THE EFFECTS OF NEUTRON RADIATION ON STRUCTURAL MATERIALS

REIC Report No. 45

RADIATION EFFECTS INFORMATION CENTER
Battelle Memorial Institute
Columbus Laboratories
Columbus, Ohio 43201

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REPORT

on

THE EFFECTS OF NEUTRON RADIATION ON STRUCTURAL MATERIALS

by

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to

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

NASW-1568

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ABSTRACT

A compilation of the data available on the effects of radiation on tensile, creep, fatigue, impact, and hardness properties of various structural materials is presented. These properties are given as a function of test temperature, irradiation temperature, and radiation fluence. Specifically the following reactor materials are covered: (1) aluminum alloys, (2) magnesium alloys, (3) beryllium, (4) zirconium alloys, (5) mild steels, (6) stainless steels, (7) nickel alloys, and (8) refractory metals. Data on the effects of radiation on the mechanical properties of selected materials at cryogenic temperatures also are included.

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FOREWORD

This report consists of a compilation of the data on radiation effects on structural materials that have been generated since the publishing of REIC Report No. 20 (September 15, 1961). The objective of the report is to provide information and significant supporting data regarding effects of radiation on structural metals, and not to summarize all the data which have been generated during the last 6 years. Before proceeding with a discussion of the effects of radiation on specific metals and alloys it is necessary to clarify some of the terms used in the report and to provide some general guide lines for interpretation of the test results.

(1) Dosimetry. The word fluence is used in this report instead of descriptions such as neutron accumulation, neutron dose, and neutron exposure which were previously used in the literature. Fluence means time integrated neutron flux and its units are n/cm^2. It can be applied to thermal, epithermal, or fast neutrons. The neutron energy should be specified. The units nvt may be used for thermal neutrons. The use of the word fluence is specifically recommended by ASTM Subcommittee E-10 on Radioisotopes and Radiation Effects. In this report, the fluences are generally for fast neutrons. Almost all investigators report their fluence in terms of fast neutrons with energies greater than 1 MeV, although some investigators use 0.1 MeV, 0.5 MeV, or fission neutrons (~3 MeV). In this report, a fast neutron is taken as a neutron having an energy larger than 1 MeV unless otherwise stated.

In recent years, numerous attempts have been made to measure radiation damage in terms other than >1 MeV neutrons because the present method does not take into account neutrons with energies of less than 1 MeV and does not differentiate between neutrons with energies of 1, 2, or 3 MeV. Radiation-induced changes in mechanical properties are a function of the number of atoms that have been displaced from equilibrium lattice sites. Since energies of only about 6 to 30 eV are required to displace an atom from the lattice site, significant numbers of displacements are caused by neutrons having energies of less than 1 MeV. All of the proposed alternative methods of measuring radiation exposure utilize the number of displaced atoms that would be expected from the total energy of all the neutrons. This means that 10 neutrons with energies of 0.1 MeV would have approximately the same effect as 1 neutron with an energy of 1 MeV. Use of the total energy relationships usually results in better correlation with radiation-effects data than does the enumeration of neutrons having energies greater than 1 MeV. These total energy relationships are quite applicable when irradiations from different reactors are compared and are especially applicable when the results from a mixed thermal, epithermal, and fast-neutron spectrum are compared with a predominantly fast-neutron spectrum, as in a fast reactor. However, none of the total energy relationships have gained wide acceptance among the various investigators. This nonacceptance can probably be attributed to the following:

(a) Although the various total energy relationships agree in principle, differences do exist in the handling and subsequent interpretation of data

(b) The flux spectrum is not well known for all irradiation experiments

(c) The handling of these total energy relationships is somewhat more complicated and time consuming than just reporting fluence numbers as obtained from dosimetry wires

(d) The resistance which normally is encountered in changing from an old method to a new method.

(2) Mechanical Properties. The variables that affect mechanical properties of the irradiated material are the composition of the material, grain size, degree of cold work, fast fluence, energy spectrum of the radiation, irradiation temperature, testing temperature, and other variables such as time at temperature and strain rate.

The reader is cautioned against direct comparisons of experimental results from different laboratories and from different countries. This pertains mostly to elongation measurements where the term ductility has been used variously to apply to total elongation, uniform elongation, change of tube diameter in burst tests, reduction in area, and nonuniform elongation. Also, quite often, no information is given concerning the method by which these measurements were made (e.g., extensometer, head travel of tensile machine, or pre- and posttest measurement of a scribe mark on the specimen). In elevated-temperature tensile tests, the strain rate becomes quite important, but often is not specified by the tester. Therefore, for the most meaningful results, it is recommended that the reader compare the relative changes in properties as obtained from the irradiated specimens and the preirradiation control specimens. Another consideration is the importance of using control specimens to separate purely thermal effects from irradiation effects. This is especially important for complex alloys such as the precipitation and age-hardenable nickel-base alloys.

SUMMARY

A summary of the effects of radiation on the mechanical properties of various metals and alloys is presented below.

Light Metals

Only minor changes in room-temperature tensile properties of aluminum and magnesium are caused by radiation. However, at very high fluences, significant increases in strength and reductions in ductility take place. At irradiation temperatures above 115 C, no changes in mechanical properties of aluminum alloys are caused by radiation and a 30-minute anneal at 200 C is sufficient to remove the radiation damage resulting from irradiation at lower temperatures. No tensile tests of significance have been performed on irradiated titanium alloys.

Beryllium

Significant amounts of helium and hydrogen are produced in beryllium by neutron irradiation. This causes swelling of beryllium at high temperatures and high fast fluences. At low temperatures, the yield strength and ultimate strength of beryllium are significantly increased by irradiation and the ductility is decreased. These radiation-induced property changes persist up to testing temperatures between 600 and 700 C. At these temperatures, the preirradiation properties are recovered, except for some permanent loss in ductility; swelling also begins to take place at these temperatures. The postirradiation mechanical-property data are conflicting for irradiations at 600 C and above. Certain test results have shown substantial reductions in elongation after irradiation, whereas other results have shown practically no change in elongation. Differences in the postirradiation tensile properties also exist, but these are less significant. The mechanical properties of warm-extruded beryllium generally have been shown to be affected less by irradiation than are those of hot-extruded beryllium. However, it is noted that there is considerable variation in the mechanical properties among different batches of beryllium. Therefore, a careful comparison must be made of pre- and postirradiation properties among batches of beryllium.

Zirconium Alloys

Radiation causes significant increases in the yield and ultimate strengths of annealed zirconium alloys and reduces both the uniform and total elongations. These radiation-induced property changes appear to be relatively more severe at higher irradiation temperatures (~300 C), and saturation of the property changes has not been reached after fast fluences of 1.5×10^{21} n/cm^2. Irradiation saturation at the lower irradiation temperature (~50 C) is not reached after a fast fluence of 2×10^{22} n/cm^2. For highly cold-worked Zircaloy-2, a saturation in irradiation effects is reached after about 10^{20} n/cm^2, after which a decrease in strength and an increase in elongation occur.

Recovery of radiation-induced property changes appears to take place between 350 and 500 C. The stronger alloys such as the zirconium-2.5 wt % niobium alloy require higher temperatures to induce recovery. The creep rate of Zircaloy-2 appears to be increased by irradiation at 300 C, while the creep rate of zirconium-2.5 wt % niobium is not affected by irradiation.

The mechanical properties of zirconium may be affected significantly by the presence of hydride platelets, particularly if the platelets are orientated in a direction perpendicular to the stress axis. These platelets, formed as a result of hydrogen pickup from water and/or steam environments, are especially embrittling at high strain rates and low (<200 C) temperatures.

Ferritic and Martensitic Steels

The most significant radiation-induced change in the mechanical properties of ferritic steels is the change or shift in the ductile to brittle transition temperature (nil ductility transition temperature or NDT). The magnitude of the radiation-induced increase in NDT increases with increasing fast fluence and decreases with higher irradiation temperatures. However, large variations in the magnitude of the NDT change have been encountered for different heats of steel having approximately the same nominal compositions. It has been found that the smallest radiation-induced increases in NDT take place in steels that are low in interstitial elements (carbon, oxygen, nitrogen), have a small grain size, and possess a quenched and tempered microstructure. Postirradiation annealing at 400 to 450 C for 1 hour results in almost complete restoration of the preirradiation ductile-to-brittle transition temperature as well as restoration of the preirradiation tensile properties.

At low testing temperatures, irradiation increases the yield and ultimate strengths and decreases the ductility of the ferritic steels. However, the ferritic stainless steels do not exhibit the radiation-induced elevated-temperature embrittlement that is common to the austenitic stainless steels and nickel alloys discussed below. Annealing irradiated AISI Type 406 stainless steel for 1 hour at 650 C results in complete recovery of the preirradiation tensile properties.

Limited tensile tests have been performed on irradiated martensitic stainless steels. Irradiation increases the room-temperature yield and ultimate strength of martensitic stainless steels and decreases the ductility.

Austenitic Stainless Steels

The maximum hardening of austenitic stainless steels by irradiation is produced at temperatures of about 300 C. The radiation hardening causes increases in the room temperature yield and ultimate strength and is accompanied by reductions in ductility. These changes in tensile properties can be removed by annealing at 500 to 600 C.

A drastic reduction in the ductility of irradiated austenitic stainless steels takes place at elevated temperatures. This ductility loss at elevated temperatures, which is presently attributed to helium bubbles formed at grain boundaries, cannot be recovered by annealing at temperatures up to 1350 C. Helium is presumed present as a result of the thermal-neutron $^{10}B(n,\alpha)^{7}Li$ reaction as well as from (n,α) reactions between fast neutrons and iron, nickel, and other alloy constituents. The magnitude of the ductility loss is increased by large grain size and low strain rates. Correlations of the ductility losses at elevated temperatures with helium content have been partially successful, as have attempts to improve the postirradiation ductility at elevated temperatures by titanium additions and placement of carbide particles near the grain boundaries. It has not been definitely established whether a predominantly fast flux affects the mechanical properties in a similar manner to that of a mixed fast and thermal flux at equivalent fast fluences.

The effect of radiation on the elevated-temperature creep of stainless steel has not been firmly established. Some investigators have found that radiation increases the creep rate, while other investigators have found that the creep rate is not changed by radiation. However, there is agreement that the elongation at failure, in stress-rupture and creep tests, is reduced by radiation.

Nickel-Base Alloys

Radiation affects the mechanical properties of nickel-base alloys in about the same way as it does those of austenitic stainless steels. At low temperatures the yield strength and ultimate strength are increased and the ductility is decreased. These radiation-induced changes in the low-temperature mechanical properties can be removed by annealing. The nickel-base alloys also undergo radiation-induced embrittlement at elevated temperatures. The embrittlement of nickel-base alloys is also attributed to helium bubble formation at the grain boundaries. The embrittlement is significantly increased by low strain rates and appears to increase with increasing nickel content. The creep rate of nickel alloys is not affected by radiation, but the rupture life is significantly reduced because of the reduction in ductility. Additions of titanium and zirconium to some nickel-base alloys have resulted in improved postirradiation ductility at elevated temperatures. Irradiation at elevated temperatures causes a large reduction in strength when compared with thermal control specimens. It appears that overaging is promoted by irradiation.

Materials for Cryogenic Applications

Irradiation and testing of selected aluminum alloys, titanium alloys, nickel alloys, and stainless steels has been performed at cryogenic temperatures. Radiation generally causes increases in yield and ultimate strength and reductions of ductility at cryogenic temperatures. However, the rather low fast fluences (1×10^{18} n/cm^2) to which the specimens have been exposed and the difficulties of testing at cryogenic temperatures make the measurement of these property changes somewhat difficult. Annealing at room temperature was found to cause restoration of the preirradiation tensile properties at -196 C for most alloys.

Refractory Metals

A limited number of tensile and stress-rupture tests have been performed on irradiated niobium, molybdenum, rhenium, tungsten, and vanadium alloys. The magnitude of radiation effects on the mechanical properties of these materials was found to be very sensitive to the interstitial content of the materials. Increased interstitial content resulted in larger 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.

LIGHT METALS

The effect of neutron radiation on the light metals aluminum, magnesium, and titanium is discussed in this section. The discussion centers around an environment of room temperature or above. The effects of radiation on aluminum, magnesium, and titanium at cryogenic temperatures are discussed on pages 199 through 206.

Both aluminum and magnesium are attractive for use as structural materials in thermal reactors because of their low cross section for thermal neutrons. However, both aluminum and magnesium are limited in their usage because of their low melting points and consequent lack of strength at elevated temperatures. Aluminum has been used as a fuel-element cladding and structural material in water-cooled reactors 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 fuel material since aluminum is a very reactive metal. Magnesium alloys have been used as fuel-element cladding material for gas-cooled reactors which operate in the 250 to 350 C temperature range. Titanium, however, has not been used in nuclear applications because of its high cross section for thermal neutrons. The chemical compositions of the aluminum, magnesium, and titanium alloys that were irradiated are given in Table 1. Results of tensile tests on these alloys in the irradiated condition are given in Table 2.

Aluminum Alloys

Test results indicate that irradiation increases the strength and decreases the ductility of aluminum at room temperature. These changes appear to be a function of fast fluence as shown in Figure 1.[10] No saturation limit in irradiation effects with increasing fluence has been found for these specimens.

Not included in Table 2 are the results of tensile tests conducted as part of an extensive irradiation program on 6061 aluminum alloy.[11] Sheet specimens from this alloy were irradiated at temperatures ranging from 43 to 485 C, with the maximum fast fluence being 10^{19} n/cm^2. Tensile tests on specimens irradiated at above 115 C indicated that no changes in mechanical properties resulted. The postirradiation tensile tests were performed at temperatures from room temperature to 485 C. Specimens in the annealed condition that were irradiated at 43 C to a fast fluence of 10^{18} n/cm^2 showed some changes in mechanical properties. The room-temperature tensile tests indicated a 10 percent increase in yield strength, a 4.5 percent increase in ultimate strength, and a 20 percent reduction in elongation. Annealing at 200 C for 30 minutes resulted in complete restoration of the preirradiation mechanical properties. In the cold-worked condition (T-6), irradiation did not affect the yield strength, but increased the tensile strength 10 percent and elongation 50 percent. The improvement in ductility indicates that some of the cold work was annealed out during irradiation.

The most severe radiation-induced changes in mechanical properties of aluminum occurred in aluminum-capsule bodies which were irradiated in the 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 2. The combined irradiation and corrosion caused a fivefold increase in tensile strength and a 90 percent reduction in elongation.[2]

TABLE 1. COMPOSITION OF ALUMINUM, MAGNESIUM, AND TITANIUM ALLOYS

Alloy Designation	Alloy Composition, percent										
	Al	Mg	Ti	Cu	Fe	Mn	Si	Zn	Cr	Zr	Other
1100	99.0[a]			0.20	0.5	0.05	0.5	0.10			0.15
2024	Bal	1.2-1.8		3.8-4.9	0.5	0.3-0.9	0.5	0.25	0.10		
6061	Bal	0.8-1.2	0.15	0.15-0.40	0.7	0.15	0.4-0.8	0.25	0.15-0.35		0.15
6063	Bal	0.45-0.9	0.10	0.10	0.35	0.10	0.2-0.6	0.10	0.10		0.15
A-356	Bal	0.2-0.4	0.20	0.20	0.60	0.10	6.5-7.5	0.10			
M-257	Bal			(6-8 wt % Al_2O_3)							
M-486[b]	Bal		0.20		8.0				0.20	0.2	
X-2219[c]	Bal			6.2	0.19	0.32	0.11				
X-8001[d]	97.5[a]			0.15	0.45-0.70		0.17				0.15
HK-31A[e]		Bal								0.5-1.0	0.30
A-40[f]			Bal							0.7	0.08
A-110AT[g]	5.0		Bal								

(a) Minimum.
(b) 0.2 wt % vanadium.
(c) 0.12 wt % vanadium.
(d) 0.9-1.3 wt % nickel.
(e) 2.5-4.0 wt % thorium.
(f) 0.10 wt % carbon.
(g) 2.5 wt % tin.

TABLE 2. EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF LIGHT METALS

Material	Condition	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp, C	Test Temp, C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, percent				Ref.
									Uniform		Total		
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Al (99.99)	Annealed	8.0×10^{20}(a)	100	RT	2.1	2.1	6.6	12.2			46	30	1
1100	Ditto	8.7×10^{21}	50	RT	4.8	48.3	9.7	55.9			33	9.7	2
1100	"	8.7×10^{21}	50	RT	4.8	43.2	9.7	53.0			33	6.6	2
1100	"	1.1×10^{22}	50	RT	4.8	50.6	9.7	55.4			33	3.0	2
1100	"	1.1×10^{22}	50	RT	4.8	46.0	9.7	53.7			33	3.2	2
1100	H-12	5.0×10^{19}	100	RT	14.0	16.0	16.0	19.0					3
1100	H-12	6.4×10^{19}	100	RT	15.5	16.6	17.4	19.8					3
1100	H-12	8.0×10^{20}(a)		RT	15.1	18.7	17.6	22.0					1
2024		2×10^{19}	50	RT	45.3	48.7	71.6	71.1			26	25	4
2024		1.2×10^{20}	50	RT	45.3	48.2	71.6	74.3			26	26	4
2024		5.6×10^{20}	50	RT	45.3	56.1	71.6	79.4			26	25	4
2024		9.8×10^{20}	50	RT	45.3	66.2	71.6	84.9			26	24	4
6061		5.2×10^{20}	50	RT	40	43.7	47.2	50.2			21	22	4
6061		1.2×10^{21}	50	RT	40	42.8	47.2	51.9			21	21	4
6063	Annealed	1.3×10^{20}	100	RT	8.0	24.0	19.0	34.0			33.0	20.0	3
6063	Annealed	8.0×10^{20}(a)	100	RT	8.2	33.9	19.4	24.2			24	13	1
6063	T-4	1.0×10^{18}	40	RT	12.0		24.5				30.4	31.2	3
6063	T-4	6.4×10^{19}	100	RT	13.8	28.6	28.5	40.7					3
6063	T-4	8.0×10^{20}(a)	100	RT	14.1	39.4	28.5	48.2					1
6063	T-6	1.0×10^{18}	40	RT	28.3	28.1	33.2	34.1			23.4	20.3	3
6063	T-6	1.0×10^{18}	40	RT	33.4	33.8	34.6	35.8			16.4	14.1	3
6063	T-6	5.0×10^{19}	100	RT	34.0	36.0	39.0	40.0					3
X-2219		1.3×10^{20}	100	RT	25.0	28.0	38.0	42.0			17	16	3
X-2219		8.0×10^{20}(a)		RT	28.5	32.8	38.1	47.0			12	12	1
X-2219		1.3×10^{20}	100	RT	38.0	50.0	58.0	68.0			17	11	3
X-2219		8.0×10^{20}(a)		RT	38.3	47.2	57.8	63.8			12	11	1
X-8001	Annealed	5.0×10^{19}	100	RT	4.0	5.0	14.0	15.0					3
X-8001	Ditto	5.4×10^{19}	40	RT	5.5	8.0	16.3	17.1			30.0	23.2	5
X-8001	"	1.0×10^{20}	40	RT	5.5	11.3	16.3	19.1			30.0	19.8	5
X-8001		1.3×10^{20}	100	RT	9.0	10.0	17.0	18.0			43	33	3
X-8001		8.0×10^{20}(a)		RT	9.4	7.6	17.2	17.2			31	27	1
X-8001	H-14	1.3×10^{20}	100	RT	20.0	21.0	21.0	22.0			21	18	3
X-8001	H-14	8.0×10^{20}(a)		RT	20.3	20.6	21.2	22.7			15	14	1

TABLE 2. (Continued)

Material	Condition	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp, C	Test Temp, C	Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, percent				Ref.
									Uniform		Total		
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
M-257	Extruded	5.4×10^{19}	40	RT	28.6	34.7	36.2	38.6			8.0	6.5	5
M-257	Ditto	1.0×10^{20}	40	RT	28.6	35.8	36.2	40.2			8.0	6.3	5
M-257	"	1.2×10^{20}	50	RT	30.0	31.4	37.4	39.7			16.8	11.8	6
M-257	"	1.2×10^{20}	50	RT	19.0	20.0	21.2	21.5			11.5	11.6	6
M-257	"	1.5×10^{20}	50	RT	30.0	32.0	37.4	41.0			16.8	11.5	6
M-257	"	1.2×10^{20}	50	RT	19.0	20.1	21.2	22.0			11.5	10.5	6
M-257	"	3.1×10^{19}	270	RT	31.8	28.6	43.3	37.4			12.3	17.2	6
M-257	"	3.1×10^{19}	270	RT	21.0	18.2	28.0	21.0			11.5	11.0	6
A-356		2.0×10^{19}	50	RT	26	29.1	33.1	36.7			4	6	4
A-356		1.2×10^{20}	50	RT	26	33.5	33.1	42.0			4	6	4
A-356		5.6×10^{20}	50	RT	26	42.4	33.1	45.9			4	6	4
A-356		9.8×10^{20}	50	RT	26	52.1	33.1	54.4			4	3	4
M-486	Extruded	7.7×10^{19}	50	RT	31.9	30.7	40.8	44.1			17.0	15.5	6
M-486	Extruded	7.7×10^{19}	50	RT	19.8	20.4	21.0	21.9			16.8	16.6	6
Alcan 57S		2.0×10^{20}	50	RT	11.2	13.4	28.7	31.8			32	32	7
Alcan 6057		2.0×10^{20}	50	RT	14.9	14.9	34.0	35.2			33	33	7
H-3XA	Annealed	7.8×10^{18}	40	RT	13.7	15.8	26.7	28.4	5.1	5.0	13	20.4	8
H-3XA	Ditto	5.4×10^{19}	40	RT	13.7	16.4	26.7	34.5	5.1	6.1	13	16.2	8
H-3XA	"	1×10^{20}	40	RT	13.7	25.1	26.7	36.6	5.1	4.2	13	9.9	8
HK-31A	H-24	5.4×10^{19}	40	RT	26.5	18.1	36.1	37.0	3.5	6.1	3.5	6.1	8
HI-31A	H-24	1.0×10^{20}	40	RT	26.5	22.1	36.1	37.4	6	4.9	3.5	4.9	8
A-40		0.8×10^{16}	250-300	RT	40.1	39.6	47.8	47.7	10	8.9	45.9	40.8	9
A-40		1.3×10^{17}	250-300	RT	40.1	41.7	47.8	50.9	10	11.1	45.9	38.1	9
A-40		2.6×10^{18}	250-300	RT	40.1	46.7	47.8	57.5	10	11.6	45.9	32.8	9
A-110AT		1.3×10^{17}	250-300	RT	122.4	124.1	132.1	133.1	10.6	11.1	19.8	20.7	9

(a) Energy greater than 0.1 MeV.

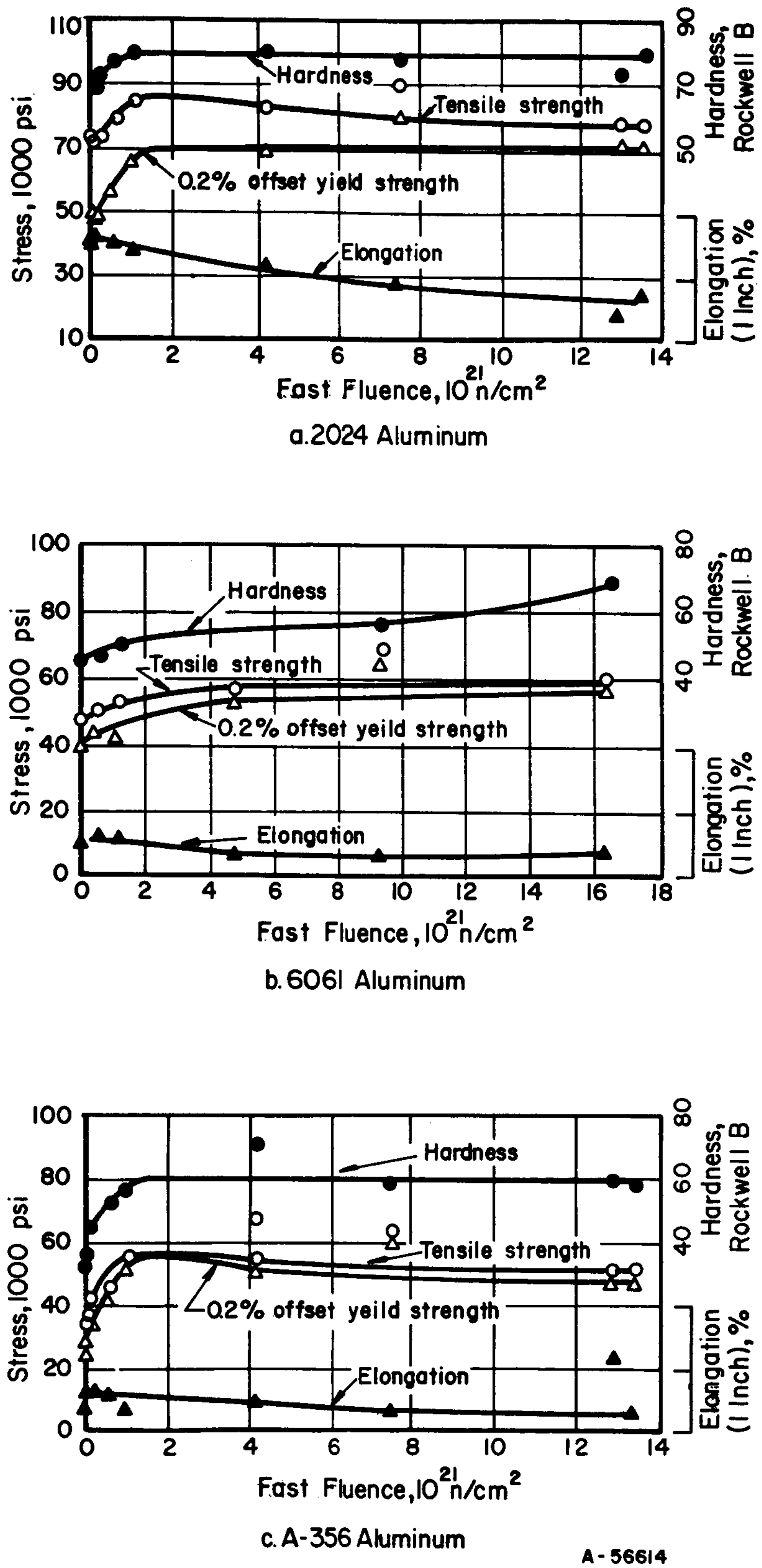

FIGURE 1. EFFECTS OF IRRADIATION ON THE ROOM-TEMPERATURE ROCKWELL HARDNESS, TENSILE STRENGTH, YIELD STRENGTH, AND ELONGATION OF ALUMINUM ALLOYS(10)

Limited tensile and impact tests have also been performed on irradiated SAP alloys (6 to 8 wt % Al_2O_3).[6] The room-temperature tensile and impact properties of the alloy were not affected by irradiation. These specimens had been irradiated to a fast fluence of 1.5×10^{20} n/cm^2 at 50 and 270 C.

Fatigue tests have been performed on irradiated aluminum by using notched specimens on a rotating-beam unit. The specimens had been irradiated to a fast fluence of 4×10^{16} n/cm^2 at 130 to 140 C. The tests showed that the irradiated specimens had a fatigue life only one-fourth that of the unirradiated specimens.[12] These results are in disagreement with those of Zamrik[13] who found a significant increase in fatigue strength of pure aluminum which had received a fast fluence of 1×10^{18} n/cm^2. The discrepancy may be due to the testing technique – Zamrik performed his tests by superimposing an alternating tension-compression stress on an original tensile stress while Brewster obtained his results on notched specimens.

Irradiation has also 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.[14] 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.[15] 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 environment. These phenomena have probably been noticed for aluminum because it has been irradiated near the 0.5 T_m temperature, while most other materials have not.

Magnesium Alloys

The few recent tensile tests on magnesium alloys indicate that fast neutron irradiation produces no significant changes in room-temperature properties. However, it must be remembered that the maximum fast fluence is only 1×10^{20} n/cm^2.[8] Figure 2 shows that irradiation improves the fatigue properties of magnesium. The specimens were irradiated at 80 C to a fast fluence of 10^{19} n/cm^2 and then tested by bending.[16]

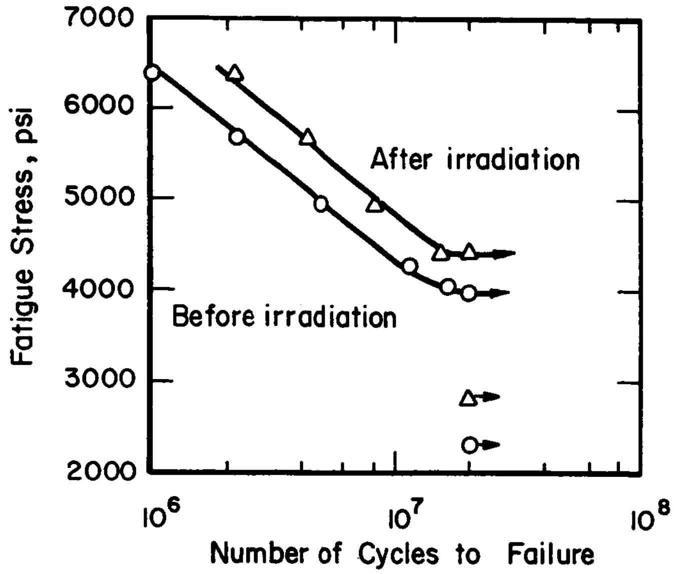

FIGURE 2. EFFECT OF FAST-NEUTRON IRRADIATION ON FATIGUE PROPERTIES OF MAGNESIUM(16)

Titanium Alloys

The limited data on titanium are insigificant since the maximum fast fluence is only 2.6×10^{18} n/cm^2. This low fast fluence causes a 20 percent reduction in total elongation.

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.

However, there are limitations 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(17, 18). 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.(18)

	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 solid solution in the matrix. However, at higher temperatures, the diffusion of helium atoms becomes significant and they agglomerate into bubbles. The diffusion coefficient of helium in beryllium has been found to be 6.8×10^{-14} cm^2/sec at 600 C and 1.13×10^{-9} cm^2/sec at 750 C.[19] 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 fast fluence of 7.6×10^{21} n/cm^2 at 70 C[20] or a fast fluence of 8×10^{20} n/cm^2 at 600 C.[21] 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 5×10^{20} n/cm^2[22, 23]. Another study showed a density decrease of 1.5 percent for beryllium irradiated to a fast fluence of 1.8×10^{20} n/cm^2 at 780 C.[24] 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. In a study performed by Idaho Nuclear Corporation, sections of the beryllium reflector from the MTR were annealed at temperatures of 600 to 1000 C.[18,25,26] These beryllium specimens had been irradiated to a fast fluence of 1×10^{21} n/cm^2 at 70 C. Results of the annealing studies are given in Figure 3. 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 1×10^{21} n/cm^2.[27] These studies also indicated that breakaway swelling was not only temperature dependent but also time dependent. This is illustrated in Figure 4 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 7.6×10^{21} n/cm^2, the breakaway swelling temperature is 600 C compared with about 725 C after irradiation to a fast fluence of 1×10^{21} n/cm^2.[27] Again, results show 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 5 illustrates the swelling threshold in irradiated beryllium as a function of temperature and fast fluence.[28]

Mechanical Properties

For a better understanding of the effects of radiation on beryllium's mechanical properties, the mechanical properties of unirradiated beryllium should be understood. A good review of the properties of beryllium has been recently issued and the mechanical properties are also covered.[29] Of main consideration in the mechanical properties of beryllium is the anisotropy of the material, due to its hexagonal structure. Of special importance is the brittleness in certain directions at low temperatures. Fracture in beryllium at room temperature occurs along three planes: basal plane (0001), the second-order prism plane (1120), and the (1012) plane. The most prominent fracture plane at

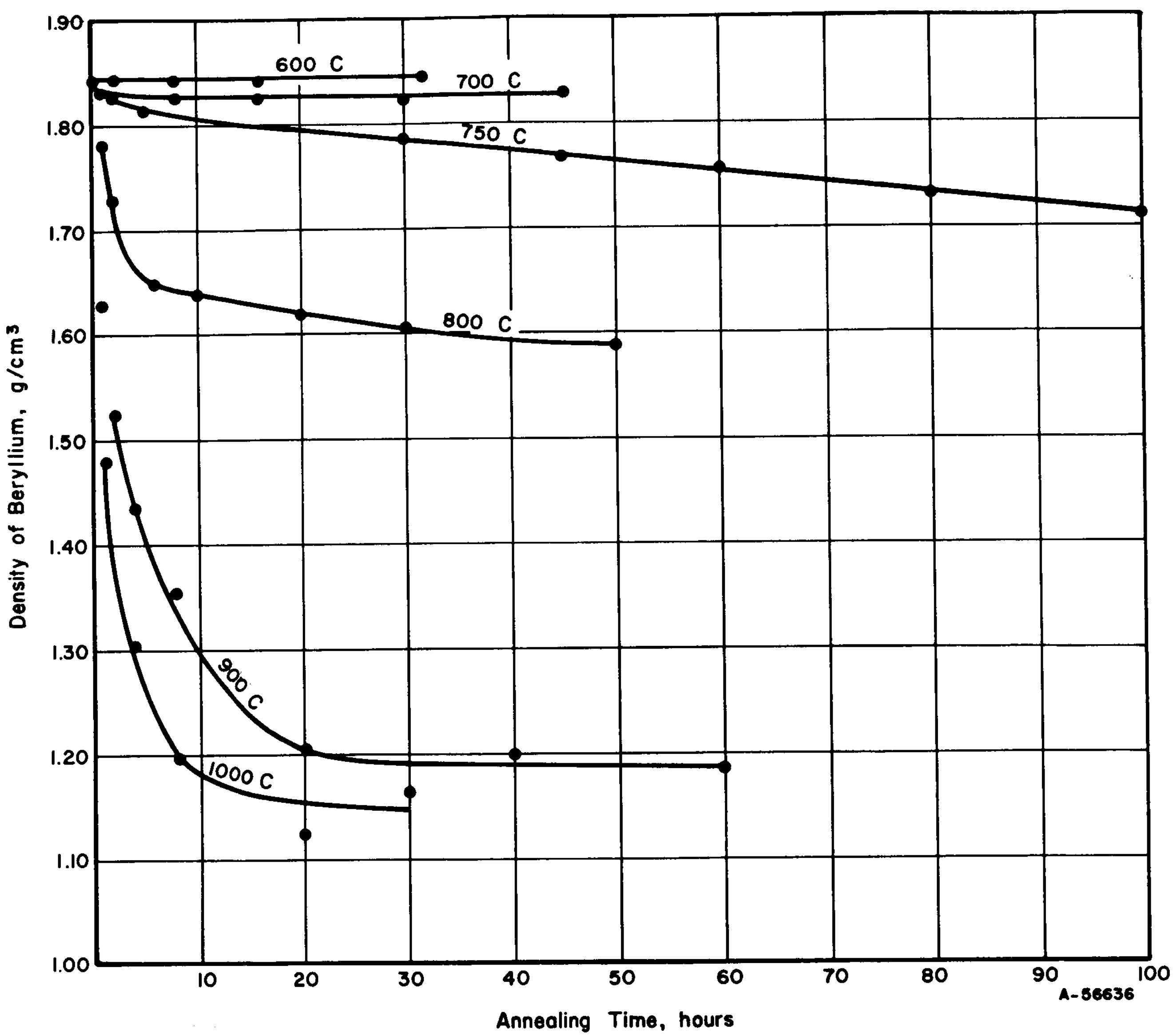

FIGURE 3. SWELLING OF BERYLLIUM IRRADIATED TO A FAST FLUENCE OF 1×10^{21} N/CM^2 AT 70 C[18,25,26] AS A FUNCTION OF ANNEALING TIME AND ANNEALING TEMPERATURE

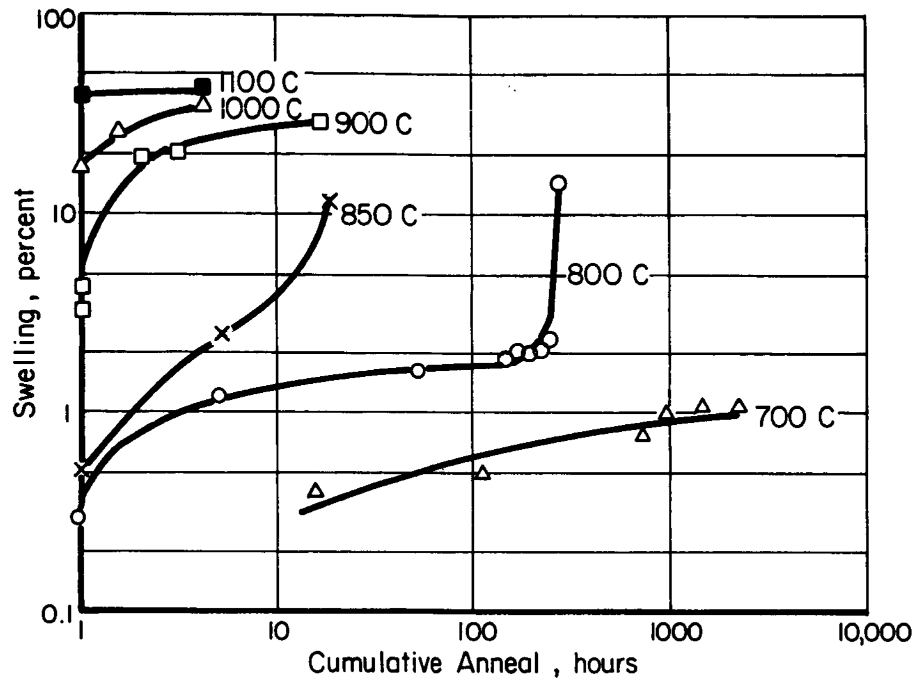

FIGURE 4. TIME DEPENDENCE OF SWELLING AT VARIOUS TEMPERATURES(27) FOR BERYLLIUM IRRADIATED AT 70 C TO A FAST FLUENCE OF 2.75×10^{21} N/CM^2

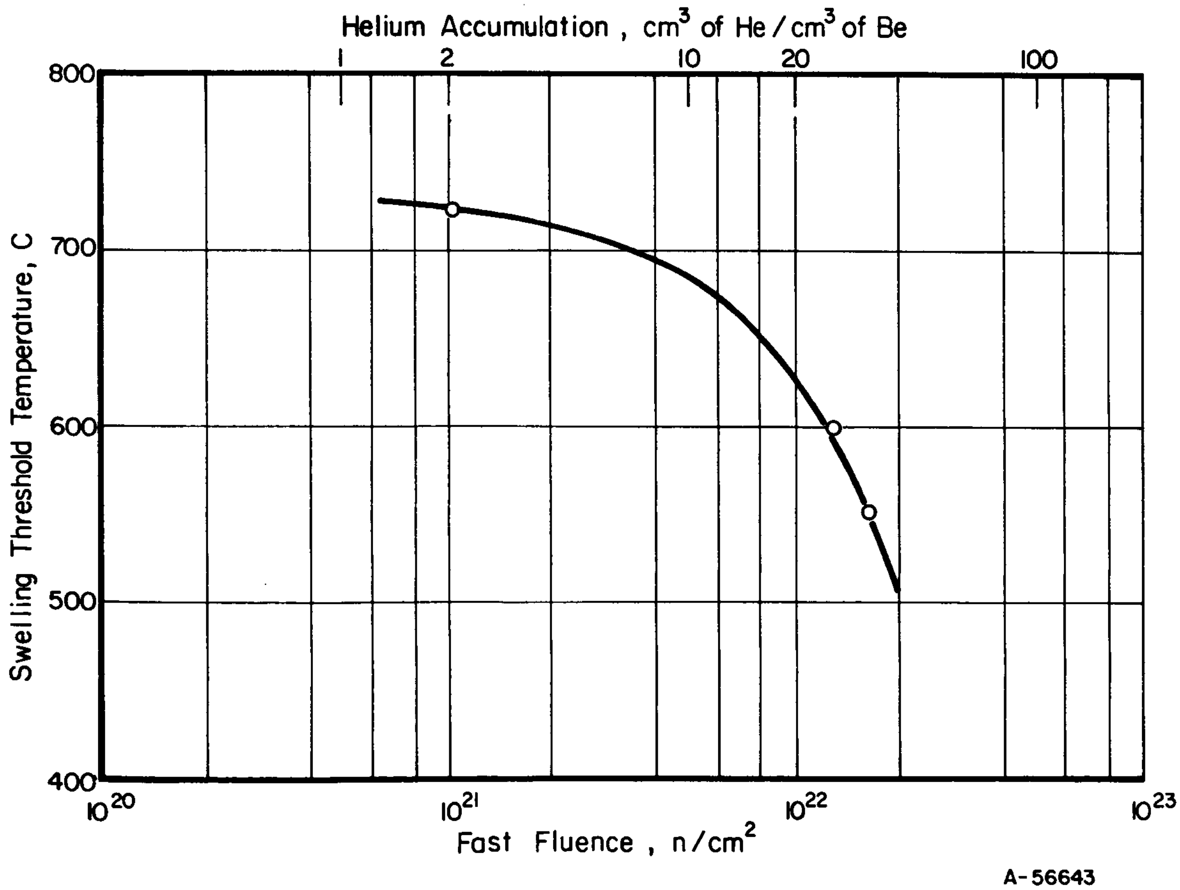

FIGURE 5. SWELLING THRESHOLD OF BERYLLIUM AS A FUNCTION OF GAS CONTENT AND FLUENCE(28)

room temperature is the basal plane where basal cleavage takes place. Since only limited basal glide can be accommodated before crack nucleation, this mode represents a limiting value in ductility. To improve beryllium's low-temperature ductility, it is absolutely necessary to orient the basal plane to minimize basal glide. At higher temperatures, fracture along the other planes becomes more prominent, and failure occurs by ductile shear at 400 C with considerable elongation (25 to 30 percent). Above that temperature, the ductility again decreases as the fracture mode becomes intergranular. Grain size and impurity content also influence the mechanical properties of beryllium. The increased grain size causes anisotropy in mechanical properties and, consequently, both ductility and strength are decreased. Beryllium metal contains a considerable amount of impurities (0.5 to 2.0 percent) – chiefly oxygen in the form of BeO. Other significant impurities are iron, silicon, aluminum, and carbon, usually present in quantities of up to 1000 ppm. Magnesium, manganese, nickel, and chromium are usually present in 100 ppm concentrations or above. These impurities generally tend to increase beryllium's strength and decrease its ductility.

Recent investigations show that annealing at 300 to 600 C for about 1000 hours significantly affects beryllium's mechanical properties. These long-time anneals generally increase ductility and decrease strength.[(22,23)] However, other investigators have found that annealing unirradiated beryllium at 600 C for 6 months did not affect the tensile properties.[(21)] It has been found that the best unirradiated-beryllium mechanical properties, especially with respect to ductility at low temperatures, are obtained by hot pressing beryllium flake into block and extruding the block at 1050 C. This is followed by a second extrusion or anneal at 750 to 850 C.[(22,23)] This fabrication technique gives good mechanical properties in the longitudinal direction, but does not improve the ductility in the transverse direction. Therefore, beryllium could be used as a uniaxially loaded structural material in the longitudinal direction; serious questions, however, are raised as to its applicability in biaxial loading situations because of its lack of ductility in the transverse direction.

Table 3 summarizes the effects of irradiation on mechanical properties of beryllium. The variables which are given in the table are: irradiation temperature, testing temperature, accumulated fast fluence, and different fabrication techniques. The data from Table 3 illustrate that irradiation of beryllium at 75 to 100 C does not change the tensile properties significantly, provided that the fast fluence is less than 5×10^{19} n/cm^2.[(22)] This holds true for all testing temperatures and for materials fabricated by different techniques. However, above a fast fluence of 5×10^{19} n/cm^2, there is a direct relationship between increased room-temperature strength and decreased ductility with increasing fast fluence, as illustrated in Figure 6.[(36)] These changes in tensile properties are more prominent at lower testing temperatures, and approach the values for unirradiated material at higher testing temperatures, as shown in Figure 7.[(21)] 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. As for the lower-temperature irradiations, the mechanical-property changes become less

TABLE 3. EFFECT OF RADIATION ON THE TENSILE PROPERTIES OF BERYLLIUM

Material	Fast Fluence, n/cm² (>1 Mev)	Irradiation Temp, C	Testing Temp, C	0.1% Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, %		Reduction in Area, %		References
				Preirr.	Postirr.	Preirr.	Postirr.	Preirr.	Postirr.	Preirr.	Postirr.	
Pechiney, sintered, extruded at 1050 C	5×10^{18}	75-100	100	31.7	30.2	58.6	55.3	18.7	21.3	20	21.5	22
			200	23	24.3	41.7	42.9	22.8	24.2	27.7	27.5	
			300	18.1	22.6	32.1	35.3	21.3	18.8	31.2	22.6	
			400	16	17.8	24.4	26.1	12.5	10.0	25.3	25.7	
			500	12	12.8	13.7	15.0	7.0	6.7	6.4	5.4	
			600	4.3	4.5	4.6	4.6	5.0	5.4	3.1	4.9	
	1×10^{19}		100	31.7	24.6	58.6	54.3	18.7	19.6	20	22.5	
			200	23	24.0	41.7	41.7	22.8	20.7	27.7	29.2	
			300	18.1	22.7	32.1	34.5	21.3	13.9	31.2	32.4	
			400	16	19.2	22.4	25.6	12.5	9.0	25.3	25.1	
			500	12	14.2	13.7	15.4	7.0	6.3	6.4	6.1	
			600	4.3	4.2	4.6	4.6	5.0	6.0	3.1	5.1	
	5×10^{19}		100	31.7	31.3	58.6	61.1	18.7	18.8	20	22.6	
			200	23	26.9	41.7	44.7	22.8	14.6	27.7	26.1	
			300	18.1	21.5	32.1	34.8	21.3	11.5	31.2	29.1	
			400	16	17.7	22.4	25.7	12.5	8.3	25.3	22.7	
			500	12	14.2	13.7	15.1	7.0	4.9	6.4	6.5	
			600	4.3	4.4	4.6	4.4	5.0	4.9	3.1	4.2	
	2×10^{20}		100	31.7	40.8	58.6	65.2	18.7	9.9	20	18.4	
			200	23	37.8	41.7	51.8	22.8	11.4	27.7	20.4	
			300	18.1	28.1	32.1	44.3	21.3	9.9	31.2	19.1	
			400	16	26.1	22.4	36.5	12.7	7.7	25.3	16.8	
			500	12	17.3	13.7	21.5	7.0	3.2	6.4	1.7	
	3×10^{20}		100	31.7	48.8	58.6	79.7	18.7	11	20	19.2	
			200	23	39.8	41.7	62	22.8	12	27.7	21.2	
			300	18.1	33.7	32.1	51.2	21.3	8.4	31.2	23.4	
			400	16	30.2	22.4	39.3	12.7	7.1	25.3	18.3	
			500	12	22.1	13.7	23.0	7.0	4.8	6.4	0.5	
			600	4.3	5.1	4.6	5.5	5.0	4.1	3.1	7.1	
Pechiney, sintered, extruded at 750 C	2×10^{20}		100	58.6	63	88.2	89.3	16.9	5.5	17.9	9.3	
			200	53.8	55	70.1	--	20.2	--	22.4	--	
			300	43.6	47.5	49	58.3	15.1	8.8	22.4	19.1	
			400	30.4	38.2	31.3	44.5	9.9	8.7	20.2	13.3	
			500	11.5	19.7	12.1	20.6	9.4	6.5	9.0	4.8	
			600	6.2	7.0	6.8	7.3	5.4	6.4	2.1	1.7	
Hot pressed, hot extruded at 1050 C with 1500 psi	3.1×10^{20}		RT	43	61.2	91	68.5	14.3	0.7	13.0	0	
			100	42.3	55.5	77.6	80.1	29.7	1.8	26.4	2.0	30
			200	35.7	45.1	60.1	65.7	30.4	10.9	37.8	26	
			300	30.1	45.2	48.5	61.1	25.1	15.6	39.9	24.7	
			400	27.0	42.9	37.7	48.2	14.7	9.4	33.7	1.5	
			500	18.9	27.2	20.6	28.7	9.8	5.2	8.0	2.8	
			600	7.3	19.4	7.7	20.2	13.7	4.7	8.3	4.8	
			700	3.7	--	3.8	--	16.9	7.3	10.9	--	

TABLE 3. (Continued)

Material	Fast Fluence, n/cm³ (>1 MeV)	Irradiation Temp, C	Testing Temp, C	0.1% Offset Yield Strength, 1000 psi Preirr.	Postirr.	Ultimate Tensile Strength, 1000 psi Preirr.	Postirr.	Elongation, % Preirr.	Postirr.	Reduction in Area, % Preirr.	Postirr.	References
Hot pressed, hot extruded at 1050 C with 1500 psi (Continued)	6×10^{20}		RT	43	78.7	91	79.6	14.3	0.2	13	0	30
			100	42.3	70.7	77.6	71.3	29.7	0.2	26.4	0.3	
			200	35.7	62.1	60.1	70.4	30.4	0.9	37.8	0.9	
			300	30.1	56.8	48.5	63.2	25.1	0.7	39.9	0.6	
			400	27.0	52.8	37.7	53.9	14.7	0.7	33.7	0.4	
			500	18.9	27.9	20.6	31.2	9.8	2.8	8.0	2.0	
			600	7.3	10.4	7.7	10.9	13.7	6.1	8.3	3.7	
			700	3.7	4.6	3.8	4.7	16.9	6.1	10.9	2.9	
Hot pressed, hot extruded at 1050 C with 800 psi	3.1×10^{20}		RT	54.7	74.6	59.2	86.1	0.2	0.8	0	1.6	
			100	--	70.8	--	86.9	--	2.2	--	1.5	
			200	47.8	64.2	73.1	87.4	22	6.0	27.7	6.3	
			300	43	56.5	55.9	74	29.6	7.9	33.9	11.8	
			400	35.9	50.8	43.2	57.5	13	7.4	22.6	14.5	
			500	26	33.6	27.9	34.7	9	1.9	5.6	1.5	
			600	13.4	20.3	14.6	20.7	4.4	2.7	1.1	1.7	
Brush N-50-B, warm extruded	1.5×10^{19}	250	100			73.5	71.2	9.5	6			30,31
			200			58.7	60.7	43	39			
			300			43.8	46	38	39			
			400			32.5	34	44	34			
			500			23.1	24	36	23			
			600			15.1	13	20	1.5			
	4.0×10^{19}	350	100			73.5	75.6	9.5	10.5			
			200			58.7	57.7	43	22			
			300			43.8	44.8	38	43			
			400			32.5	30.9	44	50			
			500			23.1	21.7	36	47			
			600			15.1	15	20	14			
Brush-2-300-B, warm extruded	1.4×10^{19}	210-280	100				80.8		8			
			200			71.2	74.6	39	31			
			300			57.2	54.7	53	41			
			400				46.3		54			
			500			31.3	31.6	31	26			
			600			21.1	19.9	16.5	16			
	1.5×10^{19}	250	100				86.6		15			
			200			71.2	70.6	39	39			
			300			57.2	54.7	53	39			
			400				42.7		43			
			430			38.3	39.0	42	37			
			500			31.3	32.3	31	29			
			600			21.1	21.3	16.5	17			
	1.0×10^{21}	430	RT			88.1	94.5	3.0	0			
			200			72.6	77.1	30.0	0			
			300			55.7	62.2	38	26			
			430			39.0	50.7	58	22			
			500			32.8	41.8	36	23			
			600			21.1	29.8	16.5	5			

TABLE 3. (Continued)

Material	Fast Fluence, n/cm^3 (>1 MeV)	Irradiation Temp, C	Testing Temp, C	0.1% Offset Yield Strength, 1000 psi Preirr.	0.1% Offset Yield Strength, 1000 psi Postirr.	Ultimate Tensile Strength, 1000 psi Preirr.	Ultimate Tensile Strength, 1000 psi Postirr.	Elongation, % Preirr.	Elongation, % Postirr.	Reduction in Area, % Preirr.	Reduction in Area, % Postirr.	References
Brush, hot pressed	3.3×10^{21}(a)	280-480	RT			31.0	38	3	0			27
			RT			35.0	50.0	3	0			
			300	12		22	50	3	0			
			300	12	55	22	58	3	0			
			450	12	43	22	48	10	0			
			450	13		23	47	9	0			
			600	10		15		4	0			
Pechiney, hot pressed, longitudinal	2.0×10^{20}	350	RT	36	55			17	9			32
			100	35	50			22	15			
			300	29	35			22	16			
			450	22	22			9	9			
			600	8	9			5	5			
Pechiney, hot pressed, transitional			RT	40	37			0	0			
			100	40	46			0.1	0.1			
			300	35	32			0.5	0.5			
			450	29	25			1.3	1.7			
			600	5	5			1.5	1.0			
Brush, hot pressed	3.3×10^{21}(a)	280-480	RT(b)		13			3	0.2			27
			450(c)	12	44			10	0.1			
			450(d)	12	28			10	0.1			
			450(c)	12	21			10	0.3			
QMV (98+%Be)	2.4×10^{20}	40	20	24.5		35.7	51.4	1.4	0.2			33
	7.2×10^{20}	40	20	24.5		35.7	50.4	1.4	0.2			
	2.6×10^{21}	40	20	24.5	>47.8	35.7	>47.8	1.4	0			
	3.8×10^{21}	40	20	24.5		35.7		1.4	0			
Pechiney, loose sintered warm extruded,	3.2×10^{19}	450	450		17.5		22.4		20.5			34
hot pressed, extruded at	3.8×10^{19}	600	600	6.0	6.0	7.3	8.8	13.6	8.2			
1050 C, extruded at	1.7×10^{19}	700	700	3.2	3.2	3.7	3.9	9.2	2.8			
750 C	3.1×10^{20}	550	RT	47.7	97.1	79.2	99.5	6.9	1.4	5	1	30
			200	36.6	74.6	60.1	86.6	51	25	58	37	
			300	58.3	36	71.7	46.3	42	40	52	42	
			400	52.2	26	56	28.2	28	25	41	28	
			500	26.2	32.5	29.2	34.6	30	17	25	16	
			600	12.9	10.6	14.6	12.4	19	16	12	12	
			700	4.7	5.3	5.7	6.2	13	7	6	13	
Hot pressed, extruded at 1050 C, extruded at 800 C, stress relieved at 750 C			RT	38.2	41.7	78.8	58.8	8	7	8	12	
			200	51	38.8	67.7	57.6	40	20	59	19	
			300	29	32.6	42.9	42.8	65	39	76	48	
			400	27.7	26.4	27.2	29.7	82	40	79	46	
			500	15.3	20.0	16.4	20.7	65	37	38	43	
			600	8.5	11.1	96	12.6	38	10	29	9	
			700	3.7	3.6	4.5	3.9	14	12	10	6	

TABLE 3. (Continued)

Material	Fast Fluence, n/cm^2 (>1 MeV)	Irradiation Temp, C	Testing Temp, C	0.1% Offset Yield Strength, 1000 psi Preirr.	0.1% Offset Yield Strength, 1000 psi Postirr.	Ultimate Tensile Strength, 1000 psi Preirr.	Ultimate Tensile Strength, 1000 psi Postirr.	Elongation, % Preirr.	Elongation, % Postirr.	Reduction in Area, % Preirr.	Reduction in Area, % Postirr.	References
B11 Vacuum Cast,	3.5-5.5 x 10^{20}	450	450	20.4	24.3	34.9	34	32	18	54	28	35
extruded at 1050 C,		550	550	16.6	19.4	22.2	21	53	7.4	64	9.6	
extruded at 850 C,		650	650	8.5	12.1	9.5	12.9	49	2.3	43	1.8	
B12 Sintered at 1220 C		450	450	11.2	17.4	12.3	18.4	33	11	35	14	
extruded at 850 C		550	550	8.0	10.8	8.5	11.1	29	2.0	31	3.5	
		650	650	4.3	4.9	4.6	5.0	13	2.2	10	4.0	
B13 Sintered at 1220 C,		450	450	20.6	22.5	24.8	25.5	26	22	62	34	
extruded at 1050 C		550	550	12.8	17.1	15.3	18.4	20	9.8	27	10	
		650	650	8.0	7.9	8.8	8.4	13	2.1	12	0.6	
B14 Sintered at 1220 C,		450	450	17.6	20.8	21.8	22.3	41	29	72	40	
extruded at 1050 C		550	550	13.2	16.4	15.5	17.1	28	11	38	14	
		650	650	6.2	5.7	7.2	6.6	21	2.7	15	1.8	
B15 Vaccum cast,		450	450	8.0	14.1	11.2	14.5	37	7.5	45	14	
extruded at 1050 C,		550	550	4.6	10.1	4.9	10.2	40	1.6	44	3.2	
extruded at 850 C, slow cooled		650	650	2.5	--	3.0	2.6	22	1.1	22	1.1	
B16 Sintered at 1220 C,		450	450	10.5	18	11.9	18.5	35	6.0	34	9.3	
extruded at 1050 C,		550	550	6.0	10.9	6.5	11.2	19	1.4	21	2.8	
extruded at 850 C, standard cool		650	650	3.2	3.1	3.9	3.1	8.2	0.6	7.1	3.0	
B21 Sintered at 1220 C,	5.5-7 x 10^{20}	450	450	12.5	22.2	13.8	23.1	24.5	2.5	30	4.7	23
extruded at 850 C	9 x 10^{20}	550	550	7.4	12.9	7.8	13.7	26.5	3.5	27	3.0	
		650	650	5.5	4.9	5.7	4.9	12.3	1.6	9.3	1.3	
B22 Hot pressed,	5.5-7 x 10^{20}	450	450	21.3	32.5	26	32.3	29.5	16	5.2	32.7	
extruded at 1050 C,	9 x 10^{20}	550	550	13.9	17.8	15.7	18.8	30.5	6.2	28	7.2	
extruded at 850 C,		650	650	4.2	7.3	5.5	7.9	21.2	4.0	18.5	4.2	
B23 Hot pressed,	5.5-7 x 10^{20}	450	450	20.1	30.6	24.9	31.1	36	20	53.5	36.2	
extruded at 1050 C,	9 x 10^{20}	550	550	13.4	17.9	14.7	18.7	26	8.7	24.5	7.7	
extruded at 850 C, slow cooled,		650	650	4.4	7.6	5.9	8.4	18	5.0	9.9	2.3	
B24 Sintered at 1220 C,	5.5-7 x 10^{20}	450	450	11.8	21.5	13.6	24	29.5	3.5	33.2	5.2	
extruded at 850 C	9 x 10^{20}	550	550	6.4	10.9	6.9	11.3	11.7	1.1	16	3.3	
		650	650	3.9	--	4.4	3.6	8.5	0.9	7.6	2.0	
B25 Sintered at 1220 C,	5.5-7 x 10^{20}	450	450	11.5	16.7	13.3	18.3	20.2	4.4	34.5	7.6	
extruded at 850 C,	9 x 10^{20}	550	550	6.4	11.8	7.3	12.3	20	1.3	19	1.2	
slow cooled		650	650	3.8	2.6	4.6	2.8	13.5	0.5	7.7	3.0	
B26 Hot pressed,	5.5-7 x 10^{20}	450	450	23	29	26.9	29.8	27	20	50	33	
extruded at 1050 C	9 x 10^{20}	550	550	13.6	19.1	15.1	20.6	21	8.3	28	7.4	
extruded at 850 C		650	650	4.7	7.4	5.6	7.7	26.5	3.1	12	0.9	

TABLE 3. (Continued)

Material	Fast Fluence, n/cm^2 (>1 MeV)	Irradiation Temp, C	Testing Temp, C	0.1% Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Elongation, %		Reduction in Area, %		References
				Preirr.	Postirr.	Preirr.	Postirr.	Preirr.	Postirr.	Preirr.	Postirr.	
Sintered, extruded at 1050 C	8.5×10^{19}	600	RT	61.5(a)	51.3(a)	96.2	66	3.9	1.8	--	1.6	22
				61.5(a)	50.5(a)		83.9		11.3	--	11	
				61.5(a)	51.5(a)		85.7		9.5	--	8.6	
				61.5(a)	50(a)		86.9		11.7	--	--	
	1.5×10^{20}	670		51.7(a)	47.6	80.2	82.3	5.3	9.0	5.6	--	
				51.7(a)	48.6		78.8		7.9		8.1	
				51.7(a)	47.9(a)		84		9.3		--	
	1.4×10^{20}	500		46.7(a)	52.1(a)	69.2	53.8	2.7	1.0	1.1	--	
				46.7(a)	59(a)		58.7		1.2		0.6	
				46.7(a)	61.5(a)		60.8		0.4		0.5	
	1.5×10^{20}			42.5	52.6(a)	91.5	58.3	19.1	0.9	18.4	1	
			100	42.5	51.9(a)		74.6		12.9		12	
			200	34.1	43.1(a)	52.6	55.9	34.3	21.1	47.5	33	
			300	27.9	37.6(a)	37.2	41.8	28.7	36.8	49.5	44.4	
			400	23.1	27.7(a)	27	28.4	22.2	19.3	41.5	40.7	
			500		19.5(a)		20.5		9.1		10.5	
			600	8.1	9.4	8.2	9.7	7.3	5.3	6.4	3.9	
	2.4×10^{20}	600	RT	46.6	53.7(a)	91.7	88.6	18.9	13.6	17.4	12.5	
			100		50.3(a)		77.5		23.1		19.6	
			200	33.2	40.6(a)	51.8	56.1	36.7	36.6	41.4	37.2	
			300	30.2	33.9(a)	41	40.6	40.8	31	46.2	43.6	
			400	24.1	29.7(a)	27	31.5	23.6	26	35.2	32.7	
			500	14.8	17.5	14.8	17.7	11.9	9.7	18.1	15.1	
			600	7.6	9.9	8.2	10.3	8.7	6.3	3.8	6.1	
		300-500	-195				64.8		0.1		0.2	
			-72				61.5		0.1		0.2	
			RT	43.3	64.6(a)	88.3	65.7	15.3	0.5	16.6	0.2	
			200	33.6	53.4(a)	51	55.5	34	22.8	38.5	34.1	
			300	25.1	39.8(a)							
			400	21.3	30.3(a)	25.7	31.6	20.3	18.3	35.1	29.2	
			500	17.5	19.7	18	19.9	13.7	9.3	14.9	18.0	
			600	6.2	9.6	6.6	9.7	5.8	7.3	2.3	2.9	

(a) Neutrons with energies above 1.4 MeV.
(b) Annealed 1 hour at 1000 C.
(c) Annealed 1 hour at 450 C.
(d) Annealed 1 hour at 850 C.
(e) Annealed 1 hour at 950 C.

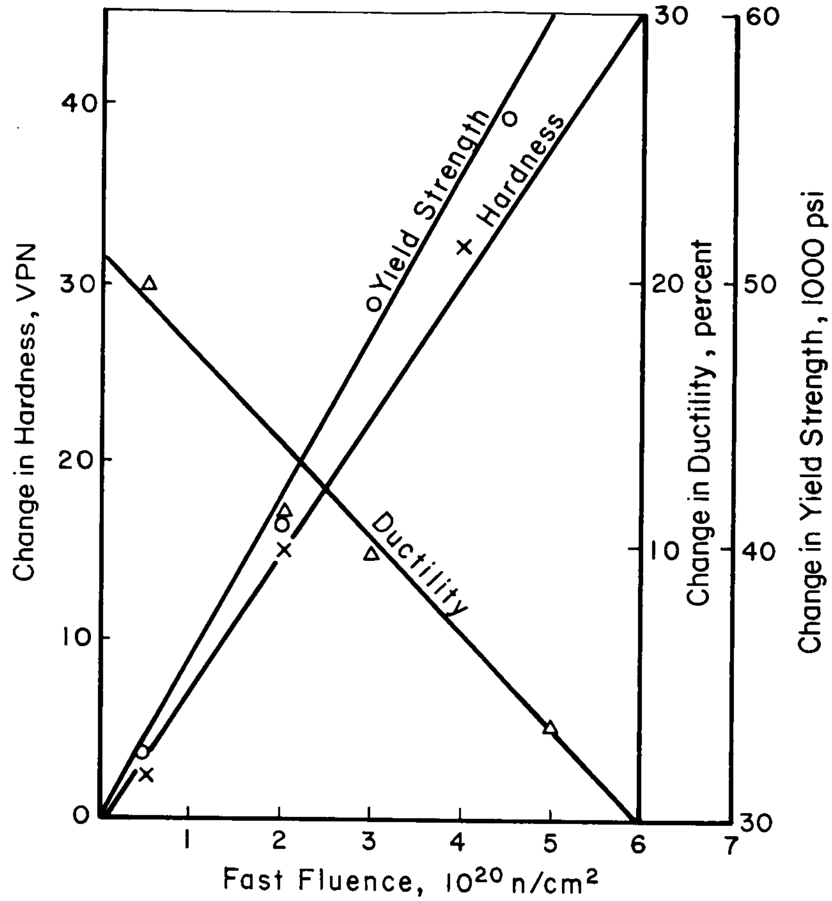

FIGURE 6. CHANGES IN YIELD STRENGTH, DUCTILITY, AND HARDNESS FOR BERYLLIUM IRRADIATED AT 75 TO 100 C(36)

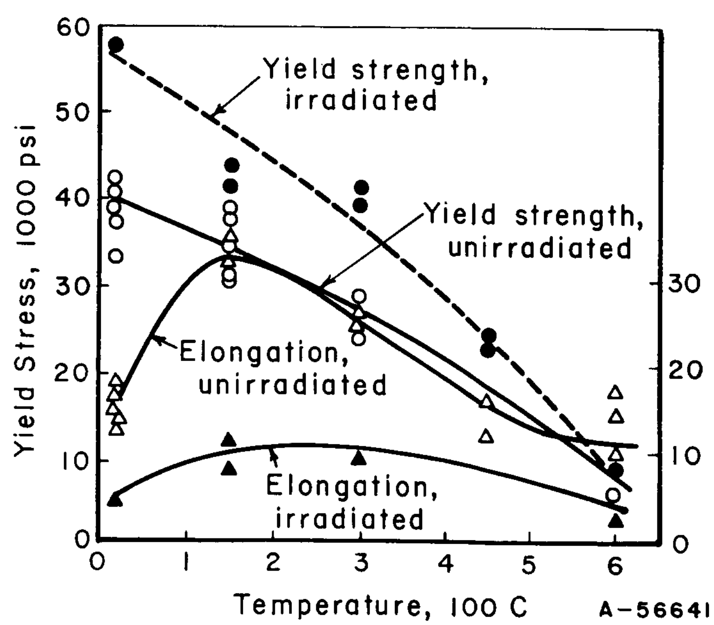

FIGURE 7. EFFECT OF IRRADIATION AT 100 C TO A FAST FLUENCE OF 4×10^{20} N/CM^2 ON TENSILE PROPERTIES OF BERYLLIUM(21)

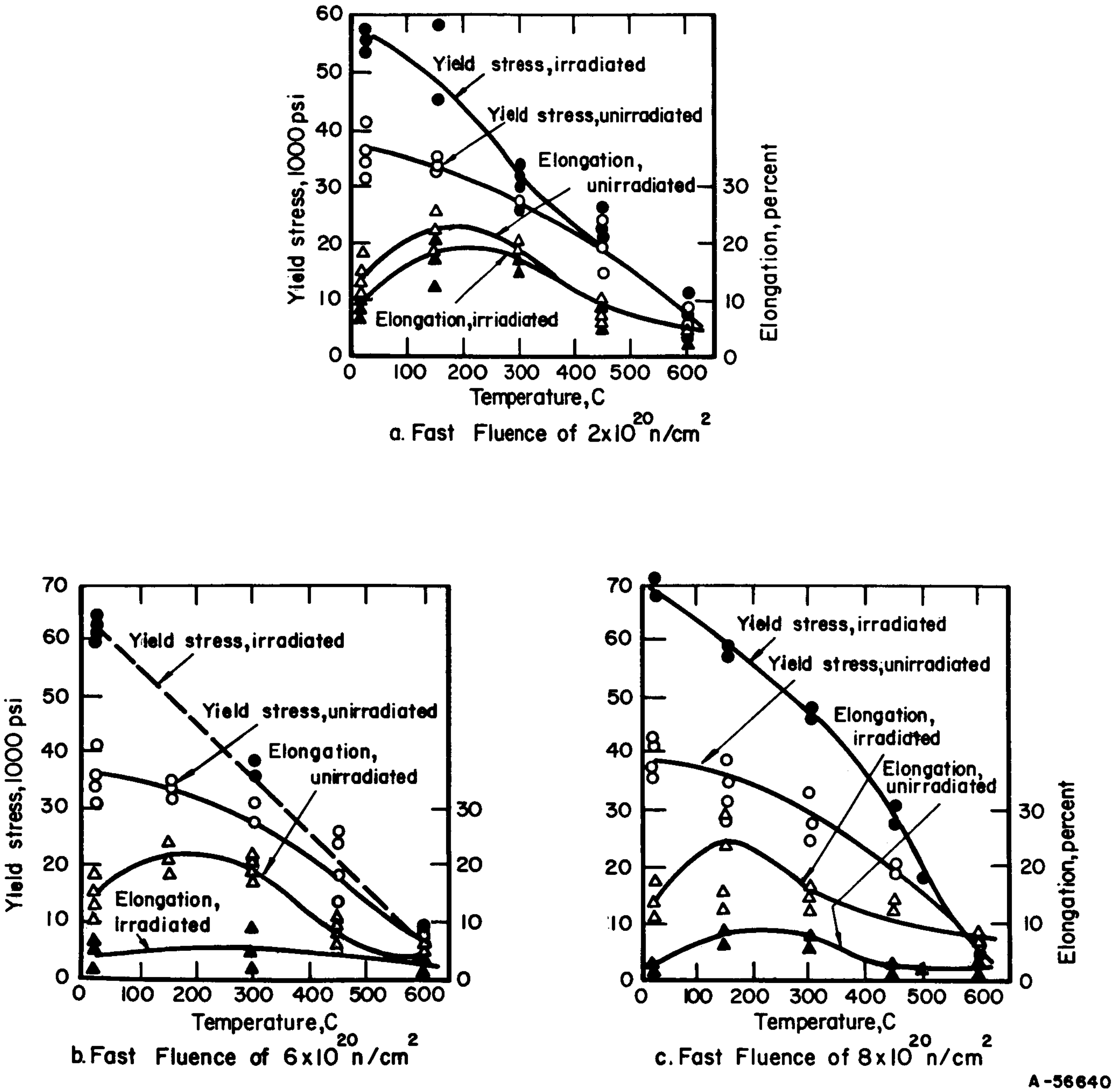

FIGURE 8. EFFECT OF IRRADIATION AT 350 C ON THE TENSILE PROPERTIES OF BERYLLIUM[21]

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. The severity of this helium-induced ductility loss increases with higher fast fluence, as shown in Figure 9. As illustrated in Table 4, higher irradiation and higher testing temperatures cause larger degrees of embrittlement.

This temperature dependence of ductility loss is due to the larger helium-bubble size found on fracture surfaces. Table 4 also illustrates the variation in irradiation effects on tensile properties of different batches of material irradiated at similar temperatures and to similar fluences. It has been found that the hot-pressed beryllium exhibits more resistance to irradiation embrittlement than does sintered material. Also, the best forming procedure is extrusion at 1050 C followed by either further extruding or heat treating in the 750 to 850 C range.[23]

Irradiation at 600 C and above results in tensile-property changes for some batches of beryllium but not for other batches. Figure 10 shows the results for a batch of beryllium whose tensile properties were not changed by irradiation at 600 C[21], while Tables 3 and 4 show the results for material whose properties were changed by irradiation at 650 C and above. 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 11, indicate that irradiation does not affect the transverse tensile properties of beryllium.[37] It should be emphasized, however, that beryllium is extremely brittle in the transverse direction despite the irradiation condition.

Barnes[38] has developed a mathematical model for predicting irradiation-induced changes in the yield strength of beryllium. This model is based on the size and number of helium bubbles in the material and assumes that these bubbles impede the movement of dislocations in the matrix. The yield strength of the irradiated material is given by the following formula:

$$\sigma_i = \sigma_v + Gb\,(2Nr)^{1/2} \quad ,$$

where

σ_v = yield strength of unirradiated material

G = shear modulus

b = Burger's vector

N = number of bubbles per cm^3

r = radius of the bubbles.

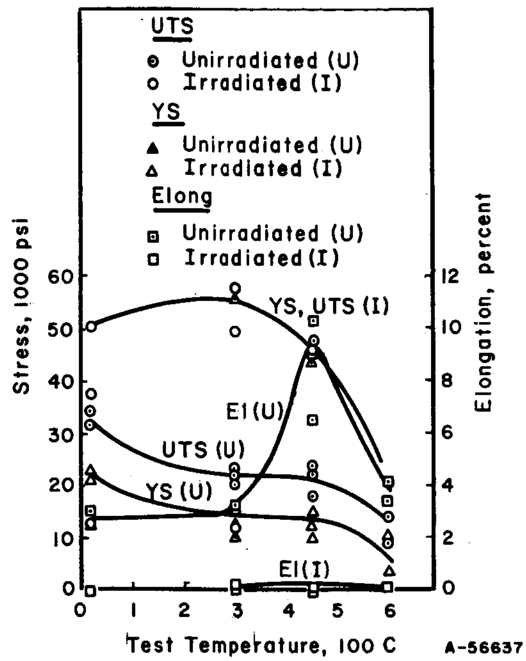

FIGURE 9. TENSILE PROPERTIES OF BERYLLIUM IRRADIATED AT 280 TO 480 C TO A FAST FLUENCE OF 3.3 x 10^{20} N/CM^2 (27)

TABLE 4. EFFECT OF TESTING AND IRRADIATION TEMPERATURE ON ELONGATION OF DIFFERENT BATCHES OF BERYLLIUM(23)

	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 1220 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 to 7 x 10^{20} n/cm^2.
(d) Received a fast fluence of 9 x 10^{20} n/cm^2.

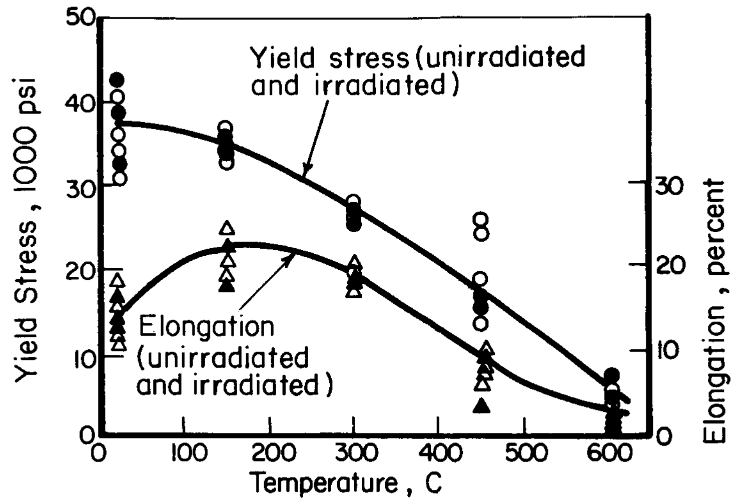

FIGURE 10. EFFECT OF 600 C IRRADIATION TO A FAST FLUENCE OF 6×10^{20} N/CM^2 ON THE LONGITUDINAL MECHANICAL PROPERTIES OF BERYLLIUM[21]

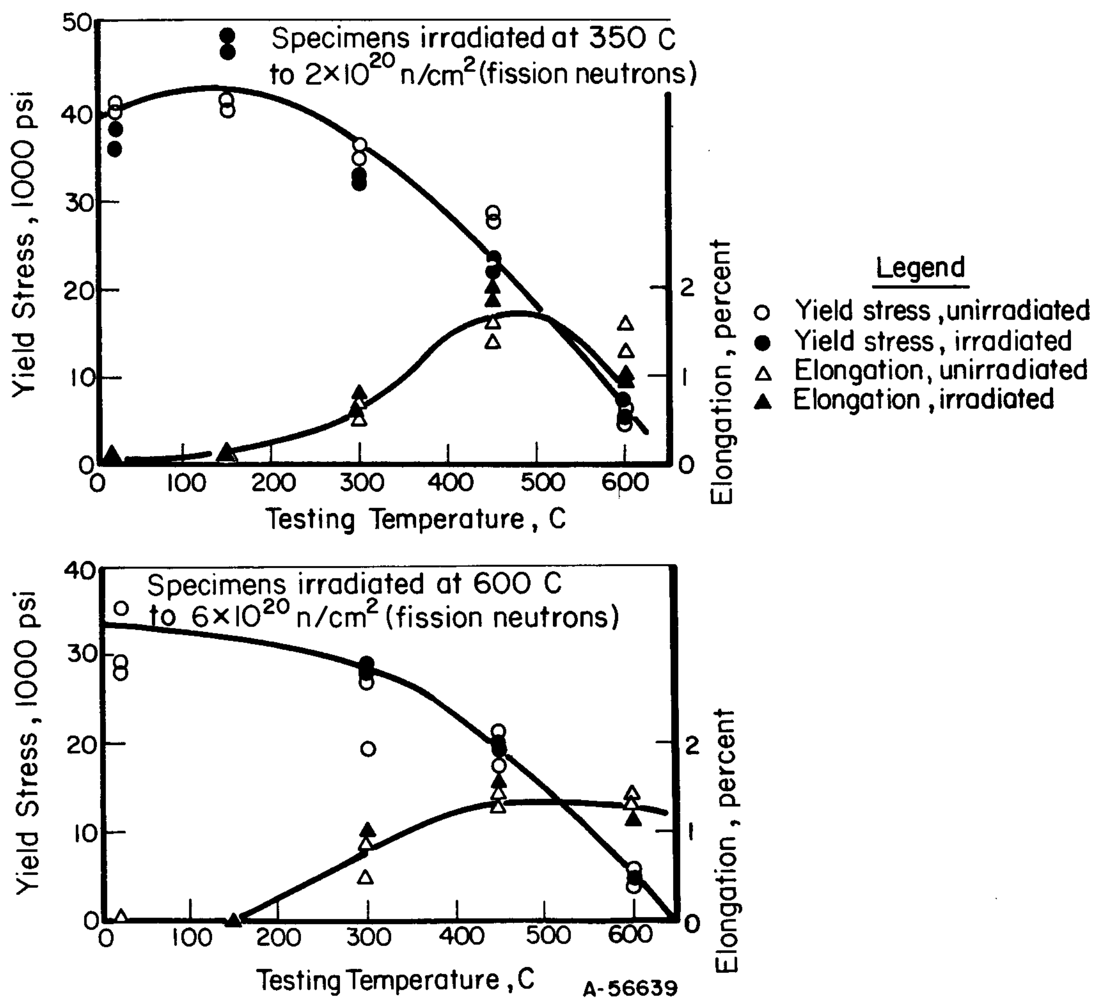

FIGURE 11. EFFECT OF IRRADIATION ON THE TRANSVERSE MECHANICAL PROPERTIES OF EXTRUDED BERYLLIUM[37]

The good agreement between the calculated and measured yield strength for the irradiated beryllium is shown in Figure 12.[21] Table 5 illustrates that the yield strength of irradiated beryllium can be predicted for different testing temperatures and even after annealing.

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. Figures 13 through 16 illustrate the restoration of yield strength, ductility, and electrical resistivity with increasing annealing temperature. The temperature where significant removal of irradiation hardening takes place appears identical for all of these properties, as well as the temperature where significant swelling in irradiated beryllium 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. It can be seen in Figure 16 that beryllium which has received a lower fast fluence reverts to its preirradiation electrical resistivity sooner than beryllium receiving the higher fast fluence.[36] Similar fast-fluence dependence on annealing out of irradiation-produced effects has been observed for tensile properties.[30] 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.

Increased hardness in beryllium has been found proportional to the fast fluence, as illustrated in Figure 6.[36] The effects of irradiation temperature and grain size on radiation-induced hardness changes are illustrated in Table 6. These results show no difference in the relative radiation-induced hardness changes for beryllium having different grain sizes. While irradiation at 450 C resulted in significant hardness changes, only minor hardness changes were produced by irradiation at 650 C.[35] Annealing of the irradiated beryllium restored the preirradiation hardness at about the same annealing temperature where yield strength and ductility are restored (Figure 14).

As shown in Table 7[39], compression tests have been performed on beryllium irradiated at 70 C. 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 resulted only in minor changes in compression properties, as indicated in Table 8.[19] These results, which show no changes in compressive properties, agree with tensile data obtained on beryllium irradiated at the same temperatures.

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.[24] Results of these bend tests, shown in Table 9, indicate no ductility changes for specimens irradiated at 500 to 700 C. This is expected since irradiation in that temperature range is not 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.[24] 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 at 700 C.[20] Results indicated that the fracture stress required for failure by

TABLE 5. CALCULATED AND OBSERVED YIELD STRESSES OF IRRADIATED BERYLLIUM[21]

Fast Fluence, n/cm^2	Irradiation Temp, C	Postirradiation Annealing	Yield Strength Calculated From Bubble Counts, psi	Unirradiated Yield Strength, psi	Calculated Yield Strength Plus Unirradiated Yield Strength, psi	Actual Irradiated Yield Strength, psi
2×10^{20}	350	1 hr at 600 C	5,080	35,000 (20 C)	40,080	44,300 (20 C)
2×10^{20}	350	1 hr at 1000 C	6,420	35,000 (20 C)	41,420	37,000 (20 C)
6×10^{20}	350	1 hr at 800 C	12,970	35,000 (20 C)	47,970	41,700 (20 C)
6×10^{20}	350	1 hr at 900 C	8,530	35,000 (20 C)	43,530	41,100 (20 C)
6×10^{20}	350	1 hr at 1000 C	15,790	35,000 (20 C)	50,790	45,600 (20 C)
6×10^{20}	600	--	5,080	35,000 (20 C)	40,080	40,000 (20 C)
1.5×10^{21}(a)	280-450	--	35,900	16,800 (300 C)	52,700	53,600 (300 C)
1.5×10^{21}(a)	280-450	1 hr at 900 C	14,370	12,700 (450 C)	27,070	28,500 (450 C)
1.5×10^{21}(a)	280-450	1 hr at 950 C	8,420	12,700 (450 C)	21,120	20,800 (450 C)

(a) Hot-pressed beryllium.

TABLE 6. EFFECT OF IRRADIATION TEMPERATURE ON HARDNESS OF BERYLLIUM[(a)][(35)]

Grain Size, μ	As Received	Annealing Temperature, C 550	Annealing Temperature, C 650	Irradiation Temperature, C 450	Irradiation Temperature, C 550	Irradiation Temperature, C 650
	Hardness, VHN					
30-150	139	125	122	153	133	131
10-50	133	122	135	151	151	137
5-25	154	152	152	172	161	156
5-25	156	155	149	169	161	156
100-400	102	104	101	130	116	109
80-200	128	128	123	150	139	125

(a) Received a fast fluence of 3.5 to 5.5 x 10^{20} n/cm^2.

TABLE 7. COMPRESSION STRENGTH OF IRRADIATED BERYLLIUM[(39)]

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 x 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.

TABLE 8. MECHANICAL PROPERTIES OF IRRADIATED BERYLLIUM TESTED IN COMPRESSION[19]

Material	Yield Strength (0.2% Offset), 1000 psi	Ultimate Strength, 1000 psi	Plastic Strain[a], %
As-received samples	44.0[b]	191.7	30.1
Capsule 1 - control samples[c] (600 C - 3532 hours - 105 cycles)	46.4	210.1	30.7
Capsule 2 – control samples (750 C – 1589 hours – 55 cycles)	47.0	212.0	26.5
Capsule 3 – control samples (600 C – 1992 hours – 52 cycles)	51.2	184.5	27.8
Capsule 6 – 600 C – 3559 hours (irradiated to 1.08 x 10^{21} n/cm^2)[d]	48.5	147.9	19.6
Capsule 7 – 750 C – 1586 hours (irradiated to 0.75 x 10^{21} n/cm^2)	49.3	147.1	19.9
Capsule 8 – 600 C – 1973 hours (irradiated to 0.88 x 10^{21} n/cm^2)	64.9	174.7	25.2

(a) Plastic strain was calculated as the strain at the fracture point minus the strain at the yield point.

(b) All values are averages of three measurements.

(c) Control samples were heated at temperatures and times indicated and thermally cycled the number of cycles equivalent to reactor scrams and shut downs.

(d) All fluence values are for neutron energies > 1 MeV.

TABLE 9. BENDING PROPERTIES OF IRRADIATED BERYLLIUM[24]

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.

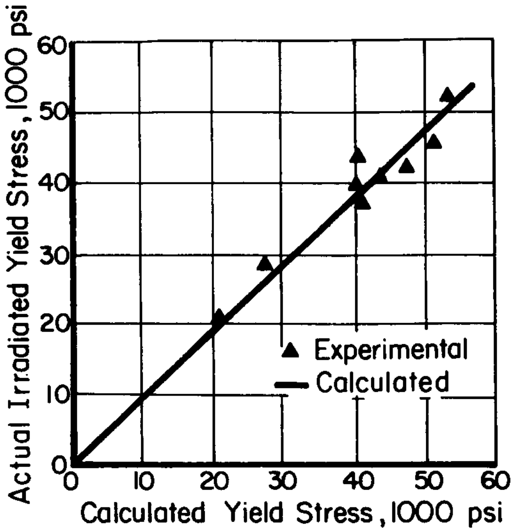

FIGURE 12. COMPARISON OF OBSERVED AND CALCULATED YIELD STRESSES OF IRRADIATED BERYLLIUM[21]

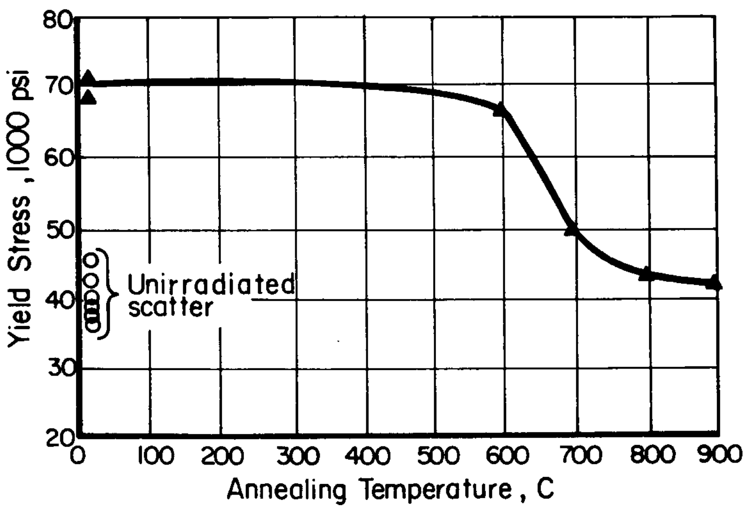

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FIGURE 13. EFFECT OF ANNEALING ON THE ROOM-TEMPERATURE YIELD STRESS OF LONGITUDINAL PECHINEY BERYLLIUM IRRADIATED TO A FAST FLUENCE OF 8 X 10^{20} N/CM^2 AT 350 C[21]

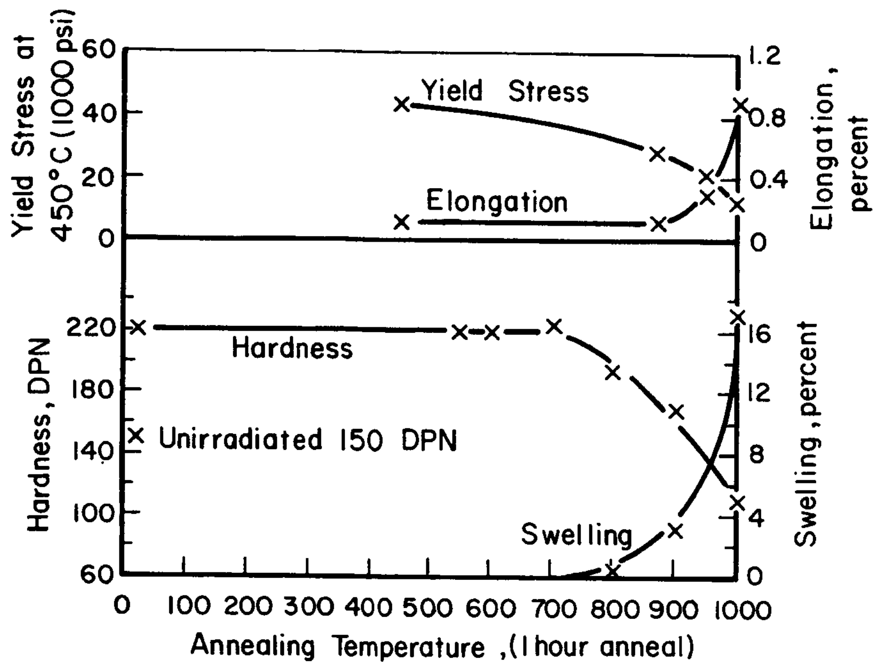


FIGURE 14. THE EFFECT OF ANNEALING ON THE HARDNESS, YIELD STRESS AND ELONGATION OF IRRADIATED BERYLLIUM

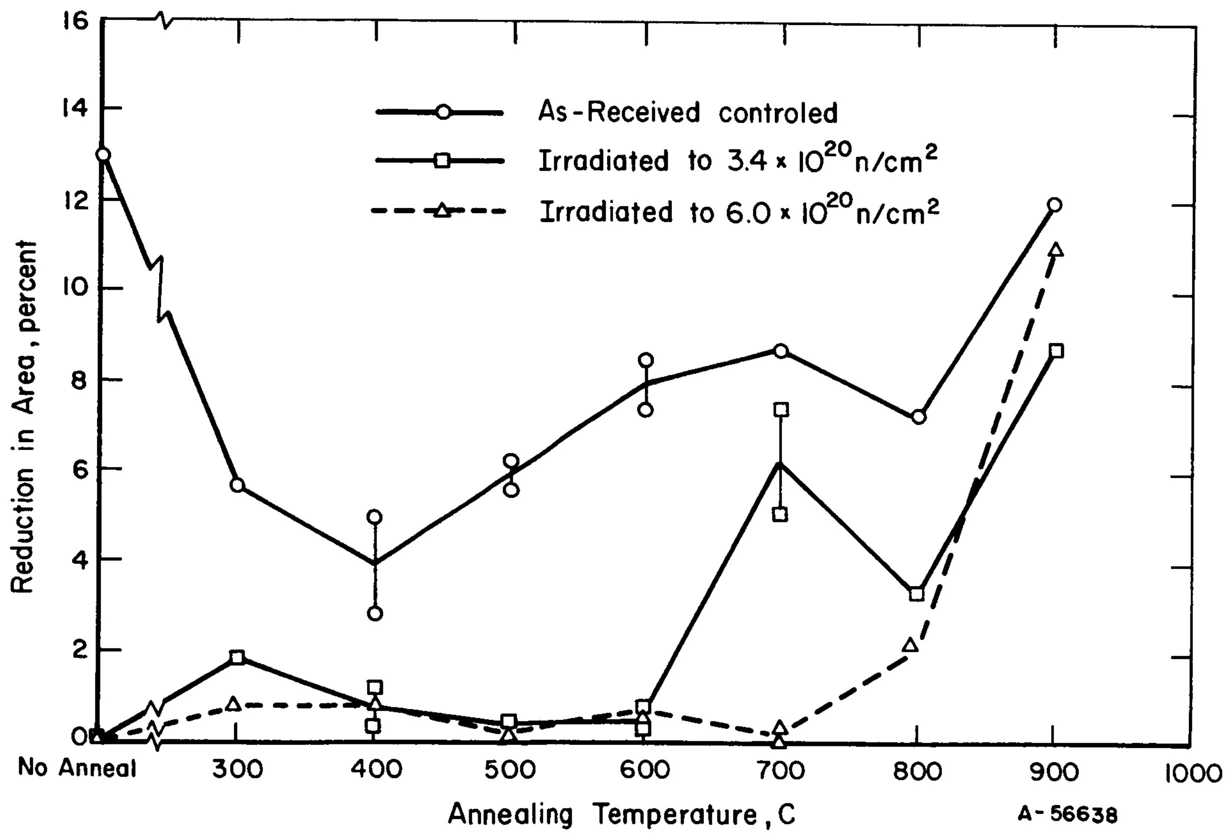


FIGURE 15. RECOVERY OF DUCTILITY WITH ANNEALING IN MATERIAL A IRRADIATED TO A FAST FLUENCE OF 3.4 X 10^{20} N/CM^2

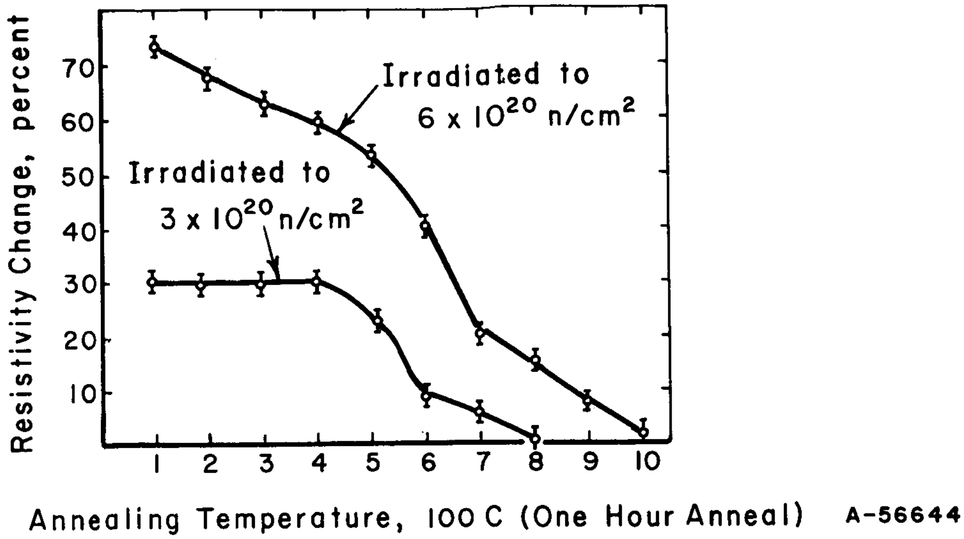


FIGURE 16. RECOVERY OF ELECTRICAL-RESISTIVITY CHANGE WITH ANNEALING FOR IRRADIATED BERYLLIUM[36]

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.

In-pile burst tests have been performed at 600 C on beryllium tubes[24] supplied by two different vendors. These tubes were 0.3 inch in diameter and had a wall thickness of 40 mils. Results of these tests are shown in Figures 17 and 18. The rupture strength of the Brush beryllium was somewhat higher owing to its higher impurity content. However, both types of tubes underwent about the same reduction in rupture life. In another series of tensile tests, the best ductility properties were exhibited by warm-extruded specimens; these lost only 10 percent of their strength, while the hot-extruded specimens lost up to 20 percent of their strength following irradiation to fast fluences of 2×10^{20} n/cm^2.[40] The warm-extruded specimens also exhibited better pre- and postirradiation strain at rupture, as shown below:

	Strain at Rupture, %	
	Irradiated	Unirradiated
Warm Extruded[a]	2-3.6	1.5-3.5
Hot Pressed[a]	0.5-2.5	1-4

(a) The temperatures were not given in the reference but generally warm extrusions are carried out at about 750 C and hot pressing at about 1000 C for beryllium.

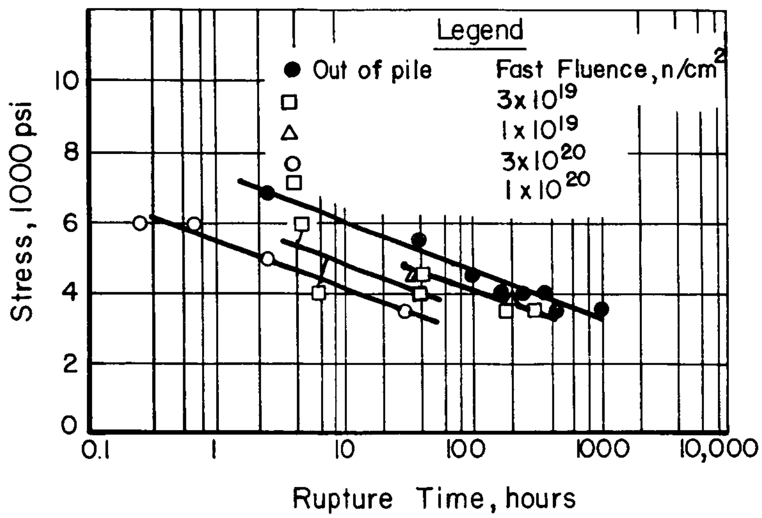


FIGURE 17. EFFECT OF IRRADIATION AT 600 C ON THE STRESS-RUPTURE PROPERTIES OF BRUSH BERYLLIUM TUBING MACHINED FROM HOT-PRESSED BLOCKS[24]

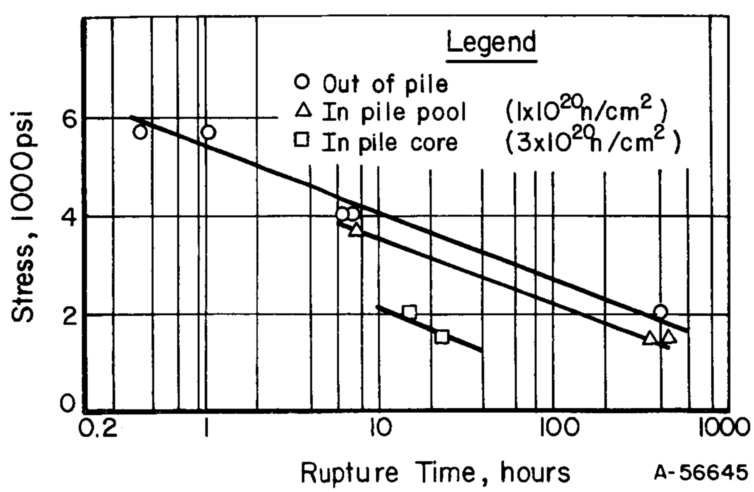


FIGURE 18. EFFECT OF IRRADIATION AT 600 C ON STRESS-RUPTURE PROPERTIES OF PECHINEY BERYLLIUM TUBING[24]

ZIRCONIUM ALLOYS

Zirconium alloys have found wide use as structural materials in the nuclear industry. 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 weight percent niobium alloy has been developed. The composition of zirconium alloys, or Zircaloys as they are called, that are used in the nuclear industry is given in Table 10.

TABLE 10. COMPOSITION OF ZIRCONIUM ALLOYS

Alloy	Composition(a), percent				Composition(b), ppm							
	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.

Tensile Properties

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. Considerable tensile data on irradiated zirconium alloys are available (see Table 11). Zirconium alloys are affected by fast-neutron irradiation much the same way as are all other materials where room-temperature strength increases and ductility decreases. The variables that affect the magnitude of radiation-induced changes in the mechanical properties of zirconium alloys are discussed below.

(1) 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 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.[58] 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

TABLE 11. EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF ZIRCONIUM ALLOYS

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				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Zirconium														
A	1.3×10^{19}	100	RT	29.3		58.6	66.2	9.8	4.4	19.8	14.3	52.1	43.8	41
A	1.6×10^{19}	100	RT	29.3	56.0	58.6	67.7	9.8		19.8		53.1	44.7	41
A	1.7×10^{19}	100	RT	29.3		58.6	66.7	9.8	3.9	19.8	13.3	53.1	41.6	41
A	1.9×10^{19}	100	RT	29.3	57.3	58.6	67.0	9.8		19.8		53.1	45.7	41
A	2.1×10^{19}	100	RT	29.3	59.8	58.6	71.3	9.8		19.8		53.1	44.1	41
A	2.7×10^{19}	100	RT	29.3		58.6	69.8	9.8	1.1	19.8	12.4	53.1	49.6	41
A	3.2×10^{19}	100	RT	29.3	58.6	58.6	66.2	9.8		19.8		53.1	46.2	41
A	3.3×10^{19}	50	RT	29.6	35.4	41.7	47.4			29.0	21.5	27.1	35.4	42
A	3.7×10^{19}	100	RT	29.3	52.2	58.6	66.2			19.8		53.1	47.0	41
A	3.8×10^{19}	100	RT	29.3		58.6	69.0	9.8	3.5	19.8	13.4	53.1	48.3	41
A	4.0×10^{19}	100	RT	29.3		58.6	68.5	9.8	3.6	19.8		53.1	49.5	41
A	4.2×10^{19}	100	RT	29.3		58.6	69.5	9.8	3.6	19.8	11.8	53.1	41.6	41
A	5.2×10^{19}	100	RT	29.3	64.1	58.6	71.8	9.8		19.8		53.1	40.6	41
A	6.3×10^{19}	100	RT	29.3		58.6	72.6	9.8	2.0	19.8	10.7	53.1	39.2	41
A	6.5×10^{19}	100	RT	29.3	56.0	58.6	66.9			19.8	13.8	53.1	43.7	41
A	7.0×10^{19}	100	RT	29.3		58.6	74.6	9.8	2.0	19.8	10.1	53.1	46.0	41
A	8.5×10^{19}	100	RT	29.3	52.2	58.6	66.2	9.8		19.8		53.1	47.0	41
A	1.0×10^{20}	100	RT	29.3	58.6	58.6	68.6			19.8	12.4	53.1	46.4	41
A	2.0×10^{20}	100	RT	29.3	64.1	58.6	69.7			19.8	12.6	53.1	45.0	41
A	2.9×10^{20}	100	RT	29.3		58.6	73.6			19.8	10.4	53.1	42.6	41
Zircaloy - 2														
A-TD	5.1×10^{19}	280	RT	54.9	71.0	63.3	71.8	11.9	1.1	32.4	11.7			43
A-RD	5.3×10^{19}	280	RT	47.4	69.0	66.5	77.1	14.2	4.4	28.7	10.1			43
A	5.9×10^{19}	<100	RT	56.4	74.2	77.4	89.5			29.0	19.0	44.0	44.0	4
A	7.7×10^{19}	<100	RT	56.4	84.4	77.4	83.3			29.0	19.0	44.0	44.0	4
A	1.1×10^{20}	<100	RT	41.0	72.0	68.0	78.0			21.0	15.0	40.0	36.0	44
A-RD	1.1×10^{20}	60	RT	47.9	72.1	68.0	78.3	16.8	4.2	33.0	14.0	51.0	49.0	45
A-TD	1.1×10^{20}	60	RT	54.4	77.5	63.1	77.5	12.4	0.8	34.4	6.0	55.0		45
A	1.2×10^{20}	<100	RT	56.4	73.9	77.4	89.6			29.0	19.0	44.0	44.0	4
A-RD	1.1×10^{20}	280	RT	47.4	70.8	66.5	78.5	14.2	4.2	28.7	11.2			43
A-TD	1.2×10^{20}	280	RT	54.9	80.3	63.3	80.5	11.9	0.8	32.4	6.2			43
A-TD	1.3×10^{20}	280	RT	54.9	82.0	63.3	82.2	11.9	1.0	32.4	8.7			43
A-TD	1.3×10^{20}	280	RT	47.4	72.5	66.5	79.0	14.2	4.2	28.7	10.4			43
A-TD	2.0×10^{20}	280	RT	54.9	76.0	63.3	76.4	11.9	0.9	32.4	11.5			43
A-RD	3.0×10^{20}	280	RT	47.4	68.4	66.5	76.7	14.2	5.0	28.7	16.8			43
A	3.6×10^{20}	<100	RT	52.9	90.2	81.5	96.0			29.0	18.0	37.0	40.0	46

TABLE 11. (Continued)

Material Condition(a)	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp, C	Test Temp, C	Offset Yield Strength, 1000 psi Unirr.	Offset 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
A-TD	3.8×10^{20}	280	RT	54.9	96.5	63.3	96.6	11.9	0.8	32.4	4.6			43
A-RD	4.0×10^{20}	280	RT	47.4	77.8	66.5	82.0	14.2	3.2	28.7	9.2			43
A-TD	5.9×10^{20}	280	RT	54.9	91.8	63.3	91.9	11.9	0.8	32.4	5.0			43
A	1.0×10^{21}	95-120	RT	45.3	82.0	70.5	89.2	16.7	3.2	26.3	12.3	39.9	39.1	47
A	1.0×10^{21}	260	RT	44.3	78.9	68.6	81.5	16.0	2.1	33.4	12.5	41.6	29.2	48
A	1.2×10^{21}	175	RT	49.2	83.3	75.0	91.6	15.0	4.5	28.7	13.4			49
A	1.2×10^{21}	175	RT	49.2	86.8	75.0	93.8	15.0	4.1	28.7	13.0			49
A	1.4×10^{21}	95-120	RT	45.3	76.0	70.5	82.0	16.7	3.1	26.3	11.8	39.9	39.3	47
A	6.0×10^{21}	260	RT	44.3	93.8	68.6	95.9	16.0	1.3	33.4	10.1	41.6	27.5	48
A	1.0×10^{21}	260	205	31.0	71.5	46.1	71.5	14.2	0.4	34.9	14.7	53.3	42.0	48
A	6.0×10^{21}	260	205	31.0	90.0	46.1	91.0	14.2	0	34.9	11.6	53.3	39.5	48
A-TD	5.1×10^{19}	280	300	17.8	36.9	26.5	37.1							43
A-RD	5.3×10^{19}	280	300	18.7	36.5	31.6	37.3	20.3		23.9				43
A-RD	1.1×10^{20}	280	300	18.7	38.2	31.6	38.2	20.3		23.9				43
A-TD	1.2×10^{20}	280	300	17.8	47.6	26.5	47.6							43
A-TD	1.3×10^{20}	280	300	17.8	38.4	26.5	38.7							43
A-RD	1.3×10^{20}	280	300	18.7	39.4	31.6	39.5	20.3		23.9				43
A-TD	2.0×10^{20}	280	300	17.8	40.0	26.5	40.4							43
A-RD	3.0×10^{20}	280	300	18.7	39.2	31.6	39.3	20.3		23.9				43
A-TD	3.8×10^{20}	280	300	17.8	45.4	26.5	45.4							43
A-RD	4.0×10^{20}	280	300	18.7	44.6	31.6	44.7	20.3		23.9				43
A-TD	5.9×10^{20}	280	300	17.8	47.6	26.5	47.6							43
A	1.0×10^{21}	260	315	22.4	61.2	34.6	61.2	11.7	0	35.4	14.5	59.9	46.7	48
A	6.0×10^{21}	260	315	22.4	73.2	34.6	73.2	11.7	0	35.4	12.6	59.9	38.4	48
10 CW-RD	0.84×10^{19}	60	RT	73.4	85.0	81.6	87.5	2.3	1.5	11.3	6.5	46.0		45
10 CW-TD	0.84×10^{19}	60	RT	69.4	83.9	79.9	87.9	2.7	1.7	11.6	4.3	52.0	50.0	45
10 CW-RD	1.1×10^{20}	60	RT	73.4	92.0	81.6	92.4	2.3	1.1	11.3	4.6	46.0		45
10 CW-TD	1.1×10^{20}	60	RT	69.4	90.9	79.9	91.2	2.7	1.0	11.6	3.9	52.0	45.0	45
10 CW-RD	1.1×10^{21}	60	RT	73.4	91.0	81.6	91.0	2.3	0.9	11.3	5.2	46.0	41.0	45
10 CW-TD	1.1×10^{21}	60	RT	69.4	92.3	79.9	92.3	2.7	0.9	11.6	4.8	52.0	49.0	45
15 CW, A-650	5.6×10^{18}	<100	RT	32.8	46.1	54.5	59.0	14.8	5.6	26.8	20.9			50
15 CW, A-650	5.6×10^{18}	<100	RT		41.5		52.4		6.7		13.3			50
15 CW	5.6×10^{18}	40	RT	69.1	89.8	74.0	86.0	2.5	1.1	12.8	6.3			50
15 CW	5.6×10^{18}	40	RT	62.3	69.6	70.8	76.2	3.7	3.1	11.4	11.6			50
15 CW	5.6×10^{18}	40	RT		60.6		72.1		5.8		14.6			50
15 Cw, A-650	5.6×10^{19}	40	RT	34.0	50.4	55.0	64.5	14.2	5.7	24.0	25.9			50
15 CW, A-650	5.6×10^{19}	40	RT	34.0	42.4	59.6	59.6	14.4	9.1	24.4	18.7			50
15 CW, A-650	5.6×10^{19}	40	RT	34.0	37.0	59.6	57.0	14.4	13.7	24.4	22.6			50

TABLE 11. (Continued)

Material Condition(a)	Fast Fluence, n/cm^2 (>1 MeV)	Irr. Temp, C	Test Temp, C	Offset Yield Strength, 1000 psi Unirr.	Offset 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
15 CW, A-650	5.6×10^{19}	40	RT	32.8	52.6	54.5	63.6	14.8	5.2	26.8	20.0			50
15 CW	5.6×10^{19}	40	RT	75.5	90.7	79.5	91.5	1.8	0.7	9.2	4.9			50
15 CW, A-650	5.6×10^{18}	<100	80	27.6	41.0	46.8	50.9	14.1	4.4	26.5	19.3			50
15 CW, A-650	5.6×10^{18}	<100	80		36.4		44.7		6.7		16.2			50
15 CW, A-650	5.6×10^{19}	40	80	27.6	46.7	46.8	57.5	14.1	4.1	26.5	14.9			50
15 CW	5.6×10^{19}	40	80	68.0	84.9	71.0	85.5	1.7	0.7	8.7	4.4			50
18 CW	6.2×10^{19}	50	RT		84.8		91.0		4.0		8.6		35.8	51
18 CW	6.9×10^{19}	280	RT	56.7	85.2	76.9	92.1	10.7	3.1	16.6	5.7	38.8	32.8	51
18 CW	6.9×10^{19}	280	RT	58.8	84.3	79.7	93.1	11.9	4.0	21.0	8.0	38.9	32.9	51
18 CW	1.1×10^{20}	280	RT		89.9		96.0		4.0		7.5		32.6	51
18 CW	1.6×10^{20}	280	RT		93.1		98.4		3.1		6.4		31.5	51
18 CW	3.5×10^{20}	280	RT	60.3	101.3	81.1	104.2	9.1	1.2	16.2	3.0	37.3	28.1	51
18 CW	5.6×10^{20}	280	RT		104.8		107.0		1.8		4.7		29.3	51
18 CW	9.5×10^{20}	280	RT	61.3	105.9	81.9	109.2	9.5	1.4	16.6	3.6	35.7	26.3	51
18 CW	1.3×10^{21}	280	RT	59.4	107.8	79.5	110.4	9.0	1.6	13.4	3.3	36.0	26.8	51
18 CW	2.2×10^{21}	280	RT	61.1	106.4	81.8	107.3	9.0	1.3	16.3	2.9	43.2	35.1	51
18 CW	6.2×10^{19}	50	300		50.5		51.4		1.2		4.6		46.4	51
18 CW	1.1×10^{20}	280	300		53.8		53.9		0.7		4.3		43.1	51
18 CW	1.6×10^{20}	280	300		56.2		57.1		1.0		4.3		36.3	51
18 CW	5.6×10^{20}	280	300		63.4		65.3		1.1		2.8		34.3	51
18 CW	1.3×10^{21}	280	300	38.0	68.4	45.6	72.5	4.0	1.0	8.2	2.3	52.0	32.5	51
18 CW	2.2×10^{21}	280	300	38.0	73.9	45.8	75.2	4.0	1.0	9.6	1.9	48.5	34.6	51
20 CW-RD	0.84×10^{19}	60	RT	79.2	91.8	88.0	96.4	2.6	1.7	9.6	6.4	42.0		45
20 CW-RD	0.84×10^{19}	60	RT	73.0	88.3	83.9	93.7	2.5	1.7	9.3	4.5	52.0		45
20 CW-TD	5.1×10^{19}	280	RT	76.7	89.4	81.2	92.2	2.7	1.2	10.9	4.2			43
20 CW-RD	5.3×10^{19}	280	RT	74.1	91.4	82.8	93.2	4.4	3.7	13.9	7.4			43
20 CW-RD	1.1×10^{20}	280	RT	74.1	93.3	82.8	95.3	4.4	1.7	13.9	5.7			43
20 CW-RD	1.1×10^{20}	60	RT	79.2	95.9	88.0	97.1	2.6	1.3	9.6	5.3	42.0	36.0	45
20 CW-TD	1.1×10^{20}	60	RT	73.0	94.4	83.9	96.1	2.5	1.2	9.3	3.2	52.0	41.0	45
20 CW-TD	1.2×10^{20}	280	RT	76.7	92.4	81.2	92.2	2.7	0.9	10.9	4.0			43
20 CW-TD	1.3×10^{20}	280	RT	76.7	91.8	81.2	92.6	2.7	2.7	10.9	6.3			43
20 CW-RD	1.3×10^{20}	280	RT	74.1	90.0	82.8	93.0	4.4	2.5	13.9	6.5			43
20 CW-TD	2.0×10^{20}	280	RT	76.7	88.4	81.2	88.8	2.7	1.0	10.9	6.0			43
20 CW-RD	3.0×10^{20}	280	RT	74.1	82.0	82.8	87.2	4.4	3.3	13.9	8.7			43
20 CW-TD	3.8×10^{20}	280	RT	76.7	94.7	81.2	97.7	2.7	0.9	10.9	3.6			43
20 CW-RD	4.0×10^{20}	280	RT	74.1	92.2	82.8	93.4	4.4	1.2	13.9	3.9			43
20 CW-TD	5.9×10^{20}	280	RT	76.7	94.2	81.2	94.7	2.7	1.1	10.9	1.9			43
20 CW-RD	1.1×10^{21}	60	RT	79.2	96.2	88.0	96.6	2.6	1.2	9.6	5.8	42.0	40.0	45

TABLE 11. (Continued)

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				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
20 CW-TD	1.1×10^{21}	60	RT	73.0	98.0	83.9	98.3	2.5	1.0	9.3	3.6	52.0	45.0	45
20 CW-TD	5.1×10^{19}	280	300	46.7	52.6	48.8	52.7							43
20 CW-RD	5.3×10^{19}	280	300	48.8	53.2	49.6	53.3	4.2		9.4				43
20 CW-RD	1.1×16^{20}	280	300	48.8	54.2	49.6	54.2	4.2		9.4				43
20 CW-TD	1.2×10^{20}	280	300	46.7	55.1	48.8	56.2							43
20 CW-TD	1.3×10^{20}	280	300	46.7	52.7	48.8	52.8							43
20 CW-RD	1.3×10^{20}	280	300	48.8	53.1	49.6	53.2	4.2		9.4				43
20 CW-TD	2.0×10^{20}	280	300	46.7	52.3	48.8	52.4							43
20 CW-TD	3.8×10^{20}	280	300	46.7	57.8	48.8	58.1							43
20 CW-RD	4.0×10^{20}	280	300	48.8	58.4	49.6	58.4	4.2		9.4				43
20 CW-RD	5.9×10^{20}	280	300	46.7	58.2	48.8	58.2							43
23 CW, A-605	5.6×10^{19}	40	RT	30.6	53.0	56.1	67.2	15.8	5.1	27.8	17.4			50
23 CW	5.6×10^{19}	40	RT	75.6	93.6	78.8	94.9	1.5	0.5	8.4	4.9			50
23 CW, A-605	5.6×10^{19}	40	80	27.7	45.5	48.6	56.3	15.4	4.7	29.9	14.8			50
23 CW	5.6×10^{19}	40	80	70.0	85.3	74.0	85.4	0.8	0.6	8.2	4.1			50
28 CW	6.9×10^{19}	280	RT	65.2	89.5	88.2	98.8	9.5	3.6	15.0	5.4	34.5	29.4	51
28 CW	6.9×10^{19}	280	RT	64.4	89.2	88.4	99.6	9.3	4.2	15.4	7.9	35.2	33.4	51
28 CW	1.1×10^{20}	50	RT		90.6		98.4		4.0		8.2		29.0	51
28 CW	1.1×10^{20}	280	RT		91.8		99.8		3.3		6.3		29.8	51
28 CW	1.6×10^{20}	280	RT		97.5		105.1		3.3		5.3		27.2	51
28 CW	3.5×10^{20}	280	RT	63.8	103.9	87.1	107.7	8.3	2.0	11.5	3.2	37.2	28.1	51
28 CW	5.6×10^{20}	280	RT		104.6		111.7		2.8		4.7		27.6	51
28 CW	9.5×10^{20}	280	RT	67.0	110.0	90.1	114.1	8.3	1.8	11.6	3.6	34.2	24.1	51
28 CW	1.3×10^{21}	280	RT	63.6	109.0	86.7	115.6	8.6	2.6	12.8	4.4	33.5	23.8	51
28 CW	2.2×10^{21}	280	RT	64.7	115.6	90.9	116.7	8.3	1.7	11.9	2.7	35	26.1	51
28 CW	1.1×10^{20}	50	300		56.6		59.2		2.2		5.3		38.4	51
28 CW	1.1×10^{20}	280	300		58.5		60.1		2.2		5.7		37.8	51
28 CW	1.6×10^{20}	280	300		60.0		62.3		1.7		3.8		27.7	51
28 CW	5.6×10^{20}	280	300		62.5		68.6		1.6		3.9		31.7	51
28 CW	1.3×10^{21}	280	300	43.6	73.1	53.2	78.4	3.6	1.2	6.6	2.2	47.1	24.5	51
28 CW	2.2×10^{21}	280	300	43.6	73.0	54.6	76.5	3.0	1.3	6.2	2.1	40.8	32.1	51
29 CW	9.8×10^{19}	50	RT		94.8		97.9		2.4		6.4		37.6	51
29 CW	9.8×10^{19}	50	300		57.0		58.7		1.0		3.8		49.8	51
30 CW	6.9×10^{19}	280	RT	68.9	93.8	88.4	102.4	8.9	3.8	18.0	8.3	45	33.3	51
30 CW	6.9×10^{19}	280	RT	68.5	94.5	89.5	101.8	9.9	3.4	17.9	6.9	39.2	33.0	51
30 CW	1.1×10^{20}	280	RT		97.2		103.8		3.8		7.6		33.4	51
30 CW	1.2×10^{20}	50	RT		98.2		103.8		3.8		7.8		36.5	51
30 CW	1.6×10^{20}	280	RT		101.4		106.4		3.6		6.9		32.7	51

TABLE 11. (Continued)

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				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
30 CW	3.5×10^{20}	280	RT	68.8	105.8	88.4	107.4	8.9	2.3	17.0	4.9	38.1	31.6	51
30 CW	5.6×10^{20}	280	RT		109.4		113.0		2.4		4.4		32	51
30 CW	9.5×10^{20}	280	RT	69.5	113.6	89.8	114.4	9.6	2.0	18.2	5.1	38.8	28.4	51
30 CW	1.3×10^{21}	280	RT	68.0	110.8	88.4	116.4	9.3	1.8	16.2	3.5	35.4	23.0	51
30 CW	2.2×10^{21}	280	RT	66.8	119.7	91.1	120.5	8.0	1.4	11.9	4.2	40.4	29.7	51
30 CW	1.1×10^{20}	280	300		59.3		59.3		0.7		4.7		40.6	51
30 CW	1.2×10^{20}	50	300		59.8		61.1		1.3		3.5		50.1	51
30 CW	1.6×10^{20}	280	300		64.8		66.5		1.1		4.1		39.7	51
30 CW	5.6×10^{20}	280	300		67.1		69.2		1.3		3.4		40.2	51
30 CW	1.3×10^{21}	280	300	42.5	72.1	52.0	74.0	4.8	1.1	9.1	1.9	52.0	31.7	51
30 CW	2.2×10^{21}	280	300	44.0	75.4	53.4	77.9	4.2	1.2	8.8	2.0	47.6	32.4	51
40 CW-RD	0.84×10^{19}	60	RT	86.0	92.9	96.3	99.7	3.0	2.2	8.7	6.8	39	41	45
40 CW-TD	0.84×10^{19}	60	RT	80.3	92.0	92.4	98.8	2.5	2.0	8.3	4.9	53		45
40 CW-TD	5.1×10^{19}	280	RT	81.3	96.2	86.6	97.9	1.9	1.5	8.6	4.5			43
40 CW-RD	5.3×10^{19}	280	RT	78.6	93.2	89.4	96.6	4.7	2.3	13.8	6.9			43
40 CW-RD	1.1×10^{20}	60	RT	86.0	99.7	96.3	104.0	3.0	1.5	8.7	3.6	39	28	45
40 CW-TD	1.1×10^{20}	60	RT	80.3	99.6	92.4	103.0	2.5	1.5	8.3	3.0	53	32	45
40 CW-RD	1.1×10^{20}	280	RT	78.6	92.2	89.4	98.8	4.7	2.5	13.8	6.3			43
40 CW-TD	1.2×10^{20}	280	RT	81.3	103.2	86.6	105.4	1.9	1.2	8.6	3.7			43
40 CW-TD	1.3×10^{20}	280	RT	81.3	97.4	86.6	99.2	1.9	1.2	8.6	3.9			43
40 CW-RD	1.3×10^{20}	280	RT	78.6	97.2	89.4	100.3	4.7	2.6	13.8	6.7			43
40 CW-TD	2.0×10^{20}	280	RT	81.3	96.0	86.6	97.2	1.9	1.2	8.6	5.0			43
40 CW-RD	3.0×10^{20}	280	RT	78.6	88.6	89.4	93.8	4.7	3.1	13.8	7.4			43
40 CW-TD	3.8×10^{20}	280	RT	81.3	105.8	86.6	107.0	1.9	1.2	8.6	3.6			43
40 CW-RD	4.0×10^{20}	280	RT	78.6	96.8	89.4	99.2	4.7	2.2	13.8	5.9			43
40 CW-RD	1.1×10^{21}	60	RT	86.0	98.3	96.3	101.0	3.0	1.8	8.7	5.0	39	39	45
40 CW-TD	1.1×10^{21}	60	RT	80.3	105.0	92.4	106.0	2.5	1.2	8.3	4.0	53	41	45
40 CW-TD	5.1×10^{19}	280	300	52.2	60.0	55.2	62.2	5.6		10.6				43
40 CW-RD	5.3×10^{19}	280	300	55.9	57.6	58.4	58.2	5.9		10.1				43
40 CW-RD	1.1×10^{20}	280	300	55.9	59.3	58.4	59.6	5.9		10.1				43
40 CW-TD	1.2×10^{20}	280	300	52.2	63.2	55.2	63.8	5.6		10.6				43
40 CW-TD	1.3×10^{20}	280	300	52.2	61.7	55.2	62.2	5.6		10.6				43
40 CW-RD	1.3×10^{20}	280	300	55.9	58.0	58.4	58.7	5.9		10.1				43
40 CW-TD	2.0×10^{20}	280	300	52.2	61.4	55.2	62.0	5.6		10.6				43
40 CW-RD	3.0×10^{20}	280	300	55.9	54.0	58.4	54.9	5.9		10.1				43
40 CW-TD	3.8×10^{20}	280	300	52.2	67.0	55.2	67.2	5.6		10.6				43
40 CW-RD	4.0×10^{20}	280	300	55.9	62.2	58.4	62.4	5.9		10.1				43
70 CW-RD	0.84×10^{19}	60	RT	92.8	103.0	109.0	114.0	3.4	2.6	8.6	4.9	34		45
70 CW-TD	0.84×10^{19}	60	RT	85.6	102.0	104.0	112.0	3.6	2.5	7.1	4.7	45		45

TABLE 11. (Continued)

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				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
70 CW-RD	1.1×10^{20}	60	RT	92.8	105.0	109.0	112.0	3.4	2.1	8.6	5.4	34	29	45
70 CW-TD	1.1×10^{20}	60	RT	85.6	108.0	104.5	116.0	3.6	1.8	7.1	2.7	45	21	45
70 CW-RD	1.1×10^{21}	60	RT	92.8	104.0	109.0	111.0	3.4	3.3	8.6	6.3	34	30	45
70 CW-TD	1.1×10^{21}	60	RT	85.6	111.0	104.0	114.0	3.6	1.4	7.1	3.4	45	23	45
CR	0.8×10^{16}	250	RT	65.2	65.3	71.2	71.4	7.6	7.4	23.5	24.1			52
?	0.8×10^{16}	~250	RT	55.4	54.9	73.5	73.6	16.9	15.5	29.0	25.8			52
?	0.8×10^{16}	~250	RT	57.5	59.0	67.0	66.5	6.3	4.0	12.2	9.7			52
CR	1.3×10^{17}	~250	RT	65.2	65.1	71.2	70.9	7.6	7.4	23.5	22.1			52
CR	2.6×10^{18}	250-300	RT	65.2	68.3	71.2	71.8	7.6	6.9	23.5	20.6			52
R	2.6×10^{11}	250-300	RT	55.4	58.1	73.5	74.9	16.9	15.8	29.0	25.5			52
BQ	2.6×10^{18}	250-300	RT	66.6	73.6	81.1	86.0	11.6	10.3	21.7	17.0			52
?	5.6×10^{18}	>100	RT	36.8	58.4	60.5	62.8	21.4	19.2	30.0	28.2			50
EX	5.6×10^{19}	40	RT	38.7	62.2	61.0	63.6	21.5	16.8	28.4	28.2			50
EX, A-750	5.6×10^{19}	40	RT	36.8	60.0	60.5	63.0	21.4	18.1	30.0	27.9			50
EX	5.6×10^{19}	40	RT	38.7	51.1	61.0	64.2	20.3	18.0	26.9	25.6			50
EX	5.6×10^{19}	40	RT	39.6	42.4	62.4	64.6	20.6	20.0	28.5	26.6			50
EX, A-750	5.6×10^{19}	40	RT	39.5	58.0	64.9	65.3	21.7	18.4	29.0	27.4			50
EX, A-750	5.6×10^{19}	40	RT	38.4	50.8	62.5	63.1	22.1	18.4	30.5	28.7			50
EX	5.6×10^{19}	40	80	32.6	48.7	52.8	53.2	20.2	17.8	28.6	29.4			50
EX	5.6×10^{19}	40	80	33.0	47.6	54.9	52.4	21.2	18.3	30.0	27.4			50
T-S	0.9×10^{21}	>100	RT	48.8	92.4	70.8	98	11.7	0.5	19.9	2.9			53
T-S	0.9×10^{21}	>100	RT	48.8	78.6	70.8	100	11.7	1.1	19.9	5.1			53
T-S	1.0×10^{21}	>100	RT	48.8	77.6	70.8	88.6	11.7	--	19.9	--			53
T-S	1.0×10^{21}	>100	RT	48.8	77.6	70.8	90.9	11.7	0.8	19.9	4.6			53
T-S	1.0×10^{21}	>100	RT	48.8	91.4	70.8	98.2	11.7	0.5	19.9	2.1			53
T-S	1.0×10^{21}	>100	RT	48.8	66.8	70.8	106	11.7	2.5	19.9	5.3			53
T-S	1.1×10^{21}	>100	RT	48.8	69.5	70.8	106.3	11.7	2.0	19.9	5.3			53
T-S	1.3×10^{21}	>100	RT	48.8	87.4	70.8	89	11.7	0.4	19.9	1.9			53
T-S	1.3×10^{21}	>100	RT	48.8	72	70.8	92.4	11.7	1.0	19.9	4.1			53
T-S	1.3×10^{21}	>100	RT	48.8	73.1	70.8	86.2	11.7	0.6	19.9	2.0			53
T-S	1.3×10^{21}	>100	RT	48.8	--	70.8	92.3	11.7	--	19.9	--			53
T-S	1.4×10^{21}	>100	RT	48.8	104	70.8	107.4	11.7	0.4	19.9	2.0			53
T-S	1.4×10^{21}	>100	RT	48.8	76.7	70.8	96.9	11.7	1.0	19.9	2.8			53
T-S	1.4×10^{21}	>100	RT	48.8	58.7	70.8	98.7	11.7	2.8	19.9	5.1			53
T-S	1.4×10^{21}	>100	RT	48.8	101	70.8	103.6	11.7	0.4	19.9	2.8			53
T-S	0.7×10^{21}	>100	345	20.5	44.4	31.5	56.6	13.1	0.8	23.3	2.3			53
T-S	0.7×10^{21}	>100	345	20.5	55.2	31.5	56.5	13.1	0.4	23.3	1.6			53
T-S	0.7×10^{21}	>100	345	20.5	58.5	31.5	58.6	13.1	0.2	23.3	1.9			53
T-S	1.0×10^{21}	>100	345	20.5	55.9	31.5	56.2	13.1	0.2	23.3	1.9			53

TABLE 11. (Continued)

Material Condition(a)	Fast Fluence, n/cm² (>1 MeV)	Irr. Temp, C	Test Temp, C	Offset Yield Strength, 1000 psi Unirr.	Offset 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
T-S	1.0×10^{21}	>100	345	20.5	61.7	31.5	62.0	13.1	0.2	23.3	1.4			53
T-S	1.2×10^{21}	>100	345	20.5	59.7	31.5	59.7	13.1	0.2	23.3	1.6			53
T-S	1.2×10^{21}	>100	345	20.5	61.2	31.5	61.2	13.1	0.2	23.3	1.6			53
T-S	1.3×10^{21}	>100	345	20.5	66.0	31.5	66.5	13.1	0.3	23.3	2.1			53
T-S	1.3×10^{21}	>100	345	20.5	57.2	31.5	57.2	13.1	0.2	23.3	1.7			53
T-S	1.3×10^{21}	>100	345	20.5	61.6	31.5	61.6	13.1	0.1	23.3	1.3			53
T-S	1.3×10^{21}	>100	345	20.5	57.0	31.5	58.1	13.1	0.3	23.3	1.8			53
T-S	1.4×10^{21}	>100	345	20.5	55.2	31.5	55.2	13.1	0.2	23.3	1.3			53
T-S	1.5×10^{21}	>100	345	20.5	51.4	31.5	52.7	13.1	0.4	23.3	1.0			53
T-S	1.5×10^{21}	>100	345	20.5	64.2	31.5	64.5	13.1	0.2	23.3	1.5			53
T-S	1.2×10^{21}	175	RT	49.3	83.3	73.0	91.5	16.4	6.8	30.8	17.3			54
T-S	1.2×10^{21}	175	RT	49.3	87.0	73.0	93.8	16.4	5.6	30.8	15.0			54
T-S	2.4×10^{21}	175	RT	49.3	93.7	73.0	100	16.4	5.6	30.8	12.1			54
T-S	2.4×10^{21}	175	RT	49.3	94.7	73.0	101	16.4	6.1	30.8	13.4			54
T-S	2.4×10^{21}	175	RT	49.3	103.1	73.0	107	16.4	5.6	30.8	10.1			54
T-S	2.4×10^{21}	175	RT	49.3	103.3	73.0	107.8	16.4	5.7	30.8	11.2			54
T-S	1.2×10^{21}	175	315	22.4	46.6	34.6	46.6	11.7	2.1	35.4	12.5			54
T-S	1.2×10^{21}	175	315	22.4	48.9	34.6	51	11.7	2.7	35.4	11.3			54
T-S	2.4×10^{21}	175	315	22.4	47.6	34.6	47.6	11.7	2.3	35.4	9.6			54
T-S	2.4×10^{21}	175	315	22.4	52.4	34.6	52.4	11.7	2.4	35.4	10.0			54
T-S	2.4×10^{21}	175	315	22.4	52.1	34.6	52.1	11.7	1.9	35.4	11.6			54
T-S	2.4×10^{21}	175	315	22.4	51.2	34.6	51.7	11.7	2.3	35.4	13.4			54
A	1.0×10^{21}	95-120	290	21.2	49.2	40.6	49.2	15.7	0.6	27.0	13.7	48.0	50.9	47
A	1.4×10^{21}	95-120	290	21.2	48.8	40.6	49.1	15.7	0.6	27.0	12.9	48.0	45.1	47
A	2.5×10^{21}	95-120	290	21.2	59.0	40.6	59.0	15.7	0.2	27.0	10.0	48.0	42.4	47
19 CW	1.1×10^{20}	<100	RT	41.6	66.0	70.0	83.0	16.5	4.5	28.0	14.0	46.2	37.5	44
19 CW	1.1×10^{20}	<100	RT	75.5	92.4	88.4	101.9	4.6	3.0	12.4	8.8	36.1	29.6	44
32 CW	1.1×10^{20}	<100	RT	87.6	103.7	96.5	109.3	2.9	2.5	10.4	6.7	27.4	18.0	44
EX	1.0×10^{21}	95-100	150	53.8	82.1	73.7	87.4	9.3	1.5	21.4	11.2	50.5	49.2	47
EX	1.4×10^{21}	95-120	150	53.8	85.8	73.7	88.0	9.3	0.6	21.4	9.4	50.5	49.7	47
EX	2.5×10^{21}	95-120	150	53.8	93.8	73.7	95.0	9.3	0.7	21.4	8.8	50.5	46.3	47
EX, A-750	1.0×10^{21}	95-120	RT	48.1	79.6	73.3	90.0	15.0	3.7	22.9	11.4	41.5	36.6	47
EX, A-750	1.4×10^{21}	95-120	RT	48.1	80.6	73.3	90.0	15.0	3.6	22.9	11.4	41.5	32.3	47
EX, A-750	1.0×10^{21}	95-120	150	35.1	67.1	56.5	70.8	13.0	2.2	23.9	12.0	42.1	39.1	47
WEX, A-750	1.4×10^{21}	95-120	150	35.1	69.0	56.5	72.2	13.0	2.2	23.9	11.6	42.1	39.9	47
WEX, A-750	2.5×10^{21}	95-120	150	35.1	74.6	56.5	76.0	13.0	1.1	23.9	9.3	42.1	39.9	47
T-S	5.6×10^{20}	285	RT	48.9	76.4	70.2	93	11.2	0.88	17.9	4.3			55
T-S	5.6×10^{20}	285	RT	48.9	80.9	70.2	96.3	11.2	0.75	17.9	2.5			55
T-S	8.2×10^{20}	285	RT	48.9	78.6	70.2	94	11.2	0.55	17.9	1.1			55

TABLE 11. (Continued)

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				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
T-S	8.5×10^{20}	285	RT	48.9	85.7	70.2	97.2	11.2	1.63	17.9	2.4			55
T-S	2.1×10^{20}	285	345	26.1	36.6	35.6	46.1	2.1	1.0	7.9	3.8			55
T-S	4.4×10^{20}	285	345	26.1	57.1	35.6	57.1	2.1	0.19	7.9	3.8			55
T-S	4.4×10^{20}	285	345	26.1	52	35.6	52	2.1	0.18	7.9	1.8			55
T-S	7.2×10^{20}	285	345	26.1	53	35.6	53	2.1	0.14	7.9	1.8			55
T-S	7.2×10^{20}	285	345	26.1	52.4	35.6	53	2.1	0.38	7.9	1.0			55
T-S	9.0×10^{20}	285	345	26.1	56.3	35.6	58.5	2.1	0.44	7.9	2.3			55
T-S	9.0×10^{20}	285	345	26.1	55.4	35.6	55.4	2.1	0.15	7.9	1.8			55
Zirconium 2.5 wt. % Niobium														
Q-B, A-500	3×10^{20}	250-325	RT	113	153	170	160	4	<1	10	1	>25	>5	56
Q-B, A-500	1×10^{20}	250	300	85	97	93	102			14.2	13.3			57
Q-B, A-500	3×10^{20}	250-325	300	81	114	133	126	3	1.5	12	2	56	<20	56
SC-B, 40 CW	3×10^{20}	250-325	RT	97	131	122	152	1	1			30	26	56
SC-B, 40 CW	3×10^{20}	250-325	300	60	101	90	116	1	1			34	26	56
Q-B, A-500	1×10^{20}	250	RT		136		142				10.5			57
SC-(A + B)	3×10^{20}	250-325	RT	61	107	132	150	12	3.0			56	37	56
SC-(A + B)	3×10^{20}	250-325	300	32	80	100	120	18	2.0	30	11	72	60	56
SC-B	3×10^{20}	250-325	RT	73	100	130	123	9	10			44	13	56
SC-B	3×10^{20}	250-325	300	34	68	92	120	5	4	12	9	67	40	56
Q-(A + B)	3×10^{20}	250-325	RT	109	142	194	210	4	1			60	55	56
A-500	3×10^{20}	250-325	300	85	116	167	175	3	<1	12	11	70	65	56
Q-B	1×10^{20}	250	300	77	108	87	115			14.5	11.3			57

(a) Explanation of designations:

A - annealed
TD - transverse direction
RD - rolling direction
10 CW - 10 percent cold worked
A-650 - annealed at 650 C for 1 hour
A-750 - annealed at 750 C for 1 hour
EX - extruded
CR - cross rolled
T-S - tubular specimen
SC - slow cooled
(A+B) - from (alpha + beta) phase
B - from beta phase
Q - quenched
A-500 - annealed at 500 C
WEX - warm extruded

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.[(58)] The effect of these zirconium hydride platelets on the strength and ductility of unirradiated Zircaloy-2 is illustrated in Figure 19. Since the zirconium-hydride platelets become plastic at about 200 C, the embrittlement ceases to be a problem at that temperature and above.[(59)] The combined effects of oxygen and hydrogen contamination and neutron irradiation on the tensile properties is illustrated in Table 12.[(60,61)] 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.

(2) Fast Fluence. Figures 20 and 21 illustrate the change in yield strength, ultimate strength, uniform elongation, and total elongation as a function of increasing fast fluence. 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.

(3) Irradiation Temperature. The effect that irradiation temperature has on the yield strength is illustrated in Figure 22. 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 piled-up dislocations, and thus the stress required for yielding is higher.

(4) 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 fast-neutron environments. Figure 23 illustrates fast-fluence effects on the yield strength and elongation of Zircaloy-2 cold worked in the testing direction. It can be seen that the yield strength of cold-worked Zircaloy-2 is increased by irradiation to fast fluences up to 1×10^{20} n/cm^2, after which the yield strength starts to decrease. Similarly, the elongation decreases up to a fast fluence of 1×10^{20} n/cm^2, but increases with increased irradiation. This behavior is attributed to a thermal recovery induced by interaction of the cold-work-caused dislocations and the irradiation-induced point defects.[(45)] No similar reversal of strength and ductility was found for the irradiated and cold-worked material in the transverse direction at fast fluences of 1×10^{21} n/cm^2. The effect of irradiation and cold work on the point of

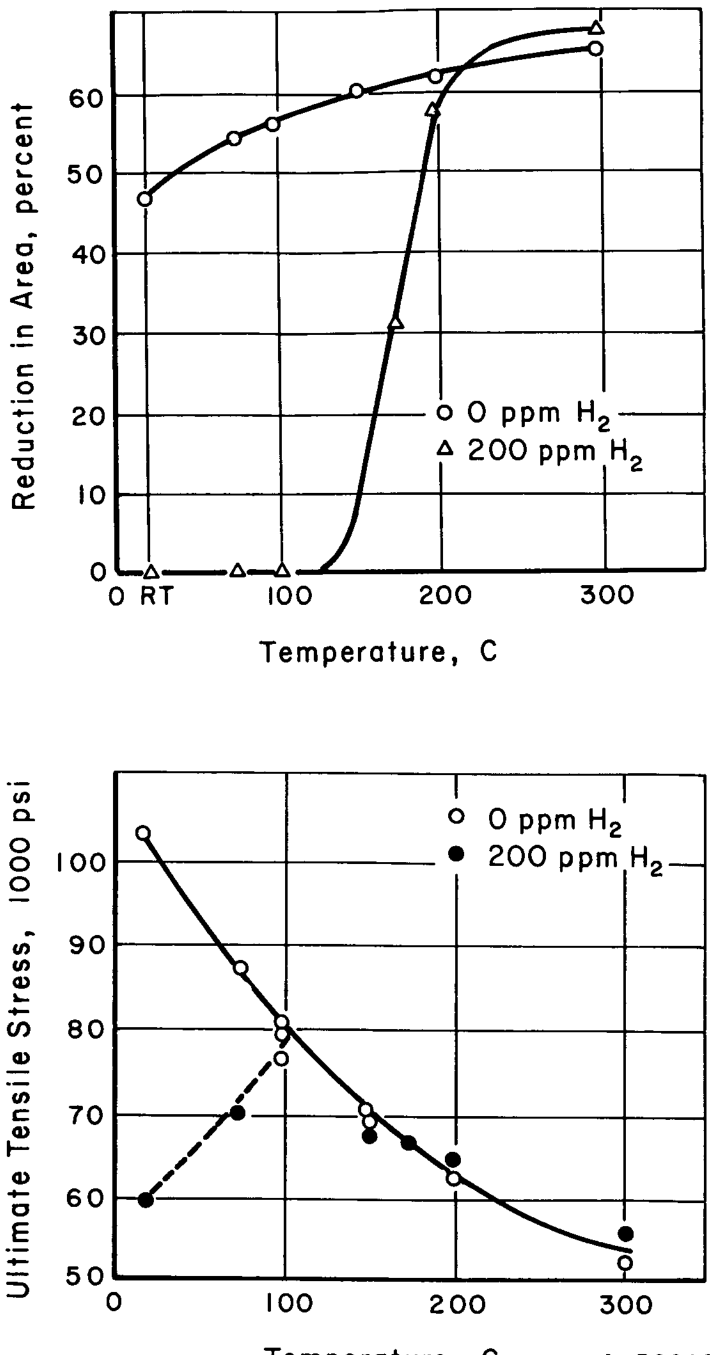


FIGURE 19. MECHANICAL PROPERTIES OF COLD-WORKED ZIRCALOY-2 CONTAINING HYDRIDES PRECIPITATED NORMAL TO TENSILE AXIS[21]

TABLE 12. EFFECT OF HYDROGEN, OXYGEN AND IRRADIATION ON MECHANICAL PROPERTIES OF ZIRCONIUM ALLOYS

Material and Condition[a]	Impurity Content, ppm	Fast Fluence n/cm^2 (>1 MeV)	Irr. 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.	
					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			60
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			60
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			60
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			60
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	47
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	47
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	61
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	61
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	61
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	61
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	61
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	61
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	61
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	61
Zircaloy-2.5 wt % Nb	0	1×10^{21}	290	400	70.3	95.5	78.1	97.6			15.3	13.5			61
Zircaloy-2.5 wt % Nb	13	1×10^{21}	290	400	67.3	98.8	75.3	101.2			14.2	12.7			61
Zircaloy-2.5 wt % Nb	100	1×10^{21}	290	400	69.6	104.2	76.5	105.4			14.6	11.7			61
Zircaloy-2.5 wt % Nb	250	1×10^{21}	290	400	71.3	98.1	78.2	99.5			14.3	12.0			61
					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	62
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	62
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	62
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	62
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	62
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	62
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	62
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	62
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	62
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	62
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	62
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	62

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

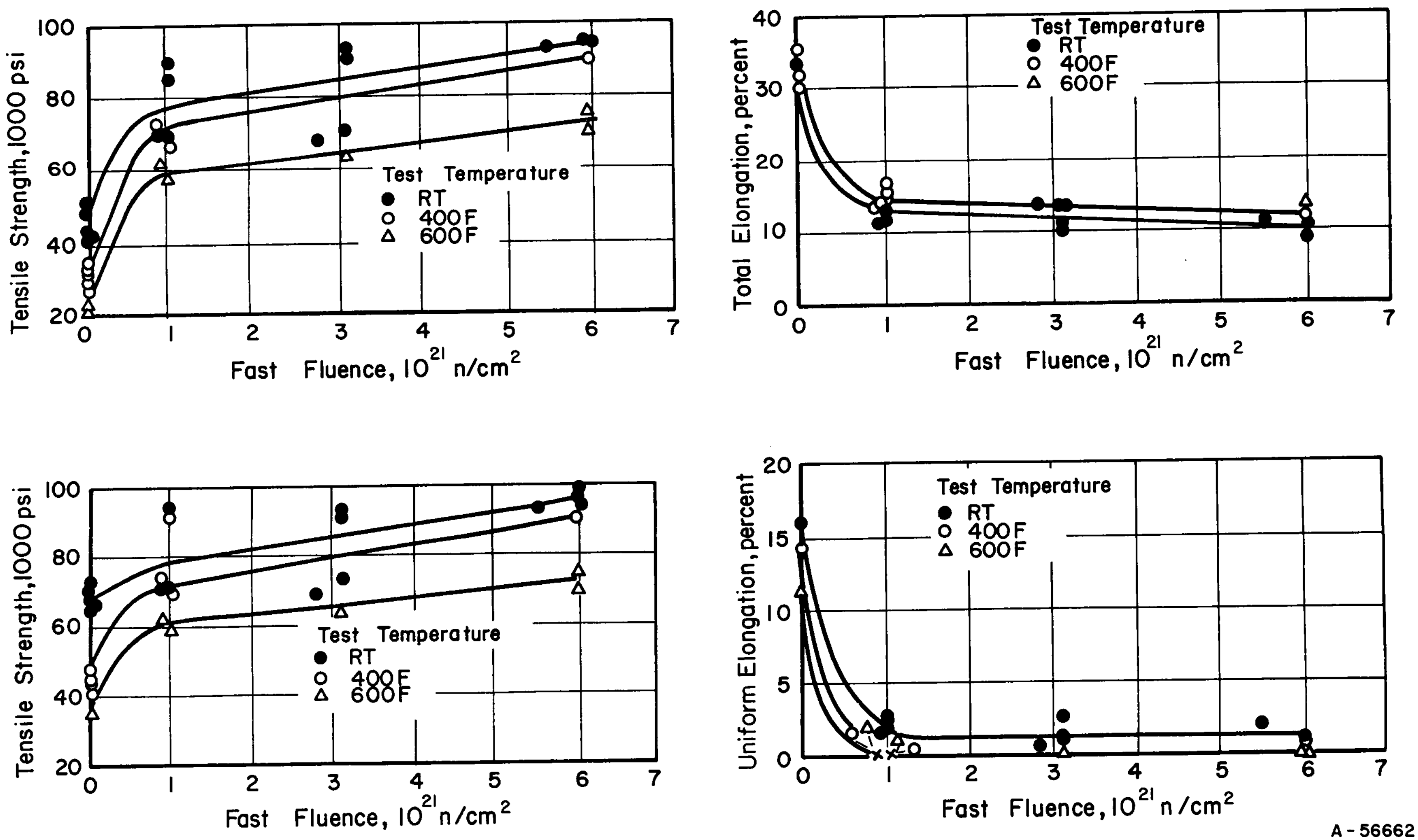


FIGURE 20. EFFECT OF IRRADIATION AT 250 C TO FAST FLUENCE ON THE MECHANICAL PROPERTIES OF ZIRCALOY-2(63)

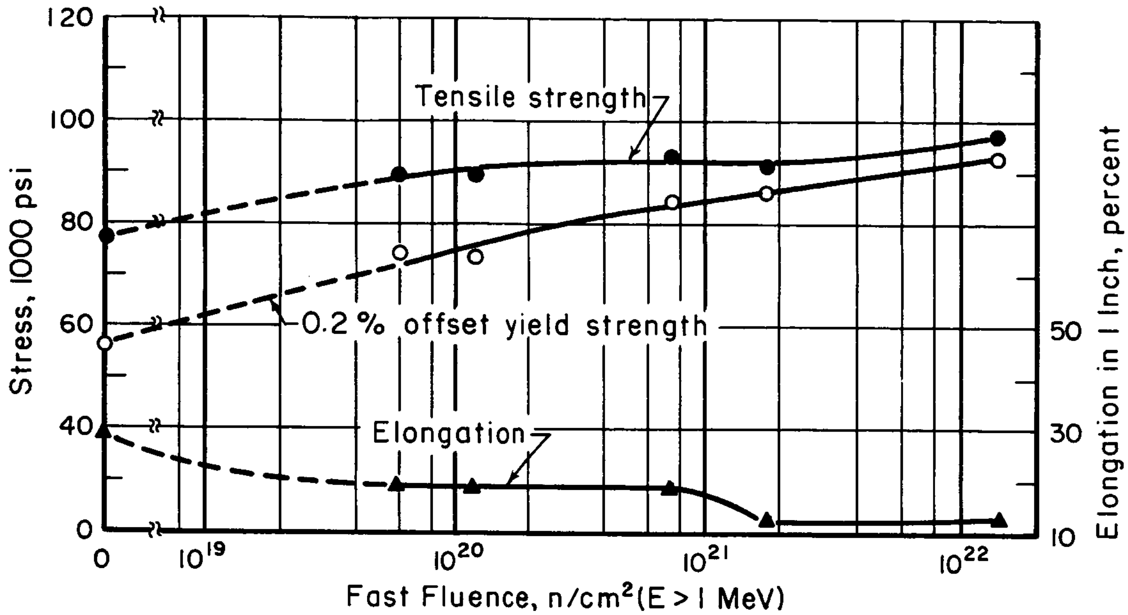


FIGURE 21. EFFECTS OF IRRADIATION AT 50 C ON ROOM-TEMPERATURE TENSILE PROPERTIES OF ZIRCALOY-2[64]

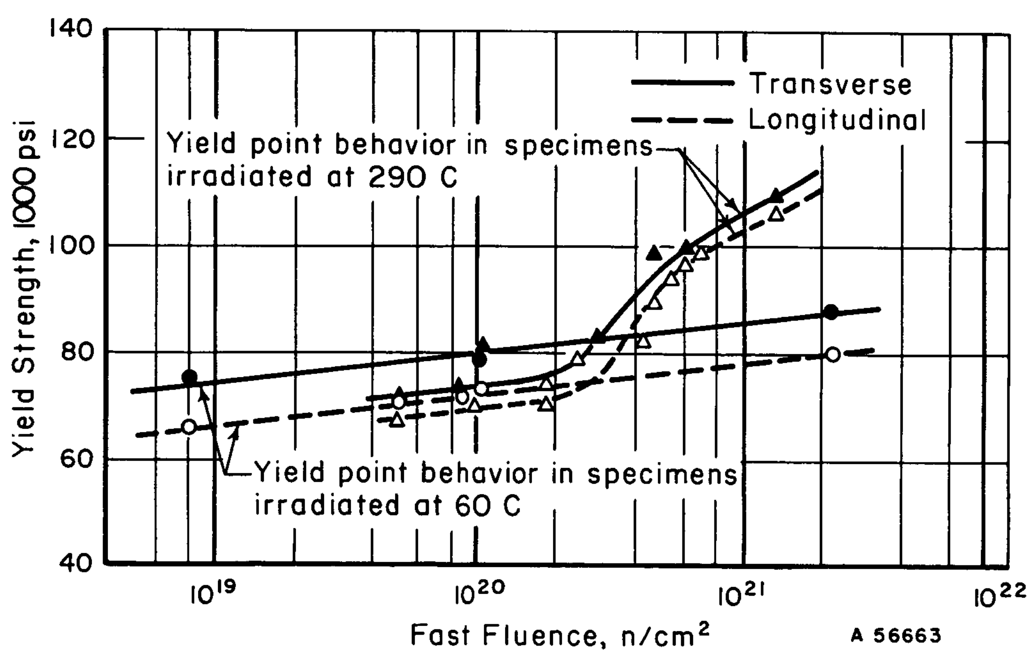


FIGURE 22. EFFECT OF FAST FLUENCE AND IRRADIATION TEMPERATURE ON YIELD STRENGTH OF ZIRCALOY-2[5]

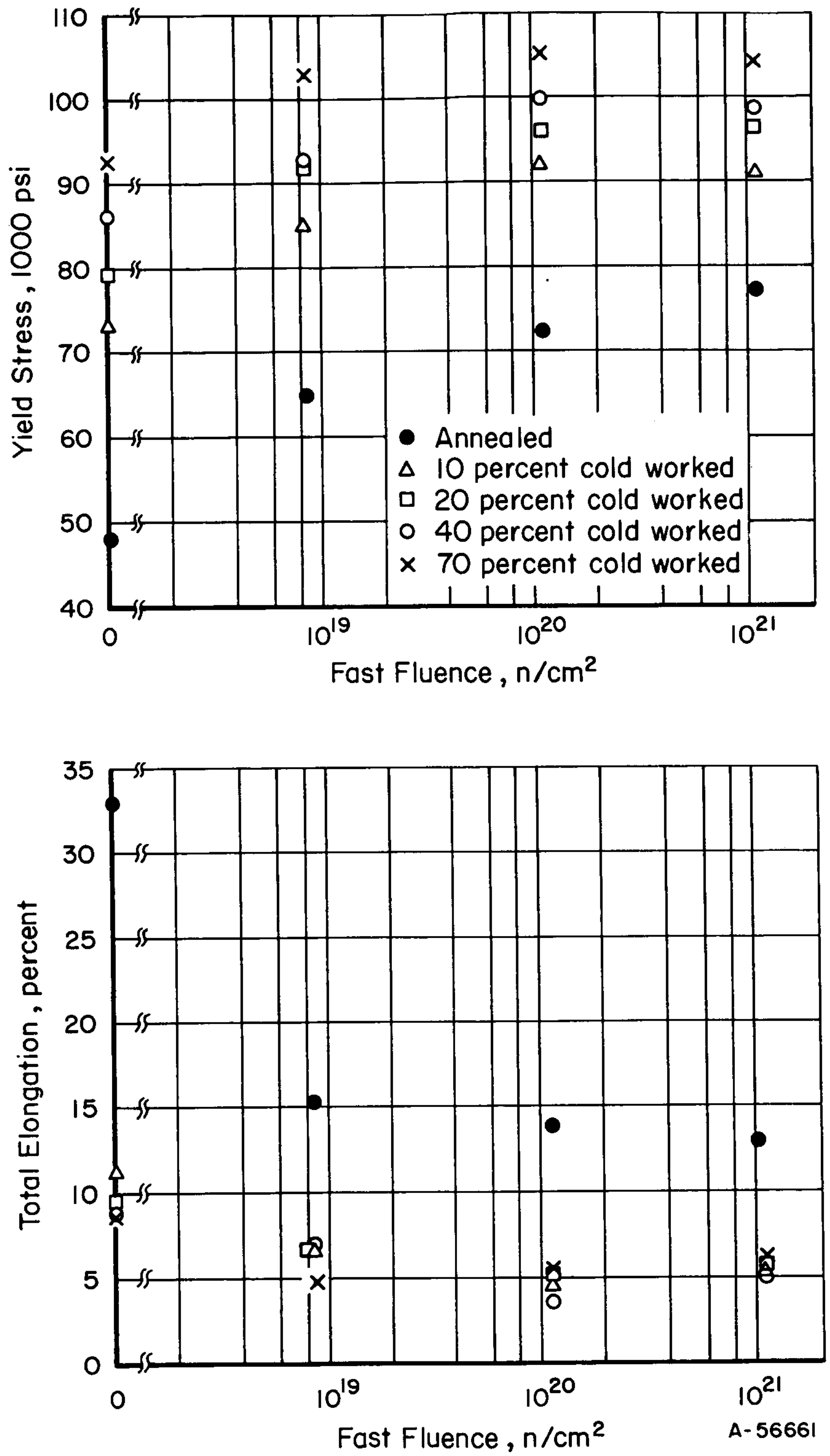


FIGURE 23. THE EFFECT OF COLD WORK AND IRRADIATION ON THE LONGITUDINAL, ROOM TEMPERATURE YIELD STRENGTH AND ELONGATION OF ZIRCALOY-2 IRRADIATED AT 60 C(45)

plastic instability in Zircaloy-2 is illustrated in Figure 24.[5] 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.

(5) 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[5] (see Figure 25). Bush concluded that:

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

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

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

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

(6) Shape Factor. No experimental work is available on the effect of specimen shape on the mechanical properties of irradiated zirconium alloys. However, experimental results with unirradiated Zircaloy-2 tubes containing randomly orientated hydride platelets indicate that the mechanical properties of the tube are dependent on tube wall thickness.[67] It was found that the yield strength increases and ductility decreases with decreasing wall thickness. This behavior was found to depend directly on the ratio of cross-sectional area of average hydride platelet to the cross-sectional area of the tubular specimens. If the cross-sectional area of the tube is small, it is highly possible that somewhere along the tube length exists a large percentage of the hydride platelets orientated perpendicularly to the stress direction. The rather low ductility values found for irradiated tubular specimens, as shown in Table 11, can probably be attributed to the presence of hydride platelets in the thin-walled specimens.

(7) Welding. The effect of irradiation on the mechanical properties of welded Zircaloy-2 is shown in Table 13. Welding does not change the mechanical properties of unirradiated Zircaloy-2, and the as-welded material exhibits properties similar to those for annealed Zircaloy-2. Cold working of the as-welded materials also results in mechanical properties similar to those for cold-worked unwelded material. 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.

The effect of irradiation on mechanical properties other than tensile properties is discussed below.

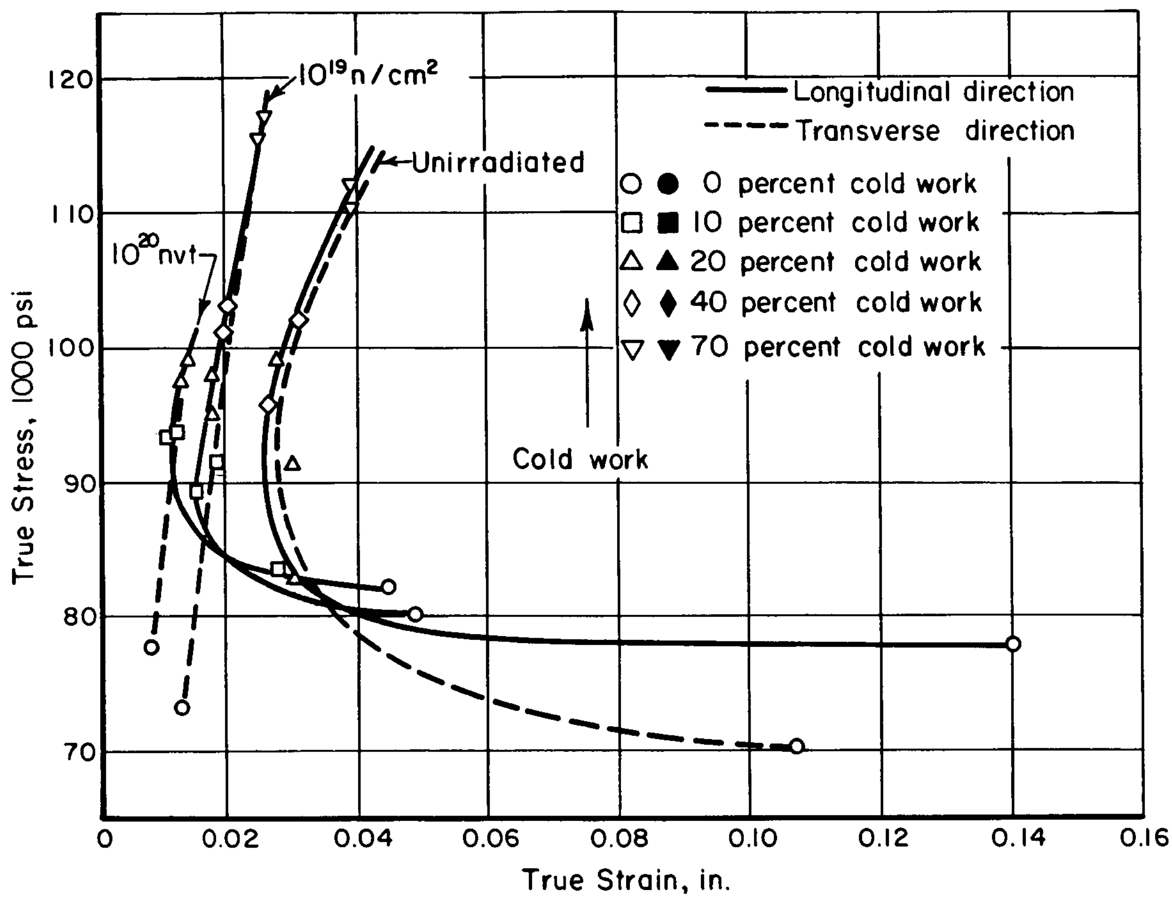


FIGURE 24. DISPLACEMENT IN POINT OF PLASTIC INSTABILITY WITH COLD WORK AND IRRADIATION FOR ZIRCALOY-2[5]

Reference[a]	Material	State	Fast Fluence, n/cm2	Irr. Temp., C
○ 42	Zirconium	Annealed	$3.3 \times 10^{19} > 1$ MeV	50
△ 57	Zr-2.5 wt % Nb	Q & T	$1 \times 10^{20} > 500$ eV	50
▲ 57	Zr-2.5 wt % Nb	Q & T	$1 \times 10^{20} > 500$ eV	250
□ 42	Zircaloy-2	Annealed	$3 \times 10^{19} > 1$ MeV	50
◩ 65	Zircaloy-2	Annealed	$7.7 \times 10^{19} > 500$ eV	280
■ 46	Zircaloy-2	Annealed	$2.7 \times 10^{20} > 500$ eV	280
		Above, all times 1 hour		
× 66	Zircaloy-2	Annealed	$1.8 \times 10^{20} > 1$ MeV	40
⊥ 50	Zircaloy-2	Extruded ($\alpha + \beta$)	$5.6 \times 10^{19} > 1$ MeV	40
T 50	Zircaloy-2	Extruded and annealed	$5.6 \times 10^{19} > 1$ MeV	40
⊢ 50	Zircaloy-2	Extruded and 50% CW	$5.6 \times 10^{19} > 1$ MeV	40
⊣ 50	Zircaloy-2	Extruded and 15% CW and annealed	$5.6 \times 10^{19} > 1$ MeV	40
		Above, all times 100 hours		

(a) 0.2% offset yield strengths were used except in Reference 1, where proportional limit was used.

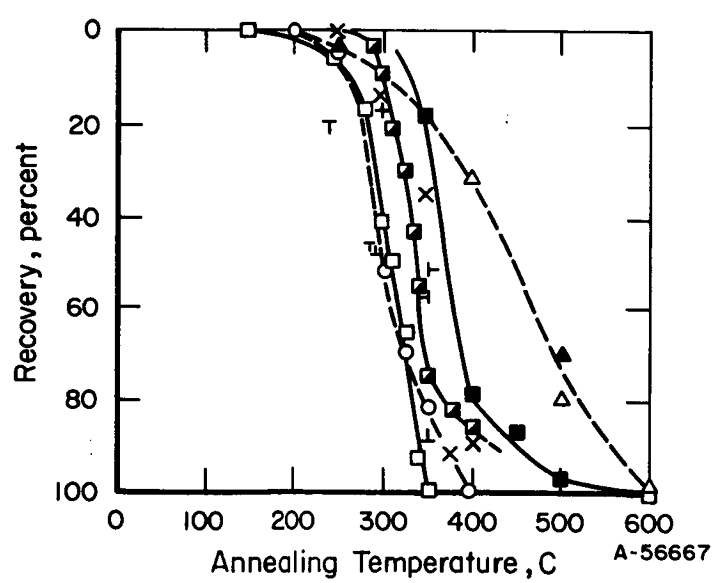


FIGURE 25. RECOVERY OF IRRADIATION-INDUCED DAMAGE IN TENSILE PROPERTIES OF ZIRCONIUM AND ITS ALLOYS BY POSTIRRADIATION ANNEALING(5)

TABLE 13. EFFECT OF FAST-NEUTRON IRRADIATION ON MECHANICAL PROPERTIES OF WELDED ZIRCALOY-2

Material Condition(a)	Fast Fluence, n/cm^2 (>1 Mev)	Irr. Temp, C	Test Temp, C	0.2% Offset 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.	
B	1.3×10^{17}	~250	RT	57.5	59.7	67.0	66.5	6.3	6.0	12.2	11.9	--	--	52
B	2.6×10^{18}	250-300	RT	57.5	61.2	67.0	68.4	6.3	5.3	12.2	10.6	--	--	52
AW	$0.3\text{-}1.9 \times 10^{14}$	260	RT	47.3	49.3	62.7	62.3	10.2	11.4	11.9	14.3	--	--	60
AW	$0.3\text{-}1.9 \times 10^{19}$	260	RT	47.3	49.3	62.7	62.3	10.2	11.4	11.9	14.3	--	--	60
A-900	$0.3\text{-}1.9 \times 10^{19}$	260	RT	39.2	54.9	67.1	74.8	13.4	12.0	25.3	24.2	34.8	49.8	60
AW	$0.3\text{-}1.9 \times 10^{14}$	260	260	24.8	30.9	37.3	39.2	12.3	9.3	21.3	18.5	--	--	60
A-900	$0.3\text{-}1.9 \times 10^{19}$	260	260	23.9	28.2	44.1	43.1	13.8	8.6	35.4	29.1	--	--	60
BQ-W-14CW-T	6.7×10^{19}	280	RT	62.3	85.2	76.0	92.1	8.4	3.1	13.0	5.7	39.6	32.8	68
BQ-W-14CW-N	6.7×10^{19}	280	RT	55.4	84.3	76.1	93.1	10.2	4.0	16.1	8.0	42.0	32.9	68
BQ-W-14CW-T	1.3×10^{20}	280	RT	62.3	90.3	76.0	97.1	8.4	4.2	13.0	6.6	39.6	32.4	68
BQ-W-14CW-N	1.3×10^{20}	280	RT	55.4	89.5	76.1	94.8	10.2	3.9	16.1	8.5	42.0	32.9	68
BW-W-14CW-T	2.2×10^{20}	280	RT	56.8	94.7	77.0	100.3	10.7	3.0	16.6	6.0	38.9	31.4	70
BQ-W-14CW-N	2.2×10^{20}	280	RT	58.8	92.8	79.7	98.4	11.9	3.1	21.0	6.1	39.0	28.7	70
BQ-W-14CW-N	5.4×10^{20}	280	RT	55.4	105.4	76.1	108.1	10.2	1.3	16.1	5.5	42.0	28.1	70
BQ-W-14CW-T	5.4×10^{20}	280	RT	62.3	104.8	76.0	107.0	8.4	1.5	13.0	4.3	39.6	28.0	70
BQ-W-16CW-T	6.7×10^{19}	280	RT	64.8	89.5	88.8	98.8	12.3	3.6	19.2	5.4	36.6	29.4	68
BQ-W-16CW-N	6.7×10^{19}	280	RT	64.4	89.2	87.6	99.6	9.5	4.2	13.3	7.9	35.9	33.4	68
BQ-W-16CW-T	1.3×10^{20}	280	RT	64.8	90.9	88.8	98.1	12.3	2.6	19.2	3.9	36.6	28.9	68
BQ-W-16CW-N	1.3×10^{20}	280	RT	64.4	92.8	87.6	101.6	9.5	4.0	13.3	8.7	35.9	30.6	68
BQ-W-16CW-T	2.2×10^{20}	280	RT	65.2	97.9	88.3	104.7	9.5	2.9	15	4.5	34.5	27.5	70
BQ-W-16CW-N	2.2×10^{20}	280	RT	64.3	98.1	88.4	106.4	9.4	3.7	15.5	6.7	35.2	29.3	70
BQ-W-16CW-T	5.4×10^{20}	280	RT	64.8	103.8	88.8	110.7	12.3	2.7	19	4.7	36.6	28.6	70
BQ-W-16CW-N	5.4×10^{20}	280	RT	64.4	104.7	87.6	110.7	9.5	2.8	13.3	4.8	35.9	28.8	70
BQ-W-16CW-T	5.6×10^{20}	280	300	--	66.5	--	70.2	--	1.5	--	3.9	--	28.7	69
BQ-W-16CW-N	5.6×10^{20}	280	300	--	66.0	--	69.8	--	1.4	--	4.6	--	31.8	69
BQ-W-20CW-T	6.7×10^{19}	280	RT	69.5	93.8	90.2	102.4	9.7	3.8	17.4	8.3	38.9	33.3	68
BQ-W-30CW-N	6.7×10^{19}	280	RT	67.2	94.5	88.3	101.8	9.8	3.4	17.3	6.9	39.6	33	68
BQ-W-30CW-T	1.3×10^{20}	280	RT	69.5	97.3	90.2	103.9	9.7	3.8	17.4	7.3	38.9	32.5	68
BQ-W-30CW-N	1.3×10^{20}	280	RT	67.2	97.2	88.3	103.9	9.8	3.8	17.3	7.9	39.6	34.4	68
BQ-W-30CW-T	1.6×10^{20}	280	300	--	62.4	--	62.6	--	1.1	--	5.3	--	46.8	69
BQ-W-30CW-N	1.6×10^{20}	280	300	--	64.3	--	64.6	--	0.9	--	4.8	--	47.1	69
BQ-W-30CW-T	2.2×10^{20}	280	RT	69	102.9	88.5	107.9	9.0	3.1	18.1	6.5	45	31.6	70
BQ-W-30CW-N	2.2×10^{20}	280	RT	69	102.7	89.5	108.2	10.0	3.7	18	7.4	39.1	35.5	70
BQ-W-30CW-N	5.4×10^{20}	280	RT	67.2	107.2	88.3	115.2	9.8	0.5	17.3	6.6	39.6	30.4	70
BQ-W-30CW-T	5.4×10^{20}	280	RT	69.5	111.5	90.2	113.3	9.7	2.5	17.4	5.9	38.9	31.8	70
BQ-W-30CW-T	5.6×10^{20}	280	300	--	69.8	--	71.1	--	1.1	--	3.7	--	37.0	69
BQ-W-30CW-N	5.6×10^{20}	280	300	--	70.8	--	71.5	--	1.0	--	3.5	--	42.4	69

(a) Explanation of designations:

B - welded according to Code B.
AW - as welded.
A-900 - annealed at 900 C.
BQ - quenched from beta phase
W - welded
14CW - 14 percent cold worked
T - tested in transverse direction
N - tested in normal direction.

Other Mechanical Properties

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 26.(71) Table 14 shows that irradiation of Zircaloy-2 tubing to higher fluences results in significant increases of strength and drastic reductions in ductility.(72)

Notch Sensitivity and Impact Properties. No significant changes in room-temperature notch sensitivity are induced by irradiation to a fast fluence of 1×10^{20} n/cm^2 if the amount of cold work in the specimens is less than 40 percent.(73) With more cold working, the notch strength increases sharply after irradiation to a fast fluence of about 9×10^{19} n/cm^2 and the notch ductility falls to zero after irradiation to a fast fluence of 4×10^{20} n/cm^2.

It has been shown that irradiation to a fast fluence of 2.5×10^{21} n/cm^2 has only minor effects on the impact properties of zirconium alloys.(47) Irradiation of welded(60) and cold-worked(41) Zircaloy-2 has not induced any change in impact properties.

The addition of hydrogen has been found to have considerable effect on the impact properties of Zircaloy-2.(69) 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 weight percent niobium alloy which contains hydrogen increases the transition temperature, as illustrated in Figure 27. 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 weight percent niobium than in Zircaloy-2 after an equal fluence.(61, 74) The transition-temperature shift in hydrided zirconium alloys does not appear to be fast-fluence dependent above 2×10^{20} n/cm^2.

Stress Relaxation. It has been found that less stress relaxation at 100 C or less takes place in annealed Zircaloy-2 during irradiation than in unirradiated material loaded at the same stress.(75) Both the unirradiated material and material irradiated to a fast fluence of 8×10^{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 specimens in-pile relaxed to 9,000 psi. The in-pile specimens received a fast fluence of 8×10^{20} n/cm^2, test temperature being less than 100 C.

Creep. Out-of-pile creep tests have been performed on Zircaloy-4 irradiated to fast fluences up to 2.5×10^{21} n/cm^2. Tests performed at 285 C with loads of 20,000 psi showed that there was no difference in the creep rates between irradiated and unirradiated Zircaloy-4.(47) In-pile creep tests have been performed on 20 percent cold-worked

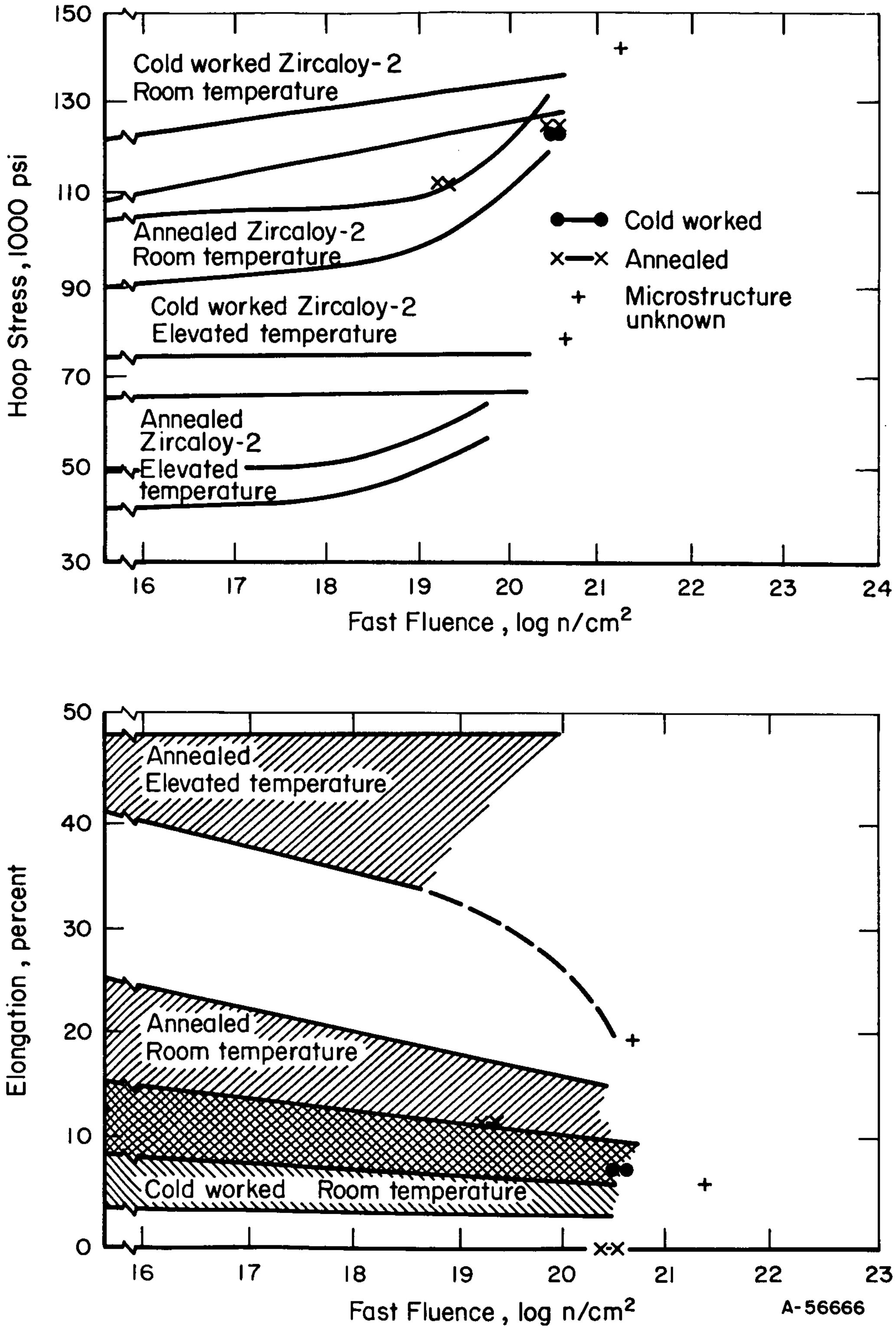


FIGURE 26. HOOP STRENGTH AND ELONGATION AS A FUNCTION OF FAST FLUENCE IN PRTR ZIRCALOY-2 PRESSURE TUBES[71]

TABLE 14. EFFECT OF FAST NEUTRON IRRADIATION ON ROOM TEMPERATURE BURST PROPERTIES OF ZIRCALOY-2 AND ZIRCALOY-4 IRRADIATED AT 286 C[72]

Material	Fast Fluence, 10^{21} n/cm^2 (>1 MeV)	0.2% Offset Yield Strength, 1000 psi	Ultimate Tensile Strength, 1000 psi	Elongation, percent	
				Uniform	Total
Zr-2	0	19.7	31.6	14.9	22.8
Zr-2	0	19.3	30.8	13.7	26.2
Zr-2	0	22.6	31.9	10.6	21.1
Zr-2	0.71	55.2	56.5	0.43	1.6
Zr-2	0.71	58.5	58.0	0.21	1.9
Zr-2	0.74	44.4	56.6	0.80	2.3
Zr-2	0.74	55.6	55.6	0.13	1.4
Zr-2	0.96	55.9	56.2	0.23	1.9
Zr-2	0.96	61.7	62.0	0.21	1.4
Zr-2	1.21	59.7	59.7	0.20	1.6
Zr-2	1.21	61.2	61.2	0.20	1.6
Zr-2	1.28	61.6	61.6	0.13	1.3
Zr-2	1.28	57.0	58.1	0.30	1.8
Zr-2	1.30	57.2	57.2	0.17	1.7
Zr-2	1.30	66.0	66.5	0.26	2.1
Zr-2	1.44	55.2	55.2	0.19	1.3
Zr-2	1.53	51.4	52.7	0.43	1.0
Zr-2	1.53	64.2	64.5	0.21	1.5
Zr-4	0.44	52.0	52.0	0.18	1.8
Zr-4	0.44	57.1	57.1	0.19	1.9
Zr-4	0.72	55.4	55.4	0.15	1.8
Zr-4	0.72	53.0	53.0	0.14	1.8
Zr-4	0.90	52.4	53.0	0.38	1.0
Zr-4	0.90	56.3	58.5	0.44	2.3

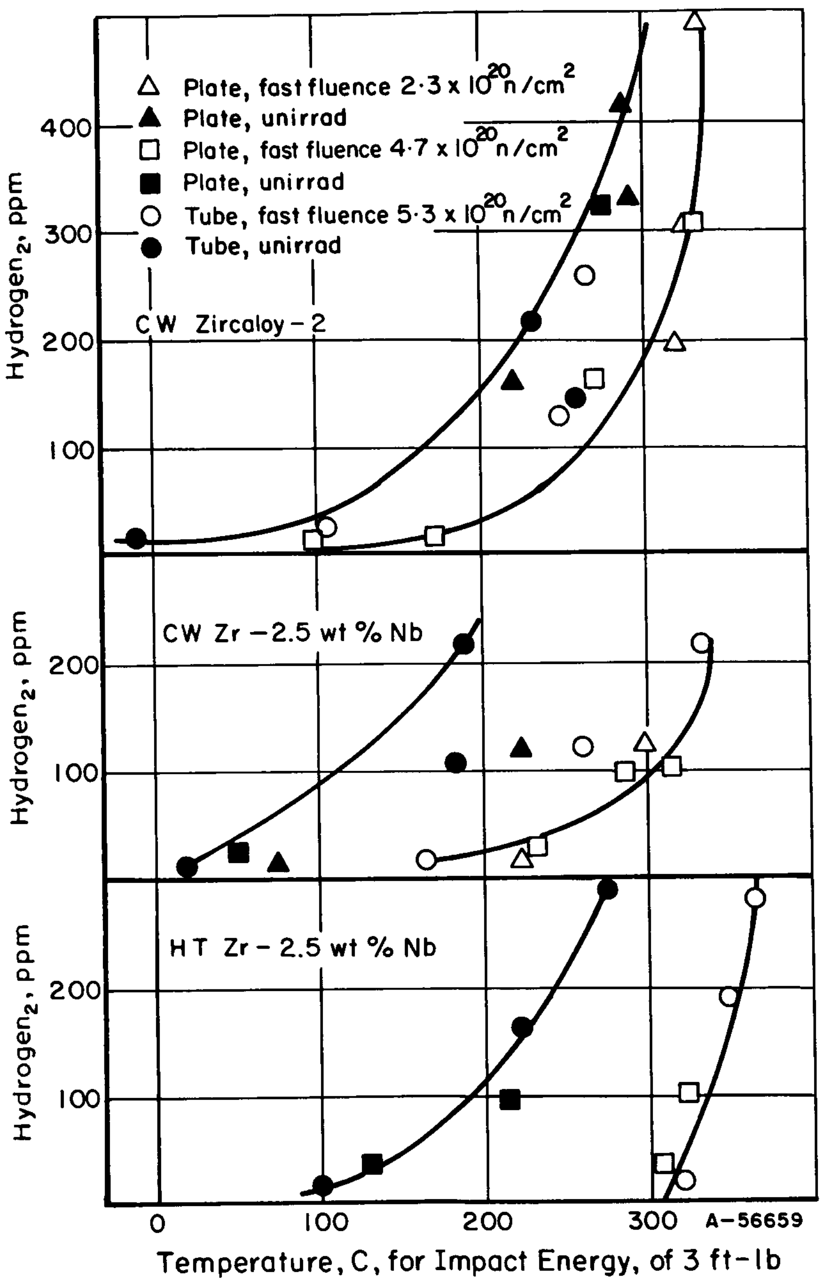


FIGURE 27. SUMMARY OF COMBINED EFFECTS OF HYDROGEN AND IRRADIATION ON IMPACT PROPERTIES OF ZIRCALOY-2 AND ZIRCONIUM-2.5 WT. % NIOBIUM ALLOY[74]

Zircaloy-2 at Battelle-Northwest and Chalk River. The tests performed at Battelle-Northwest indicated that no significant differences were found between the in-pile and out-of-pile creep rate of cold-worked Zircaloy-2.[76] Workers at Chalk River found that while the creep rate or in-pile and out-of-pile specimens is the same during the first 500 hours, the creep rate out-of-pile levels off while the in-pile creep rate remains nearly constant. Consequently, the in-pile creep was found to be larger by a factor of 10 after about 4,000 hours at 300 C with a load of 20,000 psi. In-pile and out-of-pile creep data are compared in Figure 28.[77] The results for in-pile and out-of-pile creep of zirconium alloys are summarized in Figure 29.[78] It can be seen that at stresses above 20,000 psi, the in-pile creep rate is considerably higher than the out-of-pile creep rate of control specimens.

As shown in Figure 30, Zircaloy-2 cladding can undergo considerably more strain, before failure by cracking, if the irradiation temperature is above 350 C.[79] No performance difference was noted for claddings with thicknesses of 25 and 35 mils. Strain in these claddings was induced by swelling of uranium fuel rods inside the cladding.

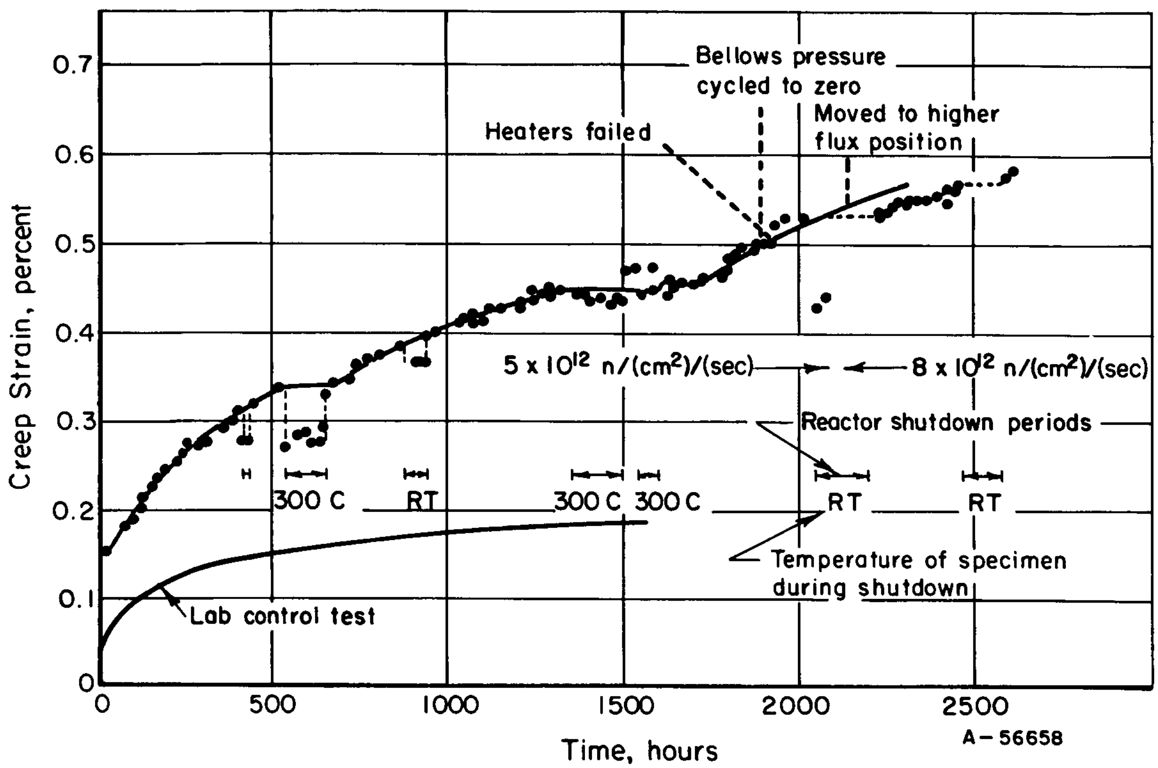


FIGURE 28. COMPARISON OF IN-PILE AND OUT-OF-PILE CREEP OF 20 PERCENT COLD-WORKED ZIRCALOY-2 AT 30,000 PSI AT 300 C[77]

Apparent specimen contractions during reactor shutdown are caused by dimensional changes in the specimen-supporting unit of the creep machine due to gamma heating.

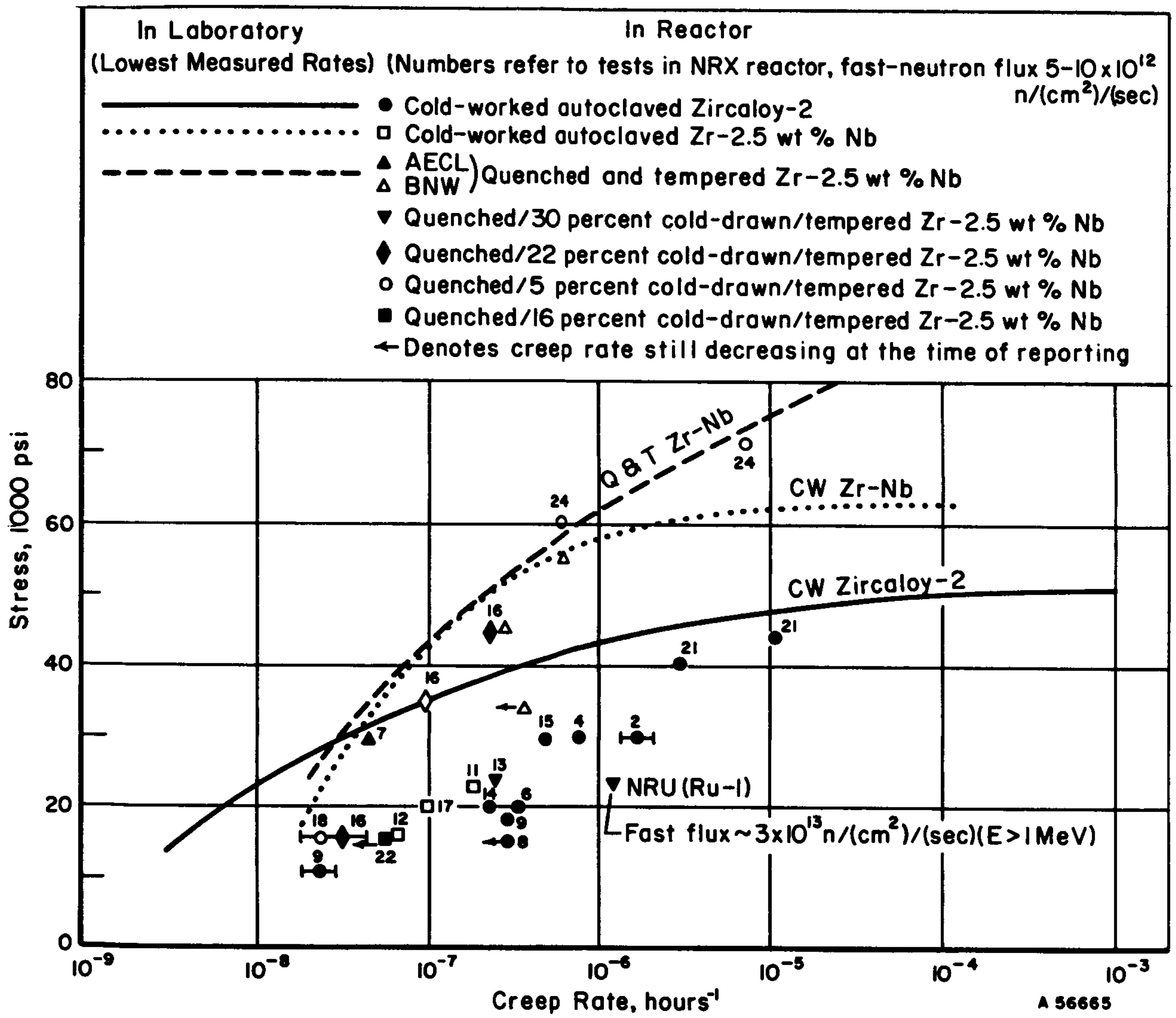


FIGURE 29. CREEP OF ZIRCONIUM ALLOYS AT 300 C(78)

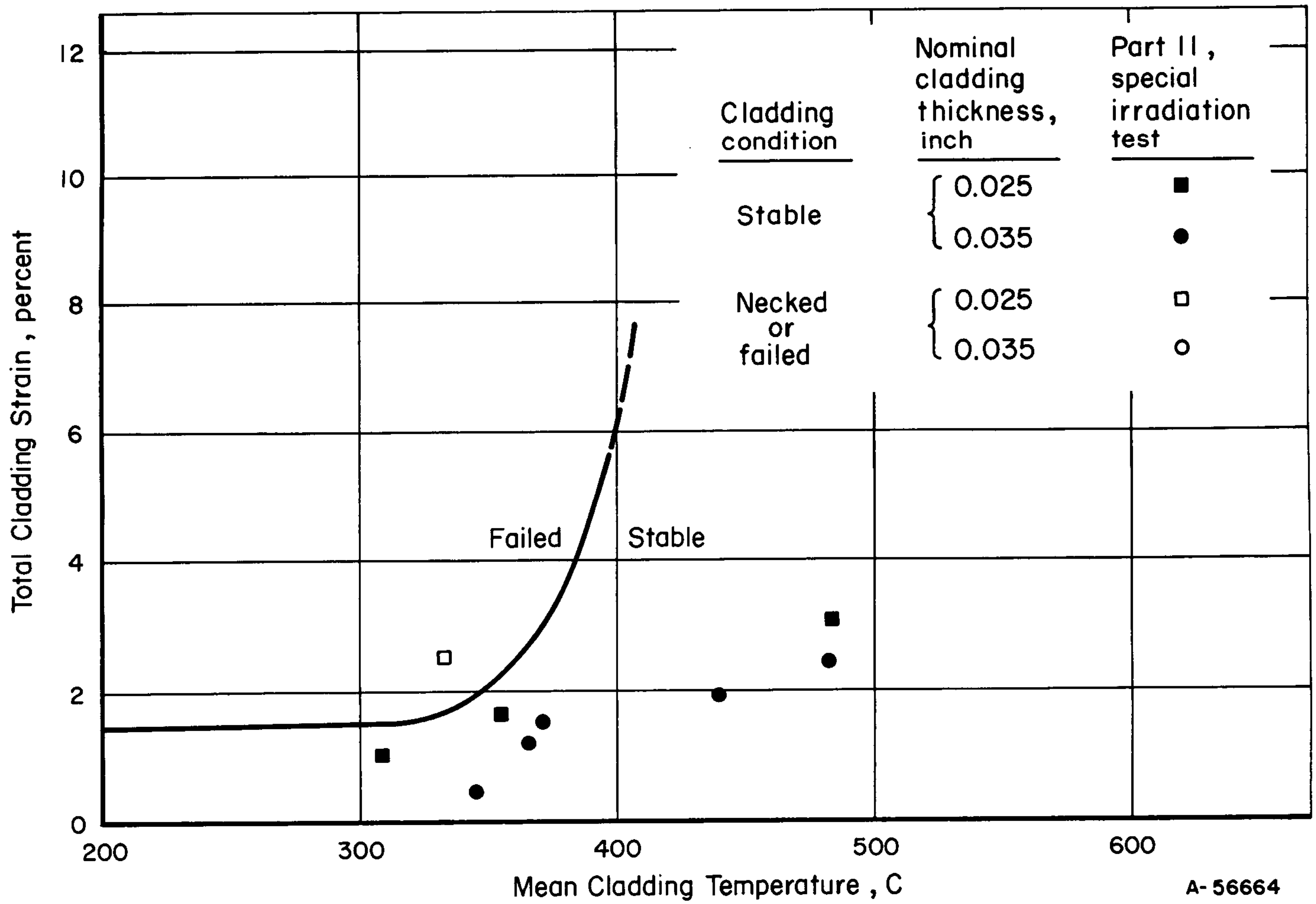


FIGURE 30. EFFECT OF CLADDING IRRADIATION TEMPERATURE ON CLADDING STABILITY(79)

FERRITIC AND MARTENSITIC STEELS

Pressure-Vessel Steels

The nuclear industry has used ferritic steels as structural materials in reactor applications for many years. 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.

Effect of Irradiation on 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 is usually 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 lower than 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.(80) 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.(81) 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 report. For clarity, a prior explanation of factors influencing the NDT temperature of an unirradiated steel will be helpful in discussing the effect of irradiation on the NDT.(82)

(1) Composition. The effect of various alloying elements on the transition temperature is illustrated in Figure 31.[83] These results are for steels containing 0.30 wt % carbon, 1.0 wt % manganese, and 0.30 wt % silicon, plus the element whose effect is being measured. It has been found that molybdenum additions in low-alloy nickel-chromium steels are very beneficial in maintaining a low NDT even after prolonged heating.[84]

(2) Grain Size. Steels with smaller grain size have been found to have lower NDT temperatures.

(3) Cold Work. Cold working raises the NDT of steels, and aging following cold work results in further increases in NDT. Cold working by cyclic fatiguing results in considerable increase in NDT with the increase being largest when the temperature is in the 400 to 580 F range.[85]

(4) Heat Treatment. A quenched and tempered structure has a considerably lower NDT than an annealed or normalized structure.

(5) Welding. Welding generally results in higher NDT because the heat-affected zone (HAZ) resulting from welding is similar to that of annealed metal.

(6) Direction. Specimens formed from the longitudinal direction, relative to rolling direction, generally show an impact strength higher than that of specimens formed from the transverse direction.

Figure 32 shows the irradiation-induced change in the NDT for a number of steels.[86] Since, as tabulated in Table 15, 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. Specific attempts have been made to identify the variables affecting the irradiation sensitivity of a steel. Figure 33 illustrates the large variation in irradiation-induced NDT shifts obtained on different heats of ASTM A302-B steel.[87] The authors termed some of the heats as "sensitive" or "insensitive" to irradiation-induced embrittlement. Since then, considerable research has been under way to determine all possible variables affecting "sensitivity" of a steel to irradiation. 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. The importance of structure has been demonstrated by giving different heat treatments to two steels (HY-80 and A353) showing large differences in the NDT shifts.[88a] From the experiments, it was found that microstructure considerably affected the transition-temperature shift. A ferritic structure, resulting from slowly cooling from the heat-treat temperature, was 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.

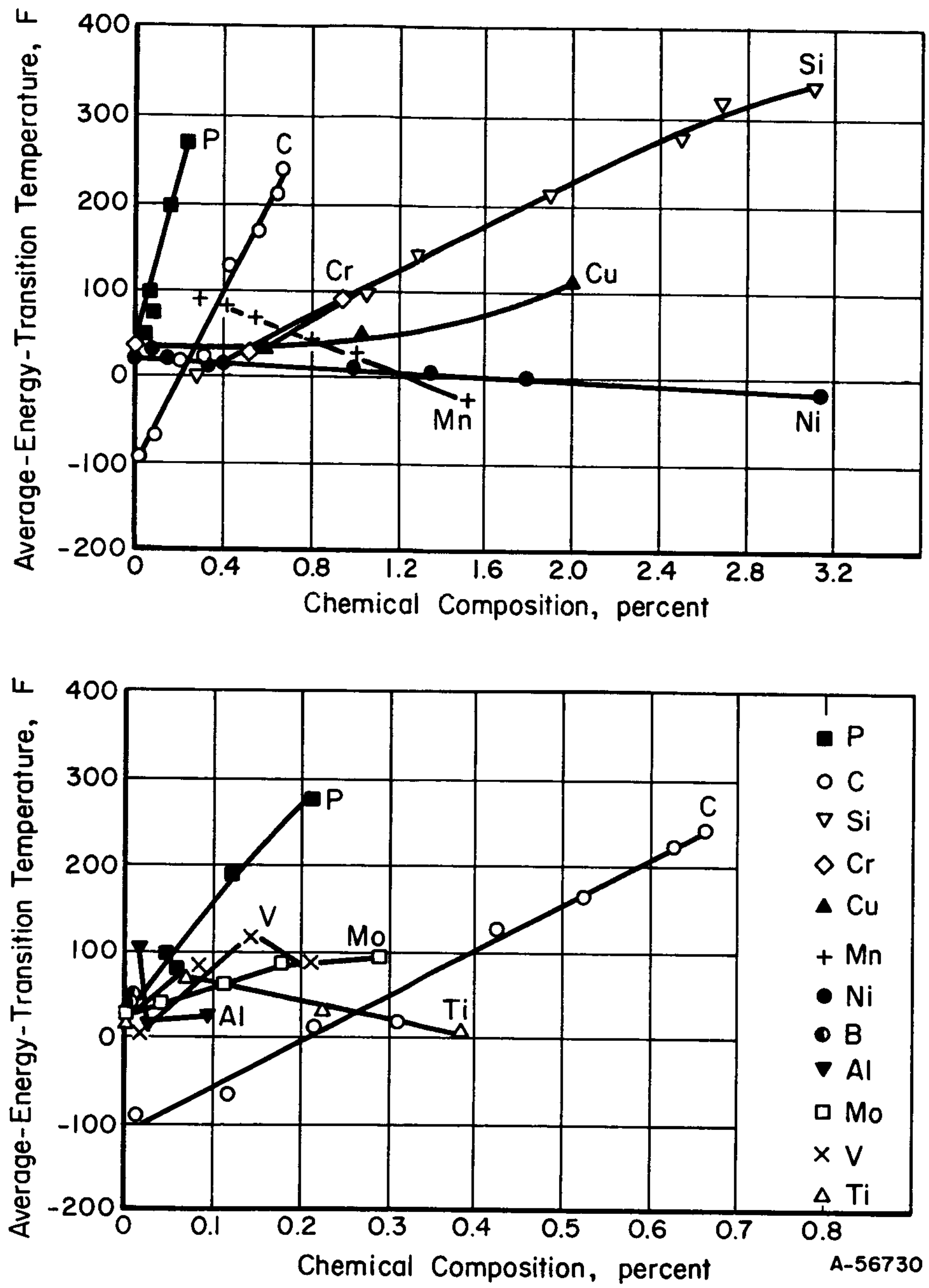


FIGURE 31. EFFECT OF CHEMICAL COMPOSITION ON AVERAGE-ENERGY TRANSITION TEMPERATURE(83)

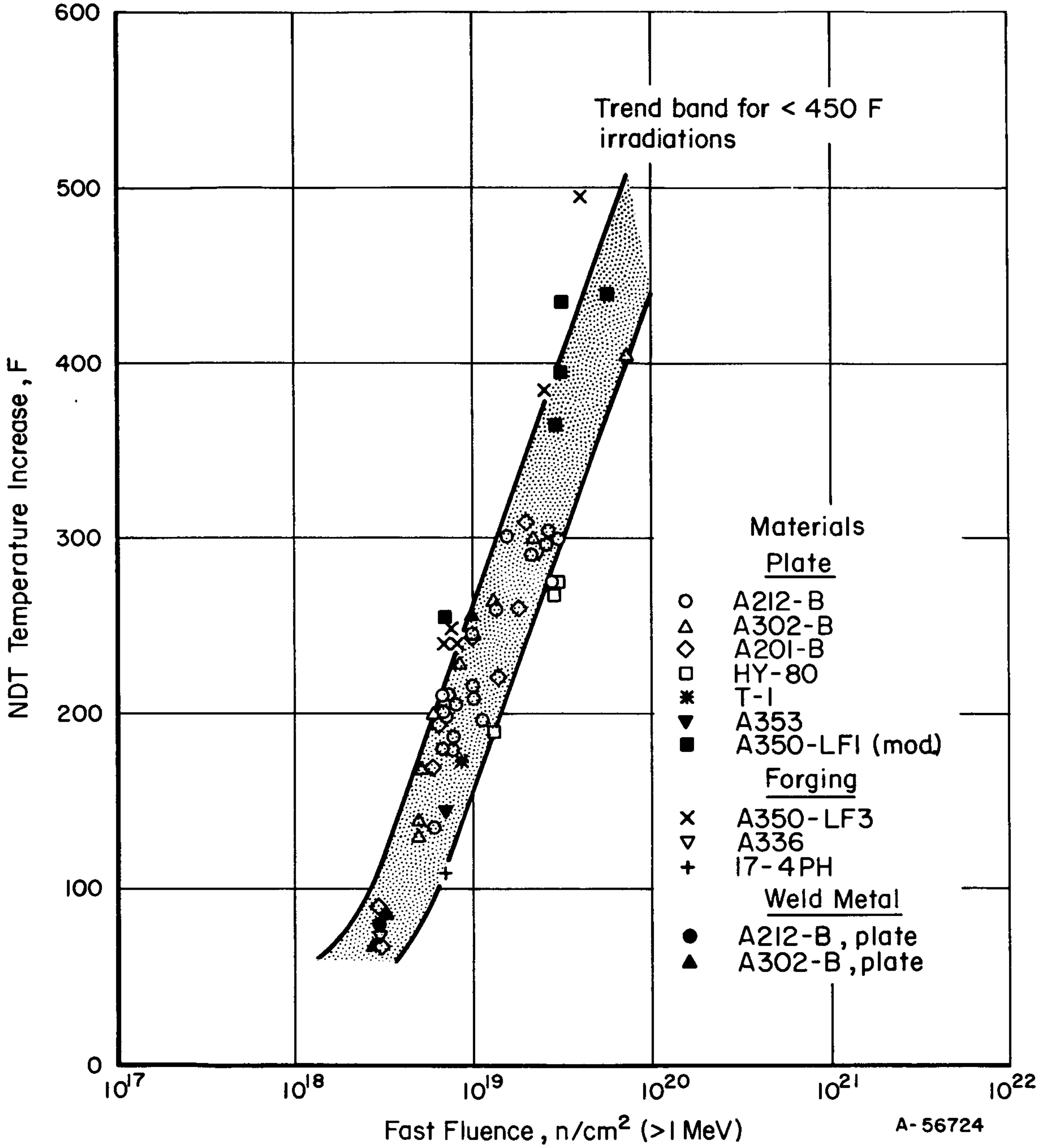


FIGURE 32. INCREASE IN THE NDT TEMPERATURES OF STEELS RESULTING FROM IRRADIATION AT TEMPERATURES BELOW 450 F[86]

TABLE 15. COMPOSITION OF THE STEELS WHOSE SENSITIVITY TO IRRADIATION EMBRITTLEMENT HAS BEEN STUDIED

Designation	Composition, weight percent														
	Fe	C	Mn	Si	Cr	Mo	Ni	Al	P	S	Cu	Ti	V	N	Nb
A212-B	Bal	0.26	0.76	0.24	0.20	0.02	0.22		0.011	0.031					
A302-B	Bal	0.20	1.31	0.25	0.17	0.47	0.20		0.012	0.023					
A336	Bal	0.19	0.65	0.26	0.40	0.64	0.79		0.011	0.014	0.12				
A350-LF 1	Bal	0.15	0.79	0.25	0.05	0.04	1.71		0.027	0.033	0.088		0.04		
A350-LF 3	Bal	0.14	0.52	0.25	0.04	0.05	3.28		0.031	0.032			0.04		
A353	Bal	0.09	0.44	0.21	0.04	0.02	8.85	0.011	0.009	0.014					
A387-B	Bal	0.12	0.49	0.23	1.02	0.52	0.10	0.04	0.010	0.019	0.19	0.004	0.003		
A387-D	Bal	0.14	0.30	0.24	2.05	0.98			0.012	0.020					
Fortiweld B	Bal	0.12	0.68	0.38	0.07	0.45	0.04	0.10	0.023	0.011	0.12			0.007	
HY-80	Bal	0.14	0.21	0.19	1.55	0.54	2.91	0.06	0.011	0.014					
HY-90	Bal	0.16	0.35	0.32	1.90	0.50	2.85	0.02	0.014	0.015				0.009	
HY-150	Bal	0.12	0.27	0.21	0.81	0.97	7.38	0.034	0.008	0.008					
OK54P	Bal	0.06	0.99	0.56	0.05		0.07		0.015	0.020	0.18			0.011	
SSS-100	Bal	0.18	0.51	0.28	1.72	0.50	0.08	0.036	0.017	0.020	0.26	0.085			
T-1	Bal	0.13	0.85	0.24	0.67	0.40	0.64		0.013	0.013			0.06		
3.5 Ni-Cr-Mo	Bal	0.17	0.38	0.29	1.88	0.51	3.65	0.02	0.013	0.023				0.010	
7.5 Ni-Cr-Mo	Bal	0.12	0.27	0.21	0.81	0.97	7.38	0.034	0.008	0.008					
17-4 PH	Bal	0.04	0.3	0.6	16.00	0.50	4.00				3.30				0.35
2103/R3	Bal	0.15	1.59	0.41	0.02	0.05	0.01	0.034	0.014	0.012	0.05			0.006	

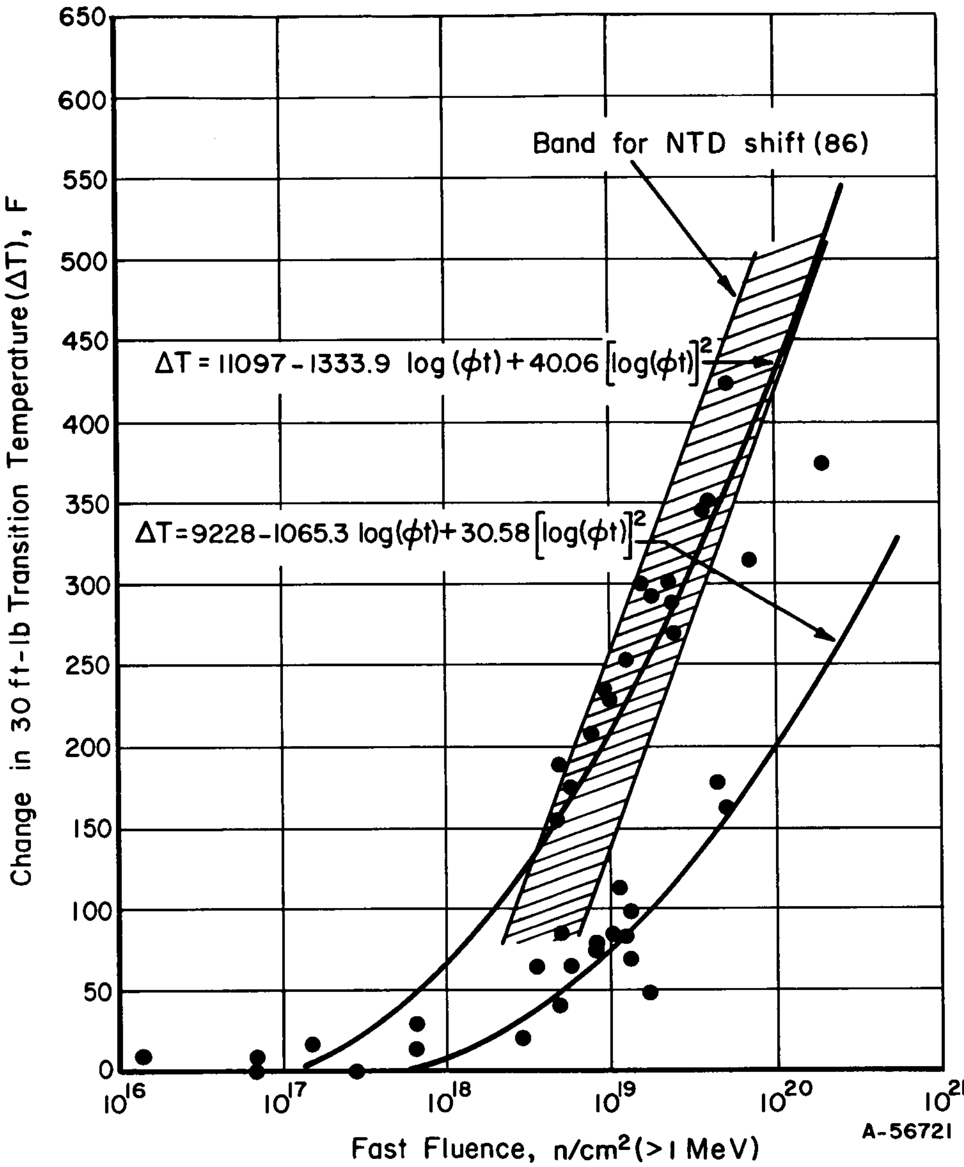


FIGURE 33. EFFECT OF IRRADIATION ON VARIOUS HEATS OF A302-B STEEL[87]

"Sensitive" and "insensitive" irradiation behaviors are demonstrated by the upper and lower curves, respectively.

(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, is important in predicting steel sensitivity to irradiation it becomes difficult to separate the effect of chemical composition from those of microstructure. Limited studies have shown that nickel content (3.8 to 8.4 wt %) affects the irradiation sensitivity of steel only by changing its heat treatment response and consequently the final microstructure.[(88a)]

Earlier results indicated that 1.1 wt % uranium added to steel increased the sensitivity to irradiation; but now the results are being attributed to the effect of uranium on the microstructure of steel.[(89)] Also, the induction-melting removal of residual elements has improved the irradiation resistance of steel.[(88a)] This finding agrees with most theoretical predictions attributing irradiation hardening to the interstitial atoms.[(90)]

Demands for higher strength in pressure vessels, owing to increased pressures and temperature requirements, have prompted investigations of high-strength steels (80,000 to 180,000-psi yield strength) for possible pressure-vessel applications. The high-strength steels investigated consisted of 3.5Ni-Cr-Mo, 7.5Ni-Cr-Mo, SSS-100, 12Ni-5Cr-3Mo, and 9Ni-4Co. These alloy steels exhibited lower preirradiation transition temperatures and were found to be less sensitive to irradiation embrittlement.[(88b)] Also, an earlier saturation limit in irradiation-induced NDT shift was reached for the high alloy steels. This saturation limit was reached still earlier at higher irradiation temperatures.

(3) Stress. Specimens of A350 were stressed up to 80 percent of their yield strength during irradiation at 430 F while receiving a fast fluence of 3×10^{19} n/cm^2.[(91)] Impact tests on these specimens, shown in Figure 34, indicated that applied stress has no effect on irradiation embrittlement of steel. In another test 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. Results of these tests are shown in Figure 35.[(92)]

(4) Direction. Radiation response of longitudinal and transverse (to rolling direction) specimens of A212-B and A302-B plate was studied. The results shown in Figure 36[(93)] indicate that both transverse and longitudinal specimens are about equally sensitive to irradiation. The impact energy of the weaker transverse specimens is reduced by a 1.1×10^{20} n/cm^2 fast fluence to markedly less than the criterion of 30 ft-lb for shift in transition temperature.

(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. To test the effect of irradiation on heat-affected zones, irradiations were carried out on welded metal and base metal from the same material.[(94)] Test results on the irradiated material are given in Table 16. These limited data suggest that base metal appears to be more sensitive to irradiation

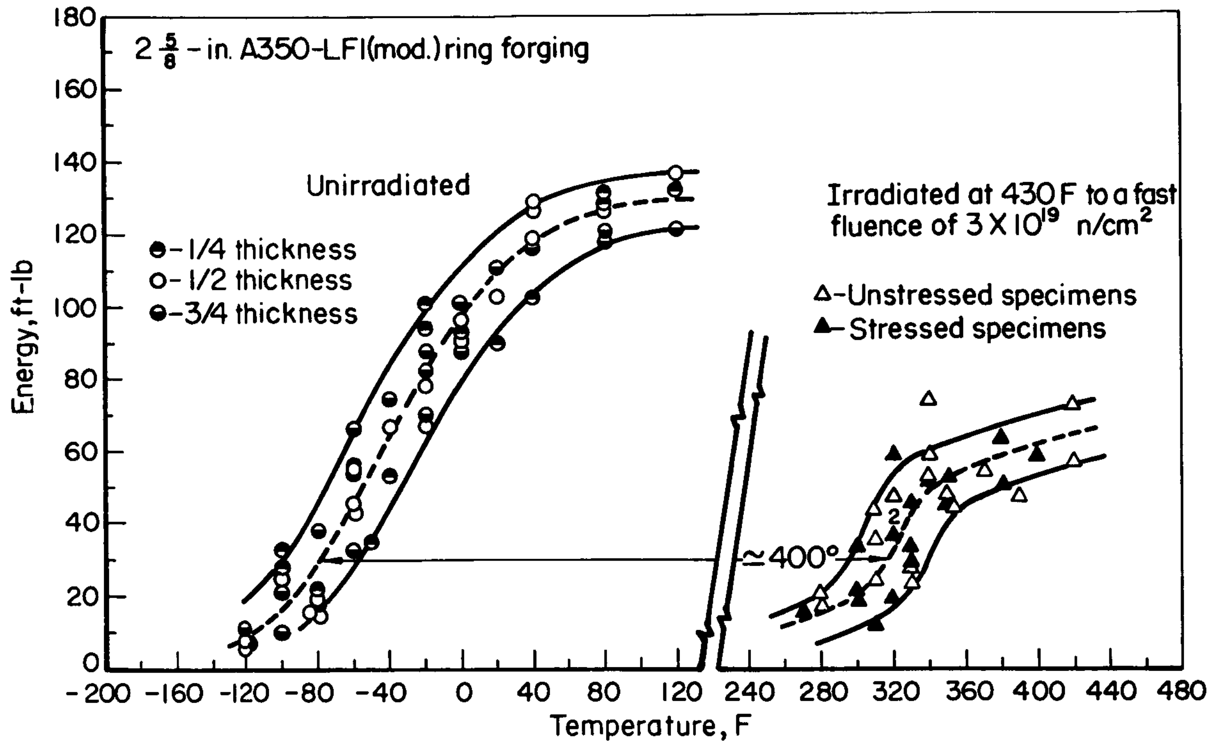


FIGURE 34. COMPARISON OF NOTCH DUCTILITY PERFORMANCE OF A350-LF1 (MODIFIED) STEEL IRRADIATED IN THE UNSTRESSED AND STRESSED CONDITIONS[91]

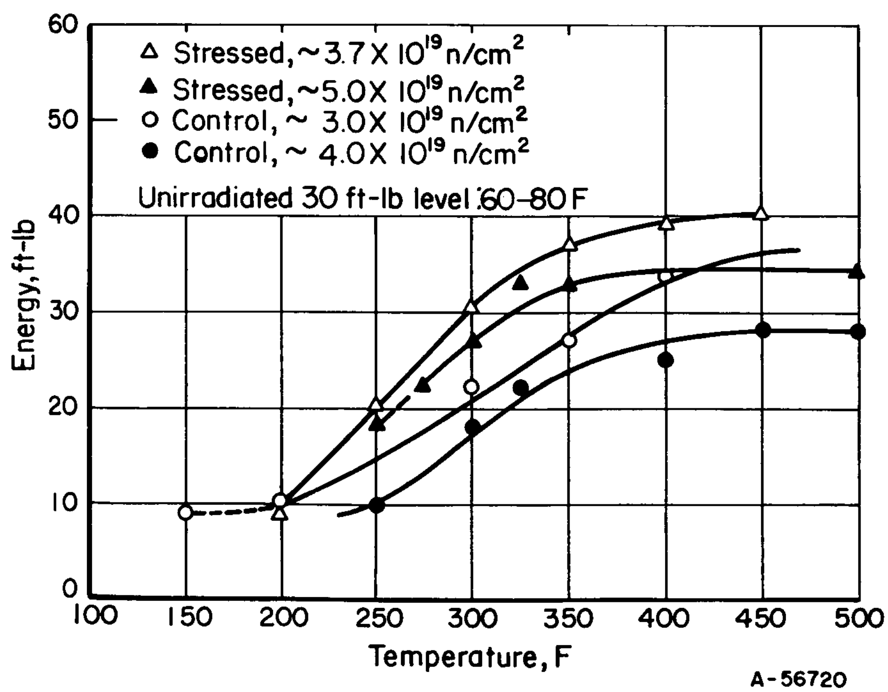
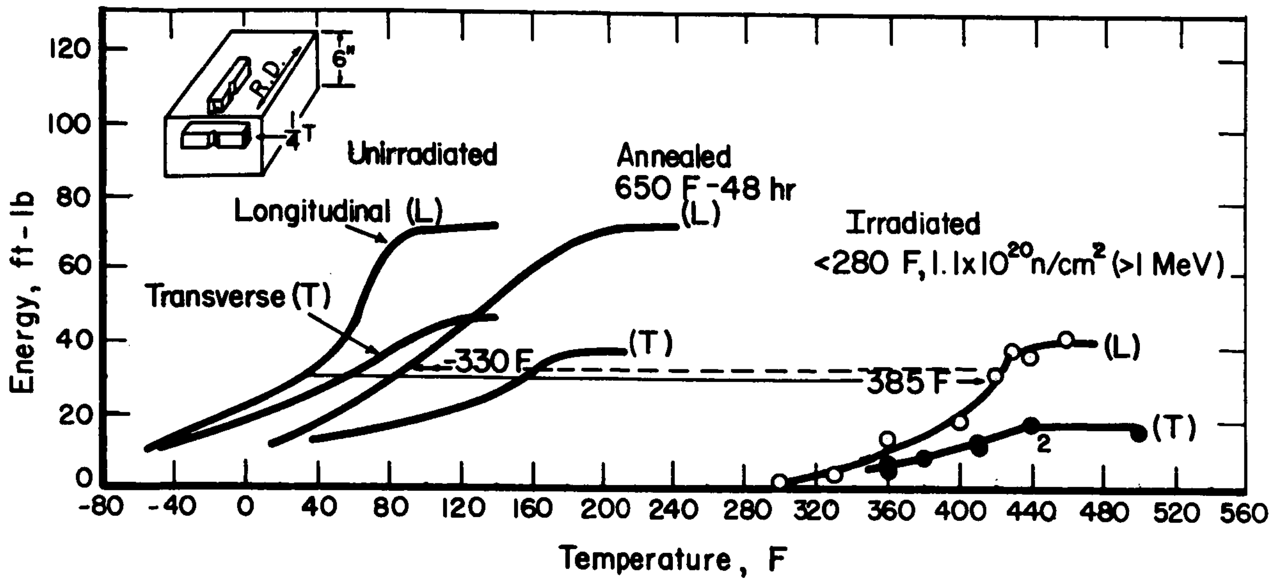
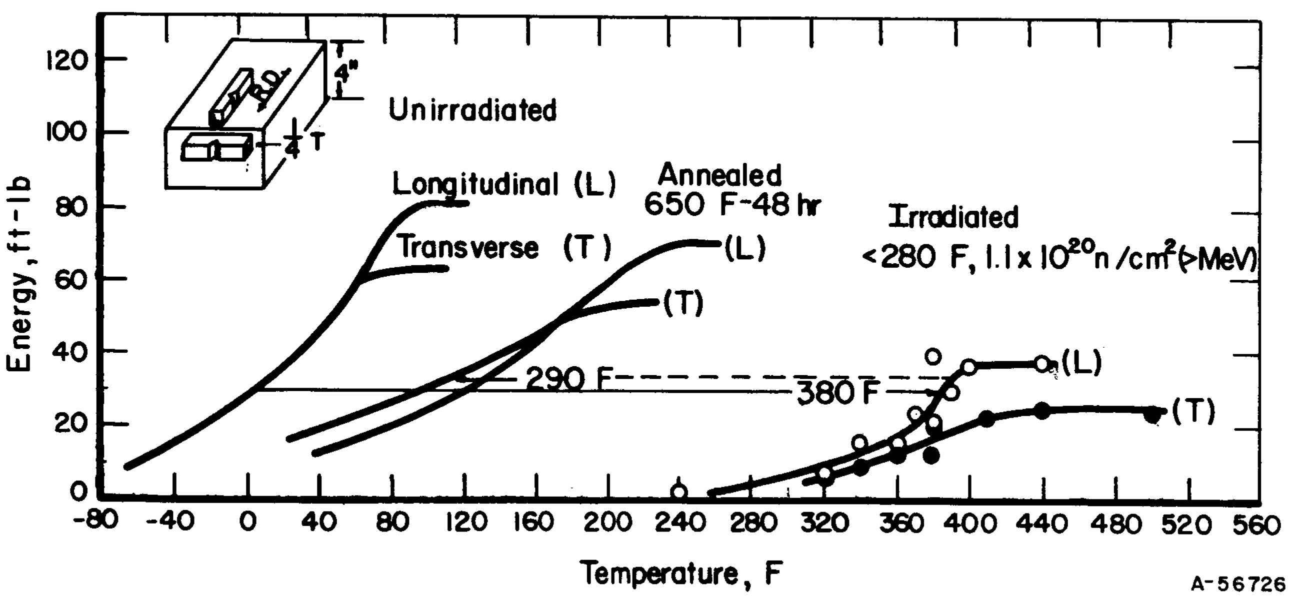


FIGURE 35. EFFECT OF STRESS ON THE IRRADIATION-INDUCED NDT SHIFT IN LOW-CARBON STEEL[92]



a. 6-Inch A302-B Plate

A NDT shift of 385 F is produced by irradiation and 330 F of this NDT shift is recovered by 1-hour anneal at 680 F.



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b. 4-Inch A212-B Plate

A NDT shift of 380 F is produced by irradiation, and 290 F of this NDT shift is recovered by 1-hour anneal at 680 F.

FIGURE 36. NOTCH-DUCTILITY BEHAVIOR OF A212-B PLATE IN THE UNIRRADIATED, IRRADIATED, AND POST-IRRADIATION ANNEALED CONDITIONS SHOWING THE RADIATION RESPONSE IN LONGITUDINAL AND TRANSVERSE DIRECTIONS[93]

than the weld metal. On the other hand, some investigators have found welds to be more sensitive to irradiation embrittlement than the base metals.[88c]

TABLE 16. NOTCH-DUCTILITY CHARACTERISTICS OF BASE PLATE AND WELD HEAT-AFFECTED ZONE OF A212-B AND A302-B STEELS BEFORE[a] AND AFTER IRRADIATION TO A FAST FLUENCE OF 3.1×10^{19} N/CM^2 (>1 Mev)[94]

Material	Irradiation Temperature, F	Form	30 Ft-Lb Transition Temperature, F			Full-Shear Energy Absorption, ft-lb		
			Initial	Irrad.	Increase	Initial	Irrad.	Decrease
A212-B	500	Base plate	5	240	235	81	51	30
		HAZ[b]	10	210	200	(a)	56	--
A302-B	550	Base plate	30	185	155	72	63	9
		HAZ[b]	65	195	130	(a)	56	--

(a) Not fully documented for unirradiated condition.
(b) HAZ - heat-affected zone.

(6) Temperature. Temperature during irradiation is a very important factor in influencing the degree of irradiation embrittlement. Figures 37, 38, and 39 illustrate how irradiation temperature affects the transition-temperature shift in various steels.[86, 95a] Maximum embrittlement has been found to be caused by irradiation temperatures below 450 F; the exact temperature where maximum embrittlement occurs is difficult to establish (Figure 37).[94] With progressively higher irradiation temperatures, the shift in transition temperature, at equivalent fast fluences, is continuously lower. Figure 39 shows that the specimens must be irradiated at temperatures in the 1200 to 1400 F range to avoid any increase in NDT.[95a] The NDT shift progressively decreases with increasing irradiation temperature because the 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) Varying of Temperature During Irradiation. The effect of low irradiation temperature followed by high irradiation temperature is illustrated in Figure 40.[96a] In this experiment, specimens were irradiated at 400 F to a fast fluence of 2.3×10^{19} n/cm^2 and reirradiated at 540 F with an additional fast fluence of 1.6×10^{19} n/cm^2. It appears that sufficient annealing takes place at 540 F to almost completely nullify the effects of irradiation at 400 F.

Another study was made by irradiating a batch of specimens at only 490 F; another batch of identical specimens was irradiated first at 490 F and later at 350 F.[96a] Both specimen batches were irradiated to the same fast fluence. Comparison of the NDT shifts indicated that the specimens irradiated last at 350 F underwent larger NDT changes. Results of the tests are given in Table 17. The magnitude of the NDT shift is not consistent among the steels, but they generally follow a previously established trend of irradiation sensitivity except for the HY-30 steel.

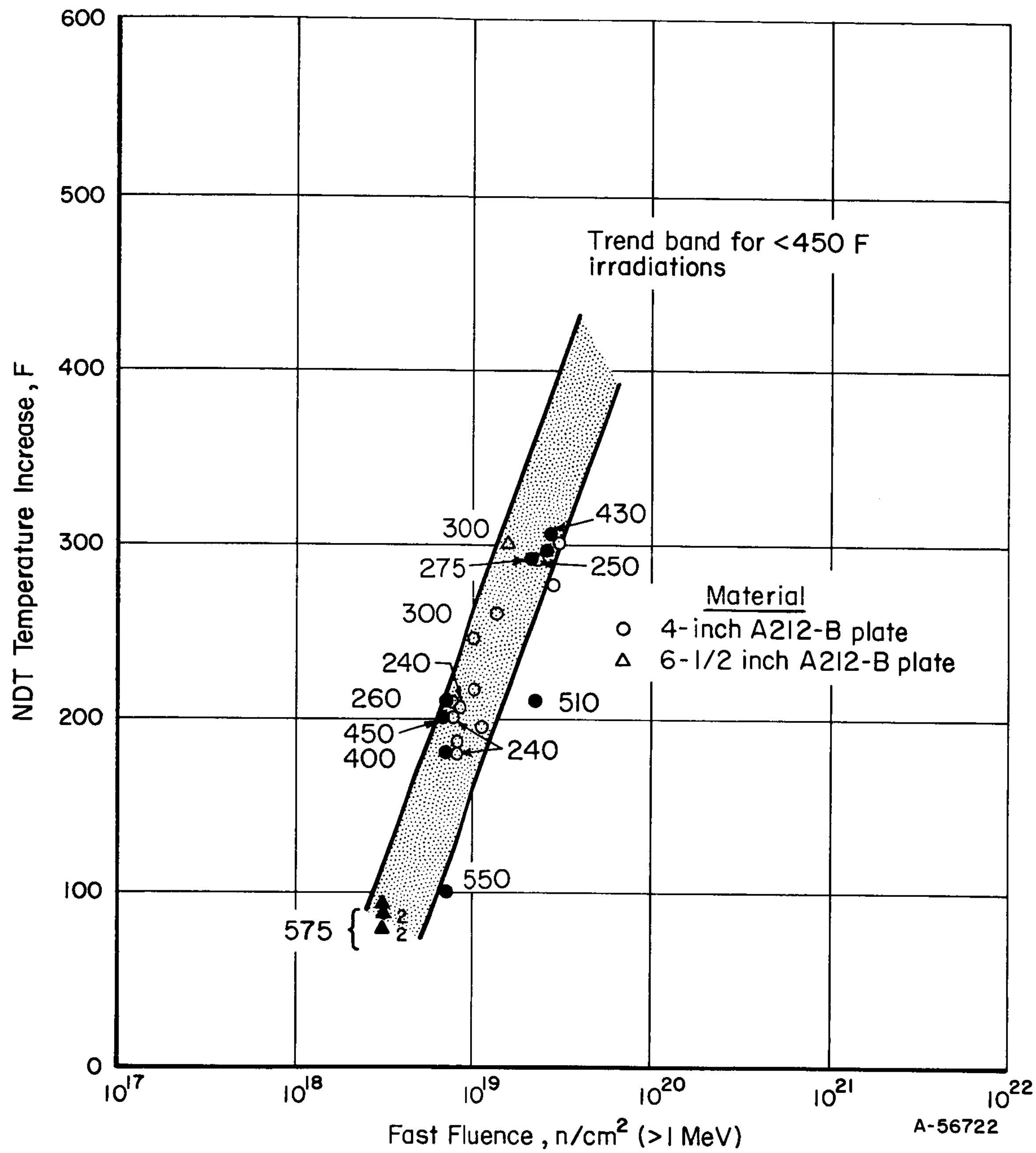


FIGURE 37. INCREASE IN THE NDT TEMPERATURES OF ASTM A212-B STEELS RESULTING FROM NEUTRON IRRADIATION AT VARIOUS TEMPERATURES(94)

Numbers indicate irradiation temperatures. Irradiation temperature is 200 F if no number is given.

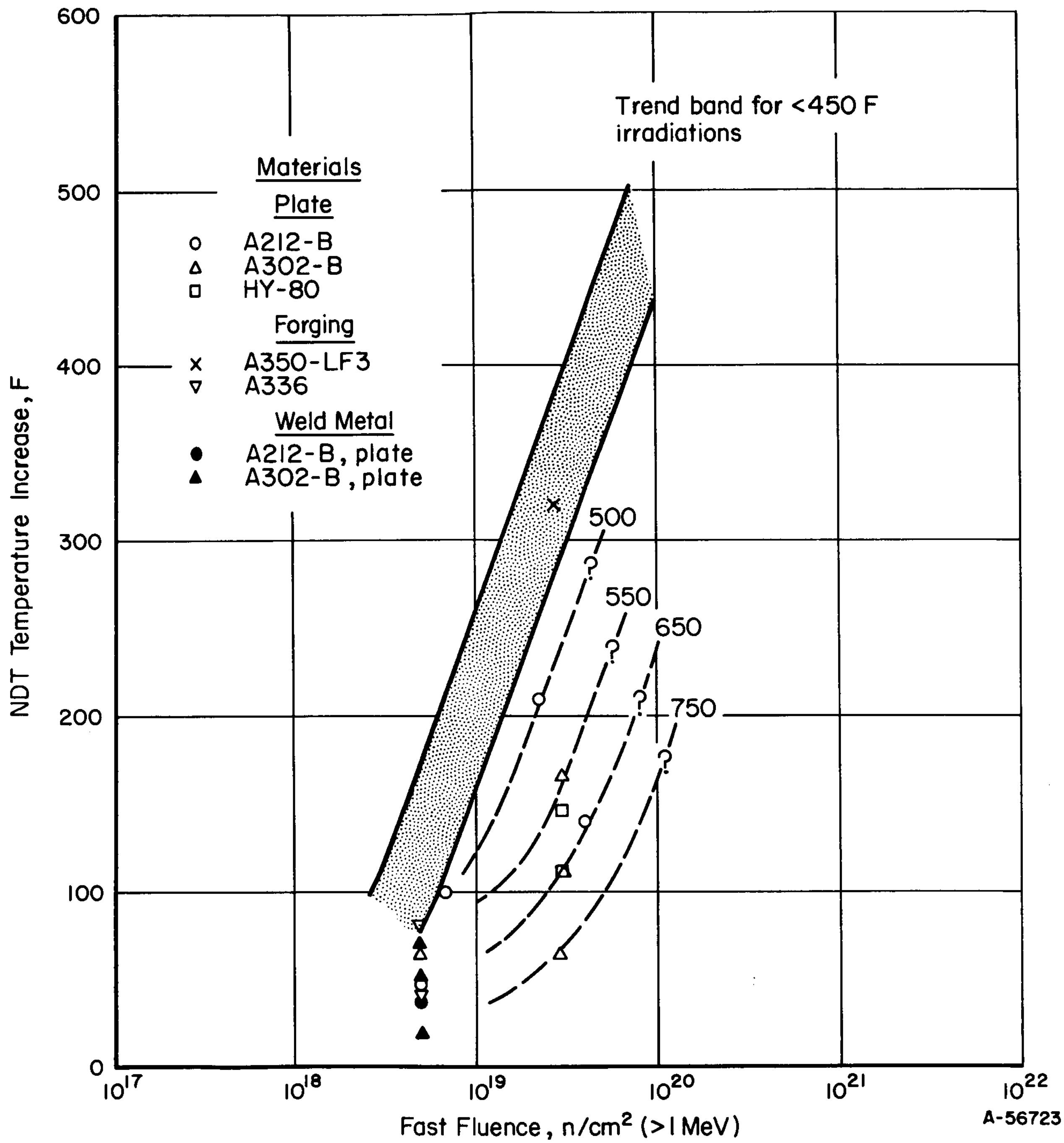


FIGURE 38. INCREASE IN THE NDT TEMPERATURES OF STEELS RESULTING FROM IRRADIATION AT TEMPERATURES ABOVE 450 F[86]

Numbers indicate irradiation temperatures.

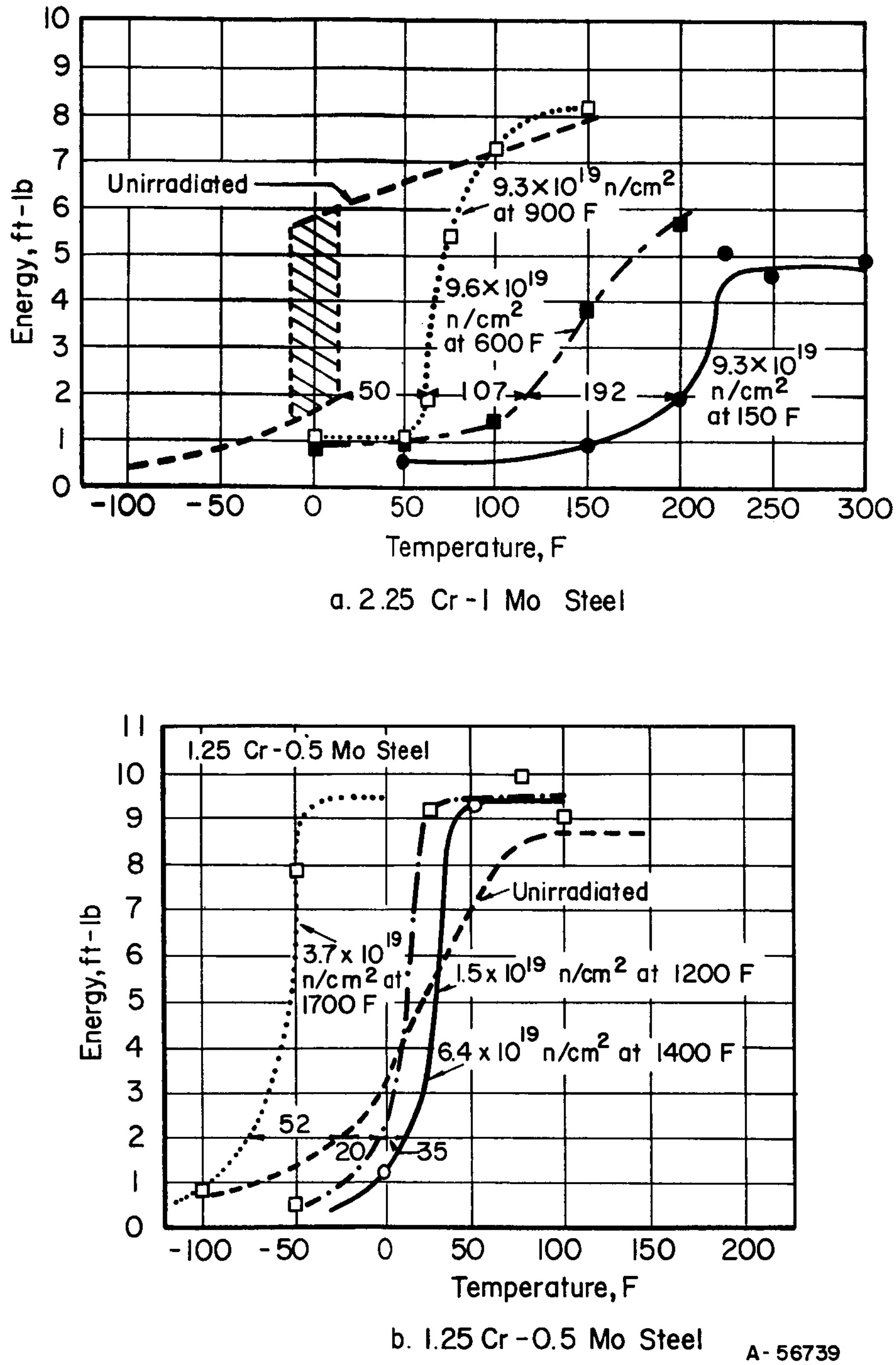


FIGURE 39. EFFECT OF NEUTRON RADIATION AND IRRADIATION TEMPERATURE ON THE IMPACT PROPERTIES OF CHROMIUM-MOLYBDENUM STEELS[95a]

TABLE 17. COMPARISON OF NOTCH DUCTILITY BEHAVIOR OF STEELS IRRADIATED CONTINUOUSLY AT 490 F WITH THEIR BEHAVIOR UNDER IRRADIATION AT 490 F FOLLOWED BY IRRADIATION AT 350 F[96a]

Material	ΔT, F, 1.4×10^{19} n/cm^2 at 490 F	ΔT, F, 1.0×10^{19} n/cm^2 at 490 F plus 5×10^{18} n/cm^2 at 350 F	Difference, ΔT, F
A212-B	230	255	25
A350-LF 3	270	335	65
A353	200	240	40
A302-B	200	230	30
HY-80	100	170	70

(8) Saturation. Figure 32 illustrates an increasing NDT shift with increasing fluence but, because of the many variables, only a band of values could be established. To determine a saturation point in the NDT shift due to fast neutrons, specimens from one heat of A302-B were irradiated to various fast fluence levels. The irradiation temperatures were kept below 250 F. Results of the Charpy V-notch tests (Figure 41) indicate a linear fast-fluence dependence to about 1×10^{19} n/cm^2. At that point, the relative effect of fast fluence on NDT decreases, but there is still a linear relationship although the slope is lower up to the maximum fast fluence of 7×10^{19} n/cm^2.[96a] It also appears that the fast fluence needed for saturation in irradiation-induced effects in steels depends on the irradiation sensitivity of the material. It was illustrated earlier that the high-alloy nickel-chromium-manganese steels, which are less sensitive to irradiation, reach a saturation in irradiation-induced effects faster than do other steels. Swedish researchers have illustrated this point by comparing the fast fluence required for saturation in a typical steel and a more irradiation-sensitive weld metal. As seen in Figure 42, the NDT shift for the 2103/R3 steel has almost reached saturation after a fast fluence of 1×10^{19} n/cm^2, while the NDT shift for the weld metal is still increasing steeply.[88c]

(9) Annealing. Results of annealing studies at various temperatures on irradiated A350-LFI are shown in Figure 43.[96a] 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 3×10^{19} n/cm^2. After irradiation, the specimens irradiated at 640 F were annealed at 800 F and the specimens irradiated at 740 F

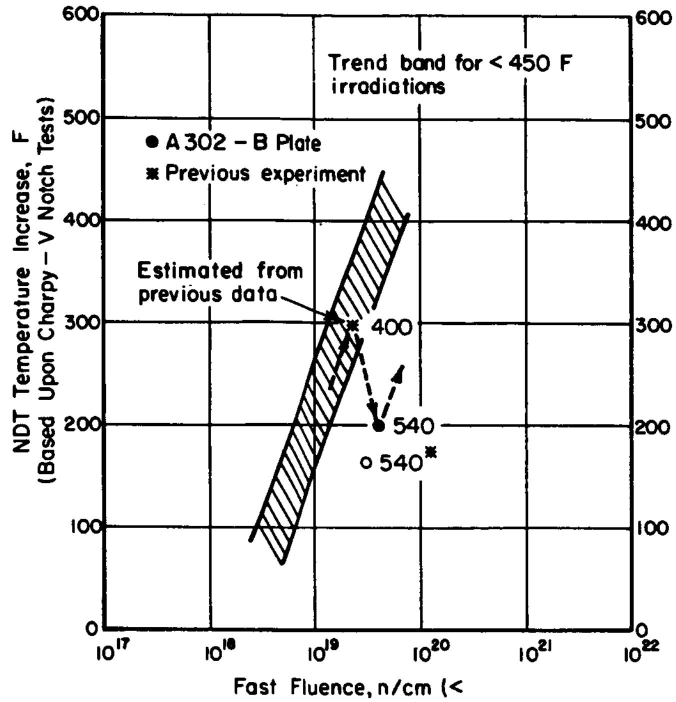


FIGURE 40. NDT TEMPERATURES OF A302-B STEEL RESULTING FROM IRRADIATION IN A TWO-PHASE (TWO-TEMPERATURE) SCHEDULE; LOW (400 F), THEN HIGHER TEMPERATURE (540 F)[96a]

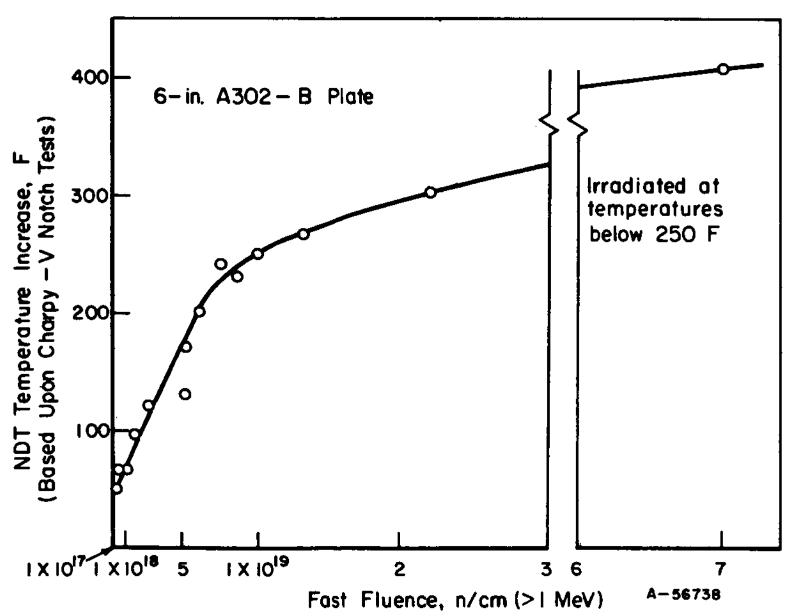


FIGURE 41. INCREASE IN NDT TEMPERATURE OF A302-B STEEL IRRADIATED TO VARIOUS FAST FLUENCES AT TEMPERATURES BELOW 250 F[96a]

Linear graph permits direct comparison of effects for various exposure levels.

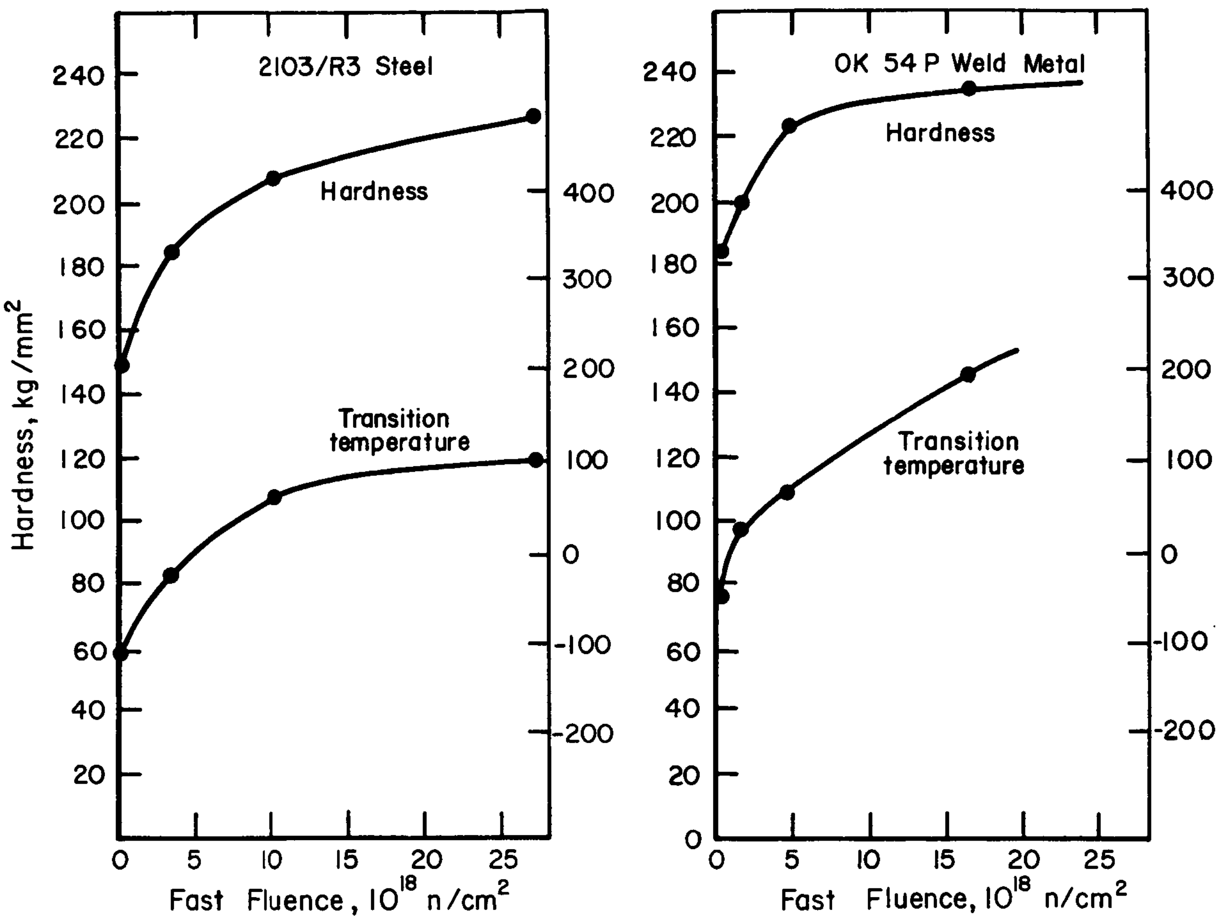


FIGURE 42. EFFECT OF IRRADIATION ON THE HARDNERS AND SHIFT OF TWO SWEDISH STEELS (2103/R3 and OK 54 P) AS A FUNCTION OF FAST FLUENCE[(88c)]

Steels were irradiated at 105 F.

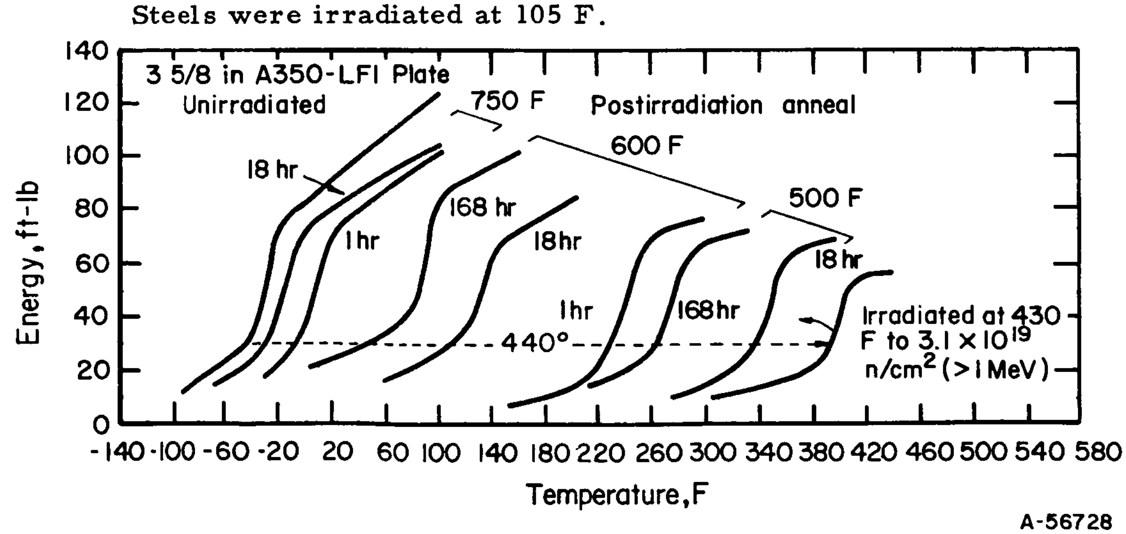


FIGURE 43. NOTCH-DUCTILITY CHARACTERISTICS OF IRRADIATED A350-LF1 STEEL[(96a)]

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

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 and HY-80 steels.(96a) 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 44. It can be seen that the cumulative increase in NDT is somewhat less for the cyclically irradiated and annealed material than for the material irradiated to the same fluence without intervening annealing. Irradiations followed by annealing again, followed by further irradiations, were also performed by Westinghouse on A302-B steel.(87) The results (shown in Figure 45) indicate that any reduction in the NDT shift obtained was very minor. The best results in minimizing NDT shifts during irradiation were achieved by annealing the maximum number of times at the maximum temperature (Table 18).(96a)

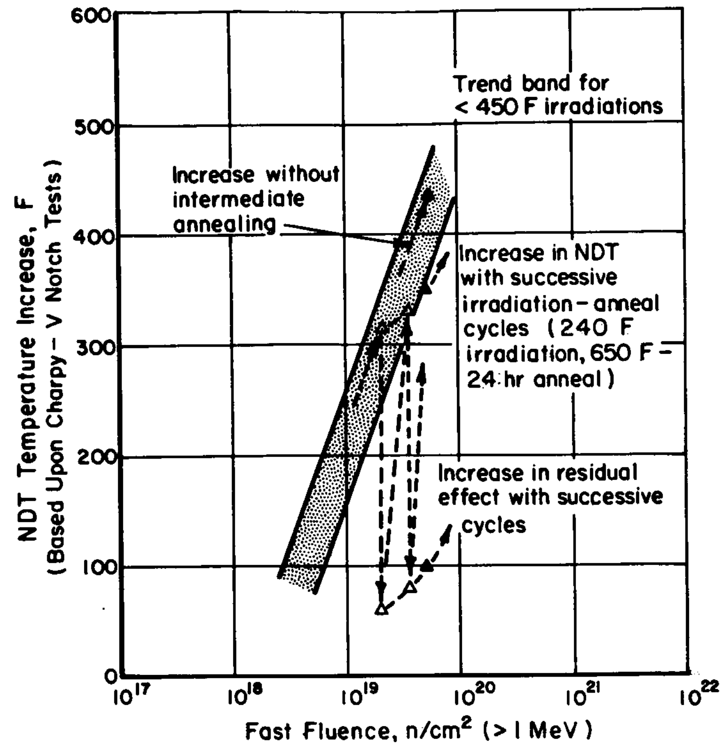
TABLE 18. EFFECT OF INTERMEDIATE ANNEALING ON THE NDT INCREASE OF IRRADIATED A350-LF1 STEEL(96a)

Material Condition	Fast Fluence, 10^{19} n/cm^2 (>1 MeV)	Measured ΔNDT, F, With Annealing	Estimated(a) ΔNDT, F, Without Annealing	Net Gain With Annealing ΔT,	Net Gain With Annealing Percent
Control	2.8	415	...	...	...
Annealed midcycle 550 F, 24 hr	3.1	405	430	25	5.8
Annealed midcycle 600 F, 24 hr	3.3	400	440	40	9.1
Annealed quarter-cycle 600 F, 24 hr (three times)	3.6	315	455	140	30.8

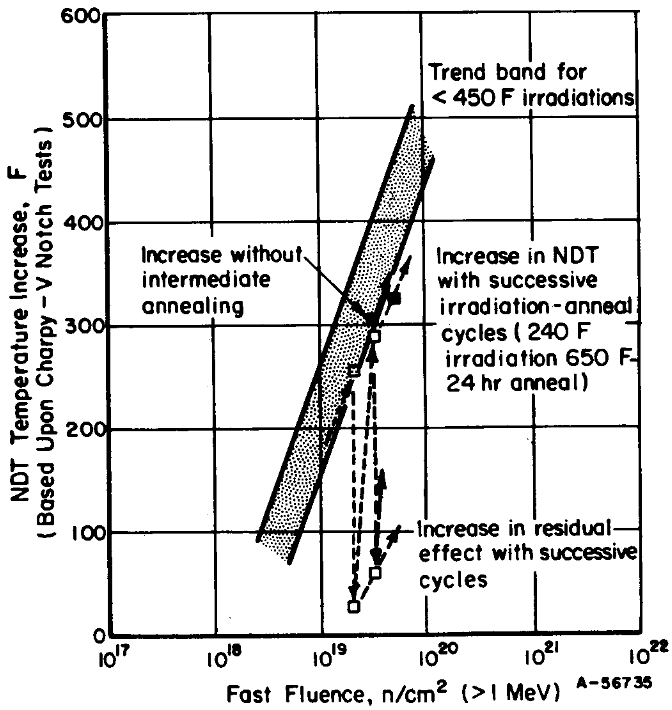
(a) Based on previous 430 F irradiations of this material.

(10) Location in Reactor. The reactor pressure vessel is of sufficient thickness (up to 12 inches) 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.(97) The specimens received a relative fast-fluence variation of about nine, depending on their location; the consequent shifts in NDT are shown in Figure 46.

(11) Grain Size. It has been demonstrated that the NDT increases considerably more in large-grained steels than in fine-grained steels during irradiation (Figure 47).(88d) The model for predicting the NDT shift was found to be dose-rate dependent at elevated temperatures.



a. 6 – Inch A 302 – B Plate



b. 3 – Inch HY – 80 Plate

FIGURE 44. NDT TEMPERATURE BEHAVIOR EXHIBITED BY TWO STEELS AT VARIOUS STAGES OF CYCLIC IRRADIATION-ANNEALING TREATMENTS[96a]

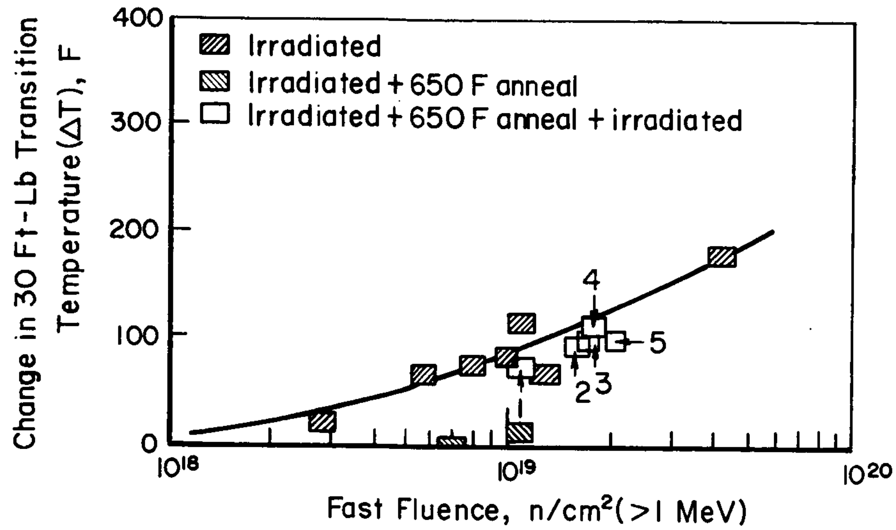


FIGURE 45. EFFECT OF IRRADIATION, POSTIRRADIATION ANNEALING, AND REIRRADIATION ON THE TRANSITION TEMPERATURE OF A302-B Mn-Mo STEEL(82)

The size of the data point indicates its uncertainty.

Estimated neutron exposures (±10 percent) of post-irradiation-annealed specimens prior to reirradiation:

(1) 8.3×10^{18} n/cm^2
(2) 8.5×10^{18} n/cm^2
(3) 1.4×10^{19} n/cm^2
(4) 7.5×10^{19} n/cm^2
(5) 1.0×10^{19} n/cm^2

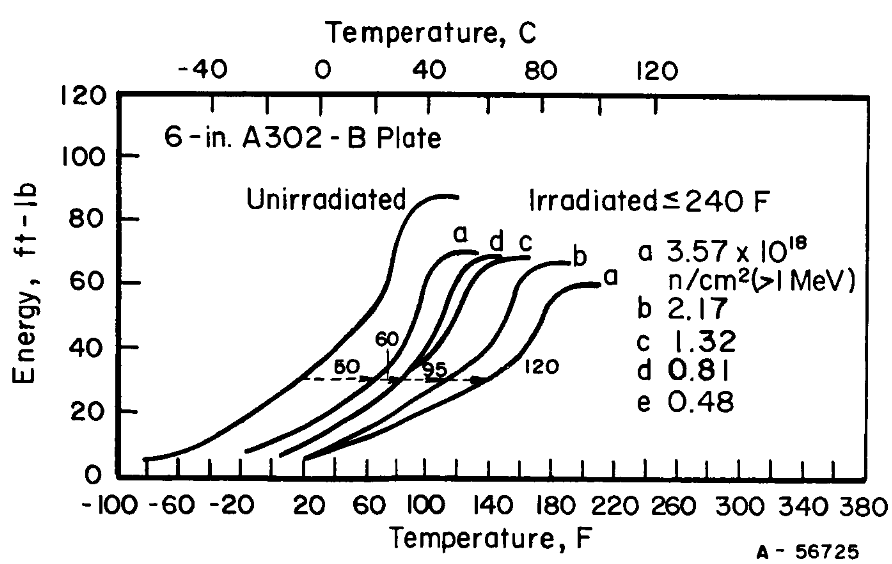


FIGURE 46. NOTCH-DUCTILITY PROPERTIES OF A302-B STEEL AT FIVE LOCATIONS INSIDE TEST BLOCK(97)

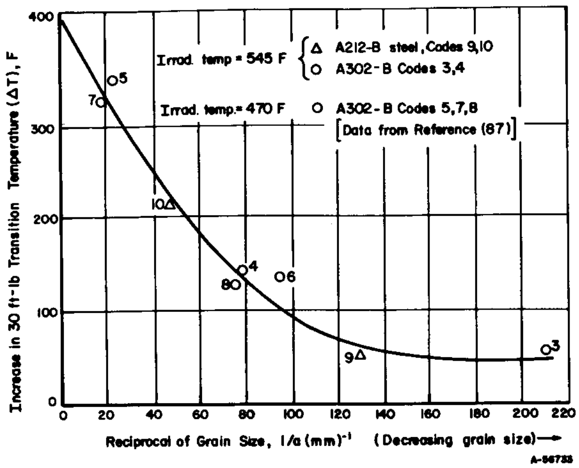


FIGURE 47. INCREASE IN 30 FT-LB TRANSITION TEMPERATURE VERSUS RECIPROCAL GRAIN RADIUS UPON IRRADIATION TO A FAST FLUENCE OF 3 x 10^{19} N/CM^2 (>1 MEV) AT 470 AND 545 F(88d)

(12) Prestrain. To test the effect of strain on irradiation-induced NDT shift, Charpy V-notch specimens of Swedish 2103/R3 steel were elongated 10 percent and aged for 1 hour at 480 F before irradiation.(98) The strain-aged specimens were irradiated along with unstrained control specimens to a fast fluence of 2.4 x 10^{18} n/cm^2. Results of the impact tests are given in Table 19.

TABLE 19. EFFECT OF STRAIN AGING ON IRRADIATION EMBRITTLEMENT OF 2103/R3 STEEL(98)

	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

From these results it was concluded that the transition-temperature shifts caused by irradiation and straining are additive.

Since the shift in NDT is of such importance for pressure-vessel integrity, it is believed obligatory to observe NDT changes during the lifetime of the reactor pressure vessel. Almost all operating power reactors include a continuous surveillance program to monitor irradiation-induced changes in properties of the pressure vessel. These surveillance programs usually consist of Charpy V-notch specimens and tensile specimens made from the same heat of metal as the pressure vessel. These specimens are located close to the pressure-vessel wall so they can receive fast fluences at temperatures equivalent to those received by the pressure vessel. However, the construction of the pressure vessel prevents exact duplication of the irradiation conditions. Usually, the specimens are placed inside the pressure vessel where they receive an accelerated fast fluence at somewhat higher temperatures than they do at the pressure-vessel wall. This situation may be unfortunate, but it is deemed more conservative than placing the specimens outside the pressure vessel at a lower fast fluence and temperature. An advantage of accelerated neutron accumulation is to keep "ahead" of the fast fluence of the pressure vessel. This means that the irradiation-induced property changes in the surveillance samples are experienced first. Of distinct disadvantage in irradiating the surveillance samples inside the pressure vessel is the somewhat higher temperature during irradiation. If the irradiation temperature of the surveillance specimens is sufficiently higher than the pressure vessel, then considerable annealing of the irradiation-induced NDT shift may take place in the specimens. This possibility should always be considered when results of a surveillance program are being evaluated. The first sets of surveillance capsules for the pressure vessels of Yankee, Army SM-1 and SM-1A, Piqua, and Big Rock Point reactors have been tested.(99) A comparison of the transition-temperature shift undergone by the surveillance specimens with the shift predicted by previous irradiation data is shown in Figures 48 and 49. The surveillance specimens for both Big Rock Point and Yankee reactors were placed between the reactor core and the pressure-vessel wall; these specimens experienced irradiation at a somewhat higher temperature than did the pressure vessel. Since the irradiation temperatures were somewhat higher, a lower shift in transition temperature would be expected. However, there appears to be good agreement between the data obtained in the surveillance program and the data for the test material irradiated at about 550 F, the estimated irradiation temperature of the surveillance specimens.

Tests on the Piqua reactor surveillance specimens indicated considerable irradiation-induced deterioration of impact strength of A212-B steel (Figure 50).(93) No unirradiated control specimens were available for impact testing so only thermal control specimens could be used for comparison. The thermal control specimens had been treated at the same temperature and time as had the irradiated specimens, but were not irradiated. Testing of the thermal control specimens indicated low impact-strength values, and annealing of the irradiated specimens at 800 F resulted in comparable values. The low impact strength was puzzling, since the composition was found to conform to ASTM specifications. However, a metallographic examination indicated that both the irradiated and thermal control specimens were transverse from the rolling direction, and would be expected to be somewhat weaker than the specimens longitudinal from the rolling direction.

Effect of Irradiation on Mechanical Properties

In connection with surveillance programs on pressure-vessel steels, there has been considerable speculation as to the effect of fast fluence accumulation rate on irradiation-induced property changes. The settling of this question is of utmost importance because a reactor pressure vessel accumulates its fast fluence in up to 40 years, while the specimens used to predict its behavior accumulate an equivalent fast fluence in a matter of months. The British irradiated three different types of steels (En-2, aluminum killed, and silicon killed) at fast fluences in the proportion of 1:4:100 to each other.(100) The consequent changes in yield and ultimate strengths and the ductility of the samples were found to be dependent on total fast fluence but not on the rate of fast fluence accumulation.

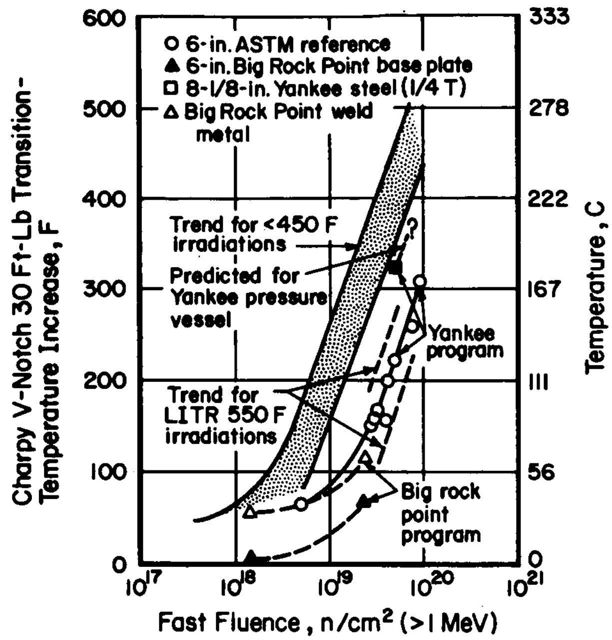


FIGURE 48. CHANGE IN TRANSITION TEMPERATURES OF YANKEE AND BIG ROCK POINT PRESSURE VESSELS(99)

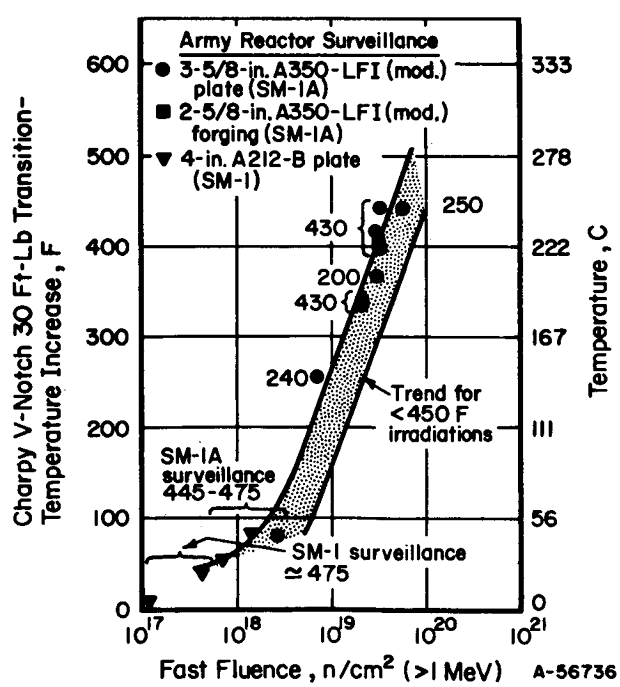


FIGURE 49. CHANGE IN TRANSITION TEMPERATURES OF SM-1 AND SM-1A PRESSURE VESSELS(99)

Numbers refer to irradiation temperatures.

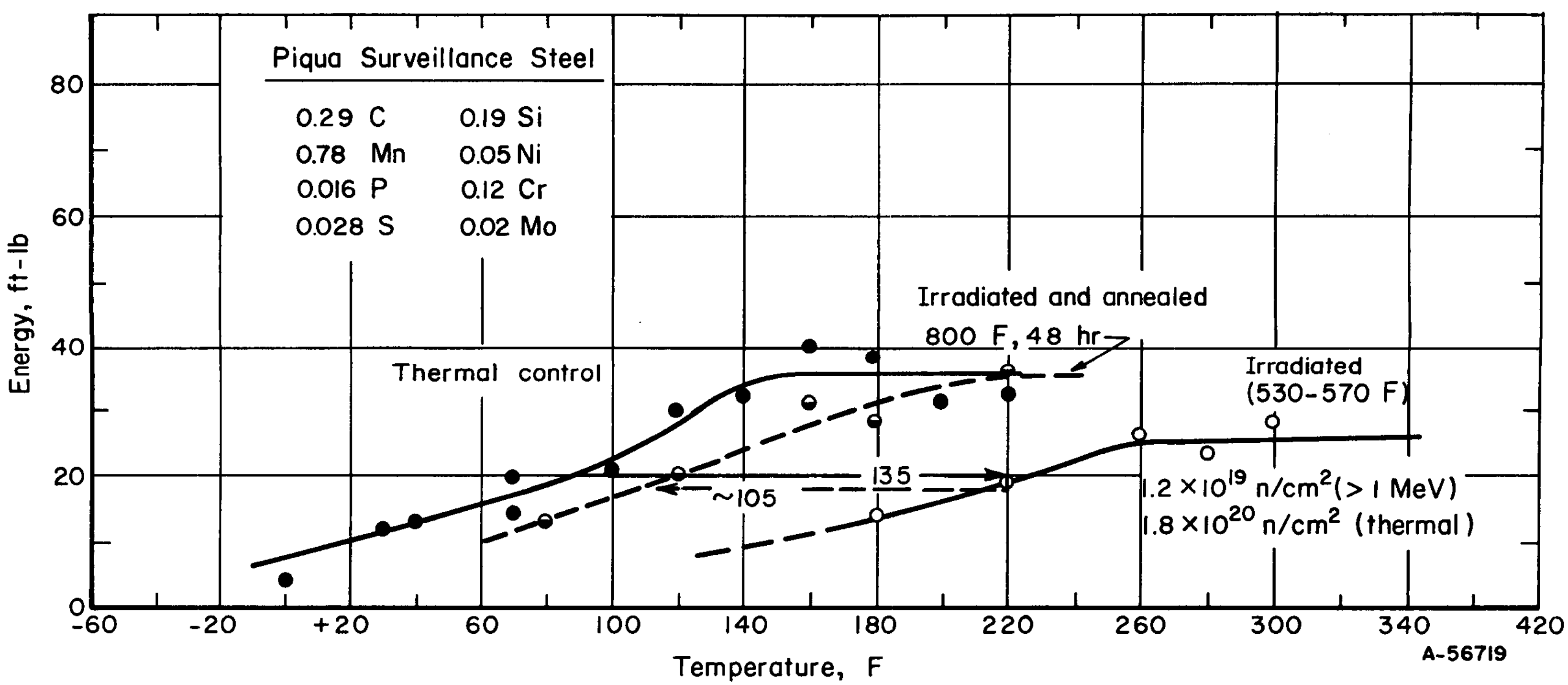


FIGURE 50. NOTCH-DUCTILITY CHARACTERISTICS OF THE TRANSVERSE PIQUA SURVEILLANCE CHARPY SPECIMENS(93)

Note the recovery of unirradiated properties by postirradiation heat treatment indicated by the half-closed circles.

A similar experiment was conducted at Oak Ridge where specimens of pure iron were irradiated at instantaneous fast fluences of 2×10^{11} to 3×10^{13} n/cm^2 at 200 F. A fast fluence of 4.6×10^{18} n/cm^2 increased yield strength threefold, but no fast fluence dependence of the yield strength increase was detected.(88e)

Very few significant tensile data on pressure-vessel steels have been generated since the publication of Report REIC 20.(101) Most of the reported tensile tests have been performed on steels irradiated at elevated temperatures. Tensile tests were performed on low-alloy steels (1.25 wt % chromium-0.5 wt % molybdenum and 2.25 wt % chromium-1 wt % molybdenum) irradiated at temperatures of 100 to 1700 F. A general trend in the effect on irradiation temperature on the tensile properties of the steels after a fast fluence of 3×10^{19} n/cm is given in Figure 51.(95a) From Figure 51 it can be seen that the maximum irradiation effect on tensile properties is achieved by irradiation at about 700 F. However, one would expect the irradiation sensitivity of tensile properties to vary with composition, microstructure, and heat treatment as was the case with impact properties. Recent annealing studies on a low-alloy steel, which had received a fast fluence of 2×10^{19} n/cm^2 below 210 F, indicated that maximum "thermal hardening" occurred at 200 C (392 F) as shown in Figure 52.(102) "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 lowest temperature for maximum "thermal hardening" reported is 210 F.(100) Generally, one would expect maximum thermal hardening temperature to be higher for high alloy steels.(103) 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 Swedes have irradiated various steels at varying temperatures with increasing fluences. Results of these studies are shown in Figure 53. These results illustrate that saturation of irradiation effects on yield strength occurs at lower fluence levels for the higher irradiation temperatures.(104) The effect of increasing fast fluence on the room-temperature strength and ductility of the same steels is shown in Figure 54. These results show that at an irradiation temperature of 105 F, no saturation of irradiation-induced mechanical-property changes appears to take place.(88c) The tests on two different steels again illustrate the variance in irradiation sensitivity of various steels which may not differ much in composition.

Annealing of irradiated steels completely restores the preirradiation mechanical properties, provided the annealing temperature is high enough. Results of annealing studies by the British are given in Figure 55.(95b) The annealing kinetics have also been found to be time dependent as illustrated in the isothermal annealing curves of Figure 56.(95b) These annealing temperatures are in reasonable agreement with the irradiation temperatures found by Lowe(95a), where no irradiation-induced changes in mechanical properties take place.

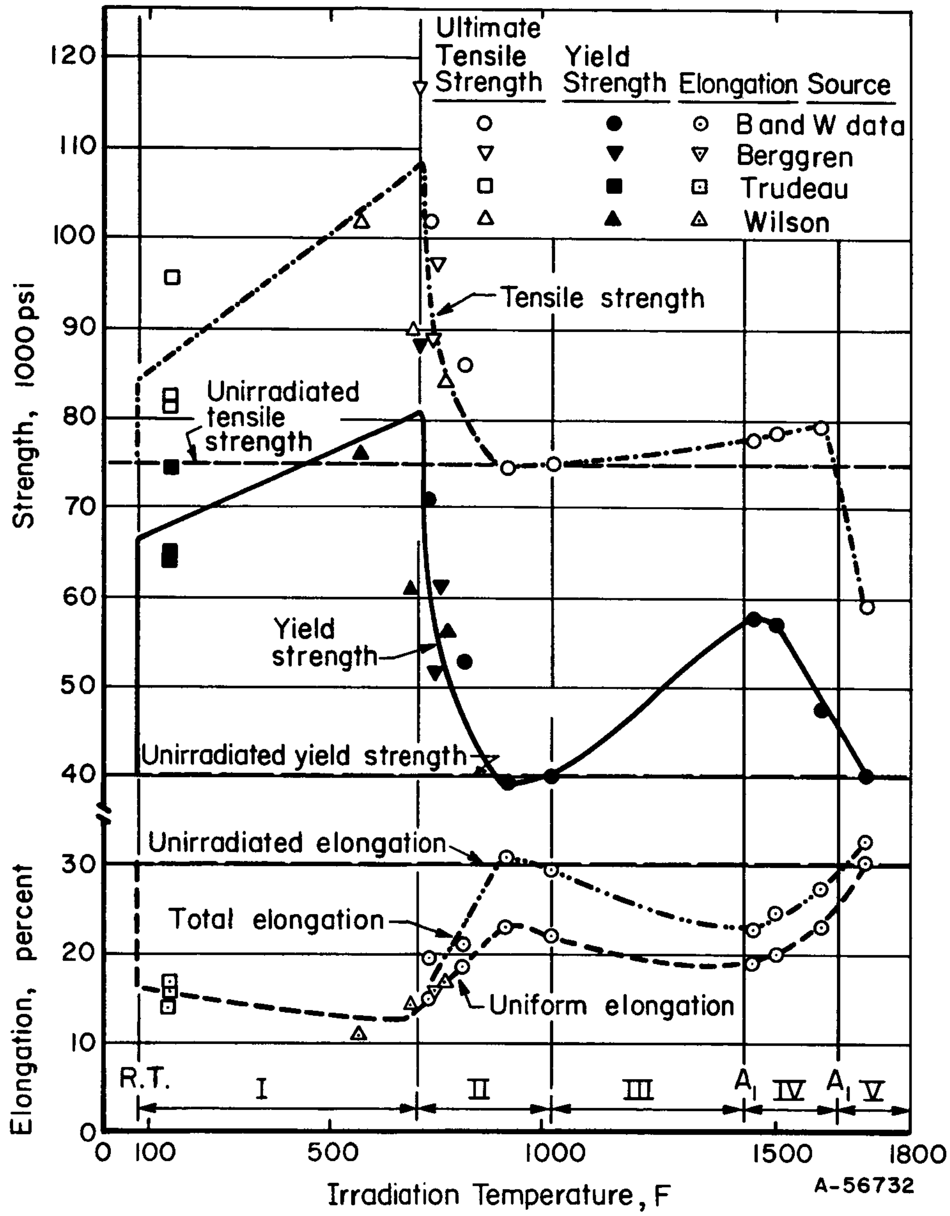


FIGURE 51. GENERAL EFFECT OF IRRADIATION TEMPERATURE ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF LOW-ALLOY STEELS IRRADIATED TO ~3 x 10^{19} N/CM^2 (> 1 MEV)(95a)

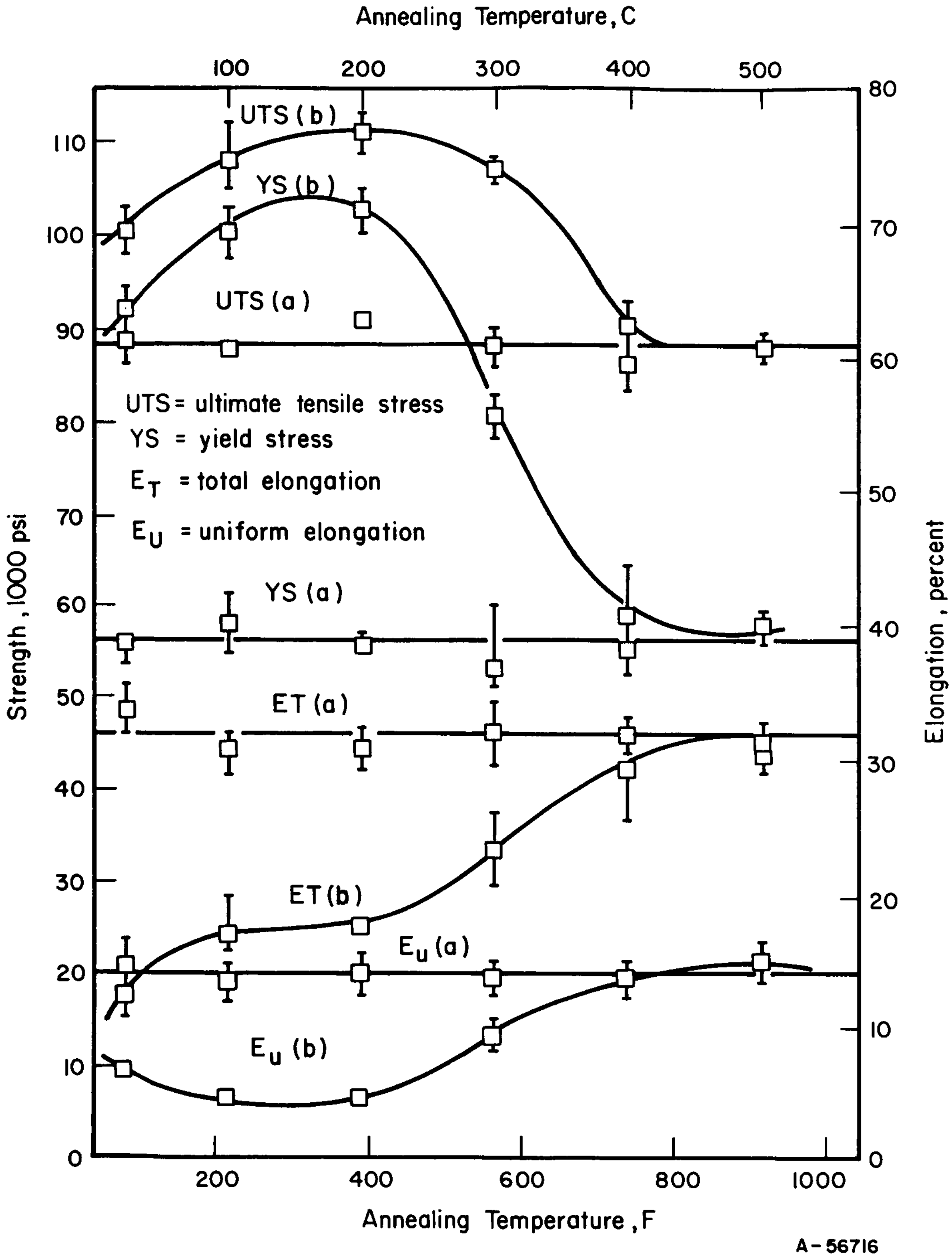


FIGURE 52. EFFECT OF POSTIRRADIATION ANNEALING ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF IRRADIATED LOW-ALLOY STEEL(102)

(a) Unirradiated.
(b) Fast fluence of 2 x 10^{19} n/cm^2.

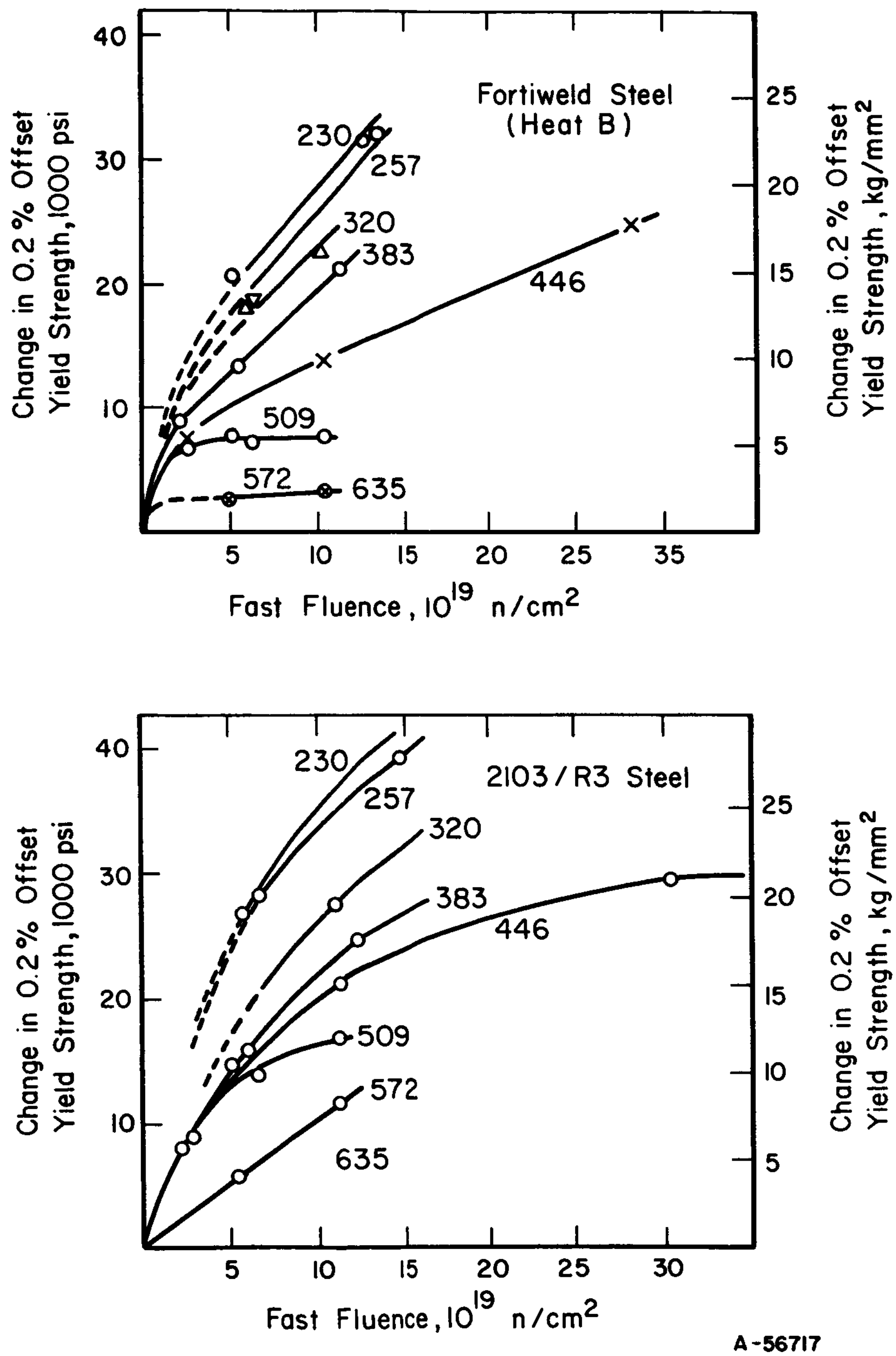


FIGURE 53. CHANGES IN 0.2 PERCENT OFFSET YIELD STRENGTH AT ROOM TEMPERATURE OF TWO STEELS AS A FUNCTION OF FAST FLUENCE AT DIFFERENT IRRADIATION TEMPERATURES[104]

Numbers indicate irradiation temperature, F.

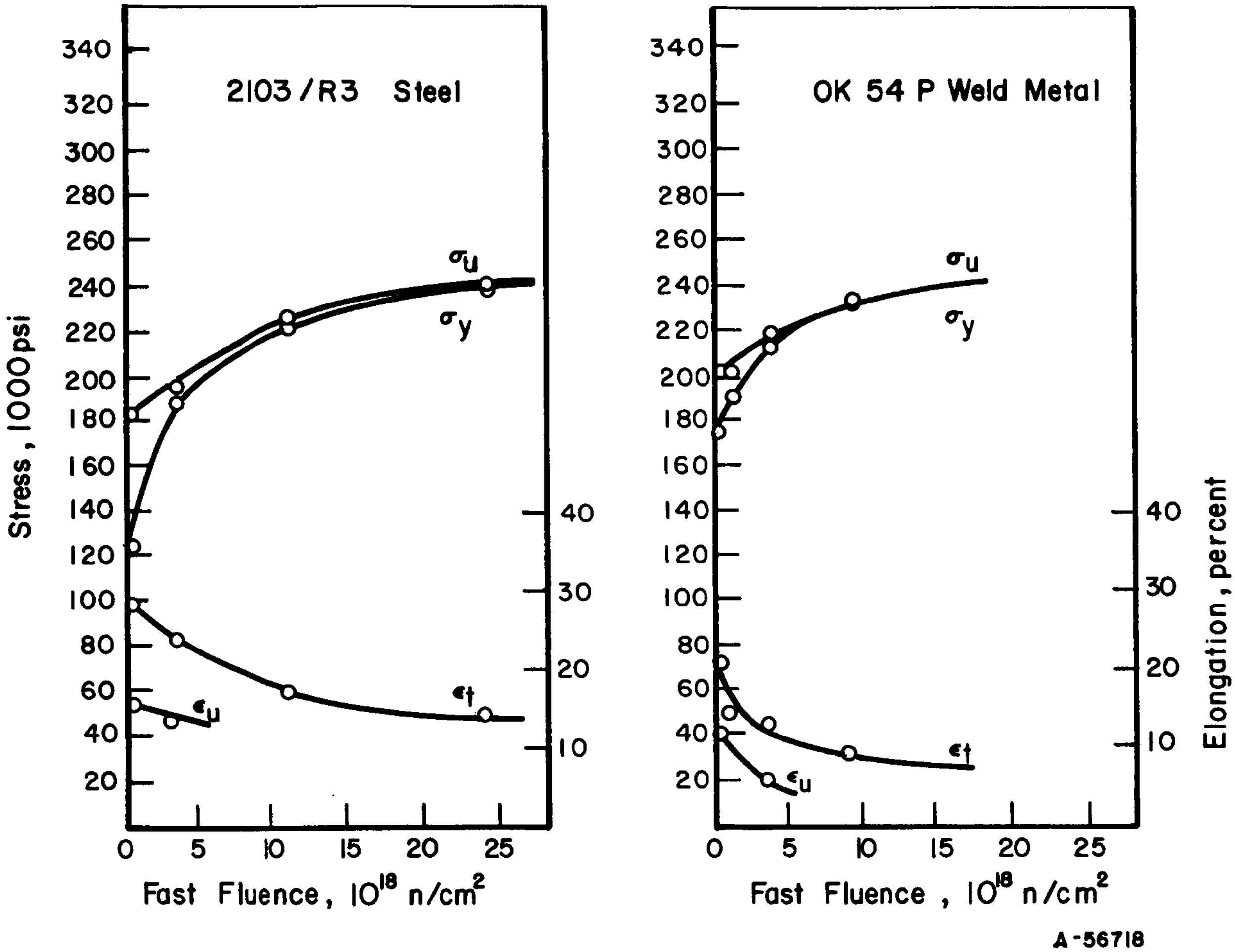


FIGURE 54. EFFECT OF IRRADIATION ON THE ROOM TEMPERATURE TENSILE PROPERTIES OF TWO SWEDISH STEELS 92103/R3 AND OK 54P) AS A FUNCTION OF FAST FLUENCE(88c)

Steels were irradiated at 105 F.

σ_u = Ultimate tensile strength.
σ_y = 0.2% offset yield strength.
E_t = Total elongation.
E_u = Uniform elongation.

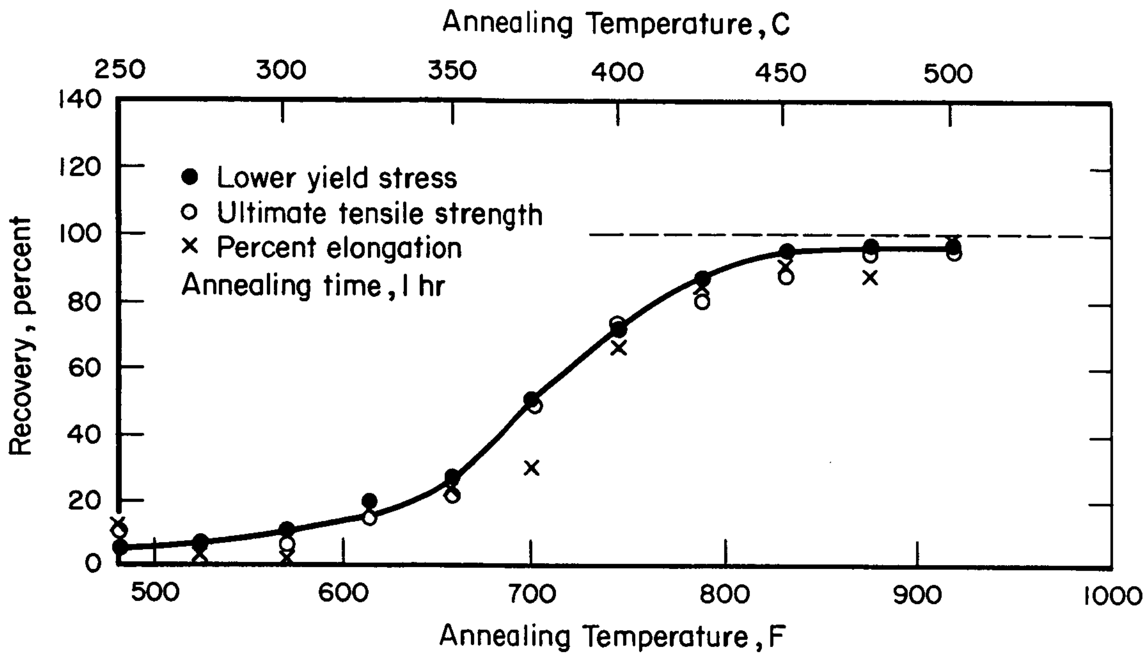


FIGURE 55. ISOCHRONAL RECOVERY CURVE FOR THE ROOM-TEMPERATURE TENSILE PROPERTIES OF A SILICON-KILLED CARBON STEEL (0.24 PERCENT CARBON, 0.15 PERCENT SILICON, 0.55 PERCENT MANGANESE) IRRADIATED IN THE CALDER REACTOR AT 150 C TO A FLUENCE OF 4.0 x 10^{18} N/CM^2 (> 1 MEV)(86)

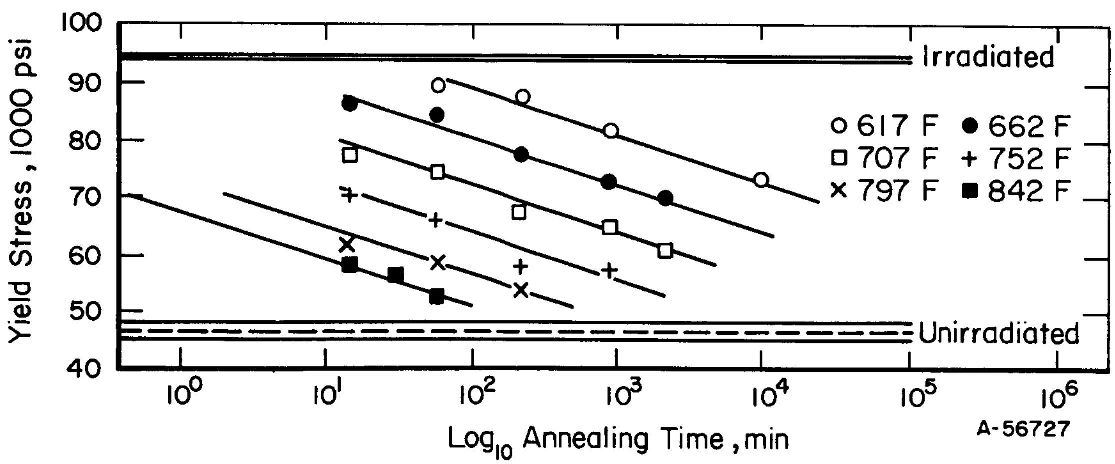


FIGURE 56. ISOTHERMAL ANNEALING CURVES FOR THE ROOM-TEMPERATURE TENSILE YIELD STRESS OF A SILICON-KILLED CARBON STEEL (0.24 PERCENT CARBON, 0.15 PERCENT SILICON, 0.55 PERCENT MANGANESE) IRRADIATED IN THE CALDER REACTOR AT 150 C TO A FAST FLUENCE OF 9.5 x 10^{18} N/CM^2 (> 1 MEV)(95b)

The British have developed a formula for predicting irradiation-induced yield-strength changes in steels. The original theory, as originated by Cottrell[105], postulated yield-strength change as a function of cube root of total fluence. However, experimental work has indicated that yield strength varies with the square root of total fluence[95b], and is in agreement with Makin's[106] theoretical work. The change in yield strength can be calculated by the formula: $\Delta y.s. = A (\theta t)^{1/2}$, where A is a material constant or "irradiation sensitivity" of a specific material and θt is the fast fluence. Since irradiation effects on yield strength are not fluence-rate dependent, it is concluded that the controlling factor is not the diffusion of point defects.[96b] Since the irradiation effects depend on total fast fluence, it is possible that these effects are caused by formation of vacancy and interstitial defect clusters. The temperature dependence of irradiation-induced property changes indicates that the stability of these clusters is also temperature dependent. At higher temperatures, only the larger clusters are stable and the smaller clusters are either annihilated or merge with the larger clusters. Therefore, the increase of irradiation or annealing temperature decreases the number of clusters; and although the hardening effect of each large cluster is more than that of the small cluster, the overall hardening effect is lessened by reducing the number of clusters with the higher irradiation temperatures. Eventually the temperature becomes high enough that even the most stable clusters are annihilated.

Considerable experimental work has been done to investigate the effect of varying interstitial content on irradiation sensitivity of steels. It has been shown that both carbon[88e, 90, 100, 107, 108] and nitrogen[88e, 109] in solution help to form complex clusters contributing to irradiation embrittlement.

Catastrophic delayed failure due to hydrogen occurs in high-strength steels but not in low-strength steels like A212-B which are used for pressure-vessel construction. Since irradiation increases the strength of A212-B, there is a possibility that A212-B would become susceptible to hydrogen embrittlement after irradiation. To determine the effect of hydrogen on irradiation embrittlement, Rossin[110] tested both unirradiated and irradiated hydrogen-impregnated specimens. For controls, unirradiated and irradiated specimens without any hydrogen were tested. The test results indicated that the embrittlement effects caused by irradiation and hydrogen are not additive. Also, not enough hydrogen is expected to be produced by (n, p) reactions in the steel during irradiation to cause embrittlement, if the fast fluence is low enough to satisfy the allowable NDT shift.

The effect of hydrogen on irradiated 1Cr-0.5Mo pressure-vessel steel has been studied.[111] Smooth and notched tensile specimens were tested after hydrogen impregnation and irradiation at room temperature, 212, 392, and 482 F. The irradiated specimens received a fast fluence of 1 to 1.4×10^{19} n/cm^2. The rather limited testing suggests that the ductile-to-brittle transition temperatures of both the unirradiated and irradiated steel are increased by hydrogen. However, the hydrogen does not appear to affect the notch strength and the irradiated steel is not susceptible to hydrogen-induced delayed fracture.

Tests on unirradiated material have shown that the ductility, in a biaxially loaded tube, is reduced one-half of that shown by a simple uniaxial tension test. Tubes of A302-B steel irradiated to a fluence of 5 to 12×10^{18} n/cm^2 were tested biaxially at temperatures of -32, 150, and 300 F.[96c, 112] The results of these tests are plotted in Figure 57. In addition, the irradiated tubes showed considerably less ductility when loaded biaxially than they did when tested in simple tension. At 150 F, no significant differences were detected in the effective biaxial stress required to fail the irradiated specimens; but at 300 F, considerable scatter in effective stress was found.

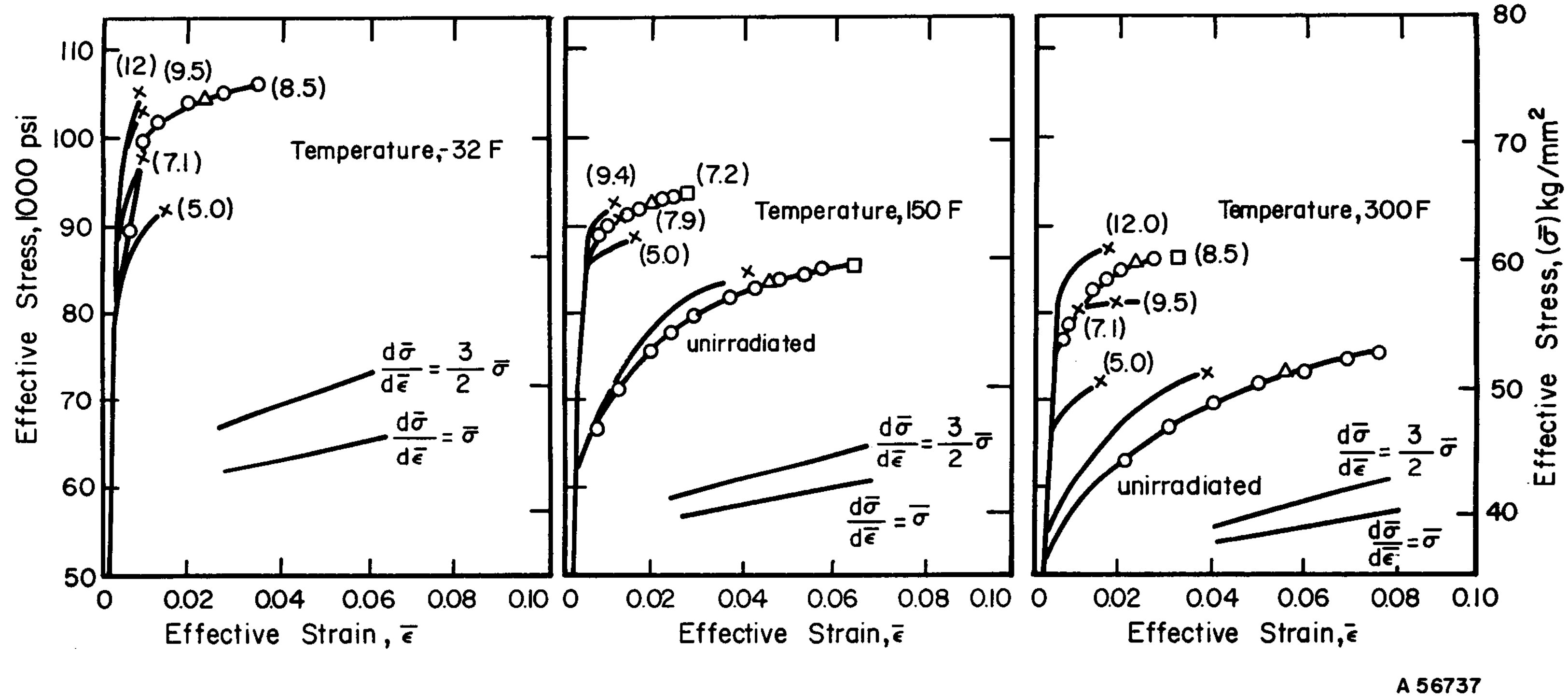


FIGURE 57. EFFECTIVE STRESS-EFFECTIVE STRAIN CURVES FOR A302-B STEEL TUBE SPECIMENS TESTED AT VARIOUS TEMPERATURES AFTER IRRADIATION TO VARIOUS FAST-FLUENCE LEVELS.(96c)

Numbers in parenthesis indicate fast-fluence levels in 10^{18} n/cm^2.

Tube, tested in simple tension
Tube, equal biaxial tension
End uniform strain
Tube burst
Instability strain predicted from $\frac{d\bar{\sigma}}{d\bar{\epsilon}} = \frac{3}{2}\bar{\sigma}$

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 58.(113) 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 while it decreased the low-cycle life. The improvement of the endurance limit takes place after about 10^4 to 10^5 cycles.(114)

Effect of Irradiation on Other Properties

Some other property changes found for mild steels are summarized below:

(1) 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.(115) On the other hand, it has been reported that a fast fluence of 1×10^{18} n/cm^2 decreased the wear resistance of carbon steels at room temperature.(116)

(2) 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.(117)

(3) Work Hardening. The rate of work hardening decreases at all strains after irradiation to a fast fluence of 1.7×10^{18} n/cm^2, as shown in Figure 59.(118) Annealing for 1 hour at 212 F further decreased the work-hardening rate. Anneals at 400 to 750 F for 1 hour increased the rate of work hardening but did not restore properties to the level of unirradiated material.

(4) 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.(119) The irradiation formed vacancies and interstitials but did not change the room-temperature strength. However, the 100-hour stress-rupture strength at 842 F was increased from 27,300 to 29,400 psi.

Structural Steels

In this section, ferritic and martensitic steels used in applications other than reactor pressure-vessel construction are discussed. Table 20 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.

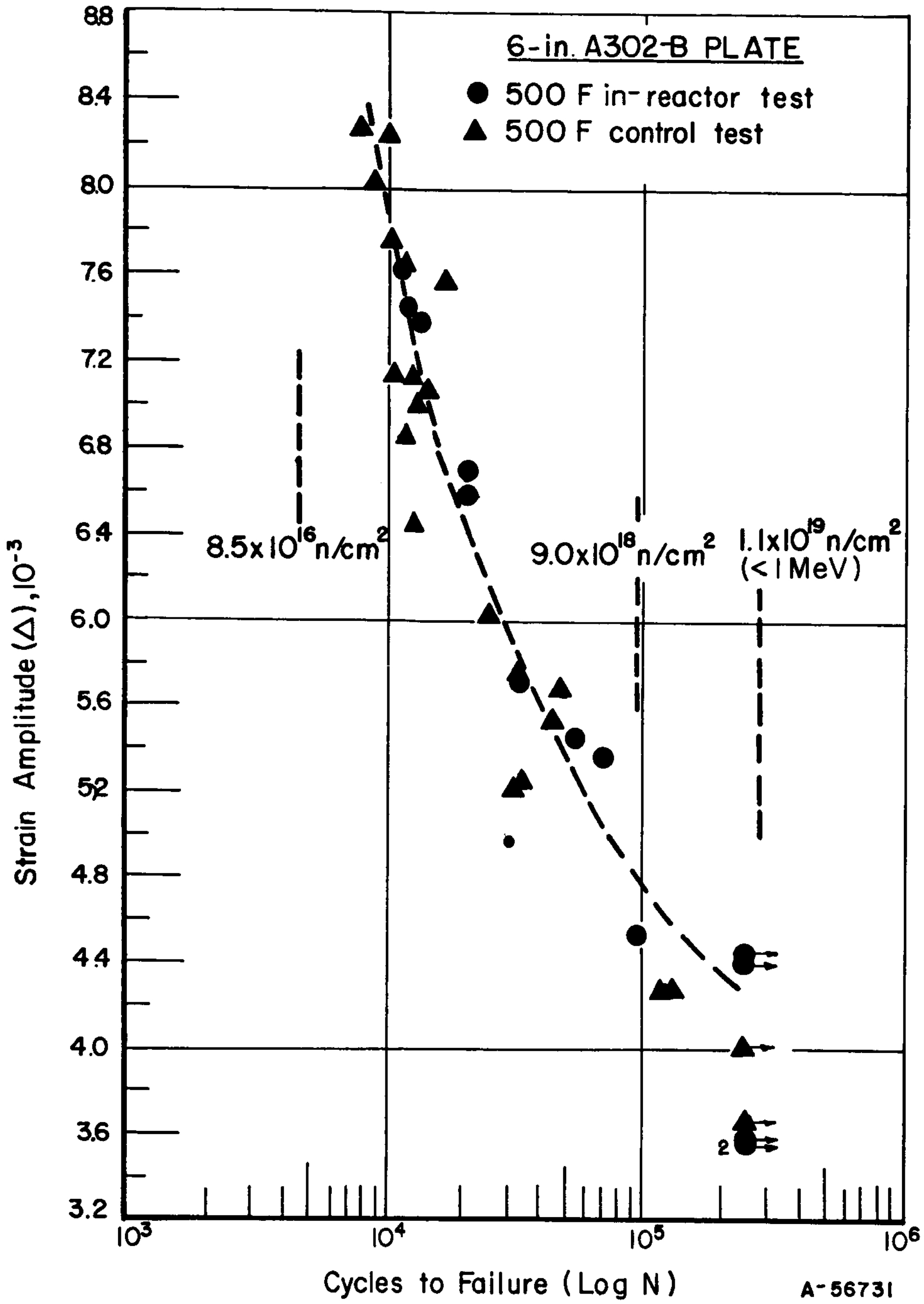


FIGURE 58. COMPARISON OF FATIGUE DATA FOR ASTM TYPE A302-B STEEL DEVELOPED BY IN-REACTOR TESTS AT 500 F WITH THE RESULTS OF OUT-OF-REACTOR CONTROL TESTS(113)

The measurements of strain amplitude were performed at room temperature.

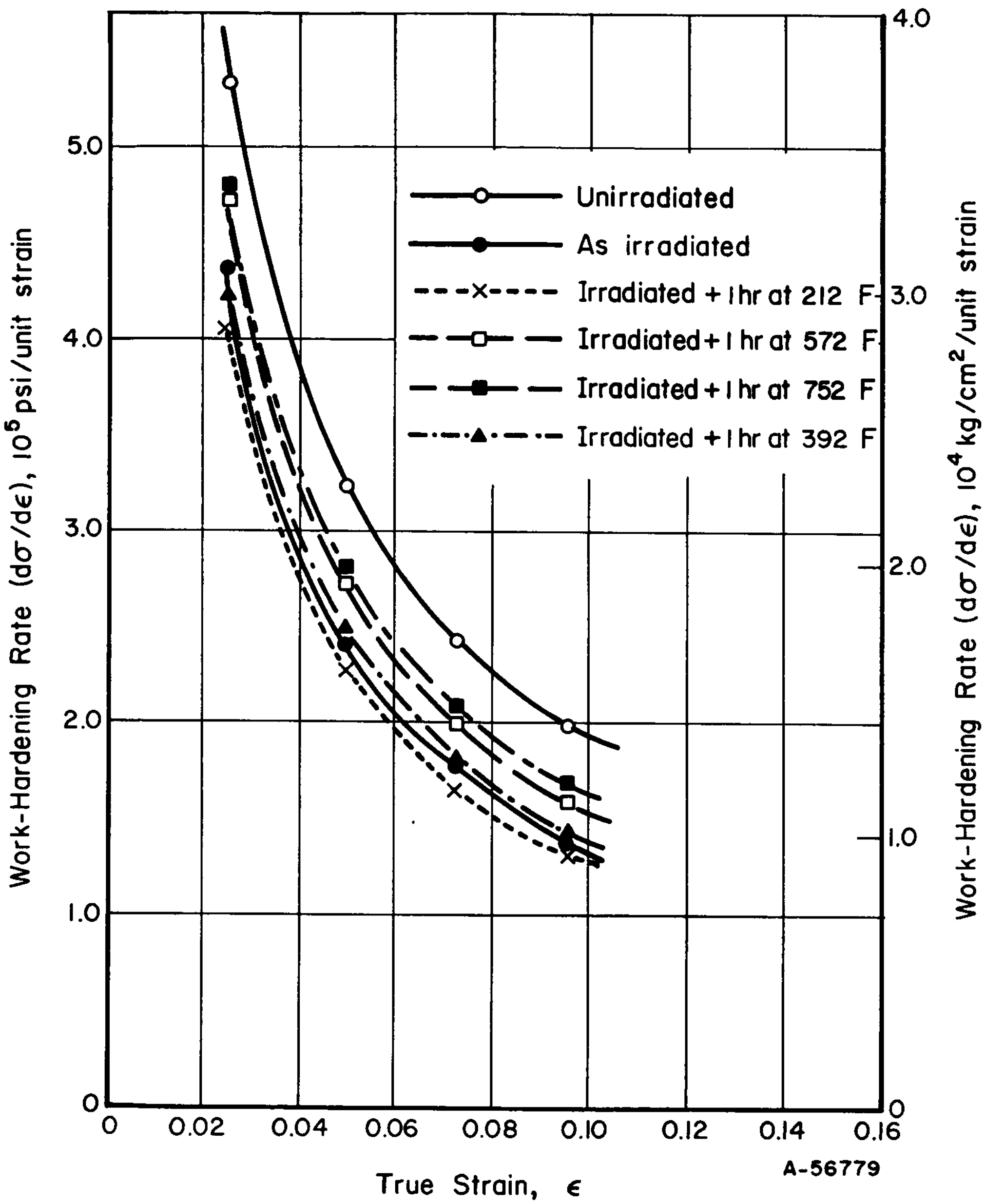


FIGURE 59. EFFECT OF IRRADIATION AND ANNEALING ON THE RATE OF WORK HARDENING OF THE ALUMINUM-GRAIN-SIZE-CONTROLLED MILD STEEL(118)

TABLE 20. CHEMICAL COMPOSITION OF FERRITIC AND MARTENSITIC STEELS

Material	Composition, weight percent												
	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 weight percent yttrium.

Effect of Irradiation on Tensile Properties

The effects of irradiation on the tensile properties of ferritic stainless steels are shown in Table 21. 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 60 illustrates the dependence of property changes in AM-350 and 17-4PH on the fast fluence.[10] These results indicate that only minor changes in tensile properties occur after fast fluences of 2 x 10^{21} n/cm^2.

A limited number of tensile tests at elevated temperatures have been performed on irradiated ferritic stainless steels. The steels which have been investigated are AISI Type 406, 17-4PH, and the General Electric alloy 1541. 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 61 and 62.[131] Figures 62 and 63 show the tensile properties of the unirradiated and irradiated material, respectively. It can be seen that a fast fluence of 1.3 x 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.

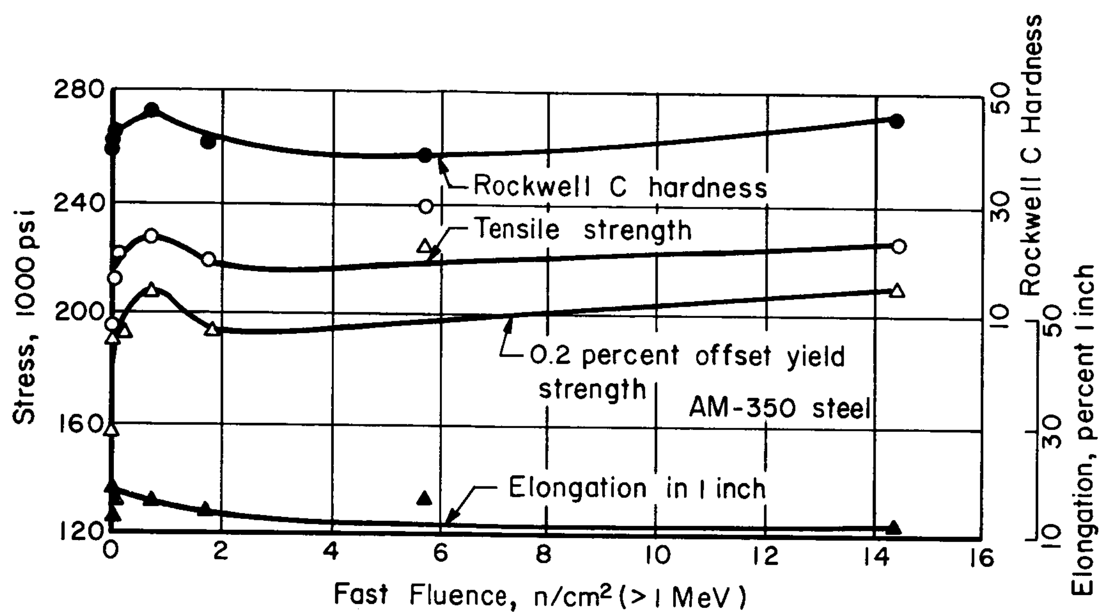
The elevated-temperature tensile properties of 17-4PH are given in Table 21[128], while those of Alloy 1541 are given in Tables 21[130] and 22[132]. These results indicate that irradiation does not significantly alter the tensile properties at elevated temperature. The effects of strain rate on Alloy 1541 are also minimal except that the strength is somewhat increased by higher strain rates for both the unirradiated and irradiated material. No significant strain-rate-dependent trends in ductility are evident for either the unirradiated or irradiated specimens.

The austenitic stainless steels exhibit an irradiation-induced ductility decrease at elevated temperatures, but the ductility of the ferritic steels at elevated temperatures is not affected by irradiation. This ductility decrease in austenitic stainless steels is attributed to helium bubble formation at the grain boundaries.[133] These helium bubbles

TABLE 21. 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			120
406 SS		4.0×10^{20}	~400	RT	62.9	100			24.8	16.5			121
406 SS		4.0×10^{20}	~400	595	29.3	33.9			54.4	32.2			121
406 SS		3.0×10^{19}	650	700	14.7	31	14	42	74	54			120
410 SS	TD-M	3.7×10^{19}	260-290	RT		58.7		72.7		18.0			120
410 SS	A	4.2×10^{19}	265	RT	38	56	59.5	77.5	32.5	27.7			122
410 SS	TD-A	4.5×10^{19}	260-290	RT	38.3	67.0	67.8	85	26.0	19.5			120
410 SS	A	4.9×10^{19}	265	RT	38	57	59.5	78	32.5	26.2			122
410 SS	A	5.6×10^{19}	265	RT	38	59	59.5	78	32.5	25.0			122
410 SS		6.0×10^{19}	265	RT	38	56.5	59.5	85.5	32.5	23.9			122
410 SS	A	7.2×10^{19}	265	RT	38	60	59.5	80.5	32.5	26.6			122
410 SS	A	8.3×10^{19}	265	RT	38	61	59.5	80	32.5	26.0			122
410 SS	TD-A	1.0×10^{20}	260-290	RT	38.9	75.2	66.6	89.8	25.7	20.7			122
410 SS	TD-A	1.0×10^{20}	260-290	RT	38.6	92.9	65.6	97.3	31.3	8.8			120
410 SS	TD-A	1.1×10^{20}	50	RT		87.8		87.8		2.8			8
410 SS	A	1.2×10^{20}	50	RT	38	89	59.5	92	32.5	15.3			122
410 SS	TD-A	2.4×10^{20}	260-290	RT	38.5	105.8	66.0	106.2	36.6	4.6			120
410 SS	TD-M	2.5×10^{20}	260-290	RT	37.6	108.2	67.6	108.2	31.5	4.0			120
410 SS	T	7.0×10^{19}	<100	RT	47	62	69	70					123
410 SS	T	7.0×10^{19}	<100	RT	44	61	70	75					123
410 SS	T	3.0×10^{20}	315-370	RT	70	86	92.3	102.8	32	21			124
410 SS	M	1.2×10^{19}	35	RT	147.7	175.5	177	225	20	17			125
410 SS	TD-M	3.7×10^{19}	260-290	RT		174.3		200.0		5.6			120
410 SS	TD-M	4.5×10^{19}	260-290	RT	140.7	176.3	174.2	223.4	6.8	8.0			120
410 SS	TD-M	1.0×10^{20}	260-290	RT	146.0	181.9	173.4	208.4	6.6	6.5			120
410 SS	TD-M	1.0×10^{20}	260-290	RT	133.9	194.0	168.9	216.7	8.9	4.9			120
410 SS	TD-A	1.2×10^{20}	260-290	RT	140.6	182.4	174.2	210.7	7.9	6.4			120
410 SS		1.5×10^{20}	35	RT	147.7	198	177	233	20	11			125
410 SS	TD-M	1.6×10^{20}	50	RT		166.2		192.0		4.3			8
410 SS	TD-M	2.4×10^{20}	260-290	RT	139.5	190.9	169.5	215.6	7.9	5.5			120
410 SS	TD-M	2.5×10^{20}	260-290	RT	138.1	203.6	127.4	222.6	9.2	4.6			120
414 SS	M	6.4×10^{19}	35	RT		206		240		15			125
420 SS	A	5.0×10^{19}	>100	RT	47	90	83	95	19	10			126
440 SS	H	7.0×10^{19}	>100	RT	185	205	199	240					123
440 SS	H	7.0×10^{19}	>100	RT	185	200	211	240					123
X-13		2.4×10^{20}	80	RT	45	98	68	98	36.5	1.2			127
17-4PH		0.2×10^{20}	50	RT	144.7	179.4	148.5	181	16	13	65	53	4
17-4PH		1.3×10^{20}	50	RT	144.7	194.5	148.5	197	16	12	65	44	4
17-4PH		5.1×10^{20}	50	RT	144.7	205.5	148.5	206	16	11	65	44	4
17-4PH		11.8×10^{20}	50	RT	144.7	209.5	148.5	210	16	10	65	41	4
17-4PH		28.0×10^{20}	50	RT	144.7	214.3	148.5	215	16	9	65	29	4
17-4PH		20×10^{20}	525	RT	162	148	148.5		10.2	12.6			128
17-4PH		20×10^{20}	525	400	158	113			6.3	9.9			128
17-4PH		20×10^{20}	525	500	110	89.5			8.2	8.0			128
17-4PH		20×10^{20}	525	600	70.5	51.3			18.8	18.1			128
17-4PH		20×10^{20}	525	700	23.1	33.5			33.4	16.2			128
17-4PH		20×10^{20}	525	800	15.9	15.2			44.0	37.8			128
17-4PH		20×10^{20}	525	900	13.8	10.3			16.3	18.3			128
1541		0.1×10^{20}	<120	RT	47.6	59.8	64.5	62.9	5.8	2.1			129
1541		3.5×10^{20}	700	600	23.0	12.0	27.0	16.0	36	37			130
1541		3.5×10^{20}	700	700	11.6	9.0	12.0	13.0	61	52			131

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



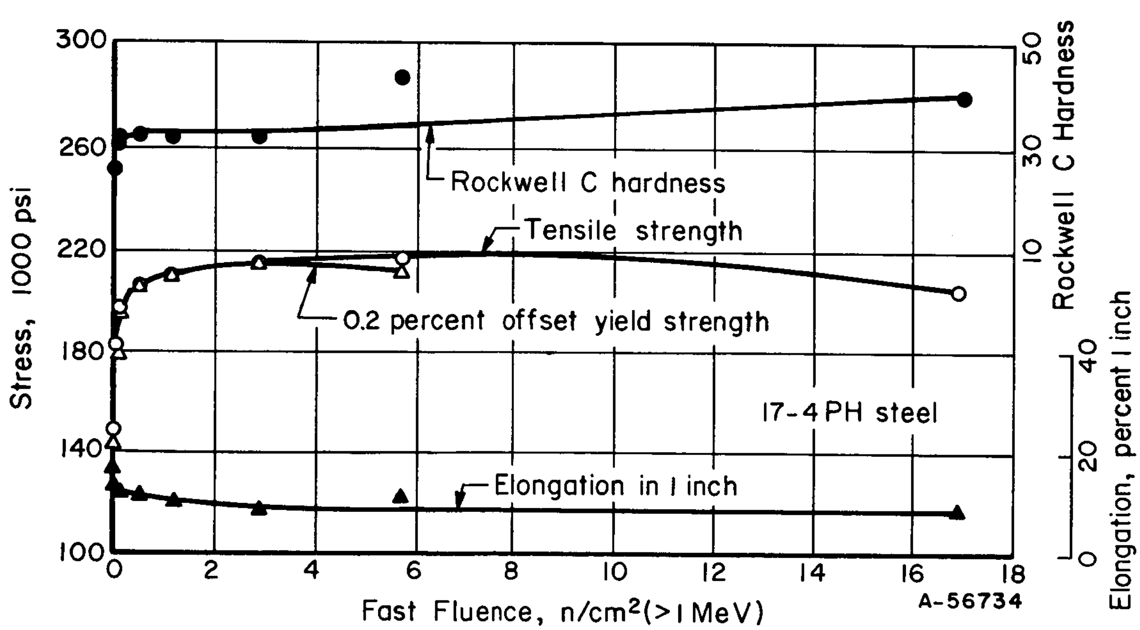


FIGURE 60. EFFECTS OF IRRADIATION ON THE ROOM-TEMPERATURE ROCKWELL HARDNESS, TENSILE STRENGTH, YIELD STRENGTH, AND ELONGATION OF TWO STAINLESS STEELS[10]

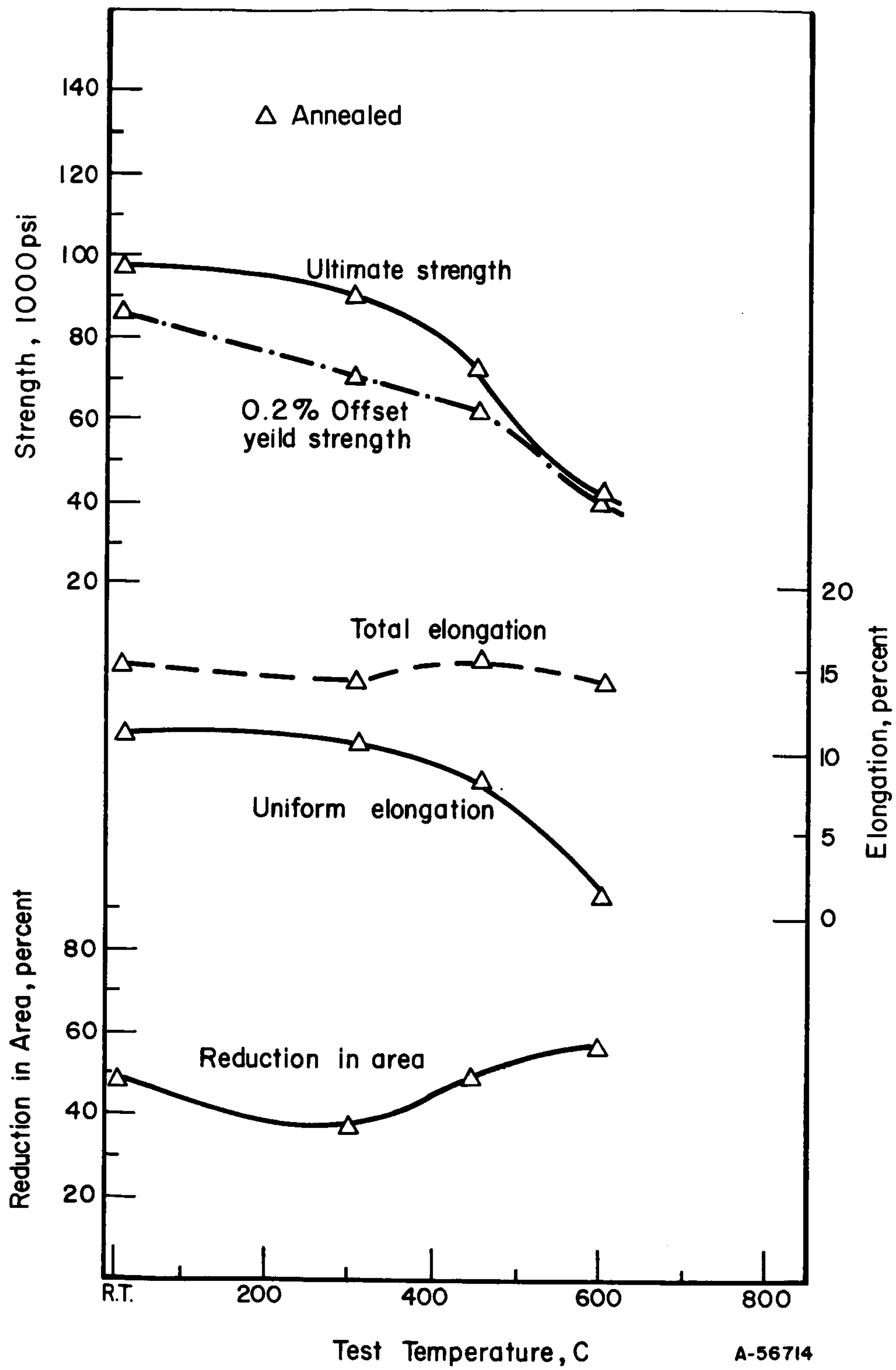


FIGURE 61. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF UNIRRADIATED AISI 406 STAINLESS STEEL(131)

Transverse specimens only.

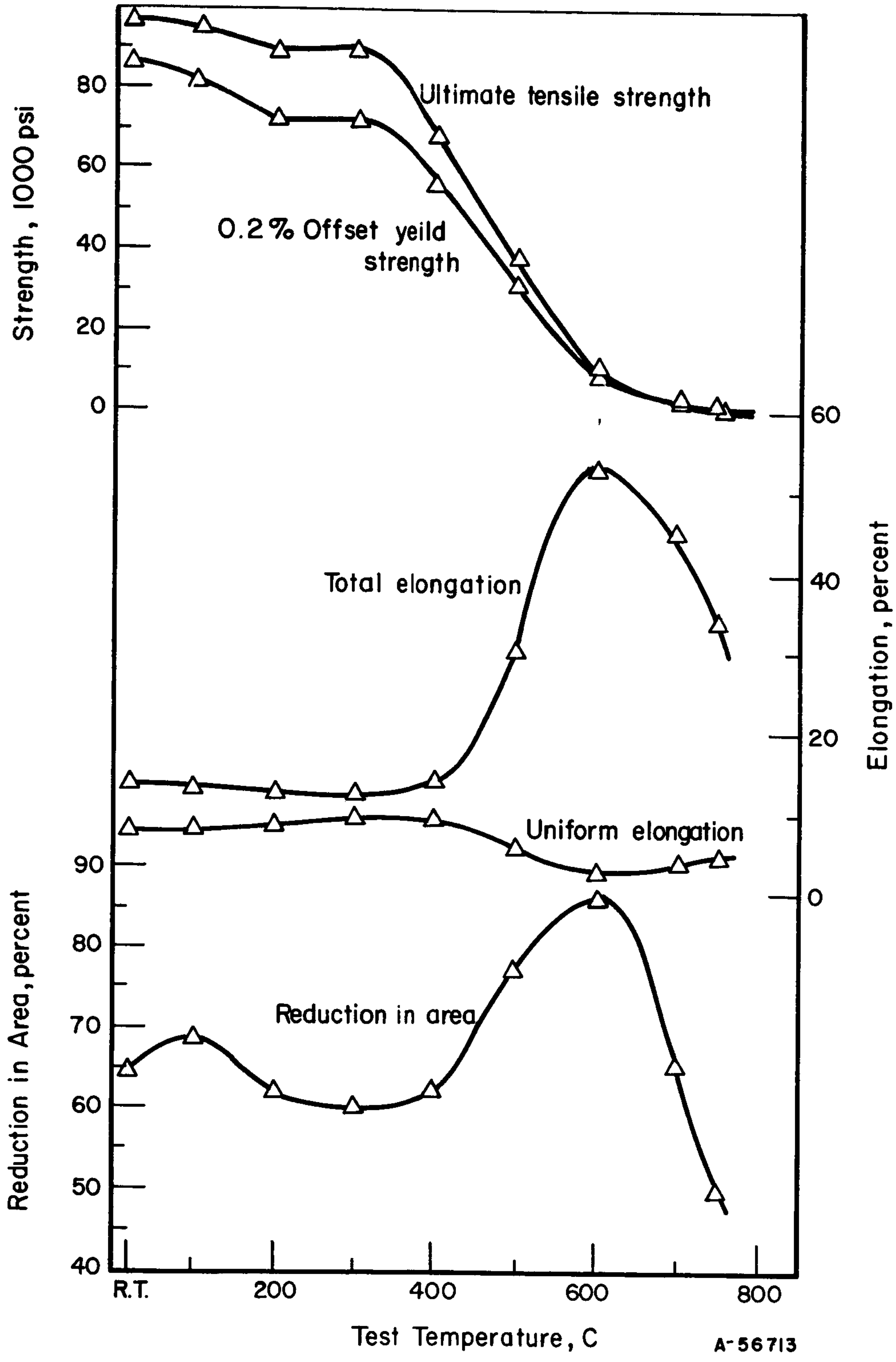


FIGURE 62. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF IRRADIATED AISI 406 STAINLESS STEEL(131)

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

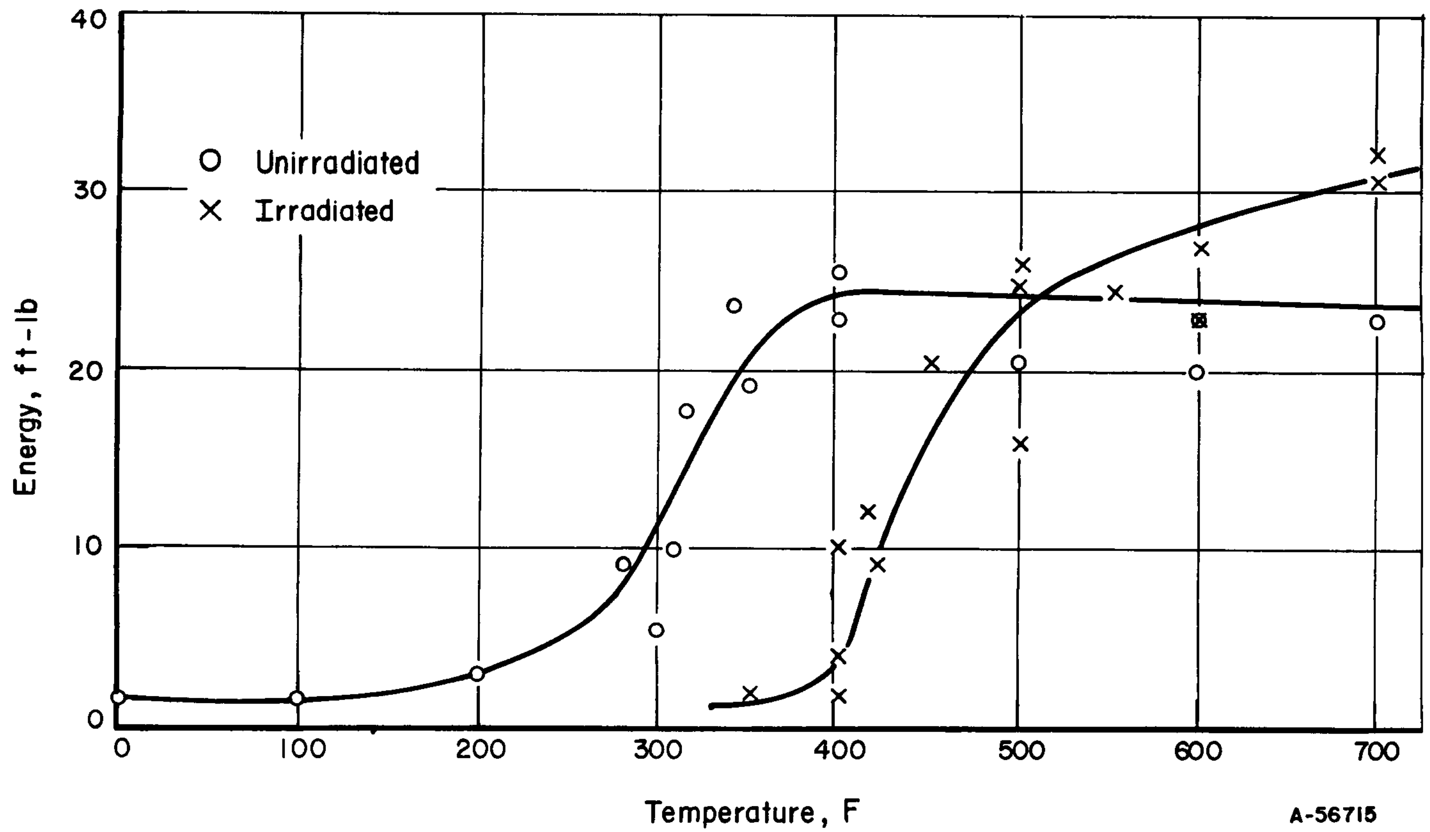


FIGURE 63. NOTCH-DUCTILITY PROPERTIES OF Fe-15Cr-4Al-1Y ALLOY ROD IRRADIATED AT < 250 F to 1 x 10^{19} N/CM^2 (> 1 MEV)(129)

TABLE 22. STRAIN-RATE SENSITIVITY OF IRRADIATED[a] AND UNIRRADIATED[b] Fe-15Cr-4Al-1Y ALLOY(132)

Deformation Temperature, C	Strain Rate, %/min	Strength, 1000 psi: 0.2% Offset Yield	Strength, 1000 psi: Engineering Ultimate	Ductility, percent: True Uniform Strain	Ductility, percent: 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.

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. The lack of a ductility minimum at elevated temperatures in irradiated ferritic steels may be due to low-boron contents or high solubility of helium in the body-centered cubic lattice, or it is possible that helium bubbles do not promote intergranular fracture in ferritic stainless steels at elevated temperatures. Annealing at 650 C for 1 hour results in complete recovery of pre-irradiation tensile properties at all temperatures of testing (up to 600 C) for AISI Type 406 stainless steel.(131) The properties that limit ferritic stainless steels for cladding application are low strength at elevated temperatures and low irradiation-induced room-temperature ductility, which will be discussed below.

Effect of Irradiation on the 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.(86) This NDT shift is in the band of values shown in Figure 32. A similar NDT shift, shown in Figure 63, occurs in General Electric Alloy 1541 after a fast fluence of 1×10^{19} n/cm^2(129), and thus renders these structural steels unattractive for reactor applications.

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.

Since there are a great many data available on the effects of irradiation on mechanical properties of stainless steels, the data are presented according to alloy. This results in some redundancy, since most stainless steels are affected by irradiation in a similar manner; but, if all data were lumped together, it would be difficult for the reader to evaluate the results for himself. With the recent interest in predominantly fast fluence irradiations, an effort is made to report these results separately for each alloy. The stainless steels covered in this report are:

(1) AISI Type 304
(2) AISI Type 304L
(3) AISI Type 347 and 348
(4) AISI Type 316
(5) AISI Type 318
(6) 20Cr-25Ni-Nb
(7) A-286
(8) Incoloy 800
(9) Russian stainless steels.

The compositions of these stainless steels are given in Table 23.

TABLE 23. COMPOSITION OF VARIOUS STAINLESS STEELS THAT HAVE BEEN IRRADIATED

Type	Composition, weight percent									
	Carbon[a]	Mn[a]	Si[a]	Cr	Ni	Mo	Nb+Ta	Fe	Al	Ti
AISI 304	0.08	2.00	1.00	18-20	8-12	--	--	Bal		
AISI 304L	0.03	2.00	1.00	18-20	8-12	--	--	Bal		
AISI 316	0.08	2.00	1.00	16-18	10-14	2-3	--	Bal		
AISI 318 (Fv548)	0.08	2.00	1.00	16-18	10-14	2-3	1.0			
AISI 347 & 348[b]	0.08	2.00	1.00	17-19	9-13	--	10 x C	Bal		
20Cr-25Ni-Nb	0.03	2.00	1.00	20	25	--	1.0	Bal		
A-286	0.08	2.00	1.00	13.5-16	24-27	1-1.75	--	Bal	0.35	1.9-2.3
Incoloy 800	0.10	1.50	1.00	19-23	30-35			Bal	0.3	.4

(a) Maximum.
(b) The only difference between AISI 347 and 348 is that the maximum allowable tantalum content in AISI 348 is 0.10 weight percent.

AISI Type 304

Mixed Thermal and Fast Fluence

Tensile Properties. A considerable amount of tensile testing has been performed on irradiated Type 304 stainless steel. However, all of these tests have been performed on rod-type specimens. Figure 64 illustrates the effect of irradiation on the room-temperature yield strength and ductility.(134) The results indicate that a saturation in irradiation effects is reached after a fast fluence of about 2 x 10^{20} n/cm^2 if the irradiation is performed at 50 C. However, no saturation in irradiation effects is reached after a fast fluence of 1.5 x 10^{21} n/cm^2 if the irradiation temperature is 290 C. The same trends, as far as saturation of irradiated effects, appear to take place for both the annealed and cold-worked (25 percent) material. The mechanical properties of Type 304 stainless steel at 300 C, irradiated at 290 C, are given in Table 24.

TABLE 24. MECHANICAL PROPERTIES AT 300 C OF TYPE 304 STAINLESS STEEL IRRADIATED AT 290 C(134)

Fast Fluence, n/cm^2	Condition	Tensile Strength, 1000 psi		Yield Strength, 1000 psi		Elongation, percent		Reduction in Area, percent	
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
4.1 x 10^{19}	Annealed	22.5	47.8	66.8	74.6	43.6	26.6	55.5	45.5
1.5 x 10^{21}	Annealed	22.5	109.5	66.8	109.5	43.6	3.3	55.5	35.5
4.1 x 10^{19}	25% CW	94.7	120.1	108.0	128.9	4.8	3.1	34.9	27.9
1.5 x 10^{21}	25% CW	94.7	147.2	108.0	151.1	4.8	2.5	34.9	22.9

The elevated-temperature mechanical properties of irradiated Type 304 stainless steel are illustrated in Figures 65 and 66.(131) It can be seen that at above 500 C there is little difference in the ultimate and yield strength of irradiated and unirradiated cold-worked materials. The same trend is apparent for the ultimate strength of the annealed material above 500 C, while the yield strength of the irradiated annealed material remains above that of the unirradiated material up to temperatures of 750 C. The ductility of the irradiated material is decreased by both the testing temperature and the increasing fast fluence. Two ductility minimums appear to be present, with one being at about 350 C and the other at 700 C. The increase in ductility between these two minimums is attributed to annealing out of displacement-type damage at temperatures above 350 C.

The effect of irradiation temperature on the room-temperature tensile properties of Type 304 stainless steel is illustrated in Figure 67.(135) It can be seen that the maximum irradiation hardening results from an irradiation temperature of 150 C, while at irradiation temperatures above 300 C, the hardening effect is considerably less. However, the minimum ductility results from an irradiation temperature of 300 C. The irradiation-induced property changes are attributed to the presence of defect clusters as observed by transmission electron microscopy. These clusters were found to be 100 to 200 A in size when the maximum irradiation effects on yield strength were detected.

At elevated temperatures, the irradiation effects on mechanical properties are characterized by minor changes in strength and severe reductions in ductility. The reduction in ductility at elevated temperatures has been found to be independent of

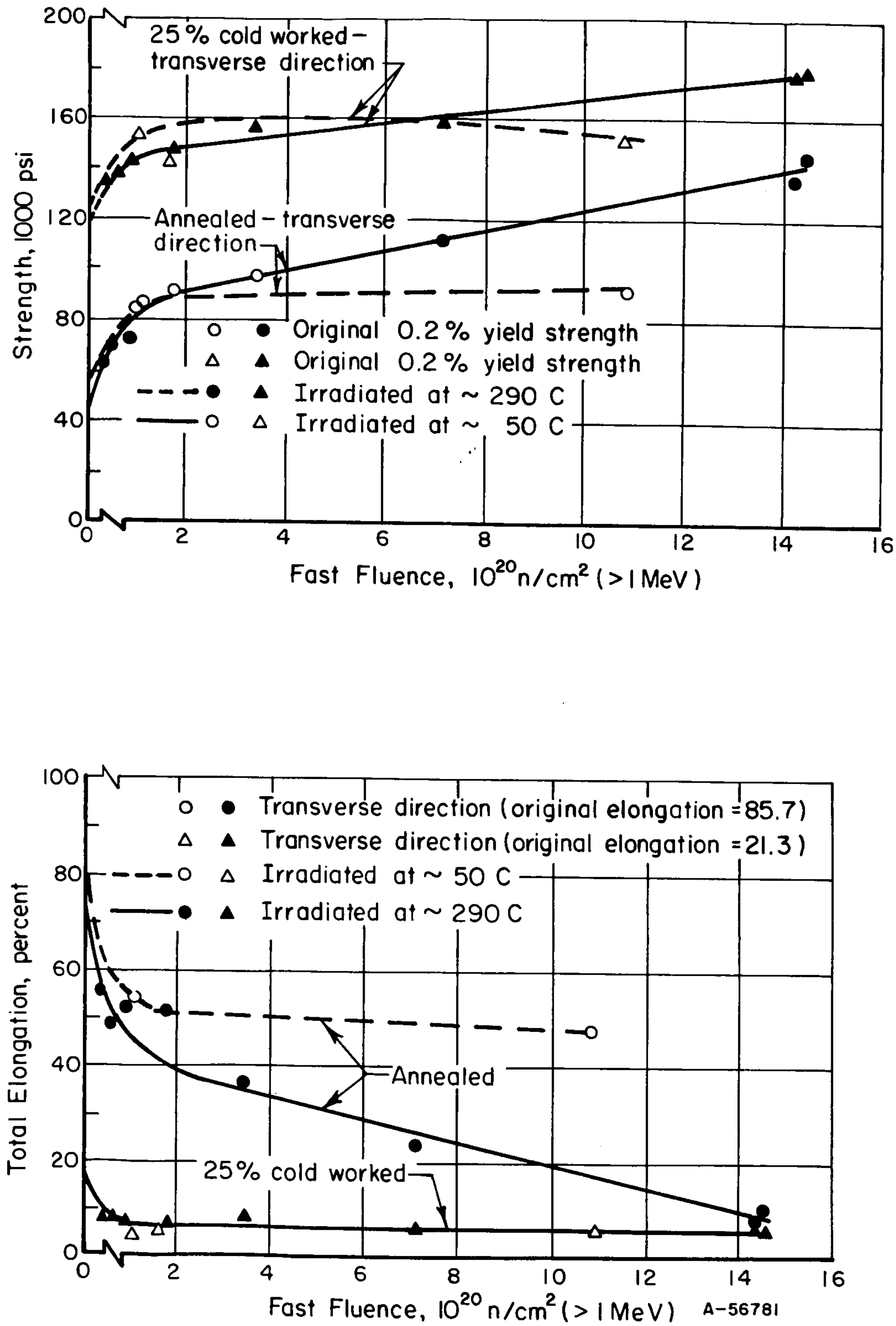


FIGURE 64. EFFECTS OF IRRADIATION ON ROOM-TEMPERATURE YIELD STRENGTH AND DUCTILITY OF TYPE 304 STAINLESS STEEL(134)

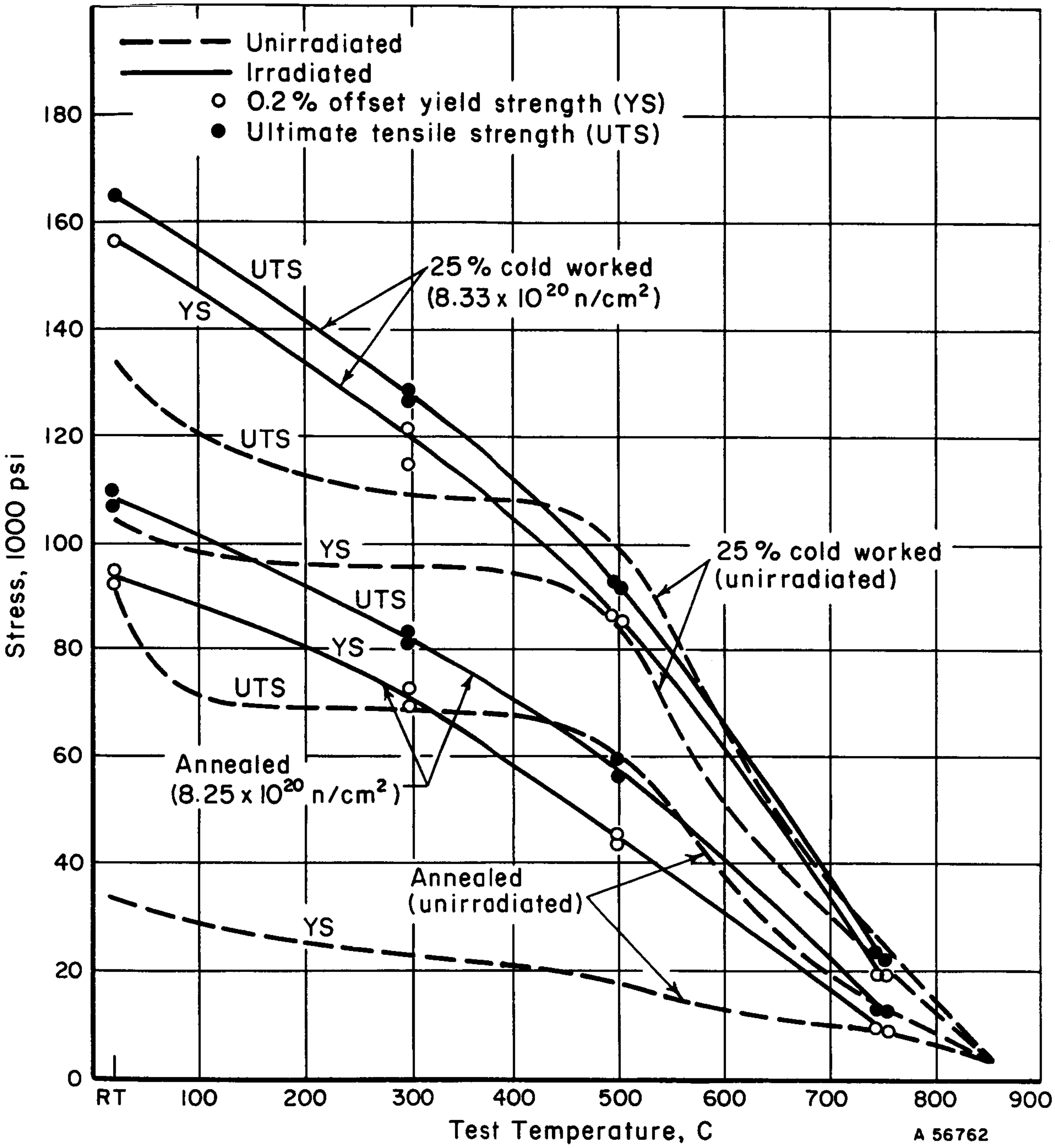


FIGURE 65. EFFECT OF TEST TEMPERATURE ON THE STRENGTH OF IRRADIATED ANNEALED AND 25 PERCENT COLD-WORKED TYPE 304 STAINLESS STEEL(131)

Irradiation temperature 60 C; fast fluence ~1.0 x 10^{21} n/cm^2; all transverse specimens.

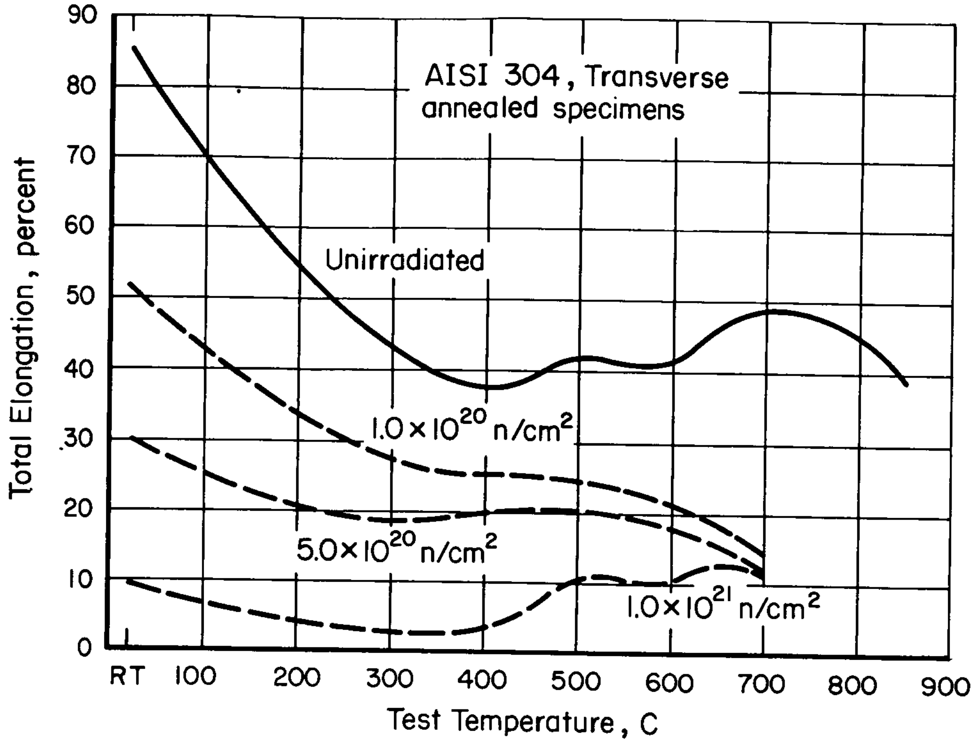


FIGURE 66. EFFECT OF TEST TEMPERATURE ON THE DUCTILITY OF TYPE 304 STAINLESS STEEL IRRADIATED AT 290 C[131]

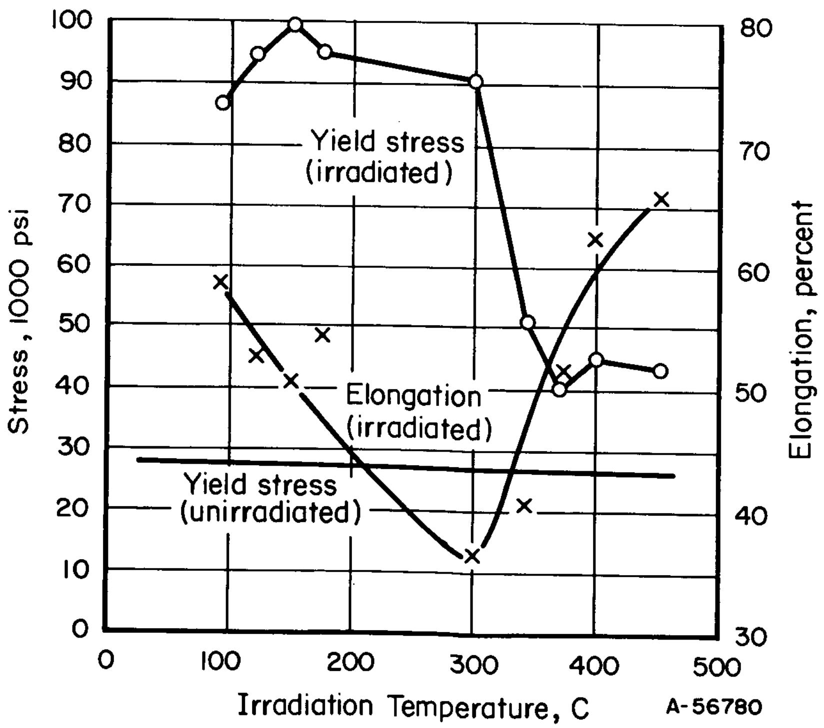


FIGURE 67. ROOM-TEMPERATURE TENSILE PROPERTIES OF IRRADIATED TYPE 304 STAINLESS STEEL AS A FUNCTION OF IRRADIATION TEMPERATURE[135]

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

irradiation temperature if the irradiation temperature is below 600 to 700 C.(131, 136) However, at irradiation temperatures above 700 C, somewhat more embrittlement takes place, as indicated in Figure 68.(137)

Other variables that have been found to have a significant effect on the irradiation-induced embrittlement of AISI Type 304 stainless steel at elevated temperatures are:

(1) Strain Rate. Decreasing the strain rate has been found to significantly decrease the elevated-temperature ductility.(136)

(2) Grain Size. Material with large grain size has been found to be more susceptible to the irradiation-induced embrittlement.(138)

(3) Alloying Additions. Additions of titanium to Type 304 stainless steel increase its resistance to irradiation-induced embrittlement at elevated temperature.(139) This is illustrated in Figure 69. The maximum improvement is obtained with a titanium addition of about 0.20 weight percent, which is somewhat less than the 0.40 weight percent titanium content of the titanium-stabilized Type 321 stainless steel. The improved resistance may be due to decreased grain size which accompanies the titanium additions.

(4) 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 25.(140) These test results indicate that the weld metal and base metal are embrittled to about the same degree by irradiation. Table 26 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.

(5) Brazes. Table 27 shows the effect of irradiation on the shear strength of nickel-13 wt % chromium-10 wt % phosphorus braze joints of Type 304 stainless steel.(140) 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.

Creep Properties. In-pile creep tests on Type 304 stainless steel have been performed at Battelle-Northwest(141) and Oak Ridge(142); postirradiation stress-rupture tests have also been performed at Oak Ridge.(142, 143) The results of the Battelle tests, shown in Table 28(141), indicated that while the actual creep rate was not affected by irradiation, the elongation at failure was significantly reduced. With the large reduction in the elongation at failure, the time to rupture was significantly reduced by irradiation.

The results of the creep and stress-rupture tests on irradiated Type 304 stainless steel obtained at Oak Ridge are shown in Tables 29(142), 30(142), and 31(143). Table 29 shows that the creep rates for Type 304 stainless steel in both the in-pile and postirradiation creep tests are significantly higher than those for the unirradiated material. It

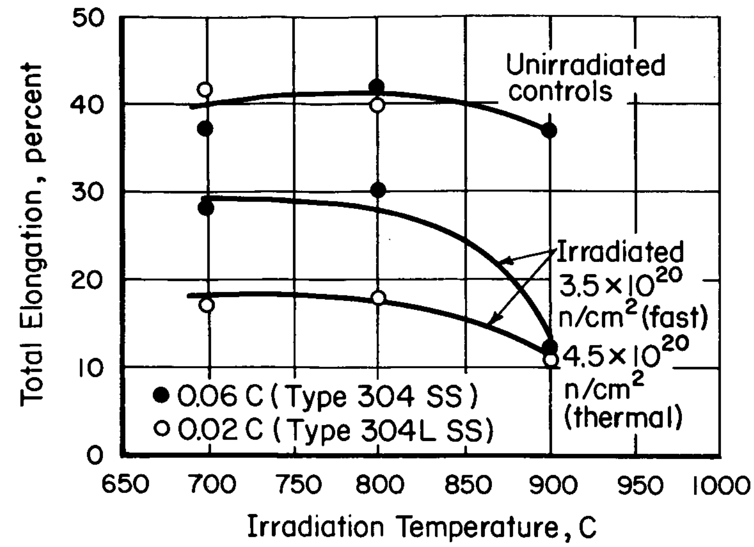


FIGURE 68. EFFECT OF IRRADIATION TEMPERATURE ON DUCTILITY OF TYPE 304 AND 304L STAINLESS STEELS WHEN TESTED AT 650 C[137]

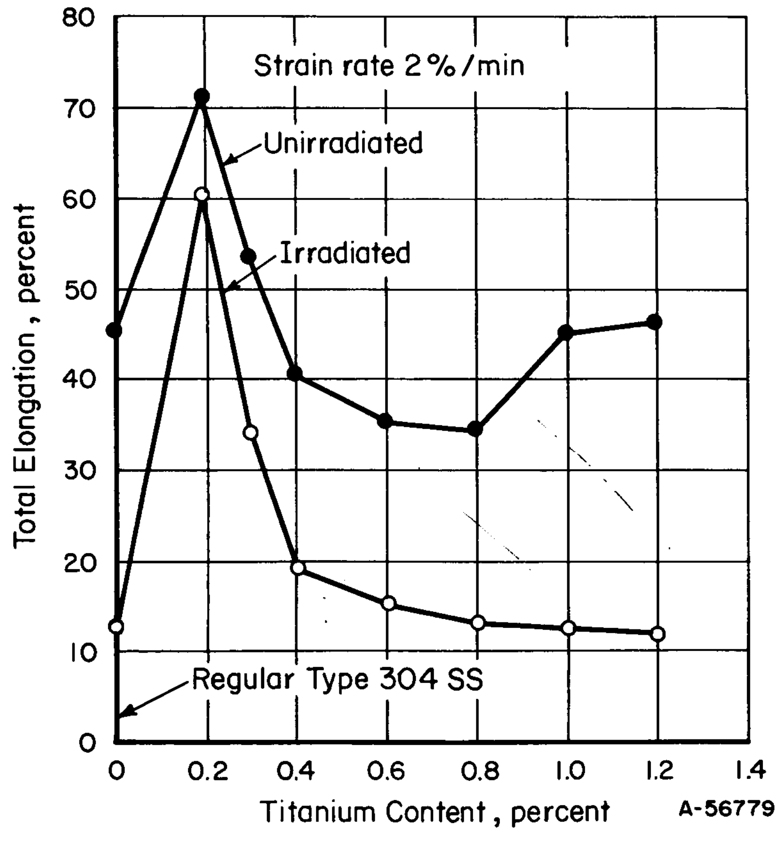


FIGURE 69. DUCTILITY AT 842 C OF IRRADIATED AUSTENITIC STAINLESS STEEL AS A FUNCTION OF TITANIUM CONTENT[139]

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

TABLE 25. COMPARISON OF STRENGTH AND DUCTILITY FOR IRRADIATED AND UNIRRADIATED TYPE 304 STAINLESS STEEL WELDS AND WROUGHT METAL[a][140]

Material	Deformation Temp, C	Strength, 1000 psi: 0.2 Percent Offset Yield, Irradiated	Strength, 1000 psi: 0.2 Percent Offset Yield, Unirradiated	Strength, 1000 psi: Ultimate Tensile, Irradiated	Strength, 1000 psi: Ultimate Tensile, Unirradiated	Ductility, percent: True Uniform Strain, Irradiated	Ductility, percent: True Uniform Strain, Unirradiated	Ductility, percent: Total Elongation, Irradiated	Ductility, percent: Total Elongation, 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.

TABLE 26. RATIO OF IRRADIATED TO UNIRRADIATED DUCTILITY OF WELDS AND WROUGHT ALLOYS[140]

Base Material	Ratio of Irradiated to Unirradiated Toughness at Indicated Temperature: 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

TABLE 27. LOAD-CARRYING CAPACITY OF IRRADIATED AND UNIRRADIATED (Ni- Cr- P) BRAZES WITH TYPE 304 STAINLESS STEEL[140]

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.

TABLE 28. IN-REACTOR CREEP OF ANNEALED TYPE 304 STAINLESS STEEL[141]

Temp, C	Stress, 1000 psi	Percent Creep Strain at Failure		Failure Time, hours	
		In-Reactor	Ex-Reactor	In-Reactor	Ex-Reactor
650	20	8-9	25-26	150	570
550	30	3-4	8	770	4500

TABLE 29. MINIMUM CREEP RATE OF TYPE-304 STAINLESS STEEL AT 650 C[a][142]

Stress, 1000 psi	Minimum Creep Rate[b], percent/hr		
	Unirradiated	In-Reactor	Postirradiation
35	2.26 (7.7)		
30	0.92 (9.0)		0.50 (12.0)
27	0.33 (44.1)		
25	0.048 (122.8)	0.138 (17.1)	0.139 (24.4)
22.25	0.042 (157.7)		0.014 (123.4)
20	0.0073 (366.6)	0.016 (98.0)	
17		0.0035 (423.0)	0.006 (314.4)
15	0.00032	0.0014	0.0018 (956.9)

(a) The instantaneous fluence for in-pile specimens was 6 x 10^{13} n/cm^2 thermal and 5 x 10^2 n/cm^2 fast (>2.9 MeV). The fluence of postirradiation specimens was 2 x 10^{20} n/cm^2 thermal and 2.9 x 10^{19} n/cm^2 fast.

(b) Numbers in parentheses are rupture life in hours.

TABLE 30. STRESS-RUPTURE DUCTILITY OF TYPE-304 STAINLESS STEEL AT 650 C[a][142]

Stress, 1000 psi	Elongation, percent			Reduction in Area, percent		
	Unirradiated	In-Reactor	Postirradiation	Unirradiated	In-Reactor	Postirradiation
35	37.2			35.1		
30	17.2		8.4	25.7		9.1
27	26.6			28.0		
25	17.2	6.7	4.9	22.3	11.1	4.2
22.25	18.4		2.9	19.9		4.0
20	17.5	2.6		13.9	3.0	
17			3.7			
15			2.8			0.2

(a) The instantaneous fluence for in-pile specimens was 6 x 10^{13} n/cm^2 thermal and 5 x 10^{12} n/cm^2 fast (>2.9 MeV). The fluence of postirradiation specimens was 2 x 10^{20} n/cm^2 thermal and 2.9 x 10^{19} n/cm^2 fast.

TABLE 31. STRESS RUPTURE AND DUCTILITY OF IRRADIATED[a] TYPE 304 STAINLESS STEEL AT 650 C PER PRETEST HEAT TREATMENT[143]

	Time to Rupture, hours			Total Elongation at Fracture, percent		
Stress, 1000 psi	Fine Grain	Coarse Grain	Aged, Coarse Grain	Fine Grain	Coarse Grain	Aged, Coarse Grain
30	11.3	1.5	14.8	44.0	11.4	24.2
25	79.0	5.5	50.8	29.5	9.0	25.1
20	191.0	109.5	664.1	17.2	3.8	14.3
15	514.4	194.4	3638.0	9.2	3.5	7.8

(a) Irradiated at 50 C to a fast fluence of 9×10^{20} n/cm^2 (> 1 MeV) and a thermal fluence of 7×10^{20} n/cm^2.

is believed that the apparently higher creep rates for the irradiated material are due to failure of specimens while they are still in the primary creep stage.[142] Lower values were obtained for elongation at failure in postirradiation and in-pile tests than in preirradiation tests. The limited creep tests at Oak Ridge suggest that the in-pile creep behavior of Type 304 stainless steel is similar to postirradiation creep behavior. The microstructure has been shown to have an effect on the stress-rupture life of irradiated Type 304 stainless steel. Table 31 illustrates the importance of grain size and the shape of the carbide particles on rupture life.[143] Considerably improved elongation at fracture can be obtained by fine grain size, but longer lives can be obtained if the material is aged sufficiently long to have spherical carbide particles at the grain boundaries.

Fatigue Properties. The cyclic-strain fatigue behavior of AISI Type 304 stainless steel has been determined in-pile at 649 C.[144] By applying gas pressure, the thin-walled tubular specimens were alternately expanded and contracted between rigid concentric mandrels. The results of the tests (Figure 70) indicate that irradiation tends to reduce fatigue life.

Hardness Properties. The effect of irradiation on the hardness of annealed and cold-worked Type 304 stainless steel is illustrated in Figure 71.[131]

Other Properties. It was found that irradiation did not cause any detectable increase in ferrite content, as measured by magnetic susceptibility.[134]

Predominantly Fast Fluence

Tensile Properties. The effect of irradiation in a predominantly fast fluence on the tensile properties of Type 304 stainless steel is illustrated in Figure 72.[145] The irradiation temperature of 540 C appears to have been high enough to eliminate the displacement-type damage, since the postirradiation yield and ultimate strength are significantly different from each other. Also, the postirradiation room-temperature elongation is similar to that expected for the unirradiated material. However, at elevated temperatures, the ductility of the irradiated material is significantly reduced. This can be attributed to the irradiation-induced embrittlement at elevated temperatures. The recovery of the irradiation-induced ductility at elevated temperatures has not been previously observed; however, this is also the highest testing temperature for highly irradiated stainless steel.

AISI Type 304L

Mixed Thermal and Fast Fluence

Tensile Properties. Only a few tensile tests have been performed on irradiated Type 304L stainless steel. The results of these tests[93] (shown in Figure 73) indicate

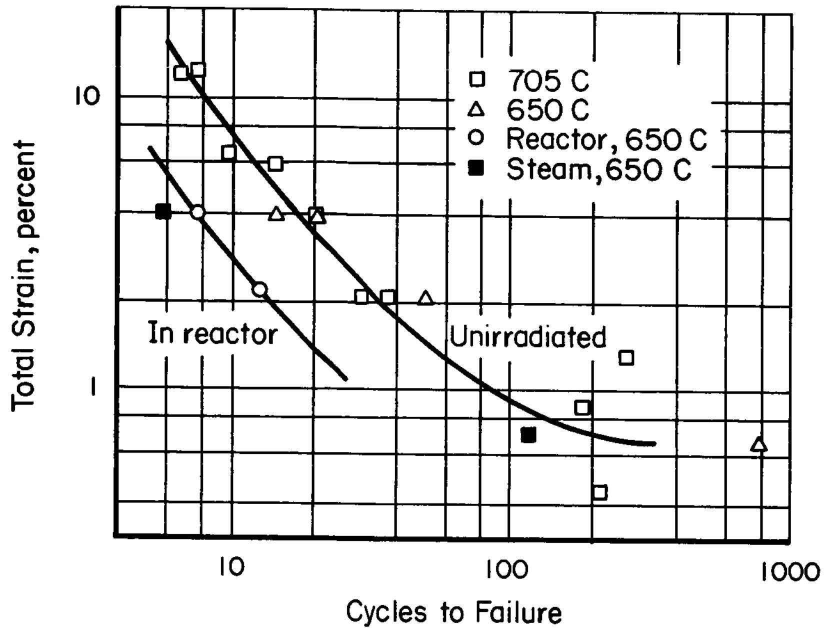


FIGURE 70. STRAIN FATIGUE LIFE OF TYPE 304 STAINLESS STEEL TUBING IN REACTOR AND OUT OF REACTOR(144)

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

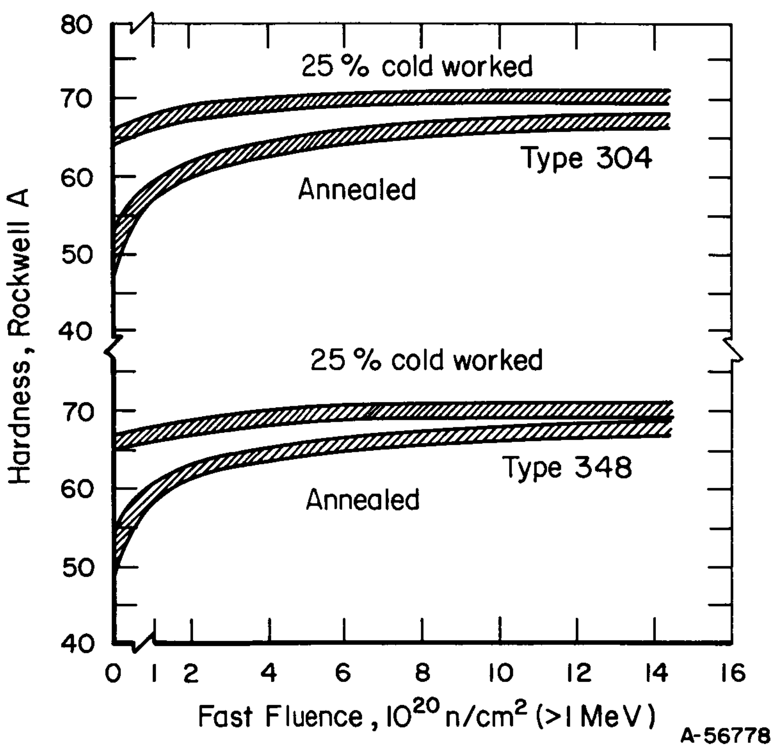


FIGURE 71. EFFECTS OF IRRADIATION ON THE HARDNESS OF TYPES 304 AND 348 STAINLESS STEEL(131)

Irradiation temperature ~290 C.

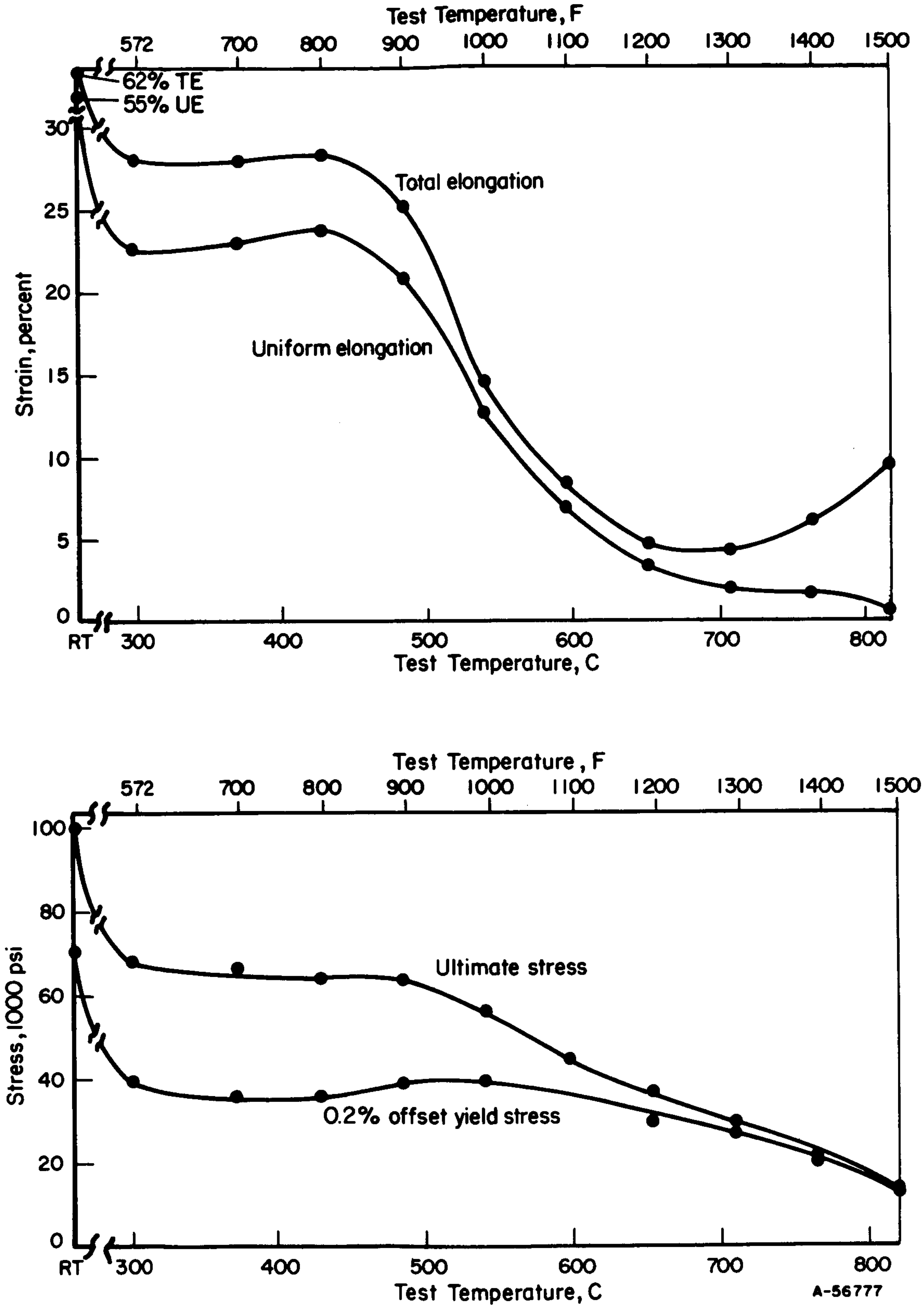


FIGURE 72. EFFECT OF TEST TEMPERATURE ON STRENGTH AND DUCTILITY OF ANNEALED TYPE 304 STAINLESS STEEL IRRADIATED IN EBR-2(145)

Irradiated at 540 C to a fast fluence of 2.0×10^{22} n/cm^2.

that Type 304L stainless steel, which is low in carbon, is somewhat more susceptible to irradiation-induced embrittlement than the Type 304 stainless steel which has a higher carbon content. The high carbon content supposedly results in more carbide particles near the grain boundary, which thereby impede the movement of helium to grain boundaries.

Impact Properties. The effect of irradiation on the impact properties of Type 304L stainless steel is illustrated in Figure 73.(93)

Predominantly Fast Fluence

Burst Tests. Extensive tube burst tests have been performed on Type 304L stainless steel tubes that had been claddings for EBR-II fuel specimens. The tangential rupture strength of these irradiated tubes is shown in Figure 74(146); the elongation (as measured by increase in diameter) is given in Figure 75.(146) The tangential rupture strength is higher for material which has received a fast fluence of 0.8×10^{22} n/cm^2 until a testing temperature of 700 C; above that temperature the unirradiated material is stronger. Irradiated tubes which had received a higher fast fluence (1.6×10^{22} n/cm^2) were stronger than the unirradiated tubes up to 800 C. The irradiated tubes exhibited their lowest ductility in the 400 to 500 C range; after this, the ductility increased somewhat, but it was still lower than that of the unirradiated tubes. Annealing at 500 C for up to 100 hours did not significantly affect the ductility of the irradiated tubes when burst tested at 500 and 900 C.(147) However, annealing of the irradiated tubes at 900 C for 1 hour resulted in reduction of the tangential rupture strength and in improvement in elongation, provided that the test temperature was less than 750 C. Figure 76 indicates that no improvement in ductility resulted from a 1-hour anneal at 900 C if tested at temperatures above 750 C.(147, 148)

AISI Types 347 and 348

Mixed Thermal and Fast Fluence

These two stainless steels, Type 347 and 348, will be treated together because the chemical composition and mechanical properties of the two are practically identical. In quite a few cases, the experiments refer to the stainless steel in question as Type 347, while actually it meets the specifications of both Type 347 and Type 348. The only difference in specifications is that the tantalum content must be less than 0.1 weight percent in Type 348, while it could be as high as 10 times the carbon content in Type 347. Usually, the tantalum content in Type 347 is low enough to qualify it for the Type 348 designation.

Tensile Properties. The effect of irradiation on the room-temperature mechanical properties of Type 347 stainless steel is illustrated in Figures 77 and 78.(149) The irradiation causes an increase in both the yield strength and the ultimate strength, but the increase in yield strength is considerably more and it approaches the ultimate strength in value. Irradiation at a low temperature such as 50 C has been found to result in

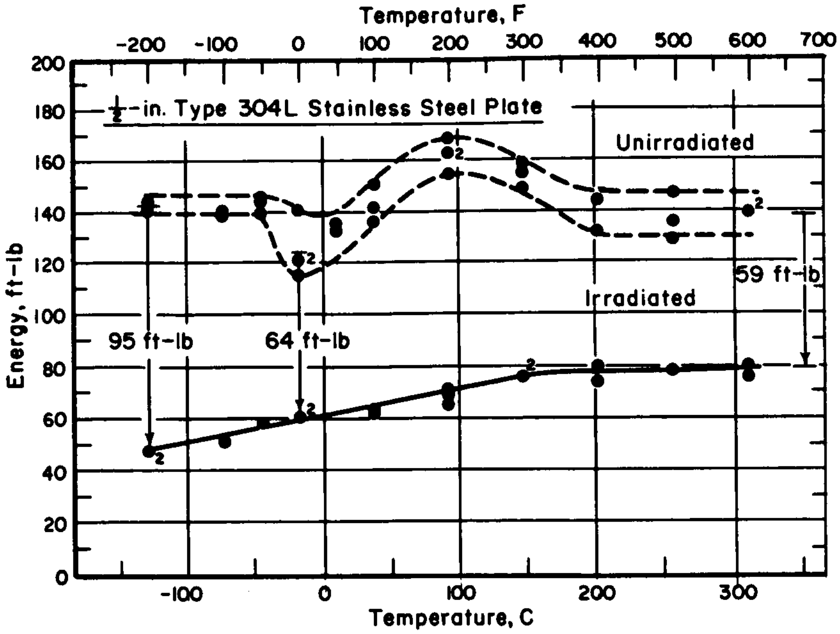


FIGURE 73. CHARPY-V NOTCH DUCTILITY CHARACTERISTICS OF TYPE 304L STAINLESS STEEL PLATE BEFORE AND AFTER IRRADIATION AT <120 C TO A FAST FLUENCE OF 1.1 X 10^{20} n/cm^2 [93]

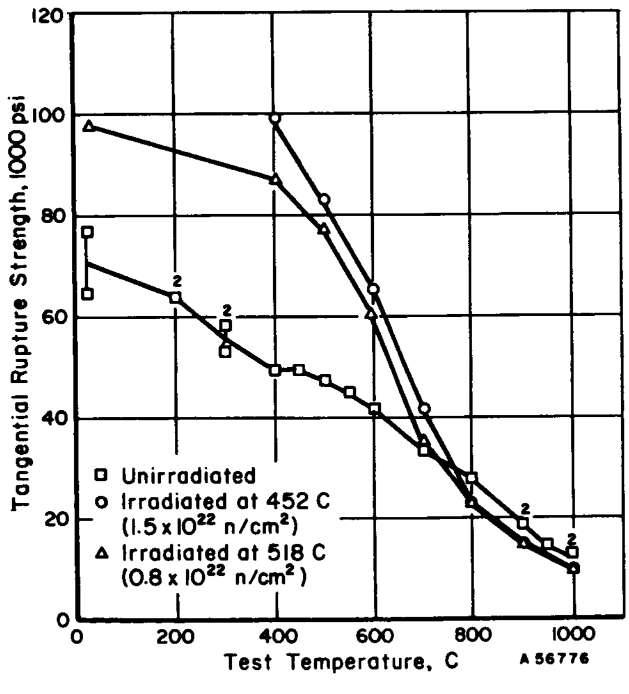


FIGURE 74. EFFECT OF TEST TEMPERATURE ON RUPTURE STRENGTH OF IRRADIATED TYPE 304L STAINLESS STEEL JACKETING FOR EBR-I FUEL[146]

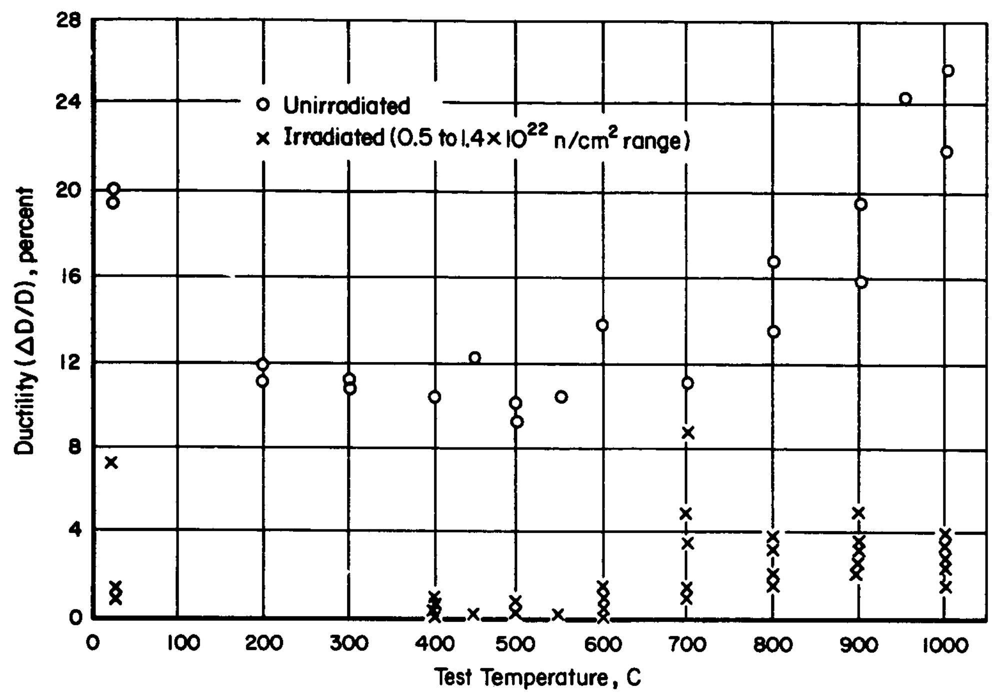


FIGURE 75. EFFECT OF TEST TEMPERATURE ON THE DUCTILITY (ΔD/D%) OF IRRADIATED TYPE 304L STAINLESS STEEL JACKETING FOR EBR-II FUEL(146)

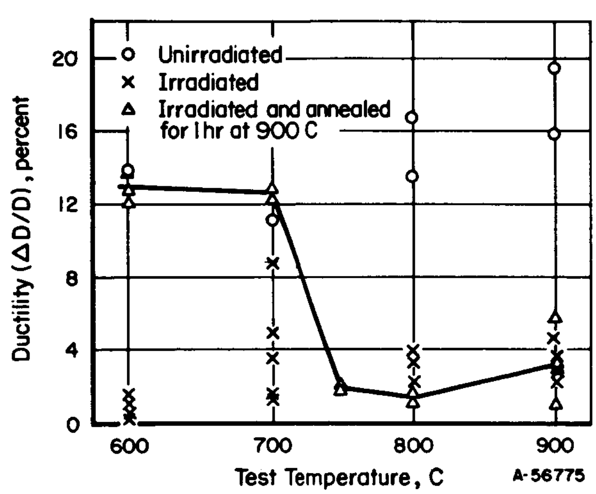


FIGURE 76. EFFECT OF TEST TEMPERATURE ON DUCTILITY FOR EBR-II TYPE 304L FUEL CLADDING(148)

Notice the restoration of preirradiation ductility by annealing at 900 C up to testing temperatures of 700 C.

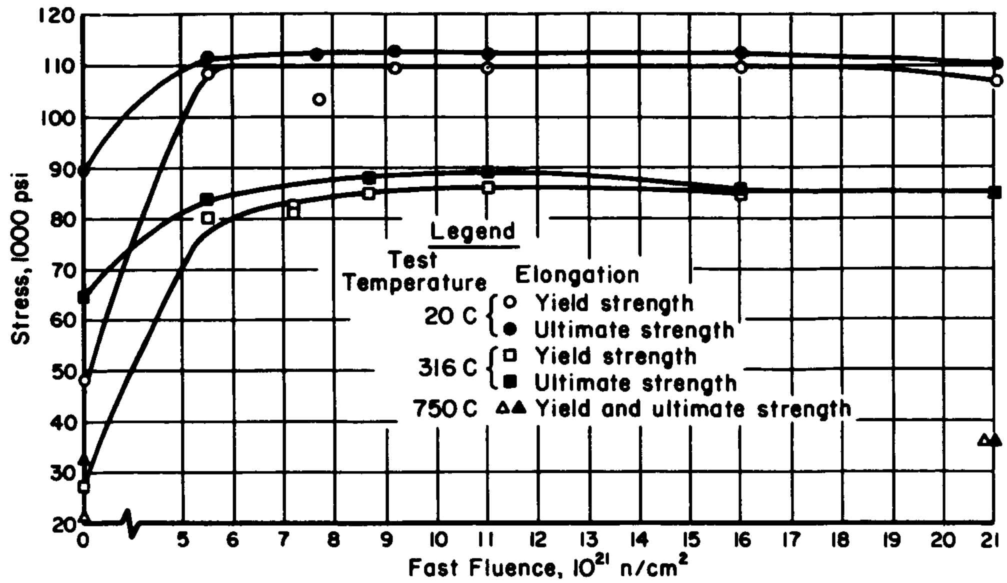


FIGURE 77. CHANGES IN YIELD AND ULTIMATE STRENGTHS OF TYPE 347 STAINLESS STEEL AT 20, 316, AND 750 C AS A FUNCTION OF NEUTRON IRRADIATION(149)

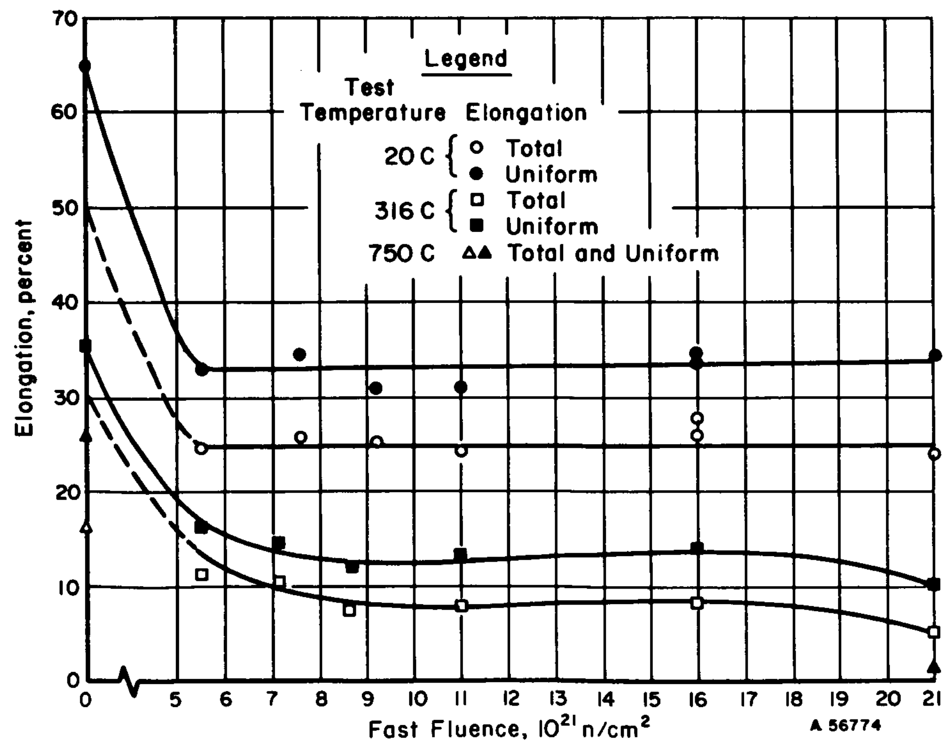


FIGURE 78. EFFECT OF IRRADIATION ON THE UNIFORM AND TOTAL ELONGATIONS OF TYPE 347 STAINLESS STEEL AT 20, 316, AND 750 C(149)

saturation of irradiation-induced changes in mechanical properties after a fast fluence of 2×10^{20} n/cm^2.(134) However, irradiation in the 300 C temperature range has been found to result in no saturation of irradiation effects even after fast fluences of 1.4×10^{21} n/cm^2.(134) These irradiations in the 280 to 400 C range have been found to result in extremely low ductilities when the material is tested either at room temperature or in the 280-400 C range. These low ductilities are illustrated in Table 32.(150) However, annealing of the irradiated material at 538 C for 1 hour resulted in considerable restoration of the preirradiation mechanical properties, while annealing at 982 C for 1 hour resulted in complete recovery of the preirradiation mechanical properties except for some ductility. The effect of testing temperature on the strength and ductility of the irradiated material is illustrated in Figure 79.(131) The yield and ultimate strengths of the cold-worked irradiated material are identical to those of the unirradiated material at temperatures above 500 C. For the annealed material, the ultimate strength of the unirradiated and irradiated materials becomes the same above 500 C, while the yield strength of the irradiated material is somewhat higher until a testing temperature of about 750 C. The elevated-temperature ductility of Types 347 and 348 stainless steel is considerably reduced by irradiation. The effect of testing temperature on the postirradiation ductility is illustrated in Figure 80.(131) It can be seen that some recovery in the ductility takes place above 400 C; this ductility restoration is attributed to annealing out of the displacement-type irradiation damage. The postirradiation ductility decreases with increasing fluence at all temperatures. After irradiation to a fast fluence of 2.1×10^{22} n/cm^2, the total elongation at 750 C has been reduced to only 0.5 percent.(149) The reduction of ductility with increasing temperature takes place very rapidly as shown in Figure 81. The uniform elongation is reduced from 16 percent at 525 C to 1.8 percent at 575 C.(151) This loss in uniform elongation is recovered by a one-hour anneal at 980 C if the testing temperature is 500 C or below. However, at testing temperatures of 600 C or above, the ductility cannot be recovered by annealing even at 1350 C.

Results of tensile tests on tubes irradiated at temperatures up to 343 C are given in Table 33.(152) These tubes were 0.34 inch in diameter and had a wall thickness of 20 mils. Comparison of the test results with results for plate specimens irradiated under similar conditions (Table 32) shows that the tubes undergo considerably more change in mechanical properties, especially with regard to reduction in ductility. As for the plate specimens, the most severe reduction in ductility for the tubes occurs when the specimen is irradiated at ~350 C and tested at about the same temperature.

Room-temperature burst tests have been performed on highly irradiated notched and unnotched Type 348 stainless steel pressure tubes.(153) These tubes had a wall thickness of about 210 mils, and the notched specimens had a notch with a 10-mil radius. Results of the burst tests are given in Table 34.

The results indicate that the stress required to burst the irradiated pressure tubes is higher than that required for the unirradiated pressure tubes. Also the calculated tensile stress is in good agreement with the measured tensile stress for both the irradiated and unirradiated material (Table 32).(150)

Room-temperature burst tests on the Type 348 stainless steel, from the Yankee cladding, showed that with increasing fast fluence, a higher stress was required for fracture. These results (Figure 82) indicate that hardness can be used as a fairly good approximation of burst strength of irradiated cladding.(154)

TABLE 32. POSTIRRADIATION MECHANICAL PROPERTIES OF TYPE 347 STAINLESS STEEL PLATE SPECIMENS IRRADIATED AT INTERMEDIATE TEMPERATURES[150]

Fluence, 10^{21} n/cm^2	Irradiation Temperature, C	Test Temperature, C	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.
1.7	282	RT	56[a]	130	92[a]	134	NA	2.0	57[a]	6.5	65[a]	59.1
1.9	266	RT		132		135		1.2		4.3		53.5
3.3	385	RT		143		147		1.0		9.4		61.5
3.9	330	RT		143		145		1.2		6.5		54.5
5.6	357	RT		140		145		1.8		10.6		59.8
7.1	390	RT		143		149		2.0		10.3		58.2
2.1	282	400	45[a]	103	75[a]	103		0.6	47[a]	4.5	NA	42.7
2.3	371	400		115		115		0.6		4.2		64
3.7	405	400		123		123.5		0.7		4.2		45.5
5.1	393	400		115		115		0.7		1.5		40.8
6.5	405	400		123.6		125.8		0.8		4.1		40.5
6.9	405	400		120		120		0.8		3.5		52.5
1.0	405	RT[b]	56[a]	72.3	92[a]	98		20.0	57[a]	44.3	65[a]	56.2
5.3	390	RT[b]		75.0		96		14.0		41.0		63.3
6.5	405	RT[c]		55.8		86.5		25.0		51.0		42.9
6.3	390	RT[c]		36.5		77.5		25.0		41.2		36.9
1.7	405	400[b]	45[a]	54.4	75[a]	69.2		11.1	47[a]	16.6	NA	55.8
2.3	355	400[b]		58.0		71.4		11.9		19.5		55.2
4.6	400	400[c]		33.0		62.5		21.5		26.2		53.6
6.3	400	400[c]		27.2		62.5		28.0		41.2		67.3

NA - not available.
(a) Averaged values.
(b) Annealed for 1 hour at 540 C.
(c) Annealed for 1 hour at 980 C.

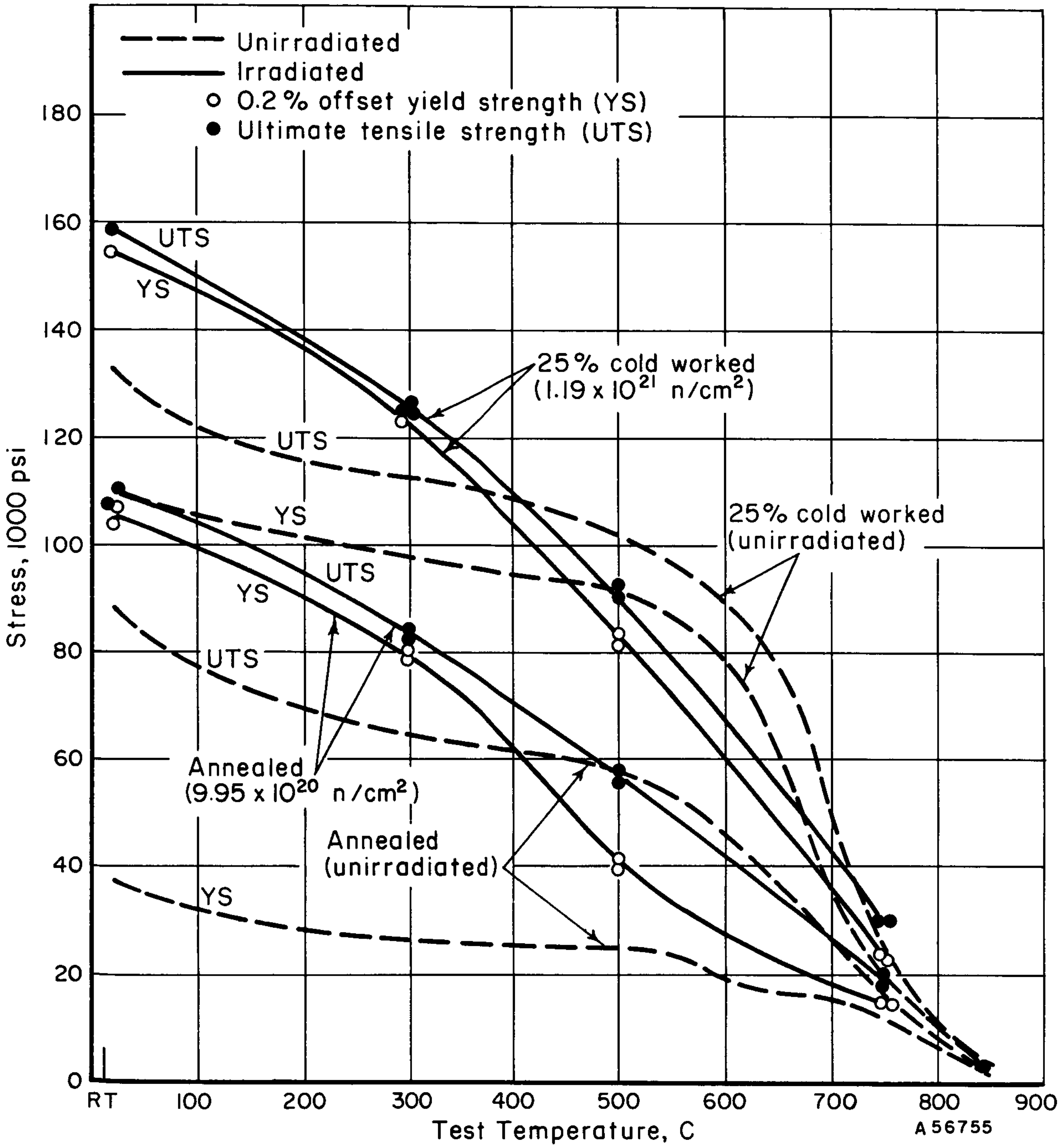


FIGURE 79. EFFECT OF TEST TEMPERATURE ON THE STRENGTH OF IRRADIATED ANNEALED AND 25 PERCENT COLD-WORKED TYPE 348 STAINLESS STEEL(131)

Irradiation temperature 60 C; fast fluence ~1.0 x 10^{21} n/cm^2; all transverse specimens.

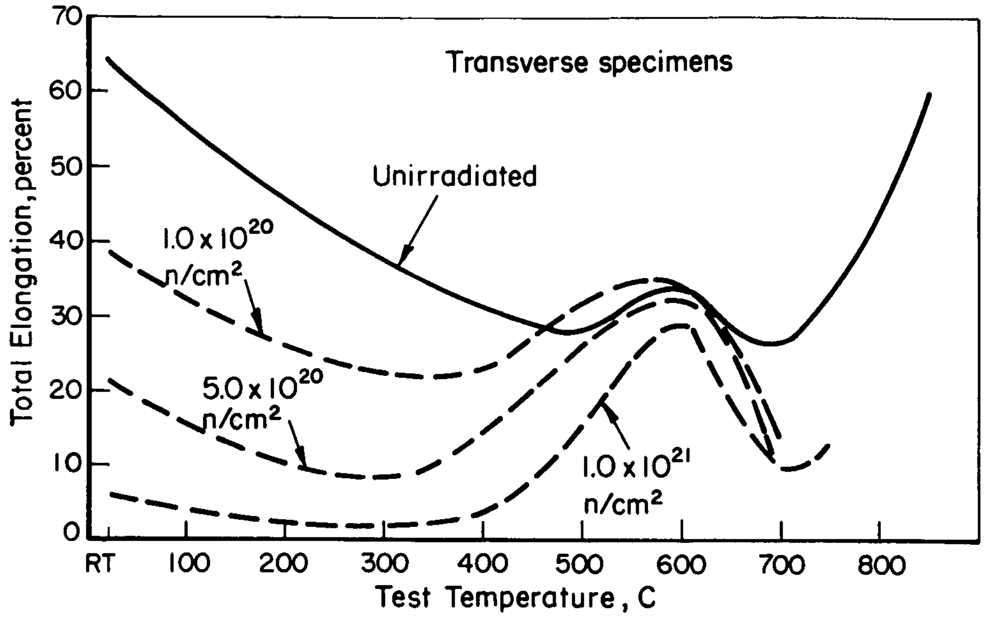


FIGURE 80. EFFECT OF TEST TEMPERATURE ON THE DUCTILITY OF IRRADIATED TYPE 348 STAINLESS STEEL IN THE ANNEALED CONDITION[131]

Irradiation temperature ~290 C.

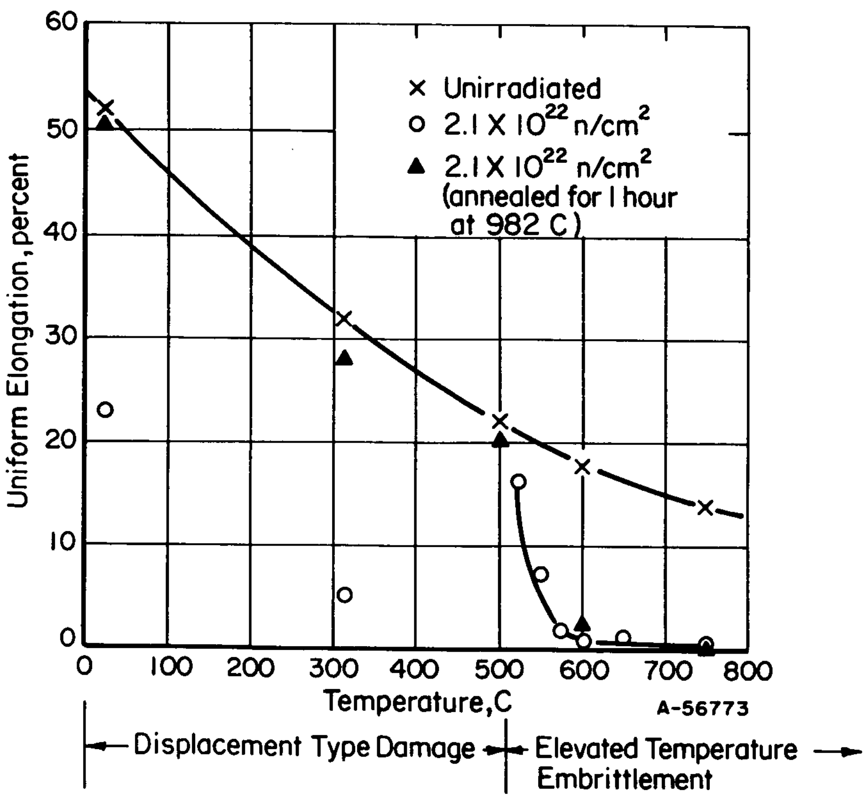


FIGURE 81. ONSET OF IRRADIATION-INDUCED ELEVATED-TEMPERATURE EMBRITTLEMENT IN TYPE 347 STAINLESS STEEL[151]

TABLE 33. POSTIRRADIATION TENSILE PROPERTIES OF SPECIMENS MACHINED FROM TYPE 348 STAINLESS STEEL CLADDINGS[a](152)

Test Temp, C	Fast Fluence, 10^{21} n/cm^2	0.2% Offset Yield Strength, 1000 psi	Ultimate Strength, 1000 psi	Elongation, percent Uniform	Elongation, percent Total
RT	0	42	100	45	51
	0.8	125	128	18.5	27.2
	2.2	147	150	0.4	6.1
	3.3	164	168	0.4	2.2
	5.7	173	177	0.3	1.2
	5.4	171	173	0.3	1.4
	7.1	(b)	170	0.2	(b)
	7.1	165	168	0.4	1.5
	7.9	166	166	0.2	2.7
	7.9	176	181	0.4	1.4
343	0	27	73	20	25
	3.2	(c)	135	0.1	0.2
	3.5	(c)	140	0.1	0.4
	5.7	(c)	145	0.1	0.3
	6.0	(c)	144	0.1	0.1
	3.6	(c)	141	0.1	0.4
	7.1	(c)	147	0.1	0.4
	7.1	(c)	141	0.2	0.4
	7.1	(c)	148	(b)	(b)
	7.9	(c)	152	0.1	0.3

(a) 0.34 inch OD, 20-mil wall thickness.
(b) Extensometer malfunctioned.
(c) Ultimate stress was reached before 0.2 percent offset strain.

TABLE 34. ROOM-TEMPERATURE BURST TESTS ON IRRADIATED PRESSURE TUBES(153)

Type	Fast Fluence, 10^{22} n/cm^2	Burst Pressure, 1000 psi Unirradiated	Burst Pressure, 1000 psi Irradiated	Calculated Tensile Strength, 1000 psi Unirradiated	Calculated Tensile Strength, 1000 psi Irradiated
Unnotched	0.9	15.6	27.8	100	150
Notched	1.1	8.1	14.2	100	140

Fatigue Properties. Irradiation to fast fluences of 5.5 and 11 x 10^{21} n/cm^2 has been found to increase the room-temperature fatigue life at total strains less than 1 percent and to decrease the fatigue life at total strains above 1 percent.[149] The improvement in fatigue life at low total strains is attributed to an increase in the proportional limit, which results in mostly elastic strain during cycling. The decreased fatigue life at high total strains is attributed to the irradiation-induced loss in ductility.

The effect of irradiation on fatigue properties of Type 348 stainless steel pressure tubes at room temperature is illustrated in Figure 83.[153] These tests were performed by cycling the internal pressure of the tubes from atmospheric to the test pressure. The test results indicate that irradiation slightly improved the room-temperature fatigue life of the pressure tubes.

Impact Properties. The impact properties of Type 347 stainless steel after irradiation to two levels of fluence are shown in Figure 84.[145, 149] 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 348 stainless steel is shown in Figure 71.[131] In Figures 85 and 86 an attempt has been made to correlate room-temperature hardness with yield strength and total elongation for stainless steels.[101, 155] It can be seen that an approximate relationship exists between these mechanical properties for both unirradiated and irradiated stainless steels.

Other Properties. Irradiation was found to cause no detectable increase in ferrite content, as measured by magnetic susceptibility.[134]

Predominantly Fast Fluence

Tensile Properties. The room-temperature tensile properties of Type 347 stainless steel which was part of the EBR-I core flow separation are given in Table 35.[156] The material was irradiated in liquid NaK which was at 228 C (inlet) and 316 C (outlet). Comparison of the results shown in Table 35 with those of Table 32 indicates that the

TABLE 35. EFFECT OF PREDOMINANTLY FAST FLUX ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF TYPE 347 STAINLESS STEEL[156]

Fast Fluence, n/cm^2	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent		Reduction in Area, percent	
	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
6.2 x 10^{20}	56.3[a]	75.6	89.7[a]	101.6	40[a]	30	54[a]	52
6.2 x 10^{20}	56.3[a]	76.8	89.7[a]	101.2	40[a]	28	54[a]	49
2 x 10^{21}	56.3[a]	95.0	89.7[a]	104.7	40[a]	19	54[a]	37
2 x 10^{21}	56.3[a]	102.7	89.7[a]	103.4	40[a]	19	54[a]	NA

(a) Average of two values.
NA - Not Available.

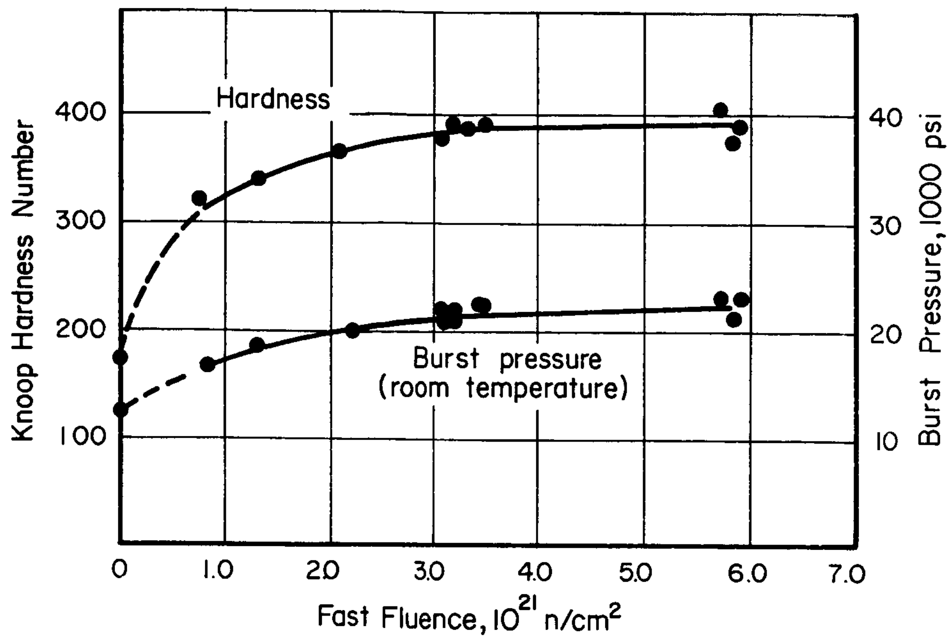


FIGURE 82. EFFECT OF IRRADIATION ON HARDNESS AND BURST PRESSURE OF YANKEE FUEL CLADDING(154)

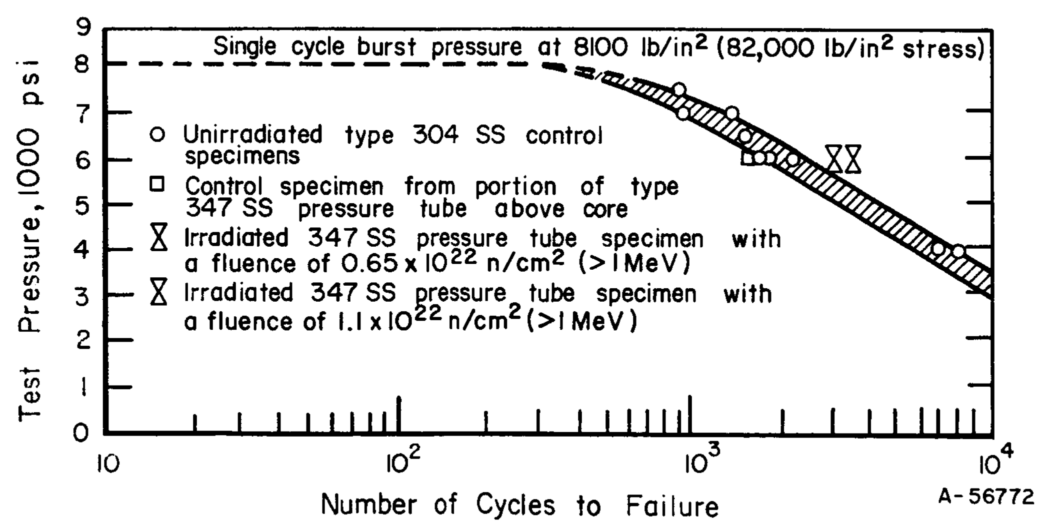


FIGURE 83. COMPARISON OF FATIGUE LIFE OF IRRADIATED AND UNIRRADIATED PRESSURE-TUBE SPECIMENS(153)

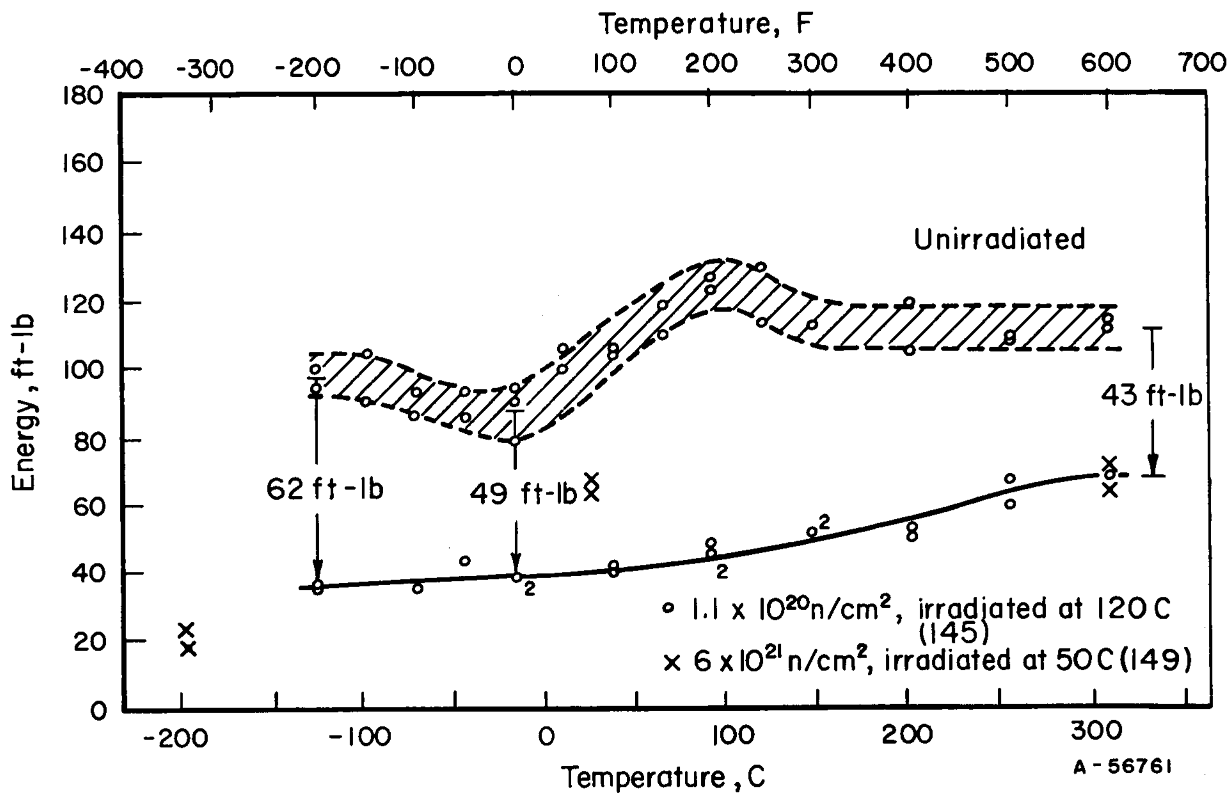


FIGURE 84. CHARPY-V NOTCH DUCTILITY CHARACTERISTICS OF TYPE 347 STAINLESS STEEL BEFORE AND AFTER IRRADIATION

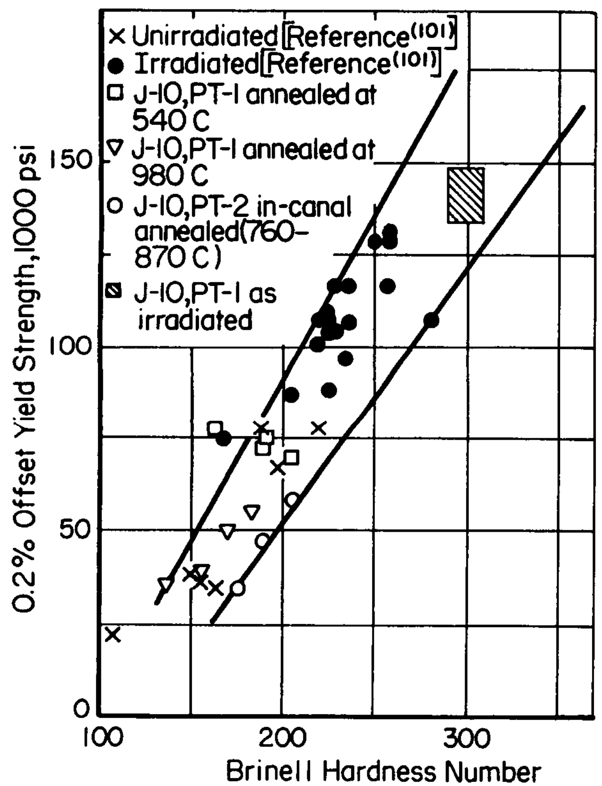


FIGURE 85. COMPARISON OF YIELD STRENGTH AND HARDNESS OF AUSTENITIC STAINLESS STEELS AT ROOM TEMPERATURE(155)

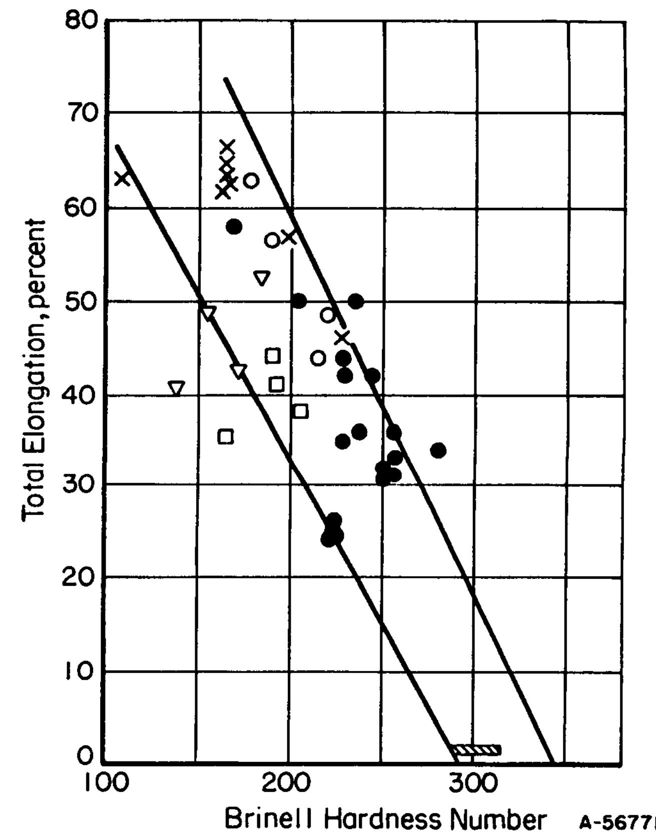


FIGURE 86. COMPARISON OF TOTAL ELONGATION AND HARDNESS OF AUSTENITIC STAINLESS STEELS AT ROOM TEMPERATURE(155)

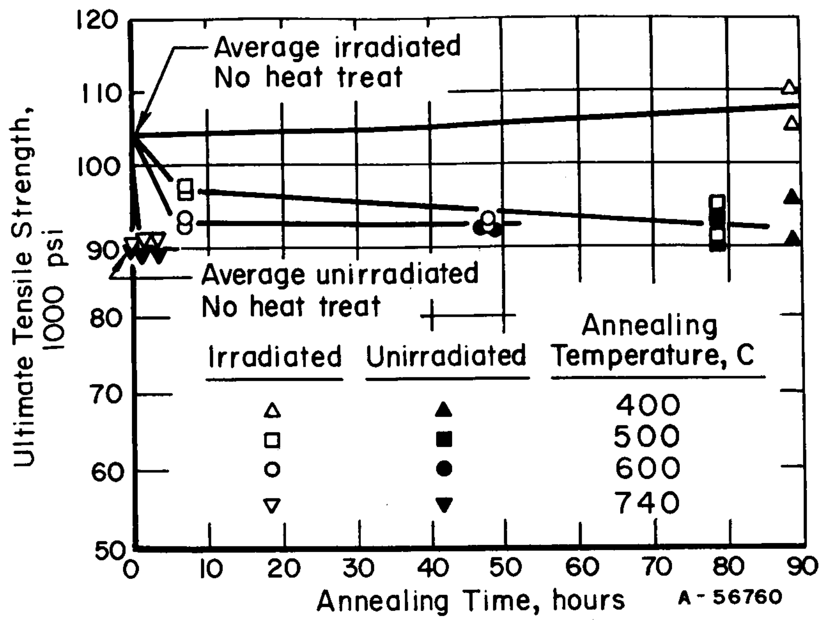


FIGURE 87. EFFECT OF ANNEALING ON ULTIMATE TENSILE STRENGTH OF TYPE 347 STAINLESS STEEL IRRADIATED TO A FAST FLUENCE OF 2 X 10^{21} N/CM^{2}(156)

mixed fast and thermal fluence causes significantly more change in mechanical properties than does a predominantly fast fluence at equivalent rates and irradiation temperatures. The effect of annealing on the tensile strength of the irradiated material is shown in Figure 87. These results show that annealing for 7 hours at 600 C and 79 hours at 500 C results in the removal of irradiation effects, while annealing at 400 C for 90 hours results in an increase in irradiation hardening.(156)

AISI Types 316 and 316L Stainless Steel

Mixed Fast and Thermal Fluence

Tensile Properties. A surprisingly small number of tensile tests have been performed on irradiated Type 316 stainless steel at room temperature. Results of the room-temperature tensile tests are given in Table 36.(156) The mechanical properties undergo the typical irradiation-induced changes, i.e., the yield and ultimate strength are increased and the ductility is decreased.

TABLE 36. ROOM-TEMPERATURE MECHANICAL PROPERTIES OF IRRADIATED COLD-WORKED TYPE 316 STAINLESS STEEL[156]

Fast Fluence, 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.
1.2×10^{19}	67.1	101.2	91.4	105.2	60.0	44.0	76	68
5.6×10^{19}	67.1	104.8	91.4	108.3	60.0	44.0	76	68
9.9×10^{19}	67.1	107.5	91.4	107.4	60.0	42.0	76	68
1.3×10^{21}	67.1	107.1	91.4	110.6	60.0	42.0	76	67

Irradiation at elevated temperatures results in only minor changes in room-temperature tensile properties as shown in Table 37. It is postulated that the irradiation temperature is sufficiently high to anneal out the displacement-type of irradiation damage. However, a tensile test at 595 C, after irradiation to a fast fluence of 6×10^{19} n/cm^2, indicated that the yield strength is not changed by irradiation at 430 C, but the total elongation is reduced from 35 to 21 percent.[121] The irradiation-induced decrease in ductility at elevated temperatures has been shown to be dependent on the thermal neutrons rather than fast neutrons.[158] This is illustrated in Figure 88 where the ductility of stainless steel irradiated to equivalent fast fluences but varying thermal fluences is compared. The elevated-temperature ductility is shown to decrease further with increasing thermal fluence as shown in Figure 89. Note that while Type 316L appears to undergo less irradiation-induced embrittlement at lower thermal fluences than does Type 316, they both suffer about the same loss in ductility at higher thermal fluences.

TABLE 37. EFFECT OF ELEVATED-TEMPERATURE IRRADIATION ON THE ROOM-TEMPERATURE MECHANICAL PROPERTIES OF TYPE 316 STAINLESS STEEL

Fast Fluence, n/cm^2	Irradiation Temperature, C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Total Elongation, percent		Reference
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
2.5×10^{19}(a)	650	30	27	81	78	62	56	157
6×10^{19}(b)	430	55.2	61	NA	NA	38.5	38	121
4.0×10^{20}(b)	430	55.2	59.8	NA	NA	38.5	36.5	121

NA - Not available.
(a) As annealed.
(b) Cold worked.

Creep Properties. The effect of irradiation on the tube burst properties of Type 316 stainless steel is illustrated in Figure 90.[158] It can be seen that burst life and elongation are reduced by increasing thermal fluence.

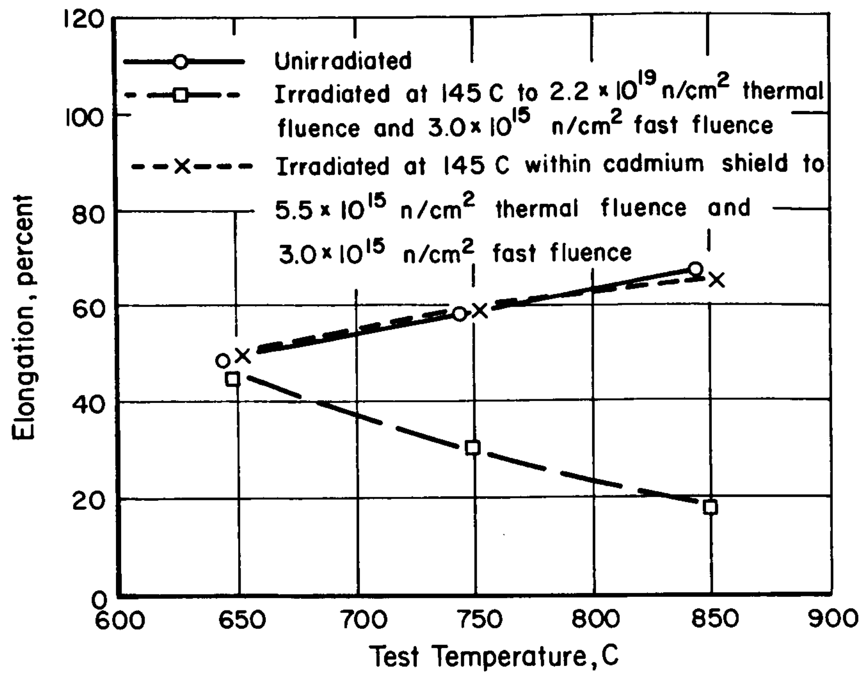


FIGURE 88. EFFECTS OF THERMAL FLUENCE ON THE ELEVATED-TEMPERATURE TENSILE DUCTILITIES OF TYPE 316 AUSTENITIC STEELS(158)

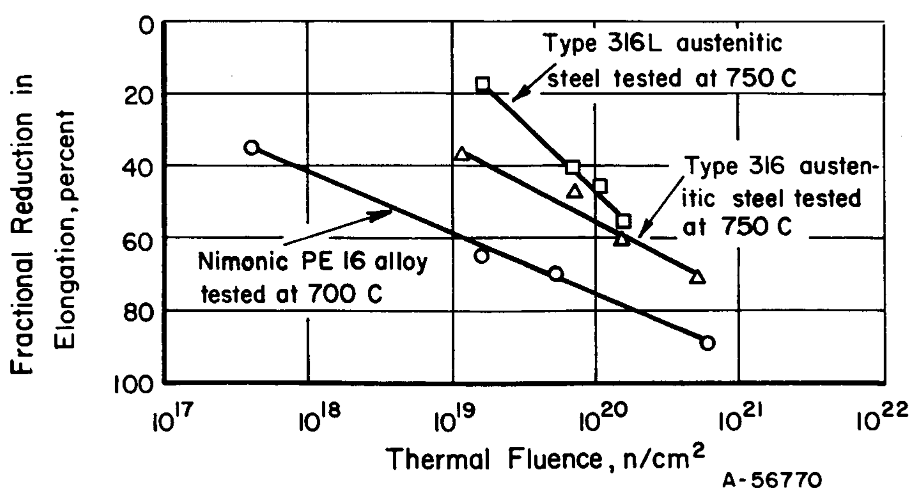


FIGURE 89. EFFECTS OF THERMAL FLUENCE ON THE REDUCTIONS IN THE ELEVATED-TEMPERATURE TENSILE DUCTILITIES OF TYPES 316 AND 316L AUSTENITIC STEELS AND NIMONIC PE16 ALLOY(158)

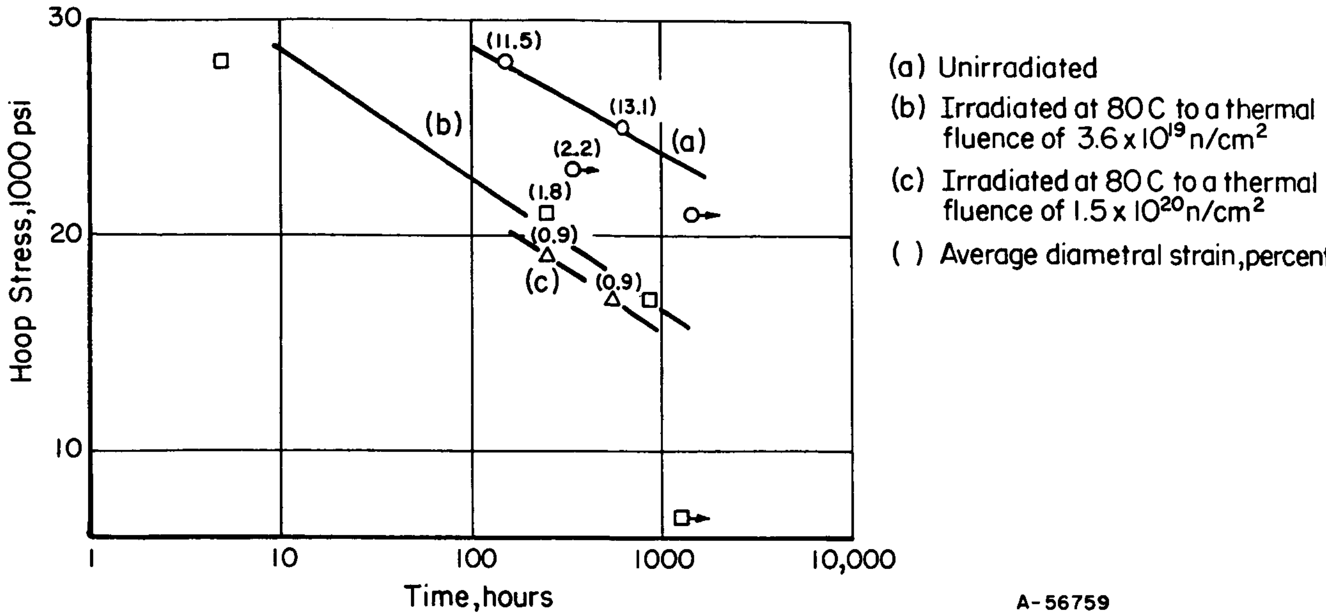


FIGURE 90. EFFECTS OF NEUTRON IRRADIATION ON THE 650 C TUBE-BURST PROPERTIES OF TYPE 316 AUSTENITIC STEEL(158)

Predominantly Fast Fluence

Tensile Properties. The results of tensile tests on Type 316L stainless steel irradiated in the LAMPRE Reactor at 525 C to a fast fluence of 2 x 10^{21} n/cm^2 are given in Table 38.(159) The British have also irradiated Type 316 stainless steel in the Dounreay Fast Reactor at temperatures of 230 to 350 C. The results of tensile tests on the irradiated specimens are shown in Figure 91.(160) These tensile test results on materials irradiated at two different temperatures (at 525 C in LAMPRE and at 250 to 350 C in DFR) indicate that at 525 C, almost all of the displacement type of damage is annealed out, since no irradiation-induced changes in yield strength or ductility are evident (Table 38). The irradiations at 250 to 350 C result in increases in yield and ultimate tensile strength up to a testing temperature of about 600 C, after which the displacement type of damage is expected to be annealed out during the performance of the tensile test. However, specimens from both irradiations show a decrease in elevated temperature ductility after 600 to 700 C. This ductility decrease is attributed to the helium generated by fast neutrons by (n, α) reactions from iron, nickel, and chromium. The ductility at 650 and 700 C is found to decrease with an increasing fast fluence as illustrated in Figure 92.(160) Prestraining of the stainless steel is expected to improve the postirradiation ductility by providing a defect structure where the helium atoms could be anchored. However, Figure 92 shows that prestraining does not have a significant effect on the postirradiation elevated-temperature ductility of Type 316 stainless steel.

Creep Properties. Figures 93 and 94 illustrate the effect of fast fluence on the creep life of pressurized tubes.(160) It is apparent that irradiation reduces both the rupture life and the ductility of the tubes, the ductility progressively decreasing with increasing fast fluence.

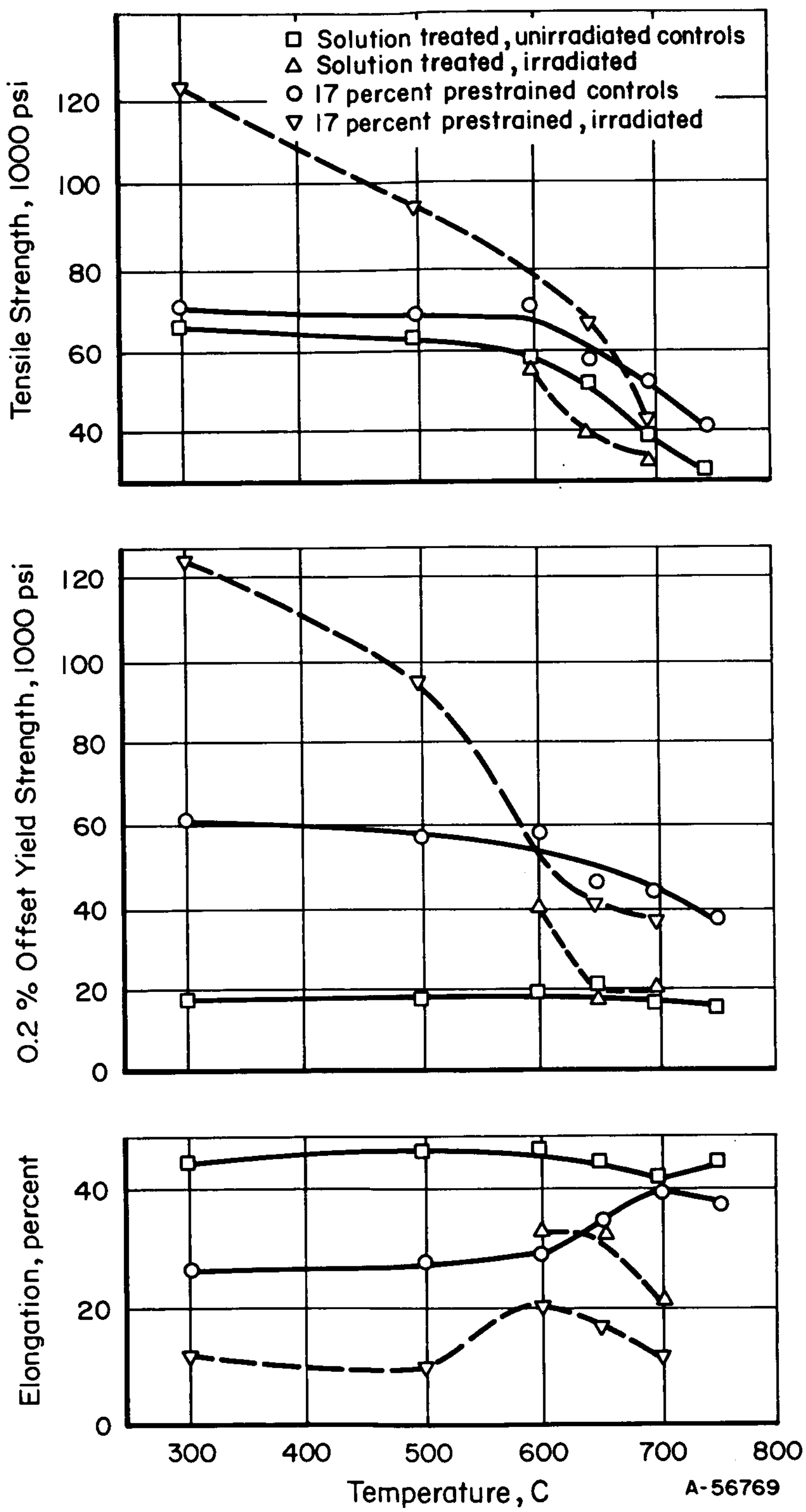


FIGURE 91. TENSILE PROPERTIES OF SOLUTION TREATED AND 17 PERCENT PRESTRAINED TYPE 316 STEEL AFTER IRRADIATION TO A FAST FLUENCE OF 3.6 X 10^{22} N/CM^2(160)

Initial strain rate 1.58 x 10^{-4} sec^{-1}.

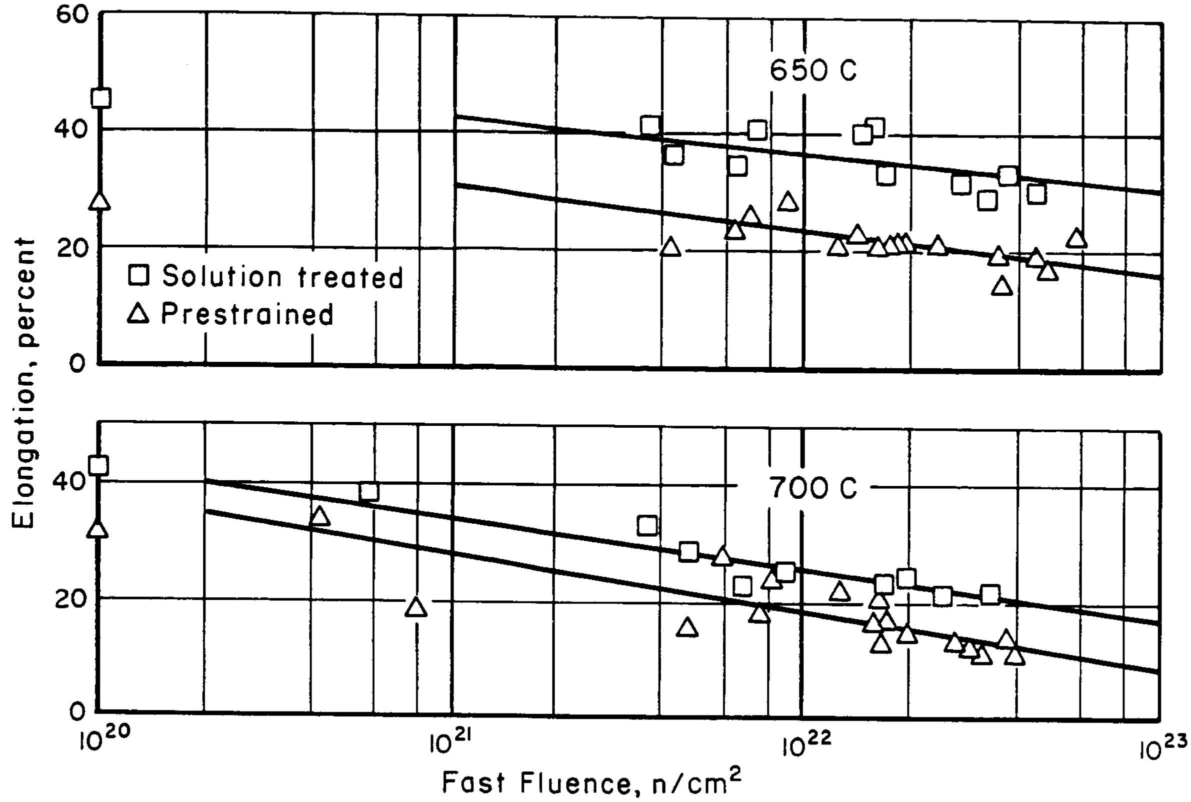


FIGURE 92. DUCTILITY OF TYPE 316 AT 650 AND 700 C AFTER IRRADIATION AT 250 TO 350 C(160)

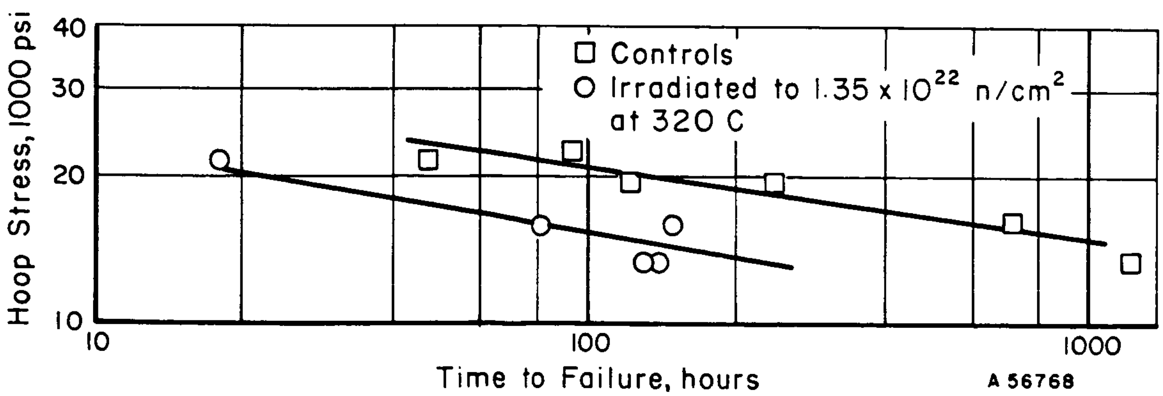


FIGURE 93. EFFECT OF IRRADIATION ON THE RUPTURE LIFE OF TYPE 316L STAINLESS STEEL TUBES AT 650 C(160)

TABLE 38. MECHANICAL PROPERTIES OF TYPE 316L STAINLESS STEEL IRRADIATED IN THE LAMPRE TO A FAST FLUENCE OF 2 x 10^{21} N/CM^2 AT 525 C(159)

Test Temperature, C	Yield Strength, 1000 psi		True Uniform Strain, percent		Total Elongation, percent	
	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
25	33.1	34.2	51.0	51.7	72.2	75.6
400	19.0	19.6	33.5	33.1	43.6	43.7
500	17.8	20.5	33.5	34.0	44.8	44.4
600	19.8	19.7	29.7	29.3	40.8	38.8
700	15.6	18.6	18.3	17.7	42.7	27.0
800	13.2	15.0	14.0	4.5	42.3	7.3
900	11.3	10.8	14.6	1.8	59.3	3.7

Hardness. The effect of annealing on the postirradiation hardness of Type 316 stainless steel at Dounreay to a fast fluence of 1.6 x 10^{22} n/cm^2 is illustrated in Figure 95.(161) The temperature where the effects of irradiation on hardness are removed corresponds well with the findings, which showed that irradiation at 525 C did not result in irradiation-induced changes in mechanical properties at room temperature.(159)

AISI Type 318 (Fv548)

Predominantly Fast Fluence

Tensile Properties. Type 318 is a niobium-stabilized modification of Type 316 stainless steel with 1 weight percent niobium added. Limited postirradiation tensile tests have been performed by the British on Type 318 stainless steel which has been irradiated in the Dounreay Fast Reactor.(161) Results of these tensile tests are shown in Figures 96 and 97. The Type 318 stainless steel in the annealed condition behaves similarly to other stainless steels after irradiation. Since the irradiation temperature was 250 to 350 C, sufficient displacement type of irradiation damage is present at testing temperatures of up to about 600 C where the displacement type of damage is annealed out. At 700 C, the steel exhibits the irradiation-induced elevated-temperature embrittlement which is typical of other austenitic stainless steels (Figure 96). However, the Type 318 stainless steel does not appear to exhibit the irradiation-induced elevated-temperature embrittlement if it is 20 percent cold worked before irradiation (Figure 97). This better performance is attributed to fine particles of NbC which prevent the helium from reaching the grain boundaries.

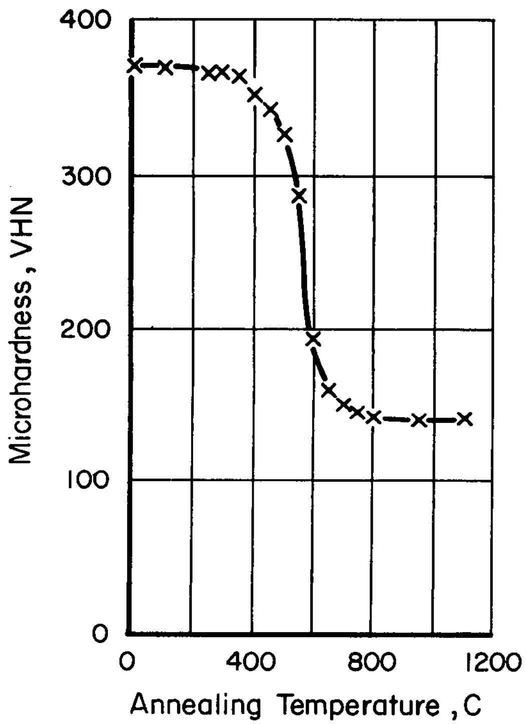


FIGURE 94. EFFECT OF IRRADIATION ON THE CREEP OF PRESSURIZED TYPE 316L STAINLESS STEEL TUBES AT 650 C(160)

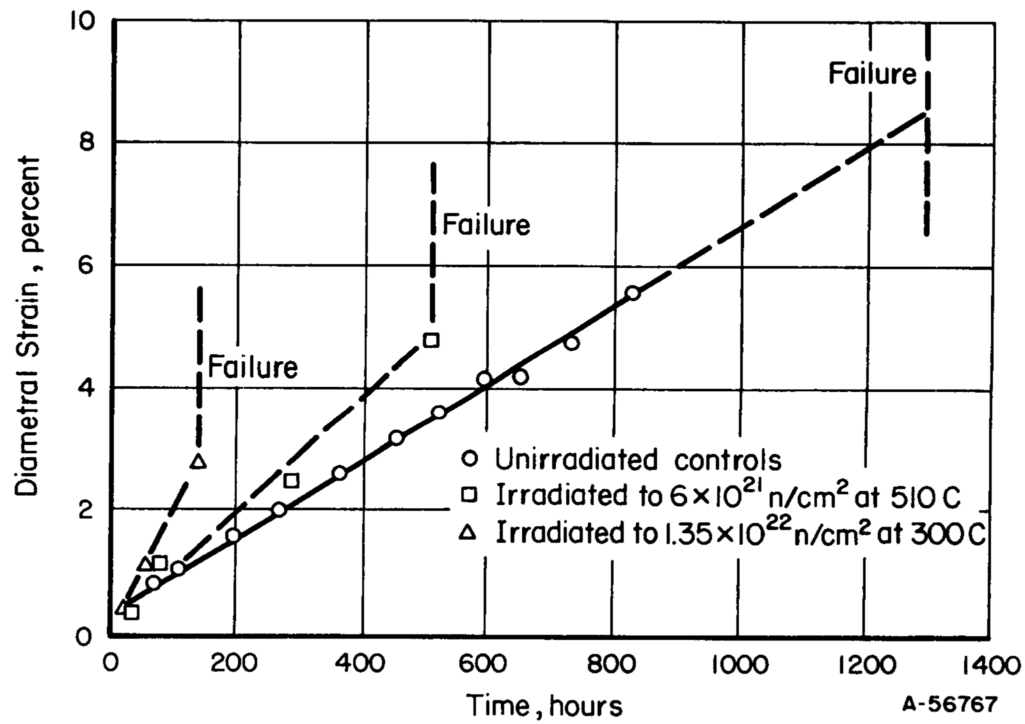


FIGURE 95. PLOT OF HARDNESS VERSUS ANNEALING TEMPERATURE FOR TYPE 316 STAINLESS STEEL IRRADIATED AT 300 C TO A FAST FLUENCE OF 1.6 X 10^{22} N/CM^2(161)

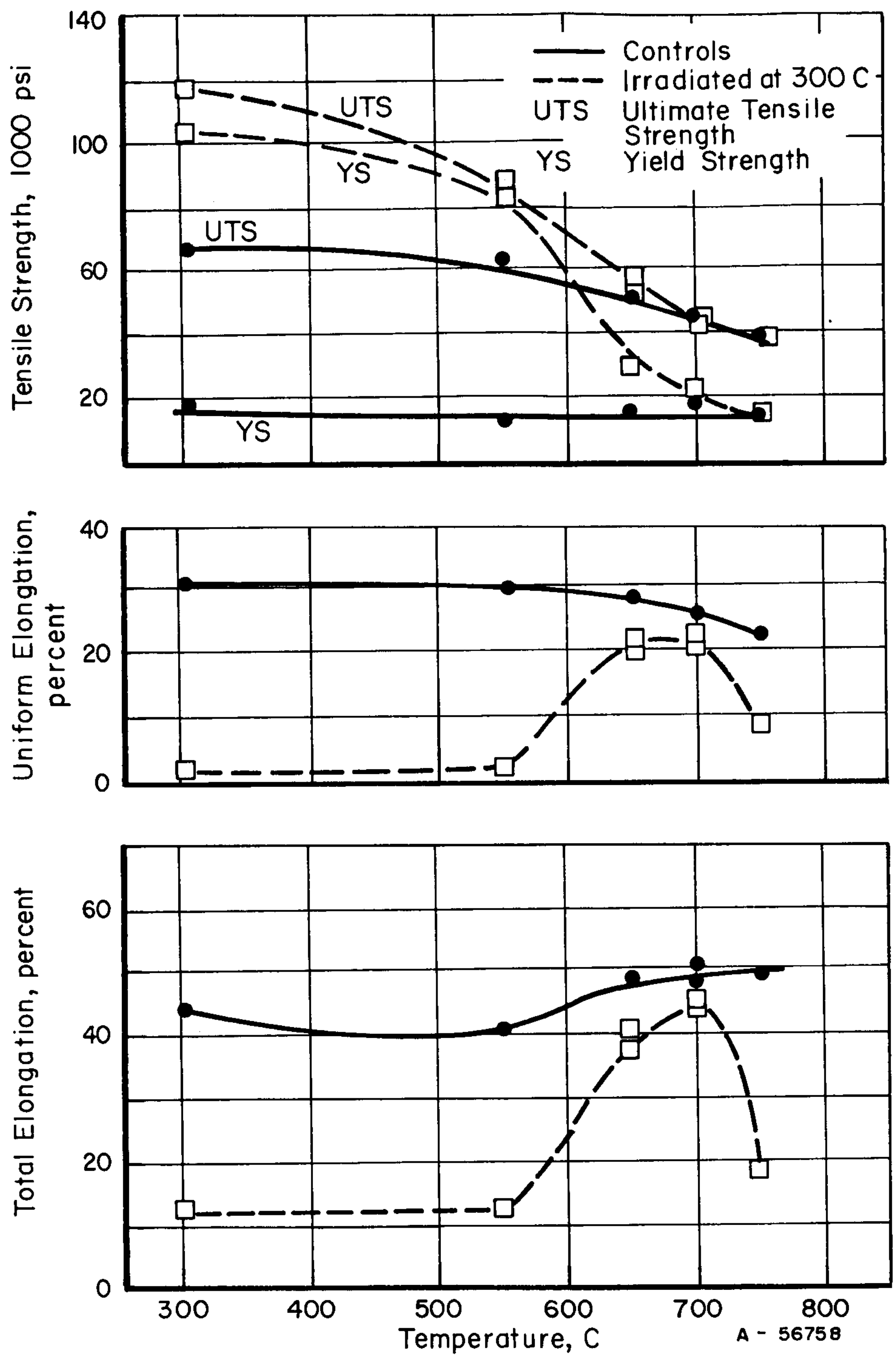


FIGURE 96. TENSILE PROPERTIES OF SOLUTION-TREATED AND 850 C AGED FV 548 AFTER IRRADIATION TO A FAST FLUENCE OF 3.8 X 10^{22} N/CM^{2}(160)

Initial strain rate was 5.8×10^{-4} sec^{-1}.

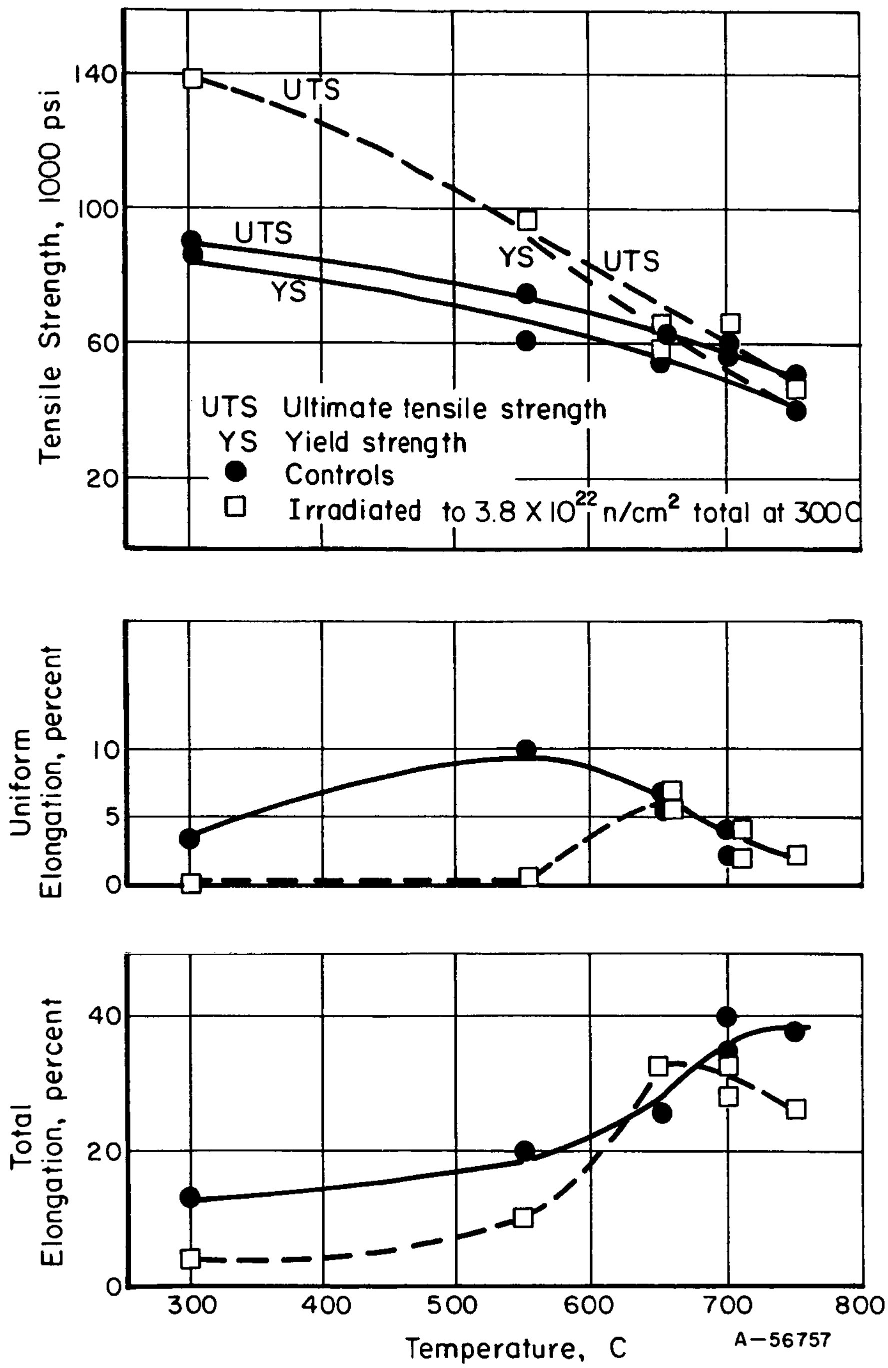


FIGURE 97. TENSILE PROPERTIES OF 20 PERCENT PRESTRAINED FV 548 AFTER IRRADIATION TO A FAST FLUENCE OF 3.8 X 10^{22} N/CM^2 (160)

Initial strain rate 5.8 x 10^{-4} sec^{-1}.

20Cr-25Ni-Nb Stainless Steel

Mixed Thermal and Fast Fluence

Tensile Properties. The effect of neutron irradiation on the tensile properties of 20Cr-25Ni-Nb stainless steel is illustrated in Figure 98.(158) It can be seen that the low temperature strength properties are increased by irradiation while the elongation is somewhat decreased by irradiation. At temperatures above 500 C there is no significant difference between the unirradiated and irradiated tensile properties. This lack of difference in tensile properties can be attributed to the annealing out of displacement type of irradiation damage by the temperature. However, at temperatures above 700 C the elongation of the irradiated material is somewhat lower. This decrease in ductility is attributed to the helium induced embrittlement which has been found in austenitic stainless steels and nickel alloys.

A-286 Stainless Steel

Mixed Thermal and Fast Fluence

Tensile Properties. A limited number of tensile tests have been performed on irradiated A-286. The results are given in Table 39.(162, 163) These results indicate that irradiation does not affect the strength properties of A-286 at most testing temperatures. No significant changes occur in the room-temperature ductility as a result of irradiation, but at 540 C, significant irradiation-induced ductility losses take place. These irradiation-induced ductility losses become more drastic at 650 C, resulting in total elongations in the range of 1 to 2 percent.

Creep Properties. Results of creep rupture tests at 650 C on unirradiated and irradiated A-286 stainless steel are shown in Figure 99.(164) These results indicate that the precipitation-hardened stainless steel is severely embrittled by irradiation, even after low fast fluences such as 10^{16} n/cm^2. This irradiation-induced embrittlement is attributed to helium which is generated from boron-10 by the (n, α) reaction. The alloy also loses considerable strength as a result of irradiation. However, some of the loss of strength may be due to overaging rather than to irradiation.

Fatigue Properties. Fatigue tests have been performed on irradiated A-286 at 690 C and the results are given in Table 40.(165, 166) Irradiation seems to improve the fatigue life significantly (up to a factor of 3) at 60,000 psi, while only minor improvements are obtained at 68,000 psi. At 75,000 psi, the irradiation causes a 6 to 7-fold increase in fatigue life.

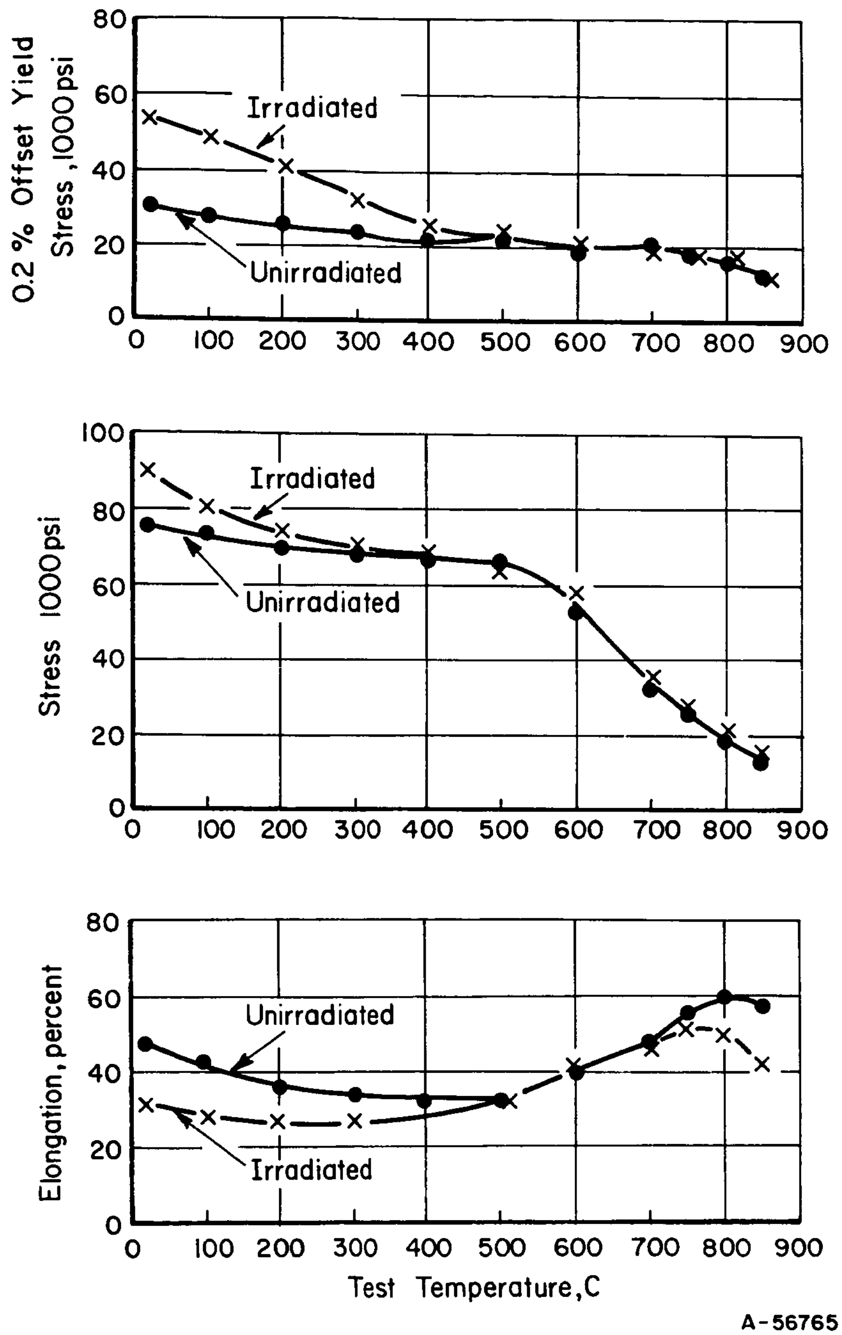


FIGURE 98. EFFECT OF NEUTRON IRRADIATION [7.6 X 10^{18} N/CM^2 (THERMAL FLUENCE), 3.8 X 10^{18} N/CM^2 (FAST FLUENCE) AT 45 C] ON THE ROOM- AND ELEVATED-TEMPERATURE TENSILE PROPERTIES OF 20Cr-25Ni, NIOBIUM-STABILIZED AUSTENITIC STEEL(158)

TABLE 39. EFFECTS OF IRRADIATION ON THE TENSILE PROPERTIES OF A-286[a][162, 163]

Fast Fluence, 10^{19} n/cm^2	Irradiation Temperature, C	Test Temperature, C	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Total Elongation, percent		Reduction in Area, percent	
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
5	650	RT	111	110.7	172.1	166.6	22.5	22.9	45.5	41.0
5	650	315		114		156.3		15.8		28.3
5	650	540	106	104.8	144.7	141.7	13.2	7.8		10.9
5	650	650	95.1	106.7	114.2	107.1	5.1	1.4	4.8	1.6
5	650	650	96	102	111	110.5	4.6	1.5	7.9	1.6
5	650	650	96	105.5	111	109.5	4.6	2.1	7.9	1.2
7	540	RT	111	122.7	172	181.4	22.5	23.4	45.5	18.3
7	540	RT	111	120.2	172	174.8	22.5	13.7	45.5	37.3
7	540	RT	118	119.4	172	167.5	21.6	19.3	43.7	25.5
7	540	RT	118	115.9	172	168.1	21.6	21.1	43.7	32.1
7	540	315		116.4		165.7		--	43.7	15.5
7	540	315		100.1		155.2		18.4		34.7
7	540	540	106	121.8	144.7	148.3	13.2	7.5	21.4	11
7	540	540	106	116.6	144.7	139.8	13.2	7.2	21.4	11.1
7	540	540	113.5	112.7	138	143.9	8.0		24	
7	540	540	113.5	113.8	138	142.0	8.0		24	
20	60-95	RT	134	139.8	164.7	148.3	24	6.3	53.8	11.9
20	60-95	540[b]		111		130		6.3		10.7
20	60-95	650	101	91.1	108.5	111	4	2.5	6.6	2.5
20	60-95	650[c]		115.7		115.7		2.0		5.0
70	60-95	540	107.4	111	136	130	10.5	6.3		
70	60-95	650	102.5	91	109.5	111	36	2.5		

(a) 2 hours at 900 C, water quenched; 16 hours at 720 C, air cooled; 16 hours at 650 C, air cooled.

(b) Held at 540 C for 24 hours before testing.

(c) Held at 650 C for 24 hours before testing.

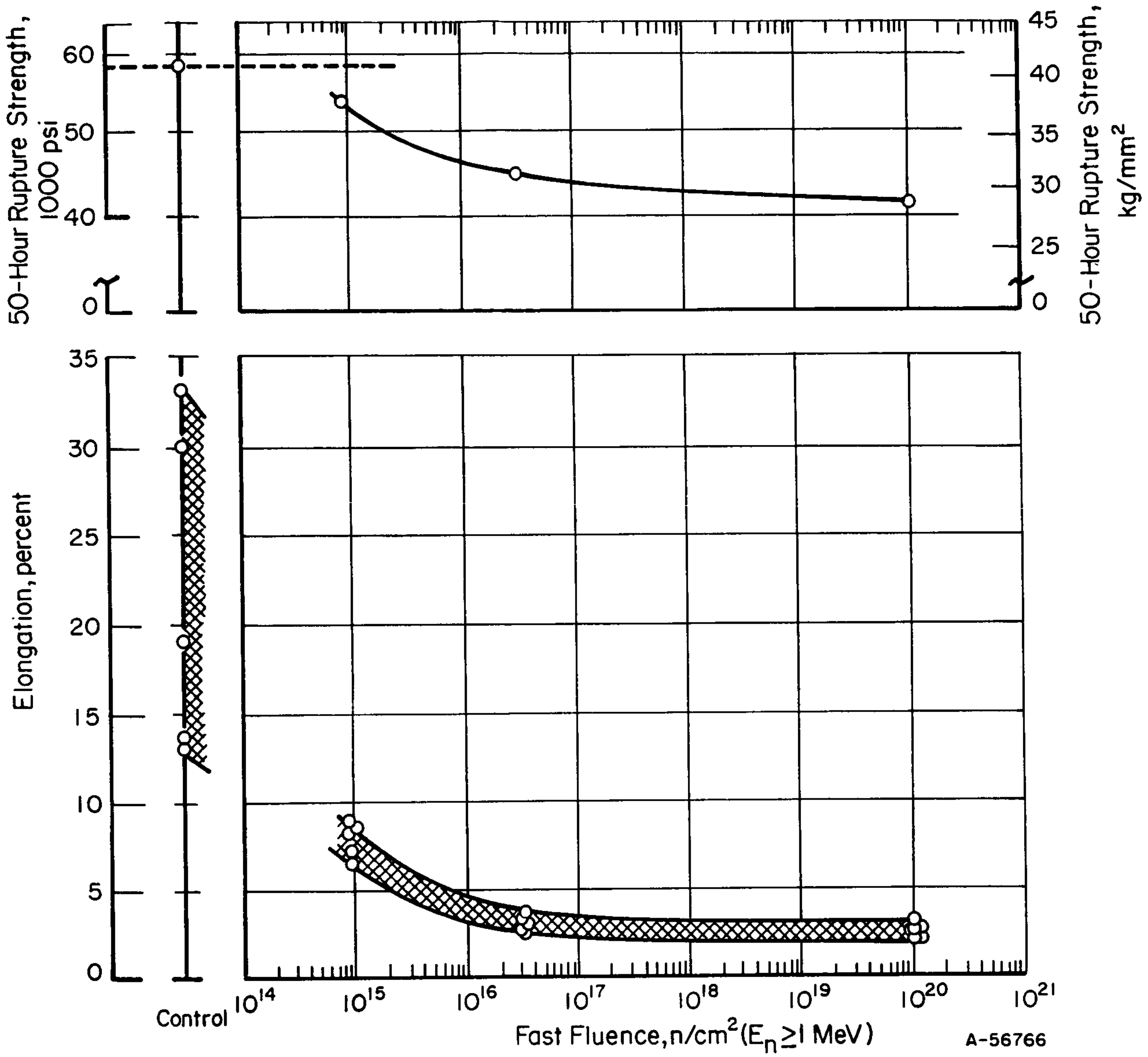


FIGURE 99. STRESS-RUPTURE STRENGTH AND ELONGATION OF A-286 ALLOY AT 650 C AS A FUNCTION OF FAST FLUENCE(164)

TABLE 40. EFFECT OF IRRADIATION ON THE FATIGUE LIFE OF A-286 AT 690 C(165, 166)

Fast Fluence, 10^{19} n/cm^2	Stress, 1000 psi	Number of Cycles to Failure, 10^6
Control	60	21.6
Control	60	13.4
4	60	72.0
4	60	49.6
Control	68	3.6
Control	68	3.4
4	68	4.3
4	68	5.7
4	68	4.0
Control	75	0.22
Control	75	0.16
4	75	1.2
4	75	1.4

Incoloy 800

Mixed Thermal and Fast Fluence

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 41. 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 42. These postirradiation tensile results exhibit trends similar to those for the room-temperature properties

TABLE 41. 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	167
1×10^{20}	50		64				39			168
1×10^{20}	740		24				37			168
4×10^{20}	400	41.3	46.8			37.7	35.7			121
0.6×10^{21}	180	44.4	86.1	86.5	104.6	41.7	25.9	50.4	34.4	169
1.2×10^{21}	180	44.4	94.9	86.5	109.1	41.7	25.0	50.4	33.3	169
2.4×10^{21}	180	44.4	113.1	86.5	120.6	41.7	20.6	50.4	27.5	169
2.5×10^{21}	180	44.4	114.1	86.5	121.0	4.17	21.5	50.4	27.6	169

TABLE 42. 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	168
9.2×10^{19}(b)	740	300	21.6	22.0	62.6	62.8	37.8	39.8	40.7	43.6	168
9.2×10^{19}(c)	740	300		27.4		73.9		31.5		36.2	168
2.2×10^{20}(a)	740	300	34.3	45.3	77.6	89.8		34.3		37.4	168
2.2×10^{20}(b)	740	300	25.7	22.9	67.4	61.6	30.8	34.1	34.4	37.9	168
2.2×10^{20}(c)	740	300	29.7	23.8	76.5	69.6	24.8	33.0	31.5	37.0	168
2.4×10^{20}(a)	740	300	35.3	30.6	76.1	69.1	24.9	30.4	30.6	35.9	168
2.4×10^{20}(b)	740	300	24.4	23.3	69.8	61.5	29.7	37.3	33.8	40.7	168
2.4×10^{20}(c)	740	300	28.4	24.3	75.7	70.4	24.3	32.6	27.2	37.2	168
6×10^{20}	180	315		58.7		84.1		25.8		31.1	169
1.2×10^{21}	180	315		72.9		90.5		23.4		28.6	169

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

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 43. Here again 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 drastic decrease in elevated-temperature ductility is believed to be due to the formation of helium by (n, α) reactions from the boron impurity which is present. To check this hypothesis, Incoloy 800 specimens with various boron contents were irradiated at 704 C and tested at 593, 704, and 816 C after irradiation.[170] The results of these tests (Table 44) indicate that only limited improvement in the postirradiation ductility of Incoloy 800 at elevated temperatures can be obtained by reducing the boron content. In another study it was found that the variation of boron content from 5 to 77 ppm had no detectable effect on the postirradiation ductility of Incoloy 800 at elevated temperatures.[171]

The addition of titanium to Type 304 stainless steel has been shown to result in improved postirradiation ductility at elevated temperatures.[139] Therefore, an attempt was made to see whether increasing the titanium content (usual content 0.2 weight percent) of Incoloy 800 would improve the postirradiation ductility at elevated temperatures.[171] An alloy which contained 0.9 weight percent titanium (Incoloy 801) was irradiated to a fast fluence of 3.5×10^{20} n/cm^2 and tested at 593 C. The postirradiation total elongation of Incoloy 801 was reduced to 1 percent, while the standard Incoloy 800 retained a total elongation of 22 percent after irradiation. The postirradiation total elongation at 704 C, after a fast fluence of 2.0×10^{21} n/cm^2, was found to be 7 percent for Incoloy 800 and less than 1 percent for Incoloy 801. The more severe irradiation embrittlement of Incoloy 801 is attributed to strengthening of the matrix by the addition of titanium, thereby resulting in relatively weakened grain boundaries.

Preirradiation heat treatment and carbon content are other variables whose effect on the postirradiation elevated-temperature ductility of Incoloy 800 has been investigated.[170] Carbon content is believed to be important in that carbide particles will precipitate in front of the grain boundaries and thus act as a barrier to prevent the helium atoms from diffusing to the grain boundaries. To prevent the helium from reaching the grain boundaries, it is not only necessary to have a sufficient carbon content, but the carbide particles have to be of certain size and distribution. Table 45 enumerates the postirradiation tensile data for Incoloy 800 which had various carbon contents and which was given different preirradiation heat treatments. It can be seen that the carbon content by itself does not influence the postirradiation ductility, but some improvement can be obtained if the specimens are annealed for 96 hours at 816 C before irradiation.

No real differences in resistance to irradiation embrittlement are apparent for the Incoloy-800 specimens which have been given various preirradiation treatments such as solution annealing, mill annealing, and 20 percent cold working.[168] All of these treatments result in about the same percentage loss of postirradiation ductility at elevated temperatures after fast fluences of 2×10^{20} n/cm^2.

Tensile tests have been performed on Incoloy-800 cladding irradiated to a fast fluence of 4×10^{20} n/cm^2 at 400 C.[121] The results (Table 46) show that the irradiation-induced ductility loss in tubes is more severe than that in regular tensile specimens irradiated and tested under identical conditions. A few burst tests have also been performed on irradiated Incoloy 800 tubing. The results (Table 47)[170] indicate that irradiation reduces the hoop strength and causes a drastic reduction in ductility.

TABLE 43. 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: Uniform		Elongation, percent: Total		Reference
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
4.0 x 10^{20}	400	593	28.2	27.7					34.5	23	121
1.5 x 10^{21}	180	593	26.3	34.5	62.8	50.2	37.3	6.5	42.5	7.2	169
1.9 x 10^{21}	180	593	26.3	29.7	62.8	48.4	37.3	7.8	42.5	8.3	169
2.2 x 10^{21}	180	593	26.3	31.4	62.8	43.7	37.3	9.0	42.5	9.8	169
2.0 x 10^{19}	40	650	17.8	16.6	43.5	35.9	40	28.5	45.0	29.0	167
9.9 x 10^{19}(a)	740	650	22.6	21.4	48.1	36.4	24.7	8.1	42.8	8.7	168
9.9 x 10^{19}(b)	740	650	15.7	11.6	46.6	27.5	23.8	11.9	32.1	12.8	168
9.9 x 10^{19}(c)	740	650	25.1	15.6	50.5	33.8	17.4	10.8	48.8	11.3	168
2.2 x 10^{20}(a)	740	650	32.6	25.6	60.3	37.5	15.8	3.7	37.3	5.5	168
2.2 x 10^{20}(b)	740	650	22.9	12.6	53.3	28.8	15.8	9.6	32.5	10.3	168
2.2 x 10^{20}(c)	740	650	25.7	18.6	54.7	35.2	17.9	7.6	39.1	8.1	168
2.4 x 10^{20}(a)	740	650	31.5	19.7	54.5	31.4	16.6	5.4	37.2	5.9	168
2.4 x 10^{20}(b)	740	650	22.3	8.1	51.6	24.5	26.3	12.2	32	13.4	168
4.0 x 10^{20}(a)	740	650		22.5		32.6		4.3		4.6	168
4.0 x 10^{20}(b)	740	650		14.0		27.2		8.6		8.7	168
4.0 x 10^{20}(c)	740	650		18.9		31.1		6.6		6.9	168
2.0 x 10^{19}	40	750	16.6	15.6	25.6	22.7	17.0	10.0	28.0	12.0	167
2.0 x 10^{19}	40	820	16.3	13.6	19.8	15.6	9.0	6.0	17.0	8.0	167

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

TABLE 44. TENSILE DATA FOR INCOLOY 800 IRRADIATED AT 704 C TO A FAST FLUENCE OF 1.7 x 10^{21} N/CM^2 (170)

Baron Impurity Content, wt %	Test Temperature, C	Thermal Controls: Yield Strength, 1000 psi	Thermal Controls: Ultimate Tensile Strength, 1000 psi	Thermal Controls: Elongation, percent Uniform	Thermal Controls: Elongation, percent Total	Irradiated: Yield Strength, 1000 psi	Irradiated: Ultimate Tensile Strength, 1000 psi	Irradiated: Elongation, percent Uniform	Irradiated: Elongation, percent Total
1.8	593	31.1	53.9	15.8	18.4	20-38	38.4-38.6	7.2-7.6	8.6-8.8
	704	25.9	43.7	13.3	16.7	14-21	14-21.5	0.8-2.3	1.6-5
	816	17.3	19.4	7.3	22.4	9-17.2	9-18.3	0.9-1.8	2.1-3.2
7.0	593	37.1	56.8	16.5	18.7	20.7-22.8	34.4-43.5	4.6-8	5.3-8.7
	704	26.9	42.2	12.1	19.8	19.2-19.7	20.3-20.5	1.1-1.9	1.3-2.2
	816	17.3	19.1	9.3	27.5	10.2	10.5	1.0	1.3

TABLE 45. EFFECT OF CARBON CONTENT AND HEAT TREATMENT ON TENSILE PROPERTIES OF INCOLOY 800 AFTER A FAST FLUENCE OF 1.1 TO 1.6 x 10^{21} N/CM^2 AT 704 C(170)

Carbon Content, wt %	Test Temperature, C	Condition	Yield Strength, 1000 psi	Ultimate Tensile Strength, 1000 psi	Elongation, percent Uniform	Elongation, percent Total
0.03	704	Annealed	17.1	18.7	1.0	5.4
0.04	"	"	21.0	24.6	3.5	4.6
0.07	"	"	19.7	32.2	4.6	5.1
0.03	704	Preaged[a]	33.3	35.0	1.7	2.0
0.04	"	"	21.2	28.3	3.2	3.7
0.07	"	"	17.2	36.8	8.6	9.6
0.03	816	Annealed	7.4	11.8	3.9	5.6
0.04	"	"	4.2	4.8	3.1	5.6
0.07	"	"	8.5	9.5	1.7	2.2
0.03	816	Preaged[a]	7.3	12.4	1.8	3.8
0.04	"	"	5.9	8.6	4.2	9.8
0.07	"	"	5.3	10.3	6.2	8.1

(a) 816 C for 96 hours.

TABLE 46. TENSILE DATA FOR IRRADIATED INCOLOY 800 FUEL CLADDING[a] (121)

Test Temperature, C	0.2 Percent Offset Yield Strength, 1000 psi		Ultimate Tensile Strength, 1000 psi		Total Elongation, percent	
	Unirradiated	Irradiated[b]	Unirradiated	Irradiated[b]	Unirradiated	Irradiated[b]
RT	37.0	55-63	83.7	98-102	45	17
593	24.6	26-34	70.2	48-57	41	9-14

(a) Specimens were irradiated at 400 C to a fast fluence of 4×10^{20} n/cm^2.
(b) Range of values for a number of tests.

TABLE 47. EFFECT OF IRRADIATION ON THE BURST PROPERTIES OF INCOLOY 800 TUBING[a] (170)

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 $1.1 \times 1.6 \times 10^{21}$ n/cm^2.

TABLE 48. COMPARISON OF THE EFFECTS OF A PREDOMINANTLY FAST FLUENCE AND A MIXED FLUENCE (FAST AND THERMAL) ON THE MECHANICAL PROPERTIES OF INCOLOY 800(171)

Test Temperature, C	Fast Fluence, 10^{20} n/cm^2	Yield Strength, 1000 psi		Tensile Strength, 1000 psi		Elongation, percent: Uniform		Elongation, percent: 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 ETR-II for 200 hours (fast fluence).
(b) Irradiated at 704 C in GETR for 630 hours (mixed fluence).

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 100.(143) These results show that irradiation tends to reduce the fatigue life of Incoloy 800.

Hardness. A fast fluence of 4 x 10^{20} n/cm^2 at 400 C increased the room-temperature hardness of Incoloy 800 from VHN 141 to 145 to VHN 165 to 179.(121)

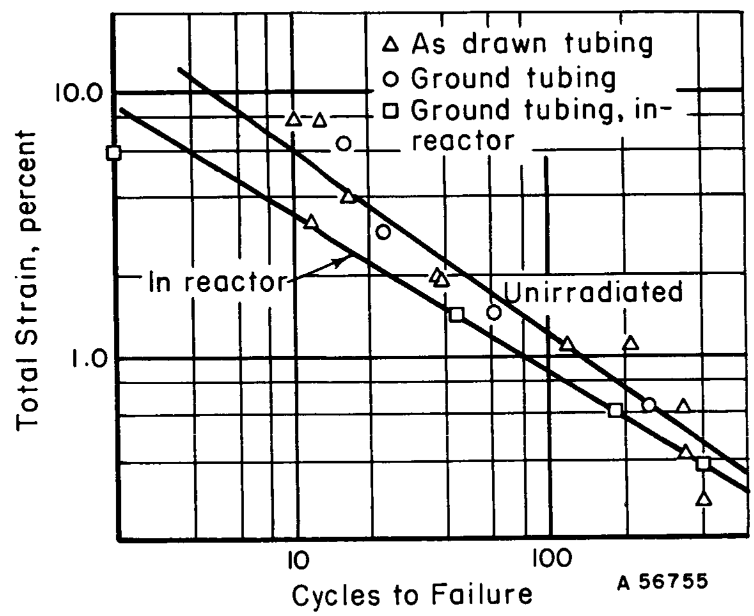


FIGURE 100. STRAIN FATIGUE LIFE OF INCOLOY 800 TUBING AT 704 C(144)

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

Predominantly Fast Fluence

Tensile Properties. Table 48 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. This finding is in conflict with the results that have been reported by the British for Type 316 stainless steel and Nimonic PE 16 alloy(158) (Fe-42.5Ni-17Cr-3.1Mo-1.2Al-1.3Ti).

Russian Stainless Steels

Mixed Fast and Thermal Fluence

Tensile Properties. The room-temperature tensile properties of irradiated Russian stainless steels are shown in Table 49.[172, 173] These results confirm the findings which are generally reported in that the yield and tensile strength are increased and the ductility is decreased. However, the room-temperature strength may be decreased and the ductility increased if the irradiation temperature is sufficiently high (~600 C). Testing of irradiated stainless steels at elevated temperatures was found to result in minor decreases in yield and ultimate strength, accompanied by a drastic reduction in ductility.[174]

Creep Properties. Figures 101 and 102 illustrate the effect of irradiation on the stress-rupture behavior of two different stainless steels.[174] It can be readily seen that irradiation to a fast fluence of 1 to 3 x 10^{20} n/cm^2 results in significant reductions in rupture life. These reductions in rupture life at elevated temperatures are attributed to helium from the (n,α) reactions in boron-10. However, it was not possible to verify this since increasing the boron content from 10 ppm to 150 ppm did not measurably affect the rupture life of the stainless steels.

TABLE 49. ROOM-TEMPERATURE TENSILE PROPERTIES OF IRRADIATED RUSSIAN STAINLESS STEELS(172, 173)

Material	Fast Fluence, n/cm^2 (>1 MeV)	Irradiation Temp, C	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Total Elongation, percent	
			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
Kh18N9T	3.0×10^{17}	80	34	52.5	92.5	102.5	66	65
Kh18N9T	1.4×10^{18}	80	34	62.5	92.5	105	65	55
Kh18N9T	2.4×10^{18}	300	37	59.5	101	104	64	50
Kh18N9T	4.4×10^{18}	600	45.5	38.5	108	108	62	70
Kh18N9T 25% CW	7.0×10^{19}	100	86.5	116.5	124	142	15	11
Kh18N9T	7.4×10^{19}	300	86.5	104	124	136	15	16
Kh18N9T	1.0×10^{20}	80	34	131	92	142	66	43
Kh18N9T	2.4×10^{20}		31.8	93.5	89	114	71	37
Kh18N9T	3.2×10^{20}		38.5	83.5	91.5	100.5	66.5	36.5
Kh18N12M2T	1.0×10^{18}	300	49.5	51	98	112	60	57
Kh18N12M2T	7.0×10^{18}	300	43	51	95	112	70	57
Kh18N12M2T	1.0×10^{19}	470	43	43	95	100	70	55
Kh18N12M2T	1.1×10^{19}	600	47	40	108	101	48	51
Kh18N12M2T	1.0×10^{20}	80	42.5	95	95	121	70	45
Kh18N22V2T2	7.0×10^{19}	300	81	78	131	122	29	26
Kh18N22V2T2	7.4×10^{19}	100	81	92	131	122	29	31
Kh18N22V2T2	2.2×10^{20}		81	116	131	120	29	24
Kh20N14S2	2.5×10^{17}		49	54.0	106.0	112.0	62	66
Kh20N14S2	1.2×10^{18}		49	71.0	106.0	117.0	62	65
Kh20N14S2	2.0×10^{19}		49	95.0	106.0		62	47
Kh20N14S2	4.4×10^{19}		49	106.0	106.0	127	62	40
Kh20N14S2	3.2×10^{20}		40	79	86	101	60.5	38.5
Kh18N14M3Nb	5.0×10^{19}	80	35.5	95	95	127	50	29
Kh18N14M3Nb	3.7×10^{18}	300	35.5	41	97	120	56	52
Kh20N404 (W + Nb)	2.5×10^{18}	300	59.5	71	117	122	39	28
Kh20N404 (W + Nb)	3.5×10^{18}	470	54	57	115	122	38	36
Kh20N404 (W + Nb)	4.4×10^{18}	600	62.5	71	114	124	39	35
Kh20N404 (W + Nb)	1.0×10^{20}	80	61	89.5	108	135	39	25
Kh18N9	2.4×10^{20}		29.5	73.5	94	95	76	56
Kh17N5M2	6.0×10^{19}	80	51.7	100	138	160	32	18.9
Kh17N5M2	6.0×10^{19}	180	51.7	88	138	152	32	24
Kh17N5M2	6.0×10^{19}	340	51.7	83.5	138	155	32	27.6

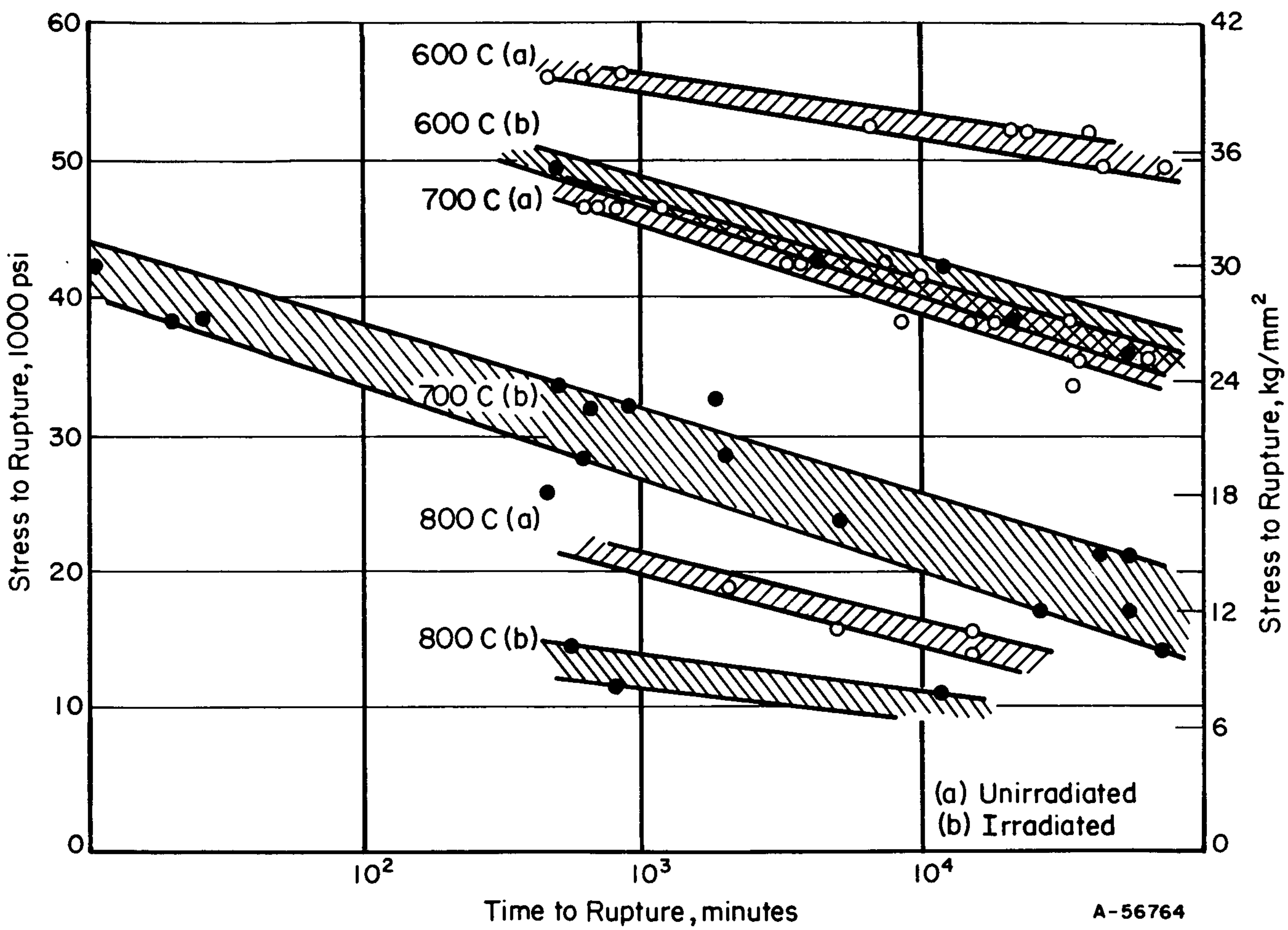


FIGURE 101. EFFECT OF IRRADIATION ON THE STRESS-RUPTURE PROPERTIES OF 18 Cr-22Ni-1.5Ti STAINLESS STEEL(174)

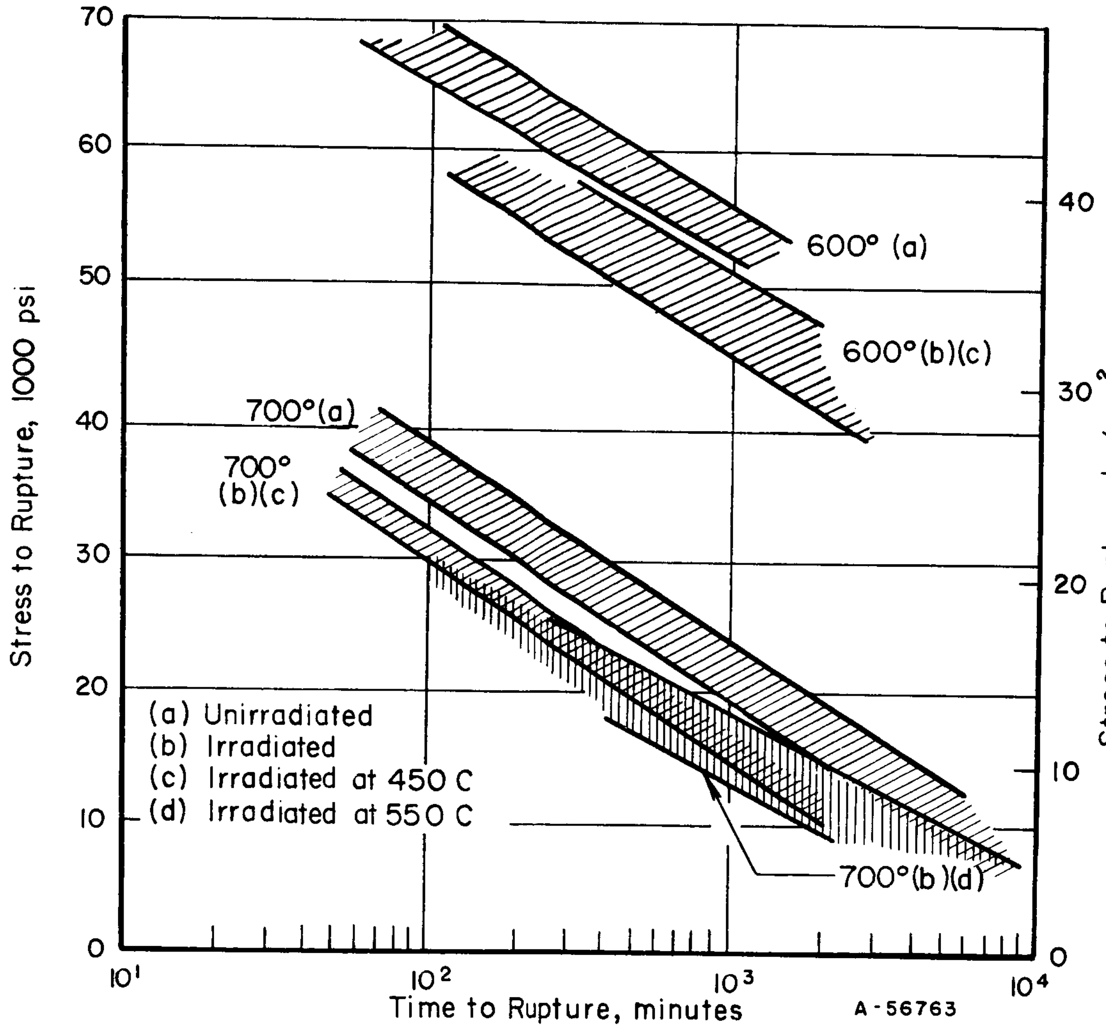


FIGURE 102. EFFECT OF IRRADIATION ON THE STRESS-RUPTURE PROPERTIES OF 18Cr-10Ni-0.5Ti-0.5Al STAINLESS STEEL(174)

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 cladding 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. Table 50 gives the composition of the nickel-base alloys that have been irradiated. These compositions should be used as only rough approximations since a large number of investigators use special heats and the composition limits of these alloys have wide variations. The nickel-base alloys are discussed in the following sequence:

(1) Nickel
(2) Hastelloy X-280
(3) Hastelloy N (INOR-8)
(4) Hastelloy C
(5) Inconel 600
(6) Inconel 625
(7) Inconel 702
(8) Inconel 718
(9) Incoloy 825
(10) Hastelloy R-235
(11) Inconel X
(12) René 41
(13) PDRL-120
(14) Nimonic alloys
(15) Russian nickel-base alloys

Since the main interest in nickel-base alloys is in their application in fast reactors, the irradiation effects of a purely fast fluence spectrum are treated separately from the more common mixed thermal and fast fluence, provided data were available.

Nickel

Mixed Thermal and Fast Fluence

Tensile Properties. The effects of irradiation on the room-temperature tensile properties of nickel are shown in Table 51.[175, 176] Irradiation increases both the yield and ultimate strength and causes significant reductions in ductility. The irradiation-induced increase in the yield strength for nickel has been shown to depend on the cube

TABLE 50. APPROXIMATE COMPOSITION OF NICKEL ALLOYS

Materials	Composition, weight percent												
	Ni	Cr	Fe	Mo	Co	W	Nb	Ti	Al	C(a)	Mn(a)	Si(a)	Other
Nickel A	99.5		(b)		(b)						(b)	(b)	
Nickel 330	99.7				(b)					(b)			
Hastelloy C	56	15	6	15	2	3.5				0.08	1.0	1.0	0.35V
Hastelloy N	71	7	5(a)	16						0.08	0.8		
Hastelloy X-280	48	22	18	9	0.5(a)	0.6				0.15	1.0	1.0	
Inconel 600	73	15	7				2.25			0.10	1.0	0.75	0.50Cu
Inconel 625	61	22	2	9			4.0			0.10	1.0	0.75	
Inconel 702	78	15	2(a)					0.6	3.2	0.10	1.0		
Inconel 718	50	19	20	3			5	0.8	0.6	0.10	0.50	0.75	0.75Cu
Inconel 825	42	21	29	3				0.9	0.2(a)	0.05	1.0	0.5	2.2Cu
Hastelloy R-235	62	15	10	5.5	2.5(a)			2.5	2.0	0.16	0.25	0.6	
PDRL-120	56	21		4	14		2	2.5	0.25	0.10			
René 41	44	18	2	10	11			3.2	1.5	0.10	0.5	0.5	
Nimonic PE-16	42	17	35	3				1.3	1.2	0.08	0.5	0.5	
Nimonic 80A	72	20			2			2.3	1.3	0.10	1.0	1.0	
Inconel X	73	15	7				0.8	2.6	0.9	0.05	0.6	0.3	

(a) Maximum.
(b) Contains minor amounts of element.

TABLE 51. 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	175
Nickel A	2.4(b)	16.6	82.8	56	85.7	45	23	175
Nickel A	4.0(b)	17.1	91.2	62.5	106	33	17	175
Nickel A	4.0(b)	16.6	61.2	56	89.0	34	13	175
Nickel 330(c)	7.8	52.5	108.6	66.5	108.6			176
Nickel 330(d)	7.8	80.5	146.0	82.3	146.0	8	3	176
Nickel 330(c)	7.8	53.2	149.0	84.1	149.0			176
Nickel 330	9.0	45.8	95.6	63.4	95.6	36	10	176
Nickel 330	9.0	45.8	98.6	63.4	98.6	36	11	176
Nickel 330(e)	9.0	37.3	98.0	61.8	98.0	30	7	176
Nickel 330(c)	9.0	40.0	104	63.0	104.0			176
Nickel 330	9.0	43.0	118	82.0	118.0			176
Nickel 330(c, e)	11.3	49.6	118	84.7	129.8			176
Nickel 330(c)	11.3	80.0	163	97.4	163			176
Nickel 330	11.3	67.2	145	87.5	145	34	5	176
Nickel 330	11.3	45.8	101.3	63.4	101.3	36	8	176

(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.

root of fast fluence in the range of 1 x 10^{17} to 1.1 x 10^{20} n/cm^2; this dependence is illustrated in Figure 103.(177)

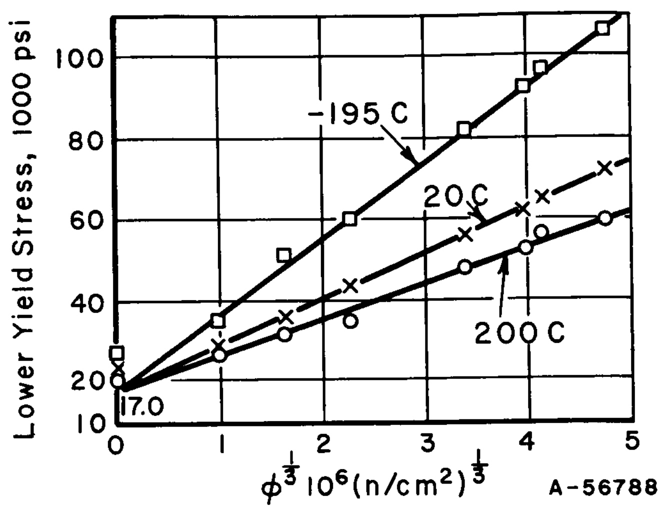
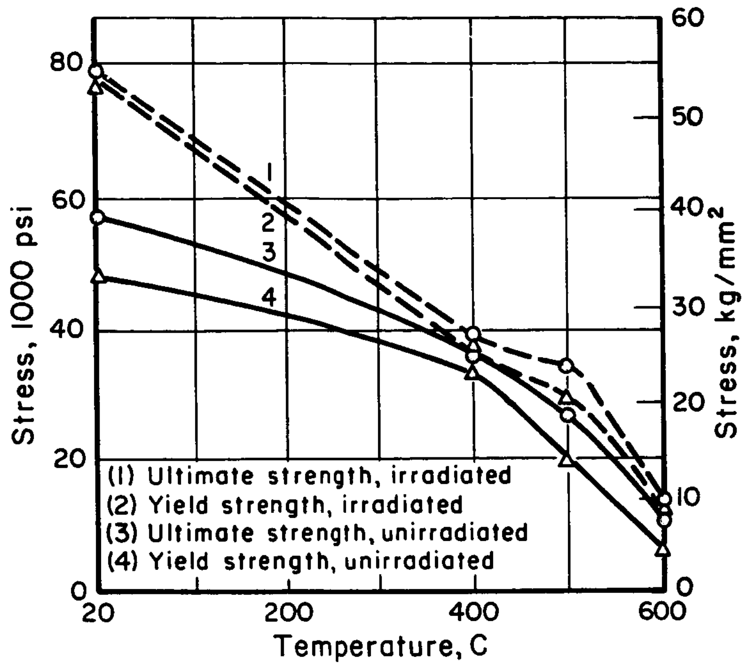


FIGURE 103. DEPENDENCE OF IRRADIATION-INDUCED YIELD-STRENGTH INCREASE ON FAST FLUENCE(177)

ϕ = fast fluence.

The ductility of irradiated nickel appears to be drastically reduced at elevated temperatures. For example, a fast fluence of 7 x 10^{20} n/cm^2 was found to reduce the total elongation at 700 C from 3 to 2 percent and that at 880 C from 9 to 2 percent.(69) Although the preirradiation ductility was quite low, these appear to be reasonable values since the Russians found similar drastic ductility reductions. The Russians irradiated 99.95 percent pure nickel at 150 to 200 C to a fast fluence of 1.7 x 10^{20} n/cm^2. Figure 104 shows that a drastic ductility reduction takes place between 400 and 500 C, and at 600 C, the ductility is almost zero. Figure 104 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 105.(174) The irradiation damage (as measured by hardness increase) is not annealed out until about 700 C.



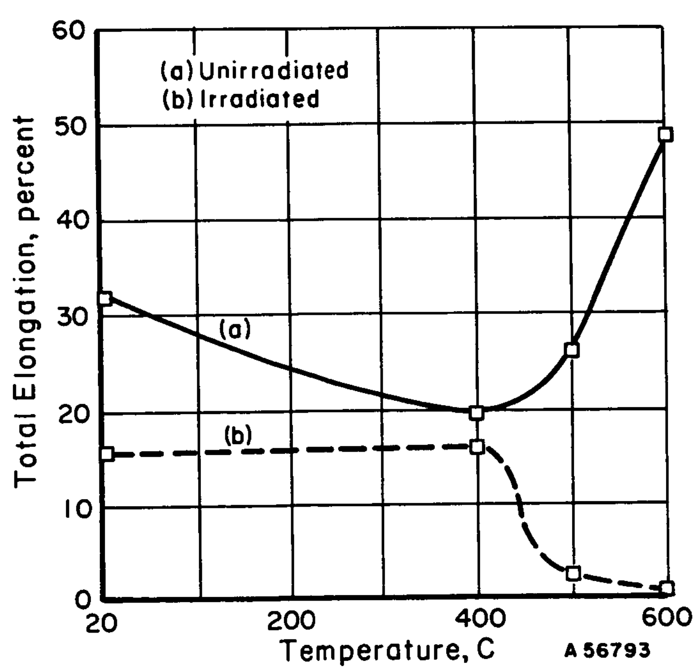


FIGURE 104. EFFECT OF IRRADIATION (1.7×10^{20} N/CM^2 ON THE DUCTILITY AND STRENGTH OF 99.95 PERCENT PURE NICKEL[174]

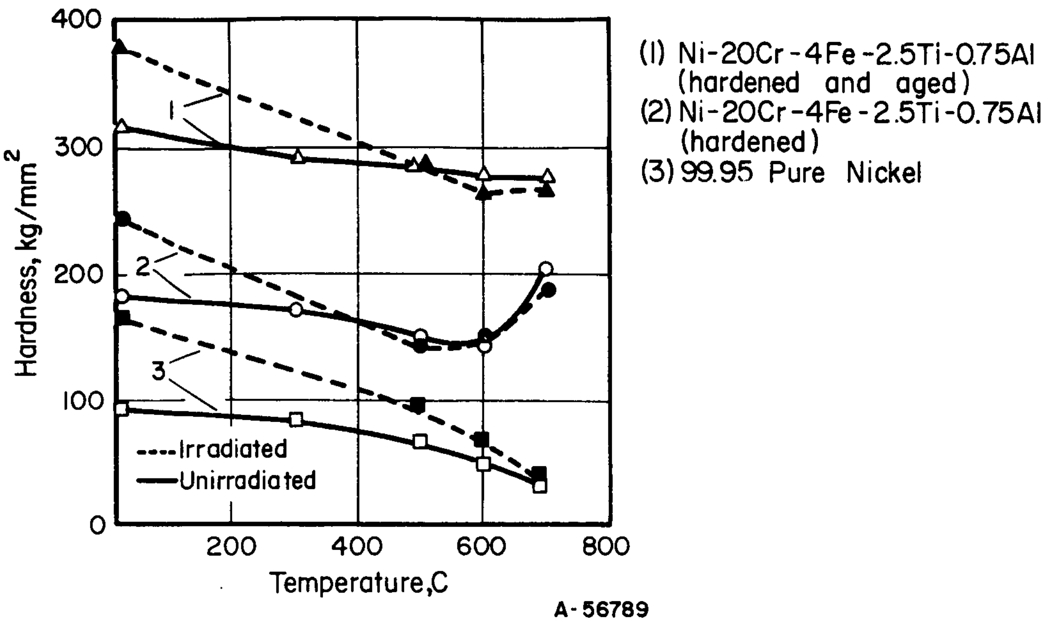


FIGURE 105. EFFECT OF IRRADIATION (1.7 x 10^{20} N/CM^2) ON THE HOT HARDNESS OF NICKEL AND NICKEL ALLOYS[174]

Hastelloy X and Hastelloy X-280

Mixed Fast and Thermal Fluence

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 52.[121, 168, 178-180] 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 drastic 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 drastically 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

TABLE 52. 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	178
280	RT	5×10^{19}	--	71	--	117	--	47	--	--	178
280	RT	7×10^{19}	--	74	--	119	--	46	--	--	178
400	RT	4×10^{20}	59.2	74.2	--	--	--	--	41.8	45.3	121
650	RT	3×10^{19}	53	46	118	105	--	--	30	25	179
650	RT	3×10^{19}	53	38	118	109	--	--	30	34	179
740	300	9×10^{19}	47.2	45.8	107.9	109.7	23.5	18.6	26.0	20.9	168
740	300	2.2×10^{20}	47.6	40.7	111	105.9	21.4	21.7	23.2	22.4	168
740	300	2.4×10^{20}	47.6	41.2	111	98.9	21.4	13.7	23.4	14.3	168
400	593	4×10^{20}	42.5	41.5	--	--	--	--	44.9	13.7	121
280	650	1×10^{20}	--	26	--	49	--	4.8	--	4.8	178
280	650	1×10^{20}	--	26	--	49	--	10.7	--	13	178
650	593	2×10^{21}	56	59	111	93	20	4	28	5	180
650	700	3×10^{19}	35	23	71	87	--	--	48	24	179
740	650	9×10^{19}	40.8	32.5	90.1	62.6	16.7	5.6	52	6.2	168
740	650	2.2×10^{20}	40.0	34.7	85.5	67.4	15.0	5.7	43.5	6.1	168
740	650	4×10^{20}	40.5	37	90.7	67.7	19.6	4.7	40.3	7.4	168

effects of elevated-temperature irradiations it must be remembered that the tensile properties are significantly changed by the aging at the irradiation temperature.[168] Unfortunately some investigators have not measured this aging by having the unirradiated controls at the temperature of irradiation for the identical time of the irradiation.

Attempts have been made to improve the tensile properties of irradiated Hastelloy X by giving the material various preirradiation thermo-mechanical treatments. These treatments consisted of solution annealing at 1177 C for 1 hour, air cooling, and then cold working it from 5 to 20 percent, followed by a 24-hour anneal at either 538 C or 816 C.[181] The purpose of these treatments is to furnish dislocation lines throughout the matrix where second-phase particles could precipitate and thus possibly influence the irradiation response of Hastelloy X. Figure 106 shows the effect of irradiation at 280 C, to a fast fluence of 1×10^{20} n/cm^2, on the mechanical properties at 732 C of specimens which had received various thermo-mechanical treatments. It can be seen that considerable increase in postirradiation strength can be obtained with these treatments, although the ductility is considerably reduced. The improvement in strength may be especially significant since irradiation at 740 C and above has resulted in considerable loss of strength.[168] It remains to be seen how the thermo-mechanically heated Hastelloy X specimens perform after being irradiated at higher temperatures.

Tensile tests have been performed on Hastelloy X cladding after the fuel was removed. These claddings were irradiated to a fast fluence of 2.6×10^{18} n/cm^2[182] at temperatures up to 815 C. Results of the tensile tests on tubular specimens are given in Table 53. It should be noted that these specimens exhibited a very low ductility at elevated temperature, considering the rather low fast fluences.

TABLE 53. MECHANICAL PROPERTIES OF IRRADIATED TUBULAR HASTELLOY X SPECIMENS[a][182]

Temperature, C	Yield Strength, 1000 psi	Ultimate Strength, 1000 psi	Total Elongation, percent
RT	61.8	113	7.6
RT	83.0	118	2.8
RT	63.0	--	--
RT	78.4	--	--
815	25.8	30.3	2.4
815	25.5	26.3	0.9
815	32.5	33.0	0.8
955	15.5	16.5	0.8
955	16.2	16.5	1.0
955	11.4	11.4	0.3
1010	5.6	5.6	0.9
1010	7.4	7.4	0.8
1010	8.8	8.8	0.4

(a) Irradiated at 815 C to a fast fluence of 2.6×10^{18} n/cm^2.

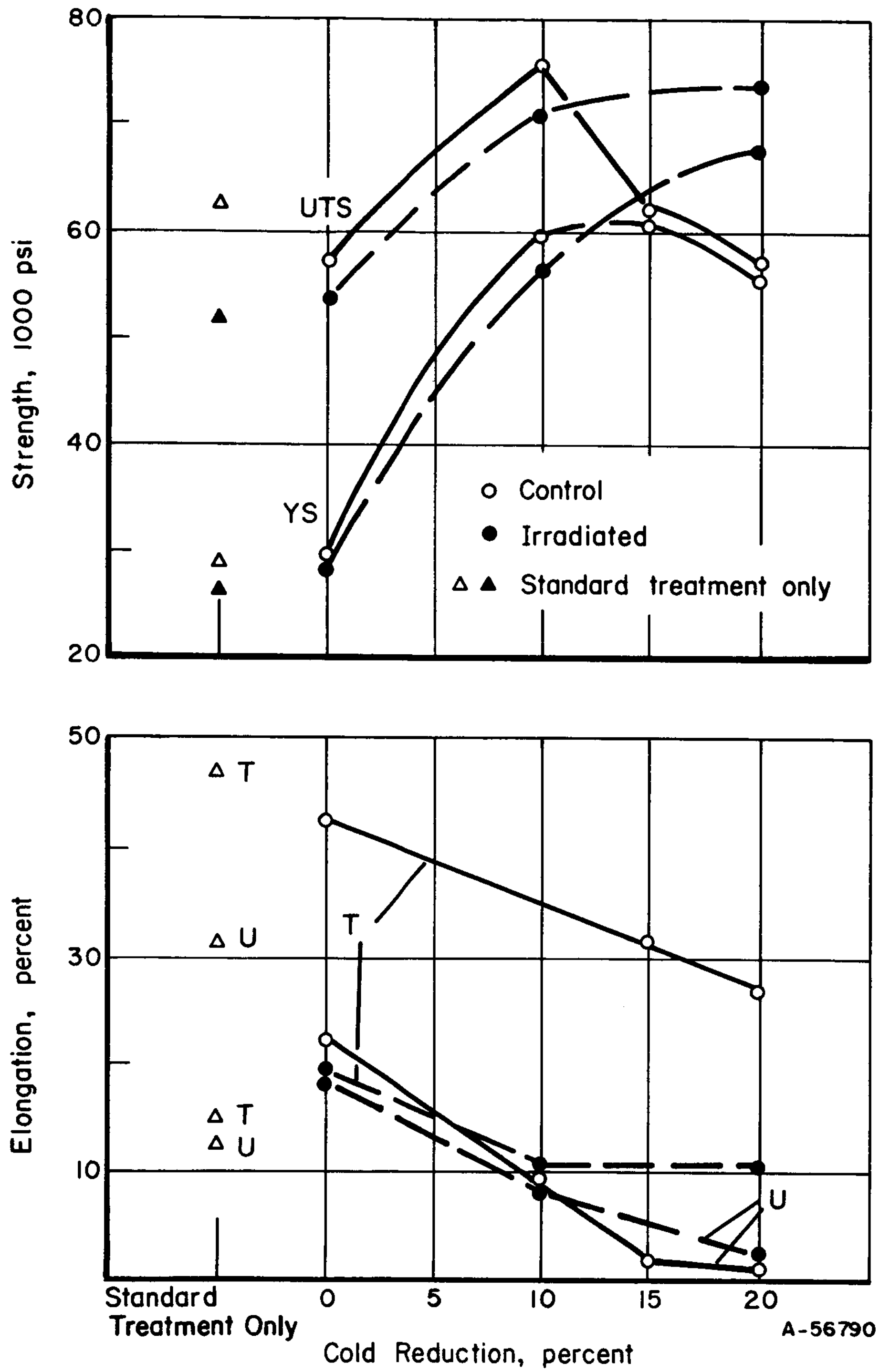


FIGURE 106. EFFECT OF IRRADIATION ON THE 732 C TENSILE PROPERTIES OF HASTELLOY X-280[181]

Standard Treatment:	1177 C/1 hour, air cooled
Experimental Treatment:	1177 C/1 hour, air cooled, cold rolled
	538 C/24 hours, air cooled

Creep Properties. The effect of irradiation on the 540 and 650 C stress-rupture properties of Hastelloy X are shown in Figure 107.[183] 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.

The effect of thermo-mechanical treatments on postirradiation stress-rupture life of Hastelloy X have also been studied. The stress-rupture specimens were given thermo-mechanical treatments similar to those given to the tensile specimens discussed above.[181] The effect of irradiation on these specially treated specimens is shown in Figure 108. It appears that the 24-hour anneal at 816 C after cold working is effective in increasing postirradiation rupture life, whereas the anneal at 538 C results in a lower rupture life than that obtained by the standard treatment. The improvement in rupture and creep properties is attributed to improved grain-boundary ductility.[184]

Hardness. The effect of irradiation at 280 C on the hardness of Hastelloy X is given in Table 54.[68]

TABLE 54. HARDNESS OF UNIRRADIATED AND IRRADIATED HASTELLOY X AT ROOM TEMPERATURE[68]

Fast Fluence, n/cm^2	Hardness, Rockwell A	
	Unirradiated[a]	Irradiated
5.3×10^{19}	11.4	61.8
7.7×10^{19}	11.8	61.2
3.5×10^{20}	11.2	60.6

(a) Measurements based on Rockwell C.

Predominantly Fast Fluence

Tensile Tests. Results of limited tensile tests performed on Hastelloy X specimens irradiated in a predominantly fast flux in EBR-II are given in Table 55.[180] Comparison of these results with those in Table 52 indicates that irradiation in a predominantly fast fluence results in changes in mechanical properties similar to those induced by irradiation in a mixed thermal and fast fluence.

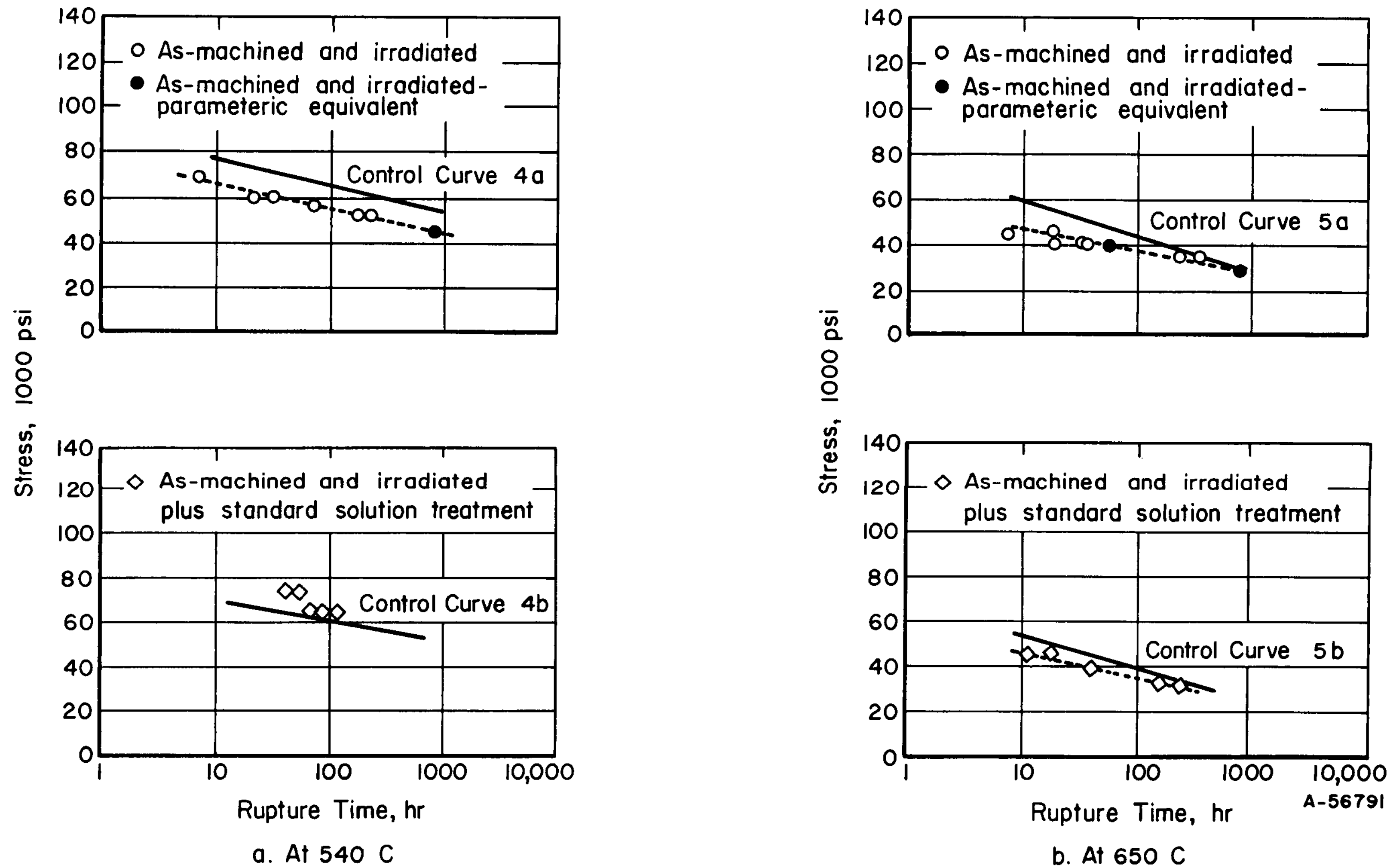


FIGURE 107. STRESS-RUPTURE STRENGTH OF HASTELLOY X ALLOY IRRADIATED SPECIMENS FROM HEAT E-9500[183]

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

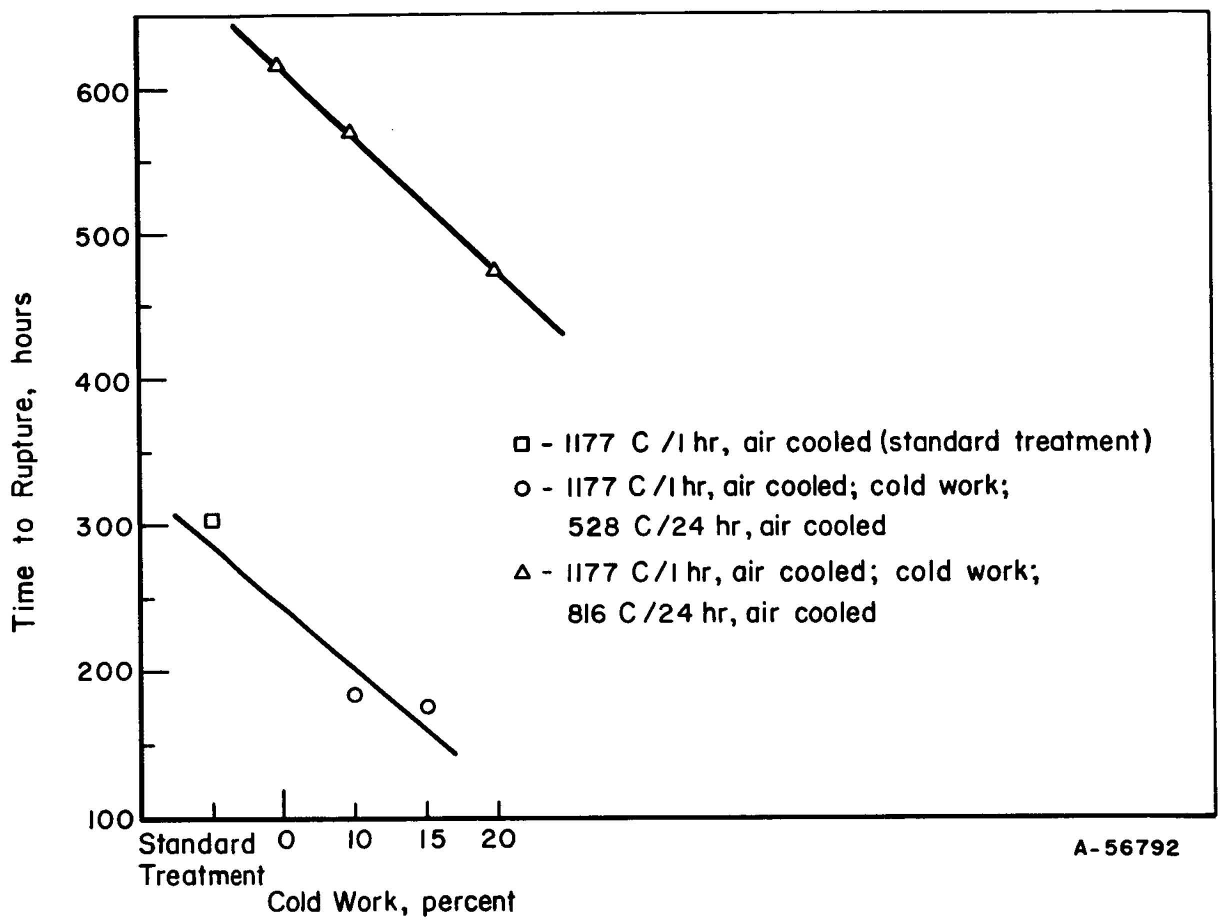


FIGURE 108. EFFECT OF PREIRRADIATION TREATMENTS ON POST-IRRADIATION TIME-TO-RUPTURE FOR HASTELLOY X-280 AT 732 C UNDER AN ORIGINAL STRESS OF 18,000 PSI[181]

Irradiated at 280 C to a fast fluence of approximately 1×10^{20} n/cm^2 (E $\leq$ 1 MeV).

TABLE 55. EFFECT OF IRRADIATION IN A PREDOMINANTLY FAST FLUENCE ON THE MECHANICAL PROPERTIES OF HASTELLOY X[a](180)

Test Temperature, C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent			
					Uniform		Total	
	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
RT	43.5	65.3	99.3	137	50.1	15.5	54.1	16.0
593	26.5	52.3	76.0	102.9	54.5	14.4	57.3	14.7
704	26.7	48.3	61.7	62.0	40.9	5.4	49.8	8.4

(a) Irradiated in EBR-II at 538 C to a fast fluence of 3.3×10^{20} n/cm^2.

Hastelloy N (INOR-8)

Mixed Thermal and Fast Fluence

Tensile Properties. The postirradiation tensile properties of Hastelloy N are given in Tables 56(168, 185-187) and 57(188). The data show that irradiation at low temperatures results in increased strength and decreased ductility when the 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.

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 58.(188)

Composition. The addition of zirconium has 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 weight percent zirconium were found to be 3.6, 7.5, and 11.5 percent, respectively.(189) 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 59.(190) The data indicate that annealing for 1 hour at 400, 650, and 871 C causes further decreases in the 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.

TABLE 56. POSTIRRADIATION TENSILE PROPERTIES OF HASTELLOY N(168, 185-187)

Irradiation Temp, C	Test Temp, C	Fast Fluence, n/cm^2	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reduction in Area, percent	
							Uniform		Total			
			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 57. TENSILE STRENGTH AND DUCTILITY OF IRRADIATED AND UNIRRADIATED HASTELLOY N[a](188)

Deformation Temperature, C	Strength, 1000 psi				Ductility, percent			
	Yield		True Tensile		True Uniform Strain		True Fracture Strain	
	Irrad.	Unirrad.	Irrad.	Unirrad.	Irrad.	Unirrad.	Irrad.	Unirrad.
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.

TABLE 58. STRAIN RATE SENSITIVITY OF IRRADIATED AND UNIRRADIATED HASTELLOY N[a](188)

Deformation Temperature, C	Strain Rate, min^{-1}	Strength, 1000 psi				Ductility, percent			
		Yield		True Tensile		True Uniform Strain		True Fracture Strain	
		Irrad.	Unirrad.	Irrad.	Unirrad.	Irrad.	Unirrad.	Irrad.	Unirrad.
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.

TABLE 59. EFFECT OF POSTIRRADIATION ANNEALING ON THE TENSILE PROPERTIES OF HASTELLOY N AT 650 C[(a, b)][(190)]

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 = 8.5×10^{20} n/cm^2.
(b) Strain rate of 0.002 in./in./minute.

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 109[(189)] and 110[(190, 191)] illustrate the effect of irradiation on the stress-rupture properties of Hastelloy N at 650 C. The effects at 760 C are shown in Figure 111.[(192)] 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 appears to diminish (Figure 109). The decrease in rupture life is due to drastic irradiation-induced reductions in ductility; however, the actual creep rate does not appear to be significantly affected by irradiation.[(191)] The drastic ductility reductions are attributed to formation of helium bubbles at the grain boundaries. The helium is supposedly formed by (n, α) reactions from the boron-10 impurity, and good correlation between the helium content and loss of ductility have been obtained (Figure 112). Figure 110 illustrates that the rupture life is decreased with increasing fast fluence, which could mean that more helium is produced at the higher fast fluence levels. In all of these irradiations, the fast and thermal fluence levels were approximately the same.

It has also been found that additions of titanium improve the rupture life and the ductility at fracture as shown in Figure 113.[(189)] Additions of zirconium increase the ductility of irradiated Hastelloy N at high strain rates (tensile tests), but have no effect on the ductility in creep tests, as illustrated in Figure 114.[(189)]

Hardness. See Table 60.

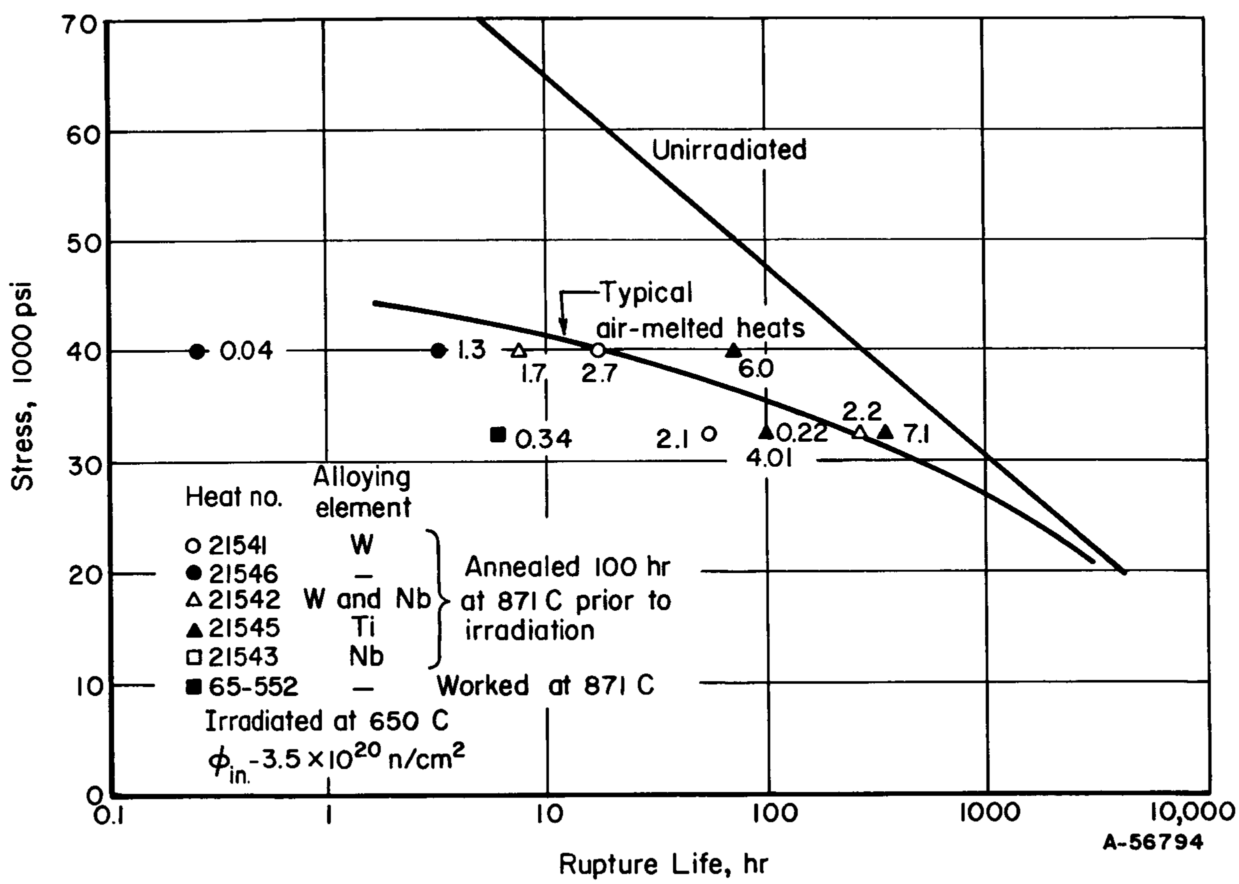


FIGURE 109. POSTIRRADIATION CREEP-RUPTURE OF SEVERAL HASTELLOY-N HEATS AT 650 C(189)

Numbers indicate fracture elongations.

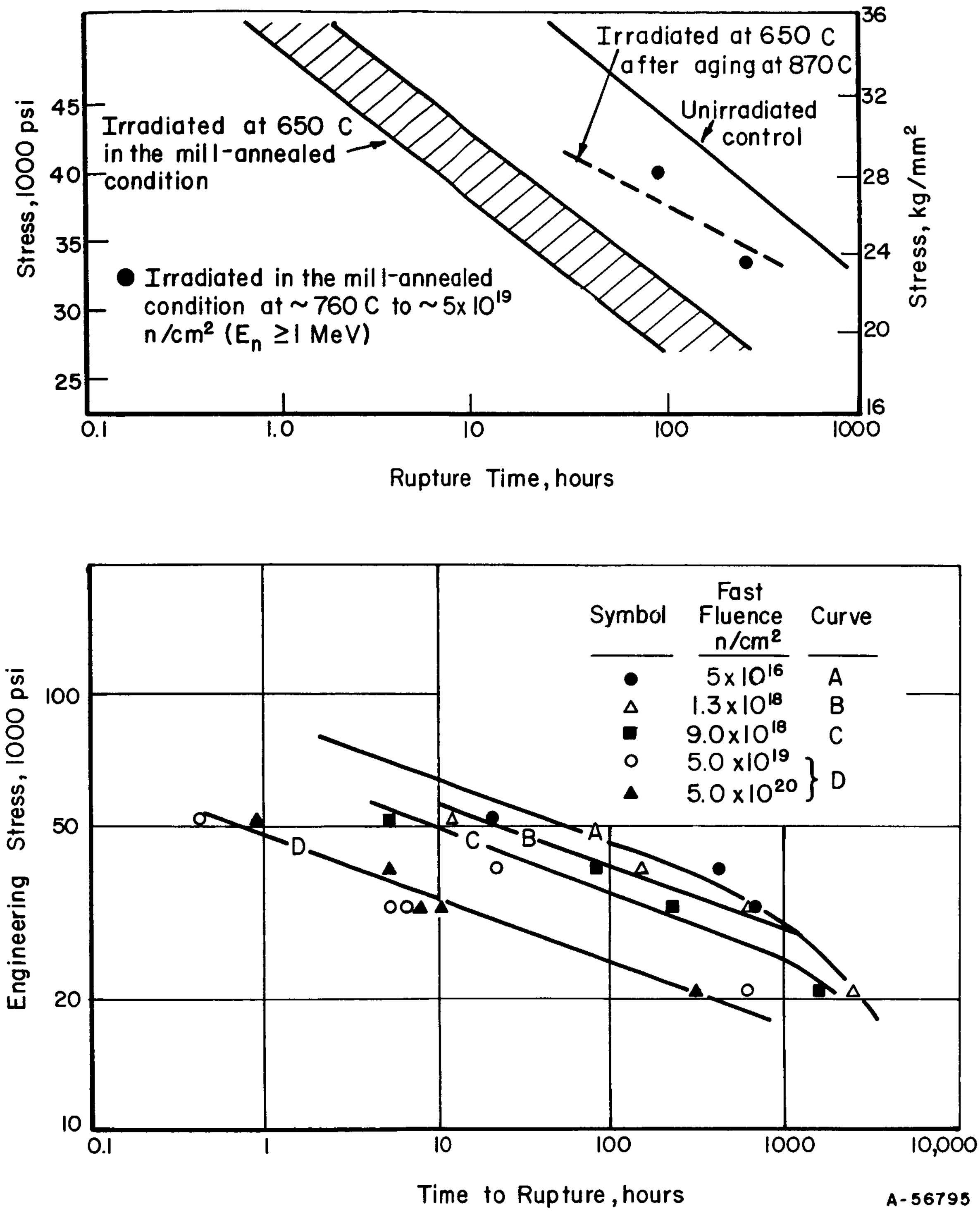


FIGURE 110. POSTIRRADIATION STRESS RUPTURE OF HASTELLOY N AT 650 C(191)

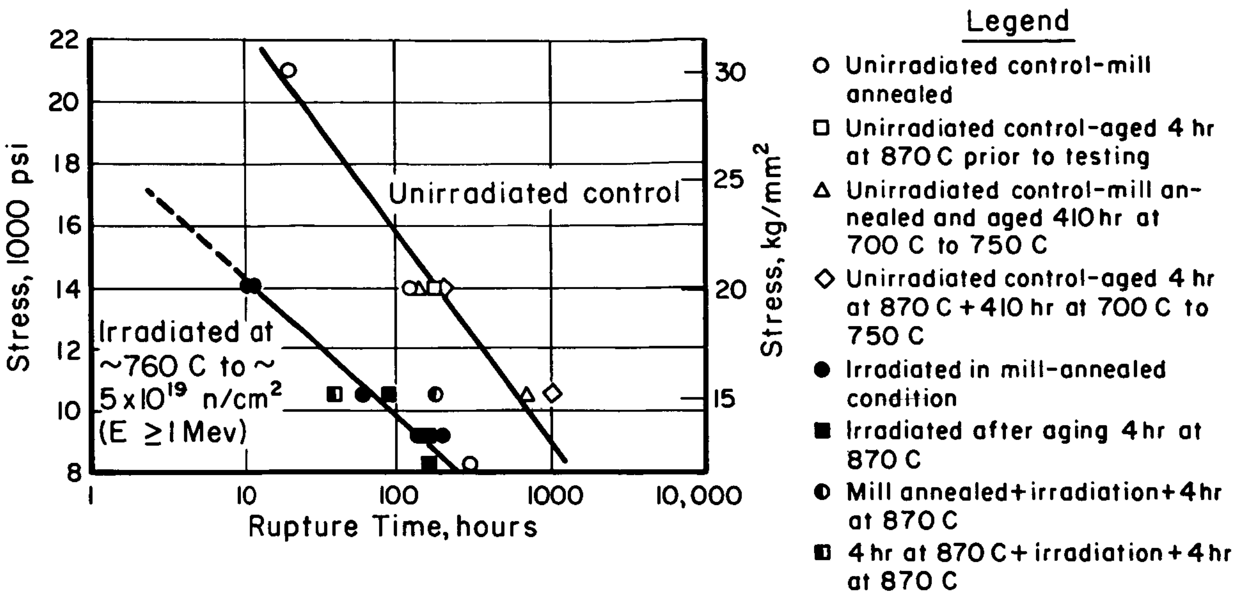


FIGURE 111. STRESS-RUPTURE PROPERTIES OF IRRADIATED AND UNIRRADIATED HASTELLOY N AT 760 C[192]

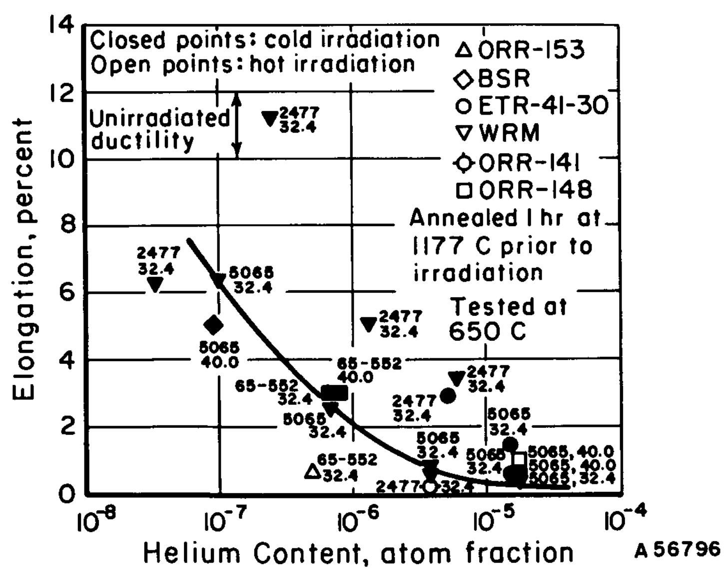


FIGURE 112. VARIATION OF POSTIRRADIATION CREEP DUCTILITY OF HASTELLOY N WITH HELIUM CONTENT[189]

The top numbers indicate different heats, while the bottom number gives the stress in 1000 psi.

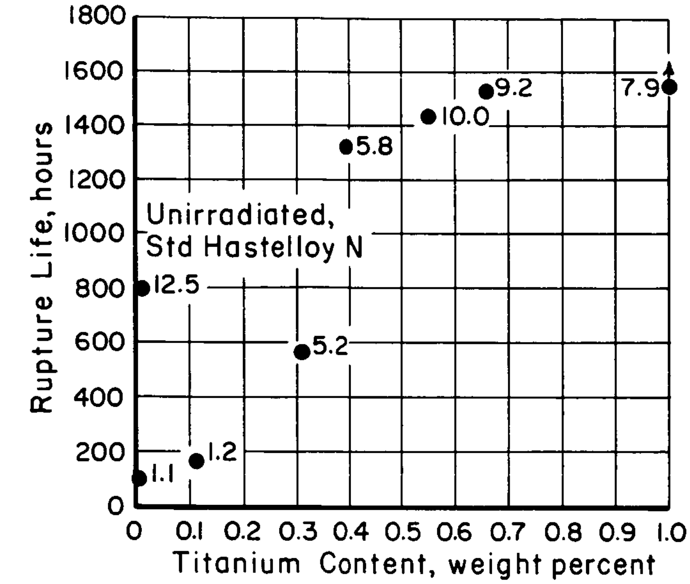


FIGURE 113. INFLUENCE OF TITANIUM ON THE POSTIRRADIATION CREEP PROPERTIES OF Ni-12Mo-7Cr-0.05C(189)

Numbers indicate fracture elongations. Annealed 1 hour at 1177 C prior to irradiation. Irradiation temperature 650 C, thermal fluence 2.5 x 10^{20} n/cm^2. Test temperature 650 C, stress 32,350 psi.

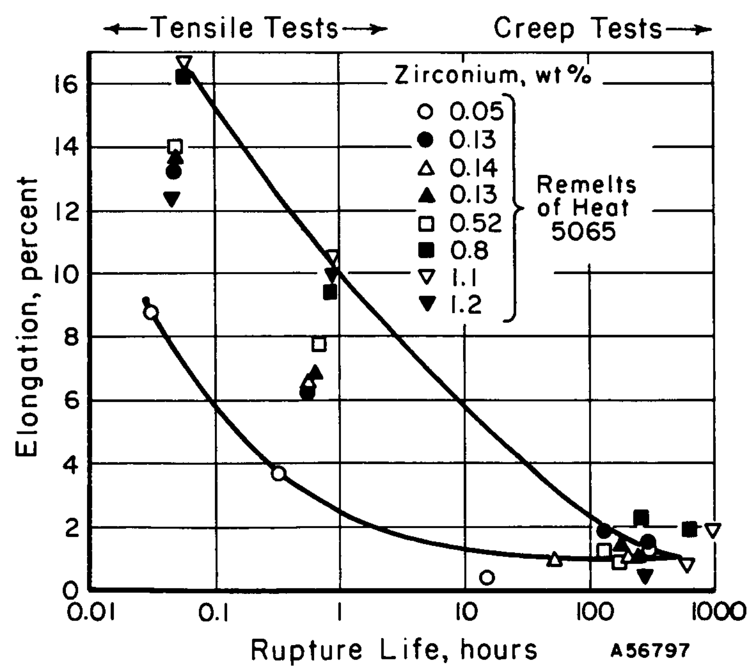


FIGURE 114. INFLUENCE OF ZIRCONIUM CONTENT ON THE POSTIRRADIATION DUCTILITY OF HASTELLOY N(189)

Thermal fluence 2 x 10^{20} n/cm^2. Irradiation temperature and test temperature 650 C.

TABLE 60. EFFECT OF IRRADIATION ON ROOM-TEMPERATURE HARDNESS OF HASTELLOY N[68]

Fast Fluence, n/cm^2	Irradiation Temp, C	Hardness, Rockwell A Unirradiated	Hardness, Rockwell A Irradiated
1.8×10^{20}	50	50.3	63.3
5.3×10^{19}	280	50.4	60.0
7.7×10^{19}	280	50.6	60.0
3.5×10^{20}	280	50.5	61.3

Hastelloy C

Mixed Thermal and Fast Fluence

Tensile Properties. Only limited tensile tests have been performed on irradiated Hastelloy C. The results are given in Table 61.[168] The irradiation at 740 C does not appear to affect the tensile properties at room temperature and 300 C. However, irradiation causes a drastic reduction in ductility and significant reductions in strength at a test temperature of 650 C.

TABLE 61. EFFECT OF IRRADIATION AT 740 C ON THE TENSILE PROPERTIES OF HASTELLOY C[168]

Fast Fluence, 10^{20} n/cm^2	Test Temp, C	0.2% Offset Yield Strength, 1000 psi Unirr.	0.2% Offset Yield Strength, 1000 psi Irr.	Ultimate Strength, 1000 psi Unirr.	Ultimate 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.
0.9	20	79.8	88.9	145.8	157.7	7.9	8.4	10.0	8.5	17.2	8.8
2.2	20	84.3	86.9	157.6	175.0	10.4	14.7	10.7	14.9	8.1	9.1
2.4	20	84.7	79.3	159	157.5	10.1	12.0	10.2	12.3	8.7	6.0
0.9	300	75.3	86.5	146.5	125.1	9.2	--	9.6	--	8.1	8.8
2.2	300	77.3	70.9	146.4	136.6	8.7	7.8	8.7	8.4	8.4	7.2
2.4	300	78.2	70.0	152.5	139.9	8.7	8.3	9.6	9.5	8.8	11.7
0.9	650	71.1	52.8	129.2	80.6	14.5	4.0	21.3	4.5	15.5	14.5
2.2	650	68.1	61.5	122.5	90.8	15.9	2.4	32.3	2.6	29.7	8.2
2.4	650	--	55.1	124.0	84.8	17.7	3.9	18.2	4.4	20.4	4.7
4.0	650		61.9		86.2		2.5		2.8		5.5

Inconel 600

Mixed Thermal and Fast Fluence

Tensile Properties. The results of postirradiation tensile tests on Inconel 600 are summarized in Table 62. These results show that if the irradiation temperature is

TABLE 62. 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			157
RT	650	1.2×10^{20}	36	34	94	94			42	37			157
RT	400	4×10^{20}	45.8	61.2					37.2	35.8			157
RT(a)	400	4×10^{20}	36.5	55-63	89.7	98-102			43	17-38			157
RT	175	6×10^{20}	45.0	105.5	93.8	118.8	41.3	25.5	48.7	31.6			169
RT	175	1.2×10^{21}	45.0	135.6	93.8	136.1	41.3	3.4	48.7	17.9			169
RT	175	2.4×10^{21}	45.0	144.7	93.8	146.4	41.3	3.8	48.7	11.4			169
RT	175	2.5×10^{21}	45.0	145.7	93.8	146.7	41.3	3.4	48.7	10.6			169
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	168
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	168
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	168
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	168
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	168
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	168
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	168
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	168
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	168
300(b)	740	4×10^{20}		31.9		61.0		16.4		17.5		32.8	168
300(d)	740	4×10^{20}		37.7		88.9		32.5		34.7		34.7	168
315(a)	400	4×10^{20}	33.2	47	87.7	72			47.0	16			121
315	175	6×10^{20}		85.2		102.3		26.6		29.8			169
315	175	1.2×10^{21}		106.7		109.8		3.2		19.4			169
315	175	1.9×10^{21}		120		120.1		3.3		11.5			169
315	175	2.2×10^{21}		119		119		3.6		9.7			169
595	400	4×10^{20}	35.0	33.7					33.0	6.0			157
595(a)	400	4×10^{20}	30.1	15-35	76.5	23-40			35.0	2.7-3.2			157
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	168
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	168
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	168
705(a)	400	4×10^{20}		14		14				0.2			157

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

below 200 C, then considerable increases in strength and decreases in ductility take place.(169) 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. The effects of irradiation temperature on the mechanical properties of Inconel 600 are illustrated in Figure 115.(168)

Cold-worked Inconel 600 was found to exhibit somewhat higher postirradiation strength at 650 C than did mill- and solution-annealed alloys, all of which had been irradiated at 280 C.(168) However, the postirradiation ductility of all Inconel 600 specimens, which had received different preirradiation treatments, was drastically reduced when the material was tested at 650 C.

The effect of irradiation on welded Inconel 600 is illustrated in Table 63.(140) These specimens were irradiated at 50 C to a fast fluence of 2.3×10^{18} n/cm^2 with the thermal fluence being 6.7×10^{19} n/cm^2. These results indicate that irradiation does not change the strength of Inconel 600 at temperatures in the 500 to 900 C range. No measurable irradiation-induced changes take place in uniform elongation, but irradiation causes a drastic change in the necking elongation (i. e., total elongation-uniform

TABLE 63. COMPARISON OF STRENGTH AND DUCTILITY FOR UNIRRADIATED AND IRRADIATED INCONEL 600 WELD AND WROUGHT METAL(a)(140)

		Strength, 1000 psi				Ductility, percent			
		0.2% Offset Yield		Ultimate Tensile		True Uniform Strain		Total Elongation	
Material	Deformation Temp, C	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
Weld metal	500	50.5	51.2	78.0	81.4	29.6	32.1	37.3	41.6
	600	47.9	48.5	70.6	70.0	24.5	23.6	30.3	28.5
	700	45.0	44.3	55.7	52.6	10.9	5.8	25.2	6.4
	800	33.8	39.6	34.2	40.5	1.4	1.6	32.3	6.5
	900	20.4	24.4	20.5	24.6	0.9	1.1	43.2	4.7
Wrought metal	500	25.8	40.8	75.0	84.0	32.8	36.0	41.2	45.3
	600	29.4	32.5	81.0	68.0	29.8	24.6	44.9	28.1
	700	24.4	25.2	52.4	44.5	16.5	9.7	42.6	10.4
	800	21.2	23.2	22.5	27.3	4.3	3.3	71.0	4.3
	900	12.1	13.0	12.5	13.3	9.4	1.2	48.5	1.8
Joints	500	37.2	47.1	83.5	82.3	34.1	30.8	43.6	38.0
	600	37.0	41.0	74.8	73.2	27.4	23.6	34.0	27.8
	700	45.5	36.2	70.8	50.0	10.9	5.2	32.2	5.3
	800	26.4	30.8	27.2	32.0	1.6	1.8	33.9	3.0
	900	15.4	16.0	15.7	16.3	1.4	1.0	39.8	1.6

(a) Irradiated to a fast fluence of 2.3×10^{18} n/cm^2.

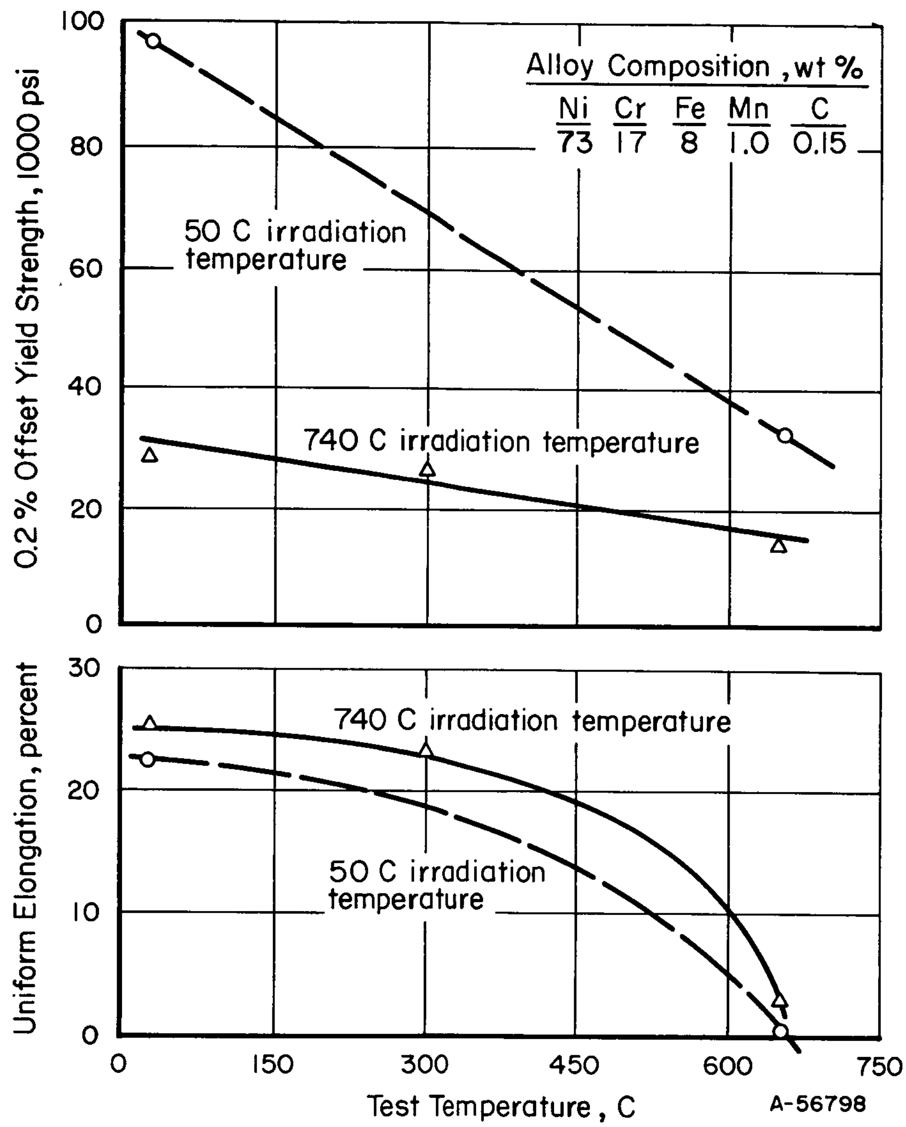


FIGURE 115. EFFECT OF TEST TEMPERATURE ON TENSILE PROPERTIES OF INCONEL 600 IRRADIATED TO A FAST FLUENCE OF $\sim 1 \times 10^{20}$ N/CM^2 (E > 1 MeV)(168)

elongation) at testing temperatures of 500 to 900 C. Table 64 illustrates the irradiation-induced deterioration of weld ductility. Up to 600 C there is relatively little difference between the unirradiated and irradiated weld ductility, but at 700 C, the irradiated weld has only 28 percent of the unirradiated weld ductility.(140)

The effect of irradiation on the shear strength of Inconel 600 brazed with Ni-13Cr-10P is shown in Table 65. At temperatures up to 600 C, irradiation increases the strength of brazes, but at higher temperatures, the braze strength is slightly decreased by irradiation.(140)

TABLE 64. RATIO OF IRRADIATED[a] TO UNIRRADIATED DUCTILITY OF WELDS AND WROUGHT ALLOYS(140)

Base material	Ratio of Irradiated to Unirradiated Toughness at Indicated Temperature				
	500 C	600 C	700 C	800 C	900 C
Inconel 600 wrought metal	1.10	0.61	0.11	0.08	0.05
ERN-62 weld metal	1.0	0.99	0.28	0.21	0.10

(a) Irradiated to a fast fluence of 2.3 x 10^{18} n/cm^2.

TABLE 65. LOAD-CARRYING CAPACITY OF IRRADIATED[a] INCONEL 600 BRAZED WITH Ni-13Cr-10P(140)

Deformation Temp, C	Shear Strength, lb		Ratio
	Unirradiated	Irradiated	Irradiated/Unirradiated
RT	377	580	1.54
100	380	536	1.42
200	364	516	1.42
300	366	440	1.20
400	366	413	1.13
500	324	340	1.05
600	323	329	1.02
700	316	276	0.88
800	227	188	0.83

(a) Irradiated to a fast fluence of 2.3 x 10^{18} n/cm^2.

Inconel 625

Mixed Fast and Thermal Fluence

Tensile Properties. A considerable number of irradiated Inconel-625 specimens have been tested at various temperatures and the test results are given in Table 66. 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 cause 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.

Hardness. Table 67 shows that no significant changes in hardness have been found in Inconel 625 after irradiation at 280 C to a fast fluence of 1.4×10^{20} n/cm^2.[68]

Predominantly Fast Fluence

Tensile Properties. A few tensile tests have been performed on Inconel 625 which has been irradiated at 538 C in a predominantly fast flux in the EBR-II to a fast fluence of 3.3×10^{20} n/cm^2. Results of these tests are given in Table 68.[180]

The specimens irradiated in a predominantly fast flux do not undergo a decrease in strength as do Inconel 625 specimens irradiated at the same temperatures to equivalent fast fluences in a mixed fast and thermal flux. However, both environments cause reduction in elevated-temperature ductility. In comparing the irradiation results from EBR-II and a thermal reactor, it must be remembered that an equivalent fast fluence is accumulated about three or four times faster in EBR-II and, consequently, the specimens would undergo a significantly longer exposure at elevated temperature in the thermal reactor. This long-time exposure at temperature can significantly alter the mechanical properties, especially in case of a complex alloy such as Inconel 625.

TABLE 66. 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	193
50	RT	1.8×10^{20}	94	138.6	150	147.5	36.4	23.1	35.5		193
50	RT	4.6×10^{20}	94	125.5	150	151.5	36.4	28.1	35.5	53.4	193
50	RT	7.6×10^{20}	94	160.6	150	166	36.4	8.4	35.5	49.5	193
280	RT	5.3×10^{19}	102.1	118.1	158.5	169.8	39.2	38.2	48.9	43.2	193
280	RT	1.3×10^{20}	102.1	110.2	158.5	153.4	39.2	40.3	48.9	52.2	193
280	RT	2.6×10^{20}	102.1	106	158.5	150.1	39.2	44.7	48.9	52.9	193
280	RT	7.7×10^{20}	102.1	100	158.5	136.3	39.2	47	48.9	54.7	193
280	RT	8.3×10^{20}	102.1	117.6	158.5	148.6	39.2	31.5	48.9	64.7	193
580	RT	9.2×10^{19}	134	172.1	179	199.4	27.3	17.6	25	31.9	193
704	RT	2.0×10^{21}	120	111	169	154	31	23	—	—	180
740(a)	RT	9×10^{19}	81.9	100.4	106.3	128.0	5.9	6.4	3.7	11.7	193
740(b)	RT	9×10^{19}	92.6	113.8	134.9	156.8	10.8	12.1	10.8	10.1	193
280	300	1×10^{20}		94.3		139.5		26.7		17.6	193
740	300	9.2×10^{19}	94.2	100	149.5	143.7	18.1	6.2	17.7	16.7	193
740	300	1.9×10^{20}	91.5	85.2	140.2	128.1	9.0	5.3	11.0	9.7	193
50	650	1.7×10^{21}	83.6	108	129.6	125.6	19.1	7.2	16.9	17.5	193
280(b)	650	1×10^{20}	85.3	73.9	133.8	101.1	5.0	8.3	18.2	15.3	193
280(b)	650	3.2×10^{20}	85.3	85.1	133.8	109.1	5.0	9.1	18.2	17.1	193
280(b)	650	5.7×10^{20}	85.3	88.6	133.8	108.6	5.0	13.2	18.2	18.2	193
280(b)	650	1.5×10^{21}	85.3	45.4	133.8	52.1	5.0	4.5	18.2	28.9	193
704	704	3.5×10^{20}	87	75	106	90	8	4			180
704	704	3.5×10^{20}	89	80	110	88	7	1			180
740(a)	650	9.2×10^{19}	80.0	72.1	112	96.8	8.2	4.8	11.5	8.0	193
740(b)	650	9.2×10^{19}	84.6	66.2	124.6	96.7	10.5	6.6	11.1	11.0	193
704	593	3.5×10^{20}	91	74	146	122	22	8			180
704	593	2.0×10^{21}	104	92	146	114	19	3			180

(a) Mill annealed.
(b) Solution annealed.

TABLE 67. EFFECT OF IRRADIATION ON THE ROOM-TEMPERATURE HARDNESS OF VARIOUS NICKEL-BASE ALLOYS(68)

Material	Irradiation Temp, C	Fast Fluence, 10^{19} n/cm^2	Hardness, Rockwell A Unirr.	Hardness, Rockwell A Irr.
Inconel 625	50	18.4	67.6	71.6
Inconel 625	280	5.3	68.7	68.1
Inconel 625	280	7.7	69.1	68.5
Inconel 625	280	35	69	68.6
Inconel 702	50	18.4	69.5	69.4
Inconel 702	280	14.0	65.2	65.2
Inconel 718	50	18.4	71.6	72.3
Inconel 718	280	5.3	72.3	71.8
Inconel 718	280	7.7	72.1	72.0
Inconel 718	280	14.0	71.4	69.1
Inconel 718	280	35.0	72.3	72.7
Hastelloy R-235	50	12.7	64.8	66.0
Hastelloy R-235	280	5.3	65.7	66.4
Hastelloy R-235	280	7.7	65.4	66.9
Hastelloy R-235	280	14.0	63.9	63.9
Hastelloy R-235	280	35.0	65.6	63.2

TABLE 68. EFFECT OF PREDOMINANTLY FAST FLUX ON THE MECHANICAL PROPERTIES OF INCONEL 625(a)(180)

Test Temp, C	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent			
					Uniform		Total	
	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
RT	66.3	102.5	141	163.9	48.7	28.7	50.0	30.0
593	51.5	75.8	119	124.5	47.9	13.8	49.5	14.1
704	48.1	73.2	96.4	92.5	24.3	5.2	36.0	5.8

(a) Irradiated at 538 C to a fast fluence of 3.3 x 10^{20} n/cm^2 in EBR-II.

Inconel 702

Mixed Thermal and Fast Fluence

Tensile Properties. A few tensile tests have been performed on Inconel 702 which has been irradiated at 740 C, and the test results are given in Table 69.[168] The tensile properties of this age-hardenable alloy are only slightly affected by irradiation when tested at room temperature and 300 C. However, irradiation causes minor reductions in strength at 650 C and the ductility is drastically reduced. In another series of postirradiation tensile tests at room temperature, it was found that the strength values reached a maximum after a fast fluence of 1 x 10^{21} n/cm^2 and then decreased.[10] However, the ductility of Inconel 702 was found to increase with increasing fast fluences above 1 x 10^{21} n/cm^2. These tensile results are shown in Figure 116.

TABLE 69. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF INCONEL 702 IRRADIATED AT 740 C[168]

Fast Fluence, 10^{19} n/cm^2	Test Temp, C	0.2% Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reduction in Area, percent	
						Uniform		Total			
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
9.2	20	81.9	77.4	137.1	168	29.0	26.7	31.7	28.0	31.0	23.6
22	20	73.1	80.2	142	151.8	31.6	23.9	35.7	34.9	35.7	36.1
24	20	76.0	70.0	143	138.8	30.7	34.8	33.2	36.4	46.5	31.5
40	20		65.2		137.8		35.8		36.5		32.8
9.2	300		69.2		130.8		22.8		23.5		23.7
22	300	69.2	62.2	135.9	122.7	24.8	23.7	27.2	25.1	49.2	23.0
24	300	65.0	62.3	130.0	127.5	25.7	36.9	28.5	38.1	41.0	32.8
22	650	66.5	56.8	107.1	57.3	7.5	1.1	7.8	1.1	8.1	
24	650	63.2	54.9	98.7	60.1	6.0	1.3	6.5	1.4	5.7	5.2

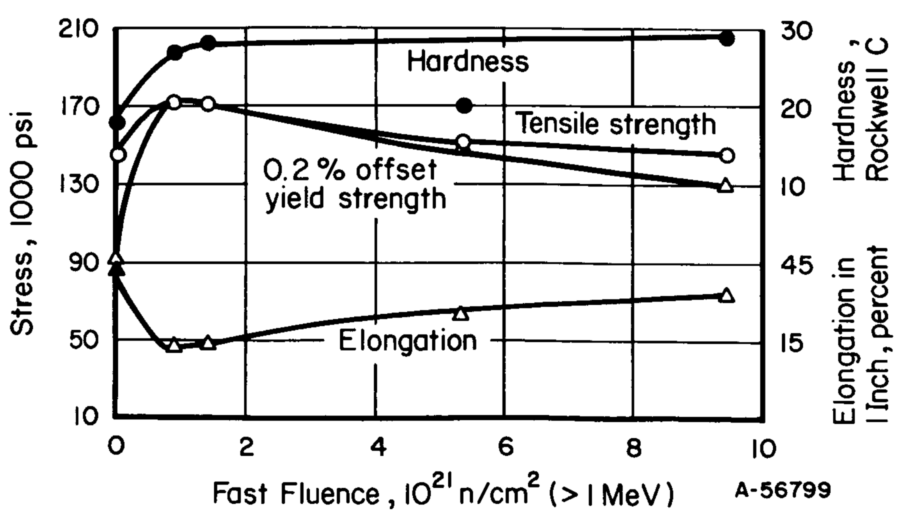


FIGURE 116. EFFECTS OF IRRADIATION ON THE ROOM-TEMPERATURE ROCKWELL HARDNESS, TENSILE STRENGTH, AND ELONGATION OF INCONEL 702[10]

TABLE 70. EFFECT OF NEUTRON IRRADIATION ON THE MECHANICAL PROPERTIES OF INCONEL 718[168,187]

Test Temp, C	Irradiation Temp, C	Fast Fluence, n/cm^2	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.
RT	280	4.5×10^{19}	158-162	178	190-200	196			12-21	9		
RT	280	5.9×10^{19}	158-162	192	190-200	205			12-21	12-14		
RT	280	7.2×10^{19}	158-162	160-170	190-200	180-188			12-21	7-11		
RT	280	1.1×10^{20}	158-162	135-150	190-200	164-179			12-21	17-25		
RT	280	5.4×10^{20}	158-162	145-163	190-200	170-185			12-21	10-20		
300	740	9.2×10^{19}	65.4	71.4	127	118.8	25.6	11.4	27.0	12.2	24.8	17.8
300	740	2.2×10^{19}	71.4	78.0	128	117.8	18.2	8.7	18.7	10.0	14.6	16.0
300	740	2.4×10^{19}	65.3	62.8	122	106.3	17.8	7.6	18.7	8.1	17.6	15.0
650	50	1.8×10^{20}	102	122	119	139	11.1	2.3	12.6	2.6	41.9	12.9
650	50	1×10^{21}	102	130	119	131	11.1	0.8	12.6	0.8	41.9	11.0
650	280	1×10^{20}	121	87.4	138	99.8	7.4	5.2	8.1	5.9	21.3	18.2
650	280	1.6×10^{20}	121	95.5	138	108	7.4	5.0	8.1	5.7	21.3	16.4
650	280	3.2×10^{20}	121	139	138	140	7.4	0.8	8.1	0.9	21.3	10.4
650	740	9.2×10^{19}	64	51.5	104	79.6	12.5	5.3	25.6	6.0	24.1	9.7
650	740	2.2×10^{20}	64	62.1	104	89.1	12.5	4.4	25.6	5.0	24.1	11.6
650	740	2.4×10^{20}	64	51.5	104	79.2	12.5	5.0	25.6	5.7	24.1	14.4
650	740	4.0×10^{20}	64	62.8	104	94.4	12.5	6.5	25.6	6.7	24.1	10.1

Hardness Properties. Irradiation causes only minor changes in hardness (see Table 67).[68] However, as can be seen in Figure 116, a fast fluence of 1 x 10^{21} n/cm^2 increases the hardness from R_C 18 to R_C 29.[10] Only minor hardness changes occur with increasing fast fluence.

Inconel 718

Mixed Thermal and Fast Fluence

Tensile Properties. The postirradiation tensile properties of Inconel 718 at room temperature and 650 C are given in Table 70.[168, 187] The data on room-temperature properties of irradiated Inconel 718 show that minor changes in strength and ductility result from irradiation at 280 C. The strength of Inconel 718 seems to be reduced significantly by thermal aging at 740 C and is reduced further by irradiation at 740 C. Inconel 718 undergoes irradiation-induced elevated-temperature embrittlement as do most nickel alloys, with the total elongation at 650 C being only 0.8 percent after irradiation to a fast fluence of 1 x 10^{21} n/cm^2. The irradiation temperature does not appear to affect significantly the severity of irradiation-induced embrittlement at 650 C.

Hardness Properties. Irradiation causes only minor changes in the hardness of Inconel 718 (see Table 67).[68]

Incoloy 825

Mixed Thermal and Fast Fluence

Tensile Properties. A few tensile tests have been performed on irradiated Incoloy 825 specimens. The results (Table 71)[180] indicate that while the long-time irradiation at elevated temperature causes some reductions in strength, the ductility at elevated temperatures is drastically reduced by even the short-term irradiation.

TABLE 71. EFFECT OF IRRADIATION[a] ON THE TENSILE PROPERTIES OF INCOLOY 825 AT 704 C[180]

Fast Fluence, n/cm^2	Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent			
					Uniform		Total	
	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
3.5 x 10^{20}	36	37	54	51	11	6	46	6
2.0 x 10^{21}	41	34	60	46	10	3	49	3

(a) Irradiated at 704 C in a mixed thermal and fast fluence.

Hastelloy R-235

Mixed Thermal and Fast Fluence

Tensile Properties. Table 72 illustrates that irradiation causes drastic reductions in ductility in the precipitation-hardenable Hastelloy R-235 alloy, even at room temperature.(168) The strength of the alloy is also reduced by irradiation at all testing temperatures.

TABLE 72. EFFECT OF TEST TEMPERATURE ON THE MECHANICAL PROPERTIES OF HASTELLOY R-235(168)

Fast Fluence, 10^{19} n/cm^2	Test Temp, C	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reduction in Area, percent	
						Uniform		Total			
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
9.2	20	95	107.9	136.7	114.2	15.6	1.1	15.5	1.2	14.3	5.3
22	20	102.5	120.1	155.8	128.7	13.8	1.4	14.0	1.5	10.1	9.3
40	20		99.8		101.5		0.6		0.7		1.5
9.2	300	97.5	108.1	157.0	128.3	15.3	3.6	16.3	4.1	18.1	6.9
22	300		102.4		124.8		4.8		5.1		6.5
24	300	99.5	97.8	165.0	109.8	18.1	2.0	19.4	2.3	18.6	8.3
9.2	650	94.8	90.9	145.3	95.6	11.8	0.8	23.4	0.9	25.6	6.6
22	650	93.4	91.1	141.1	101.5	9.8	2.4	19.8	3.6	20.8	7.9
24	650	96.5	81.7	144	100.5	10.8	2.8	22.5	3.1	18.9	5.0

Hardness. Irradiation causes only minor changes in the hardness of Hastelloy R-235 (see Table 67).(68)

Inconel X

Mixed Thermal and Fast Fluence

Tensile Properties. Figure 117 illustrates the effect of fast fluence on the room-temperature mechanical properties of Inconel X.(10) The data show that the strength increase and ductility decrease reach a maximum after a fast fluence of 1×10^{21} n/cm^2. With increasing fast fluence, the strength decreases and ductility increases. After a fast fluence of about 3×10^{21} n/cm^2, the ultimate strength value is reduced below the value for unirradiated material.

Hardness. Figure 117 shows that the hardness is only slightly affected by fast fluences. The hardness reaches a maximum after a fast fluence of 1×10^{21} n/cm^2 and then decreases.(10)

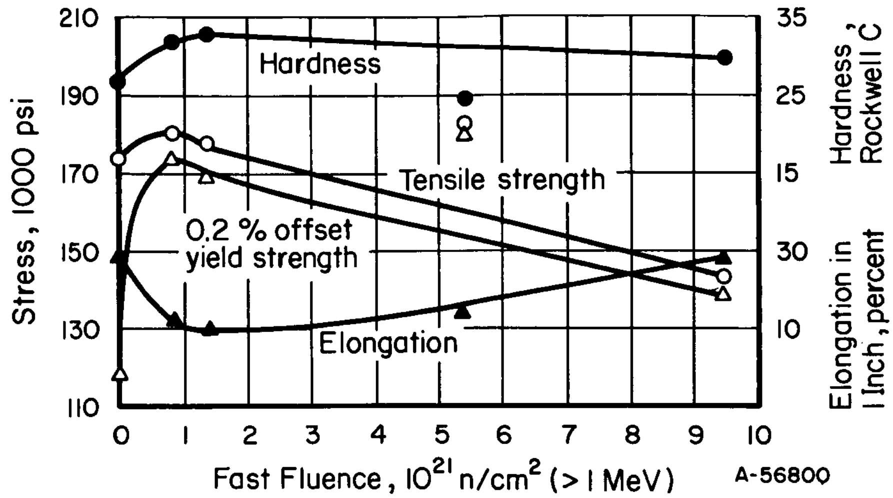


FIGURE 117. EFFECTS OF IRRADIATION ON THE ROCKWELL HARDNESS, TENSILE STRENGTH, YIELD STRENGTH, AND ELONGATION OF INCONEL X DA(10)

René 41

Mixed Thermal and Fast Fluence

Tensile Properties. The effect of irradiation on the notched and unnotched tensile properties of René 41 is illustrated in Table 73(162, 194). These test results indicate 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 118.(164) Both the rupture strength and ductility at 650 C are decreased with increasing fast fluence.

PDRL-120

Mixed Thermal and Fast Fluence

Tensile Properties. The effect of irradiation at 280 C on the tensile properties of high-strength PDRL-120 is illustrated in Figure 119.(158) Irradiation increases the yield strength up to about 550 C, after which the irradiation damage is annealed out; the irradiation-induced losses in ductility are not recovered at temperatures up to 650 C.

TABLE 73. EFFECT OF IRRADIATION ON THE TENSILE PROPERTIES OF RENÉ 41[162, 194]

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.

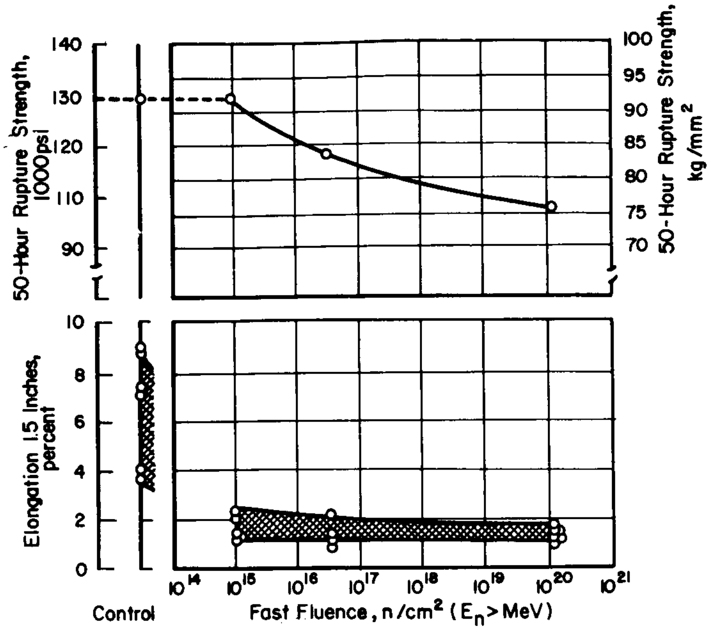


FIGURE 118. STRESS-RUPTURE STRENGTH AND ELONGATION OF RENÉ 41 AT 650 C AS A FUNCTION OF FAST FLUENCE[164]

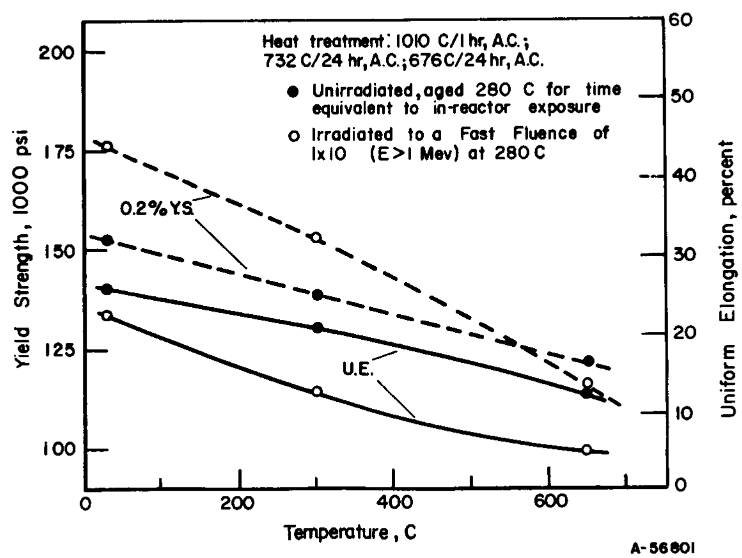


FIGURE 119. EFFECT OF VARYING TEST TEMPERATURE ON THE TENSILE PROPERTIES OF NICKEL-BASE ALLOY PDRL-120 IRRADIATED AT 280 C[158]

Nimonic Alloys

Mixed Thermal and Fast Fluence

Tensile Properties. The tensile properties of irradiated Nimonic PE16 have been determined and the results are shown in Figure 120.[160] The Nimonic PE16 alloy is affected by irradiation in the same manner as are other nickel-base alloys in that a significant reduction in ductility occurs at elevated temperatures.

Creep Properties. The effect of irradiation on the stress relaxation of Nimonic 80A has been studied. Figure 121 shows that stress relaxation due to irradiation is considerably higher than that due to purely thermal effects.[196]

Predominantly Fast Fluence

Tensile Properties. Figure 122 shows that the minor reductions in strength and the drastic reductions in ductility are caused by irradiation in a predominantly fast flux in the Dounreay Fast Reactor. These reductions become more severe with increasing fast fluence.[160] Figure 123 shows that increasing the grain size results in more severe irradiation-induced embrittlement at a testing temperature of 650 C.[160]

Russian Nickel-Base Alloys

Mixed Thermal and Fast Fluence

Tensile Properties. The effect of irradiation on the strength properties of two Russian nickel-base alloys is illustrated in Figure 124.[174] The nickel alloy containing a total of 3.25 weight percent aluminum + titanium undergoes significant reductions in strength as a result of irradiation. The alloy that contains only 1.0 weight percent aluminum + titanium does not undergo similar strength decreases as a result of irradiation. Therefore, it appears that these strength reductions may be due to overaging rather than irradiation. The effect of irradiation on the ductility of these alloys is illustrated in Figure 125. This figure shows that the ductility undergoes a drastic decrease at about 600 C and is not recovered at higher testing temperatures. It is concluded by the Russians that the loss in ductility is due to formation of helium at the grain boundaries. The helium is generated by the (n, α) reactions which take place between thermal neutrons and boron-10. However, some doubt is raised concerning the actual embrittling mechanism, since the variation of the boron content from 10 to 150 ppm did not significantly affect the elevated-temperature elongation of stress-rupture specimens. These test results and interpretations as to causes of elevated-temperature embrittlement are in general agreement with the test results and interpretations of irradiation effects on nickel-base alloys in the United States.

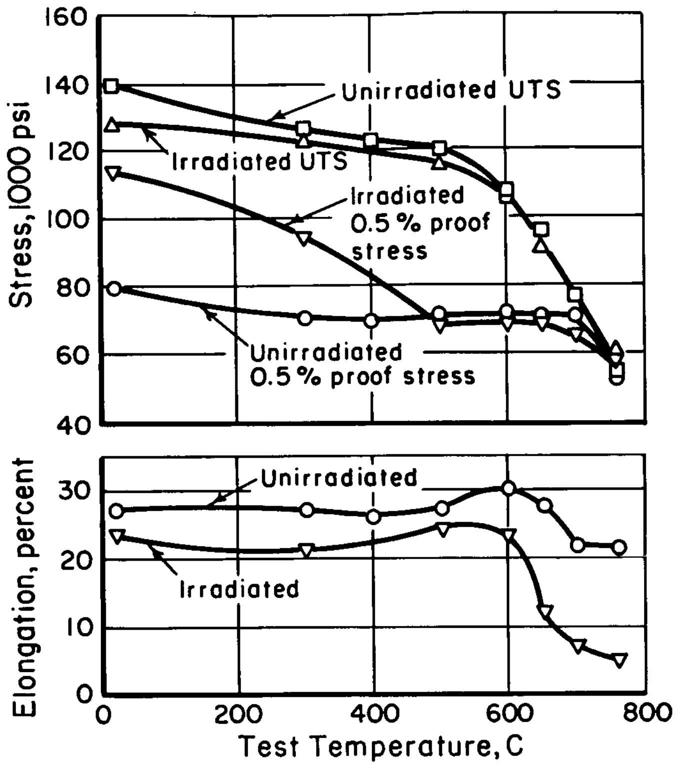


FIGURE 120. EFFECT OF NEUTRON IRRADIATION ON THE ROOM- AND ELEVATED-TEMPERATURE TENSILE PROPERTIES OF NIMONIC PE16 ALLOY[158]

Irradiated to a thermal fluence of 5×10^{19} n/cm^2 and a fast fluence of 2.5×10^{19} n/cm^2 at 45 C.

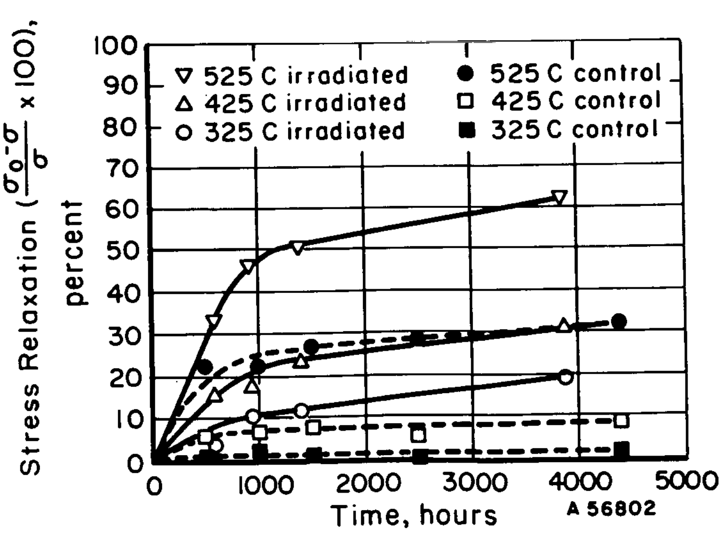


FIGURE 121. PERCENTAGE STRESS RELAXATION FOR CONTROL SPRINGS AND SPRINGS IRRADIATED AT 325, 425 AND 525 C[196]

Irradiated to a fast fluence of 0.6 to 5.9×10^{19} n/cm^2.

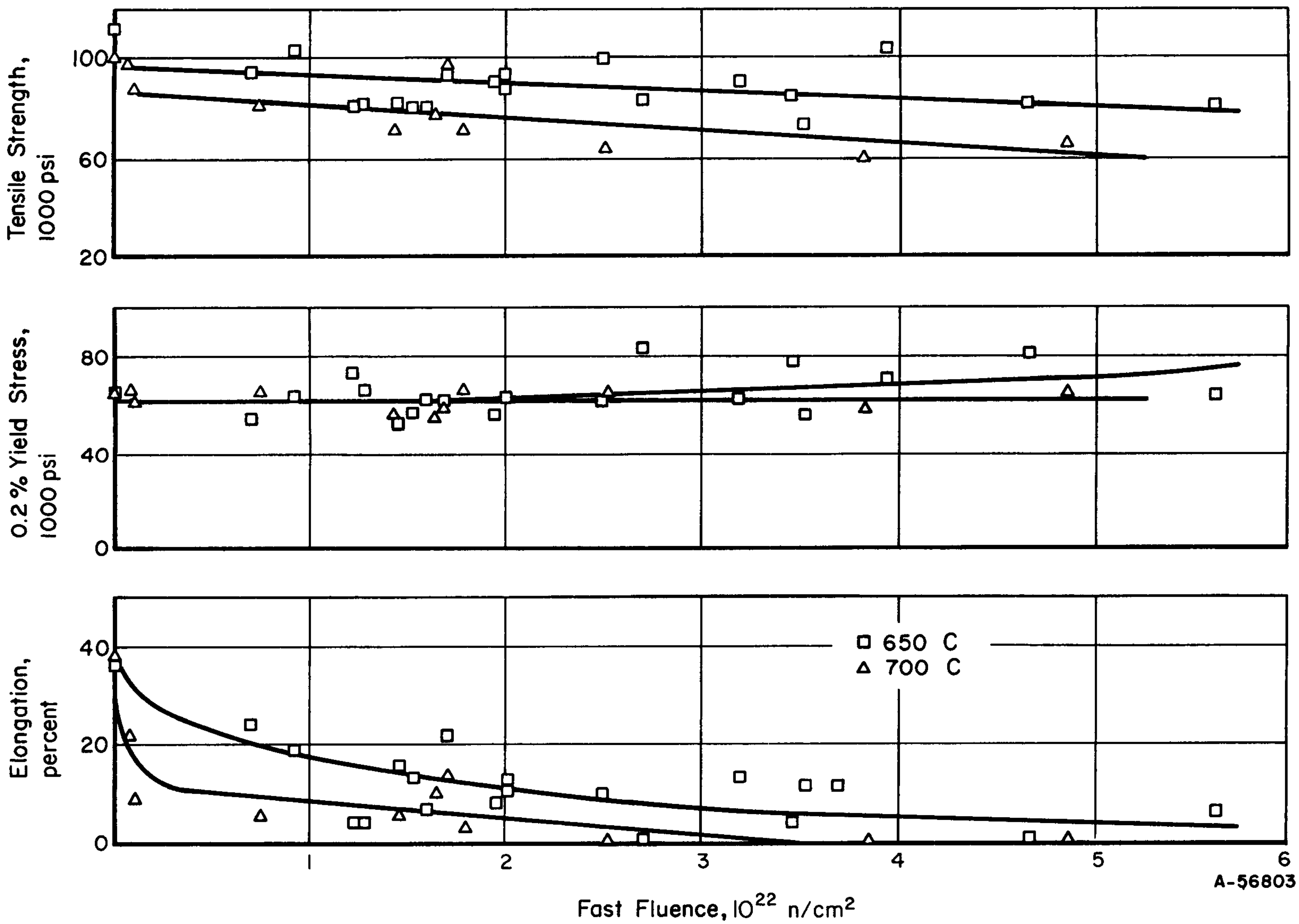


FIGURE 122. TENSILE PROPERTIES OF FULLY HEAT TREATED NIMONIC PE16 AT 650 AND 700 C AFTER IRRADIATION AT 250 TO 350 C(160)

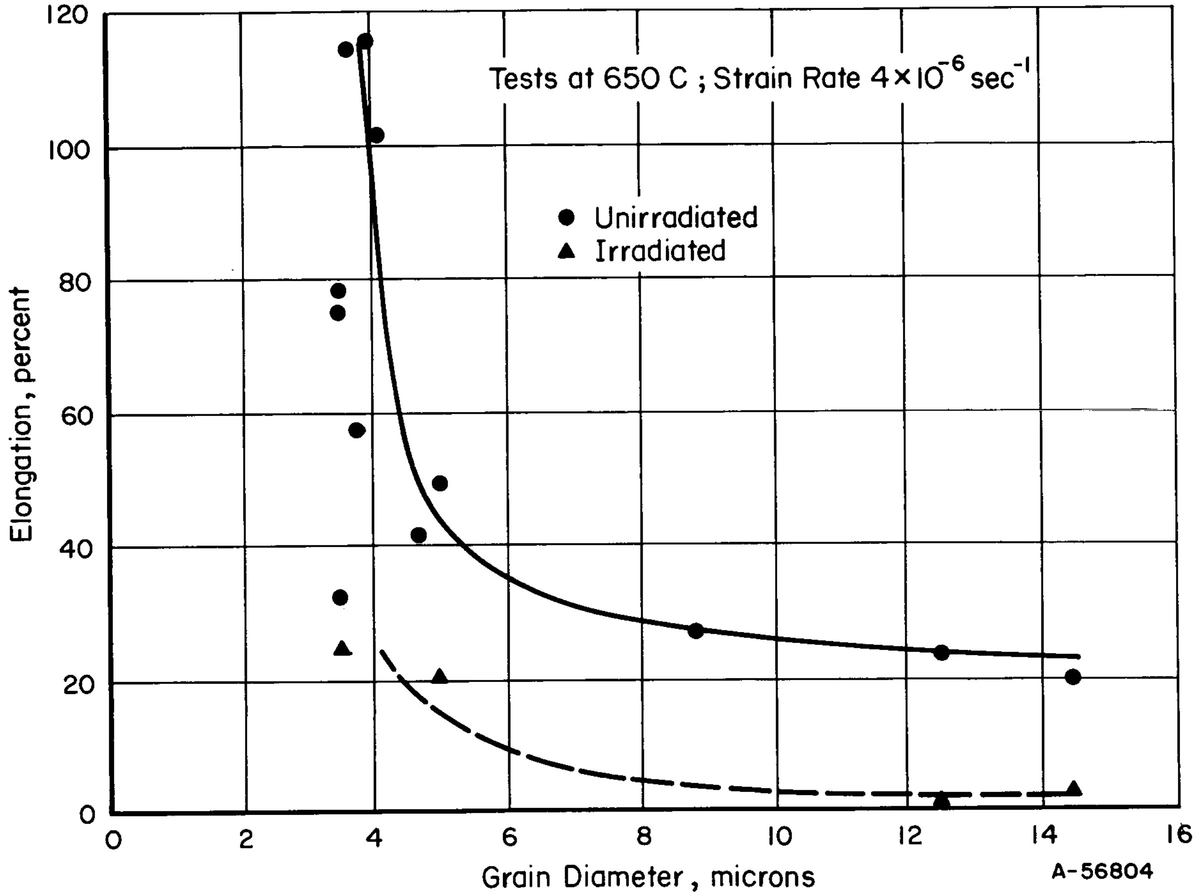
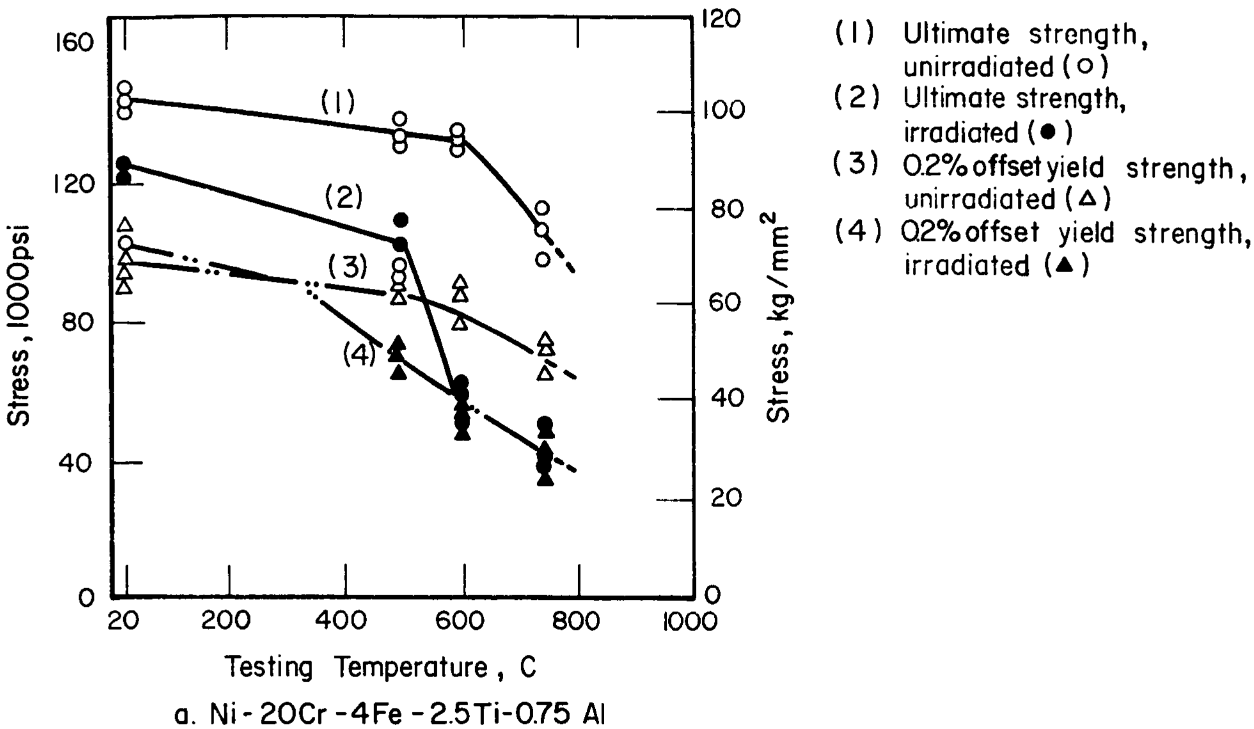
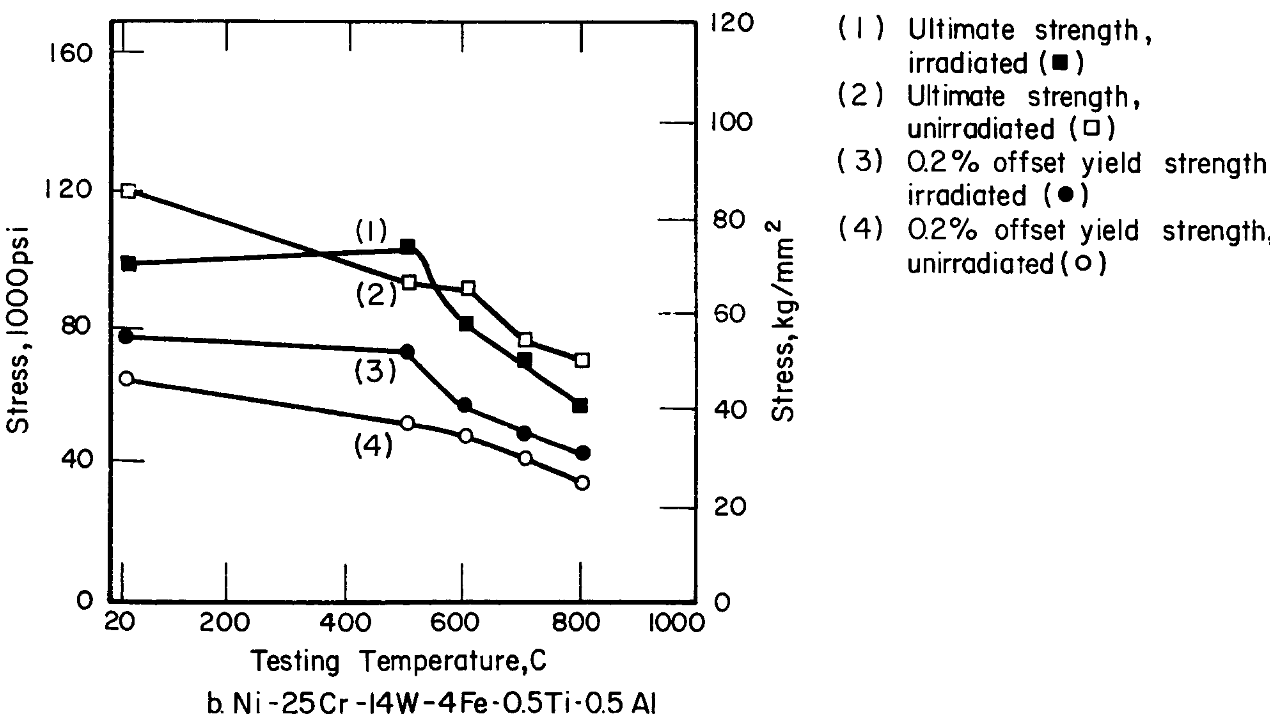


FIGURE 123. VARIATION OF ELONGATION TO FAILURE WITH GRAIN SIZE IN LOW-BORON NIMONIC PE16[160]

Thermal fluence 1.5×10^{20} n/cm^2; fast fluence 5×10^{19} n/cm^2.



a. Ni-20Cr-4Fe-2.5Ti-0.75 Al



b. Ni-25Cr-14W-4Fe-0.5Ti-0.5 Al

FIGURE 124. EFFECT OF IRRADIATION ON THE TENSILE PROPERTIES OF THE RUSSIAN NICKEL-BASE ALLOYS Ni-20Cr-4Fe-2.5Ti-0.75 Al AND Ni-25Cr-14W-4Fe-0.5Ti-0.5Al IRRADIATED TO A FAST FLUENCE OF 1 TO 3 x 10^{20} N/CM^2 AT 150 TO 200 C[174]

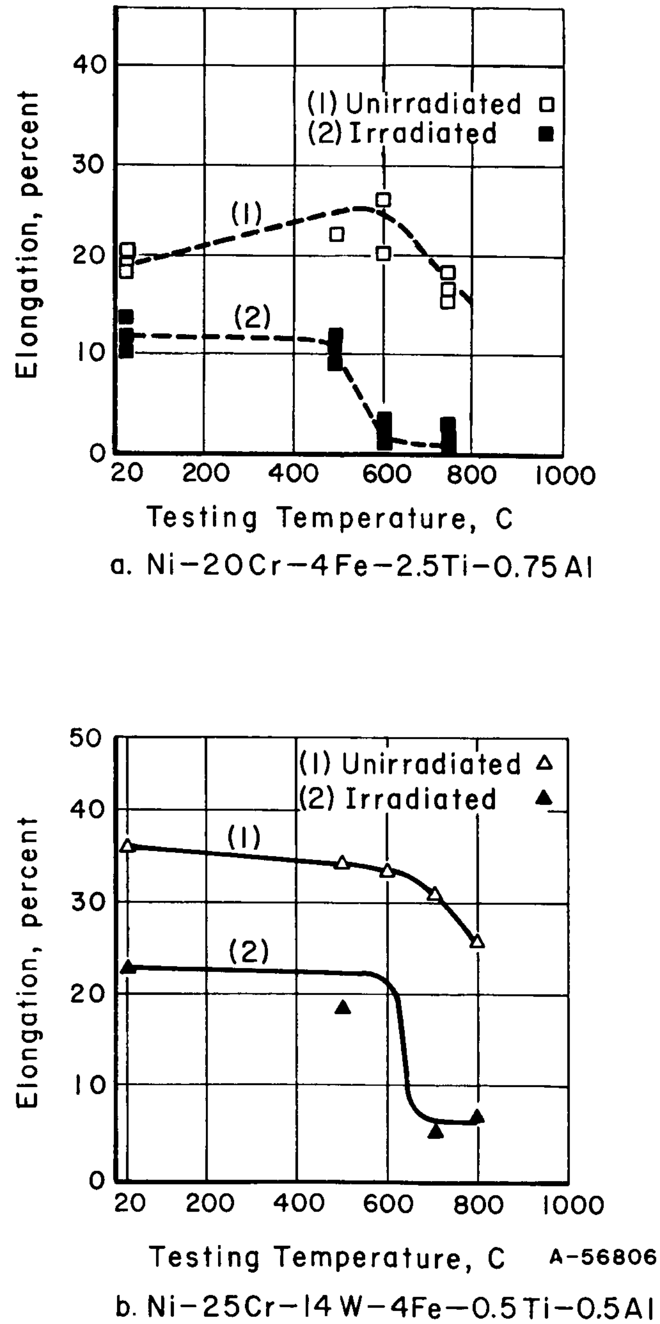


FIGURE 125. EFFECT OF IRRADIATION ON THE DUCTILITY OF Ni-20Cr-4Fe-2.5Ti-0.75Al AND Ni-25Cr-14W-4Fe-0.5Ti-0.5Al IRRADIATED TO A FAST FLUENCE OF 1 TO 3 x 10^{20} N/CM^2 AT 150 TO 200 C[174]

Creep Properties. The effect of irradiation on the stress-rupture properties of a Russian nickel-base alloy is shown in Figure 126. The alloy undergoes large reductions in both rupture life and elongation at rupture when tested at 800 C.[5]

Hardness. The effect of irradiation on the hot hardness of a Russian nickel-base alloy (in both the hardened and the aged conditions) is shown in Figure 105.[174] The preirradiation hardness of the hardened alloy is restored at about 400 C, while the aged alloy recovers its preirradiation hardness at 500 C.

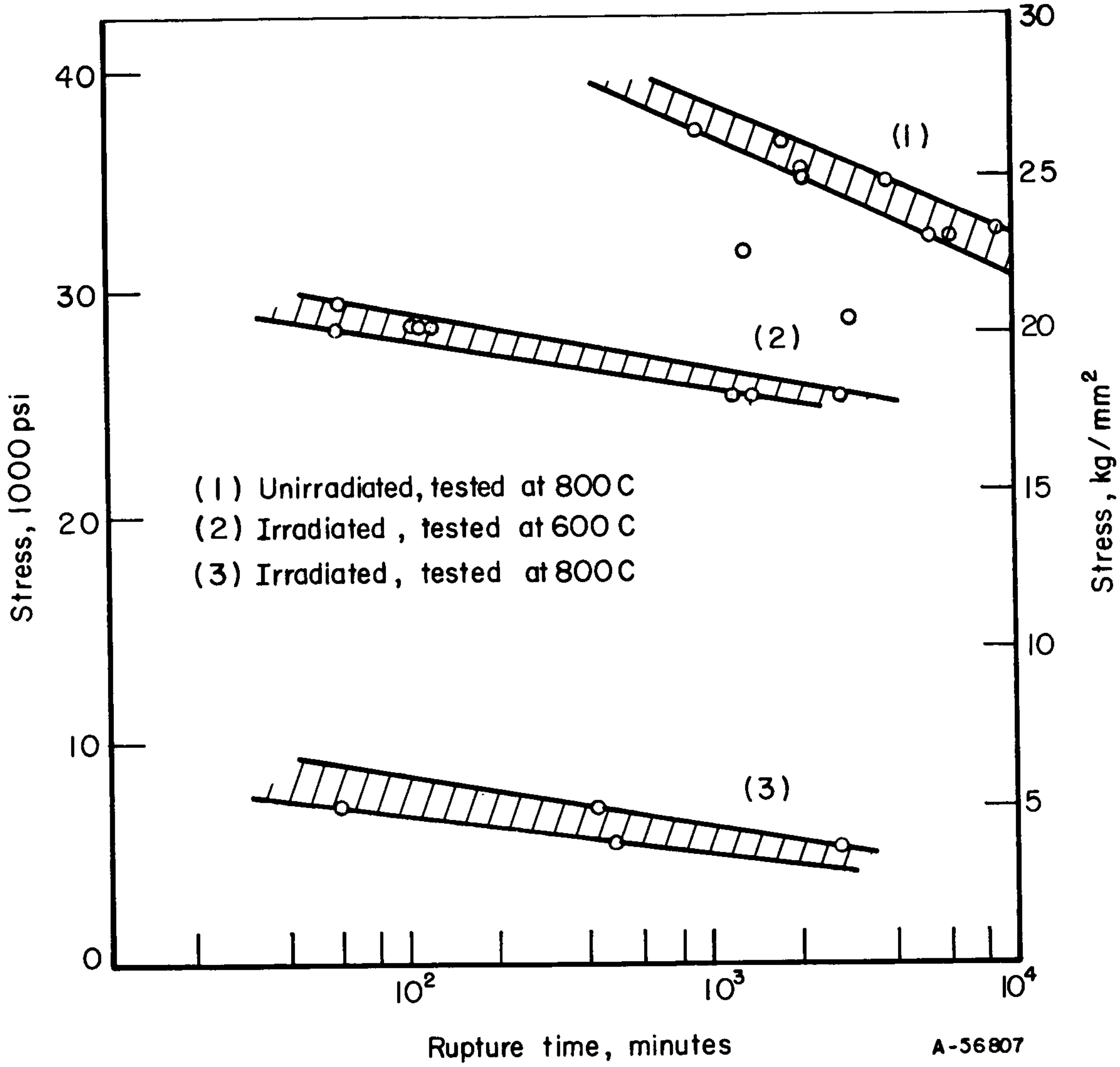


FIGURE 126. EFFECT OF IRRADIATION ON THE STRESS-RUPTURE PROPERTIES OF THE RUSSIAN NICKEL-BASE ALLOY Ni-20Cr-4Fe-2.5Ti-0.75Al IRRADIATED TO A FAST FLUENCE OF 1 TO 3 x 10^{20} N/CM^2 AT 150 TO 200 C(174)

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, and valves.

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 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.

Table 74 lists the composition of materials irradiated and tested at cryogenic temperatures. Data showing the effects of irradiation on mechanical properties of these materials at cryogenic temperatures are given in Table 75. Theoretically, irradiations at cryogenic temperatures are 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 will be annealed out at room temperature, while at the cryogenic temperatures, no significant annealing of fast-neutron-caused defects will 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. At this point, 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 of sufficient fast fluence 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 properties on specimens taken from the same heat and in specimens taken from different heats and the possible irradiation-induced changes in mechanical properties.

TABLE 74. COMPOSITION OF MATERIALS IRRADIATED AND TESTED AT CRYOGENIC TEMPERATURES

Material	Composition, weight percent											
	Al	Fe	Ni	Mn	Si	Cr	C	Mo	Ti	Cu	Mg	Zn	Other
1099	99.99[a]												
2014	Bal	1.0		0.4-1.2	0.5-1.2	0.10				3.9-5	0.2-0.8	0.25	0.15
2024	Bal	0.5		0.3-0.9	0.5	0.10				3.8-4.9	1.2-1.8	0.25	0.15
2219	Bal	0.3		0.2-0.4	0.2				0.02-.10	5.8-6.8	0.02	0.10	0.15
5083	Bal	0.4		0.3-1.0	0.4	.05-.25			0.15	0.10	4-4.9	0.25	0.15
5086	Bal	0.5		0.2-0.7	0.4	.05-.25			0.15	0.10	3.5-4.5	0.25	0.15
5456	Bal	0.4[b]		0.5-1.0	0.4[b]	.05-.20			0.20	0.20[c]	4.7-5.5	0.25	0.15
6061	Bal	0.7		0.15	0.4-0.8	.15-.35			0.15	0.15-0.40	0.8-1.2	0.25	0.15
7075	Bal	0.7[c]		0.1-0.3	0.5[c]	.18-.40			0.2[c]	1.2-2.0	2.1-2.9	5.1-6.1	
7079	Bal	0.4		0.1-0.3	0.3	.10-.25			0.10	0.4-0.8	2.9-3.7	3.8-4.8	0.15
7178	Bal	0.7		0.3	0.5	.18-.40			0.20	1.6-2.4	2.4-3.1	6.3-7.3	0.15
X-250	Bal	1.5[c]		0.1	0.6[c]				0.2[c]	9.2-10.7	0.15-0.35		0.20
13-750	Bal		0.9-1.5							1.7-2.3	0.6-0.9		5.5-7.0Sn
A-356	Bal	0.6		0.10	6.5-7.5				0.2		0.2-0.4	0.10	
55A							0.20[c]		99.8[a]				0.08N 0.15H
Ti-5Al-2.5Sn	4-6								Bal				1.5-3.5Sn
Ti-6Al-4V	5.5-6.5						0.10[c]		Bal				3.5-4.5V 0.6
Ti-8Al-1Mo-1V													
AISI 301		Bal	6-8	2.0[c]	1.0[c]	16-18	0.15[c]						
AISI 303 Se		Bal	8-10	2.0[c]	1.0[c]	17-19	0.15[c]						0.15Se
AISI 304		Bal	8-11	2.0[c]	0.75[c]	18-20	0.08[c]						
AISI 310		Bal	19-22	2.0[c]	1.5[c]	24-26	0.25[c]						
AISI 347		Bal	9-13	2.0[c]	1.0[c]	17-19	0.08[c]						0.8Ta+Nb
AISI 440 C		Bal		1.0	1.0[c]	16-18	0.95-1.2	0.75[c]					
A 286	0.2	54	26	1.4		15	0.05	1.25	2.0				0.3V
AM 350		Bal	4.5	1.80	0.25	16-17	0.10	2.75					0.1N
17-7Ph	0.75-1.5	Bal	6.5-7.5	1.0[c]	1.0[c]	16-18	0.09						
René 41	1.5		Bal			19	0.09	10	3.10				11Co
K Monel	2-4	2	63-70	1.5[c]	1.0[c]		0.25[c]		0.25-1.0	30-35			
Inconel		6-10	72[a]	1.0[c]	0.5[c]	14-17	0.15[c]						
Inconel X	0.9-1.5	5-9	70[a]	1.0[c]	0.5[c]	14-17	0.10[c]		2.0-2.6				0.7-1.2Nb
Inconel X-750	0.8	7	73	0.7	0.3	15	0.04		2.5				0.85Nb
Inconel 713C	5.5-6.5	5[c]	Bal	1.0[c]	1.0[c]	11-14	0.20[c]	3.5-5.5	0.75-1.25				1-3Nb
Inconel 718	0.6	18	52.5	0.20	0.20	19	0.04	3.0	0.8				3Nb
Hastelloy C	1.0	4-7	Bal	1.0[c]	1.0[c]	14.5-16.5	0.08[c]	15-17					(d)

(a) Minimum.
(b) Combined iron + silicon.
(c) Maximum.
(d) 3 to 4.5 weight percent tungsten, 2.5 weight percent cobalt, 0.35 weight percent vanadium.

TABLE 75. 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	197
2014		4.6×10^{16}(b)	-210	-236	85.5	90.3	105.0	97.5			6	5			198
2014		4.6×10^{16}(b)	-210	RT			78	59							198
2014		4.6×10^{16}(b)	-210	-236			101	91.2							198
2014		4.6×10^{16}(b)	-210	-236			75.3	65.3							198
2014		4.6×10^{16}(b)	-210	-236	66.0	61.0	74.0	71.0			4	5			198
2014	T-651	1.0×10^{17}(a)	-257	-257	68.3	71.8	91.1	84.6	101.2		17.7	12.7	26.3	18.3	197
2014		2.0×10^{17}(b)	-250	-250			75.3	75.8							198
2014		2.0×10^{17}(b)	-250	-250			101	96.3							198
2014		2.0×10^{17}(b)	-250	-250	85.5	87.2	105.0	95.6			6	6			198
2024	T-351	1.0×10^{17}(a)	-257	-257	77.2	79.4	106.6	100.8	95.6		22.3	16.3	20.3	18.0	197
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	197
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			199
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			199
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			199
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			199
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			199
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			199
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			199
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			199
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			199
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	197
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	197
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	199
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	199
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	199
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	197
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	199
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	197
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	197
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	199
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	197
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	197
X-250	T-4	1.0×10^{17}(a)	-257	-257	49.0	(a)	49.5	46.9	52.8		nil	nil	nil	nil	197
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	197
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	199
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	199
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	197
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	197
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			199
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			199
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			199

TABLE 75. (Continued)

Material	Condition	Fast Fluence, n/cm²	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.	Notched Tensile Strength, 1000 psi Unirr.	Notched Tensile Strength, 1000 psi Irr.	Total Elongation, percent Unirr.	Total Elongation, percent Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Ref.
Ti-6Al-4 V	A	1.0 x 10^{17}(a)	-257	-257	243.2	254	260.4	273.7	281.6	283.7	7.6	5.7	30.4	37.3	197
Ti-6Al-4 V	Aged	1.0 x 10^{17}(a)	-257	-257	275	293.3	282.2	302.3	288.8	295	6.4	5.0	25.4	22.5	197
Ti-8Al-1Mo-1V		1.0 x 10^{17}(a)	-257	-257	224.2	242.7	239	261.7	267.2	282	--	5.7	--	29.0	197
Be		2.4 x 10^{20}(c)	300-500	-196		(d)		64.8				0.1		0.2	22
Be		2.4 x 10^{20}(c)	300-500	-72		(d)		61.5				0.1		0.2	22
AlSl-301	H	4.6 x 10^{16}(b)	-210	-235	302	250	391	250			13	6			198
AlSl-301	H	4.6 x 10^{16}(b)	-210	RT	201	180	213	225			10	20			198
AlSl-301	H	2.0 x 10^{17}(b)	-248	-254	302	284	391	284			13	8			198
AlSl-301	S	5.1 x 10^{17}(c)	-196	-196	165.2	176.1	292.8	290.4	207.3	214.3	20.4	20.3			199
AlSl-301	S	5.1 x 10^{17}(c)	-196	-196 a	165.2	173.7	292.8	294.3	207.3	216.7	20.4	20.0			199
AlSl-301	S	5.1 x 10^{17}(c)	-196	27	151.0	156.3	186.9	188.8	191.2	193.8	19.2	17.4			199
AlSl-303 Se	B	6.5 x 10^{17}(c)	-196	-196	85.4	115.0	165.7	173.0	191.0	228.2	44.8	34.3			199
AlSl-303 Se	B	6.5 x 10^{17}(c)	-196	-196 a	85.4	104.4	165.7	172.4	191.0	208.5	44.8	38.7			199
AlSl-303 Se	B	6.5 x 10^{17}(c)	-196	27	49.1	60.8	92.7	96.0	121.6	131.4	31.2	26.0			199
AlSl-304		1.0 x 10^{17}(a)	-257	-257	39.6	51.9	242	260.3	193.4	173.3	34.4	33.5	38.4	28.5	197
AlSl-310	H	4.6 x 10^{16}(b)	-210	RT	158	158	180	178			1	1			198
AlSl-310	H	4.6 x 10^{16}(b)	-210	-235	246	235	297	252			6	2			198
AlSl-310		1.0 x 10^{17}(a)	-257	-257	137.8	135.7	212.2	218.7	188.6	218	42.8	48.3	56.2	21.7	197
AlSl-310	H	2.0 x 10^{17}(b)	-248	-254	246	254	297	267			6	1			198
AlSl-347-(b)	L	5.0 x 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	199
AlSl-347-(b)	LW	5.0 x 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	199
AlSl-347-(b)	L	5.0 x 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	199
AlSl-347		1.0 x 10^{17}(a)	-257	-257	51.9	64.0	237.4	250.7	214.2	238	41.3	37.0	43.8	55.0	197
AlSl-440 C		1.0 x 10^{17}(a)	-257	-257	(d)	(d)	261.2	217.7	107.7	118.7	nil	nil	nil	nil	197
A-286	L	5.0 x 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	199
A-286*		1.0 x 10^{17}(a)	-257	-257	149.6	152	235	229	216.4	245.7	34.5	33.5	46.5	16	197
A-386*		1.0 x 10^{17}(a)	-257	-257	156.8	165.7	238.2	238	255	239	31.8	31.0	42.2	39.7	197
AM-350		1.0 x 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	197
A-353		1.0 x 10^{17}(a)	-257	-257	174.8	190.0	201.6	215.7	189.6		18.0	15.7	39.6	21.3	197
T-450		1.0 x 10^{17}(a)	-257	-257	91.5	92.3	197.2	192.7	214.4		31.2	30.3	27.2	29.7	197
17-7 PH		1.0 x 10^{17}(a)	-257	-257	330.0	250.7	335.4	251.3	182.6		nil	nil	nil	nil	197
René 41		1.0 x 10^{17}(a)	-257	-257	107.6	113.3	194.4	194.7	204.8	194	60.5	55	50.5	48	197
K Monel		1.0 x 10^{17}(a)	-257	-257	121.4	138.7	187.2	188.3	210	211.3	32	33	54.8	51.3	197
Inconel		1.0 x 10^{17}(a)	-257	-257	175.8	179	186.4	191	222.4	237.3	20	25	56	51	197
Inconel X		1.0 x 10^{17}(a)	-257	-257	150.8	161.3	243.2	241	249.8	230.3	33	29	45.6	37.3	197
Inconel 713 C	L	5.0 x 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	199
Inconel 718	B	1.0 x 10^{18}(c)	-196	-196	206.7	235.2	267.9	271.6	312.6	342.0	12.5	8.4			199
Inconel 718	B	1.0 x 10^{18}(c)	-196	-196 a	206.7	222.5	267.9	273.6	312.6	338.2	12.5	12.7			199
Inconel 718	B	1.0 x 10^{18}(c)	-196	27	179.5	188.8	217.8	214.8	280.9	290.5	10.9	11.5			199
Inconel 718	S	7.0 x 10^{17}(c)	-196	-196	212.4	232.1	270.0	269.6	242.6	259.9	12.6	12.3			199
Inconel 718	S	7.0 x 10^{17}(c)	-196	-196 a	212.4	217.4	270.0	267.4	242.6	252.4	12.6	13.3			199
Inconel 718	S	7.0 x 10^{17}(c)	-196	27	184.9	188.2	218.7	213.7	219.2	218.7	11.3	11.8			199

TABLE 75. (Continued)

Material	Condition	Fast Fluence, n/cm^2	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.	Notched Tensile Strength, 1000 psi Unirr.	Notched Tensile Strength, 1000 psi Irr.	Total Elongation, 1000 psi Unirr.	Total Elongation, 1000 psi Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Ref.
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			199
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			199
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			199
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	199
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			199
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			199
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			199
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			199
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			199
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			199
Hastelloy-C	L	5.0×10^{16}	-257	-257	111.0	111.5	185.7	185.4	--	156.3	39.2	50.4	32.6	32.8	199

(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.

Lockheed Nuclear Products Company[197] investigators used designations of "real" degradation in properties if the confidence level of results was above 90 percent and "probable" degradation in properties if the confidence level of the results was above 50 percent. They concluded that the materials listed below exhibited a "real" degradation in one or more of the strength functions after irradiation to a fast fluence of 1×10^{17} n/cm^2 (>0.5 MeV) at 30 R (-257 C). The numerical value in parentheses indicates the ratio of postirradiation value to the preirradiation value at 30 R. In some cases, the magnitude of the change was minor but all of the results are included. The following designations were used: UTS – ultimate tensile strength, NTS – notch tensile strength, YS – yield strength.

Aluminum 2014 (UTS – 0.93)
Titanium-5Al-2.5Sn, ELI (UTS – 0.98)
René 41 (NTS – 0.95)
Inconel X (NTS – 0.92)
A-286 (NTS – 0.94)
AM 350 (UTS – 0.92, NTS – 0.55)
17-7PH (UTS – 0.75, YS – 0.76)
440 C (UTS – 0.83).

Materials exhibiting "probable" adverse cryogenic irradiation effects on strength functions were:

Aluminum 2024 (UTS – 0.95)
Aluminum 2219 (UTS – 0.98)
Aluminum 6061 (UTS – 0.95)
Aluminum 7079 (UTS – 0.92)
Stainless Steel 304 (NTS – 0.90)
Manganese Steel T-450 (UTS – 0.98).

Materials exhibiting significant reductions in ductility (shown in parentheses) through irradiation were:

Aluminum 7178 (from 12.4 to 6.3 percent)
Aluminum B-750 (from 7.0 to 3.3 percent)
Aluminum A-356 (from 11.6 to 8.7 percent)
Titanium-6Al-4V (annealed) (from 7.6 to 5.7 percent)
Titanium-6Al-4V (aged) (from 6.4 to 5 percent)
Stainless Steel AM-350 (from 11 to 7.3 percent).

Aluminum X-250, 17-7PH stainless steel, and Type 440 C stainless steel exhibited nil ductility at 30 R in the unirradiated condition. Therefore, it was not practical to measure the effect of irradiation on their ductility.

It was concluded that the materials which were irradiated at 30 R to a fast fluence of 1×10^{17} n/cm^2 and mechanically tested at 30 R should be divided into the following three categories:

(1) Unsuitable for application in an irradiation environment at cryogenic temperatures:

Aluminum 1099
Aluminum 7079

Aluminum X-250
Titanium-6Al-4V (annealed)
Titanium-6Al-4V (aged)
Stainless Steel 17-7PH
Stainless Steel 440 C

(2) Materials that showed some deterioration in strength or ductility and should be subjected to additional investigation before extensive use in nuclear cryogenic applications:

Aluminum Alloys – 2014, 2024, 2219, 5086, 5456, 6061, and 7178
Titanium Alloys – 5Al-2.5Sn, Std 1.5Al-2.5Sn ELI, and 8Al-1Mo-1V
Nickel Alloys – René 41 and Inconel X
Stainless Steels – 304, 347, A-286, and AM-350
Other Steels – T-450 and A-353

(3) Materials that behaved satisfactorily as they did not undergo any strength or ductility decreases resulting from irradiation at 30 R:

Titanium Alloy – 55A
Nickel Alloys – K Monel and Inconel
Stainless Steel – 310.

Table 75 also illustrates how annealing at room temperature affects the mechanical properties of alloys irradiated at -196 C. The test results indicate that annealing at room temperature and then testing at -196 C largely restores preirradiation mechanical properties. Testing of the materials irradiated at -196 C, at room temperature indicates that the irradiation has resulted in only minor property changes. In many cases, the mechanical-property changes are within experimental scatter.

Table 76 illustrates the effect of irradiation on the shear strength of various alloys at 40 R. As with other experimental results, there is considerable experimental scatter and only a few statistically significant results. No definite conclusions can be drawn because some alloys show increased shear strength, while others show decreased shear strength.

Tests carried out in hydrogen, helium, and nitrogen atmospheres suggest that the mechanical properties of materials irradiated at cryogenic temperatures are independent of the test atmosphere. Furthermore, it was found that direct extrapolation of mechanical-properties test results from liquid-nitrogen temperature (-196 C) to liquid-hydrogen temperature (-257 C) is not completely reliable. Yet, a fair extrapolation of mechanical properties from -196 C to -257 C can be made for irradiated materials if electrical-resistivity data are available for both temperatures.

TABLE 76. COMPARISON OF PRE- AND POSTIRRADIATION SHEAR STRENGTHS AT 40 R[199]

Materials	Shear Strengths, 1000 psi Control	Standard Deviation	Irradiated	Standard Deviation	Change 1000 Psi	Change Percent
Stainless Steels						
A-286	168.2	12.0	157.0	5.1	-11.1	-7
Type 347	152.4	5.3	151.5	6.0	-0.9	-1
Type 410	149.3	4.1	160.7	8.5	+11.4	+8
Nickel-Base Alloys						
Hastelloy C	170.8	2.6	145.6	1.7	-25.2	-15
Inconel X-750	152.8	1.4	162.0	5.0	+9.2	+6[a]
D-979	152.5	6.0	168.0	18.1	+15.5	+10.2
Aluminum-Base Alloys						
6061-T6	49.8	1.1	52.3	1.6	+2.5	+5.0[a]
7075-T6	95.2	4.8	80.3	3.5	-14.9	-16[a]
A-356-T6	50.4	2.3	45.6	0.6	-4.8	-10[a]
Titanium Alloy						
Ti-5Al-2.5Sn ELI	130.6	4.3	152.5	8.8	+11.9	+9

(a) Statistically significant changes; probability >0.90 percent.

REFRACTORY METALS

Niobium Alloys

Data showing the effect of irradiation on the mechanical properties of niobium alloys are given in Table 77. These data indicate that irradiation at 330 C causes more embrittlement than irradiation at 50 C - even at fast fluences as low as 1 x 10^{18} n/cm^2.[200] Irradiation at 50 C to a fast fluence of about 1 x 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.[187] Material with larger grain size will undergo greater ductility decreases when irradiated, conforming to the Petch relationship.[200] Annealing studies on irradiated niobium plotted in Figure 128 indicate that the preirradiation mechanical properties are restored by an annealing at about 600 C.[201] Preirradiation mechanical properties are recovered for specimens tested at 1090 C by the 30-minute soaking time at test temperature.[168]

The yield strength of niobium-1 weight percent zirconium and D-43 (niobium-10 weight percent tungsten-1 weight percent zirconium-0.1 weight percent carbon) were determined by bend testing after irradiation to a fast fluence of 1 x 10^{20} n/cm^2.[203] It was found that irradiation increased the yield strength of the niobium alloys, but that the irradiated material retained sufficient ductility even when tested at -75 C. Cold-worked niobium-1 weight percent zirconium specimens underwent smaller increases in yield strength, with the welded material behaving similarly to the annealed material. Figures 128 and 129 illustrate the relative irradiation-induced increases in the yield strength of annealed and cold-worked material, along with annealing-induced recovery at 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 covered.

Bend tests on D-43 showed increases in room-temperature yield strength from 117,000 to 193,000 psi due to irradiation.[203] Annealing for 1 hour at 1000 C completely restored the unirradiated yield strength. The effects of cold work, irradiation, and annealing on the yield strength of D-43 are compared in Figures 130 and 131.

Limited in-pile stress-rupture tests have been performed on niobium-1 weight percent zirconium specimens at 980 and 1095 C.[204] These specimens were irradiated in a helium atmosphere in an instantaneous fast flux of 3 x 10^{13} n/(cm^2)(sec). 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 x 10^{20} n/cm^2, while a specimen with a rupture life of 10 hours would receive a fast fluence of 1.08 x 10^{18} n/cm^2. Results of the tests (Figure 132) 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 cubic metals, and, consequently, the possibility of a

TABLE 77. 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		Ultimate Tensile Strength, 1000 psi		Elongation, percent				Reduction in Area, percent		Reference
								Uniform		Total				
				Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Niobium (99.8%)	9.1×10^{17}	77	RT	19.5	42.5	40.0	46.2			51.5	41.5			200
	9.1×10^{17}	77	RT	17.3	40.7	34.8	42.3			53.0	38.5			200
	9.1×10^{17}	77	RT	17.3		34.8	45.6			53.0	37.1			200
	9.1×10^{17}	77	RT	21.4	41.4	30.0	43.1			45.0	32.9			200
	9.1×10^{17}	330	RT	19.5		40.0	55.7			51.5	27.8			200
	9.1×10^{17}	330	RT	19.5	46.0	40.0	52.4							200
	9.1×10^{17}	330	RT	19.5		40.0	54.9			50.0	31.4			200
	9.1×10^{17}	330	RT	21.4	43.8	30.0	47.6			45.0	24.3			200
	9.1×10^{17}	330	RT	21.4		30.0	44.2			45.0	20.8			200
Niobium	1×10^{20}	125-175	RT	59.4	77.2	71.8	80.4			20.6	8.0			201
	1×10^{20}	125-175	RT	59.4	74.1	71.8	74.4			20.6	6.4			201
	1×10^{20}	125-175	RT	59.4	74.4	71.8	77.0			20.6	6.5			201
	1×10^{20}	125-175	RT(b)	59.4	92.2	71.8	92.8			20.6	6.8			201
	1×10^{20}	125-175	RT(b)	59.4	92.2	71.8	92.7			20.6	7.7			201
	1×10^{20}	125-175	RT(b)	59.4	91.2	71.8	91.7			20.6	4.2			201
	1×10^{20}	50	RT	72	91	73	91			11.0	7.0			202
Nb-0.6Zr, S	2×10^{20}	50	RT	67.4	93	71.5	93.5			19	12.8			202
Nb-0.6Zr, R	2×10^{20}	50	RT	75	102	84.7	104			15	11.8			202
	2×10^{20}	50	RT	75	106	84.7	108			15	13.2			202
	2×10^{20}	50	RT	75	99.4	84.7	101.5			15	13.2			202
	2×10^{20}	50	RT(c)		85		87				14.9			
	2×10^{20}	50	RT(d)	69.9	73.5	79.4	79.7			18.6	18.3			
Nb-0.6Zr, S	2×10^{20}	50	RT(e)	59.1	53.6	69.1	63.6			23	18.8			
Nb-0.6Zr, R	2×10^{20}	50	RT(e)		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	187
	8.8×10^{19}	50	RT	55.1	62.7	59.7	64.5	3.2	0.6			78.8	72.0	187
	1.0×10^{20}	50	RT	55.1	62.4	59.7	64.9	3.2	0.6			78.8	72.2	187
	1.0×10^{20}	50	RT	55.1	63.0	59.7	64.2	3.2	0.5			78.8	55.4	187
	1.5×10^{20}	50	RT	55.1	61.4	59.7	63.6	3.2	0.6			78.8	72.3	187
	1.5×10^{20}	50	RT	55.1	64.5	59.7	65.3	3.2	0.6			78.8	80.5	187
Nb-1Zr	5.9×10^{20}	50	1090	9.4	9.1	10.5	9.9	1.4	1.3	52.3	49.2			168
Cb-752	8.8×10^{19}	50	RT	66.7	109	81.9	109.2	14.5	0.8			56.6	58.2	187
	8.8×10^{19}	50	RT	66.7	101.4	81.9	104.9	14.5	0.7			56.6	51.2	187
	1.0×10^{20}	50	RT	66.7	98.5	81.9	104.2	14.5	1.6			56.6	57.6	187
	1.0×10^{20}	50	RT	66.7	104.6	81.9	106.5	14.5	0.8			56.6	58.7	187
	1.5×10^{20}	50	RT	66.7	106.8	81.9	107.0	14.5	0.9			56.6	60.3	187
	1.5×10^{20}	50	RT	66.7	107.8	81.9	108.1	14.5	0.9			56.6	59.0	187
	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	168

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

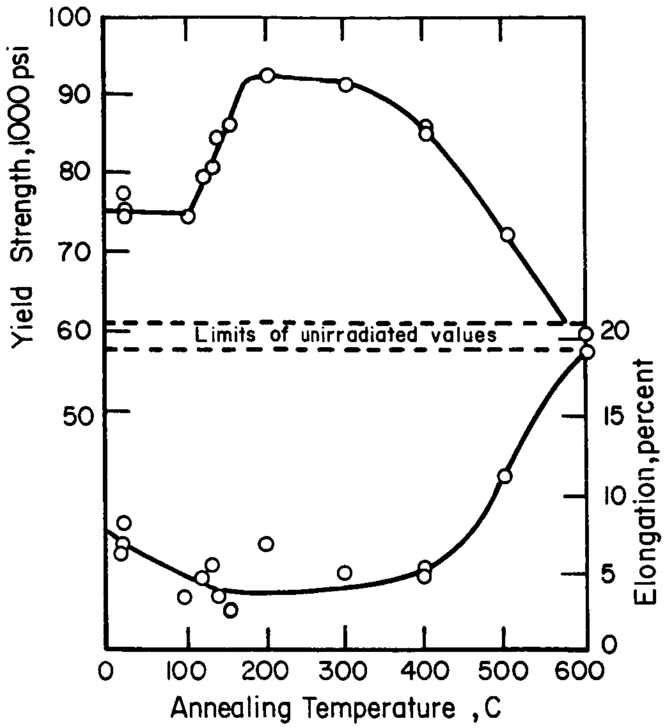


FIGURE 127. EFFECTS OF POSTIRRADIATION ANNEALING ON THE ROOM TEMPERATURE TENSILE PROPERTIES OF NIOBIUM(201)

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

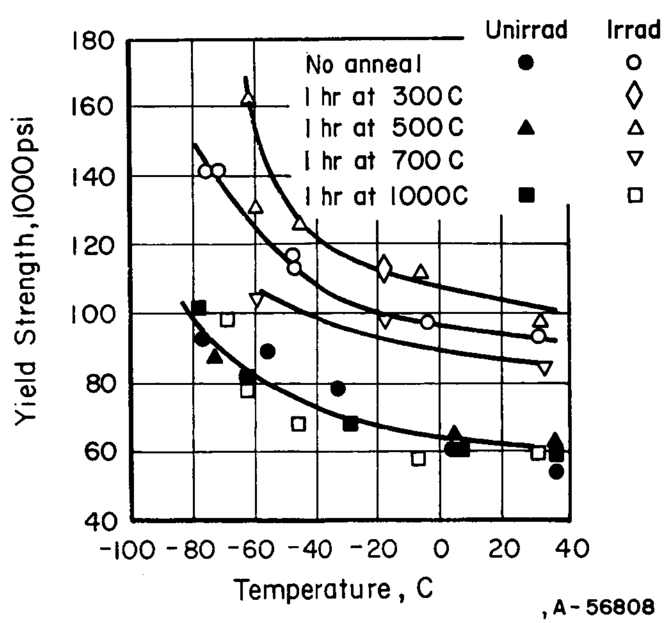


FIGURE 128. YIELD STRESS OF RECRYSTALLIZED NIOBIUM-1 WEIGHT PERCENT ZIRCONIUM BEFORE AND AFTER IRRADIATION(203)

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

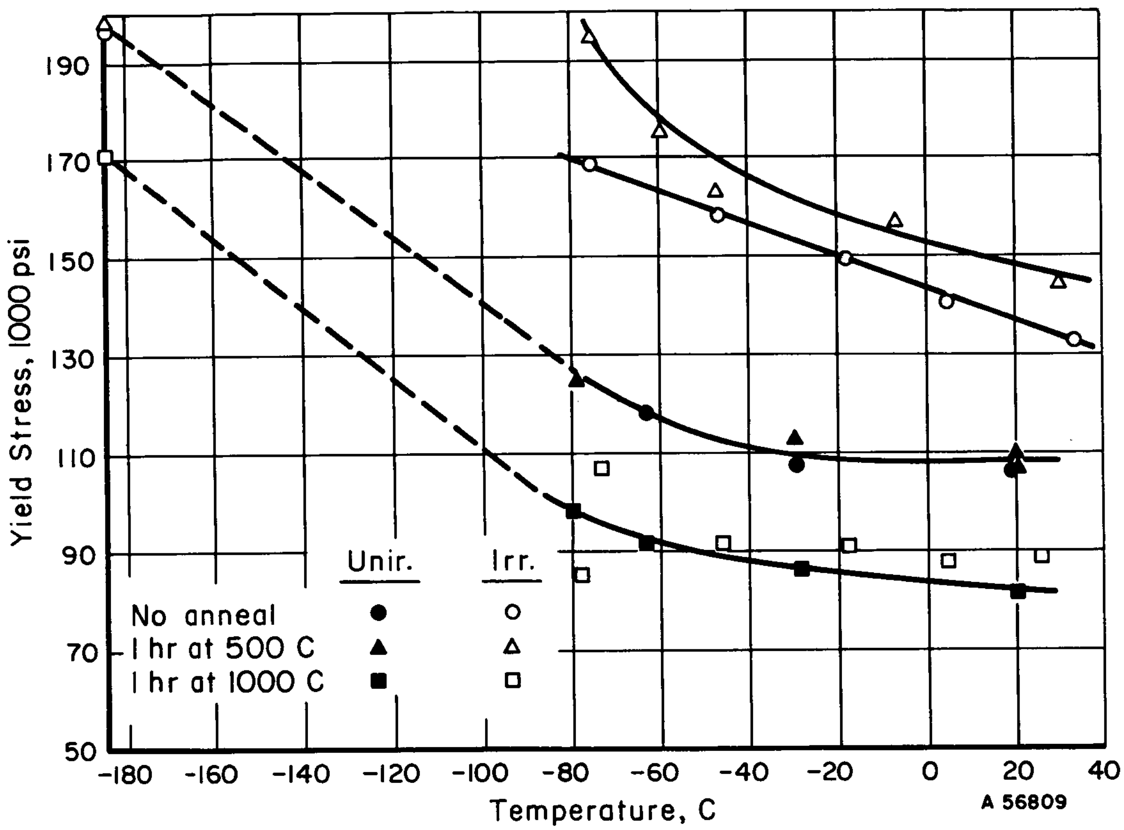


FIGURE 129. YIELD STRESS OF COLD-WORKED NIOBIUM-1 WEIGHT PERCENT ZIRCONIUM BEFORE AND AFTER IRRADIATION[203]

Specimens received fast fluences of 0.6 to 6.4 x 10^{19} n/cm^2.

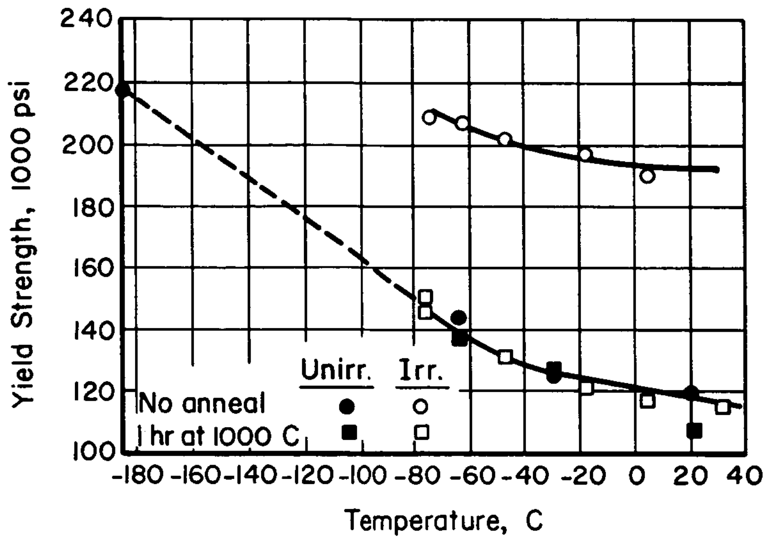


FIGURE 130. YIELD STRESS OF RECRYSTALLIZED D-43 ALLOY BEFORE AND AFTER IRRADIATION(203)

Specimens received fast fluences of 0.6 to 6.4 x 10^{19} n/cm^2.

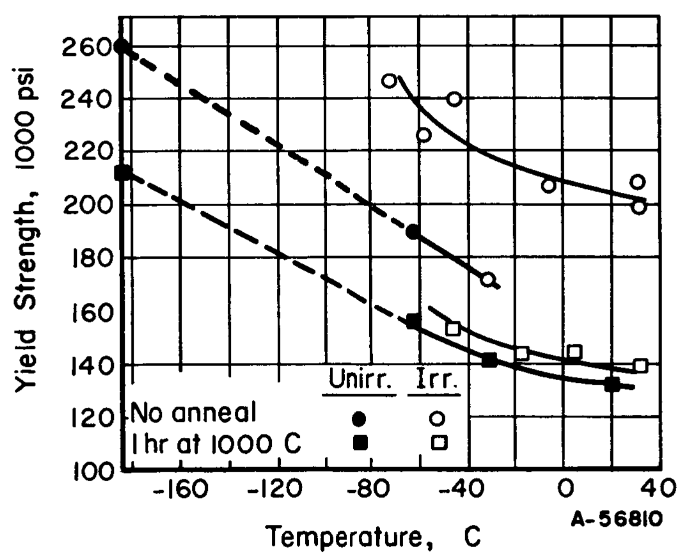


FIGURE 131. YIELD STRESS OF COLD-WORKED D-43 ALLOY BEFORE AND AFTER IRRADIATION(203)

Specimens received fast fluences of 0.6 to 6.4 x 10^{19} n/cm^2.

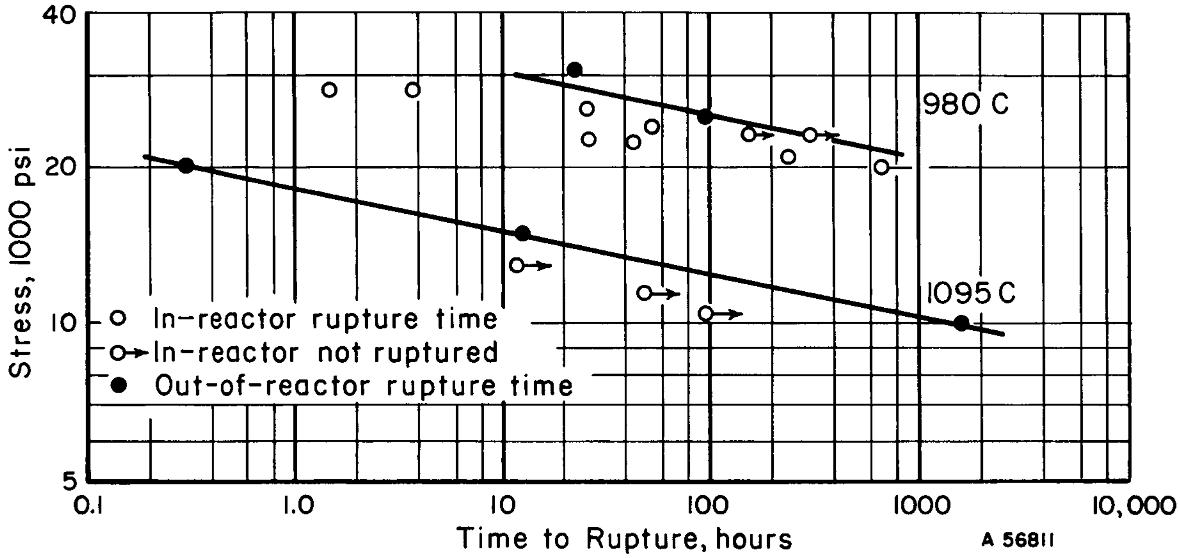


FIGURE 132. EFFECT OF NEUTRON BOMBARDMENT ON THE STRESS-RUPTURE STRENGTH OF NIOBIUM-1 PERCENT ZIRCONIUM ALLOY AT 1800 AND 2000 F[204]

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

transition-temperature increase by irradiation is of utmost importance. The irradiation-induced transition-temperature shift measured by various investigators is tabulated in Table 78. These somewhat conflicting results 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.

TABLE 78. 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	205
Molybdenum[b]	4×10^{19}	-110	-110	0	206
Molybdenum[c]	5×10^{19}	-136	-73	63	207
Molybdenum[a]	5×10^{20}	-30	+70	100	205
Mo-0.5Ti	1×10^{20}	19	138	119	203

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

The limited tensile data on irradiated molybdenum and its alloys are given in Table 79 and Figure 133. 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. Figure 134 illustrates how preirradiation annealing, irradiation, and postirradiation aging affect the mechanical properties of molybdenum.[209] These materials contained about 100 ppm of impurities, including 20 ppm of carbon. These results show that recrystallization of molybdenum makes it very susceptible to irradiation embrittlement. However, it is possible that the specimens were oxygen-contaminated during the 1500 C anneal. 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 and decreases ductility with increasing annealing temperature, eventually reaching a maximum thermal hardening temperature of about 500 C (Figure 135). Annealing at higher temperatures decreases room-temperature strength, and the preirradiation strength properties are restored by annealing at 800 C.[210] However, the annealing temperature necessary for removal of irradiation-induced property changes is very sensitive to either the material properties or fast fluence. Figure 133 shows that an anneal at 1000 C is necessary for restoration of tensile properties; Figure 136 shows that an anneal at 1200 C is necessary for restoration of preirradiation hardness properties.[192] These annealing phenomena are substantiated by tensile tests performed on irradiated molybdenum at 1090 C, which indicated no difference in mechanical properties between the irradiated and unirradiated material. However, testing at 1090 C does not restore the unirradiated mechanical-property values for TZM (molybdenum-0.5 weight percent titanium-0.08 weight percent zirconium-0.03 weight percent carbon).[168] Certain characteristics of the irradiated molybdenum-50 weight percent rhenium alloy

TABLE 79. 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			207
	5×10^{19}	100	80	80	93.3	90.5	93.5			23.8	18.5			207
	5×10^{19}	100	200	70.4	85.5	74.6	85.9			2.8	5.8			207
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	168
	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	168
Commercial molybdenum	0		-60	142		142				0		0		205
	0		-40	123		123				0		0		205
	0		-20	125		120				32.8		63.8		205
	5.1×10^{20}	90	RT	102.5	151.7	100.8	151.7			45.7	0	72.4	0.1	205
	5.1×10^{20}	90	RT	93.8		98.8	109.7			41.7	0	65	0	205
	5.8×10^{20}	90	60				148.5				0		0	205
	5.8×10^{20}	90	80		143.5		143.5				14.7		60.7	205
	5.8×10^{20}	90	100		111.5		111.5				10		59.7	205
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	168
Mo-50Re sheet	3.4×10^{19}	70	RT	115	178	142	179			28.5	13.1			208
	4.2×10^{19}	70	RT	115	212	143	214			28.5	8.6			208
	3.4×10^{19}	70	RT[b]	116	173	141	174			31.0	17.3			208
	4.2×10^{19}	70	RT[b]	116	197	141	206			31.0	9.2			208
	3.4×10^{19}	70	RT[c]	111	119	142	147			31.0	28.3			208
	4.2×10^{19}	70	RT[c]	111	134	142	155			31.0	24.0			208
	3.4×10^{19}	70	RT[d]	109	115	141	147			30.3	28.0			208
	4.2×10^{19}	70	RT[d]	109	117	141	146			30.3	28.0			208
Ta	1×10^{20}	50	RT			128	148			8.6	7			187
Ta-1.5W	3.9×10^{20}	50	RT	31.0	65.8	44.9	69.5			39	16			219
Ta-3.0W	8×10^{20}	50	RT	38.5	81.4	52.5	86.3			35	7			219
Ta-10W	3×10^{20}	50	RT			166	290			7.4	4.9			187

(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.

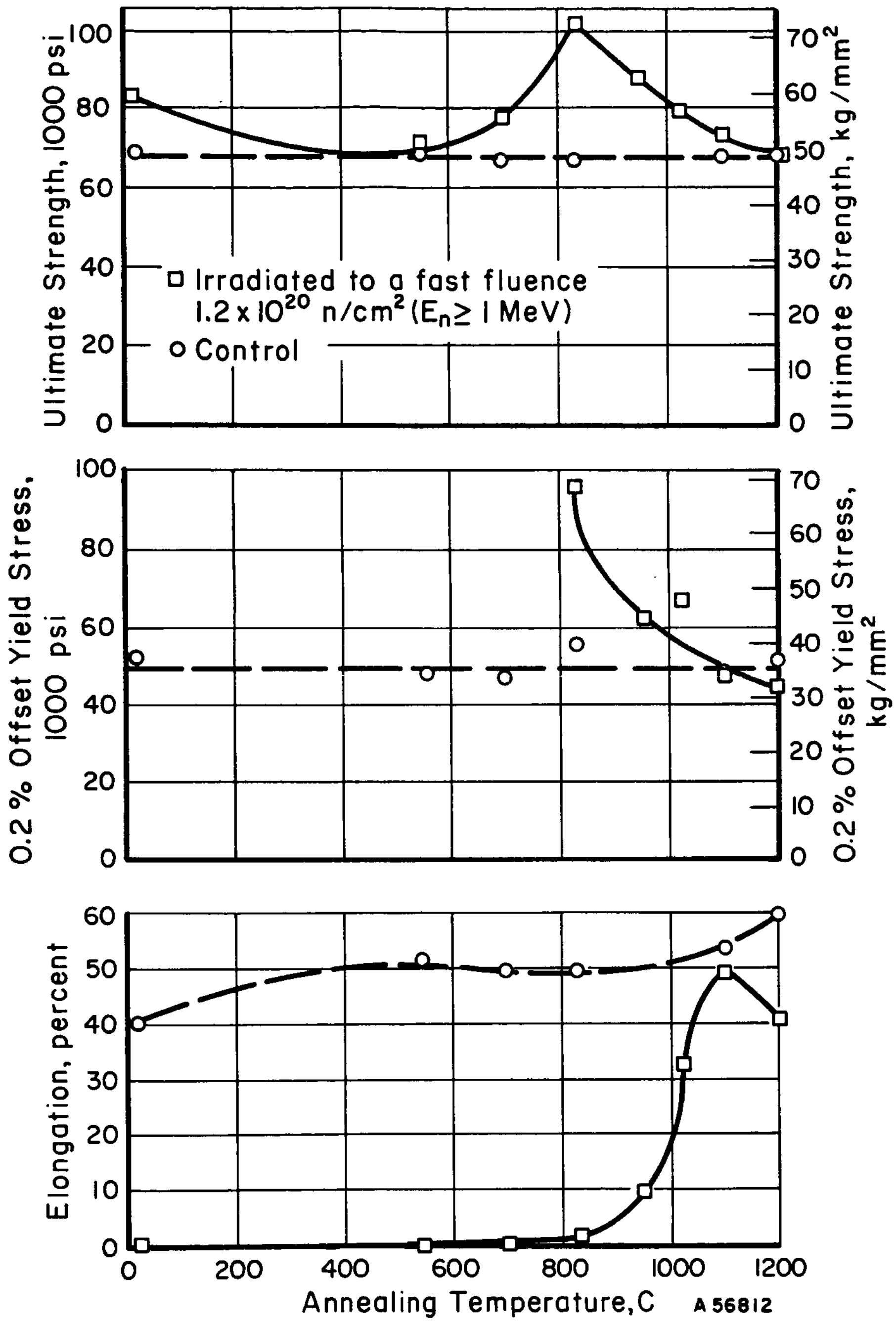


FIGURE 133. ROOM-TEMPERATURE TENSILE PROPERTIES OF MOLYBDENUM SHEET SPECIMENS VERSUS POST-IRRADIATION ANNEALING TEMPERATURE(192)

All anneals for 1 hour at indicated temperature.

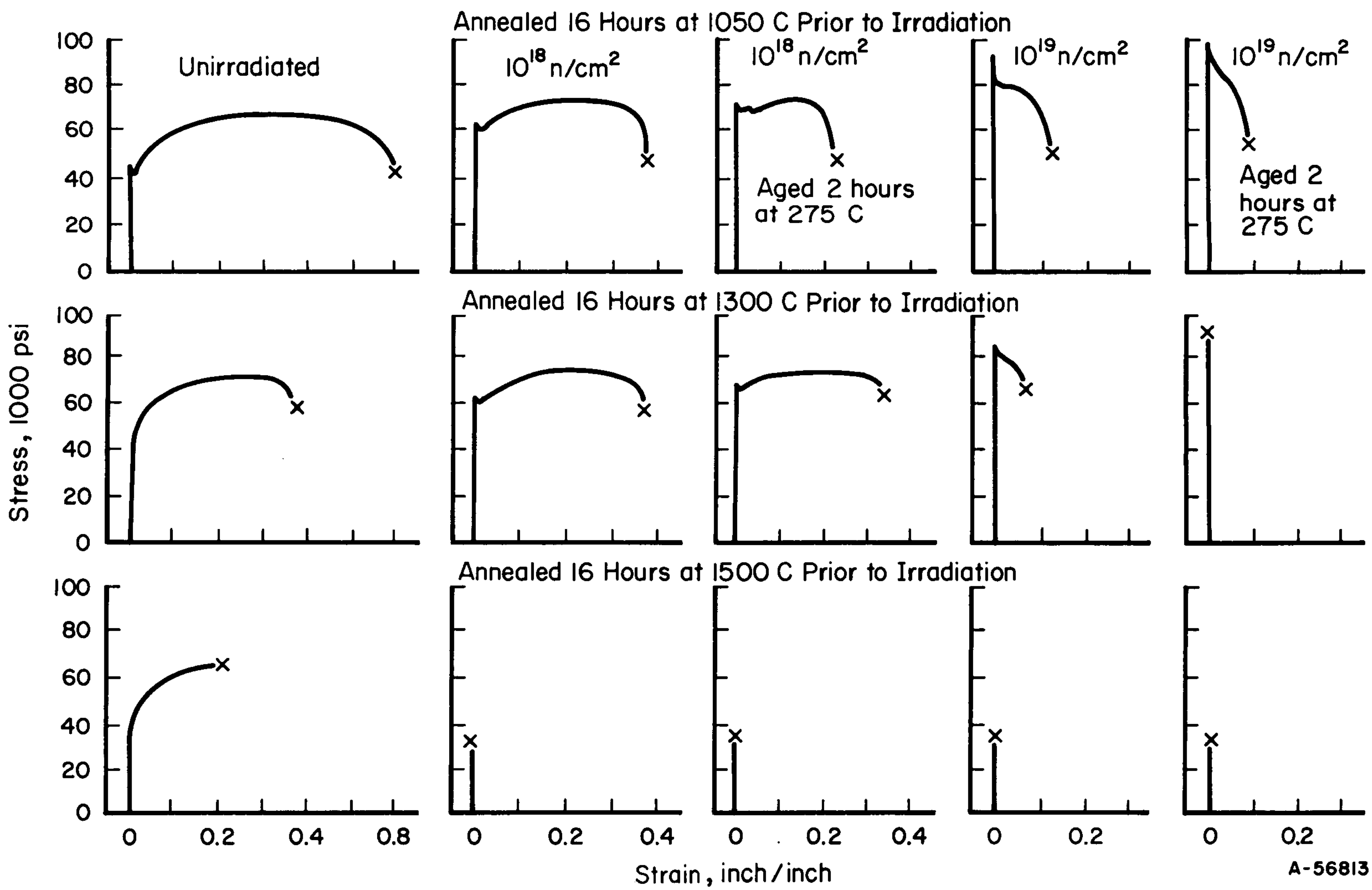


FIGURE 134. EFFECT OF IRRADIATION ON TENSILE PROPERTIES OF MOLYBDENUM CONTAINING LESS THAN 100 PPM OF IMPURITIES(209)

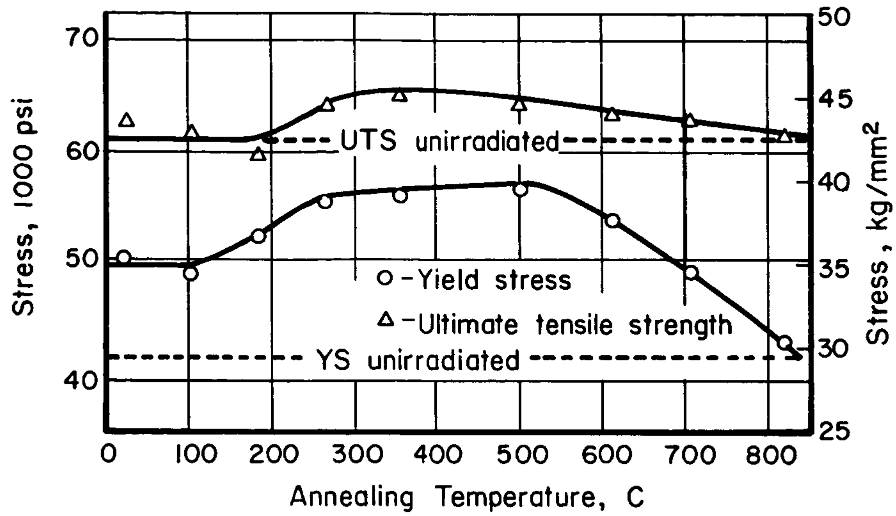


FIGURE 135. ROOM TEMPERATURE YIELD STRESSES AND ULTIMATE TENSILE STRENGTHS OF IRRADIATED MOLYBDENUM SPECIMENS AS A FUNCTION OF ANNEALING TEMPERATURE[210]

Irradiated to an epithermal fluence of 2 x 10^{18} n/cm^2.

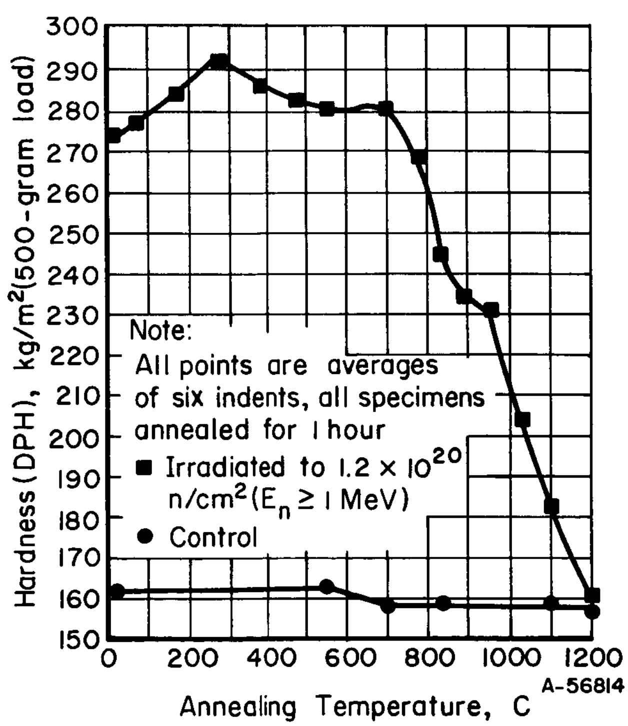


FIGURE 136. ROOM-TEMPERATURE MICROHARDNESS OF MOLYBDENUM SHEET SPECIMENS AS A FUNCTION OF POSTIRRADIATION ANNEALING TEMPERATURE[192]

cause a reduction in the irradiation-induced strength increase on annealing at temperatures as low as 250 C.(208) Annealing at 400 C further reduces the irradiation-induced strength increase, whereas annealing at 700 C completely restores preirradiation mechanical properties. These results indicate that the thermal hardening temperature for irradiated molybdenum-50 weight percent rhenium must exist below 250 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.(211) 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. The 560 to 650 C temperature range was chosen because annealing of irradiated molybdenum at 640 C (0.31 Tm*) sharply increases electrical resistivity. This increased resistivity is believed to be due to the presence of an irradiation-produced defect whose stability is favored by 0.31 Tm. Another reason for testing in the 560 to 650 C range was to determine the degree of recovery in irradiation effects at those testing temperatures. Results of these creep-rupture tests on irradiated molybdenum are shown in Figure 137. 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.

Creep-rupture tests have been performed on TZM specimens irradiated to a fast fluence of 6.8×10^{19} n/cm^2.(130) 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. The results of all the stress-rupture tests on refractory metals are given in Table 80.

Tantalum

Only minor interest has been shown in using tantalum as a structural material in reactors. Consequently, only limited data concerning the effect of irradiation on its mechanical properties are available. Results of the few tensile tests on irradiated tantalum and its alloys are given in Table 79. 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.(219) The data given in Table 79 indicate that the effects of solid-solution strengthening caused by the transmuted tungsten atoms is minor compared to fast-neutron-caused effects.

*Tm is defined as the melting temperature (absolute scale).

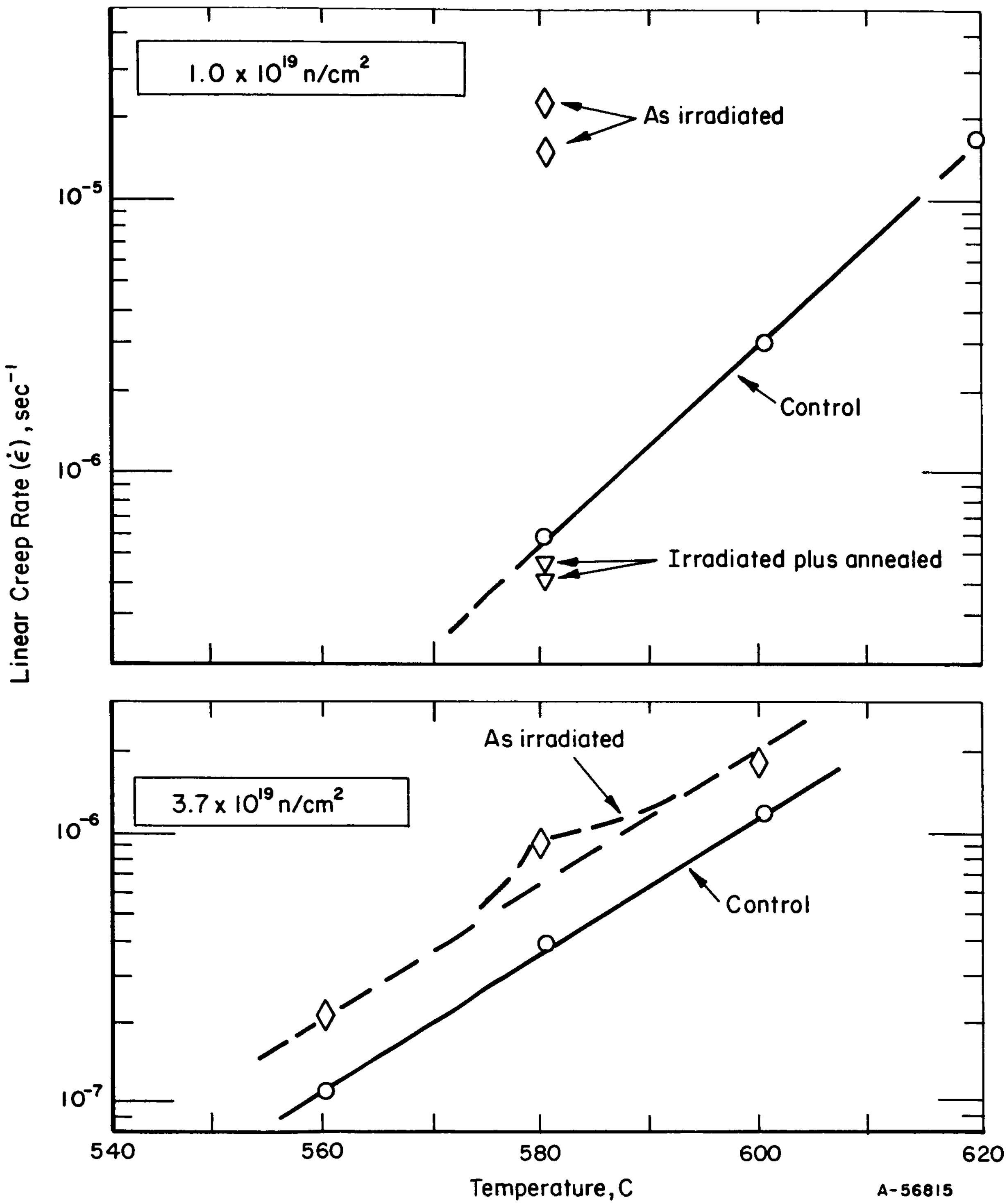


FIGURE 137. CREEP RATE OF RECRYSTALLIZED MOLYBDENUM IRRADIATED TO FAST FLUENCES OF 1.0×10^{19} AND 3.7×10^{19} N/CM^2(211)

The anneal consisted of heating the specimens for 1 hour at 770 C. The initial load in all tests was 30,000 psi.

TABLE 80. CREEP-RUPTURE DATA ON IRRADIATED REFRACTORY METALS

Material	Fast Fluence, n/cm2 (>1 MeV)	Irradiation Temp, C	Test Temp, C	Stress, 1000 psi	Rupture Life, hours Unirr.	Rupture Life, hours Irr.	Linear Creep Rate, sec^{-1} Unirr.	Linear Creep Rate, sec^{-1} Irr.	Elongation, percent Unirr.	Elongation, percent Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Reference
Tungsten	1.6×10^{19}	70	900	32.2	12.45	47.48	2.32×10^{-6}	5.48×10^{-7}	36.4	26.8	96.6	96.5	212
	1.6×10^{19}	70	900(a)	32.2	12.45	77.04	2.32×10^{-6}	5.08×10^{-7}	36.4	32.5	96.6	96.8	212
	1.6×10^{19}	70	900(b)	32.2	12.45	124.6	2.32×10^{-6}	1.96×10^{-7}	36.4	30.0	96.6	96.6	212
	1.6×10^{19}	70	900(c)	32.2	12.45	36.65	2.32×10^{-6}	8.97×10^{-7}	36.4	33.6	96.6	96.2	212
	1.6×10^{19}	70	900(d)	32.2	12.45	32.20	2.32×10^{-6}	1.01×10^{-6}	36.4	35.3	96.6	93.7	212
	1.6×10^{19}	70	900(e)	32.2	12.45	24.13	2.32×10^{-6}	1.26×10^{-6}	36.4	33.9	96.6	94.8	212
	3.2×10^{19}	70	1100	52.9	2.64	4.10	4.0×10^{-6}	2.61×10^{-6}	7.5	7.6			213
	3.2×10^{19}	70	1100	50.9	3.42	7.00	2.62×10^{-6}	1.65×10^{-6}	6.9	8.5			213
	3.2×10^{19}	70	1100	49	4.93	8.89	1.85×10^{-6}	1.4×10^{-6}	7.0	10.1			213
	3.7×10^{18}	70	1100	28.5	1.25	1.19	3.51×10^{-5}	3.68×10^{-5}	33.7	32.9	97.8	97.6	214
	3.7×10^{18}	70	1100	28.9	3.00	3.47	1.19×10^{-5}	1.24×10^{-5}	26.6	27.3			214
	3.7×10^{18}	70	1100	28.9	3.00	3.59		1.19×10^{-5}		24.3			214
	3.7×10^{18}	70	1100	28.9	2.27	2.55	1.72×10^{-5}	1.37×10^{-5}	29.1	24.3			214
	2.5×10^{19}	70	1100	28.9	1.25	2.83	3.51×10^{-5}	2.01×10^{-3}	33.7	32.1	97.8		214
	2.5×10^{19}	70	1100	28.9	3.00	7.85	1.19×10^{-5}	4.52×10^{-6}	26.6	24.9			214
	2.5×10^{19}	70	1100	28.9	2.27	6.74	1.72×10^{-5}	5.68×10^{-6}	28.7	28.1			214
	3.7×10^{18}	70	1100	27.5	5.85	5.65	7.37×10^{-6}	7.7×10^{-6}	30.8	31.2	97.0	97.8	214
	1.6×10^{19}	70	1100(a)	27.4	7.23	8.96	5.25×10^{-6}	4.46×10^{-6}	30.3	27.6	94.8	86.1	215
	2.5×10^{19}	70	1100	27.5	5.85	10.2	7.37×10^{-6}	4.18×10^{-6}	30.8	30.3	97.8	96.3	214
	5.8×10^{19}	70	1100	27.5	5.85	6.74	7.37×10^{-6}	6.36×10^{-6}	30.8	31.5	97.0	95.7	214
	8.2×10^{19}	70	1100(a)	27.4	7.23	37.6	5.25×10^{-6}	6.87×10^{-7}	30.3	27.5	94.8	91.3	215
	8.5×10^{17}	70	1100	26.0	7.72	6.52	5.35×10^{-6}	6.26×10^{-6}	31.2	33.9	98.5	95.9	208
	4.0×10^{18}	70	1100	26.0	7.99	12.65	4.46×10^{-6}	3.87×10^{-6}		35.7		90.0	208
	4.0×10^{18}	70	1100	26.0	7.72	12.61	5.35×10^{-6}	3.60×10^{-6}	31.2	28.0	98.5	90.0	208
	5.9×10^{18}	70	1100	26.0	7.72	15.92	5.35×10^{-6}	3.0×10^{-6}	31.2	32.3	98.5	95.1	208
	5.9×10^{18}	70	1100	26.0	7.99	15.22	4.46×10^{-6}	2.86×10^{-6}		29.8		94.9	208
	1.6×10^{19}	70	1100	26.0	20.82	26.63	1.80×10^{-6}	1.52×10^{-6}	31.1	27.9	94.5	93.4	215
	3.2×10^{19}	70	1100(b)	26.2	2.68	4.08	1.53×10^{-6}	1.41×10^{-5}	39.7	38.7			213
	3.8×10^{19}	70	1100	26.0	7.99	43.98	4.46×10^{-6}	1.20×10^{-6}		24.4		92.9	208
	3.8×10^{19}	70	1100	26.0	7.72	41.92	5.35×10^{-6}	1.08×10^{-6}	31.2	26.3	98.5	92.3	208
	3.9×10^{19}	70	1100	26.0	7.99	21.04	4.46×10^{-6}	5.0×10^{-7}		14.0		93.5	208
	3.9×10^{19}	70	1100	26.0	7.72	14.64	5.35×10^{-6}	3.58×10^{-6}	31.2	28.0	98.5	90.0	208
	7.9×10^{19}	70	1100	26.0	7.99	58.38	4.46×10^{-6}	1.23×10^{-7}		12.2		89.0	208
	8.2×10^{19}	70	1100	26	20.82	122.36	1.80×10^{-6}	4.43×10^{-6}	31.1	29.1	94.5	89.2	215
	8.2×10^{19}	70	1100(a)	26	20.82	63.76	1.80×10^{-6}	6.38×10^{-7}	31.1	29.1	94.5	90.6	215
	1.2×10^{20}	70	1100	26.0	7.99	352.1	4.46×10^{-6}	2.66×10^{-8}		28.9		87.4	208

TABLE 80. (Continued)

Material	Fast Fluence, n/cm^2 (>1 MeV)	Irradiation Temp, C	Test Temp, C	Stress, 1000 psi	Rupture Life, hours Unirr.	Rupture Life, hours Irr.	Linear Creep Rate, sec^{-1} Unirr.	Linear Creep Rate, sec^{-1} Irr.	Elongation, percent Unirr.	Elongation, percent Irr.	Reduction in Area, percent Unirr.	Reduction in Area, percent Irr.	Reference
	1.2×10^{20}	70	1100	26.0	7.72	194.9	5.35×10^{-6}	2.93×10^{-7}	31.2	33.5	98.5	92.4	208
	1.6×10^{19}	70	1100(a)	25	45.76	54.22	8.62×10^{-7}	6.87×10^{-7}	25.8	27.5	93.2	91.3	215
	3.7×10^{18}	70	1200	25.0	2.10	2.3	1.94×10^{-5}	2.13×10^{-5}	26.9	28.0	96.4	94.8	214
	5.8×10^{17}	70	1200	24	2.10	2.10	1.94×10^{-5}	2.29×10^{-6}	26.9	32.0			214
	5.8×10^{17}	70	1400	18.5	2.60	2.55			23.0	24.2	63.6	64.7	214
	5.8×10^{17}	70	1400	17.5	4.15	3.95			25.2	26.4	65.5	54.5	214
	3.7×10^{18}	70	1400	18.5	4.00	3.95	2.34×10^{-6}		16.3	18.0			214
	3.7×10^{18}	70	1400	18.5	4.00	3.95	2.34×10^{-6}		16.3	18.0			214
	3.7×10^{18}	70	1400	18.5	4.00	4.37	2.34×10^{-6}		16.3	18.5			214
	8.2×10^{19}	70	1400	15	18.64	26.71	2.84×10^{-7}	2.74×10^{-7}	11.8	15.1	35.6	26.7	214
	8.2×10^{19}	70	1400	14	33.67	41.93	1.59×10^{-7}	1.23×10^{-7}	11.4	11.1	33.3	17.4	215
	8.2×10^{19}	70	1500	12	18.28	20.10	2.07×10^{-7}	1.90×10^{-7}	8.6	6.1	21.5	16.2	215
	8.2×10^{19}	70	1700	8	12.57	13.67	2.39×10^{-7}	2.07×10^{-7}	7.5	5.4	8.0	10.4	215
	8.2×10^{19}	70	1700	7	30.8	24.6	1.38×10^{-7}	1.43×10^{-7}	5.3	5.4	13.8	13.6	215
W-26 Re	3.4×10^{19}	70	800	110	24.2	382.2	1.95×10^{-6}	6.31×10^{-8}		30.4			216
	3.4×10^{19}	70	800(f)	110		139		2.19×10^{-7}		30.5			216
	3.4×10^{19}	70	800(g)	107.5	30.3		1.18×10^{-6}	1.53×10^{-8}	28.8				216
			842	107.5		388		2.55×10^{-7}		25.6			
	4.2×10^{19}	70	800(g)	107.5	30.3		1.18×10^{-6}	1.35×10^{-8}	28.8				216
			840	107.5		598		2.13×10^{-8}		(k)			
	1.2×10^{19}	70	800	100	182	283.1	2.35×10^{-7}	1.31×10^{-7}	33.5	32.6			216
	1.2×10^{19}	70	800(h)	100		56.8		5.83×10^{-7}					216
	3.4×10^{19}	70	900	105	1.3	1.75	3.00×10^{-5}	2.5×10^{-5}	41.0	35.7			216
	3.4×10^{19}	70	900	100	3.87	8.63	1.34×10^{-5}	6.79×10^{-6}	33.1	38.2			216
	4.2×10^{19}	70	900(i,j)	100				7.69×10^{-8}					216
			900	105		698		2.63×10^{-7}		24			
W-25Re	8×10^{18}	70	1100	70	3.99	3.98	1.85×10^{-5}	1.52×10^{-5}	54.4	39.9			217
	8×10^{18}	70	1100	70	3.84	4.56			41.1	39.1			217
	8×10^{18}	70	1100	70	6.92	6.73	9.85×10^{-6}	9.78×10^{-6}	53.5	30.7			216
W-26Re	3.2×10^{19}	70	1100	67.5	2.90	3.13	2.28×10^{-5}	2.03×10^{-5}	57.1	45.6			216
W-25Re(i)	3.7×10^{19}	70	1100	67.5	2.90	3.13	2.28×10^{-5}	2.03×10^{-5}	57.1	45.6			215
W-26Re	4.2×10^{19}	70	1100	67.5	2.90	5.30	2.28×10^{-5}	1.37×10^{-5}	57.1	47.1			216
W-25Re(a)	8×10^{19}	70	1100	67.5		5.30		1.37×10^{-5}		47.1			166
W-25Re	8×10^{18}	70	1100	65	8.40	9.08	7.42×10^{-6}	6.75×10^{-6}	52.1	41.0			217

TABLE 80. (Continued)

Material	Fast Fluence, n/cm^2 (>1 MeV)	Irradiation Temp, C	Test Temp, C	Stress, 1000 psi	Rupture Life, hours Unirr.	Irr.	Linear Creep Rate, sec^{-1} Unirr.	Irr.	Elongation, percent Unirr.	Irr.	Reduction in Area, percent Unirr.	Irr.	Reference
W-25Re	8×10^{18}	70	1100	65	9.08	10.11	7.40×10^{-6}	6.51×10^{-6}	54.9	55.1			217
	8×10^{18}	70	1100	65	7.60	7.84	6.50×10^{-6}	5.80×10^{-6}	33.7	48.8			217
W-25Re(i)	3.7×10^{19}	70	1100	65	3.94	3.61	1.65×10^{-5}	1.73×10^{-5}	49.8	47.2			215
W-25Re(a)	8×10^{19}	70	1100	65	6.20	9.62	1.18×10^{-5}	5.67×10^{-6}	60.3	26.7			166
W-26Re	1.2×10^{19}	70	1100	62.5	7.03	10.7	9.23×10^{-6}	6.22×10^{-6}	53.3	45.1			216
W-25Re(i)	3.7×10^{19}	70	1100	62.5	7.02	10.70	9.23×10^{-5}	6.22×10^{-6}	53.3	45.1			215
W-26Re	4.2×10^{19}	70	1100	62.5	6.20	9.62	1.18×10^{-5}	5.67×10^{-6}	60.3	26.7			216
	1.2×10^{19}	70	1100	60		13.05		5.00×10^{-6}		54			216
W-25Re(i)	3.7×10^{19}	70	1100	60		13.05		5.00×10^{-6}		54.0			215
W-26Re	4.2×10^{19}	70	1100	57	17.09	25.07	4.54×10^{-6}	2.99×10^{-6}	66.1	37.7			216
W-25Re(a)	8×10^{19}	70	1100	57	17.09	25.07	4.54×10^{-6}	2.99×10^{-6}	66.1	37.7			166
Rhenium	8×10^{18}	70	1100	30			1.58×10^{-5}	1.22×10^{-5}					217
Molybdenum	3.7×10^{19}	70	560	30	487.9	237.7	1.01×10^{-7}	2.14×10^{-7}	56.9	51.9			218
	3.7×10^{19}	70	580	30	115.6	41.3	3.59×10^{-7}	1.13×10^{-6}	52.3	53.1			218
	1×10^{19}	70	580	28.5	68.20	3.62	5.78×10^{-7}	1.47×10^{-5}	53.4	51.4			212
	1×10^{19}	70	580(l)	28.5		110.48		4.22×10^{-7}		42.2			212
	1×10^{19}	70	580(m)	28.5		75+		4.16×10^{-7}					212
	3.7×10^{19}	70	600		40.3	27.3	1.03×10^{-6}	1.62×10^{-6}	49.8	49.7			218
	3.7×10^{19}	70	650		5.74	3.01	7.70×10^{-6}	1.58×10^{-5}	51.8	55.8			218
	6.9×10^{18}	70	870	22.0	3.55	3.38	2.20×10^{-5}	2.55×10^{-5}	53.1	55.5			213
	6.9×10^{18}	70	870	21.0	6.63	4.84	1.25×10^{-5}	1.70×10^{-5}	60.6	57.5			213
	6.9×10^{18}	70	980	15.0	4.40	4.85	1.77×10^{-5}	1.72×10^{-5}	67.9	77.2			213
TZM	6.8×10^{19}	70	1310	16		18.61		4.57×10^{-6}		49.8			130
	6.8×10^{19}	70	1310	15	16.64	17.95	3.89×10^{-6}	3.02×10^{-6}	41.4	45.9			130
	6.8×10^{19}	70	1310	14	17.97	19.3	2.28×10^{-6}	2.36×10^{-6}	39.5	39.2			136

(a) Annealed at 1050 C.
(b) Annealed at 1200 C.
(c) Annealed at 1300 C.
(d) Annealed at 1400 C.
(e) Annealed at 1700 C.
(f) Annealed at 1500 C for 30 hours in hydrogen.
(g) Tested at two different temperatures.
(h) Annealed at 1700 C for 30 hours in hydrogen.
(i) Annealed at 1700 C for 1 hour.
(j) Tested at two different stresses.
(k) Test discontinued before fracture.
(l) Annealed at 757 C.
(m) Annealed at 779 C.

The effect of a predominantly fast fluence on the hardness of tantalum has been determined (Table 81).[220]

TABLE 81. EFFECT OF NEUTRON IRRADIATION ON THE ROOM-TEMPERATURE HARDNESS OF TANTALUM IRRADIATED AT 550 C[220]

Fast Fluence, 10^{20} n/cm^2	Hardness, VHN
0	75
6.4	195
8.8	249
9.6	190
10.2	273

Annealing of the irradiated tantalum at 600, 700, 800, and 900 C resulted in hardness changes, with a hardness peaking at 800 C and decreasing after annealing at 900 C. This indicates that maximum thermal hardening occurs in irradiated tantalum at about 800 C.

Tungsten Alloys

Results of the few tensile tests performed on irradiated tungsten and tungsten alloys are shown in Table 82 and Figure 138. The low-temperature tensile tests on the irradiated material produced rather unusual results since the ductility was found to be improved by irradiation.[207] Tensile tests at 400 C indicate a drastic ductility decrease after a fast fluence of 1 to 2 x 10^{19} n/cm^2[212] (Figures 138 and 139). This ductility decrease was found to be more drastic for tungsten shielded by cadmium foil. This foil 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 140 illustrates the increase in 0.05 percent yield strength as a function of fast fluence. 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. Similar yield-strength increases as a function of square root of fluence have been noted for other metals.[95b] The tensile tests at 1090 C indicate that not all of the irradiation-induced property changes in tungsten-26 weight percent 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.[130] The effect of annealing on microhardness of irradiated tungsten is shown in Figure 141. Similar results have been reported for tensile tests performed on irradiated tungsten annealed at 240 to 1100 C. In these tests, the maximum yield strength was determined for irradiated tungsten specimens annealed at 750 C. Complete recovery of preirradiation strength values was obtained after an anneal at 1100 C, but the specimens still exhibited a ductility decrease of 66 percent from the preirradiation ductility.[208]

TABLE 82. 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			207
	5×10^{19}	100	200	148.0	131.0	173.2	173.0			2.4	4.2			207
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	168
	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	168
	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	168

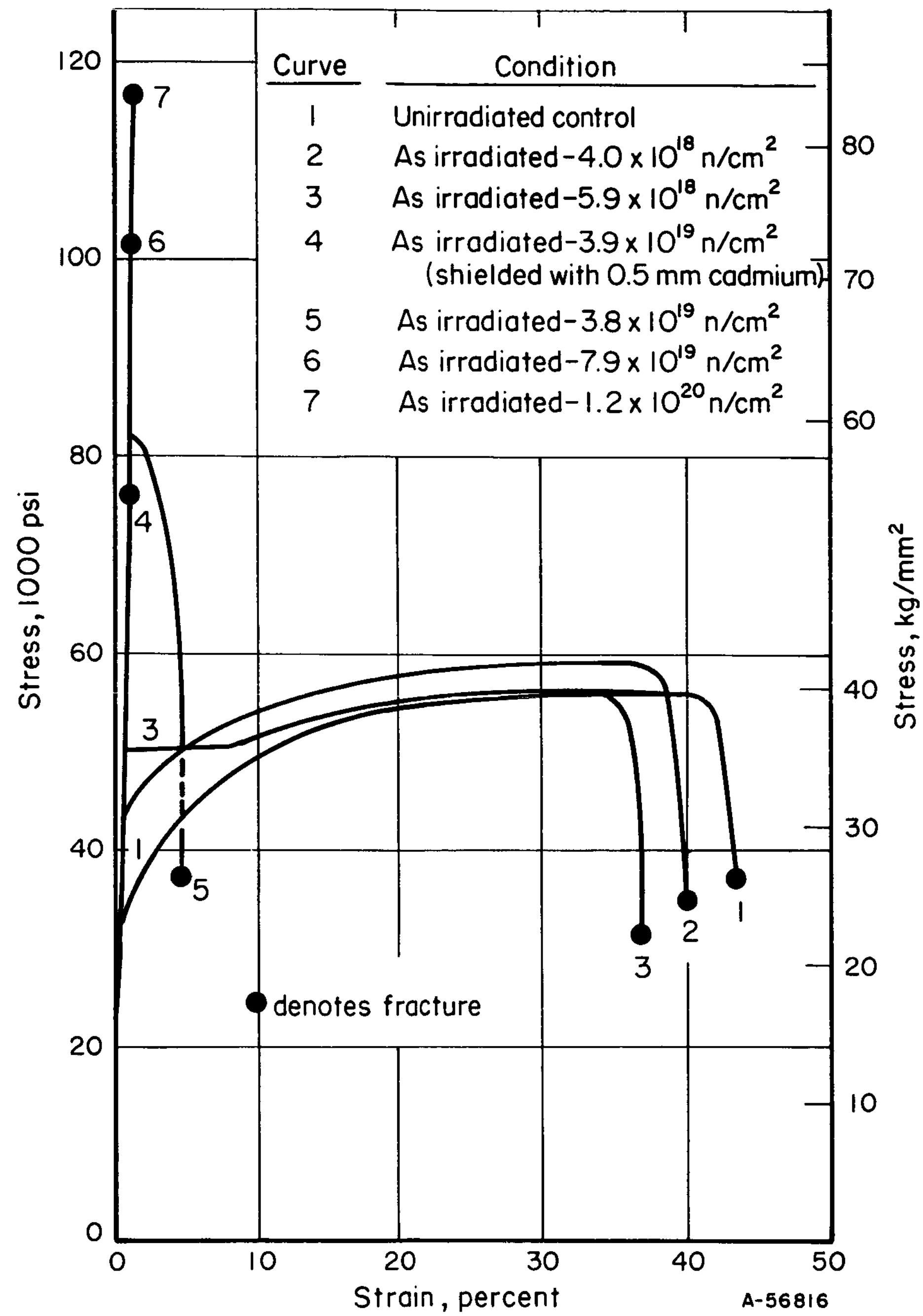


FIGURE 138. STRESS VERSUS STRAIN CURVES FOR CONTROL AND IRRADIATED TUNGSTEN SPECIMENS TESTED AT 400 C IN HELIUM(211)

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

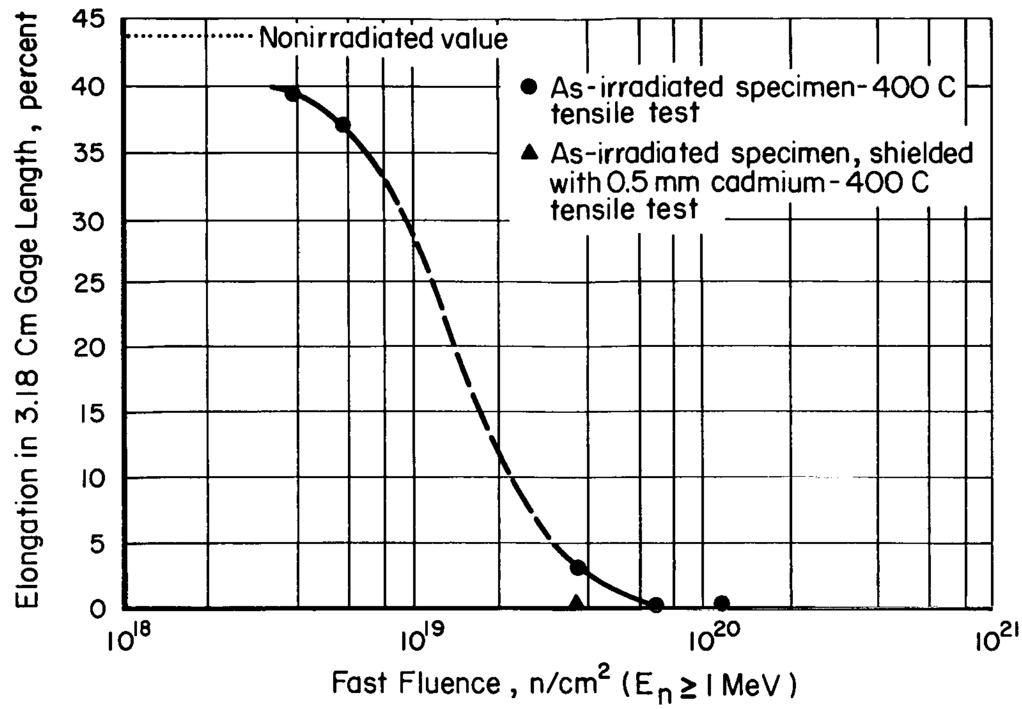


FIGURE 139. DUCTILITY VERSUS FAST NEUTRON FLUENCE FOR TUNGSTEN TENSILE SPECIMENS TESTED AT 400 C[212]

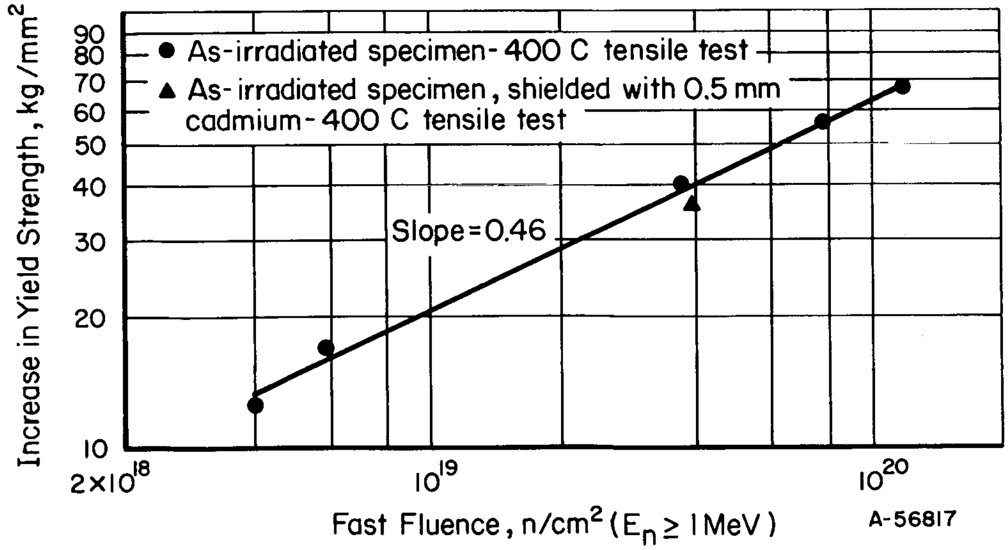


FIGURE 140. INCREASE IN 0.05 PERCENT YIELD STRENGTH VERSUS FAST NEUTRON FLUENCE FOR TUNGSTEN TENSILE SPECIMENS TESTED AT 400 C[212]

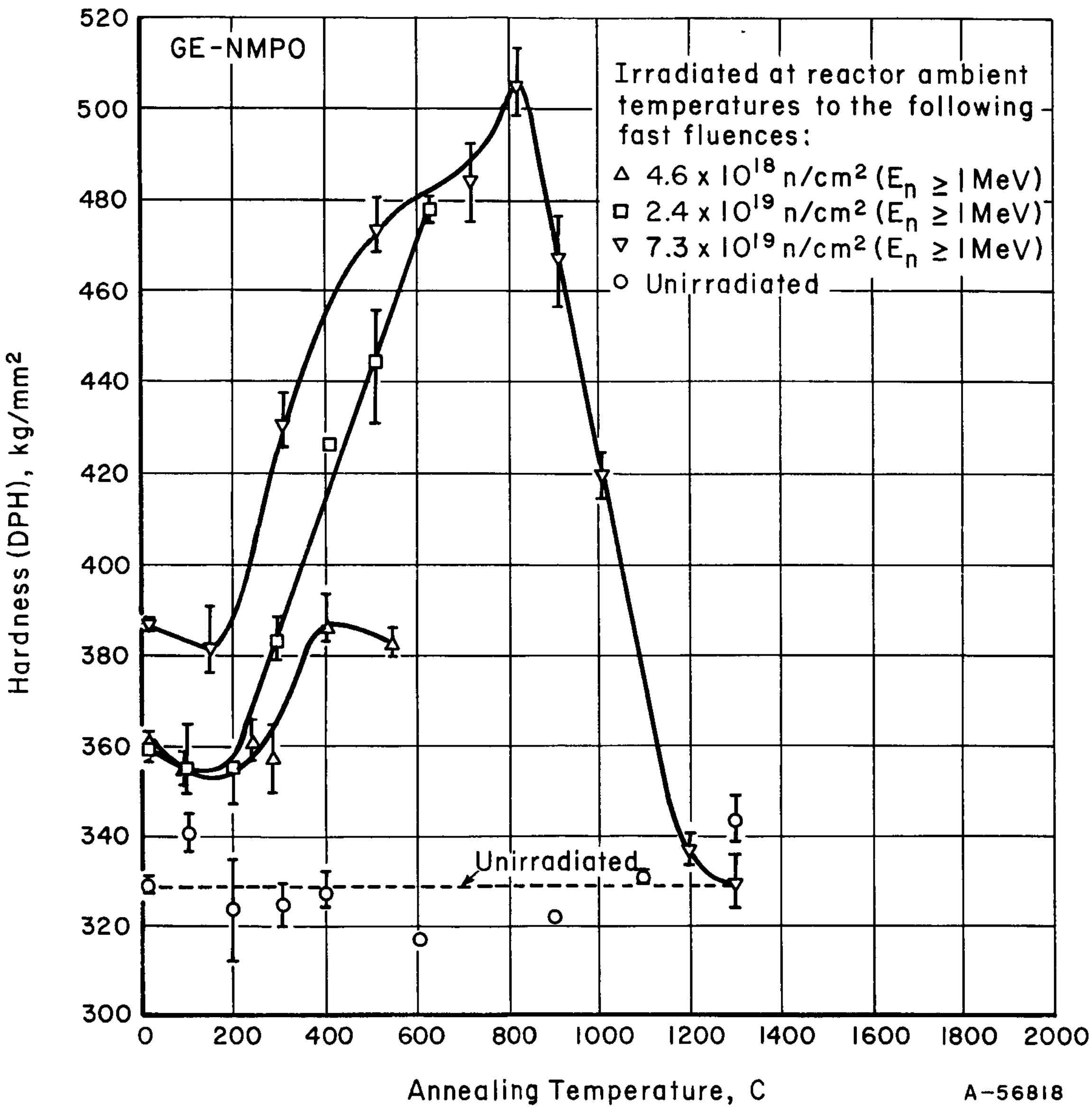


FIGURE 141. ROOM-TEMPERATURE HARDNESS OF IRRADIATED AND UNIRRADIATED SINGLE-CRYSTAL TUNGSTEN (411) AS A FUNCTION OF ANNEALING TEMPERATURE[130]

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

A considerable number of stress-rupture tests at elevated temperatures (900 to 1700 C) have been performed on irradiated tungsten and tungsten-25 weight percent rhenium. The rhenium content of these materials apparently varied from 25 to 26 weight percent for different batches, so that both of these values are used. The results of these stress-rupture tests on irradiated refractory metals and alloys are given in Table 80. Irradiation results in somewhat lower creep rates for tungsten and tungsten-25 weight percent rhenium at these temperatures. Since the elongation at rupture is not affected by irradiation at testing temperatures above 1100 C, the rupture life for the irradiated material is significantly longer. At 900 C, the elongation is somewhat reduced by irradiation but the creep rate is reduced to a larger degree and, therefore, the rupture life is still increased by irradiation.

The irradiation-induced increase in rupture life for tungsten-25 weight percent rhenium alloys has been shown to increase with increasing fast and thermal fluence. By shielding some specimens with cadmium during irradiation, thereby reducing thermal fluence, the rupture life was reduced. Therefore, it was concluded that the increase in rupture life is dependent on thermal fluence.[130] The thermal neutrons promote increased rupture life by transmuting some tungsten atoms to rhenium atoms and some of the rhenium atoms to osmium. These transmuted atoms cause increased strength, owing to solid-solution hardening, and improve ductility in tungsten.

Vanadium Alloys

Vanadium-titanium alloys have recently received consideration as fuel-element cladding materials in fast reactors. Tensile tests were performed on vanadium-titanium alloys containing chromium, niobium, or molybdenum which had received a fast fluence of 1.4×10^{21} n/cm^2 (>0.1 MeV).[221] The tests were performed at 20, 200, 400, 650, and 750 C. The results (Table 83) showed that irradiation drastically reduces elongation at room-temperature and significantly increases yield and ultimate strength. Increasing the testing temperature causes annealing out of the irradiation-induced damage, and the preirradiation mechanical properties are restored at 650 and 750 C. Increasing the titanium content in the alloys permitted more speedy recovery of preirradiation properties during annealing. Although the material contained significant quantities of boron, which was transmuted to helium by irradiation, no helium-bubble-induced intergranular fracture was observed.

An extensive vanadium-alloy development program is presently under way at Argonne National Laboratory. The alloys which are receiving most of the attention are vanadium-20 weight percent titanium and vanadium-15 weight percent titanium-7.5 weight percent chromium. The effect of elevated-temperature irradiation on the room-temperature tensile properties is illustrated in Table 84.[222] It can be seen that irradiation in the 500 to 660 C range to fast fluence levels of 1.6 to 4.6×10^{21} n/cm^2 does not significantly affect the tensile properties. The alloys which were irradiated at the lowest temperatures suffer some loss of ductility. The higher irradiation temperatures are apparently high enough to anneal out almost all of the displacement type of damage.

TABLE 83. 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)(221)

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 84. ROOM-TEMPERATURE TENSILE PROPERTIES OF VANADIUM-BASE ALLOYS IRRADIATED IN EBR-11[222]

Composition, weight percent	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	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

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*19	The Effect of Nuclear Radiation on Lubricants and Hydraulic Fluids (May 31, 1961), AD 261278 (Supersedes Report No. 4)
20	The Effect of Nuclear Radiation on Structural Metals (September 15, 1961), AD 265839 (Supersedes Report No. 5) AD 265839
*21	The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials (September 1, 1961), AD 267890 (Supersedes Reports Nos. 3, 9, 13, and 17 and Momos Nos. 1, 3, 8, 13, and 17)
*21 Addendum	The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials (August 31, 1964), AD 454056 (Supersedes Reports Nos. 3, 9, 13, and 17 and Momos Nos. 1, 3, 8, 13, and 17)
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13. ABSTRACT

Data available on the effects of radiation on tensile, creep, fatigue, impact and hardness properties of various structural materials are compiled. These properties are given as a function of test temperature, irradiation temperature, and radiation fluence. Specifically the following reactor materials are covered: (1) aluminum alloys, (2) magnesium alloys, (3) beryllium, (4) zirconium alloys, (5) mild steels, (6) stainless steels, (7) nickel alloys and (8) refractory metals. Data on the effects of radiation on the mechanical properties of selected materials at cryogenic temperatures also are included.

14. KEY WORDS	LINK A		LINK B		LINK C	
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Radiation effects						
Structural materials						
Metals						
Alloys						
Aluminum						
Magnesium						
Titanium						
Beryllium						
Zirconium						
Ferritic steels						
Martensitic steels						
Austenitic steels						
Nickel alloys						
Niobium						
Molybdenum						
Tantalum						
Tungsten						
Vanadium						
Refractory metals						
Cryogenic effects						