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GEAP-10026
JOINT U.S.-EURATOM RESEARCH
AND DEVELOPMENT REPORT
APRIL 1969

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**SPECIFIC ZIRCONIUM ALLOY
DESIGN PROGRAM
QUARTERLY PROGRESS REPORT
No. 24**

MASTER

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U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT PA 24

P2633

NUCLEAR FUELS DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA

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QUARTERLY PROGRESS REPORT NO. 24

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Prepared Under U.S. Atomic Energy Commission
Contract AT(04-3)-189
Project Agreement 24
for the
Joint U. S. - Euratom Research and Development Program

Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia
Price: \$3.00 per copy

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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.

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 March 31, 1969, the Zr-Cr rods inserted in June 1967 have accumulated an average exposure of 13,936 MWd/tU, and the replacement rods inserted in June 1968 have accumulated an average exposure of 5,936 MWd/tU.

SECTION I

INTRODUCTION

This is the 24th Quarterly Progress Report on Project Agreement 24 of USAEC Contract AT(04-3)-189 covering the period from January 1 through March 31, 1969.

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 cladding. 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 barstock 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, 4, 5) 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 can be realized on tubing produced by conventional processes.

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 test parameters and test environments were reported in detail in the last quarterly.⁽⁵⁾ A brief summary of the results of the CL-1 loop follows:

The weight gains of the isothermal saturated water, heat transfer, and isothermal steam-water specimens are illustrated in Figure 1. 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

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

Zr-Cr heat transfer specimens are comparable. The heat transfer 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 2. The results indicate the following:

- 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. ⁽⁶⁾
- 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 barstock. 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 of the corrosion tests on the second set of specimens were also reported in detail in the previous quarterly. A brief summary of the results follows:

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 3.

The data indicate the same type of behavior as the first set of specimens. However, the corrosion rates are considerably lower for the second set of specimens. The test was terminated with the completion of the corrosion tests on the first set of specimens.

The heat transfer and isothermal steam-water specimens are being sectioned to examine the oxide films and compare oxide thicknesses with the measured weight gains. The results will be compared directly with the metallographic results on the fuel rods and dummy rods.

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

FIGURE 2. OUT-OF-PILE CL-1 LOOP SATURATED STEAM CORROSION TEST DATA ON Zr-Cr ALLOY TUBING (546°F, 1000 psi). FIRST SET OF SPECIMENS.

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

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 test rods have been placed in the peripheral positions in the fuel assembly. Figure 4 shows the location of each rod at the start of the irradiation program in June 1967. Table 1 has the calculated (assuming an isotropic flux distribution across the bundle) relative power factors for each of the test fuel rods. The Zr-Cr fuel rods were placed in the highest power positions available in the bundle to obtain the highest cladding temperature possible.

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 60 hours
- Autoclaved at 400°C for 14 hours

The testing of these initial cladding conditions was prompted by BNWL data⁽⁷⁾ 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

	a	b	c	d	e	f	g	h	i	j	k
1			FR4-1	FR4-5	FR2-6*	DR2-3*		FR2-5*	FR4-11		
2		FR4-9*								FR4-15*	
3	FR4-14*										FR4-2
4	FR2-4										FR4-6
5											
6	DR4-1*										DR4-2
7											
8	FR4-8*										FR2-7
9	FR4-4*										FR4-12
10		FR4-16								FR4-10	
11			FR4-13	FR2-9		DR2-4	FR2-8*	FR4-7	FR4-3*		

ORIENTATION HOLE

* DENOTES ROD REMOVED DURING
JUNE - JULY 1968 FUEL OUTAGE

**FIGURE 4. LOCATION OF TEST FUEL AND DUMMY RODS IN ASSEMBLY B-I
AT START OF IRRADIATION PROGRAM, JUNE 1967**

TABLE 1

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

<u>Number</u>	<u>Location</u>	<u>Relative Rod Power</u>	<u>Cladding Alloy</u>	<u>Cladding* Condition</u>	<u>Removed June 1968</u>	<u>Replaced With</u>	<u>Cladding Alloy</u>	<u>Cladding* Condition</u>
<u>Fuel Rods</u>								
FR2-4	a-4	1.14	Zircaloy-2	1	-	-	-	-
FR2-5	h-1	1.14	Zircaloy-2	1	x	FR2-10	Zircaloy-2	3
FR2-6	e-1	1.10	Zircaloy-2	2	x	FR2-11	Zircaloy-2	3
FR2-7	k-8	1.14	Zircaloy-2	2	-	-	-	-
FR2-8	g-11	1.10	Zircaloy-2	3	x	FR2-12	Zircaloy-2	3
FR2-9	d-11	1.14	Zircaloy-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	x	FR4-21	Zr-Cr	1
FR4-4	a-9	1.22	Zr-Cr	1	x	FR4-22	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	x	FR4-25	Zr-Cr	2
FR4-9	b-2	1.16	Zr-Cr	2	x	FR4-26	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	x	FR4-31	Zr-Cr	3
FR4-15	j-2	1.16	Zr-Cr	3	x	FR4-32	Zr-Cr	3
FR4-16	b-10	1.16	Zr-Cr	3	-	-	-	-
<u>Dummy Rods</u>								
DR2-3	f-1	0	Zircaloy-2	1, 2, and 3	x	FR2-13	Zircaloy-2	3
DR2-4	f-11	0	Zircaloy-2	1, 2, and 3	-	-	-	-
DR4-1	a-6	0	Zr-Cr	1, 2, and 3	x	FR2-14	Zircaloy-2	3
DR4-2	k-6	0	Zr-Cr	1, 2, and 3	-	-	-	-

*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 60 hours; and
- 3 - Cladding etched, cleaned, and autoclaved at 400°C for 14 hours.

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.

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 detailed results of the latest pool inspection were reported previously and a brief summary of the results of the inspection is included here to maintain the continuity of the irradiation program.⁽⁵⁾

During the June—July 1968 reactor outage bundle B-1 was removed from the core for leak checking and visual examination. Leak checking indicated the bundle was sound with no indications of failure after 8000 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 last pool examination in February 1968.⁽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.

Nine fuel rods and two dummy rods were removed from assembly B-1 for the

post-irradiation examination now in progress. The bundle was reconstituted with Zr-2 and Zr-Cr replacement rods as per the schedule in Table 2.

Table 1 shows the location of the rods removed and the number and surface condition of the replacement rods. Assembly B-1 was returned to the core and the irradiation is continuing.

4.3 STATUS OF IRRADIATION TEST

The Zr-Cr alloy-clad fuel rods are presently accumulating exposure in the Consumers Big Rock Point reactor. Consumers Big Rock Point reactor is a boiling water reactor operating at 1325 psi and 305°C (580°F).

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 13,900 MWd/tU average exposure as of March 31, 1969. The total time of high temperature reactor exposure is approximately 13,800 hours. The Zr-Cr replacement rods inserted in the June—July 1968 outage have accumulated approximately 5900 MWd/tU (~6000 hours total time at temperature).

During the June—July 1968 reactor shutdown, nine fuel rods and two dummy rods were removed from assembly B-1. Assembly B-1 had accumulated approximately 8000 MWd/tU average exposure with a total time at temperature of approximately 7800 hours. Table 3 lists the rods removed and the approximate average rod burnup. Figure 5 shows the reactor power history from July 1967 to June 1968. At full power operation the hottest Zr-Cr rods were operating with a peak heat flux of approximately 320,000 Btu/h-ft².

Plans are being made to inspect the Zr-Cr fuel rods during the next Consumers

TABLE 2

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 60 hours; and
- 3 - Cladding etched, cleaned, and autoclaved at 400°C for 14 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.

TABLE 3

RODS REMOVED FROM ASSEMBLY B-1 AFTER ~8,000 MWd/tU AVERAGE BURNUP (7,900 h)

B-1 Assembly Position	Rod Serial No.	Cladding Alloy	Initial Cladding Surface Condition	Rod Contents	Enrichment (%)	Relative Rod Power	Average Rod Burnup (MWd/tU)
h-1	FR2-5	Zircaloy-2	1	UO ₂	2.6	1.14	9120
e-1	FR2-6	Zircaloy-2	2	UO ₂	2.6	1.10	8800
g-11	FR2-8	Zircaloy-2	3	UO ₂	2.6	1.10	8800
i-11	FR4-3	Zr-Cr	1	UO ₂	2.6	1.22	9760
a-9	FR4-4	Zr-Cr	1	UO ₂	2.6	1.22	9760
a-8	FR4-8	Zr-Cr	2	UO ₂	2.6	1.14	9120
b-2	FR4-9	Zr-Cr	2	UO ₂	2.6	1.16	9280
a-3	FR4-14	Zr-Cr	3	UO ₂	2.6	1.22	9760
i-2	FR4-15	Zr-Cr	3	UO ₂	2.6	1.16	9280
f-1	DR2-3	Zr-2	1, 2, 3	Samples	-	-	-
a-6	DR4-1	Zr-Cr	1, 2, 3	Samples	-	-	-

Initial Surface Conditions:

1. Etched and Cleaned (0 mg/dm²)
2. Etched, Cleaned, and Autoclaved 300 °C for 60 hours (2.6 mg/dm²)
3. Etched, Cleaned, and Autoclaved 400 °C for 14 hours (5.9 mg/dm²)

FIGURE 5. REACTOR POWER HISTOGRAM FROM JUNE 1967 TO JUNE 1968 (CONSUMERS BIG ROCK POINT REACTOR)

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 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 4 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. In addition, the corrosion rate of the Zr-Cr alloy can be compared directly with those of the Zircaloy-2 alloy.

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 inter-

mediate spacer positions. Figure 6 shows typical crud deposition at the spacer location of rod FR4-8. "Pimple-like" protuberances were found on all of the Zr-Cr clad rods in the area of the intermediate 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 7.

Rods FR4-3, FR4-4, FR4-8, FR4-9, and FR4-14 have been gamma scanned. Figure 8 shows the gamma scan of rod FR4-14 (one of the highest powered rods) and is typical of the gamma scans of the other four rods. The peak power position in all of the rods occurred just above the middle intermediate spacer positions. Note the depressions on the gamma scans at the axial positions corresponding to the spacer locations.

Full-length dimensional profiles (longitudinal traverse with rotation), before crud removal, have been performed on rods FR4-4, FR4-8, and FR4-14. Figure 9 shows the dimensional profile of rod FR4-14 and is typical of the profiles of the other two rods. The most prominent feature of the profiles is the sharp diameter increases at the spacer locations caused by localized crud buildups. 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,

TABLE 4

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

Rods Removed from Assembly B-1 After ~ 8,000 MWd/tU Average Burnup (7,900 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 60 h (2.6 mg/dm²)
3. Etched, Cleaned, and Autoclaved 400°C for 14 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 6. 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 7. SUPERFICIAL CRUD SPALLING ON ROD FR 4-14

FIGURE 8. GAMMA SCAN OF ROD FR4-14

FIGURE 9. DIMENSIONAL PROFILE OF ROD FR 4-14 BEFORE REMOVAL OF CRUD

the sharp jogs in the profile trace indicates the spalling is superficial and is only about 0.002 inch deep.

The sheared ends and fission gas puncture holes on rods FR2-5, FR2-6, FR2-8, FR4-4, FR4-8, and FR4-14 were filled with epoxy, and the rods were soaked in hot HCl to remove the crud. After removal of the crud rods FR4-4, FR4-8, and FR4-14 were reprofiled. Figure 10 is the profile of rod FR4-14 after crud removal. The spurious peaks were caused by epoxy or dirt on the surface of the rod. There is more ovality in the center region of the rods than there is at the ends of the rods. At the lower intermediate spacer location there is an apparent slight increase in the diameter of the rod. The profiles appeared as expected with no indications of bulging or impending failure.

After removal of the crud, the six rods were thoroughly examined. The end-plug welds were examined in detail. It was possible to see some difference in surface color at the welds on all of the rods. The heat-affected zones were slightly lighter in color on both the Zr-2 rods and the Zr-Cr rods. The Zr-2 and Zr-Cr clad rods were comparable in color at the top end plug welds. The Zr-Cr clad rods were darker in color at the bottom weld heat affected zone than the Zr-2 clad rods. This effect is shown in Figure 11. No obvious difference was noted in the color of the weld heat affected zone of the rods with different surface preparations.

The full length of each of the six initially removed rods was thoroughly examined. Outside of the spacer locations both the Zr-Cr and Zr-2 clad rods had black, lustrous adherent oxide (ZrO_2) films with no indications of post-transition corrosion. Figure 12 shows surface appearance of the Zr-Cr clad rods in the area of highest heat flux just above the middle intermediate spacer position. The Zr-Cr clad rods were comparable to the Zr-2

clad rods. No particular difference in surface appearance could be seen between the rods with different initial surface pretreatments.

The most interesting observation was the surface condition of the rods at the spacer locations. Both the Zr-Cr clad rods and the Zr-2 clad rods exhibited post-transition corrosion (in the form of white oxide) at the spacer locations. The Zr-2 clad rods exhibited a patch-like area of white oxide at the spacer spring contact areas. The Zr-Cr clad rods had similar patches at the spacer spring contact areas but to a much lesser degree than the Zr-2 clad rods. However, the Zr-Cr clad rods exhibited small localized areas of corrosion surrounding the spacer spring contact points. These small localized areas of corrosion apparently caused the "pimply" effect seen in the initial examination (before crud removal). Also, the apparent increase in diameter of the Zr-Cr at the lower intermediate spacer location on the profiles is apparently caused by the oxide buildup.

Figures 13, 14, and 15 illustrate the lower and upper intermediate spacer locations of the Zr-Cr clad rods and the Zr-2 clad rods. Several interesting observations can be made. The Zr-Cr clad rods appear to have more white corrosion products at the lower-intermediate spacer position than at the upper-intermediate spacer position. In contrast the Zr-2 clad rods exhibited less white corrosion products at the lower-intermediate spacer position than at the upper-intermediate spacer position. Also qualitatively it appears that the as-etched surface pretreatment rods exhibit less corrosion at the spacer locations than the 400°C autoclaved rods for both the Zr-Cr alloy and the Zircaloy-2 alloy. The severity of the corrosion attack at the middle-intermediate spacer location appeared to be between the two extremes of the attack at the upper and lower intermediate spacer locations. Figure 16 shows the middle spacer location

FIGURE 10. DIMENSIONAL PROFILE OF ROD FR 4-14 AFTER REMOVAL OF CRUD

BOTTOM END PLUG

TOP END PLUG

Zr-2
FR2-5
AS ETCHED

BOTTOM END PLUG

TOP END PLUG

Zr-Cr
FR4-4
AS ETCHED

FIGURE 11. PHOTOGRAPHS OF BOTTOM AND TOP END PLUG WELDS ON RODS
FR2-5 AND FR4-4

FIGURE 12. SURFACE APPEARANCE OF Zr-Cr CLAD RODS IN HIGHEST HEAT-FLUX AREA

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR4-4
Zr-Cr
AS ETCHED

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR2-5
ZR-2
AS ETCHED

FIGURE 13. PHOTOGRAPHS OF SPACER LOCATIONS ON RODS FR2-5 AND FR4-4

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR4-8
Zr-Cr
300°C PREOXIDIZED

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR2-6
Zr-2
300°C PREOXIDIZED

FIGURE 14. PHOTOGRAPHS OF SPACER LOCATIONS ON RODS FR2-6 AND FR4-8

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR4-14
Zr-Cr
400°C
PREOXIDATION

LOWER-INTERMEDIATE SPACER LOCATION

UPPER-INTERMEDIATE SPACER LOCATION

FR2-8
Zr-2
400°C
PREOXIDATION

FIGURE 15. PHOTOGRAPHS OF SPACER LOCATIONS ON RODS FR2-8 AND FR 4-14

FIGURE 16. PHOTOGRAPHS OF MIDDLE-INTERMEDIATE SPACER LOCATIONS ON RODS FR2-8 AND FR4-14

for rods FR2-8 and FR4-14. Unfortunately, the shear cut went directly through the spacer location. No evidence of white corrosion products were observed at either the upper or lower spacer location on both the Zr-Cr clad rods or the Zr-2 clad rods. The upper spacer location is in the area of the plenum region, and there is no heat flux in this area. The lower spacer location has minimal heat flux in that area.

The post-transition corrosion appears to be unique to the spacer locations for both the Zr-2 alloy and the Zr-Cr alloy. Outside of the spacer locations both alloys appear to have excellent corrosion performance. Since no evidence of post-transition corrosion was found at the upper or lower spacer locations (minimal heat flux areas), it appears that a prerequisite for the post-transition corrosion at the spacer locations is a relatively

high heat flux. However, neither alloy had the most severe attack at the middle-intermediate spacer location (area of very high heat flux). Another variable which may affect the in-reactor corrosion rates of these alloys is the oxygen content of the water and steam phases surrounding the rods. The sub-cooled water contains ~200 ppb oxygen while the steam phase contains about 25 ppm oxygen.

The post-irradiation examination is continuing. Metallographic specimens will be made from selected areas of rods FR2-5, FR2-6, FR2-8, FR4-4, FR4-8, and FR4-14, to determine oxide thickness and to perform hardness traverses. Metallographic specimens will be made from the dummy rods to obtain oxide thicknesses in order to determine the effects of heat transfer on the oxide film growth rates.

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