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**THE EFFECTS OF IRRADIATION AND ENVIRONMENT
ON THE MECHANICAL PROPERTIES
OF INCONEL-ZIRCALOY TRANSITION JOINTS**

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FEBRUARY, 1966

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THE EFFECTS OF IRRADIATION AND ENVIRONMENT ON THE MECHANICAL PROPERTIES OF INCONEL-ZIRCALOY TRANSITION JOINTS

INTRODUCTION

This report describes the effects of irradiation and environment on the mechanical properties of Zircaloy-2-Inconel 600 transition joints produced by both tandem extrusion and explosive welding. The test results will be used in the evaluation of transition joints between the Zircaloy-2 and Inconel 600 being considered for the BONUS Reactor advanced core fuel elements.

The diverse service conditions in nuclear reactors require the cladding or joining of dissimilar metals. Materials used in pressurized or boiling water and in superheated steam are of particular interest. Most reactor designers have made good use of zirconium alloys which have favorable neutron economy, good corrosion resistance, and acceptable mechanical properties. Advanced reactor designs requiring higher operating temperatures create the need for other materials. Consequently, it is not surprising that a number of successful techniques have been developed⁽¹⁻⁴⁾ for joining a variety of dissimilar metals including Zircaloy-2.

Designs being considered for the advanced cores of the BONUS Reactor incorporate both superheating and boiling in each fuel assembly.⁽⁵⁾ The first BONUS Reactor core has a centrally located boiling water region and a peripherally located super-heat

region. If the new BONUS fuel element utilizes a transition joint, the problem of satisfactorily joining Zircaloy clad boiler-surface to Inconel 600 headers or end-mounts must be solved. It is feasible to join Zircaloy-2 to stainless steel;^(1,6,7) however little data exists on joining Zircaloy-2 to Inconel 600.

An experimental program^(5,8,9) was begun to produce these transition tubes and determine the effects of irradiation and environment on the strength and integrity of the joints. The irradiation testing portion of this program was conducted by Battelle-Northwest under the Irradiation Effects on Reactor Structural Materials Program sponsored by the Fuels and Materials Development Branch of the Division of the Division of Reactor Development and Technology of the U. S. Atomic Energy Commission. The overall evaluation of the experimental tube joints is being conducted by Combustion Engineering, Inc., in Windsor, Connecticut, on the Advanced BONUS Core Development Program.

Because of the metallurgical incompatibility of Zircaloy-2 and Inconel 600, fusion welding of these two materials has not been successful. Direct welding of the material, either by the vacuum electron beam or tungsten inert gas methods results in brittle intermetallic and other second phase formations. Consequently, several processes were considered in an attempt

to produce a transition joint that would be acceptable for nuclear service. Two processes, tandem extrusion and explosive forming, were developed to produce tube joints for evaluation and testing.

Specimens from tubes produced by both processes were irradiated in an environment similar to that of the BONUS Reactor.⁽¹⁰⁻¹²⁾ The test conditions for the irradiated and unirradiated control specimens from both types of joints were equivalent; therefore, a comparison between the two different joints was made.

SUMMARY

Transition joints between Inconel 600 and Zircaloy-2 were irradiated at 550 °F to integrated exposures of 6.2×10^{20} nvt (> 1 MeV). The integrity of these joints is not impaired by exposures under conditions typical of the BONUS Reactor. Tensile tests at 550 °F and room temperature indicate the life of the transition joints will probably be limited by the Zircaloy-2 portion of the tube. A large decrease in ductility was observed after irradiation in the Zircaloy-2 portion of one type of transition joint. Hydrogen pickup was greatest in this particular joint and is believed to be partially responsible for the marked loss in ductility. Hydride gradients were observed near the transition joints. The mechanical property changes and hydrogen pickup are definitely influenced by the prior fabrication history of these transition tubes.

EXPERIMENTAL PROCEDURE

The developments leading to the explosive-welding technique used to produce the tubular transition joints used in these studies has been reported in detail.⁽¹³⁾ The tubular transition joints were formed by first machining a taper of the proper angle and surface finish on the ends of commercially fabricated Zircaloy-2 and Inconel 600 tubes and then placing the two pieces together into an assembly. The explosive-welding was accomplished by placing the assembled tube components into a mild steel die, evacuating the joint interface, and detonating an explosive charge to create the weld at the joint interface.

Two joint sizes were produced by tandem extrusion:⁽¹⁴⁾ one 3.316 in. ID by 0.100 in. wall, the other 1.604 in. OD by 1.338 in. ID. Specimens used in this investigation were from the smaller tubular section. Extrusion billets for each size were prepared by cleaning all components thoroughly and vacuum degassing the steel and Inconel 600 materials at 1600 °F. The Zircaloy-2 was etched in HNO_3 -HF solution and assembled with the Inconel 600 into the evacuated and welded billet assembly. Prior to extrusion, the Zircaloy-2 and Inconel 600 cores were 5.210 in. OD by 3.285 in. ID and 5.280 in. OD by 3.265 in. ID, respectively, for the larger transition joint. For the smaller joint these measurements were 2.860 in. OD by 1.391 in. ID and 2.848 in. OD by 1.385 in. ID, respectively. The extrusion

parameters for both size joints are summarized in Table I. An extrusion ratio of about 5 to 1 was used for both sizes of joints.

Test specimens were fabricated from experimental tubes produced by two different vendors and sent to Battelle-Northwest for irradiation testing. (8-10) Twenty-four specimens with the dimensions shown in Figure 1 were provided by each vendor. The approximate outside diameter and wall thickness is given for each type of tube. Because of the angle of the tapered interface, the length of the transition joints in the coextruded tubes were considerably longer than those in the explosively welded tubes. Therefore, the wall of the coextruded tubes was machined to about one-third of the original thickness to keep the

transition joint within the reduced gage section of the test specimen.

All of the test specimens were thoroughly examined and characterized before testing. Each specimen was stamped on both the Zircaloy-2 and Inconel 600 ends for positive identification. Hardness, weight, and dimension measurements were made on each sample. These results are given in Table II with comparable measurements on the same specimens after exposures in either the in-reactor or out-of-reactor loops. Pieces from both ends of each type of transition joint specimen were removed before testing for metallographic examination.

Tensile test grips with wedge type inserts were specifically designed and fabricated for testing both types of specimens. The test specimens had a

TABLE I. TANDEM EXTRUSION PROCESS
VARIABLES USED FOR FABRICATION OF
INCONEL 600 - ZIRCALOY-2 TRANSITION
JOINTS

	Nominal Tube Size	
	1 1/2 in.	3 1/2 in.
Extrusion Temperature, °F	1400	1400
Heating Time, hr	4	4
Maximum Force Required, tons	750	1690
Lubricant	Oil and Graphite	Oil and Graphite
Ram Speed, in./min	20	36
Container Temperature, °F	850	800
Extrusion Die Size, in.	1.730	3.775
Mandrel Size, in.	1.240-1.260	3.163-3.187
Average Tube OD with Cladding, in.	1.790	3.925
Average Tube Length with Cladding, in.	Not recorded	33
Method of Cladding Removal	Pickle in 50-50 HNO ₃ -H ₂ O	Pickle in 50-50 HNO ₃ -H ₂ O

FIGURE 2. BONUS TRANSITION JOINT IRRADIATION TEST SPECIMENS

different radius of curvature, so the knurled faces of the inserts were machined to an average radius of curvature for use on both types. The grips used for testing the unirradiated specimens are shown in Figure 2. The same jaws and inserts were used in a set of grips designed for the remote testing of the irradiated specimens. The remotely controlled grips utilize double acting air cylinders to open and close the jaws. Tensile tests were conducted at room temperature and at 550 °F for each condition of both types

of joints. All of the tensile tests were pulled at a crosshead speed of 0.005 in./min. A screw-on type extensometer was used in each test to measure the extension to a point beyond the 0.2% offset yield strength. The extension for the remainder of the curve was measured by crosshead travel. Figure 3 illustrates the manner in which the extensometer was attached to the test specimens.

The extensometer used for elevated temperature testing was of similar construction to the one shown in Figure 3;

TABLE II. SPECIMEN CHARACTERIZATION BEFORE AND AFTER LOOP EXPOSURES^(a)

Specimen Identification Assigned	Vendor's	Type of Joint ^(b)	Dimensions in Inches			Weights in Grams			Hardness Values Taken on Rockwell "A" Scale											
			Width	Thickness ^(c)	ΔThickness	Preexposure	Postexposure	Change ^(d)	Preexposure	Zry-2	Postexposure	Zry-2	Preexposure	Inconel	Postexposure	Inconel	Preexposure	Inconel	Postexposure	Inconel
175-1	1-1	CE	0.4376	0.0294-0.0328	0.0034	19.66159	19.7283	0.0667	52.4	52.1	51.9	59.0	59.5	58.7	53.5	64.0	64.0	66.5	67.9	68.6
175-2	3-1 SCJL	EW	0.4356	0.0853-0.0951	0.0098	59.18432	59.2225	0.0382	59.0	59.2	58.7	61.8	63.7	61.2	50.5	52.0	50.7	60.9	61.5	63.5
175-3	1-2	CE	0.4349	0.0329-0.0331	0.0002	20.08288	20.1114	0.0285	52.3	52.3	52.0	60.5	60.4	59.9	55.3	64.2	64.5	66.2	67.7	68.1
175-4	3-2 SCJL	EW	0.4368	0.0824-0.0855	0.0031	53.86965	53.9092	0.0395	59.0	57.5	59.0	61.9	63.2	65.5	53.0	51.3	49.0	60.8	61.0	62.7
175-5	1-3	CE	0.4349	0.0323-0.0328	0.0005	19.92549	19.9538	0.0283	52.0	51.9	51.8	59.5	60.2	60.4	64.4	64.4	64.4	65.4	67.8	67.9
175-6	3-3 SCJL	EW	0.4368	0.0870-0.0950	0.0080	58.99194	59.2714	(e)	59.3	59.2	59.2	62.7	63.5	62.8	52.3	50.8	49.5	59.9	62.5	61.8
176-1	1-4	CE	0.4379	0.0323-0.0329	0.0006	20.30766	20.3362	0.0285	52.0	52.0	49.3	57.8	57.8	57.7	64.6	64.5	64.3	67.5	66.9	67.6
176-2	3-4 SCJL	EW	0.4359	0.0860-0.0942	0.0082	58.31713	58.3438	0.0267	58.6	59.0	58.9	60.7	59.5	61.8	52.5	51.5	49.3	59.8	59.1	59.0
176-3	1-5	CE	0.4342	0.0320-0.0331	0.0011	19.95688	19.9833	0.0264	52.0	52.1	51.0	58.2	58.7	58.2	64.4	64.3	64.2	65.8	66.6	66.8
176-4	3-5 SCJL	EW	0.4372	0.0870-0.0947	0.0077	58.78734	58.8114	0.0241	58.8	59.5	58.7	61.9	62.0	61.5	52.5	51.3	49.5	58.4	59.0	58.5
176-5	1-6	CE	0.4340	0.0319-0.0322	0.0003	20.00651	20.0303	0.0238	52.0	52.0	52.0	57.8	57.4	57.8	64.5	64.5	64.5	65.6	65.3	66.0
176-6	3-6 SCJL	EW	0.4362	0.0890-0.0945	0.0055	59.10635	59.1284	0.0220	59.0	59.0	58.9	61.3	62.0	62.1	52.3	51.0	49.0	59.1	59.5	59.2
177-1	2-1	CE	0.4368	0.0320-0.0335	0.0015	20.18499	20.2023	0.0173	52.3	52.0	52.5	56.6	57.5	57.5	64.8	64.5	64.5	66.8	66.9	66.8
177-2	4-1 SCJL	EW	0.4362	0.0805-0.0856	0.0051	53.83082	53.8585	0.0187	58.8	58.9	59.0	61.5	61.9	61.9	52.2	52.1	50.5	57.2	58.1	59.1
177-3	2-2	CE	0.4365	0.0317-0.0330	0.0013	20.00320	20.0219	0.0187	52.2	52.0	50.0	58.2	58.5	58.5	64.2	64.0	64.1	66.3	66.2	66.2
177-4	4-2 SCJL	EW	0.4382	0.0884-0.0948	0.0064	58.48120	58.5006	0.0194	58.7	59.2	58.5	61.4	61.2	61.2	52.1	52.0	51.3	59.6	58.4	59.2
177-5	2-3	CE	0.4368	0.0312-0.0322	0.0010	19.74208	19.7604	0.0183	51.1	51.0	50.7	57.4	57.5	57.5	64.2	64.4	64.4	67.2	67.0	67.0
177-6	4-3 SCJL	EW	0.4360	0.0809-0.0865	0.0056	53.95875	53.9796	0.0208	58.6	57.7	59.2	62.2	62.2	62.2	52.2	50.0	48.9	57.9	58.5	58.2
178-1	2-4	CE	0.4389	0.0300-0.0310	0.0010	19.19908	19.2344	0.0353	52.1	52.2	52.0	60.1	59.4	59.8	64.4	64.0	64.0	66.2	67.5	68.0
178-2	4-4 SCJL	EW	0.4350	0.0805-0.0860	0.0055	53.76945	53.8297	0.0602	59.0	58.5	59.0	62.5	63.5	63.6	55.0	52.2	50.0	60.8	61.5	63.2
178-3	2-5	CE	0.4364	0.0299-0.0310	0.0011	19.25626	19.3019	0.0456	52.3	52.5	52.0	58.6	60.5	60.0	64.1	64.2	63.8	66.8	68.1	66.1
178-4	4-5 SCJL	EW	0.4378	0.0805-0.0862	0.0057	54.33945	54.3932	0.0537	59.0	58.7	59.0	61.5	62.5	62.0	55.0	53.0	51.0	61.0	61.5	63.0
178-5	2-6	CE	0.4365	0.0321-0.0330	0.0009	19.73794	19.7800	0.0421	52.3	52.2	51.2	60.3	60.5	60.4	64.5	64.6	64.3	67.1	68.5	68.2
178-6	4-6 SCJL	EW	0.4368	0.0805-0.0859	0.0054	53.93795	54.0033	0.0653	59.2	58.3	59.0	60.1	61.4	62.3	55.1	53.0	51.4	61.8	61.0	63.1
177-1X	-- ^(f)	CE	0.4375	0.0318-0.0334	0.0016	13.77422	13.7757	0.0015	50.0	49.9	50.2	-- ^(g)	--	--	62.5	63.3	63.8	--	--	--
177-2X	--	EW	0.4372	0.0961-0.0970	0.0009	53.80277	53.8046	0.0018	58.0	57.5	59.0	--	--	--	47.5	48.5	48.1	--	--	--
177-3X	--	CE	0.4375	0.0338-0.0346	0.0008	20.65418	20.6537	0.0005 ^(h)	46.6	48.5	49.2	49.5	50.1	50.1	63.6	63.8	63.5	63.5	63.5	--
177-4X	--	EW	0.4355	0.0961-0.0969	0.0008	53.93977	53.8138	≤	58.2	58.2	57.8	--	--	--	47.0	48.2	48.2	--	--	--
177-5X	--	CE	0.4382	0.0317-0.0325	0.0008	13.40903	13.4085	-0.0005	47.5	49.0	50.0	46.9	48.1	49.0	63.6	63.9	63.8	63.5	--	63.8
177-6X	--	EW	0.4388	0.0955-0.0968	0.0013	60.06993	60.0641	-0.0038	57.0	57.8	58.0	58.0	58.0	--	48.3	48.8	48.1	49.5	48.6	49.2
178-1X	--	CE	0.4369	0.0304-0.0330	0.0026	19.75131	19.7604	0.0091	48.9	49.9	50.3	50.4	50.0	52.0	62.1	63.1	63.6	62.5	63.0	62.5
178-2X	--	EW	0.4392	0.0960-0.0975	0.0015	60.53781	60.5412	0.0034	58.2	57.9	58.5	57.4	58.0	57.0	47.6	47.3	48.2	47.0	47.0	45.0
178-3X	--	CE	0.4368	0.0338-0.0355	0.0017	20.33183	20.3353	0.0035	48.5	50.0	51.1	50.0	50.0	51.1	63.5	63.6	64.1	63.0	63.0	63.0
178-4X	--	EW	0.4378	0.0820-0.0865	0.0045	53.78430	53.7876	0.0033	56.6	58.2	58.6	57.4	57.5	58.0	48.1	48.5	49.9	47.0	47.5	46.5
178-5X	--	CE	0.4369	0.0340-0.0352	0.0012	20.64620	20.5505	e	46.2	48.5	49.5	50.0	50.0	50.5	63.0	64.0	63.4	63.0	62.5	62.7
178-6X	--	EW	0.4341	0.0957-0.0968	0.0011	59.64277	59.6432	0.0004	57.7	58.8	59.2	57.2	57.5	58.0	47.0	49.0	49.8	48.0	47.6	47.4

^a Exposure history for both the G 7 and out-of-reactor loop as given in Table III.

^b CE designates a coextruded tube and EW represents an explosively welded tube.

^c Minimum and maximum thicknesses respectively, noted in gage length of specimen.

^d All changes reflect a weight gain unless otherwise noted.

^e This value is known to be incorrect and was not considered in overall evaluation.

^f No identification on these specimens.

^g Data were not obtained for these specimens prior to testing.

^h The minus sign indicates a weight loss.

FIGURE 2. TENSILE TEST GRIPS FOR CURVED SPECIMENS

FIGURE 3. ASSEMBLY FOR TESTING UNIRRADIATED SPECIMENS AT ROOM TEMPERATURE

but the transducer, leads, and springs were fabricated from materials suitable for service at 550 °F. A few changes were made in the design of the frame, and all of the aluminum parts were replaced by pieces fabricated from a titanium alloy. When either the room temperature or elevated temperature extensometer was used, a magnification ratio of 1000:1 was obtained. Both extensometers were the separable type that do not need to be removed during testing.

Four quadrants containing six specimens each were assembled and placed in various positions of the G-7 Hot Water Loop in the core of the Engineering Test Reactor (ETR). The G-7 Loop normally operates at 2000 psi and 540 °F when the ETR is operating at full power (i.e., 175 MW). Each quadrant, such as those shown in Figure 4, contained three specimens from each

FIGURE 4. SPECIAL QUADRANTS FOR THE IRRADIATION OF INCONEL-ZIRCALOY TRANSITION JOINTS

type of joint. The total integrated exposure and exposure time for each quadrant are given in Table III. Neutron exposures were determined from activation analysis of dosimeters placed with each quadrant. These flux monitors were placed adjacent to the gage sections of the test specimens as shown in Figure 4. Each flux monitor contained iron, nickel, and titanium dosimeters for determining fast flux and an aluminum-cobalt alloy dosimeter for monitoring thermal flux. The flux characteristics of the G-7 Loop have been determined and reported elsewhere. (15)

Two additional quadrants were assembled and exposed in an out-of-reactor loop located at Battelle-Northwest. This hot water loop is programed to duplicate the operation of the G-7

Loop minus the neutron flux. The quadrants placed in the out-of-reactor loop were designated 177-X and 178-X and were exposed for approximately 725 and 2200 hr, respectively. The comparison of data from two quadrants exposed for the same time and temperature in both the G-7 and out-of-reactor loops provides a means of separating thermal effects from combined neutron and thermal effects. At the temperatures of interest, extended thermal exposures can affect the strength and ductility of Zircaloy-2. (16)

The hydrogen content was determined for the Zircaloy-2 portion of both types of test specimens in the as-received condition and after irradiation and loop exposures. Duplicate analyses were performed for each

TABLE III. SUMMARY OF IRRADIATION CONDITIONS FOR BONUS QUADRANTS EXPOSED IN THE ETR G-7 LOOP

<u>Quadrant</u>	<u>EIR Cycles of Exposure</u>	<u>Hanger Position</u>	<u>Neutron Flux, nv^(a)</u>	<u>Neutron Exposure, nvt^(a)</u>	<u>Time at Operating Temperature, hr</u>
175	67,68,69	F ₁ -G ₁	1.0×10^{14}	6.2×10^{20}	> 1765
176	67,68	A ₁ -B ₁	4.2×10^{13}	1.9×10^{20}	> 1300
177 ^(b)	67	B ₂ -C ₂	5.5×10^{13}	1.4×10^{20}	> 725
178 ^(b)	67,68,69, 70	(B ₃ -C ₃) ^(c) (F ₁ ³ -G ₁ ³)	6.8×10^{13}	5.7×10^{20}	> 2200

- (a) Both the flux and exposure values are for neutrons having energies greater than 1 MeV.
- (b) Normal operating temperature for the EIR G-7 Loop is 540 °F. The out-of-reactor loop exposures for quadrants 177-X and 178-X were determined from the same time at temperature exposure received by the in-reactor quadrants 177 and 178, respectively. Therefore, the exposure for matching quadrants (i.e., 177 and 177-X) are the same.
- (c) Quadrant 178 was moved to position (F₁-G₁) for exposure during Cycle 70.

specimen examined. In several instances duplicate analyses were performed by two different laboratory groups on the same specimens. The results of these analyses are given in Table IV.

Both types of transition joints were examined metallographically before and after irradiation with particular emphasis on the bonded region. Sections of both the Inconel 600 and the Zircaloy-2 away from the bonded area

TABLE IV. HYDROGEN CONTENT OF THE
ZIRCALOY-2 PORTION OF THE BONUS TRANSI-
TION JOINT SPECIMENS

Specimen Identification	Exposure Location	Exposure Time, hr	Integrated Neutron Exposure, nvt >1 MeV	H ₂ - ppm	
				EW	CE
1	As-Received	0	0		<10,10 (4.8,5.1) (a)
2	As-Received	0	0	<10,10 (2.0,6.5)	
3	As-Received	0	0		<10,10
4	As-Received	0	0	<10,10	
177-1X	X-Reactor Loop	725	0		6.0,8.2
177-2X	X-Reactor Loop	725	0	2.6,3.9	
178-1X	X-Reactor Loop	2200	0		21,22 (22,22) (a)
178-2X	X-Reactor Loop	2200	0	21,23 (20,22)	
177-1	G-7 Loop	725	1.4×10^{20}		63,79
177-2	G-7 Loop	725	1.4×10^{20}	35,36	
176-1	G-7 Loop	1300	1.9×10^{20}		49,54
176-2	G-7 Loop	1300	1.9×10^{20}	23,31	
176-4	G-7 Loop	1300	1.9×10^{20}	17	
176-5	G-7 Loop	1300	1.9×10^{20}		114
175-1	G-7 Loop	1765	6.2×10^{20}		108,119 (105,118)
175-1	G-7 Loop	1765	6.2×10^{20}		92,95
175-2	G-7 Loop	1765	6.2×10^{20}	47,61 (50,50) (b)	
178-1	G-7 Loop	2200	5.7×10^{20}		104,109 (82,98) (b)
178-1	G-7 Loop	2200	5.7×10^{20}		114
178-2	G-7 Loop	2200	5.7×10^{20}	44,46 (40,43) (b)	
178-3	G-7 Loop	2200	5.7×10^{20}		(95,98) (b)

(a) The values in parenthesis were determined by the Battelle-Northwest Chemistry Analytical Laboratory and all other values were determined in the Battelle-Northwest Radiometallurgy Laboratory.

(b) These values were determined from samples taken at the specimen break (i.e., near the transition joint). All other values were taken from samples located in the tab of the tensile specimens, away from the break and transition joint.

were also examined for comparison. Typical metallographic features of these materials are shown in Figures 5 through 8. The significance of certain of these features is discussed in a later section.

RESULTS AND DISCUSSION

Tensile tests were conducted at room temperature and at 550 °F on the irradiated, as-received, and control specimens exposed in the out-of-reactor loop. Extensometers with 2 in. gage lengths were centered on each specimen so that all of the transition joint was located within the 2 in. gage section. Strain was measured with an extensometer until either the 0.2% offset yield strength was determined or the specimen failed.

After a few tests it was apparent that all of the failures in either the irradiated or unirradiated specimens would most likely occur in the Zircaloy-2 end of the specimen. Therefore, the "effective" gage length of the test specimens was really the Zircaloy-2 portion of the actual gage section located beyond the transition joint, neglecting elastic strains in the rest of the gage section. Consequently, the strain values measured represent an elongation over the 2.5 or 2.75 in. specimen gage section and do not reflect the normal elongation usually observed in Zircaloy-2.

The results obtained from tensile tests on each type of specimen are given in Table V. All of the specimens tested failed primarily through the Zircaloy-2 portion of the specimen. In one instance a specimen in the as-fabricated condition sheared along a bond interface while the Zircaloy-2

portion of the sample was necking to failure. One of the irradiated specimens showed a failure that occurred across the bond interface near the Zircaloy-2 end. The other 47 specimens tested failed entirely through the Zircaloy-2 portion of the specimen, outside of the transition joint.

In the as-received condition the coextruded specimens have less strength and greater ductility than the explosive-welded specimens for both test temperatures. Exposures in the out-of-reactor loop for 2200 hr at temperatures greater than 540 °F had very little effect on the room temperature or 550 °F tensile properties. However, after nominal exposures of between 1.4 and 6.2×10^{20} nvt (>1 MeV) the tensile properties of the coextruded specimens changed quite drastically. Changes in properties are illustrated in Figures 9 and 10. Figure 9 shows the reduction in both strength and ductility of the coextruded specimens at both room temperature and 550 °F after irradiation. Figure 10 shows the increase in strength and decrease in ductility for the explosive-welded specimens irradiated and tested under the same conditions.

A slight reduction in the strength of both types of transition joint was observed after exposure in the out-of-reactor loop. However, the corresponding increase in ductility normally associated with cold work recovery was not observed. Instead, ductility was slightly decreased at both test temperatures with increasing exposure in the out-of-reactor loop. The greatest strength and ductility

Inconel 600

Intermetallic Layer

(a) 1000X

Neg. No. 4651827C

Zircaloy-2

(b) 1000X

Neg. No. 4651827D

(c) 1000X

Neg. No. 4651827E

(d) 1000X

Neg. No. 4651827F

FIGURE 5. COEXTRUDED INCONEL-ZIRCALOY
JOINT INTERFACE SHOWING CONTINUOUS INTER-
METALLIC LAYER [(a) through (c) are as-
polished; (d) etched slightly to accentu-
ate the intermetallic layer]

(a) 1000X
Neg. No. 4651824A

(b) 1000X
Neg. No. 4651824B

(c) 1000X
Neg. No. 4651825E

FIGURE 6. EXPLOSIVE-WELDED INCONEL-ZIRCALOY JOINT INTERFACE SHOWING INTERMETALLIC POCKETS [as-polished surfaces of interfaces; (a) no intermetallic; (b) and (c) varying sized pockets]

(b) 100X
Neg. No. 4651825A

(c) 100X
Neg. No. 4651825B

(d) 100X
Neg. No. 4651825C

FIGURE 7. DISTRIBUTION OF INTERMETALLIC
POCKETS IN THE EXPLOSIVE-WELDED INCONEL-
ZIRCALOY JOINTS [(a) location of typical
intermetallic pockets; enlarged in (b)
through (d); all samples as-polished]

(a) 500X
Neg. No. 4651827A

(b) 500X
Neg. No. 4651827B

(c) 500X
Neg. No. 4651825D

FIGURE 8. TYPICAL TRANSITION INTERFACES
FOR COEXTRUDED AND EXPLOSIVE- WELDED
INCONEL-ZIRCALOY JOINTS [(a) and (b)
continuous intermetallic layer character-
istic of the coextruded joints; (c) typ-
ical explosive-welded interface, free
from intermetallic pockets; specimens
as-polished]

TABLE V. TENSILE PROPERTIES OF BONUS
TRANSITION JOINTS

Specimen Identification	Type of Joint	Specimen Condition	Test Temperature, °F	0.2% Offset Yield Strength ^(a)	Ultimate Tensile Strength ^(a)	Uniform Elongation ^(b)	Total Elongation ^(b)	Reduction of Area	Elastic Modulus ^(c)
H03483 ^(d) #1	CE ^(e)	As-Fab	RT	59.2	74.0	4.9	6.2	34.7	21.8
H03498 #2	CE	As-Fab	RT	68.2	74.5	2.5	5.5	45.2	23.0
H03497 #3	CE	As-Fab	RT	72.7	77.1	1.8	4.4	45.9	19.6
H03496 #4	CE	As-Fab	550	24.3	35.6	5.9	9.6	59.6	19.0
H03495 #5	CE	As-Fab	550	26.3	37.0	4.9	8.3	66.4	22.2
H03485 #6	CE	As-Fab	550	23.3	35.2	5.2	8.5	57.8	23.0
H03500 #1	EW ^(f)	As-Fab	RT	90.0	93.1	0.4	0.6	3.8	19.2
H03499 #2	EW	As-Fab	RT	81.3	87.3	1.1	3.4	37.2	14.3
H03494 #3	EW	As-Fab	550	59.5	59.8	0.4	3.4	50.6	14.3
H03493 #4	EW	As-Fab	550	59.2	59.5	0.4	3.8	49.4	15.5
H03484 #5	EW	As-Fab	550	51.8	52.3	0.6	3.4	48.4	14.3
-- #6 ^(g)	EW	As-Fab	--	--	--	--	--	--	--
H03487 177-1X	CE	725 hr ^(h)	RT	58.0	71.4	3.9	5.9	32.9	20.0
H03489 177-3X	CE	725 hr	550	39.0	39.2	0.2	2.4	59.6	15.9
H03491 177-5X	CE	725 hr	550	28.5	35.6	4.3	7.3	47.3	21.7
H03488 177-2X	EW	725 hr	RT	89.2	92.5	0.7	1.0	3.8	20.6
H03490 177-4X	EW	725 hr	572	57.4	57.6	0.7	3.1	46.2	16.1
H03492 177-	EW	725 hr	550	58.0	58.6	0.5	3.4	50.4	16.2
H03475 178-1X	CE	2200 hr ^(h)	RT	55.0	64.5	2.7	4.7	41.8	21.0
H03477 178-3X	CE	2200 hr	550	31.5	32.3	1.4	7.6	58.9	10.6
H03478 178-5X ⁽ⁱ⁾	CE	2200 hr	550	35.6	38.5	2.3	5.3	58.9	15.9
H03476 178-2X	EW	2200 hr	RT	79.8	84.5	1.0	1.1	6.8	17.9
-- 178-4X ^(j)	EW	2200 hr	550	--	--	--	--	55.0	--
H03479 178-6X	EW	2200	550	53.5	55.2	0.5	3.3	47.2	18.2
-- 175-1	CE	6.2 x 10 ²⁰ ^(k)	RT	--	23.6	--	--	--	17.6
-- 175-3	CE	6.2 x 10 ²⁰	550	--	28.4	--	--	--	22.1
-- 175-5	CE	6.2 x 10 ²⁰	550	--	29.5	--	--	--	19.1
H03474 175-2	EW	6.2 x 10 ²⁰	RT	113.0	114.9	0.4	1.4	16.4	18.1
H03472 175-4	EW	6.2 x 10 ²⁰	550	75.2	75.8	0.3	1.6	42.5	15.6
H03473 175-6	EW	6.2 x 10 ²⁰	550	72.9 ⁽¹⁾	72.9	0.3	1.8	50.9	15.9
-- 176-1	CE	1.9 x 10 ²⁰	RT	--	33.5	--	--	10.0	26.4
-- 176-3	CE	1.9 x 10 ²⁰	RT	--	47.6	--	--	10.0	27.1
-- 176-5	CE	1.9 x 10 ²⁰	550	--	48.8	--	~0.1	14.0	24.3
H03482 176-2	EW	1.9 x 10 ²⁰	RT	97.6	101.3	0.8	0.9	1.0	19.8
H03481 176-4	EW	1.9 x 10 ²⁰	550	67.1	67.6	0.3	2.4	41.3	16.8
H03480 176-6	EW	1.9 x 10 ²⁰	550	65.0	65.7	0.3	2.0	42.6	16.5
-- 177-1	CE	1.4 x 10 ²⁰	550	--	39.3	--	--	0.0	14.0
-- 177-3	CE	1.4 x 10 ²⁰	(m)	--	--	--	--	--	--
-- 177-5	CE	1.4 x 10 ²⁰	(m)	--	--	--	--	--	--
-- 177-2	EW	1.4 x 10 ²⁰	RT	--	84.9	--	--	5.3	17.3
H03486 177-4	EW	1.4 x 10 ²⁰	550	65.4	66.9	0.3	1.9	37.4	16.3
-- 177-6	EW	1.4 x 10 ²⁰	RT	--	75.2	--	--	16.1	17.3
-- 178-1	CE	5.7 x 10 ²⁰	RT	--	17.9	--	--	0.0	26.1
-- 178-3	CE	5.7 x 10 ²⁰	550	--	45.0	--	--	0.0	18.7
-- 178-5	CE	5.7 x 10 ²⁰	550	--	55.6	--	--	0.0	20.1
-- 178-2	EW	5.7 x 10 ²⁰	RT	--	116.7	--	~0.01	0.0	18.1
H03470 178-4	EW	5.7 x 10 ²⁰	550	76.1	76.5	0.3	1.9	46.7	14.9
H03471 178-6	EW	5.7 x 10 ²⁰	550	79.1	79.2	0.4	1.8	44.4	14.5

(a) Ultimate tensile and yield strength values are given in pounds per square inch x 10⁻³.

(b) Uniform elongation, total elongation, and reduction of area values are given in percent.

(c) Young's modulus values are given in pounds per square inch x 10⁻⁶.

(d) The designations in this column reference data from each test which is currently stored on the REM Computer Program.

(e) CE represents a coextruded joint.

(f) EW represents an explosive-welded joint.

(g) This specimen was used to develop testing methods and procedures in the hot-cell, and was not tested.

(h) All specimens in quadrants 177-X and 178-X were exposed in the out-of-reactor loop at approximately 550 °F for the time indicated.

(i) Specimen was loaded above proportional elastic limit before testing began.

(j) No chart was produced for this test due to a recorder malfunction.

(k) Approximate neutron exposures in nvt for neutrons having energies greater than 1 MeV.

(1) A drop-in-load yield point was observed.

(m) These specimens were broken during handling in the Radiometallurgy facilities.

FIGURE 9. THE CHANGE IN THE STRESS-STRAIN BEHAVIOR OF THE COEXTRUDED TRANSITION JOINTS DUE TO IRRADIATION (ENGINEERING STRESS STRAIN)

FIGURE 10. THE CHANGE IN THE STRESS-STRAIN BEHAVIOR OF THE EXPLOSIVE-WELDED TRANSITION JOINTS DUE TO IRRADIATION (ENGINEERING STRESS-STRAIN)

changes noted occurred for room temperature tests. For purposes of comparison, a graphical representation of both the irradiation and the control data are given in Figures 11 and 12. The apparent increase in the 550 °F yield strength with out-of-reactor loop exposure of the coextruded joints is not consistent with the trend observed and is considered normal scatter for these particular specimens.

Hardness readings were taken on both ends of each specimen before any exposure in either the G-7 or out-of-reactor loop. The readings were taken on the Rockwell "A" hardness scale for both the Inconel 600 and Zircaloy-2. Several hardness determinations were made on each specimen both before and after exposure, and the best average of these data is listed in Table VI. A series of hardness measurements was also made from the end of the specimen toward the transition joint at approximately 1/4 in. intervals. The hardness profile for both materials was essentially constant in most specimens. However, a few specimens did show a tendency for increasing hardness as the transition joint was approached.

A graphical comparison of the hardness data is given in Figure 13. Initially, the hardness of the Zircaloy-2 used in the explosive-welded joints was much greater than the Zircaloy-2 used in the coextruded joints. This difference persisted throughout the range of exposures employed, but was much smaller after 2200 hr in the reactor. For the Inconel, the coextruded specimens have a greater hardness initially than do the explosive-welded joints. This difference also

persists to the highest exposure employed. The greatest hardness changes with irradiation occurred in the Zircaloy-2 in the coextruded specimens and the Inconel 600 in the explosive-welded specimens.

This variation in hardness for both materials from each type of joint reflects the difference one should expect when a second fabrication history is superimposed upon another without an intermediate treatment. For the coextruded joints, the Zircaloy-2 was annealed during the 1400 °F extrusion and, therefore, had a much lower hardness than the explosive-welded Zircaloy-2. The Zircaloy-2 tubes used as starting material for the explosive-welded joints were probably tube reduced or drawn cold to about 30 to 35% reduction of area as indicated by the high hardness. There was very little hardening in the explosive forming process due to the very localized type of deformation involved. Although the coextrusions at 1400 °F resulted in a partial anneal of the Zircaloy-2, a considerable amount of residual warm work remained in the Inconel 600 judging from the relatively high hardness and the microstructure observed for this material in the as-fabricated condition.

The higher hardness values of the explosive-welded Zircaloy-2 as compared with the coextruded Zircaloy-2 are in agreement with the tensile properties shown in Table V. The increase in hardness due to neutron irradiation, however, will not explain the drastic loss of ductility in the coextruded joints. The hardness values of the coextruded Zircaloy-2 after the highest

FIGURE 11. THE EFFECTS OF IRRADIATION AND OUT-OF-REACTOR LOOP EXPOSURE ON THE STRENGTH OF INCONEL-ZIRCALOY TRANSITION JOINTS (EACH HORIZONTAL LINE ON THE BARS REPRESENTS ONE TEST)

FIGURE 12. THE EFFECTS OF IRRADIATION AND OUT-OF-REACTOR LOOP EXPOSURE ON THE DUCTILITY OF INCONEL-ZIRCALOY TRANSITION JOINTS (EACH HORIZONTAL LINE ON THE BARS REPRESENTS ONE TEST)

TABLE VI. THE EFFECTS OF IRRADIATION
AND EXPOSURE ON THE HARDNESS OF INCONEL
600 AND ZIRCALOY-2

Specimen Identification		As-Fabricated (RA)		Post-Irradiation and Out-of-Reactor Loop Exposure (RA)		Δ Hardness (RA)	
		Zry-2	Inconel	Zry-2	Inconel	Zry-2	Inconel
175-1-1	CE	52.1	63.8	59.0	67.6	+6.9	+3.8
175-3-1	EW	59.0	51.0	62.2	61.9	+3.2	+10.9
175-1-2	CE	52.2	64.7	60.3	67.3	+8.1	+2.6
175-3-2	EW	58.5	51.1	63.5	61.5	+5.0	+10.4
175-1-3	CE	51.9	64.4	60.0	67.0	+8.1	+2.6
175-3-3	EW	59.2	50.9	63.0	61.4	+3.8	+10.5
176-1-4	CE	51.9	64.5	57.8	67.3	+6.7	+2.8
176-3-4	EW	58.8	51.1	60.7	59.3	+1.9	+8.2
176-1-5	CE	51.7	64.3	58.4	66.4	+6.7	+2.1
176-3-5	EW	59.0	51.1	61.8	58.4	+2.8	+7.5
176-1-6	CE	52.0	64.5	57.7	65.6	+5.7	+1.1
176-3-6	EW	59.0	50.8	61.8	59.3	+2.8	+8.5
177-2-1	CE	52.3	64.6	57.2	66.8	+4.9	+2.2
177-4-1	EW	58.9	51.7	61.8	58.1	+2.9	+6.4
177-2-2	CE	52.1	64.1	58.4	66.2	+6.3	+2.1
177-4-2	EW	58.8	51.8	61.2	59.1	+2.4	+7.3
177-2-3	CE	50.9	64.3	57.4	67.0	+6.5	+2.7
177-4-3	EW	58.5	50.4	62.2	58.2	+3.7	+7.8
178-2-4	CE	52.1	64.2	57.9	67.2	+5.8	+3.0
178-4-4	EW	58.7	52.4	63.2	61.8	+4.5	+9.4
178-2-5	CE	52.3	64.0	59.7	67.0	+7.4	+3.0
178-4-5	EW	58.9	53.0	63.0	61.8	+4.1	+8.8
178-2-6	CE	51.9	64.5	60.4	67.9	+8.5	+3.4
178-4-6	EW	58.8	53.2	61.3	61.9	+2.5	+8.7
177-1X	CE	50.0	63.2	--	--	--	--
177-2X	EW	58.2	48.0	--	--	--	--
177-3X	CE	48.1	63.6	49.9	63.5	+1.8	-0.1
177-4X	EW	58.1	47.8	--	--	--	--
177-5X	CE	48.8	63.8	48.0	63.7	-0.8	-0.1
177-6X	EW	57.6	48.4	58.0	49.1	+0.4	+0.7
178-1X	CE	49.7	62.9	50.8	62.6	+1.1	-0.3
178-2X	EW	58.2	47.7	57.4	46.3	-0.8	-1.4
178-3X	CE	49.9	63.7	50.4	63.0	+0.5	-0.7
178-4X	EW	57.8	48.8	57.6	47.0	-0.2	-1.8
178-5X	CE	48.1	63.5	50.3	62.8	+2.2	-0.7
178-6X	EW	58.6	48.6	57.5	47.7	-1.1	-0.9

FIGURE 13. THE EFFECTS OF IRRADIATION AND OUT-OF-REACTOR LOOP EXPOSURE ON THE HARDNESS OF ZIRCALOY-2 AND INCONEL 600

neutron exposure in the G-7 Hot Water Loop was still less than the hardness of explosive-welded Zircaloy-2 in the as-received condition.

After an exposure of 1.9×10^{20} nvt (>1 MeV) the coextruded specimens failed in a very brittle manner with no sign of plastic flow at either room temperature or 550 °F. From Figure 9 it can also be seen that the breaking load became much lower for increasing irradiation exposure. Because of the residual cold work in the explosive-welded Zircaloy-2, very little strain hardenability was observed in even the

as-received material at either test temperature as shown in Figure 10. At the lowest neutron exposure a drop-in-load yield developed at the 550 °F test temperature. After an exposure of 6.2×10^{20} nvt (>1 MeV) a drop-in-load yield occurred in both room temperature and 550 °F tests. The specimens necked to failure after yielding, demonstrating the complete loss of strain hardenability.

Typical fracture surfaces and profiles for irradiated and control specimens from each type of joint are shown in Figures 14 through 16. In Figure 14

(a) Coextruded RT Test
Spec. No. 178-1 Neg. No. D-287

(b) Coextruded 550 °F Test
Spec. No. 178-3 Neg. No. D-289

(c) Explosive Welded RT Test
Spec. No. 178-2 Neg. No. D-288

(d) Explosive Welded 550 °F Test
Spec. No. 178-4 Neg. No. D-290

FIGURE 14. TYPICAL FRACTURE SURFACES OF
IRRADIATED BONUS SPECIMENS (~4.7X)

(a) Coextruded RT Test Spec. No. 178-1 Concave Neg. No. 16543

(b) Coextruded 550 °F Test Spec. No. 178-3 Concave Neg. No. 16547

(c) Explosive Welded RT Test Spec. No. 178-2 Convex Neg. No. 16546

(d) Explosive Welded 550 °F Test Spec. No. 178-4 Convex Neg. No. 16550

FIGURE 15. TYPICAL FRACTURE PROFILES OF
IRRADIATED BONUS SPECIMENS (~3/4X) (THE
SHORTEST PIECE OF THE SPECIMENS IS
ZIRCALOY-2)

Out-of-Reactor Quadrant 178-X

FIGURE 16. TYPICAL FRACTURE PROFILES OF BONUS SPECIMENS EXPOSED IN THE OUT-OF-REACTOR LOOP ($\sim 3/4$ X) (THE SHORT PIECE OF EACH SPECIMEN IS ZIRCALOY-2. THESE SPECIMENS SHOW THAT THE HIGHEST DUCTILITY IS OBSERVED IN THE 550 °F TESTS.)

it can be seen that at the highest neutron exposure the failure of the co-extruded specimens at either test temperature is the same. No macroscopic plastic flow was observed, and the fracture surfaces appear to be quite brittle in nature. The explosive-welded specimens in this same figure, however, show a strong effect of test temperature. At 550 °F the failure is extremely localized but quite ductile in nature. At room temperature the ductility is drastically reduced, but there is still evidence of both macroscopic and microscopic plastic flow. Small amounts of shear lip can be observed on both the inner and outer circumference of the explosive-welded specimen tested at room temperature.

The fracture profiles for the irradiated specimens are shown in Figure 15 and for the specimens exposed in the out-of-reactor loop in Figure 16. Both figures show that the location of the failures in all specimens from each type of joint is primarily in the Zircaloy-2 and is usually very near the transition joint itself. In the room temperature tests the low ductility is typified by the straight break perpendicular to the tensile axis. The more ductile failures at 550 °F result in a fracture trace which forms an angle of 30 to 45 ° with the tensile axis.

The low ductility observed in these tests cannot be explained by irradiation hardening alone. Therefore, metallographic examination and analytical tests were performed to determine the amount and distribution of hydride present before and after either exposure

in the out-of-reactor loop or irradiation. The results of the chemical analysis given in Table IV show that the hydrogen content in the as-received condition was nominal. After 2200 hr in the out-of-reactor loop, the total hydrogen content was still below 25 ppm which is acceptable.

The hydrogen pickup in the G-7 Hot Water Loop, however, occurred at a rate much greater than was observed in the out-of-reactor loop. The increase was approximately twice as high in the explosive-welded joints and about five times as great in the coextruded joints. Hydrogen levels between 100 and 125 ppm were observed in the coextruded joints after 1765 hr in the G-7 Loop. This phenomenon of accelerated corrosion and hydrogen pickup is not fully understood but has been experimentally observed and reported^(17,18) for other tests on Zircaloy-2 in the G-7 Loop under these same conditions. The accelerated rate is believed to result from the combined effects of high oxygen content in the loop water and neutron flux. The flux effect is observable in the results given in Table IV when Quadrant 175 is compared with Quadrant 178. Quadrant 175, which had the greater weight gain and hydrogen pickup, was exposed at a higher flux and had a higher integrated neutron exposure than did Quadrant 178, even though Quadrant 178 was exposed 435 hr longer.

The coextruded specimens have a larger surface-to-volume ratio than do the explosive-welded specimens, and would be expected to pick up hydrogen at a greater rate. This was verified in both the chemical and metallographic

results obtained. It was difficult, however, to obtain more than a qualitative agreement between chemical analysis and metallographic estimate of the hydrogen content of irradiated specimens. The standard etching techniques, which are normally used quite successfully for irradiated Zircaloy-2, were less than satisfactory for these materials. One of the main difficulties was the inability to control the rate of the etching process. The over-etching that occurred gives the microstructure the appearance of having a much higher hydride content than is actually present. Second phase material and grain boundaries were selectively attacked along with the hydride.

The hydride distribution in each type of joint after exposure in the out-of-reactor loop or irradiation in the G-7 Loop is shown in Figures 17 through 21.

Figures 17 and 18 illustrate the striking difference observed in the two joints after an exposure of 1.9×10^{20} nvt (>1 MeV). Both samples were prepared at the same time using the same metallographic technique. The specimens in Figures 17 and 18 were obtained from the fractured area of tensile samples. Portions of the fracture trace are shown in the upper sections of each photomicrograph. The voids present in the explosive-welded sample and not present in the coextruded one indicate that a greater amount of plastic flow occurred in the explosive-welded specimen. Using metallographic standards for estimating hydride content,⁽¹⁹⁾ it would appear that the total hydrogen content in the coextruded joints was several hundred ppm instead of the 50

to 100 ppm determined from chemical analysis. The results of the chemical and metallographic analyses of hydrogen content in the explosive-welded samples are in good agreement.

The hydride distributions after 2200 hr exposure in the out-of-reactor loop for both types of joints are shown in Figures 19 and 20. Both transverse and longitudinal sections were examined and are shown in each figure. Figure 19 shows that the hydride platelets in the explosive-welded sample are much larger and more specifically oriented along the tube axis than are the platelets in the coextruded tube shown in Figure 20. By comparing Figures 19 and 20 it is easy to see that the coextruded tubes contain the greater amount of hydride. The photomicrographs in Figures 19(a) and 20(a) show that a slight gradient in hydride content exists at the Zircaloy-2-Inconel 600 interface. The joint interface is shown in the upper portion of these micrographs, but is out of focus because of the difference in polishing and etching response of these two materials. Figure 19(b) shows a section of the explosive-welded joint nearer the center of the specimen where no gradient exists. Similar gradients were also observed in the irradiated samples. Slightly higher hydride concentrations were noted nearer the transition joints, but chemical analysis on samples very near the joint and at the end of the Zircaloy-2 portion of the tensile specimen gave approximately the same values. A gradient is also visible in Figure 21 where several photomicrographs were taken at various distances from the beginning of the Zircaloy-2 portion of the specimen along

(a) Neg. No. C 9095

(b) Neg. No. C 9094

(c) Neg. No. C 9096

Explosive Welded 176-4 250X

(a) Neg. No. C 9396

(b) Neg. No. C 9391

(c) Neg. No. C 9398

Coextruded 176-3 250X

FIGURE 17. HYDRIDE DISTRIBUTION IN
EXPLOSIVE-WELDED SAMPLES AFTER IRRADIATION
IN THE ETR G-7 LOOP

FIGURE 18. HYDRIDE DISTRIBUTION IN CO-
EXTRUDED SAMPLES AFTER IRRADIATION IN
THE ETR G-7 LOOP

(a)

Explosive Welded 178-2X Longitudinal
Neg. No. 4652095 A 250X

(b)

Explosive Welded 178-2X Longitudinal
Neg. No. 4652095 B 250X

(c)

Explosive Welded 178-2X Transverse
Neg. No. 465 2111 250X

FIGURE 19. HYDRIDE DISTRIBUTION IN
EXPLOSIVE-WELDED SAMPLES EXPOSED IN THE
OUT-OF-REACTOR LOOP

(a)

Coextruded 178-1X Longitudinal
Neg. No. 465 2094 250X

(b)

Coextruded 178-1X Transverse
Neg. No. 465 2110 250X

FIGURE 20. HYDRIDE DISTRIBUTION IN CO-
EXTRUDED SAMPLES EXPOSED IN THE OUT-OF-
REACTOR LOOP

(a) Coextruded 178-1
Neg. No. D-1982 250X

(b) Coextruded 178-1
Neg. No. D-1984 250X

(c) Coextruded 178-1
Neg. No. D-1985 250X

FIGURE 21. HYDRIDE DISTRIBUTION ALONG
THE ZIRCALOY-INCONEL TRANSITION JOINT
AFTER IRRADIATION IN THE ETR G-7 HOT
WATER LOOP

the transition joint. The coextruded specimen shown in Figure 21 was irradiated in the ETR G-7 Loop for 2200 hr and acquired a neutron exposure of 5.7×10^{20} nvt (>1 MeV). The photomicrograph in Figure 21(c) was taken in a section of the transition joint where the thickness of the Zircaloy-2 was about one-tenth of the total section thickness, while the section shown in Figure 21(a) is entirely Zircaloy-2 and was taken at the point of failure. The hydride concentration is much more dense in the fracture area than in the section where the Zircaloy-2 is only a small portion of the total section thickness.

The reason for the existence of a hydrogen gradient is not clear and further examination of this phenomena is outside the scope of this study. At least two possibilities are suggested, however, which may account for these observations. Accelerated hydriding is known to occur as a result of an applied chemical potential.⁽²⁰⁾ Although the potential difference between these two materials is small, the presence of the intermetallic compound at the joint interface could influence the potential. This could be especially true in the case of the coextruded joints which have a continuous intermetallic layer along the joint, as shown in Figures 5 and 8. The intermetallic compound formed in the explosive-welded joints is much more localized and is found in pockets of varying size as shown in Figures 6 and 7. Figure 7(a) shows that these pockets are formed about mid-way between the adjacent apexes of the scallops. The intermetallic compound is very brittle and fractures with little or no strain. It is also

readily dissolved by the etchants used in these examinations.

The second possibility which could cause a hydrogen gradient along the joint is the hydrogen present in the Inconel 600 tubes used as starting material and the subsequent diffusion of this hydrogen during the irradiation or exposure process. It is also possible that hydrogen might diffuse more easily through the Inconel 600 or the inter-metallic compound than through the Zircaloy-2. Since all of the specimens were exposed in the as-machined condition with relatively rough surfaces, the nature of the corrosion and hydride pickup is subject to much speculation.

Corrosion rates were approximated by weighing all specimens before and after exposure. Surface areas were determined for each type of sample and it was assumed that all of the weight gain was due to Zircaloy-2 corrosion. It was also assumed that the Zircaloy-2 surface area was equal to one-half of the total surface area. The resulting weight gains determined in this manner are given in Table VII. From the approximate weight gains determined, it seems reasonable that all of the hydrogen pickup could be accounted for by corrosion hydrogen.

Corrosion rates are also known to be influenced by the differences in metallographic texture. The typical metallographic appearance of each material from each type of tube is shown in Figures 22 and 23. The explosive-welded Zircaloy-2 in Figure 22(b) still retains the wrought texture acquired in the tube forming process. The coextruded Zircaloy-2, however, has equiaxed grains resulting from the hot extrusion

process at 1400 °F. Both materials have grain sizes between 0.01 and 0.016 which correspond to an ASTM grain size of between 9 and 11. The coextruded Zircaloy-2 appears to have a slightly smaller grain size. From the similarity exhibited, it would seem that the texture of these two Zircaloy-2 tubes would not contribute to the difference in the corrosion behavior observed.

The Inconel 600 was quite different in these two joints, however, as shown in Figure 23. The coextrusion process left the Inconel 600 in a highly wrought condition with elongated grains oriented along the extrusion direction. Figure 23(a) shows the coextruded transition joint with the Inconel 600 in the upper portion of the photomicrograph and the Zircaloy-2 in the lower portion. The Inconel 600 grains in the explosive-welded joints are equiaxed. The precipitates and inclusions are broken up and uniformly distributed in the coextruded samples, while some relatively large inclusions exist in the explosive-welded tubes. Since all of the transition joint failures occurred through the Zircaloy-2, the relative strength and ductility of the Inconel 600 in either joint was not determined. The mechanical properties of the Inconel 600 in each joint should not be the same, however, judging from the microstructure.

CONCLUSIONS AND RECOMMENDATIONS

Highly satisfactory transition joints and cladding components^(21,22) have been fabricated in a reproducible manner by a number of companies with the intention of using them for reactor service. The reliability of these transition joints between Zircaloy-2 and materials such as nickel-base alloys and stainless steels

TABLE VII. WEIGHT GAIN OF BONUS TRANSI-
TION JOINT SAMPLES

Specimen Identification		Original Weight, gm	Final Weight, gm	Net Change, mg	mg/dm ²
<u>In-Reactor</u>					
175-1-1	#1	19.66159	19.7283	+66.70	16.8
175-3-1	#2	59.18432	59.2225	+38.20	9.3
175-1-2	#3	20.08288	20.1114	+28.5	7.2
175-3-2	#4	53.86965	53.9092	+39.5	9.7
175-1-3	#5	19.92549	19.9538	+28.3	7.1
175-3-3	#6	58.99194	59.2714	+279.5	68.3
176-1-4	#1	20.30766	20.3362	+28.54	7.2
176-3-4	#2	58.31713	58.3438	+26.67	6.5
176-1-5	#3	19.95688	19.9833	+26.42	5.9
176-3-5	#4	58.78734	58.8114	+24.06	5.9
176-1-6	#5	20.00651	20.0303	+23.79	6.0
176-3-6	#6	59.10635	59.1284	+22.05	5.4
177-2-1	#1	20.18499	20.2023	+17.31	4.4
177-4-1	#2	53.83982	53.8585	+18.68	4.6
177-2-2	#3	20.00320	20.0219	+18.70	4.7
177-4-2	#4	58.48.20	58.5006	+19.40	4.7
177-2-3	#5	19.74208	19.7604	+18.32	4.6
177-4-3	#6	53.95875	53.9796	+20.85	5.1
178-2-4	#1	19.19908	19.2344	+35.3	8.9
178-4-4	#2	53.76945	53.8297	+60.2	14.7
178-2-5	#3	19.25626	19.3019	+45.6	11.5
178-4-5	#4	54.33945	54.3932	+53.7	13.1
178-2-6	#5	19.73794	19.7800	+42.1	10.6
178-4-6	#6	53.93795	54.0033	+65.3	16.0
<u>Out-of-Reactor</u>					
177-1X		19.77422	19.7757	+ 1.5	0.4
177-2X		59.80277	59.8046	+ 1.8	0.4
177-3X		20.65418	20.65365	- .5	--
177-4X		59.93977	59.8138	-12.6	--
177-5X		19.40903	19.40847	- .56	--
177-6X		60.06993	60.06412	- .58	--
178-1X		19.75131	19.76039	+ 9.08	2.3
178-2X		60.53781	60.54117	+ 3.36	0.8
178-3X		20.33183	20.33525	+ 3.42	0.9
178-4X		53.78430	53.78755	+ 3.25	0.8
178-5X		20.64620	20.55048	-95.72	--
178-6X		59.64277	59.64319	+ .4	0.1

(a) Coextruded Longitudinal
Neg. No. 465572C 250X

(b) Explosive Welded Longitudinal
Neg. No. 465577C 250X

FIGURE 22. METALLOGRAPHY OF THE
ZIRCALOY-2 FROM THE BONUS TRANSITION
JOINT SPECIMENS IN THE AS-RECEIVED
CONDITION

(a) Coextruded Longitudinal 100X
Neg. No. 465575C

(b) Explosive-Welded Longitudinal
100 X Neg. No. 465579A

(c) Coextruded Longitudinal 500X
Neg. No. 465575F

(d) Explosive-Welded Longitudinal
500X Neg. No. 465579B

FIGURE 23. METALLOGRAPHY OF THE INCONEL
600 FROM THE BONUS TRANSITION JOINT SPE-
CIMENS IN THE AS-RECEIVED CONDITION

must be demonstrated under service conditions and exposures typical of those expected in actual service. Due to the limited information concerning the exact nature of hydrogen uptake in Zircaloy-2 when coupled with dissimilar metals, it is recommended that this phenomenon be studied in greater detail before subscribing to the acceptability of a particular type of transition joint. It is also recommended that a more complete characterization of starting materials and fabrication steps encountered before the fabrication of the transition joints be obtained, since in joining methods such as explosive welding, the prior fabrication history of the starting materials can markedly influence the final mechanical properties of the transition joints.

As was stated earlier, this study was a cooperative effort coordinated through the Fuels and Materials Development Branch of the Division of Reactor Development and Technology of the U. S. Atomic Energy Commission. The final responsibility for the evaluation of the BONUS transition joints rests with Combustion Engineering, Inc. However, from the irradiation testing program conducted at Battelle-Northwest the following conclusions were drawn:

- Neutron irradiation at approximately 290 °C in the G-7 Hot Water Loop to an integrated exposure of 6.2×10^{20} nvt (>1 MeV) does not impair the integrity of the transition joints. However, irradiation hardening in both the Inconel 600 and Zircaloy-2 portions of the samples results in a decrease in ductility and an increase in strength in each material of both types of joints.

The transition joints themselves were completely sound in the as-received condition and after exposure in either the ETR G-7 Hot Water Loop or the out-of-reactor control loop. Exposures at temperatures near 500 °F for 2200 hr did not result in the formation of additional intermetallic compound at the bond interface.

- Every sample tested failed primarily through the Zircaloy-2 portion of the specimen. Only two or three samples failed within the transition joint area, and these failures were initiated in the Zircaloy-2 portion. Consequently, the life of the transition joints will probably be limited by the Zircaloy-2 portion of the tubes.
- The decrease in ductility in the Zircaloy-2 portion of each type of joint was considerably greater than was expected from the irradiation conditions. This is particularly true of the coextruded joints which exhibited the greater ductility in the as-received condition. It is felt that this accelerated decrease in ductility is strongly affected by the prior fabrication history of these joints including the fabrication of the Zircaloy-2 tube components prior to the joining process.
- Hydrogen content is believed to be at least partially responsible for the drastic reduction in ductility in the coextruded joints. Hydrogen content is not considered to be a factor in reducing the ductility in the explosive-welded joints, in the amounts observed.
- Hydrogen pickup is influenced by fabrication history as well as surface condition and specimen geometry.

The presence of a hydride gradient in certain areas of the transition joint is an indication that the hydrogen absorption may be occurring by several different processes. A study of these processes is outside the scope of this program.

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