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**SPECIFIC ZIRCONIUM ALLOY
DESIGN PROGRAM
QUARTERLY PROGRESS REPORT
Nos. 22 AND 23**

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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 20,000 MWd/tU. The rate of fuel burnup has been approximately 800 MWd/tU (average) per month of full-power operation.

The long-term, out-of-pile, loop corrosion tests have been satisfactorily completed. The Zr-Cr samples were tested in a loop which simulated a boiling-water environment including heat transfer. The samples accumulated approximately 7,800 hours of exposure.

During the June - July 1968 reactor outage, the pool examination of the B-1 assembly indicated that the Zr-Cr and Zr-2 test rods were performing satisfactorily after 8,000 MWd/tU bundle average exposure. Nine test fuel rods and two dummy rods were removed from the bundle and returned to Vallecitos Nuclear Center. The post-irradiation examination of these rods has been initiated. The assembly was reconstituted and the irradiation continued. As of December 31, 1968, the Zr-Cr rods inserted in June 1967 have accumulated an average exposure of 12,000 MWd/tU, and the replacement rods inserted in June 1968 have accumulated an average exposure of 4,300 MWd/tU.

SECTION I

INTRODUCTION

This is the combined 22nd and 23rd Quarterly Progress Report on Project Agreement 24 of USAEC Contract AT(04-3)-189 covering the period from July 1, 1968 through December 31, 1968.

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 irradiation of the alloy, and detailed post-irradiation examinations of the fuel clad-

ding. Zr-Cr alloy-clad fuel rods were successfully fabricated⁽²⁾ for irradiation in the Consumers Big Rock Point reactor and have been examined during interim reactor shutdowns. 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 into fuel cladding. Results of these evaluations have been reported. ^(2,3,4)

The fuel rods have been under irradiation in the Consumers Big Rock Point reactor since June 1967. The present program effort is devoted to completing the corrosion characterization tests on samples of the fuel cladding and following the progress of the irradiation test program.

SECTION II

ALLOY DESIGN CONTINUATION

2.1 Zr-Cr TUBING FABRICATION

A topical report⁽²⁾ described 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. ⁽⁵⁾

Also previously described⁽³⁾ 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 has been completed. The tests were 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 has been performed as a control for the irradiation test program. This test has run for the same duration as the initial phase of the irradiation test (the first group of fuel rods was removed in early July 1968 after irradiation to 8000 MWd/tU or after 7800 hours).

3.1 CORROSION TESTS - CL-1 LOOP

The out-of-pile, long-term corrosion tests on the Zr-Cr tubing have been completed. The test was terminated on August 26, 1968. The specimens accumulated up to 7800 hours exposure in the simulated boiling water reactor environment of the CL-1 loop. (The exposure time is the same exposure time as the first group of irradiated test rods removed from Consumers Big Rock Point reactor during the June - July 1968 outage.) The data previously reported⁽⁴⁾ have been updated to include the final results of the tests. A review of the test parameters and corrosion test environments is being included in this report for completeness.

The tests were designed to evaluate the corrosion behavior of Zr-Cr alloy tubing with various surface treatments in simulated reactor environments and under heat transfer conditions. Zr-Cr tubing specimens were prepared with the following surface treatments:

- Etched and NaOH cleaned

- Etched, NaOH cleaned, and preoxidized at 300°C for 60 hours
- Etched, NaOH cleaned, and preoxidized at 400°C for 12 hours

It was necessary to introduce another variable, however. The heater elements required tubing with a diameter larger than the normal fuel tubing; therefore, two different sizes of tubing were fabricated. The small-diameter tubing, designated as "Zr-Cr fuel tubing," received the same fabrication process as the tubing for Zr-Cr clad fuel rods in the Consumers Big Rock Point reactor. The large-diameter tubing, designated as "Zr-Cr heat transfer tubing," received a slightly different fabrication process. The process flow diagrams for both the Zr-Cr fuel tubing and the Zr-Cr heat transfer tubing are shown in Figure 1. Metallographic characterization of the tubing after fabrication showed that the two tubings had comparable grain size (ASTM No. 11-12), hydride content, and second-phase particle size ($< 1/2 \mu$) and distribution. There was, however, a difference in the hardness. The fuel tubing had a hardness of DPH 197 ($R_B 90.9$) compared with a hardness of DPH 185 ($R_B 88.3$) for the heat transfer tubing.

The CL-1 loop simulates boiling water environment without neutron flux. The flow diagram of the loop is shown in Figure 2. The loop was operated at 285°C, 1000 psi, with saturated water containing approximately 0.2 ppm O_2 and approximately 0.025 ppm H_2 . The steam in the saturated steam test section contained approximately 23 ppm O_2 and 1-2 ppm H_2 .

Zr-Cr tubing and Zircaloy-2 control specimens were placed in the saturated steam coupon test section and the saturated water coupon test section. Heat transfer test specimens

(1) FIRST NUMBER IS OUTSIDE DIAMETER OF TUBE, SECOND NUMBER IS WALL THICKNESS.

(2) REDUCED TUBES WERE FLASH-ETCHED BEFORE ANNEAL.

FIGURE 1. FLOW DIAGRAM OF PROCESS USED IN FABRICATION OF COMMERCIAL Zr-Cr TUBING

FIGURE 2. SCHEMATIC DIAGRAM OF THE CL-1 BOILING WATER CORROSION TEST FACILITY.

of the Zr-Cr tubing operating at over 500,000 Btu/h/ft² were placed in the boiling heat transfer test sections. Also, in the boiling heat transfer test sections, just above the heat transfer test specimens and in line with the steam-water environment generated by the heat transfer test specimens, there was an isothermal Zr-Cr tubing specimen and a Zircaloy-2 control specimen.

The results of the corrosion tests in the CL-1 loop are summarized in Table I. The weight gains of the isothermal saturated water, heat transfer, and isothermal steam-water specimens are illustrated in Figure 3. The isothermal saturated water and steam-water specimens have similar weight gains for both the Zr-Cr tubing and Zircaloy-2 control specimens. The Zr-Cr heat transfer tubing specimens have reasonable weight gains. The weight gains of the etched and the preoxidized Zr-Cr heat transfer specimens are comparable. The surface appearance of the heat transfer and isothermal steam-water specimens is shown in Figure 4. The specimens had tightly adhering, lustrous black oxide films. No evidence of gross pitting or post-transition corrosion was observed.

The weight gains of the specimens in the saturated steam environment are presented in Figure 5. The results indicate:

- The Zr-Cr tubing specimens have higher weight gains than the Zircaloy-2 control. This anomalous behavior is contrary to high temperature (400°C and 500°C) steam autoclave corrosion results. (1)
- The Zr-Cr fuel tubing specimens have higher weight gains than the Zr-Cr heat transfer tubing specimen. The cause of the higher weight gains of the fuel tubing may be associated with the slight difference in the fabrication history of the two different size tubings.

- The preoxidized Zr-Cr fuel tubing specimens have higher weight gains than the etched Zr-Cr fuel tubing.

To verify the anomalous behavior of the Zr-Cr tubings in the saturated steam environment and the difference in the weight gain between the Zr-Cr fuel and heat transfer tubing, another set of specimens was prepared from the same lot of material. In addition, specimens were prepared from Zr-Cr swaged and drawn bar stock. The surfaces of the specimens were prepared under laboratory conditions. The new set (designated as the second set of specimens) was run concurrently with the first set of specimens in the CL-1 loop. The only difference is that the isothermal steam-water specimens are located in the steam and water coupon test section instead of directly in line with the heat transfer specimens in the boiling heat transfer test sections. The results, after 6992 hours of exposure, are summarized in Table II.

The weight gains of the specimens in the saturated water and steam-water environments are lower, but similar to the weight gains of the first set of specimens. The weight gains of the specimens located in the saturated steam environment are illustrated in Figure 6. The data indicate the same type of behavior as the first set of specimens. However, the oxidation rates are considerably lower for the second set of specimens. The test has been terminated with the completion of the corrosion tests on the first set of specimens.

The out-of-pile corrosion tests have been completed except for the metallographic examination of the oxide films and correlation of weight gain and the oxide film thickness. Destructive examination of the out-of-pile corrosion specimens will be performed concurrent with and compared to the first group of irradiated test rods. After the above examination is completed, a critical review of the corrosion data will be made.

TABLE I
CL-1 LOOP CORROSION TEST RESULTS, FIRST SET OF SPECIMENS

Specimen Description	Initial Surface Condition ⁽¹⁾	Loop Environment	Specimen Number	Average Weight Gain ⁽²⁾ (mg/dm ²)																
				2549h ⁽³⁾	3149h	4146h	5134h	6134h	7128h	8130h	9134h	10142h								
Zr-2 Control	1	285°C (546°F) Saturated Water ⁽⁴⁾	1	16.8	16.6	18.4	19.1	19.6	19.7	20.5	21.1	23.3								
				629h	1229h	2226h	3214h	4214h	5208h	6209h	7213h	8221h								
Zr-Cr Fuel Tubing	1	285°C (546°F) Saturated Water ⁽⁴⁾	1	11.9	14.4	17.2	19.4	19.7	20.8	22.6	25.2	26.5								
				271h	600h	871h	1597h	1868h	2535h	2856h	3585h	3856h	4579h	4850h	5580h	5851h	6584h	6855h	7592h	7863h
Zr-Cr Heat Transfer Tubing Under Heat Transfer	1	280°C (536°F) to 285°C (546°F) at 514, 500 ⁽⁵⁾ Btu/h-ft ²	1		12.2		20.4		30.5		30.2		36.4		45.7		55.6		56.6	
Zr-Cr Heat Transfer Tubing Under Heat Transfer	3	280°C (536°F) to 285°C (545°F) at 518, 900 ⁽⁵⁾ Btu/h-ft ²	1	5.0		9.4		15.9		30.4		28.8		34.6		42.7		50.2		54.8
Zr-Cr Heat Transfer Tubing	1	285°C (546°F) Iso-thermal ⁽⁵⁾ Steam-Water	1	8.7		12.4		15.4		17.1		18.5		19.9		21.0		21.9		22.2
Zr-2 Control	1	285°C (546°F) Iso-thermal ⁽⁵⁾ Steam-Water	1	8.4		12.3		15.3		17.1		18.2		19.9		21.2		21.8		23.3
Zr-Cr Fuel Tubing	1	285°C (546°F) Saturated Steam ⁽⁶⁾	1 2 3 Avg.	7.4 7.8 7.8 7.7		10.2 16.8 10.5 12.5		19.9 70.7 17.6 36.1		64.1 128 38.6 77		133 252 83.4 156		229 380 109 240		342 503 154 333		452 624 215 430		512 528 260 433
Zr-Cr Fuel Tubing	2	285°C (546°F) Saturated Steam ⁽⁶⁾	1 2 3 Avg.		9.0 17.6 34.0 20.2		50 78.9 141 90		145 174 291 203		252 272 439 321		369 380 --- 375		496 498 --- 497		615 604 --- 610		653 637 --- 645	
Zr-Cr Fuel Tubing	3	285°C (546°F) Saturated Steam ⁽⁶⁾	1 2 3 Avg.	19.1 9.4 13.7 14		105 67.2 70.3 81		243 186 224 218		441 357 398 399		600 508 553 554		763 666 713 714		928 666 784 793		1085 945 933 988		1075 970 1024 1023
Zr-Cr Heat Transfer Tubing	1	285°C (546°F) Saturated Steam ⁽⁶⁾	1	7.2		9.9		18.2		30.5		47.6		78.3		116		161		200
Zr-2 Control	1	285°C (546°F) Saturated Steam ⁽⁶⁾	1	11.2		12.3		15.7		19.2		19.4		21.3		23.1		21.9		20.9

- (1) 1 - Etched and NaOH cleaned.
 2 - Etched, NaOH cleaned, and preoxidized at 300°C for 60 hours (2.6 mg/dm²).
 3 - Etched, NaOH cleaned, and preoxidized at 400°C for 12 hours (5.9 mg/dm²).
 (2) For preoxidized specimens, preoxidation weight gain is not included.
 (3) 282°C (539°F) subcooled water 1.5 ppm O₂ in CL-4 loop, transferred to CL-1 loop.
 (4) Isothermal section of CL-1 loop with ~0.2 ppm O₂ and ~0.025 ppm H₂.
 (5) Heat transfer section of CL-1 loop has 20 ppm O₂ and 2 ppm H₂ in steam phase.
 (6) Saturated steam section of CL-1 loop has 23 ppm O₂ and 1-2 ppm H₂.

FIGURE 3. OUT-OF-PILE CL-1 LOOP CORROSION TEST DATA ON Zr-Cr ALLOY TUBING (546°F, 1000 psi). FIRST SET OF SPECIMENS.

1. Zr-Cr HEAT TRANSFER TUBING UNDER HEAT TRANSFER, 518, 900 Btu/h-ft², PREOXIDIZED AT 400°C FOR 12 HOURS.
2. Zr-Cr HEAT TRANSFER TUBING UNDER HEAT TRANSFER, 514, 500 Btu/h-ft², ETCHED.
3. Zr-2 CONTROL, ISOTHERMAL STEAM - WATER, ETCHED
4. Zr-Cr HEAT TRANSFER TUBING, ISOTHERMAL STEAM - WATER, ETCHED

HEAT TRANSFER TESTS 500,000 Btu/h-ft²

- 1). Zr-Cr, 7863 HOURS EXPOSURE
 - 2). Zr-Cr, 7592 HOURS EXPOSURE
- #### ISOTHERMAL TEST STEAM - WATER
- 3). Zr-2, 7863 HOURS EXPOSURE
 - 4). Zr-Cr, 7863 HOURS EXPOSURE

FIGURE 4. SURFACE APPEARANCE HEAT TRANSFER AND STEAM - WATER SPECIMENS AFTER 7800 HOURS EXPOSURE

FIGURE 5. OUT-OF-PILE CL-1 LOOP SATURATED STEAM CORROSION TEST DATA ON Zr-Cr ALLOY TUBING (546°F, 1000 psi). FIRST SET OF SPECIMENS.

TABLE II

CL-1 LOOP CORROSION TEST RESULTS, SECOND SET OF SPECIMENS

Spec. Description	Initial ⁽¹⁾ Condition	No. of Specimens	Environment	Average Weight Gain ⁽²⁾ (mg/dm ²)							
				Preoxidation	997h	1985h	2985h	3979h	4980h	5984h	6992h
Zr-Cr Fuel Tubing	1	3	285°C (546°F) Saturated Water ⁽³⁾	--	8.0	9.6	10.3	10.9	12.1	13.2	13.6
Zr-Cr Fuel Tubing	2	3	285°C (546°F) Saturated Water ⁽³⁾	6.9	3.0	4.6	5.4	6.5	7.3	8.1	9.3
Zr-Cr Heat Transfer Tubing	1	3	285°C (546°F) Saturated Water ⁽³⁾	--	7.8	11.4	11.0	10.9	11.6	13.5	14.7
Zr-Cr Heat Transfer Tubing	2	3	285°C (546°F) Saturated Water ⁽³⁾	6.3	2.5	4.6	5.5	4.5	6.2	7.2	7.8
Zircaloy-2 Tubing	1	3	285°C (546°F) Saturated Water ⁽³⁾	--	8.2	10.4	11.1	11.3	12.4	13.5	14.6
Zircaloy-2 Tubing	2	3	285°C (546°F) Saturated Water ⁽³⁾	10.2	4.5	6.3	7.0	6.7	8.5	10.1	11.1
Zr-Cr Fuel Tubing	1	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	--	7.7	9.8	10.9	11.4	12.6	13.5	14.7
Zr-Cr Fuel Tubing	2	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	6.8	3.0	6.8	6.8	6.4	8.1	8.3	10.0
Zr-Cr Heat Transfer Tubing	1	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	--	8.0	10.2	10.9	12.0	12.0	13.5	14.9
Zr-Cr Heat Transfer Tubing	2	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	5.8	3.1	4.7	4.7	7.1	6.9	8.1	8.8
Zircaloy-2 Tubing	1	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	--	9.1	10.3	10.9	12.2	12.2	13.9	14.9
Zircaloy-2 Tubing	2	3	285°C (546°F) Isothermal Steam-Water ⁽⁴⁾	10.5	4.3	5.8	6.4	6.6	8.6	9.2	10.2

(1) 1 - Etched, NaOH cleaned.

2 - Etched, NaOH cleaned, and preoxidized in 400°C steam 14 hours.

(2) For preoxidized specimens, preoxidation weight gain is not included.

(3) Isothermal section of CL-1 loop with ~0.2 ppm O₂ and ~0.025 ppm H₂ in the water.(4) Heat transfer section of CL-1 loop has 20 ppm O₂ and 2 ppm H₂ in steam phase.(5) Saturated steam section of CL-1 loop has 23 ppm O₂ and 1-2 ppm H₂ in the steam.

TABLE II
CL-1 LOOP CORROSION TEST RESULTS, SECOND SET OF SPECIMENS (Continued)

Spec. Description	Initial ⁽¹⁾ Condition	No. of Specimens	Environment	Average Weight Gain ⁽²⁾ (mg/dm ²)							
				Preoxidation	997h	1985h	2985h	3979h	4980h	5984h	6992h
Zr-Cr Fuel Tubing	1	3	285°C (546°F) Saturated Steam ⁽⁵⁾	--	17.9	60.9	119	181	258	340	424
Zr-Cr Fuel Tubing	2	3	285°C (546°F) Saturated Steam ⁽⁵⁾	7.1	35.2	107	186	281	420	540	648
Zr-Cr Heat Transfer Tubing	1	3	285°C (546°F) Saturated Steam ⁽⁵⁾	--	10.6	24.9	52.8	85.2	125	175	230
Zr-Cr Heat Transfer Tubing	2	3	285°C (546°F) Saturated Steam ⁽⁵⁾	6.3	22.6	51.3	119	183	259	344	426
Zircaloy-2 Tubing	1	3	285°C (546°F) Saturated Steam ⁽⁵⁾	--	7.8	10.2	10.5	10.6	12.1	13.3	13.9
Zircaloy-2 Tubing	2	3	285°C (546°F) Saturated Steam ⁽⁵⁾	8.7	4.6	6.0	7.6	6.9	8.0	8.7	9.1
				<u>1000h</u>	<u>1994h</u>	<u>2995h</u>	<u>3999h</u>	<u>5007h</u>			
Zr-Cr Bar Stock, Swaged	1	2	285°C (546°F) Steam-Water ⁽⁴⁾	7.3	8.5	10.7	9.9	11.4			
Zr-Cr Bar Stock, Drawn	1	1	285°C (546°F) Steam-Water ⁽⁴⁾	--	7.3	11.5	10.9	12.5			

(1) 1 - Etched, NaOH cleaned.

2 - Etched, NaOH cleaned, and preoxidized in 400°C steam 14 hours.

(2) For preoxidized specimens, preoxidation weight gain is not included.

(3) Isothermal section of CL-1 loop with ~0.2 ppm O₂ and ~0.025 ppm H₂ in the water.

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

(5) Saturated steam section of CL-1 loop has 23 ppm O₂ and 1-2 ppm H₂ in the steam.

FIGURE 6. OUT-OF-PILE CL-1 LOOP SATURATED STEAM CORROSION TEST DATA ON Zr-Cr ALLOY TUBING (546°F, 1000 psi). SECOND SET OF SPECIMENS.

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 Point power plant to a target burnup of 20,000 MWd/tU assembly average. These rods were positioned in a reload fuel assembly (designated as B-1) to operate as hot as possible in order to evaluate maximum temperature irradiation conditions. Detailed post-irradiation examinations are being performed to evaluate fuel performance as compared to Zircaloy-2 control fuel irradiated in the same bundle.

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 are similar to the reload fuel design for 2.6% enriched UO_2 pellet-type fuel, except that sixteen rods are clad with Zr-Cr and six 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 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.

4.1.1 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.

Three pre-irradiation cladding conditions will be tested:

- As-etched
- Low temperature autoclaved at 300°C for 70 hours
- Autoclaved at 400°C for 12 hours

The testing of these initial cladding conditions was prompted by 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.

Three different environments exist during the irradiation testing:

- Heat transfer in reactor water (fuel cladding)
- Nonheat transfer in reactor water (corrosion samples in vented compartments in dummy rods)
- Nonheat 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 maximum target exposure is 20,000 MWd/tU average burnup. Both fuel and dummy rods were removed during the June - July 1968 reactor shutdown to obtain data at 8,000 MWd/tU average assembly exposure. Replacement rods were inserted to reconstitute the bundle and allow data to be obtained at 12,000 MWd/tU exposure when the target exposure of 20,000 MWd/tU is reached.

The irradiation schedule and the various types of fuel and dummy rods for this experiment are shown in Table III.

4.2 INSPECTION OF ASSEMBLY B-1 DURING JUNE - JULY 1968 OUTAGE

Bundle B-1 was among the highest power bundles in the Consumers Big Rock Point core during the last run. It is the carrier bundle for the Zr-Cr rods which are under test. The bundle was leak-checked and found to be sound, with no indication of failures after 8,000 MWd/tU exposure.

Using a periscope, a visual examination of all four sides of the bundle was performed in the Consumers Big Rock Point reactor pool. The visual examination indicated no basic change in the surface appearance of the rods had occurred since the February 1968 examination.⁽⁴⁾ The most obvious feature in the bundle was the fact that the dummy rods had considerably less crud deposits than the power-producing rods. All of the power-producing rods in the bundle showed a reddish crud deposit typical of all fuel bundles examined. There are no differences in appearance between the power-producing rods: the different alloys, Zircaloy-2 and Zr-Cr; and the different preconditionings: etched, 300°C autoclaved, or 400°C autoclaved.

Both the top and bottom end plugs of the fuel rods were examined in detail. There were no indications of any anomalies and no particular differences in the appearance of the weld heat-affected zone near the end plugs. On the dummy rods, it was possible to see some difference in color at the welds which joined the various sections together. The heat-affected zones were slightly lighter in color. The dummy sections have segments which have holes in them to allow water ingress for corrosion tests. The holes at the bottom spacer appeared clear and sharp with little crud buildup. The holes further up the rods were still clear and sharp; however, the holes were outlined by crud buildup. The

edges of the corrosion and tensile specimens could be seen within the dummy rods.

Near the bottom spacer, the crud deposition was relatively light. A rod was selected, removed from the bundle, and profiled. The profile showed no unusual characteristics and was similar to profiles of other rods obtained during prior shutdowns. Nine fuel rods and two dummy rods were removed from assembly B-1 for post-irradiation examination. The bundle was reconstituted with Zr-2 and Zr-Cr replacement rods as per the schedule in Table III. Photographs were taken of the reconstituted bundle (Figures 7 and 8). Assembly B-1 was returned to the core and will continue to be irradiated.

4.3 STATUS OF IRRADIATION TEST

The Zr-Cr alloy-clad fuel rods are presently accumulating exposure in the Consumers 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 fuel rods inserted into assembly B-1 in June 1967 have accumulated approximately 12,300 MWd/tU average exposure as of December 31, 1968. The total time of high temperature reactor exposure is approximately 11,700 hours. The Zr-Cr replacement rods inserted in June - July 1968 outage have accumulated approximately 4,300 MWd/tU (~ 3000 hours total time at temperature).

Plans are again being made to inspect the Zr-Cr fuel rods during the next Consumers Big Rock Point reactor shutdown, which is scheduled for April - May 1969. Again during this shutdown, the B-1 assembly will be leak-checked and a visual inspection performed on the Zr-Cr alloy-clad fuel rods.

4.4 POST-IRRADIATION EXAMINATION

Nine fuel rods and two dummy rods were removed from the B-1 assembly during the June - July 1968 outage. The fuel rods were cut in

TABLE III

Zr-Cr FUEL IRRADIATION EXPERIMENT

Rod Description							
Item	Cladding Alloy	Pre-Irradiation ⁽¹⁾ Cladding Condition	Rod Type ⁽²⁾				
1	Zr-Cr Alloy	1	Fuel Rod				
2	Zr-Cr Alloy	2	Fuel Rod				
3	Zr-Cr Alloy	3	Fuel Rod				
4	Zr-Cr Alloy	1, 2, and 3	Unfueled Dummy				
5	Zircaloy-2	1	Fuel Rod				
6	Zircaloy-2	2	Fuel Rod				
7	Zircaloy-2	3	Fuel Rod				
8	Zircaloy-2	1, 2, and 3	Unfueled Dummy				
Loading Plan							
Item	Number of Initial Rods	At 8,000 MWd/T ⁽³⁾		At 12,000 MWd/T		At 20,000 MWd/T	
		Removed	Replaced	Removed	Replaced	Removed	Replaced
1	4	2	2	2	0	2	0
2	6	2	2	2	0	4	0
3	6	2	2	2	0	4	0
4	2	1	0	0	0	1	0
5	2	1	0	0	0	1	0
6	2	1	0	0	0	1	0
7	2	1	3	1	0	4	0
8	2	1	0	0	0	1	0

(1) Zr-Cr alloy and Zircaloy-2 tubing were rocked to size and annealed. Number designation refers to:

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

(2) Fuel rods contain 2.6% enriched UO₂ pellets, 0.449 in o.d. × 0.032 in wall cladding. Each dummy rod contains 18 pre-cut tensile coupons, 24 pre-cut corrosion coupons and 3 burst samples.

(3) Completed during June - July 1968 outage.

(a)

↑
DR 4-2(Zr-Cr)

BUNDLE B-1, SIDE D, BETWEEN
SECOND SPACER FROM BOTTOM
AND MIDDLE SPACER.

NOTICE ABSENCE OF CRUD ON
CENTER DUMMY NONFUELED
Zr-Cr CLADDING (ROD DR-4:2)
CRUD AROUND VENT HOLES AND
AND WELDS.

BUNDLE B-1, SIDE D, BELOW
BOTTOM SPACER. NO EVIDENCE
OF CORROSION AT LOWER END
PLUG WELDS. NO CRUD.

(b)

↑
DR 4-2(Zr-Cr)

**FIGURE 7. BUNDLE B-1 Zr-Cr FUEL ROD CARRIER — POOL PHOTOGRAPHS
(JUNE-JULY 1968)**

BUNDLE B-1. SIDE C,
SECOND SPACER FROM TOP.
NOTICE DUMMY Zr-2 ROD IN
CENTER AND NEW REPLACE-
MENT RODS.

(a) $\uparrow$ $\uparrow$ $\uparrow$
FR 4-21(Zr-Cr) DR 2-4(Zr-2)
FR 2-12(Zr-2)

BUNDLE B-1, SIDE C,
NOTICE DUMMY Zr-2
ROD IN CENTER AND
NEW REPLACEMENT
ROD

(b) $\uparrow$ $\uparrow$
FR 2-12(Zr-2) DR 2-4(Zr-2)

**FIGURE 8. BUNDLE B-1 Zr-Cr FUEL ROD CARRIER — POOL PHOTOGRAPHS
(JUNE — JULY 1968)**

half, using an underwater shear in the Consumers Big Rock Point reactor pool, and returned to the Vallecitos Nuclear Center in a 700-EX series cask for destructive examination. The dummy rods were returned as full-length rods in another cask.

The post-irradiation examinations have been initiated. Table IV lists the rods returned and the initial scope of the post-irradiation examinations. This initial work will relate only to the corrosion behavior of the Zr-Cr alloy and will be unique because of the design of the experiment. The effects of heat transfer and irradiation on the corrosion rate of the Zr-Cr alloy will be separable as well as an unequivocal evaluation of the effect of the three pre-treatments performed on the Zr-Cr alloy surfaces in contact with the boiling water environments.

The visual examination of the test fuel rods prior to removing the crud has been completed. The surfaces of the rods were reddish-brown in color (crud deposits). At the top and bottom of the rods, the black oxide (ZrO_2) could be seen through the thin crud deposit. There were no indications of any anomalies or any particular difference between the end-plug weld heat-affected zones and the areas away from the heat-affected zones.

Relatively thick crud deposits were found at spring contact areas at the intermediate spacer positions (Figure 9). Also "simple-like" protuberances were found on several of the rods in the area of the spacer locations. These formations were most prominent at the lower intermediate spacer location. Evidence of these formations was found only at the spacer locations.

Superficial crud spalling or evidence of the crud beginning to spall was found on rods FR4-3, FR4-4, FR4-9, and FR4-14. The crud spalling occurred in areas of relatively thick crud deposition just above or below the middle intermediate spacer position. The most pronounced area of spalling occurred on rod FR4-14 and is shown in Figure 10.

Full-length dimensional profiles (longitudinal traverse with rotation), before decruding, have been performed on rods FR4-4, FR4-8, and FR4-14. Figure 11 shows the dimensional profile of rod FR4-14 and is typical of the profiles of the other two rods. The crud deposition apparently has caused an increase in the diameter of the center portion of the rod. Note that at the spacer locations, the low point of the profiles is approximately the same diameter as the ends of the rod. Considerable ovality was found in the areas of high crud deposition. In the areas of the crud spalling, the sharp jogs in the profile trace indicates the spalling is superficial and is only about 0.002 inch deep.

Rods FR4-3, FR4-4, FR4-8, FR4-9, and FR4-14 have been gamma scanned. The peak power position in all of the rods is approximately seven inches above the middle intermediate spacer position. Depressions were noted at axial positions on the gamma scan corresponding to the spacer locations.

The post-irradiation examination is continuing. The crud will be removed from selected rods to enable detailed examination of the actual fuel rod surface to be performed. After crud removal, rods FR4-4, FR4-8, and FR4-14 will be re-profiled. Regions then will be selected for metallographic examination.

TABLE IV

INITIAL SCOPE OF POST-IRRADIATION EXAMINATIONS, Zr-Cr ALLOY DEVELOPMENT

Rods Removed from Assembly B-1 After ~ 8,000 MWd/tU Average Burnup (7,800 h)

B-1 Assembly Position	Rod Serial No.	Cladding Alloy	Initial Cladding Surface Condition	Rod Contents	INITIAL POST-IRRADIATION WORK						
					Visual	γ -Scan	Profile	Remove Crud	Repeat Profile	Repeat Visual	Cut Samples
h-1	FR2-5	Zr-2	1	UO ₂	Yes	---	---	Yes	---	Yes	Yes
e-1	FR2-6	Zr-2	2	UO ₂	Yes	---	---	Yes	---	Yes	Yes
g-11	FR2-8	Zr-2	3	UO ₂	Yes	---	---	Yes	---	Yes	Yes
i-11	FR4-3	Zr-Cr	1	UO ₂	Yes	Yes	---	---	---	---	?
a-9	FR4-4	Zr-Cr	1	UO ₂	Yes	Yes	Yes	Yes	Yes	Yes	Yes
b-2	FR4-9	Zr-Cr	2	UO ₂	Yes	Yes	---	---	---	---	?
a-8	FR4-8	Zr-Cr	2	UO ₂	Yes	Yes	Yes	Yes	Yes	Yes	Yes
j-2	FR4-15	Zr-Cr	3	UO ₂	Yes	---	---	---	---	---	?
a-3	FR4-14	Zr-Cr	3	UO ₂	Yes	Yes	Yes	Yes	Yes	Yes	Yes
f-1	DR2-3	Zr-2	1,2,3	Samples	Visual		Length Measurement		Remove Samples		
					Yes		Yes		Yes		
a-6	DR4-1	Zr-Cr	1,2,3	Samples	Yes		Yes		Yes		

Initial Surface Conditions:

1. Etched and Cleaned (0 mg/dm²)
2. Etched, Cleaned, and Autoclaved 300°C for 70 h (2.6 mg/dm²)
3. Etched, Cleaned, and Autoclaved 400°C for 12 h (5.9 mg/dm²)

NOTE: "PIMPLE-LIKE" PROTUBERANCES IN THE AREA OF THE SPACER LOCATION THE OBLONG SHAPES ARE THE SPACER SPRING CONTACT AREAS. NOTE THAT THE CRUD HAS SPALLED OFF AT ONE OF THE SPACER SPRING CONTACT AREAS AND THE TUBING WALL CAN BE SEEN IN THE SPALL AREA.

FIGURE 9. VIEW OF SPACER SPRING CONTACTS AT LOWER INTERMEDIATE SPACER LOCATION ON ROD FR4-8

NOTE: THE SPIRAL LINES ARE FROM THE DIMENSIONAL PROFILE TRACE.

FIGURE 10. SUPERFICIAL CRUD SPALLING ON ROD FR 4-14

FIGURE 11. DIMENSIONAL PROFILE OF ROD FR 4-14

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