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RADIATION EFFECTS DESIGN HANDBOOK

Section 7. Structural Alloys

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16. Abstract

This document summarizes information about the effects of radiation on structural alloys. The radiation form is neutrons with a small amount of gamma information. Information is included on aluminum, beryllium, magnesium, titanium, zirconium, various steels and refractory metals. Some cryogenic work is discussed.

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PREFACE

This document is the seventh section of a Radiation Effects Design Handbook designed to aid engineers in the design of equipment for operation in the radiation environments found in space, be they natural or artificial. This section of the Handbook provides the general background and information necessary to enable designers to consider suitable types of structural materials that may be exposed to nuclear radiation from a steady-state reactor.

Other sections of the Handbook discuss such subjects as transistors, solar cells, thermal-control coatings, interactions of radiation, electrical insulating materials and capacitors.

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radiation-induced increases in the ductile-to-brittle transition temperature and also larger decreases in ductility at low temperatures. The refractory metals undergo increases in yield and ultimate strength accompanied by decreases in ductility as a result of neutron irradiation. These radiation-induced changes in tensile properties are removed by annealing at elevated temperatures. The required annealing temperature for removal of radiation-induced changes in mechanical properties generally increases with increasing melting temperature of the alloy. The refractory metals do not exhibit the radiation-induced embrittlement at elevated temperatures that is common to austenitic stainless steels and nickel alloys, and the stress-rupture properties are not significantly altered by irradiation.

ALUMINUM ALLOYS

Aluminum is attractive for use as a structural material in thermal reactors because of its low cross section for thermal neutrons. However, its use is limited because of its low melting point and consequent lack of strength at elevated temperatures. Aluminum has been used as a fuel-element cladding and structural material in water-cooled swimming pool type reactors (not appropriate for space) where the operating temperatures are less than 100 C. Any increase in temperature will stimulate considerable corrosion and increase the possibility of reactions with the metallic fuel material since aluminum is a very reactive metal.

Some recent increases in the operating temperatures of aluminum alloys have been obtained by dispersing Al_2O_3 particles in either aluminum or aluminum alloys. However, this increase in strength at elevated temperatures is accompanied by large reductions in ductility. The operating temperature for this material is too high (~400 C) for metallic fuel and therefore it has been used as a cladding for UO_2.[(1)*]

The effects of irradiation on room temperature tensile and hardness properties of aluminum alloys are illustrated in Figure 1.[(2)] Irradiation causes increases in hardness, yield strength, and ultimate strength and decreases in ductility. The magnitude of these changes appears to increase with increasing fluence with no saturation having occurred after exposure to a fast fluence of 1.6×10^{22} n/cm^2.

The most severe radiation-induced changes in room temperature mechanical properties of aluminum occurred in aluminum-capsule bodies

* References are listed at the end of the section.

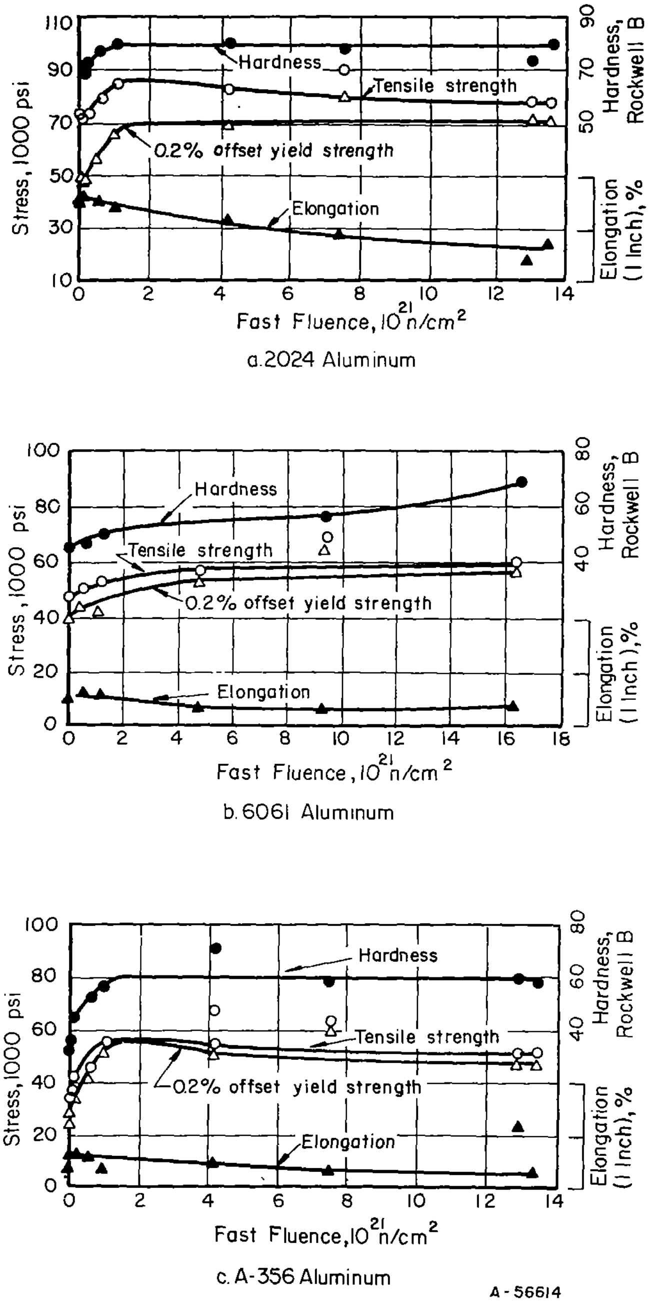

FIGURE 1. EFFECTS OF IRRADIATION ON THE ROOM-TEMPERATURE ROCKWELL HARDNESS, TENSILE STRENGTH, YIELD STRENGTH, AND ELONGATION OF ALUMINUM ALLOYS(2)

irradiated in the Engineering Test Reactor (ETR) for 5 years. These aluminum-capsule bodies were in contact with water at 50 C and received a fast fluence of 1.1×10^{22} n/cm^2. Considerable erosion-corrosion of the aluminum capsule bodies had taken place. The postirradiation room-temperature tensile properties of aluminum specimens fabricated from the capsules are given in Table 1. The combined irradiation and corrosion caused a fivefold increase in tensile strength and a 90 percent reduction in elongation.(3)

TABLE 1. ROOM TEMPERATURE TENSILE PROPERTIES OF ANNEALED 1100 ALUMINUM IRRADIATED AT 50 C(3)

Fast Fluence, 10^{21} n/cm^2	Yield Strength, 1000 psi	Ultimate Strength, 1000 psi	Elongation, percent
0	4.8	9.7	33
8.7	48.3	55.9	9.7
8.7	43.2	53.0	6.6
11	50.6	55.4	3.0
11	46.0	53.7	3.2

The elevated temperature tensile properties of irradiated aluminum are shown in Table 2.(4) The ultimate strength of the as-irradiated material was significantly higher than that of the unirradiated material at all testing temperatures. The irradiated specimens exhibited a minimum in ductility values at about 205 C with the ductility improving at higher testing temperatures. The unirradiated specimens did not exhibit a similar ductility minimum at these temperatures. The severity of ductility reduction also increases with increasing fast fluence. Other aluminum alloys like 8001 and 6061 have been found to exhibit similar behavior.

Postirradiation annealing of 8001 alloy (which had been irradiated at 60 C to 1.3×10^{22} n/cm^2) for 1 hour at ~540 C resulted in restoration of preirradiation tensile properties to testing temperatures up to 150 C. However, the ductility loss at testing temperatures above 175 C could not be recovered by postirradiation annealing.(4)

Irradiation of aluminum and its alloys also has resulted in significant density changes as illustrated in Figure 2. This density decrease is attributed to condensation of vacancies, which are produced by fast neutron-lattice atom collisions, to voids of 100 to 600 Å in diameter.(5) A 1-hour

TABLE 2. ELEVATED TEMPERATURE TENSILE PROPERTIES OF IRRADIATED 1100 ALUMINUM SPECIMENS(4)

Test Temp., C	Fast Fluence, 10^{22} n/cm^2	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Total Elongation, percent	
		Unirradiated[a]	Irradiated	Unirradiated[a]	Irradiated	Unirradiated[a]	Irradiated
RT	0.19	5.0	11.8	13.0	20.4	45	23.4
	0.39		18.6		24.1		29.8
	0.76		20.7		25.6		26.4
	1.3		22.4		26.0		26.7
	1.3		22.9		26.2		26.4
93	1.2	5.0	19.1	11.0	21.7	45	19.1
149	1.3	4.5	16.6	8.5	19.6	55	10.9
204	0.23	3.5	7.0	6.0	9.6	65	50.1
	0.44		11.0		12.6		18.2
	0.83		14.2		14.8		6.8
	0.97[b]		3.2		6.3		39.9
	1.2		14.8		15.9		5.9
260	1.2	2.0	10.4	4.0	11.2	75	5.6
316	1.1	1.5	6.6	2.5	7.1	80	10.7
371	1.2	1.0	4.1	2.0	4.4	85	9.0

(a) Handbook values.

(b) Annealed 1 hour at 538 C before testing.

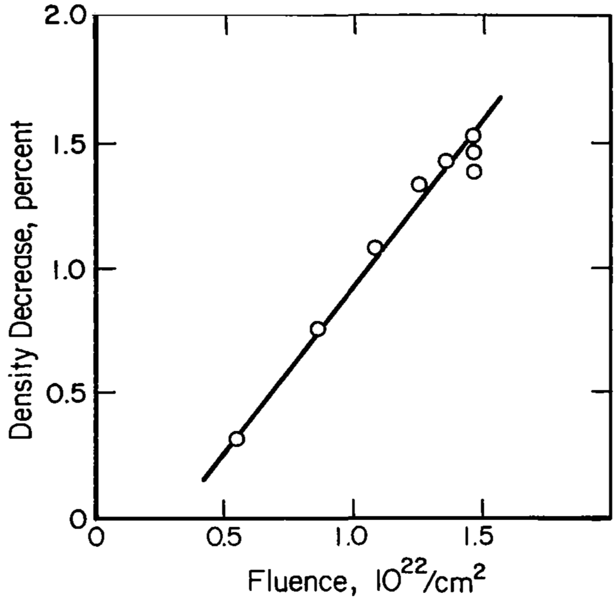

FIGURE 2. EFFECT OF TOTAL FAST-NEUTRON FLUENCE (>1 MeV) ON THE IMMERSION DENSITY OF 1100 ALUMINUM(5)

anneal at 260 C resulted in complete removal of these voids. The voids could also be eliminated by cold working the irradiated aluminum 65 percent at either room temperature or 120 C.

Tensile tests have been performed on SAP alloys (6-8 vol % Al_2O_3 in aluminum matrix) which were irradiated at 420 C to a fluence of 2×10^{20} n/cm^2. Irradiation did not change the tensile properties at any testing temperature within the range of room temperature to 400 C.(6) Impact tests at room temperature showed that the impact strength of SAP alloys irradiated to 1.5×10^{20} n/cm^2 at 50 and 270 C was not influenced by irradiation.(7)

Fatigue tests have been performed on irradiated aluminum by using notched specimens on a rotating-beam unit. The specimens had been irradiated to a fluence of 4×10^{16} n/cm^2 at 130 to 140 C. The irradiated specimens had a fatigue life only one-fourth that of the unirradiated specimens.(8) These results differ with another study(9) which found a significant increase in fatigue strength of pure aluminum that had received a fast fluence of 1×10^{18} n/cm^2. The discrepancy is due to the testing technique (Zamrik(9) performed his tests by superimposing an alternating tension-compression stress on an original tensile stress while Brewster(8) obtained his results on notched specimens).

Irradiation also has been found to effect significant changes in some processes involving aluminum. Aging in some aluminum alloys in a neutron environment has been found to occur at temperatures about 150 C lower than would be expected from purely temperature considerations.[10] Also, the reaction between UO_2 and aluminum metal has been observed to occur at considerably lower temperatures in a neutron flux than out-of-pile.[11] Both phenomena are controlled by diffusion; therefore, it would appear that diffusion in aluminum is enhanced during irradiation. Since a certain number of vacancies are required for significant diffusion to take place, the temperature of the material is usually about 0.5 T_m when enough vacancies become available owing to temperature. (T_m is defined as the melting temperature in degrees Kelvin; therefore, 0.5 T_m for aluminum would be 190 C.) However, since fast neutrons produce vacancies and interstitials, enough vacancies may become available at a temperature lower than 0.5 T_m if the material is in a sufficiently high fast-neutron flux.

MAGNESIUM ALLOYS

Magnesium alloys are attractive for nuclear applications because of their low cross section (0.00272/cm) for thermal neutrons. However, their limitations are their low melting temperature and low strength at elevated temperatures. Magnesium alloys have been used as a cladding material for British CO_2-cooled gas reactors where the clad operates at about 350 C. However, only limited data are available on the effects of radiation on the mechanical properties of magnesium alloys. The effects of radiation on the tensile properties of some magnesium alloys are shown in Table 3.

Figure 3 illustrates that irradiation to a fast fluence of 1 x 10^{19} n/cm^2 at 80 C improves the fatigue properties of magnesium.

TABLE 3. ROOM TEMPERATURE TENSILE PROPERTIES OF IRRADIATED MAGNESIUM ALLOYS

Alloy	Fast Fluence, 10^{20} n/cm^2	Irradiation Temperature, C	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent		Ref.
			Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	
H-3xA	0.078	40	13.7	15.8	26.7	28.4	13	20.4	12
H-3xA	0.54	40	13.7	16.4	26.7	34.5	13	16.2	12
H-3xA	1.0	40	13.7	25.1	26.7	36.6	13	9.9	12
HK-31A[(a)]	0.54	40	26.5	18.1	36.1	37.0	3.5	6.1	12
HK-31A[(a)]	1.0	40	26.5	22.1	36.1	37.4	3.5	4.9	12
A-12	1.0	50	16.0	17.8	21.3	22.5	5.8	5.3	13
Mg-0.8Al	1.0	50	16.6	15.7	22.4	21.6	6.2	5.8	13
AM-503S	1.0	50	24.2	25.2	31.5	28.8	3.8	2.0	13
AM503ZA	1.0	50	18.1	18.7	26.2	19.5	6.0	0.8	13
ZWI	1.0	50	24.1	23.2	32.0	31.2	14.2	17.4	13

(a) In H-24 condition.

TABLE 4. ROOM TEMPERATURE TENSILE PROPERTIES OF IRRADIATED TITANIUM ALLOYS

Material	Irradiation Temperature, C	Fast Fluence, n/cm^2 (E >1 MeV)	Test Temp.	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Uniform Elongation, percent		Total Elongation, percent		Ref.
				Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	
Titanium	100	5×10^{19}	20	79.8	88.7	83.4	92.2	--	--	10.4	8.3	15
Titanium	100	5×10^{19}	200	37.2	46.2	53.1	55.2	--	--	9.4	8.4	15
A-40	250-300	0.8×10^{16}	RT	40.1	39.6	47.8	47.7	10	8.9	45.9	40.8	16
A-40	250-300	1.3×10^{17}	RT	40.1	41.7	47.8	50.9	10	11.1	45.9	38.1	16
A-40	250-300	2.6×10^{18}	RT	40.1	46.7	47.8	57.5	10	11.6	45.9	32.8	16
A-110AT	250-300	1.3×10^{17}	RT	122.4	124.1	132.1	133.1	10.6	11.1	19.8	20.7	16

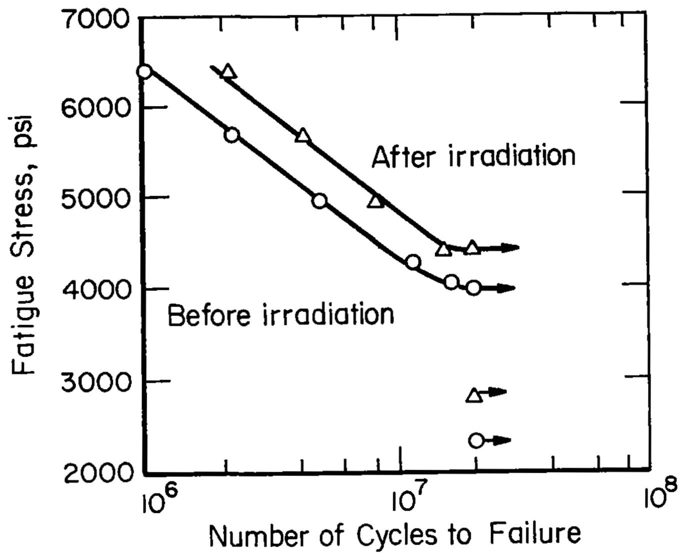

FIGURE 3. EFFECT OF FAST-NEUTRON IRRADIATION ON FATIGUE PROPERTIES OF MAGNESIUM(14)

TITANIUM ALLOYS

The effects of radiation on tensile properties of titanium alloys are summarized in Table 4. Some further data on titanium alloys are given in the section of irradiation effects at cryogenic temperatures.

BERYLLIUM

The leading beryllium application in the nuclear industry has been as a neutron-moderating or -reflector material in nuclear reactors. Its high-neutron-scattering cross section and low-neutron-absorption cross section make it very attractive for these nuclear applications. Recently, some attention has focused on beryllium as a possible fuel-element cladding material in gas-cooled reactors operating at elevated temperatures. It was hoped that the beryllium cladding would serve a dual purpose as a moderating material and a container for the nuclear fuel.

There are limitations, however, to the use of beryllium. One is its low ductility at room temperature which makes fabrication difficult; another is the large amount of helium gas generated during irradiation. At elevated temperatures, the helium gas agglomerates into bubbles and these bubbles cause swelling of beryllium. Along with the poor corrosion resistance of beryllium to H_2O and CO_2, these limitations have resulted in loss of enthusiasm for beryllium as a cladding-material candidate.

Swelling

The following gas-producing nuclear reactions occur in beryllium during irradiation:

By Fast Neutrons

$$Be^9 + n_f \rightarrow Be^8 + 2n$$

$$Be^8 \rightarrow 2He^4$$

$$Be^9 + n_f \rightarrow He^6 + He^4$$

$$He^6 \xrightarrow{\beta^-} Li^6$$

By Thermal Neutrons

$$Li^6 + n_t \rightarrow H^3 + He^4$$

$$He^3 + n_t \rightarrow H^3 + H$$

These nuclear reactions are calculated to produce about 2.2 to 2.6 cm^3 of gas/cm^3 of beryllium after a fluence of 1×10^{21} n/cm^2[16]. Most of the produced gas is either helium-4 or tritium, with minor amounts of hydrogen and helium-3 being produced. A meltdown study of irradiated beryllium gave the following results.[17]

	Cm^3 of Gas/Cm^3 of Beryllium	
	Calculated	Measured
Helium-4	2.043	1.52
Hydrogen-3 (tritium)	0.113	0.09
Hydrogen-1	0.025	0.29
Helium -3	0.0002	0.01

The swelling in irradiated beryllium is attributed to the helium gas generated during irradiation. At low temperatures, the helium atoms remain in solution in the matrix. However, at higher temperatures, the diffusion of helium atoms becomes significant and they agglomerate into bubbles. As the bubbles grow in size, considerable swelling of the irradiated beryllium takes place. Irradiation does not change the density of beryllium provided that irradiation temperatures and fluences are sufficiently low. No change in density has been found for beryllium irradiated to a fluence of 7.6 x 10^{21} n/cm^2 at 70 C[18] or a fluence of 8 x 10^{20} n/cm^2 at 600 C.[19] However, irradiations at 650 C and 700 C result in about a 0.5 to 1.0 percent density decrease after irradiation to a fast fluence of 5 x 10^{20} n/cm^2[20, 21]. Another study showed a density decrease of 1.5 percent for beryllium irradiated to a fast fluence of 1.8 x 10^{20} n/cm^2 at 780 C.[22] It seems likely that the radiation-induced swelling characteristics of beryllium vary from batch to batch.

Another approach used in the swelling studies has been to take low-temperature irradiated beryllium and anneal it at elevated temperatures. Sections of the beryllium reflector from the MTR were annealed at temperatures of 600 to 1000 C.[17, 23, 24] These beryllium specimens had been irradiated to a fast fluence of 1 x 10^{21} n/cm^2 at 70 C. Results of the annealing studies are given in Figure 4. From those results, the investigators concluded that the threshold temperature for swelling in irradiated beryllium is 725 C. These results agree with previous studies showing that significant swelling occurred in beryllium annealed between 700 to 800 C following irradiation to a fast fluence of 2.75 x 10^{21} n/cm^2.[25] These studies also indicated that breakaway swelling was not only temperature dependent but also time dependent. This is illustrated in Figure 5 where the swelling at 800 C is shown to increase drastically after 200 hours. It was also found that after irradiation to a fast fluence of 7.6 x 10^{21} n/cm^2, the breakaway swelling temperature is 600 C compared with about 725 C after irradiation to a fast fluence of 1 x 10^{21} n/cm^2. Again, results show

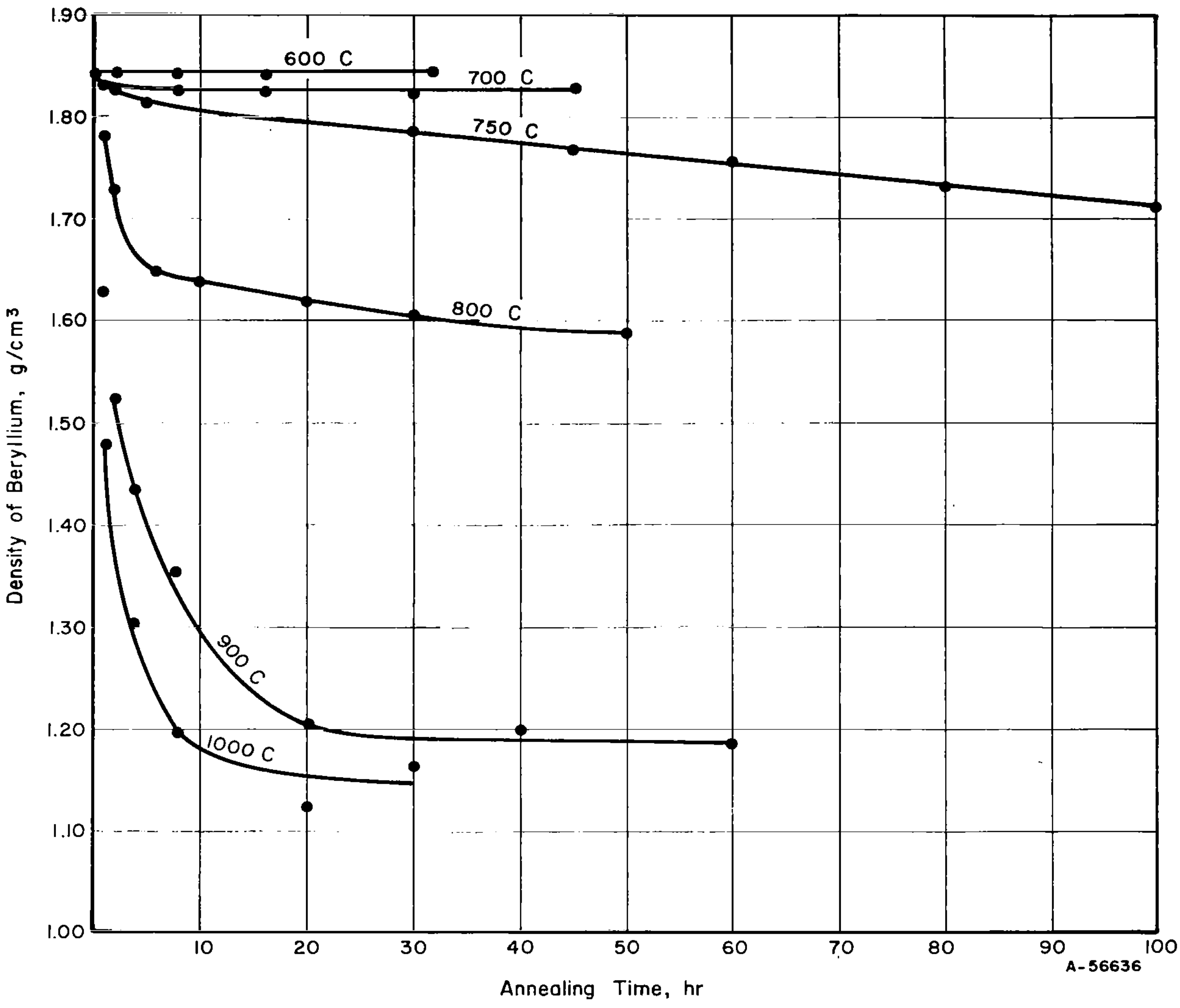

FIGURE 4. SWELLING OF BERYLLIUM IRRADIATED TO A FAST FLUENCE OF 1 x 10^{21} N/CM^2 AT 70 C(17, 23, 24) AS A FUNCTION OF ANNEALING TIME AND ANNEALING TEMPERATURE

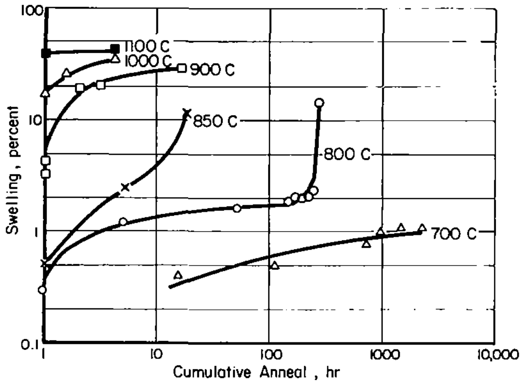

FIGURE 5. TIME DEPENDENCE OF SWELLING AT VARIOUS TEMPERATURES[25] FOR BERYLLIUM IRRADIATED AT 280 to 480 C TO A FAST FLUENCE OF 2.75 x 10^{21} N/CM^2

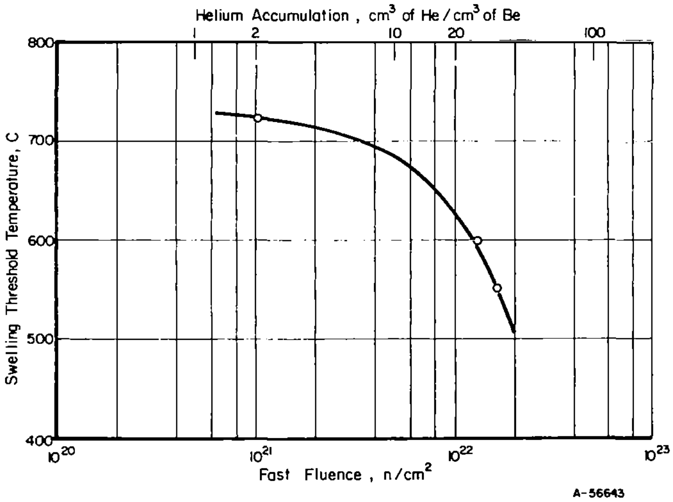

FIGURE 6. SWELLING THRESHOLD OF BERYLLIUM AS A FUNCTION OF GAS CONTENT AND FLUENCE[26]

that the breakaway swelling in irradiated beryllium is dependent on annealing temperature and annealing time, as well as on fast fluence and possibly the mechanical properties of the specific bath of beryllium metal. Figure 6 illustrates the swelling threshold in irradiated beryllium as a function of temperature and fast fluence.(26)

Mechanical Properties

In evaluating the tensile properties of beryllium it is important to know that randomly oriented beryllium has low ductility at low testing temperatures. This is because slip can occur only along basal planes and therefore the basal planes will have to be orientated parallel to the tensile axis if the material is to have any ductility. Most of the beryllium that has been irradiated has had oriented basal planes. Figure 7 illustrates the tensile properties of beryllium that has been irradiated at 100 C.(19)

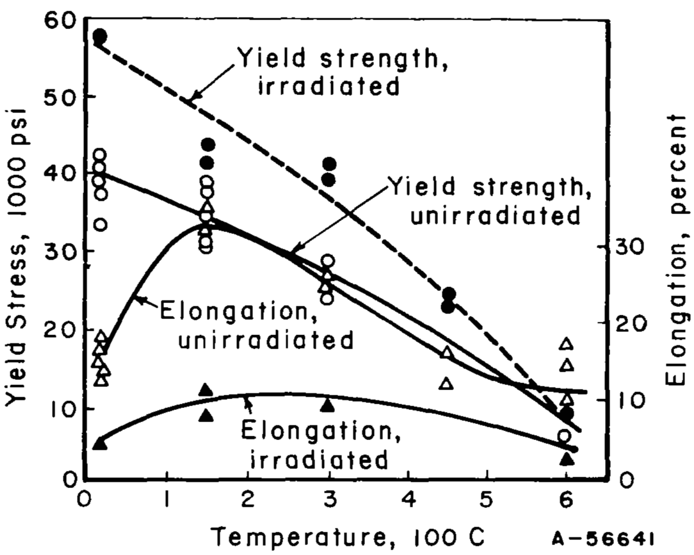

FIGURE 7. EFFECT OF IRRADIATION AT 100 C TO A FAST FLUENCE OF 4 x 10^{20} N/CM^2 ON TENSILE PROPERTIES OF BERYLLIUM(19)

The changes in tensile properties are more prominent at lower testing temperatures, and approach the values for unirradiated material at higher testing temperatures. The mechanism of radiation-induced hardening in beryllium is a combination of fast-neutron-produced interstitial and vacancy defects and transmuted helium atoms. At these low temperatures, helium atoms stay in the metal matrix as atoms and cause solid-solution hardening. Both the helium atoms and the defects impede the movement of dislocations, thus increasing the strength and decreasing the ductility. At higher testing temperatures, some of the fast-neutron-produced defects are annealed out of the beryllium matrix and thereby the radiation-induced changes in tensile properties are reduced.

Irradiations at the intermediate-temperature ranges (250 to 550 C) progressively change the tensile properties with increasing fast fluence. This is shown in Figure 8.[19] As for the lower temperature irradiations, the mechanical-property changes become less significant at higher testing temperatures owing to the annealing out of fast-neutron-produced defects. The progressively smaller ductility restoration caused by increasing fast fluence is attributed to coalescence of helium atoms to bubbles at grain boundaries. These bubbles weaken the grain boundaries and promote intergranular fracture.

Irradiation at 600 C and above results in tensile-property changes for some batches of beryllium but not for other batches. Figure 9 shows the results for a batch of beryllium whose tensile properties were not changed by irradiation at 600 C[19], while Table 5 shows the results for material whose properties were changed by irradiation at 650 C and above.[21] This discrepancy is explained by the fact that impurity sites act as traps for the helium atoms and prevent it from agglomerating as bubbles at the grain boundaries. If helium-bubble agglomeration at the grain boundaries is minimum, the mechanical properties are not changed by the elevated-temperature irradiation. It is expected that the point defects produced by fast neutrons would probably be annealed out above 600 C.

A few postirradiation tensile tests have been performed on specimens formed transverse to the extrusion direction. These tests were performed on hot-pressed and hot-extruded material irradiated at 350 and 600 C. The results, shown in Figure 10, indicate that irradiation does not affect the transverse tensile properties of beryllium.[27] It should be emphasized, however, that beryllium is extremely brittle in the transverse direction despite the irradiation condition.

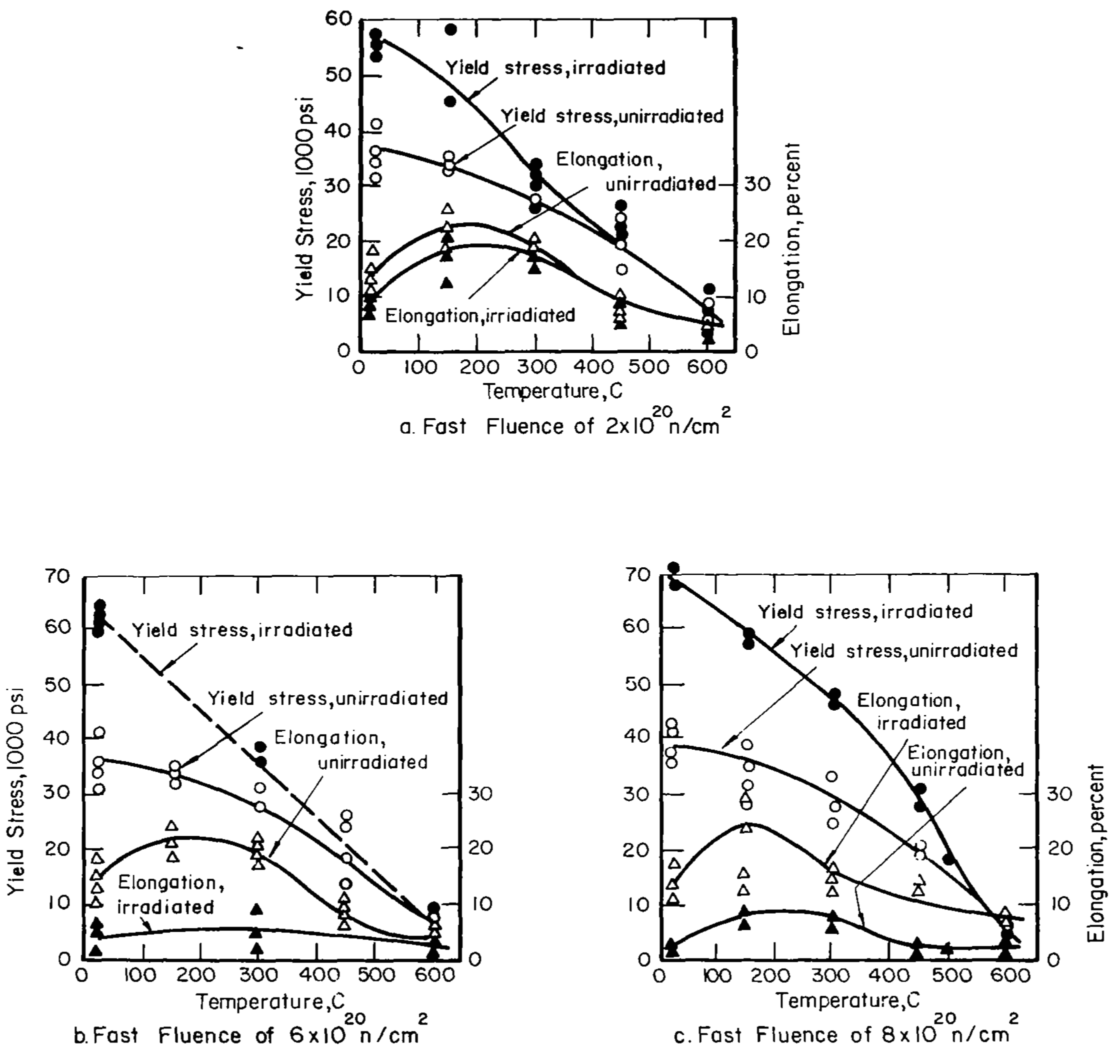

FIGURE 8. EFFECT OF IRRADIATION AT 350 C ON THE TENSILE PROPERTIES OF BERYLLIUM (19)

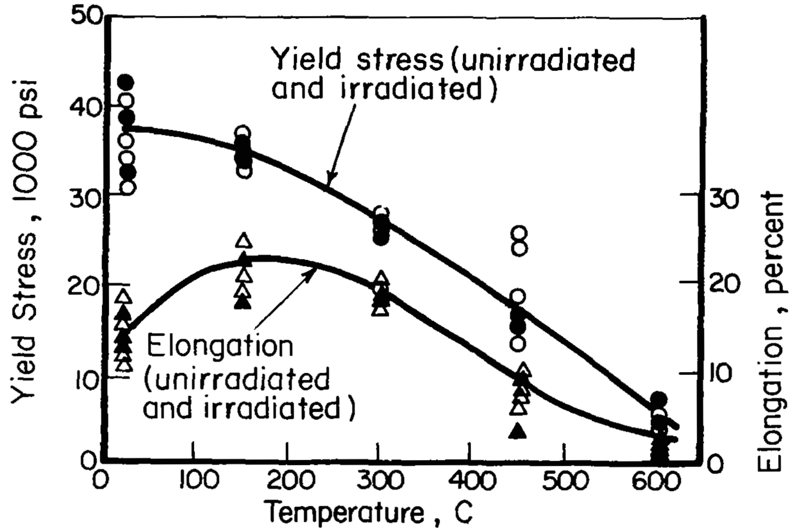

FIGURE 9. EFFECT OF 600 C IRRADIATION TO A FAST FLUENCE OF 6 x 10^{20} N/CM^2 ON THE LONGITUDINAL MECHANICAL PROPERTIES OF BERYLLIUM[19]

TABLE 5. EFFECT OF TESTING AND IRRADIATION TEMPERATURE ON ELONGATION OF DIFFERENT BATCHES OF BERYLLIUM[21]

	Elongation, percent, at Indicated Irradiation and Testing Temperature					
	450 C		550 C		650 C	
	Preirr.	Postirr.	Preirr.	Postirr.	Preirr.	Postirr.
Material A[a]	20.2	4.4[c]	20	1.3[d]	13.8	0.5[d]
Material B[b]	36.0	20.0[c]	26	8.7[d]	18.0	5.0[d]

(a) Sintered at 1200 C; extruded at 850 C.

(b) Hot pressed at 1050 to 1100 C; extruded once at 1050 C, again at 850 C.

(c) Received a fast fluence of 5.5 x 7 x 10^{20} n/cm^2.

(d) Received a fast fluence of 9 x 10^{20} n/cm^2.

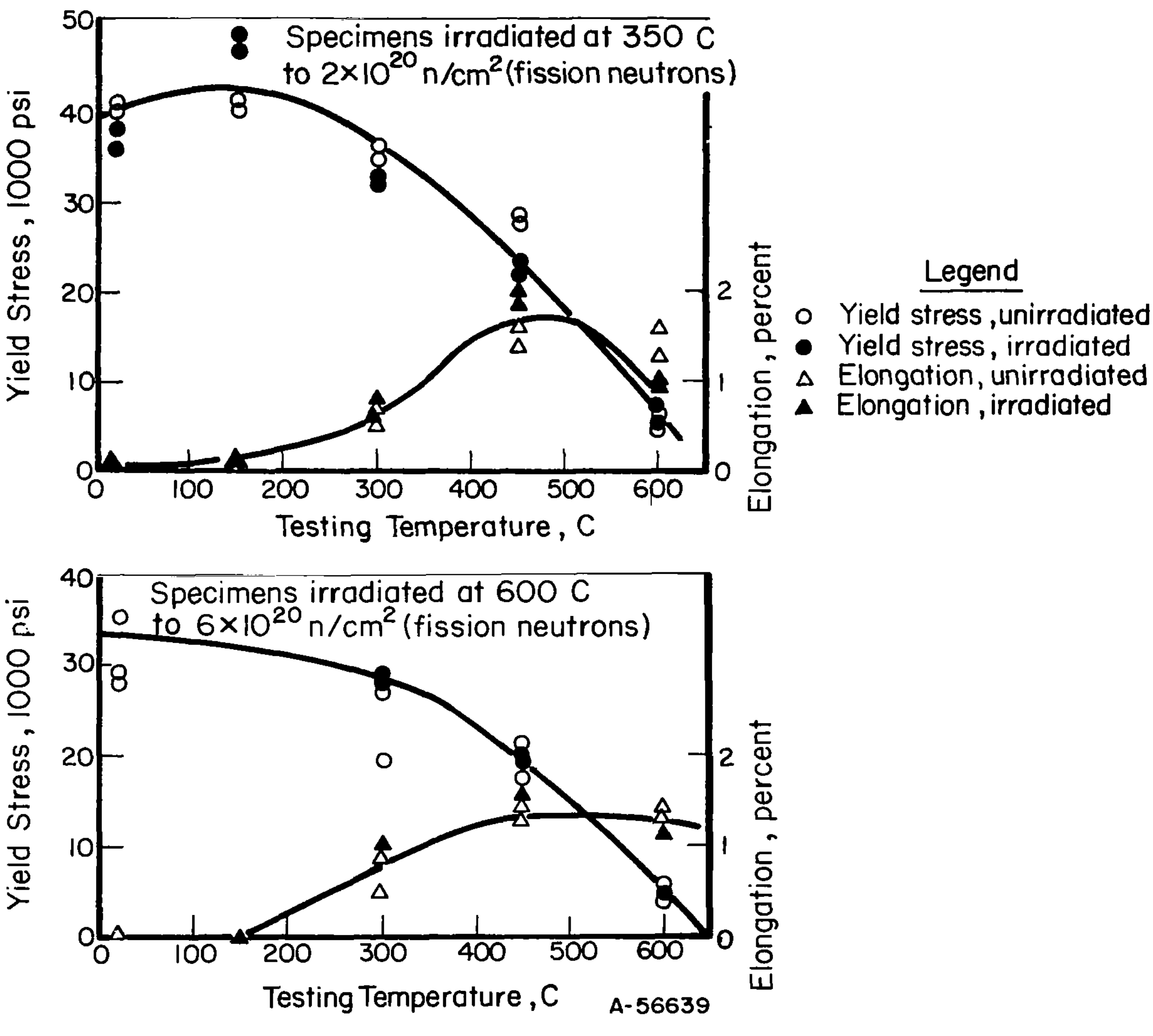

FIGURE 10. EFFECT OF IRRADIATION ON THE TRANSVERSE MECHANICAL PROPERTIES OF EXTRUDED BERYLLIUM[27]

Annealing of irradiated beryllium above 700 C restores its unirradiated properties. This restoration of unirradiated properties comes about by the removal of defect clusters and small helium bubbles which act as barriers to dislocation movement in the irradiated material. Figure 11 illustrates the restoration of preirradiation yield strength, ductility, and hardness with increasing annealing temperature.[25] The temperature where significant removal of irradiation hardening takes place appears identical for both properties. At that temperature significant swelling in irradiated beryllium also begins. Therefore, most investigators believe that at that temperature removal of point defects occurs and the helium atoms start to agglomerate into small bubbles. When the defect clusters are removed and the helium bubbles have reached a significant size, no further obstacles to dislocation movement are offered and the preirradiation properties are restored. The increased ductility is explained in terms of bubble migration to the grain boundary which allows more elongation to take place in the grain before failure. The annealing behavior of irradiated beryllium is expected to vary among different batches since almost all batches show some variation in properties.

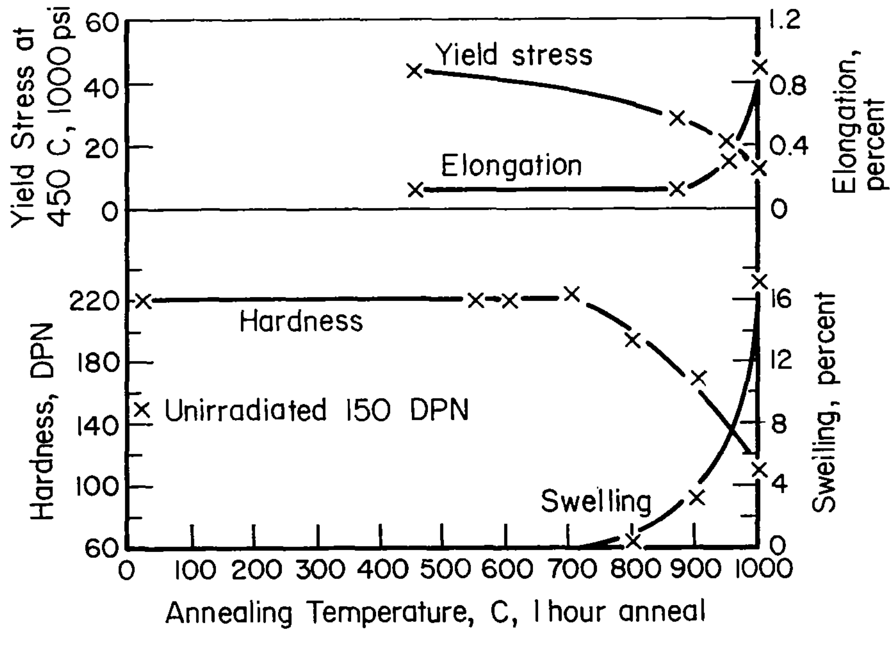

FIGURE 11. THE EFFECT OF ANNEALING ON THE HARDNESS, YIELD STRESS AND ELONGATION OF IRRADIATED BERYLLIUM[25]

As shown in Table 6, compression tests have been performed on beryllium irradiated at 70 C.[28] With increasing fast fluence, the compressive yield strength increased while the ductility decreased. A peak ultimate compressive strength was obtained after irradiation to a fast fluence of 1.6×10^{21} n/cm^2, but the strength decreased with increasing fast fluence. Irradiation at 600 and 750 C to a fast fluence 8.8×10^{20} n/cm^2 resulted only in minor changes in compression properties.[29] These results, which show no changes in compressive properties, agree with tensile data obtained on beryllium irradiated at the same temperatures.

TABLE 6. COMPRESSION STRENGTH OF IRRADIATED BERYLLIUM[28]

Position Code[(a)]	Fast Fluence[(b)], n/cm^2	Compression Yield Strength (0.2% Offset), 1000 psi	Compression Ultimate Strength, 1000 psi	Total Strain in 1.5 In., %	Plastic Strain in 1.5 In.[(c)], %
LL-6-7-8-9 (4 samples)	0.6×10^{21}	79 ± 4	153 ± 62	14.4	12.9
LT-6-7-8-10 (4 samples)	1.1	98 ± 4	184 ± 16	11.5	11.0
LL-1-2-3 (3 samples)	1.6	117 ± 6	194 ● 13	13.9	12.4
LT-3-4-5 (3 samples)	3.2	122 ± 24	129 ± 41	3.1	1.5
MT-6	5.0	163	163	2.8	0.7

(a) Average of number of samples indicated.
(b) Calculated from fluence profiles of MTR Cycle 146 for various positions in which LB-15 received irradiation (approximate values).
(c) Measured from yield point to point of fracture.

Bend tests were conducted on beryllium specimens irradiated at 500 to 700 C to a fast fluence of 1.3×10^{20} n/cm^2 and at 60 C to a fast fluence of 6×10^{20} n/cm^2.[22] Results of these bend tests, shown in Table 7, indicate no ductility changes for specimens irradiated at 500 to 700 C. This is expected since irradiation in that temperature range is not

TABLE 7. BENDING PROPERTIES OF IRRADIATED BERYLLIUM[22]

Specimen	Fluence, 10^{20} n/cm^2	Irradiation Temperature, C	Preirradiation[a] Load, lb	Preirradiation[a] Deflection, mils	Postirradiation Load, lb	Postirradiation Deflection, mils
HPB	1.3	500	79-92	5.0-6.5	74	5.1
HPB	1.3	600	79-92	5.0-6.5	96	4.6
CR	1.3	500	60-75	1.5-2.5	66	2.0
CR	1.3	560	60-75	1.5-2.5	76	2.0
CR	1.3	700	60-75	1.5-2.5	51	1.9
			Postirradiation		Postanneal	
HPB[b]	6	60	48	1.1	71	9.1
HPB[c]	6	60	41	0.9	95	3.2
CR[d]	6	60	44	1.0	57	1.5
CR[e]	6	60	38	1.0	52	2.6
CR[f]	6	60	60	1.1	69	1.5
CR[g]	6	60	57	1.2	100	2.6
CR[h]	6	60	28	0.7	62	2.2

(a) These values are ranges of five determinations.
(b) Annealed 700 C 1 hr.
(c) Annealed 800 C 1 hr.
(d) Annealed 400 C 1 hr.
(e) Annealed 500 C 1 hr.
(f) Annealed 600 C 1 hr.
(g) Annealed 700 C 1 hr.
(h) Annealed 800 C 1 hr.

expected to change the properties of beryllium. Some losses in ductility were shown by specimens irradiated at 60 C. However, annealing at 400 C restored ductility significantly.[22] In another series of tests, specimens irradiated to a fast fluence of 7.6×10^{21} n/cm^2 at 70 C were loaded to failure by bending at room temperature, 300, 600, and 700 C.[20] Results indicated that the fracture stress required for failure by bending was not changed by irradiation. However, the ductility was considerably affected. For the unirradiated specimens, some bending occurred at each temperature before fracture. However, no bending was found for the irradiated specimens at any testing temperature. All of the irradiated specimens were held at the temperature of testing for 17 hours before the test. Failure of this annealing to improve the ductility in the irradiated material indicates that 700 C is not a sufficiently high temperature for recovery of irradiation embrittlement in some batches of beryllium.

ZIRCONIUM ALLOYS

Zirconium alloys although of little interest in space have found wide use as structural materials in the nuclear industry and are included for completeness. The zirconium alloys are very attractive for nuclear considerations because of their low thermal-neutron capture cross section. These alloys also have adequate tensile strength and good corrosion resistance to water and steam at temperatures up to 400 C, the maximum temperature in boiling-and pressurized-water reactors. For additional strength at these operating temperatures, a zirconium-2.5 wt % niobium alloy has been developed. The composition of zirconium alloys ("Zircaloys") used in the nuclear industry is given in Table 8.

TABLE 8. COMPOSITION OF ZIRCONIUM ALLOYS

	Composition[a], percent				Composition[b], ppm							
Alloy	Zr	Sn	Fe	Cr	Ni	N	Al	C	Hf	Pb	Si	W
Zircaloy-2	Bal	1.2-1.7	0.07-0.20	0.05-0.15	800	80	75	270	200	130	200	100
Zircaloy-3	Bal	0.2-0.3	0.20-0.30	0.05	500	80	75	270	200	130	200	100
Zircaloy-4	Bal	1.2-1.7	0.18-0.38	0.05-0.15	70	80	75	270	200	130	200	100

(a) Maximum allowable content indicated.

(b) Maximum allowable limit of elements not given if 50 ppm or less.

Zircaloys

In considering the mechanical properties of zirconium alloys, it must be emphasized that the hexagonal structure of zirconium causes anisotropy of mechanical properties. Unlike beryllium, where slip occurs mostly at the basal plane, causing extremely low room-temperature ductility, the predominant room-temperature slip planes in zirconium are the prismatic $(10\bar{1}0)$ planes and, consequently, zirconium has considerable ductility at room temperature. In general, zirconium alloys exhibit a higher yield strength but lower tensile strength and ductility when tested in the transverse direction. There does not appear to be any significant difference in mechanical properties among the various Zircaloys. The variables which affect postirradiation mechanical properties of the Zircaloys are discussed below.

Fast Fluence

Figures 12 and 13 illustrate the change in yield strength, ultimate strength, uniform elongation, and total elongation as a function of increasing fast fluence.(30,31) While the total elongation is considerably reduced by irradiation, the uniform elongation, especially at elevated temperatures, is drastically reduced by a fast fluence of 1×10^{21} n/cm^2. Although saturation in neutron-induced damage has not been reached up to a fast fluence of 1×10^{22} n/cm^2, the rate of change in mechanical properties is significantly decreased above a fast fluence of 1×10^{20} n/cm^2.

Irradiation Temperature

The effect that irradiation temperature has on the yield strength is illustrated in Figure 14.(32) It can be seen that irradiation at 290 C increases yield strength considerably more than does irradiation at 60 C. This is attributed to the formation of larger defect clusters at higher irradiation temperatures. These larger and therefore more stable, defect clusters have a greater effect in blocking the movement of dislocations. Consequently, a higher stress is required to move the pile-up dislocations, and thus the stress required for yielding is higher.

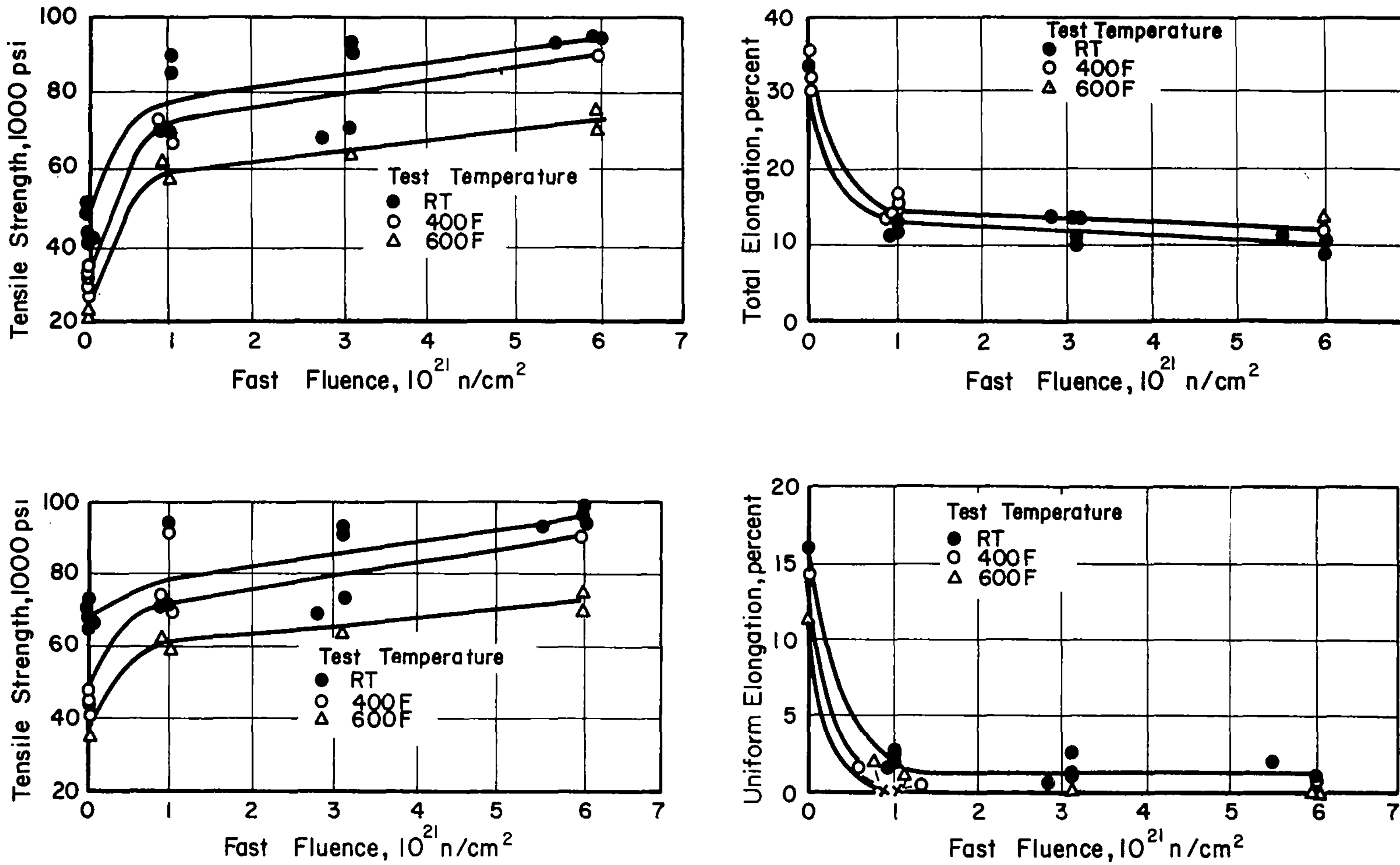

FIGURE 12. EFFECT OF IRRADIATION AT 250 C TO FAST FLUENCE ON THE MECHANICAL PROPERTIES OF ZIRCALOY-2(30)

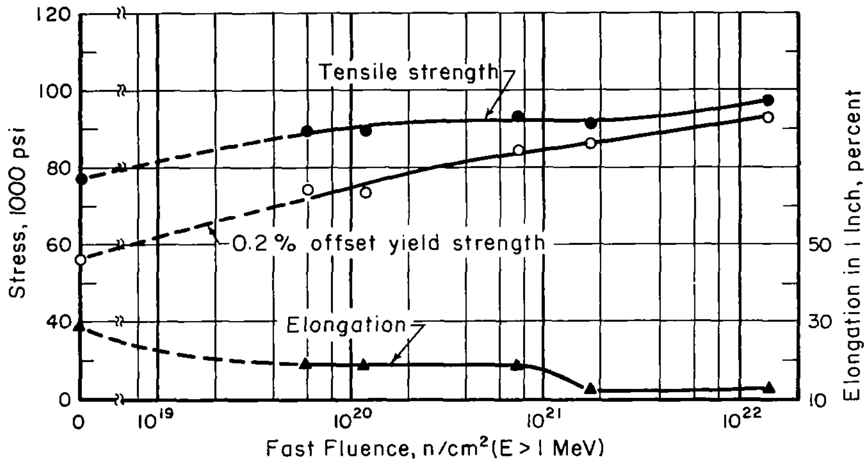

FIGURE 13. EFFECTS OF IRRADIATION AT 50 C ON ROOM-TEMPERATURE TENSILE PROPERTIES OF ZIRCALOY-2[31]

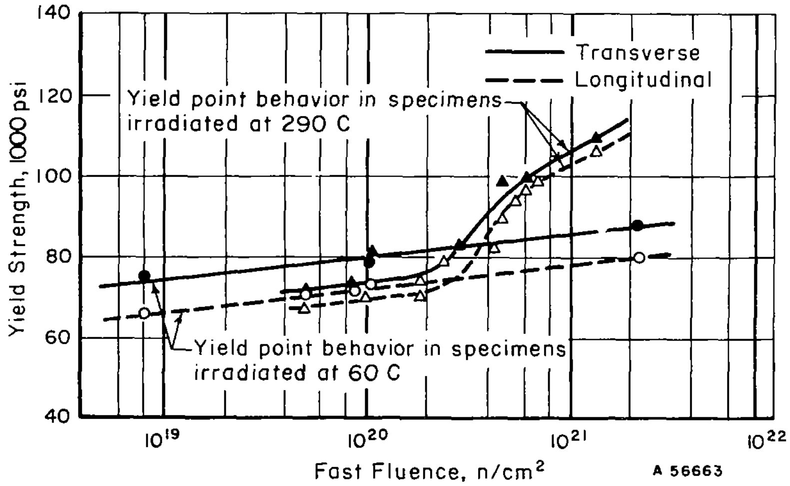

FIGURE 14. EFFECT OF FAST FLUENCE AND IRRADIATION TEMPERATURE ON YIELD STRENGTH OF ZIRCALOY-2[32]

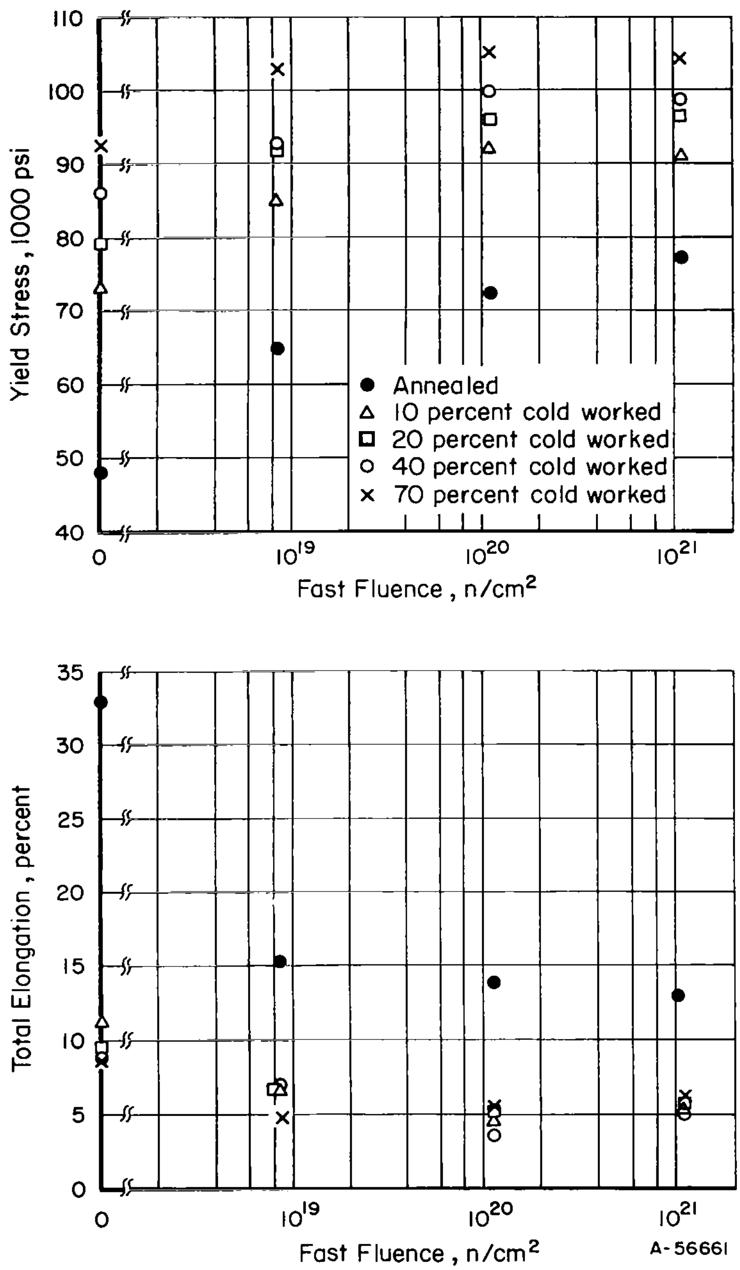

FIGURE 15. THE EFFECT OF COLD WORK AND IRRADIATION ON THE LONGITUDINAL, ROOM TEMPERATURE YIELD STRENGTH AND ELONGATION OF ZIRCALOY-2 IRRADIATED AT 60 C[33]

Cold Work

The effect of cold work on performance of zirconium alloys is important since their application is usually in cold-formed tubing. Therefore, it must be known whether the added strength can be utilized in neutron environments. Figure 15 illustrates fast-fluence effects on the yield strength and elongation of Zircaloy-2 cold worked in the testing direction.(33) It can be seen that the yield strength of cold-worked Zircaloy-2 is increased by irradiation to fast fluences up to 1 x 10^{20} n/cm^2, after which the yield strength starts to decrease. Similarly, the elongation decreases up to a fast fluence of 1 x 10^{20} n/cm^2, but increases with increased irradiation. This behavior is attributed to a thermal recovery induced by irradiation-induced point defects. No similar reversal of strength and ductility was found for the irradiated and cold-worked material in the transverse direction at fast fluences of 1 x 10^{21} n/cm^2. The effect of irradiation and cold work on the point of plastic instability in Zircaloy-2 is illustrated in Figure 16(32) The point of plastic instability is defined as the location on the stress-strain curve where plastic deformation takes place without further increase in stress.

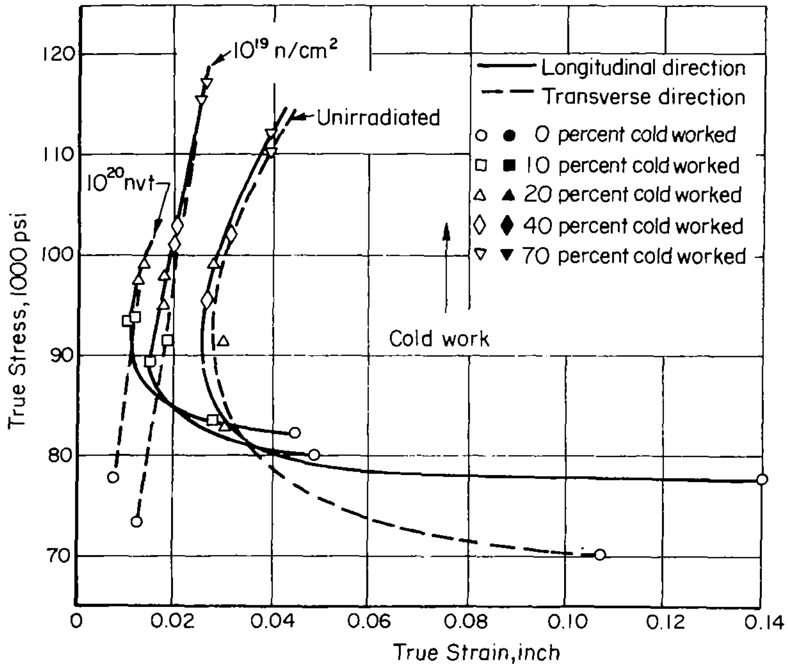

FIGURE 16. DISPLACEMENT IN POINT OF PLASTIC INSTABILITY WITH COLD WORK AND IRRADIATION FOR ZIRCALOY-2(32)

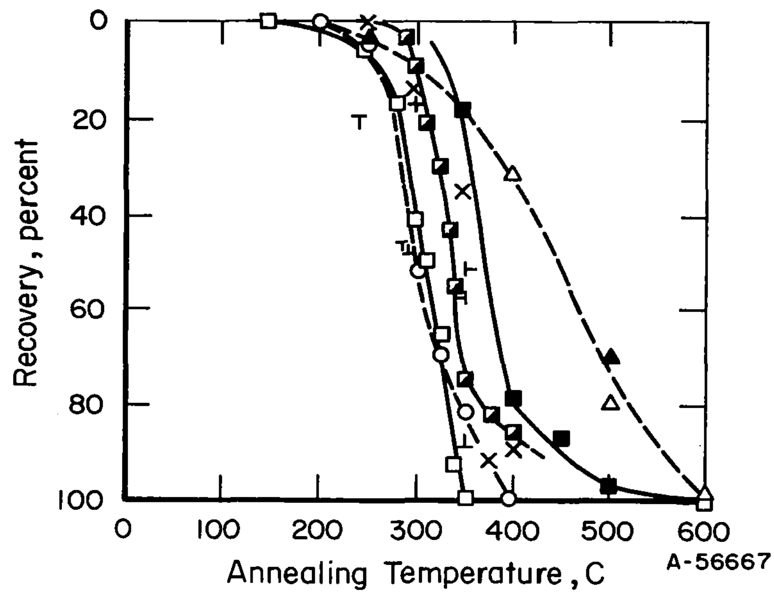

FIGURE 17. RECOVERY OF IRRADIATION-INDUCED DAMAGE IN TENSILE PROPERTIES OF ZIRCONIUM AND ITS ALLOYS BY POSTIRRADIATION ANNEALING(32)

	Material	State	Fast Fluence, n/cm^2	Irr. Temp., C
○	Zirconium	Annealed	$3.3 \times 10^{19} > 1$ MeV	50
△	Zr-2.5 wt % Nb	Q & T	$1 \times 10^{20} > 500$ eV	50
▲	Zr-2.5 wt % Nb	Q & T	$1 \times 10^{20} > 500$ eV	250
□	Zircaloy-2	Annealed	$3 \times 10^{19} > 1$ MeV	50
◪	Zircaloy-2	Annealed	$7.7 \times 10^{19} > 500$ eV	280
■	Zircaloy-2	Annealed	$2.7 \times 10^{20} > 500$ eV	280
		Above, all times 1 hour		
×	Zircaloy-2	Annealed	$1.8 \times 10^{20} > 1$ MeV	40
⊥	Zircaloy-2	Extruded ($\alpha + \beta$)	$5.6 \times 10^{19} > 1$ MeV	40
T	Zircaloy-2	Extruded and	$5.6 \times 10^{19} > 1$ MeV	40
⊢	Zircaloy-2	Extruded and 50% CW	$5.6 \times 10^{19} > 1$ MeV	40
⊣	Zircaloy-2	Extruded and 15% CW and annealed	$5.6 \times 10^{19} > 1$ MeV	40
		Above, all times 100 hours		

Testing Temperature

The testing temperature for irradiated materials becomes significant because some of the irradiation damage may be annealed out by the elevated testing temperature. Since irradiated Zircaloy-2 has been generally tested at only room temperature and 300 C, not many tensile data at other temperatures have been generated. However, considerable data are available on the recovery of tensile properties by postirradiation annealing. A good summary of the annealing behavior in irradiated zirconium alloys was presented by Bush[32] (see Figure 17). Bush concluded that:

(1) The tensile-property-recovery behavior of irradiated zirconium and Zircaloy-2 is about the same.

(2) Higher irradiation temperature stabilized the more complex defects, thereby retarding recovery.

(3) Higher fast-fluence levels also cause more complex defects and require higher annealing temperatures.

(4) The high-strength zirconium-2.5 wt % niobium alloy requires a higher annealing temperature than does Zircaloy-2 for recovery of irradiation-induced defects for the same fast-fluence value.

Results of elevated temperature tensile tests on annealed and cold-worked Zircaloy-2 are given in Table 9.[34]

Composition

Extreme care should be taken in interpreting the effects of irradiation on mechanical properties because of the possible contamination by oxygen and hydrogen. Since zirconium alloys are usually irradiated immersed in water, oxygen and hydrogen are introduced into the material. Oxygen forms as ZrO_2 particles in zirconium alloys, while hydrogen forms as zirconium hydride platelets. These brittle zirconium hydride platelets usually deposit at the grain boundaries, and result in extreme embrittlement if the stress is applied perpendicular to the axis of the platelet. The orientation of these hydride particles has been shown to be dependent on the material's

TABLE 9. ELEVATED TEMPERATURE TENSILE PROPERTIES OF ZIRCALOY-2[a][34]

Condition	Test Temp, C	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Uniform Elongation, percent		Total Elongation, percent	
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
Annealed	25	54.5	98.6	63.8	98.6	12.2	1.1	29.9	2.6
	400		61.5		61.7		1.2		3.7
	500	10.8	30.4	14.7	34.0	7.7	1.5	55.0	6.3
	600	6.5		8.2		7.1		52.5	
	650		6.5		7.6		2.5		23.1
20% CW	25	69.8	113.9	76.2	114.8	3.9	1.3	15.5	1.4
	300	42.9	82.5	45.5	87.3	1.0	1.6	1.6	2.2
	400		58.2		63.4		2.2		9.0
	500	21.5	23.2	25.3	29.8	2.2	2.3	23.4	15.4
	600	9.2		12.0		2.7		37.5	
	650		2.0		2.8		6.2		48.9
40% CW	25	80.8	108.4	85.5	112.3	2.7	1.8	9.7	2.0
	300	50.8	78.0	53.4	80.9	1.4	1.9	6.0	3.3
	400		58.6		62.7		1.7		7.8
	500	20.9	27.9	27.2	30.3	1.3	2.5	22.6	13.7
	600	9.5		11.0		2.4		47.6	
	650		4.2		5.3		2.0		12.5

(a) Irradiated at 280 C to a fast fluence.
CW = Cold Worked.

fabrication history and the stresses present when the zirconium alloy is cooled from a temperature of high hydrogen solubility to a temperature of low hydrogen solubility. The hydrogen solubility in alpha-zirconium is 160 ppm at 400 C, but only 0.005 ppm at 20 C.[35] It has been found that the hydride platelets precipitate perpendicular to the stress direction when in tension and parallel to the stress direction if in compression. The hydride platelets also have a tendency to precipitate parallel to the direction of cold working. At the present, the relative importance of the direction of cold work and direction of stress on the orientation of zirconium hydride platelets has not been definitely established. It must be emphasized that it is not the actual hydrogen content that affects the mechanical properties; rather, it is the fraction of the hydride platelets perpendicular to the stress direction that must be considered. If the platelets are parallel to the direction of stress, then these platelets behave like voids in the material. The embrittling effect of the hydride platelets has been shown to be very strain-rate sensitive even if the platelets are oriented perpendicular to the stress. This means that while the hydrides are very embrittling in an impact test, they may not be very embrittling in a tensile test.[35] Since the zirconium-hydride platelets become plastic at about 200 C, the embrittlement ceases to be a problem at that temperature and above. The combined effects of oxygen and hydrogen contamination and neutron irradiation on the tensile properties is illustrated in Table 10. It can be seen that contamination and irradiation combined result in more severe embrittlement than would be expected if only one phenomenon were contributing. On the other hand, the effects of hydrogen content and irradiation are not additive. The effects of both oxygen and hydrogen on mechanical properties are minimized if the irradiated material is tested at higher temperatures. The reader is again cautioned in interpreting the tensile results, especially at lower temperatures, since only a few investigations have reported the oxygen content or the content and orientation of hydride platelets of their specimens.

Welding

Irradiation of the as-welded Zircaloy-2 and cold-worked welded material causes irradiation-induced mechanical-property changes similar to those observed in unwelded Zircaloy-2 during irradiation.

TABLE 10. EFFECT OF HYDROGEN, OXYGEN AND IRRADIATION ON MECHANICAL PROPERTIES OF ZIRCONIUM ALLOYS

Material and Condition[a]	Impurity Content, ppm	Fast Fluence, n/cm² (>1 MeV)	Irr. Temp, C	Test Temp, C	Yield Strength, 1000 psi Unirr.	Yield Strength, 1000 psi Irr.	Ultimate Tensile Strength, 1000 psi Unirr.	Ultimate Tensile Strength, 1000 psi Irr.	Elongation, percent Uniform Unirr.	Elongation, percent Uniform Irr.	Elongation, percent Total Unirr.	Elongation, percent Total Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Reference
Containing Hydrogen															
Zircaloy-2, W	0	$0.3\text{-}1.9 \times 10^{19}$	260	RT	47.3	49.3	62.7	62.3	10.2	11.4	11.9	14.3			35
Zircaloy-2, W	100	$0.3\text{-}1.9 \times 10^{19}$	260	RT	52.2	60.3	68.4	74.7	8.4	7.7	12.0	11.9			35
Zircaloy-2, W	0	$0.3\text{-}1.9 \times 10^{19}$	260	260	24.8	30.9	37.3	39.2	12.3	9.3	21.3	18.5			35
Zircaloy-2, W	100	$0.3\text{-}1.9 \times 10^{19}$	260	260	25.0	29.4	36.1	38.0	11.7	11.6	21.5	23.5			35
Zircaloy-4, A	110	2.5×10^{21}	95-120	150	34.9	71.6	54.3	74.8	11.2	1.5	24.6	10.0	40.3	39.1	38
Zircaloy-4, E	110	2.5×10^{21}	95-120	150	55.9	95.0	72.4	96.8	6.9	0.7	20.7	9.1	45.6	47.1	38
Zircaloy-2.5 wt % Nb	0	1×10^{21}[b]	290	RT	117.0	158.0	126.0	162.0			14.2	7.5		44.0	36
Zircaloy-2.5 wt % Nb	13	1×10^{21}[b]	290	RT	115.0	157.0	124.0	161.0			14.2	7.2		43.0	36
Zircaloy-2.5 wt % Nb	100	1×10^{21}[b]	290	RT	115.0	155.0	126.0	157.0			12.3	5.3		30.0	36
Zircaloy-2.5 wt % Nb	250	1×10^{21}[b]	290	RT	122.0	161.0	134.0	162.0			8.7	2.5		10.0	36
Zircaloy-2.5 wt % Nb	0	1×10^{21}[b]	290	300	79.3	112.0	89.9	116.0			14.8	9.5		65.0	36
Zircaloy-2.5 wt % Nb	13	1×10^{21}[b]	290	300	74.7	118.0	84.7	119.0			14.2	8.7		66.0	36
Zircaloy-2.5 wt % Nb	100	1×10^{21}[b]	290	300	78.1	115.0	87.9	118.0			12.9	7.5		53.0	36
Zircaloy-2.5 wt % Nb	250	1×10^{21}[b]	290	300	82.7	115.0	93.6	118.0			13.2	5.5		33.0	36
Zircaloy-2.5 wt % Nb	0	1×10^{21}	290	400	70.3	95.5	78.1	97.6			15.3	13.5			36
Zircaloy-2.5 wt % Nb	13	1×10^{21}	290	400	67.3	98.8	75.3	101.2			14.2	12.7			36
Zircaloy-2.5 wt % Nb	100	1×10^{21}	290	400	69.6	104.2	76.5	105.4			14.6	11.7			36
Zircaloy-2.5 wt % Nb	250	1×10^{21}	290	400	71.3	98.1	78.2	99.5			14.3	12.0			36
Containing Oxygen															
Zircaloy-2	555	5.2×10^{20}	>100	RT	41.0	76.0	72.5	81.6			22.5	13.5	42.5	50.0	36a
Zircaloy-2	1395	5.2×10^{20}	>100	RT	66.5	92.3	96.0	101.0			21.0	9.5	37.5	38.2	36a
Zircaloy-2	555	5.2×10^{20}	>100	250	18.5	47.0	38.5	50.6			39.3	16.6	64.1	61.5	36a
Zircaloy-2	1395	5.2×10^{20}	>100	250	38.5	55.4	50.5	62.3			37.0	17.1	59.8	56.0	36a
Zircaloy-2	555	5.2×10^{20}	>100	350	15.5	39.8	32.5	40.2			38.2	21.6	68.8	51.4	36a
Zircaloy-2	1395	5.2×10^{20}	>100	350	21.0	44.8	39.0	48.7			36.8	25.8	64.6	53.8	36a
Zircaloy-3	640	5.2×10^{20}	>100	RT	31.3	63.5	65.0	72.2			22.8	14.7	43.0	37.0	36a
Zircaloy-3	1395	5.2×10^{20}	>100	RT	53.8	91.6	89.5	101.0			21.2	12.2	39.8	34.1	36a
Zircaloy-3	640	5.2×10^{20}	>100	250	13.0	40.4	30.8	41.3			49.6	23.8	72.0	65.6	36a
Zircaloy-3	1395	5.2×10^{20}	>100	250	21.0	52.8	43.0	57.0			42.6	26.2	65.0	55.1	36a
Zircaloy-3	640	5.2×10^{20}	>100	350	10.5	32.5	26.0	33.2			47.7	29.2	77.0	63.5	36a
Zircaloy-3	1395	5.2×10^{20}	>100	350	15.5	40.5	32.5	43.2			42.6	33.7	71.5	64.2	36a

(a) W - welded, A - annealed, E - extruded.
(b) Neutron energy >0.5 MeV.

Burst Strength

Burst tests have been performed on Zircaloy-2 tubes which had been irradiated as fuel-element claddings. At fluences of about 10^{20} n/cm^2, only minor increases in strength and decreases in ductility were found as illustrated in Figure 18.[39]

Notch Sensitivity and Impact Properties

It has been shown that irradiation to a fast fluence of 2.5 x 10^{21} n/cm^2 has only minor effects on the impact properties of zirconium alloys.[40] Irradiation of welded[36] and cold-worked[41] Zircaloy-2 has not induced any change in impact properties.

However one must be aware that hydrogen which results from irradiation may have considerable effect on the impact properties of Zircaloy-2[42]. The hydrogen causes a ductile-to-brittle transition temperature which increases with increasing hydrogen content. Irradiation of the Zircaloy-2 and the zirconium-2.5 wt % niobium alloy which contains hydrogen increases the transition temperature, as illustrated in Figure 19. The magnitude of the irradiation-induced transition-temperature shift in hydrided material decreases with increasing hydrogen content. This indicates that the transition-temperature shifts caused by hydrogen and irradiation are not additive. It can be seen that a larger irradiation-induced transition-temperature shift occurs in hydrided zirconium-2.5 wt % niobium than in Zircaloy-2 after an equal fluence.[43] The transition-temperature shift in hydrided zirconium alloys does not appear to be fast-fluence dependent above 2 x 10^{20} n/cm^2.

Stress Relaxation

In annealed Zircaloy-2, it has been found that less stress relaxation takes place at 100 C and below during irradiation than in unirradiated material loaded at the same stress.[44] Both the unirradiated material and material irradiated to a fast fluence of 8 x 10^{20} n/cm^2 were originally stressed at 25,000 psi. The final stress for the irradiated material was 19,000 psi, while that for the unirradiated material was 15,000 psi. However, cold-worked material behaved in the opposite manner. Zircaloy-2 specimens cold worked 10 percent were initially loaded at 15,000 psi. The out-of-pile unirradiated specimens relaxed to 12,000 psi, while the

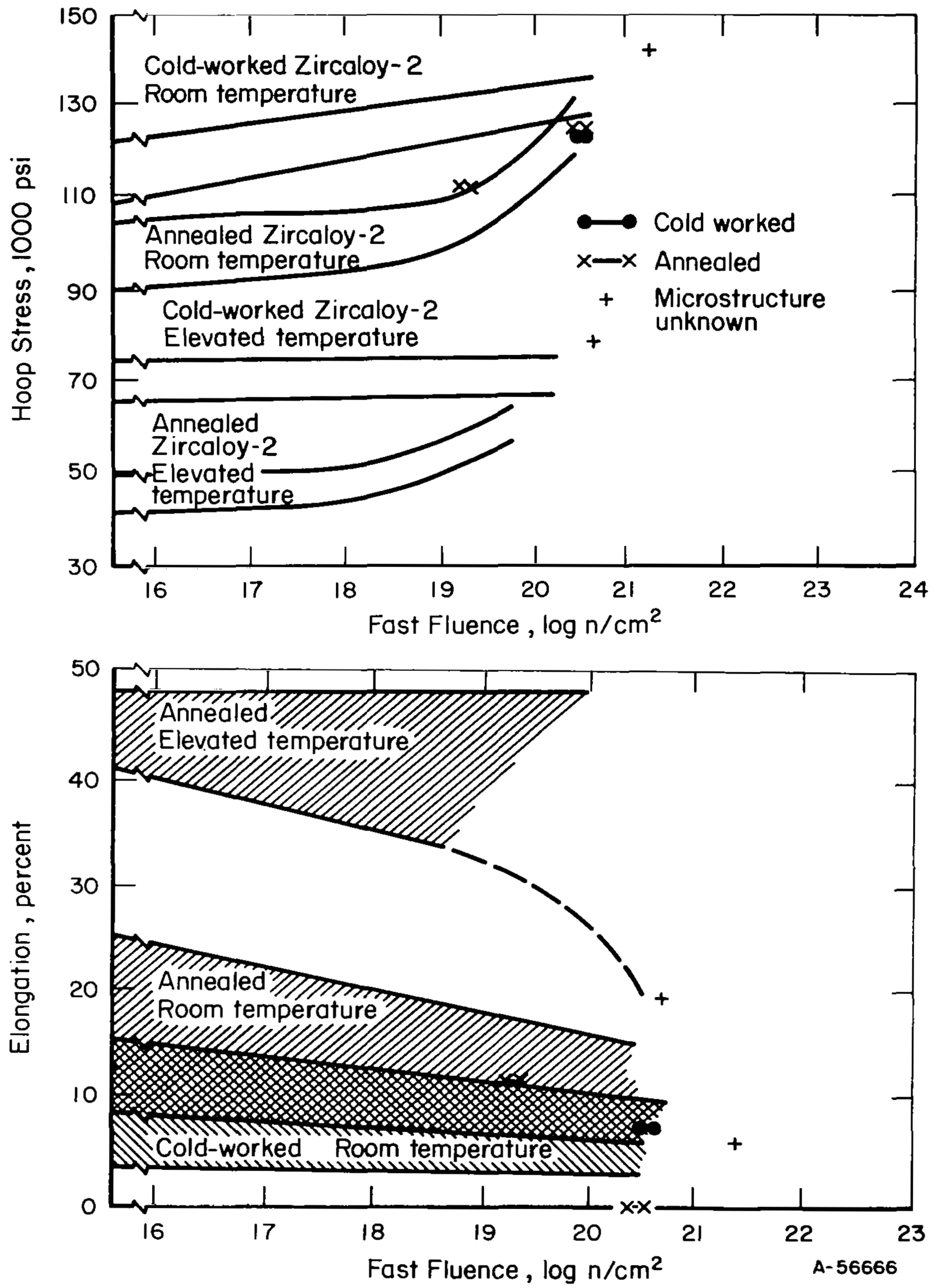

FIGURE 18. HOOP STRENGTH AND ELONGATION AS A FUNCTION OF FAST FLUENCE IN PRTR ZIRCALOY-2 PRESSURE TUBES(39)

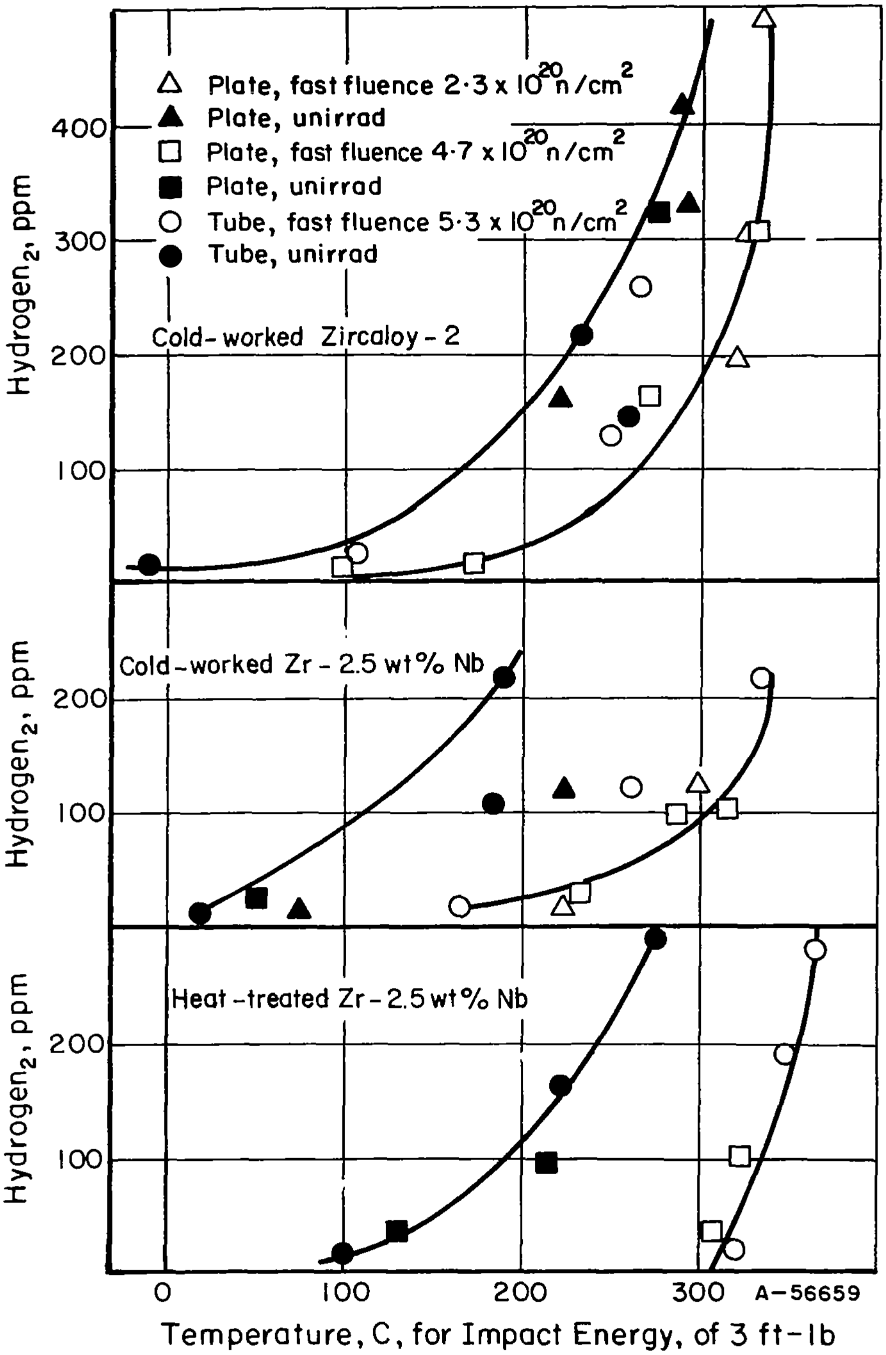

FIGURE 19. SUMMARY OF COMBINED EFFECTS OF HYDROGEN AND IRRADIATION ON IMPACT PROPERTIES OF ZIRCALOY-2 AND ZIRCONIUM-2.5 WT % NIOBIUM ALLOY[43]

specimens in-pile relaxed to 9,000 psi. The in-pile specimens received a fast fluence of 8 x 10^{20} n/cm^2, test temperature being less than 100 C.

Creep

Creep rates for 20 percent cold-worked Zircaloy-2 tubes at 258 C are shown in Figure 20 in-pile experiments.(45) It can be seen that irradiation increases the creep rate by about ten fold. The creep rate at 300 and 350 C is given in Figure 21.(46)

Zirconium-2.5 wt % Niobium

This zirconium alloy has considerably higher strength than the Zircaloys. Also it can be heat treated to obtain the desired strength and ductility.

Table 11 enumerates the tensile properties of irradiated zirconium-2.5 wt % niobium alloy.(47, 48)

The in-pile creep properties of Zirconium-2.5 wt % niobium at 300 C are illustrated in Figure 22.(46) It should be noted that at lower stresses the in-pile creep rate is higher than that of unirradiated material. However, at higher stresses the in-pile creep rate is lower than that of unirradiated material. This changeover is explained in terms of irradiation induced defect clusters acting as effective barriers to dislocation movement at higher stresses. The different creep rates as found by the two investigators apparently reflect differences in lot to lot variation in creep properties.

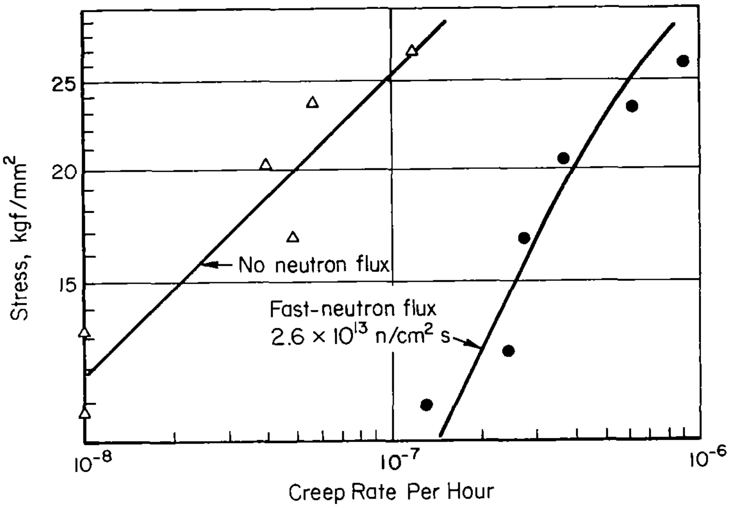

FIGURE 20. TUBULAR CREEP RATES OF ZIRCALOY-2 IN AND OUT OF FAST-NEUTRON FLUX AT 9000 HOURS AND 258 C[45]

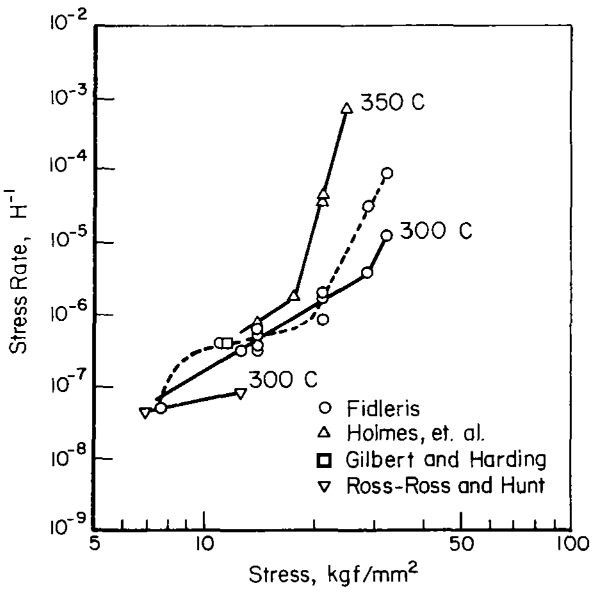

FIGURE 21. STRESS-DEPENDENCE OF THE IN-REACTOR STRAIN RATE OF COLD-WORKED ZIRCOLOY-2 AT 300 AND 350 C[46]

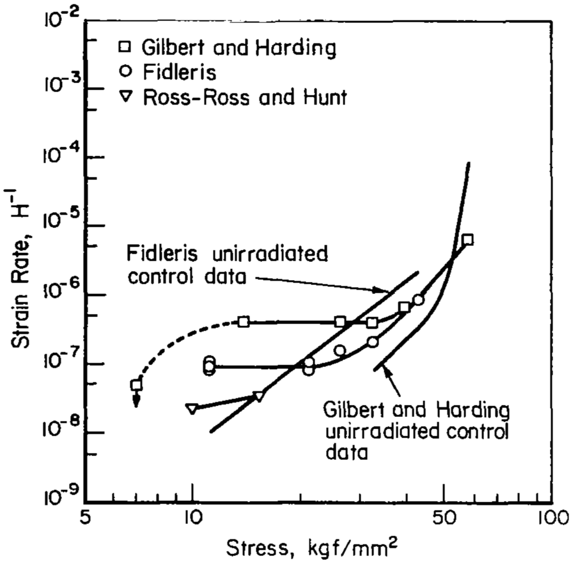

FIGURE 22. STRESS DEPENDENCE OF STRAIN RATE FOR IN-REACTOR AND UNIRRADIATED CONTROL CREEP OF Zr-2.5Nb AT 300 C

TABLE 11. TENSILE PROPERTIES OF UNIRRADIATED AND IRRADIATED ZIRCONIUM-2.5 wt % NIOBIUM ALLOY

Material Condition(a)	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp., C	Test Temp., C	Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, percent: Uniform		Elongation, percent: Total		Reduction in Area, percent		Reference
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Q-B, A-500	3×10^{20}	250-325	RT	113	153	170	160	4	<1	10	1	>25	>5	47
Q-B, A-500	1×10^{20}	250	300	85	97	93	102			14.2	13.3			48
Q-B, A-500	3×10^{20}	250-325	300	81	114	133	126	3	1.5	12	2	56	<20	47
SC-B, 40 CW	3×10^{20}	250-325	RT	97	131	122	152	1	1			30	26	47
SC-B, 40 CW	3×10^{20}	250-325	300	60	101	90	116	1	1			34	26	47
QB, A-500	1×10^{20}	250	RT		136		142				10.5			48
SC-(A + B)	3×10^{20}	250-325	RT	61	107	132	150	12	3.0			56	37	47
SC-(A + B)	3×10^{20}	250-325	300	32	80	100	120	18	2.0	30	11	72	60	47
SC-B	3×10^{20}	250-325	RT	73	100	130	123	9	10			44	13	47
SC-B	3×10^{20}	250-325	300	34	68	92	120	5	4	12	9	67	40	47
Q-(A + B)	3×10^{20}	250-325	RT	109	142	194	210	4	1			60	55	47
A-500	3×10^{20}	250-325	300	85	116	167	175	3	<1	12	11	70	65	47
Q-B	1×10^{20}	250	300	77	108	87	115			14.5	11.3			48

(a) Explanation of designations:

A - annealed; SC - slow cooled; Q - quenched
40 CW - 40 percent cold worked; (A + B) - from (alpha + beta) phase
A-500 - annealed at 500 C for 1 hour; B - from beta phase

FERRITIC AND MARTENSITIC STEELS

Pressure-Vessel Steels

Even though having little application for space systems the ferritic steels are included for completeness. Mostly, these steels have been used for pressure-vessel construction. Since the ferritic steels have a body-centered crystal structure, their fracture mode changes from ductile to brittle with decreasing temperature. In a very narrow temperature range, the impact strength of body-centered cubic materials decreases rapidly and the temperature at which the impact strength has a specific value is generally known as ductile-to-brittle transition temperature. Irradiation-induced shift of this transition temperature has been of the greatest concern for reactor-pressure-vessel design. Pressure vessels for nonnuclear applications operate in the 70 to 500 F temperature range, above the steel's brittle-to-ductile transition temperature, and, therefore, the steel's ductility is maintained. However, if irradiation increases the ductile-to-brittle transition temperature into the operating temperature range, then a possibility of catastrophic brittle failure of the pressure vessel is introduced. The ASME Pressure Vessel Code specifies that the reactor pressure vessel should operate 60 F above the ductile-to-brittle transition temperature.

Brittle-to-Ductile Transition Temperature

The magnitude of the irradiation-induced upward shift of the ductile-to-brittle transition temperature, the factors affecting it, and possible ways of minimizing it have received much of the attention by investigators in recent years. In comparing changes in ductile-to-brittle transition temperatures, great care should be taken that the temperature changes are measured by the same criterion and that the results are interchangeable. This is especially applicable when test results from different countries are compared.

The ductile-to-brittle transition temperature may be determined by testing Charpy V-notch specimens at varying temperatures and making a plot of impact strength versus temperature. A similar plot can be made by using Izod specimens instead of the Charpy V-notch specimens. The transition temperature chosen is the temperature where the impact strength has a specific value. In the United States, 30 ft-lb is usually taken as a standard for most steels, although lower values have been used for steels that have impact strengths below 30 ft-lb. Another way of determining transition temperature is by observing the amount of shear fracture on the fracture surfaces of fractured impact specimens tested at various temperatures. The transition temperature then, by definition, occurs where 50 percent of the fracture has taken place by shear. Transition temperature can also be determined by using a standard drop-weight test.[49] In this test, a weight is dropped on a weld beaded metal plate; at the nil-ductility temperature (NDT), a crack will develop in the plate as a result of the dropped weight.

Successful correlations between the energy criterion of 30 ft-lb and 50 percent shear have not been obtained for all steels. However, there are excellent correlations between NDT and the 30-ft-lb criterion for most unirradiated and irradiated steels.[50] Although most of the experimental data have been obtained with impact tests, the NDT term has gained wide acceptance and will be used interchangeably with ductile-to-brittle transition temperature throughout this discussion.

Figure 23 shows the irradiation-induced change in the NDT for a number of steels.[51] Since these steels differ in composition, and the specimens were irradiated in different reactors and tested at different sites, it is significant that the results indicated a direct dependence of the NDT shift on total fast fluence. The amount of scatter is not excessive considering all the variables involved. Attempts have been made to identify the specific variables which affect the irradiation sensitivity of a steel. It must be emphasized that "sensitivity to irradiation" refers to the magnitude of the NDT shift and not the original NDT of the material. The effects of certain variables on the irradiation-induced NDT shift in pressure vessel steels are discussed below.

(1) Structure. A ferritic structure, resulting from slowly cooling from the heat-treat temperature, is found to be most sensitive to irradiation embrittlement. A martensitic structure obtained by quenching and tempering was found to be the most resistant to irradiation-induced NDT shift.[52]

(2) Composition. Composition and heat treatment are the most important factors in determining the ductile-to-brittle transition temperature in unirradiated steels. Since microstructure, which is dependent on composition and heat treatment,

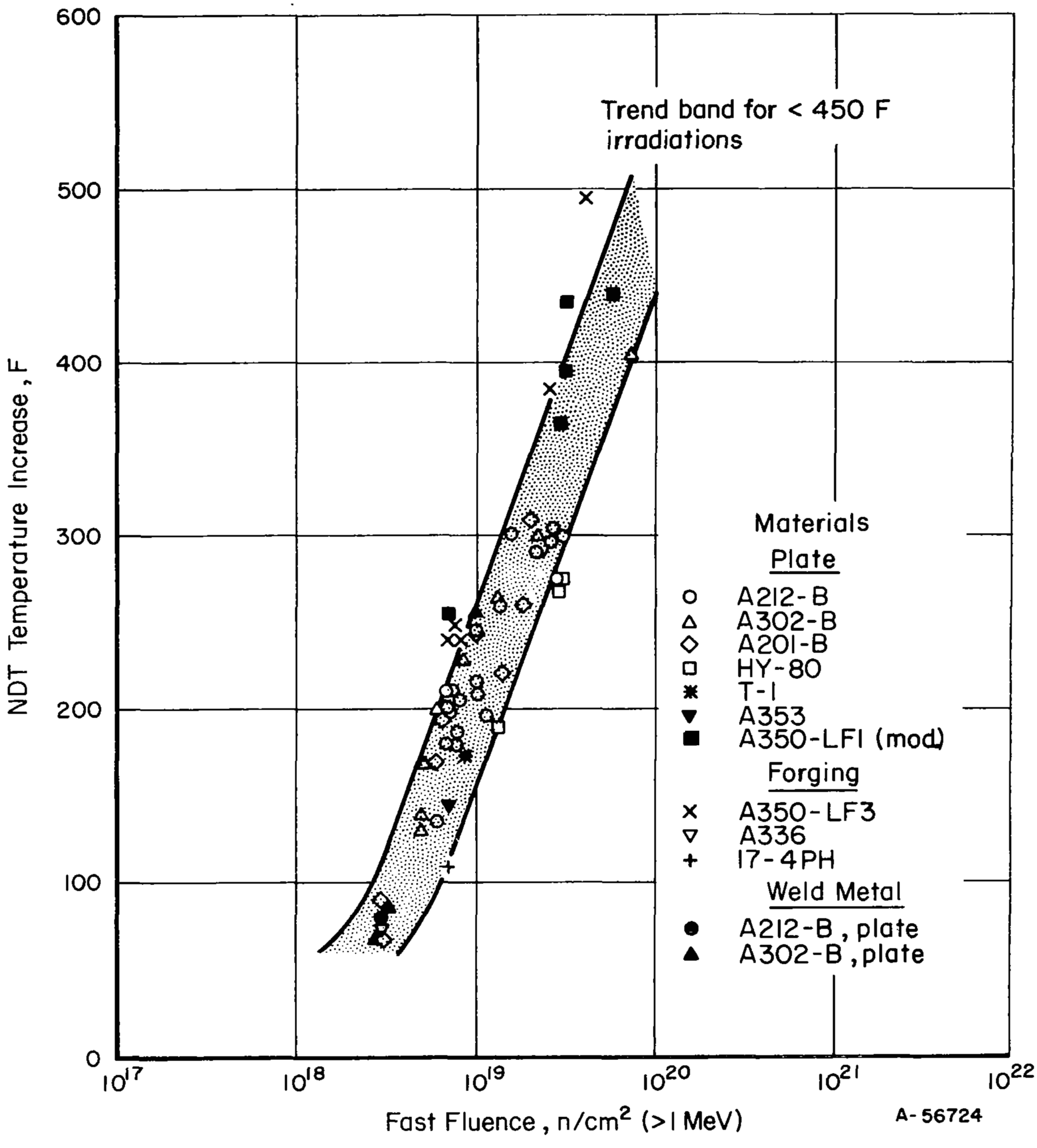

FIGURE 23. INCREASE IN THE NDT TEMPERATURES OF STEELS RESULTING FROM IRRADIATION AT TEMPERATURES BELOW 450 F(51)

is important in predicting steel sensitivity to irradiation it becomes difficult to separate the effect of chemical composition from those of microstructure. Most composition changes affect the irradiation sensitivity of the steel by changing its heat treatment response and consequently the final microstructure.[52] However, it has been definitely established that the presence of copper and phosphorus increase the irradiation induced NDT increase.[53] Also, the induction-melting removal of residual elements has improved the irradiation resistance of steel[52]. This finding agrees with most theoretical predictions attributing irradiation hardening to interstitial atoms.[54]

(3) Stress. It was found that a material stressed 20 percent of its yield strength underwent a smaller irradiation-induced NDT shift than did an unstressed material receiving the same fluence.[55] In another test it was found that stress (80 percent of yield strength) had no effect on the irradiation induced NDT shift.[56]

(4) Direction. Radiation response of longitudinal and transverse (to rolling direction) specimens of A212-B and A302-B plate have shown that both transverse and longitudinal specimens are about equally sensitive to irradiation to a fast fluence of 1.1×10^{20} n/cm^2.[57]

(5) Welding. Welding results in a heat-affected zone (HAZ) comparable to that of annealed material. As mentioned earlier, annealing at high temperatures increases NDT in unirradiated materials. Therefore the irradiation induced NDT shift is a function of the type of microstructure that results from welding.

(6) Temperature. Temperature during irradiation is a very important factor in influencing the degree of irradiation embrittlement. Figure 24 illustrates how irradiation temperature affects the transition-temperature shift in various steels.[51] Maximum embrittlement has been found to be caused by irradiation temperatures below 450 F. The NDT shift progressively decreases with increasing irradiation temperature because the

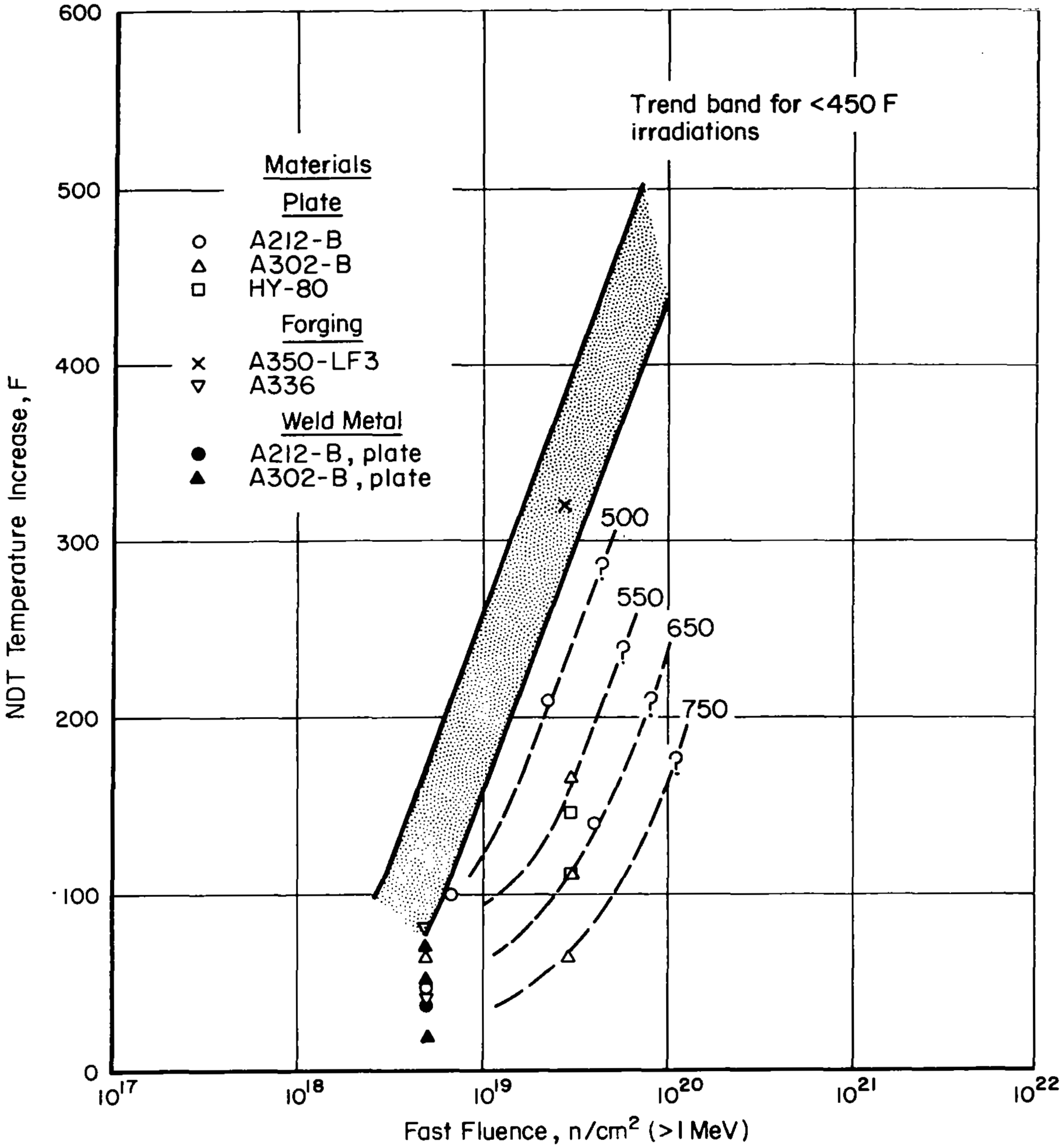

FIGURE 24. INCREASE IN THE NDT TEMPERATURES OF STEELS RESULTING FROM IRRADIATION AT TEMPERATURES ABOVE 450 F[51]

Numbers indicate irradiation temperatures.

less stable defect clusters anneal out at the higher irradiation temperature. At the higher irradiation temperatures, only the most stable defect clusters remain so and, consequently, only minor changes in the transition temperature can be expected.

(7) Saturation. The fast fluence at which saturation of the irradiation induced NDT increase occurs varies for different steels. Generally the saturation occurs at lower fluences for steels which undergo lower NDT increases.

(8) Annealing. Results of annealing studies at various temperatures on irradiated A350-LFI are shown in Figure 25.[58] In another experiment, it was found that higher irradiation temperature decreases the percentage of the NDT shift recoverable at equivalent annealing temperatures. For example, specimens of A212-B were irradiated to equivalent fast fluences, at both 275 and 510 F. After irradiation, both types of specimens were annealed for 36 hours at 750 F. The specimens irradiated at 275 F recovered 85.6 percent of their irradiation-induced NDT shift, while the specimens irradiated at 510 F recovered only 64 percent of their NDT shift. However, a larger percentage of the irradiation-induced NDT shift was recovered for specimens irradiated at higher temperatures and annealed at a fixed increment above the irradiation temperature. For example, specimens were irradiated at 640 and 750 F to a fast fluence of 3 x 10^{19} n/cm^2. After irradiation, specimens irradiated at 640 F were annealed at 800 F and the specimens irradiated at 740 F were annealed at 900 F. In this case, the specimens irradiated at 640 F recovered 64 percent of their NDT shift and the specimens irradiated at 740 F recovered 85 percent of their NDT shift, by an anneal 160 F above their respective irradiation temperatures.

A very important concept for a pressure vessel is the possibility of in-place annealing which could minimize the NDT shift during the pressure-vessel lifetime. Cyclic irradiation-annealing treatments have been performed on A302-B steel.[58] These steels were irradiated to a fast fluence of 1 x 10^{19} n/cm^2 at 240 F, annealed for 24 hours at 650 C, further irradiated, and annealed again. The results of these studies are shown in Figure 26. It can be seen that the cumulative increase in NDT

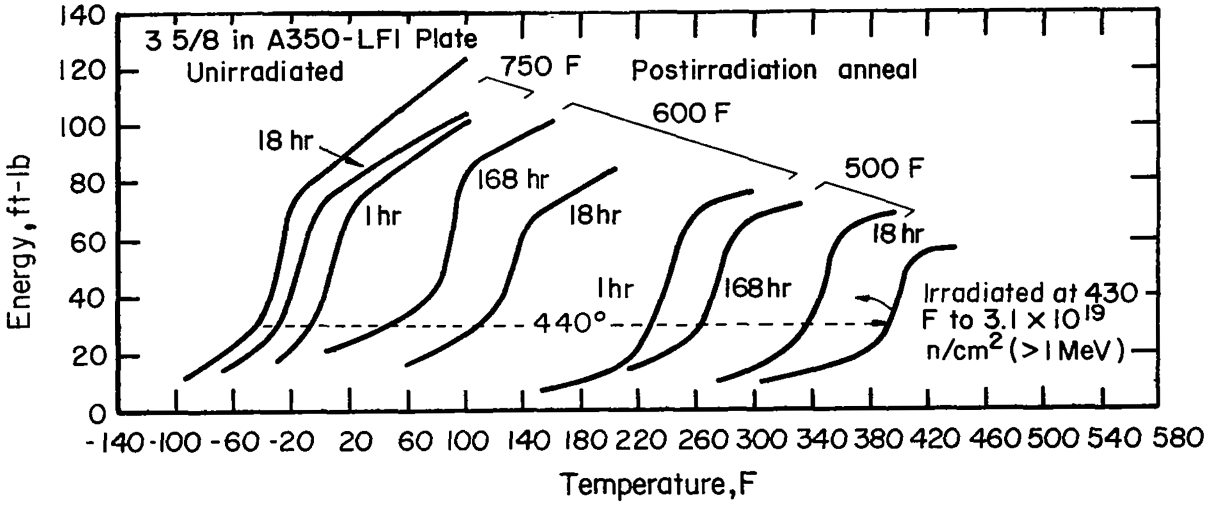

FIGURE 25. NOTCH-DUCTILITY CHARACTERISTICS OF IRRADIATED A350-LF1 STEEL(56)

Results of postirradiation annealing show effects of various heat treatment (time-temperature) combinations.

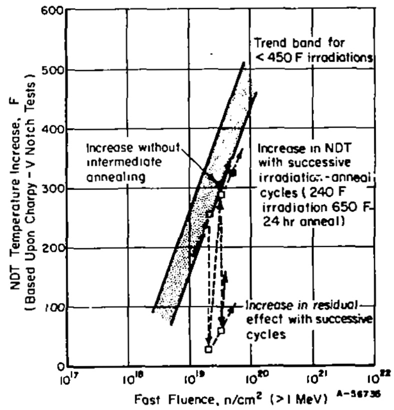

FIGURE 26. NDT TEMPERATURE BEHAVIOR EXHIBITED BY TWO STEELS AT VARIOUS STAGES OF CYCLIC IRRADIATION-ANNEALING TREATMENTS(58)

is somewhat less for the cyclically irradiated and annealed material than for the material irradiated to the same fluence without intervening annealing.

(9) Location in Reactor. A reactor pressure vessel may be sufficient thickness (up to 12 inches), not applicable to space, that the outside sections of the pressure vessel receive considerably fewer fast neutrons. This effect was demonstrated by irradiating Charpy V-notch specimens imbedded in a steel block simulating an actual pressure-vessel location.[59] The specimens received a relative fast-fluence variation of about nine, depending on their location; the consequent shifts in NDT are shown in Figure 27.

(10) Grain Size. It has been demonstrated that the NDT increases considerably more in large-grained steels than in fine-grained steels during irradiation (Figure 28).[60]

(11) Prestrain. To test the effect of strain on irradiation-induced NDT shift, Charpy V-notch specimens of Swedish 2103/R3 steel were elongated to 10 percent and aged for 1 hour at 480 F before irradiation.[61] The strain-aged specimens were irradiated along with unstrained control specimens to a fast fluence of 2.4×10^{18} n/cm^2. Results of the impact tests are given in Table 12. From these results it was concluded that the transition-temperature shifts caused by irradiation and straining are additive.

Tensile Properties

It has been definitely established that the irradiation-induced increase in tensile properties is a function of fast fluence and is normally independent of flux except at extremely high values. Specimens were irradiated at instantaneous fluxes ranging from 2×10^{11} to 3×10^{13} n/(cm$^2\cdot$ s) with the total fluence being 4.6×10^{18} n/cm^2. However, the yield strength increase was found to be the same for all specimens.[62]

The effect of fast fluence and irradiation temperature on the changes of tensile properties of mild steel are illustrated in Figure 29.[63] It must be remembered that all mild steels behave somewhat differently although the general trends are about the same.

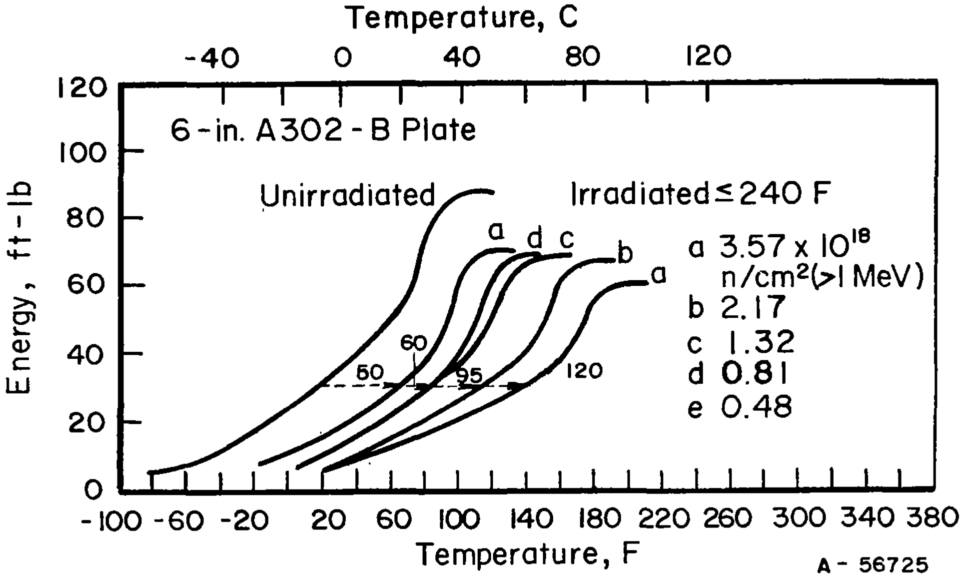

FIGURE 27. NOTCH-DUCTILITY PROPERTIES OF A302-B STEEL AT FIVE LOCATIONS INSIDE TEST BLOCK(59)

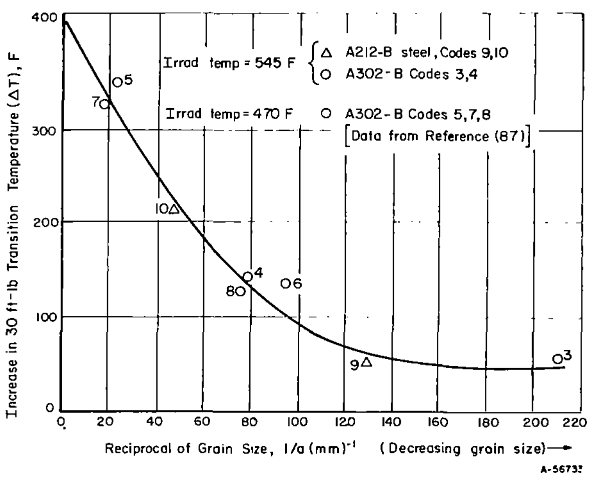

FIGURE 28. INCREASE IN 30 FT-LB TRANSITION TEMPERATURE VERSUS RECIPROCAL GRAIN RADIUS UPON IRRADIATION TO A FAST FLUENCE OF 3 x 10^{19} N/CM^2 (>1 MEV) AT 470 AND 545 F(60)

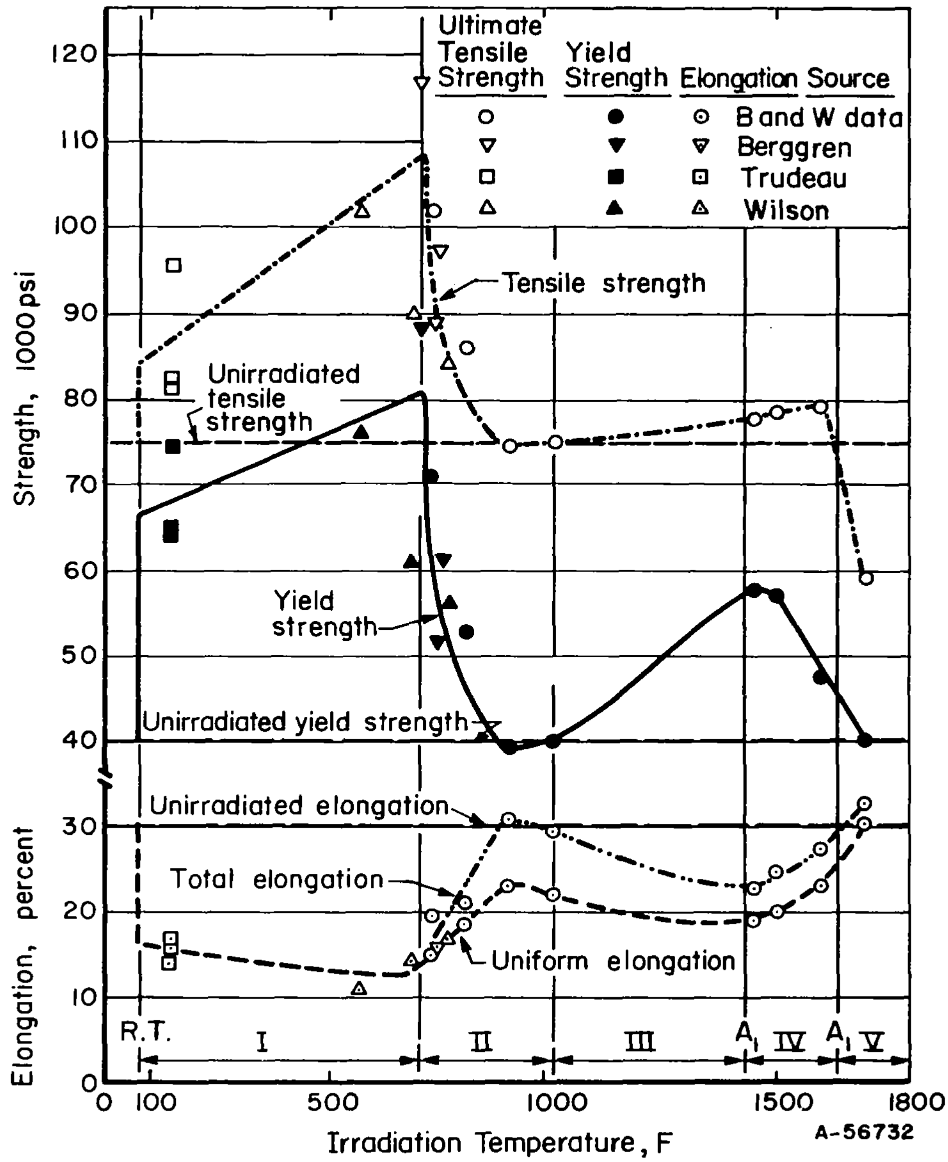

FIGURE 29. GENERAL EFFECT OF IRRADIATION TEMPERATURE ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF LOW-ALLOY STEELS IRRADIATED TO ~3 x 10^{19} N/CM^2 (> 1 MEV)[63]

TABLE 12. EFFECT OF STRAIN AGING ON IRRADIATION EMBRITTLEMENT OF 2103/R3 STEEL[61]

	Control	Strain Aged
Transition temperature (15 ft-lb), unirradiated, F	-139	-58
Transition temperature (15 ft-lb), irradiated, F	-67	32
Shift in transition temperature, F	72	90

Irradiation at increasing temperatures above 700 F induces progressively smaller changes in tensile properties until no significant changes are produced by irradiation at 900 F. However, irradiation at still higher temperatures induces a secondary irradiation hardening which reaches a maximum at an irradiation temperature of about 1400 F. This secondary irradiation hardening is attributed to changes in microstructure with increased atomic mobilities at the elevated temperature. Irradiation-induced accelerated precipitation of Mo_2C particles is believed to be the cause of increased strength and decreased ductility.

The effect of postirradiation annealing on the room temperature tensile properties of mild steel is illustrated in Figure 30.[64] It can be seen that as the annealing temperature is increased the strength of the irradiated mild steel is increased. This increase in strength is attributed to "thermal hardening". "Thermal hardening" is defined as the agglomeration of small irradiation-induced defects into more stable defects during annealing. These stabler defects are more able to impede the movement of dislocations during deformation. The "maximum thermal hardening" occurs when the optimum defect sizes and distributions for impeding dislocation movement are obtained. As the annealing temperatures are increased, the defects become larger in size but fewer in number and, consequently, their net effect on dislocation movement decreases. The temperature at which maximum "thermal hardening" occurs varies for the different steels. The lowest temperature for maximum "thermal hardening" reported is 210 F.[65] Generally, one would expect maximum thermal hardening temperature to be higher for high alloy steels.[66]

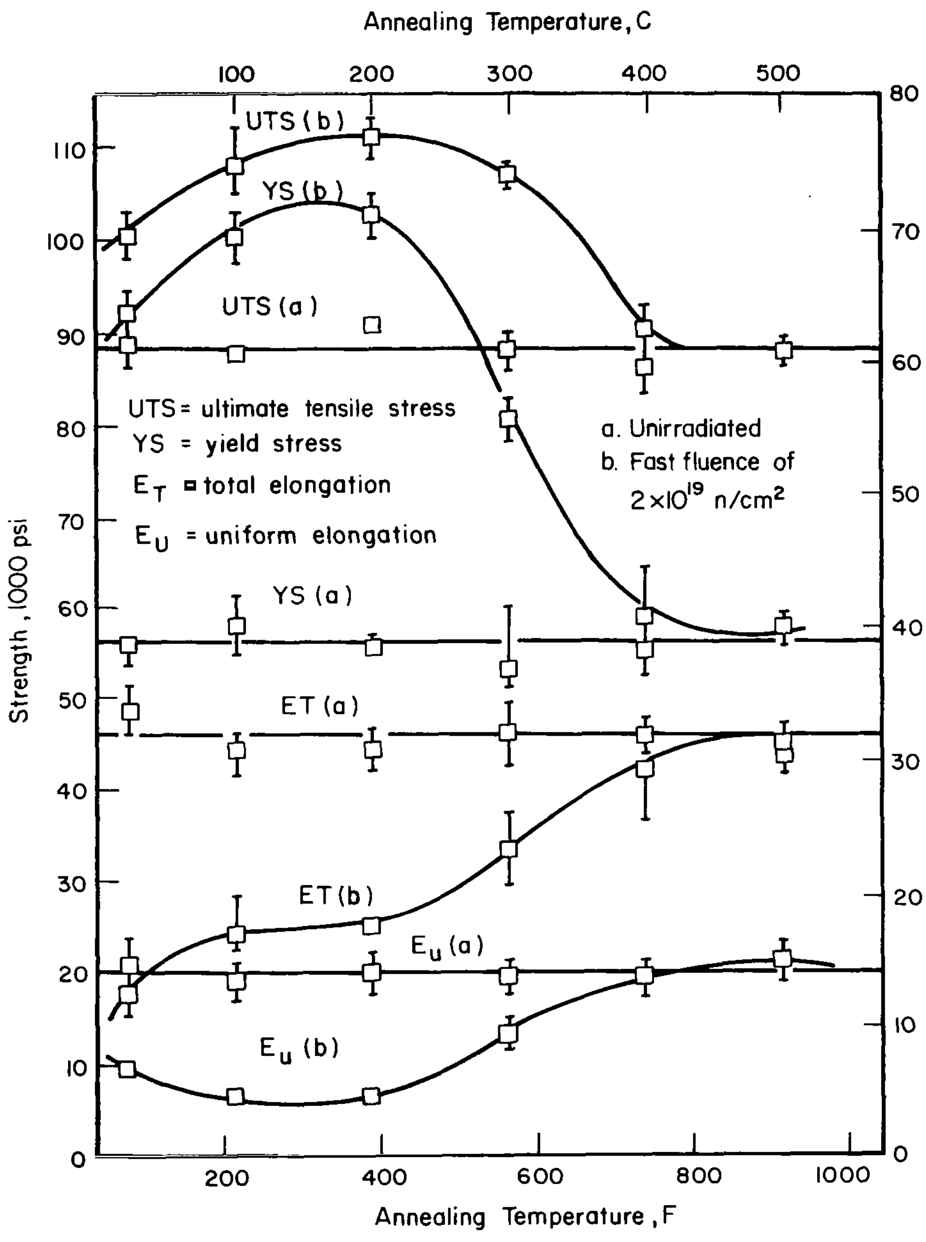

FIGURE 30. EFFECT OF POSTIRRADIATION ANNEALING ON THE ROOM-TEMPERATURE MECHNAICAL PROPERTIES OF IRRADIATED LOW-ALLOY STEEL(64)

Fatigue Properties

Bending-type fatigue tests were performed in-pile on ASTM Type A302-B steel at 500 F with maximum fast fluences being 1.1×10^{19} n/cm^2. Results of these tests are shown in Figure 31.[67] No definite conclusions can be drawn by the limited number of tests, although it appears that the unirradiated and irradiated material behave similarly. The irradiated material shows somewhat better performance above 50,000 cycles. In another study, specimens of A212-B and A302 were tested after irradiation to a fast fluence of 6×10^{19} n/cm^2. In all cases, irradiation increased the endurance limit, performance in the elastic range, while it decreased the low-cycle life, performance in plastic range. The improvement of the endurance limit takes place after about 10^4 to 10^5 cycles.[68].

Sliding Characteristics

Conflicting reports on irradiation effects on sliding characteristics of steels have been published. Tool steels were irradiated at 800 to 1200 F to a fast fluence of 1.6×10^{19} n/cm^2 without any changes in sliding characteristics.[69] It also has been reported that a fast fluence of 1×10^{18} n/cm^2 decreased the wear resistance of carbon steels at room temperature.[70]

Magnetic Properties

Magnetic properties of A212-B pressure vessel steel were unaffected by fast fluences of 1×10^{20} to 1.3×10^{21} n/cm^2.[71]

Gamma Irradiation

A steel containing 0.08 wt % carbon, 0.05 wt % silicon, and 0.26 wt % manganese was given a gamma dose of 1.73×10^7 R from a cobalt-60 source.[72] The irradiation formed vacancies and interstitials but did not change the room-temperature tensile strength. However, the 100-hour stress-rupture strength at 842 F was increased from 27,300 to 29,400 psi.

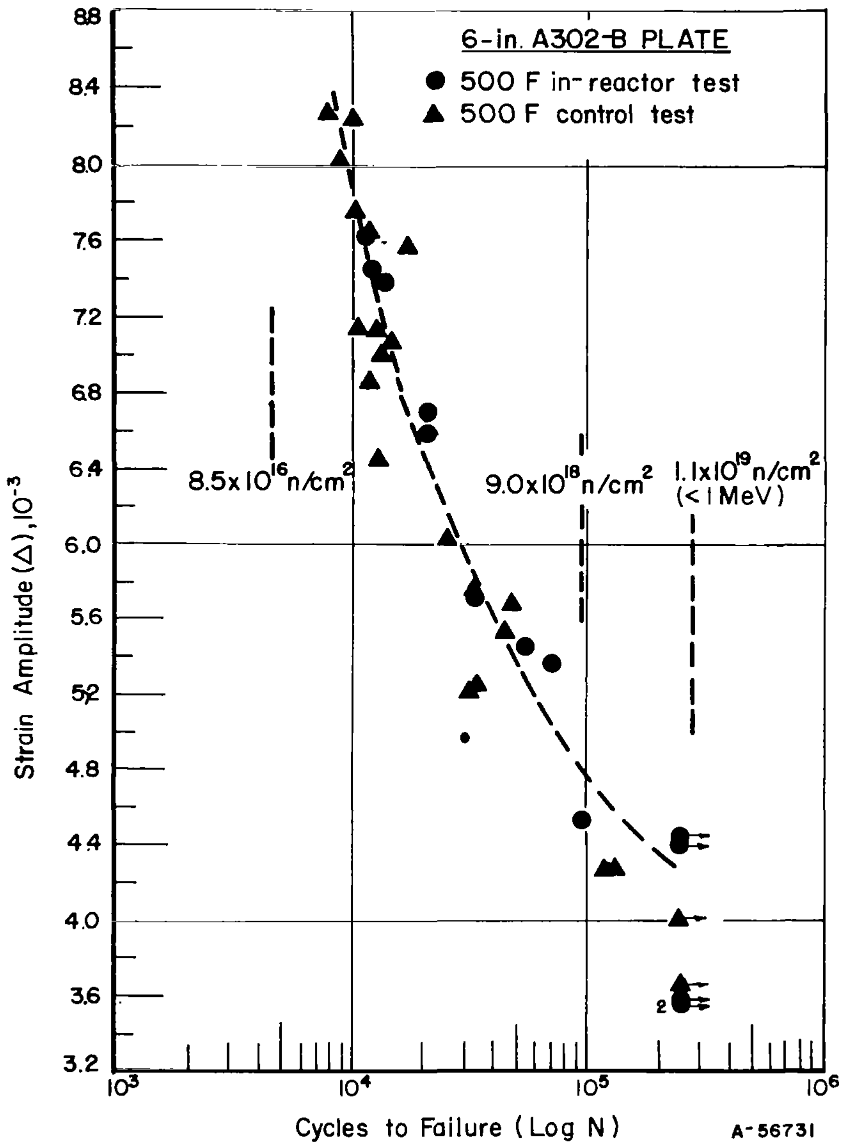

FIGURE 31. COMPARISON OF FATIGUE DATA FOR ASTM TYPE A302-B STEEL DEVELOPED BY IN-REACTOR TESTS AT 500 F WITH THE RESULTS OF OUT-OF-REACTOR CONTROL TESTS(67)

The measurements of strain amplitude were performed at room temperature.

Structural Steels

In this section, ferritic and martensitic steels used in applications other than reactor pressure-vessel construction are discussed. Table 13 lists the steel compositions. The martensitic steels and 17-4PH are used in reactor components, while AISI-406 stainless steel and Alloy 1541 are being considered as cladding materials.

TABLE 13. CHEMICAL COMPOSITION OF FERRITIC AND MARTENSITIC STEELS

	Composition, wt %												
Material	Fe	Cr	Ni	Mn	Si	Al	Co	S	Y	C	Mo	Nb	P
AISI 406	Bal	13.1	--	0.42	0.5	3.9		0.012		0.07			
AISI 410	Bal	11.5-13.5	--	1.0[a]	1.0[a]			0.03[a]		0.15[a]			0.04[a]
AISI 414	Bal	11.5-13.5	1.25-2.5	1.0[a]	1.0[a]			0.03[a]		0.15[a]			0.04[a]
AISI 420	Bal	12-14		1.0[a]	1.0[a]			0.03[a]		0.15[b]			0.04[a]
AISI 440	Bal	16-18	--	1.0[a]	1.0[a]			0.03[a]		0.6-1.2	0.75[a]		0.04[a]
AM-350	Bal	16.6	4.4	0.78	0.48					0.10		2.6	
17-4PH	Bal	17	3-5	1.0[a]	1.0[a]					0.07	0.5	0.35	
1541 alloy[c]	Bal	15				4			1.0				

(a) Maximum.
(b) Minimum.
(c) Contains 1 wt % yttrium.

Tensile Properties

The effects of irradiation on the tensile properties of ferritic stainless steels are shown in Table 14. These data show that irradiation increases the room-temperature yield and ultimate strength, while the ductility is decreased. Irradiation was found to affect AISI 410 to a greater degree in the annealed condition than in the martensitic condition. An irradiation temperature of 260 to 290 C does not appear to temper the martensite. Figure 32 illustrates the dependence of property changes in AM-350 and 17-4PH on the fast fluence.[2] These results indicate that only minor changes in tensile properties occur after fast fluences of 2×10^{21} n/cm^2.

A limited number of tensile tests at elevated temperatures have been performed on irradiated ferritic stainless steels. The effect of irradiation at 280 C on the tensile properties of Type 406 stainless steel at various temperatures can be obtained by comparing Figures 33 and 34.[84] Figures 33 and 34 show the tensile properties of the unirradiated and irradiated material,

TABLE 14. EFFECTS OF IRRADIATION ON TENSILE PROPERTIES OF FERRITIC AND MARTENSITIC STEELS

Material	Condition[a]	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp., C	Test Temp., C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Total Elongation		Reduction in Area, percent		Ref.
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
406 SS		3.0×10^{19}	650	RT	56	70	89	91	23	25			70
406 SS		4.0×10^{20}	~400	RT	62.9	100			24.8	16.5			71
406 SS		4.0×10^{20}	~400	595	29.3	33.9			54.4	32.2			71
406 SS		3.0×10^{19}	650	700	14.7	31	14	42	74	54			70
410 SS	TD-M	3.7×10^{19}	260-290	RT		58.7		72.7		18.0			70
410 SS	A	4.2×10^{19}	265	RT	38	56	59.5	77.5	32.5	27.7			72
410 SS	TD-A	4.5×10^{19}	260-290	RT	38.3	67.0	67.8	85	26.0	19.5			70
410 SS	A	4.9×10^{19}	265	RT	38	57	59.5	78	32.5	26.2			72
410 SS	A	5.6×10^{19}	265	RT	38	59	59.5	78	32.5	25.0			72
410 SS		6.0×10^{19}	265	RT	38	56.5	59.5	85.5	32.5	23.9			72
410 SS	A	7.2×10^{19}	265	RT	38	60	59.5	80.5	32.5	26.6			72
410 SS	A	8.3×10^{19}	265	RT	38	61	59.5	80	32.5	26.0			72
410 SS	TD-A	1.0×10^{20}	260-290	RT	38.9	75.2	66.6	89.8	25.7	20.7			72
410 SS	TD-A	1.0×10^{20}	260-290	RT	38.6	92.9	65.6	97.3	31.3	8.8			70
410 SS	TD-A	1.1×10^{20}	50	RT		87.8		87.8		2.8			12
410 SS	A	1.2×10^{20}	50	RT	38	89	59.5	92	32.5	15.3			72
410 SS	TD-A	2.4×10^{20}	260-290	RT	38.5	105.8	66.0	106.2	36.6	4.6			70
410 SS	TD-M	2.5×10^{20}	260-290	RT	37.6	108.2	67.6	108.2	31.5	4.0			70
410 SS	T	7.0×10^{19}	<100	RT	47	62	69	70					73
410 SS	T	7.0×10^{19}	<100	RT	44	61	70	75					73
410 SS	T	3.0×10^{20}	315-370	RT	70	86	92.3	102.8	32	21			74
410 SS	M	1.2×10^{19}	35	RT	147.7	175.5	177	225	20	17			75
410 SS	TD-M	3.7×10^{19}	260-290	RT		174.3		200.0		5.6			70
410 SS	TD-M	4.5×10^{19}	260-290	RT	140.7	176.3	174.2	223.4	6.8	8.0			70
410 SS	TD-M	1.0×10^{20}	260-290	RT	146.0	181.9	173.4	208.4	6.6	6.5			70
410 SS	TD-M	1.0×10^{20}	260-290	RT	133.9	194.0	168.9	216.7	8.9	4.9			70
410 SS	TD-A	1.2×10^{20}	260-290	RT	140.6	182.4	174.2	210.7	7.9	6.4			70
410 SS		1.5×10^{20}	35	RT	147.7	198	177	233	20	11			75
410 SS	TD-M	1.6×10^{20}	50	RT		166.2		192.0		4.3			12
410 SS	TD-M	2.4×10^{20}	260-290	RT	139.5	190.9	169.5	215.6	7.9	5.5			70
410 SS	TD-M	2.5×10^{20}	260-290	RT	138.1	203.6	127.4	222.6	9.2	4.6			70
414 SS	M	6.4×10^{19}	35	RT		206		240		15			75
420 SS	A	5.0×10^{19}	>100	RT	47	90	83	95	19	10			76
440 SS	H	7.0×10^{19}	>100	RT	185	205	199	240					73
440 SS	H	7.0×10^{19}	>100	RT	185	200	211	240					73
X-13		2.4×10^{20}	80	RT	45	98	68	98	36.5	1.2			77
17-4PH		0.2×10^{20}	50	RT	144.7	179.4	148.5	181	16	13	65	53	2
17-4PH		1.3×10^{20}	50	RT	144.7	194.5	148.5	197	16	12	65	44	2
17-4PH		5.1×10^{20}	50	RT	144.7	205.5	148.5	206	16	11	65	44	2
17-4PH		11.8×10^{20}	50	RT	144.7	209.5	148.5	210	16	10	65	41	2
17-4PH		28.0×10^{20}	50	RT	144.7	214.3	148.5	215	16	9	65	29	2
17-4PH		20×10^{20}	525	RT	162	148	148.5		10.2	12.6			78
17-4PH		20×10^{20}	525	400	158	113			6.3	9.9			78
17-4PH		20×10^{20}	525	500	110	89.5			8.2	8.0			78
17-4PH		20×10^{20}	525	600	70.5	51.3			18.8	18.1			78
17-4PH		20×10^{20}	525	700	23.1	33.5			33.4	16.2			78
17-4PH		20×10^{20}	525	800	15.9	15.2			44.0	37.8			78
17-4PH		20×10^{20}	525	900	13.8	10.3			16.3	18.3			78
1541		0.1×10^{20}	<120	RT	47.6	59.8	64.5	62.9	5.8	2.1			79
1541		3.5×10^{20}	700	600	23.0	12.0	27.0	16.0	36	37			80
1541		3.5×10^{20}	700	700	11.6	9.0	12.0	13.0	61	52			80

(a) TD - transverse direction.
A - annealed.
M - martensitic.
T - tempered martensite.
H - hardened martensite.

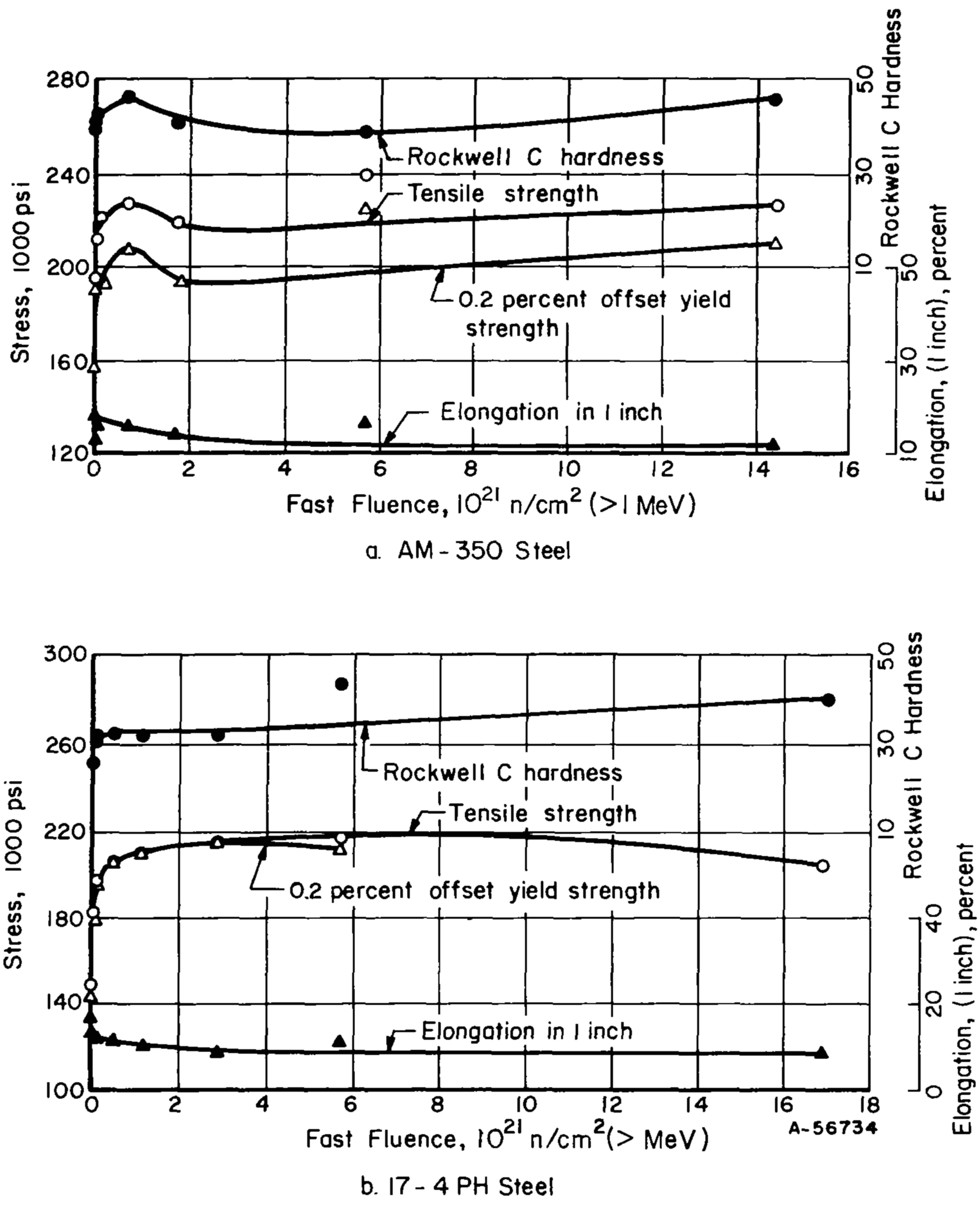

FIGURE 32. EFFECTS OF IRRADIATION ON THE ROOM-TEMPERATURE ROCKWELL HARDNESS, TENSILE STRENGTH, YIELD STRENGTH, AND ELONGATION OF TWO STAINLESS STEELS(2)

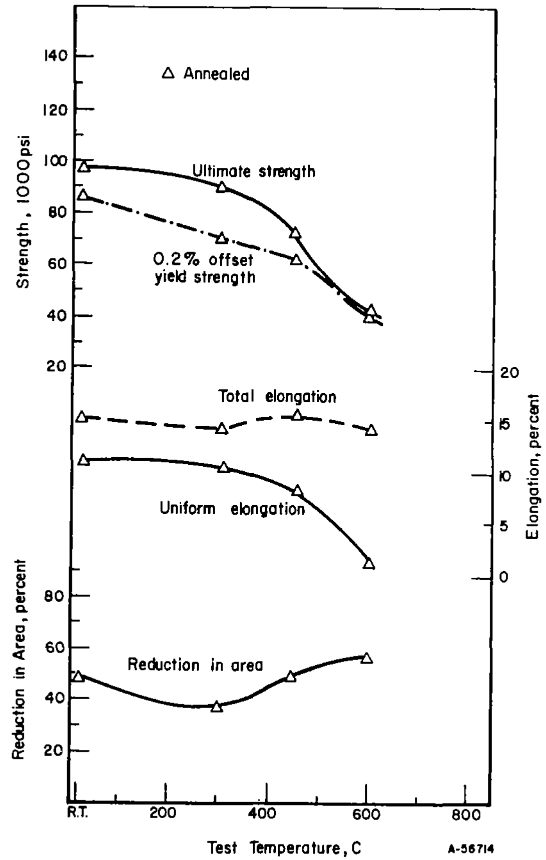

FIGURE 33. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF UNIRRADIATED AISI 406 STAINLESS STEEL(84)

Transverse specimens only.

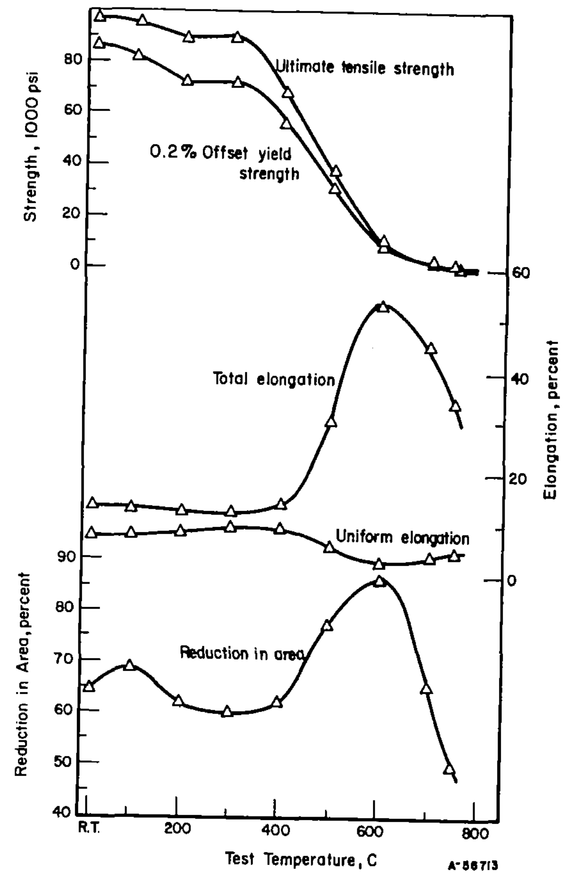

FIGURE 34. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF IRRADIATED AISI 406 STAINLESS STEEL(84)

All transverse specimens irradiated at ~280 C in the annealed condition to a fast fluence of 1.3 x 10^{20} n/cm^2.

respectively. It can be seen that a fast fluence of 1.3×10^{20} n/cm^2 does not cause any significant changes in the strength properties. The only unusual phenomenon is the rather large increase of total elongation with increasing temperature for the irradiated specimens. This increase in total elongation, for the irradiated specimens, reaches a maximum at 600 C and then starts to decrease. The elevated-temperature tensile properties of 17-4PH are given in Table 14, while those of Fe-15Cr-4Al-1Y (Alloy 1541) are given in Table 15. These results indicate that irradiation does not significantly alter the tensile properties at elevated temperature.

TABLE 15. STRAIN-RATE SENSITIVITY OF IRRADIATED[a] AND UNIRRADIATED[b] Fe-15Cr-4Al-1Y ALLOY[85]

Deformation Temperature, C	Strain Rate, %/min	Strength, 1000 psi		Ductility, percent	
		0.2% Offset Yield	Engineering Ultimate	True Uniform Strain	Total Elongation
500	20.0	32.2 (28.0)	45.4 (37.9)	8.8 (8.1)	22.5 (17.9)
	2.0	34.6 (30.7)	41.6 (34.5)	10.8 (4.2)	14.9 (19.7)
	0.2	30.8 (20.1)	35.8 (24.8)	6.0 (7.1)	23.2 (26.3)
600	20.0	29.7 (24.4)	34.8 (28.1)	6.3 (4.1)	32.7 (25.4)
	2.0	27.2 (19.3)	29.9 (21.4)	4.5 (4.3)	30.3 (31.1)
	0.2	21.8 (15.3)	22.8 (17.1)	4.0 (3.5)	30.4 (27.2)
700	20.0	18.5 (13.8)	20.3 (14.6)	6.5 (2.6)	39.0 (62.0)
	2.0	14.2 (9.8)	15.6 (9.9)	4.4 (3.2)	51.9 (51.0)
	0.2	11.4 (8.2)	12.3 (8.5)	2.6 (2.0)	37.8 (44.3)
800	20.0	12.0 (7.6)	12.4 (7.9)	4.4 (2.3)	50.9 (75.7)
	2.0	7.9 (4.3)	8.1 (4.5)	3.5 (3.3)	64.0 (74.1)
	0.2	3.9 (3.5)	4.7 (3.7)	1.9 (1.8)	78.1 (76.4)
871	20.0	6.2 (6.8)	6.5 (6.9)	2.1 (1.0)	58.3 (82.5)
	2.0	4.3 (4.0)	4.8 (4.2)	2.6 (3.3)	76.4 (79.3)
	0.2	2.8 (2.2)	3.2 (2.5)	3.5 (4.1)	93.1 (92.7)

(a) First entries are data from samples irradiated at 50 C to a thermal fluence of 1×10^{20} n/cm^2 and a fast fluence of 1.5×10^{19} n/cm^2 (>1 MeV).

(b) Values given in parentheses.

Brittle-to-Ductile Transition Temperature

Impact tests with irradiated Charpy V-notch specimens on 17-4PH indicate a NDT shift of 110 F after a fast fluence of 7×10^{18} n/cm^2.[51] This NDT shift is in the band of values shown in Figure 23. A similar NDT shift, shown in Figure 35, occurs in General Electric Alloy 1541 after a fast fluence of 1×10^{19} n/cm^2.[82]

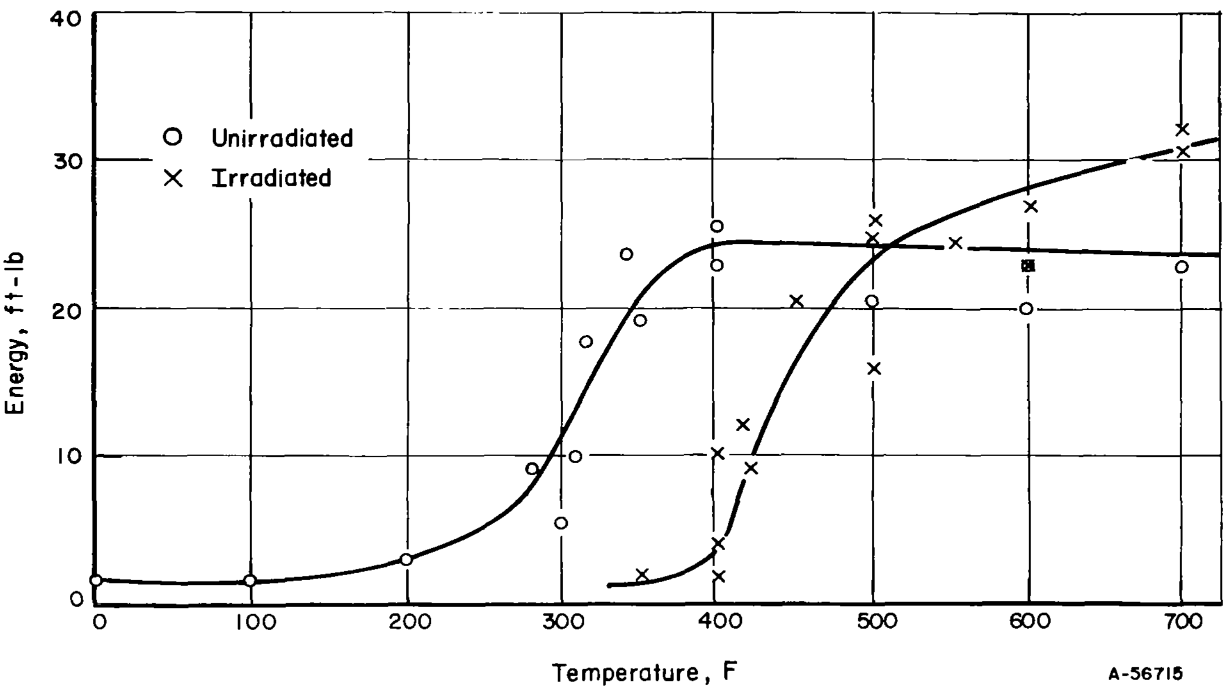

FIGURE 35. NOTCH-DUCTILITY PROPERTIES OF Fe-15Cr-4Al-1Y ALLOY ROD IRRADIATED AT < 250 F TO 1×10^{19} N/CM^2 (> 1 MEV)[82]

AUSTENITIC STAINLESS STEELS

The austenitic stainless steels have found wide application in the nuclear industry as cladding materials for the fuel elements and as structural materials for pressure tubes and coolant pipes. The stainless steels have attractive properties for nuclear applications since they are corrosion resistant and have adequate strength for both room-temperature and elevated-temperature applications. The main application of stainless steels has been for pressurized-water reactors (PWR) and boiling-water reactors (BWR) which operate at maximum temperatures of about 350 C. Presently, stainless steels are being considered as prime candidates for the Liquid Metal Fast Breeder Reactor (LMFBR) applications which require temperatures up to 700 C.

Most of the experimental work has been done on the 300 series of austenitic steels with data on Types 304, 316, 347, and 348 being the most common. It has been determined that significantly different changes in properties in these steels are caused by a mixed thermal and fast flux as opposed to a purely fast flux. Since sufficient data for both types of irradiations are available these will be treated differently.

Mixed Thermal and Fast Fluence

Tensile Properties

Fast Fluence. The effect of increasing fast fluence on the room temperature and 315 C tensile properties are illustrated in Figure 36.[86] It can be seen that most of the irradiation induced changes in tensile properties occur after a fast fluence of 1×10^{22} n/cm^2 with very little consequent changes. Actually it appears that some reduction in the irradiation induced changes occurs between fluences of 2×10^{22} and 3×10^{22} n/cm^2. This is represented by a slight drop in the irradiation induced yield strength and a slight improvement in ductility.

Irradiation Temperature. Figure 37 represents the effect of different irradiation temperatures on the room temperature yield strength and elongation[87] of Type 304 stainless steel. The maximum yield strength increase results from the production of dislocation loops during irradiation at 180 C (0.27 T_m) while irradiation at 300 C (0.35 T_m) causes the largest reduction in elongation because of vacancy clusters. The irradiation-induced property changes are attributed to the presence of defect clusters as obserbed by transmission electron microscopy. These clusters were found to be 100 to 200 Å in size when the maximum irradiation effects on yield strength were detected.

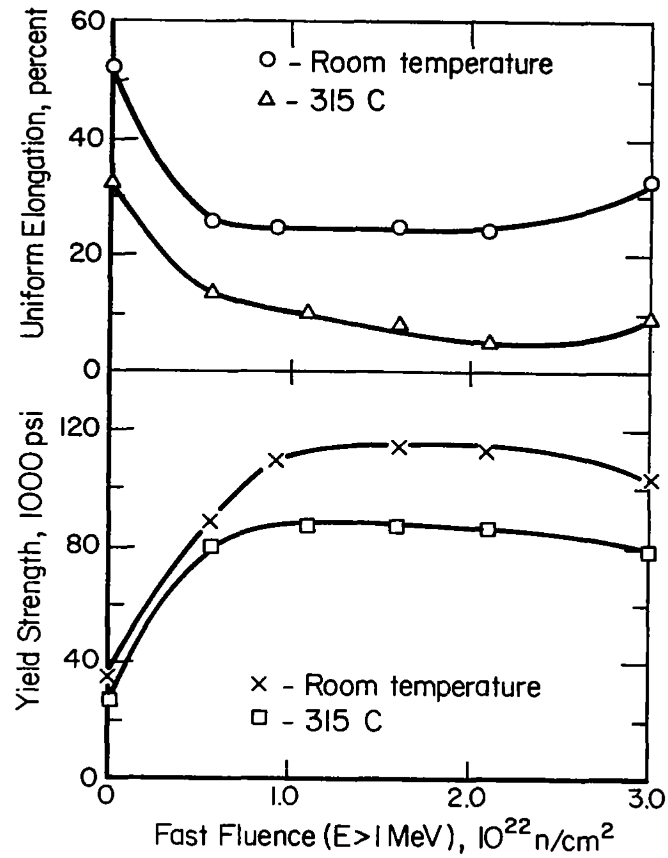

FIGURE 36. EFFECT OF FAST FLUENCE ON ROOM-TEMPERATURE TENSILE PROPERTIES OF TYPE 347 STAINLESS STEEL IRRADIATED AT 50 C[86]

The effect of irradiation at 290 C, to different fast fluence levels, on the total elongation of Type 348 stainless steel is illustrated in Figure 38[84]. At low testing temperatures there is a significant decrease in elongation, which becomes more severe with increasing fluence. As the testing temperature is increased above 400 C the displacement types of radiation damage is annealed out and the ductility is restored. However, at above about 600 C the elongation is again reduced. This ductility decrease in austenitic stainless steels is attributed to helium bubble formation at the grain boundaries.[88] These helium bubbles promote intergranular fracture and thus decrease the ductility. The main source of this helium is believed to be due to (n,α) reactions in the boron impurity; but fast neutrons also cause (n,α) reactions in iron, chromium, and nickel.

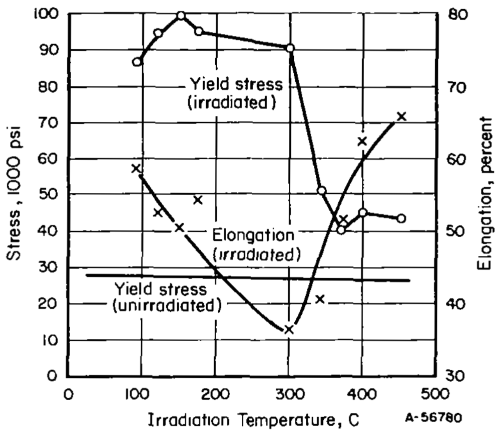

FIGURE 37. ROOM-TEMPERATURE TENSILE PROPERTIES OF IRRADIATED TYPE 304 STAINLESS STEEL AS A FUNCTION OF IRRADIATION TEMPERATURE(87)

Specimens were irradiated to a fast fluence of 7×10^{20} n/cm^2.

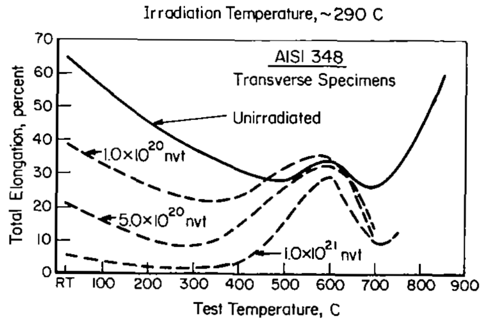

FIGURE 38. THE EFFECT OF TEST TEMPERATURE ON THE DUCTILITY OF IRRADIATED AISI TYPES 348 AND 304 STAINLESS STEEL IN THE ANNEALED CONDITION(84)

TABLE 16. COMPARISON OF STRENGTH AND DUCTILITY FOR IRRADIATED AND UNIRRADIATED TYPE 304 STAINLESS STEEL WELDS AND WROUGHT METAL[a](93)

		Strength, 1000 psi				Ductility, percent			
	Deformation	0.2 Percent Offset Yield		Ultimate Tensile		True Uniform Strain		Total Elongation	
Material	Temp., C	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated
Weld metal	500	48.2	35.8	60.6	54.6	16.0	9.2	21.3	25.0
	600	37.2	35.7	48.6	43.9	16.8	15.0	23.8	21.1
	700	32.6	30.6	36.3	32.1	6.4	4.3	13.4	27.1
	800	22.6	19.7	23.2	19.8	2.0	2.4	12.8	31.4
	900	14.6	14.9	14.8	14.9	1.3	1.1	4.1	30.2
Wrought metal	500	24.4	21.3	60.5	56.6	29.6	31.2	40.0	41.1
	600	18.3	19.4	51.2	45.5	29.0	30.2	38.0	39.4
	700	17.5	17.9	34.4	30.7	17.7	21.0	23.2	37.8
	800	14.2	13.3	19.5	18.7	6.9	12.6	9.5	21.5
	900	10.4	10.8	11.0	12.5	2.4	--	4.9	20.2
Joints	500	36.8	30.7	60.4	58.2	14.0	19.1	19.2	25.5
	600	30.5	32.8	52.5	50.0	17.4	17.0	23.0	21.6
	700	27.8	--	37.0	--	8.6	--	12.1	16.0
	800	20.2	23.9	22.5	24.8	3.1	7.8	5.7	15.8
	900	13.3	13.4	13.4	13.7	1.0	7.5	2.8	17.3

(a) Irradiated to 2.3 x 10^{18} n/cm^2 fast fluence (> 1 MeV) and 6.7 x 10^{19} n/cm^2 thermal fluence at a temperature of 52 C.

The reduction of ductility at elevated temperatures has been found to be independent of irradiation temperature if the irradiation temperature is below 700 C. However, at irradiation temperatures above 700 C somewhat more embrittlement takes place.[89] The degree of embrittlement is also greater with lower strain rates[90] and increasing grain size of the material[91].

Additions of titanium to Type 304 stainless steel increase its resistance to irradiation-induced embrittlement at elevated temperature.[92] This is illustrated in Figure 39. The maximum improvement is obtained with a titanium addition of about 0.20 wt %, which is somewhat less than the 0.40 wt % titanium content of the titanium-stabilized Type 321 stainless steel.

Weld Material. Results of tensile tests on irradiated and unirradiated weld metal, base metal, and joints of Type 304 stainless steel at temperatures of 500 to 900 C are given in Table 16.[93] These test results indicate that the weld metal and base metal are embrittled to about the same degree by irradiation. Table 17 shows that, up to 600 C, the ductility of the unirradiated and irradiated welds was about the same, but that the ductility of the irradiated welds then decreased significantly with increasing temperature. The strength of the welded joints is not significantly affected by irradiation at any of the testing temperatures.

TABLE 17. RATIO OF IRRADIATED TO UNIRRADIATED DUCTILITY OF WELDS AND WROUGHT ALLOYS[93]

	Ratio of Irradiated to Unirradiated Toughness at Indicated Temperature				
Base Material	500 C	600 C	700 C	800 C	900 C
Type 304 stainless steel, wrought metal	0.95	1.0	0.60	0.45	0.31
Type 308 stainless steel, weld metal	0.90	1.05	0.50	0.40	0.15

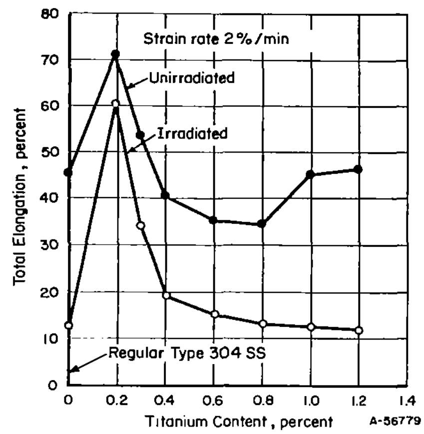

FIGURE 39. DUCTILITY AT 842 C OF IRRADIATED AUSTENITIC STAINLESS STEEL AS A FUNCTION OF TITANIUM CONTENT[92]

Irradiated to a thermal fluence of 1 x 10^{20} n/cm^2 and a fast fluence at 1.5 x 10^{19} n/cm^2.

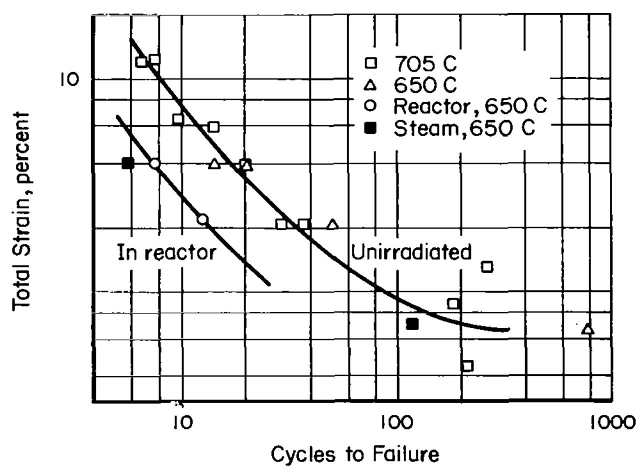

FIGURE 40. STRAIN FATIGUE LIFE OF TYPE 304 STAINLESS STEEL TUBING IN REACTOR AND OUT OF REACTOR[95]

Brazes. Table 18 shows the effect of irradiation on the shear strength of nickel-13 wt % chromium-10 wt % phosphorus braze joints of Type 304 stainless steel.[93] These results show that irradiation increases the shear strength of brazes up to about 500 C, with minor irradiation-induced decreases in shear strength occurring at temperatures above 500 C.

TABLE 18. LOAD-CARRYING CAPACITY OF IRRADIATED AND UNIRRADIATED (Ni- Cr- P) BRAZES WITH TYPE 304 STAINLESS STEEL[93]

Deformation Temp., C	Maximum Load[a], lb, During Shear Test		
	Unirradiated	Irradiated	Ratio, Irradiated/Unirradiated
RT	468	631	1.34
100	420	592	1.41
200	411	527	1.41
300	382	473	1.24
400	381	433	1.14
500	359	398	1.10
600	367	331	0.91
700	311	250	0.80
800	222	210	0.94

(a) Average of three tests.

Creep Properties

In-pile creep tests at 550 and 650 C have indicated that the creep rate of austenitic stainless steel is not affected by irradiation.[94] However, since the total elongation of stainless steel is reduced by irradiation the time to rupture is lowered by irradiation.

Fatigue Properties

The cyclic-strain fatigue behavior of AISI Type 304 steel has been determined in-pile at 650 C.[95] By applying gas pressure, the thin-walled tubular specimens were alternately expanded and contracted between rigid concentric mandrels. The results of the tests (Figure 40) indicate that irradiation tends to reduce fatigue life. Results of room temperature test on Type 347 stainless steel, which had been irradiated to 5.5×10^{21} and 1.1×10^{22} n/cm^2, are shown in Figure 41. The fatigue life was increased at strains below 1 percent and decreased at strains above 1 percent.(96)

Impact Properties

The impact properties of Type 347 stainless steel after irradiation to two levels of fluence are shown in Figure 42.(96) It is interesting to note that the irradiation temperature appears to be more significant in determining the room-temperature impact strength of the irradiated material than is the fast fluence.

Hardness

The effect of irradiation on the hardness of cold-worked and annealed Type 304 and 348 stainless steel is shown in Figure 43.(84)

Predominantly Fast Fluence

Tensile Properties

The effect of irradiation in EBR-II at 538 C on the tensile properties of Type 304 stainless steel is illustrated in Figures 44 and 45.(97) The irradiation induced strength increases and ductility decreases become more pronounced with increasing fast fluence as illustrated in Figures 46 and 47.(98)

Creep Properties

Effect of fast neutron irradiation on the uniaxial creep rate of Type 316 stainless steel is illustrated in Figure 48. The biaxial rupture life of Type 316 stainless steel is given in Figure 49. Irradiation appears to cause a slight decrease in the creep rate but the rupture times are somewhat reduced by irradiation. This reduction in rupture life is attributed to irradiation induced decrease in elongation at rupture.(99)

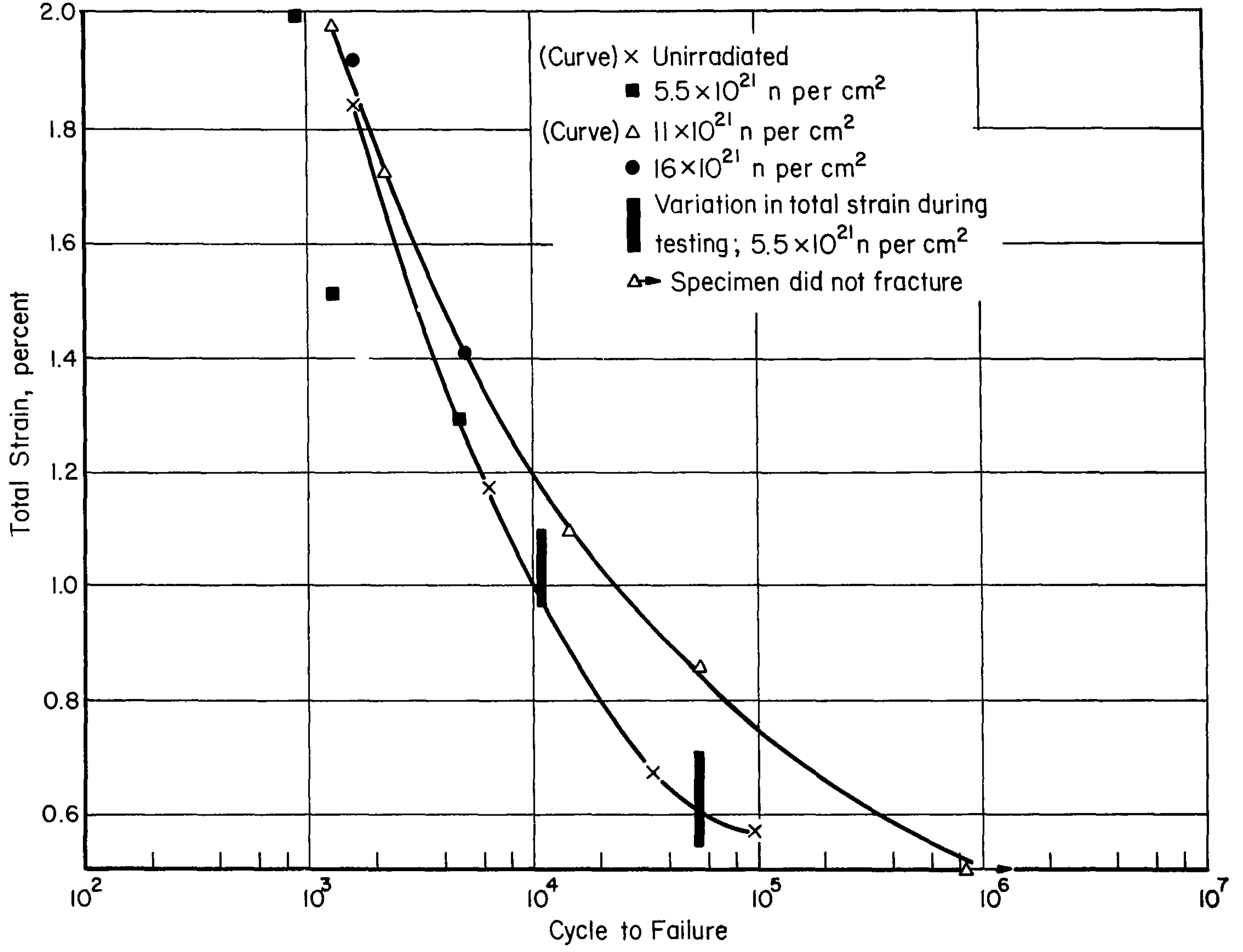

FIGURE 41. THE EFFECT OF TOTAL STRAIN PER CYCLE ON THE FATIGUE LIFE FOR IRRADIATED AND UNIRRADIATED TYPE 347 STAINLESS STEEL TESTED AT 20 C[96]

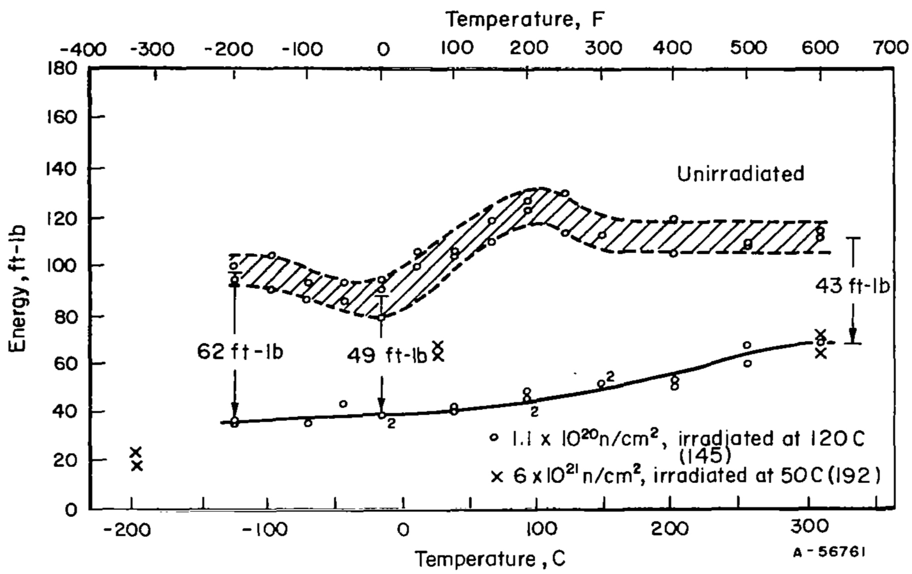

FIGURE 42. CHARPY-V NOTCH DUCTILITY CHARACTERISTICS OF TYPE 347 STAINLESS STEEL BEFORE AND AFTER IRRADIATION

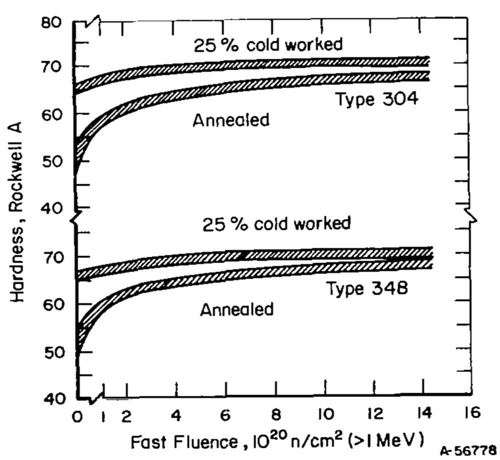

FIGURE 43. EFFECTS OFIRRADIATION ON THE HARDNESS OF TYPES 304 AND 348 STAINLESS STEEL[84]

Irradiation temperature ~290 C.

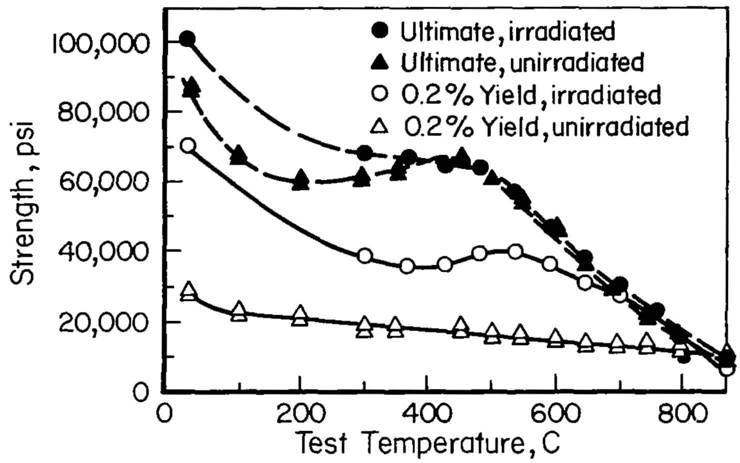

FIGURE 44. THE 0.2 PERCENT YIELD AND ULTIMATE STRENGTH FOR AISI TYPE 304 STAINLESS STEEL IRRADIATED TO 1.4 x 10^{22} N/CM^2 (E > 0.18 MEV) AT 538 C IN THE EBR-II(97)

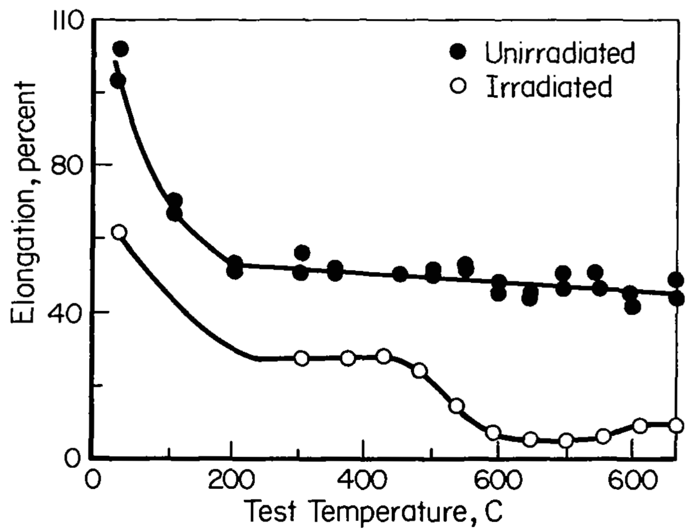

FIGURE 45. TOTAL ELONGATION IN 1.25-IN. GAGE LENGTH FOR AISI TYPE 304 IRRADIATED TO 1.4 x 10^{22} N/CM^2 (E > 0.18 MEV) IN THE EBR-11 AT 538 C(97)

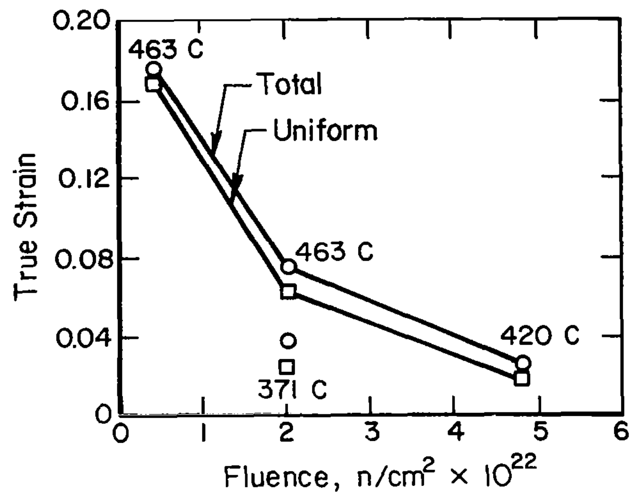

FIGURE 46. THE EFFECT OF NEUTRON EXPOSURE AND IRRADIATION TEMPERATURE ON THE TRUE UNIFORM STRAIN OF TYPE 304 STAINLESS STEEL DURING TESTING AT 450 C[98]

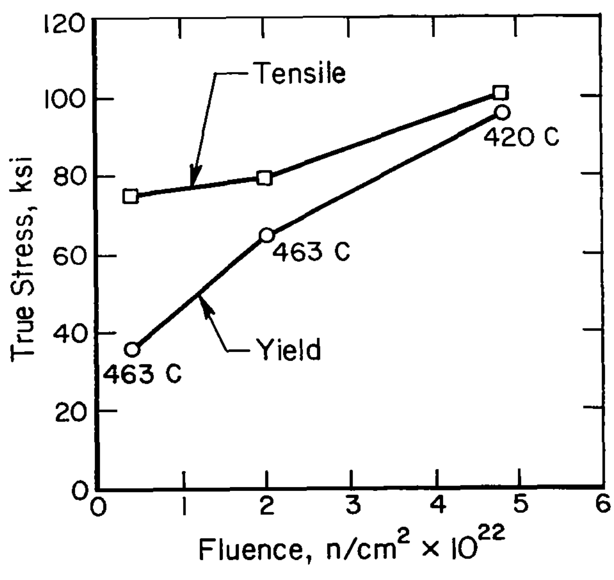

FIGURE 47. THE TRUE YIELD STRENGTH AND FRACTURE STRENGTH (IN THOUSANDS OF PSI) OF TYPE 304 STAINLESS STEEL AT A TEST TEMPERATURE OF 450 C VERSUS NEUTRON EXPOSURE AT IRRADIATION TEMPERATURE ~450 C[98]

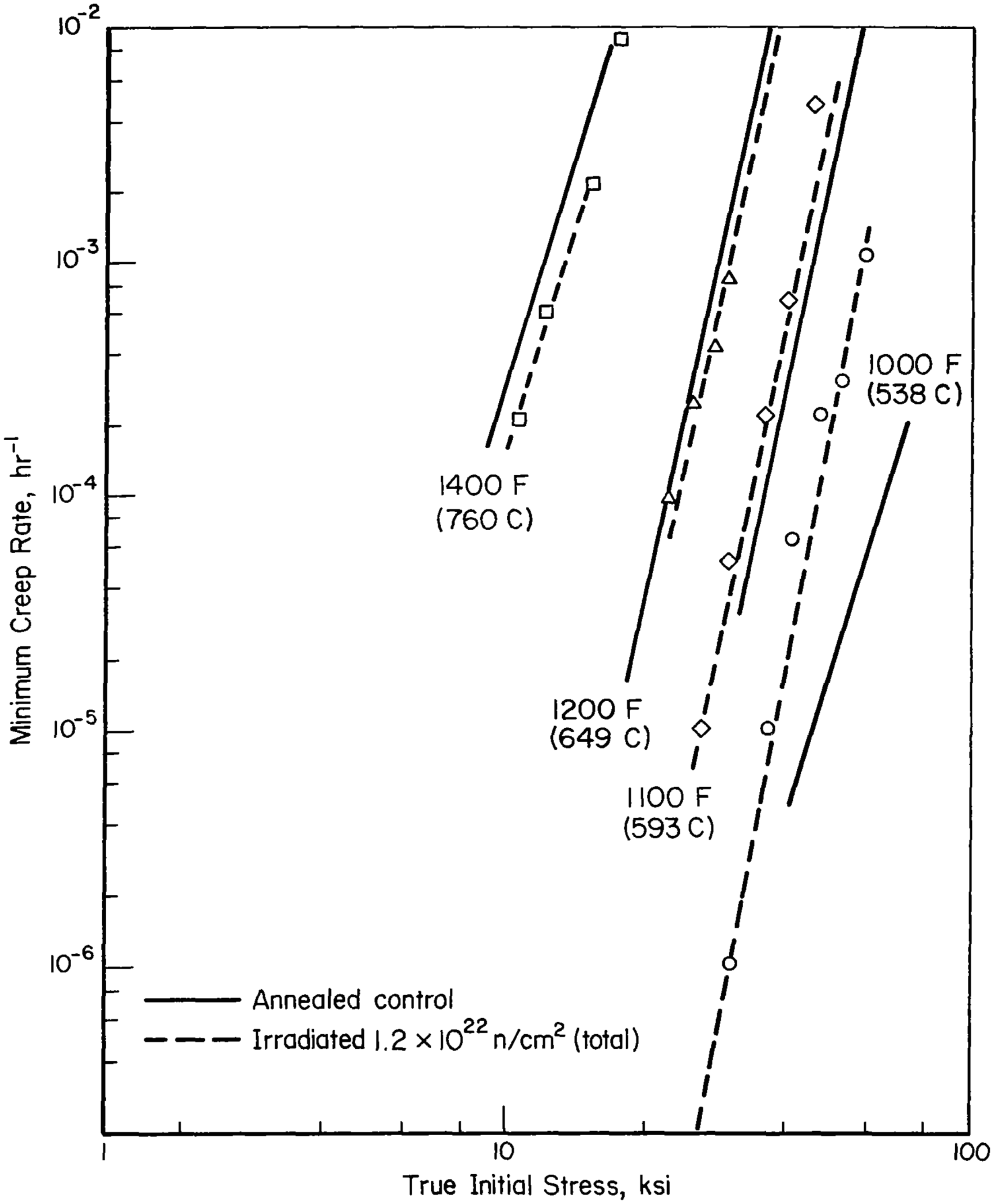

FIGURE 48. EFFECT OF EBR-II IRRADIATION ON THE MINIMUM CREEP RATE OF AISI TYPE 316 STAINLESS STEEL DETERMINED IN UNIAXIAL TESTS(99)

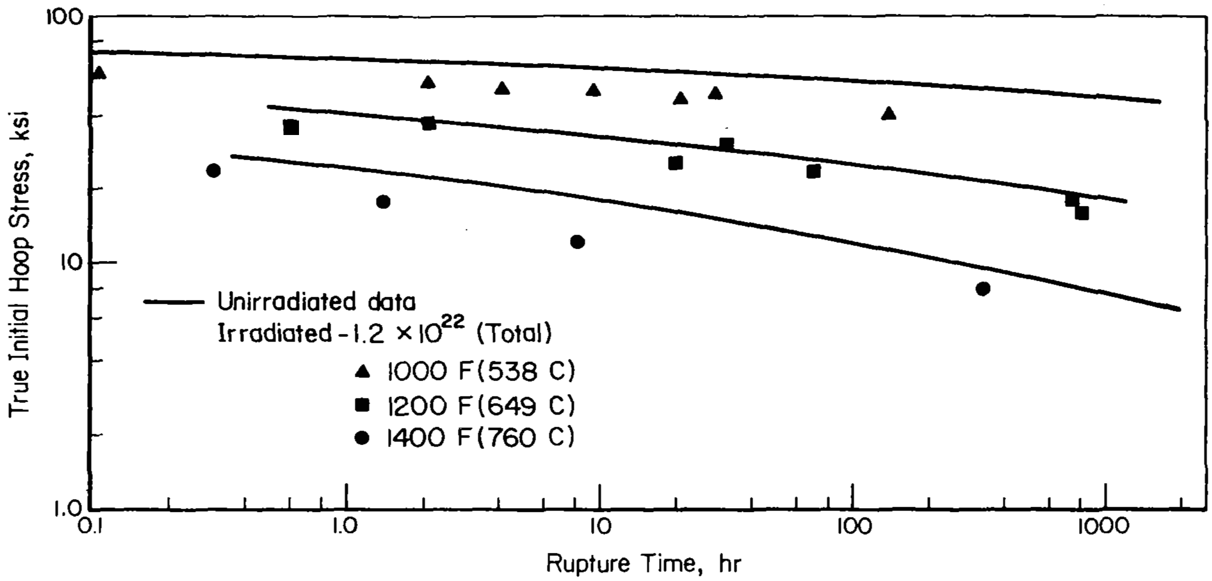

FIGURE 49. THE EFFECT OF EBR-II IRRADIATION ON THE RUPTURE LIFE OF AISI TYPE 316 STAINLESS STEEL DETERMINED IN BIAXIAL TESTS(99)

Swelling

A high instantaneous fluence has been shown to cause swelling in stainless steels in the 400 to 600 C temperature range. This swelling is attributed to condensation of vacancies into voids of 100 to 300 Å. Results indicate that cold worked stainless steel undergoes less swelling than annealed stainless steel at equivalent fast fluences.(100)

NICKEL-BASE ALLOYS

Nickel-base alloys have found only limited use in the pressurized and boiling-water reactors because of the rather high capture cross section for thermal neutrons. However, nickel-base alloys have been used as cladding materials in nuclear superheat applications where their higher strength at elevated temperatures makes them more attractive than stainless steel.

Presently, nickel-base alloys are being considered as candidate component materials in liquid-sodium-cooled fast-breeder reactors. The capture cross section of nickel for fast neutrons is not considered high relative to other candidate materials.

The nickel-base alloys can be divided into two categories: the solid-solution-hardened alloys and precipitation-hardened alloys. The solid-solution-hardened alloys derive their strength from substitutional solid-solution hardening. The precipitation-hardened alloys derive their strength from titanium and aluminum additions which result in small particles of Ni_3Ti and Ni_3Al. A considerable number of the solid-solution-hardened nickel-base alloys have been irradiated and tested. These alloys behave very similarly to the austenitic stainless steels. The precipitation hardened nickel alloys have been found to undergo extreme embrittlement as a result of irradiation.[101]

Only the most thoroughly investigated nickel-base alloys will be discussed to eliminate redundancy. These alloys are:

(1) Nickel

(2) Incoloy 800

(3) Hastelloy X and Hastelloy X-280

(4) Hastelloy N (INOR 8)

(5) Inconel 600

(6) Inconel 625

(7) René 41.

Nickel

Tensile Properties

The effects of irradiation on the room-temperature tensile properties of nickel are shown in Table 19.[102, 103] Irradiation increases both the yield and ultimate strength and causes significant reduction in ductility. The irradiation-induced increase in the yield strength for nickel has been shown to depend on the cube root of fast fluence in the range of 1×10^{17} to 1.1×10^{20} n/cm^2; this dependence is illustrated in Figure 50.[104]

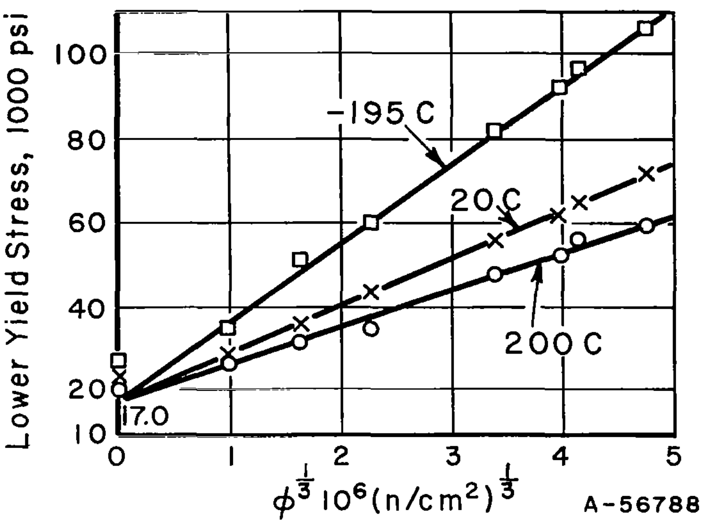

FIGURE 50. DEPENDENCE OF IRRADIATION-INDUCED YIELD-STRENGTH INCREASE ON FAST FLUENCE(99)

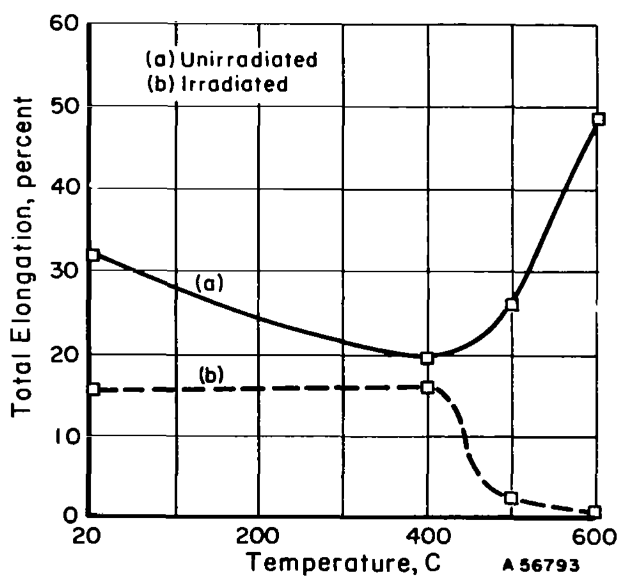

FIGURE 51. EFFECT OF IRRADIATION (1.7 x 10^{20} N/CM^2 ON THE DUCTILITY AND STRENGTH OF 99.95 PERCENT PURE NICKEL(105)

TABLE 19. EFFECT OF IRRADIATION ON THE ROOM-TEMPERATURE TENSILE PROPERTIES OF NICKEL[a]

Material	Fluence, 10^{20} n/cm^2 (>1 MeV)	0.2% Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Total Elongation, percent		Reference
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Nickel A	2.4[a]	17.1	77.7	62.5	95.3	33	17	102
Nickel A	2.4[b]	16.6	82.8	56	85.7	45	23	102
Nickel A	4.0[b]	17.1	91.2	62.5	106	33	17	102
Nickel A	4.0[b]	16.6	61.2	56	89.0	34	13	102
Nickel 330[c]	7.8	52.5	108.6	66.5	108.6			103
Nickel 330[d]	7.8	80.5	146.0	82.3	146.0	8	3	103
Nickel 330[c]	7.8	53.2	149.0	84.1	149.0			103
Nickel 330	9.0	45.8	95.6	63.4	95.6	36	10	103
Nickel 330	9.0	45.8	98.6	63.4	98.6	36	11	103
Nickel 330[e]	9.0	37.3	98.0	61.8	98.0	30	7	103
Nickel 330[c]	9.0	40.0	104	63.0	104.0			103
Nickel 330	9.0	43.0	118	82.0	118.0			103
Nickel 330[c, e]	11.3	49.6	118	84.7	129.8			103
Nickel 330[c]	11.3	80.0	163	97.4	163			103
Nickel 330	11.3	67.2	145	87.5	145	34	5	103
Nickel 330	11.3	45.8	101.3	63.4	101.3	36	8	103

(a) Material irradiated at less than 100 C.
(b) Neutron energy >0.5 MeV.
(c) Notched specimen.
(d) 75 percent cold worked.
(e) Weld material.

Pure nickel (99.95 percent) has been irradiated at 150 to 200 C to a fast fluence of 1.7 x 10^{20} n/cm^2. Figure 51 shows that a drastic ductility reduction takes place between 400 and 500 C and, at 600 C, the ductility is almost zero.[105] It has been found that this drastic reduction in ductility results from the change of fracture mode at these temperatures from transgranular to intergranular as a result of irradiation.[106] Figure 51 also shows that at a test temperature of 400 C, most of the displacement-type damage is annealed out and the yield and ultimate strengths of unirradiated and irradiated nickel are equivalent above that temperature.

Hardness

The effect of irradiation on the hot hardness of 99.95 percent pure nickel is shown in Figure 52.[105] The irradiation damage (as measured by hardness increase) is not annealed out until about 700 C.

Density

The density of nickel has been shown to undergo large decreases as a result of irradiation.[107] These density decreases become more pronounced with increasing fast fluence and are more severe for higher purity nickel (Figure 53).

Incoloy 800

Tensile Properties

A large number of tensile tests have been performed on irradiated Incoloy 800 specimens at various test temperatures. These specimens were given various preirradiation heat treatments and were irradiated at different temperatures and to various fast-fluence levels. The room-temperature mechanical properties of irradiated Incoloy 800 are given in Table 20. It becomes apparent that the yield strength increases and the ductility decreases with increasing fast fluence if the irradiation takes place at a low temperature. However, the irradiation temperature plays a major role in causing irradiation-induced effects, since significant increases in yield strength take place at lower irradiation temperatures while no changes occur at an irradiation temperature of about 400 C. At an irradiation temperature of 740 C, a considerable decrease in yield strength occurs and the reduction in ductility is somewhat less than that when the material is irradiated at lower temperatures. This dependence of strength and ductility on the irradiation temperature is probably due to overaging of the complex alloy at the higher irradiation temperatures. The tensile properties of irradiated Incoloy 800 at intermediate temperatures are given in Table 21. These postirradiation tensile results exhibit trends similar to those for the room-temperature properties in that the elevated temperature irradiations result in decreased strength and increased ductility.

The postirradiation mechanical properties of Incoloy 800 at elevated temperatures are given in Table 22. Here gain the elevated-temperature irradiations cause a loss in strength as compared with the low temperature irradiations at equivalent fast fluences. However, the elevated-temperature ductility is drastically reduced at all irradiation temperatures. The cause of this decrease in elevated-temperature ductility is attributed to the formation of helium by (n, α) reactions from the boron impurity which is present.

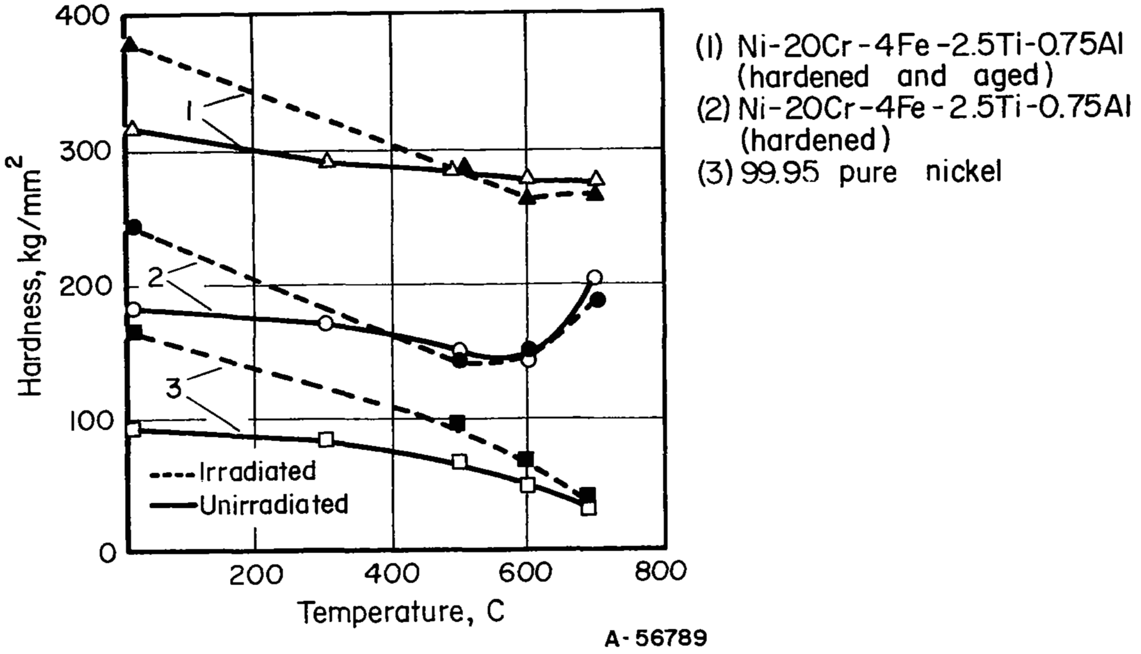

FIGURE 52. EFFECT OF IRRADIATION (1.7 x 10^{20} N/CM^2) ON THE HOT HARDNESS OF NICKEL AND NICKEL ALLOYS[105]

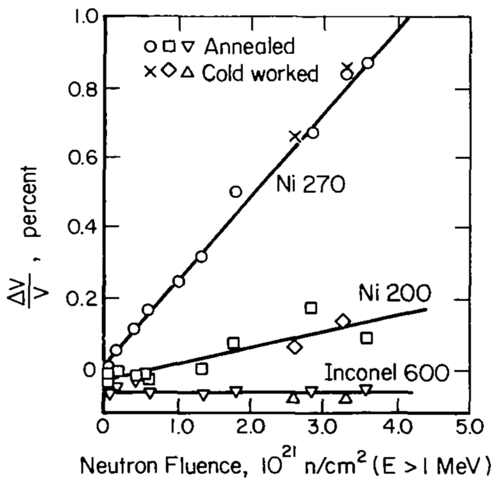

FIGURE 53. RADIATION-INDUCED SWELLING IN NICKEL-BASE ALLOYS[107]

TABLE 20. EFFECT OF IRRADIATION ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF INCOLOY 800

Fast Fluence, n/cm^2	Irradiation Temp., C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reference
						Uniform		Total		
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
2×10^{19}	40	25.6	32.6	71.7	80.5			47	47	108
1×10^{20}	50		64				39			109
1×10^{20}	740		24				37			109
4×10^{20}	400	41.3	46.8			37.7	35.7			74
0.6×10^{21}	180	44.4	86.1	86.5	104.6	41.7	25.9	50.4	34.4	110
1.2×10^{21}	180	44.4	94.9	86.5	109.1	41.7	25.0	50.4	33.3	110
2.4×10^{21}	180	44.4	113.1	86.5	120.6	41.7	20.6	50.4	27.5	110
2.5×10^{21}	180	44.4	114.1	86.5	121.0	4.17	21.5	50.4	27.6	110

TABLE 21. EFFECT OF IRRADIATION ON THE MECHANICAL PROPERTIES OF INCOLOY 800 AT INTERMEDIATE TEMPERATURES

Fast Fluence, n/cm^2	Irradiation Temp., C	Test Temp., C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Uniform Elongation, percent		Total Elongation, percent		Reference
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
9.2×10^{19}(a)	740	300	26.7	32.3	69.6	71.6	30.8	31.5	33.8	37.3	109
9.2×10^{19}(b)	740	300	21.6	22.0	62.6	62.8	37.8	39.8	40.7	43.6	109
9.2×10^{19}(c)	740	300		27.4		73.9		31.5		36.2	109
2.2×10^{20}(a)	740	300	34.3	45.3	77.6	89.8		34.3		37.4	109
2.2×10^{20}(b)	740	300	25.7	22.9	67.4	61.6	30.8	34.1	34.4	37.9	109
2.2×10^{20}(c)	740	300	29.7	23.8	76.5	69.6	24.8	33.0	31.5	37.0	109
2.4×10^{20}(a)	740	300	35.3	30.6	76.1	69.1	24.9	30.4	30.6	35.9	109
2.4×10^{20}(b)	740	300	24.4	23.3	69.8	61.5	29.7	37.3	33.8	40.7	109
2.4×10^{20}(c)	740	300	28.4	24.3	75.7	70.4	24.3	32.6	27.2	37.2	109
6×10^{20}	180	315		58.7		84.1		25.8		31.1	110
1.2×10^{21}	180	315		72.9		90.5		23.4		28.6	110

(a) Mill annealed.
(b) 20 percent cold worked.
(c) Solution annealed.

TABLE 22. EFFECT OF IRRADIATION ON THE ELEVATED-TEMPERATURE MECHANICAL PROPERTIES OF INCOLOY 800

Fast Fluence, n/cm^2	Irradiation Temp., C	Test Temp., C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reference
							Uniform		Total		
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
4.0×10^{20}	400	593	28.2	27.7					34.5	23	74
1.5×10^{21}	180	593	26.3	34.5	62.8	50.2	37.3	6.5	42.5	7.2	110
1.9×10^{21}	180	593	26.3	29.7	62.8	48.4	37.3	7.8	42.5	8.3	110
2.2×10^{21}	180	593	26.3	31.4	62.8	43.7	37.3	9.0	42.5	9.8	110
2.0×10^{19}	40	650	17.8	16.6	43.5	35.9	40	28.5	45.0	29.0	108
9.9×10^{19}(a)	740	650	22.6	21.4	48.1	36.4	24.7	8.1	42.8	8.7	109
9.9×10^{19}(b)	740	650	15.7	11.6	46.6	27.5	23.8	11.9	32.1	12.8	109
9.9×10^{19}(c)	740	650	25.1	15.6	50.5	33.8	17.4	10.8	48.8	11.3	109
2.2×10^{20}(a)	740	650	32.6	25.6	60.3	37.5	15.8	3.7	37.3	5.5	109
2.2×10^{20}(b)	740	650	22.9	12.6	53.3	28.8	15.8	9.6	32.5	10.3	109
2.2×10^{20}(c)	740	650	25.7	18.6	54.7	35.2	17.9	7.6	39.1	8.1	109
2.4×10^{20}(a)	740	650	31.5	19.7	54.5	31.4	16.6	5.4	37.2	5.9	109
2.4×10^{20}(b)	740	650	22.3	8.1	51.6	24.5	26.3	12.2	32	13.4	109
4.0×10^{20}(a)	740	650		22.5		32.6		4.3		4.6	109
4.0×10^{20}(b)	740	650		14.0		27.2		8.6		8.7	109
4.0×10^{20}(c)	740	650		18.9		31.1		6.6		6.9	109
2.0×10^{19}	40	750	16.6	15.6	25.6	22.7	17.0	10.0	28.0	12.0	108
2.0×10^{19}	40	820	16.3	13.6	19.8	15.6	9.0	6.0	17.0	8.0	108

(a) Mill annealed.
(b) Solution treated.
(c) 20% cold worked.

Table 23 compares the mechanical properties of Incoloy 800 specimens irradiated in a predominantly fast flux (EBR-II) and in a mixed thermal and fast flux (GETR). The limited tests indicate that at 593 C, there is no difference in the effects of a predominantly fast fluence and a mixed thermal and fast fluence. However, at 704 C, the predominantly fast fluence appears to induce larger losses in ductility than does an equivalent fast fluence which also includes a thermal-fluence component.(111)

TABLE 23. COMPARISON OF THE EFFECTS OF A PREDOMINANTLY FAST FLUENCE AND A MIXED FLUENCE (FAST AND THERMAL) ON THE MECHANICAL PROPERTIES OF INCOLOY 800(111)

Test Temperature, C	Fast Fluence, 10^{20} n/cm^2	Yield Strength, 1000 psi		Tensile Strength, 1000 psi		Elongation, percent			
						Uniform		Total	
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
593	3.3(a)	30.2	25.3	61.6	55.3	28	20.1	32.6	22.3
593	3.5(b)	24	21	55	48	31	19	38	22
704	3.3(a)	24.3	22.3	37.5	30.9	17.3	7.6	63.2	12.8
704	3.5(b)	21	20	37	35	18	12	47	25

(a) Irradiated at 538 C in EBR-II for 200 hours (fast fluence).
(b) Irradiated at 704 C in GETR for 630 hours (mixed fluence).

Burst Properties

Effect of irradiation on the burst properties at elevated temperature are shown in Table 24.(111) These tests indicate that irradiation reduces the hoop strength and causes drastic reductions in ductility.

TABLE 24. EFFECT OF IRRADIATION ON THE BURST PROPERTIES OF INCOLOY 800 TUBING(a) (111)

Test Temperature, C	Hoop Stress, 1000 psi		Change in Diameter, percent	
	Unirradiated	Irradiated	Unirradiated	Irradiated
704	36.5	25	12.9	0.9
816	27.8	18.5	13.2	3.6

(a) Irradiated at 704 C to a fast fluence of 1.1 x 1.6 x 10^{21} n/cm^2.

Fatigue Properties

In-pile cyclic-strain fatigue tests have been performed on Incoloy 800. The testing method utilized rigid concentric mandrels against which the thin-walled tube was alternately expanded and contracted by applying gas pressure. Results of the in-pile tests at 704 C are shown in Figure 54.[95] These results show that irradiation tends to reduce the fatigue life of Incoloy 800.[95]

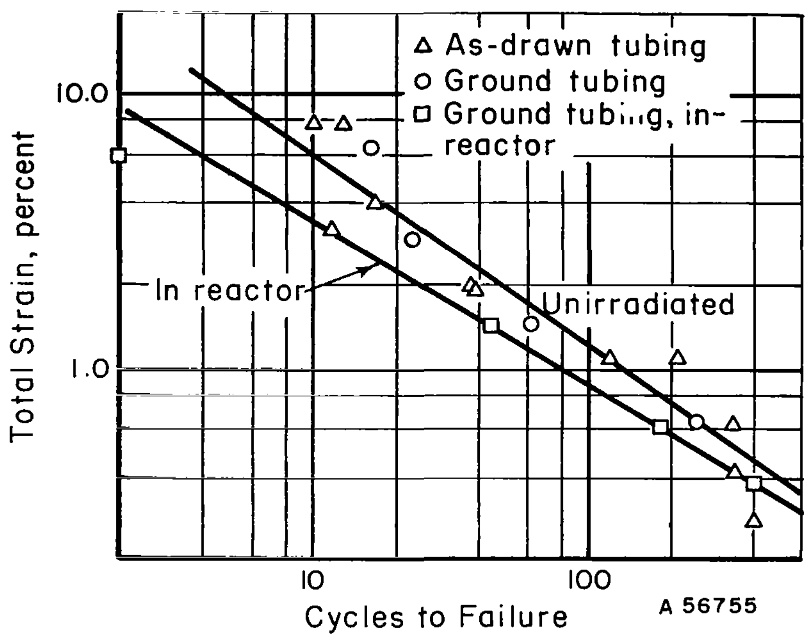

FIGURE 54. STRAIN FATIGUE LIFE OF INCOLOY 800 TUBING AT 704 C[95]

The in-reactor tubing received a fast fluence of 7×10^{19} n/cm^2.

Hastelloy X and Hastelloy X-280

Tensile Properties

The only difference between Hastelloy X and Hastelloy X-280 is that the cobalt content in the latter is limited to a maximum of 0.5 percent in order to obtain better neutron economy and also minimize the residual radioactivity after irradiation. It is believed that the difference in cobalt content does not affect the postirradiation mechanical properties.

Results of tensile tests on irradiated Hastelloy X are given in Table 25. The tensile properties of Hastelloy X are affected by irradiation in the same way as the tensile properties of stainless steels. Irradiation at low temperatures results in increases in yield and ultimate strengths and decreases in ductility when tested at low temperatures. Irradiation at an intermediate temperature (~400 C) does not change the strength or ductility except for a large reduction in ductility at test temperatures above about 600 C. Irradiation at an elevated temperature (740 C) reduces the strength at all temperatures while it increases the ductility at low temperatures and decreases the ductility at elevated temperatures. The changes in strength at lower irradiation temperatures are associated with displacement-type damage, while the loss of strength after elevated-temperature irradiations is due to possible overaging of the alloy. Drastic losses of elevated-temperature ductility are generally attributed to the presence of helium at the grain boundaries. In evaluating the effects of elevated-temperature irradiations it must be remembered that the tensile properties are significantly changed by the aging at the irradiation temperature.

Creep Properties

The effect of irradiation on the 540 and 650 C stress-rupture properties of Hastelloy X are shown in Figure 55.[115] Irradiation seems to decrease the time to rupture at both temperatures. A significant finding is that the 540 C rupture life can be restored by annealing at 1175 C for 1 hour after irradiation, but the 650 C rupture life is not affected by postirradiation annealing. The elongation at rupture, at both 540 and 650 C, was reduced by irradiation, but the ductility loss was more severe for specimens tested at 650 C.

Hastelloy N (INOR-8)

Tensile Properties

The postirradiation tensile properties of Hastelloy N are given in Tables 26[109, 116] and 27[117]. The data show that irradiation at low temperatures results in increased strength and decreased ductility when the

TABLE 25. EFFECT OF IRRADIATION ON THE TENSILE PROPERTIES OF HASTELLOY X

Irradiation Temperature, C	Test Temperature, C	Fast Fluence, n/cm^2	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent Uniform		Elongation, percent Total		Reference
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
50	RT	1×10^{20}	--	95	--	125	--	34.5	--	40.0	112
280	RT	5×10^{19}	--	71	--	117	--	47	--	--	112
280	RT	7×10^{19}	--	74	--	119	--	46	--	--	112
400	RT	4×10^{20}	59.2	74.2	--	--	--	--	41.8	45.3	74
650	RT	3×10^{19}	53	46	118	105	--	--	30	25	113
650	RT	3×10^{19}	53	38	118	109	--	--	30	34	113
740	300	9×10^{19}	47.2	45.8	107.9	109.7	23.5	18.6	26.0	20.9	109
740	300	2.2×10^{20}	47.6	40.7	111	105.9	21.4	21.7	23.2	22.4	109
740	300	2.4×10^{20}	47.6	41.2	111	98.9	21.4	13.7	23.2	14.3	109
400	593	4×10^{20}	42.5	41.5	--	--	--	--	44.9	13.7	74
280	650	1×10^{20}	--	26	--	49	--	4.8	--	4.8	112
280	650	1×10^{20}	--	26	--	49	--	10.7	--	13	112
650	593	2×10^{21}	56	59	111	93	20	4	28	5	114
650	700	3×10^{19}	35	23	71	87	--	--	48	24	113
740	650	9×10^{19}	40.8	32.5	90.1	62.6	16.7	5.6	52	6.2	109
740	650	2.2×10^{20}	40.0	34.7	85.5	67.4	15.0	5.7	43.5	6.1	109
740	650	4×10^{20}	40.5	37	90.7	67.7	19.6	4.7	40.3	7.4	109

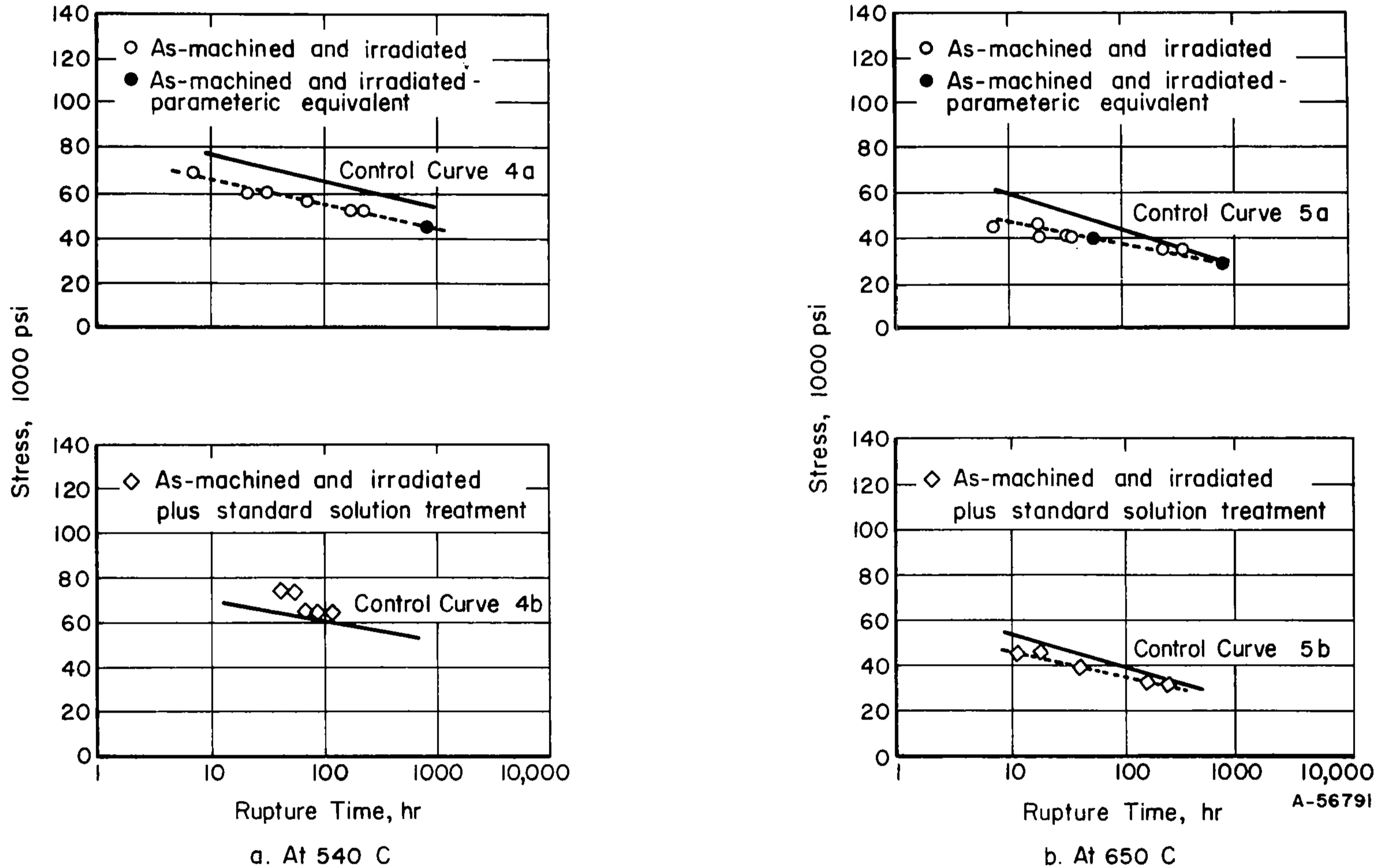

FIGURE 55. STRESS-RUPTURE STRENGTH OF HASTELLOY X ALLOY IRRADIATED SPECIMENS FROM HEAT E-9500[115]

Specimens irradiated to a fast fluence of 5 to 7 x 10^{19} n/cm^2.

TABLE 26. POSTIRRADIATION TENSILE PROPERTIES OF HASTELLOY N[109, 116]

Irradiation Temp., C	Test Temp., C	Fast Fluence, n/cm^2	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent: Uniform		Elongation, percent: Total		Reduction in Area, percent	
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
280	RT	3.7×10^{19}	40.6	71.2	110	113.3	62.9	47.5			59.1	50.0
280	RT	4.5×10^{19}	40.6	73	110	114	62.9	53.5				
280	RT	1×10^{20}	40.6	74.8	110	111.5	62.9	49.5			59.1	57.6
280	RT	1.2×10^{20}	40.6	73.7	110	116	62.9	57.1			59.1	57.3
280	RT	1.6×10^{20}	40.6	76.6	110	111.7	62.9	50.9			59.1	60.9
280	RT	5.4×10^{20}	40.6	80	110	117	62.9	52				
280	RT	8.3×10^{20}	40.6	99.6	110	131.6	62.9	42.4			59.1	51.5
740	RT	5×10^{19}		43		75.5				18.9		22.1
740	RT	1.9×10^{20}		50.3		81.3		11.0		11.2		12.8
740	300	1.9×10^{20}		36.8		67.3		14.3		15.4		21.5
50	650	8.3×10^{20}	32.0	44.7	82.5	52.7	30.5	6.1	32.1	7.1	40.1	18.0
50	650	1×10^{21}	32.0	43.8	82.5	50.6	30.5	3.6		4.9		7.0
280	650	1×10^{20}	26.0	29.8	57.8	50.5	20.2	12.7	21.3	13.9	29.3	32.1
280	650	8.3×10^{20}	26.0	NA	57.8	NA	20.2	<1.0	21.3	<1.0	29.3	16.3
280	650	1.5×10^{21}	26.0	19.6	57.8	19.8	20.2	<1.0	21.3	<1.0	29.3	7.2
740	650	9.2×10^{19}	30.5	23.3	63.1	32.5	20.8	5.0	21.6	5.4	18.6	16.2
740	650	2.2×10^{20}	30.5	30.2	63.1	36.5	20.8	3.8	21.6	3.9	18.6	9.4
740	650	2.4×10^{20}	30.5	25.1	63.1	32.6	20.8	3.3	21.6	3.6	18.6	23.3
740	650	4.0×10^{20}	30.5	32.5	63.1	37.4	20.8	1.9	21.6	2.2	18.6	12.1

TABLE 27. TENSILE STRENGTH AND DUCTILITY OF IRRADIATED AND UNIRRADIATED HASTELLOY N[a][117]

Deformation Temperature, C	Strength, 1000 psi: Yield, Irr.	Yield, Unirr.	True Tensile, Irr.	True Tensile, Unirr.	Ductility, percent: True Uniform Strain, Irr.	True Uniform Strain, Unirr.	True Fracture Strain, Irr.	True Fracture Strain, Unirr.
RT	46.3	45.5	168.6	166.5	42.3	40.6	42.5	39.0
100	43.9	43.9	159.5	161.0	40.1	40.3	44.6	37.2
200	38.4	40.7	150.6	157.5	40.3	41.9	42.5	50.7
300	36.0	40.7	154.3	147.0	42.2	37.9	44.6	41.4
400	35.0	40.7	146.9	153.0	40.2	39.3	42.5	46.9
500	35.8	35.8	129.5	144.0	35.3	42.4	--	--
600	32.5	36.2	82.4	109.0	11.8	26.7	21.9	31.6
700	31.0	34.1	53.4	102.8	8.0	30.8	11.6	42.1
800	28.5	30.9	38.4	59.9	3.7	12.2	6.9	86.6

(a) Irradiated to a fast fluence of 7×10^{20} n/cm^2.

material is tested at room temperature. Irradiation at temperatures above 700 C results in decreased strength and especially decreased ductility when tested at elevated temperatures. The postirradiation tensile properties have been found to depend on strain rate, composition, and postirradiation annealing, as follows.

Strain Rate. It has been shown that the strain rate has a considerable effect on the ductility of unirradiated Hastelloy N at elevated temperatures. The effect of strain rate on ductility at elevated temperature is increased significantly by irradiation as shown in Table 28.(117)

Composition. The addition of zirconium and titanium have been shown to improve the ductility of irradiated Hastelloy N at 650 C. With a strain rate of 0.002 in./min, the ductilities of irradiated (fast fluence of 2×10^{20} n/cm^2) Hastelloy N containing 0.05, 0.52, and 1.2 wt % zirconium were found to be 3.6, 7.5, and 11.5 percent, respectively.(118) The irradiated Hastelloy N which contains zirconium exhibits a strain-rate dependence at elevated temperatures similar to that for irradiated Hastelloy N which does not contain any zirconium.

Postirradiation Annealing. The effect of postirradiation annealing on the ductility of irradiated Hastelloy N at 650 C is illustrated in Table 29.(119) The data indicate that annealing for 1 hour at 400, 650, and 871 C causes further decreases in ductility at 650 C. However, a 1-hour anneal at 1200 C results in improved ductility. This is the first reported instance where postirradiation annealing improves the irradiation-induced embrittlement of nickel-base alloys at elevated temperatures.

Creep Properties. The postirradiation creep properties of Hastelloy N have been studied extensively since the containment vessel of MSRE (Molten Salt Reactor Experiment) is constructed of Hastelloy N. Figures 56(120) and 57(121) illustrate the effect of irradiation on the stress-rupture properties of Hastelloy N at 650 C and 760 C, respectively. The results indicate that irradiation causes a significant reduction in rupture life, but that at low stresses, the difference in rupture life between unirradiated and irradiated Hastelloy N at 650 C appears to diminish (Figure 56). The decrease in rupture life is due to drastic irradiation-induced reductions in ductility;

TABLE 28. STRAIN RATE SENSITIVITY OF IRRADIATED AND UNIRRADIATED HASTELLOY N[(a)](117)

Deformation Temperature, C	Strain Rate, min^{-1}	Strength, 1000 psi				Ductility, percent			
		Yield		True Tensile		True Uniform Strain		True Fracture Strain	
		Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.
500	0.2	32.7	34.9	136.1	145.6	41.2	42.3	46.6	53.4
500	0.02	35.8	35.8	129.5	144.0	35.3	42.4		51.4
500	0.002	34.4	37.4	122.5	131.5	32.3	33.3	36.5	34.2
600	0.2	32.9	34.1	112.7	134.4	31.2	38.9	36.5	48.7
600	0.02	32.5	36.2	82.4	109.0	17.7	26.7	21.9	31.6
600	0.002	34.2	34.6	63.1	106.0	10.3	29.9	13.2	29.7
700	0.2	30.5	30.9	66.8	106.5	13.5	32.2	19.2	39.2
700	0.02	31.0	34.1	53.4	102.8	8.0	30.8	11.6	42.1
700	0.002	32.1	33.7	47.0	80.5	5.6	20.0	7.8	29.0
800	0.2	29.3	29.3	45.7	79.8	6.7		11.6	
800	0.02	28.5	30.9	38.4	59.9	3.7	12.2	6.9	86.6
800	0.002	29.3	32.5	32.2	42.9	1.8	6.5	4.9	93.7

(a) Irradiated to a fast fluence of 7×10^{20} n/cm^2.

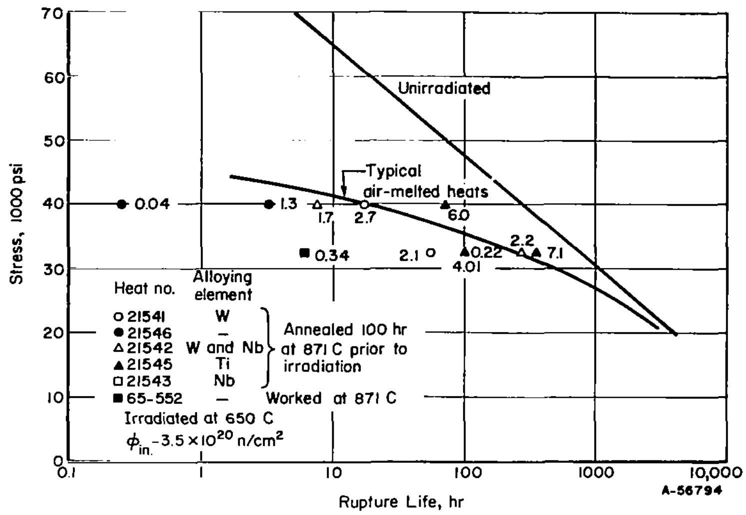

FIGURE 56. POSTIRRADIATION CREEP-RUPTURE OF SEVERAL HASTELLOY-N HEATS AT 650 C(120)

Numbers indicate fracture elongations.

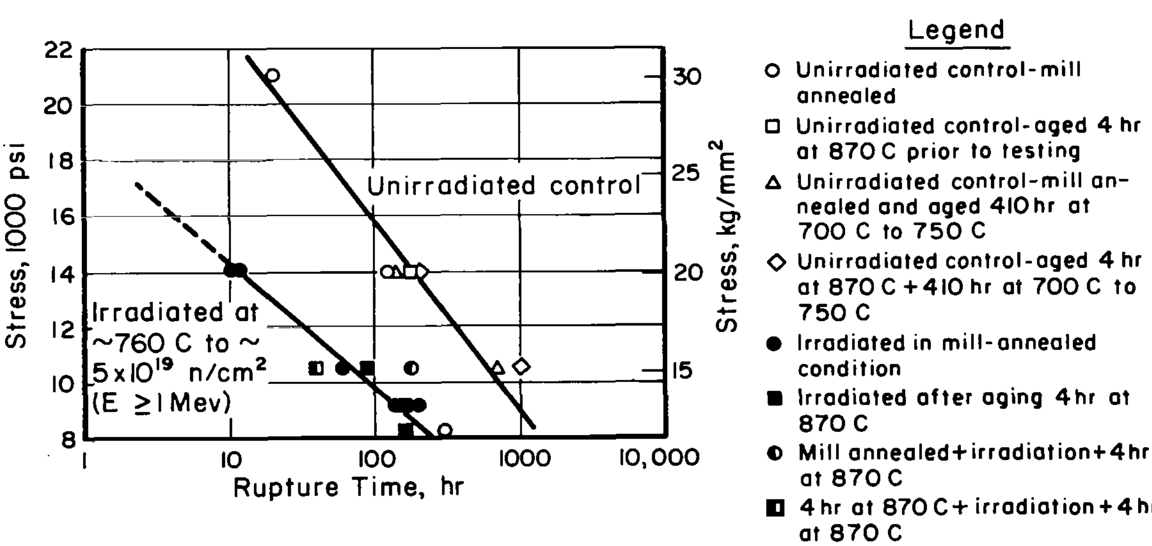

FIGURE 57. STRESS-RUPTURE PROPERTIES OF IRRADIATED AND UNIRRADIATED HASTELLOY N AT 760 C(121)

however, the actual creep rate does not appear to be significantly affected by irradiation.[120] The ductility reductions are attributed to formation of helium bubbles at the grain boundaries. The helium is formed by (n, α) reactions from the boron-10 impurity.

TABLE 29. EFFECT OF POSTIRRADIATION ANNEALING ON THE TENSILE PROPERTIES OF HASTELLOY N AT 650 C[a,b][119]

Postirradiation Anneal	Yield Strength, 1000 psi	Uniform Elongation, percent	Total Elongation, percent	Reduction in Area, percent
Heat 5065				
Unirradiated	46.3	22.8	24.0	28.1
As irradiated	40.8	12.2	13.1	21.9
400 C for 1 hr	41.9	11.0	11.5	16.3
650 C for 1 hr	42.1	10.9	12.4	15.5
871 C for 1 hr	43.9	6.6	6.7	10.7
1200 C for 1 hr	37.2	7.0	7.5	12.5
Heat 65-552				
Unirradiated	41.6	22.0	22.4	23.5
As irradiated	47.5	4.7	4.9	7.10
400 C for 1 hr	51.1	4.1	4.4	13.6
650 C for 1 hr	46.1	5.6	5.8	10.2
871 C for 1 hr	43.0	3.3	3.7	8.82
1200 C for 1 hr	38.8	12.7	12.8	18.0

(a) Irradiation temperature = 43 C; thermal fluence = 8.5 x 10^{20} n/cm^2.
(b) Strain rate of 0.002 in./in./min.

Inconel 600

Tensile Properties

The results of postirradiation tensile tests on Inconel 600 are summarized in Table 30. These results show that if the irradiation temperature is below 200 C, then considerable increases in strength and decreases in

TABLE 30. EFFECT OF IRRADIATION ON THE MECHANICAL PROPERTIES OF INCONEL 600

Test Temp., C	Irradiation Temp., C	Fast Fluence, n/cm^2	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reduction in Area, percent		Reference
							Uniform		Total				
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
RT	650	2.5×10^{19}	36	47	94	95			42	43			122
RT	650	1.2×10^{20}	36	34	94	94			42	37			122
RT	400	4×10^{20}	45.8	61.2					37.2	35.8			122
RT(a)	400	4×10^{20}	36.5	55-63	89.7	98-102			43	17-38			122
RT	175	6×10^{20}	45.0	105.5	93.8	118.8	41.3	25.5	48.7	31.6			110
RT	175	1.2×10^{21}	45.0	135.6	93.8	136.1	41.3	3.4	48.7	17.9			110
RT	175	2.4×10^{21}	45.0	144.7	93.8	146.4	41.3	3.8	48.7	11.4			110
RT	175	2.5×10^{21}	45.0	145.7	93.8	146.7	41.3	3.4	48.7	10.6			110
300(b)	740	9.2×10^{19}	32.3	32.9	84.2	86.5	40.9	41.9	44.9	47.5	53.1	57.1	109
300(c)	740	9.2×10^{19}	19.1	26.2	74.7	64.4	50.7	23.6	54.0	25.8	57.9	17.7	109
300(d)	740	9.2×10^{19}	26.9	45.1	86.5	90.9	39.9	23.2	44.5	25.5	59.3	45.1	109
300(b)	740	2.2×10^{20}	27.6	33.7	77.4	84.1	45.1	35.4	48.8	36.5	44.8	34.3	109
300(c)	740	2.2×10^{20}	24.2	20.5	75.7	57.3	46.4	23.3	50.0	24.4	43.5	27.5	109
300(d)	740	2.2×10^{20}	24.3	30.8	79.4	88.5	43.6	38.8	48.1	42.9	51.0	50.0	109
300(b)	740	2.4×10^{20}	29.7	31.7	84.2	63.8	41.3	14.7	46.2	15.1	60.5	35.2	109
300(c)	740	2.4×10^{20}	12.6	20.5	48.6	56.4	44.6	23.2	47.6	24.3	70.5	46.9	109
300(d)	740	2.4×10^{20}	25.5	29.4	84.6	88.6	40.9	39.0	43.8	45.2	56.5	57.5	109
300(b)	740	4×10^{20}		31.9		61.0		16.4		17.5		32.8	109
300(d)	740	4×10^{20}		37.7		88.9		32.5		34.7		34.7	109
315(a)	400	4×10^{20}	33.2	47	87.7	72			47.0	16			74
315	175	6×10^{20}		85.2		102.3		26.6		29.8			110
315	175	1.2×10^{21}		106.7		109.8		3.2		19.4			110
315	175	1.9×10^{21}		120		120.1		3.3		11.5			110
315	175	2.2×10^{21}		119		119		3.6		9.7			110
595	400	4×10^{20}	35.0	33.7					33.0	6.0			122
595(a)	400	4×10^{20}	30.1	15-35	76.5	23-40			35.0	2.7-3.2			122
650(c)	280	5.7×10^{20}	19.8	39.1	54.5	42.6	31.7	3.7	34.8	4.4	17.8	9.6	109
650(b)	280	5.7×10^{20}	30.2	48.1	66.8	56.2	23.3	3.3	36.4	3.8	56.8	7.5	109
650(d)	280	5.7×10^{20}	70.2	60.6	83.1	67.1	5.1	2.0	18.0	3.5	3.1	10.9	109
705(a)	400	4×10^{20}		14		14				0.2			122

(a) Specimens made from cladding.
(b) Mill annealed.
(c) Solution treated.
(d) 20 percent cold worked.

ductility take place.[110] These effects tend to increase with increasing fluence. As the irradiation temperature is increased, the displacement type of irradiation damage is annealed out and the postirradiation properties depend more on the aging characteristics of the alloy. Irradiation at 740 C results in large reductions in strength for Inconel 600, and similar reductions in strength take place in unirradiated thermal controls annealed at equivalent temperatures for equivalent times. Although the strength is considerably decreased by irradiation at 740 C owing to overaging, no expected improvement in ductility owing to overaging takes place for the irradiated material. The ductility of the irradiated Inconel 600 is drastically reduced at elevated temperatures, and the degree of the irradiation-induced embrittlement does not appear to depend on the irradiation temperature.

Inconel 625

Tensile Properties

A considerable number of irradiated Inconel-625 specimens have been tested at various temperatures and the test results are given in Table 31. The room-temperature properties appear to be primarily dependent on the irradiation temperature. However, it should be realized that a significant part of the change in properties is due to purely temperature effects. The maximum changes in room-temperature properties appear to be caused by an irradiation temperature of near 580 C. Higher irradiation temperatures causes significant annealing of irradiation-induced displacement-type damage. Irradiation at temperatures of 700 C and above seems to result in decreased elevated-temperature strength along with drastic reductions in ductility. The loss in ductility appears to depend on the fast fluence since it was not apparent after irradiation to a fast fluence of 9×10^{19} n/cm^2, but was significant after a fast fluence of 3.5×10^{20} n/cm^2, and quite drastic after a fast fluence of 2×10^{21} n/cm^2.

René 41

Tensile Properties

The effect of irradiation on the notched and unnotched tensile properties of René 41 is illustrated in Table 32[124, 125]. These test results indicate

TABLE 31. MECHANICAL PROPERTIES OF IRRADIATED INCONEL 625

Irradiation Temp., C	Test Temp., C	Fast Fluence, n/cm^2	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Uniform Elongation, percent		Reduction in Area, percent		Reference
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
50	RT	9.8×10^{19}	94	124.5	150	150	36.4	29.3	35.5	60.3	123
50	RT	1.8×10^{20}	94	138.6	150	147.5	36.4	23.1	35.5		123
50	RT	4.6×10^{20}	94	125.5	150	151.5	36.4	28.1	35.5	53.4	123
50	RT	7.6×10^{20}	94	160.6	150	166	36.4	8.4	35.5	49.5	123
280	RT	5.3×10^{19}	102.1	118.1	158.5	169.8	39.2	38.2	48.9	43.2	123
280	RT	1.3×10^{20}	102.1	110.2	158.5	153.4	39.2	40.3	48.9	52.2	123
280	RT	2.6×10^{20}	102.1	106	158.5	150.1	39.2	44.7	48.9	52.9	123
280	RT	7.7×10^{20}	102.1	100	158.5	136.3	39.2	47	48.9	54.7	123
280	RT	8.3×10^{20}	102.1	117.6	158.5	148.6	39.2	31.5	48.9	64.7	123
580	RT	9.2×10^{19}	134	172.1	179	199.4	27.3	17.6	25	31.9	123
704	RT	2.0×10^{21}	120	111	169	154	31	23	—	--	114
740[a]	RT	9×10^{19}	81.9	100.4	106.3	128.0	5.9	6.4	3.7	11.7	123
740[b]	RT	9×10^{19}	92.6	113.8	134.9	156.8	10.8	12.1	10.8	10.1	123
280	300	1×10^{20}		94.3		139.5		26.7		17.6	123
740	300	9.2×10^{19}	94.2	100	149.5	143.7	18.1	6.2	17.7	16.7	123
740	300	1.9×10^{20}	91.5	85.2	140.2	128.1	9.0	5.3	11.0	9.7	123
50	650	1.7×10^{21}	83.6	108	129.6	125.6	19.1	7.2	16.9	17.5	123
280[b]	650	1×10^{20}	85.3	73.9	133.8	101.1	5.0	8.3	18.2	15.3	123
280[b]	650	3.2×10^{20}	85.3	85.1	133.8	109.1	5.0	9.1	18.2	17.1	123
280[b]	650	5.7×10^{20}	85.3	88.6	133.8	108.6	5.0	13.2	18.2	18.2	123
280[b]	650	1.5×10^{21}	85.3	45.4	133.8	52.1	5.0	4.5	18.2	28.9	123
704	704	3.5×10^{20}	87	75	106	90	8	4			114
704	704	3.5×10^{20}	89	80	110	88	7	1			114
740[a]	650	9.2×10^{19}	80.0	72.1	112	96.8	8.2	4.8	11.5	8.0	123
740[b]	650	9.2×10^{19}	84.6	66.2	124.6	96.7	10.5	6.6	11.1	11.0	123
704	593	3.5×10^{20}	91	74	146	122	22	8			114
704	593	2.0×10^{21}	104	92	146	114	19	3			114

(a) Mill annealed.

(b) Solution annealed.

TABLE 32. EFFECT OF IRRADIATION ON THE TENSILE PROPERTIES OF RENÉ 41[124, 125]

Test Temp., C	Irradiation Temp., C	Fast Fluence, 10^{19} n/cm^2	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Total Elongation, percent		Reduction in Area, percent	
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
RT[a, b]	675	4			234.5	236				
RT[a, b]	675	4			234.5	236				
RT[a]	675	5	117	94.5	137	159.7	3.3	21.9	4.8	4.7
RT[a]	675	5	117	121	137	139	3.3		4.8	
RT[a]	870	6	135.1	120.2	161.3	141.7	2.0	1.4	3.0	6.3
RT[a]	550	7	102.3	111.5	145.5	134	11.5		12.4	
RT[a]	540	7	102.3	111.6	145.5	133.1	11.5	7.5	12.4	7.9
RT[c, b]	630	11			234.6	236.3				
RT[c]	630	11	150.7	141.6	199.0	181.1	9.9	13.0	8.3	12.5
540[a, d]	615	5	87.2	83.5	136.5	124.8	14.8	9.7	16.9	13.9
540[a]	550	7	94.7	105.9	130	140	11	17.9	15.8	19.7
540[a]	550	7	94.7	101.5	130	134	11		15.8	
650[a, b]	675	4			220.2	209				
650[a, b]	675	4			233.9	204				
650[a, b]	675	4			233.9	199				
650[a, b]	675	4			220.2	153				
650[a, b]	675	4			187	158				
650[a]	675	5	102	83.5	144	129.8	12.2		12.6	
650[a]	675	5	102	105	144	140.5				
650[a]	675	5	102	107	144	135				
650[a]	615	5	102.4	106	143.5	138	12.3	8.1	13.6	11.6
650[c]	675	6	130.6	145	191.8	183	12.1	5.6	10.9	9.4
650[c]	675	6	130.6	143	191.8	179	12.1	4.7	10.9	9.4
650[a]	870	6	114.2	125.4	180.2	165.1	4.1	4.3	5.4	6.3
650[c]	630	11			233.9	201.3				
650[c]	630	11			220.2	208.6				
650[a]	630	11	132.2	131.2	190.2	170.2	13.0	8.1	11.8	8.5
650[a]	675	11	132.2	132	190.2	175	13.0	8.8	11.8	6.3
870[a, b]	870	6			119	77.4				
870[a, b]	870	6			119	55				
870[a, b]	870	6			119	66.5				
870[d]	815	5	49.9	51.0	74	57	24.8	2.6	57.0	10.1
870[a][d]	815	5	70	60.4	77	61.7	11.5	3.5	32.1	8.8
870[c]	815	5	40.7	32.6	65	36.3	12.7	3.0	55.6	9.3
870[c]	870	6	40.7	35.9	65	40.8	12.7	3.1	55.6	7.8
870[c]	870	6	40.7	29.3	65	31.8	12.7	2.8	55.6	10.9
870[a][d]	800	6	70	63.4	78.5	65.1	12.2	3.4	32.1	8.0
870[a][d]	800	6	70	57.4	78.5	58.2	12.2	3.6	32.1	9.6
870[a]	860	6	49.9	47.3	74	53	24.8	2.6	57.0	15.4
870[a]	860	6	49.9	54.8	74	61	24.8	2.7	57.0	4.9
870[a]	870	6	54.2	49.4	79.5	53.4	26.7	3.1	60.4	4.8

(a) Heat treated:
2 hours at 1065 C, water quenched
1/2 hour at 1175 C, air cooled
4 hours at 900 C.

(b) Notched specimens.

(c) Heat treated:
2 hours at 1065 C, water quenched
1/2 hour at 1065 C, air cooled
16 hours at 760 C.

(d) Heat treated after irradiation for 1/2 hour at 1175 C, air cooled.

that irradiation at temperatures of 650 C does not significantly affect the room temperature and 540 C tensile properties. However, testing at 650 C or above results in significant decreases in ductility and minor decreases in strength. Irradiation and tensile testing at 870 C result in large reductions in both strength and ductility. The notch strength of René 41 is decreased by irradiation at all testing temperatures above 650 C.

Creep Properties

The effect of irradiation on the stress-rupture properties of René 41 is illustrated in Figure 58.(126) Both the rupture strength and ductility at 650 C are decreased with increasing fast fluence.

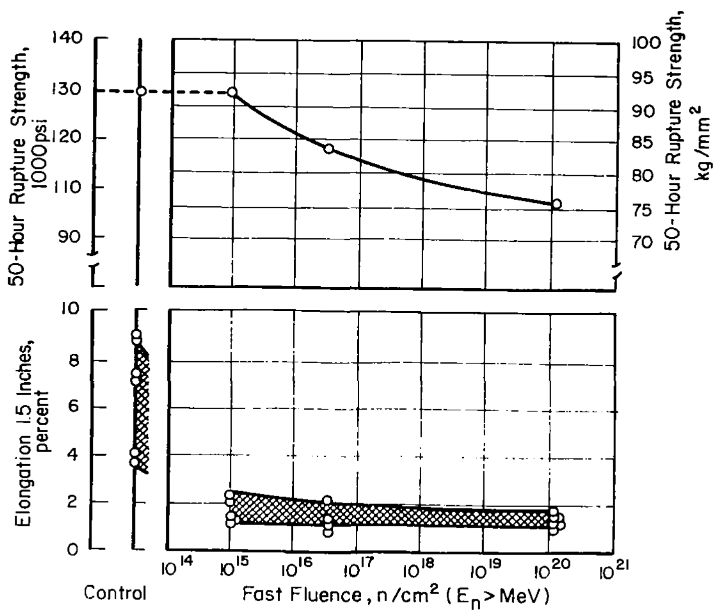

FIGURE 58. STRESS-RUPTURE STRENGTH AND ELONGATION OF RENÉ 41 AT 650 C AS A FUNCTION OF FAST FLUENCE(126)

COBALT-BASE ALLOYS

Only limited interest has been shown to cobalt-base alloys because of the high neutron capture cross section. Results of the limited tensile tests are shown in Table 33.(106, 127) The cobalt alloys behave similarly to the nickel-base alloys and austenitic stainless steels. At low testing temperatures the displacement type of radiation damage causes increases in strength and reductions in ductility. Annealing the irradiated material restores the preirradiation tensile properties for low testing temperatures. However at high testing temperatures the irradiation induced embrittlement causes large reductions in ductility. This ductility loss cannot be restored by annealing at high temperatures.

NIOBIUM ALLOYS

Data showing the effect of irradiation on the mechanical properties of niobium alloys are given in Table 34. These data indicate that irradiation at 330 C causes more embrittlement than irradiation at 50 C – even at fast fluences as low as 1×10^{18} n/cm^2.(128) Irradiation at 50 C to a fast fluence of about 1×10^{20} n/cm^2 causes minor strength increases accompanied by drastic reduction in uniform elongation. The fracture mode also changes from a ductile type to a cleavage type.(116) Material with larger grain size will undergo greater ductility decreases when irradiated, conforming to the Petch relationship.(128) Annealing studies on irradiated niobium plotted in Figure 59 indicate that the preirradiation mechanical properties are restored by an annealing at about 600 C.(129) Preirradiation mechanical properties are recovered for specimens tested at 1900 C by the 30-minute soaking time at test temperature.(109)

The yield strength of niobium-1 wt % zirconium was determined by bend testing after irradiation to a fast fluence of 1×10^{20} n/cm^2(132). It was found that irradiation increased the yield strength of the niobium alloy, but that the irradiated material retained sufficient ductility even when tested at -75 C. Cold-worked niobium-1 wt % zirconium specimens underwent smaller increases in yield strength, with the welded material behaving similarly to the annealed material. Figure 60 illustrates the relative irradiation-induced increases in the yield strength, along with annealing-induced recovery at

TABLE 33. TENSILE PROPERTIES OF IRRADIATED AND UNIRRADIATED COBALT-BASE ALLOYS

Alloy	Test Temp., C	Fast Fluence, 10^{20} ns/cm^2	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Total Elongation, percent		Ref.
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Haynes Stellite 25	RT	1.3	67.0	139.0	152.0	142	55.0	15.0	127
	650	1.3		45.6		70.6		17	127
	750	1.3		35.0		38.2		9.2	127
	800	1.3	35.9	39.2	41.5	41.5	10.5	2.3	127
	RT*	14	66.3	56.2	139	133	68	55.6	106
	315*	14	39.7	36.0	118.5	117	86	60.2	106
	450*	14	36.1	29.9	96.3	70.3	42	22.8	106
	500*	14		31.0		66.5		17.0	106
	550*	14	33.9	28.5	82.3	53.5	30	8.8	106
	650*	14	32.3	29.8	69.3	40.5	17	4.5	106
UMCo-50	RT*	14	47.5		126	127	35	27.4	106
	315*	14	33.6	29.7	121	112.1	61	56.7	106
	450*	14	28.9	26.0	82.5	75.1	24.5	30.7	106
	550*	14	25.0	24.6	72.0	49.1	27	13.9	106
UMCo-51	25	1.3	84.7	120	171	146	36	20	127
	650	1.3	37.8	46.4	72.2	68.6	14	9	127
	750	1.3		38.7		46.6		7.5	127
	800	1.3	30.0	34.2	40.3	39.8	10	6.5	127
Multimet	25	1.3	71.0	116	121	138.0	43	31	127
	650	1.3		28.1		53.5		11	127
	800	1.3	23.6	30.0	32.9	31.3	55	6.7	127
Haynes 188	25	1.3	71.6	133	142.5	167	25	44	127
	650	1.3		46.1		60.8		10	127
	750	1.3		38.6		38.6		3	127
	800	1.3	37.5	37.4	45.5	37.4	65	1.5	127
S-1	25	1.3	47.0	114	95.2	118	18.5	4	127
	650	1.3		37.9		47.4		5	127
	750	1.3		30.5		31.7		2	127
	800	1.3	19.5	23.6	33.8	23.6	29.0	1.5	127

*Specimens were annealed for 1 hr at 800°C before testing.

TABLE 34. EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF NIOBIUM ALLOYS

Material[a]	Fast Fluence, n/cm^2 (>1 Mev)	Irradiation Temp., C	Test Temp., C	Yield Strength, 1000 psi Unirr.	Yield Strength, 1000 psi Irr.	Ultimate Tensile Strength, 1000 psi Unirr.	Ultimate Tensile Strength, 1000 psi Irr.	Elongation, percent Uniform Unirr.	Elongation, percent Uniform Irr.	Elongation, percent Total Unirr.	Elongation, percent Total Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Reference
Niobium (99.8%)	9.1×10^{17}	77	RT	19.5	42.5	40.0	46.2			51.5	41.5			128
	9.1×10^{17}	77	RT	17.3	40.7	34.8	42.3			53.0	38.5			128
	9.1×10^{17}	77	RT	17.3		34.8	45.6			53.0	37.1			128
	9.1×10^{17}	77	RT	21.4	41.4	30.0	43.1			45.0	32.9			128
	9.1×10^{17}	330	RT	19.5		40.0	55.7			51.5	27.8			128
	9.1×10^{17}	330	RT	19.5	46.0	40.0	52.4							128
	9.1×10^{17}	330	RT	19.5		40.0	54.9			50.0	31.4			128
	9.1×10^{17}	330	RT	21.4	43.8	30.0	47.6			45.0	24.3			128
	9.1×10^{17}	330	RT	21.4		30.0	44.2			45.0	20.8			128
Niobium	1×10^{20}	125-175	RT	59.4	77.2	71.8	80.4			20.6	8.0			129
	1×10^{20}	125-175	RT	59.4	74.1	71.8	74.4			20.6	6.4			129
	1×10^{20}	125-175	RT	59.4	74.4	71.8	77.0			20.6	6.5			129
	1×10^{20}	125-175	RT[b]	59.4	92.2	71.8	92.8			20.6	6.8			129
	1×10^{20}	125-175	RT[b]	59.4	92.2	71.8	92.7			20.6	7.7			129
	1×10^{20}	125-175	RT[b]	59.4	91.2	71.8	91.7			20.6	4.2			129
	1×10^{20}	50	RT	72	91	73	91			11.0	7.0			130
	2.1×10^{20}	70	RT	18.7	39.2	31.6	40.1	24.4	0.5	35.5	18.6			131
	2.1×10^{20}	70	RT[c]		41.8		44.2		2.5		15.9			131
	2.1×10^{20}	70	RT[d]		33.2		41.2		5.4		13.5			131
	2.1×10^{20}	70	RT[e]	20.1	19.6	30.6	31.5	25.3	28.3	35.3	43.0			131
	2.1×10^{20}	70	300[c]	21.6	32.7	28.7	33.0	9.2	0.4	13.4	5.9			131
	2.1×10^{20}	70	650[d]	9.8	17.5	15.3	17.5	9.0	1.0	14.7	12.6			131
Nb-0.6Zr, S	2×10^{20}	50	RT	67.4	93	71.5	93.5			19	12.8			130
Nb-0.6Zr, R	2×10^{20}	50	RT	75	102	84.7	104			15	11.8			130
	2×10^{20}	50	RT	75	106	84.7	108			15	13.2			130
	2×10^{20}	50	RT	75	99.4	84.7	101.5			15	13.2			130
	2×10^{20}	50	RT[f]		85		87				14.9			
	2×10^{20}	50	RT[g]	69.9	73.5	79.4	79.7			18.6	18.3			
Nb-0.6Zr, S	2×10^{20}	50	RT[h]	59.1	53.6	69.1	63.6			23	18.8			
Nb-0.6Zr, R	2×10^{20}	50	RT[h]		65.7		73.4				19.0			
Nb-1Zr	8.8×10^{19}	50	RT	55.1	59.0	59.7	62.8	3.2	0.5			78.8	68.7	116
	8.8×10^{19}	50	RT	55.1	62.7	59.7	64.5	3.2	0.6			78.8	72.0	116
	1.0×10^{20}	50	RT	55.1	62.4	59.7	64.9	3.2	0.6			78.8	72.2	116
	1.0×10^{20}	50	RT	55.1	63.0	59.7	64.2	3.2	0.5			78.8	55.4	116
	1.5×10^{20}	50	RT	55.1	61.4	59.7	63.6	3.2	0.6			78.8	72.3	116
	1.5×10^{20}	50	RT	55.1	64.5	59.7	65.3	3.2	0.6			78.8	80.5	116
Nb-1Zr	5.9×10^{20}	50	1090	9.4	9.1	10.5	9.9	1.4	1.3	52.3	49.2			109
Cb-752	8.8×10^{19}	50	RT	66.7	109	81.9	109.2	14.5	0.8			56.6	58.2	116
	8.8×10^{19}	50	RT	66.7	101.4	81.9	104.9	14.5	0.7			56.6	51.2	116
	1.0×10^{20}	50	RT	66.7	98.5	81.9	104.2	14.5	1.6			56.6	57.6	116
	1.0×10^{20}	50	RT	66.7	104.6	81.9	106.5	14.5	0.8			56.6	58.7	116
	1.5×10^{20}	50	RT	66.7	106.8	81.9	107.0	14.5	0.9			56.6	60.3	116
	1.5×10^{20}	50	RT	66.7	107.8	81.9	108.1	14.5	0.9			56.6	59.0	116
	9×10^{20}	50	1090	17.9	17.9	24.0	23.8	5.4	8.7	30.3	33.3	87.6	87.8	109

(a) S - swaged, R - rolled.
(b) Annealed at 200 C for 1 hour.
(c) Annealed at 300 C for 1 hour.
(d) Annealed at 650 C for 1 hour.
(e) Annealed at 1000 C for 1 hour.
(f) Annealed at 595 C for 1 hour.
(g) Annealed at 760 C for 1 hour.
(h) Annealed at 870 C for 1 hour.

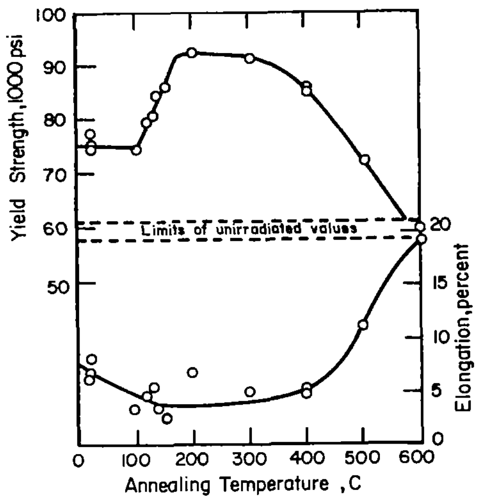

FIGURE 59. EFFECTS OF POSTIRRADIATION ANNEALING ON THE ROOM TEMPERATURE TENSILE PROPERTIES OF NIOBIUM(129)

Specimens irradiated to a fast fluence of 9.1×10^{17} n/cm^2.

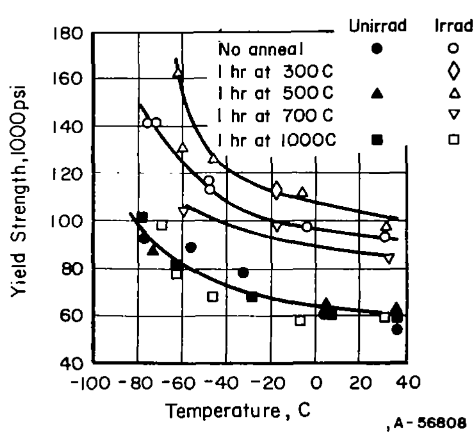

FIGURE 60. YIELD STRESS OF RECRYSTALLIZED NIOBIUM-1 WEIGHT PERCENT ZIRCONIUM BEFORE AND AFTER IRRADIATION(132)

Specimens received fast fluences of 0.3 to 1.1×10^{19} n/cm^2.

various temperatures. The yield strength reaches a maximum after an annealing temperature of 500 C but is reduced below the as-irradiated value after an anneal of 700 C. After annealing at 1000 C, the unirradiated yield strength is completely recovered.

Limited in-pile stress-rupture tests have been performed on niobium-1 wt % zirconium specimens at 980 and 1095 C.(133) These specimens were irradiated in a helium atmosphere in an instantaneous fast flux of 3×10^{13} n/(cm$^2\cdot$ s). Therefore, the fast fluence received by any specimen would depend on its life before rupture. For example, a specimen with a rupture life of 1000 hours would receive a fast fluence of 1.08×10^{20} n/cm^2, while a specimen with a rupture life of 10 hours would receive a fast fluence of 1.08×10^{18} n/cm^2. Results of the tests (Figure 61) indicate that irradiation may cause a minor reduction in rupture life, although the effect may be due to testing of the irradiated material in helium atmosphere and of the unirradiated material in vacuum.

MOLYBDENUM ALLOYS

Molybdenum exhibits a ductile-to-brittle transition temperature similar to that for other body-centered metals, and, consequently, the possibility of an increase in transition temperature by irradiation is of utmost importance. The irradiation induced transition temperature shift measured by various investigators is tabulated in Table 35. These somewhat conflicting results

TABLE 35. EFFECT OF NEUTRON IRRADIATION ON THE TRANSITION-TEMPERATURE SHIFT OF MOLYBDENUM ALLOYS

Material Condition	Fast Fluence, n/cm^2	Transition Temperature, C		Temperature Shift, C	Reference
		Unirr.	Irr.		
Molybdenum[a]	2×10^{18}	-30	-40	26	134
Molybdenum[b]	4×10^{19}	-110	-110	0	135
Molybdenum[c]	5×10^{19}	-136	-73	63	136
Molybdenum[a]	5×10^{20}	-30	+70	100	134
Mo-0.5Ti	1×10^{20}	19	138	119	132

(a) Recrystallized.
(b) Swaged.
(c) Stress relieved at 1030 C.

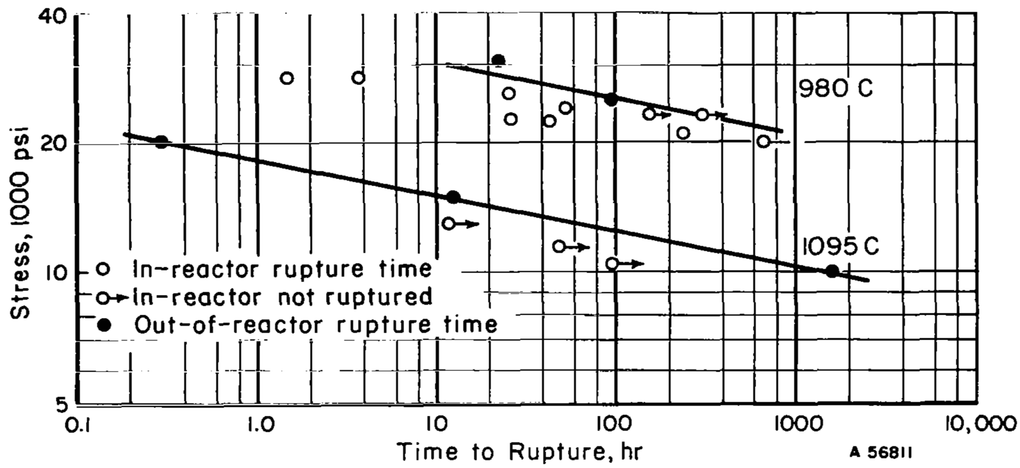

FIGURE 61. EFFECT OF NEUTRON BOMBARDMENT ON THE STRESS-RUPTURE STRENGTH OF NIOBIUM-1 PERCENT ZIRCONIUM ALLOY AT 1800 AND 2000 F[133]

Instantaneous fast neutron flux was 3×10^{13} n/(cm^2)(sec).

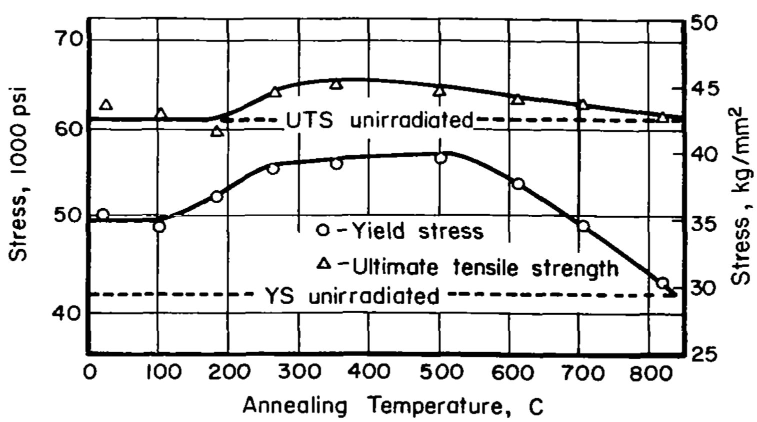

FIGURE 62. ROOM TEMPERATURE YIELD STRESSES AND ULTIMATE TENSILE STRENGTHS OF IRRADIATED MOLYBDENUM SPECIMENS AS A FUNCTION OF ANNEALING TEMPERATURE[138]

indicate that variables other than irradiation effects, such as interstitial impurities, may be the overriding factors determining the ductile-to-brittle transition temperature of the material after irradiation.

The limited tensile data on irradiated molybdenum and its alloys are given in Table 36 and Figure 62. The room-temperature yield and tensile strength are increased by fast-neutron irradiation and the ductility is decreased, as is true of most irradiated metals. The postirradiation aging is believed to cause embrittlement by providing the thermal energy for formation of larger and more stable defect clusters, either among the defects caused by irradiation or the interstitial impurity atoms present. The agglomeration of defects in irradiated molybdenum increases room-temperature strength with increasing annealing temperature, eventually reaching a maximum thermal hardening temperature of about 500 C (Figure 62). Annealing at higher temperatures decreases room-temperature strength, and the preirradiation strength properties are restored by annealing at 800 C.

Creep-rupture tests at 870 and 780 C have been performed on molybdenum specimens which had received a fast fluence of 6.9×10^{18} n/cm^2.[139] No differences in stress-rupture properties were found between the irradiated and unirradiated material, indicating that the test temperature was sufficiently high to remove any irradiation effects. Creep-rupture studies on irradiated molybdenum were also performed in the 560 to 650 C range. Results of these creep-rupture tests on irradiated molybdenum are shown in Figure 63. It can be seen that irradiation to a fast fluence of 3.7×10^{19} n/cm^2 considerably increases creep rate (60-270 percent) at those testing temperatures. Since the elongation at rupture is also reduced, the life to rupture is significantly decreased by irradiation. Specimens tested at 580 C after irradiation to a fast fluence of 1×10^{19} n/cm^2 showed a 25-fold increase in creep rate when compared with the unirradiated specimens. Since the elongation to rupture was also reduced by irradiation, a drastic reduction in time to rupture resulted. Annealing irradiated molybdenum specimens above 0.31 Tm (770 C) restored the preirradiation creep rates. The reasons for this reduction in creep-rate acceleration with increasing fast fluence are not presently clear.

The effect of irradiation temperature on the creep-rupture properties of polycrystalline molybdenum irradiated to 1.6×10^{20} n/cm^2 is illustrated in Figure 64.[140] For a stress of 25,600 psi at 750 C the rupture life increases with increasing irradiation temperature.

TABLE 36. EFFECT OF IRRADIATION ON THE MECHANICAL PROPERTIES OF MOLYBDENUM AND TANTALUM ALLOYS

Material	Fast Fluence, n/cm^2 (> 1 MeV)	Irradiation Temp., C	Test Temp., C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, percent				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Molybdenum[a]	5×10^{19}	100	-60	93.7	99.4	99.8	104.3			23.6	22			136
	5×10^{19}	100	80	80	93.3	90.5	93.5			23.8	18.5			136
	5×10^{19}	100	200	70.4	85.5	74.6	85.9			2.8	5.8			136
Molybdenum	3×10^{20}	50	1090	24.2	22.6	24.4	22.8	0.56	0.42	33.8	27.4	92.4	91.5	109
	3.2×10^{20}	50	1090	24.2	23.8	24.4	24.0	0.56	0.49	33.8	38.6	92.4	88.1	109
Commercial molybdenum	0		-60	142		142				0		0		134
	0		-40	123		123				0		0		134
	0		-20	125		120				32.8		63.8		134
	5.1×10^{20}	90	RT	102.5	151.7	100.8	151.7			45.7	0	72.4	0.1	134
	5.1×10^{20}	90	RT	93.8		98.8	109.7			41.7	0	65	0	134
	5.8×10^{20}	90	60				148.5				0		0	134
	5.8×10^{20}	90	80		143.5		143.5				14.7		60.7	134
	5.8×10^{20}	90	100		111.5		111.5				10		59.7	134
TZM	2.4×10^{20}	50	1090	51.4	57.1	53.3	60.5	1.1	1.1	11.6	8.0	66	47.5	109
Mo-50Re sheet	3.4×10^{19}	70	RT	115	178	142	179			28.5	13.1			137
	4.2×10^{19}	70	RT	115	212	143	214			28.5	8.6			137
	3.4×10^{19}	70	RT[b]	116	173	141	174			31.0	17.3			137
	4.2×10^{19}	70	RT[b]	116	197	141	206			31.0	9.2			137
	3.4×10^{19}	70	RT[c]	111	119	142	147			31.0	28.3			137
	4.2×10^{19}	70	RT[c]	111	134	142	155			31.0	24.0			137
	3.4×10^{19}	70	RT[d]	109	115	141	147			30.3	28.0			137
	4.2×10^{19}	70	RT[d]	109	117	141	146			30.3	28.0			137
Ta	1×10^{20}	50	RT			128	148			8.6	7			116
Ta-1.5W	3.9×10^{20}	50	RT	31.0	65.8	44.9	69.5			39	16			141
Ta-3.0W	8×10^{20}	50	RT	38.5	81.4	52.5	86.3			35	7			141
Ta-10W	3×10^{20}	50	RT			166	290			7.4	4.9			116

(a) Stress relieved at 1030 C before irradiation.
(b) Annealed at 250 C for 1 hour.
(c) Annealed at 400 C for 1 hour.
(d) Annealed at 700 C for 1 hour.

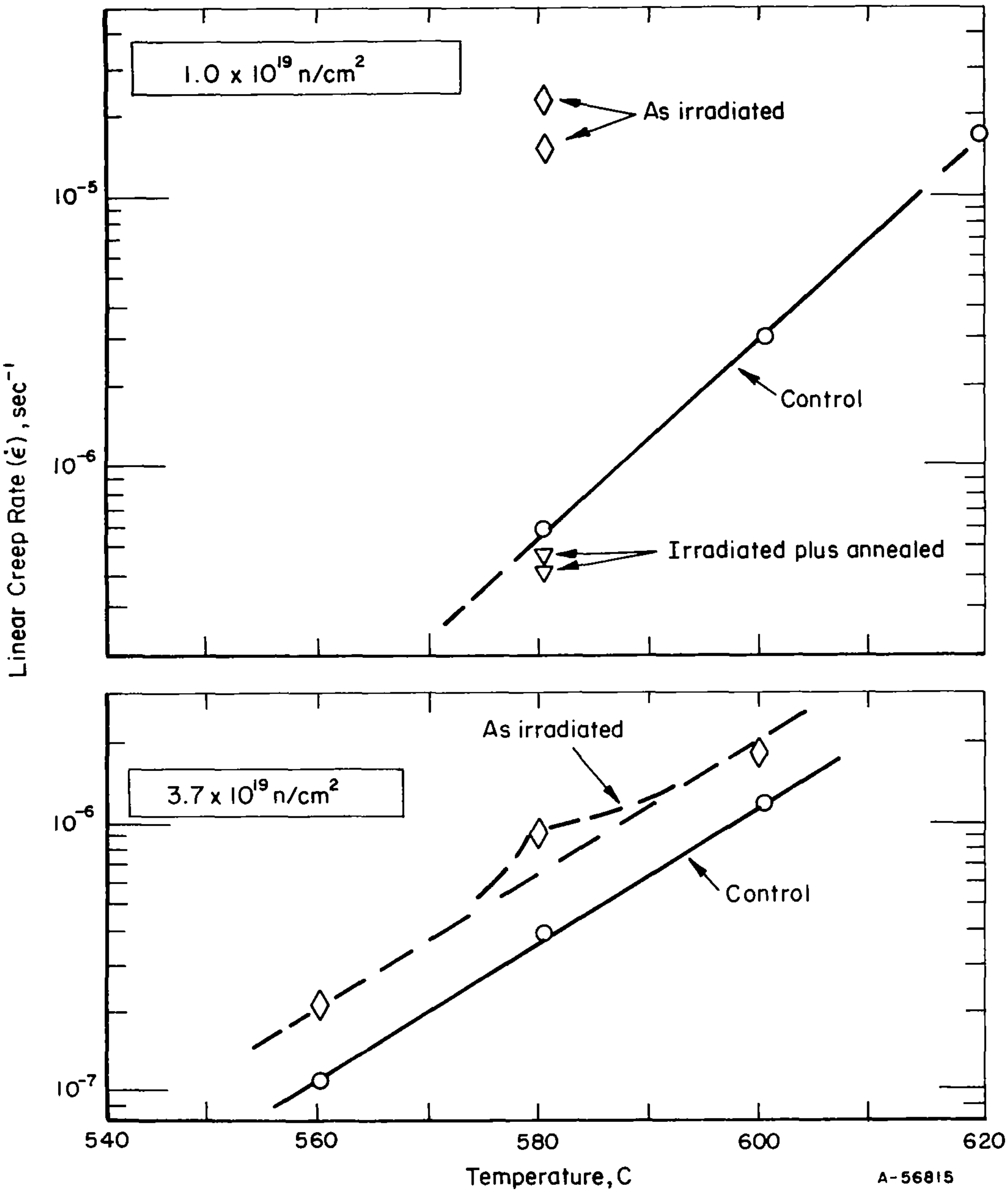

FIGURE 63. CREEP RATE OF RECRYSTALLIZED MOLYBDENUM IRRADIATED TO FAST FLUENCES OF 1.0 x 10^{19} AND 3.7 x 10^{19} N/CM^2 (139)

The anneal consisted of heating the spe ~ens for 1 hour at 770 C. The initial load in all tests v 000 psi.

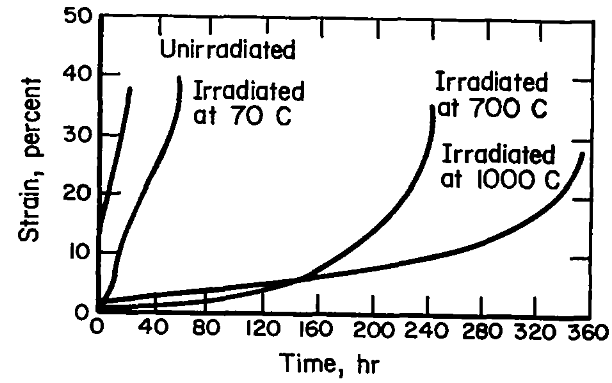

FIGURE 64. CREEP-RUPTURE PROPERTIES OF POLYCRYSTALLINE MOLYBDENUM AT A TEST TEMPERATURE OF 750 C AND AN INITIAL STRESS LEVEL OF 25,600 PSI FOLLOWING NEUTRON IRRADIATION AT THREE TEMPERATURES TO 1.4 - 1.8 x 10^{20} NEUTRONS/CM^2 (E> 1 MEV) IN A WATER-MODERATED REACTOR[140]

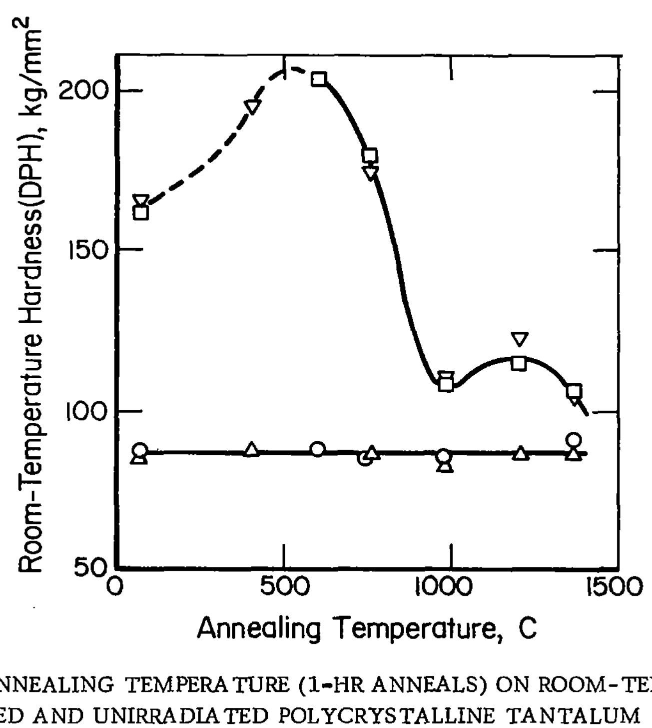

FIGURE 65. EFFECT OF ANNEALING TEMPERATURE (1-HR ANNEALS) ON ROOM-TEMPERATURE HARDNESS OF IRRADIATED AND UNIRRADIATED POLYCRYSTALLINE TANTALUM

Irradiated to 1.2 x 10^{20} neutrons/cm^2 (E> 1 MeV) at 700 C: □, 500-g load; ▽, 100-g load - normalized data. Unirradiated: ○, 500-g load; △, 100-g load - normalized data.[142]

Creep-rupture tests have been performed on TZM specimens irradiated to a fast fluence of 6.8×10^{19} n/cm^2.(83) The rupture life of irradiated specimens tested at 1310 C increased slightly. The elongation at rupture was about the same for the unirradiated and irradiated specimens, indicating that the irradiation effects were mostly annealed out at the testing temperature.

TANTALUM

Only limited data concerning the effect of irradiation on the mechanical properties of tantalum are available. Results of the few tensile tests on irradiated tantalum and its alloys are given in Table 36. A study was performed to distinguish the two different irradiation-induced mechanisms that affect the mechanical properties of tantalum. One of these effects is due to fast-neutron-caused displacement-type damage, while the other is due to solid-solution hardening from tungsten atoms introduced by transmutation from tantalum by thermal neutrons.(141) The data given in Table 36 suggest that the effects of solid-solution strengthening caused by the transmuted tungsten atoms is minor compared to fast-neutron-caused effects.

The effect of postirradiation annealing on the hardness of tantalum is illustrated in Figure 65. Almost all of the radiation-induced hardness increase is removed by 1 hr at 1000 C.(142)

TUNGSTEN ALLOYS

Results of the few tensile tests performed on irradiated tungsten and tungsten alloys are shown in Table 37 and Figure 66.(143) The low-temperature tensile tests on the irradiated material produced rather unusual results since the ductility was found to be improved by irradiation.(136) Tensile tests at 400 C indicate a drastic ductility decrease after a fast fluence of 1 to 2×10^{19} n/cm^2 (Figure 67). This ductility decrease was found to be more drastic for tungsten shielded by cadmium foil. This foil

TABLE 37. EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF TUNGSTEN ALLOYS

Material	Fast Fluence, n/cm^2 (> 1 MeV)	Irradiation Temp., C	Test Temp., C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Uniform Elongation, percent		Total Elongation, percent		Reduction in Area, percent		Reference
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Tungsten	5×10^{19}	100	100	137.0	152.0	Fractured at yield				0	0			136
	5×10^{19}	100	200	148.0	131.0	173.2	173.0			2.4	4.2			136
W-25Re	2.4×10^{20}	50	1090	88.2	108	95	113.6	1.8	0.3	23.3	14.3	75.2	65.6	109
	3.4×10^{20}	50	1090	88.2	107.1	95	114.7	1.8	2.6	23.3	13.9	75.2	65.1	109
	5.6×10^{20}	50	1090	88.2	121.5	95	147.2	1.8	12.6	23.3	16.3	75.2	33.2	109

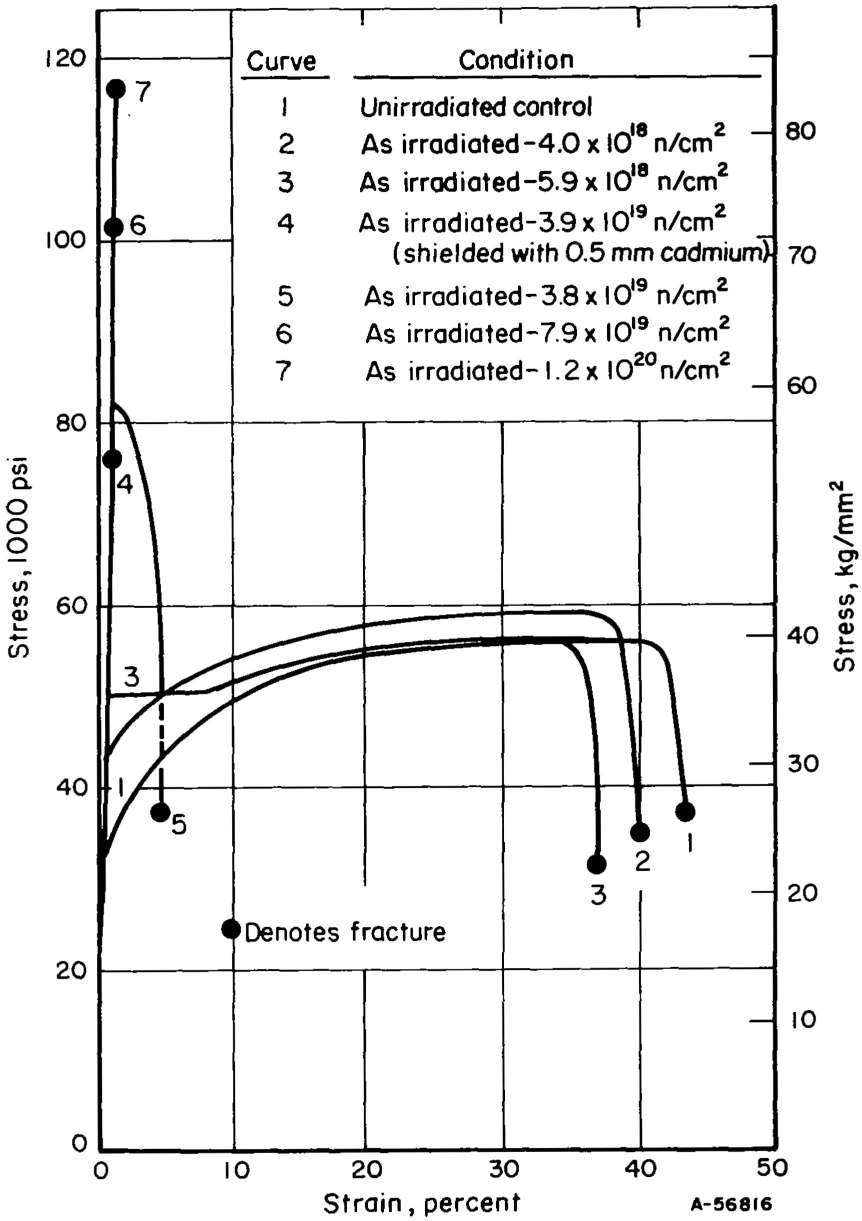

FIGURE 66. STRESS VERSUS STRAIN CURVES FOR CONTROL AND IRRADIATED TUNGSTEN SPECIMENS TESTED AT 400 C IN HELIUM[143]

All fluence levels are for $E_n \geq 1$ MeV.

absorbed the thermal neutrons which otherwise would have transmuted some of the tungsten to rhenium. Even minor additions of rhenium have been found to result in significant improvement in ductility for unirradiated tungsten and may also apply for irradiated tungsten. Figure 68 illustrates the increase in 0.05 percent yield strength as a function of fast fluence.[143] Since the slope of this increase is 0.46, it means that the yield strength of tungsten increases as a function of square root of fluence. The tensile tests at 1090 C indicate that not all of the irradiation-induced property changes in tungsten-26 wt % rhenium are annealed out by soaking at that temperature for 30 minutes before testing. Microhardness tests performed on irradiated tungsten indicate hardness increases with increased annealing temperature to a peak hardness at about 800 C, after which hardness decreases with higher annealing temperatures.[83] The effect of annealing on microhardness of irradiated tungsten is shown in Figure 69.

A considerable number of creep-rupture tests have been performed on tungsten. These tests were performed at temperatures ranging from 900 to 1900 C. Figure 70 shows the effect of irradiation on the creep-rupture ductility.[142] Figure 71 compares the rupture lives for unirradiated and irradiated specimens at various testing temperatures. At lower testing temperatures the creep rate is decreased by irradiation while at temperatures above 1700 C the creep rate is unaffected by irradiation. Generally the rupture life is increased by irradiation although the strain at rupture is decreased.

Neutron irradiation was found to decrease the creep rate of tungsten-26 wt % rhenium at temperatures of 800 to 1100 C as shown in Figure 72. However, the elongation at rupture was somewhat reduced as shown in Figure 73.

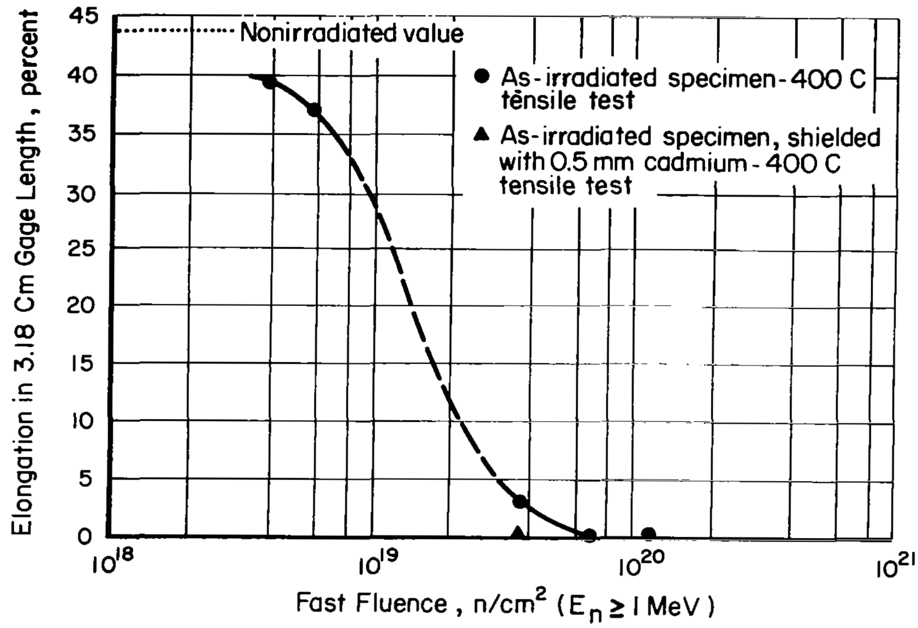

FIGURE 67. DUCTILITY VERSUS FAST NEUTRON FLUENCE FOR TUNGSTEN TENSILE SPECIMENS TESTED AT 400 C[143]

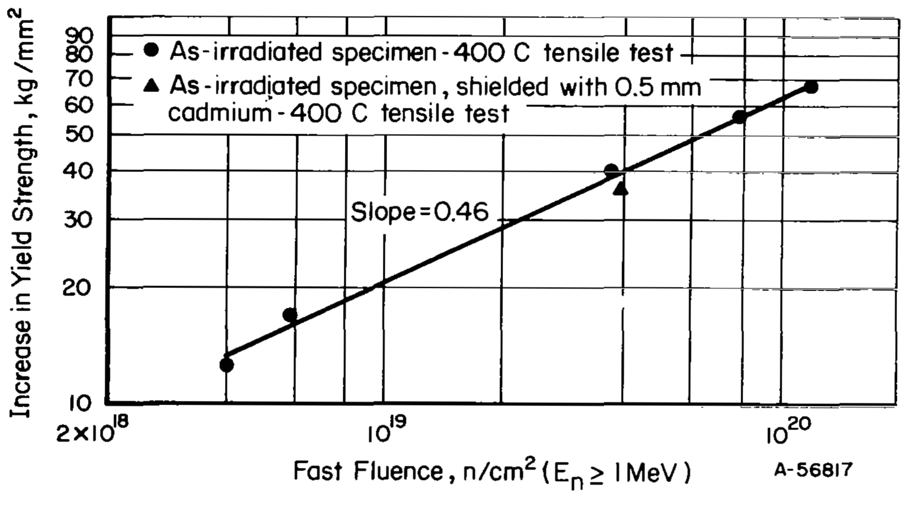

FIGURE 68. INCREASE IN 0.05 PERCENT YIELD STRENGTH VERSUS FAST NEUTRON FLUENCE FOR TUNGSTEN TENSILE SPECIMENS TESTED AT 400 C[143]

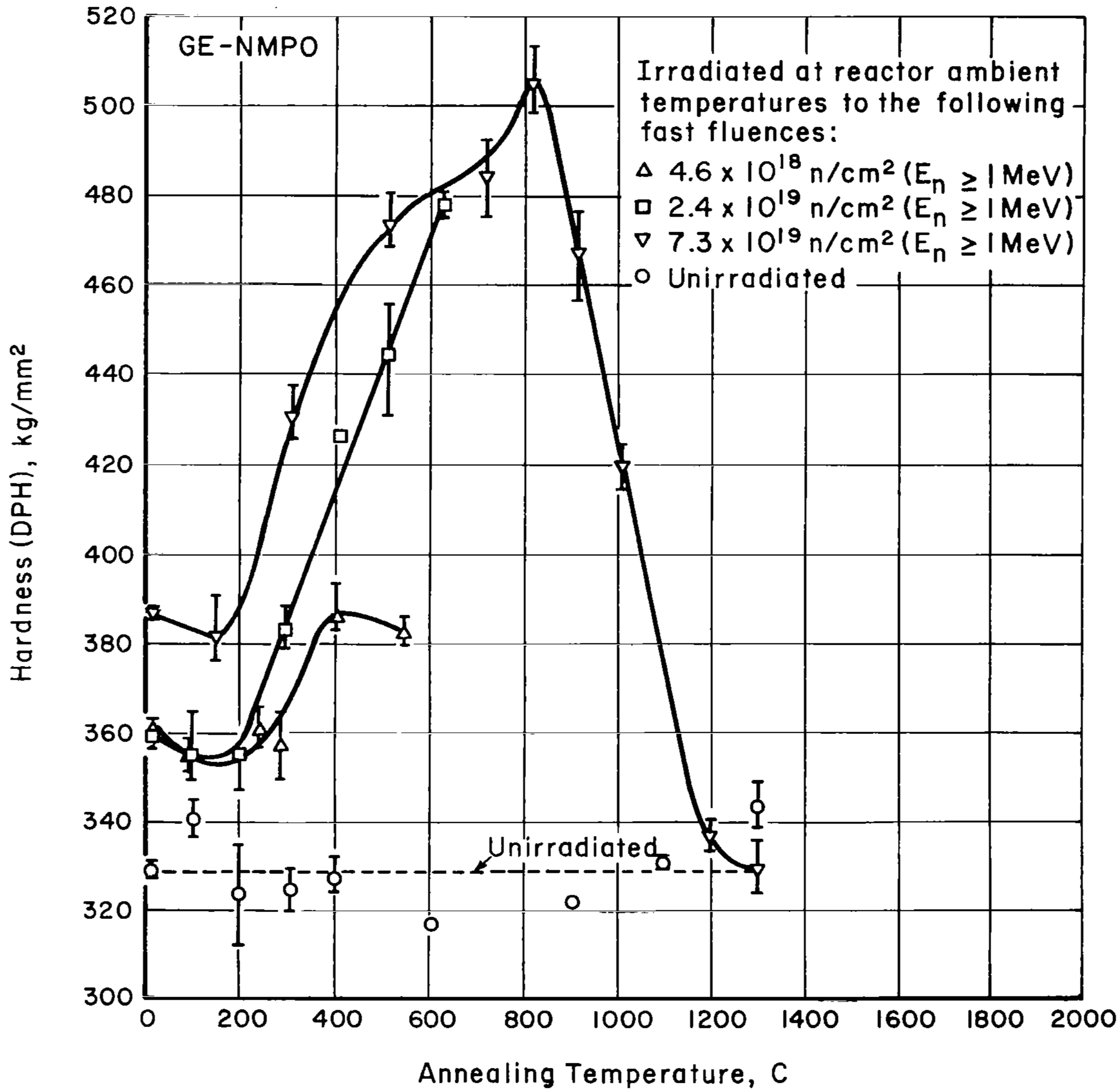

FIGURE 69. ROOM-TEMPERATURE HARDNESS OF IRRADIATED AND UNIRRADIATED SINGLE-CRYSTAL TUNGSTEN (411) AS A FUNCTION OF ANNEALING TEMPERATURE[83]

Specimens annealed for 1 hour at each temperature. Hardness measurements: 500-g load, 30 seconds.

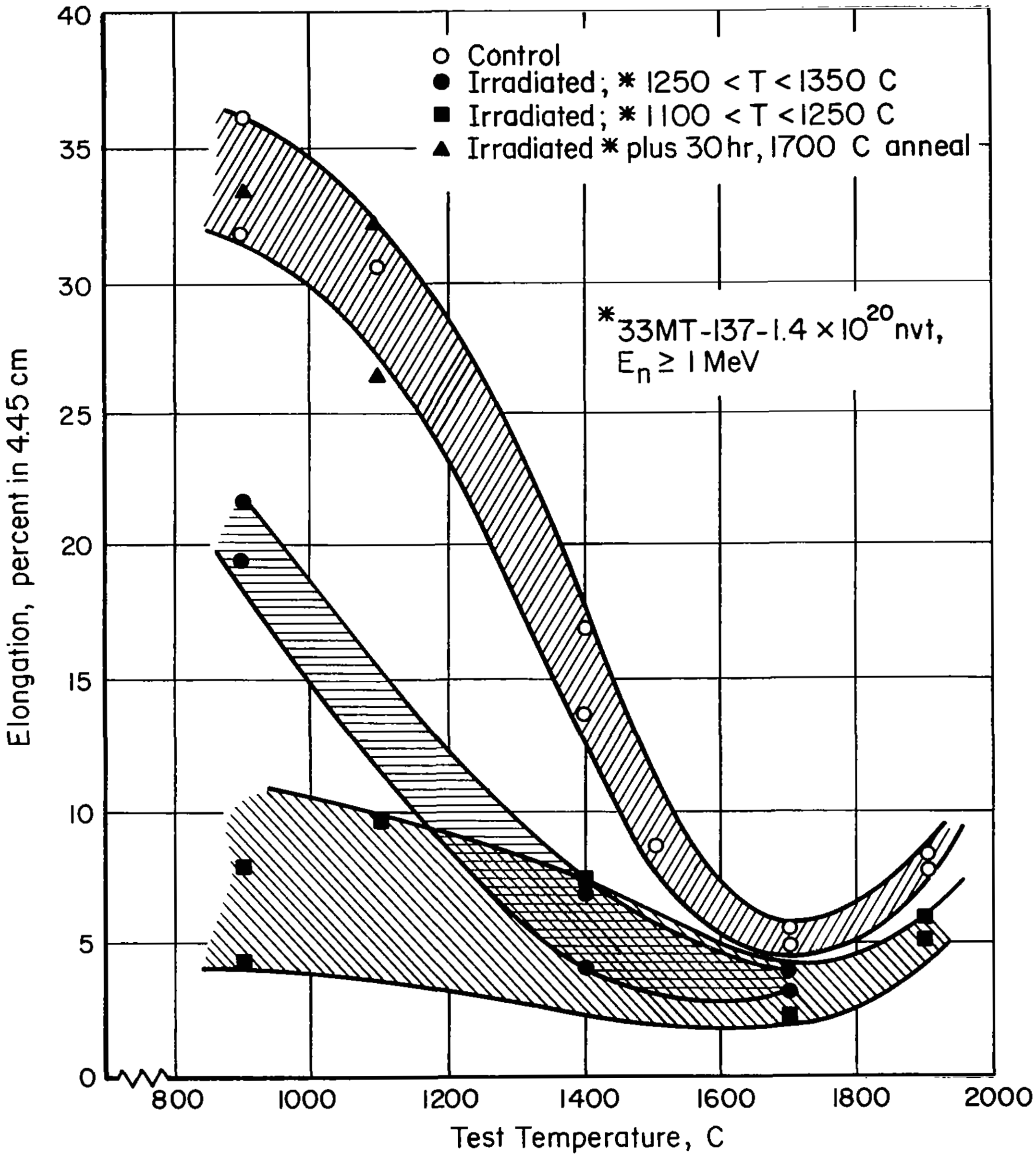

FIGURE 70. CREEP-RUPTURE DUCTILITY OF TUNGSTEN AS A FUNCTION OF TEST TEMPERATURE FOR SPECIMENS IRRADIATED AT ELEVATED TEMPERATURE AND CONTROL SPECIMENS (142)

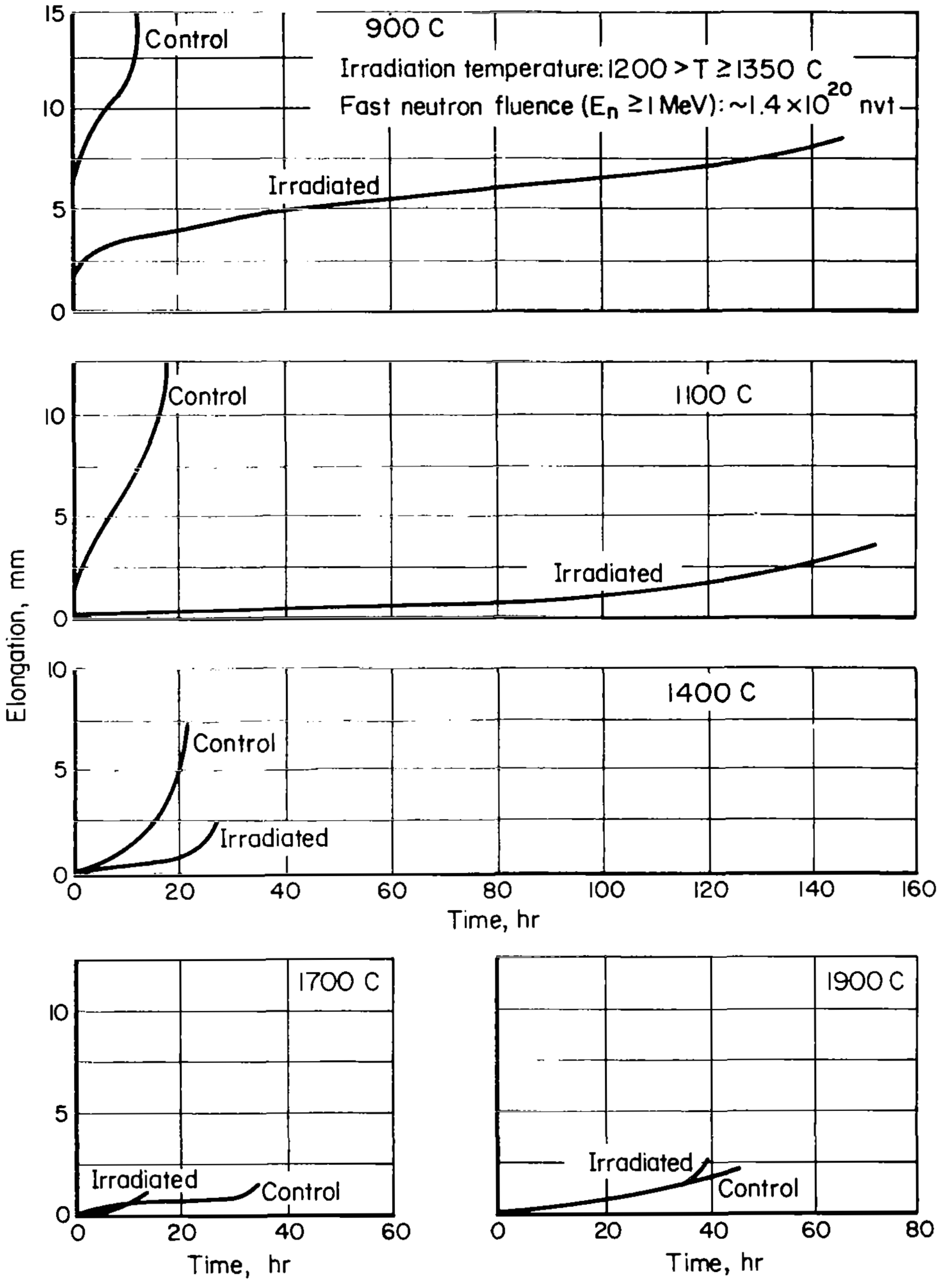

FIGURE 71. CREEP CURVES FOR CONTROL AND IRRADIATED POLYCRYSTALLINE TUNGSTEN SPECIMENS TESTED AT VARIOUS TEMPERATURE AND STRESS LEVELS (142)

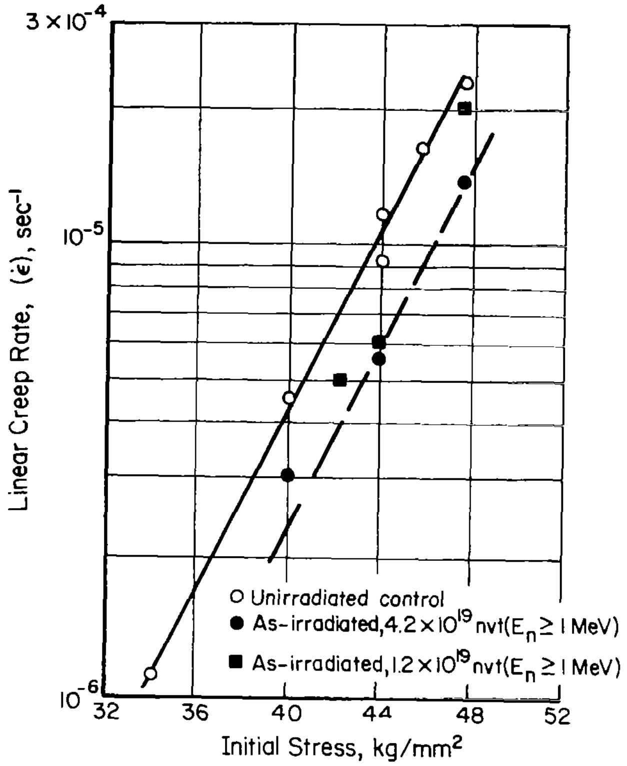

FIGURE 72. LINEAR CREEP RATES OF W-25Re SPECIMENS TESTED AT 1100 C

Irradiated at reactor ambient temperature to two fast (E > 1 MeV) neutron fluences.[144]

VANADIUM ALLOYS

Vanadium has recently received some attention as a possible cladding material for fast breeder reactors. Figure 74 illustrates that increasing fast fluence causes progressively higher strength and lower ductility.[145]

However,most interest has been on the elevated temperature tensile properties of various vanadium alloys. These tensile properties are summarized on Tables 38[146] and 39[147]. At low testing temperatures irradiation causes increases in strength and reductions in ductility. As the testing

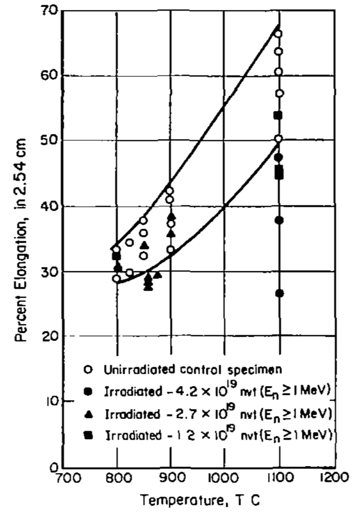

FIGURE 73. DUCTILITY VERSUS TEST TEMPERATURE FOR UNIRRADIATED AND IRRADIATED W-25RE CREEP-RUPTURE SPECIMENS

Irradiations conducted at reactor ambient temperature to a fast (E > 1 MeV) neutron fluence range of of 1.2 to 4.2 x 10^{19} cm^{-2}(144)

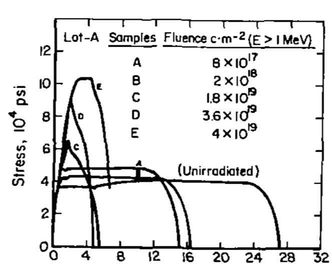

FIGURE 74. THE STRESS-STRAIN CURVES OBTAINED FROM ROOM TEMPERATURE TENSILE TESTS FOR VANADIUM AFTER IRRADIATION TO VARIOUS FLUENCES AT 107 C (145)

TABLE 38. EFFECT OF IRRADIATION AT 50 TO 100 C ON MECHANICAL PROPERTIES OF VANADIUM ALLOYS RECEIVING A FAST FLUENCE OF 1.4×10^{21} n/cm^2 (>0.1 MeV)(146)

Material	Test Temp., C	0.2% Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Total Elongation, percent	
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
V-3Ti	RT	63.1	108.5	79.6	109	17.1	1.5
	650	47.0	54.7	64.1	64.6	10.5	8.2
V-5Ti	RT	60.2	109	74.8	109	17.1	1.5
	650	43.0	52.3	65.2	64.6	10.5	8.4
V-10Ti	RT	67.7	124.5	81.3	124.5	21.1	1.5
	650	46.1	45.7	72.5	73.5	13.3	13.8
V-20Ti	RT	93.0	134.2	102.5	134	18.4	1.8
	650	57.0	55.8	80.5	79.1	15.7	15.8
V-5Ti-20Nb	RT	92.4	164.1	104.5	164	20.5	1.4
	650	69.2	76.0	102.3	101	15.3	12.8
V-10Ti-20Nb	RT	106.5	164	117	165	20.4	2.5
	650	67.4	69.2	96.5	91.5	13.5	14.5
V-20Ti-20Nb	RT	107.6	162.4	124.6	163.5	17.1	2.4
	650	72.1	76.0	96.5	94.5	13.7	11.7
V-10Ti-2Cr	RT	67.1	122.2	79.5	124	19.8	1.5
	200	57.0	105.0	70.7	106.5	16.3	1.8
	400	53.7	89.5	69.5	91.7	11.2	2.0
	650	43.1	48.5	72	78	17.8	15.8
	750	38.6	39.9	64.1	66.7	16.8	16
V-5Ti-5Cr	RT	58.9	118.5	71.4	119	21.8	1.4
	200	47.4	109.2	63.7	110	19.9	1.9
	400	41.2	111	61.5	113.2	14.4	2.3
	650	37.0	62.9	65.7	76.9	14.8	7.7
	750	39.4	36.9	58.9	63.1	14.8	12.7
V-10Ti-5Cr	RT	74.4	148	88.8	149	18.9	2.0
	200	63.4	118	77.8	112	15.6	3.3
	400	61.3	110	77.5	113.3	11.7	3.6
	650	48.2	53.8	79.9	85.2	16.3	15.8
	750	44.9	45.4	70.0	72.7	14.8	15.6
V-5Ti-5Mo	RT	69.9	117	96.5	117	14.5	1.7
	650	51.0	55.6	74.6	64.6	11.9	7.1
V-10Ti-5Mo	RT	73.7	128	85.5	122.2	18.5	1.8
	200		118		118.8		1.9
	400	56.5	105.5	72.5	109	14.3	4.3
	650	44.2	45.4	73	77.5	15	14.8
	750	40.9	40.9	67.8	66.5	16.3	16.8
V-5Ti-3Mo-2Cr	RT	58.9	113.4	73.5	114	21.8	1.3
	200	48.8	109.5	60.6	111.8	16.7	2.8
	400	36.0	89.5	55.8	93.0	16.3	4.3
	650	33.6	62.0	60.0	72.8	13.6	5.6
	750	38.6	40.9	59.2	61.5	9.7	11.0
V-5Ti-5Mo-5Nb	RT	70.7	126	85.8	127.8	19.3	2.3
	200	56.6	125	73.4	126	18.0	1.3
	400	52.4	92.5	67.0	96.5	11.2	4.1
	650	43.7	68.0	72.0	81.6	13.8	8.7
	750	44.9	43.7	66.2	67.2	12.2	13.0

TABLE 39. ROOM-TEMPERATURE TENSILE PROPERTIES OF VANADIUM-BASE ALLOYS IRRADIATED IN EBT-II(147)

Composition, wt %	Condition	Fast Fluence, 10^{21} n/cm^2	Irradiation Temperature, C	0.2% Offset Yield Strength, 1000 psi	Ultimate Tensile Strength, 1000 psi	Elongation, percent Uniform	Elongation, percent Total
V-20Ti	Unirradiated	0	--	89.8	107	20.0	25.8
V-20Ti	Unirradiated, 30 days--550 C	0	--	74.7	92.0	15.0	20.4
V-20Ti	Irradiated	1.8	540	70.3	86.1	15.8	22.5
V-20Ti	Irradiated	3.2	600	76.0	91.7	15.4	21.7
V-20Ti	Irradiated	4.8	650	82.0	95.7	14.2	19.9
V-20Ti	Irradiated	4.6	660	80.8	96.1	13.0	18.2
V-15Ti-7.5 Cr	Unirradiated	0	--	91.5	103	15.0	20.6
V-15Ti-7.5 Cr	Unirradiated, 30 days--550 C	0	--	95.5	108	15.0	19.9
V-15Ti-7.5 Cr	Irradiated	1.6	500	87.0	103	10.4	14.8
V-15Ti-7.5 Cr	Irradiated	3.2	570	82.5	97.9	10.0	14.2
V-15Ti-7.5 Cr	Irradiated	4.9	610	97.3	116	13.6	17.8
V-15Ti-7.5 Cr	Irradiated	4.6	630	93.1	111	13.6	18.3

temperature is increased these effects are removed and the preirradiation tensile properties are restored. Vanadium alloys do not exhibit the irradiation induced elevated temperature embrittlement which is prevalent in austenitic stainless steels and nickel base alloys.

Creep rates for unirradiated and irradiated vanadium-20 wt % titanium at 650 C are compared in Figure 75. These results indicate that at 650 C the minimum creep rate of the vanadium alloy is not affected by irradiation.[148]

Figure 76 compares the burst properties of unirradiated and irradiated vanadium-15 wt % titanium-7.5 wt % chromium alloy at room temperature.[149 Also shown on the figure are the fast fluence and irradiation temperature of the specimens. It appears that the lower irradiation temperatures cause appreciable increases in burst strength accompanied by decreases in tangential strain. Higher irradiation temperatures do not significantly affect the room temperature burst properties.

MATERIALS FOR CRYOGENIC APPLICATIONS

With the planned construction of nuclear rocket engines where liquid hydrogen is used as a propellant, some effort has been directed toward measuring the effects of fast-neutron irradiation on the mechanical properties of selected materials at liquid-hydrogen temperatures of 30 R (-257 C). The reactor-core temperatures are expected to be at about 4500 R (2200 C) and, therefore, the highest fast-neutron levels received by materials will be at elevated temperatures. However, since the shielding on the reactor will be minimal, some fast fluence can be expected in sections at cryogenic temperatures. The sections expected to be at cryogenic temperatures are the liquid-hydrogen storage tanks and the various system lines, pumps, valves, and the upper end of the fuel element tie rods.

In selecting materials for cryogenic applications, the prime considerations are strength, ductility, and toughness. By these criteria, the body-centered cubic materials which exhibit the ductile-to-brittle transition temperature are eliminated since this transition temperature occurs above liquid-hydrogen temperature for most body-centered cubic materials. Thus, the materials used at cryogenic temperatures are the face-centered cubic aluminum alloys, austenitic stainless steels, and nickel alloys. Since materials such as titanium, beryllium, and magnesium alloys have a hexagonal crystal structure, these have also been considered. The mechanical

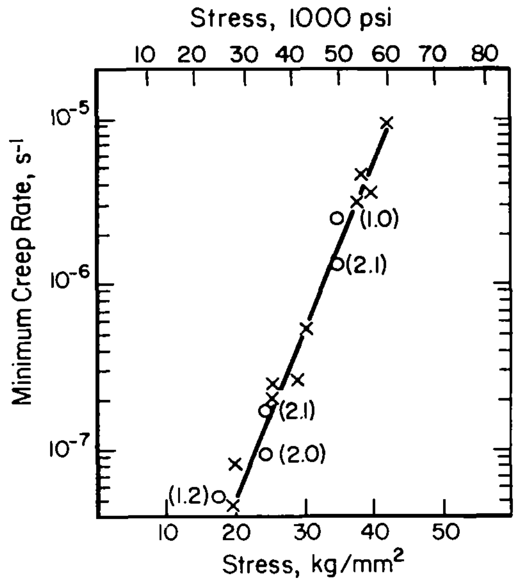

FIGURE 75. MINIMUM CREEP RATE OF UNIRRADIATED AND IRRADIATED V-20 TI ALLOY 34C AT 650 C

×, unirradiated; o, irradiated. Numbers in parentheses indicate the fluence in 10^{22} n/cm^2 (148)

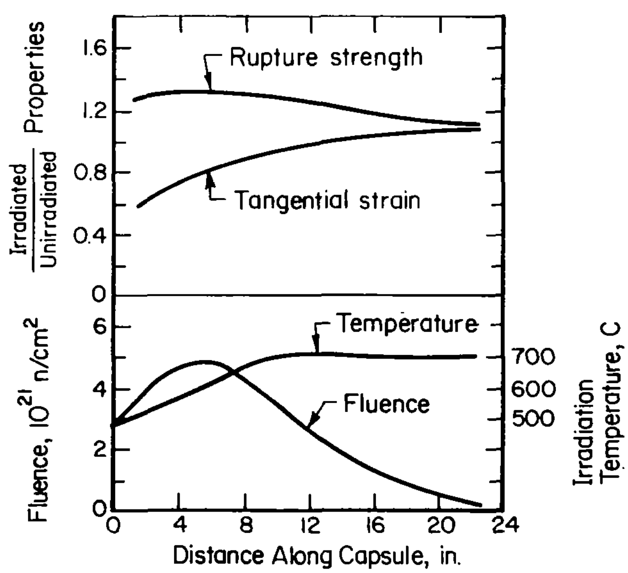

FIGURE 76. EFFECT OF IRRADIATION ON THE TUBE-BURST PROPERTIES OF V-15 Ti-7.5 Cr ALLOY (149).

properties of fcc materials at cryogenic temperatures differ from the room-temperature properties in that the yield strength, ultimate strength, and notch strength are higher at lower temperatures. Generally, ultimate strength increases to a larger proportion than does notch strength, and the notch strength/ultimate strength ratio decreases with decreasing temperature. The effect of low temperature on the ductility of face-centered cubic materials is not easily explained. In almost all cases, the reduction in area is decreased by lower temperatures, while the total elongation may be either increased or decreased. The elongation at cryogenic temperature depends on grain size, composition, degree of prior cold work, and temperature.

Results of tensile tests on various materials at cryogenic temperatures are presented in Table 40. Theoretically, irradiations at cryogenic temperatures would be expected to cause greater changes in mechanical properties than do irradiations at room temperature for the same levels of fast fluence. This is based on the supposition that some of the vacancies and interstitials produced by fast neutrons would be annealed out at room temperature while, at the cryogenic temperatures, minimal annealing of fast-neutron-caused defects would occur. Since these fast-neutron-caused defects prevent the movement of dislocations, an increase in yield strength and a decrease in ductility would be expected. Whether the ultimate strength is increased or decreased depends largely on the reduction of elongation by irradiation. Up to this time, the maximum irradiation levels at cryogenic temperatures have been limited to fast fluences of 1×10^{18} n/cm^2; this level may not be sufficient to change significantly the mechanical properties in many materials. It should be emphasized that the variations in unirradiated mechanical properties at cryogenic temperatures between two different heats of the same material were about the same as those in the irradiation-induced changes in mechanical properties. It becomes difficult to evaluate the data obtained because of the variations in the properties of specimens taken from the same heat as well as in specimens taken from different heats and the possible irradiation-induced changes in mechanical properties. Figures 77 and 78 illustrate the effect of increasing test temperature on the yield strength of irradiated beryllium and aluminum.(152) As the testing temperatures are increased the irradiation induced yield strength increase is removed and the preirradiation value is restored.

TABLE 40. THE EFFECT OF IRRADIATION AND CRYOGENIC TEMPERATURE ON THE TENSILE PROPERTIES OF SELECTED MATERIALS

Material	Condition	Fast Fluence, n/cm^2	Irradiation Temp., C	Test Temp., C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Notched Tensile Strength, 1000 psi		Total Elongation, percent		Reduction in Area, percent		Ref.
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
1099	H-14	1.0×10^{17}(a)	-257	-257	6.7	43.3	33.8	49.2	47.2	--	61.4	46.0	69.2	54.0	150
2014	T-651	1.0×10^{17}(b)	-257	-257	68.3	71.8	91.1	84.6	101.2		17.7	12.7	26.3	18.3	150
2024	T-351	1.0×10^{17}(b)	-257	-257	77.2	79.4	106.6	100.8	95.6		22.3	16.3	20.3	18.0	150
2219	T-87	1.0×10^{17}(a)	-257	-257	68.2	74.1	95.9	93.7	98.1		16.4	15.3	27.2	18.3	150
2219	S	7.5×10^{17}(c)	-196	-196	49.2	63.1	73.1	75.0	67.5	75.4	9.1	8.0			151
2219	S	7.5×10^{17}(c)	-196	-196 a	49.2	51.8	73.1	74.2	67.5	67.9	9.1	10.1			151
2219	S	7.5×10^{17}(c)	-196	27	42.0	41.5	59.7	58.4	56.5	53.5	7.0	6.8			151
2219	FA	7.5×10^{17}(c)	-196	-196	53.6	79.0	72.3	80.7	88.2	91.4	7.6	2.3			151
2219	FA	7.5×10^{17}(c)	-196	-196 a	53.6	54.6	72.3	72.1	88.2	88.1	7.6	6.7			151
2219	FA	7.5×10^{17}(c)	-196	27	47.3	47.2	62.1	61.4	75.5	72.2	7.4	6.6			151
2219	FR	7.5×10^{17}(c)	-196	-196	51.7	78.3	74.8	80.5	93.0	105.4	14.3	5.6			151
2219	FR	7.5×10^{17}(c)	-196	-196 a	51.7	53.5	74.8	74.7	93.0	92.3	14.3	13.8			151
2219	FR	7.5×10^{17}(c)	-196	27	46.3	46.8	61.8	61.7	77.3	77.7	10.9	11.4			151
5086	H-32	1.0×10^{17}(a)	-257	-257	36.2	59.5	92.0	94.2	68.4		30.0	22.3	25.0	20.0	150
5456	H-321	1.0×10^{17}(a)	-257	-257	43.9	66.7	92.2	93.2	66.8		18.2	14.0	16.8	16.5	150
6061	T6-L	5.0×10^{16}(c)	-257	-257	46.5	55.0	64.7	69.0	61.4	65.8	24.1	27.5	34.4	32.6	151
6061	T6-TW	5.0×10^{16}(c)	-257	-257	47.4	57.6	67.3	71.6	56.5	65.2	9.0	11.2	16.4	21.5	151
6061	T6-LW	5.0×10^{16}(c)	-257	-257	56.9	52.6	68.9	66.7	57.2	67.6	6.0	10.2	11.9	10.8	151
6061	T-6	1.0×10^{17}(a)	-257	-257	50.4	59.4	68.1	64.6	73.0		30.0	30.0	41.4	34.0	150
7075	T6-L	5.0×10^{16}(c)	-257	-257	99.1	104.3	110.4	113.4	70.5	75.5	6.4	5.9	7.0	7.9	151
7079	T-6	1.0×10^{17}(a)	-257	-257	129.8	127.7	145.2	133.7	151.8		5.8	5.3	5.8	4.3	150
7178	T-651	1.0×10^{17}(a)	-257	-257	108	121.3	129	134.3	128.2		12.4	6.3	13.0	4.3	150
A-356	T6-L	5.0×10^{16}(c)	-257	-257	31.0	42.8	44.7	52.8	39.7	43.2	1.5	1.4	2.8	0.8	151
A-356		1.0×10^{17}(a)	-257	-257	37.6	46.1	64.5	62.1	65.9		11.6	8.7	9.4	8.0	150
B-750		1.0×10^{17}(a)	-257	-257	25.2	43.5	42.6	46.2	31.1		7.0	3.3	4.0	1.0	150
X-250	T-4	1.0×10^{17}(a)	-257	-257	49.0	(a)	49.5	46.9	52.8		nil	nil	nil	nil	150
55 A		1.0×10^{17}(a)	-257	-257	122	131.7	169.4	192.3	167.2	187.3	33.3	34	53	53	150
Ti-5Al-2.5 Sn	T	5.0×10^{16}(c)	-257	-257	199.5	205.1	216.4	226.5	171.0	153.7	17.1	18.8	17.8	16.8	151
Ti-5Al-2.5 Sn	TW	5.0×10^{16}(c)	-257	-257	198.6	204.2	214.6	217.1	118.1	121.2	5.9	6.1	9.4	11.4	151
Ti-5Al-2.5 Sn	(Std 1)	1.0×10^{17}(a)	-257	-257	205.3	218	224.8	239	249	273	13.8	11.5	30	36	150
Ti-5Al-2.5 Sn	Eli	1.0×10^{17}(a)	-257	-257	214.2	213	228.4	223.3	267.2	270	9.7	11.0	32.3	31	150
Ti-5Al-2.5 Sn	B	6.0×10^{17}(c)	-196	-196	177.6	195.0	183.3	197.5	245.0	227.1	16.3	4.6			151
Ti-5Al-2.5 Sn	B	6.0×10^{17}(c)	-196	-196 a	177.6	188.8	183.3	191.8	245.0	228.5	16.3	9.5			151
Ti-5Al-2.5 Sn	B	6.0×10^{17}(c)	-196	27	119.2	128.6	121.0	129.0	189.3	193.2	16.5	12.8			151
Ti-6Al-4 V	A	1.0×10^{17}(a)	-257	-257	243.2	254	260.4	273.7	281.6	283.7	7.6	5.7	30.4	37.3	150
Ti-6Al-4 V	Aged	1.0×10^{17}(a)	-257	-257	275	293.3	282.2	302.3	288.8	295	6.4	5.0	25.4	22.5	150
Ti-8Al-1Mo-1V		1.0×10^{17}(a)	-257	-257	224.2	242.7	239	261.7	267.2	282	--	5.7	--	29.0	150
Be		2.4×10^{20}(c)	300-500	-196		(d)		64.8				0.1		0.2	20
Be		2.4×10^{20}(c)	300-500	-72		(d)		61.5				0.1		0.2	20
AISI-301	S	5.1×10^{17}(c)	-196	-196	165.2	176.1	292.8	290.4	207.3	214.3	20.4	20.3			151
AISI-301	S	5.1×10^{17}(c)	-196	-196 a	165.2	173.7	292.8	294.3	207.3	216.7	20.4	20.0			151
AISI-301	S	5.1×10^{17}(c)	-196	27	151.0	156.3	186.9	188.8	191.2	193.8	19.2	17.4			151
AISI-303 Se	B	6.5×10^{17}(c)	-196	-196	85.4	115.0	165.7	173.0	191.0	228.2	44.8	34.3			151
AISI-303 Se	B	6.5×10^{17}(c)	-196	-196 a	85.4	104.4	165.7	172.4	191.0	208.5	44.8	38.7			151
AISI-303 Se	B	6.5×10^{17}(c)	-196	27	49.1	60.8	92.7	96.0	121.6	131.4	31.2	26.0			151
AISI-304		1.0×10^{17}(a)	-257	-257	39.6	51.9	242	260.3	193.4	173.3	34.4	33.5	38.4	28.5	150
AISI-310		1.0×10^{17}(a)	-257	-257	137.8	135.7	212.2	218.7	188.6	218	42.8	48.3	56.2	21.7	150
AISI-347-(b)	L	5.0×10^{16}(c)	-257	-257	87.4	84.5	259.9	227.1	139.4	135.2	40.2	41.0	28.0	27.9	151
AISI-347-(b)	LW	5.0×10^{16}(c)	-257	-257	97.0	63.0	224.2	229.5	136.5	126.8	27.1	30.1	20.8	23.6	151
AISI-347-(b)	L	5.0×10^{16}(c)	-257	-257	83.0	80.2	117.3	115.4	99.2	102.3	7.7	10.0	8.8	6.4	151

TABLE 40. (Continued)

Material	Condition	Fast Fluence, n/cm^2	Irradiation Temp., C	Test Temp., C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Notched Tensile Strength, 1000 psi		Total Elongation, percent		Reduction in Area, percent		Ref.
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
AISI-347		1.0×10^{17}(a)	-257	-257	51.9	64.0	237.4	250.7	214.2	238	41.3	37.0	43.8	55.0	150
AISI-440 C		1.0×10^{17}(a)	-257	-257	(d)	(d)	261.2	217.7	107.7	118.7	nil	nil	nil	nil	150
A-286	L	5.0×10^{16}(c)	-257	-257	135.9	137.0	223.6	219.7	185.8	188.6	37.5	30.4	31.0	32.2	151
A-286*		1.0×10^{17}(a)	-257	-257	149.6	152	235	229	216.4	245.7	34.5	33.5	46.5	16	150
A-286*		1.0×10^{17}(a)	-257	-257	156.8	165.7	238.2	238	255	239	31.8	31.0	42.2	39.7	150
AM-350		1.0×10^{17}(a)	-257	-257	332.3	308.3	340.6	313.3	268.4	148.3	11.0	7.3	36.0	20.7	150
A-353		1.0×10^{17}(a)	-257	-257	174.8	190.0	201.6	215.7	189.6		18.0	15.7	39.6	21.3	150
T-450		1.0×10^{17}(a)	-257	-257	91.5	92.3	197.2	192.7	214.4		31.2	30.3	27.2	29.7	150
17-7 PH		1.0×10^{17}(a)	-257	-257	330.0	250.7	335.4	251.3	182.6		nil	nil	nil	nil	150
René 41		1.0×10^{17}(a)	-257	-257	107.6	113.3	194.4	194.7	204.8	194	60.5	55	50.5	48	150
K Monel		1.0×10^{17}(a)	-257	-257	121.4	138.7	187.2	188.3	210	211.3	32	33	54.8	51.3	150
Inconel		1.0×10^{17}(a)	-257	-257	175.8	179	186.4	191	222.4	237.3	20	25	56	51	150
Inconel X		1.0×10^{17}(a)	-257	-257	150.8	161.3	243.2	241	249.8	230.3	33	29	45.6	37.3	150
Inconel 713 C	L	5.0×10^{16}(c)	-257	-257	108.0	133.2	111.6	135.3	127.7	135.4	3.0	0.8	7.0	1.7	151
Inconel 718	B	1.0×10^{18}(c)	-196	-196	206.7	235.2	267.9	271.6	312.6	342.0	12.5	8.4			151
Inconel 718	B	1.0×10^{18}(c)	-196	-196 a	206.7	222.5	267.9	273.6	312.6	338.2	12.5	12.7			151
Inconel 718	B	1.0×10^{18}(c)	-196	27	179.5	188.8	217.8	214.8	280.9	290.5	10.9	11.5			151
Inconel 718	S	7.0×10^{17}(c)	-196	-196	212.4	232.1	270.0	269.6	242.6	259.9	12.6	12.3			151
Inconel 718	S	7.0×10^{17}(c)	-196	-196 a	212.4	217.4	270.0	267.4	242.6	252.4	12.6	13.3			151
Inconel 718	S	7.0×10^{17}(c)	-196	27	184.9	188.2	218.7	213.7	219.2	218.7	11.3	11.8			151
Inconel 718	WS	6.0×10^{17}(c)	-196	-196	195.7	211.6	206.3	211.6	162.0	165.8	1.0	0.9			151
Inconel 718	WS	6.0×10^{17}(c)	-196	-196 a	195.7	205.8	206.3	214.2	162.0	165.0	1.0	1.0			151
Inconel 718	WS	6.0×10^{17}(c)	-196	27	162.6	167.8	173.3	168.9	148.2	149.6	1.0	1.5			151
Inconel X-750	L	5.0×10^{16}(c)	-257	-257	135.8	146.8	253.3	227.6	191.6	197.2	31.2	33.4	27.1	26.6	151
Inconel X-750	B	9.0×10^{17}(c)	-196	-196	119.9	166.5	210.5	209.2	244.3	299.3	24.9	19.1			151
Inconel X-750	B	9.0×10^{17}(c)	-196	-196 a	119.9	151.6	210.5	213.8	244.3	280.3	24.9	21.1			151
Inconel X-750	B	9.0×10^{17}(c)	-196	27	105.4	131.0	169.1	166.8	217.6	241.8	16.3	16.1			151
Inconel X-750	S	5.5×10^{17}(c)	-196	-196	125.1	170.1	210.1	213.0	182.7	217.8	27.0	21.9			151
Inconel X-750	S	5.5×10^{17}(c)	-196	-196 a	125.1	153.2	210.1	213.2	182.7	205.1	27.0	23.9			151
Inconel X-750	S	5.5×10^{17}(c)	-196	27	107.7	129.4	166.4	164.9	155.4	171.3	18.8	18.3			151
Hastelloy-C	L	5.0×10^{16}(c)	-257	-257	111.0	111.5	185.7	185.4	--	156.3	39.2	50.4	32.6	32.8	151

(a) > 0.5 MeV.
(b) > 0.33 MeV.
(c) > 1 MeV.
(d) Failed at less than 0.2% plastic strain.

L = longitudinal direction.
T = transverse direction.
W = welded.
* = different heats.
T6 = specifies heat treatment.
H = hardened.
S = sheet.
FA = forging, axial direction.
FR = forging, radial direction.
B = bar.

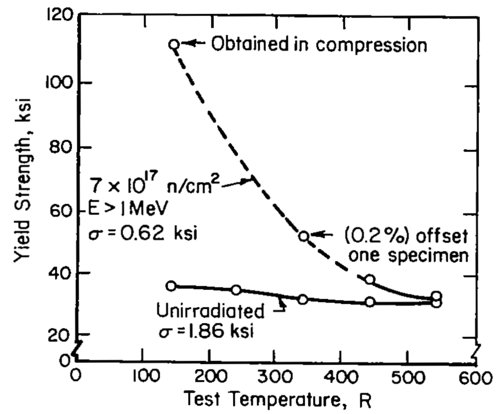

FIGURE 77. EFFECT OF IRRADIATION (7 x 10^{17} N/CM^2) ON THE 0.2 PERCENT OFFSET YIELD STRENGTH OF BERYLLIUM(152)

Irradiation temperature = 140 R – average of 4 specimens per point.

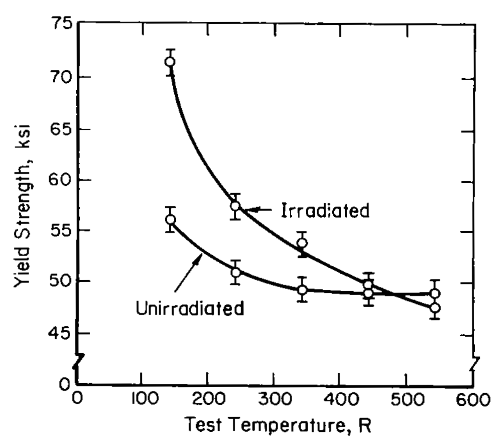

FIGURE 78. EFFECT OF IRRADIATION (8 x 10^{17} N/CM^2) ON 0.2 PERCENT OFFSET YIELD STRENGTH OF 2219 ALLUMINUM ALLOY(152)

Irradiation temperature = 140 R – average of 3 tests per point with 1 σ band.

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