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EFFECT OF NUCLEAR RADIATION AT CRYOGENIC TEMPERATURES ON THE TENSILE PROPERTIES OF TITANIUM AND TITANIUM-BASE ALLOYS

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16. Abstract

An experimental investigation of the effects of reactor irradiation at 17 K on tensile properties of titanium and titanium-base alloys has been conducted in special test loops at the Plum Brook Reactor Facility. Commercially pure titanium, titanium - 5 aluminum - 2.5 tin, titanium - 6 aluminum - 4 vanadium, and titanium - 8 aluminum - 1 molybdenum - 1 vanadium alloys showed significant increases in tensile strength and probable decreases in ductility following reactor exposures up to 10×10^{17} fast neutrons per square centimeter. The influences of impurities, preirradiation heat treatment, and postirradiation heat treatment on radiation damage were also evaluated.

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SUMMARY

An experimental investigation of the effect of reactor irradiation at 17 K on the tensile properties of titanium and titanium-base alloys has been conducted at the Plum Brook Reactor Facility. The purpose of this investigation was to determine the fluence following which changes in tensile properties become significant.

Commercially pure titanium, titanium - 5 aluminum - 2.5 tin, titanium - 6 aluminum - 4 vanadium, and titanium - 8 aluminum - 1 molybdenum - 1 vanadium were exposed to irradiations up to 10×10^{17} neutrons per square centimeter (neutron energy greater than 0.5 MeV (80 femtojoule). Test specimens were exposed to reactor irradiation and then tensile tested in test loops which maintained specimen temperature at 17 K throughout irradiation exposure and tensile testing.

Following irradiation, all test materials show increases in yield and ultimate tensile strengths and slight decreases in ductility. The threshold for irradiation damage at 17 K occurs near 1×10^{17} neutrons per square centimeter and varies with the alloy. The thresholds for titanium - 6 aluminum - 4 vanadium and titanium - 8 aluminum - 1 molybdenum - 1 vanadium alloys are less than 1×10^{17} neutrons per square centimeter. Commercially pure titanium and titanium - 5 aluminum - 2.5 tin alloy exhibited the threshold after 1×10^{17} neutrons per square centimeter.

Results also indicate that the irradiation effect may be independent of impurity content and/or heat-treated condition. Impurity content does, however, appear to influence the materials' variability following irradiation. The higher the impurity content, the greater the variability following irradiation.

Thermal treatments following irradiation at 17 K of commercially pure titanium show that about 50 percent of the irradiation damage to the yield strength is recovered by an intermediate warming to 178 K. The ultimate strength shows 50-percent recovery at 78 K and indicates additional recovery in the 78 to 178 K range.

INTRODUCTION

The titanium-base alloys are finding wide acceptance for use at liquid hydrogen temperature (20 K). These alloys are also candidate materials for nuclear rocket applications where the nuclear environment is imposed on a structural material which is at the cryogenic temperature.

The tensile behavior of the various titanium-base alloys as a function of cryogenic temperature is now fairly well established (e.g., ref. 1). In general, most titanium-base alloys experience a marked decrease in elongation and reduction of area as the temperature is reduced to 20 K. At temperatures below about 78 K, only a few alloys experience plastic deformation prior to failure.

In 1956, Makin and Minter (ref. 2) published results showing that cryogenic temperatures accelerate the onset of brittle behavior for irradiated commercially pure titanium (CP Ti). These researchers exposed wire specimens of CP Ti to a fast neutron fluence of 5.1×10^{19} neutrons per square centimeter at 373 K and then performed tensile tests at various temperatures in the range from 78 to 473 K. For tests at 293 and 195 K, the elongation of unirradiated material was the same, 10.5 percent. Corresponding tests on irradiated material showed 8.3-percent elongation at 293 K and 4.8-percent elongation at 195 K. At 78 K, both unirradiated and irradiated specimens were brittle which precludes determination of the effect of irradiation on ductility.

To further investigate the effect of nuclear irradiation on embrittlement of titanium and titanium alloys at cryogenic temperature, an experimental program was undertaken at NASA's Plum Brook Reactor Facility. Test materials included in the program were selected to be representative of the various titanium-base alloys suitable for use at cryogenic temperatures. Tests were conducted using specially designed test equipment capable of maintaining the test specimen at 17 K throughout irradiation exposure and postirradiation tensile testing. The objectives of this test program were to determine the radiation damage threshold - that is, the fluence at which changes in tensile properties first become significant for each alloy - and to investigate the effects of impurities and heat treatment on this threshold.

TEST MATERIALS

Test materials were CP Ti in the annealed condition, titanium - 5 aluminum - 2.5 tin (Ti-5Al-2.5Sn) with normal impurity (NI) content in the annealed condition, Ti-5Al-2.5Sn with extra low impurity (ELI) content in the annealed condition, titanium - 6 aluminum - 4 vanadium (Ti-6Al-4V) in both annealed and age-hardened (aged) conditions, and titanium - 8 aluminum - 1 molybdenum - 1 vanadium (Ti-8Al-1Mo-1V) in the duplex annealed condition. Material compositions, actual analyses on as-received bars,

TABLE I. - CHEMICAL COMPOSITION AND FABRICATION HISTORY OF TEST MATERIALS

Test material	Temper	Heat treatment[a]	Chemical composition by weight percent[b]									
			Titanium	Aluminum	Tin	Vanadium	Molybdenum	Iron	Carbon	Nitrogen	Oxygen	Hydrogen
Commercially pure titanium	Annealed	A	Balance	----	----	----	----	0.190	0.032	0.023	0.218	0.006
Titanium - 5 aluminum - 2.5 tin, NI	Annealed	B		5.10	2.47	----	----	.110	.032	.019	.116	.012
Titanium - 5 aluminum - 2.5 tin, ELI	Annealed	B		5.36	2.35	----	----	.025	.022	.010	.059	.006
Titanium - 6 aluminum - 4 vanadium	Annealed	A		5.95	----	4.00	----	.170	.010	.022	.065	.006
Titanium - 6 aluminum - 4 vanadium	Aged	C		5.80	----	3.90	----	.150	.010	.035	.102	.010
Titanium - 8 aluminum - 1 molybdenum - 1 vanadium	Duplex annealed	D		8.02	----	1.14	1.00	.120	.030	.013	.091	.009

[a]Heat treatment:

A - Annealed at 978 K for 2 hr, air cooled.

B - Annealed at 1033 K for 2 hr, air cooled.

C - Solution treated at 1214 K for 0.5 hr, water quenched; aged at 811 K for 4 hr, air cooled.

D - Annealed at 1172 K for 1 hr, air cooled; annealed at 867 K for 8 hr, air cooled.

[b]Actual analyses on as-received bar.

are given in table I.

All test materials were obtained as 0.5-inch- (1.27-cm-) diameter round bar stock. Stock materials were prepared by consumable electrode vacuum arc melting, forging to 1.5-inch (3.81-cm) square bar at 1200 to 1422 K, rolling to 0.5-inch- (1.27-cm-) diameter round bar at 978 to 1256 K, and then heat treating. Heat treatments for each test material are given in table I.

TEST PROCEDURES

Tensile test data were obtained using miniature round tension test specimens and specially designed test equipment installed at the Plum Brook Reactor Facility. The program was conducted, as nearly as feasible, in accordance with the provisions of ASTM Standards E199 (ref. 3), E8 (ref. 4), and E184 (ref. 5).

Test Specimens

The tensile test specimen (fig. 1) was a geometrically similar miniaturization of the

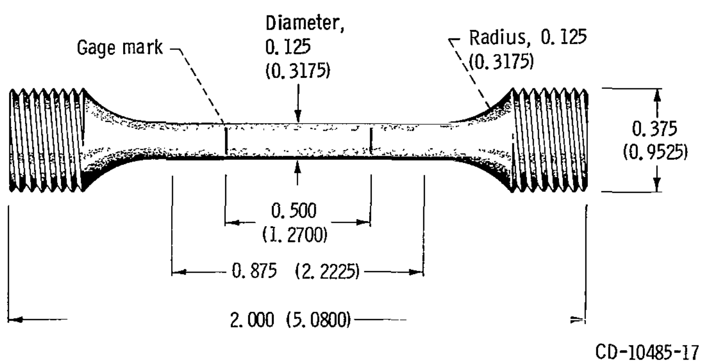

Figure 1. - Miniature round tensile test specimen. All dimensions are in inches (cm).

standard 0.5-inch (1.27-cm) round tension specimen of ASTM E8 (ref. 4). This specimen had a gage uniform section of 0.125-inch (0.3175-cm) diameter by 0.875-inch (2.2225-cm) length. Gage marks were inscribed by light sandblasting to delineate a 0.500-inch (1.27-cm) gage length. Ratios of significant dimensions were the same as for the standard ASTM specimen.

Test Equipment

Irradiations were conducted with the test specimen positioned adjacent to the beryllium reflector of the reactor core. Access to the irradiation zone was through a horizontal beam hole (HB-2) located approximately 20 feet (6.1 m) below the surface of a pool of demineralized water. Figure 2 shows the major components of the test equipment. The test machine was incorporated into a test loop having a 6-inch (15.24-cm) outside diameter and about a 9-foot (2.7-m) length. This loop also contained equipment necessary for precise temperature control and fracture of the specimen at temperature without removal from the irradiation field. The test loop was securely attached to a carriage which, along with tables capable of translation and rotation, provided the means for test

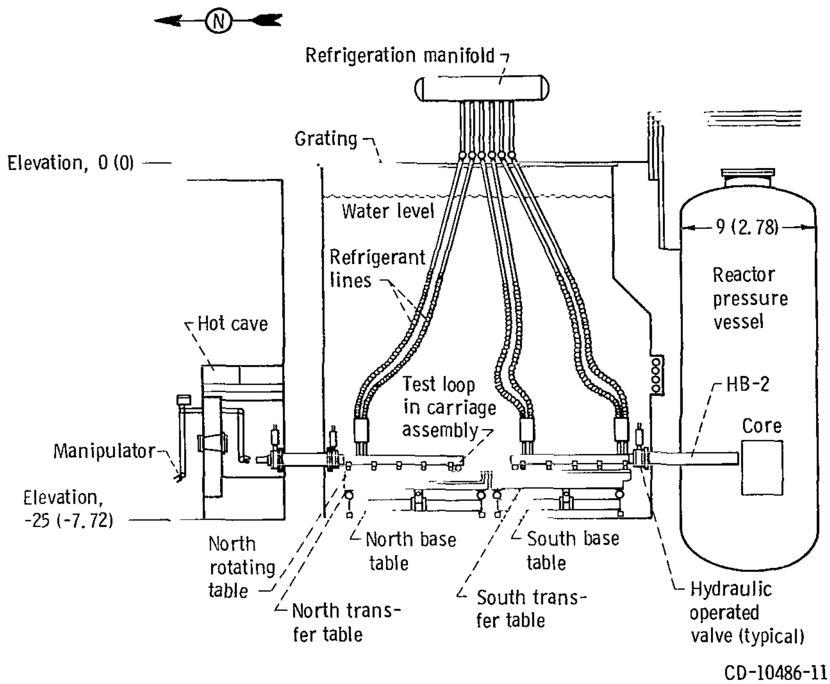

Figure 2. - Cryogenic irradiation test equipment. All dimensions are in feet (m).

loop insertion, retraction, translation, and rotation operations between a hot cave and the irradiation field. Two test loops and the ancillary equipment shown in figure 2 permitted continuous utilization of the neutron environment and did not require reactor shutdown for specimen replacement in the test loops.

The test loop body (fig. 3) contained a horizontally placed 5000-pound (22 241-N) capacity test machine together with the necessary load actuation components, stress-strain monitoring instrumentation, and vacuum insulated refrigerant transfer lines. The forward section, or head assembly, served both as a fixed crosshead of the test machine

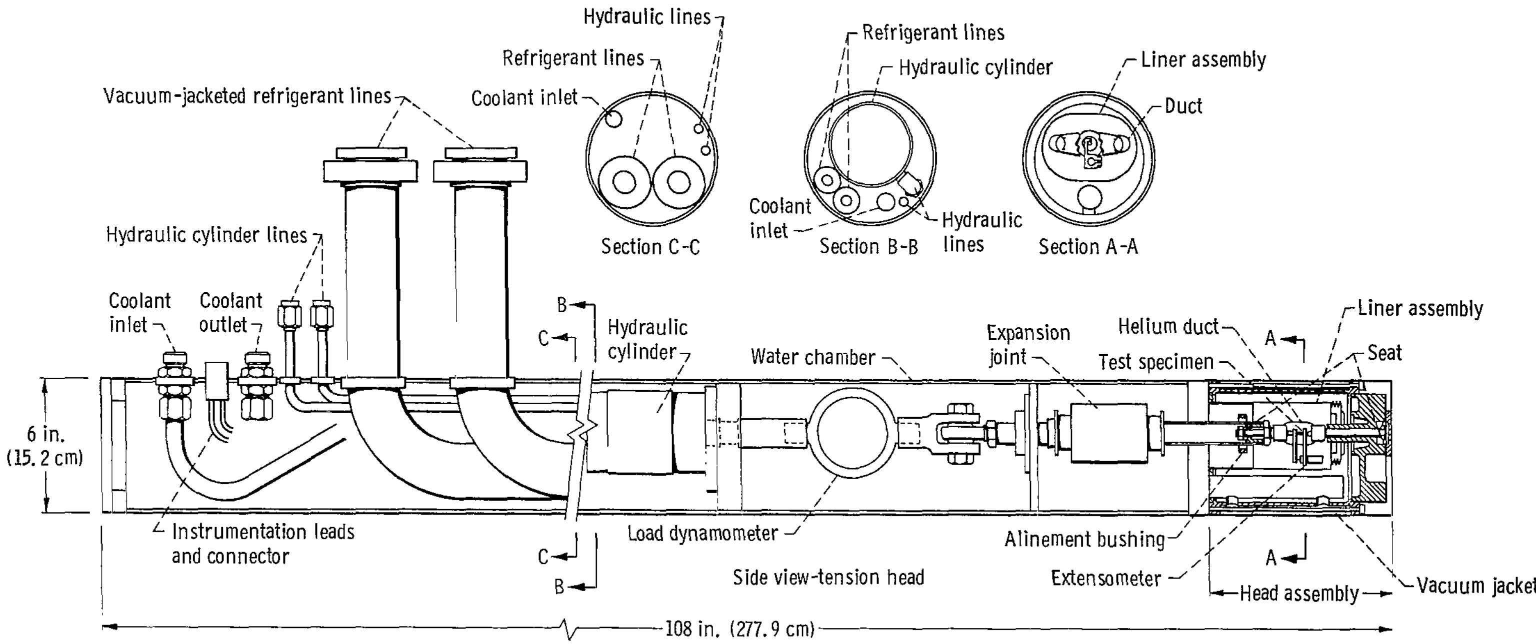

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Figure 3. - Cryogenic irradiation test loop.

and a cryostat for temperature control. The head assembly was removable by remote handling methods to allow specimen change. Alinement studies showed no significant bending moments due to the horizontal attitude of the test machine. A detailed discussion of the design features of the test loop has been reported elsewhere (refs. 6 and 7).

The load applied to the test specimen was monitored by a proving ring type dynamometer using a linear variable differential transformer (LVDT) to measure the ring deflection resulting from loading. Dynamometers in each test loop were calibrated to within 2 percent of a National Bureau of Standards certified reed type proving ring.

Strain was measured using an extensometer which measured the increase of the separation between two knife edges initially 0.5-inch (1.27-cm) apart. The measurement was accomplished through the use of a LVDT specially constructed to be resistant to radiation effects. This extensometer had a range of reliable accuracy of approximately 0.010 inch (0.025 cm), which was sufficient to record strains to well beyond the yield strength (0.2 percent offset method). The extensometer was verified in accordance with ASTM Specification E83 (ref. 8) and the error in indicated strain was less than 0.0001. Installation of the extensometer by remote means, however, introduces the possibility for increased error in the indicated strain.

An X-Y recorder was employed to automatically plot the load-strain curve to approximately 0.02 total strain. The X-axis recorder was then switched to time travel and a load-time curve through fracture was obtained.

Test Specimen Temperature Control

Temperature control of the test specimen was achieved with a closed cycle refrigeration system in which helium gas was continuously circulated. This system had a maximum refrigeration capacity of 1100 watts when operating for 17 K refrigeration. The design features of this refrigerator have been reported elsewhere (ref. 9) and are not repeated here, except to note a few important features. Figure 4 is a schematic diagram of the system. Refrigeration was accomplished by expansion engines from which the cold helium gas was discharged into a refrigeration manifold for use in test loop temperature control. From the manifold, the cold helium gas was routed through the test loops at a rate controlled by the readings of platinum resistance thermometers located in the inlet and return legs of the manifold transfer line connections.

Direct measurement of the specimen temperature was not feasible so the platinum resistance thermometers were calibrated to provide test specimen temperature control. This was accomplished by using a typical specimen of titanium with thermocouples attached to the midpoint and each end of the gage length. The instrumented specimen was calibrated to serve as a working standard using a platinum resistance bulb calibrated by the National Bureau of Standards. After calibration, this specimen was installed in the

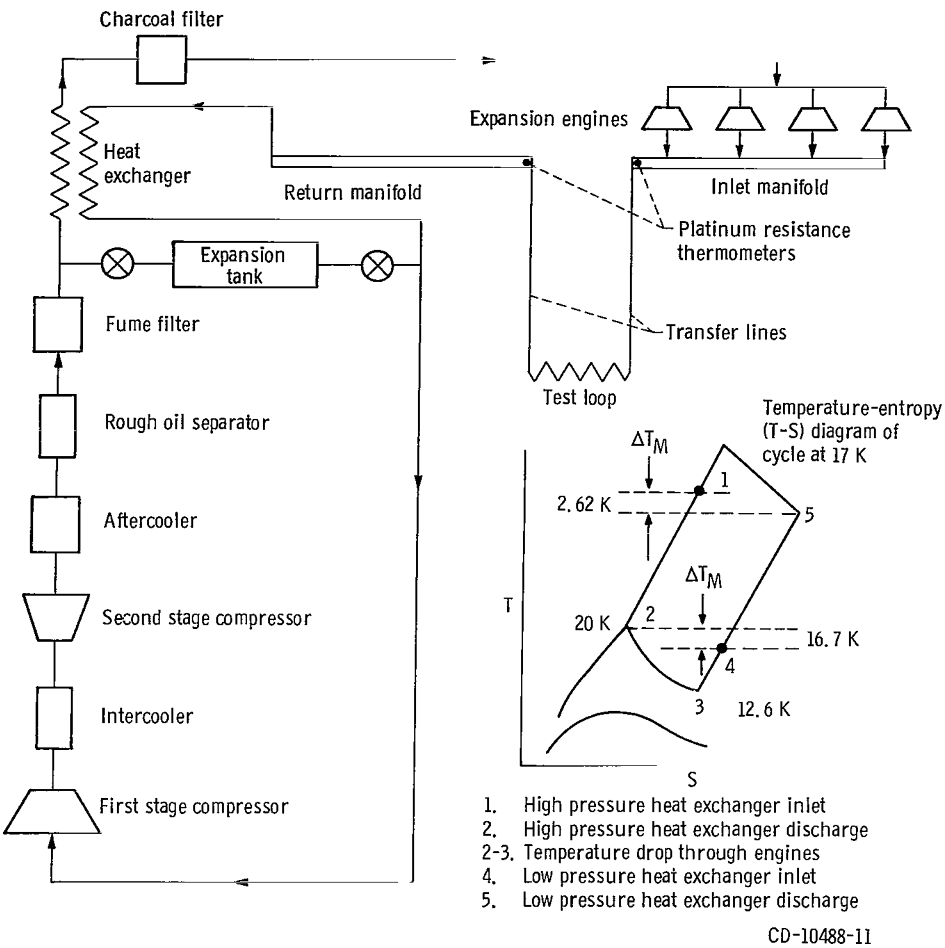


Figure 4. - Schematic of refrigeration system.

normal test position in a test loop. Readings of the inlet and outlet platinum resistance thermometers were taken which corresponded to specimen temperature under both out-of-pile and in-pile steady-state conditions as measured by the instrumented specimen. The correction constants determined for the platinum resistance thermometers were used during test program performance to maintain the specimen temperature within ±0.5 K.

Measurement of Neutron Environment

The fast neutron spectrum in the test location in HB-2 was determined using foil measurement techniques. Sets of foils, as shown in table II, were placed in an aluminum foil holder which was then installed in a test loop at the test specimen location. The loop was inserted into HB-2 and irradiated for 1800 seconds during full power reactor operation. Following irradiation, the foils were counted and evaluated by standard techniques.

TABLE II. - FOILS USED FOR SPECTRAL MEASUREMENTS OF FAST NEUTRON FLUENCE IN HB-2

Type of foil	Nuclear reaction	Threshold energy		Cross section, cm^2
		MeV	fJ	
Indium	In^{115} (neutron, neutron) In^{115m}	0.45	72	0.20×10^{-24}
Neptunium	Np^{237} (neutron, fission) Ba^{140}	.75	120	1.52
Uranium	U^{238} (neutron, fission) Ba^{140}	1.45	232	.54
Thorium	Th^{232} (neutron, fission) Ba^{140}	1.75	280	[a].10
Sulfur	S^{32} (neutron, proton) P^{32}	2.9	464	.284
Nickel	Ni^{58} (neutron, proton) Co^{58}	5.0	800	1.67
Magnesium	Mg^{24} (neutron, proton) Na^{24}	6.3	1008	.0715
Aluminum[b]	Al^{27} (neutron, alpha) Na^{24}	8.1	1296	.110
Aluminum	Al^{27} (neutron, alpha) Na^{24}	8.6	1376	.23

[a]Cross section for thorium is not considered reliable.
[b]Al^{27} (neutron, proton) Mg^{27} reaction with a threshold energy of 5.3 MeV (848 fJ) is not included because of short (9.8 min) half life of product.

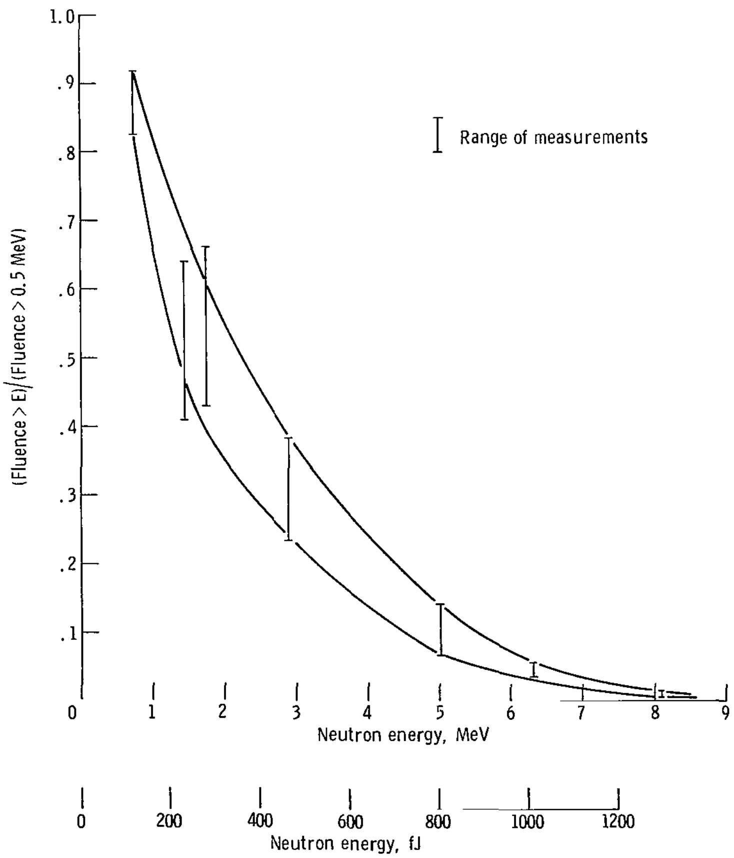


Figure 5. - Ratio of fluence greater than E to fluence greater than 0.5 MeV (80 fJ).

From these data and the reactor operating conditions at the time of foil exposure, the fast neutron fluence, with energy greater than 0.5 MeV, ($E > 0.5$ MeV (80 fJ)) was determined.

The preceeding procedure was repeated periodically throughout the life of a reactor power cycle in order to determine the variation in neutron flux with control rod position. During test program performance the exposure rate for test specimens varied from about 2.0×10^{12} to 2.5×10^{12} neutrons per square centimeter per second, $E > 0.5$ MeV (80 fJ). The ratio of the fluence with neutron energies greater than E to the fluences with energies greater than 0.5 MeV (80 fJ) is given in figure 5. This figure is based on 27 sets of foils irradiated during the test program.

Data Acquisition

Performance of the data acquisition phase of the test program can readily be followed by reference to figure 2. In a typical testing sequence, a test loop was inserted into the hot cave for specimen installation. After specimen installation, the loop was withdrawn from the hot cave to the north table in quandrant "D". Refrigerant flow was started and the table holding the loop was rotated 180°. The loop was then transferred to the south table and positioned in-line with HB-2 and, after stabilization of specimen temperature at 17 K, inserted into HB-2. The loop was held at this position with the specimen maintained at 17 K until the accumulated fast neutron fluence was attained. The time for this operation varied from 10 to 140 hours, depending on the desired fluence. When the specimen had accumulated the required fast neutron fluence, the test loop was retracted approximately 4 feet (1.2 m) and then an axial tensile load was applied to the specimen. During elastic behavior, the tensile load was applied at a strain rate which did not exceed 2.5×10^{-5} per second. After specimen failure, the loop was returned to the hot cave for specimen replacement.

For some tests, the specimens were thermally treated following accumulation of the fast neutron fluence and prior to tensile load application. The thermal treatments consisted of either a warming operation or an annealing operation. For the warming operation, the test specimen temperature was raised from 17 K to either 78 or 178 K, held at temperature for 1 hour, and then fractured. The annealing operation followed essentially the same procedure with the exception that following 1 hour at the annealing temperature, the specimen was recooled to and held for 1 hour at 17 K prior to application of the tensile load.

After removal of the specimen from the test loop, the fractured gage length and minimum diameter were measured. The broken halves of the test specimens were fit together and measurements were obtained by means of a micrometer stage and hairline apparatus accurate to ±0.0001 inch (0.00025 cm).

Unirradiated control tests were conducted in a test loop under the same conditions as their irradiated counterparts.

For each of the test materials included in the program, at least three specimens were tested for each exposure condition.

Data Reduction and Analysis

The load-strain/load-time curve developed by the X-Y recorder during testing and the initial specimen dimensions provided data for the determination of the ultimate tensile strength and the tensile yield strength (0.2-percent offset method). The modulus of elasticity was also approximated from these curves. Elongation and reduction of area values were calculated from the original specimen dimension and the dimensions following fracture.

The test data were compiled and subjected to statistical analysis in accordance with methods 1-6, 2-1.4.1, 3-3.1.2, and 4-2.1 given by Natrella (ref. 10). These analyses included determination of average values and estimated standard deviations for each material and test condition; determinion of the differences between values for irradiated and unirradiated test conditions, the 95-percent confidence interval associated with these differences; and estimation of the difference in variability following irradiation. Confidence intervals were not calculated for the differences in variability since the number of test specimens used per test condition was small.

DISCUSSION OF TEST RESULTS

Tensile test data obtained during performance of the program are compiled in tables VI to XIII of the appendix. These tables contain individual test specimen values, average values and estimated standard deviation for each test condition, and the statistical analyses of differences in the average values and the variability. The 95-percent confidence interval for each difference is also included in the tables. The test results obtained are now discussed according to individual alloy.

Commercially Pure Titanium

Some CP Ti was employed at test material for investigating the synergistic effect of reactor irradiation and cryogenic temperature on the tensile properties. The objective of this investigation was to determine the radiation damage threshold - that is, the fluence

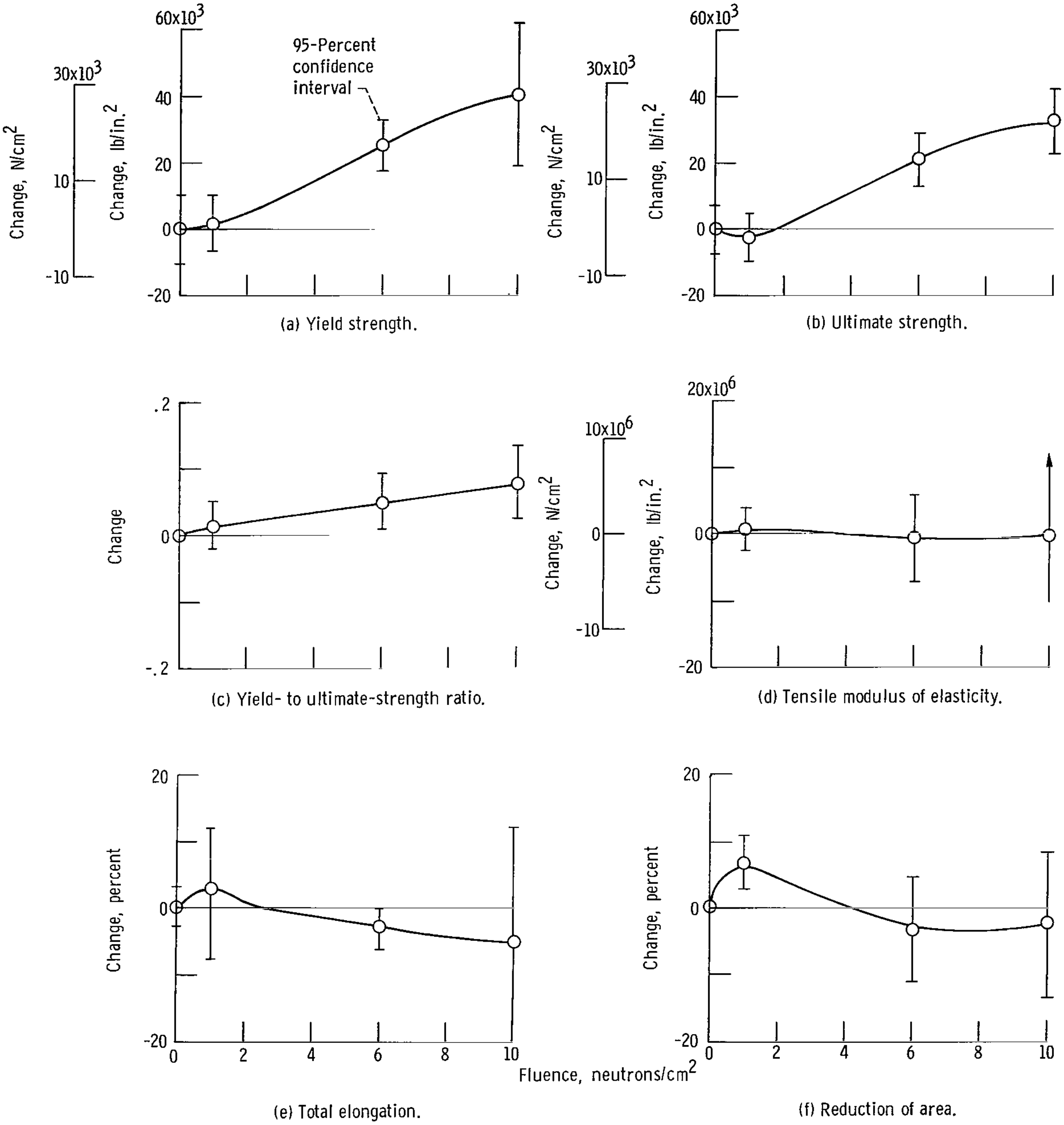


Figure 6. - Effect of reactor irradiation at 17 K on 17 K tensile properties of commercially pure titanium. Neutron energy > 0.5 MeV (80 fJ).

level at which significant changes in tensile properties first occurred - and to study the reduction of this damage during postirradiation annealing.

Radiation damage threshold. - The radiation damage threshold was determined from tests conducted at 17 K following reactor irradiation at 17 K to fluence levels of 1×10^{17}, 6×10^{17}, and 10×10^{17} neutrons per square centimeter, E > 0.5 MeV (80 fJ). Tests conducted at 17 K on unirradiated material were used as base-line control data. Results of the data analyses are shown in figures 6 and 7.

As may be seen from figure 6, irradiation to 10×10^{17} neutrons per square centimeter at 17 K increased strength parameters, decreased ductility parameters, but did not alter the tensile modulus of elasticity. The increases in yield and ultimate strengths are approximately linear with increasing fluence with the yield strength increasing more rapidly than the ultimate strength, as may be observed from the plot of yield- to ultimate-strength ratio. The changes in total elongation and reduction of area are small and in most cases are not statistically significant at the 0.05 level of significance. (In order for the difference to be statistically significant at the 0.05 level of significance, the 95-percent confidence interval must not include zero.) Only the increase in the reduction of area following 1×10^{17} neutrons per square centimeter exposure and the decrease in total elongation following 6×10^{17} neutrons per square centimeter exposure are statistically significant.

The general data trend for irradiations to 10×10^{17} neutrons per square centimeter at 17 K indicates that the radiation damage threshold occurs in the region between 1×10^{17} and 6×10^{17} neutrons per square centimeter. This threshold is defined by a significant in-

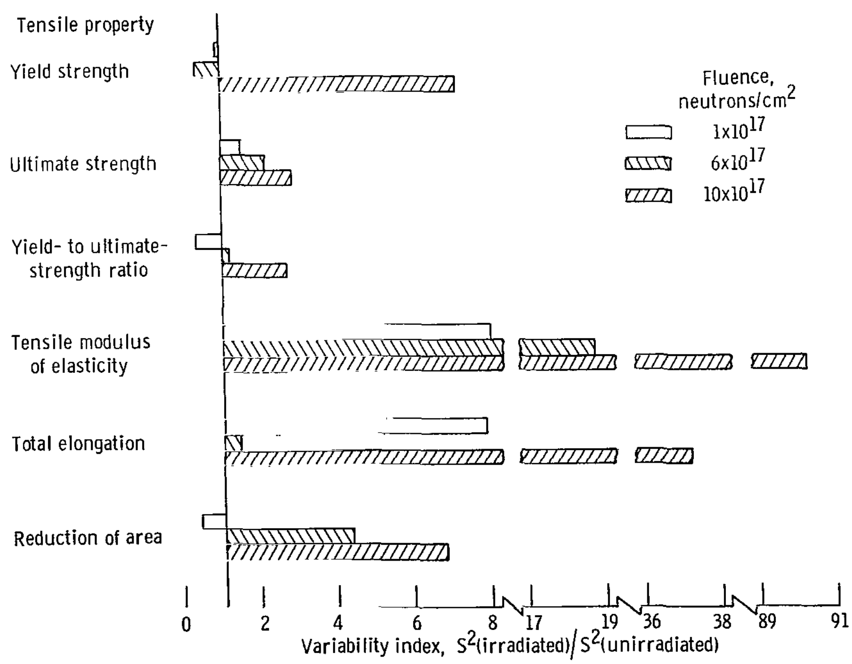


Figure 7. - Effect of reactor irradiation at 17 K on variability of 17 K tensile properties of commercially pure titanium. Neutron energy > 0.5 MeV (80 fJ).

crease in both yield strength and ultimate strength and possibly a small decrease in ductility. The increase in reduction of area following irradiation to 1×10^{17} neutrons per square centimeter appears to be anomolous.

Figure 7 is an estimate, defined as the variability index, of the effect of irradiation at 17 K on variability of 17 K tensile properties. The variability index is defined as the ratio of the square of the estimated standard deviation for the irradiated condition, s^2 (irradiated), to the square of the estimated standard deviation for the corresponding unirradiated condition, s^2 (unirradiated). The "no change" condition is indicated by 1.0. For indices greater than 1.0, the variability of the property increases, and conversely, for indices less than 1.0, the variability of the property decreases following irradiation.

As may be seen from figure 7, the variability of the 17 K tensile properties of CP Ti apparently increases with increasing fluence at 17 K. This indicated trend is interesting even though confidence cannot be established because of the small number of specimens employed per test condition. Since test procedures were essentially identical except for irradiation exposure, the increase in variability with increasing fluence is probably real, however, it should be noted that exceptions to this generalization do occur. For example, the variability index of the yield strength, yield- to ultimate-strength ratio, the reduction of area following 1×10^{17} neutrons per square centimeter, and the yield strength following 6×10^{17} neutrons per square centimeter indicate a decrease in variability due to irradiation. Another exception which should be noted is the difference in relative magnitude of the increase in variability indices of total elongation following 1×10^{17} and 6×10^{17} neutrons per square centimeter exposures. The lower fluence exposure shows a large increase in variability whereas the higher fluence exposure shows only a slight increase in variability.

Reduction of radiation damage. - The reduction of radiation damage was investigated using heat treatments following irradiation at 17 K to 6×10^{17} neutrons per square centimeter. Unirradiated specimens were subjected to the same heat treatment conditions except for the time of initial 17 K exposure. The unirradiated specimens were held for 1 hour at 17 K prior to heat treatment, whereas the irradiated specimens were held approximately 40 hours (irradiation time) at 17 K prior to heat treatment.

Two cases are considered to be pertinent to the reduction of radiation damage. The first case consists of simply warming the specimen from 17 K to a higher temperature and then testing at this higher temperature. This provides information relative to the reduction of the irradiation induced defects influence on the metal lattice, but does not separate the annihilation of defects from the reduction in lattice friction due to higher thermal oscillation of lattice ions. The second case, defined as annealing, duplicated the previous warming condition, but specimen temperatures were subsequently reduced to and held for 1 hour at 17 K prior to testing at 17 K. The combination of these two cases permits a separation of defect annihilation from defect resistance. Figures 8 to 10 illustrate the results.

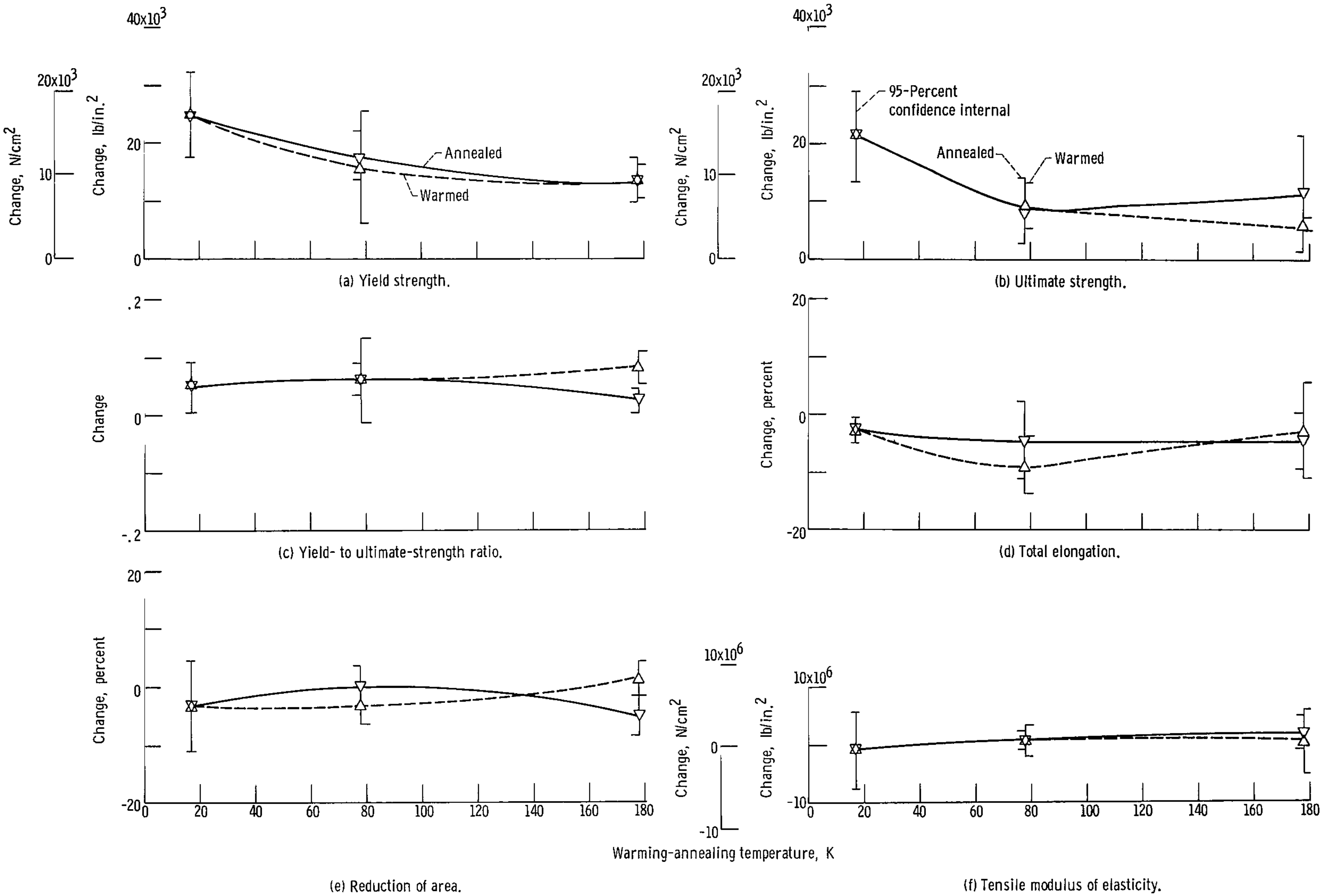


Figure 8. - Effect of postirradiation warming and annealing on tensile properties of commercially pure titanium irradiated at 17 K to 6×10^{17} neutrons per square centimeter. Energy > 0.5 MeV (80 fJ).

As can be seen from the dashed curves of figure 8, warming the specimens from 17 to 178 K resulted in a recovery of approximately one-half of the radiation damage for the yield strength and three-fourths for the ultimate strength. The greater portion of this recovery occurred between 17 and 78 K. The reduction of area and tensile modulus of elasticity are not significantly different at 178 K than at 17 K although the trend for the reduction of area does indicate some damage at 17 K which has fully recovered at 178 K. The total elongation indicates further damage at 78 K; however, the 95-percent confidence intervals for this property overlap at all temperatures, suggesting an equal probability that the indicated trend is within material variability limits.

Annealing the test material resulted in essentially the same recovery as observed for the warming case, as may be noted from the solid curves of figure 8. The ultimate strength following 178 K annealing appears somewhat higher than the corresponding 178 K warming case; however, the 95-percent confidence interval is also greater and includes the confidence interval for the warming case. Such a condition suggests that there is no difference between the two thermal treatments. This conclusion is somewhat shadowed, however, by the distinct separation of the confidence intervals for the yield- to ultimate-strength ratio following thermal treatment at 178 K. The results for total elongation following the 78 K treatment and the reduction of area following the 78 and 178 K treatments show some deviations; however, for each case there is an overlap of the 95-percent confidence intervals which indicates no significant differences.

The data compared in figure 8 lead to the conclusion that thermally cycling CP Ti to 178 K following irradiation at 17 K to 6×10^{17} neutrons per square centimeter reduces the irradiation damage by annihilation of irradiation-induced defects. As a result of this annihilation, the irradiation damage to the yield and ultimate strengths is reduced by about 50 percent. Furthermore, for the ultimate strength, there is an indication that additional recovery, attributable to reduced lattice resistance, occurs between 78 and 178 K.

Inspection of the variability analysis for each thermal cycle condition (figs. 9 and 10) shows that recovery of the irradiation effect is probably occurring here also. The general trend with increasing warming and/or annealing temperature for each tensile property's variability index is an approach to 1.0, the no change condition. In most cases, exposure to 178 K decreases the variability index below 1.0.

Comparison of test results from various sources. - The results obtained from irradiation of CP Ti during the program reported herein and by other investigations reported in the literature (refs. 2 and 11 to 16) are compiled in table III. These data represent the differences between average tensile property values for irradiated and unirradiated material and in many cases an average value represents only one specimen. Fluence values are for neutrons with energies greater than 1 MeV (160 fJ). In a number of instances, noted on table III, fluence values for neutrons with energies greater than 1 MeV (160 fJ) were calculated and/or estimated from data reported by the investigator.

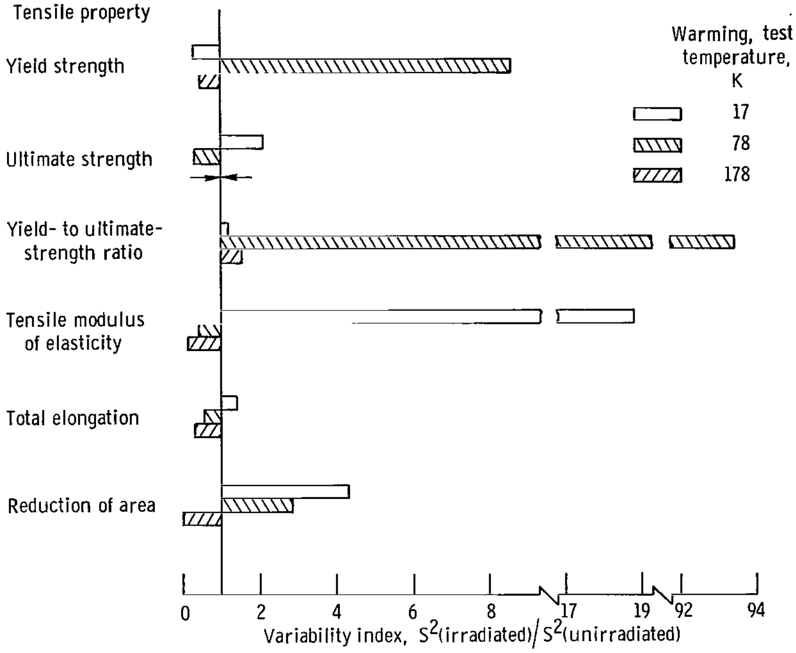


Figure 9. - Effect of postirradiation warming on variability of tensile properties of commercially pure titanium irradiated at 17 K to 6×10^{17} neutrons per square centimeter. Energy > 0.5 MeV (80 fJ).

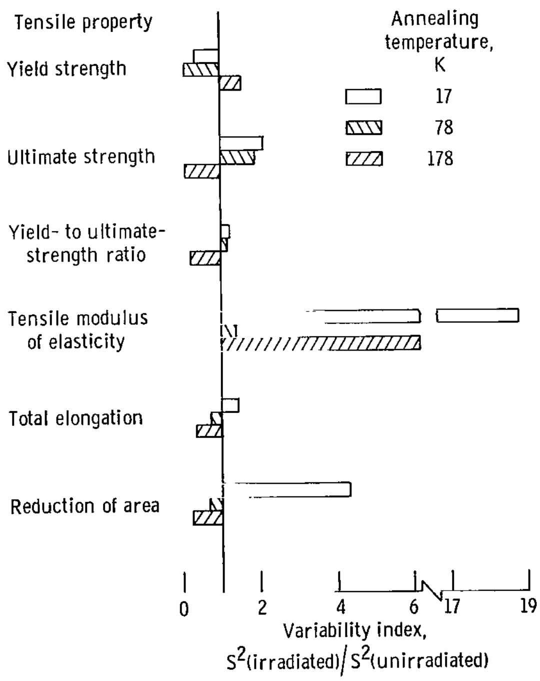


Figure 10. - Effect of postirradiation annealing on variability of tensile properties of commercially pure titanium irradiated at 17 K to 6×10^{17} neutrons per square centimeter. Energy > 0.5 MeV (80 fJ).

TABLE III. - EFFECT OF REACTOR IRRADIATION ON TENSILE PROPERTIES OF COMMERCIALLY PURE TITANIUM AS REPORTED BY VARIOUS INVESTIGATORS

Entry	Code	Test material		Exposure			Post-exposure temperature, K		Change in 0.2-percent offset yield strength		Change in ultimate strength		Change in total elongation, percent	Change in reduction of area, percent	Remarks	Reference
		Form	Condition	Medium	Temperature, K	Fast fluence, neutrons/cm^2 (a)	Pretest	Test	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2				
1	A	Bar	Annealed	Helium	17	7.6×10^{16}	17	17	1.4×10^3	1.0×10^3	-2.3×10^3	-1.6×10^3	3.0	6.7	(b)	Present investigation
2						4.6×10^{17}	17		25.0	17.2	21.3	14.7	-2.7	-3.3		
3						7.6	17		40.0	27.6	32.6	22.5	-5.0	-2.6		
4						4.6	78		17.8	12.3	8.4	5.8	-4.3	0		
5							178		13.5	9.3	11.3	7.8	-4.3	-5.0		
6							78	78	15.7	10.8	9.1	6.3	-9.0	-3.0		
7							178	178	13.4	9.2	5.9	4.1	-2.3	1.4		
8	B	Bar	Annealed	------	338	1.0×10^{18}	300	300	11.8×10^3	8.1×10^3	6.8×10^3	3.9×10^3	-1.1	-0.9	(c)	11, 12
9				------	373	1.0			12.1	8.3	7.7	4.5	-1.1	-1.3	(c)	11, 12
10				------	573	1.0			9.2	7.1	9.0	5.2	-.8	-1.3	(c)	11, 12
11				Water	353	1.0×10^{17}			11.0	7.6	6.0	4.1	-4.5	-3.0		12, 13
12						9.0			29.0	20.0	18.0	12.4	-13.0	-6.5		
13						4.5×10^{18}			43.5	30.0	28.5	19.6	-15.0	-9.0		
14						9.0			42.5	29.3	22.5	15.5	-14.0	-6.5		
15						1.2×10^{20}			63.0	43.4	42.0	28.9	-18.5	-20.5		
16					363	2.7×10^{19}			42.2	29.1	20.9	14.4	-12.9	-6.3	(d)	11, 12
17					363	5.3			43.4	29.9	24.3	16.7	-15.1	-8.1	(d)	11, 12
18					363	1.2×10^{20}			41.3	28.5	24.3	16.7	-14.4	-6.4	(d)	11, 12
19	C	Bar	Annealed	D_2O	523	1.2×10^{17}	300	300	1.6×10^3	1.1×10^3	3.5×10^3	2.4×10^3	[e]-5.0	3.8		14
20				UO_2SO_2		2.6×10^{18}	300	300	5.9	4.1	9.5	6.5	[f]-11.9	1.0		14
21				UO_2SO_2		5.0	300	300	[g]5.1	[g]3.5	5.9	4.1	-----	-4.4		15
22				UO_2SO_2		5.0	573	573	[g]9.4	[g]6.5	9.7	6.7	-----	-12.6		15
23	D	Wire	Annealed	Argon	373	5.1×10^{19}	78	78	$\geq14.6\times10^3$	$\geq10.1\times10^3$	$\geq14.6\times10^3$	$\geq10.1\times10^3$	-----	-----	(h)	2
24							195	195	8.0	5.5	5.1	3.5	-5.8	-----		
25							293	293	8.9	6.1	8.8	6.1	-2.1	-----		
26							473	473	[i]9.0	6.2	2.1	1.4	-1.0	-----		

[a]Fast fluence is for $E > 1$ MeV (160 fJ).

[b]Fast fluence > 1 MeV (160 fJ) $\approx$ 0.76 times fast fluence > 0.5 MeV (80 fJ).

[c]Investigators (ref. 11) reported fluence as 8.0 to 9.4×10^{18} thermal neutrons/cm^2 obtained in Brookhaven National Laboratory reactor. Fluence converted to 1 MeV (160 fJ) using data reported in ref. 16 for hole E-45 in Brookhaven National Laboratory reactor.

[d]Investigators (ref. 13) reported fluence as 2.2 to 6.6×10^{20} thermal neutrons/cm^2 obtained in Materials Testing Reactor along with a thermal neutron flux profile against specimen irradiation position. Fleunce converted to 1 MeV (160 fJ) using corresponding data for AISI 347 stainless steel and high purity iron as reported in survey article by Porter (ref. 16) and flux profile against specimen irradiation position reported by investigators (ref. 13).

[e]Investigators (ref. 14) also report 0.2-percent decrease in uniform elongation and 2.8-percent decrease in necking elongation.

[f]Investigators (ref. 14) also report 0.3-percent increase in uniform elongation and 12.7-percent decrease in necking elongation.

[g]Change in proportional limit rather than 0.2-percent offset yield strength.

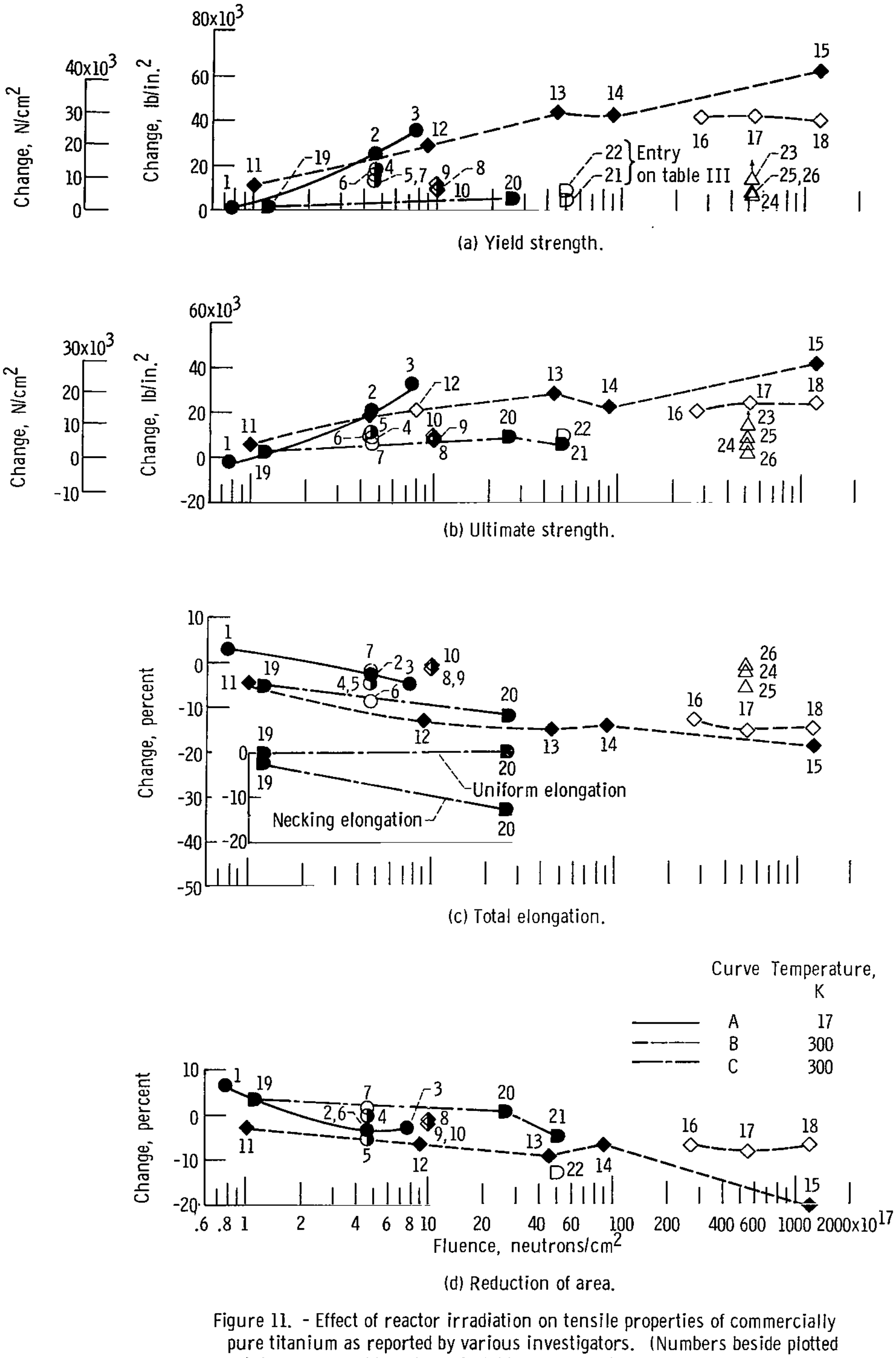
[h]Investigators (ref. 2) report fluence as slow neutrons/cm^2 with statement that fast fluence is approximately equal to slow fluence. Fast fluence is assumed to be for $E > 1$ MeV (160 fJ).

[i]Investigators (ref. 2) report a drop-in-load yield point for irradiated material.

The various test results compiled in table III are plotted for comparison purposes in figure 11. On this figure, the numbers beside the plotted points correspond to the entry on table III. Curves A, B, and C characterize the results for three different test materials for which data were available for two or more irradiation exposures. The comparisons show several interesting characteristics even though valid conclusions cannot be drawn because of the considerable variation between test materials and experimental techniques.

First, from the comparison of yield and ultimate strengths, it appears that the irradiation effect on both of these properties depends on irradiation and postirradiation temperatures. The effect of room temperature conditions - that is, irradiation near room temperature followed by testing at room temperature - is illustrated by curve B. Increasing the irradiation temperature to about 570 K decreases the irradiation effect on the strength properties for both room and elevated temperature test conditions, as indicated by curve C and point 22. Decreasing the irradiation temperature to 17 K increases the irradiation effect when tests are conducted at the irradiation temperature as indicated by curve A; however, this is not necessarily the case when the material is warmed from 17 K to a higher temperature for testing. As a matter of fact, raising the test temperature from 17 to 178 K reduces the irradiation effect to levels comparable with room temperature test conditions. This may be seen by constructing a line between 0 change at 1×10^{17} and point 7 for each tensile property. It can then be seen that for both yield and ultimate strengths the constructed line and curve B are parallel over the fluence region from about 7.2×10^{16} to 4.3×10^{17} neutrons per square centimeter. The irradiation effect on the ultimate strength for 523 K irradiation (curve C) is not only approximately parallel, but is virtually identical with the line constructed through point 7 over this same fluence range. Correspondingly, the irradiation effect on the yield strength for 523 K irradiation (curve C) is lower than the other two conditions and does not increase as rapidly with increasing fluence.

A second feature of the yield and ultimate strength plots is the irradiation effect for postirradiation test temperatures of 78, 178, and 195 K. It appears that recovery of the cryogenic temperature irradiation effect of test material A (points 6 and 7) following exposure to 4.3×10^{17} neutrons per square centimeter at 17 K is essentially complete for the ultimate tensile strength but not for the yield strength. This is based on comparison with the data for test material D (points 23 and 24) following exposure to 5.1×10^{19} neutrons per square centimeter. For the 78 K test temperature, the irradiation effect on both materials is essentially the same order of magnitude for the yield strength (point 6 against point 23). For the ultimate strength, point 23 is slightly greater than point 6. For the 178 and 195 K cases, the yield strength difference for point 7 is greater than point 24 which is essentially identical with the room temperature condition (point 25). This indicates further recovery may occur for material A in the temperature range from 178 K to



(a) Yield strength.

(b) Ultimate strength.

(c) Total elongation.

(d) Reduction of area.

Figure 11. - Effect of reactor irradiation on tensile properties of commercially pure titanium as reported by various investigators. (Numbers beside plotted points correspond to entrees in table III.) Neutron energy > 1 MeV (160 fJ).

room temperature. The difference in ultimate strengths (point 7 against points 24 and 25) indicates fair agreement, or possibly an increase of the irradiation damage. A combination of annealing data (point 5) and results from test material D when applied to curve A at 7.2×10^{17} neutrons per square centimeter shows that recovery should reduce the yield and ultimate strength differences to a level comparable with room temperature test conditions - that is, curve B.

The third feature of the strength comparisons shown in figure 11 is an apparent saturation effect in the interval between about 2.5×10^{18} and 9×10^{18} neutrons per square centimeter. While this may very well be due to material variability, it is interesting to note that the characteristic is present for materials of two different impurity levels, materials B and C.

As a final point of interest in the yield strength comparisons, it can be shown from curve B that on either side of the apparent saturation region the difference against the log of the fluence may be characterized by straight lines of the same slope. A similar relation exists for the difference comparisons of the ultimate strengths. Applying this to curve C at point 21 results in projected lines which pass through point 25, which is the room temperature test condition for material D at 5.1×10^{19} neutrons per square centimeter.

The comparison of ductility differences shown in figure 11 indicates a trend which is less complex than the comparison of strength differences. In almost all cases, the decreases in ductility with increasing fluence fall on essentially straight lines which are parallel. These data might very well be approximated by a single straight line for each of the ductility parameters.

One significant point with regard to the effect of irradiation on the change in total elongation is shown by the insert in figure 11. Measurements separating the total elongation into its constituents (the uniform elongation and the necking elongation) show that the uniform elongation is unchanged by irradiation, whereas the necking elongation is reduced and accounts for the decrease in total elongation. Applying this phenomenon to the data for test material D at 5.1×10^{19} neutrons per square centimeter could readily account for the small irradiation effect shown for 0.040-inch- (0.1016-cm-) diameter specimens. The same size effect may also be responsible for the smaller decreases in total elongation of test material A.

The results compared in figure 11 show that the irradiation damage threshold of CP Ti is not substantially different for irradiation temperatures in the range from 17 to about 573 K. This threshold occurs at about 1×10^{17} neutrons per square centimeter ($E > 1$ MeV (160 fJ)) and is indicated by an increase in the yield and ultimate strengths. This increase may also be accompanied by a slight decrease in ductility. Following attainment of the threshold, the radiation damage with increasing fluence appears to be markedly dependent on postirradiation temperature conditions. Irradiation temperature may also be an important factor in the magnitude of the irradiation effect; however, the data available for analysis are insufficient to establish a general trend.

Titanium - 5 Aluminum - 2.5 Tin Alloy

The titanium alloy containing 5 percent aluminum and 2.5 percent tin (Ti-5Al-2.5Sn) was employed as test material for investigating the influence of impurity elements on radiation damage at cryogenic temperature. Two heats of the alloy (table I), one having the normal level of impurity content and one having an extra low level of impurity content, were used.

Tests using each heat of material were conducted at 17 K following reactor irradiation at 17 K to 1×10^{17} and 10×10^{17} neutrons per square centimeter ($E > 0.5$ MeV (80 fJ)). Tests conducted at 17 K on unirradiated material were used as base line control data. Results of the data analyses are shown in figures 12 and 13.

Radiation damage threshold. - As may be seen from the solid curves of figure 12, irradiation of the normal impurity material at 17 K to 10×10^{17} neutrons per square centimeter increased the yield and ultimate strengths, with the yield strength increasing more rapidly than the ultimate strength as evidenced by the plot of yield- to ultimate-strength ratio. The total elongation was decreased slightly by the irradiation, while the reduction of area and tensile modulus of elasticity remained unchanged. These data indicate that the threshold for radiation damage at 17 K occurs prior to 1×10^{17} neutrons per square centimeter as evidenced by the marked increases in yield and ultimate strengths. Figure 13 shows that variability of the tensile modulus of elasticity, yield strength, and ultimate strength of normal impurity material probably increase with increasing fluence. The variability of ductility parameters, however, apparently decreases with increasing fluence.

Irradiation of the extra low impurity material at 17 K to 10×10^{17} neutrons per square centimeter caused essentially the same changes in 17 K tensile properties (dashed curves shown in fig. 12) with the exception of the reduction of area. The reduction of area shows a significant decrease following irradiation to 10×10^{17} neutrons per square centimeter. It should be noted also that following irradiation to 1×10^{17} neutrons per square centimeter the tensile modulus of elasticity decreased. This decrease is small, but it is statistically significant at the 0.05 level of significance. The threshold for radiation damage is higher than the normal impurity material and occurs in the interval between 1×10^{17} and 10×10^{17} neutrons per square centimeter. Figure 13 indicates that, in general, irradiation reduces the variability of tensile properties of extra low impurity material. However, following 10×10^{17} neutrons per square centimeter exposure, increases in the variability of the ultimate strength and reduction of area are indicated.

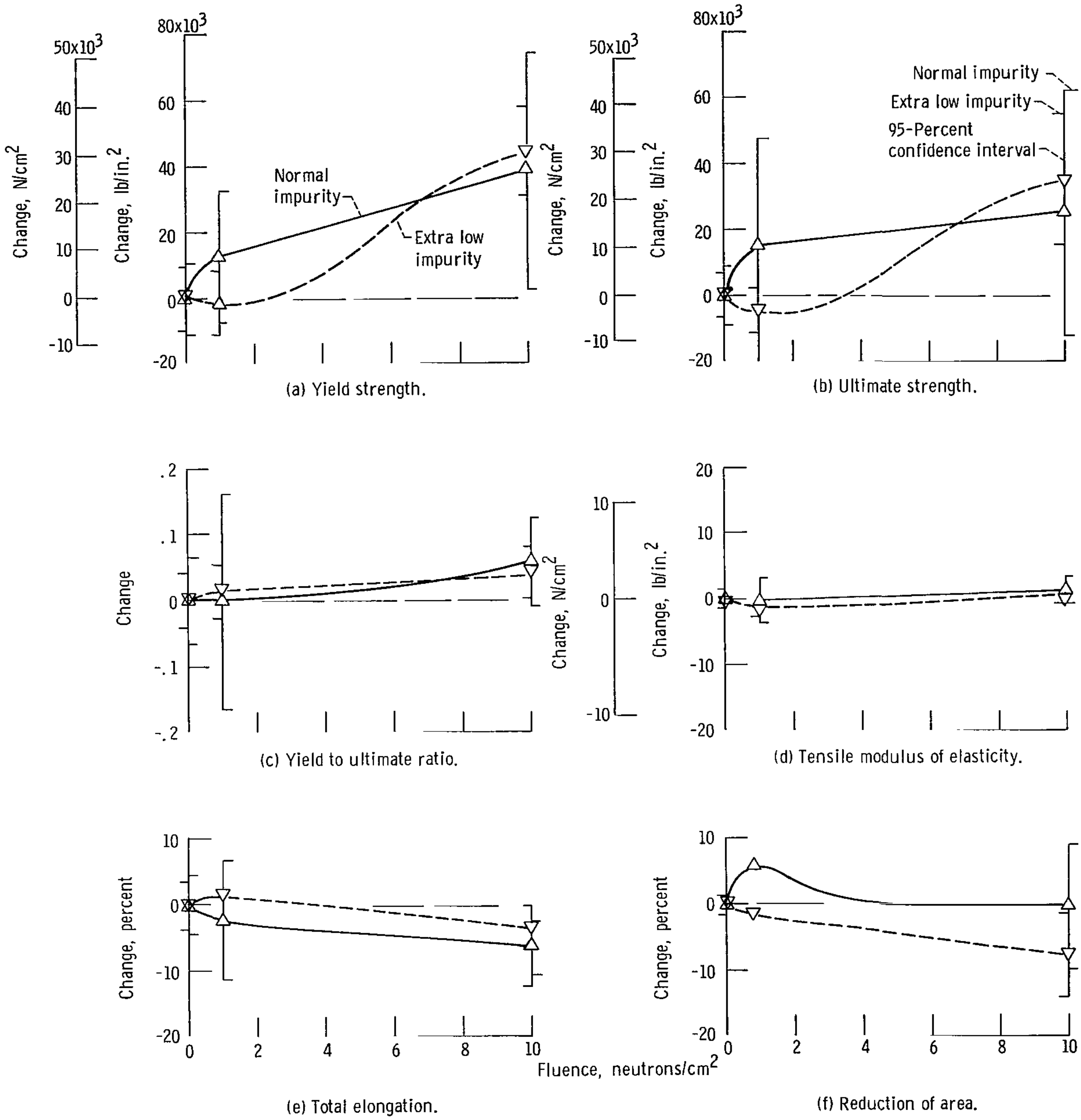


Figure 12. - Effect of reactor irradiation at 17 K on 17 K tensile properties of titanium - 5 aluminum - 2.5 tin alloy. Neutron energy > 0.5 MeV (80 fJ).

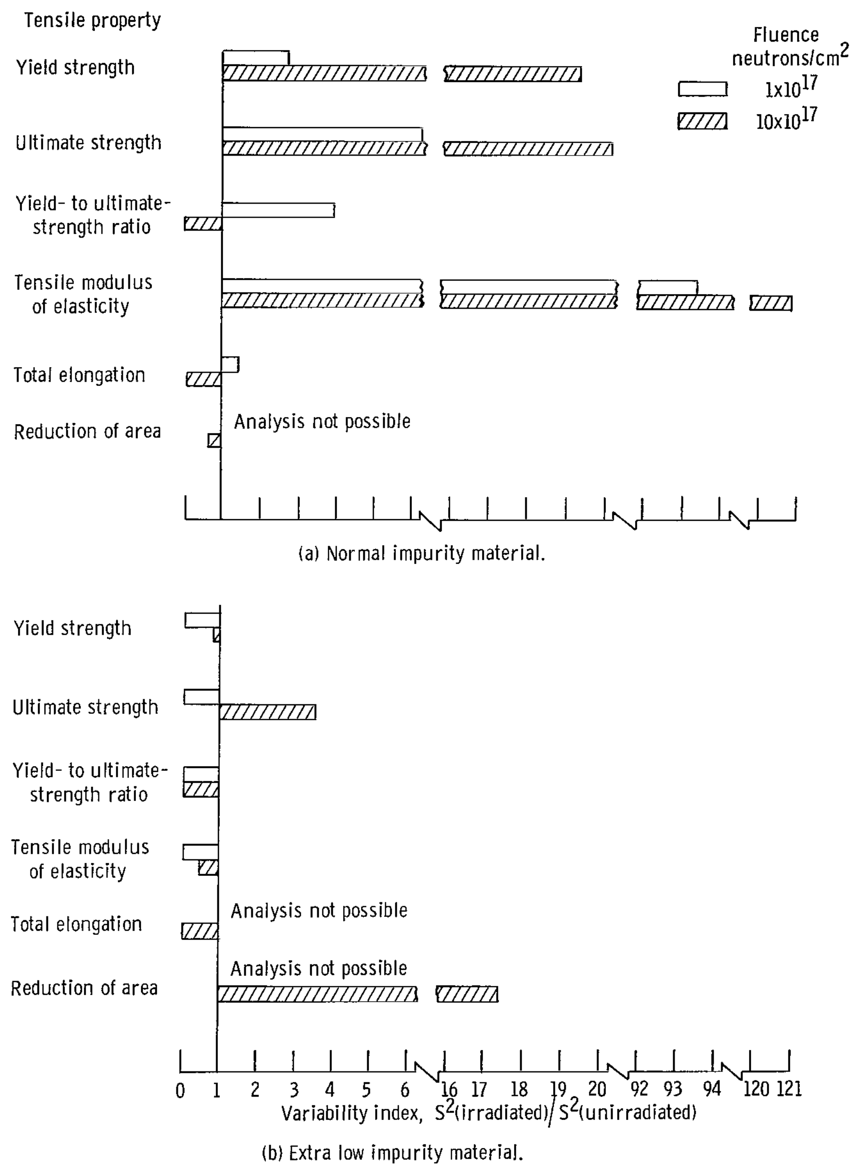


Figure 13. - Effect of reactor irradiation at 17 K on variability of 17 K tensile properties of titanium - 5 aluminum - 2.5 tin alloy. Neutron energy > 0.5 MeV (80 fJ).

Influence of impurity elements on radiation damage. - As for the effects of impurity elements on radiation damage, figure 12 shows that in every case there is an overlap of the 95-percent confidence intervals and in most cases the overlapped regions include the mean values for both normal impurity and extra low impurity materials. Notable exceptions occur for yield and ultimate strengths following 1×10^{17} neutrons per square centimeter. Here the 95-percent confidence intervals do overlap, but the difference in mean values of the yield and ultimate strengths for the two materials do not fall within the overlapped intervals. It would be presumptious to conclude from these data that there is a difference in the radiation damage attributable to impurity content except possibly for the yield and ultimate strengths following 1×10^{17} neutrons per square centimeter exposure.

The variability analyses shown in figure 13, however, suggest that impurity elements do influence the radiation effect. Based on this observation, it can be concluded that, although impurities may not necessarily alter the mean value, they may influence the variability.

Comparison of test results from various sources. - The results obtained from irradiation of Ti-5Al-2.5Sn alloy reported herein and by various investigations reported in the literature (refs. 14, 15, and 17 to 21) are compiled in table IV. These data represent the differences between average tensile property values for irradiated and unirradiated material and in many cases an average value represents only one specimen. Fluence values are for neutrons with energies greater than 1 MeV (160 fJ).

The results obtained for 20 and 78 K irradiation and test temperature are compared in figure 14 with a room temperature (300 K) test condition following irradiation at about 550 K. These data were generated on a variety of material forms, and in most cases, indicate that radiation damage is not particularly dependent on the form of the material. The data illustrated in figure 14 do indicate that the irradiation damage is sensitive to irradiation temperature, followed by testing at this temperature. For exposures greater than about 5×10^{17} neutrons per square centimeter, there appears to be an increase of damage as the irradiation and test temperature are reduced from 300 to 20 K. This increased irradiation damage is evidenced by a substantially greater increase in both yield and ultimate strengths and a decrease in total elongation for lower temperatures.

The changes in the reduction of area following irradiation, although small, show an interesting series of curves (fig. 14). Irradiation at about 550 K, followed by testing at room temperature (300 K) decreased the reduction of area. Lowering the irradiation and test temperature to 78 K resulted in little or no radiation effect. Further reduction in irradiation and test temperature to 20 K caused an increase in the reduction of area for exposures less than 5×10^{17} neutrons per square centimeter ($E > 1$ MeV (160 fJ)). (It will be recalled that an increase of comparable magnitude occurred for the reduction of area of CP Ti following irradiation and testing at 17 K.)

TABLE IV. - EFFECT OF REACTOR IRRADIATION ON TENSILE PROPERTIES OF TITANIUM - 5 ALUMINUM - 2.5 TIN ALLOY AS REPORTED BY VARIOUS INVESTIGATORS

Entry	Code	Test material		Exposure			Post-exposure temperature, K		Change in 0.2-percent offset yield strength		Change in ultimate strength		Change in total elongation, percent	Change in reduction of area, percent	Remarks	Reference
		Form	Condition	Medium	Temperature, K	Fluence, neutrons/cm^2 (E > 1 MeV (160 fJ))	Pretest	Test	$lb/in.^2$	N/cm^2	$lb/in.^2$	lb/cm^2				
1	A	Bar, NI	Annealed	Helium	17	7.6×10^{16}	17	17	12.7×10^3	8.8×10^3	15.8×10^3	10.9×10^3	-2.3	6.0	---	Present investigation
2	A	Bar, NI	Annealed	Helium	17	7.6×10^{17}	17	17	38.1	26.3	25.4	17.5	-6.4	-.2	---	
3	B	Bar, ELI	Annealed	Helium	17	7.6×10^{16}	17	17	-1.2×10^3	$-.8\times10^3$	-5.1×10^3	-3.5×10^3	1.3	-1.3	---	Present investigation
4	B	Bar, ELI	Annealed	Helium	17	7.6×10^{17}	17	17	44.2	30.5	35.4	24.4	-3.7	-7.6	---	
5	C	Bar, NI	Annealed	D_2O	523	1.2×10^{17}	300	300	1.7×10^3	1.2×10^3	1.0×10^3	0.7×10^3	[b]0.4	-4.9	---	14
6		↓	↓	UO_2SO_4	553	2.5×10^{18}	300	300	[a].9	[a].6	-.4	-.3	----	-8.2	---	15
7		↓	↓	↓	↓	5.0	300	300	[a]4.8	[a]3.3	2.3	1.6	----	-6.4	---	↓
8		↓	↓	↓	↓	2.5	573	573	[a]-2.0	[a]-1.4	-4.0	-2.8	----	-14.9	---	↓
9		↓	↓	↓	↓	5.0	573	573	[a]7.4	5.1	8.7	6.0	----	-23.0	---	↓
10	D	Bar, ELI	Annealed	Nitrogen	78	6.0×10^{17}	78	78	17.4×10^3	12.0×10^3	14.2×10^3	9.8×10^3	-11.7	-----	---	17
11	D	Bar, ELI	Annealed	Nitrogen	78	6.0	300	78	11.2	7.7	8.5	5.9	-6.8	-----	---	17
12	D	Bar, ELI	Annealed	Nitrogen	78	6.0	300	300	9.4	6.5	8.0	5.5	-3.7	-----	---	17
13	E	Sheet, NI	Annealed	Hydrogen	519	4.2×10^{16}	303	303	-1.5×10^3	-1.0×10^3	-3.0×10^3	-2.1×10^3	1.0	-----	(c)	18
14	E	Sheet, NI	Annealed	Hydrogen	63	3.2×10^{16}	303	303	-2.0	-1.4	2.0	1.4	.5	-----	(c)	18
15	E	Sheet, NI	Annealed	Hydrogen	24	1.4×10^{17}	20	20	-15.0	-10.3	-6.0	-4.1	-2.0	-----	(c)	18
16	F	Sheet, ELI	Annealed	Nitrogen	78	8.0×10^{17}	78	78	8.0×10^3	5.5×10^3	7.2×10^3	5.0×10^3	-2.4	-2.0	---	19
17	↓	↓	↓	↓	↓	↓	[d]300	↓	6.0	4.1	5.4	3.7	-2.4	-3.0	---	↓
18	↓	↓	↓	↓	↓	↓	[e]300	↓	6.0	4.1	5.5	3.8	-1.8	5.0	---	↓
19	↓	↓	↓	↓	↓	↓	[f]556	↓	4.0	2.8	1.9	1.3	-1.4	3.0	---	↓
20	↓	↓	↓	↓	↓	↓	[f]833	↓	0	0	.1	.1	1.0	3.0	---	↓
21	↓	↓	↓	↓	↓	↓	[g]300	300	1.0	.6	.2	.1	.2	2.0	---	↓

22	G	Plate, ELI (transverse)	Annealed	Hydrogen	20	5.0×10^{16}	20	20	5.6×10^{3}	3.9×10^{3}	10.1×10^{3}	7.0×10^{3}	1.7	-1.0	---	17
23	H	Plate, ELI	Annealed	Hydrogen	20	2.6×10^{17}	20	20	9.4×10^{3}	6.5×10^{3}	-0.4×10^{3}	-0.3×10^{3}	---	8.2	---	20
24				Hydrogen	20	3.5	20	20	11.4	7.9	12.8	8.8	1.3	9.2	---	20
25				Hydrogen	20	2.4	78	20	-1.0	-.7	-4.6	-3.2	---	5.2	---	21
26				Nitrogen	78	2.5	78	78	7.0	4.8	9.2	6.3	-.7	3.0	---	20
27				Nitrogen	78	3.4	78	78	.5	.3	0	0	2.6	1.0	---	20
28	I	Forging, ELI	Annealed	Nitrogen	78	1.1×10^{18}	78	78	14.8×10^{3}	10.2×10^{3}	12.7×10^{3}	8.8×10^{3}	-0.7	-1.2	---	29
29	J	Forging, ELI	As-forged	Nitrogen	78	1.1×10^{18}	78	78	14.0×10^{3}	9.6×10^{3}	12.2×10^{3}	8.4×10^{3}	-7.9	-3.0	---	19
30						1.1	189		13.0	9.0	11.6	8.0	-5.5	-7.0	---	
31						9.7×10^{17}	[d]300		11.0	7.6	8.3	5.7	-5.7	-6.0	---	
32						5.6	[g]300		6.0	4.1	3.5	2.4	-9.7	-2.0	---	
33						5.6	[e]300		5.0	3.4	4.7	3.2	-6.3	-4.0	---	
34						9.3	[f]556		-2.0	-1.4	-1.7	-1.2	-.4	-6.0	---	
35						9.3	[f]833		-6.0	-4.1	-7.5	-5.2	-6.9	-13.0	---	
36						8.5	189	189	5.0	3.4	-4.1	-2.8	-1.8	-10.0		
37						8.5	300	300	2.0	1.4	1.7	1.2	-2.0	-2.0	---	
38	K	Forging, ELI	Annealed	Nitrogen	78	7.9×10^{17}	78	78	16.7×10^{3}	11.5×10^{3}	14.4×10^{3}	9.9×10^{3}	----	-3.3	---	21
39	K	Forging, ELI	Annealed	Nitrogen	78	7.9	300	78	11.2	7.7	9.2	6.3	----	-4.8	---	21

[a]Proportional limit rather than 0.2-percent offset yield strength.

[b]Investigators also report 0.5-percent increase in uniform elongation and 0.4-percent increase in necking elongation.

[c]Investigator reported fluence as E > 0.33 MeV (53 fJ). Converted using relation E > 1 MeV ≈ 0.7 E > 0.33 MeV (E > 160 fJ ≈ 0.7 E > 53 fJ).

[d]Fifteen minutes at annealing temperature.

[e]Twenty-four hours at annealing temperature.

[f]One hour at annealing temperature.

[g]Five minutes at annealing temperature.

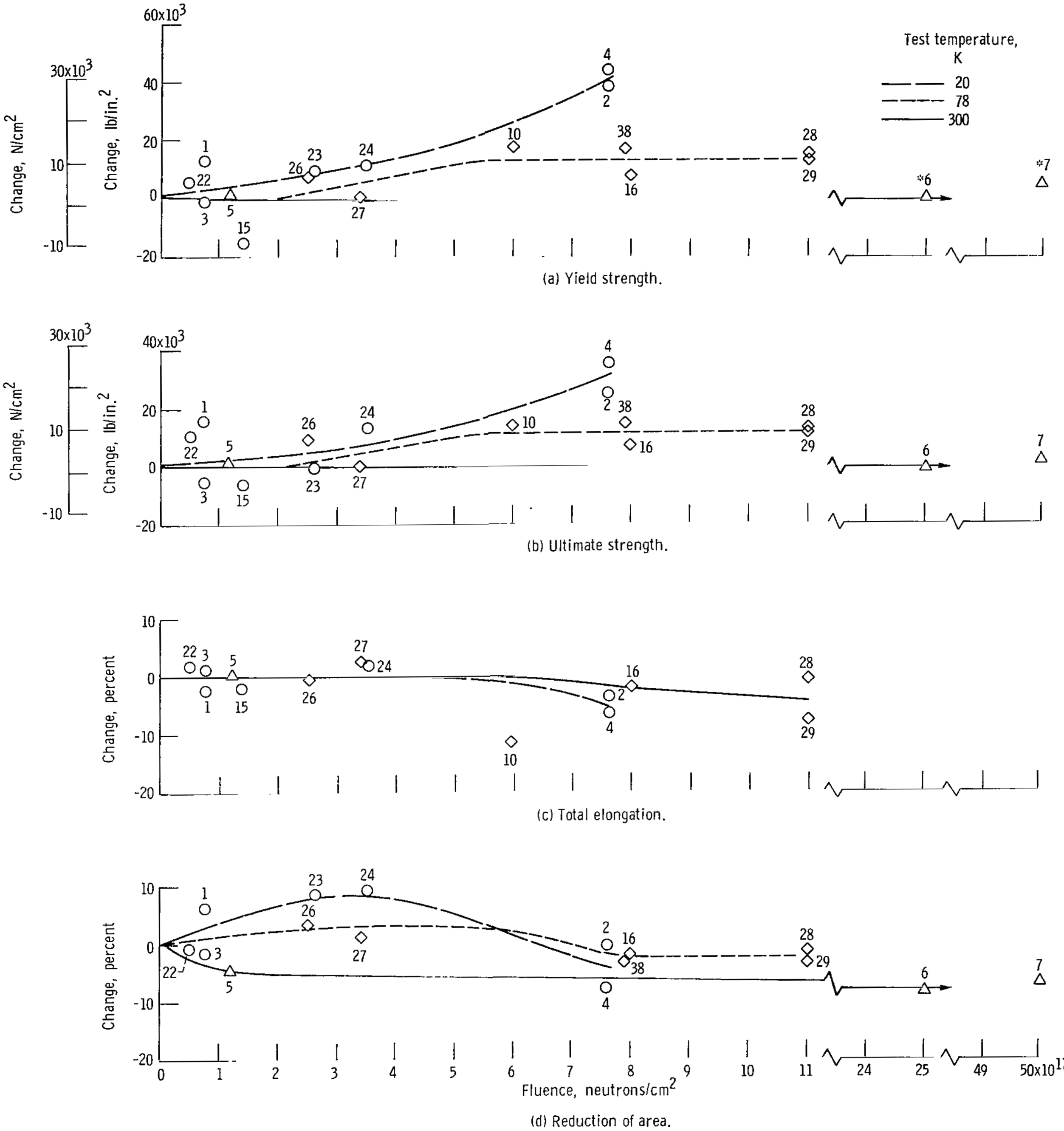


Figure 14. - Effect of reactor irradiation on tensile properties of titanium - 5 aluminum - 2.5 tin alloy as reported by various investigators. Neutron energy > 1 MeV (160 fJ). (Numbers beside plotted points correspond to entries in table IV; *, proportional limit.)

Finally, from table IV, it is noted that recovery of the radiation damage for the Ti-5Al-2.5Sn alloy requires postirradiation temperatures of the order of 800 K.

Titanium - 6 Aluminum - 4 Vanadium Alloy

Two heats of the titanium alloy containing 6 percent aluminum and 4 percent vanadium (Ti-6Al-4V) were used for cryogenic irradiation studies. One heat of the material was in the annealed condition, while the second heat of material was solution treated and then aged.

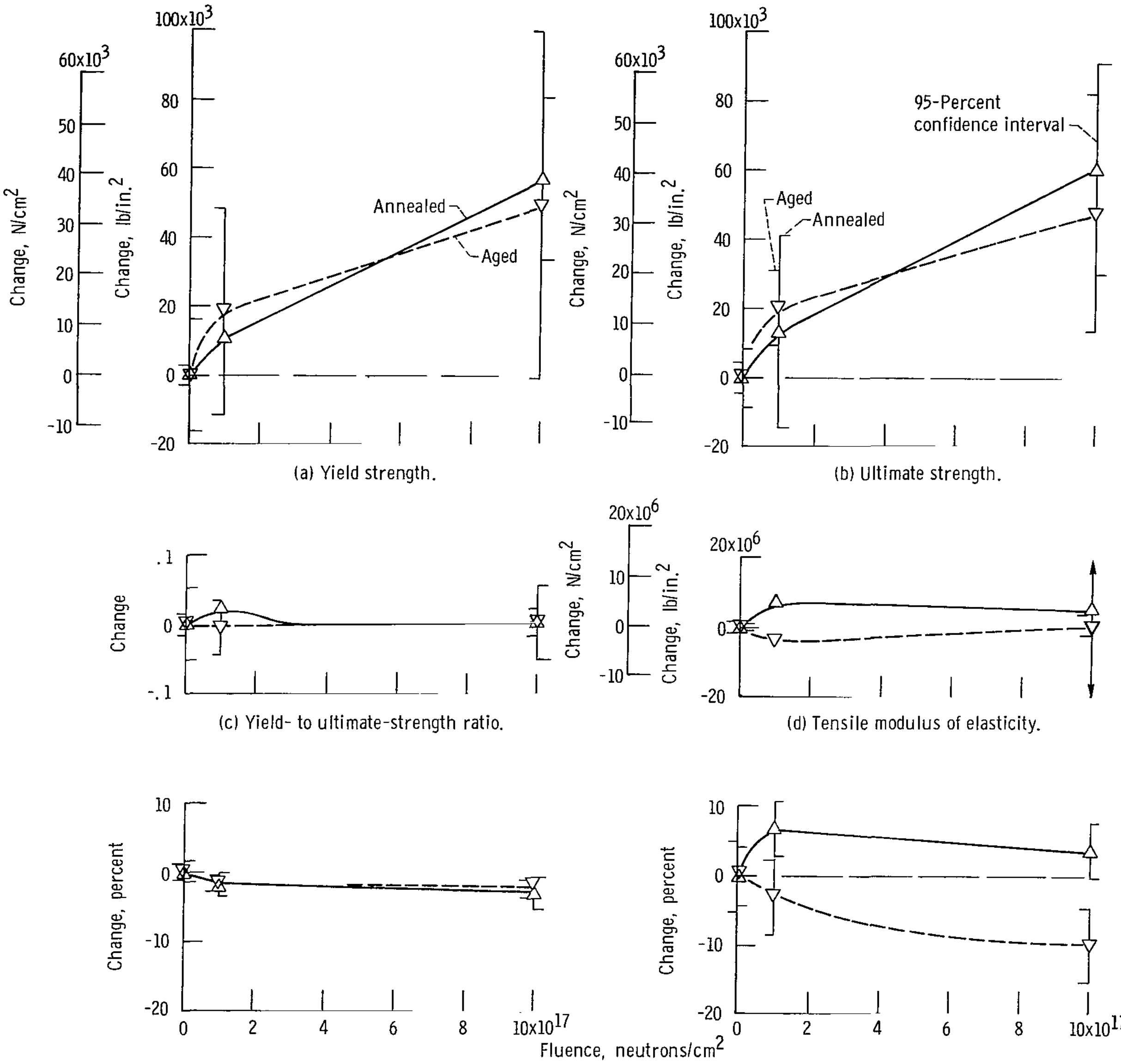
Tests using each heat of material were conducted at 17 K following reactor irradiation at 17 K to 1×10^{17} and 10×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). Tests conducted at 17 K on unirradiated materials were used as base-line control data. The results of data analysis are shown in figures 15 and 16.

Radiation damage threshold. - As may be seen from figure 15, with the exception of the reduction of area, there is little difference in the irradiation effect on annealed material and aged material. For both material conditions the radiation damage threshold occurs for exposures less than to 1×10^{17} neutrons per square centimeter, as evidenced by increases in yield and ultimate strengths, a decrease in total elongation, and changes in the reduction of area. The decreases in total elongation are small; however, they are statistically significant at the 0.05 level of significance.

The reduction of area shows that irradiation of annealed material causes an increase, whereas irradiation of the aged material causes a decrease. The 95-percent confidence intervals are approximately equal for all differences. Following irradiation to 1×10^{17} and 10×10^{17} neutrons per square centimeter, these confidence intervals do not overlap, which indicates that there is a difference in the effect of irradiation at 17 K on annealed material and aged material.

The difference indicated by the tensile modulus of elasticity is probably exaggerated since the variability of the annealed material following irradiation to 10×10^{17} neutrons per square centimeter is unusually large and data from only one specimen for each heat treated condition were obtained following irradiation to 1×10^{17} neutrons per square centimeter.

Variability analyses, illustrated by figure 16, indicate that the variability of the strength parameters increases with increasing fluence for both the annealed material and the aged material. Ductility parameters, on the other hand, show that following irradiation the variability in ductility tends to decrease for both the annealed material and the aged material. The trends in the variability indices for both annealed and aged materials are similar. The magnitude of the increase in the variability of the yield and ultimate strengths for aged material appears greater than that for annealed material. This may be due to a higher oxygen content in the aged material, 0.102 weight percent as op-



(e) Total elongation.

(f) Reduction of area.

Figure 15. - Effect of reactor irradiation at 17 K on 17 K tensile properties of titanium - 6 aluminum - 4 vanadium alloy. Neutron energy > 0.5 MeV (80 fJ).

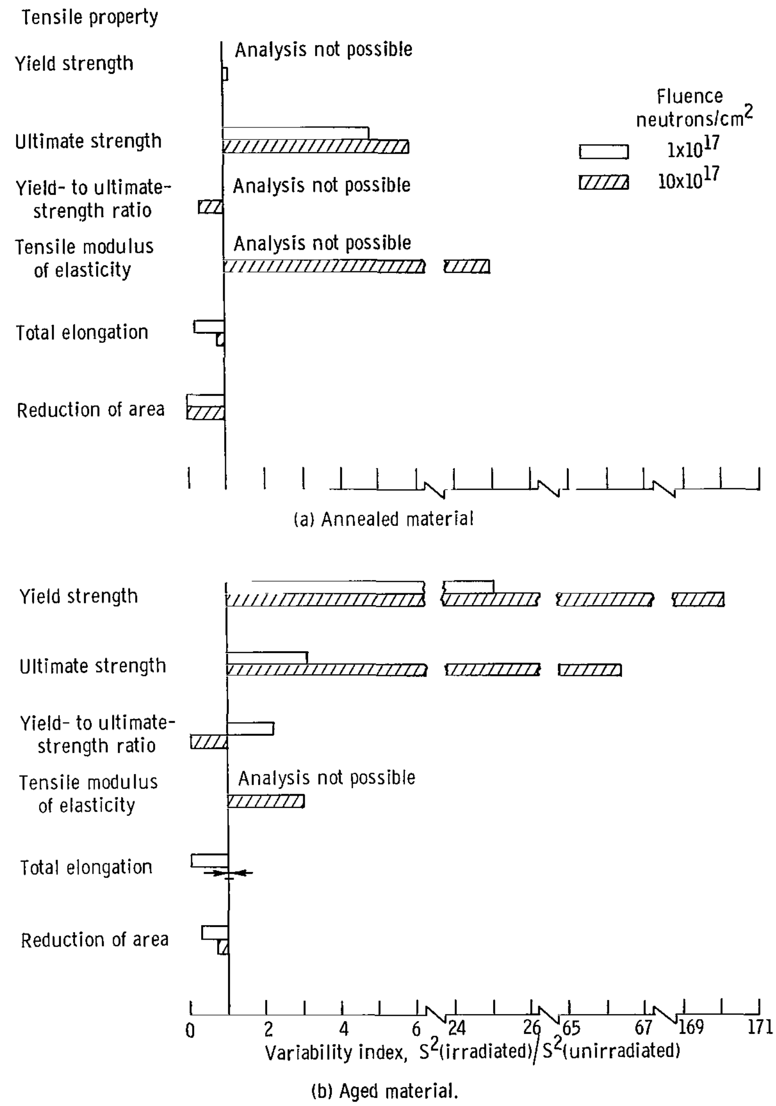


Figure 16. - Effect of reactor irradiation at 17 K on variability of 17 K tensile properties of titanium - 6 aluminum - 4 vanadium alloy. Neutron energy > 0.5 MeV (80 fJ).

posed to 0.065 weight percent for annealed material.

From these data, the only indication that metallurgical condition influences radiation damage at 17 K is a small difference in the reduction of area following exposure to 10×10^{17} neutrons per square centimeter. This difference represents a decrease in the ductility of aged material which is not evident in the annealed material.

Wallack (ref. 22) reports results on the Ti-6Al-4V alloy which indicate that differences due to the metallurgical condition may occur for strength parameters following higher fluence exposures. Test results reported include hardness measurements at room temperature on both annealed and aged conditions of Ti-6Al-4V unirradiated and following reactor irradiation at about 373 K to 3×10^{18}, 2×10^{19}, and 8.5×10^{19} neutrons per square centimeter ($E > 1$ MeV (160 fJ)). Irradiation caused an increase in the hardness for both material conditions, the increase being greater for the aged material. Since an increase in hardness implies an increase in ultimate tensile strength, it may be concluded that the ultimate tensile strength of aged material increased more than that of annealed material. Furthermore, X-ray diffraction measurements indicated that irradiation caused lattice parameter and line breadth changes in the aged material similar to an overaging treatment of unirradiated material.

Comparison of test results from various sources. - Only one other investigation of the effects of reactor irradiation on the tensile properties of the Ti-6Al-4V alloy was found in the literature. Results from this investigation (ref. 14) are compiled in table V along with results from the present investigation converted to fluence values for neutrons with energies greater than 1 MeV (160 fJ). The comparison of test results show that for the strength properties, irradiation at 17 K to about 1×10^{17} neutrons per square centimeter followed by testing at 17 K causes damage equivalent to irradiation at 523 K to 2.6×10^{18} neutrons per square centimeter (followed by testing at room temperature). Total elongation and reduction of area, however, decreased considerably more for the higher fluence exposure (followed by testing at room temperature). This result is consistent with what has been observed for other titanium alloys, although the magnitude of the decrease in reduction of area is greater for the Ti-6Al-4V alloy.

The effect of irradiation on room temperature elongation reported by Bomar et al. (ref. 14) shows that both uniform and necking elongation are decreased by irradiation. This is contrary to the effect noted previously for CP Ti (fig. 11) which showed only the necking elongation to decrease following irradiation. This effect may be due to alloy content and/or metallurgical structure.

TABLE V. - EFFECT OF REACTOR IRRADIATION ON TENSILE PROPERTIES OF TITANIUM - 6 ALUMINUM - 4 VANADIUM ALLOY

AS REPORTED BY VARIOUS INVESTIGATORS

Code	Test material		Exposure			Post-exposure temperature, K		Change in 0.2-percent offset yield strength		Change in ultimate strength		Change in total elongation, percent	Change in reduction of area, percent	Reference
	Form	Condition	Medium	Temperature, K	Fast fluence, neutrons/cm^2 (a)	Pretest	Test	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
A	Bar	Annealed	Helium	17 17	7.6×10^{16} 7.6×10^{17}	17 17	17 17	10.8×10^{3} 56.3	7.4×10^{3} 38.8	13.3×10^{3} 59.8	9.2×10^{3} 41.2	-1.9 -2.9	6.9 3.6	Present investigation
B	Bar	Aged	Helium	17 17	7.6×10^{16} 7.6×10^{17}	17 17	17 17	18.3×10^{3} 49.0	12.6×10^{3} 33.8	20.1×10^{3} 47.3	13.8×10^{3} 32.6	-1.4 -2.0	-2.9 -10.0	Present investigation
C	Bar	Annealed	UO_2SO_4	523	2.6×10^{18}	300	300	18.6×10^{3}	12.8×10^{3}	16.6×10^{3}	11.4×10^{3}	[b]-6.3	-28.8	14

[a]Fast fluence is for neutrons with $E > 1$ MeV (160 fJ).

[b]Investigators also report a 2.4-percent decrease in uniform elongation and a 3.4-percent decrease in necking elongation.

Titanium - 8 Aluminum - 1 Molybdenum - 1 Vanadium Alloy

The titanium alloy containing 8 percent aluminum, 1 percent molybdenum, and 1 percent vanadium (Ti-8Al-1Mo-1V) was irradiated at 17 K to a fluence of 1×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)) to determine the threshold for radiation damage. Tests conducted at 17 K on unirradiated material were used as base-line control data. Data analyses are shown in figures 17 and 18.

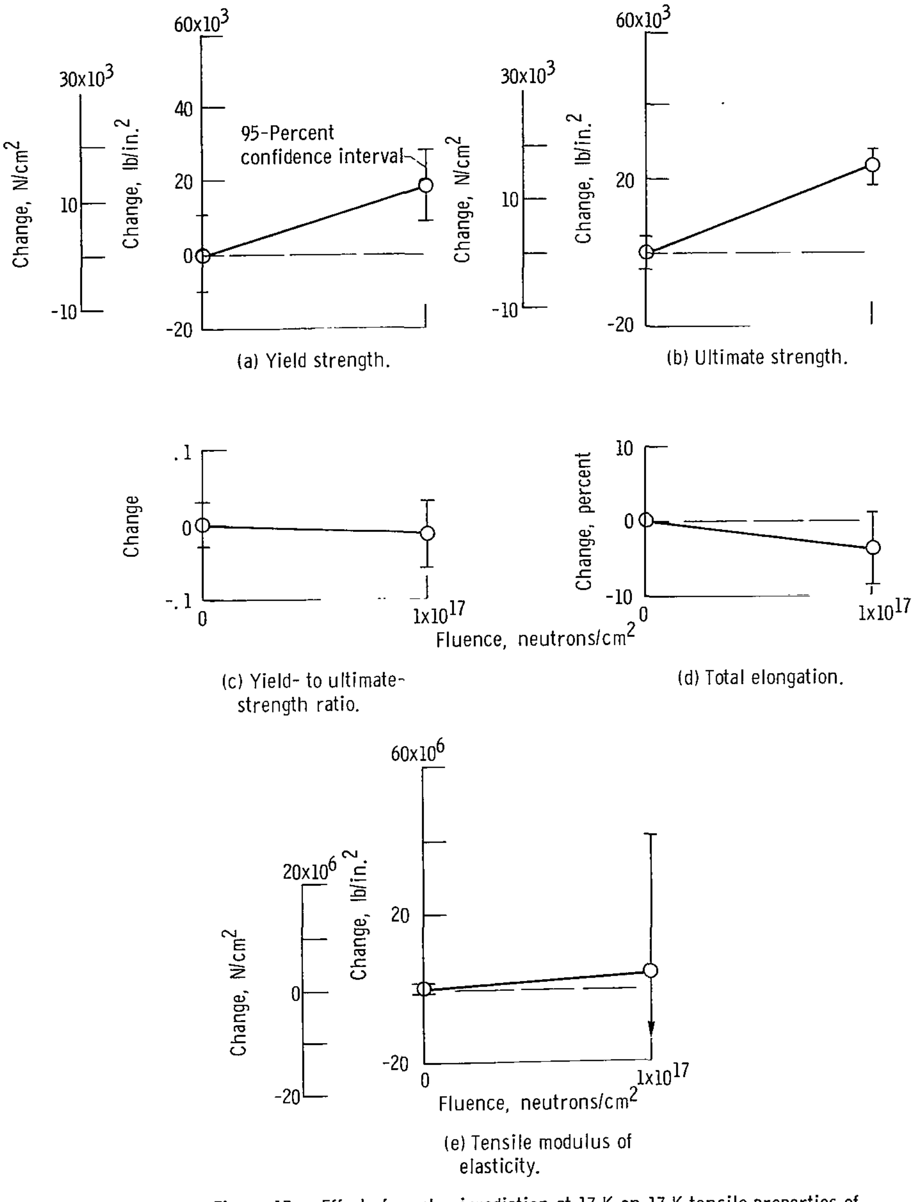


Figure 17. - Effect of reactor irradiation at 17 K on 17 K tensile properties of titanium - 8 aluminum - 1 molybdenum - 1 vanadium alloy. Neutron energy > 0.5 MeV (80 fJ).

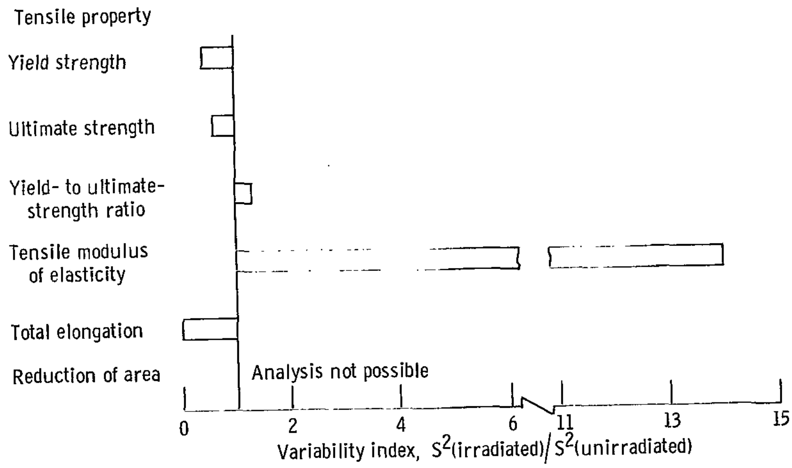


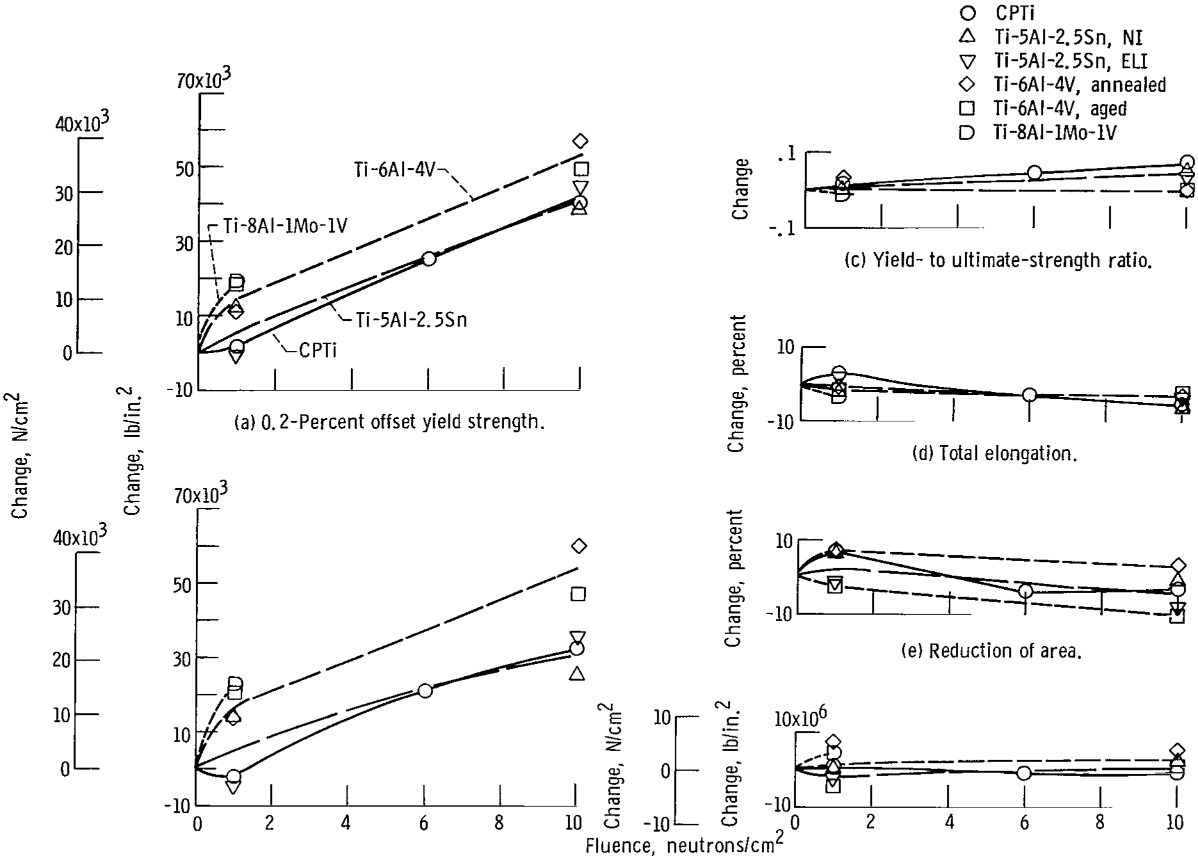
Figure 18. - Effect of reactor irradiation at 17 K to 1×10^{17} neutrons per square centimeter (energy > 0.5 MeV (80 fJ) on variability of 17 K tensile properties of titanium - 8 aluminum - 1 molybdenum - 1 vanadium alloy.

Radiation damage threshold. - As can be seen from figure 17, the threshold for radiation damage at 17 K occurs for exposures less than 1×10^{17} neutrons per square centimeter as evidenced by the increases in both yield and ultimate strengths. The total elongation following this exposure appears to decrease; however, the decrease is small and not statistically significant at the 0.05 level of significance. The indicated increase in the tensile modulus of elasticity is subject to considerable uncertainty since it is based on only two irradiated specimen values, both of which appear to be unusually high. The irradiated values and the unusually large 95-percent confidence interval are most probably due to test techniques rather than irradiation. The variability of the irradiated material (fig. 18) appears to decrease as evidenced by the variability indices being less than 1.0.

Comparison of test results from various sources. - Test results reported herein for the Ti-8Al-1Mo-1V alloy are the only known studies of the effect of reactor irradiation on this material.

Comparison of Radiation Damage to Titanium-Base Alloys

The effect of reactor irradiation at 17 K on 17 K tensile properties of CP Ti and the three titanium-base alloys is shown in figure 19. The points plotted on this figure are the differences in the mean values of each tensile property for each alloy and alloy condition. The curves for Ti-5Al-2.5Sn are the average for normal impurity and extra low impurity heats of material. The curves for Ti-6Al-4V are the average for annealed and aged conditions (with the exception of reduction of area, where curves representing the



(b) Ultimate strength.

(f) Tensile modulus of elasticity.

Figure 19. - Effect of reactor irradiation at 17 K on 17 K tensile properties of titanium and titanium-base alloys. Neutron energy > 0.5 MeV (80 fJ).

two material conditions are shown).

The comparisons of figure 19 show that irradiation at 17 K to 10×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)) causes similar changes in the 17 K tensile properties of all alloys. Tensile strengths increase (with the yield strength increasing more than the ultimate strength) and ductilities decrease slightly. There is also an indication that the magnitude of the change due to irradiation depends on total alloy content and/or metallurgical structure.

SUMMARY OF RESULTS AND CONCLUSIONS

Commercially pure titanium and three titanium-base alloys have been subjected to reactor irradiation at 17 K for fluence levels up to 10×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). Postirradiation tensile tests were conducted at 17 K without intervening warmup, except for a few tests using CP Ti. Test results show the following:

1. The irradiation damage threshold at 17 K for CP Ti occurs in the region between 1×10^{17} and 6×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). This threshold is defined by a significant increase in both yield strength and ultimate strength and possibly a small decrease in ductility.

2. Thermally cycling CP Ti to 178 K following irradiation at 17 K to 6×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)) reduces the irradiation damage by annihilation of irradiation-induced defects. As a result of this annihilation, the irradiation damage to the yield and ultimate strengths is reduced by about 50 percent. Furthermore, for the ultimate strength, there is an indication that additional recovery, attributable to reduced lattice resistance, occurs between 78 and 178 K.

3. Comparison of results reported herein with results from other sources shows that the irradiation damage threshold of CP Ti is not substantially different for irradiation temperatures in the range from 17 to about 573 K. This threshold occurs at about 1×10^{17} neutrons per square centimeter (E > 1 MeV (160 fJ)). Following attainment of the threshold, the radiation damage with increasing fluence appears to be markedly dependent on postirradiation temperature conditions.

4. The threshold for radiation damage at 17 K of Ti-5Al-2.5Sn alloy with normal impurity content occurs for exposures less than 1×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). For the same alloy with extra low impurity content, the threshold for radiation damage at 17 K is higher than the normal impurity material and occurs in the interval between 1×10^{17} and 10×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). These thresholds are indicated by significant increases in both yield and ultimate strengths and decreases in ductility. The differences in radiation damage thresholds shown by the Ti-5Al-2.5Sn alloys may be due to material variability. This is indicated by a comparison of results following 10×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)) which shows changes that are essentially the same for both impurity content materials.

5. Comparison of test results reported herein for the Ti-5Al-2.5Sn alloy with results of investigations reported in the literature suggest that irradiation damage is sensitive to irradiation temperature, followed by testing at this temperature. For exposures greater than about 5×10^{17} neutrons per square centimeter (E > 0.5 MeV (160 fJ)) there appears to be an increase of damage as the irradiation and test temperature are reduced from room temperature to 20 K.

6. The irradiation damage threshold at 17 K for the Ti-6Al-4V alloy occurs for exposures less than 1×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). The threshold is evidenced by significant increases in yield and ultimate strengths and decreases in ductility parameters. Differences in the reduction of area following irradiation are the only indications that the heat-treated condition influences the irradiation effect.

7. Comparison of test results for the Ti-6Al-4V alloy reported herein with results reported in the literature indicate that irradiation at 17 K to about 1×10^{17} neutrons per square centimeter (E > 1 MeV (160 fJ)), followed by testing at 17 K, causes damage equivalent to irradiation at 523 K to 2.6×10^{18} neutrons per square centimeter followed by testing at room temperature.

8. The irradiation damage threshold at 17 K for the Ti-8Al-1Mo-1V alloy occurs for exposures less than 1×10^{17} neutrons per square centimeter (E > 0.5 MeV (80 fJ)). This threshold is indicated by significant increases in both the yield and ultimate strengths and probable decrease in ductility.

9. The effect of reactor irradiation at 17 K on 17 K tensile properties of CP Ti and the three titanium-base alloys investigated are similar. Irradiation damage is evidenced by significant increases in strength parameters and probable decreases in ductility parameters. The magnitude of radiation damage appears to depend on the total alloy content and/or metallurgical structure.

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National Aeronautics and Space Administration,
Cleveland, Ohio, June 25, 1969,
122-29.

APPENDIX - TABULATED TEST DATA AND SUMMARY OF STATISTICAL ANALYSES

Tensile test data obtained during performance of the data acquisition phase of the program are compiled in tables VI to XIII. Summaries of the statistical analyses performed in connection with data analyses are also included in the tables. Symbols used on the tables for statistical analysis are defined as follows:

$X_A = \frac{1}{n_A} \sum_{i=1}^{n} X_i$	arithmetic mean of n_A measurements yielding property values of $X_1, X_2, \ldots$
$s_A = \sqrt{\frac{\sum_{i=1}^{n} (X_i - \bar{X}_A)^2}{n_A - 1}}$	estimated standard deviation of n_A measurements yielding property values of $X_1, X_2, \ldots$
$X_L = (\bar{X}_A - \bar{X}_A) - u_A$ or $(\bar{X}_B - \bar{X}_A) - u_{BA}$	lower limit of the 95-percent confidence interval of the difference between arithmetic means
$X_U = (\bar{X}_A - \bar{X}_A) + u_A$ or $(\bar{X}_B - \bar{X}_A) + u_{BA}$	upper limit of 95-percent confidence interval of difference between arithmetic means
$u_A = (t_{0.975}) \frac{s_A}{\sqrt{n_A}}$	for $n_A - 1$ degrees of freedom

$$u_{BA} = (t_{0.975}) \sqrt{\frac{s_A^2}{n_A} + \frac{s_B^2}{n_B}}$$

for $\dfrac{\left(\dfrac{s_A^2}{n_A} + \dfrac{s_B^2}{n_B}\right)^2}{\dfrac{\left(s_A^2/n_A\right)^2}{n_A + 1} + \dfrac{\left(s_B^2/n_B\right)^2}{n_B + 1}} - 2$ effective number of degrees of freedom

$t_{0.975}$ distribution value for "t" test of significance (e.g., ref. 10, table A-4)

TABLE VI. - EFFECT OF REACTOR IRRADIATION AT 17 K ON 17 K TENSILE PROPERTIES

OF COMMERCIALLY PURE TITANIUM[a]

Specimen	Fast fluence, neutrons/cm^2 (b)	Statistical analysis (c)	Tensile modulus of elasticity, lb/$in.^2$	Tensile modulus of elasticity, N/cm^2	Yield strength (0.2-percent offset), lb/$in.^2$	Yield strength (0.2-percent offset), N/cm^2	Ultimate strength, lb/$in.^2$	Ultimate strength, N/cm^2	Yield- to ultimate-strength ratio	Total elongation, percent (d)	Reduction of area, percent
1Aa211	0	--------	19.0×10^6	13.1×10^6	126.0×10^3	86.8×10^3	185.0×10^3	127.5×10^3	0.681	32.0	44.0
1Aa3	0	--------	18.0	12.4	131.0	90.3	182.0	125.4	.719	31.0	48.0
1Aa2	0	--------	19.0	13.1	134.0	92.3	188.0	129.5	.712	30.0	47.0
		$\bar{X}_A$	18.7×10^6	12.9×10^6	130.3×10^3	89.8×10^3	185.0×10^3	127.5×10^3	0.704	31.0	46.3
		s_A	.6	.4	4.1	2.8	3.0	2.1	.020	1.0	2.1
		$\bar{X}_A - \bar{X}_A$	0	0	0	0	0	0	0	0	0
		X_L	-1.5	-1.0	-10.2	-7.0	-7.4	-5.1	-.050	-2.5	-5.2
		X_U	1.5	1.0	10.2	7.0	7.4	5.1	.050	2.5	5.2
1Aa138	1×10^{17}	--------	18.4×10^6	12.7×10^6	128.0×10^3	88.2×10^3	181.0×10^3	124.7×10^3	0.706	----	---
1Aa148	1	--------	18.5	12.7	131.0	90.3	180.0	124.0	.728	36.0	54.0
1Aa144	1	--------	21.4	14.7	136.0	93.7	187.0	128.8	.726	32.0	52.0
		$\bar{X}_B$	19.4×10^6	13.4×10^6	131.7×10^3	90.7×10^3	182.7×10^3	125.9×10^3	0.720	34.0	53.0
		s_B	1.7	1.2	4.0	2.8	3.7	2.6	.012	2.8	1.4
		$\bar{X}_B - \bar{X}_A$	.7	.5	1.4	.9	-2.3	-1.6	.016	3.0	6.7
		X_L	-2.6	-1.8	-6.7	-4.6	-9.4	-6.5	-.020	-5.9	2.7
		X_U	4.0	2.8	9.5	6.5	4.8	3.3	.052	11.9	10.7
		${}^e s_B^2/s_A^2$	8.03	9.00	.95	1.00	1.52	1.53	.36	7.84	.44
1Aa200	6×10^{17}	--------	20.0×10^6	13.8×10^6	154.0×10^3	106.1×10^3	203.0×10^3	139.9×10^3	0.757	29.0	45.0
1Aa153	6	--------	15.0	10.3	154.0	106.1	211.0	145.4	.729	27.0	38.0
1Aa203	6	--------	19.0	13.1	158.0	108.9	204.0	140.6	.773	29.0	46.0
		$\bar{X}_C$	18.0×10^6	12.4×10^6	155.3×10^3	107.0×10^3	206.0×10^3	142.0×10^3	0.753	28.3	43.0
		s_C	2.6	1.8	2.3	1.6	4.4	3.0	.022	1.2	4.4
		$\bar{X}_C - \bar{X}_A$	-.7	-.5	25.0	17.2	21.0	14.5	.049	-2.7	-3.3
		X_L	-7.3	-5.1	17.5	12.1	13.4	9.2	.007	-4.9	-11.1
		X_U	5.9	4.1	32.5	22.4	29.2	20.2	.091	-.5	4.5
		${}^e s_C^2/s_A^2$	18.78	22.56	.31	.32	2.15	2.02	1.21	1.44	4.38
1Aa152	10×10^{17}	--------	14.2×10^6	9.8×10^6	159.0×10^3	109.6×10^3	213.0×10^3	146.8×10^3	0.746	30.0	49.0
1Aa205	10	--------	22.2	15.3	171.0	117.8	216.0	148.8	.791	29.0	44.0
1Aa206	10	--------	----	----	181.0	124.7	223.0	153.6	.811	19.0	38.0
		$\bar{X}_D$	18.2×10^6	12.5×10^6	170.3×10^3	117.4×10^3	217.3×10^3	149.7×10^3	0.783	26.0	43.7
		s_D	5.7	3.9	11.0	7.6	5.1	3.5	.033	6.1	5.5
		$\bar{X}_D - \bar{X}_A$	-.5	-.4	40.0	27.6	32.3	22.2	.079	-5.0	-2.6
		X_L	[f]-52.0	[f]-35.9	18.4	12.7	23.1	15.9	.022	-20.4	-13.4
		X_U	51.0	35.1	61.6	42.4	42.1	29.0	.136	10.4	8.2
		${}^e s_D^2/s_A^2$	90.25	95.06	7.19	7.36	2.89	2.78	2.72	37.21	6.85

[a]Specimen exposed in gaseous helium and maintained at 17 K throughout test.

[b]Fast fluence is for $E > 0.5$ MeV (80 fJ).

[c]Arithmetic mean $\bar{X}$; estimated standard deviation s; lower and upper limits of 95 percent confidence interval of the difference, X_L, X_U.

[d]Total elongation in 0.5-in. (1.27-cm) gage length.

[e]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two systems of units.

[f]Value exceeds lower limit of possible change.

TABLE VII. - EFFECT OF WARMING FOLLOWING REACTOR IRRADIATION AT 17 K ON

TENSILE PROPERTIES OF COMMERCIALLY PURE TITANIUM[a]

Specimen	Fast fluence, neutrons/cm^2 (b)	Test temperature, K (c)	Statistical analysis (d)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (e)	Reduction of area, percent
				lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
1Aa201	0	78	--------	20.0×10^6	13.8×10^6	103.0×10^3	71.0×10^3	134.0×10^3	92.3×10^3	0.768	52.0	66.0
1Aa195	0	78	--------	17.0	11.7	106.0	73.0	137.0	94.4	.773	51.0	68.0
1Aa11	0	78	--------	18.0	12.4	106.0	73.0	138.0	95.1	.767	47.0	67.0
			$\bar{X}_E$	18.3×10^6	12.6×10^6	105.0×10^3	72.3×10^3	136.4×10^3	93.9×10^3	0.769	50.0	67.0
			s_E	1.5	1.0	1.7	1.2	2.1	1.5	.003	2.6	1.0
1Aa212	6×10^{17}	78	--------	18.0×10^6	12.4×10^6	116.0×10^3	79.9×10^3	145.0×10^3	99.9×10^3	0.800	43.0	63.0
1Aa1	6	78	--------	19.0	13.1	120.0	82.7	145.0	99.9	.827	39.0	66.0
1Aa42	6	78	--------	20.0	13.8	126.0	86.8	147.0	101.3	.857	41.0	63.0
			$\bar{X}_F$	19.0×10^6	13.1×10^6	120.6×10^3	83.1×10^3	145.7×10^3	100.4×10^3	0.828	41.0	64.0
			s_F	1.0	.7	5.0	3.4	1.2	.8	.029	2.0	1.7
			$\bar{X}_F - \bar{X}_E$	.7	.5	15.6	10.7	9.3	6.5	.059	-9.0	-3.0
			X_L	-2.0	-1.4	5.9	4.1	5.4	3.7	-.014	-13.6	-6.2
			X_U	3.4	2.3	25.3	17.4	13.2	9.1	.132	-4.4	.2
			$f_{s_F^2/s_E^2}$	.44	.49	8.64	8.02	.33	.27	93.44	.59	2.89
1Aa59	0	178	--------	17.0×10^6	11.7×10^6	76.9×10^3	53.0×10^3	94.2×10^3	64.9×10^3	0.817	30.0	64.0
1Aa51	0	178	--------	21.0	14.5	78.9	54.4	94.3	65.0	.837	26.0	62.0
1Aa17	0	178	--------	15.0	10.3	79.9	55.1	95.2	65.6	.839	35.0	61.0
			$\bar{X}_G$	17.7×10^6	12.2×10^6	78.6×10^3	54.2×10^3	94.6×10^3	65.2×10^3	0.831	30.3	62.3
			s_G	3.1	2.1	1.5	1.1	.6	.4	.012	4.5	1.5
1Aa23	6×10^{17}	178	--------	17.0×10^6	11.7×10^6	90.7×10^3	62.5×10^3	101.0×10^3	69.6×10^3	0.897	25.0	64.0
1Aa45	6	178	--------	18.0	12.4	92.6	63.8	100.0	68.9	.926	30.0	64.0
1Aa35	6	178	--------	20.0	13.8	92.7	63.9	101.0	69.6	.917	29.0	63.0
			$\bar{X}_H$	18.3×10^6	12.6×10^6	92.0×10^3	63.4×10^3	100.7×10^3	69.4×10^3	0.913	28.0	63.7
			s_H	1.5	1.0	1.1	.8	.6	.4	.015	2.6	.6
			$\bar{X}_A - \bar{X}_G$	.6	.4	13.4	9.2	6.1	4.2	.082	-2.3	1.4
			X_L	-4.9	-3.4	10.6	7.3	4.9	3.4	.055	-10.6	-1.5
			X_U	6.1	4.2	16.2	11.2	7.3	5.0	.109	6.0	4.3
			$f_{s_H^2/s_G^2}$	.23	.23	.54	.53	1.00	1.00	1.56	.33	.16

[a]Specimen exposed in gaseous helium through test.

[b]Unirradiated specimen cooled to 17 K and held for 1 hr prior to warming and testing. Irradiated specimen cooled to and held at 17 K throughout irradiation exposure. Fast fluence is for neutron energies greater than 0.5 MeV (80 fJ).

[c]Specimen warmed to and held for 1 hr at test temperature, then fractured at test temperature.

[d]Arithmetic mean, $\bar{X}$; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.

[e]Total elongation in 0.5-in. (1.27-cm) gage length.

[f]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two systems of units.

TABLE VIII. - EFFECT OF ANNEALING FOLLOWING REACTOR IRRADIATION AT 17 K ON TENSILE PROPERTIES OF COMMERCIALLY PURE TITANIUM[a]

Specimen	Fast fluence, neutrons/cm^2 (b)	Annealing temperature, K (c)	Statistical analysis (d)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total-elongation, percent (e)	Reduction of area, percent
				lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
1Aa34	0	78	--------	19.0×10^6	13.1×10^6	131.0×10^3	90.3×10^3	184.0×10^3	126.8×10^3	0.711	30.0	46.0
1Aa16	0	78	--------	18.0	12.4	135.0	93.0	188.0	129.5	.718	35.0	44.0
1Aa19	0	78	--------	19.0	13.1	136.0	93.7	184.0	126.8	.738	28.0	48.0
			$\bar{X}_I$	18.7×10^6	12.9×10^6	134.0×10^3	92.3×10^3	185.3×10^3	127.7×10^3	0.722	31.0	46.0
			s_I	.6	.4	2.6	1.8	2.3	1.6	.014	3.6	2.0
1Aa47	6×10^{17}	78	--------	19.0×10^6	13.1×10^6	151.0×10^3	104.0×10^3	196.0×10^3	135.0×10^3	0.770	30.0	47.0
1Aa4	6	78	--------	--------	--------	152.0	104.7	190.0	130.9	.799	24.0	47.0
1Aa14	6	78	--------	20.0	13.8	152.0	104.7	195.0	134.4	.779	26.0	44.0
			$\bar{X}_J$	19.5×10^6	13.4×10^6	151.7×10^3	104.5×10^3	193.7×10^3	133.4×10^3	0.783	26.7	46.0
			s_J	.7	.5	.6	.4	3.2	2.2	.015	3.1	1.7
			$\bar{X}_J - \bar{X}_I$	.8	.5	17.7	12.2	8.4	5.7	.061	-4.3	0
			X_L	-.9	-.6	11.1	7.6	2.6	1.8	.032	-11.0	-3.7
			X_U	2.5	1.7	24.3	16.8	14.2	9.8	.090	2.4	3.7
			$^f s_J^2/s_I^2$	1.36	1.56	.05	.05	1.94	1.89	1.15	.74	.72
1Aa25	0	178	--------	18.0×10^6	12.4×10^6	131.0×10^3	90.3×10^3	176.0×10^3	121.3×10^3	0.744	32.0	52.0
1Aa53	0	178	--------	19.0	13.1	134.0	92.3	184.0	126.8	.728	31.0	54.0
1Aa24	0	178	--------	18.0	12.4	134.0	92.3	186.0	128.2	.720	36.0	50.0
			$\bar{X}_K$	18.3×10^6	12.6×10^6	133.0×10^3	91.6×10^6	182.0×10^3	125.4×10^3	0.731	33.0	52.0
			s_K	.6	.4	1.7	1.2	5.3	3.7	.012	2.6	2.0
1Aa28	6×10^{17}	178	--------	19.0×10^6	13.1×10^6	145.0×10^3	99.9×10^3	192.0×10^3	132.3×10^3	0.755	29.0	48.0
1Aa50	6	178	--------	20.0	13.8	146.0	100.6	194.0	133.7	.752	30.0	47.0
1Aa41	6	178	--------	22.0	15.2	149.0	102.7	195.0	134.4	.764	27.0	46.0
			$\bar{X}_L$	20.3×10^6	14.0×10^6	146.7×10^3	101.1×10^3	193.7×10^3	133.5×10^3	0.757	28.7	47.0
			s_L	1.5	1.0	2.1	1.5	1.5	1.1	.006	1.5	1.0
			$\bar{X}_L - \bar{X}_K$	2.0	1.4	13.7	9.5	11.7	8.1	.026	-4.3	-5.0
			X_L	-.9	-.6	9.9	6.9	1.6	1.1	.004	-9.1	-8.6
			X_U	4.9	3.4	17.5	12.1	21.8	15.1	.048	.5	-1.4
			$^f s_L^2/s_K^2$	6.25	6.25	1.52	1.56	.08	.09	.25	.33	.25

[a]Specimen exposed in gaseous helium throughout test.

[b]Unirradiated specimen cooled to 17 K and held for 1 hr prior to annealing. Irradiated specimen cooled to and held at 17 K throughout irradiation exposure. Fast fluence is for E > 0.5 MeV (80 fJ).

[c]Specimen warmed from 17 K to annealing temperature, held 1 hr at annealing temperature, cooled to 17 K, held 1 hr at 17 K.

[d]Arithmetic mean, X; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.

[e]Total elongation in 0.5-in. (1.27-cm) gage length.

[f]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two systems of units.

TABLE IX. - EFFECT OF REACTOR IRRADIATION AT 17 K on 17 K TENSILE PROPERTIES

ON NORMAL IMPURITY TITANIUM - 5 ALUMINUM - 2.5 TIN ALLOY

Specimen	Fast fluence, neutrons/cm^2 (a)	Statistical analysis (b)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (c)	Reduction of area, percent
			lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
3Aa55	0	--------	18.0×10^6	12.4×10^6	200.0×10^3	137.8×10^3	231.0×10^3	159.2×10^3	0.865	18.0	21.0
3Aa54	0	--------	17.5	12.1	201.0	138.5	213.0	146.8	.943	12.0	30.0
3Aa30	0	--------	18.2	12.5	-----	-----	225.0	155.0	----	12.0	35.0
3Aa21	0	--------	18.4	12.7	205.0	141.2	229.0	157.8	.895	13.0	34.0
3Aa29	0	--------	18.2	12.5	215.0	148.1	226.0	155.7	.951	----	----
		$\bar{X}_A$	18.1×10^6	12.5×10^6	205.3×10^3	141.4×10^3	224.8×10^3	154.9×10^3	0.914	13.8	30.0
		s_A	.3	.2	6.9	4.7	7.0	4.8	.041	2.9	6.4
		$\bar{X}_A - \bar{X}_A$	0	0	0	0	0	0	0	0	0
		X_L	-.4	-.3	-11.0	-7.5	-8.7	-6.0	-.065	-4.6	-10.2
		X_U	.4	.3	11.0	7.5	8.7	6.0	.065	4.6	10.2
3Aa58	1×10^{17}	--------	14.8×10^6	10.2×10^6	211.0×10^3	145.4×10^3	257.0×10^3	177.0×10^3	0.821	---	----
3Aa52	1	--------	18.1	12.5	212.0	146.1	222.0	153.0	.955	9.0	----
3Aa57	1	--------	20.5	14.1	231.0	159.2	238.0	164.0	.970	14.0	36.0
		$\bar{X}_B$	17.8×10^6	12.3×10^6	218.0×10^3	150.2×10^3	239.0×10^3	164.7×10^3	0.915	11.5	36.0
		s_B	2.9	2.0	11.3	7.8	17.5	12.0	.082	3.5	----
		$\bar{X}_B - \bar{X}_A$	-.3	-.2	12.7	8.8	14.2	9.8	.001	-2.3	6.0
		X_L	-7.5	-5.2	-7.8	-5.4	-19.5	-13.4	-.164	-11.4	----
		X_U	6.9	4.8	33.2	22.9	47.9	33.0	.166	6.8	----
		$^{d}s_B^2/s_A^2$	93.44	100.00	2.68	2.76	6.25	6.24	4.00	1.46	----
3Aa73	10×10^{17}	--------	20.0×10^6	13.8×10^6	205.3×10^3	141.5×10^3	211.8×10^3	145.9×10^3	0.969	8.0	34.0
3Aa168	10	--------	20.0	13.8	220.6	152.0	226.8	156.3	.971	8.0	34.0
3Aa167	10	--------	16.0	11.0	248.2	171.0	250.8	172.8	.990	7.0	27.0
3Aa169	10	--------	22.0	15.2	262.8	181.1	270.8	186.6	.970	8.0	33.0
3Aa63	10	--------	25.0	17.2	279.9	192.9	289.2	199.3	.967	6.0	21.0
		$\bar{X}_C$	20.6×10^6	14.2×10^6	243.4×10^3	167.7×10^3	249.9×10^3	172.2×10^3	0.973	7.4	29.8
		s_C	3.3	2.3	30.4	21.0	31.5	21.7	.009	.9	5.3
		$\bar{X}_C - \bar{X}_A$	2.5	1.7	38.1	26.3	25.1	17.3	.059	-6.4	-.2
		X_L	-1.6	-1.1	2.0	1.4	-12.0	-8.3	-.008	-10.6	-9.7
		X_U	6.6	4.5	74.2	51.2	62.2	42.9	.126	-2.2	9.3
		$^{d}s_C^2/s_A^2$	121.00	132.25	19.39	19.96	20.25	20.42	.05	.10	.68

[a]Specimen exposed in gaseous helium at 17 K throughout test. Fast fluence is for $E > 0.5$ MeV (80 fJ).

[b]Arithmetic mean, $\bar{X}$; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.

[c]Total elongation in 0.5-in. (1.27-cm) gage length.

[d]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two systems of units.

TABLE X. - EFFECT OF REACTOR IRRADIATION AT 17 K ON 17 K TENSILE PROPERTIES OF EXTRA LOW IMPURITY TITANIUM - 5 ALUMINUM - 2.5 TIN ALLOY

Specimen	Fast fluence, neutrons/cm^2 (a)	Statistical analysis (b)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (c)	Reduction of area, percent
			lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
8Aa18	0	--------	--------	--------	203.0×10^3	139.9×10^3	231.0×10^3	159.2×10^3	0.879	---	----
8Aa32	0	--------	22.5×10^6	15.5×10^6	213.0	146.8	223.0	153.6	.955	8.0	33.0
8Aa35	0	--------	--------	--------	213.0	146.8	225.0	155.0	.947	11.0	32.0
8Aa34	0	--------	20.8	14.3	217.0	149.5	227.0	156.4	.956	10.0	32.0
8Aa27	0	--------	22.4	15.4	225.0	155.0	236.0	162.6	.954	----	----
		$\bar{X}_D$	21.9×10^6	15.1×10^6	214.2×10^3	147.6×10^3	228.4×10^3	157.4×10^3	0.938	9.7	32.3
		s_D	.9	.7	8.0	5.5	5.2	3.6	.033	1.5	.6
		$\bar{X}_D - \bar{X}_D$	0	0	0	0	0	0	0	0	0
		X_L	-2.2	-1.7	-9.9	-6.8	-6.4	-4.4	-.041	-3.7	-1.5
		X_U	2.2	1.7	9.9	6.8	6.4	4.4	.041	3.7	1.5
8Aa25	1×10^{17}	--------	18.7×10^6	12.9×10^6	211.0×10^3	145.4×10^3	222.0×10^3	153.0×10^3	0.950	11.0	31.0
8Aa24	1	--------	18.5	12.7	213.0	146.8	225.0	155.0	.947	----	----
8Aa12	1	--------	18.7	12.9	215.0	148.1	223.0	153.6	.964	----	----
		$\bar{X}_E$	18.6×10^6	12.8×10^6	213.0×10^3	146.8×10^3	223.3×10^3	153.9×10^3	0.953	11.0	31.0
		s_E	.1	.1	2.0	1.4	1.5	1.0	.009	----	----
		$\bar{X}_E - \bar{X}_D$	-3.3	-2.3	-1.2	-.8	-5.1	-3.5	.015	1.3	-1.3
		X_L	-5.5	-3.8	-10.7	-7.4	-11.2	-7.7	-.026	---	----
		X_U	-1.1	-.8	8.3	5.7	1.0	.7	.056	---	----
		[d]s_E^2/s_D^2	.01	.02	.06	.06	.08	.09	.07	---	----
8Aa60	10×10^{17}	--------	22.0×10^6	15.2×10^6	250.0×10^3	172.3×10^3	252.6×10^3	174.0×10^3	0.990	6.0	27.0
8Aa49	10	--------	22.0	15.2	262.1	180.6	268.0	184.7	.978	6.0	22.0
8Aa55	10	--------	23.0	15.8	263.1	181.3	270.9	186.7	.971	6.0	25.0
		$\bar{X}_F$	22.3×10^6	15.4×10^6	258.4×10^3	178.1×10^3	263.8×10^3	181.8×10^3	0.980	6.0	24.7
		s_F	.6	.4	7.3	5.0	9.8	6.8	.010	0	2.5
		$\bar{X}_F - \bar{X}_D$	.4	.3	44.2	30.5	35.4	24.4	.042	-3.7	-7.6
		X_L	-1.2	-.8	31.2	21.5	15.7	10.8	.003	-7.4	-14.0
		X_U	2.0	1.4	57.2	39.5	55.1	38.0	.081	0	-1.2
		[d]s_F^2/s_D^2	.44	.33	.83	.83	3.55	3.57	.09	0	17.36

[a]Specimen exposed in gaseous helium at 17 K throughout test. Fast fluence is for $E > 0.5$ MeV (80 fJ).

[b]Arithmetic mean, $\bar{X}$; estimated standard deviation, s; lower and upper limits of 95 percent confidence interval of the difference, X_L, X_U.

[c]Total elongation in 0.5-in. (1.27-cm) gage length.

[d]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two systems of units.

TABLE XI. - EFFECT OF REACTOR IRRADIATION AT 17 K ON 17 K TENSILE PROPERTIES OF ANNEALED TITANIUM - 6 ALUMINUM - 4 VANADIUM ALLOY

Specimen	Fast fluence, neutrons/cm^2 (a)	Statistical analysis (b)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (c)	Reduction of area, percent
			lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
2Ac1	0	--------	17.6×10^6	12.1×10^6	228.0×10^3	157.1×10^3	249.0×10^3	171.6×10^3	0.915	10.0	30.0
2Ac9	0	--------	16.3	11.2	230.0	158.5	264.0	181.9	.871	7.0	30.0
2Ac71	0	--------	17.7	12.2	250.0	172.3	265.0	182.6	.943	↓	36.0
2Ac10	0	--------	17.9	12.3	253.0	174.3	261.0	179.8	.969	↓	27.0
2Ac12	0	--------	16.8	11.6	255.0	175.7	263.0	181.2	.969	↓	29.0
		$\bar{X}_A$	17.3×10^6	11.9×10^6	243.2×10^3	167.6×10^3	260.4×10^3	179.4×10^3	.933	7.6	30.4
		s_A	.7	.5	13.1	9.0	6.5	4.5	.041	1.3	3.4
		$\bar{X}_A - \bar{X}_A$	0	0	0	0	0	0	0	0	0
		X_L	-.9	-.6	-16.2	-11.2	-8.1	-5.6	-.051	-1.5	-4.2
		X_U	.9	.6	16.2	11.2	8.1	5.6	.051	1.5	4.2
2Ac59	1×10^{17}	--------	--------	--------	---------	---------	265.0×10^3	182.6×10^3	-----	6.0	37.0
2Ac61	1	--------	24.8×10^6	17.1×10^6	254.0×10^3	175.0×10^3	266.0	183.3	0.955	5.0	37.0
2Ac72	1	--------	--------	--------	---------	---------	290.1	199.8	-----	6.0	38.0
		$\bar{X}_B$	24.8×10^6	17.1×10^6	254.0×10^3	175.0×10^3	273.7×10^3	188.6×10^3	0.955	5.7	37.3
		s_B	--------	--------	---------	---------	14.2	9.7	-----	.6	.6
		$\bar{X}_B - \bar{X}_A$	7.5	5.2	10.8	7.4	13.3	9.2	.022	-1.9	6.9
		X_L	--------	--------	---------	---------	-14.4	-9.9	-----	-3.5	2.9
		X_U	--------	--------	---------	---------	41.0	28.2	-----	-.3	10.9
		[d]s_B^2/s_A^2	--------	--------	---------	---------	4.76	4.64	-----	.21	.03
2Ac54	10×10^{17}	--------	20.0×10^6	13.8×10^6	289.4×10^3	199.4×10^3	302.7×10^3	208.6×10^3	0.956	4.0	34.0
2Ac56	10	--------	--------	--------	294.7	203.0	325.0	223.9	.907	6.0	34.0
2Ac55	10	--------	25.0	17.2	314.5	216.7	332.9	229.4	.945	4.0	34.0
		$\bar{X}_C$	22.5×10^6	15.5×10^6	299.5×10^3	206.4×10^3	320.2×10^3	220.6×10^3	0.936	4.7	34.0
		s_C	3.5	2.4	13.2	9.1	15.7	10.8	.026	1.2	0
		$\bar{X}_C - \bar{X}_A$	5.2	3.6	56.3	38.8	59.8	41.2	.003	-2.9	3.6
		X_L	[e]-26.6	[e]-17.8	32.8	22.6	29.5	20.3	-.052	-5.1	-.6
		X_U	37.0	25.0	79.8	55.0	90.1	62.1	.058	-.7	7.8
		[d]s_C^2/s_A^2	25.00	23.03	1.02	1.02	5.83	5.75	.40	.85	0

[a]Specimen exposed in gaseous helium at 17 K throughout test. Fast fluence is for E > 0.5 MeV (80 fJ).

[b]Arithmetic mean, X; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.

[c]Total elongation in 0.5-in. (1.27-cm) gage length.

[d]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two system of units.

[e]Value exceeds lower limit of possible change.

TABLE XII. - EFFECT OF REACTOR IRRADIATION AT 17 K ON 17 K TENSILE PROPERTIES OF AGED TITANIUM - 6 ALUMINUM - 4 VANADIUM ALLOY

Specimen	Fast fluence, neutrons/cm^2 (a)	Statistical analysis (b)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (c)	Reduction of area, percent
			lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
2Aa30	0	--------	20.4×10^6	14.1×10^6	273.0×10^3	188.1×10^3	286.0×10^3	197.1×10^3	0.954	6.0	22.0
2Aa11	0	--------	19.7	13.6	274.0	188.8	277.0	190.9	.989	7.0	28.0
2Aa29	0	--------	17.1	11.8	274.0	188.8	281.0	193.6	.975	7.0	24.0
2Aa10	0	--------	17.4	12.0	275.0	189.5	284.0	195.7	.969	7.0	31.0
2Aa13	0	--------	18.1	12.5	279.0	192.2	283.0	195.0	.986	5.0	22.0
		$\bar{X}_D$	18.5×10^6	12.8×10^6	275.0×10^3	189.5×10^3	282.2×10^3	194.5×10^3	0.975	6.4	25.4
		s_D	1.5	1.0	2.4	1.6	3.4	2.4	.014	.9	4.0
		$\bar{X}_D - \bar{X}_D$	0	0	0	0	0	0	0	0	0
		X_L	-1.9	-1.3	-3.0	-2.0	-4.2	-2.9	-.017	-1.1	-5.0
		X_U	1.9	1.3	3.0	2.0	4.2	2.9	.017	1.1	5.0
2Aa43	1×10^{17}	--------	--------	--------	281.0×10^3	193.6×10^3	296.0×10^3	203.9×10^3	0.949	5.0	24.0
2Aa44	1	--------	14.6×10^6	10.1×10^6	294.0	202.6	303.0	208.8	.970	---	----
2Aa77	1	--------	--------	--------	305.0	210.1	308.0	212.2	.990	5.0	21.0
		$\bar{X}_E$	14.6×10^6	10.1×10^6	293.3×10^3	202.1×10^3	302.3×10^3	208.3×10^3	0.970	5.0	22.5
		s_E	--------	--------	12.0	8.3	6.0	4.1	.021	0	2.1
		$\bar{X}_E - \bar{X}_D$	-3.9	-2.7	18.3	12.6	20.1	13.8	-.005	-1.4	-2.9
		X_L	--------	--------	-11.9	-8.2	9.6	6.6	-.044	-2.5	-8.4
		X_U	--------	--------	48.5	33.4	30.6	21.1	.034	-.3	2.6
		$^{d}s_E^2/s_D^2$	--------	--------	25.00	26.85	3.11	2.92	2.25	0	.28
2Aa78	10×10^{17}	--------	20.0×10^6	13.8×10^6	289.5×10^3	199.5×10^3	296.1×10^3	204.0×10^3	0.978	5.0	21.0
2Aa87	10	--------	15.0	10.3	---------	---------	324.8	223.8	-----	4.0	13.0
2Aa89	10	--------	19.0	13.1	319.0	219.8	325.2	224.1	.980	5.0	16.0
2Aa82	10	--------	20.0	13.8	322.1	221.9	328.3	226.2	.981	5.0	14.0
2Aa81	10	--------	22.0	15.2	365.5	251.8	373.2	257.1	.979	3.0	13.0
		$\bar{X}_F$	19.2×10^6	13.2×10^6	324.0×10^3	223.3×10^3	329.5×10^3	227.0×10^3	0.979	4.4	15.4
		s_F	2.6	1.8	31.3	21.5	27.7	19.1	.002	.9	3.4
		$\bar{X}_F - \bar{X}_D$	.7	.4	49.0	33.8	47.3	32.5	.004	-2.0	-10.0
		X_L	-2.4	-1.7	-.9	-.6	12.7	8.7	-.013	-3.3	-15.3
		X_U	3.8	2.6	98.9	68.2	81.9	56.3	.021	-.7	-4.7
		$^{d}s_F^2/s_D^2$	3.00	3.24	170.09	180.57	66.37	63.34	.02	1.00	.72

[a]Specimen exposed in gaseous helium at 17 K throughout test. Fast fluence is $E > 0.5$ MeV (80 fJ).
[b]Arithmetic mean, $\bar{X}$; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.
[c]Total elongation in 0.5-in. (1.27-cm) gage length.
[d]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two system of units.

TABLE XIII. - EFFECT OF REACTOR IRRADIATION AT 17 K ON 17 K TENSILE PROPERTIES OF

TITANIUM - 8 ALUMINUM - 1 MOLYBDENUM - 4 VANADIUM ALLOY

Specimen	Fast fluence, neutrons/cm^2 (a)	Statistical analysis (b)	Tensile modulus of elasticity		Yield strength (0.2-percent offset)		Ultimate strength		Yield- to ultimate-strength ratio	Total elongation, percent (c)	Reduction of area, percent
			lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2	lb/in.^{2}	N/cm^2			
4Aa3	0	--------	19.2×10^6	13.2×10^6	217.0×10^3	149.5×10^3	236.0×10^3	162.6×10^3	0.920	12.1	----
4Aa1	↓	--------	18.3	12.6	220.0	151.6	242.0	166.7	.909	5.8	----
4Aa4		--------	16.4	11.3	224.0	154.3	236.0	162.6	.950	7.8	----
4Aa5		--------	18.5	12.7	224.0	154.3	238.0	164.0	.940	7.8	----
4Aa2		--------	18.9	13.0	236.0	162.6	243.0	167.4	.970	15.2	----
		$\bar{X}_A$	18.3×10^6	12.6×10^6	224.2×10^3	154.5×10^3	239.0×10^3	164.7×10^3	0.938	9.5	----
		s_A	1.1	.7	7.2	5.0	3.3	2.3	.024	3.8	----
		$\bar{X}_A - \bar{X}_A$	0	0	0	0	0	0	0	0	----
		X_L	-1.4	-1.0	-8.9	-6.1	-4.2	-2.9	-.030	-4.7	----
		X_U	1.4	1.0	8.9	6.1	4.2	2.9	.030	4.7	----
4Aa34	1×10^{17}	--------	19.6×10^6	13.5×10^6	238.0×10^3	164.0×10^3	264.0×10^3	181.9×10^3	0.901	6.0	31.0
4Aa43	1	--------	25.4	17.5	243.0	167.4	262.0	180.5	.927	5.0	26.0
4Aa38	1	--------	--------	--------	247.0	170.2	259.0	178.5	.954	6.0	30.0
		$\bar{X}_B$	22.5×10^6	15.5×10^6	242.7×10^3	167.2×10^3	261.7×10^3	180.3×10^3	0.927	5.7	29.0
		s_B	4.1	2.8	4.5	3.1	2.5	1.7	.027	.6	2.6
		$\bar{X}_B - \bar{X}_A$	4.2	2.9	18.5	12.7	22.7	15.6	-.011	-3.8	----
		X_L	[e]-33.2	[e]-22.9	9.0	6.2	17.9	12.3	-.057	-8.6	----
		X_U	41.6	28.7	28.0	19.3	27.5	18.9	.035	1.0	----
		[d]s_B^2/s_A^2	13.90	16.00	.39	.38	.57	.55	1.26	.02	----

[a]Specimen exposed in gaseous helium at 17 K throughout test. Fast fluence is for E > 0.5 MeV (80 fJ).

[b]Arithmetic mean, $\bar{X}$; estimated standard deviation, s; lower and upper limits of 95-percent confidence interval of the difference, X_L, X_U.

[c]Total elongation on 0.5-in. (1.27-cm) gage length.

[d]Values for tensile modulus of elasticity, yield strength, and ultimate strength differ because of significant figures maintained in comparative calculations between the two system of units.

[e]Value exceeds lower limit of possible change.

REFERENCES

1. Schwartzberg, F. R.; Osgood, S. H.; Keys, R. D.; and Kiefer, T. F.: Cryogenic Materials Data Handbook. Martin Co. (AFML-TDR-64-280, DDC No. AD-609562), Aug. 1964.
2. Makin, M. J.; and Minter, F. J.: The Effect of Neutron Irradiation on the Mechanical Properties of Titanium and Zirconium. J. Inst. Metals, vol. 85, 1956-1957, pp. 397-402.
3. Anon.: Tentative Recommended Practice for Measuring Neutron Flux Environment for Reactor Irradiated Specimens. ASTM Designation: E199-62T. 1962 Supplement to Book of ASTM Standards, Part 3, Methods of Testing Metals (Except Chemical Analysis). ASTM, 1962, pp. 72-85.
4. Anon.: Tentative Methods of Tension Testing of Metallic Materials. ASTM Designation: E8-61T. 1961 Book of ASTM Standard, Part 3, Methods of Testing Metals (Except Chemical Analysis). ASTM, 1961, pp. 165-181.
5. Anon.: Recommended Practice for Effect of High-Energy Radiation on the Tensile and Impact Properties of Metallic Materials. ASTM Designation: E184-62. 1962 Supplement to Book of ASTM Standards, Part 3, Methods of Testing Metals (Except Chemical Analysis). ASTM, 1962, pp. 29-33.
6. Bridges, W. L.; and Liebschutz, A. M.: Tension-Compression-Shear Loop for Irradiation Testing at Cryogenic Temperatures. Advances in Cryogenic Engineering. Vol. 7. K. D. Timmerhaus, ed., Plenum Press, 1962, pp. 440-447.
7. Aseff, G. V., Sr.; Callaway, R. F.; and Liebschutz, A. M.: Tensile, Compression, and Shear Loop for Irradiation Testing at Cryogenic Temperatures. Advances in Cryogenic Engineering. Vol. 8, K. D. Timmerhaus, ed., Plenum Press, 1963, pp. 624-630.
8. Anon.: Tentative Method of Verification and Calssification of Extensometers. ASTM Designation: E83-57T. 1961 Book of ASTM Standards, Part 3, Methods of Testing Metals (Except Chemical Analysis). ASTM, 1961, pp. 202-209.
9. Liebschutz, A. M.; and Schulte, C. A.: In-Pile Loop Cryogenic Refrigerator. Advances in Cryogenic Engineering. Vol 7. K. D. Timmerhaus, ed., Plenum Press, 1963, pp. 181-188.
10. Natrella, Mary Gibbons: Experimental Statistics. Handbook 91, National Bureau of Standards, Aug. 1, 1963.

11. Eckel, J. F.; Bruch, C. A.; Levy, A.; Willis, A. H.; and Seymour, W. E.: Radiation Effects on Reactor Materials - A Review of KAPL Work. Radiation Effects Review Meeting, Congress Hotel, Chicago, July 31-August 1, 1956, AEC Rep. TID-7515 (Part 2) (Del.), Aug. 1956, pp. 170-221.

12. Bruch, C. A.; McHugh, W. E.; and Hockenbury, R. W.: Variations in Radiation Damage to Metals. Trans. AIME, vol. 206, 1956, pp. 1362-1372.

13. Anderson, W. K.; Beck, C. J.; Kephart, A. R.; and Theilacker, J. S.: Reactor Structural Materials: Engineering Properties as Affected by Nuclear Reactor Service. Spec. Tech. Publ. No. 314, ASTM, 1962. (Data reported by Bruch, C. A.; and McHugh, W. E.: Radiation Damage Studies of Seven Nonfissionable Metals. Paper 58-NESC-37, presented at AIChE Nuclear Engineering and Scientific Conference, Chicago, Ill., Mar. 17-21, 1958.)

14. Bomar, E. S.; Cooke, F. W.; Leonard, W. J.; Prislinger, J. J.; and Wodtke, C. H.: Engineering Metallurgy. Metallurgy Division Annual Progress Report for Period Ending September 1, 1959. Rep. ORNL-2839, Union Carbide Corp., Dec. 16, 1959, pp. 127-143.

15. Adamson, G. M.: Metallurgy - Mechanical Metallurgy - Tensile Properties of Irradiated Titanium Alloys. Homogeneous Reactor Program Quarterly Progress Report for Period Ending October 31, 1957. Rep. ORNL-2432, Union Carbide Corp., Jan. 21, 1958, pp. 135-136.

16. Porter, Lew F.: Radiation Effects in Steel. Materials in Nuclear Applications. Spec. Tech. Publ. No. 276, ASTM, 1960, pp. 147-196.

17. Lombardo, J. J.; Dixon, C. E.; and Begley, J. A.: Cryogenic Radiation Effects on NERVA Structural Materials. Effects of Radiation on Structural Metals. Spec. Tech. Publ. No. 426, ASTM, 1967, pp. 625-652.

18. Watson, J. F.; Christian, J. L.; and Allen, J. W.: A Study of the Effects of Nuclear Radiation on High-Strength Aerospace Vehicle Materials at the Boiling Point of Hydrogen (-423° F). Rep. ERR-AN-085, General Dynamics/Astronautics, Sept. 27, 1961.

19. Shogan, R. P.: Tensile Properties of Irradiated Ti-5% Al-2.5% Sn ELI at Cryogenic Temperatures. Rep. WANL-TME-1860, Westinghouse Astronuclear Lab., Nov. 1968.

20. Drennan, J. E.; and Hamman, D. J.: Supplemental Analysis of GTR-19 Data. Rep. BMI-NRE-10, Battelle Memorial Inst., July 31, 1968.

21. Smith, E. T.: NERVA Materials Irradiation Program - GTR Test 19. Vol. 2 - Comprehensive Presentation of Material Property Data. Rep. FZK-342-2, General Dynamics/Fort Worth, Apr. 1, 1968.

22. Wallack, Stanley: The Effect of Radiation on the Physical and Mechanical Properties of Metals and Alloys. Universal Winding Co., Inc., (WADC TR 58-605, DDC No. AD-215540), Feb. 1959.

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