gains for exposures in 300, 400, and 500°C low flow steam autoclaves (20 to 23 ppm O_2 + 1 to 3 ppm H_2 added) to 2000 hours are shown in Table I. These data confirm the superior corrosion resistance of all Zr-Cr wrought products as compared to Zircaloy-2 at temperatures above 300°C. Although the corrosion rates for the wrought products are slightly higher than the best laboratory melts, the corrosion rates do indicate that the high-temperature corrosion potential of the alloy was realized in all wrought products.

Data to longer times, 4146 hours, in the isothermal section of the C1-1 loop (280°C saturated water containing 0.2 ppm O_2) were also obtained for the fuel tubing (see Table II). The weight gains for the Zr-Cr fuel tubing are comparable with the Zircaloy-2 control for this environment.

Heat transfer corrosion tests are also in progress. These tests are being performed in the C1-1 loop which simulates actual boiling water reactor fuel rod conditions without neutron flux. This loop operates with 285°C saturated steam (steam quality 4%) similar to boiling water reactor environments (0.2 ppm O_2). Data after exposures of 2856 hours were obtained (Table II) and indicate normal weight gains for heat transfer specimens operated at 500,000 Btu/h-ft². Isothermal specimens were also included in the same environment. Similar weight gains were experienced for both Zr-Cr and Zircaloy-2. These tests will be continued for 4500 to 6500 hours for comparison with reactor fuel cladding.

3.2 MECHANICAL PROPERTIES

Mechanical properties of fine-grain Zr-Cr tubing were determined by tensile tests on tubing and by burst and stress rupture pressure tests on tubing specimens at various temperatures. The properties included: longitudinal tensile properties, burst hoop stress at different strain rates, stress rupture performance, circumferential fracture elongations, and Larson-Miller parameter creep constant.

A series of mechanical tests was performed on tubes of a fine-grain Zr-Cr material to determine mechanical properties of interest for reactor applications. Zr-Cr is a zirconium alloy designed for superior corrosion resistance at higher temperatures as compared to conventional Zircaloy-2 type alloys. The grain size of the material tested was ASTM 10 or less [< 0.0127 mm (< 0.5 mil) mean diameter] as compared with ASTM 7 or less [< 0.0304 mm (< 1.2 mil) mean diameter] for normal Zircaloy-2 alloy tubing.⁽²⁾

TABLE I

Zr-Cr TUBING AND BAR CORROSION WEIGHT GAINS
Low Flow Steam Autoclaves⁽¹⁾

Specimen Description and Initial Conditions (2)	Average Weight Gain (mg/dm ²)										
500°C Test	24 h	48 h	72 h	144 h	421 h	569 h	966 h	990 h	1579 h	1976 h	2000 h
Zr-Cr Fuel Tube, 1	32.7	---	50.5	----	97.4	----	-----	130	-----	-----	205
Zr-Cr Heat Transfer Tube, 1	32.4	---	49.6	----	93.4	----	-----	125	-----	-----	246
Zr-Cr Bar Stock, 1	---	---	----	----	-----	94.5	-----	-----	167	-----	-----
Zr-Cr Fuel Tube, 2	---	47.0	---	67.8	-----	-----	129	-----	-----	239	-----
Zr-Cr Fuel Tube, 3	21.6	---	38.3	----	81.0	----	-----	116	-----	-----	222
Zircaloy-2 Control, 1	74.6	---	105	----	234	----	-----	646	-----	-----	Disintegrated
400°C Test	72 h	96 h	168 h	341 h	413 h	569 h	910 h	982 h	1587 h	1928 h	2000 h
Zr-Cr Fuel Tube, 1	13.2	---	19.9	----	28.7	----	-----	41.5	-----	-----	55.5
Zr-Cr Heat Transfer Tube, 1	15.0	---	21.9	----	29.4	----	-----	44.2	-----	-----	57.7
Zr-Cr Bar Stock, 1	---	---	----	----	-----	27.8	-----	-----	43.3	-----	-----
Zr-Cr Fuel Tube, 2	---	21.1	----	33.6	-----	-----	48.9	-----	-----	60.9	-----
Zr-Cr Fuel Tube, 3	7.5	---	14.9	----	23.0	----	-----	35.3	-----	-----	49.4
Zircaloy-2 Control, 1	19.4	32.0	25.2	42.2	33.2	54.8	67.1	53.8	85.5	95.7	84.3
300°C Test	72 h	96 h	168 h	349 h	421 h	918 h	990 h	1928 h	2000 h		
Zr-Cr Fuel Tubes, 1	5.5	---	5.9	----	9.0	----	15.9	-----	30.4	-----	-----
Zr-Cr Heat Transfer Tube, 1	5.1	---	5.4	----	8.1	----	15.6	-----	31.8	-----	-----
Zr-Cr Fuel Tube, 2	---	2.2	----	6.7	-----	16.0	-----	35.2	-----	-----	-----
Zr-Cr Fuel Tube, 3	1.1	---	1.3	----	2.8	----	8.2	-----	20.8	-----	-----
Zircaloy-2 Control, 1	9.5	9.5	11.7	12.9	14.2	15.7	16.6	19.1	19.7	-----	-----

(1) 20-23 ppm O₂, 1-3 ppm H₂ added.

(2) 1 - Etched and NaOH cleaned.

2 - Etched, NaOH cleaned, and pre-oxidized at 300°C for 60 hours (2.6 mg/dm²).

3 - Etched, NaOH cleaned, and pre-oxidized at 400°C for 12 hours (5.9 mg/dm²).

TABLE II
CL-1 LOOP CORROSION TEST RESULTS

Specimen Description and Initial Condition ⁽¹⁾	Loop Environment	Average Weight Gain (mg dm ⁻²)						
		271 h	600 h	871 h	1597 h	1868 h	2585 h	2856 h
Zr-Cr Heat Transfer Tube Under Heat Transfer, 1	280 °C (533 °F) to 285 °C (546 °F) at 514,500 ⁽²⁾ Btu/h-ft ²	--	12.2		20.4		--	
Zr-Cr Heat Transfer Tube Under Heat Transfer, 2	280 °C (536 °F) to 285 °C (546 °F) at 518,900 ⁽²⁾ Btu/h-ft ²	5.0		9.4		15.9		30.4
Zr-Cr Heat Transfer Tube, 1	285 °C (546 °F) Isothermal ⁽²⁾ Steam-Water	8.7	--	12.4		15.4	30.5	
Zircaloy-2 Control, 1	285 °C (546 °F) Isothermal ⁽²⁾ Steam-Water	8.4	--	12.3		15.3		17.1
		2549 h ⁽⁴⁾		3149 h		4146 h		5134 h
Zircaloy-2 Control, 1	285 °C (546 °F) Saturated ⁽³⁾ Water	16.8		16.6		18.4		19.1
		629 h		1229 h		2226 h		3214 h
Zr-Cr Fuel Tubing, 1	285 °C (546 °F) Saturated ⁽³⁾	11.9		14.4		17.2		19.4

(1) 1 - Etched and NaOH cleaned.

2 - Etched, NaOH cleaned, and pre-oxidized at 300 °C for 60 hours (2.6 mg/dm²).

3 - Etched, NaOH cleaned, and pre-oxidized at 400 °C for 12 hours (5.9 mg/dm²).

(2) Heat transfer section of Cl-1 loop has 20 ppm O₂ and 2 ppm H₂ in steam phase.

(3) Isothermal section of Cl-1 loop with ~ 0.2 ppm O₂ and ~ 0.025 ppm H₂.

(4) 282 °C (539 °F) subcooled water 1.5 ppm O₂ in Cl-4 loop, transferred to Cl-1 loop.

3.2.1 Longitudinal Tensile Tests

Tensile tests were performed on Zr-Cr fuel tubing and bar stock at ambient temperature and at various temperatures to 600°C (1112°F). These tests (Table III) indicate longitudinal tensile properties similar to stress-relieved Zircaloy-2. Adequate ductility was also demonstrated by total tensile elongations of 20% or greater. Properties for stress-relieved Zircaloy-2 tubing are also shown in Table III for comparison.

3.2.2 Burst Tests

The mechanical tests essentially consisted of the pressurization of the tubular test specimens at various test temperatures. Burst pressure tests were conducted by increasing the internal pressures at prescribed rates until rupture occurred. The rates were chosen so that the tube test results could be compared with results of longitudinal specimen tests conducted elsewhere. The stress rupture tests were conducted at constant pressures until rupture occurred.

There were three mechanical property values to be established by these tests: one was the burst hoop stress, the second was the stress rupture performance, and the third was the circumferential fracture elongation of the failed specimens.

The fine-grain Zr-Cr tubing was fabricated into tubular test specimens approximately 12.7 cm (5 in.) long. The tubing size was approximately 1.14 cm (0.450-in.) o.d. and 0.081 cm (0.032 in.) wall thickness.

The tubes were placed in safety containers within furnaces that were maintained at the specified test temperatures, which ranged from 25°C (77°F) to 600°C (1112°F). The temperatures were monitored continuously by use of thermocouples within the safety containers adjacent to the specimens. Argon pressure

was applied internally to the specimens either at an increasing rate for burst tests, or at a constant value for the stress rupture tests. The pressure source was a Haskel gas booster in conjunction with a gas receiver to supply pressure without surges. (For some very high rate burst tests at room temperature, hydraulic pressure was utilized.) The specimen test pressure was indicated by pressure gages and monitored by a pressure cell-recorder combination.

The circumferential fracture elongation was determined by pre- and post-test measurements—the former by diameter measurements, the latter by arc length measurements using a narrow thin foil strip of steel.

The results of the burst tests are listed in Table IV.

3.2.3 Loading Rate Effect

To evaluate any potential different burst stress performance which may depend on the rate of application of the load, the pressure loading rates were varied in different tests from 0.035 kg/mm²-min (50 psi/min) to 23 kg/mm²-min (33,000 psi/min). For purposes of data evaluation, these were arbitrarily categorized as "fast rates" and "slow rates," with the pressure rate of 0.35 kg/mm²-min (500 psi/min) being the point of separation. For these specimens, the pressure rate of 0.35 kg/mm²-min (500 psi/min) corresponds to both an elastic hoop stress rate of approximately 2.34 kg/mm²-min (3350 psi/min) and a circumferential elastic strain rate of approximately 0.024 %/min.

As shown in Figure 1, the burst stresses at the slower rates are approximately 10% less than those at the higher rates. This rate effect is typical of many materials.

Early in the testing program, prior to receipt of the longitudinal tensile test data, it

TABLE III
MECHANICAL PROPERTIES OF Zr-Cr TUBING AND BAR

Material	Type of Sample	Test Temp.		0.2% Offset Yield Stress ⁽¹⁾		Ultimate or Burst Stress ⁽²⁾		Elongation		Red. in Area (%)
		(°C)	(°F)	(psi)	(kg/mm ²)	(psi)	(kg/mm ²)	Uniform	Total	
								(%)	(%)	
Bar Stock	Long. Tensile	25	77	67,000	46.8	90,600	63.3	9.0	23.4	49
Fuel Tube	Long. Tensile, Entire Tube	25	77	65,300	45.6	90,500	63.2	---	19.4	---
Fuel Tube	Long. Tensile, Cut Section	25	77	69,500	48.5	90,900	63.5	7.7	14.3	42
Bar Stock	Long. Tensile	100	212	58,000	40.5	79,000	55.1	9.1	24.4	54
Fuel Tube	Long. Tensile, Cut Section	100	212	60,000	41.9	79,500	55.5	10.4	22.1	45
Bar Stock	Long. Tensile	200	392	48,700	34.0	64,800	45.3	7.5	32.5	61
Fuel Tube	Long. Tensile, Cut Section	200	392	50,600	35.4	66,400	46.3	7.3	18.0	45
Bar Stock	Long. Tensile	300	572	41,200	28.8	54,400	38.0	7.5	27.2	68
Fuel Tube	Long. Tensile Cut Section	300	572	45,800	32.0	55,200	38.6	4.3	16.3	53
Fuel Tube	Long. Entire Tube	343	650	33,000	23.1	43,600	30.5	---	22.4	---
Bar Stock	Long. Tensile	400	752	37,000	26.0	44,300	31.0	5.7	37.4	79
Fuel Tube	Long. Tensile, Cut Section	400	752	35,000	24.4	42,800	30.0	4.2	20.4	65
Bar Stock	Long. Tensile	500	932	26,200	18.3	30,900	21.6	4.1	51.7	97
Fuel Tube	Long. Tensile, Cut Section	500	932	26,300	18.8	30,200	21.1	2.3	28.3	91
Bar Stock	Long. Tensile	600	1112	15,200	10.6	18,400	12.9	4.8	52.6	100
Fuel Tube	Long. Tensile Cut Section	600	1112	11,400	7.7	14,300	10.0	5.2	57.7	97
Zircaloy-2	Tensile ⁽³⁾	25	77	>60,000	>42	>70,000	>49	---	>16	---
Zircaloy-2	Tensile	343	650	>32,000	>22.5	----	----	---	----	---

- (1) All Tensile Tests at 0.02 min⁻¹ strain rate.
(2) Hoop Stress based on original tube dimensions.
(3) Values for stress-relieved Zircaloy-2.

TABLE IV
FINE GRAIN Zr-Cr ALLOY BURST TEST RESULTS

Specimen No.	Test Temp.		Pressurization Rate			Burst Pressure		Burst Hoop Stress		Circumferential Fracture Elongation (CFE) (%)	Time to Failure (hours)	Remarks
	(°C)	(°F)	F/S*	psi/min	kg/mm ² -min	psi	kg/mm ²	psi	kg/mm ²			
1	25	77	F	24,500	17.1	19,240	13.40	122,800	86.0	8.1	----	Pressurized to 9050 psi. Repressurized hydraulically to failure. -----
5	25	77	F	33,000	23.0	19,260	13.45	122,800	86.0	9.9	----	
6	25	77	F	1,540	1.08	18,620	13.00	118,600	83.0	12.0	----	
15	25	77	F	3,550	2.48	18,250	12.70	107,200	75.0	9.5	----	
8	200	392	S	222	1.55	12,000	8.40	75,000	52.5	14.4	----	-----
30	200	392	F	1,365	0.955	12,720	8.90	81,500	57.0	6.5	----	
2	300	572	S	63	0.044	9,570	6.70	60,900	42.6	11.2	0.12	Pressurized to 9050 psi. Repressurized to 9570 psi and held to rupture. Pressurized to 9900 psi. Held to rupture. ----- Pressurized to 9620 psi. Held to rupture. -----
7	300	572	S	55	0.038	9,900	6.94	63,200	44.4	6.1	0.23	
9	300	572	S	253	0.177	10,770	7.54	68,300	47.8	8.2	----	
10	300	572	F	652	0.45	9,620	6.75	61,600	43.2	1.8	0.43	
14	300	572	F	3,240	2.26	10,920	7.65	69,200	48.5	7.1	----	
3	400	752	S	48	0.033	7,270	5.10	46,300	32.4	10.3	----	-----
11	400	752	F	584	0.41	8,070	5.65	51,600	36.2	14.1	----	
13	500	932	F	725	0.51	5,650	3.96	37,000	25.9	25.2	----	-----
4	600	1112	S	60	0.042	2,475	1.73	15,800	11.1	45.2	----	-----
12	600	1112	F	560	0.39	3,100	2.17	19,300	13.5	32.2	----	

* F/S = Fast or slow rate of pressurization.
Slow rate < 500 psi/min (< 0.35 kg/mm²-min).
Fast rate > 400 psi/min (> 0.35 kg/mm²-min).

FIGURE 1. Zr-Cr FINE-GRAIN TUBE BURST HOOP STRESS VERSUS TEST TEMPERATURE

was established that the fine-grain Zr-Cr was stronger than originally expected.⁽³⁾ At room temperature, the pressure required to burst the tubes exceeded the capacity of the available pneumatic equipment. To obtain data at an early date, two specimens (Nos. 1 and 5) were loaded hydraulically to burst at pressures of approximately 13.5 kg/mm^2 (19,250 psi). Also, other specimens at 300°C (572°F) (Nos. 2, 7, and 10) were loaded at prescribed rates to the then maximum available pressures of approximately 6.94 kg/mm^2 (9900 psi). These pressures were maintained at the values shown in Table IV until rupture occurred. The times to rupture at those pressures varied from 7 to 26 minutes. When the equipment was later modified to provide increased pressure capability, these tests were repeated and the results shown in Table IV.

3.2.4 Comparison of Longitudinal and Tube Burst Data

Data from longitudinal tensile tests of the Zr-Cr tube material shown in Table III were compared with the results of the burst tests, and are illustrated in Figure 2. The longitudinal tests were conducted at strain rates which varied from 0.002 %/min to 2.0 %/min at most temperatures, and at 0.02 %/min at all temperatures. The range of ultimate stress values reported are indicated in the figure by vertical lines.

It can be seen that the ultimate stress values from the longitudinal tests are consistently lower than the burst stress values from the tube tests. This also is expected when considering the biaxial effect of the stresses. To improve the basis of comparison, the longitudinal data was corrected for the biaxial conditions by utilizing an effective stress and assuming isotropic properties. (This means either multiplying the burst hoop stress by 0.866 and comparing it with the longitudinal tensile stress or, as was done here, dividing the tensile stress by 0.866 and comparing the

result with the burst hoop stress.) The corrected value is almost coincidental with the slow rate burst test data. This would appear to indicate that the fine-grain Zr-Cr is nearly isotropic.

3.2.5 Circumferential Fracture Elongation (CFE)

The circumferential fracture elongation values were established for the specimens, and are illustrated in Figure 3. These data indicate a minimum at approximately 300°C (572°F) and then at higher temperatures, an increase of the CFE with increasing temperature. The CFE data do not indicate significant differences associated with fast and slow strain rates, though several slow rate CFE values are higher than corresponding fast rate values. Circumferential fracture elongations from burst tests indicate lower transverse ductility, 10%, compared to longitudinal ductility. However, 10% circumferential fracture elongation under closed-end burst test conditions is considered quite adequate for the proposed fuel cladding. The results of the stress rupture tests are summarized in Table V.

3.2.6 Stress Rupture Values

The stress rupture data are shown in Figure 4 for tests conducted at temperatures between 400 and 600°C . In general, these data are consistent. As indicated, the test of specimen 19 was terminated at 1000 hours without failure. Also shown in Figure 4 are comparable stress rupture test results reported for cold-worked Zr-2⁽⁴⁾ and annealed Zr-2.⁽⁵⁾

3.2.7 Larson-Miller Parameter

The assessment of the stress rupture performance of the Zr-Cr tubing can be improved if a determination of the standard parameters can be made. The Larson-Miller Parameter (LMP), and particularly the constant of the LMP, are typical parameters that are

FIGURE 2. Zr-Cr FINE-GRAIN BIAXIAL TEST BURST HOOP STRESS AND LONGITUDINAL TEST ULTIMATE STRESS VERSUS TEST TEMPERATURE

FIGURE 3. Zr-Cr FINE-GRAIN CIRCUMFERENTIAL FRACTURE ELONGATION VERSUS TEST TEMPERATURE

TABLE V
FINE-GRAIN Zr-Cr ALLOY STRESS RUPTURE TEST RESULTS

Specimen No.	Test Temp.		Internal Pressure		Initial Hoop Stress		Time to Failure (hours)	Circumferential Fracture Elongation (CFE) (%)	Remarks
	(°C)	(°F)	(psi)	(kg/mm ²)	(psi)	(kg/mm ²)			
16	400	752	4,720	3.30	29,950	21.00	29.7	13.0	----
17	400	752	4,000	2.80	25,300	17.70	91.0	19.0	----
18	400	752	2,980	2.08	19,150	13.40	840.4	26.4	----
19	400	752	2,130	1.49	13,600	9.51	>1000	(2.9)	Not failed after 1000 hours: CFE value shown is diametral expansion.
25	400	752	4,000	2.80	24,900	17.40	71.8	19.3	----
20	500	932	1,500	1.05	9,500	6.65	147.9	41.7	----
21	500	932	1,830	1.28	11,720	8.21	52.2	53.8	----
24	535	995	1,500	1.05	9,500	6.65	34.0	41.6	To check Larson-Miller Parameter (LMP) constant.
22	600	1112	1,500	1.05	9,550	6.68	2.7	45.9	----
23	600	1112	1,000	0.70	6,370	4.46	16.4	52.0	----

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FIGURE 4. Zr-Cr FINE-GRAIN INITIAL HOOP STRESS VERSUS TIME TO RUPTURE

useful for this purpose. The LMP is of the form.

$$\text{LMP} = T (c + \log t_R),$$

where

T = Temperature, °R.

t_R = Time to rupture, hours, and

c = Material constant.

The test conditions were selected to determine the value of c which best fits the data. The data are shown in Figure 5, which contains plots of the applied hoop stress versus the LMP for values of c of 10, 15, 20, and 25. These data fit the $c = 15$ curve best. Specimens 20, 22, and 24, tested at approximately 6.65 kg/mm^2 (9500 psi) hoop stress were chosen to help determine c . The curve with minimum scatter of these points is that associated with the most probable value of c .

3.2.8 Circumferential Fracture Elongation (CFE)

The CFE values of stress rupture test specimens are illustrated in Figure 3. In general, these indicate that over the range of 400°C (752°F) to 600°C (1112°F), the CFE values increase with increasing temperature, and also increase as the time to rupture increases. Also, for the test times investigated the CFE values from rupture tests are consistently larger than those from burst tests. This is as expected, since the least strain rate of those tested is $\sim 3 \times 10^{-5} \text{ h}^{-1}$ (specimen 19) and the reverse of lesser CFE values at longer times would not usually be anticipated until the strain rates decreased to less than 10^{-6} h^{-1} .

Photographs of the specimens illustrating the failure regions are shown in Figures 6 through 17. The latter ten figures are of one of the various test groups.

3.3 CONCLUSIONS

The following conclusions may be drawn from this work.

- a. In general, based on burst and tensile test results, the fine-grain Zr-Cr tube material is stronger and less ductile than the usual Zircaloy-2 tubing materials. Also, the stress rupture properties of the Zr-Cr were lower than those of the Zircaloy-2 materials.
- b. Fine-grain Zr-Cr tubing pressurized at circumferential strain rates (comparable to those tested) greater than $0.024 \text{ \%/min}$, burst at hoop stresses approximately 10% greater than those pressurized at lower rates.
- c. The circumferential fracture elongation (CFE) of the fine-grain Zr-Cr tubing is less than 15% in the temperature range from room temperature to 350°C (662°F). At 400°C (752°F) and higher temperatures, the CFE increases as the temperature is increased.
- d. The CFE values of the stress rupture test specimens (for the test times evaluated) are usually greater than those of the burst test specimens at the same temperature.
- e. The constant of the Larson-Miller Parameter for the Zr-Cr material is 15.

FIGURE 5. Zr-Cr FINE-GRAIN HOOP STRESS VERSUS LARSON-MILLER PARAMETER (LMP)

FIGURE 6. MECHANICAL TEST SPECIMENS SHOWING FAILURE REGIONS

BURST TESTS

- A 25°C (77°F)
- B 200°C (392°F)
- C 300°C (572°F)
- D 400°C (752°F)
- E 600°C (1112°F)
- F 500°C (932°F)

STRESS RUPTURE TESTS

- G 500°C (932°F)
- H 600°C (1112°F)
- I 535°C (995°F)
- J 400°C (752°F)

FIGURE 7. ALL SPECIMENS - BY GROUPS

FIGURE 8. SPECIMENS 1, 5, 6, AND 15 (GROUP A) BURST-TESTED AT 25°C (77°F)

FIGURE 9. SPECIMENS 8 AND 30 (GROUP B) BURST-TESTED AT 200°C (392°F)

FIGURE 10. SPECIMENS 2, 7, 9, 10, AND 14 (GROUP C) BURST-TESTED AT 300°C (572°F)

FIGURE 11. SPECIMENS 3 AND 11 (GROUP D) BURST-TESTED AT 400°C (752°F)

FIGURE 12. SPECIMEN 13 (GROUP F) BURST-TESTED AT 500°C (932°F)

FIGURE 13. SPECIMENS 4 AND 12 (GROUP E) BURST-TESTED AT 600°C (1112°F)

FIGURE 14. SPECIMENS 16, 17, 18, 19, AND 25 (GROUP J) STRESS-RUPTURE-TESTED AT 400°C (752°F)

FIGURE 15. SPECIMENS 20 AND 21 (GROUP G) STRESS-RUPTURE-TESTED
AT 500°C (932°F)

FIGURE 16. SPECIMEN 24 (GROUP I) STRESS-RUPTURE-TESTED AT 535°C (995°F)

FIGURE 17. SPECIMENS 22 AND 23 (GROUP H) STRESS - RUPTURE-TESTED AT 600°C (1112°F)

SECTION IV

IRRADIATION TEST

4.1 DESCRIPTION OF TEST

The primary objective of this program is to evaluate Zr-Cr clad fuel rods in Consumers Big Rock Power Plant to a target of 15,000 to 20,000 MWd/T average burnup. These rods were positioned in a reload fuel assembly designated as B-01 so that they operate as hot as possible in order to evaluate maximum temperature irradiation conditions. Detailed post-irradiation examinations will be performed to evaluate fuel performance as compared to Zircaloy-2 control fuel irradiated in the same bundle.

4.1.1 Materials

Two specific materials were used in this experiment:

- a. Commercial Zr-Cr (Vendor Ingot No. 2), and
- b. Zircaloy-2 (Heat No. WC-643).

A summary of the Zr-Cr tubing mechanical property and corrosion test data was presented in Section III of this report.

The irradiation is being performed in a Consumers Big Rock Point reload fuel assembly designated as B-01. This bundle was un-irradiated, and was available at the Big Rock Point Reactor site. Twenty-two fuel rods and four nonfueled dummy rods were used in the experiment and replaced twenty-six existing fuel rods in this bundle.

All twenty-two fuel rods (Figure 18) are similar to the reload fuel design for 2.6% enriched UO_2 pellet-type fuel except that sixteen are clad with Zr-Cr and six rods with annealed Zircaloy-2.

The external geometry of the dummy rods resembles the reload fuel rods, and they are readily interchangeable with 2.6% enriched fuel rods. Each dummy rod (Figure 19) contains seven individual compartments. Three of these compartments are nonvented and can be separated after irradiation for use as burst or stress-rupture type specimens. These compartments also contain six tensile specimens in each, mounted on a tantalum rod. The four remaining compartments are vented by holes in the cladding to allow reactor water to come into contact with specimens inside. These specimens are mounted on a tantalum rod, and provide a means for non-heat-transfer corrosion to be measured by weight gain. Autoclave tests in high-oxygen steam at 300°C (572°F) (20 ppm O_2) show that tantalum exhibits excellent corrosion resistance at reactor temperatures.

The basic design of these dummy rods is as simple as possible, to facilitate easy access to tensile and corrosion specimens and provide post-irradiation tube samples for burst testing.

4.1.2 Test Variables

Although the primary purpose of this experiment is evaluation of Zr-Cr fuel cladding as a function of exposure, other variables were incorporated into this experiment to make it more complete. These variables include pre-irradiation condition of cladding and the irradiation environment.

4.1.2.1 Exposure

The maximum target exposure is 15,000 to 20,000 MWd/T average burnup. However, both fuel and dummy rods will be removed at a scheduled reactor shutdown to obtain data at about 7500 MWd/T average. Duplicate rods will be inserted to reconstitute the bundle and

FIGURE 18. FUEL ROD

12950401

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GENERAL ELECTRIC

107C5-7

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:			
APPLIED PRACTICES	SURFACES	TOLERANCES ON DIMENSIONS	ANGLES
145A5481	✓	FRACTIONS	DECIMALS

PART NO.		NAME		DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT	
1	1	END PLUG	129B2144 P1		
2	2		129B2144 P2		
3	3		129B2144 P3		
4	4		129B2144 P4		
5	5		129B2138 P1		
6	6		129B2138 P2		
7	7		129B2138 P3		
8	8	END PLUG	129B2138 P4		
9	9	TUBE	129B2139 P1		
10	10		129B2139 P2		
11	11		129B2139 P3		
12	12	TUBE	129B2139 P4		
13	13	ROD	129B2149 P1		
14	14	TENSILE SPECIMEN	129B2145 P1		
15	15		129B2145 P2		
16	16		129B2145 P3		
17	17	TENSILE SPECIMEN	129B2145 P4		
18	18	WIRE	.020 DIA. WIRE (TYPE 302 STN. STL.) #7A2.3A

NOTES:

1. SCRATCHES, NICKS, ETC., INTRODUCED IN THE TUBING FABRICATION & HANDLING SHALL NOT EXCEED .003 IN DEPTH.
2. FILL VOIDS WITH GRADE 99.995% PURE) HELIUM. ESCAPE THROUGH TUBES & END PLUGS OR WELDED REGIONS SHALL NOT EXCEED 10^{-8} STD. CM³/SEC. OF HELIUM WITH A VACUUM OF 20 MICRONS OF MERCURY.
3. ALL END PLUGS, TENSILE SPECIMENS, WIRES, AND ROD (P13) SURFACES SHALL BE FREE OF ALL VOLATILE CONTAMINANTS AT ASSEMBLY.
4. OUTSIDE DIAMETER OF END PLUG WELD SHALL NOT EXCEED .468 INCHES. THICKNESS OF TUBING OR WELD AT WELD ZONE SHALL NOT BE LESS THAN .029 INCHES.
5. S1 & G2 - PBG-Y90
G3 & G4 - LATER.
6. TENSILE SPECIMENS MAY BE WIRED AT RANDOM

G1 G2 G3 G4

DESCRIPTION OF GROUPS	REVISIONS	PRINTS TO
G1 - COLD DRAWN Zr-2 G2 - ROCKED Zr-2 G3 - COLD DRAWN Zr-Cr G4 - ROCKED Zr-Cr	1. REVISED 6.6.64 NE-8366	
MADE BY: R. Cate OCT. 2, 1964 CHECKED: R. G. Graham OCT. 7, 1964	APPROVED: A. P. E. D. SAN JOSE	REV. NO. 1

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FIGURE 19. DUMMY FUEL ROD

allow data at 12,500 MWd/T exposure to be obtained when the target exposure is reached. The irradiation schedule and the various types of fuel and dummy rods for this experiment are shown in Table VI.

4.1.2.2 Pre-Irradiation Cladding Condition

Three pre-irradiation cladding conditions will be tested:

- a. As etched,
- b. Low temperature autoclaved at 300°C for 70 hours, and
- c. Autoclaved at 400°C for 12 hours.

The testing of these initial cladding conditions was prompted by recent BNWL data⁽⁶⁾ where it was observed that in-reactor corrosion rates for Zircaloy-2 and Zr-Cr varied considerably with pre-irradiation conditioning. Since this variable has not been systematically tested in actual fuel rod experiments, it is considered equally important in this experiment.

4.1.2.3 Irradiation Environment

Three different environments will exist during irradiation testing:

- a. Heat transfer in reactor water (fuel cladding),
- b. Non-heat-transfer in reactor water (corrosion samples in vented compartments in dummy rods), and
- c. Non-heat-transfer in helium (tensile samples in sealed compartments in dummy rods).

These environments will allow separation of effects due to heat transfer during irradiation on corrosion rates and mechanical properties.

The actual rod positions for the initial 26 rods are shown in Table VII, with relative

power factors for each fuel rod. The Zr-Cr fuel rods were placed in the highest power positions available in the bundle to obtain the highest cladding temperature possible. In this manner, the Zr-Cr fuel cladding will be tested to evaluate corrosion resistance in-pile for comparison with that observed for this alloy in out-of-pile tests.

4.1.3 Post-Irradiation Tests

The post-irradiation examinations will be performed on specific rods both in the Consumers fuel storage pool and in hot cells at the Vallecitos Nuclear Center. Interim non-destructive tests will include leak checking, visual examinations, and profilometry in the Consumers spent fuel pool. Selected rods will be sent back to Vallecitos for extensive destructive examination. These include weight gain measurements on precut corrosion specimens, metallography for corrosion film measurement and determination of structural changes in cladding, and burst and tensile tests for measurements of irradiated cladding mechanical properties. Out-of-pile mechanical property and corrosion data are available from characterization studies for direct comparison with post-irradiation data.

4.2 STATUS OF IRRADIATION TEST

The Zr-Cr alloy-clad fuel rods accumulated exposure in the Big Rock Point reactor. The rate of fuel burnup has been approximately 800 MWd/TU (average) per month of full-power operation. The Zr-Cr alloy-clad fuel rods have accumulated a total of approximately 6000 MWd/TU average exposure as of January 31, 1968. This corresponds to about 5100 hours of high-temperature, full-power reactor operation.

Plans are being made to inspect the Zr-Cr fuel rods during the next Big Rock Point Reactor shutdown, which is scheduled for February 1968. During this shutdown, the B-01 assembly will be leak-checked, a visual

TABLE VI

Zr-Cr FUEL IRRADIATION EXPERIMENT

<u>Rod Description</u>			<u>Pre-Irradiation^(a)</u>	<u>Rod Type</u>
<u>Item</u>	<u>Cladding Alloy</u>	<u>Design Drawing</u>		
1	Zr-Cr	107C5326G4	1	Fuel Rod, 2.6% UO ₂ Pellet
2	Zr-Cr	107C5326G4	2	Fuel Rod, 2.6% UO ₂ Pellet
3	Zr-Cr	107C5326G4	3	Fuel Rod, 2.6% UO ₂ Pellet
4	Zr-Cr	161F263G2	1, 2, and 3	Unfueled Dummy Rod
5	Zircaloy-2	107C5326G2	1	Fuel Rod, 2.6% UO ₂ Pellet
6	Zircaloy-2	107C5326G2	2	Fuel Rod, 2.6% UO ₂ Pellet
7	Zircaloy-2	107C5326G2	3	Fuel Rod, 2.6% UO ₂ Pellet
8	Zircaloy-2	161F263G1	1, 2, and 3	Unfueled Dummy Rod

Initial Loading

<u>Item</u>	<u>Number of Rods</u>	<u>At 5000 to 7500 MWd/T</u>		<u>At 10,000 to 12,500 MWd/T</u>		<u>At 15,000 to 20,000 MWd/T</u>	
		<u>Removed^(b)</u>	<u>Replaced</u>	<u>Removed</u>	<u>Replaced</u>	<u>Removed</u>	<u>Replaced^(c)</u>
1	4	1 or 2	1 or 2	1 or 2	0	3 or 2	0
2	6	1 or 2	1 or 2	1 or 2	0	5 or 4	0
3	6	1 or 2	1 or 2	1 or 2	0	5 or 4	0
4	2	1	1	1	0	1	0
5	2	1	1	1	0	1	0
6	2	1	1	1	0	1	0
7	2	1	1	1	0	1	0
8	2	1	1	1	0	1	0

Fabrication Schedule

<u>Item</u>	<u>Number of Rods to be Fabricated</u>	<u>Identification of 26 Initial Rods</u>
1	4 plus 1 or 2 spares	FR4-1, FR4-2, FR4-3, and FR4-4
2	6 plus 1 or 2 spares	FR4-5, through FR4-10
3	6 plus 1 or 2 spares	FR-11, through FR4-16
4	2 plus 1 spare	DR1-1 and DR1-2
5	2 plus 1 spare	FR2-4 and FR2-5
6	2 plus 1 spare	FR2-6 and FR2-7
7	2 plus 1 spare	FR2-8 and FR2-9
8	2 plus 1 spare	DR2-3 and DR2-4

Notes

(a) Both Zr-Cr and Zircaloy-2 tubing used in this experiment was rocked to size and annealed. The Zircaloy-2 was annealed rocked tubing. Number designation refers to:

1. Cladding etched and cleaned.
2. Cladding etched, cleaned, and autoclaved at 300°C for 70 hours.
3. Cladding etched, cleaned, and autoclaved at 400°C for 12 hours.

(b) Decision whether 1 or 2 rod removal will be made at 5000 to 7500 MWd/T interim examination.

(c) Final replacement rods will be made of whatever cladding condition proves best from interim examination.

inspection performed, and diameter profiles obtained on selected Zr-Cr alloy-clad fuel rods.

being tested in the loop under heat transfer conditions, and have accumulated about 3800 hours exposure as of January 17, 1968.

The long-term out-of-pile loop corrosion tests are continuing. Zr-Cr alloy samples are

TABLE VII

ROD LOCATIONS FOR Zr-Cr ALLOY EXPERIMENT IN B-01 BUNDLE

Rod Number	Bundle Location	Relative Rod Power	Cladding Alloy	Cladding* Condition
<u>Fuel Rods</u>				
FR2-4	a-4	1.14	Zirc-2	1
FR2-5	h-1	1.14	Zirc-2	1
FR2-6	c-1	1.10	Zirc-2	2
FR2-7	k-8	1.14	Zirc-2	2
FR2-8	g-11	1.10	Zirc-2	3
FR2-9	d-11	1.14	Zirc-2	3
FR4-1	c-1	1.22	Zr-Cr	1
FR4-2	k-3	1.22	Zr-Cr	1
FR4-3	i-11	1.22	Zr-Cr	1
FR4-4	a-9	1.22	Zr-Cr	1
FR4-5	d-1	1.14	Zr-Cr	2
FR4-6	k-4	1.14	Zr-Cr	2
FR4-7	h-11	1.14	Zr-Cr	2
FR4-8	a-8	1.14	Zr-Cr	2
FR4-9	b-2	1.16	Zr-Cr	2
FR4-10	j-10	1.16	Zr-Cr	2
FR4-11	i-1	1.22	Zr-Cr	3
FR4-12	k-9	1.22	Zr-Cr	3
FR4-13	c-11	1.22	Zr-Cr	3
FR4-14	a-3	1.22	Zr-Cr	3
FR4-15	j-2	1.16	Zr-Cr	3
FR4-16	b-10	1.16	Zr-Cr	3
<u>Dummy Rods</u>				
DR2-3	f-1	0	Zirc-2	1, 2, and 3
DR2-4	f-11	0	Zirc-2	1, 2, and 3
DR4-1	a-6	0	Zr-Cr	1, 2, and 3
DR4-2	k-6	0	Zr-Cr	1, 2, and 3

*See Note (a) in Table VI

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