OAK RIDGE NATIONAL LABORATORY
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RESULTS OF TENSILE TESTS PERFORMED ON MATERIALS EXPOSED IN THE HOMOGENEOUS REACTOR EXPERIMENT NO. 2 BLANKET AND LOW-FLUX-CORE REGION

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METALS AND CERAMICS DIVISION

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SUMMARY

Tensile specimens of various alloys of zirconium and titanium and a variety of austenitic stainless steels and Incoloy were exposed in the blanket and low-flux core regions of the Homogeneous Reactor Experiment (HRE) No. 2. The purpose of this exposure was to determine the effect of HRE-2 environment on mechanical properties and to provide corrosion data.

Specimens were exposed in the low-flux core region and in two blanket loadings. While the specimens were in the low-flux core region, a maximum estimated fast (> 1 Mev) neutron dose of 0.8×10^{16} nvt was accumulated. The calculated doses in the first and second blanket loadings were 1.3×10^{17} and 2.6×10^{18} nvt, respectively.

Only insignificant changes were observed in the mechanical properties as a result of the smallest exposure (i.e., 0.8×10^{16} nvt). At 1.3×10^{17} nvt, some small decreases in ductility for Zircaloy-2, type 347 stainless steel, and A-40 titanium were observed. At 2.6×10^{18} nvt, small but definite changes in some of the mechanical properties were observed in all of the materials exposed, usually as small increases in the tensile and yield strengths and slight reduction in elongation, with the greatest changes occurring in the A-40 titanium. Negligible corrosion was observed during the three exposures.

INTRODUCTION

Tensile tests have been conducted on various alloys of zirconium and titanium and on a variety of austenitic stainless steels and Incoloy.

The materials included in this study are Zircaloy-2 (both cross-rolled commercial material and straight-rolled HRP material), Zircaloy-2 weldments, two titanium alloys (A-40 and A-110 AT), stainless steels (types 347, 347 weldments, type 304L, and type 309 SCb), and Incoloy.

Specimens of these alloys were exposed to the environment of a low-flux core position and to the blanket region of HRE-2. These tests were part of a program to determine the effects of the actual homogeneous reactor environment, during reactor operation, on the corrosion rate and mechanical properties of various materials of interest. Specimens have been exposed for various intervals in the center of the core, in the outlet pipe (i.e., low neutron flux but same solutions), and in the blanket. This report will cover only specimens exposed during loading No. 1 (runs 13 and 14) in which specimens were in the blanket, and loading No. 2 (run 17) in which specimens were placed in the blanket and low-flux-core region. Information related to the effect of the environment on the corrosion resistance of these materials has been reported by other sections.[1] The theoretical treatment of the effect of bombarding metals with fast neutrons has also been discussed by others[2,3] and will not be covered in this report.

The two groups of specimens (i.e., loadings Nos. 1 and 2) were exposed to different solutions. The first loading was exposed during run Nos. 13 and 14 from December 30, 1957, to April 8, 1958.[1] During this period, the reactor accumulated 375 Mwhr. The temperature was between 100 and 250°C for 227 hr and was greater than 250°C for 752 hr. Except for a few minutes near the end of the run, the blanket solution was D_2O (without added oxygen). Run No. 14 was terminated by the formation

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[1] A. R. Olsen, HRE-2 Corrosion Specimens - Blanket Region of Pressure Vessel (Loading No. 1) - Weight Data and Scale Analysis, ORNL-CF-58-10-83.

[2] G. J. Dienes and G. H. Vineyard, Radiation Effects in Solids, Interscience Publishers, New York, 1957.

[3] D. S. Billington, McGraw-Hill Encyclopedia of Science and Technology, Vol 11, pp 223–25, McGraw-Hill, New York, 1960.

of a hole in the core tank which exposed the blanket specimens to dilute uranyl sulfate solution during the shutdown period. The maximum estimated integrated fast (> 1 Mev) neutron dose was 1.3×10^{17} nvt. The flux calculations are believed to be accurate to within ± 25%.

The second loading was exposed from July 19, 1958, to September 22, 1958, during run No. 17. During this period, the reactor operated with the hole present in the core vessel so that the blanket specimens were exposed to fissioning uranyl sulfate. During run No. 17, the reactor developed 2426 Mwhr, 40% of which were attributed to the blanket. The temperature was below 250°C for 195 hr and between 250 and 300°C for 890 hr. The maximum estimated fast (> 1 Mev) integrated dose was 2.6×10^{18} nvt. During this run, specimens in the low-flux core region of the outlet pipe were exposed to a maximum estimated fast dose of 0.8×10^{16} nvt. Additional information on the reactor operation during these periods has been reported elsewhere.[4–6]

Subsize tensile and impact specimens were included in the overall program but only the results of the tensile tests are reported in this memorandum. A detailed drawing of the tensile specimen is shown in Fig. 1. The corrosion rate during exposure was low enough so that all of the specimens were still within the ± 0.001-in. machining tolerance when measured just before testing. All of the specimens except those of commercial Zircaloy-2 were machined so that the longitudinal axis of the specimen was parallel to the rolling direction. The Zircaloy-2 specimens were machined from scrap material remaining from the HRE-2 core-tank development and were not always identifiable as to the rolling direction. However, previous tensile tests conducted on this material revealed very little difference in the longitudinal or transverse direction of this plate.

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[4] J. R. Engel _et al._, _Summary of HRE-2 Run 13 (Initial Power Operation)_, ORNL-CF-58-10-115.

[5] J. R. Engel _et al._, _Summary of HRT Run 14_, ORNL-CF-59-6-96.

[6] P. N. Haubenreich _et al._, _Summary of HRT Run 17_, ORNL-CF-60-8-151.

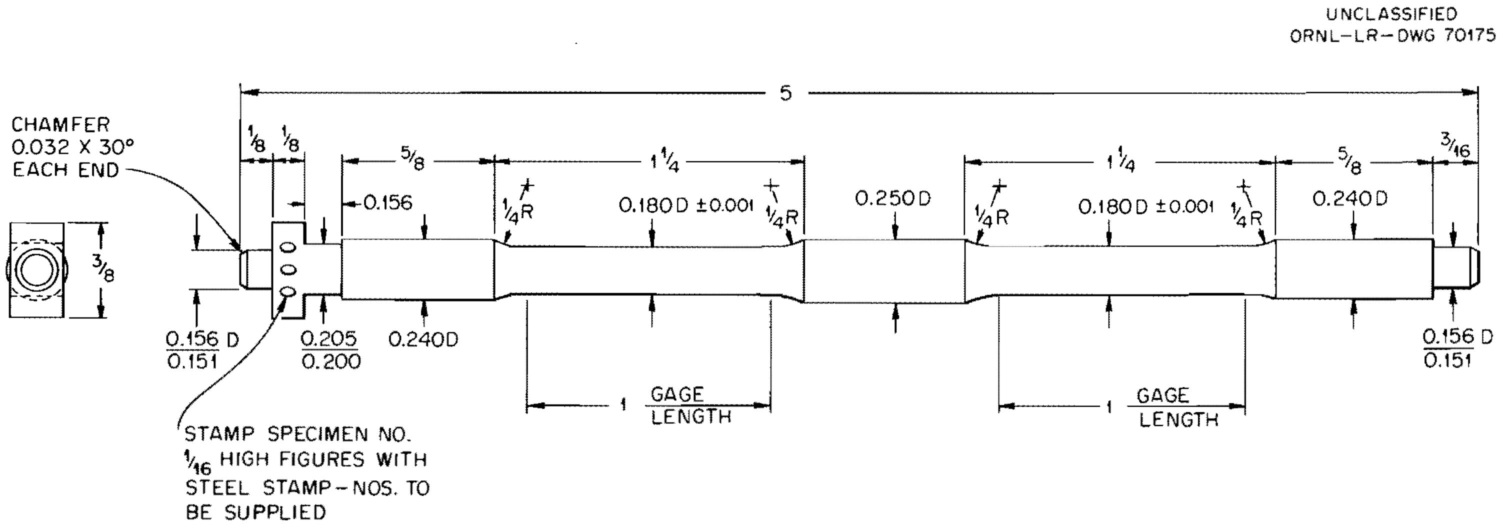

Fig. 1. Subsize Tensile Specimen Used in HRE-2 Exposure Studies.

The unirradiated and irradiated tensile specimens were measured and broken remotely in the hot-cell facility of the Solid State Division. All of the tests were conducted at room temperature, using a constant crosshead rate of 0.05 in./min. Stress-strain curves were obtained from the movement of the crosshead and, in some cases, extensometer curves were also obtained.

The purpose of the extensometer measurements was to check the accuracy of the elongation values obtained from the crosshead motion. The shoulders of the tensile specimens were not threaded (to facilitate in-cell testing); consequently, the specimens were pulled using chuck jaws. As the chuck jaws bite into the tensile specimens, this motion is recorded as elongation when monitoring is performed at the crossheads. In the case of the extensometer, the motion that is recorded is that which takes place only within the gage length. It would be expected then that motion recorded from the crossheads would yield higher elongation values. This was observed in the case of type 347 stainless steel but, in the case of crystal-bar zirconium (cross rolled), Zircaloy-2 code B, and A-40 titanium, the differences were small.

DISCUSSION AND RESULTS

Shown in Tables 1–4 are the average values of the tensile properties for the various groups of metals studied. Also shown are the maximum and minimum values, the percentage change of the mean value, the standard deviation, and whether or not a statistically significant change (at the 5% significance level) has taken place. The procedure used to determine the standard deviation and whether or not a statistically significant change has occurred are explained in the Appendix.

The values for uniform elongation shown in Tables 2 and 3 are taken from the recorded stress-strain curves and are described as the change in gage length per unit length between the 0.2% yield stress and the point of maximum load (i.e., the ultimate tensile strength). The necking elongation is obtained from the stress-strain curves and is the change in gage length per unit length from the maximum load to the point of fracture.

Table 1. Yield and Tensile Strengths of Various Materials as Influenced by Exposure in the HRE-2

			0.2% Offset Yield Strength					Tensile Strength				
Material	Irradiated or Unirradiated	No. of Breaks Averaged	0.2% Yield Strength (mean) (psi)	Max and Min Values, 0.2% Yield Strength (psi)	Percentage Change of Mean as Result of Exposure	Standard Deviation (psi)	Significant Change	Ultimate Tensile Strength (mean) (psi)	Max and Min Tensile Strength (psi)	Percentage Change of Mean as Result of Exposure	Standard Deviation (psi)	Significant Change
Crystal-Bar Zirconium	Unirradiated	4	30,000	30,000 29,600		330		48,400	48,900 47,700		512	
	Core (low flux)[a]	2	28,300	29,000 27,500	−2.3	522	Yes	46,800	47,100 46,500	−3.3	426	Yes
Zircaloy-2 (Core Tank)	Unirradiated	8	65,200	66,700 63,700		1005		71,200	72,500 70,700		564	
	Core (low flux)	6	65,300	66,300 64,700	+0.2	463	No	71,400	71,800 71,000	+0.3	250	No
	1st Blanket[b] loading	6	65,100	66,300 64,300	−0.2	493	No	70,900	71,400 70,600	−0.4	252	No
	2nd Blanket[c] loading	2	68,300	68,400 68,300	+5.1	750	Yes	71,800	72,900 70,700	+0.8	603	No
Zircaloy-2 (Code B)	Unirradiated	6	55,400	56,900 54,900		776		73,500	74,100 72,900		521	
	Core (low flux)	2	54,900	56,100 53,700	−0.9	809	No	73,600	73,800 73,300	−0.1	406	No
	2nd Blanket loading	2	58,100	58,400 57,800	+4.9	596	Yes	74,900	75,100 74,700	+1.6	399	Yes
Zircaloy-2 (Code B) Beta-Quenched	Unirradiated	2	66,600	66,600 66,600		0		81,100	81,800 80,400		990	
	2nd Blanket loading	2	73,600	74,600 72,500	+10.5	324	Yes	86,000	86,100 85,800	+6.0	716	Yes
Zircaloy-2 Welded (Cobe B)	Unirradiated	6	57,500	58,800 54,800		1582		67,000	69,000 65,300		1365	
	Core (low flux)	2	59,000	59,200 58,800	+2.6	1184	No	66,500	67,100 65,900	−0.7	1055	No
	1st Blanket loading	2	59,700	60,500 58,800	+3.8	1247	No	66,500	67,400 65,500	−0.7	1112	No
	2nd Blanket loading	2	61,200	62,600 59,800	+6.4	1352	Yes	68,400	69,200 67,600	+2.1	1085	No

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[a] Estimated fast (>1 Mev) neutron dose is 0.8×10^{16} nvt.

[b] Estimated fast (>1 Mev) neutron dose is 1.3×10^{17} nvt.

[c] Estimated fast (>1 Mev) neutron dose is 2.6×10^{18} nvt.

Table 1 (continued)

Material	Irradiated or Unirradiated	No. of Breaks Averaged	0.2% Offset Yield Strength					Tensile Strength				
			0.2% Yield Strength (mean) (psi)	Max and Min Values, 0.2% Yield Strength (psi)	Percentage Change of Mean as Result of Exposure	Standord Deviation (psi)	Significant Change	Ultimate Tensile Strength (mean) (psi)	Max and Min Tensile Strength (psi)	Percentage Change of Mean as Result of Exposure	Standard Deviation (psi)	Significant Change
A-40 Titanium	Unirradiated	9	40,100	40,800 38,500		676		47,800	48,800 46,800		622	
	Core (low flux)	2	39,600	40,000 39,200	– 1.2	520	No	47,700	48,100 47,200	– 0.2	541	No
	1st Blanket loading	7	41,700	44,700 40,600	+4.0	529	Yes	50,900	52,000 50,000	+6.5	367	Yes
	2nd Blanket loading	2	46,700	48,400 44,900	+16.5	815	Yes	57,500	58,000 57,000	+20.3	547	Yes
A-110 AT Titanium	Unirradiated	2	122,400	123,000 121,800		85		132,100	133,200 121,000		156	
	1st Blanket loading	2	124,100	124,200 124,000	+1.4	608	No	133,100	133,200 133,000	+0.7	111	No
Type 347 Stainless Steel	Unirradiated	8	45,000	48,000 42,000		2187		90,800	95,800 86,300		4214	
	Core (low flux)	6	43,300	45,900 40,800	– 3.8	1167	No	87,700	88,700 85,400	– 3.4	1783	No
	1st Blanket loading	4	47,700	50,000 45,500	+6.8	1287	No	97,000	98,000 95,800	+6.8	2185	Yes
Type 347 Stainless Steel (welded)	Unirradiated	4	53,900	54,900 52,900		1155		96,900	96,900 96,900		0	
	Core (low flux)	4	56,800	58,800 55,300	+5.2	947	Yes	97,300	98,000 96,500	+0.4	338	Yes
	2nd Blanket loading	2	62,400	62,700 62,100	+15.8	671	Yes	100,800	101,500 100,000	+4.0	325	Yes
Type 304 Stainless Steel	Unirradiated	2	64,300	65,100 63,500		1131		93,500	94,100 92,900		849	
	Core (low flux)	2	62,800	64,700 60,800	– 2.3	2108	No	91,800	93,700 89,800	– 1.8	2040	No
Type 304 L Stainless Steel	Unirradiated	2	64,900	65,100 64,700		283		94,500	94,800 94,100		495	
	2nd Blanket loading	2	67,700	67,800 67,600	+4.3	224	Yes	96,200	96,400 95,900	+1.8	430	No
Type 309 SCb Stainless Steel	Unirradiated	4	58,400	60,300 56,800		1658		96,100	96,500 95,500		526	
	2nd Blanket loading	2	59,500	59,800 59,200	+1.9	1257	No	99,800	100,200 99,400	+3.7	464	Yes
Incoloy	Unirradiated	2	48,400	48,600 48,200		283		86,300	86,500 86,100		283	
	2nd Blanket loading	2	48,600	49,300 47,800	+0.4	776	No	93,000	93,200 92,800	+7.8	283	Yes

Table 2. Elongation Values of Various Materials Obtained from Crosshead Motion as Influenced by Exposure in the HRE-2

Material	Irradiated or Unirradiated	No. of Breaks Averaged	Elongation Values Obtained from Crosshead Motion														
			Total Elongation					Uniform Elongation					Necking Elongation				
			Total Elongation (mean) (%)	Max and Min Values of Total Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Uniform Elongation (mean) (%)	Max and Min Values of Uniform Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Necking Elongation (mean) (%)	Max and Min Values of Necking Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change
Crystal-Bar Zirconium	Unirradiated	4	42.0	44.8 37.2		3.34		27.0	29.5 24.8		2.01		15.0	16.6 12.4		1.83	
	Core (low flux)	2	38.5	40.6 36.3	– 8.3	2.83	No	25.8	27.2 24.3	– 4.4	1.75	No	12.7	13.4 12.0	– 15.3	1.44	No
Zircaloy-2 (Core Tank)	Unirradiated	8	23.5	24.8 21.6		1.09		7.6	9.0 6.4		0.84		15.9	17.8 14.5		1.11	
	Core (low flux)	6	24.1	30.4 21.0	+2.6	1.27	No	7.4	8.2 6.7	– 2.6	0.41	No	16.7	22.2 14.1	+5.0	1.11	No
	1st Blanket loading	6	22.1	24.1 20.4	– 6.0	0.64	Yes	7.4	8.3 6.2	– 2.6	0.46	No	14.7	16.0 14.2	– 7.5	0.51	Yes
	2nd Blanket loading	2	20.6	21.0 20.2	– 12.3	0.82	Yes	6.9	7.2 6.5	– 9.2	0.64	No	13.7	14.5 13.0	– 13.8	0.87	Yes
Zircaloy-2 (Code B)	Unirradiated	6	29.0	32.5 26.0		2.75		16.9	19.5 15.0		1.63		12.1	15.8 10.1		2.17	
	Core (low flux)	2	25.8	26.2 25.4	– 11.0	2.14	No	15.5	16.0 15.0	– 8.3	1.38	No	10.3	10.4 10.2	– 14.9	1.62	No
	2nd Blanket loading	2	25.5	26.5 24.5	– 12.1	2.11	No	15.8	16.5 15.0	– 6.5	1.26	No	9.7	10.0 9.5	– 19.8	1.62	No
Zircaloy-2 (Code B) Beta-Quenched	Unirradiated	2	21.7	22.3 21.1		0.85		11.6	12.5 10.6		1.34		10.1	10.5 9.8		0.67	
	2nd Blanket loading	2	17.0	17.5 16.5	– 21.7	0.78	Yes	10.3	11.0 9.5	– 12.6	1.71	No	6.7	7.0 6.5	– 33.7	0.54	Yes
Zircaloy-2 Welded (Code B)	Unirradiated	6	12.2	15.9 9.7		2.17		6.3	9.0 4.0		1.69		5.9	6.9 5.1		0.73	
	Core (low flux)	2	9.7	11.2 8.2	– 20.0	1.77	No	4.0	4.0 4.0	– 36.5	1.26	No	5.7	7.2 4.2	– 3.4	0.89	No
	1st Blanket loading	2	11.9	13.2 10.5	– 2.5	1.74	No	6.0	6.7 5.2	– 4.8	1.31	No	5.9	6.5 5.3	0	0.62	No
	2nd Blanket loading	2	10.6	11.1 10.1	– 13.0	1.64	No	5.3	5.5 5.2	– 15.9	1.26	No	5.3	5.6 4.9	– 10.2	0.57	No
A-40 Titanium	Unirradiated	9	45.9	51.3 40.5		3.93		10.0	14.6 8.2		2.29		35.9	42.3 31.8		3.38	
	Core (low flux)	2	40.8	40.8 40.8	– 11.1	2.90	No	8.9	9.0 8.7	– 11.0	1.69	No	31.9	32.1 31.8	– 11.1	2.49	No
	1st Blanket loading	7	38.1	42.0 32.8	– 17.0	1.96	Yes	11.1	14.4 9.0	+11.0	1.07	No	27.0	30.0 23.8	– 24.8	1.63	Yes
	2nd Blanket loading	2	32.8	35.0 30.7	– 26.3	3.00	Yes	11.6	13.5 9.7	+16.0	1.83	No	21.2	25.3 17.2	– 46.5	2.91	Yes

Table 2 (continued)

Material	Irradiated or Unirradiated	No. of Breaks Averaged	Elongation Values Obtained from Crosshead Motion														
			Total Elongation					Uniform Elongation					Necking Elongation				
			Total Elongation (mean) (%)	Max and Min Values of Total Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Uniform Elongation (mean) (%)	Max and Min Values of Uniform Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Necking Elongation (mean) (%)	Max and Min Values of Necking Elongation (%)	Percentage Change of Mean as Result of Exposure	Standord Deviation (%)	Significant Change
A-110 AT Titanium	Unirradiated	2	19.8	21.0 18.6		1.70		10.6	10.9 10.2		0.49		9.2	10.1 8.4		0.67	
	1st Blanket loading	2	20.7	21.1 20.3	+4.5	1.27	No	11.1	11.6 10.6	+4.7	0.61	No	9.6	9.7 9.5	+4.2	0.48	No
Type 347 Stainless Steel	Unirradiated	8	81.8	87.4 77.4		3.25		70.4	75.4 66.7		3.32		11.4	12.3 10.0		0.82	
	Core (low flux)	5	82.1	89.6 72.0	+0.4	2.69	No	71.3	79.0 62.0	+1.3	2.56	No	10.8	11.5 10.0	− 5.3	0.38	No
	1st Blanket loading	4	74.4	77.2 69.6	− 9.0	6.38	No	63.3	67.0 58.0	− 10.1	2.14	Yes	11.1	11.6 10.2	− 2.6	0.48	No
Type 347 Stainless Steel (welded)	Unirradiated	4	55.8	60.6 53.0		4.02		46.7	50.2 44.6		2.64		9.1	10.4 8.3		0.91	
	Core (low flux)	4	56.0	61.0 50.0	+4.0	3.09	No	47.0	49.3 43.7	+0.8	1.84	No	9.0	11.7 6.3	− 1.1	0.67	No
	2nd Blanket loading	2	52.6	53.9 51.3	− 5.4	2.20	No	44.3	46.0 42.7	− 5.1	1.57	No	8.3	8.6 7.9	− 8.8	0.51	No
Type 304 Stainless Steel	Unirradiated	2	67.7	70.4 65.0		3.82		55.7	57.8 53.5		3.04		12.0	12.6 11.5		0.78	
	Core (low flux)	2	63.7	64.1 63.2	− 5.9	8.73	No	51.7	52.0 51.3	− 7.2	0.971	Yes	12.0	12.8 11.2	0	2.17	No
Type 304 L Stainless Steel	Unirradiated	2	70.9	72.5 69.2		2.33		59.1	61.0 57.2		2.69		11.8	12.0 11.5		0.35	
	2nd Blanket loading	2	61.6	63.1 60.0	− 13.1	2.26	No	50.7	53.2 48.1	− 14.2	1.03	Yes	10.9	11.9 9.9	− 7.6	3.18	No
Type 309 SCb Stainless Steel	Unirradiated	4	59.1	61.3 56.4		2.15		46.2	47.8 43.0		2.22		12.9	13.8 11.0		1.33	
	2nd Blanket loading	2	49.9	51.5 48.2	− 15.6	1.90	Yes	39.3	41.3 37.2	− 14.9	1.024	Yes	10.6	11.0 10.2	− 17.8	2.09	No
Incolay	Unirradiated	2	57.8	61.5 54.0		5.30		46.5	50.5 42.5		5.66		11.3	11.5 11.0		0.35	
	2nd Blanket loading	2	55.5	56.0 55.0	− 4.0	3.78	No	45.3	46.0 44.5	− 2.8	0.354	No	10.2	10.5 10.0	− 9.7	4.07	No

Table 3. Elongation Values of Various Materials Obtained from Extensometer Data as Influenced by Exposure in the HRE-2

Material	Irradiated or Unirradiated	No. of Breaks Averaged	Elongation Values Obtained from Extensometer Data														
			Total Elongation					Uniform Elongation					Necking Elongation				
			Total Elongation (mean) (%)	Max and Min Values of Total Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Uniform Elongation (mean) (%)	Max and Min Values of Uniform Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Necking Elongation (mean) (%)	Max and Min Values of Necking Elongation (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change
Crystal-Bar Zirconium	Unirradiated	3	39.9	41.2 38.6		1.31		20.7	20.8 20.7		0.059		19.2	20.4 17.8		1.15	
	Core (low flux)	2	36.3	39.8 32.8	−8.3	8.28	No	20.5	22.2 18.9	−1.0	1.23	No	15.8	17.6 13.9	−17.7	1.62	No
Zircaloy-2 (Code B)	Unirradiated	4	29.1	32.8 25.0		3.55		14.3	16.7 13.2		1.59		14.8	17.7 11.8		2.62	
	Core (low flux)	2	25.3	26.0 24.6	−13.1	2.70	No	13.4	14.6 12.2	−6.3	1.40	No	11.9	12.4 11.4	−19.5	1.99	No
A-40 Titanium	Unirradiated	3	46.3	54.5 40.5		7.32		7.9	8.5 7.2		0.65		38.4	46.6 32.0		7.46	
	Core (low flux)	[a]	43.0	43	−7.1	6.90	No	7.1	7.1 6.8	−10.1	0.50	No	35.9	35.9	−6.5	7.04	No
	1st Blanket loading	3	34.8	37.8 33.2	−24.8	5.78	No	7.8	8.6 7.5	−1.3	0.68	No	27.0	29.2 25.7	−29.7	5.75	No
Type 347 Stainless Steel	Unirradiated	4	58.5	62.0 55.3		1.12		48.8	51.2 47.5		1.67		9.7	13.7 4.1		4.21	
	Core (low flux)	6	59.0	66.0 49.0	+0.9	3.38	No	49.7	52.7 43.0	1.8	1.96	No	9.3	14.9 5.2	−4.1	2.62	No

[a] One break for total elongation, 2 breaks for uniform elongation. Specimen was damaged during testing.

Table 4. Reduction of Area and Anisotropy of Various Materials as Influenced by Exposure in the HRE-2

			Reduction of Area					Ratio of Fracture Diameters					k_{xy} Values				
Material	Irradiated or Unirradiated	No. of Breaks Averaged	Reduction of Area (%)	Max and Min Values of Reduction of Area (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	Ratio of Fracture Diameters	Max and Min Values of Ratio of Fracture Diameters (%)	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change	k_{xy} Values (mean)	Max and Min Values of k_{xy}	Percentage Change of Mean as Result of Exposure	Standard Deviation (%)	Significant Change
Zircaloy-2 (Core Tank)	Unirradiated	4	55.2	57.6 53.7		1.75		1.627	1.685 1.588		0.04		−0.8028	−0.8013 −0.8060		0.0024	
	1st Blanket loading	2	57.2	59.2 55.2	3.6	1.8	No	1.660	1.71 1.61	+2.0	0.5	No	−0.7989	−0.7984 −0.7993	+0.5	0.0018	No
	2nd Blanket loading	2	53.7	54.7 52.6	−2.7	1.5	No	1.583	1.589 1.577	−2.7	0.5	No	−0.8009	−0.7945 −0.8073	+0.2	0.0043	No
Zircaloy-2 (Code B)	Unirradiated	2	37.4	38.0 36.8		1.65		1.127	1.127 1.126		0.0007		−0.6271	−0.6239 −0.6302		0.0044	
	2nd Blanket loading	2	35.5	36.3 34.6	−5.1	1.4	No	1.132	1.144 1.119	+0.4	0.025	No	−0.6391	−0.6292 −0.6489	−1.9	0.0103	No
Zircaloy-2 (Code B) Beta-Quenched	Unirradiated	2	28.8	31.7 25.8		4.17		1.106	1.116 1.096		0.014		−0.6500	−0.6450 −0.6549		0.0070	
	2nd Blanket loading	2	26.7	27.6 25.7	−7.3	3.1	No	1.106	1.115 1.096	0.0	0.0	No	−0.6599	−0.6549 −0.6648	−1.5	0.0070	No
Zircaloy-2 Welded (Code B)	Unirradiated	4	37.5	40.3 33.7		2.83		1.081	1.10 1.05		0.026		−0.5829	−0.5589 −0.5936		0.0165	
	1st Blanket loading	2	38.4	40.8 36.0	+2.4	2.6	No	1.080	1.119 1.04	−0.1	0.03	No	−0.5759	−0.5437 −0.6081	+1.2	0.02328	No
	2nd Blanket loading	2	37.7	40.5 34.9	+0.5	2.7	No	1.087	1.113 1.06	−0.6	0.03	No	−0.5863	−0.5686 −0.6039	−0.6	0.0520	No
A-40 Titanium	Unirradiated	4	67.6	70.1 65.0		2.38		2.008	2.05 1.96		0.048		−0.8093	−0.7973 −0.8205		0.0101	
	1st Blanket loading	4	71.4	75.6 67.0	+5.6	2.16	No	2.113	2.37 1.89	+5.3	0.10	No	−0.7963	−0.7890 −0.8047	+1.61	0.00648	No
	2nd Blanket loading	2	68.6	69.0 68.1	+1.5	1.80	No	1.948	1.973 1.923	−3.0	0.04	No	−0.7861	−0.7814 −0.7907	+2.9	0.00809	Yes
A-110 AT Titanium	Unirradiated	2	44.1	45.0 43.1		1.34		1.131	1.137 1.125		0.032		−0.6067	−0.5996 −0.6138		0.0100	
	1st Blanket loading	2	39.2	40.0 38.4	−11.1	1.76	No	1.100	1.10 1.10	−2.7	0.02	No	−0.5964	−0.6012 −0.5915	+1.7	0.0085	No
Type 347 Stainless Steel	Unirradiated	4	61.4	63.5 60.0		1.62		1.029	1.050 1.010		0.018		−0.5151	−0.5033 −0.5271		0.102	
	1st Blanket loading	4	62.6	67.5 58.0	+2.0	2.42	No	1.025	1.030 1.020	−0.4	0.01	No	−0.5132	−0.5102 −0.5148	+0.4	0.005	No
Type 304 L Stainless Steel	Unirradiated	2	74.1	76.4 71.7		3.32		1.020	1.020 1.020		0		−0.5073	−0.5064 −0.5082		0.001	
	2nd Blanket loading	2	75.5	78.0 73.0	+1.8	3.43	No	1.012	1.014 1.010	−0.8	0.002	No	−0.5043	−0.5037 −0.5047	+0.6	0.001	No
Type 309 SCb Stainless Steel	Unirradiated	4	64.8	67.4 61.1		2.66		1.024	1.032 1.015		0.008		−0.5109	−0.5072 −0.5144		0.0032	
	2nd Blanket loading	2	69.8	71.7 67.8	+7.7	2.32	No	1.015	1.020 1.010	−0.9	0.007	No	−0.5061	−0.5039 −0.5082	+0.9	0.0027	No
Incoloy	Unirradiated	2	61.5	61.9 61.1		0.57		1.021	1.023 1.018		0.004		−0.5098	−0.5076 −0.5119		0.0032	
	2nd Blanket loading	2	65.7	66.6 64.7	+6.8	1.46	No	1.038	1.045 1.030	+1.7	0.11	No	−0.5173	−0.5126 −0.5219	−1.5	0.0052	No

When titanium and zirconium alloys are strained in tension, the reduction that occurs in the specimen is not as uniform as it is in steels. The metal usually contracts less in the thickness direction of the plate than in directions in the plane of the plate, thus leading to an elliptical cross section in originally round specimens. The sizes and shapes of such ellipses are reproducible and are related to the thermal and mechanical history, composition, and specimen orientation. The averaged ratio of the fracture diameters is shown in Table 4. Variations in the elliptical axes are a sensitive indication of anisotropy. Shown in the last column in Table 4 are the k_{xy} values.[7] A completely isotropic material would have a k_{xy} value of -0.5. As the anisotropy increases, the k_{xy} value approaches -1.0. The k_{xy} is the ratio of the contractile true strain in the plane of the plate to the total axial true strain. This ratio is constant for any axial true strain greater than approximately 0.02.

It should be noted that these specimens have been exposed to relatively small-to-moderate neutron doses. The results must therefore be considered as screening tests and evaluated for possible trends rather than for the actual magnitude of the changes.

When statistically significant changes occurred as indicated in Tables 1–4, the following yardstick was used to measure the extent of damage as related to the percentage change:

Extent of Change	% Change[8]
Very small	< 7.0
Small	7.1 to 17.0
Moderate	17.1 to 35.0
Large	35.1 to 50.0
Very large	> 50.0

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[7] P. L. Rittenhouse and M. L. Picklesimer, Metallurgy of Zircaloy-2 Part II. The Effects of Fabrication Variables on the Preferred Orientation and Anisotropy of Strain Behavior, ORNL-2948 (Jan. 11, 1961).

[8] Percentage changes were calculated to the closest 0.1%.

It should be mentioned that in many cases there were considerable percentage changes, but they were not statistically significant since the variability of the unirradiated specimens was also large.

Crystal-Bar Zirconium

Low-Flux-Core Region. A very small but statistically significant decrease was observed in the yield and tensile strengths.

The original material was cold reduced about 90% and vacuum annealed at 525°C for 3 hr. As was mentioned previously, the temperature was below 250°C for 190 hr and between 250 and 300°C for 890 hr. The long time between 250 and 300°C might have caused additional annealing over the 3 hr at 525°C since it is believed that the radiation strengthening at 0.8×10^{16} nvt is negligible. There were no significant changes in ductility, i.e., from the elongation values as obtained using crosshead and/or extensometer motion. This material was practically unaffected by the exposure condition.

Zircaloy-2 (Core Tank)

Low-Flux-Core Region (0.8×10^{16} nvt). No significant change was noted in any of the mechanical properties.

First Blanket Loading (1.3×10^{17} nvt). No significant changes were observed in the yield or tensile strengths or reduction of area; however, there was a small decrease in the necking and total elongation, as shown in Table 2. Kemper and Zimmerman[9] report a greater reduction in uniform strain than for necking elongation. This is opposite to the findings in this report. However, their irradiations were performed at 40 to 60°C to an estimated integrated thermal neutron flux of 1.4×10^{21} nvt.

Second Blanket Loading (2.6×10^{18} nvt). A very small increase in the yield strength and a small decrease in the necking and total elongation were noted. No significant change was observed in the reduction of area.

Except for a very small decrease in ductility for the first and second blanket loading and a very small increase in the yield strength

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[9] R. S. Kemper and D. L. Zimmerman, Neutron Irradiation Effects on the Tensile Properties of Zircaloy-2, HW-52323 (Aug. 22, 1957).

for the second blanket loading, this material was practically unaffected by the various exposure conditions.

Zircaloy-2 (Code B)

Low-Flux-Core Region. No significant changes were observed in any of the mechanical properties.

Second Blanket Loading. Small but significant changes were observed in the yield and tensile strengths, but no changes were observed in the ductility, i.e., from elongation or reduction of area values.

Percentagewise, there was a small-to-moderate decrease in elongation values under both conditions of exposure but, statistically, these decreases were not significant due to the large spread in values for both the unirradiated and irradiated conditions.

Zircaloy-2 (Code B) Beta Quenched

Second Blanket Loading. Small but significant increases occurred in the yield and tensile strengths. Moderate decreases were observed in the necking and total elongation. The beta-quenched material appeared to undergo greater changes in tensile properties than did the unquenched material.

Zircaloy-2 (Code B) Welded

No significant changes were observed for specimens exposed to the low-flux-core region or to the first blanket loading. However, a very small but statistically significant increase was noted in the tensile strength as a result of exposure in the second blanket loading.

A-40 Titanium

Low-Flux-Core Region. No significant change was observed in any of the mechanical properties.

First Blanket Loading. Very small but significant increases were observed in the yield and tensile strengths and a small-to-moderate decrease was noted in the necking and total elongation. It should be noted that the percentage changes in elongation using the extensometer were greater than those obtained from the crosshead motion. The variability of the unirradiated specimens using the extensometer was greater than that obtained from the crosshead motion for the total and necking elongation so that no significant change was observed in the extensometer data.

The fact that the larger significant changes occurred in the A-40 titanium than in any of the materials tested is probably related to the fact that this material has the lowest initial yield and ultimate strengths.

Second Blanket Loading. Small-to-moderate increases were observed in the yield and tensile strengths with moderate-to-large increases in the total and necking elongation. The reduction in area was not affected; however, a very small decrease in anisotropy was noted. Of all the materials exposed in this loading, the greatest changes occurred in the A-40 titanium.

A-110 AT Titanium

First Blanket Loading. No significant change was observed in any of the mechanical properties. This material appeared to be the least sensitive to the reactor environment of the three titanium alloys tested. A slight yield point was observed for both the control and irradiated conditions.

Type 347 Stainless Steel

Low-Flux-Core Region. No significant changes were observed.

First Blanket Loading. No significant change was observed in the yield strength but a very small increase was observed in the tensile strength. A small decrease was observed in the uniform elongation but the reduction of area was unaffected. About 86% of the total elongation is necking elongation so that there is not a serious loss of ductility as a result of irradiation.

Type 347 Stainless Steel (Welded)

Low-Flux-Core Region. A very small increase was observed in the yield strength. The variability of the tensile strength of the un-irradiated specimens was very small. All four breaks had the same tensile strength so that, even though there was only an 0.4% increase in the irradiated specimens, it was a very small but statistically significant increase.

Second Blanket Loading. A small increase in the yield strength and a very small increase in the tensile strength were observed. There were no significant changes in the ductility.

Type 304 Stainless Steel

Low-Flux-Core Region. No significant changes were observed except for a small decrease in the uniform elongation.

Type 304L Stainless Steel

Second Blanket Loading. A very small but significant increase was observed in the yield strength and a small decrease was observed in the uniform elongation.

Type 309 SCb Stainless Steel

Second Blanket Loading. A very small but significant increase was observed in the tensile strength and a small decrease was observed in the total and uniform elongation.

Incoloy

Second Blanket Loading. A very small increase was observed in the tensile strength but no significant decrease was observed in the ductility as measured by percent elongation.

CONCLUSIONS

Some very small changes were observed in the tensile properties as a result of the smallest exposure (i.e., 0.8×10^{16} nvt), but these were almost insignificant. At 1.3×10^{17} nvt, some small decreases in ductility were observed for Zircaloy-2 core tank, type 347 stainless steel, and A-40 titanium. At 2.6×10^{18} nvt, small but definite changes in some of the mechanical properties, usually as slight increases in tensile and yield strengths and small reductions in elongation, were observed in all of the materials exposed with the greatest changes occurring in the A-40 titanium. The extent of corrosion was practically negligible for the three exposures.

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APPENDIX

In order to determine if the observed changes in the mechanical properties were statistically significant, the following procedure was used.

The sum of the squares of the deviation from the mean was determined for the various mechanical properties of materials of interest in the unirradiated and irradiated conditions. This quantity is identified as follows:

$$SS_{un} = \left[\sum_{i=1}^{n} X_i^2 - \frac{\left(\sum_{i=1}^{n} X_i \right)^2}{n} \right]_{un} \tag{1}$$

where the subscript un represents the unirradiated condition and X_1, X_2, $X_3 \ldots X_n$ are the individual values of the particular mechanical property being evaluated.

The standard deviation for the difference between the two means (i.e., $\bar{X}_{un} - \bar{X}_{irr}$ where the subscript irr represents the irradiated condition) is estimated by

$$s_{(\bar{X}_{un} - \bar{X}_{irr})} = \left[\left(\frac{SS_{un} + SS_{irr}}{n_{un} + n_{irr} - 2} \right) \left(\frac{1}{n_{un}} + \frac{1}{n_{irr}} \right) \right]^{\frac{1}{2}} \tag{2}$$

where n_{un} and n_{irr} are equal, respectively, to the number of unirradiated and irradiated tests. The standard deviation for the unirradiated condition is estimated by

$$s_{un} = \left[\frac{SS_{un}}{n_{un} - 1} \right]^{\frac{1}{2}} \tag{3}$$

The procedure for testing the significance of the difference between two means was to compute the ratio

$$t_c = \frac{\bar{X}_{un} - \bar{X}_{irr}}{s_{(\bar{X}_{un} - \bar{X}_{irr})}} \tag{4}$$

and compare t_c with the 5% value of t(with $n_{un} + n_{irr} - 2$ degrees of freedom) as tabulated in tables of the "Student's t Distribution".[10] If $|t_c| \geq t$, the difference between the two means was declared statistically significant at the 5% level. If, on the other hand, $|t_c| < t$, the conclusion was that there was insufficient evidence to detect any difference between the two means. An example illustrating the above is shown below.

Consider the case for crystal-bar zirconium where it is desirable to calculate the standard deviation of the 0.2% yield strength for the unirradiated and the irradiated conditions and also whether or not a statistically significant change has occurred as a result of the exposure to the reactor environment.

The standard deviation for the unirradiated condition is estimated by Eq. (3), where SS_{un} = the sum of the squares of the deviation from the mean as defined in Eq. (1). Also found in Eq. (1) are X_1, X_2, $X_3 \ldots X_n$ which are experimentally obtained individual values of the 0.2% yield strength.

Two double-break tensile specimens (i.e., four tests) of crystal-bar zirconium were pulled and the following results were obtained for the yield strength from unirradiated specimens:

$\times 10^2$	$\times 10^4$
302	91,204
298	88,804
296	87,616
303	91,809
1,199	359,433

also

$\overline{X} = \frac{1199}{4} \times 10^2 = 29,975$; rounded off to the closest 100 psi, $\overline{X} = 30,000$.

Then

$$\sum_{i=1}^{4} X_i = 1,199 \times 10^2$$

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[10] A. Hald, Statistical Tables and Formulas, p 39, Wiley, New York (1952).

and

$$\sum_{i=1}^{4} X_i^2 = 359{,}433 \times 10^4.$$

Substituting into Eq. (1)

$$SS_{un} = \left[359{,}433 \times 10^4 - \frac{(1199 \times 10^2)^2}{4}\right] = 32.75 \times 10^4 \text{ (psi)}^2$$

and from Eq. (3)

$$s_{un} = \left(\frac{32.75 \times 10^4}{3}\right)^{\frac{1}{2}} = 330 \text{ psi.}$$

The standard deviation between the two means (i.e., $\overline{X}_{un} - \overline{X}_{irr}$) is denoted by Eq. (2), where n_{un} = number of tests made on unirradiated material, n_{irr} = number of tests made on irradiated material.

In order to calculate $s_{(\overline{X}_{un} - \overline{X}_{irr})}$, it is necessary that $SS_{\ell fc}$ (i.e., with reference to irradiated specimens from the low-flux core region) be calculated.

$\times 10^2$	$\times 10^4$
275	75,625
290	84,100
565	159,725

From Eq. (1) $SS_{\ell fc} = \left[159{,}725 \times 10^4 - \frac{(565 \times 10^2)^2}{2}\right] = 112.5 \times 10^4 \text{ (psi)}^2.$

Substituting into Eq. (2)

$$s_{(\overline{X}_{un} - \overline{X}_{\ell fc})} = \left[\left(\frac{32.75 + 112.50}{4 + 2 - 2}\right)\left(0.25 + 0.50\right)\right]^{\frac{1}{2}} 10^2 = 522 \text{ psi}$$

also

$$\overline{X}_{\ell fc} = 28{,}300.$$

In order to determine if the difference between the two means (i.e., $\overline{X}_{un} - \overline{X}_{irr}$) was statistically significant, it was necessary to calculate t_c, as defined by Eq. (4). In this case

$$t_c = \frac{30,000 - 28,300}{522} = 3.26.$$

From tables of the t distribution where the degree of freedom is 4 (i.e., 4 + 2 - 2) and corresponding to the 5% significance level, $t = 2.776$.

In this example $|t_c| > t$, indicating that a statistically significant change has occurred as a result of the exposure. The decrease in the yield strength from a mean value of 30,000 to 28,300 psi indicates that, at this low dose, the effect of annealing was more predominant as compared to the strengthening effect produced by the neutron bombardment.

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