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EVALUATION OF IRON-AND NICKEL-BASE ALLOYS
FOR MEDIUM AND HIGH TEMPERATURE
REACTOR APPLICATIONS: PART II

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EVALUATION OF IRON- AND NICKEL-BASE ALLOYS
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REACTOR APPLICATIONS: PART II

By

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EVALUATION OF IRON- AND NICKEL-BASE ALLOYS
FOR MEDIUM AND HIGH TEMPERATURE
REACTOR APPLICATIONS: PART II

INTRODUCTION

An evaluation of several nickel- and iron-base alloys was conducted at the Pacific Northwest Laboratory as part of a program to determine the irradiation effects upon the mechanical properties of several iron- and nickel-base alloys. The corrosion resistance and good mechanical properties at elevated temperature make these alloys candidate materials for high temperature applications in many nuclear reactor components. Examples of such applications are the Advanced Test Reactor (ATR) and Plutonium Recycle Test Reactor (PRTR) high-temperature-gas-loop irradiation facilities.

By using preliminary surveys and screening procedures, a series of austenitic and ferritic stainless steels, experimental iron-chromium-aluminum-base alloys, and solution and precipitation hardenable nickel-base alloys were selected for evaluation. Determination of out-of-reactor corrosion rates in high temperature gaseous and aqueous environments, fabricability, and effects of neutron irradiation on the structural and mechanical properties of these alloys were the main objectives of the program.

Part I of AEC Research and Development Report HW-67715,^{*} under the above title in February, 1961, contains (in addition to the alloy property characteristics) the results of corrosion tests in air, carbon dioxide, helium, and nitrogen up to 1060 °C and in high purity water of 360 °C and 3200 psi pressure.

Conclusions from Part I of this report justified further evaluation of the selected alloys. More evaluation of the effects of radiation under various environmental conditions upon their properties was especially needed.

* General Electric Company, Richland, Washington

Because of the growing trend toward higher temperatures and pressures in water-cooled reactors, several alloys whose properties were suitable for nuclear superheat applications were included in the evaluation.

Part II of this report summarizes the results which were obtained by the cooperative work of various departments of the Pacific Northwest Laboratory and of several offsite research centers referred to in the appropriate sections of the report.

SUMMARY

When alloys were exposed up to 1060 °C for 500 hr in flowing helium and nitrogen that had been first passed through graphite at 450 to 500 °C, AISI 406 SS, PDRL-102, Hastelloy X and N, and the Fe-30 Cr-1 Y type experimental alloy showed the lowest weight gains and increased in the order listed.

When exposed for 100 hr to pure, dry carbon dioxide while in contact with powdered graphite at 100 °F temperature intervals up to 1093 °C, Hastelloy X gained the least in weight and was followed by Inconel X-750, Hastelloy R-235 and AISI 406 SS.

Extensive autoclave testing of the selected alloys in oxygenated and deoxygenated superheated steam at 550 °C and 1000, 3000, and 5000 psi pressure; and at 650 °C and 600 psi pressure, respectively, established the following tentative grouping of alloys with increasing resistance to corrosion:

1. AISI 304, 316, 406, and 430 SS showed weight gains from about 50 to 300 mg/dm², or over 0.2 mil of penetration;
2. Fe-25 Cr-3 Al-0.6 Y, Hastelloy N, PDRL-102 and Inconel X-750 showed weight gains from about 8 to 75 mg/dm², or 0.1 to 0.2 mil penetration; and
3. Incoloy 800, AISI 446 SS, Hastelloy X-280, Hastelloy R-235, and Fe-2 Cr-5 Al showed weight gains below 15 mg/dm², or less than 0.1 mil penetration.

The oxygen content and pressure of the steam appear to affect the various alloys in different ways. While the ferritic stainless steels and the Fe-Cr-Al-base alloys seem to offer better resistance in oxygenated steam, the austenitic and martensitic stainless steels, and the nickel-base superalloys are more resistant to deoxygenated steam. The steam pressure also appears to influence the corrosion rates of various alloys in different ways. However, the overall rating of resistance to the corrosion media and the environmental conditions used remains generally the same as the rating previously established in Part I.

Susceptibility to stress corrosion cracking of the plastically and elastically prestressed alloys in 42% boiling, aqueous, magnesium-chloride solution was severest for the 300 type stainless steels, and considerably less severe for the 400 series stainless steels, the Fe-Cr-Al-base alloys, and the nickel-base alloys Hastelloy X and PDRL-102. AISI 406 SS and Incoloy 800 appear to be most resistant to stress condition.

The effects of neutron exposure upon the mechanical properties of several iron- and nickel-base alloys have been determined. Specimens were irradiated and tested at ambient and elevated temperatures after a maximum exposure of 1.21×10^{20} nvt. These test results show that it was not possible to classify the irradiation effects according to material category. An individual evaluation is required in terms of exposure and testing conditions. Generally, however, the comparative ambient pre- and postirradiation room temperature test results do not indicate any pronounced property changes because of irradiation, except for slight losses in strength and elongation. Actually, a slight increase in these properties was noted in AISI 406 SS. When tested at 700 °C, radically different effects of irradiation are observed. In addition to a general deterioration of the tensile properties (which is to be expected at this testing temperature), most alloys suffer varying severe losses of yield and ultimate strength due to irradiation and a most pronounced decrease in elongation. These effects are least noticeable in the stainless steels where the 406 SS shows greatly increased elongation caused by irradiation.

Biaxial Erichsen cup test results are also adversely affected by irradiation. The chemical composition of materials evaluated are listed in Table I. With the exception of the addition of Incoloy 800 and the elimination of several less promising alloys, the table contains the alloys selected for evaluation in Part I of this report.

DISCUSSION OF TEST RESULTS

Corrosion Tests

The shroud tube, of an in-reactor coolant testing facility, resting on the reactor graphite lattice blocks, was expected to tolerate prolonged contact with nitrogen. The nitrogen was contaminated with up to 1% CO and CO₂ on the outside and pure helium of ambient pressure on the inside at a maximum temperature of 950 °C. For evaluation, promising alloys, tentatively selected from those listed in Table I, were exposed for 500 hr at 1060 °C in both helium and nitrogen flowing at a rate of 0.75 ft³/hr. Both gases, prior to reaching the specimen coupons, were first passed over graphite heated to 450 to 500 °C. The resulting weight changes are recorded in Table II, and the alloy test specimens after exposure are shown in Figures 1 and 2. In comparing the results shown in these tables; AISI 406 SS, Hastelloy N, Hastelloy X-280, and PDRL-102 alloy were least affected under these exposure conditions. Hastelloy R-235, AISI 316 SS, and AISI 304 SS were grossly affected as shown by weight gains of 1.56, 1.85, and 3.32 mg/dm²/hr, respectively. As compared with various stainless steels and high nickel-chrome-base alloys, ^(1, 2) AISI 406 SS has been found to be of superior resistance to fuel oil flue gases at 565 to 850 °C and H₂S at 400 °C, respectively. These data are summarized in Tables III and IV for comparison with the present work.

In continuation of the high temperature CO₂-metal compatibility studies of the alloy evaluation program, a carbon dioxide exposure test series similar to those described in Part I of this report was carried out by the Materials Research and Service Section, Pacific Northwest Laboratory. In the present investigation one surface of the alloy specimens was in direct contact with powdered graphite while heated in a stream of CO₂ at from 538 to 1093 °C for 100 hr.

TABLE I
CHEMICAL COMPOSITIONS OF STAINLESS STEELS,
IRON-ALUMINUM-CHROMIUM-BASE ALLOYS,
AND NICKEL-BASE SUPERALLOYS

Alloy	Al, max	C, max	Si, max	Mn, max	P	S	Cr	Mo	Ni	Cb	Co, max	Cu	Ti	Fe	Zr	Y	V	W	wt%
302 SS	-	0.150	1.00	2.00	-	-	18.00	-	9.00	-	-	-	-	Balance	-	-	-	-	-
304 SS	-	0.100	1.00	2.00	-	-	19.00	2.50	10.00	-	-	-	-	Balance	-	-	-	-	-
316 SS	-	0.100	1.00	2.00	-	-	17.00	2.50	12.00	-	-	-	-	Balance	-	-	-	-	-
347 SS	-	0.030	1.00	2.00	-	-	18.00	-	11.00	10 x C	-	-	-	Balance	-	-	-	-	-
405 SS	0.30	0.080	1.00	2.00	-	-	11.5 to 13.5	-	-	-	-	-	-	Balance	-	-	-	-	-
406 SS (Al)	4.15	0.120	0.36	0.35	0.018	0.014	13.15	-	0.24	-	-	-	-	Balance	-	-	-	-	-
406 SS (Carpenter)	3.90	0.070	0.48	0.40	0.014	0.012	13.50	0.03	0.12	-	-	0.05	0.32	Balance	-	-	-	-	-
446 SS	-	0.350	1.00	2.00	-	-	23 to 27	-	-	-	-	-	-	Balance	-	-	-	-	-
410 SS	-	0.150	1.00	1.00	-	-	12.50	-	0.50	-	-	-	-	Balance	-	-	-	-	-
430 SS	-	0.120	1.00	1.00	-	-	16.00	-	0.50	-	-	-	-	Balance	-	-	-	-	-
502 SS	-	0.100	1.00	1.00	-	-	4 to 6	-	-	-	-	-	-	Balance	-	-	-	-	-
Fe-Cr-Al	5.60	0.030	0.10	0.10	0.009	0.014	24.00	-	0.10	-	-	-	-	Balance	-	-	-	-	-
Fe-Cr-Y	0.12	0.010	0.11	-	0.005	0.012	30.00	-	-	-	-	-	-	Balance	-	1.00	-	-	-
Fe-Cr-Al-Y	3.00	0.033	0.02	-	-	-	25.40	-	-	-	-	-	-	Balance	-	0.58	-	-	-
Hastelloy F	-	0.050	-	-	-	-	22.00	6.50	48.00	-	-	-	-	13.00	-	-	-	1.00	-
Hastelloy X-280	-	0.150	1.00	-	-	-	22.00	9.00	47.00	-	2.50	-	-	19.00	-	-	-	1.00	-
Hastelloy C	-	0.080	1.00	1.00	-	-	15.00	16.00	56.00	-	-	-	-	6.00	-	-	0.35	4.00	-
Hastelloy R-235	2.00	0.160	1.00	1.00	-	-	14.00	5.50	65.00	-	-	-	2.50	8.00	-	-	-	-	-
Hastelloy N (Inor 8)	0.50 max	0.020	0.32	0.34	0.009	0.040	7.00	15.82	70.50	-	0.51	0.03	-	4.85	-	-	-	0.35	-
Incoloy 800	-	0.100	0.60	1.00	-	-	20.50	-	32.00	-	-	-	-	Balance	-	-	-	-	-
Inconel 600	-	0.150	0.50	1.00	-	-	17.00	-	73.00	-	-	0.50	-	8.00	-	-	-	-	-
Inconel X-750	1.00	0.080	0.50	1.00	-	-	17.00	-	70.00	1.0	-	0.50	2.50	7.00	-	-	-	-	-
Inconel 702	3.75	0.100	0.70	1.00	-	-	17.00	-	75.00	-	-	0.50	1.00	2.00	-	-	-	-	-
D-979	1.15	0.054	0.30	0.30	0.005	0.009	15.32	4.30	45.54	-	-	0.06	2.84	26.00	-	-	-	3.70	B 0.011
PDRL-102 Alloy	0.50	0.060	-	-	-	-	15.00	3.00	68.00	3.0	-	-	0.50	7.00	-	-	-	3.00	B 0.004 Mg 0.02
DB-3-35	7.40	0.040	-	-	-	-	4.80	-	-	2.8	-	-	-	Balance	0.40	-	-	-	-
DB-3-38	7.20	0.030	-	-	-	-	4.70	-	-	2.6	-	-	-	Balance	0.50	-	-	-	-
201 SS	-	0.150	1.00	5.5 to 7.5	-	-	16.0 to 18.0	-	3.5 to 5.5	-	-	-	-	Balance	-	-	-	-	N max 0.25
202 SS	-	0.150	1.00	7.5 to 10.0	-	-	17.0 to 19.0	-	4.0 to 6.0	-	-	-	-	Balance	-	-	-	-	N max 0.25

TABLE II

WEIGHT GAINS OF IRON AND NICKEL-BASE ALLOYS EXPOSED FOR 500 HR IN HELIUM AND NITROGEN AT 1060 °C

No.	Type Alloy	Atmosphere: Flowing Helium (.75 ft ³ /hr) Passed Through Graphite at 450 to 500 °C	Atmosphere: Flowing Nitrogen (.75 ft ³ /hr) Passed Through Graphite at 450 to 500 °C
		Mg/Dm ² /hr	Mg/Dm ² /hr
15 _{III}	AISI 406 SS (A. L.)	0.142	0.114
15C _{III}	AISI 406 (Carp)	0.160	0.241
20	Fe-30 Cr-1Y	0.373	1.827
22	Hastelloy X-280	0.397	0.229
24	Hastelloy R-235	1.563	0.991
25	Hastelloy-N	0.241	0.256
26	Inconel-600	0.805	0.392
29	PDRL-102	0.227	0.387
32	AISI 304 SS	3.327	0.676
33	AISI 316 SS	1.851	2.903

15^{III} AISI 406 SS
(A.L.)

15^{CII} AISI 406 SS
(Carp.)

20 Fe-30Cr-1Y

22 Hastelloy-X

24 Hastelloy R-235

25 Hastelloy-N

26 Inconel

29 PDRL-102

32 AISI 304 SS

33 AISI 316 SS

FIGURE 1

Iron- and Nickel-Base Alloys Exposed for 500 hr at 1060 °C
to Flowing Helium (0.75 ft³/hr) After Passing Through Graphite
at 450 to 500 °C

Neg. 061787-1

15_{III} AISI 406 SS
(A.L.)

15_{CII} AISI 406 SS
(Carp.)

20 Fe-30 Cr-1Y

22 Hastelloy-X

24 Hastelloy R-235

25 Hastelloy-N

26 Inconel

29 PDRL-102

32 AISI 304 SS

33 AISI 316 SS

FIGURE 2

Iron- and Nickel-Base Alloys Exposed for 500 hr at 1060 °C
to Flowing Nitrogen (0.75 ft³/hr). After Passing Through Graphite
at 450 to 500 °C

TABLE III
CORROSION RATES OBTAINED
ON SAMPLES EXPOSED IN BOILER BURNING BUNKER C FUEL OIL *

Temperature Range	565 to 850 °C
Duration of Test	648 hr
<u>Material, AISI Type</u>	<u>Corrosion Rate, in/yr</u>
406 (12 Cr-4 Al)	0.108
446 (28 Cr)	0.189
302 (18 Cr-8 Ni)	0.406
410 (12 Cr)	0.555
321 (18 Cr-8 Ni-Ti)	0.690
431 (17 Cr-2 Ni)	0.906

* Reference 1

TABLE IV
CORROSION RATES OF Fe-Cr-Ni AND Fe-Cr-Al ALLOYS
IN H₂S AT ELEVATED TEMPERATURE *

Temperature	400 °C	
Hydrogen Pressure	485 psig	
H ₂ S Concentration in H ₂	1.5%	
Length	730 hr	
<u>Alloy</u>	<u>Composition</u>	<u>Mils/yr</u>
Type 304	18 Cr-9 Ni	7.8
Type 347	18 Cr-11 Ni	6.4
Type 309	22 Cr-14 Ni	4.9
Incoloy	21 Cr-34 Ni	2.9
Inco 804	30 Cr-42 Ni	5.0
Inconel	15 Cr-78 Ni	10.1
Type 406	12 Cr-4 Al	Nil
Croloy 20-2 Al	20 Cr-2 Al	3.0

* Reference 1

The results of this test series, which are plotted in $\text{mg}/\text{dm}^2/\text{hr}$ weight gain against temperature in Figure 3, agree with the results of the earlier test series, except for generally lower weight gains of the present test series. The lower weight gain may be because of the graphite preventing free access of CO_2 gas to one-half of the specimens. It is interesting to note that between 870 and 980 °C, the Hastelloy alloys X and R-235 show a more rapid weight gain than the other alloys. The Carpenter experimental Ferral alloy (14C) has the highest initial gain up to 700 °C, followed by a decrease to 980 °C, and a steep increase above that temperature. Between 980 and 1090 °C, alloys D979, Inconel 600, Hastelloy X and Hastelloy R-235, after leveling off, reverse their previous trend and experience a sharp drop in weight gain. The remaining three alloys, Inconel X-750, Hastelloy F, and at a much slower rate, AISI 406 SS, continue to gain weight. No explanation can be given presently for this difference in behavior and especially for the difference of such closely related alloys as Hastelloy X and F, or Inconel 600 and Inconel X-750. The commercial AISI 406 SS (Carpenter Alloy 1-JR) shows a continuous moderate weight gain from a minimum of 0.025 to a maximum of 0.60 $\text{mg}/\text{dm}^2/\text{hr}$ over a temperature range of 700 to 1090 °C. A decrease in weight gain was found in the temperature range of 540 to 700 °C.

Tubular sections of Hastelloy X, Hastelloy X-280, and AISI 406 SS (Carpenter alloy 1-JR) were exposed to dry purified CO_2 at 816, 927, and 1025 °C up to 400 hr (Figure 4). Flow rate of the gas was 0.5 ft^3/hr . At 816 °C, the AISI 406 SS alloy has a relatively low weight gain over the exposure times tested. At 1023 °C, an almost opposite effect (relatively extreme weight gains) noted for the AISI 406 SS alloy as compared to the Hastelloy X seamless tubing tested at the same temperature. In comparing the Hastelloy X and Hastelloy X-280 specimens tested at 927 °C, the Hastelloy X-280 had significantly better oxidation resistance.

Susceptibility to stress-corrosion cracking of fuel cladding candidate materials is a severe deterrent to nuclear superheat applications of such alloys. Stress-corrosion susceptibility was determined by

FIGURE 3

Weight Gain of Iron- and Nickel-Base Alloys in Contact with Powdered Graphite During 100 hr Exposure to Pure, Dry Carbon Dioxide

FIGURE 4

Weight Gain (mg)
of Hastelloy X, Hastelloy X-280 and AISI 406 SS (Carp. 1-JR Seamless) Tube Sections
Exposed to Dry Purified Carbon Dioxide at 1500 °F (816 °C),
1700 °F (927 °C) and 1875 °F (1025 °C),
at a Flow Rate of 0.5 ft³/hr

immersing, plastically and elastically, prestressed strips of the alloys in a 42% boiling magnesium chloride solution (Figure 5). The 200 and 300 types of austenitic stainless steel fractured in less than 20 hr of exposure, while the 400 and 500 types of stainless steel, the Fe-Cr-Al-base alloys and the nickel-base alloys, did not fracture even after 150 hr immersion. The Fe-Cr-Al experimental alloys were partially corroded by localized or edge attack. Hastelloy X, PDRL-102, AISI 406 SS, and Incoloy-800 appear to be most resistant to stress corrosion.

A series of corrosion tests were performed by the Chemical Metallurgy section of the Pacific Northwest Laboratory using an autoclave of about 8 liters volume, and a design capacity of 670 °C and 5000 psi pressure. The tests were carried out in deionized, oxygenated (3 to 4 ppm O₂) and deoxygenated (0.05 ppm O₂) steam at 550 °C and at 1000, 3000, and 5000 psi pressure. Make-up water was added during the tests at a rate of 1 liter/hr. The corrosion weight gain results are listed in Table V and are plotted as a function of logarithm exposure time in Figure 6a. Superheated steam corrosion tests have also been carried out at 650 °C under 600 psi pressure by Greenberg and associates of Argonne National Laboratory with a comparable group of alloys.⁽³⁾ The results of these tests are summarized in Table VI and are shown in Figure 6b. From these data, it appears that the alloys tested may be placed into three distinct groups of different corrosion resistance up to 1000 hr exposure time. The austenitic stainless steels 304 and 316 and the ferritic stainless steels 406 and 430 are least resistant with weight gains of 50 to 300 mg/dm² (0.2 mil penetration). The Fe-25 Cr-3 Al-0.6 Y, AISI 304, 309, 310, Hastelloy N, PDRL-102, and Inconel X-750 have weight gains from about 8 to 75 mg/dm² (0.1 to 0.2 mil penetration); and those with the most resistance, Incoloy 800, 446 SS, Hastelloy X-280, Hastelloy R-235, and Fe-24 Cr-5 Al have weight gains below 15 mg/dm² (0.1 mil penetration).

FIGURE 5

Stress Corrosion Test. Plastically and Elastically Prestressed
Alloy Specimens (0.394 x 3.94 x 0.010 in.)
Immersed in 42 wt% MgCl_2 Aqueous Solution at 154 °C.

Neg. 062173-2

FIGURE 6a

Weight Gain of Iron- and Nickel-Base Alloys
in Superheated Steam of 550 °C
and (1000), 3000 psi Pressure (mg/dm²)

FIGURE 6b

Weight Gain of Iron- and Nickel-Base Alloys
in Oxygenated, and Oxygen Free, Superheated Steam
of 650 °C and 600 psi Pressure (mg/dm²)

TABLE V

WEIGHT GAIN OF IRON- AND NICKEL- BASE ALLOYS IN SUPERHEATED STEAM, mg/dm^2 ,
 AT 550 °C (1022 °F) AND 1000 OR 3000 psi PRESSURE

ALLOY	EXPOSURE HOURS	72	78	96	168	174	262	336	340	599	678	720	984	1062
	PRESSURE, psi	1000	3000	3000	1000	3000	3000	1000	3000	3000	3000	1000	3000	3000
ASTM A 212-B		260.0			305.0			730.0			915.0			
AISI 304 SS		62.0	64.0		78.0	94.0		117.0	121.0		160.0	219.0		220.0**
AISI 430 SS		(141.)	124.0		(160.)	156.0		(209.)	177.0		196.0	(226.)		240.0
AISI 446 SS		3.2	5.3		4.1	7.6		6.3	9.4		10.6	6.7		11.0
FERRAL			124.0			186.0			250.0		342.0			583.0
AISI 406 SS (AL)		(155.)	121.0		(186.)	148.0		(259.)	167.0		185.0	(280.)		219.0
AISI 406 SS (Carp.)			45.0			92.0		119.0			143.0			190.0
AISI 406 SS (Carp.) Wd*			143.0			181.0		212.0			247.0			310.0
Fe-24CR-5.6 Al.			2.2			3.5			5.0		10.0			(56.)
Fe-25CR-3 Al-0.6Y			22.0			34.0			43.0		52.0			73.0
Hastelloy X-280				4.5			6.0			9.0			13.0	
Hastelloy N			8.0			12.0			16.0		51.0			57.0
Inconel X			9.0			11.0			14.0		18.0			26.0
Incoloy				6.4			8.4			11.0			14.0	
PDRL-102			8.5			10.5			12.4		31.0			42.0

* Weldrawn

** 304-L SS

TABLE VI

ARGONNE NATIONAL LABORATORY

BEHAVIOR OF STAINLESS STEELS WITH DIFFERENT SURFACE TREATMENTS IN OXYGENATED AND OXYGEN FREE SUPERHEATED STEAM

AT 650 °C (1202 °F) AND 600 psi PRESSURE - WEIGHT CHANGE IN mg/dm²

Exposure Hours	67	120	139	144	161	283	300	379	422	583	706	744	820	888	1176	1320	1471	1519	1896	2184	2229	2328	2507	2602	2640	2760	3360
Days	2.8	4.9	5.8	6	6.7	11.8	12.5	15.8	17.6	24.3	29.4	31	34.2	37	49	55	61	63.3	79	91	93	97	103	108.4	110	115	140
ALLOY																											
AISI 304 SS																											
Wet Ground	3				45		-197			-563								-732					-1002		-1011		
Pickled		(240)				-309		(312)	-246		-200		(*)					-335	(*)		(*)			-668		-678	
Wet Grd. Pickled				-223										-679				-586									
AISI 304-L SS			-46						-29							69						103					
AISI 309 SS																											
Wet Ground						27								42						38		38					
Pickled						18								26						19		20					
Wet Grd. Pickled				55								57						51									
AISI 310 SS																											
Wet Ground						0								26						25		26					
Pickled						90								104						103		104					
Wet Grd. Pickled				9.6								34						33									
AISI 316 SS																											
Wet Ground	0	(50)			5		52	(69)		171			(*)				299	(*)		(*)			270		261		
Pickled						-260			-138		52							-734						-863		-840	
Wet Grd. Pickled				118										-688				-1154									
AISI 347 SS																											
Wet Ground	0	(45)			3.6		53	(83)		78			(108)				152	(129)		(140)			218		230		
Pickled						-129			-107		-100							-68						-57		-53	
Wet Grd. Pickled				131										235					278								
AISI 406 SS																											
AL OHMALOY	134	(136)			139		144	(143)		139			(149)				145	(150)	(150)		(153)		145		145		150
AISI 406 SS																											
Carp. 1-JR	-2.6	(1.8)				0	2.6	(5.3)		0			(11.4)				8.6	13.6		(16)			6		8		12
AISI 406 SS																											
Seamless Tube			32						35							38						38	38				39
Weldrawn Tube			121						132							139						139	140				143

() Oxygen free steam

(*) Severe spalling

SOURCE OF DATA: Private Communication with S. Greenberg, Argonne National Laboratory - Metallurgy Division

The oxygen content of the steam has a varying effect on the corrosion rate of different groups of alloys. Ferritic stainless steels and Fe-Cr-Al-base alloys offer better resistance in oxygenated steam. The opposite applies to the austenitic and martensitic stainless steels and to the nickel-base alloys which are more resistant to deoxygenated steam.

A comparison of the data given in Tables V and VI indicates a slight decrease in the corrosion rate when the steam pressure is increased from 1000 to 3000 psi. The corrosion penetrations at the various exposures were calculated from the weight gains. Agreement between the calculated and measured penetrations was found by stripping the oxide from the samples following their exposure to the 1000 psi superheated steam.

To determine the actual corrosion, the oxides formed on the surface must be removed to allow accurate weight loss measurements of the base metal. Oxide films formed on iron- and nickel-base alloys during exposure to water or steam up to a temperature of 400 °C can be readily removed. The removal of oxides formed during exposure to 550 °C steam has been much less satisfactory.

A stripping procedure, originated by the Stellite Division of the Union Carbide Corporation, was successfully applied to iron- and nickel-base alloys exposed to 550 °C and 3000 to 5000 psi steam for 100 days. The total corrosion weight losses found after stripping were converted into mils of penetration.

Incoloy 800, Hastelloy X, Inconel X-750, Hastelloy N, PDRL-102, Hastelloy R-235, Nichrome V, and AISI 446 SS all had corrosion penetration depths of 0.1 mil after 100 days of exposure (Figure 7) to 550 °C, 5000 psi deoxygenated steam at the Pacific Northwest Laboratory.⁽⁴⁾

A general trend of lower corrosion rates for the high pressure case is observed when comparing the high pressure data to the lower 3000 psi data. Similar comparisons of corrosion rates of deoxygenated steam and oxygenated steam are given (Figure 8) and only small differences in corrosion rate were noted.

FIGURE 7

Descaled Corrosion Penetrations of Various Alloys
Following 100 Days in 3000 psi and 5000 psi 500 °C Deoxygenated Steam

FIGURE 8

Corrosion of Various Alloys
Following 100 Days in 550 °C - 3000 psi Steam

Tube Burst Tests at Ambient and Elevated Temperatures

Sections of welded, weldrawn and seamless tubes of various stainless steels, Incoloy 800, and Zircaloy-2 were hydrostatically burst tested at atmospheric and elevated temperatures. The burst pressures and hoop tension stresses for test temperatures between room temperature and 400 °C are listed in Table VII and shown in Figure 9. According to these test results the AISI 406 SS, weldrawn tubes have a higher average strength, i.e., a less pronounced temperature dependent decrease of hoop tension stress than either the austenitic stainless tubes or the Incoloy 800 and Zircaloy-2 tubes of comparable dimensions. The Zircaloy-2 tubes suffer a severe loss in strength between ambient and 200 to 220 °C, and a less severe loss between 300 and 400 °C test temperatures. None of the burst fractures occurred in the welded end closures or the welded areas of any of the tubes.

Weldrawn 406 SS tube sections (0.563 in. OD x 0.010 in. wall x 8 in. long) with tungsten inert gas (TIG) arc welded 406 SS end caps and magnetic force resistance welded 406 SS end caps were internally pressurized. This pressurizing was done primarily to test the quality of the welds.

All of the five specimens with arc welded end caps, which were pressure tested, failed in the tube section (Figure 10). These tubes burst at an average pressure of 3150 psi, which is approximately the tensile strength of the weldrawn 406 SS tubes in the "as-received" condition. There were no indications of leaks or failures in any of the end cap welds before or during the burst tests.

The pressure test results of the tube sections with magnetic force resistance welded end caps, (Figure 11) indicate that four specimens failed in the end cap welds at lower pressures and four fractured in the tube walls at practically the same average pressure as those of the first test series. These test results indicate that satisfactory end cap welds can be obtained by either TIG-arc or magnetic force resistance welding of 406 SS end caps to weldrawn 406 SS tubing.

TABLE VII

HYDROSTATIC BURST PRESSURES AND HOOP TENSION STRENGTHS
OF THINWALLED, SEAMLESS, AND WELDRAWN ALLOY TUBES *

<u>MATERIAL</u>	<u>HEAT NUMBER</u>	<u>DIMENSIONS OF TUBE, in.</u>	<u>TEST TEMPERATURE, °C</u>	<u>BURST PRESSURE, lb</u>	<u>HOOP STRESS, lb</u>
AISI 304 SS (Trentweld)	890409	0.562 OD x 0.035 wall	RT 280 300 320	9,455 7,055 7,060 7,260	71,700 53,500 53,500 54,000
AISI 304 SS (Low Co, Seamless)	46657	0.566 OD 0.503 ID x 0.031 wall	RT 291 320 360	12,150 9,800 9,660 9,560	110,000 79,500 79,600 79,000
AISI 304 SS (Seamless)		0.561 OD x 0.009 wall	RT 276 314 386	2,165 1,595 1,570 1,520	73,700 54,400 53,500 51,800
AISI 316 SS (Seamless) (High P - INCO)		0.434 OD x 0.0312 wall	RT 177 340	12,280 10,310 9,500	83,000 64,000 59,500
AISI 406 SS (Carpenter-Superior Weldrawn Tubing)		0.505 OD x 0.030 wall	RT 207 218 320 329	12,270 10,265 10,200 10,000 9,800	92,100 78,650 78,100 76,600 75,100
AISI 406 SS (Carpenter-Superior Weldrawn Tubing)		0.563 OD x 0.010 wall	RT 200 24	3,070 2,543 2,457	95,850 79,330 76,633
AISI 406 SS PO (NTH Welded Tubing)	77578	1 OD x 0.021 wall	RT 280	4,015 3,835	101,000 87,400
AISI 406 SS PO (Welded Tubing)	70120	0.563 OD x 0.020 wall	RT 275	6,600 6,240	84,000 79,500
AISI 406 SS (Weldrawn) PO	X-52600 70120	0.563 OD x 0.010 wall	RT 276	3,645 3,070	99,000 83,300
AISI 406 SS (Weldrawn)		0.580 OD x 0.010 wall	RT 176 300	3,965 3,473 3,248	111,067 97,200 91,033
AISI 406 SS Carpenter	803549	0.505 OD x 0.010 wall	RT 354	3,825 2,910	108,000 82,000
AISI 406 SS Carpenter	46012	0.505 OD x 0.10 wall	RT 346	3,560 3,220	100,000 90,600
Incoloy		0.3125 OD x 0.020 wall	RT 176 348	11,520 10,170 9,430	78,550 69,250 64,250
Zircaloy-2 (Low Ni)		0.495 ID x 0.035 wall	RT 228 305 342 370	12,413 7,480 6,195 6,013 5,910	103,600 63,100 54,700 50,000 48,700
Zircaloy-2		0.505 ID x 0.020 wall	RT 235 300 320 340 355 370 390	11,150 6,240 5,535 5,290 5,170 5,140 4,790 4,710	112,533 62,933 56,100 53,300 52,166 51,800 48,300 47,500

* Average values from 2 to 3 test results by P. M. Jackson, Pacific Northwest Laboratory

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BNWL-154

NWL-1

355	5,140	51,800
370	4,790	48,300
390	4,710	47,500

FIGURE 9

Hydrostatic Burst Pressures and Hoop Tension Strengths
of Thin Walled, Seamless and Weldrawn Alloy Tubes

FIGURE 10
TIG Arc-Welded End Caps of Weldrawn AISI 406 SS
Thin-Walled Tube Sections Burst Tested

Irradiation Exposure Effects on Structural and Mechanical Properties

The effects of neutron irradiation on the mechanical properties of four promising nickel-base alloys and four stainless steels were determined by testing subsize tensile specimens. Tensile tests were performed both before and after irradiation in a 650 °C helium-graphite environment to exposures of 2.53×10^{19} nvt and 1.21×10^{20} (E > 1 Mev) nvt, respectively. The unirradiated control specimens were placed on graphite supports in sealed, evacuated, and helium-refilled stainless steel containers and heated at 650 °C to simulate reactor environment exposure conditions. The four nickel-base alloys include Hastelloy X-280, Hastelloy R-235, Inconel 600, and Inconel 702. The four stainless steel alloys include AISI 316, 347, and 406, and a modified ferral alloy (Tables VIII, IX).

Figure 12 shows the properties obtained for the solution hardenable alloy, Hastelloy X-280. In the room temperature tests, a loss of yield strength was observed at both exposure levels while little or no change in ductility was found as measured by the percent total elongation. In the 700 °C tests, only slight increases in yield strength were found and they amounted to about 10% for both exposures. The ultimate strength decreases at the higher exposure with no change at the lower exposure, but wide scatter in the data makes analysis difficult. The decrease in the ductility of this alloy is of extreme consequence. A 48% elongation was decreased to 25% and 5%, respectively, after the two irradiation exposures and little if any changes in hardness were found.

Data for Hastelloy R-235, an age hardenable alloy, are shown in Figure 13. Room temperature properties of this alloy are similar to the Inconel 702 (Figure 14), but the ductility properties are generally lower. In the 700 °C tests, the Hastelloy R-235 retained its ductility in the unirradiated condition. However, the percent elongation decreased with increasing exposure and was only about 4% after an exposure of 1.2×10^{20} nvt (E > 1 Mev). Only small increases

FIGURE 11
Magnetic Force Resistance Welded End Caps
of Weldrawn AISI 406 SS Thin-Walled Tube Sections Burst Tested

TABLE VIII
EFFECTS OF NEUTRON IRRADIATION IN CONTAMINATED HELIUM ATMOSPHERE AT 650 °C (1202 °F) MAX.
UPON THE ROOM TEMPERATURE HARDNESS AND TENSILE PROPERTIES
AT ATMOSPHERIC AND ELEVATED TEMPERATURE OF SOME NICKEL-BASE SUPERALLOYS

ALLOY	UNIRRADIATED				IRRADIATED				IRRADIATED			
	HARDNESS R _B	YS [*] 1000 psi	US ^{**}	ELONG. %	2.53 x 10 ¹⁹ nvt (E > 1 Mev)				1.21 x 10 ²⁰ (E > 1 Mev)			
					HARDNESS R _B	YS 1000 psi	US	ELONG. %	HARDNESS R _B	YS 1000 psi	US	ELONG. %
TESTED AT ROOM TEMPERATURE												
Hastelloy X	92	52	116	30	89	33	108	34	91	46	105	25
Hastelloy R-235	105	91	158	17	100	97	159	19	105	91	143	13
Inconel 600	77	36	94	42	80	47	95	43	70	34	94	37
Inconel 702	100	97	155	27	94	97	154	31	101	99	138	14
TESTED AT 700 °C (1292 °F)												
Hastelloy X	89	36	72	48	82	42	64	13	93	40	58	9
Hastelloy R-235	105	85	127	19	101	85	117	7	103	87	88	4
Inconel 600	76	28	46	29	72	32	70	16	71	32	33	3
Inconel 702	102	87	95	5	103	55	65	5	100	11	21	17

* YS - Yield Strength

** US - Ultimate Strength

TABLE IX

EFFECTS OF NEUTRON IRRADIATION IN CONTAMINATED HELIUM ATMOSPHERE AT 650 °C (1202 °F) MAX.
UPON THE ROOM TEMPERATURE HARDNESS AND TENSILE PROPERTIES
AT ATMOSPHERIC AND ELEVATED TEMPERATURE OF SOME IRON-BASE ALLOYS

UNIRRADIATED					IRRADIATED			
ALLOY	HARDNESS* R _B	YS 1000 psi	US 1000 psi	ELONG. %	HARDNESS* R _B	2.53 x 10 ¹⁹ nvt (E>1 Mev)		
						YS 1000 psi	US 1000 psi	ELONG. %
TESTED AT ROOM TEMPERATURE								
AISI 316 SS	66	30	81	62	63	27	78	56
AISI 347 SS	76	53	100	44	72	40	95	37
AISI 406 SS	87	67	89	23	--	70	91	25
FERRAL MOD.	95	74	98	16	95	77	77	3
TESTED AT 700 °C (1292 °F)								
AISI 316 SS	67	13	36	48	66	25	67	46
AISI 347 SS	75	33	43	34	74	39	68	27
AISI 406 SS	86	14	14	74	76	31	42	54
FERRAL MOD.	90	16	19	53	100	42	54	29

* Hardness tested at room temperature

FIGURE 12

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum on the RT Hardness, and the Tensile Properties of Hastelloy X at Atmospheric and Elevated Test Temperature

FIGURE 13

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum
on the RT Hardness and the Tensile Properties of Hastelloy R-235
at Atmospheric and Elevated Test Temperature

FIGURE 14

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum
on the RT Hardness and the Tensile Properties of Inconel 702
at Atmospheric and Elevated Test Temperature

in yield strength were observed in the material that maintained a value of 83,000 psi throughout the irradiation. The ultimate tensile strength decreased from 125,000 psi (prior to irradiation) to 90,000 psi after the longer exposure. A slight increase in room temperature hardness was found at the higher exposure.

Data obtained from Inconel 600, a solution hardenable alloy, are illustrated in Figure 15. Results somewhat similar to the Hastelloy X-280 were obtained for this alloy in that changes in mechanical properties were small for the room temperature tests. However, tests at 700 °C showed a marked decrease in ductility. After an exposure of 1.2×10^{20} nvt ($E > 1$ Mev) only a 3 to 5% elongation remains, compared to 28% elongation prior to irradiation. In addition, an increase in ultimate tensile strength was found at the lower exposure, and a decrease in room temperature hardness was noted at the higher exposure. A marked decrease in the ultimate strength of Inconel 600 was observed at the higher exposure and may be associated with the loss in ductility as indicated by the decrease in percent elongation.

The results for an age hardenable alloy, Inconel 702, are shown in Figure 14. The room temperature tests indicated that a very slight increase in yield strength (about 3%) occurs for exposures up to 1.2×10^{20} nvt, while an 11% decrease in ultimate tensile strength occurs. The percent elongation increased about 12% at the lower exposure, but decreased again with higher exposure to 19% below the preirradiation value. At the 700 °C test temperature, the preirradiation elongation of only 5% was maintained at both exposure levels. Drastic effects on the strength properties of the alloy were discovered. The yield strength decreased from about 90,000 psi in the unirradiated case, to 55,000 psi for the lower exposure. A further decrease to only 5000 psi was found after an exposure of 1.2×10^{20} nvt, and similar decreases in the ultimate strength values were noticed.

The stainless steels irradiated at 650 °C in a helium-graphite environment and tested at room temperature and 700 °C were AISI 406,

FIGURE 15

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum
on the RT Hardness and the Tensile Properties of Inconel 600
at Atmospheric and Elevated Test Temperature

316, and 347. These alloys were irradiated to one exposure, 2.53×10^{19} nvt ($E > 1$ Mev) (Figures 16 to 18), and the data collected are similar in many ways to the nickel-base alloy data.

Figure 16 shows the effects of the irradiation and environment upon AISI 406. As in the nickel-base alloys, only small changes in properties are found in the ambient or room temperature tests where the yield strength increases only about 10%. The 700 °C tests disclosed significant changes in strength and ductility. After irradiation, a yield strength of about 15,000 psi was measured as compared to about 8000 psi in the unirradiated condition. Percent total elongation remained above 53% after the irradiation which indicated the material was still quite ductile. Very little change in hardness accompanied the increase in strength values.

Tensile test results on AISI 316 stainless steel disclosed little change in the room temperature properties after irradiation (Figure 17). After testing at 700 °C, significant increases in yield and ultimate strength of the material were noted, with no important changes in material hardness or ductility. Similar results found after irradiation for AISI 347 (Figure 18) were a 12% increase in yield strength and a 60% increase in ultimate strength at 700 °C. The total elongation of this alloy (AISI 347) at 700 °C remained above 20% after irradiation as compared to a 35% total elongation prior to irradiation.

Tensile specimens of the same series of stainless steel and nickel-base alloys were charged in a sealed aluminum tube, placed into a reactor coolant water channel testing facility, and irradiated at 20 to 30 °C to an exposure of 9×10^{20} nvt ($E > 1$ Mev). The room temperature and 700 °C hardness and tensile properties of these materials irradiated under the above conditions are listed in Tables X and XI and plotted on bar charts in Figures 19 to 25.

The Hastelloy X room temperature tests show a large (93%) increase in the yield strength of the material and a smaller increase (12%) in the ultimate strength (Figure 19). Accompanying these increases in material strength is a substantial decrease (34 to 23%) in ductility as measured by percent total

FIGURE 16

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum on the RT Hardness, and the Tensile Properties of AISI 406 SS at Atmospheric and Elevated Test Temperature

FIGURE 17

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum on the RT Hardness and the Tensile Properties of AISI 316 SS at Atmospheric and Elevated Test Temperature

FIGURE 18

Effects of Neutron Radiation in Contaminated Helium at 650 °C Maximum
on the RT Hardness and the Tensile Properties of AISI 347 SS
at Atmospheric and Elevated Test Temperature

TABLE X

EFFECTS OF NEUTRON IRRADIATION IN REACTOR COOLANT WATER OF 20 - 30 °C (68 - 86 °F)
ON THE HARDNESS AND TENSILE PROPERTIES
AT AMBIENT AND ELEVATED TEMPERATURE OF SOME NICKEL-BASE ALLOYS

ALLOY	UNIRRADIATED				IRRADIATED 9 x 10 ²⁰ nvt (E >1 Mev)				
	HARDNESS R _B	YS 1000 psi	US	ELONG %	HARDNESS R _B	YS 1000 psi	US	ELONG. %	RA %
TESTED AT ROOM TEMPERATURE									
Hastelloy X	94	55	116	34	62	106	130	23	
Hastelloy R-235	104	99	168	28	61	125	132	15	
Inconel-600	98	31	80	61	63	133	135	15	
Inconel-702	99	79	145	33	60	138	138	11	
TESTED AT 700 °C (1292 °F)									
Hastelloy X	91	36	70	39	81	42	56	3	15
Hastelloy R-235	105	99	104	31	101	92	93	23	14
Inconel	98	65	67	11	96	72	72	21	8
Inconel - 702	100	75	84	13	100	60	60	20	1

TABLE XI

EFFECTS OF NEUTRON IRRADIATION IN REACTOR COOLANT WATER OF 20 - 30 °C (68 - 86 °F)
 ON THE HARDNESS AND TENSILE PROPERTIES
 AT AMBIENT AND ELEVATED TEMPERATURE OF SOME IRON-BASE ALLOYS

ALLOY	UNIRRADIATED				IRRADIATED				
	HARDNESS R _B	YS 1000 psi	US 1000 psi	ELONG. %	9 x 10 ¹⁹ nvt (E >1 Mev)				
					HARDNESS R _B	YS 1000 psi	US 1000 psi	ELONG. %	RA %
<u>TESTED AT AMBIENT TEMPERATURE</u>									
AISI 304 SS	100	77	112	38	62	132	132	13	
AISI 347 SS	99	66	101	39	55	115	117	19	
AISI 406 SS	86	59	78	23	54	89	93	13	
<u>TESTED AT 700 °C (1292 °F)</u>									
AISI 304 SS	95	39	42	25	97	47	48	25	11
AISI 347 SS	97	42	45	24	94	45	46	43	29
AISI 406 SS	83	11	12	68	81	12	13	120	96

FIGURE 19

Effects of Neutron Irradiation in Reactor Coolant Water of 20 to 30 °C
on the RT Hardness and Tensile Properties of Hastelloy X
at Ambient and Elevated Temperature

FIGURE 20

Effects of Neutron Irradiation in Reactor Coolant Water of 20 to 30 °C
on the RT Hardness and Tensile Properties of Hastelloy R-235
at Ambient and Elevated Temperature

FIGURE 21

Effects of Neutron Irradiation in Reactor Coolant Water of 20 to 30 °C
 on the RT Hardness and Tensile Properties of Inconel 600
 at Ambient and Elevated Temperature

FIGURE 22

Effects of Neutron Irradiation in Reactor Coolant Water of 20 to 30 °C
on the RT Hardness and Tensile Properties of Inconel 702
at Ambient and Elevated Temperature

FIGURE 23

Effects of Neutron Radiation in Reactor Coolant Water of 20 to 30 °C upon the RT Hardness, and the Tensile Properties of AISI 406 SS at Atmospheric and Elevated Temperature

FIGURE 24

Effects of Neutron Radiation in Reactor Coolant Water of 20 to 30 °C upon the RT Hardness, and the Tensile Properties of AISI 304 SS at Atmospheric and Elevated Test Temperature

FIGURE 25

Effects of Neutron Irradiation in Reactor Coolant Water of 20 to 30 °C on the RT Hardness and Tensile Properties of AISI 347 SS at Ambient and Elevated Temperature

elongation. Tests performed at 700 °C indicate a smaller percentage increase (16%) in yield strength and an actual decrease (20%) in ultimate strength. This decrease in ultimate strength may be associated with the marked decrease (39 to 3%) in ductility as measured by the percent elongation.

The changes in mechanical properties of Hastelloy R-235 due to irradiation are shown in Figure 20. The room temperature data indicate a 25% increase in yield strength after an exposure of 9×10^{19} nvt at ambient temperature. In contrast to the Hastelloy X, there was a decrease in ultimate strength, and the ductility decreased from 28 to 15% in total elongation after the irradiation.

Smaller changes in the mechanical properties of Hastelloy R-235 were found at the 700 °C test temperature. Decreases in both yield and ultimate strength of the irradiated material were evident as compared to the unirradiated control specimens. Also, while a 20% decrease in percent elongation was noted, a 23% elongation remained after an exposure of 9×10^{10} nvt.

Figures 21 and 22 show the data for Inconel 600 and 702 tested at room temperature and 700 °C after an exposure of 9×10^{19} nvt ($E > 1$ Mev) at ambient temperatures. The room temperature data for both alloys show substantial increases in strength and decreases in percent elongation because of the irradiation. The postirradiation ductility, as measured by percent total elongation of both Inconel 600 and 702, were about the same, 11 and 13%, respectively. As expected, the preirradiation value for the solution hardenable alloy, Inconel 600, was substantially higher (61%) than that of the precipitation hardenable alloy, Inconel 702 (33%).

In comparing the 700 °C test data for Inconel 600 and 702, a significant difference in the strength properties exist. Irradiation caused a small increase in both the yield and ultimate strength of the Inconel 600 material while a decrease in these same properties occurred in Inconel 702. The apparent improvement in ductility of both alloys as evidenced by the increase in percent total elongation is of particular

interest. The lower value of reduction in area indicates that the materials did not exhibit a large amount of plastic instability prior to fracture; nor did they exhibit any strain hardening at this test temperature.

In recent years, interest has increased in the mechanical behavior of the ferritic and austenitic stainless steels at elevated temperatures. AISI 406, 304, and 347 stainless steels have been irradiated at ambient temperatures to an exposure of 9×10^{19} nvt ($E > 1$ Mev) and tested at room temperature and 700°C (Figures 23 to 25). Comparison of the room temperature irradiated and control data shows an increase in yield and ultimate strength for the three materials. Both austenitic steels increased in yield strength by about 70%. The AISI 406 ferritic stainless increased in yield strength by about 50%. The percent elongation to fracture for all the materials was sharply reduced by factors of 2 to 3.

The changes in mechanical strength properties of the stainless steel material tested at 700°C were generally smaller. In comparing the irradiated and unirradiated control data, changes in the yield strength data ranged from 9% for the AISI 406 material to about 20% for the AISI 304. Similar percent changes in ultimate strength were also found. The elevated temperature ductility data was especially revealing. In the AISI 406, the unirradiated control specimen had a 68% elongation at fracture. After irradiation to an exposure of 9×10^{19} nvt, the value increased to 120% which showed a marked increase in material ductility. A similar increase in percent elongation was attributed to AISI 347 where the pre- and post-irradiation values were 24 and 43%, respectively. The pre- and postirradiation values of percent elongation for AISI 304 were 25% and revealed no change because of irradiation.

Effects of Irradiation on the Erichsen Cup Test Results

Mechanical behavior under biaxial loading after exposure to irradiation is an essential alloy characteristic for thin-wall, fuel cladding materials. Mechanical behavior can be determined by the so-called Erichsen cupping test.

In the modified Erichsen-Olsen Ductomatic Testing Machine, a spherical punch forms a cup-shaped penetration in a tightly clamped, disc-

shaped, 10- to 20-in. specimen of 2 1/2 in. in diameter until fracture occurs. Two control specimens of each alloy were tested at room temperature, both in the unirradiated condition and after irradiation, to an exposure of 3.45×10^{20} nvt ($E > 1$ Mev) in a reactor, water-cooled, test facility at 20 °C (Figures 26, 27). Serrations at the top of the specimens were for specimen identification (Figures 26, 27).

The duplicate test values agreed favorably with each other except for some of the Fe-Cr-Al-base alloys that have a wide scatter range. Usually, neutron flux exposure under the listed environment conditions decreases both the stress to fracture and the depth of the cup. Two Fe-Cr-Al-base alloys, namely the Martin Company's experimental DB3-35 and DB3-42 alloys, show increases of these properties in the irradiated condition.

CONCLUSIONS

The data in this report conclude this particular series of tests. It was the purpose of Part I of the report to evaluate the behavior of several stainless steel, iron-chromium-aluminum-base alloys, and nickel-base precipitation and solution-hardenable alloys, in elevated-temperature, simulated, reactor environments. The conclusions drawn from Part I are as follows:

- For reactor structural and core components exposed to dry carbon dioxide as coolant gas up to 950 °C and low pressures, the Hastelloys, X and N, preferably with low cobalt contents, are the most promising materials. They are closely followed in oxidation resistance by Hastelloy F, PDRL-102 alloy, and the Fe 30 Cr 1Y alloy.
- Some of the Fe-Cr-Al-base alloys, including AISI 406 SS, offer very good oxidation and carburization resistance up to 816 °C. Their medium neutron capture cross section, and resistance to intergranular and stress corrosion make these alloys especially attractive for reactor applications where their elevated temperature strength is sufficient.
- Both the austenitic and ferritic stainless steels, except AISI 316 SS and AISI 347 SS, are attacked rapidly by CO₂ at temperatures

FIGURE 26

Effects of Radiation upon the Erichsen Cup Text Results of Cold Rolled and Annealed .010 in. Thick Specimens of Various Types of Alloys

FIGURE 27

Interior of Cup Fracture of Irradiated Cup Tested
Disc of D-979 Alloy

Neg. 0612088

above 700 °C. The oxidation resistance of Inconel alloys is better than that of the stainless steels and the ferral alloys; however their high nickel content and neutron capture cross section are handicaps.

- Alloys offering maximum resistance to corrosion in deionized water of pH 10 to 7 at 290 to 360 °C and 1500 to 3200 psi pressure include the austenitic AISI 304 SS and 347 SS, the ferritic AISI 406 SS and 446 SS, Hastelloy X, and the Fe-Cr-Al-base alloys.
- Manganese, nickel, and cobalt, which are high neutron absorbers, often cause objectionably high neutron capture cross sections and postirradiation gamma activities in the austenitic stainless steels and in the nickel-base alloys. Moreover, they are strategic elements. Fe-Cr-Al-base alloys, however, are made from the most abundant, nonstrategic, raw materials with medium and low neutron capture cross sections, resulting in alloy cross sections one-half and two-thirds those of the nickel-base alloys and the austenitic stainless steels, respectively. Fe-Cr-Al-base alloys now developed, specifically the commercially available AISI 406 SS, are more economical than the austenitic stainless steels and the nickel-base alloys. Seamless and welded tubes have been fabricated recently on a production basis from AISI 406 SS.
- Fe-Cr-Al-base alloys have lower susceptibility toward carburization and intergranular and stress corrosion than austenitic stainless steels. These qualities of the Fe-Cr-Al-base alloys often compensate for their lower room temperature ductility, higher creep rate above 650 °C, and fabrication procedures calling for more rigid quality controls than are required for either austenitic stainless steels or nickel-base alloys.

The data presented in this report are the final result of this particular series of tests. Through the use of screening studies, literature surveys, and actual experimental data, a series of austenitic and ferritic stainless steels, iron-chromium-aluminum alloys, and solution and precipitation hardenable nickel-base alloys have been studied to determine their applicability for reactor components operating at elevated temperatures. These studies have primarily determined the corrosion and

oxidation resistance of the materials and have determined the effects of ambient and elevated temperature irradiation upon the room temperature and elevated temperature mechanical properties of the selected materials.

Results of the corrosion and oxidation tests showed some advantage for the iron-chromium-aluminum alloys in helium, nitrogen, and carbon dioxide over the more expensive nickel base alloys. Experience in the fabrication of these alloys into tubular products has not, however, met with consistent success, thus, materials such as AISI 406 require further development.

For tests where materials were exposed to high-pressure, de-oxygenated and oxygenated, superheated steam, the high nickel alloys showed excellent resistance to corrosion. Incoloy 800, Hastelloy X-280, and Hastelloy R-235 were among those alloys showing the least amount of corrosion penetration under these test conditions. As might be expected, the 300 series stainless steels were among the alloys showing the least amount of resistance.

Also, in tests using prestressed specimens in 42% magnesium chloride, the 300 series stainless were found to exhibit the highest susceptibility to stress corrosion cracking.

The ultimate goal in reactor construction is to apply a material that will not be affected by its surrounding thermal and chemical environment nor will be affected by long term exposure to neutron irradiation. Several of the alloys were irradiated at ambient and 650 °C and tested in tension at room temperature and 700 °C. The ductility, measured by percent elongation of the nickel base alloys irradiated in the study at 650 °C and tested at room temperature, was not affected by exposures of 2.53×10^{19} nvt. Some loss in ductility was observed at an exposure of 1.21×10^{20} nvt. The precipitation hardenable alloys, Inconel 702 and Hastelloy R-235, were found to have the lowest ductility and highest strength of the alloys tested. Only minor changes in room temperature, tensile and yield strength were observed.

In general, the elevated temperature ductility of the nickel-base alloys irradiated at 650 °C was substantially decreased at an exposure

of 2.53×10^{19} nvt and further reduced at an exposure of 1.21×10^{20} nvt. One exception was Inconel 702 where the percent elongation of the unirradiated control specimen was about 5%. Substantial scatter in the data for irradiated specimens makes definite conclusions difficult, since a maximum value for percent elongation of 17% was observed in a specimen that received an exposure of 1.21×10^{20} nvt. One important observation must be made in comparing the room temperature ductility to that at 700 °C in the unirradiated specimens. A substantial decrease can be observed, which indicates that the elevated test temperature alone is sufficient to reduce the ductility. While the room temperature yield strengths of Inconel 702 was not greatly affected by the irradiation, the elevated temperature yield was substantially reduced at both exposures.

The room temperature properties observed for the nickel base alloys irradiated at ambient temperature are characterized by increases in strength and marked decreases in ductility. With specimens tested at 700 °C, increases in yield strength were seen for the two solution hardenable alloys, Inconel 600, and Hastelloy X. Decreases in percent elongation were also noted for Hastelloy X and for Hastelloy R-235. A substantial increase in ductility was seen in Inconel 600 and Inconel 702. This improvement of ductility after irradiation was also observed for AISI 406 and 347 stainless steel alloys. Further investigation of the material microstructure and fabrication history must be completed to fully understand these observations.

In comparing the stainless steel data for the ambient temperature irradiations, it can be seen that the unirradiated room temperature yield strengths are high, indicating the presence of some cold work. Similar control specimens tested at 700 °C show large reductions in strength as expected. Comparing the data for the irradiated specimens, substantially larger changes in strength properties are found for the room temperature tests. This indicates the possibility of the 700 °C tests annealing out a substantial amount of the displacement damage to the materials.

The relationship between the changes in the alloy properties and the neutron exposure, neutron spectra, associated neutronic reactions

such as the (n, α) and (n,p) reactions, boron effects, and the metallurgical instability of the alloys at elevated temperatures must be determined before such understanding of the observed data can be accomplished.

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